ENCLOSURE

ATTACHMENT 3

NEDO-33351, Safety Analysis Report for Nine Mile Point Nuclear Station Unit 2 Constant Pressure Power Uprate (PUSAR) (non-proprietary version)

Nine Mile Point Nuclear Station, LLC
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HITACHI

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Non-Proprietary Information

SAFETY ANALYSIS REPORT

FOR

NINE MILE POINT NUCLEAR STATION UNIT 2

CONSTANT PRESSURE POWER UPRATE

INFORMATION NOTICE

This is a non-proprietary version of the document NEDC-33351P, Revision 0, which has the proprietary information removed. Portions of the document that have been removed are indicated by an open and closed bracket as shown here [[ ]].

IMPORTANT NOTICE REGARDING

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Please Read Carefully

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ACRONYMS AND ABBREVIATIONS

Term	Definition
AAC	Alternate AC Sources
ABA	Amplitude Based Algorithm
AC	Alternating Current
ADHR	Alternate Decay Heat Removal
ADS	Automatic Depressurization System
AFIL	Acoustic and Flow Induced Loads
AHC	Access Hole Cover
AL	Analytical Limit
ALARA	As Low As reasonable Achievable
ALT	As Left Tolerance
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOO	Anticipated Operational Occurrence (moderate frequency transient event)
AOP	Alternate Operating Procedure
AOR	Analysis of Record
AOV	Air-Operated Valve
AP	Annulus Pressurization
APRM	Average Power Range Monitor
ARI	Alternate Rod Insertion
ARS	Acceleration Response Spectra
ARTS	APRM/RBM/Technical Specifications
ASME	American Society of Mechanical Engineers
ASDC	Alternate Shutdown Cooling
AST	Alternate Source Term
ATU	Analog Trip Unit
ATWS	Anticipated Transient Without Scram
AV	Allowable Value
AWLZ	Above Water Level Zero

Term	Definition
BHP	Brake Horsepower
BIIT	Boron Injection Initiation Temperature
BOC	Beginning of Cycle
BOP	Balance-of-Plant
B&PV	Boiler and Pressure Vessel
BPWS	Banked Position Withdrawal Sequence
BSP	Backup Stability Protection
BSW	Biological Shield Wall
BTU	British Thermal Unit
BWR	Boiling Water Reactor
BWROG	BWR Owners Group
BWRVIP	BWR Vessel and Internals Project
CAD	Containment Atmospheric Dilution
CBP	Condensate Booster Pump
CDF	Core Damage Frequency
CFD	Condensate Filter Demineralizer
CFR	Code of Federal Regulations
CFS	Condensate and Feedwater System
CGCS	Combustible Gas Control System
CGG	Constellation Generation Group LLC
CLTP	Current Licensed Thermal Power
CLTR	Constant Pressure Power Uprate Licensing Topical Report
CMTR	Certified Material Test Report
CO	Condensation Oscillation
COLR	Core Operating Limits Report
CPPU	Constant Pressure Power Uprate
CPR	Critical Power Ratio
CRD	Control Rod Drive
CRDA	Control Rod Drop Accident

Term	Definition
CRAVS	Control Room Area Ventilation System
CREF	Control Room Emergency Filtration System
CREVS	Control Room Emergency Ventilation System
CRGT	Control Rod Guide Tube
CRHZ	Control Room Habitability Zone
CS	Core Spray
CSC	Containment Spray Cooling
CSH	High Pressure Core Spray
CSL	Low Pressure Core Spray
CST	Condensate Storage Tank
CT	Current Transformer
CUF	Cumulative Usage Factors
CWS	Circulating Water System
DBA	Design Basis Accident
DBLOCA	Design Basis Loss-of-Coolant Accident
DC	Direct Current
DFFR	Dynamic Forcing Function Report
DIVOM	Delta CPR over Initial CPR Versus Oscillation Magnitude
DLO	Dual (Recirculation) Loop Operation
DVS	Drywell Ventilation System
DW	Dry Well
EAB	Exclusion Area Boundary
ECCS	Emergency Core Cooling System
EECW	Emergency Equipment Cooling Water
EFDS	Equipment and Floor Drainage System
EFPY	Effective Full Power Years
ELLLA	Extended Load Line Limit Analysis
ELTR1	Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate Licensing Topical Report
ELTR2	Generic Evaluations of General Electric Boiling Water Reactor Extended

Term	**Definition**
	Power Uprate Licensing Topical Report
EOC	End of Cycle
EOP	Emergency Operating Procedure
EPRI	Electric Power Research Institute
EPU	Extended Power Uprate
EQ	Environmental Qualification
ESF	Engineered Safety Feature
ESFAS	Engineered Safety Feature Actuation System
ESFVS	Engineered Safety Feature Ventilation System
FAC	Flow Accelerated Corrosion
FCV	Flow Control Valve
FFWTR	Final Feedwater Temperature Reduction
FHA	Fuel Handling Accident
FIV	Flow Induced Vibration
FLIM	Failure Likelihood Index Methodology
FLL	Fuel Lift Loads
FPC	Fuel Pool Cooling
FPCCS	Fuel Pool Cooling and Cleanup System
FPP	Fire Protection Program
FSAR	Final Safety Analysis Report
FV	Fussel-Vesely
FW	Feedwater
FWCF	Feedwater Controller Failure Maximum Demand
FWHOOS	Feedwater Heater Out-of-Service
FWP	Feedwater Pump
FWS	Feedwater System
FWTR	Feedwater Temperature Reduction
GDC	General Design Criteria
GE	General Electric
GEH	GE-Hitachi Nuclear Energy Americas LLC

Term	**Definition**
GL	Generic Letter
GNF	Global Nuclear Fuel LLC
GRA	Growth Rate Based Algoritham
GSF	Generic Shape Function
GSU	Generator Step Up
GWMS	Gaseous Waste Management (Offgas) System
HCR	Human Cognitive Reliability
HCTL	Heat Capacity Temperature Limit
HELB	High Energy Line Break
HEP	Human Error Probability
HEPA	High Efficiency Particulate Air
HFCL	High Flow Control Line
Hg_a	Inches of Mercury Absolute
HPCS	High Pressure Coolant Spray
HPT	High Pressure Turbine
HRA	Human Reliability Analysis
HVAC	Heating Ventilating and Air Conditioning
HWL	High Water Level
HX	Heat Exchanger
IASCC	Irradiation-Assisted Stress Corrosion Cracking
ICA	Interim Corrective Action
ICF	Increased Core Flow
IEB	Inspection & Enforcement Bulletins
ICS	Integrated Computer System
IEEE	Institute of Electrical and Electronics Engineers
IGSCC	Intergranular Stress Corrosion Cracking
ILBA	Instrument Line Break Accident
IORV	Inadvertent Opening of a Relief Valve
IPB	Isolated Phase Bus

Term	Definition
IPE	Individual Plant Examination
IPEEE	Individual Plant Examination of External Events
IRM	Intermediate Range Monitor
ISLOCA	Interfacing System Loss-of-Coolant Accident
ISP	Integrated Surveillance Program
ISI	In-Service Inspection
IST	In-Service Testing
JR	Jet Reaction
LAT	Leave Alone Tolerance
LCS	Leakage Control System
LDS	Leak Detection System
LER	Licensee Event Report
LERF	Large Early Release Frequency
LFWH	Loss of Feedwater Heater
LHGR	Linear Heat Generation Rate
LLHS	Light Load Handling System
LOC	Loss of Condenser
LOCA	Loss-of-Coolant Accident
LOCV	Loss of Condenser Vacuum
LOFW	Loss of Feedwater
LOOP	Loss of Offsite Power
LPCI	Low Pressure Coolant Injection
LPCS	Low Pressure Core Spray
LPRM	Local Power Range Monitor
LPSP	Low Power Setpoint
LRNBP	Generator-Load Rejection with no Steam Bypass Failure
LTR	Licensing Topical Report
LWL	Low Water Level
LWMS	Liquid Waste Management System

Term	**Definition**
MAAP	Modular Accident Analysis Program
MAPLHGR	Maximum Average Planar Linear Heat Generation Rate
MBTU	Millions of BTUs
MC	Main Condenser
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 (ratio MCPR to limit)
MFLPD	Maximum Fraction of Limiting Power Density (ratio MLHGR to limit)
Mlb	Millions of Pounds
MLOCA	Medium Loss-of-Coolant Accident
MOC	Middle of Cycle
MOV	Motor Operated Valve
MS	Main Steam
MSIV	Main Steam Isolation Valve
MSIVC	Main Steam Isolation Valve Closure
MSIVF	Main Steam Isolation Valve Closure with Scram on High Flux
MSL	Main Steam Line
MSLB	Main Steam Line Break
MSLBA	Main Steam Line Break Accident
MSRV	Main Steam Relief Valve
MSS	Main Steam System
MSVV	Main Steam Valve Vault
MVA	Million Volt Amps
Mvar	Megavar
MWd	Energy Units MWd Thermal Energy
MWd/ST	Exposure Units MWd Thermal Energy Per Core Weight Short Tons

Term	Definition
MWe	Megawatts-Electric
MWt	Megawatt-Thermal
NA	Not Applicable
NCL	Natural Circulation Line
NDE	Non-Destructive Testing
NMP2	Nine Mile Point Nuclear Station Unit 2
NMPC	Niagra Mohawk Power Corporation
NMPNS	Nine Mile Point Nuclear Station, LLC
NPSH	Net Positive Suction Head
$NPSH_R$	Net Positive Suction Head Required
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
NTSP	Nominal Trip Set Point
NUREG	Nuclear Regulatory Commission Technical Report Designation
OFS	Orificed Fuel Support
OLMCPR	Operating Limit Minimum Critical Power Ratio
OLTP	Original Licensed Thermal Power
OOS	Out-of-Service
OPRM	Oscillation Power Range Monitor
ΔP	Differential Pressure - psi
P_{25}	25% of EPU Rated Thermal Power
PBDA	Period Based Detection Algorithm
PCPL	Primary Containment Pressure Limit
PCS	Pressure Control System
PCT	Peak Clad Temperature
PF	Power Factor
PLOF	Partial Loss of Feedwater Initiating Event
PRA	Probabilistic Risk Assessment
PRFD	Pressure Regulator Failure Downscale

Term	Definition
PRFO	Pressure Regulator Failure 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
PSP	Pressure Suppression Pressure
PSPL	Primary Suppression Pressure Limit
P-T	Pressure-Temperature
PUSAR	Power Uprate Safety Analysis Report
RAVS	Radwaste Area Ventilation System
RAW	Risk Achievement Worth
RBCCW	Reactor Building Closed Cooling Water
RBCLC	Reactor Building Closed Loop Cooling
RBM	Rod Block Monitor
RCIC	Reactor Core Isolation Cooling
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
RCW	Raw Cooling Water
RG	Regulatory Guide
RHS	Residual Heat Removal System
RHR	Residual Heat Removal
RHRSW	Residual Heat Removal Service Water
RIPD	Reactor Internal Pressure Difference
RLA	Reload Licensing Analysis
RPT	Recirculation Pump Trip
RPV	Reactor Pressure Vessel
RRRB	Reactor Recirculation Runback

Term	**Definition**
RRS	Reactor Recirculation System
RSLB	Recirculation System Line Break
RSP	Remote Shutdown Panel
RT_{NDT}	Reference Temperature of Nil-Ductility Transition
RTP	Rated Thermal Power
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 Limits
SAR	Safety Analysis Report
SBO	Station Blackout
SCM	Steam Condensing Mode
SDC	Shutdown Cooling
SER	Safety Evaluation Report
SFC	Spent Fuel Pool Cooling
SFP	Spent Fuel Pool
SFPC	Spent Fuel Pool Cooling
SFPAVS	Spent Fuel Pool Area Ventilation System
SGTS	Standby Gas Treatment System
SHB	Shroud Head Bolts
SIL	Service Information Letter
SJAE	Steam Jet Air Ejectors
SLCS	Standby Liquid Control System
SLMCPR	Safety Limit Minimum Critical Power Ratio
SLO	Single Loop Operation
SLOCA	Small Loss-of-Coolant Accident
SORV	Stuck Open SRV
SOV	Solenoid-Operated Valve

Term	**Definition**
SP	Suppression Pool
SPC	Suppression Pool Cooling
SPDS	Safety Parameter Display System
SPDES	State Pollutant Discharge Elimination System
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	Systems Structures Components
SSE	Safe Shutdown Earthquake
SSP	Supplemental Surveillance Capsule Program
SSV	Spring Safety Valve
STP	Simulated Thermal Power
SWMS	Solid Waste Management Systems
SWS	Station Service Water System
TAF	Top of Active Fuel
TAVS	Turbine Area Ventilation System
TBCCW	Turbine Building Closed Loop Cooling Water
TBS	Turbine Bypass System
TCV	Turbine Control Valve
TEDE	Total Effective Dose Equivalent
TFSP	Turbine First-Stage Pressure
T-G	Turbine-Generator
TIP	Traversing Incore Probe
TLO	Two Loop Operation
TSV	Turbine Stop Valve
TSVC	Turbine Stop Valve Closure
TT	Turbine Trip

Term	**Definition**
TTNBP	Turbine Trip with no Steam Bypass Failure
T_w	Time Available
UHS	Ultimate Heat Sink
UPS	Uninterruptible Power Supply
USAR	Updated Safety Analysis Report
USE	Upper Shelf Energy
VB	Vacuum Breaker
WCS	Reactor Water Cleanup System
WW	Wet Well

EXECUTIVE SUMMARY

This report summarizes the results of safety evaluations performed that justify uprating the licensed thermal power at Nine Mile Point Nuclear Station Unit 2 (NMP2). The requested licensed power level is an increase to 3988 Megawatt-Thermal (MWt) from the current licensed reactor thermal power of 3467 MWt.

GEH has previously developed and implemented Extended Power Uprate (EPU) at several nuclear power plants. Based on EPU experience, GEH 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 (CPPU) and was approved by the NRC in the Licensing Topical Report (LTR) NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," hereafter referred to as the CLTR. The CLTR was approved for Boiling Water Reactor (BWR) plants containing GE fuel types and using GEH accident analysis methods. NMP2 contains only GE fuel types and this evaluation uses only GEH accident analysis methods. By performing the power uprate in accordance with the CLTR Safety Evaluation Report (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 NMP2, 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 licensee 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. NMP2, 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 original 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 some 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 NMP2 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 this report:

- All safety aspects of NMP2 that are affected by the increase in thermal power were evaluated;
- Evaluations were performed using NRC-approved or industry-accepted analysis methods;
- Systems and components affected by EPU were reviewed to ensure there is no significant challenge to any safety system;
- 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);

1 INTRODUCTION

1.1 Report Approach

This report summarizes the results of safety evaluations that were performed to justify uprating the licensed thermal power at Nine Mile Point Nuclear Station Unit 2 (NMP2). The requested license power level is an increase to 3988 MWt from the current licensed reactor thermal power (CLTP) of 3467 MWt.

GEH has previously developed and implemented Extended Power Uprate (EPU) at several nuclear power plants. 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 (CPPU) 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 US Nuclear Regulatory Commission (NRC) approved the CLTR in the staff Safety Evaluation Report (SER) contained in Reference 1 for BWR plants containing GE fuel types and using GEH accident analysis methods. NMP2 contains only GE 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 3988 MWt, which corresponds to 120% of the original licensed thermal power (OLTP) for NMP2. This report is presented in a format consistent with the Template Safety Evaluation Report contained in Section 3.2 of the NRC, Office of Nuclear Reactor Regulation, Review Standard for Extended Power Uprates, RS-001, December 2003. The Regulatory Evaluations from the Template Safety Evaluation have been modified to reflect the licensing basis of NMP2.

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 by the NRC. 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 assessments provided in previous EPU licensing topical reports that included a pressure increase (References 2 and 3 also referred as 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. [[

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]] 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. 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 NMP2 design for operation at 120% of OLTP for a cycle length of 24 months.

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 in the original license.

The method for achieving higher power is to extend the power/flow map (Figure 1-1) along the maximum extended load line limit developed as part of 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 pre-EPU 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

NMP2 is currently licensed at the 100% CLTP level of 3467 MWt. The EPU RTP level included in this evaluation is 120% of the OLTP. 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 (Reference 5) 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 NMP2 licensing basis except where noted in this report.

1.2.3 Approach

The planned approach to achieving the higher power level consists of the change to the NMP2 licensing and design basis to increase the licensed power level to 3988 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 the MELLLA upper boundary
- No change to source term methodology
- No new fuel product line introduction
- No change to fuel cycle length
- No additions to currently licensed operational enhancements

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 uprate for NMP2. For this increase in power, the conclusions of system/component acceptability stated in the CLTR are bounding and have been confirmed for NMP2. 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:

- **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.

- **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.

- **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.

- **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, evaluations have been performed to determine the need for any Technical Specification allowable values changes for various functions (e.g., main steam line high flow isolation setpoints).

- **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.
- **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.
- **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.
- **Reactor Safety Performance Evaluations:** The limiting Updated Safety Analysis Report (USAR) analyses for design basis events have been addressed as part of 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. [[ ]]
- **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 NMP2 Probabilistic Risk Assessment (PRA) have been analyzed to demonstrate that there are no new vulnerabilities to severe accidents.

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 105% core flow with dome pressure at 1050 psia. Figure 1-4 shows the EPU heat balance at 102% of EPU RTP and 100% core flow, with dome pressure 1055 psia.

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 assure 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/flow map region along the upper boundary extension.

The NMP2 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	Y(2)	NEDE-24011-P Rev. 0 SER
Reactor Core and Fuel Performance	TGBLA PANACEA ISCOR	06 11 09	Y Y Y(2)	NEDE-30130-P-A (4) NEDE-30130-P-A (4) NEDE-24011P Rev. 0 SER
Thermal Hydraulic Stability	ISCOR PANACEA ODYSY OPRM TRACG	09 11 05 01 04	Y(2) Y Y Y(15) Y	NEDE-24011-P Rev. 0 SER NEDE-30130-P-A (4) NEDC-32992P-A, Class III, July 2001 NEDO-32465-A
RPV Fluence	TGBLA DORTG01V	06 01	Y N	See note 14 See notes 12 and 13
RPV Internals Structural Integrity Evaluation	SAP4G07V	01	NA	NEDO-10909 (1)
Reactor Internal Pressure Differences	ISCOR LAMB TRACG	09 07 02	Y(2) (3) Y	NEDE-24011P Rev. 0 SER NEDE-20566-P-A NEDE-32176P Rev. 2 NEDC-32177P Rev. 2 NRC TAC No. M90270
Transient Analysis	PANACEA ISCOR ODYN SAFER	11 09 09 04	Y Y(2) Y (5)	NEDE-30130-P-A (4) NEDE-24011-P Rev. 0 SER NEDO-24154-A NEDC-32424P-A, NEDC-32523P-A (8) (9) (10)
Anticipated Transient Without Scram	ODYN STEMP PANACEA ISCOR	09 04 11 09	Y (6) Y Y(2)	NEDE-24154P-A Supp. 1, Vol. 4 NEDE-30130-P-A NEDE-24011-P Rev. 0 SER
Containment System Response	SHEX M3CPT LAMB	05 05 08	Y Y (3)	(7) NEDO-10320, Apr. 1971 NEDE-20566-P-A September 1986
Appendix R Fire Protection	GESTR SAFER SHEX	08 04 04	(5) (5) Y	NEDE-23785-1-PA Rev. 1 (8) (9) (10) (7)
Reactor Recirculation System	BILBO	04V	NA	NEDE-23504, February 1977 (1)

Task	Computer Code*	Version or Revision	NRC Approved	Comments
ECCS-LOCA	LAMB GESTR SAFER ISCOR TASC	08 08 04 09 03A	Y Y Y Y(2) Y	NEDO-20566A NEDE-23785-1-PA Rev. 1 (8) (9) (10) NEDE-24011-P Rev. 0 SER NEDC-32084P (11)
Fission Product Inventory	ORIGEN	2.1	N	Isotope Generation and Depletion Code

* 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) 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 (GE) 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, Reactor Core and Fuel Performance and LOCA applications is consistent with the approved models and methods.

(3) The LAMB code is approved for use in ECCS-LOCA applications (NEDE-20566-P-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.

(4) 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 (GE) 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.

TGBLA06/PANAC11 was used consistent with Limitation 1 of the Safety Evaluation for Licensing Topical Report NEDC-33173P (Reference 6) in the NMP2 Core Design analysis.

(5) 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 (NRC) to J.F. Quirk

(GE), "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.

(6) 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.

(7) The application of the methodology in the SHEX code to the containment response is approved by NRC in the letter to G. L. Sozzi (GE) from A. Thadani (NRC), "Use of the SHEX Computer Program and ANSI/ANS 5.1-1979 Decay Heat Source Term for Containment Long-Term Pressure and Temperature Analysis," July 13, 1993 (Reference 7).

(8) Letter, J.F. Klapproth (GE) to NRC, 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.

(9) 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.

(10) "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.

(11) 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.

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

(13) Letter, H. N. Berkow (NRC) to G. B. Stramback (GE), "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.

(14) Letter, S.A. Richards (NRC) to G. A. Watford (GE), "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.

(15) The OPRM code is not Level 2. However, the methodology as implemented in the OPRM code has been approved by the NRC.

Table 1-2 Current and EPU Plant Operating Conditions

Parameter	Current Licensed Value[1]	EPU Value[4]
Thermal Power (MWt)	3467	3988
Vessel Steam Flow (Mlb/hr) [2]	15.002	17.636
Full Power Core Flow Range		
Mlb/hr	86.8 to 113.9	107.4 to 113.9
% Rated	80.0 to 105.0	99.0 to 105.0
Maximum Nominal Dome Pressure (psia)	1035	1035
Maximum Nominal Dome Temperature (°F)	548.8	548.8
Pressure at upstream side of turbine stop valve (TSV) (psia)	1003	991
Full Power Feedwater		
Flow (Mlb/hr)	14.970	17.604
Temperature (°F)	425.1	440.5
Core Inlet Enthalpy (Btu/lb) [3]	529.2	528.9

Notes:

1. Based on current reactor heat balance.
2. At normal FW heating.
3. At 100% core flow condition.
4. Currently licensed performance improvement features and/or equipment OOS that are included in EPU evaluations:
   (a) Single Loop Operation (SLO)
   (b) 2 SRVs Out-of-Service (OOS)
   (c) APRM/RBM/Technical Specifications (ARTS)
   (d) Maximum Extended Load Line Limit Analysis (MELLLA)
   (e) Increased core flow (ICF)
   (f) End-of-Cycle Recirculation Pump Trip (EOC RPT) OOS (RPTOOS)
   (g) Turbine Bypass OOS (TBVOOS)
   (h) MSIV Out-of-Service (MSIVOOS)
   (i) ADS Out-of-Service OOS
   (j) 20 °F FW Operational Temperature Band
   (k) 24 Month Cycle
   (l) 60-Year Plant Life

Figure 1-1 Power/Flow Operating Map for EPU

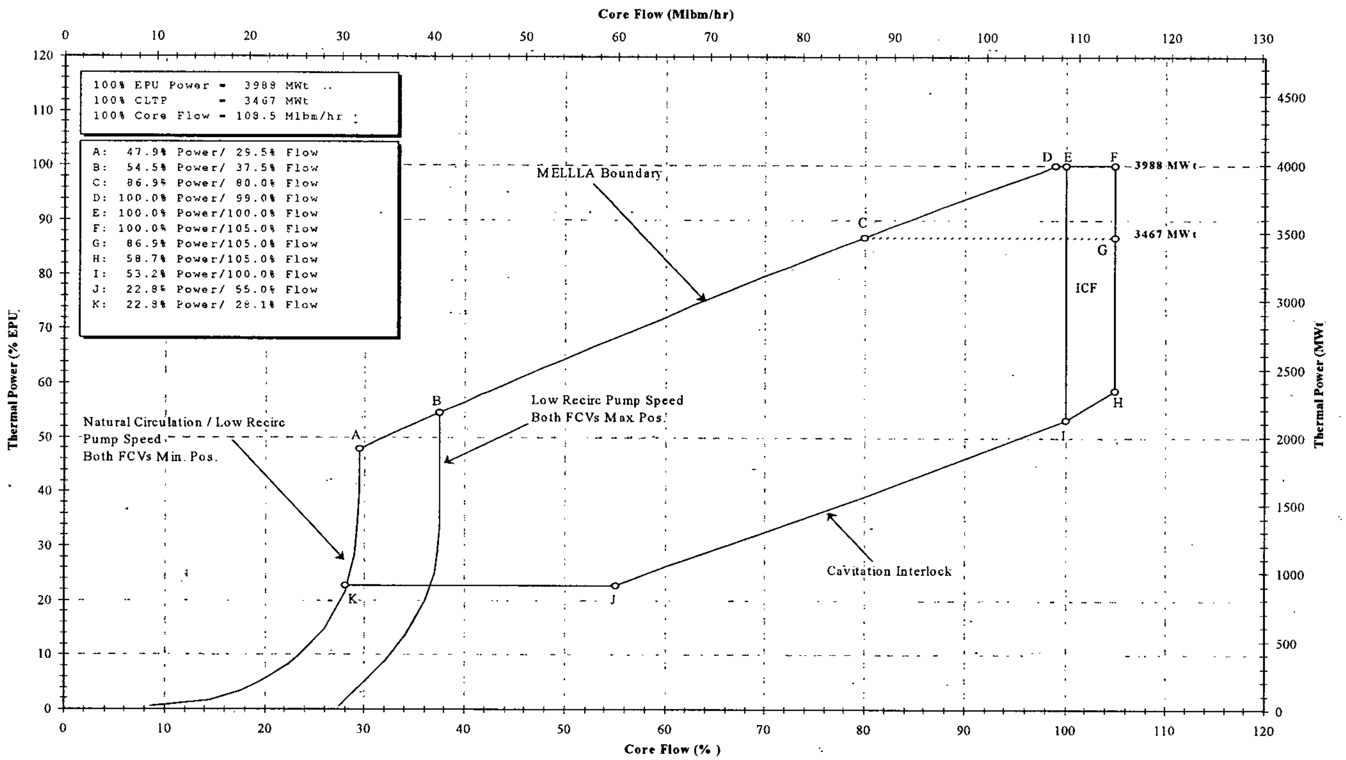
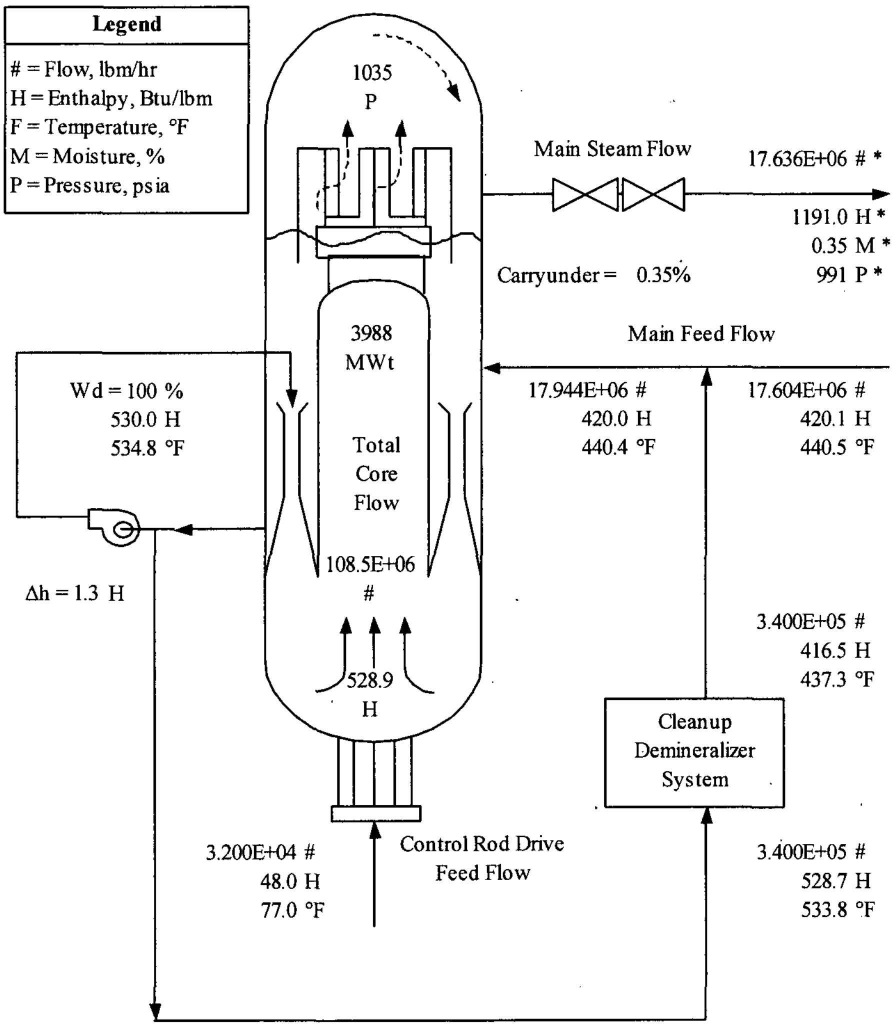


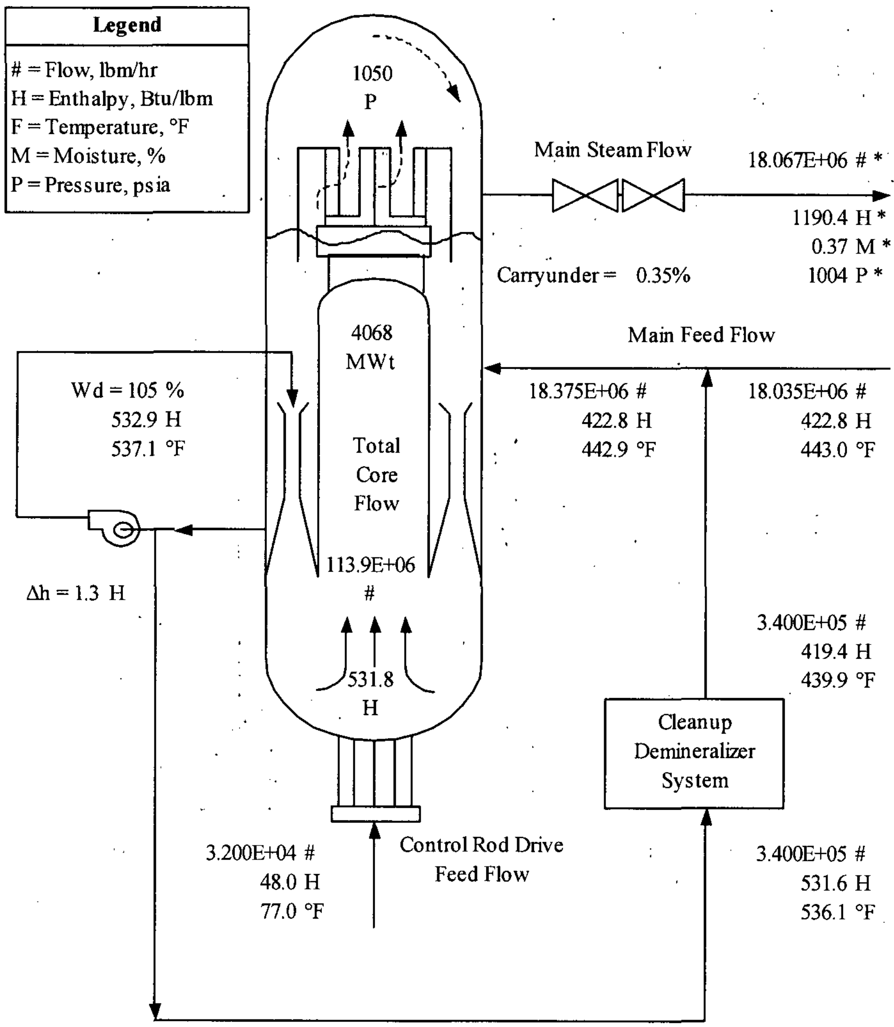
Figure 1-2 EPU Heat Balance – Nominal
(@ 100% Power and 100% Core Flow)



*Conditions at upstream side of TSV

Core Thermal Power	3988.0
Pump Heating	12.4
Cleanup Losses	-11.2
Other System Losses	-1.1
Turbine Cycle Use	3988.1 MWt

Figure 1-3 EPU Heat Balance - Overpressure Protection Analysis
(@ 102% Power and 105% Core Flow); Dome Pressure = 1050 psia

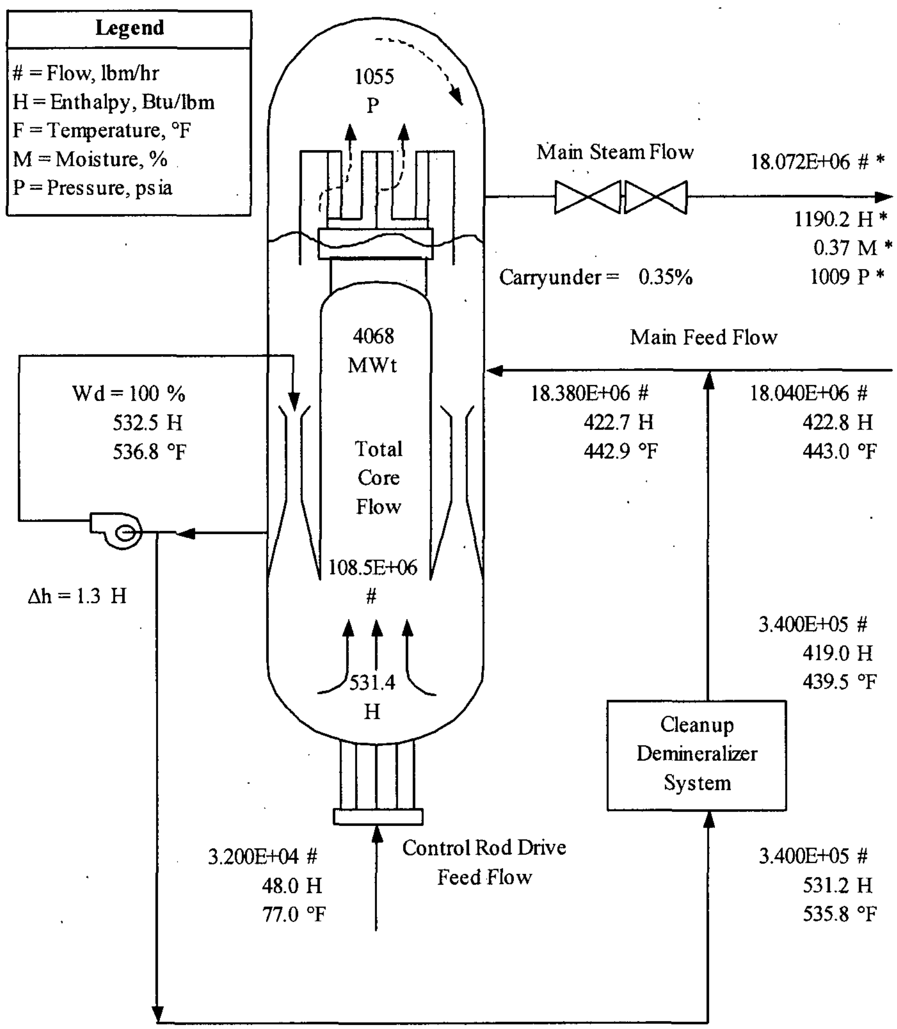


*Conditions at upstream side of TSV

Core Thermal Power	4068.0
Pump Heating	12.6
Cleanup Losses	-11.2
Other System Losses	-1.1
Turbine Cycle Use	4068.3 MWt

Figure 1-4 EPU Heat Balance - Accident (LOCA) Analysis

(@102% Power and 100% Core Flow; Dome Pressure = 1055 psia)



*Conditions at upstream side of TSV

Core Thermal Power	4068.0
Pump Heating	12.4
Cleanup Losses	-11.2
Other System Losses	-1.1
Turbine Cycle Use	4068.1 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 Nine Mile Point Nuclear Station (NMPNS) review primarily focused on the effects of the proposed EPU on the reactor vessel surveillance capsule withdrawal schedule. 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) GDC31, insofar as it requires that the RCPB be designed with margin sufficient to assure that, under specified conditions, it will behave in a nonbrittle manner and the probability of a rapidly propagating fracture is minimized; (3) 10 CFR 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 50, Appendix H.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of Constant Pressure Power Uprates. Section 3.2.1 of the CLTR addresses the effect of Constant Pressure Power Uprate on the reactor vessel material surveillance program. The results of this evaluation are described below.

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 x 10^{17} n/cm^2. This region is defined as the "beltline" region. Operation at 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 8.72 effective full power years (EFPY) of operation. The remaining two capsules have been in the reactor vessel since plant startup. NMP2 is a participant in the Integrated Surveillance Program (ISP), currently administrated by Electric Power Research Institute (EPRI), and is not designated as a representative plant; therefore, no capsules are slated for removal. 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 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

NMPNS has reviewed the effects of the proposed EPU on the reactor vessel surveillance withdrawal schedule and concludes that the changes in neutron fluence and their effects on the schedule have been adequately addressed. NMPNS further concludes that the ISP is appropriate to ensure that the material surveillance program will continue to meet the requirements of 10 CFR 50, Appendix H, and 10 CFR 50.60, and will provide NMPNS with information to ensure continued compliance with GDC14 and GDC31 in this respect following implementation of the proposed EPU. Therefore, NMPNS has determined that 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-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 NMPNS review of P-T limits covered the P-T limits methodology and the calculations for the number of effective full power years specified for the proposed EPU, considering neutron embrittlement effects and using linear elastic fracture mechanics. The NRC's acceptance criteria for P-T limits are based on (1) GDC14, 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) GDC31, insofar as it requires that the RCPB be designed with margin sufficient to assure that, under specified conditions, it will behave in a nonbrittle manner and the probability of a rapidly propagating fracture is minimized; (3) 10 CFR 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 50, Appendix G.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of Constant Pressure Power Uprates. Section 3.2.1 of the CLTR addresses the effect of Constant Pressure Power Uprate on Pressure-Temperature (P-T) Limits and Upper-Shelf Energy (USE). The results of this evaluation are described below.

The neutron fluence is recalculated using both the NRC approved GEH neutron fluence methodology (Reference 8) and the NRC approved NMP2 neutron fluence methodology (Reference 9). Both methods are consistent with RG 1.190. The revised fluence is used to evaluate the vessel against the requirements of 10 CFR 50, Appendix G. The results of these evaluations indicate that:

(a) USE will remain greater than 50 ft-lb for the design life of the vessel and maintain the margin requirements of 10 CFR 50, Appendix G. The minimum USE for the beltline materials is 61 ft-lb for 54 EFPY. These values are provided in Table 2.1-1.

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

(c) The CLTP P-T curves remain bounding for EPU, limited to the currently approved fluence. The current adjusted reference temperature values for the beltline plates and welds remain bounding for EPU due to the elimination of the conservative σ_I term. The Low Pressure Coolant Injection (LPCI) and Water Level Instrumentation nozzles that occur within the beltline region are bounded by the Upper Vessel (Feedwater nozzle) curve. The hydrotest pressure for EPU is 1035 psig.

(d) The 54 EFPY shift is decreased, and consequently, results in a change in the adjusted reference temperature, which is the initial RT_{NDT} plus the shift. These values are provided in Table 2.1-2.

(e) The 54 EFPY beltline circumferential weld material RT_{NDT} remains bounded by the requirements of BWRVIP-74. This comparison is provided in Table 2.1-3.

Conclusion

NMPNS has reviewed the effects of the proposed EPU on the P-T limits for the NMP2 and concludes that the changes in neutron fluence and their effects on the P-T limits have been adequately addressed. NMPNS further concludes the validity of the existing P-T limits remained valid through the currently approved maximum fluence for operation under the proposed EPU conditions. Based on this, NMPNS concludes that the proposed P-T limits will continue to meet the requirements of 10 CFR 50, Appendix G, and 10 CFR 50.60 and will enable NMP2 to continue to comply with GDC14 and GDC31 following implementation of the proposed EPU. Therefore, NMPNS finds the proposed EPU acceptable with respect to the proposed P-T limits.

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 NMPNS review covered the materials' specifications and mechanical properties, welds, weld controls, nondestructive examination procedures, corrosion resistance, and susceptibility to degradation. The NRC's acceptance criteria for reactor internal and core support materials are based on GDC1 and 10 CFR 50.55a for material specifications, controls on welding, and inspection of reactor internals and core supports.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of Constant Pressure Power Uprates. Sections 3.3 and 10.7 of the CLTR addresses the effect of Constant Pressure Power Uprate on Reactor Internal and Core Support Materials. The results of this evaluation are described below.

