Enclosure 7 to L-MT-08-052

Safety Analysis Report For Monticello Constant Pressure Power Uprate (Non-Proprietary Version of Enclosure 5)

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SAFETY ANALYSIS REPORT
FOR
MONTICELLO
CONSTANT PRESSURE POWER UPRATE

NON-PROPRIETARY INFORMATION NOTICE

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IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT

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Revisions

No.	Change	
0	NA	
1	Clarified text regarding increased power level analysis bases in the Executive Summary, Introduction, and Section 2.4.1.	
2	Unmarked minor editorial changes, corrected reference pointer, renumbered tables, grammatical corrections, NSPM name change, etc.	
	Section 2.1.4	amplified text regarding components that exceed fluence threshold values.
	Section 2.1.6	added text to discuss GL 89-13 program.
	Section 2.1.7	added discussion of Reactor Water Iron and conductivity changes with EPU and added new Table 2.1-5 with comparative values.
	Section 2.2.2.1	BOP sub-section, expanded pipe break location discussion.
	Section 2.2.3	added CLTR as the evaluation basis and added pointer to new Table 2.2-3 FIV Component Analysis Results. Added expanded scaling methodology description.
	Section 2.2.3 (h)	amplified discussion of channel/control blade interference.
	Section 2.2.5	clarified text regarding Mechanical EQ.
	Section 2.3.2	added information regarding the degraded voltage time delay.
	Table 2.3-2	Updated Guaranteed Generator Output Value.
	Section 2.3.5	added pointers to new Reference35 and Table 2.3-4, SBO Sequence of Events.
	Section 2.5.1.2.2	clarified text regarding turbine stop valve capability.
	Section 2.5.3.1.3	amplified SFP Radiation Levels discussion.
	Section 2.5.5.2	added numerical value of Radioactive Waste system capacity at EPU.
	Section 2.5.5.3	amplified solid radioactive waste discussion.
	Section 2.6.1	added reference pointer to new Reference 34.
	Section 2.6.5	modified ECCS NPSH Discussion. added basis for ECCS Strainer loading.
	Tables 2.6-3 through 2.6-9 updated based on latest analyses.	
	Figures 2.6-2 through 2.6-7 updated based on latest analyses.	
	Figure 2.6-8	added new figure.
	Section 2.8.2.6	added pointer to new Appendix A, IMLTR Limitations.
	Section 2.8.4.2	added ODYN computer code from Table 1-1 to overpressure text.
	Section 2.8.5.2.3	added decay heat basis to LOFW event description.
	Section 2.8.5.7	added pointer to three new tables for ATWS Sequence of Events; Tables 2.8-6, 2.8-7, and 2.8-8, renumbered other tables.
	Section 2.9.2	clarified text for MSLBA dose rate analysis.
	Section 2.11.1	clarified operator training completion requirements for EPU changes.
	Section 3.0	added new References 34 and 35.
	Appendix A	added new appendix to provide the dispositions of the Safety Evaluation Limitations from the Safety Evaluation for the Interim Methods Licensing Topical Report, NEDC-33173P.

No.	Change	
3	Section 2.3.1	updated section with most recent evaluation results and deleted Table 2.3-1, renumbered remaining tables.

Revision bars in the right hand margins indicate the changes to the document since it was originally submitted to the NRC.

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Glossary of Terms

Term	Definition
AC	Alternating Current
ADS	Automatic Depressurization System
AL	Analytical Limit
AEC	Atomic Energy Commission
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOO	Anticipated Operational Occurrence (moderate frequency transient event)
APRM	Average Power Range Monitor
ARI	Alternate Rod Insertion
ARTS	APRM/RBM/Technical Specifications
ASME	American Society of Mechanical Engineers
AST	Alternate Source Term
ATWS	Anticipated Transient Without Scram
AV	Allowable Value
BHP	Brake Horse Power
BIIT	Boron Injection Initiation Temperature
BOC	Beginning of Cycle
BOP	Balance-of-Plant
BPWS	Banked Position Withdrawal Sequence
BWR	Boiling Water Reactor
BWROG	BWR Owners Group
BWRVIP	BWR Vessel and Internals Project
CDF	Core Damage Frequency
CFD	Condensate Filter Demineralizer
CFR	Code of Federal Regulations
CLTP	Current Licensed Thermal Power
CLTR	Constant Pressure Power Uprate Licensing Topical Report
CO	Condensation Oscillation
COLR	Core Operating Limits Report
CRAVS	Control Room Area Ventilation System

Term	Definition
CRD	Control Rod Drive
CRDA	Control Rod Drop Accident
CREF	Control Room Emergency Filtration System
CRGT	Control Rod Guide Tube
CSC	Containment Spray Cooling
CS	Core Spray
CSS	Core Support Structure
CST	Condensate Storage Tank
CUF	Cumulative Usage Factors
CWS	Circulating Water System
DBA	Design Basis Accident
DC	Direct Current
dP	Differential Pressure
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EFDS	Equipment and Floor Drainage System
EFPY	Effective Full Power Years
EOC	End of Cycle
EOP	Emergency Operating Procedure(s)
EPRI	Electric Power Research Institute
EPU	Extended Power Uprate
EQ	Environmental Qualification
ESF	Engineered Safety Feature
ESFVS	Engineered Safety Feature Ventilation System
ESW	Emergency Service Water
FAC	Flow Accelerated Corrosion
FFWTR	Final Feedwater Temperature Reduction
FHA	Fuel Handling Accident
FIV	Flow Induced Vibration
FPCC	Fuel Pool Cooling and Cleanup
FPP	Fire Protection Program

Term	**Definition**
FW	Feedwater
FWHOOS	Feedwater Heater Out Of Service
GDC	General Design Criteria
GE	General Electric Company
GEH	GE-Hitachi Nuclear Energy
GEZIP	Zinc Injection System
GL	NRC Generic Letter
GNF	Global Nuclear Fuel LLC
HX	Heat Exchanger
HELB	High Energy Line Break
Hg_a	Inches of Mercury Absolute
HPCI	High Pressure Coolant Injection
HPT	High Pressure Turbine
HVAC	Heating Ventilating and Air Conditioning
IASCC	Irradiation-Assisted Stress Corrosion Cracking
ICF	Increased Core Flow
ICS	Integrated Computer System
IEEE	Institute of Electrical and Electronics Engineers
IGSCC	Intergranular Stress Corrosion Cracking
ILBA	Instrument Line Break Accident
IPE	Individual Plant Examination
IRM	Intermediate Range Monitor
ISP	Integrated Surveillance Program
LCS	Leakage Control System
LDS	Leak Detection System
LERF	Large Early Release Frequency
LHGR	Linear Heat Generation Rate
LOCA	Loss-Of-Coolant Accident
LOFW	Loss Of Feedwater
LOOP	Loss of Offsite Power
LPCI	Low Pressure Coolant Injection

Term	**Definition**
LPRM	Local Power Range Monitor
LPSP	Low Power Setpoint
LTR	Licensing Topical Report
MAAP	Modular Accident Analysis Program
MAPLHGR	Maximum Average Planar Linear Heat Generation Rate
MBTU	Millions of BTUs
MCES	Main Condenser Evacuation System
MCPR	Minimum Critical Power Ratio
MCR	Main Control Room
MELB	Moderate Energy Line Break
MELLLA	Maximum Extended Load Line Limit Analysis
MeV	Million Electron Volts
MFLCPR	Maximum Fraction of Limiting Critical Power Ratio
MFLPD	Maximum Fraction of Limiting Power Density
MISO	Midwest Independent System Operator
Mlb	Millions of pounds
Monticello	Monticello Nuclear Generating Plant
MOC	Middle of Cycle
MOV	Motor Operated Valve
MS	Main Steam
MSIV	Main Steam Isolation Valve
MSL	Main Steam Line
MSLB	Main Steam Line Break
MSLBA	Main Steam Line Break Accident
MSRV	Main Steam Relief Valve
MSVV	Main Steam Valve Vault
Mvar	Megavar
MWe	Megawatts-electric
MWt	Megawatt-thermal
MVA	Million Volt Amps
NA	Not Applicable

Term	Definition
NSPM	Northern States Power-Minnesota
NPSH	Net Positive Suction Head
NPSHR	Net Positive Suction Head Required
NRC	Nuclear Regulatory Commission
NSP	Northern States Power Company
NSSS	Nuclear Steam Supply System
NUREG	Nuclear Regulatory Commission Technical Report
OLMCPR	Operating Limit Minimum Critical Power Ratio
OLTP	Original Licensed Thermal Power
OOS	Out-of-Service
ΔP	Differential Pressure - psi
P_{25}	25% of EPU Rated Thermal Power
PCS	Pressure Control System
PCT	Peak Cladding Temperature
PRA	Probabilistic Risk Assessment
PRFO	Pressure Regulator Fail Open
PSA	Probabilistic Safety Analysis
PSF	Performance-Shaping Factor
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 and Temperature
PUSAR	Power Uprate Safety Analysis Report
RAVS	Radwaste Area Ventilation System
RBCCW	Reactor Building Closed Cooling Water
RBM	Rod Block Monitor
RCS	Reactor Coolant System
RCIC	Reactor Core Isolation Cooling
RCIS	Rod Control and Information System
RCPB	Reactor Coolant Pressure Boundary

Term	**Definition**
RCW	Raw Cooling Water
RG	Regulatory Guide
RHR	Residual Heat Removal
RHRSW	Residual Heat Removal Service Water
RIPD	Reactor Internal Pressure Difference(s)
RPT	Recirculation Pump Trip
RLA	Reload Licensing Analysis
RPV	Reactor Pressure Vessel
RSLB	Recirculation System Line Break
RRS	Reactor Recirculation System
RTP	Rated Thermal Power
RT_{NDT}	Reference Temperature Of Nil-Ductility Transition
RWCU	Reactor Water Cleanup
RWE	Rod Withdrawal Error
RWM	Rod Worth Minimizer
S_{alt}	EPU alternating stress intensity
S_m	Code allowable stress limit
SAFDL	Specified Acceptable Fuel Design Limit
SAR	Safety Analysis Report
SBO	Station Blackout
SCF	Scaling Factor for Stresses
SCF_P	Pressure Scaling Factor
SCF_T	Temperature Scaling Factor
SDC	Shutdown Cooling
SER	Safety Evaluation Report
SFP	Spent Fuel Pool
SFPAVS	Spent Fuel Pool Area Ventilation System
SGTS	Standby Gas Treatment System
SJAE	Steam Jet Air Ejector
SLCS	Standby Liquid Control System
SLMCPR	Safety Limit Minimum Critical Power Ratio

Term	Definition
SLO	Single-Loop Operation
SRLR	Supplemental Reload Licensing Report
SRM	Source Range Monitor
SRP	Standard Review Plan
SRV	Safety Relief Valve(s)
SRVDL	Safety Relief Valve Discharge Line
SSC	Structures, Systems, and Components
SSP	Supplemental Surveillance Capsule Program
SW	Plant Service Water
TAF	Top of Active Fuel
TAP	Torus Attached Piping
TAVS	Turbine Area Ventilation System
TBS	Turbine Bypass System
TLO	Two Recirculation Loop Operation
TSV	Turbine Stop Valve
TTNBB	Turbine Trip with Steam Bypass Failure
T_w	Time available
UHS	Ultimate Heat Sink
UPS	Uninterruptible Power Supply
USAR	Updated Safety Analysis Report
USE	Upper Shelf Energy

EXECUTIVE SUMMARY

This report summarizes the results of all significant safety evaluations performed that justify uprating the licensed thermal power at the Monticello Nuclear Generating Plant (Monticello), hereafter referred to as Monticello. The analysis provided herein, supports an increase of up to 2004 MWt from the current licensed reactor thermal power of 1775 MWt.

GEH has previously developed and implemented Extended Power Uprate (EPU). Based on EPU experience, GEH has developed an approach to uprate reactor power that maintains the current plant maximum normal operating reactor dome pressure. This approach is referred to as Constant Pressure Power Uprate and is contained in the Licensing Topical Report (LTR) NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," hereafter referred to as CLTR. The NRC approved the CLTR in the staff Safety Evaluation Report (SER) contained in the letter, William H. Ruland (NRC) to James F. Klapproth (GEH), "Review of GE Nuclear Energy Licensing Topical Report, NEDC-33004P, Revision 3, 'Constant Pressure Power Uprate' (TAC No. MB2510)," dated March 31, 2003, for Boiling Water Reactor (BWR) plants containing GEH fuel types and using GEH accident analysis methods. Monticello contains only GEH fuel types and this evaluation uses only GEH accident analysis methods. By performing the power uprate in accordance with the CLTR SER, the evaluation of the plant safety analyses and system performance is reduced, thus allowing for a more streamlined process.

This report provides systematic application of the CLTR approach to Monticello, including performance of plant specific engineering assessments and confirmation of the applicability of the CLTR generic assessments required to support an EPU.

It is not the intent of this report to explicitly address all the details of the analyses and evaluations described herein. For example, only previously NRC-approved or industry-accepted methods were used for the analyses of accidents and transients, as referred to in the CLTR. Therefore, the safety analysis methods have been previously addressed, and thus, are not explicitly addressed in this report. Also, event and analysis descriptions that are already provided in other licensing reports or the Updated Safety Analysis Report (USAR) are not repeated within this report. This report summarizes the significant evaluations needed to support a licensing amendment to allow for uprated power operation.

Uprating the power level of nuclear power plants can be done safely within plant specific limits and is a cost-effective way to increase installed electrical generating capacity. Many light water reactors have already been uprated worldwide, including many BWR plants.

An increase in the electrical output of a BWR plant is accomplished primarily by generating and supplying higher steam flow to the turbine generator. Monticello, as originally licensed, has an as-designed equipment and system capability to accommodate steam flow rates above the current rating. Also, the plant has sufficient design margins to allow the plant to be safely uprated significantly beyond its originally licensed power level.

A higher steam flow is achieved by increasing the reactor power along specified control rod and core flow lines. A limited number of operating parameters are changed, some setpoints are adjusted and instruments are recalibrated. Plant procedures are revised, and tests similar to many of the original startup tests are performed.

Detailed evaluations of the reactor, engineered safety features, power conversion, emergency power, support systems, and design basis accidents were performed. This report demonstrates that Monticello can safely operate at the requested EPU level. However, non-safety power generation modifications must be implemented in order to obtain the electrical power output associated with the uprate power. Until these modifications are completed, the non-safety, balance of plant equipment may limit the electrical power output, which in turn may limit the operating thermal power level to less than the rated thermal power (RTP) level.

The evaluations and reviews were conducted in accordance with the CLTR. The results of these evaluations and reviews are presented in the succeeding sections of this report:

- All safety aspects of Monticello that are affected by the increase in thermal power were evaluated;
- Evaluations were performed using NRC-approved or industry-accepted analysis methods;
- No changes, which require compliance with more recent industry codes and standards, are being requested;
- The USAR will be updated for the EPU related changes, after EPU is implemented, per the requirements in 10 CFR 50.71(e);
- Limited hardware modifications are required to meet safety requirements, and any modification to power generation equipment will be implemented per 10 CFR 50.59;
- Systems and components affected by EPU were reviewed to ensure there is no significant challenge to any safety system;
- A review of open, long-term, and programmatic NRC commitments for Monticello was performed. No plant unique commitments were identified that require special consideration for EPU; and
- Planned changes not yet implemented have also been reviewed for the effects of EPU.

1 INTRODUCTION

1.1 REPORT APPROACH

This report summarizes the results of all significant safety evaluations that were performed to justify uprating the licensed thermal power at Monticello. The analysis provided herein, supports an increase of up to 2004 MWt from the current licensed reactor thermal power of 1775 MWt.

GEH has previously developed and implemented Extended Power Uprate (EPU). Based on EPU experience, GEH has developed an approach to uprating reactor power that maintains the current plant maximum normal operating reactor dome pressure. This approach is referred to as Constant Pressure Power Uprate and is contained in the Licensing Topical Report (LTR) NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," (Reference 1) hereafter referred to as the "CLTR." The NRC approved the CLTR in the staff Safety Evaluation Report (SER) contained in Reference 1 for BWR plants containing GEH fuel types and using GEH accident analysis methods. Monticello contains only GEH fuel types and this evaluation uses only GEH accident analysis methods. By performing the power uprate in accordance with the CLTR and within the constraints of the NRC SER, the evaluation of the plant safety analyses and system performance is reduced, thus allowing for a more streamlined process.

This evaluation justifies an EPU to 2004 MWt, which corresponds to 120% of the original licensed thermal power (OLTP) for Monticello. This report is presented in the topical subject review sequence for EPU licensing reports as described in Section 3 of the US NRC, Office Of Nuclear Reactor Regulation, Review Standard for Extended Power Uprates, RS-001, December 2003.

Monticello was not originally licensed to the 10 CFR 50 Appendix A General Design Criteria (GDC), and the Regulatory Evaluations in RS-001 are not generally applicable. Therefore, a brief description of the licensing basis for the topic and a reference to the USAR section where the licensing basis is located is provided in each section. A declarative statement is provided regarding any change(s) to the licensing basis for EPU.

1.1.1 Generic Assessments

Many of the component, system, and performance evaluations contained within this report have been generically evaluated in the CLTR, and found to be acceptable. The plant specific applicability of these generic assessments is identified and confirmed in the applicable sections of this report. Generic assessments are those safety evaluations that can be dispositioned for a group or all BWR plants by:

- A bounding analysis for the limiting conditions,
- Demonstrating that there is a negligible effect due to EPU, or
- Demonstrating that the required plant cycle specific reload analyses are sufficient and appropriate for establishing the EPU licensing basis.

Bounding analyses may be based on either a demonstration that previous pressure increase power uprate assessments provided in Reference 2 or 3 (ELTR1 and ELTR2, respectively) are bounding or on specific generic studies provided in the CLTR. For these bounding analyses, the current EPU experience is provided in the CLTR along with the basis and results of the assessment. For those EPU assessments having a negligible effect, the current EPU experience plus a phenomenological discussion of the basis for the assessment is provided in the CLTR. For generic assessments that are fuel design dependent, the assessments are applicable to GEH / Global Nuclear Fuel LLC (GNF) fuel designs up through GE14, analyzed with GEH methodology.

Some of the safety evaluations affected by EPU are fuel cycle (reload) dependent. Reload dependent evaluations require that the reload fuel design, core loading pattern, and operational plan be established so that analyses can be performed to establish core operating limits. The reload analysis demonstrates that the core design for EPU meets the applicable NRC evaluation criteria and limits documented in Reference 4. [[

]] Therefore, the reload fuel design and core loading pattern dependent plant evaluations for EPU operation will be performed with the reload analysis as part of the standard reload licensing process. No plant can implement a power uprate unless the appropriate reload core analysis is performed and all criteria and limits documented in Reference 4 are satisfied. Otherwise, the plant would be in an unanalyzed condition. Based on current requirements, the reload analysis results are documented in the Supplemental Reload Licensing Report (SRLR), and the applicable core operating limits are documented in the plant specific Core Operating Limits Report (COLR).

1.1.2 Plant Specific Evaluation

Plant specific evaluations are assessments of the principal evaluations that are not addressed by the generic assessments described in Section 1.1.1. The relative effect of EPU on the plant specific evaluations and the methods used for their performance are provided in this report. Where applicable, the assessment methodology is referenced. If a specific computer code is used, the name of this computer code is provided in the subsection. If the computer code is identified in Reference 1, 2, 3, or 4, these documents may be referenced rather than the original report. Table 1-1 provides a summary of the computer codes used.

The plant specific evaluations performed and reported in this document use plant specific values to model the actual plant systems, transient response, and operating conditions. These plant specific analyses are considered reload independent and are performed using a conservative core representative of Monticello design for operation at 120% of OLTP for a cycle length of 24 months.

1.1.3 Report Generation and Review Process

GEH Scope

This Power Uprate Safety Analysis Report (PUSAR) represents several years of project planning activities, engineering analysis, technical verification, and technical customer review. The final stages of the PUSAR preparation include PUSAR integration, additional customer review, on-site review committee review, and submittal to NRC. The Monticello EPU project relied on the generic power uprate licensing topical reports (References 1, 3, and 4) submitted to and approved by NRC.

The project begins with the respective GEH and NSPM Project Managers creating a Project Work Plan (PWP). This PWP, developed in accordance with GEH engineering procedures, was used to define the plant specific work scope, inputs and outputs required for Project activities. A Division of Responsibility (DOR) between NSPM and GEH was used to further develop the work scope and assign responsible engineers (REs) from each organization. A Task Scoping Document (TSD) applicable for each GEH task was created, reviewed, and approved by NSPM prior to any technical work being performed. Each GEH task RE submitted a Design Input Request (DIR) to the NSPM task RE interface to define the correct plant information for use in the GEH task analysis and evaluation. Additional DIRs were submitted as the project continued. A plant specific PUSAR "shell" was created that contains the appropriate depth of information (but not the specifics) expected in the final PUSAR.

All pertinent information is captured in an individual task Design Record File (DRF) maintained by the GEH RE with oversight by the respective engineering manager. Each DRF contains the Quality Assurance records applicable to the task, including evidence of design verification.

A Draft Task Report (DTR) was created for every GEH task; the DTR includes a description of the analysis performed, inputs, methods, and results obtained, and includes input to the applicable PUSAR section(s). The DTR was design verified, in accordance with the GEH Quality Assurance Program, by a GEH technical verifier and a GEH Regulatory Services verifier, with oversight by the responsible GEH technical manager and GEH Project Manager. The DTR was transmitted by the GEH Project Manager to NSPM and reviewed by the NSPM RE and other NSPM engineers, as appropriate. Subsequent comments were resolved between the GEH and the NSPM REs and a Final Task Report (FTR) was developed. The FTR was again design verified (whether or not there were changes to the document), in accordance with the GEH Quality Assurance Program, by a GEH technical verifier and a GEH Regulatory Services verifier, with oversight by the responsible GEH technical manager and GEH Project Manager. The GEH Project Manager transmitted the FTR to the customer.

For the Monticello EPU, NSPM personnel:

1. Conducted multidisciplinary technical reviews of GEH evaluation reports (DTRs and FTRs) to ensure:
    i. Appropriate use of design inputs;
    ii. Consistency with the CLTR; and

iii. Design basis and licensing basis requirements were addressed.

2. Provided technical review results, in the form of detailed comments, to GEH performers;
3. Participated in discussions with GEH REs to address and resolve comments; and
4. Controlled the application of the NSPM control of off-site services process to GEH.

The Regulatory Services RE integrated the individual PUSAR sections creating a Draft PUSAR that was design verified, in accordance with the GEH Quality Assurance Program, by another GEH Regulatory Services engineer, with oversight by the GEH Regulatory Services Manager and the GEH Project Manager. The GEH Project Manager transmitted the verified Draft PUSAR to NSPM where it received another complete review by NSPM's technical personnel, project staff, and Licensing staff.

NSPM personnel generated questions and comments, which were responded to by GEH's technical and Regulatory Services personnel. The Draft PUSAR was revised and presented to the NSPM's review committee. Another round of comments and responses resulted in another revision to the Draft PUSAR. After a final round of comments and responses, the Final PUSAR was submitted to the NRC.

Technical assessment of GEH's work was performed during a review conducted at the GEH offices during March 19 & 20, 2007. Additional reviews were performed by accessing the GEH Design Record Files from the Monticello site June 18-20, 2007 and the week of December 10, 2007. The scope of these assessments included work performed by GE Hitachi Nuclear Energy, Global Nuclear Fuels (GNF), and GE Energy Services (GEES) in support of the Monticello EPU project. Participating in those activities were representatives of Monticello mechanical/structural, nuclear, system, and program engineering disciplines, and project engineering. The Monticello team reviewed selected design inputs, analysis methodologies, and results in the GEH Design Record Files. The reviews included discussion with GEH technical task performers to obtain a thorough understanding of GEH analysis methods.

In addition, NSPM retained a consultant to provide an independent technical assessment of task reports. This review was completed in December of 2007.

NSPM Scope

As noted in Section 1.1.3 above, a DOR between NSPM and GE was used to further develop the work scope and assign responsible engineers (REs) from each organization. Tasks assigned to NSPM responsible engineers were performed under the NSPM 10 CFR 50, Appendix B Quality Assurance Program. The NSPM assigned tasks were performed internally by NSPM engineers or contracted out to engineering consulting firms on the NSPM approved supplier list. Where applicable, the contractors applied a 10 CFR 50, Appendix B Quality Assurance Program.

NSPM internal tasks were prepared, reviewed, and approved in accordance with applicable procedure(s). A Task Scoping Document applicable for each task was created, reviewed, and approved by NSPM prior to any technical work being performed. This work scope formed the basis for the EPU task. The design inputs were then collected, reviewed, and forwarded to the engineering consultant, in accordance with applicable procedures.

Draft task reports were created that included a description of the analysis performed, inputs, methods, results obtained, and input to the applicable PUSAR section(s). NSPM engineering personnel, EPU Project personnel, and NSPM subject matter experts, as appropriate, reviewed the draft task report. An integrated set of comments on the draft task reports were forwarded for comment resolution and incorporation into the final task report. Appropriate information for NSPM tasks was captured in PassPort SharePoint files associated with each task. Final task reports when issued are processed through the NSPM Engineering Change process as a final verification of acceptability and retained as quality records in the NSPM nuclear records management system.

1.2 PURPOSE AND APPROACH

An increase in electrical output of a BWR is accomplished primarily by generation and supply of higher steam flow to the turbine generator. Most BWRs, as originally licensed, have an as-designed equipment and system capability to accommodate steam flow rates at least 5% above the original rating. In addition, continuing improvements in the analytical techniques (computer codes) based on several decades of BWR safety technology, plant performance feedback, operating experience, and improved fuel and core designs have resulted in a significant increase in the design and operating margin between the calculated safety analyses results and the current plant licensing limits. The available margins in calculated results, combined with the as-designed excess equipment, system, and component capabilities (1) have allowed many BWRs to increase their thermal power ratings by 5% without any Nuclear Steam Supply System (NSSS) hardware modification, and (2) provide for power increases up to 20% with some non-safety hardware modifications. These power increases involve no significant increase in the hazards presented by the plants as approved by the NRC at the original license stage.

The method for achieving higher power is to extend the Power to Flow map (Figure 1-1) along the Maximum Extended Load Line Limit Analysis (MELLLA). However, there is no increase in the maximum normal operating reactor vessel dome pressure or the maximum licensed core flow over their CLTP values. EPU operation does not involve increasing the maximum normal operating reactor vessel dome pressure, because the plant, after modifications to non-safety power generation equipment, has sufficient pressure control and turbine flow capabilities to control the inlet pressure conditions at the turbine.

1.2.1 Uprate Analysis Basis

Monticello is currently licensed at the 100% CLTP level of 1775 MWt. The EPU RTP level included in this evaluation is 120% of the OLTP (1670 MWt). Plant specific EPU parameters are listed in Table 1-2. The EPU safety analyses are based on a power level of 1.02 times the EPU power level unless the Regulatory Guide (RG) 1.49 two percent power factor is already accounted for in the analysis methods consistent with the methodology described in Reference 4, or RG 1.49 does not apply (e.g., Anticipated Transient Without Scram (ATWS) and Station Blackout (SBO) events).

1.2.2 Computer Codes

NRC-approved or industry-accepted computer codes and calculational techniques are used to demonstrate compliance with the applicable regulatory acceptance criteria. The application of these codes to the EPU analyses complies with the limitations, restrictions, and conditions specified in the approving NRC SER where applicable for each code. The limitations on use of these codes and methods as defined in the NRC staff position letter reprinted in ELTR2 were followed for this EPU analysis. Any exceptions to the use of the code or conditions of the applicable SER are noted in Table 1-1. The application of the computer codes in Table 1-1 is consistent with the current Monticello licensing basis except where noted in this report.

1.2.3 Approach

The planned approach to achieving the higher power level consists of changes to the Monticello licensing and design basis that support an increase of the licensed power level of up to 2004 MWt, consistent with the approach outlined in the CLTR, except as specifically noted in this report. Consistent with the CLTR, the following plant specific exclusions are exercised:

- No increase in maximum normal operating reactor dome pressure
- No increase to maximum licensed core flow
- No increase to currently licensed MELLLA upper boundary
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The plant specific evaluations are based on a review of plant design and operating data, as applicable, to confirm excess design capabilities; and, if necessary, identify required modifications associated with EPU. All changes to the plant licensing basis have been identified in this report. For specified topics, generic analyses and evaluations in the CLTR demonstrate plant operability and safety. The dispositions in the CLTR are based on a 20% of OLTP increase, which is the requested power increase for Monticello. For this increase in power, the conclusions of system/component acceptability stated in the CLTR are bounding and have been confirmed for Monticello. The scope and depth of the evaluation results provided herein are established based on the approach in the CLTR and unique features of the plant. The results of the following evaluations are presented in this report:

a) **Reactor Core and Fuel Performance:** Specific analyses required for EPU have been performed for a representative fuel cycle with the reactor core operating at EPU conditions. Specific core and fuel performance is evaluated for each operating cycle, and will continue to be evaluated and documented for the operating cycles that implement EPU.

b) **Reactor Coolant System and Connected Systems:** Evaluations of the NSSS components and systems have been performed at EPU conditions. These evaluations confirm the acceptability of the effects of the higher power and the associated change in process variables (i.e., increased steam and feedwater flows). Safety-related equipment performance is the primary focus in this report, but key aspects of reactor operational capability are also included.

c) **Engineered Safety Feature Systems:** The effects of EPU power operation on the Containment, Emergency Core Cooling System (ECCS), Standby Gas Treatment system and other Engineered Safety Features have been evaluated for key events. The evaluations include the containment responses during limiting Anticipated Operational Occurrences (AOOs) and special events, ECCS- Loss-Of-Coolant Accident (LOCA), and safety relief valve (SRV) containment dynamic loads.

d) **Control and Instrumentation:** The control and instrumentation signal ranges and analytical limits (ALs) for setpoints have been evaluated to establish the effects of the changes in various process parameters such as power, neutron flux, steam flow and feedwater (FW) flow. As required, setpoint evaluations have been performed to determine the need for any Technical Specification setpoint changes for various functions (e.g., main steam line high flow isolation setpoints).

e) **Electrical Power and Auxiliary Systems:** Evaluations have been performed to establish the operational capability of the plant electrical power and distribution systems and auxiliary systems to ensure that they are capable of supporting safe plant operation at the EPU power level.

f) **Power Conversion Systems:** Evaluations have been performed to establish the operational capability of various non-safety balance-of-plant (BOP) systems and components to ensure that they are capable of delivering the increased power output, and/or the modifications necessary to obtain full EPU power.

g) **Radwaste Systems and Radiation Sources:** The liquid and gaseous waste management systems have been evaluated at limiting conditions for EPU to show that applicable release limits continue to be met during operation at higher power. The radiological consequences have been evaluated for EPU to show that applicable regulations have been met for the EPU power conditions. This evaluation includes the effect of higher power level on source terms, on-site doses and off-site doses, during normal operation.

h) **Reactor Safety Performance Evaluations:** The limiting Updated Safety Analysis Report (USAR) analyses for design basis events have been addressed as part of the EPU evaluation. All limiting accidents, AOOs, and special events have been analyzed or generically dispositioned consistent with the CLTR and show continued compliance with regulatory requirements. [[

]]

i) **Additional Aspects of EPU:** High-energy line break (HELB) and environmental qualification evaluations have been performed at bounding conditions for EPU to show the continued operability of plant equipment under EPU conditions. The effects of EPU on the Monticello Individual Plant Examination (IPE) have been analyzed to demonstrate that there are no new vulnerabilities to severe accidents.

1.2.4 Concurrent Changes Unrelated to EPU

Consistent with the conditions and limitations on the use of the CLTR, NSPM is not requesting a concurrent review of any changes listed among the restrictions applicable to the CLTR.

1.3 EPU PLANT OPERATING CONDITIONS

1.3.1 Reactor Heat Balance

The operating pressure, the total core flow, and the coolant thermodynamic state characterize the thermal hydraulic performance of a BWR reactor core. The EPU values of these parameters are used to establish the steady state operating conditions and as initial and boundary conditions for the required safety analyses. The EPU values for these parameters are determined by performing heat (energy) balance calculations for the reactor system at EPU conditions.

The reactor heat balance relates the thermal-hydraulic parameters to the plant steam and FW flow conditions for the selected core thermal power level and operating pressure. Operational parameters from actual plant operation are considered (e.g., steam line pressure drop) when determining the expected EPU conditions. The thermal-hydraulic parameters define the conditions for evaluating the operation of the plant at EPU conditions. The thermal-hydraulic parameters obtained for the EPU conditions also define the steady state operating conditions for equipment evaluations. Heat balances at appropriately selected conditions define the initial and boundary conditions for plant safety analyses.

Figure 1-2 shows the EPU heat balance at 100% of EPU RTP and 100% rated core flow. Figure 1-3 shows the EPU heat balance at 102% of EPU RTP and 100% core flow.

Table 1-2 provides a summary of the reactor thermal-hydraulic parameters for the current rated and EPU conditions. At EPU conditions, the maximum nominal operating reactor vessel dome pressure is maintained at the current value, which minimizes the need for plant and licensing changes. With the increased steam flow and associated non-safety BOP modifications, the current dome pressure provides sufficient operating turbine inlet pressure to ensure good pressure control characteristics.

1.3.2 Reactor Performance Improvement Features

The reactor performance improvement features and the equipment allowed to be out-of-service (OOS) are listed in Table 1-2. When limiting, the input parameters related to the performance improvement features or the equipment OOS have been considered in the safety analyses for EPU, and as applicable, will be included in the reload core analyses. The use of these performance improvement features and allowing for equipment OOS are allowed during EPU operation. Where appropriate, the evaluations that are dependent upon cycle length are performed for EPU assuming a 24-month fuel cycle length.

1.4 SUMMARY AND CONCLUSIONS

This evaluation has covered an EPU to 120% of OLTP. The strategy for achieving higher power is to extend the MELLLA Power to Flow map region along the upper boundary extension.

The Monticello licensing bases have been reviewed to demonstrate how this uprate can be accommodated without a significant increase in the probability or consequences of an accident previously evaluated, without creating the possibility of a new or different kind of accident from any accident previously evaluated, and without exceeding any existing regulatory limits or design allowable limits applicable to the plant which might cause a reduction in a margin of safety. The EPU described herein involves no significant hazard consideration.

Table 1-1 Computer Codes Used For EPU

Task	Computer Code*	Version or Revision	NRC Approved	Comments
Nominal Reactor Heat Balance	ISCOR	09	(3)	NEDE-24011P Rev. 0 SER
Reactor Core and Fuel Performance	TGBLA PANACEA ISCOR	06 11 09	Y(2) Y(2) (3)	NEDE-30130-P-A NEDE-30130-P-A NEDE-24011P Rev. 0 SER
Thermal Hydraulic Stability	ODYSY TRACG OPRM	05 04 01	Y Y Y	NEDC-32992P-A, Class III, July 2001 NEDO-32465-A, Class I, August 1996
RPV Fluence	TGBLA DORTG	06 01	Y(2) Y	See notes 13 and 14
Reactor Internal Pressure Differences	ISCOR LAMB TRACG	09 07 02	(3) (4) (15)	NEDE-24011P Rev. 0 SER NEDE-20566-P-A NEDE-32176P, Rev. 2, Dec. 1999 NEDC-32177P, Rev. 2, Jan 2000 NRC TAC No M90270, Sep 1994
Transient Analysis	PANACEA ISCOR ODYN SAFER	11 09 10 04	Y (3) Y (6)	NEDE-30130-P-A (5) NEDE-24011P Rev. 0 SER NEDO-24154-A NEDC-32424P-A, NEDC-32523P-A, (9), (10) (11)
Anticipated Transient Without Scram	ODYN STEMP PANACEA	10 04 11	Y (7) Y	NEDE-24154P-A Supp. 1, Vol. 4 NEDE-30130-P-A
Containment System Response	SHEX M3CPT LAMB	06 05 08	Y Y (4)	(8) NEDO-10320, Apr. 1971 NEDE-20566-P-A September 1986
Appendix R Fire Protection	GESTR SAFER SHEX	08 04 06	(6) (6) Y	NEDE-23785-1-PA, Rev. 1 (9) (10) (11) (8)
Reactor Recirculation System	BILBO	04V	NA	(1) NEDE-23504, February 1977
ECCS-LOCA	LAMB GESTR SAFER ISCOR TASC	08 08 04 09 03A	Y Y Y (3) Y	NEDO-20566A NEDE-23785-1-PA, Rev. 1 (6) (9) (10) (11) NEDE-24011P Rev. 0 SER NEDC-32084P (12)
Station Blackout (SBO)	SHEX	06	Y	(8)
Fission Product Inventory	ORIGEN	2.1	N	Isotope Generation and Depletion Code
Plant Life	CHECWORKS™	2.1	N	Industry Standard

Task	Computer Code*	Version or Revision	NRC Approved	Comments
High Energy Line Break (HELB) Subcompartment Evaluation	GOTHIC	7.1	N	

* The application of these codes to the EPU analyses complies with the limitations, restrictions, and conditions specified in the approving NRC SER where applicable for each code. The application of the codes also complies with the SERs for the EPU programs.

(1) Not a safety analysis code that requires NRC approval. The code application is reviewed and approved by GEH for "Level-2" application and is part of GEH's standard design process. Also, the application of this code has been used in previous power uprate submittals.

(2) Letter, S.A. Richards (USNRC) to G. A. Watford (GEH), "Amendment 26 to GE Licensing Topical Report NEDE-24011-P-A, GESTAR II – Implementing Improved GE Steady State Methods (TAC No. MA6481)," November 10, 1999.

(3) The ISCOR code is not approved by name. However, the SER supporting approval of NEDE-24011P Rev. 0 by the May 12, 1978 letter from D. G. Eisenhut (NRC) to R. Gridley (GEH) finds the models and methods acceptable, and mentions the use of a digital computer code. The referenced digital computer code is ISCOR. The use of ISCOR to provide core thermal-hydraulic information in reactor internal pressure differences, Transient, ATWS, Stability, and LOCA applications is consistent with the approved models and methods.

(4) The LAMB code is approved for use in ECCS-LOCA applications (NEDE-20566P-A and NEDO-20566A), but no approving SER exists for the use of LAMB in the evaluation of reactor internal pressure differences or containment system response. The use of LAMB for these applications is consistent with the model description of NEDE-20566P-A.

(5) The physics code PANACEA provides inputs to the transient code ODYN. The improvements to PANACEA that were documented in NEDE-30130-P-A were incorporated into ODYN by way of Amendment 11 of GESTAR II (NEDE-24011-P-A). The use of TGBLA Version 06 and PANACEA Version 11 in this application was initiated following approval of Amendment 26 of GESTAR II by letter from S.A. Richards (NRC) to G.A. Watford (GEH) Subject: "Amendment 26 to GE Licensing Topical Report NEDE-24011-P-A, GESTAR II Implementing Improved GE Steady State Methods," (TAC NO. MA6481), November 10, 1999.

(6) The ECCS-LOCA codes are not explicitly approved for Transient or Appendix R usage. The staff concluded that SAFER is qualified as a code for best estimate modeling of loss-of-coolant accidents and loss of inventory events via the approval letter and evaluation for NEDE-23785P, Revision 1, Volume II. (Letter, C.O. Thomas (See NRC) to J.F. Quirk (GEH), "Review of NEDE-23785-1 (P), "GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident, Volumes I and II," August 29, 1983). In addition, the use of SAFER in the analysis of long term Loss-of-Feedwater (LOFW) events is specified in the

approved LTRs for power uprate: "Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate," NEDC-32424P-A, February 1999 and "Generic Evaluations of General Electric Boiling Water Reactor Extended Power Uprate," NEDC-32523P-A, February 2000. The Appendix R events are similar to the loss of FW and small break LOCA events.

(7) The STEMP code uses fundamental mass and energy conservation laws to calculate the suppression pool heatup. The use of STEMP was noted in NEDE-24222, "Assessment of BWR Mitigation of ATWS, Volume I & II (NUREG-0460 Alternate No. 3) December 1, 1979." The code has been used in ATWS applications since that time. There is no formal NRC review and approval of STEMP or the ATWS topical report.

(8) The NRC approved the application of the methodology in the SHEX code to containment response applications in the CLTR, (Reference 1, Section 4.1). The NRC approval of SHEX for containment analysis applications at Monticello is described in USNRC, Issuance of Amendment responding to Monticello Nuclear Generating Plant, License Amendment Request dated June 2, 2004, Revised Analysis of Long-Term Containment Response and Overpressure Required for Adequate NPSH for Low Pressure ECCS Pumps, (TAC No. MB7185), Amendment 139 to DPR-22.

(9) Letter, J.F. Klapproth (GEH) to USNRC, Transmittal of GE Proprietary Report NEDC-32950P "Compilation of Improvements to GENE's SAFER ECCS-LOCA Evaluation Model," dated January 2000 by letter dated January 27, 2000.

(10) Letter, S.A. Richards (NRC) to J.F. Klapproth, "General Electric Nuclear Energy (GENE) Topical Reports GENE (NEDC)-32950P and GENE (NEDC)-32084P Acceptability Review," May 24, 2000.

(11) "SAFER Model for Evaluation of Loss-of-Coolant Accidents for Jet Pump and Non-Jet Pump Plants," NEDE-30996P-A, General Electric Company, October 1987.

(12) The NRC approved the TASC-03A code by letter from S. A. Richards, NRC, to J. F. Klapproth, GE Nuclear Energy, Subject: "Review of NEDC-32084P, TASC-03A, A Computer Code for Transient Analysis of a Single Fuel Channel," TAC NO. MB0564, March 13, 2002. The acceptance version has not yet been published.

(13) CCC-543, "TORT-DORT Two-and Three-Dimensional Discrete Ordinates Transport Version 2.8.14," Radiation Shielding Information Center (RSIC), January 1994.

(14) Letter, H. N. Berkow (USNRC) to G. B. Stramback (GEH), "Final Safety Evaluation Regarding Removal of Methodology Limitations for NEDC-32983P-A, General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluations (TAC No. MC3788)," November 17, 2005.

(15) NRC has reviewed and accepted the TRACG application for the flow-induced loads on the core shroud as stated in NRC SER TAC No. M90270.

Table 1-2 Current and EPU Plant Operating Conditions

Parameter	Current Licensed Value [1.]	EPU Value
Thermal Power (MWt)	1775	2004
Vessel Steam Flow (Mlb/hr)	7.26	8.34
Full Power Core Flow Range		
Mlb/hr	47.5 to 60.5	57.0 to 60.5
% Rated	82.4 to 105.0	99.0 to 105.0
Maximum Normal Dome Pressure (psia)	1025	No Change
Maximum Normal Dome Temperature (°F)	547.6	No Change
Pressure at upstream side of turbine stop valve (TSV) (psia)	970	952
Full Power Feedwater		
Flow (Mlb/hr)	7.24	8.31
Temperature (°F)	383.0	395.8
Core Inlet Enthalpy (Btu/lb) [2]	523.7	523.0

Notes:

1. Based on current reactor heat balance.
2. At 100% core flow condition.

Currently licensed performance improvement features and/or equipment OOS that are included in EPU evaluations:

a) Maximum Extended Load Line Limit Analysis (MELLLA),

b) Single Loop Operation (SLO)

c) Three SRV OOS

d) 3% SRV Setpoint tolerance

e) Increased core flow (ICF)

f) APRM/RBM/Technical Specifications (ARTS)

Figure 1-1 Power to Flow Operating Map for EPU

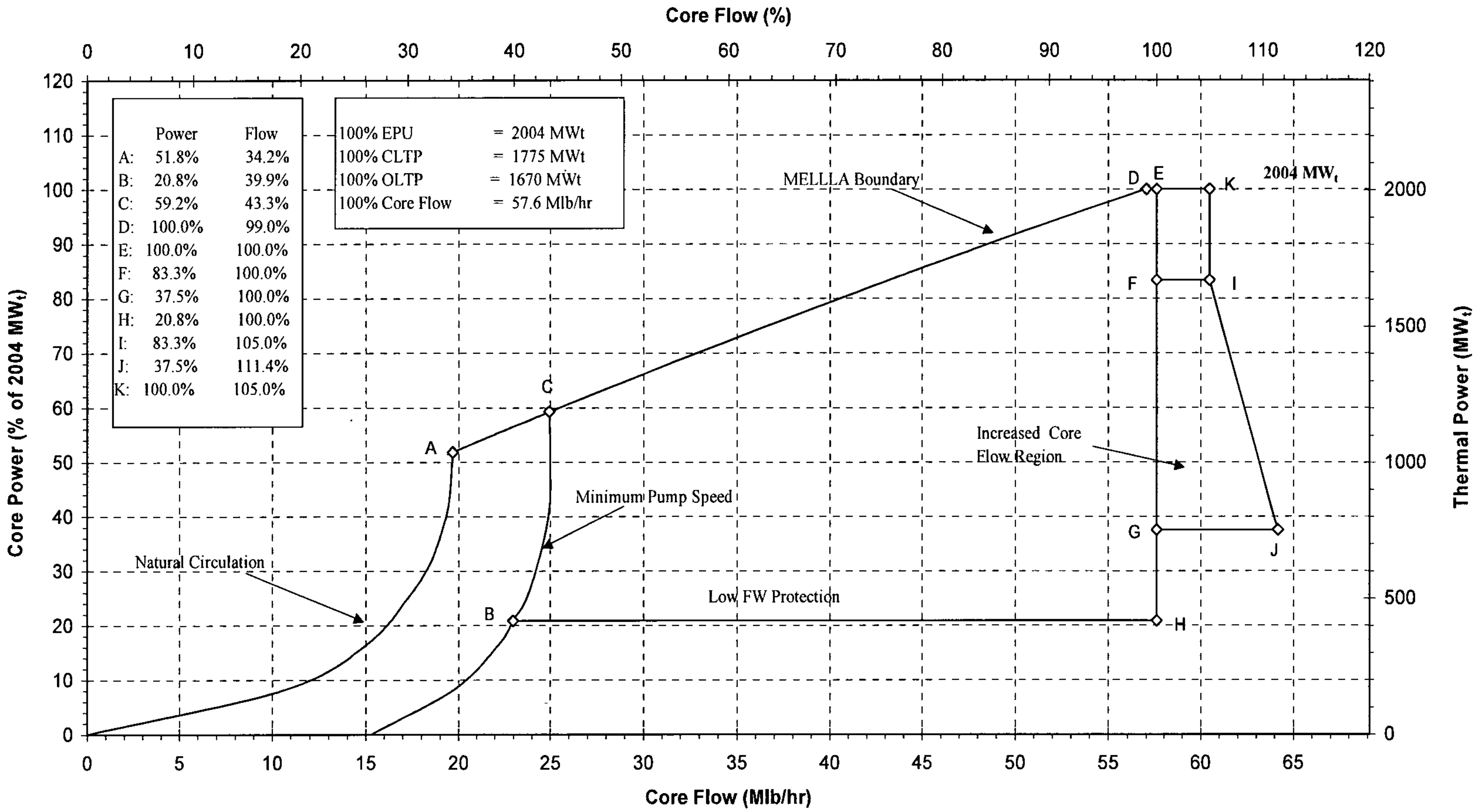
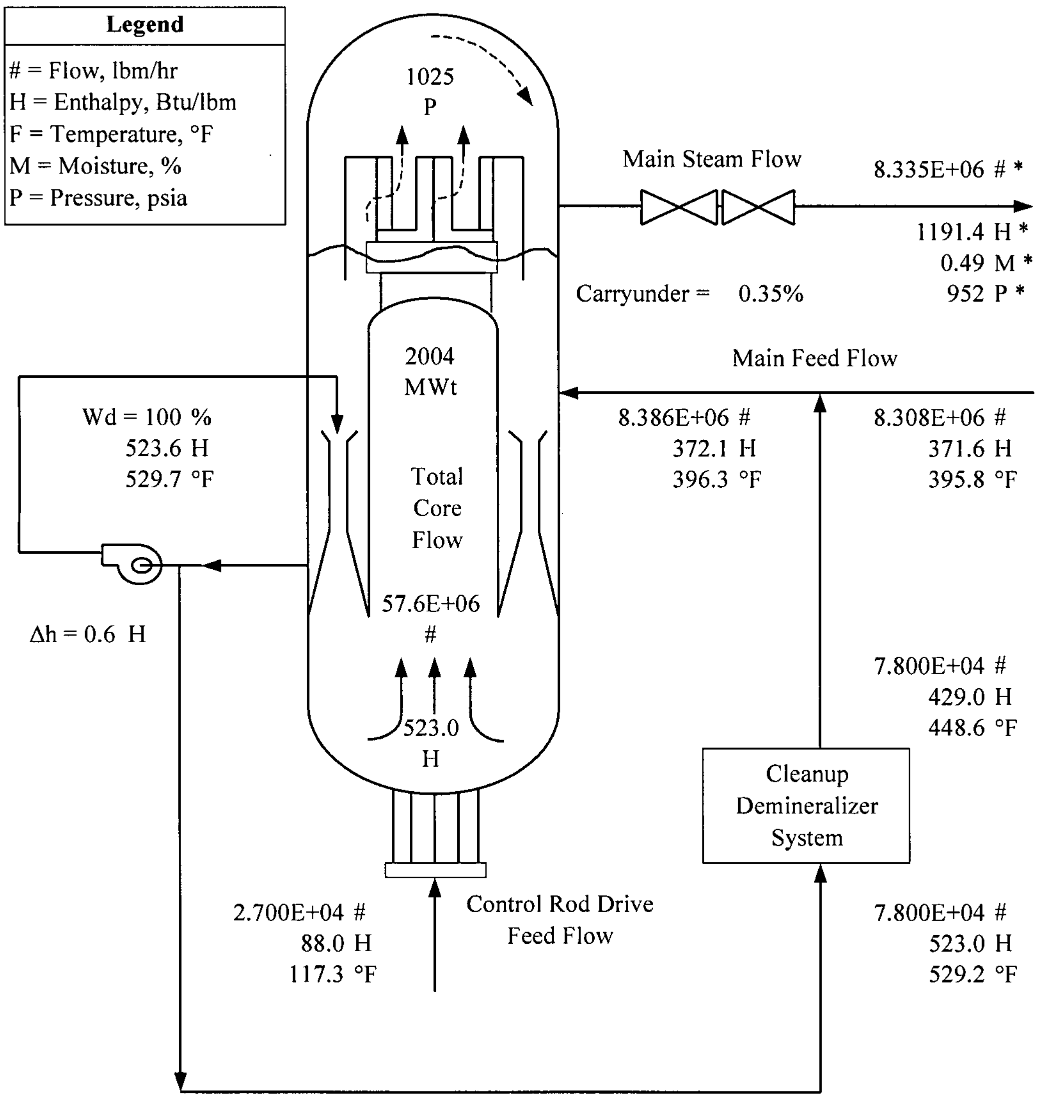


Figure 1-2 EPU Heat Balance – Nominal

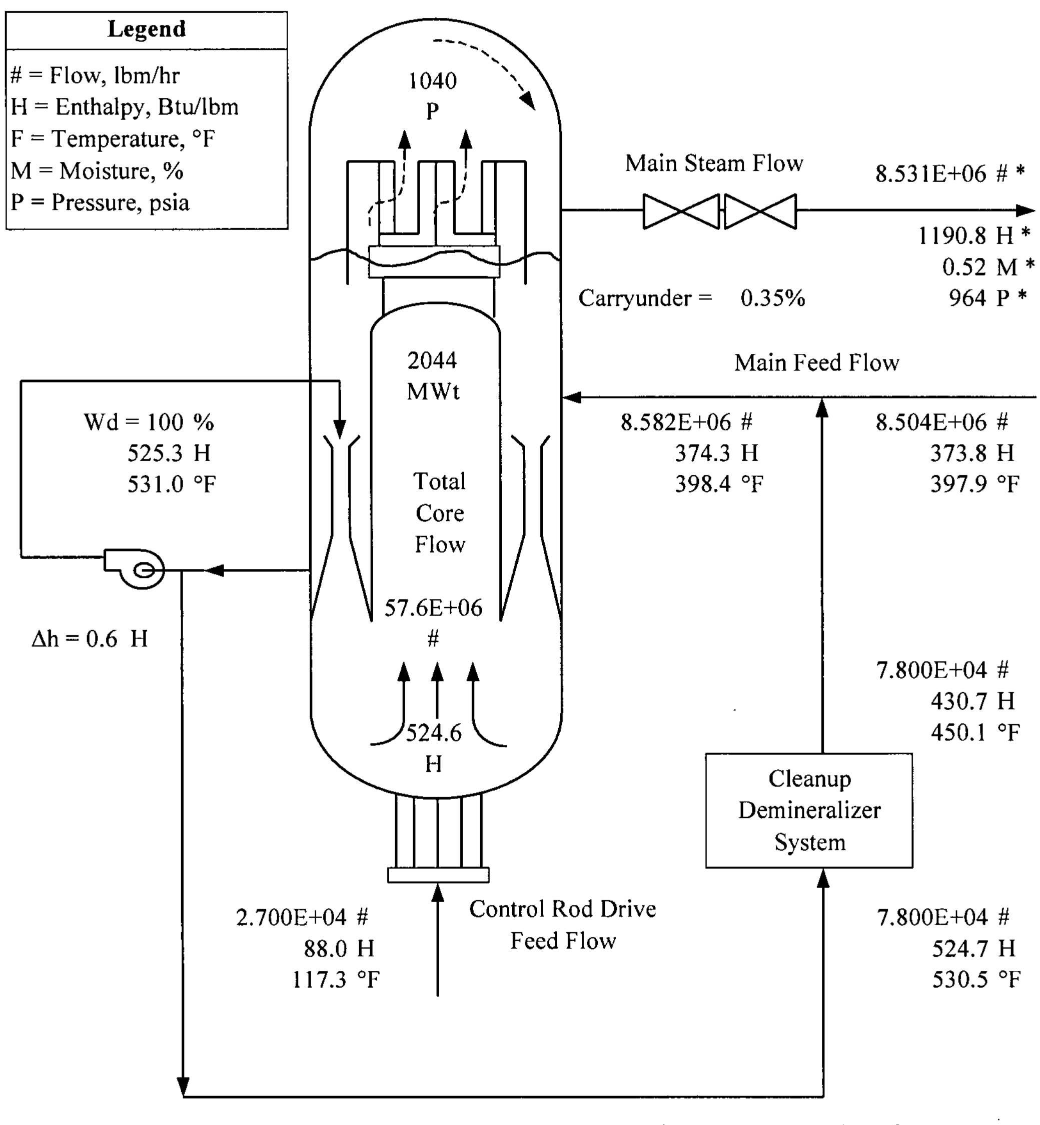
(@ 100% Power and 100% Core Flow)



*Conditions at upstream side of TSV

Core Thermal Power	2004.0
Pump Heating	3.9
Cleanup Losses	-2.1
Other System Losses	-0.8
Turbine Cycle Use	2005.0 MWt

Figure 1-3 EPU Heat Balance - Overpressure Protection Analysis
(@ 102% Power and 100% Core Flow)



Core Thermal Power	2044.0
Pump Heating	3.9
Cleanup Losses	-2.1
Other System Losses	-0.8
Turbine Cycle Use	2045.0 MWt

2 SAFETY EVALUATION

2.1 Materials and Chemical Engineering

2.1.1 Reactor Vessel Material Surveillance Program

Regulatory Evaluation

The reactor vessel material surveillance program provides a means for determining and monitoring the fracture toughness of the reactor vessel beltline materials to support analyses for ensuring the structural integrity of the ferritic components of the reactor vessel.

The NRC's acceptance criteria are based on (1) General Design Criterion (GDC)-14, insofar as it requires that the reactor coolant pressure boundary (RCPB) be designed, fabricated, erected, and tested so as to have an extremely low probability of rapidly propagating fracture; (2) GDC-31, insofar as it requires 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; (3) 10 CFR Part 50, Appendix H, which provides for monitoring changes in the fracture toughness properties of materials in the reactor vessel beltline region; and (4) 10 CFR 50.60, which requires compliance with the requirements of 10 CFR Part 50, Appendix H.

Specific NRC review criteria are contained in Standard Review Plan (SRP) Section 5.3.1 and other guidance provided in Matrix 1 of RS-001.

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC listed in the Regulatory

Evaluation above, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-9, draft GDC-33, draft GDC-34, and draft GDC-35.

The Reactor Vessel Material Surveillance Program is described in Monticello USAR Section 4.2, "Reactor Vessel," and the Bases to TS 3.4.9, "RCS Pressure and Temperature (P-T) Limits."

In addition to the evaluations described in the Monticello USAR, Monticello's systems and components were evaluated for License Renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Monticello Nuclear Generating Plant License Renewal Safety Evaluation Report (SER), NUREG-1865, dated October 2006 (Reference 5). The license renewal evaluation associated with the Reactor Vessel Material Surveillance Program is documented in NUREG-1865, Section 3.0.3.2.21.

Technical Evaluation

The CLTR, Section 3.2.1 describes the RPV fracture toughness evaluation process. RPV embrittlement is caused by neutron exposure of the wall adjacent to the core including the regions above and below the core that experience fluence greater than or equal to 1×10^{17} n/cm^2. This region is defined as the "beltline" region. Operation at the EPU conditions results in a higher neutron flux, which increases the integrated fluence over the period of plant life.

The surveillance program consists of three capsules. One capsule containing Charpy specimens was removed from the vessel after 7.08 EFPY of operation. One set of specimens was tested. A second set was re-encapsulated and placed in the Prairie Island RPV for accelerated irradiation and testing. The remaining two capsules have been in the reactor vessel since plant startup. One of these two capsules was removed during the refueling outage in 2007, after 26.5 EFPY of operation, and the other is classified as Standby. EPU has no effect on the existing surveillance schedule.

The maximum normal operating dome pressure for EPU is unchanged from that for original power operation. Therefore, the hydrostatic and leakage test pressures are acceptable for the EPU. Operation with EPU does not have an adverse effect on the reactor vessel fracture toughness because the vessel remains in compliance with the regulatory requirements as demonstrated in Section 2.1.2.

Conclusion

NSPM has evaluated the effects of the proposed EPU on the reactor vessel surveillance withdrawal schedule and has addressed changes in neutron fluence and their effects on the withdrawal schedule. The evaluation indicates that the material surveillance program continues to meet the requirements of 10 CFR Part 50, Appendix H, and 10 CFR 50.60, and will ensure continued compliance with the current licensing basis in this respect following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to the reactor vessel material surveillance program.

2.1.2 Pressure-Temperature Limits and Upper-Shelf Energy

Regulatory Evaluation

Pressure and Temperature (P-T) limits are established to ensure the structural integrity of the ferritic components of the RCPB during any condition of normal operation, including anticipated operational occurrences and hydrostatic tests.

The NRC's acceptance criteria for P-T limits are based on (1) GDC-14, 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; (2) GDC-31, insofar as it requires 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; (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 NRC review criteria are contained in SRP Section 5.3.2 and other guidance provided in Matrix 1 of RS-001.

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC listed in the Regulatory Evaluation above, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-9, draft GDC-33, draft GDC-34, and draft GDC-35.

The Pressure-Temperature Limits and Upper Shelf Energy is described in Monticello USAR Section 4.2.3.2, "Fracture Toughness of Reactor Pressure Vessel," and the Bases to TS 3.4.9, "RCS Pressure and Temperature (P/T) Limits."

In addition to the evaluations described in the Monticello USAR, Monticello's systems and components were evaluated for License Renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Monticello Nuclear Generating Plant License Renewal Safety Evaluation Report (SER), NUREG-1865, dated October 2006 (Reference 5). The license renewal evaluation associated with the Pressure Temperature Limits and Upper Shelf Energy is documented in NUREG-1865, Section 4.2.

Technical Evaluation

The neutron fluence for EPU is calculated using 2-dimensional neutron transport theory. The neutron transport methodology is consistent with RG 1.190. The revised fluence is used to evaluate the vessel against the requirements of 10CFR50, Appendix G. The results of these evaluations indicate that:

(a) The USE remains bounded by the BWROG equivalent margin analysis, thereby demonstrating compliance with Appendix G.

(b) The beltline material reference temperature of the nil-ductility transition (RT_{NDT}) remains below 200°F.

(c) The current P-T curves will be revised considering the increase in shifts affecting the beltline portion of the curves. The hydrotest pressure for EPU is 1010 psig.

(d) The 54 effective full power year (EFPY) shift is increased, and consequently, requires a change in the adjusted reference temperature, which is the initial RT_{NDT} plus the shift. These values are provided in Table 2.1-1.

(e) The 54 EFPY beltline circumferential weld material RT_{NDT} remains bounded by the requirements of BWRVIP-74, "BWR Reactor Pressure Vessel Inspection and Flaw Evaluation Guidelines." This comparison is provided in Table 2.1-2.

Conclusion

NSPM has evaluated the effects of the proposed EPU on the P-T limits for the plant and addressed changes in neutron fluence and their effects on the P-T limits. Revised P-T curves have been generated and will be submitted per 10 CFR 50.90 consistent with the guidance of the GE CLTR as a separate license amendment request.

2.1.3 Reactor Internal and Core Support Materials

Regulatory Evaluation

The reactor internals and core supports 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's acceptance criteria for reactor internal and core support materials are based on GDC-1 and 10 CFR 50.55a for material specifications, controls on welding, and inspection of reactor internals and core supports.

Specific NRC review criteria are contained in SRP Section 4.5.2 and Boiling Water Reactor Vessel and Internals Project (BWRVIP)-26.

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC listed in the Regulatory Evaluation above, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-1 and draft GDC-5.

The Reactor Internal and Core Support Materials are described in Monticello USAR Section 3.6, "Other Reactor Vessel Internals."

In addition to the evaluations described in the Monticello USAR, Monticello's systems and components were evaluated for License Renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Monticello Nuclear Generating Plant License Renewal Safety Evaluation Report (SER), NUREG-1865, dated October 2006 (Reference 5). The license renewal evaluation associated with the Reactor Internal and Core Support Materials is documented in NUREG-1865, Section 3.0.3.2.11.

Technical Evaluation

The reactor internal and core support materials evaluation includes the materials' specifications and mechanical properties, welds, weld controls, nondestructive examination procedures, corrosion resistance, and susceptibility to degradation of the reactor internals and core supports include structures, systems, and components (SSCs) that perform safety functions or whose failure could affect safety functions performed by other SSCs. None of these requirements, specifications, or controls is changed as a result of the EPU; therefore, these continue to be acceptable.

Conclusion

NSPM has evaluated the effects of the proposed EPU on the integrity of reactor internal and core support materials. The evaluation indicates that the reactor internal and core support materials will continue to be acceptable and will continue to meet the requirements of the current licensing basis and 10 CFR 50.55a. Therefore, the proposed EPU is acceptable with respect to reactor internal and core support materials.

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's acceptance criteria for RCPB materials are based on (1) 10 CFR 50.55a and GDC-1, 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-4, 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; (3) GDC-14, 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; (4) GDC-31, insofar as it requires 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, which specifies fracture toughness requirements for ferritic components of the RCPB.

Specific NRC review criteria are contained in SRP Section 5.2.3 and other guidance provided in Matrix 1 of RS-001. Additional review guidance for primary water stress-corrosion cracking of dissimilar metal welds and associated inspection programs is contained in Generic Letter (GL)

97-01, Information Notice 00-17, Bulletins 01-01, 02-01, and 02-02. Additional review guidance for thermal embrittlement of cast austenitic stainless steel components is contained in a letter from C. Grimes, NRC, to D. Walters, Nuclear Energy Institute, dated May 19, 2000.

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC listed in the Regulatory Evaluation above, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-1, draft GDC-5, draft GDC-9, draft GDC-33, draft GDC-34, draft GDC-35, draft GDC-40, and draft GDC-42.

The Reactor Coolant Pressure Boundary Materials is described in Monticello USAR Sections 4.2, "Reactor Vessel," and 4.3, "Recirculation System."

In addition to the evaluations described in the Monticello USAR, Monticello's systems and components were evaluated for License Renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Monticello Nuclear Generating Plant License Renewal Safety Evaluation Report (SER), NUREG-1865, dated October 2006 (Reference 5). The license renewal evaluation associated with the reactor coolant pressure boundary is documented in NUREG-1865, Sections 2.3.1, 3.0.3.2.24, and 4.3.

<u>Technical Evaluation</u>

The peak reactor vessel fluence increases as a result of EPU. This increase in fluence can create the potential for additional irradiation-assisted stress corrosion cracking (IASCC). To address this potential, Monticello has a procedurally controlled program for the augmented nondestructive examination (NDE) of selected RPV 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 in the adjacent base material. Monticello belongs to and has implemented the BWR Vessel and Internals Project (BWRVIP) augmented inspection program for reactor internals. The inspection strategies recommended by the BWRVIP-25, 26 and 47 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 and Enforcement Bulletins, BWRVIP documents, and recommendations provided by General Electric Service Information Letters (GE SILs). 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
- Vessel ID attachment welds
- Instrumentation penetrations
- Steam dryer

Fluence calculations performed at EPU conditions indicate that only the top guide and shroud exceed the 5E20 n/cm^2 threshold value for IASCC. The core plate fluence was calculated to be 4.43E20 n/cm^2 and as such remains beneath the IASCC threshold. Incore instrumentation dry tubes and guide tubes are included in the evaluation due to an existing identification as being susceptible to IASCC in BWRVIP-47.

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

The service life of most equipment is not affected by EPU. [[

]] The current inspection strategy for the reactor internal components is adequate to manage any potential effects of EPU.

Conclusion

NSPM has evaluated the effects of the proposed EPU on the integrity of RCPB materials. The evaluation indicates that the RCPB materials will continue to be acceptable following implementation of the proposed EPU and will continue to meet the requirements of the current licensing basis, 10 CFR Part 50, Appendix G, and 10 CFR 50.55a. Therefore, the proposed EPU is acceptable with respect to RCPB materials.

2.1.5 Protective Coating Systems (Paints) - Organic Materials

Regulatory Evaluation

Protective coating systems (paints) provide a means for protecting the surfaces of facilities and equipment from corrosion and contamination from radionuclides and also provide wear protection during plant operation and maintenance activities.

The NRC's acceptance criteria for protective coating systems are based on (1) 10 CFR Part 50, Appendix B, which states quality assurance requirements for the design, fabrication, and construction of safety-related SSCs and (2) Regulatory Guide 1.54, Revision 1, for guidance on application and performance monitoring of coatings in nuclear power plants.

Specific NRC review criteria are contained in SRP Section 6.1.2.

Monticello Current Licensing Basis

NSPM's current licensing basis regarding coatings is described in NSP's letter to the NRC dated November 11, 1998, "Response to Generic Letter 98-04: Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System After a Loss-of-Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment."

Technical Evaluation

The protective coating systems used inside the containment were evaluated for their continued suitability for and stability under design-basis loss-of-coolant accident conditions, considering radiation and chemical effects at the EPU conditions. The post LOCA containment environmental conditions temperature, pressure, and radiation, (see Sections 2.6.1, 2.6.3, 2.6.4, 2.8.5, and 2.9.2) do not significantly change as a result of EPU and the chemical constituency does not change at all. Therefore the containment protective coating systems remain acceptable for EPU operation.

Conclusion

NSPM has evaluated the effects of the proposed EPU on the protective coatings. The evaluation indicates that the protective coatings will continue to be acceptable following implementation of the proposed EPU and will continue to meet the requirements of 10 CFR Part 50, Appendix B. Therefore, the proposed EPU is acceptable with respect to protective coatings 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- 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 velocity of flow, fluid temperature, steam quality, oxygen content, and pH. During plant operation, control of these parameters is limited and the optimum conditions for minimizing FAC effects, in most cases, cannot be achieved. Loss of material by FAC will, therefore, occur.

The licensee's FAC program is based on NUREG-1344, GL 89-08, and the guidelines in Electric Power Research Institute (EPRI) Report NSAC-202L-R2. It consists of predicting loss of material using the CHECWORKS™ computer code, and 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.

Monticello Current Licensing Basis

The Monticello program for addressing Flow Accelerated Corrosion is described in a Northern States Power letter, dated July 24, 1989, that provided the response to Generic Letter 89-08, "Erosion/Corrosion Induced Pipe Wall Thinning." This response provided information regarding administrative controls, procedures, and engineering activities associated with this program.

The FAC program was also evaluated for License Renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Monticello Nuclear Generating Plant License Renewal Safety Evaluation Report (SER), NUREG-1865, dated October 2006 (Reference 5). The license renewal evaluation associated with the FAC program is documented in NUREG-1865, Sections 3.0.3.1.2 and 3.2.2.1.1.

Technical Evaluation

Monticello implements a flow accelerated corrosion (FAC) program that uses selective component inspections to provide a measure of confidence in the condition of systems susceptible to FAC. These selective inspections are the basis for qualifying components that are not inspected for further service. This approach is based upon program guidelines developed by EPRI using ASME material allowables. The criteria for selecting components for inspection after the EPU will be the same as used under CLTP. In addition to this long-term monitoring program, selected piping replacements have been performed to maintain suitable design margins. FAC resistant replacement materials are used to mitigate future occurrences of FAC as part of the modification process.

A CHECWORKS™ FAC model (in accordance with the CHECWORKS™ FAC users guide and EPRI modeling guidelines) has been developed for Monticello to predict the FAC wear rate (single and two-phase fluids) and the remaining service life for each piping component. As a minimum, the controlled 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.

Variables that influence FAC include:

- Moisture content
- Water chemistry
- Temperature
- Oxygen
- Flow path geometry and velocity
- Material composition

Monticello has predicted EPU system operating conditions that will be used as inputs to the CHECWORKS™ FAC model. Implementation of EPU will affect some of the variables that influence FAC. However, they are expected to remain within the CHECWORKS™ FAC model parameter bounds. For example, final feedwater dissolved oxygen level is currently modeled at approximately 65 ppb and the CHECWORKS™ FAC model allowable input range for this parameter is 0 – 1,000 ppb. Selected portions of some system piping are predicted to have temperature increases of approximately 15 °F and velocity increases of approximately 20%.

The Monticello CHECWORKS™ FAC model is capable of accepting these EPU related parameter changes. Based on experience at CLTP operating conditions and previous FAC modeling results, it is anticipated that the EPU operating conditions may result in the need for additional FAC monitoring points. The CHECWORKS™ FAC modeling techniques allow for the identification of additional monitoring points required for EPU. The CHECWORKS™ FAC program targets FAC susceptible piping and components. The modification process includes the installation of FAC resistant material as appropriate.

Table 2.1-3 compares key parameter values (CLTP and EPU) affecting FAC.

The increased MS and FW flow rates at EPU conditions do not significantly affect the potential for FAC in these systems. Therefore, the Monticello program for FAC is adequate to manage any potential effects of the EPU on NSSS, turbine generator, and BOP components. The reactor internals inspection and FAC programs do not significantly change for EPU. In addition, the Maintenance Rule provides oversight for the other mechanical and electrical components, important to plant safety, to guard against age-related degradation.

In response to Generic Letter 89-13 a periodic non-destructive examination program was established to inspect safety-related piping and heat exchangers at known or suspected high corrosion, biofouling, or silt buildup areas. This program is supplemented by visual inspections of opened piping and heat exchangers whenever possible.

The descriptions of the BWRVIP, FAC, and GL 89-13 Programs that are included in USAR Appendix K, "Renewed Operating License – USAR Supplement," regarding Aging Management Programs remain applicable for EPU.

Conclusion

NSPM has evaluated the effect of the proposed EPU on the FAC analysis for the plant and has addressed changes in the plant operating conditions on the FAC analysis. The evaluation indicates that the updated analyses will predict the loss of material by FAC and will ensure timely repair or replacement of degraded components following implementation of the proposed EPU. Therefore, the proposed EPU is 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 system comprise the RCPB.

The NRC's acceptance criteria for the RWCU system are based on (1) GDC-14, 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; (2) GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents; and (3) GDC-61, insofar as it requires that systems that contain radioactivity be designed with appropriate confinement.

Specific NRC review criteria are contained in SRP Section 5.4.8.

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC listed in the Regulatory Evaluation above, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-9, draft GDC-33, draft GDC-67, draft GDC-68, draft GDC-69, and draft GDC-70.

The Reactor Water Cleanup System is described in Monticello USAR Section 10.2.3, "Reactor Cleanup Demineralizer System."

In addition to the evaluations described in the Monticello USAR, Monticello's systems and components were evaluated for License Renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Monticello Nuclear Generating Plant License Renewal Safety Evaluation Report (SER), NUREG-1865, dated October 2006 (Reference 5). The license renewal evaluation associated with the Reactor Water Cleanup System is documented in NUREG-1865, Section 2.3.3.15. Management of aging effects on the Reactor Water Cleanup System is documented in NUREG-1865, Section 3.3.2.3.15.

Technical Evaluation

RWCU system operation at the EPU RTP level slightly decreases the temperature (< 1°F) within the RWCU system. This system is designed to remove solid and dissolved impurities from recirculated reactor coolant, thereby reducing the concentration of radioactive and corrosive species in the reactor coolant. The system is capable of performing this function at the EPU RTP level.

RWCU flow is usually selected to be in the range of 0.8% to 1.0% of FW flow based on operational history. The existing RWCU flow slightly exceeds this range (1.08% of FW flow). The RWCU flow analyzed for EPU is within this range. Furthermore, the EPU review included evaluation of water chemistry, heat exchanger performance, pump performance, flow control valve capability, and filter / demineralizer performance. Performance of each was found to be within the design of RWCU system at the analyzed flow. The RWCU analysis concludes:

- There is negligible heat load effect.
- A small increase (≈15%) in filter / demineralizer backwash frequency occurs, but this is within the capacity of the Radwaste system.
- The slight changes in operating system conditions result from a decrease in inlet temperature and increase in FW system operating pressure.
- The RWCU filter / demineralizer control valves may operate in a slightly more open position to compensate for the increased FW pressure. These valves do not have position indication, preventing quantification of this change. However, there are two

valves and each valve is designed to provide a flow rate of 0 to 100 gpm. Typically total RWCU flow is divided equally through each valve.

- No changes to instrumentation are required for EPU, and no setpoint changes are expected due to the negligible system process parameter changes.

Previous operating experience has shown that the FW iron input to the reactor increases for EPU as a result of the increased FW flow. This predicts an increase in the typical reactor water iron concentration from < 1.7 ppb to < 2.0 ppb. However, this change is considered insignificant, and does not affect RWCU.

The effects of EPU on the RWCU system functional capability have been reviewed, and the system can perform adequately at EPU RTP with the original RWCU system flow. Using the original RWCU system flow at EPU RTP results in a slight increase in the calculated reactor water conductivity (from 0.100 μS/cm to 0.115 μS/cm) because of the increase in FW flow. The current reactor water conductivity limits are unchanged for EPU and the actual conductivity remains within these limits.

Table 2.1-4 shows that the changes in RWCU system operating conditions are small. The system flow rate is unchanged.

The reactor water iron and conductivity parameters at Monticello are maintained well below the EPRI BWRVIP-130: BWR Water Chemistry Guidelines – 2004 Revision guidelines for these parameters. Table 2.1-5 shows typical values for these parameters based on CLTP five year monthly averages. The estimated EPU values are included. This estimated increase is proportional to the RWCU System flow capacity as a percentage of feedwater flow at EPU conditions. No credit is assumed for passive removal mechanisms such as source term reduction.

Table 2.1-5 shows that the estimated increase in these parameters is not significant and that sufficient operating margin to the conservative limits remains under EPU conditions.

The increase in FW line pressure has a slight effect on the system operating conditions. The effect of this increase is included in Section 2.6.1.3 Containment Isolation.

Conclusion

NSPM has evaluated the effects of the proposed EPU on the RWCU system. The evaluation indicates that the RWCU system will continue to be acceptable following implementation of the proposed EPU and will continue to meet the requirements of the current licensing basis. Therefore, the proposed EPU is acceptable with respect to the RWCU system.

Table 2.1-1 Adjusted Reference Temperatures

60-Year License (54 EFPY)

Lower Shell

Thickness in inches = 5.06

54 EFPY Peak I.D. fluence = 4.39E+18 n/cm^2
54 EFPY Peak 1/4 T fluence = 3.24E+18 n/cm^2

Lower-Intermediate Shell and All Welds

Thickness in inches= 5.06

54 EFPY Peak I.D. fluence = 6.43E+18 n/cm^2
54 EFPY Peak 1/4 T fluence = 4.75E+18 n/cm^2

N2 Nozzle

Thickness in inches= 5.06

54 EFPY Peak I.D. fluence = 9.32E+17 n/cm^2
54 EFPY Peak 1/4 T fluence = 6.88E+17 n/cm^2

COMPONENT	HEAT	%Cu	%Ni	CF	Adjusted CF	Initial RTndt °F	1/4 T Fluence n/cm^2	54 EFPY Δ RTndt °F	σ_I	σ_Δ	Margin °F	54 EFPY Shift °F	54 EFPY ART °F
PLATES:													
Lower-Intermediate													
1-14	C2220-1	0.17	0.65	131		27	4.75E+18	104	0	17	34	138	165
1-15	C2220-2	0.17	0.65	131		27	4.75E+18	104	0	17	34	138	165
Lower													
1-16	A0946-1	0.14	0.56	100		27	3.24E+18	69	0	17	34	103	130
1-17	C2193-1	0.17	0.50	121		0	3.24E+18	83	0	17	34	117	117
WELDS:													
Limiting	SMAW	0.10	0.99	138.5		-65.6	4.75E+18	110	12.7	28	61	171	106
NOZZLES:													
N2 (1)	E21VW	0.18	0.86	141.9		40	6.88E+17	49	0	17	34	83	123
INTEGRATED SURVEILLANCE PROGRAM (2):													
Plate (3)	C2220	0.17	0.64	128		27	4.75E+18	101	0	17	34	135	162
Weld (4)	5P6756	0.06	0.93	82		-65.6	4.75E+18	65	12.7	28	61	126	61

(1) In the absence of Cu data for this nozzle, 0.18% is based upon heats of materials used for beltline nozzles at other plants. The mean from nine nozzles (0.119) plus one standard deviation (0.0617) was used to determine the value of 0.18%.
CMTR data for the ten (10) Monticello N2 nozzles was averaged to determine the Ni content.
CMTR data for the ten (10) Monticello N2 nozzles was used to determine the initial RT_{NDT}.

(2) Procedures defined in RG1.99 Rev 2 are applied to determine the ART considering the Integrated Surveillance Program.

(3) The ISP plate is the identical heat and is presented using the ISP chemistry with the vessel plate Initial RT_{NDT} and fluence. As defined in RG1.99, as there is only one set of capsule data available for this material, the CF is obtained from RG1.99 Position 1.1.

(4) The ISP weld is not the identical heat and is presented using the ISP chemistry and CF applied to the limiting Monticello weld.
The CF is not adjusted as defined by RG1.99.

Table 2.1-2 54 EFPY Effects of Irradiation on RPV Circumferential Weld Properties

Group	CB&I * 64 EFPY	Monticello 54 EFPY
Cu%	0.10	0.10
Ni%	0.99	0.99
CF (See Note 1)	134.9	138.5
Fluence at clad/weld interface (10^{19} n/cm^2)	1.02	0.64
ΔRT_{NDT} w/o margin (°F)	135.6	121
$RT_{NDT(U)}$ (°F)	-65	-65.6
Mean RT_{NDT} (°F)	70.6	55.8
P (F/E) NRC (See Note 2)	1.78E-05	(Note 3)

* This column represents the limits as defined in BWRVIP-74.

Notes:

[1] The value of 109.5 originally presented in BWRVIP-05 was corrected to 134.9 in the SE Supplement dated March 7, 2000.

[2] P (F/E) stands for "Probability of a failure event".

[3] Although a conditional failure probability has not been calculated, the Monticello values at the end of license are less than the 64 EFPY value provided by the NRC and therefore it can be concluded that the Monticello RPV P (F/E) is bounded by the NRC analysis, consistent with the guidelines of BWRVIP-74 and Appendix A of BWRVIP.-74

Table 2.1-3 FAC Parameter Comparison for EPU

Parameter	Allowable Input[1]	CLTP Typical Range of Values	EPU Typical Range of Values	Comments
Steam Flow (lbm/hr)	1 - 100,000,000	160,000 to 7,240,000	210,000 to 8,510,000	Values Within Allowable
Velocity (ft/sec)	Calculated in program	130 to 183	145 to 188	Values Within Allowable
Steam Quality (%)	0 to 100	63 to 99.8	65 to 99.7	Values Within Allowable
Operating Temperature (°F)	0 to 750	176 to 542	183 to 540	Values Within Allowable

1. Allowable Input ranges from CHECWORKS™ FAC model users guide (Version 2.1)

Table 2.1-4 RWCU System Parameter Comparison for EPU

RWCU System Parameter	CLTP	EPU
RWCU Inlet Temperature, °F	530.2	529.7
RWCU Inlet Pressure (RPV dome pressure, neglecting head), psig	1010	1010
RWCU Outlet Temperature, °F	449.2	448.6
RWCU Outlet Pressure (at the feedwater line), psig	1045	1057
Design RWCU Flow, lbm/hr	80,000	80,000
Maximum RWCU Flow, lbm/hr	85,000	85,000

Table 2.1-5 Estimated EPU Effect on Reactor Water Parameters

Parameter	Units	CLTP Value*	Estimated EPU Value	Operating Guideline/ Limit	Projected EPU Margin
Conductivity	μS/cm	0.1	0.115	0.3 μS/cm (BWR Action L1 Guideline) ≤ 5μS/cm (Technical Requirements Manual)	0.185 μS/cm
Iron	ppb	< 1.7 ppb (typically < 1.0 ppb)	< 2.0 ppb	< 5 ppb (BWR Diagnostic)	~ 3.0 ppb
* Based on data from 5 year monthly averages					

2.2 Mechanical and Civil Engineering

2.2.1 Pipe Rupture Locations and Associated Dynamic Effects

Regulatory Evaluation

SSCs important to safety could be affected by the pipe-whip dynamic effects of a pipe rupture.

The NRC's acceptance criteria are based on GDC-4, which requires SSCs important to safety to be designed to accommodate the dynamic effects of a postulated pipe rupture.

Specific NRC review criteria are contained in SRP Section 3.6.2.

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC listed in the Regulatory Evaluation above, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-40 and draft GDC-42.

Monticello Pipe Rupture Locations and Associated Dynamic Effects are described in Monticello USAR Section 12.2, "Plant Principal Structures and Foundations," 12.2.4, "High Energy Line Failures Outside Containment," and Appendix I, "Evaluation of High Energy Line Breaks Outside Containment."

Technical Evaluation

No changes to the implementation of the existing criteria for defining pipe break and crack locations and configurations are being made for EPU and no new break or crack locations are postulated as a result of EPU.

No changes to the implementation of the existing criteria dealing with special features, such as augmented in-service inspection programs or the use of special protective devices such as pipe-whip restraints are being made for EPU.

Pipe-whip analyses results are provided in Section 2.2.2

The adequacies of supports relative to pipe whip and jet impingement loads are provided in Section 2.2.2.

High Energy Line Breaks (HELB)

EPU has no effect on the steam pressure or enthalpy at the postulated break locations. Therefore, EPU has no effect on the mass and energy releases from an HELB in a steam line. Therefore, no plant specific evaluation is required for steam line breaks. The results of the Monticello evaluation of liquid line breaks are provided in Table 2.2-1.

Changes in Methods of Analysis

The results provided for HELB events affected by EPU, specifically, the liquid line breaks in the Feedwater, Condensate, and RWCU systems show much larger changes than would be expected due to the small changes in pump discharge pressures and small enthalpy changes as a result of EPU. The results are driven by conservative changes in analysis methods resulting from corrective actions underway to perform HELB analysis upgrades at Monticello.

Comparison of results from CLTP to EPU conditions

Because of these changes in methodology, a comparison of the results between EPU and CLTP conditions shows a significantly larger change than would normally be expected based on the small changes in process fluid temperatures and enthalpy resulting from EPU based on previous industry experience.

Monticello has chosen not to perform a full re-analysis of these specific liquid line HELBs at CLTP conditions because it was determined that our effort should be focused on completing the corrective actions using bounding conditions. Thus, a detailed breakdown of the magnitude of the change is caused by EPU versus the change resulting from the changes in methods and correction of errors is not provided. 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.

2.2.1.1 Steam Line Breaks

Steam Line HELB

In accordance with the CLTR and NRC SER, a Constant Pressure EPU has no effect on the steam pressure or enthalpy at the postulated break locations. Therefore, EPU has no effect on the mass and energy releases from an HELB in a steam line. Therefore, no plant-specific evaluation is required for steam line breaks to support EPU.

However, because of the changes in methodology described above, Monticello has determined that the steam line HELB analyses also need to be upgraded. A full re-baselining of steam line HELB analyses is being performed under the corrective action process at Monticello.

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[illegible]	[illegible]	[illegible]
		]]

Main Steam Line Breaks

As a Constant Pressure EPU, there is no increase in reactor dome pressure and enthalpy conditions. These conditions are also assumed to exist at the postulated break locations in the Main Steam system. Therefore, EPU has no effect on the mass and energy releases from a HELB in a MSL.

HPCI Steam Line Breaks

As a Constant Pressure EPU, there is no increase in the reactor dome pressure at EPU. The pressure and enthalpy conditions at break and crack locations in the HPCI steam system are unchanged. Therefore the CLTP analysis of the HPCI steam line breaks is bounding for EPU conditions.

RCIC Steam Line Breaks

As a Constant Pressure EPU, there is no increase in the reactor dome pressure at EPU. The pressure and enthalpy conditions at break and crack locations in the RCIC steam system are unchanged. Therefore, the CLTP analysis of the RCIC steam line breaks is bounding for EPU conditions.

The steam line HELB events in the Monticello licensing basis were evaluated [[ ]].

2.2.1.2 Liquid Line Breaks

Operation at EPU conditions requires an increase in the MS and FW flow rates, which results in a slight increase in downcomer subcooling. This increase in subcooling may lead to increased break flow rates for liquid line breaks. The mass and energy releases for HELBs in the RWCU, FW, Condensate, CRD, Standby Liquid Control, and Zinc Injection System (GEZIP) systems and instrument and sample lines may be affected by EPU and were re-evaluated at EPU conditions. [[ ]] evaluations of liquid line breaks have been performed at EPU conditions. The results of the Monticello evaluation of liquid line breaks are provided in Table 2.2-1.

RWCU Line Breaks

Monticello has undertaken a complete re-analysis of HELB's in the reactor building updating break and crack analysis originally performed in response to the guidelines of the letter from A. Giambusso, Deputy Director for Reactor Projects, to NSP, Consequences of Piping Failures Outside of the Containment Structure, December 18, 1972 (Giambusso Letter), ISEQ # RAC00159, MO2943-0075, now incorporated as Appendix B to Branch Technical Position SPLB 3-1 in NUREG-0800 Standard Review Plan Section 3.6.1. During the 6.3 percent rerate in 1996, only one new case was reanalyzed at CLTP for the RWCU system – a break in the system suction piping at the outboard isolation valve. For this reason a detailed comparison of CLTP and EPU results for HELBs in the RWCU system is not possible. For the break location that was analyzed during rerate, 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% and an increase in integrated energy release of 63 percent.

The Reactor Building pressure, temperature and relative humidity profiles at EPU conditions were evaluated for the effect on equipment qualification as discussed in Sections 2.2.5 and 2.3.1.

Feedwater System Line Breaks

The mass and energy releases for double-ended breaks and critical cracks in the FW lines were re-analyzed at EPU conditions. At EPU peak break and crack mass releases, flow rates increase by up to 23 percent due to assumed increased pump discharge pressure and changes in fluid enthalpy based on the EPU BOP heat balance. Integrated mass flow does not change significantly due to conservative assumptions that basically drain the hotwells. Based on the small changes in fluid enthalpy, integrated energy release increases for breaks by about four percent. The Reactor Building and turbine building pressure, temperature, flooding, and relative humidity profiles at EPU conditions were evaluated for the effect on equipment qualification as discussed in Sections 2.2.5 and 2.3.1.

