Enclosure 2

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 176 TO

RENEWED FACILITY OPERATING LICENSE NO. DPR-22

NORTHERN STATES POWER COMPANY

MONTICELLO NUCLEAR GENERATING PLANT

DOCKET NO. 50-263

(Non-Proprietary - Redacted Version)

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

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NORTHERN STATES POWER COMPANY – MINNESOTA (NSPM)

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Proprietary information is identified by text enclosed within double brackets ([[ ]]).

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1.0 INTRODUCTION

1.1 Application

By letter dated November 5, 2008,[1] as supplemented by additional letters,[2] Northern States Power Company, a Minnesota corporation (NSPM or the licensee), submitted an application for amendment for an extended power uprate (EPU) for the Monticello Nuclear Generating Plant (MNGP). The proposed amendment would increase the authorized maximum licensed thermal power level by approximately 13 percent (%), from the current licensed thermal power (CLTP) of 1,775 megawatts thermal (MWth) to 2,004 MWth.

The first six additional letters provided in Footnote 2 are referenced in Enclosure 16 of the licensee's November 5, 2008, application. Enclosure 16 includes a table of docketed NRC acceptance review questions and licensee response letters associated with the March 31, 2008, MNGP EPU submittal.

The supplemental letters received between December 11, 2008, and November 8, 2013, contained clarifying information that did not change the initial no significant hazards consideration determination noticed in the *Federal Register* on January 23, 2009 (74 FR 4252), and did not expand the scope of the original application.

1.2 Background

The MNGP site is located in Monticello, Minnesota, along the southern bank of the

[1] Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML083230111, ML083230112, ML083230114, and ML083230124.

[2] May 20, 2008 (ADAMS Accession No. ML081430494); May 28, 2008 (ADAMS Accession No. ML081490639); May 30, 2008 (ADAMS Accession No. ML081550504); June 3, 2008 (ADAMS Accession No. ML081550640); June 5, 2008 (ADAMS Accession No. ML081570467); June 12, 2008 (ADAMS Accession No. ML081640435); June 25, 2008 (ADAMS Accession No. ML081770562); December 11, 2008 (ADAMS Accession No. ML083500099); January 29, 2009 (ADAMS Accession No. ML090300303); February 4, 2009 (2 letters, ADAMS Accession Nos. ML090360545 and ML093620023); February 17, 2009 (ADAMS Accession No. ML090710679); February 24, 2009 (ADAMS Accession No. ML090560464); March 19, 2009 (ADAMS Accession No. ML090790388); April 22, 2009 (ADAMS Accession No. ML091130634); May 13, 2009 (ADAMS Accession No. ML091410117); May 26, 2009 (ADAMS Accession No. ML091470559); May 28, 2009 (ADAMS Accession No. ML081490639); May 29, 2009 (ADAMS Accession No. ML091520133); June 12, 2009 (ADAMS Accession No. ML091670410); June 16, 2009 (ADAMS Accession No. ML091671787); July 13, 2009 (ADAMS Accession No. ML092170404); July 23, 2009 (ADAMS Accession No. ML092090219); August 12, 2009 (2 letters, ADAMS Accession Nos. ML092260436 and ML092260132); August 19, 2009 (ADAMS Accession No. ML092320064); August 21, 2009 (2 letters, ADAMS Accession Nos. ML092390341 and ML092430088); August 26, 2009 (ADAMS Accession No. ML092390326); August 31, 2009 (ADAMS Accession No. ML092440171); October 1, 2009 (ADAMS Accession No. ML092790191); January 25, 2010 (ADAMS Accession No. ML100270020); April 6, 2010 (ADAMS Accession No. ML101020021); December 21, 2010 (ADAMS Accession No. ML103570026); June 30, 2010 (ADAMS Accession No. ML102010462); April 5, 2011 (ADAMS Accession No. ML11081A046); July 7, 2011 (ADAMS Accession No. ML11144A085); August 30, 2011 (ADAMS Accession No. ML11249A045); November 11, 2011 (ADAMS Accession No. ML11321A332); January 13, 2012 (ADAMS Accession No. ML12019A246); July 19, 2012 (ADAMS Accession Nos. ML12207A541, ML12207A542, ML12207A543, ML12207A544, and ML12207A545); July 19, 2012 (ADAMS Accession No. ML122070642); September 28, 2012 (ADAMS Accession No. ML12276A057); October 21, 2012 (ADAMS Accession No. ML12307A036); October 22, 2012 (ADAMS Accession Nos. ML12298A032 and ML12298A033); October 30, 2012 (ADAMS Accession No. ML12307A036); November 30, 2012 (ADAMS Accession No. ML123380435); January 21, 2013 (ADAMS Accession No. ML130390220); January 31, 2013 (ADAMS Accession Nos. ML13037A200 and ML13037A201), February 22, 2013 (ADAMS Accession No. ML13057A034); February 27, 2013 (ADAMS Accession No. ML130640494); March 7, 2013 (ADAMS Accession No. ML13071A615); March 18, 2013 (ADAMS Accession No. ML13078A390); March 21, 2013 (ADAMS Accession No. ML13085A033); March 29, 2013 (ADAMS Accession No. ML130920389); April 10, 2013 (ADAMS Accession No. ML131050224); May 13, 2013 (ADAMS Accession No. ML13134A301; May 30, 2013 (ADAMS Accession No. ML13154A011); June 26, 2013 (ADAMS Accession No. 13191B126); July 8, 2013 (ADAMS Accession No. ML13191A568); July 18, 2013 (2 letters, ADAMS Accession Nos. ML13199A487 and ML13205A110); August 2, 2013 (ADAMS Accession No. ML13218A339); September 30, 2013 (ADAMS Accession No. ML13275A063); and November 8, 2013 (ADAMS Accession No. ML13316B025).

Mississippi River, approximately 30 miles northwest of Minneapolis/St. Paul, and east of Interstate Highway 94. The site consists of 2 miles of frontage on both banks of the Mississippi River, within portions of Wright and Sherburne Counties. The plant and its supporting facilities are located in Wright County.

The U.S. Nuclear Regulatory Commission (NRC) issued the construction permit for MNGP on June 19, 1967. The NRC issued the operating license for MNGP on January 9, 1981. MNGP is a single-cycle, forced circulation, General Electric (GE) boiling-water reactor (BWR), BWR-3, producing steam for direct use in a steam turbine. The General Electric Corporation supplied the nuclear steam supply system (NSSS) and Bechtel Corporation originally designed and constructed the balance of the plant. MNGP operates at a current licensed power output of 1,775 MWth, with a gross electrical output of approximately 600 megawatts electric (MWe). The MNGP Updated Safety Analysis Report (USAR) contains details concerning the plant and the site.

The Atomic Energy Commission (AEC, predecessor of the NRC) originally issued an operating license to MNGP for a thermal power level of 1,670 MWth. In September 1998, the NRC approved a 6.3 percent power uprate to increase the power output to 1,775 MWth.

In the application dated November 5, 2008, the licensee indicated its intention to implement the proposed EPU in two phases to coincide with two refueling outages: the first refueling outage was scheduled for late 2009, with a corresponding increase in power of approximately 50 MWth to a total of 1,825 MWth; the second refueling outage was scheduled for 2011, and the power level will be increased to the maximum of 2,004 MWth. Due to the length of the review and the licensee's installation of EPU hardware changes, the licensee has stated its intent to begin implementation of the EPU upon receipt of an approved license amendment from the NRC.

As discussed in the MNGP USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," the licensee had previously made an assessment of each of the 70 criteria of the AEC Design Criteria. In Appendix E, the licensee had provided references to locations in the MNGP USAR, where further information pertinent to each criterion can be found.

1.3 Licensee's Approach

The licensee's application for the proposed EPU was prepared following the guidelines contained in GE Licensing Topical Report (LTR) NEDC-33004P-A, "Constant Pressure Power Uprate," Revision 4, dated July 31, 2003. The constant pressure power uprate (CPPU) LTR, hereafter referred to as the CLTR, was approved by the NRC in a final safety evaluation (SE) dated March 31, 2003. The CLTR provided appropriate guidelines for CPPU applications with a core exclusively using GE fuel. Some topics in the CLTR are directly fuel-dependent, because the fuel type affects the resulting evaluation or the consequences of transients or accidents.

For the fuel-dependent topics, the evaluation methods from GE LTR NEDC-32424P-A, dated February 1999, "Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate," (ELTR1), and in Section 4.8 of Supplement 1 of GE LTR, NEDC-32523P-A, dated February 2000, "Generic Evaluations of General Electric Boiling Water Reactor Extended Power Uprate" (ELTR2), are applied. In general, the licensee's plant-specific engineering evaluations supporting the power uprate were performed in accordance with guidance contained in ELTR1. This topical report was previously reviewed and endorsed by the NRC staff. For some items, bounding analyses and evaluations provided in GE LTR ELTR2 were cited. The NRC staff has also previously approved ELTR2.

An increase in the electrical output of a BWR plant is accomplished primarily by generating and supplying higher steam flow to the turbine-generator. As currently licensed, most BWR plants, including MNGP, have an as-designed equipment and system capability to accommodate steam flow rates above the original rating. In addition, continuing improvements in the analytical techniques (computer codes and data) based on several decades of BWR safety technology, plant performance feedback, and improved fuel and core designs have resulted in a significant increase in the design and operating margins between calculated safety analysis results and the licensing limits. These available safety analyses differences, combined with the excess as-designed equipment, system and component capabilities, provide BWR plants the capability to achieve an increase in thermal power ratings of between 5 and 20 percent without major NSSS hardware modifications.

The licensee proposes that a higher steam flow be achieved by increasing the reactor power along specified control rod and core flow lines. For such, a limited number of operating parameters will be changed, some setpoints will be adjusted, and instruments will require recalibration. Plant procedures will be revised, and tests similar to some of the original startup tests will be performed. Modifications to power generation equipment will be implemented, as necessary. The licensee provided a detailed list of planned modifications in Enclosure 8 to the November 5, 2008, application.

Enclosure 5 to the licensee's application dated November 5, 2008, contains GE Report NEDC-33322P, Revision 3, which is the "Safety Analysis Report for Monticello Constant Pressure Power Uprate" (PUSAR). This report summarizes the results of safety analyses and evaluations performed by GE, justifying the proposed MNGP EPU. The PUSAR follows the generic content and format using the CPPU approach for a reactor power uprate, as described in the CLTR. A non-proprietary (i.e., publicly available) version of the PUSAR is contained in Enclosure 7 to the licensee's application dated November 5, 2008.

The licensee referenced GE LTR NEDC-33173P (Reference 13) in its application. This report is based on the NRC staff-approved approach taken by the Vermont Yankee Nuclear Generating Station for applying the GE analytical methods for CPPU operating domains. The NRC staff's safety evaluation report (SER) for NEDC-33173P, dated January 17, 2008, specifies the limitations that apply to NEDC-33173P.

The licensee referenced NEDC-33173P to justify application of GE methods to the proposed EPU. Each limitation specified in the NRC staff's SER dated January 17, 2008, was evaluated by the licensee for acceptability for the proposed EPU. The NRC staff's evaluation of applicability of NEDC-33173P can be found in Section 2.8.7 of this safety evaluation.

Table 1-2 of the PUSAR provides a summary comparing the plant operating conditions under CLTP and under the proposed plant operating conditions and CPPU.

The licensee plans to implement the proposed EPU following license amendment approval. Since full EPU power is not possible due to core flow limitations, ascension to full EPU power will be done in conjunction with NRC approval of the Maximum Extended Load Line Limit Analysis Plus (MELLLA+) license amendment or other options to improve core flow capability.

1.4 Plant Modifications

The licensee determined that plant modifications were necessary to implement the proposed MNGP EPU. As previously mentioned in Section 1.3, the licensee tabulated these planned modifications in Enclosure 8 of its November 5, 2008, application.

The NRC staff's evaluation of the licensee's plant modifications, within the scope of the areas of review, is provided in Section 2.5 of this SE.

1.5 Method of NRC Staff Review

The NRC staff's review of the MNGP EPU application is based on NRC Review Standard RS-001, "Review Standard for Extended Power Uprates," Revision 0, dated December 2003. RS-001 contains guidance for evaluating each area of review in the application, including the specific General Design Criteria (GDC) used as the NRC's acceptance criteria. The guidance in RS-001 is based on the final GDCs. In addition to RS-001, the NRC staff used applicable rules, regulatory guides, Standard Review Plan (SRP) sections, and NRC staff positions on the topics being evaluated.

The NRC staff requested that the licensee identify all codes and methodologies used to obtain safety limits (SLs) and operating limits and explain how they verified these limits were correct for the uprate reactor core. The NRC staff also requested that the licensee identify and discuss any limitations imposed by the NRC staff on the use of these codes and methodologies.

Pursuant to 10 CFR 50.92(a), in determining whether to issue an amendment to a license, the Commission will be guided by the considerations which govern the issuance of initial licenses to the extent applicable and appropriate. The considerations for issuance of an operating license are provided in 10 CFR 50.57, and include the following findings: (1) there is reasonable assurance that the activities authorized by the operating license can be conducted without endangering the health and safety of the public; (2) there is reasonable assurance that such activities will be conducted in compliance with the Commission's regulations; and (3) the issuance of the license will not be inimical to the common defense and security or to the health and safety of the public. Thus, when considering this amendment to the license, the NRC staff used these same three considerations to the extent applicable and appropriate.

The purpose of the NRC staff's review is to evaluate the licensee's assessment of the impact of the proposed MNGP EPU on design-basis analyses. The NRC staff evaluated the licensee's application and supplements. The NRC staff also performed audits, independent calculations, analyses, and evaluations as noted in details of evaluation below.

In areas where the licensee and its contractors used NRC-approved or widely accepted methods in performing analyses related to the proposed MNGP EPU, the NRC staff reviewed relevant material to ensure that the licensee/contractor used the methods consistently with the limitations and restrictions placed on the methods. In addition, the NRC staff considered the effects of the changes in plant operating conditions on the use of these methods to ensure that the methods are appropriate for use at the proposed MNGP EPU operating conditions. The details of the NRC staff's review are provided in Section 2.0 of this SE.

Audits supporting the proposed MNGP EPU were conducted by the NRC staff and its contractors in relation to the following topics:

- Steam dryer structural integrity analyses
- Reactor core long-term stability solution

During the audits, the NRC staff and its contractors conducted confirmatory calculations, analyses, and evaluations related to the aforementioned topics.

2.0 EVALUATION

2.1 Materials and Chemical Engineering

2.1.1 Reactor Vessel Material Surveillance Program

Regulatory Evaluation

The reactor vessel (RV) material surveillance program provides a means for monitoring the fracture toughness of the RV beltline materials to support analyses for ensuring the structural integrity of the ferritic components of the RV. Title 10 of the *Code of Federal Regulations*, Part 50 (10 CFR Part 50), Appendix H, provides the NRC's requirements for the design and implementation of the RV material surveillance program. The NRC staff's review primarily focused on the effects of the proposed EPU on the licensee's RV surveillance capsule withdrawal schedule. The NRC's acceptance criteria are based on: (1) GDC-14, "Reactor coolant pressure boundary," which requires that the reactor coolant pressure boundary (RCPB) be designed, fabricated, erected, and tested so as to have an extremely low probability of rapidly propagating failure; (2) GDC-31, "Fracture prevention of reactor coolant pressure boundary," which requires that the RCPB be designed with margin sufficient to assure that, under specified conditions, it will behave in a non-brittle manner and that the probability of a rapidly propagating fracture is minimized; (3) 10 CFR Part 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements," which provides requirements for monitoring changes in the fracture toughness properties of materials in the RV beltline region; and (4) 10 CFR 50.60, "Acceptance criteria for fracture prevention measures for light-water nuclear power reactors for normal operations," which requires compliance with the requirements of 10 CFR Part 50, Appendix H. Specific review criteria are contained in the SRP, Section 5.3.1.

Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company, the predecessor to NSPM, performed a comparative evaluation of the design basis of MNGP with the AEC proposed GDC of 1967. The MNGP comparative evaluation to the 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") associated with this review is contained in Appendix E of the MNGP USAR: draft GDC-9, draft GDC-33, draft GDC-34, and draft GDC-35.

Technical Evaluation

Requirements of 10 CFR Part 50, Appendix H, invoke, by reference, the guidance in American Society for Testing and Materials (ASTM) Standard Practice E 185, "Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels." ASTM Standard Practice E 185 provides guidelines for designing and implementing the RV materials surveillance programs for operating light-water reactors, including guidelines for determining RV surveillance capsule withdrawal schedules based on the vessel material predicted transition temperature shifts (ΔRT_{NDT}).

The licensee discussed the impact of the 12.9 percent EPU on the RV material surveillance program in Section 2.1.1 of the MNGP EPU Licensing Report, submitted as part of the November 5, 2008, application. The licensee stated, in part, that the MNGP RV surveillance program consists of three capsules. One capsule, containing Charpy impact test specimens, was removed from the RV after 7.08 effective full power years (EFPY) of facility operation. One set of specimens from this capsule was tested. A second set of specimens from this capsule was re-encapsulated and placed in the Prairie Island Nuclear Generating Plant RV for accelerated irradiation and testing. The remaining two capsules have been in the RV since plant startup. One of these two capsules was removed during the refueling outage in 2007, after 26.5 EFPY of facility operation, and the other capsule is designated as a standby capsule. The licensee concluded that the current surveillance capsule withdrawal schedule is still valid for the EPU conditions.

The Boiling Water Reactor Vessels and Internals Program (BWRVIP) developed an Integrated Surveillance Program (ISP) for the reactor vessel base metal and weld materials in all operating BWRs. The BWRVIP ISP is designed to comply with the requirements for ISPs in Appendix H to 10 CFR Part 50. The ISP is described in proprietary topical reports BWRVIP-78, "BWR Integrated Surveillance Program (ISP) Plan," and BWRVIP-86, "BWR Vessel and Internals Project: BWR Integrated Surveillance Program (ISP) Implementation." The NRC approved these proprietary reports applying the design and implementation of the ISP by BWRs during their first 40-year operating period in its February 1, 2002, final SER to the BWRVIP (Reference 88). The BWRVIP issued proprietary topical report BWRVIP-116, "BWR Vessel and Internals Project Integrated Surveillance Program (ISP) Implementation for License Renewal," to address ISP changes necessary for License Renewal Applicants for operating BWRs. This report was approved by the NRC in a March 1, 2009, letter from M.A. Mitchell (NRC) to B. Eaton (BWRVIP) (Reference 71).

The NRC staff verified that the licensee's Reactor Vessel Surveillance Program is implemented in accordance with the BWRVIP ISP as described and discussed in above cited BWRVIP-78, BWRVIP-86, and BWRVIP-116 for the extended license operating period. The NRC staff approved the application of the BWRVIP ISP to the MNGP RV for the original 40-year license term in its SE dated April 22, 2003, in which the staff concurred that the BWRVIP ISP, as approved in BWRVIP-78 and BWRVIP-86-A (the staff-approved version of BWRVIP-86), met the requirements of 10 CFR Part 50, Appendix H, for the RV. The NRC staff documented its review and approval of the application of the BWRVIP ISP to the MNGP RV for the extended licensed operating period (54 EFPY) as part of its October 2006 SER for license renewal. The ISP provides for a number of surveillance capsules to be removed from specified BWRs and to be available for testing during the license renewal period for the BWR fleet. The ISP establishes acceptable technical criteria for capsule withdrawal and testing. The NRC staff verified that the proposed EPU will have no impact on the licensee's effective implementation of the BWRVIP ISP.

The NRC staff reviewed the licensee's description of the RV material surveillance program under EPU conditions and finds it acceptable because the surveillance program will continue to be implemented in accordance with the BWRVIP ISP under EPU conditions. Therefore, the staff finds that the licensee's RV material surveillance program for MNGP will remain in compliance with the requirements specified in 10 CFR Part 50, Appendix H, under EPU conditions.

The staff also reviewed the licensee's projected fluence values for the 54 EFPY licensed operating period, accounting for EPU conditions. These fluence values were provided in Table 2.1-1 of the MNGP EPU Licensing Report. The NRC staff found that these fluence values were calculated using the GE methodology for neutron flux calculations documented in LTR NEDC-32983P-A, "General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluations," dated January 2006. This methodology has been reviewed and approved by the NRC staff, and it adheres to the guidance of Regulatory Guide (RG) 1.190, "Calculational and Dosimetry Methods for determining Pressure Vessel Neutron Fluence," dated March 2001. Therefore, the NRC staff found that the EPU fluence projections were acceptable.

Conclusion

The NRC staff concludes that the licensee has adequately addressed the impact of the proposed EPU on the RV material surveillance program at MNGP. The NRC staff further concludes that the licensee's implementation of the BWRVIP ISP at MNGP is appropriate to ensure that the material surveillance program will continue to meet the regulatory requirements set forth above following implementation of the proposed EPU.

Based on the above, the NRC staff finds the proposed EPU acceptable with respect to the RV material surveillance program.

2.1.2 Upper Shelf Energy, Pressure-Temperature Limits, and RV Circumferential Weld Properties

Regulatory Evaluation

Appendix G to 10 CFR Part 50 provides fracture toughness requirements for ferritic materials (low alloy steel or carbon steel) in the RCPB, including upper shelf energy (USE) requirements for ensuring adequate safety margins against ductile tearing, as well as requirements for calculating pressure-temperature (P-T) limits for the plant. Appendix G to 10 CFR Part 50 requires that RCPB materials satisfy the criteria in Appendix G of Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code) in order to ensure the structural integrity of the RCPB during any condition of normal operation, including anticipated operational occurrences and hydrostatic tests.

The NRC's acceptance criteria are based on: (1) GDC-14, which requires that the RCPB be designed, fabricated, erected, and tested so as to have an extremely low probability of rapidly propagating fracture; (2) GDC-31, which requires that the RCPB be designed with a safety margin sufficient to assure that, under specified conditions, it will behave in a nonbrittle manner and the probability of a rapidly propagating fracture is minimized; (3) 10 CFR Part 50, Appendix G, which specifies fracture toughness requirements for ferritic components of the RCPB; and (4) 10 CFR 50.60, which requires compliance with the requirements of 10 CFR Part 50, Appendix G. Specific review criteria are contained in SRP Section 5.3.2.

Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company, the predecessor to NSPM, performed a comparative evaluation of the design basis of MNGP with the AEC proposed GDC of 1967. The MNGP comparative evaluation to the 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") associated with this review is contained in Appendix E of the MNGP USAR: draft GDC-9, draft GDC-33, draft GDC-34, and draft GDC-35.

Technical Evaluation

Upper Shelf Energy Calculations

Appendix G to 10 CFR Part 50 provides the NRC staff's criteria for maintaining acceptable levels of USE for the RV beltline materials of operating reactors throughout the licensed operational lives of the facilities. It requires RV beltline materials to have a minimum USE value of 75 foot-pounds (ft-lb) in the unirradiated condition, and to maintain a minimum USE value above 50 ft-lb throughout the operating life of the facility, unless it can be demonstrated through analysis that lower values of USE would provide acceptable margins of safety against fracture equivalent to those required by Appendix G of the ASME Code, Section XI. The regulation also mandates that the methods used to calculate USE values must account for the effects of neutron radiation on the USE values for the materials and must incorporate any relevant RV surveillance capsule data that are reported through implementation of a plant's 10 CFR Part 50, Appendix H RV materials surveillance program. The NRC staff's recommended guidelines for calculating the effects of neutron radiation on the USE values for the RV beltline materials are specified in RG 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials."

Projected USE values for RV materials are calculated based on the projected neutron fluence at a postulated flaw depth at a location of one-quarter of the vessel wall thickness (1/4T) from the clad/base metal interface of the RV 1/4T, weight percentage (wt%) of copper (Cu) in the material, and the initial USE value for the material prior to exposure to neutron radiation. Therefore, the direct calculation of projected USE values requires that initial USE values are available for the plant's RV beltline materials. For many BWRs, MNGP included, these initial USE values are not available, and direct calculation of projected USE values is not possible. However, the projected percentage decreases in USE may still be calculated based on the projected neutron fluence and copper content. Therefore, the BWR Owners Group (BWROG) developed acceptance criteria for projected percentage decreases in BWR RV materials USE, based on an equivalent margins analysis (EMA). This EMA is described in the GE BWRVIP report 74-A, "BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Inspection and Flaw Evaluation Guidelines (BWRVIP-74)," dated June 2003. The NRC staff previously reviewed and approved the EMA and associated acceptance criteria in an SE dated October 18, 2001. Therefore, the EMA acceptance criteria are now deemed suitable by the NRC staff for evaluating the projected percentage decreases in USE for BWR RV materials and determining compliance with the USE requirements of 10 CFR Part 50, Appendix G.

The licensee discussed the impact of the 12.9% EPU on the USE analysis for the RV beltline materials in Section 2.1.2 of the MNGP EPU Licensing Report. The licensee demonstrated that all RV beltline materials at MNGP will remain bounded by the EMA acceptance criteria in BWRVIP-74-A through the end-of-life (EOL), accounting for EPU conditions, thereby satisfying the requirements of Appendix G to 10 CFR Part 50. The licensee provided the wt% Cu values and projected 54 EFPY peak fluence values at the 1/4T location for all RV beltline materials in Table 2.1-1 of the MNGP EPU Licensing Report. The projected 54 EFPY peak fluence values at the 1/4T location are 3.24×10^{18} n/cm^2 (neutrons per square centimeter] (E > 1.0 MeV [Energy greater than 1.0 Mega electron-volts]) for the lower shell plates and 4.75×10^{18} n/cm^2 (E > 1.0 MeV) for the lower-intermediate shell plates and all welds, accounting for EPU conditions. The licensee calculated the projected percentage decrease in the USE for all RV beltline materials based on these fluence values using Regulatory Position 1.2 from RG 1.99, Revision 2. Regulatory Position 1.2 utilizes Figure 2 of RG 1.99 to determine a percentage drop in the USE based on neutron fluence and copper content when no credible surveillance data is available for determining the USE for the materials.

The NRC staff performed independent calculations of the projected percentage decrease in the USE at the 1/4T location for all RV beltline materials, accounting for EPU conditions, based on Regulatory Position 1.2 from RG 1.99, Rev. 2, using the wt % Cu, and the 54 EFPY neutron fluence values provided above. The NRC staff calculated a projected percentage USE decrease of 22% for the limiting RV plate material and 21% for all RV beltline weld materials at EOL. This was in general agreement with the licensee's USE calculations for these materials. These values meet the BWRVIP-74 EMA acceptance criteria for the allowable percentage decrease in USE – specifically, 23.5% for RV plates and 39% for welds fabricated by the shielded metal arc welding process in GE Type III BWR RVs. Therefore, based on its independent calculations, the NRC staff determined that the MNGP RV beltline materials will remain acceptable, with respect to the USE, under the EPU conditions, for the remainder of the extended licensed operating period.

P-T Limit Calculations

Section IV.A.2 of 10 CFR Part 50, Appendix G, requires that the P-T limits for operating reactors be at least as conservative as those that would be generated using the calculation methods specified in the ASME Code, Section XI, Appendix G. The rule also requires that the P-T limit calculations account for the effects of neutron radiation on the material properties of the RV beltline materials and that P-T limit calculations incorporate any relevant RV surveillance capsule data that are required to be reported as part of the licensee's implementation of its 10 CFR Part 50, Appendix H RV materials surveillance program. The NRC staff's recommended guidelines for calculating the effects of neutron radiation on the adjusted reference temperature (ART) values used for P-T limit calculations are specified in RG 1.99, Rev. 2.

When the licensee submitted the EPU LAR, the P-T limit curves for the MNGP reactor coolant system were contained in the MNGP Technical Specifications. By letter dated January 20, 2012 (ADAMS Accession No. ML12033A175), as supplemented by letter dated December 7, 2012 (ADAMS Accession No. ML12349A210), the licensee submitted an LAR to revise the MNGP TS requirements related to the P-T limits. The proposed TS revisions included the implementation of new P-T limits curves for 54 EFPY, corresponding to the end of the 60-year extended license. The P-T limit curves would be relocated from the TS Section 3.4.9, "RCS Pressure and Temperature (P-T) Limits," to a newly-created Pressure-Temperature Limits Report (PTLR), the content of which is governed by TS Section 5.6, "Administrative Controls." To support the implementation of the PTLR, TS Section 5.6.6 was added to the administrative controls to specify requirements for governing the content and implementation of the PTLR, including the specification of the analytical methods used to generate the P-T limits. The 54 EFPY P-T limit curves were generated using the analytical methods documented in the NRC-approved BWROG Topical Report, SIR-05-044-A, "Pressure-Temperature Limits Report [PTLR] Methodology for Boiling Water Reactors," Revision 0, dated April 2007 (ADAMS Accession No. ML072340283).

By letter dated February 27, 2013 (ADAMS Accession No. 13025A155), the NRC staff issued License Amendment 172, authorizing the implementation of the PTLR containing new P-T limit curves that are valid through 54 EFPY. As discussed in the SE for License Amendment 172, the staff determined that the 54 EFPY PTLR for MNGP was correctly developed using the NRC-approved methodology described in SIR-05-044-A, Revision 0. The NRC staff's SE for SIR-05-044-A, Revision 0, which is included with the topical report, describes how the SIR-05-044-A methodology satisfies the seven technical criteria established in NRC GL 96-03, "Relocation of Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits." The staff's SE for License Amendment 172 also describes how the MNGP PTLR was found to satisfy the seven criteria for an acceptable plant-specific PTLR described in GL 96-03. Therefore, the NRC staff found that the implementation of the MNGP PTLR is acceptable. The implementation of the PTLR allows for the P-T limit curves to be managed and updated, as necessary, in accordance with the TS administrative controls without the need for a license amendment under 10 CFR 50.90. With regard to the actual 54 EFPY P-T limit curves established in the PTLR, the staff performed an independent evaluation of the curves. Based on its independent evaluation, the staff verified that the 54 EFPY P-T limit curves were appropriately developed using the SIR-05-044-A methodology and that they satisfied the criteria of ASME Code, Section XI, Appendix G, as required by 10 CFR Part 50, Appendix G. As part of its independent evaluation, the NRC staff also confirmed that the 54 EFPY P-T limit curves are bounding for all ferritic reactor coolant pressure boundary (RCPB) components and that the curves were developed taking into consideration RV beltline and non-beltline components, including stress concentrators (e.g., RV nozzles) outside of the RV beltline shell region, as required by 10 CFR Part 50, Appendix G.

As further discussed in the NRC staff's SE for License Amendment 172, the neutron fluence used for generating the 54 EFPY P-T limit curves was calculated using the NRC-approved methods documented in NEDO-32983-A. However, as also documented in the License Amendment 172 SE, the NRC staff indicated that the neutron fluence calculations used as the basis for the new 54 EFPY P-T limit curves are acceptable insofar as they conservatively reflect currently licensed plant operation. In its SE supporting License Amendment 172, the NRC staff made no statements and drew no conclusions regarding the acceptability of the 54 EFPY neutron fluence values and corresponding P-T limit curves for EPU conditions.

By letter dated May 13, 2013 (ADAMS Accession ML13134A301), the licensee provided supplemental information concerning License Amendment 172 and its application to the EPU LAR. The licensee's May 13, 2013, letter included information for demonstrating that the P-T limit curves approved in License Amendment 172 are bounding for EPU conditions, and are therefore acceptable for operation under EPU conditions through the end of the period of extended operation (54 EFPY). For the P-T limit curves approved in License Amendment 172, the licensee stated in its May 13, 2013, letter that these curves were generated using 54 EFPY neutron fluence values that account for EPU conditions.

In its January 20, 2012, LAR to revise and relocate the P-T limit curves to a PTLR, the licensee revised the projected 54 EFPY peak fluence value for the lower intermediate shell plates at the 1/4T location to 3.29 x 1018 n/cm^2 (E > 1.0 MeV). The licensee stated that this fluence estimate includes the effects of the EPU and a 1.3 conservative multiplier to account for future cycle-to-cycle variation. The licensee also stated that the fluence values had been determined in accordance with NEDO-32983-A, "General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluations," which is NRC-approved and adherent to RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence." Since the licensee's fluence projections account for the EPU, include a 1.3 conservative factor to account for future operation, and adhere to RG 1.190 guidance, the NRC determined that the calculations are acceptable.

Based on the above finding, that the 54 EFPY neutron fluence projections provided in the January 20, 2012, LAR, that were used as the basis for generating the P-T limits as approved in License Amendment 172 are acceptable for EPU conditions, the NRC staff determined that the 54 EFPY ART values and supporting calculations provided in the licensee's January 20, 2012, LAR submittal and accompanying PTLR are also acceptable for EPU conditions. The maximum projected ART value at the 1/4T location in the RV, accounting for EPU conditions, occurs at the Lower-Intermediate Shell Plates 1-14 (Heat No. C2220-1) and 1-15 (Heat No. C2220-2). The NRC staff's ART values for these limiting beltline materials at the 1/4T location are consistent with the values of 147.4 °F, 156.0 °F, and 186.6 °F, reported for the lower intermediate shell plates (Heat No. C2220-1 and C2220-2) for 36, 40, and 54 EFPY, respectively.

Therefore, since the 54 EFPY P-T limit curves approved by the NRC staff in License Amendment 172 are based on ART values that are acceptable for EPU conditions; that the curves were generated using the NRC-approved methodology documented in SIR-04-044-A; and they were found by the NRC staff to be in compliance with the requirements of 10 CFR Part 50, Appendix G, the NRC staff finds that the current P-T limit curves, as established in the MNGP PTLR, and as approved through the issuance of License Amendment 172, are also valid for EPU conditions. Thus, the NRC staff determined that the licensee has adequately addressed the impact of the EPU on the 54 EFPY ART values and P-T limit curves.

RV Circumferential Weld Properties

The ASME Code, Section XI, Table IWB-2500-1, requires inspection of all RV welds at regular intervals. In a relief request dated July 27, 2001 (Reference 89), the NRC staff granted the licensee relief from performing ASME Code, Section XI-required examinations of the MNGP RV circumferential welds for the original 40-year licensed operating period, under pre-EPU operating conditions. The basis for this relief was the BWRVIP-05 topical report, "BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendation (BWRVIP-05)," in which the BWRVIP Committee concluded that the conditional failure probabilities for BWR RV circumferential shell welds are orders of magnitude lower than those of the axial shell welds. The NRC staff evaluated the BWRVIP-05 topical report and found that BWR licensees may utilize the report as a technical basis for requesting relief from circumferential shell weld examinations if the licensees demonstrate that their plant-specific RV circumferential shell weld parameters are bounded by those in the BWRVIP-05 report. The NRC staff evaluation of the BWRVIP-05 report is discussed in the NRC staff's final SER concerning the BWRVIP-05 report, enclosed in a letter dated July 30, 1998, from Mr. G.C. Lainas (NRC) to Mr. C. Terry (BWRVIP Chairman). The MNGP RV circumferential weld parameters under pre-EPU operating conditions were found to be bounded by the parameters used in the subject topical report; this finding was used as the basis for granting relief to the licensee from performing volumetric examinations of the RV circumferential welds for the remainder of the original 40-year licensed operating term.

In its final SER for the MNGP license renewal application, the NRC staff documented its review of the Time-Limited Aging Analysis (TLAA) of the RV circumferential weld properties for the extended licensed operating period (54 EFPY) at MNGP for the same pre-EPU operating conditions discussed above. The NRC staff found in its evaluation that the MNGP RV circumferential weld parameters will continue to be bounded by the BWRVIP-05 parameters discussed above. Furthermore the NRC staff found that the MNGP RV circumferential welds will remain bounded by the NRC circumferential weld conditional failure probability analysis from Appendix A to BWRVIP-74-A under pre-EPU conditions, which applied the BWRVIP-05 analysis to extended licensed operating periods at BWRs applying for license renewal. The NRC staff noted in its evaluation of this TLAA that the licensee would still need to apply for plant-specific relief from ASME Code, Section XI, circumferential shell weld volumetric examination requirements for the extended licensed operating period (54 EFPY) at MNGP, prior to the beginning of the period of extended operation.

The licensee calculated EOL (54 EFPY) mean RT_{NDT} values for the MNGP RV circumferential welds to determine whether these welds will remain bounded by the NRC circumferential weld conditional failure probability analysis from Appendix A to BWRVIP-74-A after implementation of the proposed EPU. These calculations were provided in Table 2.1-2 of the MNGP EPU Licensing Report. The mean RT_{NDT} value for this analysis is calculated using the peak neutron fluence at the clad/base metal interface of the RV and is equal to the initial RT_{NDT} value for the material plus the shift in the RT_{NDT} value (ΔRT_{NDT}) due to neutron irradiation. The licensee calculated a 54 EFPY mean RT_{NDT} value of 55.8°F for the RV circumferential welds. The NRC staff independently confirmed the validity of this calculation. This value is bounded by the 64 EFPY limits from the NRC analysis in Appendix A to BWRVIP-74 for welds fabricated by Chicago Bridge and Iron (the RV manufacturer for MNGP). Specifically, the NRC analysis in Appendix A to BWRVIP-74 indicates that a circumferential weld mean RT_{NDT} value of 70.6°F will result in a weld failure probability of 1.78×10^{-5} per reactor operating year at 64 EFPY.

Therefore, the NRC staff found that the MNGP RV circumferential weld properties are acceptable for plant operation through 54 EFPY, accounting for EPU conditions. The licensee will need to apply for specific relief from performing the ASME Code, Section XI-required, circumferential weld volumetric examinations for the period of extended licensed operation accounting for EPU conditions.

Conclusion

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed EPU on the USE, P-T limits, and RV circumferential weld properties. The NRC staff concludes that the licensee has adequately addressed the impact of the EPU on the MNGP USE, P-T limits, and RV circumferential weld properties. Specifically, the NRC staff finds that: (1) the MNGP RV beltline materials will remain acceptable, with respect to the USE, under EPU conditions, through the expiration of the extended operating license for the facility (54 EFPY); (2) the licensee has addressed the impact of the EPU on the ART values for the RV beltline materials and has submitted revised P-T limits that are valid for operation under the proposed EPU conditions; and (3) the RV circumferential weld properties will remain bounded by the NRC failure probability analysis from Appendix A to BWRVIP 74-A under EPU conditions through 54 EFPY. Based on this assessment, the NRC staff concludes that MNGP will continue to meet the requirements set forth above following implementation of the proposed EPU.

2.1.3 Reactor Internal and Core Support Materials

Regulatory Evaluation

The reactor internal components include structures, systems, and components (SSCs) that perform safety functions or whose failure could affect safety functions performed by other SSCs. These safety functions include reactivity monitoring and control, core cooling, and fission product confinement (within both the fuel cladding and the reactor coolant system (RCS)). The NRC staff reviewed the material specifications, mechanical properties, welds, weld controls, nondestructive examination procedures, corrosion resistance, and susceptibility to degradation for these components. The NRC's acceptance criteria for reactor internal and core support materials are based on GDC-1, "Quality standards and records," and 10 CFR 50.55a., "Codes and standards." Specific review criteria are contained in SRP Section 4.5.2.

Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company, the predecessor to NSPM, performed a comparative evaluation of the design basis of MNGP with the AEC proposed GDC of 1967. The MNGP comparative evaluation to the 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") associated with this review is contained in Appendix E of the MNGP USAR: draft GDC-1 and draft GDC-5.

Technical Evaluation

The licensee discussed the impact of the EPU on the structural integrity of the MNGP reactor internal components in Sections 2.1.3 and 2.1.4 of the MNGP EPU Licensing Report, included in the November 5, 2008, application. In this section the licensee assessed the reactor internal components and found them acceptable for continued operation through the end of the extended licensed operating period (54 EFPY) under EPU conditions.

The licensee's reactor internal and core support materials evaluation addresses the material specifications and mechanical properties, welds, weld controls, nondestructive examination procedures, corrosion resistance, and susceptibility to degradation of the reactor internal and core supports. The licensee stated that the evaluation indicated that the reactor internal and core support materials will continue to be acceptable under EPU conditions and will continue to meet the requirements of the current licensing basis and 10 CFR 50.55a.

The licensee discussed the potential for irradiation-assisted stress corrosion cracking (IASCC) in the subject components. The licensee stated that the increased neutron fluence resulting from the EPU can create the potential for additional IASCC susceptibility in these components. To address this potential, the licensee has a procedurally controlled program for the augmented nondestructive examination (NDE) of selected reactor internal components in order to ensure their continued structural integrity. The inspection techniques utilized are primarily for the detection and characterization of service-induced, surface connected planar discontinuities, such as intergranular stress corrosion cracking (IGSCC) and IASCC in welds and adjacent base material. MNGP has implemented the BWRVIP augmented inspection program for reactor internal components. The inspection programs recommended by BWRVIP-25, "BWR Vessel and Internals Project, BWR Core Plate Inspection and Flaw Evaluation Guidelines (BWRVIP-25)," dated December 1996; BWRIP-26, "BWR Vessel and Internals Project, BWR Top Guide Inspection and Flaw Evaluation Guidelines (BWRVIP-26)," dated November 2004; and BWRVIP-47, "BWR Vessel and Internals Project, BWR Lower Plenum Inspection and Flaw Evaluation Guidelines (BWRVIP-47)," dated November 2004, consider the effects of fluence on applicable components and are based on component configuration and field experience.

Components selected for inspection include those that are identified as susceptible to in-service degradation, and augmented examination is conducted for verification of structural integrity. These components have been identified through the review of NRC Inspection Bulletins, BWRVIP documents, and recommendations provided by General Electric Service Information

Letters. The inspection program provides performance frequency for NDE and associated acceptance criteria. Components inspected include the following:

- Core spray piping
- Core spray spargers
- Core shroud and core shroud support
- Jet pumps and associated components
- Core plate
- Top guide
- Standby liquid control system
- Control rod drive guide tubes
- Reactor vessel internal diameter attachment welds
- Instrument penetrations
- Steam dryer

The licensee stated that fluence calculations performed at EPU conditions indicate that only the top guide and core shroud will exceed the 5 x 10^{20} n/cm^2 (E > 1 MeV) fluence threshold value for IASCC susceptibility at 54 EFPY. The core plate fluence was calculated to be 4.43 x 10^{20} n/cm^2 (E > 1 MeV) at 54 EFPY and, as such, remains beneath the IASCC threshold.

Continued implementation of the current inspection program assures the prompt identification of any degradation of reactor internal components after implementation of the EPU. Additionally, MNGP utilizes hydrogen water chemistry application to mitigate the potential for IGSCC and IASCC in reactor internal components. RV water chemistry conditions are also maintained consistent with the Electric Power Research Institute (EPRI) and established industry guidelines.

The licensee concluded that the peak fluence increase experienced by the reactor internal components as a result of the EPU does not represent a significant increase in the potential for IASCC. The current inspection programs for the reactor internal components at MNGP are adequate to manage any potential service-induced degradation under EPU conditions. The NRC staff found that the licensee performed an adequate assessment of the reactor internal components under EPU conditions and that the licensee's implementation of the BWRVIP programs for inspection and flaw evaluation of the reactor internal components will ensure that the effects of aging are adequately managed under EPU conditions for the extended period of operation at MNGP.

Conclusion

The NRC staff concludes that the licensee has demonstrated that the reactor internal components will continue to meet the regulatory requirements set forth above following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the structural integrity of the reactor vessel components.

2.1.4 Reactor Coolant Pressure Boundary Materials

Regulatory Evaluation

The RCPB defines the boundary of systems and components containing the high-pressure fluids produced in the reactor. The NRC staff's review of RCPB materials covered their specifications, compatibility with the reactor coolant, fabrication and processing, susceptibility to degradation, and degradation management programs.

The NRC's acceptance criteria for RCPB materials are based on: (1) 10 CFR 50.55a and draft GDC-1 and 5, insofar as they require that SSCs important to safety be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety functions to be performed; (2) draft GDC-40 and 42, insofar as they require that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents; (3) draft GDC-9 and 33, insofar as they require that the RCPB be designed, fabricated, erected, and tested so as to have an extremely low probability of rapidly propagating fracture; (4) draft GDC-33, 34, and 35, insofar as they require that the RCPB be designed with margin sufficient to assure that, under specified conditions, it will behave in a non-brittle manner and the probability of a rapidly propagating fracture is minimized; and (5) 10 CFR Part 50, Appendix G, " Fracture Toughness Requirements," which specifies fracture toughness requirements for ferritic components of the RCPB. Specific review criteria are contained in SRP Section 5.2.3 and other guidance provided in Matrix 1 of RS-001. Additional review guidance for IGSCC is contained in Generic Letter (GL) 88-01 and NUREG-0313, as modified by BWRVIP-75-A, "Technical Basis for Revisions to GL 88-01 Inspection Schedules." Additional review guidance for thermal embrittlement of cast austenitic stainless steel components is contained in a letter from C. Grimes (NRC), to D. Walters (NEI), dated May 19, 2000.

Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company, the predecessor to NSPM, performed a comparative evaluation of the design basis of MNGP with the AEC proposed GDC of 1967. The MNGP comparative evaluation to the 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") associated with this review is contained in Appendix E of the MNGP USAR: draft GDC-9, draft GDC-33, draft GDC-34, draft GDC-35, draft GDC-40, and draft GDC-42.

Technical Evaluation

The RCPB piping at MNGP that was evaluated for EPU included the following systems: reactor recirculation system (RRS), control rod drive (CRD) system, standby liquid control (SLC) system, reactor pressure vessel (RPV) bottom head drain line, main steam (MS) and attached branch piping, safety relief valve discharge line (SRVDL), reactor core isolation cooling (RCIC), high-pressure coolant injection (HPCI), RPV vent, main steam isolation valve (MSIV) drains, feedwater (FW), RPV nozzles, and MSIVs. The licensee stated that the RRS, CRD system, SLC system injection line and the RPV bottom head drain line were dispositioned in the CLTR as unaffected by EPU, as these systems experience an insignificant increase in temperature, pressure, flow rate, and mechanical loading. The NRC staff finds the licensee's conclusion acceptable because the CLTR has previously been reviewed and approved by the NRC staff.

In its review of Section 3.5.1, "Reactor Coolant Pressure Boundary Piping," of GE LTR NEDC-33006P, Revision 1, the NRC staff set forth an Action Item which stated that power uprate applicants must identify all other than Category "A" materials, as defined in NUREG-0313, Revision 2, that exist in its RCPB piping, and discuss the adequacy of the augmented inspection programs in light of the power uprate on a plant-specific basis. In a May 29, 2008, e-mail (Reference 72) requesting additional information on the licensee's original EPU application of

March 31, 2008, the NRC staff requested the licensee to provide a response to several questions asked by the NRC staff in previous EPU applications.

The licensee responded to those questions in Enclosure 1 of its letter dated June 12, 2008 (Reference 73). These responses were then included in the MNGP's November 5, 2008, application. The licensee stated that the materials of construction for the RCPB piping and safe-end materials at MNGP are identified in Appendices A and B of the NRC staff's evaluation of MNGP response to GL 88-01. The NRC SE noted that all welds in the RCPB within the scope established by GL 88-01 are Category "A." Therefore, the licensee indicated, all RCPB welds at MNGP are resistant to sensitization and IGSCC.

The licensee indicated that since all subject welds at MNGP meet NUREG-0313, Rev. 2, Category "A" and are considered resistant to IGSCC, there are no augmented inspection programs implemented at MNGP. The licensee continued by indicating that the current inservice inspection (ISI) program examinations are adequate for the configuration and degradation mechanisms present.

The licensee indicated that no weld overlays have been installed to mitigate flaws within the RCPB. The licensee further stated that no ASME flaw evaluations have been performed on components within the RCPB as a result of indications discovered during ISI examinations. The NRC staff agrees that since all subject welds are Category "A" are resistant to IGSCC, the current ISI program examinations are adequate.

The licensee described several mitigation processes that have been applied to MNGP to reduce the RCPB component's susceptibility to IGSCC and ensure the continued acceptable performance of the RCPB welds. These processes include: the materials of construction, the use of solution heat treatment, induction heating stress improvement (IHSI), or corrosion resistant cladding (CRC), and the presence of hydrogen water chemistry (HWC) with electrochemical potential (ECP) verified by BWR Vessel and Internals Application (BWRVIA) modeling.

The licensee stated that the NRC staff's letter to MNGP, "Monticello Nuclear Generating Plant-Staff Evaluation of Response to Generic Letter 88-01," dated December 7, 1989, describes the RCPB welds that were solution heat treated or were stress improved using the IHSI. Corrosion resistant cladding was applied to the internal surfaces of the welds in the head spray nozzles and the head vent nozzles. The flow, pressure, temperature, and mechanical loading for most of the RCPB piping systems do not increase for EPU. Consequently, the licensee stated that there are no changes in stress. Thus, construction processes such as solution heat treatment, IHSI or CRC are not impacted by EPU conditions.

MNGP is currently operating with HWC. The licensee stated that the HWC system reduces the susceptibility of RCPB components to IGSCC in the primary system piping and improves the resistance to IGSCC in vessel internal components. Additionally, the implementation of HWC further reduces the probability of degradation of pressure boundary welds to environmental effects. The licensee stated that the HWC system was installed in accordance with the recommendations of the BWROG, "Guidelines for Permanent BWR Hydrogen Water Chemistry Installation – 1987 Revision." MNGP is a Category 2 plant using moderate HWC. Category 2 plants use the BWRVIP, BWRVIA version 2.0, for Radiolysis and ECP Analysis (BWRVIP-112) model to estimate the total oxidant and ECP at various locations. Hydrogen injection rates will be increased to maintain hydrogen concentration in feedwater at a constant level and maintain

ECP within acceptable limits. ECP is verified at MNGP to be <-330 millivolts (mV) standard hydrogen electrode (SHE), which provides margin to the IGSCC mitigation value of -230 mV SHE. The licensee indicated that these actions will ensure that EPU will not affect the water chemistry controls used for IGSCC mitigation. Lastly, the licensee indicated that the ECP probes have not been used in the recent past within MNGP's RCPB. However, the mitigation processes described above ensure the continued acceptable performance of the RCPB welds.

As set forth above, the NRC staff finds that the licensee has taken comprehensive measures at MNGP to mitigate the RCPB components' susceptibility to IGSCC and ensure the continued acceptable performance of the RCPB welds.

Conclusion

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed EPU on the susceptibility of RCPB materials to known degradation mechanisms and concludes that the licensee has identified appropriate degradation management programs to address the effects of changes in system operating temperature on the integrity of RCPB materials. The NRC staff further concludes that the licensee has demonstrated that the RCPB materials will continue to be acceptable following implementation of the proposed EPU and will continue to meet the regulatory requirements set forth above. Therefore, the NRC staff finds the proposed EPU acceptable with respect to RCPB materials.

2.1.5 Protective Coating Systems (Paints) – Organic Materials

The NRC staff reviewed the information provided in the licensee's November 5, 2009, application, in addition to information provided by the licensee in letters dated May 28, 2008, and January 21, 2013 (References 74 and 75, respectively).

Regulatory Evaluation

Protective coating systems (paints) protect the surfaces of facilities and equipment from corrosion and radionuclide contamination. The coatings also provide wear protection during plant operation and maintenance activities. The NRC staff's review covered protective coating systems used inside containment, including the coating's suitability for, and stability under, design-basis loss-of-coolant accident (DBLOCA) conditions, considering radiation and chemical effects. The NRC's acceptance criteria for protective coating systems are based on the following: (1) 10 CFR Part 50, Appendix B, "Quality Assurance Criteria For Nuclear Power Plants and Fuel Reprocessing Plants," which covers quality assurance requirements for design, fabrication, and construction of safety-related structures, systems, and components (SSCs); and (2) RG 1.54, Revision 1, "Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants," dated July 2000, which covers application and performance monitoring of coatings in nuclear power plants. Specific review criteria are contained in SRP Section 6.1.2, "Protective Coating Systems (Paints) – Organic Materials Review Responsibilities."

Technical Evaluation

The licensee stated that the protective systems used inside the containment were evaluated for their continued suitability for, and stability under, DBLOCA conditions. The evaluation considered radiation and chemical effects at EPU conditions. In its May 28, 2008, letter regarding coating qualification parameters, the licensee provided information illustrating that the post-LOCA containment environmental conditions, such as temperature, pressure, and radiation, do not significantly change as a result of the EPU. The licensee stated that the MNGP Service Level 1 protective coatings are subject to requirements of the ANSI N101.4-1972 to the extent specified in ANSI N18.7 and as modified by RG 1.54, "Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants," dated June 1973. The licensee stated that the Service Level 1 protective coating systems for the MNGP drywell are qualified to a peak temperature of 320°F. Under EPU conditions, the peak DBLOCA wall temperature is calculated to be 278°F and, therefore, the coating systems for the drywell would still perform their function and remain bounded for design-basis accident (DBA) conditions. Further, the licensee stated that the Service Level 1 protective coating systems, CZ11/368WG and 368WG/368WG, in the MNGP torus are qualified to peak temperatures of 340°F and 281°F, respectively. Under EPU conditions, the peak DBLOCA temperature is calculated to be 207°F, and therefore the coating systems in the torus would still perform their function and remain bounded for DBA conditions.

The NRC staff has reviewed the licensee's evaluation and verified that the applicable regulatory guidance was followed. The NRC staff concurs that the post-LOCA containment environmental conditions do not significantly change under EPU conditions, that the chemical constituency does not change under EPU conditions, and that the licensee has demonstrated the protective coating systems remain acceptable for EPU operation.

Conclusion

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed EPU on protective coating systems. The NRC staff concludes that the licensee has appropriately addressed the changes in conditions following a DBLOCA and their effects on the protective coatings. The NRC staff further concludes that the licensee has demonstrated that the protective coatings will continue to be acceptable following implementation of the proposed EPU. Specifically, the protective coatings will continue to meet the requirements of 10 CFR Part 50, Appendix B. Therefore, the NRC staff finds the proposed EPU acceptable with respect to protective coating systems.

2.1.6 Flow-Accelerated Corrosion

Regulatory Evaluation

Flow-accelerated corrosion (FAC) is a corrosion mechanism occurring in carbon steel components exposed to flowing single-phase or two-phase water. Components made from stainless steel are immune to FAC, and FAC is significantly reduced in components containing small amounts of chromium or molybdenum. The rates of material loss due to FAC depend on flow velocity, fluid temperature, steam quality, oxygen content, and pH (potential of hydrogen, a numerical measure of the acidity or alkalinity of a solution). During plant operation, flexibility to control these parameters to minimize FAC is limited. Loss of material by FAC will, therefore, occur. The NRC staff has reviewed the effects of the proposed EPU on FAC and the adequacy of the licensee's FAC program. The intent of the FAC program is to predict the rate of loss so that repair or replacement of damaged components can be made before they reach critical thickness. The licensee's FAC program is based on NUREG-1344, "Erosion/Corrosion-Induced Pipe Wall Thinning in U.S. Nuclear Power Plants," dated April 1989; NRC Generic Letter 89-08 (GL 89-08), "Erosion/Corrosion - Induced Pipe Wall Thinning," dated May 2, 1989; and the guidelines in EPRI Report NSAC-202L-R2, "Recommendations for an Effective Flow-

Accelerated Corrosion Program," dated April 1999. It consists of predicting loss of material using the CHECWORKS™ (Version 2.1) FAC computer code, visual inspection, and volumetric examination of the affected components. The NRC's acceptance criteria are based on the structural evaluation of the minimum acceptable wall thickness for the components undergoing degradation by FAC.

Technical Evaluation

The licensee stated that the FAC program implemented at MNGP utilizes selective component inspections, and measures the confidence in the condition of systems susceptible to FAC. This method of selective inspection is based upon guidelines developed by EPRI using allowable material values from the ASME Code. The method of inspection currently used will be the same method of inspection used after implementation of the EPU. The licensee performed piping replacements to maintain suitable design margins, and has used FAC-resistant replacement materials to mitigate future occurrences of FAC as part of the modification process.

The licensee stated that a CHECWORKS™ FAC model has been developed for MNGP to predict the FAC wear rate and the remaining service life for each piping component. The CHECWORKS™ FAC model is updated after each refueling outage. The FAC models are also used to identify FAC examination locations for the outage examination list and uses empirical data input to the model.

The licensee stated that the EPU will affect some variables that influence FAC, such as moisture content, water chemistry, temperature, oxygen, material composition, and flow path geometry and velocity. The licensee indicated that all of the affected variables are expected to remain within the CHECWORKS™ FAC model parameter bounds. The licensee anticipates that the EPU operating conditions may result in the need for additional FAC monitoring points. The CHECWORKS™ FAC modeling techniques allow for additional monitoring points required for the EPU.

In its May 28, 2008, letter, the licensee provided a sample list of components for which wall thinning was predicted and measured in order to assess the accuracy of the FAC predictions from CHECWORKS™. The results show that the CHECWORKS™ FAC model predicts the measured thickness within 5 percent of the actual average thickness. In addition, the licensee provided a table of the most susceptible systems and the predicted increase in wear rate associated with each system. The system that is predicted to experience the greatest increase in wear rate as a result of the EPU is a section of straight pipe off of the heater extraction steam inlet nozzles. The increase in predicted wear associated with the piping is 29 percent, and is due to increases in pressure, temperature, enthalpy, flow, and oxygen, and a decrease in quality. The licensee additionally states that, at EPU conditions, main steam and feedwater flow rates do not significantly affect the potential for FAC.

The NRC staff has reviewed the licensee's evaluation and has verified that the applicable regulatory guidance was followed. The licensee has demonstrated that the FAC program is adequate for managing the potential effects on the NSSS, turbine generator, and balance-of-plant (BOP) components. The NRC staff concurs that the MNGP FAC program is adequate in predicting the rate of material loss. Thus, repair or replacement of damaged components can be made before they reach a critical thickness.

Conclusion

The NRC staff has reviewed the licensee's evaluation of the effect of the proposed EPU on the FAC analysis for the plant and concludes that the licensee has adequately addressed changes in the plant operating conditions on the FAC analysis. The licensee has demonstrated that the updated analyses will predict the loss of material by FAC, and allow for timely repair or replacement of degraded components following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to FAC.

2.1.7 Reactor Water Cleanup System

Regulatory Evaluation

The reactor water cleanup (RWCU) system provides a means for maintaining reactor water quality by filtration and ion exchange and a path for removal of reactor coolant when necessary. Portions of the RWCU comprise the RCPB. The NRC staff's review of the RWCU included component design parameters for flow, temperature, pressure, heat removal capability, impurity removal capability, and the instrumentation and process controls for proper system operation and isolation. The review consisted of evaluating the adequacy of the plant's technical specifications (TSs) in these areas under proposed EPU conditions. The NRC's acceptance criteria for the RWCU are based on the following: (1) 10 CFR Part 50, Appendix A, GDC-14, "Reactor Coolant Pressure Boundary," as it requires that the RCPB be designed, fabricated, erected, and tested to have an extremely low probability of rapidly propagating fracture; (2) GDC-60, "Control of Releases of Radioactive Materials to the Environment," as it requires that the plant design include means to control the release of radioactive effluents; and (3) GDC-61, "Fuel Storage and Handling and Radioactivity Control," as it requires systems that contain radioactivity to be designed with appropriate confinement. Specific review criteria are contained in SRP Section 5.4.8, "Reactor Water Cleanup System (BWR)."

Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company, the predecessor to NSPM, performed a comparative evaluation of the design basis of MNGP with the AEC proposed GDC of 1967. The MNGP comparative evaluation to the 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") associated with this review is contained in Appendix E of the MNGP USAR: draft GDC-9, draft GDC-33, draft GDC-67, draft GDC-68, draft GDC-69 and draft GDC-70.

Technical Evaluation

The RWCU system will operate at a slightly decreased temperature under EPU rated thermal power (RTP). The temperature decrease is less than 1°F from the temperature under the CLTP. Under the lower EPU temperature, the RWCU system is capable of performing its function of removing solid and dissolved impurities from recirculated reactor coolant. The removal process reduces the concentration of radioactive and corrosive species in the reactor coolant.

The RWCU system flow analyzed for the EPU is within the range of 0.8% to 1% of feedwater flow. The flow rate is consistent with the original system specification requirement. The licensee stated that the EPU review included evaluation of water chemistry, heat exchanger performance, pump performance, flow control valve capability, and filter/demineralizer performance. The performance of each of the above mentioned parameters was found to be within the design of the RWCU system at the analyzed flow. No changes to instrumentation are required for the EPU, and no setpoint changes are expected due to the negligible system process parameter changes.

The licensee reviewed the effects of the EPU on the RWCU system functional capabilities and determined that it can adequately perform at the EPU power level with an upgraded RWCU system flow rate of 90,000 pounds-mass per hour (lbm/hr), a 12.5 percent increase from the current RWCU system flow rate. There is a slight increase in the calculated reactor water conductivity from 0.100 microsiemens per centimeter (μS/cm) to 0.115 μS/cm, because of the increase in feedwater flow. As a result of the EPU, the pressure in the feedwater line increases and has a slight effect on the system operating conditions. The estimated increase in these parameters is not significant and sufficient operating margin to the conservative limits remain under the EPU conditions.

The NRC staff has reviewed the licensee's evaluation and verified that the applicable regulatory guidance was followed. The NRC staff concurs that the proposed EPU will introduce only insignificant changes in RWCU system operating parameters, which will not affect satisfactory performance of its intended function.

Conclusion

The NRC staff has reviewed the licensee's evaluation of the effects the proposed EPU will have on the RWCU and concludes that the licensee has adequately addressed changes in impurity levels and pressure, and their effects on the RWCU system. The NRC staff further concludes that the licensee has demonstrated that the RWCU system will continue to be acceptable following implementation of the proposed EPU. Specifically, the RWCU system will continue to meet the regulatory requirements set forth above. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the RWCU system.

2.2 Mechanical and Civil Engineering

2.2.1 Pipe Rupture Locations and Associated Dynamic Effects

Regulatory Evaluation

Structures, systems and components (SSCs) important to safety at nuclear power plants could be impacted by the pipe-whip dynamic effects of a pipe rupture. The NRC staff conducted a review of pipe rupture analyses to ensure that SSCs important to safety at MNGP are adequately protected from the effects of pipe ruptures. The NRC staff's review covered: (1) the implementation of criteria for defining pipe break and crack locations and configurations; (2) the implementation of criteria dealing with special features, such as augmented ISI programs or the use of special protective devices such as pipe-whip restraints; (3) pipe-whip dynamic analyses and results, including the jet thrust and impingement forcing functions and pipe-whip dynamic effects; and (4) the design adequacy of supports for SSCs provided to ensure that the intended design functions of the SSCs will not be impaired to an unacceptable level as a result of pipe-whip or jet impingement loadings. The NRC staff's review focused on the effects that the proposed CPPU may have on items (1) through (4) above.

The NRC staff's acceptance criteria are based on 10 CFR 50, Appendix A, GDC-4, "Environmental and dynamic effects design basis," which requires SSCs important to safety to be designed to accommodate the dynamic effects of a postulated pipe rupture. However, MNGP is not licensed to the Appendix A GDCs or the 1967 AEC proposed GDCs, but its design conforms with the intent of the 1967 AEC draft GDCs. The MNGP principal design criteria are listed in Section 1.2, "Principle Design Criteria," of the USAR. As provided in Appendix E to the USAR, a comparative evaluation indicates that MNGP conforms to the intent of draft GDC-40 and draft GDC-42, which are comparable to the current GDC-4. Specific review criteria are contained in SRP Sections 3.6.1 and 3.6.2 and SRP Branch Technical Positions (BTP) 3-3 and 3-4.

Technical Evaluation

The licensee's review of the effects of CPPU on the postulated pipe rupture locations and associated dynamic effects for MNGP is documented in the MNGP PUSAR, which follows the CPPU approach of the NRC approved GE CLTR. The current licensing basis (CLB) evaluation criteria for high energy line breaks are contained in MNGP USAR Appendix I, "Evaluation Criteria", which specifies that the criteria used for the determination of the high energy lines and the effects of the postulated breaks on these lines on safe shutdown equipment are addressed in the AEC's December 18, 1972, Giambusso letter, as clarified by SRP Section 3.6.1, Rev. 1, dated July 1981; SRP Section 3.6.2, Rev. 1, dated July 1981; and NRC GL 87-11, which allows elimination of the arbitrary intermediate pipe breaks. These criteria are utilized as the basis for the determination of the high energy line break locations, and the evaluation of their associated dynamic effects on safe shutdown equipment.

The NRC staff reviewed the PUSAR and the licensee's response to the staff's request for additional information (RAI)-1 (Reference 64). Based on the staff's review, the majority of RCPB piping systems experience no increase in pressure, temperature, flow or mechanical loading for CPPU, except for the main steam and feedwater piping systems, which exhibit flow increases of approximately 15% from CLTP and 23% from original licensed thermal power (OLTP). The staff noted that the licensee's PUSAR, Section 2.2.1, identifies that corrective actions are underway to perform high energy line break (HELB) analysis upgrades at MNGP due to changes in pipe break methodology. The NRC staff requested that the licensee clearly identify the changes in pipe break methodology at MNGP. In summary, the licensee's response to the NRC staff's RAI-3 and RAI-6 in Reference 64 clarified that there is no change to the existing pipe break methodology and criteria at MNGP, and that the changes involve re-analyses of breaks using more conservative assumptions of mass and energy (M&E) release at pipe break locations and upgrade of the GOTHIC computer code utilized for the analysis of M&E release, from GOTHIC Version 4.0 to later versions of GOTHIC. GOTHIC is a state-of-the-art general purpose thermal-hydraulics computer code for performing containment analyses. GOTHIC is widely used by the nuclear industry and applications of this code have been previously approved by the NRC staff on a case-by-case basis.

The NRC staff's further review of the M&E release at pipe break locations is contained in Section 2.6, Containment Review Considerations. Non-conservative HELB analyses at MNGP were discovered during the preparation for the EPU license amendment request (LAR). This resulted in corrective actions to perform HELB analysis upgrades using more conservative assumptions of M&E release at pipe break locations. HELB program deficiencies were documented in the MNGP corrective action program (action request AR1131913). The licensee submitted a letter dated April 6, 2010 (Reference 66), in response to a conference call with the NRC staff, stating that it is not seeking NRC approval with the EPU LAR for changes in the HELB analysis methods resulting from corrective actions at MNGP. NSPM also stated that changes in HELB analysis methods discussed in the EPU LAR, and any required modifications (if necessary), will be reviewed using the 10 CFR 50.59 process. Also, the licensee stated in its letters dated April 6, 2010, and January 21, 2013, that there was no fundamental change to the HELB methodology, and that the program remains in compliance with both the Giambusso letter and IEB 79-01B, as implemented in the MNGP design and licensing bases. As such, the licensee determined that a 10 CFR 50.59 evaluation was not required for the reconstitution because the HELB program remained in compliance with the Giambusso letter as implemented in the MNGP design and licensing bases without introducing any changes to the existing pipe break methodology.

The NRC staff finds the licensee responses acceptable because the current licensing bases have been utilized for EPU without changes for postulating piping failures.

Steam Line High Energy Line Breaks

The licensee's PUSAR indicated that the main steam piping analysis was not completed and needed further reconciliation (PUSAR pages 2-22, 2-63). The licensee's response to NRC staff's RAI-7 and RAI-17(a) (in Reference 64, with an update in Reference 67) indicates that since submitting the EPU LAR, the MS piping analysis has been completed and the licensee provided stress summary results at EPU conditions which show that the code equation stresses are satisfied for design-basis break limit stress criteria. Therefore, the licensee concluded that there is no new limiting main steam break postulated outside containment based on pipe stress criteria, including EPU conditions. The licensee noted that there are postulated break locations based on configuration (e.g., terminal ends) which have not changed due to the EPU. Section 2.2.1 of the licensee's PUSAR stated that no new break or crack locations are postulated as a result of EPU. In a later calculation, a new high energy crack location was identified. The licensee provided a response (Reference 65), indicating that the newly identified crack has been evaluated and is bounded by an existing HELB calculation. The licensee's response also shows that the above PUSAR statement was revised to indicate that no new limiting break or crack locations are postulated as a result of EPU. The NRC staff finds the licensee's response acceptable, as the newly identified crack has been categorized as non-limiting because it has been evaluated and found to be bounded by an existing HELB. Based on the above review, the NRC staff finds the licensee's responses and evaluations to be acceptable for steam line ruptures.

The licensee evaluated steam line HELB for the main steam, HPCI, and RCIC systems at CPPU. The licensee concluded that the EPU has no effect on the M&E releases due to HELB in the main steam, HPCI, and RCIC systems. The NRC staff finds this acceptable, as it is in accordance with the staff-approved CLTR and that CPPU has no effect on the steam pressure or enthalpy at the postulated break locations.

Liquid Line High Energy Line Breaks

The licensee's PUSAR identifies that increased MS and FW flows may lead to increased break flow rates for liquid line breaks. The licensee re-evaluated the M&E releases at EPU conditions for the following systems: reactor water cleanup (RWCU), FW, condensate, CRD, Standby Liquid Control (SLC), and GEZIP (GE zinc injection process). The PUSAR states that a review of the results from several recent EPU submittals concluded that, in most cases, environmental conditions are bounded by previous analyses, confirming that EPU produces relatively minor effects.

With regard to RWCU HELB evaluation, the PUSAR states the following:

> New mass and energy release calculations considered additional blowdown sources that had not been considered in the previous 1996 analysis. This resulted in an increase in integrated mass release of about 90 percent and an increase in integrated energy release of 63 percent.

In response to NRC staff RAI-6(a) requesting clarification (Reference 64), the licensee stated that the 90 percent and 63 percent increases are not due to the proposed EPU but rather due to change in assumptions of HELB M&E re-analyses (see above). The licensee also stated in its response that if the CLTP HELB cases were run using similar assumptions, the changes in M&E releases would have been minor as a result of EPU. The licensee, in response to staff RAI-5 with regard to RWCU, stated that the CLTP analysis of RWCU HELBs evaluated the effects from the terminal end break and crack case at the inlet to the RWCU heat exchanger. These were considered the bounding cases and other cases were not run at CLTP conditions. For EPU, HELB locations were evaluated covering all possible breaks and cracks. The licensee, in its response to RAI-8, also indicated that these HELB locations have not changed for EPU because there is no change in RWCU temperature or design pressure due to EPU that would affect the pipe break postulation stress evaluations. The NRC staff noted that the M&E releases at the RWCU pipe break location mentioned above are much greater than previously analyzed and requested a justification for their effects on plant SSCs. The licensee, in response to staff's RAI-6(b), indicated that re-analyses of all HELBs have been completed and changes in M&E releases have been evaluated using the GOTHIC code. This allowed a determination of time histories for all plant areas to evaluate effects on temperature, pressure and flooding. The licensee also considered differential pressures due to HELB M&E releases between plant areas and verified that acceptable margins exist for plant structures including block walls. In its updated RAI-responses (RAI-3(a) and RAI-6(b)) (Reference 67), the licensee stated that the effects of changes to temperature, pressure and flooding have been evaluated for impact on the environmental qualification (EQ) of equipment and evaluation of affected EQ components has been completed. The NRC staff finds the licensee's responses to be acceptable, as the effect of M&E releases at EPU conditions, including those of differential pressure, have been evaluated and found to have no adverse effect on the structural integrity of plant SSCs.

The licensee also identified in its PUSAR submittal (Reference 7), and in response to NRC staff RAIs in Reference 64, that the configuration of FW and condensate piping from the condensate pump suction to the containment isolation valves will be changed subject to the FW and condensate pump, and heater replacement modifications supporting the EPU. The staff notes that piping configuration changes could potentially affect pipe stresses and pipe supports. In its response to RAI-7 (Reference 64), the licensee indicated that re-analysis of piping and affected SSCs due to the FW and condensate pump and heater modifications have not been completed. Since the submittal of References 7 and 68, the licensee has completed the structural evaluations for the FW and condensate modifications which, as indicated in References 71 and 72, resulted in one new postulated crack on the 14"-line at the inlet to the #14 FW heater. According to the licensee, this new crack is characterized as limited for flooding, does not impact EQ equipment, and does not result in any new jet impingement targets. The structural review of the FW piping due to the EPU modifications is provided in Section 2.2.2, Balance-of-Plant Piping, Components, and Supports, on page 30 of this SE.

For further review of the evaluations of the effects of postulated pipe breaks, including that of M&E releases and flooding at postulated pipe rupture (breaks and cracks) locations, see

Section 2.5, "Plant Systems," for outside containment, and Section 2.6, "Containment Review Considerations."

Pipe Whip and Jet Impingement

The NRC staff notes that pipe whip and jet impingement loads resulting from high energy pipe breaks are directly proportional to system pressure and pipe break area. According to the CLTR, CPPU has no effect on the steam pressure or enthalpy at the postulated break locations. According to the licensee's evaluations, there are no new limiting steam pipe break or crack locations postulated due to EPU which is a CPPU. Therefore, pressure and break area have not changed for postulated steam pipe failures at EPU conditions. Hence, the MNGP EPU has no effect on steam line breaks pipe whip and jet impingement loads. It can be shown that for EPU, as provided in the licensee's response to staff RAI-1, the FW temperature increases by approximately 5% from both OLTP and CLTP.

The licensee initially reviewed (see Reference 7) existing configuration FW pipe stress calculations and determined that the small increase in temperature will not result in pipe stress levels above the thresholds required for postulating HELBs (except at existing postulated HELBs) and that the EPU FW pressure is bounded by the existing piping analysis (PUSAR pages 2-31 and 2-33). Completion of the FW and condensate piping analysis included evaluation of the dynamic effects, pipe whip, and jet impingement from postulated pipe failures. The piping analysis for the configuration of the FW and condensate piping from the condensate pump suction to the containment isolation valves which needed to be changed due to the FW and condensate pump and heater replacement modifications for the proposed EPU was pending when the EPU LAR was submitted, References 1 and 2. Subsequently, as indicated in References 71 and 72, the licensee completed the pipe stress and pipe support evaluations for this piping. As stated above, a new postulated crack (on the 14"-line at the inlet of the #14 feedwater heater) was identified, which did not result in any new jet impingement targets. As part of the planned FW and condensate EPU modifications, the licensee performed a HELB target impact evaluation.

The licensee also evaluated the torus attached structures. The bounding design-basis accident loss-of-coolant accident (DBA LOCA) hydrodynamic loads, including the pool swell loads, vent thrust loads, condensation oscillation loads and chugging loads were originally defined and evaluated for MNGP. The evaluation of the structures attached to the torus shell, such as piping system, vent penetrations, and valves are based on these bounding DBA LOCA hydrodynamic loads. Because the bounding hydrodynamic loads did not change for EPU, there are no resulting effects on the torus shell attached structures. The licensee also determined that the safety relief valve (SRV) discharge loads used in the existing analyses are not affected by the proposed EPU.

Based on the above review, the NRC staff finds that reasonable assurance has been provided that appropriate protection exists for SSCs important to safety against postulated pipe failures and their associated dynamic effects and effects of DBA LOCA and SRV loads at EPU conditions.

<u>Conclusion</u>

The NRC staff reviewed the licensee's evaluations related to determinations of rupture locations and associated dynamic effects and concludes that the licensee adequately addressed the effects of the proposed EPU on them using current licensing and design basis methods and criteria. The NRC staff further concludes that the licensee has demonstrated that SSCs important to safety will continue to meet the intent of draft GDC-40 and 42 following implementation of the proposed CPPU. Therefore, the NRC staff finds the proposed CPPU acceptable with respect to the determination of rupture locations and dynamic effects associated with the postulated rupture of piping.

2.2.2 Pressure-Retaining Components and Component Supports

Regulatory Evaluation

The NRC staff reviewed the structural integrity of pressure-retaining components (and their supports) designed in accordance with the ASME Code, Section III, Division 1; ASME/ANSI B31.1 and GDCs 1, 2, 4, 14, and 15. The NRC staff's review focused on the effects of the proposed CPPU on the design input parameters and the design-basis loads and load combinations for normal operating, upset, emergency, and faulted conditions.

The NRC staff's review covered: (1) the analyses of flow-induced vibration; and (2) the analytical methodologies, assumptions, ASME Code editions, and computer programs used for these analyses. The staff's review also included a comparison of the resulting stresses and cumulative fatigue usage factors (CUFs) against the code-allowable limits. The NRC's acceptance criteria are based on: (1) GDC-1, "Quality standards and records," insofar as it requires that SSCs important to safety be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety functions to be performed; (2) GDC-2, "Design basis for protection against natural phenomena," insofar as it requires that SSCs important to safety be designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions; (3) GDC-4, "Environmental and dynamic effects design basis," insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents; (4) GDC-14, "Reactor coolant pressure boundary," insofar as it requires that the RCPB be designed, fabricated, erected, and tested so as to have an extremely low probability of rapidly propagating fracture; and (5) GDC-15, "Reactor coolant system design," insofar as it requires that the RCS be designed with margin sufficient to ensure that the design conditions of the RCPB are not exceeded during any condition of normal operation. However, as stated in the introduction, MNGP is not licensed to the Appendix A GDC but its design conforms with the intent of the 1967 AEC draft GDC.

While MNGP is not explicitly licensed to the current General Design Criteria or the 1967 AEC-proposed GDC, the licensee has made a comparison of the current GDC to the applicable AEC-proposed General Design Criteria as described in Appendix E of the USAR. MNGP conforms to the intent of draft GDC- 1, draft GDC-2, draft GDC-5, draft GDC-9, draft GDC-33, draft GDC-40, and draft GDC-42, which are comparable to the GDCs listed above with the exception of current GDC-15. There is no draft GDC directly associated with current GDC-15. According to Section 2.2.2 of the PUSAR, current GDC-15 is applicable to MNGP for certain events as described in USAR Section 14.4. Specific review criteria are contained in SRP Sections 3.9.1, 3.9.2, 3.9.3, and 5.2.1.1; and other guidance provided in RS-001, "Review Standard for Extended Power Uprates".

Technical Evaluation

Nuclear Steam Supply System Piping, Components, and Supports

The Reactor Coolant Pressure Boundary (RCPB) piping consists of a number of safety-related piping subsystems that move fluid through the reactor and other safety systems. The licensee performed structural evaluations for RCPB piping. Specifically, the following items were evaluated by the licensee for EPU:

- FW piping
- MS piping
- MS Safety Relief Valve (SRV) discharge piping (SRVDL)
- Reactor Core Isolation Cooling (RCIC)
- High Pressure Coolant Injection (HPCI)
- RPV vent (RCPB portion)
- MSIV drain piping
- Reactor Water Cleanup (RWCU) piping
- Residual Heat Removal (RHR) Low Pressure Coolant Injection (LPCI) lines
- Core Spray (CS) injection lines.

The following systems were dispositioned in accordance with the CLTR as unaffected by the EPU.

- Reactor Recirculation System (RRS)
- Control Rod Drive (CRD) system
- Standby Liquid Control (SLC) system injection line
- Reactor Pressure Vessel (RPV) bottom head drain line

The licensee's evaluation addressed branch lines, piping supports (snubbers, hangers and struts), nozzles, penetrations, flanges, and valves. The licensee also evaluated the safety-related thermowells in the MS and FW systems and the sample probe in the FW system for CPPU.

The licensee evaluated the above RCPB piping systems in accordance with the methodology documented in the GE NRC-approved topical report CLTR (Reference 8). In response to NRC staff RAI, the licensee verified that all structural evaluations of SSCs, required for EPU, were performed in accordance with the DB codes of record for piping and pipe supports. The licensee's piping evaluation methodology is described in Appendix K of the GE NRC-approved topical report ELTR1 (Reference 9). Appendix K provides guidance in determining pipe stress increases by the use of scaling factors from pressure, temperature, and flow increases for CPPU. Therefore, the NRC staff finds the licensee's methodology acceptable.

The licensee's PUSAR indicates that loadings due to pressures, temperatures, flows and mechanical loads that affect piping and pipe supports do not increase or change at EPU conditions for most of the RCPB piping systems. This assessment is consistent with the staff approved CLTR. In addition, seismic loads are not affected by EPUs and the licensee has determined that the SRV discharge loads are also not affected by the proposed CPPU. The staff finds this acceptable, as it compares well with previous CPPUs the staff has previously reviewed and approved. The licensee reviewed the RRS system, CRD system, SLC system injection line, and RPV bottom head drain line, and dispositioned these in accordance with the CLTR as unaffected by the EPU. The NRC staff agrees with the licensee's disposition as these systems, with the exception of RRS, do not experience a change in temperature, pressure, and flow rate from CLTP to EPU conditions as shown in Section 2.2.2 of the PUSAR. The only change that the RRS system experiences at EPU conditions is a small increase in flow rate of 1.7 percent, which is not significant enough to affect the structural integrity of the RRS system. In its response to staff RAI-15 regarding RHR LPCI, the licensee indicated that the loads used in the current analysis bound those at EPU conditions. Therefore, the structural integrity of RHR LPCI is not affected by EPU. Based on the above, the NRC staff finds that the licensee has provided reasonable assurance that the structural integrity of these RCPB systems will be maintained at EPU conditions.

The licensee indicated in its PUSAR that it evaluated the MS piping, its RPV nozzles, and branch lines connected to the MS line headers (safety/relief valve discharge line (SRVDL), RCIC, HPCI, RPV Vent, MSIV Drain) at EPU conditions. The licensee evaluated the above lines inside containment for piping loads at EPU conditions, including the increased load due to the turbine stop valve (TSV) fast closure transient, in accordance with the DB Code of Record ASME Section III, Division 1, 1977 Edition with Addenda up to and including winter 1978. The licensee concluded in its PUSAR that the calculated pipe stresses in the design analyses satisfy design basis code requirements at EPU conditions with the exception of one small bore branch line that did not meet displacement criteria. The licensee stated in its PUSAR that additional detailed analysis will be performed to qualify this line or the piping modified prior to operation at EPU conditions (see below, licensee's response to staff's RAI).

The NRC staff requested the licensee to submit maximum pipe stress and support evaluation result summaries (including evaluations at critical locations such as nozzles and penetrations) for piping, inside containment, for which design parameters such as temperature, pressure, flow or mechanical loads have been increased due to EPU to validate its conclusion (RAI-13). In its response, the licensee indicated that in addition to the MS piping, the only other piping inside containment for which piping loads are not bounded in the existing design basis analyses is the CS system due to the CS piping analysis temperature increase at EPU by 17.8 percent from OLTP and 7.8 percent from CLTP. The licensee in its response to RAI-15 provided maximum pipe stress and support load summaries for the CS line which show that the calculated stresses and loads meet code allowables at EPU conditions. The licensee noted that the MS line pressures, temperatures, and SRV discharge loads are unaffected by the EPU. The increase in MS flow though (approximately 23 percent from OLTP and 15 percent from CLTP) results in increased loads in the MS piping system due to the TSV fast closure transient. The TSV fast closure loads bound the MSIV closure loads because the TSVs close more rapidly than the MSIVs. Therefore, the licensee reanalyzed the MS system piping using the EPU TSV load case. For the MS inside containment, the licensee in its response to staff's RAI-17 provided maximum pipe stress and support evaluation summaries. The maximum EPU MS pipe stress ratio (calculated over allowable) due to thermal expansion is shown to be 1.00 and 0.99 due to sustained loads load combination from pressure, deadweight and operational basis earthquake. These stress ratios are less than or equal to one and, thus, are acceptable according to the requirement of the Code of Record (stated above). The licensee included evaluation result summaries for pipe support loads and loads at the SRV inlet and outlet flanges, RPV nozzle loads, and containment penetration flued anchor loads. The analysis summaries indicate that the licensee's evaluations for the MS inside containment meet ASME Code of Record and DB allowables and, therefore, are acceptable.

In response to an NRC staff RAI-regarding the above mentioned small bore branch line that did not meet displacement criteria, the licensee stated that this is a 1-inch instrument sensing line which provides a safety-related input function to the high flow Group 1 Containment Isolation logic will automatically isolate the MSIV's in the event of a main steam line break. The licensee indicated that pipe support span field data required to evaluate this line was not available when the PUSAR was submitted. These field data was collected during a subsequent refueling outage and the line has been found to meet code allowables without any modification necessary.

Based on its review, as shown above, the staff finds that the MS piping inside containment including its branch lines and its RPV nozzles, pipe supports and associated components remain structurally adequate for EPU.
With regard to FW inside containment (from outboard IV to the RPV), the licensee indicated in the PUSAR and in response to RAI-14 that design basis loads used in the current CLTP analyses of record (AOR) were modified by increasing piping temperature to reflect piping loads at EPU conditions. Therefore, this section of the FW piping is appropriately analyzed by the EPU.

The licensee dispositioned the MS line flow elements (restrictors) for structural integrity. The licensee's structural integrity review of the MS line flow restrictors is documented in the proprietary portion of Section 2.5.4.1, "Main Steam," of PUSAR which finds that there is no effect on the structural integrity of the MS line flow restrictors due to the EPU. The NRC staff finds the licensee's review of the MS line flow restrictors acceptable, as it is in accordance with the staff approved CLTR which concluded that the main steam flow restrictors are not affected by a CPPU.

The licensee evaluated the flow-induced vibration (FIV) levels associated with the MS and FW piping systems that are projected to increase for CPPU. The NRC staff's evaluation of FIV and power ascension and testing programs for CPPU are documented in Sections 2.2.6 and 2.12 of the staff's SER.

The licensee also evaluated the effects of FIV at EPU conditions on the safety-related MS and FW thermowells and the FW sample probes. The licensee in its response to staff's RAI-28 (Reference 64) provided a discussion and a tabulated summary of the analyses results for these instruments. The calculated EPU stresses due to FIV for fatigue consideration, although higher than CLTP due to EPU higher flows, are well below the material endurance limit of the material for the thermowells and probes. The table indicates that 4,536 psi is the maximum calculated alternating stress from all thermowells and probes which is less than the ASME code fatigue endurance limit of 13,600 psi for austenitic stainless steel materials. The licensee's response to the staff's RAI also reported ratios of the flow induced vortex shedding frequency (Fs) to the natural frequency (Fn) of the instruments. The NRC staff notes that according to industry guidance when the Fs and the Fn are sufficiently close they can become synchronized (the Fs "locks-in" to the structural Fn) and lead in to resonant response which produces large motions. According to industry guidance (also contained in Appendix N of ASME Section III), "lock-in" can be avoided if Fn < 0.7Fs or Fn > 1.3Fs. Review of the response's data shows that the main steam thermowells are of concern at EPU as the periodic Fs appears to be in the range of "lock-in" (Fn = 1.22Fs). As stated in Reference 64, to avoid the potential of the main steam thermowells going into resonance, the licensee plans to reduce the length of the thermowells (by replacing them with shorter ones) and thus increase their Fn or remove the thermowells altogether. In an updated response (Reference 67), the licensee stated that during the spring 2013 refueling outage a plant modification will remove the existing main steam thermowells and install plugs in their place, thus resolving the potential of resonance. The NRC staff finds the licensee's plan to be practical and acceptable.

NRC RIS 2008-30, "Fatigue Analysis of Nuclear Power Plant Components," identified a concern with the simplified single-stress methodology used by some license renewal applicants to perform fatigue calculations, and as input for on-line fatigue monitoring programs, in lieu of the ASME Code, Section III, Subsection NB, Subarticle NB-3200 method which requires consideration of all six stress components. Approval of license renewal for MNGP was issued prior to RIS 2008-30 on November 8, 2006 (see NUREG-1865, Reference 112), the NRC staff requested that the licensee demonstrate compliance with ASME Section III when stress-based fatigue monitoring is utilized. The licensee indicated the following in Item 23 of its January 21, 2013, letter (Reference 76):

> RIS 2008-30 was evaluated in January 2009 for impact on the Monticello Fatigue Monitoring Program. Monticello does not currently use FatiguePro or any other fatigue monitoring software that uses the simplified approach of a single stress component as input to the Green's Function for determining fatigue usage. Monticello performs manual cycle counting in accordance with ASME Code, Section III requirements and all thermal transient calculations used in monitoring fatigue accumulation are derived from the six stress components as required by the ASME Code.

The NRC staff finds the licensee's response acceptable, as it demonstrates compliance with Section III of the ASME Code when performing fatigue evaluations. Thus, there is no impact of RIS 2008-30.

Based on its review above, the NRC staff finds that the licensee has provided reasonable assurance that the structural integrity of the RCPB piping, supports, and associated components will remain structurally adequate for the proposed EPU.

Balance-of-Plant Piping, Components, and Supports

The licensee evaluated the structural integrity of the Balance of Plant (BOP) piping, components and supports to assess the impact of temperature, pressure and flow rate changes that will result due to the implementation of EPU. In response to a staff RAI, the licensee provided the following list of piping systems outside containment for which piping loads due to EPU conditions of temperature, pressure and or flow are not bounded in the existing design basis structural analyses.

> Main steam, extraction steam, feedwater, condensate, torus attached piping, emergency service water, heater drains, cross around piping, cross-around relief valve piping and moisture separator drain lines.

The licensee evaluated the BOP piping and pipe supports in accordance with the current design basis codes (and code year editions and addendums) of record as referenced in the appropriate calculations with the code allowable values and analytical techniques that were used without introducing any new assumptions. The licensee evaluated the BOP EPU affected piping and in Section 2.2 of its PUSAR provided tables with evaluation summary results. The NRC staff reviewed the licensee's evaluation and in its RAI noted that the PUSAR tables present percentage increases for pipe stresses and pipe support loads, varying from 9 to 72 percent increases, due to EPU increases in pressure, temperature or flow, and that these percentages are not indications that piping and pipe supports meet design basis code allowable values. The licensee in its response to staff's RAI-13 provided a tabulated summary of maximum pipe stresses and pipe support loads. The staff reviewed the pipe stress and pipe support load summaries presented in the licensee's response and found that the maximum reported pipe stresses and pipe support loads are within the design basis and Code of Record allowable values and, therefore, are acceptable. The maximum EPU stress ratio (calculated over allowable) for the main steam outside containment is 0.86 (less than the allowable value of one) due to service level B (upset condition) loads at the turbine connection node. The maximum main steam drywell penetration flued head anchor load interaction ratio (IR) (tension/allowable + shear/allowable) is 0.92 (less than the allowable value of one) due to faulted service level D loads which include SSE and pipe break loads. All pipe stress ratios and pipe support load IRs are less than one and, therefore, are acceptable.