The reactor internal and core support materials evaluation included the materials' specifications and mechanical properties, welds, weld controls, nondestructive examination procedures, corrosion resistance, and susceptibility to degradation. This evaluation of the reactor internals and core supports includes 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 EPU; therefore, these continue to be acceptable.

NMP2 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 irradiation-assisted stress corrosion cracking (IASCC), in welds and in the adjacent base material. NMP2 belongs to the BWR Vessel and Internals Project (BWRVIP) organization and implementation of the procedurally controlled program is consistent with the BWRVIP issued documents. The inspection strategies recommended by the BWRVIP 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 (IEBs), 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:

1. Core spray piping
2. Core spray spargers
3. Core shroud and core shroud support
4. Jet pumps and associated components
5. Top guide
6. Lower plenum
7. Vessel Inner Diameter attachment welds
8. Instrumentation penetrations
9. Steam dryer drain channel welds
10. FW spargers

Inspected components are considered as being potentially susceptible to IASCC if the end-of-life fluence is in excess of 5 x 10^{20} n/cm^2. Three (3) components have been identified as being potentially susceptible to IASCC, based upon the projected 54 EFPY fluence: (1) Top Guide, 4.04 x 10^{22} n/cm^2; (2) Shroud, 5.1 x 10^{21} n/cm^2; and (3) Core Plate, 6.61 x 10^{20} n/cm^2. The BWRVIP inspection recommendations which provide the scope, sample size, inspection method, and frequency of examination used to manage the effects of IASCC are as follows:

- Top Guide (BWRVIP-26-A)
- Shroud (BWRVIP-76-A)
- Core Plate (BWRVIP-25-A)

Continued implementation of the current procedure program assures the prompt identification of any degradation of reactor vessel internal components experienced during EPU operating conditions. To mitigate the potential for IGSCC and IASCC, NMP2 utilizes hydrogen water chemistry and noble metals applications. Reactor vessel water chemistry conditions are also maintained consistent with the EPRI and established industry guidelines.

The service life of most equipment is not affected by EPU. The peak fluence increase experienced by the reactor internals does not represent a significant increase in the potential for IASCC. The current inspection strategy for the reactor internal components is expected to be adequate to manage any potential effects of EPU.

Conclusion

NMPNS has reviewed the effects of the proposed EPU on the susceptibility of reactor internal and core support materials to known degradation mechanisms and concludes that the appropriate degradation management programs to address the effects of changes in operating temperature and neutron fluence on the integrity of reactor internal and core support materials have been identified. NMPNS further concludes that the reactor internal and core support materials will continue to be acceptable and will continue to meet the requirements of GDC1 and 10 CFR 50.55a with respect to material specifications, welding controls, and inspection following implementation of the proposed EPU. Therefore, NMPNS finds the proposed EPU acceptable with respect to reactor internal and core support materials.

2.1.4 Reactor Coolant Pressure Boundary Materials

Regulatory Evaluation

The Reactor Coolant Pressure Boundary (RCPB) defines the boundary of systems and components containing the high-pressure fluids produced in the reactor. The NMPNS review of RCPB materials covered their specifications, compatibility with the reactor coolant, susceptibility to degradation, and degradation management programs. The NRC's acceptance criteria for RCPB materials are based on (1) 10 CFR 50.55a and GDC1, 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) GDC4, 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) GDC14, 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) GDC31, insofar as it requires that the RCPB be designed with margin sufficient to assure that, under specified conditions, it will behave in a nonbrittle manner and the probability of a rapidly propagating fracture is minimized; and (5) 10 CFR 50, Appendix G, which specifies fracture toughness requirements for ferritic components of the RCPB.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of Constant Pressure Power Uprates. Sections 3.2 and 10.7 of the CLTR address the effect of Constant Pressure Power Uprate on Reactor Coolant Pressure Boundary Materials. The results of this evaluation are described below.

The NMP2 ISI program for all ASME Code Class 1 and 2 reactor coolant pressure boundary piping is in accordance with an NRC staff approved alternate risk-informed inspection program utilizing the NRC approved EPRI methodology, Technical Report TR-112657, Rev. B-A (Reference 10). In addition to the ASME Code, Section XI and the alternate Risk-Informed programs, NMP2 implements an augmented IGSCC inspection program in accordance with Generic Letter (GL) 88-01 (Reference 11), NUREG-0313, and as modified by BWRVIP-75 (Reference 12) for IGSCC Category D weld examination frequency using normal water chemistry. NMP2 implements ASME Section XI, Appendix VIII for the performance demonstration for ultrasonic examination systems administrated through the EPRI PDI program. Appendix VIII provided the requirements for the performance demonstration for ultrasonic examination procedures, equipment, and personnel to detect and size flaws. All of the above programs have been credited as an aging management program during the NMP2 License Renewal process.

Continued implementation of the current program assures the prompt identification of any degradation of RCPB components experienced during EPU operating conditions.

The augmented inspection program is designed to detect potential degradation from IGSCC. For IGSCC to occur, three conditions must be present: (1) a susceptible material (for a list of materials in the RCPB, see Table 5.2-5 of the NMP2 USAR); (2) the presence of residual stress (such as from welding); and (3) aggressive environment. Operation at EPU conditions results in an insignificant change to temperature and flow conditions for portions of the RCPB piping and does not impact the other susceptibility factors associated with IGSCC (this is consistent with the conclusions presented in Section 3.6.1 of ELTR2). EPU does increase the fluence for the reactor vessel beltline region; however the EPU fluence for the reactor vessel nozzle safe-end welds and piping remains well below the 5.0×10^{20} n/cm^2 fluence threshold for IASCC concerns for stainless steel.

The NMP2 augmented inspection program frequency is based on BWRVIP-75 (Reference 12) normal water chemistry. While NMP2 has implemented hydrogen water chemistry with noble metals, the augmented program has not implemented the HWC frequency as allowed by BWRVIP-75A (Reference 13) at this time. NMP2 is planning to implement BWRVIP-75A HWC frequencies in the near future when NRC issues a final safety evaluation for BWRVIP-62 (Reference 14). Table 2.1-4 provides a summary of the critical BWRVIP-75A category D and E dissimilar metal welds for the reactor vessel nozzle to safe-end locations.

Conclusion

NMPNS has reviewed the effects of the proposed EPU on the susceptibility of RCPB materials to known degradation mechanisms and concludes that the appropriate degradation management programs have been identified to address the effects of changes in system operating temperature on the integrity of RCPB materials. NMPNS further concludes that the RCPB materials will continue to be acceptable following implementation of the proposed EPU and will continue to meet the requirements of GDC1, GDC4, GDC14, GDC31, 10 CFR 50, Appendix G, and 10 CFR 50.55a. Therefore, NMPNS finds the proposed EPU 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 NMPNS review covered protective coating systems used inside the containment for their suitability for and stability under design basis loss-of-coolant accident (DBLOCA) conditions considering radiation and chemical effects. The NRC's acceptance criteria for protective coating systems are based on (1) 10 CFR 50, Appendix B, which states quality assurance requirements for the design, fabrication, and construction of safety-related SSCs and (2) RG 1.54 (Reference 15), for guidance on application and performance monitoring of coatings in nuclear power plants.

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 (DBLOCA) conditions, considering radiation and chemical effects at 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.

The Service Level 1 protective coatings used at NMP2 were qualified per ANSI N101.2-1972 to a radiation level of 1×10^9 Rads and a temperature of 340 degrees. Comparisons of the test results to the bounding conditions for a design basis accident (DBA) under EPU conditions are shown below.

EQ Zone	Peak Temperature (°F)	Peak Pressure (psia)	Irradiation (Rads)
General Containment: short term, DBA	285	52.9	n/a
Bounding Environmental Qualification (EQ) Zones – 100 day: PC175101, PC19912, PC215121	<285	<52.9	9.32×10^8

For NMP2, monitoring and maintaining protective coatings inside the primary containment are performed to ANSI N101.2-1972, ANSI N101.4-1972, ANSI N5.12-1974, RG 1.54, June 1973 (Reference 15), and ANSI/American Society of Mechanical Engineers (ASME) NQA-1-1983. Coating condition assessments of Service Level 1 coatings inside the primary containment (drywell) are also conducted every refueling outage.

The NMP2 Protective Coating Monitoring and Maintenance Program is an existing program that is described in the NMP2 response to GL 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," (Reference 16). The program was developed in accordance with ANSI N101.4-1972 referenced in RG 1.54, June 1973, along with ANSI/ASME NQA-1-1983. The NMP program is a "comparable programs" as described in NUREG-1801 (Reference 17), Chapter XI, Program XI.S8, Protective Coating Monitoring and Maintenance Program.

The program applies to Service Level 1 protective coatings inside the primary containment. The NMP2 suppression pool (wetwell) is not included because it is stainless steel lined reinforced concrete and does not have Service Level 1 coatings. Coating conditions monitored by this program include blistering, cracking, peeling, loose rust, and physical/mechanical damage. When localized degradation of a coating is identified, the affected area is evaluated and scheduled for repair, replacement, or removal, as needed. The condition assessments and resulting repair, replacement, or removal activities ensure that the amount of coatings subject to detachment from the substrate during a LOCA is minimized to ensure post-accident operability of the ECCS suction strainers.

Conclusion

NMPNS has reviewed the effects of the proposed EPU on protective coating systems and concludes that the effect of changes in conditions following a DBLOCA and their effects on the protective coatings have been appropriately addressed. NMPNS further concludes 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 50 Appendix B. Therefore, NMPNS finds the proposed EPU 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.

NMPNS has reviewed the effects of the proposed EPU on FAC and the adequacy of the NMP2's FAC program to predict the rate of loss so that repair or replacement of damaged components could be made before they reach critical thickness. The NMP2 FAC program is based on NUREG-1344, GL 89-08, and the guidelines in EPRI Report NSAC-202L-R3. 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.

Technical Evaluation

The FAC program at Nine Mile Point Unit 2 (NMP2) is based on:

- NRC I&E Bulletin 87-01, "Thinning Pipe Walls in Nuclear Power Plants"
- GL 89-08, "Erosion/Corrosion-Induced Pipe Wall Thinning" (Reference 18)
- EPRI NSAC-202L-R3, *Recommendations for an Effective Flow Accelerated Corrosion Program* (Revision 3, May 2006).

The NMP2 FAC program monitors all FAC susceptible piping—both small bore and large bore—to ensure the structural integrity and functionality are maintained. FAC susceptible piping can be divided into two categories: lines that meet the requirements to be modeled using EPRI CHECWORKS™ SFA, and those that do not. For those that meet the requirements, NMP2 uses CHECWORKS™ SFA, in conjunction with volumetric examination to predict FAC wear rates and remaining service life for components in single phase and two phase systems.

The FAC susceptible lines that do not meet the minimum requirements for modeling and analysis by CHECWORKS™ SFA are referred to as "Susceptible Non-Modeled" (SNM). This group is comprised of lines with unknown or widely varying operating conditions that prevent the development of accurate predictive models. It includes bypass lines, recirculation lines, vent lines, high level dumps, and socket welded piping. Small bore piping and piping susceptible to wall thinning mechanisms other than FAC are also included in this group. Selection of this piping for inspection is typically the result of industry operating experience, NMP2 experience, or engineering judgment.

One of the most import aspects of the NMP2 FAC program is the proper selection of locations for FAC inspection and replacement of degraded piping. This is accomplished using the following:

- CHECWORKS™ SFA predictive wear analysis
- Susceptibility ranking of SNM piping
- Operating Experience (OE)
- NMP2-specific experience
- Trending of historical inspection data
- Sound engineering judgment combining all of the above

The proposed EPU will impact the following aspects of the NMP2 FAC program.

- FAC system susceptibility evaluation - this may include the addition of new lines in the FAC program based on changes in operating conditions.
- Wear rates - changes in operating conditions will result in some components wearing at an accelerated rate, while others will wear at a slower rate.
- Selection of component inspection and replacement locations and subsequent evaluation of inspection results (trending) - there could be a short term increase in the number of inspections performed.

These are evaluated as follows:

FAC System Susceptibility Evaluation

NMPNS performed a system susceptibility screening based on the revised EPU heat balance, and determined that no additional lines were required to be added to the FAC program.

CHECWORKS™ SFA Model Update to Reflect EPU Conditions

The proposed EPU will result in changes to several variables that may directly influence FAC wear rates. The variables include operating temperature, steam quality, velocity and oxygen content. To account for these changes, NMP2 updated the affected parameters in the CHECWORKS™ SFA model based on the EPU heat balance diagram.

Table 2.1-5 contains a listing of the CHECWORKS™ SFA run definitions (i.e., compilations of lines with similar operating conditions, water chemistry and usage for analysis). A comparison of pre- and post-EPU wear rate predictions identified changes ranging from a decrease of 17.7% to an increase of 35.8%. Of the 24 run definitions listed, nine had an increase in the predicted wear rate while the remaining fifteen exhibited a decrease or no change. Based on a review of the changes in operating conditions, NMPNS found the resulting predicted wear rates to be consistent with EPU conditions.

Selection of Inspection and Replacement Locations

The current approach to select locations for FAC inspection does not change as a result of EPU. However, there could be an increase in the number of FAC inspections performed on both CHECWORKS™ SFA-modeled and SNM piping over the next several refueling outages to ensure the impact of power uprate is understood. Inspections will be selected considering the changes in predicted wear rates, actual component thicknesses, operating time since last examination and design margin. This approach will ensure that FAC susceptible components are inspected or replaced prior to reaching code minimum wall thickness. Based on the evaluation of the impact of EPU, no immediate component replacements were required prior to EPU implementation (RFO-13).

This data will be used to further calibrate the CHECWORKS™ SFA model and susceptibility rankings for SNM piping.

Benchmarking CHECWORKS™ SFA Predicted Component Thickness

Table 2.1-6 presents a comparison of CHECWORKS™-predicted thicknesses to measured thicknesses for a sample component from each of the 24 run definitions. The selection process includes components with the highest predicted wear rates prior to EPU. The measured thicknesses were determined by ultrasonic testing NDE performed between RFO-9 and RFO-11, the three most recent refueling outages.

The table shows that in all cases the measured thickness from inspection was greater than the predicted thickness, indicating that CHECWORKS™ SFA predictions were conservative.

<u>**Conclusion**</u>

NMPNS has reviewed the effect of the proposed EPU on the FAC analysis for the plant and concludes that the changes in the plant operating conditions on the FAC analysis have been adequately addressed. NMPNS further concludes 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, NMPNS finds the proposed EPU acceptable with respect to FAC.

2.1.7 Reactor Water Cleanup System

<u>**Regulatory Evaluation**</u>

The reactor water cleanup system (WCS) 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 WCS comprise the RCPB. The NMPNS review of the WCS included component design parameters for flow, temperature, pressure, heat removal capability, and impurity removal capability; and the instrumentation and process controls for proper system operation and isolation. The review consisted of evaluating the adequacy of the NMP2 Technical Specifications in these areas under the proposed EPU conditions. The NRC's acceptance criteria for the WCS are based on (1) GDC14, 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) GDC60, insofar as it requires that the plant design include means to control the release of radioactive effluents; and (3) GDC61, insofar as it requires that systems that contain radioactivity be designed with appropriate confinement.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of Constant Pressure Power Uprates. Section 3.11 of the CLTR addresses the effect of Constant Pressure Power Uprate on the reactor water cleanup system. The results of this evaluation are described below.

WCS operation at EPU RTP level slightly decreases the temperature within the WCS. 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. Table 2.1-7 provides the magnitude of changes in WCS operating conditions due to a decrease in inlet temperature and increase in operating pressure.

WCS flow is usually selected to be in the range of 0.8% to 1.0% of Feedwater System (FWS) flow based on operational history. The existing WCS flow (and that analyzed for EPU) is within the BWR operational history range and has additional margin. Furthermore, the EPU review included evaluation of water chemistry, heat exchanger performance, pump performance, flow control valve capability and filter/demineralizer performance. All aspects of performance were found to be within the design of the WCS at the analyzed flow at EPU conditions. The WCS analysis concludes that:

1. There is negligible heat load effect.
2. A small increase in filter / demineralizer backwash frequency occurs, but this is within the capacity of the Radwaste system.
3. The slight changes in operating system conditions result from a decrease in inlet temperature and increase in FWS operating pressure.
4. The WCS filter / demineralizer control valve operates in a slightly more open position to compensate for the increased FWS pressure.
5. No changes to instrumentation are required; and setpoint changes are not expected due to the negligible system process parameter changes.

Previous operating experience has shown that the FWS iron input to the reactor increases for EPU as a result of the increased FWS flow. This input increases the calculated reactor water iron concentration from 21 ppb to 24 ppb, as shown in Table 2.1-8. However, this change is considered insignificant, and does not affect WCS.

The EPU WCS flow rate is not increased to match the increase in FWS flow; therefore, the sulfate and chloride concentrations will increase above the CLTP levels. The current average levels of chlorides are 0.70 ppb. The expected reactor water chloride level for EPU, considering the FWS flow increase, is 0.82 ppb for the nominal case. The current average levels of sulfates are 2.34 ppb. The expected reactor water sulfate level for EPU, considering the FWS flow increase, is 2.75 ppb. These anticipated concentration levels remain below the administrative limits of 5.0 ppb for chlorides and 5.0 ppb for sulfates.

The effects of EPU on the WCS functional capability have been reviewed, and the system can perform adequately at EPU RTP with the original WCS flow. As can be seen from Table 2.1-6, the changes in WCS operating conditions are very small. This WCS flow results in a slight increase in the calculated reactor water conductivity (from 0.102 μS/cm to 0.110 μS/cm) because of the increase in FW system flow. The current reactor water conductivity limits are unchanged for EPU and the actual conductivity remains within these limits.

Table 2.1-7 provides a summary of the chemistry values. As this table indicates, the NMP2 WCS has sufficient capacity to respond to the EPU conditions and maintain the chemistry parameters within administrative limits.

The increase in FW system line pressure has a slight effect on the system operating conditions. The effect of this increase is included in the containment isolation assessment in Sections 2.2.4 and 2.6.1.3.

Conclusion

NMPNS has reviewed the effects of the proposed EPU on the WCS and concludes that the changes in impurity levels and pressure and their effects on the WCS have been adequately addressed. NMPNS further concludes that the WCS will continue to be acceptable following implementation of the proposed EPU and will continue to meet the requirements of GDC14, GDC60, and GDC61. Therefore, NMPNS finds the proposed EPU acceptable with respect to the WCS.

Table 2.1-1 Upper Shelf Energy

60 Year Life (54 EFPY)

Material	Heat Number	Initial Transverse USE (ft-lb)	%Cu	54 EFPY 1/4T Fluence (n/cm^2)	% Decrease USE[(1)]	54 EFPY Transverse USE[(2)] (ft-lb)
PLATES:						
Lower-Intermediate Shell						
	C3065-1	94	0.06	1.12E+18	9.5	85
	C3121-2	71	0.09	1.12E+18	11	63
	C3147-1	70	0.11	1.12E+18	12.5	61
Lower Shell						
	C3065-2	83	0.06	1.09E+18	9.5	75
	C3066-2	80	0.07	1.09E+18	10	72
	C3147-2	86	0.11	1.09E+18	12.5	75
WELDS: [(3)]						
Circumferential						
	4P7216(S)/0751	89	0.045	1.09E+18	11.5	78
	4P7216(T)/0751	98	0.035	1.09E+18	11	87
	4P7465(S)/0751	102	0.02	1.09E+18	10	91
	4P7465(T)/0751	110	0.02	1.09E+18	10	99
Axial						
Lower Shell	5P6214B(S)/0331	88	0.02	1.09E+18	10	79
Lower Shell	5P6214B(T)/0331	96	0.014	1.09E+18	10	86
Lower-Intermediate Shell	5P5657(S)/0931	85	0.07	1.12E+18	13	73
Lower-Intermediate Shell	5P5657(T)/0931	88	0.04	1.12E+18	11	78
NOZZLES: [(4)]						
N6: LPCI	Q2QL3W	85	0.065	3.68E+17	8	78
N12 Water Level Instrumentation [(5)]	717456	70	0.11	2.52E+17	8	64
INTEGRATED SURVEILLANCE PROGRAM: [(6)]						
Plate	C2761-2	127.2	0.10	1.12E+18	11.5	112
Weld	5P6214B	90.9	0.027	1.12E+18	10	81

Notes:
(1) % Decrease USE is obtained from RG1.99 Figure 2 (Reference 50).
(2) 54 EFPY Transverse USE = Initial Transverse USE * [1 - (% Decrease USE /100)].
(3) This evaluation includes both Single (S) and Tandem (T) Wire materials.
(4) Chemistries for the nozzle forgings were obtained from CMTR data.
(5) *The N12 nozzle is classified as a partial penetration in Shell Ring #2. Because the forging is <2.5" thick, fracture toughness* is not required (ASME Code, Appendix G, Section G2223(c)). Therefore, this nozzle is evaluated considering the Shell Ring #2 material properties. Because the fluence is lower at the nozzle location, Shell Ring #2 remains the limiting component.
(6) Representative materials as defined by the Integrated Surveillance Program (BWRVIP-135, Revision 1; Reference 49).

Table 2.1-2 Adjusted Reference Temperatures

60-Year License (54 EFPY)

Lower Shell, Shell 1 to Shell 2 Girth Weld and Lower Shell Axial Welds

Thickness in inches = 6.1875 | 54 EFPY Peak I.D. fluence = 1.58E+18 n/cm^2
54 EFPY Peak 1/4 T fluence = 1.09E+18 n/cm^2

Lower-Intermediate Shell and Axial Welds

Thickness in inches = 6.1875 | 54 EFPY Peak I.D. fluence = 1.62E+18 n/cm^2
54 EFPY Peak 1/4 T fluence = 1.12E+18 n/cm^2

N6 LPCI Nozzle

Thickness in inches = 6.1875 | 54 EFPY Peak I.D. fluence = 5.34E+17 n/cm^2
54 EFPY Peak 1/4 T fluence = 3.68E+17 n/cm^2

N12 Water Level Instrumentation Nozzle

Thickness in inches = 6.1875 | 54 EFPY Peak I.D. fluence = 3.65E+17 n/cm^2
54 EFPY Peak 1/4 T fluence = 2.52E+17 n/cm^2

COMPONENT	HEAT	%Cu	%Ni	CF	Adjusted CF	Initial RT_{NDT} °F	1/4 T Fluence n/cm^2	54 EFPY ΔRT_{NDT} °F	σ_I (Note 8)	σ_Δ	Margin °F	54 EFPY Shift °F	54 EFPY ART °F
PLATES:													
Lower-Intermediate Shell													
	C3065-1	0.06	0.63	37	-	-10	1.12E+18	16.3	0	8	16.3	32.5	23
	C3121-2	0.09	0.65	58	-	0	1.12E+18	25.5	0	13	25.5	51.0	51
	C3147-1	0.11	0.63	74.5	-	0	1.12E+18	32.7	0	16	32.7	65.5	66
Lower Shell													
	C3065-2	0.06	0.63	37	-	10	1.09E+18	16.1	0	8	16.1	32.1	42
	C3066-2	0.07	0.64	44	-	-20	1.09E+18	19.1	0	10	19.1	38.2	18
	C3147-2	0.11	0.63	74.5	-	0	1.09E+18	32.4	0	16	32.4	64.7	65
WELDS: (1)													
Circumferential													
	4P7216(S)/0751	0.045	0.80	61	-	-50	1.09E+18	26.5	0	13	26.5	53.0	3
	4P7216(T)/0751	0.035	0.82	47.5	-	-80	1.09E+18	20.6	0	10	20.6	41.3	-38
	4P7465(S)/0751	0.02	0.82	27	-	-60	1.09E+18	11.7	0	6	11.7	23.5	-36
	4P7465(T)/0751	0.02	0.80	27	-	-60	1.09E+18	11.7	0	6	11.7	23.5	-36
Axial													
Lower Shell	5P6214B(S)/0331	0.02	0.82	27	-	-50	1.09E+18	11.7	0	6	11.7	23.5	-26
Lower Shell	5P6214B(T)/0331	0.014	0.70	22.8	-	-40	1.09E+18	9.9	0	5	9.9	19.8	-20
Lower-Intermediate Shell	5P5657(S)/0931	0.07	0.71	95	-	-60	1.12E+18	41.8	0	21	41.8	83.5	24
Lower-Intermediate Shell	5P5657(T)/0931	0.04	0.89	54	-	-60	1.12E+18	23.7	0	12	23.7	47.5	-12
NOZZLES:													
N6 LPCI (2)	Q2QL3W	0.07	0.86	44	-	-20	3.68E+17	10.9	0	5	10.9	21.8	2
N12 Water Level Instrumentation (2,3)	717456	0.11	0.65	75	-	0	2.52E+17	14.8	0	7	14.8	29.6	30
INTEGRATED SURVEILLANCE PROGRAM: (4)													
Plate (5)	C2761-2	0.10	0.54	65	-	10	1.12E+18	29	0	14	28.6	57.1	67
Weld (6,7)	5P6214B	0.027	0.94	36.8	53.7	-40	1.12E+18	24	0	12	23.6	47.2	7

Notes:
(1) This evaluation includes both Single (S) and Tandem (T) Wire materials.
(2) Chemistries and Initial RT_{NDT} for the nozzle forgings were obtained from CMTR data.
(3) The N12 nozzle is classified as a partial penetration in Shell Ring #2. Because the forging is <2.5" thick, fracture toughness is not required (ASME Code, Appendix G, Section G2223(c)). Therefore, this nozzle is evaluated considering the Shell Ring #2 material properties.
(4) Representative materials as defined by the Integrated Surveillance Program, BWRVIP-135, Revision 1 (Reference 49).
(5) The ISP plate material is not the same heat number as the target plate as defined in the ISP (BWRVIP-135, Revision 1). Therefore, the CF from RG1.99, Position 1.1 (Reference 50) is used to determine the ART. This information is not required per BWRVIP-135, Revision 1 (Reference 49), to dictate the ART used for the NMP2 PT curves, and is provided for information only. Plate C3147-1 remains the limiting beltline material for the purpose of developing PT curves.
(6) The ISP weld material is not the same heat number as the target weld (5P5657) as defined in the ISP (Reference 49). However, the ISP weld material is the same heat as another beltline weld. Therefore, the surveillance data is considered and the CF is adjusted as defined in RG1.99 Position 2.1 (Reference 50).
(7) The Adjusted CF was conservatively calculated using the limiting parameters for this heat because there are two (2) sets of data provided by the ISP (from Perry and SSP materials that originated from Grand Gulf). CFAdjusted = (27/20) * 39.75 = 53.7
(8) Previous submittals included a conservative σ_I = 14.5°F. BWROG NRC-approved methodology (Reference 46) has been applied, allowing reduction such that σ_I = 0°F.

Table 2.1-3 54 EFPY Effects of Irradiation on RPV Circumferential Weld Properties

Group	CB&I * 64 EFPY	NMP2 54 EFPY
Cu%	0.10	0.045
Ni%	0.99	0.82
CF (See Note 1)	134.9	61
Fluence at clad/weld interface (10^{19} n/cm^2)	1.02	0.158
ΔRT_{NDT} w/o margin (°F) (See Note 4)	135.6	31.4
$RT_{NDT(U)}$ (°F)	-65	-50
Mean RT_{NDT} (°F)	70.6	-18.6
P (F/E) NRC (See Note 2)	1.78E-05	(Note 3)

* This column represents the limits as defined in BWRVIP-74 (Reference 48).

Notes:

[1] The value of 109.5 originally presented in BWRVIP-05 (Reference 47) 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 fact that the NMP2 values at the end of license are less than the 64 EFPY value provided by the NRC leads to the conclusion that the NMP2 RPV conditional failure probability is bounded by the NRC analysis, consistent with the requirements of Reference 48.

[4] For the NMP2-specific calculations, Monte Carlo methods were previously used to calculate the Mean RT_{NDT}. RG1.99 Rev 2 (Reference 50) methods were used to calculate the Mean RT_{NDT} for EPU conditions presented in this table.

Table 2.1-4 Summary of Critical BWRVIP-75A Category D & E Dissimilar Metal Welds

Weld ID and Description	Plant System in Which Weld is Located	BWRVIP-75A Weld Category
2RPV-KB-32 (nozzle to safe end)	N16 High Pressure Core Spray	D
2RPV-KC-32 (SE to SE ext)	N16 High Pressure Core Spray	E
2RPV-KB-01 (nozzle to safe end)	N1A Recirc	D
2RPV-KB-02 (nozzle to safe end)	N1B Recirc	D
2RPV-KB-03 (nozzle to safe end)	N2A Recirc	D
2RPV-KB-04 (nozzle to safe end)	N2B Recirc	D
2RPV-KB-05 (nozzle to safe end)	N2C Recirc	D
2RPV-KB-06 (nozzle to safe end)	N2D Recirc	D
2RPV-KB-07 (nozzle to safe end)	N2E Recirc	D
2RPV-KB-08 (nozzle to safe end)	N2F Recirc	D
2RPV-KB-09 (nozzle to safe end)	N2G Recirc	D
2RPV-KB-10 (nozzle to safe end)	N2H Recirc	D
2RPV-KB-11 (nozzle to safe end)	N2J Recirc	D
2RPV-KB-12 (nozzle to safe end)	N2K Recirc	D
2RPV-KB-17 (nozzle to safe end)	N4A Feedwater	D
2RPV-KB-18 (nozzle to safe end)	N4B Feedwater	D

Weld ID and Description	Plant System in Which Weld is Located	BWRVIP-75A Weld Category
2RPV-KB-19 (nozzle to safe end)	N4C Feedwater	D
2RPV-KB-20 (nozzle to safe end)	N4D Feedwater	D
2RPV-KB-21 (nozzle to safe end)	N4E Feedwater	E
2RPV-KB-22 (nozzle to safe end)	N4F Feedwater	D
2RPV-KB-23 (nozzle to safe end)	N5 Core Spray	D
2RPV-KB-24 (nozzle to safe end)	N6A Core Spray	D
2RPV-KB-25 (nozzle to safe end)	N6B Core Spray	D
2RPV-KB-26 (nozzle to safe end)	N6C Core Spray	D
2RPV-KC-23 (SE to SE ext)	N5 Core Spray	D
2RPV-KC-24 (SE to SE ext)	N6A Core Spray	D
2RPV-KC-25 (SE to SE ext)	N6B Core Spray	D
2RPV-KC-26 (SE to SE ext)	N6C Core Spray	D
2RPV-KB-29 (nozzle to safe end)	N9A JPI	D
2RPV-KB-30 (nozzle to safe end)	N9B JPI	D

Table 2.1-5 Comparison of Key Parameters Influencing FAC Wear Rate, NMP2

Pre-EPU vs. Post-EPU

CHECWORKS™ Wear Rate Analysis Run Definition Name	Temperature (deg F)		% Change	Velocity (ft/sec)		% Change	Oxygen (ppb)		% Change	Quality		% Change	Percent Change in Predicted Wear Rate Due to Power Uprate
	Current (105%)	EPU (120%)		Current (105%)	EPU (120%)		Current (105%)	EPU (120%)		Current (105%)	EPU (120%)		
Cond Htr 2 To Htr 3	220.6	223.5	1.3	10.925	12.573	15.1	72.038	70.422	-2.2	0.000	0.000	No Change	11.9
Cond Htr 3 To Htr 4	288.3	294.8	2.3	22.264	25.681	15.3	72.038	70.422	-2.2	0.000	0.000	No Change	11.8
Cond Htr 4 To HD Tee	322.9	330.5	2.4	11.516	13.298	15.5	72.038	70.422	-2.2	0.000	0.000	No Change	1.0
Cond HD Tee To Htr 5	323.9	332.0	2.5	13.601	15.754	15.8	53.374	53.375	No Change	0.000	0.000	No Change	-1.0
Cond Htr 5 To Header	365.2	378.2	3.6	13.969	16.247	16.3	53.374	53.374	No Change	0.000	0.000	No Change	-8.0
Cond RFP Suction	365.2	378.2	3.6	14.537	16.907	16.3	53.374	53.374	No Change	0.000	0.000	No Change	-8.0
FW Pmp To Balance Ln	368.2	381.1	3.5	17.947	20.874	16.3	53.374	53.374	No Change	0.000	0.000	No Change	-8.9
FW Balance Line - 1	368.2	381.1	3.5	6.034	7.011	16.2	53.374	53.374	No Change	0.000	0.000	No Change	-4.8

CHECWORKS™ Wear Rate Analysis Run Definition Name	Temperature (deg F)		% Change	Velocity (ft/sec)		% Change	Oxygen (ppb)		% Change	Quality		% Change	Percent Change in Predicted Wear Rate Due to Power Uprate
	Current (105%)	EPU (120%)		Current (105%)	EPU (120%)		Current (105%)	EPU (120%)		Current (105%)	EPU (120%)		
FW Balance Line - 2	368.2	381.1	3.5	6.034	7.011	16.2	53.374	53.374	No Change	0.000	0.000	No Change	-4.8
FW Balance Ln - Htr 6	368.2	381.1	3.5	17.362	20.194	16.3	53.374	53.374	No Change	0.000	0.000	No Change	-8.9
FW Htr 6 To Reactor	425.4	440.5	3.5	18.786	21.927	16.7	53.374	53.374	No Change	0.000	0.000	No Change	13.7
Htr 6 Drain To Htr 5	384.1	393.5	2.4	4.913	5.782	17.7	1060.534	1356.321	27.9	0.000	0.000	No Change	19.6
Htr 5 Drn To 4 USCV	337.0	344.2	2.1	6.217	7.352	18.3	488.922	623.817	27.6	0.000	0.000	No Change	0.0
Htr 4 Drn To CNM Sys	327.5	335.5	2.4	11.276	13.138	16.5	10.721	13.219	23.3	0.000	0.000	No Change	-15.2
Htr 3 Drn To Drn Clr	233.8	235.5	0.7	3.153	3.868	22.7	124.011	137.357	10.8	0.000	0.000	No Change	10.8
Htr 2 Drn To Drn Clr	222.8	228.5	2.6	3.259	3.755	15.2	33.279	34.599	4.0	0.000	0.000	No Change	4.8
Moist Sep Drns To Tk	372.1	385.1	3.5	0.799	0.802	0.4	962.294	943.392	-2.0	0.000	0.000	No Change	-16.8
Ms Drn Tk4 To Htr 4	372.1	385.1	3.5	3.263	3.285	0.7	4.400	5.431	23.4	0.000	0.000	No Change	-17.7
Rhtr Drains To Tank	544.4	541.8	-0.5	1.682	1.605	-4.6	664.500	633.827	-4.6	0.000	0.000	No	0.0

CHECWORKS™Wear Rate Analysis Run Definition Name	Temperature (deg F)		% Change	Velocity (ft/sec)		% Change	Oxygen (ppb)		% Change	Quality		% Change	Percent Change in Predicted Wear Rate Due to Power Uprate
	Current (105%)	EPU (120%)		Current (105%)	EPU (120%)		Current (105%)	EPU (120%)		Current (105%)	EPU (120%)		
												Change	
Rhtr Drn Tk To Htr 6	544.4	541.7	-0.5	3.782	3.641	-3.7	149.981	137.507	-8.3	0.000	0.000	No Change	4.2
WCS Pump Suctn/Disch	535.8	535.8	No Change	11.742	11.742	No Change	53.374	53.374	No Change	0.000	0.000	No Change	No Change
WCS Regen HX to NR HX	235.6	235.6	No Change	5.287	5.287	No Change	53.374	53.374	No Change	0.000	0.000	No Change	No Change
WCS Regen HX to FWS	440.2	440.2	No Change	6.458	6.458	No Change	53.374	53.374	No Change	0.000	0.000	No Change	No Change
Main Steam	547.3	543.5	-0.7	9.089	15.632	72.0	0.001	0.001	0.0	0.999	0.998	-0.1	35.8

Notes

1. All NMP2 Extraction Steam piping is constructed of FAC resistant material.
2. The NMP2 Cold Reheat piping is constructed of FAC resistant material.
3. All piping downstream of the level control valves on the Heater 5 drain line to Heater 4 is constructed of FAC resistant material.

Table 2.1-6 Sample of Components with Highest Predicted Wear Rates, NMP2 CHECWORKS™ SFA-Predicted Thickness vs. Measured Thickness

CHECWORKS™ Wear Rate Analysis Run Definition Name	Component Name	Component Type	Nominal Pipe Size (inches)	Nominal Thickness Tnom (inches)	Measured Thickness Tmeas (inches)	Predicted Thickness Tpred (inches)	Ratio of Measured Thickness to Predicted Thickness	Time of Inspection*
Cond Htr 2 To Htr 3	23-33EL03	90 Elbow	18	0.75	0.796	0.65	1.22	RFO-10
Cond Htr 3 To Htr 4	23-29EL02	90 Elbow	12	0.5	0.411	0.395	1.04	RFO-11
Cond Htr 4 To HD Tee	23-26EL01	90 Elbow	18	0.75	0.888	0.646	1.37	RFO-10
Cond HD Tee To Htr 5	23-26EL08	90 Elbow	20	0.812	0.785	0.606	1.30	RFO-10
Cond Htr 5 To Header	23-20EL08-DS	Pipe	20	0.812	0.782	0.664	1.18	RFO-11
Cond RFP Suction	23-24EL05	45 Elbow	24	0.969	0.895	0.855	1.05	RFO-10
FW Pmp To Balance Ln	47-01VA03-DS	Pipe	24	2.062	1.92	1.84	1.04	RFO-9
FW Balance Line - 1	47-04EL06	90 Elbow	24	2.062	2.198	1.742	1.26	RFO-9
FW Balance Line - 2	47-05EL03	90 Elbow	24	2.062	2.21	1.914	1.15	RFO-10
FW Balance Ln - Htr 6	47-01EL12	90 Elbow	20	1.75	1.781	1.376	1.29	RFO-10
FW Htr 6 To Reactor	47-14TR01	Tee	18	0.938	0.922	0.882	1.05	RFO-11
Htr 6 Drain To Htr 5	11-06EL01	90 Elbow	12	0.375	0.335	0.303	1.11	RFO-9
Htr 5 Drn To 4 USCV	10-35RD01	Reducer	12 x 8	0.375 / 0.322	0.357 / 0.318	0.309 / 0.239	1.15 / 1.33	RFO-11

CHECWORKS™ Wear Rate Analysis Run Definition Name	Component Name	Component Type	Nominal Pipe Size (inches)	Nominal Thickness Tnom (inches)	Measured Thickness Tmeas (inches)	Predicted Thickness Tpred (inches)	Ratio of Measured Thickness to Predicted Thickness	Time of Inspection*
Htr 4 Drn To CNM Sys	23-27EL12	90 Elbow	12	0.5	0.486	0.457	1.06	RFO-9
Htr 3 Drn To Drn Clr	10-24NZ01	Exit Nozzle	8	0.594	0.574	0.471	1.22	RFO-11
Htr 2 Drn To Drn Clr	10-20NZ01	Exit Nozzle	8	0.5	0.477	0.383	1.25	RFO-11
Moist Sep Drns To Tk	34-01EL03	90 Elbow	20	0.5	0.53	0.401	1.32	RFO-11
Ms Drn Tk4 To Htr 4	34-06TR01	Tee	12	0.375	0.381	0.217	1.76	RFO-11
Rhtr Drains To Tank	33-25EL06	90 Elbow	24	1.219	1.204	1.12	1.08	RFO-11
Rhtr Drn Tk To Htr 6	33-10EL05-DS	Pipe	12	0.688	0.61	0.577	1.06	RFO-11
WCS Pump Suctn/Disch	09-05EL01	90 Elbow	4	0.337	0.35	0.217	1.61	RFO-10
WCS Regen HX to NR HX	09-10EL01	90 Elbow	8	0.594	0.512	0.455	1.13	RFO-10
WCS Regen HX to FWS	09-14TR01	Tee	8	0.906	0.861	0.603	1.43	RFO-11
Main Steam	01-14EL03	90 Elbow	26	1.266	1.241	1.066	1.16	RFO-10

* Inspection timeframe: RFO-9 – Spring 2004; RFO-10 – Spring 2006; and RFO-11 – Spring 2008.