Condensate System Line Breaks

The mass and energy releases for double-ended breaks and critical cracks in the high energy portions of condensate lines were re-analyzed at EPU conditions. At EPU peak mass releases flow rates increase by up to 28 percent due to conservative assumptions of increased pump discharge pressure and changes in fluid enthalpy based on the EPU heat balance. Integrated mass flow for breaks does not change significantly due to conservative assumptions that basically drain the hotwells. Based on the small changes in fluid enthalpy, integrated energy release for breaks is essentially unchanged. Based on increased flow rates and energy releases assumed for critical cracks results show increases in integrated mass release of 20 percent and on integrated energy releases up to 4 percent[1]. The Turbine Building pressure, temperature and relative humidity profiles at EPU conditions were evaluated for the effect on equipment qualification as discussed in Sections 2.2.5 and 2.3.1.

CRD System Line Breaks

Within the CRD system, only a small portion of the CRD return line to the RWCU return line is high-energy piping. Due to its location break or crack in this 3" piping is indistinguishable from a break or crack in the 3" RWCU piping in this area. A separate Mass and Energy Calculation for the CRD system is not necessary.

Standby Liquid Control System Breaks

Breaks or cracks in SLCS involve an isolated section of piping between two check valves in SLCS. The results of this break do not create harsh environmental conditions for temperature, pressure or liquid flooding level. It is a fixed volume and the current analysis assumed the entire contents of this volume are vented into the room in 1 second. The enthalpy of the release was assumed constant at 536.8 BTU/lbm. This remains bounding with respect to reactor coolant conditions at EPU.

GEZIP (Zinc Injection)

The portions of the GEZIP system with postulated HELBs are small piping (1 ½") located near the Feedwater and Condensate lines in the Turbine Building. Due to the proximity to the postulated Feedwater and Condensate Systems HELBs, any adverse effects from a GEZIP HELB would be enveloped by a postulated HELB on one of the Feedwater or Condensate lines. Because pathways to safe shutdown have been demonstrated for the Feedwater and Condensate postulated HELBs, these same pathways would exist for the much smaller GEZIP HELB. The Feedwater and Condensate system HELBs will remain bounding at EPU conditions.

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[1] One scenario evaluated had a long isolation time of 8 hours and resulted in significantly larger release of energy over a longer time, however, this case is not limiting due to its small mass flow rate. This scenario was omitted from the evaluation of EPU effects because it is a methods change and not an EPU effect.

Offgas System

The [[ ]] for steam lines in 2.2.1.1 was not used for the Offgas System. For convenience, the effect of EPU on both steam and liquid lines in the offgas system is included here.

A small portion of the Offgas System was excluded from consideration for HELB under the 2% criterion. EPU will not change that basis for exemption.

The steam supply line for the Offgas system is located in the condenser bay area, and the SJAE room in the Turbine Building. The effects of postulated breaks on this line in the condenser area are bounded by the postulated breaks on other lines such as Feedwater and Main Steam. There is no safe shutdown equipment in the SJAE room to be affected by a pipe whip or jet impingement. Postulated breaks on the steam piping are bounded by other postulated HELBs, for which paths to safe shutdown have been demonstrated. Because EPU will not increase steam pressure, this basis for exemption will not change.

Other high-energy lines in the Offgas System are located in the Offgas Storage and Compressor Building, in which no safe shutdown equipment is located. Because the occurrence of a HELB in this area can have no effect on systems required to mitigate accidents, no further evaluations need to be made. There will be no modifications or changes in safe-shutdown equipment at EPU conditions that would affect this conclusion.

Instrument and Sample Lines

The instrument sensing lines from the primary cooling system and reactor vessel to the instrumentation represent high-energy lines for the portion of the instrument sensing line routing outside of the Primary Containment. Because these lines are 1 inch or smaller in nominal size, neither circumferential breaks nor longitudinal breaks are required to be postulated. In addition, these lines are equipped with excess flow check valves that will mitigate any break within a few seconds of the break occurring. The basis for exempting these lines from further consideration will remain valid at EPU conditions.

Instrument and sample lines, which tee from other high-energy lines, are identified and evaluated with the specific system.

Conclusion

NSPM has evaluated the effects of the proposed EPU on rupture locations and associated dynamic effects. The evaluation indicates that SSCs important to safety will continue to meet the requirements of the current licensing basis with respect to the dynamic effects of a postulated pipe rupture following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to the dynamic effects associated with the postulated rupture of piping.

2.2.2 Pressure-Retaining Components and Component Supports

Regulatory Evaluation

The NRC's acceptance criteria are based on (1) 10 CFR 50.55a and GDC-1, 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, 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, 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, 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, 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.

Specific NRC 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 Matrix 2 of RS-001.

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC listed in the Regulatory Evaluation above, with the exception of current GDC-15, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-1, draft GDC-2, draft GDC-5, draft GDC-9, draft GDC-33, draft GDC-40, and draft GDC-42. There is no draft GDC directly associated with current GDC-15. Current GDC-15 is applicable to Monticello for certain events as described in USAR Section 14.4.

The Pressure Retaining Components and Component Supports are described in Monticello USAR Section 3.6, "Other Reactor Vessel Internals," and Section 12, "Plant Structures and Shielding."

In addition to the evaluations described in the Monticello USAR, Monticello's systems and components were evaluated for License Renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Monticello Nuclear Generating Plant License Renewal Safety Evaluation Report (SER), NUREG-1865, dated October 2006 (Reference 5). Systems and system components, programs used to manage aging effects, and time limited aging analyses are documented in NUREG-1865, Sections 2.3, 3.1, 3.2, 3.3, 3.4, 3.5, and 4.3.

Technical Evaluation

Flow Induced Vibration (FIV)

The FIV evaluation addresses the influence of an increase in flow during EPU on reactor coolant pressure boundary (RCPB) piping and RCPB piping components. The topics addressed in this evaluation are:

- Structural Evaluation of Recirculation Piping [[ ]]
- Structural Evaluation of MS and FW Piping [[ ]]
- Safety-Related Thermowells and Probes

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[illegible]	[illegible]	[illegible]
		]]

Key applicable structures include the Main Steam (MS) system piping and suspension, the FW system piping and suspension, and the RRS piping and suspension. In addition, branch lines attached to the MS system piping or FW system piping are considered.

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The MS piping and the FW piping have increased flow rates and flow velocities under EPU conditions. As a result, the MS and FW piping experience increased vibration levels, approximately 50 percent for a 20 percent power uprate. The ASME Code and nuclear regulatory guidelines require vibration test data be taken and evaluated for these high energy piping systems during initial operation at EPU conditions. Vibration data for the MS and FW piping inside containment will be acquired using remote sensors. A piping vibration startup test program, which meets the ASME Code and regulatory requirements, will be performed.

The FIV effect on the MS, FW, and RRS piping inside containment at Monticello is confirmed to be consistent with the generic description provided in the CLTR [[

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The effects of FIV induced stresses at EPU conditions on safety-related thermowells in the MS and FW system and the sample probe in the FW system were evaluated. Although the non-dimensional "reduced velocity" parameters have increased for EPU compared to CLTP conditions for these three components, the FIV induced zero-to-peak stress at the root of the MS and FW thermowells and at the root of the FW sample probe are well within their endurance limits.

The ratios of the vortex shedding frequency and the structural natural frequency are greater than the CLTP value for the MS thermowell. If the estimated ratio of the shedding frequency to the natural frequency is kept below the maximum value estimated for the CLTP conditions, resonance due to vortex shedding should not occur. The length of the MS thermowell will be reduced in order to increase the natural frequency so that the estimated ratio of the vortex shedding frequency to the natural frequency is no greater than the maximum CLTP value. The ratio of vortex shedding frequency to the natural frequency for the FW thermowell and FW sample probe remain acceptable under EPU conditions.

Reactor Coolant Pressure Boundary Structural (Non-FIV)

The Reactor Coolant Pressure Boundary (RCPB) piping systems evaluation consists of a number of safety-related piping subsystems that move fluid through the reactor and other safety systems.

The flow, pressure, temperature, and mechanical loading for most of the RCPB piping systems do not increase for EPU. Consequently, there are no changes in stress.

The following systems are [[ ]] dispositioned in the CLTR as unaffected by EPU:

- RRS
- Control Rod Drive System
- SLCS injection line
- RPV Bottom Head Drain line

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[illegible]	[illegible]	[illegible]

[illegible]	[illegible]	[illegible]
		]]

The piping systems described above are [[ ]] the temperature, pressure, flow rate, and mechanical loading are unchanged for the Monticello EPU.

Section 2.8.4.2 demonstrates that the RCPB piping remains below the ASME pressure limit during the most severe pressurization transient.

IGSCC (NRC Generic Letter 88-01 and NUREG-0313, Rev. 2)

The augmented inspection program for intergranular stress corrosion cracking (IGSCC), as addressed in NRC Generic Letter 88-01 and NUREG-0313, Rev. 2, has been resolved by Monticello's pipe replacement program wherein all susceptible material was replaced with resistant material. All welds are therefore classified as IGSCC Category "A". In accordance with EPRI TR-112657, piping welds identified as Category "A" are considered resistant to IGSCC, and as such are assigned a low failure potential provided no other damage mechanisms are present. Examination criteria for these welds will be in accordance with the RI-ISI process.

2.2.2.1 Pipe Whip and Jet Impingement

Pipe whip and jet impingement loads resulting from high energy pipe breaks are directly proportional to system pressure. Because EPU conditions do not result in an increase in the pressure considered in the high-energy piping evaluations, there is no increased pipe whip or jet impingement loads on HELB targets or pipe whip restraints. Additionally, a review of pipe stress calculations determined that the FW temperature increases associated with EPU conditions will not result in pipe stress levels above the thresholds required for postulating HELBs, except at locations already evaluated for breaks. As a result, EPU conditions do not result in new HELB locations, nor affect existing HELB evaluations of pipe whip restraints and jet targets.

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.

Main Steam and Associated Piping System Evaluation

The CLTR, Section 3.5.1, requires a plant specific evaluation for the MS and FW piping because the MS and FW piping and associated branch piping up to the first anchor or support experience an increase in flow, pressure and/or temperature due to EPU, resulting in an increase in stress

The MS piping system and associated branch piping (inside containment) were evaluated for compliance with the ASME Section III, Division I, 1977 Edition with Addenda up to and including Winter 1978 Piping Code stress criteria, including the effects of EPU on piping stresses, piping supports including the associated building structure, piping interfaces with the RPV nozzles, containment penetrations, flanges, and valves.

Because the MS piping pressures and temperatures are not affected by EPU, there is no effect on the analyses for these parameters. Seismic inertia loads, seismic building displacement loads, and SRV discharge loads are not affected by EPU, thus, there is no effect on the analyses for these load cases. The increase in MS flow results in increased forces from the turbine stop valve closure (TSVC) transient. The turbine stop valve closure loads bound the MSIV closure loads because the MSIV closure time is significantly longer than the stop valve closure time (CLTR). Due to the magnitude of the TSVC transient load increase, the transient event was reanalyzed. The main steam piping was then reanalyzed using this revised load definition.

Pipe Stresses

The results of the Main Steam system piping analysis indicate that piping load changes do not result in load limits being exceeded for the MS system and attached branch piping (SRVDL, Reactor Core Isolation Cooling (RCIC), High Pressure Coolant Injection (HPCI), RPV Vent, and MSIV Drain) or for the RPV nozzles. The design analyses demonstrates that the calculated stresses meet ASME Section III, Division I, 1977 Edition with Addenda up to and including Winter 1978 Piping Code allowable limits to justify operation at EPU conditions except for one small bore branch line that did not meet displacement criteria. Additional detailed analysis will be performed to qualify this line or the piping modified prior to operation at EPU conditions. No new postulated break locations were identified.

Pipe Supports

The MS piping (inside containment) was evaluated for the effects of flow increase on the piping snubbers, hangers and struts. A review of the increase in MS flow associated with EPU indicates that piping load changes do not result in any load limit being exceeded. Similarly, the drywell steel was evaluated for changes in loading and determined to be adequate for EPU.

Feedwater and Associated Piping System Evaluation

The RCPB portion of FW was evaluated for changes in temperature, pressure and flow. The FW CLTP analyzed temperatures and pressures envelope the EPU conditions. Therefore FW inside containment and associated attached lines outside containment including HPCI, RCIC and Reactor Water Cleanup are acceptable for EPU conditions. FW between the second and third isolation valves outside containment and associated attached lines, HPCI, RCIC and RWCU are not part of the RCPB boundary; however, they were evaluated with the RCPB because they are addressed in the same analytical model as the RCPB FW piping.

Main Steam Isolation Valves

The inboard MSIVs have been evaluated, as discussed in Section 4.7 of ELTR2, Supplement 1. The evaluation covers both the effects of the changes to the structural capability of the inboard MSIV to meet pressure boundary requirements, and the potential effects of EPU-related changes to the safety functions of the inboard MSIVs. The generic evaluation from ELTR2 is based on (1) a 20% thermal power increase, (2) an increased operating dome pressure to 1095 psia, (3) a reactor temperature increase to 556°F, and (4) steam and FW increases of about 24%. The evaluation from ELTR2 is confirmed applicable to Monticello inboard MSIVs. An increase in flow rate assists inboard MSIV closure, which results in a slightly faster inboard MSIV closure time. The Monticello inboard MSIV has design features that ensure the MSIV closure time is not reduced below the stroke time limit. The closing time of the inboard MSIVs is controlled by the design of the hydraulic control valves and the function of the hydraulic damper or dashpot. Prior to EPU implementation, the hydraulic control valve of the inboard MSIV will be adjusted for the required closing time. The solenoid valves for the inboard MSIVs will be replaced with valves designed to function with a lower differential pressure to atmosphere. The inboard MSIVs are designed for 1250 psig at 575°F. Hence, the pressure integrity of the valves is not affected by operation at EPU conditions. The inboard MSIV is designed to close at normal closing time of 3 to 10 seconds at 200% of original design steam flow. The flow restrictor cross-sectional area controls the maximum steam flowrate (choke flow). The flow restrictors for EPU are not being changed or modified, so the maximum flow rate will not change for EPU. Therefore, the normal closing time of 3 to 10 seconds is not affected for EPU. Therefore, the Monticello EPU is bounded by conclusions of the evaluation in Section 4.7 of ELTR2, and the inboard MSIVs are acceptable for EPU operation.

The outboard MSIVs are a double disc gate valve type with air/spring actuator to close.

The EPU conditions for the outboard MSIV analysis is 1040 psia dome pressure with steam at saturated conditions. The steam flow associated with this condition is 2.133 Mlbm/hr per outboard MSIV. These conditions provide the most limiting conditions of operation for EPU. The outboard MSIVs are designed for 1250 psig at 575°F. Hence, the pressure integrity of the valves is not affected by operation at EPU conditions.

The outboard MSIVs are designed to close with a maximum differential pressure of 1000 psid. The maximum differential pressure for outboard MSIV closure is an outside steam line break. The differential pressure associated with this break for EPU operation is below 1000 psia and thus valve closure is assured for EPU.

Therefore, the outboard MSIVs are acceptable for EPU operation.

Feedwater Evaluation

The FW system outside containment from the isolation valve to the motor operated valve (MOV) downstream of high pressure heaters was evaluated for compliance with the ANSI-B31.1-1977, including Winter 1978 Addenda Power Piping Code stress criteria, and for the effects of deadweight, pressure, seismic and thermal expansion displacements on the piping snubbers, hangers, and struts. Piping interfaces with penetrations, flanges, equipment nozzles, and valves were also evaluated. The remaining FW piping from the MOVs to the condensate pumps will be modified as a result of the replacement of the feedwater and condensate pumps, and will be qualified for full EPU operation as part of the modification. The current piping and associated components are adequate for operation within the capability of the existing feedwater and condensate pumps.

Pipe Stresses

A review of the small increases in pressure, temperature and flow associated with EPU indicates that the EPU temperature, pressure and flow conditions are bounded by the existing analyses. The original design analyses have sufficient design margin between calculated stresses and ANSI-B31.1-1977, including Winter 1978 Addenda Code allowable limits to justify operation at EPU conditions.

Therefore, EPU does not have an adverse effect on the FW piping design. No new postulated pipe break locations were identified.

Pipe Supports

The FW system was evaluated for the effects of seismic, deadweight and thermal expansion displacements on the piping snubbers, hangers, and struts. A review of the increases in temperature and FW flow associated with EPU indicates that the EPU conditions are bounded by the existing analyses.

Other RCPB Piping Evaluation

This section addresses the adequacy of the other RCPB piping designs, for operation at the EPU conditions. The nominal operating pressure and temperature of the reactor are not changed by EPU. Aside from MS and FW, no other system connected to the RCPB experiences an increased flow rate at EPU conditions. Only minor changes to fluid conditions are experienced by these systems due to higher steam flow from the reactor and the subsequent change in fluid conditions within the reactor.

Additionally, dynamic piping loads for SRV discharge at EPU conditions are unchanged from those used in the existing analyses. The increase in MS flow results in increased forces from the turbine stop valve closure (TSVC) transient. These effects have been evaluated for the RCPB piping, including attached lines and SRV discharge piping, as required.

Balance-of-Plant Piping (BOP) systems

Other Balance-of-Plant Piping (BOP) system evaluations consist of a number of piping subsystems that move fluid through systems outside the RCPB piping.

The flow, pressure, temperature, and mechanical loading for some BOP piping systems do not increase for EPU. Consequently, there is no change in stress evaluations and these BOP piping systems are generically evaluated in the CLTR.

The following BOP piping systems at Monticello [[
]] the flow, temperature, pressure, and other mechanical loads do not change for the Monticello EPU:

- HPCI (except Torus and FW connection, see Tables 2.2-2a and 2.2-2b);
- RCIC (except Torus and FW connection, see Tables 2.2-2a and 2.2-2b).
- Neutron Monitoring Piping
- Containment Air Monitoring Piping (except for Torus Attached Piping (TAP) sections, see Table 2.2-2d)

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[illegible]	[illegible]	[illegible]

[illegible]	[illegible]	[illegible]
		]]

Per the CLTR, [[ ]] is required for BOP Piping systems where the loads and temperatures used in the analyses are dependent on the containment hydrodynamic loads and short/long term temperature evaluation results (Section 2.6.1). Bounding hydrodynamic loads and short/long term torus/suppression pool temperatures due to a design basis loss-of-coolant accident (LOCA) were defined for CLTP. The bounding hydrodynamic loads did not change for EPU. However, the short/long term suppression pool temperature results change for EPU.

Large bore and small bore ANSI B31.1, Class I and Class II piping and supports [[ ]] were evaluated for acceptability at EPU conditions. The evaluation of the BOP piping and supports was performed in a manner similar to the evaluation of RCPB piping systems and supports, using applicable B31.1 Power Piping Code and/or ASME Boiler and Pressure Vessel Code – Section III, Division I equations. The Codes of Record (as referenced in the appropriate calculations) code allowable values, and analytical techniques were used and no new assumptions were introduced.

The Design Basis Accident (DBA)-LOCA dynamic loads, including the pool swell loads, vent thrust loads, condensation oscillation (CO) loads, and chugging loads were originally defined and evaluated for Monticello. The structures attached to the torus shell such as piping systems, vent penetrations, and valves are based on these DBA LOCA hydrodynamic loads. Because dynamic loads do not change for EPU, there are no resulting effects on the torus shell attached structures.

The effects of the EPU conditions have been evaluated for the following piping systems:

- MS (outside containment)
- Turbine Bypass Line
- Extraction Steam, Heater Vents and Drains
- FW and Condensate – (Piping from the condensate pump to the feedwater flow control valve will be qualified as part of FW and Condensate Pump Modifications)
- RWCU - Outside Containment
- RHR - Outside Containment
- RHR Service Water
- CS - Outside Containment
- HPCI – Outside Containment
- RCIC – Outside Containment
- SLCS - Outside Containment
- Service Water
- CRD
- Emergency Service Water
- Reactor Building Closed Cooling Water
- Spent Fuel Cooling
- Circulating Water
- Off Gas
- Torus Attached Piping Including ECCS Suction Strainers
- Post Accident Sampling
- Cross Around Piping
- Cross Around Relief Valve (CARV) Piping
- Moisture Separator Drain lines
- Hard Piped Vent

Operation at the EPU conditions increases stresses on piping and piping system components on some BOP piping systems due to slightly higher operating temperatures, pressures and flow rates internal to the pipes. For those systems with analysis, the maximum stress level analysis results were reviewed based on specific increases in temperature, pressure and flow rate (see Tables 2.2-

2a and 2.2-2b). These piping systems have been evaluated using the process defined in Appendix K of ELTR1 and found to meet the appropriate code criteria for the EPU conditions, based on the design margins between actual stresses and code limits in the existing design. The original construction code was USAS B31.1.0 – 1967 Power Piping Code. The existing code of record for many systems is ANSI B31.1.0, 1977 Edition with Addenda up to and including Winter 1978 and ASME Boiler and Pressure Vessel Code – Section III, Division I 1977 Edition through the Winter 1978 Addenda for torus attached piping. The existing code of record for other specific systems includes other versions of ANSI B31.1.0 and/or ASME Section III, Division I. The Codes of Record as referenced in the appropriate calculations, code allowable values, and analytical techniques were used and no new assumptions were introduced. For those systems that do not require a detailed analysis, pipe routing and flexibility were evaluated and determined to be acceptable.

Pipe break criteria were evaluated in accordance with Monticello Design Criteria, which are based on the Giambusso letter, SRP 3.6.2 and Generic Letter 87-11. Where required, percentage increases were applied to the calculated stress levels at applicable piping system node points. The combination of stresses was evaluated to meet the requirement of pipe break criteria. Based on these criteria, no new postulated pipe break locations were identified.

Pipe Supports

Operation at the EPU conditions increases the pipe support loadings on some BOP piping systems due to increases in the temperature of the affected piping systems (see Tables 2.2-2a, 2.2-2b, and 2.2-2c).

The pipe supports for the systems affected by EPU loading increases were reviewed to determine if there is sufficient margin to code acceptance criteria to accommodate the increased loadings. This review shows that, in most cases, support loads under EPU conditions are in compliance with the appropriate Code criteria. Additional detailed analyses will be performed and/or the supports will be modified prior to operation at EPU conditions (see Table 2.2-2d) to ensure code limits are not exceeded.

Main Steam and Associated Piping System Evaluation (Outside containment)

The MS piping system (outside containment) was evaluated for compliance with Monticello criteria, including the effects of EPU on piping stresses, piping supports, and the associated building structure, turbine nozzles, and valves.

Because the MS piping pressures and temperatures outside containment are not affected by EPU, there was no effect on the analyses for these parameters. The increase in MS flow results in increased forces from the turbine stop valve closure transient (TSVC). The turbine stop valve closure loads bound the MSIV valve loads because the MSIV closure time is significantly longer than the stop valve closure time. Due to the magnitude of the TSVC transient load increase, the transient event was reanalyzed. The MS piping was then reanalyzed using this revised load definition. The MS turbine stop valve closure transient analysis pipe stress and support results are provided in Table 2.2-2c.

Pipe Stresses

The results of the Main Steam system piping analysis indicate that piping load changes do not result in load limits being exceeded for the MS piping system outside containment except for a few small bore lines. Additional detailed analyses will be prepared and/or the piping will be modified for these small bore lines prior to EPU implementation to ensure code limits are not exceeded (See Table 2.2-2d). No new postulated pipe break locations were identified.

Pipe Supports

The pipe supports and turbine nozzles for the MS piping system outside containment were evaluated for the increased loading and movements associated with the turbine stop valve closure transient at EPU conditions. The evaluations demonstrate that the supports and turbine nozzles have adequate design margin to accommodate the increased loads and movements resulting from EPU except for a few supports. Additional detailed analyses will be prepared and/or the supports will be modified prior to EPU implementation to ensure code limits are not exceeded (See Table 2.2-2d). Based on existing margins available for the outside containment MS piping supports, except for those supports that may require modification, it was concluded that EPU does not result in reactions on existing structures in excess of the current design capacity. Structural capacity associated with modified supports will be evaluated prior to EPU implementation to ensure design capacity is not exceeded.

Conclusion

NSPM has evaluated the structural integrity of pressure-retaining components and their supports and has addressed the effects of the proposed EPU on these components and supports. The evaluation indicates that pressure-retaining components and their supports will continue to meet the requirements of 10 CFR 50.55a and Monticello's current licensing basis following implementation of the proposed EPU. Therefore, the proposed EPU is 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

Reactor pressure vessel internals consist of all the structural and mechanical elements inside the reactor vessel, including core support structures.

The NRC's acceptance criteria are based on (1) 10 CFR 50.55a and GDC-1, 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, 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, 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, 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.

Specific NRC 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.

Monticello Current Licensing Basis:

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC listed in the Regulatory Evaluation above, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-1, draft GDC-2, draft GDC-5, draft GDC-6, draft GDC-40, and draft GDC-42. Current GDC-10 is applicable to Monticello for certain events as described in USAR Section 14.4.

The Reactor Pressure Vessel Internals and Core Supports are described in Monticello USAR Sections 3.6, "Other Reactor Vessel Internals," and 4.2, "Reactor Vessel."

In addition to the evaluations described in the Monticello USAR, Monticello's systems and components were evaluated for License Renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Monticello Nuclear Generating Plant License Renewal Safety Evaluation Report (SER), NUREG-1865, dated October 2006 (Reference 5). The reactor internals and core support structural components evaluation for license renewal are discussed in NUREG-1865, Section 3.1.

Technical Evaluation

[[

	]]	Maximum core flow at OLTP and EPU is 60.5 Mlbs/hr (105% of rated core flow)

The FIV evaluation of the RPV internals addresses the influence of an increase in flow during EPU and consists of the following:

- Structural Evaluation of core flow dependent RPV Internals [[ ]]
- Structural Evaluation of other RPV Internals.

The core flow dependent RPV internals (in-core guide tube and control rod guide tube components) are [[ ]] the maximum core flow does not change for EPU.

The required RPV internals vibration assessment of the other RPV internals is described in the CLTR. EPU operation increases the steam production in the core, resulting in an increase in the core pressure drop. There is only a slight increase (1.7%) in maximum drive flow at EPU conditions for Monticello as compared to CLTP. The increase in power may increase the level of reactor internals vibration. Analyses were performed to evaluate the effects of FIV on the reactor internals at EPU conditions. This evaluation used a bounding reactor power of 2004 MWt and 105% of rated core flow. This assessment was based on vibration data obtained during startup testing at Monticello and of the similar prototype plant. For components requiring an evaluation but not instrumented in the vibration data obtained during the startup testing, then data from similar plants or acquired outside the RPV was used. The expected vibration levels for EPU were estimated by extrapolating the vibration data recorded in the plant startup testing of similar plants and on GEH BWR operating experience as described in the CLTR (Reference 1). These expected vibration levels were then compared with the established vibration acceptance limits. The following components were evaluated:

- Shroud Head and Separator
- Jet Pumps
- Feedwater Sparger
- Jet Pump Sensing Lines

- Incore Guide Tube
- Control Rod Guide Tube
- Fuel Assembly, Top Guide, and Core Plate
- Guide Rods
- Shroud Head Bolts
- RPV Top Head Spare Instrument Nozzle
- RPV Top Head Vent Nozzle
- RPV Head Spray Pipe and Head Spray Nozzle
- Core Spray Piping (internal to RPV)

The results of the vibration evaluation show that continuous operation at a reactor power of 2004 MWt and 105% of rated core flow does not result in any detrimental effects on the safety-related reactor internal components. See Table 2.2-3.

In order to apply the vibration criteria, a structural dynamic analysis was performed to relate peak stresses to measured strains or displacements at sensor locations. Finite element models for each component were developed to calculate the natural frequencies and mode shapes for these components. The locations and magnitudes of the peak stress intensity, including the effects of stress concentration factors, are identified. The acceptance criteria for each mode are then determined by calculating the modal strains and displacements at sensor locations by normalizing the peak stress intensity to 10,000 psi. The percent criteria for each significant natural mode are then determined by obtaining a ratio of the measured modal response to the calculated acceptance criteria for that mode. The percent contributions of the various modes are absolute summed for conservatism. The component is deemed acceptable if the modal sum is less than 100%. A value of less than 100% confirms that the maximum vibration stress is less than 10,000 psi and therefore no fatigue usage is accumulated by the component due to FIV.

During EPU, the components in the upper zone (plenum) of the reactor, such as the moisture separators and dryer, are mostly affected by the increased steam flow. Components near the shroud head, such as Shroud Head Bolts and Guide Rods, are mainly affected by the increase of feedwater flow. Components in the core region and components such as the Core Spray line are primarily affected by the core flow. Components in the annulus region such as the jet pump are primarily affected by the recirculation pump drive flow and core flow. Maximum licensed core flow at Monticello remains unchanged as a result of EPU. Hence there is no change in the vibrations of core flow dependent components from OLTP. Maximum recirculation drive flow increases negligibly (approximately 1.7%) due to increases in core differential pressure.

The steam dryer and steam separators are non safety-related components. Failure of a dryer component does not represent a safety concern, but can result in a large economic effect.

A proprietary evaluation has been performed to characterize dryer stress at EPU conditions considering dynamic loading conditions. This evaluation is provided as enclosures 11 (proprietary) and 12 (non-proprietary). It concludes that the Monticello steam dryer is structurally adequate for operation at EPU conditions.

The calculations for EPU conditions indicate that vibrations of all safety-related reactor internal components are within the GEH acceptance criteria of 10,000 psi.

The analysis is conservative for the following reasons:

- The GEH criteria of 10,000 psi peak stress intensity is less than the ASME Code criteria of 13,600 psi;
- The modes are absolute summed; and
- The maximum vibration amplitude in each mode is used in the absolute sum process, whereas in reality the peak vibration amplitudes are unlikely to occur at the same time.

Based on the above, it is concluded that FIV effects are expected to remain within acceptable limits at EPU conditions.

[[

[illegible]	[illegible]	[illegible]

[illegible]	[illegible]	[illegible]

		]] The above components are unaffected by EPU operating conditions.

[[

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Certain reactor vessel components are [[

]]

The Recirculation Outlet Nozzle, Steam Outlet Nozzle, Core Spray Nozzle, Top Head Spray, Instrument Nozzles and Vent Nozzles, Jet Pump Instrumentation Nozzles, CRD hydraulic system return nozzle, Core ΔP and Liquid Control Nozzle, 4” Instrumentation Nozzles, Drain Nozzle, Bottom Head Support Skirt, CRD Penetrations, Main Closure Region Flange, Shroud

Support, Top Head and Shell Insulation Support Brackets, Steam Dryer Hold Down Brackets, Guide Rod Brackets, Steam Dryer Support Brackets, Feedwater Sparger Brackets, Core Spray Brackets, Upper and Lower Surveillance Brackets, Dry Tube, IRM/SRM Dry Tube, Power Range Detector, and In-Core Detector Assembly components are [[ ]] for Monticello. Therefore these components are considered acceptable for EPU based on the EPU evaluation methodology.

The Top Head and Cylindrical Shell and the Stabilizer Bracket were not evaluated for fatigue at the time that the OLTP evaluation was performed, and have not been evaluated for EPU.

High and Low Pressure Seal Leak Detection Nozzles, Top Head Lifting Lugs, and the Jet Pump Riser Support Pads were not considered to be pressure boundary components at the time that the OLTP evaluation was performed, and have not been evaluated for EPU.

The effect of EPU was evaluated to ensure that the reactor vessel components continue to comply with the existing structural requirements of the ASME Boiler and Pressure Vessel Code. For the OLTP components under consideration, the 1965 code with addenda to and including Summer 1966, with the following exceptions: i.) ASME SA-533 plate material allowed by Addenda Summer 1967 and ii.) Inconel material allowed by Addenda Summer 1967. These were used as the governing code and are considered the code of construction. However, if a component's design has been modified, the governing code for that component was the code used in the stress analysis of the modified component. The following components were modified since the original construction of Monticello:

- Recirculation Outlet Nozzle Safe Ends: This component was modified and the governing Code for the evaluation/modification is the ASME Boiler and Pressure Vessel Code, Section III, 1980 Edition with Addenda to and including Summer 1982.
- Recirculation Inlet Nozzle: This component was modified and the governing Code for the evaluation/modification is the ASME Boiler and Pressure Vessel Code, Section III, 1980 Edition with Addenda to and including Winter 1980.
- Feedwater Nozzle: This component was modified and the governing Code for the evaluation/modification is the ASME Boiler and Pressure Vessel Code, Section III, 1977 Edition with Addenda to and including Summer 1978.
- Core Spray Nozzle Safe End Extension: This component was modified and the governing Code for the evaluation/modification is the ASME Boiler and Pressure Vessel Code, Section III, 1977 Edition with Addenda to and including Summer 1978. Further analysis was performed using the 1980 Edition with Addenda to and including Summer 1982.
- Top Head Spray, Instrument, and Vent Nozzles: This component was modified and the governing Code for the evaluation/modification is the ASME Boiler and Pressure Vessel Code, Section III, 1980 Edition with Addenda to and including Winter 1980.

- Jet Pump Instrumentation Nozzle Penetration Seal: This component was modified and the governing Code for the evaluation/modification is the ASME Boiler and Pressure Vessel Code, Section III, 1980 Edition with Addenda to and including Winter 1980.
- CRD Hydraulic System Return Nozzles: This component was modified and the governing Code for the evaluation/modification is the ASME Boiler and Pressure Vessel Code, Section III, 1980 Edition with Addenda to and including Winter 1980.
- Core ΔP and Liquid Control Nozzle Safe End: This component was modified and the governing Code for the evaluation/modification is the ASME Boiler and Pressure Vessel Code, Section III, 1980 Edition with Addenda to and including Winter 1981.
- Dry Tube: This component was modified and the governing Code for the evaluation/modification is the ASME Boiler and Pressure Vessel Code, Section III, 1971 Edition with Addenda to and including Summer 1973.
- IRM/SRM Dry Tube: This component was modified and the governing Code for the evaluation/modification is the ASME Boiler and Pressure Vessel Code, Section III, 1977 Edition with Addenda to and including Summer 1977.
- Power Range Detector: This component was modified and the governing Code for the evaluation/modification is the ASME Boiler and Pressure Vessel Code, Section III, 1971 Edition with Addenda to and including Summer 1973 as well as 1977 Edition with Addenda to and including Summer 1977.
- In-Core Detector Assembly: This component was modified and the governing Code for the evaluation/modification is the ASME Boiler and Pressure Vessel Code, Section III, 1971 Edition with Addenda to and including Summer 1973.

New stresses are determined by scaling the "original" stresses based on the EPU conditions (pressure, temperature, and flow). [[

]] The analyses were performed for the design, the normal and upset, and the emergency and faulted conditions. If there is an increase in annulus pressurization, jet reaction, pipe restraint or fuel lift loads, the changes are considered in the analysis of the components affected for normal, upset, emergency and faulted conditions.

Design Conditions

Because there are no changes in the design conditions due to EPU, the design stresses are unchanged and the Code requirements are met.

Normal and Upset Conditions

The reactor coolant temperature and flows (except core flow) at EPU conditions are only slightly changed from those at current rated conditions. Evaluations were performed at conditions that bound the slight change in operating conditions. The type of evaluations is mainly reconciliation of the stresses and usage factors to reflect EPU conditions. A primary plus secondary stress analysis was performed showing EPU stresses still meet the requirements of the ASME Code, Section III, Subsection NB. Lastly, the fatigue usage was evaluated for the limiting location of components with a usage factor greater than 0.33. The Monticello fatigue analysis results for the limiting components are provided in Table 2.2-4. The Monticello analysis results for EPU show that all components meet their ASME Code requirements.

Emergency and Faulted Conditions

The stresses due to Emergency and Faulted conditions are based on loads such as peak dome pressure, which are unchanged for EPU. Therefore, Code requirements are met for all RPV components under emergency and faulted conditions.

Reactor Internal Pressure Differences

The increase in core average power alone would result in higher core loads and reactor internal pressure differences (RIPDs) due to the higher core exit steam quality. The maximum acoustic and flow-induced loads, following a postulated recirculation system line break (RSLB), were shown to be unaffected by the EPU.

The RIPDs are calculated for Normal (steady state operation), Upset, Emergency and Faulted conditions for all major reactor internal components. For minor components (jet pump sensing lines, dryer/separator guide rods, and in-core guide tube braces), the pressure drops during Normal, Upset, Faulted, and Emergency conditions are minimal and represent insignificant portions of the loads because of the small surface area, and thus are not affected by EPU and are not evaluated for EPU.

Tables 2.2-4 through 2.2-8 compare results for the various loading conditions between original analysis results and operation with EPU for the vessel internals that are affected by the changed RIPDs.

Steam Dryer RIPD Methodology

The CLTP analysis for steam dryer dP at normal condition used a different methodology than the current GEH method. The CLTP analysis method applied the more conservative BWR4-6 correlation based on the air test data for BWR6 steam dryers, which resulted in a steam dryer dP of 0.49 psid irreversible, (Table 3-1 of Reference 7) at 1880 MWt.

The current GEH methodology is based on a more realistic correlation for a BWR3 steam dryer, which results in a steam dryer dP of 0.30 psid (irreversible) at EPU. In addition to the irreversible term, the elevation loss of 0.08 psid was calculated for EPU per the current GEH methodology.

For the upset condition, the CLTP analysis calculated a steam dryer dP (irreversible) of 1.23 psid by [[ ]] as documented in Table 3-2 of Reference 7.

The EPU analysis is based on the current method [[ ]], resulting in upset condition dPs of 0.62 psid (irreversible) and 0.16 psid (elevation). [[ ]].

For faulted condition, CLTP analysis conservatively selected the peak steam dryer dPs (5.2 psid for high power and 9.0 psid for interlock as shown in Table 3-3 of Reference 7) that [[ ]] compared to the realistic peak values of 2.8 psid (high power) and 3.5 psid (interlock) [[ ]] for EPU based on the current GEH methodology.

The current GEH methodology for determining steam dryer dPs has been used for previous EPU projects.

Reactor Internals Structural Evaluation (Non-FIV)

The RPV internals consist of the Core Support Structure (CSS) components and non-CSS components. The RPV Internals are not ASME Code components, however, the requirements of the ASME Code are used as guidelines in their design/analysis. The evaluations/stress reconciliation in support of the EPU was performed consistent with the design basis analysis of the components. The reactor internal components evaluated are:

Core Support Structure Components

- Shroud
- Shroud Support
- Core Plate
- Top Guide
- Control Rod Drive Housing

- Control Rod Guide Tube
- Orificed Fuel Support
- Fuel Channel

Non-Core Support Structure Components

- Steam Dryer
- FW Sparger
- Jet Pumps
- Core Spray Line and Sparger
- Access Hole Cover
- Shroud Head and Steam Separator Assembly
- In-Core Housing and Guide Tube
- Jet Pump Instrument Penetration Seal
- Core Differential Pressure and Standby Liquid Control Line

The original configurations of the internal components are considered in the EPU evaluation unless a component has undergone permanent structural modifications, in which case, the modified configuration is used as the basis for the evaluation. Structural modifications at Monticello include the following:

- Shroud head bolts replaced with new bolts. This made the system structurally the same as the original condition.
- Modification to the T-box for the Core Spray Lines and Spargers. This modification was installed in 1994 to address a crack in the vicinity of the T-box of one of the lines.

The effects on the loads as a result of the thermal-hydraulic changes due to EPU were evaluated for the reactor internals. All applicable Normal, Upset, Emergency, and Faulted service condition loads were considered consistent with the existing design basis analysis. These loads include the RIPDs, dead weight, seismic loads, acoustic and flow induced loads, fuel lift loads and thermal loads.

EPU has no effect on the dead weight and seismic loads. As a result of EPU, the RIPDs increased for some components. The fuel lift loads due to the combined effect of uplift pressures and dynamic loads were evaluated for the EPU-specific conditions and determined to be negligible. The effects of changes in acoustic and flow induced loads; thermal and flow conditions are also taken into consideration in the evaluation of the internal components.

EPU loads are compared to those used in the existing design basis analysis. If the load conditions do not change or if the loads do not increase due to EPU, the existing analysis results are assumed valid and bounding for EPU conditions. No further analysis/qualification is required. [[

]]

Table 2.2-9 presents the governing stresses for the various reactor internal components as affected by EPU. All stresses are within allowable limits, and the RPV internal components are demonstrated to be structurally adequate for operation in the EPU condition.

The following reactor vessel internals are evaluated for the effects of changes in loads due to EPU:

a) **Shroud**: [[

]] Based on the evaluation it was concluded that the stresses in the shroud in Normal, Upset, Emergency, and Faulted conditions remain within the design basis ASME Code allowable limits. Hence, the shroud remains qualified in its original configuration for operation in at EPU conditions.

b) **Shroud Support**: [[

]] Based on the evaluation, it was concluded that the stresses in the shroud support assembly in the Normal, Upset, Emergency, and Faulted conditions remain within the design basis ASME Code allowable limits. Hence, the shroud support assembly remains qualified in its original configuration for operation at EPU conditions.

c) **Core Plate**: [[

]]. Based on the above it was shown that the stresses and the buckling strength of the core plate longest beam in the Normal, Upset, Emergency and Faulted conditions remain within the ASME Code allowable limits. Hence, the core plate remains qualified in its original configuration for operation at EPU conditions.

d) **Top Guide**: [[

]] Based on the above it was shown that the stresses in the Normal, Upset, Emergency, and Faulted conditions remain within the ASME Code allowable limits. Hence, the top guide remains qualified in its original configuration for operation at EPU conditions.

e) **Control Rod Drive Housing: (inside RPV):** [[

]] Based on the above, the Control Rod Drive Housing (inside RPV) remains qualified in its original configuration for operation at EPU conditions. The portion of the Control Rod Drive Housing outside the RPV remains unaffected by EPU because the design pressure and seismic loads remain unaffected.

f) **Control Rod Guide Tube:** [[

]] Based on the above, the CRGT remains qualified in its original configuration for operation at EPU conditions.

g) **Orificed Fuel Support:** [[

]] Based on the above, the Orificed Fuel Support remains qualified in its original configuration for operation at EPU conditions.

h) **Fuel Channel:** The fuel channel RIPDs are within the design basis limits for GE14 fuel for all service levels. Channel/Control blade interference is not affected by EPU, because for constant pressure EPU operation, the effects of channel-control rod interference are the same as for CLTP operation. Monticello has implemented the recommended GEH surveillance program for monitoring channel-control blade interference. The effects of channel bow and channel-control rod interference are included in the reload core designs. The GE14 fuel channel is structurally qualified for EPU.

i) **Steam Dryer:** The Monticello licensing basis dryer analysis is a static calculation that considers bounding faulted differential pressure loads created by the guillotine rupture of a main steam line outside of containment. The limiting structures on the Monticello dryer for this analysis are the lifting rods. The calculated differential pressure that is likely to cause buckling of the dryer lifting rods is 12 psi. The differential pressure value currently used in the analysis for this event at CLTP is 9.0 psi. The methodology for calculating the faulted differential pressure under EPU conditions (Table 2.2-8) yields a substantially lower value. Nonetheless, this value is calculated at an off-rated plant condition (hot standby) that does not change at EPU. This differential pressure remains the bounding condition for the Monticello dryer under EPU conditions. Therefore, the limiting static dryer stress for the Monticello dryer is unaffected by EPU.

j) **FW Sparger:** [[

]]. Based on the above assessment, the feedwater sparger remains qualified in its original configuration for operation at EPU conditions.

k) **Jet Pump Assembly:** [[

]] Hence, the CLTP value of stresses remains valid and applicable for EPU conditions also. The load tending to disassemble the beam bolt was calculated conservatively by considering the increase in hydraulic flow loads and then compared to end-of-life preload of the bolt. The analysis shows that the stresses for the jet pump components in the Normal, Upset, Emergency and Faulted conditions remain within ASME Code limits. Hence, the jet pump assembly remains qualified in its original configuration for operation at EPU conditions.

l) **Core Spray Lines and Spargers:** [[

]] Based on the above, the Core spray line and sparger assembly remains qualified for operation at EPU conditions.

m) **Access Hole Cover:** [[

]] Based on the above, the access hole cover is qualified for operation at EPU conditions.

n) **Shroud Head and Steam Separator Assembly (including Shroud Head Bolts):** [[

]]
Based on the above, it was shown that the stresses in the shroud head bolts, in the lugs, and in the flange remain within the ASME Code allowable values. Hence, the shroud head and steam separator assembly is qualified for operation at EPU conditions.

o) **In-Core Housing and Guide Tube:** [[

]] Based on the above, the in-core housing and guide tube remain qualified for operation at EPU conditions.

p) **Jet Pump Instrument Penetration Seal:** [[

]] Therefore, the existing design basis evaluation remains applicable at EPU condition. Hence, the jet pump instrument penetration seal is qualified for operation at EPU conditions.

q) **Core Differential Pressure and Liquid Control Line:** [[

]] Based on the

above, the core differential pressure and liquid control line remain qualified for operation at EPU conditions.

Steam Dryer/Separator Performance

EPU results in an increase in saturated steam generated in the reactor core. For constant core flow, this in turn results in an increase in the separator inlet quality and dryer face velocity and a decrease in the water level inside the dryer skirt. These factors, in addition to the radial power distribution affect the steam separator-dryer performance. The performance of the steam separators and dryer has been evaluated to assess their capability to provide the quality of the steam leaving the reactor pressure vessel necessary to meet operational criteria at EPU conditions.

The evaluation of steam separator and dryer performance at EPU conditions indicates an increase in moisture carryover will occur. The effect of increasing steam moisture content on the radiation source terms is addressed in Section 2.10. Steam separator-dryer performance operational testing is included in the CPPU implementation plan as described in Section 2.12 to ensure adequate operating limitations are implemented as required.

Conclusion

NSPM has evaluated the effects of the proposed EPU on the reactor internals and core supports. The evaluation indicates that the reactor internals and core supports will continue to meet the requirements of 10 CFR 50.55a and Monticello's current licensing basis following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to the design of the reactor internal and core supports.

2.2.4 Safety-Related Valves and Pumps

Regulatory Evaluation

The NRC's acceptance criteria are based on (1) GDC-1, 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-37, GDC-40, GDC-43, and GDC-46, insofar as they require that the emergency core cooling system (ECCS), 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, 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 NRC review criteria are contained in SRP Sections 3.9.3 and 3.9.6; and other guidance provided in Matrix 2 of RS-001.

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC listed in the Regulatory Evaluation above, with the exception of current GDC-46, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-1, draft GDC-5, draft GDC-38, draft GDC-46, draft GDC-47, draft GDC-48, draft GDC-51, draft GDC-57, draft GDC-59, draft GDC-60, draft GDC-61, draft GDC-63, draft GDC-64, and draft GDC- 65. There is no draft GDC directly associated with current GDC-46.

The inservice testing of safety-related valves and pumps is described in Monticello USAR Section 13.4.6, "10CFR50.55a Inservice Inspection and Testing Programs." MOV programs related to GL 89-10, GL 95-07 are described in USAR Section 8.11, "Power Operated Valves."

In addition to the evaluations described in the Monticello USAR, Monticello's safety-related valves and pumps were evaluated for License Renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Monticello Nuclear Generating Plant License Renewal Safety Evaluation Report (SER), NUREG-1865, dated October 2006 (Reference 5). The safety-related valves and pumps are addressed within NUREG-1865 under the systems that contain them.

Technical Evaluation

Containment Isolation

Motor-Operated and Air Operated Valves

For the majority of the GL 89-10 MOVs and safety-related air operated valves (AOVs), the EPU operating conditions are bounded by the existing design inputs used to establish thrust/torque margins for these valves. For the remaining valves, the EPU operating conditions result in small changes to design inputs, such as differential pressure and maximum ambient temperature, depending on the valve-operating scenario. These changes have been identified for each affected valve. A bounding engineering evaluation was performed that enveloped the EPU changes to the design inputs. The evaluations determined that sufficient design margin exists such that all valve actuator capabilities remain within the acceptance criteria for safe operation without causing spurious trips or valve damage. Therefore, the MOVs and AOVs remain capable of performing their design basis functions at EPU conditions without modification. A field adjustment to a torque switch setting was identified for one MOV. This change will be made prior to implementing EPU.

EPU does not introduce any changes to the plant specific analysis or compliance methodology for GL-95-07, Pressure Locking and Thermal Binding of Safety Related Power-Operated Valves. EPU does not result in additional valves that are susceptible to pressure locking and thermal binding, and the existing set of potentially susceptible valves continue to perform their associated safety functions.

Conclusion

NSPM has evaluated the effects of the proposed EPU on safety-related valves. The evaluation addressed the effects of the proposed EPU on its MOV programs related to GL 89-10 and GL 95-07. The evaluation indicates that safety-related valves will continue to meet the requirements of 10 CFR 50.55a(f) and the current licensing basis following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to safety-related valves.

2.2.5 Seismic & Dynamic Qualification of Mechanical & 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's acceptance criteria are based on (1) GDC-1, 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, 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, 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, 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, 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, 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, which sets quality assurance requirements for safety-related equipment.

Specific NRC review criteria are contained in SRP Section 3.10.

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC listed in the Regulatory Evaluation above, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: 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.

The Seismic and Dynamic Qualification of Mechanical and Electrical Equipment is described in Monticello USAR Section 12.2, "Plant Principal Structures and Foundations."

Technical Evaluation

The effect of dynamic forces (pipe whip and jet impingement) is minimal because there is no pressure increase for the EPU (see Section 2.2.2). As stated above, the primary input motions due to the safe shutdown earthquake are not affected by an EPU and therefore, there are no consequences to the existing seismic analyses. No quality standards related to the design, fabrication, erection, and testing of the RCPB or SSCs important to safety are relaxed or removed as a result of the EPU and no changes have been made to the plant design bases established in consideration of the seismic and geologic characteristics of the plant site.

The temperatures, accident radiation level, and the normal radiation level increase due to EPU. These effects are not considered to have an adverse effect on the functional capability of non-metallic components in the mechanical equipment both inside and outside containment.

The Monticello design and licensing bases do not require a formal mechanical EQ program like the EQ program applied to electrical equipment. The design control program ensures that mechanical components are specified and procured for the environment in which they are intended to function. Periodic maintenance and testing are performed in accordance with industry operating experience and vendor recommendations to ensure continued functionality. Causes of failures are investigated as part of the Monticello Maintenance Rule programs and incorporated into equipment reliability improvement efforts. Aging Management, Long Term Planning, and Life Cycle Management strategies are used to manage equipment aging and obsolescence.

The mechanical design of equipment/components (e.g., pumps, heat exchangers) in certain systems is affected by operation at EPU due to slightly increased temperatures and in some cases, flows. The revised operating conditions do not significantly affect the CUFs of mechanical components.

The effects of increased fluid induced loads on safety-related components are described in Sections 2.6.1.2. Increased nozzle loads and component support loads due to the revised operating conditions were evaluated within the piping assessments. These increased loads are insignificant, and become negligible (i.e., remain bounded) when combined with the governing dynamic loads. Therefore, the mechanical components and component supports are adequately designed for EPU conditions.

Conclusion

NSPM has evaluated the effects of the proposed EPU on the qualification of mechanical and electrical equipment and addressed the effects of the proposed EPU on this equipment. The evaluation indicates that the equipment will continue to meet the requirements of 10 CFR Part 100, Appendix A; 10 CFR Part 50, Appendix B; and its current licensing basis following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to the qualification of the mechanical and electrical equipment.

Table 2.2-1 Liquid Line Breaks

Location	EPU Analysis Results[1] Compared to Current EQ Analysis			
	Mass and Energy Release	**Room Flooding**	**Room Pressure**	**Room Temperature**
Reactor Building		Increase $\leq$ 2.15 ft	Increase $\leq$ 1.25 psi	Increase $\leq$ 86.5 °F
Turbine Building		Increase $\leq$ 4.35 ft	Increase $\leq$ 0.25 psi	Increase $\leq$ 102.5 °F
RWCU System (affects Reactor Building)	Peak mass flow rate increases.			
FW system (affects Reactor Building and Turbine Building)	Peak mass flow rate increases by up to 23%. Total mass release does not change significantly. Integrated energy release increases by about 4%.			
Condensate system (affects Turbine Building)	Peak mass flow rate increases by up to 28%. Total mass release increases by up to 20%. Integrated energy release increases by about 4%.[2]			

Notes:

1. As discussed in Section 2.2.1, EPU Analysis Results include the composite effects of EPU and changes in methodology.
2. One scenario evaluated had a long isolation time of 8 hours and resulted in significantly larger release of energy over a longer time, however, this case is not limiting due to its small mass flow rate. This scenario was omitted from the evaluation of EPU impacts because it is a methods change and not an EPU effect.

Table 2.2-2a BOP Piping FW, Extraction Steam, FW Heater Drains & Vents, Condensate and MSR Drains

System	FW[1]	Extraction Steam	Heater Drains & Vents	Condensate[2]	MSR Drain Lines
Maximum pipe stress increase from:					
Temperature expansion	0%	11%	72%	N/A	19%
Pressure	0%	64%	Note 3	N/A	Note 3
Fluid Transients	N/A	N/A	N/A	N/A	N/A
Maximum pipe support loading increase (due to thermal expansion loading):	0%	11%	72%	N/A	19%

Notes:

1. FW section from containment flued head penetration to first isolation valve. The remaining BOP FW will be analyzed and qualified for EPU in conjunction with the FW and Condensate pump modifications prior to operation at EPU conditions.
2. The condensate piping from the FW pumps to the condensate pumps will be analyzed and qualified for EPU in conjunction with the FW and Condensate pump modifications prior to operation at EPU conditions.
3. Calculations for existing pressure stresses were not available. New calculations were prepared to qualify these lines.

Table 2.2-2b BOP Piping

CS, RCIC, HPCI and RHR (Outside Containment)

System	CS	RHR Shutdown Cooling	RHR injection	RCIC injection[1]	HPCI injection[1]
Maximum pipe stress increase due to:					
Temperature expansion	8.9%	0%	8.9%	8.9%	8.9%
Pressure	0%	0%	0%	0%	0%
Fluid Transients	N/A	N/A	N/A	N/A	N/A
Maximum pipe support loading increase (due to thermal expansion loading):	8.9%	8.9%	8.9%	8.9%	8.9%

Notes:

1. Portion connecting to Torus or Torus header.

Table 2.2-2c BOP Piping
Main Steam System (Outside Containment)

Maximum pipe stress at EPU due to:	
Temperature expansion	No change
Pressure	No change
Fluid Transients	38% (Note 1)
Maximum pipe support loading:	
EPU (due to thermal expansion loading)	No change
EPU (due to fluid transient loading)	38% (Note 1)

Notes: 1. The fluid transient load and stress increases shown above are based on a 17.5% flow increase. A refined analysis was performed to evaluate the TSVC loads resulting in much lower loads and stresses.

Table 2.2-2d Piping Components Requiring Further Reconciliation

Item	System
1	Main Steam (Outside Containment)[1]
2	Feedwater and Condensate (from condensate pump to the feedwater MO valves downstream of the HP Heaters), due to pending pump changes
3	Torus Attached Piping
4	RHR (BOP Condensate Service Water Lines)
5	Cross Around Piping [1,2]
6	CARV Discharge Piping [1,3]

Notes:

1. Walkdowns in Hi Radiation Areas are required to complete calculations.
2. Scope of Cross Around Piping Analysis is being determined based upon turbine modification.
3. Scope of CARV Analysis is being determined based upon turbine modification.

Table 2.2-3 FIV Component Analysis Results

Analysis	Natural Frequency (Hz)	Parameter	Units	EPU/OLTP Change	1.02×EPU Value	Evaluation
Shroud Head and Separator		Peak Stress	psi	172% increase	<500	GEH acceptance criterion of 10,000 psi met.
Jet Pumps		Peak Stress	psi	54% or 34% increase, for Jet Pumps with or without cracked Brace	<7,000	Stress is within the GEH acceptance criterion of 10,000 psi, and less than the ASME Fatigue Curve C criterion of 13,600 psi..
Feedwater Sparger		Peak Stress	psi	67% increase	1,670	GEH acceptance criterion of 10,000 psi met; no FIV concern if interference fit is maintained.
Jet Pump Sensing Line		Vane Passing Frequency Resonance	NA	No change	Acceptable	No FIV concern.
Incore Guide Tube		Peak Stress	psi	59% increase	<1,100	No FIV concern.
Control Rod Guide Tube		Peak Stress	psi	51% increase	<500	No FIV concern.

Analysis	Natural Frequency (Hz)	Parameter	Units	EPU/OLTP Change	1.02×EPU Value	Evaluation
Fuel Assembly, Top Guide and Core Plate		Bundle Flow	NA	Increased slightly	Acceptable	No FIV concern. - Bundle flows from 0 gpm to 500 gpm were tested and found to be acceptable. Because the bundle flow for the EPU condition is less than 400 gpm, it is also acceptable. Fuel assembly vibration measured in plants at OLTP is less than 5% of the acceptance criterion.
Guide Rods	10	Peak Stress (cross flow 5 ft/sec assumed)	psi	NA	<600	No FIV concern - The FIV stress (<600 psi) is well within the 10,000 psi acceptance criterion.
Shroud Head Bolts		Peak Stress	psi	No Change	210	No FIV concern.
RPV Top Head Spare Instrument Nozzle	178	Peak Stress	psi	Insignificant Increase	Acceptable	No FIV concern – The natural frequency is 178 Hz and steam flow velocity is essentially zero near the nozzle.

Analysis	Natural Frequency (Hz)	Parameter	Units	EPU/OLTP Change	1.02×EPU Value	Evaluation
RPV Top Head Vent Nozzle	188	Steam Velocity	ft/sec	Insignificant Increase	Acceptable	No FIV concern - The natural frequency is 188 Hz and steam velocity is essentially zero near the nozzle.
RPV Head Spray Pipe and Head Spray Nozzle	88	Steam Velocity	ft/sec	Insignificant Increase	Acceptable	No FIV concern – The natural frequency is 88 Hz and steam velocity is essentially zero near the nozzle.
Core Spray Piping	19	Vortex Shedding Frequency	Hz	No Change	Acceptable	No FIV concern.

Table 2.2-4 CUFs of Limiting Components

Component[1]	P + Q Stress (ksi)			CUF[5]		
	Current (1775 MWt)	EPU (2044 MWt)[4]	Allowable (ASME Code Limit)	Current (1775 MWt)	EPU (2044 MWt)[4]	Allowable
Recirculation Inlet Nozzle	49.16	49.63	51.75	0.226	0.556	1.0
Feedwater Nozzle	62.3 / 45.94[2]	74.75 / 52.13[2]	55.80	0.621	0.9138	1.0
Main Closure Region Studs	89.824	90.722	108.975	0.573	0.534[3]	1.0
Refueling Bellows	51.737	52.25	58.91	0.861	0.833[3]	1.0
Bottom Head Support Skirt	[6]	[6]	[6]	0.40 / 0.2832[6]	[6]	1.0

Notes:

1. Only components with usage factors greater than 0.33 are included in this table.
2. Thermal bending included/Thermal bending removed. P + Q stresses are acceptable per CLTP elastic-plastic analysis, which is valid for EPU conditions.
3. This component was re-evaluated considering a more representative and less conservative treatment of the duty cycles.
4. EPU was conservatively evaluated for 102% of EPU (2004 MWt * 1.02).
5. Only the limiting CUF is presented.
6. As part of the EPU evaluation, excessive conservatism was removed from the 40-year CLTP CUF, resulting in the CLTP CUF = 0.2832. Because the 40-year CLTP CUF < 0.33, no further analysis is performed per the criteria defined in Section 2.2.3 above.

Table 2.2-5 RIPDs for Normal Conditions (psid)

Parameter	113 %OLTP (1)(2)	EPU (2)
Shroud Support Ring and Lower Shroud	26.41	26.63
Core Plate and Guide Tube	19.77	20.17
Upper Shroud	6.63	6.65
Shroud Head	6.78	7.21
Shroud Head to Water Level (Irreversible (3))	8.93	9.47
Shroud Head to Water Level (Elevation (3))	0.81	0.61
Top Guide	0.52	0.48
Steam Dryer	Not Calculated	0.30
Fuel Channel Wall	9.19	9.67

Notes:

1. Current RIPD reference base is 113 % Original Licensed Thermal Power (OLTP). OLTP is 1670 MWt.
2. 105% core flow.
3. Irreversible loss is the loss across the separators; the elevation loss or reversible head loss is the loss between the inside shroud to the exit of the separators.

Table 2.2-6 RIPDs for Upset Conditions (psid)

Parameter	113 %OLTP (1)(2)	EPU (2)
Shroud Support Ring and Lower Shroud	28.81	29.03
Core Plate and Guide Tube	22.17	22.57
Upper Shroud	9.95	9.98
Shroud Head	10.17	10.82
Shroud Head to Water Level (Irreversible (3))	13.39	14.21
Shroud Head to Water Level (Elevation (3))	1.21	0.92
Top Guide	<1.0	0.53
Steam Dryer	Not Calculated	0.62
Fuel Channel Wall	<11.3	<11.8

Notes:

1. Current RIPD reference base is 113 % Original Licensed Thermal Power (OLTP). OLTP is 1670 MWt.
2. 105% core flow.
3. Irreversible loss is the loss across the separators; the elevation loss or reversible head loss is the loss between the inside shroud to the exit of the separators.

Table 2.2-7 RIPDs for Emergency Conditions (psid)

Parameter	113 %OLTP (1)(2)	EPU (2)
Shroud Support Ring and Lower Shroud	37	38
Core Plate and Guide Tube	26.0	26.5
Upper Shroud	11.5	12.1
Shroud Head	12.2	12.8
Shroud Head to Water Level (Irreversible (3))	13.5	14.1
Shroud Head to Water Level (Elevation (3))	1.1	1.0
Top Guide	0.12	0.12
Steam Dryer	Not Calculated	N/C
Fuel Channel Wall	11.3	11.8

Notes:

1. Current RIPD reference base is 113 % Original Licensed Thermal Power (OLTP). OLTP is 1670 MWt.
2. 105% core flow.
3. Irreversible loss is the loss across the separators; the elevation loss or reversible head loss is the loss between the inside shroud to the exit of the separators.

Table 2.2-8 RIPDs for Faulted Conditions (psid)

Parameter	113 %OLTP (1)(2)	EPU (2)
Shroud Support Ring and Lower Shroud	47	45
Core Plate and Guide Tube	28.0	28.0
Upper Shroud	26.0	24.5
Shroud Head	26.0	25.0
Shroud Head to Water Level (Irreversible (3))	26.5	26.0
Shroud Head to Water Level (Elevation (3))	1.8	1.7
Top Guide	0.35	0.38
Steam Dryer	Not Calculated	3.5
Fuel Channel Wall	12.6	12.9

Notes:

1. Current RIPD reference base is 113 % Original Licensed Thermal Power (OLTP). OLTP is 1670 MWt.
2. Values are the maximum results from either the cavitation interlock power and flow or the high power and 105% core flow points.
3. Irreversible loss is the loss across the separators; the elevation loss or reversible head loss is the loss between the inside shroud to the exit of the separators

Table 2.2-9 Reactor Internal Components - Summary of Stresses

No	Component	CLTP				EPU					
		Location	Service Level	Units	Value	Location	Service Level	Stress Category /Other	Units	Value	Allowable
1	Shroud [1]	Lower Shroud	D	psi	39,500	Shroud Support Plate	D	Pm	psi	5,000	32,000
2	Shroud Support	Shroud Support Cylinder	B	psi	22,900	Shroud Support Cylinder	A, B	P_L	psi	22,900	34,950
3	Core Plate	Buckling [2]	B	psi	5,560	Longest Beam	B	Buckling	psi	5,560	6,200
4	Top Guide	Not Available	Not Available	psi	Not Available	End Connection Pins	A, B	Pure Shear	psi	9,067	10,140
5	Control Rod Drive Housing	Not Available	Not Available	psi	Not Available	RPV Bottom Head	A, B	Pm	psi	6,441	16,600
6.a	Control Rod Guide Tube	Not Available	Not Available	psi	Not Available	--	D	Pm	psi	5,911	16,600
6.b	Control Rod Guide Tube	Collapse	B	ratio	0.37	--	A, B	Collapse	ratio	0.38	0.45
7	Orificed Fuel Support	Not Available	Not Available	psi	Not Available	--	B	Pm+Pb	psi	9,679	15,580
8	Fuel Channel	Not Available	Not Available	--	Not Available	--	Qualified By GNF				

No	Component	CLTP				EPU					
		Location	Service Level	Units	Value	Location	Service Level	Stress Category /Other	Units	Value	Allowable
9	Feedwater Sparger	Not Available	Not Available	ratio	Not Available	Header to Sparger Weld	A, B	CUF	ratio	0.32	1.00
10.a	Jet Pump Assembly	Riser Elbow	B	psi	10,156	Riser Elbow	B	Pm+Pb	psi	10,156	25,350
10.b	Jet Pump Assembly	Not Available	Not Available	lbs	Not Available	Beam Bolt	A, B	Load	lbs	14,708	21,080
11	Core Spray Line and Sparger	Not Available	Not Available	psi	Not Available	Latch Bolt	A, B	Pm+Pb	psi	28,400	29,385
12	Access Hole Cover	Not Available	Not Available	psi	Not Available	--	D	Pm	psi	22,598	36,348 [3]
13	Shroud Head and Steam Separators Assembly	Shroud Head Bolt Lugs	D	psi	14,900	Shroud Head Bolts [4]	B	Pm	psi	11,400	16,000
14	In-Core Housing and Guide Tube	Not Available	Not Available	psi	Not Available	--	A, B, C, D	Pm	psi	5,967	16,000
15	Jet Pump Instrument Penetration Seal	Not Available	Not Available	psi	Not Available	--	C, D	Pm+Pb	psi	22,186	25,440

No	Component	CLTP				EPU					
		Location	Service Level	Units	Value	Location	Service Level	Stress Category /Other	Units	Value	Allowable
16	Core Differential Pressure and Liquid Control Line	Not Available	Not Available	Hz	Not Available	--	A, B	Natural Frequency	Hz	23	63.33

(1) The CLTP faulted stress value of 39,500 psi for shroud was obtained from the evaluations for a larger BWR4, 251" size New Loads unit, and was used as-is. The acoustic load for the EPU conditions increased by an amount in excess of the available stress margin. Therefore, a reduction in conservatism became necessary. An EPU-specific re-evaluation was performed, and the stress values obtained from this re-evaluation is provided for the EPU conditions.

(2) The CLTP value of stress for longest beam was based on a larger BWR4, 251" size unit. These values remain bounding for Monticello EPU conditions. The value of allowable stress is based on the stress-at-buckling evaluation for a BWR4 unit of the same size as Monticello.

(3) This allowable includes a weld factor of 0.65.

(4) The shroud head bolt is subjected to mechanical as well as thermal pre-load. Conservatively, the stress due to combined load is treated as primary membrane and the corresponding normal condition allowable is used.

2.3 Electrical Engineering

2.3.1 Environmental Qualification of Electrical Equipment

Regulatory Evaluation

Environmental qualification (EQ) of electrical equipment involves demonstrating that the equipment is capable of performing its safety function under significant environmental stresses that could result from DBAs.

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 that is located in a harsh environment.

Specific NRC review criteria are contained in SRP Section 3.11.

Monticello Current Licensing Basis

The Monticello program for environmental qualification of electrical equipment is described in Monticello USAR Section 8.9, "Environmental Qualification of Safety-Related Electrical Equipment."

In addition to the evaluations described in the Monticello USAR, the Monticello's environmental qualification of electrical equipment was evaluated for License Renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Monticello Nuclear Generating Plant License Renewal Safety Evaluation Report (SER), NUREG-1865, dated October 2006 (Reference 5). The environmental qualification of electrical equipment for license renewal is discussed in NUREG-1865, Sections 3.0.3.1.9 and 4.7.

Technical Evaluation

The Monticello EQ Program was developed to the guidance and requirements contained in the Division of Operating Reactors (DOR) Guidelines and Category II of NUREG 0588 for equipment that predates the issuance of 10CFR50.49 as delineated in 10CFR50.49 paragraph (k). For other equipment in the EQ program and for general guidance, NRC Regulatory Guide 1.89 contains methods for complying with regulatory requirements of 10CFR50.49. The safety-related electrical equipment was reviewed consistent with these requirements to determine if the existing qualification for the normal and accident conditions expected in the area where the devices are located remains adequate. The acceptance criteria including pressure, temperature, and radiation were used in making this determination. The evaluation of the EQ effects and parameter changes associated with the EPU are provided separately.

Inside Containment

EQ for safety-related electrical equipment located inside the containment is based on MSLB and/or DBA/LOCA conditions and their resultant temperature, pressure, humidity, and radiation consequences, and includes the environments expected to exist during normal plant operation. Normal temperatures are expected to increase slightly, but remain bounded by the normal temperatures used in the EQ analyses. The post-accident peak and long-term temperature and pressure for CLTP conditions increase slightly for EPU. However, the increase was determined not to adversely affect the qualification of safety-related electrical equipment.

The current radiation levels under normal plant conditions were evaluated to increase in proportion to the increase in RTP. The accident radiation levels increase above the levels used in the current EQ Program. The total integrated doses (normal plus accident) for EPU conditions were determined not to adversely affect qualification of the equipment located inside containment.

Outside Containment

Accident temperature, pressure, and humidity environments used for qualification of equipment outside containment result from an MSLB, or other HELBs, whichever is limiting for each plant area. The temperature, pressure and humidity profiles that are not bounded by the CLTP conditions were evaluated and do not adversely affect the qualification of safety related electrical equipment.

The accident temperature resulting from a LOCA/MSLB inside containment increased for some Reactor Building areas due to the additional heat load resulting from the increase in drywell and wetwell temperatures. However, the increase in long-term post-accident temperatures was evaluated and determined not to adversely affect the qualification of safety-related electrical equipment with the exception of two pressure transmitters that will be replaced. The normal temperature, pressure, and humidity conditions do not change significantly as a result of EPU. The current normal and post-accident radiation levels were evaluated to increase. The total integrated doses (normal plus accident) for EPU conditions were evaluated and determined not to adversely affect qualification of the EQ equipment located outside of containment with the exception of the two pressure transmitters that will be replaced. The evaluation of the environmental parameter changes for EPU is provided separately.

Conclusion

NSPM has evaluated the effects of the proposed EPU on the environmental conditions for the qualification of electrical equipment. The evaluation indicates that the electrical equipment will continue to meet the relevant requirements of 10 CFR 50.49 following implementation of the proposed EPU. Therefore, the proposed EPU is 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's acceptance criteria for offsite power systems are based on GDC-17.

Specific NRC review criteria are contained in SRP Sections 8.1 and 8.2, Appendix A to SRP Section 8.2, and Branch Technical Positions PSB-1 and ICSB-11.

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC listed in the Regulatory Evaluation above, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-24 and draft GDC-39.

The Monticello offsite power system is described in Monticello USAR Section 8.2, "Transmission System." Monticello is not licensed to the GDC-17 design criteria.

Monticello's Offsite Power System was evaluated for License Renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Monticello Nuclear Generating Plant License Renewal Safety Evaluation Report (SER), NUREG-1865, dated October 2006 (Reference 5). The offsite power system is discussed in the NUREG-1865, Sections 2.5.1.9 and 3.6.

Technical Evaluation

The Monticello alternating current (AC) power supply includes both offsite and onsite power supplies. The on-site power distribution system consists of transformers, buses, and switchgear. AC power to the distribution system is provided from the transmission system or from onsite Diesel Generators. The EPU effect on the onsite AC power system is addressed in Section 2.3.3 and the EPU effect on the onsite DC electrical system is addressed in Section 2.3.4.

The off-site power system starts at the output from the main generator and includes the isolated phase bus, the main transformer, switchyard, power grid and off-site power supplies to the switchyard. Plant electrical characteristics are given in Tables 2.3-1 and 2.3-2. Monticello USAR Section 8.2 describes the separation and independence of the offsite power supplies. The detailed description of the network interconnections will change with the electrical modifications planned for EPU, however, the adequacy of the independence and separation of the offsite power supplies will be maintained.

EPU can affect the grid and AC loads served by the offsite power supply system. Several modifications to 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 uprated plant electrical output as shown in Table 2.3-2. The review concluded the following:

- The continuous current rating of the isolated phase bus will be upgraded from 18.7 kA to support the higher Generator output of 19.834 kA at EPU condition. This will be accomplished by modification of the forced air-cooling system.
- The main transformer will be replaced for EPU operation and 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.
- With modification of the 1R & 2R supplies and onsite non-safety distribution, the offsite AC power sources will be adequate to accomplish required ECCS functions under postulated design basis accident conditions with the 115KV & 345KV grid voltages within the operating limits described in USAR Section 8.10.
- The Midwest Independent Transmission System Operator, Inc. performed a grid stability study for the increased EPU generator output. The results indicate that the electrical output can be increased without compromising the capability of the off-site power sources supplying the in-plant loads. A summary of this study is provided as a separate enclosure in the License Amendment Request.

The current licensing basis addresses onsite and offsite electrical supply and distribution systems for safety-related components. There is no significant effect on grid stability or reliability. There is no increase in safety-related loads at EPU conditions.

At EPU conditions, the modified Offsite Power System will have sufficient capacity to start and operate the required safety-related AC loads that are postulated to operate during design basis events. The capacity of the offsite sources will be such that a degraded bus voltage transfer will not occur for the limiting design basis load cases.

At EPU conditions, the modified Offsite Power System will supply power within the existing design voltage ranges for starting and steady state operation of AC electrical equipment, selective coordination will be maintained, and steady state currents and fault currents will be within the design ratings of the AC electrical equipment.

Conclusion

NSPM has evaluated the effects of the proposed EPU on the offsite power system. The evaluation indicates that with some modifications, the offsite power system will continue to meet the requirements of the current licensing basis following implementation of the proposed EPU. Adequate physical and electrical separation exists and will be maintained via the modifications. The offsite power system will have the capacity and capability to supply power to all safety loads and other required equipment. Therefore, the proposed EPU is acceptable with respect to the offsite power system.

2.3.3 Onsite AC Power System

Regulatory Evaluation

The alternating current (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's acceptance criteria for the AC onsite power system are based on GDC-17, insofar as it requires the system to have the capacity and capability to perform its intended functions during anticipated operational occurrences and accident conditions.

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

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release.

As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC listed in the Regulatory Evaluation above, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-24 and draft GDC-39.

The Monticello onsite AC power system is described in Monticello USAR Sections 8.3, "Auxiliary Power System," and 8.4, "Plant Standby Diesel Generator Systems."

Monticello's onsite power system was evaluated for License Renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Monticello Nuclear Generating Plant License Renewal Safety Evaluation Report (SER), NUREG-1865, dated October 2006 (Reference 5). The AC onsite power system was determined to be within the scope of license renewal and the components subject to age management review are evaluated on a plant wide basis as commodities. The electrical commodity groups are described in NUREG-1865, Section 2.5.2, and aging management for electrical commodities is described in NUREG-1865, Section 3.6.2.3. The onsite power supplies are described in NUREG-1865, Section 2.3.3.6. Aging management for onsite power supplies is documented in NUREG-1865, Section 3.3.2.6.

Technical Evaluation

The Emergency Diesel Generator (EDG) and Class 1E Uninterruptible Power Supply (UPS) provide power to essential AC loads including adequate distribution, protection, and control for design basis events with a simultaneous loss of offsite power (LOOP). The essential AC System provides power distribution and control of loads during these events.

EPU Effect on the Emergency Diesel Generator (EDG) System

EPU 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 hp 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 hp requirements. The pump operating hp 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.

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 the 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 EDG design basis loading is not affected by EPU. The EDG continuous load rating of 2500kW envelopes the initial and steady state loading. In addition, EDG transient voltage and frequency performance is not affected.

Essential AC System during EDG Operation

There are no EPU changes to the ratings for safety-related loads and no new safety-related loads normally powered from the EDG. The control logic for bus transfers and load shedding do not change including control logic for loads that operate during a DBA LOCA or an Appendix R fire event.

Some electrical supply and distribution equipment within the Monticello essential AC system may change to maintain protection and coordination and maintain voltages from offsite sources. Any changes will be to accommodate increases in non-safety related loads that may affect safety-related loads (e.g., voltage range, frequency range or short circuit capability). These changes will be performed in accordance with the Monticello modification process to assure that the changes are in accordance with the design basis for the essential AC System including system operation during design events that require EDG operation.

Class 1E UPS System

The Class 1E UPS System is capable of supplying adequate power to the connected AC loads. The 250V DC Battery remains capable of supplying adequate power to the inverters under EPU conditions for postulated design basis events.

At EPU conditions, there are no increases in safety-related loads and no new safety-related loads powered from the Class 1E UPS System. The inverter loads are the only AC loads that operate during the four hour SBO coping period. There are no increases to the inverter loads that are postulated to operate during a SBO at EPU conditions. There are no increases to the AC loads, including inverter loads that are assumed to operate during an Appendix R event. There are no increases to the operation, loads, or service conditions of the Class 1E UPS Systems under EPU conditions.

EPU has no effect on the functional requirements for the instrumentation and control subsystems of the safety-related UPS System and there are no changes to the instrumentation and control systems of the essential AC systems.

No increase in flow or pressure is required of any AC-powered ECCS equipment for EPU. Therefore, the amount of power required of the UPS to perform safety-related functions is not increased with EPU, and the current emergency UPS System remains adequate. The system has sufficient capacity; to support required loads for safe shutdown, to maintain a safe shutdown condition, and to operate the engineered safety feature equipment following postulated accidents.

Conclusion

NSPM has evaluated the effects of the proposed EPU on the AC onsite power system including the effects of the proposed EPU on the system's functional design. The evaluation indicates that the AC onsite power system will continue to meet the requirements of Monticello's current licensing basis following implementation of the proposed EPU. Therefore, the proposed EPU is 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's acceptance criteria for the DC onsite power system are based on GDC-17, insofar as it requires the system to have the capacity and capability to perform its intended functions during anticipated operational occurrences and accident conditions.

Specific NRC review criteria are contained in SRP Sections 8.1 and 8.3.2.

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC listed in the Regulatory Evaluation above, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-24 and draft GDC-39.

The Monticello onsite DC power system is described in Monticello USAR Section 8.5, "DC Power Supply Systems."

Monticello's onsite DC power system was evaluated for License Renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Monticello Nuclear Generating Plant License Renewal Safety Evaluation Report (SER), NUREG-1865, dated October 2006 (Reference 5). The onsite DC power system was determined to be within the scope of license renewal and the components subject to age management review are evaluated on a plant wide basis as commodities. The onsite DC power supplies are described in NUREG-1865, Section 2.5.1.6. The electrical commodity groups are described in NUREG-1865, Section 2.5.2, and aging management for electrical commodities is described in NUREG-1865, Section 3.6.2.3.

Technical Evaluation

The Monticello direct current (DC) power distribution system provides control and motive power for various systems and components within the plant. The DC loads are used as inputs for the computation of system load, equipment voltage drop, and available short circuit current. The DC loading and battery requirements were reviewed for the design basis worst-case loading scenario, which occurs during Station Blackout (SBO) mitigation.

The changes to the DC system under EPU conditions have been evaluated and are not significant. At EPU conditions, the integrated safety-related and SBO DC loads are not increasing and remain bounded by the existing battery capacity. The EPU changes do not increase the magnitude of the individual DC loads, but do result in changes to the timing of certain loads, such as the loads that support HPCI System operation. The timing changes were incorporated into the DC system load profile and evaluated against the DC System design criteria. The evaluation demonstrated that the changes do not result in significant reductions in battery capacity margin and do not reduce DC equipment voltages below operating limits.

The safety-related Monticello DC power systems will continue to have sufficient capacity to start and operate all connected DC loads that are postulated to operate during design basis events and the system will continue to meet all applicable design criteria for battery capacity and DC equipment operation with adequate margin under EPU conditions.

The DC Power System may be slightly changed to accommodate EPU equipment modifications. For instance, the anticipated modification to the 4kV switchgear that supplies the reactor feedwater motors may require a small change to the DC Power System such as an additional relay load or a change to a fuse setting. These types of changes regularly occur during normal plant operation and maintenance. Existing plant design processes control these changes. The associated plant changes require detailed design reviews that include an evaluation of the effect, if any, on the DC System. The design reviews provide assurance that the EPU modifications will be completed without significantly reducing the DC System design margins.

Conclusion

NSPM has evaluated the effects of the proposed EPU on the onsite DC power system and has accounted for the effects of the proposed EPU on the system's functional design. The evaluation indicates that the DC onsite power system will continue to meet the requirements of the current licensing basis following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to the DC onsite power system.

2.3.5 Station Blackout

Regulatory Evaluation

Station blackout (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 AC sources".

The NRC's acceptance criteria for SBO are based on 10 CFR 50.63.

Specific NRC review criteria are contained in SRP Section 8.1 and Appendix B to SRP Section 8.2; and other guidance provided in Matrix 3 of RS-001.

Monticello Current Licensing Basis

The licensing basis for station blackout is described in Monticello USAR Section 8.12, "Station Blackout."

Station blackout coping equipment was evaluated for License Renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Monticello Nuclear Generating Plant License Renewal Safety Evaluation Report (SER), NUREG-1865, dated October 2006 (Reference 5). Station blackout is discussed in NUREG-1865, Sections 2.4.9 and 2.5.2.4. The station blackout coping equipment was determined to be within the scope of license renewal and the components subject to age management review are evaluated on a plant wide basis as commodities. The electrical commodity groups are described in NUREG-1865, Section 2.5.2, and aging management for electrical commodities is described in NUREG-1865, Section 3.6.

Technical Evaluation

The SBO containment response was evaluated at EPU conditions using the SHEX computer code and the guidance provided by NUMARC 87-00, "Guidelines and Technical Bases for

NUMARC Initiatives Addressing Station Blackout at Light Water Reactors," (Reference 35) and Regulatory Guide 1.155, Station Blackout. The CLTP SBO containment response was evaluated using the Modular Accident Analysis (MAAP) code. The input to the SBO analysis included the following assumptions.

- Initial reactor power is 2004 MWt.
- Decay heat is EOC (24 month) GE 14 fuel.
- The HPCI system is the only credited injection source.
- An MSIV closure signal is generated at t = 0. No credit is taken for decay heat removal via the turbine bypass valves after the MSIVs close.
- Recirculation pump seal leakage is 18 gpm per pump.
- The primary HPCI suction source is the CST in accordance with the Monticello Emergency Operating Procedures (EOPs).
- Automatic and manual CST-torus HPCI suction transfers are included in the model.
- The model includes cases for automatic initiation of the HPCI System on low low reactor level or high drywell pressure.

The sequence of events for the SBO are provided in Table 2.3-3.

The plant response to and coping capabilities for an SBO event are affected slightly by operation at EPU RTP, due to the increase in the initial power level and decay heat. There are no changes to the systems and equipment used to respond to an SBO, nor is the required coping time changed.

Areas containing equipment necessary to cope with an SBO event were evaluated for the effect of loss-of-ventilation due to an SBO. The evaluation shows that equipment operability is bounded due to conservatism in the existing design and qualification bases. The battery capacity remains adequate to support HPCI operation after EPU. Adequate compressed gas capacity exists to support the SRV actuations.

The required condensate inventory for decay heat removal calculated using the method of NUMARC 87-00 Section 7.2.1 (Reference 35), is 44,329 gallons. This value is within the available CST inventory. The available Condensate Storage Tank (CST) inventory provides adequate water volume to remove decay heat and maintain reactor vessel level above the top of active fuel. Peak containment pressure and temperature remain within design values.

Consistent with the DBA LOCA condition, the required NPSH margin for the ECCS pumps has been evaluated (see Section 2.8.5.6.2) and a component acceptability review has been completed (see Section 2.8.4.4).

Based on the above evaluations, Monticello continues to meet the requirements of 10 CFR 50.63 after the EPU.

Conclusion

NSPM has evaluated 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 evaluation indicates that the plant will continue to meet the requirements of 10 CFR 50.63 following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to SBO.

Table 2.3-1 EPU Electrical Characteristics

Parameter	EPU
Guaranteed Generator Output (MWe)	691.4
Rated Voltage (kV)	22
Power Factor	0.963
Guaranteed Generator Output (MVA)	718
Current Output (kA)	18.8
Isolated Phase Bus Duct Rating (kA)	19.834*
Main Transformer Rating (MVA)	800
EPU Transformer Output (MVA)	745

* Generator rated output at 0.95 pu (per unit) rated voltage.

Table 2.3-2 Offsite Electric Power System

Component	Rating	EPU Output
Generator (MVA)	718	718
Isolated Phase Bus Duct (kA)	≥ 19.834[1]	19.834[2]
Main Transformer (MVA)	800[3]	745
Auxiliary Transformer (MVA)	N/A	N/A
Switchyard (limiting) (MVA)	848	745[4]

Notes:

1 Modification of the isolated phase bus is required to support EPU output

2 Generator rated output at 0.95 pu (per unit) rated voltage.

3 Rating indicated is for the replacement Main Transformer required for EPU

4 EPU output with a leading power factor of 0.95

Table 2.3-3 Monticello EPU Station Blackout Sequence of Events

Time (sec)	Event or Action Description
0	Reactor scram. Feedwater trip. MSIV positions to close at 0 seconds. SRVs automatically cycle for pressure control. Automatic operation of HPCI to maintain reactor water level (L2-L8) until end of coping period t = 14,400 Sec (4 hrs). HPCI suction is aligned to CST for first and second injection cycles and is aligned to SP for the third injection cycle.
3.0	MSIV valve shut
5.0	Feedwater Flow = 0
5.5	SRV Open
6.4	SRV Open
7.5	SRV Open
44.9	SRV Shut
49.6	SRV Shut
63.5	SRV Shut
159.9	SRV Open
225.6	SRV Shut
336.0	SRV Open
396.0	SRV Shut
516.0	SRV Open
572.6	SRV Shut
586.9	HPCI Aligned to CST On
866.3	HPCI Off
2083.7	SRV Open
2138.4	SRV Shut
2354.0	SRV Open
2409.8	SRV Shut
2623.9	SRV Open
2675.9	SRV Shut
2890.3	SRV Open
2939.0	SRV Shut
3148.3	SRV Open
3201.7	SRV Shut
3413.1	SRV Open
3463.2	SRV Shut
3677.4	SRV Open
3726.2	SRV Shut
3942.8	SRV Open
3989.7	SRV Shut
4204.0	SRV Open
4212.5	HPCI Aligned to CST On

Time (sec)	Event or Action Description
4227.8	SRV Shut
4509.0	HPCI Off
6573.2	SRV Open
6625.9	SRV Shut
6925.3	SRV Open
6978.8	SRV Shut
7274.2	SRV Open
7326.4	SRV Shut
7614.3	SRV Open
7663.7	SRV Shut
7949.5	SRV Open
7999.6	SRV Shut
8285.3	SRV Open
8333.6	SRV Shut
8612.4	SRV Open
8661.0	SRV Shut
8932.7	SRV Open
8978.7	SRV Shut
9249.2	SRV Open
9257.1	HPCI Aligned to SP On
9274.3	SRV Shut
9550.8	HPCI OFF
11969.2	SRV Open
12022.8	SRV Shut
12372.0	SRV Open
12423.4	SRV Shut
12763.2	SRV Open
12813.0	SRV Shut
13144.5	SRV Open
13193.7	SRV Shut
13516.2	SRV Open
13563.5	SRV Shut
13884.8	SRV Open
13931.3	SRV Shut
14256.3	SRV Open
14300.6	SRV Shut

2.4 Instrumentation and Controls

2.4.1 Reactor Protection, Safety Features Actuation, and Control Systems

Regulatory Evaluation

Instrumentation and control systems are provided (1) to control plant processes having a significant effect on plant safety, (2) to initiate the reactivity control system (including control rods), (3) to initiate the engineered safety features (ESF) systems and essential auxiliary supporting systems, and (4) for use to achieve and maintain a safe shutdown condition of the plant. Diverse instrumentation and control systems and equipment are provided for the express purpose of protecting against potential common-mode failures of instrumentation and control protection systems.

The NRC's acceptance criteria related to the quality of design of protection and control systems are based on 10 CFR 50.55a(a)(1), 10 CFR 50.55a(h), and GDCs 1, 4, 13, 19, 20, 21, 22, 23, and 24.

Specific NRC review criteria are contained in SRP Sections 7.0, 7.2, 7.3, 7.4, 7.7, and 7.8.