Since the submittal of the EPU LAR, which included the licensee's PUSAR, the licensee completed the main steam analysis and found that all piping, pipe components, and supports meet Code of Record allowable values (see response to RAI-12 and RAI-13 in Reference 64). As a result of the torus attached piping analyses for EPU, it was identified that pipe support TWH-143 was overstressed. The licensee indicated in Reference 69 that this support has been replaced with a structurally adequate support to correct the overstressed condition.

Since the submittal of the EPU LAR, the licensee completed the FW and condensate systems piping structural evaluations. In Reference 67, the licensee provided its results of the structural evaluations that included maximum pipe stress and pipe support loads, and maximum pump nozzle stresses, compared to the Code of Record and design basis established allowable limits. The reported maximum stress ratios and maximum pipe support load interaction ratio are less than one and, therefore, are acceptable.

The MS and FW piping have increased flow rates and flow velocities in order to accommodate the power uprate. As a result, the MS and FW piping experience increased vibration levels, approximately proportional to the square of the flow velocities. Enclosure 10 to licensee's letter L-MT-08-052 (Reference 1), "Piping Flow Induced Vibration Monitoring Program," provides information on the plant system piping and components, including MS and FW piping and components, which could be subject to increased FIV due to EPU. The licensee's acceptance criteria for its piping flow induced vibration monitoring program follows guidance documented in ASME OM-SG Part 3, "Requirements for Preoperational and Initial Start-Up Vibration Testing of Nuclear Power Plant Piping Systems." The licensee evaluated the FIV levels associated with the MS and FW piping systems that are projected to increase for EPU. The NRC staff's review of the licensee's FIV and power ascension and testing programs for EPU are documented in Sections 2.2.6 and 2.12 of the staff's SER.

The NRC staff finds that the licensee adequately addressed the effects of the proposed EPU on the BOP piping, pipe components, and pipe supports. Based on its review, as summarized above, the NRC staff concludes that the proposed EPU does not adversely affect the structural integrity of the BOP piping, pipe components, and pipe supports.

Reactor Vessel and Supports

The licensee evaluated the effects of the proposed EPU on the RPV structure and support components for the design, normal, upset, emergency and faulted conditions in accordance with the plant's current design basis. In its evaluation, the licensee utilized the methodology documented in the NRC approved power uprate GE-Hitachi Nuclear Energy (GEH) LTRs (CLTR, ELTR1 and ELTR2). In accordance with this methodology, the licensee compared the proposed power uprate conditions (pressure, temperature and flow) against those used in the current design basis evaluations and reviewed existing component stress reports to identify components having a 40-year fatigue CUF greater than 0.33. Identified components where the power uprate conditions of pressure, temperature and/or flow rate did not increase or where the 40-year component fatigue CUF was equal to or less than 0.33 were dispositioned as acceptable, per the CLTR recommendation, and no further evaluation was performed.

The licensee, in accordance with the CLTR, performed plant specific evaluations for components having an OLTP 40-year CUF greater than 0.33 and experience an increase in pressure, temperature and/or flow rate due to the proposed EPU. It is noted that the CLTR recommends that plant specific evaluations be performed for components having CUFs greater than 0.5 and that experience an increase in pressure, temperature and/or flow rate due to the proposed EPU. The operating license for MNGP has been extended from 40 to 60 years and, therefore, the 0.5 CUF requirement has been scaled down by 1.5 (to reflect the 60-year plant life) which results in a CUF threshold of 0.33. The NRC staff finds the licensee's methodology is acceptable, as it is in accordance with the NRC-approved power uprate GEH LTRs and adjustments have been made to account for the 60-year plant life due to the plant renewed license.

Discussion of the conclusions of the licensee's RPV structural evaluations is presented in the proprietary portion of Section 2.2.3 of the PUSAR. For plant normal and upset conditions, the maximum stress and fatigue evaluation summary results from evaluations for RPV limiting components, affected by the power uprate, are presented in PUSAR Table 2.2-4, which indicates that the Code of Record allowable limits for stress and fatigue usage factors have been met for these components. There is no change in the plant design, emergency and faulted conditions. Therefore, RPV component stresses remain unchanged for these conditions. The NRC staff's review of the structural integrity of the RPV internals is discussed in Section 2.2.3 of this SE.

Based on its review, as summarized above, the NRC staff concludes that the licensee's evaluations have provided acceptability of the structural integrity of the RPV structure and its components.

Control Rod Drive Mechanism

The licensee's evaluation of the control rod drive mechanism (CRDM) for CPPU is documented in the proprietary portion of Sections 2.2.2 and 2.8.4.1.3 of PUSAR. The PUSAR states that the CRDM has been analyzed for an abnormal pressure operation that bounds the ASME reactor overpressure limit of 1375 psig. The peak RPV bottom head pressure due to EPU is 1335 psig versus the limit of 1375 psig. According to the CLTR, the reactor operating condition for a CPPU does not affect the CRD pump discharge pressure. The staff previously accepted the CLTR's conclusion that the maximum calculated stress for the limiting CRDM component (CRD system pressure-regulating valve that applies the maximum pump discharge pressure to the CRDM internal components) is not affected by the CPPU. The licensee, in its PUSAR, provided confirmation that MNGP CRD system integrity is consistent with the description provided in the CLTR because the reactor overpressure limit is not exceeded and dispositioned the CRD system as unaffected by the EPU.

Based on its review, as summarized above, the NRC staff agrees that the structural integrity of the CRDM is maintained for the proposed EPU conditions.

Recirculation Pumps and Supports

The NRC staff reviewed the licensee's evaluation of the recirculation pumps and supports documented in the proprietary portion of Section 2.2.2 of the PUSAR. For the proposed MNGP EPU operation, the maximum core flow rate remains unchanged at EPU operating conditions and the RRS drive flow increases slightly (approximately 1.7 percent) with no changes in operating pressure and temperature. Therefore, the NRC staff finds that the licensee has provided reasonable assurance that RRS system pumps and supports will remain structurally adequate at EPU conditions.

Conclusion

The NRC staff has reviewed the licensee's evaluations related to the structural integrity of pressure-retaining components and their supports. For the reasons set forth above, the NRC staff concludes that the licensee has adequately addressed the effects of the proposed EPU on these components and their supports. Based on the above, the NRC staff further concludes that the licensee has provided reasonable assurance that pressure-retaining components and their supports are structurally adequate to perform their intended design function under EPU conditions and remain in compliance with MNGP's current licensing basis following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the structural integrity of the pressure-retaining components and their supports.

2.2.3 Reactor Pressure Vessel Internals and Core Supports

Regulatory Evaluation

The Reactor Pressure Vessel (RPV) internals consist of all the structural and mechanical elements inside the reactor vessel, including core support structures. The NRC staff reviewed the effects of the proposed CPPU on the design input parameters and the design-basis loads and load combinations for the reactor internals for normal operation, upset, emergency, and faulted conditions. These include pressure differences and thermal effects for normal operation, transient pressure loads associated with loss-of-coolant accidents (LOCAs), and the identification of design transient occurrences. The staff's review included (1) the analyses of flow-induced vibration for safety-related and non-safety-related reactor internal components (steam dryer review is discussed in Section 2.2.6 of the staff's SER) and (2) the analytical methodologies, assumptions, ASME Code editions, and computer programs used for these analyses. The staff's review also included a comparison of the resulting stresses and CUFs against the corresponding Code-allowable limits.

The NRC's staff's acceptance criteria are based on (1) GDC-1, "Quality standards and records," insofar as it requires that SSCs important to safety be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety functions to be performed; (2) GDC-2, "Design bases for protection against natural phenomena," insofar as it requires that SSCs important to safety be designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions; (3) GDC-4, "Environmental and dynamic effects design bases," insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents; and (4) GDC-10, "Reactor design," insofar as it requires that the reactor core be designed with appropriate margin to assure that specified acceptable fuel design limits (SAFDLs) are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

As previously stated, MNGP is not licensed to the Appendix A GDCs but its design conforms with the intent of the 1967 AEC draft GDCs. According to MNGP's current licensing basis (see USAR Appendix E), MNGP conforms to the intent of draft GDC-1, draft GDC-2, draft GDC-5, draft GDC-6, draft GDC-40, and draft GDC-42; which are comparable to the GDCs listed above. Specific review criteria are contained in SRP Sections 3.9.1, 3.9.2, 3.9.3, and 3.9.5; and other guidance provided in Matrix 2 of RS-001.

Technical Evaluation

The RPV internals consist of core support structure and non-core support structure components. The licensee evaluated the RPV internals for the normal, upset, emergency and faulted conditions due to EPU (steam dryer review is addressed in Section 2.2.6 of the staff's SER). The loads considered in the evaluation were consistent with the existing design basis and include deadweight, seismic, reactor internal pressure differences, acoustic and flow induced loads, fuel lift loads and thermal loads. EPU loads are compared to those used in the existing design basis analyses. For cases where the loads due to EPU conditions are bounded by the existing design basis loads, no further evaluation is required. If the loads increase due to the EPU, the effect of the load increase is evaluated further and new stresses are determined by scaling up the existing design basis stresses in proportion to the loads. The resulting stresses are compared against the design basis code allowable values. The NRC staff finds the methodology used by the licensee acceptable, as it is consistent with the NRC-approved methodology in Appendix I of ELTR1 (Reference 9).

The licensee performed qualitative and quantitative assessments of the RPV internals. The licensee's discussion of the results of the qualitative and quantitative assessments is presented in Section 2.2.3 of its PUSAR (Reference 7). The licensee's results of the quantitative assessments are contained in the proprietary portion of Section 2.2.3 of the PUSAR. Quantitative summaries of governing stress and fatigue evaluation results for the RPV internals are summarized on PUSAR Table 2.2-9. For the feedwater sparger, this table indicates an EPU fatigue usage factor of 0.32 at the header to sparger weld. This value is less than the ASME Section III code allowable of 1.00 and, therefore, acceptable. The table also indicates a general primary membrane stress intensity (P_m) value for the Shroud of 39,500 psi for CLTP service level D which exceeds its level D code allowable value of 32,000 psi. The PUSAR states that the stress value of 39,500 psi was conservatively obtained from the evaluations for a larger BWR4, 251" size "New Loads Unit," and was used as-is. The licensee's PUSAR states that the acoustic load for EPU conditions has increased by an amount in excess of the available stress margin. To reduce conservatism, an EPU reevaluation was performed which resulted in a P_m

stress intensity value of 5,000 psi which is less than the 32,000 psi allowable and, therefore, is acceptable. Stress values for the remaining critical internals listed in the table are all within code allowable values, and, therefore are acceptable.

GEH issued a 10 CFR Part 21 Safety Information Communication (SIC) 09-03 on the subject of Shroud Screening Criteria Reports. SC 09-03 lists MNGP as one of the affected plants for the shroud screening criteria flaw evaluations due to the omission of the reactor recirculation line break loads in the shroud screening criteria reports. The NRC staff requested that the licensee address GEH SC 09-03 and its effect on the MNGP core shroud. The licensee stated in Reference 67 that NSPM entered GEH SC 09-03 into the MNGP corrective action program and requested that GEH review the MNGP faulted core shroud loading evaluation. GEH provided revised faulted shroud loading values, applicable to EPU conditions, which included RLB loads. NSPM updated the shroud inspection criteria evaluation for MNGP and determined that there was no effect on the shroud inspection criteria for MNGP, and determined that there was not effect on the inspection interval for the shroud as a result of the additional RLB loads. The NRC staff finds the licensee's response acceptable because it demonstrates that NSPM successfully resolved GEH SC 09-03 by incorporating the RLB loads in the core shroud inspection criteria evaluation.

As stated in Enclosure 10 of the proposed EPU LAR (Reference 1), when power is increased from CLTP to EPU conditions, steady state FIV levels are expected to increase approximately in proportion to the increase in the square of the fluid velocity. This is also described in the CLTR. With respect to the effects of FIV on the RPV internal components, the licensee indicated that the steam moisture separators and dryer in the upper elevations of the RPV are the components most affected by the increased steam flow at EPU conditions. Components near the core shroud head, such as shroud head bolts and Guide Rods, are mainly affected by the increase in feedwater flow. Components in the core region are primarily affected by core flow. Components in the annulus region, such as the jet pumps, are primarily affected by recirculation pump drive flow and core flow. This assessment is consistent with previous BWR power uprates that the NRC staff has reviewed. The licensee indicated that the maximum core flow rate remains unchanged and that recirculation pump drive flow only increases slightly for EPU, approximately 1.7 percent compared to CLTP. Therefore, the changes in FIV due to EPU in the core and annulus regions are negligible. The required RPV internals vibration assessments of the other RPV internals affected by the EPU are described in the CLTR (Reference 8).

The licensee performed FIV evaluations of the EPU affected internals which included the feedwater sparger, shroud head and separator, shroud head bolts, guide rods, incore guide tube, control rod guide tube, RPV top head spare instrument, nozzle, and RPV top head vent nozzle. These evaluations were performed at EPU conditions and 105 percent of rated core flow. The EPU FIV evaluations were based on given recorded vibration data obtained during startup testing of MNGP and from an instrumented prototype plant. For components requiring an evaluation, but not instrumented in the prototype plant, vibration data acquired during the startup testing from similar plants or acquired outside the RPV was used (from BWR operating experience). The expected vibration levels for EPU were then estimated by extrapolation of the "given" recorded vibration data, and on GE Nuclear Energy BWR operating experience, in accordance with the CLTR. These expected EPU vibration levels were then compared with established vibration acceptance limits. The established vibration level acceptance limits are based on the GE criterion which limits FIV alternating stress intensity to 10,000 psi for austenitic stainless steels. The NRC staff finds this criterion acceptable, as it is conservative when compared to the ASME Section III design fatigue endurance limit for austenitic stainless steel material of 13,600 psi which is further reduced for steady state vibration by a factor of 0.8 to 10,880 psi, following the guidance of Part 3 (Requirements for Preoperational and Initial Start-up Vibration Testing of Nuclear Power Plant Piping Systems) of the ASME OM-SG Code, "Standards and Guides for the Operation and Maintenance of Nuclear Plants."

Summaries of the licensee's structural evaluations of the RPV internals due to EPU FIV are presented in Section 2.2.3 of the PUSAR. The staff considers the licensee's methodology to be acceptable for reasons noted above and because it is in accordance with the staff approved CLTR and is similar to methodologies used in previously approved BWR power uprates. FIV analysis results for critical RPV internals are summarized in PUSAR Table 2.2-3, which shows that RPV critical internal components were conservatively evaluated for FIV at 102 percent of EPU RTP. The maximum FIV stress increase, EPU over OLTP, is reported for the shroud head and separator (a 172 percent increase) and it resulted in a stress value below 500 psi, which is less that the GE acceptance criteria of 10,000 psi. The jet pumps are reported to have the maximum calculated EPU FIV stress at a value below 7,000 psi. EPU FIV stresses for all reactor internal components in Table 2.2-3 are within the GE-established acceptance criteria of 10,000 psi and, therefore, are acceptable.

Based on its review, as summarized above, the NRC staff concludes that the RPV internals will continue to maintain their structural integrity at EPU conditions. The steam dryer assembly is addressed separately in Section 2.2.6 of the staff's SER.

Conclusion

The NRC staff reviewed the licensee's evaluations related to the structural integrity of reactor internals and core supports. For the reasons set forth above, the staff concludes that the licensee has adequately addressed the effects of the proposed EPU on the reactor internals and core supports. The staff further concludes that the licensee has demonstrated that the reactor internals and core supports will continue to meet the requirements of the MNGP current licensing basis following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the structural integrity of the reactor pressure vessel internals and core support structures.

2.2.4 Safety-Related Valves and Pumps

Regulatory Evaluation

The NRC staff's review included certain safety-related pumps and valves typically designated as Class 1, 2, or 3 under Section III of the ASME B&PV Code, and within the scope of Section XI of the ASME B&PV Code and the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code), as applicable. The NRC staff's review focused on the effects of the proposed EPU on the required functional performance of valves and pumps at MNGP. The review also covered any impacts that the proposed EPU might have on the licensee's motor-operated valve (MOV) program related to GL 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance;" GL 96-05, "Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves;" and GL 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves." The NRC staff also evaluated the licensee's consideration of lessons learned from the MOV program and the application of those lessons learned to other safety-related power-operated valves. The NRC's acceptance criteria are based on: (1) GDC-1, "Quality standards and records," insofar as it requires those systems and components which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed; (2) GDC-37, "Testing of emergency core cooling system," GDC-40, "Testing of containment heat removal system," GDC-43, "Testing of containment atmosphere cleanup systems," and GDC-46, "Testing of cooling water system," insofar as they require that the emergency core cooling system, the containment heat removal system, the containment atmospheric cleanup systems, and the cooling water system, respectively, be designed to permit appropriate periodic testing to ensure the leak-tight integrity and performance of their active components; (3) GDC-54, "Piping systems penetrating containment," insofar as it requires that piping systems penetrating containment be designed with the capability to periodically test the operability of the isolation valves to determine if valve leakage is within acceptable limits; and (4) 10 CFR 50.55a(f), insofar as it requires that pumps and valves subject to that section must meet the inservice testing program requirements identified in that section. Specific review criteria are contained in SRP Sections 3.9.3 and 3.9.6, and Power Uprate Review Standard RS-001.

Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company, the predecessor to NSPM, performed a comparative evaluation of the design basis of MNGP with the AEC proposed GDC of 1967. The MNGP comparative evaluation to the 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") associated with safety-related valves and pumps is contained in Appendix E of the MNGP USAR: draft GDC-1 and draft GDC-5.

Technical Evaluation

In the application dated November 5, 2008, as supplemented on January 21, 2013, the licensee discussed its evaluation of safety-related valves and pumps to perform their intended functions under EPU conditions. The NRC staff's review of the impact of EPU conditions on safety-related valves and pumps is summarized in the following paragraphs.

In response to GL 89-10 and GL 96-05, MNGP established a testing and surveillance program for MOVs. The NRC acceptance of the MOV program for MNGP was documented in a letter dated May 11, 1995. In a letter dated April 12, 1999, the NRC conveyed the SE for MNGP's response to GL 96-05, and stated that MNGP had established an acceptable program to periodically verify the design-basis capability of the safety-related MOVs. In its request for the EPU license amendment, the licensee described its evaluation of the MOVs within the scope of GL 89-10 for the effects of the proposed EPU, including those related to pressure locking and thermal binding as addressed in GL 95-07. The licensee's review of affected systems indicates that the existing maximum operating conditions (e.g., flow rates, pressures and temperatures) remain valid for the EPU. Therefore, no changes were identified to the design functional requirements for all GL 89-10 MOVs. The licensee evaluated all GL 89-10 MOVs for the effects of changes in system pressures and environmental temperatures due to EPU conditions. Based on the evaluation, the licensee concludes that all MOVs will perform their safety-related functions under EPU conditions with the following exceptions:

- MO-2009, 12 RHR Torus Cooling Injection Valve
- MO-2014, 11 LPCI Inboard Injection Valve
- MO-2015, 12 LPCI Inboard Injection Valve

- MO-2020, 11 Containment Spray Outboard Valve
- MO-2021, 12 Containment Spray Outboard Valve
- MO-2023, 12 Containment Spray Inboard Valve
- MO-2034, HPCI Steam Line Isolation Valve
- MO-2035, HPCI Steam Line Isolation Outboard Valve
- MO-2061, HPCI Torus Suction Inboard Isolation Valve
- MO-2062, HPCI Torus Suction Outboard Isolation Valve

The licensee notes that the MOVs listed above require switch adjustments to fully comply with the EPU conditions and requirements of GL 89-10 and GL 96-05. The licensee also states that the switch adjustments are scheduled for completion in the 2013 refueling outage. Based on the licensee's commitment to complete the switch adjustments in the 2013 refueling outage, as outlined in Enclosure 1 of the January 21, 2013, letter, the NRC staff concludes that the above MOVs are acceptable and will be fully capable of performing their EPU post-event safety functions. The MOVs were also evaluated for pressure locking and thermal binding under EPU conditions, and no new MOVs were determined to be susceptible to pressure locking or thermal binding.

MNGP has in place a program to ensure that safety-related air-operated valves (AOVs) are selected, set, tested and maintained so that AOVs will be operated under normal, abnormal, or emergency operating design-basis conditions. Furthermore, the AOV program will ensure continued AOV reliability for the life of the plant. The licensee has reviewed system level design-basis calculations for the Automatic Depressurization System, Residual Heat Removal System, Standby Liquid Control System, High Pressure Injection System, and Core Spray System. The results of the evaluation show that the EPU does not affect the maximum differential pressures, flow rates, or fluid temperatures for the design-basis conditions. Therefore, the EPU has no impact on the associated AOVs, and the existing design pressures and temperatures are adequate for these valves. The licensee also stated that the MOV and AOV programs are governed by site and fleet procedures which require operating experience to be evaluated and incorporated for the corresponding programs.

The licensee's review of affected systems indicates that the existing maximum operating conditions, i.e., flow rates, pressures and temperatures, remain valid for the EPU. As such, no change in pump head performance is required for the affected safety-related pumps at EPU conditions. Therefore, pump design and Inservice Testing (IST) Program requirements for these pumps are not affected by the EPU.

In the application dated November 5, 2008, the licensee described its review of the IST Program for safety-related pumps and valves at MNGP for EPU operations. The Code of Record for MNGP is the 2004 Edition with the 2005 and 2006 Addenda of the ASME OM Code. The IST Program at MNGP assesses the operational readiness of pumps and valves within the scope of the OM Code. The scope of the IST Program at MNGP, and the testing frequencies, will not be affected by the EPU. The IST program must be periodically updated to meet applicable ASME OM Code requirements specified in 10 CFR 50.55a. However, in support of the EPU request, no effects are anticipated in the revised IST Program at MNGP.

Conclusion

The NRC staff has reviewed the licensee's assessments related to the functional performance of safety-related valves and pumps at MNGP in support of the proposed EPU amendment. Based on the review set forth above, the NRC staff has determined that the licensee adequately addressed the effects of the proposed EPU on safety-related pumps and valves. The NRC staff further concludes that the licensee has adequately evaluated the effects of the proposed EPU on its MOV programs related to GL 89-10, GL 96-05, and GL 95-07, and considered the lessons learned from those programs to other safety-related power-operated valves. Therefore, the NRC staff concludes that the licensee has demonstrated that safety-related valves and pumps will continue to meet the regulatory requirements as set forth above following implementation of the proposed EPU at MNGP. As a result, the NRC staff finds the proposed EPU for MNGP to be acceptable with respect to safety-related valves and pumps.

2.2.5 Seismic and Dynamic Qualification of Mechanical and Electrical Equipment

Regulatory Evaluation

Mechanical and electrical equipment covered by this section includes equipment associated with systems that are essential to emergency reactor shutdown, containment isolation, reactor core cooling, and containment and reactor heat removal. Equipment associated with systems essential to preventing significant releases of radioactive materials to the environment are also covered by this section. The NRC staff's review focused on the effects of the proposed CPPU on the qualification of the equipment to withstand seismic events and the dynamic effects associated with pipe-whip and jet impingement forces. The primary input motions due to the safe shutdown earthquake (SSE) are not affected by a CPPU.

The NRC's acceptance criteria are based on: (1) GDC-1, "Quality standards and records," insofar as it requires that SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed; (2) GDC-30, "Quality of reactor coolant pressure boundary," insofar as it requires that components that are part of the RCPB be designed, fabricated, erected, and tested to the highest quality standards practical; (3) GDC-2, "Design bases for protection against natural phenomena," insofar as it requires that SSCs important to safety be designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions; (4) 10 CFR Part 100, Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants," which sets forth the principal seismic and geologic considerations for the evaluation of the suitability of plant design bases established in consideration of the seismic and geologic characteristics of the plant site; (5) GDC-4, "Environmental and dynamic effects design bases," insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents; (6) GDC-14, "Reactor coolant pressure boundary," insofar as it requires that the RCPB be designed, fabricated, erected, and tested so as to have an extremely low probability of rapidly propagating fracture; and (7) 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," which sets quality assurance requirements for safety-related equipment.

As previously stated, MNGP is not licensed to the 10 CFR 50, Appendix A, GDCs, but its design conforms to the intent of the 1967 AEC draft GDC. According to MNGP's current licensing basis (see USAR Appendix E), MNGP conforms to the intent of draft GDC-1, draft GDC-2, draft GDC-5, draft GDC-9, draft GDC-16, draft GDC-33, draft GDC-40, and draft GDC-42; which are comparable to the GDC-listed above.

Specific review criteria are contained in SRP Section 3.10.

Technical Evaluation

The licensee evaluated safety-related SSCs subject to CPPU conditions. Seismic loads are not affected by power uprates. The licensee has considered DBA LOCA conditions, main steam line break (MSLB), and other HELBs that could affect safety-related mechanical and electrical equipment and components. In Section 2.2.1 of this SER, the staff's review of the licensee's evaluations concluded that SSCs important to safety are adequately protected from the dynamic effects of postulated pipe failures, including pipe whip and jet impingement, at EPU conditions. As shown in the staff's input in Sections 2.2.1 and 2.2.2 of this SER, containment hydrodynamic inertia loads due to a DBA LOCA and SRV discharge are not affected by the proposed EPU. The licensee's evaluation of containment hydrodynamic loads, which also shows that these loads are not affected by the EPU, is presented in PUSAR Section 2.6.1.2. The staff's review of this section is provided by the Containment and Ventilation Branch of the Division of Safety Systems in Section 2.6 of the staff's SER.

The licensee also evaluated safety-related mechanical equipment subject to increased fluid-induced loads, nozzle loads and component support loads due to increased temperatures, flows or pressures for EPU. As shown in Section 2.2.2 of this SER, the staff's review of the licensee's evaluations found that the mechanical components and component supports are adequately designed for the proposed EPU conditions.

The licensee noted that normal temperature, pressure, and humidity conditions either do not significantly change due to EPU or remain bounded by values used in the current analyses. Accident temperature, pressure and humidity profiles, that are not bounded by the CLTP conditions, were evaluated by the licensee and found that they do not adversely affect the qualification of safety-related electrical equipment. The licensee's evaluation for the qualification of safety-related electrical equipment subject to DBA LOCA conditions, MSLB and other HELBs is documented in PUSAR section 2.3.1. The NRC staff's review of this section is provided by the Electrical Engineering Branch of the Division of Engineering in Section 2.3 of the staff's SER.

With regard to non-metallic components found in mechanical equipment, the licensee determined that increases in temperature and accident and normal radiation levels due to the EPU do not adversely affect the functional capability of these components.

The licensee, in its PUSAR and in its response to staff RAIs, indicated that the following programs provide reasonable assurance that important SSCs will be capable of performing their intended functions: MNGP design control program; periodic preventive maintenance and testing; and investigation of causes of failures, part of the MNGP Maintenance Rule which also incorporates industry operating experience. The NRC staff finds the licensee's response to be acceptable, as it provides additional assurance that the reliability of plant equipment will be maintained following implementation of the proposed EPU.

Conclusion

The NRC staff has reviewed the licensee's evaluations of the effects of the proposed EPU on the qualification of mechanical and electrical equipment and, based on its review above, concludes that the licensee has: (1) adequately addressed the effects of the proposed EPU on these equipment; and (2) demonstrated that these equipment will continue to meet the requirements 10 CFR Part 100, Appendix A; and 10 CFR Part 50, Appendix B, and MNGP's current licensing basis following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed CPPU acceptable with respect to the qualification of the mechanical and electrical equipment.

2.2.6 <u>Additional Review Areas – Evaluation of Replacement Steam Dryer Integrity</u>

<u>Background</u>

The MNGP is a GE-designed BWR/3 plant with a Mark I containment. The AEC issued a provisional operating license for full commercial operation at 1670 MWt for MNGP in June 1971. The NRC issued a full-term operating license for 1670 MWt in January 1981. In 1998, the NRC approved a power uprate to 1775 MWt. The NRC granted license renewal for MNGP in November 2006.

After submitting the MNGP EPU license amendment request in March 2008, which was based on the original GE parallel vane bank, square hood type, steam dryer, NSPM withdrew the LAR due to acceptance review questions from the NRC staff related to the steam dryer stress margin and noise subtraction. By letter to the NRC dated November 5, 2008 (Reference 1), NSPM re-submitted its EPU LAR pursuant to the requirements of 10 CFR 50.90. The proposed amendment consists of an increase licensed thermal power from 1,775 MWt to 2,004 MWt. This increase corresponds to approximately 13 percent above the CLTP and 20 percent above the OLTP of 1670 MWt.

The licensee elected to replace the original steam dryer with a Westinghouse concentric, octagonal shaped, three parallel vane bank, steam dryer, to reduce the moisture carry over (MCO) to ≤ 0.1 percent. This replacement steam dryer was installed during spring of 2011, and has been operating at CLTP power level.

<u>Regulatory Evaluation</u>

The steam dryer is a reactor internal component and is located in the steam dome portion of the RPV. MNGP was licensed to the draft GDCs published for comment in the *Federal* Register (32 FR 10213 on July 11, 1967, and MNGP USAR, Appendix E, contains discussions on how the plant meets each of the 70 Draft GDCs. Since the steam dryer is a safety significant component, the NRC's acceptance criteria is based on: (1) 10 CFR 50.55a, "Codes and standards," and GDC-1, "Quality standards and records," insofar as they require that SSCs important to safety be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety functions to be performed; (2) GDC-2, "Design bases for protection against natural phenomena," insofar as it requires that SSCs important to safety be designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions; (3) GDC-4, "Environmental and dynamic effects design bases," insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents; and (4) GDC-40, "Testing of containment heat removal system," and GDC-42, "Inspection of containment atmosphere cleanup systems," insofar as they require that protection be provided for engineered safety features (ESFs) against the dynamic effects and missiles that might result from plant equipment failures, as well as the effects of a LOCA. Specific NRC review criteria are contained in NRC SRP Sections 3.9.1, 3.9.2, 3.9.3, and 3.9.5; and other guidance provided in Matrix 2 of RS-001 and RG 1.20.

MNGP Current Licensing Basis

The GDCs listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable MNGP principal design criteria predate these criteria. The MNGP principal design criteria are listed in Appendix E, "Principal Design Criteria," of the USAR. In July 1967, the AEC published in the *Federal Register* for public comment a revised set of proposed GDC (32 FR 10213). Although not explicitly licensed to the AEC-proposed GDC, the licensee performed a comparative evaluation of the design basis of MNGP with the AEC-proposed GDC (hereafter referred to as "draft GDC"). Appendix E of the USAR contains a comparative evaluation of the MNGP principal design criteria with each of the draft GDC. With regard to each group of criteria, there is a statement of the licensee's understanding of the intent of the draft GDC in that group and a discussion of the plant design conformance with the intent of the draft GDC. Based on this evaluation, the licensee determined that MNGP is in conformance with the intent of the draft GDC.

The RPV internals and core supports are described in Section 3.6, "Other Reactor Vessel Internals," and Section 4.2, "Reactor Vessel," of the MNGP USAR. In addition to the evaluations described in the USAR, systems and components were evaluated during license renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated and documented in the MNGP License Renewal SER, NUREG-1865, dated October 31, 2006 (Reference 111). The reactor internals and core support structural components evaluation for license renewal are discussed Section 3.1 of NUREG-1865.

Technical Evaluation

Plant operation at EPU conditions can result in adverse flow effects on the MS, FW, and condensate systems and their components, and the steam dryers in BWR plants from increased system flow and flow-induced vibration. Some plant components, such as the steam dryer, do not perform a safety function but must retain their structural integrity to avoid the generation of loose parts that might adversely impact the capability of other plant equipment to perform their safety functions. Therefore, a BWR steam dryer is a safety significant component located inside the reactor pressure vessel. The NRC staff reviewed the evaluation by NSPM of the potential adverse flow effects for the proposed EPU at MNGP, including consideration of the design input parameters and the design-basis loads and load combinations for the MNGP steam dryer for normal operation, upset, emergency, and faulted conditions. The staff's review covered the analytical methodologies, assumptions, and computer modeling used in the evaluation of the MNGP steam dryer, and also included a comparison of the resulting stresses against the applicable limits.

The NRC staff reviewed the licensee's flow-induced vibration evaluation of the MS, FW, and condensate system components, and the MNGP RSD for susceptibility to adverse flow effects due to increased flow at EPU operation. The staff's detailed technical evaluation of the MNGP RSD, including the Power Ascension Test Plan and License Conditions, is provided in Appendix A of this Safety Evaluation.

Conclusion

The NRC staff reviewed the licensee's evaluations of potential adverse flow effects on the MS, FW, and condensate systems and their components (including the steam dryer) for the operation of the MNGP RSD at EPU conditions. The staff concluded that the licensee has provided reasonable assurance that the flow-induced and mechanically-induced effects on the steam dryer and other plant equipment are within the structural limits at the CLTP conditions and the extrapolated EPU conditions.

Based on the above evaluation, the NRC staff concluded that the proposed license amendment to operate MNGP at the proposed EPU conditions is acceptable with respect to potential adverse flow effects. The NRC staff further concludes that the licensee has demonstrated that the replacement steam dryer will continue to meet the requirements of draft GDCs 1, 2, 40, and 42, following implementation of the proposed EPU at MNGP subject to the license conditions in this SE. There is reasonable assurance that the MNGP RSD will maintain its structural integrity at the projected EPU conditions.

2.3 Electrical Engineering

Background

The licensee provided additional information to this section in two letters submitted before the November 5, 2008, application on May 28, 2008, and June 5, 2008 (References 77 and 78, respectively). As a result of subsequent NRC staff RAIs, the licensee provided additional information in letters dated May 26, 2009, January 21, 2013, April 10, 2013, and June 26, 2013 (References 79, 67, 80 and 81, respectively).

2.3.1 Environmental Qualification of Electrical Equipment

Regulatory Evaluation

Environmental Qualification (EQ) of electrical equipment demonstrates that the equipment is capable of performing its safety function under significant environmental stresses which could result from DBAs. The NRC staff's review focused on the effects of the proposed EPU on the environmental conditions that the electrical equipment will be exposed to during normal operation, anticipated operational occurrences, and accidents. The NRC staff's review was conducted to ensure that the electrical equipment will continue to be capable of performing its safety functions following implementation of the proposed EPU. The NRC's acceptance criteria for EQ of electrical equipment are based on 10 CFR 50.49, which sets forth requirements for the qualification of electrical equipment important to safety which is located in a harsh environment. Specific review criteria are contained in SRP Section 3.11.

Technical Evaluation

Inside Containment

EQ for safety-related electrical equipment located inside containment is based on DBAs for a MSLB, and LOCA conditions and their resultant temperature, pressure, humidity, and radiation consequences. The EQ also includes the environment expected to exist during normal plant operation. The NRC staff reviewed the licensee's EPU application, including the licensee's EQ evaluation report, "Task T1004, Revision 1," which was submitted as Enclosure 17 of the November 5, 2008, application. The NRC staff also reviewed the licensee's supplemental letter dated May 26, 2009.

The licensee, in their letter dated January 21, 2013, provided a Gap Analysis of the EQ in comparison with the information provided in the original LAR. The licensee stated that the EQ program has been reconstituted to incorporate the environmental conditions associated with the EPU increased thermal power and revised environmental conditions were incorporated into EQ Environmental Specifications. In letter dated April 10, 2013, the licensee provided revised information of the normal and accident levels for radiation, temperature, pressure, submergence, chemical spray effects, and humidity for all areas in which environmentally qualified equipment is installed showing the pre-EPU and EPU conditions, as well as the qualified values of EQ equipment in those areas. The tabulations for the EQ parameters provided by the licensee indicate these details for two types of qualification categories, one for the equipment qualified per Division of Operating Reactors (DOR) guidelines, and the other qualified per 10 CFR 50.49 guidelines. The licensee stated that the EQ Program at MNGP was developed to the guidance and requirements contained in the DOR guidelines and Category II of NUREG-0588 for equipment that predates the issuance of 10 CFR 50.49 with incorporation of EPU plant conditions as provided in Section 8.9 of the USAR. Therefore the replacement equipment installed subsequent to February 22, 1983, has been qualified in accordance with the provisions of 10 CFR 50.49, using the guidance of RG 1.89, Revision 1.

The staff reviewed the EQ details provided in the Enclosure 1 (Responses to the Gap Analysis) and Enclosure 2 (Marked-up Page Changes to EPU Documentation Based on the Gap Analysis) of the letter dated January 21, 2013, and subsequent RAI-responses provided in letter dated April 10, 2013. In its April 10, 2013, RAI-response, the licensee provided the EPU Peak Temperature vs. Qualification Temperature of applicable EQ equipment along with the equipment type, location, accident type, and the margin maintained for qualified equipment. Table 2-1 of the licensee's RAI-response showed that for certain equipment which are qualified in accordance with 10 CFR 50.49 in the reactor building (RB), the peak EPU temperature was not bounded by the test peak temperature by the recommended margin of 15°F (per IEEE 323-1974). One reason provided by the licensee is that plant conditions are adequately bounded by the test since it has a dual transient exposure. During a June 18, 2013, conference call with the licensee, the NRC staff clarified that while IEEE 323-1983 permits the peak environmental transients to be applied twice without the margins on peak values this guidance has not been endorsed by the NRC. Similarly, in Table 2-2, the licensee provided the EPU Peak Temperature vs. Qualification Temperature of applicable EQ equipment. Based on its review of this table, the staff identified that the margin for certain equipment was not bounded by the qualified pressure. Based on the above observations, the staff requested the licensee to provide additional information to justify having temperature margins less than 15°F under EPU conditions demonstrating that adequate margin has been established for all locations of the RB. The NRC staff also asked the licensee to provide justification on the peak accident pressure margin for certain equipment.

The licensee indicated that there are 13 solenoid valves and 9 Limitorque MOVs located in the drywell (DW) area. According to the licensee, the DW initial peak accident temperature of 338°F reduces to below 300°F within 700 seconds. The EQ test for the solenoid valves and Limitorque MOVs was maintained at a peak temperature of 346°F for 2.86 hours. Although the peak DW temperature is not bounded by the recommended margin of 15°F, the staff determined that the testing was conducted at a temperature higher than the expected peak temperature during design basis events for a time period substantially longer than what is expected without any failures, therefore demonstrating that adequate margin is available to ensure that the solenoid valves and Limitorque MOVs can perform their design function under EPU conditions. Similarly the Rockbestos Coax Cable, Firewall SR Cable, and Firewall III/SIS Cable in DW, which have margin less than 15°F, were tested at 346°F, 351.2°F, and 342°F respectively, each for 2.91 hours. These cables were tested at a temperature higher than the expected peak temperature during design basis events for a time period substantially longer than what is expected without any failures, therefore demonstrating that adequate margin is available to ensure that these cables can perform their design function under EPU conditions.

Additionally, there are 12 Static O-ring pressure switches located within the RB that are qualified under DOR guidelines that are expected to see a peak pressure of 0.26 psig during the worst design basis event. However, the data provided by the licensee showed that the Static O-ring pressure switches were tested to a pressure of 0.25 psig. The licensee indicated that the peak pressure in RB volumes 22 and 33 during any accident is 0.16 psig, which is adequately bounded by the 6 hours of testing conducted at 0.25 psig. Also in RB volumes 1 and 3, during a DBA LOCA, there is no postulated increase in pressure. Based on the pressure profiles provided by the licensee, the staff concludes that the peak pressure (0.26 psig) does not occur during a period when a harsh environment exists in any of the RB volumes where the Static O-ring pressure switches are located and the peak pressure experienced, during a design basis accident, when a harsh environment exists is 0.16 psig which is adequately bounded by the 0.25 psig test pressure.

Based on the above review, the NRC staff verified that the normal operating temperatures will continue to be bounded by the temperatures used in the licensee's EQ analyses. Furthermore, the NRC staff verified that the post-accident peak temperature and pressure will continue to be bounded by the peak temperature and pressure conditions used in the licensee's EQ analyses. The radiation EQ for safety-related electrical equipment inside containment is based on the radiation environment expected to exist during normal operations, post-LOCA conditions, and the resultant cumulative radiation doses. The licensee noted that the radiation levels would increase above the levels used in its current EQ program. The NRC staff reviewed the licensee's EQ evaluation and supplemental responses, and confirmed that the increase should not affect the qualification of the EQ equipment located inside containment.

Based on the licensee's application and supplemental responses, the NRC staff finds that the total integrated radiation doses (normal plus accident) for EPU conditions would not adversely affect the qualification of equipment inside containment.

Outside Containment

The licensee stated that accident temperature, pressure, and humidity environments used for qualification of equipment outside containment result from MSLB or other HELBs, whichever is limiting for each plant area. The licensee evaluated the temperature, pressure, and humidity profiles that were not bounded by current licensed thermal power conditions to ensure that the new profiles do not adversely affect the qualification of safety-related electrical equipment. The accident temperature resulting from a LOCA or MSLB inside containment increased for some RB areas due to the additional heat load resulting from the increase in drywell and wetwell temperatures. The NRC staff reviewed the licensee's EQ evaluation provided in the LAR; subsequent RAI-responses dated May 28, 2008, June 5, 2008, May 26, 2009; letter dated January 21, 2013 (Enclosure 1 and 2 of the Gap Analysis, Item 27); and subsequent

RAI-response dated April 10, 2013. The staff verified that the long-term post-accident temperatures would not adversely affect the qualification of safety-related electrical equipment. Additionally, in letter dated June 26, 2013 (Enclosure 6), the licensee confirmed that it had replaced two level transmitters in the torus compartment (LT-7338A, and LT-7338B) to maintain the qualification of the equipment. The licensee stated that the normal temperature, pressure, and humidity conditions do not change significantly as a result of EPU.

Based on its review of the licensee's application and supplemental responses, the NRC staff verified that the change of the normal operating temperature, pressure, submergence, and humidity conditions will not adversely affect the qualification of safety-related electrical equipment. The licensee noted that the radiation levels would increase above the levels used in their current EQ program. The NRC staff reviewed the licensee's EQ evaluation and supplemental responses and confirmed that the increase should not affect the qualification of the EQ equipment located outside of containment. The staff finds that the total integrated radiation doses (normal plus accident) for EPU conditions would not adversely affect the qualification of the EQ equipment located outside containment with the exception of the two level transmitters that will be replaced.

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the EQ of electrical equipment, and concludes that the licensee has adequately addressed the effects of the proposed EPU on the environmental conditions inside and outside containment and the qualification of electrical equipment. The NRC staff further concludes that the electrical equipment will continue to meet the relevant requirements of 10 CFR 50.49 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the EQ of electrical equipment.

2.3.2 Offsite Power System

Regulatory Evaluation

The offsite power system includes two or more physically independent circuits capable of operating independently of the onsite standby power sources. The NRC staff's review covered the descriptive information, analyses, and referenced documents for the offsite power system; and the stability studies for the electrical transmission grid. The NRC staff's review focused on whether the loss of the nuclear unit, the largest operating unit on the grid, or the most critical transmission line will result in the loss of offsite power (LOOP) to the plant following implementation of the proposed EPU.

The NRC's acceptance criteria for offsite power systems are set forth as GDC-17, "Electric power systems." The applicable Principal Design Criteria for MNGP predates this criterion. The MNGP Principal Design Criteria is listed in USAR Section 1.2, "Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed GDC (*Federal Register*, 32 FR 10213, dated July 11, 1967). Although not explicitly licensed to the AEC-proposed GDCs published in 1967, the licensee performed a comparative evaluation of the design basis of the MNGP with the AEC-proposed GDC of 1967. This evaluation is included in MNGP USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria."

Based on a review of the MNGP's USAR, the NRC staff identified the following Principal Design

Criteria as being applicable to the proposed EPU application:

The USAR Principal Design Criteria 24 and 39 require that alternate power systems shall be provided and designed with adequate independency, redundancy, capacity, and testability to permit the functioning required of the engineered safety features. As a minimum, the on-site power system and the off-site power system shall each, independently, provide this capacity assuming a failure of a single active component in each power system. In the event of the loss of all off-site power, sufficient alternate sources of power shall be provided to permit the required functioning of protection systems.

The specific review criteria are contained in SRP Sections 8.1 and 8.2, Appendix A to SRP Section 8.2, and BTPs PSB-1 and ICSB-11.

Technical Evaluation

The MNGP offsite power system starts at the output from the main generator and includes the isolated phase bus (IPB), the main step-up transformer, switchyard, power grid, and the offsite power supplies to the switchyard. The offsite power system is designed to provide adequate power to site loads given that the 345 kilo-volt (kV) and 115 kV grid voltages are within the ranges specified by plant procedures. The ranges are derived from the plant alternating current (AC) load studies. Operation within these ranges provides adequate voltage for operability of safety-related equipment, provides for proper operation of various automatic voltage regulating equipment (e.g., load tap changers), and will result in avoidance of inadvertent bus transfers of the safety-related buses due to degraded voltage when starting plant equipment.

The licensee stated that several modifications to the existing onsite and offsite electrical equipment are necessary to assure the system is adequate for operation with increased non-safety-related in-plant loads and updated electrical output. These modifications will be controlled by the Monticello Modification Process. This process requires compliance with site work instructions for the Fuse/Breaker Coordination Study and AC Electrical Load Study. Conformance to the MNGP licensing bases is controlled by required load studies for changes to the site AC electrical system. The AC load study is described in the MNGP USAR.

While loads shed by ECCS load shedding are not included in the Offsite AC System loading determination for the DBA LOCA loads, the licensee noted that EPU does not involve any changes to load shedding circuits. The licensee further stated that EPU does not affect any of the timing associated with ECCS load sequencing. The AC load studies include minimum and maximum equipment voltages for steady-state operation and motor starting. It also includes, by reference, the degraded voltage setpoints.

The MNGP AC load study established voltage limits based on equipment design. The licensee stated that EPU does not change these limits. All of the new EPU AC motors will be designed to start and operate within the existing voltage limits or, if operated at a different voltage base, new limits will be established based on equipment design. Based on this information, the proposed EPU does not require any changes to the MNGP setpoints for degraded bus voltage and loss of voltage logic.

The Midwest Independent System Operator (MISO) is the North American Electric Reliability Corporation Regional Transmission Organization that has jurisdiction of MNGP. As the Regional Transmission Organization (RTO), MISO is responsible for the operation of the transmission grid. The MISO coordinates the planning process for connection of new generation, coordinates the reliability studies for operation of new generation, and oversees the construction of the required Interconnection Facilities. The licensee provided a summary of the MISO grid stability study for the proposed EPU at MNGP in Enclosure 14 of the licensee's November 5, 2008, application. The summary of the MISO grid stability study demonstrates that the MNGP electrical output can be increased to 705.7 Megawatts (MW) electric gross without compromising the offsite power grid or its capability to supply in-plant loads. As a result of the power uprate, the electrical output of the main generator will increase from 631.2 MWe at a power factor (pf) of 0.95 to 691.4 MWe at 0.963 pf. In a January 21, 2013, supplemental letter, the licensee identified that a new 345 kV transmission line has been installed in the plant substation. This addition resulted in a reconfiguration of the plant output connection to the grid. Due to these changes to the grid, the licensee noted that MISO commissioned a restudy evaluation of projects with permanent Generator Interconnection Agreements. No adverse impacts were identified in this study. Since the proposed increase is within the limit identified in the MISO load study, the NRC staff finds that the proposed power uprate would not adversely affect the stability of the electric power grid.

As mentioned previously, the licensee stated that several modifications to the existing onsite and offsite electrical equipment are necessary to assure the system is adequate for operation with increased non-safety-related in-plant loads and updated electrical output. The licensee noted the following modifications:

- The continuous current rating of the IPB will be upgraded from 18.7 kilo-amperes (kA) to support a maximum generator output of 19.834 kA at EPU conditions. This will be accomplished by modification of the forced air-cooling system. The licensee also stated that it will add a redundant isolated phase bus cooling skid to increase reliability.

- The main transformer will be replaced for EPU operation. The associated switchyard components (rated for maximum transformer output) are adequate for the uprated transformer output. The protective relaying for the main generator is adequate for the uprated generator output with some changes in protective relay setpoints.

- Replace the existing 11 and 12 4-kV buses with 13.8 kV buses, including replacement of the 1R and 2R transformers. This will require replacing all motors associated with the new bus to provide motors rated at 13.8 kV. These modifications will insure compliance with design requirements as defined in the Technical Evaluation of PUSAR Section 2.3.2. With modification of the 1R and 2R supplies and onsite non-safety distribution, the offsite AC power sources will be adequate to accomplish required emergency core cooling system functions under postulated design basis accident conditions with the 115 kV and 345 kV grid voltages within the operating limits described in MNGP USAR Section 8.10.

- Add remote reactivity capability to the grid to meet the 0.95 lead/lag power factor requirements of the MISO interconnection tariff. The size and location of such devices will be identified in the Interconnection Agreements negotiated with MISO.

- Replace the Reactor Recirculation Motor-Generator (RRMG) Set Motors with new 13.8 kV motor or adjustable speed drives.

- Replacement of the 1AR Transformer due to aging not related to the EPU.
- Replace the Reactor Feed Pump motors with new higher horsepower 13.8 kV motor.
- Upgrade Condensate Pump with new higher horsepower 13.8 kV motor.

In its May 28, 2008, submittal, the licensee stated that increases in required condensate and feedwater pump capacity for EPU result in electrical loads for onsite non-safety-related AC power systems that exceed the capacity of the existing system. The modifications listed above provide upgrades to plant non-safety-related AC electrical distribution systems to correct this deficiency. The NRC staff verified that there are no changes required to safety-related buses for EPU.

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the offsite power system and concludes that the offsite power system will continue to meet the MNGP USAR Principal Design Criteria 24 and 39 following implementation of the proposed EPU. Adequate physical and electrical separation exists and the offsite power system has the capacity and capability to supply power to all safety loads and other required equipment.

The NRC staff further concludes that the impact of the proposed EPU on grid stability is negligible. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the offsite power system.

2.3.3 AC Onsite Power System

Regulatory Evaluation

The AC onsite power system includes those standby power sources, distribution systems, and auxiliary supporting systems provided to supply power to safety-related equipment. The NRC staff's review covered the descriptive information, analyses, and referenced documents for the ac onsite power system. The NRC's acceptance criteria for the ac onsite power system are set forth as GDC-17, "Electric power systems," insofar as it requires the system to have the capacity and capability to perform its intended functions during anticipated operational occurrences and accident conditions. The applicable Principal Design Criteria for MNGP predates this criterion. The MNGP Principal Design Criteria are listed in USAR Section 1.2, "Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed GDC (*Federal Register*, 32 FR 10213, July 11, 1967). Although not explicitly licensed to the AEC-proposed GDCs published in 1967, the licensee performed a comparative evaluation of the design basis of the MNGP with the AEC-proposed GDC of 1967. This evaluation is included in MNGP USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria." Based on a review of the MNGP's USAR, the NRC staff identified the following Principal Design Criteria as being applicable to the proposed EPU application:

USAR Principal Design Criteria 24 and 39 require that alternate power systems shall be provided and designed with adequate independency, redundancy, capacity, and testability to permit the functioning required of the engineered safety features. As a minimum, the on-site power system and the off-site power system shall each, independently, provide this capacity assuming a failure of a single active component in each power system. In the event of the loss of all off-site power, sufficient alternate sources of power shall be provided to permit the required functioning of protection systems.

Specific review criteria are contained in SRP Sections 8.1 and 8.3.1.

Technical Evaluation

The NRC staff reviewed the licensee's submittals to determine whether the emergency diesel generators (EDGs) would remain capable of performing their intended function at EPU conditions. The MNGP EDGs and Class 1E uninterruptible power supply (UPS) provide power to essential AC loads including adequate distribution, protections, and control for design-basis events with a simultaneous LOOP. The essential AC system provides power distribution and control of loads during these events.

There are no changes to the ratings of safety-related loads and no new safety-related loads normally powered from the EDG as a result of EPU. EPU also does not involve any changes to load shedding circuits or essential bus transfers.

The EDG load is based on the nameplate equipment rating of the loads. In general, the motor rated horsepower is determined assuming a conservative motor efficiency of 0.9 or less. Non-motor loads are conservatively included by either assuming load operating time is maximized or by including extra load margin.

The ECCS motors are sized to provide sufficient torque to operate pumps and valves according to the pump and valve horsepower requirements. The pump operating horsepower is a function of flow, head, and pump efficiency. The EPU does not involve changes to pump variables or torque requirements for the ECCS loads and the loading does not increase.

The EPU does not affect the timing associated with ECCS load sequencing and has no effect on EDG transient performance. There are no changes to the sequencing and timing of AC ECCS loads during a DBA LOCA. EPU has no effect on the functional requirements for instrumentation and control subsystems of the safety-related EDG power systems and there are no changes to the instrumentation and control systems of the essential AC systems.

The EDGs have a continuous load rating of 2500 kilowatts, which envelopes both initial and steady-state loading. In addition, EDG transient voltage and frequency performance is not affected. Based on this information, the NRC staff finds that the EDG design-basis loading would not be affected by EPU.

The licensee stated that there are no increases in safety-related loads, and no new safety-related loads powered from the Class 1E UPS system as a result of EPU. No increase in flow or pressure is required of any AC-powered ECCS equipment as result of EPU operation. Therefore, the amount of power required of the UPS to perform safety-related functions is not increased with EPU. Based on this information, the NRC staff finds that the existing Class 1E UPS system remains adequate to support required loads for safety shutdown, to maintain a safe shutdown condition, and to operate the engineered safety feature equipment following postulated accidents.

In letter dated January 21, 2013, the licensee stated that in order to implement the EPU, the reactor feed water flow needs to be increased and this requires additional pumping/motor capacity for the condensate and feedwater systems. Based on this power increase, the licensee stated that the existing reserve 1R and auxiliary 2R transformers need to be replaced and a new 13.8 kV distribution system needs to be installed. The new 13.8 kV Buses (11 and 12) will continue to supply the reactor feed pump and RRMG set drive motors. The new voltage at these buses requires replacing the RRMG drive motor, although there is no change in motor horsepower. In addition, the new condensate pump motors are also being relocated to new 13.8 kV Buses 11 and 12. The load increases in condensate and feedwater pump capacity for EPU result in electrical loads for onsite non-safety-related AC power systems that exceed the capacity of the existing system therefore requiring upgrades/replacements of the equipment. Based on its review, the staff finds that the increase in power demand will not impact any Class 1E buses or system as a result of the proposed EPU.

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the AC onsite power system and concludes that the licensee has adequately accounted for the effects of the proposed EPU on the system's functional design. The NRC staff further concludes that the AC onsite power system will continue to meet the MNGP USAR Principal Design Criteria 24 and 39 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the AC onsite power system.

2.3.4 DC Onsite Power System

Regulatory Evaluation

The direct current (DC) onsite power system includes the DC power sources and their distribution and auxiliary supporting systems that are provided to supply motive or control power to safety-related equipment. The NRC staff's review covered the information, analyses, and referenced documents for the DC onsite power system. The NRC's acceptance criteria for the DC onsite power system are based on GDC-17, "Electric power systems," insofar as it requires the system to have the capacity and capability to perform its intended functions during anticipated operational occurrences and accident conditions. The applicable Principal Design Criteria for MNGP predates this criterion. The MNGP Principal Design Criteria are listed in USAR Section 1.2, "Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed GDC (*Federal Register*, 32 FR 10213, July 11, 1967). Although not explicitly licensed to the AEC-proposed GDC published in 1967, the licensee performed a comparative evaluation of the design basis of MNGP with the AEC-proposed GDC of 1967. This evaluation is included in MNGP USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria."

Based on a review of the MNGP's USAR, the NRC staff identified the following Principal Design Criteria as being applicable to the proposed EPU application:

USAR Principal Design Criteria 24 and 39 require that alternate power systems shall be provided and designed with adequate independency, redundancy, capacity, and testability to permit the functioning required of the engineered safety features. As a minimum, the on-site power system and the off-site power system shall each, independently, provide this capacity assuming a failure of a single active component in each power system. In the event of the loss of all off-site power, sufficient alternate sources of power shall be provided to permit the required functioning of protection systems.

Specific review criteria are contained in SRP Sections 8.1 and 8.3.2

Technical Evaluation

The NRC staff reviewed the licensee's submittals to determine whether the DC system and its components would remain capable of performing their intended design function at EPU conditions. The licensee stated that at EPU conditions the integrated safety-related and SBO DC loads are not increasing and remain bounded by the existing battery capacity. While the EPU changes do not increase the magnitude of the individual DC loads, EPU changes do result in a change of timing sequence for certain loads such as the loads that support high pressure coolant injection system operation. The licensee stated that it has incorporated these changes into the DC system load profile and evaluated the changes against the DC system design criteria.

In its May 28, 2008, letter, the licensee provided the results of its DC battery calculations for MNGP to show the available battery margins before and after the proposed EPU. In its letter dated January 21, 2013 (Reference 67, Enclosure 1, Item 2), the licensee noted that DC Onsite Power System changes remain bounded by the available battery capacity and that a revision of station DC battery cell sizing calculation verified that acceptable margin remains after EPU implementation. The NRC staff reviewed the table summarizing the final battery sizing margins and finds that the capacity margins of the 125 V DC Division I and II and 250 V DC Division I and II batteries, which range from 8.11 percent to 22.81 percent under EPU conditions, are acceptable.

Based on its review of this information, the NRC staff finds that adequate margin remains in the MGNP batteries to support EPU conditions. Additionally, the NRC staff reviewed the licensee's general design loading assumptions for the MNGP batteries, and finds that the licensee has adequately demonstrated that sufficient power will be available to mitigate the consequences of an SBO event, which the licensee considers to be the design-basis loading scenario. Therefore, the NRC staff finds that the MNGP DC power system remains adequate to supply safety-related systems at EPU levels.

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the DC onsite power system and concludes that the licensee has adequately accounted for the effects of the proposed EPU on the system's functional design. The NRC staff further concludes that the DC onsite power system will continue to meet the MNGP USAR Principal Design Criteria 24 and 39 following implementation of the proposed EPU. Adequate physical and electrical separation exists and the system has the capacity and capability to supply power for all safety loads and other required equipment. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the DC onsite power system.

2.3.5 Station Blackout

Regulatory Evaluation

An SBO refers to a complete loss of AC electric power to the essential and nonessential switchgear buses in a nuclear power plant. SBO involves the LOOP concurrent with a turbine trip and failure of the onsite emergency AC power system. SBO does not include the loss of available AC power to buses fed by station batteries through inverters or the loss of power from alternate alternating current (AAC) sources. The NRC staff's review focused on the impact of the proposed EPU on MNGP's ability to cope with and recover from an SBO event for the period of time established in the plant's licensing basis. The NRC's acceptance criteria for SBO are based on 10 CFR 50.63. Specific review criteria are contained in SRP Sections 8.1 and Appendix B to SRP Section 8.2; and other guidance provided in Matrix 3 of RS-001.

Technical Evaluation

The licensee re-evaluated SBO using the guidelines of NUMARC [Nuclear Management and Resource Council, Inc.] 87-00, "Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors." The licensee stated that MNGP's response to and coping capabilities for an SBO event would be affected slightly by operation at EPU due to the increase in the initial power level and decay heat. However, the licensee indicated that no changes are necessary to the systems and equipment used to respond to an SBO and that the SBO coping duration does not change under EPU conditions.

The licensee stated that areas containing equipment necessary to cope with an SBO event were evaluated for the effect of loss-of-ventilation due to an SBO. The licensee's evaluation showed that equipment operability is bounded due to conservatism in the existing design and qualification bases. The battery capacity remains adequate to support high pressure coolant injection operation at EPU conditions. In addition, adequate compressed gas capacity exists to support main steam relief valve (MSRV) actuations.

Having adequate condensate inventory ensures that adequate water volume is available to remove decay heat and maintain reactor vessel level above the top of active fuel. The licensee calculated the required condensate inventory for decay heat removal (44,329 gallons) using the method described in NUMARC 87-00. The NRC staff confirmed that this quantity is within the available condensate storage tank (CST) inventory.

Conclusion

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

Based on the above, the NRC staff has evaluated the effect of the EPU on the necessary electrical systems and environmental qualification of electrical components. The NRC staff's evaluations show that the MNGP electrical systems design will continue to meet the MNGP USAR principal design criteria 24 and 39, 10 CFR 50.49, and 10 CFR 50.63 at EPU conditions.

Therefore, the proposed power uprate is acceptable with respect to SBO.

2.4 Instrumentation and Controls

This NRC staff reviewed the licensee's analysis of the suitability of existing instruments for EPU operation, the determination of instrument uncertainties, and setpoint determinations for systems affected by the EPU. The NRC staff predicated its evaluation of the identified instrumentation for the new power level upon the assumption that the licensee's analytical limits are based on the application of approved design codes.

Regulatory Evaluation

Instrumentation and control (I&C) systems are provided: (1) to control plant processes that have a significant impact on plant safety; (2) to initiate the reactivity control system, including control rods; (3) to initiate the engineered safety features and essential auxiliary supporting systems; and (4) for use to achieve and maintain the plant in a safe-shutdown condition. Diverse I&C systems and equipment are provided for the express purpose of protecting against potential common-mode failures of I&C protection systems. The NRC staff reviewed the reactor trip system, the engineered safety feature actuation system, safe-shutdown systems, control systems, and diverse I&C systems for the proposed EPU to ensure that the systems and any changes necessary for the proposed EPU are adequately designed so that the systems continue to meet their safety functions. The NRC staff also conducted its review to ensure that system failures do not affect safety functions. The NRC's acceptance criteria related to the design quality of protection and control systems are based on Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(a)(1), 10 CFR 50.55a(h), and draft General Design Criteria 1, 5, 11, 12, 13, 14, 15, 19, 20, 22, 23, 25, 26, 40, and 42. Specific review criteria appear in Sections 7.0, 7.2, 7.3, 7.4, 7.7, and 7.8 of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," issued March 2007.

Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company, the predecessor to NSPM, performed a comparative evaluation of the design basis of MNGP with the AEC proposed GDC of 1967. The MNGP comparative evaluation to the 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") associated with this review is contained in Appendix E of the MNGP USAR.

Technical Evaluation

The uprate application frequently references GE Nuclear Energy LTR NEDC-33004P, Revision 3, "Constant Pressure Power Uprate," which the NRC approved in a letter dated March 31, 2003 (Reference 82). MNGP also recently upgraded its power range monitoring system in preparation for this proposed EPU. The NRC approved the licensing amendment request for the power range monitoring system upgrade in Amendment No. 159, dated January 30, 2009 (Reference 70).

Suitability of Existing Instruments

For the proposed EPU, the licensee evaluated the existing instruments of the affected nuclear steam supply systems and balance-of-plant systems to determine their suitability for the revised operating range of the affected process parameters. Where operation at the power uprate condition affected safety analysis limits, the licensee verified that the acceptable safety margin continues to exist under all power uprate conditions. Where necessary, the licensee revised the setpoint and uncertainty calculations for the affected instruments. Apart from a few devices that needed changing, the licensee's evaluations found most existing instrumentation acceptable for proposed power uprate operation. The licensee's evaluation resulted in the changes outlined in the table below:

Instrument/Parameter	EPU Impact/Change
Average power range monitors (APRMs)	Will be calibrated to read 100% at EPU RTP level. Dispositioned by the constant pressure power uprate (CPPU) licensing topical report (CLTR).
Intermediate range monitors (IRMs)	Will be adjusted, in accordance with normal plant procedures, to ensure that overlap with APRMs and source range monitors is adequate. Dispositioned by the CPPU CLTR.
Main steamline high flow	Setpoint changes reflected in terms of percent of rated steam flow to maintain current absolute allowable value.
Turbine first stage pressure scram bypass instruments	Recalibrate or replace turbine first-stage pressure instruments and verify/validate relationship between instruments and RTP during startup testing following replacement of the high-pressure turbine.
Feedwater control system	Instrumentation will be recalibrated (or replaced) before EPU implementation. Dispositioned by the CPPU CLTR.
Reactor steam dome pressure permissive – bypass timer (pump permissive)	Setpoint change from $\leq$ 22 minutes to $\leq$ 18 minutes to maintain margin for EPU conditions

The licensee stated that it will make these changes to accommodate the revised process parameters. Discussion of instrumentation and parameter changes that are covered by the TSs occur later in this section. Since these changes are based on the system review and analysis reviewed by the NRC staff (see other sections of this safety evaluation), and because the licensee will confirm the acceptability of these changes during power ascension testing, the NRC staff agrees with the licensee's conclusion that when these modifications and changes are implemented, MNGP's I&C systems will accommodate the proposed power uprate without compromising safety. None of the above changes affects the licensee's compliance with the existing plant licensing basis; therefore, the NRC staff finds that MNGP will continue to meet its current regulatory basis.

Instrument Setpoint Methodology

As noted above, the NRC staff approved the power range monitoring system upgrade with Amendment No. 159, which included a number of TS and setpoint revisions associated with the APRMs. These changes were based on NEDC-31336P, "General Electric Instrument Setpoint Methodology," to calculate revised values. Section 3.3 of the safety evaluation associated with Amendment No. 159 discusses the acceptability of the MGNP setpoint methodology.

The Main Steam Line-High Flow setpoint is used to detect and isolate main steamline breaks outside of containment, and to do so through closure of the main steam isolation valves. The setpoint is based on the differential pressure across the flow restrictors, which corresponds to the steam flow rate in the pipe. Calculation CA-95-075, Revision 1, "Main Steam Line High Flow Setpoint" (enclosed with the licensee's February 27, 2013, submittal), documents the licensee's evaluation of the change. The values presented in Section 6.5.4 of the document, which include appropriate consideration of expected uncertainties, demonstrate that the operating setpoint may be set as high as 148 psid at EPU conditions; however, the licensee is conservatively keeping the setting at 143 psid which corresponds to a TS allowable value of 116.9 percent rated steam flow. These values properly account for a 10 CFR Part 21 Communication from GEH regarding an error in the Main Steam Line High Flow Calculational Methodology.

The licensee's methodology for calculating acceptable as-found (AAF) and acceptable as-left (AAL) values meets the guidance in Regulatory Issue Summary (RIS) 2006-17, "NRC Staff Position on the Requirement of 10 CFR 50.36, Technical Specifications, Regarding Limiting System Settings During Periodic Testing and Calibration of Instrument Channels," dated August 24, 2006, and, therefore, is acceptable to the NRC staff.

The Turbine First Stage Pressure setpoint serves to reduce scrams at low-power levels, where the steam bypass system is in effect for turbine trips and generator load rejections. The current setpoint is 125 psig and is being revised to 95 psig for EPU operations. Calculation CA-96-054, Revision 5, "Turbine Stop Valve Closure/Generator Load Reject Scram Bypass," dated May 6, 2009 (included in the licensee's May 13, 2009, letter), documents the licensee's evaluation of the change. The licensee specifically proposed to alter the TS from which the setpoint was derived to allow the bypass to enact at an equivalent megawatt thermal value. The values presented in Section 6.5.4 of the document, which include appropriate consideration of expected uncertainties, demonstrate that the operating setpoint may be set as high as 100.6 psig at EPU conditions; however, the licensee is conservatively setting the value at 95 psig. The licensee committed to confirm the relationship of turbine first-stage pressure to reactor power during startup testing and to make any adjustments to this calculation at that point, if necessary. The licensee's methodology for calculating AAF and AAL values meets the guidance in RIS 2006-17 and, therefore, is acceptable to the NRC staff.

The Reactor Vessel Water Level-Low setpoint is used to scram the reactor in the event that the water level drops too low in the reactor vessel. Because of the increased steam flow under EPU operations, the licensee revised the analytical limit from 0 inches to -2.5 inches. The licensee's evaluation of the change is documented in Calculation CA-95-073, Revision 4, "Reactor Low Water Level SCRAM Setpoint," dated December 19, 2008 (included in letter dated May 13, 2009). Although the calculation demonstrates that the setpoint could be set below the current value for EPU conditions, including appropriate consideration of expected uncertainties, the licensee is conservatively maintaining the current setpoint (+9 inches; see Section 6.6.4). Therefore, this setpoint will not change. The licensee's methodology for calculating acceptable AAF and acceptable AAL values meets the guidance in RIS 2006-17 and, therefore, is acceptable to the NRC staff.

The Reactor Steam Dome Pressure Permissive - Bypass Timer (Pump Permissive) (called ADS bypass timer) setpoint allows the CS and LPCI pumps to start on a Reactor Vessel Water Level-Low signal after a time delay, even if the reactor steam dome pressure is above its permissive setpoint. This ensures that the starting of low pressure ECCS subsystem pumps will occur on a Reactor Vessel Water Level - Low signal after a time delay. The current setpoint is ≤ 22 minutes and is being revised to ≤ 18 minutes for EPU operation. Calculation CA-03-036, Revision 2, "Instrument Setpoint Calculation - Reactor Low Pressure Permissive Bypass Timer," dated April 9, 2013 (included as Enclosure 3 of Reference 68), documents the licensee's evaluation of the change. The licensee specifically proposed to alter the TS from which the setpoint was derived to allow the bypass to occur at an equivalent time value for EPU conditions. The values presented in Section 6.5.4 of the document, which include appropriate consideration of expected uncertainties, demonstrate that the operating setpoint may be set as high as 15.48 minutes at EPU conditions; however, the licensee is conservatively setting the value at 15 minutes. The licensee's methodology for calculating AAF and AAL values meets the guidance in RIS 2006-17 and, therefore, is acceptable to the NRC staff.

Based on the above, the NRC staff concludes that there is reasonable assurance that MNGP will operate in accordance with the licensee's safety analysis, and that operability of the instrumentation is ensured. Therefore, the NRC staff finds that the proposed changes meet the requirements of 10 CFR 50.36, "Technical Specifications," and the guidance in RG 1.105, Revision 3, "Setpoints for Safety-Related Instrumentation," issued in December 1999.

Technical Specification Changes Related to the Power Uprate

The licensee proposed TS changes to I&C-related systems. These changes are further evaluated in Section 3.2 of this SE.

<u>Conclusion</u>

The NRC staff reviewed the licensee's application, as supplemented, related to the effects of the proposed EPU on the functional design of the reactor trip system, engineered safety feature actuation system, safe-shutdown system, and control systems. The NRC staff concludes that the licensee adequately addressed the effects of the proposed EPU on these systems, and that the changes necessary to achieve the proposed EPU are consistent with MNGP's design basis. Furthermore, the NRC staff concludes that the systems will continue to meet the requirements of 10 CFR 50.55a(a)(1), 10 CFR 50.55(a)(h), and draft GDCs-1, 5, 11, 12, 13, 14, 15, 19, 20, 22, 23, 25, 26, 40 and 42. Therefore, the NRC staff finds the licensee's proposed EPU acceptable with respect to instrumentation and controls.

2.5 <u>Plant Systems</u>

Enclosure 8 of the licensee's application dated November 5, 2008, lists planned design changes and modifications. The modifications listed in Enclosure 8 are not regulatory commitments. The licensee stated that modifications listed in Enclosure 8 would be implemented in accordance with the requirements of 10 CFR 50.59. Changes to the list of planned design changes and modifications were identified in the licensee's letter dated January 21, 2013. The NRC staff identified the following modifications related to the balance-of-plant:

Main Steam

The licensee plans to replace the existing cross-around relief valves and associated discharge piping to provide increased relieving capacity under CPPU conditions. The licensee also plans to replace the solenoid valves on the inboard MSIVs to increase the margin between maximum containment pressure and minimum nitrogen supply pressure and modify or replace the main steam flow transmitters to accommodate increased flows under CPPU conditions. Finally, the licensee plans to upgrade MSRV actuators to address obsolescence issues.

Main Turbine

The licensee plans to replace the existing high pressure (HP) turbine steam path with a new rotor and diaphragms to accommodate increased steam flow under CPPU conditions. The licensee also plans to replace several diaphragm sets and one set of buckets in each low pressure (LP) turbine to accommodate increased steam flow under CPPU conditions.

Condensate and Feedwater

The condensate and feedwater flow rates will increase approximately in proportion to the uprate power increase. The licensee plans to replace the condensate pump motors and main feedwater pump motors with motors rated for CPPU operation. Also, the licensee plans to replace the condensate pump internals and the entire main feedwater pump assembly to support CPPU operation. In addition, the licensee plans to replace the existing feedwater regulating valves with new ones sized for operation under CPPU conditions. Finally, the licensee plans to modify or replace the condensate and feedwater flow transmitters and pressure control instrumentation to maintain functionality with increased flows and pressure drops under CPPU conditions.

Feedwater Heaters, Heater Drains, and Condensate Demineralizers

The CPPU will result in increases in the temperatures, pressures, and flows in the various feedwater heaters, heater drains and other components. The licensee plans to rerate the 11 and 12 feedwater heaters, and replace the existing 13, 14, and 15 feedwater heaters with new heaters sized for CPPU conditions. The licensee also plans to replace, re-analyze, or modify the existing drain coolers for the 11 and 12 feedwater heaters and the moisture separator to maintain margin under CPPU conditions. In addition, the licensee plans to replace the existing condensate demineralizer vessels with new vessels to accommodate increased flow under CPPU conditions.

2.5.1 <u>Internal Hazards</u>

2.5.1.1 <u>Flooding</u>

Flood Protection

For proposed power uprates, the NRC staff reviews flood protection to ensure that SSCs important to safety are adequately protected from the consequences of internal flooding that result from postulated failures of tanks and vessels; flooding due to pipe failures is evaluated in Section 2.5.1.3. Because this portion of the NRC staff's review focuses on increases in fluid volumes in tanks and vessels that will occur as a result of the power uprate, the NRC staff reviewed proposed modifications related to the proposed CPPU that could increase the volumes contained in tanks and vessels. The NRC staff reviewed the proposed modifications listed in Enclosure 8 of the November 5, 2008, application and determined that the volume of tanks and vessels will not increase as a result of the CPPU. Therefore, the NRC staff has determined that an evaluation of this particular section is not required.

Equipment and Floor Drains

The function of the equipment and floor drainage system (EFDS) is to assure that waste liquids, valve and pump leakoff, and tank drains are directed to the proper area for processing or disposal. The EFDS consists of the radioactive and nonradioactive waste drainage and collection systems. The radioactive and nonradioactive drainage systems are segregated to prevent the transfer of radioactive contamination to the nonradioactive liquid wastes and uncontrolled access areas. The licensee indicated that the EFDS has adequate capacity to accept the small increase in liquid volumes directed to system drains that result from the proposed CPPU (Section 2.5.1.1.2 of the MNGP PUSAR). The licensee also stated in Section 2.5.1.1.2 of the PUSAR that the EDFS backflow at maximum flood levels will not change as a result of the CPPU. Therefore, the NRC staff has determined that an evaluation of the EFDS is not required.

Circulating Water System

The circulating water system (CWS) provides a continuous supply of cooling water to the main condenser to remove excess heat from the turbine cycle and auxiliary systems. For the proposed CPPU, the NRC staff's review of the CWS focuses on the impact that the proposed uprate will have on existing flooding analyses due to any increases that may be necessary in fluid volumes and installation of larger capacity CWS pumps or piping. MNGP is not installing larger CWS pumps or CWS piping for CPPU operation. Therefore, the NRC staff has determined that an evaluation for the CWS in this section is not required because the proposed power uprate will not affect the licensee's flooding analysis for the CWS.

2.5.1.2 Missile Protection

Internally Generated Missiles

Regulatory Evaluation

The NRC staff's review concerns the protection of SSCs important to safety from missiles that could result from in-plant component overspeed conditions and high-pressure system ruptures. Potential missile sources include pressurized systems and components, and high-speed rotating machinery. The purpose of the NRC staff's review is to confirm that SSCs important to safety are protected from internally generated missiles. The NRC staff's review for proposed power uprates focuses on modifications that affect the location of important-to-safety SSCs relative to postulated missile sources and the adequacy of existing missile barriers for potential increases in the energy of postulated missiles. The criteria that are most applicable to the NRC staff's review of the protection of SSCs important to safety from the effects of internally generated missiles for proposed power uprates are based on GDC-4, "Environmental and Dynamic Effects Design Basis," insofar that SSCs important to safety should be protected from the effects of internally generated missiles, and other licensing basis considerations that are applicable.

The NRC staff's review related to internally generated missiles is performed in accordance with the guidance in Section 2.1 of RS-001, Matrix 5. Acceptability for EPU operation is judged based upon conformance with existing licensing-basis considerations as discussed primarily in Sections 5.2.3.5.3, "Drywell Missile Protection," and 8.8.2, "Original Separation Criteria for the Primary Containment Isolation System (PCIS) and the Engineered Safeguards Systems," of the MNGP USAR, except where proposed changes are found to be acceptable based upon the specified review criteria.

While MNGP is not explicitly licensed to the current General Design Criteria or the 1967 AEC-proposed GDC, the licensee has made a comparison of the current GDC to the applicable AEC-proposed General Design Criteria. For the current GDC-4, the licensee's evaluation of the analogous 1967 AEC-proposed General Design Criteria is also contained in MNGP USAR, Appendix E: draft GDC-40 and draft GDC-42.

Technical Evaluation

The licensee evaluated the impact of the proposed CPPU on the possibility of higher system pressures or changes in existing system configuration to affect the missile protection afforded equipment important-to-safety. The licensee determined that the CPPU does not result in any condition (significant system pressure increase or equipment overspeed) that could result in an increase in the generation of internally generated missiles. In addition, the licensee determined that the CPPU does not entail any changes in equipment configurations that could change the effect of internally generated missiles on safety-related or non-safety-related equipment.

The reactor feed pressure will increase a small amount to accommodate the increased feedwater flow for CPPU operation. However, this pressure change does not affect the basis for acceptance of the existing missile barriers. The energy of postulated missiles with the potential to affect containment integrity, as described in Section 5.2.3.5.3 of the MNGP USAR, was based in part on the velocity of choked flow though a postulated break, which is not affected by the pressure increase. Similarly, Section 8.8.2 of the MNGP USAR described criteria for train separation that is independent of the energy of postulated missiles. The train separation criteria specified either a 20 foot separation or a 6-inch reinforced concrete missile barrier as providing adequate train separation. For locations inside the drywell where limited space may prevent attainment of the minimum separation, Section 8.8.2 of the MNGP USAR states that care was taken to locate redundant cable raceways so that a single missile will not damage both channels. Therefore, consistent with the MNGP licensing basis, SSCs important to safety will continue to be adequately protected from internally generated missiles, consistent with the facility's licensing basis, following CPPU implementation.

Conclusion

The NRC staff has reviewed the licensee's assessment of changes necessary to support the proposed CPPU and finds that SSCs important to safety will continue to be protected from the effects of internally generated missiles in accordance with licensing-basis assumptions. Therefore, the NRC staff concludes that implementation of the proposed CPPU is acceptable with respect to the protection of SSCs important to safety from internally generated missiles.

Turbine Generator

Regulatory Evaluation

The turbine generator (TG) does not perform a safety function and it is not safety-related. However, the TG is of regulatory significance because the large steam turbines of the TG set have the potential for producing high energy missiles, especially if the turbines exceed their rated speed. The turbine control system, main stop valves, control valves, intercept and intermediate stop valves control the turbine speed and include design features that prevent turbine overspeed conditions. The NRC staff's review of the TG for proposed power uprates focuses on the effects of the proposed EPU on the turbine overspeed protection features to confirm that adequate turbine overspeed protection will be maintained. The criteria that are most applicable to the NRC staff's review of the TG for proposed power uprates are based on GDC-4, "Environmental and Dynamic Effects Design Basis," insofar that SSCs important to safety should be protected from the effects of turbine missiles, and other licensing basis considerations that are applicable.

The NRC staff's review related to the TG is performed in accordance with the guidance provided in Section 2.1 of RS-001, Matrix 5. Acceptability for EPU operation is judged based upon conformance with existing licensing-basis considerations as discussed primarily in Sections 11.2, "Turbine-Generator System;" 7.7, "Turbine-Generator System Instrumentation and Control;" and 12.2.3, "Turbine Missile Analysis," of the MNGP USAR, except where proposed changes are found to be acceptable based upon the specified review criteria.

While MNGP is not explicitly licensed to the current General Design Criteria or the 1967 AEC-proposed GDC, the licensee has made a comparison of the current GDC to the applicable AEC-proposed GDC. For the current GDC-4, the licensee's evaluation of the analogous 1967 AEC-proposed General Design Criteria is also contained in MNGP USAR, Appendix E: draft GDC-40 and draft GDC-42.

Technical Evaluation

The licensee's evaluation of the impact that the proposed CPPU will have on the capability to prevent overspeed of the main turbine is provided in Section 2.5.1.2.2 of the MNGP PUSAR. The high-pressure and low-pressure turbine rotors at MNGP (for both current licensed thermal power and CPPU) have integral, non-shrunk-on wheels. Per CLTR Section 7.1, a separate rotor missile analysis is not required for plants with integral wheels.

The overspeed calculation compares the entrapped steam energy contained within the turbine and the associated piping after the stop valves trip to the sensitivity of the rotor train to overspeed. The entrapped energy increases slightly for the EPU conditions. The hardware modification design and implementation process establishes the overspeed trip settings to provide protection for a turbine trip.

The slight increase in entrapped energy may result in a small increase in the peak turbine speed following load rejection events. However, Section 12.2.3 of the MNGP USAR states that both the rotor and bucket failure speeds are well above the redundant turbine overspeed protection feature trips of the turbine, which have current setpoints of 110 percent and 112 percent of rated speed. Therefore, the small increase in peak turbine speed would not affect the probability of turbine damage due to overspeed, and the existing design basis of the main turbine remains acceptable for CPPU operation.

Conclusion

The NRC staff has reviewed the licensee's assessment of the impact that the proposed CPPU will have on overspeed protection of the main turbine and finds that the existing overspeed trip setpoints will continue to protect against damaging main turbine overspeed conditions consistent with main turbine missile design-basis considerations. Therefore, the proposed CPPU is considered to be acceptable with respect to the TG.

2.5.1.3 Pipe Failures

Regulatory Evaluation

The failure of high-energy piping can cause pipe whip, jet impingement, and harsh environmental conditions that can result in damage and render SSCs inoperable. The NRC staff's review for EPUs is concerned with the impact that the proposed power uprate will have on the capability that is credited for mitigating the failure of high and moderate energy fluid piping located outside containment and for safely shutting down the plant in accordance with the plant licensing basis. The NRC staff's review focuses on those system modifications and increases in system pressures and temperatures that are necessary in order to implement the proposed power uprate to confirm that the limitations and assumptions of previous pipe failure analyses remain valid or are otherwise addressed. The acceptance criteria that are most applicable to the NRC staff's review of postulated pipe failures for proposed power uprates are based on GDC-4, "Environmental and Dynamic Effects Design Bases," insofar that SSCs important to safety should be appropriately protected against the dynamic effects of postulated pipe ruptures, including the effects of pipe whipping and discharging fluids, and other licensing-basis considerations that are applicable.

The NRC staff's review related to postulated pipe failures is performed in accordance with the guidance provided in Matrix 5 of RS-001. Piping failures outside containment are described in Appendix I, "Evaluation of High Energy Line Breaks Outside Containment," of the MNGP USAR. Acceptability for EPU operation is judged based upon conformance with existing licensing-basis considerations, except where proposed changes are found to be acceptable based upon the specified review criteria.

While MNGP is not explicitly licensed to the current General Design Criteria or the 1967 AEC-proposed GDC, the licensee has made a comparison of the current GDC to the applicable AEC-proposed GDC. For the current GDC-4, the licensee's evaluation of the analogous 1967 AEC-proposed General Design Criteria is also contained in MNGP USAR, Appendix E: draft GDC-40 and draft GDC-42.

Technical Evaluation

The licensee's evaluation of the impact of postulated high energy line breaks (HELBs) outside containment is provided in PUSAR Sections 2.5.1.1.1, "Flood Protection;" 2.2.2.1, "Pipe Whip and Jet Impingement;" 2.2.5, "Seismic and Dynamic Qualification of Mechanical and Electrical Equipment;" and 2.3.1, "Environmental Qualification of Electrical Equipment." Because the power uprate is performed at constant pressure with no changes in reactor steam pressure or enthalpy, the effects of postulated HELBs in steam piping will not change for CPPU. For postulated high energy liquid line breaks, the licensee determined that use of a more conservative analysis assumption was appropriate. The change in assumptions combined with the increased pressure in some lines as a result of the EPU produced an increase in the calculated mass and energy release rates for postulated breaks in the main feedwater, condensate, and reactor water cleanup (RWCU) systems. The licensee determined that postulated breaks in the control rod drive and zinc injection systems will be bounded by postulated breaks in the RWCU, feedwater, and condensate systems, respectively. The licensee also determined that existing evaluations and analyses, addressing postulated high energy line breaks in the standby liquid control system, the offgas system, and instrumentation and sampling lines will remain valid at CPPU operating conditions. Therefore, the NRC staff's evaluation was limited to HELB events originating in the main feedwater, condensate, and RWCU systems.

For the main feedwater, condensate, and RWCU systems, the licensee determined that the peak mass release rate increases substantially, largely due to the change in analysis assumptions. The change in the analysis involved consideration of fire sprinkler actuation as a result of the HELB in certain plant areas. The licensee also determined that the integrated energy release for postulated feedwater and condensate system HELBs would increase by a small amount, and the integrated mass release for the condensate system break would increase due to changes in the assumed initial condenser hotwell level. These calculated changes resulted in small increases in the liquid line HELB flood elevation, peak pressure, and peak temperature in both the reactor building and the turbine building. The licensee evaluated the temperature, pressure and humidity profiles that were not bounded by the CLTP conditions, and determined that the changes do not adversely affect the qualification of safety-related electrical equipment. The licensee's evaluation indicated that SSCs important to safety will continue to be protected from flooding and will continue to meet the requirements of the current licensing basis following implementation of the proposed EPU. As described in the supplementary information provided by letter dated June 12, 2009, the licensee based this determination on evaluations demonstrating that at least one division of engineered safety systems would remain operable following postulated internal flooding events, which is consistent with the evaluations provided in Section I.5 in Appendix I of the MNGP USAR. Therefore, the licensee concluded the proposed EPU is acceptable with respect to protection from flooding associated with HELB events.

The licensee found that because the pressure considered in the high-energy piping evaluations encompasses pressures at EPU conditions with the existing feedwater and condensate pumps, there are neither increased jet impingement loads on HELB targets nor increased pipe whip loads on HELB targets and pipe whip restraints. The licensee stated that installation of new condensate and feedwater pumps with associated piping modifications will include an evaluation of HELB target impact as part of the planned modification, which will be implemented pursuant to 10 CFR 50.59. The installation of new pumps would affect the peak mass release rates due to the potential for higher discharge pressures from the higher capacity pumps. However, factors that affect peak flooding elevations, such as the integrated mass release, are more dependent on the assumed initial inventory of the condensate and feedwater systems than on the design characteristics of system pumps.

Based on a review of the information provided, the NRC staff found that the licensee adequately evaluated and addressed the impact of the proposed CPPU on the consequences of postulated high energy and moderate energy pipe failures, including flooding considerations. The licensee determined that the proposed CPPU will not result in any new limiting pipe failure locations and the consequences of postulated pipe failures will not exceed plant design limitations that were previously recognized and used in the existing flood evaluations. Therefore, the NRC staff agrees that the capability to mitigate postulated HELBs in accordance with the licensing-basis considerations will not be compromised by operating at the proposed CPPU power level.

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed CPPU on the consequences of postulated high-energy pipe failures and finds that protection of SSCs important to safety from the effects of HELBs will continue to satisfy the MNGP licensing basis. Therefore, the proposed changes associated with the CPPU, with the exception of the replacement of the feedwater and condensate pumps that the licensee stated would be evaluated separately, are acceptable with respect to the consequences of postulated high energy pipe failures outside containment.

2.5.1.4 Fire Protection

Enclosure 5, "Safety Analysis Report for Monticello Constant Pressure Power Upate," Section 2.5.1.4, "Fire Protection," of the November 5, 2009, application provides the technical information associated with the fire protection program.

Regulatory Evaluation

The purpose of the fire protection program is to provide assurance, through a defense-in-depth design, that a fire will not prevent the performance of necessary plant safe-shutdown functions nor will it significantly increase the risk of radioactive releases to the environment. The NRC staff's review focused on the effects of the increased decay heat on the plant's safe-shutdown analysis to ensure that SSCs required for safe-shutdown of the plant are protected from the effects of the fire and will continue to be able to achieve and maintain safe-shutdown following a fire.

The NRC staff's acceptance criteria for the fire protection program are based on (1) 10 CFR 50.48, "Fire protection," insofar as it requires the development of a fire protection program to ensure, among other things, the capability to safely shutdown the plant; (2) GDC-3, "Fire protection," insofar as it requires that (a) SSCs important to safety be designed and located to minimize the probability and effect of fires, (b) noncombustible and heat resistant materials be used, and (c) fire detection and suppression systems be provided and designed to minimize the adverse effects of fires on SSCs important to safety; and (3) GDC-5, "Sharing of structures, systems, and components," insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions. Specific review criteria are contained in Appendix D of NUREG-0800, Revision 5, "Standard Review Plan," Section 9.5.1, as supplemented by the guidance provided in Attachment 2 to Matrix 5 of Section 2.1 of RS-001, Revision 0, "Review Standard for Extended Power Uprates."

The MNGP fire protection program describes the fire protection features of the plant necessary to comply with BTP 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants." The safety evaluation reports (SERs) dated August 29, 1979, February 12, 1981, and October 2, 1985, describe the approved fire protection program for MNGP. These SERs are listed in the MNGP Unit No. 1 Operating License Condition 2.C(4).

While MNGP is not explicitly licensed to the current General Design Criteria or the 1967 AEC-proposed GDC, the licensee has made a comparison of the current GDCs to the applicable AEC-proposed GDC. For the current GDCs listed above, the licensee's evaluation of the analogous 1967 AEC-proposed General Design Criteria is also contained in MNGP USAR, Appendix E: draft GDC-3 and draft GDC-4.