Table 2.1-7 Comparison of WCS System Operating Conditions

Parameter	Units	CLTP	EPU
WCS System	MWt	3467	3988
WCS Inlet Temperature	°F	534.0	533.8
WCS Inlet Pressure (RPV dome pressure, neglecting head)	psia	1035	1035
WCS Outlet Temperature	°F	437.5	437.3
WCS Outlet Pressure (at the feedwater line)	psig	1111.7	1143.0
Design WCS Flow	lbm/hr	340,000	340,000
Maximum WCS Flow	lbm/hr	340,000	340,000

Table 2.1-8 Comparisons of Chemistry Parameters for CLTP and EPU Cases

Item	Parameter	Units	CLTP Values		EPU Values	
			Average WCS Flow	WCS Design Flow	Average WCS Flow	WCS Design Flow
1	Average reactor water iron concentration	ppb	20.74	16.08	24.39	18.92
2	Average reactor water conductivity	μS/cm	0.102	0.091	0.110	0.098
3	Average Chloride concentration	ppb	0.70	0.54	0.82	0.64
4	Average Sulfate concentration	ppb	2.34	1.81	2.75	2.13

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. NMPNS conducted a review of pipe rupture analyses to ensure that SSCs important to safety are adequately protected from the effects of pipe ruptures. NMPNS conducted a review of (1) the implementation of criteria for defining pipe break and crack locations and configurations, (2) the implementation of criteria dealing with special features, such as augmented in-service inspection (ISI) programs or the use of special protective devices such as pipe-whip restraints, (3) pipe-whip dynamic analyses and results, including the jet thrust and impingement forcing functions and pipe-whip dynamic effects, and (4) the design adequacy of supports for SSCs provided to ensure that the intended design functions of the SSCs will not be impaired to an unacceptable level as a result of pipe-whip or jet impingement loadings. The NMPNS review focused on the effects that the proposed EPU may have on items (1) through (4) above. The NRC's acceptance criteria are based on GDC4, which requires SSCs important to safety to be designed to accommodate the dynamic effects of a postulated pipe rupture.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of Constant Pressure Power Uprates. Section 10.1 of the CLTR addresses the effect of Constant Pressure Power Uprate on High Energy Line Breaks. The results of this evaluation are described below.

High-energy piping systems inside and outside containment are listed in USAR Table 3.6A-73.

Inside containment, the high-energy piping systems affected by EPU are: Main Steam, Main Steam Drains, reactor core isolation cooling (RCIC) steam line, Feedwater, Main Steam vent lines, and Main Steam safety relief valve piping (between the main steam line and each SRV). Main Steam Drains and RCIC steam line flow rates, pressures and temperatures are unchanged from CLTP to EPU operating conditions. However, since these piping systems are connected directly to Main Steam piping, stresses in these systems were conservatively increased commensurate with the increase in Main Steam system piping stresses.

Outside containment the high-energy piping systems affected by EPU are Main Steam, Main steam drains, Feedwater, and WCS.

A review was performed of piping stresses that increased due to EPU and postulated pipe break locations. The review was conducted in accordance with the requirements of the original license basis methodology. No changes to the implementation of the existing criteria for defining pipe break and crack locations and configurations are being made for EPU. No new break or crack locations are required to be postulated as a result of the increased piping stresses associated with EPU.

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

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. As such, no plant-specific evaluation is required for steam line breaks. The results of the NMP2 evaluation of HELBs are provided in Table 2.2-1.

Steam Line Breaks

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Thus, the steam line HELB events in the NMP2 licensing basis were evaluated and confirmed to be consistent with the [[ ]] provided in the CLTR.

Liquid Line Breaks

Operation at EPU conditions requires an increase in the main steam (MS) and FW flows, which results in an increase in FW system pressures. This increase in pressure may lead to increased break flow rates for liquid line breaks. Only the mass and energy releases for HELBs in the WCS and FW systems may be affected by EPU and were re-evaluated at EPU conditions.

WCS Line Breaks

An evaluation of the mass and energy releases for WCS line breaks at CLTP and EPU conditions determined that the EPU mass releases remain bounded by the existing (CLTP) mass releases. The increase in system operating pressure due to EPU is more than offset by existing analytical conservatisms. No changes are being made to the automatic leak detection logic or to any leak detection system settings as a result of EPU.

Feedwater System Line Break

The CLTP mass and energy releases for FW line breaks are affected by changes in the FW system including increased FW flow rate and modifications to the FW pumps. The mass and energy releases for double-ended breaks and critical cracks in the FW lines were re-analyzed at EPU conditions. At EPU, the flashing portion of the mass release into the Main Steam Tunnel was found to increase by approximately 4%. However, the effects of a Feedwater System Line Break on Main Steam Tunnel peak pressures and temperatures are bounded by a Main Steam Line Break in the Main Steam Tunnel. For the portion of the smaller WCS piping attached to the Feedwater piping in the Main Steam Tunnel, mass and energy releases from breaks in the smaller WCS piping are bounded by the Feedwater break mass and energy releases.

Pipe Whip and Jet Impingement

Pipe whip and jet impingement loads resulting from high energy pipe breaks are a function of system pressure, temperature, and size, as well as proximity to relatively constant pressure sources connected to the line, and the effect of friction or line area restrictions between the break and the constant pressure source.

Inside containment, the only high-energy piping that experiences an increase in operating pressure due to EPU is in the Feedwater system. Outside containment, the only high-energy piping experiencing an increase in operating pressure due to EPU is in the FWS and WCS.

The potential impact of increased FWS and WCS operating pressures at the existing HELB break locations relative to the subsequent effects of pipe whip (targets) and jet impingement loads were evaluated. The resulting EPU pipe whip (targets) and jet impingement loads are bounded by the current licensing basis pipe whip and jet impingement loads.

Conclusion

NMPNS has reviewed the evaluations related to determinations of rupture locations and associated dynamic effects and addressed the effects of the proposed EPU on them. NMPNS further concludes that the SSCs important to safety will continue to meet the requirements of GDC4 following implementation of the proposed EPU. Therefore, NMPNS finds the proposed EPU acceptable with respect to the determination of rupture locations and dynamic effects associated with the postulated rupture of piping.

2.2.2 Pressure-Retaining Components and Component Supports

Regulatory Evaluation

NMPNS has reviewed the structural integrity of pressure-retaining components (and their supports) designed in accordance with the ASME Boiler and Pressure Vessel Code (B&PV Code), Section III, Division 1 (Reference 19), and GDCs 1, 2, 4, 14, and 15. NMPNS review focused on the effects of the proposed EPU on the design input parameters and the design-basis loads and load combinations for normal operating, upset, emergency, and faulted conditions. The NRC's acceptance criteria are based on (1) 10 CFR 50.55a and GDC1, 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) GDC2, 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) GDC4, 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) GDC14, 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) GDC15, 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.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of Constant Pressure Power Uprates. Section 3.4 and 3.5 of the CLTR addresses the effect of Constant Pressure Power Uprate on Flow-Induced Vibration and Piping Evaluation, respectively. The results of this evaluation are described below.

Flow-Induced Vibration (FIV)

The FIV evaluation addresses the influence of an increase in flow during EPU on RCPB piping and RCPB piping components. Safety related thermowells and probes are addressed in this evaluation.

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The methodology used to evaluate components for FIV for EPU is described in Section 3.4.1 of "Constant Pressure Power Uprate," NEDC-33004P-A, Revision 4, Class III, July 2003, which has been approved by NRC. This evaluation utilizes SAP4G07 to develop the dynamic finite element models of the MSS/FWS Thermowell and Sample Probe.

Three-dimensional beam elements are used to model the Thermowell & Sample Probe, Sockolet/Pipe weld. At each nodal point of the beam elements six degrees of freedom are assumed; three translations and three rotations. At the Sockolet/Pipe weld to the outer pipe wall, all six degrees of freedom are fixed. The masses of the Thermowell/Sample Probe and Sockolet/Pipe weld are lumped at the nodal points, which include both the structural mass and fluid mass displaced by the Thermowell/Sample Probe. These added masses are used to account for the effects of fluid on the Thermowell/Sample Probe vibration responses.

The non-dimensional quantity defined as Vr, termed reduced velocity ASME Pressure Vessel Code, 1998 Section III, Division 1, Appendices, N-1300 (Figure N-1323-1), is used to assess whether or not high FIV response is likely. To assess whether or not synchronization of vortex shedding frequency and tube natural frequency occurs, a damping parameter (Figure N-1323-1) is calculated. To calculate the structural response, a non-dimensional parameter, termed reduced damping (N-1324.1 Equation 76), was calculated:

For off resonance (non lock-in) condition, the structural response is ordinarily small and was calculated using the standard method (N-1324.2, first paragraph). For Thermowell and Sample Probe resonant structural response, ASME Pressure Vessel Code, 1998 Section III, Division 1, Appendices, N-1300 Table N-1324.2(a)-1 was used.

The total vibratory stress was calculated by using the SRSS (square root of the sum of the squares) of the oscillating lift and drag forces. These are unlikely to occur at the same time and hence it is conservative to use the SRSS method. The results of the analyses are presented below:

Item	Component Analyzed	Unit	EPU Value	Allowable
1	MSS Thermowell (TE102)	psi	22.5	7,690 psi for Carbon steel
2	MSS Thermowell (TE117)	psi	444.8	7,690 psi for Carbon steel
3	MSS Thermowell (TE2A)	psi	241.6	7,690 psi for Carbon steel
4	FWS Thermowell (TE108)	psi	24.4	7,690 psi for Carbon steel
5	FWS Thermowell (TE19A)	psi	1,531.7	7,690 psi for Carbon steel
6	FWS Thermowell (TW15A)	psi	44.3	7,690 psi for Carbon steel
7	FWS Sample Probe (SMPT1)	psi	1,235.7	10,880 psi for stainless steel

In conclusion, the safety-related Thermowells and probe in the MS and FW piping systems were evaluated and found to be adequate for the increased MS and FW flows as a result of EPU.

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 [[ ]]

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

RRS drive flow is not significantly increased (< 2%) during EPU operation. Consequently, the FIV levels of the RRS system components are expected to remain essentially the same. Because RRS flow rates for EPU are essentially the same as previously experienced and tested, no further evaluation or testing of the FIV levels of the RRS system piping, branch piping (e.g., attached Residual Heat Removal piping), or its suspension system is required.

The MS piping and the FW piping have increased flow rates and flow velocities in order to accommodate EPU. As a result, the MS and FW piping experience increased vibration levels, approximately proportional to the square of the flow velocities. The ASME Code (NB-3622.3) and nuclear regulatory guidelines require some 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, such as displacement probes, velocity sensors, and accelerometers. 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 NMP2 is confirmed to be consistent with the generic description provided in the CLTR because the nominal reactor dome pressure remains the same, the RRS maximum drive flow does not increase more than 5%, and FIV testing of the MS and FW piping system will be performed during EPU power ascension.

Nuclear Steam Supply System Piping, Components and Supports (Inside Containment)

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

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

- RRS (Conditions verified below)
- Control Rod Drive (CRD) System (No change in operating condition, unaffected by EPU)

The following piping system segments from the RPV to the normally closed containment isolation valve are also [[ ]] in the CLTR as unaffected by EPU:

- Residual Heat Removal (RHR) LPCI lines
- Core Spray (CS) injection lines
- Standby Liquid Control System (SLCS) injection line
- RPV Bottom Head Drain line

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[illegible]	[illegible]	[illegible]

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The piping systems described above are confirmed to be consistent with the [[ ]] provided in the CLTR because the temperature, pressure, flow rate, and mechanical loading changes are insignificant for the NMP2 EPU. Annulus Pressurization (AP) loads were determined to increase as discussed in Section 2.6.2. The changes in AP loads and resulting effects on piping inside and outside primary containment are discussed separately, at the end of this section.

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

Operation at EPU conditions increases stresses on piping and piping system components due to slightly higher operating temperatures, pressures and flow rates internal to the pipes. For all systems, the maximum stress levels and fatigue analysis results were reviewed based on specific increases in temperature, pressure, and flow rate (see Tables 2.2-2a and 2.2-2b). Simple, conservative scaling factors are used to evaluate the effects of EPU pressure, temperature, and flow changes. Pipe stress increase is proportional to pressure increase, temperature increase, and flow increase. EPU operation also increases the pipe support loads due to the above effects as well as increased fluid transient Turbine Stop Valve Closure (TSVC) loads that result from the increased steam flow rates (see Tables 2.2-2a and 2.2-2b).

The factors in Tables 2.2-2a and 2.2.-2b were derived using NRC approved GEH methodology in accordance with the CLTR. Application of these factors resulted in four main steam pipe supports that exceeded allowable stresses. A more detailed plant-specific steam hammer analysis was performed to determine the main steam system TSVC loads at EPU conditions. This analysis determined that the maximum piping support load increase inside containment due to TSV closure was 27%. Therefore, this factor was conservatively applied to the CLTP TSV closure support load to determine EPU loads. This evaluation resulted in all main steam pipe supports inside containment meeting the acceptance critieria.

The piping systems affected by EPU have been evaluated and found to meet the appropriate code criteria for EPU conditions, based on the design margins between actual stresses and code limits in the original design. All piping is below the code allowables of the plant code of record.

The pipe supports of 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 there is adequate design margin between the original design stresses and code limits of all supports inside containment and all but one condensate support outside containment (see below) to accommodate the load increase. Other than this support, the original design analyses have sufficient design margin to justify operation at the EPU conditions.

Main Steam and Associated Piping System Evaluation (Inside Containment)

The CLTR, Section 3.5.1, requires a [[ ]] 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 and fatigue. For NMP2, an increase in flow, pressure, temperature and mechanical loads was evaluated on a plant specific basis consistent with the methods specified in Appendix K of ELTR1. Plant-specific evaluations are required to demonstrate that the calculated stresses and fatigue usage factors are less than the code allowable limits in accordance with the requirements of the applicable code of record in the existing design basis stress report.

The MS and associated branch piping (inside and Class 1 piping outside containment) was evaluated for compliance with the original code of record, ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, 1974 Edition stress criteria, including the effects of EPU on piping stresses, piping supports including the associated building structure, piping interfaces with the RPV nozzles, 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. Other external loading conditions, such as Chugging and Condensation Oscillation also are not changed by EPU (see Section 2.6.1.2.1). The effect of the AP external loading condition is addressed at the end of this section. 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.

The Turbine Stop Valve Closure (TSVC) transient loading is considered one of the most significant loads for the qualification of piping and supports to EPU conditions because the TSVC load was already a significant load at transient CLTP conditions. The increase in the main steam flow rate for EPU will increase this load. The EPU evaluations used very conservative scaling factors (see Table 2.2-2a) to identify the components, pipe stress, and support loads that might exceed their allowable.

Pipe Stresses

As well as the MS piping inside containment, this discussion also pertains to that portion of the MS Class 1 piping that is located outside containment. A review of the increase in flow associated with EPU indicates that piping load changes do not result in load limits being exceeded for the MS system and attached branch piping or for RPV nozzles and containment penetrations. The original design analyses have sufficient margin between calculated stresses and ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, 1974 Edition allowable limits (see Tables 2.2-3a through 2.2-3f) to justify operation at EPU conditions. The pressure and temperature of the MS piping are unchanged for EPU.

Similarly, the branch pipelines (Safety Relief Valve Discharge Line (SRVDL), Reactor Core Isolation Cooling (RCIC), RPV Vent, and Main Steam drains including MSIV Drain) connected to the MS headers were evaluated to determine the effect of the increased MS flow on the lines. This evaluation concluded that there is no adverse effect on the existing MS branch line qualifications due to the increased flows resulting from EPU. As with the MS piping, the pressures and temperatures for these branch pipelines do not change as a result of EPU. A review was performed of postulated pipe break locations. The review was conducted in accordance with the requirements of the current license basis methodology. As a result of this review, no new postulated break locations were identified. Based on existing margins available for the MS piping, it was concluded that EPU does not result in reactions in excess of the current design capacity.

Pipe Supports

The MS piping (inside and outside containment) was evaluated for the effects of flow increase and vibration 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 exceed component or structure allowables.

The Turbine Stop Valve Closure (TSVC) transient load is one of the individual loads experiencing the most significant increase for the qualification of piping and supports to EPU conditions. The increased load is due to the increase in the main steam flow rate for EPU. The EPU evaluations used very conservative scaling factors (see Table 2.2-2a) to identify the components, pipe stress, and support loads that might exceed their allowable. Applying these conservative EPU scaling factors to the CLTP loads initially resulted in 4 supports inside containment that exceeded the allowable stresses.

A computer program and plant-specific model was used to analyze the TSVC EPU mass flows and corresponding pressures. The program and model was initially benchmarked against the original TSVC transient results. Then the benchmarked program and model was used to determine the TSVC loadings at EPU conditions. These results were used to remove unnecessary conservatisms that were inherent in the initial scaling factors and to develop more refined plant-specific scaling factors. The results of the subsequent plant-specific detailed computer analysis at EPU conditions showed that the scaling factors that had been applied to the CLTP TSVC loads were overly conservative for the supports. Consequently, using the results from the detailed computer analysis, no supports exceed the allowable stress.

Main Steam Isolation Valves

The MSIVs are part of the RCPB, and perform the safety function of steam line isolation during certain abnormal events and accidents. The MSIVs must be able to close within a specified time range at all design and operating conditions. They are designed to satisfy leakage limits set forth in the plant Technical Specifications. These design requirements are not adversely impacted by increased EPU flow, thus the original design remains adequate for EPU conditions.

The MSIVs have been evaluated, as discussed in Section 4.7 of ELTR2, Supplement 1 (Reference 3). The evaluation covers both the effects of the changes to the structural capability of the MSIV to meet pressure boundary requirements, and the potential effects of EPU-related changes to the safety functions of the 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 flow increases of about 24%. Table 1-2 provides the Maximum Nominal Dome Pressure and Temperature as well as the changes in steam and FW flows. From these parameters, it can be determined that the [[ ]] from ELTR2 is applicable to NMP2.

The Hydraulic Control System controls the MSIV closure time. Adjusting the pressure compensated flow control valve controls the stroke time. Once the flow control valve is adjusted for a particular extend speed, the pressure compensator feature of the valve will maintain this set speed even if external forces, such as valve disc thrust vary. The hydraulic damper senses the combined driving force of the pneumatic cylinder, the external closing springs, the steam drag force, the deadweight of the moving components, and the friction force. The steam drag force applied on the main disc increases due to an increase in steam flowrate. This force change is transmitted from the main disc to the valve stem, and then to the connecting hydraulic damper rod. It is then transmitted to the hydraulic damper and the hydraulic control circuit. As the driving force increases due to the higher steam flowrate, a spring inside the hydraulic control valve reduces the opening of an internal variable orifice in order to compensate for the higher closing force. The net driving force stays unchanged due to this compensating mechanism. The self-compensating feature of the hydraulic control valve will ensure the MSIV closing time will stay essentially unchanged as a result of changes in the steam flow rate. Therefore the MSIV performance is bounded by conclusions of the evaluation in Section 4.7 of ELTR2, and the MSIVs are acceptable for EPU operation.

Feedwater System Evaluation (Inside Containment)

The FWS and associated branch piping (inside containment) was evaluated for compliance with the original code of record, ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, 1974 Edition stress criteria, and for the effects of EPU on piping supports including the associated building structure. Piping interfaces with RPV nozzles, penetrations, flanges, and valves were also evaluated.

Because the FWS piping pressures and temperatures increase due to EPU, the effect of these parameters on the existing analyses was evaluated. Seismic inertia loads, and seismic building displacement loads are not affected by EPU; thus, there is no effect on the analyses for these load cases. Other external loading conditions, such as Chugging, and Condensation Oscillation also are not changed by EPU (See Section 2.6.1.2.1). The effect of the AP external loading condition is addressed at the end of this section. The increase in FWS flow results in increased forces from the assumed transient cases.

Pipe Stresses

As well as the FW piping inside containment, this discussion also pertains to that portion of the FW Class 1 piping that is located outside containment. A review of the increase in flow, operating pressure, and temperature associated with EPU indicates that piping load changes do not result in load limits being exceeded for the FWS and attached branch piping or for RPV nozzles. The original design analyses have sufficient margin between calculated stresses and ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, 1974 Edition, allowable limits to justify operation at EPU conditions.

A review was performed of postulated pipe break locations. The review was conducted in accordance with the requirements of the original license basis methodology. As a result of this review, no new postulated break locations were identified. Based on existing margins available for the FWS piping (see Tables 2.2-4a through 2.2-4c), it was concluded that EPU does not have an adverse effect on the FW piping design.

Pipe Supports

The FWS piping (inside containment) was evaluated for the effects of flow, operating pressure, and temperature increase on the piping snubbers, hangers, and struts. A review of the existing analyses for the system condition changes associated with EPU indicates that piping load changes do not result in any load limit being exceeded (see Table 2.2-4d). Therefore, the existing analyses bound the EPU conditions.

Other Piping Evaluation (Inside Containment)

As previously noted, 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. Other external loading conditions, such as Chugging, and Condensation Oscillation also are not changed by EPU (see Section 2.6.1.2.1). The effect of the AP external loading condition is addressed at the end of this section. Additionally, dynamic piping loads for RV and SRV at EPU conditions are bounded by those used in the existing analyses. These effects have been evaluated for the RCPB portion of the RPV head vent line, RV/SRV discharge piping and Reactor Water Cleanup piping, as required.

These systems were previously evaluated for compliance with the ASME Code stress criteria as required. Because none of these piping systems experience any significant change in operating conditions, they are all acceptable as currently designed.

A number of piping subsystems located inside containment move fluid through systems outside the RCPB piping. The flow, pressure, temperature, and mechanical loading for some of these piping systems do not increase for EPU. Consequently, there is no change in stress and fatigue evaluations and these piping systems are [[ ]] in the CLTR.

Large bore and small bore piping and supports not addressed in Section 3.5.1 of the CLTR were evaluated for acceptability at EPU conditions. The evaluation of the piping and supports was performed in a manner similar to the evaluation of RCPB piping systems and supports, using applicable code equations. The original codes of record (as referenced in the appropriate calculations), code allowable values, and analytical techniques were used and no new assumptions were introduced.

The CLTR requires a [[ ]] for piping systems where the loads and temperatures used in the CLTP analyses do not remain bounding for EPU containment hydrodynamic loads and short/long-term temperatures. 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 NMP2. The structures attached to the containment (wetwell) such as piping systems, vent penetrations, and valves are based on the DBLOCA hydrodynamic loads. For EPU conditions, the DBLOCA containment (wetwell) response loads were evaluated and found to be unchanged by EPU (see Section 2.6.1.2.1) and thus, there are no resulting effects on the containment wetwell attached structures.

EPU short/long-term suppression pool temperatures are evaluated in Section 2.6 and reported in Table 2.6-1. It was confirmed that the existing piping and pipe support analyses contain adequate margin to accommodate the effects of these small changes in short/long-term suppression pool temperatures for the following piping systems:

- RHR Low Pressure Coolant Injection lines
- High Pressure Core Spray/Injection lines (beyond the closed valve)
- Low Pressure Core Spray
- RCIC (water segment beyond the isolation valve)

Annulus Pressurization Load Increase – Piping, Components and Supports

During the review of the impact of the EPU conditions on the AP load break energy, several non-conservative assumptions were discovered related to the original design basis for mass energy release and the limiting mass energy release (see Section 2.6.2 for relative impacts on AP loads due to EPU and the non-conservative assumptions). As a result, new AP acceleration response spectra (ARS) were developed for use in the evaluation of piping, components and supports. The new AP ARS, at some nodal locations, demonstrated shifts in frequency, and/or increased accelerations over the CLTP AP ARS.

Annulus pressurization between the RPV and biological shield wall (BSW) most directly affects piping that is inside the drywell, attached to the RPV and/or supported from the BSW. RPV attached (RCPB) piping (ASME Class 1) systems include FWS, MSS, CSH, CSL, ICS, RCS, RHS, SLCS and WCS. Of these, only the FWS and MSS systems experienced changes in flows, pressures or temperatures as a result of EPU. The effect of the EPU related increased flows, pressures and/or temperatures as well as the effects of increased AP loads are provided in Table 2.2-3a; Tables 2.2-3c thru 2.2-3f; and Tables 2.2-4a thru 2.2-4d for the "A" loops of MSS and FWS. RCS (including the attached RHS shutdown cooling piping), CSL, WCS and ICS head spray piping inside primary containment, while not affected by EPU (no increase in flow, pressure or temperature), was also evaluated for the increased AP load. These systems were selected based on the locations with the minimum margin to design allowable at the vessel nozzle and include all the Class 1 systems discussed in USAR Section 6A.9.1.1.4, Results of Piping System Evaluation. The quantitative analysis determined that the piping, components and supports for these systems are within the design basis required margins. Based on this analysis, the other Class 1, 2 and 3 systems inside the drywell were assessed qualitatively to validate that all design margins are maintained at EPU conditions.

In addition to piping inside the drywell, the increased AP ARS were used to evaluate (1) piping inside the wetwell (primary containment below the drywell floor), and (2) piping outside, but connecting to primary containment penetrations. Even with the increased AP loads, other hydrodynamic loads remain governing or dominant in the determination of limiting faulted condition loads and stresses. Thus, the limiting faulted condition loads and stresses at these locations are unaffected by the increased AP loads.

Balance-of-Plant Piping (BOP) Components, and Supports (Outside Containment)

Operation at the EPU conditions increases stresses on some piping and piping system components due to slightly higher operating temperatures, pressures and flow rates internal to the pipes. The NMP2 piping systems determined to be impacted by EPU operation include:

- MSS – Main Steam System (outside containment)
- ESS - Extraction Steam System

- ASR – Radwaste Auxiliary Steam
- FWS – Feedwater (outside containment)
- FWR – Feedwater Pump Recirculation Balance Drum Leakoff
- CNM – Condensate
- CND – Condensate Demineralizers
- TME – Turbine Generator Gland Seal and Exhaust Steam
- DSM – Moisture Separator Vents and Drains
- DSR – Moisture Separator Reheater Vents and Drains
- HDL – Low Pressure Feedwater Heater Drains
- HDH – High Pressure Feedwater Heater Drains
- CWS – Circulating Water
- CCS – Turbine Building Closed Loop Cooling Water System
- RDS – Control Rod Drive Hydraulics
- ASS – Auxiliary Steam
- FWP – Feedwater Pump Seals and Leakoff
- DTM – Turbine Plant Miscellaneous Drains
- ARC – Condenser Air Removal
- OFG – Off Gas
- SVH – Feedwater Heater Relief Vents and Drains
- CNA – Auxiliary Condensate
- CNS – Condensate Makeup and Draw-off
- WCS – Reactor Water Cleanup System (outside containment)
- ZIP – Zinc Injection
- HRS – Hot Reheat System

For those systems with analysis, the maximum stress levels, minimum wall thickness evaluations, and fatigue analysis results were reviewed based on specific increases in temperature, pressure and flow rate (see Tables 2.2-5a through 2.2-5g). The above listed piping systems have been evaluated for the appropriate code criteria for the EPU conditions, based on the design margins between actual stresses and code limits in the original design. All piping stresses have been found to be below the code allowable of the present code of record: ANSI B31.1 Power Piping Code, 1973 Edition with addenda through Addendum C and ASME Boiler and Pressure Vessel Code – Section III, Division I, 1974 Edition. No new postulated pipe break locations were identified. For those systems that do not require a detailed analysis, pipe routing and flexibility was determined to remain acceptable.

Main Steam and Associated Piping System Evaluation (Outside Containment)

The MS piping system (outside containment) was evaluated for compliance with all codes and standards that are captured under NMP2 criteria, including the effects of EPU on piping stresses, piping supports, and the associated building structure, equipment nozzles, pipe break postulation, flanges and valves.

Because the effect of EPU on MS piping pressures and temperatures outside containment is minimal, there was no effect on the analyses for these parameters. The increase in MS flow results in increased forces from the turbine stop valve (TSV) and isolation valve closure transients. 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. For all piping and supports that are affected by this occasional event, the TSV closure contribution to the stress/load was increased by the appropriate factor, then recombined with the other dynamic load cases in the same manner as in the original calculation. The new combined stresses/loads were then compared to the Code allowables. The MS analysis results are provided in Table 2.2-5a.

Pipe Stresses

A review of the increase in flow associated with EPU was performed to determine if code allowable limits would be exceeded for the MS piping system outside containment. The original design has sufficient design margin to justify operation at EPU conditions. The effect of EPU on pressure and temperature of the MS piping is minimal. No new postulated pipe break locations were identified. The main steam stress analysis results for the non-Class 1 piping that interfaces with the Class 1 piping outside containment are provided in Table 2.2-5a. Piping stress increases do not result in code allowable limits being exceeded for the MS piping system outside containment.

Pipe Supports

The pipe supports and turbine nozzles for the MS piping system outside containment were evaluated for the increased loading associated with the turbine valve closure transients at EPU conditions. The evaluations demonstrate that all supports and turbine nozzles have adequate design margin to accommodate the increased loads and movements resulting from EPU. Based on existing margins available for the outside containment MS piping supports, it was concluded that EPU does not result in reactions on existing structures in excess of the current design capacity.

Feedwater System Evaluation (Outside Containment)

The design pressure of the feedwater piping will increase from 2,200 psig to 2,250 psig as a result of the reactor feed pump modification. The existing class break between the 1500 lb class piping and components which are subject to full feedwater pressure and the 900 lb class piping and components within the overpressure protection boundary of the RPV, currently occurs at the locked open stop valves 2FWS*HCV54A and B inside containment. This boundary will be moved upstream to the RPV side of outboard containment isolation valves 2FWS*MOV21A and B, in accordance with the provisions of ASME Section III, Division I, Article NB-7142. Piping and components between the new boundary at the outboard containment isolation valves and the RPV will be classified for service at the 900 lb class design conditions (1,300 psig at 575°F). The design pressure increase on the remaining 1500 lb class portions of the system including containment isolation valves 2FWS*MOV21A and B has been reviewed and determined to be acceptable for all impacted piping and feedwater components.

Pipe Stresses

Operation at EPU conditions increases stresses on piping and piping system components due to slightly higher operating temperatures, pressures and flow rates internal to the pipes. The piping systems outside containment have been evaluated for the appropriate code criteria for EPU conditions, based on the design margins between actual stresses and applicable code limits. All piping is below the code allowable of the present code of record: ANSI B31.1 Power Piping Code, 1973 Edition with addenda through Addendum C for all piping other than isolated segments of MS and FW piping at the non-safety to safety related interface, and ASME Boiler and Pressure Vessel Code – Section III, Division I, Subsection NB for these lines. This review shows that piping stresses under EPU conditions are in compliance with appropriate Code criteria. No new postulated pipe break locations were identified. The feedwater stress analysis results for the non-Class 1 piping that interfaces with the Class 1 piping outside containment are provided in Table 2.2-5b.

Details regarding analyses pertaining to high-energy piping failures outside containment are provided in Section 2.2.1 and the effects of the failures are discussed in Section 2.5.1.3.

Pipe Supports

Operation at EPU conditions increases the pipe support loadings due to increases in the temperature and transient loads of the affected piping systems (see Tables 2.2-5a through 2.2-5g).

The pipe supports of the systems affected by EPU loading increases (see listing above) were reviewed to determine if there is sufficient margin to support capacity to accommodate the increased loadings. This review shows that, in all cases except one, support loads under EPU conditions are in compliance with appropriate Code criteria. The condensate support requiring modification is located on piping that experiences a maximum summer design temperature change from 146°F at CLTP conditions to 159.4°F at EPU conditions. This temperature change results in a 17.6% increase in the thermal portion of the support load. Because a weld on this support currently has only 7% margin to its allowable load, the thermal load increase due to EPU will cause the load at this weld to exceed the allowable by about 10%, as shown in the following table.

Supports That Exceed Capacity		CLTP Load	EPU Load	Allowable Load	Ratio	Modification/Limiting Component
2CNM-PSR085A4	Fx	822	968	882	1.098	Weld between tube steel and existing steel
	Fy	9751	10504	10461	1.004	

Reactor Water Cleanup System Evaluation (Outside Containment)

The operating pressure of the Class 1 portion of the Reactor Water Cleanup piping will increase as a result of the reactor feed pump modification. However, the current analysis for this Class 1 piping uses an operating pressure that is higher than what is expected at EPU. The design pressure, design temperature and operating temperature of this piping are not changing due to EPU. Therefore, the current analysis remains acceptable for all Class 1 piping and supports in the Reactor Water Cleanup System.

Reactor Vessel and Supports

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Certain reactor vessel components are [[ ]] without detailed structural analysis. For components with no increase in flow, temperature, reactor internal pressure difference (RIPD), or other mechanical loads, no further evaluation is required. In addition, previous GE BWR power uprate experience has demonstrated that, using a modified version of the evaluation process documented in Appendix I of ELTR1 (Reference 2) to account for the 60-year license, numerous components do not exceed the 60-year 0.33 CUF criteria, thus requiring no further evaluation. The 60-year 0.33 CUF is a result of scaling the 40-year 0.5 CUF to account for the Time Limiting Aging Analysis (TLAA) factor of 1.5 as is the case for NMP2.

The 10 inches Welded Thermal Sleeve Nozzles, Closure Flange, Closure Flange Bolts, Core Spray Nozzle, CRD HSR Nozzle CRD Penetration, In-Core Detector, IRM/SRM Dry Tube, Power Range Detector, Core Spray Bracket, Steam Dryer Bracket, Core ΔP Nozzle, Recirculation Inlet Nozzle, Recirculation Outlet Nozzle, Refueling Bellows, RHR – LPCI Nozzle, Shroud Support and Support Skirt, Stabilizer Bracket, Steam Outlet Nozzle, Vent Nozzle, Jet Pump Instrumentation Nozzle, Water level Instrumentation Nozzle, Drain Nozzle, Feedwater Sparger Bracket, Guide Rod Bracket, Steam Dryer Hold Down Bracket, Top Head Lifting Lug Bracket, Head Cooling Spray Nozzle and Spare Nozzle components are confirmed to be consistent with the [[ ]] provided in the CLTR (Reference 1), ELTR1 (Reference 2), and ELTR2 (Reference 3) for NMP2. Therefore, these components are considered acceptable for EPU based on the EPU evaluation methodology.

The Top Head and Cylindrical Shell 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 were not considered to be pressure boundary components at the time that the OLTP evaluation was performed, and have not been evaluated for EPU.

ASME Section XI flawed components identified at NMP2 are: the Recirculation Inlet, Core Spray, and Feedwater Nozzles. Existing evaluations and repairs, including weld overlay of the Feedwater Nozzle, have been reviewed and found to remain acceptable for EPU operating conditions.

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, ASME BPV 1971 Code with addenda to and including Winter 1972 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 NMP2:

- CRD HSR Nozzle: This component was modified and the governing Code for the evaluation/modification is the ASME Boiler and Pressure Vessel Code, Section III, 1974 Edition with Addenda to and including Winter 1975.
- Feedwater Nozzle: This component was modified and the governing Code for the modification is the ASME Boiler and Pressure Vessel Code, Section III, 1974 Edition with Addenda to and including Summer 1976. Further analysis was performed using the 1989 Edition.
- 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.
- Recirculation Inlet Nozzle: This component was modified and the governing Code for the modification is the ASME Boiler and Pressure Vessel Code, Section III, 1974 Edition with Addenda to and including Summer 1976. In addition, the modification is to satisfy the requirements of ASME Boiler and Pressure Vessel Code, Section III, 1971 Edition with Addenda to and including Winter 1972, Article NB-3000.
- Universal 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.
- 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, 1986 Edition with Addenda to and including Winter 1987.
- 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.
- Head Cooling Spray 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. In addition, the RCIC system was modified (including the Head Cooling Spray) and the governing code for the evaluation/modification is the ASME Boiler and Pressure Vessel Code, Section III, 1977 Edition with No Addenda. The modification is to also satisfy the requirements of ASME Boiler and Pressure Vessel Code, Section III, 1974 Edition with Addenda up to and including Summer 1975 and ASME Boiler and Pressure Vessel Code, Section III, 1971 Edition with Addenda up to and Including Winter 1972.
- HPCS Injection Nozzle: This component was modified and the governing Code for the evaluation/modification is the ASME Boiler and Pressure Vessel Code, Section XI, 1989 Edition.

- Jet Pump Instrumentation Penetration Seal: This component was modified and the governing Code for the evaluation/modification is the ASME Boiler and Pressure Vessel Code, Section XI, 1974 Edition.
- Fatigue Monitoring System: Many components were modified and the governing code for the evaluation is the applicable code listed for the latest modification. If there are no modifications then the original Code of Construction may be used.

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, feedwater flow, and main steam flow) at EPU conditions are only slightly changed from those at current rated conditions. Evaluations were performed at conditions that bound the change in operating conditions. The evaluation type 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 NMP2 fatigue analysis results for the limiting components are provided in Table 2.2-6. The NMP2 analysis results for EPU show that components requiring evaluation do not exceed ASME code allowable values and no further analysis is required before these 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. Annulus pressurization loads increase for a number of components, all other emergency and faulted loads remain unchanged for EPU conditions. Components evaluated for increases in annulus pressurization do not exceed ASME code allowable values and no further analysis is required. Therefore, Code requirements are met for all RPV components under emergency and faulted conditions.

Conclusion

NMPNS has reviewed the evaluations related to the structural integrity of pressure-retaining components and their supports. For the reasons set forth above, NMPNS concludes that the effects of the proposed EPU on these components and their supports are adequately addressed. Based on the above, NMPNS further concludes that pressure-retaining components and their supports will continue to meet the requirements of 10 CFR 50.55a, GDC1, GDC2, GDC4, GDC14, and GDC15 following implementation of the proposed EPU. Therefore, NMPNS finds the proposed EPU acceptable with respect to the structural integrity of the pressure-retaining components and their supports.

2.2.3 Reactor Pressure Vessel Internals and Core Supports

Regulatory Evaluation

Reactor pressure vessel internals consist of all the structural and mechanical elements inside the reactor vessel, including core support structures. NMPNS reviewed the effects of the proposed EPU on the design input parameters and the design-basis loads and load combinations for the reactor internals for normal operation, upset, emergency, and faulted conditions. These include pressure differences and thermal effects for normal operation, transient pressure loads associated with LOCAs, and the identification of design transient occurrences. The NMPNS review covered (1) the analyses of flow-induced vibration for safety-related and non-safety related reactor internal components and (2) the analytical methodologies, assumptions, ASME Code editions, and computer programs used for these analyses. The NMPNS review also included a comparison of the resulting stresses and Cumulative Usage Factors (CUFs) against the corresponding Code-allowable limits. The NRC's acceptance criteria are based on (1) 10 CFR 50.55a and GDC1, 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) GDC2, 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) GDC4, 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) GDC10, 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.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of Constant Pressure Power Uprates. Sections 3.2 and 3.3 of the CLTR address the effect of Constant Pressure Power Uprate on Reactor Vessel and Reactor Internals, respectively. The results of this evaluation are described below.

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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. [[ ]] 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 3988 MWt and 105% of rated core flow. [[ ]] For components requiring an evaluation but not instrumented in the prototype plant, [[ ]] The expected vibration levels for EPU were estimated by extrapolating the vibration data recorded in the prototype plant or similar plants and on GEH BWR operating experience. These expected vibration levels were then compared with the established vibration acceptance limits. The following components were evaluated:

a) Shroud Head and Separator Assembly

b) Jet Pumps

c) Core Delta P Line

d) Guide Rods

e) In-Core Guide Tubes and Control Rod Guide Tubes

f) Jet Pump Sensing Lines

g) FW Sparger

h) Fuel Assembly, Top Guide, and Core Plate

i) RPV Top Head Spare Instrument Nozzle

j) RPV Top Head Vent Nozzle

k) RPV Head Spray Pipe and Head Spray Nozzle

l) Core Spray Piping

The results of the vibration evaluation show that continuous operation at a reactor power of 3988 MWt and 105% of rated core flow does not result in any detrimental effects on the safety-related reactor internal components. Flow induced vibration of critical reactor internal components at EPU is predicted based on the available startup test data at [[

]] Vibration amplitudes are also adjusted by a [[

]] The extrapolated vibration amplitude response under EPU conditions is compared with the acceptance criterion for each mode. [[

]] The summary of the evaluation methods and results for the following components are:

Shroud Head and Separator Assembly

For the shroud head, [[

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Jet Pumps

Results from strain gage measurements [[

]]

In 2004, slip joint clamps were installed on seven jet pumps (JP # 5, 6, 13, 15, 16, 19 and 20) and in 2006, slip joint clamps were installed on the remaining 13 jet pumps. The high vibrations that would have been caused by slip joint leakage flow instability have therefore been eliminated with the slip joint clamp installation. Hence, only the normal vibrational stresses currently exist and these are less than the acceptance criteria, as shown above. The slip joint clamps will be maintained during EPU operation to prevent slip joint leakage flow instability and if they are removed, other devices, which fulfill the same function, will be installed.