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of the design basis of Monticello Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC listed in the Regulatory Evaluation above, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-1, draft GDC-5, draft GDC-11, draft GDC-12, draft GDC-13, draft GDC-14, draft GDC-15, draft GDC-19, draft GDC-20, draft GDC-22, draft GDC-23, draft GDC-25, draft GDC-26, draft GDC-40 and draft GDC-42. Current GDC-13 and current GDC-20 are applicable to Monticello for certain events as described in USAR Chapter 14. Current GDC-13 is applicable to Monticello as described in USAR Section 14.7.4. Current GDC-19 is applicable to Monticello as described in USAR Sections 5.3.5, 6.7.2, 12.3.1.6, and 14.7. Current GDC-20 is applicable to Monticello as described in USAR Section 14.4.3.3.

Monticello instrumentation and control systems are described in Monticello USAR Section 7, "Plant Instrumentation and Control Systems."

Monticello's instrumentation and control systems were evaluated for license renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Monticello Nuclear Generating Plant License Renewal Safety Evaluation Report (SER), NUREG-1865, dated October 2006 (Reference 5). The instrumentation and control systems were determined to be within the scope of license renewal and the components subject to age management review are evaluated on a plant wide basis as commodities. The electrical commodity groups are described in NUREG-1865, Section 2.5.2, and aging management for electrical commodities is described in NUREG-1865, Section 3.6.

Technical Evaluation

2.4.1.1 Nuclear Steam Supply System Monitoring and Control Instrumentation

The instruments and controls used to monitor and directly interact with or control reactor parameters are usually within the NSSS. Changes in process variables and their effects on instrument performance and setpoints were evaluated for EPU operation to determine any related changes. Process variable changes are implemented through changes in normal plant operating procedures. Technical Specifications address instrument Allowable Values (AV) and/or setpoints for those NSSS sensed variables that initiate protective actions. The effects of EPU on Technical Specifications are addressed in Section 2.4.1.3.

EPU affects the performance of the Neutron Monitoring System and is generically dispositioned in the CLTR. These performance effects are associated with the Average Power Range Monitors (APRMs), Intermediate Range Monitors (IRMs), Local Power Range Monitors (LPRMs), Rod Block Monitor (RBM), and Rod Worth Minimizer (RWM).

Average Power Range Monitors, Intermediate Range Monitors and Source Range Monitors

[[

[illegible]	[illegible]	[illegible]
		]]

The increase in power level due to EPU increases the average flux in the core and at the in-core detectors. The APRM power signals are calibrated to read 100% at the EPU RTP level. EPU has little effect on the IRM overlap with the SRMs and the APRMs. Using normal plant surveillance procedures, the IRMs may be adjusted, as required, so that overlap with the SRMs and APRMs remains adequate.

The SRM, IRM, and APRM Systems installed at Monticello are in accordance with the requirements established by the GEH design specifications. [[
]]

Local Power Range Monitors

[[

[illegible]	[illegible]	[illegible]
		]]

At EPU RTP, the average flux experienced by the detectors increases due to the average power increase in the core. The maximum flux experienced by an LPRM remains approximately the same because the peak bundle power does not appreciably increase. Due to the increase in neutron flux experienced by the LPRMs and traversing incore probes (TIPs), the neutronic life of the LPRM detectors may be reduced and radiation levels of the TIPs may be increased. LPRMs are designed as replaceable components. The LPRM accuracy at the increased flux is within specified limits, and LPRM lifetime is an operational consideration that is handled by routine replacement. TIPs are stored in shielded rooms. The radiation protection program for normal plant operation accommodates a small increase in radiation levels.

The LPRMs and TIPs installed at Monticello are in accordance with the requirements established by the GEH design specifications. [[
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Rod Block Monitor

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		]]

The increase in power level at the same APRM reference level results in increased flux at the LPRMs that are used as inputs to the RBM. The RBM instrumentation is referenced to an APRM channel. Because the APRM has been rescaled, there is only a small effect on the RBM performance due to the LPRM performance at the higher average local flux. The change in performance does not have a significant effect on the overall RBM performance.

The RBMs installed at Monticello are in accordance with the requirements established by the GEH design specifications. [[
]] In addition, the RBM is not credited in any safety analysis for the Monticello EPU. The RBM is not safety-related, but provides defense in depth for the response to a Rod Worth Withdrawal Error event as described in USAR 7.2.1.1.2.2.2.

Rod Worth Minimizer/Rod Control and Information System

The Rod Control and Information System (RCIS) is not applicable to Monticello.

[[

[illegible]	[illegible]	[illegible]
	]]	• Monticello uses all GE14 or earlier fuel. • Thermal power increase: OLTP – 1670 MWt EPU – 2004 MWt Thermal power increase is equal to 20% of OLTP. [[ ]]

[[ ]]

The RWM is a normal operating system that does not perform a safety-related function. The function of the RWM is to support the operator by enforcing rod patterns until reactor power has reached appropriate levels. [[ ]] The power-dependent instrument setpoints for the RWM are included in the plant Technical Specifications (see Section 2.4.1.3).

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2.4.1.2 BOP Monitoring and Control

Operation of the plant at EPU conditions has minimal effect on the BOP system instrumentation and control devices and these instruments are generically dispositioned in the CLTR. Based on EPU operating conditions for the power conversion and auxiliary systems, most process control valves and instrumentation have sufficient range/adjustment capability for use at the EPU conditions. However, some (non-safety) modifications may be needed to the power conversion systems to obtain EPU RTP. No safety-related BOP system setpoint change is required as a result of the EPU, with the exception of MSL high flow discussed in Section 2.4.1.3. The topics considered in this section are:

Pressure Control System

[[

[redacted]	[redacted]	[redacted]
		]]

The PCS is a normal operating system to provide fast and stable responses to system disturbances related to steam pressure and flow changes to control reactor pressure within its normal operating range. This system does not perform a safety function. Pressure control operational testing is included in the EPU implementation plan as described in Section 2.12 to ensure adequate turbine control valve pressure control and flow margin is available.

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Turbine Steam Bypass System

[[

[illegible]	[illegible]	[illegible]
		]]

The Turbine Steam Bypass System is a normal operating system that is used to bypass excessive steam flow. The absolute flow capacity of the bypass system is unchanged.

The Turbine Steam Bypass System at Monticello is [[ ]] the bypass valve flow capacity is not changed (in terms of lbm/hr) and the turbine steam bypass system is non-safety related.

The Turbine Steam Bypass System at Monticello is [[ ]] the bypass valve flow capacity is used in some AOO evaluations and when used in a limiting safety analyses in the reload analysis, the analysis utilize the actual bypass flow.

Feedwater Control System

[[

[illegible]	[illegible]	[illegible]
		]]

The Feedwater Control System is a normally operating system to control and maintain the reactor vessel water level. EPU results in an increase in FW flow. FW control operational testing is included in the EPU implementation plan as described in Section 2.12 to ensure that the FW response is acceptable. Failure of this system is evaluated in the reload analysis for each reload core with the FW controller failure-maximum demand event. A Loss of Feedwater (LOFW) event can be caused by downscale failure of the controls. The LOFW is discussed in Section 2.8.

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]].

Leak Detection System

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[illegible]	[illegible]	[illegible]
		]]

The only effect on the LDS due to EPU is a slight increase in the FW temperature and steam flow. [[

]] The increased FW temperature results in a small increase in the MS tunnel temperature (< 1°F). [[

]]. MSL high flow is discussed in Section 2.4.1.3.

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2.4.1.3 Technical Specification Instrument Setpoints

Technical Specifications instrument Allowable Values (AVs) and/or setpoints are those sensed variables, which initiate protective actions and are generally associated with the safety analysis. Technical Specification AVs are highly dependent on the results of the safety analysis. The safety analysis generally establishes the Analytical Limits (ALs). The determination of the Technical Specification AVs and other instrument setpoints includes consideration of measurement uncertainties and is derived from the ALs. The settings are selected with sufficient margin to minimize inadvertent initiation of the protective action, while assuring that adequate operating margin is maintained between the system settings and the actual limits. There is typically substantial margin in the safety analysis process that should be considered in establishing the setpoint process used to establish the Technical Specification AVs and other setpoints.

Increases in the core thermal power and steam flow affect some instrument setpoints. These setpoints are adjusted to maintain comparable differences between system settings and actual limits, and are reviewed to ensure that adequate operational flexibility and necessary safety functions are maintained at the EPU RTP level. Where the power increase results in new instruments being employed, an appropriate setpoint calculation is performed and Technical Specification changes are implemented, as required. If there is no change in the instrument equipment, the simplified process outlined in the CLTR may be used to determine the instrument AV and setpoint.

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Monticello has elected to not use the simplified methodology and has applied the existing setpoint methodology to the Technical Specification instrument setpoints. Monticello has implemented the GEH Setpoint Methodology (Reference 6) that has been reviewed and approved by the NRC. All Technical Specification instruments were evaluated for effects from EPU. This evaluation included a review of environmental (i.e., radiation and temperature) effects, process (i.e., measured parameter) effects and analytical (i.e., AL and margins) effects on the subject instruments.

Setpoint Evaluations

Table 2.4-1 summarizes the affected current and EPU ALs and AVs for Monticello.

The Setpoint Calculation Methodology for each topic addressed in the section is generically dispositioned in the CLTR while the setpoint value is plant specifically determined.

APRM Simulated Thermal Power (STP) Scram

EPU maintains the MELLLA CLTP slope of the APRM STP scram AL lines in terms of absolute core power versus recirculation drive flow. This slope in terms of relative core power and recirculation drive flow at the EPU condition was obtained by applying the ratio of the CLTP to the EPU RTP to the CLTP slope in the STP scram equation. The APRM STP scram AL intercept is adjusted to match the associated clamped AL at the minimum operating core flow (approximating the recirculation drive flow) corresponding to the new rated power. In effect, the APRM STP setpoint rescaling maintains comparable margin between the operating region and the APRM trips. The clamped setpoint retains its current value (in percent power). See table 2.4-1.

APRM Simulated Thermal Power Control Rod Block

APRM Control Rod Block setpoints were revised similarly to the SCRAM setpoint changes described above.

APRM Setdown in Startup Mode

The value for the Technical Specification safety limit for reduced pressure or low core flow condition, is established to satisfy the fuel thermal limits monitoring requirements.

APRM Setdown setpoint is unchanged in terms of percentage of rated power at EPU conditions.

Rod Block Monitor

The severity of the rod withdrawal error (RWE) during power operation event is dependent upon the RBM rod block setpoint. This setpoint is only applicable to the control rod withdrawal error. The power-dependent setpoints for EPU are maintained at the same percent power. The cycle specific reload analysis establishes the rod block setpoint.

The value in the plant Technical Requirements Manual for the RBM power-biased setpoints is maintained the same in terms of percent power. The trip setpoints (corresponding to the various power dependent setpoint levels) are evaluated as part of the cycle specific reload analysis. Therefore the CLTR generic disposition of this setpoint is applicable to this setpoint. RBM rod block setpoint will be established by the reload analysis for the first operating cycle at EPU.

Rod Worth Minimizer Setpoint

The Rod Worth Minimizer LPSP is used to bypass the rod pattern constraints established for the control rod drop accident (CRDA) at greater than a pre-established low power level. The measurement parameters are FW and steam flow.

For EPU, the LPSP in the plant Technical Specifications is conservatively kept at the same value in terms of percent power and no Technical Specification change is required. This approach does not affect the limitations on the sequence of control rod movement to the absolute core power level for the LPSP associated with the requirements of the CRDA.

Main Steam Line High Flow Isolation

EPU results in increased steam flow at the increased power level.

The MSL high flow isolation setpoint is used to initiate the isolation of the Group 1 primary containment isolation valves. The MSLBA is the only safety analysis event that credits this trip. For the MSLBA, there are diverse trips from high area temperatures. For Monticello, the choke flow point for the main steamline flow restrictors is used as the AL for the setpoint and for the value of mass released in the evaluation of the radiological consequences of the MSLBA.

A new AV has been calculated to maintain the AV at the same absolute steam flow by calculating the new value of percent steam flow at EPU conditions (123.64%) that provides the same differential pressure (151.95 psid) across the flow sensors as the current AV (142%). This approach increases margin to the AL and reduces spurious trip avoidance margin.

Turbine First-Stage Pressure Scram

EPU results in an increased power level and the high-pressure turbine (HPT) modifications result in a change to the relationship of turbine first-stage pressure to reactor power level. The turbine first-stage pressure setpoint is used to reduce scrams at low power levels where the turbine steam bypass system is effective for turbine trips (TT) and generator load rejections. In the safety analysis, this trip bypass only applies to events at low power levels that result in a TT or load rejection. The AL for EPU is maintained at approximately the same absolute power as for the current setpoint, maintaining the same transient analysis basis and scram avoidance range of the bypass valves. Transient analysis confirmed that the new AL met acceptance criteria.

Because the HPT will be modified to support achieving the EPU RTP level, a new AL (in psig) corresponding to the same absolute power as the current AL was established. Therefore, a new setpoint was calculated using the methodology described above, and the Technical Specification applicable condition in percent RTP has been changed. To assure that the new value is appropriate, EPU power ascension startup testing or normal plant surveillance will be used to validate that the actual plant interlock is cleared consistent with the safety analysis.

Reactor Water Level – Low (SCRAM)

Increased steam flow through the steam dryer creates an increased differential pressure across the steam dryer. If reactor water level drops below the level of the steam dryer skirt, resulting steam bypassing the dryer flows past the variable leg reactor water level instrument tap and creates a Bernoulli effect pressure reduction indicated as a non-conservative increase in reactor water level. The analytical limit for this setpoint has been revised applying this effect as a bias. Transient analyses confirmed that the new AL meets acceptance criteria (Reactor water level remains above Top of Active Fuel during a Loss of Feedwater Event). No change is being made to the Allowable Value or Nominal Trip Setpoint.

2.4.1.4 Changes to Instrumentation and Controls

In the CLTR SER, the staff requested that the plant specific submittal address all EPU-related changes to instrumentation and controls, such as scaling changes, changes to upgrade obsolescent instruments, and changes to the control philosophy. Table 2.4-2 provides this information. No obsolescent instrument changes are required as a result of EPU and there are no changes to instrument philosophy as a result of EPU.

Conclusion

NSPM has evaluated the effects of the proposed EPU on the functional design of the reactor trip system, safe shutdown system, and control systems. The evaluation indicates that the changes that are necessary to achieve the proposed EPU are consistent with the plant's design basis and that the systems will continue to meet the requirements of 10 CFR 50.55a(a)(1), 10 CFR 50.55a(h), and the current licensing basis. Therefore, the proposed EPU is acceptable with respect to instrumentation and controls.

Table 2.4-1 Analytical Limits and Allowable Values for Setpoints

Parameter	Current	EPU
APRM Calibration Basis (MWt)	1775	2004
APRM Neutron Flux High Scram AL	125	No Change
APRM STP (Scram) AVs[1,2]		
TLO (%RTP)	0.66W + 61.6	0.55W + 61.5
SLO (%RTP)	0.66(W-delta W) + 61.6	0.55(W-delta W) + 61.5
Clamp (%RTP)	116	No Change
APRM STP (Rod Block) AVs [1, 2]		
TLO (%RTP)	0.66W + 55.6	0.55W + 55.5
SLO (%RTP)	0.66(W-delta W) + 55.6	0.55(W-delta W) + 55.5
Clamp (%RTP)	110	No Change
APRM Setdown in Startup Mode AVs		
Scram (%RTP)	20	No Change
Rod Block (%RTP)	15	No Change
Rod Block Monitor AVs	See note 3	No Change
Rod Worth Minimizer LPSP AV (%RTP)	10	No Change
Main Steam Line High Flow Isolation AL (% rated steam flow)	142	123.5
Turbine First-Stage Pressure Scram Bypass AL (%RTP)	45.0%	40.0%
Reactor Water Level – Low (SCRAM) (inches above indicated zero) (AL)	0	-2.5

Notes:

1. No credit is taken in any safety analysis for the flow referenced setpoints.
2. The EPU APRM STP Scram and Rod Block clamps remain the same in terms of percent rated power.
3. The cycle specific reload analysis is used to determine any change in the rod block trip setpoint. The RBM trip setpoints listed are based on an Operating Limit Minimum Critical Power Ratio (OLMCPR) of 1.30.

Table 2.4-2 Changes to Instrumentation and Controls

Parameter	Change
APRM STP SCRAM and Rod Block	Rescaled for EPU conditions based on methodology described in CLTR
MSL High Flow	Setpoint changes for new setpoints to scale the isolation in % of rated steam flow to maintain current AV at the same absolute steam flow as current conditions
Turbine First-Stage Pressure Scram Bypass	Recalibrate or replace Turbine 1st Stage Pressure instruments and Validate/Verify relationship between Turbine 1st Stage Pressure and Reactor Thermal Power during startup testing following replacement of the High Pressure Turbine.

2.5 Plant Systems

2.5.1 Internal Hazards

2.5.1.1 Flooding

2.5.1.1.1 Flood Protection

Regulatory Evaluation

The NRC staff conducted a review in the area of flood protection to ensure that SSCs important to safety are protected from flooding.

The NRC's acceptance criteria for flood protection are based on GDC-2.

Specific NRC review criteria are contained in SRP Section 3.4.1.

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of the design basis of Monticello Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC listed in the Regulatory Evaluation above, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-2.

Monticello internal flooding hazards are described in Monticello USAR Section 12.2.1.7.2, "Internal Flooding."

In addition to the evaluations described in the Monticello USAR, Monticello's systems and components were evaluated for License Renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Monticello Nuclear Generating Plant License Renewal Safety Evaluation Report (SER), NUREG-1865, dated October 2006 (Reference 5). The license renewal evaluation associated with the flood protection barriers is documented in NUREG-1865, Section 2.4. During plant license renewal evaluations, tanks, and pipes which were not already in scope pursuant to 10CFR54.4(a)(1) or (a)(3) were evaluated to ensure they were not "non-safety equipment whose failure could affect a safety function" (criterion (a)(2)). Components that met the inclusion criteria were evaluated within the system that contained them. Additionally, civil features whose function was to control, abate, or minimize the effects of flooding were identified and evaluated within the structure that contained them.

Technical Evaluation

Internal Flooding from Feedwater Line Break

Plant flooding is mainly dependent upon the maximum water levels in the hotwells and not EPU reactor vessel conditions. The existing feedwater and condensate systems were evaluated for the effects of EPU and no adverse effects due to EPU were identified. The design of the plant incorporates features to ensure that a postulated internal flooding event (via spray or submergence) will not prevent safe shutdown or the availability of core cooling. USAR Section 10.3 discusses internal flooding for selected events, including torus failures that affect the ECCS corner rooms and USAR Section 12.2 discusses internal flooding for selected events that protects safe shutdown paths.

Planned feedwater and condensate system hardware changes have been evaluated using conservative estimates for the effects of these changes on line breaks. The evaluation results were acceptable. Normal hotwell level may be increased for EPU to provide additional NPSH for the condensate pumps. Condensate demineralizers may be replaced for EPU. Similarly, planned Feedwater and Condensate pump replacements may potentially increase system flow rates and pump discharge pressures. The predicted effects on mass and energy release calculations and HELB calculations from these modifications will be confirmed as the final modification designs are finalized.

Conclusion

NSPM has evaluated the effects of the proposed EPU on internal flooding hazards. The evaluation indicates 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. Therefore, the proposed EPU is acceptable with respect to flood protection.

2.5.1.1.2 Equipment and Floor Drains

Regulatory Evaluation

The function of the equipment and floor drainage system (EFDS) is to assure that waste liquids, valve and pump leakoffs, and tank drains are directed to the proper area for processing or disposal. The EFDS is designed to handle the volume of leakage expected, prevent a backflow of water that might result from maximum flood levels to areas of the plant containing safety-related equipment, and protect against the potential for inadvertent transfer of contaminated fluids to an uncontaminated drainage system.

The NRC's acceptance criteria for the EFDS are based on GDCs 2 and 4 insofar as they require the EFDS to be designed to withstand the effects of earthquakes and to be compatible with the environmental conditions (flooding) associated with normal operation, maintenance, testing, and postulated accidents (pipe failures and tank ruptures).

Specific NRC review criteria are contained in SRP Section 9.3.3.

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC listed in the Regulatory Evaluation above, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-2, draft GDC-40, and draft GDC-42.

The equipment and floor drains are described in Monticello USAR Section 10.3.6, "Plant Equipment and Floor Drainage Systems."

In addition to the evaluations described in the Monticello USAR, Monticello's systems and components were evaluated for License Renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Monticello Nuclear Generating Plant License Renewal Safety Evaluation Report (SER), NUREG-1865, dated October 2006 (Reference 5). During plant license renewal evaluations, tanks, and pipes which were not already in scope pursuant to 10CFR54.4(a)(1) or (a)(3) were evaluated to ensure they were not "non-safety equipment whose failure could affect a safety function" (criterion (a)(2)). Components that met the inclusion criteria were evaluated within the system that contained them. Additionally, civil features whose function was to control, abate, or minimize the effects of flooding were identified and evaluated within the structure that contained them.

Technical Evaluation

The floor drain collector subsystem and the waste collector subsystem both receive periodic inputs from a variety of sources. Neither subsystem is expected to experience a large increase in the total volume of liquid and solid waste due to operation at the EPU condition. Monticello has sufficient capacity to handle added liquid increases expected, i.e., it can collect and process the drain fluids. The drainage systems backflow at maximum flood levels does not change as a result of EPU. The drainage systems design capability to withstand the effects of earthquakes and to be compatible with environmental conditions does not change as a result of EPU. Therefore, EPU does not affect system operation or equipment performance.

Conclusion

NSPM has evaluated the effects of the proposed EPU on the EFDS. The evaluation indicates that the EFDS has sufficient capacity to (1) handle any additional expected leakage resulting from the plant changes, (2) does not affect the backflow of water to areas with safety-related equipment. The EFDS will continue to meet the requirements of the current licensing basis following implementation of the proposed EPU. Therefore the proposed EPU is acceptable with respect to the EFDS.

2.5.1.1.3 Circulating Water System

Regulatory Evaluation

The Circulating Water System (CWS) provides a continuous supply of cooling water to the main condenser to remove the heat rejected by the turbine cycle and auxiliary systems.

The NRC's acceptance criteria for the CWS are based on GDC-4 for the effects of flooding of safety-related areas due to leakage from the CWS and the effects of malfunction or failure of a component or piping of the CWS on the functional performance capabilities of safety-related SSCs.

Specific NRC review criteria are contained in SRP Section 10.4.5.

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC listed in the Regulatory Evaluation above, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-40 and draft GDC-42.

The Circulating Water System is described in Monticello USAR Section 11.5, "Circulating Water System."

In addition to the evaluations described in the Monticello USAR, Monticello's systems and components were evaluated for License Renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Monticello Nuclear Generating Plant License Renewal Safety Evaluation Report (SER), NUREG-1865, dated October 2006 (Reference 5). The license renewal evaluation associated with the Circulating Water System is documented in NUREG-1865, Section 2.3.3.3. Management of aging effects on the Circulating Water System is documented in NUREG-1865, Section 3.3.2.3.3.

Technical Evaluation

The main condenser, circulating water, and normal heat sink systems are not being modified for EPU operation. The performance of these systems was evaluated for EPU by assessing the effects of increasing power to 120% of OLTP on the heat rejection systems and comparing to the existing NPDES permit limits. This evaluation used plant, river and atmospheric data for the years 2001-2006 and adjusted thermal power per the EPU heat balance. Key parameters were evaluated to determine the effects of EPU operating conditions upon plant parameters. In general, the EPU effects are within the existing limits. If temperatures approach the existing limits, then reactor thermal power reduction would be required to reduce the heat rejected to the condenser and maintain temperatures within the existing limits. The effect of EPU on the flooding analyses is addressed in Section 2.5.1.1.1.

Conclusion

There are no EPU related modifications to the CWS. Performance was analyzed with respect to EPU power levels. If temperatures approach existing NPDES permit limits, a reactor power reduction would be required to reduce the heat rejected via the CWS. The effect of EPU on the flooding analyses is addressed in Section 2.5.1.1.1.

2.5.1.2 Missile Protection

2.5.1.2.1 Internally Generated Missiles

Regulatory Evaluation

The NRC staff's review concerns missiles that could result from in-plant component overspeed failures and high-pressure system ruptures.

The NRC's acceptance criteria for the protection of SSCs important to safety against the effects of internally generated missiles that may result from equipment failures are based on GDC-4.

Specific NRC review criteria are contained in SRP Sections 3.5.1.1 and 3.5.1.2.

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC listed in the Regulatory Evaluation above, the Monticello comparative evaluation of the comparable 1967 AEC proposed

General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-40 and draft GDC-42.

The missile protection for internally generated missiles is described in Monticello USAR Sections 5.2.3.5.3, "Drywell Missile Protection," and 12.2.3, "Turbine Missile Analysis."

In addition to the evaluations described in the Monticello USAR, Monticello's systems and components were evaluated for License Renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Monticello Nuclear Generating Plant License Renewal Safety Evaluation Report (SER), NUREG-1865, dated October 2006 (Reference 5). The equipment and components credited with mitigating the effect of missiles are documented in NUREG-1865, Section 2.4, and the programs credited with managing that equipment aging are documented in NUREG-1865, Section 3.5.

Technical Evaluation

This review criterion is applicable to EPUs that result in substantially higher system pressures or changes in existing system configuration. The Monticello EPU does not result in any condition (system pressure increase or equipment overspeed) that could result in an increase in the generation of internally generated missiles. In addition, the Monticello EPU does not entail any modifications that change the effect of internally generated missiles on safety-related or non-safety related equipment.

Conclusion

NSPM has evaluated changes in system pressures, configurations, and equipment rotational speeds necessary to support the proposed EPU. The evaluation indicates that SSCs important to safety will continue to be protected from the effects of internally generated missiles in accordance with current licensing basis assumptions. Therefore, the proposed EPU is acceptable with respect to the protection of SSCs important to safety from internally generated missiles.

2.5.1.2.2 Turbine Generator

Regulatory Evaluation

The turbine control system, steam inlet stop and control valves, low pressure turbine steam intercept and inlet control valves control the speed of the turbine under normal and abnormal conditions, and are thus related to the overall safe operation of the plant.

The NRC's acceptance criteria for the turbine generator are based on GDC-4, and relates to protection of SSCs important to safety from the effects of turbine missiles by providing a turbine overspeed protection system (with suitable redundancy) to minimize the probability of generating turbine missiles.

Specific NRC review criteria are contained in SRP Section 10.2.

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC listed in the Regulatory Evaluation above, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-40 and draft GDC-42.

The turbine generator is described in Monticello USAR Sections 11.2, "Turbine-Generator System," and 7.7, "Turbine-Generator System Instrumentation and Control."

In addition to the evaluations described in the Monticello USAR, Monticello's systems and components were evaluated for License Renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Monticello Nuclear Generating Plant License Renewal Safety Evaluation Report (SER), NUREG-1865, dated October 2006 (Reference 5). The license renewal evaluation associated with the turbine generator is documented in NUREG-1865, Section 2.3.4.5. Management of aging effects on the turbine generator is documented in NUREG-1865, Section 3.4.2.3.5.

Technical Evaluation

The turbine and generator were originally designed with a maximum flow-passing capability and generator output in excess of rated conditions to ensure that the original rated steam-passing capability and generator output is achieved. This excess design capacity ensures that the turbine and generator meet rated conditions for continuous operating capability with allowances for variations in flow coefficients from expected values, manufacturing tolerances, and other variables that may adversely affect the flow-passing capability of the units. The difference in the steam-passing capability between the design condition and the rated condition is called the flow margin.

The current high-pressure turbine is designed for operation at CLTP with a flow margin of approximately 3%. The current rated throttle steam flow is 7.22 Mlb/hr at a throttle pressure of 965 psia. The current generator rating is 664.4 MVA, which results in a rated electrical output (gross) of 631.2 MWe at a power factor of 0.95.

At the EPU RTP and reactor dome pressure of 1025 psia, the turbine operates at an increased rated throttle steam flow of 8.31 Mlb/hr and at a throttle pressure of 952 psia. To maintain control capability GE uses a minimum target value of approximately 97% throttle flow ratio, with controllability confirmed by unit testing as described in Section 2.12. For operation at EPU, the high-pressure turbine has been redesigned with new diaphragms and buckets with a throttle flow margin of approximately 3%, to increase its flow passing capability. The turbine main steam stop valves are designed such that the steam flow assists their closure and therefore maintain their capability to close with increased main steam flow.

The generator will be uprated to 718 MVA at 0.963 power factor for EPU. A new stator winding, improved core end baffles, and new hydrogen coolers will be installed for EPU.

The expected environmental changes such as daytime to nighttime heating and cooling effects changing cycle efficiency could periodically require management of reactor power to remain within the generator rating. The required variations in reactor power do not approach the magnitude of changes periodically required for surveillance testing and rod pattern alignments and other occasional events requiring de-rating, such as equipment out of service for maintenance.

The high-pressure and low-pressure turbine rotors at Monticello (for both CLTP and EPU RTP) 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, and the sensitivity of the rotor train for the capability of overspeeding. 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 Monticello EPU does not result in increases in system pressures, configurations, or equipment overspeed that would affect the evaluation of internally generated missiles on safety-related or nonsafety-related equipment.

Conclusion

NSPM has evaluated the effects of the proposed EPU on the turbine generator. The evaluation indicates that the turbine generator will continue to provide adequate turbine overspeed protection to minimize the probability of generating turbine missiles and will continue to meet the requirements of the current licensing basis following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to the turbine generator.

2.5.1.3 Pipe Failures

Regulatory Evaluation

The NRC staff conducted a review of the plant design for protection from piping failures outside containment to ensure that (1) such failures would not cause the loss of needed functions of safety-related systems and (2) the plant could be safely shut down in the event of such failures.

The NRC's acceptance criteria for pipe failures are based on GDC-4, which requires, in part, that SSCs important to safety be designed to accommodate the dynamic effects of postulated pipe ruptures, including the effects of pipe whipping and discharging fluids.

Specific NRC review criteria are contained in SRP Section 3.6.1.

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC listed in the Regulatory Evaluation above, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-40 and draft GDC-42.

Piping failures outside containment are described in Monticello USAR Appendix I, "Evaluation of High Energy Line Breaks Outside Containment."

Technical Evaluation

Piping failures outside containment are addressed in Sections 2.2.2.1, Pipe Whip and Jet Impingement, 2.2.5, Seismic and Dynamic Qualification of Mechanical and Electrical Equipment, 2.3.1, Environmental Qualification of Electrical Equipment, and 2.9.2, Radiological Consequences Analyses Using Alternative Source Terms.

Conclusion

NSPM has evaluated the changes that are necessary for the proposed EPU. The evaluation indicates that SSCs important to safety will continue to be protected from the dynamic effects of postulated piping failures in fluid systems outside containment and will continue to meet the requirements of the current licensing basis following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to protection against postulated piping failures in fluid systems outside containment.

2.5.1.4 Fire Protection

Regulatory Evaluation

The purpose of the Fire Protection Program (FPP) is to provide assurance, through a defense-in-depth design, that a fire will not prevent the performance of necessary safe plant shutdown functions and will not significantly increase the risk of radioactive releases to the environment.

The NRC's acceptance criteria for the FPP are based on (1) 10 CFR 50.48 and associated Appendix R to 10 CFR Part 50, insofar as they require the development of an FPP to ensure, among other things, the capability to safely shut down the plant; (2) GDC-3, 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-5, 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 NRC review criteria are contained in SRP Section 9.5.1, as supplemented by the guidance provided in Attachment 2 to Matrix 5 of Section 2.1 of RS-001.

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of

the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC listed in the Regulatory Evaluation above, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-3 and draft GDC-4.

Fire Protection is described in Monticello USAR Appendix J, "Fire Protection Program."

In addition to the evaluations described in the Monticello USAR, Monticello's systems and components were evaluated for License Renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Monticello Nuclear Generating Plant License Renewal Safety Evaluation Report (SER), NUREG-1865, dated October 2006 (Reference 5). The fire protection systems are documented in NUREG-1865, Section 2.3.3.9. Fire barrier materials are addressed as a commodity group, while walls, floors, doors, structural steel etc., are evaluated within the building that contains them. Components credited with achieving safe shutdown following a fire are evaluated within the system that contains them. Management of aging effects on the fire protection systems is documented in NUREG-1865, Section 3.3.2.3.9.

Technical Evaluation

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]] Any changes in physical plant configuration or combustible loading as a result of modifications to implement the EPU, will be evaluated in accordance with the plant modification and fire protection programs. The safe shutdown systems and equipment used to achieve and maintain cold shutdown conditions do not change, and are adequate for the EPU conditions. The operator actions required to mitigate the consequences of a fire are not affected. Therefore, the fire protection systems and analyses are not affected by EPU.

The reactor and containment response to the postulated 10 CFR 50 Appendix R fire event at EPU conditions was evaluated. The results show that the peak fuel cladding temperature, reactor pressure, and containment pressures and temperatures are below the acceptance limits and demonstrate that there is sufficient time available for the operators to perform the necessary actions to achieve and maintain cold shutdown conditions.

A plant specific evaluation was performed to demonstrate safe shutdown capability in compliance with the requirements of 10 CFR 50 Appendix R assuming EPU conditions. The limiting Appendix R fire event from the current analysis was reanalyzed assuming EPU. The fuel heatup analysis was performed using the SAFER/GESTR-LOCA analysis model. The containment analysis was performed using the SHEX model. These are the same analysis methodologies that were used for the existing Appendix R Fire event analysis at Monticello. This evaluation determined the effect of EPU on fuel cladding integrity, reactor vessel integrity, and containment integrity as a result of the fire event.

The postulated Appendix R fire event using the Safe Shutdown System was analyzed for the two cases described below, consistent with the current analysis basis:

> Case 1: One Relief Valve (RV) is assumed to be stuck open until the manual opening of the two SRVs 17 minutes into the event. This case assumes one Core Spray (CS) pump and one Residual Heat Removal (RHR) pump with one heat exchanger operating. Suppression Pool Cooling is initiated manually at 40 minutes into the event.
>
> Case 2: No stuck open RV is assumed. 2 SRVs are used to depressurize the vessel after being initiated manually 17 minutes into the event. This case assumes one Core Spray (CS) pump and one Residual Heat Removal (RHR) pump with one heat exchanger operating. Suppression Pool Cooling is initiated manually at 40 minutes into the event.

These cases were evaluated for EPU with no addition of any new operator actions. The operator action times were not changed from the Reference 7 evaluation (40 minutes).

The results of the Appendix R evaluation for EPU provided in Table 2.5-1 demonstrate that the fuel cladding integrity, reactor vessel integrity, and containment integrity are maintained and that sufficient time is available for the operator to perform the necessary actions. EPU does not affect any exemptions described in the current Monticello Appendix R Fire Protection Analysis (Reference 7). No changes are necessary to the equipment required for safe shutdown for the Appendix R event. One train of systems remains available to achieve and maintain safe shutdown conditions from either the main control room or the remote shutdown panel. Therefore, EPU has no adverse effect on the ability of the systems and personnel to mitigate the effects of an Appendix R fire event, and satisfies the requirements of Appendix R with respect to achieving and maintaining safe shutdown in the event of a fire.

Conclusion

NSPM has evaluated fire-related safe shutdown requirements and has accounted for the effects of the increased decay heat on the ability of the required systems to achieve and maintain safe shutdown conditions. The evaluation indicates that the FPP will continue to meet the requirements of 10 CFR 50.48, Appendix R to 10 CFR Part 50, and the current licensing basis following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to fire protection.

2.5.2 Fission Product Control

2.5.2.1 Fission Product Control Systems and Structures

Regulatory Evaluation

The NRC's acceptance criteria are based on GDC-41, insofar as it requires that the containment atmosphere cleanup system be provided to reduce the concentration of fission products released to the environment following postulated accidents.

Specific NRC review criteria are contained in SRP Section 6.5.3.

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. However, there is not a draft GDC directly associated with the current GDC listed in the Regulatory Evaluation above (GDC-41).

The Standby Gas Treatment System is described in Monticello USAR Section 5.3.4.1, "Standby Gas Treatment System (SGTS)."

In addition to the evaluations described in the Monticello USAR, Monticello's systems and components were evaluated for License Renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Monticello Nuclear Generating Plant License Renewal Safety Evaluation Report (SER), NUREG-1865, dated October 2006 (Reference 5). The license renewal evaluation associated with the Standby Gas Treatment System is documented in NUREG-1865, Section 2.3.2.8. Management of aging effects on the Standby Gas Treatment System is documented in NUREG-1865, Section 3.2.2.3.8.

Technical Evaluation

The Standby Gas Treatment System (SGTS) is designed to maintain secondary containment at a negative pressure and to filter the exhaust air for removal of fission products potentially present during abnormal conditions. By limiting the release of airborne particulates and halogens, the SGTS limits off-site dose following a postulated DBA. Flow capacity of the SGTS is discussed in Section 2.6.6.

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The charcoal adsorber removal efficiency for radioiodine is not affected by EPU. The total (radioactive plus stable) post-LOCA iodine loading on the charcoal adsorbers increases proportionally with the increase in core iodine inventory, which is proportional to core thermal power. Sufficient charcoal mass is present so that the post-LOCA iodine loading on the charcoal remains below the guidance provided by RG 1.52.

While decay heat from fission products accumulated within the system filters and charcoal adsorbers increases in proportion to the increase in thermal power, the cooling air flow required to maintain components below operating temperature limits is well below the cooling flow capability of the system.

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The results of the bounding AST evaluation show that the maximum charcoal loading, based on only 50 pounds of charcoal per adsorber train, is approximately 0.26 mg of total iodine per gram of charcoal. This is well below the 2.5 mg/gm maximum value in RG 1.52. The maximum component temperature is approximately 168°F with normal flow conditions and 500°F under conditions of a failed fan with minimum cooling flow, which is well below the 625°F charcoal ignition temperature from RG 1.52.

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Conclusion

NSPM has evaluated the effects of the proposed EPU on the Standby Gas Treatment System. The evaluation indicates that the system will continue to provide adequate fission product removal in post accident environments following implementation of the proposed EPU and will continue to meet the requirements of the current licensing basis. Therefore, the proposed EPU is acceptable with respect to the fission product control systems and structures.

2.5.2.2 Main Condenser Evacuation System

Regulatory Evaluation

The main condenser evacuation system (MCES) generally consists of two subsystems: (1) the "hogging" or startup system that initially establishes main condenser vacuum and (2) the system that maintains condenser vacuum once it has been established.

The NRC's acceptance criteria for the MCES are based on (1) GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents; and (2) GDC-64, 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 anticipated operational occurrences and postulated accidents.

Specific NRC review criteria are contained in SRP Section 10.4.2.

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed

AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC listed in the Regulatory Evaluation above, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-17 and draft GDC-70.

The Main Condenser Evacuation System is described in Monticello USAR Section 11.3.2, "Main Condenser Gas Removal System."

In addition to the evaluations described in the Monticello USAR, Monticello's systems and components were evaluated for License Renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Monticello Nuclear Generating Plant License Renewal Safety Evaluation Report (SER), NUREG-1865, dated October 2006 (Reference 5). The license renewal evaluation associated with the Main Condenser Evacuation System is documented in NUREG-1865, Section 2.3.4.3. Management of aging effects on the Main Condenser Evacuation System is documented in NUREG-1865, Section 3.4.2.3.3.

Technical Evaluation

The main condenser "hogging" (mechanical vacuum pump) and the SJAE functions are required for normal plant operation and are not safety-related.

The design of the condenser air removal system is not adversely affected by EPU and no modification to the system is required. The following aspects of the condenser air removal system were evaluated for this determination:

- Non-condensable gas flow capacity of the SJAE system;
- Capability of the SJAE to operate satisfactorily with available dilution / motive steam flow;
- SJAEs' and inter-condensers' performance at the higher expected non-condensable flow and condenser pressure conditions for EPU, considering water vapor carryover and the maximum expected condensate temperature and flow rate; and
- Mechanical vacuum (hogging) pump capability to remove required non-condensable gases from the condenser at EPU conditions.

The physical size of the main condenser and evacuation time are the primary factors in establishing the capabilities of the mechanical vacuum pump. These parameters do not change. Because flow rates do not change, there is no change to the holdup time in the pump discharge line routed to the reactor building vent stack. The capacity of the SJAEs is adequate because they are designed for operation at flows greater than those required at EPU conditions.

Conclusion

There are no EPU related changes to the MCES and the MCES will continue to maintain its ability to control and provide monitoring for releases of radioactive materials to the environment following implementation of the proposed EPU. The MCES will continue to meet the requirements of the current licensing basis. Therefore, the proposed EPU is acceptable with respect to the MCES.

2.5.2.3 Turbine Gland Sealing System

Regulatory Evaluation

The turbine gland sealing system is provided to control the release of radioactive material from steam in the turbine to the environment.

The NRC's acceptance criteria for the turbine gland sealing system are based on (1) GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents; and (2) GDC-64, 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 anticipated operational occurrences and postulated accidents.

Specific NRC review criteria are contained in SRP Section 10.4.3.

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC listed in the Regulatory Evaluation above, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-17 and draft GDC-70.

The turbine gland sealing system is included in Monticello USAR Section 11.3.2, "Main Condenser Gas Removal System."

In addition to the evaluations described in the Monticello USAR, Monticello's systems and components were evaluated for License Renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Monticello Nuclear Generating Plant License Renewal Safety Evaluation Report (SER), NUREG-1865, dated October 2006 (Reference 5). The license renewal evaluation associated with the turbine gland sealing system is included in NUREG-1865, Section 2.3.4.5. Management of aging effects on the turbine gland sealing system is documented in NUREG-1865, Section 3.4.2.3.5.

Technical Evaluation

The evaluation of the turbine gland seal system, taking into account the modification of the Monticello main turbine to accept the increased steam flow at EPU operating conditions, demonstrated that the system is capable of adequately performing its design function without major modification. Recent uprates have required new springs for the steam seal regulators. This will be determined during the HP steam path final design phase and would be included as part of this installation.

Conclusion

There are no EPU related changes to the turbine gland sealing system. The turbine gland sealing system will continue to maintain its ability to control and provide monitoring for releases of radioactive materials to the environment consistent with the current licensing basis. Therefore, the proposed EPU is acceptable with respect to the turbine gland sealing system.

2.5.2.4 Main Steam Isolation Valve Leakage Control System

Regulatory Evaluation

Redundant quick-acting isolation valves are provided on each main steam line. The leakage control system is designed to reduce the amount of direct, untreated leakage from the main steam isolation valves (MSIVs) when isolation of the primary system and containment is required.

The NRC's acceptance criteria for the MSIV leakage control system are based on GDC-54, insofar as it requires that piping systems penetrating containment be provided with leakage detection and isolation capabilities.

Specific NRC review criteria are contained in SRP Section 6.7.

Monticello Current Licensing Basis

The Monticello design does not include a Main Steam Isolation Valve Leakage Control System.

Technical Evaluation

Not applicable.

Conclusion

Not applicable.

2.5.3 Component Cooling and Decay Heat Removal

2.5.3.1 Spent Fuel Pool Cooling and Cleanup System

Regulatory Evaluation

The spent fuel pool provides wet storage of spent fuel assemblies. The safety function of the spent fuel pool cooling and cleanup system is to cool the spent fuel assemblies and keep the spent fuel assemblies covered with water during all storage conditions.

The NRC's acceptance criteria for the spent fuel pool cooling and cleanup system are based on (1) GDC-5, 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, (2) GDC-44, 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 and accident conditions be provided, and (3) GDC-61, insofar as it requires that fuel storage systems be designed with RHR capability reflecting the importance to safety of decay heat removal, and measures to prevent a significant loss of fuel storage coolant inventory under accident conditions.

Specific NRC review criteria are contained in SRP Section 9.1.3, as supplemented by the guidance provided in Attachment 1 to Matrix 5 of Section 2.1 of RS-001.

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release.

As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC listed in the Regulatory Evaluation above, with the exception of current GDC-44, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-4, draft GDC-67, draft GDC-68, and draft GDC-69. There is no draft GDC directly associated with current GDC-44.

The Spent Fuel Pool Cooling and Cleanup System is described in Monticello USAR Section 10.2.2, "Spent Fuel Pool Cooling and Demineralizer System."

In addition to the evaluations described in the Monticello USAR, Monticello's systems and components were evaluated for License Renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Monticello Nuclear Generating Plant License Renewal Safety Evaluation Report (SER), NUREG-1865, dated October 2006 (Reference 5). The license renewal evaluation associated with the Spent Fuel Pool Cooling and Cleanup System is documented in NUREG-1865, Section 2.3.3.10. Management of aging effects on the Spent Fuel Pool Cooling and Cleanup System is documented in NUREG-1865, Section 3.3.2.3.10.

Technical Evaluation

The Monticello fuel pool systems consist of storage pools, fuel racks, and the Spent Fuel Pool Cooling and Demineralizer system. The RHR system can also be used to provide additional cooling capacity. The objectives of the fuel pool systems are to provide underwater storage space for the spent fuel assemblies, to remove the decay heat from the fuel assemblies, and maintain the fuel pool water within specified temperature limits. The effects of EPU on the Monticello fuel pool are addressed in the following evaluation:

2.5.3.1.1 Fuel Pool Cooling

The Monticello spent fuel pool (SFP) bulk water temperature must be maintained below the licensing limit of 140°F. 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 decay heat used in offload evaluations assumes a 24-month fuel cycle, ANSI/ANS 5.1-1994 standard with a one sided 95% confidence interval, and GE14 fuel. EPU decay heat loads are higher than the CLTP heat load.

The limiting condition (emergency heat load) is a heat load value of 24.7 MBtu/hr. The design basis limiting heat load does not change at EPU conditions. The emergency heat load condition assumes that fuel pool cooling is provided by the RHR system in spent fuel pool cooling assist mode. The increase in heat load resulting from EPU will continue to be managed by performing the required cycle-specific heat load calculation prior to moving fuel to the pool.

EPU does not affect the heat removal capability of the fuel pool cooling system or the fuel pool cooling mode of the RHR system. EPU results in higher core decay heat loads during refueling but can be managed by various methods including extending the time after shutdown before discharging fuel to the SFP. The full core offload heat load in the SFP reaches a maximum immediately after the full core discharge. Consistent with the current licensing basis, Monticello uses appropriate administrative controls to ensure the SFP temperature is maintained less than the licensing limit.

In the unlikely event of a complete loss of SFP cooling capability, the SFP is calculated to reach the boiling temperature in 6.5 hours in the worst case conditions after the limiting full core offload. The calculated boil off rate at design conditions is ~ 53 gpm.

Available sources of water for makeup include the following:

Seismic I System

RHRSW system (~ 3000 gpm)

Seismic II Systems

Filter demineralizer backwash connection (~100 gpm)

Condensate service water (~25 gpm/hose station)

Demineralized water system (~25 gpm/hose station)

Fire Water System (~ 100 gpm/hose station)

These systems are not affected by EPU and remain adequate for the boil off rate of 53 gpm at EPU conditions.

Prior to each fuel movement that could increase the spent fuel pool heat load, calculations are performed to determine the actual pool heat load and determine which equipment must be placed in service to maintain pool temperature. Administrative controls are used to ensure that the fuel pool cooling capacity is not exceeded during core offload. Existing plant instrumentation and procedures provide adequate indications and direction for monitoring and controlling SFP temperature and level during normal batch offloads and the unexpected case of the limiting full core offload.

2.5.3.1.2 Crud Activity and Corrosion Products

SFP crud activity and corrosion products will increase slightly due to EPU conditions. However, the increase is insignificant and the SFP water quality will be maintained by the spent fuel pool cooling and cleanup system.

2.5.3.1.3 Radiation Levels

The normal radiation levels around the SFP may increase slightly due to EPU spent fuel conditions. The increase in radiation levels result primarily during fuel handling operations. Current Monticello radiation procedures and radiation monitoring program will detect any changes in radiation levels and initiate appropriate actions and controls.

The spent fuel pool was designed to provide sufficient shielding to maintain dose rates below 10 CFR 20 limits for normal building occupancy. Radiation surveys for areas adjacent to the spent fuel pool show that existing dose rates are well below these limits. The table below shows dose rates from general radiation surveys taken at a reactor power of 1775 MWt for the refuel floor and the two floors beneath the refuel floor. The bottom of the spent fuel pool is located at elevation 988' 11''. The general areas dose rates for accessible areas above and below the spent fuel pool are low and only one of these areas meets the 10 CFR 20.1003 definition of a Radiation Area.

Plant Area	General Area Dose Rate
Refuel Floor (Elev. 1027 ft)	< 3 mr/hr
Reactor Bldg. (Elev. 1001 ft)	< 4 mr/hr
Reactor Bldg. (Elev. 985 ft)	< 21 mr/hr (peak, near hot spot) Most readings below 5 mr/hr

A slight increase to the above rates will not result in a significant dose. Monticello does not plan on implementing special spent fuel placements within the spent fuel pool to reduce personnel dose in adjacent areas. Monticello procedures provide for routine radiation surveys to detect increases in the normal general radiation levels for these areas and for controls to limit personnel dose during fuel transfers.

2.5.3.1.4 Fuel Racks

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[redacted]	[redacted]	[redacted]
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The increased decay heat from the EPU results in a higher heat load in the fuel pool during long-term storage. The fuel racks are designed for higher temperatures (150°F) than the licensing limit of 140°F. There is no effect on the design of the Monticello fuel racks because the original fuel pool design temperature is not exceeded.

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Conclusion

NSPM has evaluated the spent fuel pool cooling and cleanup system and accounted for the effects of the proposed EPU on the spent fuel pool cooling function of the system. The evaluation concludes that the system will continue to provide sufficient cooling capability to cool the spent fuel pool following implementation of the proposed EPU and will continue to meet the requirements of the current licensing basis. Therefore, the proposed EPU is acceptable with respect to the spent fuel pool cooling and cleanup system.

2.5.3.2 Station Service Water System

Regulatory Evaluation

The station service water system provides essential cooling to safety-related equipment and may also provide cooling to non-safety related auxiliary components that are used for normal plant operation.

The NRC's acceptance criteria are based on (1) GDC-4, 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, including flow instabilities and loads (e.g., water hammer), maintenance, testing, and postulated accidents; (2) GDC-5, 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; and (3) GDC-44, 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 and accident conditions be provided.

Specific NRC review criteria are contained in SRP Section 9.2.1, as supplemented by GL 89-13 and GL 96-06.

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of

1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC listed in the Regulatory Evaluation above, with the exception of current GDC-44, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-4, draft GDC-40, and draft GDC-42. There is no draft GDC directly associated with current GDC-44.

The Monticello design includes three open loop cooling water systems. The Plant Service Water System supplies water to the Reactor and Turbine Buildings for cooling. Cross connections in the Plant Service Water System are provided to the main condenser hotwell, the ECCS room air coolers, and the ECCS pump motor coolers. Plant Service Water, when used as a standby coolant supply, provides an additional supply of screened river water to the main condenser hotwell for injection into the reactor vessel by the Feedwater System. Plant Service Water can also be used as an additional source of cooling water for the ECCS room air coolers and the ECCS pump motors. The Plant Service Water System is described in Monticello USAR Section 10.4.1, "Plant Service Water System."

The Residual Heat Removal Service Water System is provided to remove the heat rejected by the residual heat removal system during normal shutdown and accident operations. In addition this system provides a source of water for the RHR-RHR Service Water cross connection. The Residual Heat Removal Service Water System is described in Monticello USAR Section 10.4.2, "Residual Heat Removal System Service Water System."

The Emergency Service Water System is provided to remove the heat rejected by the equipment that must operate under accident conditions. The Emergency Service Water System is described in Monticello USAR Section 10.4.4, "Emergency Service Water System."

Monticello's current licensing basis regarding GL 89-13 is discussed in NSP's response to the NRC by letter dated January 29, 1990, "Response to Generic Letter 89.13 Service Water Problems Affecting Safety-Related Equipment," and NRC's letter to Northern States Power Company, "Monticello Nuclear Generating Plant - Response To Generic Letter 89-13, 'Service Water System Problems Affecting Safety-Related Equipment (TAC 74028),'" dated March 6, 1990. Monticello's current licensing basis regarding GL 96-06 is NRC's letter to Northern States Power Company, "Completion of Licensing Action for Generic Letter 96-06 - Assurance of Equipment Operability and Containment Integrity During Design Basis Accident Conditions (TAC No. M96835)," dated May 18, 2000, and the correspondence referenced in that letter.

In addition to the evaluations described in the Monticello USAR, Monticello's systems and components were evaluated for License Renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Monticello Nuclear Generating Plant License Renewal Safety Evaluation Report (SER), NUREG-1865, dated October 2006 (Reference 5). The license renewal evaluation associated with the Emergency Service Water System and Residual Heat Removal Service Water System is documented in NUREG-1865, Section 2.3.3.8. The license renewal evaluation associated with the Plant Service Water System is documented in NUREG-1865, Section 2.3.3.16. Management of aging effects on the Emergency Service Water System and Residual Heat Removal Service Water System is documented in NUREG-1865, Section 3.3.2.3.8. Management of aging effects on the Plant Service Water System is documented in NUREG-1865, Section 3.3.2.3.16.

The effect of EPU on actions taken for GL 96-06 is addressed in Section 2.6.1.5.

Technical Evaluation

2.5.3.2.1 Plant Service Water System

The Plant Service Water (SW) System cools various plant components that have heat loads that are either power dependent or unaffected by EPU. The Plant Service Water 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. The required service water flow to cool these components will be evaluated in conjunction with these modifications.

The SW System has sufficient capacity to supply adequate cooling to the remaining components at EPU conditions.

2.5.3.2.2 Residual Heat Removal Service Water System

The safety-related residual heat removal service water system provides a reliable supply of cooling water for the following essential equipment, systems and functions:

- RHR heat exchangers;
- Long term core and containment cooling, as necessary;
- SFP emergency make-up, as necessary.

The containment cooling analysis shows that the post-LOCA RHR heat load increases due to an increase in the maximum suppression pool temperature that occurs following a LOCA. 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 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 spent fuel pool cooling. The RHRSW is capable of maintaining a positive pressure across the RHR heat exchanger between the tube side (RHRSW) and the shell side (RHR). In addition, the RHRSW system has sufficient capacity to serve as a standby coolant supply for long term core and containment cooling, as well as emergency makeup to the SFP, as required for EPU conditions. The RHRSW system flow rate is not changed.

2.5.3.2.3 Emergency Service Water

The safety-related Emergency Service Water (ESW) system provides a reliable supply of cooling water for the following essential equipment:

- ECCS (RHR and CS) pump motor coolers;
- ECCS (RHR and CS) room coolers;
- Emergency Diesel Generators
- MCR Air Conditioning System

Additionally, the ESW system provides a reliable supply of cooling water to the Division II ECCS pump electric motors and ECCS room coolers during a postulated Appendix R fire event.

ESW also supplies normal cooling water to the HPCI room coolers. Because the heat loads from the HPCI room coolers do not change at EPU, the ESW continues to provide adequate heat removal capacity.

Heat loads from the RHR and CS ECCS room coolers increase less than 3 percent at EPU conditions. The remaining heat loads are unchanged at EPU conditions. Sufficient heat removal capacity is available to accommodate the small increases in heat loads during operation at EPU design conditions.

Conclusion

NSPM has evaluated the effects of the proposed EPU on the station service water system including any increased heat loads on system performance that would result from the proposed EPU. The evaluation indicates that the station service water systems will continue to provide sufficient cooling for SSCs important to safety following implementation of the proposed EPU. Therefore, the station service water systems will continue to meet the requirements of the current licensing basis. Additionally, the Monticello GL 89-13 Program (i.e., scope, maintenance, and testing) to manage and monitor raw water cooling systems is not affected by the proposed EPU. Based on the above, the proposed EPU is acceptable with respect to the station service 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 ECCS.

The NRC's acceptance criteria for the reactor auxiliary cooling water system are based on (1) GDC-4, 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 including flow instabilities and attendant loads (i.e., water hammer), maintenance, testing, and postulated accidents; (2) GDC-5, 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; and (3) GDC-44, 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 and accident conditions be provided.

Specific NRC review criteria are contained in SRP Section 9.2.2, as supplemented by GL 89-13 and GL 96-06.

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC listed in the Regulatory Evaluation above, with the exception of current GDC-44, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-4, draft GDC-40, and draft GDC-42. There is no draft GDC directly associated with GDC-44.

The Monticello design includes one closed loop cooling water system. The Reactor Building Closed Cooling Water System is designed to remove heat from the reactor auxiliary systems equipment and their accessories.

The Reactor Building Closed Cooling Water System is described in Monticello USAR Section 10.4.3, "Reactor Building Closed Cooling Water System."

Monticello's current licensing basis regarding GL 89-13 is discussed in NSP's response to the NRC by letter dated January 29, 1990, "Response to Generic Letter 89.13 Service Water Problems Affecting Safety-Related Equipment," and NRC's letter to Northern States Power Company, "Monticello Nuclear Generating Plant - Response To Generic Letter 89-13, 'Service Water System Problems Affecting Safety-Related Equipment (TAC 74028),'" dated March 6, 1990. Monticello's current licensing basis regarding GL 96-06 is NRC's letter to Northern States Power Company, "Completion of Licensing Action for Generic Letter 96-06 - Assurance of Equipment Operability and Containment Integrity During Design Basis Accident Conditions (TAC No. M96835)," dated May 18, 2000, and the correspondence referenced in that letter.

In addition to the evaluations described in the Monticello USAR, Monticello's systems and components were evaluated for License Renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Monticello Nuclear Generating Plant License Renewal Safety Evaluation Report (SER), NUREG-1865, dated October 2006 (Reference 5). The license renewal evaluation associated with the Reactor Building Closed Cooling Water System is documented in NUREG-1865, Section 2.3.3.14. Management of the effects of aging on the RBCCW system is documented in NUREG-1865, Section 3.3.2.3.14.

The effect of EPU on actions taken for GL 96-06 is addressed in Section 2.6.1.5.

Technical Evaluation

2.5.3.3.1 Reactor Building Closed Cooling Water System

The heat loads on the non-safety related RBCCW system increase insignificantly (< 0.1%) 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 Spent Fuel Pool Heat Exchangers. The Drywell Atmosphere Cooler heat load increases approximately 9,360 BTU/hr (< 0.3%), which is insignificant when compared to the RBCCW system total heat load of 36E6 BTU/hr. The heat loads of the RWCU Non-Regenerative Heat Exchangers and Spent Fuel Pool Heat Exchangers are not changed by EPU because the fluid conditions in these systems are unchanged. The heat load of the Spent Fuel Pool increases due to increased decay heat resulting from operation at a higher core power. However, the increased heat load doesn't increase RBCCW design heat load because the RHR system would be used, if necessary, to supplement the Spent Fuel Pool Heat Exchangers (see Section 2.5.3.1 for the Spent Fuel Pool cooling evaluation). The cooling loads for the remaining systems served by RBCCW (e.g. seal and bearing coolers, sample coolers, and drain sump cooler) are not changed by EPU.

The RBCCW system has sufficient heat removal capacity to assure adequate heat removal capability for all serviced components. Therefore, sufficient heat removal capacity is available to accommodate the small increase in heat load due to EPU.

Conclusion

NSPM has evaluated the effects of the proposed EPU on the reactor auxiliary cooling water systems including any increased heat loads from the proposed EPU on system performance. The evaluation indicates that the reactor auxiliary cooling water systems will continue to provide sufficient cooling for SSCs important to safety following implementation of the proposed EPU. Therefore, the reactor auxiliary cooling water systems will continue to meet the requirements of the current licensing basis. Additionally, the Monticello GL 89-13 Program (i.e., scope, maintenance, and testing) to manage and monitor raw water cooling systems is not affected by the proposed EPU. Based on the above, the proposed EPU is acceptable with respect to the reactor auxiliary cooling water systems.

2.5.3.4 Ultimate Heat Sink

Regulatory Evaluation

The ultimate heat sink (UHS) is the source of cooling water provided to dissipate reactor decay heat and essential cooling system heat loads after a normal reactor shutdown or a shutdown following an accident.

The NRC's acceptance criteria for the UHS are based on (1) GDC-5, 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; and (2) GDC-44, 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 and accident conditions be provided.

Specific NRC review criteria are contained in SRP Section 9.2.5.

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC listed in the Regulatory Evaluation above, with the exception of current GDC-44, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-4. There is no draft GDC directly associated with current GDC-44.

The Mississippi River serves as the ultimate heat sink for the plant. The ultimate heat sink temperature limit is described in Monticello USAR Section 5.2.3.2.4, "Licensing Basis Ultimate Heat Sink Limit."

Technical Evaluation

The Ultimate Heat Sink (UHS) for the Monticello Nuclear Generating Station is the Mississippi River. The UHS intake temperature is unaffected by operations at EPU conditions. The Service Water, Residual Heat Removal Service Water, Emergency Service Water, and Emergency Diesel Generator-Emergency Service Water systems were reviewed to evaluate the effect of the increase in core decay heat on the UHS for EPU. The review concludes that the flow rate requirements at the maximum supply water temperature (90^0F) for these systems are unchanged for EPU and therefore the ability of the UHS to serve its safety function is unaffected by EPU.

Conclusion

NSPM has evaluated the effects that the proposed EPU would have on the UHS safety function, including the design-basis UHS temperature limit. The evaluation indicates that the proposed EPU will not compromise the design-basis safety function of the UHS, and that the UHS will continue to satisfy the requirements of the current licensing basis following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to the UHS.

2.5.4 Balance-of-Plant Systems

2.5.4.1 Main Steam

Regulatory Evaluation

The main steam supply system (MS) transports steam from the NSSS to the power conversion system and various safety-related and non-safety related auxiliaries.

The NRC's acceptance criteria for the MS are based on (1) GDC-4, insofar as it requires that SSCs important to safety be protected against dynamic effects, including the effects missiles, pipe whip, and jet impingement forces associated with pipe breaks; and (2) GDC-5, 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 NRC review criteria are contained in SRP Section 10.3.

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC listed in the Regulatory Evaluation above, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-4, draft GDC-40, and draft GDC-42.

Main steam piping is discussed in several USAR Sections including Section 4, "Reactor Coolant System," Section 6.3, "Main Steam Line Flow Restrictions," and Section 12.2.2.10, "Main Steam Line Restraints."

In addition to the evaluations described in the Monticello USAR, Monticello's systems and components were evaluated for License Renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Monticello Nuclear Generating Plant License Renewal Safety Evaluation Report (SER), NUREG-1865, dated October 2006 (Reference 5). The license renewal evaluation associated with the main steam piping is documented in NUREG-1865, Section 2.3.4.4. Management of the effects of aging on the main steam piping is documented in NUREG-1865, Section 3.4.2.3.4.

Technical Evaluation

The heat balance for the EPU conditions is provided in Section 1.3.1. The heat balance shows that the capability to transport steam to the power conversion equipment, the heat sink, and to steam driven components is acceptable. Flow induced vibration and structural loading of the main steam system piping and supports is addressed in Sections 2.2.2. Dynamic loading from water hammer is discussed below. Safety Relief Valve (SRV) dynamic loads are discussed in Sections 2.2.2 and 2.2.3. The function and capability of the Main Steam Isolation Valves are discussed in Section 2.2.2. SRV setpoint tolerance and Flow Induced Vibration effects are discussed below.

Because the MS piping pressures and temperatures are not affected by EPU, there is no effect on the analyses for these parameters. Seismic inertia loads, seismic building displacement loads, and SRV discharge loads are not affected by EPU, thus, there is no effect on the analyses for these load cases. The increase in MS flow results in increased forces from the turbine stop valve closure transient. The turbine stop valve closure loads bound the MSIV closure loads because the MSIV closure time is significantly longer than the stop valve closure time.

SRV setpoint tolerance is independent of EPU. EPU evaluations are performed using the existing SRV setpoint tolerance AL of 3% as a basis. Actual historical in-service surveillance of SRV setpoint performance test results are monitored separately for compliance to the Technical Specification requirements.

The in-service surveillance testing of the plant's SRVs have not shown a significant propensity for high setpoint drift greater than 3%. Out of 14 SRV tests, from the "as found" setpoint lift verification tests performed from 1998 to 2005; all of the "as found" setpoints were within ± 3%.

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Flow-induced vibration (FIV) may increase incidents of valve leakage. However, Monticello currently has procedures to address a leaking SRV. FIV on the Target Rock 3-Stage SRV design may result in an inadvertent SRV opening and a "stuck open" SRV event. This characteristic has previously been identified and is addressed in plant procedures. The consequences of a stuck open SRV have been previously considered in the plant specific safety analyses and have been demonstrated to be non-limiting.

Increased main steam line (MSL) flow may affect flow-induced vibration of the piping and SRVs during normal operation. The vibration frequency, extent, and magnitude depend upon plant specific parameters, valve locations, the valve design, and piping support arrangements. The flow-induced vibration of the piping will be addressed by vibration testing during EPU power ascension.

The FIV effect on the MS piping inside containment at Monticello is confirmed to be consistent with the [[

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Main Steam Line Flow Restrictors

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The increase in steam flow rate has no significant effect on flow restrictor erosion. There is no effect on the structural integrity of the MSL flow element (restrictor) due to the increased differential pressure because the restrictors were designed and analyzed for the choke flow condition.

After a postulated steam line break outside containment, the fluid flow in the broken steam line increases until it is limited by the MSL flow restrictor. Because the maximum operating dome pressure does not change, the resulting break flow rate is unchanged from the current analysis and the operational stresses are not affected. Therefore, the MSL flow restrictors are not significantly affected by EPU.

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Conclusion

NSPM has evaluated the effects of the proposed EPU on MS including the effects of changes in plant conditions on the design of MS. The evaluation indicates that the system will continue to meet the requirements of the current licensing basis. Therefore, the proposed EPU is acceptable with respect to MS.

2.5.4.2 Main Condenser

Regulatory Evaluation

The main condenser system is designed to condense and deaerate the exhaust steam from the main turbine and provide a heat sink for the turbine bypass system. For BWRs without an MSIV leakage control system, the main condenser system may also serve an accident mitigation function to act as a holdup volume for the plate out of fission products leaking through the MSIVs following core damage.

The NRC's acceptance criteria for the main condenser system are based on GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents.

Specific NRC review criteria are contained in SRP Section 10.4.1.

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC listed in the Regulatory Evaluation above, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-70.

The main condenser system is described in Monticello USAR Section 11.3, "Main Condenser System."

In addition to the evaluations described in the Monticello USAR, Monticello's systems and components were evaluated for License Renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Monticello Nuclear Generating Plant License Renewal Safety Evaluation Report (SER), NUREG-1865, dated October 2006 (Reference 5). The license renewal evaluation associated with the main condenser system is documented in NUREG-1865, Section 2.3.4.3. The management of the effects of aging on the main condenser system is documented in NUREG-1865, Section 3.4.2.3.3.

Technical Evaluation

The main condenser, circulating water, and normal heat sink systems are designed to remove the heat rejected to the condenser and thereby maintain adequately low condenser pressure as recommended by the turbine vendor. Maintaining adequately low condenser pressure assures the efficient operation of the turbine generator and minimizes wear on the turbine last stage buckets.

EPU operation increases the heat rejected to the condenser and, therefore, reduces the difference between the operating pressure and the recommended maximum condenser pressure. If condenser pressures approach the main turbine backpressure limitation, then reactor thermal power reduction would be required to reduce the heat rejected to the condenser and maintain condenser pressure within the main turbine requirements.

An evaluation also assessed the effects of increasing power to 120% of OLTP on the heat rejection systems and comparing to the existing NPDES permit limits. This evaluation used plant, river and atmospheric data for the years 2001-2006 and adjusted thermal power per the EPU heat balance. Key parameters were evaluated to determine the effects of EPU operating conditions upon plant parameters. In general, the EPU effects are within the existing limits. If temperatures approach the existing limits, then reactor thermal power reduction would be required to reduce the heat rejected to the condenser and maintain temperatures within the existing limits.

Because the turbine bypass system is not being modified for EPU and the operating pressure is not being changed, the turbine bypass capacity and the blowdown effects of steam from the turbine bypass system on the main condenser are not affected by EPU.

Because the main condenser is not being modified for EPU, its use as a holdup volume is not affected by EPU.

The absolute value in lbm/hr of the steam bypassed to the main condenser during a load rejection event is not increased for EPU. The turbine steam bypass system is discussed in Section 2.5.4.3.

Conclusion

NSPM has evaluated the effects of the proposed EPU on the main condenser system including the effects of changes in plant conditions on the design of the main condenser system. The evaluation indicates that the main condenser system will continue to maintain its ability to withstand the blow down effects of the steam from the turbine bypass system and thereby continue to meet the current licensing bases. Therefore, the proposed EPU is acceptable with respect to the main condenser system.

2.5.4.3 Turbine Bypass

Regulatory Evaluation

The Turbine Bypass System (TBS) is designed to discharge a stated percentage of rated main steam flow directly to the main condenser system, bypassing the turbine. This steam bypass enables the plant to take step-load reductions up to the TBS capacity without the reactor or turbine tripping. The system is also used during startup and shutdown to control reactor pressure. For a BWR without an MSIV leakage control system, the TBS could also provide an accident mitigation function. A TBS, along with the main steam supply system and main condenser system, may be credited for mitigating the effects of MSIV leakage during a LOCA by the holdup and plate out of fission products.

The NRC's acceptance criteria for the TBS are based on (1) GDC-4, 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 (including pipe breaks or malfunctions of the TBS), and (2) GDC-34, insofar as it requires that a RHR system be provided to transfer fission product decay heat and other residual heat from the reactor core at a rate such that SAFDLs and the design conditions of the RCPB are not exceeded.

Specific NRC review criteria are contained in SRP Section 10.4.4.

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of

the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC listed in the Regulatory Evaluation above, with the exception of current GDC-34, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-40 and draft GDC-42. There is no draft GDC directly associated with current GDC-34.

The TBS is described in Monticello USAR Section 11.4, "Main Turbine Bypass System." Section 4.0 in the NRC's Safety Evaluation Report for License Amendment 102, September 16, 1998 summarizes the review and acceptance methodology and conclusions regarding the integrity of the MSIV leakage collection path at Monticello.

In addition to the evaluations described in the Monticello USAR, Monticello's systems and components were evaluated for License Renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Monticello Nuclear Generating Plant License Renewal Safety Evaluation Report (SER), NUREG-1865, dated October 2006 (Reference 5). The TBS is included in the discussion of the license renewal evaluation for the Main Steam System. That discussion can be found in NUREG-1865, Section 2.3.4.4. Management of aging effects on the Main Steam System is documented in NUREG-1865, Section 3.4.2.3.4.

Technical Evaluation

The Turbine Steam Bypass System provides a means of accommodating excess steam generated during normal plant maneuvers and transients. The turbine bypass valves are currently rated for a total steam flow capacity of not less than 13.3% of the current rated reactor steam flow, or ~0.97 Mlb/hr. Each of two bypass valves is designed to pass a steam flow of ~485,000 lbm/hr and this does not change at EPU RTP. At EPU conditions, rated reactor steam flow is 8.34 Mlb/hr, resulting in a bypass capacity of ~11.6 % of EPU rated steam flow.

The Turbine Steam Bypass System is a normal operating system that is used to bypass excessive steam flow. The absolute flow capacity (mass flow rate) of the bypass system is unchanged. The bypass flow capacity is included in some AOO evaluations (Section 2.8.5).

Conclusion

NSPM has evaluated the effects of the proposed EPU on the TBS. The evaluation indicates that the same absolute value of steam flow bypass capacity will exist at EPU. The relative bypass capability with respect to rated steam flow at EPU conditions is reduced slightly. The TBS will continue to provide a means of accommodating excess steam generation during normal plant maneuvers and transients. Therefore, the proposed EPU is acceptable with respect to the TBS.

2.5.4.4 Condensate and Feedwater

Regulatory Evaluation

The condensate and feedwater system provides feedwater at a particular temperature, pressure, and flow rate to the reactor. The only part of the condensate and feedwater system classified as safety-related is the feedwater piping from the NSSS up to and including the outermost containment isolation valve.

The NRC's acceptance criteria for the condensate and feedwater system are based on (1) GDC-4, 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 including possible fluid flow instabilities (e.g., water hammer), maintenance, testing, and postulated accidents; (2) GDC-5, 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; and (3) GDC-44, 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 and accident conditions be provided, and that the system be provided with suitable isolation capabilities to assure the safety function can be accomplished with electric power available from only the onsite system or only the offsite system, assuming a single failure.

Specific NRC review criteria are contained in SRP Section 10.4.7.

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC listed in the Regulatory Evaluation above, with the exception of current GDC-44, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-4, draft GDC-40 and draft GDC-42. There is no draft GDC directly associated with current GDC-44.

The condensate and feedwater system is described in Monticello USAR Section 11.8, "Condensate and Reactor Feedwater Systems." The condensate demineralizer system is described in Monticello USAR Section 11.7, "Condensate Demineralizer System."

In addition to the evaluations described in the Monticello USAR, Monticello's systems and components were evaluated for License Renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Monticello Nuclear Generating Plant License Renewal Safety Evaluation Report (SER), NUREG-1865, dated October 2006 (Reference 5). The license renewal evaluation associated with the condensate and feedwater system is documented in NUREG-1865, Section 2.3.4.2. The management of the effects of aging on the condensate and feedwater system is documented in NUREG-1865, Section 3.4.2.3.2.

Technical Evaluation

The FW and Condensate systems provide the source of makeup water to the reactor to support normal plant operation. The increase in power level increases the feedwater requirements of the reactor.

The FW and Condensate systems are designed to provide a reliable supply of FW at the temperature, pressure, quality, and flow rate as required by the reactor. The FW and Condensate systems do not perform a system level safety-related function; however, their performance has an effect on plant availability and capability to operate at the EPU conditions.

For EPU, the FW and Condensate systems will meet their performance requirements with modifications to the following non-safety related equipment for increased capacity:

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

For life cycle management (i.e., existing equipment will operate within their ratings at the EPU conditions), modifications are anticipated for the following non-safety related equipment:

- FW heaters 13A, 13B, 14A, 14B, 15A, and 15B
- Drain coolers 11 and 12

- Drain and dump valves for FW heaters and drain coolers

Normal Operation

System operating flows at EPU increase to approximately 115% of rated flow at the CLTP. The FW and Condensate systems will be modified to assure acceptable performance with the new operating conditions.

The FW heaters are acceptable for the higher FW heater flows, temperatures and pressures at EPU. The performance of the FW heaters will be monitored during the EPU power ascension program.

Transient Operation

To account for FW transients, the modified FW and Condensate pumps will provide a minimum of 5% margin above the EPU rated FW flow. This is consistent with the CLTP design, 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.

Condensate Demineralizers

The effect of the higher condensate flow rate required for EPU on the condensate filter demineralizers (CFDs) was reviewed. The Condensate Demineralizer System will be modified to support full flow CFD operation without requiring a plant power reduction during CFD regenerations. The system modifications may affect the run time between CFD regenerations and the volume of radioactive waste (liquid and solid) associated with CFD regeneration. The frequency of CFD regenerations will depend on the filtering capacity of the new vessels and the higher flow rate through the system. The effects on the liquid and solid waste management systems are discussed in Sections 2.5.5.2 and 2.5.5.3.

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[redacted]	[redacted]	[redacted]
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Key applicable structures include the FW system piping and suspension. In addition, branch lines attached to the FW system piping are considered.

The FW piping experiences increased flow rates and flow velocities under EPU conditions. As a result, the FW piping experiences increased vibration levels, approximately 50 percent for a 20 percent power uprate. The ASME Code and nuclear regulatory guidelines require vibration test data be taken and evaluated for these high energy piping systems during initial operation at EPU conditions. Vibration data for the MS and FW piping inside containment will be acquired using remote sensors. A piping vibration startup test program, which meets the ASME Code and regulatory requirements, will be performed.

The FIV effect on the FW piping inside containment at Monticello is confirmed to be consistent with the [[

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Conclusion

NSPM has evaluated the effects of the proposed EPU on the condensate and feedwater system. The evaluation indicates that the condensate and feedwater systems will continue to meet their performance requirements following modifications to several non-safety related components. Additionally, the modified condensate and feedwater pumps will provide a minimum of 5 percent margin above the EPU rated flow to account for feedwater transients. Therefore, the proposed EPU is acceptable with respect to the condensate and feedwater system.

2.5.5 Waste Management Systems

2.5.5.1 Gaseous Waste Management Systems

Regulatory Evaluation

The gaseous waste management systems involve the gaseous radwaste system, which deals with the management of radioactive gases collected in the offgas system or the waste gas storage and decay tanks. In addition, it involves the management of the condenser air removal system; the gland seal exhaust and the mechanical vacuum pump operation exhaust; and the building ventilation system exhausts.

The NRC's acceptance criteria for gaseous waste management systems are based on (1) 10 CFR 20.1302, 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, 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, insofar as it requires that the plant design include means to control the release of radioactive effluents; (4) GDC-61, insofar as it requires that systems that contain radioactivity be designed with appropriate confinement; and (5) 10 CFR Part 50, Appendix I, 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.

Specific NRC review criteria are contained in SRP Section 11.3.

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC listed in the Regulatory Evaluation above, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-3, draft GDC-67, draft GDC-68, draft GDC-69, and draft GDC-70.

The gaseous waste management system is described in Monticello USAR Section 9.3, "Gaseous Radwaste System."

Technical Evaluation

The CLTR (Section 8.2) [[ ]] the offgas system performance at EPU conditions.

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[illegible]	[illegible]	[illegible]
		]] Further, none of the identified parameters are significantly affected by EPU.
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The primary function of the Gaseous Waste Management (Offgas) System is to process and control the release of gaseous radioactive effluents to the site environs so that the total radiation exposure of persons in offsite areas is within the guideline values of 10 CFR 50, Appendix I. The Offgas System involves the management of the main condenser air removal system; gland seal exhaust and mechanical vacuum pump operation exhaust; and building ventilation system exhausts. Plant procedures exist to test for air infiltration (e.g., main condenser) and repair as needed to maintain the Offgas System functional.

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. None of these parameters are significantly affected by EPU.

The administrative controls mentioned above to maintain the offgas radiological release rate below limits include power reduction or shutdown, reducing main condenser air inleakage (increasing delay tank holdup time), and local power suppression (inserting control rods near fuel leaker). Monticello has Technical Specification requirements and administrative controls to 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 Monticello to maintain the offgas limits. These procedures are not affected by EPU.

Because EPU affects only the flow rate of radiolytic hydrogen and oxygen to the Offgas System, only the catalytic recombiner temperature and offgas condenser heat load are affected. The design of these components is based on a flow rate of radiolytic gas conservatively chosen from measurements performed at many BWRs. The chosen value used for the CLTR design basis is 0.0670 cfm/MWt (130°F and 1 atm). The CLTR design value represents a margin of more than 50% over the 0.0450 cfm/MWt actual radiolysis rate determined for Monticello. Thus, the recombiner and condenser, as well as downstream system components, are designed to handle an average increase in thermal power of more than 50% relative to OLTP, without exceeding the design basis temperatures, flow rates, or heat loads.

The EPU evaluation of the Offgas System and those systems and components connected to the Offgas System concludes that sufficient capacity exists without modification to process expected offgas. [[

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Conclusion

NSPM has evaluated the gaseous waste management systems and the increase in fission product and amount of gaseous waste on the abilities of the system to control releases of radioactive materials and preclude the possibility of an explosion if the potential for explosive mixtures exists. The evaluation indicates that the gaseous waste management systems will continue to meet their design functions following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to the gaseous waste management systems.

2.5.5.2 Liquid Waste Management Systems

Regulatory Evaluation

The NRC's acceptance criteria for the liquid waste management systems are based on (1) 10 CFR 20.1302, 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-60, insofar as it requires that the plant design include means to control the release of radioactive effluents; (3) GDC-61, insofar as it requires that systems that contain radioactivity be designed with appropriate confinement; and (4) 10 CFR Part 50, Appendix I, Sections II.A and II.D, which set numerical guides for dose design objectives and limiting conditions for operation to meet the ALARA criterion.

Specific NRC review criteria are contained in SRP Section 11.2.

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC listed in the Regulatory Evaluation above, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-67, draft GDC-68, draft GDC-69, and draft GDC-70.

The Liquid Waste Management System is described in Monticello USAR Section 9.2, "Liquid Radwaste System."

In addition to the evaluations described in the Monticello USAR, Monticello's systems and components were evaluated for License Renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Monticello Nuclear Generating Plant License Renewal Safety Evaluation Report (SER), NUREG-1865, dated October 2006 (Reference 5). The license renewal evaluation associated with the Liquid Waste Management System is documented in NUREG-1865, Section 2.3.3.13. Management of aging effects on the Liquid Waste Management System is documented in NUREG-1865, Section 3.3.2.3.13.

Technical Evaluation

The Monticello Liquid Radwaste System collects, monitors, processes, stores, and returns processed radioactive waste to the plant for reuse or for discharge.

The single largest source of liquid and wet solid waste is from the backwash of condensate demineralizers. EPU results in an increased flow rate through the condensate demineralizers, resulting in a reduction in the average time between backwashes. This reduction does not affect plant safety. Similarly, the RWCU filter-demineralizer requires more frequent backwashes due to higher levels of impurities as a result of the increased FW flow.

The increased loading of soluble and insoluble species increases the volume of the liquid processed wastes by $\leq$ 2%. The total volume of liquid processed waste does not increase appreciably (as compared to the Radwaste System capacity) because the only increase in processed waste is due to more frequent backwashes of the condensate demineralizers and RWCU filter demineralizers. The total liquid increases are within the Radwaste System capacity. The Radwaste System will operate at 55% of design capacity under EPU conditions. Therefore, EPU does not have an adverse effect on the processing of liquid radwaste, and there are no significant environmental effects.

The increases in the liquid processed waste are based on the increase due to the FW flow increase. The percentage bounding value for the increase in liquid processed waste is equal to or less than that of the FW flow percentage increase.

Conclusion

NSPM has evaluated the liquid waste management systems including the effects of the increase in fission product and amount of liquid waste on the ability of the liquid waste management systems to control releases of radioactive materials. The evaluation indicates that the liquid waste management systems will continue to meet their design functions following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to the liquid waste management systems.

2.5.5.3 Solid Waste Management Systems

Regulatory Evaluation

The NRC's acceptance criteria for the Solid Waste Management System are based on (1) 10 CFR 20.1302, 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-60, insofar as it requires that the plant design include means to control the release of radioactive effluents; (3) GDC-63, insofar as it requires that systems be provided in waste handling areas to detect conditions that may result in excessive radiation levels, (4) GDC-64, 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; and (5) 10 CFR Part 71, which states requirements for radioactive material packaging.

Specific NRC review criteria are contained in SRP Section 11.4.

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC listed in the Regulatory Evaluation above, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-17, draft GDC-18, draft GDC-67, and draft GDC-70.

The Solid Waste Management System is described in Monticello USAR Section 9.4, "Solid Radwaste System."

In addition to the evaluations described in the Monticello USAR, Monticello's systems and components were evaluated for License Renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Monticello Nuclear Generating Plant License Renewal Safety Evaluation Report (SER), NUREG-1865, dated October 2006 (Reference 5). The license renewal evaluation associated with the Solid Waste Management System is documented in NUREG-1865, Section 2.3.3.13. Management of aging effects on the Solid Waste Management System is documented in NUREG-1865, Section 3.3.2.3.13.

Technical Evaluation

The single largest source of wet solid waste is from the backwash of condensate demineralizers. EPU results in an increased flow rate through the condensate demineralizers, resulting in a reduction in the average time between backwashes. This reduction does not affect plant safety. Similarly, the RWCU filter-demineralizer requires more frequent backwashes due to higher levels of impurities as a result of the increased FW flow.

The increased loading of soluble and insoluble species increases the volume of the solid processed wastes by ≤ 18%. The total volume of solid processed waste does not increase appreciably (as compared to the Radwaste System capacity) because the only increase in processed waste is due to more frequent backwashes of the condensate demineralizers and RWCU filter demineralizers. The total solid increases are within the Radwaste System capacity. NSPM continually tracks the volume of solid radwaste generated at Monticello. Significant volume reductions have occurred over the years. In the 1994-95 timeframe, approximately 50 cubic meters/year was shipped. For calendar years 2001 through 2006, the average volume of solid radwaste (spent resin, filter sludge, evaporator bottoms, etc.) shipped per year was less than 20 cubic meters. The increased volume of resins due to power uprate (estimated at approximately 3 cubic meters/year) could be accommodated in one additional truck shipment per year. Therefore, EPU does not have an adverse effect on the processing of solid radwaste, and there are no significant environmental effects.

The increases in the solid processed waste are based on the increase due to the FW flow increase. The percentage bounding value for the increase in solid processed waste is equal to or less than that of the FW flow percentage increase.

Conclusion

NSPM has evaluated the effects of the increase in fission product and amount of solid waste on the ability of the Solid Waste Management System to process the waste. The evaluation indicates that the Solid Waste Management System will continue to meet its design functions following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to the Solid Waste Management System.

2.5.6 Additional Considerations

2.5.6.1 Emergency Diesel Engine Fuel Oil Storage and Transfer System

Regulatory Evaluation

Nuclear power plants are required to have redundant onsite emergency power supplies of sufficient capacity to perform their safety functions (e.g., power diesel engine-driven generator sets), assuming a single failure.

The NRC's acceptance criteria for the emergency diesel engine fuel oil storage and transfer system are based on (1) GDC-4, insofar as it requires that SSCs important to safety be protected against dynamic effects, including missiles, pipe whip, and jet impingement forces associated with pipe breaks; (2) GDC-5, 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; and (3) GDC-17, insofar as it requires onsite power supplies to have sufficient independence and redundancy to perform their safety functions, assuming a single failure.

Specific NRC review criteria are contained in SRP Section 9.5.4.

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC listed in the Regulatory Evaluation above, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-4, draft GDC-24, draft GDC-39, draft GDC-40, and draft GDC-42.

The Diesel Engine Fuel Oil Storage and Transfer capability is described in Monticello USAR Section 8.4, "Plant Standby Diesel Generator Systems."

In addition to the evaluations described in the Monticello USAR, Monticello's systems and components were evaluated for License Renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Monticello Nuclear Generating Plant License Renewal Safety Evaluation Report (SER), NUREG-1865, dated October 2006 (Reference 5). The license renewal evaluation associated with the Diesel Engine Fuel Oil Storage and Transfer capability is documented in NUREG-1865, Section 2.3.3.6. Management of aging effects on the Diesel Engine Fuel Oil Storage and Transfer capability is documented in NUREG-1865, Section 3.3.2.3.6.

Technical Evaluation

The EDG design basis loading is not affected by EPU. The EDG continuous load rating of 2500 kW envelopes the initial and steady state load. Loads and mission times are not increased for EPU because there are no changes to variables for ECCS loading. Therefore, no changes are necessary to the Emergency Diesel Engine Fuel Oil Storage and Transfer System.

Conclusion

NSPM has evaluated the required fuel oil for the emergency diesel generators and the effects of any increased electrical demand on fuel oil consumption. The evaluation indicates that the fuel oil storage and transfer system will continue to provide an adequate amount of fuel oil to allow the diesel generators to meet the onsite power requirements of the current licensing basis.

Therefore, the proposed EPU is acceptable with respect to the fuel oil storage and transfer system.

2.5.6.2 Light Load Handling System (Related to Refueling)

Regulatory Evaluation

The light load handling system includes components and equipment used in handling new fuel at the receiving station and the loading of spent fuel into shipping casks.

The NRC's acceptance criteria for the light load handling system are based on (1) GDC-61, insofar as it requires that systems that contain radioactivity be designed with appropriate confinement and with suitable shielding for radiation protection; and (2) GDC-62, insofar as it requires that criticality be prevented.

Specific NRC review criteria are contained in SRP Section 9.1.4.

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC listed in the Regulatory Evaluation above, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-66, draft GDC-67, draft GDC-68, and draft GDC-69.

The light load handling system is described in Monticello USAR Section 10.2.1, "Fuel Storage and Fuel Handling Systems."

In addition to the evaluations described in the Monticello USAR, Monticello's systems and components were evaluated for License Renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Monticello Nuclear Generating Plant License Renewal Safety Evaluation Report (SER), NUREG-1865, dated October 2006 (Reference 5). The license renewal evaluation associated with the light load handling system is documented in NUREG-1865, Section 3.0.3.2.18.