Technical Evaluation

In the application, the licensee evaluated the applicable SSCs and safety analyses at the proposed EPU core power level of 2,004 MWth. The NRC staff's review of Enclosure 5, Section

2.5.1.4, identified areas in which additional information was necessary to complete the review of the proposed EPU amendment. By letter dated May 13, 2009, NSPM responded to the NRC staff's request for additional information (RAI) as discussed below.

In RAI-1, the NRC staff noted that RS-001, Revision 0, "Review Standard for Extended Power Uprates," Attachment 1 to Matrix 5, "Supplemental Fire Protection Review Criteria," states that:

> Power uprates typically result in increases in decay heat generation following plant trips. These increases in decay heat usually do not affect the elements of a fire protection program related to (1) administrative controls, (2) fire suppression and detection systems, (3) fire barriers, (4) fire protection responsibilities of plant personnel, and (5) procedures and resources necessary for the repair of systems required to achieve and maintain cold shutdown. In addition, an increase in decay heat will usually not result in an increase in the potential for a radiological release resulting from a fire. However, the licensee's license amendment request should confirm that these elements are not impacted by the extended power uprate.

The licensee stated that the increase in decay heat from the proposed EPU does not affect the fire protection program elements: administrative controls, fire suppression and detection systems, fire barriers, fire protection responsibilities of plant personnel, and procedures and resources necessary for the repair of systems required to achieve and maintain cold shutdown. Further, the licensee stated that there are no design-basis events for MNGP that involve radiological releases that result from fires at CLTP or EPU operating conditions, and the increased decay heat from EPU will not result in an increase in the potential for a radiological release resulting from a fire from a design-basis event. The licensee's response to RAI-1 satisfactorily addresses the NRC staff's concerns, and this RAI-issue is considered resolved on the basis that the proposed EPU would not revise fire protection program elements, i.e., fire suppression and detection systems, fire barriers, responsibilities of plant personnel, and resources for the repair of systems required to achieve and maintain cold shutdown. The licensee indicated that for the proposed EPU condition, there is no increase in the potential for a radiological release resulting from a fire.

In RAI-2, the NRC staff noted that the results of the Appendix R evaluation for EPU are provided in Section 2.5.1.4, "Fire Protection," of Enclosure 5 to the application. However, this section does not discuss the time necessary for the repair of systems required to achieve and maintain cold shutdown nor the increase in decay heat generation following plant trips. The NRC staff requested the licensee to verify that the plant can meet the 72-hour requirements in 10 CFR Part 50, Appendix R, Sections III.G.1.b and III.L, with increased decay heat at EPU conditions. By letter dated May 13, 2009, the licensee provided its response, stating that the Appendix R safe-shutdown analysis does not require or credit any repair of the plant systems for CLTP or EPU conditions. Further, the licensee indicated that, at the EPU condition for the Appendix R event, which included the decay heat effects associated with the power increase, the reactor can be brought to cold shutdown conditions within 44.7 hours of the event. Cold shutdown can then be maintained using the existing plant systems previously designated for this purpose at CLTP conditions. In addition, the reactor can be brought to a cold shutdown condition within 72 hours at EPU conditions using the existing alternate shutdown cooling systems designated for mitigation of the Appendix R event and then maintained in cold shutdown, for compliance with the 10 CFR Part 50, Appendix R, Sections III.G.1.b and III.L.

The licensee's response satisfactorily addresses the NRC staff's concern and RAI-2 is considered resolved based on the following discussion. For the EPU condition, the licensee reviewed its systems to achieve and maintain the plant in cold shutdown condition. The results demonstrate that additional decay heat generation would not impact the ability to repair systems necessary to achieve and maintain cold shutdown (10 CFR Part 50, Appendix R, Section III.G.1.b) or achieve cold shutdown conditions (10 CFR Part 50, Appendix R, Section III.L) within 72 hours.

In RAI-3, the NRC staff noted that Enclosure 5 Section 2.5.1.4 states that the Appendix R fire event was analyzed for the two cases at EPU conditions. The licensee stated that the operator actions required to mitigate the consequences of a fire are not affected nor is there a need for any new operator actions. The NRC staff requested the licensee to verify that additional heat in the plant environment from the EPU will not: (1) interfere with required operator manual actions being performed at their designated time; or (2) require any new operator actions. By letters dated May 13 and August 12, 2009, the licensee provided following response:

> The governing procedure for operator actions during an Appendix R event is Abnormal Operating Procedure C.4-C, "Shutdown Outside of the Control Room." The remote shutdown is accomplished at Alternate Shutdown System Panel C-292 on the third floor of the Emergency Filtration Train (EFT) Building, which is accessed via the Plant Administration Building. Operator actions to maintain hot shutdown and place the reactor in a cold shutdown condition are performed at this panel. The EPU does not present conditions that interfere with or change the operator actions necessary to achieve hot or cold shutdown for the design-basis Appendix R event. The Appendix R procedure includes a step to implement Procedure C.4-B.08.07.A (Ventilation System Failure) as appropriate to provide ventilation to Division II vital electrical spaces. The ventilation procedure includes steps to monitor plant areas in the event ventilation is lost, and open doors or use pre-positioned fans if necessary. The need for performing these compensatory actions depends upon whether hotter than normal outside air temperatures exist. The licensee stated that the EPU has no effect on implementing this contingency procedure or the execution of the compensatory actions.

The licensee's response satisfactorily addresses the NRC staff's concerns. For the EPU condition, the licensee updated the fire safe-shutdown analysis as described in Enclosure 5, Section 2.5.1.4, "Fire Protection," of the application. The licensee identified that the proposed EPU does not present conditions that interfere with or change the operator actions necessary to achieve hot or cold shutdown for the Appendix R event. Based on its review, the NRC staff concludes that the proposed EPU does not impact operator manual actions.

In RAI-4, the NRC staff noted that some plants credit aspects of their fire protection system for other than fire protection activities, (e.g., using the fire water pumps and water supply as backup cooling or inventory for non-primary reactor systems). If the MNGP credits its fire protection system in this way, the EPU application should identify the specific situations and discuss to what extent, if any, the EPU affects these "non-fire-protection" aspects of the plant fire protection system. If the MNGP does not take such credit, the NRC staff requested that the licensee verify this as well. By letter dated May 13, 2009, the licensee provided the following response.

The licensee stated that MNGP does not credit the fire protection system in mitigation sequences for any design-basis event; however the fire protection system can be used as a backup to the primary sources of cooling water in severe accident sequences as shown in the table that was provided. The use of the fire protection system in these sequences would be due to failures of various primary makeup systems, and this secondary cooling function is not dependent on reactor power or the plant changes that occur due to EPU. None of these events where the primary cooling functions are lost are postulated to occur simultaneously with a fire event.

The licensee's response satisfactorily addresses the NRC staff's concerns, and RAI- 4 is considered resolved based on the following. The licensee indicated that the fire protection system is not credited in any mitigation sequences for any design-basis event. The licensee has credited the fire protection system as a backup to the primary source of cooling water in severe accidents for the following three accident sequences: (1) loss of Intake Structure - Backup cooling water to the "A" Residual Heat Removal (RHR) Heat Exchanger to bring the reactor to cold shutdown if RHR Service Water flow is lost; (2) Reactor Pressure Vessel (RPV) Makeup Alternate Injection - Alternate injection of cooling water to the reactor via the fire water cross-tie to Low-Pressure Coolant Injection (LPCI); and (3) Emergency Fuel Pool Cooling - Emergency cooling to the spent fuel pool (SFP) via a hose station or via the RHR System. The licensee has evaluated these scenarios and concluded that backup cooling for mitigating actions using the fire water system is not impacted by EPU conditions.

The information provided in the application, as supplemented, satisfactorily demonstrates that compliance with the fire protection and safe-shutdown program will not be affected, i.e., the NRC staff's EPU evaluation did not identify changes to design or operating conditions that will impact the post-fire safe-shutdown capability. The EPU does not change the credited equipment necessary for post-fire safe-shutdown nor does it require reroute of essential cables or relocation of essential components/equipment credited for post-fire safe-shutdown. The licensee has made no changes to the plant configuration or combustible loading as a result of modifications necessary to implement the EPU that affect the MNGP fire protection program.

Conclusion

The NRC staff has reviewed the licensee's fire-related safe-shutdown assessment and concludes that the licensee has adequately accounted for the effects of the 13 percent increase in decay heat on the ability of the required systems to achieve and maintain safe-shutdown conditions. The NRC staff finds that the capability of the associated SSCs to perform their design basis functions at an increased core power level of 2,004 MWth acceptable with respect to fire protection.

2.5.2 Fission Product Control

Fission Product Control Systems and Structures

The purpose of the NRC staff's review of fission product control systems and structures is to confirm that the current analyses remain valid or have been revised, as appropriate, to properly reflect the proposed EPU conditions. Consequently, the NRC staff's review focuses primarily on any adverse effects that the proposed EPU might have on the assumptions that were used in analyses that were previously completed. Because the impact of the proposed CPPU on fission product control systems and structures are encompassed by the evaluations that are completed in Section 2.6, "Containment Review Considerations," Section 2.7, "Habitability, Filtration, and Ventilation," and Section 2.9, "Source Terms and Radiological Consequences," a separate evaluation in this section is not required.

Main Condenser Evacuation System

The main condenser evacuation system (MCES) is a non-safety-related system that is used for establishing a vacuum in the condenser during startup and for maintaining the vacuum during normal plant operation. It also removes the non-condensable gases from the main condenser and air ejectors during normal operation and discharges these gases to the gaseous radwaste system. The Main Condenser Evacuation System is described in MNGP USAR Section 11.3.2, "Main Condenser Gas Removal System." The MCES is sized based upon the volume of the condenser and desired evacuation time, neither of which is impacted by the proposed CPPU, and the licensee proposed no modification to the system for CPPU. Consequently, the existing capability to monitor the MCES effluent is also not affected by the proposed CPPU and therefore, NRC review of the MCES is not required.

Turbine Gland Sealing System

The turbine gland sealing system (TGSS) is a non-safety-related system that provides sealing steam for the main turbine shafts and selected valve stem packing to prevent air in-leakage and the escape of steam, thereby preventing the uncontrolled release of radioactive material in the steam to the environment. A discussion of the turbine gland sealing system is included in MNGP USAR Section 11.3.2, "Main Condenser Gas Removal System." Because no modifications are being made to the TGSS that are of consequence and non-condensable gases will continue to be monitored for radiation, the function of the TGSS will not be adversely affected by the proposed power uprate and therefore, an evaluation of the TGSS is not required.

Main Steam Isolation Valve Leakage Control System

Because MNGP does not have a main steam isolation valve leakage control system, this review section is not applicable.

2.5.3 Component Cooling and Decay Heat Removal

2.5.3.1 Fuel Pool Cooling and Cleanup System

Regulatory Evaluation

The SFP provides wet storage of spent fuel assemblies. The safety function of the fuel pool cooling and cleanup system (FPCCS) is to cool the spent fuel assemblies and keep the spent fuel assemblies covered with water during all storage conditions. The NRC staff's review of the FPCCS for the proposed EPU focused on the effects of the proposed uprate on the capability of the system to provide adequate cooling for the spent fuel during all operating and accident conditions. The criteria that are most applicable to the NRC staff's review of the FPCCS for proposed power uprates are based primarily on GDC-61, "Fuel Storage and Handling and Radioactivity Control," insofar as it specifies that fuel storage systems be designed to prevent significant reduction in fuel storage coolant inventory under accident conditions and with residual heat removal capability reflecting the importance to safety of decay heat removal.

The NRC staff's review of the FPCCS is performed in accordance with the guidance provided in Section 2.1 of RS-001, Matrix 5. Acceptability for EPU operation is judged based upon conformance with existing licensing-basis considerations as discussed primarily in Section 10.2.2, "Spent Fuel Pool Cooling and Demineralizer System," of the MNGP USAR, except where proposed changes are found to be acceptable based upon the specified review criteria.

While MNGP is not explicitly licensed to the current General Design Criteria or the 1967 AEC-proposed GDC, the licensee has made a comparison of the current GDC to the applicable AEC-proposed GDC. For the current GDC-61, the licensee's evaluation of the analogous 1967 AEC-proposed General Design Criteria is also contained in MNGP USAR, Appendix E: draft GDC-4, draft GDC-67, draft GDC-68, and draft GDC-69.

Technical Evaluation

The licensee evaluated the FPCCS in Section 2.5.3.1 of the PUSAR for MNGP. The components that are necessary for performing the cooling function include a skimmer surge tank, two half-capacity water pumps, two half-capacity fuel pool heat exchangers, two demineralizers, and associated piping, valves, and instrumentation. The system also has a cross-connection with the RHR system which allows the RHR system to provide SFP cooling.

Table 10.2-1, "Reactor Auxiliary Systems - Spent Fuel Pool Cooling and Demineralizer System Principal Design Parameters," of the MNGP USAR indicates that the SFP temperature would be maintained at pool temperatures less than or equal to 140°F. Consistent with the description in Section 10.2.2 of the MNGP USAR, the licensee describes in Section 2.5.3.1 of the PUSAR that administrative controls are used to ensure that the fuel pool temperature does not exceed 140°F during a normal batch or full core offload. The licensee states that the decay heat used in offload evaluations assumes use of GE14 fuel for 24-month fuel cycle and is calculated using the ANSI/ANS 5.1-1994 Standard with a one-sided 95 percent confidence. For fuel batches with equivalent power histories and decay times, the EPU decay heat loads are higher than the CLTP heat load.

Section 2.5.3.1 of the PUSAR describes that the EPU results in higher core decay heat loads during refueling and that these higher heat loads can be managed by various methods including extending the time after shutdown before discharging fuel to the SFP. Consistent with the current licensing basis, the licensee stated that appropriate administrative controls would be used to ensure the SFP temperature is maintained below the licensing limit of 140°F. By letter dated June 12, 2009, the licensee stated that cycle-specific calculations are procedurally controlled. The MNGP methods and assumptions used for decay heat calculations were previously described to the NRC staff by letter dated December 15, 2006 (Reference 83). The licensee stated that fuel pool heat load would be maintained within the heat removal capabilities of the fuel pool cooling and RHR systems using cycle-specific calculations and procedural controls, as described in the existing licensing basis.

Section 2.5.3.1 of the PUSAR describes that the limiting condition (emergency heat load) is a heat load value of 24.7 MBtu/hr and that this design basis limiting heat load does not change at EPU conditions. This heat load value is within the heat removal capacity of the RHR system in the fuel pool cooling mode at all credible service water supply temperatures. However, Section 10.2.2 of the MNGP USAR states that the emergency heat load condition assumes that a full core discharge that fills the last 484 spaces in the pool is required 30 days following the last refueling discharge and the full core discharge is complete 150 hours after shutdown. With those parameters unchanged, the heat load resulting from the emergency heat load case would increase. By letter dated June 12, 2009, as revised by letter dated December 21, 2010, the licensee restated that the limiting heat load would remain 24.7 MBtu/hr and clarified that the emergency heat load of 24.7 MBtu/hr would occur approximately 192 hours after shutdown from EPU conditions. The licensee committed to revise the USAR prior to implementation of the EPU to indicate the increase in the time after shutdown for the emergency heat load case.

In Section 2.5.3.1 of the PUSAR also describes that, in the unlikely event of a total loss of SFP cooling, the SFP would reach the boiling temperature in 6.5 hours under the limiting heat load conditions and assuming an initial pool temperature of 140°F. The corresponding makeup rate would be 53 gpm. These values are unchanged from the current licensing basis because the peak heat load and pool temperature limits imposed through administrative controls are unchanged. Available sources of water for makeup include the safety-related residual heat removal service water (RHRSW) system and several backup systems. In addition to the RHRSW system, two of the backup systems, the filter demineralizer backwash connection and the fire water system (hose station); provide makeup at rates exceeding the maximum boil-off rate calculated for EPU.

Based on a review of the information that was provided, the NRC staff is satisfied that the licensee has adequately evaluated and addressed the potential impact of the proposed EPU on the capability of the FPCCS (with the availability of RHR for backup heat removal) to adequately cool the spent fuel. The licensee will continue to administratively control the SFP heat load following EPU implementation within the heat removal capacity of the FPCCS at the maximum SFP temperature of 140°F, consistent with the current licensing basis. The time to boil following a loss of SFP cooling for the most limiting full core offload case will continue to afford plant operators sufficient time to take corrective actions, and available SFP makeup sources continue to have capacities exceeding the maximum calculated boil-off rate at EPU conditions. Therefore, the NRC staff concludes that the capability to remove decay heat from the SFP and to prevent a substantial reduction in SFP coolant inventory under accident conditions will be maintained in accordance with the plant licensing basis following EPU implementation.

Conclusion

The NRC staff has reviewed the licensee's assessment related to the impact that the proposed EPU will have on the FPCCS and finds that the FPCCS will continue to be capable of performing its cooling function, and that the SFP makeup capability will continue to be adequate in accordance with licensing basis considerations. Therefore, the NRC staff considers the proposed EPU to be acceptable with respect to SFP cooling and makeup capability.

2.5.3.2 Station Service Water System

Regulatory Evaluation

The station service water system (SWS) provides essential cooling for safety-related equipment and may also provide cooling for non-safety-related auxiliary components that are used for normal plant operation. The NRC staff's review of proposed power uprates focuses on the impact that the proposed EPU will have on the capability of the SWS to perform its safety functions. The criteria most applicable to the NRC staff's review of the SWS are based primarily on GDC-44, "Cooling water," insofar as it specifies that a system with the capability to transfer heat loads from safety-related SSCs to a heat sink under both normal-operating and accident conditions be provided, and other licensing-basis considerations that are applicable.

The NRC staff's review of the SWS is performed in accordance with the guidance provided in Section 2.1 of RS-001, Matrix 5, and acceptability for EPU operation is judged based upon conformance with existing licensing-basis considerations as discussed primarily in Section 10.4, "Plant Cooling System," of the MNGP USAR, except where proposed changes are found to be acceptable based upon the specified review criteria.

While MNGP is not explicitly licensed to the current General Design Criteria or the 1967 AEC-proposed GDC, the licensee has made a comparison of the current GDC to the applicable AEC-proposed GDC. There is no draft GDC directly associated with the current GDC-44. The licensee's evaluation of the analogous 1967 AEC-proposed General Design Criteria is contained in MNGP USAR, Appendix E: draft GDC-4, draft GDC-40, and draft GDC-42.

Technical Evaluation

The licensee's evaluation of the SWS is provided in Section 2.5.3.2 of the PUSAR. The MNGP design includes three open-loop cooling water systems. The plant service water (SW) system supplies water to the reactor and turbine buildings for cooling. The RHRSW system removes the heat rejected by the residual heat removal system during normal shutdown and accident operations. The emergency service water (ESW) system removes the heat rejected by the equipment that must operate under accident conditions.

The SW system cools various plant components that have heat loads that are either power dependent or unaffected by EPU. The SW system is not required during or immediately subsequent to a design-basis accident and, therefore, performs no safety-related functions. The heat loads on the non-safety-related SW system, which are power-dependent and increased by EPU, are: the main generator hydrogen coolers, the turbine lube oil coolers, the exciter air cooler, the isolated phase bus cooler, the condensate pump motor and bearing coolers, the reactor feed pump motors and lube oil coolers and the condensate pump area ventilation units. The capability of the non-safety-related SW system to provide adequate cooling to these components is affected by the EPU. Plant modifications (e.g. turbine replacement, generator rewind, feedwater pump motor replacement and condensate pump motor replacement) are required to implement EPU. Section 2.5.3.2.1 of the PUSAR describes that the licensee would evaluate changes to the service water flow necessary to cool these non-safety-related components in conjunction with these modifications, which the licensee stated would be implemented pursuant to the requirements of 10 CFR 50.59.

In Section 2.5.3.2.2 of the MNGP PUSAR, the licensee described that the post-LOCA containment and suppression pool responses have been calculated based on an energy balance between the post-LOCA heat loads and the existing heat removal capacity of the RHR and RHRSW systems. The licensee determined that the RHRSW system has sufficient capacity at EPU to supply adequate cooling to the RHR heat exchangers for post-accident and Appendix R suppression pool cooling, shutdown cooling, and supplemental SFP cooling.

The safety-related ESW system provides a reliable supply of cooling water for the following essential equipment: RHR and core spray (CS) pump motor coolers; RHR and CS room coolers; HPCI room cooler; EDGs; and the control room air conditioning system. Heat loads from the RHR and CS room coolers increase less than 3 percent at EPU conditions. The remaining heat loads are unchanged at EPU conditions. The licensee concluded that sufficient heat removal capacity would be available to accommodate the small increases in ESW system heat loads during operation at EPU design conditions.

The licensee determined that the MNGP program (i.e., scope, maintenance, and testing) to manage and monitor raw water cooling systems developed in response to Generic Letter (GL) 89-13, "Service Water System Problems Affecting Safety-Related Equipment" (Reference 36) is not affected by the proposed EPU.

The NRC staff found that the licensee adequately evaluated the impact of the proposed CPPU on the capability of the SW, RHRSW, and ESW systems to perform their safety functions. The proposed EPU has a minimal effect on the ESW system heat load, and the adequacy of the RHRSW system heat removal capacity would be established by the post-accident containment heat removal analysis, which is reviewed in Section 2.6 of this safety evaluation.

Conclusion

The NRC staff reviewed the licensee's assessment of the impact that the proposed CPPU will have on the SW, RHRSW, and ESW systems and found that the SWS will continue to be capable of performing their safety functions in accordance with licensing-basis considerations. Therefore, the NRC staff concludes that the proposed power uprate is considered to be acceptable with respect to open cycle cooling water systems.

2.5.3.3 Reactor Auxiliary Cooling Water Systems

Regulatory Evaluation

These systems include closed-loop auxiliary cooling water systems for reactor system components, reactor shutdown equipment, ventilation equipment, and components of the emergency core cooling system (ECCS). The MNGP design includes one closed loop cooling water system, the reactor building closed-loop cooling water (RBCCW) system, which is designed to remove heat from the reactor auxiliary systems. The NRC staff s review for proposed power uprates focuses on the continued capability of the RBCCW system to provide adequate cooling for critical plant equipment in accordance with its licensing basis. The criteria most applicable to the NRC staff's review of the RBCCW system are based on GDC-44, "Cooling water," insofar as it requires that a system with the capability to transfer heat loads from safety-related SSCs to a heat sink under both normal operating conditions and accident conditions be provided, and other licensing-basis considerations that are applicable.

The NRC staff's review of the reactor auxiliary cooling water systems is performed in accordance with the guidance provided in Section 2.1 of RS-001, Matrix 5, and acceptability for EPU operation is judged based upon conformance with existing licensing-basis considerations as discussed primarily in Section 10.4.3, "Reactor Building Closed Cooling Water System," of the MNGP USAR, except where proposed changes are found to be acceptable based upon the specified review criteria.

While MNGP is not explicitly licensed to the current General Design Criteria or the 1967 AEC-proposed GDC, the licensee has made a comparison of the current GDC to the applicable AEC-proposed General Design Criteria. There is no draft GDC directly associated with the current GDC-44. The licensee's evaluation of the analogous 1967 AEC-proposed General Design

Criteria is contained in MNGP USAR, Appendix E: draft GDC-4, draft GDC-40, and draft GDC-42.

Technical Evaluation

The licensee's evaluation of the RBCCW system is provided in 2.5.3.3.1, "Reactor Building Closed Cooling Water System," of the MNGP PUSAR. The licensee determined that the heat loads on the non-safety-related RBCCW system increase an insignificant amount (< 0.1 percent) as a result of EPU. The RBCCW heat loads are mainly dependent on the reactor vessel temperature and the conditions (temperature and flow rates) in the systems cooled by RBCCW. The largest heat loads on RBCCW are the drywell atmosphere coolers, RWCU non-regenerative heat exchangers, and the SFP heat exchangers. The drywell atmosphere cooler heat load increases an insignificant amount when compared to the RBCCW system total heat load. The design heat loads of the RWCU non-regenerative heat exchangers and SFP heat exchangers are not changed by EPU because the fluid conditions assumed to establish the heat removal rate from these systems are unchanged.

The licensee addressed the issues identified in GL 96-06, "Assurance of Equipment Operability and Containment Integrity during Design-Basis Accident Conditions," through a combination of analysis, procedural changes, administrative controls, and modifications. The licensee reviewed the actions identified in the responses to GL 96-06 for continued applicability at EPU conditions and determined that they are unaffected by EPU. Therefore, the NRC staff accepts that the licensee's response to Generic Letter 96-06 remains valid for EPU.
The NRC staff is satisfied that the licensee has adequately evaluated and addressed the impact of CPPU on the capability of the RBCCW system to perform its specified safety functions. The licensee has confirmed that the capability of the RBCCW system to accommodate the specified heat loads in accordance with the plant licensing basis will not be affected by the proposed power uprate. Also, the licensee's resolution of the GL 96-06 issues would not be affected by the proposed CPPU.

Conclusion

The NRC staff reviewed the licensee's assessment of the effects of the proposed CPPU on the RBCCW system and concluded that the licensee has adequately accounted for the increased heat loads from the proposed CPPU on system performance. Therefore, the NRC staff finds the proposed CPPU acceptable with respect to the reactor auxiliary cooling water systems.

2.5.3.4 Ultimate Heat Sink

The ultimate heat sink (UHS) provides the cooling medium for dissipating the heat removed from the reactor and its auxiliaries during normal operation, refueling, transient, and accident conditions. The Mississippi River serves as the ultimate heat sink for MNGP and provides an essentially unlimited supply of cooling water. The ultimate heat sink temperature limit is described in MNGP USAR Section 5.2.3.2.4, "Licensing Basis Ultimate Heat Sink Limit." The UHS temperature and level considerations relative to CPPU operation are evaluated primarily in Sections 2.5.3.1, 2.5.3.2, and 2.5.3.3. Therefore, the NRC staff concludes that the UHS is unaffected by the proposed power uprate.

2.5.4 Balance-of-Plant Systems

Main Steam

The main steam supply system (MSSS) transports steam from the NSSS to the power conversion system and to various auxiliary steam loads. The NRC staff's review of the MSSS for proposed power uprates focuses primarily on any changes in the design or operation of the MSSS that could impact the capability of steam-driven equipment to function in accordance with safe shutdown and accident analysis assumptions, impact the capacity of the steam dump system, or could otherwise result in increased off-site releases or challenges to reactor safety systems. Therefore, the NRC staff concludes that an evaluation of the MSSS is not required since no changes of this nature are being made.

Main Condenser

The main condenser system (MCS) is designed to condense and deaerate the exhaust steam from the main turbine and provide a heat sink for the turbine steam bypass system, and is typically credited for providing sufficient condensate retention time to allow short-lived radioactive isotopes to decay. For BWRs without an MSIV leakage control system, the MCS may also be credited for providing holdup and plate-out of radioactive iodine through the MSIV bypass leakage pathway following core damage. The NRC staff's review of the MCS for proposed power uprates focuses primarily on the capability of the main condenser system to accommodate the steam bypass flow rates and on any changes that are being made to the MSIV bypass leakage pathway to confirm that the isolation boundary has been properly established. Because the proposed EPU will not affect the steam bypass flow rate and MSIV bypass leakage pathway boundaries, this area of review is not affected by the proposed EPU. The main condenser will be operated with a higher hotwell level to support the NPSH requirements for the new condensate pumps. This will reduce the condenser free volume, which is addressed in Section 2.9.2 of this SE. Therefore, the NRC staff concludes that an evaluation of the MCS is not required.

Turbine Bypass

The turbine bypass system (TBS) is a non-safety-related system designed to discharge a stated percentage of rated main steam flow directly to the main condenser, bypassing the turbine and enabling the plant to take step-load reductions up to the capacity of the TBS without causing the reactor or turbine to trip. The NRC staff's review of the TBS for proposed power uprates focuses primarily on any modifications that are being made to the TBS that may warrant the performance of confirmatory testing. The licensee did not propose to credit additional steam bypass capacity beyond what was previously assumed, and no modifications are being made to the steam bypass system for EPU operation. The nominal turbine bypass flow rate at EPU operating conditions will remain 0.97 Mlb/hr. Therefore, the NRC staff concludes that an evaluation of the TBS is not required since no changes were proposed in the design and operation of the TBS for EPU operation.

Condensate and Feedwater

Regulatory Evaluation

The condensate and feedwater system (CFS) provides feedwater at a particular temperature, pressure, and flow rate to the reactor. The scope of review in this section includes the part of the CFS that is outside containment beyond the outermost containment isolation valves. The NRC's acceptance criteria for the condensate and feedwater system are based on GDC-4, "Environmental and dynamic effects design bases," insofar as it requires that SSCs important to safety be protected against dynamic effects such as possible fluid flow instabilities (e.g., water hammer) that develop during normal operation or upset conditions, and GDC-44, "Cooling water," insofar as it requires provision of a system with the capability to transfer heat loads from safety-related SSCs to a heat sink under both normal operating and accident conditions. Inadequate system design and operational capability could result in loss of the capability to transfer heat from safety-related SSCs to a heat sink under normal operating conditions and increased challenges to safety systems.

The NRC staff's review of the CFS for proposed power uprates focuses primarily on system modifications, design limitations, and reductions in operational flexibility that could result in less reliable CFS operation. The acceptance criteria that are most applicable to the NRC staff's review of the CFS for proposed power uprates are based on existing plant licensing-basis considerations, especially with respect to maintaining CFS reliability and minimizing loss of FW event occurrences. The NRC staff's review of the CFS is performed in accordance with the guidance provided in Section 2.1 of RS-001, Matrix 5, and acceptability for EPU operation is judged based upon conformance with existing licensing-basis considerations as discussed primarily in MNGP USAR Section 11.8, "Condensate and Reactor Feedwater Systems," except where proposed changes are found to be acceptable based upon the specified review criteria. The condensate demineralizer system is described in MNGP USAR Section 11.7, "Condensate Demineralizer System."

Technical Evaluation

The licensee's evaluation of the CFS for EPU operation is provided in Section 2.5.4.4 of the PUSAR. The CFS is designed to provide a reliable supply of feedwater at the temperature, pressure, quality, and flow rate as required by the reactor. The licensee indicated that the CFS does not perform a system level safety-related function. However, the system provides the normal source of water for reactor heat removal when operating at EPU conditions and following certain transients. For EPU, the feedwater and condensate systems will meet their performance requirements with modifications to the following non-safety-related equipment for increased capacity:

- Feedwater pumps and motors
- Condensate pumps and motors
- Moisture separator drain tank discharge piping (improve sub-cooling to reduce two-phase flow)

The licensee indicated that to account for feedwater transients, the modified feedwater and condensate pumps will provide a minimum of 5 percent margin above the EPU-rated feedwater flow. This is consistent with the design at the CLTP, thus the capability to supply the transient flow requirements is not decreased. For system operation with all system pumps available, the predicted operating parameters are acceptable and within the component capabilities.

The NRC staff reviewed the proposed modifications and found that, since the proposed modifications maintain the existing margin above full power feedwater flow and industry operating experience with these modifications has been good, there is reasonable assurance of adequate feedwater performance under normal operating conditions. However, the NRC staff found that analysis and testing of system transient response is not necessary to ensure the system response to transient conditions is consistent with performance capabilities described in the licensing basis. Section 2.12 of this safety evaluation provides details of this review.

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the CFS and finds that the CFS will remain capable of satisfying the reactor feedwater demands for normal operation at EPU conditions. The proposed power ascension testing and transient response of the system are necessary to provide reasonable assurance that the CFS would respond to transients at EPU operating conditions consistent with licensing-basis considerations.

2.5.5 Waste Management Systems

2.5.5.1 Gaseous Waste Management Systems

Regulatory Evaluation

The gaseous waste management system (GWMS) includes those systems that process potential sources of airborne releases of radioactive gases during normal operation and anticipated operational occurrences (AOOs). These systems typically include the off-gas system, the condenser air removal system, the gland seal exhaust, and building ventilation system exhausts. The NRC staff's review of the GWMS focuses on the effects that the proposed EPU may have on: (1) the design criteria of the gaseous waste management systems; (2) methods of treatment; (3) expected releases; (4) principal parameters used in calculating the releases of radioactive materials in gaseous effluents; and (5) design features for precluding the possibility of an explosion if the potential for explosive mixtures exists. The criteria that are most applicable to the NRC staff's review of the GWMS for proposed power uprates are based on: (1) 10 CFR 20.1302, "Compliance with dose limits for individual members of the public," insofar as it provides for demonstrating that annual average concentrations of radioactive materials released at the boundary of the unrestricted area do not exceed specified values; (2) GDC-3, "Fire protection," insofar as it requires that (a) SSCs important to safety be designed and located to minimize the probability and effect of fires, (b) noncombustible and heat-resistant materials be used, and (c) fire detection and fighting systems be provided and designed to minimize the adverse effects of fires on SSCs important to safety; (3) GDC-60, "Control of releases of radioactive materials to the environment," insofar as it requires that the plant design include means to control the release of radioactive effluents; (4) GDC-61, "Fuel storage and handling and radioactive control," insofar as it requires that systems that contain radioactivity be designed with appropriate confinement; and (5) 10 CFR Part 50, Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion "As Low as is Reasonably Achievable" for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents," Sections II.B, II.C, and II.D, which set numerical guides for design objectives and limiting conditions for operation to meet the "as low as is reasonably achievable" (ALARA) criterion.

The NRC staff's review of the GWMS is performed in accordance with the guidance in Section 2.1 of RS-001, Matrix 5; and acceptability for EPU operation is judged based upon conformance with existing licensing basis considerations as discussed primarily in MNGP USAR Section 9.3, "Gaseous Radwaste System," except where proposed changes are found to be acceptable based upon the specified review criteria.

While MNGP is not explicitly licensed to the current General Design Criteria or the 1967 AEC-proposed GDC, the licensee has made a comparison of the current GDC to the applicable AEC-proposed General Design Criteria. For the current GDC-3, GDC-60 and GDC-61, the licensee's evaluation of the analogous 1967 AEC-proposed General Design Criteria is also contained in MNGP USAR, Appendix E: draft GDC-3, draft GDC-67, draft GDC-68, draft GDC-69, and draft GDC-70.

Technical Evaluation

As discussed in Section 2.5.5.1 of the MNGP PUSAR, the licensee evaluated the impact of the proposed EPU on the capability of the GWMS to perform its functions and determined that sufficient capacity exists without modification to process the increase in gaseous waste that will result from EPU operation. The licensee stated that the radiological release rate is administratively controlled to remain within existing site release rate limits, and is a function of fuel cladding performance, main condenser air inleakage, and compressed gas storage tank volume. The administrative controls include power reduction or shutdown, reducing main condenser air inleakage, and local power suppression (inserting control rods near a leaking fuel assembly). The MNGP TSs and administrative controls require that the licensee limit fission gas releases to the environment. Plant procedures for reducing power, identifying and suppressing power near leaking fuel, and repairing condenser air inleakage exist and have been used at MNGP to maintain the offgas limits. These procedures are not affected by EPU.

The NRC staff accepted a CLTR design value of 0.0677 cubic feet per minute per megawatts thermal (cfm/MWth) (at 130°F and 1 atmosphere) for the rate of radiolytic gas production. The CLTR design value represents a margin of more than 50 percent over the 0.0450 cfm/MWth actual radiolysis rate determined for MNGP. Thus, the recombiner and main condenser, as well as downstream system components, are designed to handle an average increase in thermal power of more than 50 percent relative to OLTP, without exceeding the design basis temperatures, flow rates, or heat loads.

The NRC staff found that the licensee has adequately evaluated the impact of the proposed EPU on the capability of the GWMS to perform its functions. Because the radiolytic gas flow rates and concentrations will remain within the design capability of the GWMS and radiological release rates will continue to be administratively controlled during EPU operation, the NRC staff agrees that the GWMS will continue to satisfy the plant licensing basis following implementation of EPU.

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the capability of the GWMS to perform its functions, and finds that the GWMS will continue to control the release of radioactive materials and preclude the possibility of waste gas explosions in accordance with licensing basis considerations. Therefore, the NRC staff concludes that the proposed power uprate is considered to be acceptable with respect to the GWMS.

2.5.5.2 Liquid Waste Management Systems

Regulatory Evaluation

The liquid waste management system (LWMS) is designed to collect, store, process, and dispose of or recycle all radioactive or potentially radioactive liquid waste generated by plant operation or maintenance. Major components include floor and equipment drains, transfer pumps, and various waste system tanks. The NRC staff's review of the LWMS focuses on the effects that the proposed EPU may have on previous analyses and considerations used in estimating the increase in volume of the liquid radioactive waste. The criteria that are most applicable to the NRC staff's review of the LWMS are based on: (1) 10 CFR 20.1302, "Compliance with dose limits for individual members of the public," insofar as it places specific limitations on the annual average concentrations of radioactive materials released at the boundary of the unrestricted area; (2) 10 CFR Part 50, Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion "As Low as is Reasonably Achievable" for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents," Sections II.A and II.D, which set numerical guides for dose design objectives and limiting conditions for operation to meet the ALARA criteria; and (3) other licensing basis considerations that are applicable.

The NRC staff's review of the LWMS is performed in accordance with the guidance in Section 2.1 of RS-001, Matrix 5, and acceptability for EPU operation is judged based upon conformance with existing licensing-basis considerations as discussed primarily in Section 9.2, "Liquid Radwaste System," of the MNGP UFSAR, except where proposed changes are found to be acceptable based upon the specified review criteria.

While MNGP is not explicitly licensed to the current General Design Criteria or the 1967 AEC-proposed GDC, the licensee has made a comparison of the current GDC to the applicable AEC-proposed GDC. The licensee's evaluation of the analogous 1967 AEC-proposed General Design Criteria is also contained in MNGP USAR, Appendix E: draft GDC-3, draft GDC-67, draft GDC-68, draft GDC-69, and draft GDC-70.

Technical Evaluation

As discussed in Section 2.5.5.2 of the PUSAR, the licensee determined that the largest EPU effect on the LWMS is the increase in liquid and wet solid waste that will result from more frequent backwashing of the condensate demineralizers. More frequent demineralizer backwashing will be necessary due to the increased condensate flow that will be required for EPU operation. Similarly, the RWCU filter-demineralizer requires more frequent backwashes due to higher levels of impurities as a result of the increased feedwater flow. The licensee estimated that the resultant increase in liquid radiological waste is insignificant when compared to the LWMS capacity. Since the design and operation of the LWMS will not change and the volume of fluid flowing into the liquid radwaste system will not increase significantly as a result of EPU operation, the licensee concluded that the capacity of the LWMS will continue to be adequate.

The NRC staff found that the licensee has adequately evaluated and addressed the impact of the proposed EPU on the capability of the LWMS to perform its functions. Because the increase in additional radioactive waste being generated due to EPU operation is expected to be minimal and well within the capacity of the LWMS, any increase in offsite dose projections as a consequence is expected to be inconsequential and remain well below established plant release limits.

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the capability of the LWMS to perform its functions and found that the LWMS will continue to control the release of liquid radioactive materials in accordance with licensing basis considerations. Therefore, the NRC staff concludes that the proposed EPU is considered to be acceptable with respect to the LWMS.

2.5.5.3 Solid Waste Management Systems

Solid radioactive waste consists of wet and dry waste. Wet waste consists mostly of low specific activity spent secondary and primary resins and filters, and oil and sludge from various contaminated systems. The NRC staff's review relates primarily to the wet waste dewatering and liquid collection processes, and focuses on the impact that the proposed EPU will have on the release of radioactive material to the environment via gaseous and liquid effluents. The NRC staff concludes that because Sections 2.5.5.1 and 2.5.5.2 above had fully encompassed these considerations, a separate evaluation of solid waste management systems is not required.

2.5.6 Additional Considerations

2.5.6.1 Emergency Diesel Engine Fuel Oil Storage and Transfer System

Nuclear power plants are required to have redundant onsite emergency power supplies of sufficient capacity to perform their safety functions (e.g., diesel engine-driven generator sets). The NRC staff's review of the emergency diesel fuel oil storage and transfer system for proposed power uprates focuses on the effects that the proposed power uprate may have on the fuel oil storage requirements for the EDGs. In Section 2.5.6.1 of the PUSAR, the licensee indicated that the electrical rating and loading of the EDGs are not altered by the proposed EPU and, consequently, the fuel oil consumption rate and fuel oil storage requirements are not affected. Therefore, the NRC staff concludes that an evaluation of the EDG fuel oil storage requirements for the proposed EPU is not required.

2.5.6.2 Light Load Handling System (Related to Refueling)

The light load handling system (LLHS) includes components and equipment used for handling new fuel at the receiving station and for loading spent fuel into shipping casks. The licensee is not introducing a new fuel design in conjunction with the proposed EPU, and as indicated in Table 2.5-3 of the PUSAR, cranes, hoists, and fuel handling systems are not affected by the proposed power uprate. Because this area of review is not affected by the proposed EPU, an evaluation of the LLHS is not required.

2.6 Containment Review Considerations

2.6.1 Primary Containment Functional Design

Regulatory Evaluation

The containment encloses the reactor system and is the final barrier against the release of significant amounts of radioactive fission products in the event of an accident. The NRC staff's review of the primary containment functional design covered: (1) the temperature and pressure conditions in the drywell and wetwell that would result from a spectrum of postulated LOCAs; (2) suppression pool dynamic effects during a LOCA or following the actuation of one or more Reactor Coolant System (RCS) Safety Relief Valves (SRVs); (3) the consequences of a LOCA occurring within the containment; (4) the capability of the containment to withstand the effects of steam bypassing the suppression pool; (5) the suppression pool temperature limit during RCS SRV operation; and (6) the analytical models used for containment analysis. The NRC staff acceptance criteria for the primary containment functional design are based on the following GDCs in 10 CFR 50, Appendix A:

GDC-4, "Environmental and dynamic effects design bases," insofar as it requires that Structure, System, and Components (SSCs) important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents and that such SSCs be protected against dynamic effects.

GDC-13, "Instrumentation and Control," insofar as it requires that instrumentation be provided to monitor variables and systems over their anticipated ranges for normal operation and for accident conditions, as appropriate, to assure adequate safety.

GDC-16, "Containment design," insofar as it requires that the containment and associated systems be designed to establish an essentially leak tight barrier against the uncontrolled release of radioactivity to the environment, and to assure that the containment design conditions important to safety are not exceeded as long as postulated accident conditions require.

GDC-50, "Containment Design Basis," insofar as it requires that the containment and its associated heat removal systems be designed so that the containment structure can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated temperature and pressure conditions resulting from any LOCA.

GDC-64, "Monitoring Radioactivity Releases," insofar as it requires that means be provided to monitor the reactor containment atmosphere for radioactivity that may be released from normal operations and from postulated accidents.

NUREG-0800, Section 6.2.1.1.C contains specific review criteria.

The GDCs discussed above are those currently specified in Appendix A of 10 CFR Part 50. The applicable MNGP principal design criteria predate these Appendix A criteria. These MNGP principal design criteria are listed in Section 1.2, "Principal Design Criteria," of the MNGP Updated Safety Analysis Report (USAR). In 1967, the AEC published for public comment a revised set of proposed GDCs (32 *FR* 10213, July 11, 1967). An evaluation comparing the MNGP design basis to the AEC proposed GDC of 1967 is presented in Appendix E, "Plant

Comparative Evaluation with the Proposed AEC 70 Design Criteria," of the MNGP USAR.

While MNGP is not explicitly licensed to the current General Design Criteria or the 1967 AEC proposed GDC, the licensee stated that current GDCs 4, 16, 50, and 64 are comparable to the applicable AEC proposed GDCs 10, 12, 13, 17, 40, 42, and 49. Regarding the current GDC-13, the licensee stated that it is applicable as described in the USAR, Section 14.7.4.

Technical Evaluation

The MNGP is a BWR-3 with a Mark I pressure suppression type primary containment. As described in Section 5.1 of the USAR (Revision 24), the primary containment encloses the RV, the reactor coolant recirculation loops, and other branch connections of the RCS. The major elements of the primary containment are the drywell, the pressure suppression chamber (or wetwell) that stores a large volume of water (suppression pool), the connecting vent pipe system between the drywell and the wetwell, isolation valves, the vacuum relief system, the containment cooling systems and other service equipment.

The drywell is a steel pressure vessel with a spherical lower portion and a cylindrical upper portion. The wetwell is a toroid-shaped steel pressure vessel located below and encircling the drywell. The drywell-to-wetwell vents are connected to a vent header contained within the airspace of the wetwell. Downcomer pipes project downwards from the vent header and terminate below the water surface of the suppression pool so that in the event of any pipe failure in the drywell, the released steam would pass directly to the water where it would be condensed. The vacuum relief system consists of eight vacuum breakers which equalize the pressure between the wetwell and the drywell to prevent a backflow of water from the suppression pool into the vent system.

The proposal to operate at EPU conditions requires that safety analyses for those Design-Basis Accidents (DBAs) whose results depend on power level be recalculated at the higher power level. The containment design basis is primarily established based on the LOCA and the actuation of the RV SRVs and their discharge into the suppression pool. The RV steam dome pressure remains constant at its pre-EPU value and, therefore, the EPU is regarded as a Constant Pressure Power Uprate (CPPU).

The MNGP USAR documents the current results of short-term and long-term containment analyses. The short-term analysis is directed primarily at determining the drywell pressure and gas temperature response during the initial blowdown of the RV inventory to the containment following a design basis LOCA. The long-term analysis is directed primarily at the suppression pool temperature response, considering the decay heat addition to the suppression pool. The effect of power uprate on the events yielding the limiting containment pressure and temperature responses are described below.

The licensee performed the EPU analysis in accordance with guidelines in Reference 25 using General Electric-Hitachi (GEH) computer codes (References 32, 33, and 34). For the plant specific use of GEH Super Hex (SHEX) code for MNGP, the licensee performed a confirmatory calculation of the peak suppression pool temperature using the NRC-accepted HXSIZ code and found a difference of less than 1°F. By performing this calculation, the licensee confirmed compliance with NRC requirement in Section 2.6a of Reference 25 for the use of SHEX for calculating the suppression pool response under EPU conditions.

Short-Term LOCA Analysis for Drywell Pressure Response

The short-term analysis for drywell pressure response covers the blowdown period during which the maximum drywell pressure and differential pressure between drywell and wetwell occur. The peak drywell pressure occurs within the first 100 seconds of the blowdown period. The short-term DBA LOCA analysis which assumes a large double-ended guillotine break of a recirculation suction line resulted in maximum drywell pressure. This analysis used analytic methods approved for EPUs. The decay heat model used is ANSI/ANS 5.1-1971 plus 20-percent which is the same as used in the current USAR analysis. The licensee used the LAMB computer code (Reference 32) for the short-term mass and energy release and the M3CPT computer code (Reference 33) for the containment response. The power uprate methods approved by the NRC permit the use of either the M3CPT computer code or the LAMB computer code to calculate the mass and energy release from the postulated pipe break into the drywell (Reference 8) and M3CPT for the short-term containment response.

The licensee made the following key conservative assumptions in performing the analysis: (a) The reactor is assumed to be operating at two percent above the EPU Reactor Thermal Power (RTP) to include instrument uncertainty effects; (b) The initial value of containment pressure is assumed to be 3 psig compared to 2 psig used in the current analysis; (c) The current and proposed initial value of drywell relative humidity is assumed to be at its minimum value of 20-percent; (d) The suppression pool level and mass are at values corresponding to the maximum TS limit; (e) The vessel depressurization flow rates are calculated using the Moody critical flow model (Reference 35) which maximizes the mass flow into the drywell, which is conservative compared to more realistic prediction methods such as the Homogeneous Equilibrium Model (HEM) (Reference 44); (f) The fluid flowing through the drywell-to-wetwell vents is assumed to be a homogenous mixture of the fluid in the drywell, which means the flow contains liquid droplets thereby increasing the pressure drop of the flow through the vents and consequently maximizing the drywell pressure; and (g) The proposed and the current USAR analyses assume that there is no heat loss from the gases inside the primary containment. In reality, condensation of steam on the drywell surfaces would be expected. Neglecting this heat transfer is conservative for the short-term analyses.

The NRC staff requested in an RAI that the licensee confirm if the assumption for feedwater (FW) coastdown time for the short-term LOCA analysis was the same as in current analysis or more conservative. In its response to RAI-2 (Reference 47), the licensee stated that the EPU short-term LOCA analysis with LAMB blowdown model for peak drywell pressure analysis assumes a 5-second coastdown, whereas the current analysis assumes a 4-second coastdown. The NRC staff considers the licensee's response acceptable because the assumption of a greater FW coastdown time for short-term LOCA analysis is more conservative.

The NRC staff requested in an RAI that the licensee confirm if the assumption for MSIV closure time for the short-term LOCA analysis was the same as in current analysis or more conservative. In its response to RAI-3 (Reference 47), the licensee stated that the MSIV closure time used in the EPU short-term LOCA analysis is more conservative than the closure time used in the current short-term LOCA analysis. The EPU short-term LOCA analysis for peak drywell pressure uses the LAMB vessel model for calculating break flow assumes a MSIV closure time of 3 seconds, whereas the current short-term analysis assumes a MSIV closure time of 5 seconds. The licensee further stated that the faster closure time results in maintaining RV pressure higher longer, with a resulting higher break flow calculation, which results in a more conservative estimate for peak drywell pressure. The NRC staff agrees with the licensee response.

Table 2.6-1 of the Reference 7 presents the results of the short-term LOCA analyses at EPU and its design limits. The short-term portion of this table is reproduced in Table 1 below:

Table 1: EPU Short-Term LOCA Containment Pressure Response Results

Parameter	At CLTP from USAR	EPU Analysis	Design Limit
Peak drywell pressure (psig)	39.5	44.1	56

Containment leakage rate testing pressure P_a is the pressure at which containment leakage rate testing is performed as per 10 CFR Part 50 Appendix J. It is defined in 10 CFR Part 50 Appendix J as the calculated peak containment internal pressure related to the design-basis LOCA. In MNGP TS 5.5.11, "Primary Containment Leakage Rate Testing Program," the licensee proposed to revise P_a to 44.1 psig. The NRC staff finds this acceptable since P_a, the calculated peak containment internal pressure related to the design-basis LOCA for the EPU, is determined with acceptable methods and assumptions.

Based on the use of acceptable calculation methods and conservative assumptions and results less than the design containment pressure, the NRC staff finds the licensee's short-term drywell pressure response results at EPU to be acceptable.

Long-Term Small Steam Line Break Containment Analysis for Environmental Qualification

The licensee stated that the drywell gas temperature analysis for environmental qualification uses the same methodology as was used in the current analysis for environmental qualification. The GEH SHEX computer code is used to evaluate containment response to the various assumed steam line breaks. The licensee used assumptions which maximized the drywell gas temperature response. The maximum drywell gas temperature is reached during the short term blowdown from a small steam line break, of area 0.5 square feet, in which reactor steam at EPU conditions expands adiabatically (at constant enthalpy) to a superheated state at a drywell pressure of 35 psig. The licensee calculated the EPU maximum drywell gas temperature of 338°F compared to the current value of 335°F reported in USAR Revision 24, Table 5.2-8. The licensee, therefore, proposed to revise its design limit from its current value of 335°F to 338°F, which is the same as the analytical result of 338°F. The increase in drywell gas temperature design limit affects the environmental qualification of equipment which is discussed in Section 2.3.1 of Reference 7, and is being reviewed by the Electrical Engineering Branch (EEEB). Based on the above analysis, with a drywell gas temperature of 338°F, the licensee calculated the maximum drywell wall temperature of 278°F using 1xUCHIDA condensation heat transfer correlation. The current structural drywell design temperature is 281°F which bounds the maximum drywell wall temperature under EPU conditions.

Table 2.6-1 of Reference 7, as revised in Reference 65, presents the results of the long-term small steam line break analysis at EPU and its design limits. The drywell peak temperature results of this table are reproduced in Table 2 below:

Table 2: EPU Long-Term Small Steam Line Break Drywell Temperature Response for Equipment Qualification

Parameter	At CLTP from USAR	EPU Analysis	Design Limit
Peak drywell gas temperature (°F)	335	338	338
Peak drywell wall temperature (°F)	273	278	281

Based on the use of acceptable calculation methods and conservative assumptions and results at or below the new design limit, the NRC staff finds the licensee's long-term small steam line break containment analysis EPU results to be acceptable.

Long-Term LOCA Analysis for Suppression Pool Temperature Response

The long-term analysis for suppression pool temperature covers the transient period during which the maximum suppression pool temperature occurs. The long-term DBA LOCA analysis which assumes a large double-ended guillotine break of a recirculation suction line resulted in maximum suppression pool temperature. The licensee used ANSI/ANS 5.1-1979 decay heat model with 2 standard deviation (2σ) uncertainty added which is same as in current USAR analysis. In its response to a RAI-(Reference 20, RAI-1), the licensee stated that the decay heat under EPU conditions incorporated the guidance of GE Service Information Letter (SIL) 636, Revision 1, which recommends accounting for additional actinides and activation products, which further increases the predicted decay heat. This analysis was performed using analytic methods approved for EPU. The SHEX computer code (Reference 34) is used for the analysis of the peak suppression pool temperature.

The analysis makes several conservative assumptions. The reactor is assumed to be operating at 2-percent above the EPU RTP to include instrument uncertainty effects, consistent with RG 1.49. The initial value of containment pressure is assumed to be at its maximum at 3 psig compared to 2 psig used in the current analysis. The current and the proposed initial value of drywell relative humidity are assumed to be at its minimum at 20-percent which is conservative. The RV depressurization flow rates are calculated using the homogeneous equilibrium break flow model which is conservative for suppression pool temperature and wetwell pressure response. The suppression pool level and mass are at values corresponding to the minimum TS limit which is conservative. The analysis resulted in a peak suppression pool temperature of 203°F which is greater than the current design temperature of 196.7°F (Reference 37) for wetwell attached piping. Therefore, for EPU the licensee revised the design temperature of wetwell attached piping from 196.7°F to 212°F. The wetwell attached piping design temperature change is discussed in Section 2.2 of Reference 7, as revised in Reference 65, and is being reviewed by the Mechanical and Civil Engineering Branch. The proposed and the current USAR analyses assume that there is no heat loss from the gases inside the primary containment. In reality, condensation of steam on the drywell surfaces would be expected. Neglecting this heat transfer is conservative for the long-term analyses.

The evaluation of the long-term containment DBA LOCA response reflects four changes from the MNGP current licensing basis analysis in USAR. These changes are as follows: (1) the steel structures in the drywell, wetwell air space, and the suppression pool are modeled and credited as passive heat sinks; (2) thermal equilibrium is assumed between the suppression pool and the wetwell airspace for the first 30 seconds, and subsequently the heat and mass transfer between the suppression pool and wetwell airspace is mechanistically modeled; (3) the RHR heat exchanger performance factor (or K-value) is assumed to vary with its hot side inlet temperature; and (4) the service water temperature to the RHR heat exchanger is 90°F instead of 94°F used in the current analysis. The changes are evaluated below.

The licensee currently credits the suppression pool as the only passive heat sink available in the containment system. For the EPU, the licensee proposes to additionally take credit of heat transfer from the containment atmosphere to steel structures in the drywell, wetwell air space, and the suppression pool and model these as passive heat sinks. The NRC staff has reviewed the licensee's approach and finds it acceptable because these structures already exist and therefore represent a more realistic configuration compared to configuration in the current analysis.

The licensee currently assumes thermal equilibrium between the wetwell airspace and the suppression pool during the entire transient period. In the EPU analysis the licensee assumed thermal equilibrium for the first 30 seconds and subsequently mechanistically modeled heat and mass transfer between the wetwell airspace and the suppression pool. The NRC staff requested in an RAI-that the licensee provide justification for this assumption and reasons for it being more conservative than in the current licensing basis given in USAR Revision 24, Table 5.2-7, item number 6. In response to RAI-8 (Reference 47), the licensee stated that during the early blowdown period of a DBA LOCA event, agitation of the suppression pool surface due to pool swell and later due to steam condensation enhances mixing between the wetwell airspace and the suppression pool, which results in significant heat and mass transfer between the pool and the airspace such that thermal equilibrium adequately models the wetwell airspace and suppression pool response during this period of the event. To model the higher expected mixing that occurs during this early blowdown period, it is assumed that, for the first 30 seconds, thermodynamic equilibrium conditions exist in the wetwell. After 30 seconds, it is assumed that pool surface agitation is reduced, resulting in reduced heat and mass transfer, and a mechanistic model for heat and mass transfer is used instead. The assumption of mechanistic heat and mass transfer results in less heat transfer and less mass transfer (by evaporation) from the suppression pool to the wetwell airspace. This results in a slight increase in the energy retained in the suppression pool, and consequently a (slightly) higher pool temperature. However, the effect of this modeling assumption on the suppression pool temperature is small due to the relatively small energy transferred to the wetwell airspace gas with either modeling assumption. The NRC staff has reviewed the licensee's approach and finds it acceptable because the assumption of mixing and thermal equilibrium between the suppression pool and the wetwell gas space during the first 30 seconds is realistic and the subsequent mechanistic modeling still provides a conservative long term suppression pool temperature response.

An important parameter characterizing the performance of the suppression pool is the K-value of the RHR heat exchanger. The licensee currently uses a K-value of 147 British thermal units per second-degrees Fahrenheit (Btu/s-°F) for each RHR heat exchanger and assumes it to be constant throughout the event. In the EPU analysis, the licensee used a variable K-value for the RHR heat exchanger which is a change from the current analysis. In the EPU analysis, the K-value increases linearly with the hot side inlet (suppression pool) temperature from 146.5 Btu/s-°F at 110°F to 151.6 Btu/s-°F at 195°F, and has constant values of 146.5 Btu/s-°F below 110°F, and 151.6 Btu/s-°F above 195°F. The licensee stated that the K-values have been conservatively derived using vendor design assumptions including fouling factors. Confirmation of the ability of the RHR heat exchangers to support the K-values used is verified by performance of a heat exchanger efficiency test. In a RAI, the licensee was requested to verify if the testing is performed as per NRC GL 89-13 (Reference 36). In its response to RAI-9 (Reference 47), the licensee stated that the RHR heat exchangers are periodically tested according to the recommendations of GL 89-13 to ensure that the heat exchangers meet or exceed this K-value. The NRC staff considers the use of variable K-value for the RHR heat exchanger acceptable because the licensee conservatively derived these values using vendor design assumptions, and further confirming the ability of the RHR heat exchanger to support the K-values by performing heat exchanger test as recommended in GL 89-13.

The licensee currently uses 94°F as service water temperature inlet to RHR heat exchanger. In the EPU analysis, the service water temperature to RHR heat exchanger is 90°F. As per PUSAR Section 5.2.3.2.4, a service water temperature of 90°F is an accepted licensing basis. The service water temperature current licensing basis value is 90°F (Reference 37). The NRC staff finds the service water temperature of 90° F acceptable for long term containment analysis.

In the EPU analysis, the licensee assumes one CS loop (one pump per loop) operates continuously during the accident. The licensee also assumes one RHR loop (one pump per loop) operates up to first 600 seconds into the accident in the RV injection mode. After 600 seconds, it is assumed that the operator actuates one RHR loop consisting of one pump and one heat exchanger operating in the direct suppression pool cooling mode. During the transient, when the energy removal rate of the RHR system exceeds the energy addition rate from the decay heat and pump heat, the containment pressure and temperature reach a second peak value and then decrease gradually.

The long-term DBA LOCA analysis demonstrates that the peak suppression pool temperature remain below its design limit. Table 2.6-1 in Reference 7, as revised in Reference 65, presents the EPU analysis results and the acceptance criteria. The long term peak suppression pool temperature of this table is reproduced in Table 3 below:

Table 3: EPU Long-Term LOCA
Suppression Pool Temperature Response Results

Parameter	At CLTP from USAR	EPU Analysis	Design Limit
Peak Bulk Suppression Pool Temperature (°F)	194.2	203	212

The licensee has revised the suppression pool design temperature from its current value of 196.7°F to the EPU value 212°F. The EPU peak suppression pool temperature of 203°F is less than the suppression pool newly selected design temperature of 212°F. Since the licensee used acceptable calculation methods and conservative assumptions and the calculated values are below the design limits, the long-term containment calculation for EPU conditions is acceptable.

Hydrodynamic Loads

Part of the containment design basis is the acceptable response of the containment to hydrodynamic loads associated with the discharge of reactor steam and drywell nitrogen into the suppression pool following a DBA LOCA or the discharge of reactor steam following actuation of the SRVs. The licensee used analytical and empirical methods developed by the Mark I Owners' Group and approved by the NRC staff in Reference 39 to address these issues for MNGP.

The licensee must ensure, as part of the power uprate evaluation, that these analyses remain bounding for operation at EPU conditions. This is done by the pressure and temperature calculations for short-term DBA LOCA which assumes a large double-ended guillotine break of a recirculation suction line. The key parameters are the drywell and wetwell pressure, vent flow rates, and the suppression pool temperature.

The licensee considered DBA LOCA-induced loads such as the vent thrust loads during vent clearing, pool swell loads, Condensation Oscillation (CO) loads, and chugging loads. Vent clearing refers to the ejection of water in the downcomers caused by drywell pressurization as a result of the DBA LOCA.

The pool swell and CO loads are a function of the initial drywell pressurization rate during a DBA LOCA. The licensee stated that the short term containment response conditions are within the range of test conditions used to define the pool swell and CO loads. The long-term containment response conditions, when chugging would occur, are within the range of conditions used to define the chugging loads. The licensee calculated EPU vent thrust loads and found them to be less than the MNGP plant specific values calculated during the Mark I containment long term program. The licensee was requested to explain why the vent thrust loads at EPU conditions are less than the MNGP plant specific values calculated for the Mark I containment long term program. In its response to RAI-20 (Reference 47), the licensee provided the following information:

> The original MNGP Plant Unique Load Definition (PULD) analysis and the MNGP EPU analysis for vent thrust loads both use the same methodology for evaluating containment response and vent thrust loads. This methodology uses the GEH M3CPT containment code to evaluate the containment response used to evaluate the vent thrust loads in accordance with the Mark I Load Definition Report.
>
> The original PULD analysis calculated break flow rates using the homogeneous equilibrium model (HEM) for evaluating critical break flow rate, and the vessel model internal to M3CPT. The vessel model internal to M3CPT is very simplistic, and requires very conservative assumptions to account for subcooled liquid break flows that maximize the containment pressure response required for evaluating bounding vent thrust loads.
>
> The EPU analysis of the DBA LOCA containment response for calculating vent thrust loads uses the GEH LAMB code for calculating break flow rates as input to the M3CPT code. Break flow rates calculated with LAMB use the same break flow model (HEM) as used with the M3CPT internal vessel model. But the LAMB vessel model is more detailed than the M3CPT internal vessel model, and can

therefore, provide break flow rates more consistent with the GEH BWR vessel, especially with regard to subcooled liquid break flows.

The vent thrust loads at EPU conditions are therefore less than the MNGP PULD values because the containment response for the EPU conditions uses LAMB break flow rates rather than the break flow rates calculated with the vessel model in M3CPT.

The NRC staff considers the above explanation acceptable. The licensee's evaluation of containment hydrodynamic loads as a result of a DBA LOCA is in accordance with the EPU topical report (Reference 8) and shows acceptable results.

Safety/Relief Valve (SRV) Loads

The dynamic loads on the suppression pool due to the discharge of steam from SRVs are part of the containment design basis. The loads evaluated for the initial and subsequent SRV actuations under EPU conditions are SRV discharge lines loads, suppression pool boundary loads, and loads on the submerged structures in the suppression pool. The licensee stated that the parameters that affect the SRV load are the RV pressure, the SRV opening setpoints, SRV discharge line air and water volumes, and the configuration of the submerged structures in the suppression pool. The NRC staff requested that the licensee verify that these parameters will not change under EPU conditions. In its response the licensee stated that for the initial SRV actuation the parameters will not change, and therefore the loads due to initial actuation are not impacted by the EPU. The loads due to subsequent SRV actuations depend primarily on the SRV discharge line reflood height at the time of SRV opening and SRV setpoints. The licensee stated that the number of SRV cycles will increase with EPU due to a higher steaming rate at increased decay power levels. EPU will reduce the time between actuations to about 12 seconds. The time at which equilibrium height is re-established remains less than 6 seconds after the SRV closes, which is independent of reactor power level. The current SRV low-low set logic includes a minimum 8-second delay after valve closure. The current SRV low-low set logic therefore prevents subsequent SRV actuations until after the SRV discharge reflood level stabilizes to the equilibrium height. Therefore, the current specified loads due to initial and subsequent SRV actuations are not affected by EPU.

Local Pool Temperature with SRV Discharge

NUREG-0783 (Reference 41) specifies a local pool temperature limit for SRV discharge because of concerns resulting from unstable condensation observed at high pool temperatures in BWRs without quenchers. By Reference 42, the NRC approved an amendment that eliminated local suppression pool temperature limits from the MNGP USAR as the basis for limiting suppression pool mechanical loads due to unstable steam condensation during SRV actuations.

Instrumentation and Control for Containment Monitoring

The licensee has not proposed any changes to instrumentation and controls provided to monitor and maintain variables within prescribed operating ranges. The licensee also has not proposed any changes to instrumentation used to monitor the containment atmosphere for radioactivity that may be released from normal operations and from postulated accidents.

Conclusion

The NRC staff has reviewed the licensee's assessment of the containment temperature and pressure transient and concludes that the licensee has adequately accounted for the increase of mass and energy resulting from the proposed EPU. The NRC staff further concludes that containment systems will continue to provide sufficient pressure and temperature mitigation capability to ensure that containment integrity is maintained. The NRC staff also concludes that containment systems and instrumentation will continue to be adequate for monitoring containment parameters and release of radioactivity during normal and accident conditions and the containment and associated systems will continue to meet the requirements of GDC-4, 13, 16, 50, and 64 following implementation of the proposed EPU.

Therefore, the NRC staff finds the proposed EPU acceptable with respect to primary containment functional design subject to approval of:

- The change in design temperature of wetwell attached piping from 196.7°F to 212°F (Reference 7, as revised in Reference 65), by EMCB in Section 2.2.
- The equipment qualification of equipment inside containment due to change in the drywell gas temperature design limit from 335°F to 338°F (Reference 7, as revised in Reference 65), by EEEB in Section 2.3.1.

2.6.2 Subcompartment Analyses

Regulatory Evaluation

A subcompartment is defined as any fully or partially enclosed volume within the primary containment that houses high-energy piping and would limit the flow of fluid to the main containment volume in the event of a postulated pipe rupture within the volume. The NRC staff's review of subcompartment analyses covered the determination of the design differential pressure values for containment subcompartments. The staff's review focused on the effects of the increase in mass and energy release into the containment caused by operation at EPU conditions and the resulting increase in pressurization. The NRC staff acceptance criteria for the subcompartment analyses are based on the following GDCs in 10 CFR 50, Appendix A:

GDC-4, "Environmental and dynamic effects design bases," insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents and that such SSCs be protected against dynamic effects.

GDC-50, "Containment Design Basis," insofar as it requires that containment subcompartments be designed with sufficient margin to prevent fracture of the structure resulting from the calculated pressure differential conditions across the walls of the subcompartments. SRP Section 6.2.1.2 contains specific review criteria.

The GDC discussed above are those currently specified in Appendix A of 10 CFR Part 50. The applicable MNGP principal design criteria predate these Appendix A criteria. These MNGP principal design criteria are listed in the MNGP USAR, Section 1.2, "Principal Design Criteria". In 1967, the AEC published for public comment a revised set of proposed GDC (32 *FR* 10213, July 11, 1967). An evaluation comparing the MNGP design basis to the AEC proposed GDC of

1967 is presented in MNGP USAR, Appendix E "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria."

While MNGP is not explicitly licensed to the current General Design Criteria or the 1967 AEC proposed GDC, the licensee's stated that the current GDCs-4 and 50 are comparable to the applicable AEC proposed GDC-40, 42, and 49.

Technical Evaluation

The licensee stated that the break that caused maximum pressurization within the reactor shield annulus region is the recirculation suction nozzle safe end break. A break at this location results in higher mass and energy releases than any other break, including breaks of the FW lines. The main steam lines are located above the top elevation of the annulus region and therefore do not impact its pressurization. The licensee stated that the mass and energy releases that affect annulus pressurization (AP) loads on the biological shield wall (BSW) caused by a postulated recirculation suction line break at EPU conditions and Maximum Extended Load Line Limit Analysis (MELLLA), and 105 percent core flow was compared to the pressure load at current/MELLLA conditions. The methods used for the evaluation are consistent with those used in the current analysis, including the evaluation performed for MELLLA. The licensee stated that current/MELLLA annulus pressure of 40 psid remains bounding and has considerable margin with the BSW structural design pressure of 58 psid. For a more limiting case of recirculation suction line break with minimum recirculation pump speed point on the MELLLA line, which is not a part of current design basis, the annulus pressure developed is 42.3 psid which has sufficient margin from the BSW structural design pressure of 58 psid.

In Reference 57, Item 20, the licensee provided evaluation of the impact of various plant improvements on the BSW doors due to AP loads. The licensee confirmed that the design differential pressure capability of the BSW doors (54 psid) bounds the expected AP LOCA differential pressure during the evaluated off-normal conditions which was postulated to release higher M&E release than the release at the design power and core flow operating point. The MNGP configuration before the EPU application had bricks installed in the penetration gaps between the pipe and the BSW structure. In its July 13, 2009, response to RAI-17 (Reference 47), the licensee stated that three of the BSW piping penetrations had shielding bricks installed in the gap between the pipe and the structure. As stated in Reference 57, Item 20, and also in the above licensee's response, the bricks were permanently removed during the refueling outage that ended in May 2009. Removal of these bricks eliminated the potential of high energy brick missiles caused by the AP resulting from a break.

The drywell head region is subject to steam breaks only. Since the current and EPU reactor operating pressures are the same, the pressurization from steam line breaks, including the head vent line does not change as a result of EPU. Dynamic effects from these breaks are also not changed at EPU conditions.

Conclusion

The NRC staff has reviewed the subcompartment assessment performed by the licensee and the change in predicted pressurization resulting from the increased mass and energy release. The staff concludes that containment SSCs important to safety will continue to be protected from the dynamic effects resulting from pipe breaks and that the subcompartments will continue to have sufficient margins to prevent fracture of the structure as the result of pressure difference across the walls following implementation of the proposed EPU. Based on these findings, the NRC staff concludes that the plant will continue to meet GDCs-4 and 50 for the proposed EPU and, therefore, finds the proposed EPU acceptable with respect to subcompartment analyses.

2.6.3 Mass and Energy Release

2.6.3.1 Mass and Energy Release Analysis for Postulated Loss of Coolant Accident

Regulatory Evaluation

The release of high-energy fluid into containment from pipe breaks could challenge the structural integrity of the containment, including subcompartments and systems within the containment. The NRC staff's review covered the energy sources that are available for release to the containment and the mass and energy release rate calculations for the initial blowdown phase of the accident. The NRC staff acceptance criteria for the mass and energy release analysis for postulated loss of coolant accident are based on 10 CFR 50: Appendix A, GDC-50, "Containment design basis," insofar as it requires that sufficient conservatism be provided in the mass and energy release analysis to assure that containment design margin is maintained.

Appendix K, "Emergency Core Cooling System (ECCS) Evaluation Models," insofar as it identifies sources of energy during a LOCA.

SRP Section 6.2.1.3 contains specific review criteria.

The GDC discussed above are those currently specified in Appendix A of 10 CFR Part 50. The applicable MNGP principal design criteria predate these Appendix A criteria. These MNGP principal design criteria are listed in the MNGP USAR, Section 1.2, "Principal Design Criteria". In 1967, the AEC published for public comment a revised set of proposed GDC (32 *FR* 10213, July 11, 1967). An evaluation comparing the MNGP design basis to the AEC proposed GDC of 1967 is presented in the MNGP USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria."

While MNGP is not explicitly licensed to the current General Design Criteria or the 1967 AEC proposed GDC, the licensee's stated that the current GDC-50 is comparable to the applicable AEC proposed General Design Criterion 49. The licensee stated that the mass and energy release analysis is applicable as described in USAR Section 5.2.3.2, which provides the sources of mass and energy.

Technical Evaluation

Section 2.6.1 provides a technical evaluation of the mass and energy release following a High Energy Line Break (HELB) in containment.

Conclusion

The NRC staff has reviewed the licensee's mass and energy release assessment and concludes that the licensee has adequately addressed the effects of the proposed EPU and adequately accounts for the sources of energy identified in Appendix K to 10 CFR Part 50. Based on this, the staff finds that the mass and energy release analysis meets the requirements in GDC-50 for ensuring that the analysis is conservative. Therefore, the NRC staff finds the proposed EPU acceptable with respect to mass and energy release for a postulated LOCA.

2.6.4 Combustible Gas Control in Containment

Regulatory Evaluation

Following a LOCA, hydrogen and oxygen may accumulate inside the containment as the result of chemical reactions between the fuel rod cladding and steam, corrosion of aluminum and other materials, and radiolytic decomposition of water. If excessive hydrogen is generated, it may form a combustible mixture in the containment atmosphere. The NRC staff's review included: (1) the production and accumulation of combustible gases; (2) the capability to prevent high concentrations of combustible gases in local areas; (3) the capability to monitor combustible gas concentrations; and (4) the capability to reduce combustible gas concentrations. The NRC staff's review primarily focused on any impact that the proposed EPU may have on hydrogen release assumptions and the mitigation of any increases in hydrogen release. The NRC staff acceptance criteria for the combustible gas control in containment are based on the following:

In 10 CFR 50, Appendix A, GDC-5, "Sharing of structures, systems, and components," insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions.

In 10 CFR 50, Appendix A, GDC-41, "Containment atmosphere cleanup," insofar as it requires that systems be provided to control the concentration of hydrogen or oxygen that may be released into the reactor containment following postulated accidents to ensure that containment integrity is maintained.

In 10 CFR 50, Appendix A, GDC-42, "Inspection of containment atmosphere cleanup systems," insofar as it requires that systems required by GDC-41 be designed to permit appropriate periodic inspection.

In 10 CFR 50, Appendix A, GDC-43, "Testing of containment atmosphere cleanup systems," insofar as it requires that systems required by GDC-41 be designed to permit appropriate periodic testing.

In 10 CFR 50.44, "Combustible Gas Control for Nuclear Power Reactors," insofar as it requires that plants be provided with the capability of controlling combustible gas concentrations in the containment atmosphere.

SRP Section 6.2.5 contains specific review criteria.

The GDCs discussed above are those currently specified in Appendix A of 10 CFR Part 50. The applicable MNGP principal design criteria predate these Appendix A criteria. These MNGP principal design criteria are listed in MNGP USAR Section 1.2, "Principal Design Criteria". In 1967, the AEC published for public comment a revised set of proposed GDC (32 *FR* 10213, July 11, 1967). An evaluation comparing the MNGP design basis to the AEC proposed GDC of 1967 is presented in MNGP USAR, Appendix E "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria."

While MNGP is not explicitly licensed to the current General Design Criteria or the 1967 AEC proposed GDC, the licensee's stated that the current GDC-5, -42, and -43 are comparable to the applicable AEC proposed GDC-4, 62, 63, 64, and 65. The licensee stated that there is no AEC proposed GDC that associates with the current GDC-41.

Technical Evaluation

The NRC has revised 10 CFR 50.44, "Standards for Combustible Gas Control System in Light-Water-Cooled Power Reactors." The amended standards eliminated the requirements for hydrogen recombiners and relaxed the requirements for hydrogen and oxygen monitoring in containment. The revised 10 CFR 50.44 no longer defines a DBLOCA hydrogen release, and eliminates requirements for hydrogen control systems to mitigate such a release.

The installation of hydrogen recombiners and/or vent and purge systems required by 10 CFR 50.44(b)(3) was intended to address the limited quantity and rate of hydrogen generation that was postulated from a DBLOCA. The NRC approved and issued the MNGP license amendment request (Reference 43) that removed the requirements for hydrogen recombiners. The licensee has subsequently abandoned this equipment in-place. Therefore, plant operation under EPU conditions does not affect the current combustible gas control system.

Conclusion

The NRC staff has reviewed the licensee's assessment related to combustible gas and concludes that the plant design is consistent with the requirements in 10 CFR 50.44 and 10 CFR 50.46 for systems required to control the concentration of hydrogen or oxygen that may be released into the reactor containment following postulated accidents to ensure containment integrity is maintained at extended power. Therefore, the NRC staff finds the proposed EPU acceptable with respect to combustible gas control in containment.

2.6.5 Containment Heat Removal

Regulatory Evaluation

Fan cooler systems, spray systems, and RHR systems are provided to remove heat from the containment atmosphere and from the water in the containment wetwell. The NRC staff's review in this area focused on: (1) the effects of the proposed EPU on the analyses of the Net Positive Suction Head (NPSH) available to the RHR and CS pumps available for containment heat removal; and (2) the analyses of the heat removal capabilities of the spray water system and the fan cooler heat exchangers. The NRC staff's acceptance criteria for containment heat removal are based on 10 CFR 50 Appendix A.

GDC-38, "Containment heat removal," insofar as it requires that a containment heat removal system be provided and that its function shall be to rapidly reduce the containment pressure and temperature following a LOCA and maintain them at acceptably low levels.

SRP Section 6.2.2, as supplemented by Draft Guide 1107, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," contains specific review criteria.

The GDC discussed above are those currently specified in Appendix A of 10 CFR Part 50. The applicable MNGP principal design criteria predate these Appendix A criteria. These MNGP principal design criteria are listed in MNGP USAR Section 1.2, "Principal Design Criteria." In 1967, the Atomic Energy Commission (AEC) published for public comment a revised set of proposed GDC (32 *FR* 10213, July 11, 1967). An evaluation comparing the MNGP design basis to the AEC proposed GDC of 1967 is presented in MNGP USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria." While MNGP is not explicitly licensed to the current General Design Criteria or the 1967 AEC proposed General Design Criteria, the licensee's stated that the current GDC-38 is comparable to the applicable AEC proposed GDC-41 and -52.

Technical Evaluation

Systems supporting containment heat removal are partly evaluated in Section 2.6.1 from a heat removal capability and the ability to maintain the containment pressure and temperature within design limits during design basis events under EPU conditions. The following sections provide evaluation of the ECCS and containment heat removal system pump operation from the standpoint of having adequate available NPSH at pump suction inlet during design basis and special events. The ECCS and the containment heat removal pumps are the RHR and CS system pumps.

The heat addition to the suppression pool is increased under EPU conditions due to increase in the reactor decay heat. This increased heat input increases the peak suppression pool water temperature, which affects the available NPSH at the suction inlet of the RHR and CS pumps, thereby affecting their operation. The licensee performed NPSH analyses for the following events: (1) DBA LOCA, and (2) non-design basis events which are (a) Small Steam Line Break Accident (SBA), (b) Appendix R Fire, (c) Anticipated Transient Without Scram (ATWS), and (d) Station Blackout (SBO). The licensee's analyses for each event consisted of two steps: (a) containment analysis using the SHEX computer code to calculate the transient wetwell pressure and the corresponding transient suppression pool temperature, (b) calculate the NPSH available (NPSHa) at the inlet of the RHR and CS pumps using the transient suppression pool temperature, and use either full, or partial, or none of the transient wetwell pressure as inputs, and check if NPSHa is greater than or equal to the NPSH required (NPSHr) to confirm its adequacy. As stated in Reference 52, the transient wetwell pressure developed during the above events is termed as Containment Accident Pressure (CAP).

Reference 52 provides NRC staff guidance for determination of plant site value of NPSHr designated as NPSHreff from the shop tested value of NPSHr designated as NPSHr3%, and the uncertainty in NPSHr3% designated as "uncertainty." Reference 52 defines NPSHr3% as that value of NPSH that results in a 3-percent drop in pump discharge head. The relationship between NPSHreff, NPSHr3% and 'uncertainty' is as follows:

NPSHreff = (1 + uncertainty) x NPSHr3%

For the design basis events, Reference 52 requires uncertainty to be determined and included in the above equation. For special events, Reference 52 allows NPSHreff equal to NPSHr3%.

DBA LOCA Analysis for NPSHa

The DBA LOCA containment analysis for NPSH consists of a short-term analysis, up to 600 second into the event, and a long-term analysis, after 600 seconds into the event. The short-term analysis covers that period from the time of the break until operator action is taken to throttle the RHR and CS system pumps and establish containment cooling. The analysis assumes reactor operating at 2-percent above the EPU RTP to include instrument uncertainty effects. The licensee used ANS 5.1-1979 decay heat model including 2-sigma uncertainty and guidance in SIL 636, Revision 1 as used in the EPU long term suppression pool temperature response analysis. For conservatism, the licensee used assumptions and biased the input parameters to minimize the wetwell pressure and maximize the suppression pool temperature. The containment leakage TS limit of 1.2 percent weight per day to the environment is assumed in the analysis which is conservative for minimizing the wetwell pressure. The licensee used a K-value of 147 Btu/s-°F for each RHR heat exchanger and assumed it as constant throughout the transient consistent with the current analysis of record. In a RAI-the licensee was requested to provide reasons for using a constant K-value of RHR heat exchanger instead of a variable K-value as used in the long-term suppression pool temperature response analysis. In its response to RAI-7 (Reference 47), the licensee stated that the version of SHEX code used for the long-term analysis with RHR operating in spray mode does not have the capability to use a variable K-value.

The NPSH analysis conservatively assumes containment cooling via the containment spray mode, and therefore could not use the variable K-value. The containment spray mode is conservative because it gives lower values of NPSHa, which is more limitinig compared to the NPSHa results obtained from RHR direct suppression pool cooling mode.

The licensee's calculated suppression pool temperature response used for its NPSHa evaluation has a peak temperature of 207.1°F (Reference 49) for the DBA LOCA event.

The licensee has credited CAP in the NPSHa analysis for the RHR and CS under EPU conditions. The current analysis for DBA LOCA NPSHa for the RHR and CS pumps also credits CAP (Reference 38) to justify adequate NPSHa at the suction inlet of RHR and CS pumps. For the proposed EPU analysis, the licensee used NRC staff guidance on use of CAP in Sections 6.6.1 through 6.6.10 of Reference 52. The licensee responded to the NRC staff guidance in Sections 6.6.1, 6.6.2, 6.6.3, 6.6.5, 6.6.6, 6.6.8, 6.6.9, and 6.6.10 of Reference 49, and to Sections 6.6.4 and 6.6.7 of Reference 49, for operating under EPU conditions in the MELLLA+ domain. The NRC staff's evaluation of the licensee's response to Sections 6.6.1 through 6.6.10 is given below:

Reference 52, Section 6.6.1, states the following:

> For DBAs, a value of NPSHreff should be used in the analyses concerning the use of containment accident pressure. NPSHreff includes the uncertainty in the value of NPSHr3% based on vendor testing and installed operation. The effects of motor slip, suction piping configuration, and air content, and wear ring leakage should be included.
>
> NPSHreff = (1 + uncertainty) x NPSHr3%
>
> For non-DBAs, NPSHr3% may be used.

NRC Staff Evaluation of Licensee's Response in Reference 49

The licensee used NRC staff guidance for determining the installed operational NPSHreff by quantifying the uncertainty in the value of NPSHr3% based on pump vendor testing. The licensee used the Boiling Water Reactor Owners' Group (BWROG) evaluation (Reference 54) of the uncertainty in the NPSHr3% for the CVDS model RHR and CS pumps which are similar in design. The BWROG assessment of uncertainties included the effects of pump suction piping configuration, air content in water, pump speed, test instrumentation accuracy, and pump wear ring leakage. The licensee stated that the BWROG initial approach was to quantify the uncertainty in the NPSHr3% using the Computational Fluid Dynamics (CFD) method. Using the CFD analysis results in attempting to benchmark against the original vendor test results for the RHR pump, it was determined that the CFD analysis results for NPSHr were not representative of the original test and therefore BWROG abandoned this approach. The BWROG then used an alternate approach using a qualitative analysis method to quantify the NPSHr uncertainty.

Regarding the uncertainty due to motor speed, the licensee justified that consideration of motor slip speed impact on NPSHr uncertainty is conservative as the currently installed motors will not operate at speeds above the original factory test for nominal frequencies of 60 Hertz. The licensee stated that consideration of frequency variation due to the emergency diesel generator speed uncertainty bounds the pump speed uncertainty. For the RHR pumps, the TS limits the required speed correction due to EDG output frequency to 2-percent, which results in an impact on NPSHr uncertainty for speed of +/- 4-percent.

Regarding the uncertainty due to air content, the licensee stated that for the long term, the Emergency Operating Procedures (EOPs) require the lowest suppression pool level be maintained to ensure that air entrainment will not occur. However, during the short term (for the first 10 minutes of the event) when the EOPs do not apply, the licensee stated that the suppression pool does not lose enough water volume to cause its level to drop below its air entrainment level. Therefore, the short-term air content consideration is limited to the impact of uncertainty due to dissolved gas in the suppression pool water.

Based on the BWROG reports (Reference 55), the licensee stated that uncertainty obtained using the alternative approach for the pump suction piping configuration, plus the uncertainties due to pump speed, factory test instrumentation, dissolved gas, wear ring leakage and temperature support a conservative total uncertainty of 21 percent for the DBA LOCA long-term analysis.

The NRC staff considers the licensee's response to NRC staff guidance in Section 6.6.1 acceptable because the licensee considered all items contributing to the uncertainty in NPSHr3% in calculating the NPSHreff. For the DBA LOCA analysis, the licensee increased the NPSHr3% by a factor of 1.21 and calculated the NPSHreff for the RHR and CS pumps. For non-design basis events, the licensee used the NPSHr3% in accordance with the NRC staff guidance.

Reference 52, Section 6.6.2, states the following:

> The maximum flow rate chosen for the NPSHa analysis should be greater than or equal to the flow rate assumed in the safety analyses that demonstrate adequate core and containment cooling. This ensures that the safety analysis and the NPSH analysis are consistent. If the NPSHa is assumed to equal the NPSHr3% (the usual assumption for determining the amount of containment accident pressure used), then the flow rate used in the core and containment cooling analyses should also be equal to or greater than the flow rate resulting from a 3-

percent decrease in pump TDH [total dynamic head].

NRC Staff Evaluation of Licensee's Response in Reference 49

In response to Section 6.6.2 of Reference 52, the licensee revised long term DBA LOCA NPSHa analyses from the previous analysis reported in Reference 7. For the DBA LOCA NPSHa short term analysis, the licensee stated that the CS flow assumed is the same as in the previous analysis reported in Reference 7, Section 2.6.5, which is the same as assumed in the current ECCS core cooling analysis.

For the DBA LOCA NPSHa long term analysis, the licensee conservatively revised the CS flow delivered to the core to 3020 gallons per minute (gpm) (Reference 57, Item 8). The revised flow is based on the design rating of the CS system. In addition, the licensee accounted for leakages and flow through the pump minimum flow line which increased the total required CS pump flow to 3388 gpm (Reference 57, Item 8). The licensee stated that the NPSHr3% at the revised flow of 3388 gpm is 23.3 feet which is not significantly greater from the previously used value of 23 feet for NPSHr3% at 3029 gpm.

For the DBA LOCA NPSHa long term analysis, the licensee conservatively increased the RHR pump flow to 4178 gpm (Reference 57, Item 8) from its previous value of 4000 gpm reported in Reference 1, Section 2.6.5, while maintaining the RHR heat exchanger flow at 4000 gpm. The increased flow of 178 gpm is the flow through the pump minimum flow line.

Evaluation of Response to Appendix R fire NPSHa analysis is provided in Section 2.6.5.2.3 of this SER.

Evaluation of Response to ATWS NPSHa analysis is provided in Section 2.6.5.2.4 of this SER.

Table 6.6.2-1 of Reference 49 provides the licensee's NPSH analyses results using the revised RHR and CS pump flow rates for the DBA LOCA, ATWS, and Appendix R fire events. This table supersedes the information in the table provided by the licensee in response to the NRC staff RAI-number SCVB-RAI-5 in Reference 48 for these events.

The NRC staff considers the response to the guidance in Section 6.6.2 of Reference 52 to be acceptable because the licensee revised the previous DBA LOCA, ATWS and Appendix R Fire NPSHa analysis reported in Reference 7, Section 2.6.5, by using more conservative CS and RHR pump flow rates to maximize the pump suction piping head loss.

Reference 52, Section 6.6.3, states the following:

> A 95/95 lower tolerance limit should be used to calculate the containment accident pressure used to determine the NPSHa.

NRC Staff Evaluation of Licensee's Response in Reference 49

The NRC staff guidance requires the licensees proposing the use of CAP for determining NPSHa to perform containment DBA LOCA analyses using the Monte Carlo approach to demonstrate great NPSH margin than obtained from the conservative DBA LOCA analysis. To satisfy the NRC staff guidance, the BWROG presented the Monte Carlo analysis using MNGP plant-specific data and assumptions. The input values for some parameters are sampled from statistical distributions, and conservative (bounding) values are used for others. An acceptance criterion of a 95-percent probability at a 95-percent confidence level (95/95) is used for the Monte Carlo analysis. However, since conservative values are used for other input to the NPSHa calculation, the tolerance limit on NPSHa is actually greater than the 95/95 valve. The Monte Carlo DBA LOCA analysis for MNGP provided in Reference 53 is acceptable to the NRC staff as noted in Reference 52. The Monte Carlo approach has the advantage of defining and quantifying the degree of conservatism in the value of NPSHa obtained from the conservative analysis described above in this Section. The licensee demonstrated (see Figure 6.6.9-3 in Reference 49) the conservatism in the conservative DBA LOCA analysis by determining greater NPSH margin than that obtained in the Monte Carlo approach. The NRC staff considers the response to the NRC staff guidance in Section 6.6.3 of Reference 52 to be acceptable because the licensee quantified the conservatism in the conservative approach used for the DBA LOCA analysis.