Core Delta P Line

The Core Delta P line is evaluated based on the instrumented prototype plant [[

]]

Guide Rods

The guide rod is subjected to cross flow vibration and the procedure for the lock-in phenomena using ASME Code Section III is used. [[

]]

In-Core Guide Tubes and Control Rod Guide Tubes

The flow-induced vibrations of these components are not affected by EPU as they are a function of the core flow and the core flow does not change during EPU. Hence there will be no increase in FIV stresses due to EPU. Maximum stresses during OLTP are well within the acceptance criteria and will remain about the same at EPU conditions.

Jet Pump Sensing Lines

Resonance of the Recirculation Pump vane passing frequency (VPF) with the natural frequency of the jet pump sensing line (JPSL) is the cause of the JPSL stress. [[

]]

FW Sparger

Feedwater sparger in NMP2 is of the [[

]]. Therefore, the NMP2 feedwater sparger is acceptable under FIV for EPU conditions.

Fuel Assembly, Top Guide and Core Plate

NMP2 uses the [[

]] Therefore, the NMP2 fuel assembly is acceptable under FIV for EPU conditions. The vibratory inertia forces of the fuel assembly are reacted by the top guide and core plate. In-reactor test results of GEH fuel show that fuel assembly vibration is less than 5% of the acceptance criteria. Thus, the vibratory loads produced by the fuel bundle on the top guide and core plate are deemed acceptable.

RPV Top Head Spare Instrument Nozzle

[[

]] Thus, the stress due to FIV at EPU conditions is deemed to be negligible.

RPV Top Head Vent Nozzle

[[

]] Therefore, the top head vent nozzle will be structurally adequate from a vibration viewpoint at EPU conditions.

RPV Head Spray Pipe and Head Spray Nozzle

[[

]] Thus, the stress due to FIV at EPU conditions is deemed to be negligible.

Core Spray Piping

[[

]] Therefore, the FIV stress due to vortex shedding at EPU conditions is minimal.

During EPU, the components in the upper zone of the reactor, such as the moisture separators, are mostly affected by the increased steam flow. Components in the core region and components such as the CS 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. For EPU conditions at NMP2, there is no change in the maximum licensed core flow in comparison to the CLTP condition, resulting in negligible changes in FIV on the components in the annular and core regions.

The calculations for EPU conditions indicate that vibrations of all safety-related reactor internal components are within the GEH acceptance criteria. The analysis is conservative for the following reasons:

- The GEH criteria of [[ ]] stress intensity is less than the ASME Section III Code criteria of 13,600 psi;
- The modes are [[ ]]; 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.

Steam Dryer

The NMP2 steam dryer has been evaluated for EPU steam flow conditions consistent with the guidance provided in BWRVIP-182, "Guidance for Demonstration of Steam Dryer Integrity for Power Uprate," issued January, 2008 (Reference 21). The analysis determined that two locations on the steam dryer require reinforcement of selected attachment welds to meet the 100% margin on alternating stress for the normal EPU operating condition. The locations are the inner and middle hood end cover welds and the lifting rod upper brace to vane bank weld. Additional detail for the steam dryer evaluation and the proposed dryer modifications has been provided in Attachment 13 to the EPU license amendment request.

Reactor Internal Pressure Differences

The increase in core average power alone would result in higher core loads and reactor internals 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, 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, and Faulted conditions are minimal and represent insignificant portions of the loads because of the small surface area, thus are not affected by EPU and are not evaluated for EPU.

Tables 2.2-7 through 2.2-9 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.

The core plate DP slightly increases from CLTP to EPU. The EPU is evaluated on the basis of operation at the same dome pressure but higher core power and steam flow. Therefore, pressures up-stream from the dome, such as above the core plate region, should increase slightly (consistent with higher steam flow resistance of approximately 1 psi in the steam dryers and separators), additionally, pressures downstream from the dome, such as the turbine inlet region, should decrease slightly (consistent with higher steam flow resistance in the steam lines).

Reactor Internals Structural Evaluation (Non-FIV)

The RPV internals consist of the Core Support Structure components and Non-Core Support Structure 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 EPU was performed consistent with the design basis analysis of the components. The reactor internal components evaluated are:

Core Support Components

- Shroud
- Shroud Support
- Core Plate
- Top Guide
- CRD Housing
- Control Rod Guide Tube
- Orificed Fuel Support
- Fuel Channel

Non-Core Support Components

- Shroud Head and Separators Assembly (including Shroud Head Bolts)
- Jet Pump Assembly
- Access Hole Cover
- Core Spray Line & Sparger
- Feedwater Sparger
- In-Core Housing and Guide Tube
- Core Differential Pressure Line
- LPCI Coupling
- Steam Dryer

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.

The effects of the thermal-hydraulic changes due to EPU on the reactor internals were evaluated. All applicable Normal (Service Level A), Upset (Service Level B), Emergency (Service Level C), and Faulted condition (Service Level D) loads were considered consistent with the existing design basis analysis. These loads include the RIPDs, deadweight, seismic loads, hydrodynamic loads such as SRV, LOCA, Annulus Pressurization (AP) and Jet Reaction (JR) loads, acoustic and flow induced loads, fuel lift loads, SCRAM and thermal loads.

EPU loads are compared to those used in the existing design basis analysis. If the EPU loads are bounded by the design basis loads for the RPV internals, the existing design basis qualification is valid for EPU. In such cases, no further evaluations are required or performed. For RPV internals exhibiting increases in loads, the method of analysis is to linearly scale the critical/governing stresses based on increase in loads as applicable, and compare the resulting stresses against the allowable stress limits, consistent with the design basis. Typically, new loads adequacy stress data are used as the baseline data. Conservative assessment is the initial approach; however, if required, excessive conservatism is removed from the existing assessment and/or the design basis analysis, as appropriate, and if justifiable.

Tables 2.2-10 and 2.2-11 present the governing stresses for the various reactor internal components as affected by EPU. All stresses and fatigue usage factors are within the design basis ASME Code allowable limits, and the RPV internal components are demonstrated to be structurally qualified for operation at EPU conditions.

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

a) **Shroud:** A quantitative evaluation of the shroud was performed for the changes in loads associated with EPU conditions. RIPDs, Deadweight, Seismic, SRV, LOCA, Annulus Pressurization (AP), Jet Reaction (JR), Acoustic and Flow Induced Loads (AFIL), and Fuel Lift Loads (FLL) are applicable loads for the shroud. Deadweight and Seismic loads remain unchanged for EPU with respect to the current design basis loads. SRV and LOCA loads remain bounded by the current design basis loads. EPU-based AP, JR and Acoustic loads increased with respect to the CLTP conditions. Fuel lift loads increased by up to 10% relative to the current design basis loads. RIPDs increased for Normal, Upset and Faulted conditions with respect to CLTP conditions. The structural integrity evaluation of the shroud was performed with respect to the new loads adequacy evaluation. Based on this evaluation, it was concluded that the Normal, Upset, Emergency and Faulted condition stresses in the shroud remain within the design basis ASME Code allowable stress limits. The Upset condition was found to be limiting in terms of available stress margin. Hence, the shroud remains qualified for EPU.

b) **Shroud Support:** A quantitative evaluation of the shroud support was performed for the changes in loads associated with EPU conditions. RIPDs, Deadweight, Seismic, SRV, LOCA, AP, JR, Acoustic, and Fuel lift loads are applicable loads for the shroud support. Deadweight and Seismic loads remain unchanged for EPU with respect to the current design basis loads. SRV and LOCA loads remain bounded by the current design basis loads. EPU-based AP, JR and Acoustic loads increased with respect to CLTP conditions. Fuel lift loads increased by up to 10% relative to the current design basis loads. RIPDs increased for Normal, Upset and Faulted conditions with respect to CLTP conditions. The structural integrity evaluation of the shroud support was performed with respect to the new loads adequacy evaluation. Based on this evaluation, the Normal, Upset, Emergency, and Faulted condition loads for EPU were found to be bounded by the original design basis loads to which the shroud support plate and the legs were designed. It was concluded that the Normal, Upset, Emergency and Faulted condition stresses in the shroud support remain within the design basis ASME Code allowable stress limits. The Upset condition was found to be limiting in terms of available stress margin. Hence, the shroud support remains qualified for EPU.

c) **Core Plate:** A quantitative evaluation of the core plate was performed for the changes in loads associated with EPU conditions. RIPDs, Deadweight, Seismic, SRV, LOCA, AP, JR, and FLL are applicable loads to the core plate. Deadweight and Seismic loads remain unchanged for EPU with respect to the current design basis loads. SRV and LOCA loads remain bounded by the current design basis loads. A comparison of EPU-based AP, JR and the original design basis AP and JR loads showed that the original design basis AP and JR loads remain bounding. Fuel lift loads increased by up to 10% relative to the current design basis loads. RIPDs increased for Normal and Upset conditions and remained bounded for the Faulted conditions with respect to the CLTP conditions. The end-of-life (60 years) preload in the core plate bolts was used considering the relaxation in the bolt preload due to irradiation to evaluate the potential for the core plate to slide or lift off. It was concluded that there is no concern for the core plate to lift off or slide in the Normal, Upset, Emergency and Faulted conditions. The core plate longest beam was evaluated for its ability to resist buckling under all loading conditions for EPU. It was shown that buckling of the longest beam is not a concern since the buckling criteria were met for Normal, Upset, Emergency, and Faulted conditions. The core plate bolts were also evaluated for fatigue. It was shown that the 40 and 60 year fatigue usage is negligible. The structural integrity evaluation of the core plate was performed with respect to the new loads adequacy evaluation. Based on this evaluation, it was concluded that the Normal, Upset, Emergency and Faulted condition stresses in the core plate remain within the design basis ASME Code allowable stress limits. The Upset condition was found to be limiting in terms of available stress margin. Hence, the core plate remains qualified for operation at EPU conditions.

d) **Top Guide:** A qualitative evaluation of the top guide was performed for the changes in loads associated with EPU conditions. RIPDs, Deadweight, Seismic, SRV, LOCA, AP, JR, and Fuel Lift loads are applicable to the top guide. Deadweight and Seismic loads remain unchanged for EPU with respect to the current design basis loads. SRV and LOCA loads remain bounded by the current design basis loads. A comparison of EPU-based AP, JR and the original design basis AP and JR loads showed that the original design basis AP and JR loads remain bounding. Fuel lift loads increased by up to 10% relative to the current design basis loads. RIPDs for Normal, Upset and Faulted conditions remained bounded with respect to CLTP conditions. The structural integrity evaluation of the top guide was performed with respect to the new loads adequacy evaluation. Based on this evaluation, it was concluded that the Normal, Upset, Emergency and Faulted condition stresses in the top guide remain within the design basis ASME Code allowable stress limits, and remain bounded by the new loads adequacy evaluation. The Upset condition was found to be limiting in terms of available stress margin. Hence, the top guide remains qualified for operation at EPU conditions.

e) **CRD Housing:** A quantitative evaluation of the CRD housing was performed for the changes in loads associated with EPU conditions. Reactor pressure, SCRAM loads, Deadweight, Seismic, SRV, LOCA, AP, JR, FLL and flow impingement loads are applicable to the CRD housing. Flow impingement loads corresponding to ICF conditions were considered. Reactor pressure, SCRAM loads, Deadweight and Seismic loads remain unchanged for EPU with respect to the current design basis loads. SRV and LOCA loads remain bounded by the current design basis loads. EPU-based AP and JR loads increased with respect to the original design basis loads. Fuel lift loads increased by up to 10% relative to the current design basis loads. The structural integrity evaluation of the CRD housing was performed with respect to the new loads evaluation. Based on this evaluation, it was concluded that the Normal, Upset, Emergency and Faulted condition stresses in the CRD housing remain within the design basis ASME Code allowable stress limits, and remain bounded by the CLTP evaluation. The Upset condition was found to be limiting in terms of available stress margin. Hence, the CRD Housing remains qualified for operation at EPU conditions.

f) **Control Rod Guide Tube:** A quantitative evaluation of the CRGT was performed for the changes in loads associated with EPU conditions. RIPDs, Deadweight, Seismic, SRV, LOCA, AP, JR, FLL and flow impingement loads are applicable to the CRGT. Flow impingement loads corresponding to ICF conditions were considered. Deadweight and Seismic loads remain unchanged for EPU with respect to the current design basis loads. RIPDs increased for Normal and Upset conditions and remained bounded for the Faulted conditions. SRV and LOCA loads remain bounded by the current design basis loads. EPU-based AP and JR loads increased with respect to the original design basis loads. Fuel lift loads increased by up to 10% relative to the current design basis loads. The structural integrity (in terms of stress and buckling capability) evaluation of the CRGT was performed with the design basis evaluation. Based on this evaluation, it was concluded that the Normal, Upset, Emergency and Faulted condition stresses in the CRGT remain within the design basis ASME Code allowable stress limits, and the original design basis buckling criteria for Normal, Upset, Emergency, and Faulted conditions are also satisfied. The Upset condition was found to be limiting in terms of available stress margin. Hence, the CRGT remains qualified for operation at EPU conditions.

g) **Orificed Fuel Support (OFS):** A quantitative evaluation of the OFS was performed for the changes in loads associated with EPU conditions. Deadweight (including the deadweight of the fuel bundles), Seismic, SRV, LOCA, AP, JR, and fuel lift loads are applicable to the OFS. Deadweight and Seismic loads remain unchanged for EPU with respect to the current design basis loads. The RIPDs increased for the Normal and Upset conditions with respect to CLTP, and remained bounded for the Faulted conditions. SRV and LOCA loads remain bounded by the current design basis loads. EPU-based AP and JR loads increased with respect to the original design basis loads. Fuel lift loads increased by up to 10% relative to the current design basis loads. The structural integrity evaluation of the OFS was performed consistent with the design basis evaluation. Based on this evaluation, it was concluded that the Normal, Upset, Emergency and Faulted condition stresses in the OFS remain within the design basis ASME Code allowable stress limits. The Upset condition was found to be limiting in terms of available stress margin. Hence, the OFS remains qualified for operation at EPU conditions.

h) **Fuel Channel:** The fuel channel is qualified by proprietary methodology by GNF. The Normal, Upset, Emergency, and Faulted condition RIPDs for the fuel channel are within the respective design limits for the channel. Additionally, the channel/control blade interference is not impacted by EPU.

i) **Shroud Head and Separators Assembly (including Shroud Head Bolts):** A quantitative evaluation of the shroud head bolts (SHBs) was performed for the changes in loads associated with EPU conditions. RIPDs, Deadweight, Seismic loads, SRV, LOCA, AP, JR and thermal effects are applicable loads to the SHBs. Deadweight and Seismic loads remain unchanged for EPU with respect to the current design basis loads. RIPDs across the shroud head increased in the Normal and Upset conditions, and remained bounded in the Faulted condition with respect to CLTP conditions. SRV and LOCA loads remain bounded by the current design basis loads. A comparison of EPU-based AP, JR and the original design basis AP and JR loads showed that the original design basis AP and JR loads remain bounding. The largest reduction in the shroud/RPV annulus temperature was found between the 100% EPU/105% Core flow and the 100% EPU/100% Core flow – 20°F feedwater temperature reduction (FWTR) conditions. The 3°F temperature reduction is <1% with respect to CLTP conditions. Hence, the effect on the SHB preload is insignificant. The structural integrity evaluation of the SHBs (most limiting component of the shroud head and steam separator assembly) was performed with respect to the new loads adequacy evaluation. This evaluation was performed assuming 29-functioning or good SHBs (or up to 7 non-functioning SHBs) consistent with the previous evaluation. Based on the evaluation, it was concluded that the Normal, Upset, Emergency and Faulted condition stresses in the SHBs remain within the design basis ASME Code allowable stress limits. The Upset condition was found to be limiting in terms of available stress margin. Hence, the Shroud Head and Separators Assembly (with 29-functioning and uniformly spaced SHBs) remains qualified for operation at EPU conditions.

j) **Jet Pump Assembly:** A quantitative evaluation of the jet pump assembly was performed for the changes in loads associated with EPU conditions. RIPDs, Reactor pressure loads, Deadweight, Seismic loads, SRV, LOCA, AP, JR, Hydraulic loads, thermal loads, and Acoustic loads are applicable to the jet pump assembly components. Deadweight and Seismic loads remain unchanged for EPU with respect to the current design basis loads. The RIPDs across the shroud support plate increased for EPU for the Normal, Upset and Faulted conditions with respect to CLTP conditions. SRV and LOCA loads remain bounded by the current design basis loads. EPU-based AP and JR loads increased with respect to the original design basis loads. The recirculation loop drive flow for EPU increased (at 100% flow control valve position) with respect to CLTP. This drive flow was considered in the beam bolt calculations for EPU in conjunction with other applicable loads. The calculated beam bolt load was compared to the end-of-life bolt preload. The faulted condition was found to be limiting for the beam bolt load calculations. The largest reduction in the shroud/RPV annulus temperature was found between the 100% EPU/105% Core flow and the 100% EPU/100% Core flow – 20°F feedwater temperature reduction conditions. The 3°F temperature reduction is <1% with respect to CLTP conditions. Therefore, the effect on the thermal stress is insignificant. Based on the stress and fatigue evaluation of the jet pump assembly, it was concluded that the Normal, Upset, Emergency and Faulted condition stresses remain within the design basis ASME Code allowable stress limits. The Faulted condition was found to be limiting in terms of available stress margin for the riser brace. Hence, the jet pump assembly remains qualified for operation at EPU conditions.

k) **Access Hole Cover (AHC):** A qualitative evaluation of the access hole cover was performed for the changes in loads associated with EPU conditions. Both AHC designs ("flat plate" and "top hat") were evaluated. RIPDs, differential thermal and pressure loads, Deadweight, Seismic, SRV, LOCA, AP, JR, and Acoustic loads are applicable to the AHCs. RIPDs across the shroud support plate are applicable to the "flat plate" AHC. RIPDs increased for EPU in the Normal, Upset and Faulted conditions with respect to CLTP. Acoustic loads for the AHCs increased for EPU conditions. Deadweight and Seismic loads remain unchanged for EPU conditions with respect to the current design basis loads. SRV and LOCA loads remain bounded by the current design basis loads. EPU-based AP and JR loads increased with respect to the original design basis loads. The largest reduction in the shroud/RPV annulus temperature was found between the 100% EPU/105% Core flow and the 100% EPU/100% Core flow – 20°F feedwater temperature reduction conditions. The 3°F temperature reduction is <1% with respect to CLTP conditions. Therefore, the effect on the thermal stress is insignificant. The structural integrity evaluation of the AHC was performed consistent with the design basis evaluation. Both AHC designs were qualitatively assessed by comparison with the identical designs for similar plants for which detailed stresses are available, and adjusted for the applicable NMP2 loads. Based on this evaluation, it was concluded that the Normal, Upset, Emergency and Faulted condition stresses in the AHC remain within the design basis ASME Code allowable stress limits. The Upset condition was found to be limiting in terms of available stress margin. Hence, both access hole cover designs are qualified for operation at EPU conditions.

l) **Core Spray Line & Sparger:** A qualitative evaluation of the core spray line and sparger was performed for the changes in loads associated with EPU conditions. Deadweight, Seismic, SRV, LOCA, AP, JR and thermal effects are applicable to the core spray line and sparger. Deadweight and Seismic loads remain unchanged for EPU conditions with respect to the current design basis loads. The system flow and temperature for EPU conditions remain unaffected with respect to CLTP. SRV and LOCA loads remain bounded by the current design basis loads. EPU-based AP and JR loads increased with respect to the original design basis loads. The largest reduction in the shroud/RPV annulus temperature was found between the 100% EPU/105% Core flow and the 100% EPU/100% Core flow – 20°F feedwater temperature reduction conditions. The 3°F temperature reduction is <1% with respect to the CLTP condition. Therefore, the effect on the thermal stress is insignificant. The structural integrity evaluation of the core spray line and sparger was performed with respect to the new loads evaluation. Based on this evaluation, it was concluded that the Normal, Upset, Emergency and Faulted condition stresses in the core spray line and sparger remain bounded by the previous new loads adequacy and CLTP evaluations. The Upset condition was found to be limiting in terms of available stress margin. Hence, the core spray line and sparger remain qualified for operation at EPU conditions.

m) **Feedwater Sparger:** A qualitative evaluation of the feedwater sparger was performed for the changes in loads associated with EPU conditions. Deadweight, Seismic, SRV, LOCA, AP, JR, Hydraulic, and thermal loads are applicable to the feedwater sparger. The feedwater flow at EPU conditions increased with respect to CLTP. However, the hydraulic loads are small compared to other primary loads, and the increase in the stress is small compared to the stress margin available. Deadweight and Seismic loads remain unchanged at EPU conditions with respect to the current design basis loads. SRV and LOCA loads remain bounded by the current design basis loads. EPU-based AP and JR loads increased with respect to the original design basis loads. The primary stresses in the feedwater sparger are small compared to the thermal stresses. Thermal stresses in the feedwater sparger are high and of a cyclical nature due to the system thermal transients (e.g., turbine roll). Hence, fatigue is the most limiting parameter for the feedwater sparger. Given that large margin of safety exists in the fatigue usage factor, it is concluded that sufficient margin exists to offset a small increase in the fatigue usage due to the 20°F reduction in the feedwater temperature, the 3°F reduction in the annulus temperature, and an increase of ~14% in the feedwater flow. Based on this evaluation, it was concluded that the Normal, Upset, Emergency and Faulted condition stresses and fatigue usage in the feedwater sparger remain within the design basis ASME Code allowable limits. The Upset condition was found to be limiting in terms of available stress margin. Hence, the feedwater sparger remains qualified for operation at EPU conditions.

n) **In-Core Housing and Guide Tube:** A qualitative evaluation of the in-core housing and guide tube was performed for the changes in loads associated with EPU conditions. Deadweight, Seismic, SRV, LOCA AP, JR, reactor pressure, and flow impingement loads are applicable for this component. Deadweight and Seismic loads remain unchanged at EPU conditions with respect to the current design basis loads. SRV and LOCA loads remain bounded by the current design basis loads. EPU-based AP and JR loads increased with respect to the original design basis loads. The flow impingement loads are bounded by those of the ICF condition. The structural integrity evaluation of the in-core housing and guide tube was performed with respect to the new loads evaluation. Based on this evaluation, it was concluded that the Normal, Upset, Emergency and Faulted condition stresses remain bounded by the CLTP evaluation. The Upset condition was found to be limiting in terms of available stress margin. Hence, the in-core housing and guide tube remain qualified for operation at EPU conditions.

o) **Core Differential Pressure Line:** A qualitative evaluation of the core differential pressure line was performed for the changes in loads associated with EPU conditions. Deadweight, Seismic, SRV, LOCA, AP, JR, and flow impingement loads are applicable for this component. Deadweight and Seismic loads remain unchanged at EPU conditions with respect to the current design basis loads. SRV and LOCA loads remain bounded by the current design basis loads. EPU-based AP/JR loads increased with respect to the original design basis loads. The flow impingement loads are bounded by those of the ICF condition. The structural integrity evaluation of the core differential pressure line was performed with respect to the new loads evaluation. Based on this evaluation, it was concluded that the Normal, Upset, Emergency and Faulted condition stresses remain bounded by the new loads evaluation. The Upset condition was found to be limiting in terms of available stress margin. Hence, the core differential pressure line remains qualified for operation at EPU conditions.

p) **LPCI Coupling:** A qualitative evaluation of the LPCI coupling was performed for the changes in loads associated with EPU conditions. Deadweight, Seismic, SRV, LOCA, AP and JR loads are applicable for the LPCI coupling. Deadweight and Seismic loads remain unchanged at EPU conditions with respect to the current design basis loads. SRV and LOCA loads remain bounded by the current design basis loads. EPU-based AP/JR loads increased with respect to the original design basis loads. The structural integrity evaluation of the LPCI coupling was performed with respect to the new loads evaluation. Based on this evaluation, it was concluded that the Normal, Upset, Emergency and Faulted condition stresses remain within the design basis ASME Code allowable limits. The Emergency condition was found to be limiting in terms of available stress margin. Hence, the LPCI coupling remains qualified for operation at EPU conditions.

q) **Steam Dryer**: The steam dryer was evaluated for the changes in loads associated with EPU conditions. RIPDs, Deadweight, Seismic, and SRV loads are applicable to the steam dryer. Deadweight, SRV and Seismic loads remain unchanged for EPU with respect to the current design basis loads. The RIPDs increased at EPU conditions for the Normal and Upset conditions with respect to CLTP conditions. The structural integrity evaluation of the steam dryer was performed consistent with the EPU loads and it was concluded that the Normal, Upset, Emergency and Faulted condition stresses in the steam dryer remain within the design basis ASME Code allowable stress limits. See Attachment 13 to the EPU license amendment request for details.

Steam Dryer/Separator Performance

At NMP2, the performance of the steam separators and dryer has been evaluated to ensure that the quality of the steam leaving the reactor pressure vessel continues to meet existing operational criteria at EPU conditions. 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 results of the evaluation demonstrate that the steam separator-dryer performance remains acceptable (e.g., moisture content $\leq$ 0.1 weight %) at EPU conditions.

Conclusion

NMPNS has reviewed the structural integrity of reactor internals and core supports and concludes that the effects of the proposed EPU on the reactor internals and core supports have been adequately addressed. NMPNS further concludes that the reactor internals and core supports will continue to meet the requirements of 10 CFR 50.55a, GDC1, GDC2, GDC4, and GDC10 following implementation of the proposed EPU. Therefore, NMPNS finds the proposed EPU acceptable with respect to the design of the reactor internal and core supports.

2.2.4 Safety-Related Valves and Pumps

Regulatory Evaluation

NMPNS review included certain safety-related pumps and valves typically designated as Class 1, 2, or 3 under Section III of the ASME B&PV Code and within the scope of the ASME Operations and Maintenance (O&M) Code, as applicable. NMPNS review focused on the effects of the proposed EPU on the required functional performance of the valves and pumps. The review also covered any effects that the proposed EPU may have on the NMPNS's motor-operated valve (MOV) programs related to GL 89-10, GL 96-05, and GL 95-07. NMPNS also evaluated the lessons learned from the MOV program and the application of those lessons learned to other safety-related power-operated valves. The NRC's acceptance criteria are based on (1) GDC1, 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) GDC37, GDC40, GDC43, and GDC46, 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) GDC54, 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 in-service testing program requirements identified in that section.

Technical Evaluation

2.2.4.1 Background

In-Service Testing of Safety-Related Pumps and Valves

In-Service Testing of Safety-related Pumps and Valves is addressed in USAR Section 3.9A.6 and documented in the NMPNS In-Service Testing Program. Nine Mile Point Nuclear Station Pump and Valve In-Service Testing Program, hereafter referred to as the IST Program, has been prepared to summarize the test program for certain pumps and valves pursuant to the requirements of the 10 CFR 50.55a(f).

NMP2 Improved Technical Specifications 5.5.6 In-Service Testing Program, states that this program provides controls for in-service testing of ASME Class 1, 2 and 3 pumps and valves.

The IST Program was prepared in accordance with the ASME O&M Code – 2004 edition.

Containment Leakage Rate Testing Program

Containment Leakage Rate Testing is addressed in USAR Section 6.2.6. The NMP2 Containment Leakage Rate Testing Program implements testing requirements in accordance with 10CFR 50 Appendix J, Option B, as modified by any approved exemptions, and guidelines contained in RG 1.163, Performance-Based Containment Leak Test Program (dated September

1995). Tests that measure containment isolation valve leak rates (Type C tests) are performed using the Technical Specification value for Pa of 39.75 psig. From the containment analysis at the EPU conditions, the peak containment pressure is 52.9 psia (38.2 psig) for a LOCA (PUSAR Section 2.6.1). Because the containment peak pressure for EPU is lower than the Technical Specification Pa, the current test criterion bounds the peak LOCA containment pressure for the proposed EPU. Thus, the leak rate testing requirements for containment isolation valves are not impacted by the proposed EPU.

Pumps in the IST Program

The scope of the IST Program is derived from the OM-6 requirements for the ASME Code Class 1, 2 and 3 pumps providing a safety-related function. Table 2.2-12 lists the systems with pumps in the IST Program.

Valves in the IST Program

The scope of the IST Program is derived from the OM-10 requirements for the active and passive ASME Code Class 1, 2, and 3 valves. The IST program includes Motor-Operated Valves (MOVs), Air-Operated Valves (AOVs), Solenoid-Operated Valves (SOVs), check valves, pressure relief valves and thermal relief valves. Table 2.2-12 lists the systems with valves in the IST Program.

Motor Operated Valve Program

The NMP2 Motor Operated Valve Program implements the recommendations and requirements of GL 89-10, "Safety-Related Motor Operated Valve Testing and Surveillance". The scope of the program also includes the requirements of GL 96-05, "Verification of Design-Basis Capability of Safety-Related Motor Operated Valves". Table 2.2-12 indicates the systems which contain GL 89-10 (Reference 23) and GL 96-05 (Reference 24) MOVs.

Air-Operated Valve Program

The NMP2 Air Operated Valve Program addresses aspects of AOV design, performance monitoring and maintenance that are necessary to provide reasonable assurance that design bases functions will be accomplished. The AOV Program valve population includes valves as described in INPO document NX-1018 JOG AOV Program. This includes safety significant AOVs that are required to change position during a design basis event, i.e., those NMP2 designated as "CAT 1A" and "CAT 2A." Table 2.2-12 indicates the systems which contain AOV Program valves.

Lessons Learned

NMP2 IST Program, Containment Leak Rate Program, MOV Program and AOV Program utilize the NMPNS Corrective Action Program to evaluate and resolve non-conforming conditions identified during program performance. The purpose of the NMPNS Corrective Action Program is to stimulate and manage continuous improvement of station and organizational performance through identification, evaluation, correction and prevention of reoccurrence of unwanted and/or unexpected conditions, deviations, events, or issues that have the potential for affecting the safe, reliable, and efficient operation of NMPNS. Included in the program is recognition of any lessons learned or improvement opportunities identified from an assessment of missed opportunities to avoid the event/condition/issue. CNG-CA-1.01-1000 is the administrative procedure that implements the requirements of the corrective action process from 10CFR50 Appendix B.

2.2.4.2 Description of Analyses and Evaluations

This section addresses the impact of EPU on the performance requirements of NMP2 Safety-Related pumps and valves in the IST, Motor Operated Valve, and Air Operated Valve Programs. The discussion is organized by system or groups of systems and the respective Program pumps and valves are discussed therein. Each system was analyzed to define any parameter changes such as pressure, flow and process & ambient temperatures resulting from implementation of EPU. The minor impact to normal operating and DBA temperatures due to EPU does not require a change to the temperature assumptions used in the MOV voltage drop calculations, therefore there is no effect on these calculations.

Systems Not Significantly Affected by EPU

The Breathing Air, Instrument Air, Service Air, Nitrogen, Reactor Building Equipment Drains, Reactor Building Floor Drains, Nuclear Boiler Instrumentation, and Containment Leakage Monitoring systems are addressed in Section 2.5.7. The Emergency Diesel Generator and Auxiliaries (Sections 2.3.3 and 2.5.6.1), Control Building HVAC and Control Room Chilled Water Systems (Section 2.7.3) and Fire Protection Water System (Section 2.5.1.4) are not significantly affected by EPU. There are no system parameter changes, and no modifications are being made to these systems as a result of EPU. Therefore the NMP2 program valves in these systems are not impacted by EPU.

Nuclear Steam Supply Systems

The Nuclear Steam Supply System (NSSS) systems include the Low Pressure Core Spray (Section 2.8.5.6.2), High Pressure Core Spray (Section 2.8.5.6.2), Reactor Core Isolation Cooling (Section 2.8.4.3), Residual Heat Removal (PUSAR section 2.8.4.4), Reactor Water Cleanup (Section 2.1.7) and Standby Liquid Control (Section 2.8.4.5). Evaluations show that the EPU has no impact on system operating pressures, flow rates, and pump head performance for Low Pressure Core Spray, High Pressure Core Spray, Reactor Core Isolation Cooling, and Residual Heat Removal systems. No modifications are being made to these systems. Based on these evaluations, EPU has no impact on the performance characteristics and IST Plan requirements for safety-related pumps and valves in the Low Pressure Core Spray, High Pressure Core Spray, Reactor Core Isolation Cooling, or Residual Heat Removal systems.

Reactor Water Cleanup System

The Reactor Water Cleanup system changes due to EPU are addressed in Section 2.1.7. The slight increase in the system operating pressure is due to the increase in Feedwater system operating pressure. The MOV affected by EPU is evaluated in Section 2.2.4.3. The ability of the IST Program check valves to perform their safety functions is not impacted by EPU.

Standby Liquid Control System

The Standby Liquid Control system changes due to EPU are addressed in Section 2.8.4.5. The Standby Liquid Control maximum pump discharge pressure is increased by 1 psig, which remains below the pressure at which the pump is currently tested to satisfy Technical Specification requirements. The ability of the IST Program valves to perform their safety functions is not impacted by EPU.

Balance of Plant Systems

Reactor Building Closed Loop Cooling System

The Reactor Building Closed Loop Cooling system changes due to EPU are addressed in Section 2.5.3.3.1. No modifications are being made as a result of EPU. The NMP2 program valves are not impacted by EPU.

Feedwater / Feedwater Pump Recirculation Systems

The Feedwater system changes due to EPU are addressed in Section 2.5.4.4. The temperature, flow, operating pressure and design pressure will increase. The piping downstream of the feedwater pumps will be rerated. The NMP2 Program MOVs and AOVs affected by EPU are evaluated in Section 2.2.4.3. The ability of the IST Program check valves to perform their safety functions is not impacted by EPU.

Standby Gas Treatment System

The Standby Gas Treatment system changes due to EPU are addressed in Section 2.5.2.1. No modifications are being made as a result of EPU. The NMP2 program valves are not impacted by EPU.

Post-LOCA Combustible Gas Control System

The Post-LOCA Combustible Gas Control system includes the Containment Monitoring, Containment Purge and Hydrogen Recombiner systems. The systems changes due to EPU are addressed in Section 2.6.4. No modifications are being made to these systems as a result of EPU. The NMP2 program valves are not impacted by EPU.

Main Steam System

The proposed EPU results in main steam flow increase. Main steam header pressure will remain the same or slightly lower due to increased friction losses at higher EPU flow rates. The Main Steam Safety Relief Valves are addressed in Section 2.8.4.2. The Main Steam Isolation Valves are addressed in Section 2.2.2.1. The ability of the NMP2 program MOVs to perform their safety function(s) is not impacted by EPU. The system check valves are not adversely impacted by EPU.

Reactor Recirculation System

The Reactor Recirculation system changes due to EPU are addressed in Section 2.8.4.6. No modifications are being made to the system valves as a result of EPU. The IST Program valves are not impacted by EPU.

Control Rod Drive Hydraulic System

The Control Rod Drive Hydraulic system changes due to EPU are addressed in Section 2.8.4.1. No modifications are being made to the system as a result of EPU. The NMP2 program valves are not impacted by EPU.

Spent Fuel Pool Cooling System and Alternate Heat Decay Removal System

The Spent Fuel Pool Cooling and Alternate Decay Heat Removal systems are addressed in Section 2.5.3.1. No modifications are being made to the systems as a result of EPU. The IST program pumps and NMP2 program valves are not impacted by EPU.

Service Water System

The Service Water system changes due to EPU are addressed in Section 2.5.3.2. The non-safety related portion of the Service Water system will be modified to provide cooling water to four additional area coolers in the turbine building. System components within the IST, MOV and AOV Programs are not affected. The IST Program pumps and NMP2 program valves are not impacted by EPU.

Turbine Building Closed Loop Cooling

The Turbine Building Closed Loop Cooling system changes due to EPU are addressed in Section 2.5.3.3.2. No modifications are being made, but system flows will be optimized to address the additional heat load as a result of EPU. The NMP2 program valves are not impacted by EPU.

Generic Letter 95-07

GL 95-07 addresses "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves", 8/17/1995 (Reference 25). Pressure locking and thermal binding had been previously evaluated for all NMP2 safety related gate valves, and MOVs were modified to provide mitigation of pressure locking occurrences. Review of the safety related gate valves indicates that they are not susceptible to pressure locking and thermal binding as a result of EPU.

2.2.4.3 Individual Component Evaluations

Reactor Water Cleanup system valve 2WCS*MOV200

This MOV is a containment isolation valve that has a safety function to close remotely by the operator to provide long-term leakage protection. Review of valve calculations indicates that the maximum DP in which the valve needs to close against is equal to the Feedwater system operating pressure between 2FWS*MOV21A &B and the reactor pressure vessel. This pressure will increase by approximately 32 psig under EPU conditions. To verify the acceptability of the valve actuator for operation under EPU conditions, the sizing calculation was performed at the new maximum expected differential pressure (MEDP). The results of this analysis verified that the MOV will continue to perform reliably at the new EPU pressure with the GL 89-10 (Reference 23) required 10% margin. There are no physical changes to the valve as a result of EPU.

Reactor Feedwater system valves 2FWS*MOV21A & 2FWS*MOV21B

These MOVs are containment isolation valves that are normally opened and have a safety function to close to provide long-term leakage protection. The system temperature, flow, operating pressure and design pressure will increase due to EPU and the valves will be rerated to 2,250 psig. The design required MEDP used in the GL 89-10 (Reference 23) analysis is based on static head from the RPV which is not affected by EPU. EPU has no impact on the safety function of these valves. The physical configuration of the associated piping is not altered and there are no physical changes to the valves as a result of EPU.

2FWR-FV2A, 2FWR-FV2B, & 2FWR-FV2C

The Feedwater Pump Recirculation (FWR) Program AOVs (2FWR-FV2A, 2B & 2C) are non-safety related valves in the minimum flow recirculation lines for the main feedwater pumps. The valves are used when the normal flow path to the reactor vessel is unavailable or insufficient to sustain recommended flow conditions for the pumps (i.e., during plant start up, etc.). EPU operation requires modifications to the feedwater pumps and upstream heater drain pumps, both of which will contribute to an increase in the operating pressure of all feedwater lines on the discharge side of the pumps. Piping will be rerated to 2,250 psig to accommodate the pump modifications described above. A conservative component level assessment was performed based on the combined shutoff head of the condensate, condensate booster, and feedwater pumps. The results of this evaluation confirm the acceptability of these valves to operate at the rerated design pressure. Therefore the AOVs will continue to operate reliably and require no physical changes as a result of EPU.

Conclusion

NMPNS has reviewed the functional performance of safety-related valves and pumps and concludes that the effects of the proposed EPU on safety-related pumps and valves have been adequately addressed. NMPNS further reviewed the effects of the proposed EPU on its MOV programs related to GL 89-10, GL 96-05, and GL 95-07, and the lessons learned from those programs to other safety-related, power-operated valves. Based on this, NMPNS concludes that the safety-related valves and pumps will continue to meet the requirements of GDC1, GDC37, GDC40, GDC43, GDC46, GDC54, and 10 CFR 50.55a(f) following implementation of the proposed EPU. Therefore, NMPNS finds the proposed EPU acceptable with respect to safety-related valves and pumps.

2.2.5 Seismic and Dynamic Qualification of Mechanical and Electrical Equipment

Regulatory Evaluation

Mechanical and electrical equipment covered by this section includes equipment associated with systems that are essential to emergency reactor shutdown, containment isolation, reactor core cooling, and containment and reactor heat removal. Equipment associated with systems essential to preventing significant releases of radioactive materials to the environment are also covered by this section. The NMPNS review focused on the effects of the proposed EPU on the qualification of the equipment to withstand seismic events and the dynamic effects associated pipe-whip and jet impingement forces. The primary input motions due to the safe shutdown earthquake (SSE) are not affected by an EPU. The NRC's acceptance criteria are based on (1) GDC1, 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) GDC30, 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) GDC2, 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) GDC4, 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) GDC14, 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.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of Constant Pressure Power Uprates. Sections 10.1 and 10.3 of the CLTR address the effect of Constant Pressure Power Uprate on the seismic and dynamic qualification of mechanical and electrical equipment. The results of this evaluation are described below.

The effect of dynamic forces (pipe whip and jet impingement) is minimal because there is a negligible effect due to EPU (see Section 2.2.1). As stated above, the primary input motions due to the SSE 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 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 NMP2 Mechanical EQ (MEQ) Program is based upon the response to FSAR question F270.3. The MEQ Program qualification documentation was reviewed based on changes to the radiological conditions and to the long-term portion of the inside primary containment accident temperature profile. The evaluation concluded that the EPU normal and accident radiation dose changes have little or no impact on the radiation tolerance of the mechanical equipment materials. The MEQ program excludes the effects of beta airborne radiation due to the encapsulation and shielding provided by the equipment to the non-metallic parts and materials. The review concluded that there is no change in qualification status for MEQ Program components and this equipment has sufficient life for plant operation past the first outage following EPU implementation.