Technical Evaluation

Section 6.8 of the CLTR addresses the effect of EPU on plant systems that are not significantly affected. The Light Load Handling System (related to Refueling) is one of the systems that are not significantly affected. (Reference Table 2.5-3, Item 18)

Conclusion

Implementing EPU does not require introducing any new fuel designs. Therefore, the fuel handling analysis is not affected by the EPU. An evaluation of the light load handling system for the proposed EPU is not required. The proposed EPU is acceptable with respect to the light load handling system.

2.5.7 Additional Review Areas (Plant Systems)

Section 6.8 of the CLTR addresses the effect of EPU on plant systems that are not significantly affected.

[[

[redacted]	[redacted]	[redacted]
		]]

Based on experience and previous NRC reviews, all systems that are significantly affected by EPU are addressed in this report. Other systems, listed in Table 2.5-3, that are not addressed by this report are not significantly affected by EPU. The effect of EPU on the other systems at Monticello was confirmed to be consistent with the CLTR.

Table 2.5-1 Appendix R Fire Event Evaluation Results

Parameter	CLTP[1]	EPU	App. R Criteria
Cladding Heatup (PCT) (°F)	596	984[2]	≤ 1500
Primary System Pressure (psig) [3]	1273	1335	≤ 1375
Primary Containment Pressure (psig)	12.6	9.9[4]	≤ 56
Suppression Pool Bulk Temperature (°F)	193	197	≤ 281[5] ≤ 208 [6.]
Net Positive Suction Head [7.]	Yes	Yes	Adequate for ECCS systems using suppression pool water source

Notes:

1. As reported in Reference 7.
2. Due to longer core uncovery as a result of more decay heat at EPU, the PCT at EPU is higher than that at CLTP while operator action times remain unchanged for both CLTP and EPU
3. Bounded by peak pressure from MSIV closure with flux scram event.
4. This decrease from the CLTP to EPU value is due to the modeling and coding changes to the SHEX code used for the containment analysis since the CLTP analyses was performed.
5. Containment structural design limit.
6. Limited by analysis temperature of torus attached piping.
7. NPSH demonstrated adequate, see Section 2.6.5.

Table 2.5-2 SGTS Iodine Removal Capacity Parameters

[[		
		]]

Table 2.5-3 Basis for Classification of No Significant Effect

Item	[[			

Item	[[[illegible]	[illegible]	[illegible]	[illegible]

Item	[[			

Item	[[			

Item	[[[redacted]	[redacted]	[redacted]	[redacted]

Item	[[			

Item	[[			

Item	[[[illegible]	[illegible]	[illegible]	[illegible]
				]]

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's acceptance criteria for the primary containment functional design are based on (1) GDC-4, 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; (2) GDC-16, insofar as it requires that reactor containment be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment; (3) GDC-50, 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; (4) GDC-13, 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; and (5) GDC-64, 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.

Specific NRC review criteria are contained in SRP Section 6.2.1.1.C.

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC listed in the Regulatory Evaluation above, with the exception of current GDC-13, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-10, draft GDC-12, draft GDC-13, draft GDC-17, draft GDC-40, draft GDC-42, and draft GDC-49. Current GDC-13 is applicable to Monticello as described in USAR Section 14.7.4.

The primary containment is described in Monticello USAR Sections 5.2, "Primary Containment System" and 7.6.3, "Primary Containment Isolation System."

In addition to the evaluations described in the Monticello USAR, Monticello's systems and components were evaluated for License Renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Monticello Nuclear Generating Plant License Renewal Safety Evaluation Report (SER), NUREG-1865, dated October 2006 (Reference 5). The license renewal evaluation associated with the primary containment is documented in NUREG-1865, Sections 2.3.2.5 and 2.4.13. Management of aging effects on the primary containment is documented in NUREG-1865, Section 3.2.2.35 and 3.5.2.3.13.

Technical Evaluation

The USAR provides the containment responses to various postulated accidents that validate the design basis for the containment. Operation at the EPU RTP causes changes to some of the conditions for the containment analyses. For example, the short-term DBA LOCA containment response during the reactor blowdown is governed by the blowdown flow rate. This blowdown flow rate is dependent on the reactor initial thermal-hydraulic conditions, such as vessel dome pressure and the mass and energy of the vessel fluid inventory, which change slightly at the EPU RTP. Also, the long-term heatup of the suppression pool following a LOCA or a transient is governed by the ability of the RHR system to remove decay heat. Because the decay heat depends on the initial reactor power level, the long-term containment response is affected by EPU. The containment pressure and temperature responses have been reanalyzed, as described in Section 2.6.1.1, to demonstrate the Monticello's capability to operate at EPU RTP.

The analyses were performed in accordance with RG 1.49 and ELTR1 using GEH codes and models (References 8 through 11 and 34). Confirmatory calculations with the SHEX code and the NRC-accepted HXSIZ code show a difference of less than 1°F in peak suppression pool temperature between the two codes. Therefore, the use of the SHEX code for Monticello complies with the NRC requirements for use in EPU analyses presented in Reference 12.

The EPU containment analyses were performed assuming a maximum service water temperature of 90°F, which is consistent with the limiting CLTP design basis for this parameter.

The effect of EPU on the containment dynamic loads due to a LOCA or SRV discharge has also been evaluated as described in Section 2.6.1.2. These loads were previously defined generically during the Mark I Containment Long Term Program (LTP) as described in Reference 13. Plant specific dynamic loads were also defined (References 14 and 15), which were accepted by the NRC in Reference 16. The evaluation of the LOCA containment dynamic loads is based primarily on the results of the short-term analysis described in Section 2.6.3.1.2. The SRV discharge load evaluation is based on no changes in the SRV opening setpoints at EPU conditions.

2.6.1.1 Containment Pressure and Temperature Response

Short-term and long-term containment analyses results are reported in the USAR. The short-term analysis is directed primarily at determining the drywell pressure response during the initial blowdown of the reactor vessel inventory to the containment following a large break inside the drywell. The long-term analysis is directed primarily at the suppression pool temperature response, considering the decay heat addition to the suppression pool. Peak values of the containment pressure and temperature responses to the DBA LOCA are given in Table 2.6-1. The effect of EPU on the events yielding the limiting containment pressure and temperature responses are provided below.

2.6.1.1.1 Long-Term Suppression Pool Temperature Response

a) Bulk Pool Temperature

The long-term bulk pool temperature response with EPU was evaluated for the DBA LOCA. The analysis was performed at 102% of the EPU RTP. Table 2.6-1 compares the calculated peak values for LOCA bulk pool temperature. The current analyses have been performed using an RHR containment cooling capability (K-value) of 147 BTU/sec°F, as is identified in Reference 17. However, the design basis analysis, which assumes containment cooling using the suppression pool cooling mode of RHR, uses a slightly modified RHR containment cooling capability from that used in the USAR. For this analysis, a heat exchanger K-value was used that improves with the temperature of the hot inlet liquid as shown in the table below. Below 110°F and above 195°F the K-value is assumed constant, and varies linearly with inlet temperature between the values shown in the table.

Hot Inlet Temperature (°F)	Heat Exchanger K-Value (BTU/sec°F) (based on 90°F service water)	
	1 RHR / 1 RHRSW	2 RHR / 2 RHRSW
≤ 110	146.5	190.8
125	147.6	
160	149.7	
165		194.5
≥195	151.6	196.1

This approach provides a means to account for the variation in the heat exchanger performance with suppression pool temperature, which can affect the long-term containment response. The difference between the maximum and minimum calculated value for K using this approach is only 3.5%. Consequently, the effect on heat exchanger performance of using this approach versus using a constant value for K is relatively small.

Additionally, there is no difference between the methodology used to calculate the varying K values and the constant K values. In either case the values for K 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.

The EPU analysis was performed using a realistic decay heat model (ANS/ANSI 5.1) as was used in the current USAR analysis, with the addition of a 2σ uncertainty for additional conservatism. The EPU analysis also credits passive heat sinks in the containment as identified in Reference 2, whereas the current USAR analysis does not credit passive heat sinks in the containment. As a final change, although the USAR methodology assumes thermal equilibrium between the suppression pool and the wetwell airspace throughout the event, the EPU analysis assumes thermal equilibrium between the suppression pool and the wetwell airspace only for the first 30 seconds of the event when the suppression pool is greatly agitated due to vent flow to the wetwell. After 30 seconds, heat and mass transfer between the suppression pool and wetwell airspace is mechanistically modeled. Because heat is transferred to the wetwell through the suppression pool, this change is acceptable because it provides a more conservative estimate for the suppression pool response. Benchmark calculations were made as requested by the NRC in Reference 12. The Monticello calculated peak bulk suppression pool temperatures are provided in Table 2.6-1 for both 102% of current rated power and 102% of EPU RTP. This comparison shows that EPU results in an increase of 10°F in peak suppression pool temperature, based on current methodology.

Based on the analysis and limit values shown in Table 2.6-1, the peak bulk pool temperature with EPU is acceptable from a structural design standpoint.

b) Local Pool Temperature with SRV Discharge

The local pool temperature limit for SRV discharge is specified in NUREG-0783 (Reference 17). The NUREG guidance was developed due to concerns regarding unstable condensation observed at high pool temperatures in plants without quenchers. Reference 18 has eliminated the local pool temperature limit for SRV discharge for Monticello.

2.6.1.2 Containment Dynamic Loads

2.6.1.2.1 Loss-of-Coolant Accident Loads

The LOCA containment dynamic loads analysis for EPU is based primarily on the short-term LOCA analyses. These analyses were performed as described in Section 2.6.3.1.2, using the Mark I LTP method, except that the break flow was calculated using a more detailed RPV model (Reference 11). The application of this model to EPU containment evaluations is identified in ELTR1. These analyses provide calculated values for the controlling parameters for the dynamic loads throughout the blowdown. The key parameters are drywell and wetwell pressure, vent flow rates and suppression pool temperature. The LOCA dynamic loads for EPU include pool swell, condensation oscillation (CO), and chugging loads. For Mark I plants like Monticello, the vent thrust loads were also evaluated.

The short-term containment response conditions with EPU are within the range of test conditions used to define the pool swell and condensation oscillation loads for Monticello. The peak drywell pressure from these analyses is given in Table 2.6-1. The long-term response conditions at EPU conditions when chugging would occur are within the conditions used to define the chugging loads. The vent thrust loads at EPU conditions were calculated to be less than the plant specific values calculated during the Mark I Containment LTP. Therefore, the LOCA dynamic loads are not affected by EPU.

2.6.1.2.2 Safety Relief Valve Loads

The Safety Relief Valve (SRV) air-clearing loads include SRV discharge line (SRVDL) loads, suppression pool boundary pressure loads and drag loads on submerged structures. The SRV opening setpoint pressure, the initial water leg in the SRVDL, SRVDL geometry, and suppression pool geometry, influences these loads. The SRV loads were evaluated for two different actuation phases: initial actuation and subsequent actuation.

For the initial SRV actuation following an event involving RPV pressurization, the only parameter change potentially introduced by EPU, which can affect the SRV loads definition, is an increase in SRV opening setpoint pressure. However, the changes proposed for EPU do not include an increase in the SRV opening setpoint pressure. EPU may reduce the time between subsequent SRV actuations, which may affect the load definition for subsequent actuations.

The SRV opening values, which are the basis for the SRVDL loads and the SRV loads on the suppression pool boundary and submerged structures are not changed. The effect of EPU on the load definition for subsequent SRV actuations was evaluated. The load definition for subsequent SRV actuations is not affected because SRV low-low set logic has been incorporated at Monticello to ensure subsequent actuations occur after the water level in the SRVDL returns to normal (Reference 20). Therefore, EPU does not affect the SRV loads or load definitions.

2.6.1.3 Containment Isolation

The system designs for containment isolation are not affected by EPU. The capabilities of isolation actuation devices to perform during normal operations and under post-accident conditions have been determined to be acceptable. Therefore, the Monticello containment isolation capabilities are not adversely affected by the EPU.

2.6.1.4 Generic Letter 89-16

In response to GL 89-16, Monticello installed a hardened wetwell vent system. The Monticello design of the hardened wetwell vent was based on 1670 MWt (OLTP). Therefore, at the EPU RTP conditions, the existing hardened wetwell vent will exhaust a smaller percentage of RTP. Based on the as-built design, the hardened wetwell vent will exhaust approximately 1.05% RTP at 2004 MWt (EPU RTP) and is designed to be operational during an SBO.

2.6.1.5 Generic Letter 96-06

The NRC staff acceptance of the NSP response to Generic Letter 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions," is provided in Reference 22. These responses were reviewed for continued applicability at EPU conditions. The issues identified in the GL 96-06 review were addressed through a combination of analysis, procedural changes, administrative controls, and modifications that are unaffected by EPU. Therefore, the accepted NSP response to Generic Letter 96-06 remains valid for EPU.

Conclusion

NSPM has evaluated the containment temperature and pressure transient and accounted for the increase of mass and energy resulting from the proposed EPU. The evaluation indicates that containment systems will continue to provide sufficient pressure and temperature mitigation capability to ensure that containment integrity is maintained. The evaluation further indicates 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 the current licensing basis following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to primary containment functional design.

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's acceptance criteria for subcompartment analyses are based on (1) GDC-4, 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, and (2) GDC-50, insofar as it requires that containment subcompartments be designed with sufficient margin to prevent fracture of the structure due to the calculated pressure differential conditions across the walls of the subcompartments.

Specific NRC review criteria are contained in SRP Section 6.2.1.2.

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern

States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC listed in the Regulatory Evaluation above, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-40, draft GDC-42, and draft GDC-49.

The primary containment is described in Monticello USAR Sections 5.2, "Primary Containment System" and 12.2.2.1.1, "Structure Description."

In addition to the evaluations described in the Monticello USAR, Monticello's systems and components were evaluated for License Renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Monticello Nuclear Generating Plant License Renewal Safety Evaluation Report (SER), NUREG-1865, dated October 2006 (Reference 5). The license renewal evaluation associated with the primary containment is documented in NUREG-1865, Sections 2.3.2.5 and 2.4.13. The management of the effects of aging on the primary containment is documented in NUREG-1865, Section 3.3.3.3.5 and 3.5.2.3.13.

Technical Evaluation

The subcompartment of concern is the annular space between the RPV and the biological shield wall. The annulus pressure load on the biological shield wall due to a postulated recirculation line break at EPU conditions (2044 MWt (102 percent RTP) and MELLLA, and 105 percent Core Flow) is compared to the load at OLTP/MELLLA. The OLTP/MELLLA annulus pressure load (40 psi) remains bounding compared to the load and energy at EPU conditions. The biological shield wall design remains adequate at EPU conditions because there is substantial margin between the annulus pressure load and the biological shield wall structural design value of 58 psid.

In addition, this analysis evaluated operation at the Minimum Recirculation Pump Speed point on the MELLLA line (1210.4 MWt/43.3% Core Flow). Although not part of the current design basis, this power/flow point is more limiting for the annulus pressurization. The annulus pressure load on the biological shield wall at this power/flow point is 42.3 psi, which remains below the design value of 58 psid.

This evaluation also included consideration of missile energy for shield bricks installed in biological shield wall penetrations. These bricks are postulated to be ejected by annulus pressure and impact the primary containment liner. Missile energies below 18.4 ft. kips will not penetrate the liner. Missile energies for MELLLA, 100% and 105% core flows remain below the allowable energy of 18.4 ft. kips. Missile energies at the Minimum Recirc Pump Speed point, not part of the current design basis, reach energies of 19.0 ft. kips. This point is within the Controlled Entry Region for core stability and therefore prompt exit is required. To increase margins, these shield bricks will be removed by modification.

Conclusion

NSPM has evaluated the change in predicted pressurization resulting from the increased mass and energy release. The evaluation indicates 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 failure of the structure due to pressure difference across the walls at design basis conditions following implementation of the proposed EPU. Missile energy for shield bricks exceeding the threshold was identified at a non-current design basis operating point on the power/flow map and a modification will remove these shield bricks. Based on this, the plant will continue to meet the current licensing basis for the proposed EPU. Therefore, the proposed EPU is 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

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's acceptance criteria for mass and energy release analyses for postulated LOCAs are based on (1) GDC-50, insofar as it requires that sufficient conservatism be provided in the mass and energy release analysis to assure that containment design margin is maintained and (2) 10 CFR Part 50, Appendix K, insofar as it identifies sources of energy during a LOCA.

Specific NRC review criteria are contained in SRP Section 6.2.1.3.

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of

the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC listed in the Regulatory Evaluation above, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-49.

The mass and energy release analysis is described in Monticello USAR Section 5.2.3.2, "Containment Response to a Loss of Coolant Accident."

Technical Evaluation

2.6.3.1.1 Short-Term Gas Temperature Response

The drywell airspace temperature limit is specified in Table 2.6-1. The limit is increased for EPU from 335°F to 340°F. The revised limit is based on a bounding analysis of the superheated gas temperature reached during an adiabatic (isenthalpic) expansion of reactor steam with the physically highest initial enthalpy to a drywell at a pressure of 35 psig. The changes in the reactor vessel conditions at EPU increase the calculated peak drywell gas temperature following a LOCA by 2°F, but remains bounded by the new drywell airspace temperature limit.

Short-term containment response analyses for DBA LOCA demonstrate that operation at EPU RTP does not result in exceeding the containment design limits. These analyses cover the blowdown period when the maximum drywell airspace temperature occurs. The analyses were performed at 102% of EPU RTP, using the methods reviewed and accepted by the NRC during the Mark I containment LTP. The calculated peak drywell airspace temperatures are provided in Table 2.6-1. Table 2.6-1 also shows the values from calculations using CLTP using the same methods. EPU increases the calculated peak drywell gas temperature 2°F.

A short-term wetwell gas space peak temperature response is not reported for EPU, being bounded by the long-term wetwell gas space peak temperature response.

2.6.3.1.2 Short-Term Containment Pressure Response

Short-term containment response analyses were performed for the limiting DBA LOCA, which assumes a double-ended guillotine break of a recirculation suction line, to demonstrate that EPU does not result in exceeding the containment design limits. The short-term analysis covers the blowdown period during which the maximum drywell pressures and differential pressures between the drywell and wetwell occur. These analyses were performed at 102% of the EPU RTP, using methods reviewed and accepted by the NRC during the Mark I Containment LTP with the break flow calculated using a more detailed RPV model (Reference 11) previously approved by the NRC. The EPU analysis uses LAMB (Reference 11) with Moody's Slip critical flow model (Reference 10) to generate the blowdown flow rates, which are then used as inputs to M3CPT. This approach differs from that for the current USAR analysis, which uses the Homogeneous Equilibrium Model. Application of the LAMB blowdown model for the EPU analysis is identified in Reference 2. The results of these short-term analyses are summarized in Table 2.6-1 for comparison to the drywell design pressure. Also included in Table 2.6-1 is a comparison of the peak drywell pressure calculated using the current method at CLTP and at EPU RTP with equivalent input values. The drywell pressure increases slightly as a result of EPU. Finally, the peak pressure for the current rated power reported in the USAR is included. As shown by these results, the design pressure bounds the maximum drywell pressure values at the EPU conditions.

Conclusion

NSPM has evaluated mass and energy release and accounted for the sources of energy identified in 10 CFR Part 50, Appendix K. Based on this, the mass and energy release analysis meets the requirements in the current licensing basis for ensuring that the analysis is conservative. Therefore, the proposed EPU is acceptable with respect to mass and energy release for postulated LOCA.

2.6.4 Combustible Gas Control in Containment

Regulatory Evaluation

Following a LOCA, hydrogen and oxygen may accumulate inside the containment due to 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's acceptance criteria for combustible gas control in containment are based on (1) 10 CFR 50.44, insofar as it requires that plants be provided with the capability for controlling combustible gas concentrations in the containment atmosphere; (2) GDC-5, 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; (3) GDC-41, 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; (4) GDC-42, insofar as it requires that systems required by GDC-41 be designed to permit appropriate periodic inspection; and (5) GDC-43, insofar as it requires that systems required by GDC-41 be designed to permit appropriate periodic testing. Additional requirements based on 10 CFR 50.44 for control of combustible gas apply to plants with a Mark III type of containment that do not rely on an inerted atmosphere to control hydrogen inside the containment.

Specific NRC review criteria are contained in SRP Section 6.2.5.

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC listed in the Regulatory Evaluation above, with the exception of current GDC-41, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-4, draft GDC-62, draft GDC-63, draft GDC-64, and draft GDC-65. There is no draft GDC directly associated with current GDC-41.

Combustible gas control in containment is described in Monticello USAR Sections 5.2.2.6, "Primary Containment Atmospheric Control System," 5.2.2.7, "Containment Atmosphere Monitoring System," and 5.2.3.4, "Hydrogen and Oxygen Generation in Containment." Monticello's containment is inerted during power operation.

Technical Evaluation

10 CFR 50.44 was revised in September 2003 and no longer defines a design basis LOCA hydrogen release and eliminates the requirements for hydrogen control systems to mitigate such releases.

Monticello has adopted the revised ruling per Monticello License Amendment Number 138, issued in May 2004, which eliminated the requirements for hydrogen recombiners. The hydrogen recombiners have since been abandoned in place.

However, NSPM made commitments to maintain the hydrogen and oxygen monitoring systems capable of diagnosing beyond design basis accidents. EPU has no effect on the design of these systems or on the ability of these systems to perform their intended functions.

Conclusion

NSPM has evaluated the containment hydrogen and oxygen monitoring systems with respect to EPU and determined that there is no effect on the design of these systems or on the ability of these systems to perform their intended functions. Therefore, the proposed EPU is acceptable with respect to combustible gas control in containment.

2.6.5 Containment Heat Removal

Regulatory Evaluation

Fan cooler systems, spray systems, and residual heat removal (RHR) systems are provided to remove heat from the containment atmosphere and from the water in the containment wetwell.

The NRC's acceptance criteria for containment heat removal are based on GDC-38, 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.

Specific NRC review criteria are contained in SRP Section 6.2.2, as supplemented by Draft Guide (DG) 1107.

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed

General Design Criteria can usually be made. For the current GDC listed in the Regulatory Evaluation above, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-41 and draft GDC-52.

The containment heat removal systems are described in Monticello USAR Sections 5.2.2.5.1, "Spray Cooling System," and 6.2.3, "Residual Heat Removal System (RHR)."

In addition to the evaluations described in the Monticello USAR, Monticello's systems and components were evaluated for License Renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Monticello Nuclear Generating Plant License Renewal Safety Evaluation Report (SER), NUREG-1865, dated October 2006 (Reference 5). The license renewal evaluation associated with the containment heat removal system is documented in NUREG-1865, Section 2.3.2.7. The management of the effects of aging on the containment heat removal system is documented in NUREG-1865, Section 3.2.2.3.7.

Technical Evaluation

Long-Term Suppression Pool Temperature Response

The long-term bulk pool temperature response with EPU was evaluated for the DBA LOCA. The analysis was performed at 102% of the EPU RTP. Table 2.6-1 compares the calculated peak values for LOCA bulk pool temperature. The current analyses have been performed using the same RHR containment cooling capability used in the USAR Section 4.1.1.1 analysis, a K-value of 147 BTU/sec°F per heat exchanger. However, the design basis analysis, which assumes containment cooling using the suppression pool cooling mode of RHR, uses a slightly modified RHR containment cooling capability from that used in the USAR (see Section 2.6.1.1.1.a).

The EPU analysis was performed using a realistic decay heat model (ANS/ANSI 5.1), as was used in the current USAR analysis, with the addition of a 2σ uncertainty for additional conservatism. The EPU analysis also credits passive heat sinks in the containment as identified in Reference 2, whereas the current USAR analysis does not credit passive heat sinks in the containment. As a final change, although the USAR methodology assumes thermal equilibrium between the suppression pool and the wetwell airspace throughout the event, the EPU analysis assumes thermal equilibrium between the suppression pool and the wetwell airspace only for the first 30 seconds of the event when the suppression pool is greatly agitated due to vent flow to the wetwell. After 30 seconds, heat and mass transfer between the suppression pool and wetwell airspace is mechanistically modeled. Because heat is transferred to the wetwell through the suppression pool, this change is acceptable because it provides a more conservative estimate for the suppression pool response. Benchmark calculations were made as requested by the NRC in Reference 12. The Monticello calculated peak bulk suppression pool temperatures are provided in Table 2.6-1 for both 102% of current rated power and 102% of EPU RTP. This comparison shows that EPU results in an increase of approximately 10°F in peak suppression pool temperature, based on current methodology.

The containment response used for NPSH evaluations was calculated using Monticello-specific inputs to maximize suppression pool temperature and minimize containment pressure for the DBA LOCA analysis. The suppression pool temperature and corresponding wetwell pressure for the short-term and long-term NPSH containment analyses were used in the evaluation of the available NPSH for the CS and the RHR pumps. The containment responses used for NPSH evaluations for Special Events (such as ATWS, SBO, and Appendix R) used Monticello-specific nominal inputs to provide realistic maximized suppression pool temperatures and corresponding realistic minimized wetwell pressures.

ECCS Net Positive Suction Head

Basis of Limiting ECCS Configuration for NPSH Evaluations

The ECCS pump suction piping was modeled to evaluate the effect on ECCS pump NPSH required for the events of concern. These events are described below. The piping configuration of RHR pump "C" and Core Spray pump "A" causes those pumps to be limiting for evaluation of NPSH required. The Appendix R events are mitigated by use of Alternate Shutdown Systems that use RHR pump "B" and Core Spray pump "B".

DBA LOCA

Following a LOCA, the RHR and CS pumps operate to provide the required core and containment cooling. Adequate NPSH margin (NPSH available minus NPSH required) is required during this period to assure the essential pump operation. The NPSH margins for the ECCS pumps were evaluated for the limiting conditions following a DBA LOCA. The limiting NPSH conditions occur during either short-term or long-term post-LOCA pump operation and depend on the total pump flow rates, debris loading on the suction strainers, and suppression pool temperature.

Monticello is currently licensed for containment overpressure credit at CLTP for short-term and long-term DBA LOCA ECCS pump operation per Reference 33.

The NPSH margins were calculated based on the design required flow rates during the short-term and long-term DBA LOCA ECCS pump operation. The pump flow rates for the short-term case are 17282 gpm total RHR flow (four RHR pumps at runout flow) and 8489 gpm total CS flow (two CS pumps at runout flow). The pump flow rates for the long-term case are 4000 gpm total RHR flow and 3035 gpm for CS pump "A" and 3029 gpm for CS pump "B". The debris loading on the suction strainers for EPU is the same as the CLTP condition. The assumptions in the ECCS NPSH calculations for friction loss, static head, flow, and NPSH required have not been changed since the Monticello responses to NRC GL 97-04 (Reference 23). EPU RTP operation increases the reactor decay heat, which increases the heat addition to the suppression pool following a LOCA. As a result, the suppression pool water temperature and containment pressure increase. Therefore, only changes in vapor pressures corresponding to the increases in suppression pool temperatures affect the NPSH margins.

For the CLTP and the EPU analyses, the assumptions used in containment response analyses maximized the suppression pool temperature and minimized the containment pressure. These include operation of the RHR pumps for suppression pool cooling after 10 minutes. The analysis then assumes that the operators establish long-term containment cooling to control ECCS flow and containment pressure until overpressure is no longer required to meet NPSH requirements.

Short-term and long-term containment analyses were performed for EPU conditions (short-term from 0 to 600 seconds and long-term from 600 seconds until the end of the event). The short and long-term analysis indicates that overpressure is available from the beginning of the event until the end of the event with Technical Specification containment leakage assumed (1.2 weight % of containment air/day).

During the event, it is shown that the ECCS pumps (RHR and CS) require overpressure to be credited to satisfy pump NPSH requirements.

Table 2.6-2 and Figure 2.6-1A and Figure 2.6-1B provide the results of the short-term and long-term containment response, including suppression pool temperature, required containment pressure to satisfy the NPSH Required (NPSHR), and available wetwell pressure based on the use of 3% NPSHR curves. Figure 2.6-1C is provided for comparison and shows the results if based on the use of 1% NPSHR curves except the short term CS NPSHR that utilized the 3% curve value.

Based on the above, Monticello is requesting continued approval of overpressure credit to meet both the short-term and long-term NPSH requirements for the LOCA.

Appendix R Fire

One RHR pump and one CS pump are required to operate during the Appendix R fire event. EPU reactor thermal power (RTP) operation increases the reactor decay heat, which increases the heat addition to the suppression pool following this event (see Section 2.5.1.4). As a result, the peak suppression pool water temperature and peak containment pressure increase. The NPSH margins for the ECCS pumps were evaluated for the limiting conditions following two Appendix R events (Case 1 and Case 2). Case 1 assumes one (1) Stuck Open Relief Valve (SORV) and Case 2 assumes no SORV. The NPSH evaluation for these events shows that adequate NPSH margin exists during the limiting Appendix R event if the available containment pressure is credited.

Monticello is currently licensed for containment overpressure credit at CLTP for Appendix R ECCS pump operation per Reference 33.

Containment analyses were performed for the Appendix R event at EPU conditions. The analysis indicates that suppression pool overpressure is available during the event. CS pump "B" was assumed to be operating at 3029 gpm and RHR pump "B" was assumed to be operating at 4000 gpm.

Table 2.6-3 and Figure 2.6-2 (Case 1) and Table 2.6-4 and Figure 2.6-3 (Case 2) provide the results of the containment response, including suppression pool temperature, the containment pressure necessary to satisfy the NPSHR, and the available wetwell pressure. All NPSHR values are based on the 3% NPSHR curves versus the 1% NPSHR curves used in previous evaluations.

The 1% NPSHR curves are provided for information. NPSHR remains sufficient for EPU conditions and no increase in the amount of credited pressure is necessary. Adequate margin to the peak containment overpressure value of 20.36 psia previously approved on June 2, 2004 under Amendment No. 139 to Facility Operating License No. DPR-022 is available.

Based on the above, Monticello is requesting continued approval of overpressure credit to meet NPSH requirements during an Appendix R event.

Station Blackout (SBO)

HPCI system operation is credited during the SBO event. The HPCI pump will initially take suction from the CST until the suppression pool temperature exceeds the CST temperature. At this point, the HPCI pump will take suction from the suppression pool until three (3) hours into the event, at which point the HPCI pump suction will be placed back on the CST. HPCI flowrate was assumed to be 3000 gpm.

EPU RTP operation increases the reactor decay heat, which increases the heat addition to the suppression pool following this event (see Section 2.3.5).

The NPSH evaluation for this event shows that adequate NPSH is available during the limiting SBO event without requiring credit for containment overpressure. The evaluation results are provided in Table 2.6-5.

Anticipated Transient Without Scram (ATWS)

Following a postulated Anticipated Transient Without Scram (ATWS), the RHR and CS pumps operate to provide the required core and containment cooling. Adequate NPSH margin (NPSH available minus NPSH required) is required during this period to assure the essential pump operation. The NPSH margins for the ECCS pumps were evaluated for the limiting conditions for the following three ATWS events (PRFO Case 1, PRFO Case 2, and LOOP).

The PRFO Case 1 analysis assumed CS pump "A" is operating at 3035 gpm and four RHR pumps are operating in Suppression Pool Cooling mode at 4000 gpm each. The PRFO Case 2 analysis assumed CS pump "A" is operating at 3035 gpm, four RHR pumps are operating, two in Suppression Pool Cooling mode and two in Containment Spray Mode. Each RHR pump has a flow rate of 4000 gpm. The LOOP analysis assumed CS pump "A" is operating at 3035 gpm and two RHR pumps are operating in Containment Spray Mode.

EPU RTP operation increases the reactor decay heat, which increases the heat addition to the suppression pool following this event (see Section 2.8.5.7). As a result, the peak suppression pool water temperature and peak containment pressure increase. Containment analyses were performed for the ATWS event at EPU conditions. The analysis indicates that required overpressure is available during the event. Thus, adequate NPSH margin exists during the ATWS event if the available containment overpressure is credited.

Tables 2.6-6, 2.6-7, and 2.6-8 and Figures 2.6-4, 2.6-5 and 2.6-6 provide the results of the containment response including suppression pool temperature, required containment pressure to satisfy the NPSHR, and available wetwell pressure, for PRFO Case 1, PRFO Case 2, and the LOOP Case respectively. NSPM is requesting that the staff review be based on the use of 3% NPSH required curves. Both 1% and 3% NPSHR curves are provided for information. Adequate margin to the peak containment overpressure value of 20.36 psia previously approved for DBA and Appendix R events is available.

Based on the above, Monticello is requesting approval of overpressure credit to meet NPSH requirements during an ATWS event.

Small Steam Line Break Accident (SBA)

Following a 0.01 ft^2 Small Steam Line Break Accident (SBA), the RHR and CS pumps operate to provide the required core and containment cooling as well as maintain RPV water level. Adequate NPSH margin (NPSH available minus NPSH required) is required during this period to assure the essential pump operation. The NPSH margins for the ECCS pumps were evaluated for the limiting conditions following an SBA.

EPU RTP operation increases the reactor decay heat, which increases the heat addition to the suppression pool following this event. As a result, the peak suppression pool water temperature and peak containment pressure increase. Containment analyses were performed for the SBA event at EPU conditions. The analysis indicates that overpressure is available during the event. Thus, adequate NPSH margin exists during the SBA event if the available containment overpressure is credited.

The SBA analysis assumed one RHR pump in LPCI injection mode from 0-600 seconds at a flow of 4320 gpm. At 600 seconds LPCI injection is secured and one RHR pump is operating in Containment Spray mode for the remainder of the event at a flow of 4000 gpm. One CS pump is assumed to be operating at 3020 gpm. The CS pump is expected to maintain water level during this event, and actual flow rate is expected to be significantly less (approximately 200 gpm).

Table 2.6-9 and Figures 2.6-7 and 2.6-8 provide the results of the containment response including suppression pool temperature, required containment pressure to satisfy the NPSHR, and available wetwell pressure, for the SBA event. All NPSHR values are based on the 3% NPSHR curves. The amount of overpressure credit requested for this event is below the 20.36 psia peak value for DBA LOCA. NSPM is requesting approval of overpressure credit to meet NPSH requirements during an SBA event.

Suction Strainer Debris Loading

The methodology used by Monticello to determine the amount of debris generated and transported to the strainers is generally based on NEDO-32686, the BWROG Utility Resolution Guidance for ECCS Suction Strainer Blockage (Reference 24). The assumption used for protective coatings, specifically inorganic zinc with epoxy topcoat, was 85 lbm. This is the bounding value recommended by NEDO-32686, Section 3.2.2 and is not affected by EPU.

The ECCS strainer loading calculations for Monticello are based on the zone of influence and debris generation developed in NEDO-32686. EPU conditions do not affect the variables used in the determination of the zone of influence and calculated debris volumes. Therefore, ECCS pump head losses due to debris accumulation on ECCS suction strainers located in the suppression pool are not affected by EPU.

Conclusion

NSPM has evaluated the containment heat removal systems and addressed the effects of the proposed EPU. The evaluation indicates that the systems will continue to meet their operational criteria with respect to rapidly reducing the containment pressure and temperature following a LOCA and maintaining them at acceptably low levels. Therefore, the proposed EPU is acceptable with respect to containment heat removal systems.

2.6.6 Secondary Containment Functional Design

Regulatory Evaluation

The secondary containment structure and supporting systems of dual containment plants 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's acceptance criteria for secondary containment functional design are based on (1) GDC-4, 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; and (2) GDC-16, 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.

Specific NRC review criteria are contained in SRP Section 6.2.3.

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC listed in the Regulatory Evaluation above, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-10, draft GDC-40, and draft GDC-42.

The secondary containment systems are described in Monticello USAR Sections 5.3, "Secondary Containment System and Reactor Building," and 12.2.2.1, "Reactor Building."

In addition to the evaluations described in the Monticello USAR, Monticello's systems and components were evaluated for License Renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Monticello Nuclear Generating Plant License Renewal Safety Evaluation Report (SER), NUREG-1865, dated October 2006 (Reference 5). The license renewal evaluation associated with the secondary containment systems is documented in NUREG-1865, Sections 2.3.2.8 and 2.4.15. Management of aging effects on the secondary containment systems is documented in NUREG-1865, Sections 3.2.2.3.8 and 3.5.2.3.15.

Technical Evaluation

The Standby Gas Treatment System (SGTS) is designed to maintain secondary containment at a negative pressure and to filter the exhaust air for removal of fission products potentially present during abnormal conditions. By limiting the release of airborne particulates and halogens, the SGTS limits off-site dose following a postulated DBA. The fission product control and removal function evaluation is described in Section 2.5.2.

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The design flow capacity of the SGTS was selected to maintain the secondary containment at the required negative pressure to minimize the potential for exfiltration of air from the reactor building. [[

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Conclusion

NSPM has evaluated the ability of the secondary containment to provide an essentially leak-tight barrier against uncontrolled release of radioactivity to the environment. The evaluation indicates 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 secondary containment and associated systems will continue to meet the requirements of the current licensing basis. Therefore, the proposed EPU is acceptable with respect to secondary containment functional design.

Table 2.6-1 Containment Performance Results

Parameter	CLTP from USAR	CLTP with EPU Method[1]	EPU	Limit
Peak Drywell Pressure (psig)	39.5	43.4	44.1[2]	56[3]
Peak Drywell Temperature (°F)	335	336	338[2]	340[4]
Peak Drywell Wall Temperature (°F)	273[5]	277[5]	278[5]	281
Peak Bulk Suppression Pool Temperature (°F)	194.2	193[6]	203 / 207[7]	208[8]
Peak Wetwell Pressure (psig)	31.2	31.3	32.7	56[3]

Notes:

1. The EPU Method, which was used for the EPU analysis, uses the EPU RTP analysis method with CLTP inputs. The EPU Method includes a more bounding initial containment pressure of 3.0 psig as compared with the CLTP of the USAR, which assumed an initial containment pressure of 2.0 psig. The EPU method also assumes the initial reactor power is at 102% of the RTP.
2. Includes an increase in the assumed initial containment pressure from 2.0 psig of the method of the USAR analysis to 3.0 psig for the EPU Method.
3. The design pressure for the drywell and wetwell is 56 psig. Maximum internal pressure is 62 psig, as shown in USAR Table 5.2-1.
4. Limit for the drywell environmental temperature is increased for EPU from 335°F shown in USAR Table 5.2-8 to 340°F.
5. Calculated assuming a 0.50 sqft steam break into the drywell with UCHIDA condensing heat transfer to the drywell wall to the saturation temperature at the drywell pressure, and initiation of drywell sprays at 10 minutes.
6. Reduction in peak bulk pool temperature from 194.2°F shown in USAR Table 5.2-4- to 193°F shown above for CLTP with EPU Method is primarily due to use of a K-value that increases with increasing hot inlet water temperature.
7. The first value is the peak suppression pool temperature for the DBA LOCA with direct suppression pool cooling, 90°F service water temperature, and an RHR heat exchanger K-value that increases with increasing hot inlet water temperature. The second number is the peak suppression pool temperature for the same DBA LOCA and 90°F service water temperature, but with containment cooling using containment sprays and a constant K-value of 147 BTU/sec°F, used for NPSH evaluation.
8. The limit for peak bulk pool temperature, determined as the design temperature for the torus-attached piping, is increased for EPU from 196.7°F (Reference 19) to 208°F.

Table 2.6-2 Design Basis Accident (DBA) Loss of Coolant Accident (LOCA)

Time	EPU Suppression Pool Temperature	Required Containment Pressure at EPU Conditions	CLTP Licensed Containment Overpressure Credit	Margin between Current Licensed COP credit and Required Containment Pressure at EPU Conditions	Minimum Wetwell Pressure Available at EPU Conditions	Margin between Available Wetwell Pressure and Required Containment Pressure, at EPU Conditions
				(Licensed COP minus Required Containment Pressure)		(Available Wetwell Pressure at EPU minus Required Containment Pressure at EPU)
Seconds	°F	psia	psia	psia	psia	psia
96	134.0	13.10	16.26	3.16	30.41	17.31
185	138.0	13.53	16.26	2.73	24.57	11.04
358	150.0	14.69	16.26	1.57	16.62	1.93
476	155.0	15.37	16.26	0.89	16.57	1.20
590	158.0	15.82	16.26	0.44	16.96	1.14
978	158.3	11.35	16.26	4.91	16.98	5.63
5558	184.0	15.15	18.26	3.11	20.24	5.09
21517	205.0	19.71	20.36	0.65	23.88	4.17
34748	207.1	20.27	20.36	0.09	24.26	3.99
46321	206.0	19.98	20.36	0.38	24.06	4.08
80325	197.0	17.82	19.26	1.44	22.11	4.29
233659	166.6	12.73	15.26	2.53	16.99	4.26
472096	149.8	10.99	15.26	4.27	15.05	4.06

Table 2.6-3 Appendix R Fire (Case #1)

Time	Suppression *Pool* Temperature 120% OLTP	Required *Containment* Pressure 120% OLTP	Licensed *Containment* Overpressure CLTP	Margin between Licensed COP and Required Containment Pressure	Minimum Wetwell Pressure Available 120 % OLTP	Margin between Available Wetwell Pressure and Required Containment Pressure
Seconds	°F	psia	psia	(Licensed COP – Required Containment Pressure, psia)	psia	(Available Wetwell Pressure – Required Containment Pressure, psia)
2023	133.9	8.72	18.26	9.54	14.83	6.11
4003	151.3	10.25	18.26	8.01	15.24	4.99
6010	164.9	11.70	18.26	6.56	16.00	4.30
8010	172.9	12.75	18.26	5.51	16.84	4.09
10008	178.2	13.54	19.56	6.02	17.67	4.13
12001	182.2	14.20	19.56	5.36	18.37	4.17
14003	185.3	14.74	19.56	4.82	18.95	4.21
16007	187.8	15.20	20.36	5.16	19.44	4.24
18013	189.8	15.59	20.36	4.77	19.85	4.26
20001	191.3	15.89	20.36	4.47	20.18	4.29
22001	192.5	16.13	20.36	4.23	20.45	4.32
24012	193.4	16.32	20.36	4.04	20.67	4.35
26005	194.1	16.46	20.36	3.90	20.84	4.38
28006	194.7	16.59	20.36	3.77	20.99	4.40
30002	195.0	16.65	20.36	3.71	21.10	4.45
32011	195.3	16.72	20.36	3.64	21.18	4.46
34002	195.4	16.74	20.36	3.62	21.25	4.51
36005	195.4	16.74	20.36	3.62	21.30	4.56
38005	195.2	16.70	20.36	3.66	21.30	4.60

Time	Suppression Pool Temperature 120% OLTP	Required Containment Pressure 120% OLTP	Licensed Containment Overpressure CLTP	Margin between Licensed COP and Required Containment Pressure	Minimum Wetwell Pressure Available 120 % OLTP	Margin between Available Wetwell Pressure and Required Containment Pressure
Seconds	°F	psia	psia	(Licensed COP – Required Containment Pressure, psia)	psia	(Available Wetwell Pressure – Required Containment Pressure, psia)
40016	195.0	16.65	20.36	3.71	21.29	4.64
42006	194.7	16.59	20.36	3.77	21.25	4.66
44010	194.4	16.52	20.36	3.84	21.19	4.67
46003	194.1	16.46	20.36	3.90	21.11	4.65
48022	193.7	16.38	20.36	3.98	21.03	4.65
50014	193.2	16.27	20.36	4.09	20.94	4.67
52006	192.8	16.19	20.36	4.17	20.84	4.65
54011	192.4	16.11	20.36	4.25	20.74	4.63
56001	192.0	16.03	19.86	3.83	20.64	4.61
58004	191.5	15.93	19.86	3.93	20.55	4.62
60006	191.0	15.82	19.86	4.04	20.45	4.63
80015	185.7	14.82	19.26	4.44	19.99	5.17
100004	180.4	13.90	18.46	4.56	19.50	5.60
150009	169.8	12.33	18.46	6.13	18.59	6.26
200008	162.3	11.40	16.56	5.16	18.01	6.61
250015	156.9	10.82	15.26	4.44	17.58	6.76
300002	152.6	10.39	15.26	4.87	17.23	6.84

Table 2.6-4 Appendix R Fire (Case #2)

Time	Suppression Pool Temperature 120% OLTP	Required Containment Pressure 120% OLTP	Licensed Containment Overpressure CLTP	Margin between Licensed COP and Required Containment Pressure	Minimum Wetwell Pressure Available 120 % OLTP	Margin between Available Wetwell Pressure and Required Containment Pressure
Seconds	°F	psia	psia	(Licensed COP – Required Containment Pressure, psia)	psia	(Available Wetwell Pressure – Required Containment Pressure, psia)
2011	136.8	9.08	18.26	9.18	14.65	5.57
4006	157.9	10.90	18.26	7.36	15.38	4.48
6019	168.7	12.18	18.26	6.08	16.22	4.04
8000	174.4	12.97	18.26	5.29	17.05	4.08
10006	178.7	13.63	19.56	5.93	17.80	4.17
12012	182.1	14.19	19.56	5.37	18.42	4.23
14006	184.9	14.67	19.56	4.89	18.93	4.26
16004	187.1	15.07	20.36	5.29	19.37	4.30
18008	188.9	15.41	20.36	4.95	19.72	4.31
20009	190.3	15.68	20.36	4.68	20.04	4.36
22009	191.5	15.93	20.36	4.43	20.26	4.33
24007	192.4	16.11	20.36	4.25	20.46	4.35
26013	193.2	16.27	20.36	4.09	20.65	4.38
28011	193.8	16.40	20.36	3.96	20.79	4.39
30005	194.2	16.48	20.36	3.88	20.94	4.46
32012	194.5	16.55	20.36	3.81	21.02	4.47
34014	194.6	16.57	20.36	3.79	21.09	4.52

Time	Suppression Pool Temperature 120% OLTP	Required Containment Pressure 120% OLTP	Licensed Containment Overpressure CLTP	Margin between Licensed COP and Required Containment Pressure	Minimum Wetwell Pressure Available 120 % OLTP	Margin between Available Wetwell Pressure and Required Containment Pressure
Seconds	°F	psia	psia	(Licensed COP – Required Containment Pressure, psia)	psia	(Available Wetwell Pressure – Required Containment Pressure, psia)
36014	194.7	16.59	20.36	3.77	21.15	4.56
38009	194.6	16.57	20.36	3.79	21.18	4.61
40004	194.4	16.52	20.36	3.84	21.18	4.66
42007	194.1	16.46	20.36	3.90	21.14	4.68
44010	193.9	16.42	20.36	3.94	21.08	4.66
46002	193.5	16.33	20.36	4.03	21.03	4.70
48008	193.2	16.27	20.36	4.09	20.94	4.67
50011	192.8	16.19	20.36	4.17	20.88	4.69
52007	192.4	16.11	20.36	4.25	20.80	4.69
54013	192.0	16.03	20.36	4.33	20.72	4.69
56015	191.6	15.95	19.86	3.91	20.64	4.69
58014	191.2	15.87	19.86	3.99	20.56	4.69
60005	190.7	15.77	19.86	4.09	20.47	4.70
80006	185.6	14.80	19.26	4.46	19.96	5.16
100009	180.5	13.92	18.46	4.54	19.50	5.58
150009	169.8	12.33	18.46	6.13	18.64	6.31
200005	162.4	11.41	16.56	5.15	18.09	6.68
250012	157.0	10.82	15.26	4.44	17.69	6.87
300000	152.7	10.40	15.26	4.86	17.36	6.96

Table 2.6-5 Station Blackout (SBO)

Peak Suppression Pool *Temperature during HPCI pump* operation 120% OLTP	HPCI NPSHA	*HPCI* NPSHR	*Required Containment Pressure* 120% OLTP
			NPSHR + System Resistance
°F	ft	ft	psia
157.4	28.4*	17	9.42

* Based on use of atmospheric pressure at 3 hours into event

Table 2.6-6 Anticipated Transient Without Scram (ATWS) – Pressure Regulator Failed – Open (PRFO) Case #1

Time	Suppression Pool Temperature 120% OLTP	Required Containment Pressure 120% OLTP	Minimum Wetwell Pressure Available 120 % OLTP	Margin between Available Wetwell Pressure and Required Containment Pressure
Seconds	°F	psia	psia	(Available Wetwell Pressure – Required Containment Pressure, psia)
600	171.50	13.60	16.92	3.32
735	172.80	13.78	17.23	3.45
815	172.60	13.75	17.21	3.46
903	172.30	13.70	17.20	3.50
1006	171.60	13.61	17.22	3.61
1207	177.10	14.38	17.38	3.00
1406	183.10	15.33	17.60	2.27
1620	185.30	15.70	17.79	2.09
1808	185.10	15.66	17.91	2.25
2021	183.80	15.43	17.98	2.55
2202	182.40	15.19	18.07	2.88
2415	180.70	14.91	18.18	3.27
2620	180.80	14.92	18.29	3.37
2826	182.30	15.16	18.42	3.26
3017	183.70	15.39	18.56	3.17
3202	184.90	15.59	18.68	3.09
3420	186.20	15.81	18.84	3.03
3617	187.30	16.01	18.97	2.96
3820	188.30	16.18	19.12	2.94
4034	187.80	16.08	19.26	3.18
4226	186.50	15.84	19.33	3.49
4410	185.10	15.59	19.42	3.83

Time	Suppression Pool Temperature 120% OLTP	Required Containment Pressure 120% OLTP	Minimum Wetwell Pressure Available 120 % OLTP	Margin between Available Wetwell Pressure and Required Containment Pressure
Seconds	°F	psia	psia	(Available Wetwell Pressure – Required Containment Pressure, psia)
4608	183.50	15.31	19.52	4.21
4801	183.40	15.28	19.62	4.34
5024	184.40	15.45	19.75	4.30
5236	185.30	15.60	19.88	4.28
5412	186.10	15.74	19.98	4.24
5614	186.80	15.86	20.09	4.23
5814	187.50	15.98	20.21	4.23
6020	188.20	16.10	20.32	4.22
6224	188.80	16.20	20.43	4.23
6444	188.60	16.16	20.54	4.38
6637	187.30	15.92	20.56	4.64
6831	185.80	15.64	20.60	4.96
7035	185.30	15.55	20.66	5.11
7216	185.80	15.63	20.72	5.09
7457	186.50	15.75	20.80	5.05
7601	186.90	15.81	20.84	5.03
7846	187.40	15.90	20.92	5.02
8019	187.80	15.97	20.98	5.01
8227	188.20	16.03	21.03	5.00
8413	188.60	16.10	21.09	4.99
8620	188.90	16.15	21.15	5.00
8809	188.10	16.06	20.94	4.88
9019	186.50	15.80	20.88	5.08

Time	Suppression Pool Temperature 120% OLTP	Required Containment Pressure 120% OLTP	Minimum Wetwell Pressure Available *120 % OLTP*	Margin between Available Wetwell Pressure and Required Containment Pressure
Seconds	°F	psia	psia	(Available Wetwell Pressure – Required Containment Pressure, psia)
9218	185.80	15.67	20.94	5.27
9409	185.40	15.59	20.99	5.40
9608	185.00	15.52	21.03	5.51
9801	184.60	15.44	21.06	5.62
10007	184.20	15.37	21.07	5.70
11018	182.10	14.99	21.04	6.05
11063	182.00	14.97	21.04	6.07
12034	179.90	14.60	20.87	6.27
13032	177.90	14.27	20.63	6.36
14021	176.10	13.97	20.38	6.41
15006	170.70	13.30	19.63	6.33
16020	168.70	13.02	19.50	6.48
17013	167.40	12.83	19.39	6.56
18011	166.00	12.64	19.30	6.66
19033	164.70	12.46	19.23	6.77
20026	163.50	12.30	19.18	6.88
21012	162.30	12.15	19.13	6.98
22030	157.60	11.74	18.89	7.15
23018	155.70	11.53	18.79	7.26
24001	155.00	11.45	18.73	7.28

Table 2.6-7 Anticipated Transient Without Scram (ATWS) – Pressure Regulator Failed – Open (PRFO) Case #2

Time	Suppression Pool Temperature 120% OLTP	Required Containment Pressure 120% OLTP	Minimum Wetwell Pressure Available 120 % OLTP	Margin between Available Wetwell Pressure and Required Containment Pressure
Seconds	°F	psia	psia	(Available Wetwell Pressure – Required Containment Pressure, psia)
500	167.20	13.05	16.55	3.50
633	172.70	13.88	16.43	2.55
739	173.30	14.06	16.88	2.82
839	173.40	14.16	17.14	2.98
910	173.30	14.21	17.27	3.06
1018	172.90	14.26	17.37	3.11
1203	179.60	15.35	17.84	2.49
1402	186.20	16.45	18.46	2.01
1603	188.10	16.79	18.90	2.11
1801	187.80	16.73	19.29	2.56
2003	186.20	16.43	19.58	3.15
2202	184.30	16.09	19.77	3.68
2402	182.50	15.78	19.72	3.94
2601	182.50	15.77	19.74	3.97
2803	184.30	16.07	19.99	3.92
3004	185.90	16.34	20.24	3.90
3205	187.30	16.59	20.50	3.91
3402	188.50	16.81	20.76	3.95,

Time	Suppression Pool Temperature 120% OLTP	Required Containment Pressure 120% OLTP	Minimum Wetwell Pressure Available 120 % OLTP	Margin between Available Wetwell Pressure and Required Containment Pressure
Seconds	°F	psia	psia	(Available Wetwell Pressure – Required Containment Pressure, psia)
3604	189.60	17.01	21.02	4.01
3800	190.60	17.19	21.25	4.06
4003	190.00	17.07	21.38	4.31
4204	188.50	16.78	21.37	4.59
4405	186.70	16.44	21.28	4.84
4604	185.00	16.13	21.20	5.07
4805	185.20	16.16	21.24	5.08
5003	186.40	16.37	21.41	5.04
5203	187.40	16.55	21.60	5.05
5406	188.30	16.71	21.79	5.08
5603	189.00	16.83	21.96	5.13
5806	189.80	16.98	22.12	5.14
6002	190.40	17.09	22.27	5.18
6201	191.00	17.20	22.41	5.21
6403	190.80	17.15	22.51	5.36
6604	189.20	16.84	22.48	5.64
6804	187.50	16.52	22.34	5.82
7001	186.80	16.38	22.25	5.87
7205	187.60	16.52	22.33	5.81

Time	Suppression Pool Temperature 120% OLTP	Required Containment Pressure 120% OLTP	Minimum Wetwell Pressure Available 120 % OLTP	Margin between Available Wetwell Pressure and Required Containment Pressure
Seconds	°F	psia	psia	(Available Wetwell Pressure – Required Containment Pressure, psia)
7400	188.20	16.63	22.42	5.79
7600	188.80	16.74	22.52	5.78
7804	189.30	16.82	22.61	5.79
8001	189.80	16.91	22.69	5.78
8206	190.20	16.99	22.77	5.78
8401	190.50	17.04	22.84	5.80
8602	190.80	17.09	22.91	5.82
8800	190.10	17.00	22.72	5.72
9006	188.30	16.70	22.58	5.88
9201	187.40	16.53	22.56	6.03
9404	187.00	16.45	22.54	6.09
9604	186.60	16.37	22.52	6.15
9807	186.20	16.29	22.49	6.20
10007	185.80	16.22	22.46	6.24
11002	183.60	15.81	22.28	6.47
12003	181.40	15.41	22.08	6.67
13003	179.30	15.05	21.85	6.80
14003	177.30	14.72	21.61	6.89
15002	171.40	13.96	20.82	6.86

Time	Suppression Pool Temperature 120% OLTP	Required Containment Pressure 120% OLTP	Minimum Wetwell Pressure Available 120 % OLTP	Margin between Available Wetwell Pressure and Required Containment Pressure
Seconds	°F	psia	psia	(Available Wetwell Pressure – Required Containment Pressure, psia)
15007	171.40	13.96	20.82	6.86
15013	171.40	13.96	20.81	6.85
15018	171.30	13.94	20.81	6.87
15025	171.30	13.94	20.81	6.87
15032	171.20	13.93	20.80	6.87
16004	169.70	13.71	20.40	6.69
17003	168.20	13.50	20.11	6.61
18003	166.80	13.30	19.88	6.58
19002	165.40	13.11	19.69	6.58
20002	164.10	12.94	19.53	6.59
21001	162.90	12.78	19.39	6.61
22003	158.20	12.36	18.80	6.44
23002	156.20	12.15	18.51	6.36
24000	155.50	12.06	18.35	6.29

Table 2.6-8 Anticipated Transient Without Scram (ATWS) Loss of Offsite Power (LOOP)

Time	Suppression Pool Temperature 120% OLTP	Required Containment Pressure 120% OLTP	Available Wetwell Pressure 120 % OLTP	Margin between Available Wetwell Pressure and Required Containment Pressure
Seconds	°F	psia	psia	(Available Wetwell Pressure – Required Containment Pressure, psia)
600	156.4	11.86	16.36	4.50
691	159.4	12.16	16.57	4.41
791	162.3	12.47	16.80	4.33
891	165.1	12.79	17.01	4.22
900	165.3	12.82	17.03	4.21
1000	168.1	13.20	15.68	2.48
1101	171.0	13.62	15.99	2.37
1241	174.0	14.09	16.48	2.39
1414	174.3	14.21	16.82	2.61
1622	173.7	14.22	16.98	2.76
1808	175.1	14.51	17.14	2.63
2008	178.5	15.11	17.46	2.35
2205	181.5	15.64	17.79	2.15
2402	183.9	16.04	18.15	2.11
2601	186.0	16.41	18.56	2.15
2802	187.3	16.64	18.98	2.34
3007	186.5	16.49	19.29	2.80
3205	185.0	16.21	19.40	3.19
3401	183.5	15.95	19.43	3.48

Time	Suppression Pool Temperature 120% OLTP	Required Containment Pressure 120% OLTP	Available Wetwell Pressure 120 % OLTP	Margin between Available Wetwell Pressure and Required Containment Pressure
Seconds	°F	psia	psia	(Available Wetwell Pressure – Required Containment Pressure, psia)
3607	182.0	15.69	19.43	3.74
3806	183.4	15.92	19.55	3.63
4003	185.1	16.21	19.75	3.54
4200	186.7	16.49	19.93	3.44
4406	188.3	16.78	20.12	3.34
4604	189.7	17.04	20.32	3.28
4803	191.0	17.29	20.53	3.24
5005	192.2	17.52	20.75	3.23
5013	192.3	17.54	20.76	3.22
5204	192.5	17.57	20.93	3.36
5406	191.3	17.33	21.01	3.68
5603	189.8	17.03	21.03	4.00
5807	188.3	16.74	21.00	4.26
6005	187.9	16.66	21.00	4.34
6205	189.2	16.90	21.14	4.24
6405	190.4	17.13	21.31	4.18
6604	191.5	17.34	21.48	4.14
6804	192.5	17.53	21.63	4.10
7003	193.4	17.71	21.78	4.07
7205	194.3	17.89	21.94	4.05
7402	195.1	18.06	22.09	4.03

Time	Suppression Pool Temperature 120% OLTP	Required Containment Pressure 120% OLTP	Available Wetwell Pressure 120 % OLTP	Margin between Available Wetwell Pressure and Required Containment Pressure
Seconds	°F	psia	psia	(Available Wetwell Pressure – Required Containment Pressure, psia)
7604	195.8	18.20	22.24	4.04
7801	195.3	18.09	22.34	4.25
8005	195.4	18.10	22.41	4.31
8205	196.1	18.25	22.53	4.28
8407	196.8	18.39	22.66	4.27
8606	195.9	18.26	22.49	4.23
8805	194.4	17.97	22.48	4.51
9007	194.0	17.88	22.50	4.62
9202	194.8	18.05	22.58	4.53
9406	195.4	18.17	22.67	4.50
9604	195.9	18.27	22.75	4.48
9804	196.2	18.33	22.82	4.49
10007	196.1	18.30	22.87	4.57
11003	194.6	17.96	22.95	4.99
12003	193.1	17.62	22.88	5.26
13008	191.4	17.28	22.70	5.42
14003	186.2	16.36	22.12	5.76
15004	185.1	16.14	21.85	5.71
16008	183.9	15.91	21.62	5.71
17007	182.7	15.69	21.42	5.73
18000	181.6	15.49	21.23	5.74

Time	Suppression Pool Temperature 120% OLTP	Required Containment Pressure 120% OLTP	Available Wetwell Pressure 120 % OLTP	Margin between Available Wetwell Pressure and Required Containment Pressure
Seconds	°F	psia	psia	(Available Wetwell Pressure – Required Containment Pressure, psia)
19008	180.4	15.27	21.05	5.78
20001	177.4	14.89	20.46	5.57
21007	173.8	14.34	20.05	5.71
22000	173.1	14.23	19.83	5.60
23003	172.3	14.10	19.70	5.60
24007	171.5	13.97	19.60	5.63
25004	170.7	13.84	19.50	5.66
26007	169.9	13.72	19.41	5.69
27006	169.2	13.62	19.33	5.71
28007	164.9	13.17	19.01	5.84
29002	163.3	12.97	18.76	5.79
30000	163.0	12.92	18.68	5.76
31005	162.6	12.86	18.61	5.75
32004	162.2	12.80	18.56	5.76
33002	161.7	12.73	18.52	5.79
34006	161.3	12.67	18.49	5.82
35008	160.9	12.61	18.47	5.86
36000	158.2	12.42	18.25	5.83

Table 2.6-9 Small Steamline Break Accident (SBA)

Time	Suppression Pool Temperature 120% OLTP	Required Containment Pressure 120% OLTP	Available Wetwell Pressure 120 % OLTP	Margin between Available Wetwell Pressure and Required Containment Pressure
Seconds	°F	psia	psia	(Available Wetwell Pressure – Required Containment Pressure - psia)
600	108.0	9.58	24.10	14.52
700	110.0	7.61	23.90	16.29
801	110.0	7.61	24.70	17.09
901	111.0	7.64	25.50	17.86
1001	112.0	7.78	19.30	11.52
1210	114.0	7.94	15.60	7.66
1400	114.0	8.03	15.60	7.57
1608	116.0	8.19	15.50	7.31
1807	117.0	8.30	15.30	7.00
2004	117.0	8.31	15.40	7.09
2201	120.0	8.43	15.50	7.07
2404	122.0	8.52	15.60	7.08
2603	124.0	8.60	15.70	7.10
2805	127.0	8.75	15.80	7.05
3001	130.0	8.91	16.00	7.09
3005	130.0	8.91	16.00	7.09
3202	133.0	9.08	16.10	7.02

Time	Suppression Pool Temperature 120% OLTP	Required Containment Pressure 120% OLTP	Available Wetwell Pressure 120 % OLTP	Margin between Available Wetwell Pressure and Required Containment Pressure
Seconds	°F	psia	psia	(Available Wetwell Pressure – Required Containment Pressure - psia)
3402	136.0	9.27	16.30	7.03
3601	138.0	9.39	16.40	7.01
3804	141.0	9.60	16.60	7.00
4006	144.0	9.82	16.80	6.98
4204	147.0	10.05	17.00	6.95
4406	151.0	10.39	17.20	6.81
4607	154.0	10.67	17.30	6.63
4801	156.0	10.86	17.50	6.64
5000	158.0	11.05	17.60	6.55
5200	161.0	11.37	17.80	6.43
5404	164.0	11.71	17.90	6.19
5601	165.0	11.82	18.10	6.28
5803	167.0	12.06	18.20	6.14
6002	170.0	12.44	18.40	5.96
6203	171.0	12.57	18.50	5.93
6404	173.0	12.84	18.70	5.86
6605	175.0	13.12	18.80	5.68
6804	177.0	13.42	19.00	5.58
7002	178.0	13.62	18.90	5.28

Time	Suppression Pool Temperature 120% OLTP	Required Containment Pressure 120% OLTP	Available Wetwell Pressure 120 % OLTP	Margin between Available Wetwell Pressure and Required Containment Pressure
Seconds	°F	psia	psia	(Available Wetwell Pressure – Required Containment Pressure - psia)
7201	177.0	13.55	18.50	4.95
7403	177.0	13.65	18.30	4.65
7602	176.0	13.50	18.30	4.80
7800	176.0	13.50	18.40	4.90
8005	176.0	13.50	18.50	5.00
8200	175.0	13.35	18.60	5.25
8403	175.0	13.35	18.60	5.25
8603	175.0	13.34	18.70	5.36
8804	177.0	13.64	18.90	5.26
9004	179.0	13.95	19.00	5.05
9202	181.0	14.26	19.20	4.94
9402	183.0	14.60	19.40	4.80
9602	184.0	14.76	19.50	4.74
9804	187.0	15.30	19.70	4.40
10000	188.0	15.47	19.90	4.43
11000	194.0	16.69	20.60	3.91
12005	196.0	17.15	21.00	3.85
13002	197.0	17.36	21.40	4.04
14004	199.0	17.79	21.70	3.91

Time	Suppression Pool Temperature 120% OLTP	Required Containment Pressure 120% OLTP	Available Wetwell Pressure 120 % OLTP	Margin between Available Wetwell Pressure and Required Containment Pressure
Seconds	°F	psia	psia	(Available Wetwell Pressure – Required Containment Pressure - psia)
15003	201.0	18.24	22.10	3.86
16001	202.0	18.46	22.40	3.94
17006	200.0	18.08	22.10	4.02
18004	202.0	18.54	22.30	3.76
19001	203.0	18.76	22.50	3.74
20005	204.0	19.00	22.70	3.70
21005	205.0	19.23	22.80	3.57
22001	206.0	19.48	23.00	3.52
23004	205.0	19.30	22.70	3.40
24005	204.0	19.04	22.80	3.76
25002	205.0	19.27	22.90	3.63
26004	205.0	19.26	23.00	3.74
27003	206.0	19.50	23.10	3.60
28006	206.0	19.49	23.20	3.71
29003	207.0	19.73	23.30	3.57
30001	206.0	19.55	23.00	3.45
31000	205.0	19.29	22.80	3.51
32004	205.0	19.27	22.90	3.63
33004	205.0	19.26	23.00	3.74

Time	Suppression Pool Temperature 120% OLTP	Required Containment Pressure 120% OLTP	Available Wetwell Pressure 120 % OLTP	Margin between Available Wetwell Pressure and Required Containment Pressure
Seconds	°F	psia	psia	(Available Wetwell Pressure – Required Containment Pressure - psia)
34004	206.0	19.51	23.10	3.59
35004	206.0	19.49	23.20	3.71
36001	206.0	19.48	23.30	3.82
37003	206.0	19.46	23.30	3.84
38006	203.0	18.79	22.70	3.91
39005	204.0	19.03	22.60	3.57
40002	204.0	19.01	22.60	3.59
41002	204.0	19.00	22.60	3.60
42005	204.0	18.98	22.70	3.72
43005	204.0	18.97	22.70	3.73
44005	204.0	18.96	22.80	3.84
45001	204.0	18.95	22.80	3.85
46001	201.0	18.30	22.20	3.90
47004	201.0	18.28	22.00	3.72
48002	201.0	18.27	22.00	3.73
49001	201.0	18.26	22.10	3.84
50004	201.0	18.25	22.10	3.85
51004	201.0	18.24	22.10	3.86
52000	201.0	18.22	22.20	3.98

Time	Suppression Pool Temperature 120% OLTP	Required Containment Pressure 120% OLTP	Available Wetwell Pressure 120 % OLTP	Margin between Available Wetwell Pressure and Required Containment Pressure
Seconds	°F	psia	psia	(Available Wetwell Pressure – Required Containment Pressure - psia)
53002	201.0	18.21	22.20	3.99
54004	199.0	17.83	21.70	3.87
55002	198.0	17.59	21.40	3.81
56002	198.0	17.57	21.40	3.83
57005	198.0	17.56	21.40	3.84
58001	198.0	17.55	21.40	3.85
59004	198.0	17.53	21.40	3.87
60002	198.0	17.52	21.40	3.88
61004	198.0	17.51	21.50	3.99
62004	198.0	17.50	21.50	4.00
63001	195.0	16.93	20.90	3.97
64001	195.0	16.92	20.70	3.78
65003	195.0	16.90	20.70	3.80
66004	195.0	16.89	20.70	3.81
67004	195.0	16.88	20.70	3.82
68005	195.0	16.87	20.70	3.83
69003	195.0	16.86	20.80	3.94
70005	195.0	16.84	20.80	3.96
71005	195.0	16.83	20.80	3.97

Time	Suppression Pool Temperature 120% OLTP	Required Containment Pressure 120% OLTP	Available Wetwell Pressure 120 % OLTP	Margin between Available Wetwell Pressure and Required Containment Pressure
Seconds	°F	psia	psia	(Available Wetwell Pressure – Required Containment Pressure - psia)
72001	193.0	16.50	20.30	3.80
73000	191.0	16.08	20.10	4.02
74003	191.0	16.07	20.00	3.93
75004	191.0	16.06	20.00	3.94
76004	191.0	16.05	20.00	3.95
77003	191.0	16.04	20.10	4.06
78001	191.0	16.03	20.10	4.07
79003	191.0	16.01	20.10	4.09
80004	191.0	16.00	20.10	4.10
81001	191.0	15.99	20.20	4.21
82003	188.0	15.51	19.60	4.09
83003	188.0	15.49	19.40	3.91
84000	188.0	15.48	19.40	3.92
85002	188.0	15.47	19.40	3.93
86002	188.0	15.45	19.40	3.95
86400	188.0	15.45	19.50	4.05

Figure 2.6-1A Design Basis Accident (DBA) Loss of Coolant Accident with 3% NPSHR Curves (LOCA)

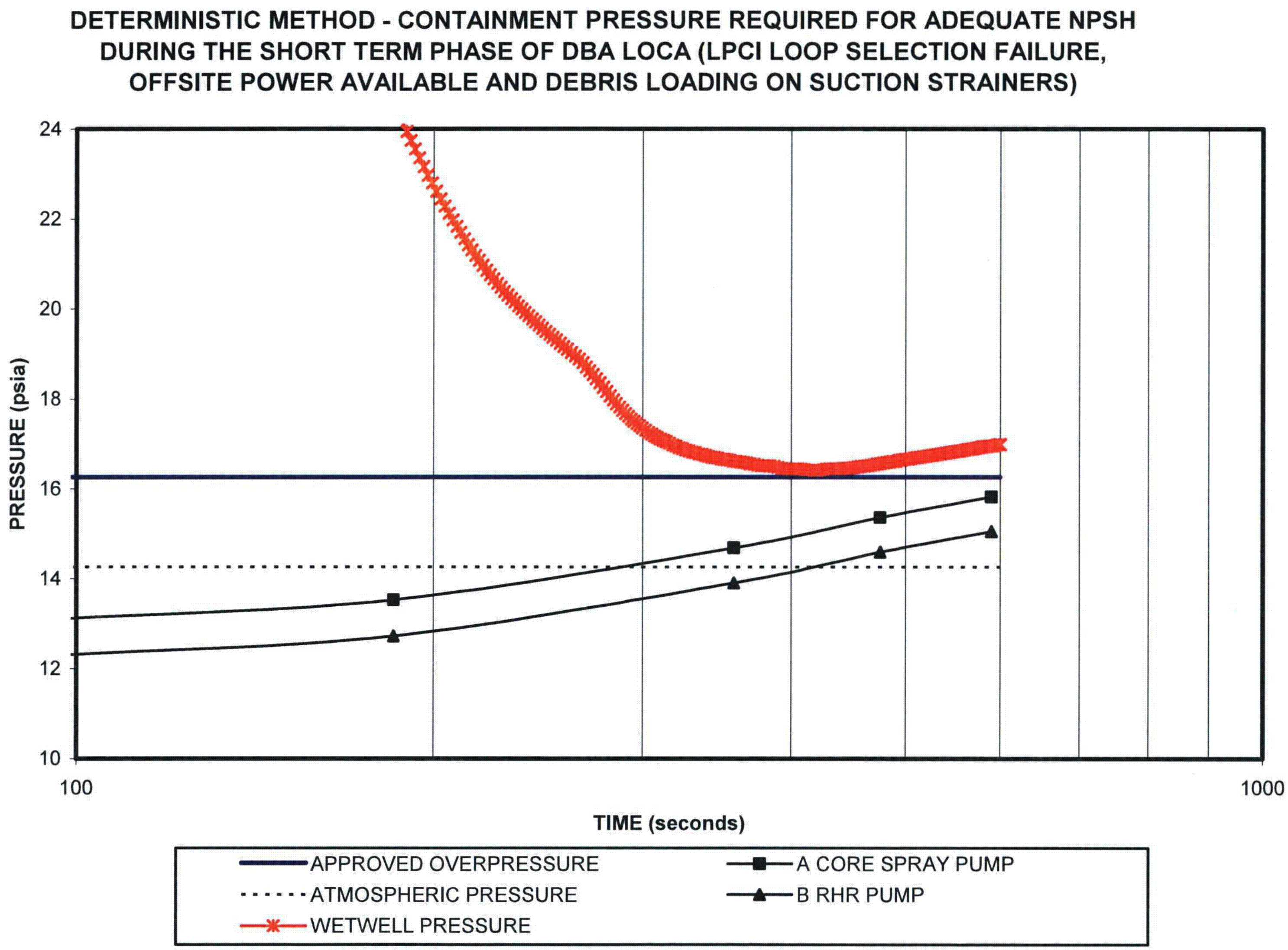


Figure 2.6-1B Design Basis Accident (DBA) Loss of Coolant Accident with 3% NPSHR Curves (LOCA)

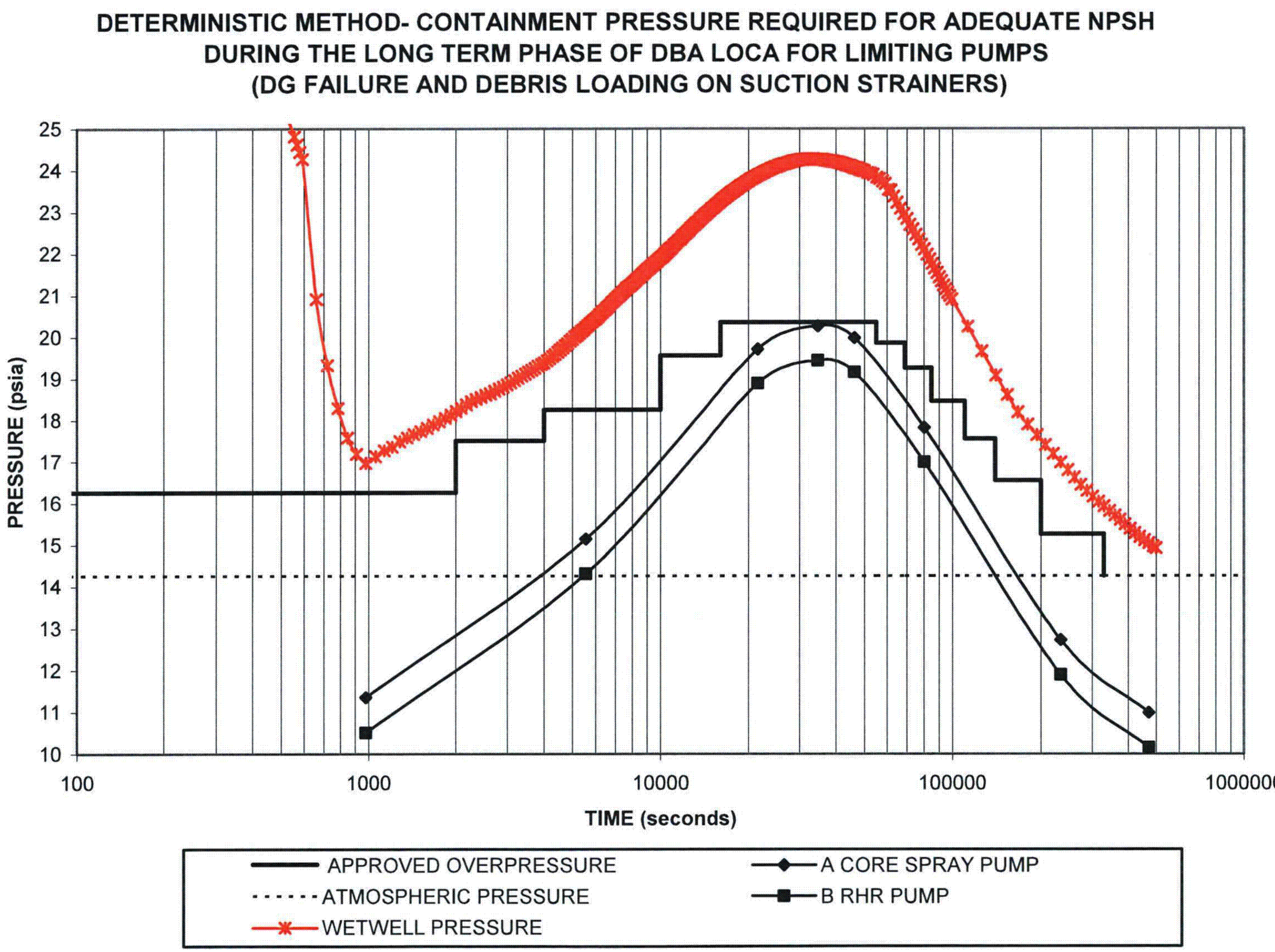


Figure 2.6-1C Design Basis Accident (DBA) Loss of Coolant Accident with 1% NPSHR Curves (LOCA)

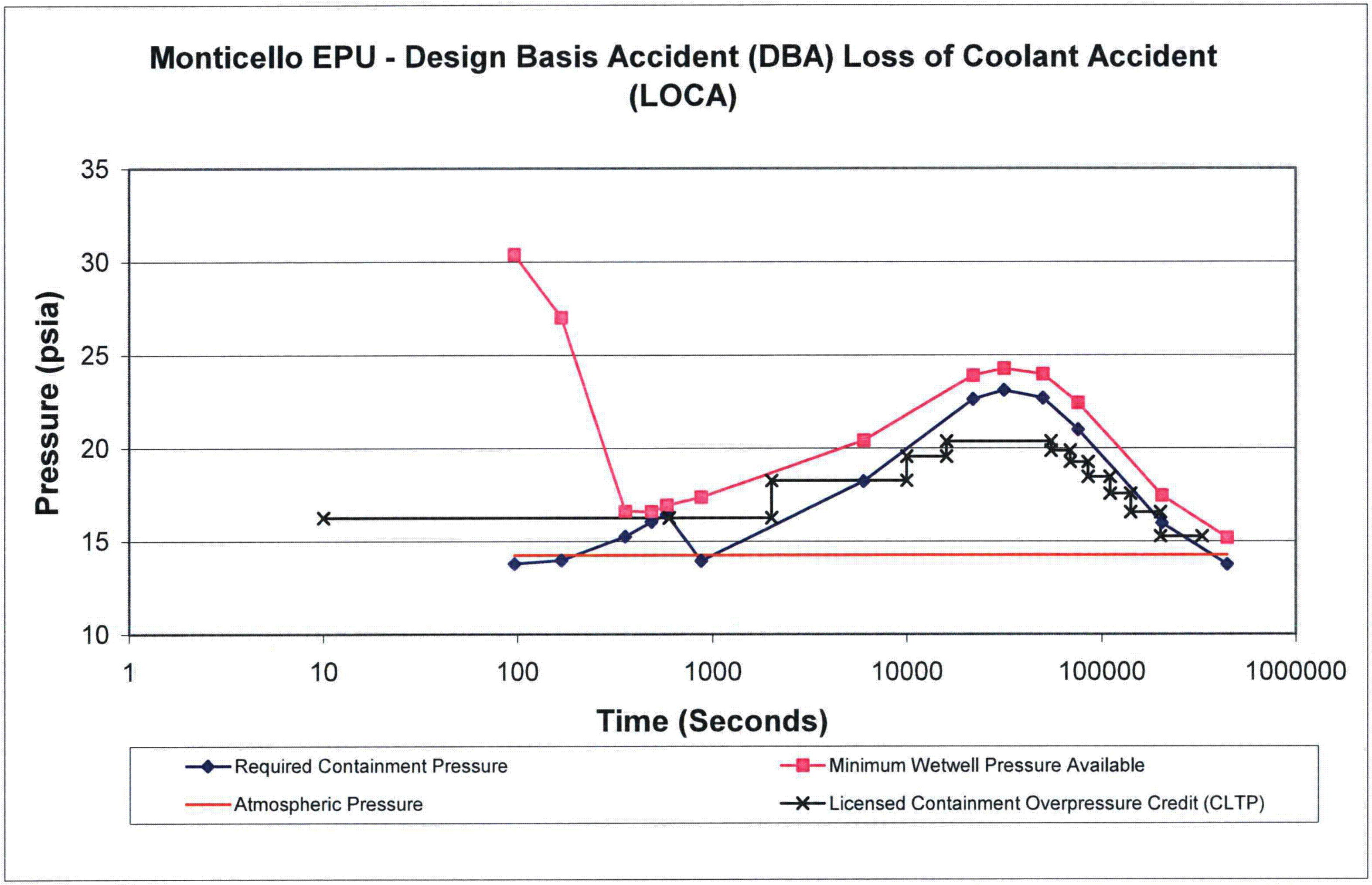


Figure 2.6-2 Appendix R Fire (Case #1)

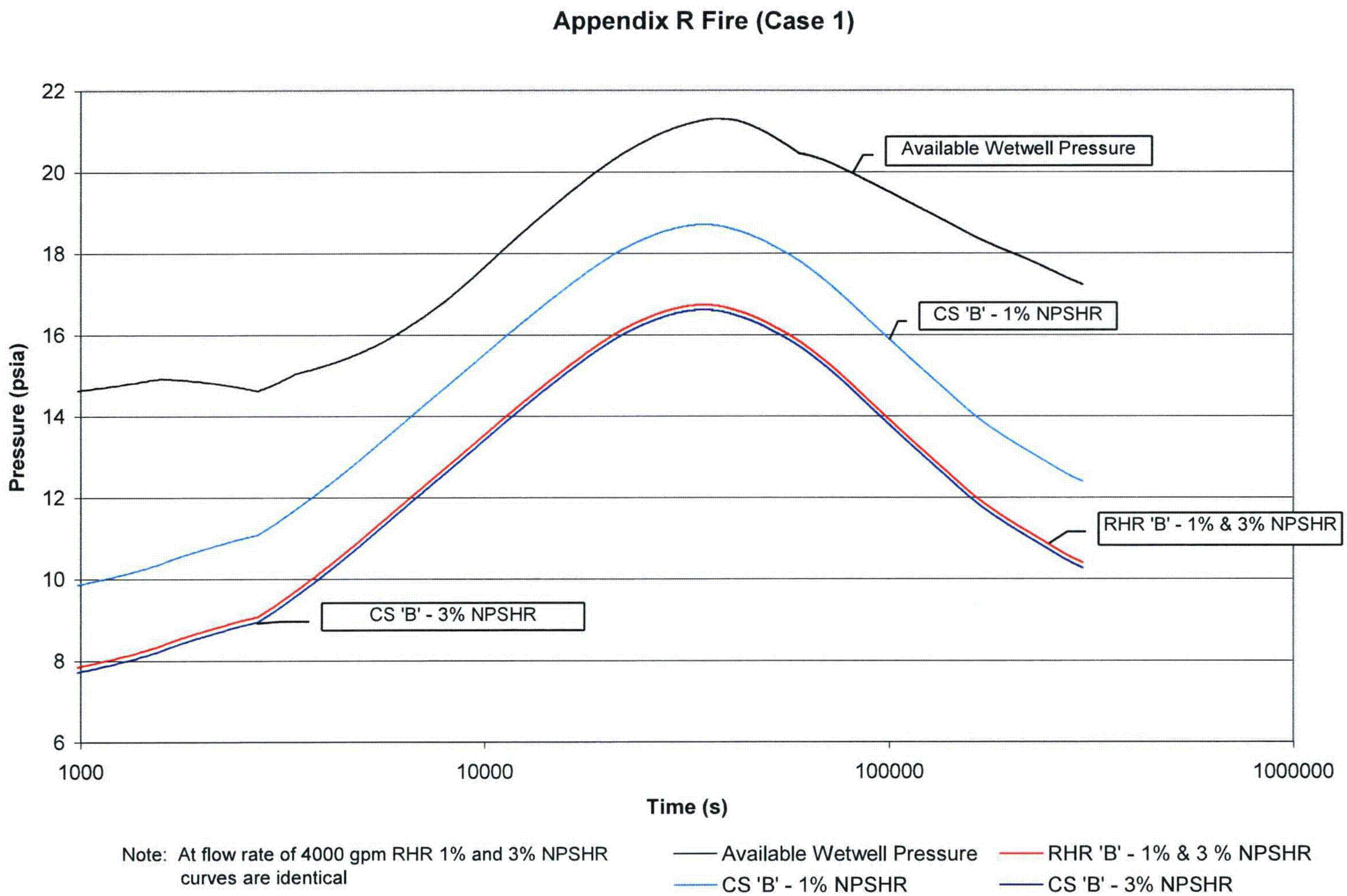


Figure 2.6-3 Appendix R Fire (Case #2)

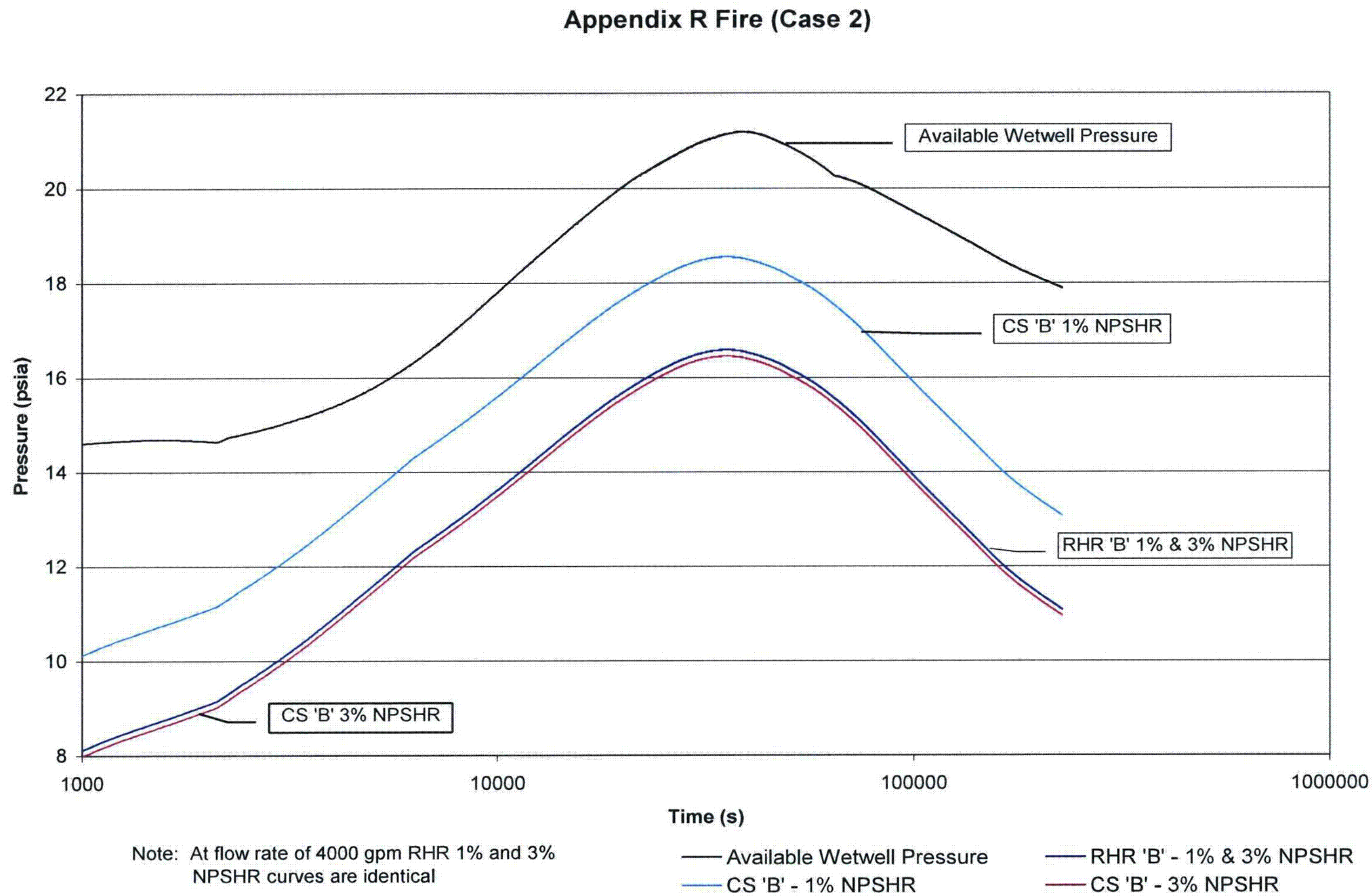


Figure 2.6-4 Anticipated Transient Without Scram (ATWS) – Pressure Regulator Failed – Open (PRFO) Case #1