Reference 52, Section 6.6.4

Section 6.6.4 of Reference 52 pertains to an Appendix R Fire event. Refer to "Appendix R Fire Analysis for NPSHa" on Page 100 of this SE for the NRC staff's evaluation of the licensee's response.

Reference 52, Section 6.6.5, states the following:

> Operator action to control containment accident pressure is acceptable. The NRC staff should approve any operator actions, and the appropriate plant procedures (e.g., emergency, abnormal) should include them.

NRC Staff Evaluation of Licensee's Response in Reference 49

The EOPs provide operator actions to control wetwell temperature and pressure, drywell temperature and pressure, and hydrogen concentration. The licensee stated that no changes are required in the EOPs to support operation under EPU condition in the MELLLA Plus (MELLLA+) domain. The licensee stated that a caution is included in the EOPs that if the containment pressure falls below the value required for CAP based on NPSHr3% then this may result in inadequate NPSHa. In a RAI-the licensee was requested to justify the use of NPSHr3% instead of NPSHreff, where NPSHreff = (1 + uncertainty) x NPSHr3%. The NRC staff considers NPSHreff is the required pump NPSH under the site conditions. In its response (Reference 55, NRC Question 1), the licensee stated that the existing EOP caution will be revised to identify to the operators that if containment pressure falls below 8.6 psig that inadequate NPSH may exist. This value corresponds to the amount of containment pressure required for NPSHreff with a DBA LOCA.

Reference 25, Section 6.6.6 states the following:

> It is possible that the NPSHa may be less than NPSHreff (LOCA) or NPSHr3% (non-DBA).
>
> Operation in this mode is acceptable if appropriate tests are done to demonstrate that the pump will continue to perform its safety functions. The following conditions should apply:

a. Predicted operation during the postulated accident below NPSHreff (LOCA) or NPSHr3% (nondesign-basis event) is of limited duration (less than 100 hours).

b. The tests are conducted on the actual pump with the same mechanical shaft seal (including flush system) or at least a pump of the same model, size, impeller diameter, materials of construction, and pump seal and flush system.

c. The test is conducted at the same (field application) speed.

d. The test is conducted at the actual predicted NPSHa since testing at a lower NPSHa can actually reduce, rather than increase, the cavitation erosion rate in some cases.

e. The test duration should be for the time NPSHa is predicted to be less than NPSHreff (LOCA) or NPSHr3% (nondesign-basis event).

f. The flow rate and discharge head must remain above the values necessary to provide adequate core and containment cooling.

NRC Staff Evaluation of Licensee's Response in Reference 49

In response to item (a) above, the licensee stated that there is no accident or event that results in operation with NPSHa below NPSHreff. The results for DBA LOCA short term analysis show that for a short duration, from about 360 second to 600 seconds (for 4-minutes approximately) from the event, the pump will operate with NPSHa below NPSHreff. The licensee justified the conditions associated with the test to demonstrate that the pumps will perform the required safety function, as discussed in responses to items (b) through (f), have been met.

In response to item (b) above, the licensee stated that operating a pump in the vicinity of the NPSHr3% condition requires consideration of the potential for damage that water vapor or entrained air, or both, could do within the pump to the mechanical shaft seal faces, which could fail in a very short period of time if the seal faces run dry. Both RHR and CS pumps have a similar design for the seal flushing system. The licensee refers to Attachment 6 of Reference 55 which provides the pump manufacturer's (Sulzer) evaluation of the effect of non-condensable gases that come out of the solution and migrate to the seal purge piping of the RHR pump. Based on this evaluation the licensee concluded that sufficient flushing flow exists to remove bubbles that could be created in the process stream from the gas coming out of solution. The vertical configuration of the pump seals also supports removal of any non-condensable gas and therefore the gas will not cause damage to pump internals. The licensee confirmed the applicability of this analysis to the CS pump.

In response to item (c) above, the licensee stated that the factory tests for RHR and CS pumps were done at approximately the same speed as the speed in its installed condition. However motor replacements for the pumps have resulted in some very small variations in motor slip for some motors since original installation.

In response to item (d) above, the licensee refers to Attachment 5 of Reference 55 for the evaluation of erosion rate for the RHR pump impeller due to cavitation. According to this analytical evaluation, there is substantial margin (more than 6200 days compared to its 30 days of mission time) in the impeller life while operating under the worst conditions postulated for impeller erosion. The NRC staff requested in an RAI that the licensee describe the worst operating conditions postulated for impeller erosion. In its March 21, 2013, response to NRC Question 2, the licensee stated that this condition is shown in Attachment 5, Figure 2, of Reference 55, indicating that the maximum erosion rate occurs in the range of 1.75 to 2.5 times the ratio of NPSHa/NPSH3%. Based on the evaluation, the licensee concludes that the worst conditions will not challenge the ability of the pumps to operate for their required mission time. The licensee also confirmed the applicability of the evaluation to the CS pump. This licensee stated that the analysis bounds the results from the requested testing for NPSHa.

In response to item (e) above, the licensee refers to Attachment 4 of Reference 55 for the evaluation of RHR pump at reduced NPSHa and stated that that the RHR pump have been tested with a NPSH at the pump inlet where the dynamic heads drops by 5-percent from the NPSH when there no head drop. The licensee concludes that the cavitation induced vibration does not lead to the pump failure during its 30 days operation under DBA LOCA conditions. The licensee also confirmed the applicability of the evaluation to the CS pump.

In response to item (f) above, the licensee demonstrated that in the NPSHa analysis the flow rate and discharge head for both the CS and RHR pumps assumed are above the values necessary to provide adequate core and containment cooling functions.
The NRC staff requested in an RAI that the licensee correct an inconsistency in two different figure captions given for Figure 6.6.6-5. In its March 21, 2013, response to Question 9, the licensee corrected the figure caption indicating that Figure 6.6.6-5 is for RHR pump B long-term DBA LOCA NPSH. The licensee also provided a similar NPSH graph for RHR pump C.

As an overall conclusion to the evaluation of Section 6.6.6 of Reference 52, the NRC staff considers that the licensee has demonstrated by analyses pump seal failure will not occur during operation of the RHR and CS pumps at slightly reduced NPSHa from NPSHreff conditions for four minutes during the first ten minutes of operation. Also, the licensee demonstrated by testing of the RHR and CS pumps that no detectable degradation of the pump occurred and there is large margin with respect to the RHR and CS pumps achieving their mission time.

Reference 52, Section 6.6.7, states the following:

> Licensees and applicants should consider a loss of containment isolation that could compromise containment integrity. Possible losses of containment integrity include containment venting required by procedures or loss of containment isolation from a postulated Appendix R fire. It should be demonstrated conservatively that, for the plant examined, loss of containment integrity from these causes cannot occur or that they would occur only after use of containment accident pressure is no longer needed.
>
> To reduce the likelihood of a preexisting leak, licensees proposing to use containment accident pressure in determining NPSH margin should do the following:
>
> (1) Determine the minimum containment leakage rate sufficient to lose the containment accident pressure needed for adequate NPSH margin.
>
> (2) Propose a method to determine whether the actual containment leakage rate

exceeds the leakage rate determined in (1) above. For inerted containments, this method could consist of a periodic quantitative measurement of the nitrogen makeup performed at an appropriate frequency to ensure that no unusually large makeup of nitrogen occurs. Monitoring oxygen content is another method. For subatmospheric containments, a similar procedure might be used.

(3) Propose a limit on the time interval that the plant operates when the actual containment leakage rate exceeds the leakage rate determined in (1) above.

NRC Staff Evaluation of Licensee's Response in Reference 49

To determine the minimum containment leakage rate sufficient to lose the CAP needed for adequate NPSH, the licensee performed a containment analysis using GOTHIC code. The analysis used conservative inputs with the exception of the use of a temperature dependent K-value of the RHR heat exchanger. The results of the licensee's analysis for the NPSH margin (NPSHa minus NPSHreff) are shown in Figure 6.6.7-1 of Reference 22 for containment leakages of 1La, 20La, 30La, and 40La. The TS leakage of 1La is equivalent to a leakage rate of 7.6 standard cubic feet per minute (scfm) at CLTP. The results indicate that a containment leakage greater than 228 scfm will result in a complete loss of the NPSH margin.

For the on-line containment leakage monitoring, the licensee has proposed a procedure for detection of a large containment integrity failures where consideration of instrument uncertainty is not necessary. The proposed procedure consists of: (a) an on-line leakage test that determines the containment leakage rate during power operation, and (b) an on-line monitoring of the parameters on which the operator currently relies to determine abnormal containment leakage conditions. The proposed on-line leakage test (a) will be performed after an outage at the beginning of a new cycle when the plant is stabilized at full power. The test is based on a quantitative measurement of the nitrogen (N_2) makeup while the N_2 system is operated with a known vent release rate for eight hours. The test makes use of a computer point that calculates the N_2 gas mass inside the inerted primary containment. This leakage rate test will be a benchmark quantitative test which will provide a baseline that would identify a significant change in containment leakage rate at any time during power operation. This licensee intends to repeat the test at any time during the cycle if inputs identified below for monitoring during normal operation warrant another measurement. For the online monitoring (b), the licensee stated that following control room inputs are available that would indicate an increase in the containment leak rate:

1. A computer point is provided that continually calculates the N_2 mass in containment and provides a computer alarm in the control room if the N_2 mass is too low or too high. The low inventory alarm corresponds to the minimum non-condensable gas mass assumed for the ECCS pump NPSH analysis. Calculated values below the minimum assumed in the NPSH analysis will result in operator action to declare the ECCS pumps inoperable. The computer alarm response procedure includes a note that frequent use of the Nitrogen Makeup System may indicate an increase in N_2 leakage from containment.

2. A control room annunciator that alarms on drywell high or low pressure is available. The low pressure setpoint is 0.1 psig. The annunciator response procedure includes actions that would require investigation if low pressure exists.

3. A flow indicator that measures the N_2 flow in the supply to the containment air system is monitored. Data are taken four times per day to validate that the makeup flow is within the normal range.

The licensee stated that the plant operating procedures include a general precaution that any condition that may indicate an increase in containment nitrogen leakage rate during normal operation is to be carefully and promptly assessed.

In a RAI-the licensee was requested to describe the action to be taken after the operator declares ECCS pumps inoperable. In its March 21, 2013, response to NRC Question 3, the licensee stated that the N_2 inventory is maintained above the low inventory alarm point and the required operator action for low N_2 inventory, which indicates a containment leak, is declaring RHR and CS pumps inoperable and TS 3.5.1, "ECCS –Operating" is applicable. If all ECCS pumps are inoperable under TS 3.5.1.O, the required action is to enter TS Limiting Condition for Operation 3.0.3 immediately, which requires initiation of a reactor shutdown.

The licensee justified the continuous monitoring to ensure CAP is available by stating that the drywell continuous air monitor detection capability is less than 5 scfm. The leakage rate that could challenge the NPSH margin for the ECCS and containment heat removal pumps is greater than 228 scfm which is well above the leakage that can be detected by the proposed procedure. The NRC staff considers the licensee's method to detect loss of containment integrity during power operation acceptable because the proposed on-line test at the beginning of the cycle along with on-line monitoring of the above parameters will ensure the availability of CAP so that NPSH margin is available for the ECCS and containment heat removal pumps during design basis and non-design basis accidents.

The licensee stated that the leakage rates greater than the acceptance criteria will be investigated by the licensee's Corrective Action Program. The licensee did not propose a time limitation for correcting the measured leakage if it is above the acceptance criteria because the proposed leak monitoring procedure does not meet the requirements and controls of a TS-required 10 CFR 50, Appendix J, Type A test. The NRC staff requested that the licensee provide reasons for not specifying an appropriate time limit for performing the correction in case the containment leakage rate determined by the on-line test and monitoring exceeds the acceptance limits. In its March 21, 2013, response to NRC Question 4, the licensee stated that the proposed leak monitoring method does not have sufficient instrument accuracy to measure leakage rates for the 10 CFR 50 Appendix J criteria. The licensee proposed to implement changes in the plant procedure requiring the on-line leakage test to meet the acceptance criterion of 150 scfm. If the acceptance criteria is not met, the plant will immediately enter TS 3.5.1.O (ECCS - Operating) for ECCS inoperable. TS 3.5.1.O requires immediate entry into TS LCO 3.0.3. The NRC staff accepts the licensee's response because it ensures that the plant will not operate when the actual containment leakage rate exceeds the leakage rate requirement.

Reference 52, Section 6.6.8, states the following:

> The zone of maximum erosion rate should be considered to lie between NPSH margin ratios of 1.2 to 1.6. The permissible time in this range, for very-high-suction energy pumps, should be limited unless operating experience, testing, or analysis justifies a longer time. Realistic calculations should be used to determine the time within this band of NPSH ratio values.

NRC Staff Evaluation of Licensee's Response in Reference 49

In its response the licensee referred to an evaluation performed by Sulzer, the RHR pump manufacturer. The evaluation is documented in a BWROG report, provided as Attachment 5, "Task 4 – Operation in the Maximum Erosion Rate Zone (CVDS pump)," of Reference 55. The MNGP RHR pumps manufacturer model number is CVDS. The report provides an evaluation of the potential effects of operating the RHR pump in the range of NPSHa that causes the maximum erosion rate of the pump impeller. The NRC staff has not performed, and does not intend to perform, a SE of this report because it presents a manufacturer's evaluation of the impeller material erosion resistance capability for the RHR pump. The cavitation erosion and the impeller service life calculations for the maximum erosion zone show that the MNGP RHR pump impeller would operate for at least 6200 days while operating at the flow rate and NPSH margin corresponding to the maximum erosion rate. This service life is more than 200 times the minimum required 30-day service life. Based on the evaluation presented in the BWROG report, the licensee stated that the pump manufacturer has assured that the impeller integrity while the pump operates with NPSHa in the zone of maximum erosion. The licensee stated that the CS pump is similar to the RHR pump and its impeller life also significantly exceeds its required mission time while it operates with NPSHa in the maximum erosion zone and under EPU conditions. The NRC staff considers the issue of RHR and CS pump operation in the zone of maximum erosion resolved, as the pump manufacturer's evaluation demonstrated that the impeller life far exceeds the pump mission time while operating with NPSHa in the maximum erosion zone.

Reference 52, Section 6.6.9, states the following:

> A realistic calculation of NPSHa should be performed to compare with the NPSHa determined from the Monte Carlo 95/95 calculation.

NRC Staff Evaluation of Licensee's Response in Reference 49

The NRC staff guidance requires that licensees proposing to use CAP in determining NPSHa should also perform a realistic calculation of NPSHa to compare with the conservative calculation or the Monte Carlo 95/95 calculation. To satisfy the NRC staff guidance, the licensee performed containment DBA LOCA analysis for NPSHa using realistic inputs and assumptions, and using the best-estimate computer code GOTHIC version 7.2b and compared with the NPSHa results from the Monte Carlo 95/95 analysis and conservative analysis using SHEX code. The realistic inputs and their comparison with the conservative inputs are shown in Table 6.6.9-1 of Reference 49. The licensee stated that selected realistic input values are met 98-percent of the time at MNGP. The licensee results for CAP credit given in Table 6.6.9-3 of Reference 49 demonstrates that the calculated CAP credit from the realistic analysis using the GOTHIC code is approximately 70-percent of the CAP credit calculated using the statistical Monte Carlo 95/95 approach, and is approximately 50-percent of the CAP credit calculated from the conservative analysis using SHEX code. The NRC staff considers the response to the guidance in Section 6.6.9 of Reference 25 acceptable because the licensee demonstrated the conservatism in the conservative analysis by using the realistic analysis as well the statistical analysis approach.

Reference 52, Section 6.6.10, states the following:

> The necessary mission time for a pump using containment accident pressure

should include not only the duration of the accident when the NPSH margin may be limited, but any additional time needed for operation of the pump after recovery from the accident when the pump is needed to maintain the reactor or containment, or both, in a stable, cool condition but at a much greater NPSH margin. This additional time is usually taken as 30 days.

NRC Staff Evaluation of Licensee's Response in Reference 49

In its response the licensee stated that the mission time used for the evaluation of the DBA LOCA was 30 days. The licensee confirmed that other events were evaluated until CAP credit was no longer required to mitigate the event. The NRC staff considers the response to the guidance in Section 6.6.10 of Reference 52 to be acceptable.

Small Steam Line Break Accident Analysis for NPSHa

The licensee stated that the most limiting break area for a small steam line break accident (SBA) for NPSHa evaluation is 0.01 ft^2. In a RAI, the licensee was requested to confirm that 0.01 ft^2 break is the most limiting for the NPSH analysis. In its March 21, 2013, response to RAI-5, the licensee stated that break sizes of 0.5 ft^2, 0.1 ft^2, and 0.01 ft^2 were analyzed. The calculated peak suppression pool temperature for the 0.01 ft^2 break was 2°F higher than the peak suppression pool temperature obtained from either of the other two breaks. The smaller flow from the 0.01 ft^2 break maximized the direct transfer of vessel and decay heat energy to the suppression pool with SRV discharges and minimized the energy transferred to the drywell airspace, therefore minimizing the wetwell pressure and maximizing the suppression pool temperature response. The NRC staff accepts the licensee's evaluation because the 0.01 ft^2 break resulted in bounding and conservative conditions of minimum wetwell pressure with maximum suppression pool temperature used in evaluating the NPSH margins.

The licensee conservatively considered one CS pump operating at a flow of 3020 gpm, and one RHR pump operating up to 600 seconds in the RV injection mode at a flow of 4320 gpm. After 600 second one RHR pump operates at 4000 gpm in the containment spray mode. The licensee used ANS 5.1-1979 decay heat model including 2σ uncertainty and guidance in SIL 636, Revision 1. For conservatism, the licensee biased the input parameters to minimize the transient wetwell pressure, and maximize the transient suppression pool temperature. Since the version of SHEX code used cannot directly use variable K-values while in containment spray mode, the licensee used an alternate method to simulate the variable K-value for evaluation of the containment response. In its March 21, 2013, response to RAI-- 6, the licensee explained the alternate method to simulate the variable K-value which the NRC staff considers acceptable.

The licensee's calculated suppression pool temperature response used for its NPSHa evaluation has a peak temperature of 207°F (Reference 1) for the SBA event.

Using the analysis results, the licensee calculated the required wetwell pressure profile that meets the NPSHr3%. Table 2.6-9 of Reference 7 tabulates the available and required wetwell pressures and the margins. The required wetwell pressure exceeded the atmospheric pressure after about 9000 seconds, the maximum reached is 19.73 psia. The NRC staff considers the use of available wetwell pressure for calculating the NPSHa acceptable because the required wetwell pressure is conservative and there is sufficient margin between the available and the required wetwell pressure.

Appendix R Fire Analysis for NPSHa

The licensee performed containment NPSHa analysis for the Appendix R Fire event for two cases: (1) considering a stuck open relief valve (SORV),; and (2) considering no SORV. The licensee used ANS 5.1-1979 decay heat model, and guidance in SIL 636, Revision 1. Consistent with the current analysis, the EPU analyses assumes one CS division with one pump and one RHR division with one pump operating at design flow rates for containment heat removal.

MNGP is currently licensed with a CAP credit in the calculation of NPSHa for Appendix R Fire event (Reference 38). For the EPU analysis, the licensee calculated the suppression pool temperature, minimum wetwell pressure available, and the required wetwell pressure profile to meet the NPSHr3% at the pump inlet. Table 6.6.2-1 in Reference 49 provides the analysis results under EPU conditions which are conservatively revised from the previously reported results given in Tables 2.6-3 and 2.6-4 of Reference 7. The licensee increased the RHR pump flow to account for the flow through the pump minimum flow line while maintaining the same flow through the RHR heat exchanger. Table 6.6.2-1 of Reference 49 supersedes the information in the table provided by the licensee in its August 21, 2009, response to NRC staff RAI-SCVB-RAI-5 for the Appendix R fire cases. A markup of the revisions of the response to SCVB-RAI-5 was provided by the licensee in its January 21, 2013, letter.

The NRC staff requested that the licensee describe the Appendix R fire scenario under EPU conditions which results in most limiting NPSHa. In its July 13, 2009, response to RAI-27, the licensee stated that the limiting fire zones are the cable spreading room and the control room. The licensee stated that the remaining zones are not limiting because for a fire scenario in these zones, both divisions of RHR will be available providing more effective suppression pool cooling resulting in lower temperature and therefore higher NPSHa as compared to the suppression pool cooling with one RHR division. In fire scenarios of both limiting zones, the safe shutdown is accomplished from the Alternate Shutdown System (ASDS) panel using the Alternate Shutdown Cooling Method (ASCM). The minimum equipment available is one train of CS system, one RHR pump, one RHR heat exchanger, and one RHR service water pump and two SRVs. In the ASCM, the reactor is depressurized below the shutoff head of the CS pump or RHR pump using the automatic depressurization system (ADS), and subsequently flooding the RV using either RHR pump in the LPCI mode or the CS pump. The RV water is discharged to the suppression pool through the SRVs. The suppression pool is cooled by the RHR operating in the suppression pool cooling mode. The licensee stated that the plant emergency procedure requires the drywell spray be initiated when the drywell temperature exceeds 281°F. In the EPU Appendix R fire analysis, the drywell temperature does not exceed 281°F, therefore the RHR operation in the spray mode is not used for containment cooling. The licensee used a variable heat exchanger K-value which is a change from the current Appendix R Fire containment analysis. The justification of the heat exchanger variable K-value is provided in Section 2.6.1 of this SE.

The licensee provided a response to NRC staff guidance in Section 6.6.2 of Reference 52 for the NPSHa analysis for an Appendix R fire event. The licensee's proposed evaluation in Reference 49 is the same as previously submitted in Reference 7, Section 2.6.5., with the exception of a conservatively increased RHR pump flow rate from 4000 gpm to 4178 gpm, accounting for the flow through the pump minimum flow line assuming the a fail-open minimum flow valve.

The licensee's calculated suppression pool temperature response used for its NPSHa evaluation has a peak temperature of 195.4°F (Reference 49) for the Appendix R Fire event.

The licensee provided a response to NRC staff guidance in Section 6.6.4 of Reference 52.

The NRC staff's evaluation of the licensee's response is provided below.

Reference 52, Section 6.6.4, states the following:

> It should be demonstrated conservatively that, for the plant examined, loss of containment integrity from containment venting, circuit issues associated with an Appendix R fire, or other causes cannot occur or that they would occur only after use of containment accident pressure is no longer needed.

NRC Staff Evaluation of Licensee's Response in Reference 51

In response to the above guidance, the licensee stated that the most limiting Appendix R Fire scenario is a fire occurring in a cable spreading room or control room for which the alternate shutdown cooling capability for the reactor is provided as per 10 CFR 50 Appendix R, III.G.3. In this scenario, the ASDS panel and the ASCM is used for the safe shutdown cooling of the reactor. For the EPU analysis, the licensee evaluated the effect of Multiple Spurious Operations (MSOs) during the Appendix R limiting fire scenario using the guidance given in Reference 7, and Regulatory Guide 1.189, Revision 2, "Fire Protection for Nuclear Power Plants." The NRC staff has endorsed Reference 59 in Reference 58 as an acceptable method for performing circuit analysis. The licensee considered a combined spurious operation of maximum of four components and modeled the containment cooling using GOTHIC, Version 7.2b, computer code to calculate the transient wetwell pressure and the transient suppression pool temperature.

Using the output suppression pool temperature and the wetwell pressure transients, the licensee calculated the transient NPSHa at the suction inlet of the CS and RHR pumps. Out of the 13 MSO cases analyzed, the most limiting case shows a margin (NPSHa minus NPSHr3%) of 1.7 feet. The licensee describes this case as follows:

> At 600 seconds into the event, 3 RHR non credited pumps start and the wetwell spray valve opens. The system failures continue for the duration of the event. The credited RHR pump also starts. Heat is added to the suppression pool from all 4 RHR pumps.

The NRC staff considers GOTHIC code, which is a best estimate thermal hydraulic analysis code, acceptable for Appendix R Fire analysis because the cases analyzed are non-design basis events. By performing the thermal-hydraulic analysis, the licensee identified the need for the following modifications: (a) to preclude MSO of drywell spray valves; and (b) to preclude MSO of the main steam line drain valves. The licensee stated that modification (a) is complete, and modification (b) is in progress.

In an RAI, the licensee was requested to address the Appendix R fire induced failure of associated circuits that could result in a loss of containment integrity due to containment venting. The licensee was also requested to provide the results of safe shutdown analysis showing that adequate NPSH will be available for the RHR and CS pumps under a loss of containment integrity due to fire-induced MSO, or provide justification that a loss of containment integrity cannot occur under an Appendix R Fire scenario. In its March 21, 2013, response to NRC Question 7a, the licensee stated that the Appendix R fire analysis included consideration of impact of MSOs that could result in loss of containment integrity due to containment venting. The licensee stated that MSO scenarios and combinations of MSO scenarios per guidelines of Reference 63 that could challenge Appendix R fire-required CAP, were precluded from occurring through modifications and configuration changes. Those scenarios and combinations of scenarios that were identified as not precluded from occurring were demonstrated by analysis to show that they will not challenge the required CAP. The licensee's evaluation identified the applicable generic MSOs in Appendix G of Reference 63, as well as plant specific MSOs. The licensee performed valve modifications and configuration changes in fuses to preclude Appendix R fire-induced MSOs from adversely affecting safe shutdown. The licensee also performed evaluation of containment penetrations identified in USAR Table 5.2-3a and validated that the potential diversion flow paths out of primary containment were properly addressed.

The NRC staff requested the licensee to address the applicable scenarios listed in Reference 63, Table G-1, under "Decay Heat Removal," and provide the NPSH margin results. Specifically for scenarios 4r, 4s, and 4t, the licensee was requested to discuss the potential fire-induced impact on CAP and possible NPSH loss which should be addressed. The licensee was also requested to justify why the remaining scenarios in Reference 63, Table G-1, listed for BWR-3 under "Decay Heat Removal," are not applicable to MNGP. In its March 21, 2013, response to NRC for Question 7b, the licensee stated that applicable scenarios to MNGP were considered and the NPSH margins are provided in Table 6.6.4-1 of Reference 23. By cross-referencing scenarios 4r, 4s, and 4t with the cases analyzed and listed in Table 6.6.4-1 of Reference 51, the licensee provided additional details for these scenarios. The licensee identified the following generic MSOs that were applicable to BWR-3, but not applicable to MNGP: (a) 4l and 4m are not applicable because high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) are not credited for post Appendix R fire operation; (b) scenarios 4n, 4o, 4p, and 4q are not applicable because MNGP does not have an isolation condenser; and (c) scenario 4w is not applicable because the MNGP design basis assumes that all pumps aligned to the suppression pool start simultaneously.

The NRC staff accepts that the licensee evaluated the Appendix R fire-induced MSOs that could result in a loss of containment integrity due to containment venting and provided the most limiting combination of postulated vent paths or other system spurious operation in Reference 51. The NRC staff considers the licensee's approach for meeting the guidance in Section 6.6.4 of Reference 52 acceptable because the licensee's considered loss of containment integrity due to Appendix R fire-induced MSOs and demonstrated that adequate NPSHa for the RHR pump is available with spurious operation of up to four components caused by an Appendix R fire scenario.

ATWS Analysis for NPSHa

The licensee used May-Witt decay heat model for the EPU analysis, same as used in the current ATWS containment analysis. The licensee used nominal values of inputs to obtain realistic transients for wetwell pressure and suppression pool temperature. The licensee analyzed four cases for an ATWS event which are: (a) ATWS/Pressure Regulator Failed Open (PRFO) Case -1; (b) ATWS/PRFO Case-2; (c) ATWS/ loss of offsite power (LOOP); and (d) ATWS/LOOP/MELLLA+. The results of cases (a), (b), and (c) are reported in Tables 2.6-6, 2.6-7 and 2.6-8 of Reference 63. The results of case (d) are reported in Table 6.6.2-1 of Reference 49. ATWS cases (a), (b) and (c) are under EPU conditions only, whereas ATWS case (d) is under EPU MELLLA+ conditions. For ATWS case (a), the licensee considered one CS pump operating at its design flow and two RHR loops (two pumps in each loop) operating at their design flow in the direct suppression pool cooling mode. In case (b), the licensee considered one CS pump operating at its design flow, one RHR loop (two pumps) operating at design flow in the direct suppression pool cooling mode, and the other RHR loop (two pumps) operating at design flow in the containment spray mode. In case (c), the licensee considered one CS pump operating at its design flow and one RHR loop (two pumps) operating at design flow in the containment spray mode. In case (d), the licensee considered one CS pump and four RHR pumps at a higher flow which includes the flow through the minimum flow line assuming a fail-open minimum flow valve.

The licensee stated that there is no current NPSHa analysis for PRFO/ATWS cases. For the EPU 'PRFO/ATWS' NPSHa analysis cases, the licensee stated that the drywell temperature does not exceed 281°F, and all RHR loops are available. The analysis assumes operation of both RHR loops in the direct suppression pool cooling mode for containment cooling. The licensee used a temperature dependent RHR heat exchanger K-value for calculating the containment response.

The licensee stated that there is no current NPSHa analysis for LOOP/ATWS case. For the EPU 'LOOP/ATWS' NPSHa analysis, the licensee stated that the drywell temperature exceeds 281°F after 15 minutes from the initiation of the event. The analysis assumes containment cooling begins with RHR operating in the suppression pool cooling mode starting at 10 minutes from the event, and switches to drywell spray cooling when the drywell temperature exceeds 281°F. For the suppression pool cooling mode analysis, the licensee used a constant RHR heat exchanger K-value of 149.46 Btu/sec-°F which is based on a suppression pool temperature (hot fluid inlet temperature) of 156°F at 10 minutes from the event. When the suppression pool cooling mode is switched to drywell spray mode at 15 minutes from the event, the licensee used a constant K-value of 150.0 Btu/sec-°F which is based on a suppression pool temperature (hot fluid inlet temperature) of 165.5°F when drywell spray is initiated.

The licensee provided response to NRC staff guidance in Section 6.6.2 of Reference 52 for the NPSHa analysis for the ATWS event concurrent with LOOP under EPU conditions in Reference 49. The licensee's proposed evaluation in Reference 49 is based on the containment response under MELLLA+ conditions. The licensee stated that an ATWS is the only event under MELLLA+ conditions that affects the NPSHa, and is limiting compared to EPU ATWS events (a), (b), and (c) mentioned above.

The licensee's calculated suppression pool temperature response used for its NPSHa evaluation has a peak temperature of 202.8°F (Reference 49) for the ATWS event.

Using the containment analysis results, the licensee calculated the required wetwell pressure profile that meets the NPSHr3% at the pump inlet. ATWS/LOOP/MELLLA+ and is the most limiting event under EPU conditions.
The NRC staff considers licensee's ATWS NPSHa analysis results acceptable because the licensee showed margin between the available and the required wetwell pressure to meet NPSHr3% for the most limiting ATWS event while following the NRC staff guidance in Reference 52.

SBO Analysis for NPSHa

The SBO scenario and NPSHa containment analysis description was provided by the licensee in its August 21, 2009, response to RAI-3. The containment cooling is not available for the entire coping period of the SBO event which is therefore assumed in the SBO NPSHa. In the EPU analysis, the licensee used ANS 5.1-1979 decay heat model guidance of SIL 636, Revision 1, and nominal values of input parameters to obtain a realistic suppression pool temperature and wetwell pressure transients. One HPCI system loop operates at its design flow and injects makeup water into the RV. The licensee stated that for EPU, it has upgraded the SBO NPSHa analysis model from its current model by using the SHEX code modeling the HPCI pump suction transfer from the condensate storage tank to suppression pool on receiving a transfer signal based on temperature. The current analysis uses the 1995 version of Modular Accident Analysis Program (MAAP) code in which the HPCI pump suction source is suppression pool only because the code capability did not allow dynamic modeling of the suction transfer. The licensee modeled the initiation of a SBO event, in the SHEX code, with the HPCI pump suction from the CST assuming at its maximum water temperature of 135°F. The suction transfer within the model takes place when the suppression pool temperature reaches 135°F in about one hour. The model then assumes HPCI suction from the suppression pool for three hours. The licensee stated that all of the design basis HPCI cycles are complete within three hours with no credit for a manual suction transfer, and with the HPCI suction from the source with the most limiting temperature.

The licensee's calculated suppression pool temperature response used for its NPSHa evaluation has a peak temperature of 154.7°F (Reference 1) for the SBO event.

Table 2.6-5 of Reference 32 provides the results of the NPSHa, and its comparison with NPSHr3%, for the HPCI pump with suppression pool as its suction source. The results show that CAP credit is not used for adequate NPSHa at HPCI pump suction. The NRC staff considers the licensee's proposed EPU analysis and results acceptable because: (a) the licensee followed the NRC staff guidance in Reference 52; (b) the proposed analysis model is more representative of the actual plant response to an SBO event than the current MAAP model; and (c) the results show there is sufficient margin between the NPSHa and the NPSHr3% without CAP credit.

RHR and CS Pump Suction Strainers Head Loss

An important item in the evaluation of the NPSHa is the head loss in the pump suction strainers. A higher head loss negatively affects the NPSHa at the pump suction inlet. The licensee has installed new strainers to the suction lines in the suppression pool. Each strainer has two modules; each module is a 40-inch nominal diameter, convoluted cylindrical strainer approximately seven feet long and having approximately 40 percent open area. In the current licensing basis, the licensee evaluated suction strainer debris loading using NRC-approved guidance in Reference 46, and the 85 pounds of protective coating which is a bounding value. The licensee stated that the proposed EPU does not affect the volume of debris generated in a postulated DBA LOCA because the existing evaluation is based on assumptions that are bounding for EPU conditions. Therefore the ECCS pump head losses due to debris accumulation on their suppression pool suction strainer are not affected under EPU conditions.

Conclusion

The NRC staff has reviewed the licensee's assessment of those systems required to support containment heat removal and concludes that the licensee adequately addressed the effects of the proposed EPU. The NRC staff finds that the licensee has implemented the guidance for using CAP credit in Reference 52 for the assessment of NPSHa and NPSHreff for the ECCS and containment heat removal pumps. The NRC staff concludes that the ECCS and containment heat removal systems will continue to meet GDC-38, with respect to rapidly reducing the containment pressure and temperature following the design basis and non-design basis events and maintaining these parameters at acceptably low levels. Therefore, the NRC staff concludes that the proposed operation under EPU conditions in MELLLA+ domain acceptable with respect to containment heat removal systems.

2.6.6 Secondary Containment Functional Design

Regulatory Evaluation

The secondary containment structure and supporting systems are provided to collect and process radioactive material that may leak from the primary containment following an accident. The supporting systems maintain a negative pressure within the secondary containment and process this leakage. The NRC staff's review covered the following: (1) analyses of the pressure and temperature response of the secondary containment following accidents within the primary and secondary containments; (2) analyses of the effects of openings in the secondary containment on the capability of the depressurization and filtration system to establish a negative pressure in a prescribed time; (3) analyses of any primary containment leakage paths that bypass the secondary containment; (4) analyses of the pressure response of the secondary containment resulting from inadvertent depressurization of the primary containment when there is vacuum relief from the secondary containment; and (5) the acceptability of the mass and energy release data used in the analysis. The review primarily focused on the effects that the proposed EPU may have on the pressure and temperature response and drawdown time of the secondary containment and the impact this may have on offsite dose. The NRC staff acceptance criteria for the secondary containment functional design are based on 10 CFR 50, Appendix A:

GDC-4, "Environmental and dynamic effects design bases," insofar as it requires that SSCs important to safety be designed to accommodate the effects of environmental conditions associated with normal operation, maintenance, testing, and postulated accidents and be protected from dynamic effects (e.g., the effects of missiles, pipe whipping, and discharging fluids) that may result from equipment failures.

GDC-16, "Containment design," insofar as it requires that reactor containment and associated systems be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment.

SRP Section 6.2.3 contains specific review criteria.

While MNGP is not explicitly licensed to the current General Design Criteria GDC or the 1967 AEC-proposed GDC, the licensee has made a comparison of the current GDC to the applicable AEC-proposed General Design Criteria. For the current GDC-4 and GDC-16, the licensee's evaluation of the analogous 1967 AEC-proposed GDC is also contained in MNGP USAR,

Appendix E: draft GDC-10, draft GDC-40, and draft GDC-42.

Technical Evaluation

An increase in RTP increases the heat load on the secondary containment and may affect the drawdown time of the secondary containment. The drawdown time is the time period following the start of the accident during which loss of offsite power causes loss of secondary containment vacuum (relative to atmospheric pressure) which is assumed to result in releases from the primary containment directly to the environment without filtering.

In an RAI, the NRC staff requested that the licensee provide an evaluation of the effect of the EPU on the secondary containment drawdown time and dose evaluation. In its July 13, 2009, response to RAI-18, the licensee stated that in the analysis of record for the Alternate Source Term (AST), the estimated drawdown time for the positive pressure period was 5 minutes during which the radionuclide removal from standby gas treatment system (SGTS) operation is not credited. The licensee's drawdown calculation using GOTHIC code and a single lumped node determined that the positive pressure period was less than 2-minutes. Under the EPU conditions, the licensee determined that the decrease in air density due to slight increase in the reactor building temperature is not significant and does not affect the calculated drawdown time, and would not approach the 5 minute positive pressure assumption used in the AST analysis. The licensee concluded that the EPU does not affect the secondary containment drawdown time and the AST dose analysis. Based on the above, the NRC staff considers the licensee's evaluation of the effect of EPU on the drawdown time to be acceptable.

The licensee stated that the current design flow capacity of the SGTS maintains the secondary containment at the required negative pressure to minimize the potential for ex-filtration of air from the reactor building during an accident. This capability is unaffected by EPU because the primary and secondary leak rates are not affected by the EPU, and the HEPA filters have sufficient design margin to accommodate additional fission product loading without restricting flow rate.

Conclusion

The NRC staff has reviewed the licensee's assessment related to the secondary containment pressure and temperature transient and the ability of the secondary containment to provide an essentially leak-tight barrier against uncontrolled release of radioactivity to the environment. The staff concludes that the licensee has adequately accounted for the increase of mass and energy that would result from the proposed EPU and further concludes that the secondary containment and associated systems will continue to provide an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment following implementation of the proposed EPU. Based on this, the staff also concludes that the secondary containment and associated systems will continue to meet the requirements of GDC-4 and -16. Therefore, the NRC staff finds the proposed EPU acceptable with respect to secondary containment functional design.

2.6.7 Additional Review Items (Containment Review Considerations)

2.6.7.1 Containment Isolation

Regulatory Evaluation

The NRC staff acceptance criteria for the containment isolation are based on 10 CFR 50, Appendix A:

GDC-50, "Containment design basis," insofar as the containment structure, including penetrations, shall be designed to accommodate, without exceeding design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any LOCA.

Technical Evaluation

The licensee reviewed the containment isolation portions of the systems penetrating the primary containment and determined that EPU does not affect the containment isolation devices and the capability to isolate the primary containment during normal or accident conditions. However EPU has resulted in changes in temperature response both in the drywell and wetwell. The NRC staff requested that the licensee explain the following: (a) why pipe penetration integrity of water filled isolated piping that is susceptible to thermally induced over-pressurization is unaffected by the EPU; and (b) why the higher temperatures under EPU conditions will not affect the calculated leakage pressure through the valve bonnet gaskets and discs for each of the penetrations. In its July 13, 2009, response to RAI-14, the licensee noted 13 lines containing air or water that penetrate the primary containment were in the scope of equipment subject to response to GL 96-06. The licensee stated that no new air or water lines have since been routed through primary containment, thus the potentially affected lines within the scope of GL 96-06 do not change at EPU conditions. The licensee evaluated each of the 13 lines and determined that the EPU does not impact these lines and the response to GL 96-06.

The licensee also discussed the effects of EPU on its motor-operated containment isolation valve programs related to GL 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," and GL 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves," in Section 2.2.4 of Reference 1, and letter dated January 21, 2013.

Conclusion

Based on the above, the NRC staff finds that the EPU does not adversely affect system designs for containment isolation capabilities, which continue to meet the requirements of GDC-50. Therefore, the NRC staff finds the proposed EPU acceptable with respect to containment isolation.

2.6.7.2 Hardened Vent

Regulatory Evaluation

Generic Letter 89-16, "Installation of a Hardened Wetwell Vent," discusses the advantages of installing a hardened containment (wetwell) vent and requested information from licensees on installation of such a vent. This was a result of the NRC's BWR Mark-I Containment

Performance Improvement Program.

Technical Evaluation

Section 5.2.1.5 of the MNGP USAR (Revision 24) states that the hardened vent design criterion is to prevent containment pressure from increasing under conditions of constant heat input at a rate equal to 1 percent of RTP, and containment pressure equal to its primary containment pressure limit. The licensee stated that the current design of the MNGP hardened wetwell vent was based on the 1670 MWth OLTP. The licensee stated that the as-built design of the wetwell vent will exhaust approximately 1.05 percent of the EPU RTP of 2004 MWth. The hardened wetwell vent is designed to be operational during a SBO event.

Conclusion

The existing MNGP hardened wetwell vent meets the intent of GL 89-16 for EPU conditions. The NRC issued Order EA-13-109 on June 6, 2013, requiring BWR Mark I and Mark II containment types to install a severe accident capable, reliable hardened vent. As a result, the licensee is expected to either modify the existing vent or install a new vent from the wetwell to comply with the Order. The Order requires that the severe accident capable, reliable hardened vent, be operational no later than startup from the second refueling outage that beings after June 30, 2014, or June 30, 2018, whichever comes first.

Based on the above, the NRC staff finds that the existing MNGP hardened wetwell vent meets the intent of GL 89-16 for EPU conditions.

2.7 Habitability, Filtration, and Ventilation

2.7.1 Control Room Habitability System

Regulatory Evaluation

The NRC staff reviewed the control room habitability system and control building layout and structures to ensure that plant operators are adequately protected from the effects of accidental releases of toxic and radioactive gases. Another objective of the NRC staff's review was to ensure that the control room can be maintained as the backup center from which technical support center personnel can safely operate in the case of an accident. The NRC staff's review focused on the effects of the proposed EPU on radiation doses, toxic gas concentrations, and estimates of dispersion of airborne contamination. The NRC's acceptance criteria for the control room habitability system are based on: (1) GDC-4, "Environmental and dynamic effects design bases," insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with postulated accidents, including the effects of the release of toxic gases; and (2) GDC-19, "Control room," insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 roentgen equivalent man (rem)) total effective dose equivalent (TEDE) as defined in 10 CFR 50.2, "Definitions," for the duration of the accident. SRP Section 6.4 and other guidance in Matrix 7 of "Review Standard for Extended Power Uprates," RS-001, Rev. 0, contain specific review criteria.

The GDCs discussed herein are those currently specified in 10 CFR Part 50, Appendix A. The applicable MNGP Principal Design Criteria predate these Appendix A criteria. These MNGP Principal Design Criteria are listed in MNGP USAR Section 1.2, "Principal Design Criteria." In 1967, the Atomic Energy Commission (AEC) published for public comment a revised set of proposed General Design Criteria (32 *FR* 10213, July 11, 1967). An evaluation comparing the MNGP design basis to the AEC-proposed General Design Criteria of 1967 is presented in MNGP Updated Safety Analysis Report (USAR), Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria." While MNGP is not explicitly licensed to the current GDC or the 1967 AEC-proposed General Design Criteria, the licensee has made a comparison of the current GDC to the applicable AEC-proposed General Design Criteria. For the current GDC-4 the licensee's evaluation of the analogous 1967 AEC-proposed General Design Criteria is also contained in MNGP USAR, Appendix E: draft GDC-11, 40, and 42. Current GDC-19 is applicable to MNGP as described in USAR Section 5.3.5, 6.7.3, 12.3.1.6, and 14.7.

Technical Evaluation

The main control room (MCR) area, the EFT Building, and the technical support center (TSC) ventilation system is addressed in 2.7.3 of this report. As described in USAR, Revision 25, Section 6.7, under high radiation emergency conditions, a filtration train will start to provide filtered outside air to the MCR and portions of the EFT building. Section 2.7.1.1 of Enclosure 5 of the licensee's application describes this as control room emergency filtration (CREF) system which provides a radiologically-controlled environment from which the plant can be operated safety following a design-basis accident (DBA). For the area which the CREF system serves, it isolates the unfiltered air intake, and pressurizes the area with filtered air during a DBA LOCA.

The licensee evaluated the EPU effect on the CREF due to an increase in the core iodine activity released during the DBAs using RG 1.3 (derived from technical information document (TID)-14844 "Calculation of Distance Factors for Power and Test Reactor Sites"), "Assumptions used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors," source term.

The licensee used RG 1.183 for control room personnel dose evaluation, which has been approved for MNGP by the NRC via Amendment No. 148 (Reference 84) and a letter dated April 17, 2007 (Reference 85), and adjusted the source term for 102-percent of the EPU power of 2004 MWth. The licensee accounted for the increase in primary containment leak rate due to increased drywell pressure under EPU conditions. In addition, the licensee considered the EPU increase in main steam piping temperature which results in a decrease in radionuclide deposition within the piping and main condenser.

The licensee's evaluation concluded that the EPU post-accident increase in iodine loading of the filtration system remains less than half the allowable limit of RG 1.52, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants." The licensee also evaluated the effect of radiolytic heating of the filters and concluded that the filter temperature stays within its design requirement. Therefore, the licensee stated that EPU does not affect the iodine filtration efficiency. The NRC staff agrees with licensee's evaluation.

In 2004, the licensee evaluated control room habitability in response to NRC GL 2003-01, "Control Room Habitability." The licensee stated that operation at EPU does not introduce any new toxic chemicals sources and, therefore, the analysis for toxic chemicals at EPU operation does not change from the current licensing basis. The licensee also stated that operator actions in response to a toxic chemical release at EPU will not change from the current licensing basis. The NRC staff agrees that EPU operation does not affect control room habitability during a toxic chemical release event.

Conclusion

The NRC staff has reviewed the licensee's assessment related to the effects of the proposed EPU on the ability of the control room habitability system to protect plant operators against the effects of accidental releases of toxic and radioactive gases. The NRC staff concludes that the licensee has adequately accounted for the increase of toxic and radioactive gases that would result from the proposed EPU. The NRC staff also concludes that the control room habitability system will continue to provide the required protection following implementation of the proposed EPU insofar as equipment operability is concerned. The review of EPU control room dose assessment analysis can be found at Section 2.9.2 of this safety evaluation. Based on this, the NRC staff concludes that the control room habitability system will continue to meet the requirements of GDCs 4 and 19. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the control room habitability system.

2.7.2 Engineered Safety Feature Atmosphere Cleanup

Regulatory Evaluation

Engineered Safety Feature (ESF) atmosphere cleanup systems are designed for fission product removal in post-accident environments. These systems generally include primary systems (e.g., in-containment recirculation) and secondary systems (e.g., SGTS and emergency or post-accident air cleaning systems) for the fuel handling building, control room, shield building, and areas containing ESF components. For each ESF atmosphere cleanup system, the NRC staff's review focused on the effects of the proposed EPU on system functional design, environmental design, and provisions to preclude temperatures in the adsorber section from exceeding design limits. The NRC's acceptance criteria for ESF atmosphere cleanup systems are based on: (1) GDC-19, "Control room," insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess 0.05 sievert (Sv) (5 rem [roentgen equivalent man]) TEDE as defined in 10 CFR 50.2, for the duration of the accident; (2) GDC-41, "Containment atmosphere clean-up," insofar as it requires that systems to control fission products released into the reactor containment be provided to reduce the concentration and quality of fission products released to the environment following postulated accidents; (3) GDC-61, "Fuel storage and handling and radioactive control," insofar as it requires that systems that may contain radioactivity be designed to assure adequate safety under normal and postulated accident conditions; and (4) GDC-64, "Monitoring radioactivity releases," insofar as it requires that means be provided for monitoring effluent discharge paths and the plant environs for radioactivity that may be released from normal operations, including AOOs and postulated accidents. Specific review criteria are contained in SRP Section 6.5.1.

The GDCs discussed herein are those currently specified in 10 CFR Part 50, Appendix A. The applicable MNGP Principal Design Criteria predate the Appendix A criteria. The MNGP Principal Design Criteria are listed in MNGP USAR Section 1.2, "Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (32 *FR* 10213, July 11, 1967). An evaluation comparing the MNGP design basis to the AEC-proposed General Design Criteria of 1967 is presented in MNGP USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria." While MNGP is not explicitly licensed to the current GDCs or the 1967 AEC-proposed General Design Criteria, the licensee has made a comparison of the current GDC to the applicable AEC-proposed GDC. For the current GDC-61 and 64, the licensee's evaluation of the analogous 1967 AEC-proposed General Design Criteria is also contained in MNGP USAR, Appendix E: draft GDC-11, 17, 67, 68, and 69. The intent of current GDC-41 is described in USAR Section 5.3.4.1. Current GDC-19 is applicable to MNGP as described in USAR Section 5.3.5, 6.7.3, 12.3.1.6, and 14.7.

Technical Evaluation

The SGTS is the ESF atmosphere cleanup system that provides fission product control during DBA conditions. Sections 2.5.2.1, 2.6.6, and 2.9 of Enclosure 5 of the licensee's November 5, 2008, application present the licensee's SGTS evaluation. The NRC staff evaluation of the SGTS is given in Section 2.6.6 "Secondary Containment Functional Design" and Section 2.9.2 "Radiological Consequences Analyses Using Alternative Source Terms" of this safety evaluation.

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the ESF atmosphere cleanup systems. The NRC staff concludes that the licensee has accounted for the increase of fission products and changes in expected environmental conditions that would result from the proposed EPU, and the NRC staff also finds that the ESF atmosphere cleanup systems will continue to provide fission product removal in post-accident environments following implementation of the proposed EPU insofar as the equipment operability is concerned. Based on this, the NRC staff concludes that the ESF atmosphere cleanup systems will continue to meet the requirements of GDC-19, 41, 61, and 64. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the ESF atmosphere cleanup systems.

2.7.3 Control Room Area Ventilation System

Regulatory Evaluation

The function of the control room area ventilation system (CRAVS) is to provide a controlled environment for the comfort and safety of control room personnel and to support the operability of control room components during normal operation, AOOs, and DBA conditions. The NRC's review of the CRAVS focused on the effects that the proposed EPU will have on the functional performance of safety-related portions of the system. The review included the effects of radiation, combustion, and other toxic products and the expected environmental conditions in areas served by the CRAVS. The NRC's acceptance criteria for the CRAVS are based on: (1) GDC-4, "Environmental and dynamic effects design bases," insofar as it requires that 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; (2) GDC-19, "Control room," insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) TEDE as defined in 10 CFR 50.2, for the duration of the accident; and (3) GDC-60, "Control of release of radioactive materials to the environment," insofar as it requires that the plant design include means to control the release of radioactive effluents. SRP Section 9.4.1 contains specific review criteria.

The general design criteria discussed herein are those currently specified in 10 CFR Part 50, Appendix A. The applicable MNGP Principal Design Criteria predate these Appendix A criteria. These MNGP Principal Design Criteria are listed in MNGP USAR Section 1.2, "Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (32 FR 10213, July 11, 1967). An evaluation comparing the MNGP design basis to the AEC-proposed General Design Criteria of 1967 is presented in MNGP USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria." While MNGP is not explicitly licensed to the current GDC or the 1967 AEC-proposed General Design Criteria, the licensee has made a comparison of the current GDC to the applicable AEC-proposed General Design Criteria. For the current GDC-4 and 60, the licensee's evaluation of the analogous 1967 AEC-proposed General Design Criteria is also contained in MNGP USAR, Appendix E: draft GDC-11, 40, 42, and 70. The current GDC-19 is applicable to MNGP as described in USAR Section 5.3.5, 6.7.3, 12.3.1.6, and 14.7.

Technical Evaluation

The CRAVS provides ventilation to the MCR and portions of the EFT building. The heat sources for these areas include equipment, lights, ambient outside air temperature. Heat loads from these sources do not change for the EPU. As the heat loads do not change for the EPU, the existing control room area cooling system remains adequate to control the temperature. Section 2.7.1 of this document addresses the effects of radioactive gases.

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the ability of the CRAVS to provide a controlled environment for the comfort and safety of control room personnel and to support the operability of control room components. The NRC staff concludes that the licensee has adequately accounted for the increase of toxic and radioactive gases that would result from a DBA under the conditions of the proposed EPU and associated changes to parameters affecting environmental conditions for control room personnel and equipment. Accordingly, the NRC staff concludes that the CRAVS will continue to provide an acceptable control room environment for safe operation of MNGP following implementation of the proposed EPU. Based on this, the NRC staff concludes that the CRAVS will continue to meet the requirements of GDC-4, 19, and 60. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the CRAVS.

2.7.4 Spent Fuel Pool Area Ventilation System

Regulatory Evaluation

The function of the spent fuel pool area ventilation system (SFPAVS) is to maintain ventilation in the SFP equipment areas, permit personnel access, and control airborne radioactivity in the area during normal operation, AOOs, and following postulated fuel-handling accidents. The NRC staff's review focused on the effects of the proposed EPU on the functional performance of the safety-related portions of the system.

The NRC's acceptance criteria for the SFPAVS are based on: (1) GDC-60, "Control of releases of radioactive materials to the environment," insofar as it requires that the plant design include means to control the release of radioactive effluents; and (2) GDC-61, "Prevention of criticality in fuel storage and handling," insofar as it requires that systems that contain radioactivity be designed with appropriate confinement and containment.

SRP Section 9.4.2 contains specific review criteria.

Technical Evaluation

As indicated in Section 2.7.4 of Enclosure 5 of the licensee's November 3, 2008, application, MNGP does not have a separate SFPAVS. During normal conditions, the SFP area is ventilated by the reactor building heating, ventilation, and air conditioning (HVAC) system, and when required, the SGTS maintains ventilation for this area. In its July 13, 2009, letter, the licensee states that although normal SFP EPU decay heat loads are higher than the current licensing basis heat loads, the existing FPCCS has the capacity to maintain the SFP below the design limit of 140°F. Therefore there is no change to the reactor building HVAC system design heat load due to the SFP area as a result of the EPU.

Conclusion

MNGP does not have a separate SFPAVS. The NRC staff's concludes that the reactor building HVAC system, which provides ventilation and cooling of the SFP area during normal operation, will maintain the area within design limits. The FPCCS is evaluated in Section 2.5.3.1 of this safety evaluation.

2.7.5 Primary Containment, Radwaste Area and Turbine Area Ventilation Systems

Regulatory Evaluation

The function of the radwaste area ventilation system (RAVS) and the turbine area ventilation system (TAVS) is to maintain ventilation in the auxiliary and radwaste equipment and turbine areas, permit personnel access, and control the concentration of airborne radioactive material in these areas during normal operation, during AOOs, and after postulated accidents. The NRC staff's review focused on the effects of the proposed EPU on the functional performance of the safety-related portions of these systems. The NRC's acceptance criteria for the RAVS and TAVS are based on GDC-60, "Control of releases of radioactive materials to the environment," insofar as it requires that the plant design include the means to control the release of radioactive effluents. SRP Sections 9.4.3 and 9.4.4 contain specific review criteria.

The general design criteria discussed herein are those currently specified in 10 CFR Part 50, Appendix A. The applicable MNGP Principal Design Criteria predate these Appendix A criteria. These MNGP Principal Design Criteria are listed in MNGP USAR Section 1.2, "Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (32 *FR* 10213, July 11, 1967). An evaluation comparing the MNGP design basis to the AEC-proposed General Design Criteria of 1967 is presented in MNGP USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria."

While MNGP is not explicitly licensed to the current General Design Criteria or the 1967 AEC-proposed GDC, the licensee has made a comparison of the current GDC to the applicable AEC-proposed GDC. For the current GDC-60, the licensee's evaluation of the analogous 1967 AEC-proposed General Design Criteria is also contained in MNGP USAR, Appendix E: draft

GDC-70.

Technical Evaluation

The RAVS and TAVS consist mainly of heating, cooling, supply, exhaust, and recirculation units serving the radwaste and turbine buildings, respectively. The EPU results in slightly higher process temperatures and a small increase in the heat load because of higher electrical power requirements for some motors.

The areas affected by the EPU in the turbine building are feedwater and condensate pump areas and associated switchgear. The HVAC systems that provide cooling to these areas are affected. Other areas of turbine building and radwaste building are unaffected by the EPU because the process temperatures remain relatively constant.

Regarding the turbine building ventilation system, the licensee states that this system will be affected because the modification of the feedwater and condensate pump motors is necessary for EPU operation. In its July 13, 2009, letter, the licensee states that the condensate pump motors and feedwater pump motors are being replaced due to EPU. The motor selection for this modification which may affect the heat load is currently not completed. The licensee has committed to evaluate the changes in heat load upon completion of the final designs, including motor selection and the verification that the condensate pump and the reactor feedwater pump area temperatures remain within design limits of less than 130°F and 104°F, respectively. In case these design limits are exceeded, the licensee has committed to modify the TAVS in order to meet the design limits.

In its July 13, 2009, letter, the licensee stated that the effect of EPU on the drywell atmosphere heat loads was evaluated and was found to increase by 0.26 percent, which is insignificant. The primary increase in the drywell heat loads is due to feedwater piping and the biological shield gamma heating. The NRC staff agrees with the licensee's evaluation that the existing primary containment cooling system will be able to maintain the containment atmosphere below its design limit of 135°F bulk average.

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the RAVS, TAVS, and the primary containment cooling system. The NRC staff agrees with the licensee's conclusion that the containment and radwaste building will have higher heat loads which are insignificant and that the HVAC system will not be affected. The turbine building HVAC is affected and will be evaluated by the licensee in more detail after the modification of the feedwater and condensate pump motors is confirmed. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed EPU on the capability of reactor building and radwaste building HVAC systems to maintain ventilation in the respective areas and will continue to meet the requirements of GDC-60. For the turbine building, prior to EPU implementation, the licensee will provide for NRC staff's review and approval the evaluation of the revised heat loads and the necessary modification to the TAVS for meeting the room temperature design limits.

2.7.6 Engineered Safety Feature Ventilation System

Regulatory Evaluation

The function of the engineered safety feature ventilation system (ESFVS) is to provide a suitable and controlled environment for ESF components following certain anticipated transients and DBAs. The NRC staff's review of the ESFVS focused on the effects of the proposed EPU on the functional performance of the safety-related portions of the system. The NRC staff's review also covered: (1) the ability of the ESF equipment in the areas being serviced by the ventilation system to function under degraded ESFVS performance; (2) the capability of the ESFVS to circulate sufficient air to prevent accumulation of flammable or explosive gas or fuel-vapor mixtures from components (e.g., storage batteries and stored fuel); and (3) the capability of the ESFVS to control airborne particulate material (dust) accumulation. The NRC's acceptance criteria for the ESFVS are based on: (1) GDC-4, "Environmental and dynamic effects design bases," insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents; (2) GDC-17, "Electric power systems," insofar as it requires that onsite and offsite electric power systems be provided to permit functioning of SSCs important to safety; and (3) GDC-60, "Control of releases of radioactive materials to the environment," insofar as it requires that the plant design include the means to control the release of radioactive effluents. The specific review criteria are contained in SRP Section 9.4.5.

The general design criteria discussed herein are those currently specified in 10 CFR Part 50, Appendix A. The applicable MNGP Principal Design Criteria predate these Appendix A criteria. The MNGP GDCs are listed in MNGP USAR Section 1.2, "Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (32 *FR* 10213, July 11, 1967). An evaluation comparing the MNGP design basis to the AEC-proposed General Design Criteria of 1967 is presented in MNGP USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria."

While MNGP is not explicitly licensed to the current General Design Criteria or the 1967 AEC-proposed GDC, the licensee has made a comparison of the current GDC to the applicable AEC-proposed GDC. For the current GDCs 4, 17, and 60, the licensee's evaluation of the analogous 1967 AEC-proposed General Design Criteria is also contained in MNGP USAR, Appendix E: draft GDCs 24, 39, 40, 42, and 70.

Technical Evaluation

The ESF HVAC systems consist mainly of heating, cooling, supply, exhaust, and recirculation units serving the HPCI, RHR and CS pump rooms in the reactor building, control room, and the diesel generator building, and the SGTS. The affected areas in the reactor building due to EPU are steam tunnel, HPCI room, RHR and CS pumps rooms. In its July 13, 2009, letter, the licensee states that the steam tunnel area temperature is expected to increase by less than one degree Fahrenheit. In the same letter the licensee states that the higher heat loads in the HPCI, RHR and CS pump rooms is due to higher suppression pool water temperature carried in the system piping. In its January 21, 2013, letter, the licensee states that the results of an evaluation of increased RHR and CS pump room temperature during system operation were revised to a maximum of 2.9°F, which does not exceed the design temperature for this area. For the HPCI room, the licensee states that its temperature is expected to remain within design limit without taking credit for HVAC operation. In the same letter the licensee listed the conservatisms in the current licensing basis HPCI room temperature calculation. The NRC staff accepts the licensee's explanations and agrees that the HPCI room temperature will remain within its design limits because of sufficient conservatism in the calculation which is even more conservative because the HVAC system operation is not credited.

The control room and diesel generator building HVAC are not affected because the heat load in these areas is not affected by the EPU.

The SGTS is one of the fission product control systems and structures that provide fission products control during DBA conditions. Refer to Section 2.7.2 for the SGTS evaluation.

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the ESFVS. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed EPU on the ability of the ESFVS to provide a suitable and controlled environment for ESF components. The NRC staff further concludes that the ESFVS will continue to assure a suitable environment for the ESF components following implementation of the proposed EPU. The NRC staff also concludes that the ESFVS will continue to suitably control the release of gaseous radioactive effluents to the environment following implementation of the proposed EPU. Based on these findings, the NRC staff concludes that the ESFVS will continue to meet the requirements of GDC-4, 17, and 60. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the ESFVS.

2.8 Reactor Systems

The NRC staff's review largely follows the guidance contained in Nuclear Regulatory Commission Review Standard RS-001, "Review Standard for EPUs."

Similarly, the licensee's technical basis supporting the power uprate request NEDC-33322P, Revision 3, "Safety Analysis Report for Monticello Constant Pressure Power Uprate," is formatted consistent with RS-001. The licensee and the NRC staff refer to this report as the PUSAR and it is provided as Enclosure 5 of the November 5, 2008, application for amendment.

The NRC staff's review of reactor-core-related technical areas for the MNGP EPU application is based on either generic assessment or the licensee's plant-specific evaluation, as noted in each subsection (below) under Section 2.8.

The PUSAR is based on NEDC-33004P-A, "Licensing Topical Report: Constant Pressure Power Uprate" (CPPU) (CLTR), which is an NRC-approved LTR describing the generic and plant-specific evaluations that support boiling water reactor (BWR) power uprates. The material contained in NEDC-33004P-A (Reference 8) is based on two previously approved LTRs that had been used to support BWR EPUs: (1) NEDC-32424P-A, "Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate;" and (2) NEDC-32523P-A, "Generic Evaluations of General Electric Boiling Water Reactor Extended Power Uprate." These three LTRs are referenced heavily by the licensee in its application, and a large portion of the NRC staff's review effort focused on confirming the dispositions contained in these reports.

Additionally, GE obtained NRC approval for the interim use of its analytic methods for BWR power uprate analyses until GE improves its NRC-reviewed and approved experimental and operating data bases that support the analytic methods. The licensee obtained analytic support for its uprate from GE, and GE used the interim analytic approach as described in LTR NEDC-33173P-A, "Applicability of GE Methods to Expanded Operating Domains." The NRC staff reviewed the licensee's application to confirm that the analytic methods supporting the power uprate have been applied in a manner consistent with the conditions and limitations delineated in the safety evaluation approving NEDC 33173P-A. This confirmation is discussed in Section 2.8.7.1 below.

2.8.1 Fuel System Design

Regulatory Evaluation

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

1. The fuel system is not damaged as a result of normal operation and AOOs;

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

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

4. Coolability is always maintained.

The NRC staff's review covered fuel system damage mechanisms, limiting values for important parameters, and performance of the fuel system during normal operation, AOOs, and postulated accidents. The NRC's acceptance criteria are based on:

1. 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for ligh-water nuclear power reactors," insofar as it establishes standards for the calculation of emergency core cooling system (ECCS) performance and acceptance criteria for that calculated performance;

2. Draft GDC-6 and current GDC-10, "Reactor design," insofar as they require that the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to assure that specified acceptable fuel design limits (SAFDLs) are not exceeded during any condition of normal operation, including the effects of AOOs; and

3. Draft GDC-37, 41, and 44, insofar as they require that a system to provide abundant emergency core cooling be provided to prevent fuel damage following a LOCA.

Specific review criteria are contained in SRP Section 4.2 and guidance provided in Matrix 8 of RS-001.

Technical Evaluation

Under the auspices of CPPU, the disposition for the fuel system design at EPU conditions is generic, provided that the requesting licensee uses approved GEH fuel designs through the GE14 fuel product line.

NSPM stated that no new fuel products will be introduced to implement the EPU. The MNGP core has been comprised entirely of GE14 fuel assemblies since the start of Cycle 24 (currently in Cycle 27), and this will continue to be the case during implementation of the EPU (see NEDC-33322P, Reference 7). This information confirms the disposition regarding fuel system design for MNGP at EPU conditions, and the NRC staff finds this disposition acceptable.

The CLTR states that additional energy requirements are met by an increase in bundle enrichment, an increase in the reload fuel batch size, and/or changes in fuel loading pattern to maintain the desired plant operating cycle length (Reference 8). The licensee's confirmation of a relatively constant bundle average discharge burnup supports this disposition, because it is an indication that the operating cycle length does not change appreciably.

To confirm the licensee's statements, the NRC staff requested additional information about the MNGP fuel system design to compare pre-EPU and post-EPU fuel design. Specifically, the NRC staff requested that the licensee provide a comparison of fuel average discharge burnup levels (RAI-2.8.1-1) and fuel enrichment levels (RAI-2.8.1-2) from current licensed thermal power level to EPU power level.

In response to RAI-2.8.1-1, in its July 23, 2009, letter, NSPM stated that the current end-of-cycle average discharge exposure is predicted to be 44904.5 megawatt-days per metric ton (MWd/MT) (Reference 5). The predicted average discharge exposure for the EPU equilibrium cycle is predicted to be 45693.9 MWd/MT, a difference on the order of 2-percent. Both of these comply with the GE14-specific limit on discharge exposure of 50000.0 MWd/MT, and are hence acceptable with respect to the fuel design-specific limits on discharge burnup for GE14 fuel.

In response to RAI-2.8.1-2, in its July 23, 2009, letter, NSPM stated that the predicted Cycle 24 weighted average fresh bundle enrichment is predicted to be 3.92-percent and the predicted Cycle 25 weighted average fresh bundle enrichment is predicted to be 3.90-percent (Reference 5). While the EPU core weighted average fresh bundle enrichment is predicted to be 4.11-percent, the maximum licensed bundle enrichment for GE14 fuel is 5.0-percent.

While the EPU will require some modifications to the core design, the fuel design itself does not change for the uprate. The parameters provided by the licensee in response to the NRC staff's RAI-confirm that there is no significant or fundamental change to the fuel assembly design.

The NRC staff recognizes that, while there may be no fundamental change to the fuel design, the core loading, design and operation will change appreciably to allow for the loading of increased energy into the core. The core design changes are discussed in Section 2.8.2, "Nuclear Design."

<u>Conclusion</u>

The NRC staff has reviewed the licensee's disposition related to the effects of the proposed EPU on the design of the fuel assemblies, control systems, and reactor core. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed EPU on the fuel system and demonstrated that:

1. The fuel system will not be damaged as a result of normal operation and AOOs;

2. The fuel system damage will never be so severe as to prevent control rod insertion when it is required;

3. The number of fuel rod failures will not be underestimated for postulated accidents; and

4. Coolability will always be maintained.

These considerations are based, in large part, on the fact that the fuel design does not change for the EPU, and that the fuel design can withstand the MNGP-specific EPU operating conditions.

Based on these considerations, the NRC staff concludes that the fuel system and associated analyses will continue to meet the requirements of 10 CFR 50.46, current GDC-10, and draft GDC-6, 37, 41, and 44 following implementation of the proposed EPU.

2.8.2 Nuclear Design

Regulatory Evaluation

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

The NRC staff's review covered the following:

1. Core power distribution,

2. Reactivity coefficients,

3. Reactivity control requirements and control provisions,

4. Control rod patterns and reactivity worths,

5. Criticality,

6. Burnup, and

7. Vessel irradiation.

The NRC's acceptance criteria are based on the following:

1. Draft GDC-6 and current GDC-10, "Reactor design," insofar as they require that the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs;

2. Draft GDC-7 and current GDC-12, "Suppression of reactor power oscillations," insofar as they require that the reactor core be designed to ensure that power oscillations, which could cause damage in excess of acceptable fuel damage limits, are not possible or can be readily suppressed;

3. Draft GDC-8, insofar as it requires that the reactor core be designed so that the overall power coefficient in the power operating range shall not be positive;

4. Draft GDC-12, insofar as it requires that instrumentation and controls be provided as required to monitor and maintain variables within prescribed operating ranges;

5. Draft GDC-13, insofar as it requires that means shall be provided for monitoring and maintaining control over the fission process throughout core life and for all conditions that can reasonably be anticipated to cause variation in reactivity of the core, such as indication of position of control rods and concentration of soluble reactivity control poisons;

6. Draft GDC-14 and 15, insofar as they require that the protection system be designed to initiate the reactivity control systems automatically to prevent or suppress conditions that could result in exceeding acceptable fuel damage limits and to initiate operation of ESFs under accident situations;

7. Draft GDC-27 and 28, insofar as they require that at least two independent reactivity control systems be provided, with both systems capable of making and holding the core subcritical from any hot standby or hot operating condition sufficiently fast to prevent exceeding acceptable fuel damage limits;

8. Draft GDC-29, insofar as it requires that at least one of the reactivity control systems be capable of making the core subcritical under any condition sufficiently fast to prevent exceeding acceptable fuel damage limits;

9. Draft GDC-30, insofar as it requires that at least one of the reactivity control systems provided shall be capable of making and holding the core subcritical under any conditions with appropriate margins for contingencies;

10. Draft GDC-31, insofar as it requires that the reactivity control systems be capable of sustaining any single malfunction without causing a reactivity transient which could result in exceeding acceptable fuel damage limits;

11. Draft GDC-32, insofar as it requires that limits, which include considerable margin, be placed on the maximum reactivity worth of control rods or elements and on rates at which reactivity can be increased to ensure that the potential effects of a sudden or large change of reactivity cannot:

    a. Rupture the reactor coolant pressure boundary, or

    b. Disrupt the core, its support structures, or other vessel internals sufficiently to impair the effectiveness of emergency core cooling; and

12. Current GDC-20, "Protection system functions," GDC-25, "Protection system requirements for reactivity control malfunctions," and GDC-26, "Reactivity control system redundancy and capability," insofar as they prescribe requirements for the reactor protection and reactivity control system requirements, redundancy and capability.

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

Technical Evaluation

NSPM addresses several aspects of the nuclear design for uprated conditions, including:

1. Core design
2. Fuel thermal margin monitoring
3. Thermal limits
4. Reactivity characteristics
5. Interim methods applicability

Items 1-4 are addressed below, and Item 5 is addressed in Section 2.8.7 of this SE.

Core Design

The licensee confirmed the generic disposition set forth in the CLTR, stating that implementation of the EPU will increase the average power density of the core by increasing bundle enrichment and reload fuel batch size, and/or changing the fuel loading pattern (see Enclosure 5 of Reference 1). The required changes are implemented in such fashion as to limit the impact on fuel safety parameters, which include the minimum critical power ratio (MCPR), the linear heat generation rate (LHGR) and maximum average planar LHGR (MAPLHGR).

In order to invoke the CLTR disposition, the licensee is required to confirm that no changes are made to the fuel design limits. The licensee stated that there is no change to the fuel design, and that no fuel design limit changes are necessary from CLTP to EPU. The NRC staff observes, therefore, that the licensee has confirmed the generic disposition, and agrees that the CLTR disposition is applicable to MNGP at EPU conditions. The NRC staff's agreement is based on the fact, as discussed in Section 2.8.1 of this SE, in that the licensee is using the GE14 fuel product line which is approved for use with the CLTR.

The NRC staff requested that the licensee provide predicted reload batch fractions for each of the EPU transition cycles and compare to the most recent operating cycle and to the EPU equilibrium core design (RAI-2.8.2-1). The NRC staff requested this information to confirm that the core design remains within the staff's experience base for EPU cores, which would serve to validate the referenced disposition.

The licensee stated in response to RAI-2.8.2-1 that the batch fraction for Cycle 24 was 0.31. For Cycle 25, the batch fraction was 0.34 (Reference 5). The Cycles 26 and 27 Supplemental Reload Licensing Reports (SRLRs) (References 86 and 87, respectively) indicate that the batch fraction for both cycles was 0.31. The predicted equilibrium EPU core has a batch fraction of 0.36 (Reference 5). These fractions are consistent with the CLTR, which states that the EPU energy requirements are achieved by increases to the bundle average enrichment, increases in reload fuel batch size, and/or changes to the fuel loading pattern. In the case of MNGP, although there is a slight increase in the fresh bundle average enrichment, the EPU energy is largely being added by the insertion of a larger fresh bundle loading (approximately 15-percent more fresh fuel per cycle).

The NRC staff reviewed current loading strategies at other uprated BWRs of MNGP's vintage (Reference 14). MNGP's EPU batch fractions are consistent with those for which the NRC staff was able to locate data.

The acceptability of the core nuclear design also relies on acceptable results of AOO transient and accident analyses, which are acceptable for uprated conditions as evaluated by the NRC staff in Section 2.8.5 of this SE.

Based on these considerations, the NRC staff determined that: (1) NSPM has acceptably invoked the CLTR disposition for core design; (2) accident and transient analyses evaluated herewith or appropriately disposed have shown acceptable results; and (3) the NRC staff has confirmed that the nuclear design characteristics are, to a reasonable extent, consistent with both limitations on the GE14 bundle design characteristics (see NRC staff evaluation of RAIs 2.8.1-1 and 2.8.1-2) and the NRC staff's CPPU experience base. The NRC staff concludes that NSPM has acceptably accounted for the core design at EPU conditions, and in this respect, that the proposed EPU at MNGP is acceptable.

Fuel Thermal Margin Monitoring

The MNGP TSs require monitoring for margin to the fuel thermal limits. For example, limiting condition for operation (LCO) 3.2.1 requires that all average planar LHGRs (APLHGRs) be less than or equal to the limits specified in the Core Operating Limits Report (COLR). This LCO, as are the others that pertain to the fuel thermal limits, is applicable when the thermal power is greater than or equal to 25-percent RTP. This applicability requirement is the fuel thermal margin monitoring threshold.

The CLTR requires that licensees confirm whether 25 percent rated average bundle power exceeds 1.2 MWth per bundle. At MNGP EPU conditions, the licensee stated, the average bundle power at 25-percent RTP will be 1.0 MWth per bundle (Reference 7). The licensee concluded, therefore, that because the 25-percent, uprated condition, average bundle power at MNGP will be less than 1.2 MWth per bundle, the thermal limits monitoring threshold need not be rescaled.

The NRC staff researched the basis for this generic disposition and concluded that, because the disposition is based on the thermal limits monitoring threshold of the highest power density BWR operating at original licensed thermal power level, it may not be directly applicable to MNGP. The NRC staff's concern arises from the fact that the other plant is of a different design class, and has a significantly larger core, than MNGP. The NRC staff, therefore, requested additional justification of MNGP's conclusion on this matter. The NRC staff's request focused on the direct impact of a lack of monitoring for margin to thermal limits in the 20- to 25-percent power range.

In response to NRC staff RAI-2.8.2-2 related to the above concern, the licensee provided additional clarification for the basis of the disposition regarding the margin for thermal limits monitoring. The licensee stated that, in the 20- to 25-percent power range, there is a very large margin on critical power (Reference 5). With such large margins, the licensee concluded, no transients would have limiting consequences when initiated from the 20- to 25-percent power range.

The NRC staff determined that the licensee's response was acceptable, since the threshold is established based on large margins on critical power. On this basis, the NRC staff agrees with the licensee that the fuel thermal margin monitoring threshold need not be rescaled for EPU conditions.

Thermal Limits Assessment

Section 2.8.2.3 of the PUSAR addresses the effect of EPU on the MCPR safety and operating limits and on the MAPLHGR and LHGR limits. The NRC's acceptance criteria require that the reactor core and the associated control and instrumentation systems be designed with appropriate margin to ensure that the specified acceptable fuel design (or, in the case of the proposed AEC GDC, damage) limits SAFDLs are not exceeded during normal operation, including anticipated AOOs. Operating limits are established to assure that regulatory or safety limits are not exceeded for a range of postulated events (transients and accidents).

Safety and Operating Limit Minimum Critical Power Ratios

The safety limit minimum critical power ratio (SLMCPR) ensures that 99.9 percent of the fuel rods are protected from boiling transition during steady-state operation, with adequate margin for uncertainties and AOOs. The operating limit minimum critical power ratio (OLMCPR) ensures that, in steady-state operations, the plant has sufficient thermal margin to accommodate the effects of AOOs without challenging the SLMCPR. The SLMCPR is controlled by the MNGP TSs; the OLMCPR is set on a cycle-specific basis and reported in the Core Operating Limits Report.

<u>Safety Limit Minimum Critical Power Ration (SLMCPR)</u>

The licensee stated that the SLMCPR can be affected slightly by EPU due to a flatter power distribution, but that the resultant increase in SLMCPR is typically less than 0.01 (Reference 7). NSPM will analyze the SLMCPR on a cycle-specific basis to confirm this observation.

The licensee confirmed the generic disposition in the CLTR by stating that [[

]]. The licensee also stated that the SLMCPR will include a 0.02 adder for increased core flow uncertainties during single recirculation loop operation.

The NRC staff reviewed the uncertainty values contained in the SLMCPR evaluation topical reports to confirm the parameters' applicability to MNGP at EPU conditions.[3] Based on its review, the staff did not identify any uncertainly values that were inappropriate for MNGP's proposed operating condition, with the exception of the SLMCPR adders required for adherence to the conditions and limitations contained in the safety evaluation approving NEDC-33173P-A

[3] The GEH SLMCPR basis documents, which describe how the SLMCPR is calculated and verified, include the following NRC-approved licensing topical reports: (1) NEDC-32601P-A, "Methodology and Uncertainties for Safety Limit MCPR Evaluations" (Reference 28), and (2) NEDC-32694P-A, "Power Distribution Uncertainties for Safety Limit MCPR Evaluation" (ADAMS Accession No. ML003740151).

(Reference 25). The NRC staff issued a request for additional information regarding the validity of the SLO adder, which is evaluated in Section 2.8.7.1 of this SE.

By letter dated May 4, 2011 (Reference 90), the NRC staff issued an amendment to the MNGP facility operating license which changed the SLMCPR set forth in TS 2.1 from ≥ 1.10 to ≥ 1.15 for two recirculation loop operation (TLO), and from ≥ 1.12 to ≥ 1.15 for one (i.e., single) recirculation loop operation (SLO). This TS change supports cycle designs that include the EPU operating domain, and includes the adders required to satisfy the limitations and conditions related to SLMCPR contained in NEDC-33173P-A. Although the SLO SLMCPR value is typically higher to account for increased core flow uncertainties, this SLMCPR license amendment provides adders required not only for EPU, but also for the MELLLA+ operating domain, which is not currently approved for MNGP. Additional information describing why the two values are equivalent is provided in Item 2 of NSPM's response to a request for additional information. For the EPU, the NRC staff concludes that the 1.15 SLMCPR limit values are acceptable because they bound (i.e., they are greater than or equal to) the previous SLMCPR values that were applicable to the EPU operating domain.

The NRC staff determined that the licensee's evaluation of the SLMCPR for MNGP was acceptable for the proposed EPU. The NRC staff's conclusion in this regard is based on the fact that the SLMCPR is analyzed using the NRC-approved methods described in Reference 28, and its applicability will be confirmed on a cycle-specific basis.

<u>Operating Limit Minimum Critical Power Ration (OLMCPR)</u>

NRC staff experience with several power uprate amendments has shown that the change in OLMCPR resulting solely from the EPU is small. The OLMCPR will be determined for MNGP cycle-specific core design parameters using approved methods, as discussed in Chapter 1 of the PUSAR.[4] As required by the CLTR and the cycle-specific reload licensing requirements, the licensee will perform cycle-specific reload analyses to determine the OLMCPR.

The licensee stated that it will evaluate the OLMCPR as part of the reload licensing analysis performed for the cycle-specific core design (Reference 7). The licensee stated that the EPU operating conditions have only a small effect on the MCPR operating limit. The OLMCPR is calculated by adding the change in MCPR due to the limiting AOO event to the SLMCPR. The OLMCPR is determined on a cycle-specific basis using NRC-approved methods, and the method does not change with the EPU.

The NRC staff accepts the licensee's disposition regarding the OLMCPR because the OLMCPR will be reassessed on a cycle-specific basis using NRC-approved reload licensing methods. The OLMCPR assessment is acceptable for uprate operation at MNGP.

Additional conservatisms in the OLMCPR required for the interim implementation of GE/GNF analytic methods at the EPU expanded operating domain will be added and are addressed by the NRC staff in Section 2.8.7.1 of this SE.

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[4] During the course of the EPU review, the licensee has implemented new safety analysis methods. The licensee now uses NEDC-32906P, Supplement 3-A, Revision 1, "Migration to TRACG04/PANAC11 from TRACG02/PANAC10 for TRACG AOO and ATWS Overpressure Transients." Although the EPU analyses were based on prior methods, the staff evaluated the adequacy of the TRACG04 migration with respect to adherence to NEDC-33173P-A (ML12313A107) conditions and limitations in Section 2.8.7 of this SE.

The MAPLHGR operating limit is based on the most limiting LOCA conditions, and ensures compliance with the ECCS acceptance criteria in 10 CFR 50.46. For every reload, licensees confirm that the MAPLHGR operating limit for each reload fuel bundle design remains applicable (topical report NEDC-33004P-A). The generic disposition contained in the CLTR is based on the fact that, not only do cycle-specific reload analyses confirm the applicability of the MAPLHGR operating limits, but the MAPLHGR operating limits are generally unaffected by EPU implementation.

As addressed by the NRC staff in Section 2.8.5.6 of this SE, the licensee has recently elected to remove an upper bound peak cladding temperature (PCT) limitation that will result in a relaxation of the MAPLHGR operating limits. In light of this change, the NRC staff asked the licensee to address the changes that the relaxation of the upper bound PCT limitation will have on the MAPLHGR limits, and what impact these changes will have on other operating limits as power distribution limit relaxations propagate through the AOO analyses (RAI-2.8.2-3).

The licensee responded to NRC staff RAI-2.8.2-3 by stating that the MAPLHGR limits are now set as determined by fuel operation limits, and the application of the MAPLHGR limits would be consistently applied throughout the safety analyses (Reference 5). The licensee's response clarifies that the relaxed MAPLHGR limits are incorporated into the remaining safety analyses.

Linear Heat Generation Rate (LHGR)

The licensee stated in the PUSAR that the Maximum LHGR Operating Limit is determined by the fuel rod thermal mechanical design and is not affected by EPU.

Since the licensee submitted its EPU application, the NRC issued Information Notice (IN) 2009-23, "Nuclear Fuel Thermal Conductivity Degradation [TCD]" (References 91 and 92). IN 2009-23 describes that legacy codes may not account for the burnup-dependent degradation of thermal conductivity in uranium dioxide fuel, and that the results of downstream safety analyses that rely on such legacy codes may be less conservative than previously understood. For use of the GESTR-M-based analytic methods that support the EPU application, the licensee applied penalties consistent with the conditions and limitations contained in NEDC-33173P-A (Reference 27), as discussed in Section 2.8.7.1 of this SE.

A supplemental letter dated July 8, 2013 (Reference 93) provided additional detail explaining how the licensee accounts for TCD in its safety analyses, in light of a transition from GESTR-M-based analytic methods, which do not account for burnup-dependent nuclear fuel TCD, to PRIME-based analytic methods (Reference 87).

Because the LHGR operating limits are determined by the fuel rod thermal mechanical design, and unaffected by EPU, the NRC staff determined that the licensee's disposition was acceptable for the proposed EPU.

Additional NRC Staff Evaluation of Thermal Limits

In general, the licensee must ensure that plant operation is in compliance with the cycle-specific thermal limits (SLMCPR, OLMCPR, MAPLHGR, and maximum LHGR) and specify the thermal limits in a cycle-specific COLR as required by the MNGP TSs.

The NRC staff requested that the licensee demonstrate the validity of the constant pressure power uprate licensing topical report (CLTR) conclusions in this regard by providing confirmation regarding the OLMCPR, MAPLHGR, and LHGR operating limits in the form of a SRLR for Cycle 25 (RAI-2.8.2-4).

In response to the NRC staff's RAI, the licensee provided the SRLR for Cycle 25, noting that the full EPU SRLR will not be available prior to November 2009 (Reference 5). To provide additional information regarding the full EPU performance, the licensee provided a table of uncorrected ΔCPR values for transient analyses performed using the equilibrium EPU core design.

The NRC staff reviewed the SRLR for Cycle 25 and observed that there were not significant differences between it, the current Cycle 24 design (MNGP USAR), and the equilibrium EPU core design. The NRC staff also reviewed the table of uncorrected ΔCPR [change in critical power ratio] values and observed that, while some critical power ratio (CPR) transients underwent a small increase in ΔCPR, others reflected a small decrease. Most importantly, the limiting CPR transient remains the inadvertent high pressure coolant injection actuation with a turbine trip on Level 8. For this transient, the limiting uncorrected ΔCPR value did not change.

Upon implementation of the TRACG04 transient analysis methodology, the licensee began characterizing the effects of the pressurization transients (e.g., inadvertent HPCI initiation) in its SRLRs in terms of uncorrected ΔCPR per initial CPR. The NRC staff compared these quantities for Cycles 25, 26, and 27, where the values were 0.247, 0.234, and 0.248, respectively. Since the Cycle 25 SRLR did not reflect the EPU operating domain, these parameters confirm that the effect of the proposed EPU on thermal-hydraulic margin is within the cycle-to-cycle variation (References 5, Reference 86, and 87).

The information provided by the licensee in response to RAI-2.8.2-4 provides additional confirmation that the dispositions set forth in the CLTR regarding thermal limits are applicable to the MNGP uprated core design. This information, therefore, confirms the licensee's disposition.

Reactivity Characteristics

The licensee stated that it will maintain all minimum shutdown margin requirements without change (Reference 7). The licensee checked for adequate margin to cold shutdown, evaluating shutdown using both the standby liquid control system and the control rods.

The higher core energy requirements of a power uprate may affect the hot excess core reactivity and can also affect operating shutdown margins. The general effect of a power uprate on core reactivity, as described in Section 5.7.1 of ELTR1, is applicable to an EPU (Reference 9). Based on experience with previous plant-specific power uprate submittals, the required hot excess reactivity and shutdown margin can typically be achieved for power uprates through the standard approved fuel and core reload design process. Plant shutdown and reactivity margins must meet NRC-approved limits established in GESTAR-II [GE Standard Application for Reactor Fuels] (topical report NEDC-24011P-A) on a cycle-specific basis and are evaluated for each plant reload core. Additional hot excess reactivity and shutdown margin analyses are not specifically required for the EPU.

The reload core analysis will ensure that the minimum shutdown margin requirements are met for each core design and that the current design and TS cold shutdown margin will be met. Since the licensee will continue to confirm that the TS cold shutdown requirements will be met for each reload core operation, the NRC staff finds this acceptable, and concludes that the NRC's acceptance criteria outlined in the *Regulatory Evaluation* section above will continue to be satisfied.

GEH stated in the ELTR and reaffirmed in the CLTR, and the NRC staff agreed, that the fuel reactivity characteristics for power uprate can be generically disposed. The licensee, therefore, confirmed that the MNGP reactivity characteristics are consistent with the generic description discussed in the ELTR and the CLTR, and that the licensee will evaluate the shutdown margin for each uprated reload.

Conclusion

The NRC staff has reviewed the licensee's assessment for MNGP, and concludes that it is consistent with the information and disposition described in the CLTR. In addition, the licensee committed to continue performing plant-specific reload analyses to confirm that SAFDLS and RCPB pressure limits will not be exceeded during the planned cycles. Based on this, and in coordination with the reviews of the fuel system design, thermal and hydraulic design, and transient and accident analyses, the NRC staff concludes that the nuclear design of the fuel assemblies, control systems, and reactor core will continue to meet the applicable requirements of draft GDC-6, 7, 8, 12, 14, 15, 27, 28, 29, 31, and 32, and, therefore, is acceptable to the NRC staff.

2.8.3 Thermal and Hydraulic Design

Regulatory Evaluation

The NRC staff reviewed the thermal and hydraulic design of the core and the RCS to confirm that the design is consistent with the following:

1. Has been accomplished using acceptable analytical methods,

2. Is equivalent to, or a justified extrapolation from, proven designs,

3. Provides acceptable margins of safety from conditions which would lead to fuel damage during normal reactor operation and AOOs, and

4. Is not susceptible to thermal-hydraulic instability.

The NRC's acceptance criteria are based on:

1. Draft GDC-6 and current GDC-10, "Reactor design," insofar as they require that the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs;

2. Draft GDC-7 and current GDC-12, "Suppression of reactor power oscillations," insofar as they require that the reactor core be designed to ensure that power oscillations, which could cause damage in excess of acceptable fuel damage limits, are not possible or can be readily suppressed.

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

Technical Evaluation

Summary of Technical Information

MNGP provided a technical evaluation of the proposed power uprate from the thermal-hydraulic point of view as specified by RS-001. The licensee stated that reload safety analyses will continue to be performed for EPU using approved methods to demonstrate compliance with thermal and hydraulic safety limits. Based on the evaluation, MNGP concludes that EPU affects two thermal-hydraulic items that require special attention: (1) stability; and (2) ATWS [anticipated transient without scram]-stability. The NRC staff audited MNGP on May 21, 2009. During the audit, the staff reviewed the implementation of long term stability solution Option III and found it acceptable. Information obtained during this audit supplements the information in NEDC-33322P for this review (Reference 17).