Conclusion

NMPNS has reviewed the evaluations of the effects of the proposed EPU on the qualification of mechanical and electrical equipment and concludes that the evaluations (1) adequately addressed the effects of the proposed EPU on this equipment and (2) demonstrated that the equipment will continue to meet the requirements of GDCs 1, 2, 4, 14, and 30; 10 CFR Part 100, Appendix A; and 10 CFR Part 50, Appendix B, following implementation of the proposed EPU. Therefore, NMPNS finds the proposed EPU acceptable with respect to the qualification of the mechanical and electrical equipment.

Table 2.2-1 Liquid Line Breaks

	Change Due to EPU			
Break Location	**Total Mass Release (Used for Flooding Evaluation)**	**Mass Release Rate**	**Pressure**	**Temperature**
WCS Line Break in Main Steam Tunnel	Bounded by Feedwater Line Break	Bounded by CLTP	Bounded by Main Steam Line Break	Bounded by Main Steam Line Break
Feedwater Line Break in Main Steam Tunnel (MST)	Bounded by CLTP analysis (release of entire hotwell inventory)	Results in approximately 4% increase in flashing energy. However, resulting MST pressures and temperatures remain bounded by Main Steam Line Break	Bounded By Main Steam Line Break	Bounded By Main Steam Line Break
WCS Line Break in Secondary Containment	Bounded by CLTP	Bounded by CLTP	Bounded by CLTP	Bounded by CLTP

Table 2.2-2a

Percentage Increase In Class 1 Pipe Stresses, Usage Factor, Interface Loads, and Thermal Displacements for NMP2 Piping Systems Due to EPU Conditions

ASME CODE EQUATION NO.	PIPING CATEGORIES						
	A			B			
	Main Steam			Feedwater			
	Flow	PR & T	Total	Flow	Temp	Press	Total
9A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
9B	5.10	0.00	5.10	5.10	N/A	N/A	5.10
9C	5.10	0.00	5.10	5.10	N/A	N/A	5.10
9D	5.10	0.00	5.10	5.10	N/A	N/A	5.10
10	2.64	0.08	2.72	2.64	4.41	1.10	8.15
12	N/A	0.04	Negligible Impact	N/A	6.30	0.05	6.35
13	N/A	0.09	0.09	N/A	2.10	1.46	3.56
14 (1)	1.32	0.26	1.58	1.32	15.90	5.82	23.04
Interface Loads	26.35	0.00	26.35 (2)	26.35	8.21	0.40	34.96
	40.84	0.00	40.84 (3)	Negligible Impact	Negligible Impact	Negligible Impact	Negligible Impact
Thermal Displacement	N/A	0.03	Negligible Impact	N/A	6.30	0.04	6.34

NOTES:
N/A Not affected due to EPU
(1) Fatigue – Cumulative Usage Factor
(2) Interface load factor = 26.35 to be applied to the load combination associated with TSV load case, or
(3) Interface load factor = 40.84 to be applied to the load case alone and then recombine the respective load combination.

Table 2.2-2b

Percent Increase In Class 2 and/or 3 Pipe Stresses, Interface Loads, and Displacements for NMP2 Piping Systems Due To EPU Conditions

ASME CODE EQUATION NO.	PIPING CATEGORIES						
	A			B			
	Main Steam			Feedwater			
	FLOW	PR & T	TOTAL	Flow	TEMP	PRESS	TOTAL
8	N/A	N/A	N/A	N/A	N/A	N/A	N/A
9B	5.10	0.00	5.10	5.10	N/A	N/A	5.1
9C	5.10	0.00	5.10	5.10	N/A	N/A	5.1
9D	5.10	0.00	5.10	5.10	N/A	N/A	5.1
10	N/A	0.04	0.04	N/A	6.30	0.05	6.35
Interface Loads	26.35	0.00	26.35 (2)	26.35	8.21	0.40	34.96
	40.84	0.00	40.84 (3)	Negligible Impact	Negligible Impact	Negligible Impact	Negligible Impact
Thermal Displacement	N/A	0.00	0.00	N/A	6.30	0.04	6.34

Notes:

N/A Not affected due to EPU

(1) Fatigue – Cumulative Usage Factor

(2) Interface load factor = 26.35 to be applied to the load combination associated with TSV load case, or

(3) Interface load factor = 40.84 to be applied to the load case alone and then recombine the respective load combination.

Table 2.2-3a Summary of Main Steam ASME Class 1 Piping, Pipe Stresses, and Cumulative Usage Factor (CUF)

MSS Loop A Inside Containment - AX-002AY Rev 02 Disp 02C pg 5

	Equation	Case	Node	Element	Original Max Calc Stress (psi)	Factor	EPU Stress (psi)	Allowable (psi)	Ratio	
	9	N/U	33	Elbow	23,957	5.1	25179	30,491	0.826	
		Emerg.	114	Tee	32,368	5.1	34019	45,736	0.744	
		Faulted	114	Tee	36,796	5.1	38673	60,981	0.634	
NOT In Break Exclusion Area	10		33	Elbow	52,209	2.72	53629	60,981	0.879	
	12		33	Elbow	27,697	0.00	27697	60,981	0.454	
	13		380	Branch	35,579	0.09	35611	60,981	0.584	
	Fatigue		8	Ahand	0.0116	1.58	0.0118	< 1.0	OK	
In Break Exclusion Area	10		51	Ahand	67,598	2.72	69,437	48,785	1.423	Original Stress Qualified by Eqn 12 and 13
	12		114	Tee	16,966	0.00	16,966	48,785	0.348	
	13		114	Tee	45,390	0.09	45,431	48,785	0.931	
	Fatigue		51	Ahand	0.0898	1.58	0.0912	<.1	OK	

MSS Loop A A/SVV Sweepolets- AX-002AY Rev 02 Disp 02C pg 6

	Equation	Case	Node	Element	Original Max Calc Stress (psi)	Factor	EPU Stress (psi)	Allowable (psi)	Ratio	
	9	N/U	25	Tee	25,738	5.1	27051	27,492	0.984	
		Emerg.	25	Tee	36,466	5.1	38326	41,238	0.929	
		Faulted	25	Tee	37,018	5.1	38906	57,984	0.671	
In Break Exclusion Area	10		503	Taper	60,225	2.72	61863	43,987	1.406	Original Stress Qualified by Eqn 12 and 13
	12		31	Tee	15,983	0.00	15983	43,987	0.363	
	13		503	Taper	42,722	0.09	42760	43,987	0.972	
	Fatigue		203	Taper	0.0376	1.58	0.0382	<.1	OK	

Note: Qualified by Equations 12 and 13 from ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, 1974 Edition

Table 2.2-3b Main Steam Class 1 Piping Outside Primary Containment Stress Summary

MSS Piping Outside Containment - AX-002EY, Rev 02, pg 102

	Equation	Case	Node	Element	Original Max Calc Stress (psi)	EPU Stress (psi)	Allowable (psi)	Ratio	
	9	N/U	7	Grun	18,705	19,659.0	27,030	0.727	
		Emerg.	7	Grun	18,731	19,686.3	40,545	0.486	
		Faulted	7	Grun	20,572	21,621.2	54,060	0.400	
In Break Exclusion Area	10		28	Valve	66,453	68,260.5	43,248	1.578	Note
	12		469	Grun	3,910	3,910.0	43,248	0.090	
	13		28	Valve	38,601	38,635.7	43,248	0.893	
	Fatigue		28	Valve	0.0566	0.0575	<1	OK	

AX-139AY, Rev 0, Attachment 4 pages 1 - 8

	Equation	Case	Node	Element	Original Max Calc Stress (psi)	EPU Stress (psi)	Allowable (psi)	Ratio	
	9	N/U	150	Grun	12,877	13,533.7	27,102	0.499	
		Emerg.	150	Grun	14,658	15,405.6	40,653	0.379	
		Faulted	150	Grun	15,255	16,033.0	54,204	0.296	
In Break Exclusion Area	10		4000	Srun	22,130	22,731.9	43,363	0.524	
	12		4000	Srun	725	725.0	43,363	0.017	
	13		4000	Srun	15,939	15,953.3	43,363	0.368	
	Fatigue		7000	Srun	0.0081	0.0082	<.1	OK	
NOT In Break Exclusion Area	10		125	Taper Transition Jt.	58,712	60,309.0	54,204	1.113	Note
	12		145	Taper Transition Jt.	5,131	5,131.0	54,204	0.095	
	13		125	Taper Transition Jt.	22,173	22,193.0	54,204	0.409	
	Fatigue		125	Taper Transition Jt.	0.0076	0.0077	< 1.0	OK	

Note: Qualified by Equations 12 and 13 from ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, 1974 Edition

Table 2.2-3c Main Steam Penetration Stress Summary

Calculation: EM3.342Y, Disp 01D, pg 4
Title: Stress Analysis of Piping Penetration Z-1A

Equation	Design Condidtion	Out F1 psi	Out F2 psi	Out F3 psi	Out F4 psi	Out F5 psi	EPU Out F1 psi	EPU Out F2 psi	EPU Out F3 psi	EPU Out F4 psi	EPU Out F5 psi	Allowable psi
9	Design	14,581	6,378	11,908	13,742	3,159	14,581	6,378	11,908	13,742	3,159	26,550
9	Emergency	14,624	6,397	12,496	15,245	3,666	15,370	6,723	13,133	16,022	3,853	39,825
9	Faulted 1	13,905	6,015	19,735	36,840	14,603	14,614	6,322	20,741	38,719	15,348	40,653
(Ins. Rupture)	Faulted 2	13,846	5,990	19,691	36,732	14,516	14,552	6,295	20,695	38,605	15,256	40,653
9	Faulted 1	39,068	16,232	10,846	13,756	14,994	41,060	17,060	11,399	14,458	15,759	40,653
(Out. Rupture)	Faulted 2	39,060	16,228	10,456	12,825	14,908	41,052	17,056	10,989	13,479	15,668	40,653
10	Normal/upset 1	38,941	29,639	46,432	33,578	28,338	40,000	30,445	47,695	34,491	29,109	42,480
12				2,825			0		2,825			42,480
13				39,462			0		39,498			42,480
14	Fatigue (CUF)	0.0400	0.0400	0.0400	0.0400	0.0400	0.0406	0.0406	0.0406	0.0406	0.0406	0.1

SUMMARY		Out	Max EPU Stress	Allowable	Ratio	Ok?	Notes
9	Design	F1	14,581	26,550	0.5492	OK	
9	Emergency	F4	16,022	39,825	0.4023	OK	
9	Faulted	F1	39,064	40,653	0.9609	OK	
10		F3	47,695	42,480	1.1228	Not OK	Note
12		F3	2,825	42,480	0.0665	OK	
13		F3	39,498	42,480	0.9298	OK	
14		F3	0.0406	0.1		OK	

Note: Qualified by Equations 12 and 13 from ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, 1974 Edition

Table 2.2-3d Main Steam Nozzle Loading Summary

RPV Nozzle N3A Loads - AX-002AY Rev 02 Disp 02A pg 3 and 02C pg 11

Loading Condition	K (ft-lbs)	H (lbs) CLTP	H (lbs) EPU	H allowable	Ratio	M (ft-lbs) CLTP	M (ft-lbs) EPU	M allow.	Ratio
Deadweight	44668	11,767		110,000	**0.1070**	45,159		254,167	**0.1777**
Thermal	216989	19,641		509,000	**0.0386**	232,538		1,058,333	**0.2197**
OBEA	44258	10,060		237,000	**0.0424**	46,684		355,833	**0.1312**
OBEI, OCCU	65114	14,512	18,336	237,000	**0.0774**	67,966	85,876	355,833	**0.2413**
OBEI, OCCE	150806	33,631	42,493	474,000	**0.0896**	156,932	198,283	711,666	**0.2786**
SSEI, OCCF	152710	34,032	48,150	474,000	**0.1016**	160,848	231,001	711,666	**0.3246**

Table 2.2-3e Main Steam Relief Valve Flange Summary

Worst Case Flange Loadings Loop A

Description	Node Pt.	Load Level	Direction	Current Loads	Factor	EPU Loads	Allowable	Ratio	
2SVV*PSV123 Bottom Flange	504	B	Mt	30644	26.35	38719	115469	0.34	OK
2SVV*PSV123 Bottom Flange	504	D	Mt	56306	26.35	83403	115469	0.72	OK
2SVV*PSV123 Top Flange	510	B	Mt	11873	26.35	15002	43301	0.35	OK
2SVV*PSV123 TopFlange	510	D	Mt	16521	26.35	22497	43301	0.52	OK

Table 2.2-3f Main Steam Pipe Support Load Summary

Calculation: Z2-0086-4								
Description	Node Pt.	Load Condition	Current Loads (lbs)	Factor	EPU Loads (lbs)	Allowable (lbs)	Ratio	OK?
PSSP330A1 Snubber	34	Normal/Upset	32,244	27.00	32,449	32,683	0.993	OK
		Emergency	33,960	27.00	34,155	45,283	0.754	OK
		Faulted	45,567	27.00	48,085	54,340	0.885	OK

Calculation: Z2-0089-5								
Description	Node Pt.	Load Condition	Current Loads (lbs)	Factor	EPU Loads (lbs)	Allowable (lbs)	Ratio	OK?
PSSP337A1 Snubber	9	Normal/Upset	13,722	27.00	14,521	21,287	0.682	OK
		Emergency	22,310	27.00	22,805	25,260	0.903	OK
		Faulted	22,621	27.00	23,109	30,312	0.762	OK

Calculation: Z2-0169-4								
Description	Node Pt.	Load Condition	Current Loads (lbs)	Factor	EPU Loads (lbs)	Max Spt. Load (lbs)	Support Ratio	OK?
PSSP335A1 Snubber	24	Normal/Upset	20,333	27.00	20,440	21,760	0.939	OK
		Emergency	24,366	27.00	24,455	25,542	0.957	OK
		Faulted	25,748	27.00	30,981	32,945	0.940	OK

Table 2.2-4a Feedwater ASME Class 1 Piping Stress Summary and Cumulative Usage Factor (CUF)

Primary Containment North Loop -- AX-017B-07 pg. 88

Equation	Case	Node	Element	Max Calc Stress (psi)	Factor	EPU Stress (psi)	Allowable (psi)	Ratio	
9	N/U	20	Elbow	23,632	5.1	24,837	26,598	0.934	
	Emerg.	20	Elbow	24,321	5.1	25,561	39,897	0.641	
	Faulted	20	Elbow	28,584	5.1	30,042	53,196	0.565	
10		565	Elbow	63,605	8.15	68,789	42,557	1.616	Original Stress Qualified by Eqn 12 and 13
12		565	Elbow	36,354	6.35	38,662	42,557	0.908	
13		20	Elbow	29,947	3.56	31,013	42,557	0.729	
Fatigue		580	Ahand	0.0306	23.04	0.0377	<0.1	OK	

Note: Qualified by Equations 12 and 13 from ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, 1974 Edition

Table 2.2-4b Feedwater Penetration Stress Summary

Calculation: EM3.334 Rev 02, Disp 02E, pg 4
Title: Stress Analysis of Piping Penetration Z-4A

Equation	Design Condidtion	Cut F1 psi	Cut F2 psi	Cut F3 psi	Cut F4 psi	Cut F5 psi	EPU Cut F1 psi	EPU Cut F2 psi	EPU Cut F3 psi	EPU Cut F4 psi	EPU Cut F5 psi	Allowable psi
9	Design	12,805	10,322	7,616	10,746	3,437	13,458	10,848	8,004	11,294	3,612	40,050
9	Emergency	9,493	7,391	5,542	8,209	4,232	9,977	7,768	5,825	8,628	4,448	60,075
9	Faulted 1	8,903	6,713	13,405	28,435	13,856	9,357	7,055	14,089	29,885	14,563	60,075
(Ins. Rupture)	Faulted 2	8,413	6,391	13,367	28,380	13,586	8,842	6,717	14,049	29,827	14,279	60,075
9	Faulted 1	34,147	21,600	4,731	7,235	18,301	35,888	22,702	4,972	7,604	19,234	60,075
(Out. Rupture)	Faulted 2	34,081	21,553	4,381	6,605	18,100	35,819	22,652	4,604	6,942	19,023	60,075
10	Normal/upset 1	82,497	56,482	58,281	79,504	26,657	89,221	61,085	63,031	85,984	28,830	64,080
12		42,704			43,412		45,416			46,169		64,080
13		40,210			36,774		41,641			38,083		64,080
14	Fatigue (CUF)	0.0481	0.0481	0.0481	0.0481	0.0481	0.0592	0.0592	0.0592	0.0592	0.0592	0.1

SUMMARY		Cut	Max EPU Stress	Allowable	Ratio	Ok?	Notes
9	Design	F1	13,458	40,050	0.33603	OK	
9	Emergency	F1	9,977	60,075	0.16608	OK	
9	Faulted	F1	35,888	60,075	0.59739	OK	
10		F1	89,221	64,080	1.39233	Not OK	Note
12		F1	45,416	64,080	0.70873	OK	
13		F1	41,641	64,080	0.64984	OK	
14		F1	0.0592	0.1		OK	

Note: Qualified by Equations 12 and 13 from ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, 1974 Edition

Table 2.2-4c Feedwater Nozzle Stress Summary

Reactor Nozzle 30deg Azimuth - AX-017B-07 pg. 123

	Fx (lb)	Fy (lb)	Fz (lb)	Mx (ft-lb)	My (ft-lb)	Mz (ft-lb)	EPU K (ft-lb)	EPU H (lb)	H_{allow}	H_{ratio}	EPU M (ft-lb)	M_{allow}	M_{ratio}
			CLTP Loadings										
(OBEI, OCCU)	2776	2598	6652	1836	23326	14024							
(OBEI, OCCE)	4797	4382	10678	2589	36857	20280							
(SSEI, OCCF)	5181	4738	11414	2915	39514	22614							
			*EPU Loadings**										
(OBEI, OCCU)	3746	3506	8978	2478	31481	18927	36732	9638	27400	**0.3517**	37667	41167	**0.9150**
(OBEI, OCCE)	6474	5914	14411	3494	49742	27370	56775	15577	54800	**0.2843**	58371	82334	**0.7090**
(SSEI, OCCF)	8223	6670	19375	4028	67831	32066	75028	20491	54800	**0.3739**	77042	82334	**0.9357**

* Factor is 34.96 for all Feedwater Interface Loads. It was applied to the loads in each direction, then combined.

Table 2.2-4d Feedwater Pipe Support Load Summary

Calculation: Z17-0065-2
Disposition: 02A

Support Number	Node Pt.	ASME Load Level	Loading (lbs.)	EPU Load (lbs.)	Max. Structural Load (lbs)	Structure Ratio	Results	Notes
PSSP183A1	157	Level A & B	18,050	18,431	29,255	0.63	OK	
		Level C	23,792	24,082	31,686	0.76	OK	
		Level D	34,171	40,075	45,539	0.88	OK	

Calculation: Z17-0275-0
Disposition: 00A

Support Number	Node Pt.	ASME Load Level	Loading (lbs.)	EPU Load (lbs.)	Capacity (lbs.)	Ratio= Max/Capacity	Results	notes
PSSP185A1	180	Level A & B	17,671	18,113	21,704	0.835	OK	case 4
		Level C	24,659	24,978	26,542	0.941	OK	
		Level D	30,658	33,668	36,383	0.925	OK	

Table 2.2-5a BOP Piping Main Steam System (Outside Containment)

Maximum Stress for Class 4 Piping in AX-002EY							
Criteria Per ASME III NB 3600		Node	Element Type	CLTP	EPU	Allowable	Ratio
EQ. 11	Deadweight	236	Pipe Run	8310	8310	15000	0.55
EQ. 12	Normal/Upset	NA*	Pipe Run	16793	17400	18000	0.97
EQ. 12	Emergency	NA*	Pipe Run	16799	17400	27000	0.64
EQ. 12	Faulted	1600	Pipe Run	20962	21100	36000	0.59
EQ. 13	Thermal Expansion	209	Elbow	13144	13144	22500	0.58
EQ. 14	Deadload + Thermal Expansion	1600	Pipe Run	19897	19897	37500	0.53

* Max stress originally occurred at node 1600, but EPU max stress occurs at node 125.

Maximum pipe stress increase:	
Temperature Expansion	No change
Pressure	No change
Fluid Transients (Between RPV & TSV's)	27%
Fluid Transients (Between TSV's & Turbine)	38%
Maximum pipe support loading:	
EPU (due to thermal expansion loading)	No change
EPU (due to FT loads between RPV & TSV's)	27%
EPU (due to FT loads between TSV's & turbine)	38%

Table 2.2-5b BOP Piping Feedwater System (Outside Containment)

Maximum Stress for Class 4 Piping in AX-017C							
Criteria Per ASME III NB 3600		Node	Element Type	CLTP	EPU	Allowable	Ratio
EQ. 11	Deadweight	173	Pipe Run	9325	9471†	15000	0.63
EQ. 12	Normal/Upset	175	Pipe Run	13237	13383†	18000	0.74
EQ. 12	Emergency	175	Pipe Run	13323	13469†	27000	0.50
EQ. 12	Faulted	175	Pipe Run	14947	15093†	36000	0.42
EQ. 13	Thermal Expansion	100	Pipe Run	30784	30784*	22500	1.37¥
EQ. 14	Deadload + Thermal Expansion	100	Pipe Run	37245	37245*	37500	0.99

* Thermal Stratification is the dominating thermal load case. According to Calc. PX-01752, there is more than 15% conservatism in the thermal stratification calculation for this piping. Therefore, the design basis analysis envelopes EPU loads.

† Based on a 50 psi increase on a 24 inch diameter pipe with 2.062 inch wall thickness.

¥ Qualified by EQ. 14

Maximum pipe stress increase:	
Temperature expansion	5.9%
Pressure	2.3%
Fluid Transients	0%
Maximum pipe support loading increase (due to thermal expansion loading):	5.9%

Table 2.2-5c BOP Piping

Extraction Steam

Maximum pipe stress increase:	
Temperature expansion	5.1%
Pressure	12.5%
Fluid Transients	0%
Maximum pipe support loading increase (due to thermal expansion loading):	5.1%

Table 2.2-5d BOP Piping

FW Heater Drains & Vents (HDL and HDH)

Maximum pipe stress increase:	
Temperature expansion	6.1%
Pressure	12.5%
Fluid Transients	0%
Maximum pipe support loading increase (due to thermal expansion loading):	6.1%

Table 2.2-5e BOP Piping

Condensate

Maximum pipe stress increase:	
Temperature expansion	17.6%
Pressure	0%
Fluid Transients	0%
Maximum pipe support loading increase (due to thermal expansion loading):	17.6%

Table 2.2-5f BOP Piping

Moisture Separator Reheater Vents and Drains (DSM and DSR)

Maximum pipe stress increase:	
Temperature expansion	5.0%
Pressure	12.5%
Fluid Transients	15.4%
Maximum pipe support loading increase:	
EPU (due to thermal expansion loading)	5.0%
EPU (due to fluid transient loading)	15.4%

Table 2.2-5g BOP Piping

Auxiliary Condensate

Maximum pipe stress increase:	
Temperature expansion	6.2%
Pressure	5.6%
Fluid Transients	0%
Maximum pipe support loading increase (due to thermal expansion loading):	6.2%

Table 2.2-6 CUFs and S_{p+q} Values of Limiting Components

	P + Q Stress (ksi)			CUF [4]		
Component[1]	**Current (3467 MWt)**	**EPU (4068 MWt)[3]**	**Allowable (ASME Code Limit)**	**Current (3467 MWt)**	**EPU (4068 MWt)[3]**	**Allowable**
Feedwater Nozzle - Carbon Steel Replacement Safe End	63.0/ 16.6[2]	77.9/ 16.8[2]	54.3	0.965	0.6537[5]	1.0
Feedwater Nozzle - Stainless Steel Clad Replacement Safe End	69.2/ 16.0[2]	91.0/ 17.4[2]	50.7	0.916	0.8299[5]	1.0
Steam Outlet Nozzle – Nozzle to Shell Junction	32.49	39.23	80.1	0.540	0.868	1.0
Steam Outlet Nozzle – Carbon Steel Safe End	54.8/ 51.4[2]	60.3/ 53.7[2]	54.3	0.025	0.094	1.0

Notes:

1. Only components with usage factors greater than 0.33 and experiencing an increase in flow, temperature, reactor internal pressure differences or other mechanical loads are shown.
2. Thermal Bending included/Thermal bending removed. P + Q stresses are acceptable per CLTP elastic-plastic analysis, which is valid for EPU conditions.
3. EPU was conservatively evaluated for 102% of EPU (3988 MWt * 1.02).
4. Only the limiting CUF is presented.
5. Values listed for the Feedwater Nozzle are for 40-year license. For 60-year license the calculated CUF value exceeds 1.0. Fatigue monitoring of the Feedwater Nozzle using the FatiguePro fatigue monitoring software is anticipated to be adequate to maintain the margin below 1.0 for 60-year license. If fatigue-monitoring trending predicts that fatigue usage cannot be maintained below 1.0, then corrective actions such as re-analysis, enhanced inspection, or repair/replacement will be implemented.

Table 2.2-7 RIPDs for Normal Conditions

Parameter	CLTP [1] (psid)	EPU [1] (psid)
Shroud Support Ring and Lower Shroud	29.03	31.45
Core Plate and Guide Tube	21.68	22.84
Upper Shroud	7.65	8.89
Shroud Head	8.48	9.53
Shroud Head to Water Level (Irreversible [2])	10.96	12.39
Shroud Head to Water Level (Elevation [2])	0.90	0.80
Top Guide	0.96	0.96
Steam Dryer	0.36	0.48
Fuel Channel Wall	12.45	13.76

Notes:

1. 105% core flow.
2. 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 Upset Conditions

Parameter	**CLTP [1] (psid)**	**EPU [1] (psid)**
Shroud Support Ring and Lower Shroud	31.43	33.85
Core Plate and Guide Tube	< 23.0	< 24.5
Upper Shroud	11.47	13.34
Shroud Head	12.71	14.30
Shroud Head to Water Level (Irreversible [2])	16.44	18.58
Shroud Head to Water Level (Elevation [2])	1.35	1.20
Top Guide	1.11	1.11
Steam Dryer	0.47	0.71
Fuel Channel Wall	< 14.9	< 15.9

Notes:

1. 105% core flow.
2. 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 RIPDs for Faulted Conditions

Parameter	CLTP [1] (psid)	EPU [1] (psid)
Shroud Support Ring and Lower Shroud	46	47
Core Plate and Guide Tube	27.0	27.0
Upper Shroud	26.5	27.0
Shroud Head	27.0	27.0
Shroud Head to Water Level (Irreversible [2])	28.5	28.5
Shroud Head to Water Level (Elevation [2])	2.0	2.0
Top Guide	2.5	2.7
Steam Dryer [3]	6.3	4.8
Fuel Channel Wall	14.9	15.9

Notes:

1. Values are the maximum results from either the cavitation interlock power with 105 % or 110 % core flow or the high power with 105 % core flow points.
2. 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.
3. The steam dryer ΔP was calculated conservatively for CLTP whereas the steam dryer ΔP was calculated more realistically for EPU.

Table 2.2-10 Governing Stress Results for RPV Internal Components

No	Component	CLTP			EPU					
		Service Level	Unit	Value	Location	Service Level	Stress Category /Other	Unit	Value[2]	Allowable
1	Shroud	B	psi	10,814	Top Guide Wedge	B	P_m+P_b	psi	14,707	21,450
2	Shroud Support	B	psi	28,030	Legs	B	P_m+P_b	psi	28,030	28,125
3	Core Plate	B	psi	18,550	Ligament	B	P_m+P_b	psi	20,250	25,350
4	Top Guide	B	psi	23,724	Longest Beam	B	P_m+P_b	psi	23,724	25,350
5.a	CRD Housing	B	psi	13,993	CRD Housing @ RPV Penetration	B	P_m	psi	13,993	16,600
5.b	CRD Housing	B	psi	11,594	CRD Housing @ RPV Stub Tube	B	P_m	psi	12,217	16,600
6.a	Control Rod Guide Tube	B	psi	15,056	CRGT Base	B	P_m+P_b	psi	16,562	24,000
6.b	Control Rod Guide Tube	See Note 3			Mid-Span	B	Buckling Criteria	N/A	0.436	0.45
7	Orificed Fuel Support	B	lbs	2,109[1]	OFS Body	B	P_m+P_b	psi	12,162	15,580
8	Fuel Channel				Qualified By GNF proprietary method					
9	Shroud Head and Separators Assembly (Incl. Shroud Head Bolts)	B	psi	7,980	Shroud Head Bolt Tee Bar	B	Bearing	psi	13,466	18,800
10.a	Jet Pump	B	psi	12,649	Riser Brace	D	P_m+P_b	psi	40,602	60,840
10.b		D	Lbs	17,800	Beam Bolt	D	Load	lbs	17,729	20,825
11.a	Access Hole Cover Flat Plate	See Note 4			Rim	B	P_m+P_b	psi	7,093	34,950

No	Component	CLTP			EPU					
		Service Level	Unit	Value	Location	Service Level	Stress Category /Other	Unit	Value[2]	Allowable
11.b	Access Hole Cover Top Hat				Ring	B	P_m+P_b	psi	13,188	20,580
12.a	Core Spray Line	B	psi	15,150	Elbow	B	P_m+P_b	psi	15,150	20,920
12.b	Core Spray Sparger	B	psi	6,560	Tee Junction	B	P_m	psi	6,560	21,450
13	Feedwater Sparger	See Note 5			Header Pipe to Spray Nozzle Adapter Weld	A, B	See Table 2.2-11 for fatigue usage factor			
14	In-Core Housing and Guide Tube	B	psi	16,575	In-Core Housing @ RPV Penetration	B	P_m	psi	<16,575	16,660
15	Core Differential Pressure Line	B	psi	46,319	Unknown	B	P_m+P_b+Q	psi	46,319	49,200
16	LPCI Coupling	C	psi	22,320	Support Ring	C	P_m+P_b	psi	27,677	38,025
17	Steam Dryer	D	kips	75.18	Lifting Rod	D	Buckling	kips	75.18	88.99

P_m = Primary Membrane Stress Intensity, P_b = Primary Bending Stress Intensity, Q = Secondary Membrane Plus Bending Stress Intensity

Notes given after Table 2.2-11

Table 2.2-11 Fatigue Usage for RPV Internal Components for Plant Life of 40-Years and 60-Years

No.	Component	CLTP Value of CUF	CUF 40-yr Life	CUF 60-yr Life	Allowable	See Note
1	Shroud	0.43	0.507	0.80	1.0	
2	Shroud Support	0.053	0.053	0.0795	1.0	
3.a	Core Plate – Rim Junction	0.005	0.005	0.008	1.0	
3.b	Core Plate Studs	0.930	See Note 6		1.0	
4	Top Guide	0.169	0.169	0.254	1.0	
5	CRD Housing	See Note 7			1.0	
6	Control Rod Guide Tube	See Note 8			1.0	
7	Orificed Fuel Support	0.047	0.047	0.071	1.0	
8	Shroud Head and Separators Assembly (Incl. Shroud Head Bolts)	0.049	0.0576	0.190	1.0	
9	Jet Pump (Riser Brace)	0.67	0.441	0.662	1.0	
10	Access Hole Cover – Flat Plate		0.004	0.006	1.0	See Note 5
11	Access Hole Cover- Top Hat		0.330	0.495	1.0	
12.a	Core Spray Line	0.167	0.167	0.250	1.0	
12.b	Core Spray Sparger	0.20	0.20	0.30	1.0	
13	Feedwater Sparger		0.32	0.48	1.0	See Note 6
14	In-Core Housing and Guide Tube		See Note 9		1.0	
15	Core Differential Pressure Line	0.02	0.02	0.03	1.0	
16	LPCI Coupling		See Note 10		1.0	
17	Steam Dryer		See Note 11		1.0	

Notes for Tables 2.2-10 and 2.2-11:

1 For the OFS, the corresponding allowable value is 4,495 lbs. Hence, the Calculated to Allowable value ratio is 0.47.

2 Stresses reported are for the limiting loading condition, with the least margin of safety.

3 CLTP evaluation of CRGT for buckling was based on generic GEH CRGT Stress analysis report 22A4678, Revision 4, June 1976 (Item 10 of Reference 26). CLTP value will be bounded by the EPU value provided herein.

4 For Access Hole Cover, all applicable CLTP loads remain unaffected or bounded by those for EPU. Hence, the CLTP stress value will be bounded by those for EPU provided herein. The AHC evaluation was based on a stress analysis of an identical design for a similar plant.

5 Fatigue evaluation of the feedwater sparger for CLTP and EPU was based on generic GEH Report 385HA736, Revision 0, December 1976 (Item 11 of Reference 27) and reconciled for NMP2-specific thermal transients. Fatigue Usage Factor for CLTP will be bounded by that for EPU, since the thermal transients remain essentially the same, but the feedwater flow is ~14% less than that for CLTP.

6 The fatigue usage factor of 0.93 for 40-year life was specified in the New Loads Adequacy Summary Report. In order to qualify the core plate bolts for 60-year life, new loads summary report calculations were reviewed to see if any conservatism exists in the calculations. It was found that the calculations were performed with overly conservative assumptions. For example, zero preload in the bolts was assumed as a result of which sliding of core plate and flexing of core plate bolts were predicted to occur under Upset conditions. In reality, the Core plate bolts have adequate pre-load to prevent sliding under Normal and Upset conditions. Therefore, flexing of the bolts is completely prevented. Also, high thermal stress was calculated based on overly conservative assumptions. By removing the conservatisms in the new loads adequacy summary report calculations, it was shown that core plate bolts have negligible (the alternating stress intensity, S_a, would yield an infinite number of allowable fatigue cycles) fatigue usage for 40 and 60 years life.

7 The CRD Housing is primarily subject to mechanical loadings, and thermal/secondary stresses are small. The small magnitude of the alternating stress intensity (S_a) yields an infinite number of allowable fatigue cycles resulting in a negligible fatigue usage factor.

8 The CRGT is primarily subject to mechanical loadings, and thermal/secondary stresses are small. The small magnitude of the S_a yields an infinite number of allowable fatigue cycles.

9 The effect of temperature change in the lower plenum on the thermal stress is deemed to be small compared to the stress margin available for the primary plus secondary stress limit of $3S_m$. The thermal stresses are small (temperature difference through the in-core guide tube wall is small); hence, the fatigue usage factor is deemed to be negligible.

10 The maximum temperature difference in the RPV/shroud annulus region is 3°F for EPU, which is less than 1% relative to CLTP temperature. Hence, the thermal stresses are small. All other applicable loads except the RIPDs remain unaffected; hence, the primary plus

secondary stresses are low compared to the $3S_m$ allowable stress limit. Therefore, fatigue is deemed to be negligible for the LPCI coupling.

11 The primary plus secondary stresses in the dryer are expected to be small compared to the allowable stress limit of $3S_m$ considering the original design basis loads during normal and upset conditions. Hence, the fatigue usage is deemed to be negligible for the dryer given the scope of this evaluation.

Table 2.2-12 Systems with Pumps and Valves in the IST Program

System Designator	IST Pumps	IST Valves	GL 89-10 Valves	AOV Program	System Impacted by EPU
Breathing Air (AAS)		X			
Alternate Decay Heat Removal (ADH)		X			X
Rx Bldg Closed Loop Cooling (CCP)		X	X	X	X
Tb Bldg Closed Loop Cooling (CCS)				X*	X
Cont. Atmos. Monitoring (CMS)		X			X
Containment Purge (CPS)		X		X	X
High Pressure Core Spray (CSH)	X	X	X		X
Low Pressure Core Spray (CSL)	X	X	X		X
Drywell Equipment Drains (DER)		X	X		
Drywell Floor Drains (DFR)		X	X		
EDG Starting Air (EGA)		X			
EDG Fuel Oil (EGF)	X	X			
EDG Lubricating Oil (EGO)	X	X			
EDG Jacket Cooling Water (EGS)	X	X			
Fire Protection Water (FPW)		X			
Feedwater System (FWS)		X	X		X
Feedwater Pump Recirculation (FWR)				X*	X
Instrument Nitrogen (GSN)		X		X*	
Standby Gas Treatment (GTS)		X	X	X	X
Hydrogen Recombiner (HCS)		X	X		X
Control Building HVAC (HVK)	X	X			

System Designator	IST Pumps	IST Valves	GL 89-10 Valves	AOV Program	System Impacted by EPU
C.R. Chilled Water (HVC)			X		
Instrument Air (IAS)		X		X*	
Rx Core Isolation Cooling (ICS)	X	X	X	X	X
Rx Vessel Instrumentation (ISC)		X			X
Containment Leakage Monitoring (LMS)		X			
Main Steam (MSS)		X	X		X
Neutron Monitoring (NMS)		X			
Reactor Recirculation (RCS)		X			X
Control Rod Drive (RDS)		X		X	X
Residual Heat Removal (RHS)	X	X	X	X	X
Service Air (SAS)		X			
Spent Fuel Pool Cooling (SFC)	X	X		X	X
Standby Liquid Control (SLS)	X	X	X		X
MSL SRV Vacuum Breakers (SVV)		X			X
Service Water (SWP)	X	X	X	X	X
Reactor Water Cleanup (WCS)		X	X		X

* Designates Non Safety-Related Components

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, which could result from DBAs. The NMPNS review focused on the effects of the proposed EPU on the environmental conditions that the electrical equipment will be exposed to during normal operation and accidents. The NMPNS review was conducted to ensure that the electrical equipment within the EQ program will continue to be capable of performing its safety functions following implementation of the proposed EPU. The NRC's acceptance criteria for EQ of electrical equipment are based on 10 CFR 50.49, which sets forth requirements for the qualification of electrical equipment important to safety, located in a harsh environment.

Technical Evaluation

All electrical equipment in the EQ Program was evaluated by reviewing the list of components that are identified as being in the electrical EQ program.

The safety-related electrical equipment was reviewed consistent with the requirements of 10 CFR 50.49 and guidance of NRC RG 1.89 to assure the existing qualification for the normal and accident conditions expected in the area where the devices are located remains adequate. Acceptance criteria related to pressure, temperature, relative humidity, and radiation, were used in making this determination.

The EQ Program equipment qualification basis was evaluated using the changes to existing normal and accident radiation doses expected when operating at the EPU increased reactor power level. The normal and post-LOCA radiation dose value changes are based on scaling factors and apply to plant areas inside and outside primary containment. The multiplying factors are for gamma and beta radiation liquid and airborne components and increase or decrease over the post accident time intervals from 0 hour to 100 days. For example, environmental zones located inside and outside primary containment show an increase in the post-LOCA gamma airborne and beta doses for all time intervals, while the gamma liquid and beta liquid 100 day radiation dose component decreases. The net effect of these changes is that zones that have both gamma airborne and liquid dose components show a net decrease in the overall 100 day accident gamma dose because of the larger decrease in the liquid dose, and zones that only have a gamma airborne component show an overall increase in the accident gamma dose that ranges from 16% to 20% over the post-accident time period. For all zones with an airborne beta radiation component, the 100 day total integrated post-accident dose increased by approximately 20%. The cumulative effect on the dose applied to equipment is, therefore, dependent on location, the post-accident operating time and installed equipment configuration (i.e., sealed equipment does not consider beta radiation). Once the impact of post-EPU radiation dose values were determined the equipment was categorized into three groups as follows:

Group I – Not impacted by implementation of EPU. The post-EPU parameters are bounded by the existing qualification levels of the equipment. This group includes equipment with sufficient life to demonstrate radiation qualification through the end of plant life (60 years).

Group II – Partially impacted by implementation of EPU due to postulated higher radiological conditions or plant life extension to 60 years. The EQ equipment is still qualified with reduced radiation life and increased replacement frequency; however, no additional work is required to operate the plant at EPU limits starting May 2012. Table 2.3-1 provides a listing of components which are expected to require disposition in accordance with the NMP2 EQ program prior to end of plant life. For reference purposes, the plant will have operated for approximately 24 years at the time of EPU implementation in spring of 2012.

Group III – Evaluation of the existing equipment in this group indicated that the revised post-accident EPU radiation dose is higher than the currently qualified dose. There are 17 component IDs in this group. Table 2.3-2 provides a listing of this equipment along with component identification information, manufacturer, model, zone/location and current and post-EPU radiation doses. All equipment in this group is located within two EQ zones. Equipment in zones SG261355 and SG261356 are in close proximity to the Standby Gas Treatment filters 1A and 1B.