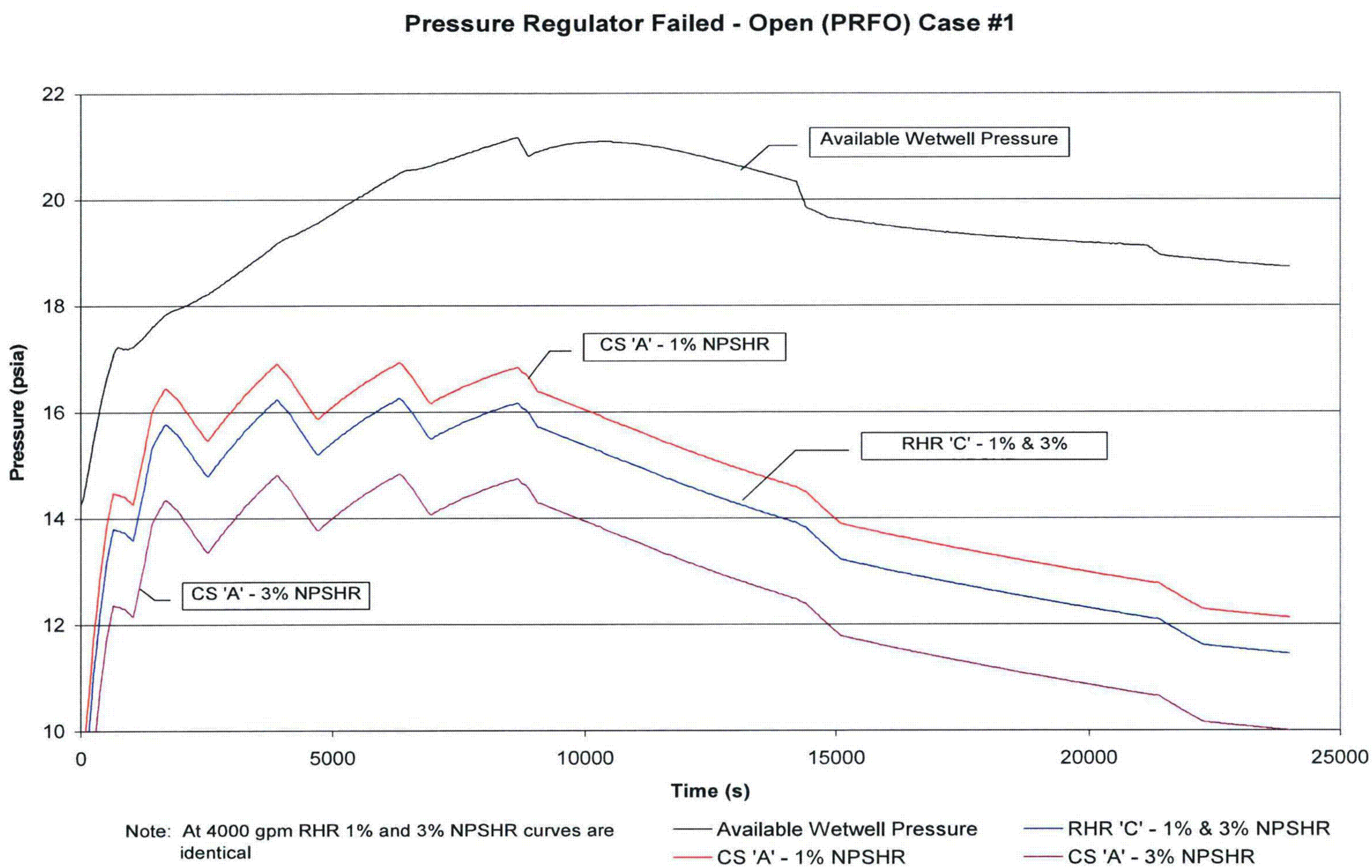


Figure 2.6-5 Anticipated Transient Without Scram (ATWS) – Pressure Regulator Failed – Open (PRFO) Case #2

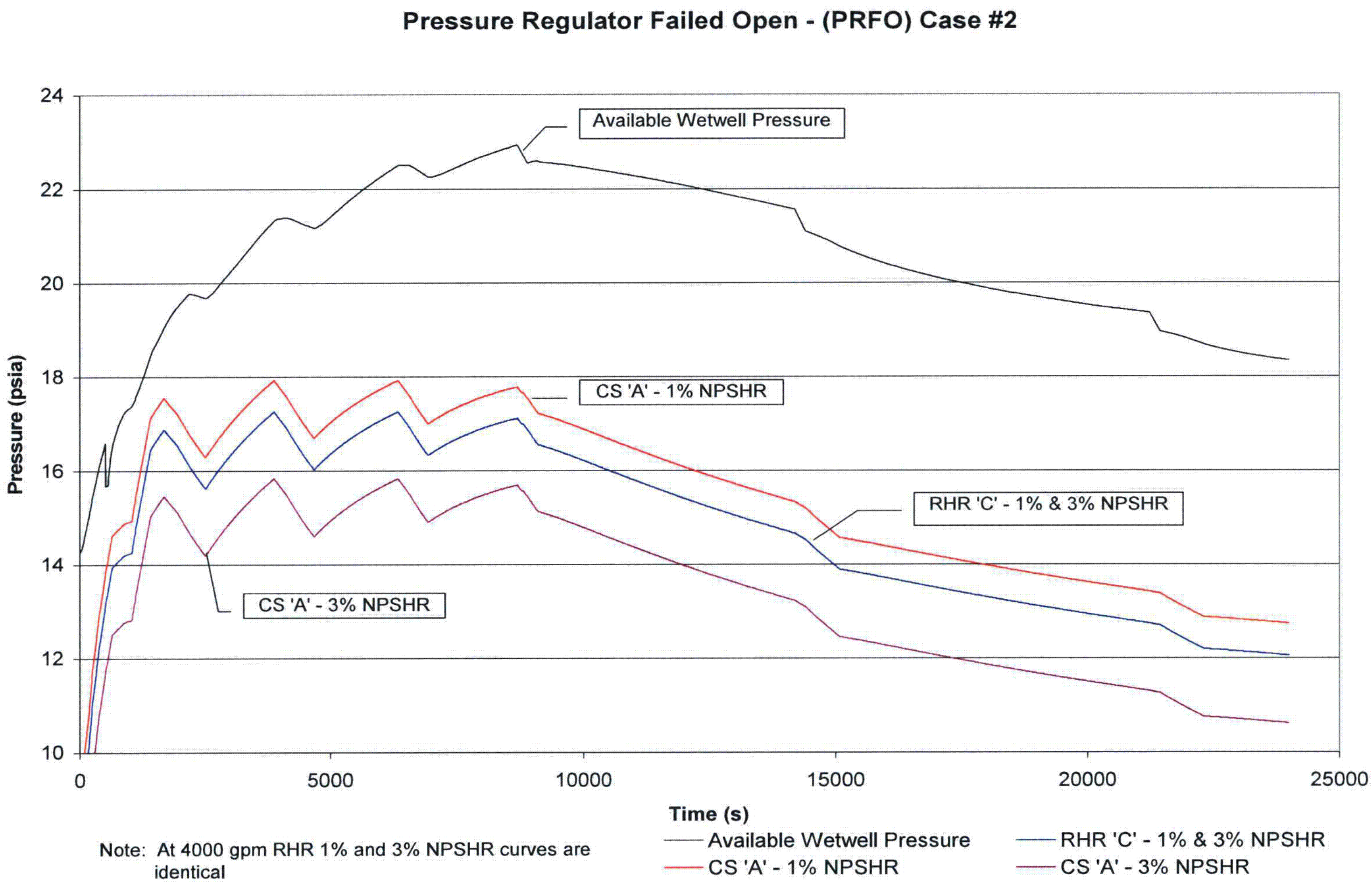


Figure 2.6-6 Anticipated Transient Without Scram (ATWS) Loss of Offsite Power (LOOP)

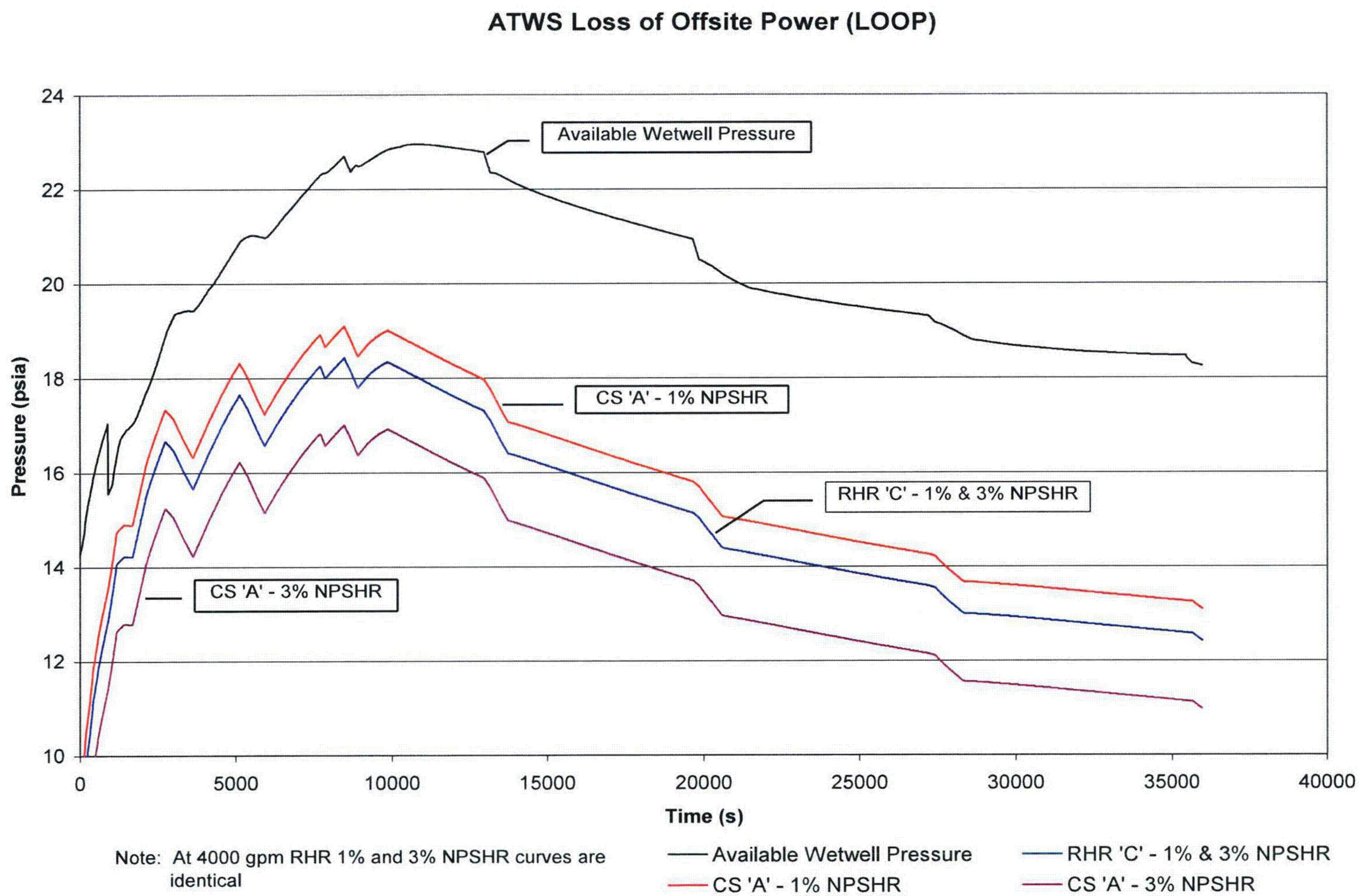


Figure 2.6-7 Small Steamline Break Accident (SBA) - RHR

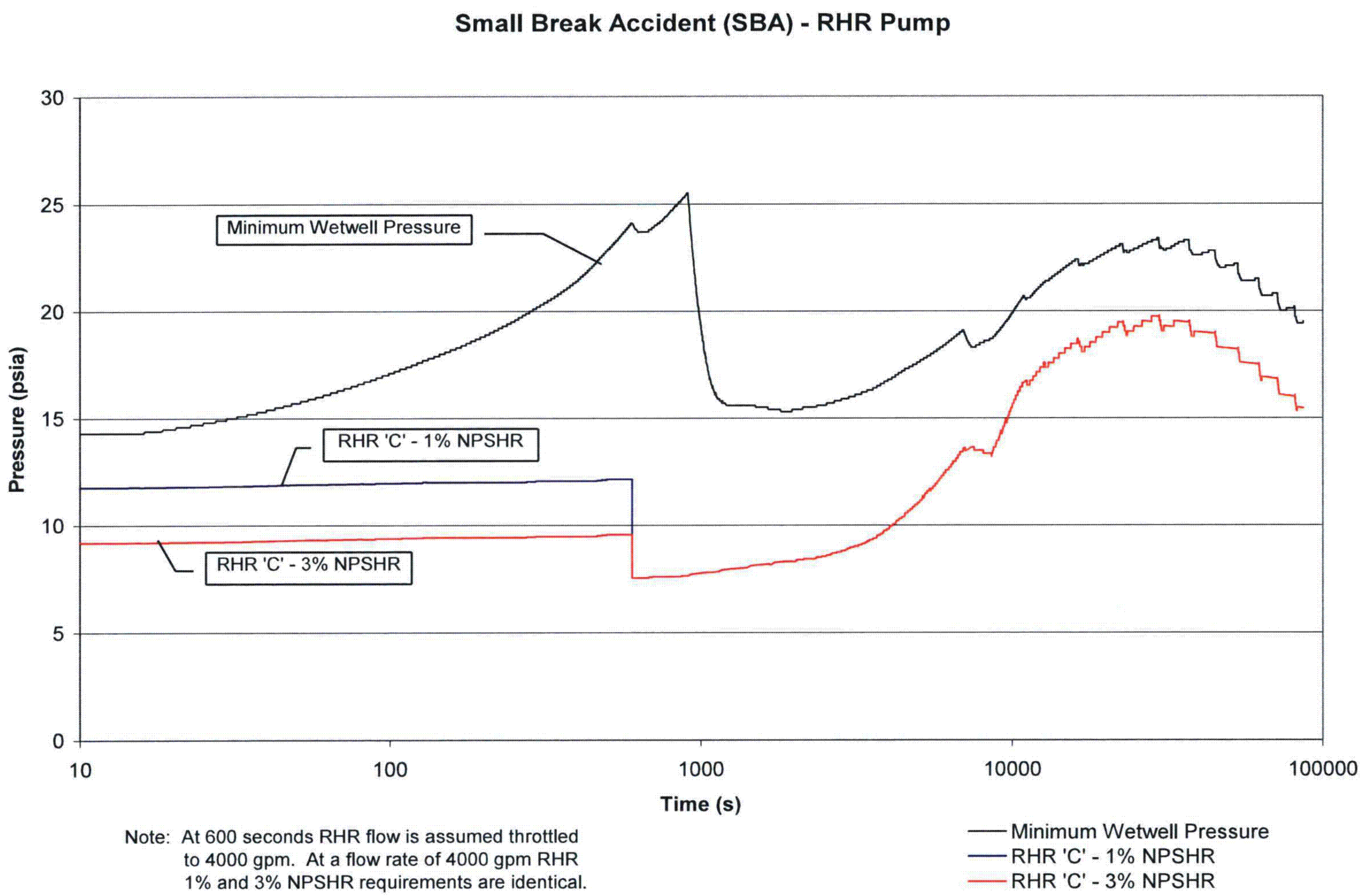
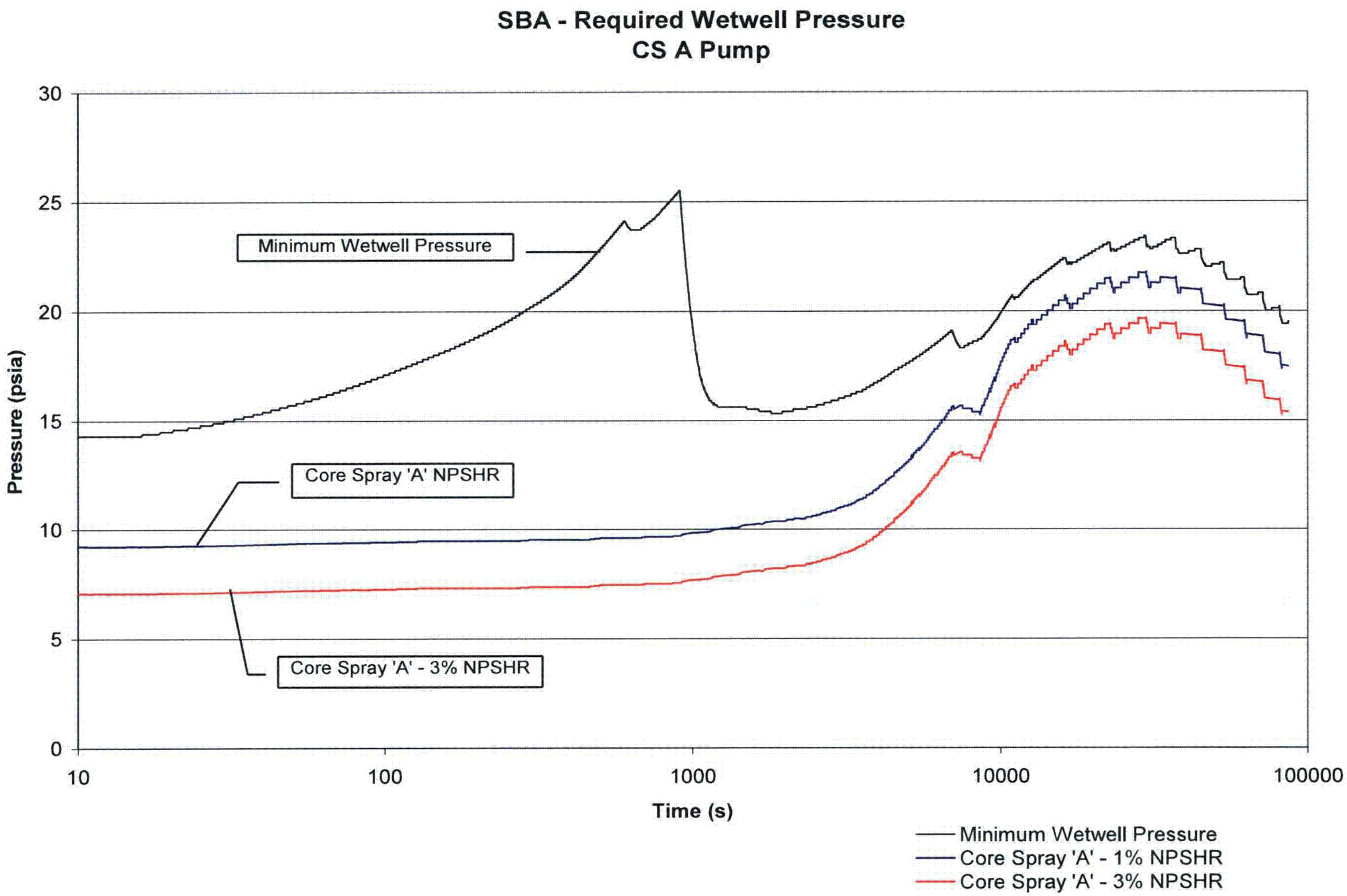


Figure 2.6-8 Small Steamline Break Accident (SBA) - CS



2.7 Habitability, Filtration, and Ventilation

2.7.1 Control Room Habitability System

Regulatory Evaluation

The NRC's acceptance criteria for the control room habitability system are based on (1) GDC-4, 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, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent, to any part of the body, for the duration of the accident.

Specific NRC review criteria are contained in SRP Section 6.4 and other guidance provided in Matrix 7 of RS-001.

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC-4 listed in the Regulatory Evaluation above, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-11, draft GDC-40, and draft GDC-42. Current GDC-19 is applicable to Monticello as described in USAR Sections 5.3.5, 6.7.2, 12.3.1.6, and 14.7.

The Control Room Habitability System is described in Monticello USAR Section 6.7, "Main Control Room, Emergency Filtration Train Building and Technical Support Center Habitability."

In addition to the evaluations described in the Monticello USAR, Monticello's systems and components were evaluated for License Renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Monticello Nuclear Generating Plant License Renewal Safety Evaluation Report (SER), NUREG-1865, dated October 2006 (Reference 5). The license renewal evaluation associated with the Control Room Habitability System is documented in NUREG-1865, Section 2.3.3.7. Management of aging effects on the Control Room Habitability System is documented in NUREG-1865, Section 3.3.2.3.7.

Technical Evaluation

2.7.1.1 Control Room Emergency Filtration (CREF) System

The CREF System provides a radiologically controlled environment from which the unit can be safely operated following a Design Basis Accident (DBA). The system is a standby system, parts of which also operate during normal unit operations to maintain the control room boundary environment. In the high radiation mode of operation, the Emergency Filtration Treatment Train will pressurize the control room area with filtered air to inhibit infiltration of contamination. The high radiation mode of operation is initiated by reactor building ventilation plenum high radiation, refueling floor high radiation, drywell high pressure, or low-low Reactor water level, or outside air high radiation. Upon receipt of an initiation signal, the system automatically switches to the pressurization mode of operation to prevent infiltration of contaminated air into the control room boundary. Outside air is taken in and passed through the charcoal adsorber filter subsystems for removal of airborne radioactive particles. This air is then combined with return air from the control room that is then supplied back into the control room.

The CREF System is credited for functions to isolate the control room from unfiltered air intake, pressurize the control room to minimize unfiltered inleakage, and provide filtration to reduce operator dose consequences during the LOCA Design Basis Accident as described in Section 2.9. No credit for system operation is assumed for the other DBAs (FHA, MSLBA, and CRDA).

Control Room Habitability was reevaluated in 2004 in response to NRC Generic Letter 2003-01. Operation at EPU will not introduce any new toxic chemical sources. Operation at EPU does not alter the basis for analysis of any on-site or off-site toxic chemical releases. Operation at EPU will not create any need to change operator response time for donning breathing apparatus. Therefore, operation at EPU will not affect control room habitability during a toxic gas event.

The EPU effects on the CREF are due to an increase in the radiological source term released during an accident (primarily iodine). The effect of EPU on the post-LOCA iodine loading of the control room charcoal filters was evaluated using the Regulatory Guide 1.3 (TID-14844) iodine source term including release of stable isotopes of iodine. (Note: Monticello has adopted the AST methodology in accordance with 10 CFR 50.67, but has not changed the Monticello licensing basis with respect to equipment qualification). With the increase in iodine loading resulting from EPU, the post-accident iodine loading on the control room charcoal filters remains less than half of the allowable limit of RG 1.52. The effect of radiolytic heating of the charcoal and particulate filters was evaluated and remains within system design specifications. Therefore, the control room iodine filter efficiency is not affected by EPU.

Conclusion

NSPM has evaluated 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 and any increase of toxic and radioactive gases that would result from the proposed EPU. The evaluation indicates that the control room habitability system will continue to provide the required protection following implementation of the proposed EPU. Based on this, the control room habitability system will continue to meet the requirements of the current licensing basis. Therefore, the proposed EPU is acceptable with respect to the control room habitability system.

2.7.2 Engineered Safety Feature Atmosphere Cleanup

Regulatory Evaluation

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., standby gas treatment systems and emergency or post accident air-cleaning systems) for the fuel-handling building, control room, shield building, and areas containing ESF components.

The NRC's acceptance criteria for ESF atmosphere cleanup systems are based on (1) GDC-19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent, to any part of the body, for the duration of the accident; (2) GDC-41, 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, 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, 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 anticipated operational occurrences (AOOs), and postulated accidents.

Specific NRC review criteria are contained in SRP Section 6.5.1.

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed

AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC-61 and GDC-64 listed in the Regulatory Evaluation above, with the exception of current GDC-41, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-11, draft GDC-17, draft GDC-67, draft GDC-68, and draft GDC-69. There is no draft GDC directly associated with current GDC-41. Current GDC-19 is applicable to Monticello as described in USAR Sections 5.3.5, 6.7.2, 12.3.1.6, and 14.7.

The ESF atmosphere cleanup system at Monticello is the Standby Gas Treatment System. The Standby Gas Treatment System is described in Monticello USAR Section 5.3.4.1, "Standby Gas Treatment System (SGTS)."

In addition to the evaluations described in the Monticello USAR, Monticello's systems and components were evaluated for License Renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Monticello Nuclear Generating Plant License Renewal Safety Evaluation Report (SER), NUREG-1865, dated October 2006 (Reference 5). The license renewal evaluation associated with the Standby Gas Treatment System is documented in NUREG-1865, Section 2.3.2.8. The management of the effects of aging on the Standby Gas Treatment System is documented in NUREG-1865, Section 3.2.2.3.8.

Technical Evaluation

The ESF atmosphere cleanup system at Monticello is the Standby Gas Treatment System and its evaluation at EPU conditions is described in Section 2.5.2.

Conclusion

NSPM has evaluated the effects of the proposed EPU on the ESF atmosphere cleanup systems and accounted for any increase of fission products and changes in expected environmental conditions that would result from the proposed EPU. The evaluation indicates that the ESF atmosphere cleanup systems will continue to provide adequate fission product removal in post accident environments following implementation of the proposed EPU. The ESF atmosphere cleanup systems will continue to meet the requirements of the current licensing basis. Therefore, the proposed EPU is 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 acceptance criteria for the CRAVS are based on (1) GDC-4, 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-19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident; and (3) GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents.

Specific NRC review criteria are contained in SRP Section 9.4.1.

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC-4 and GDC-60 listed in the Regulatory Evaluation above, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-11, draft GDC-40, draft GDC-42, and draft GDC-70. Current GDC-19 is applicable to Monticello as described in USAR Sections 5.3.5, 6.7.2, 12.3.1.6, and 14.7.

The control room area ventilation system is described in Monticello USAR Section 6.7, "Main Control Room, Emergency Filtration Train Building and Technical Support Center Habitability."

In addition to the evaluations described in the Monticello USAR, Monticello's systems and components were evaluated for License Renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Monticello Nuclear Generating Plant License Renewal Safety Evaluation Report (SER), NUREG-1865, dated October 2006 (Reference 5). The license renewal evaluation associated with the control room area ventilation system is documented in NUREG-1865, Section 2.3.3.7. Management of aging effects on the control room area ventilation system is documented in NUREG-1865, Section 3.3.2.3.7.

Technical Evaluation

The heating, ventilation and air conditioning (HVAC) systems discussed in the CLTR are only those that have power dependent heat loads. Power dependent HVAC systems require plant specific evaluations for EPU. The Main Control Room (MCR) and Emergency Filtration Train (EFT) Building Ventilation System provides ventilation to the MCR and the first and second floors of the EFT Building. Heat loads for the MCR and EFT Buildings include conduction through the walls, lights, solar loads, and equipment loads from the control room panels. These heat loads are not affected by the slightly higher process temperatures that may result from EPU, thus they are not power dependent. The control of the concentration of airborne radioactive material in the MCR and EFT Building during normal operation, during AOOs, and after postulated accidents is accomplished in conjunction with the MCR and EFT Building Ventilation System using the Main Control Room Atmosphere Control System described in Section 2.7.1.

Conclusion

NSPM has evaluated 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 evaluation indicates that the CRAVS will continue to provide an acceptable control room environment for safe operation of the plant following implementation of the proposed EPU and will continue to meet the requirements of the current licensing basis. Therefore, the proposed EPU is 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 spent fuel pool equipment areas, permit personnel access, and control airborne radioactivity in the area during normal operation, AOOs, and following postulated fuel handling accidents.

The NRC's acceptance criteria for the SFPAVS are based on (1) GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents, and

(2) GDC-61, insofar as it requires that systems which contain radioactivity be designed with appropriate confinement and containment.

Specific NRC review criteria are contained in SRP Section 9.4.2.

Monticello Current Licensing Basis

The Monticello design does not include a separate spent fuel pool area ventilation system. Ventilation in this area is provided by the Reactor Building HVAC system under normal conditions. When required, the Standby Gas Treatment System maintains ventilation for this area.

Technical Evaluation

The Monticello design does not include a separate spent fuel pool area ventilation system. Ventilation in this area is provided by the Reactor Building HVAC system under normal conditions. When required, the Standby Gas Treatment System maintains ventilation for this area. The evaluation of the Standby Gas Treatment System at EPU conditions is described in Sections 2.5.2.1 and 2.6.6.

Conclusion

Not applicable. The Monticello design does not include a separate spent fuel pool area ventilation system.

2.7.5 Auxiliary and Radwaste Areas 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's acceptance criteria for the RAVS and TAVS are based on GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents.

Specific NRC review criteria are contained in SRP Sections 9.4.3 and 9.4.4.

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of

the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC listed in the Regulatory Evaluation above, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-70.

The Radwaste Area and Turbine Area Ventilation Systems are described in Monticello USAR Section 10.3.2, "Plant Heating Ventilating and Air Conditioning Systems."

In addition to the evaluations described in the Monticello USAR, Monticello's systems and components were evaluated for License Renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Monticello Nuclear Generating Plant License Renewal Safety Evaluation Report (SER), NUREG-1865, dated October 2006 (Reference 5). The license renewal evaluation associated with the auxiliary and radwaste area ventilation system is documented in NUREG-1865, Section 2.3.3.11. The management of the effects of aging on the auxiliary and radwaste area ventilation system is documented in NUREG-1865, Section 3.3.2.3.11.

Technical Evaluation

The HVAC systems discussed in the CLTR are only those that have power dependent heat loads. Power dependent HVAC systems require plant specific evaluations for EPU. The power dependent HVAC systems serve the Turbine Building, Radwaste Building and consist mainly of heating, cooling supply, exhaust, and recirculation units. EPU may affect the HVAC systems serving these areas as a result of slightly higher process temperatures and small increases in the heat load due to higher electrical power requirements of certain motors. Ventilation within secondary containment is provided by the Reactor Building HVAC system under normal conditions. When required, the Standby Gas Treatment System maintains ventilation for this area. The evaluation of the Standby Gas Treatment System at EPU conditions is described in Sections 2.5.2.1 and 2.6.6.

The plant areas with higher heat loads due to EPU are:

- Reactor Building - Steam Tunnel, HPCI room, RHR and Core Spray pump rooms (Section 2.7.6)

- Turbine Building – Feedwater and Condensate pump areas, and associated switchgear

The increase in temperature in the Reactor Building areas will be ≤1.8°F as a result of minor heat load increases and is within the design temperatures for the areas. Modifications for the condensate and feedwater pumps/motors are necessary for full EPU operation, which will increase heat loads in the Turbine Building. The ventilation systems in the condensate and feedwater pump areas, and associated switchgear, will be evaluated in more detail when the modification designs are confirmed and the ventilation systems will be modified for EPU to accommodate the increased heat loads to maintain these area temperatures within acceptable values if necessary.

Conclusion

NSPM has evaluated the effects of the proposed EPU on the power dependent HVAC systems that serve the Turbine Building and Radwaste Building. Several plant areas will have higher heat loads but HVAC system operation is not adversely affected. The HVAC systems in the condensate and feedwater pump areas, and associated switchgear, will be evaluated in more detail and modified if necessary to support EPU operation as a result of the modifications to those systems for EPU. Therefore, the proposed EPU is acceptable with respect to HVAC system operation in the Turbine Building, Reactor Building, and drywell.

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's acceptance criteria for the ESFVS are based on (1) GDC-4, 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, insofar as it requires onsite and offsite electric power systems be provided to permit functioning of SSCs important to safety; and (3) GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents.

Specific NRC review criteria are contained in SRP Section 9.4.5.

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of

1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC listed in the Regulatory Evaluation above, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-24, draft GDC-39, draft GDC-40, draft GDC-42, and draft GDC-70.

The engineered safety feature ventilation system is discussed in Monticello USAR Section 10.4.1, "Plant Service Water System," and USAR Appendix I, Section 4.3.10, "Emergency Service Water Systems."

In addition to the evaluations described in the Monticello USAR, Monticello's systems and components were evaluated for License Renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Monticello Nuclear Generating Plant License Renewal Safety Evaluation Report (SER), NUREG-1865, dated October 2006 (Reference 5). The license renewal evaluation associated with the engineered safety feature ventilation system is documented in NUREG-1865, Section 2.3.3.8. Management of aging effects on the engineered safety feature ventilation system is documented in NUREG-1865, Section 3.3.2.3.8.

Technical Evaluation

The Engineered Safety Feature (ESF) HVAC systems consist mainly of heating, cooling supply, exhaust, and recirculation units serving the HPCI, RHR and Core Spray pump rooms in the Reactor Building, the Control Room and the Diesel Generator Building as well as the Standby Gas Treatment System (SGTS). EPU may affect the HVAC serving these areas as a result of slightly higher process temperatures.

The power dependent areas are the RHR and Core Spray pump rooms and the HPCI Room in the Reactor Building due to higher accident suppression pool temperature. The increased heat loads in the RHR and Core Spray pump rooms during system operation result in a $\leq 1.8^{\circ}F$ increase. The increase in heat load and the resultant room temperature will not exceed the design temperature for this area. HPCI Room ventilation is not credited for maintaining room temperature. Room temperature is expected to remain within the design temperature without taking credit for HVAC operation.

HVAC systems for the Diesel Generator Building and MCR are unaffected by EPU because the heat loads in these areas are not affected by EPU, so there is no expected change to the room temperatures in these areas.

Ventilation within secondary containment is provided by the Reactor Building HVAC system under normal conditions. When required, the Standby Gas Treatment System maintains ventilation for this area. The evaluation of the Standby Gas Treatment System at EPU conditions is described in Sections 2.5.2.1 and 2.6.6.The control of the concentration of airborne radioactive material in the MCR during normal operation, during AOOs, and after postulated accidents is described in Section 2.7.1.1, Main Control Room Atmosphere Control System.

Based on a review of design basis calculations and design temperatures, the design of the HVAC in these areas is adequate for EPU.

Conclusion

NSPM has evaluated the effects of the proposed EPU on the ESFVS and the effects of the proposed EPU on the ability of the ESFVS to provide a suitable environment for ESF components. The evaluation indicates that the ESFVS will continue to assure a suitable environment for the ESF components following implementation of the proposed EPU and will continue to meet the requirements of the current licensing basis. Therefore, the proposed EPU is acceptable with respect to the ESFVS.

2.8 Reactor Systems

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's acceptance criteria are based on (1) 10 CFR 50.46, insofar as it establishes standards for the calculation of emergency core cooling system (ECCS) performance and acceptance criteria for that calculated performance; (2) GDC-10, insofar as it requires that the reactor core be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs; (3) GDC-27, insofar as it requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to assure the capability to cool the core is maintained; and (4) GDC-35, insofar as it requires that a system to provide abundant emergency core cooling be provided to transfer heat from the reactor core following any LOCA.

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

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC listed in the Regulatory Evaluation above, with the exception of current GDC-27, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft

GDC") is contained in Monticello USAR Appendix E: draft GDC-6, draft GDC-37, draft GDC-41, and draft GDC-44. There are no draft GDC directly associated with current GDC-27. Current GDC-10 is applicable to Monticello as described in USAR Section 14.4.

The Fuel System Design is described in Monticello USAR Chapter 3, "Reactor."

Technical Evaluation

The effect and [[ ]] for EPU on the fuel system design at Monticello is described below.

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]] Monticello transitioned to GE14 fuel in Cycle 21 and will continue to use only GEH fuel types through EPU implementation. No new fuel product line designs are introduced and there are no changes to fuel design limits required by EPU. The fuel design limits are established for all new fuel product line designs as a part of the fuel introduction and reload analyses. Therefore, no additional fuel and core design evaluations are required for EPU [[ ]] The fuel channels and control rod blades are not affected by EPU as long as the fuel product line is not changed. The fuel channels are structurally evaluated in the subsection titled Reactor Internals Structural Evaluation (Non-FIV), in Section 2.2.3. The Control Rod Drive system is evaluated in Section 2.8.4.1.

Conclusion

NSPM has evaluated the effects of the proposed EPU on the design of the fuel assemblies, control systems, and reactor core. The evaluation indicates that the fuel system and associated analyses will continue to meet the requirements of 10 CFR 50.46 and the current licensing basis following implementation of the proposed EPU. In addition, NSPM will perform plant specific reload analyses to confirm that fuel design limits and RCPB pressure limits will not be exceeded under EPU conditions. Therefore, the proposed EPU is acceptable with respect to the fuel system design.

2.8.2 Nuclear Design

Regulatory Evaluation

The NRC's acceptance criteria are based on (1) GDC-10, insofar as it requires that the reactor core be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs; (2) GDC-11, insofar as it requires that the reactor core be designed so that the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity; (3) GDC-12, insofar as it requires that the reactor core be designed to assure that power oscillations, which can result in conditions exceeding SAFDLs, are not possible or can be reliably and readily detected and suppressed; (4) GDC-13, insofar as it requires that instrumentation and controls be provided to monitor variables and systems affecting the fission process over anticipated ranges for normal operation, AOOs and accident conditions, and to maintain the variables and systems within prescribed operating ranges; (5) GDC-20, insofar as it requires that the protection system be designed to initiate the reactivity control systems automatically to assure that acceptable fuel design limits are not exceeded as a result of AOOs and to automatically initiate operation of systems and components important to safety under accident conditions; (6) GDC-25, insofar as it requires that the protection system be designed to assure that SAFDLs are not exceeded for any single malfunction of the reactivity control systems; (7) GDC-26, insofar as it requires that two independent reactivity control systems be provided, with both systems capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes; (8) GDC-27, insofar as it requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to assure the capability to cool the core is maintained; and (9) GDC-28, insofar as it requires that the reactivity control systems be designed to assure that the effects of postulated reactivity accidents can neither result in damage to the RCPB greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals so as to significantly impair the capability to cool the core.

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

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release.

As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC listed in the Regulatory Evaluation above, with the exception of current GDC-27, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-6, draft GDC-7, draft GDC-8, draft GDC-12, draft GDC-13, draft GDC-14, draft GDC-15, draft GDC-27, draft GDC-29, draft GDC-30, draft GDC-31, and draft GDC-32. There is no draft GDC directly associated with current GDC-27. Current GDC-10, current GDC-20, current GDC-25, and current GDC-26 are applicable to Monticello as described in USAR Section 14.4. Current GDC-12 is applicable to Monticello as described in USAR Section 14.6. Current GDC-13 is applicable to Monticello as described in USAR Section 14.7.4.

Nuclear design is described in Monticello USAR Section USAR Chapter 3, "Reactor."

Technical Evaluation

2.8.2.1 Core Design

The effect and [[ ]] for EPU on the core design at Monticello is described below.

[[[illegible]	[illegible]	[illegible]
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EPU increases the average power density proportional to the power increase and has some effects on operating flexibility, reactivity characteristics and energy requirements. The additional energy requirements for EPU 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. The power distribution in the core is changed to achieve increased core power, while limiting the minimum critical power ratio (MCPR), linear heat generation rate (LHGR), and maximum average planar linear heat generation rate (MAPLHGR) in any individual fuel bundle to be within its allowable value (AV) as defined in the COLR.

2.8.2.2 Fuel Thermal Margin Monitoring Threshold

The percent power level above which fuel thermal margin monitoring is required may change with EPU. The original plant operating licenses set this monitoring threshold at a typical value of 25% of rated thermal power (RTP). Because the fuel thermal margin monitoring is a fuel bundle requirement, it is more appropriate to consider the monitoring threshold in terms of the average bundle power. The average bundle power for the highest power density BWR with the plant operating at OLTP is 4.8 MWt. At a power level of 25% of OLTP, the average bundle power for this plant is 1.2 MWt. Therefore, the fuel thermal margins must be monitored when the average bundle power exceeds 1.2 MWt. Consequently, below an average bundle power of 1.2 MWt, the bundle powers are low enough such that thermal margin monitoring is not required. For Monticello at an EPU power level of 25%, the average bundle power is 1.0 MWt.

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]], to ensure that monitoring is initiated by the time the average bundle power reaches 1.2 MWt. Specifically, if the average bundle power at 25% EPU RTP (P_{25}) increases above 1.2 MWt, then the existing power threshold value must be lowered by a factor of $1.2/P_{25}$.

For Monticello, the fuel thermal monitoring threshold is established at 25% of EPU RTP. Thus, there is no change in the fuel thermal monitoring threshold and no change to the Technical Specification reactor core safety limit for reduced pressure or low core flow.

2.8.2.3 Thermal Limits Assessment

The effect of EPU on the MCPR safety and operating limits and on the MAPLHGR and LHGR limits for Monticello is addressed below.

Operating limits ensure that regulatory and/or safety limits are not exceeded for a range of postulated events (e.g., transients, LOCA). This section addresses the effects of EPU on thermal limits. A representative cycle core is used for the EPU evaluation. Cycle-specific core configurations, evaluated for each reload, confirm EPU capability, and establish or confirm cycle-specific limits, as is currently the practice.

2.8.2.3.1 Safety Limit MCPR

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The Safety Limit MCPR (SLMCPR) can be affected slightly by EPU due to the flatter power distribution inherent in the increased power level. Experience has shown that the power uprate flatter power distribution results in an increase in the SLMCPR of ≤ 0.01. This effect is not changed by the EPU approach (Reference 1). The SLMCPR analysis reflects the actual plant core-loading pattern and is performed for each plant reload core (see Reference 4). [[

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]] The calculated values will be reported in the Supplemental Reload Licensing Report (SRLR) for the EPU core. The SLMCPR will include a 0.02 adder for SLO as required by Reference 25.

2.8.2.3.2 MCPR Operating Limit

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[illegible]	[illegible]	[illegible]
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Per the CLTR, the EPU operating conditions have only a small effect on the MCPR Operating Limit. The MCPR Operating Limit is calculated by adding the change in MCPR due to the limiting AOO event to the SLMCPR and is determined on a cycle specific basis. EPU does not change the method used to determine this limit. The effect of EPU on AOO events is addressed in Section 2.8.5. Based on experience with previous plant specific power uprate submittals, the effect on the MCPR Operating Limit due to EPU is small and does not significantly affect plant operation. [[

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2.8.2.3.3 MAPLHGR and LHGR Operating Limits

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The MAPLHGR and Maximum LHGR Operating Limits ensure that the plant does not exceed regulatory limits established in 10 CFR 50.46 or the fuel design limits. The MAPLHGR Operating Limit is determined by analyzing the limiting loss-of-coolant accident (LOCA) for the plant. As discussed in Section 4.3 of the CLTR, EPU does not usually affect the MAPLHGR Operating Limit for plant operation. The Maximum LHGR Operating Limit is determined by the fuel rod thermal-mechanical design and is not affected by EPU. [[

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2.8.2.4 Reactivity Characteristics

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The general effect of power uprate on core reactivity is described in Section 5.7.1 of ELTR1, (Reference 2) and is also applicable to EPU. Based on experience with previous plant specific power uprate submittals, the required hot excess reactivity and shutdown margin can be achieved for EPU through appropriate fuel and core design. [[

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The Monticello reactivity characteristics are confirmed to be consistent with the generic description provided in the CLTR because the reactivity characteristics are evaluated for the uprated reload core prior to EPU implementation.

2.8.2.5 Core Exit High Powered Bundle Void Fraction and Power to Flow Ratio Evaluations

The peak bundle exit void fractions at 3 points: 120% OLTP/100% CF, 120% OLTP/80% CF, and 96.8% OLTP/55% CF, where 120% OLTP is the EPU power (2004 MWt) are provided in Table 2.8-1. The results are for the beginning, middle and end of an EPU RTP core cycle and are compared to a recent Monticello cycle (24) at 106.29% OLTP/82.4% core flow (currently licensed minimum CF). The maximum bundle exit void fraction for EPU is 0.896 compared to the current maximum bundle exit void fraction of 0.845.

The GE14 bundle R-factors are consistent with Monticello hot channel axial void conditions for EPU operation. The axial profile used for the R-factor calculations has an average void fraction of 0.60 and an exit void fraction of 0.86.

The core thermal power to total core flow ratio is reported in Table 2.8-2. The EPU RTP is 2004 MWt, the 100% rated flow is 57.6 Mlbm/hr. As shown in Table 2.8-2, the core thermal power to total core flow ratio does not exceed 50 MWt/Mlbm/hr.

2.8.2.6 Monticello Specific Data Requested in the Interim Methods LTR

The following parameters: (1) Maximum Bundle Power; (2) Flow for Peak Bundle; (3) Exit Void Fraction for Peak Power Bundle; (4) Maximum Channel Exit Void Fraction; (5) Core Average Exit Void Fraction; (6) Peak LHGR; and (7) Peak Nodal Exposure are shown in Figures 2.8-1 through 2.8-7. The Monticello data are plotted with the available EPU experience base (Reference 26).

Quarter core maps with mirror symmetry are plotted in Figures 2.8-8 through 2.8-18 showing bundle power, bundle operating LHGR, and MCPR for BOC, MOC, and EOC. Because the minimum margins to specific limits occur at exposures other than the traditional BOC, MOC, and EOC, the data is provided at these other exposures as applicable (Figures 2.8-17 and 2.8-18). The bundle power is dimensionless. To obtain the bundle powers in MWt, multiply each value by the average bundle power of 4.141. The average bundle power is equal to 2004/484, where 2004 MWt is the EPU RTP and 484 is the total number of bundles in the core (Reference 26).

A listing of all of the Limitations in the Safety Evaluation for the Interim Methods LTR and their disposition for Monticello is provided in Appendix A.

Conclusion

NSPM has evaluated the effect of the proposed EPU on the nuclear design of the fuel assemblies, control systems, and reactor core. The evaluation indicates that the nuclear design of the fuel assemblies, control systems, and reactor core will continue to meet the current licensing basis. In addition, NSPM will perform plant specific reload analyses to confirm that fuel design limits and RCPB pressure limits will not be exceeded under EPU conditions. Therefore, the proposed EPU is acceptable with respect to the nuclear design.

2.8.3 Thermal and Hydraulic Design

Regulatory Evaluation

The NRC's acceptance criteria are based on (1) GDC-10, insofar as it requires that the reactor core be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs; and (2) GDC-12, insofar as it requires that the reactor core and associated coolant, control, and protection systems be designed to assure that power oscillations, which can result in conditions exceeding SAFDLs, are not possible or can reliably and readily be detected and suppressed.

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

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC listed in the Regulatory Evaluation above, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-6 and draft GDC-7. Current GDC-10 is applicable to Monticello as described in USAR Section 14.4. Current GDC-12 is applicable to Monticello as described in USAR Section 14.6.

The Thermal and Hydraulic Design is described in Monticello USAR Section 3.2, "Thermal and Hydraulic Characteristics." Power oscillations are addressed in USAR Section 14.6, "Plant Stability Analysis."

Technical Evaluation

Section 3.2 of ELTR1 documents interim corrective actions and four long-term stability options. Monticello has adopted Option III (Reference 29). Option III evaluations are core reload dependent and are performed for each reload fuel cycle. The Monticello Option III hardware will be installed and connected to the Reactor Protection System. In the event that the OPRM system is declared inoperable, Monticello will operate under the BWROG Guidelines for Backup Stability Protection (BSP) as described in Reference 31. When the EPU is implemented, cycle specific setpoints will be determined and documented in the same Supplemental Reload Licensing Report (SRLR).

2.8.3.1 Option III

The Option III solution combines closely spaced LPRM detectors into "cells" to effectively detect either core-wide or regional (local) modes of reactor instability. These cells are termed Oscillation Power Range Monitor (OPRM) cells and are configured to provide local area coverage with multiple channels. Plants implementing Option III have hardware to combine the LPRM signals and to evaluate the cell signals with instability detection algorithms. The Period Based Detection Algorithm (PBDA) is the only algorithm credited in the Option III licensing basis. Two defense-in-depth algorithms, referred to as the Amplitude Based Algorithm (ABA) and the Growth Rate Based Algorithm (GRBA), offer a high degree of assurance that fuel failure will not occur as a consequence of stability related oscillations.

The Option III trip is armed only when plant operation is within the Option III trip-enabled region. The Option III trip-enabled region is generically defined as the region on the power/flow map with power ≥ 30% of OLTP and core flow ≤ 60% of rated core flow. For EPU, the Option III trip-enabled region is rescaled to maintain the same absolute power/flow region boundaries. Because the rated core flow is not changed, the 60% core flow boundary is not rescaled. The 30% of OLTP boundary changes by the following equation:

EPU Region Boundary = 30% OLTP * (100% ÷ EPU (% OLTP))

Thus, for a 120% of OLTP EPU:

EPU Region Boundary = 30% OLTP * (100% ÷ 120%) = 25% EPU

The Monticello OPRM trip-enabled region is shown in Figure 2.8-19. The Backup Stability Protection (BSP) evaluation described in Section 2.8.3.2 shows that the generic Option III Trip Enabled Region is adequate. The adequacy of the OPRM armed Region will be confirmed for each reload.

Stability Option III provides SLMCPR protection by generating a reactor scram if a reactor instability, which exceeds the specified trip setpoint, is detected. The OPRM setpoint is determined per a NRC approved methodology (References 29 and 32).

The Option III stability reload licensing basis calculates the limiting OLMCPR required to protect the SLMCPR for both steady state and transient stability events as specified in the Option III methodology. These OLMCPRs are calculated for a range of OPRM setpoints for EPU operation. Selection of an appropriate instrument setpoint is then based upon the OLMCPR required to provide adequate SLMCPR protection. This determination relies on the DIVOM curve (Delta CPR Over Initial CPR Versus Oscillation Magnitude) to determine an OPRM setpoint that protects the SLMCPR during an anticipated instability event. The DIVOM slope was developed based on a TRACG evaluation in accordance with the BWROG Regional DIVOM Guideline (Reference 32).

The generic analyses for the Option III hot channel oscillation magnitude and the OPRM hardware were designed to be independent of core power. [[

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A demonstration analysis was performed based on an equilibrium GE14 core using nominal core simulator wrap-ups at limiting conditions. For this analysis, a DIVOM slope of 0.45 was used. As shown in Table 2.8-3, with an assumed rated OLMCPR of 1.50 and an off-rated OLMCPR (at 45% Flow) of 1.72, an OPRM setpoint of 1.15 is the highest setpoint that may be used without stability setting the OLMCPR. The actual setpoint will be established in accordance with Monticello Technical Specifications.

Per the SER condition imposed in Reference 26, the OPRM system will incorporate a 5% relative penalty on the OPRM setpoints to address the issue of the bypass voiding at low-flow conditions. As an illustration, the OPRM setpoint of 1.15 shown above will be reduced to 1.14.

2.8.3.2 BSP Evaluation

Monticello will be implementing Backup Stability Protection (BSP) (Reference 31) as the stability licensing basis if the Option III OPRM system is declared inoperable. The BSP evolved from the stability interim corrective actions (ICAs), which restrict plant operation in the high power, low core flow region of the BWR power/flow operating map. The ICAs provide guidance which reduces the likelihood of an instability event by limiting the period of operation in regions of the power/flow map most susceptible to thermal hydraulic instability. The ICAs also specify operator actions, which are capable of detecting conditions consistent with the onset of oscillations, and additional actions, which mitigate the consequences of oscillations consistent with degraded thermal hydraulic stability performance of the core.

The BSP regions consist of two regions (I-Scram and II-Controlled Entry), which are reduced from the three ICA regions (I-Scram, II-Exit and III-Controlled Entry) (References 30 and 31). The standard ICA region state points on the High Flow Control Line (HFCL) and on the Natural Circulation Line (NCL) define the base BSP region state points on the HFCL and on the NCL. The bounding plant specific BSP region state points must enclose the corresponding base BSP region state points on the HFCL and the NCL. If a calculated BSP region state point is located inside the corresponding base BSP region state point, it must be replaced by the corresponding base BSP region state point. If a calculated BSP region state point is located outside the corresponding base BSP region state point, this point is acceptable for use. That is, the selected points will result in the largest, or most conservative, region sizes. The proposed BSP Scram and Controlled Entry region boundaries are constructed by connecting the corresponding bounding state points on the HFCL and the NCL using the Generic Shape Function (GSF).

The GE14 equilibrium demonstration analysis was used to determine the ODYSY calculated BSP boundaries as shown in Table 2.8-4. These ODYSY-calculated BSP boundaries are all smaller than the corresponding Base BSP boundaries and hence the Base BSP boundaries are adopted for the demonstration analysis as shown in Figure 2.8-20.

2.8.3.3 ATWS with Core Instability

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The ATWS with core instability event occurs at natural circulation following an RPT. Therefore, it is initiated at approximately the same power level as a result of EPU operation because the MELLLA upper boundary is not increased. The core design necessary to achieve EPU operations may affect the susceptibility to coupled thermal-hydraulic/neutronic core oscillations at the natural circulation condition, but would not significantly affect the event progression.

Several factors affect the response of an ATWS instability event, including operating power and flow conditions and core design. The limiting ATWS core instability evaluation presented in References 27 and 28 was performed for an assumed plant initially operating at OLTP and the MELLLA minimum flow point. The results showed that for the unmitigated case, a small fraction of the core experiences extended dryout, but the maximum energy deposition meets the licensing limit for reactivity insertion events. For the mitigated case, extended dryout did not occur. EPU allows plants to increase their operating thermal power but does not allow an increase in control rod line. Therefore, as compared to the event initiated from CLTP condition, the event initiated from the EPU condition on the same rod line will end up at approximately the same power level at natural circulation. The conclusion of Reference 28 and the associated NRC SER that the analyzed operator actions effectively mitigate an ATWS instability event are applicable to the operating conditions expected for EPU at Monticello.

The EPU effect on ATWS with core instability at Monticello has been confirmed to be consistent with the generic evaluation in the CLTR, because the maximum rod line is unchanged and operator actions are expected to mitigate an ATWS instability event at EPU conditions.

Conclusion

NSPM has evaluated the effects of the proposed EPU on the thermal and hydraulic design of the core and the RCS. In addition, NSPM will perform plant specific reload analyses to confirm that fuel design limits and RCPB pressure limits will not be exceeded under EPU conditions, and that the core design is not susceptible to thermal-hydraulic instability. The evaluation indicates that the thermal and hydraulic design will continue to meet the requirements of the current licensing basis following implementation of the proposed EPU. Therefore, the proposed EPU is 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's acceptance criteria are based on (1) GDC-4, 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-23, insofar as it requires that the protection system be designed to fail into a safe state; (3) GDC-25, insofar as it requires that the protection system be designed to assure that SAFDLs are not exceeded for any single malfunction of the reactivity control systems; (4) GDC-26, insofar as it requires that two independent reactivity control systems be provided, with both systems capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes; (5) GDC-27, insofar as it requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to assure the capability to cool the core is maintained; (6) GDC-28, insofar as it requires that the reactivity control systems be designed to assure that the effects of postulated reactivity accidents can neither result in damage to the RCPB greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals so as to significantly impair the capability to cool the core; (7) GDC-29, insofar as it requires that the protection and reactivity control systems be designed to assure an extremely high probability of accomplishing their safety functions in event of AOOs; and (8) 10 CFR 50.62(c)(3), insofar as it requires that all BWRs have an alternate rod injection (ARI) system diverse from the reactor trip system, and that the ARI system have redundant scram air header exhaust valves.

Specific NRC review criteria are contained in SRP Section 4.6.

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC listed in the Regulatory Evaluation above, with the exception of current GDC-27 and GDC-29, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-26, draft GDC-27, draft GDC-29, draft GDC-30, draft GDC-31, draft GDC-32, draft GDC-40, and draft GDC-42. There is no draft GDC directly associated with current GDC-27 or current GDC-29. Current GDC-25 and current GDC-26 are applicable to Monticello as described in USAR Section 14.4.

The design of Control Rod Drive System is described in Monticello USAR Section 3.5.3, "Control Rod Drive System."

In addition to the evaluations described in the Monticello USAR, Monticello's systems and components were evaluated for License Renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Monticello Nuclear Generating Plant License Renewal Safety Evaluation Report (SER), NUREG-1865, dated October 2006 (Reference 5). The license renewal evaluation associated with the Control Rod Drive System is documented in NUREG-1865, Section 2.3.3.4. Management of aging effects on the Control Rod Drive System is documented in NUREG-1865, Section 3.3.2.3.4.

Technical Evaluation

The Control Rod Drive (CRD) system is used to control core reactivity by positioning neutron absorbing control rods within the reactor and to scram the reactor by rapidly inserting withdrawn control rods into the core. No change is made to the control rods due to the EPU. The Monticello Alternate Rod Insertion (ARI) system is not affected by EPU. The topics addressed in this evaluation for Monticello are:

2.8.4.1.1 Control Rod Scram

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[redacted]	[redacted]	[redacted]
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For pre-BWR/6 plants, the scram times are decreased by the transient pressure increase, [[ ]] At normal operating conditions, the CRD Hydraulic Control Unit accumulator supplies the initial scram pressure and, as the scram continues, the reactor becomes the primary source of pressure to complete the scram. [[

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2.8.4.1.2 *Control Rod Drive Positioning and Cooling*

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[illegible]	[illegible]	[illegible]

[illegible]	[illegible]	[illegible]
		]]

[[

]] and the automatic operation of the CRD system flow control valve maintains the required drive water pressure and cooling water flow rate. Therefore, the CRD system positioning and cooling functions are not affected. The CRD system cooling and normal positioning functions are operational considerations, not safety-related functions, and are not affected by EPU operating conditions.

Plant operating data regarding the operating position of the CRD system flow control valve have confirmed that the valve has sufficient operating margin. [[

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2.8.4.1.3 Control Rod Drive Integrity Assessment

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[illegible]	[illegible]	[illegible]
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The postulated abnormal operating condition for the CRD design assumes a failure of the CRD system pressure-regulating valve resulting in the application of the maximum pump discharge pressure to the CRD mechanism internal components. This postulated abnormal pressure bounds the ASME reactor overpressure limit. [[

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Loads from reactor internal pressure differences, flows, and acoustics due to reactor transients do not affect the CRD mechanism because it is located in the CRD Housing and Control Rod Guide Tube. Control rod weight and seismic loads remain unaffected by EPU.

CRD system piping is addressed in Section 2.2.2 of this report.

The CRD system integrity at Monticello [[

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Conclusion

NSPM has evaluated the effects of the proposed EPU on the CRDS. The evaluation indicates that the system's ability to affect a safe shutdown, respond within acceptable limits, and prevent or mitigate the consequences of postulated accidents will be maintained following the implementation of the proposed EPU. Based on this, NSPM concludes that the fuel system and associated analyses will continue to meet the requirements of the current licensing basis and 10 CFR 50.62(c)(3) following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to the functional design of the CRDS.

2.8.4.2 Overpressure Protection During Power Operation

Regulatory Evaluation

Relief and safety valves and the reactor protection system provide overpressure protection for the RCPB during power operation.

The NRC's acceptance criteria are based on (1) GDC-15, insofar as it requires that the RCS and associated auxiliary, control, and protection systems be designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including AOOs; and (2) GDC-31, insofar as it requires that the RCPB be designed with sufficient margin to assure that it behaves in a non-brittle manner and that the probability of rapidly propagating fracture is minimized.

Specific NRC review criteria are contained in SRP Section 5.2.2.

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC listed in the Regulatory Evaluation above, with the exception of current GDC-15, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-33, draft GDC-34, and draft GDC-35. There is no draft GDC directly associated with current GDC-15. Current GDC-15 is applicable to Monticello as described in USAR Section 14.4.

Overpressure protection during power operation is described in Monticello USAR Section 4.4, "Reactor Pressure Relief System."

In addition to the evaluations described in the Monticello USAR, Monticello's systems and components were evaluated for License Renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Monticello Nuclear Generating Plant License Renewal Safety Evaluation Report (SER), NUREG-1865, dated October 2006 (Reference 5). The license renewal evaluation associated with overpressure protection is documented in NUREG-1865, Section 2.3.2.1. Management of aging effects on overpressure protection is documented in NUREG-1865, Section 3.2.2.3.1.

Technical Evaluation

The nuclear system pressure relief system prevents overpressurization of the nuclear system during AOOs, the plant ASME Upset overpressure protection event, and postulated ATWS events. The plant SRVs along with other functions provide this protection. An evaluation was performed in order to confirm the adequacy of the pressure relief system for EPU conditions. The adequacy of the pressure relief system is also demonstrated by the overpressure protection evaluation performed for each reload core and by the ATWS evaluation performed for EPU (Section 2.8.5.7). The ODYN computer code is used for both overpressure protection evaluations.

For Monticello, no SRV setpoint increase is needed because there is no change in the dome pressure or simmer margin. Therefore, there is no effect on valve functionality (opening/closing).

Two potentially limiting overpressure protection events are typically analyzed for EPU: (1) Main Steam Isolation Valve Closure with Scram on High Flux (MSIVF) and (2) Turbine Trip with Bypass Failure and Scram on High Flux (ELTR1, Section 5.5.1.4). However, based on both plant initial core analyses and subsequent power uprate evaluations, the MSIVF is more limiting than the turbine trip event with respect to reactor overpressure. Recent EPU evaluations show a 24 to 40 psi difference between these two events. Only the MSIVF event was performed because it is limiting. In addition, an evaluation of the MSIVF event is performed with each reload analysis.

The design pressure of the reactor vessel and reactor coolant pressure boundary (RCPB) remains at 1250 psig. The acceptance limit for pressurization events is the ASME Code allowable peak pressure of 1375 psig (110% of design value). The overpressure protection analysis description and analysis method are provided in ELTR1. The overpressurization event is conservatively analyzed assuming a failure of the valve position scram. The analyses also assume that the event initiates at a reactor dome pressure of 1040 psia (which is higher than the maximum normal EPU dome pressure), and three SRVs out-of-service (OOS). Starting from 102% of EPU RTP, the calculated peak reactor pressure vessel (RPV) pressure, located at the bottom of the vessel, is 1335 psig. The corresponding calculated maximum reactor dome pressure is 1317 psig. The peak calculated RPV pressure remains below the 1375 psig ASME limit, and the maximum calculated dome pressure remains below the Technical Specification 1332 psig Safety Limit. Therefore, the results are acceptable and within AVs. The results of the EPU overpressure protection analysis for the Monticello MSIVF event are consistent with the generic analysis in ELTR2. The Monticello response to the MSIVF event is provided as Figure 2.8-21.

Conclusion

NSPM has evaluated the effects of the proposed EPU on the overpressure protection capability of the plant during power operation. The evaluation indicates that the plant will continue to have sufficient pressure relief capacity to ensure that pressure limits are not exceeded. Based on this, NSPM concludes that the overpressure protection features will continue to meet the current licensing basis following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to overpressure protection during power operation.

2.8.4.3 Reactor Core Isolation Cooling System

Regulatory Evaluation

The reactor core isolation cooling (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 condensate storage tank, with a secondary supply from the suppression pool.

The NRC's acceptance criteria are based on (1) GDC-4, insofar as it requires that SSCs important to safety be protected against dynamic effects; (2) GDC-5, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be demonstrated that sharing will not impair its ability to perform its safety function; (3) GDC-29, insofar as it requires that the protection and reactivity control systems be designed to assure an extremely high probability of accomplishing their safety functions in event of AOOs; (4) GDC-33, insofar as it requires that a system to provide reactor coolant makeup for protection against small breaks in the RCPB be provided so the fuel design limits are not exceeded; (5) GDC-34, insofar as it requires that a residual heat removal system be provided to transfer fission product decay heat and other residual heat from the reactor core at a rate such that SAFDLs and the design conditions of the RCPB are not exceeded; (6) GDC-54, 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 (7) 10 CFR 50.63, insofar as it requires that the plant withstand and recover from an SBO of a specified duration.

Specific NRC review criteria are contained in SRP Section 5.4.6.

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC listed in the Regulatory Evaluation above, with the exception of current GDC-29 and current GDC-34, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-4, draft GDC-37, draft GDC-40, draft GDC-42, draft GDC-51, and draft GDC-57. There is no draft GDC directly applicable to current GDC-29 or current GDC-34.

The Reactor Core Isolation Cooling System is described in Monticello USAR Section 10.2.5, "Reactor Core Isolation Cooling System (RCIC)."

In addition to the evaluations described in the Monticello USAR, Monticello's systems and components were evaluated for License Renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Monticello Nuclear Generating Plant License Renewal Safety Evaluation Report (SER), NUREG-1865, dated October 2006 (Reference 5). The license renewal evaluation associated with the RCIC system is documented in NUREG-1865, Section 2.3.2.6. Management of aging effects on the RCIC system is documented in NUREG-1865, Section 3.2.2.3.6.

Technical Evaluation

The RCIC system evaluation for EPU at Monticello addressed the following topics:

- System performance and hardware
- Net positive suction head[1]
- Adequate core cooling for limiting LOFW events (Section 2.8.5.2.3)

[[

[redacted]	[redacted]	[redacted]
		]] SRV Nominal Trip Setpoints (NTSPs) – 1109 psig.

The design basis event for RCIC system operation is the limiting LOFW transient with reactor isolation. The RCIC system is assumed to inject 400 gpm from the CST or torus following a LOFW event with reactor isolation and the HPCI system unavailable with rated flow available in 30 seconds. For this event, the reactor water level is required to remain above the TAF to maintain adequate core cooling. The reactor system response to an LOFW transient with RCIC is discussed in Section 2.8.5.2.3. The results of the Monticello plant specific evaluation indicate adequate water level margin above TAF at the EPU conditions. Thus, the RCIC injection rate is adequate for this design basis event.

For the EPU, there is no change to the normal reactor operating pressure and the SRV setpoints remain the same. There is no change to the RCIC system performance parameters. Therefore, no changes are required to meet the performance requirements for the RCIC system or to limit the maximum startup transient speed peak.

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[1] [[

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Some BWR plants have an operational requirement for RCIC to prevent the level decrease during a LOFW transient from initiating ADS. This operational requirement does not apply to Monticello because both RCIC and ADS initiation are activated at the same reactor water level.

The RCIC system has the capability of using the CST or the torus as a suction source. At CLTP and EPU conditions, the CST provides additional head over that provided by the torus for the RCIC pump, and the CST is not subject to the heat addition from reactor blowdown that reduces suction head. Consequently, torus suction under EPU conditions is more limiting for RCIC NPSH. Operation from the CST is not credited for RCIC operation. For Monticello, the torus provides the safety-related suction source for the RCIC system. The torus temperature increases at EPU conditions and therefore the available NPSH is affected.

Monticello calculations demonstrated that the RCIC pump would have adequate NPSH and low suction pressure trip margins given a torus water temperature of 170°F. Other than suction water temperature, EPU has no affect on the assumptions for NPSH such as piping configuration, flow rate, etc.

The higher EPU decay heat will increase the torus water suction temperature for both the LOFW and SBO transients. Although the RCIC system is not credited in the design basis mitigation of the SBO event at Monticello, the SBO torus water temperature response is useful for the purposes of evaluating RCIC NPSH during a LOFW transient. This is because the events are similar with respect to heat transfer to the torus, and the SBO torus temperature response well bounds the LOFW torus temperature response as torus cooling is available during a LOFW. The resultant torus temperature determined for SBO (~141°F at two hours) demonstrates that the 170°F torus water temperature limit for RCIC NPSH will not be approached during the design basis LOFW event.

The RCIC system performance and hardware at Monticello is confirmed to be consistent with the [[ ]]. A [[ ]] of RCIC NPSH and the LOFW transient indicates the RCIC system as acceptable for operation at EPU conditions.

Conclusion

NSPM has evaluated 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. The evaluation indicates that the RCIC system will continue to provide sufficient decay heat removal and makeup for this event following implementation of the proposed EPU. Based on this, NSPM concludes that the RCIC system will continue to meet the requirements of the current licensing basis following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to the RCIC system.

2.8.4.4 Residual Heat Removal System

Regulatory Evaluation

The RHR system is used to cool down the RCS following shutdown. The RHR system is typically a low-pressure system that takes over the shutdown cooling function when the RCS temperature is reduced.

The NRC's acceptance criteria are based on (1) GDC-4, insofar as it requires that SSCs important to safety be protected against dynamic effects; (2) GDC-5, 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; and (3) GDC-34, which specifies requirements for an RHR system.

Specific NRC review criteria are contained in SRP Section 5.4.7 and other guidance provided in Matrix 8 of RS-001.

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC listed in the Regulatory Evaluation above, with the exception of current GDC-34, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-4, draft GDC-40, and draft GDC-42. There is no draft GDC directly associated with current GDC-34.

The Residual Heat Removal system is described in Monticello USAR Section 6.2.3, "Residual Heat Removal system (RHR)."

In addition to the evaluations described in the Monticello USAR, Monticello's systems and components were evaluated for License Renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Monticello Nuclear Generating Plant License Renewal Safety Evaluation Report (SER), NUREG-1865, dated October 2006 (Reference 5). The license renewal evaluation associated with the RHR system is documented in NUREG-1865, Section 2.3.2.7. Management of aging effects on the RHR system is documented in NUREG-1865, Section 3.2.2.3.7.

Technical Evaluation

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. The EPU effect on the RHR system is a result of 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. For Monticello, the RHR system is designed to operate in the LPCI mode, Shutdown Cooling (SDC) mode, Suppression Pool Cooling (SPC) mode, Containment Spray Cooling (CSC) mode, and Fuel Pool Cooling (FPC) Assist (Supplemental Spent Fuel Pool Cooling) mode. Steam Condensing Mode of RHR is not installed at Monticello.

The LPCI mode, as it relates to the LOCA response, is discussed in Section 2.8.5.6.2.

The SPC mode is manually initiated following isolation transients and a postulated LOCA to maintain the containment pressure and suppression pool temperature within design limits. The CSC mode reduces drywell pressure, drywell temperature, and suppression chamber pressure following an accident. The adequacy of these operating modes is demonstrated by the containment analysis (Section 2.6.5).

The higher suppression pool temperature and containment pressure during a postulated LOCA do not affect hardware capabilities of RHR equipment to perform the LPCI, SPC, and CSC functions.

The FPC Assist (Supplemental Spent Fuel Pool Cooling) mode, using existing RHR heat removal capacity, provides supplemental fuel pool cooling capability in the event that the fuel pool heat load exceeds the heat removal capability of the Fuel Pool Cooling and Demineralizer System. The adequacy of fuel pool cooling, including use of the Supplemental Spent Fuel Pool Cooling mode, is addressed in Section 2.5.3.1.1.

The effects of EPU on the remaining modes are discussed in the following subsections.

Shutdown Cooling Mode

The SDC mode removes the sensible and decay heat from the reactor primary system during a normal reactor shutdown.

Regulatory Guide (RG) 1.139, "Guidance for Residual Heat Removal," recommends demonstrating the capability of achieving cold shutdown conditions (≤ 212 °F reactor fluid temperature) within

36 hours. For the EPU, an alternate shutdown cooling analysis based on the criteria in RG 1.139 was performed. The results of this analysis show that the reactor can be cooled to ≤ 212 °F in less than the 36-hour criterion.

EPU increases the reactor decay heat, which requires a longer time for cooling down the reactor. The SDC analysis for the EPU determined that the time needed for cooling the reactor to 125 °F during normal reactor shutdown, with two SDC loops (i.e., two RHR heat exchangers) in service, is increased to approximately 26.5 hours. This calculated normal reactor shutdown time exceeds the OLTP SDC time criterion of 24 hours, which was selected, based on engineering judgment so that the SDC mode of operation will not become a critical path during refueling operations. This SDC design criterion was used as one of the bases for sizing the RHR heat exchangers and does not constitute a plant operational parameter. The increase in the normal reactor shutdown time for EPU indicates that a normal reactor shutdown may take longer, which could delay the start of an outage. This delay may have an effect on plant availability, but has no effect on plant safety, or the design operating margins and therefore, requires no change to the RHR system.

Conclusion

NSPM has evaluated the effects of the proposed EPU on the RHR system. The evaluation indicates that the RHR system will maintain its ability to cool the RCS following shutdown and provide decay heat removal. Based on this, NSPM concludes that the RHR system will continue to meet the requirements of the current licensing basis following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to the RHR system.

2.8.4.5 Standby Liquid Control System

Regulatory Evaluation

The Standby Liquid Control System (SLCS) provides backup capability for reactivity control independent of the control rod system. The SLCS injects a boron solution into the reactor to effect shutdown.

The NRC's acceptance criteria are based on (1) GDC-26, 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) GDC-27, insofar as it requires that the reactivity control systems have a combined capability, in conjunction with poison addition by the ECCS, to reliably control reactivity changes under postulated accident conditions; and (3) 10 CFR 50.62(c)(4), insofar as it requires that the SLCS 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 NRC review criteria are contained in SRP Section 9.3.5 and other guidance provided in Matrix 8 of RS-001.

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC listed in the Regulatory Evaluation above, with the exception of current GDC-27, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-27, draft GDC-29, and draft GDC-30. There is no draft GDC directly associated with current GDC-27. Current GDC-26 is applicable to Monticello as described in USAR Section 14.4.

The SLCS is described in Monticello USAR Section 6.6, "Standby Liquid Control System."

In addition to the evaluations described in the Monticello USAR, Monticello's systems and components were evaluated for License Renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Monticello Nuclear Generating Plant License Renewal Safety Evaluation Report (SER), NUREG-1865, dated October 2006 (Reference 5). The license renewal evaluation associated with the SLCS System is documented in NUREG-1865, Section 2.3.3.17. Management of aging effects on the SLCS is documented in NUREG-1865, Section 3.3.2.3.17.

Technical Evaluation

The SLCS is designed to shut down the reactor from rated power conditions to cold shutdown in the postulated situation that some or all of the control rods cannot be inserted. This manually operated system pumps a highly enriched sodium pentaborate solution into the vessel, to provide neutron absorption and achieve a subcritical reactor condition. SLCS is designed to inject over a wide range of reactor operating pressures.

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[illegible]	[illegible]	[illegible]
		]] Monticello only uses GE14 fuel.

The ability of the SLCS boron solution to achieve and maintain safe shutdown is not a direct function of core thermal power, and therefore, is not affected by the EPU. SLCS shutdown capability (in terms of the required reactor boron concentration) is reevaluated for each fuel reload. No new fuel product line designs are introduced for EPU. The boron shutdown concentration of 660 ppm does not change for EPU. No changes are necessary to the solution volume / concentration or the boron-10 enrichment for EPU to achieve the required reactor boron concentration for shutdown. [[

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The SLCS is designed for injection at a maximum reactor pressure equal to the upper AL for the lowest group of SRVs operating in the safety relief mode. For the EPU, the nominal reactor dome pressure and the SRV setpoints are unchanged. Therefore, the capability of the SLCS to provide its backup shutdown function is not affected by the EPU. The SLCS is not dependent upon any other SRV operating modes.

The boron injection rate requirement for maintaining the peak suppression pool water temperature limits, following the limiting ATWS event with SLCS injection, is not increased for EPU.

Based on the results of the plant specific ATWS analysis, the maximum reactor lower plenum pressure following the limiting ATWS event reaches 1205.3 psig (1220 psia) during the time the SLCS is analyzed to be in operation. This occurs at CLTP conditions. Consequently, there is no corresponding increase in the maximum pump discharge pressure and no decrease in the operating pressure margin for the pump discharge relief valves. The operation of the pump discharge system was analyzed to confirm that the pump discharge relief valves reclose in the event that the system is initiated before the time that the reactor pressure recovers from the first transient peak. The evaluation compared the calculated maximum reactor pressure needed for the pump discharge relief valves to reclose with the lower reactor pressure expected during the time the SRVs are cycling open and closed prior to the time when rated SLCS injection is assumed in the ATWS analysis. Consideration was also given to 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. The pump discharge relief valves are periodically tested to maintain this tolerance. Therefore, the current SLCS process parameters associated with the minimum boron injection rate are not changed.

The SLCS ATWS performance is evaluated in Section 2.8.5.7 for a representative core design for EPU. The evaluation shows that EPU has no adverse effect on the ability of the SLCS to mitigate an ATWS.

Conclusion

NSPM has evaluated the effects of the proposed EPU on the SLCS. The evaluation indicates 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 this, NSPM concludes that the SLCS will continue to meet the requirements of the current licensing basis and 10 CFR 50.62(c)(4) following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to the SLCS.

2.8.4.6 Reactor Recirculation System Performance

The Reactor Recirculation System (RRS) performance is not specifically addressed in RS-001; however, it is addressed in Section 3.6 of the CLTR. Evaluation of the RRS as described in CLTR Section 3.6 is provided below.

The EPU power condition is accomplished by operating along extensions of current rod lines on the Power to Flow map with no increase in the maximum core flow. Extrapolation of the RRS performance to full EPU power indicates that the practical core flow margin is likely to be very limited and may be non-existent. Thus, RRS performance will be monitored to confirm actual capabilities at higher power levels. Technical Specification 3.4.1, Recirculation Loops Operating, requires operation within the normal region of the Power to Flow map specified in the COLR. The Power to Flow map in the COLR limits reactor power based on core flow.

The cavitation protection interlock remains the same in terms of absolute flow rates. This interlock is based on subcooling in the external recirculation loop and thus is a function of absolute FW flow rate and FW temperature at less than full thermal power operating conditions. Therefore, the interlock is not changed by EPU.

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[illegible]	[illegible]	[illegible]
		]] This is unchanged with EPU.

[[

]] This was confirmed for Monticello.

The Monticello recirculation loop jet pump flow mismatch Technical Specification limits do not change because these limits are based on rated core flow, which is not affected by EPU. As stated in the CLTR, the flow mismatch limits must be reviewed only if a detailed ECCS evaluation was required. The flow mismatch limits are not affected because a detailed ECCS evaluation was not required (see Section 2.8.5.6.2).

SLO is limited to offrated conditions and is not affected by EPU.

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2.8.5 Accident and Transient Analyses

Fuel Thermal Margin Events

The Anticipated Operational Occurrence (AOO) events that determine the operating limit MCPR do not change significantly due to an increase in reactor power up to 20% above the OLTP. This characteristic was established by the initial and reload core analyses for different power level and power density plants and confirmed by the results from subsequent power uprate evaluations. These limiting events are defined in ELTR1. Other events listed in Table E-1 of ELTR1 do not establish the operating limit MCPR and do not have to be analyzed to establish this limit.

The operating limit MCPR is not significantly affected by EPU. Table 3-1 of ELTR1 shows an effect on the operating limit MCPR of less than 0.03 for a 20% power uprate with an increase in pressure. This small effect is due to the small changes in transient void and scram reactivity response and the flatter radial power distribution at the EPU RTP. GEH BWR experience to date for power uprates up to 120% of OLTP confirms this assessment with changes in the operating limit MCPR of +0.018 to –0.013. Therefore, the 0.03 increase in operating limit MCPR documented in ELTR1 is expected to bound the Monticello EPU.

The results of the limiting thermal margin event analyses are dependent upon the reference core loading pattern and will, therefore, be analyzed for actual reload core. [[

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[illegible]	[illegible]	[illegible]
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Acceptable fuel rod thermal-mechanical performance for both UO_2 and GdO_2 fuel rods was demonstrated. Results for all AOO pressurization transient events analyzed, including equipment out of service, showed at least 10 percent margin to the fuel centerline melt and the one percent cladding circumferential plastic strain acceptance criteria. The minimum calculated margin to the fuel centerline melt criterion was 26%. The minimum calculated margin to the cladding strain criterion was 35%. Fuel rod thermal-mechanical performance will be evaluated as part of the RLA performed for the cycle specific core. Documentation of acceptable fuel rod thermal-mechanical response will be included in the SRLR or COLR.

Power and Flow Dependent Limits

The operating limit MCPR, LHGR, and/or MAPLHGR thermal limits are modified by a flow factor when the plant is operating at less than 100% core flow. This flow factor is primarily based upon an evaluation of the slow recirculation increase event. The current Monticello analysis is based upon a conservative flow runup rod line that bounds operation to the rod line documented in Section 1.2. Therefore, these flow-dependent limits do not change due to steady state operation at EPU RTP.

Similarly, the thermal limits are modified by a power factor when the plant is operating at less than 100% power. This power factor was generically developed for all plants and is referenced to the power level used in the plant reload transient analysis. The change in this factor at different percent power levels remains the same at EPU RTP as at CLTP.

In addition, the operating thermal limits of less than 100% power and flow are confirmed as part of the reload process because they are evaluated for the actual EPU reload core.

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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's acceptance criteria are based on (1) GDC-10, insofar as it requires that the RCS be designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations including AOOs; (2) GDC-15, insofar as it requires that the RCS and its associated auxiliary systems be designed with margin sufficient to ensure that the design condition of the RCPB are not exceeded during any condition of normal operation; (3) GDC-20, insofar as it requires that the reactor protection system be designed to initiate automatically the operation of appropriate systems, including the reactivity control systems, to ensure that SAFDLs are not exceeded during any condition of normal operation, including AOOs; and (4) GDC-26, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operation, including AOOs, SAFDLs are not exceeded.

Specific NRC review criteria are contained in SRP Section 15.1.1-4 and other guidance provided in Matrix 8 of RS-001.

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC listed in the Regulatory Evaluation above, with the exception of current GDC-15, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-6, draft GDC-14, draft GDC-15, draft GDC-27, draft GDC-29, and draft GDC-30. There is no draft GDC directly associated with current GDC-15. Current GDC-10, current GDC-15, current GDC-20, and current GDC-26 are applicable to Monticello as described in USAR Section 14.4.

The analysis of a loss of feedwater heating transient is described in Monticello USAR Section 14.4.2, "Loss of Feedwater Heating." The analysis of a feedwater controller failure with maximum demand is described in Monticello USAR Section 14.4.4, "Feedwater Controller Failure - Maximum Demand."

__Technical Evaluation__

Section 9 of the CLTR and the response to CLTR NRC RAI Set 9 Number 14 RXSB (contained in pages A-77 through A-85 of the CLTR) provide the disposition of the AOOs for EPU. The NRC staff in the SE for NEDC-33004P-A accepted this disposition. The following is a summary of the evaluation provided in the above-named sources for the excessive heat removal events:

The Decrease in Feedwater Temperature limiting events ([[ ]]) and the Increase in Feedwater Flow limiting event ([[ ]])

are confirmed to be within the Monticello reload evaluation scope. [[

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The Increase in Steam Flow event ([[ ]]) is [[ ]] within the reload evaluation scope. [[

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The Inadvertent Opening of a Safety Relief Valve event is [[

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Conclusion

NSPM has completed a generic assessment and found Monticello is consistent with the approach described in the CLTR. NSPM will perform plant specific reload analyses to confirm that fuel design limits and RCPB pressure limits will not be exceeded under EPU conditions. The CLTR requires that approved analytical methods be used for the EPU core reload analysis. Based on this, the plant will continue to meet the requirements of the current licensing basis following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to a decrease in reactor water temperature event.

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 unplanned decreases in heat removal by the secondary system. 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's acceptance criteria are based on (1) GDC-10, insofar as it requires that the RCS be designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations, including AOOs; (2) GDC-15, insofar as it requires that the RCS and its associated auxiliary systems be designed with margin sufficient to ensure that the design condition of the RCPB are not exceeded during any condition of normal operation; and (3) GDC-26, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operation, including AOOs, SAFDLs are not exceeded.

Specific NRC review criteria are contained in SRP Section 15.2.1-5 and other guidance provided in Matrix 8 of RS-001.

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC listed in the Regulatory Evaluation above, with the exception of current GDC-15, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-6, draft GDC-27, draft GDC-29, and draft GDC-30. There is no draft GDC directly associated with current GDC-15. Current GDC-10, current GDC-15, current GDC-20, and current GDC-26 are applicable to Monticello as described in USAR Section 14.4.

The analysis of a generator load rejection is described in Monticello USAR Section 14.4.1, "Generator Load Rejection Without Bypass." The analysis of a turbine trip without bypass is described in Monticello USAR Section 14.4.5, "Turbine Trip Without Bypass." The analysis of an MSIV closure event is described in Monticello USAR Section 14.5.1, "Vessel Pressure ASME Code Compliance Model - MSIV Closure."

Technical Evaluation

Section 9 of the CLTR and the response to CLTR NRC RAI Set 9 Number 14 RXSB (contained in pages A-77 through A-85 of the CLTR) provide the disposition of the AOOs for EPU. The

NRC staff in the SE for NEDC-33004P-A accepted this disposition. The following is a summary of the evaluation provided in the above-named sources for the decreased heat removal events:

The Loss of External Load limiting event (Generator Load Rejection with Steam Bypass Failure (LRNBP)) and the Turbine Trip limiting event (Turbine Trip with Steam Bypass Failure (TTNBP)) are confirmed to be within the Monticello reload evaluation scope. [[

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For all BWRs, the Loss of Condenser Vacuum (LOCV) event is [[

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The Closure of Main Steam Isolation Valves with Failure of Direct Scram (MSIVF) event is evaluated in Section 2.8.4.2. This event is also confirmed as evaluated during the Monticello reload evaluation scope.

The Pressure Regulator Failure Closed (Pressure Regulator Failure Downscale (PRFD)) event is [[ ]] (Please see Table 3, item 4 of the response to NRC RAI Set 9 Number 14 RSXB contained in the CLTR.)

Conclusion

NSPM has completed a generic assessment and found Monticello is consistent with the approach described in the CLTR. NSPM will perform plant specific reload analyses to confirm that fuel design limits and RCPB pressure limits will not be exceeded under EPU conditions. Based on this, the plant will continue to meet the requirements of the current licensing basis following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to an increase in reactor pressure event.

2.8.5.2.2 Loss of Non-emergency AC Power to the Station Auxiliaries

Regulatory Evaluation

The loss of non-emergency AC power is assumed to result in the loss of all power to the station auxiliaries and the simultaneous tripping of all reactor coolant circulation pumps. This causes a flow 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's acceptance criteria are based on (1) GDC-10, insofar as it requires that the RCS be designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations, including AOOs; (2) GDC-15, insofar as it requires that the RCS and its associated auxiliary systems be designed with margin sufficient to ensure that the design condition of the RCPB are not exceeded during any condition of normal operation; and (3) GDC-26, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operation, including AOOs, SAFDLs are not exceeded.

Specific NRC review criteria are contained in SRP Section 15.2.6 and other guidance provided in Matrix 8 of RS-001.

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC listed in the Regulatory Evaluation above, with the exception of current GDC-15, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-6, draft GDC-27, draft GDC-29, and draft GDC-30. There is no draft GDC directly associated with current GDC-15. Current GDC-10, current GDC-15, and current GDC-26 are applicable to Monticello as described in USAR Section 14.4.

The Monticello USAR does not include an analysis for loss of non emergency AC power to the station auxiliaries.

Technical Evaluation

Section 9 of the CLTR and the response to CLTR NRC RAI Set 9 Number 14 RXSB (contained in pages A-77 through A-85 of the CLTR) provide the disposition of the AOOs for EPU. The NRC staff in the SE for NEDC-33004P-A accepted this disposition. The following is a summary of the evaluation provided in the above-named sources for the loss of non-emergency AC power to the Station Auxiliaries event:

The Loss of Non-Emergency AC Power to the Station Auxiliaries event is [[

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Conclusion

NSPM has completed a generic assessment and found Monticello is consistent with the approach described in the CLTR. NSPM will perform plant specific reload analyses to confirm that fuel design limits and RCPB pressure limits will not be exceeded under EPU conditions. Based on this, the plant will continue to meet the requirements of the current licensing basis following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to a loss of non emergency AC power to station auxiliaries event.

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 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 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's acceptance criteria are based on (1) GDC-10, insofar as it requires that the RCS be designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations, including AOOs; (2) GDC-15, insofar as it requires that the RCS and its associated auxiliary systems be designed with margin sufficient to ensure that the design condition of the RCPB are not exceeded during any condition of normal operation; and (3) GDC-26, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operation, including AOOs, SAFDLs are not exceeded.