Stability Long-Term Solution

MNGP is a small core plant with a tight inlet orifice (see Reference 17). As such, it qualified for stability Long-Term Solution Option I-D. MNGP has been operating under Option I-D for a number of years and its experience has been positive. However, as part of EPU and the upgrade to a digital neutron monitoring system, NSPM has decided to update MNGP's stability long-term solution to Option III, which provides: (1) more operating flexibility; and (2) the possibility to upgrade to MELLLA+ methodology in the future by enabling the Detect and Suppress Solution – Confirmation Density (DSS/CD) stability solution.

NEDC-33322P, Revision 3, describes the Option III implementation at MNGP (Reference 7). The armed region in Option III is defined as percent power and flow (greater than 30 percent power and less than 60 percent flow). However, with the power uprate to 120 percent, the percent power for the armed region is set to 25 percent to maintain the same region in terms of megawatts. This is an acceptable and recommended deviation of the approved Option III because the original Option III approval did not envision the possibility of power uprate. This modification maintains the same level of stability protection.

Option III requires the combination of local power range monitor (LPRM) signals in a series of oscillation power range monitor (OPRM) channels, which are similar in nature to the existing average power range monitor (APRM) channels, differing only on the LPRM grouping. APRM channels attempt to average LPRM signals from all over the core. OPRM channels utilize average LPRM signals from specific regions in the core, so that they can detect regional or out-of-phase oscillations. APRM channels are not sensitive to out-of-phase oscillations because they average them out. The LPRM groupings in the OPRM channels are designed to avoid this problem. MNGP installed Option III during the April - May 2009 reload to support the power uprate. All hardware is now in place and operational. Installation of the OPRM system for MNGP took approximately 20 days, followed by approximately 10 days of testing. No events were reported.

The MNGP OPRM system has implemented the lessons learned from the Nine Mile Point and Fitzpatrick stability events (Reference 5). The low-pass corner frequency of the OPRM algorithm and period tolerance values are set to the recommended values of 1 Hertz and 100 milliseconds, respectively.

In the MNGP implementation for the EPU, the licensing basis protection is provided by the standard Solution III Period Based Detection Algorithm. As with all standard Solution III implementations, the other two defense-in-depth algorithms (Growth Rate Based and Amplitude Based) are present and would scram the reactor; however, no analysis is required to ensure that the defense-in-depth algorithms protect against SAFDLs for every possible scenario. Only the Period Based Detection Algorithm (PBDA) setpoint value is determined to ensure that SAFDLs are protected with a high likelihood. The licensee evaluated the Solution III hardware for 90 days before arming the OPRM system. This is a standard procedure, and all Solution III licensees use an evaluation period to familiarize themselves with the new OPRM system operation and avoid spurious scrams. MNGP armed the OPRM system in September 2009. During the 90-day evaluation period, MNGP used a plant-specific backup stability protection (BSP), which is based on the DSS/CD BSP solution and the old "interim corrective actions" (ICAs). The MNGP BSP is based on stability calculations of the exclusion regions, which require either immediate exit or scram depending on the region and special circumstances. Per TSs, during this 90-day period, the OPRM was declared inoperable. Entry into Region II required an immediate exit. Entry into Region I required an immediate scram. The MNGP TSs rely on the ICAs when the Solution III licensed application is unavailable. The ICA exclusion regions are specified in the SRLR and are available in the control room via procedures. MNGP TSs allow operation under ICAs for up to 120 days. MNGP uses plant-specific exclusion regions for its ICAs, which are verified for adequacy at every reload. The ICAs are enforced manually and are an acceptable temporary solution when the primary solution (e.g., Solution III) is unavailable under the current operating domain. The NRC staff reviewed the ICAs and found the actions acceptable as a backup for up to 120 days. The specific MNGP ICA procedures are contained in Abnormal Procedure C.4-B.05.01.02.A "Control of Neutron Flux Oscillations," which was reviewed by the NRC staff during the May 21, 2009, audit, and is similar to others in the industry.

In its January 21, 2013, letter (Reference 67), the licensee states that the OPRM-based Option III long term stability solution equipment has been installed and was turned over to the Operations in September 2009. The monitoring and evaluation period has been completed.

The only deviations from the standard Solution III implementation in MNGP relates to the high growth and decay ratio alarms, the voting logic for the OPRM upscale and APRM inoperable functions, and a modification to the Option III PBDA. The NRC previously evaluated these deviations, summarized in Reference 70, and found them acceptable. First, MNGP determined that a plant operator would not have time to act to prevent a scram if any of these two alarms were triggered during an oscillation event. Therefore, MNGP has disabled these alarms to simplify operator training. Note that the trips are still enabled for both high growth and high decay ration. This action is acceptable since the alarm is a trip preventing measure that serves no safety function. Second, the MNGP PRNM system has modified APRM upscale, OPRM upscale, and APRM INOP function logic. The logic change eliminates the occurrence of two half-trips in each of the 2-out-of-4 voter channels when a combination of an inoperable APRM INOP function in one APRM channel and an inoperable OPRM upscale function in another channel occurs. The modified combination at MNGP now results in RPS trip outputs in all 2-out-of--4 voter channels when this inoperability combination occurs. Finally, the MNGP installation defines the base period differently from the Option III licensing basis. The topically defined base period is the average of all successively confirmed periods. The MNGP application defines the successive base period as equal to the previous period that is within the

PBDA algorithm upper and lower time limits of the oscillation period. This change maximizes the ability of the PBDA to recognize the initiation of oscillations following a fast flow reduction event.

A 5 percent OPRM setpoint reduction due to bypass voiding has been applied to account for the possible presence of voids in the bypass region. This is consistent with GEH EPU Interim Methods as discussed in Section 2.8.7.1.

Anticipated Transient Without Scram (ATWS) - Stability

Background

MNGP has performed an evaluation of the ATWS-Stability event. For this event, a turbine trip with bypass is assumed, followed by failure to scram. When the extraction steam is lost as a result of the turbine trip, the feedwater temperature cools down, which causes a significant power increase and very large unstable power oscillations may develop. The ATWS stability mitigation actions were designed to minimize the impact of this very severe event.
In Enclosure 5 of the November 5, 2008, submittal, MNGP evaluates the ATWS-Stability event at EPU conditions and concludes that the ATWS-Stability analysis of record in NEDO-32164 is applicable to MNGP under EPU conditions. The licensee based this conclusion on the fact that the maximum rod line is unchanged and, thus, following the recirculation pump trip (RPT) prescribed by the ATWS rule, the reactor will be in similar conditions before or after EPU is implemented. In addition, MNGP is a very stable plant because of its small core size and tight inlet orifice; therefore, ATWS-Stability oscillations are expected to be of smaller amplitude in MNGP than in the analysis of record.

The NRC staff conducted an audit on May 21, 2009, and reviewed the MNGP ATWS procedures and witnessed three ATWS events in the plant simulator (Reference 17). All events were handled properly by the operators and the reactor was successfully shutdown without violating the ATWS criteria, which are based on core coolability, pressure boundary limits, and radiation release from containment.

Technical Evaluation

The NRC staff's findings in regard to thermal-hydraulic design are based on the following considerations:

1. For the first EPU implementation operating cycle, MNGP will have a full core loading of GEH fuel. MNGP uses approved GEH analytical methods for its analysis.

2. GEH has used the approved interim methods for EPU applications. Specifically, a 5 percent penalty is applied to the OPRM setpoint to account for the possibility of bypass voiding.

3. The proposed MNGP power uprate is similar to those implemented in other plants, and it is based on extending the maximum rod line to full flow.

4. In Enclosure 5 of the November 5, 2008, application, NSPM indicates that it will perform plant-specific reload analysis to confirm for the first EPU core that fuel design limits and RCPB pressure limits will not be exceeded under EPU conditions.

5. Compliance with stability of MNGP under EPU conditions is accomplished through a "detect and suppress" option that satisfies GL 94-02 and GDC-12. The stability solution implemented in MNGP (Option III) has been in operation in similar plants with success. Option III is an approved solution up to EPU conditions that satisfies the requirements of GL 94-02.

6. ATWS-stability has been evaluated by MNGP. The analysis of record in NEDO-32164 is applicable to MNGP under EPU conditions because the EPU upgrade does not change significantly the end point in the power-flow map once the recirculation pumps trip. In addition, MNGP is a very stable plant because of its small core and tight inlet orifice. The analysis of record in NEDO-32164 assumed a plant with a more unstable configuration (larger core and looser orifice), which resulted in very large unstable power oscillations. An ATWS-stability event at MNGP should have significantly lower oscillation amplitude.

Conclusion

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

2.8.4 Emergency Systems

2.8.4.1 Functional Design of Control Rod Drive System

Regulatory Evaluation

The NRC staff's review covered the functional performance of the control rod drive (CRD) system to confirm that the system can affect a safe shutdown, respond within acceptable limits during AOOs, and prevent or mitigate the consequences of postulated accidents. The review also covered the CRD system to ensure that it will continue to meet its design requirements.

The NRC's acceptance criteria are based on:

1. Draft GDC-26, insofar as it requires that the protection system be designed to fail into a safe state;

2. Draft GDC-27 and 28, insofar as they require that at least two independent reactivity control systems be provided, with both systems capable of making and holding the core subcritical from any hot standby or hot operating condition sufficiently fast to prevent exceeding acceptable fuel damage limits;

3. Draft GDC-29, insofar as it requires that at least one of the reactivity control systems be capable of making the core subcritical under any condition sufficiently fast to prevent exceeding acceptable fuel damage limits;

4. Draft GDC-30, insofar as it requires that at least one of the reactivity control systems provided shall be capable of making and holding the core subcritical under any conditions with appropriate margins for contingencies;

5. Draft GDC-31, insofar as it requires that the reactivity control systems be capable of sustaining any single malfunction without causing a reactivity transient which could result in exceeding acceptable fuel damage limits;

6. Draft GDC-32, insofar as it requires that limits, which include considerable margin, be placed on the maximum reactivity worth of control rods or elements and on rates at which reactivity can be increased to ensure that the potential effects of a sudden or large change of reactivity cannot:

    a. Rupture the reactor coolant pressure boundary, or

    b. Disrupt the core, its support structures, or other vessel internals sufficiently to impair the effectiveness of emergency core cooling;

7. Draft GDC-40 and 42, insofar as they require that protection be provided for ESFs against the dynamic effects and missiles that might result from plant equipment failures, as well as the effects of a loss-of-coolant accident;

8. Current GDC-25, "Protection system requirements for reactivity control malfunctions," and GDC-26, "Reactivity control system redundancy and capability," insofar as they prescribe functional, redundancy, and capability requirements for the reactivity control system; and

9. 10 CFR 50.62, "Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants," Section (c)(3), insofar as it requires that all BWRs have an alternate rod insertion (ARI) system diverse from the reactor trip system, and that the ARI system has redundant scram air header exhaust valves.

Specific review criteria are contained in SRP Section 4.6 and guidance provided in Matrix 8 of RS-001; the licensee provided a link to the applicable, proposed draft GDC. The licensee also indicated that current GDC-25 and 26 are relevant to this review insofar as they are incorporated in the MNGP licensing basis as described in USAR Section 14.4.

Technical Evaluation

The MNGP CRD system is described in Section 3.5 of the MNGP USAR. The CRD system is used to position movable rods in 6-inch steps to control the neutron flux distribution in the core. The basic drive mechanism is a double-acting, mechanically latched, hydraulic cylinder that uses water as the operating fluid. The water also serves to cool the drive mechanism. The hydraulic drive is used for controlled insertion and withdrawal of control rods.

The rods also have a scram function. In the event of a reactor scram, the rods obtain the drive fluid for rapid insertion from a bank of scram accumulators. The accumulator pressure produces a large upward force on an index tube and the control rod, which causes the control rods to accelerate rapidly, then insert at a rate of approximately five feet per second. Note that the initial drive force is provided by the scram accumulator discharge, but that, according to Topical Report NEDC-32523P-A, Supplement 1, Volume 1, the control rods rely on reactor pressure to complete the scram (Reference 11).

The licensee referenced the generic disposition of the control rod drive system contained in the CLTR (Reference 7). The CLTR provides this disposition for **[[**
]]. The licensee addressed three topics in its evaluation of the functional design of the control rod drive system:

- Control Rod Scram
- Control Rod Drive Positioning and Cooling
- Control Rod Drive Integrity

Control Rod Scram

The licensee stated that the scram times are decreased by the transient pressure increase, which causes the **[[** **]]** (Reference 7). This is because, as indicated in the PUSAR, while the CRD hydraulic control unit supplies the initial scram pressure, the reactor becomes the primary source of pressure to complete the scram.

Because the steady-state operating pressure does not change due to the power uprate, the initial pressure against which the hydraulic control units (HCUs) must provide drive pressure to the control rod to attain the scram function would not change appreciably. Therefore, the initial rapid acceleration of the control rod for which the HCU is required would still be attained, and the reactor pressure would provide the motive necessary to complete the scram at uprated conditions. The NRC staff agrees, therefore, that **[[**
]].

Technical Specification 3.1.4 provides requirements and acceptance criteria for scram time testing. The licensee demonstrates in accordance with the surveillance requirements of TS 3.1.4 that the scram performance of the CRD system is within the analyzed capability of the scram system. The licensee has not requested to change these requirements in concert with the EPU.

The licensee concluded that the CRD system control rod scram at Monticello is confirmed to be consistent with the generic description provided in the CLTR for pre-BWR/6 plants, **[[**
]]. The NRC staff, as described above, agrees with this disposition. The NRC staff also notes that the scram function of the control rod drive system must also be verified in accordance with TS SR 3.1.4, and that the licensee has requested no change to the TS. Based on these two considerations, the NRC staff finds the control rod scram performance acceptable for the requested EPU.

Control Rod Drive Positioning and Cooling

The NRC staff's SE approving the CLTR states that the normal CRD positioning function is an operational consideration and not a safety-related function (Reference 8).

Notwithstanding this information, the CLTR states that the increase in reactor power at the CPPU operating condition results in a [[ ]]. General Electric has concluded that this [[ ]] from the CRD system to the CRDs during normal plant operation, and thus, that [[ ]] by CPPU implementation. The PUSAR states that automatic operation of the CRD system flow control valve maintains the required drive water pressure and cooling water flow rate (Reference 7).

To offer some order of magnitude for this change at the core plate, the NRC staff has observed that other BWR licensees implementing CPPU have quantified the change as [[ ]], which is also consistent with the change identified in Enclosure 5 of the licensee's November 5, 2008, application.

The licensee confirmed the CLTR generic disposition, adding that [[ ]]. The NRC staff estimates that the valve has adequate margin to compensate for the changes expected at the core plate. In light of the changes that occur at the core plate during uprated operation, and the licensee's confirmation of adequate margin to compensate for these changes, the NRC staff agrees with the licensee's adoption of the CLTR generic disposition and finds the requested CPPU acceptable with respect to control rod drive positioning and cooling.

Control Rod Drive Integrity Assessment

The CLTR states that the constant pressure power uprate causes an increased transient pressure response, which poses a potential to create higher pressure loadings (Reference 8). With respect to the CRD design, according to the CLTR, the postulated abnormal operating condition assumes a failure of the CRD system pressure-regulating valve that applies the maximum pump discharge pressure to the CRDM internal components. This postulated abnormal pressure bounds the ASME reactor overpressure limit.

The CLTR states further that [[ ]]. The disposition for other mechanical loadings is provided in Section 3.3.2 of the CLTR, and 2.2.2 of both the licensee's PUSAR and the NRC staff's SER.

The licensee confirmed that the pressure for the ASME RPV overpressure condition is 1335 psig, compared to the ASME limit of 1375 psig (Reference 7). Therefore, the licensee confirmed the CLTR generic disposition and the NRC staff agrees that this disposition is acceptable. The NRC staff finds the proposed EPU acceptable with respect to the integrity of the CRD system.

Conclusion

The NRC staff has reviewed the licensee's evaluation related to the effects of the proposed EPU on the functional design of the CRD system. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed EPU on the system and demonstrated that the system's ability to perform a safe shutdown, respond within acceptable limits, and prevent or mitigate the consequences of postulated accidents following the implementation of the proposed EPU.

The NRC staff further concludes that the licensee has demonstrated that sufficient technical basis exists to ensure the system's design bases will continue to be followed upon implementation of the proposed EPU. The present design satisfies the draft GDCs under which MNGP was licensed. No system changes are required for EPU, so the system design will continue to meet draft GDCs and current licensing bases in this technical area. Based on these considerations, the NRC staff concludes that the CRD system and associated analyses will continue to meet the requirements of draft GDC-26, 27, 28, 29, 31, 32, 40, and 42, current GDC-25 and 26, and 10 CFR 50.62(c)(3) following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the functional design of the CRD system.

2.8.4.2 Overpressure Protection During Power Operation

Regulatory Evaluation

Overpressure protection for the RCPB during power operation is provided by relief and safety valves and the reactor protection system. The NRC staff's review covered relief and safety valves on the main steam piping from these valves to the suppression pool. The NRC's acceptance criteria are based on the following:

1. Current GDC-15, "Reactor coolant system design," insofar as it requires that the RCS and associated auxiliary, control, and protection systems be designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including AOOs;

2. Draft GDC-33, 34, and 35, insofar as they require that the RCPB be designed to assure that it behaves in a non-brittle manner and that the probability of rapidly propagating type failures is minimized.

Specific review criteria are contained in SRP Section 5.2.2 and guidance provided in Matrix 8 of RS-001. The licensee indicated that current GDC-15 was applicable as described in USAR Section 14.4.

Technical Evaluation

The reactor pressure relief system is discussed in Section 4.4 of the MNGP USAR. The safety/relief valves (SRVs) provide over-pressure protection for the NSSS, preventing failure of the nuclear system pressure boundary and uncontrolled release of fission products. The MNGP USAR indicates that the MNGP main steam system is equipped with eight MSRVs. These SRVs provide the mitigating capability for the over-pressure transient, which is terminated by the reactor scram function.

The licensee stated in the PUSAR that no SRV setpoint increase is needed for the requested EPU because there is no change in the dome pressure or simmer margin (Reference 7). Because of this, there is no effect on the valve functionality. The NRC staff accepts this conclusion.

Consistent with GE's analytic experience with EPU application since the approval of ELTR1 and ELTR2, the licensee evaluated one of two possibly limiting overpressure transients. The licensee evaluated the main steam isolation valve (MSIV) closure with scram on high flux (MSIVF), which has been shown to be the limiting event for overpressure when compared to the other potentially limiting event, the turbine trip with bypass failure and scram on high flux (TTNBP).

GE's analyses have shown, as discussed in the CLTR, that the MSIVF typically exceeds the TTNBP in limiting pressure by about 25 to 40 psi (Reference 8). The NRC staff accepted this conclusion as set forth in its safety evaluation for the CLTR. Based on the licensee's disposition of event selection, the NRC staff accepts the licensee's overpressure evaluation based on the MSIVF event.

The SRV set points are established to provide the over-pressure protection function while ensuring that there is adequate pressure difference (simmer margin) between the reactor operating pressure and the SRV actuation set points. The SRV set points are also selected to be high enough to prevent unnecessary SRV actuations during normal plant maneuvers.

MNGP-Specific Analytic Assumptions

The licensee's EPU analysis employs several conservative assumptions (Reference 7). First, the licensee assumes that the direct scram on MSIV position indication fails, which delays the initiation of the reactor trip until the ensuing flux peak is detected. Second, the event initiates at a dome pressure of 1040 psia, which is higher than the nominal dome pressure of 1025 psia. Third, the licensee assumes that three SRVs are out of service, and the MNGP TSs require the operability of seven SRVs. With eight SRVs installed at MNGP, the analysis assumes the availability of two fewer SRVs than required at the plant.

The overpressure protection analysis is performed assuming a starting power level of 102 percent of the EPU RTP.

Using these assumptions, the licensee used the ODYN code as described in NRC-approved licensing topical report NEDO-24154-A.

Analytic Acceptance Criteria

The licensee stated that the design pressure for the reactor vessel and reactor coolant pressure boundary remains unchanged at 1250 psig, with the acceptance limit remaining at 110-percent of the design value, 1375 psig.

MNGP TS 2.1.2, "Reactor Coolant System Pressure SL," provides the safety limit for the maximum calculated reactor steam dome pressure which is 1332 psig. As discussed below, the analysis demonstrates acceptable performance relative to this safety limit.

Evaluation of Analytic Results

The NRC staff issued one technical request for additional information relating to the at-power overpressure protection analysis. The NRC staff observed oscillations in the vessel steam flow and requested that the licensee clarify why these oscillations were occurring while the other parameters shown remained relatively constant (RAI-2.8.4.2-2).

The NRC staff also requested for clarification regarding the scaling of a figure in the PUSAR (RAI-2.8.4.2-1) and any change in safety relief valves assumed out of service (RAI-2.8.4.2-3). Since these RAIs requested clarification, they are not discussed extensively in this SE.

In response to NRC RAI-2.8.4.2-2 regarding vessel steam flow oscillations, the licensee stated that the ODYN model is capturing the effect of a pressure wave traversing between the MSIV and the reactor dome plenum region (Reference 3). The licensee's response indicates that the flow oscillations are a modeled parameter that is attributable to physical phenomena associated with the flow interactions with installed plant hardware during the postulated transient. The licensee's response provides confirmation that the model has performed adequately and, therefore, is acceptable.

In response to NRC RAI-2.8.4.2-3 (Reference 3) regarding the number of SRVs assumed out of service, the licensee confirmed that there is no change from the current analysis basis for MNGP.

The licensee stated that the maximum reactor dome pressure is 1317 psig, with a corresponding peak reactor vessel pressure, located at the bottom of the reactor vessel, of 1335 psig. The peak pressure calculated for this transient remains below 1375 psig, and the calculated peak dome pressure remains below the TS 2.1.2 safety limit of 1332 psig. Based on the predicted peak pressures remaining below their respective limits, the NRC staff concludes that the overpressure protection analysis demonstrates that the proposed EPU is acceptable with respect to overpressure protection during power operation.

Conclusion

The NRC staff has reviewed the licensee's generic and plant-specific analyses related to the effects of the proposed EPU on the overpressure protection capability of the plant during power operation. In addition, the licensee will continue to perform plant-specific reload analyses for each cycle to confirm that SAFDLS and RCPB pressure limits will not be exceeded during the planned cycle.

Based on this information, the NRC staff concludes that the overpressure protection features will continue to meet draft GDC-33, 34, and 35, and current GDC-15 following implementation of the proposed EPU and, therefore, is acceptable. The NRC staff also finds that the licensee has demonstrated that the proposed EPU will not challenge the safety limit contained in TS 2.1.2, the Reactor Coolant System Pressure Safety Limit.

2.8.4.3 Reactor Core Isolation Cooling System

Regulatory Evaluation

The RCIC system serves as a standby source of cooling water to provide a limited decay heat removal capability whenever the main feedwater system is isolated from the reactor vessel. In addition, the RCIC system may provide decay heat removal necessary for coping with a station blackout. The water supply for the RCIC system comes from the CST, with a secondary supply from the suppression pool.

The NRC staff's review covered the effect of the proposed EPU on the functional capability of the system. The NRC's acceptance criteria are based on the following:

1. Draft GDC-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 loss of coolant accident;

2. Draft GDC-37, insofar as it requires that ESFs be provided to back up the safety provided by the core design, the RCPB, and their protective systems;

3. Draft GDC-51 and 57, insofar as they require that piping systems penetrating containment be designed with appropriate features as necessary to protect from an accidental rupture outside containment and the capability to periodically test the operability of the isolation valves to determine if valve leakage is within acceptable limits; and

4. 10 CFR 50.63, "Loss of all alternating current power," insofar as it requires that the plant withstand and recover from a station blackout of a specified duration.

Specific review criteria are contained in SRP Section 5.4.6 and guidance provided in Matrix 8 of RS-001.

Technical Evaluation

MNGP is equipped with a RCIC system which is described in Section 10.2 of the MNGP USAR. The system is designed to serve as a standby source of cooling water to provide decay heat removal capability whenever the main feedwater system is isolated from the reactor vessel. In addition, the RCIC system may provide decay heat removal necessary for coping with a station blackout. The water supply for the RCIC system comes from the CST, with a secondary supply from the suppression pool. For the purposes of design basis evaluation, only the suppression pool water source is considered.

The licensee addresses three evaluations in the PUSAR concerning the RCIC system's capability at EPU conditions. First, the licensee evaluated system performance and hardware. Second, the licensee considered available net positive suction head. Finally, the licensee considered whether the RCIC could provide adequate core cooling for limiting loss of feedwater (LOFW) events.

System Performance and Hardware

The licensee's system performance and hardware evaluation confirms the generic CLTR disposition for the RCIC system (Reference 7). The CLTR states that there is no change to the normal reactor operating pressure, and the SRV setpoints remain the same. There is no change to the maximum specified reactor pressure for RCIC system operation. As this is the case for the MNGP proposed EPU, the NRC staff agrees with the disposition. The RCIC system performance and hardware are acceptable for the proposed EPU.

The MNGP RCIC and ADS initiations are activated at the same reactor water level. This is significant because the CLTR discusses the fact that some BWR plants have an operational requirement for RCIC to prevent the level decrease during a LOFW transient from initiating ADS. This discussion is not applicable to MNGP, as the operational requirement does not apply.

RCIC Net Positive Suction Head Requirements

The CLTR provides a generic disposition for the net positive suction head requirements for the RCIC pump, but this disposition is not applicable at MNGP for EPU conditions. This is because the higher decay heat load associated with the EPU will increase the torus water suction temperature. The licensee therefore evaluated the torus temperature under SBO conditions to confirm that the torus would remain at an acceptable temperature relative to the RCIC net positive suction head (NPSH) requirement limit.

In RAI-2.8.4.3-1, the NRC staff requested that the licensee address how the RCIC NPSH evaluation aligns with or differs from the current design-basis RCIC NPSH evaluation, based on the fact that NSPM did not use the CLTR disposition for the RCIC NPSH requirements.

In response to RAI-2.8.4.3-1 (Reference 3), the licensee clarified that existing calculations for the RCIC system to confirm acceptable NPSH are based on conservative design assumptions for parametric values including flow rates, torus level, and torus temperature, and that these assumptions are selected to bound those that arise during RCIC operation during the design-basis events of interest. The licensee stated that EPU changes to the significant design parameters were re-evaluated, and the current licensing basis analysis was confirmed to remain valid. Therefore, the licensee concluded, no design-basis changes were being made to the RCIC system.

The licensee's response clarifies that no design-basis changes are being made to the RCIC system associated with the EPU, which is responsive to the NRC staff's request, and is hence acceptable. Furthermore, the licensee's response also provides a confirmation of the CLTR disposition in that there is no effect on RCIC NPSH requirements associated with the CPPU.

The licensee's analysis confirmed that, at a peak of 141°F at two hours, the torus temperature was within the net positive suction head requirement for the RCIC system of 170°F. The licensee selected the SBO event because torus cooling is available during the LOFW, and not for the SBO.

The NRC staff requested that the licensee explain how the torus temperature was determined in the SBO evaluation (RAI-2.8.4.3-2). In Reference 3, the licensee responded that the torus temperature was evaluated using the SHEX containment response program and reflect initial conditions that are consistent with the EPU plant configuration (Reference 3). The licensee also indicated that the torus response during the SBO is discussed further in Section 2.3.5 of the PUSAR, and as such, it is evaluated by the NRC staff in Section 2.3.5 of this SE.
The NRC staff requested that the licensee explain why the SBO analysis evaluates torus temperature for a mission time of two hours, rather than four hours (RAI-2.8.4.3-3). In Reference 3, the licensee responded that, while the SBO coping period is four hours, the mission time for the RCIC system is two hours, based on coping with a loss of all offsite power

event. Therefore, the licensee considered the torus temperature at two hours from the SBO analysis, because torus cooling is available under LOOP conditions. The NRC staff agrees that this approach is acceptable because it is within the mission requirements of the RCIC system and also because it is conservative.

Loss of Feedwater Transient

The licensee stated that an evaluation of the LOFW transient confirms that the RCIC system performs adequately at power uprated conditions. The NRC staff requested that the licensee provide additional detail regarding the LOFW analysis that was performed to confirm RCIC performance (RAI-2.8.4.3-4). This transient analysis is discussed further in Section 2.8.5.2.3 of the PUSAR, and the NRC staff's evaluation of the licensee's response is discussed in Section 2.8.5.2.3 of this SE.

Because the licensee has analyzed the LOFW transient for EPU operation, and has conservatively evaluated the pressure performance requirements of the MNGP RCIC system, the NRC staff accepts the licensee's assessment that the RCIC will continue to meet the NRC's acceptance criteria as delineated in the Regulatory Evaluation section above.

Conclusion

The NRC staff has reviewed the licensee's generic and plant-specific analyses related to the effects of the proposed EPU on the ability of the RCIC system to provide decay heat removal following an isolation of main feedwater event and the ability of the system to provide makeup to the core following a small break in the RCPB. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed EPU on these events and demonstrated that the RCIC system will continue to provide sufficient decay heat removal and makeup for these events following implementation of the proposed EPU.

Based on these considerations, the NRC staff concludes that the RCIC system will continue to meet the requirements of draft GDC-37, 40, 42, 51, and 57, and 10 CFR 50.63 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the RCIC system.

2.8.4.4 Residual Heat Removal System

Regulatory Evaluation

The residual heat removal (RHR) system is used to cool down the RCS following shutdown. The RHR system is a low pressure system which takes over the shutdown cooling function when the RCS pressure and temperature are reduced.

The NRC staff's review covered the effect of the proposed EPU on the functional capability of the RHR system to cool the RCS following shutdown and provide decay heat removal.

The NRC's acceptance criteria are based on draft GDC-40 and 42, insofar as they require that ESFs be protected against dynamic effects. Specific review criteria are contained in SRP Section 5.4.7 and guidance provided in Matrix 8 of RS-001.

Technical Evaluation

The MNGP RHR system is described in Section 6.2.3 of the MNGP UFSAR. The RHR system is designed to restore and maintain the reactor coolant inventory following a LOCA and remove reactor decay heat following reactor shutdown for normal, transient, and accident conditions. For MNGP, the RHR system is designed to operate in the LPCI mode, shutdown cooling mode, suppression pool cooling mode, containment spray cooling mode, and fuel pool cooling assist mode. This section of the NRC staff's safety evaluation addresses the shutdown cooling mode of the residual heat removal system.

Other operational and safety objectives of the RHR system are evaluated in different sections of this SE. The LPCI mode is discussed in Section 2.8.5.6.2 of the PUSAR and in this SE. Suppression pool cooling and containment spray cooling are addressed in Section 2.6.5 of the PUSAR and SE. The fuel pool cooling assist mode of RHR operation is addressed in Section 2.5.3.1.1 of the PUSAR and SE.

The licensee stated that the steam condensing mode of RHR is not installed at MNGP, and the NRC staff requested that the licensee clarify some discussion in the USAR that refers to steam condensing capabilities (RAI-2.8.4.4-1). In Reference 3, the licensee responded to the NRC staff's RAI-by clarifying that this language in the USAR refers to the suppression pool cooling mode of RHR, and confirmed that the steam condensing mode of RHR is not installed at MNGP.

According to the CLTR, the CPPU effect on the RHR system is caused by the higher decay heat in the core corresponding to the uprated power and the increased amount of reactor heat discharged into the containment during a LOCA (Reference 8). Higher decay loads will result in a longer time required to attain the shutdown cooling objective, which is to remove sensible and decay heat within a certain time objective.

The licensee stated that the shutdown cooling analysis is performed using guidance contained in RG 1.139, "Guidance for Residual Heat Removal." RG 1.139 recommends that shutdown cooling systems be capable of placing the plant in cold shutdown conditions within 36 hours, which is an objective that the licensee states can be obtained at MNGP at EPU conditions.

The NRC staff notes that RG 1.139 has been withdrawn and requested that the licensee explain in greater detail its residual heat removal evaluation (RAI-2.8.4.4-2). In Reference 3, the licensee clarified that the Reference to RG 1.139 should not have been included in this section of the PUSAR. The Reference refers to an alternate shutdown cooling analysis that was performed in support of Appendix R alternate shutdown requirements. Further details on cooldown with use of alternate shutdown were provided in Reference 51 to note that the time to reach cold shutdown is 44.7 hours.

The licensee stated that the OLTP shutdown cooling (SDC) analyses were performed to confirm that the reactor could be cooled to 125°F during normal reactor shutdown with two SDC loops in-service within about 24 hours. At EPU conditions, the licensee predicts that the same evolution would take 26.5 hours. The licensee stated that this is not a safety-related function, and affects only plant availability.

Based on the following considerations: (1) the licensee has determined the effects of the EPU on RHR shutdown cooling in both an operational and an alternative evaluation; (2) the alternate evaluation confirms that cold shutdown relying only on safety-related systems can be attained within 45 hours; (3) the increase in time to perform the operational shutdown cooling evolution is expected to increase by a small amount; and (4) the operational shutdown cooling time does not affect plant safety; the NRC staff concludes that the licensee has acceptably evaluated shutdown cooling and demonstrated that the proposed EPU is acceptable with respect to the shutdown cooling mode of the RHR system.

Conclusion

The NRC staff has reviewed the licensee's evaluation related to the effects of the proposed EPU on the RHR system. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed EPU on the system and demonstrated that the RHR system will maintain its ability to cool the RCS following shutdown and provide decay heat removal.

Based on these considerations, the NRC staff concludes that the RHR system will continue to meet the requirements of draft GDC-40 and 42 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the RHR system.

2.8.4.5 Standby Liquid Control (SLC) System

Regulatory Evaluation

The SLC system provides backup capability for reactivity control independent of the control rod system. The SLC system functions by injecting a boron solution into the reactor to affect shutdown. The NRC staff's review covered the effect of the proposed EPU on the functional capability of the system to deliver the required amount of boron solution into the reactor.

The NRC's acceptance criteria are based on the following:

(1) Current GDC-26, "Reactivity control system redundancy and capability," insofar as it requires that two independent reactivity control systems of different design principles be provided, and that one of the systems be capable of holding the reactor subcritical in the cold condition;

(2) Draft GDC-27, insofar as it requires at least two independent reactivity control systems, preferably of different principles, to be provided;

(3) Draft GDC-29, insofar as it requires at least one of the reactivity control systems provided be capable of making the core subcritical under any condition (including anticipate operational transients) sufficiently fast to prevent exceeding acceptable fuel damage limits;

(4) Draft GDC-30, insofar as it requires at least one of the reactivity control systems capable of making and holding the core subcritical under any condition with appropriate margin for contingencies; and

(5) 10 CFR 50.62, "Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants," Section (c)(4), insofar as it requires that the SLC system be capable of reliably injecting a borated water solution into the reactor pressure vessel at a boron

concentration, boron enrichment, and flow rate that provides a set level of reactivity control.

Specific review criteria are contained in SRP Section 9.3.5 and guidance provided in Matrix 8 of RS-001. The licensee indicated that current GDC-26 is applicable as described in MNGP USAR Section 14.4.

Technical Evaluation

The MNGP SLC system is described in Section 6.6 of the MNGP USAR. The MNGP SLC system is a manually operated system that pumps an isotopically enriched sodium pentaborate solution into the reactor vessel to effect neutron absorption and is capable of bringing the reactor to a subcritical condition from RTP at any time in a cycle with the reactor in the most reactive xenon-free state with all of the control rods in the full-out condition.

In Enclosure 5 of the November 5, 2008, application the licensee stated that the ability of the SLC system boron solution to achieve and maintain safe shutdown is not a direct function of the core thermal power, and therefore is not affected by EPU (Reference 7). The SLC system shutdown capability (in terms of the required reactor boron concentration) is reevaluated for each fuel load. No new fuel product line designs were introduced for EPU. The boron shutdown concentration of 660 ppm did not change for EPU. No changes were necessary to the solution volume or concentration or the boron-10 enrichment for EPU to achieve the required reactor boron concentration for shutdown. Thus the licensee confirms that the SLC system shutdown margin capability is consistent with the generic description provided in the CLTR.

The licensee performed an EPU ATWS analysis and stated that the peak reactor lower plenum pressure following the limiting ATWS event reaches 1205.3 psig during the time the SLC system is analyzed to be in operation (see Enclosure 5 of the November 5, 2008, application). In response to RAI-2.8.4.5-1, the licensee provided graphs of this limiting ATWS for both CLTP and EPU conditions and the NRC staff was able to confirm the peak lower plenum pressure when the SLC system was analyzed to be in operation (Reference 5). This pressure peak occurs at CLTP conditions and, therefore, there is neither an increase in the maximum pump discharge pressure nor a decrease in the operating pressure margin for the pump discharge relief valves.

In the event that the SLC system is initiated before the time that reactor pressure recovers from the first transient peak, resulting in opening of the SLC system relief valves, the reactor pressure must reduce sufficiently to ensure SLC system relief valve closure. The licensee stated that the analytical results indicate pump discharge relief valve would reclose before the time that the reactor pressure recovers from the first transient peak. The licensee also stated that consideration was given to the system flow, head losses for full injection, and cyclic pressure pulsations due to the positive displacement pump operation in determining the pressure margin to the opening set point for the pump discharge relief valves.

10 CFR 50.62(c)(4) requires that each BWR must have a SLC system with a minimum flow capacity and boron content equivalent in control capacity to 86 gallon per minute (gpm) of 13 weight percent (wt%) sodium pentaborate decahydrate solution at the natural boron-10 isotope abundance into a 251-inch inside diameter RPV for a given core design. For ATWS, the equivalency requirement of the rule can be met if the following relationship is satisfied:

(Q/86) x (M251/M) X (C/13) X (E/19.8) > 1

where:

Q = expected SLC system flow rate, gpm
M = mass of water in the RPV and recirculation system at hot rated condition, lbs
C = sodium pentaborate solution concentration, wt%
E = Boron-10 isotope enrichment (19.8% of natural boron)
M251 = mass of water in a BWR/4 251-inches diameter RPV (lbs) = 628300 lbs

The licensee performed calculations to verify that the SLC system complies with the ATWS rule referred above. Using the following MNGP-specific values to satisfy the relationship given above, the licensee established the bases for meeting the ATWS rule.

Q = 24 gpm
C = 10.7 wt%
M = 400000 lbs
E = 55.0

(24 / 86) x (628300 / 400000) X (10.7 / 13) X (55.0 / 19.8) > 1

1.0022 > 1

Conclusion

The NRC staff has reviewed the licensee's analyses related to the effects of the proposed EPU on the SLC system and concludes that the licensee has adequately accounted for the effects of the proposed EPU on the system and demonstrated that the system will continue to provide the function of reactivity control independent of the control rod system following implementation of the proposed EPU.

Based on these considerations, the NRC staff concludes that the SLC system will continue to meet the requirements of draft GDCs 27, 29 and 30, current GDC-26, and 10 CFR 50.62(c)(4) following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the SLC system.

2.8.4.6 Reactor Recirculation System Performance

The licensee provided Section 2.8.4.6, "Reactor Recirculation System Performance," which is evaluated in other sections of this SE, as appropriate.

2.8.5 Accident and Transient Analyses

Anticipated operational occurrences (AOOs) are abnormal transients which are expected to occur one or more times in the life of a plant and are initiated by a malfunction, a single failure of equipment, or a personal error. The applicable acceptance criteria for the AOOs are based on 10 CFR Part 50, Appendix A, as follows:

1. GDC-10, "Reactor design," insofar as it requires that the reactor core and associated coolant, control, and protection systems to be designed with appropriate margin to

assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including effects of AOOs;

2. GDC-15, "Reactor coolant system design," insofar as it requires that the reactor core and associated auxiliary, control, and protection systems to be designed with sufficient margin to assure that design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including of AOOs; and

3. GDC-20, "Protection system functions," insofar as it requires that 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 AOOs; and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

The Standard Review Plan (SRP) provides further guidelines that:

1. Pressure in the reactor coolant and main steam system should be maintained below 110 percent of the design values according to the ASME Code, Section III, Article NB-7000, "Overpressure Protection";

2. Fuel cladding integrity should be maintained to ensure that SAFDL are not exceeded during normal operating conditions and AOOs;

3. An incident of moderate frequency should not generate a more serious plant condition unless other faults occur independently; and

4. An incident of moderate frequency, in combination with any single active component failure or single operator error, should not result in the loss of function of any fission product barrier other than fuel cladding. A limited number of fuel cladding perforations are acceptable.

Chapter 14 of the MNGP UFSAR identifies eight parameters which are used to evaluate the effects of AOOs: (1) nuclear system pressure increase; (2) reactor vessel water (moderator) temperature decrease; (3) positive reactivity insertion; (4) reactor vessel coolant inventory decrease; (5) reactor core coolant flow decrease; (6) reactor core coolant flow increase; (7) core coolant temperature increase; and (8) excess of coolant inventory. To determine the plant system disturbances caused by single operator error or a single equipment malfunction, the initial MNGP FSAR analyzed 16 transients, each relating to one of the above parameters. Even though the consequences of each transient would be cycle-specific, subsequent reload analyses did not require each of the 16 transients to be re-analyzed as most of the 16 transients resulted in a fairly mild plant disturbance and only a small subset were found to be potentially limiting. However, those transients found to be potentially limiting would have to be re-analyzed for each re-load analysis.

The transients found most potentially limiting were those which involved significant change in power. Large power changes were determined to have the most significant effect on MCPR. Those transients are given as follows: (1) [[ ]]; (2) [[ ]]; (3) [[ ]]; (4) generator load rejection without bypass; (5) turbine trip without bypass; and (6) rod withdrawal error. These same limiting transients are part of those identified in Appendix E, Table E-1 of

ELTR1.

2.8.5.1 Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Main Steam Relief or Safety Valve

Regulatory Evaluation

Excessive heat removal causes a decrease in moderator temperature which increases core reactivity and can lead to a power level increase and a decrease in shutdown margin. Any unplanned power level increase may result in fuel damage or excessive reactor system pressure. Reactor protection and safety systems are actuated to mitigate the transient.

The NRC staff's review covered the following:

1. Postulated initial core and reactor conditions,

2. Methods of thermal and hydraulic analyses,

3. The sequence of events,

4. Assumed reactions of reactor system components,

5. Functional and operational characteristics of the reactor protection system,

6. Operator actions, and

7. The results of the transient analyses.

The NRC's acceptance criteria are based on the following:

1. Draft GDC-6 and current GDC-10, "Reactor design," insofar as they require that the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs;

2. Draft GDC-14 and 15, insofar as they require that the core protection system be designed to act automatically to prevent or suppress conditions that could result in exceeding acceptable fuel damage limits and that protection systems be provided for sensing accident situations and initiating the operation of necessary ESFs;

3. Draft GDC-27 and 28, insofar as they require that at least two reactivity control systems be provided and be capable of making and holding the core subcritical from any hot standby or hot operating condition sufficiently fast to prevent exceeding acceptable fuel damage limits;

4. Draft GDC-29, insofar as it requires that at least one of the reactivity control systems be capable of making the core subcritical under any condition sufficiently fast to prevent exceeding acceptable fuel damage limits; and

5. Draft GDC-30, insofar as it requires that at least one of the reactivity control systems provided shall be capable of making and holding the core subcritical under any conditions with appropriate margins for contingencies.

The licensee also stated that current GDC-15, "Reactor coolant system design," GDC-20, "Protection system functions," and GDC-26, "Reactivity control system redundancy and capability," are applicable as described in USAR Section 14.4. Specific review criteria are contained in SRP Section 15.1.1-4 and guidance provided in Matrix 8 of RS-001.

Technical Evaluation

The limiting events for the decrease in feedwater temperature are [[

]]. The limiting event for the increase in feedwater flow is the [[
]]. The licensee confirmed that each of these events is within the MNGP reload evaluation scope. As such, the licensee [[

]].

In Enclosure 5 of the licensee's November 5, 2008, application, the licensee provided a table of methods used for analysis, and confirmed that the same methods were used for transient analyses as those discussed in ELTR1; however, ELTR1 does not specifically mention the analytic method used to analyze the loss of feedwater heating (LOFWH). The NRC staff issued RAI-2.8.5.1-1 to determine which computer code or method was used to analyze the LOFWH event. The licensee responded to RAI-2.8.5.1-1 (Reference 5) by stating that the LOFWH event will be analyzed in the cycle-specific reload licensing analyses using the methods described in GESTAR II. The licensee stated further that the computer code used is PANACEA. This information confirms that the 3D simulator, listed in ELTR1 table, is indeed an NRC-approved computer code (PANACEA). The licensee's response clarifies that the LOFWH analysis is performed using NRC-approved codes and methods, and the response is, therefore, acceptable.

In RAI-2.8.5.1-2, the NRC staff requested that the licensee evaluate the LOFWH transient at EPU conditions to confirm acceptance criteria relative to fuel thermal-mechanical performance. The licensee responded (Reference 5) by stating that this analysis was performed, and confirmed that the results show acceptable results with respect to the fuel centerline melt and 1-percent cladding strain limits with more than 10-percent margin. The results are further discussed in the licensee's response to RAI-2.8.3-10, in Section 2.8.7.2 of this SE.

The limiting increase in steam flow event, according to the licensee, is [[

]]. Therefore, the increase in steam flow event was [[
]], and is [[ ]] within the reload evaluation scope.

An inadvertent safety relief valve opening is a non-limiting transient that results in a very slight increase in reactor power. The steam pressure regulator remains in service during the transient, which controls the reactor pressure. This event is not analyzed for the EPU.

In summary, the licensee applied the CLTR generic disposition for each event in the excessive heat removal category. The limiting events are within reload evaluation scope, and need not specifically be evaluated for the requested EPU. The NRC staff has accepted this disposition, consistent with the approach set forth in the CLTR. The licensee will perform plant-specific reload analyses, using NRC-approved methods, to confirm that fuel design limits and RCPB pressure limits will not be exceeded under EPU conditions for this class of transients.

Based on the above, the NRC staff accepts the licensee's disposition of the excessive heat removal transients.

Conclusion

The NRC staff has reviewed the licensee's disposition regarding the excess heat removal events described above and concludes that the licensee's disposition has adequately accounted for operation of the plant at the proposed power level and is based on acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the SAFDLs and the RCPB pressure limits will not be exceeded as a result of these events. Based on these considerations, the NRC staff concludes that MNGP will continue to meet the requirements of AEC proposed General Design Criteria 6, 14, 15, 27, 28, 29 and 30, and current GDC-10, 15, 20 and 26 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the events stated.

2.8.5.2 Decrease in Heat Removal by the Secondary System

2.8.5.2.1 Loss of External Load; Turbine Trip; Loss of Condenser Vacuum; Closure of Main Steam Isolation Valve; and Steam Pressure Regulator Failure (Closed)

Regulatory Evaluation

A number of initiating events may result in an unplanned increase in reactor pressure and decrease in heat removal from the core. These events result in a sudden reduction in steam flow and, consequently, result in pressurization events. Reactor protection and safety systems are actuated to mitigate the transient.

The NRC staff's review covered the following:

1. The sequence of events,

2. The analytical models used for analyses,

3. The values of parameters used in the analytical models, and

4. The results of the transient analyses.

The NRC's acceptance criteria are based on the following:

1. Draft GDC-6 and current GDC-10, "Reactor design," insofar as they require that the reactor core and associated coolant, control, and protection systems be designed with

appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs;

2. Draft GDC-27 and 28, insofar as they require that at least two reactivity control systems be provided and be capable of making and holding the core subcritical from any hot standby or hot operating condition sufficiently fast to prevent exceeding acceptable fuel damage limits;

3. Draft GDC-29, insofar as it requires that at least one of the reactivity control systems be capable of making the core subcritical under any condition sufficiently fast to prevent exceeding acceptable fuel damage limits; and

4. Draft GDC-30, insofar as it requires that at least one of the reactivity control systems provided shall be capable of making and holding the core subcritical under any conditions with appropriate margins for contingencies.

The licensee also stated that current GDC-15, "Reactor coolant system design," GDC-20, "Protection system functions," and GDC-26, "Reactivity control system redundancy and capability," are applicable to MNGP as described in USAR Section 14.4. Specific review criteria are contained in SRP Section 15.2.1-5 and guidance provided in Matrix 8 of RS-001.

Technical Evaluation

The transients evaluated in this group included the following:

- Loss of external load
- Turbine trip
- Loss of condenser vacuum
- Closure of main steam isolation valve
- Steam pressure regulator failure (closed)

The limiting events in the loss of external load and turbine trip categories are acceptable for generic disposition because they are within the MNGP reload evaluation scope. Specifically, NSPM will evaluate the generator load rejection with steam bypass failure (LRNBP) and the TTNBP as a part of the cycle-specific reload analysis process.

The licensee stated that for all BWRs, the loss of condenser vacuum (LOCV) event is [[

]]. By comparison, the NRC staff expects that the LOCV transient would result in a milder pressurization due to the availability of the turbine bypass system, and hence agrees with the licensee's disposition. The NRC staff finds that the LOCV need not be analyzed for the EPU because it is bounded by the TTNBP event, and is not within the MNGP reload evaluation scope. This is consistent with the CLTR generic disposition, which the NRC staff finds acceptable.

The limiting main steam isolation valve (MSIV) closure event is one with failure of direct scram. This transient is analyzed in support of the requested EPU, as evaluated by the NRC staff in Section 2.8.4.2 of this SE.

Because MNGP is a BWR/3, the pressure regulator failure need not be analyzed. This is because, as stated by the licensee, this event is **[[ ]]**. The NRC staff has previously accepted this disposition as indicated in the SER approving the CLTR and, therefore, it is considered acceptable for the MNGP EPU.

Conclusion

The NRC staff has reviewed the licensee's analyses of the decrease in heat removal events described above and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the specified acceptable fuel design limits and the RCPB pressure limits will not be exceeded as a result of these events.

Based on these considerations, the NRC staff concludes that MNGP will continue to meet the intent of the proposed draft design criteria 6, 27, 28, 29 and 30, and current GDC-10, 15, 20 and 26, following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the events stated.

2.8.5.2.2 Reactor Vessel Coolant Inventory Decrease - Loss of Auxiliary Power

Regulatory Evaluation

The loss of nonemergency AC power is assumed to result in the loss of all power to the station auxiliaries and simultaneous tripping of both reactor coolant pumps. This causes a flow coast down as well as a decrease in heat removal by the secondary system, a turbine trip, an increase in pressure and temperature of the coolant, and a reactor trip. Reactor protection and safety systems are actuated to mitigate the transient.

The NRC staff's review covered the following:

1. The sequence of events,
2. The analytical model used for analyses,
3. The values of parameters used in the analytical model, and
4. The results of the transient analyses.

The NRC's acceptance criteria are based on the following:

1. Draft GDC-6 and current GDC-10, "Reactor design," insofar as they require that the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs;
2. Draft GDC-27 and 28, insofar as they require that at least two reactivity control systems be provided and be capable of making and holding the core subcritical from any hot standby or hot operating condition sufficiently fast to prevent exceeding acceptable fuel damage limits;

3. Draft GDC-29, insofar as it requires that at least one of the reactivity control systems be capable of making the core subcritical under any condition sufficiently fast to prevent exceeding acceptable fuel damage limits; and

4. Draft GDC-30, insofar as it requires that at least one of the reactivity control systems provided shall be capable of making and holding the core subcritical under any conditions with appropriate margins for contingencies.

The licensee also indicated that current GDC-10, "Reactor design," GDC-15, "Reactor coolant system design," and GDC-26, "Reactivity control system redundancy and capability," are applicable to MNGP as described in USAR Section 14.4. Specific review criteria are contained in SRP Section 15.2.6 and guidance provided in Matrix 8 of RS-001.

Technical Evaluation

The reactor is subjected to a complex sequence of events when the station loses all auxiliary power. This can occur if all external grid connections are lost or if faults occur in the auxiliary power system itself. A Loss of Auxiliary Power to the Station Auxiliaries is a non-limiting event for all GE BWRs. The TTNBP event bounds this event because the loss of non-emergency AC power event has a delayed turbine trip with a recirculation pump trip. This event is not analyzed.

The load rejection and turbine trip events, both with failures of the bypass system, are addressed in Section 2.8.5.2.1 of this SE.

Conclusion

The NRC staff has reviewed the licensee's disposition regarding analysis of the loss of nonemergency AC power to the station auxiliaries event and concludes that the licensee's disposition adequately accounts for operation of the plant at the proposed power level and is based on analyses performed using acceptable analytical models. The NRC staff further concludes that the license has demonstrated that the reactor protection and safety systems will continue to ensure that the specified acceptable fuel design limits and the RCPB pressure limits will not be exceeded as a result of this event. Based on these considerations, the NRC staff concludes that the plant will continue to meet the intent of proposed draft criteria 6, 27, 28, 29 and 30, and current GDC-10, 15 and 26, following implementation of the proposed EPU. Therefore, the NRC staff finds that the proposed EPU is acceptable with respect to the event associated with a loss of non-emergency AC power to the station auxiliaries.

2.8.5.2.3 Loss of Normal Feedwater Flow

Regulatory Evaluation

A loss of normal feedwater flow could occur from pump failures, valve malfunctions, or a loss-of-off-site power (LOOP). Loss of feedwater flow results in an increase in reactor coolant temperature and pressure which eventually requires a reactor trip to prevent fuel damage. Decay heat must be transferred from the fuel following a loss of normal feedwater flow. Reactor protection and safety systems are actuated to provide this function and mitigate other aspects of the transient.

The NRC staff's review covered the following:

1. The sequence of events,
2. The analytical model used for analyses,
3. The values of parameters used in the analytical model, and
4. The results of the transient analyses.

The NRC's acceptance criteria are based on the following:

1. Draft GDC-6 and current GDC-10, "Reactor design," insofar as they require that the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs;
2. Draft GDC-27 and 28, insofar as they require that at least two reactivity control systems be provided and be capable of making and holding the core subcritical from any hot standby or hot operating condition sufficiently fast to prevent exceeding acceptable fuel damage limits;
3. Draft GDC-29, insofar as it requires that at least one of the reactivity control systems be capable of making the core subcritical under any condition sufficiently fast to prevent exceeding acceptable fuel damage limits; and
4. Draft GDC-30, insofar as it requires that at least one of the reactivity control systems provided shall be capable of making and holding the core subcritical under any conditions with appropriate margins for contingencies.

Specific review criteria are contained in SRP Section 15.2.7 and guidance provided in Matrix 8 of RS-001.

<u>Technical Evaluation</u>

Feedwater Control System failures or reactor feedwater pump trips can lead to partial or complete LOFW flow. A LOFW results in a situation where the mass of steam leaving the reactor vessel exceeds the mass of water entering the vessel, resulting in a decrease in the coolant inventory available to cool the core. Consistent with dispositions provided in Enclosure 5 of the licensee's November 5, 2008, application, this loss-of-level event is evaluated on a plant-specific basis to assure that, for the higher decay heat load, coolant inventory remains to provide adequate core coverage (Reference 7).

The licensee performed a calculation to support the PUSAR with a representative equilibrium core for LOFW flow. The NRC staff requested that the licensee identify the analytic methods used to analyze this transient (RAI-2.8.5.2-1) and address more specifically the EPU conditions analyzed (RAI-2.8.5.2-2; RAI-2.8.4.3-4).

In its response to RAI-2.8.5.2-1 (Reference 5), the licensee stated that, consistent with the CLTR approach, the SAFER04 model was used to model the LOFW analysis. This approach was approved by the NRC staff and, therefore, the licensee's response is acceptable.

In its response to RAI-2.8.5.2-2/2.8.4.3-4 (Reference 5), the licensee stated that the analysis was performed assuming operation at 102-percent of the EPU power level when a LOFW occurs. The model assumes that initial level is at the low-level scram setpoint, and reactor feedwater is instantaneously isolated at initiation of the event. On level decrease to the low-low level setpoint, the RCIC system and MSIV closure are initiated. Only RCIC flow is credited to recover reactor water level. NSPM stated that, specific to the MNGP analysis, the LOFW analysis for EPU uses the ANS 5.1-1979 decay heat model with an additional 10 percent uncertainty. This assumption bounds the generic approach, which assumes 2σ uncertainty.

The results of the analysis demonstrate that the RCIC system, under LOFW conditions at the EPU power level, can maintain minimum reactor water level throughout the transient greater than 77 inches above the top of active fuel.

The increased decay heat due to EPU operation requires more time for the automatic systems to restore water level. This analysis assumed failure of the HPCI system and used only the RCIC system to restore the reactor water level. The reactor level is automatically maintained above the top of the active fuel without any operator action. The results of the LOFW analysis show that the minimum water level inside the core shroud is 77 inches above the top of the fuel. Because the licensee's analysis shows that an acceptable core water level is maintained, the NRC staff finds the licensee's analysis acceptable.

The licensee also evaluated operator actions that are required in this event. These actions include manual control of the water level, reduction of reactor pressure, and initiation of RHR shutdown cooling. The licensee stated that the transient requires no new operator actions or shorter operator responses times. Because there is no significant change in operator actions required by the EPU, the NRC staff finds the licensee's evaluation of the operator actions acceptable.

Regarding the loss of a single feedwater pump, the licensee invoked the CLTR generic disposition. The licensee stated that the loss of a single feedwater pump addresses operational considerations to avoid reactor scram on low water level. The generic disposition is acceptable for the loss of a single feedwater pump, provided that **[[**

]]. This is because this transient results in a reduction in total core flow, causing the power to coast down along the flow control line within the analyzed power-to-flow operating domain. The NRC staff finds the licensee's disposition of the loss of a single feedwater pump event acceptable.

<u>Conclusion</u>

The NRC staff reviewed the licensee's analyses and dispositions of the decrease in reactor coolant flow events and concludes that the licensee's analyses and dispositions have adequately accounted for operation of the plant at the proposed power level and were either based on or performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the specified acceptable fuel design limits and the RCPB pressure limits will not be exceeded as a result of this event. Based on these considerations, the NRC staff concludes that MNGP will continue to meet the intent of draft design criteria 6, 27, 28, 29, and 30. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the loss of normal feedwater flow event.

2.8.5.3 Decrease in Reactor Coolant System Flow

2.8.5.3.1 Loss of Forced Reactor Coolant Flow

Regulatory Evaluation

A decrease in reactor coolant flow occurring while the plant is at power could result in a degradation of core heat transfer. An increase in fuel temperature and accompanying fuel damage could then result if SAFDLS are exceeded during the transient. Reactor protection and safety systems are actuated to mitigate the transient.

The NRC staff's review covered the following:

1. The postulated initial core and reactor conditions,

2. The methods of thermal and hydraulic analyses,

3. The sequence of events,

4. Assumed reactions of reactor systems components,

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

6. Operator actions, and

7. The results of the transient analyses.

The NRC's acceptance criteria are based on the following:

1. Draft GDC-6 and current GDC-10, "Reactor design," insofar as they require that the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs;

2. Draft GDC-27 and 28, insofar as they require that at least two reactivity control systems be provided and be capable of making and holding the core subcritical from any hot standby or hot operating condition sufficiently fast to prevent exceeding acceptable fuel damage limits. Specific review criteria are contained in SRP Section 15.3.1-2 and guidance provided in Matrix 8 of RS-001;

3. Draft GDC-29, insofar as it requires that at least one of the reactivity control systems be capable of making the core subcritical under any condition sufficiently fast to prevent exceeding acceptable fuel damage limits; and

4. Draft GDC-30, insofar as it requires that at least one of the reactivity control systems provided shall be capable of making and holding the core subcritical under any conditions with appropriate margins for contingencies.

Technical Evaluation

Events in this group include Recirculation Flow Control Failure, Trip of One Recirculation Pump and Trip of Two Recirculation Pumps. Several varieties of recirculation flow control malfunctions can cause a decrease in core coolant flow. Although the manual loading station output values are adjustable based on selectable high and low limits, it could malfunction in such a way that a zero speed signal is generated for both recirculation flow control loops. This scenario is no more severe than the simultaneous trip of both recirculation pumps.

Normal trip of one recirculation loop is caused by the drive motor breaker. This transient is bounded by the trip of two recirculation pumps.

When the drive motor breakers are tripped, the motor-generators will continue to supply some reduced power to their respective recirculation pump motors, due to the time required for the motor-generator sets to coast down. As the core flow decreases, additional voids will be formed, causing a decrease in reactor power. Reactor power will decrease approximately 50 percent within a short time. The time constants of the fuel will cause thermal power to lag behind the neutron flux and core flow decay and the mismatch between reactor thermal power and recirculation flow results in a decrease in CPR. The MCPR would reach its lowest value in a very short time, but would not reach the MCPR safety limit. The fuel thermal margin is provided, in part, by the rotating inertia of the motor-generator sets.

Analyses performed for several BWRs have shown that the events in this category are not limiting events and are bounded by the more limiting transients, which, as the licensee stated, [[ ]].

Conclusion

The NRC staff has reviewed the licensee's disposition regarding the decrease in reactor coolant flow event and concludes that the licensee's disposition adequately accounts for operation of MNGP at the proposed EPU level. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the specified acceptable fuel design limits and the RCPB pressure limits will not be exceeded as a result of this event. Based on these considerations, the NRC staff concludes that MNGP will continue to meet the intent of proposed draft criteria 6, 27, 28, 29 and 30. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the decrease in reactor coolant flow event.

2.8.5.3.2 Recirculation Pump Rotor Seizure and Reactor Coolant Pump Shaft Break

Regulatory Evaluation

The event postulated is an instantaneous seizure of the rotor or break of the shaft of a recirculation pump. Flow through the affected loop is rapidly reduced, leading to a reactor and turbine trip. The sudden decrease in core coolant flow while the reactor is at power results in a degradation of core heat transfer which could result in fuel damage. The initial rate of reduction of coolant flow is greater for the rotor seizure event. However, the shaft break event permits a greater reverse flow through the affected loop later during the transient and, therefore, results in a lower core flow rate at that time. In either of these cases, reactor protection and safety systems are actuated to mitigate the transient.

The NRC staff's review covered the following:

1. The postulated initial and long-term core and reactor conditions,
2. The methods of thermal and hydraulic analyses,
3. The sequence of events,
4. The assumed reactions of reactor system components,
5. The functional and operational characteristics of the reactor protection system,
6. Operator actions, and
7. The results of the transient analyses.

The NRC's acceptance criteria are based on the following:

1. GDC-10, "Reactor design," insofar it requires that the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs;
2. Draft GDC-32, insofar as it requires that limits, which include considerable margin, be placed on the maximum reactivity worth of control rods or elements and on rates at which reactivity can be increased to ensure that the potential effects of a sudden or large change of reactivity cannot
    a. Rupture the reactor coolant pressure boundary, or
    b. Disrupt the core, its support structures, or other vessel internals sufficiently to impair the effectiveness of emergency core cooling; and
3. Draft GDC-33, 34, and 35, insofar as they require that the RCPB be designed to assure that it behaves in a non-brittle manner and that the probability of rapidly propagating type failures is minimized.

<u>Technical Evaluation</u>

The recirculation pump rotor seizure and shaft break events are design-basis accidents. Flow through the affected loop is rapidly reduced, leading to a reactor and turbine trip. The sudden decrease in core flow results in a degradation of core heat transfer; however, core uncovery is not expected during this accident.

Generic analyses performed for several BWRs have shown that the accidents in this category are not limiting events and are bounded by the more limiting accidents and, hence, these accidents are not included in the reload analyses. The licensee stated that **[[**

]].

Also, the MNGP USAR (Section 14.7.5) states that the one recirculation pump seizure accident was evaluated on a cycle-independent basis for MNGP against the acceptance criteria for plant transients (i.e., AOOs). Appendix A to Chapter 14 of the USAR is a cycle-specific safety analysis report, which indicates that the single recirculation pump seizure was evaluated for the Cycle 24 reload.

The information contained both in the PUSAR and in the MNGP USAR differs slightly from the information provided in GE's response to RAI-Set 9, Number 14, from the CLTR. The information in the CLTR identifies those transients which are within the scope of the plant-specific reload, and those which are evaluated for power uprate applications. The CLTR RAI-response indicates that [[ ]]. The information provided by NSPM appeared to contradict the information contained in the CLTR, and the NRC staff issued an RAI-for clarification (RAI-2.8.5.3-1). The NRC staff requested that the licensee explain whether the analysis discussed in the PUSAR was specific to MNGP, or generic. The NRC staff also requested that the licensee explain why the accident is discussed for the MNGP uprate request, but not in the CLTR. Finally, the NRC staff also requested that the licensee explain the apparent difference between the USAR information and the PUSAR information (specifically, the PUSAR statement that [[ ]]).

The licensee stated in its response (Reference 5) that the analysis discussed in the PUSAR is one performed to support the initial loading of GE14 fuel in the MNGP core for Cycle 22. The licensee also stated that the accident was evaluated to provide a cycle-independent MCPR limit, which is reviewed on a cycle-by-cycle basis to ensure that the limit continues to be met.

The licensee clarified that the analysis was discussed to provide a MNGP-specific supplement to the generic information that is available in the PUSAR, and that, because only the single pump seizure from single loop operation was re-analyzed at MNGP, the cited disposition remains applicable. Technically, this is because the initial condition for the transient is bounded by the Maximum Extended Load Line Limit Analysis (MELLLA) boundary at the maximum single-loop operation core flow, and these two boundaries are not changed for the EPU.

In evaluating the licensee's response to RAI-2.8.5.3-1, the NRC staff first considered that the licensee re-evaluates the single RCP rotor seizure/shaft break from SLO conditions to determine that cycle-independent MCPR limits do not need to be changed. The NRC staff then considered that the transient itself is reasonably unaffected by EPU implementation, since the transient is evaluated for MCPR performance, and the initial conditions of core flow and maximum rod line are unchanged for the transient. Based on these two considerations, the NRC staff accepts the licensee's response.

<u>Conclusion</u>

The NRC staff has reviewed the licensee's dispositions and analyses of the sudden decrease in core coolant flow events and concludes that the licensee's dispositions and analyses have adequately accounted for operation of MNGP at the proposed power level, and based on or performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the ability to insert control rods is maintained, the RCPB pressure limits will not be exceeded, the RCPB will behave in a non-brittle manner, the probability of propagating fracture of the RCPB is minimized, and adequate core cooling will be provided.

Note that the NRC staff's assurance that MNGP will meet accident acceptance criteria for this event is based on the fact that it has been demonstrated that these events can be sustained and meet AOO acceptance criteria, which are more stringent than the accident acceptance criteria listed in this evaluation.

Based on this, the NRC staff concludes that MNGP will continue to meet the intent of GDC-10 and proposed draft criteria 32-35 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the sudden decrease in core coolant flow events.

2.8.5.4 Reactivity and Power Distribution Anomalies

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

Regulatory Evaluation

An uncontrolled control rod assembly withdrawal from subcritical or low-power startup conditions may be caused by a malfunction of the reactor control or rod control systems. This withdrawal will uncontrollably add positive reactivity to the reactor core, resulting in a power excursion.
The NRC staff's review covered:

1. The description of the causes of the transient and the transient itself,
2. The initial conditions,
3. The values of reactor parameters used in the analysis,
4. The analytical methods and computer codes used, and
5. The results of the transient analyses.

The NRC's acceptance criteria are based on:

1. Draft GDC-6 and current GDC-10, "Reactor design," insofar as they require that the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs;

2. Draft GDC-14 and 15, insofar as they require that the core protection systems be designed to act automatically to prevent or suppress conditions that could result in exceeding acceptable fuel damage limits and that protection systems be provided for sensing accident situations and initiating the operation of necessary ESFs; and

3. Draft GDC-31, insofar as it requires that the reactivity control systems be capable of sustaining any single malfunction without causing a reactivity transient which could result in exceeding acceptable fuel damage limits.

The licensee indicated that that current GDC-20, "Protection system functions," and GDC-25, "Protection system requirements for reactivity control malfunctions," are applicable to MNGP as described in USAR Section 14.4. Specific review criteria are contained in SRP Section 15.4.1 and guidance provided in Matrix 8 of RS-001.

Technical Evaluation

The uncontrolled control rod withdrawal event at subcritical or low power startup conditions is a localized, low power event. The [[

]]. If the peak fuel rod enthalpy calculated for this event at the originally licensed power is increased by a factor of 1.2 to take into account EPU conditions, the peak fuel enthalpy calculated for the uncontrolled rod withdrawal event at subcritical or low power conditions is 72 cal/g. However, because the transient is a localized, low-power event, and because the RWM affords the same protection as in the pre-EPU plant design, this scaling treatment is conservative. Also, this scaled value remains below the acceptance criterion of 170 cal/g.

In the course of its review, the NRC staff communicated to the licensee RAIs regarding the evaluation of the uncontrolled control rod withdrawal event at zero or low power conditions. The staff's evaluation of the licensee's responses appears in the following paragraphs. RAI-2.8.5.4-1 identified an incorrectly referenced document, and the licensee clarified in Reference 5 that the correct Reference is NEDO-23842, "Continuous Control Rod Withdrawal Transient in the Startup Range." This report provides the correct, generic disposition of the subject transient. Therefore, the information that the licensee provided is responsive to the NRC staff's request and hence is acceptable.

The NRC staff requested, in RAI-2.8.5.4-2, that the licensee identify the analytical methods used to analyze the rod withdrawal error (RWE) events, both at-power and from a low-power or subcritical condition. The licensee responded that the startup analysis is performed as described in NEDO-23842 (discussed above). The methods applied are consistent with those used also for the control rod drop accident analysis described in Licensing Topical Report NEDO-10527, "Control Rod Drop Accident Analysis for Large Boiling Water Reactors," March 1972.

The licensee explained that the generic analysis remains applicable to EPU conditions at MNGP because the RWE at startup is a localized, low-power event. Also, generic core design and operational strategies are employed at boiling water reactors to minimize the reactivity worth of any single control rod, meaning, as the licensee stated, [[

]].
While the licensee acknowledged that EPU fuel and core designs can lead to a generally higher rod worth distribution, and therefore, a higher peak fuel enthalpy at low power, [[
]]. Finally, the licensee stated that increasing the peak fuel enthalpy by a factor of 1.2 as an approach to account conservatively for the EPU effects and yielded a startup

RWE fuel enthalpy of 72 cal/g, a value significantly below the acceptance criterion of 170 cal/g.

The licensee also stated that the startup RWE analysis is performed using PANACEA, Version 11. The NRC staff evaluated the licensee's response, and found it acceptable based on the following two considerations. First, the response is acceptable because it identifies the method used to evaluate the transient, and the method is acceptable. Second, the licensee provided further justification regarding the EPU applicability of the analysis and its results:

- Limitations on in-sequence rod worths and shutdown margin serve to limit peak fuel enthalpy on the startup RWE; this consideration does not change for the EPU; and
- Increasing the peak fuel enthalpy by a factor of 1.2 still leaves significant margin to the licensing limit for this transient.

Conclusion

The NRC staff has reviewed the disposition of the uncontrolled rod withdrawal event at subcritical or low-power conditions presented in the MNGP EPU application, as supplemented, and concludes that the information presented pertaining to this event is consistent with the expectations delineated in the SER associated with the CLTR. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that SAFDLs and RCPB pressure limits will not be exceeded as a result of these events. Based on these considerations, the NRC staff concludes that MNGP will continue to meet the intent of draft GDC-6, 14, 15, and 31, and current GDC-10, 20, and 25, following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the uncontrolled control rod assembly withdrawal from a subcritical or low-power startup condition.

2.8.5.4.2 Continuous Control Rod Withdrawal during Power Range Operation

Regulatory Evaluation

An uncontrolled control rod assembly withdrawal at power may be caused by a malfunction of the reactor control or rod control systems. This withdrawal will uncontrollably add positive reactivity to the reactor core, resulting in a power excursion. The NRC staff's review covered the consistency of the licensee's disposition of the uncontrolled control rod assembly withdrawal at power with the generic disposition approved in the CLTR SER.

The NRC's acceptance criteria are based on:

1. Draft GDC-6 and current GDC-10, insofar as they require that the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs;
2. Draft GDC-14 and 15, insofar as they require that the core protection systems be designed to act automatically to prevent or suppress conditions that could result in exceeding acceptable fuel damage limits and that protection systems be provided for sensing accident situations and initiating the operation of necessary ESFs; and

3. Draft GDC-31, insofar as it requires that the reactivity control systems be capable of sustaining any single malfunction without causing a reactivity transient which could result in exceeding acceptable fuel damage limits.

The licensee indicated that current GDC-20 and 25 are applicable to MNGP as described in USAR Section 14.4. Specific review criteria are contained in SRP Section 15.4.2 and guidance provided in Matrix 8 of RS-001.

Technical Evaluation

While operating in the power range, it is assumed that the reactor operator makes a procedural error and fully withdraws the maximum worth control rod. Due to the positive reactivity insertion, the core average power increases. If the rod withdrawal error (RWE) is severe enough, the RBM will sound alarms, at which time the operator will take corrective actions. Even for extremely severe conditions i.e., for highly abnormal control rod patterns, operating conditions, and assuming that the operator ignores all the alarms and warnings and continues to withdraw the control rod), the fuel cladding integrity safety limit (i.e., MCPR) and fuel rod mechanical overpower limits will not be exceeded.

The NRC staff has reviewed MNGP's disposition of the RWE, and agrees with the assessment that RWE analysis is within the MNGP reload scope as defined by the SER associated with the CLTR. This disposition is acceptable for the following reasons: [[

]]. It is expected that this analysis will be carried out with NRC staff-approved methods and codes and the results documented in the SRLR.

Conclusion

The NRC staff has reviewed the disposition of the uncontrolled control rod withdrawal error at power presented in the MNGP EPU application, and concludes that the content of the application pertaining to this event are consistent with the expectations delineated in the SER associated with CLTR. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that SAFDLs and RCPB pressure limits will not be exceeded as a result of these events. Based on this, the NRC staff concludes that MNGP will continue to meet the intent of draft GDC-6, 14, 15, and 31, and current GDC-10, 20, and 25, following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the uncontrolled control rod withdrawal error at power.

2.8.5.4.3 Core Coolant Flow Increase, Startup of Idle Recirculation Pump, Recirculation Flow Controller Failure

Regulatory Evaluation

A transient due to startup of an inactive loop may result in either an increased core flow or the introduction of cooler water into the core. This event causes an increase in core reactivity due to decreased moderator temperature and core void fraction.

The NRC staff's review covered the consistency of the licensee's disposition of the uncontrolled control rod assembly withdrawal at power with the disposition approved in the CLTR SER.

The NRC's acceptance criteria are based on:

1. Draft GDC-6 and current GDC-10, "Reactor design," insofar as they require that the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs;

2. Draft GDC-14 and 15, insofar as they require that the core protection systems be designed to act automatically to prevent or suppress conditions that could result in exceeding acceptable fuel damage limits and that protection systems be provided for sensing accident situations and initiating the operation of necessary ESFs;

3. Draft GDC-27, insofar as it requires that at least two independent reactivity control systems, preferably of different principles, be provided;

4. Draft GDC-29, insofar as it requires that at least one of the reactivity control systems provided shall be capable of making the core subcritical under any condition, including anticipated operational transients, sufficiently fast to prevent exceeding acceptable fuel damage limits;

5. Draft GDC-30, insofar as it requires that at least one of the reactivity control systems provided shall be capable of making and holding the core subcritical under any conditions with appropriate margins for contingencies;

6. Draft GDC-32, insofar as it requires that limits, which include considerable margin, be placed on the maximum reactivity worth of control rods or elements and on rates at which reactivity can be increased to ensure that the potential effects of a sudden or large change of reactivity cannot:

   a. Rupture the reactor coolant pressure boundary, or;

   b. Disrupt the core, its support structures, or other vessel internals sufficiently to impair the effectiveness of emergency core cooling.

The licensee indicated that current GDC-15, "Reactor coolant system design," GDC-20, "Protection system functions," and GDC-26, "Reactivity control system redundancy and capability," are also applicable to MNGP as described in USAR Section 14.4. Specific review criteria are contained in SRP Section 15.4.4-5 and guidance provided in Matrix 8 of RS-001.

<u>Technical Evaluation</u>

Events in this category include recirculation flow controller failure leading to increasing flow, and start-up of idle recirculation pump. Failure of the controller can result in fast recirculation increase. This event is non-limiting.

Start-up of an idle recirculation pump is a non-limiting transient for GE BWRs that have the APRM/RBM/TS (ARTS) plant performance option.

The CLTR provides the basis for disposing this event, based on the fact that [[

]].

The disposition of the increase in core flow events is confirmed because the MELLLA domain is unchanged with the EPU.

Conclusion

The NRC staff has reviewed the disposition of the recirculation flow increase events presented in the MNGP EPU application, and concludes that the content of the application pertaining to this event are consistent with the expectations delineated in the SER associated with CLTR. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that SAFDLS and RCPB pressure limits will not be exceeded as a result of these events. Based on this, the NRC staff concludes that MNGP will continue to meet the intent of draft GDC-6, 14, 15, 27, 29, 30, and 32, and current GDC-10, 15, 20, and 26, following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the recirculation flow increase events.

2.8.5.4.4 Control Rod Drop Accident

Regulatory Evaluation

The NRC staff evaluated the consequences of a control rod drop accident (CRDA) in the area of reactor physics. The NRC staff's review covered the occurrences that lead to the accident, 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 draft GDC-32, insofar as it requires that limits, which include considerable margin, be placed on the maximum reactivity worth of control rods or elements and on rates at which reactivity can be increased to ensure that the potential effects of a sudden or large change of reactivity cannot: (a) rupture the reactor coolant pressure boundary; or (b) disrupt the core, its support structures, or other vessel internals sufficiently to impair the effectiveness of emergency core cooling.

Specific review criteria are contained in SRP Sections 4.2 and 15.4.9 and guidance provided in Matrix 8 of RS-001.

Technical Evaluation

The CRDA is analyzed in Section 14.7.1 of the MNGP USAR.

The CRDA is a design-basis accident. The initiating event is separation of a control rod from its drive. The blade remains stuck in its channel while the drive is withdrawn under it. At some point the blade becomes unstuck and drops from the core at a speed controlled by the velocity limiter, adding reactivity and increasing power. High neutron flux trips the reactor protection system. The energy added heats up the fuel, producing negative Doppler feedback, which reduces power. Finally a scram shuts the reactor down.

According to Section 14.7.1.4 of the MNGP USAR, a bounding evaluation of the CRDA for all BWRs using the Banked Position Withdrawal Sequence (BPWS) has been performed and it was estimated that the resultant peak full enthalpy would be less than 157.8 cal/g, which is less than the 280 cal/g criterion.

The MNGP EPU application states [[

]].

No change in peak fuel enthalpy is expected due to the uprate itself; however, because of the fuel and core design changes necessary to sustain reactor operation at EPU conditions, there may be a resulting increase in control rod worth. Thus the reactivity insertion due to a CRDA would be higher, with a correspondingly higher energy deposition in the fuel (i.e., a higher peak fuel enthalpy).

The increase in rod worth as a result of core design changes necessary to operate under EPU conditions is not expected to cause an increase in peak fuel enthalpies excessively. If the peak fuel rod enthalpy determined for BPWS plants is increased by a factor of 1.2, the peak fuel rod enthalpy at EPU will be 162 cal/g. This value is below the acceptance criterion of 280 cal/g, and thus from a reactor physics standpoint the consequences of the CRDA are acceptable.

As a result of a fuel failure during a test at the CABRI reactor in France in 1993, and one in 1994 at the NSRR test reactor in Japan, the NRC recognized that high burnup fuel cladding might fail during a reactivity insertion accident (RIA), such as a Control Rod Drop event, at lower enthalpies than the limits currently specified in Section 4.2 of the 1981 Revision of the Standard Review Plan. However, analyses performed by all of the reactor vendors have indicated that the fuel enthalpy during RIAs will be much lower than the SRP 4.2 limits, based on their 3D neutronics calculations. For high burnup fuel which has been burned so long that it no longer contains significant reactivity, the fuel enthalpies calculated using the 3D models are expected to be much less than 100 cal/g.

The NRC staff has concluded that although the SRP 4.2 limits may not be conservative for cladding failure, the analyses performed by the vendors, which have been confirmed by NRC-sponsored calculations, provide reasonable assurance that the effects of postulated RIAs in operating plants with fuel burnups up to 60 gigawatt days per metric ton uranium will neither: (1) result in damage to the RCPB; nor (2) sufficiently disturb the core, its support structures, or other RPV internals to impair significantly the capability to cool the core as specified in current regulatory requirements.

The NRC staff requested that the licensee explain what methods and analytical codes are used to analyze the control rod drop accident (RAI-2.8.5.4-5). In Reference 5, the licensee responded by stating that the evaluation is based on the generic study contained in the original BPWS topical report, NEDO-21231. The licensee clarified further that the analysis was not updated because the CRDA is a localized, low-power event. The licensee also noted that EPU fuel and core designs can lead to generally higher rod worth distributions and therefore higher peak fuel enthalpy at low power. The licensee stated that [[

]] associated with the BPWS.

In RAI-2.8.4.5-6, the NRC staff requested that the licensee justify the conservatism of a 120 percent multiplier on peak fuel enthalpy for the CRDA. In response (Reference 5), the licensee reiterated the above information.

The licensee stated in the PUSAR (Reference 7) that if a conservative multiplier is applied to the peak fuel enthalpy at the current thermal power, the peak fuel enthalpy at EPU conditions will be 162 cal/g. The result reported in the MNGP USAR for the CLTP is 158 cal/g. The NRC staff requested that the licensee explain what peak fuel enthalpy value was being multiplied by 1.2 (RAI-2.8.5.4-7). The licensee responded (Reference 5) by stating that the 1.2 multiplier was applied to results obtained from the analysis contained in NEDO-21231, which predicted a peak fuel enthalpy of 132 cal/g. The result contained in the MNGP USAR, which is higher, is based on an upper bound enthalpy of a limiting worth derived from many CRDA calculations. The rod worths used in the USAR calculations exceed those acceptable from the BPWS and are, therefore, very conservative. Since MNGP is a BPWS plant, the NRC staff considers the licensee's response to be acceptable.

Conclusion

The NRC staff has reviewed the licensee's disposition of the control rod drop accident and concludes that the licensee's disposition has adequately taken into account the EPU impact on the control rod drop accident. Based on this consideration, the NRC staff concludes that MNGP will continue to meet the intent of draft GDC-32 following implementation of the EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the control rod drop accident.

2.8.5.5 Core Coolant Flow Increase, Inadvertent Operation of ECCS or Malfunction that Increases Reactor Coolant Inventory – Feedwater Controller Failure

Regulatory Evaluation

Equipment malfunctions, operator errors, and abnormal occurrences could cause unplanned increases in reactor coolant inventory. Depending on the temperature of the injected water and the response of the automatic control systems, a power level increase may result and, without adequate controls, could lead to fuel damage or overpressurization of the RCS. Alternatively, a power level decrease and depressurization may result. Reactor protection and safety systems are actuated to mitigate these events.

The NRC staff's review covered:

1. The sequence of events,

2. The analytical model used for analyses,

3. The values of parameters used in the analytical model, and

4. The results of the transient analyses.

The NRC's acceptance criteria are based on:

1. Draft GDC-6 and current GDC-10, "Reactor design," insofar as they require that the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs;

2. Draft GDC-27 and 28, insofar as they require that at least two reactivity control systems be provided and be capable of making and holding the core subcritical from any hot standby or hot operating condition sufficiently fast to prevent exceeding acceptable fuel damage limits;

3. Draft GDC-29, insofar as it requires that at least one of the reactivity control systems be capable of making the core subcritical under any condition sufficiently fast to prevent exceeding acceptable fuel damage limits; and

4. Draft GDC-30, insofar as it requires that at least one of the reactivity control systems provided shall be capable of making and holding the core subcritical under any conditions with appropriate margins for contingencies.

The licensee indicated that current GDC-15 and 26 are also applicable to MNGP as described in USAR Section 14.4. Specific review criteria are contained in SRP Section 15.5.1-2 and guidance provided in Matrix 8 of RS-001.

Technical Evaluation

Events in this category result in an increase in core coolant inventory. The increase in coolant inventory that is associated with these events also results in an increase in subcooling. One of the potentially limiting events in this category is the feedwater controller failure maximum demand, which the licensee addressed in Section 2.8.5.1 of the PUSAR. The NRC staff addresses this event in the corresponding section of this safety evaluation.

The other potentially limiting event in this category is the inadvertent actuation of the high pressure coolant injection system. This event is [[

]]. This is consistent with the NRC-approved generic disposition contained in the CLTR and, hence, is acceptable to the NRC staff.

Conclusion

The NRC staff has reviewed the licensee's disposition regarding the analysis of the inadvertent operation of emergency core cooling system (ECCS) or malfunction that increases reactor coolant inventory and concludes that the licensee's disposition adequately accounts for operation of MNGP at the proposed power level and is based on acceptable analytic models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the specified acceptable fuel design limits and the RCPB pressure limits will not be exceeded as a result of this event. Based on these considerations, the NRC staff concludes that MNGP will continue to meet the regulatory intent of proposed draft criteria 6, 27, 28, 29, and 30, and current GDC-10, 15, and 26, following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the inadvertent operation of ECCS or malfunction that increases reactor coolant inventory.

2.8.5.6 Decrease in Reactor Coolant Inventory

2.8.5.6.1 Inadvertent Opening of a Pressure Relief Valve

Regulatory Evaluation

The inadvertent opening of a pressure relief valve results in a reactor coolant inventory decrease and a decrease in RCS pressure. The pressure relief valve discharges into the suppression pool. Normally there is no reactor trip. The pressure regulator senses the RCS pressure decrease and partially closes the turbine control valves to stabilize the reactor at a lower pressure. The reactor power settles out at nearly the initial power level. The coolant inventory is maintained by the feedwater control system using water from the CST via the main condenser hotwell.

The NRC staff's review covered:

1. The sequence of events,
2. The analytical model used for analyses,
3. The values of parameters used in the analytical model, and
4. The results of the transient analyses.

The NRC's acceptance criteria are based on:

1. Draft GDC-6 and current GDC-10, "Reactor design," insofar as they require that the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs;
2. Draft GDC-27 and 28, insofar as they require that at least two reactivity control systems be provided and be capable of making and holding the core subcritical from any hot standby or hot operating condition sufficiently fast to prevent exceeding acceptable fuel damage limits;
3. Draft GDC-29, insofar as it requires that at least one of the reactivity control systems be capable of making the core subcritical under any condition sufficiently fast to prevent exceeding acceptable fuel damage limits; and
4. Draft GDC-30, insofar as it requires that at least one of the reactivity control systems provided shall be capable of making and holding the core subcritical under any conditions with appropriate margins for contingencies.

The licensee also indicated that current GDC-15, "Reactor coolant system design," and GDC-26, "Reactivity control system redundancy and capability," are applicable to MNGP as described in USAR Section 14.4. Specific review criteria are contained in SRP Section 15.6.1

and guidance provided in Matrix 8 of RS-001.

Technical Evaluation

Inadvertent opening of a safety/relief valve will cause a decrease in reactor coolant inventory and result in mild depressurization. The pressure regulator senses the reactor pressure decrease and closes the turbine control valves far enough to maintain constant reactor vessel pressure. Reactor power settles out at nearly the initial power level. This event will have only a slight effect on fuel thermal margins. The change in fuel rod surface heat flux is expected to be negligible, causing an insignificant change in the MCPR. Thus, this transient is bounded by more severe transients and hence is not analyzed.

Conclusion

The NRC staff has reviewed the licensee's disposition regarding the inadvertent opening of a pressure relief valve event and concludes that the licensee's disposition has adequately accounted for operation of MNGP at the proposed power level. The NRC staff further concludes that the licensee has demonstrated that the reactor pressure relief and control systems will continue to ensure that the specified acceptable fuel design limits and the RCPB pressure limits will not be exceeded as a result of this event. Based on these considerations, the NRC staff concludes that MNGP will continue to meet the regulatory intent of proposed draft criteria 6, 27, 28, 29, and 30, and current GDC-10, 15, and 26, following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the inadvertent opening of a pressure relief valve event.

2.8.5.6.2 Emergency Core Cooling System and Loss-of-Coolant Accidents (LOCAs)

Regulatory Evaluation

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

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 models used for the analyses and calculations of the reactor power, pressure, flow, and temperature transients,

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

6. Functional and operational characteristics of the reactor protection and ECCS systems, and

7. Operator actions.

The NRC's acceptance criteria are based on:

1. 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," insofar as it establishes standards for the calculation of ECCS performance and acceptance criteria for that calculated performance;

2. 10 CFR 50, Appendix K, "ECCS Evaluation Models," 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. Draft GDC-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. Draft GDC-37, 41, 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 Sections 6.3 and 15.6.5, and guidance provided in Matrix 8 of RS-001.

<u>Technical Evaluation</u>

The ECCS for MNGP is described in Section 6.2 of the MNGP UFSAR. ECCS components are designed to provide protection in the event of a LOCA due to a rupture of the primary system piping. Although DBAs are not expected to occur during the lifetime of a plant, plants are designed and analyzed to ensure that the radiological dose from a DBA will not exceed the dose limits specified in 10 CFR Part 100.

For a LOCA, 10 CFR 50.46 specifies design acceptance criteria based on: (a) the PCT; (b) local cladding oxidation; (c) total hydrogen generation; (d) coolable core geometry; and (e) long-term coolability. The LOCA analysis considers a spectrum of break sizes and locations against these acceptance criteria, including a rapid circumferential rupture of the largest recirculation system pipe. Assuming a single failure of the ECCS, the LOCA analysis identifies the break sizes that most severely challenge the ECCS systems and the primary containment. The MAPLHGR operating limit is based on the most limiting LOCA analysis, and licensees perform LOCA analyses for each new fuel type to demonstrate that the 10 CFR 50.46 acceptance criteria can be met.