Shielding will be installed to reduce the post-accident dose to these components enough to extend the qualified life sufficient to meet EQ program requirements. As it is impractical to install shielding around the affected components, shielding will be installed around the two Standby Gas Treatment filters. This shielding will be sufficient to reduce the radiation exposure from the filters to less than that expected under CLTP conditions. The reduction in dose due to the newly installed shielding will be a minimum of 32% which is sufficient to offset the expected increases due to EPU.

Inside Containment

EQ for safety-related electrical equipment located inside the containment is based on main steam line break (MSLB) and/or DBLOCA conditions and their resultant temperature, pressure, humidity, and radiation consequences, and includes the environments expected to exist during normal plant operation. Normal temperatures are essentially unchanged and remain bounded by the normal temperatures used in the EQ analyses. The worst case plant post-LOCA temperature profiles were revised due to EPU changes. These changes occurred at the end of the 100 day profile when the accident temperature has decayed to 210°F, necessitating the evaluation of the new requirements based on the series of individual component type tests available to the EQ Program. However, the revised post-LOCA accident temperatures were determined not to adversely affect the qualification of safety-related electrical and mechanical equipment. The revised worst case inside primary containment and worst case plant EQ enveloping profiles are shown in Figure 2.3-1.

The current radiation levels under normal plant conditions were evaluated to increase in proportion to the increase in RTP. The total integrated doses (normal plus accident) for EPU conditions were determined not to adversely affect qualification of the equipment located inside containment at the time of EPU implementation. As shown in Table 2.3-3, several primary containment EQ zones experience an increase in normal gamma dose rates. The increases are limited to 17% and are highlighted in the table. As indicated in Table 2.3-1, some components within primary containment are expected to exceed EQ limits prior to end of plant life. These components will be addressed within the NMP2 EQ program as warranted. The changes to the NMP2 EQ program brought about by the implementation of EPU will be documented and administered per NMPNS Nuclear Engineering Procedure NMP-CM-7.01-2000, Environmental Qualification Program Administration.

Outside Containment

Accident temperature, pressure, and humidity environments were used for qualification of equipment outside containment resulting from an MSLB, or other HELBs, whichever is limiting for each plant area. The existing plant profiles were extended or pushed forward in time as shown in Figure 2.3-1, necessitating the evaluation of the new requirements based on the series of individual component type tests available to the EQ Program. Based on the evaluations performed and documented under the EPU program, the revised EPU post-accident temperatures do not result in any additional impact to the EQ Program for NMP2. The normal temperature, pressure, and humidity conditions do not change as a result of EPU.

The current radiation levels under normal plant conditions were conservatively evaluated to increase in proportion to the increase in reactor thermal power. The qualification basis for the EQ program equipment was evaluated based on the revised EPU normal and accident radiation dose values. This evaluation used environmental area dose values and equipment-specific dose values where necessary. The evaluation determined that the post-EPU radiation dose changes result in changes in existing equipment qualification status as noted below.

There are 17 component IDs for which the existing qualification is challenged due to EPU radiation values and is not adequate to support equipment operation at EPU limits starting in the spring of 2012. These components are listed in Table 2.3-2. This equipment is located in the Standby Gas Treatment system (SGTS) Building. Equipment in zones SG261355 and SG261356 are in close proximity to the Standby Gas Treatment filters 1A and 1B. Shielding will be installed sufficient to reduce the post-accident dose to these instruments enough to maintain qualification and extend the qualified life.

Shielding will be installed in equipment zones SG261355 and SG261356 to reduce the post-accident dose to these components sufficient to meet EQ program requirements. As it is impractical to install shielding around the affected components, shielding will be installed around the two Standby Gas Treatment filters. This shielding will be sufficient to reduce the radiation exposure from the filters to less than that expected under CLTP conditions. The reduction in dose due to the newly installed shielding will be a minimum of 32%, which is sufficient to offset the expected increases due to EPU.

The environmental qualification of most equipment was not impacted by EPU changes. In some cases the equipment is still qualified, but with a reduced radiation life. However, no additional work is required for plant operation at EPU limits starting in the spring of 2012 beyond that described above.

Conclusion

NMPNS has reviewed the effects of the proposed EPU on the EQ of electrical equipment and concludes that the effects of the proposed EPU on the environmental conditions for and the qualification of electrical equipment have been adequately addressed. NMPNS further concludes that the electrical equipment will continue to meet the relevant requirements of 10 CFR 50.49 following implementation of the proposed EPU. Therefore, NMPNS finds the proposed EPU acceptable with respect to the EQ of electrical equipment.

2.3.2 Offsite Power System

Regulatory Evaluation

The offsite power system includes two or more physically independent circuits capable of operating independently of the onsite standby power sources. The NMPNS review covered the descriptive information, analyses, and referenced documents for the offsite power system and the stability studies for the electrical transmission grid. The NMPNS review focused on whether the loss of the nuclear unit, the largest operating unit on the grid, or the most critical transmission line will result in the loss of offsite power (LOOP) to the plant following implementation of the proposed EPU. The NRC's acceptance criteria for offsite power systems are based on GDC17.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of Constant Pressure Power Uprates. Section 6.1 of the CLTR addresses the effect of Constant Pressure Power Uprate on the AC Power System. The results of this evaluation are described below.

The NMP2 main generator is connected to the main generator transformers. The 345kV switchyard consists of the 345kV generator step-up transformers, buswork, 345kV disconnect switch and the associated control and protection systems. The 115kV offsite power sources originate from Scriba Substation. The 115kV offsite power sources consist of 345/115kV transformers, circuit breakers, disconnect switches, reserve station service transformers and transmission lines. The protective relaying schemes are designed to protect the equipment from electrical faults. Electrical ratings and margins associated with major components of the offsite power system are given in Table 2.3-4.

The existing off-site electrical equipment was determined to be adequate for operation with the uprated electrical output and increased electrical loading. The review concluded the following:

- The Generator Step-Up (GSU) transformer rating is adequate for the increased generator output associated with EPU under normal operations with the Station Service Transformer in service.

- The 345kV Switchyard components (i.e. bus, breakers, switches, current transformers (CTs), and lines) are adequate for increased generator output associated with EPU.
- The 115kV Switchyard components (i.e. bus, breakers, switches, circuit switchers and lines) are adequate for operation under EPU conditions.

The GSU transformer cooling system will be modified by replacing the existing cooling coils with more efficient coolers to increase transformer thermal margin prior to operation at EPU conditions. In addition, the isolated phase bus (IPB) duct will be modified to increase the continuous current rating to provide additional margin for operation at EPU output and reduced voltage.

The existing protective relay settings for the main generator, 115kV equipment and 345kV equipment were determined to be adequate for operation with the EPU electrical output. The review included the protective relaying for the main generator, 345kV step-up transformers, 345kV transmission line and the Scriba Substation relays. The existing protective relay set points were developed and validated based on equipment ratings, which are not being changed for EPU. Therefore, the settings are unaffected by operation at EPU conditions.

Grid studies have been performed, considering the increase in electrical output, to demonstrate conformance to GDC17 (10 CFR 50 Appendix A). Details of these studies are provided in Attachment 8 to the EPU license amendment request, "Grid Stability Evaluation Summary." The analysis has determined that the power uprate will not adversely impact bulk power transmission system steady-state power flow (thermal ratings and voltage), stability, short circuit duty or power transfer levels. Grid events analyzed included loss of the largest generator, loss of NMP2 and loss of the most critical transmission line due to fault with the unit operating at full power uprate capacity. Pre-event line outages were also considered. Stability simulations were transiently stable and exhibited positive damping with the power uprate. NMP2 offsite power steady state and transient voltages resulting from critical transmission line faults or loss of NMP2 generation are adequate to operate loads required for safe shutdown and will preclude the inadvertent separation from the offsite supply. Reactive power will be maintained within acceptable limits analyzed in the grid studies. This will be accomplished by utilizing the existing generator-exciter control system and governed by operational procedures as described in Section 2.5.1.2.2.

Conclusion

NMPNS has reviewed the assessment of the effects of the proposed EPU on the offsite power system and concludes that the offsite power system will continue to meet the requirements of GDC17 following implementation of the proposed EPU. Adequate physical and electrical separation exists and the offsite power system has the capacity and capability to supply power to all safety loads and other required equipment. NMPNS further concludes that the effect of the proposed EPU on grid stability is insignificant. Therefore, NMPNS finds the proposed EPU acceptable with respect to the offsite power system.

2.3.3 AC Onsite 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 NMPNS review covered the descriptive information, analyses, and referenced documents for the AC onsite power system.

The NRC's acceptance criteria for the AC onsite power system are based on GDC17, insofar as it requires the system to have the capacity and capability to perform its intended functions during anticipated operational occurrences and accident conditions.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of Constant Pressure Power Uprates. Section 6.1 of the CLTR addresses the effect of Constant Pressure Power Uprate on the AC Power System. The results of this evaluation are described below.

The NMP2 on-site power distribution system consists of transformers, numerous buses, and switchgear. AC power to the distribution system can be supplied from the transmission system, main generator and from onsite diesel generators.

The onsite AC power system consists of equipment and systems required to provide AC power to unit auxiliaries and service loads under all conditions of plant operation. This includes 13.8kV switchgears, 4.16kV switchgears, 600V load centers and motor control centers, 120VAC distribution panels, uninterruptible power supply (UPS) systems, and standby diesel generators.

The on-site AC power distribution system loads were reviewed under both normal and emergency operating scenarios. In both cases, loads are computed based on equipment nameplate data or brake horse power (BHP), with conservative demand factors applied. These loads are used as inputs for the computation of load, voltage drop and short circuit current values in electrical analysis software called Electrical Transient Analysis Program (ETAP). The significant change in electrical load demand is associated with main power generation system pump motors: Feedwater, Condensate, and Heater Drain pumps. For EPU, the Reactor Recirculation pump motor horsepower was re-analyzed at the nameplate value; however, the actual change in load demand at rated conditions for EPU will increase by less that 1% of the original analyzed value. The NMPNS review covered the AC power components with respect to their functional performance as affected by various configurations and loading conditions including full operation and unit trip with LOCA. The NMPNS review focused on the additional electric load that would result from the proposed EPU.

Operation at the EPU power level is achieved in both normal and emergency conditions by operating equipment at or below the nameplate ratings. Table 2.3-5 provides a summary of the loading changes to the onsite power analysis model due to EPU operation.

Load flow, voltage drop and short circuit current calculations were performed to verify the adequacy of the on-site AC system for the proposed changes. Analyzed EPU BHP loads as discussed above are within the electrical distribution equipment capabilities (i.e., Normal Station Service Transformer, Reserve Station Service Transformers, Aux. Transformers, Bus ducts, etc.). The running and starting voltages for motors are within the acceptable values (i.e., 90% running and 80% starting voltages).

The existing protective relay settings, except feedwater pump and new heater drain pump motor protection, are adequate to accommodate the increased load on the non-safety 13.8kV and 4.16kV systems. Selective coordination is maintained between the pump motor breakers and 13.8kV and 4.16kV switchgear main feeder breakers. The existing protective relay settings for feedwater pump motors were based on the original feedwater pump motors with a 14,100 HP rating. In 2004, feedwater pump motors were replaced with 16,500 HP motors to accommodate future plant power uprates; however, cables and protective relays were not changed. Since the feedwater pump motor BHP required during EPU operation is approximately 16,064 HP; the feedwater motor feed cables will be replaced with larger cables. This will require changes to overcurrent relay settings and instrument meter ranges prior to EPU operation. The existing 1,500 HP heater drains pump motors will be replaced with 1,750 HP motors and this will require changes to overcurrent relay settings. Two additional 13.8 kV breakers will be added to support a modification to add swing bus capability for operational flexibility for feedwater pumps 2FWS-P1A and 1B.

No increase in flow or pressure is required of any AC-powered ECCS equipment for EPU. Therefore, the amount of power required to perform safety-related functions (pump and valve loads) is not increased with EPU. The increased load on each reserve station service transformer (RSST) 13.8kV winding due to the increase in power generation pump load has an insignificant effect on the safety related buses supplied from the 4.16kV winding of the RSST. Starting and running voltages are impacted by < 0.1%, and the existing degraded voltage relay settings are acceptable for EPU. Therefore, the current emergency power system remains adequate. The electrical systems have sufficient capacity to support all required loads for safe shutdown, to maintain a safe shutdown condition, and to operate the engineered safety feature equipment following postulated accidents.

Conclusion

NMPNS has reviewed the effects of the proposed EPU on the onsite AC power system and concludes the effects of the proposed EPU on the system's functional design have been adequately evaluated. NMPNS further concludes that the AC onsite power system will continue to meet the requirements of GDC17 following implementation of the proposed EPU. Therefore, NMPNS finds the proposed EPU acceptable with respect to the AC onsite power system.

2.3.4 DC Onsite Power System

Regulatory Evaluation

The direct current (DC) onsite power system includes the DC power sources and their distribution and auxiliary supporting systems that are provided to supply motive or control power to safety-related equipment. The NMPNS review covered the information, analyses, and referenced documents for the DC onsite power system. The NRC's acceptance criteria for the DC onsite power system are based on GDC17, insofar as it requires the system to have the capacity and capability to perform its intended functions during anticipated operational occurrences and accident conditions.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of Constant Pressure Power Uprates. Section 6.2 of the CLTR addresses the effect of Constant Pressure Power Uprate on DC Power. The results of this evaluation are described below.

The NMP2 direct current (DC) power distribution system provides control and motive power for various systems/components within the plant. The DC power loading requirements in the USAR were reviewed and no reactor power dependent loads were identified.

The DC power system provides DC power to protective relaying, control, instrumentation and other DC loads. This system includes the station batteries, battery chargers and DC distribution system. The DC power system is divided into two distinct categories. The components of the DC system and the loads that are required to safely shutdown the reactor in case of a DBA are designated nuclear safety-related or class 1E; the others are non-safety related or non-class 1E. In both the normal and emergency operating conditions, the DC loads are summarized in the battery sizing calculations based on the equipment ratings. These loads are used as inputs to determine the batteries and battery chargers sizing, voltage drop and short circuit current calculations. There will be small DC load changes in the non-safety related batteries (two additional indication lights on each battery) due to a proposed modification to add swing bus capability for operational flexibility for feedwater pumps 2FWS-P1A and 1B. This DC load addition is negligible compared to the existing margin in both non-safety related batteries and associated battery chargers. The safety-related DC power system is not affected by EPU conditions.

Conclusion

NMPNS has reviewed the assessment of the effects of the proposed EPU on the DC onsite power system and concludes that the effects of the proposed EPU on the system's functional design have been adequately evaluated. NMPNS further concludes that the DC onsite power system will continue to meet the requirements of GDC17 following implementation of the proposed EPU. Adequate physical and electrical separation exists and the system has the capacity and capability to supply power to all safety loads and other required equipment. Therefore, NMPNS finds the proposed EPU 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" (AACS). The review focused on the effect of the proposed EPU on the plant's ability to cope with and recover from an SBO event for the period of time established in the plant's licensing basis. The NRC's acceptance criteria for SBO are based on 10 CFR 50.63.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of Constant Pressure Power Uprates. Section 9.3.2 of the CLTR addresses the effect of Constant Pressure Power Uprate on Station Blackout. The results of this evaluation are described below.

SBO was re-evaluated using the guidelines of NUMARC 87-00, "Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors" (Reference 28), and RG 1.155, Station Blackout (Reference 29).

The major characteristics that affect the ability to cope with a station blackout event are identified in NUMARC 87-00 Rev. 1 as:

1. Condensate inventory for decay heat removal
2. Class 1E battery capacity
3. Compressed gas capacity
4. Effects of loss of ventilation
5. Containment isolation

By satisfying the criteria used in assessing the above characteristics, the plant is able to show satisfactory response to an SBO event.

NUMARC 87-00 Rev. 1 (Section 7) provides two methods for conducting the assessment. The first method, the AC Independent approach is used in the NMP2 SBO assessment.

In the AC Independent approach the plant relies on available process steam, DC power and compressed gas to operate equipment necessary to achieve and maintain hot shutdown.

The four-hour coping duration criteria for AC Independent plants continue to apply to NMP2. Thus, NMP2 must meet the SBO requirements for at least four hours.

Condensate Inventory for Decay Heat Removal

Analyses have shown that the NMP2 condensate inventory is adequate to meet the SBO coping requirement for EPU conditions. The current condensate storage tank (CST) inventory reserve (135,000 gallons) for RCIC use ensures that adequate water volume is available to remove decay heat, depressurize the reactor and maintain reactor vessel level above the top of active fuel (approximately 105,000 gallons required).

Class 1E Battery Capacity

Evaluation of the NMP2 Class 1E Battery Capacity has shown that NMP2 has adequate battery capacity to support decay heat removal during a station blackout for the required coping duration. The battery capacity remains adequate to support RCIC operation after EPU.

Compressed Gas Capacity

NMP2 meets the requirement for compressed gas assessment. An evaluation has shown that the NMP2 air operated SRVs associated with the ADS required for decay heat removal have sufficient compressed gas for the required automatic and manual operation during the SBO event for EPU conditions. Sufficient capacity remains to perform emergency RPV depressurization, if it is required. Adequate compressed gas capacity exists to support the SRV actuations because the maximum number of ADS valve operations is less than the capacity of the ADS pneumatic supply (accumulators).

Effects of Loss of Ventilation

The effect of loss of ventilation in dominant areas of concern containing equipment necessary to achieve and maintain safe shutdown during a station blackout is performed using results from SHEX analysis and NMP2 station operations procedure for SBO.

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.

These areas for NMP2 included:

1. Control Room
2. Switchgear Room
3. Drywell
4. RCIC Room

Containment Isolation

Containment Isolation capability is not adversely affected by SBO event for EPU.

SBO Sequence of events is given in Table 2.3-6. 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. Decay heat was evaluated assuming end-of-cycle (24-month) and GE14 fuel. There are no changes to the systems and equipment used to respond to an SBO, nor is the required coping time of four hours changed.

The battery capacity remains adequate to support RCIC operation after EPU. Adequate compressed gas capacity exists to support the SRV actuations.

The SBO evaluation at EPU conditions shows a need for an additional ~15% over CLTP of Condensate Storage Tank water for RCIC use to ensure that adequate water volume is available to remove decay heat, depressurize the reactor, and maintain reactor vessel level above the top of active fuel. This increases the total Condensate Storage Tank volume required to approximately 105,000 gallons which is well within the current Condensate Storage Tank inventory reserve of 135,000 gallons. For reactor vessel pressure control, one additional SRV manual pressure reduction cycle is required as compared to the CLTP SBO evaluation.

The SBO event calculations for CLTP and EPU conditions were performed using the NRC approved SHEX Computer Program and nominal ANSI/ANS 5.1-1979 Decay Heat Source Term at 100% rated power for Containment Long-Term Pressure and Temperature Analysis (Reference 7).

The key parameters for the SBO calculations for containment response at the CLTP, EPU conditions and design limit are provided in the following table.

Key Containment Parameters Comparison

Parameter	Units	CLTP	EPU	Design Limit
Drywell Pressure	psia	43.9	45.7	<59.7
Suppression Pool (SP) Temperature	°F	176	186.3	212

The containment response comparison is based on a scenario that provides conservative containment parameters as compared to the SBO procedures that are used for reactor pressure control.

Based on the above evaluations, NMP2 continues to meet the requirements of 10 CFR 50.63 after the EPU.

Conclusion

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

Table 2.3-1 Group II Partially Qualified Components

COMPONENT ID	EQ Zone	Manufacturer	Model	Spec. No.	Qualified Radiation Level (rads)	Radiation Life (Years)	60 Year Normal Dose w/EPU (rads)	100 Day Accident Dose at EPU (w/10% Margin) (rads)
2MSS*AOV6A-33-2	PC250618	NAMCO	EA740-80120	P800A	5.00E+07	**14.82**	6.37E+07	3.43E+07
2MSS*AOV6A-33-5	PC250618	NAMCO	EA740-80120	P800A	5.00E+07	**14.82**	6.37E+07	3.43E+07
2MSS*AOV6A-33-6	PC250618	NAMCO	EA740-80120	P800A	5.00E+07	**14.82**	6.37E+07	3.43E+07
2MSS*AOV6B-33-1	PC250618	NAMCO	EA740-80120	P800A	5.00E+07	**14.82**	6.37E+07	3.43E+07
2MSS*AOV6B-33-2	PC250618	NAMCO	EA740-80120	P800A	5.00E+07	**14.82**	6.37E+07	3.43E+07
2MSS*AOV6B-33-5	PC250618	NAMCO	EA740-80120	P800A	5.00E+07	**14.82**	6.37E+07	3.43E+07
2MSS*AOV6B-33-6	PC250618	NAMCO	EA740-80120	P800A	5.00E+07	**14.82**	6.37E+07	3.43E+07
2MSS*AOV6C-33-1	PC250618	NAMCO	EA740-80120	P800A	5.00E+07	**14.82**	6.37E+07	3.43E+07
2MSS*AOV6C-33-2	PC250618	NAMCO	EA740-80120	P800A	5.00E+07	**14.82**	6.37E+07	3.43E+07
2MSS*AOV6C-33-5	PC250618	NAMCO	EA740-80120	P800A	5.00E+07	**14.82**	6.37E+07	3.43E+07
2MSS*AOV6C-33-6	PC250618	NAMCO	EA740-80120	P800A	5.00E+07	**14.82**	6.37E+07	3.43E+07
2MSS*AOV6D-33-1	PC250618	NAMCO	EA740-80120	P800A	5.00E+07	**14.82**	6.37E+07	3.43E+07

COMPONENT ID	EQ Zone	Manufacturer	Model	Spec. No.	Qualified Radiation Level (rads)	Radiation Life (Years)	60 Year Normal Dose w/EPU (rads)	100 Day Accident Dose at EPU (w/10% Margin) (rads)
2MSS*AOV6D-33-2	PC250618	NAMCO	EA740-80120	P800A	5.00E+07	**14.82**	6.37E+07	3.43E+07
2MSS*AOV6D-33-5	PC250618	NAMCO	EA740-80120	P800A	5.00E+07	**14.82**	6.37E+07	3.43E+07
2MSS*AOV6D-33-6	PC250618	NAMCO	EA740-80120	P800A	5.00E+07	**14.82**	6.37E+07	3.43E+07
2MSS*SOV6A-1	PC250618	AUTOMATIC VALVE (AVCO)	6910-074	E169	5.00E+07	**39.48**	6.37E+07	8.08E+06
2MSS*SOV6A-2	PC250618	AUTOMATIC VALVE (AVCO)	6910-074	E169	5.00E+07	**39.48**	6.37E+07	8.08E+06
2MSS*SOV6A-3	PC250618	AUTOMATIC VALVE (AVCO)	6910-074	E169	5.00E+07	**39.48**	6.37E+07	8.08E+06
2MSS*SOV6B-1	PC250618	AUTOMATIC VALVE (AVCO)	6910-074	E169	5.00E+07	**39.48**	6.37E+07	8.08E+06
2MSS*SOV6B-2	PC250618	AUTOMATIC VALVE (AVCO)	6910-074	E169	5.00E+07	**39.48**	6.37E+07	8.08E+06
2MSS*SOV6B-3	PC250618	AUTOMATIC VALVE (AVCO)	6910-074	E169	5.00E+07	**39.48**	6.37E+07	8.08E+06
2MSS*SOV6C-1	PC250618	AUTOMATIC VALVE (AVCO)	6910-074	E169	5.00E+07	**39.48**	6.37E+07	8.08E+06
2MSS*SOV6C-2	PC250618	AUTOMATIC VALVE (AVCO)	6910-074	E169	5.00E+07	**39.48**	6.37E+07	8.08E+06

COMPONENT ID	EQ Zone	Manufacturer	Model	Spec. No.	Qualified Radiation Level (rads)	Radiation Life (Years)	60 Year Normal Dose w/EPU (rads)	100 Day Accident Dose at EPU (w/10% Margin) (rads)
2MSS*SOV6C-3	PC250618	AUTOMATIC VALVE (AVCO)	6910-074	E169	5.00E+07	**39.48**	6.37E+07	8.08E+06
2MSS*SOV6D-1	PC250618	AUTOMATIC VALVE (AVCO)	6910-074	E169	5.00E+07	**39.48**	6.37E+07	8.08E+06
2MSS*SOV6D-2	PC250618	AUTOMATIC VALVE (AVCO)	6910-074	E169	5.00E+07	**39.48**	6.37E+07	8.08E+06
2MSS*SOV6D-3	PC250618	AUTOMATIC VALVE (AVCO)	6910-074	E169	5.00E+07	**39.48**	6.37E+07	8.08E+06
2CMS*SOV23A	PC289681	TARGET ROCK	76P-001	P304X	4.20E+07	**54.41**	1.40E+07	2.93E+07
2CMS*SOV23A-33	PC289681	TARGET ROCK	-0-	P304X	4.20E+07	**54.41**	1.40E+07	2.93E+07
2CMS*SOV23B	PC289681	TARGET ROCK	76P-001	P304X	4.20E+07	**54.41**	1.40E+07	2.93E+07
2CMS*SOV23B-33	PC289681	TARGET ROCK	-0-	P304X	4.20E+07	**54.41**	1.40E+07	2.93E+07
2CMS*SOV23C	PC289681	TARGET ROCK	76P-001-1	P304X	4.20E+07	**54.41**	1.40E+07	2.93E+07
2CMS*SOV23C-33	PC289681	TARGET ROCK	-0-	P304X	4.20E+07	**54.41**	1.40E+07	2.93E+07
2CMS*SOV23D	PC289681	TARGET ROCK	76P-001	P304X	4.20E+07	**54.41**	1.40E+07	2.93E+07
2CMS*SOV23D-33	PC289681	TARGET ROCK	-0-	P304X	4.20E+07	**54.41**	1.40E+07	2.93E+07

COMPONENT ID	EQ Zone	Manufacturer	Model	Spec. No.	Qualified Radiation Level (rads)	Radiation Life (Years)	60 Year Normal Dose w/EPU (rads)	100 Day Accident Dose at EPU (w/10% Margin) (rads)
2CMS*SOV23E	PC289681	TARGET ROCK	76P-001	P304X	4.20E+07	**54.41**	1.40E+07	2.93E+07
2CMS*SOV23E-33	PC289681	TARGET ROCK	-0-	P304X	4.20E+07	**54.41**	1.40E+07	2.93E+07
2CMS*SOV23F	PC289681	TARGET ROCK	76P-001	P304X	4.20E+07	**54.41**	1.40E+07	2.93E+07
2CMS*SOV23F-33	PC289681	TARGET ROCK	-0-	P304X	4.20E+07	**54.41**	1.40E+07	2.93E+07
2CMS*SOV24A	PC289681	TARGET ROCK	76P-001	P304X	4.20E+07	**54.41**	1.40E+07	2.93E+07
2CMS*SOV24A-33	PC289681	TARGET ROCK	-0-	P304X	4.20E+07	**54.41**	1.40E+07	2.93E+07
2CMS*SOV24B	PC289681	TARGET ROCK	76P-001	P304X	4.20E+07	**54.41**	1.40E+07	2.93E+07
2CMS*SOV24B-33	PC289681	TARGET ROCK	-0-	P304X	4.20E+07	**54.41**	1.40E+07	2.93E+07
2CPS*SOV122	PC289681	TARGET ROCK	76P-027	P304X	4.20E+07	**54.41**	1.40E+07	2.93E+07
2CPS*SOV122-33	PC289681	TARGET ROCK	-0-		4.20E+07	**54.41**	1.40E+07	2.93E+07
2IAS*SOV180	PC289681	TARGET ROCK	76P-019	P304X	4.20E+07	**54.41**	1.40E+07	2.93E+07
2IAS*SOV180-33	PC289681	TARGET ROCK	-0-		4.20E+07	**54.41**	1.40E+07	2.93E+07
2IAS*SOV184	PC289681	TARGET ROCK	76P-019	P304X	4.20E+07	**54.41**	1.40E+07	2.93E+07
2IAS*SOV184-33	PC289681	TARGET ROCK	-0-		4.20E+07	**54.41**	1.40E+07	2.93E+07

COMPONENT ID	EQ Zone	Manufacturer	Model	Spec. No.	Qualified Radiation Level (rads)	Radiation Life (Years)	60 Year Normal Dose w/EPU (rads)	100 Day Accident Dose at EPU (w/10% Margin) (rads)
2CMS*LT11A	SC175102	ROSEMOUNT	1153DB4	C071M	2.20E+07	**54.58**	1.65E+07	6.99E+06
2CMS*LT9A	SC175102	ROSEMOUNT	1153DB5PA	C071M	2.20E+07	**54.58**	1.65E+07	6.99E+06
2CSL*FT107	SC175102	ROSEMOUNT	1153DB3	P800A	2.20E+07	**54.58**	1.65E+07	6.99E+06
2CSL*FT126	SC175102	ROSEMOUNT	1153DB5	P800A	2.20E+07	**54.58**	1.65E+07	6.99E+06
2CSL*PDT132	SC175102	ROSEMOUNT	1153DB8	P800A	2.20E+07	**54.58**	1.65E+07	6.99E+06
2CSL*PT109	SC175102	ROSEMOUNT	1153GB7	P800A	2.20E+07	**54.58**	1.65E+07	6.99E+06
2CSL*PT110	SC175102	ROSEMOUNT	1153GB7	P800A	2.20E+07	**54.58**	1.65E+07	6.99E+06
2CSL*PT129	SC175102	ROSEMOUNT	1153GB6	P800A	2.20E+07	**54.58**	1.65E+07	6.99E+06
2CSL*PT130	SC175102	ROSEMOUNT	1153GB6	P800A	2.20E+07	**54.58**	1.65E+07	6.99E+06
2RHS*FT14A	SC175102	ROSEMOUNT	1153DB5	P800A	2.20E+07	**54.58**	1.65E+07	6.99E+06
2RHS*FT60A	SC175102	ROSEMOUNT	1153DB5	C071M	2.20E+07	**54.43**	1.65E+07	7.03E+06
2RHS*FT60B	SC175102	ROSEMOUNT	1153DB5	C071M	2.20E+07	**54.43**	1.65E+07	7.03E+06
2RHS*FT86A	SC175102	ROSEMOUNT	1153DB3	P800A	2.20E+07	**54.58**	1.65E+07	6.99E+06
2RHS*PT133	SC175102	ROSEMOUNT	1153GB7	P800A	2.20E+07	**54.58**	1.65E+07	6.99E+06

COMPONENT ID	EQ Zone	Manufacturer	Model	Spec. No.	Qualified Radiation Level (rads)	Radiation Life (Years)	60 Year Normal Dose w/EPU (rads)	100 Day Accident Dose at EPU (w/10% Margin) (rads)
2RHS*PT3A	SC175102	ROSEMOUNT	1153GB7	P800A	2.20E+07	**54.58**	1.65E+07	6.99E+06
2RHS*PT3B	SC175102	ROSEMOUNT	1153GB7	P800A	2.20E+07	**54.58**	1.65E+07	6.99E+06
2RHS*PT3C	SC175102	ROSEMOUNT	1153GB7	P800A	2.20E+07	**54.58**	1.65E+07	6.99E+06
2RHS*PT5A	SC175102	ROSEMOUNT	1153GB7	P800A	2.20E+07	**54.58**	1.65E+07	6.99E+06
2RHS*PT6A	SC175102	ROSEMOUNT	1153GB7	P800A	2.20E+07	**54.58**	1.65E+07	6.99E+06
2RHS*PT76A	SC175102	ROSEMOUNT	1153GB8	P800A	2.20E+07	**54.58**	1.65E+07	6.99E+06
2SWP*FT13A	SC175102	ROSEMOUNT	1153DB5	P800A	2.20E+07	**54.58**	1.65E+07	6.99E+06
2HVR*MST403B-X	SC196116	MICRON INDUSTRIES	B150WZ13	P412M	1.00E+07	**56.13**	9.15E+06	1.44E+06
2HVR*MST404A-X	SC196116	GOULD	D23T5115	P412M	1.00E+07	**54.02**	9.15E+06	1.76E+06
2HVR*MST404D-X	SC196116	GOULD	D23T5115	P412M	1.00E+07	**56.4**	9.15E+06	1.40E+06
2HVR*TIS25A	SC196116	UNITED ELECTRIC CONTROLS CO	802P-5AS	C071A	1.00E+07	**58.93**	9.15E+06	1.01E+06
49-2HVRD14(2)	SC196116	GOULD		P412M	1.00E+07	**56.4**	9.15E+06	1.40E+06
49-2HVRD14(3)	SC196116	GOULD		P412M	1.00E+07	**56.4**	9.15E+06	1.40E+06
2HVR*MST411A	SC261145	GOULD	A203B	P412M	1.00E+07	**47.19**	1.19E+07	6.81E+05

COMPONENT ID	EQ Zone	Manufacturer	Model	Spec. No.	Qualified Radiation Level (rads)	Radiation Life (Years)	60 Year Normal Dose w/EPU (rads)	100 Day Accident Dose at EPU (w/10% Margin) (rads)
2HVR*MST411A-X	SC261145	GOULD	D23T5115	P412M	1.00E+07	**43.11**	1.19E+07	1.49E+06
2HVR*MST411B	SC261145	GOULD	A203B	P412M	1.00E+07	**47.19**	1.19E+07	6.81E+05
2HVR*MST411B-X	SC261145	GOULD	D23T5115	P412M	1.00E+07	**43.38**	1.19E+07	1.43E+06
2HVR*MST411C	SC261145	GOULD	A203B	P412M	1.00E+07	**47.19**	1.19E+07	6.81E+05
2HVR*MST411C-X	SC261145	GOULD	D23T5115	P412M	1.00E+07	**42.28**	1.19E+07	1.65E+06
2HVR*MST414A	SC261145	GOULD	A203B	P412M	1.00E+07	**47.19**	1.19E+07	6.81E+05
2HVR*MST414A-X	SC261145	MICRON INDUSTRIES	B150WZ13	P412M	1.00E+07	**43.11**	1.19E+07	1.49E+06
2HVR*MST414B	SC261145	GOULD	A203B	P412M	1.00E+07	**47.19**	1.19E+07	6.81E+05
2HVR*MST414B-X	SC261145	MICRON INDUSTRIES	B150WZ13	P412M	1.00E+07	**38.62**	1.19E+07	2.37E+06
2HVR*TIS28A	SC261145	UNITED ELECTRIC CONTROLS CO	802P-6AS	C071A	1.00E+07	**46.9**	1.19E+07	7.37E+05
2HVR*TIS28B	SC261145	UNITED ELECTRIC CONTROLS CO	802P-6AS	C071A	1.00E+07	**47.07**	1.19E+07	7.04E+05
2HVR*TIS28C	SC261145	UNITED ELECTRIC CONTROLS CO	802P-6AS	C071A	1.00E+07	**46.07**	1.19E+07	9.02E+05

COMPONENT ID	EQ Zone	Manufacturer	Model	Spec. No.	Qualified Radiation Level (rads)	Radiation Life (Years)	60 Year Normal Dose w/EPU (rads)	100 Day Accident Dose at EPU (w/10% Margin) (rads)
2HVR*TIS35A	SC261145	UNITED ELECTRIC CONTROLS CO	802P-5AS	C071A	1.00E+07	**46.9**	1.19E+07	7.37E+05
2HVR*TIS35B	SC261145	UNITED ELECTRIC CONTROLS CO	802P-5AS	C071A	1.00E+07	**46.44**	1.19E+07	8.27E+05
2RDS*LTX12A	SC261145	GOULD	PD3218	P800A	1.00E+07	**39.99**	1.19E+07	2.10E+06
2RDS*LTX12B	SC261145	GOULD	PD3218	P800A	1.00E+07	**39.99**	1.19E+07	2.10E+06
2RDS*LTY12A	SC261145	GOULD	PD3218	P800A	1.00E+07	**39.99**	1.19E+07	2.10E+06
2RDS*LTY12B	SC261145	GOULD	PD3218	P800A	1.00E+07	**39.99**	1.19E+07	2.10E+06
49-2HVRX01(2)	SC261145	GOULD			1.00E+07	**43.11**	1.19E+07	1.49E+06
49-2HVRX01(3)	SC261145	GOULD			1.00E+07	**43.11**	1.19E+07	1.49E+06
49-2HVRY01(2)	SC261145	GOULD			1.00E+07	**43.38**	1.19E+07	1.43E+06
49-2HVRY01(3)	SC261145	GOULD			1.00E+07	**43.38**	1.19E+07	1.43E+06
F-2HVRA86	SC261145	GOULD	ATM3	P412M	1.00E+07	**47.19**	1.19E+07	6.81E+05
F-2HVRB86	SC261145	GOULD	ATM3	P412M	1.00E+07	**47.19**	1.19E+07	6.81E+05
F-2HVRC86	SC261145	GOULD	ATM3	P412M	1.00E+07	**47.19**	1.19E+07	6.81E+05

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F-2HVRX01	SC261145	GOULD	ATM3	P412M	1.00E+07	**47.19**	1.19E+07	6.81E+05
F-2HVRY01	SC261145	GOULD	ATM3	P412M	1.00E+07	**47.19**	1.19E+07	6.81E+05
2WCS*TE75A	SC289158	PYCO	102-9039-08	P800A	2.00E+08	**30.68**	3.90E+08	5.81E+05
2WCS*TE75B	SC289158	PYCO	102-9039-08	P800A	2.00E+08	**30.68**	3.90E+08	5.81E+05
2E*13	WCOPC	MARATHON	1500 NUC	E061A	1.87E+08	**56.9**	1.65E+08	3.05E+07
2E*30	WCOPC	ROSEMOUNT	353C	E061A	1.07E+08	**27.81**	1.65E+08	3.05E+07
NJM-04	WCOPC	KERITE	FR/FR	E024A	4.69E+07	**30.37**	6.45E+07	1.43E+07
NJM-05	WCOPC	KERITE	FR/FR	E024A	4.69E+07	**30.37**	6.45E+07	1.43E+07
NJM-06	WCOPC	KERITE	FR/FR	E024A	4.69E+07	**30.37**	6.45E+07	1.43E+07
NJM-10	WCOPC	KERITE	FR/FR	E024A	4.69E+07	**30.37**	6.45E+07	1.43E+07
NJM-13	WCOPC	KERITE	FR/FR	E024A	4.69E+07	**30.37**	6.45E+07	1.43E+07
NJM-14	WCOPC	KERITE	FR/FR	E024A	4.69E+07	**30.37**	6.45E+07	1.43E+07
NJN-07	WCOPC	KERITE	FR/FR	E024A	4.69E+07	**30.37**	6.45E+07	1.43E+07
NJN-08	WCOPC	KERITE	FR/FR	E024A	4.69E+07	**30.37**	6.45E+07	1.43E+07

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NJN-09	WCOPC	KERITE	FR/FR	E024A	4.69E+07	**30.37**	6.45E+07	1.43E+07
NJN-10	WCOPC	KERITE	FR/FR	E024A	4.69E+07	**30.37**	6.45E+07	1.43E+07
NJN-18	WCOPC	KERITE	FR/FR	E024A	4.69E+07	**30.37**	6.45E+07	1.43E+07
NJN-19	WCOPC	KERITE	FR/FR	E024A	4.69E+07	**30.37**	6.45E+07	1.43E+07
NJN-20	WCOPC	KERITE	FR/FR	E024A	4.69E+07	**30.37**	6.45E+07	1.43E+07
NJN-21	WCOPC	KERITE	FR/FR	E024A	4.69E+07	**30.37**	6.45E+07	1.43E+07
NJN-22	WCOPC	KERITE	FR/FR	E024A	4.69E+07	**30.37**	6.45E+07	1.43E+07
NJN-23	WCOPC	KERITE	FR/FR	E024A	4.69E+07	**30.37**	6.45E+07	1.43E+07
NJN-31	WCOPC	KERITE	FR/FR	E024A	4.69E+07	**30.37**	6.45E+07	1.43E+07
NJN-32	WCOPC	KERITE	FR/FR	E024A	4.69E+07	**30.37**	6.45E+07	1.43E+07
NJP-01	WCOPC	BRAND REX	T-7936		1.93E+08	**59.15**	1.65E+08	3.05E+07
NJP-02	WCOPC	BRAND REX	T-10744		1.93E+08	**59.15**	1.65E+08	3.05E+07
NJP-07	WCOPC	BRAND REX	T-10744		1.93E+08	**59.15**	1.65E+08	3.05E+07
NJP-22	WCOPC	BRAND REX	T-10744		1.93E+08	**59.15**	1.65E+08	3.05E+07