Specific NRC review criteria are contained in SRP Section 15.2.7 and other guidance provided in Matrix 8 of RS-001.

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern

States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC listed in the Regulatory Evaluation above, with the exception of current GDC-15, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-6, draft GDC-27, draft GDC-29, and draft GDC-30. There is no draft GDC directly associated with current GDC-15.

The analysis of the loss of normal feedwater flow transient is described in Monticello USAR Sections 10.2.5, Reactor Core Isolation Cooling System (RCIC) and 14.10.1, "Adequate Core Cooling for Transients with a Single Failure."

Technical Evaluation

For the LOFW event, adequate transient core cooling is provided by maintaining the water level inside the core shroud above the top of active fuel. A plant specific analysis was performed for Monticello at EPU conditions. The ANS/ANSI 5.1 model was used to calculate the decay heat value used in the analysis as was used in the current USAR analysis, with the addition of a 2σ uncertainty for additional conservatism (see Sections 2.6.1.1 and 2.6.5). The LOFW event analysis assumed failure of the HPCI system and used only the RCIC system to restore the reactor water level. Because of the extra decay heat from EPU, slightly more time is required for the automatic systems to restore water level. Operator action is only needed for long-term plant shutdown. The results of the LOFW analysis for Monticello show that the minimum water level inside the shroud is 77 inches above the top of active fuel at EPU conditions. After the water level is restored, the operator manually controls the water level, reduces reactor pressure, and initiates RHR shutdown cooling. This sequence of events does not require any new operator actions or shorter operator response times. Therefore, the operator actions for an LOFW transient do not significantly change for EPU.

The results of the Monticello LOFW analysis also showed that the RCIC system restores the reactor water level while avoiding the level at which the Emergency Operating Procedures (EOPs) require the operator to initiate ADS. Therefore, unnecessary initiation of ADS is avoided.

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The loss of one FW pump event only addresses operational considerations to avoid reactor scram on low reactor water level (Level 3). This requirement is intended to avoid unnecessary reactor shutdowns. Because the MELLLA region is extended along the existing upper boundary to the EPU RTP, there is no increase in the highest flow control line for the Monticello EPU. [[

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Conclusion

NSPM has evaluated the loss of normal feedwater flow event and accounted for operation of the plant at the proposed power level using acceptable analytical models. NSPM has found Monticello is consistent with the approach described in the CLTR. NSPM will perform plant specific reload analyses to confirm that fuel design limits and RCPB pressure limits will not be exceeded under EPU conditions. Based on this, the plant will continue to meet the requirements of the current licensing basis following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to a 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's acceptance criteria are based on (1) GDC-10, insofar as it requires that the RCS be designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations, including AOOs; (2) GDC-15, insofar as it requires that the RCS and its associated auxiliary systems be designed with margin sufficient to ensure that the design condition of the

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

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

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC listed in the Regulatory Evaluation above, with the exception of current GDC-15, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-6, draft GDC-27, draft GDC-29, and draft GDC-30. There is no draft GDC directly associated with current GDC-15. Current GDC-10, current GDC-15, and current GDC-26 are applicable to Monticello as described in USAR Section 14.4.

Technical Evaluation

Section 9 of the CLTR and the response to CLTR NRC RAI Set 9 Number 14 RXSB (contained in pages A-77 through A-85 of the CLTR) provide the disposition of the AOOs for EPU. The NRC staff in the SE for NEDC-33004P-A accepted this disposition. The following is a summary of the evaluation provided in the above-named sources for the Loss of Forced Reactor Coolant Flow event:

The Loss of Forced Reactor Coolant Flow Including Trip of Pump Motor and Flow Controller Malfunctions events are events that result in a decrease in reactor core coolant flow rate. Events in this category are [[

]] (Please see Table 2 of the response to NRC RAI Set 9 Number 14 RSXB contained in the CLTR.)

Conclusion

NSPM has completed a generic assessment and found Monticello is consistent with the approach described in the CLTR. NSPM will perform plant specific reload analyses to confirm that fuel design limits and RCPB pressure limits will not be exceeded under EPU conditions. Based on this, the plant will continue to meet the requirements of the current licensing basis following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to a loss of forced reactor coolant flow event.

2.8.5.3.2 Reactor Recirculation Pump Rotor Seizure and Reactor Recirculation Pump Shaft Break

Regulatory Evaluation

The events postulated are an instantaneous seizure of the rotor or break of the shaft of a reactor 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 case, reactor protection and safety systems are actuated to mitigate the transient.

The NRC's acceptance criteria are based on (1) GDC-27, insofar as it requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to assure the capability to cool the core is maintained; (2) GDC-28, insofar as it requires that the reactivity control systems be designed to assure that the effects of postulated reactivity accidents can neither result in damage to the RCPB greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals so as to significantly impair the capability to cool the core; and (3) GDC-31, insofar as it requires that the RCPB be designed with 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.

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

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The

Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC listed in the Regulatory Evaluation above, with the exception of current GDC-27, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-32, draft GDC-33, draft GDC-34, and draft GDC-35. There is no draft GDC directly associated with current GDC-27. Current GDC-10, current GDC-15, and current GDC-26 are applicable to Monticello as described in USAR Section 14.4.

The analysis of a one recirculation pump seizure accident is described in Monticello USAR Section 14.7.5, "One Recirculation Pump Seizure Accident Analysis."

Technical Evaluation

Section 9 of the CLTR and the response to CLTR NRC RAI Set 9 Number 14 RXSB (contained in pages A-77 through A-85 of the CLTR) provide the disposition of the AOOs for EPU. The NRC staff in the SE for NEDC-33004P-A accepted this disposition. The following is a summary of the evaluation provided in the above-named sources for the Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break events:

The Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break events are events that result in a decrease in reactor core coolant flow rate. Events in this category, [[

]] (Please see Table 2 of the response to NRC RAI Set 9 Number 14 RSXB contained in the CLTR.)

Conclusion

NSPM has completed a generic assessment and found Monticello is consistent with the approach described in the CLTR. NSPM will perform plant specific reload analyses to confirm that fuel design limits and RCPB pressure limits will not be exceeded under EPU conditions. Based on this, the plant will continue to meet the requirements of the current licensing basis following implementation of the proposed EPU. Therefore, the proposed is EPU acceptable with respect to a sudden recirculation pump rotor seizure and reactor recirculation pump shaft break event.

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's acceptance criteria are based on (1) GDC-10, insofar as it requires that the RCS be designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations, including AOOs; (2) GDC-20, insofar as it requires that the reactor protection system be designed to initiate automatically the operation of appropriate systems, including the reactivity control systems, to ensure that SAFDLs are not exceeded as a result of AOOs; and (3) GDC-25, insofar as it requires that the protection system be designed to assure that SAFDLs are not exceeded for any single malfunction of the reactivity control systems.

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

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC listed in the Regulatory Evaluation above, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-6, draft GDC-14, draft GDC-15, and draft GDC-31. Current GDC-10, current GDC-20, and current GDC-25 are applicable to Monticello as described in USAR Section 14.4.

The analysis of a rod withdrawal error transient is described in Monticello USAR Section 14.4.3, "Rod Withdrawal Error."

Technical Evaluation

The evaluation of the Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Conditions event for Monticello is a comparison of the expected maximum increase in peak fuel enthalpy for a 20% EPU with the acceptance criterion of 170 cal/gram. The CLTP RWE analysis for Monticello is based on Reference 30. The Monticello EPU core consists only of GEH fuel assemblies and the EPU is limited to 120% of OLTP. There is also no change to the Monticello reactor manual control system or control rod hydraulic control units for EPU. [[

]] No change in peak fuel enthalpy is expected due to EPU because an RWE is a localized low-power event. If the peak fuel rod enthalpy is conservatively increased by a factor of 1.2, the RWE peak fuel enthalpy at EPU will be 72 cal/gram. This enthalpy is well below the acceptance criterion of 170 cal/gram.

Conclusion

NSPM has evaluated the uncontrolled control rod assembly withdrawal from a subcritical or low power startup condition and accounted for the core design changes necessary for operation of the plant at the proposed power level. NSPM has completed a generic assessment and found Monticello is consistent with the approach described in the CLTR. NSPM will perform plant specific reload analyses to confirm that fuel design limits and RCPB pressure limits will not be exceeded under EPU conditions. Based on this, the plant will continue to meet the requirements of the current licensing basis following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to an uncontrolled control rod assembly withdrawal from a subcritical or low power startup condition event.

2.8.5.4.2 Uncontrolled Control Rod Assembly Withdrawal at Power

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's acceptance criteria are based on (1) GDC-10, insofar as it requires that the RCS be designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations, including AOOs; (2) GDC-20, insofar as it requires that the reactor protection system be designed to initiate automatically the operation of appropriate systems, including the reactivity control systems, to ensure that SAFDLs are not exceeded as a result of AOOs; and (3) GDC-25, insofar as it requires that the protection system be designed to assure that SAFDLs are not exceeded for any single malfunction of the reactivity control systems.

Specific NRC review criteria are contained in SRP Section 15.4.2 and other guidance provided in Matrix 8 of RS-001.

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC listed in the Regulatory Evaluation above, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-6, draft GDC-14, draft GDC-15, and draft GDC-31. Current GDC-10, current GDC-20, and current GDC-25 are applicable to Monticello as described in USAR Section 14.4.

The analysis of a rod withdrawal error transient is described in Monticello USAR Section 14.4.3, "Rod Withdrawal Error."

Technical Evaluation

Section 9 of the CLTR and the response to CLTR NRC RAI Set 9 Number 14 RXSB (contained in pages A-77 through A-85 of the CLTR) provide the disposition of the AOOs for EPU. The NRC staff in the SE for NEDC-33004P-A accepted this disposition. The following is a summary of the evaluation provided in the above-named sources for the uncontrolled control rod assembly withdrawal at power event:

Control Rod Withdrawal Error at Power is confirmed to be within the Monticello reload evaluation scope. [[

]]

Conclusion

NSPM has completed a generic assessment and found Monticello is consistent with the approach described in the CLTR. NSPM will perform plant specific reload analyses to confirm that fuel design limits and RCPB pressure limits will not be exceeded under EPU conditions. Based on this, the plant will continue to meet the requirements of the current licensing basis following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to a continuous rod withdrawal during power range operation event.

2.8.5.4.3 Startup of a Recirculation Loop at an Incorrect Temperature and Flow Controller Malfunction Causing an Increase in Core Flow Rate

Regulatory Evaluation

A startup of an inactive loop transient 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's acceptance criteria are based on (1) GDC-10, insofar as it requires that the RCS be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs; (2) GDC-20, insofar as it requires that the protection system be designed to initiate automatically the operation of appropriate systems to ensure that SAFDLs are not exceeded as a result of operational occurrences; (3) GDC-15, insofar as it requires that the RCS and its associated auxiliary systems be designed with margin sufficient to ensure that the design condition of the RCPB are not exceeded during AOOs; (4) GDC-28, insofar as it requires that the reactivity control systems be designed to assure that the effects of postulated reactivity accidents can neither result in damage to the RCPB greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals so as to significantly impair the capability to cool the core; and (5) GDC-26, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operation, including AOOs, SAFDLs are not exceeded.

Specific NRC review criteria are contained in SRP Section 15.4.4-5 and other guidance provided in Matrix 8 of RS-001.

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC listed in the Regulatory Evaluation above, with the exception of current GDC-15, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-6, draft GDC-14, draft GDC-15, draft GDC-27, draft GDC-29, draft GDC-30, and draft GDC-32. There is no draft GDC directly applicable to the current GDC-15. Current GDC-10, current GDC-15, current GDC-20, and current GDC-26 are applicable to Monticello as described in USAR Section 14.4.

The Monticello USAR does not include an analysis of the startup of a recirculation loop at an incorrect temperature or a flow controller malfunction causing an increase in core flow rate.

Technical Evaluation

Section 9 of the CLTR and the response to CLTR NRC RAI Set 9 Number 14 RXSB (contained in pages A-77 through A-85 of the CLTR) provide the disposition of the AOOs for EPU. The NRC staff in the SE for NEDC-33004P-A accepted this disposition. The following is a summary of the evaluation provided in the above-named sources for the Startup of an Idle Recirculation Pump and the Failure of the Recirculation Flow Controller events:

The Failure of the Recirculation Flow Controller can result in either a slow or fast recirculation increase. The disposition of these events for EPU indicates that [[

]] (Please see Table 3, items 8 and 9 of the response to NRC RAI Set 9 Number 14 RSXB contained in CLTR.) [[

]]

The Startup of an Idle Recirculation Pump event is [[
]]

Conclusion

NSPM has completed a generic assessment and found Monticello is consistent with the approach described in the CLTR. NSPM will perform plant specific reload analyses to confirm that fuel design limits and RCPB pressure limits will not be exceeded under EPU conditions. Based on this, the plant will continue to meet the requirements of the current licensing basis following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to startup of an idle recirculation pump or recirculation flow controller failure event.

2.8.5.4.4 Spectrum of Rod Drop Accidents

Regulatory Evaluation

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

Specific NRC review criteria are contained in SRP Section 15.4.9 and other guidance provided in Matrix 8 of RS-001.

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC listed in the Regulatory Evaluation above, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-32.

The analysis of a control rod drop accident is described in Monticello USAR Section 14.7.1, "Control Rod Drop Accident Evaluation."

Technical Evaluation

The spectrum of CRDAs does not change with EPU. The evaluation of a CRDA for the Monticello EPU is a comparison of the expected maximum increase in peak fuel enthalpy with 20% EPU with the acceptance criterion of 280 cal/gram. The CLTP CRDA for Monticello is based on Reference 30. The Monticello EPU core consists only of GEH fuel assemblies and the EPU is limited to 120% of OLTP. Control Rod Sequencing at Monticello for CLTP and EPU follows the BPWS. There is also no change to the Monticello reactor manual control system or control rod hydraulic control units for EPU. [[

]] No change in peak fuel enthalpy is expected due to EPU because EPU by itself, does not increase peak fuel enthalpy for this localized low-power event. However, indirectly, EPU fuel and core designs can lead to higher rod worth and therefore higher peak fuel enthalpy at low power. If the peak fuel rod enthalpy is conservatively increased by a factor of 1.2, the CRDA peak fuel enthalpy at EPU will be 162 cal/gram. This enthalpy is well below the acceptance criterion of 280 cal/gram.

Conclusion

NSPM has evaluated the CRDA and accounted for operation of the plant at the proposed power level. NSPM has found Monticello is consistent with the approach described in the CLTR. NSPM will perform plant specific reload analyses to confirm that fuel design limits and RCPB pressure limits will not be exceeded under EPU conditions. Based on this, the plant will continue to meet the requirements of the current licensing basis following implementation of the EPU. Therefore, the proposed EPU is acceptable with respect to a CRDA.

2.8.5.5 Inadvertent Operation of ECCS or Malfunction that Increases Reactor Coolant Inventory

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's acceptance criteria are based on (1) GDC-10, insofar as it requires that the RCS be designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations, including AOOs; (2) GDC-15, insofar as it requires that the RCS and its associated auxiliary systems be designed with margin sufficient to ensure that the design conditions of the RCPB are not exceeded during AOOs; and (3) GDC-26, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operation, including AOOs, SAFDLs are not exceeded.

Specific NRC review criteria are contained in SRP Section 15.5.1-2 and other guidance provided in Matrix 8 of RS-001.

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed

General Design Criteria can usually be made. For the current GDC listed in the Regulatory Evaluation above, with the exception of current GDC-15, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-6, draft GDC-27, draft GDC-29, and draft GDC-30. There is no draft GDC directly applicable to the current GDC-15. Current GDC-10, current GDC-15, and current GDC-26 are applicable to Monticello as described in USAR Section 14.4.

The Monticello USAR does not include an analysis of an inadvertent operation of ECCS or a malfunction that increases reactor coolant inventory.

Technical Evaluation

Section 9 of the CLTR and the response to CLTR NRC RAI Set 9 Number 14 RXSB (contained in pages A-77 through A-85 of the CLTR) provide the disposition of the AOOs for EPU. The NRC staff in the SE for NEDC-33004P-A accepted this disposition. The following is a summary of the evaluation provided in the above-named sources for the Inadvertent Operation of ECCS or Malfunction that Increases Reactor Coolant Inventory events:

In the Inadvertent Operation of ECCS or Malfunction that Increases Reactor Coolant Inventory category, the limiting event, [[ ]] is confirmed to be within the Monticello reload evaluation scope. [[

]]

Conclusion

NSPM has completed a generic assessment and found Monticello is consistent with the approach described in the CLTR. NSPM will perform plant specific reload analyses to confirm that fuel design limits and RCPB pressure limits will not be exceeded under EPU conditions. Based on this, the plant will continue to meet the requirements of the current licensing basis following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to an inadvertent operation of the ECCS or a 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 (TCVs) to stabilize the reactor at a lower pressure. The reactor power settles out at nearly the initial power level. The feedwater control system maintains the coolant inventory using water from the condensate storage tank via the condenser hotwell.

The NRC's acceptance criteria are based on (1) GDC-10, insofar as it requires that the RCS be designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations, including AOOs; (2) GDC-15, insofar as it requires that the RCS and its associated auxiliary systems be designed with margin sufficient to ensure that the design conditions of the RCPB are not exceeded during AOOs; and (3) GDC-26, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operation, including AOOs, SAFDLs are not exceeded.

Specific NRC review criteria are contained in SRP Section 15.6.1 and other guidance provided in Matrix 8 of RS-001.

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC listed in the Regulatory

Evaluation above, with the exception of current GDC-15, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-6, draft GDC-27, draft GDC-29, and draft GDC-30. There is no draft GDC directly applicable to the current GDC-15. Current GDC-10, current GDC-15, and current GDC-26 are applicable to Monticello as described in USAR Section 14.4.

The Monticello USAR does not include an analysis of an inadvertent opening of a pressure relief valve.

Technical Evaluation

Section 9 of the CLTR and the response to CLTR NRC RAI Set 9 Number 14 RXSB (contained in pages A-77 through A-85 of the CLTR) provide the disposition of the AOOs for EPU. The NRC staff in the SE for NEDC-33004P-A accepted this disposition. The following is a summary of the evaluation provided in the above-named sources for the Inadvertent Opening of a Safety Valve event:

The Inadvertent Opening of a Safety Valve event is [[

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Conclusion

NSPM has completed a generic assessment and found Monticello is consistent with the approach described in the CLTR. NSPM will perform plant specific reload analyses to confirm that fuel design limits and RCPB pressure limits will not be exceeded under EPU conditions. Based on this, the plant will continue to meet the requirements of the current licensing basis following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to an inadvertent opening of a pressure relief valve event.

2.8.5.6.2 Emergency Core Cooling System and Loss-of-Coolant Accidents

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's acceptance criteria are based on (1) 10 CFR 50.46, insofar as it establishes standards for the calculation of ECCS performance and acceptance criteria for that calculated performance; (2) 10 CFR Part 50, Appendix K, insofar as it establishes required and acceptable features of evaluation models for heat removal by the ECCS after the blowdown phase of a LOCA; (3) GDC-4, insofar as it requires that SSCs important to safety be protected against dynamic effects associated with flow instabilities and loads such as those resulting from water hammer; (4)

GDC-27, insofar as it requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to assure the capability to cool the core is maintained; and (5) GDC-35, insofar as it requires that a system to provide abundant emergency core cooling be provided to transfer heat from the reactor core following any LOCA at a rate so that fuel clad damage that could interfere with continued effective core cooling will be prevented.

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

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC listed in the Regulatory Evaluation above, with the exception of current GDC-27, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-37, draft GDC-40, draft GDC-41, draft GDC-42, and draft GDC-44. There is no draft GDC directly applicable to current GDC-27.

The analysis of a loss-of-coolant accident is described in Monticello USAR Section 14.7.2, "Loss-of-Coolant Accident."

In addition to the evaluations described in the Monticello USAR, Monticello's systems and components were evaluated for License Renewal. Systems and component materials materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Monticello Nuclear Generating Plant License Renewal Safety Evaluation Report (SER), NUREG-1865, dated October 2006 (Reference 5).

The license renewal evaluations associated with HPCI, Core Spray, RHR, and ADS are located in NUREG-1865 sections 2.3.2.4 and 3.2.2.3.4, 2.3.2.3 and 3.2.2.3.3, 2.3.2.7 and 3.2.2.3.7, and 2.3.2.1 and 3.2.2.3.1, respectively.

Technical Evaluation

The Monticello EPU LOCA analyses are based on NRC-approved GEH LOCA analysis methods and are in full compliance with 10 CFR 50.46. No new fuel designs are being introduced. No ECCS changes are required to meet LOCA analysis acceptance criteria. The radiological consequences are evaluated in Section 2.9.2.

Each ECCS is discussed in the following subsections. The effect on the functional capability of each system, due to EPU is addressed. The assumption of constant pressure minimizes the effect of EPU for ECCS evaluation.

High Pressure Coolant Injection

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[illegible]	[illegible]	[illegible]
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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 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 adequacy of the HPCI system is demonstrated in the ECCS performance discussion at the end of this section.

For EPU, there is no change to the maximum nominal reactor operating pressure and the SRV analytical limits remain the same. [[

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Because the maximum normal operating pressure and the SRV setpoints do not change for EPU, the HPCI system performance requirements do not change. [[
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Core Spray

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The Core Spray (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 CS system meets all applicable safety criteria for the EPU.

The slight change in the system operating condition due to EPU for a postulated LOCA does not affect the hardware capabilities of the CS system. The generic core spray distribution assessment provided in ELTR2, Section 3.3, continues to be valid for the EPU as described in CLTR Section 4.2.3.

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Low Pressure Coolant Injection

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The Low Pressure Coolant Injection (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.

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Automatic Depressurization System

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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. This allows the CS and LPCI systems to inject coolant into the reactor vessel. The adequacy of the ADS is demonstrated by the margins discussed below. The ADS initiation logic and valve control is not affected by EPU conditions.

The EPU does not change the conditions at which the ADS must function.

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Emergency Core Cooling System Performance

The Monticello ECCS is designed to provide protection against postulated LOCAs caused by ruptures in the primary system piping. The ECCS performance characteristics are not changed for EPU. ECCS-LOCA performance analyses demonstrate that the 10 CFR 50.46 requirements continue to be met at the EPU RTP conditions.

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[illegible]	[illegible]	[illegible]
		]]

The basic break spectrum response is not affected by EPU. There are two limiting points on the break spectrum; the full sized RSLB, and the worst small break with failure of the high pressure ECCS. The break spectrum response is determined by the ECCS network design and is common to all BWRs. GEH BWR power uprate evaluation experience shows that the basic break spectrum response is not affected by changes in core power. The PCT for the limiting large break LOCA is determined primarily by the hot bundle power, which is unchanged with EPU. In the analysis, the hot bundle is assumed to be operating at the thermal limits (MCPR, MAPLHGR, and LHGR); these limits are not changed for EPU. GEH BWR experience has shown that a power uprate with no pressure increase has only a small effect on the Licensing Basis PCT (typically less than 20°F). Because EPU has only a small effect on PCT, there is a negligible effect on compliance with the other acceptance criteria of 10 CFR 50.46 (local cladding oxidation, core-wide metal-water reaction, coolable geometry and long-term cooling). The local fuel conditions are not significantly changed with EPU, because the hot bundle operation is still constrained by the same operating thermal limits. EPU affects the relative flow distribution between the hot and average channel. As the average channel power increases with EPU, the fraction of the flow passing through the hot channel increases. The increased flow keeps the cladding temperature from increasing with EPU. Because an EPU has such a small effect on the PCT, the system response over the large break spectrum is not affected.

The effects of EPU on the large and small break PCT are evaluated on a plant specific basis. The generic disposition of the local oxidation and core wide metal water reaction are confirmed by the small change in the plant Licensing Basis PCT due to EPU. Coolable geometry and long-term cooling have been dispositioned generically for BWRs. These generic dispositions are not affected by EPU.

The Licensing Basis PCT is based on the most limiting Appendix K case plus a plant variable uncertainty term that accounts statistically for the uncertainty in parameters that are not specifically addressed by 10 CFR 50 Appendix K. The Appendix K results confirm that the limiting break is the recirculation suction line DBA and that the LPCI Injection valve (LPCIIV) failure is the limiting single failure. [[

]] For both EPU and CLTP, the GE14 Licensing Basis PCT is 2140°F and is based on the operating conditions at CLTP power and MELLLA core flow. The results of these analyses are provided in Table 2.8-5.

In addition to the large break LOCA analysis, the small break LOCA response was reviewed in order to assure adequate ADS capacity. The increased decay heat associated with EPU results in a longer ADS blowdown and a higher PCT for the small break LOCA. 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. Also, the plant performance improvement of three SRVs OOS remains valid with EPU.

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For SLO, a multiplier is applied to the Two-Loop LHGR and MAPLHGR Operation limits. The operating conditions for SLO are not changed with EPU; therefore, the current SLO analysis remains acceptable for EPU.

ARTS limits are unaffected by EPU. Also, the effect of ICF on PCT is negligible with EPU. Thus the ARTS limits, as well as the ICF domain, remain valid with EPU.

Conclusion

NSPM has evaluated the LOCA events and the ECCS. The evaluation concludes that operation of the plant at the proposed power level is acceptable In addition, NSPM will perform cycle specific reload analyses to confirm that the peak cladding temperature, total oxidation of the cladding, total hydrogen generation, and changes in core geometry and long-term cooling will remain within acceptable limits. Based on this, the evaluation concludes that the plant will continue to meet the requirements of the current licensing basis and 10 CFR 50.46 following implementation of the proposed EPU, and is, therefore, acceptable.

2.8.5.7 Anticipated Transients Without Scrams

Regulatory Evaluation

ATWS is defined as an AOO followed by the failure of the reactor portion of the protection system specified in GDC-20. The regulation at 10 CFR 50.62 requires that:

- each BWR have an 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 standby liquid control system (SLCS) 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 weight-percent 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.

Review NRC guidance is provided in Matrix 8 of RS-001.

Monticello Current Licensing Basis

The analysis of anticipated transients without scram is described in Monticello USAR Section 14.8, "Anticipated Transients without Scram (ATWS)."

Technical Evaluation

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[redacted]	[redacted]	[redacted]

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The overpressure evaluation includes consideration of the most limiting RPV overpressure case. Previous evaluations considered four ATWS events. Based on experience and the CLTR generic analyses, only two of these cases need to be performed: (1) MSIVC; and (2) PRFO. As shown in Section 3.7 of ELTR2, the vessel overpressure response for these two cases bounds the IORV and the LOOP cases.

For Monticello, the LOOP event results in a reduction in the RHR pool cooling capability relative to the MSIVC and PRFO cases. Due to the reduction in the RHR pool cooling capability for the LOOP, the containment response is evaluated for the MSIVC, PRFO, and LOOP cases. The IORV is not limiting for any plant and is not evaluated. [[

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Monticello meets the ATWS mitigation requirements as follows:

- Installation of an Alternate Rod Insertion (ARI) system;
- Boron injection equivalent to 86 gpm; and
- Installation of automatic Recirculation Pump Trip (RPT) logic (i.e., ATWS-RPT).

In addition, plant specific ATWS analysis is performed to ensure that the following ATWS acceptance criteria are met:

- Peak vessel bottom pressure less than ASME Service Level C limit of 1500 psig;
- Peak suppression pool temperature less than 281°F (Wetwell shell design temperature); and

- Peak containment pressure less than 56 psig (Drywell design pressure).

The limiting events for the acceptance criteria discussed above are the PRFO, MSIVC, and LOOP events. The sequence of events for each of the three events are provided in Tables 2.8-6, 2.8-7, and 2.8-8.

A plant specific ATWS analysis was performed for CLTP and for EPU RTP to demonstrate the effect of the EPU on the ATWS acceptance criteria. There are no changes to the assumed operator actions for the EPU ATWS analysis and there is no change to the required hot shutdown boron weight. The key inputs to the ATWS analysis are provided in Table 2.8-9. The results of the analysis are provided in Table 2.8-10.

The results of the ATWS analysis meet the above ATWS acceptance criteria. Therefore, the Monticello response to an ATWS event at EPU is acceptable.

Coolable core geometry is assured by meeting the 2200°F PCT and the 17% local cladding oxidation acceptance criteria of 10 CFR 50.46. Previous ATWS analyses have demonstrated that there is significant margin to the acceptance criteria of 10 CFR 50.46. The calculated PCTs for ATWS events using the methodology described in Section 3.7 of ELTR2 have been consistently less than 1500°F. EPU has a negligible effect on the PCT or local cladding oxidation. The local fuel conditions are not changed with EPU, because the hot bundle operation is still constrained by the same operating thermal limits. EPU affects the relative flow distribution between the hot and average channel. Because the average channel power increases with EPU, the fraction of the flow passing through the hot channel increases. The increased flow keeps the cladding temperature from increasing with EPU. [[ ]].

[[ ]]

Conclusion

NSPM has evaluated ATWS and accounted for the effects of the proposed EPU on ATWS. The evaluation confirmed that ARI, SLCS, and recirculating pump trip systems will continue to meet the requirements of 10 CFR 50.62. Therefore, the proposed EPU is acceptable with respect to ATWS.

2.8.6 Fuel Storage

2.8.6.1 New Fuel Storage

Regulatory Evaluation

Nuclear reactor plants include facilities for the storage of new 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.

The NRC's acceptance criteria are based on GDC-62, insofar as it requires the prevention of criticality in fuel storage systems by physical systems or processes, preferably utilizing geometrically safe configurations.

Specific NRC review criteria are contained in SRP Section 9.1.1.

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC listed in the Regulatory Evaluation above, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-66.

New Fuel Storage is described in Monticello USAR Section 10.2.1, "Fuel Storage and Fuel Handling Systems."

In addition to the evaluations described in the Monticello USAR, Monticello's systems and components were evaluated for License Renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Monticello Nuclear Generating Plant License Renewal Safety Evaluation Report (SER), NUREG-1865, dated October 2006 (Reference 5). The license renewal evaluation associated with the fuel storage is documented in NUREG-1865, Section 2.3.3.10. Management of aging effects on fuel storage is documented in NUREG-1865, Section 3.3.2.3.10.

Technical Evaluation

There are no effects due to EPU on the new fuel storage facilities. This is because the CLTR limits the EPU per the following:

- No new fuel product line introduction
- No change to fuel cycle length

The capability of the new fuel storage facility is evaluated whenever a change to fuel design is introduced but that is not the case for this EPU. Therefore, the new fuel storage facility is adequate for EPU.

Conclusion

NSPM has evaluated the effect of EPU on the fuel storage facilities and concludes that the fuel storage facilities will continue to meet the requirements of the current licensing basis following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to the new fuel storage.

2.8.6.2 Spent Fuel Storage

Regulatory Evaluation

Nuclear reactor plants include storage facilities for the wet storage of spent fuel assemblies. The safety function of the spent fuel pool and storage racks is to maintain the 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) GDC-4, 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 (2) GDC-62, 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.

Specific NRC review criteria are contained in SRP Section 9.1.2.

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC listed in the Regulatory Evaluation above, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-40, draft GDC-42, and draft GDC-66.

Spent Fuel Storage is described in Monticello USAR Section 10.2.1, "Fuel Storage and Fuel Handling Systems."

In addition to the evaluations described in the Monticello USAR, Monticello's systems and components were evaluated for License Renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Monticello Nuclear Generating Plant License Renewal Safety Evaluation Report (SER), NUREG-1865, dated October 2006 (Reference 5). The license renewal evaluation associated with the fuel storage is documented in NUREG-1865, Section 2.3.3.10. Management of aging effects on the fuel storage is documented in NUREG-1865, Section 3.3.2.3.10.

Technical Evaluation

There are no effects due to EPU on the spent fuel storage facilities or the ability to transfer spent fuel to shipping casks. This is because the CLTR limits the EPU per the following:

- No new fuel product line introduction
- No change to fuel cycle length

The capability of the spent fuel storage facility is evaluated whenever a change to fuel design is introduced but that is not the case for this EPU. Therefore, the spent fuel storage facility is adequate for EPU.

The EPU effect on the spent fuel cooling capability, fuel pool radiation levels, crud effects, and fuel rack thermal properties are evaluated in Section 2.5.3.1.

Conclusion

NSPM has evaluated the effects of the proposed EPU on the spent fuel storage capability and accounted for the effects of the proposed EPU. The evaluation concludes that the spent fuel pool design will continue to ensure an acceptably low temperature and an acceptable degree of subcriticality following implementation of the proposed EPU. Based on this NSPM concludes that the spent fuel storage facilities will continue to meet the requirements of the current licensing basis following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to spent fuel storage.

Table 2.8-1 Core Exit High Powered Bundle Void Fractions

Power (% OLTP)	Rated Flow	Bundle Void Fractions		
		BOC	MOC	EOC
96.8% (80.6275% EPU)	55%	0.859	0.896	0.885
120% (EPU)	80%	0.851	0.884	0.878
120% (EPU)	100%	0.820	0.821	0.838
106.29% (CLTP)	82.4%	0.838	0.844	0.845

Table 2.8-2 Core Power to Flow Ratio

Power (% OLTP)	Rated Flow	Power-to-Flow Ratio (MWt/Mlbm/hr)
120% (EPU)	80%	43.49
120% (EPU)	100%	34.79

Table 2.8-3 Option III Setpoint Demonstration

OPRM Amplitude Setpoint	OLMCPR(SS)	OLMCPR(2PT)
1.05	1.202	1.063
1.06	1.223	1.082
1.07	1.245	1.102
1.08	1.269	1.122
1.09	1.293	1.143
1.10	1.318	1.166
1.11	1.342	1.187
1.12	1.367	1.210
1.13	1.394	1.233
1.14	1.421	1.257
1.15	1.450	1.282
Acceptance Criteria	Off-rated OLMCPR at 45% Flow	Rated Power OLMCPR

Table 2.8-4 ODYSY Decay Ratios at BSP Region Boundary Endpoints for Nominal Feedwater Temperature Operation

Point*	Power (%)	Flow (%)	Core Decay Ratio	Channel Decay Ratio
Controlled Entry (Region II), NCL Runs				
B2	39.89	33.52	0.795	0.282
B2-ICA	28.6	31.2	<0.795	
Scram (Region I), NCL Runs				
B1	48.81	34.06	0.796	0.260
B1-ICA	42.6	33.7	<0.796	
Controlled Entry (Region II), HFCL Runs				
A2	58.90	42.93	0.794	0.253
A2-ICA	64.5	50.0	<0.794	
Scram (Region I), HFCL Runs				
A1	51.81	34.18	0.778	0.290
A1-ICA	56.6	40.0	<0.778	

* The Power/Flow state points shown above (A1, B1, A2, B2) define the BSP region boundary endpoints on the HFCL and the NCL. The region boundaries can be specified by using the GSF.

$$P = P_1\left[\frac{P_2}{P_1}\right]^{\frac{1}{2}\left[\left(\frac{W-W_1}{W_2-W_1}\right)+\left(\frac{W-W_1}{W_2-W_1}\right)^2\right]}$$

where

P = percent rated power

P_2 = percent rated power at point 2 (HFCL)

P_1 = percent rated power at point 1 (NCL)

W = percent rated core flow

W_2 = percent rated core flow at point 2

W_1 = percent rated core flow at point 1

Table 2.8-5 ECCS Performance Results

Parameter	CLTP	EPU	10 CFR 50.46 Limit
Method	SAFER/GESTR	SAFER/GESTR	
Thermal Power (MWt)	1775	2004	
Licensing Basis PCT (°F)	< 2140	< 2140	≤ 2200
Cladding Oxidation (% Original Clad Thickness)	< 9.0	< 9.0	≤ 17
Hydrogen Generation, Core wide Metal-Water Reaction (%)	< 0.2	< 0.2	≤ 1.0
Coolable Geometry	OK	OK	PCT< 2200 °F, and Local Oxidation <17%
Core Long Term Cooling	OK	OK	Core flooded to TAF or Core flooded to jet pump suction elevation and at least one core spray system is operating at rated flow.
Upper Bound PCT (°F)	< 1670	< 1670	Less than LBPCT
Limiting Appendix K Large Break PCT (°F)	2123	2123	
Limiting Nominal Large Break PCT (°F)	1275	1310	
Limiting Appendix K Small Break PCT (°F)	Bounded by EPU	1607	
Limiting Nominal Small Break PCT (°F)	Bounded by EPU	1179	

Table 2.8-6 Monticello PRFO ATWS Sequence of Events

Step	Event	CLTP Time (sec)	EPU Time (sec)
1	TCV and Bypass Valves Start Open	0.1	0.1
2	MSIV Closure Initiated by Low Steamline Pressure	24.9	40.0
3	MSIVs Fully Closed	28.9	44.0
4	Peak Neutron Flux	28.9	44.0
5	High Pressure ATWS Setpoint	31.4	45.1
6	Opening of the First Relief Valve	31.8	45.5
7	Recirculation Pumps Trip	31.9	45.6
8	Peak Heat Flux	32.1	45.8
9	Peak Vessel Pressure	36.2	51.9
10	BIIT Reached	83.0	88.0
11	Feedwater Reduction Initiated	120.0	130.0
12	SLCS Pumps Start	151.4	165.1
13	RHR Cooling Initiated	600	600
14	Hot Shutdown Achieved (Neutron Flux remains < 0.1%)	1572	1539
15	Peak Suppression Pool Temperature	1391	1401

Table 2.8-7 Monticello MSIVC ATWS Sequence of Events

Step	Event	CLTP Time (sec)	EPU Time (sec)
1	MSIV Isolation Initiated	0.0	0.0
2	MSIVs Fully Closed	4.0	4.0
3	Peak Neutron Flux	4.1	4.0
4	High Pressure ATWS Setpoint	4.5	4.3
5	Opening of the First Relief Valve	4.9	4.7
6	Recirculation Pumps Trip	5.0	4.8
7	Peak Heat Flux	5.3	5.1
8	Peak Vessel Pressure	8.7	10.4
9	BIIT Reached	58.0	52.0
10	Feedwater Reduction Initiated	90.0	90.0
11	SLCS Pumps Start	124.5	124.3
12	RHR Cooling Initiated	600	600
13	Hot Shutdown Achieved (Neutron Flux remains < 0.1%)	1568	1542
14	Peak Suppression Pool Temperature	1396	1392

Table 2.8-8 Monticello LOOP ATWS Sequence of Events

Step	Event	CLTP Time (sec)	EPU Time (sec)
1	Loss of All Auxiliary Power Transformers	0.0	0.0
2	Recirculation Pumps Trip	0.0	0.0
3	Feedwater / Condensate Pumps Trip	0.0	0.0
4	Peak Neutron Flux	0.5	0.5
5	Peak Heat Flux	0.7	0.6
6	High Pressure ATWS Setpoint	1.2	1.0
7	Opening of the First Relief Valve	1.7	1.3
8	MSIV Isolation Initiated	2.0	2.0
9	MSIVs Fully Closed	6.0	6.0
10	Peak Vessel Pressure	3.7	5.6
11	BIIT Reached	62.0	56.0
12	SLCS Pumps Start	121.2	121.0
13	RHR Cooling Initiated	600	600
14	Hot Shutdown Achieved (Neutron Flux remains < 0.1%)	1574	1530
15	Peak Suppression Pool Temperature	4410	4795

Table 2.8-9 Monticello Key Inputs for ATWS Analysis

Input Variable	CLTP	EPU
Reactor power (MWt)	1775	2004
Analyzed power (MWt)	1775	2044
Reactor dome pressure (psia)	1025	1025
Each SRV capacity at 1120 (Mlbm/hr)	0.821	0.821
High pressure ATWS-RPT (psig)	1162	1162
Number of SRVs Out-of-service (OOS)	1	1

Table 2.8-10 Monticello Results for ATWS Analysis [1]

Acceptance Criteria	CLTP [2]	EPU
Peak vessel bottom pressure (psig)	1385	1489
Peak suppression pool temperature (°F)	187	189
Peak containment pressure (psig)	11.1	11.6

Notes:

1. Cladding temperature and oxidation remain below their 10 CFR 50.46 limits.
2. To maximize the effect of EPU, a baseline is established at the CLTP level, assuming the current licensed equipment performance assumptions and plant parameters.

Figure 2.8-1 Power of Peak Bundle versus Cycle Exposure

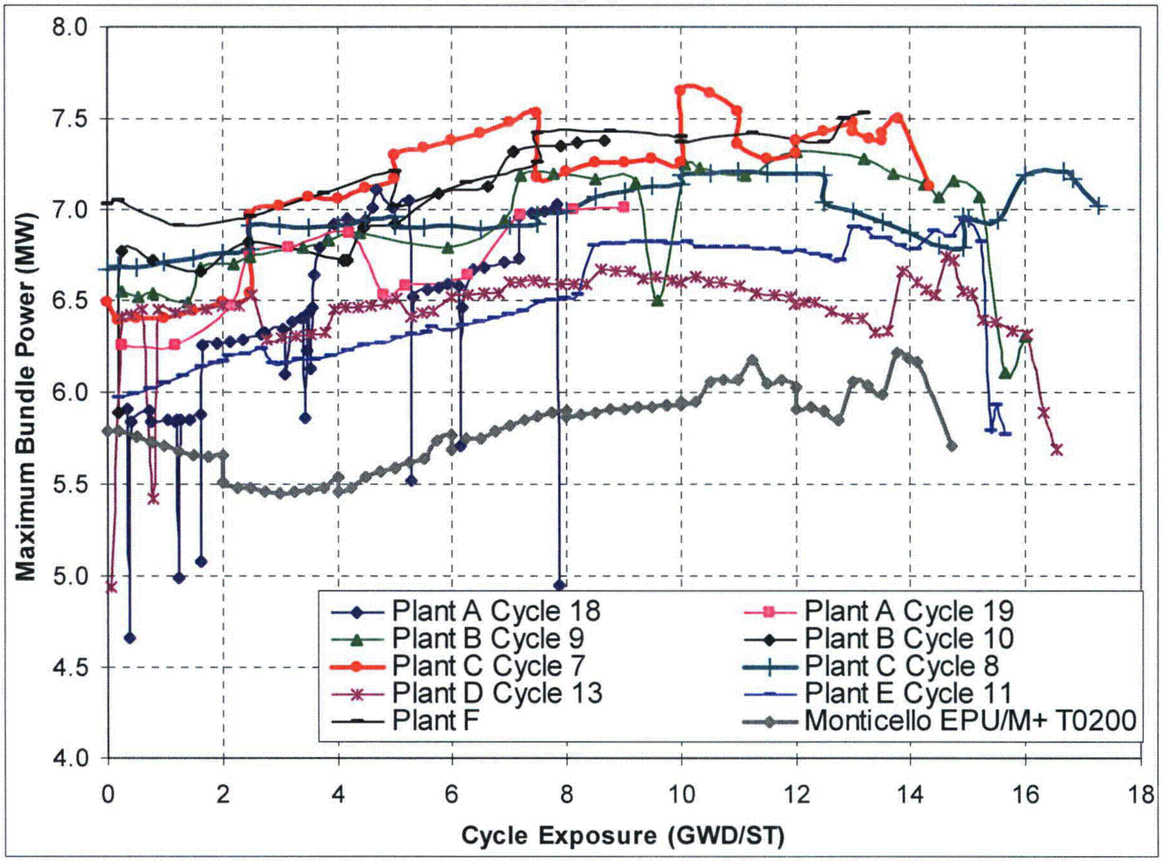


Figure 2.8-2 Coolant Flow for Peak Bundle versus Cycle Exposure

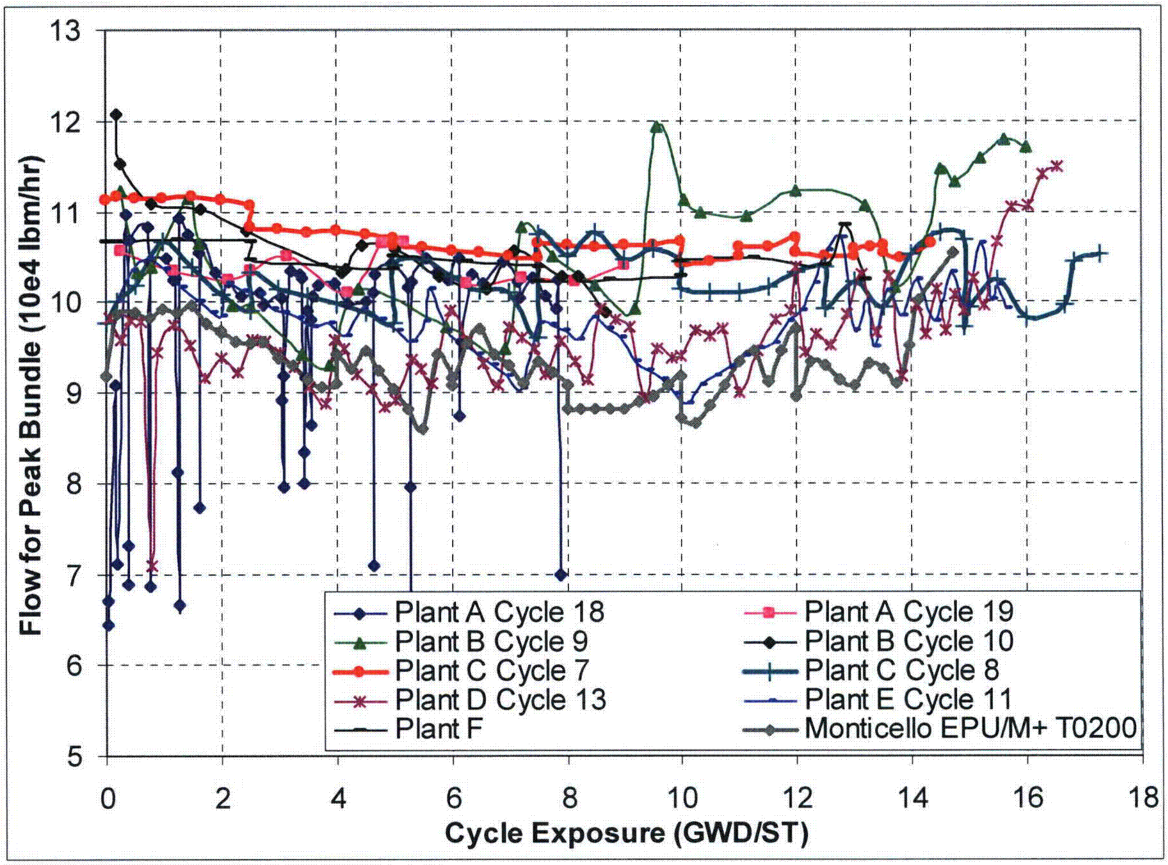


Figure 2.8-3 Exit Void Fraction for Peak Power Bundle versus Cycle Exposure

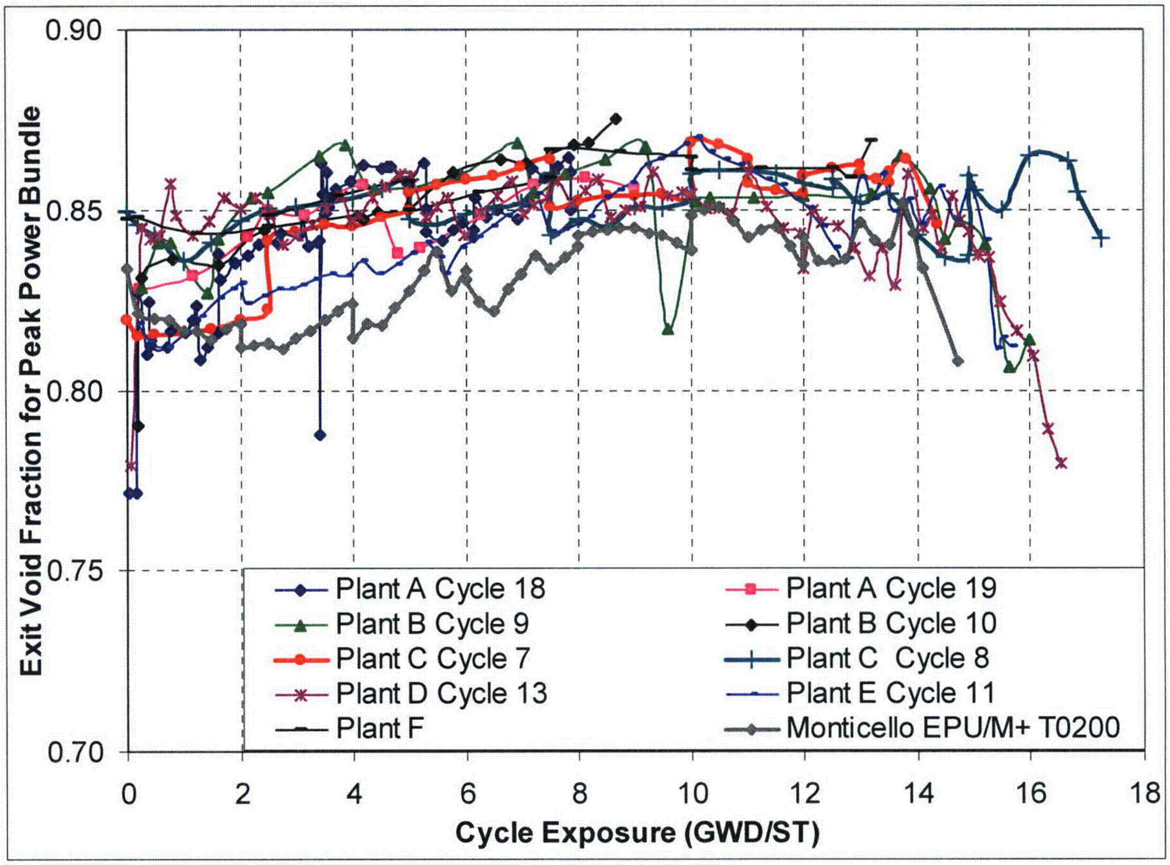


Figure 2.8-4 Maximum Channel Exit Void Fraction versus Cycle Exposure

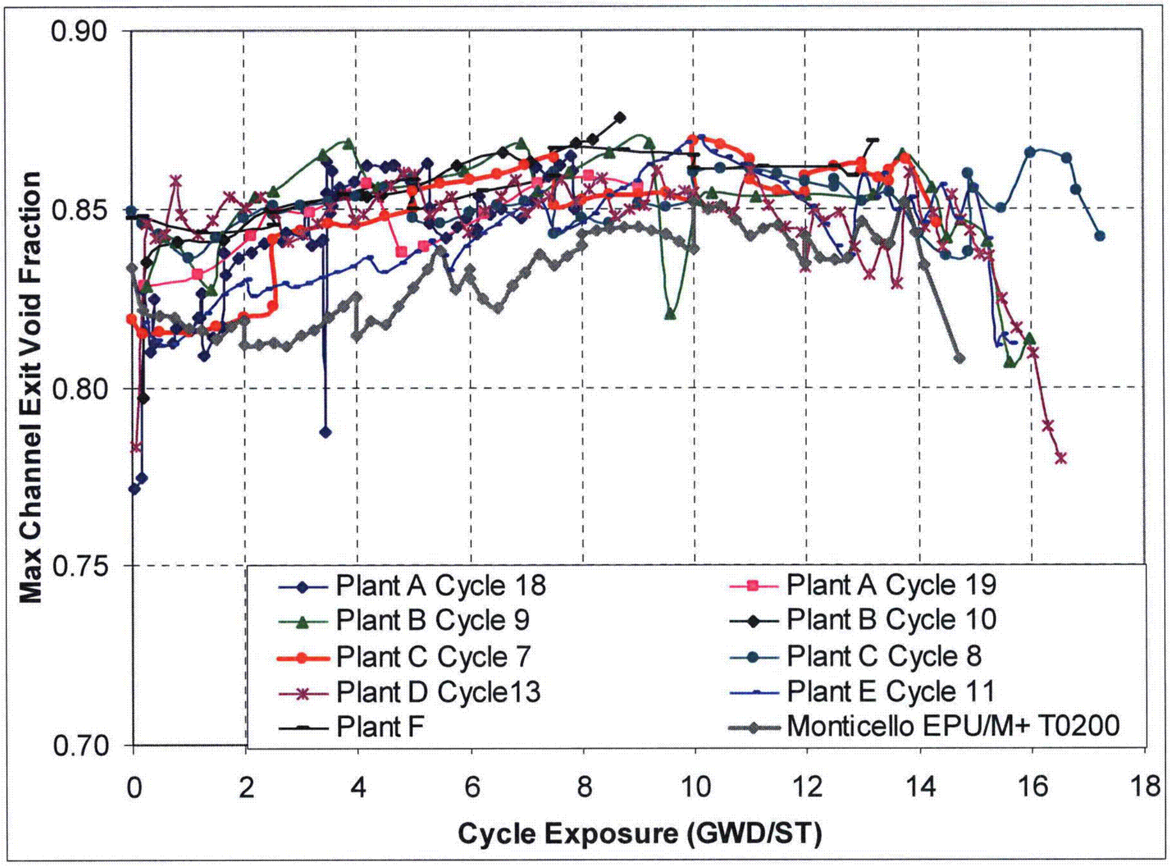


Figure 2.8-5 Core Average Exit Void Fraction versus Cycle Exposure

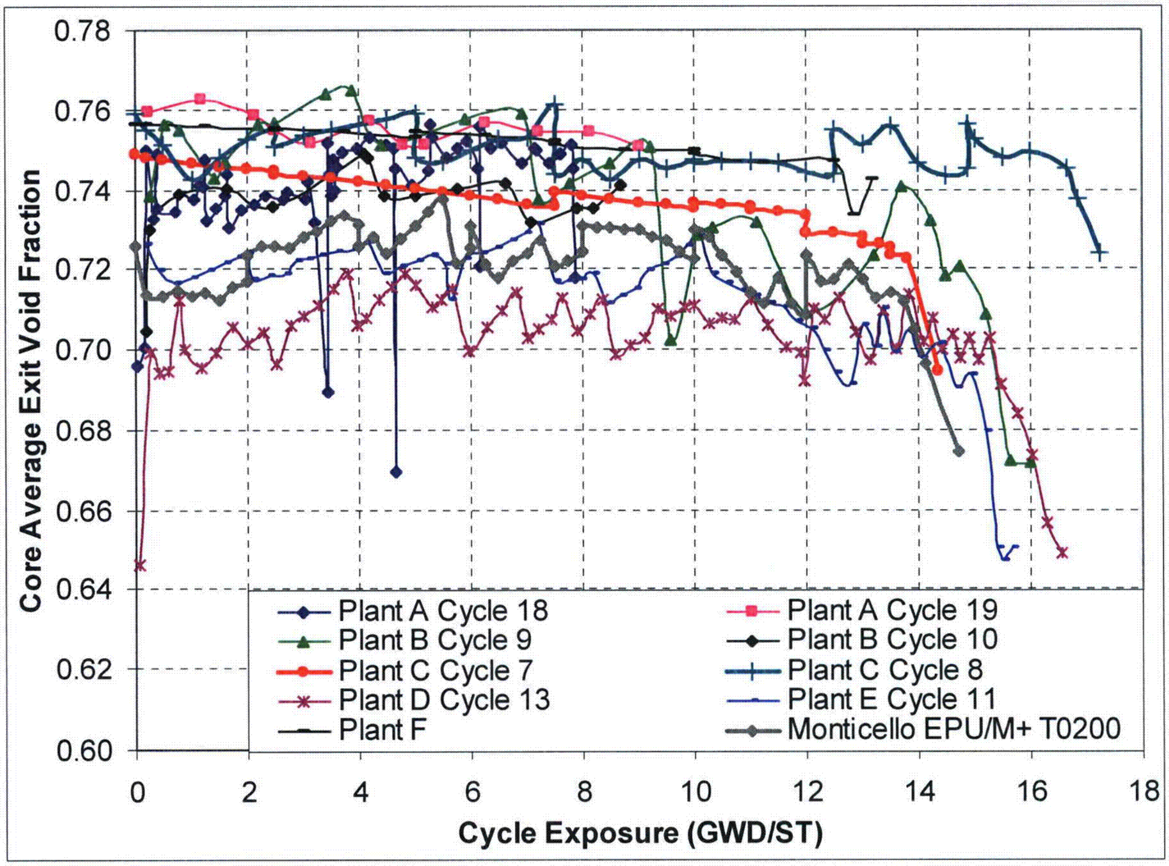


Figure 2.8-6 Peak LHGR versus Cycle Exposure

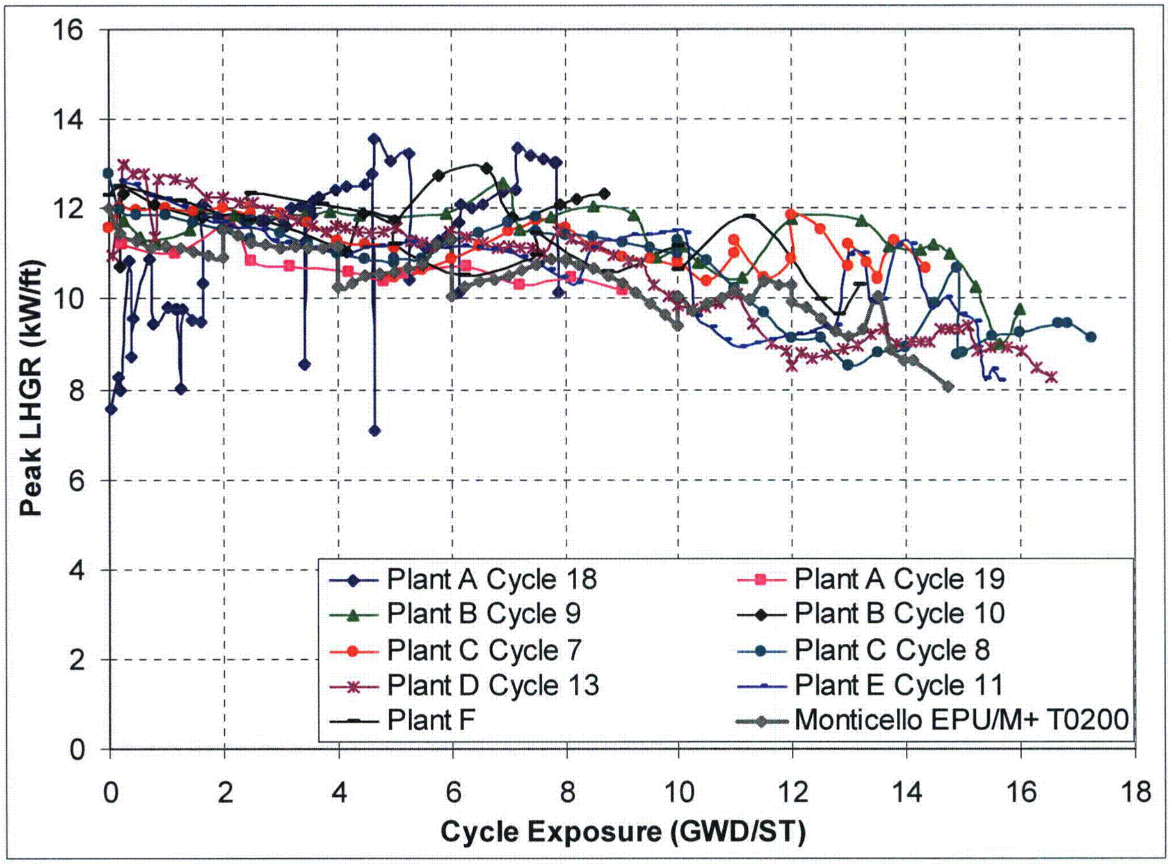


Figure 2.8-7 Peak Nodal Exposures

EOC Peak Nodal Exposures

Plant	Cycle	Peak Nodal Exposure (GWD/ST)
A	18	38.849
A	19	43.784
B	9	56.359
B	10	51.544
C	7	53.447
C	8	47.766
D	13	56.660
E	11	55.387
F		51.174
Monticello EPU T0200		55.050

Figure 2.8-8 Dimensionless Bundle Power at BOC (200 MWd/ST)

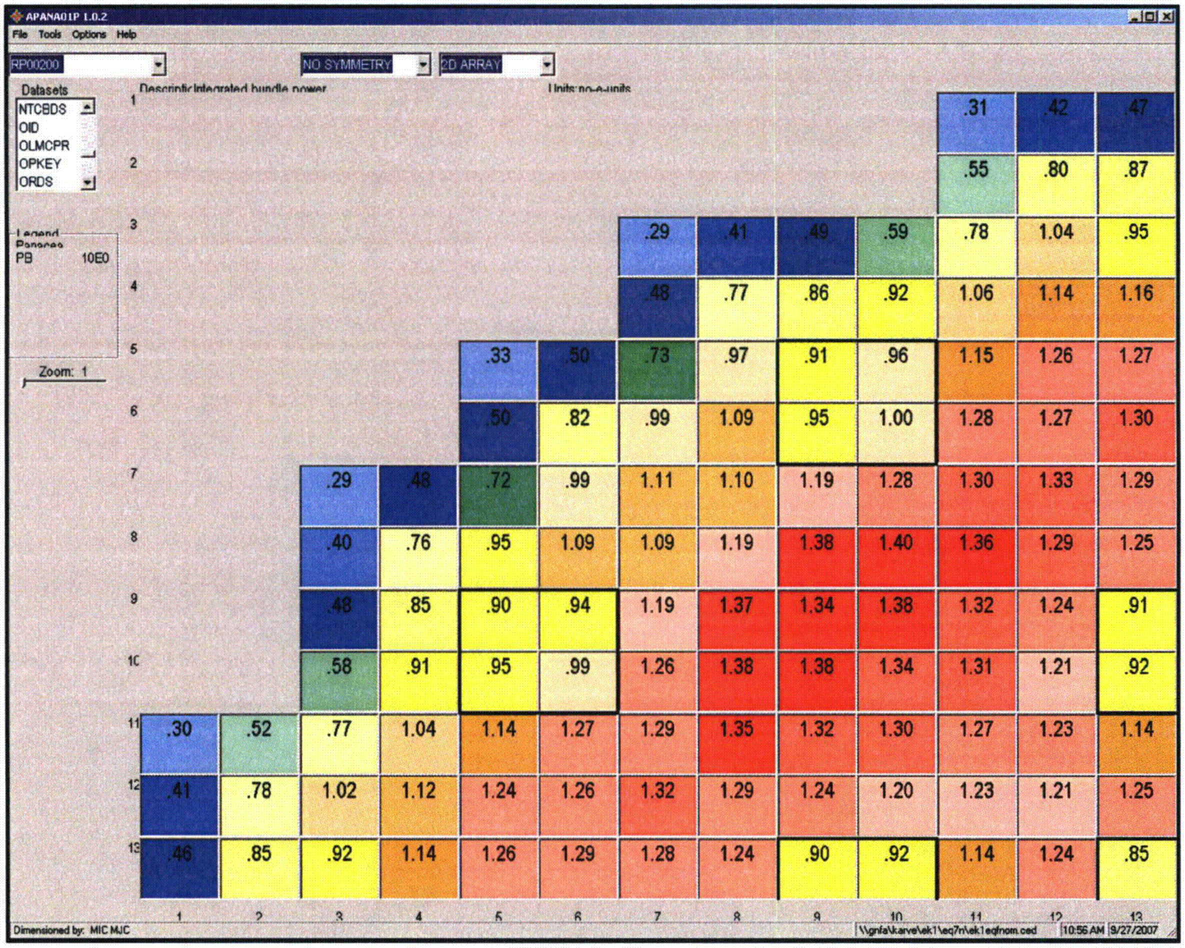


Figure 2.8-9 Dimensionless Bundle Power at MOC (8500 MWd/ST)

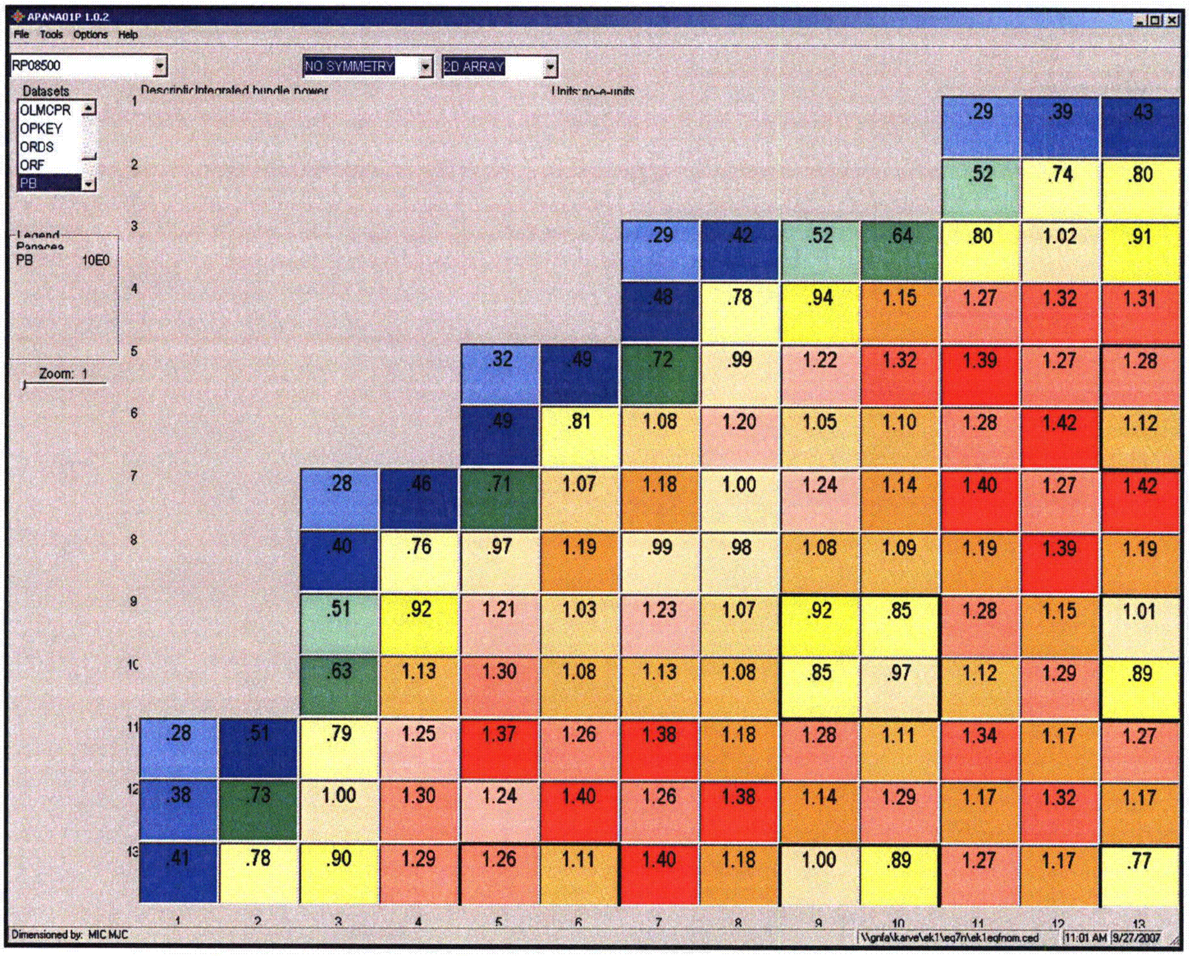
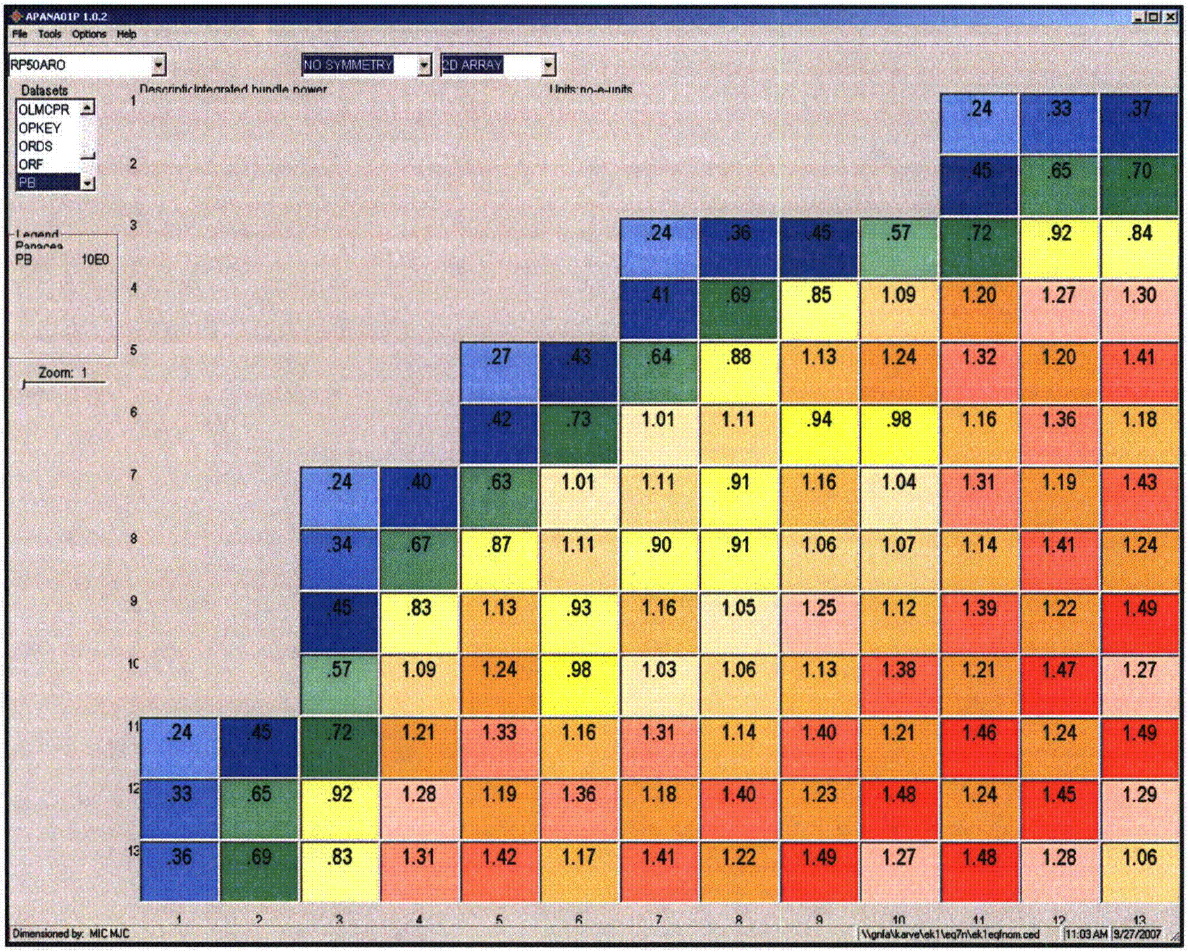


Figure 2.8-10 Dimensionless Bundle Power at EOC (13946 MWd/ST)



	1	2	3	4	5	6	7	8	9	10	11	12	13
1											.24	.33	.37
2											.45	.65	.70
3							.24	.36	.45	.57	.72	.92	.84
4							.41	.69	.85	1.09	1.20	1.27	1.30
5					.27	.43	.64	.88	1.13	1.24	1.32	1.20	1.41
6					.42	.73	1.01	1.11	.94	.98	1.16	1.36	1.18
7			.24	.40	.63	1.01	1.11	.91	1.16	1.04	1.31	1.19	1.43
8			.34	.67	.87	1.11	.90	.91	1.06	1.07	1.14	1.41	1.24
9			.45	.83	1.13	.93	1.16	1.05	1.25	1.12	1.39	1.22	1.49
10			.57	1.09	1.24	.98	1.03	1.06	1.13	1.38	1.21	1.47	1.27
11	.24	.45	.72	1.21	1.33	1.16	1.31	1.14	1.40	1.21	1.46	1.24	1.49
12	.33	.65	.92	1.28	1.19	1.36	1.18	1.40	1.23	1.48	1.24	1.45	1.29
13	.36	.69	.83	1.31	1.42	1.17	1.41	1.22	1.49	1.27	1.48	1.28	1.06

Figure 2.8-11 Bundle Operating LHGR (kW/ft) at BOC (200 MWd/ST)

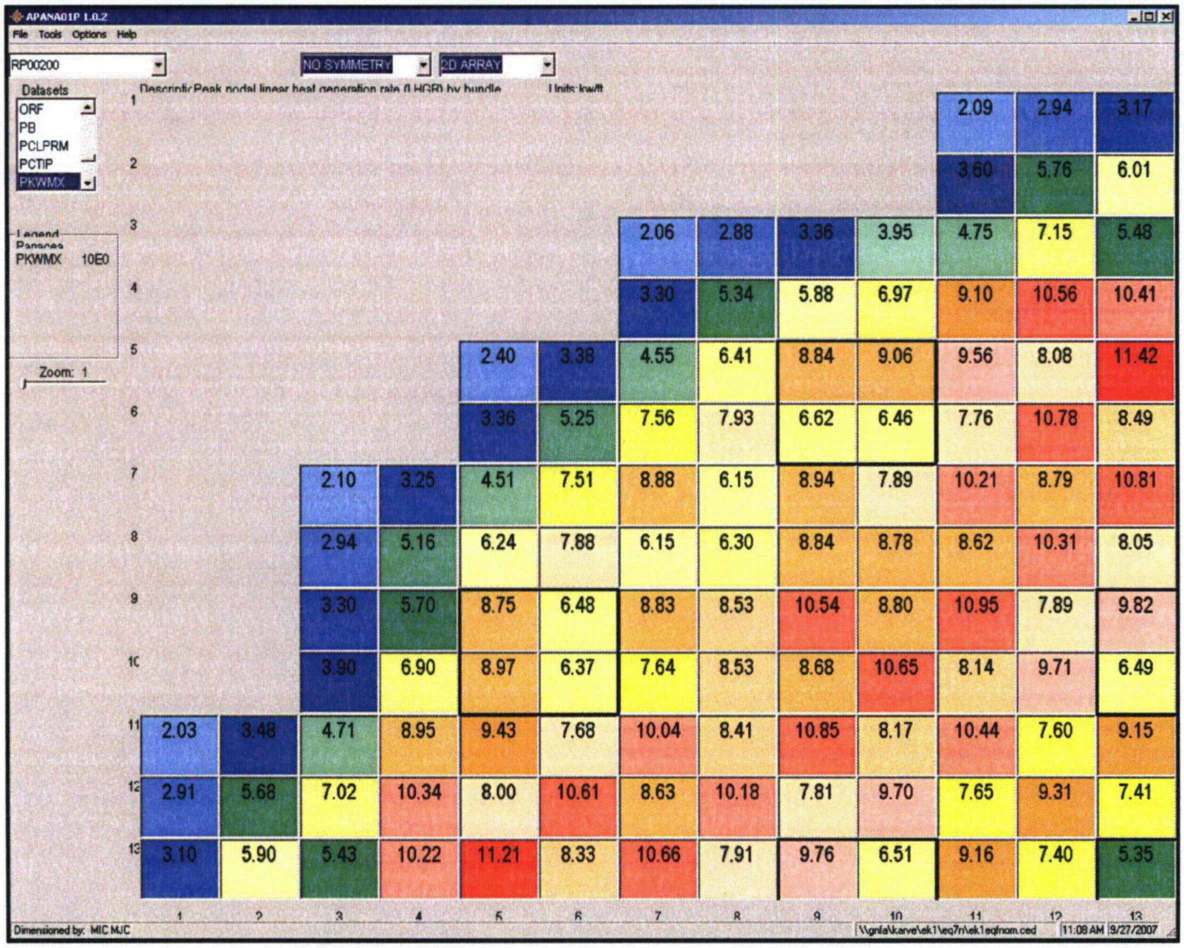


Figure 2.8-12 Bundle Operating LHGR (kW/ft) at MOC (8500 MWd/ST)

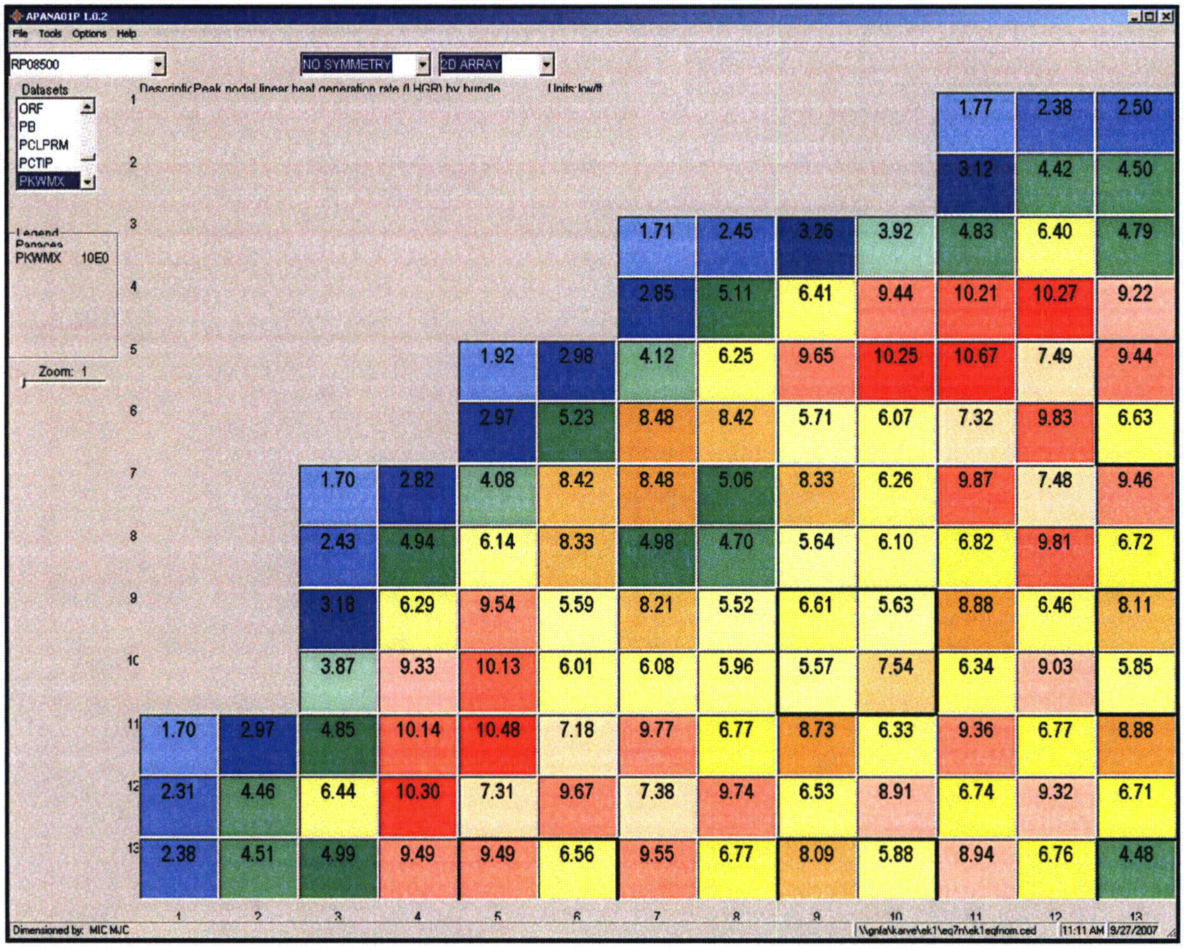


Figure 2.8-13 Bundle Operating LHGR (kW/ft) at EOC (13946 MWd/ST)

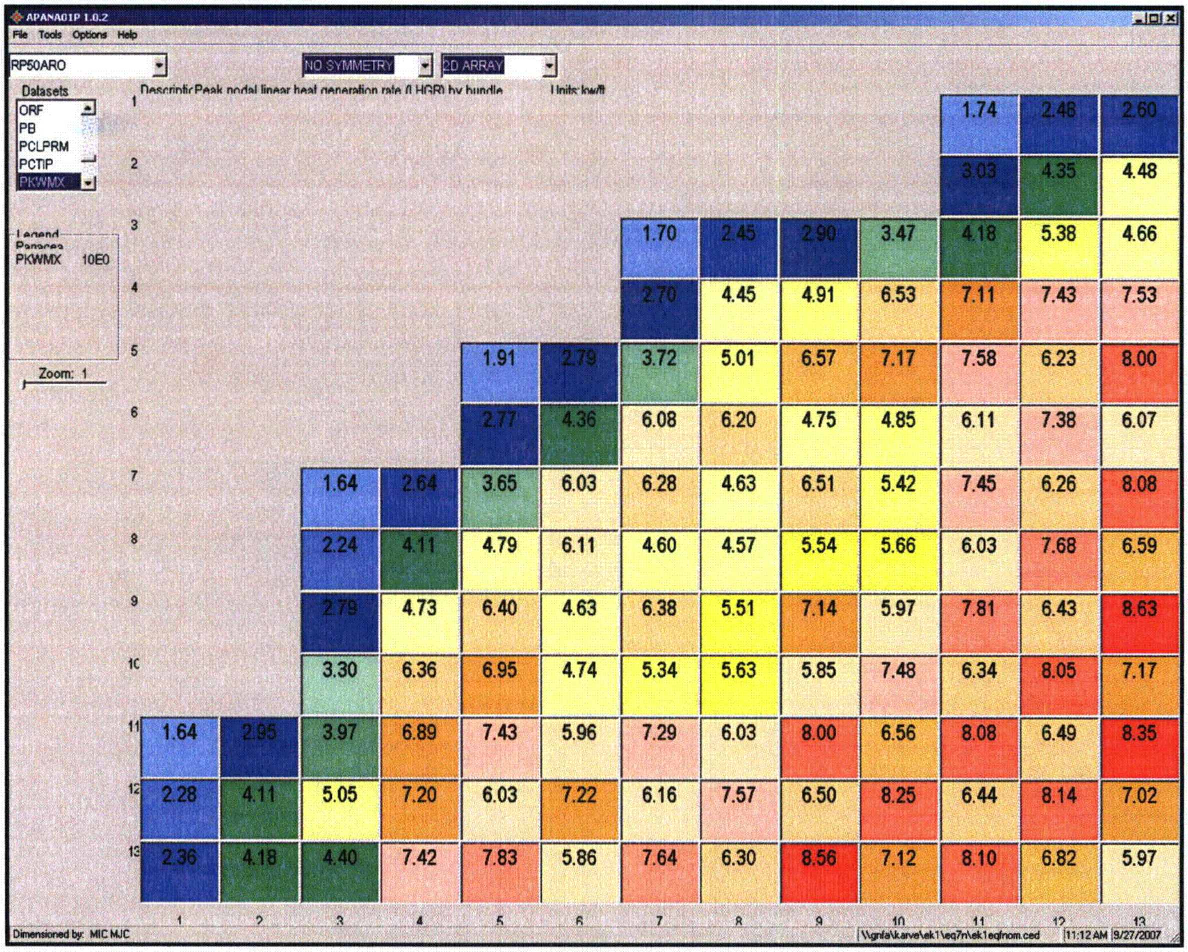


Figure 2.8-14 Bundle operating MCPR at BOC (200 MWd/ST)

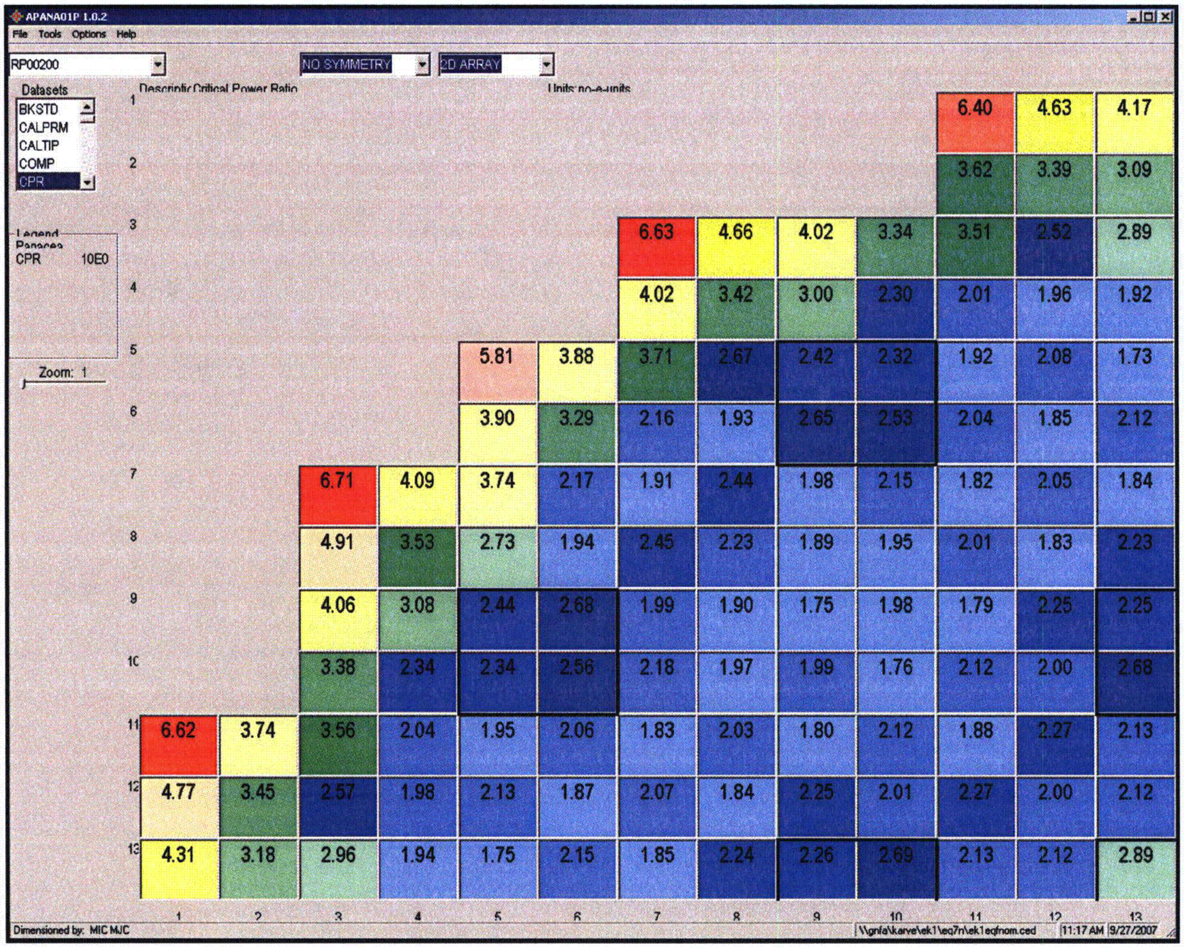


Figure 2.8-15 Bundle operating MCPR at MOC (8500 MWd/ST)