The ECCS for MNGP includes the HPCI system, the LPCI mode of the RHR, the CS system and ADS. The CLTR provides for plant-specific disposition of the NPSH requirements, about which the NRC staff requested additional information via RAI-2.8.5.6-7.

In response to RAI-2.8.5.6-7, the licensee stated that NPSH requirements for the ECCS are evaluated in PUSAR Section 2.6.5. NPSH requirements are, therefore, not evaluated in this section of the NRC staff's SER.

High Pressure Coolant Injection (HPCI)

The HPCI system is designed to pump water into the reactor vessel over a wide range of operating pressures. The primary purpose of the HPCI system is to maintain reactor vessel coolant inventory in the event of a small-break LOCA that does not immediately depressurize the reactor vessel. In this event, the HPCI system maintains reactor water level and helps depressurize the reactor vessel.

The CLTR provides for generic disposition of HPCI performance, provided that licensees confirm the following requirements:

[[

]]

The licensee confirmed that these requirements are met in concert with the EPU request. The NRC staff finds that the licensee's adoption of the CLTR disposition for HPCI at EPU condition is acceptable, because the requirements have been met.

Core Spray (CS)

The CS system is automatically initiated in the event of a LOCA. When operating in conjunction with other ECCS, the CS system is required to provide adequate core cooling for all LOCA events. There is no change in the reactor pressures at which the CS is required.

The CS system sprays water into the reactor vessel after it is depressurized. The primary purpose of the CS system is to provide reactor vessel coolant inventory makeup for a large-break LOCA and for any small-break LOCA after the reactor vessel has depressurized. It also provides long-term core cooling in the event of a LOCA.

The increase in decay heat due to EPU could increase the calculated PCT following a postulated LOCA by a small amount. The ECCS performance evaluation demonstrates that the existing CS system performance capability, in conjunction with the other ECCS as required, is adequate to meet the post-LOCA core cooling requirement for the EPU conditions. The licensee stated that **[[**

]].

The NRC staff, therefore, accepts the licensee's assessment that EPU does not significantly impact operation of the CS system. Since the licensee's ECCS-LOCA analysis (see section below titled, "ECCS Performance") is based on the current CS system capability and demonstrates that the system provides adequate core cooling, the NRC staff finds the evaluation acceptable and agrees with the licensee's assessment that the CS system will continue to meet the NRC's acceptance criteria.

Low Pressure Coolant Injection (LPCI)

The LPCI mode of the RHR system is automatically initiated in the event of a LOCA. The primary purpose of the LPCI mode is to help maintain reactor vessel coolant inventory for a large-break LOCA and for any small-break LOCA after the reactor vessel has depressurized. The LPCI operating requirements are not affected by EPU. The increase in decay heat due to EPU could increase the calculated PCT following a postulated LOCA by a small amount. [[ ]]. The ECCS performance evaluation demonstrates that the existing LPCI mode performance capability, in conjunction with the other ECCS, is adequate to meet the post-LOCA core cooling requirement for EPU reactor thermal power conditions. The licensee stated that [[ ]] consistent with the generic disposition set forth in the CLTR.

Since the licensee's ECCS-LOCA analysis (see section below titled, "ECCS Performance") is based on the current LPCI system capability and demonstrates that the system provides adequate core cooling, the NRC staff finds the evaluation acceptable and agrees with the licensee's assessment that the LPCI system will continue to meet the NRC's acceptance criteria.

Automatic Depressurization System (ADS)

The ADS uses SRVs to reduce the reactor pressure following a small-break LOCA when it is assumed that the high-pressure systems have failed. After a specified delay, the ADS actuates either on low water level plus high drywell pressure or on sustained low water level alone. This allows the CS and LPCI to inject coolant into the reactor vessel.

Plant design requires a minimum flow capacity for the SRVs, and that ADS initiates following confirmatory signals and associated time delay(s). MNGP's ability to initiate ADS on appropriate signals is not affected by EPU. In References 65 and 66, the licensee discussed changes to the sustained low water level initiation logic used to actuate ADS. This changed Function 1.e and 2.e, "Reactor Steam Dome Pressure Permissive – Bypass Timer (Pump Permissive)," in Table 3.3.5.1-1, "Emergency Core Cooling System Instrumentation." With these changes, the licensee stated that the ADS initiation logic and ADS valve control are not affected, and are adequate for EPU conditions.

The NRC staff requested additional information based on what appears to be an increase in assumed ADS valve relief capability. This is discussed in the evaluation of the LOCA analysis and, specifically, through RAI-2.8.5.6-4. The EPU analysis assumes three ADS valves are operable, and this is reflected by changes to TS 3.5.1 in the November 8, 2008, application.

Since the licensee's ECCS-LOCA analysis (see section below titled, "ECCS Performance"), based on the proposed ADS capability, demonstrates that the system provides adequate core cooling, the NRC staff finds the evaluation acceptable, and agrees with the licensee's assessment that the ADS will continue to meet the NRC's acceptance criteria.

The EPU does not affect the protection provided for any of the ECCS features (HPCI, CS, LPCI, and ADS) against the dynamic effects and missiles that might result from plant equipment failures.

ECCS Performance

The ECCS is designed to provide protection against postulated LOCAs caused by ruptures in the primary system piping. The ECCS performance under all LOCA conditions and the analysis models must satisfy the requirements of 10 CFR 50.46 and 10 CFR Part 50, Appendix K.

The following staff-approved codes were used for the LOCA analysis (GE topical report NEDE-23785-1-PA, 1984):

SAFER -- The SAFER code is used to calculate the long-term-thermal-hydraulic behavior of the coolant in the vessel during a LOCA. Some important parameters calculated by SAFER are vessel pressure, vessel water level, and ECCS flow rates. The SAFER code also calculates PCT and local maximum oxidation.

LAMB -- The LAMB code is used to analyze the short-term thermal-hydraulic behavior of the coolant in the vessel during a postulated LOCA. In particular, LAMB predicts the core flow, core inlet enthalpy, and core pressure during the initial phase of the LOCA event (i.e. the first 5 seconds)

GESTR-LOCA -- The GESTR-LOCA code is used to provide best-estimate predictions of the thermal performance of GE nuclear fuel rods experiencing variable power histories. For LOCA analysis, the GESTR code is used to initialize the fuel stored energy and fuel rod fission gas inventory at the onset of a postulated LOCA.

TASC -- The TASC code has been accepted for transient analysis and LOCA analysis. TASC is a functional replacement of the SCAT code. TASC is an improved version of the NRC-approved SCAT code, with the added capability to model advanced fuel features (partial length rods and new critical power correlation). TASC is a detailed model of an isolated fuel channel. It is used to predict the time to boiling transition for a large-break LOCA. This value is used in subsequent codes to turn off nucleate boiling heat transfer models and turn on transition boiling models. Because there is significant experience with GE's application of the SAFER/GESTR-LOCA methodology, and appreciable experience with the application of this methodology to EPU plants, the NRC staff's review focused on the results of the analysis, and how they may have changed for the EPU.

The following paragraphs are excerpted from a tutorial on the SAFER/GESTR analysis process that was presented to the NRC staff in October 2001 (Reference 23). They are included to provide a clearer understanding of the differences in results between the CLTP-analyzed core and the EPU core.

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MNGP's LOCA Analysis Results

The LOCA analysis for EPU builds on the existing SAFER/GESTR LOCA analyses for a plant. The NRC staff evaluations of past EPU at BWRs have shown that the basic break spectrum is not affected by EPU and EPU is expected to have a small effect on the licensing basis PCT. Because the EPU implementation has only a small effect on PCT, the limiting single failure will not change for EPU conditions in a plant.

The licensing basis PCT is based on the Appendix K PCT. The effect of EPU on the licensing basis PCT will be based on the delta PCT change from the large-break and small-break evaluation such that the licensing basis PCT is maximized. Use of the most limiting of the nominal or Appendix K PCT changes for the licensing basis PCT will ensure continued compliance with the requirements for the SAFER/GESTR LOCA application methodology as approved by the NRC.

The licensing basis PCT was determined based on the calculated Appendix K PCT at MELLLA core flow, [[ ]] with an adder to account for uncertainties. For the EPU, the GE14 licensing basis PCT is ≤ 2140°F at MELLLA core flow, with transient cladding oxidation not exceeding 9.0 percent of the original cladding thickness, and hydrogen generation not exceeding 0.2 percent of the core-wide metal-water reaction.

Long-term cooling is assured when the core remains flooded to the jet pump top elevation and when a core spray system is operating.

The PUSAR states that the increased decay heat associated with EPU results in a longer ADS blowdown and a higher PCT for the small-break LOCA. The NRC staff requested clarification of this statement in light of the requested change in assumed ADS availability (RAI-2.8.5.6-4). The licensee responded to RAI-2.8.5.6-4 (Reference 4) by stating that increasing the assumed ADS capability mitigates small-break LOCA sensitivity to power shape. Therefore, MNGP's small-break LOCA PCT decreases with EPU implementation, not because of the EPU itself, but because of the change in ADS configuration.

The PUSAR states "plant specific analyses demonstrate that there is sufficient ADS capacity at EPU conditions with all ADS valves available. With two ADS valves available, a LHGR multiplier is applied to ensure that the small break is not limiting." The NRC issued RAI-2.8.5.6-5 for clarification of this statement. The licensee provided clarification (Reference 4), stating that an additional analysis was provided to support plant operation with two ADS valves available, but that this analytic option is not currently being used to support any licensing requests. In consideration of the licensee's response to RAI-2.8.5.6-5, the NRC staff disregarded the PUSAR statement concerning plant operation with two ADS valves available.

The NRC staff inquired about oxidation sources included in the licensee's LOCA analyses (RAI-2.8.5.6-8). The NRC staff inquired about both pre-existing oxidation, and about oxidation on both surfaces of the fuel cladding.

Regarding pre-existing oxidation, the licensee stated that the LOCA analyses consider only transient oxidation (Reference 5). However, GE14 fuel studies have concluded that: (1) at the time of maximum stored energy, the pre-existing oxidation at MNGP would be on the order of 1.19 percent; and (2) highly exposed bundles indicate oxidation levels as high as 3.53 percent. Since the predicted oxidation level at MNGP is <9.0 percent, the results maintain significant margin to the 17 percent regulatory limit, even in consideration of the pre-existing oxidation. The NRC staff finds this clarification acceptable because there is still significant margin to the regulatory limits.

Regarding cladding inside oxidation, the licensee confirmed that the inside surface cladding is calculated as a part of the total transient cladding oxidation. Because the calculation considers both cladding surfaces, the NRC staff finds the licensee's response acceptable.

Based on the licensee's LOCA analysis, and because the licensee will perform plant- and cycle-specific evaluations of ECCS-LOCA performance for each fuel reload at the EPU conditions using approved methods, the NRC staff agrees with the licensee that the MNGP ECCS-LOCA performance complies with 10 CFR 50.46 and Appendix K requirements. The EPU analyses are acceptable for the following seven reasons:

(1) The NRC staff evaluations of several requests for stretch power increase and extended power uprate at BWRs have shown that the change of PCT for power uprates is not significant. The maximum increase in the PCT was small (in consideration of MNGP's LHGR limit relaxation, it [[ ]]), and was well within the acceptance criteria of 10 CFR 50.46. Since there is only a small change in PCT, an EPU has a negligible effect on the adders used to determine the licensing basis PCT.

    The NRC staff requested additional information concerning what appears to be a significant increase in PCT to clarify that the increase is not directly a result of the requested EPU (RAI-2.8.5.6-1).

    In response, the licensee provided a table indicating the key differences in the LOCA model when comparing: (1) the current licensing basis; (2) the CLTP case evaluated in the PUSAR; and (3) the EPU case evaluated in the PUSAR (Reference 4). For convenience, a simplified version of this table is included below.

    The licensee also clarified that the change in PCT is largely due to the elimination of a 1600°F limitation on the Upper Bound PCT. The 1600°F limitation was imposed by the

NRC staff in its approving safety evaluation of the SAFER/GESTR-LOCA methodology. It was based on the availability of sufficient experimental data to justify the uncertainties used in determining the Upper Bound PCT. This original limitation was permitted for removal based on improvements to the SAFER/GESTR-LOCA methodology, as discussed in an NRC staff safety evaluation (Reference 24).

MNGP has recently eliminated this limitation. The elimination of the limitation allows an increase in linear heat generation rates, which in turn causes the initial conditions for the LOCA analysis to change. While in the upper-bound case, the difference in PCT is on the order of 100°F, the difference is slightly greater (on the order of 160°F) in the Appendix K and licensing basis cases. This is because the Appendix K heatup modeling is more sensitive to the higher linear heat rate. The licensee stated that, because the Appendix K model indicates a faster increase in large-break PCT than does the upper-bound PCT, the Appendix K PCT prediction is demonstrated to remain adequately conservative.

When considering that the upper-bound PCT is used to indicate the 95 percent upper bound on achievable PCTs (a measure of statistical confidence), the licensee's assertion that the Appendix K PCT's greater increase is a demonstration of conservatism is reasonable.

The key results provided by the licensee in response to RAI-2.8.5.6-1 were as follows:

Parameter	Current Licensing Basis	CLTP/PUSAR	EPU/PUSAR
Nominal Thermal Power	1775 MWt	1775 MWt	2004 MWt
LHGR Setdown Limit[5]	15%	10%	10%
Limiting Appendix K Large Break PCT	1966°F	2123°F	2119°F
Upper Bound PCT	<1600°F	<1670°F	<1670°F
Licensing Basis PCT	<1970°F	<2140°F	<2140°F

These results demonstrate that, while the relaxation of the LHGR setdown limit contributes to an increase in PCT, the EPU itself does not. This is because the licensee analyzed a case assuming the relaxed LHGR setdown limits with a core operating at the current licensed thermal power level, as well as a similar case operating at the uprated power level. The results show that there is marginal difference in PCTs between the CLTP and EPU-analyzed cases, even though the current licensing basis PCTs are less than those presented in the PUSAR. Based on the licensee's responses and the NRC staff's evaluation described above, the NRC staff confirmed the licensee's adoption of the PUSAR disposition regarding increases in LBLOCA PCT associated with EPU.

[5] The LHGR Setdown Limit is a restriction placed on the peak linear heat generation rate to assure that the analyzed PCT following a postulated LOCA complies with applicable requirements. For instance, a limit may be applied to force the upper bound PCT to comply with the 1600°F licensing restriction, or to cause the licensing basis PCT to remain within 10 CFR 50.46 limits.

(2) The ECCS performance characteristics and basic break spectrum response are largely unaffected by an EPU.

The NRC staff requested additional information because the small-break LOCA is currently the limiting LOCA (RAI-2.8.5.6-2). The licensee's response to RAI-2.8.5.6-2 (Reference 4) indicates that the small-break LOCA analysis currently yields the limiting PCT due to the assumption of the availability of only two automatic depressurization system (ADS) valves. Because of this assumption, the limiting failure of the high-pressure coolant injection system causes a more drastic effect on the PCT resulting from the inability of the plant to depressurize as quickly to the shutoff head of the low-pressure coolant injection system.

While the current licensing-basis analysis assumes that one ADS valve is out of service, the PUSAR analyses, both the CLTP and the EPU cases, assume that no ADS valves are out of service. This change has little impact on the large-break analyses; however, the unavailability of high-pressure coolant injection in combination with degraded depressurization capability will have a significant impact on the small-break analyses. According to the licensee's response to RAI-2.8.5.6-1, this change affects the small-break PCT by more than 500°F.

In response to RAI-2.8.5.6-2 (Reference 4), the licensee added that this change returns the MNGP LOCA analyses to the trends associated with the historical patterns for the break spectrum, and based on the results presented in response to RAI-2.8.5.6-1, this statement is confirmed.

(3) The limiting break sizes are well known and have been shown not to be a function of reactor power level.

The licensee stated that "The Appendix K results confirm that the limiting break is the recirculation suction line DBA and that the LPCI injection valve failure is the limiting single failure." The NRC staff requested additional information to determine how the analyses confirmed the limiting break and failure assumptions (RAI-2.8.5.6-3).

In its response to RAI-2.8.5.6-3 (Reference 4), the licensee stated that an inherently conservative modeling assumption is the open modeling of the recirculation loop, which allows hydraulic communication from LPCI flow and the vessel (via bottom head drain line and vessel downcomer) to and out of the broken recirculation loop. This modeling assumption causes the vessel inventory loss to dominate the accident sequence, and it causes the suction location to be limiting because of its larger area. A break spectrum analysis confirms this for MNGP.

Different single failures were also modeled to confirm that the LPCI injection valve failure remains the limiting failure at EPU conditions.

These modeling approaches confirm the generic disposition regarding break location and limiting failure insofar as the results confirm that the break location and limiting failure remain the double-ended guillotine break on the recirculation suction line with a LPCI injection valve failure. Based on this, the NRC staff confirmed that the generic disposition is applicable to MNGP.

(4) The analyses assume the hot bundle continues to operate at the thermal limits (MCPR, MAPLGHR, and LHGR) which are not changed by the EPU.

(5) The PCT for the limiting large-break LOCA is determined primarily by the hot bundle power, which is not expected to increase with power uprate.

(6) The reload evaluation confirms that the MAPLHGR for each fuel type in the specific reload core is bounded by the MAPLHGR used in the ECCS-LOCA performance analysis.

(7) Because the plant is MAPLHGR-limited, a detailed plant-specific analysis for the licensing basis PCT was performed.

Conclusion

The NRC staff has reviewed the licensee's analyses of the LOCA events and the ECCS. The NRC staff concludes that the licensee's analyses have adequately accounted for operation of MNGP at the proposed power level and that the analyses were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection system and the ECCS will continue to ensure that the peak cladding temperature, total oxidation of the cladding, total hydrogen generation, and changes in core geometry, and long-term cooling, will remain within acceptable limits. Based on these considerations, the NRC staff finds the proposed EPU acceptable with respect to the ECCS-LOCA.

On September 12, 2011, the NRC staff requested (Reference 94) that the licensee reanalyze the ECCS performance per 10 CFR 50.46. A reanalysis was performed to account for 10 CFR 50.46, Notifications 2011-02 and 2011-03. The results were provided in Reference 67, and reflected on corrected page 2-296 to the updated PUSAR (i.e., NEDC-33322P, Revision 3).

In its July 8, 2013, letter, the licensee notified the NRC that the predicted peak cladding temperature for the limiting LBLOCA would increase by [[ ]] when GESTR-M-based fuel conditions are replaced with PRIME-based fuel conditions in the ECCS evaluation (Reference 87). This conclusion was based on a "single effect sensitivity" study, in which the same limiting LBLOCA case was calculated using the PRIME model input as an explicit replacement. Based on the fact that this study relied on explicit analyses comparing the effects of GESTR-M and PRIME use, the NRC staff determined that the ECCS evaluation, as updated by the sensitivity study, acceptably accounts for nuclear fuel thermal conductivity degradation.

2.8.5.7 Anticipated Transients without Scram (ATWS)

Regulatory Evaluation

An ATWS is defined as an AOO followed by the failure of the reactor portion of the protection system specified in GDC-20, Protection system functions." The regulations at 10 CFR 50.62, "Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants," require that:

Each BWR have an alternate rod injection (ARI) system that is designed to perform its function in a reliable manner and be independent (from the existing reactor trip system) from sensor output to the final actuation device;

Each BWR have a SLC system with the capability of injecting into the reactor vessel a borated water solution with reactivity control at least equivalent to the control obtained by injecting 86 gpm of a 13 wt% sodium pentaborate decahydrate solution at the natural boron-10 isotope abundance into a 251-inch inside diameter reactor vessel. The system initiation must be automatic;

Each BWR have equipment to trip the reactor coolant recirculation pumps automatically under conditions indicative of an ATWS.

The NRC staff's review was conducted to ensure that:

1. The above requirements are met;

2. Sufficient margin is available in the setpoint for the SLC system pump discharge relief valve such that SLC system operability is not affected by the proposed EPU; and

3. Operator actions specified in the plant's Emergency Operating Procedures are consistent with the generic emergency procedure guidelines/severe accident guidelines (EPGs/SAGs), insofar as they apply to the plant design.

In addition, the NRC staff reviewed the licensee's ATWS analysis to ensure that:

1. The peak vessel bottom pressure is less than the ASME Service Level C limit of 1500 psig;

2. The peak clad temperature is within the 10 CFR 50.46 limit of 2200°F;

3. The peak suppression pool temperature is less than the design limit; and

4. The peak containment pressure is less than the containment design pressure.

The NRC staff also evaluated the potential for thermal-hydraulic instability in conjunction with ATWS events in Section 2.8.3. Specific review criteria are provided in SRP Section 15.8 and additional guidance is provided in Matrix 8 of RS-001.

Technical Evaluation

The licensee's analysis of the ATWS is described in the MNGP USAR Section 14.8, "Anticipated Transients without Scram (ATWS)."

The licensee stated that MNGP meets the ATWS requirements defined in 10 CFR 50.62 because: (a) an ARI system is installed; (b) the boron injection capability is equivalent to 86 gpm; and (c) there is an automatic RPT logic (i.e. ATWS-RPT). In addition, an ATWS analysis was performed at EPU conditions to confirm that: (a) the peak vessel bottom pressure is less than ASME Service Level C limit of 1500 psig; (b) the peak suppression pool temperature is less than 281°F (wetwell shell design temperature); and (c) the peak containment pressure is less than 56 psig (drywell design pressure). Section 3.7 of ELTR2 discusses the ATWS analyses and provides an evaluation of the following limiting ATWS events in terms of overpressure and suppression pool cooling: (a) MSIV closure; (b) pressure regulator failure - open (PRFO), loss of offsite power (LOOP), and (4) inadvertent opening of a relief valve.

The licensee performed an ATWS analyses for an equilibrium core at the EPU operating condition to demonstrate that MNGP meets the ATWS acceptance criteria. Based on experience, only the limiting cases (MSIV closure and PRFO) were analyzed. [[

]]. Therefore, the NRC staff agrees with the licensee that there is significant margin to the PCT and oxidation criteria of 10 CFR 50.46.

Table 2.8-6 of the PUSAR lists the key input parameters used in the ATWS analyses, and Table 2.8-7, lists the corresponding results (peak vessel bottom pressure, peak suppression pool temperature, and peak containment pressure).

Conclusion

The NRC staff reviewed the information submitted by the licensee related to ATWS and concludes that the licensee has adequately accounted for the effects of the proposed EPU on ATWS. The NRC staff concludes that the licensee has demonstrated that ARI, SLC, and RPT systems are installed and will continue to meet the requirements of 10 CFR 50.62 and the analysis acceptance criteria following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to ATWS.

2.8.6 Fuel Storage

2.8.6.1 New Fuel Storage

Although RS-001 provides for the separate evaluation of both new and spent fuel storage, the regulatory requirements for each is effectively the same, with the exception that GDC-4, "Environmental and dynamic effects design basis" is applicable to spent fuel storage, but not necessarily to new fuel storage.

This consideration, in combination with the fact that MNGP's licensing basis for fuel storage combines both new and spent fuel storage, provides sufficient justification to combine the NRC staff's evaluation of new and spent fuel storage. NSPM's disposition is the same for both new and spent fuel storage.

RS-001 also notes that a review of fuel storage is not applicable unless an EPU application requests approval for a new fuel design, which is not the case for MNGP. Therefore, the NRC staff reviewed information provided by the licensee and has added additional, summary information pertinent to the current GE14 fuel design in use at MNGP, and its subcriticality analysis.

Regulatory Evaluation

Nuclear reactor plants include facilities for storage of both new and spent fuel. The quantity of new fuel to be stored varies from plant to plant, depending upon the specific design of the plant and the individual refueling needs. For MNGP in particular, the fuel storage design basis provides for the storage of 150 new fuel assemblies. The safety function of the fuel pool and storage racks is to maintain the new or spent fuel assemblies in a safe and subcritical array during all credible storage conditions and to provide a safe means of loading the assemblies into shipping casks.

The NRC's acceptance criteria are based on:

1. Draft GDC-40 and 42, insofar as they require that protection be provided for ESFs against the dynamic effects and missiles that might result from plant equipment failures, as well as the effects of a loss of coolant accident; and

2. Draft GDC-66, insofar as it requires that criticality in the fuel storage systems be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

Technical Evaluation

The licensee stated that the CLTR limits the EPU because there is no new product line introduction and no change to fuel cycle length. The licensee also stated that the spent fuel storage facility is evaluated whenever a change to the fuel design is introduced, but that this is not the case for EPU.

Specific information pertinent to the NRC staff's review regarding the fuel bundle design is available in MNGP's current licensing basis. The MNGP USAR describes the SFP, which is comprised predominately of a General Electric-designed High Density Fuel Storage System (HDFSS), with a single, original, low-density rack and two control blade racks. The design basis criterion associated with the storage of both irradiated and new fuel is that the effective multiplication factor of fuel stored under normal conditions (flooded pool, dry vault) will be 0.90 or less for the regular density racks and 0.95 or less for the high density fuel storage racks. Per the GE topical report (Reference 16), these criteria are met when the uncontrolled infinite lattice multiplication factors for all current and future reload fuel designs are 1.33 or less for 20°C to 100°C, for the high density racks, or 1.31 or less for 20°C to 100°C, for the new fuel vault storage rack. The low-density rack criteria is met when the uncontrolled infinite lattice multiplication factors for all current and future reload designs are 1.31 or less for 20°C to 100°C for regular spent fuel storage racks with an interrack spacing $\geq$ 11.875 inches, or 1.30 or less for 20°C to 100°C for regular spent fuel storage racks with an interrack spacing $\geq$ 11.71 inches.

The licensee stated that, because new fuel designs are not being introduced, no further analysis was necessary. The NRC staff verified this claim by reviewing the compliance document for GE14 fuel (Reference 31).

Indeed, GE14 fuel has been confirmed to meet the subcriticality criterion for GE-designed fuel storage racks, both high- and low-density. [[

]]. Compliance with this limit is confirmed for each GE14 lattice as part of the design process.

The NRC staff confirmed, as discussed above, that the GE14 fuel design meets the MNGP licensing bases for fuel storage, and this consideration is unaffected by the requested EPU.

Conclusion

The NRC staff reviewed the licensee's generic and plant-specific assessment related to the effects of the proposed EPU on the spent fuel storage capability and concludes that the licensee has adequately accounted for the effects of the proposed EPU on the spent fuel criticality analyses. Specifically, as provided in RS-001, no review is required because the EPU request is not combined with a fuel design change request. Based on this consideration, as augmented by the NRC staff's own technical review, the NRC staff concludes that the spent fuel storage facilities will continue to meet the requirements of draft GDC-40, 42, and 66 following implementation of the proposed EPU, and is acceptable.

2.8.7 Additional Review Areas

2.8.7.1 Methods Evaluation

Regulatory Evaluation

The licensee's analyses supporting safe operation at EPU conditions are performed using NRC-approved licensing methodology, analytical methods, and codes. In general, the accuracy of the analytical methods and codes are assessed and benchmarked against measurement data, comparisons to actual nuclear plant test data, and research reactor measurement data. The uncertainties and biases associated with specific correlations simulating physical phenomena, with key parameters or with integral code calculations modeling a design-bases event, are determined. The identified uncertainties associated with the analytical methods, the measured quantities used to simulate the core conditions, and the manufacturing tolerances (e.g., fuel manufacturing tolerances) are accounted for in the analyses. The NRC-approved licensing methodology, topical reports, and codes specify the applicability ranges.

The LTR covering specific analytical methods or code system quantify the accuracy of the methods or the code used. The SER approving the topical report includes limitations that delineate the conditions that warrant specific actions, such as obtaining measurement data or when new NRC approval is required. In general, the use of NRC-approved analytical methods is contingent upon application of these methods and codes within the ranges for which the data was provided and against which the methods were evaluated. Thus, in general, the plant-specific application does not entail review of the NRC-approved analytical methods and codes.

The NRC staff reviewed the referenced Interim Methods Licensing Topical Report (IMLTR) NEDC-33173P to verify the following:

- The analytical methods and codes used to perform the design-bases safety analyses will be applied within the applicable NRC-approved validation ranges. The calculation and measurement uncertainties applied to the thermal limit calculations and the models simulating physical phenomena will remain valid for the predicted neutronic and thermal hydraulic core and fuel conditions during steady-state, transient, and accident conditions. The qualification database supporting analytical models simulating physical phenomena remains valid and applicable to the conditions under which it is applied, including those models and key parameters in which specific uncertainties are not applied.

- If the NRC-approved analytical methods and codes are extended outside the applicability ranges, the extension of the specific models are demonstrated to be acceptable or additional margins are applied to the affected downstream safety analyses until such time the supporting qualification data is extended.

The NRC staff's SER for NEDC-33173P, "Applicability of GE Methods to Expanded Operating Domains," dated January 17, 2008, specifies the limitations that apply to NEDC-33173P (letter from H. K. Nieh of NRC to R. E. Brown of GEH, dated January 17, 2008 (Reference 95). The NRC staff's revised final SE for NEDC-33173P was issued in a letter dated July 21, 2009 (Reference 96).

Technical Evaluation

The licensee referenced NEDC-33173P to justify application of GE methods to the MNGP EPU application. Each condition specified in the NRC staff's SER for NEDC-33173P was evaluated for acceptability for MNGP EPU by NSPM in Appendix A of the PUSAR (Enclosure 5 of the licensee's November 5, 2008, application). The NRC staff's review of these conditions is discussed below.

Condition 1: TGBLA/PANAC Version

IMLTR SER Condition

The neutronic methods used to simulate the reactor core response and that feed into the downstream safety analyses supporting operation at EPU/MELLLA+ will apply to TGBLA06/PANAC11, or a later NRC-approved version of the neutronic method.

PUSAR Disposition

Appendix A of the MNGP PUSAR states that TGBLA06/PANAC11 methods are used in the safety analysis. The NRC staff requested additional information in RAI-SNPB-1 regarding the version of TGBLA06 used in the analysis. Specifically the NRC staff requested that NSPM clarify if the modified TGBLA06 code was used in the analysis. The licensee's March 19, 2009, response states that the modified TGBLA06 code will be used to analyze the fresh fuel bundles in the first EPU core design. However, the partially burnt bundles in the core will be analyzed based on the historically developed nuclear libraries that were generated using the unmodified TGBLA06 code. The response states that the sensitivity of the analysis results to the modification made in TGBLA06 is insignificant. The response references the results of sensitivity studies performed for GE14 lattices as part of the NRC staff's ESBWR review (General Electric, "Supplemental Response to Portion of NRC Request for Additional

Information Letter No. 53 Related to ESBWR Design Certification Application – DCD Chapter 4 and GNF Topical Reports – RAI-Number 4.3-3," MFN 06-297 Supplement 1, November 8, 2006). The NRC staff has reviewed these results as part of its ongoing review of the ESBWR design certification application. In the conduct of this review the NRC staff found that the results of the analyses confirm that the extrapolated high void fraction nuclear parameters are essentially identical when either code is used. As the results are demonstrated to be essentially the same the NRC staff finds that it is acceptable to use the previously calculated nuclear data libraries for the partially burnt fuel bundles. The response states that all future fuel loads will be analyzed using the most recent version of the TGBLA06 code. Based on the above, the NRC staff finds that the disposition of the condition is adequate.

Condition 2: 3D MONICORE

IMLTR SER Condition

For EPU/MELLLA+ applications, relying on TGBLA04/PANAC10 methods, the bundle RMS difference uncertainty will be established from plant-specific core-tracking data, based on TGBLA04/PANAC10. The use of plant-specific trendline based on the neutronic method employed will capture the actual bundle power uncertainty of the core monitoring system.

PUSAR Disposition

Appendix A of the MNGP PUSAR states that there is no reliance of TGBLA04/PANAC10 methods (see Enclosure 5 of the licensee's November 5, 2008, application). As such, this condition is not applicable to MNGP. Therefore, the NRC staff finds that the disposition of the condition is acceptable.

Condition 3: Power to Flow Ratio

IMLTR SER Condition

Plant-specific EPU and expanded operating domain applications will confirm that the core thermal power to core flow ratio will not exceed 50 MWth/Mlbm/hr at any statepoint in the allowed operating domain. For plants that exceed the power-to-flow value of 50 MWth/Mlbm/hr, the application will provide power distribution assessment to establish that neutronic methods axial and nodal power distribution uncertainties have not increased.

PUSAR Disposition

Appendix A of the PUSAR states that the power to flow ratio is less than 50 MWth/Mlbm/hr. The NRC staff confirmed that the power to flow ratio at the highest thermal power at the minimum flow point (100 percent EPU power / 99 percent rated core flow) is less than 50 MWth/Mlbm/hr based on the plant information provided in Table 1-2 of the PUSAR. The Appendix A disposition references the 100 percent EPU power / 100 percent RCF statepoint. The NRC staff requested clarification regarding this disposition in RAI-SNPB-2. The licensee's March 19, 2009, response to RAI-SNPB-2 states that the disposition of the condition is consistent with the guidance provided in MFN 08-693 regarding the power to flow ratio (Reference 27). The power/flow operating map does not change from cycle to cycle and, therefore, the NRC staff finds that the power-to-flow ratio is within the limit imposed by Condition 3. As the power to flow ratio remains below 50 MWth/Mlbm/hr at the minimum flow point at the highest power level, the NRC staff finds that this condition is met and, therefore, is acceptable.

Condition 4: SLMCPR 1

IMLTR SER Condition

For EPU operation, a 0.02 value shall be added to the cycle-specific SLMCPR value. This adder is applicable to SLO, which is derived from the dual loop SLMCPR value.

PUSAR Disposition

Appendix A of the PUSAR states that 0.02 is added to the SLO SLMCPR (Enclosure 5 of the licensee's November 5, 2008, application). The NRC staff requested additional information in RAI-SNPB-3. Specifically the NRC staff requested that the licensee clarify that the 0.02 adder is applied to both the dual-loop operation, as well as the SLO, SLMCPR to account for nuclear methods uncertainties. The licensee responded in its March 19, 2009, letter to RAI-SNPB-3, stating that the adder has been applied to both SLMCPR values. The NRC staff finds that the disposition is consistent with the condition and, therefore, is acceptable.

The NRC staff requested additional information regarding the core flow and feedwater flow uncertainties that are used to determine the SLMCPR in RAI-SNPB-9. The licensee's March 19, 2009, response states that the **[[ ]]**. The NRC staff agrees that the **[[ ]]** and agrees with the basis in the response to perform the SLMCPR analysis **[[ ]]**. The response states that the rated core flow uncertainty used in the SLMCPR analysis is consistent with the NRC-approved values (GE topical report "Methodology and Uncertainties for Safety Limit MCPR Evaluations," NEDC-32601P-A, Class III, August 1999). Therefore, the NRC staff finds that the treatment of the flow uncertainties for SLMCPR analysis is appropriate and acceptable.

For the SLO statepoint the licensee's response confirms that the increased SLO core flow uncertainty is applied in the analysis (see Reference 27). The NRC staff finds that this approach is consistent with the approved SLMCPR methodology. The licensee's response states that the conditions of **[[ ]]** and that the same uncertainties are applied. The NRC staff agrees with this basis.

Therefore, the NRC staff finds that the uncertainties applied in the analysis are consistent with the previously approved values, and that these values are applicable to the conditions analyzed at MGNP.

Condition 5: SLMCPR 2

IMLTR SER Condition

For operation at MELLLA+, including operation at the EPU power levels at the achievable core flow statepoint, a 0.03 value shall be added to the cycle-specific SLMCPR value.

PUSAR Disposition

Appendix A of the PUSAR states that the current amendment application is for EPU operation and approval for operation in the MELLLA+ domain is not currently sought. Therefore, Condition 5 is not applicable to the MNGP EPU application.

Condition 6: R-Factor

IMLTR SER Condition

The plant specific R-factor calculation at a bundle level will be consistent with lattice axial void conditions expected for the hot channel operating state. The plant-specific EPU/MELLLA+ application will confirm that the R-factor calculation is consistent with the hot channel axial void conditions.

PUSAR Disposition

Appendix A of the PUSAR refers to Section 2.8.2.5 regarding the consistency between the R-factor analysis and core conditions. The NRC staff requested additional information regarding the consistency between the axial void profile used to generate the R-factor and the predicted axial void profiles for the limiting bundles. RAI-SNPB-4 specifically requests that the licensee justify the statement that the axial profiles are consistent by comparison of the limiting bundles with the generic axial profile used in the analysis. The licensee's response provides a comparison of the in-core predicted void profiles and the generic void profile. The licensee's March 19, 2009, response provides a figure depicting the predicted limiting bundle void content and the MCPR. Based on the figure, the NRC staff agrees that the void conditions are reasonably consistent with expected void conditions for the limiting channels and is, therefore, acceptable for use in calculating the R-factor. Therefore, the NRC staff finds that the disposition of this condition in the PUSAR is adequate and acceptable.

Condition 7: ECCS-LOCA 1

IMLTR SER Condition

For applications requesting implementation of EPU or expanded operating domains, including MELLLA+, the small- and large-break ECCS-LOCA analyses will include top-peaked and mid-peaked power shape in establishing the MAPLHGR and determining the PCT. This limitation is applicable to both the licensing bases PCT and the upper bound PCT. The plant-specific applications will report the limiting small- and large- break licensing basis and upper bound PCTs.

PUSAR Disposition

Condition 7 of the NRC SER for the IMLTR requires that the ECCS-LOCA performance analyses consider both top-peaked and mid-peaked power distributions. Appendix A of the PUSAR states that these two power shapes were considered in the analysis. Consistent with Condition 7, Table 2.8-5 of the PUSAR provides the upper bound, limiting Appendix K, limiting nominal, and licensing basis PCTs. Section 2.8.5.6.2 provides a discussion regarding the maximum MAPLHGR analyses that are consistent with Condition 7.

In RAI-SNPB-5, the NRC staff requested NSPM to clarify the analysis results provided in Table 2.8-5 of the PUSAR. The PUSAR reports only one PCT for each of the LOCA analyses. The NRC staff requested that NSPM provide the limiting power shape for each PCT as well as the analogous PCT for the other power shape. The licensee's response provides Tables SNPB 5-1 and SNPB 5-2. These tables convey the results of the analyses for both power shapes. The NRC staff finds that the additional information is adequate to provide reasonable assurance that the condition has been met. Therefore, the NRC staff finds the disposition of the Condition 7 to be acceptable.

Condition 8: ECCS-LOCA 2

IMLTR SER Condition

The ECCS-LOCA will be performed for all statepoints in the upper boundary of the expanded operating domain, including the minimum core flow statepoints, the transition statepoint as defined in GE topical report NEDE24011P-A (Reference 16), and the 55 percent core flow statepoint. The plant-specific application will report the limiting ECCS-LOCA results as well as the rated power and flow results. The Supplemental Reload Licensing Report (SRLR) will include both the limiting statepoint ECCS-LOCA results and the rated conditions ECCS-LOCA results.

PUSAR Disposition

Condition 8 of the NRC SER for the IMLTR is applicable to MELLLA+ operation. As the current application for amendment does not request approval to operate in the MELLLA+ domain this condition is not applicable to MNGP.

Condition 9: Transient LHGR 1

IMLTR SER Condition

Plant-specific EPU and MELLLA+ applications will demonstrate and document that during normal operation and core-wide AOOs, the T-M acceptance criteria as specified in Amendment 22 to GESTAR II will be met. Specifically, during an AOO, the licensing application will demonstrate that the: (1) loss of fuel rod mechanical integrity will not occur due to fuel melting; and (2) loss of fuel rod mechanical integrity will not occur due to pellet–cladding mechanical interaction. The plant-specific application will demonstrate that the T-M acceptance criteria are met for the both the UO_2 [uranium dioxide] and the limiting Gd_2O_3 [gadolinium oxide].

PUSAR Disposition

Appendix A of the PUSAR refers to Section 2.8.5 regarding thermal-mechanical analyses. The disposition in Appendix A provides the results of the analyses demonstrating compliance with the fuel centerline melt and cladding plastic strain criteria. Therefore, the NRC staff finds that the condition is acceptably met.

Condition 10: Transient LHGR 2

IMLTR SER Condition

Each EPU and MELLLA+ fuel reload amendment application will document the calculation results of the analyses demonstrating compliance to transient T-M acceptance criteria. The plant T-M response will be provided with the SRLR or COLR, or it will be reported directly to the NRC as an attachment to the SRLR or COLR.

PUSAR Disposition

Appendix A of the PUSAR states that the results of the analyses will be documented in the cycle-specific SRLR. The NRC staff finds that this is acceptable to meet the condition.

Condition 11: Transient LHGR 3

IMLTR SER Condition

To account for the impact of the void history bias, plant-specific EPU and MELLLA+ applications using either TRACG or ODYN will demonstrate an equivalent to 10 percent margin to the fuel centerline melt and the 1 percent cladding circumferential plastic strain acceptance criteria due to pellet-cladding mechanical interaction for all of limiting AOO transient events, including equipment out-of-service. Limiting transients in this case refers to transients where the void reactivity coefficient plays a significant role (such as pressurization events). If the void history bias is incorporated into the transient model within the code, then the additional 10 percent margin to the fuel centerline melt and the 1 percent cladding circumferential plastic strain are no longer required.

PUSAR Disposition

The PUSAR provides the minimum calculated margin to the fuel centerline melt and cladding plastic strain criteria of 26 percent and 35 percent, respectively. These analyses demonstrate greater margin than the 10 percent required by Condition 11. Therefore, the NRC staff finds that the condition has been acceptably met.

Updated Information – TRACCG04 Implementation

In its July 8, 2013, letter, the licensee stated that the void history bias was incorporated into the transient model within the TRACG04 code, obviating the need to maintain 10-percent margin to the fuel centerline melt and the 1-percent cladding circumferential plastic strain acceptance criteria. The licensee also stated that the MNGP Cycle 27 analyses meet the conditions of the Void Reactivity Coefficient Correction Model Condition (Limitation 21 of NEDC-32906P, Supplement 3-A, Revision 1) and the Void Reactivity Coefficient Correction Model Basis Condition (Limitation 22 of NEDC-32906P, Supplement 3-A, Revision 1); therefore, the pressurization transient events are not required to demonstrate the aforementioned margins to the safety analysis limits.

In its review of the SE for NEDC-32906P, Supplement 3-A, Revision 1, "Migration to TRACG04/PANAC11 from TRACG02/PANAC10 for TRACG A and ATWS Overpressure Transients," the NRC staff verified that it is appropriate to remove the 10-percent margin requirements, as stated above. Specifically, Condition 23 of the approving SE for NEDC-32906P, Supplement 3-A, Revision 1, states, in part:

> When the Void Reactivity Coefficient Correction Model Condition [i.e., Limitation 21 discussed above] and the Void Reactivity Coefficient Correction Model Basis Condition [i.e., Limitation 22 discussed above] specified in this SE are met, the additional 10 percent margin to the fuel centerline melt and the one percent cladding circumferential plastic strain criteria is no longer required for TRACG04.

Based on the above passage and the information provided by the licensee, the NRC staff concluded that the appropriate limitations for use of TRACG04 had been satisfied to permit removing the 10-percent margin requirement from the fuel centerline melt and cladding circumferential plastic strain acceptance criteria. The staff notes that the licensee refers specifically to the pressurization transient events; therefore, this conclusion applies similarly. Any remaining AOOs not analyzed using TRACG04 would be subject to the above condition. The NRC staff observed, as documented in "Supplemental Reload Licensing Report for Monticello Reload 26 Cycle 27," that core-wide transients were also analyzed using TRACG04.

The Cycle 27 SRLR was provided by letter dated July 8, 2013 (Reference 87).

Condition 12: LHGR and Exposure Qualification

IMLTR SER Condition

In MFN 06-481, GE committed to submit plenum fission gas and fuel exposure gamma scans as part of the revision to the T-M licensing process. The conclusions of the plenum fission gas and fuel exposure gamma scans of GE 10x10 fuel designs as operated will be submitted for NRC staff review and approval. This revision will be accomplished through amendment to General Electric Standard Application for Reactor Fuels (GESTAR) II or in a T-M licensing topical report. PRIME (a newly developed T-M code) has been submitted to the NRC staff for review. Once the PRIME licensing topical report and its application are approved, future license applications for EPU and MELLLA+ referencing NEDC-33173P must utilize the PRIME T-M methods.

PUSAR Disposition

At the time of the EPU application submittal, the PRIME topical report was under review by the NRC staff. Therefore, the MNGP EPU application is based on the GSTRM T-M methodology. The staff finds that this is consistent with the condition based on the state of its review of the PRIME T-M methods.

RAI-SNPB-12 requested that the licensee describe how conditions 12 and 14 of the SER for the IMLTR will be met in subsequent cycle analyses. Limitation 12 involves the use of updated T-M analysis methods for future EPU and MELLLA+ license applications. The response states that the updated T-M methods (PRIME) are currently under NRC review, however, GEH has committed to submit a supplement to the IMLTR describing the implementation of the updated T-M models into the safety analysis codes; the transmittal letter to this supplement will provide the schedule for the upgrade (see Reference 5). The IMLTR supplement was submitted by GEH by letter dated July 10, 2009 (Reference 29). Therefore, the NRC staff has reasonable assurance that if the NRC staff approves the updated T-M models, the code upgrade approach taken through the IMLTR will ensure that the condition is met for future cycle reload analyses through the approved GESTAR II process. For the current MNGP EPU application, compliance with Condition 14 (see below) is sufficient to address adequacy of the current licensing analysis methodology.

Updated Information – PRIME Sensitivities

On December 26, 2013, the licensee submitted a report pursuant to 10 CFR 50.46(a)(3) (Reference 97). In the report, the licensee stated that the estimated effect of a transition to a PRIME-based ECCS evaluation model would increase the predicted peak cladding temperature from the limiting loss of coolant accident by approximately 45°F.

The licensee stated in its July 8, 2013, letter, that while PRIME was not generally used for the MNGP EPU application, a single limiting LOCA case was performed with PRIME fuel properties to demonstrate the conservatism in the above estimate. In the limiting large break case from the EPU submittal, a comparison between the submitted GESTR-M-based ECCS evaluation and a PRIME-based evaluation showed that the resulting PCT increased by 10°F, meaning that the previous 45°F estimate was conservative.

For the Cycle 27 reload safety analysis, the licensee observed that the PRIME transition was not complete at the time the analyses were performed. The licensee also stated that the pressurization transient analyses and stability analyses performed using TRACG04 used PRIME-based fuel properties in the Cycle 27 analyses.

Since the licensee submitted its EPU request significantly prior to the NRC's approval of PRIME, and has provided information to indicate both that the licensee is transitioning to PRIME-based safety analysis methods and to estimate the effects of an upgrade to PRIME in the ECCS evaluation, the NRC staff concludes that the supplemental information submitted by the licensee, regarding the status of its PRIME implementation, does not affect the acceptability of the requested EPU.

Finally, it should be noted that the licensee continues to observe appropriate IMLTR penalties on the use of GESTR-based models, as discussed in the NRC staff evaluation of Condition 14, below.

Condition 13: Application of 10 Weight Percent Gadolinia

IMLTR SER Condition

Before applying 10 wt% Gd [gadolinium, loaded as burnable absorber] to licensing applications, including EPU and expanded operating domain, the NRC staff needs to review and approve the T-M licensing topical report demonstrating that the T-M acceptance criteria specified in GESTAR II and Amendment 22 to GESTAR II can be met for steady-state and transient conditions. Specifically, the T-M application must demonstrate that the T-M acceptance criteria can be met for TOP and MOP conditions that bounds the response of plants operating at EPU and expanded operating domains at the most limiting statepoints, considering the operating flexibilities (e.g., equipment out-of-service). Before the use of 10 wt% Gd for modern fuel designs, NRC must review and approve the TGBLA06 qualification submittal. Where a fuel design refers to a design with Gd-bearing rods adjacent to vanished or water rods, the submittal should include specific information regarding acceptance criteria for the qualification and address any downstream impacts in terms of the safety analysis. The 10 wt% Gd qualifications submittal can supplement this report.

PUSAR Disposition

Appendix A of the PUSAR states that the MNGP EPU bundle design will utilize less than 10 wt% Gd in the fuel. Therefore, the NRC staff finds that the licensee's application is consistent with this condition, and the disposition is acceptable.

Condition 14: Part 21 Evaluation of GSTRM Fuel Temperature Calculation

IMLTR SER Condition

Any conclusions drawn from the NRC staff evaluation of the GE's Part 21 report will be applicable to the GESTR-M T-M assessment of this SE for future license application. The NRC staff determined that until such time that GE benchmarks the GSTRM methodology, the critical pressure ($P_{critical}$) acceptance criteria will be reduced by 350 psi. This adjusted $P_{critical}$ must be used to verify that the LHGR limit for the current fuel designs remains applicable with burnup.

PUSAR Disposition

Appendix A of the PUSAR states that Condition 14 is not applicable. The disposition of Condition 14 from the NRC SER for the IMLTR is not consistent with the SER. This SER incorporates Appendix F, which discusses the findings of the NRC staff review of GE's Part 21 evaluation of non-conservatisms in the GSTRM thermal mechanical (T-M) methodology. The NRC staff concludes in Appendix F that an additional margin of 350 psi is required in the critical pressure analysis.

The NRC staff requested additional information in RAI-SNPB-6 regarding Condition 14, asking the licensee to confirm that the additional margin of 350 psi has been included in the safety analysis. The licensee's March 19, 2009, response states that the penalty has not been incorporated (Reference 3).

The response to RAI-SNPB-6 states that the licensee expects that the critical pressure penalty will be removed on the basis of supplemental information provided to GEH's notification of a Part 21 evaluation. However, the NRC staff has found that the basis provided in the supplemental information is not sufficient to justify the removal of the critical pressure penalty (Reference 30). Therefore, the NRC staff requested supplemental information in RAIs SNPB-10, SNPB-11, and SNPB-12.

RAI-SNPB-10 requested that the PUSAR be updated to incorporate the 350 psi. The licensee's July 23, 2009, response states that the penalty has been included in the generic thermal-mechanical operating limit (TMOL) for GE14 as reported in the revised GESTAR II compliance document (Reference 31). The response further states that the current core design accommodates the revised TMOL and that future cycle designs with GE14 fuel will adopt the generic TMOL. The NRC staff finds that this approach is acceptable to incorporate the 350 psi margin.

RAI-SNPB-11 requested that the licensee provide details of the specific actions taken to address the penalty. The information provided in the response to RAI-SNPB-10 is sufficient to describe how the penalty is taken into account and how it is applied in future reload cycle licensing evaluations.

RAI-SNPB-12 requested that NSPM describe how conditions 12 and 14 of the SER for the IMLTR are met for subsequent cycles. The response to RAI-SNPB-10 is sufficient to describe how limitation 14 will be met for subsequent cycles and is consistent with the approved GESTAR II reload licensing process (Reference 16).

Updated Information – Continued Use of Critical Pressure Penalty

The licensee stated in its July 8, 2013, supplemental letter, that the GE14 thermal-mechanical operating limit applied to Cycle 27 analyses incorporated the 350 psi penalty on fuel rod critical pressure in the fuel rod internal pressure design ratio (Reference 87).

Condition 15: Void Reactivity 1

IMLTR SER Condition

The void reactivity coefficient bias and uncertainties in TRACG for EPU and MELLLA+ must be representative of the lattice designs of the fuel loaded in the core.

PUSAR Disposition

Appendix A of the PUSAR states that TRACG methods are not utilized in the current application. Therefore, the NRC staff concurs with the disposition that this condition does not apply to the MNGP EPU LAR.

Updated Information – TRACG04 Migration

The licensee stated in its July 8, 2013, supplemental letter, that for the use of TRACG methods associated with the Cycle 27 safety analyses, the TRACG04 model has characterized the void coefficient biases and uncertainties for the GE14 and GNF-2 lattice types over an encompassing domain of operational conditions as a function of instantaneous voids, void history, and exposure (Reference 87). During the TRACG evaluations, these three transient inputs to the model are calculated and provided locally for the plant and cycle-specific analysis conditions to determine the local biases and uncertainties to be applied during the course of the transient calculation. This information applies also to Conditions 16 and 20, below.

Based on the fact that the licensee is using the GE14 bundle design, and void reactivity coefficient biases and uncertainties based on instantaneous voids, void history, and exposure are calculated for plant- and cycle-specific analysis conditions, the NRC staff concludes that Condition 15 is satisfied for the TRACG04 migration at MNGP. Therefore, the NRC staff determined that the TRACG04 migration would compromise the acceptability of the requested EPU.

Condition 16: Void Reactivity 2

IMLTR SER Condition

TRACG internally models the response surface for the void coefficient biases and uncertainties for known dependencies due to the relative moderator density and exposure on nodal basis.

Therefore, the void history bias determined through the methods review can be incorporated into the response surface "known" bias, or through changes in lattice physics/core simulator methods for establishing the instantaneous cross-sections. Including the bias in the calculations negates the need for ensuring that plant-specific applications show sufficient margin. For application of TRACG to EPU and MELLLA+ applications, the TRACG methodology must incorporate the void history bias. The manner in which this void history bias is accounted for will be established by the NRC staff's SE approving NEDE-32906P, Supplement 3, "Migration to TRACG04/PANAC11 from TRACG02/PANAC10," May 2006. This limitation applies until the new TRACG/PANAC methodology is approved by the NRC staff.

PUSAR Disposition

Appendix A of the PUSAR states that TRACG methods are not utilized in the current application. Therefore, the NRC staff concurs with the disposition that this condition does not apply to the licensee's application for EPU.

Updated Information – TRACG04 Migration

Refer to Condition 15, above, for the impact of TRACG04 migration on adherence to this condition.

Condition 17: Steady State Five Percent Bypass Voiding

IMLTR SER Condition

The instrumentation specification design bases limit the presence of bypass voiding to 5 percent (local power range monitor (LPRM) levels). Limiting the bypass voiding to less than 5 percent for long-term steady operation ensures that instrumentation is operated within the specification. For EPU and MELLLA+ operation, the bypass voiding will be evaluated on a cycle-specific basis to confirm that the void fraction remains below 5 percent at all LPRM levels when operating at steady-state conditions within the MELLLA+ upper boundary. The highest calculated bypass voiding at any LPRM level will be provided with the plant-specific SRLR.

PUSAR Disposition

The disposition of Condition 17 is consistent with the NRC SER for the IMLTR, however, the predicted steady state bypass void fraction has not been provided in Appendix A of the PUSAR. The NRC staff requested additional information in RAI-SNPB-7 regarding the bypass void analysis. In particular the NRC staff requested additional information to garner reasonable assurance that the 5 percent steady state bypass void fraction limit will be maintained on a cycle-specific basis.

The licensee's March 19, 2009, response to RAI-SNPB-7 provides a description of the analysis methodology that is applied for each cycle reload (Reference 3). The analysis assumptions are conservative and consistent with the NRC staff's generic review. The response provides the NRC staff with reasonable assurance that the steady-state bypass void fraction will be evaluated on a cycle-specific basis, evaluated using appropriately conservative analysis assumptions, checked against the 5 percent limit, and reported in the supplemental reload licensing report (SRLR). Therefore, the NRC staff finds that the IMLTR condition is acceptably met.

Condition 18: Stability Setpoints Adjustment

IMLTR SER Condition

The NRC staff concludes that the presence of bypass voiding at the low-flow conditions where instabilities are likely can result in calibration errors of less than 5 percent for oscilation power range monitor (OPRM) cells and less than 2 percent for APRM signals. These calibration errors must be accounted for while determining the setpoints for any detect-and-suppress long-term methodology. The calibration values for the different long-term solutions are specified in the associated sections of the SER for the IMLTR, discussing the stability methodology.

PUSAR Disposition

The disposition of Condition 18 in Appendix A of the PUSAR is not sufficiently detailed for the NRC staff to ensure consistency between the disposition and the NRC staff's SER for the IMLTR. The NRC staff reviewed a description of the operating limit minimum critical power ratio (OLMCPR) used in the OPRM setpoint analysis, however, required some additional clarification based on the statements in Section 2.8.3.1. The NRC staff requested additional information (RAI-SNPB-8) regarding the OPRM setpoint determination.

The licensee's March 19, 2009, response to RAI-SNPB-8 provides additional detailed information regarding the consistency between the MNGP's EPU OPRM setpoint calculation (Reference 3) and the guidance provided in a November 8, 2006, letter from D. H. Hinds of GE to the NRC (Reference 26). The licensee's response states that the OPRM illustrative example does not include the OLMCPR adder. The licensee's response also confirms that the MNGP EPU operating cycle OPRM setpoint calculation will not incorporate the OLMCPR adder; thereby the cycle analysis is consistent with MFN 08-693. The NRC staff found that the MNGP approach is consistent with the guidance and, therefore, is acceptable.

Condition 19: Void Quality Correlation 1

IMLTR SER Condition

For applications involving PANACEA/ODYN/ISCOR/TASC for operation at EPU and MELLLA+, an additional 0.01 will be added to the OLMCPR, until such time that GEH expands the experimental database supporting the Findlay-Dix void-quality correlation to demonstrate the accuracy and performance of the void-quality correlation based on experimental data representative of the current fuel designs and operating conditions during steady-state, transient, and accident conditions.

PUSAR Disposition

Pending the GEH resolution of Condition 19, Appendix A of the PUSAR states that the 0.01 OLMCPR adder will be applied. The NRC staff finds this acceptable.

Updated Information – TRACG04 Migration

The licensee stated in its July 8, 2013, supplemental letter, that this limitation is directed to the determination of the OLMCPR when PANACEA/ODYN/ISCOR/TASC is used (Reference 87). Since TRACG is being used in the Cycle 27 analysis, the limitation is not applicable and the

0.01 adder has not been included. Because the licensee is using updated methods, the NRC staff agrees that this adder may be removed, as discussed on Page 60 of the NRC safety evaluation approving NEDC-32906P, Supplement 3-A, Revision 1.

Condition 20: Void Quality Correlation 2

IMLTR SER Condition

The NRC staff is currently reviewing Supplement 3 to NEDE-32906P, "Migration to TRACG04/PANAC11 from TRACG02/PANAC10," dated May 2006. The adequacy of the TRACG interfacial shear model qualification for application to EPU and MELLLA+ will be addressed under this review. Any conclusions specified in the NRC staff SE approving Supplement 3 to NEDC-32906P will be applicable as approved.

PUSAR Disposition

The MNGP EPU amendment application does not rely on TRACG. Therefore, this condition is not applicable to the current review.

Updated Information – TRACG04 Migration

Refer to Condition 15, above, for the impact of TRACG04 migration on adherence to this condition.

Condition 21: Mixed Core Method 1

IMLTR SER Condition

Plants implementing EPU or MELLLA+ with mixed fuel vendor cores will provide plant-specific justification for extension of GE's analytical methods or codes. The content of the plant-specific application will cover the topics addressed in this SE as well as subjects relevant to application of GE's methods to legacy fuel. Alternatively, GE may supplement or revise topical report NEDC-33173P for mixed core application.

PUSAR Disposition

The MNGP EPU core will consist entirely of GE14 fuel. Therefore, the mixed core method condition is not applicable to the current review.

Condition 22: Mixed Core Method 2

IMLTR SER Condition

For any plant-specific applications of TGBLA06 with fuel type characteristics not covered in this review, GE needs to provide assessment data similar to that provided for the GE fuels. The Interim Methods review is applicable to all GE lattices up to GE14. Fuel lattice designs, other than GE lattices up to GE14, with the following characteristics are not covered by this review:

- Square internal water channels or water crosses
- Gd rods simultaneously adjacent to water and vanished rods

- 11x11 lattices
- Mixed oxide (MOX) fuel

The acceptability of the modified epithermal slowing down models in TGBLA06 has not been demonstrated for application to these or other geometries for expanded operating domains.

Significant changes in the Gd rod optical thickness will require an evaluation of the TGBLA06 radial flux and Gd depletion modeling before being applied. Increases in the lattice Gd loading that result in nodal reactivity biases beyond those previously established will require review before the GE methods may be applied.

PUSAR Disposition

The licensee stated that the MNGP EPU core will consist entirely of GE14 fuel. Therefore, the mixed core method condition is not applicable to the current review.

Condition 23: MELLLA+ Eigenvalue Tracking

IMLTR SER Condition

In the first plant-specific implementation of MELLLA+, the cycle-specific eigenvalue tracking data will be evaluated and submitted to NRC to establish the performance of nuclear methods under the operation in the new operating domain. The following data will be analyzed:

- Hot critical eigenvalue,
- Cold critical eigenvalue,
- Nodal power distribution (measured and calculated traversing incore probe (TIP) comparison),
- Bundle power distribution (measured and calculated TIP comparison),
- Thermal margin,
- Core flow and pressure drop uncertainties, and
- The MCPR Importance Parameter Criterion (e.g., determine if core and fuel design selected is expected to produce a plant response outside the prior experience base).

Provision of evaluation of the core-tracking data will provide the NRC staff with bases to establish if operation at the expanded operating domain indicates: (1) changes in the performance of nuclear methods outside the EPU experience base; (2) changes in the available thermal margins; (3) need for changes in the uncertainties and NRC-approved criterion used in the SLMCPR methodology; or (4) any anomaly that may require corrective actions.

PUSAR Disposition

The scope of the current amendment application does not request approval for operation in the MELLLA+ domain. Therefore, the condition is not applicable to the current review.

Condition 24: Plant Specific Application

IMLTR SER Condition

The plant-specific applications will provide prediction of key parameters for cycle exposures for operation at EPU (and MELLLA+ for MELLLA+ applications). The plant-specific prediction of these key parameters will be plotted against the EPU Reference Plant experience base and MELLLA+ operating experience, if available. For evaluation of the margins available in the fuel design limits, plant-specific applications will also provide quarter core map (assuming core symmetry) showing bundle power, bundle operating LHGR, and MCPR for beginning of cycle (BOC), middle of cycle (MOC), and end of cycle (EOC). Since the minimum margins to specific limits may occur at exposures other than the traditional BOC, MOC, and EOC, the data will be provided at these exposures.

PUSAR Disposition

Appendix A of the PUSAR states that the required information is provided in the PUSAR in Figures 2.8-1 through 2.8-18. The NRC staff reviewed these figures and found that the information provided in the figures is sufficient to meet the requirements of Condition 24.

Conclusion

The NRC staff has reviewed the information provided in the PUSAR and the licensee's responses to the NRC staff's requests for additional information. On the basis of the disposition of IMLTR SE conditions contained in Appendix A of the PUSAR, and the licensee's RAI-responses, the NRC staff has concluded that the MNGP safety analyses were performed consistent with the approval of the GEH analytical methods described in NEDC-33173P.

Therefore, the NRC staff finds that the analysis methods are acceptable.

2.8.7.2 Additional Stability-Related Requests for Additional Information (RAI)

On March 23, 2009, the NRC staff issued an RAI (ADAMS Accession No. ML090820145 (proprietary)) containing, among other things, thirteen (13) items associated with the licensee's stability evaluation. The RAIs served several purposes in addition to developing a clearer understanding of the regulatory basis for the acceptability of NSPM's stability evaluation. This included providing additional information and clarification regarding the current status of NSPM's stability solution implementation and documentation of information discussed during the NRC staff's audit of the MNGP stability solution. The licensee responded by its July 23, 2009, letter. Since not every item contained in the RAI-is specifically evaluated in the sections above, the licensee's responses which are not evaluated above are summarized below:

RAI-2.8.3-1 NEDC-33322P refers to a demonstration analysis performed to determine the limiting OPRM setpoint that results in a non-OLMCPR setting stability transient. Discuss whether the initial conditions in this analysis are the same as those used in any analyses supporting the Option III license amendment request. Address any differences.

The licensee stated that the OPRM III license amendment (Amendment No. 159, dated January 30, 2009 (Reference 70)) analyses were performed at EPU conditions. The final

Option III setpoint is affected by three components: the hot channel oscillation magnitude (HCOM), delta over initial CPR versus oscillation magnitude (DIVOM), and the dual pump trip delta CPR (DIRPT). The HCOM analysis is not affected by operating power. The DIVOM was calculated on a cycle-specific basis, [[ ]]; therefore, there was no change from OLTP to EPU. [[ ]] the setpoint values when compared to OLTP.

In the case of MNGP, the largest allowable Option III setpoint of 1.15 with 16 confirmation counts was used because Option III does not affect the OLMCPR limit. With the maximum allowed Option III setpoints, the [[ ]].

RAI-2.8.3-2 What version of emergency operating procedures [EOPs] is currently implemented at MNGP? Provide a short description of the process used to ensure that the EPG variables (e.g. Hot Shutdown Boron Weight (HSBW) and Heat Capacity Temperature Limit (HCTL)) are adequate under CPPU conditions.

The licensee stated that it has implemented EOPs in accordance with the BWROG Emergency Procedure Guidelines (EPG)/Severe Accident Guidelines (SAG), Rev 2. In particular, the EOPs require immediate water level reduction below the sparger and prompt initiation of SLC upon detection of oscillations.

The EPG variables, including the HSBW and HCTL, are performed for each cycle of operation to account for the fuel design and loading.

RAI-2.8.3-3 Provide a short description of how the Stability Mitigation Actions (e.g. immediate water level reduction and early boron injection) are implemented at MNGP. Does operation at CPPU conditions require modification of any operator instructions?

The licensee provided a copy of EOP flowchart C.5-2007, "Failure to Scram," and stated that the Stability Mitigation Actions have been incorporated. The licensee stated that no modifications are necessary to accommodate EPU operation.

The licensee provided the most recent revision to C.5-2007 in a letter dated January 21, 2013, in response to Gap Analysis Item 17 (Reference 57). The licensee indicated in its response that no steps had been eliminated, but only moved to a separate procedure.

RAI-2.8.3-4 What is the current status of LTSS [Long-Term Stability Solution] Option III implementation? When will it be armed in the plant?

The licensee stated, in part, that Option III was installed during the April-May 2009 outage. At the time of the NRC staff audit (May 21, 2009), all the hardware was installed and operational on the plant and the simulator. The trip, however, was not armed. MNGP underwent a 90-day testing period to ensure that false scrams don't occur. At the end of the testing period, the Option III trips were armed. For the testing period, MNGP was operating on backup stability protection (BSP), which is similar to the old Boiling Water Reactor Owners' Group (BWROG) Interim Corrective Actions (ICA). The NRC staff reviewed the specific ICA procedures, which are contained in Abnormal Procedure C.4-B.05.01.02.A "Control of Neutron Flux oscillations."

In its January 21, 2013, letter (Reference 57), the licensee states that the OPRM-based Option III long term stability solution equipment has been installed and was turned over to the Operations in September 2009. The monitoring and evaluation period has been completed.

RAI-2.8.3-5 Is Option III hardware implemented in the Monticello simulator? What are the plans and overall schedule for operator training?

The licensee stated, in part, that operator training would be completed before completion of the 90-day testing period. However, during the audit on May 21, 2009, the NRC staff was informed that operator training for the new Option III had been completed ahead of schedule.

RAI-2.8.3-6 Will the Option III hardware implemented in Monticello have the DSS/CD software installed for testing purposes? What are the testing plans?

The licensee stated, in part, that DSS/CD (Detect and Suppress Solution – Confirmation Density) software is installed and is in the process of being tested. DSS/CD will not be armed during this cycle and it will continue to collect data in preparation for a possible MELLLA+ upgrade in the future.

RAI-2.8.3-7 Will the DIVOM curve be implemented as cycle-specific in Monticello? If the generic DIVOM slope will not be used, provide a Reference to the DIVOM analysis methodology that will be used.

The licensee stated, in part, that the DIVOM [delta over initial CPR versus oscillation magnitude] curve is evaluated on a cycle-specific basis. The NRC staff notes that per BWR guideline GE-NE-0000-0028-9714-R1, "Plant Specific Regional Mode DIVOM Procedure Guideline," if the TRACG04 cycle-specific model of the plant cannot be made to oscillate (i.e., the plant is very stable) the generic DIVOM slope is used. In addition, if a cycle-specific DIVOM slope is calculated that is smaller than the generic 0.45 slope, the generic is also used. [[

]].

RAI-2.8.3-8 The Nine Mile Point 2 Instability Event showed some reduced sensitivity to low-level oscillations if the Option III parameters were set at minimum sensitivity settings. Have the lessons-learned from this event been incorporated in Monticello?

The licensee stated, in part, that the lessons learned from Nine Mile Point 2 have been incorporated. At MNGP, the cutoff frequency is set to 1 Hertz and the period tolerance to 100 milliseconds.

RAI-2.8.3-9 In September 2006, the Hope Creek plant experienced a half-scram indication from the Option III hardware while withdrawing peripheral control rods in low-power bundles. Hope Creek implemented recommendations for speed of rod withdrawal inside the armed region. Have these recommendations been incorporated in the Monticello operator training?

The licensee stated, in part, that it evaluated the Hope Creek event (INPO OE23808), and concluded that no additional operator training is required for control rod motion and spurious scram avoidance. The reason is that [[ ]]. The Hope Creek event was, in part, caused by a very small Option III setpoint setting of 1.06. [[ ]].

RAI-2.8.3-10 Assuming a conservative OPRM setpoint of 1.15, provide the hot-spot fuel temperature as function of time before the scram. Evaluate this fuel temperature oscillation against pellet-clad interaction (PCI) limits. Assume the steady-state fuel conditions before the oscillations are those of point C of Figure 2.8.3-20 of NEDC-33322P (the highest power point in the BSP scram region).

The licensee provided the acceptance criteria for GE14 fuel in terms of LHGR. The LHGR is designed to prevent the ultimate SAFDL of fuel centerline temperature and clad plastic strain <1 percent. The licensee combined the answer to this RAI-with the answer to a related loss of feedwater heating (LOFWH) RAI-(RAI-2.8.5.1-2). The licensee stated that for both events (LOFWH and stability) the maximum fraction of linear power density is [[ ]], which is well below the acceptance criteria.

The licensee argued that PCI limits [[ ]]. The licensee's argument is based on the fact that the [[ ]].

[[ ]].

Given the [[ ]], the NRC staff finds that the licensee's response is acceptable.

RAI-2.8.3-11 Provide the following information relevant to anticipated transient without scram (ATWS)-stability: (1) turbine bypass capacity; (2) percent of feedwater (FW) flow that is driven by electric or steam turbine pumps; (3) location of the extraction steam that feeds the feedwater heaters; (4) location of the extraction steam that feeds the FW steam-driven pumps (if any); (5) FW sparger elevation with respect to top of active fuel; and (6) location of the SLC injection point in the vessel.

The licensee provided the requested information. MNGP has approximately 11.5 percent bypass flow capacity for the turbine (at EPU power level). It also has 100 percent motor-driven feedwater pumps. While this configuration makes water level control easier from the point of view of the operator (feedwater pumps are easier to control than high pressure coolant injection pumps), the operator is required to lower the water level manually per the emergency operating procedures. If the feedwater pumps were partially steam-driven, the water level reduction would occur automatically.

During the NRC staff audit, a turbine trip ATWS was studied in the simulator. The operators lowered the water level as directed within 120 seconds of event initiation.

RAI-2.8.3-12 Following a turbine trip with full bypass and failure to scram, provide; (1) the maximum FW flow that the available pumps can deliver; and (2) the ultimate FW temperature after the FW heaters reach equilibrium with the new steam extraction conditions.

Since all FW pumps are motor-driven, essentially 100 percent feedwater flow will be available. The licensee notes that as the pressure increases due to the small bypass capacity, the FW flow may be reduced.

RAI-2.8.3-13 Discuss any control system actions that are relevant to ATWS-stability events. Examples are: automatic switching of extraction steam for steam driven pumps, flow runbacks on high pressure ...

MNGP does not have steam-driven FW pumps or a FW runback on high pressure. ATWS actions are those required by the ATWS rule in 10 CFR 50.62. It includes the automatic RPT on high pressure.

2.8.7.3 Additional Reactor Systems Review Area – TS Changes

2.8.7.3.1 TS 3.5.1, "Emergency Core Cooling System and Reactor Core Isolation Cooling System," Actions M and N

The licensee proposed to eliminate Action M, which permits one Automatic Depressurization System valve to be inoperable under any of the following conditions:

- With one LPCI pump inoperable,
- With one LPCI subsystem inoperable for reasons other than an inoperable pump, or with one core spray subsystem inoperable, or
- With one LPCI pump in both LPCI subsystems inoperable.

The licensee proposed this elimination because the EPU LOCA analyses do not support this inoperability assumption. Because the elimination of this action is necessary to support the EPU safety analyses, the NRC staff finds this proposed change acceptable.

The licensee also proposed to eliminate associated language referring to the operability of certain ADS valves from the last condition entry for Action N to reflect the assumption of full ADS capability analyzed in the LOCA analyses. The licensee also proposed to add an entry condition that becomes the new third condition entry for Action N to reflect a requirement to place the unit in MODE 3 with reactor steam pressure ≤150 psig when an ADS valve in combination with other ECCS components or subsystems becomes inoperable without regard to HPCI System operability.

Based on the above, the NRC staff determined that modification of these entry conditions preserves the necessary level of ECCS quality reflected in the above evaluated EPU safety analyses is acceptable.

2.8.7.3.2 TS 3.3.5.1, "ECCS Instrumentation," Function 1.e, Reactor Steam Dome Pressure Permissive - Bypass Timer (Pump Permissive)

In its October 30, 2012, supplement, the licensee proposed to modify the Reactor Steam Dome Pressure Permissive – Bypass Timer (Pump Permissive) allowable value from ≤ 22 minutes to ≤ 18 minutes (Reference 61). The NRC staff received the revised calculation to support the setpoint change in a letter dated January 31, 2013. In a May 30, 2013, letter, the licensee submitted a revised version of CA-03-036 (Revision 2). The licensee stated on Page 10 of 16 of CA-03-036, that "an analytical limit of 1208 seconds ensures actuation of ADS at approximately 1700 degrees F [peak cladding temperature], well before reaching the 2200 degrees F limit [prescribed by 10 CFR 50.46(b)(1)]. The analytical limit will be defined as 1200 seconds, or 20.0 minutes" (Reference 68).

The analyzed reactor water cleanup system (RWCU) line break is considered a small break LOCA. In the licensee's 10 CFR 50 Appendix K based ECCS evaluation model, a phase of the postulated small break loss of coolant accident, for which the ADS is necessary, occurs when the reactor coolant has uncovered the core, and the reactor pressure remains too high to permit successful operation of the low-pressure ECCS. This phase is known as the adiabatic heatup, and ADS actuation ends this phase by reducing the reactor pressure to cut-in pressure for the low pressure ECCS (LPCI and LPCS). At that point in time, water will enter the core and refill the vessel, ending the cladding temperature transient.

The licensee's evaluation showed that actuating the ADS after the analytical limit – 20 minutes – will result in acceptable calculated ECCS cooling performance because the predicted PCT will not exceed 1700°F at that time. In the licensee's ECCS evaluations for demonstrating compliance with 10 CFR 50.46 requirements, the predicted PCT for the most severe postulated loss of coolant accident is 2140°F. The licensee's assessment indicates that the results of a postulated RWCU line break are significantly less severe than: (1) the analysis of record for ECCS performance; and (2) the acceptance criterion set forth at 10 CFR 50.46(b)(1). Based on this significant margin, the NRC staff determined that the licensee's evaluation was adequate and that the proposed 20 minute analytical limit is acceptable.

2.9 Source Terms and Radiological Consequences Analyses

This evaluation addresses the impact of the proposed changes on previously analyzed design-basis accident radiological consequences and the acceptability of the revised analysis results. The regulatory requirements on which the staff based its acceptance are the design-basis accident dose acceptance criteria in Title 10 of the Code of Federal Regulations Part 50.67 (10 CFR 50.67), as supplemented by Regulatory Position 4.4 of Regulatory Guide 1.183, Alternative Radiological Source Terms for Evaluating Design-Basis Accidents at Nuclear Power Reactors," and 10 CFR Part 50 Appendix A, GDC-19, "Control Room," as supplemented by Section 6.4 of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (SRP). Except where the licensee proposed a suitable alternative, the NRC staff used the regulatory guidance provided in the following documents in performing this review:

- Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design-Basis Accidents at Nuclear Power Reactors"

- SRP Section 15.0.1, "Radiological Consequence Analysis Using Alternative Source Terms"

- SRP Section 11.1, "Source Terms"

The NRC staff also considered relevant information in the MNGP USAR and the TSs.

2.9.1 Source Terms Analyses

Regulatory Evaluation

The NRC staff reviewed the radioactive source term associated with the EPU to ensure the adequacy of the sources of radioactivity used by the licensee as input to calculations, and to verify that the radioactive waste management systems have adequate capacity for the treatment of radioactive liquid and gaseous wastes. The NRC staff's review included the parameters used to determine: (1) the concentration of each radionuclide in the reactor coolant; (2) the fraction of fission product activity released to the reactor coolant; (3) concentrations of all radionuclides other than fission products in the reactor coolant; (4) leakage rates and associated fluid activity of all potentially radioactive water and steam systems; and (5) potential sources of radioactive materials in effluents that are not considered in MNGP's USAR related to liquid waste management systems and gaseous waste management systems. The NRC's acceptance criteria for source terms are based on: (1) 10 CFR Part 20, insofar as it establishes requirements for radioactivity in liquid and gaseous effluents released to unrestricted areas; (2) 10 CFR Part 50, Appendix I, insofar as it establishes numerical guides for design objectives and limiting conditions for operation to meet the "as low as is reasonably achievable" criterion; and (3) GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents. Specific review criteria are contained in SRP Section 11.1.

The general design criteria discussed herein are those currently specified in 10 CFR Part 50, Appendix A. The applicable MNGP Principal Design Criteria predate these Appendix A criteria. These MNGP Principal Design Criteria are listed in the MNGP USAR Section 1.2, "Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (32 *FR* 10213, July 11, 1967). An evaluation comparing the MNGP design basis to the AEC-proposed General Design Criteria of 1967 is presented in MNGP USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria." While MNGP is not explicitly licensed to the current General Design Criteria or the 1967 AEC-proposed GDC, the licensee has also made a comparison of the current GDC to the applicable AEC-proposed General Design Criteria. For the current GDC-60, the licensee's evaluation of the analogous 1967 AEC-proposed General Design Criteria is also contained in MNGP USAR, Appendix E: draft GDC-70.

Technical Evaluation

The core isotopic inventory is a function of the core power level, while the reactor coolant isotopic activity concentration is a function of the core power level as well as leakage from the fuel, radioactive decay, and removal by coolant purification systems. The licensee's analyses supporting the EPU amendment request included a core isotopic inventory calculated for the EPU conditions. The assumed inventory of fission products in the reactor core and available for release to the containment is based on the maximum power level of 2,004 MWth, corresponding to current fuel enrichment and fuel burnup, which is 1.20 times the MNGP original licensed thermal power and 1.13 times the CLTP of 1,775 MWth. Design-basis accident analyses for radiological consequences are performed at 1.02 times the proposed increase in licensed thermal power to account for a 2-percent instrumentation uncertainty.

During reactor operation, the coolant passing through the core region becomes radioactive as a result of exposure to neutron flux. The coolant activation, especially nitrogen-16 (N-16) activity, is the dominant source in the turbine building and in the lower regions of the drywell. The increase in activation of the water in the core region due to the power increase is approximately proportional to the increase in thermal power. The licensee states, and the NRC staff agrees, that the margin in the MNGP design-basis for reactor coolant activation product concentration significantly exceeds potential increases due to operation at EPU conditions. Therefore, no change is required in the design-basis reactor coolant activation product concentration for operation at EPU.

The reactor coolant contains activated corrosion products, which result from metallic materials entering the water and being activated in the reactor region. Under the EPU conditions, both the feedwater flow and the activation rate in the reactor region increase with power. The licensee has determined the net result to be an increase in the activated corrosion product production. Further, the licensee states that the total activated corrosion product activity as a result of the EPU is approximately 41 percent of the total corrosion product activity currently assumed as a design basis. An evaluation of steam fission and corrosion products based upon American National Standards Institute (ANSI) 18.1 methodology at assumed uprate conditions show the plant design basis to be bounding with respect to predicted concentrations. Therefore, no change is required in the design-basis corrosion product activity for the EPU.

Fission products in the reactor coolant are assumed to partition between the steam and the reactor coolant. Noble gas activity released from the core is assumed to transport with the steam. This activity is the noble gas offgas that is included in the plant design. The licensee stated that the offgas rates at EPU conditions are well below the original design-basis value. Therefore, the NRC staff agrees that no change is required to the MNGP design-basis for offgas activity as the result of the EPU.

The fission product activity in the reactor coolant is the result of releases from damaged fuel rods. The licensee calculated the reactor coolant fission product activity level to be less than 2 percent of the cumulative design-basis fission product activity levels. Therefore, the NRC staff agrees that no change to the MNGP design-basis fission product activity is required as the result of the EPU.

During power operation, radiation sources in the core are directly related to the fission rate. The sources include radiation from the fission process, accumulated fission products, and neutron reactions as a secondary result of fission. Post-operation, the dominant source of radiation is from the aforementioned accumulated fission products in the core. Typically, and for MNGP, these sources are defined in terms of activity released per unit of reactor power, i.e., Curies per megawatt-thermal (Ci/MWth). Therefore, for an EPU with all else being equal, the percent increase in the operating source term should be no greater than the percent increase in power.

Post-operation core radiation source (activity) data is needed for post-accident evaluations, which apply different release and transport assumptions to different fission products. The licensee based the core fission product inventories for these evaluations on an assumed fuel irradiation time (typically approximately 3 years), which analytically establishes "equilibrium" activities in the fuel. The licensee asserts, and the NRC staff agrees, that most radiologically significant fission products reach equilibrium within a 60-day period. The licensee's evaluation, which relies on the GE LTR NEDC-33004P-A, "Constant Pressure Power Uprate", uses bounding fuel parameters to calculate the equilibrium core isotopic inventory. The licensee states that the GE evaluation bounds all GEH BWR fuel product lines through GE14. The NRC staff considered GE proprietary information to make its determination, and agrees that the source terms calculated by the licensee are acceptable.

Conclusion

The NRC staff has reviewed the radioactive source term in the reactor coolant and steam associated with the proposed EPU, and concludes that the proposed parameters and resultant composition and quantity of radionuclides are appropriate for the evaluation of the radioactive waste management systems. The NRC staff further concludes that the proposed radioactive source term meets the requirements of 10 CFR Part 20, 10 CFR Part 50, Appendix I, and the current MNGP licensing basis. Therefore, the NRC staff finds the proposed EPU acceptable with respect to source terms for radwaste systems and DBA analyses.

2.9.2 Radiological Consequences Analyses Using Alternative Source Terms

Regulatory Evaluation

The NRC staff reviewed the DBA radiological consequence analyses submitted by the licensee in support of the EPU. The radiological consequence analyses reviewed are the loss-of-coolant accident (LOCA), fuel handling accident (FHA), control rod drop accident (CRDA), and main steam line break accident (MSLBA). The NRC staff's review of each accident analysis included: (1) the sequence of events; and (2) models, assumptions, and values of parameter inputs used by the licensee for the calculation of the total effective dose equivalent (TEDE). The NRC staff's acceptance criteria for radiological consequence analyses using an alternative source term are based on: (1) 10 CFR 50.67, "Accident source term," insofar as it sets standards for radiological dose consequence of a postulated accident; and (2) GDC-19, "Control room," insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem TEDE, as defined in 10 CFR 50.20, for the duration of the accident. Specific review criteria are contained in SRP Section 15.0.1.

The general design criteria discussed herein are those currently specified in 10 CFR Part 50, Appendix A. The applicable MNGP Principal Design Criteria predate these criteria. These Principal Design Criteria are listed in MNGP USAR Section 1.2, "Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (Federal Register 32 FR10213, July 11, 1967). An evaluation comparing the MNGP, Unit 1, design basis to the AEC-proposed General Design Criteria of 1967 is presented in MNGP USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria." While MNGP is not explicitly licensed to the current General Design Criteria or the 1967 AEC proposed GDC, the licensee has also made a comparison of the current GDC to the applicable AEC-proposed General Design Criteria. For the current GDC-19, the licensee's evaluation of the analogous 1967 AEC-proposed General Design Criteria is also contained in MNGP USAR, Appendix E: draft GDC-11.

Technical Evaluation

In its previous review for Amendment No. 148 regarding alternative source term (AST), dated December 7, 2006 (Reference 84), the NRC staff compared the doses estimated by the licensee to the applicable regulatory acceptance criteria and found with reasonable assurance that the licensee's estimates of the offsite and control room doses will continue to comply with the applicable regulatory criteria. The safety evaluation for the AST amendment concludes that the NRC staff found the radiological consequences of DBAs to remain bounding up to a thermal power of 1,880 MWth, or approximately 1.06 times the CLTP. However, the requested EPU license amendment proposes to increase the RTP to 2,004 MWth, or approximately 1.13 times the CLTP. Therefore, the licensee re-analyzed each DBA to determine the effect of the proposed increase in power over that which was previously analyzed. This calculated impact of the EPU on the radiological dose consequence of DBAs is discussed in Section 2.9.2 of the PUSAR. The applicable re-analyzed events are the LOCA, FHA, CRDA, and MSLBA.

As stated in the previous Section 2.9.1, radiation sources in the core are directly related to the fission rate. So, the percent increase in the source term should be no greater than the percent increase in power. As a result, all accidents resulting in damage to the fuel, i.e., LOCA, FHA, and CRDA, can typically be re-evaluated by applying this ratio of increase to the dose consequences, assuming no other changes are made to the accident analysis. For MNGP, the licensee made slight revisions to the accident analyses in addition to incorporating the new EPU power, which in turn resulted in increases in dose consequences that are not directly proportional to the increase in power. Because the MSLBA is not a fuel damage accident, and is instead dependent upon the TS reactor coolant activity concentration limit, the proposed EPU does not affect the resulting dose consequence. However, the licensee also revised the MSLBA analysis of record to incorporate conservatisms in addition to those approved in the previous AST analysis.

The changes to the DBA analyses made by the licensee, and the evaluation by the NRC staff are discussed in further detail in the following sections.

Radiological Consequences of the Design-Basis LOCA

The licensee updated the design-basis LOCA analysis performed for the MNGP implementation of a full-scope conversion to the AST methodology reviewed and approved as Amendment No. 148. In addition to incorporating the proposed change for EPU conditions, the revised analysis implements the source term inventory approved for use in the CLTR. The licensee also updated the assumed containment leakage rates versus time, based upon the new containment analysis pressure response timing calculated for EPU conditions. The suppression pool pH response was also revised, and the licensee has determined that the suppression pool coolant remains basic, which prevents re-evolution of iodine from the pool. The software model for natural deposition effects inside the drywell was revised to account for an error in the version of the RADTRAD software used in Amendment No. 148. The licensee asserts and confirms that no other analysis methods or inputs were changed from those used in Amendment No. 148. Table 1 indicates the calculated design-basis LOCA dose consequence as currently analyzed, while Table 2 indicates the newly calculated dose incorporating the proposed design-basis changes.

The NRC staff evaluated the licensee's revisions to the DBA analysis of radiological consequences of a design-basis LOCA, and concludes that the licensee has appropriately accounted for the effects of the proposed EPU on this analysis. The NRC staff further concludes that the plant site and the dose-mitigating engineered safety features (ESFs) remain acceptable with respect to the radiological consequences of a postulated LOCA, as the calculated offsite and onsite doses at the exclusion area boundary (EAB), the low-population zone (LPZ) outer boundary, the technical support center, and in the control room are within the applicable acceptance criteria.

Radiological Consequences of the Design-Basis Fuel Handling Accident

The licensee updated the design-basis FHA analysis performed for the MNGP implementation of a partial scope conversion to the AST methodology reviewed and approved as Amendment No. 145 (Reference 98), dated April 24, 2006. In addition to incorporating the proposed change to EPU power, the revised analysis implements the source term inventory approved for use in the CLTR. The licensee asserts and confirms that no other analysis methods or inputs were changed from those used in Amendment No. 145.

Table 1 provides the calculated design basis FHA dose consequence as currently analyzed, while Table 2 provides the newly calculated dose incorporating the proposed design basis changes.

The NRC staff has evaluated the licensee's revisions to the DBA analysis of radiological consequences of an FHA, and concludes that the licensee has appropriately accounted for the effects of the proposed EPU conditions on this analysis. The NRC staff further concludes that all credible plant site and the dose-mitigating engineered safety features remain acceptable with respect to the radiological consequences of a postulated FHA, as the calculated offsite and onsite doses at the EAB, the LPZ outer boundary, and in the control room are within the applicable acceptance criteria.

Radiological Consequences of Design Basis Control Rod Drop Accident

The licensee updated the design basis CRDA analysis performed for the MNGP implementation of a full-scope conversion to the AST methodology reviewed and approved as Amendment No. 148 to the MNGP license (Reference 84). In addition to incorporating the proposed change to EPU power, the revised analysis implements the source term inventory approved for use in the CLTR. The licensee revised the activity concentration available for release from the condenser to account for a reduction in condenser free volume as a result of increasing normal hotwell level for EPU conditions. The licensee asserts and confirms that no other analysis methods or inputs were changed from those used in Amendment No. 148. Table 1 provides the calculated design basis CRDA dose consequence as currently analyzed, while Table 2 provides the newly calculated dose incorporating the proposed design basis changes.

The NRC staff has evaluated the licensee's revisions to the DBA analysis of radiological consequences of a Control Rod Drop Accident, and concludes that the licensee has appropriately accounted for the effects of the proposed EPU on this analysis. The NRC staff further concludes that all credible plant site and the dose-mitigating ESFs remain acceptable with respect to the radiological consequences of a postulated CRDA, as the calculated offsite and onsite doses at the EAB, the LPZ outer boundary, and in the control room are within the applicable acceptance criteria.

Radiological Consequences of the Design-Basis MSLBA

The licensee updated the design-basis MSLBA analysis performed for the MNGP implementation of a full-scope conversion to the AST methodology reviewed and approved as Amendment No. 148 (Reference 84). The revised analysis incorporates an assumption of a cesium activity concentration in the reactor coolant that is available for release to the environment following the postulated MSLBA. Though this assumption adds conservatism to the current analysis, it is not a very significant analytical change and, as the licensee has shown in the reported dose results, will do little to increase the radiological consequence of the MSLBA. The licensee asserts and confirms that no other analysis methods or inputs were changed from those used in Amendment No. 148. Table 1 provides the calculated design-basis MSLBA dose consequence as currently analyzed, while Table 2 provides the newly calculated dose incorporating the proposed change.