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NJP-28	WCOPC	ROCKBESTOS	RSS-6-200	E024B	1.71E+08	**51.08**	1.65E+08	3.05E+07
NJP-51	WCOPC	ROCKBESTOS	RSS-6-207	E024B	1.71E+08	**51.08**	1.65E+08	3.05E+07
NJP-63	WCOPC	BRAND REX	CS 75285		1.93E+08	**59.15**	1.65E+08	3.05E+07
2ANY*SL1	WCPLT	RAYCHEM CORP	PART NO. EPPA-109N-3	M070A	2.20E+08	**41.87**	1.55E+08	1.12E+08
2E*19	WCPLT	RAYCHEM CORP	WCSF-N,2" MIN	E061A	2.20E+08	**41.87**	1.55E+08	1.12E+08
2E*20	WCPLT	RAYCHEM CORP	N-MCK	E061A	2.20E+08	**41.87**	1.55E+08	1.12E+08
2E*21	WCPLT	RAYCHEM CORP	NPKV	E061A	2.20E+08	**41.87**	1.55E+08	1.12E+08
2E*22	WCPLT	RAYCHEM CORP	NHVT-U	E061A	2.20E+08	**41.87**	1.55E+08	1.12E+08
2E*23	WCPLT	RAYCHEM CORP	NMCK-8	E061A	2.20E+08	**41.87**	1.55E+08	1.12E+08
2E*31	WCPLT	CONAX BUFFALO	ECSA	E061A	2.25E+08	**43.8**	1.55E+08	1.12E+08
2E*37	WCPLT	RAYCHEM CORP	NESK	E0904	2.20E+08	**41.87**	1.55E+08	1.12E+08
2E*40	WCPLT	RAYCHEM CORP	NPKX	E0904	2.20E+08	**41.87**	1.55E+08	1.12E+08
2E*43	WCPLT	RAYCHEM CORP	NPKC	E0904	2.20E+08	**41.87**	1.55E+08	1.12E+08
2E*44	WCPLT	RAYCHEM CORP	NEIS	E061A	2.20E+08	**41.87**	1.55E+08	1.12E+08

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2E*45	WCPLT	NAMCO CONTROLS	EC210-290XX,EC210-19001	E061A	2.04E+08	**35.66**	1.55E+08	1.12E+08
2E*46	WCPLT	LOCTITE	PST-580	E061A	2.00E+08	**34.11**	1.55E+08	1.12E+08
2E*49	WCPLT	CURTIS	TYPE 'L'	E061A	1.87E+08	**29.07**	1.55E+08	1.12E+08
2E*50	WCPLT	MARATHON	300	E061A	1.87E+08	**29.07**	1.55E+08	1.12E+08
2E*51	WCPLT	EGS	880701/913601 SERIES	E061A	2.00E+08	**34.11**	1.55E+08	1.12E+08
NAF-50	WCPLT	ROCKBESTOS	KXL-760	E024B	1.84E+08	**27.91**	1.55E+08	1.12E+08
NAF-51	WCPLT	ROCKBESTOS	KXL-760	E024B	1.84E+08	**27.91**	1.55E+08	1.12E+08
NAF-52	WCPLT	ROCKBESTOS	KXL-760G	E024B	1.84E+08	**27.91**	1.55E+08	1.12E+08
NJM-01	WCPLT	ROCKBESTOS	KXL-760	E024B	1.84E+08	**27.91**	1.55E+08	1.12E+08
NJM-02	WCPLT	ROCKBESTOS	KXL-760	E024B	1.84E+08	**27.91**	1.55E+08	1.12E+08
NJM-03	WCPLT	ROCKBESTOS	KXL-760	E024B	1.84E+08	**27.91**	1.55E+08	1.12E+08
NJM-08	WCPLT	OKONITE	OKONITE/OKOLON	E023C	2.00E+08	**34.11**	1.55E+08	1.12E+08
NJM-12	WCPLT	OKONITE	OKONITE/OKOLON	E023C	2.00E+08	**34.11**	1.55E+08	1.12E+08
NJM-15	WCPLT	OKONITE	OKONITE/OKOLON	E023C	2.00E+08	**34.11**	1.55E+08	1.12E+08

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NJM-16	WCPLT	OKONITE	OKONITE/OKOLON	E023C	2.00E+08	**34.11**	1.55E+08	1.12E+08
NJM-17	WCPLT	OKONITE	OKONITE/OKOLON	E023C	2.00E+08	**34.11**	1.55E+08	1.12E+08
NJM-18	WCPLT	OKONITE	OKONITE/OKOLON	E023C	2.00E+08	**34.11**	1.55E+08	1.12E+08
NJM-25	WCPLT	OKONITE	OKONITE/OKOLON	E023C	2.00E+08	**34.11**	1.55E+08	1.12E+08
NJM-28	WCPLT	OKONITE	OKONITE/OKOLON	E023C	2.00E+08	**34.11**	1.55E+08	1.12E+08
NJM-30	WCPLT	OKONITE	OKONITE/OKOLON	E023C	2.00E+08	**34.11**	1.55E+08	1.12E+08
NJM-31	WCPLT	OKONITE	OKONITE/OKOLON	E023C	2.00E+08	**34.11**	1.55E+08	1.12E+08
NJM-33	WCPLT	OKONITE	OKONITE/OKOLON	E023C	2.00E+08	**34.11**	1.55E+08	1.12E+08
NJM-34	WCPLT	OKONITE	OKONITE/OKOLON	E023C	2.00E+08	**34.11**	1.55E+08	1.12E+08
NJM-36	WCPLT	OKONITE	OKONITE/OKOLON	E023C	2.00E+08	**34.11**	1.55E+08	1.12E+08
NJM-37	WCPLT	OKONITE	OKONITE/OKOLON	E023C	2.00E+08	**34.11**	1.55E+08	1.12E+08
NJM-40	WCPLT	OKONITE	OKONITE/OKOLON	E023C	2.00E+08	**34.11**	1.55E+08	1.12E+08
NJM-41	WCPLT	OKONITE	OKONITE/OKOLON	E023C	2.00E+08	**34.11**	1.55E+08	1.12E+08
NJM-42	WCPLT	ROCKBESTOS	KXL-760	E024B	1.84E+08	**27.91**	1.55E+08	1.12E+08

COMPONENT ID	EQ Zone	Manufacturer	Model	Spec. No.	Qualified Radiation Level (rads)	Radiation Life (Years)	60 Year Normal Dose w/EPU (rads)	100 Day Accident Dose at EPU (w/10% Margin) (rads)
NJM-44	WCPLT	OKONITE	OKONITE/OKOLON	E023C	2.00E+08	**34.11**	1.55E+08	1.12E+08
NJM-45	WCPLT	OKONITE	OKONITE/OKOLON	E023C	2.00E+08	**34.11**	1.55E+08	1.12E+08
NJM-46	WCPLT	OKONITE	OKONITE/OKOLON	E023C	2.00E+08	**34.11**	1.55E+08	1.12E+08
NJM-47	WCPLT	OKONITE	OKONITE/OKOLON	E023C	2.00E+08	**34.11**	1.55E+08	1.12E+08
NJM-48	WCPLT	OKONITE	OKONITE/OKOLON	E023C	2.00E+08	**34.11**	1.55E+08	1.12E+08
NJM-49	WCPLT	OKONITE	OKONITE/OKOLON	E023C	2.00E+08	**34.11**	1.55E+08	1.12E+08
NJM-50	WCPLT	ROCKBESTOS	KXL-760	E024B	1.84E+08	**27.91**	1.55E+08	1.12E+08
NJM-51	WCPLT	ROCKBESTOS	KXL-760	E024B	1.84E+08	**27.91**	1.55E+08	1.12E+08
NJM-52	WCPLT	ROCKBESTOS	KXL-760	E024B	1.84E+08	**27.91**	1.55E+08	1.12E+08
NJM-53	WCPLT	ROCKBESTOS	KXL-760	E024B	1.84E+08	**27.91**	1.55E+08	1.12E+08
NJM-54	WCPLT	ROCKBESTOS	KXL-760		1.84E+08	**27.91**	1.55E+08	1.12E+08
NJM-55	WCPLT	ROCKBESTOS	KXL-760	E024B	1.84E+08	**27.91**	1.55E+08	1.12E+08
NJN-34	WCPLT	ROCKBESTOS	KXL-760	E024B	1.84E+08	**27.14**	1.55E+08	1.14E+08
NJN-35	WCPLT	ROCKBESTOS	KXL-760	E024B	1.84E+08	**27.91**	1.55E+08	1.12E+08

COMPONENT ID	EQ Zone	Manufacturer	Model	Spec. No.	Qualified Radiation Level (rads)	Radiation Life (Years)	60 Year Normal Dose w/EPU (rads)	100 Day Accident Dose at EPU (w/10% Margin) (rads)
NJN-36	WCPLT	ROCKBESTOS	KXL-760	E024B	1.84E+08	**27.91**	1.55E+08	1.12E+08
NJN-37	WCPLT	ROCKBESTOS	KXL-760	E024B	1.84E+08	**27.91**	1.55E+08	1.12E+08
NJN-38	WCPLT	ROCKBESTOS	KXL-760	E024B	1.84E+08	**27.91**	1.55E+08	1.12E+08
NJN-39	WCPLT	ROCKBESTOS	KXL-760	E024B	1.84E+08	**27.91**	1.55E+08	1.12E+08
NJN-59	WCPLT	ROCKBESTOS	KXL-760	E024B	1.84E+08	**27.91**	1.55E+08	1.12E+08
NJP-13	WCPLT	ROCKBESTOS	RSS-6-112/LE	E024B	1.71E+08	**51.08**	1.65E+08	3.05E+07
NJP-16	WCPLT	ROCKBESTOS	KXL-760	E024R	1.84E+08	**27.91**	1.55E+08	1.12E+08
NJP-20	WCPLT	ROCKBESTOS	KXL-760	E024R	1.84E+08	**27.91**	1.55E+08	1.12E+08
NJP-21	WCPLT	ROCKBESTOS	KXL-760	E024R	1.84E+08	**27.91**	1.55E+08	1.12E+08
NJP-23	WCPLT	ROCKBESTOS	KXL-760	E024R	1.84E+08	**27.91**	1.55E+08	1.12E+08
NJP-24	WCPLT	ROCKBESTOS	KXL-760	E024R	1.84E+08	**27.91**	1.55E+08	1.12E+08
NJP-25	WCPLT	ROCKBESTOS	KXL-760	E024B	1.84E+08	**27.91**	1.55E+08	1.12E+08
NJP-26	WCPLT	ROCKBESTOS	KXL-760	E024R	1.84E+08	**27.91**	1.55E+08	1.12E+08
NJP-27	WCPLT	ROCKBESTOS	KXL-760	E024R	1.84E+08	**27.91**	1.55E+08	1.12E+08

COMPONENT ID	EQ Zone	Manufacturer	Model	Spec. No.	Qualified Radiation Level (rads)	Radiation Life (Years)	60 Year Normal Dose w/EPU (rads)	100 Day Accident Dose at EPU (w/10% Margin) (rads)
NJP-29	WCPLT	ROCKBESTOS	RSS-6-104/LE	E024B	1.71E+08	**52.34**	6.42E+07	1.15E+08
NJP-31	WCPLT	ROCKBESTOS	RSS-6-110A/LE	E024B	1.71E+08	**51.08**	1.65E+08	3.05E+07
NJP-32	WCPLT	ROCKBESTOS	RSS-6-116/LE	E024B	1.71E+08	**51.08**	1.65E+08	3.05E+07
NJP-33	WCPLT	ROCKBESTOS	KXL-760	E024B	1.84E+08	**27.91**	1.55E+08	1.12E+08
NJP-35	WCPLT	ROCKBESTOS	KXL-760	E024B	1.84E+08	**27.91**	1.55E+08	1.12E+08
NJP-36	WCPLT	ROCKBESTOS	KXL-760	E024B	1.84E+08	**27.91**	1.55E+08	1.12E+08
NJP-37	WCPLT	ROCKBESTOS	KXL-760	E024B	1.84E+08	**27.91**	1.55E+08	1.12E+08
NJP-38	WCPLT	ROCKBESTOS	KXL-760	E024B	1.84E+08	**27.91**	1.55E+08	1.12E+08
NJP-52	WCPLT	ROCKBESTOS	RSS-6-105/LE	E024B	1.71E+08	**51.08**	1.65E+08	3.05E+07
NJP-57	WCPLT	ROCKBESTOS	KXL-760	E024B	1.84E+08	**27.91**	1.55E+08	1.12E+08
NJP-61	WCPLT	ROCKBESTOS	KXL-760	E024B	1.84E+08	**27.91**	1.55E+08	1.12E+08
NJP-67	WCPLT	ROCKBESTOS	KXL-760	E024B	1.84E+08	**27.91**	1.55E+08	1.12E+08
NJP-68	WCPLT	ROCKBESTOS	KXL-760	E024B	1.84E+08	**27.91**	1.55E+08	1.12E+08
NJP-69	WCPLT	ROCKBESTOS	KXL-760	E024B	1.84E+08	**27.91**	1.55E+08	1.12E+08

COMPONENT ID	EQ Zone	Manufacturer	Model	Spec. No.	Qualified Radiation Level (rads)	Radiation Life (Years)	60 Year Normal Dose w/EPU (rads)	100 Day Accident Dose at EPU (w/10% Margin) (rads)
NJP-70	WCPLT	ROCKBESTOS	KXL-760	E024B	1.84E+08	**27.91**	1.55E+08	1.12E+08
NJP-72	WCPLT	ROCKBESTOS	KXL-760	E024B	1.84E+08	**27.91**	1.55E+08	1.12E+08
NJP-73	WCPLT	ROCKBESTOS	KXL-760	E024B	1.84E+08	**27.91**	1.55E+08	1.12E+08
NJP-74	WCPLT	ROCKBESTOS	KXL-760	E024B	1.84E+08	**27.91**	1.55E+08	1.12E+08
NJP-75	WCPLT	ROCKBESTOS	KXL-760	E024B	1.84E+08	**27.91**	1.55E+08	1.12E+08
NJP-76	WCPLT	ROCKBESTOS	KXL-760	E024B	1.84E+08	**27.91**	1.55E+08	1.12E+08
NJP-77	WCPLT	ROCKBESTOS	KXL-760	E024B	1.84E+08	**27.91**	1.55E+08	1.12E+08
NJP-78	WCPLT	ROCKBESTOS	KXL-760	E024B	1.84E+08	**27.91**	1.55E+08	1.12E+08
NJP-79	WCPLT	ROCKBESTOS	KXL-760	E024B	1.84E+08	**27.91**	1.55E+08	1.12E+08
NJP-80	WCPLT	ROCKBESTOS	KXL-760	E024B	1.84E+08	**27.91**	1.55E+08	1.12E+08
NJP-81	WCPLT	ROCKBESTOS	KXL-760	E024B	1.84E+08	**27.91**	1.55E+08	1.12E+08
NJP-82	WCPLT	ROCKBESTOS	KXL-760	E024B	1.84E+08	**27.91**	1.55E+08	1.12E+08
NJP-83	WCPLT	ROCKBESTOS	KXL-760	E024B	1.84E+08	**27.91**	1.55E+08	1.12E+08
NJP-84	WCPLT	ROCKBESTOS	KXL-760	E024B	1.84E+08	**27.91**	1.55E+08	1.12E+08

COMPONENT ID	EQ Zone	Manufacturer	Model	Spec. No.	Qualified Radiation Level (rads)	Radiation Life (Years)	60 Year Normal Dose w/EPU (rads)	100 Day Accident Dose at EPU (w/10% Margin) (rads)
NJP-85	WCPLT	ROCKBESTOS	KXL-760	E024B	1.84E+08	**27.91**	1.55E+08	1.12E+08
NJP-86	WCPLT	ROCKBESTOS	KXL-760	E024B	1.84E+08	**27.91**	1.55E+08	1.12E+08
NJP-87	WCPLT	ROCKBESTOS	KXL-760	E024B	1.84E+08	**27.91**	1.55E+08	1.12E+08
NJP-88	WCPLT	ROCKBESTOS	KXL-760	E024B	1.84E+08	**27.91**	1.55E+08	1.12E+08
NJP-90	WCPLT	ROCKBESTOS	KXL-760	E024B	1.84E+08	**27.91**	1.55E+08	1.12E+08
NJP-91	WCPLT	ROCKBESTOS	KXL-760	E024B	1.84E+08	**27.91**	1.55E+08	1.12E+08
NJP-94	WCPLT	ROCKBESTOS	KXL-760	E024B	1.84E+08	**27.91**	1.55E+08	1.12E+08
NJQ-01	WCPLT	ROCKBESTOS	KXL-760	E024B	1.84E+08	**27.91**	1.55E+08	1.12E+08
NMP-08	WCPLT	ROCKBESTOS	KXL-760	E024B	1.84E+08	**27.91**	1.55E+08	1.12E+08
NMP-16	WCPLT	ROCKBESTOS	KXL-760	E024P	1.84E+08	**27.91**	1.55E+08	1.12E+08

Table 2.3-2 Group III, Non-qualified Components

EQ Zone	COMPONENT ID	Manufacturer	Model	Spec. No.	Qualified Radiation Level (rads)	60 Year Normal Dose w/EPU (rads)	EPU 100 Day Accident Dose (w/10% Margin) (rads)
SG261355	2GTS*PNL30B	NUTHERM INTERNATIONAL	1023-50723-33	P243U	1.10E+06	4.20E+04	1.17E+06
SG261355	2GTS*PNL30B-DS	AMERICAN SOLENOID CO INC	K25	P243U	1.10E+06	4.20E+04	1.17E+06
SG261355	2GTS*PNL30B-LITE1	MICRO SWITCH	PTL2111 W/PTLZ16	P243U	1.10E+06	4.20E+04	1.17E+06
SG261355	2GTS*PNL30B-LITE2	MICRO SWITCH	PTL2111 W/PTLZ16	P243U	1.10E+06	4.20E+04	1.17E+06
SG261355	2GTS*PNL30B-LITE3	MICRO SWITCH	PTL2241 W/PTLZ13	P243U	1.10E+06	4.20E+04	1.17E+06
SG261355	2GTS*PNL30B-TX	SOLITECH	K3-480-30-1119	P243U	1.10E+06	4.20E+04	1.17E+06
SG261355	2GTS*TC26B	MOORE INDUSTRIES	RBT/3WS-10/4-20MA/117AC/EZ115-RO(STD)	P243U	1.10E+06	4.20E+04	1.17E+06
SG261355	2GTS*TIS9B[1]	UNITED ELECTRIC CONTROLS CO	802P-7BS	P243U	1.00E+07	1.18E+06	3.15E+08
SG261355	2GTS*XD1B	SQUARE D	STYLE NO. 161702-2	P243U	1.10E+06	6.60E+04	1.13E+06

SG261356	2GTS*PNL30A	NUTHERM INTERNATIONAL	1023-50723-33	P243U	1.10E+06	5.55E+04	1.32E+06
SG261356	2GTS*PNL30A-DS	AMERICAN SOLENOID CO INC	K25	P243U	1.10E+06	5.55E+04	1.32E+06
SG261356	2GTS*PNL30A-LITE1	MICRO SWITCH	PTL2111 W/PTLZ16	P243U	1.10E+06	5.55E+04	1.32E+06
SG261356	2GTS*PNL30A-LITE2	MICRO SWITCH	PTL2111 W/PTLZ16	P243U	1.10E+06	5.55E+04	1.32E+06
SG261356	2GTS*PNL30A-LITE3	MICRO SWITCH	PTL2241 W/PTLZ13	P243U	1.10E+06	5.55E+04	1.32E+06
SG261356	2GTS*PNL30A-TX	SOLITECH	K3-480-30-1119	P243U	1.10E+06	5.55E+04	1.32E+06
SG261356	2GTS*TC26A	MOORE INDUSTRIES	RBT/3WS-10/4-20MA/117AC/E2115-RO(STD)	P243U	1.10E+06	5.55E+04	1.32E+06
SG261356	2GTS*TIS9A[1]	UNITED ELECTRIC CONTROLS CO	802P-7BS	P243U	1.00E+07	1.18E+06	3.15E+08

(1) 2GTS*TIS9A and 2GTS*TIS9B have been retired in place. They will be removed from the EQ program.

Table 2.3-3 Normal Primary Containment (Drywell and Suppression Chamber) 40 Year Neutron and Gamma Radiation Doses with EPU Conditions (92% Capacity Factor)

EQ Zone	Current Normal γ Dose Rate (rads/hr)	EPU Normal γ Dose Rate (rads/hr)	Current Normal Neutron Flux (neutrons/cm^2-sec)	EPU Normal Neutron Flux (neutrons/cm^2-sec)	60 yr Normal Dose w/EPU (γ + η) (92% CF) (rads)
PC175101	2.17E+01	2.17E+01	0.00E+00	0.00E+00	1.03E+07
PC199112	2.17E+01	2.17E+01	0.00E+00	0.00E+00	1.03E+07
PC215121	2.17E+01	2.17E+01	0.00E+00	0.00E+00	1.03E+07
PC240208	9.74E+01	9.74E+01	0.00E+00	0.00E+00	4.63E+07
PC240600	1.35E+02	1.35E+02	0.00E+00	0.00E+00	6.42E+07
PC240601	2.23E+01	2.23E+01	0.00E+00	0.00E+00	1.06E+07
PC240602	3.67E+01	3.67E+01	0.00E+00	0.00E+00	1.75E+07
PC240603	9.85E+00	9.85E+00	0.00E+00	0.00E+00	4.68E+06
PC240604	1.35E+02	1.35E+02	0.00E+00	0.00E+00	6.42E+07
PC240605	1.04E+01	1.04E+01	0.00E+00	0.00E+00	4.95E+06
PC240606	1.52E+02	1.52E+02	0.00E+00	0.00E+00	7.23E+07
PC240607	5.64E+01	5.64E+01	0.00E+00	0.00E+00	2.68E+07
PC240608	4.29E+01	4.29E+01	0.00E+00	0.00E+00	2.04E+07
PC240609	8.29E+01	8.29E+01	0.00E+00	0.00E+00	3.94E+07
PC240610	4.52E+01	4.52E+01	0.00E+00	0.00E+00	2.15E+07
PC240611	4.81E+01	4.81E+01	0.00E+00	0.00E+00	2.29E+07
PC240612	9.06E+00	9.06E+00	0.00E+00	0.00E+00	4.31E+06
PC250618	1.34E+02	1.34E+02	0.00E+00	0.00E+00	6.37E+07
PC250619	2.31E+01	2.31E+01	0.00E+00	0.00E+00	1.10E+07

EQ Zone	Current Normal γ Dose Rate (rads/hr)	EPU Normal γ Dose Rate (rads/hr)	Current Normal Neutron Flux (neutrons/cm^2-sec)	EPU Normal Neutron Flux (neutrons/cm^2-sec)	60 yr Normal Dose w/EPU (γ + η) (92% CF) (rads)
PC250620	1.52E+02	1.52E+02	0.00E+00	0.00E+00	7.23E+07
PC250621	1.07E+02	1.07E+02	0.00E+00	0.00E+00	5.09E+07
PC250622	1.41E+02	1.41E+02	0.00E+00	0.00E+00	6.70E+07
PC250623	1.07E+02	1.07E+02	0.00E+00	0.00E+00	5.09E+07
PC250624	1.36E+02	1.36E+02	0.00E+00	0.00E+00	6.47E+07
PC250625	1.19E+01	1.19E+01	0.00E+00	0.00E+00	5.66E+06
PC250626	3.78E+01	3.78E+01	0.00E+00	0.00E+00	1.80E+07
PC250627	1.08E+01	1.08E+01	0.00E+00	0.00E+00	5.14E+06
PC250628	1.90E+02	1.90E+02	0.00E+00	0.00E+00	9.03E+07
PC250629	1.54E+02	1.54E+02	0.00E+00	0.00E+00	7.32E+07
PC250630	1.20E+01	1.20E+01	0.00E+00	0.00E+00	5.71E+06
PC261207	9.74E+01	9.74E+01	0.00E+00	0.00E+00	4.63E+07
PC261636	2.48E+02	2.48E+02	0.00E+00	0.00E+00	1.18E+08
PC261637	6.95E+01	6.95E+01	0.00E+00	0.00E+00	3.30E+07
PC261638	2.75E+01	2.75E+01	0.00E+00	0.00E+00	1.31E+07
PC261639	2.17E+02	2.17E+02	0.00E+00	0.00E+00	1.03E+08
PC261640	7.84E+01	7.84E+01	0.00E+00	0.00E+00	3.73E+07
PC261641	2.84E+01	2.84E+01	0.00E+00	0.00E+00	1.35E+07
PC261642	2.14E+02	2.14E+02	0.00E+00	0.00E+00	1.02E+08
PC261643	8.72E+01	8.72E+01	0.00E+00	0.00E+00	4.15E+07
PC261644	2.47E+01	2.47E+01	0.00E+00	0.00E+00	1.17E+07
PC261645	2.24E+02	2.24E+02	0.00E+00	0.00E+00	1.07E+08

EQ Zone	Current Normal γ Dose Rate (rads/hr)	EPU Normal γ Dose Rate (rads/hr)	Current Normal Neutron Flux (neutrons/cm^2-sec)	EPU Normal Neutron Flux (neutrons/cm^2-sec)	60 yr Normal Dose w/EPU (γ + η) (92% CF) (rads)
PC261646	2.54E+02	2.54E+02	0.00E+00	0.00E+00	1.21E+08
PC261647	8.73E+01	8.73E+01	0.00E+00	0.00E+00	4.15E+07
PC261648	3.88E+01	3.88E+01	0.00E+00	0.00E+00	1.84E+07
PC261649	2.47E+01	2.47E+01	0.00E+00	0.00E+00	1.17E+07
PC261650	1.94E+02	1.94E+02	0.00E+00	0.00E+00	9.22E+07
PC261651	7.03E+01	7.03E+01	0.00E+00	0.00E+00	3.34E+07
PC261714	*see note*	*see note*	*see note*	*see note*	*see note*
PC279657	5.94E+02	6.93E+02	1.17E+06	1.17E+06	3.17E+08
PC279658	3.59E+02	4.19E+02	6.01E+05	6.01E+05	1.91E+08
PC279659	5.72E+02	6.67E+02	1.17E+06	1.17E+06	3.06E+08
PC280663	3.62E+02	4.22E+02	6.01E+05	6.01E+05	1.93E+08
PC280664	3.56E+02	4.15E+02	6.01E+05	6.01E+05	1.90E+08
PC280665	3.63E+02	4.23E+02	6.01E+05	6.01E+05	1.93E+08
PC287669	9.23E+01	1.08E+02	1.99E+05	1.99E+05	4.94E+07
PC287670	6.05E+01	7.05E+01	1.33E+05	1.33E+05	3.24E+07
PC287671	5.33E+01	6.21E+01	1.33E+05	1.33E+05	2.87E+07
PC287672	8.52E+01	9.93E+01	1.99E+05	1.99E+05	4.57E+07
PC287673	5.23E+01	6.10E+01	1.33E+05	1.33E+05	2.82E+07
PC287674	5.83E+01	6.80E+01	1.33E+05	1.33E+05	3.13E+07
PC289679	2.67E+02	3.11E+02	2.29E+06	2.29E+06	1.55E+08
PC289680	7.67E+01	8.94E+01	2.05E+05	2.05E+05	4.14E+07
PC289681	2.60E+01	3.03E+01	6.46E+04	6.46E+04	1.40E+07

EQ Zone	Current Normal γ Dose Rate (rads/hr)	EPU Normal γ Dose Rate (rads/hr)	Current Normal Neutron Flux (neutrons/cm^2-sec)	EPU Normal Neutron Flux (neutrons/cm^2-sec)	60 yr Normal Dose w/EPU (γ + η) (92% CF) (rads)
PC289682	7.96E+01	9.28E+01	4.16E+05	4.16E+05	4.43E+07
PC289683	2.68E+02	3.12E+02	2.29E+06	2.29E+06	1.55E+08
PC289684	2.70E+02	3.15E+02	2.30E+06	2.30E+06	1.56E+08
PC289685	2.69E+02	3.14E+02	2.29E+06	2.29E+06	1.56E+08
PC289686	7.83E+01	9.13E+01	4.16E+05	4.16E+05	4.36E+07
PC297691	2.52E+02	2.94E+02	2.24E+06	2.24E+06	1.47E+08
PC297692	2.49E+02	2.90E+02	2.24E+06	2.24E+06	1.45E+08
PC297693	2.52E+02	2.94E+02	2.24E+06	2.24E+06	1.47E+08
PC299697	2.42E+02	2.82E+02	2.21E+06	2.21E+06	1.41E+08
PC299698	5.94E+01	6.93E+01	1.77E+05	1.77E+05	3.22E+07
PC299699	1.87E+01	2.18E+01	5.34E+04	5.34E+04	1.01E+07
PC299700	2.46E+02	2.87E+02	2.21E+06	2.21E+06	1.43E+08
PC299701	2.46E+02	2.87E+02	2.21E+06	2.21E+06	1.43E+08
PC303705	2.60E+01	3.03E+01	7.91E+04	7.91E+04	1.41E+07
PC303706	1.55E+01	1.81E+01	7.91E+04	7.91E+04	8.62E+06
PC303707	2.61E+01	3.04E+01	7.91E+04	7.91E+04	1.41E+07
PC306711	3.72E+01	4.34E+01	2.10E+05	2.10E+05	2.08E+07
PC306712	1.49E+01	1.74E+01	3.33E+04	3.33E+04	7.99E+06
PC306713	1.24E+01	1.45E+01	2.21E+04	2.21E+04	6.61E+06
PC328185	2.09E+01	2.44E+01	5.44E+04	5.44E+04	1.13E+07

Note: This zone is the area between the Reactor Vessel and the Biological Shield Wall from El 265' to El 314'. No EQ Equipment is located in this area and normal and accident environmental conditions are not established for this area in any environmental basis calculations. Therefore, no environmental conditions are specified and no Tables or Figures are provided.

Table 2.3-4 Electrical Equipment Ratings and Margins

Component	Component Rating	CLTP Duty	CLTP Margin (%)	EPU Duty	EPU Margin (%)
Main Generator (MVA Capability /power factor)	1399.2/0.9	1310/0.9	6.4	1399.2/0.98	0
Isolated Phase Bus Duct (Amps)	32,800	30,260 (Note 4)	7.7	32,310 (Note 4)	1.5 (Note 5)
Main Generator Step-Up Transformers (MVA)	1,371	1238 (Note 6)	9	1327 (Note 6)	3
Reserve Station Service Transformers (Max. MVA)	70	54.2	22.6	56.6	19.1
Limiting Component - 345kV Switchyard: NMP Line Drops (Amps)	2,544	2,193 (Note 1)	13	2,341 (Note 2)	8
Limiting Component - 115kV Switchyard: Scriba Breaker CTs (Amps)	1,000	843 (Note 3)	15	843 (Note 3)	15

Note 1 Current has been calculated using 100% voltage, generator output of 1310 MVA and assuming station service transformer is out-of-service (represents worst case, generator output is supplied to grid).

Note 2 Current has been calculated using 100% voltage, rated generator output of 1399 MVA and assuming station service transformer is out-of-service (represents worst case, rated generator output is supplied to grid).

Note 3 The worst case load on either of the 115 kV lines and associated equipment would be the load of both Reserve Station Service Transformers (2 x 70 = 140 MVA) plus the load of the Auxiliary Boiler Transformer (27.56 MVA) resulting in a total load of approximately 168 MVA or approximately 843 amperes at 115 kV.

Note 4 Calculated at 100% generator output voltage using generator output defined in NOTE 1 and NOTE 2.

Note 5 The isolated phase bus duct will be modified or re-rated to increase margin to account for voltages less than 100%.

Note 6 Normal operation with the station service transformer in service (i.e. unit output = generator output - 72 MVA for station loads).

Table 2.3-5 Electrical Distribution System Load Changes

Motor Description	Nameplate HP	Required BHP		Analyzed BHP	
		CLTP	EPU	CLTP	EPU
Feedwater Pump	16,500	11,608	16,064	14,100	16,500
Condensate Booster Pump	3,000	2,238	2,410	3,000	3,000
Condensate Pump	1,750	1,556	1,624	1,846[2]	1,750
Heater Drain Pump	1,750[1]	1,406	1,516	1,536	1,750
Reactor Recirculation Pump (100% core flow)	8,900	8,043	8,104	8,750	8,900

Notes

1. The existing Nameplate rating for these motors is 1,500 HP. These motors will be replaced with 1,750 HP for EPU operation.
2. Conservative CLTP analyzed value based on abnormal operational configuration.

Table 2.3-6 NMP2 Station Blackout Sequence of Events

Time (sec)	Description of Event
0	Loss of Offsite Power Reactor scram MSIVs start to close Loss of Feedwater Loss of Service Water
3	MSIVs closed
5	Feedwater flow stops
5.3 to 6.9	SRVs open (relief mode)
120	Begin RPV manual pressure control reducing to 500 psig with one SRV
142	RCIC first startup w/ CST suction to maintain RPV Level (159 – 202")
1,211	RPV pressure reaches 500 psig SRV closed
4,222	RPV pressure reaches 1000 psig 1 SRV opened
5,028	RPV pressure reaches 500 psig SRV closed
8,035	RPV pressure reaches 1000 psig 1 SRV opened
8,831	RPV pressure reaches 500 psig SRV closed
~12,140	RPV pressure reaches 1000 psig 1 SRV opened
~12,922	RPV pressure reaches 500 psig SRV closed
14,400	End of Coping Period

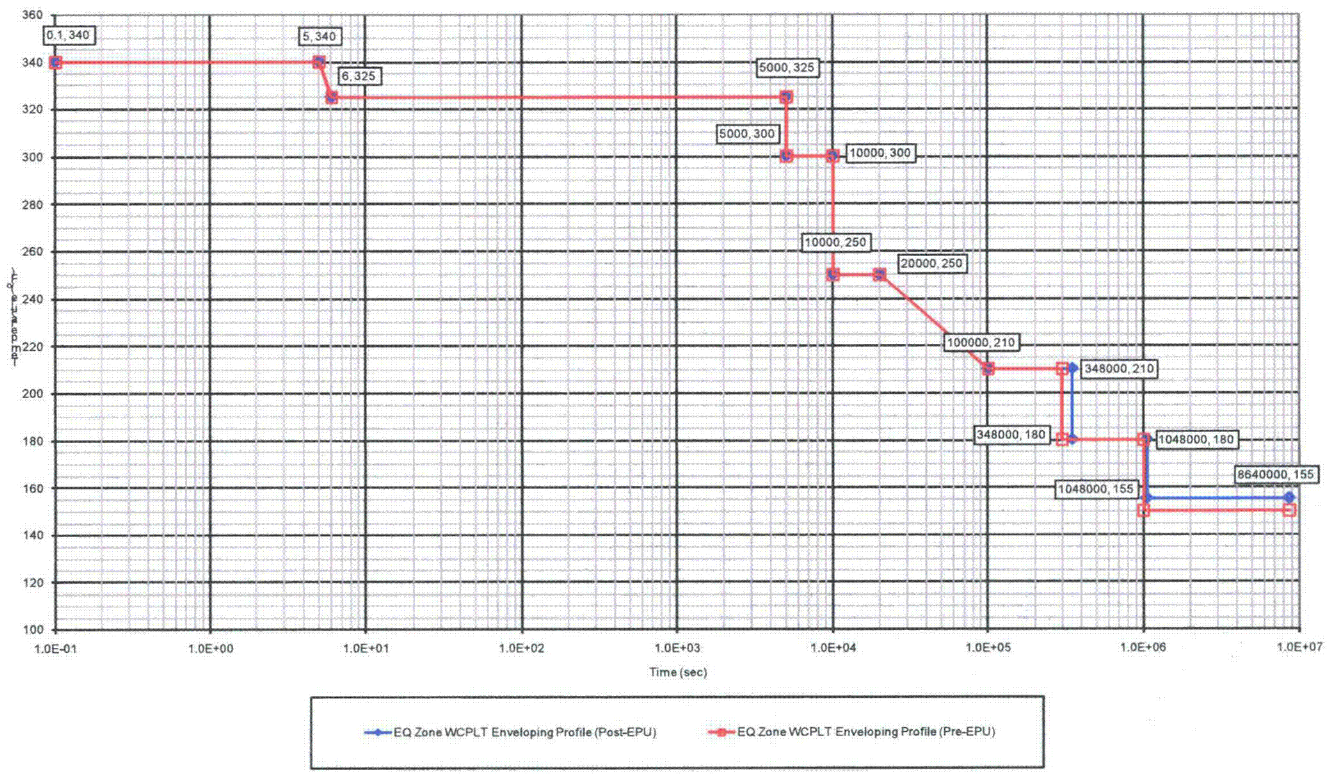

Figure 2.3-1 Worst Case Plant EQ Enveloping Temperature Profile EQ Zone WCPLT (All Plant EQ Zones and Elevations)

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. NMPNS conducted a review of the reactor trip system, engineered safety feature actuation system (ESFAS), safe shutdown systems, control systems, and diverse instrumentation and control systems for the proposed EPU to ensure that the systems and any changes necessary for the proposed EPU are adequately designed such that the systems continue to meet their safety functions. 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.

Technical Evaluation

The setpoint calculation methodology, safety limit-related limiting safety system setting determination, and instrument setpoint controls are discussed in Section 3.2 of the NMP2 License Amendment Request. A sample calculation is provided in PUSAR Section 2.4.2.

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of Constant Pressure Power Uprates. Section 5 of the CLTR addresses the effect of Constant Pressure Power Uprate on Reactor Protection, Safety Features Actuation, and Control Systems. The results of this evaluation are described below.

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 APRMs, Intermediate Range Monitors (IRMs), Local Power Range Monitors (LPRMs), RBM, and Rod Worth Minimizer (RWM).

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

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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 new licensed power (i.e., EPU RTP). 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 NMP2 are in accordance with the requirements established by the GEH design specifications. The specifications provide confirmation that the [[ ]] in the CLTR is met at EPU.

Local Power Range Monitors

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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 in-core 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. A small increase in radiation levels is accommodated by the radiation protection program for normal plant operation.

Reliability of LPRM instrumentation and accurate prediction of in-bundle pin powers typically requires operation with bypass voids lower than 5% at nominal conditions. The steady-state 5% bypass voiding evaluation is provided in Section 2.8.2.5.1.

The LPRMs and TIPs installed at NMP2 are in accordance with the requirements established by the GEH design specifications. The specifications provide confirmation that the [[
]] provided in the CLTR is met at EPU.

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 will be 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 NMP2 are in accordance with the requirements established by the GEH design specifications. The specifications provide confirmation that the [[ ]] provided in the CLTR is met at EPU.

Rod Worth Minimizer

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[illegible]	[illegible]	[illegible]
	• • •]]	• Nine Mile Point 2 uses all GE14 or earlier fuel. • Thermal power increase: OLTP – 3323 MWt CPPU – 3988 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. Therefore, no additional plant-specific information for the performance of this system relative to the normal operational function is required. The power-dependent instrument setpoints for the RWM are included in the plant Technical Specifications (see Section 2.4.1.3).

The RWM installed at NMP2 provides the same level of protection as described in the [[ ]] provided in the CLTR, because the fuel is GE14 or earlier, the power increase is ≤ 20%, the Banked Position Withdrawal Sequence (BPWS) is used, and the EPU LPSP is conservatively maintained at the same percent of RTP as the CLTP value.

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 EPU conditions. However, some (non-safety) modifications are needed to the power conversion systems to obtain EPU RTP. No safety-related BOP system setpoint change is required as a result of 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

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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

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The Turbine Steam Bypass System is a normal operating system that is used to bypass excessive steam flow. [[

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Feedwater Control System

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The FW 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. An LOFW event can be caused by downscale failure of the controls. The LOFW is discussed in Section 2.8.5.2.3.

The FW Control System at NMP2 is consistent with the generic description provided in the CLTR because the FW flow instrumentation and steam flow instrumentation will be recalibrated or replaced as shown in Table 2.4-2.

Leak Detection System

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The only effect on the LDS due to EPU is a slight increase in the FW system temperature and an increase in steam flow. The instrument setpoints associated with primary system leak detection are not affected by EPU, because the break flow does not change with the constant pressure assumption and the local temperatures are not significantly changed. The increased feedwater temperature does not result in a significant increase in the MS tunnel temperature. The normal operating drywell area temperature experiences negligible change for EPU conditions. The LDS temperature setpoints remain unchanged. There is no increase in WCS temperature and only a slight increase in system pressure, so there is no effect on the WCS temperature based leak detection. There is no change to the RCIC system temperature or pressure; therefore, the RCIC temperature based LDS is not affected. There is no increase in the RHS system temperature or pressure, therefore the RHS LDS is not affected. The flow-based LDS including MSL high flow are not affected by EPU. MSL high flow is discussed in Section 2.4.1.3.