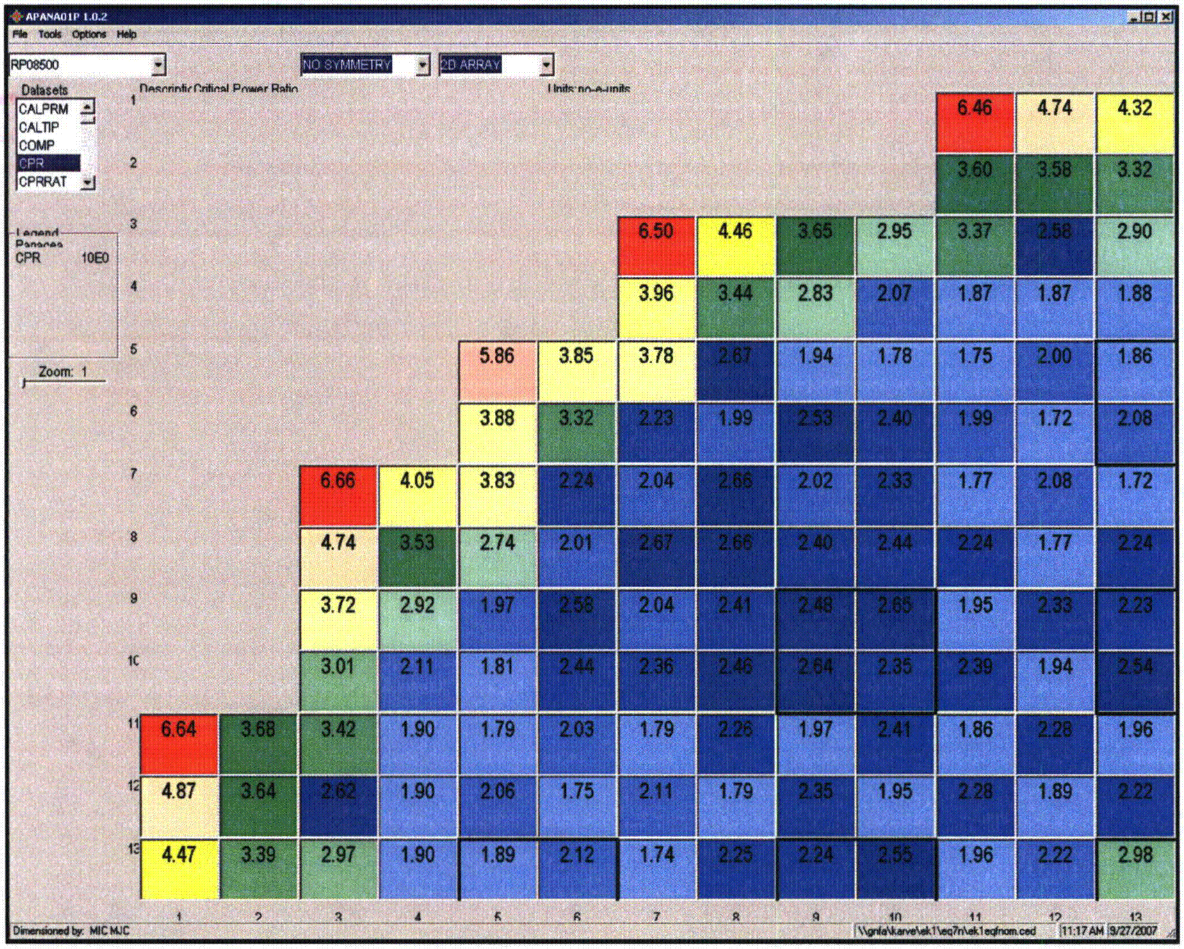


Figure 2.8-16 Bundle operating MCPR at EOC (13946 MWd/ST)

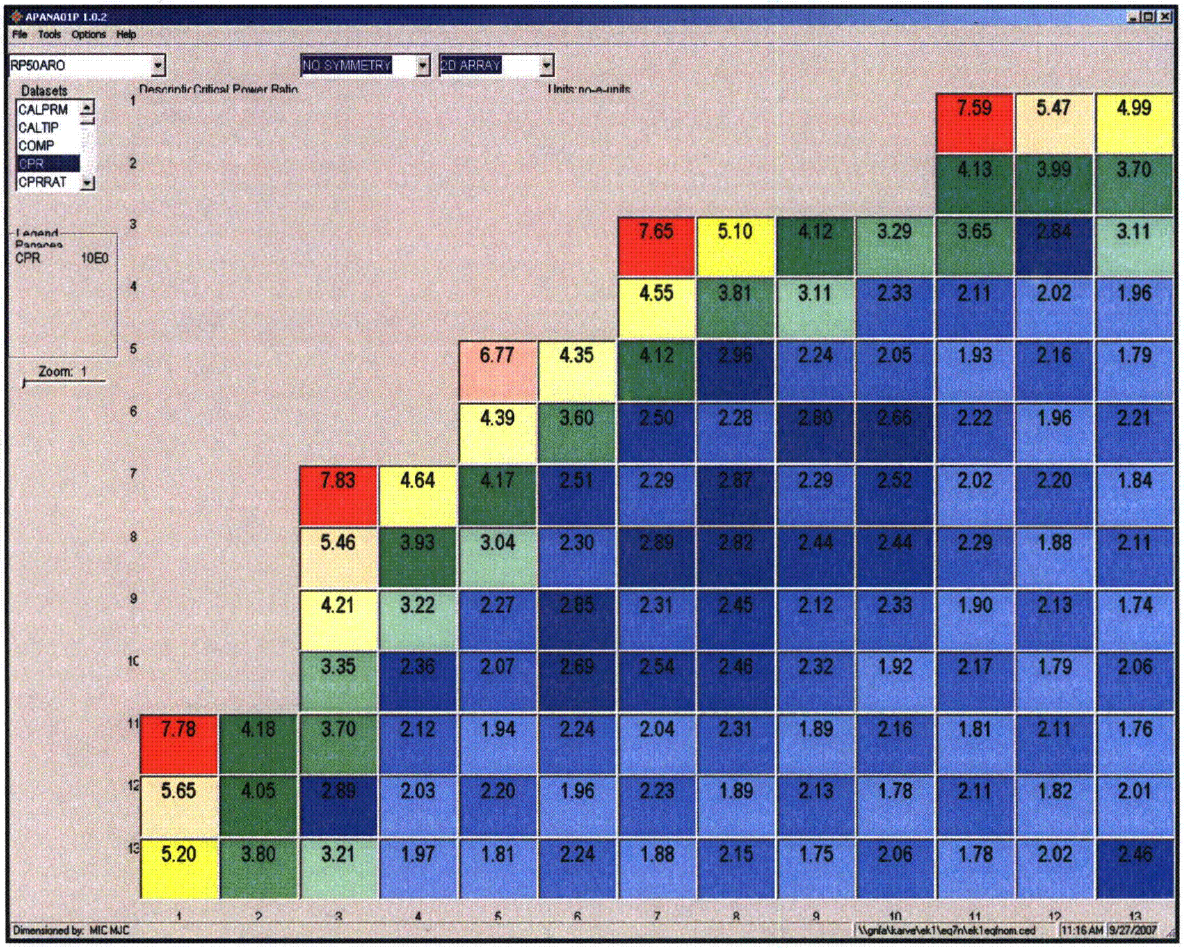


Figure 2.8-17 Bundle Operating LHGR (kW/ft) at 12000 MWd/ST (peak MFLPD)

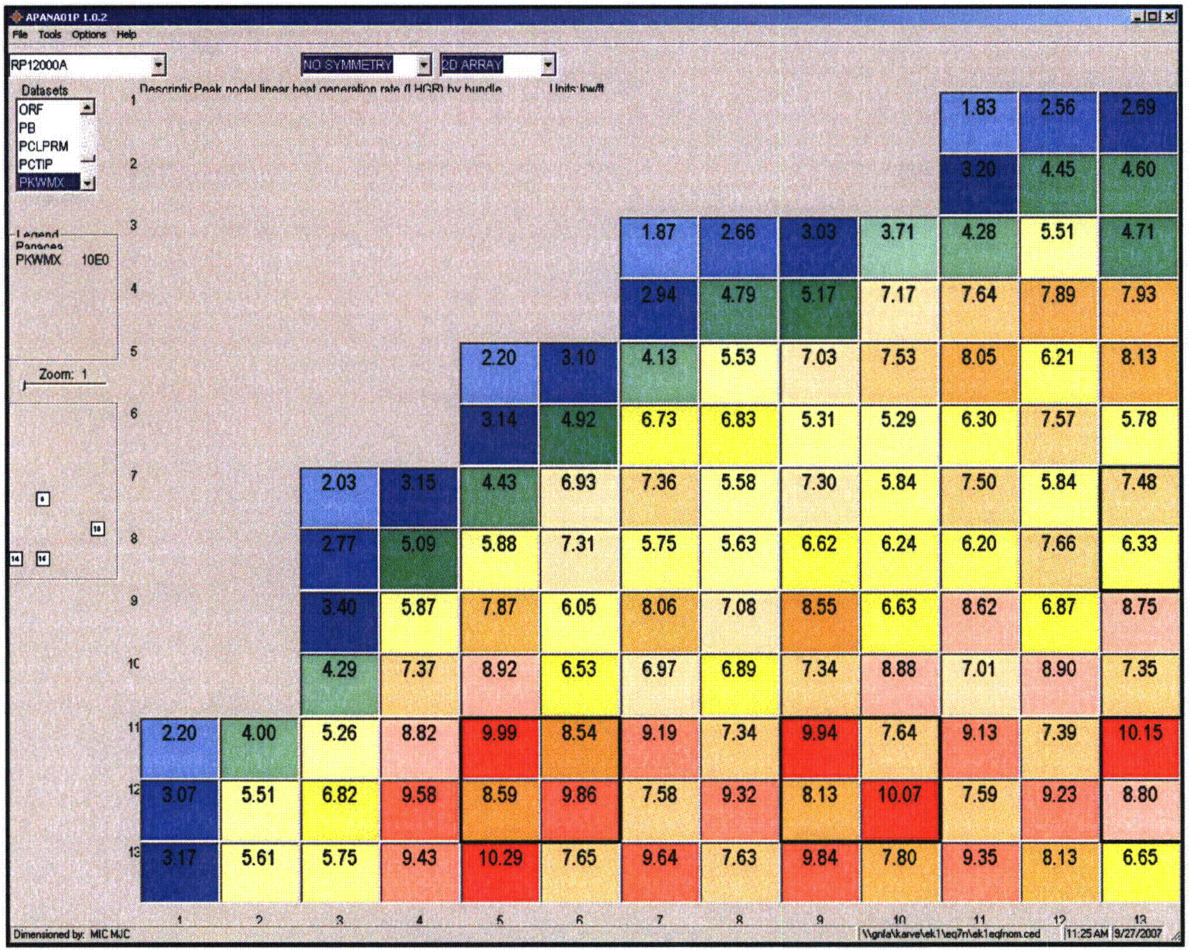


Figure 2.8-18 Bundle operating MCPR at 13750 MWd/ST (peak MFLCPR point)

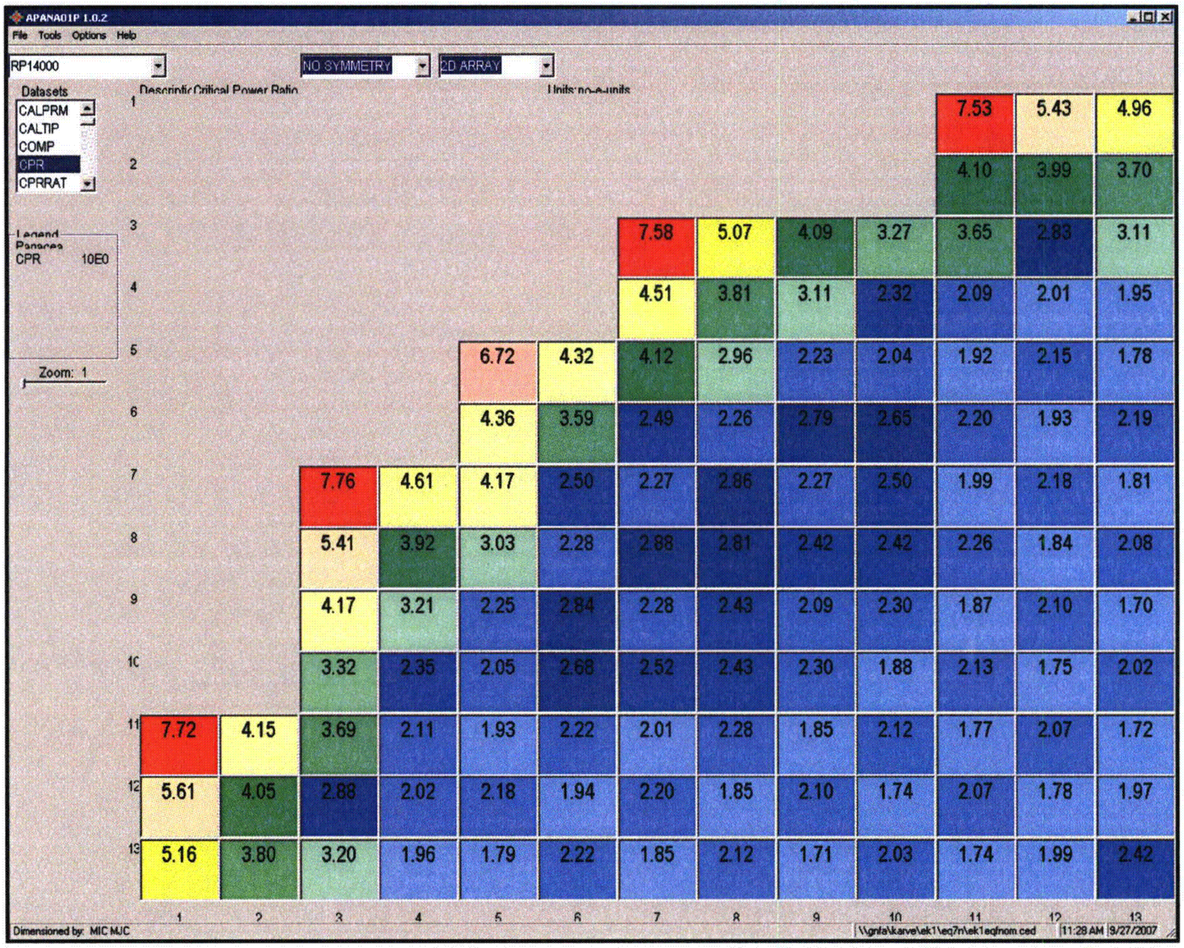


Figure 2.8-19 Illustration of OPRM Trip Enabled Region for Monticello EPU

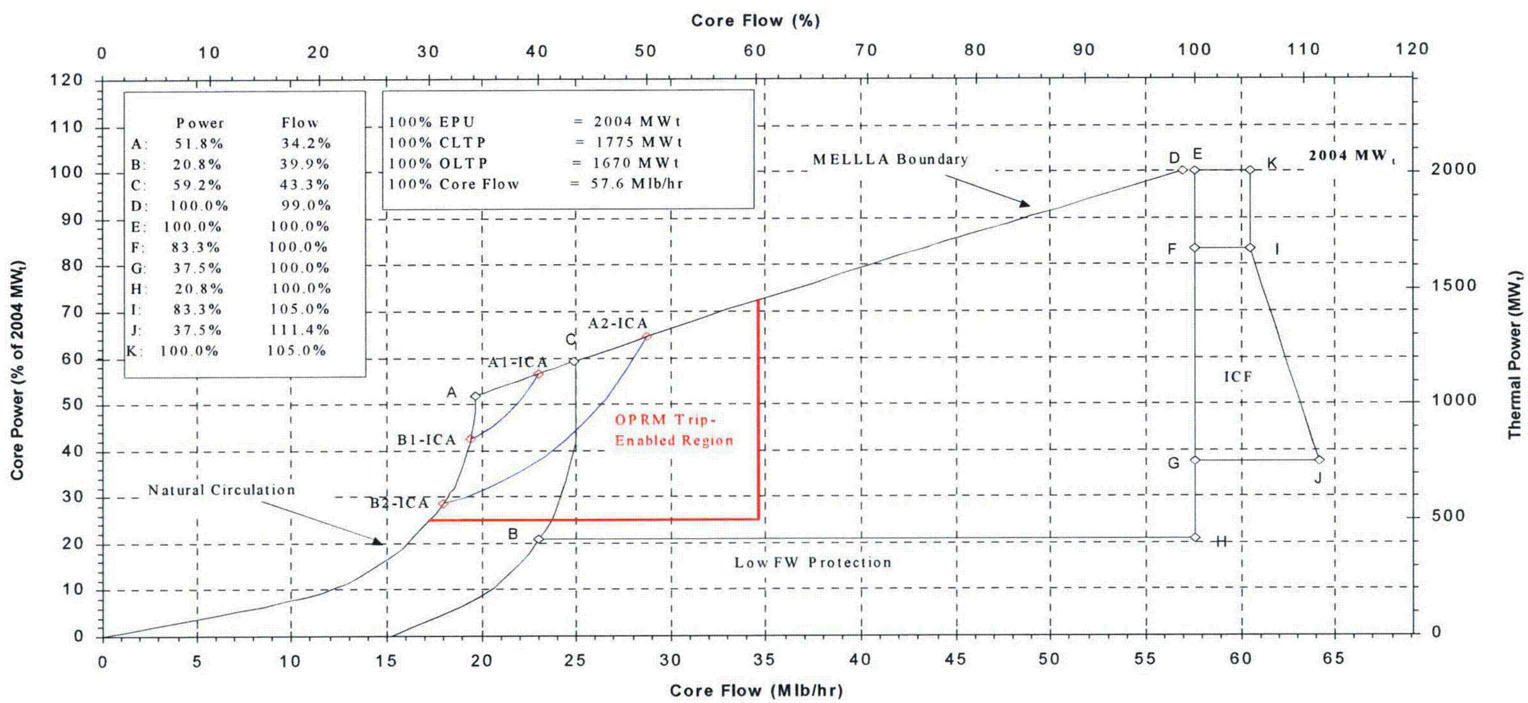


Figure 2.8-20 BSP Regions for Nominal Feedwater Temperature for Equilibrium GE14 Core

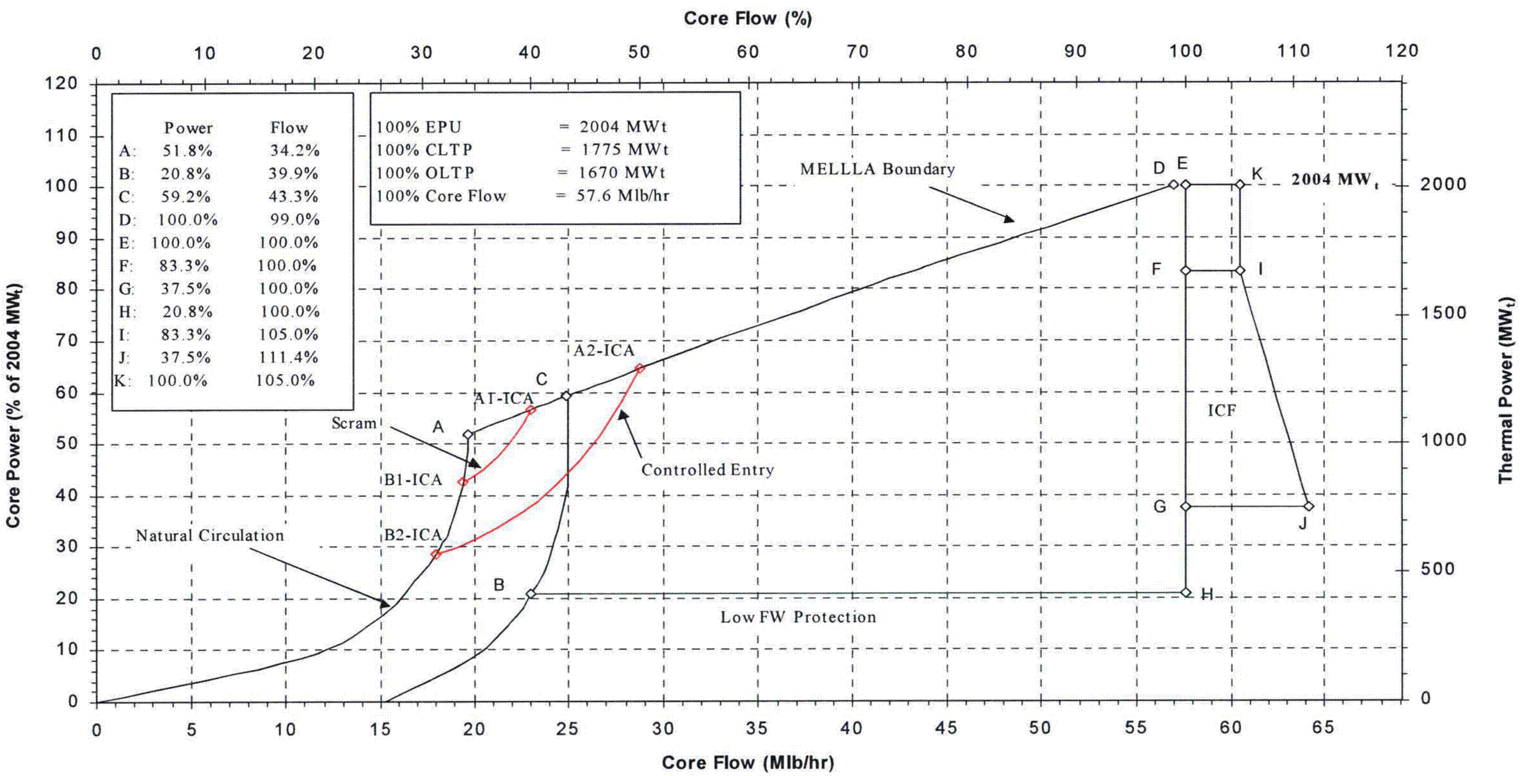
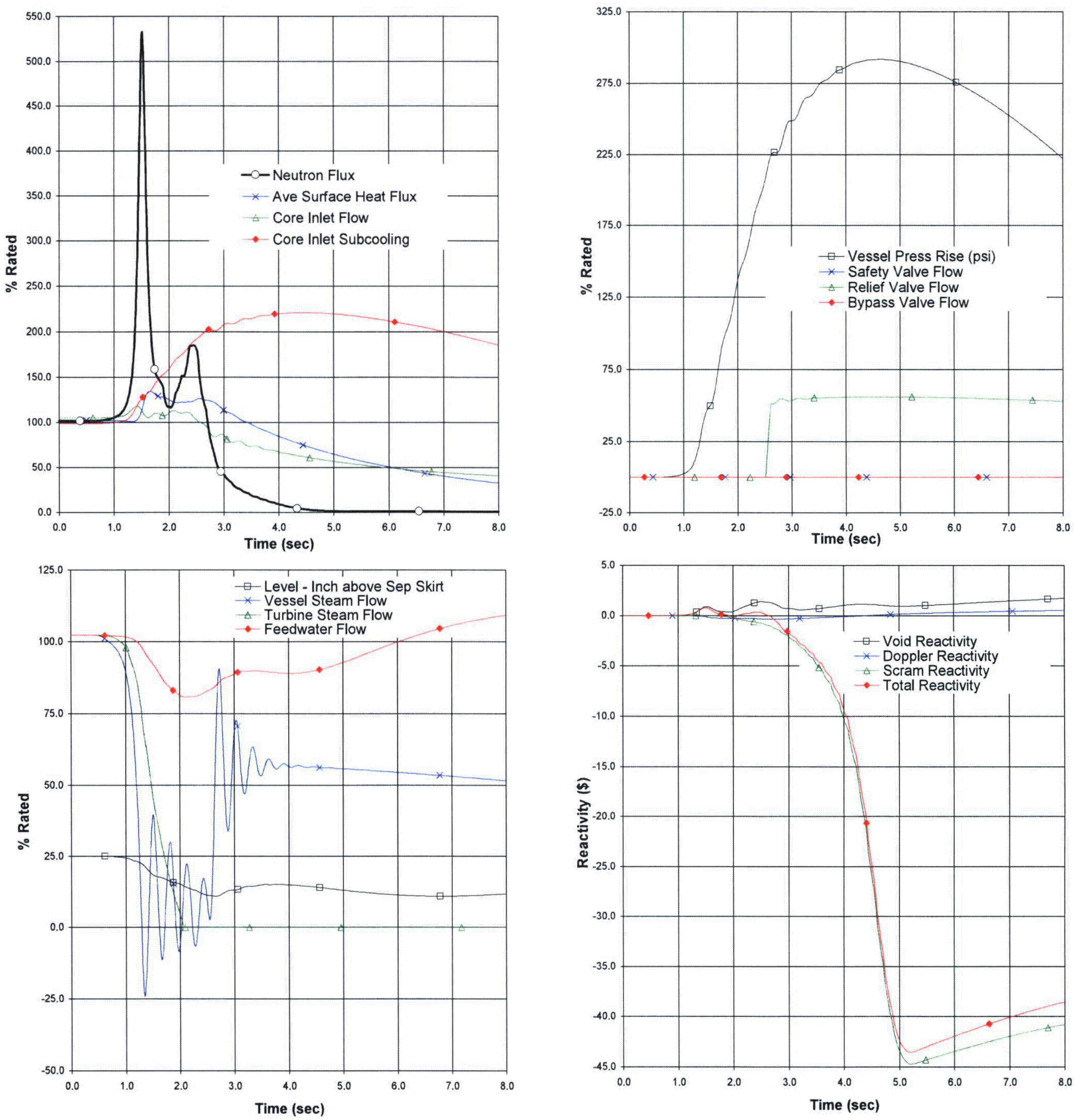


Figure 2.8-21 Response to MSIV Closure with Flux Scram

(102% EPU power, 105% core flow, and 1040 psia initial dome pressure)



2.9 Source Terms and Radiological Consequences Analyses

2.9.1 Source Terms for Radwaste Systems Analyses

Regulatory Evaluation

The NRC staff reviewed the radioactive source term associated with EPUs to ensure the adequacy of the sources of radioactivity used by the licensee as input to calculations to verify that the radioactive waste management systems have adequate capacity for the treatment of radioactive liquid and gaseous wastes.

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 NRC review criteria are contained in SRP Section 11.1.

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC listed in the Regulatory Evaluation above, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-70.

The radioactive waste systems are described in Monticello USAR Chapter 9, "Plant Radioactive Waste Control Systems."

In addition to the evaluations described in the Monticello USAR, Monticello's systems and components were evaluated for License Renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Monticello Nuclear Generating Plant License Renewal Safety Evaluation Report (SER), NUREG-1865, dated October 2006 (Reference 5). The license renewal evaluation associated with the solid and liquid radioactive waste systems are documented in NUREG-1865, Section 2.3.3.13. Management of aging effects on the solid and liquid radioactive waste systems is documented in NUREG-1865, Section 3.3.2.3.13.

Technical Evaluation

Radiation sources in the reactor coolant at Monticello include activation products, activated corrosion products and fission products.

Coolant Activation Products

During reactor operation, the coolant passing through the core region becomes radioactive as a result of nuclear reactions. The coolant activation, especially 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 in approximate proportion to the increase in thermal power. [[

]] The activation products in the steam from the proposed EPU are bounded by the existing design basis concentration. The margin in the Monticello plant design basis for reactor coolant activation concentrations significantly exceeds potential increases due to EPU. Therefore, no change is required in the design basis reactor coolant activation product concentrations for the EPU.

Activated Corrosion Products and Fission Products

The reactor coolant contains activated corrosion products, which are the result of metallic materials entering the water and being activated in the reactor region. Under the EPU conditions, the FW flow increases with power and the activation rate in the reactor region increases with power. The net result is an increase in the activated corrosion product production. Total activated corrosion product activity as a result of EPU is approximately 41% of the total design basis activity. Therefore, the activated corrosion product activity remains unchanged due to EPU.

Fission products in the reactor coolant are separable into the products in the steam and the products in the reactor water. The activity in the steam consists of noble gases released from the core plus carryover activity from the reactor water. This activity is the noble gas offgas that is included in the plant design. The calculated offgas rates for EPU after thirty minutes decay are well below the original design basis of 0.26 Curies/sec. Therefore, no change is required in the design basis for offgas activity for the EPU.

The fission product activity in the reactor water, like the activity in the steam, is the result of minute releases from the fuel rods. The cumulative calculated reactor water fission product activity levels are less than 2% of the cumulative design basis reactor water fission product activity levels. Therefore, the fission product activity design basis for Monticello is unchanged for the EPU.

Radiation Sources in the Reactor Core

During power operation, the radiation sources in the core are directly related to the fission rate. These sources include radiation from the fission process, accumulated fission products and neutron reactions as a secondary result of fission. Historically, these sources have been defined in terms of energy or activity released per unit of reactor power. Therefore, for an EPU, the percent increase in the operating source terms is no greater than the percent increase in power.

[[

[redacted]	[redacted]	[redacted]
		]]

The post-operation radiation sources in the core are primarily the result of accumulated fission products. Two separate forms of post-operation source data are normally applied. The first of these is the core gamma-ray source, which is used in shielding calculations for the core and for individual fuel bundles. This source term is defined in terms of MeV/sec per Watt of reactor thermal power (or equivalent) at various times after shutdown. The total gamma energy source, therefore, increases in proportion to reactor power.

The second set of post-operation source data consists primarily of nuclide activity inventories for fission products in the fuel. These data are needed for post-accident and SFP evaluations, which are performed in compliance with regulatory guidance that applies different release and transport assumptions to different fission products. The core fission product inventories for these evaluations are based on an assumed fuel irradiation time, which develops "equilibrium" activities in the fuel (typically 3 years). Most radiologically significant fission products reach equilibrium within a 60-day period. [[

]] The radionuclide inventories are determined in terms of Curies per megawatt of reactor thermal power at various times after shutdown.

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]] The results of the Monticello plant specific radiation sources are included in the Loss of Coolant Accident, Fuel Handling Accident, and CRDA radiological analyses presented in Section 2.9.2.

Conclusion

NSPM has evaluated the radioactive source term for radwaste systems associated with the proposed EPU. The evaluation concludes that the proposed radioactive source term meets the requirements of 10 CFR Part 20, 10 CFR Part 50, Appendix I, and the current licensing basis. Therefore, the proposed EPU is acceptable with respect to source terms.

2.9.2 Radiological Consequences Analyses Using Alternative Source Terms

Regulatory Evaluation

The NRC's acceptance criteria for radiological consequences analyses using an alternative source term are based on (1) 10 CFR 50.67, insofar as it sets standards for radiological consequences of a postulated accident, and (2) GDC-19, 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 50.2, for the duration of the accident.

Specific NRC review criteria are contained in SRP Section 15.0.1.

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC listed in the Regulatory Evaluation above, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-11. Current GDC-19 is applicable to Monticello as described in USAR Sections 5.3.5, 6.7.2, 12.3.1.6, and 14.7.

Radiological consequences associated with potential Monticello accidents are addressed in USAR Section 14.7, "Accident Evaluation Methodology."

Technical Evaluation

The 100-day post-accident radiation doses are expected to increase. Accident radiation dose due to gamma radiation inside containment increases by up to 8.3 percent. Beta radiation dose at EPU inside containment is bounded by previous analysis. Reactor and Turbine Building Post Accident doses increase by up to 13 percent.

With the elimination of Post-Accident Sampling under Amendment 136 dated June 17, 2003, the Monticello design and licensing basis does not include any post accident missions requiring plant-specific dose analysis under NUREG-0737, Item II.B.2.

The CLTR, Section 9.2 requires a plant specific evaluation for a LOCA inside Containment, Fuel Handling Accident (FHA), Control Rod Drop Accident (CRDA) and Main Steam Line Break Accident (MSLBA) Outside Containment. The total effective dose equivalent (TEDE) was calculated at the exclusion area boundary (EAB), low population zone (LPZ), and in the main control room (CR). Additionally for a LOCA, the TEDE was calculated in the Technical Support Center (TSC). The doses resulting from the accidents analyzed are compared with the applicable dose limits in Tables 2.9-1 through 2.9-5, for both the EPU and CLTP RTP levels. The results for the EPU remain below established regulatory limits.

The magnitude of radiological consequences of a DBA is proportional to the quantity of radioactivity released to the environment. This quantity is a function of the fission products released from the core as well as the transport mechanism between the core and the release point.

Loss of Coolant Accident

The post-LOCA doses were re-analyzed for EPU conditions. This analysis revision was performed based on plant operation at 102% of the EPU power level of 2004 MWt using the Alternative Source Term (AST) in accordance with the guidance provided by the Nuclear Regulatory Commission (NRC) in Regulatory Guide 1.183 (July 2000) and as allowed by 10 CFR 50.67. This analysis updates analysis for the Monticello implementation of a full-scope conversion to the AST methodology reviewed and approved as License Amendment 148. The analysis used the generic source term inventory approved for use under the CLTR. Containment leakage rates versus time were revised based on the containment analysis pressure response timing for EPU. The suppression pool pH response was revised and remains basic, which prevents re-evolution of iodine from the pool. Analysis methods and other inputs were not changed from those used in Amendment 148.

The post-LOCA EAB, LPZ, CR, and TSC doses are within the applicable regulatory limits with the results and regulatory criteria summarized in Table 2.9-1.

Fuel Handling Accident (FHA)

The post-FHA doses were re-analyzed for EPU conditions. This analysis revision was performed based on plant operation at 102% of the EPU power level of 2004 MWt using the AST in accordance with the guidance provided by the NRC in Regulatory Guide 1.183 (July 2000) and as allowed by 10 CFR 50.67. This analysis updates analysis for the Monticello implementation of a full-scope conversion to the AST methodology reviewed and approved as Monticello License Amendment 145 (partial scope AST conversion for the FHA event) . The analysis used the generic source term inventory approved for use under the CLTR. Analysis methods and other inputs were not changed from those used in Amendment 145.

The post-FHA EAB, LPZ, and CR doses are within the applicable regulatory limits with the results and regulatory criteria summarized in Table 2.9-2.

Control Rod Drop Accident (CRDA)

The post-CRDA EAB, LPZ, and CR doses were re-analyzed for EPU conditions. This analysis revision was performed based on plant operation at 102% of the EPU power level of 2004 MWt using AST in accordance with the guidance provided by the NRC in Regulatory Guide 1.183 (July 2000) and as allowed by 10 CFR 50.67. This analysis updates analysis for the Monticello implementation of a full-scope conversion to the AST methodology reviewed and approved as Monticello License Amendment 148. The analysis used the generic source term inventory approved for use under the CLTR. Analysis methods and other inputs were not changed from those used in Amendment 148.

The post-CRDA EAB, LPZ, and CR doses are within the applicable regulatory limits with the results and regulatory criteria summarized in Table 2.9-3.

Main Steam Line Break Accident (MSLBA) Outside Containment

The post-MSLBA EAB, LPZ, and CR doses were re-analyzed for EPU conditions. Because no fuel damage occurs during a MSLBA at Monticello, the released activity is associated with the steam and liquid discharged from the break. The normal liquid coolant and main steam noble gas; fission product and activation product source terms are affected by the EPU. However these normal sources are not significant contributors to the radiological consequences of the MSLBA. The doses were calculated at the technical specification limits for equilibrium, or pre-incident spiked radioiodine concentrations and included consideration of noble gases.

During the development of the Monticello license application for conversion to the Alternative Source term, NRC Regulatory Issue Summary RIS 2006-04 "Experience With Implementation Of Alternative Source Terms" had not been issued and the analysis at that time did not consider the release of coolant cesium. The calculations were revised to include cesium in the coolant in accordance with the RIS and it was confirmed that these minor changes in coolant chemistry do not affect the dose consequences of the MSLBA.

Analysis methods and other inputs were not changed from those used in Amendment 148. The post MSLBA EAB, LPZ, and CR doses are within the applicable regulatory limits with the results and regulatory criteria summarized in Tables 2.9-4 and 2.9-5.

Conclusion

NSPM has evaluated and accounted for the effects of the proposed EPU on the accident analyses. The evaluation concludes that the plant site and the dose-mitigating ESFs remain acceptable with respect to the radiological consequences of postulated DBAs because, as set forth above, the calculated total effective dose equivalent (TEDE) at the exclusion area boundary (EAB), at the low population zone (LPZ) outer boundary, and in the control room meet the exposure guideline values specified in 10 CFR 50.67 and the current licensing basis. Therefore, the proposed EPU is acceptable with respect to the radiological consequences of DBAs.

Table 2.9-1 LOCA Radiological Consequences

	TEDE Dose (REM)			
	Receptor Location			
	CR	EAB	LPZ	TSC
Calculated Dose CLTP[1]	3.40	1.31	1.72	0.77
Calculated Dose EPU[2]	3.80	1.46	1.99	0.83
Allowable TEDE Limit[3]	5	25	25	5

Table 2.9-2 FHA Radiological Consequences

	TEDE Dose (REM)		
	Receptor Location		
	CR	EAB	LPZ
Calculated Dose CLTP[1]	4.29	1.61	0.31
Calculated Dose EPU[2]	4.67	1.74	0.34
Allowable TEDE Limit[3]	5	6.3	6.3

Table 2.9-3 CRDA Radiological Consequences

	TEDE Dose (REM)		
	Receptor Location		
	CR	EAB	LPZ
Calculated Dose CLTP[1]	1.70	1.73	0.79
Calculated Dose EPU[2]	1.86	1.96	0.90
Allowable TEDE Limit[3]	5	6.3	6.3

Tables 2.9-1, 2.9-2 and 2.9-3 notes:

1. CLTP Power Level Assumption = 1880 MWt x 1.02 = 1918 MWt
2. EPU power Level Assumption = 2004 MWt x 1.02 = 2044 MWt
3. RG 1.183 Table 6

Table 2.9-4 MSLBA Pre-Incident Iodine Spike Radiological Consequences

	TEDE Dose (REM) Receptor Location		
	0 – 2 hr CR	30 day EAB	30 day LPZ
Calculated Dose CLTP[1]	3.25	1.05	0.20
Calculated Dose EPU[2]	3.25	1.05	0.20
Allowable TEDE Limit[3]	5	25	25

Table 2.9-5 MSLBA Equilibrium Iodine Concentration Radiological Consequences

	TEDE Dose (REM) Receptor Location		
	0 – 2 hr CR	30 day EAB	30 day LPZ
Calculated Dose CLTP[1]	0.33	0.11	0.02
Calculated Dose EPU[2]	0.33	0.11	0.02
Allowable TEDE Limit[3]	5	2.5	2.5

Tables 2.9-4 and 2.9-5 notes:

1. CLTP Power Level Assumption = 1880 MWt x 1.02 = 1918 MWt
2. EPU power Level Assumption = 2004 MWt x 1.02 = 2044 MWt
3. RG 1.183 Table 6

2.10 Health Physics

2.10.1 Occupational and Public Radiation Doses

Regulatory Evaluation

The NRC's acceptance criteria for occupational and public radiation doses are based on 10 CFR Part 20 and GDC-19.

Specific NRC review criteria are contained in SRP Sections 12.2, 12.3, 12.4, and 12.5, and other guidance provided in Matrix 10 of RS-001.

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC listed in the Regulatory Evaluation above, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-11. Current GDC-19 is applicable to Monticello as described in USAR Sections 5.3.5, 6.7.2, 12.3.1.6, and 14.7.

Technical Evaluation

Onsite Radiation Levels

Onsite radiation levels will increase by approximately the percentage increase in power level as a result of EPU. For conservatism, many aspects of the plant were originally designed for higher-than-expected radiation sources. Thus, the increase in radiation levels is not expected to affect radiation zoning or shielding in the various areas of the plant because it is offset by conservatism in the original design, source terms used and analytical techniques. The topics addressed in this evaluation are normal operational radiation levels, post-operating (normal shutdown) radiation levels, and post-accident radiation levels.

Normal operational radiation levels

The current normal operating radiation levels evaluated for Monticello are based on dose rate measurements at various locations during plant operation at CLTP conditions. USAR 12.3.1.6 described the design bases for shielding, "The offgas system shielding is based on a stack release rate of 260,000 μCi/sec. Reactor water fission product concentrations and activated corrosion products were assumed to be the maximum values expected; 8.0 μCi/cc, and 0.07 μCi/cc, respectively." These design criteria are not approached during normal plant operation and will remain conservative at EPU. Based on this design margin, the increase in radiation levels is not expected to affect radiation zoning. Table 2.10-1 provides the results of the evaluation of operation at EPU conditions on area radiation levels during normal operation.

The normal operating radiation levels at CLTP conditions were evaluated and determined to generally increase by approximately 13 percent. Individual worker exposures can be maintained within acceptable limits by controlling access to radiation areas using the site ALARA (As Low as Reasonably Achievable) program. Procedural controls compensate for the increased radiation levels.

Monticello has previously implemented zinc injection to limit the increase in normal radiation doses from the implementation of hydrogen water chemistry. An evaluation was performed of potential changes due to increased production rate and shorter transit times for N-16 in reactor steam. The evaluation concluded that areas close to the reactor will see no change in N-16 dose rates due to the effects of dilution from the increased steam flow rate. The effects of transit time become more significant further down the steam lines, in the turbines and condenser. The evaluation predicts about a 3 percent dose rate increase at the HP Turbine and about 9 percent entering the condenser. Sufficient holdup time will still be provided in the condenser to result in low N-16 radiation levels in condensate and feedwater systems. SJAE and offgas system steam piping N-16 radiation levels are expected to increase by 31-34 percent, but these areas are shielded based on the design fission product offgas rate which is several orders of magnitude above current operating levels, so this effect is not considered significant.

The increased normal radiation levels were evaluated and determined to have no adverse effect on safety-related plant equipment as indicated in Sections 2.2.5 and 2.3.1.

Post-operation radiation levels

Post-operation radiation levels in most areas of the plant increase by no more than 13 percent. Post-Operation Radiation levels are generally much lower than present during operation. Evaluations were conservatively performed assuming a large increase in moisture carryover with increased carryover of radioactivity and deposition in BOP systems. This could result in increased radiation levels in local areas of BOP piping equipment by as much as 1130 percent (assumes a 13 percent increase in production of contaminants and a ten-fold increase in carryover and deposition). This buildup would occur over time. Plant radiation surveys should provide prompt detection of these conditions. Work planning using ALARA principles will permit occupational doses to be managed within the limits of 10 CFR 20. Table 2.10-2 provides the results of the evaluation on shutdown radiation levels after operation at EPU conditions.

Individual worker exposures can be maintained within acceptable limits by controlling access to radiation areas using the site ALARA program. Procedural controls compensate for increased radiation levels. Radiation measurements will be made at selected power levels and following plant shutdown to ensure the protection of personnel.

Section 2.9.2 addresses the accident doses for the Main Control Room.

Off-Site Doses

The primary sources of normal operation offsite doses at Monticello are airborne releases from the Offgas System and gamma shine from the plant turbines. Additional potential sources of offsite doses include liquid effluent releases and doses resulting from storage and transfer of radioactive materials and wastes. The effects of EPU on offsite doses have been evaluated.

The increase in gaseous emissions at EPU is expected to result from the increased production rate of gaseous fission products and activation products. This increase is proportional to the increase in rated thermal power or approximately 12.9 percent. The rate of production of gaseous waste remains within the design capacity of the gaseous waste processing and storage systems. Current gaseous emission release rates are documented in annual radioactive effluent release reports and are a small fraction of the regulatory limits of 10 CFR 50 Appendix I and 40 CFR 190. An increase of 12.9 percent in dose would remain a small fraction of the regulatory limits.

The EPU increase in steam flow results in higher levels of N-16 and other activation products in the turbines. The increased flow rate and velocity, which result in shorter travel times to the turbine and less radioactive decay in transit, lead to higher radiation levels in and around the turbines and offsite skyshine dose. The evaluation indicates that EPU may result in a maximum sky shine source dose rate increase of up to 34.4 percent. The evaluation determined a very conservative upper bound for any increase in offsite dose due to sky shine at EPU conditions is less than 6 mrem/yr, which is well within the 40 CFR 190 limit of 25 mrem/yr. Maximum dose rates at plant boundaries are expected to remain below the 10 CFR 20 10.1302 maximum dose rate of 2 mrem/hr.

Any discernible increase in radiation as a result of increased N-16 would be measured on the site environmental dosimeter monitoring stations. Current offsite radiation levels are reported in annual Radiological Environmental Monitoring reports, which have concluded that operation of Monticello has no effect on ambient gamma radiation levels, and is not distinguishable from background radiation. This supports a conclusion that the evaluation discussed above is conservative and it is likely that there will be no measurable increase in offsite doses at EPU conditions. Offsite doses will remain well within the limits of 10 CFR 20, 10 CFR 50, Appendix I, and 40 CFR 190.

No radioactive liquid effluent has been intentionally released from MNGP since 1972. The evaluation indicates liquid radwaste processing systems will retain significant excess processing margin at EPU conditions. Therefore, the plant capability of maintaining its zero-discharge policy (USAR 9.2.3.1) for liquid effluents is not affected by operation at EPU.

Operation at EPU conditions may increase the need for truck transportation for disposal of solid radwaste by up to one additional truck per year. The solid radwaste system is designed to process, package, store, monitor, and provide shielded storage facilities for solid wastes to allow for radioactive decay and/or temporary storage prior to shipment from the plant for off-site disposal. The solid radioactive wastes are shipped off-site in vehicles equipped with adequate shielding to comply with Department of Transportation (DOT) regulations. Code of Federal Regulations Title 10, Parts 20, 61, 70 and 71 also apply. Based on this, implementation of EPU is not expected to result in any significant increase in offsite doses or exposures due to the storage, transportation, and disposal of radioactive materials and waste.

Conclusion

NSPM has evaluated the effects of the proposed EPU on radiation source terms and plant radiation levels. The evaluation concludes that any increases in radiation doses will be maintained as low as reasonably achievable. The evaluation further concludes that the proposed EPU meets the requirements of 10 CFR Part 20 and current licensing basis. Therefore, the proposed EPU is acceptable with respect to radiation protection and ensuring that occupational radiation exposures will be maintained as low as reasonably achievable.

Table 2.10-1 Monticello Area Radiation Levels During Normal Operation

Area	EPU Result
Reactor Building	Radiation levels generally increase in proportion to power increase of 13 percent.
RCIC Room	No change during system operation.
HPCI Room	No change during system operation.
Turbine Building Low Dose Areas	Areas may see 10 percent increase due to N-16. Some very low dose areas may see increased deposition due to moisture carryover.
Air Ejector Room	Increase by up to 25 percent to 33 percent.
Turbine Basement Condenser Area	Steam areas increase by up to 9 percent due to N-16.
Turbine Volume from 951′ to 961′ El	Increase by 2 percent to 9 percent due to N-16.
Turbine Volume from 961′ El to 1004′ El	Increase by 2 percent to 9 percent due to N-16.

Table 2.10-2 Monticello Post Operation Area Radiation Levels

Area	EPU Increase
Reactor Building	Shutdown radiation levels generally increase in proportion to power increase of 13 percent.
BOP	Shutdown radiation levels are expected to remain low. Due to increased moisture carryover some areas may see increased deposition of radioactivity that could create localized increases up to 1130 percent.

2.11 Human Performance

2.11.1 Human Factors

Regulatory Evaluation

The area of human factors deals with programs, procedures, training, and plant design features related to operator performance during normal and accident conditions.

The NRC's acceptance criteria for human factors are based on GDC-19, 10 CFR 50.120, 10 CFR Part 55, and the guidance in GL 82-33.

Specific NRC review criteria are contained in SRP Sections 13.2.1, 13.2.2, 13.5.2.1, and 18.0.

Monticello Current Licensing Basis

The general design criteria listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Monticello principal design criteria predate these criteria. The Monticello principal design criteria are listed in 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 (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC listed in the Regulatory Evaluation above, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-11. Current GDC-19 is applicable to Monticello as described in USAR Sections 5.3.5, 6.7.2, 12.3.1.6, and 14.7.

Technical Evaluation

In accordance with RS-001, Review Standard for Extended Power Uprates, Revision 0, December 2003 Section 2.11.1, five specific questions are identified associated with the human factors area. Each question has been included below with the applicable response.

1. Changes in Emergency and Abnormal Operating Procedures

Describe how the proposed EPU will change the plant emergency (EOP) and abnormal (AOP) operating procedures.

Response:

The Monticello 10 CFR 50 Appendix B plant procedure program governs changes to the AOPs and EOPs. The procedure change program and operator training program (discussed in question 5) will assure that operator performance will not be adversely affected by the proposed EPU. The following describes the procedure changes that will be implemented prior to operation at uprated conditions and/or installation of the associated modification.

The following are the AOP procedural changes:

- Turbine backpressure limits have changed as a result of modifications to the low-pressure turbines. These requirements will be incorporated into the Decreasing Condenser Vacuum AOP. The new turbine backpressure limits are slightly less restrictive at full EPU conditions, are slightly more restrictive at intermediate power conditions and are unchanged at low power conditions.
- The Station Blackout (SBO) analysis was changed to include using the HPCI suction from the Condensate Storage Tanks (CST). The AOP will be revised to require the operator to align the HPCI suction to the Condensate Storage Tanks 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.

The following are the EOP procedural 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 Pressure Suppression Pressure curve in the EOPs will be revised to reflect the increase in reactor power and increase in decay heat loading.

2. Changes to Operator Actions Sensitive to Power Uprate

Describe any new operator actions needed as a result of the proposed EPU. Describe changes to any current operator actions related to emergency or abnormal operating procedures that will occur as a result of the proposed EPU. (SRP Section 18.0) (i.e., Identify and describe operator actions that will involve additional response time or will have reduced time available. Your response should address any operator workarounds that might affect these response times. Identify any operator actions that are being automated or being changed from automatic to manual as a result of the power uprate. Provide justification for the acceptability of these changes).

Response:

There are no new operator actions required as a result of the proposed EPU. One action that was performed as part of the EOP actions is now included in the SBO analysis (Align HPCI suction from 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 requirement as all actions are performed in the main control room. The action is to open a knife switch and align three motor operated valves at the HPCI panel.

There are no operator workarounds created, modified or affected as a result of EPU.

3. Changes to Control Room Controls, Displays and Alarms

Describe any changes the proposed EPU will have on the operator interfaces for control room controls, displays, and alarms. For example, what zone markings (e.g. normal, marginal and out-of-tolerance ranges) on meters will change? What setpoints will change? How will the operators know of the change? Describe any controls, displays, alarms that will be upgraded from analog to digital instruments as a result of the proposed EPU and how operators will be tested to determine they could use the instruments reliably. (SRP Section 18.0).

Response:

- 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.
- The Plant Process Computer alarm values for monitoring reactor power are being increased to reflect the EPU rated thermal power (RTP) levels.

4. Changes on the Safety Parameter Display System

Describe any changes to the safety parameter display system (SPDS) resulting from the proposed EPU. How will the operators know of the changes? (SRP Section 18.0)

Response:

SPDS equipment is not being modified for the EPU. There may be minor changes to the SPDS displays, these include:

- The Heat Capacity Temperature Limit 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.
- The turbine exhaust pressure limit display to reflect the change in turbine backpressure requirements.

SPDS changes will be communicated to the operators as part of the larger EPU training.

5. Changes to the Operator Training Program and the Control Room Simulator

Describe any changes to the operator training program and the plant referenced control room simulator resulting from the proposed EPU, and provide the implementation schedule for making the changes. (SRP Sections 13.2.1 and 13.2.2)

Response:

The Operations Training Group provides training on all modifications installed that affect unit operation prior to operation with the modification. The operator training is presented in the classroom and on the simulator. The content of the training for the EPU is dependent on the EPU power ascension plan and the EPU related modification implementation schedule.

The major EPU related change for the control room operators involves the installation of the 13.8 kv electrical buses.

Operator training, licensed and non-licensed operator training, will be provided during the training cycle prior to the refueling outage (RFO) and will focus on plant modifications, procedure changes, startup test procedures, and other aspects of 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 performance of the new high-pressure turbine and 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.

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, are 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 HP 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.

Conclusion

NSPM has evaluated the changes to operator actions, human-system interfaces, procedures, and training required for the proposed EPU and (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 evaluation concludes that Monticello will continue to meet the requirements of the current licensing basis, 10 CFR 50.120, and 10 CFR Part 55 following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to 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 purpose of the 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 design criteria at EPU conditions.

The NRC's acceptance criteria for the proposed EPU test program are based on 10 CFR Part 50, Appendix B, Criterion XI, which requires establishment of a test program to demonstrate that SSCs will perform satisfactorily in service.

Specific NRC review criteria are contained in SRP Section 14.2.1.

Monticello Current Licensing Basis

USAR, Appendix D, "Pre-Operational and Startup Tests," provides an overview of the initial power ascension test program.

Technical Evaluation

Testing is required for the initial power ascension following the implementation of EPU. The topics addressed in this section are:

[[

[illegible]	[illegible]	[illegible]
		]]

Based on the analyses and GEH BWR experience with uprated plants, a standard set of tests has been established for the initial power ascension steps of EPU. These tests, which supplement the normal Technical Specification testing requirements, are as follows:

- Testing will be performed in accordance with the Technical Specifications Surveillance Requirements on instrumentation that is re-calibrated for EPU conditions. Overlap between the IRM and APRM will be assured.
- Steady state data will be taken at points from 90% up to 100% of the CLTP, so that system performance parameters can be projected for EPU power before the CLTP is exceeded.
- EPU power increases above the 100% CLTP will be made along an established flow control/rod line in increments of equal to or less than 5% power. Steady state operating data, including fuel thermal margin, will be taken and evaluated at each step. Routine measurements of reactor and system pressures, flows, and vibration will be evaluated from each measurement point, prior to the next power increment. Radiation measurements will be made at selected power levels to ensure the protection of personnel.
- Control system tests will be performed for the reactor FW/reactor water level controls, pressure controls, and recirculation flow controls, if applicable. These operational tests will be made at the appropriate plant conditions for that test at each of the power increments, to show acceptable adjustments and operational capability.
- Steam separator-dryer performance will be confirmed to be within limits by determination of steam moisture content as required during power ascension testing.
- Testing will be performed to confirm the power level near the turbine first-stage scram bypass setpoint.
- Steam dryer/separator performance will be confirmed within limits by determination of steam moisture content as required during power ascension testing.

The same performance criteria will be used as in the original power ascension tests, unless they have been replaced by updated criteria since the initial test program. Because steam pressure and core flow have not been changed and recirculation flow may only slightly increase for EPU, testing of system performance affected by these parameters is not necessary with the exception of the tests listed above.

The EPU testing program at Monticello, which is based on the Monticello-specific EPU power ascension and Technical Specifications, is confirmed to be consistent with the generic description provided in the CLTR.

Monticello does not plan to perform large transient testing as part of EPU implementation. The justification for not performing large transient testing is provided as a stand-alone attachment to the EPU LAR.

Further, the important nuclear characteristics required for transient analysis are confirmed by the steady state physics testing. Transient mitigation capability is demonstrated by other tests required by the Technical Specifications. In addition, the limiting transient analyses are included as part of the reload licensing analysis.

Conclusion

NSPM has provided the EPU test 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. The proposed EPU test program provides adequate assurance that the plant will operate in accordance with design criteria and that SSCs affected by the proposed EPU, or modified to support the proposed EPU, will perform satisfactorily in service. Therefore, the proposed EPU test program is acceptable.

2.13 Risk Evaluation

2.13.1 Risk Evaluation of EPU

Regulatory Evaluation

The licensee conducted a risk evaluation to (1) demonstrate that the risks associated with the proposed EPU are acceptable and (2) determine if "special circumstances" are created by the proposed EPU. As described in Appendix D of SRP Chapter 19, special circumstances are present if any issue would potentially rebut the presumption of adequate protection provided by the licensee to meet the deterministic requirements and regulations.

The NRC's risk acceptability guidelines are contained in RG 1.174.

Specific Monticello review guidance is contained in Matrix 13 of RS-001 and its attachments.

Monticello Current Licensing Basis

The USAR includes a summary description of the evaluation of plant risk including the Individual Plant Examination (IPE) and various updates in Section 14.1, Summary Description. NEDC-32546P, Revision 1, "Power Rerate Safety Analysis Report," December 1997 included an evaluation of the effect of the previous rerate on the IPE and the Individual Plant Examination of External Events (IPEEE) in Section 10.5, "Individual Plant Evaluation (IPE)."

Technical Evaluation

A Probabilistic Risk Assessment (PRA) of the EPU operating and shutdown conditions has been conducted. The revised PRA model includes the changes that arise from EPU conditions. The PRA report is included as an enclosure to this submittal. The PRA report provides the detailed analysis for the PRA results reported below. The report addresses Initiating Event Frequencies, Success Criteria, PRA Quality, and Component Reliability in detail. The report results are consistent with the CLTR description and analysis of this topic.

The key results from the EPU PRA are as follows:

- The PRA determined that the changes to the Monticello risk profile due to the effect of EPU operations and shutdown conditions are acceptably small.
- The ΔCDF and ΔLERF results are within Region III (very small change) of the regulatory guidelines of Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis." The EPU is estimated to increase the Monticello internal events PRA CDF from the base value of 7.32 E-6/yr to 7.89 E-6/yr, an increase of 5.67 E-7 (7.8 %). The LERF increases from the base value of 3.64 E-7/yr to 3.94 E-7/yr, an increase of 3.00 E-8/yr (8.2 %).
- The Level 1 PRA success criteria effects due to the EPU are as follows:
  - o 7 of 8 SRVs are required for the EPU condition for RPV initial overpressure protection during an ATWS scenario.

  - CRDH as the only early injection source using 2 CRDH pumps at nominal flow now requires that the RPV be depressurized (the use of enhanced flow CRDH with a single CRDH pump is unchanged for the EPU).
- Operator action times for some mitigation sequences are reduced slightly; however, the associated increases in failure probabilities are acceptably small.
- The PRA included conservative sensitivity studies to determine the risk associated with loss of containment overpressure that supports ECCS pump NPSH at EPU conditions. These studies show that both the ΔCDF and ΔLERF remain within Region III of Regulatory Guide 1.174.

The PRA methodology was also used to determine the effect of additional failures on transient response for the purposes of item II.K.3.44 of NUREG-0737. This is in accordance with staff guidance contained in the SER enclosed within NEDC-32523P-A, Supplement 1. The previous Monticello EPU included an evaluation of these failures based on design-basis containment analyses techniques. See Section 14.10.1 of the Monticello USAR. In keeping with the SER, PRA calculations were performed to evaluate a Loss of Feedwater Event at EPU conditions assuming a Stuck Open Relief Valve and using the RCIC System as the high-pressure injection source. The result of these evaluations demonstrated that adequate core cooling and containment integrity are maintained throughout the mitigation sequence. See Appendix E of the PRA report for additional information.

Conclusion

NSPM has evaluated the risk implications associated with the implementation of the proposed EPU. The evaluation indicates that the risks associated with the proposed EPU are acceptable and do not create the "special circumstances" described in Appendix D of SRP Chapter 19. Therefore, the risk implications of the proposed EPU are acceptable.

3 REFERENCES

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2. GE Nuclear Energy, “Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate,” Licensing Topical Reports NEDC-32424P-A, Class III (Proprietary), February 1999; and NEDO-32424, Class I (Non-proprietary), April 1995.
3. GE Nuclear Energy, “Generic Evaluations of General Electric Boiling Water Reactor Extended Power Uprate,” Licensing Topical Reports NEDC-32523P-A, Class III, February 2000; NEDC-32523P-A, Supplement 1 Volume I, February 1999; and Supplement 1 Volume II, April 1999.
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Appendix – A

Limitations from Safety Evaluation for LTR NEDC-33173P

Number	Title	Limitation Description	Disposition
1	TGBLA/PANAC Version	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 TGBLA06/PANAC11 or later NRC-approved version of neutronic method.	As indicated in Table 1-1, TGBLA06/PANAC11 was used to support the plant-specific Monticello EPU analysis. These codes were used to simulate the reactor core response and that feed into the downstream safety analyses supporting operation at EPU conditions for Monticello.
2	3D Monicore	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.	No reliance on TGBLA04/PANAC10 for Monticello.
3	Power to Flow Ratio	Plant-specific EPU and expanded operating domain applications will confirm that the core thermal power to core flow ratio will not exceed 50 MWt/Mlbm/hr at any statepoint in the allowed operating domain. For plants that exceed the power-to-flow value of 50 MWt/Mlbm/hr, the application will provide power distribution assessment to establish that neutronic methods axial and nodal power distribution uncertainties have not increased.	As discussed in Section 2.8.2.5 and Table 2.8-2, GEH has confirmed that the core thermal power to total core flow ratio does not exceed 50 MWt/Mlbm/hr at 100% EPU power and 100% rated flow in the new operating domain. This disposition is consistent with Reference A-5.

Number	Title	Limitation Description	Disposition
4	SLMCPR1	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.	As stated in Section 2.8.2.3.1, the SLMCPR will include a 0.02 adder for single loop operation.
5	SLMCPR2	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.	Not applicable to EPU.
6	R-Factor	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.	As stated in Section 2.8.2.5, the GE14 bundle R-factors are consistent with Monticello hot channel axial void conditions for EPU operation.
7	ECCS-LOCA 1	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.	As stated in Section 2.8.5.6.2, the small and large ECCS-LOCA analyses conducted for Monticello EPU includes top-peaked and mid-peaked power shapes in establishing the MAPLHGR and determining both the licensing bases and upper bound PCTs. The limiting small and large break licensing basis and upper bound PCTs are reported in PUSAR Section 2.8.5.6.2 and Table 2.8-5.

Number	Title	Limitation Description	Disposition
8	ECCS-LOCA 2	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 Reference A-2 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 SRLR will include both the limiting statepoint ECCS-LOCA results and the rated conditions ECCS-LOCA results.	Not applicable to EPU.
9	Transient LHGR 1	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 and the limiting GdO_2 rods.	As stated in PUSAR Section 2.8.5, acceptable fuel rod thermal-mechanical performance for both UO_2 and GdO_2 fuel rods was demonstrated. Results for all AOO pressurization transient events analyzed, including equipment out of service, showed at least 10 percent margin to the fuel centerline melt and the one percent cladding circumferential plastic strain acceptance criteria. The minimum calculated margin to the fuel centerline melt criterion was 26%. The minimum calculated margin to the cladding strain criterion was 35%. Fuel rod thermal-mechanical performance will be evaluated as part of the reload analysis performed for the cycle specific core. Documentation of acceptable fuel rod thermal-mechanical response will be included in the SRLR or COLR.

Number	Title	Limitation Description	Disposition
10	Transient LHGR 2	Each EPU and MELLLA+ fuel reload 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.	GEH will document the calculation results of the analyses demonstrating compliance to transient T-M acceptance criteria in the plant-specific SRLR.
11	Transient LHGR 3	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 is no longer required.	See response to Limitation 9.

Number	Title	Limitation Description	Disposition
12	LHGR and Exposure *Qualification*	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 GESTAR II or in a T-M licensing LTR. PRIME (a newly developed T-M code) has been submitted to the NRC staff for review (Reference A-3). Once the PRIME LTR and its application are approved, future license applications for EPU and MELLLA+ referencing LTR NEDC-33173P must utilize the PRIME T-M methods.	The Monticello EPU license application is *not based on the PRIME LTR. Therefore,* the Limitation is not applicable to Monticello.

Number	Title	Limitation Description	Disposition
13	Application of 10 Weight Percent Gd	Before applying 10 weight percent Gd to licensing applications, including EPU and expanded operating domain, the NRC staff needs to review and approve the T-M LTR 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 weight percent Gd for modern fuel designs, NRC must review and approve 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 weight percent Gd qualifications submittal can supplement this report.	The Monticello EPU bundle design will utilize less than 10% gadolinium. Therefore, the Limitation is not applicable to Monticello.
14	Part 21 Evaluation of GESTR-M Fuel Temperature Calculation	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. GE submitted the T-M Part 21 evaluation, which is currently under NRC staff review. Upon completion of its review, NRC staff will inform GE of its conclusions.	Not applicable.

Number	Title	Limitation Description	Disposition
15	Void Reactivity 1	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.	The Monticello EPU license application is not based on TRACG. Therefore this limitation is not applicable.
16	Void Reactivity 2	A supplement to TRACG /PANAC11 for AOO is under NRC staff review (Reference A-4). 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 SE approving NEDE-32906P, Supplement 3, "Migration to TRACG04/PANAC11 from TRACG02/PANAC10," May 2006 (Reference A-4). This limitation applies until the new TRACG/PANAC methodology is approved by the NRC staff.	The Limitation is not applicable to the current Monticello EPU license application because the application is not based on Supplement 3 to NEDC-32906P.

Number	Title	Limitation Description	Disposition
17	Steady-State 5 Percent Bypass Voiding	The instrumentation specification design bases limit the presence of bypass voiding to 5 percent (LRPM 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.	GEH will provide the highest calculated bypass voiding at any LPRM level with the plant-specific SRLR consistent with Reference A-5.
18	Stability Setpoints Adjustment	The NRC staff concludes that the presence bypass voiding at the low-flow conditions where instabilities are likely can result in calibration errors of less than 5 percent for 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 this SE, discussing the stability methodology.	As stated in PUSAR Section 2.8.3.1, the OPRM setpoints will account for the 5% calibration errors to address the issue of the bypass voiding at low-flow conditions. As an illustration, the OPRM setpoint of 1.15 based on the PUSAR demonstration analysis would be reduced to 1.14 when accounting for the 5% calibration error. The APRM penalty is not applicable for the OPRM system.

Number	Title	Limitation Description	Disposition
19	Void-Quality Correlation 1	For applications involving PANCEA/ODYN/ISCOR/TASC for operation at EPU and MELLLA+, an additional 0.01 will be added to the OLMCPR, until such time that GE 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.	The Monticello reload core analysis will include a 0.01 adder to the OLMCPR consistent with Reference A-5, pending GEH's resolution of the void quality correlation.
20	Void-Quality Correlation 2	The NRC staff is currently reviewing Supplement 3 to NEDE-32906P, "Migration to TRACG04/PANAC11 from TRACG02/PANAC10," dated May 2006 (Reference A-4). 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 LTR NEDC-32906P (Reference A-4) will be applicable as approved.	The Limitation is not applicable to the current Monticello EPU license application because the application does not rely upon the ongoing NRC review.
21	Mixed Core Method 1	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 LTR NEDC-33173P (Reference A-1) for mixed core application.	This limitation is not applicable to Monticello because the EPU core will consist of GE14 fuel exclusively and is not a mixed vendor core.

Number	Title	Limitation Description	Disposition
22	Mixed Core Method 2	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 water crosses • Gd rods simultaneously adjacent to water and vanished rods • 11x11 lattices • 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.	This limitation is not applicable to Monticello because the EPU core will consist of GE14 fuel exclusively and is not a mixed vendor core.

Number	Title	Limitation Description	Disposition
23	MELLLA+ Eigenvalue Tracking	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 TIP comparison), • Bundle power distribution (measured and calculated TIP comparison), • Thermal margin, • Core flow and pressure drop uncertainties, and • The MIP 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.	The Limitation is applicable to MELLLA+ and is not applicable to the current Monticello EPU license application.

Number	Title	Limitation Description	Disposition
24	Plant Specific Application	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 BOC, MOC, and 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.	The requested plots of the key parameters and the quarter core maps are provided in PUSAR Figures 2.8-1 thru 2.8-18.

References:

A-1 MFN 08-089, Ho K. Nieh, Deputy Director, Division of Policy and Rulemaking, Office of Nuclear Reactor Regulation to Robert E. Brown (GEH), “Final Safety Evaluation For General Electric (GE)-Hitachi Nuclear Energy Americas, LLC (GHNE) Licensing Topical Report (LTR) NEDC-33173P, "Applicability Of GE Methods To Expanded Operating Domains" (TAC NO. MD0277),” January 17, 2008.

A-2 GE Letter (MFN 05-141), L. M. Quintana to NRC, NEDC-33006P, Revision 2, “General Electric Boiling Water Reactor Maximum Extended Load Line Limit Analysis Plus,” November 28, 2005. (ADAMS Accession No. ML053360526).

A-3 GNF Letter (FLN-2007-001), A. A. Lingenfelter to NRC, “The PRIME Model for Analysis of Fuel Rod Thermal-Mechanical Performance,” January 19, 2007. (ADAMS Package Accession No. ML070250414).

A-4 Licensing Topical Report NEDE-32906P, Supplement 3, “Migration to TRACG04/PANAC11 from TRACG02/PANAC10,” May 2006.

A-5 GEH Letter (MFN 08-693), “Implementation of Methods Limitations - NEDC-33173P (TAC No. MD0277),” September 18, 2008.

Enclosure 8 to L-MT-08-052

Planned Modifications for Monticello Extended Power Uprate

ENCLOSURE 8

Planned Modifications for Monticello Extended Power Uprate

Monticello began preparation for the Extended Power Uprate (EPU) project during refueling outage (RFO) 23 in 2007 with the installation of instrumentation necessary to gather information for various evaluations required to support the EPU license amendment request. Monticello plans to install modifications in two phases to support EPU implementation. The modifications that were installed and are planned for future installation, including a summary description of the scope for each modification, are listed in the following three tables:

Table 8-1, "Pre-EPU Modifications Installed During RFO23 (2007)"
Table 8-2, "EPU Phase I Modifications Planned for 2009 (primarily RFO24)"
Table 8-3, "EPU Phase II Modifications Planned for 2011 (primarily RFO25)"

These tables also include modifications that are not required for EPU but have been approved as part of the life cycle management (LCM) program. These LCM modifications are coordinated with the EPU project and will include design criteria that incorporate EPU conditions to maintain or improve performance margin of the respective systems.

These tables are provided for information only and are not commitments. The timing and scope of the modifications may change as detailed design and outage plans progress.

Table 8-1

Pre-EPU Modifications Installed During RFO23 (2007)	
Modification	**Description**
Steam Dryer Acoustic Monitoring	Installed strain gages on Main Steam Lines (MSL) for steam dryer acoustic monitoring.
Vibration Monitoring	Installed accelerometers on Main Steam (MS) and Feedwater (FW) piping for vibration monitoring.

Table 8-2

EPU Phase I Modifications Planned for 2009 (primarily RFO24)	
Modification	**Description**
1AR Transformer Replacement	Replace the existing 1AR transformer due to aging concerns. This is a life cycle management (LCM) modification and is not related to EPU
Power Range Neutron Monitoring System (PRNM) (Included for completeness only. NRC approval has been requested under a separate, prior submittal.)	Replace the existing GE analog system with a GE digital system. This is an LCM modification that includes appropriate design considerations to allow implementation of EPU.
GEZIP	Replace the existing zinc injection (GEZIP) skid with a new passive injection skid. This is an LCM modification.
Piping Requalification	Revise documentation to incorporate revised pressure and temperature ratings for specific piping systems affected by EPU. Modify supports as required by the analyses. Main Steam Relief Valve (MSRV) actuator upgrades due to obsolescence issues. Reweld the Main Steam lead drain pipe connection to the Navy nipple.
HP Turbine Replacement	Replace the existing High Pressure (HP) turbine steam path with a new rotor and diaphragms to accommodate increased steam flow under EPU conditions.
LP Turbine Modifications	Replacement of several diaphragm sets and one set of buckets in each low pressure (LP) turbine to accommodate increased steam flow under EPU conditions. Replacement of selected casing bolts. Evaluate and replace the extraction steam expansion joints.
Isophase Bus Cooling	Replace the existing isophase cooling skid with a new one sized for increased EPU heat loads. Add a new redundant isophase cooling skid to increase reliability.

EPU Phase I Modifications Planned for 2009 (primarily RFO24)	
Modification	**Description**
Electronic Pressure Regulator	Replace or respan the electronic pressure regulator to accommodate the increased main steam line pressure drop.
Cross Around Relief Valve (CARV) Replacement	Replace the existing CARVs and associated discharge piping to provide increased relieving capacity under EPU conditions.
Torus Attached Piping	Revise the documentation to incorporate new analyses for EPU conditions. Modify some existing supports to maintain stress limits under EPU conditions.
Main Steam Flow Transmitters	Respan or replace the main steam flow transmitters to accommodate increased flows under EPU conditions.
Feedwater Flow Transmitters and Pressure Control Instrumentation	Respan or replace the FW flow transmitters to maintain functionality with increased flow under EPU conditions. Respan or replace the existing Feedwater pressure control instrumentation to maintain functionality with increased flows and pressure drops under EPU conditions.
Feedwater Regulating Valves	Adjust the stroke of the feedwater regulating valves to provide additional range for operation at increased flow during Phase I of EPU.
Reactor Feedwater Pump Motor	Rerate the reactor feedwater pump motor to allow operation at increased flow during Phase I of EPU.
Feedwater Heater Drain and Dump Valve Replacement.	Replace the drain and dump valves on the feedwater heaters and drain coolers due to obsolescence issues. This is an LCM modification that will consider EPU conditions to enhance margin.
Inboard Main Steam Isolation Valve (MSIV) Solenoid Valve Replacement	Replace the solenoid valves on the inboard MSIVs to increase the margin between maximum containment pressure and minimum nitrogen supply pressure.
11 & 12 Drain Cooler and Feedwater Heater Rerate	Rerate the 11 and 12 drain coolers and feedwater heaters for EPU conditions.

EPU Phase I Modifications Planned for 2009 (primarily RFO24)	
Modification	**Description**
Main Transformer and Isophase Duct	Replace the existing main generator step-up transformer to provide increased operating margins under EPU conditions. Remove a branch connection on the isophase bus duct to remove a hot spot and reduce overall temperatures under EPU conditions. Modify main transformer fire protection system to support the new design.
Bricks in Bioshield	Remove bricks from the bioshield to improve margin for potential missiles in the drywell.
Thermowells in Main Steam Piping	Replace or remove the thermowells in main steam piping to insure appropriate margin for flow induced vibration.
Drywell Spray Flow	Evaluate and modify the system, if necessary, to provide capability to throttle drywell spray flow consistent with design bases analytical assumptions.
EQ Modifications	Replace torus wide range water level indication transmitters
Simulator Upgrades (1)	Upgrade the simulator to include new core and containment operational response in addition to the other EPU plant modifications.
Grid Modification	Add remote reactive 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 Agreement negotiated with MISO for Project G725 that adds the first phase EPU electrical output of approximately 620 MW (net).

(1) This modification will be installed prior to EPU LAR implementation

Revise Setpoints (1)	Revise main steam line (MSL) high flow isolation setpoint to maintain the current setpoint in terms of % MSL flow. Revise the turbine first stage pressure setpoint to accommodate the new HP turbine and EPU inlet conditions. Revise the maximum combined flow limiter setpoint to maintain the setting at 110% of steam flow. Revise main generator protective relay setpoints to accommodate EPU conditions. Revise the rod worth minimizer low, intermediate and high power setpoints to maintain settings in terms of % rated thermal power. Revise reactor water level scram setpoint to accommodate the increased differential pressure across the steam dryer. Revise the Average Power Range Monitor (APRM) simulated power scram setpoint to maintain the setting in terms of % rated thermal power. (2) Revise the APRM neutron flux high setdown setpoint to maintain the setting in terms of % rated thermal power. (2) Revise the APRM neutron flux high scram setpoint to maintain the setting in terms of % rated thermal power. (2)

(1) This modification will be installed prior to EPU LAR implementation
(2) (Included for completeness only. NRC approval has been requested under a separate, prior submittal.)

Table 8-3

EPU Phase II Modifications Planned for 2011 (primarily RFO25)	
Modification	**Description**
Reactor Feed Pump Replacement	Replace the existing reactor feedwater pumps with new pumps sized for EPU conditions. Replace the existing 4KV motors with new 13.8KV motors sized for EPU conditions. Upgrade the minimum flow piping and valves as necessary for the new pumps.
Reactor Feed Pump Discharge Check Valve Replacement	Replace the existing reactor feed pump discharge check valves due to obsolescence issues and to maintain flow margin under EPU conditions. This is an LCM modification.
Feedwater Regulating Valve Replacement	Replace the existing feedwater regulating valves with new ones sized for operation under EPU conditions.
Condensate Pump Upgrades	Replace the existing condensate pump internals with new assemblies sized for increased EPU flow rates. Replace the existing 4KV motors with a new 13.8KV motors sized for EPU operating conditions. Upgrade the minimum flow piping and valves as necessary for the new pumps. Raise hotwell water level to mitigate potential flashing and vortexing issues.
Condensate Demineralizer Replacement	Replace the existing condensate demineralizer vessels with new vessels to accommodate increased flow under EPU conditions. Replace the existing control panel with a new digital control panel.
Condensate Flow Transmitters	Respan or replace the condensate flow transmitters to accommodate increased flows under EPU conditions.

EPU Phase II Modifications Planned for 2011 (primarily RFO25)	
Modification	**Description**
New 13.8KV Bus Installation	This modification is an LCM modification to increase margin in the on site distribution system. Install a new 13.8 KV bus to supply the new FW and Condensate pump motors and new 13.8 KV reactor recirculation motor-generator (M-G) set motors or adjustable speed drives pending evaluation. Replace the existing 1R and 2R transformers with new transformers, to feed two 4.8KV and two 13.8KV busses. Install new 13.8/480V transformer to feed 131 and 141 motor control centers.
Replace the Recirculation M-G Set Motors	Replace the existing recirculation M-G set motors with new 13.8KV motors or adjustable speed drives pending the results of an evaluation. This is an LCM upgrade.
Generator Rewind	Rewind the existing main generator stator and rotor to provide increased capacity required for EPU operation.
Generator Hydrogen Coolers	Replace the generator hydrogen coolers to provide additional capacity for EPU operation.
Main Exciter Replacement	Replace the existing main generator exciter with a new one to maintain operating margin under EPU conditions. This is an LCM modification.
Feedwater Heater Replacement	Replace the existing 13, 14 and 15 feedwater heaters with new ones sized for EPU conditions.
Drain Cooler Replacement/Reanalysis	Replace, re-analyze or modify the existing 11 and 12 drain coolers to maintain margin under EPU conditions.
Moisture Separator Drain Tank Cooling	Provide condensate injection into the moisture separator drain tanks discharge piping to increase sub-cooling under EPU conditions. This will stabilize flow and eliminate control issues for the drain valves.

ENCLOSURE 8

EPU Phase II Modifications Planned for 2011 (primarily RFO25)	
Modification	**Description**
Grid Modification	Add remote reactive 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 Agreement negotiated with MISO for Project G929 that adds the second phase EPU electrical output of approximately 671 MW (net).

Enclosure 9 to L-MT-08-052

Monticello Nuclear Generating Plant
Extended Power Uprate
Startup Test Plan

ENCLOSURE 9

MONTICELLO NUCLEAR GENERATING PLANT
EXTENDED POWER UPRATE STARTUP TEST PLAN

Table of Contents

1.0 Introduction

The following information supplements the Monticello Nuclear Generating Plant (MNGP) EPU License Amendment Request and provides additional information about startup testing using SRP 14.2.1 - Generic Guidelines for Extended Power Uprate Testing Programs, as a guide.

2.0 Purpose

2.1 Background

This document provides detailed information on the testing Northern States Power Company, a Minnesota corporation (NSPM) intends to perform following the Extended Power Uprate (EPU) implementation outages. The first implementation outage will be in 2009 and will upgrade plant equipment and load fuel sufficient to support operation at a maximum of 1870 MWt. The second implementation outage will be in 2011 and will upgrade plant equipment and load fuel sufficient to support operation at 2004 MWt.

Following each of these two implementation outages, NSPM will conduct testing to ensure the safe operation of the plant. The tests that NSPM intends to perform are described herein.

The required EPU startup testing for MNGP was determined using information from several sources:

- The startup and power ascension testing that was done during initial plant startup (1971). Information on this testing was obtained from the test plan and test procedure documents used at the time of initial startup.

- The testing that was done at the time of MNGP's stretch power uprate (1998). Information on this testing was obtained from the test plan and procedure documents used at the time.

- The guidance for extended power uprates in GE Topical Report NEDC 32424P-A "Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate" (also referred to as ELTR1), including the NRC's Requests for Additional Information (RAIs) and GE responses, and the NRC's staff position on the ELTR1 documented in NRC letter dated February 8, 1996.

- The guidance for extended power uprates in GE Topical Report NEDC 32523P-A "Generic Evaluations of General Electric Boiling Water Reactor Extended Power Uprate" (also referred to as ELTR2), including Supplement 1, Volumes 1 and 2 and the NRC's staff position on the ELTR2 documented in NRC letter dated September 14, 1998.

- The guidance for constant pressure power uprates (CPPUs) in GE Topical Report NEDC 33004P-A "Constant Pressure Power Uprate" (also referred to as CLTR), including the NRC's RAIs and GE responses, and the NRC's safety evaluation of the CLTR documented in NRC letter dated March 31, 2003.

- Information and data from plant transients that have occurred during MNGP's lifetime, as applicable.
- The NRC Standard Review Plan (SRP), NUREG-0800, Section 14.2.1 "Generic Guidelines for Extended Power Uprate Testing Programs." (The NRC's Review Standard for Extended Power Uprates, RS-001, identifies SRP Section 14.2.1 as guidance for EPU power ascension and testing.)
- Experience from other BWR plants that have implemented EPUs.

The use of multiple GE Topical Reports (ELTR1, ELTR2 and CLTR) comes about as a result of the manner in which GE developed this information. The ELTR1 and ELTR2 were developed first and envisioned that EPUs would include an increase in reactor pressure. However, EPU implementation is simplified if reactor pressure is kept constant, and BWR EPUs to date in the US have used a CPPU approach. To clarify this approach, GE prepared the CLTR.

With regard to startup testing, a key NRC comment on the CLTR (as expressed in their safety evaluation) is that the need for "large transient" testing (such as Main Steam Isolation Valve (MSIV) closure tests and turbine trip/generator load rejection tests) needs to be addressed on a plant-specific basis rather than a generic basis. Accordingly, this document addresses large transient testing.

2.2 Objective

This Enclosure describes the startup testing that MNGP will conduct associated with implementation of its EPU, including justification for not performing some large transient testing. The information is organized in a manner similar to SRP Section 14.2.1.

3.0 Summary of Conclusions

Based on the discussions in Section 4 and Tables 1 and 2 of this enclosure, MNGP will perform EPU testing appropriate to show that it will operate safely after implementation of the EPU and that SSCs with safety-related requirements or functions will perform satisfactorily after implementation of the EPU.

Initial Plant Startup Testing and Comparison to EPU Testing
Table 1 to this enclosure shows the initial plant startup testing performed at Monticello and compares it to the EPU startup testing. The test activities in Table 1 of this enclosure identify all startup testing that was performed during the initial plant startup in 1971. Per the guidance of SRP 14.2.1, Table 1 to this enclosure identifies all of the tests that were performed during initial startup and the power levels at which the tests were performed. There were no tests at lower power levels that would be invalidated by EPU. For each initial startup test at power $\geq$ 80 percent Original Licensed Thermal Power (OLTP), Table 1 indicates how that test is addressed in EPU testing or indicates the justification for not performing that test.

For MNGP, large transient testing (MSIV closure and turbine trip/generator load rejection) is not needed or warranted. The justification for this conclusion is a combination of plant experience/data from past transients, analysis results that predict acceptable response for these transients with margin, the absence of new thermal-hydraulic phenomena or system interactions, guidance in GE Topical Reports, and risk evaluations. Importantly, plant response to these two large transients has been demonstrated by data from MNGP transients including a 2001 MSIV closure event (from 1740 MWt or 98% CLTP) and a 2002 generator load rejection (from 1773 MWt or 100% Current Licensed Thermal Power (CLTP)). The plant responded as expected during both events. For both events, the power level equivalent to EPU power (87 percent for the MSIV closure event and 88.5 percent for the generator load rejection) exceeds the percentage power (75 percent OLTP for the MSIV closure and 50 percent OLTP for the generator load rejection) during initial startup testing. As a result, the transients experienced in 2001 and 2002 bound testing that would be performed to repeat that conducted during initial startup testing.

This enclosure provides a detailed discussion of this conclusion and justification (Section 4.3).

Tests Associated with EPU Plant Modifications
Modifications associated with EPU are described in Enclosure 8 of the EPU License Amendment Request. Testing related to changes will be performed in accordance with the plant's modification program. The intent of the testing associated with each modification is to make sure that the structures, systems and components (SSCs) affected by the modification will perform satisfactorily after the modification is implemented.

The use of a constant pressure power uprate (CPPU) approach minimizes the changes in operating conditions that plant systems will experience with EPU. Further, this approach minimizes new interdependencies or interactions between systems. The guidance in the CLTR was utilized to determine the extent of testing associated with the EPU modifications.

EPU Testing
Table 2 summarizes the testing that will be performed for EPU.

4.0 Testing Evaluations

4.1 Comparison to MNGP Startup Test Program [SRP 14.2.1, III.A]

Table 1 to this enclosure identifies all startup testing that was performed during initial plant startup in 1971. The tests include all those that were performed with the vessel head off, during heat-up, and during power ascension. The tests performed during power ascension are then sub-divided into two groups; tests performed below 80 percent power, and those performed at or above 80 percent power.

Tests performed below 80 percent OLTP were reviewed to ensure that none of those tests will be invalidated by the EPU. Tests that were performed at or above 80 percent OLTP were evaluated to determine that a) the original tests are not invalidated, and b) EPU testing adequately addresses these tests.

Power ascension tests performed at <80 percent OLTP

In accordance with SRP 14.2.1, paragraph III.A.2, the startup tests of Table 1 of this enclosure were reviewed for tests that would potentially be invalidated by EPU. The following tests were performed exclusively below 80 percent OLTP: STP-3, STP-4, STP-6, STP-7, STP-8, STP-9, STP-10, STP-11, STP-12, STP-13, STP-17, STP-22, STP-31, and STP-37.

In accordance with SRP 14.2.1, paragraph III.A.2, the initial startup tests conducted at power levels below 80 percent OLTP were reviewed for tests that would potentially be invalidated by EPU. No such testing was identified for MNGP.

The MSIV Closure Test (STP-11) and the Generator Load Rejection Test (STP-17) were performed only at power levels less than 80 percent OLTP and are not invalidated by the EPU. Despite the fact that these tests were performed at <80 percent OLTP, they are large transient tests. The need for these tests as part of EPU startup testing is discussed later in Section 4.3.

Power ascension startup tests performed at ≥ 80 percent of OLTP

Table 1 to this enclosure provides comparisons of initial startup tests and startup tests for the 6.3 percent uprate to 1775 MWt, to planned testing for CPPU startup. As seen in Table 1, the following tests were performed at 80 percent of OLTP or greater during initial startup: STP-1, STP-2, STP-5, STP-14, STP-15, STP-16, STP-18, STP-19, STP-20, STP-21, STP-23, STP-24, STP-25, STP-26, STP-27, STP-28, STP-29, STP-30, STP-32, and STP-34. Planned testing for CPPU is indicated in Table 1 for tests that are being repeated from the original plant testing, with additional details provided in Table 2, which includes a description of all tests that will be performed. Justifications for exemption from certain transient testing are provided in Section 4.3 below.

Power ascension transient tests performed at < 80 percent of OLTP

Table 1 to this enclosure provides a complete comparison of initial startup tests to the startup tests performed for the uprate to CLTP (1,775 MWt) and the tests planned for CPPU (2,004 MWt). The following large transient tests, shown in Table 2 of SRP 14.2.1, were not performed during MNGP initial startup at power levels equal to or greater than 80 percent.

Initial Transient Test	Test No.	Power Level (%OLTP)
MSIV Closure	STP-11	25% 50% 75%
Generator Trip	STP-17	15% 50%

None of these tests will be invalidated by the EPU, and it is not necessary to include these tests in the EPU startup testing, as documented in Table 1 of this enclosure and in Section 4.3.

Power ascension transient tests performed at ≥ 80 percent of OLTP

Table 1 to this enclosure provides a complete comparison of initial startup tests to the startup tests performed for the uprate to CLTP (1,775 MWt) and the tests planned for CPPU (2,004 MWt). As seen in Table 1 to this enclosure, the following table shows those startup transient tests performed at 80 percent of OLTP or greater.

Initial Transient Test	Test No.	Power Level (%OLTP)
Recirculation System • One Pump Trip • Two Pump Trip	STP-14	 100% 100%
Recirculation Flow Control	STP-15	88% 100%
Turbine Trip	STP-16	100%
Pressure Regulator • Set Point Change • Backup Regulator	STP-18	 88% 100% 100%
Bypass Valve	STP-19	88% 100%
Feedwater System • One Pump Trip • Set Point Change	STP-20	 100% 88% 100%

As shown in Table 1 to this enclosure, STP-18 and STP-20 (set point change only) will be included as part of the EPU startup testing. The turbine trip (STP-16), recirculation pump trip (STP-14), recirculation flow control (STP-15), bypass valve testing (STP-19), and feedwater pump trips (STP-20) will not be tested and are discussed in Section 4.3.

4.2 Post Modification Testing Requirements [SRP 14.2.1, III.B]

Enclosure 8 of the EPU License Amendment Request provides a listing of EPU implementation modifications that are currently anticipated and that are being prepared for implementation between 2009 and 2011. NSPM plans to implement EPU over two fuel cycles. In view of this two step process, implementation of the design changes listed in Enclosure 8 of the EPU License Amendment Request will occur throughout the period.

Modification Aggregate Impact

As can be seen from inspection of the modifications list of Enclosure 8 of the EPU License Amendment Request, the aggregate impact of most of these modifications on plant operations is minimal. The majority of the modifications are minor changes that will be segmented over two refueling outages. There are some significant modifications (e.g. HP turbine replacement, LP turbine modifications, Bus 11 and 12 replacement) that have a minor impact on the reactor plant, both individually and in aggregate. With some of the changes that are more interrelated (e.g. piping changes in feedwater), the extent of the changes themselves are minor (piping changes or pipe support modifications). An overall aggregate impact of these changes is not anticipated.