The NRC staff has evaluated the licensee's revisions to the DBA analysis of radiological consequences of an MSLBA, and concludes that the licensee has appropriately accounted for the effects of the proposed EPU conditions on this analysis. The NRC staff further concludes that all credible plant site and the dose-mitigating ESFs remain acceptable with respect to the radiological consequences of a postulated MSLBA, as the calculated offsite and onsite doses at the EAB, the LPZ outer boundary, and in the control room are within the applicable acceptance criteria.

Conclusion

The NRC staff has evaluated the licensee's revised accident analyses performed in support of the proposed EPU and concludes that the licensee has appropriately accounted for the effects of the proposed EPU. The NRC staff further concludes that the plant site and the dose-mitigating ESFs remain acceptable with respect to the radiological consequences of postulated DBAs since, as set forth above, the calculated TEDE at the EAB, at the outer boundary of the LPZ, and in the control room meet the exposure guideline values specified in 10 CFR 50.67 and the current MNGP licensing basis, as well as applicable acceptance criteria denoted in SRP 15.0.1. Therefore, the NRC staff finds the licensee's proposed EPU acceptable with respect to the radiological consequences of DBAs.

Table 1

Licensee Calculated Radiological Consequences of MNGP Design Basis Accidents at Assumed Current Licensed Thermal Power[6]

	Control Room		[7]EAB		LPZ	
Design Basis Accident	Total Dose	Acceptance Criteria	Total Dose	Acceptance Criteria	Total Dose	Acceptance Criteria
	(rem TEDE)	(rem TEDE)	(rem TEDE)	(rem TEDE)	(rem TEDE)	(rem TEDE)
LOCA	3.40E+00	5.0	1.31E+00	25	1.72E+00	25
FHA	4.29E+00	5.0	1.61E+00	6.3	3.10E-01	6.3
CRDA	1.70E+00	5.0	1.73E+00	6.3	7.90E-01	6.3
MSLBA						
Spike	3.25E+00	5.0	1.05E+00	25	2.00E-01	25
Equilibrium	3.30E-01	5.0	1.10E-01	2.5	2.00E-02	2.5

Table 2

Licensee Calculated Radiological Consequences of MNGP Design Basis Accidents at Extended Power Uprate Conditions[8]

	Control Room		[5]EAB		LPZ	
Design Basis Accident	Total Dose	Acceptance Criteria	Total Dose	Acceptance Criteria	Total Dose	Acceptance Criteria
	(rem TEDE)	(rem TEDE)	(rem TEDE)	(rem TEDE)	(rem TEDE)	(rem TEDE)
LOCA	3.80E+00	5.0	1.46E+00	25	1.99E+00	25
FHA	4.67E+00	5.0	1.74E+00	6.3	3.40E-01	6.3
CRDA	1.89E+00	5.0	2.00E+00	6.3	9.1E-01	6.3
MSLBA						
Spike	3.25E+00	5.0	1.05E+00	25	2.00E-01	25
Equilibrium	3.30E-01	5.0	1.10E-01	2.5	2.00E-02	2.5

2.10 Health Physics

2.10.1 Occupational and Public Radiation Doses

Regulatory Evaluation

The NRC staff conducted its review in this area to ascertain what overall effects the proposed EPU will have on both occupational and public radiation doses, and to determine that the licensee has taken the necessary steps to ensure that any dose increases will be maintained within applicable regulatory limits and as low as is reasonably achievable (ALARA). The NRC staff's review included an evaluation of any increases in radiation sources and how this may affect plant area dose rates, plant radiation zones, and plant area accessibility. The NRC staff evaluated how personnel doses needed to access plant vital areas following an accident are affected. The NRC staff considered the effects of the proposed EPU on N-16

[6] For the current DBA analyses, the licensee assumed a core power of 1880 MWth x 1.02 = 1918 MWth

[7] The licensee evaluated the maximum 2-hour total effective dose equivalent (TEDE) to an individual located at the exclusion area boundary (EAB) for the worst 2-hour period of the accident duration.

[8] For the revised DBA analyses, the assumed EPU power is 2004 MWth x 1.02 = 2044 MWth

levels in the plant and any effects this increase may have on radiation doses outside the plant and at the site boundary from skyshine. The NRC staff also considered the effects of the proposed EPU on plant effluent levels and any effect this increase may have on radiation doses at the site boundary.

The NRC's acceptance criteria for occupational and public radiation doses are based on 10 CFR Part 20, "Standards for Protection Against Radiation," 10 CFR 50.67, "Accident source term," 10 CFR Part 50, Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion "As Low as is Reasonably Achievable" for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents," and Appendix A, GDC-19, ""Control Room." Specific review criteria are contained in SRP Sections 12.2, 12.3, 12.4, and 12.5, NUREG-0737, item II.B.2, and other guidance provided in Matrix 10 of RS-001.

Technical Evaluation

Source Terms

The proposed EPU to operate the MNGP reactor at 2,004 megawatts-thermal is approximately a 13 percent increase above the currently licensed rated power. In general, the production of radiation and radioactive material (either fission or activation products) in the reactor core is directly dependent on the neutron flux and power level of the reactor. Therefore, as a first order approximation, a 13 percent increase in power level is expected to result in a proportional increase in the direct (i.e., from the reactor fuel) and indirect (i.e., from the reactor coolant) radiation source terms. However, due to the physical and chemical properties of the different radioactive materials that reside in the reactor coolant, and the various processes that transport these materials to locations in the plant outside the reactor, several radiation sources encountered in the balance of plant are not expected to change in direct proportion to the increased reactor power. The most significant of these are:

(1) The concentration of noble gas and other volatile fission products in the main steam line will not change. The increased production rate (13 percent) of these materials is offset by the corresponding increase in steam flow (13 percent). Although the concentration of these materials in the steam line remains constant, the increased steam flow results in a 13 percent increase in the rate these materials are introduced into the Main Condenser and Off Gas systems.

(2) For the very short-lived activities, most significantly N-16, the decreased transit (and decay time) in the main steam line, and the increased mass flow of the steam results in a larger increase of these activities in the major turbine building components. In general, the dose changes due to N-16 in the equipment above grade will be the most significant factor in skyshine offsite, although radiation scatter from other sources may be present. The equipment above grade at MNGP includes steam piping, turbines, feedwater heaters, the upper portions of moisture separators, and the transition between the turbines and condenser. The largest increase due to a reduced transit and decay time (17.1 percent) and the increased N-16 production (14.8 percent) is 34.4 percent at the outlet of the 15 feedwater heaters.

(3) The concentrations of non-volatile fission products, actinides, and corrosion and wear products in the reactor coolant are expected to increase proportionally with the power increase. However, the increased steam flow is expected to result in an increased

moisture carryover in the steam, resulting in an increased transport of these activities to the balance of the plant. The licensee has conservatively estimated that the 13 percent increase in steam flow will result in a ten-fold increase in the moisture carry over (from 0.05 percent to 0.5 percent) resulting in an overall increase of the radionuclides in the condensate system by a factor of 11.3. This dose estimate was based on the original steam dryer carryover performance. By letter dated June 30, 2012, the licensee indicated that the replacement steam dryer moisture carryover performance was such that post-shutdown radiation levels would be similar to the pre-EPU conditions with no appreciable increase. Overall, the radiation from these non-volatile radioactive materials provides only a small contribution to the dose rates around balance of plant systems during normal power operations.

Radiation Protection Design Features

Occupational and Onsite Radiation Exposures

The radiation sources in the core are expected to increase in proportion to the increase in power. This increase, however, is bounded by the existing safety margins of the plant design. Due to the design of the shielding and containment surrounding the reactor vessel, and since the reactor vessel is inaccessible to plant personnel during operation, a 13 percent increase in the radiation sources in the reactor core will have no effect on occupational worker personnel doses during power operations. Similarly, the radiation shielding provided in the balance of plant (i.e., around radioactive waste systems, main steam lines, the main turbine, etc.) is conservatively sized such that the increased source terms discussed above are not expected to significantly increase the dose rates in the normally occupied areas of the plant. The licensee has calculated that if the full 11.3 fold dose rate increase in the balance of the plant from moisture carryover was realized, the radiation zoning in four areas of the Turbine Building would be affected. The zones for three locations in the reactor feedwater and lube oil reservoir corridor and a fourth area in the feedwater pipe and cable penetration area, would be revised from the current 40 hour occupancy zones (dose less than 1.0 millirem per hour (mr/hr)) to 5 hour occupancy zones (dose less than 12 mr/hr). Actual radiation surveys of these areas during full power operations indicate a maximum general area dose rate of 0.2 mr/hr. A dose rate increase by a factor of 11.3 would result in a maximum dose rate of 2.2 mr/hr. This dose rate, and associated zoning, is acceptable since these areas of the plant do not require continuous occupancy during reactor operation.

Operating at a 13 percent higher power level will result in an increased core inventory of radioactive material that is available for release during postulated accident conditions. The plant shielding design must be sufficient to provide control room habitability, per GDC-19, and operator access to vital areas of the plant, per NUREG-0737, item II.B.2, during the accident. Currently the only vital areas (as defined in NUREG-0737, item II.B.2) requiring post-accident access are the control room (CR) and the technical support center (TSC). The licensee has calculated the expected post-accident doses for the CR and TSC, to be 3.8 rem TEDE, and 0.92 rem TEDE, respectively. These post-accident doses are well within the 5 rem TEDE criteria in GDC-19.

Public and Offsite Radiation Exposures

There are two factors associated with this proposed EPU that may impact public and offsite radiation exposures during plant operations. These are the possible increase in gaseous and liquid effluents released from the site, and the increase in direct radiation exposure from radioactive plant components and solid wastes stored onsite. As described above, this proposed EPU will result in a 13 percent increase in gaseous effluents released from the plant during operations. This increase is a minor contribution to the radiation exposure of the public. The nominal annual public dose from plant gaseous effluents for MNGP is less than 1 mrem. A 13 percent increase in this nominal dose is still well within the design criteria of 10 CFR 50, Appendix I.

The proposed EPU will also result in increased generation of liquid and solid radioactive waste. The increased condensate feed flow associated with the EPU results in faster loading of the condensate demineralizers. Similarly, the higher feed flow introduces more impurities into the reactor resulting in faster loading of the RWCU system demineralizers. Therefore, the demineralizers in both of these systems will require more frequent backwashing to maintain them. The licensee has estimated that these more frequent backwashes will increase the volume of liquid waste needing processing, by less than 18 percent. This increase is well within the processing capacity of the MNGP radwaste system and is not expected to noticeably increase the liquid effluents or solid radioactive waste released from the plant. Therefore, these increases will have a negligible impact on occupational or public radiation exposure.

To determine the potential impact of the increased N-16 production could have on the dose to an offsite member of the public, the licensee compared the environmental monitoring data (thermo-luminescence dosimetry readings from calendar years 1997 to 2007) at the site boundary with the data for locations 4 to 5 miles from the plant. The maximum difference of the average quarterly monitoring results, for this ten-year period, was 1.7 mrem per quarter. Assuming that the entire difference is due to N-16 shine from the plant indicates a maximum contribution of 6.8 mrem per year to the current offsite dose. Adjusting this value for the maximum increase in N-16 expected in the turbine building during EPU operations (i.e., increasing by 34.4 percent) gives an annual offsite dose from skyshine of less than 9.1 mrem; and adding the dose contribution from liquid and gaseous effluents (less than 1.0 mrem per year) to this skyshine dose results in a maximum total offsite dose of approximately 10 mrem per year. This annual dose is within the applicable 40 CFR 190 annual limit of 25 mrem to an actual member of the public, as referenced by 10 CFR 20.1301 (e).

Operational Radiation Protection Programs

The increased production of non-volatile fission products, actinides and corrosion and wear products in the reactor coolant may result in proportionally higher plate-out of these materials on the surfaces of, and low flow areas in, reactor systems. The corresponding increase in dose rates associated with these deposited materials will be an additional source of occupational exposure during the repair and maintenance of these systems. However, the current ALARA program practices at MNGP (i.e., work planning, source term minimization, etc.), coupled with existing radiation exposure procedural controls, will be able to compensate for the anticipated increases in dose rates associated with the proposed EPU. Therefore, the increased radiation sources resulting from this proposed EPU, as discussed above, will not adversely impact the licensee's ability to maintain occupational and public radiation doses resulting from plant operation to within the applicable limits in 10 CFR 20 and as low as is reasonably achievable.

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on radiation source terms and plant radiation levels. The NRC staff concludes that the licensee has taken the necessary steps to ensure that any increases in radiation doses will be maintained as low as is reasonably achievable. The NRC staff further concludes that the proposed EPU meets the requirements or guidance in 10 CFR Part 20, 10 CFR Part 50, Appendix I, NUREG-0737, and draft GDC-19. Therefore, the NRC staff finds the licensee's proposed EPU acceptable with respect to radiation protection and ensuring that occupational radiation exposures will be maintained as low as is reasonably achievable.

2.11 Human Performance

2.11.1 Human Factors

Regulatory Evaluation

The NRC staff reviewed the licensee's human factors evaluation to confirm that changes made to implement the proposed EPU will not adversely affect operator performance. The NRC staff reviewed changes to operator actions, human-system interfaces, procedures, and training identified by the licensee as needed for the proposed EPU. The NRC's acceptance criteria for human factors are based on GDC-19, "Control room," and the guidance in GL 82-33. Specific review criteria are contained in NUREG-0800 (Rev. 1), "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," Chapter 18.0, and RS-001, "Review Standard For Extended Power Uprates."

While MNGP is not explicitly licensed to the current General Design Criteria or the 1967 AEC-proposed GDC, the licensee has made a comparison of the current GDC to the applicable AEC-proposed General Design Criteria. The current GDC-19 is applicable to MNGP as described in USAR Sections 5.3.5, 6.7.2, 12.3.1.6, and 14.7.

Technical Evaluation

The NRC staff has developed a standard set of topics for the human factors assessment of BWR power uprates, i.e., RS-001, Section 3.2, Insert 9, Subsection 2.9. NSPM has addressed these topics in its application. The following are NSPM's description of these topics and the NRC staff's evaluation.

Emergency and Abnormal Operating Procedures

This section includes a summary of the licensee's assessment of how the proposed EPU will change the plant emergency and abnormal operating procedures, and the NRC staff's evaluation of that assessment.

NSPM identified in its application the following changes.

Abnormal Operating Procedures (AOP) changes:

- The station blackout (SBO) analysis was changed to include using the HPCI suction from the CSTs. The AOP will be revised to require the operator to align the HPCI suction to the CST from the main control room, prior to the three-hour point in the event. This action was previously performed by the operators within the EOPs and is not a new action.

- Installation of new non-safety-related 13.8 kv electrical buses and switchgear will result in changes to the electrical failure AOPs.

Emergency Operating Procedures (EOP) changes:

- The EPU will result in additional heat being added to the suppression pool during certain accident scenarios. The Heat Capacity Temperature Limit (HCTL) curve in the EOPs will be revised to reflect the increase in decay heat loading on the suppression pool.

- The existing EOP Caution will be revised to identify to operators that inadequate NPSH may exist if containment pressure lowers below 8.6 psig. The current limit is 7 psig.

- The Pressure Suppression Pressure curve in the EOPs will be revised to reflect the increase in reactor power and increase in decay heat loading.

The changes identified above will be incorporated in compliance with the MNGP 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants." The procedure changes and the associated training will be implemented prior to operation at uprated conditions. The NRC staff finds these proposed actions to be acceptable.

Operator Actions Sensitive to Power Uprate

The licensee stated that there are no new operator actions required to support the proposed EPU. As described above in *Emergency and Abnormal Operating Procedures*, there is one action that was previously performed as part of the EOP actions that are now included in the SBO analysis (align HPCI suction from the CST). This action will be implemented within the SBO procedure and has been assumed in the SBO analysis to be completed within three hours. Three hours is a reasonable time to perform this operation (open a knife switch and align three motor-operated valves at the HPCI panel in the main control room). There are no operator workarounds created as a result of EPU. The NRC staff finds the licensee's statements and proposed action to be acceptable.

Changes to Control Room Controls, Displays, and Alarms

This section includes the review of any changes that the proposed EPU will have on the operator interfaces for control room controls, displays, and alarms.

NSPM stated that changes to the MNGP control room controls and displays would not be extensive and will include:

- Reactor feedwater flow and steam flow control room indicators will be modified to increase the usable range;

- Installation of new 13.8 kv electrical buses and removal of the existing 4 kv electrical buses 11 and 12 will modify control switches, modify controls and indications, change computer displays and modify the annunciator alarms; and

- The Plant Process Computer alarm values for monitoring reactor power are being raised to reflect the EPU RTP levels.

The licensee stated that training related to the EPU modifications and resulting control board and procedure changes will be provided to the operators. The operators will also be provided station modification review packages as well as classroom and simulator training where appropriate. The NRC staff finds these actions to be acceptable.

Changes on the Safety Parameter Display System (SPDS)

This section includes the review of the changes to the SPDS resulting from the proposed EPU and how the licensee will make the operators aware of the proposed SPDS changes. NSPM stated that no significant SPDS changes are anticipated as a result of the proposed EPU. NSPM identified the following changes needed to support the EPU:

- The HCTL display to reflect the additional decay heat from the EPU;
- The Pressure Suppression Pressure display to reflect the increase in reactor power and increase in decay heat loading;
- AC electrical displays to reflect the new 13.8 kv buses; and
- Turbine exhaust pressure limit display reflects the change in turbine backpressure requirements. The licensee subsequently determined that this change was not required.

The NRC staff reviewed the proposed changes to the SPDS as described and finds the proposed changes to the SPDS acceptable.

Control Room Plant Reference Simulator and Operator Training

This section includes the review of changes to the operator training program and the plant-referenced control room simulator resulting from the proposed EPU and the implementation schedule for making the changes.

NSPM stated that it will ensure that adequate training is provided prior to EPU implementation per its normal training program. Operator training, licensed and non-licensed operator training, will be provided during the training cycle prior to the refueling outage and will focus on plant modifications, procedure changes, startup test procedures, and other aspects of the EPU including changes to parameters, set points, scales, and systems. The applicable lesson plans will be revised to reflect changes as a result of the EPU. Simulator training during this phase will also include training on power ascension to current maximum power. Prior to startup following the refueling outage, the operators will be given classroom and simulator Just-In-Time (JIT) training to cover last-minute training items and perform startup training and startup testing evolutions on the simulator. Successful completion of training is verified, as required by plant procedure, as part of the turnover of the modification to operations. NSPM also stated that the simulator is a duplicate of the main control room and, as such, is modified when modifications affecting simulator fidelity are installed in the plant. Installation of the EPU changes to the simulator will be performed in accordance with ANSI/ANS-3.5 1998, "Nuclear Power Plant Simulators for Use in Operator Training and Evaluation." The simulator changes will include hardware changes for new and modified control room instrumentation and controls, software updates for modeling changes due to EPU (i.e., 13.8 kV, reactor feed pump, condensate pump and high-pressure turbine modifications), set point changes, and re-tuning of the core physics model for cycle-specific data. The simulator process computer will be updated for EPU modifications. Operating data will be collected during EPU implementation and start-up testing. This data will be compared to simulator data as required by ANSI/ANS- 3.5 1998. Additionally, simulator acceptance testing will also be conducted to benchmark the simulator performance based on design and engineering analysis data. Lessons learned from power ascension testing and operation at EPU conditions will be fed back into the training process to update the training material and processes as required.

The NRC staff concludes that NSPM's proposed changes to the operator training program, including simulator training, are acceptable for the proposed EPU. The NRC staff also finds that these changes are being made in accordance with 10 CFR Parts 55.59 and 50.120.

Conclusion

The NRC staff has reviewed the licensee-identified changes to operator actions, human-system interfaces, procedures, and training required for the proposed EPU and concludes that NSPM has: (1) appropriately accounted for the effects of the proposed EPU on the available time for operator actions; and (2) taken appropriate actions to ensure that operator performance is not adversely affected by the proposed EPU. The NRC staff further concludes that the licensee will continue to meet the requirements of GDC-19, 10 CFR 50.120(b)(2)(i), 10 CFR 50.120(b)(3), and 10 CFR 55.59(c) following implementation of the proposed EPU. Therefore, the NRC staff finds the licensee's proposed EPU acceptable regarding the human factors aspects of the required system changes.

2.12 Power Ascension and Testing Plan

2.12.1 Approach to EPU Power Level and Test Plan

Regulatory Evaluation

The technical bases for this application in the subject area follow the guidelines contained in the NRC-approved GE LTRs for EPU safety analysis: NEDC-33004P-A, "Constant Pressure Power Uprate" (CLTR); NEDC-32424P-A, "Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate" (ELTR1), which the NRC determined to be an acceptable methodology for requesting EPUs; and NEDC-32523P-A, "Generic Evaluations of General Electric Boiling Water Reactor Extended Power Uprate" (ELTR2).

NRC staff guidance for reviewing EPU test programs is described in NUREG-0800, SRP 14.2.1, "Generic Guidelines for EPU Testing Programs." The NRC staff review focused on NSPM adequately addressing the guidance described in the SRP.

The purpose of the EPU test program is to demonstrate that structures, systems, and components (SSCs) will perform satisfactorily in service at the proposed EPU power level. The test program also provides additional assurance that MNGP will continue to operate in accordance with design criteria at EPU conditions. The NRC staff's review included an evaluation of: (1) plans for the initial approach to the proposed maximum licensed thermal power level, including verification of adequate plant performance; (2) transient testing necessary to demonstrate that plant equipment will perform satisfactorily at the proposed increased maximum licensed thermal power level; and (3) the test program's conformance with applicable regulations.

The NRC's acceptance criteria for the proposed EPU test program are based on 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," which requires establishment of a test program to demonstrate that SSCs will perform satisfactorily in service. Additionally, specific review criteria are contained in Section III of NUREG-0800, Section 14.2.1. Other guidance is also provided in Section 2 and Insert 12 of Review Standard for Extended Power Uprates (RS-001). The NRC staff's review focused on NSPM adequately addressing the guidance described in the SRP. NSPM's proposed power ascension and test plan (PATP) follows the guidelines contained in NRC-approved GE LTRs which the NRC staff determined to be an acceptable methodology for licensees requesting EPUs.

Technical Evaluation

SRP 14.2.1, Section III.A, Comparison of Proposed EPU Test Program to the Initial Plant Test Program

SRP 14.2.1 Section III.A specifies the guidance and acceptance criteria which the licensee should use to compare the proposed EPU testing program to the original power-ascension test program performed during initial plant licensing. The scope of this comparison should include: (1) all initial power-ascension tests performed at a power level of equal to or greater than 80 percent OLTP level; and (2) initial test program tests performed at lower power levels if the EPU would invalidate the test results. The licensee shall either repeat initial power-ascension tests within the scope of this comparison or adequately justify proposed test deviations. The following specific criteria should be identified in the EPU test program:

- all power-ascension tests initially performed at a power level of equal to or greater than 80 percent of the OLTP level;

- all initial test program tests performed at power levels lower than 80 percent of the OLTP level that would be invalidated by the EPU; and,

- differences between the proposed EPU power-ascension test program and the portions of the initial test program identified by the previous criteria.

The NRC staff reviewed applicable sections of the MNGP USAR, Appendix D, "Pre-Operational and Startup Tests," Section D.5, "Startup and Power Test Program," which provided general requirements and an overview of initial startup tests performed. The NRC staff also reviewed information in Sections D.5.1 and D.5.5 which described the general requirements and startup and power ascension testing performed from initial plant startup to the full rated power of 1,670 MWt to demonstrate that the unit was capable of operating safely and satisfactorily. The NRC staff also reviewed the following information in the November 5, 2008, application:

- Enclosure 7 to NSPM letter L-MT-08-052, "Safety Analysis Report for Monticello Constant Pressure Power Uprate (non-proprietary version)," contained the power uprate safety analysis report (PUSAR) formatted in accordance with RS-001. The PUSAR is an integrated summary of the results of the safety analysis and evaluations performed specifically for the MNGP EPU and follows the guidelines contained in General Electric (GE) Licensing Topical Report NEDC-33004P-A, "Constant Pressure Power Uprate" (CLTR). The NRC staff has approved the use of this Licensing Topical Report for

Reference as a basis for an EPU amendment application with the exception of the CLTR's proposed elimination of large transient testing.

- Enclosure 8 to the November 5, 2008, application, "Planned Modifications for Monticello Extended Power Uprate," provided a list of modifications planned for EPU implementation which do not constitute regulatory commitments by NSPM. The planned modifications will be implemented in accordance with the requirements of 10 CFR 50.59.

- Enclosure 9 to the November 5, 2008, application, "Monticello Nuclear Generating Plant Extended Power Uprate Startup Test Plan," provided a discussion of the EPU testing planned and provided a comparison of initial startup and EPU testing. Section 4.3 provided a justification for not performing large transient testing. This enclosure supplements PUSAR Section 2.12.

The NRC staff also found that all transient tests described in the initial startup test program were listed in Table 1 of Enclosure 9, and that Section 4.1 provided a discussion of power ascension startup tests initially performed at 80 percent or greater of OLTP. However, the NRC staff noted that the two large transient tests, STP-11 and STP-17, were initially performed at power levels less than 80 percent OLTP are not invalidated by the EPU. Large transient test STP-11, closure of all MSIVs, was initially performed at 75 percent OLTP (1,670 MWt) and STP-17, a turbine generator load rejection test, was performed at 50 percent OLTP. These tests follow the tests described in Attachment 2 of the NRC staff's SRP 14.2.1.

The MNGP PATP does not include performing large transient tests at full EPU power as part of the application. The justification for not performing such tests was presented by MNGP in Enclosure 9 which provides an overview of the PATP covering power ascension up to the full 120 percent OLTP (2,004 MWt) condition to verify acceptable performance. Table 1 of Enclosure 9 provided a comparison of the initial startup tests and planned EPU testing; and Table 2 summarized the planned EPU power ascension testing. MNGP's justification for a test program that does not include all of the power-ascension testing that would normally be performed is further discussed in SRP 14.2.1, Section III.C, of this SE.

The PATP is primarily an initial power ascension test plan designed to assess steam dryer and selected piping system performance from the CLTP of 1,775 MWt to 2,004 MWt, the final EPU power level. The licensee also plans to perform confirmatory inspections for a period of time following initial and continued operation at EPU levels. Testing will be performed in accordance with the TSs and applicable procedures on instrumentation re-calibrated to EPU conditions. Steady-state data will be taken during power ascension and continuing at each EPU power increase increment. EPU power increases above 100 percent CLTP will be made along an established flow/control rod line in increments of equal to or less than 5 percent power. Steady-state data will be taken at points from 90 percent up to 100 percent of CLTP so that system performance parameters can be projected for EPU power before the CLTP is exceeded. Power ascension will occur over a period of time with gradual increases in power and hold periods. The licensee is also performing post-modification testing, calibration and normal surveillance, as required, to ensure that systems will operate in accordance with their design requirements.

The NRC staff concludes, through comparison of the documents referenced above, including a review of the initial startup tests and planned EPU tests described in Table 1 of Enclosure 9 and applicable sections of Volume 7, Appendix D, of the MNGP USAR, that the proposed power ascension test program conforms to the NRC's acceptance criteria of 10 CFR Part 50,

Appendix B, Criterion XI, "Test Control," including specific review criteria contained in SRP 14.2.1 and other NRC staff guidance provided in RS-001. Therefore, the proposed power ascension and test plan is acceptable.

SRP 14.2.1, Section III.B, Post-Modification Testing Requirements for Functions Important to Safety Impacted by EPU-Related Plant Modifications

Section III.B of SRP 14.2.1 specifies the guidance and acceptance criteria which the licensee should use to assess the aggregate impact of EPU plant modifications, setpoint adjustments, and parameter changes that could adversely impact the dynamic response of the plant to an anticipated operational occurrence (AOO). AOOs include those conditions of normal operation that are expected to occur one or more times during the life of the plant and include events such as loss of all offsite power, tripping of the main turbine generator set, and loss of power to all reactor coolant pumps. The EPU test program should adequately demonstrate the performance of SSCs important to safety that meets all of the following criteria: (1) the performance of the SSC is impacted by EPU-related modifications; (2) the SSC is used to mitigate an AOO described in the plant-specific design basis; and (3) involves the integrated response of multiple SSCs.

The NRC staff reviewed Enclosure 8 to the November 5, 2009, application which described the planned modifications necessary to support the EPU which will be implemented prior to restart from Refueling Outage 25 (RFO25), currently scheduled for fall 2011. The NRC staff also reviewed Section 4.2 of Enclosure 9 which described NSPM's aggregate impact analysis of the modifications necessary to support the proposed EPU. Post-modification testing associated with the proposed modifications include functional performance checks, component performance measurements, equipment calibrations and pressure drop measurements at full flow-conditions. NSPM stated that plant modifications, set-point adjustments, and parameter changes will be demonstrated by a test program established for a BWR EPU in accordance with startup test specifications as described in PUSAR Section 2.12. The startup test specifications are based upon analyses and GE BWR experience with uprated plants to establish a standard set of tests for initial power ascension for CPPU.

NSPM stated that most modifications will have been implemented for one to two full operating cycles in advance of EPU implementation and therefore, the aggregate impact of these improvements, if any, should not be a factor in power ascension to EPU. Some of the planned modifications considered by NSPM for EPU Phase I and II include the high pressure main turbine, low pressure turbine, condensate pump upgrades and flow transmitters, generator rewind, and feedwater heater replacement.

The NRC staff concludes that the PATP proposed by NSPM demonstrates that EPU-related modifications will be adequately implemented. Specifically, the NRC staff concludes that, based on a review of the listing of completed and planned modifications, including post-maintenance testing associated with these modifications, the proposed EPU test program should adequately demonstrate the performance of SSCs. The NRC staff also concludes that the proposed PATP adequately identified plant modifications necessary to support operation at the uprated power level and complies with the criteria established in Section III.B of SRP 14.2.1.

SRP 14.2.1, Section III.C, Use of Evaluation to Justify Elimination of Power-Ascension Tests

Section III.C. of SRP 14.2.1 specifies the guidance and acceptance criteria the licensee should use to provide justification for a test program that does not include all of the power-ascension testing that would normally be performed, provided that proposed exceptions are adequately justified in accordance with the criteria provided in Section III.C.2. The proposed EPU test program shall be sufficient to demonstrate that SSCs will perform satisfactorily in service. The following factors should be considered, as applicable, when justifying elimination of power-ascension tests:

- Previous operating experience,
- Introduction of new thermal-hydraulic phenomena or identified system interactions,
- Facility conformance to limitations associated with analytical analysis methods,
- Plant staff familiarization with facility operation and trial use of operating and EOPs,
- Margin reduction in safety analysis results for AOOs,
- Guidance contained in vendor topical reports, and
- Risk implications.

The NRC staff's review is intended to provide reasonable assurance that the performance of plant equipment important to safety that could be affected by integrated plant operation or transient conditions is adequately demonstrated prior to extended operation at the requested EPU power level. The NRC staff recognizes that licensees may propose a test program that does not include all of the power-ascension testing referred to in Sections III.A and III.B of SRP 14.2.1 that would normally be performed, provided that proposed exceptions are adequately justified in accordance with the criteria provided in SRP Section III.C.2. If a licensee proposes to omit certain original startup tests from the EPU testing program based on favorable operating experience, the applicability of the operating experience to the specific plant must be demonstrated. Plant design details such as configuration, modifications, and relative changes in setpoints and parameters, equipment specifications, operating power level, test specifications and methods, operating and EOPs, and adverse operating experience from previous EPUs, should be considered and addressed.

The PATP is relied upon as a quality check to: (a) confirm that analyses and any modifications and adjustments that are necessary for proposed EPUs have been properly implemented; and (b) benchmark the analyses against the actual integrated performance of the plant. This is consistent with 10 CFR Part 50, Appendix B, which states that design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate calculational methods, or by the performance of a suitable testing program; and requires that design changes be subject to design control measures commensurate with those applied to the original plant design, which includes power ascension testing.

SRP 14.2.1 specifies that the EPU test program should include steady-state and transient performance testing sufficient to demonstrate that SSCs will perform satisfactorily at the requested power level and that EPU-related modifications have been properly implemented. The SRP provides guidance to the NRC staff in assessing the adequacy of the licensee's evaluation of the aggregate impact of EPU plant modifications, setpoint adjustments, and parameter changes that could adversely impact the dynamic response of the plant to AOOs.

The NRC staff reviewed NSPM's justification for not performing certain original startup tests against the review criteria established in SRP 14.2.1. NSPM presented its justification in Enclosure 9 (specifically, Section 4 and Tables 1 and 2). The NSPM PATP does not include all the power ascension large transient testing that would typically be performed during initial startup of a new plant. NSPM provided a detailed discussion of the basis for elimination of certain large transient tests (e.g., MSIV full closure and generator load rejection) pursuant to the NRC staff review criteria established in Section III.C.2 of SRP 14.2.1. The following large transient tests were performed in 1971 during initial startup as discussed in Appendix D, Section D.5, of the MNGP USAR:

Closure of All MSIVs

This initial startup test (STP-11) required a simultaneous full closure of all MSIVs and was performed at a maximum of 75 percent OLTP. The test objectives were to functionally check the MSIVs for proper operation at selected power levels, determine isolation valves' closure times, and to determine reactor transient behavior during and following simultaneous closure of all MSIVs. As reported in Enclosure 9, all acceptance criteria for the event were satisfied; proper MSIV operation was demonstrated; and proper closure times were measured during testing at selected power levels.

Turbine Trip / Generator Load Rejection

This initial startup test (STP-17) was performed to determine reactor response and turbine overspeed following a generator trip; and also to demonstrate the proper response of the reactor and its control systems following trips of the turbine and generator. During the test, the TSVs are tripped at selected reactor power levels and simultaneous opening of the main generator output breakers. The test was performed from a maximum power level of 50 percent OLTP. NSPM stated that all acceptance criteria were satisfied.

Other Industry (BWR) Post-EPU Large Transient Experience

With respect to the review criteria established in SRP Section III.C.2, NSPM cited industry transient events that occurred at greater than original power levels at several BWR-3/4 units that are similar in design to MNGP (a BWR 3 with a Mark I containment). The NRC staff review of the licensee event reports (LERs) associated with these events identified that all systems functioned as expected. Several events at Hatch Units 1 and 2 (BWR-4s with Mark I containment) included a turbine trip and a generator load reject event subsequent to its uprate, as reported in LERs 2000-004 and 2001-002. According to the NRC staff's review of the LERs, the primary safety systems functioned as designed in response to the events. In LER 2000-004, a turbine trip and reactor scram occurred while operating at 99.7 percent of RTP (2754 MWt) and was caused by the failure of a vibration instrument located on the Number-10 bearing. The LER reviewed by the NRC staff reported that the event had no adverse impact on nuclear safety. In the discussion for LER 2001-002, Unit 1 was at 100 percent RTP of 2763

MWt (full EPU approved power level of 113 percent OLTP) at the time of the main turbine trip. In May 1999, Hatch Unit 2 experienced an unplanned event that resulted in a generator load reject from 98.3 percent of uprated power (approximately 112.7 percent OLTP). The NRC staff review of LER 1999-005 identified that all systems functioned as expected and per design given the water level and pressure transients caused by the turbine trip and reactor scram. In 1998, the NRC approved an EPU for 113 percent OLTP (2763 MWt) for both units.

Brunswick Unit 2 (a BWR-4 with a Mark I containment), licensed by the NRC to 120 percent OLTP in May 2002, experienced an unplanned generator and turbine trip on November 4, 2003, which occurred at 115.2 percent OLTP (96 percent of uprated thermal power) and resulted in reactor protection system actuation. As noted by the NRC staff in LER 2003-04, plant systems responded as designed to the transient and the event was fully bounded by the analyses in Chapter 15 of the FSAR. In another example, on January 30, 2004, the Dresden Nuclear Power Station, Unit 3 (a BWR 3 with a Mark I containment), experienced an automatic scram due to a main turbine trip from low lube oil pressure while the plant was operating at 97 percent power (approximately 113 percent OLTP), as discussed in LER 2004-002. In December 2001, the NRC approved an EPU for 117 percent OLTP (2868 MWt) for both units.

Plant-Specific Large Transient Experience

NSPM provided information supporting its basis for not performing large transient testing, including actual plant transients experienced at MNGP. As documented in Enclosure 9 of the application, on October 23, 2001, MNGP recorded an MSIV closure event (SCRAM 112) while operating at 1740 MWt (98 percent CLTP; 87 percent of EPU). NSPM stated that the data recorded during the event demonstrated that the plant responded as expected and that the power level for the transient exceeded the percentage power (75 percent OLTP) during initial startup testing in 1971. Also, since the MNGP PUSAR (Section 2.2.2.1) indicates that the evaluation for MSIV closure, identified in guidance contained in NRC-approved vendor topical report GE ELTR2 (Section 4.7), is bounding and applicable to MNGP; and since MNGP is performing a CPPU without a corresponding pressure increase, NSPM does not recommend performance of an MSIV closure test.

Additionally, on January 21, 2002, a generator load rejection event (SCRAM 113) occurred while operating at 1773 MWt (100 percent of CLTP; 88.5 percent of EPU). All rods fully inserted, all safety systems functioned as designed, and the plant response was consistent with expectations. In Enclosure 9, NSPM provided graphs of reactor pressure and level for the two events and concluded that the transients bound testing that would be performed to repeat tests conducted during initial startup testing. Based on a review of aforementioned information, the NRC staff concludes that NSPM adequately justifies its basis for not performing large transient testing.

The NRC staff also noted that since the percent increase to EPU for SCRAM 113 was less than 15 percent above any previously recorded generator load rejection transient (13 percent: 1773 MWt versus 2004 MWt at full EPU power level), no new test is required as recommended by guidance in vendor topical report GE LTR ELTR1.

Plant Transient Evaluation

Transient experience at high power and for a wide range of operating power levels at operating BWR plants have shown an acceptable correlation of the plant transient data to the predicted response. The operating history of MNGP, which includes both a recent MSIV closure event and a generator load rejection event, both initiated above the 95 percent of CLTP level, demonstrates that previous transient events from full power are within expected peak limiting values. The transient analysis performed for the MNGP CPPU demonstrated that all safety criteria are met and that this uprate did not cause any previous non-limiting events to become limiting. This issue is further discussed in Section 2.8.5 above.

Based on the similarity of plants, past transient testing, past analyses, and the evaluation of test results, the effects of the EPU RTP level can be analytically determined on a plant-specific basis. No new design functions that would necessitate modifications and no large transient testing validation were required of safety-related systems for the EPU. The instrument setpoints that were changed do not contribute to the response to large transient events. No physical modification or setpoint changes were made to the safety relief valves and no new systems or features were installed for mitigation of rapid pressurization AOOs for this EPU. Since a scram from high power level results in an unnecessary and undesirable transient cycle on the primary system, additional transient testing involving a scram from high power levels is not justifiable. Should any future large transients occur, MNGP procedures require identification of any anomalous plant response and verification that all key safety-related equipment, required to function during the event, operated as anticipated or expected. Existing plant event data recorders are capable of acquiring the necessary data to confirm the actual versus expected response. Transient mitigation capability is demonstrated by other tests required by the TS. In addition, the limiting transient analyses are included as part of the reload licensing analysis.

The generator load rejection and turbine trip events are considered potentially limiting events and are re-analyzed for each reload. The re-analysis of these events is performed with the failure of the main steam bypass system. These events, without the operability of this system, are more limiting so the generator load rejection and the turbine trip event with main steam bypass system operable are not re-analyzed. With these two events re-analyzed for each reload and the turbine bypass system (TBS) not required for these events, a generator load rejection and turbine trip test for EPU testing of the TBS is not deemed necessary.

MNGP TS LCO 3.7.7 requires that the TBS must be Operable when thermal power is equal to or greater than 25 percent of rated thermal power (RTP). The basis for the LCO is to meet the plant response criteria for the feedwater controller failure - maximum demand, described in Section 14.4.4 of the USAR which is considered a potentially limiting event and is re-analyzed for each reload. This event results in a high reactor vessel level turbine trip, a reactor feedwater pump trip, and reactor scram due to turbine stop valve (TSV) closure. Pursuant to TS Surveillance Requirement 3.7.7 for surveillance testing, the response time and automatic actuation of the TBS is tested each refueling cycle to verify proper operation consistent with this analysis. Since this event is re-analyzed for each reload, a generator load rejection and turbine trip large transient test is not necessary.

The NRC staff reviewed the licensee's basis for not performing certain original startup tests against the review criteria established in SRP 14.2.1. NSPM addressed several factors discussed in SRP Section III.C.2. These factors included a discussion of previous industry operating experience at recently uprated BWRs, plant response to actual turbine and generator trip tests for other similar BWRs, and experience gained from actual plant transients. Additionally, NSPM followed the NRC staff-approved guidance contained in GE LTRs which the NRC staff concluded meets the objectives of a suitable test program for CPPU, with exception of the recommendation to eliminate large transient testing.

The NRC staff evaluation of the licensee's justification for not performing large transient testing was found to be acceptable based on the following review criteria discussed in SRP Section III.C.2:

- Previous operating experience has demonstrated acceptable performance of SSCs under a variety of steady-state and transient conditions;

- No new thermal-hydraulic phenomena or identified system interactions are expected to be introduced at the EPU conditions. Because this EPU is a CPPU, the effects on SSCs due to changes in thermal-hydraulic phenomena are limited;

- MNGP is in conformance with the limitations associated with applicable computer codes and analytical methods;

- MNGP plant staff familiarization with facility operation and use of operating and EOPs;

- Availability of adequate margin in safety analysis results for AOOs; and

- Compliance with NRC staff-approved guidance contained in GE LTRs which the NRC staff concluded meets the objectives of a suitable test program for CPPU, with exception of the recommendation to eliminate large transient testing.

The NRC staff concludes that NSPM's power ascension and testing program provides reasonable assurance that plant SSCs that are affected by the proposed EPU will perform satisfactorily in service at the proposed power uprate level, and that the program complies with the quality assurance requirements of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control."

SRP 14.2.1, Section III.D, Evaluate the Adequacy of Proposed Transient Testing Plans

This Section specifies the guidance and acceptance criteria the licensee should use to include plans for the initial approach to the increased EPU power level and testing that should be used to verify that the reactor plant operates within the values of EPU design parameters. The test plan should assure that the test objectives, test methods, and the acceptance criteria are acceptable and consistent with the design basis for the facility. The predicted testing responses and acceptance criteria should not be developed from values or plant conditions used for conservative evaluations of postulated accidents. During testing, safety-related SSCs relied upon during operation shall be verified to be operable in accordance with existing TS and quality assurance program requirements. The following should be identified in the EPU test program:

- The method in which initial approach to the uprated EPU power level is performed in an incremental manner including steady-state power hold points to evaluate plant performance above the original full-power level;

- Appropriate testing and acceptance criteria to ensure that the plant responds within design predictions including development of predicted responses using real or expected values of items such as beginning-of-life core reactivity coefficients, flow rates, pressures, temperatures, response times of equipment, and the actual status of the plant;

- Contingency plans if the predicted plant response is not obtained; and
- A test schedule and sequence to minimize the time untested SSCs important to safety are relied upon during operation above the original licensed full-power level.

The NRC staff reviewed Enclosure 9 of the application which provided additional information about startup testing using SRP 14.2.1, and Enclosure 7, "Safety Analysis Report for Monticello Constant Pressure Power Uprate," which provided a description of the required testing necessary for the initial power ascension following implementation of the EPU. The main elements of the PATP include power ascension, monitoring and analysis, and post-EPU monitoring. The NRC staff also determined that the licensee adequately addressed post-EPU operating experience for similar designed plants which have previously received an approved EPU amendment from the NRC staff. These plants include Hatch Units 1 and 2 (13-percent EPU); Dresden, Unit 3 (17-percent EPU); and Brunswick, Unit 2 (20-percent EPU).

As stated previously, the technical bases for the EPU request follows the guidelines contained in the following staff approved GE LTRs for EPU safety analysis: NEDC-33004P-A, "Constant Pressure Power Uprate" (CLTR); NEDC-32424P-A, "Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate" (ELTR1); and NEDC-32523P-A, "Generic Evaluations of General Electric Boiling Water Reactor Extended Power Uprate" (ELTR2). PUSAR Section 2.12, "Power Ascension and Testing Plan," submitted with the licensee's application, provides additional information relative to power uprate testing and describes a standard set of tests, which supplement the normal TS testing requirements established for the initial power ascension steps of CPPU. The test schedule would be performed in an incremental manner, with appropriate hold points for evaluation, and contingency plans would be utilized if predicted plant response is not obtained.

The NRC staff found that all transient tests described in the initial startup test program were listed in Table 1of Enclosure 9. Table 1 provided a listing of these tests which were initially performed during initial plant startup, as discussed in Section 4.3 of Enclosure 9. The tests included closure of all MSIVs (STP-11) at 75 percent OLTP (USAR Appendix D, Paragraph D.5.5(d)) and a generator trip test (STP-17) performed at 50 percent OLTP (USAR Appendix D, Paragraph D.5.5(j)). These tests follow the tests described in Attachment 2 of SRP 14.2.1.

The NRC staff has reviewed the licensee's EPU PATP including its conformance with applicable regulations and the staff guidance discussed in SRP 14.2.1. The NRC staff concludes that the proposed EPU test plan will adequately assure that the test objectives, test methods, and test acceptance criteria are consistent with the design basis for the facility.

Balance-of-Plant (BOP) Systems Testing Review

The NRC staff reviewed the licensee's power ascension and testing plan as it relates to BOP systems included within the scope of the original MNGP pre-operational test program or subject to extensive modification to support operation at the EPU power level. With regard to BOP systems, the original pre-operational test program included performance tests for the feedwater system and the turbine bypass system, as well as integrated plant testing (e.g., generator load rejection and turbine trip tests). Licensees commonly modify BOP systems, especially the feedwater system, to support operation at the EPU power level.

The turbine bypass control system is designed to control reactor pressure, when the main turbine is unavailable, by discharging steam to the main condenser, as assumed in the Feedwater Controller Failure - Maximum Demand transient analyses in Chapter 14 of the MNGP USAR. The licensee did not propose to credit additional steam bypass capacity beyond what was previously assumed, and no modifications are being made to the steam bypass system for EPU operation. The nominal turbine bypass flow rate at EPU operating conditions will remain 0.97 Mlb/hr (13.3 percent of the CLTP steam flow, or 11.5 percent of the post-EPU rated steam flow). Therefore, transient testing for the purpose of demonstrating acceptable performance of the turbine bypass control system is not required.

The condensate and reactor feedwater systems provide feedwater to the reactor vessel during normal operation and following certain anticipated operational occurrences, such as a turbine trip or a main generator load rejection. The feedwater system controls the rate of feedwater flow to maintain an appropriate water level in the reactor vessel during these conditions. The feedwater pumps automatically trip on high water level to reduce the potential for main steam-line flooding, and the feedwater pumps shut down automatically on low suction pressure, motor fault, low lube oil pressure, or low suction flow (with time delay). The modifications to the condensate and reactor feedwater systems proposed by the licensee for implementation of the MNGP EPU include replacement of the condensate pump internals, replacement of the entire main feedwater pump assembly, and the replacement of the condensate pump motors and main feedwater pump motors with motors rated for EPU operation. Consequently, the scope of the modifications has the potential to affect the reliability of the feedwater system and the integrated plant response to various transients.

In Enclosure 9 to the November 5, 2008, application, the licensee described proposed EPU power ascension testing that is partially consistent with the MNGP pre-operational test program. The proposed testing includes feedwater control system response to step reactor water level set point changes in the automatic level control modes (i.e., three element and single element) and to step demand flow changes in the manual flow control mode. The licensee proposed to exclude from the EPU power ascension test program the feedwater pump trip and the main turbine trip tests, which were part of the MNGP pre-operational test program. Appendix D, "Pre-Operational and Startup Tests," of the MNGP USAR describes that turbine trip tests were conducted to determine the effects of turbine trips on the reactor and the auxiliaries of the unit. Although the purpose of the feedwater pump preoperational trip test was to evaluate the reactor response to changes in sub-cooling of water in the reactor, the test also provided information regarding the transient response of the feedwater system.

The licensee provided justification for the exclusion of the large transient tests from the EPU test program in Section 4.3 of Enclosure 9 to the November 5, 2008, application. The licensee identified testing performed at greater than 80 percent power during the startup test program and EPU modifications that could significantly affect the performance of the associated system. The licensee determined that a generator load reject that occurred at MNGP in 2002 provided representative operating experience. This load reject occurred at 100 percent of CLTP (88.5 percent of the post-EPU power level), which exceeded the 50 percent minimum power level for the load reject startup test.

The licensee described some automatic actions associated with the 2002 trip response. Following the trip, the initial decrease in water level caused a Level 2 containment isolation signal. Operators tripped one feedwater pump as directed by procedure, but the subsequent reactor vessel water level swell combined with feedwater regulating valve leakage resulted in a high vessel level trip of the remaining feedwater pump. After the water level returned to the normal control band, operators restarted the feedwater pump and directed flow through the low-flow feedwater regulating valve to control level through the remainder of the transient.

By letter dated June 12, 2009, the licensee described that EPU modifications to the condensate and feedwater systems include new feedwater regulating valves (Reference 99). These new valves will reduce feedwater leakage when closed with the reactor feedwater pumps running. This improved isolation capability will reduce the potential to challenge the high level feedwater pump and turbine trip setpoint following plant scrams. Current post-scram level control operator actions include placing the feedwater low flow valve in auto, closing the feedwater regulating valves and closing the feedwater block valves. The new feedwater regulating valves will improve vessel level control and reduce operator actions required to restart a reactor feedwater pump after a high level trip.

The licensee stated that the existing reactor high water level trip logic for the feedwater pumps will be retained. This is a single-failure-proof one-out-of-two-twice logic scheme that provides a trip signal to the feedwater pump motor breakers. It is calibrated to trip within the TSs allowable value of < 49 inches; and instrument uncertainties are within 1 inch of indicated level. The bottom of the steam lines are at 108.5 inches. Upon sensing high reactor water level, the feedwater pump motors are tripped, terminating injection, thus providing almost 60-inch of margin before the steam line will start to flood. This is considered acceptable for EPU conditions.

The licensee also provided justification for elimination of the main feedwater pump trip portion of the main feedwater system tests. The licensee explained that the reactor core isolation cooling system has the capability to recover reactor vessel water level following a total loss of feedwater event. The NRC staff found that this justification lacked a discussion regarding the effect of the EPU-related modifications on the potential for system interactions related to the feedwater system. Accordingly, the NRC staff requested additional information about the potential for system interactions involving the feedwater system.

In the letter dated June 12, 2009, the licensee described analyses and testing that will be conducted to ensure the modified feedwater and condensate systems perform in a manner that avoids unexpected system interactions. The feedwater pumps will still have the pump protection trips described in the MNGP USAR, although setpoints may be revised based on pump testing. Certified performance curves and test data will be provided for each pump, and the pumps will be tested after installation to verify performance under operating conditions. These tests will be part of the overall post-modification testing to assure that the modified feedwater and condensate systems will perform as predicted under EPU operating conditions. In the June 12, 2009, letter, the licensee included a commitment to perform an analysis to predict combined feedwater system performance for normal operation and for transients including single feedwater pump trip, feedwater control system failure and single condensate pump trip (Reference 99). Proposed acceptance criteria included maintenance of an adequate margin to preclude loss of both reactor feedwater pumps from low suction pressure or flow.

In Enclosure 1, Item 4, of letter dated February 27, 2013 (ADAMS Accession No. ML13064A433), the licensee modified the above commitment. The licensee completed evaluations of several FW system transients and determined that loss of a condensate pump or loss of an electrical bus supplying one FW pump and one condensate pump would result in a sustained flow mismatch following the expected recirculation pump runback. The flow mismatch would likely cause a loss of FW system flow. However, MNGP is equipped with motor-operated

FW pumps that allow for prompt restoration of FW system flow. Therefore, the licensee changed the commitment from maintaining margin to prevent a loss of FW pumps to revising operating procedures for transient FW system events to direct prudent actions for recovering FW system flow and place the reactor in a safe and stable condition. The NRC staff found the revised commitment acceptable to mitigate the slight increase in potential for a loss of FW system flow because the motor-driven FW pumps can be recovered promptly following a FW system transient.

Conclusion

The NRC staff has reviewed the licensee's EPU power ascension and testing program, including plans for the initial approach to the proposed maximum licensed thermal power level, transient testing necessary to demonstrate that plant equipment will perform satisfactorily at the proposed increased maximum licensed thermal power level, and the test program's conformance with applicable regulations. The review included an evaluation of the licensee's plans for the initial approach to the proposed maximum licensed thermal power level, including verification of adequate plant performance, and the test program's conformance with applicable regulations. NSPM's test program primarily includes steady-state testing with no large transient testing proposed. The NRC staff also reviewed the licensee's justification for not performing large transient testing (as discussed in Enclosure 9 of the application and the licensee's June 12, 2009, letter). Based on the above, the NRC staff concludes that the licensee's justification is acceptable based on the applicable review criteria discussed in Section III.C.2 of SRP 14.2.1.

2.13 Risk Evaluation

Regulatory Evaluation

The licensee did not request the relaxation of any deterministic requirements for the proposed EPU, and the NRC staff's approval is primarily based on the licensee meeting the current deterministic engineering requirements. Per Review Standard RS-001, Section 13, a risk evaluation is conducted to determine if "special circumstances" are created by the proposed EPU. As described in Appendix D of SRP Section 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Bases: General Guidance," dated June 2007, special circumstances are any issues that would potentially rebut the presumption of adequate protection provided by the licensee meeting the currently specified regulatory requirements. Specific review guidance is contained in Matrix 13 of Review Standards RS-001 and its attachments.

The NRC staff's review addresses the risk associated with operating at the proposed EPU conditions in terms of changes in core damage frequency (CDF) and large early release frequency (LERF) from internal events, external events, and shutdown operations. In addition, the NRC staff's review addresses the quality of the risk analyses used by the licensee to support the application for the proposed EPU. This includes a review of licensee actions to address issues or weaknesses that may have been raised in previous staff reviews of the licensee's individual plant examination (IPE), individual plant examinations of external events (IPEEE), or by industry peer reviews. The NRC staff used the guidance provided in Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," to focus the review of this non-risk-informed submittal.

Technical Evaluation

The NRC staff reviewed the risk evaluation submitted by the licensee as part of its November 5, 2008, application for MNGP, as supplemented by letters dated February 4 and May 29, 2009. The licensee provided an estimate of the increase in risk (CDF and LERF) assuming EPU conditions. The licensee used a combination of quantitative and qualitative methods to assess the risk impact of the proposed EPU. The following sections set forth the NRC staff's technical evaluation of the risk information provided by the licensee. The NRC staff's evaluation did not involve an in-depth review of the licensee's risk evaluation.

A sensitivity analysis was performed by the licensee to characterize the risk associated with containment accident pressure (CAP) credit. The licensee did not seek approval to change the current NRC-approved CAP credit (Amendment No. 139, June 2, 2004; Reference 37), and the licensee's risk values associated with CAP do not create the "special circumstances" described in Appendix D of SRP Chapter 19 for a non-risk-informed application. Therefore, the NRC staff did not pursue further the risk analysis associated with CAP credit. Deterministic analyses pertaining to CAP credit is contained in Section 2.6.5, "Containment Heat Removal.

Probabilistic Risk Assessment (PRA) Model Quality

The quality of the licensee's PRA used to support a license application needs to be commensurate with the role the PRA results play in the decision-making process. The NRC staff's approval is based on the licensee meeting the current deterministic requirements, with the risk assessment providing confirmatory insights and ensuring that the EPU creates no new vulnerabilities.

IPE / IPEEE

The licensee previously submitted the MNGP IPE, which is based on a full-scope level 2 PRA performed in fulfillment of Generic Letter 88-20, on February 27, 1992 (Reference 100). On May 26, 1994, the NRC staff issued a staff evaluation report (SER),(Reference 101), stating that the licensee did not identify any severe accident vulnerabilities associated with either core damage or containment failure. The IPE submittal identified changes to the plant, procedures, and training as part of the IPE process and the licensee stated that these changes have been incorporated into the PRA model.

The NRC staff noted that an element identified in the IPE relating to EPU assessment was addressed appropriately. Modification to the bottled nitrogen supply for the SRV solenoid valves was considered in order to preclude dependency on non-essential AC power. The current PRA model reflects this change in plant design.

Based on its review of dispositions of topics outstanding from the IPE assessment, the NRC staff concludes that all items have been addressed appropriately and, therefore, do not impact the EPU risk assessment.

On March 1, 1995, the licensee submitted to the NRC the MNGP IPEEE (Reference 102) in response to Supplement 4 of GL 88-20. On April 14, 2000, the NRC issued an SER (Reference 103), concluding that the licensee's IPEEE identifies most likely severe accidents and severe accident vulnerabilities from external events.

The following issues were identified relating to the IPEEE submittal:

1. Perform additional analyses to identify if a single service water pump would provide adequate service water, instead of the two assumed in the IPEEE.

2. Revise the PRA to account for the eliminated dependency of the SRVs on AC power that was not accounted for in the submitted IPEEE.

3. Additional consideration of seismic effects on the turbine and generator lube oil tank.

4. Consideration to bypass the load shed logic for the control rod drive pumps could provide adequate core coolant and reduce the frequency of Class 2 accidents, the dominant accident class.

In its letter dated February 4, 2009, the licensee stated that the MNGP EPU could be affected by items 1, 2, and 4 (Reference 104). The licensee indicated that it has adequately addressed items significant to EPU implementation and has incorporated these items into the PRA model. The NRC staff requested additional information regarding assumptions related to the adequacy of a single service water pump for post-EPU requirements. The licensee stated in its May 29, 2009, response that the change in service water system loads at EPU conditions is not significant with respect to supplying sufficient service water flow to the dependent loads during post-transient conditions. An additional load of 38 gpm is expected for the single service water pump due to the EPU. Based on its review, the NRC staff concludes that the licensee has adequately addressed the outstanding items.

Peer Review of the MNGP PRA

The licensee's application stated that the MNGP internal events PRA received a formal industry Nuclear Energy Institute (NEI) PRA peer review in October 1997. The peer review team used the "BWROG PSA Peer Review Certification Implementation Guidelines," Revision 3, January 1997. The licensee stated that all A (i.e., findings that are extremely important and necessary to address the technical adequacy of the PRA) and B (i.e., findings that are extremely important and necessary to address but that may be deferred until the next PRA update) priority peer review comments for all 11 elements were addressed and incorporated into the PRA model. EPU-related facts and observations from the NEI PRA peer review included a finding that stated critical safety functions such as SRV Fail to Open, SRV Fail to Re-close, Vapor Suppression, and the Decay Heat Removal functions were not explicitly considered in some of the event trees. The licensee has since updated the transient event tree to address those functions.

The licensee stated that the MNGP PRA was compared against the ASME PRA standard in 2004 by Applied Reliability Engineering (ARE), Inc. The licensee stated that all open items identified in the 2004 ARE Self Assessment of the 2003 version of the MNGP PRA model have been addressed and incorporated into the current model utilized for the EPU risk assessment, with a few exceptions. These exceptions are addressed below.

An open item related to human reliability analysis (HRA) was identified in this review. It was recommended that a sensitivity study be re-performed to identify any change to the list of key pre-initiator operator actions identified in the IPE. The EPU implementation will have no impact on pre-initiator human error probability (HEP) values. If values were modified for some pre-initiator HEPs, these values would apply unchanged to both the pre-EPU and post-EPU risk quantification, and, therefore, the NRC staff finds that this open item would have no potential impact on the risk conclusions of the EPU assessment.

Additional open items relate to verifying data used to generate some initiating event frequencies accounting for plant unavailability and Bayesian updating. The licensee recognized that the elimination of non-operational time may result in moderate increases in calculated initiating event frequencies. If plant unavailability data and Bayesian modeling was updated, it would apply equally to pre- and post-EPU quantification, and, therefore, the NRC staff finds that this open item would have no potential impact on the risk conclusions of the EPU assessment.

Conclusions Regarding the Quality of the MNGP PRA

The NRC staff's evaluation of the licensee's submitted information focused on the capability of the licensee's PRA and other risk evaluations (e.g., for external events) to analyze the risks stemming from pre- and post-EPU plant operations and conditions. The NRC staff's evaluation did not involve an in-depth review of the licensee's PRA; instead, it: (1) involved an evaluation of the information provided by the licensee in its application, as supplemented; (2) considered the review findings for the MNGP IPE and IPEEE; (3) reviewed the BWROG peer review open facts and observations and their dispositions for this application; and (4) considered the licensee's self-assessment using the NRC's guidance in RG 1.200.

Based on its evaluation, the NRC staff finds that the MNGP PRA models used to support the risk evaluation for this application have sufficient scope, level of detail, and technical adequacy to support the evaluation of the EPU.

Internal Events Risk Evaluation

The licensee assessed the risk impacts from internal events resulting from the proposed EPU by reviewing the changes in plant design and operations resulting from the proposed EPU, mapping these changes onto appropriate PRA elements, modifying affected PRA elements as needed to capture the risk impacts of the proposed EPU, and re-quantifying the MNGP PRA to determine the CDF and LERF of the post-EPU plant.

Initiating Event Frequencies

The MNGP PRA models include 29 initiating event categories, including transient initiating events, LOCA initiators, and internal flooding initiators. The initiating event frequencies were not changed for the EPU risk assessment.

Transients – The licensee stated that the evaluation of the plant conditions and procedural changes for EPU conditions do not result in any new transient initiators, nor directly impact transient initiator frequencies significantly. The licensee performed sensitivity calculations that increased the non-isolation transient initiator frequency to bound the various changes to the BOP side of MNGP.

Loss of Offsite Power (LOOP) – The licensee does not expect a change in LOOP initiating event frequency due to EPU. A grid stability analysis conducted by the licensee indicated no significant impacts on grid stability due to the MNGP power uprate.

Support System – The licensee states that no significant changes to support systems are planned in support of the EPU and no significant impact on support system initiating event frequencies due to the EPU are postulated.

Loss of Coolant (LOCA) – The licensee did not identify any impact on LOCA frequencies resulting from the EPU. However, the licensee did acknowledge that increased flow rates for the EPU can cause increased piping erosion/corrosion rates. A sensitivity calculation that conservatively doubled the LOCA initiating event frequency for large LOCA showed very small increase in risk.

Internal Flooding – The licensee evaluated the effect of the proposed EPU on the Flow Accelerated Corrosion (FAC) analysis for MNGP and determined that increased main steam (MS) and feedwater (FW) flow rates at EPU conditions do not significantly affect the potential for FAC in these systems. The licensee conducted a sensitivity study that conservatively doubled the high energy line break (HELB) frequencies for MS and FW, and the quantitative results showed very minimal ΔCDF and ΔLERF changes. The ΔCDF and ΔLERF results remained within the acceptance guidelines (Region III of Figure 3 and 4) of RG 1.174. The licensee concluded that operation under EPU condition will not adversely impact the frequency of internal flooding events.

The NRC staff finds that the licensee adequately addressed internal initiating event frequencies based on the licensee properly implementing the equipment modifications and replacements it identified in its application, as supplemented. Furthermore, based on the licensee's sensitivity calculation, any short-term risk impact from break-in failures caused by the numerous BOP equipment changes is expected to be very small. Finally, the NRC staff notes that any changes observed in the future in initiating event frequencies will be identified and tracked under MNGP's existing performance monitoring programs and processes, and will be reflected in future updates of the PRA, based on actual plant operating experience.

The NRC staff has not identified any issues associated with the licensee's evaluation of internal initiating event frequencies that would significantly alter the overall risk results or conclusions for this license amendment. Therefore, the NRC staff concludes that there are no issues with the evaluation of internal initiating event frequencies associated with the MNGP internal event PRA that would rebut the presumption of adequate protection or warrant denial of this license amendment. The expectation is that initiating event frequencies will not change as a result of the EPU.

Component Failure Rates

The licensee concluded in its submittal that the EPU would not significantly impact long term equipment reliability due to the replacement/modification of plant components. The majority of hardware changes in support of the EPU may be characterized as either replacement of components or upgrade of existing components. The licensee described no planned operational modifications as part of the EPU that involve operating equipment beyond design ratings.

The NRC staff finds that the licensee adequately addressed equipment reliability based on the licensee properly implementing the equipment modifications and replacements it identified in its application. Further, any short-term risk impact of the numerous BOP equipment changes due to break-in failures, is expected to be very small. Finally, the NRC staff notes that the licensee's component monitoring programs, including equipment modifications and/or replacement, are being relied upon to maintain the current reliability of the equipment.

The NRC staff has not identified any issues associated with the licensee's evaluation of component reliability that would significantly alter the overall results or conclusions for this proposed EPU amendment. Therefore, the NRC staff concludes that there are no issues with component reliabilities/failure rates modeled in the MNGP internal events PRA that would rebut the presumption of adequate protection or warrant denial of this license amendment, and that the expectation is that there will be no change in component reliability as a result of the EPU.

Accident Sequence Delineation and Success Criteria

The licensee evaluated the impact of the proposed EPU on PRA accident sequence delineation and success criteria. The PRA success criteria are affected by the increased boil off rate, the increased heat load to the suppression pool, and the increase in containment pressure and temperature. The response to an initiator is represented in the PRA models by a set of discrete requirements for the operation of individual systems and the performance of specific operator actions. These scenario-specific requirements define the success criteria for system operation and operator action to fulfill the critical safety functions necessary to maintain the reactor fuel in a safe condition. The licensee assessed the critical safety functions of reactivity control, RPV pressure control, containment heat removal, depressurization, and RPV inventory makeup at EPU conditions using the Modular Accident Analysis Program (MAAP) thermal hydraulic computer code. The impact on success criteria and accident sequence delineation was compared to the pre-EPU conditions as modeled in the PRA model.

The licensee noted the following Level 1 PRA success criteria impacts due to the EPU:

1. 7 of 8 SRVs are required to open for the EPU conditions for RPV initial overpressure protection during an ATWS scenario.

2. Control Rod Drive Hydraulics (CRDH) as the only early injection source using two CRDH pumps at nominal flow now requires that the RPV be depressurized (use of enhanced flow CRDH with a single CRDH pump is unchanged for the EPU).

The SRV setpoints were not changed as a result of the EPU; however, the base probability of a stuck-open SRV due to additional cycling was increased in the MNGP PRA by 13 percent by using the conservative upper bound approach of increasing SRV probability by a factor equal to the increase in reactor power. The approach assumes that the stuck open relief valve probability is linearly related to the number of SRV cycles, and that the number of SRV cycles is linearly related to the reactor power increase. Two additional, less-conservative approaches, were also considered by the licensee: one that considered the number of cycles having a non-linear relationship to reactor power increase and another that assumed the stuck open relief valve probability is dominated by the initial cycle and that subsequent cycles have a much lower failure rate.

The licensee stated that, based on EPU ATWS analysis, 7 of 8 SRVs are required for the EPU condition for RPV initial overpressure protection during an ATWS scenario. Pre-EPU conditions required 6 of 8 SRVs to meet success criteria. In its May 29, 2009 submittal the licensee described changes made to the PRA to reflect the change in SRV success criteria. The licensee indicated that changes were made to the random failure probability and to the common cause failure (CCF) probability based on the Idaho National Laboratory CCF database.

The PRA success criteria for RPV makeup remain the same for the post-EPU configuration, except for CRDH. The licensee stated that both high pressure (HP) and low pressure (LP) injection systems have more than adequate flow margin for the post-EPU configuration. In its May 29, 2009, submittal the licensee clarified the explanation regarding changes to CRDH success criteria (Reference 105). Two CRDH pumps at nominal flow are not successful at EPU. If CRDH using two pumps is the only HP injection source available, RPV level will continue to drop and the EOPs direct initiation of RPV emergency depressurization. If RPV emergency depressurization is successfully initiated, then two CRDH pumps alone will be successful to maintain adequate core cooling; if RPV emergency depressurization is not initiated, then RPV level will continue to drop unless another injection source is aligned.

The licensee stated that no EOP needs to be changed and no special or new requirement for operator action pertaining to this PRA success criterion adjustment need to be imposed for EPU conditions. The licensee stated that timing changes have been identified for the level 1 PRA and can impact HEPs for operator actions; such changes have been factored into revised HEP values for EPU conditions as described in the section on HRA.

The licensee noted a negligible impact on the level 2 PRA safety functions and results, and concluded that no change to the success criteria has been identified with regard to the level 2 containment evaluation.

The NRC staff concurs with the licensee's changes to the accident sequence delineation and success criteria made to reflect the post-EPU conditions.

Operator Actions and LOOP Recovery

Human Reliability Analysis – EPU has the general effect of reducing the time available for the operators to complete recovery actions, because of the higher decay heat level after EPU implementation. The licensee stated that no new operator action or operator workaround was created as a result of the proposed EPU.

The licensee stated that MNGP is dependent on the operating crew actions for successful accident mitigation. The success of these actions is, in turn, dependent on a number of performance-shaping factors and that the performance-shaping factor that is principally influenced by the EPU is the time available within which to detect, diagnose, and perform required actions. The higher power level results in reduced times available for some operator actions.

MAAP calculations were performed for the pre- and post-EPU configurations to determine the change in allowable operator action timing. To minimize the resources required to re-quantify all operator actions in the PRA due to the EPU, a screening process was performed to identify those operator actions that have an impact on the PRA results. The operator actions identified for explicit review were selected based on Fussell Vesely (F-V) and Risk Achievement Worth (RAW) metrics. F-V is defined as the fractional decrease in CDF when the plant feature is assumed perfectly reliable and available. RAW is defined as the increase in risk if the feature is assumed to be failed at all times. The operator actions identified for explicit review were selected based on the following criteria:

1. F-V (with respect to CDF and LERF) importance measure ≥ 5E-3
2. RAW (with respect to CDF and LERF) importance measure ≥ 2.0

3. Time critical (≤ 30 minutes available) action

The licensee evaluated the impact of the power-level increase for 45 operator actions. The licensee stated that given the significant HEPs modified for this study results in increasing the plant risk profile by about seven percent, the non-significant HEPs, if adjusted, would be expected to impact the risk profile by a fraction of a percent.

For operator actions that the licensee identified as having the potential to be significantly impacted by the EPU, a detailed HRA was performed. This analysis was based on the NUREG/CR-4772, "Accident Sequence Evaluation Program Human Reliability Analysis Procedure," and added Accident Sequence Evaluation Program time reliability correlation HEPs when the response time was short (i.e., less than 1 hour).

Knowledge of the context surrounding each of the modeled operator actions (e.g., the sequences that are addressed and the additional equipment failures that have occurred) is important to ensure that the correct HEPs have been assigned. The NRC staff agrees with the licensee's conclusion that the main impact of the proposed EPU on the post-initiator operator actions is the reduction in time available for the plant operators to detect, diagnose, and perform required actions.

The licensee's use of thermal hydraulic analyses and knowledge of equipment capacities to determine the change in the time available for diagnosis and decision-making for the post-initiator operator actions is consistent with good PRA practices. The NRC staff observes that the apparent small changes in available times, and the corresponding changes in the post-initiator HEP values, should not be taken literally since the parameters and models used to obtain them are uncertain. However, the NRC staff believes that the licensee's analysis is adequate to conclude that the change in post-initiator HEP values due to the EPU is small.

In its May 29, 2009, submittal the licensee stated that EOP and severe accident management guidelines (SAMG) impacts due to EPU are minimal (Reference 105). All EOP and SAMG impacts have been identified, and the changes are limited to figures. There are no changes to EOP or SAMG actions due to EPU, and the PRA results are only minimally impacted. The EOP changes will be completed on a schedule that supports completion of all required training prior to EPU implementation.

Based on the licensee's submitted information, the NRC staff finds that it is reasonable to expect that the main impact of the EPU is to reduce the time available for some operator actions, which will increase the associated HEPs. However, these increased HEPs are not expected to create significant impacts, unless a number of critical operator actions cannot be performed at the increased power levels. The NRC staff has not identified any issues associated with the licensee's evaluation of operator actions that would significantly alter the overall results or conclusions for this proposed amendment. Therefore, the NRC staff concludes that there are no issues with the operator actions evaluation associated with the MNGP internal events PRA that would rebut the presumption of adequate protection or warrant denial of this license amendment.

Internal Events Risk Results

Table 1: Internal Events CDF and LERF Risk Metrics

	Pre-EPU	Post-EPU	Delta Change	Percent Increase
CDF	$7.3x10^{-6}$/year	$7.9x10^{-6}$/year	$5.7x10^{-7}$	7.8
LERF	$3.6x10^{-7}$/year	$3.9x10^{-7}$/year	$3.0x10^{-8}$	8.2

The increases in internal events CDF and LERF shown above are within the RG 1.174 acceptance guidelines for being "very small", and therefore do not raise concerns of adequate protection.

Level 2 PRA calculates the containment response under postulated severe accident conditions and provides an assessment of the containment adequacy. The licensee states and the staff concurs that the EPU change in power represents a relatively small change to the overall challenge to containment under severe accident conditions.

The NRC staff finds the licensee's evaluation of the impact of the proposed EPU on at-power risk from internal events is reasonable and concludes that the base risk due to the proposed EPU is acceptable and that there are no issues that rebut the presumption of adequate protection provided by the licensee meeting the currently specified regulatory requirements.

External Events Risk Evaluation

The licensee does not have fire or seismic PRA models. The IPEEE studies used the Electric Power Research Institute (EPRI) Fire Induced Vulnerability Evaluation methodology and EPRI Seismic Margins methodology to address external risk from these sources. High winds, external flooding, and other external events (e.g., transportation and nearby facility accidents) were addressed by reviewing the plant environs against regulatory requirements. The licensee provided a qualitative assessment of the impact of EPU implementation on external event risk, which is discussed below.

Internal Fire Risk

For the IPEEE fire analysis, the licensee performed a fire PRA by implementing a fire-induced vulnerability evaluation methodology. The NRC staff evaluation notes that the licensee analyzed all fire areas and compartments using a reasonable screening methodology.

The licensee stated the CDF risk increase due to fires is estimated to be consistent with the conclusion of the re-rate assessment conducted in the late 1990s; approximately one third of the internal events increase, or 2 to 3 percent (one third of 7.8 percent). The re-rate study (letter from the licensee dated December 4, 1997 (Reference 106) was conducted in the late 1990s and assessed the risk impact of a 12 percent power increase from 1670 MWth to 1880 MWth. The actual license submittal requested a 6.3 percent power increase to 1775 MWth, which is the current operating license limit for MNGP. The level 1 internal event assessment applied to the re-rate study estimated that CDF increased 17.5 percent from a baseline value of about 1.4E-05/year to about 1.6E-05/year. A further assessment of internal fires based on a conservative fire analysis model estimated that fire risk increased 5.5 percent from a baseline value of about 8.3E-06/year to 8.8E-06/year. In each assessment, re-rate and EPU, the major contribution to change in CDF is dominated by a reduction in available reaction time and the corresponding increase in HEPs for both the internal events and fire assessments. For addressing the impact of fires at EPU conditions, the licensee used the same ratio to estimate the fire CDF.

For a risk-informed application, the NRC staff would have pursued further the methodology for calculating fire risk. However, fire frequencies and fire mitigation are not related to reactor power level and there is no change in cable routing. Therefore, the NRC staff does not expect the post-EPU risk increase due to fire to exceed RG 1.174 guidelines and create the "special circumstances" described in Appendix D of SRP Chapter 19.2 for a non-risk-informed application.

Seismic Risk

MNGP seismic IPEEE used the seismic margins type approach that shows the plant has adequate safe shutdown paths assuming an operating basis earthquake peak ground acceleration of 0.06g and safe shutdown earthquake peak ground acceleration of 0.12g. The licensee's IPEEE used a review level earthquake of 0.3g and did not identify any vulnerabilities or weaknesses. The licensee stated that EPU modifications do not affect the structures or component anchoring, and that no new vulnerabilities to a seismic event are introduced by implementation of the EPU. The licensee stated that the conclusions of the seismic margins analysis would not be affected by EPU.

Other External Events Risk

The MNGP IPEEE addresses events other than seismic and fires, including high winds/tornadoes, external floods, and transportation and nearby facility accidents. Consistent with the IPEEE guidance, the licensee reviewed the plant environs against regulatory requirements regarding these hazards and concluded that MNGP meets the applicable NRC SRP guidance and, therefore, has an acceptably low risk with respect to these hazards.

External Events Risk Conclusion

The NRC staff has not identified any issues associated with the licensee's evaluation of the risks related to external events that would significantly alter the overall results or conclusions for this proposed amendment. Therefore, the NRC staff concludes that there are no issues with the external events risk evaluation that would rebut the presumption of adequate protection or warrant denial of this license amendment. The expectation is that the risk impact from external events resulting from the proposed EPU will be very small, based on the licensee's current risk evaluations.

Shutdown Risk Evaluation

The primary impact of the EPU on risk during shutdown operations is associated with the decrease in allowable operator action times in response to events. The licensee stated that the reductions are on the order of 10 to 15 percent. However, the licensee stated that these allowable operator action times to respond to loss of heat removal scenarios during shutdown operations are many hours long, and such small changes in response time result in negligible changes in HEPs.

The aspects of shutdown risk that the licensee identified as being impacted by EPU conditions included greater decay heat generation, longer times to shutdown, longer times before alternate decay heat removal (DHR) systems can be used, shorter times to boiling, and shorter times for operator responses. All of these aspects result from the increased decay heat generation created by the EPU.

The increased power level decreases the boildown time. However, because the reactor is already shut down, the boildown times are relatively long compared to the at-power PRA. The licensee stated that, at one day into an outage with the RPV level at the flange, the time to core uncovery for EPU conditions is 10.5 hours compared to 11.8 hours pre-EPU. These changes in timing are expected to have a negligible impact on operator responses and associated HEPs.

The increased decay heat loads associated with the EPU do not affect the success criteria for the systems normally used to remove decay heat, but the licensee stated that the EPU does impact the time when low-capacity DHR systems can be considered successful alternate DHR systems.

Other success criteria are stated as being marginally impacted by the EPU. The EPU has a minor impact on shutdown RPV inventory makeup during loss of DHR scenarios in shutdown because of the low decay heat level. The heat load to the suppression pool during loss of DHR scenarios is also lower than at power because of the low decay heat level, such that the margins for the suppression pool cooling capacity are adequate for EPU conditions. The licensee stated that the impact of the EPU on the success criteria for blowdown loads, RPV overpressure margin, and SRV actuation is negligible because of the low RPV pressure and low decay heat level during shutdown.

The licensee stated that procedural controls are in place to ensure that the risk impacts of EPU on shutdown operations are not significant, and that requirements of NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management" and Section 11 of NUMARC 93-01 Rev. 3, "Assessment of Risk Resulting from Performance of Maintenance Activities," are implemented to assure risk is assessed and that structures, systems, and components that perform key safety functions are available when needed.

The NRC staff has not identified any issues associated with the licensee's evaluation of shutdown risks that would significantly alter the overall results or conclusions for this proposed amendment. Therefore, the NRC staff concludes that there are no issues with the shutdown operations risk evaluation that would rebut the presumption of adequate protection or warrant denial of this license amendment. The expectation is that the impact on shutdown risk resulting from the proposed EPU will be negligibly small, based on the licensee's current shutdown risk management process.

Conclusions

The NRC staff concludes that there are no issues with the licensee's risk evaluation for the proposed EPU that would create the "special circumstances" described in Appendix D of SRP Chapter 19. Therefore, the NRC staff finds the risk implications of the proposed EPU acceptable.

3.0 RENEWED FACILITY OPERATING LICENSE AND TECHNICAL SPECIFICATION CHANGES

The licensee proposed changes to Renewed Facility Operating License DPR-22 and its Appendix A, Technical Specifications, in order to implement EPU. The technical bases for these changes have been evaluated in detail and set forth in the sections above. Therefore, Sections 3.1 and 3.2 below only describe the proposed facility operating license (FOL) and TS changes.

3.1 Renewed Facility Operating License DPR-22

License Condition 2.C.1 will be changed to reflect an approved maximum thermal power level of 2,004 MWth. There are also two new license conditions. License Condition 2.C.14 is added to allow specified leak rate testing SRs to be considered to be performed per SR 3.0.1, upon implementation of the license amendment approving the proposed EPU, until the next scheduled performance. License Condition 2.C.15 is added to reflect the testing requirements for the MNGP replacement steam dryer during power ascension, and is detailed further in Appendix A of this SE.

License Condition 2.C.1

The licensee proposed to change the maximum power level from 1,775 MWth to 2,004 MWth.

This change reflects the proposed 13 percent increase in the thermal power level for the plant and is consistent with the licensee's supporting safety analyses. The various technical aspects of this proposed change had been evaluated and found acceptable in the above sections of this SE; therefore, the NRC staff finds this proposed change acceptable.

New Operating License Condition 2.C.14

The licensee proposed to add the following new license condition:

> Leak rate tests required by surveillance requirements (SR) 3.6.1.1.1, SR 3.6.1.2.1, SR 3.6.1.3.11, SR 3.6.1.3.12, and 3.6.1.3.13 are not required to be performed until their next scheduled performance. The next scheduled performance is due at the end of the first surveillance interval that begins on the date the SR was last performed prior to implementation of Amendment No. 177.

SR 3.6.1.1.1 specifies performance of required visual examinations and leakage rate testing except for primary containment air lock testing, in accordance with the Primary Containment Leakage Rate Testing Program.

SR 3.6.1.2.1 specifies performance of the required primary containment air lock leakage rate testing in accordance with the Primary Containment Leakage Rate Testing Program.

SR 3.6.1.3.11 specifies performance of leakage rate testing for each 18-inch primary containment purge and vent valve with resilient seals.

SR 3.6.1.3.12 requires verification of leakage rate through each main steam isolation valve is: (a) less than or equal to 100 standard cubic feet per hour (scfh) when tested at greater than or equal to 44.1 psig (Pa); or (b) less than or equal to 75.3 scfh when tested at greater than or equal to 25 psig.

SR 3.6.1.3.13 requires verification of leakage rate through each main steam pathway is: (a) less than or equal to 200 scfh when tested at greater than or equal to 44.1 psig (Pa); or (b) less than or equal to 150.6 scfh when tested at greater than or equal to 25 psig.

Proposed License Condition 2.C.14 would allow leak rate tests required by these SRs to be considered to be performed per SR 3.0.1, upon implementation of the license amendment approving the proposed EPU, until the next scheduled performance. This would preclude having to perform the affected leak rate tests before their next scheduled performance solely for the purpose of documenting compliance. The allowance provided in License Condition 2.C.14 would not supersede that aspect of SR 3.0.1 that governs cases where it is believed that, if the SR were performed, it would not be met.

The licensee states that performance of the leak rate tests merely to document compliance would unnecessarily divert resources, interfere with plant operations, potentially incur additional personnel dose, and would not improve plant safety. The licensee stated that the results of the integrated leak rate testing and local leak rate testing performed in the 2007 refueling outage indicated significant margin to acceptance limits.

The NRC staff finds that the proposed surveillance interval for performance of leak rate testing described above provides reasonable assurance of containment integrity; therefore, the NRC staff finds this proposed change acceptable.

New Operating License Condition 2.C.15

The licensee proposed to add a new license condition for monitoring, evaluating, and initiating prompt action in response to potential adverse flow effects as a result of power uprate operation on plant structures, systems, and components (including verifying the continued structural integrity of the steam dryer) for power ascension from CLTP (1775 MWt) to 113 percent of CLTP (2004 MWt). The proposed license condition and the NRC staff's evaluation are discussed in Appendix A.

Based on its detailed review, the NRC staff finds this proposed change acceptable.

3.2 <u>Technical Specifications</u>

TS 1.1 – "Definitions"

The licensee proposed to change the definition of "RATED THERMAL POWER" (RTP) from the currently licensed RTP of 1,775 MWth to 2,004 MWth. This proposed change is consistent with the proposed change to License Condition 2.C.1. The change reflects the actual value in the proposed application and is consistent with the results of the NRC staff's review contained in Section 2 above. Therefore, the NRC staff finds the proposed change acceptable based on Section 2 above.

TS 3.3.1.1, "Reactor Protection System Instrumentation," Required Action E.1

The licensee proposed to revise the value for the required action from 45% RTP to 40% RTP in order to maintain this value at an equivalent absolute value in terms of megawatt thermal, given the new EPU RTP. At the current RTP, 45% equates to 798.8 MWth; and at the EPU RTP, 40% equates to 801.6 MWth. Rescaling the percent RTP maintains the same absolute thermal power level that was evaluated and authorized for CLTP. The NRC staff finds that this approach affords an equivalent level of protection to the CLB and is acceptable.

TS Section 3.3.1.1, "RPS Instrumentation," Surveillance Requirement (SR) 3.3.1.1.6

The licensee proposed to revise the frequency from its current value of 2,000 effective full-power hours. In its August 31, 2009, letter, the licensee committed to change the frequency to 1,000 megawatt days per ton, consistent with NUREG-1433, Revision 3.0, "Standard Technical Specifications, General Electric Plants, BWR/4." Since the licensee will now perform the calibration twice as often, thus accounting for more than instrument uncertainty at the higher EPU power level, the NRC staff finds this approach conservative and, therefore, acceptable.

TS Section 3.3.1.1, "RPS Instrumentation," SR 3.3.1.1.13

The licensee proposed to revise the value of this SR from 45% RTP to 40% RTP. The reasoning is the same as that cited above for TS 3.3.1.1, "Reactor Protection System Instrumentation," Required Action E.1. The NRC staff finds this approach affords an acceptable level of protection to the CLB and is acceptable.

TS Section 3.3.1.1, "RPS Instrumentation," Table 3.3.1.1-1, Function 2.b

The licensee proposed to revise the allowable value for Simulated Thermal Power High (for two-loop operation) from 0.66W + 61.6% RTP to 0.55W + 61.5% RTP. The setpoints were determined using an approved methodology based on the change to the analytical limit. The licensee previously calculated this value for the power range monitoring system upgrade, which the NRC approved by Amendment No. 159, dated January 30, 2009 (Reference 70). The new value was calculated using the same methodology. Based on the above, the NRC staff finds that this approach is acceptable.

TS Section 3.3.1.1, "RPS Instrumentation," Table 3.3.1.1-1, Note (b)

The licensee proposed to revise the allowable value for Simulated Thermal Power High (for single-loop operation) from 0.66(W - DeltaW) + 61.6% RTP to 0.55(W - DeltaW) + 61.5% RTP. As discussed above, the setpoints were determined using an approved methodology based on the change to the analytical limit. The licensee previously calculated this value for the power range monitoring system upgrade, which the NRC approved by Amendment No. 159, dated January 30, 2009 (Reference 70). The new value was calculated using the same methodology. Based on the above, the NRC staff finds that this approach is acceptable.

TS Section 3.3.1.1, "RPS Instrumentation," Table 3.3.1.1-1, Function 8

The licensee proposed to revise the applicable modes or other specified conditions from 45% RTP to 40% RTP. The reasoning is the same as that cited above for TS 3.3.1.1, "Reactor Protection System Instrumentation," Required Action E.1. The NRC staff finds this approach affords an acceptable level of protection to the CLB and is acceptable.

TS Section 3.3.1.1, "RPS Instrumentation," Table 3.3.1.1-1, Function 9

The licensee proposed to revise the applicable modes or other specified conditions from 45% RTP to 40% RTP. The reasoning is the same as that cited above for TS 3.3.1.1, "Reactor Protection System Instrumentation," Required Action E.1. The NRC staff finds this approach affords an acceptable level of protection to the CLB and is acceptable.

TS Section 3.3.5.1, "ECCS Instrumentation," Table 3.3.5.1-1, Functions 1.e and 2.e.

In its October 30, 2012, supplement, the licensee proposed to modify the Reactor Steam Dome Pressure Permissive – Bypass Timer (Pump Permissive) allowable value from ≤ 22 minutes to ≤ 18 minutes (Reference 61). The NRC staff received the revised calculation to support the setpoint change in a letter dated January 31, 2013 (Reference 62). The staff reviewed MNGP Calculation CA-03-036 which addressed relevant calculations related to allowable value (AV), setpoints, as-found, and as-left tolerances supporting the TS change.

The NRC staff requested clarification from the licensee on April 4, 2013. Based on discussions, it was determined that a revision to Calculation CA-03-036 was required. In May 30, 2013, letter, the licensee submitted a revised version of CA-03-036 (Revision 2) (Reference 68).

The NRC staff reviewed the revised AV calculation. Based on the analysis presented in Section 6.5.1 of the CA-03-036, Revision 2, the analytical limit was defined as 1200 seconds.

The NRC staff reviewed the calculation of the new AV which used a root-sum-of-the-squares calculation to combine various uncertainties. The staff noted that the corrected value for loop accuracy under trip conditions (A_{LT}) was used for the AV calculation in Revision 2. The calculation yielded a value of 18.16 minutes, which the licensee rounded down (~10 seconds) to 18 minutes. The NRC staff finds this approach to be conservative and, therefore, acceptable.

Finally, the licensee did not add the footnote used to demonstrate instrument operability in the TS for this function. The licensee's position is consistent with the NRC staff's agreement with the industry, which was documented in a letter from the BWR Owner's Group dated February 23, 2009 (Reference 108). The NRC staff finds this proposed change to be acceptable.

Based on the above, the NRC staff finds this proposed change to be acceptable.