Condensate system and feedwater system upgrades do represent significant plant modifications, such as replacement of condensate pumps and motors, replacement of reactor feedwater pumps and motors, and enlargement of condensate demineralizers. These modifications will have an aggregate impact on the plant. These changes will be adequately addressed during post modification testing and the aggregate impact will be addressed by feedwater system power ascension testing (See Table 2 for a description of the performance and functional testing). Aggregate impact of EPU plant modifications, setpoint adjustments, and parameter changes will be demonstrated by a test program established for BWR EPU in accordance with startup test specifications as described in Power Uprate Safety Analysis Report (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. These tests, which supplement the normal Technical Specification testing requirements, are summarized below:

- Testing will be performed in accordance with the Technical Specifications Surveillance Requirements on instrumentation that is recalibrated for CPPU conditions. Overlap between the IRM and APRM will be assured. Data will be taken at points from 90 percent up to 100 percent of CLTP so that system performance parameters can be projected for CPPU power before CLTP is exceeded. CPPU power increases will be made in predetermined increments of ≤5 percent power (the planned increment is ~2.5 percent). Operating data, including fuel thermal margins, will be taken and evaluated at each step. Routine measurements of reactor and system pressures, flows, and vibration will be evaluated from each measurement point, prior to the next power increment. Radiation measurements will be made at selected power levels to ensure the protection of personnel.
- Control system tests will be performed for the reactor feedwater/reactor water level controls, and pressure controls, as applicable. These operational tests will be

performed at the appropriate plant conditions for that test at each of the power increments, to show acceptable adjustments and operational capability.

- Steam dryer/separator performance will be confirmed within limits by determination of steam moisture content as required during power ascension testing.
- Vibration monitoring of main steam and feedwater piping will be performed to permit a thorough assessment of the effect of EPU on this piping.

Because steam pressure and maximum licensed core flow have not changed and recirculation flow may only slightly increase for CPPU, testing of system performance affected by these parameters is not necessary with the exception of the tests listed above.

Multiple Structure Systems and Components (SSC)

Functions important to safety that rely on integrated operation of multiple SSCs following plant events (such as plant load swings) will be adequately addressed for MNGP through the normal testing and modification processes and through EPU testing identified in this enclosure.

4.3 Justifications for Elimination of Power Ascension Tests [SRP 14.2.1, III.C]

For steady-state or small transient tests that were performed at ≥80 percent OLTP during initial startup testing, that will not be re-performed during EPU startup testing, the justification is summarized in Table 1. This approach applies to the following tests: STP-2, STP-15, STP-21, STP-23, STP-26, STP-27, STP-28, STP-29, STP-30, STP-32, and STP-34.

For large transient tests that were performed during initial startup, that will not be re-performed during EPU startup, the justification is summarized in Section 4.3. This applies to the following tests: STP-11, STP-14, STP-16, STP-17, STP-19, and STP-20.

Guidelines of SRP 14.2.1, Paragraph III.C.2

Paragraph III.C.2 of SRP 14.2.1 provides specific guidance to be considered to justify elimination of large transient testing. The following table provides a cross reference between the guidance of SRP Paragraph III.C.2 and this enclosure.

Paragraph III.C.2	Guidance Criteria	Discussion/Location in This Document
(a)	Previous operating experience	Contained in paragraph 4.3.1/2.a where applicable, considering MNGP experience and EPU industry experience.
(b)	New thermal hydraulic phenomena or system interactions	No new thermal hydraulic phenomena or new system interactions were identified as a result of MNGP CPPU, as discussed in paragraph 4.3.1/2.b.
(c)	Conformance with limitation of analytical method	MNGP has no unique limitations associated with conformance to analytical methods. Where analytical results support the justification for eliminating the test, this aspect is discussed in paragraph 4.3.1/2.c.
(d)	Plant staff familiarization with facility operation and EOPs	Contained in paragraph 4.3.1/2.d
(e)	Margin reduction in safety analysis for AOOs	Provided in paragraph 4.3.1/2.e for specific tests, where applicable. Discussed in the section on EPU analyses results.
(f)	Guidance in Vendor topical reports	Discussed in paragraph 4.3.1/2.f
(g)	Risk implications	Provided in paragraph 4.3.1/2.g

ELTR1 states MSIV closure test should be performed for EPU if the power uprate is more than 10% above any previously recorded MSIV closure transient. ELTR1 also states a generator load rejection test should be performed if the uprate is more than 15 percent above any previously recorded generator load rejection transient. ELTR1 applies to extended power uprates whether constant pressure or otherwise.

With regard to these specific ELTR1 requirements, MNGP recorded an MSIV closure event on October 23, 2001 and a generator load rejection event on January 21, 2002. Based on these events, the ELTR1 criteria apply to MNGP as follows:

Event	Date	Power Level	CPPU Power Level	Percent Increase to CPPU	Recommendation by ELTR1
MSIV Closure	10-23-2001	1740 MWt	2,004 MWt	15	Because the increase is >10%, new test should be performed.
Generator Load Rejection	1-21-2002	1773 MWt	2,004 MWt	13	Because the increase is <15%, new test is not needed.

4.3.1 MSIV Closure Event (STP-11)

The MSIV closure event startup test functionally checks the Main Steam Isolation Valves for proper operation at selected power levels, determines reactor transient behavior during and following simultaneous full closure of all MSIVs, determines isolation valve closure time and determines the maximum power at which a single valve closure can be made without a scram.

The MSIV closure tests for Monticello during initial startup were not performed at power levels above 80 percent. The maximum power full closure test was performed at 75 percent (1258 MWt) of OLTP. Above 75 percent, the MSIVs were only stroked partially to ensure the valve would stroke and that partial stroke testing did not result in a SCRAM.

During an MSIV closure, the reactor is "bottled up" as feedwater continues to enter the reactor, as regulated by the feedwater regulating valve. Feedwater will continue to enter the reactor until either the operators trip the feedwater pumps, or the pumps trip automatically on a high level. Due to decay heat, steam continues to be produced, and the safety relief valves lift cyclically to relieve pressure, until pressure is stabilized below the lowest SRV setpoint. This set of events is the basic pattern for the transient regardless of the power level at which the transient initiates.

(a) Previous operating experience:

Monticello Test/Operating Experience

STP-11 (1971)

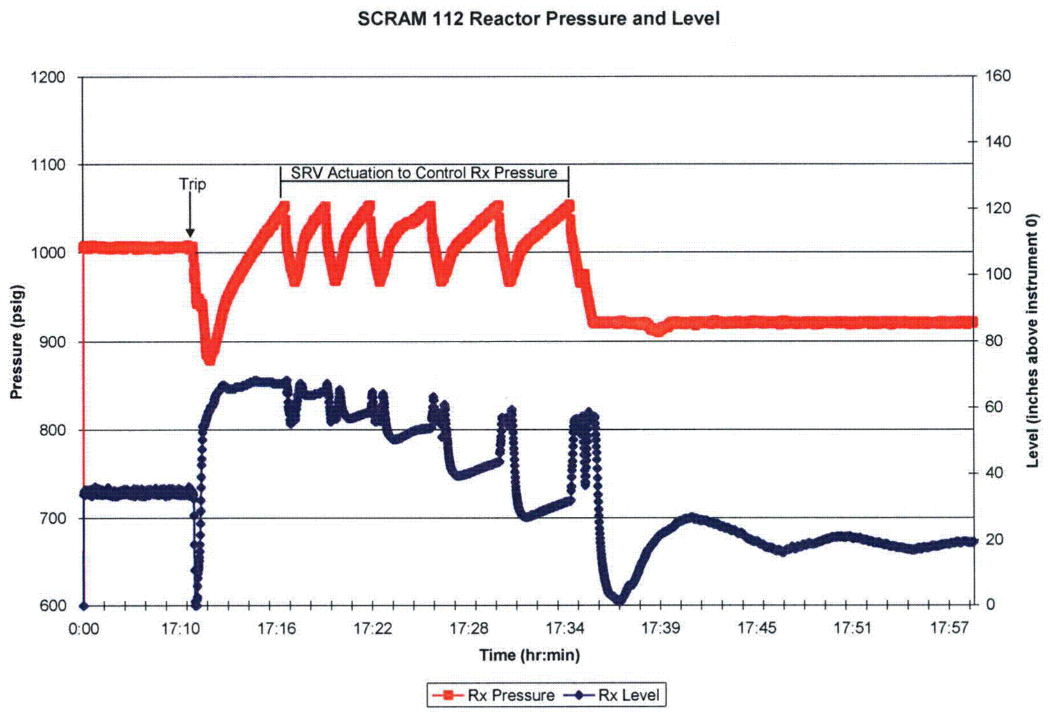
All Acceptance Criteria for MSIV Closure Event startup testing were satisfied. Proper MSIV operation was demonstrated and proper closure times were measured during testing at selected power levels.

Functionally check MSIVs for Proper Operation at Selected Power Levels
During startup testing MSIVs were closed and tested individually during initial heatup at rated pressure, and during the test condition at approximately 75 percent power. Proper operation was demonstrated and closure times were within limits. Neutron flux, reactor pressure, heat flux, and steam flow margins to scram or isolation were calculated and results were within limits.

Determine Reactor Behavior (during and following full and simultaneous closure of all MSIVs
A full MSIV isolation was initiated from 75 percent power and the parameters of MSIV closing time and reactor pressure were recorded and compared to predicted values. The actual pressure rise experienced during this test was such that no safety/relief valves lifted.

SCRAM 112 – October 2001

On October 23, 2001, a full MSIV closure event occurred at 98 percent CLTP (or 87 percent of CPPU). The event was initiated by a plant technician inadvertently contacting a critical instrument rack while working in the Reactor Building. The vibration of the workers foot contacting the rack caused a Group I Isolation and subsequent Reactor Scram. Data recorded during the event demonstrated that the plant responded as expected and that resulting parameters were within guidelines and requirements. The following figure shows the plant response for reactor pressure and level. The expected behavior is observed, i.e., SRV actuation regulates pressure within a prescribed band, after an initial level surge, level gradually decreases until stable conditions are reached.



Note: Instrument 0 is 477.5 inches, which is 126 inches above top of active fuel.

The plant response following the MSIV closure transient was in accordance with expectations. Note that the power level for this transient (87 percent of CPPU) exceeds the percentage power (75 percent OLTP) during initial startup testing.

Other BWR (post-EPU) experience

A review of industry transient events that occurred at greater than original power levels at BWR-3/4 units that are similar in design to MNGP resulted in the following examples of plant response to MSIV closure events. As indicated in the example below, the plant responded as expected in accordance with its design features. No unexpected conditions were experienced nor were any latent defects uncovered in these events beyond the specific failures that actually initiated the events. The event provides further evidence that Large Transient Testing is unnecessary.

Dresden Nuclear Power Station – 17 Percent Approved Power Uprate

LER 2005-02

On March 24, 2005, at 0529 hours (CST), with Unit 2 at approximately 96 percent power, two unexpected control room alarms were received for exceeding the Electro-Hydraulic Control System maximum combined flow limit setpoint and open Turbine Bypass Valves. Several seconds later, high flow in the Main Steam System resulted in a signal to close the Main Steam Isolation Valves (MSIVs) that initiated an automatic reactor scram. All control rods fully inserted and all Group 1 Isolation Valves closed as designed. The Group 2 and 3 Isolation Valves closed as expected and the Isolation Condenser was manually initiated to control reactor pressure. All MSIVs fully closed within the required time.

The experience above indicates that other plants that have previously performed EPUs have successfully responded as planned to MSIV closure events. Based on the experience of other similar plants, it is likely that MNGP will perform similarly to these transients.

(b) New thermohydraulic phenomena or system interactions:

No new thermohydraulic phenomena or system interactions have been identified as a result of the EPU. Additionally, PUSAR Section 2.2.2.1 indicates that the generic evaluation for MSIV closure identified in ELTR2 is applicable to Monticello and is bounding. The MSIV maximum flow and pressure conditions are unchanged from the current licensed power. Based on the results of the generic evaluation in ELTR2 and the limits placed on the maximum steam flow by the flow restrictors, the MSIVs are determined to be acceptable for EPU conditions.

(c) Conformance with limitations of analytical methods:

N/A.

(d) Plant staff familiarization with facility operation and EOPs:

The EPU will not change any plant operations or EOP actions. Since the dome pressure will not change, no SRV set point changes are required. The inboard MSIV closure time is assisted by the increased steam flow. The closing times will be readjusted to ensure that the MSIV closure times satisfy plant technical specifications. However, these changes do not change the operation of the plant to address increased steam flow.

(e) Margin reduction in safety analysis for Anticipated Operational Occurrences (AOOs):

Analysis indicates that the predicted peak dome pressure for limiting events increases by up to 40 psi. The analysis is updated each fuel reload. Adequate margins exist to accommodate cycle specific variances and to ensure that all ASME Code requirements continue to be satisfied.

(f) Guidance in vendor topical report:

PUSAR Section 2.2.2.1 indicates that the generic evaluation for MSIV closure identified in ELTR2, Section 4.7 is applicable to Monticello and is bounding. ELTR2, Section 4.7 uses a pressure increase of 75 psi; however, Monticello is performing a CPPU, which will result in a full power steam flow increase of 15 percent without a corresponding pressure increase. Based on the guidance in ELTR2, MSIV closure testing is not recommended.

(g) Risk Implications:

MNGP conducted an assessment of the risk impact of performing an MSIV isolation transient test at EPU conditions.

The MNGP EPU PRA was used to estimate the additional risk of testing by determining the Conditional Core Damage Probability (CCDP) and Conditional Large Early Release Probability (CLERP) associated with the individual tests. These values represent the additional probabilities given that the test has occurred. The CCDP values for Core Damage Frequency (CDF) were calculated by multiplying the base CDF by the PRA Fussell-Vesely risk importance measure of the initiating event and then dividing that result by the initiator frequency. The CLERP for Large Early Release Frequency (LERF) is calculated similarly.

The PRA results are presented in the tables below.

PRA Model	CDF	LERF
Base EPU PRA	7.89E-06	3.94E-7

PRA Initiating Event	Fussell-Vesely* CDF	Fussell-Vesely* LERF	Initiating Event Frequency (/yr)
Isolation event (MSIV Closure)	1.22E-03	7.98E-03	3.80E-02

Results		
Initiator	CCDP	CLERP
MSIV Isolation test	2.53E-7	8.27E-8

The results indicate a small but non-trivial increase in core damage and large early release probabilities. The increase in risk does not justify the need to perform the tests for demonstration purposes only, since no new information, that cannot otherwise be obtained, will be gained from the test.

(h) Conclusion:

MNGP has reviewed the initial startup testing, recent operating experience, and plant experience from other BWRs, as well as analysis and PRA results. The plant response to a MSIV closure event has been previously demonstrated by plant transient data. Supporting analysis included as part of the EPU and fuel reload analyses also demonstrate the continued safe operation and expected plant behavior to the MSIV closure event after the EPU. Lastly, the power level for the MSIV closure event (87 percent of CPPU) exceeds the percentage power (75 percent OLTP) during initial startup testing. Therefore the objective of this test is satisfied without requiring new or additional plant transient testing.

4.3.2 Generator Load Rejection (STP-17)

The startup testing for Generator Load Rejection demonstrates the response of the reactor and turbine overspeed following a generator trip. The startup tests for Turbine Trip (STP-16) and Generator Load Rejections (STP-17) at MNGP, in addition to recent plant transients at CLTP, adequately demonstrated this response and further testing is not considered necessary.

During a generator load rejection, the turbine will speed up due to the loss of the load provided by the generator. Upon receiving a load rejection signal, the turbine control valves will close, and the reactor will receive a SCRAM signal. Excess steam will be automatically bypassed to the main condenser by the bypass system, which is automatically controlled. If the pressure increases above the lowest SRV setpoint, the SRV will open to relieve pressure to the suppression pool.

A turbine trip is similar to a generator load rejection, with the exception that the turbine speed increase will not occur.

(a) Previous operating experience:

Monticello Test/Operating Experience

STP-17

A generator load rejection was performed from a maximum power level of 50 percent. All Acceptance Criteria for the generator load rejection startup testing were satisfied.

SCRAM 113 – January 21, 2002

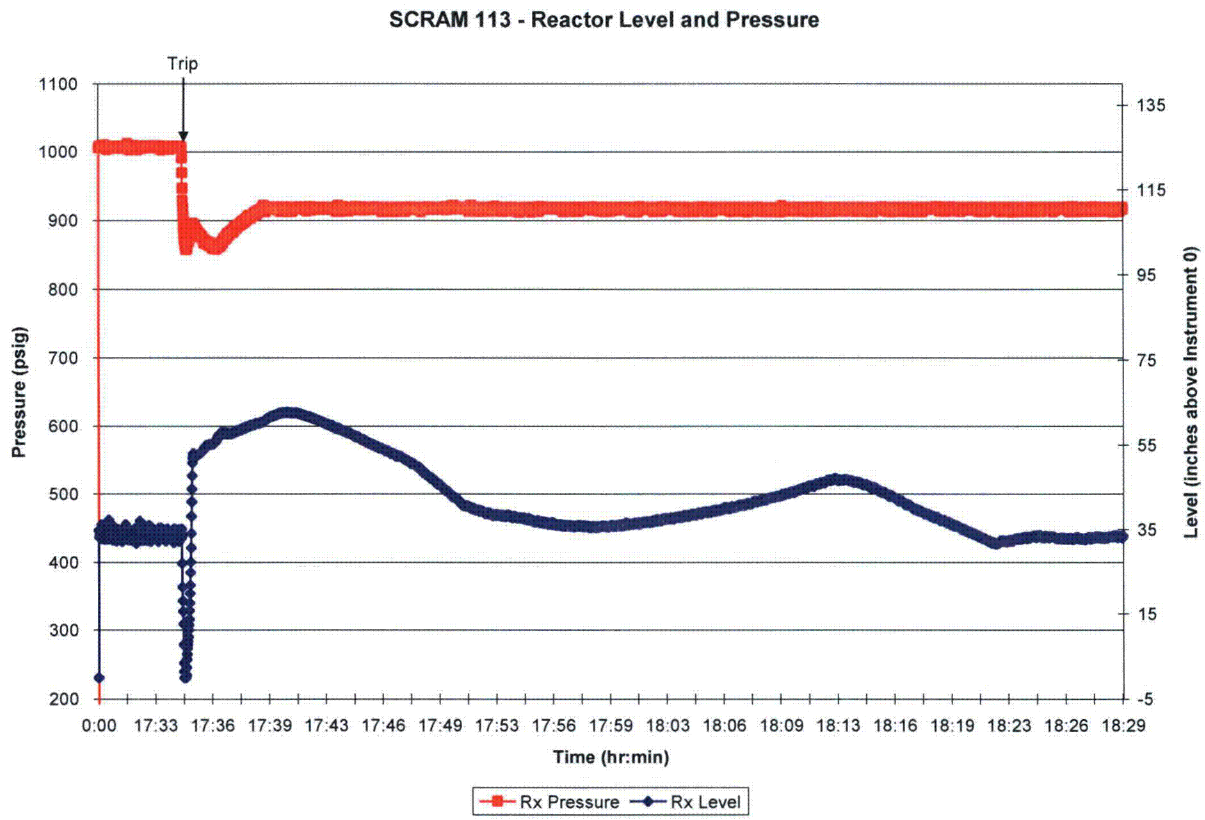
On January 21, 2002, a generator load rejection event occurred. While operating at 100 percent CLTP a turbine control valve fast closure (load rejection) signal resulted in a reactor scram. All rods fully inserted and all safety systems functioned as designed. A

ENCLOSURE 9

Group II containment isolation occurred, as expected, on a reactor low water level signal following the scram.

Following the scram, No. 11 Reactor Feedwater Pump (RFP) was manually tripped in accordance with plant procedures. Before No. 12 RFP could be manually tripped, an automatic trip on high reactor water level occurred. The No. 12 RFP was restarted, the feedwater block valves closed, and reactor water level was controlled using the low flow feedwater regulating valve. Operator actions were determined to be timely, consistent with procedures, and reflected an appropriate sensitivity to operating conservatism. All major plant and substation equipment functioned as designed in response to the scram.

The following figure shows the plant response for reactor pressure and level.



Note: Instrument 0 is 477.5 inches, which is 126 inches above top of active fuel.

The plant response for the transient was consistent with expectations. The power level for the transient is equivalent to 88.5 percent of EPU power.

Other BWR (post-EPU) experience (Includes Turbine Trips and Generator Load Rejections)

A review of industry transient events that occurred at greater than original power levels at BWR-3/4 units that are similar in design to MNGP resulted in the following examples of plant response to generator load rejection and turbine trip events. As indicated in the examples below, the plants responded as expected in accordance with its design features. No unexpected conditions were experienced nor were any latent defects uncovered in these events beyond the specific failures that actually initiated the events. These events provide further evidence that Large Transient Testing is unnecessary.

Dresden Nuclear Power Station – 17 Percent Approved Power Uprate

<u>LER 2004-002</u>

On January 30, 2004, the Shift Manager decided to swap the Unit 3 Main Turbine Lube Oil Coolers as the Turbine Oil Continuous Filter Differential Pressure had been increasing for several days. On January 30, 2004, at 1155 hours (CST), with Unit 3 at 97 percent power in Mode 1, an automatic scram occurred due to a Main Turbine trip from low lube oil pressure. The event occurred during a swapping of lube oil coolers. Immediately following the scram, the position of the Feedwater Regulating Valves (FRVs) increased from 56 percent open to 63 percent. The increase in the position of the FRVs, combined with the post-scram decreasing reactor pressure, caused an increase in total feedwater flow that led to the trip of the "B" Reactor Feedwater Pump (RFP) on low suction pressure. Additionally, subsequent FRVs response to increasing reactor vessel level was not fast enough to prevent the level from reaching the RFP High Level trip set point and resulted in the tripping of the "A" and "C" RFPs. Reactor water level was subsequently restored to normal and the RFPs were restarted. All rods inserted and other than the feedwater response, all other system responded as expected to the automatic scram.

Edwin I. Hatch Nuclear Plant – 13 Percent Approved Power Uprate

<u>LER 99-05</u>

On May 5, 1999, Hatch Unit 2 was at 98.3 percent of rated power (2,716 CMWT). The main generator tripped on a ground fault, followed by a turbine trip. The reactor scrammed and the reactor recirculation pumps tripped automatically on turbine control valve fast closure caused by the turbine trip. The reactor feed water pumps maintained water level higher than eight inches above instrument zero. No safety system actuations on low level were received nor were any required. Pressure reached a maximum value of 1,124 psig. Plant and system responses were as expected.

<u>LER 2001-02</u>

On March 28, 2001, Plant Hatch Unit 1 was at 100 percent rated thermal power. At that time, the reactor automatically scrammed on turbine control valve fast closure caused by a main turbine trip. The main turbine tripped when actuation of phase two and phase three differential relays monitoring a unit auxiliary transformer resulted in actuation of a lockout relay. Actuation of this lockout relay generated a direct turbine trip signal and the main turbine tripped per design.

Reactor Feedwater Pumps recovered reactor vessel water level within 30 seconds of the scram. As a result, the HPCI and RCIC system low water level initiation signals cleared before either system could inject makeup water to the reactor vessel. Vessel pressure reached a maximum value of 1,127 psig after receipt of the scram. All systems functioned as expected and per their design given the water level and pressure transients caused by the turbine trip and reactor scram. Vessel water level was maintained well above the top of the active fuel throughout the transient.

Brunswick Steam Electric Plant – 20 Percent Approved Power Uprate

LER 2003-04

On November 4, 2003, Brunswick Steam Electric Plant Unit 2 was operating at approximately 96 percent of rated thermal power when a loss of generator excitation caused a generator and turbine trip. The generator/turbine trip resulted in RPS actuation. The voltage transient also resulted in PCIS Valve Group I (MSIVs, Main Steam Line Drain valves, and Reactor Recirculation Sample valves), Group 3 (Reactor Water Cleanup isolation valves), and Group 6 (Containment Atmosphere Control/Dilution, Containment Atmosphere Monitoring, and Post Accident Sampling System isolation valves) isolations.

All control rods fully inserted into the core. Plant response to the transient also resulted in High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) System actuations on low reactor pressure vessel (RPV) coolant level, with injection into the RPV. All four Emergency Diesel Generators (EDGs) automatically started but did not load because electrical power was not lost to the emergency buses.

The experience above indicates that other plants that have previously performed EPUs have successfully responded as planned to generator load rejections and turbine trips. Based on the experience of other similar plants, it is likely that MNGP will perform similarly to these transients.

(b) New thermohydraulic phenomena or system interactions:

No new thermohydraulic phenomena or system interactions have been identified as a result of the EPU.

(c) Conformance with limitations of analytical methods:

N/A.

(d) Plant staff familiarization with facility operation and EOPs:

The EPU will not change any plant operations or EOP actions.

(e) Margin reduction in safety analysis for Anticipated Operational Occurrences (AOOs):
A generator load rejection is not a limiting event and does not result in a reduction to the margin of safety.

(f) Guidance in vendor topical report:

Full load reject testing is not required under the guidelines of ELTR1 as shown below:

Event	Date	Power Level	CPPU Power Level	Percent Increase to CPPU	Recommendation by ELTR1
Generator Load Rejection	1-21-2002	1773 MWt	2,004 MWt	13	Because the increase is <15%, new test is not needed.

(g) Risk Implications:

MNGP conducted an assessment of the risk impact of performing a turbine trip transient test at EPU conditions (which would be representative of a load rejection transient).

The MNGP EPU PRA was used to estimate the additional risk of testing by determining the Conditional Core Damage Probability (CCDP) and Conditional Large Early Release Probability (CLERP) associated with the individual tests. These values represent the additional probabilities given that the test has occurred. The CCDP values for Core Damage Frequency (CDF) were calculated by multiplying the base CDF by the PRA Fussell-Vesely risk importance measure of the initiating event and then dividing that result by the initiator frequency. The CLERP for Large Early Release Frequency (LERF) is calculated similarly.

The PRA results are presented in the table below.

PRA Model	CDF	LERF
Base EPU PRA	7.89E-06	3.94E-7

PRA Initiating Event	Fussell-Vesely* CDF	Fussell-Vesely* LERF	Initiating Event Frequency (/yr)
Non-isolation event (Turbine Trip)	4.00E-02	2.21E-01	9.90E-01

Results		
Initiator	**CCDP**	**CLERP**
Turbine trip test	3.19E-7	8.80E-8

The results indicate a small but non-trivial increase in core damage and large early release probabilities. The increase in risk does not justify the need to perform the tests for demonstration purposes only, since no new information, that cannot otherwise be obtained, will be gained from the test.

(h) Conclusion:

MNGP has reviewed the initial startup testing, recent operating experience, and plant experience from other BWRs, as well as analysis and PRA results. The plant response to a generator load rejection has been previously demonstrated by plant transient data. Supporting analysis included as part of the EPU and fuel reload analyses also demonstrate the continued safe operation and expected plant behavior to the generator load rejection after the EPU. Lastly, the power level for the generator load rejection (88.5 percent of CPPU) exceeds the percentage power (50 percent OLTP) during initial startup testing. Therefore the objective of this test is satisfied without requiring new or additional plant transient testing.

4.3.3 Recirculation Pump Trips (STP-14)

As part of the CPPU, the recirculation flow rate to provide 100 percent core flow is only slightly increased. The increase is approximately 1.7 percent of the CLTP recirculation rate. Due to the small increase in recirculation flow rate, the plant response to recirculation flow rate changes is not significantly affected. No testing is required.

4.3.4 Recirculation Flow Control (STP-15)

The recirculation pump motor generators will be upgraded. The recirculation flow will be increased by 1.7 percent of the OLTP as a result of the EPU. Due to the small increase in recirculation rate, the plant response is not significantly affected. No testing is required.

4.3.5 Turbine Trip (STP-16)

A turbine trip is similar to a generator load rejection (see Section 4.3.2). Additionally, turbine trips with and without bypass are considered and evaluated as appropriate for each fuel reload. The reload analysis is then added as part of the revised safety analysis report.

4.3.6 Bypass Valve (STP-19)

The bypass valves are not being modified or replaced as part of the EPU. The flow rate under normal operating conditions will remain as 0.97 Mlb/hr (13.3 percent of the current steam flow, or 11.6 percent of the post EPU steam flow). Additionally, the bypass valves are non-safety related and are not required under accident conditions. Accordingly, no startup testing of the bypass valves is warranted.

4.3.7 Feedwater Pump Trips (STP-20)

For the loss of feedwater (LOFW) event, adequate transient core cooling is provided by maintaining the water level inside the core shroud above the top of active fuel. A plant-specific analysis was performed for Monticello at EPU conditions. This analysis assumed failure of the HPCI system and used only the RCIC system to restore the reactor water level. Because of the extra decay heat from EPU, slightly more time is required for the automatic systems to restore water level. Operator action is only needed for long-term plant shutdown. The results of the LOFW analysis for Monticello show that the minimum water level inside the shroud is 77 inches above the top of active fuel at EPU conditions. After the water level is restored, the operator manually controls the water level, reduces reactor pressure, and initiates RHR shutdown cooling. This sequence of events does not require any new operator actions or shorter operator response times. Therefore, the operator actions for an LOFW transient do not significantly change for EPU.

The results of the Monticello LOFW analysis also showed that the RCIC system restores the reactor water level while avoiding the level at which the Emergency Operating Procedures (EOPs) require the operator to initiate ADS.

Since there are no adverse impacts on the ability of the plant to adequately control reactor vessel level, and no changes to the plant operation are required, transient testing of the feedwater system (other than that identified in Table 2) is not necessary.

5.0 Operator Training/Large Transient Simulations

For EPU, MNGP will benchmark the Monticello simulator against the EPU analyzed transients and to subsequently perform testing to confirm the adequacy of simulation of the various transients. Once the simulator is benchmarked, MNGP operators will be trained on various plant upset conditions, from postulated accident conditions to anticipated transients. In this way, plant operators will be prepared for the nature, timeline, and extent of the plant response to simulated transients.

Table 1 – Comparison of MNGP Initial Startup Testing and Planned EPU Testing

Orig. Test No.	Original Test Description	Startup Test Power % OLTP	1775 MWt Rerate Testing (Expressed in %OLTP) (Note 2)	Test Planned for CPPU	Evaluation Justification Notes	CPPU Test Conditions Percent of 1775 MWt (% CLTP)							
						≤90	100	102.5	105	107.5	110	112.5	EPU
STP-1	Chemical and Radiochemical: Chemical and radiochemical tests were conducted to establish water conditions prior to initial operation and to maintain these throughout the test program.	<15% 15% 25% 50% 75% 88% 100%	None	Yes EPU Test 1A	Test will be performed. See Table 2 for details.		X	X	X	X	X	X	X
N/A	Steam Dryer: The purpose of this test is to measure moisture content in main steam.	Not performed	None	Yes EPU Test 1B	Dryer performance monitoring. See Table 2 for details.		X	X	X	X	X	X	X
STP-2	Control Rod Drive: The purpose of this test is (1) Demonstrate that the hydraulic system and CRDs operate properly over the full range of primary coolant temperatures and pressures from ambient to operating, and particularly that thermal expansion of core components does not bind or significantly slow control rod movement, (2) Determine the initial operating characteristics of the entire system.	<15% 25% 50% 100%	None	None	Not necessary to retest. No changes in temperature or pressure will occur as a result of the EPU.								
STP-3	Fuel Loading: To load fuel safely and efficiently to the full core size	<15% Note 1	None	None	Fuel loading is performed in accordance with standard procedures.								
STP-4	Shutdown Margin: To demonstrate that the reactor is adequately shutdown with the strongest single control rod fully withdrawn	<15% Note 1	None	None	Shutdown margin is determined as part of every reload.								
STP-5	Radiation Measurement: The purpose of this test is to determine gamma and neutron radiation levels in the plant environs prior to operation and at selected power levels to ensure the protection of personnel.	<15% 15% 25% 50% 88% 100%	106.3% Determine gamma and neutron radiation levels in the plant environs	Yes EPU Test 2	Test will be performed. See Table 2 for details		X	X	X	X	X	X	X
STP-6	Control Rod Sequence: The purpose of this test is to verify the acceptability of the specified control rod withdrawal sequence	<15% Note 1	None	None	Control rod sequences are developed in accordance with approved procedures.								
STP-7	SRM Performance: The purpose of this test is to provide data to demonstrate that the operational sources, Source Range Monitoring instrumentation, and rod withdrawal sequences provide adequate information to the operator during normal startup	<15% Note 1	None	None	Not necessary to retest. Normal surveillances provide assurance.								
STP-8	IRM Calibration: The purpose of this test is to calibrate the IRM system to read percent of core thermal power.	<15% Note 1	None	Yes EPU Test 10	Adjust IRM to APRM overlap during first controlled shutdown following APRM calibration at EPU								

Orig. Test No.	Original Test Description	Startup Test Power % OLTP	1775 MWt Rerate Testing (Expressed in %OLTP) (Note 2)	Test Planned for CPPU	Evaluation Justification Notes	CPPU Test Conditions Percent of 1775 MWt (% CLTP)							
						≤90	100	102.5	105	107.5	110	112.5	EPU
STP-9	Reactor Vessel Temperature: The purpose of this procedure is to obtain RPV temperature during rapid heatup and cooldown to confirm thermal analysis model	<15% Note 1	None	None	The operating temperature and pressure for the EPU is the same as pre-EPU conditions.								
STP-10	System Expansion: The purpose of this test is to verify that the (1) reactor drywell piping system is free and unrestrained in regard to thermal expansion and that suspension components are functioning in the specified manner (2) provides data for calculation of stress levels in nozzles and weldments.	<15% Note 1	None	None	The operating temperature and pressure changes associated with the EPU are insignificant for these components								
STP-11	Main Steam Isolation Valves: The purpose of this test is to (1) Functionally check the main steam isolation valves for proper operation at different power levels, (2) Demonstrate the capability to perform isolation valve test closures without threatening reactor safety or causing a reactor scram, (3) Determine reactor transient behavior following simultaneous full closure of all MSIV, (4) Determine isolation valve closure times	25% 50% 75%	None	None	See PUSAR Section 2.2.2.1 for MSIV closure analysis and Section 4.3.1 of the enclosure for justification of elimination.								
STP-12	RCIC: The purpose of this test is to demonstrate the ability of the RCIC system to provide the required flow at various turbine steam supply and pump discharge pressures and to start from cold standby conditions	<15% Note 1	None	None	Since there is not a change in dome pressure, additional startup testing of the RCIC system is not necessary.								
STP-13	HPCI: The purpose of this test is to demonstrate the ability of the HPCI system to provide the required flow at various turbine steam supply and pump discharge pressures and to start from cold standby conditions	<15% 50%	None	None	Since there is not a change in dome pressure, additional startup testing of the HPCI system is not necessary.								
STP-14	Recirculation System: The purpose of this test is to (1) Evaluate the recirculation flow and power level transients following trips of one or both of the recirculation pumps, (2) Calibrate the reactor core flow measurement system, (3) Measure the reactor core flow by performing mass and energy balances on the reactor downcomer	25% 50% 75% 100%	106.3% Calibrate the reactor core flow measurement system. Functional testing Note 2	None	The increase in recirculation flow rate is approximately 1.7% from CLTP. Due to the small increase in recirculation rate, the plant response to a recirculation pump trip is not significantly affected.								

Orig. Test No.	Original Test Description	Startup Test Power % OLTP	1775 MWt Rerate Testing (Expressed in %OLTP) (Note 2)	Test Planned for CPPU	Evaluation Justification Notes	CPPU Test Conditions Percent of 1775 MWt (% CLTP)							
						≤90	100	102.5	105	107.5	110	112.5	EPU
STP-15	Flow Control: This test will determine (1) the plant response to changes in the recirculation flow, (2) Demonstrate plant load following capability	50% 75% 88% 100%	90% 98% Functional testing Note 2	None	The recirculation pump motor generators will be upgraded. The increase in recirculation flow rate is approximately 1.7% from CLTP. Due to the small increase in recirculation rate, the plant response to a recirculation flow change is not significantly affected.								
STP-16	Turbine Trip: The purpose of this test is to determine the response of the reactor system to a turbine trip. The parametric responses of particular interest are the peak values and rate of change of both reactor power and reactor steam dome pressure.	50% 75% 100%	None	None	None. Turbine trips with and without bypass are limiting transients for fuel reloads and are addressed in the reload analysis.								
STP-17	Generator Trip: The purpose of this test is to determine and demonstrate reactor response to generator trip, with particular attention to the rates of changes and the peak values of reactor power level, dome pressure and turbine speed.	15% 50%	None	None	See Section 4.3.2 of the enclosure for justification of elimination.								
STP-18	Pressure Regulator: The purpose of this test is to (1) Determine the reactor and pressure control system response to pressure regulator setpoint changes, (2) Demonstrate the stability of the reactivity-void feedback loop to pressure perturbations, (3) Demonstrate the control characteristics of the load limiter, (4) Demonstrate the take-over capabilities of the backup pressure regulator, (5) Optimize the pressure regulator setpoint	15% 25% 50% 75% 88% 100%	90% 98% Functional testing Note 2	Yes EPU Test 22	Test will be performed. See Table 2 for details	X	X	X		X		X	X
STP-19	Bypass Valves: The purpose of this test is (1) Demonstrate the ability of the pressure regulator to minimize the reactor pressure disturbance during a small step change in reactor steam flow, (2) Demonstrate that the bypass valve can be tested for proper functioning at rated power without causing a high flux scram.	15% 50% 75% 88% 100%	None	None	Not necessary to retest. No modifications are being made to the bypass valve as part of the EPU or to the pressure conditions at the bypass valve. See Section 4.3.6.								

ENCLOSURE 9

Orig. Test No.	Original Test Description	Startup Test Power % OLTP	1775 MWt Rerate Testing (Expressed in %OLTP) (Note 2)	Test Planned for CPPU	Evaluation Justification Notes	CPPU Test Conditions Percent of 1775 MWt (% CLTP)							
						≤90	100	102.5	105	107.5	110	112.5	EPU
STP-20	Feedwater System: The purpose of this test is to determine the effect of changes in subcooling on reactor power and steam pressure, and to demonstrate that reactor response to changes in subcooling are stable at all power levels. This test also includes feedwater pump trips.	15% 25% 50% 75% 88% 100%	90% 98% 104% Functional testing NSP committed to the following prior to testing: - area ambient temperatures will be monitored to confirm that design temperatures are not exceeded - confirm that the level in the high pressure and high intermediate pressure feedwater heaters is adequate to prevent bypassing of steam into the subcooling zone of the feedwater heaters - walkdown will be performed to verify that there is no unexpected flow induced vibration or motion of feedwater and condensate lines including small bore branch connections - measure feed and condensate pump motor currents to verify load study assumptions	Yes EPU Test 23	Functional and Performance tests will be performed. See Table 2 for details. Pump trip testing will not be performed. See Section 4.3.7.	X	X	X		X		X	X
STP-21	Flux Response to Rods: The purpose of this test is to demonstrate the relative stability of the power-reactivity feedback loop with regard to small perturbations in reactivity caused by rod movement with increasing power.	15% 25% 50% 75% 88% 100%	None	None	Reactor core is loaded in accordance with approved procedures. No changes have been made to reactor internals that would invalidate previous tests.								
STP-22	Relief Valves: The purpose of this test is to (1) Demonstrate the operability of the primary system relief valves, (2) Verify the capacity of the relief valves, (3) Demonstrate leak tightness of the relief valves after operation.	25%	None	None	SRV setpoints are not being changed as part of the EPU. Therefore, there is no effect on functionality.								
STP-23	LPRM Calibration: The purpose of this test is to calibrate the local power range monitor system.	15% 25% 50% 75% 100%	None	None	Tests performed in accordance with the surveillance program.								
STP-24	APRM Calibration: The purpose of this test is to present the methods for calibrating the Average Power Range Monitor Channels.	<15% 15% 25% 50% 75% 88% 100%	Flow biased scram setpoints and rod withdrawal Note 2	Yes EPU Test 12	Test will be performed. See Table 2 for details		X						
STP-25	Core Performance Evaluation: The purpose of this test is to evaluate the core and thermal hydraulic performance.	15% 25% 50% 75% 88% 100%	90% 100% 102% 104% 106.3% Note 2	Yes EPU Test 19	Test will be performed. See Table 2 for details	X	X	X	X	X	X	X	X

Orig. Test No.	Original Test Description	Startup Test Power % OLTP	1775 MWt Rerate Testing (Expressed in %OLTP) (Note 2)	Test Planned for CPPU	Evaluation Justification Notes	CPPU Test Conditions Percent of 1775 MWt (% CLTP)							
						≤90	100	102.5	105	107.5	110	112.5	EPU
STP-26	Calibration of Rods: The purpose of this test is to obtain the effect on reactor power of control rod motion for typical rod movements.	15% 50% 75% 88%	None	None	Reactor core is loaded in accordance with approved procedures. No changes have been made to reactor internals that would invalidate previous tests.								
STP-27	Axial Power Distribution: The purpose of this test is (1) obtain axial power distributions at various conditions of rod patterns, power levels, recirculation flow rate, and subcooling, (2) Verify the reproducibility of each TIP system, (3) Determine power distribution symmetry of octant-symmetric control rod patterns.	25% 50% 75% 88% 100%	None	None	Reactor core is loaded in accordance with approved procedures. No changes have been made to reactor internals that would invalidate previous tests.								
STP-28	Rod Pattern Exchange: The purpose of this test is to perform a representative change in basic rod pattern at a higher reactor power level.	100%	None	None	Reactor core is loaded in accordance with approved procedures. No changes have been made to reactor internals that would invalidate previous tests								
STP-29	Process Computer: The purpose of this test is to verify the performance of the process computer under plant operating conditions.	<15% 15% 25% 50% 75% 88% 100%	None	None	No changes are being made to the process computer that would invalidate original test.								
STP-30	Electrical Output and heat Rate: The purpose of this test is to demonstrate that the requirements of the Net Plant Electrical Outlet Warranty and the Net Plant Heat Rate Warranty are satisfied.	Warranty Test	None	None	No warranty test is planned. Provisions for performance are included in the contract with the vendor.								
STP-31	Loss of Turbine Generator and Offsite Power: The purpose of this test is to demonstrate that the reactor can safely with stand a loss of the turbine-generator and all off-site power.	25%	None	None	The EPU modifications do not change the ability of the safety systems to initiate and function properly nor change the ability of the EDGs to function properly.								
STP-32	Vibration Test: The purpose of the vibration tests is to (1) Measure the response of key reactor internal components as well as the recirculation pumps and piping to dynamic forces, (2) Demonstrate the mechanical integrity of the system to vibratory motion.	<15% 50% 75% 100%	None	None	No tests will be performed. The full power core flow range will change, but the maximum licensed core flowrate (60.5 Mlb/hr) will remain unchanged.								
STP-34	LPRM Response: The purpose of this test is to demonstrate the response characteristic of the LPRM chambers to typical flux levels encountered in operation and to the neutron flux near the maximum steady state heat flux limit.	50% 75% 100%	None	None	Reactor core is loaded in accordance with approved procedures. No changes have been made to reactor internals that would invalidate previous tests								

Orig. Test No.	Original Test Description	Startup Test Power % OLTP	1775 MWt Rerate Testing (Expressed in %OLTP) (Note 2)	Test Planned for CPPU	Evaluation Justification Notes	CPPU Test Conditions Percent of 1775 MWt (% CLTP)							
						≤90	100	102.5	105	107.5	110	112.5	EPU
STP-37	Recirculation Loop Control: The purpose of this test is to determine the "as built" characteristics of the recirculation control system (i.e., Drive Motor, Fluid Coupler, Generator, Drive Pump and Jet Pumps) and to adjust control systems parameters for optimum performance.	<15% 50%	None	None	The as-built characteristics of the Recirculation System are not changing.								
N/A	Main Steam and Feedwater Piping Vibration: The purpose of this test is to monitor vibration in the Main Steam and Feedwater System.	N/A	100% 102% 104% 106.3% Visual inspection of Feedwater and Steam Piping	Yes EPU Test 100	Test will be performed. See Table 2 for details. Main Steam and Feedwater system piping both inside and outside of containment		X	X		X		X	X
N/A	Plant Parameter Monitoring and Evaluation: The purpose of this test is to compare Power-dependant parameters that will be calculated using accepted methods to ensure current licensed and operational practices are maintained	N/A	90% 98% 100% 102% 104% 106.3%	Yes EPU Test 101	Power dependent performance parameters of systems and components remain within limits	X	X	X		X		X	X

Note 1: These tests were performed with the vessel open or during initial heat-up (prior to power testing).

Note 2: NSPM (NSP at the time) committed to performing power ascension testing in accordance with the generic power uprate guidelines provided in NEDC-32424P. This testing is designed to verify the following:

Plant systems affected by the power rerate are within design limits.
Nuclear fuel thermal limits are maintained within expected margins.
The response of the main steam pressure control system is stable.
The response of the reactor water level control system is stable.
The reactor core flow is within design limits and stable.

The power increase to the rerate power level was made along a constant rod line by increasing reactor recirculation flow. Steady state conditions were established after each increment of power increase, and test data was obtained to confirm the response of plant parameters and system performance. The power ascension consisted of three power increase increments of approximately 35 MWt or 2.1%. The incremental increases provide for a controlled approach to the rerate power level. The power increase increments are listed below.

Increment 1 1670 MWt to 1705 MWt
Increment 2 1705 MWt to 1740 MWt
Increment 3 1740 MWt to 1775 MWt

Table 2 – Planned EPU Power Ascension Testing

Title	Test Number	Test Description
Chemical and Radiochemical	Initial STP-1 EPU Test 1A	Samples will be taken and measurements will be made at selected EPU power levels to determine 1) the chemical and radiochemical quality of reactor water and reactor feedwater and 2) gaseous release.
Steam Dryer/Separator Performance	EPU Test 1B	Samples will be taken and measurements will be made at selected EPU power levels to determine steam dryer/separator performance (i.e., moisture carryover). For this testing main steam line moisture content is considered equivalent to the steam separator-dryer moisture carryover. Sampling and analysis will be in accordance with existing plant procedures.
Radiation Measurements	Initial STP-5 EPU Test 2	At selected EPU power levels, gamma dose rate measurements and, where appropriate, neutron dose rate measurements will be made at specific limiting locations throughout the plant to assess the impact of the uprate on actual plant area dose rates. USAR radiation zones will be monitored for any required changes.
IRM Performance	Initial STP-8 EPU Test 10	After the APRM calibration for EPU, the IRM gains will be adjusted as necessary to assure the IRM overlap with the APRMs.
APRM Calibration	Initial STP-24 EPU Test 12	Each APRM channel reading will be adjusted to be consistent with the core thermal power, referenced to the LPU level, as determined from the heat balance.

Title	Test Number	Test Description
Core Performance	Initial STP-25 EPU Test 19	Routine measurements of reactor parameters are taken near 90% and 100% of CLTP along a constant flow control line to be used to increase to maximum EPU power. Power increase is along this constant rod pattern line in incremental steps of 5% or less to ensure a careful, monitored approach to maximum EPU power. Measure reactor parameters and calculated core performance parameters are utilized to project those values at the next power level step. Core thermal power and core performance parameters are calculated using accepted methods to ensure current license and operational practices are maintained. Each test condition's actual parameters will be evaluated against their projected values and operational limits before increasing power to the next step and the final increase to maximum EPU power.
Pressure Regulator	Initial STP-18 EPU Test 22	The pressure regulator requires only the following changes for EPU: 1) Those settings identified in the Turbine Generator Performance Evaluations 2) Pressure Control System, (i.e. Pressure Regulator Setpoint), 3) Confirming the dynamic tuning parameters. Before EPU, while the plant is shutdown, the pressure regulator will be tested and dynamically calibrated. The Control and Transient Safety Report (C&TSR), NEDC-10069, and GE Service Information Letter (SIL) 589 discuss the settings for the dynamic parameters for the pressure regulator. Pressure control system response to pressure set point change is tested at various test conditions. Testing the system requires a 5 psi down set point change followed by a 5 psi up set point change when conditions stabilize. Following the 5 psi pressure change the same testing is performed for a 7 psi pressure change. Pressure regulators are tested individually and sequentially.
Feedwater System	Initial STP-20 EPU Test 23	The feedwater control system response to reactor water level set point changes (for level set point change tests) are evaluated in the indicated control mode (i.e., three element, single element). At each test condition, level set point change testing is performed by first making an up set point value change, which effects the level set point change desired, followed

Title	Test Number	Test Description
		by a down set point value change of the same value, after conditions stabilize, in accordance with the following set point change sequence. 1) + 2 inches; 3) + 3 inches; 5) + 4 inches 2) – 2 inches; 4) – 3 inches; 6) – 4 inches The 2 and 3 inch level set point steps are informational and recommended to demonstrate the level control response prior to performing the formal level set point steps (i.e., 4 inches). The results from the informational level set point steps are utilized to anticipate the responses to the formal demonstration test steps, so that effects on the reactor of the 4 inch steps may be anticipated (i.e., power increases, level alarms). The normal feedwater control system mode is three-element control, with single element control being used primarily during plant startup and under certain degraded conditions. The feedwater control system in three-element control mode should be adjusted, not only for stable operational transient level control (i.e., decay ratio), but also for stable steady-state level control (i.e., minimize reactor water limit cycles). In single element control mode, the system adjustments must achieve the operational transient level control criteria, but for steady state level control the temporary backup nature of this mode should be considered. For tests calling for manual flow step changes, at each test condition the feedwater control system is placed in a manual/auto configuration (i.e., one feedwater flow control valve (FCV) in manual and the other in automatic controlling water level). Preferably the flow step changes are made by inserting the step demand change into the feedwater FCV controller in manual or alternately by changing the set point of that controller in accordance with the following set point change sequence expressed in percent of rated EPU feedwater flow. After completion of testing on one controller, the manual/auto configuration is switched and

Title	Test Number	Test Description
		the sequence is repeated on the other controller. 1) Increase 5% 2) Decrease 5% 3) Increase 10% 4) Decrease 10% The 5% flow step changes are informational and recommended to demonstrate the feedwater turbine response prior to performing the formal test flow step changes (i.e., 10%). The results from the smaller informational flow steps are utilized to anticipate the responses to the formal demonstration test, so that any effects on the reactor may be anticipated (i.e., level changes, power increases).
Main Steam and Feedwater Piping Vibration	Initial STP-N/A EPU Test 100	During the EPU power ascension, designated main steam and feedwater piping points (i.e., locations and directions) will be monitored for vibration. Vibration monitoring points will be designated based on EPU piping vibration analysis and engineering judgment. Monitoring points may be coincidental with those in the initial startup piping vibration test or be selected as those points with the highest predicted vibration. Alternately, vibration monitoring points can be coincidental with exposed piping attachments provided that acceptance criteria is established for those points based on piping system vibration analysis. Vibration measurements taken above CLTP will permit a thorough assessment of the effect of the EPU in comparison to any previous piping vibration analysis or evaluation.

Title	Test Number	Test Description
Plant Parameter Monitoring and Evaluation	Initial STP-N/A EPU Test 101	Routine measurements of the power-dependent parameters from systems and components, affected by the EPU, are taken at 90% and 100% CLTP on a constant flow control line that will be used to increase to maximum EPU power. Power increase is along this constant rod pattern line in incremental steps of 5% or less to ensure a careful, monitored approach to maximum EPU power. Power-dependant parameters that are calculated will be calculated using accepted methods to ensure current licensed and operational practices are maintained. Measured and calculated power-dependant parameters are utilized to project those values at the next power level step prior to increasing to the next EPU test condition. Each step's projected values will be evaluated to have satisfactorily confirmed the actual values before advancing to the next step and the final increase to maximum EPU power.

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Piping Flow Induced Vibration

Monitoring Program

Enclosure 10

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Acronyms and Abbreviations

No.	Short Form	Description
1	ALARA	As Low As Reasonably Achievable (radiation dose concern)
2	ARS	Amplified Response Spectrum
3	ASME	American Society of Mechanical Engineers
4	BWR	Boiling Water Reactor
5	BWROG	Boiling Water Reactor Owners' Group
6	CLTP	Current Licensed Thermal Power
7	CPPU	Constant Pressure Power Uprate
8	EPU	Extended Power Uprate
9	FFT	Fast Fourier Transform
10	FIV	Flow Induced Vibration
11	FW	Feedwater
12	LTR	Licensing Topical Report
13	MNGP	Monticello Nuclear Generating Plant
14	MS	Main Steam
15	MSIV	Main Steam Isolation Valve
16	MSL	Main Steam Line
17	OE	Operating Experience
18	RMS	Root Mean Squared
19	SRSS	Square Root Sum of the Squares
20	SRV	Safety/Relief Valve
21	TSV	Turbine Stop Valve

1. Introduction

Sections 2.2.2 and 2.5.4.1 of Enclosure 5 to the Extended Power Uprate (EPU) submittal briefly discuss the EPU effects upon Flow Induced Vibration (FIV) for the Main Steam (MS) System and the Feedwater (FW) System. This Enclosure to the submittal provides a more detailed discussion of the analyses and testing program undertaken to provide assurance that unacceptable FIV issues are not experienced at Monticello Nuclear Generating Plant (MNGP) due to EPU implementation.

Increased flow rates and flow velocities during operation at EPU conditions are expected to produce increased FIV levels in some systems. As discussed in Section 3.4.1 of Licensing Topical Report (LTR) NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," the MS and FW piping vibration levels should be monitored because their system flow rates will be significantly increased (Reference 4). While a review of industry EPU operating experience identified very few component failures that can be attributed to EPU, most of these failures were related to FIV.

In January 2007, the Boiling Water Reactor Owners' Group (BWROG) issued NEDO-33159, Revision 1, "Extended Power Uprate (EPU) Lessons Learned and Recommendations" based on operating experience (OE) and evaluations from Boiling Water Reactor (BWR) plants that have previously implemented EPUs and from plants currently performing pre-EPU evaluations (Reference 1). NEDO-33159 states:

> "Since the majority of EPU-related component failures involve flow induced vibration, the BWROG EPU Committee held a vibration monitoring and evaluation information exchange meeting of industry experts in June 2004. The committee determined that the current process of monitoring large bore piping systems in accordance with the requirements of ASME O&M Part 3 is sufficient to preclude challenges to safe shutdown. Increases in large bore piping vibration levels are a precursor to increased vibration levels in attached small bore piping and components."

During Monticello's 23rd refueling outage, in 2007, a vibration monitoring program was implemented to support the MNGP Extended Power Uprate Project. Piping systems both inside and outside the drywell are being monitored using accelerometers. Monitoring occurs inside the drywell, turbine building and steam tunnel. The following piping is being monitored for vibration to establish baseline data prior to uprate and to ensure that the vibration levels of the selected piping systems are within acceptable limits during operation at EPU conditions:

Main Steam (Drywell and Turbine Building)
Feedwater (Drywell and Turbine Building)

The current results of the EPU Vibration Monitoring Program indicate no abnormal vibration levels exist within the MS and FW systems. Continued vibration monitoring of these systems during EPU power ascension will be performed. The same acceptance criteria established at CLTP will be applied to ensure that potential effects of flow induced vibration are captured under EPU conditions.

2. Susceptibility and Monitoring

The MS System piping and the FW System piping will have higher mass flow rates and flow velocities under EPU conditions. When power is increased from CLTP to EPU conditions, steady state FIV levels are expected to be approximately proportional to the mass flow rate squared. Thus, the vibration levels of the MS and the FW system piping are expected to increase by approximately 32% based upon a steam flow increase of 14.8%. Hence, a startup vibration monitoring program using accelerometers mounted on representative portions of the MS and FW piping located inside the containment will be required during the initial implementation of EPU.

In addition, the accessible large bore MS and FW piping outside of containment will be monitored by performing visual observations and by taking vibration measurements using hand-held vibration instruments during walkdowns of this piping. These walkdowns will be performed during initial plant operation at the EPU conditions. MS and FW piping outside of containment that is inaccessible to plant personnel when the plant is at high power levels required the installation of remote vibration monitoring sensors (completed in 2007).

Small bore piping attached to the MS and FW systems is susceptible to the effects of FIV. As stated in Section 1, the small bore piping will be evaluated as a function of the large bore piping FIV results. If the vibration level in the main piping in these systems is greater than 50% of the acceptance criteria, then an engineering evaluation of the small bore piping will be performed to ensure that the steady state stresses are within the endurance limit.

3. Remote Monitoring Program

During the spring 2007 Monticello refueling outage, accelerometers were installed on the MS and FW piping (and selected components) inside and outside of the drywell to monitor the steady state vibration levels. The purpose of collecting this data was to determine the baseline vibration levels in these systems in support of planned operation at EPU conditions. The steady state vibration levels of these two systems may increase due to EPU operating conditions. The collection of baseline data enables extrapolations to EPU operating conditions for steady state vibration levels. Data was collected at several power levels during power ascension following the outage.

3.1. Inside the Drywell Monitoring Information

The MS and FW systems are to be monitored because of their significant increases in flow to achieve increases in thermal power. The current scope monitors 16 piping locations using 39 accelerometers and three components using nine accelerometers (see Table 4-1 for locations). A modal analysis was performed on the as-modeled piping system to determine natural frequencies and mode shapes. The accelerometer locations were determined based on a review of the mode shapes. The accelerometer locations correspond to node points with high-calculated modal displacements.

3.2. Outside the Drywell Monitoring Information

50 accelerometers at 21 locations are being monitored in the steam tunnel and turbine building (see Tables 4-3 and 4-5 for locations). Similar to the drywell accelerometers, the locations and number of accelerometers in the steam tunnel and turbine building were determined based on performing modal analyses of the MS and FW piping systems.

3.3. Piping Vibration Acceptance Criteria

3.3.1. Methodology

Determination of the acceptance criteria is based on the guidance of ASME OM-S/G Part 3 (OM-3) (Reference 2). The methodology provides a pass/fail mechanism for the piping system such that, if the values are met, no further justification of the measured vibration levels is required.

3.3.2. Calculation

Detailed models of the MS and FW piping systems were developed for this evaluation. A 1g broad-band amplified response spectrum (ARS) was applied up to 250 Hz in each of the three orthogonal directions. Static loads, such as weight and thermal expansion, are not considered since these loads do not contribute to cyclic loading of the piping system. Additionally, seismic (inertia and anchor movements) and turbine stop valve loads are not considered, since these loads are transient dynamic loads that do not contribute to the steady-state cyclic loading of the system.

The results of the piping analysis are provided in terms of accelerations, displacements, and stresses at each node. The overall values at each node were obtained by combining the results for all three orthogonal directions using the SRSS method. Adjustment factors (calculated using maximum stress values and the guidance of ASME O&M-S/G Part 3) and maximum stress values (from the piping analysis) for each of these segments are presented in Table 3-1.

Table 3-1: Maximum Stresses and Adjustment Factors for Various Piping Segments at CLTP

Pipe Name		Inside Containment			Outside Containment		
		Maximum Stress (psi)	Adjustment Factor	Reference	Maximum Stress (psi)	Adjustment Factor	Reference
MS	A	13,929	0.552	PS1A Max	16,603	0.463	MSSIA Max
	B	24,461	0.314	PS2A Max			
	C	18,484	0.416	PS3A Max			
	D	11,899	0.646	PS4A Max			
FW	A	21,633	0.356	FWSIA Node 364	69,452	0.111	FWSIA Node 201
	B	14,536	0.529	FWSIA Node 1			

The acceptance criteria are then calculated by multiplying the accelerations and displacements by the adjustment factors in Table 3-1. Sample calculations at Node 25 on the FW-B drywell piping are provided below:

$$A_x = a_{calculated} * F_{adjust} = 0.776g * 0.529 = 0.410g$$

$$A_y = a_{calculated} * F_{adjust} = 1.664g * 0.529 = 0.881g$$

$$A_z = a_{calculated} * F_{adjust} = 1.332g * 0.529 = 0.705g$$

$$D_x = d_{calculated} * F_{adjust} = 0.884in * 0.529 = 0.468in$$

$$D_y = d_{calculated} * F_{adjust} = 2.219in * 0.529 = 1.174in$$

$$D_z = d_{calculated} * F_{adjust} = 2.752in * 0.529 = 1.456in$$

3.4. Piping Vibration Data at CLTP

Baseline vibration data was obtained following the spring 2007 refueling outage, using three Structural Integrity Associates Versatile Data Acquisition Systems (SI-VersaDAS™). One SI-VersaDAS™ was used to monitor the drywell accelerometers, and the other two units monitored the steam tunnel and turbine building accelerometers independently. The data was processed as described in Section 4.2. The processed data at 100% CLTP reactor power was then compared to the calculated acceptance criteria. This comparison is provided in tabular form in Section 4.

4. Vibration Monitoring Program Results

4.1. Data Acquisition Parameters

The accelerometer data (time histories) was recorded on an SI-VersaDAS™. Each data set was recorded using a sample rate of 2500 samples per second (sps) for the duration of 2 minutes. The data is time stamped for comparison to plant process data. Data from the Drywell and Steam Tunnel are synchronized to each other as well.

4.2. Data Reduction Methodology

The accelerometer time histories were first filtered using a Chebyshev bandpass filter (data from 2-250 Hz was allowed to pass). Once the signal was bandpass filtered, each time history was converted from the time domain to the frequency domain (frequency spectra) using a Fast Fourier Transform (FFT) algorithm within MATLAB (Reference 3). An FFT was generated for each group and then all FFT groups were summed together, and divided by the number of groups to provide linearly averaged frequency spectra. Plots for each averaged frequency spectrum (amplitude, g-RMS versus frequency, Hz) were generated for each channel.

4.2.1. Drywell

Of the 48 accelerometer channels, seven are located on FW loop A, five on FW loop B, ten on main steam line (MSL) A, five on MSL B, five on MSL C, seven on MSL D, six on SRVs, and three on MSIVs. The channel number versus accelerometer location is summarized in Table 4-1. The piping accelerations versus allowable values are provided Table 4-2. The analysis of the data was done using MATLAB, and the results are summarized below:

- In all cases, the magnitude of the vibration is low. The RMS magnitudes are generally below 0.06 g with the exception of channels 14, 23, and 24 having magnitudes residing below 0.09 g, which is considered low steady state vibration levels.
- At 100% reactor power MS A, MS B, and SRV showed consistent low magnitude vibration in the 2-50 Hz range. Similarly MSIV shows a low magnitude vibration at approximately 128 Hz.
- Channel 14 experienced elevated broadband noise in upper reactor power levels, 80%-95%.
- Channel 1 shows an isolated 0.028 g-RMS response at approximately 195 Hz at 50% reactor power. Likewise, channel 11 recorded a 0.048 g-RMS response at approximately 115 Hz and 85% power.
- Channels 23 and 24 experienced an elevated response at approximately 170 Hz and 50% power with the latter at lower overall amplitude, below 0.06 g-RMS.

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Table 4-1: Drywell Accelerometer Locations

Ch No	Node No	Channel Name	Dir	System	OD (in)	Location Description
1	10	ACC-FWB-10X	X	FW, Loop B riser	10.75	6 ft 5 in upstream of support FW39B
2		ACC-FWB-10Z	Z			
3	25	ACC-FWB-25X	X	FW, Loop B header	10.75	6 ft 8 in upstream of support FW38B
4		ACC-FWB-25Y	Y			
5		ACC-FWB-25Z	Z			
6	340	ACC-FWA-340X	X	FW, Loop A header	10.75	7 ft upstream of support FW38A
7		ACC-FWA-340Y	Y			
8		ACC-FWA-340Z	Z			
9	356	ACC-FWA-356X	X	FW, Loop A riser	10.75	3 ft 1 in upstream of support FW39A
10		ACC-FWA-356Z	Z			
11	376	ACC-FWA-376X	X	FW, Loop A riser	10.75	2 ft 3 in upstream of support FW36A
12		ACC-FWA-376Z	Z			
13	111, PS1A	ACC-MSA-111X	X	MSL A riser	18	13 ft 9 in downstream of support MSH1A
14		ACC-MSA-111Z	Z			
15	177, PS1A	ACC-MSA-177X	X	MSL A, disch line	10.75	4 ft 11 in upstream of support RVH70
16		ACC-MSA-177Y	Y			
17		ACC-MSA-177Z	Z			
18	242, PS1A	ACC-MSA-242X	X	MSL A, header	10.75	9 ft 11 in downstream of support 24AH2
19		ACC-MSA-242Y	Y			
20		ACC-MSA-242Z	Z			
21	256, PS1A	ACC-MSA-256X	X	MSL A, header	10.75	1 ft 3 in upstream of support 24AH3
22		ACC-MSA-256Y	Y			
23	134, PS2A	ACC-MSB-134X	X	MSL B, riser	18	7 ft 4 in downstream of support MSH3B
24		ACC-MSB-134Z	Z			
25	241, PS2A	ACC-MSB-241X	X	MSL B, disch line	10.75	5 ft 2 in upstream of support 25H2
26		ACC-MSB-241Y	Y			
27		ACC-MSB-241Z	Z			
28	111, PS3A	ACC-MSC-111X	X	MSL C, riser	18	11 ft 1 in downstream of support MSH1C
29		ACC-MSC-111Z	Z			
30	173, PS3A	ACC-MSC-173X	X	MSL C, disch line	10.75	4 ft 1 in upstream of support RVH77
31		ACC-MSC-173Y	Y			
32		ACC-MSC-173Z	Z			
33	140, PS4A	ACC-MSD-140X	X	MSL D, header	18	7 ft 6 in downstream of support MSH4D
34		ACC-MSD-140Z	Z			
35	184, PS4A	ACC-MSD-184X	X	MSL D, disch line	10.75	7 ft 4 in downstream of support 27H6
36		ACC-MSD-184Z	Z			
37	261, PS4A	ACC-MSD-261X	X	MSL D, header	10.75	4 ft 9 in downstream of support 27AH5
38		ACC-MSD-261Y	Y			
39		ACC-MSD-261Z	Z			
40	212, PS2	ACC-SRV-212X	X	MSL B, SRV	15.5	Inlet flange of SRV RV 2-71B
41		ACC-SRV-212Y	Y			
42		ACC-SRV-212Z	Z			
43	149, PS3	ACC-SRV-149X	X	MSL C, SRV	15.5	Inlet flange of SRV RV 2-71C
44		ACC-SRV-149Y	Y			
45		ACC-SRV-149Z	Z			
46	142, PS4	ACC-MSIV-142X	X	MSL D, MSIV	6	MSIV stuffing box
47		ACC-MSIV-142Y	Y			
48		ACC-MSIV-142Z	Z			

Table 4-2: Drywell Piping Accelerometer Data Comparison for CLTP

Channel No	Channel Name	Acceptance Criteria	Measured Value (100% Power)	Measured % of Acceptable Value
1	ACC-FWB-10X	0.9413	0.0579	6.15%
2	ACC-FWB-10Z	0.9375	0.0580	6.19%
3	ACC-FWB-25X	0.4105	0.0268	6.53%
4	ACC-FWB-25Y	0.8807	0.0492	5.58%
5	ACC-FWB-25Z	0.7050	0.0680	9.65%
6	ACC-FWA-340X	0.2441	0.0296	12.12%
7	ACC-FWA-340Y	0.5501	0.0510	9.28%
8	ACC-FWA-340Z	0.4911	0.0696	14.18%
9	ACC-FWA-356X	0.6805	0.0532	7.82%
10	ACC-FWA-356Z	0.5842	0.0616	10.55%
11	ACC-FWA-376X	0.6620	0.0591	8.93%
12	ACC-FWA-376Z	0.4566	0.0167	3.67%
13	ACC-MSA-111X	0.9147	0.0802	8.77%
14	ACC-MSA-111Z	0.8416	0.0000	N/A
15	ACC-MSA-177X	0.4952	0.0540	10.91%
16	ACC-MSA-177Y	0.3819	0.0241	6.30%
17	ACC-MSA-177Z	1.4204	0.0409	2.88%
18	ACC-MSA-242X	0.5439	0.0236	4.34%
19	ACC-MSA-242Y	0.9331	0.0311	3.33%
20	ACC-MSA-242Z	0.8315	0.0361	4.35%
21	ACC-MSA-256X	1.0117	0.0000	N/A
22	ACC-MSA-256Z	0.9335	0.0000	N/A
23	ACC-MSB-134X	0.2806	0.0907	32.32%
24	ACC-MSB-134Z	0.4843	0.0000	N/A
25	ACC-MSB-241X	0.7518	0.0009	0.12%
26	ACC-MSB-241Y	0.5216	0.0181	3.47%
27	ACC-MSB-241Z	0.3643	0.0167	4.60%
28	ACC-MSC-111X	0.5387	0.0625	11.60%
29	ACC-MSC-111Z	0.5958	0.0858	14.40%
30	ACC-MSC-173X	0.6608	0.0129	1.96%
31	ACC-MSC-173Y	0.9322	0.0109	1.16%
32	ACC-MSC-173Z	0.6516	0.0363	5.57%
33	ACC-MSD-140X	0.2929	0.0609	20.79%
34	ACC-MSD-140Z	0.7017	0.0775	11.05%
35	ACC-MSD-184X	0.7458	0.0261	3.50%
36	ACC-MSD-184Z	1.3138	0.0220	1.68%
37	ACC-MSD-261X	0.9577	0.0048	0.50%
38	ACC-MSD-261Y	1.2356	0.0180	1.45%
39	ACC-MSD-261Z	1.0258	0.0249	2.43%

Note: A field with N/A indicates that the measured value for that particular channel was invalid. Invalid values were set to zero in the tables and subsequent plots.

4.2.2. Steam Tunnel

Of the 24 accelerometer channels, two are located on FW loop A, two on FW loop B, two on MSL A, and six on each of the remaining three MSL (B, C, and D). The channel number versus accelerometer location is summarized in Table 4-3. The piping accelerations versus allowable values are provided Table 4-4. The analysis of the data was done using MATLAB, and the results are summarized below:

- In all cases, the magnitude of the vibration is low. The RMS magnitudes are all below 0.2142 g, and the Max-Min values are all below 2.0 g.
- FW RMS acceleration trends show a gradual increase in vibration from low to high power levels.
- MS A RMS acceleration trend shows almost equal vibration level at 39% and 100% power with lower levels in between.
- MS B, C, and D RMS acceleration trends show fairly constant vibration levels across all power levels with 100% vibrations being generally the highest.

Table 4-3: Steam Tunnel Accelerometer Channels and Locations

Ch No	Node No	Channel Name	Dir	System	OD (in)	Location Description
33	89	ACC-FWB-89Y	Y	FW, Loop B	14	3 ft 5 in upstream of support FW29
34		ACC-FWB-89Z	Z			
35	296	ACC-FWA-296Y	Y	FW, Loop A	14	2 ft 2 in upstream of support FW21
36		ACC-FWA-296Z	Z			
37	115	ACC-MSA-115X	X	MSL A	18	11 ft 7 in upstream of support PS-2
38		ACC-MSA-115Z	Z			
39	197	ACC-MSB-197X	X	MSL B	18	6 ft 2 in downstream of support PS-6
40		ACC-MSB-197Z	Z			
41	240	ACC-MSC-240X	X	MSL C	18	3 ft 3 in downstream of support PS-11
42		ACC-MSC-240Z	Z			
43	277	ACC-MSD-277Y	Y	MSL D	18	2 ft 6 in downstream of support PS-17
44		ACC-MSD-277Z	Z			
45	207	ACC-MSB-L1X	X	Location 1	20.5	Outboard MSIV B
46		ACC-MSB-L1Z	Z			
47		ACC-MSB-L2X	X	Location 2	20	
48		ACC-MSB-L2Z	Z			
49	247	ACC-MSC-L1X	X	Location 1	20.5	Outboard MSIV C
50		ACC-MSC-L1Z	Z			
51		ACC-MSC-L2X	X	Location 2	20	
52		ACC-MSC-L2Z	Z			
53	289	ACC-MSD-L1X	X	Location 1	20.5	Outboard MSIV D
54		ACC-MSD-L1Z	Z			
55		ACC-MSD-L2X	X	Location 2	20	
56		ACC-MSD-L2Z	Z			

Table 4-4: Steam Tunnel Piping Accelerometer Data Comparison for CLTP

Channel No	Channel Name	Acceptance Criteria	Measured Value (100% Power)	Measured % of Acceptable Value
33	ACC-FWB-89Y	0.1059	0.0452	42.63%
34	ACC-FWB-89Z	0.1479	0.0000	N/A
35	ACC-FWA-296Y	0.1143	0.0343	30.05%
36	ACC-FWA-296Z	0.2341	0.0305	13.01%
37	ACC-MSA-115X	0.4202	0.1197	28.50%
38	ACC-MSA-115Z	1.0776	0.0000	N/A
39	ACC-MSB-197X	0.4032	0.1327	32.91%
40	ACC-MSB-197Z	0.7628	0.1994	26.15%
41	ACC-MSC-240X	0.2168	0.0000	N/A
42	ACC-MSC-240Z	0.6433	0.0715	11.11%
43	ACC-MSD-277Y	0.6951	0.0000	N/A
44	ACC-MSD-277Z	1.1037	0.1086	9.84%

Note: A field with N/A indicates that the measured value for that particular channel was invalid. Invalid values were set to zero in the tables and subsequent plots.

4.2.3. Turbine Building

Of the 26 accelerometer channels, four are located on MSL A, four on MSL B, two on MSL C, and two on MSL D. Eight accelerometers are installed on the FW piping and six accelerometers are installed on turbine stop valves (TSV) (see Figure 4-1).

Table 4-5 lists the accelerometer channel numbers, description, and direction for the accelerometers installed on the MS and FW systems in the Turbine Building. The piping accelerations versus allowable values are provided Table 4-6. The analysis of the data was done using MATLAB, and the results are summarized below:

- The RMS magnitudes for all the piping locations (Channels 1 through 20) are below 0.3 g, and the corresponding maximum-minimum values at 100% power are less than 2.5 g.
- No significant acoustic response is apparent in the frequency spectra. Some channels show peaks in the 15-45 Hz range, which may be acoustic, but these are of very low magnitude. This indicates that there is sufficient separation between the acoustic and vortex shedding frequencies in the main steam safety relief valve branch lines.
- The measured acceleration values for the TSV location 1 are below 0.5 g-RMS. The measured acceleration values for TSV location 2 vertical direction decrease from 0.9 g-RMS at 39% power to 0.26 g-RMS at 100% power. The other two orthogonal directions never exceed 0.03 g-RMS. Therefore, it is suspected that the cable connections for Channel 25 may

be bad and it is planned to check these cable connections during the next outage (of sufficient duration) to confirm if the signal for Channel 25 is valid.

Table 4-5: Turbine Building Accelerometer Channels and Locations

Ch No	Channel name	System	Details	Direction	Description
1	Ch.1-FWB-94X	FW Loop B Riser	14"	X	4 ft 3 in downstream of support FW28
2	Ch.2-FWB-94Z			Z	
3	Ch.3-FWB-105X	FW Loop B	14"	X	5 ft 7 in from support FX201 downstream
4	Ch.4-FWB-105Y			Y	
5	Ch.5-FWX-147Y	Header between Loops A & B	14"	Y	4 ft 1 in from Loop A towards support FW30
6	Ch.6-FWX-147Z			Z	
7	Ch.7-FWA-152X	Riser, Loop A	14"	X	4 ft 3 in from support FW20 upstream
8	Ch.8-FWA-152Z			Z	
9	Ch.9-MSA-120Y	MSL A	18"	Y	7 ft 3 in downstream from support PS-2
10	Ch.10-MSA-120Z			Z	
11	Ch.11-MSA-126X	MSL A	18"	X	6 ft 6 in upstream of support PS-4
12	Ch.12-MSA-126Y			Y	
13	Ch.13-MSB-186X	MSL B	18"	X	6 ft downstream of support PS-8
14	Ch.14-MSB-186Y			Y	
15	Ch.15-MSB-192Y	MSL B	18"	Y	5 ft 6 in downstream of support PS-7
16	Ch.16-MSB-192Z			Z	
17	Ch.17-MSC-233Y	MSL C	18"	Y	7 ft 3.5 in downstream of support PS-12
18	Ch.18-MSC-233Z			Z	
19	Ch.19-MSD-300X	MSL D	18"	X	13 ft 3 in downstream of support PS-19
20	Ch.20-MSD-300Y			Y	
21	Ch.21-TSV1-SV4X	Location 1 SV-4	27"	X	See Figure 4-1
22	Ch.22-TSV1-SV4Y			Y	
23	Ch.23-TSV1-SV4Z			Z	
24	Ch.24-TSV2-SV4X	Location 2 SV-4	18"	X	See Figure 4-1
25	Ch.25-TSV2-SV4Y			Y	
26	Ch.26-TSV2-SV4Z			Z	

Table 4-6: Turbine Building Piping Accelerometer Data Comparison for CLTP

Channel No	Channel Name	Acceptance Criteria	Measured Value (100% Power)	Measured % of Acceptable Value
1	ACC-FWB-94X	0.1196	0.0277	23.16%
2	ACC-FWB-94Z	0.1173	0.0229	19.48%
3	ACC-FWB-105X	0.1374	0.0227	16.50%
4	ACC-FWB-105Y	0.1349	0.0253	18.73%
5	ACC-FWX-147Y	0.1438	0.0311	21.64%
6	ACC-FWX-147Z	0.1371	0.0198	14.45%
7	ACC-FWA-152X	0.1068	0.0221	20.68%
8	ACC-FWA-152Z	0.1149	0.0008	0.72%

Channel No	Channel Name	Acceptance Criteria	Measured Value (100% Power)	Measured % of Acceptable Value
9	ACC-MSA-120Y	0.9021	0.1061	11.76%
10	ACC-MSA-120Z	0.9471	0.0000	N/A
11	ACC-MSA-126X	0.7980	0.1126	14.11%
12	ACC-MSA-126Y	0.7448	0.0518	6.95%
13	ACC-MSB-186X	0.7335	0.1601	21.83%
14	ACC-MSB-186Y	0.5855	0.1156	19.74%
15	ACC-MSB-192Y	0.9248	0.0829	8.97%
16	ACC-MSB-192Z	0.5933	0.1362	22.96%
17	ACC-MSC-233Y	1.0651	0.1121	10.53%
18	ACC-MSC-233Z	0.7107	0.0135	1.90%
19	ACC-MSD-300X	0.7276	0.2472	33.98%
20	ACC-MSD-300Y	1.5793	0.1395	8.83%

Note: A field with N/A indicates that the measured value for that particular channel was invalid. Invalid values were set to zero in the tables and subsequent plots.

4.3. Results for All Accelerometers

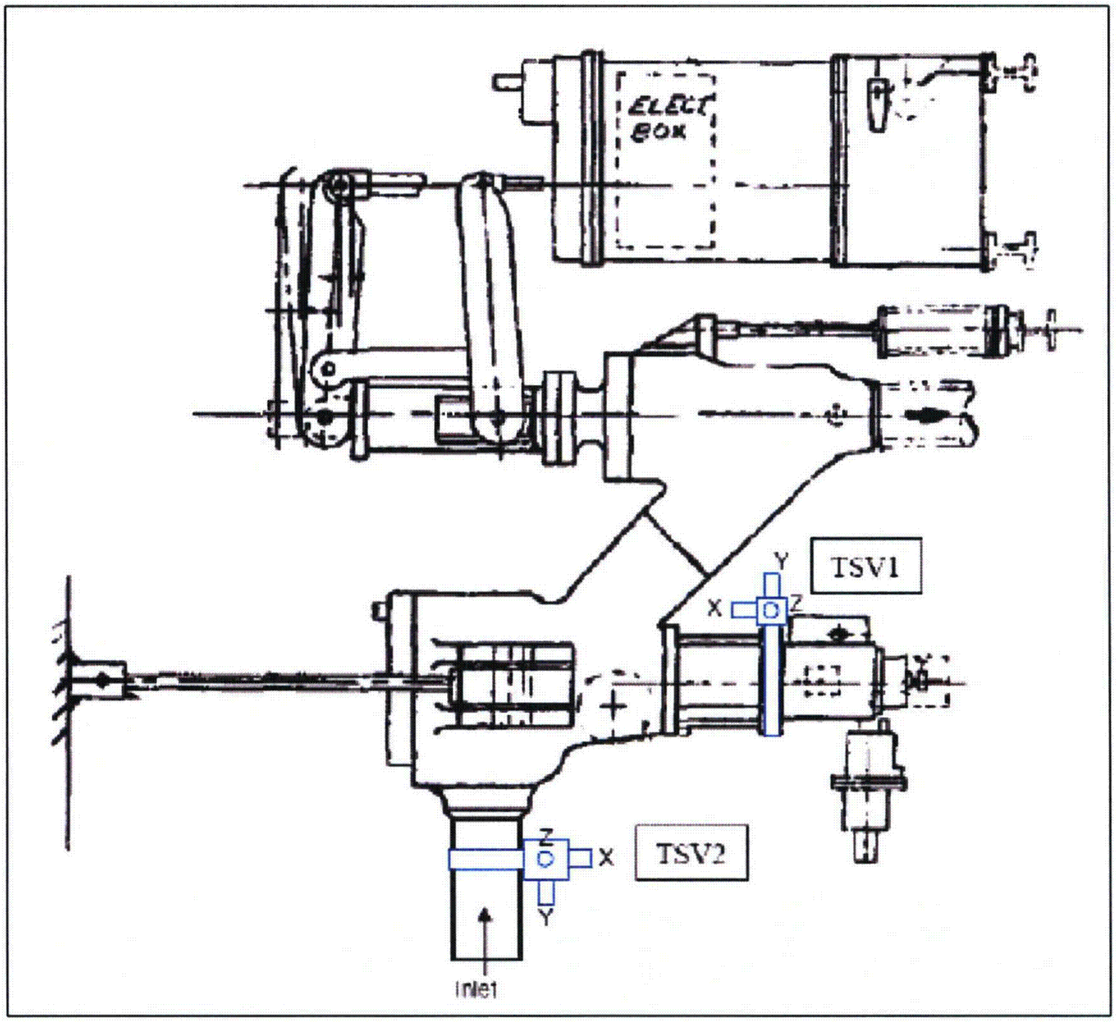
In summary, the maximum acceleration observed for the 100% (CLTP) power level in the FW piping inside the containment was 14% of the criterion. The maximum acceleration of the MS piping inside the containment was 32% of the criterion. The corresponding percentages for the FW and MS systems outside the containment were 43% and 34%, respectively.

4.4. Projected Results for EPU

Applying the expected increase of approximately 32% (based upon a steam flow increase of 14.8%) to the maximum acceleration as a percentage of the acceptance criterion (43% from Section 4.3) predicts the maximum acceleration at EPU conditions will be less than 57% of the acceptance criterion. Therefore, MS and FW piping vibration levels at EPU conditions are expected to be acceptable and vibration monitoring, as part of power ascension testing, will verify acceptable vibration levels.

Enclosure 10

Figure 4-1: Accelerometer positions on TSV



5. References

1. BWR Owners' Group EPU Committee, Extended Power Uprate (EPU) Lessons Learned and Recommendations, NEDO-33159 Revision 1, January 2007
2. ASME O&M-S/G, Standards and Guides for Operation and Maintenance of Nuclear Power Plants, Part 3, 1994 Edition, "Requirements for Preoperational and Initial Start-Up Vibration Testing of Nuclear Power Plant Piping Systems."
3. MATLAB, Version 7.0.4.365, Release 14, Mathworks, January 2005 (Macro: UniPro Version 2.3.4).
4. GE Nuclear Energy, "Constant Pressure Power Uprate," Licensing Topical Report NEDC-33004P-A, Revision 4, Class III (Proprietary), July 2003; and NEDO-33004, Class I (Non-proprietary), July 2003.

Enclosure 12 to L-MT-08-052

Continuum Dynamics, Inc
Affidavit

Continuum Dynamics, Inc.

(609) 538-0444 (609) 538-0464 fax 34 Lexington Avenue Ewing, NJ 08618-2302

5 November 2008

Mr. Steve Hammer
Monticello Nuclear Generating Plant
Northern States Power Company
West Colony Road 75
Monticello, MN 55369

SUBJECT: C.D.I. Report 07-25P "Acoustic and Low Frequency Hydrodynamic Loads at CLTP Power Level on Monticello Steam Dryer to 200 Hz," Revision 4;
C.D.I. Report 07-26P "Stress Assessment of Monticello Steam Dryer," Revision 2; and
C.D.I. Technical Note 08-12P "Limit Curve Analysis with ACM Rev. 4 for Power Ascension at Monticello," Revision 2

To Whom It May Concern:

Enclosed please find copies of the above referenced documents which are being submitted to Nuclear Management Company, LLC in accordance with the Non-Disclosure Secrecy Agreement between Continuum Dynamics, Inc. and Nuclear Management Company, LLC dated December 5, 2006. Should the documents be submitted to the Nuclear Regulatory Commission, we have also enclosed an Affidavit requesting withholding of disclosure, in accordance with 10 CFR 2.390.

Very truly yours,

Alan J. Bilanin

Alan J. Bilanin
President and Senior Associate

08200
Enclosures

Continuum Dynamics, Inc.

(609) 538-0444 (609) 538-0464 fax 34 Lexington Avenue Ewing, NJ 08618-2302

AFFIDAVIT

Re: C.D.I. Report 07-25P "Acoustic and Low Frequency Hydrodynamic Loads at CLTP Power Level on Monticello Steam Dryer to 200 Hz," Revision 4; C.D.I. Report 07-26P "Stress Assessment of Monticello Steam Dryer," Revision 2; and C.D.I. Technical Note 08-12P "Limit Curve Analysis with ACM Rev. 4 for Power Ascension at Monticello," Revision 2

I, Alan J. Bilanin, being duly sworn, depose and state as follows:

1. I hold the position of President and Senior Associate of Continuum Dynamics, Inc. (hereinafter referred to as C.D.I.), and I am authorized to make the request for withholding from Public Record the Information contained in the documents described in Paragraph 2. This Affidavit is submitted to the Nuclear Regulatory Commission (NRC) pursuant to 10 CFR 2.390(a)(4) based on the fact that the attached information consists of trade secret(s) of C.D.I. and that the NRC will receive the information from C.D.I. under privilege and in confidence.

2. The Information sought to be withheld, as transmitted to Nuclear Management LLC as attachments to C.D.I. Letter No. 08200 dated 5 November 2008, C.D.I. Report 07-25P "Acoustic and Low Frequency Hydrodynamic Loads at CLTP Power Level on Monticello Steam Dryer to 200 Hz," Revision 4; C.D.I. Report 07-26P "Stress Assessment of Monticello Steam Dryer," Revision 2; and C.D.I. Technical Note 08-12P "Limit Curve Analysis with ACM Rev. 4 for Power Ascension at Monticello," Revision 2.

3. The Information summarizes:

   (a) a process or method, including supporting data and analysis, where prevention of its use by C.D.I.'s competitors without license from C.D.I. constitutes a competitive advantage over other companies;
   (b) Information which, if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product;
   (c) Information which discloses patentable subject matter for which it may be desirable to obtain patent protection.

   The information sought to be withheld is considered to be proprietary for the reasons set forth in paragraphs 3(a), 3(b) and 3(c) above.

4. The Information has been held in confidence by C.D.I., its owner. The Information has consistently been held in confidence by C.D.I. and no public disclosure has been made and it is not available to the public. All disclosures to third parties, which have been limited, have been made pursuant to the terms and conditions contained in C.D.I.'s Nondisclosure Secrecy Agreement which must be fully executed prior to disclosure.

5. The Information is a type customarily held in confidence by C.D.I. and there is a rational basis therefore. The Information is a type, which C.D.I. considers trade secret and is held in confidence by C.D.I. because it constitutes a source of competitive advantage in the competition and performance of such work in the industry. Public disclosure of the Information is likely to cause substantial harm to C.D.I.'s competitive position and foreclose or reduce the availability of profit-making opportunities.

I declare under penalty of perjury that the foregoing affidavit and the matters stated therein are true and correct to be the best of my knowledge, information and belief.

Executed on this 5th day of November 2008.

Alan J. Bilanin
Continuum Dynamics, Inc.

Subscribed and sworn before me this day: November 5, 2008

Eileen P. Burmeister, Notary Public

EILEEN P. BURMEISTER
NOTARY PUBLIC OF NEW JERSEY
MY COMM. EXPIRES MAY 6, 2012