TS Table 3.3.6.1-1, "Primary Containment Isolation Instrumentation," Item 1.c

In its letter dated August 31, 2009, the licensee proposed to modify the main steam line high flow isolation allowable value from 142% to 123.6% (Reference 107). The licensee subsequently modified the proposed main steam line high flow isolation allowable value to 116.9% in its letter dated February 27, 2013 (Reference 109). The licensee proposed to change the allowable value based on correction of a 10 CFR Part 21 Communication SC 12-18, Revision 1, "Error in Main Steam Line High Flow Calculational Methodology," from GEH. The licensee provided revised calculations which addressed the error and established a new allowable value based on a revised differential pressure measurement at the instrument. The licensee's Calculation CA-95-075, Revision 1 (included in the February 27, 2013, letter

Reference 109), contains the relevant calculations to support the change. The licensee elected to maintain the allowable value (and associated nominal trip setpoint) at the same absolute steam flow (and differential pressure values) as currently implemented. The NRC staff finds this approach conservative and, therefore, acceptable. The licensee has not added the footnote used to demonstrate instrument operability in the TS for this function; the licensee's position is consistent with the NRC staff's agreement with the industry, which was documented in a letter from the BWR Owner's Group dated February 23, 2009 (Reference 108). The NRC staff finds this proposed change to be acceptable.

TS Section 3.5.1, "Emergency Core Cooling System (ECCS) and Reactor Core Isolation Cooling (RCIC)," Action M

This proposed change has been evaluated and found acceptable in Section 2.8.7.3.1 above.

The ECCS/LOCA analysis supporting EPU does not support inoperability of an ADS valve in combination with inoperability of other ECCS components or subsystem. The CLTP analysis assumes only two of the three ADS valves are operable and applies the single failure criterion from that point. Using this assumption, the CLTP analysis includes one inoperable ADS valve in combination with other ECCS components. The EPU analysis assumes the three ADS valves are operable and applies the single failure criterion from that point. Using this assumption, the EPU analysis does not include an ADS valve inoperable in combination with inoperability of any other ECCS component. Since it is not addressed in the EPU analysis, this TS allowance cannot be retained. Accordingly, the NRC staff finds the deletion of current Action M to be acceptable.

TS Section 3.5.1, "Emergency Core Cooling System (ECCS) and Reactor Core Isolation Cooling (RCIC)," entry conditions for Action N (renumbered to Action M)

This proposed change has been evaluated and found acceptable in Section 2.8.7.3.1 above.

The licensee proposed to modify the third entry condition for Action N (renumbered to Action M) to require placing the unit in Mode 3 with reactor steam pressure less than or equal to 150 psig when the HPCI in combination with other ECCS components less than fully operable. The licensee also proposed to add a fourth entry condition to current Action N to require placing the unit in Mode 3 with reactor steam pressure less than or equal to 150 psig when an ADS valve in combination with other ECCS components or subsystems inoperable without regard to the operability of the HPCI system.

TS Section 3.6.1.3, "Primary Containment Isolation Valves (PCIVs), SR 3.6.1.3.12

The licensee proposed to modity this SR as follows:

Verify leakage rate through each MSIV is:
(a) ≤ 100 scfh when tested at ≥~~42~~ **44.1** psig (P_a); or
(b) ≤~~77~~ **75.3** scfh when tested at ≥ 25 psig (P_a).

The containment leakage rate testing pressure (P_a) is the pressure at which containment leakage rate testing is performed as per 10 CFR Part 50 Appendix J. It is defined in 10 CFR Part 50 Appendix J as the calculated peak containment internal pressure related to the design-basis LOCA. The licensee proposed to revise P_a to 44.1 psig, as discussed in Section

2.6.1 of this SE in which the NRC staff finds the change acceptable based on P_a being determined using acceptable methods and assumptions. The proposed change to leakage rate of 75.3 scfh is a proportional change based on EPU conditions and reduced pressure testing.

Based on the aforementioned discussion and the licensee's use of acceptable calculation methods and conservative assumptions related to the design containment pressure, the NRC staff finds the changes to be acceptable.

TS Section 3.6.1.3, "Primary Containment Isolation Valves (PCIVs), SR 3.6.1.3.13

The licensee proposed to modify this SR as follows:

Verify leakage rate through main steam pathway is:

(a) ≤200 scfh when tested at ≥~~42~~ **44.1** psig (Pa); or
(b) ≤~~154~~ **150.6** scfh when tested at ≥ 25 psig (Pa).

The containment leakage rate testing pressure (P_a) is the pressure at which containment leakage rate testing is performed as per 10 CFR Part 50 Appendix J. It is defined in 10 CFR Part 50 Appendix J as the calculated peak containment internal pressure related to the design-basis LOCA. The licensee proposed to revise P_a to 44.1 psig, as discussed in Section 2.6.1 of this SE in which the NRC staff finds the change acceptable based on P_a being determined using acceptable methods and assumptions. The proposed change to leakage rate of 150.6 scfh is a proportional change based on EPU conditions and reduced pressure testing.

Based on the aforementioned discussion and the licensee's use of acceptable calculation methods and conservative assumptions related to the design containment pressure, the NRC staff finds the changes to be acceptable.

TS Section 5.5.11, "Primary Containment Leakage Rate Testing Program

The licensee proposed changes to modify TS Section 5.5.11, "Primary Containment Leakage Rate Testing Program". Specifically, the licensee proposed a change to subpart "a," which modified "exception" to "exceptions." This change is editorial in nature. Therefore, the NRC staff finds this change to be acceptable.

The licensee also proposed a change to subpart "b," which modified the P_a value from 42 psig to 44.1 psig. This proposed change is consistent with the changes to SRs 3.6.1.3.12 and 3.6.1.3.13, described above. Therefore, the NRC staff finds this change to be acceptable."

4.0 REGULATORY COMMITMENTS

The licensee submitted a list of regulatory commitments in a letter dated August 30, 2011 (Reference 69). In a letter dated February 27, 2013 (Reference 109), the licensee submitted a change to Commitment No. 7, the details of which are discussed in Enclosure 1, Item 4, of the February 27, 2013, letter. A final update to the list of regulatory commitments was provided in a letter dated November 8, 2013 (Reference 110). A list of the regulatory commitments and status are provided in the table below.

Extended Power Uprate Regulatory Commitments			
	Letter Number and Date of Submittal	Commitment Summary	Status (Closure Letter)
1	L-MT-08-052 November 5, 2008	NSPM will inspect the steam dryer during the next refueling outage to confirm no unexpected changes in crack length on the steam dryer.	Complete (L-MT-10-046)
2	L-MT-09-017 March 19, 2009	The steady-state bypass void fraction for the EPU core will be calculated using the method described by the NSPM response to NRC RAI-SNPB-7 of L-MT-09-017.	Complete (L-MT-10-046)
3	L-MT-09-043 August 12, 2009	NSPM will provide the evaluation of steam dryer structural integrity to the NRC staff prior to further increases in reactor power when increasing to power levels above CLTP.	Deleted (L-MT-10-046)
4	L-MT-09-043 August 12, 2009	NSPM will perform outage steam dryer inspections based on the guidance of BWRVIP.	Active
5	L-MT-09-044 August 21, 2009	Confirmation that Feedwater and Condensate pump and heater replacement modifications are complete and meet the code allowables will be provided to the NRC prior to implementation of the EPU license amendment request.	Complete (L-MT-13-109)
6	L-MT-09-044 August 21, 2009	Confirmation that modification of support TWH-143 is complete will be provided to the NRC prior to implementation of the EPU license amendment request.	Complete (L-MT-11-044)
7	L-MT-11-044 August 30, 2011	NSPM will perform an analysis prior to EPU implementation to predict combined Condensate and Feedwater system performance for normal operation and for transients including Single Feedwater Pump Trip, Feedwater Control System Failure, and Single Condensate Pump Trip. Acceptance criteria will include adequate margin to preclude loss of both reactor feedwater	Revised (L-MT-13-020)

		pumps from low suction pressure or flow.	
	L-MT-13-020 February 27, 2013	Prior to EPU implementation NSPM will revise operating procedures for Condensate/Feedwater (CFW) transient events, to take prudent actions to recover CFW flow, and place the reactor in a safe and stable condition.	Complete (L-MT-13-092)
8	L-MT-10-072 December 21, 2010	Prior to implementation of the EPU, the USAR will be revised to indicate that the emergency heat load of 24.7 MBTU/hr occurs approximately 192 hours after shutdown.	Complete (L-MT-13-092)
9	L-MT-11-044 August 30, 2011	NSPM commits to evaluating the changes in condensate and feed pump area heat load to confirm temperatures remain within design limits prior to EPU implementation. If necessary, modifications to the HVAC system for this area will be implemented to maintain these areas with the design limits.	Complete (L-MT-13-092)
10	L-MT-09-100 October 28, 2009	If NRR agrees to review the MELLLA+ LAR concurrent with the EPU LAR, NSPM will commit in the MELLLA+ LAR to resolve the CAP section in the same manner as the issue is resolved for the delayed EPU amendment.	Complete (L-MT-10-046)
11	L-MT-10-046 June 30, 2010	As part of the MNGP restart following installation of the replacement steam dryer, NSPM will implement the Power Ascension Test Plan found in Enclosure 1, Appendix 5, of L-MT-10-046.	Active

5.0 RECOMMENDED AREAS FOR INSPECTION

The NRC staff has conducted an extensive review of the licensee's plans and analyses related to the proposed EPU and concluded that they are acceptable. NRC Inspection Procedure 71004, "Power Uprates," provides guidance for conducting inspections associated with power uprate amendments including considerations for selecting inspection samples.

6.0 STATE CONSULTATION

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

7.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.30, 51.31, 51.32, 51.33, 51.35, and 51.119, the NRC staff published a draft Environmental Assessment (EA) and draft finding of no significant impact (FONSI) in the *Federal Register* on September 15, 2009 (74 FR 47281). The draft FONSI included a 30-day opportunity for public comment on the proposed action and on the draft finding. The NRC staff received no comments. The NRC staff published the final FONSI in the *Federal Register* on January 15, 2010 (75 FR 2565). For the reasons presented in the FONSI, the issuance of this amendment will not have a significant effect on the quality of the human environment.

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) 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

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3. Letter from Timothy J. O'Connor, Northern States Power Company, to USNRC, "Monticello Extended Power Uprate: Response to NRC Reactor Systems Branch and Nuclear Performance & Code Review Branch Request for Additional Information (RAI) dated January 16, 2009 (TAC No. MD9990)," dated March 19, 2009 (ADAMS Accession No. ML090790388).

4. Letter from Timothy J. O'Connor, Northern States Power Company, to USNRC, "Monticello Extended Power Uprate: Response to NRC Reactor Systems Branch and Nuclear Performance & Code Review Branch Request for Additional Information (RAI) dated February 23, 2009 (TAC No. MD9990)," dated April 22, 2009 (ADAMS Accession No. ML091130636).

5. Letter from Timothy J. O'Connor, Northern States Power Company, to USNRC, "Monticello Extended Power Uprate: Response to NRC Reactor Systems Branch and Nuclear Code and Performance Review Branch Request for Additional Information (RAI) dated March 23, 2009 and Nuclear Code and Performance Review Branch Request for Additional Information Dated April 27, 2009," dated July 23, 2009 (ADAMS Accession No. ML092090219).

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7. General Electric Hitachi Nuclear Energy, "Safety Analysis Report for Monticello Constant Pressure Power Uprate," Revision 3 (Non-Proprietary NEDO-33322NP/Proprietary NEDC-33322P), Wilmington, NC, dated October 2008 (ADAMS Accession Nos. ML083230112 (Publicly Available) and ML083230125 (Proprietary Version)).

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101. U.S. Nuclear Regulatory Commission Staff Evaluation Report of plant IPE in response to GL 88-20, "IPE for Severe Accident Vulnerabilities" dated May 26, 1994 (ADAMS Accession No. 9406100263).

102. MNPG letter to U.S. NRC - Submittal of Monticello Individual Plant Examination of External Events (IPEEE) Report dated March 1, 1995 (ADAMS Accession No. ML080720497).

103. Monticello, Review of Individual Plant Examination of External Events (IPEEE) Submittal dated April 14, 2000 (ADAMS Accession No. ML003703631).

104. Monticello - Response to NRC Probabilistic Risk Assessment (PRA) Branch Requests for Additional Information (RAIs) dated February 4, 2009 (ADAMS Accession No. ML090360545).

105. Monticello Extended Power Uprate: Response to NRC Probabilistic Risk Assessment (PRA) Licensing Branch Request for Additional Information (RAI) dated May 29, 2009 (ADAMS Accession No. ML091520133).

106. MNPG letter to U.S. NRC "Rev 1 to application for amend to license DPR-22, increasing plant max power level from 1670 MWt to 1775 MWt" dated December 4, 1997 (ADAMS Accession No. 9712100342).

107. Monticello, Supplement to Extended Power Uprate License Amendment Request: Revision to Proposed Technical Specification Changes in Response to Staff Comments dated August 31, 2009 (ADAMS Accession No. ML092440171).

108. BWR Owner's Group letter to U.S. NRC Industry Plan to Resolve TSTF-493, "Clarify Application of Setpoint Methodology for LSSS Functions" dated February 23, 2009 (ADAMS Accession No. ML090540849).

109. Monticello Extended Power Uprate: Second Supplement for Gap Analysis, dated February 27, 2013 (ADAMS Accession No. ML130640494).

110. Monticello Extended Power Uprate (EPU): Completion of EPU Commitment for Piping Meeting Code Requirements, dated November 8, 2013 (ADAMS Accession No. ML13316B025).

111. U.S. Nuclear Regulatory Commission NUREG-1865, dated October 31, 2006, "Safety Evaluation Report Related to the License Renewal of the Monticello Nuclear Generating Plant" (ADAMS Accession No. ML063050414).

Attachment: List of Acronyms

Principal Contributors:

Chakrapani Basavaraju	Timothy Mossman
Andrew Bielen	Carol Nove
Aleem Boatright	Aloysius Obodoako
Briana Grange	Benjamin Parks
Tai Huang	Jigar Patel
Yun-Seng Huang	Nitin Patel
Naeem Iqbal	Roger Pedersen
William Jessup	Robert Pettis
Steve Jones	Ahsan Sallman
Joshua Kaizer	Swagata Som
George Lapinsky	Christopher Sydnor
Richard Lobel	Alexander Tsigoritis
Matthew McConnell	Peter Yarsky

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APPENDIX A

REPLACEMENT STEAM DRYER

DETAILED TECHNICAL EVALUATION

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Steam Dryer Evaluation Summary

The original Monticello Nuclear Generating Plant (MNGP) steam dryer was a General Electric (GE)-designed boiling-water reactor (BWR)/3, parallel vane bank, square hood design, which does not have perforated plates at the inlet and outlet sides of the vane banks. In 2011, the licensee replaced its original GE dryer with a Westinghouse steam dryer that consists of three parallel vane banks of octagonal shape and a cylindrical skirt. This Westinghouse replacement steam dryer (RSD) design is colloquially referred to as a "Nordic Dryer," since it was installed in 8 units in Swedish nuclear plants outside the United States. No adverse incidents with those installations have been reported. In a Northern States Power Company – Minnesota (NSPM) response to a U.S. Nuclear Regulatory Commission (NRC) staff request for information (RAI)-MNGP EPU-EMCB-RSD-RAI-96 (Reference 20), several precautions were taken during the fabrication of the RSD to make it less susceptible to intergranular stress corrosion cracking (IGSCC).

NSPM summarized its assessment of the original MNGP steam dryer stresses at Extended Power Uprate (EPU) (References 1 and 2), and the RSD stress analysis at EPU conditions using the analytical techniques described in Section 3 of BWRVIP [BWR Vessel and Internals Project]-182, "Guidance for Demonstration of Steam Dryer Integrity for Power Uprate," dated May 2010 (Reference 16). Following a set of RAIs regarding the use of BWRVIP-182, the licensee modified its approach to use an improved version of the acoustic load estimating software, now called the Acoustic Circuit Model Enhanced (ACE), and decided to install instrumentation on the RSD. The NRC staff conducted an audit on April 7 - 8, 2011 (see Reference 17), to review the revised draft documents for the replacement steam dryer. Following NRC staff feedback from the audit, NSPM submitted its revised EPU dryer structural analysis reports to the NRC in January 2012 (Reference 5). The revised analysis includes a limited set of on-dryer fluctuating pressure and strain measurements taken at current licensed thermal power (CLTP) conditions. The RSD hood was instrumented with **[[** **]]** strain gauges, **[[** **]]** pressure transducers, and **[[** **]]** accelerometers. The RSD skirt, however, was not instrumented. The RSD has been operating at CLTP conditions since the spring of 2011. However, the dryer instrumentation is no longer functional and was removed during the 2013 refueling outage (RFO).

After NRC staff review and MNGP responses to RAIs, the following reports were issued in support of the MNGP dryer stress analysis at EPU conditions:

- WCAP-17548-P, Rev. 1 (Reference 10), "Signal Processing Performed on Monticello MSL Strain Gauge and RSD Instrumentation Data," dated March 2013.

  The report describes measurements and processing of main stream line (MSL) signals used as inputs to the dryer alternating stress estimation procedure; and RSD signals used to support end-to-end benchmarking of the procedure.

- WCAP-17251-P, Rev. 1 (Reference 10), "Monticello Replacement Steam Dryer Four-Line Acoustic Subscale Testing Report," dated March 2013.

  The report is used to establish plant conditions where the onset of flow-induced safety relief valve (SRV) standpipe resonance occurs and to estimate bump-up-factors of loads between CLTP and EPU conditions.

- WCAP-17716, Rev. 0 (Reference 10), "Benchmarking of the Acoustic Circuit Enhanced Revision 2.0 for the Monticello Steam Dryer Replacement Project," dated March 2013.

  The report describes ACE acoustic loads definition procedure, and end-to-end bias errors and uncertainties.

- WCAP-17252-P, Rev. 3 (Reference 10), "Acoustic Loads Definition for the Monticello Steam Dryer Replacement Project," dated March 2013.

  The report defines actual loads applied to the RSD.

- WCAP-17549-P, Rev. 1 (Reference 10), "Monticello Replacement Steam Dryer Structural Evaluation for High Cycle Acoustic Loads using ACE," dated March 2013.

  The report explains dryer structural finite element (FE) modeling, stress analysis procedure, and final alternating stresses and stress ratios.

- Letter L-MT-12-056, Proprietary Attachment from Westinghouse Letter LTR-A&SA-09-32, Revision 5, "Limit Curves for Monticello Power Ascension," dated June 2012 (Reference 6).

  The report provides MSL limit curves used to monitor dryer alternating stresses during power ascension to EPU. The limit curves were revised and provided as a response to RAI-110 in L-MT-13-074 (Reference 19).

A key part of all steam dryer alternating stress evaluations is assessing the effects of acoustic loads induced by flow-induced resonances at the various MSL valves. The acoustic modes in the valve standpipes are strongly excited when the mode frequencies coincide with those of flow instability modes across the standpipe openings driven by the MSL flow. There are specific flow rates which drive these acoustic modes, which are usually quite high, such as the plants at the Quad Cities Nuclear Power Station (Quad Cities, QC). The MNGP MSL flow velocities are 149 feet per sec (ft/s) at original licensed thermal power (OLTP), and 159 ft/s at CLTP, and the estimated EPU steam velocity is 179 ft/s. In comparison to other nuclear power plants that have received NRC-approved EPU license amendments, the MNGP MSL flow velocity is generally higher at EPU conditions: Susquehanna Steam Electric Station – 153 ft/s; Grand Gulf Nuclear Station – 161 ft/s; Hope Creek Generating Station (Hope Creek) – 167 ft/s; Vermont Yankee Nuclear Power Station (VY) – 168 ft/s; and Nine Mile Point Nuclear Station, Unit 2 (NMP2) – 177 ft/s. The one exception is the MSL velocity at EPU conditions for QC2 – 202 ft/s, which is higher than that for MNGP. [[
]].

The MSL flow velocities, however, do not uniquely determine whether valve resonances will be excited. The valve standpipe dimensions, as well as general MSL geometry, also affect resonance excitation. NSPM, therefore, constructed and tested a scale model of the MNGP RPV and MSLs to determine whether valve resonance would be excited at EPU conditions. The results of the testing revealed strong excitation by flow-induced valve standpipe resonances is not expected, but initiation of the resonance could occur below EPU conditions. The limit curves on dryer excitation will be monitored during power ascension to ensure such resonances will not challenge steam dryer alternating stress intensity limits.

The on-dryer instrumentation produced a mixed set of results, with pressure measurements compromised by flaws in the sensor wiring and the omission of strain gauges on the dryer skirt region. The dryer hood strain measurements, however, provided a means of benchmarking the MNGP procedures with end-to-end data at CLTP. This means that the overall dryer alternating stress procedure, which includes the following – MSL strain measurements (converted to MSL internal acoustic pressures); the ACE dryer load simulation procedure; the mapping of acoustic loads to the dryer surfaces; and the dryer structural FE analysis – is benchmarked against final dryer strain measurements. The end-to-end benchmarking provides more confidence in the procedure than the usual piecemeal approach, where each analysis component is benchmarked individually. Bias errors based on the benchmarking, as well as uncertainties, are still included in the overall stress estimates. The NRC staff concludes that since the final alternating stress intensity calculations meet a stress ratio of 1.0 for the instrumented hood region, this is acceptable given the end-to-end benchmarking of actual dryer strains.

NSPM performed end-to-end benchmarking using CLTP measurements, and then applied the resulting bias errors and uncertainties (B/Us) to the predicted stresses at EPU. This may not be conservative if an internal pressure source that is not detected by the MSL strain gauges is present, and it grows at a higher rate than the acoustic pressures in the MSLs during power ascension from CLTP to EPU. Measurements taken at other plants indicate a presence of such internal pressure sources. Also, the CLTP benchmarking does not include the onset of SRV resonance effects, which may occur at MNGP between CLTP and EPU conditions. Therefore, the end-to-end benchmarking of the MNGP dryer analysis methodology at CLTP may not be fully conservative at EPU conditions, and a dryer alternating stress ratio somewhat higher than 1.0 is required. The reported stress ratio is [[ ]], which is significantly higher than 1.0. The NRC staff considers that the licensee has satisfied a more conservative requirement and finds that dryer hoods are not expected to experience any fatigue cracking under EPU conditions and, therefore, are acceptable.

The end-to-end benchmarking, however, is applied only to the upper dryer (hood region). The lack of strain gauge measurements on the MNGP RSD skirt required the licensee to use an alternate approach to establish that the skirt alternating stresses are below allowable limits. The Quad Cities, Unit 2 (QC2) skirt pressure measurements were compared to a modified ACE Skirt Protection Model (ACE 2.0-SPM) calculation based on QC2 MSL measurements, and an acoustic model of the QC2 MSLs and RPV dome. The ACE 2.0-SPM model parameters were adjusted slightly to optimize agreement with the QC2 skirt pressures, particularly at low frequencies where the skirt stresses are expected to be highest. Also, a more accurate semi-analytic method was developed to better estimate pressures within the narrow annulus between the dryer skirt and RPV wall. Bias errors and uncertainties based on the QC2 skirt benchmark differ from those of the regular ACE 2.0. The ACE 2.0-SPM model was used to compute skirt loads for the MNGP RSD, and skirt alternating stresses were then computed. The recommendation for a minimum alternating stress ration (MASR) of [[ ]] for the skirt is because there are no on-dryer skirt strain measurements. The MASR for the acoustic stresses in the skirt is [[ ]]. This [[ ]] ratio uses all the B&Us including a [[ ]] uncertainty for the frequency response amplitude obtained from shaker test results on a spare GE steam dryer and the corresponding prediction from finite element analysis, and is likely conservative for MNGP's Westinghouse RSD, which is of a different design – 3-ring octagonal shape vane bank type. The licensee also performed skirt evaluation with a smaller uncertainty of [[ ]] in place of [[ ]] and the MASR for skirt for acoustic stresses is [[ ]], the corresponding MASR with conservatively computed vane passing frequency (VPF) effects included is [[ ]]. The skirt analysis is based on [[ ]] frequency shift and the FEA of the dryer predicted natural frequencies of the dryer within [[ ]] of the natural frequencies measured during the hammer test conducted on MNGP RSD.

Based on the above rationale, the NRC staff concludes that the MASR ratio for the skirt is acceptable, and that there is reasonable assurance that the dryer skirt will not experience any fatigue cracking during operation at EPU conditions.

Mechanical loads also drive steam dryers, mostly from the tones emanated by recirculation pumps. The tones occur at the vane passing frequency of the pumps, and drive the recirculation piping connected to the RPV. MNGP uses the on-dryer strain data at the VPF frequencies to infer radial mechanical load amplitudes at the dryer supports, driving their structural FE model and computing worst-case alternating stresses. The vane passing frequency (VPF) stresses, when included in the stress ratio calculations, reduce the upper dryer ratio from [[ ]], and the skirt ratio from [[ ]]. The NRC staff considers the analysis for VPF effects performed by NSPM is quite conservative because it was assumed that all the measured strain in on-dryer strain gauges at VPF frequency is all due to VPF effects. In reality there will be some acoustic component. The analysis also used conservatively bounding value of stress from [[ ]]. The NRC staff considers that the reactor recirculation pump (RRP) flow rate and revolutions per minute (rpm) will not change from CLTP to EPU, and consideration of the VPF contribution to that specific frequency corresponding to RRP speed at CLTP/EPU (rather than a conservative bounding value) is adequate.

The ANSYS finite element code is used to perform the stress analysis of the RSD subject to acoustic loads under EPU operating conditions. A portion of the skirt with a slot, a small geometric feature that was not modeled for the global analysis, was analyzed using submodels. The RSD fabrication mainly includes full penetration welds with fillet welds in the vane banks. The weld stresses calculated using weld factors bound the stresses calculated according to the ASME Code, Subsection NG. The stresses due to acoustic loads were [[ ]] to those due to RRP VPF loads. The resulting alternating stress intensities satisfy the requirement of minimum stress ratio of [[ ]] for the hood. The minimum stress ratio from acoustic loads is [[ ]] is marginally below [[ ]] for the skirt and with VPF effects included conservatively the minimum alternating stress ration (MASR) for the skirt is [[ ]], which is slightly below [[ ]]. However, based on the rationale provided above, the NRC staff concludes that the MASR ratio for the skirt is acceptable, and that there is reasonable assurance that the dryer skirt will not experience any fatigue cracking during operation at EPU conditions.

<u>Main Steam Line and Replacement Steam Dryer Instrumentation</u>

MNGP instrumented the MSLs with strain gauges and the upper portion of the RSD with accelerometers, strain gauges and pressure transducers. The MSL strain gauge signals are used to [[ ]]. The RSD instrumentation is used to benchmark the end-to-end dryer stress analysis methodology.

To measure acoustic pulsations within the MSLs, NSPM instrumented MNGP MSLs with strain gauges at [[ ]]. These strain gauges measure the hoop strain that is used to obtain the acoustic pressure inside the MSL. The acoustic pressures are used in the ACE tools to infer left and right traveling pulsations. The hoop strain is also influenced by bending strains within the MSLs. To minimize the bending effects on the measured strains (which are unrelated to the acoustic pressures), the strain gauge pairs are [[

]]. For each MSL location, the [[ ]].

The upper portion of the RSD (i.e., hood) was instrumented with [[ ]] pressure transducers, [[ ]] strain gauges, and [[ ]] accelerometers. While the on-dryer strain gauge data appears reasonable, and is the basis for the ACE end-to-end benchmarking, the on-dryer pressure sensor data is unreliable, and not usable to support the MNGP EPU application. In its response to staff RAI-MNGP EPU-EMCB-RSD-RAI-86 requesting analysis of the RSD pressure data, MNGP explains that the instrumentation wiring connections to many of the pressure transducers resulted in unreliable data due to moisture intrusion and weld heat during installation. Therefore, the licensee qualified the skirt using MNGP MSL strain gauge data and QC2 ACE 2.0-SPM benchmark. The NRC staff compared MSL strain gauge instrumentation at MNGP with previous EPU applications and finds this acceptable, as there is adequate redundancy in the number of strain gauges at each MSL location. The staff also finds that the dryer has adequate number of strain gauges on MNGP RSD in the upper portion of the dryer for end-to-end benchmarking. Since there are no strain gauges on the dryer skirt, the NRC staff requested the licensee demonstrate the structural adequacy of the dryer skirt under EPU conditions using the QC2 ACE 2.0-SPM benchmark. The NRC staff reviewed the QC2 ACE 2.0-SPM benchmark and found it acceptable.

<u>Measurements and Signal Processing</u>

MNGP uses measured data acquired during May-December 2011 testing to benchmark its dryer alternating stress analysis procedures. Any older MSL data submitted from 2007 and 2008 testing is no longer used since the 2011 data is more accurate and based on a more complete set of sensors. The 2007 data was corrupted by electrical noise from the drywell fan power cables, and several sensors in the 2008 measurement had failed. The 2011 data is clean, complete, and acquired with the RSD installed.

Data was acquired for [[

]]. The staff notes that the subsequent stress analyses, [[ ]], will ensure that dryer stress calculations are not artificially lowered by the filtering. The NRC staff finds the notch filtering used by MNGP [[ ]] acceptable.

Although care was taken to ensure the measured signals were as accurate as possible, background noise is inevitable in plant measurements, particularly at higher frequencies where the broad-band signals tend to be low. To avoid applying loads to the dryer which are caused by MSL background noise, [[ ]] is applied to the measured signals. Both the MSL and on-dryer sensor signals were [[ ]] described in WCAP-17716-P, Revision 0, "Signal Processing Performed on

Monticello MSL Strain Gauge and RSD Instrumentation Data," dated March 2013 (Reference 10). [[

]].

[[

]].

The NRC staff finds the [[ ]] technique acceptable for application to the MNGP RSD.

The pressure sensors installed on the RSD led to excessive and unreliable pressure measurements, particularly at low frequencies, including an unexplained strong [[ ]] peak. The peak is not observed in the upper dryer strain gauges, or in the MSL strain gauge measurements. MNGP was requested in RAIs MNGP EPU-EMCB-RSD-RAI-43 and 98 to examine their RPV acoustic model to determine if any low frequency modes might exist [[ ]]. In response to NRC staff's request (Reference 20), NSPM states that the natural frequencies of the lowest order dome acoustic modes range from [[

]]. Nonetheless, the NRC staff considers the skirt stresses are not high enough to exceed the allowable fatigue margin.

The on-dryer pressure measurements are not required for benchmarking the ACE procedure (on-dryer strain gauges are used instead, for a more reliable end-to-end approach). However, according to the response to staff's clarification question, any strain caused by acoustic energy [[ ]] has been included in the end-to-end benchmarking of ACE 2.0, as is clear in the cumulative stress plots showing strong [[ ]] content. Therefore, the [[ ]] loading is included appropriately in the MNGP RSD stress analysis, and the NRC staff finds this acceptable.

Evaluation of Standpipe Resonance

A four-line subscale model of MNGP was constructed to investigate the flow-induced resonance response of the Target Rock SRVs, and the resulting increase in dryer acoustic loads at EPU. Similar to previous EPU applications, the model was built to represent the plant from the RPV exit to the turbine inlet. The main objectives of the scale model test (SMT) program were to check the resonance response of the SRV standpipes and obtain frequency dependent scaling factor (bump-up factors, or BUFs) which can be used to scale the pressure spectra in the MSLs from CLTP to EPU plant conditions. All SRVs in the MNGP plant are Target Rock valves of similar size and standpipe length.

Based on the model scale (1/7.04) and the ratio of the speeds of sound in the actual plant and subscale test, the Target Rock SRV resonance frequency near [[ ]] corresponds to [[ ]] in the scale model tests. The Mach number at CLTP condition is [[ ]] and at EPU is [[ ]]. The model tests were performed with the Westinghouse-designed RSD and over a Mach number range [[ ]], which covers CLTP and EPU conditions.

WCAP-17251-P (Reference 4) provides pressure spectra measured in the MSLs for various Mach numbers as well as total and band-passed (centered at the SRV resonances) pressure root mean squared (RMS) amplitudes as functions of Mach number. The licensee asserts that the operating conditions between CLTP and EPU are within the acoustically benign range of the SRV standpipe resonance and, therefore, suggests that it is appropriate to use the dynamic head in the MSLs (i.e. the square of the steam velocity ratio) to scale the dryer acoustic loading measured at CLTP to EPU conditions. The NRC staff reviewed the SMT results and requested the licensee to provide additional data on the pressure RMS amplitude integrated over a narrower frequency range centered on the SRV resonance frequency. The supplied data indicated that the acoustic resonance is likely to be weakly initiated between CLTP and EPU operating conditions and therefore the dryer load scaling factor is likely to be higher than the square of velocity ratio near the resonance frequencies. For example, Table 5-4 of WCAP-17251 shows that the RMS amplitude scaling from CLTP to EPU is as high as [[

]]. Therefore, the licensee was requested to develop a bump-up factor (BUF) at the SRV resonance frequency.

The NRC staff recently reanalyzed the available data from literature to evaluate whether it is conservative or not to use an SMT-based BUF to scale plant measured dryer load from CLTP to EPU conditions. The staff concluded that the [[

]]. For MNGP, however, the SMTs indicated that the acoustic response of the SRV standpipes is limited to the initial region of the lock-in range, an SMT-based BUF factor is acceptable. The licensee was therefore requested to develop, and use in the dryer stress analyses, a BUF at the SRV resonance frequency. The licensee reexamined their subscale test data, and provided frequency-dependent BUFs for each MSL location in their response to staff EMCB-RSD-RAI-91. The lowest BUF for all locations is [[

]].

In earlier EPU applications using ACM methodology, licensee were requested to use additional B/Us at the SRV resonance frequencies because the dryer pressure at the resonance frequencies was underpredicted by the ACM method. The licensee, while benchmarking their ACE 2.0-SPM model [[ ]], also used the ACE 2.0-SPM model to compute acoustic dryer loads [[ ]]. The B/Us from the ACE 2.0-SPM benchmarking [[ ]] will be applied at the MNGP Target Rock SRV resonance onset frequencies should any MSL limit curves be violated during power ascension and additional MNGP RSD stress analyses be required. This is acceptable to the NRC staff because NSPM has adequately addressed and considered the Target Rock SRV resonance onset in its steam dryer evaluation.

Dryer Loads and Benchmarking

Two computer programs are used to calculate dryer acoustic and hydrodynamic loads: [[

]]. The ACE methodology is described in WCAP-17716-P, Rev. 0 (Reference 10). The ACE 2.0-SPM methodology is described in the response to NRC Clarification Question 2 (Reference 20). The actual MNGP RSD loads acting on the hood are described in WCAP-17252-P, Revision 3 (Reference 10).

The ACE methodology used for analyzing the RSD hood differs from the ACM used in several previous BWR EPU applications. [[

]]. The ACE approach, therefore, is based on true end-to-end benchmarking, whereas bias errors and uncertainties associated with dryer stresses based on ACM loading are summed over multiple components (a piecemeal approach). The ACE end-to-end method eliminates concerns over uncertainties regarding benchmarking of the FE model (done previously using hammer testing in non-operational conditions) and the actual dryer differential loads (the differences between exterior and interior pressures, which are very difficult to measure). Also, the ACE procedure uses a [[ ]] procedure to remove background noise from the MSL and RSD instrumentation signals, whereas the ACM approach used [[ ]] filtering and in some earlier applications, removal of signals based on [[ ]] tests.

[[

]].

The simulated pressures are not compared to those measured on the MNGP RSD outer surfaces, since ACE B/Us are based on comparisons of final calculated and measured on-dryer strains. Also, the measured pressures are unreliable due to instrumentation connection and moisture problems. MNGP does provide the measured dryer surface pressures in response to staff MNGP EPU-EMCB-RSD-RAI-111, [[

]]. The QC2 comparisons provide additional confidence in the conservatism of the ACE 2.0 methodology.

The final B/Us for the ACE 2.0 method as applied to the upper dryer in the MNGP plant are shown in Table 1.1 of WCAP-17716, and are subdivided into several frequency bands. The results of the end-to-end benchmarking are demonstrated in Figures 1-1 through 1-12, and indicate that [[

]].

Although the end-to-end benchmarking is well supported in WCAP-17716 for the upper dryer, the lack of strain gauge instrumentation on the lower dryer (skirt) required a different benchmarking approach to ensure conservatism of skirt stresses. In Section 6 of WCAP-17716, [[

]].

All benchmarking and calculations to date have been performed for MNGP power levels of 100 percent CLTP or less. To estimate dryer loading at EPU conditions, model scale testing described in WCAP 17251P and MNGP's response to staff EMCB-RSD-RAI-92 (Reference 20) is used. [[

]]. The NRC staff considers this to be acceptable because it is higher than the scaling factor based on the EPU to CLTP power ratio squared, [[ ]]. The scaling factors are conservative for broad-band frequencies, and appear to be conservative at the SRV resonance frequencies. Limit curves will be used to check MSL spectra at eight locations (upper and lower locations on each of the four MSLs) during power ascension to ensure that the SRV resonances, should they appear, will not be strong enough to compromise the integrity of the RSD.

Since the [[

]], the NRC staff finds the [[ ]] procedures acceptable for use in the MNGP EPU dryer stress assessments.

<u>IGSCC Susceptibility of the RSD</u>

Several precautions were taken during the fabrication of the RSD to make it less susceptible to IGSCC, as explained in the response to EMCB-RSD-RAI-96 (Reference 20). The RSD material [[ ]] has low susceptibility to IGSCC. [[

]]. The NRC staff finds that there were adequate design features in Westinghouse-designed steam dryers to minimize IGSCC cracking.

<u>Dryer Stress Analysis</u>

The following topics are discussed in this section:

- Stress Analysis for Acoustic Loads
- Stress Analysis of Hood (Upper Part of the Dryer)
- Stress Analysis of Skirt (Lower Part of the Dryer)
- Fatigue Assessment of Welds
- Finite Element Analysis of RSD
- Stresses Induced by Reactor Recirculation Pump Vane Passing Frequency Tones
- Steam Dryer Stress Ratios at EPU

Stress Analysis for Acoustic Loads

The stress analysis of the RSD is presented in WCAP-17549-P, Rev. 1 (Reference 10). NSPM employs a computationally efficient stress analysis approach for calculating the transient stress response of the MNGP steam dryer to pressure fluctuations in the steam dome. This approach was previously used by PSEG Nuclear, LLC, in the stress analysis of the Hope Creek steam dryer stress analysis under EPU conditions, and found to be acceptable. The traditional direct time-history analysis requires long computation times and includes the transient solution associated with inaccurate initial conditions (typically, zero displacement and velocity), while the more computationally efficient approach based on harmonic analysis conducted in the frequency domain provides the steady state solution. In addition, the harmonic analysis allows for applying specified damping (one percent of the critical modal damping) for the whole range of the natural frequencies of the steam dryer.

[[

]]. MNGP provided a detailed description of its load mapping procedure in the response to NRC staff EMCB-RSD-RAI-94 (a) (Reference 19), [[ ]]. The NRC staff concludes that the procedure appears to be rigorous and is acceptable.

[[

]].

Stress Analysis of Hood (Upper Part of the Dryer)

The stress analysis for the upper part of the dryer is based on [[

]] data collected during 2011 at CLTP power level.

Stress Analysis of Skirt (Lower Part of the Dryer)

[[

]].

The licensee used [[

]].

Fatigue Assessment of Welds

The licensee used stress concentration factors (SCFs), [[

]]. The licensee demonstrated that the stresses calculated by using the SCFs are higher compared to those using ASME Boiler and Pressure Vessel Code, Subsection NG, Table NG-3352-1. If the stresses calculated using the SCFs were higher than allowable, then the licensee used the ASME Code (Reference 21) for the fatigue assessment.

Finite Element Analysis of RSD

Consistent with previous acceptable EPU applications, MNGP computed dryer stresses [[
]]. The dryer loading time histories were expanded or contracted to decrease or increase the loading frequencies, [[

]]. The worst-case stresses are then used to determine the minimum alternating stress ratio. MNGP's FE model of the RSD is sufficiently accurate [[

]].

The NRC staff reviewed the FE stress analysis of the MNGP RSD, in addition to the B/Us and utilized weld factors, and determined that the structural analysis of the dryer was adequate.

Stresses Induced by Reactor Recirculation Pump Vane Passing Frequency Tones

Experiences with other BWR plants have shown that tonal pulsations emanated by the RRPs propagate through piping and the RPV and into the dryer via the dryer support brackets and via the MSL supports. The pulsations occur at the VPF of the pumps, and the tones are clearly visible in strain gauges mounted to steam dryers. In some cases, the alternating stresses induced in a dryer by the VPF tones have been comparable to those caused by acoustic loading on the dryer surfaces. Also, since the RRP VPFs shift in frequency as operating conditions change, their influence on dryer stresses can vary considerably as the tones may align with different dryer resonances. This shifting effect was not captured by the broad-band B/Us associated with the acoustic dryer loading calculations. Therefore, EPU applicants are now asked to account separately for the dryer stresses caused by RRP VPF tones.

Since the MNGP RSD was instrumented with strain gauges on the hoods, the measurements may be used to develop a conservative means of estimating RCP VPF induced dryer alternating stresses. In Section 2.2 of WCAP-17549, Rev. 1 (Reference 10), NSPM describes its VPF stress analysis procedure and provides clarifying comments in the response to NRC staff EMCB-RSD-RAI-115 (Reference 20). **[[**

]]. The NRC staff finds this to be acceptable, since the licensee conservatively accounted for the VPF effects.

Steam Dryer Stress Ratios at EPU

MNGP provided details on the three high stressed regions of the upper and lower portions of the dryer in its response to NRC staff EMCB-RSD-RAI-116, and subsequent NRC clarifying questions. **[[**

]].

For the Upper Dryer (Hood) portion, which was instrumented, the minimum alternating stress ratio is **[[** **]]** at the projected EPU conditions. **[[**

]]. The NRC staff determined that the upper dryer portion MASR based on end-to-end benchmark was acceptable.

For the Lower Dryer (Skirt) portion, which was not instrumented, the minimum alternating stress ratio, including SRV resonance, is **[[**

]].

Based on the above evaluation, the NRC staff determined that the MASR ratio for the skirt was acceptable, and concluded that there is reasonable assurance that the dryer skirt will not experience any fatigue cracking during operation at EPU conditions.

Limit Curves

In a letter dated July 19, 2012 (Reference 6), the licensee provided, as Enclosure 6, a Westinghouse letter LTR-A&SA-09-32, Revision 5, "Limit Curves for Monticello Power Ascension." **[[**

]].

The licensee generated the limit curves using EPU data instead of CLTP results as described above. This was acceptable **[[**

]].

The highest alternating stress, as reported in Enclosure 7 to NSPM Letter L-MT-13-029 (Reference 10), was used to compute a minimum (most conservative) alternating stress ratio **[[**

]]. The limit curves provided in response to EMCB-RSD-RAI-84 (Reference 10) were updated, and the final Level 1 and Level 2 limit curves were provided in response to EMCB-RSD-RAI-110 (Reference 19).

The development of the limit curves for use in monitoring during the power ascension phase was based on the minimum allowable alternating stress ratio of 1.0. It should be noted that the recommended minimum alternating stress ratio for the steam dryer at EPU is 2.0 during the steam dryer stress analyses phase (if the dryer is not instrumented). This is acceptable because it is extremely unlikely that the dryer stress ratio will reach close to 1.0 or even decrease below 2.0 without significantly violating the limit curves. There were two reasons for this assessment: **[[**

]]. The main purpose of the limit curves was to monitor for such unanticipated increases in the strain gauge measurements during power ascension.

If any of the Level 1 limit curves are violated, NSPM has committed to reanalyze the dryer alternating stresses. [[

]]. The MNGP limit curve approach for use during power ascension is similar to what has been used by the other licensee's previously during power ascension to monitor steam dryer structural integrity and, therefore, the NRC staff finds it acceptable.

Replacement Steam Dryer Stresses for Service Levels A, B, C and D

In addition to the evaluation for FIV loading and high cycle fatigue, the licensee has also evaluated the steam dryer for load combinations for service levels A through D to demonstrate its structural integrity. The licensee utilized subsection NG of the ASME Code Section III (Reference 21) for guidance. Plant specific load combinations were followed. [[

]].

[[

]]. The allowable stress limit for Levels A and B primary plus secondary stress range was $3S_m$. The Code Edition utilized for this analysis was the 2004 ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NG.

For membrane and membrane plus bending stresses, the upset and faulted combinations are more limiting than the normal and emergency combinations. The ratios of the allowable membrane stress intensity to the computed stress intensity for the dryer for the most limiting component are as follows:

For Upset [[

]]

For Faulted [[

]]

The ratios of the allowable membrane plus bending stress intensity to the computed membrane plus bending stress intensity for the dryer at the most limiting component for EPU conditions are as follows:

For Upset [[

]]

For Faulted [[

]]

The ratio of primary plus secondary stress intensity range to the computed primary plus secondary stress intensity range for the most limiting component was
[[ ]].

The licensee performed an evaluation to demonstrate that all of the conditions in NG-3222.4 (d)(l) - (d)(4) were fulfilled and, therefore, concluded that an explicit analysis for cyclic service or fatigue was not required.

Based on a review of the above results, the NRC staff determined that the results for steam dryer stress intensities were acceptable for the normal, upset, emergency, and faulted load combinations under EPU conditions [[

]].

Power Ascension Test Plan

The EPU Startup Test Plan was provided in Enclosure 9 of the November 5, 2008, EPU application (Reference 2), as revised in supplemental letters dated January 13, 2012, and September 30, 2013 (References 5 and 23, respectively). For implementation of EPU at MNGP, the comprehensive startup testing that NSPM will conduct is included in this plan. The startup test specifications described in the Power Uprate Safety Analysis Report (PUSAR) are based upon GE BWR experience with uprated plans to establish a standard set of tests for initial power ascension to constant pressure power uprate (CPPU). The purpose of EPU test program is to demonstrate that SSCs will perform satisfactorily in service at the proposed EPU power level. The test program also provides additional assurance that the plant will continue to operate in accordance with the design criteria at EPU conditions. The program describes plans for the initial approach to verify plant performance at EPU, needed transient testing, and the test program's conformance to 10 CFR Part 50, Appendix B, Criterion XI related to the establishment of test program to demonstrate the satisfactory performance of the SSCs in service. The Monticello Startup Test Plan describes that EPU power increases will be made in pre-determined increments of ≤ 5 percent power (the planned increment is approximately 2.5 percent).

Steam dryer/separator performance will be confirmed to be within limits by determination of steam moisture content during power ascension testing. Vibration monitoring of main steam and feedwater piping will be performed to assess the effects of the EPU. NSPM provided a MNGP Replacement Steam Dryer Power Ascension Test Plan (PATP) in Appendix 5 to Enclosure 1 of the June 30, 2010, submittal (Reference 4), and a revised MNGP RSD PATP was provided in Enclosure 5 to Reference 5. A final revision to the MNGP RSD PATP was provided as Enclosure 3 to Reference 23. The three main elements of the RSD PATP are as follows: (1) a slow and deliberate power ascension with defined hold points and durations allowing time for monitoring and analysis; (2) a detailed power ascension monitoring and analysis program to trend steam dryer performance through the monitoring of the MSL strain gauges, and moisture carryover; and (3) an inspection and analysis program to verify steam dryer and piping system performance. Relevant data and evaluations will be transmitted to the NRC staff during the power ascension.

This plan includes specific hold points and durations during power ascension; activities to be accomplished during hold points; data to be collected; required inspections and walk downs; data evaluation methods; and acceptance criteria for monitoring and trending plant parameters. This plan incorporates requirements from Regulatory Guide (RG) 1.20, Revision 3, dated March 2007.

In preparation for EPU power ascension, NSPM will prepare a Startup Test Plan to include: (a) stress limit curves to be applied for evaluating steam dryer performance; (b) specific hold points and their duration during EPU power ascension; (c) activities to be accomplished during hold points; (d) plant parameters to be monitored; (e) inspections and walkdowns to be conducted for steam, FW, and condensate systems and components during the hold points; (f) methods to be used to trend plant parameters; (g) acceptance criteria for monitoring and trending plant parameters, and conducting the walkdowns and inspections; (h) actions to be taken if acceptance criteria are not satisfied; and (i) verification of the completion of commitments and planned actions specified in its application and all supplements to the application in support of the EPU LAR pertaining to the steam dryer prior to power increase above 1775 MWt. NSPM will submit the flow-induced vibration related portions of the EPU startup test procedure to the NRC, including the methodology for updating the limit curves, prior to initial power ascension above 1775 MWt.

NSPM replaced the GE dryer with a Westinghouse dryer during the 2011 RFO. The replacement dryer was instrumented with strain gauges, accelerometers and strain gauges. The data from the on-dryer instruments was collected during the 2011 power ascension coming out of the 2011 RFO up to CLTP and was used by the NSPM in the RSD analysis. MNGP has been operating at CLTP with the RSD in place since 2011. The dryer sensors have been removed, and will not be monitored during EPU power ascension.

NSPM's implementation and Power Ascension Test Plan is carried out in 3 phases: A, B, and C, of which A and B have already been completed.

Phase A included: 1) a collection of data from 0 MWt to approximately 1420 MWt; 2) a power ascension rate equivalent to normal operational practices; and 3) data evaluated against acceptance criteria at every approximately 20 percent power step increase (355 MWt).

Phase B included: 1) a collection of data from 1420 MWt to approximately 1775 MWt; 2) a power ascension rate equivalent to normal operational practices; and 3) data evaluated against acceptance criteria at every approximately 6.6 percent power step (118 MWt) increase.

Phase C of the RSD PATP includes: 1) a collection of data from 1775 MWt to 2004 MWt; 2) a power ascension rate of 2 percent per hour above 1775 MWt; and 3) data to be evaluated against acceptance criteria at every approximately 2.5 percent power step (44 MWt) increase.

Power ascension above 1775 MWt will be achieved using the following guidelines:

- Obtain baseline observations at 1775 MWt;
- Maximum hourly power ascension rate of approximately 2 percent (35MWt);
- At each approximately 2.5 percent power ascension step (44 MWt), compare vibration data to acceptance criteria, obtain & evaluate moisture carryover data, perform plant walk downs, review data evaluation and walk down results.

The MNGP PATP will provide for power ascension monitoring and analysis to trend steam dryer performance. Under the PATP, power will be increased at a rate of no more than 2 percent CLTP per hour. Steam line strain gauge and accelerometer vibration data will be collected hourly during power ascension. At every 2.5 percent CLTP step, MSL strain gauge and accelerometer data, and moisture carryover data, will be evaluated against acceptance criteria. At every 5 percent CLTP plateau, the data will be evaluated against the acceptance criteria, plant walkdowns will be conducted, and information will be forwarded to the NRC in accordance with the MNGP License Condition 15. RSD stress for all power ascension steps above CLTP conditions will be monitored using MSL strain gauge readings. Evaluation of the strain gauge data will be by comparison against the limit curves, which will be provided to the NRC prior to power ascension. The stress and moisture carryover criteria will have two threshold action levels, where exceeding Level 1 criteria requires that power be reduced to a previous acceptable level and exceeding Level 2 criteria requires that power be held at that level with a re-evaluation of the data

Upon completion of the power ascension to EPU, NSPM will prepare a report on the performance of the steam dryer and plant systems during the EPU power ascension. The report will include evaluations or corrective actions that were required to obtain satisfactory steam dryer performance. The report will also include relevant data collected at each power step, comparisons to performance criteria (design predictions), and evaluations performed in conjunction with steam dryer structural integrity monitoring. NSPM will submit this report to the NRC.

The PATP also includes visual monitoring of piping, valves, and other related components outside the drywell, either by walk down or cameras at each test power level. If visual observation indicates significant vibration, the noted condition will be evaluated in more detail. Plant data such as reactor water level, steam flow, feed flow, steam flow distribution between the individual steam lines that may be indicative of off-normal steam dryer and/or piping system performance will also be monitored during power ascension. This data can provide an early indication of unacceptable steam dryer/system performance.

In accordance with the MNGP License Conditions during EPU power ascension, NSPM will monitor hourly the MSL strain gauge data during power ascension above 1775 MWt for increasing pressure fluctuations in the steam lines. NSPM will hold the unit at 105 percent and 110 percent of 1775 MWt to collect data from the MSL strain gauges, conduct plant inspections and walkdowns, and evaluate steam dryer performance based on these data. NSPM will provide the evaluation to the NRC staff upon completion of the evaluation, and will not increase power above each hold point until 96 hours after the NRC confirms receipt of the evaluation or until verbal approval by the NRC to increase power is provided, whichever comes first.

If any frequency peak from the MSL strain gauge data exceeds a Level 1 limit curve, NSPM will return the facility to a lower power level at which the limit curve is not exceeded. NSPM will resolve the uncertainties in the steam dryer analysis; evaluate the continued structural integrity of the steam dryer ensuring that the minimum alternating stress ratio is greater than 1.0 for the upper dryer portion and 2.0 for the skirt, and provide that evaluation to the NRC staff. NSPM will obtain NRC approval of that evaluation prior to further increases in reactor power. In the event that acoustic signals are identified that challenge the limit curves during power ascension, NSPM will evaluate dryer loads and re-establish the limit curves based on the new strain gauge data, and perform a frequency-specific assessment of ACE uncertainty at the acoustic signal frequency including application of appropriate bias error and 10 percent uncertainty to all the SRV acoustic resonances.

NSPM will monitor RPV water level instrumentation and MSL piping accelerometers on an hourly basis during power ascension above 1775 MWt. If resonance frequencies are identified as increasing above nominal levels in proportion to strain gauge instrumentation data, NSPM will stop power ascension, evaluate the continued structural integrity of the steam dryer, and provide that evaluation to the NRC staff.

After reaching 105 percent, 110 percent, and 113 percent of 1775 MWt, respectively, NSPM will obtain measurements from the MSL strain gauges and establish the steam dryer flow-induced vibration load fatigue margin for the facility, update the dryer stress report, and re-establish the limit curves with the updated ACE load definition, which will be provided to the NRC staff. If an engineering evaluation is required because a Level 1 acceptance criterion is exceeded, NSPM will perform the structural analysis to address frequency uncertainties up to ± 10 percent and assure that peak responses that fall within this uncertainty band are addressed.

NSPM will submit a report with the results of the MNGP PATP following completion of the power ascension. As part of the post EPU monitoring program, NSPM will monitor plant parameters indicative of degradation of the steam dryer or plant systems during EPU operation. For example, moisture carryover will be monitored with the results reviewed and evaluated. As MSL strain gauges and accelerometers remain operable, data collection may be performed during the remainder of the operating cycle following EPU implementation. Steam dryer inspections and monitoring of plant parameters potentially indicative of steam dryer failure will be conducted for all accessible, susceptible locations with considerations of BWRVIP-139-A, "BWR Steam Dryer Integrity" (Reference 22) and Electric Power Research Institute (EPRI) Technical Report 1011463, "BWR Vessel and Internals Project, Steam Dryer Inspection and Flaw Evaluation Guidelines," in conjunction with the new dryer structural weld configurations. Monitoring of plant parameters potentially indicative of steam dryer failure is included in the PATP. The results of the visual inspections of the steam dryer will be reported to the NRC staff within 90 days following startup from the respective refueling outage.

The NRC staff reviewed the MNGP PATP for its ability to provide a slow and deliberate power ascension that allows for monitoring of plant data, evaluating steam dryer and system performance, and taking corrective action in the event that plant data reveal such action is appropriate. Further, the NRC staff compared the proposed license conditions for MNGP with those applied during the Hope Creek, Vermont Yankee, and Nine Mile Point Unit 2 power ascensions. The NRC staff finds that the MNGP PATP and the applicable license conditions provide an acceptable power ascension process that is consistent with the successful approach employed during power ascension at the aforementioned plants.

License Conditions on Potential Adverse Flow Effects

15. In conjunction with the license amendment to revise paragraph 2.C.1 of Renewed Facility Operating License No. DPR-22 to reflect the new maximum licensed reactor core power level of 2004 megawatts thermal (MWt), the license is also amended to add the following license conditions. These license conditions provide for monitoring, evaluating, and taking prompt action in response to potential adverse flow effects as a result of power uprate operation on plant structures, systems, and components (including verifying the continued structural integrity of the steam dryer). These license conditions are applicable to the initial power ascension from 1775 MWt to 2004 MWt (EPU) conditions:

(a) The following requirements are placed on the initial operation of the facility above the thermal power level of 1775 MWt for the power ascension to 2004 MWt. These conditions are applicable until the first time full EPU conditions (2004 MWt) are achieved. If the number of active strain gauges is less than two strain gauges (180 degrees apart) at any of the eight MSL locations, NSPM will stop power ascension and repair/replace the damaged strain gauges and only then resume power ascension.

   1. NSPM shall monitor the MNGP main steam line (MSL) strain gauges during power ascension above 1775 MWt for increasing pressure fluctuations in the steam lines. Upon the initial increase of power above 1775 MWt until reaching 2004 MWt, NSPM shall collect data from the MSL strain gauges at nominal 2.5 percent thermal power increments and evaluate steam dryer performance based on this data.

   2. During power ascension at each nominal 2.5 percent power level above 1775 MWt, the licensee shall compare the MSL data to the approved limit curves and determine the minimum alternating stress ratio. A summary of the results shall be provided for NRC review at approximately 105 percent and 110 percent of 1775 MWt.

   3. NSPM shall hold the facility at approximately 105 percent and 110 percent of 1775 MWt to perform the following:

      a. Collect strain data from the MSL strain gauges;

      b. Collect vibration data from the accelerometers in the following locations: MSLs (including those in the drywell, turbine building and in the steam tunnel), Feedwater Lines (FWLs) (including those in the drywell and turbine building), Safety Relief Valves (SRVs), Main Steam Isolation Valves (MSIVs) in the drywell, and Turbine Stop Valves (TSVs);

      c. Evaluate steam dryer performance based on MSL strain gauge data;

      d. Evaluate the measured vibration data collected from the vibration monitoring instruments at that power level, data projected to EPU conditions, trends, and to the acceptance limits;

e. Provide the steam dryer evaluation and the vibration evaluation, including the data collected, to the NRC staff by facsimile or electronic transmission to the NRC project manager upon completion of the evaluation;

f. NSPM shall not increase power above each hold point until 96 hours after the NRC project manager confirms receipt of the evaluations transmission or until verbal approval by NRC to increase power is provided, whichever comes first.

4. If any frequency peak from the MSL strain gauge data exceeds the Level 1 limit curves, NSPM shall return the facility to a power level at which the limit curve is not exceeded. NSPM shall resolve the discrepancy, evaluate and document the continued structural integrity of the steam dryer, and provide that documentation by facsimile or electronic transmission to the NRC project manager prior to further increases in reactor power. If a revised stress analysis is required to be performed and new limit curves are developed, then NSPM shall not further increase power above each hold point until 96 hours after the NRC project manager confirms receipt of the transmission or until verbal approval by NRC to increase power is provided, whichever comes first.

5. In addition to evaluating the MSL strain gauge data, NSPM shall monitor reactor pressure vessel water level instrumentation, and MSL piping accelerometers when power levels are increasing. If resonance frequencies are identified as increasing above nominal levels in proportion to strain gauge instrumentation data, NSPM shall stop power ascension, evaluate and document the continued structural integrity of the steam dryer, and provide that documentation to NRC staff by facsimile or electronic transmission to the NRC project manager prior to further increases in reactor power.

(b) NSPM shall implement the following actions for the initial power ascension from 1775 MWt to 2004 MWt condition.

1. In the event that acoustic signals (in MSL strain gauge signals) are identified that exceed the Level 1 limit curves during power ascension above 1775 MWt, NSPM shall evaluate dryer loads, and stresses, and re-establish the limit curves. In the event that stress analyses are re-performed based on new strain gauge data to address paragraph 15(a)4 above, the revised load definition, stress analysis, and limit curves shall include:

   a. Application of the ACE 2.0 and ACE 2.0-SPM values for percent bias error and for percent uncertainty to all the SRV acoustic resonances.

   b. Use of bump-up factors associated with all the SRV acoustic resonances as determined from the scale model test results.

   c. Evaluation of the effects of $\pm$ 10 percent frequency shifts in increments of 2.5 percent.

2. After reaching 2004 MWt, NSPM shall obtain measurements from the MSL strain gauges and establish the steam dryer flow-induced vibration load fatigue margin for the facility, update the dryer stress report, and re-establish the limit

curves with the updated load definition. This data will be provided to the NRC staff as described in license condition 15(e).

(c) NSPM shall prepare the EPU power ascension test procedure to include:

1. The stress limit curves to be applied for evaluating steam dryer performance;
2. Specific hold points and their durations during EPU power ascension;
3. Activities to be accomplished during the hold points;
4. Plant parameters to be monitored;
5. Inspections and walkdowns to be conducted for steam, feedwater, and condensate systems and components during the hold points;
6. Methods to be used to trend plant parameters;
7. Acceptance criteria for monitoring and trending plant parameters, and conducting the walkdowns and inspections;
8. Actions to be taken if acceptance criteria are not satisfied; and
9. Verification of the completion of commitments and planned actions specified in its application and all supplements to the application in support of the EPU license amendment request pertaining to the steam dryer prior to power increase above 1775 MWt. NSPM shall provide the related EPU startup test procedure sections to the NRC by facsimile or electronic transmission to the NRC project manager prior to increasing power above 1775 MWt.

(d) The following key attributes of the program for verifying the continued structural integrity of the steam dryer shall not be made less restrictive without prior NRC approval:

1. During initial power ascension testing above 1775 MWt, each test plateau increment shall be approximately 5 percent of 1775 MWt.
2. Level 1 performance criteria; and
3. The methodology for establishing the limit curves used for the Level 1 and Level 2 performance

(e) The results of the power ascension testing to verify the continued structural integrity of the steam dryer shall be submitted to the NRC staff in a report that includes a final load definition and stress report of the steam dryer, including the results of a complete re-analysis using the ACE 2.0 and ACE 2.0-SPM specific bias and uncertainties. The report will be provided within 90 days of the completion of EPU power ascension testing.

(f) During the first two scheduled refueling outages after reaching EPU conditions, a visual inspection shall be conducted of all accessible, susceptible locations of the steam dryer in accordance with the inspection guidelines provided to the NRC.

(g) The results of the visual inspections of the steam dryer shall be reported to the NRC staff within 90 days following startup from the respective refueling outage.

(h) At the end of the second refueling outage, following the implementation of the EPU, the licensee shall submit a long-term steam dryer inspection plan based on industry operating experience along with the baseline inspection results for NRC review and approval.

The license conditions described above shall expire (1) upon satisfaction of the requirements in Paragraphs 15(f) and 15(g), provided that a visual inspection of the steam dryer does not reveal any new unacceptable flaw(s) or unacceptable flaw growth that is due to fatigue, and (2) upon satisfaction of the requirements specified in Paragraph 15(h).

References

1. Letter L-MT-08-018 from T. J. O'Connor (NSPM) to USNRC Document Control Desk, "License Amendment Request: Extended Power Uprate," dated March 31, 2008 (ADAMS Accession No. ML081010189)

Enclosure 13 - Steam Dryer Dynamic Stress Evaluation (non-proprietary version of Enclosure 11) (ADAMS Accession No. ML081010198)

Enclosure 11 - Steam Dryer Dynamic Stress Evaluation (proprietary) (ADAMS Accession No. ML081010202)

2. Letter L-MT-08-052 from T. J. O'Connor (NSPM) to USNRC Document Control Desk, "License Amendment Request Extended Power Uprate (TAC MD9990)," dated November 5, 2008 (ADAMS Accession No. ML083230113 (package))

Enclosure 8 - Planned Modifications for Monticello Power Uprate (ADAMS Accession No. ML083230112 (non-proprietary) and ML083230125 (proprietary))

Enclosure 9 - Monticello Nuclear Generating Plant Extended Power Uprate Startup Test Plan (ADAMS Accession No. ML083230112)

Enclosure 11 - Steam Dryer Dynamic Stress Evaluation (proprietary) (ADAMS Accession Nos. ML083230115, ML083230116, and ML083230117)

Enclosure 13 - Steam Dryer Dynamic Stress Evaluation (non-proprietary) (ADAMS Accession No. ML083230114)

3. Title 10 of the *Code of Federal Regulations,* Part 50 (10 CFR Part 50)

4. Letter L-MT-10-046 from T. J. O'Connor (NSPM) to USNRC Document Control Desk, "Monticello Extended Power Uprate: Replacement Steam Dryer Supplement (TAC MD9990)," dated June 30, 2010 (ADAMS Accession No. ML102010462)

Enclosure 1 - Overview of the design and analyses performed for the RSD

Enclosure 2 - Westinghouse Electric Company, LLC (WEC) WCAP-17085-P, Revision 1, "Monticello Replacement Steam Dryer Structural Evaluation for High-Cycle Acoustic Loads"

Enclosure 3 - WEC Report, SES 09-127-P, Revision 2, "Monticello Steam Dryer Replacement - Structural Verification of Steam Dryer"

Enclosure 5 - WEC Report, WCAP-17251-P, Revision 0 "Monticello Replacement Steam Dryer Four Line Acoustic Subscale Testing Report"

Enclosure 6 - WEC Report WCAP-17252-P, Revision 0, "Acoustic Loads Definition for the Monticello Steam Dryer Replacement Project"

Enclosure 7 - Letter LTR-A&SA-09-32, Revision 2, from WEC dated June 21, 2010. Power ascension test plan (PATP) limit curves

5. Letter L-MT-11-004 from T. J. O'Connor (NSPM) to USNRC Document Control Desk, "Monticello Extended Power Uprate: Replacement Steam Dryer Supplement (TAC MD9990)," dated January 13, 2012 (ADAMS Accession Nos. ML12019A246 and ML12019A247)

   Enclosure 1 - Response to EMCB-SD-RAI-15-S02 and Supplemental Information & Changes to PATP

   Enclosure 2 - Response Audit Action Item 17-Fabrication and Weld qualification of the RSD

6. Letter L-MT-12-056 from M. A. Schimmel (NSPM) to USNRC Document Control Desk, "Monticello Extended Power Uprate: Replacement Steam Dryer – Second Set of Responses to Requests for Additional Information (TAC MD9990)," dated July 19, 2012 (ADAMS Accession No. ML122070642)

   Enclosure 1 - Westinghouse Electric Company, LLC (WEC) letter LTR-A&SA-12-10, Revision 1, P - Attachment, "Monticello Replacement Steam Dryer RAI-and Action Item Responses for Acoustic/Structural Evaluation" (proprietary) (ADAMS Accession No. ML12207A546)

   Enclosure 2 - WEC Report WCAP-17548-P, Revision 0, "Signal Processing Performed on Monticello MSL Strain Gauge and RSD Instrumentation Data" (proprietary) (ADAMS Accession No. ML12207A547); (non-proprietary) (ADAMS Accession No. ML12207A542)

   Enclosure 3 - WEC Report WCAP-17540-P, Revision 0, "Monticello Replacement Steam Dryer Program Acoustic Load Definition Methodology" (proprietary) (ADAMS Accession No. ML12207A548); (non-proprietary) (ADAMS Accession No. ML12207A543)

   Enclosure 4 - WEC Report WCAP-17252-P, Revision 2, "Acoustic Loads Definition for the Monticello Steam Dryer Replacement Project" (proprietary) (ADAMS Accession No. ML12207A549); (non-proprietary) (ADAMS Accession No. ML12207A544)

   Enclosure 5 - WEC Report WCAP-17549-P, Revision 0, "Monticello Replacement Steam Dryer Structural Evaluation for High-Cycle Acoustic Loads Using ACE" (proprietary)

(ADAMS Accession No. ML12207A550); (non-proprietary) (ADAMS Accession No. ML12207A545)

Enclosure 6 - WEC letter LTR-A&SA-09-32, Revision 5, "Limit Curves for Monticello Power Ascension" (ADAMS Accession No. ML12207A550).

7. Letter L-MT-13-023 from M. A. Schimmel (NSPM) to USNRC Document Control Desk, Monticello Extended Power Uprate: Replacement Steam Dryer - Responses to Requests for Additional Information (TAC MD9990), dated February 22, 2013 (ADAMS Accession Nos. ML13057A034 and ML13057A035).

   Enclosure 1 - Westinghouse Electric Company, LLC (WEC) letter LTR-A&SA-13-1, "Monticello Replacement Steam Dryer RAI-Responses for Acoustic/Structural Analyses," dated February 19, 2013. Response to MNGP EPU-EMCB-RSD-RAIs 42, 44, 50, 51(a), 52, 65, 67 and 81

   Enclosure 4 - Structural Integrity Associates, Inc. Report No. 1200978.402.RO, "Response to Monticello RAI-- EPU-EMCB-RSD-RAI-68"

8. Letter L-MT-13-027 from M. A. Schimmel (NSPM) to USNRC Document Control Desk, Monticello Extended Power Uprate: Replacement Steam Dryer - Responses to Requests for Additional Information (TAC MD9990), dated March 7, 2013 (ADAMS Accession Nos. ML13071A615 and ML13071A616)

   Enclosure 1 contains Westinghouse Electric Company, LLC (WEC) letter LTR-A&SA-13-2, Revision 1, "Monticello Replacement Steam Dryer RAI-Responses for Acoustic/Structural Analyses Set #2," dated March 4, 2013. Response to MNGP EPU-EMCB-RSD-RAIs 43, 45, 46, 48, 54, 59, 60, 63, 64, 72(a), 74, 75, and 79

9. Letter L-MT-13-028 from M. A. Schimmel (NSPM) to USNRC Document Control Desk, Monticello Extended Power Uprate: Replacement Steam Dryer - Responses to Requests for Additional Information (TAC MD9990), dated March 18, 2013 (ADAMS Accession Nos. ML13078A390 and ML13078A393)

   Enclosure 1 - Westinghouse Electric Company, LLC (WEC) letter LTR-A&SA-13-6, "Monticello Replacement Steam Dryer RAI-Responses for Acoustic/Structural Analyses Set #3," dated March 4, 2013. Response to MNGP EPU-EMCB-RSD-RAIs 47, 49, 51(b), 53, 55, 56, 57, 58, 61, 62, 66, 71, 73, 77, 78 and 80

10. Letter L-MT-13-029 from M. A. Schimmel (NSPM) to USNRC Document Control Desk, Monticello Extended Power Uprate: Replacement Steam Dryer - Responses to Requests for Additional Information (TAC MD9990), dated March 29, 2013 (ADAMS Accession Nos. ML130920389)

   Enclosure 1 - Westinghouse Electric Company, LLC (WEC) letter LTR-A&SA- 13-7, P-Attachment, Revision 1, "Monticello Replacement Steam Dryer RAI-Responses for Acoustic/Structural Analyses Set #4," dated March 27, 2013 (proprietary) (ADAMS Accession No. ML13092A354) – Response to MNGP EPU-EMCB-RSD-RAIs 44(c), 69, 70, 72(b), 72(c), 76, 82, 83 and 84

Enclosure 2 - Westinghouse Electric Company, LLC (WEC) WCAP-17548- P, Rev. 1, "Signal Processing Performed on Monticello MSL Strain Gauge and RSD Instrumentation Data," dated March 2013 (proprietary) (ADAMS Accession No. ML13092A355); (non-proprietary) (ADAMS Accession No. 13092A349)

Enclosure 3 - WEC WCAP-17251-P, Revision 1, "Monticello Replacement Steam Dryer Four-Line Acoustic Subscale Testing Report," dated March 2013 (proprietary) (ADAMS Accession No. ML13092A356); (non-proprietary) (ADAMS Accession No. ML13092A350)

Enclosure 4 - WEC WCAP-17252-P, Revision 3, "Acoustic Loads Definition for the Monticello Steam Dryer Replacement Project," dated March 2013 (proprietary) (ADAMS Accession No. ML13092A357); (non-proprietary) (ADAMS Accession No. ML13092A351)

Enclosure 5 - WEC WCAP-17549-P, Revision 1, "Monticello Replacement Steam Dryer Structural Evaluation for High-Cycle Acoustic Loads Using ACE," dated March 2013 (proprietary) (ADAMS Accession No. ML13092A358); (non-proprietary) (ADAMS Accession No. ML13092A352)

Enclosure 6 - WEC WCAP-17716, Revision 0, "Benchmarking of the Acoustic Circuit Enhanced Revision 2.0 for the Monticello Steam Dryer Replacement Project," dated March 2013 (proprietary) (ADAMS Accession No. ML13092A359); (non-proprietary) (ADAMS Accession No. ML13092A353)

Enclosure 7 - WEC letter LTR-A&SA-09-32, Revision 6, "Limit Curves for Monticello Power Ascension," dated March 20, 2013 (proprietary) (ADAMS Accession No. ML13092A359).

11. NUREG-800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition"

12. Review Standard for Extended Power Uprates, RS-001, Revision 0, December 2003

13. Regulatory Guide 1.20, Revision 3, "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing," dated March 2007

14. Monticello Updated Safety Analysis Report (USAR), Revision 27 (non-public) (ADAMS Accession No. ML112301566)

15. NUREG-1865, "Safety Evaluation Report Related to the License Renewal of the Monticello Generating Plant," dated October 2006 (ADAMS Accession No. ML063050414)

16. BWRVIP-182, "Guidance for Demonstration of Steam Dryer Integrity for Power Uprate," dated May 2010

17. Audit Report Replacement Steam Dryer – Nordic Steam Dryer Supplied by Westinghouse, Monticello Nuclear Generating Plant Extended Power Uprate License Amendment Request (TAC NO. MD9990), Audit Dates - April 7-8, 2011 (ADAMS Accession Nos. ML11144A085 and ML11144A096)

18. Letter L-MT-11-044 from T. J. O'Connor (NSPM) to USNRC Document Control Desk, "Monticello Extended Power Uprate: Update on EPU Commitments (TAC MD9990)," dated August 30, 2011, Enclosure 4 (ADAMS Accession No. ML11249A045)

19. Letter L-MT-13-074 from M. A. Schimmel (NSPM) to USNRC Document Control Desk, Monticello Extended Power Uprate: Replacement Steam Dryer - Responses to Requests for Additional Information (TAC MD9990), dated July 18, 2013 (ADAMS Accession No. ML13205A110)

Enclosure 1 - Westinghouse Letter, LTR-A&SA-13-14, P-Attachment, Revision 0, Responses to the U.S. NRC Request for Additional Information Relative to the Monticello Replacement Steam Dryer Acoustic/Structural Analyses Set #5 – Response to MNGP EPU-EMCB-RSD-RAIs 85-90, 94, 95, 99-101, 105-108, 110-114, 116-118 (ADAMS Accession No. ML13205A087)

20. Letter L-MT-13-076 from M. A. Schimmel (NSPM) to USNRC Document Control Desk, "Monticello Extended Power Uprate: Replacement Steam Dryer - Responses to Requests for Additional Information (TAC MD9990)," dated August 2, 2013 (ADAMS Accession No. ML13218B338)

Enclosure 1 - Westinghouse Letter, LTR-A&SA-13-10, P-Attachment, Responses to the U.S. NRC Request for Additional Information Relative to the Monticello Replacement Steam Dryer Acoustic/Structural Analyses Set #6 – Response to MNGP EPU-EMCB-RSD-RAIs 91, 92, 96, 98, 102-104, 110, 115, 117, and 119 (ADAMS Accession No. ML13218B340)

Enclosure 3 - Northern States Power Company - Minnesota – Response to MNGP EPU-EMCB-RSD-RAIs 93, 97, and 100 (ADAMS Accession No. ML13218B339)

Enclosure 5 - Westinghouse Letter, LTR-A&SA-13-15, P-Attachment, Responses to the U.S. NRC Clarification Questions 2 and 3 (ADAMS Accession No. ML13218B340)

21. ASME Boiler and Pressure Vessel Code, Section III, Subsection NG, 2004

22. BWRVIP-139-A, BWR Vessel and Internals Project, Steam Dryer Inspection and Flaw Evaluation Guidelines, dated July 2009

23. Letter L-MT-13-092 from K. D. Fili (NSPM) to USNRC Document Control Desk, "Monticello Extended Power Uprate (EPU): Completion of EPU Commitments, Proposed License Conditions and Revised Power Ascension Test Plan (TAC MD9990)," dated September 30, 2013 (ADAMS Accession No. ML13275A063)

24. Monticello, Enclosure 1 to L-MT-13-076, Westinghouse Letter, LTR-A&SA-13-10, P-Attachment, Responses to US NRC Request for Additional Information Relative to Monticello Replacement Steam Dryer, Acoustic/Structural Analyses Set #6 and Enclosure 5 dated August 2, 2013 (ADAMS Accession No. ML13218B340)

ATTACHMENT – LIST OF ACRONYMS

ACRONYM	DEFINITION
A	amperes
AAC	alternate alternating current
AAF	acceptable as found
AAL	acceptable as left
AC	alternating current
ACE	Acoustic Circuit Model Enhanced
ACE-SPM	ACE-Skirt Protection Model
ACM	Acoustic Circuit Model
ACRS	Advisory Committee on Reactor Safeguards
ADAMS	Agencywide Documents Access and Management System
ADS	automatic depressurization system
AEC	Atomic Energy Commission
ALARA	as low as reasonably achievable
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOO	anticipated operational occurrence
AOP	abnormal operating procedure
AOR	analyses of record
AOV	air-operated valve
AP	annulus pressurization
APLHGR	average planar linear heat generation rate
APRM	average power range monitor
ARAVS	auxiliary and radwaste area ventilation system
ARE	Applied Reliability Engineering, Inc.
ARI	alternate rod injection
ART	adjusted Reference temperature
ARTS	average power range monitor, rod block monitor technical specifications
ASCM	alternate shutdown cooling method
ASDS	alternate shutdown system

ATTACHMENT – LIST OF ACRONYMS

ACRONYM	DEFINITION
ASHRAE	American Society of Heating, Refrigerating and Air-Conditioning Engineers
ASME	American Society of Mechanical Engineers
ASP	accident sequence precursor
AST	alternative/alternate source term
ASTM	American Society for Testing and Materials
ATWS	anticipated transient without scram
AV	allowable value
B&PV	[ASME] Boiler and Pressure Vessel [Code]
BOC	beginning of cycle
BOP	balance-of-plant
BPWS	banked position withdrawal sequence
BSP	backup stability protection
BSW	biological shield wall
BTP	Branch Technical Position
BTU/lbm	British thermal units per pounds mass
B/U	bias errors and uncertainty
BUF	bump-up factor
BWR	boiling-water reactor
BWROG	Boiling Water Reactor Owners' Group
BWRVIA	BWR Vessel and Internals Application
BWRVIP	Boiling Water Reactor Vessel and Internals Project
cal/g	calories per gram
CAP	containment accident pressure
CCDP	conditional core-damage probability
CCF	common cause failure
CCFP	conditional containment failure probability
CDF	core damage frequency
CEQ	Council on Environmental Quality
CFD	computational fluid dynamics

ATTACHMENT – LIST OF ACRONYMS

ACRONYM	DEFINITION
CFR	Code of Federal Regulations
CFS	condensate and feedwater system
CLB	current licensing basis
CLTP	current licensed thermal power (1775 MWth)
CLTR	constant pressure power uprate licensing topical report
CO	condensation oscillation
COLR	Core Operating Limits Report
CPPU	constant pressure power uprate
CPR	critical power ratio
CR	control room
CRAVS	control room area ventilation system
CRC	corrosion resistant cladding
CRD	control rod drive
CRDA	control rod drop accident
CRDH	control rod drive hydraulics
CRDM	control rod drive mechanism
CREF	control room emergency filtration
CS	core spray
CST	condensate storage tank
CUF	cumulative fatigue usage factor
CWS	circulating water system
DBA	design-basis accident
DBLOCA	design-basis loss-of-coolant accident
DC	direct current
DE	dose equivalent
DHR	decay heat removal
DIRPT	dual pump trip delta CPR
DIVOM	delta over initial CPR versus oscillation magnitude
DOR	Division of Operating Reactors

ATTACHMENT – LIST OF ACRONYMS

ACRONYM	DEFINITION
DSS/CD	Detect and Suppress Solution – Confirmation Density
DW	drywell
EA	environmental assessment
EAB	exclusion area boundary
ECCS	emergency core cooling system
ECP	estimated critical position
EDG	emergency diesel generator
EEEB	Electrical Engineering Branch
EFDS	equipment and floor drainage system
EFPY	effective full-power years
EFT	emergency filtration train
EHC	electrohydraulic control
EIS	environmental impact statement
ELTR1	GE Licensing Topical Report NEDC-32424P-A
ELTR2	GE Licensing Topical Report NEDC-32523P-A
EMA	equivalent margins analysis
EMCB	Mechanical and Civil Engineering Branch
EOC	end-of-cycle
EOL	end-of-life
EOP	emergency operating procedure
EPG	emergency procedure guidelines
EPRI	Electric Power Research Institute
EPU	extended power uprate
EQ	environmental qualification
ESF	engineered safety features
ESFAS	engineered safety features actuation system
ESFVS	engineered safety feature ventilation system
ESW	emergency service water
FAC	flow-accelerated corrosion

ATTACHMENT – LIST OF ACRONYMS

ACRONYM	DEFINITION
FE	finite element
FHA	fuel handling accident
FIC	flow-induced corrosion
FIV	flow-induced vibration
Fn	natural frequency
Fs	shedding frequency
FOL	facility operating license
FONSI	Finding of No Significant Impact
FPCCS	fuel pool cooling and cleanup system
FPP	fire protection program
FR	Federal Register
Fs	shedding frequency
ft-lb	foot-pound (force)
F-V	Fussell-Vesely
FW	feedwater
Gd	gadolinia
GDC	General Design Criteria (or Criterion)
GE	General Electric
GEH	GE-Hitachi Nuclear Energy
GESTAR	General Electric Standard Application for Reactor Fuels
GEZIP	GE zinc injection process
GL	Generic Letter
gmp	gallons per minute
GWd/MTU	gigawatt days per metric ton uranium
GWMS	gaseous waste management systems
HCOM	hot channel oscillation magnitude
HCTL	heat capacity temperature limit
HCU	hydraulic control unit
HDFSS	High Density Fuel Storage System

ATTACHMENT – LIST OF ACRONYMS

ACRONYM	DEFINITION
HELB	high energy line break
HEM	Homogeneous Equilibrium Model
HEP	human error probability
HEPA	high efficiency particulate air
HP	high pressure
HPCI	high pressure coolant injection
HRA	human reliability analysis
HSBW	hot shutdown boron weight
HVAC	heating, ventilation, and air conditioning
HWC	hydrogen water chemistry
IASCC	irradiation assisted stress-corrosion cracking
I&C	instrumentation and control
ICA	interim correction action
ICPR	initial critical power ratio
ID	inside diameter
IEEE	Institute of Electrical and Electronics Engineers
IGSCC	intergranular stress-corrosion cracking
ILRT	integrated leak rate test
IMLTR	Interim Methods Licensing Topical Report
IN	Information Notice
IPB	isolated phase bus
IPE	individual plant examinations
IPEEE	individual plant examinations of external events
IR	interaction ratio
IR	Inspection Report
ISHI	induction heating stress improvement
ISI	inservice inspection
ISP	integrated surveillance program
IST	inservice testing

ATTACHMENT – LIST OF ACRONYMS

ACRONYM	DEFINITION
JIT	Just In Time
kA	kilo-amperes
kV	kilovolt
LAR	license amendment request
LBLOCA	large-break loss-of-coolant accident
LCO	limiting condition for operation
LER	licensee event report
LERF	large early release frequency
LHGR	linear heat generation rate
LLHS	light load handling system
LLRT	local leak rate test
LOCA	loss-of-coolant accident
LOCV	loss of condenser vacuum
LOFW	loss of feedwater
LOFWH	loss of feedwater heating
LOOP	loss of offsite power
LP	low pressure
LPCI	low pressure coolant injection
LPRM	local power range monitor
LPSP	low power set point
LPZ	low population zone
LRNBP	load rejection with steam bypass
LSSS	limited safety system setting
LTR	Licensing Topical Report
LWMS	liquid waste management system
M&E	mass and energy
MAAP	Modular Accident Analysis Program
MAPLHGR	maximum average planar linear heat generation rate
MASR	minimum alternating stress ratio

ATTACHMENT – LIST OF ACRONYMS

ACRONYM	DEFINITION
MBTU/hr	million British thermal units per hour
MCC	motor control center
MCO	moisture carryover
MCES	main condenser evacuation system
MCPR	minimum critical power ratio
MCR	main control room
MCS	main condenser system
MELB	moderate energy line break
MELLLA	Maximum Extended Load Line Limit Analysis
MELLLA+	Maximum Extended Load Line Limit Analysis Plus
MeV	megaelectron-volt
MISO	Midwest Independent System Operator
MLHGR	maximum linear heat generation ratio
MNGP	Monticello Nuclear Generating Plant
MOC	middle of cycle
MOV	motor-operated valve
MOX	mixed-oxide
mr/hr	millirem per hour
mrem	millirem
MS	main stream
MSIV	main steam isolation valve
MSIVF	main steam isolation valve (MSIV) closure with scram on high flux
MSLB	main steam line break
MSLBA	main steam line break accident
MSO	multiple spurious operations
MSRV	main steam relief valve
MSSS	main steam supply system
MWd/MT	megawatt-day per metric ton
MWe	megawatts electric

ATTACHMENT – LIST OF ACRONYMS

ACRONYM	DEFINITION
MWth	megawatts thermal
N-16	nitrogen-16
n/cm^2	neutrons per squared centimeter (measure of fluence)
NDE	nondestructive examination
NDT	nil ductility temperature
NEI	Nuclear Energy Institute
NEPA	National Environmental Policy Act
NPSH	net positive suction head
NPSHa	net positive suction head available
NPSHr	net positive suction head required
NRC	U.S. Nuclear Regulatory Commission
NRR	NRC's Office of Nuclear Reactor Regulation
NSHC	no significant hazards consideration
NSPM	Northern States Power Company – Minnesota
NSSS	nuclear steam supply system
NUMARC	Nuclear Management and Resource Council, Inc.
OLMCPR	operating limit minimum critical power ratio
OLTP	original licensed thermal power
OM Code	[ASME] Operations and Maintenance of Nuclear Power Plants Code
OPRM	oscillation power range monitor
ORNL	Oak Ridge National Laboratory
PATP	power ascension and test plan
PBDA	Period Based Detection Algorithm
PCI	pellet-clad interaction
PCIS	primary containment isolation system
PCT	peak cladding temperature
pH	potential of hydrogen (measure of the acidity or alkalinity of a solution)
ppb	parts per billion
ppm	parts per million

ATTACHMENT – LIST OF ACRONYMS

ACRONYM	DEFINITION
PRA	probabilistic risk assessment
PRFO	Pressure regulator failure - open
PSA	probabilistic safety assessment
psi	pounds per square inch
psia	pounds per square inch absolute
psid	pounds per square inch differential
psig	pounds per square inch gauge
P-T	pressure-temperature
PULD	plant unique load definition
PUSAR	Power Uprate Safety Analysis Report
RAI	request for additional information
RAVS	radwaste area ventilation system
RAW	risk achievement worth
RB	reactor building
RBCCW	reactor building closed-loop cooling water
RBM	rod block monitor
RCIC	reactor core isolation cooling
RCPB	reactor coolant pressure boundary
RCS	reactor coolant system
rem	roentgen equivalent man
RG	Regulatory Guide
RHR	residual heat removal
RHRSW	residual heat removal service water
RIA	reactivity insertion accident
RIS	Regulatory Issue Summary
RLB	recirculation line break
RMS	root mean square
RPS	reactor protection system
RPT	recirculation pump trip

ATTACHMENT – LIST OF ACRONYMS

ACRONYM	DEFINITION
RPV	reactor pressure vessel
RRMG	reactor recirculation motor-generator
RRS	reactor recirculation system
RSD	replacement steam dryer
RTP	rated thermal power
RTO	Regional Transmission Organization
RV	reactor vessel
RWCS	reactor water cleanup system
RWCU	reactor water cleanup
RWE	rod withdrawal error
SAFDL	specified acceptable fuel design limits
SAGs	severe accident guidelines
SAMG	severe accident management guidelines
SBA	steam line break accident
SBO	station blackout
SCC	stress-corrosion cracking
SCF	stress concentration factor
scfh	standard cubic feet per hour
scfm	standard cubic feet per minute
SDC	shutdown cooling
SE	safety evaluation
SER	safety evaluation report
SFP	spent fuel pool
SFPAVS	spent fuel pool area ventilation system
SGTS	standby gas treatment system
SHE	standard hydrogen electrode
SHEX GEH	Super Hex General Electric Hitachi
SIC	Safety Information Communication
SIL	Services Information Letter

ATTACHMENT – LIST OF ACRONYMS

ACRONYM	DEFINITION
SL	safety limit
SLC	standby liquid control
SLMCPR	safety limit minimum critical power ratio
SLO	single recirculation loop operation
SMT	scale model test
SPDS	safety parameter display system
SORV	stuck open relief valve
SR	surveillance requirement
SRLR	Supplemental Reload Licensing Report
SRP	Standard Review Plan
SRV	safety relief valve
SRVDL	safety relief valve discharge line
SSCs	structures, systems, and components
SSE	safe shutdown earthquake
Sv	sievert
SW	service water
SWS	service water system
TAVS	turbine area ventilation system
TBS	turbine bypass system
TCD	thermal conductivity degradation
TEDE	total effective dose equivalent
TG	turbine generator
TGSS	turbine gland sealing system
TID	technical information document
TIP	traversing incore probe
TLAA	time-limited aging analysis
TLO	two recirculation loop operation
TS	Technical Specifications
TSC	technical support center

ATTACHMENT – LIST OF ACRONYMS

ACRONYM	DEFINITION
TSV	turbine stop valve
TTNBP	turbine trip with bypass failure and scram on high flux
USAR	Updated Safety Analysis Report
UHS	ultimate heat sink
USAR	Updated Safety Analysis Report
UPS	uninterruptible power supply
USE	upper shelf energy
UT	ultrasonic testing
VPF	vane passing frequency
wt%	weight percent
X/Q	atmosphere dispersion factor
°C	degrees Celsius or degrees Centigrade (measure of temperature)
°F	degrees Fahrenheit (measure of temperature)
1/4T	one-quarter thickness
%	percent
μS/cm	microsiemens per centimeter (measure of conductivity)