The LDS at NMP2 is consistent with the [[ ]] provided in the CLTR because the temperatures and pressures for the monitored leak detection areas are unchanged or experience only a slight increase.

2.4.1.3 Technical Specification Instrument Setpoints

Technical Specification 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 reviewed to ensure that adequate operational flexibility and necessary safety functions are maintained at the EPU RTP level. An appropriate setpoint calculation is performed and Technical Specification changes are implemented, as required. [[

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]] The justification for implementing this simplified process for the individual Technical Specification setpoints is provided for each instrument below, as applicable. Implementing the constant maximum operating pressure requirement for EPU [[

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In addition, the following restrictions are imposed on the use of the simplified process to assure its validity. Its use is limited to:

- NRC approved GEH or plant-specific methodology.
- [[
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These restrictions are satisfied for NMP2, except where instrumentation is changed affecting the instrumentation errors. For the turbine first-stage pressure (TFSP) Scram Bypass Permissive, the setpoint function is affected by modifications to the high-pressure turbine. For this setpoint function, a new setpoint was calculated using the GEH methodology per Reference 30.

Table 2.4-1 summarizes the current and EPU ALs for NMP2.

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.

Main Steam Line High Flow Isolation

The MSL high flow isolation setpoint is used to initiate the isolation of the Group 1 primary containment isolation valves. The only safety analysis event that credits this trip is the MSLB. For this accident, there are diverse trips from high area temperature and high area differential temperature in the main steam tunnel. For NMP2, there is sufficient margin to choke flow, so the AL for EPU is maintained at the current percent of rated steam flow in each MSL.

For NMP2, the AL of 140% of rated steam flow is not changed and no new instrumentation is required (the existing instrumentation has the required upper range limit and calibration span the instrument loops need to accommodate the new setpoint). A new setpoint was calculated using the GEH methodology per Reference 30 and a Technical Specification AV change is required to change the differential pressure at the allowable steam flow.

Turbine First-Stage Pressure Scram and Recirculation Pump Trip Bypass

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 TFSP setpoint is used to reduce scrams and recirculation pump trips at low power levels where the turbine steam bypass system is effective for Turbine Trips (TTs) 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. [[

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Therefore, a new setpoint was calculated using the GEH methodology per Reference 30, and [[ ]] The AV (in psig) for NMP2 will be revised prior to EPU implementation.

To assure that the new value is appropriate, an EPU plant ascension startup test will be used to validate that the actual plant interlock is cleared consistent with the safety analysis.

APRM Flow Biased Scram

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Rod Worth Minimizer Low Power Setpoint

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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 parameter is steam flow.

[[

]] associated with the requirements of the CRDA.

Rod Block Monitor

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The severity of 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. [[

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NMP2 is confirmed to be consistent with the generic description provided in the CLTR and [[

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APRM Setdown in Startup Mode

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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.

Oscillation Power Range Monitor

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The Oscillation Power Range Monitor (OPRM) is designed to provide the Option III automatic scram and was designed to be independent of core power.

The OPRM trip is armed only when plant operation is within the OPRM trip-enabled region. The OPRM trip-enabled region is generically defined as the region on the power/flow map with power ≥ 30% of RTP and core flow < 60% of rated core flow. For EPU, the NMP2 OPRM 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% CLTP boundary is rescaled to the 26% EPU thermal power limit.

2.4.1.4 Changes to Instrumentation and Controls

In the CLTR SER, the NRC 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

NMPNS has reviewed the effects of the proposed EPU on the functional design of the reactor trip system, ESFAS, safe shutdown system, and control systems. NMPNS concludes that the effects of the proposed EPU on these systems have been adequately addressed and that the changes that are necessary to achieve the proposed EPU are consistent with the plant's design basis. NMPNS further concludes that the systems will continue to meet the requirements of 10 CFR 50.55a(a)(1), 10 CFR 50.55a(h), and GDCs 1, 4, 13, 19, 20, 21, 22, 23, and 24. Therefore, NMPNS finds the proposed EPU acceptable with respect to instrumentation and controls.

2.4.2 MSL High Flow Instrument Setpoint Sample Calculation

2.4.2.1 Function

Setpoint Characteristics:	Definition	Comments
Event Protection:	Limiting event for the setpoint: Detect and mitigate the consequences of a main steam line break by sending a signal to the Primary Containment Isolation System (PCIS) for Group 1 isolation (closure of the Main Steam Isolation Valves, MSIVs).	None
Function After Earthquake	☒ Required ☐ Not Required	None
Setpoint Direction	☒ Increasing ☐ Decreasing	None
Single or Multiple Channel	☐ Single ☒ Multiple	None
LER Calculation Basis if Multiple Channel	Standard (Conservative) LER Calculation ☒, or Configuration Specific LER Calculation ☐	None
Trip Logic for Configuration Specific LER Calculation	n/a	None

n/a: not applicable

Current Function Limits:	Value/Equation		Comments	
	Present Calculation (CLTP Conditions)	Power Uprate Conditions		
Analytical Limit	126.4 psid 140% flow	194.4 psid 140% flow	None	None
Tech Spec Allowable Value	122.8 psid	Results in Section 2.4.2.3	None	
Setpoint	121.5 psid	Results in Section 2.4.2.3	None	
Operational Limit	100.5 psid 127% flow	149.24 psid 127% flow	None	None

Plant Data:	Value Present Calculation	Value Power Uprate Condition	Sigma if not 2	Comments	
Primary Element • Accuracy (APEA)	± 0.75% of point = ± 1.7 psid	± 0.75% of point = ± 4.26 psid			
• Drift (DPEA)	negligible	negligible		None	Comment 7
Process Measurement Accuracy (PMA)	± 2.9 psid	5.59 psid [[ ]]		None	None

Components (or Devices) in Setpoint Function Instrument Loop:

- Venturi Flow Nozzle
- Flow Transmitter
- Analog Trip Module (Trip Unit)

2.4.2.2 Components

2.4.2.2.1 Flow Transmitter

Component Information:	Value/Equation	Comments
Plant Instrument ID No.	E31-N086-89A-D	None
Instrument vendor	Rosemount	None
Model ID No. (including Range Code)	1153DB7RJ (Range Code 7)	None
Plant Location(s)	Secondary Containment, Elevation 215 ft, EQ Zone 122	None
Process Element	Venturi	None

Inputs:

Vendor Specifications:	Value / Equation	Sigma if not 2	Comments
Top of Scale	+250 psid (20 mAdc, 5 Vdc)	n/a	Comment 8
Bottom of Scale	-50 psid (4 mAdc, 1Vdc)	n/a	Comment 8
Upper Range Limit	300 psid	n/a	
Accuracy	± 0.25% span	3	Comment 17
Temperature Effect	± {[(0.75% URL) + (0.5 % span)] / 100degF}*(dT)	3	
Seismic Effect	± 0.5% URL during and after 4g ZPA		Comment 9

Radiation Effect	± 4% URL during and after $2.2x10^7$ rads, TID Gamma		Comment 9
Humidity Effect	Included in accuracy	3	Comment 4
Power Supply Effect	< 0.005% span/volt	3	
RFI/EMI Effect	Included in accuracy	3	Comment 4
Insulation Resistance Effect	Negligible		Comment 4
Over-pressure Effect	± 3% URL after exceeding 2000 psi	3	
Mounting Position Effect: • Span Effect • Zero Shift	None ≤ 1.5 inH_2O (Calibrated out)	3	Comment 11
Static Pressure Effect • Zero Effect	± 0.5% URL / 1000 psi (Calibrated out for operating pressure after installation)	3	
• Span Effect	Systematic (Calibrated out for a particular pressure before installation)	n/a	
• Span Effect - Correction Uncertainty	± 0.5 %reading / 1000 psig	3	Comment 12; Comment 13

URL: Upper Range Limit
TID: Total Integrated Dose
ZPA: Zero Period Acceleration

Plant Data:	**Value**	**Sigma if not 2**	**Comments**
Calib Temperature Range	75 - 104 °F	n/a	None
Normal Temperature Range	65 - 104°F	n/a	None
Trip Temperature range	75 - 104°F	n/a	None
Plant seismic value	2g	n/a	None
Plant Radiation value	0.1 Mrad/hr; negligible (mild environment)	n/a	None
Plant Humidity value	20% - 50% RH (normal)	n/a	None
Power Supply Variation value	± 4.5 Vdc assumed	n/a	Comment 14
RFI/EMI value	Not provided	n/a	Comment 4
Over-pressure value	1325 psig	n/a	Comment 10
Static Pressure value	1500 psig	n/a	

RH: Relative Humidity

Drift:	Value	Sigma if not 2	Comments
Current Calib. Interval	24 Mo. ☐Includes extra 25%	n/a	None
Desired Calib. Interval	24 Mo. ☐Includes extra 25%	n/a	None
Drift Source	☐ Calculated ☒ Vendor	n/a	None
Drift Value	± 0.2% URL / 30 months		Comment 9

Calibration:	Value / equation	Sigma if not 3	Comments
As Left Tolerance	±0.25% of 4 Vdc + 0.01 Vdc (+0.04 mA)		None
Leave Alone Tolerance	±0.25% of 4 Vdc + 0.01 Vdc (+0.04 mA)		None
Input Calibration Tool:	Test Gauge		None
Accuracy	Assumed equal to Flow Transmitter Accuracy		None
Resolution / Readability			None
Minor Division		n/a	None
Upper Range		n/a	None
Temperature Effect			None
Input Calibration Standard:	Not provided		None
Accuracy	Assumed equal to $^{1}/_{4}$ Flow Transmitter Accuracy		Comment 6
Resolution / Readability			None
Minor Division		n/a	None
Upper Range		n/a	None
Temperature Effect			None
Output Calibration Tool:	Digital MultiMeter (DMM)		None
Accuracy	Assumed equal to Flow Transmitter Accuracy		None
Resolution / Readability			None
Minor Division		n/a	None
Upper Range		n/a	None
Temperature Effect			None

Output Calibration Standard:	Not provided		None
Accuracy	Assumed equal to $^1/_4$ Flow Transmitter Accuracy		Comment 6
Resolution / Readability			None
Minor Division		n/a	None
Upper Range		n/a	None
Temperature Effect			None

2.4.2.2.2 Analog Trip Module (Trip Unit)

Component Information:	**Value/Equation**	**Comments**
Plant Instrument ID No.	E31-N686-89A·D	None
Instrument vendor	Rosemount	None
Model ID No. (including Range Code)	510DU	None
Plant Location(s)	Control Building, Elevation 306 ft, EQ Zone 312	None
Process Element	n/a	None

Inputs:

Vendor Specifications:	**Value / Equation**		**Sigma if not 2**	**Comments**	
Top of Scale	20 mAdc (5 Vdc)/ +250 psid		n/a	None	
Bottom of Scale	4 mAdc (1 Vdc)/ -50 psid		n/a	None	
Upper Range Limit	300 psid		n/a	None	
Accuracy	<u>510DU</u>	<u>710DU</u>			
▪ Adjustability	± 0.01 mAdc	± 0.01 mAdc			
▪ Trip Repeatability	± 0.13% span to ± 0.3% span (depends on Operating Condition & Environment)	± 0.13% span			
▪ Analog Output	n/a	n/a	3	Comment 15; Comment 16	Comment 15; Comment 16
Temperature Effect	Included in accuracy (between 40-104°F for 510DU; not provided for 710DU).			Comment 4	

Seismic Effect	Included in accuracy (none <11g for 510DU; not provided for 710DU)		Comment 4
Radiation Effect	Negligible (mild environment)		Comment 4
Humidity Effect	Included in accuracy (none 20-90 RH for 510DU; not provided for 710DU)		Comment 4
Power Supply Effect	Negligible		Comment 4
RFI/EMI Effect	Included in accuracy		Comment 4
Insulation Resistance Effect	Negligible		Comment 4
Over-pressure Effect	n/a		Comment 5
Static Pressure Effect	n/a		Comment 5

Plant Data:	**Value**	**Sigma if not 2**	**Comments**
Calib Temperature Range	68 - 79 °F	n/a	None
Normal Temperature Range	40 - 156 °F	n/a	None
Trip Temperature range	60 - 90 °F	n/a	None
Plant seismic value	4 g	n/a	None
Plant Radiation value	n/a	n/a	None
Plant Humidity value	20 - 50 % RH (normal)	n/a	None
Power Supply Variation value	Not provided	n/a	Comment 4
RFI/EMI value	Negligible (mild environment)	n/a	Comment 4
Over-pressure value	n/a	n/a	Comment 5
Static Pressure value	n/a	n/a	Comment 5

Drift:	**Value**	**Sigma if not 2**	**Comments**
Current Calib. Interval	92 days. ☐Includes extra 25%	n/a	None
Desired Calib. Interval	92 days ☐Includes extra 25%	n/a	None

Drift Source	☐Vendor ☒Calculated		n/a	None	
Drift Value	510DU ± 0.13% span / 6 months	710DU ± 0.13% span / 6 months		None	Comment 9; Comment 15

Calibration:	Value / equation	Sigma if not 3	Comments
As Left Tolerance	0.13% span		None
Leave Alone Tolerance	0.13% span		None
Input Calibration Tool:	RCU & Readout		
Accuracy	Assumed equal to Analog Trip Unit (ATU) Accuracy		None
Resolution / Readability			None
Minor Division		n/a	None
Upper Range		n/a	None
Temperature Effect			None
Input Calibration Standard:	Not provided		
Accuracy	Assumed equal to $^1/_4$ ATU Accuracy		None
Resolution / Readability			None
Minor Division		n/a	None
Upper Range		n/a	None
Temperature Effect			None
Output Calibration Tool:	Light Emitting Diode (LED)		
Accuracy	n/a		None
Resolution / Readability	n/a		None
Minor Division	n/a		None
Upper Range	n/a		None
Temperature Effect	n/a		None
Output Calibration Standard:	n/a		
Accuracy	n/a		None
Resolution / Readability	n/a		None
Minor Division	n/a		None
Upper Range	n/a		None
Temperature Effect	n/a		None

RCU: Rosemount Calibration Unit

2.4.2.3 Summary Results

Setpoint Function	Analytic Limit (from Section 1)	Allowable Value	Setpoint†	Meets LER Avoidance Criteria	Meets Spurious Trip Avoidance Criteria
Main Steam Line High Flow Group 1 Isolation	194.4 psid 140% flow	184.4 psid	183.0 psid	Y	Y

†Excludes head correction

2.4.2.4 Comments and Recommendations

1. Unless specifically identified as "bias" errors in this document, all instrument uncertainty errors will be considered to be random in nature, even when the "±" symbol is not shown.
2. Some plant specific information has not been provided in the current Nine Mile Point-2 setpoint calculation(s) or documents and is considered unnecessary because the impact of this information is included within the instrument accuracy values within the current Nine Mile Point-2 setpoint calculation(s) or documents.
3. Unless specified otherwise, all calibration tool errors are considered to include resolution / readability errors and temperature effect errors. For the errors of assumed calibration standards, the temperature effect term is considered to be included in the accuracy of the assumed standard. The resolution/readability error for assumed calibration standards does not apply, because it is not actually read.
4. Temperature effect, radiation effect, seismic effect, humidity effect, power supply effect, Radio Frequency Interference/ Electromagnetic Interference (RFI/EMI) effect, and Insulation Resistance Effect (IRE) errors are marked "negligible," "not provided," or "included in accuracy" and are considered to have negligible or no impact on the manufacturer's accuracy terms when they are not identified separately.
5. Overpressure effects are only applicable to certain pressure measurement devices (e.g., differential pressure transmitters), and static pressure effects are only applicable to certain differential pressure measurement devices. These effects are marked "n/a" for other devices.
6. Where a specific input was not provided on the Calibration Standards, an assumed inaccuracy ratio of one-quarter (i.e., $^1/_4$) that of the Device inaccuracy is used. "Accuracy ratio between the Lab Standards utilized to calibrate the device and the device under test is a minimum of 4 to 1 or, if less than 4 to 1 the actual ratio noted and a reason (basis) indicated as to why the 4 to 1 could not be met."
7. The PEA is assumed to be independent of time because it is assumed that the flow primary elements do not degrade with time. Therefore, the drift component of the PEA is considered to be negligible.
8. The calibrated span for the Flow Transmitter in the current setpoint calculation of record is from 0 psid to 150 psid, resulting in a Calibrated Span (SP) equal to 150 psid. However, based on, the lower limit is -50 psid and the upper limit is +250 psid. This calibrated span,

equal to 300 psid, encompasses the Analytical Limit (AL), which is well within the positive span of 250 psid. Therefore, the AL, Allowable Value (AV), and the Nominal Trip Setpoint (NTSP) will be well within instrument scale.

9. The current approach in GEH setpoint calculation methodology treats the Drift Effect, Radiation Effect and the Seismic Effect for this instrument to be a 2 sigma values.
10. For conservatism, 1325 psig, the Reactor Pressure Vessel safety limit is used as the possible over-pressure value for the Flow Transmitters.
11. The Mounting Position Effect for Zero Shift can be calibrated out. It has been calibrated out at NMP2.
12. The zero shift with static pressure can be trimmed out after installation of the Model 1153 Series B Pressure Transmitter with the unit at operating pressure, if the calibrated range includes 0 psid, as does this application for MSL Flow measurements. It has been trimmed out at NMP2.
13. The Static Pressure Span Effect is systematic and can be calibrated out for a particular pressure before installation. It has been trimmed out at NMP2. If a differential transmitter is calibrated with the low side at ambient pressure, but will be used at high line pressure, the span adjustment is corrected to compensate for the effect of static pressure on the unit. If zero is elevated or suppressed, the zero adjustment is also corrected. This applies at NMP2 and has been corrected.
14. For conservatism, and based on historical observations, a Power Supply voltage variation of ± 4.5 Vdc is assumed for the Flow Transmitters.
15. The inaccuracy for Adjustability of the 510DU Trip Unit is the same as for the 710DU Trip Unit.
16. The analog output inaccuracy for the Trip Unit does not apply because there is no Slave Trip Unit related to the trip setpoint signal sent to the Primary Containment Isolation System (PCIS), but instead a digital output from the Master Trip Unit sent to a PCIS relay.
17. The accuracy for the Differential Pressure Transmitter includes the combined effects of linearity, hysteresis, and repeatability.
18. Because the calibration tolerance terminology at NMP2 differs slightly from GEH Instrument Setpoint Methodology, the definitions for the terms used in this document are provided here.
    a. As Left Tolerance (ALT): This is the tolerance within which the device calibration reading is left after calibration.
    b. Leave Alone Tolerance (LAT): This is the tolerance within which calibration need not be performed, and is intended to allow for normal variations in instrument readings due to accuracy and drift. Note that at NMP2, the LAT is equal to the ALT.
19. Transfer functions used in this calculation:

Flow Transmitter	Output (mA) linearly converted from input (psid).

Analog Trip Module: Comparison of flow signal (mA, equivalent to psid) with a reference.

Table 2.4-1 Analytical Limits for Technical Specification Setpoints

	Analytical Limits			
Parameter	**Current**		**EPU**	
APRM Calibration Basis (MWt)	3467		3988	
APRM Neutron Flux High Scram (% RTP)	123		No Change	
APRM High Flux Flow Biased Simulated Thermal Power (STP) Scram ALs[1]				
Two Loop Operation (TLO) (%RTP)	$0.64\ W + 66.8$[4]		$0.55\ W + 63.5$[4]	
SLO (%RTP)	$0.58\ (W - \Delta W) + 65$[4] $0.58\ W + 62.1$[4]		$0.50\ (W - \Delta W) + 56.5$[4,6] $0.50\ W + 54.0$[4,6]	
Clamp (% RTP)	118		No Change[2]	
APRM Setdown in Startup Mode (%RTP)				
Scram AV	20		No Change	
Rod Block Monitor	Unfiltered	Filtered	Unfiltered	Filtered
Low Power Setpoint (Enable) (%RTP)	30	30	No Change	
Intermediate Power Setpoint (%RTP)	65	65	No Change	
High Power Setpoint (%RTP)	85	85	No Change	
Low Trip Setpoint (% Reference Level)	127	125.8	No Change[3]	
Intermediate Trip Setpoint (% Reference Level)	122	121	No Change[3]	
High Trip Setpoint (% Reference Level)	117	116	No Change[3]	
RBM Downscale Setpoint (% Reference Level)	75.8		No Change	
Rod Worth Minimizer LPSP (%RTP)	**10**		**No Change**	
Main Steam Line High Flow Isolation (% rated steam flow)	140		No Change	
Turbine First-Stage Pressure Scram Bypass (%RTP)	30		26	
Reactor Vessel Water Level – Low, Level 3 Scram (in. H_20 AWLZ)[5]	154.3		148.0	

Notes:

1. No credit is taken in any safety analysis for the flow referenced setpoints.
2. The EPU APRM STP scram 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 OLMCPR of 1.35. The trip setpoints corresponding to other OLMCPR values also would remain the same for EPU.
4. W is the Recirculation Drive Flow in percent of Rated flow. ΔW is the difference in % Drive flow between the TLO and SLO Recirculation Drive flow at the same Core flow. The TLO ΔW is 0%, and the SLO ΔW is 5%.
5. The AV and NTSP are not changed for EPU for this setpoint function. This AL change is to resolve the issue with Steam Flow Induced Error (SFIE; also called "Bernoulli error") in the case that the Steam Dryer skirt becomes uncovered for a Loss of Feedwater (LOFW), per the related Safety Communication SC04-14 (Reference 31). AWLZ means Above Water Level Zero. Water Level Zero = 380.7 inches Above Vessel Zero. Units used in the NMP2 Technical Specifications are "inches," equivalent to "in. H_2O AWLZ" here.
6. The Analytical Limits for SLO operation are unchanged in terms of MWt.

Table 2.4-2. Changes to Instrumentation and Controls

Parameter	EPU Change
MSL High Flow	Respan transmitters and associated loop components Revise setpoints
1st Stage Turbine Pressure	Respan transmitters, indicators and associated loop instruments Revise setpoints
APRM flow biased STP scram	Revise APRM setpoints
APRM flow biased STP rod block	Revise APRM setpoints
RWM LPSP	Revise setpoints
Crossaround Steam Pressure	Respan transmitters, indicators and associated loop instruments Adjust EHC power to load comparator
High Pressure Turbine (HPT) Exhaust Pressure	Respan transmitters, indicators and associated loop instruments
MS Drain Receiver Outlet Temperature	Respan indicators and associated loop instruments
Inlet Press to Low Pressure Turbine (LPT)	Respan transmitters, recalibrate instrument loop
MSR Outlet Pressures	Replace transmitters and respan associated loop instruments Revise alarm setpoints
Main Steam Temperature	Respan indicators and associated loop instruments
Condensate Polisher Flow Low Alarm	Revise setpoints
Condensate Polisher and Strainer ΔP	Revise setpoints
Condensate, Condensate Booster Pump, and FW Pump Pressures, Temperatures, & Flows	Respan indicators and associated loop instruments Revise setpoints
SRV Discharge Temperature	Respan indicators Revise alarm setpoint
Turbine Steam Bypass Outlet Temperature	Respan indicators Revise alarm setpoint
Turbine Condenser Vacuum (Alarm Low)	Revise low vacuum alarm setpoint
MSR Outlet Temperatures	Respan indicators Revise alarm setpoint
Main Steam Inlet Header Pressure	Replace transmitters and respan associated loop components Revise alarm setpoint

Parameter	EPU Change
MSL Flow	Replace transmitters and respan associated loop components Revise alarm setpoint
FWH Temperatures	Respan indicators and computer points Revise alarm setpoints
RMS/CMS/MSS/Various Radiation Monitors	Setpoints are based on background radiation input which will be evaluated and revised as required during EPU power ascension
Feedwater Pump Motors	Replace ammeters and revise protective relay settings
Feedwater Flow to Reactor	Respan transmitters, indicators and associated loop instruments
Final Feedwater Pressure to Reactor	Respan transmitters, indicators and associated loop instruments
Feedwater Flow Differential Pressure (for H2 flow control module)	Respan transmitters, indicators and associated loop instruments
Feedwater to Reactor Temperatures	Respan indicators and revise alarm setpoints
RFP Recirculation Temperatures	Respan indicators and revise alarm setpoints
Reheater Drain Temperatures	Respan indicators and revise alarm setpoints
FWH Extraction Steam Pressures	Respan indicators and revise alarm setpoints
FWH 4 HDP 1A/B/C Suction Pressure	Revise setpoints
HDP Recirculation Control	Respan instrument loop
Scavenging Steam Line Pressure and Temperatures	Respan transmitters, indicators and associated loop instruments Revise alarm setpoint
Reheater Shell Pressures	Respan instrument loop Revise computer points
Off Gas Recombiner Outlet Temperature	Revise high alarm and trip setpoints
Main Turbine Load and Load Set Meters	Respan indicator scales

2.5 Plant Systems

2.5.1 Internal Hazards

2.5.1.1 Flooding

2.5.1.1.1 Flood Protection

Regulatory Evaluation

NMPNS conducted a review in the area of flood protection to ensure that SSCs important to safety are protected from flooding. The NMPNS review covered flooding of SSCs important to safety from internal sources, such as those caused by feedwater line breaks and moderate energy line cracks. The NMPNS review focused on increases of fluid volumes in tanks and vessels assumed in flooding analyses to assess the effect of any additional fluid on the flooding protection that is provided. The NRC's acceptance criteria for flood protection are based on GDC2.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of Constant Pressure Power Uprates. Section 10.2 of the CLTR addresses the effect of Constant Pressure Power Uprate on moderate energy line breaks. The results of this evaluation are described below.

Internal Flooding from Feedwater Line Break

Components and/or equipment required for safe shutdown of the reactor were evaluated for the effects of flooding from breaks and cracks in high-energy lines. The evaluations verified that the plant can be safely shut down, assuming a concurrent single active failure in systems necessary to mitigate the consequences of the postulated component failure. Plant flooding due to internal piping failures in the FW system is evaluated for changes due to EPU. Flooding is conservatively evaluated based on the entire hotwell volume being released in the Main Steam Tunnel and then draining to the reactor building. Since no changes are made to the existing hotwell inventory, draining systems, and flood barriers; the flood levels in the reactor building due to a FW break are unchanged. The current design basis requires break detection and manual isolation of the steam tunnel drain lines in 100 minutes. This requirement is unchanged for EPU. Evaluations of the remaining high energy systems determined that flooding effects from high-energy pipe breaks and cracks outside of containment and failure of non-seismic tanks and vessels are enveloped by moderate-energy crack flooding. This is primarily due to rapid detection and isolation of high-energy pipe breaks and cracks based on automatic isolation on area high temperature. WCS break isolation for WCS pipe breaks is not impacted and therefore, the flooding effects remain bounded by moderate energy cracks. The remaining systems evaluated are not impacted by EPU and remain bounded by the current flooding analyses. Therefore, internal flooding due to postulated failures in piping systems is not impacted by EPU.

Moderate Energy Line Cracks

Moderate energy line cracks (MELCs) are evaluated for their effects on flooding [[ ]] and equipment qualification (see Section 2.3.1).

[[

		]]

System design limits (design pressure) used as input to the MELC flooding analyses are not changed by EPU. Therefore, the NMP2 MELC internal flooding evaluations are not affected by EPU.

Conclusion

NMPNS has reviewed the EPU impact to flooding of SSCs important to safety from internal sources and has determined that failure of non-seismic tanks and vessels are enveloped by moderate-energy crack flooding which is not affected by EPU. NMPNS concludes that SSCs important to safety will continue to be protected from flooding and will continue to meet the requirements of GDC2 following implementation of the proposed EPU. Therefore, NMPNS finds the proposed EPU 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 leak offs, 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 NMPNS review of the EFDS included the collection and disposal of liquid effluents outside containment. The NMPNS review focused on any changes in fluid volumes or pump capacities that are necessary for the proposed EPU and are not consistent with previous assumptions with respect to floor drainage considerations. 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).

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of Constant Pressure Power Uprates. Section 8.1 of the CLTR addresses the effect of Constant Pressure Power Uprate on CLTR Liquid and Solid Waste Management. The results of this evaluation are described below.

The floor drain collector subsystem and the waste collector equipment drain 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. The design of the NMP2 equipment and floor drains inside and outside of containment has been evaluated to ensure any EPU-related liquid radwaste increases can be processed. NMP2 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 and infiltration of radioactive water into non-radioactive water drains 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

NMPNS has reviewed the assessment of the effects of the proposed EPU on the EFDS and concludes that the plant changes resulting in increased water volumes and larger capacity pumps or piping systems have been adequately addressed. NMPNS concludes that the EFDS has sufficient capacity to (1) handle the additional expected leakage resulting from the plant changes, (2) prevent the backflow of water to areas with safety-related equipment, and (3) ensure that contaminated fluids are not transferred to non-contaminated drainage systems. Based on this, NMPNS concludes that the EFDS will continue to meet the requirements of GDCs 2 and 4 following implementation of the proposed EPU. Therefore, NMPNS finds the proposed EPU 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. NMPNS's review of CWS focused on the impact to flooding analysis due to the proposed EPU. The NRC's acceptance criteria for the CWS are based on GDC4 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.

Technical Evaluation

The main condenser, circulating water, and heat sink systems are not being modified for EPU operation. The performance of these systems was evaluated for EPU. This evaluation was based on a design duty over the actual range of circulating water inlet temperatures, and confirms that the condenser, circulating water system, and heat sink are adequate for EPU operation. The evaluation of the circulating water system for normal environmental conditions at EPU power indicates sufficient system capacity to ensure that the plant maintains adequate condenser backpressure while meeting all environmental permit conditions related to the ultimate heat sink (UHS) and the plant cooling tower. Condenser hotwell temperature limitations may require load reductions at the upper range of the anticipated circulating water inlet temperatures. The effect of EPU on the flooding analyses is addressed in Section 2.5.1.1.1.

Conclusion

NMPNS has determined that modifications are not required for the CWS to perform its design function. For this reason, NMPNS concludes that, consistent with the requirements of GDC4, fluid leakage from the CWS will not result in the failure of safety-related SSCs following implementation of the proposed EPU. Therefore, NMPNS finds the proposed EPU acceptable with respect to the CWS.

2.5.1.2 Missile Protection

2.5.1.2.1 Internally Generated Missiles

Regulatory Evaluation

The NMPNS review concerns missiles that could result from in-plant component overspeed failures and high-pressure system ruptures. The NMPNS review of potential missile sources covered pressurized components and systems, and high-speed rotating machinery. The NMPNS review was conducted to ensure that safety-related SSCs are adequately protected from internally generated missiles. In addition, for cases where safety-related SSCs are located in areas containing non-safety related SSCs, NMPNS reviewed the non-safety related SSCs to ensure that their failure will not preclude the intended safety function of the safety-related SSCs. The NMPNS review focused on any increases in system pressures or component overspeed conditions that could result during plant operation, anticipated operational occurrences, or changes in existing system configurations such that missile barrier considerations could be affected. 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 GDC4.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of Constant Pressure Power Uprates. Section 7.1 of the CLTR addresses the effect of

Constant Pressure Power Uprate on the turbine generator. The results of this evaluation regarding turbine missiles are described below.

The high-pressure and low-pressure turbine rotors at NMP2 (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; however the new rotor train has been evaluated to verify that the probability of turbine missile generation remains within the guidelines of NUREG-1048 (Reference 32) and RG 1.115 (Reference 33).

This review criterion is applicable to EPUs that result in substantially higher system pressures or changes in existing system configuration. The NMP2 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 NMP2 EPU does not entail any changes in equipment configurations that could change the effect of internally generated missiles on safety-related or non-safety related equipment.

Conclusion

NMPNS has reviewed the changes in system pressures and configurations that are required for the proposed EPU and concludes that SSCs important to safety will continue to be protected from internally generated missiles and will continue to meet the requirements of GDC4 following implementation of the proposed EPU. Therefore, NMPNS finds the proposed EPU acceptable with respect to 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 combined intercept valves, and extraction steam non-return 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 NMPNS review of the turbine generator focused on the effects of the proposed EPU on the turbine overspeed protection features to ensure that a turbine overspeed condition above the design overspeed is very unlikely. The NRC's acceptance criteria for the turbine generator are based on GDC4, 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.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of Constant Pressure Power Uprates. Section 7.1 of the CLTR addresses the effect of Constant Pressure Power Uprate on the turbine generator. The results of this evaluation are described below.

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 was 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 turbine-generator was originally designed with a flow margin of 4.8%. The current rated throttle steam flow is 13.58 Mlbm/hr at a throttle pressure of 1003 psia. The generator is rated at 1399.22 MVA, which results in a rated electrical output (gross) of 1259.3 MWe at a power factor of 0.90 and a reactive power of 609.9 MVAR.

At the EPU RTP and reactor dome pressure of 1035 psia, the turbine operates at an increased rated throttle steam flow of 16.12 Mlbm/hr and at a throttle pressure of 991 psia. A flow margin of 5% is used in designing the new high pressure turbine section. The design point of the new turbine will include this flow margin in order to ensure that the turbine will be able to pass the rated throttle, as well as to allow sufficient margin for reactor pressure control. The VWO condition therefore refers to the turbine supply steam flow at 5% over rated condition (i.e. rated flow + 5%). For operation at EPU, the high-pressure turbine has been redesigned with new diaphragms and buckets for at least the minimum target throttle flow margin, to increase its flow passing capability.

Generator components were evaluated to identify the impact of the steam turbine uprate on the generator. This generator has the highest rating for this frame size generator and has no additional capability above the present generator rating of 1399 MVA. The generator will support the steam turbine uprate to rated 120% OLTP - 1368.9 MWe at approximately 0.98 power factor (PF) with a reactive power output of approximately 240 MVAR. It can be seen from the Reactive Capability Curve (Figure 2.5-5) that the generator would be operating at a power factor for the 120% Steam Turbine uprate of approximately 0.98 PF.

The expected environmental changes such as diurnal heating and cooling effects changing cycle efficiency may 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. Operating in a generator-limited mode will require increased operator attention to these limits. NMPNS will establish appropriate administrative controls to maintain the generator within operating limits.

The current NMP2 LP rotors are monoblock and the replacement high pressure rotor will be monoblock as well. Monoblock rotors cause no increase in missile failure probability due to the uprate. Per CLTR Section 7.1, a separate rotor missile analysis is not required for plants with integral wheels; therefore, the probability of turbine missile generation remains within the limits of NUREG-1048 (Reference 32) and RG 1.115 (Reference 33).

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 EPU conditions. The hardware modification design and implementation process establishes the overspeed trip settings to provide protection for a turbine trip.

The NMP2 EPU turbine design 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 non-safety related equipment.

Conclusion

NMPNS has reviewed the assessment of the effects of the proposed EPU on the turbine generator and concludes that the effects of changes in plant conditions on turbine overspeed have been adequately addressed. NMPNS concludes 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 GDC4 following implementation of the proposed EPU. Therefore, NMPNS finds the proposed EPU acceptable with respect to the turbine generator.

2.5.1.3 Pipe Failures

Regulatory Evaluation

NMPNS 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 NMPNS review of pipe failures included high and moderate energy fluid system piping located outside of containment. The NMPNS review focused on the effects of pipe failures on plant environmental conditions, control room habitability, and access to areas important to safe control of post-accident operations where the consequences are not bounded by previous analyses. The NRC's acceptance criteria for pipe failures are based on GDC4, 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.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of Constant Pressure Power Uprates. Sections 9.2.1, 10.1, 10.2, and 10.3 of the CLTR address the effects of Constant Pressure Power Uprate on Piping Failures. The results of this evaluation are described below.

High Energy Piping Outside Containment

Where EPU resulted in increased piping stresses in high energy piping outside containment the increased stresses were evaluated against existing line break criteria to identify any potential new break locations. The results of that evaluation (see Section 2.2.1) determined that there are no new high energy line break locations outside containment due to operation at EPU conditions.

Existing high energy line break locations outside containment that are affected by EPU are identified in Section 2.2.1 with the effects summarized in Table 2.2-1. Since the EPU post-HELB mass release and environmental conditions (pressures and temperatures) are bounded by the existing CLTP analyses (see Table 2.2-1), there is no adverse affect to post-HELB control room habitability or on access to areas important to safe control of post-accident operations.

The CLTP mass and energy releases for FW line breaks are affected by changes in the FW system including increased FW flow rate and modifications to the FW pumps. The mass and energy releases for double-ended breaks and critical cracks in the FW lines were re-analyzed at EPU conditions. At EPU, the flashing portion of the mass release into the Main Steam Tunnel was found to increase by approximately 4%. However, the effects of a Feedwater System Line Break on Main Steam Tunnel peak pressures and temperatures are bounded by a Main Steam Line Break in the Main Steam Tunnel.

The ability of the plant to cope with the flooding effects from HELBs outside containment that are affected by EPU is evaluated in Section 2.5.1.1.

Moderate Energy Piping Outside Containment

As stated in Section 2.5.1.1, system design limits (design pressure) used as input to the Moderate Energy Line Crack (MELC) flooding analyses and are not changed by EPU. Since, the NMP2 MELC mass releases and environmental conditions (pressures and temperatures) are not affected by the EPU there is no adverse impact to post-MELC control room habitability or on access to areas important to safe control of post-accident operations.

Environmental Conditions

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 HELB pressure profiles for CLTP conditions were determined to be bounding for EPU conditions. The peak HELB temperatures at EPU RTP are bounded by the values used for equipment qualification at CLTP conditions.

Details regarding analyses pertaining to the above environmental conditions are addressed in Section 2.3.1.

Radiological Consequences

The radiological consequences of a Main Steam Line Break outside containment are dispositioned in the CLTR.

EPU does not have a significant effect on the radiological consequences for an MSLBA.

Details regarding analyses pertaining to the above radiological consequences are addressed in Section 2.9.2. It concludes that EPU does not have a significant effect on the radiological consequences of an MSLBA because:

Activity concentration in the primary coolant is limited by Technical Specifications which are unaffected by power uprate.

The liquid and steam masses released during the accident are unchanged due to EPU.

Conclusion

NMPNS has reviewed the changes that are necessary for the proposed EPU and the NMP2 proposed operation of the plant, and concludes 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 GDC4 following implementation of the proposed EPU. Therefore, NMPNS finds the proposed EPU 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 NMPNS review focused on the effects of the increased decay heat on the plant's safe shutdown analysis to ensure that SSCs required for the safe shutdown of the plant are protected from the effects of the fire and will continue to be able to achieve and maintain safe shutdown following a fire. The NRC'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) GDC3, 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) GDC5, 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.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of Constant Pressure Power Uprates. Section 6.7 of the CLTR addresses the effect of Constant Pressure Power Uprate on the fire protection program. The results of this evaluation are described below.

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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 defined. With these implemented, the fire protection systems and analyses are sufficient to support EPU. EPU is found to not effect the elements of the fire protection plan related to (1) administrative controls, (2) fire suppression and detection systems, (3) fire barriers, (4) fire protection responsibilities of plant personnel, and (5) procedures and resources necessary for the repair of systems required to achieve and maintain cold shutdown. In addition, the increase in decay heat will not result in an increase in the potential for a radiological release resulting from a fire.

The reactor and containment response to the postulated 10 CFR 50 Appendix R fire event at EPU conditions is evaluated in Section 2.5.1.4.1. 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. It should be noted that this analysis represents a change in acceptance criterion from water level never going below top of active fuel, to assurance that no fuel perforation occurs by assuring that PCT remains below 1500°F. Therefore, the fire protection systems and analyses are not adversely affected by EPU.

Section 2.6.1.1 documents that cold shutdown is achieved within the 54-hour design value under Alternate Shutdown Cooling. Upon pseudo-LPCI injection (NMP2 USAR 9B.4.3) the long term cooling performance of the Appendix R fire event is analogous to Alternate Shutdown Cooling, and as such it can be concluded that the 72-hour cold shutdown as stipulated by Appendix R is met.

2.5.1.4.1 10 CFR 50 Appendix R Fire Event

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]]. The limiting Appendix R fire event was analyzed under both CLTP and EPU conditions. The fuel heatup analysis was performed using the SAFER/GESTR-LOCA analysis model. The containment analysis was performed using the SHEX model. 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 results of the Appendix R evaluation for CLTP and EPU provided in Table 2.5-1 and Figures 2.5-1 through 2.5-4 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. 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 (MCR) 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

NMPNS has reviewed the fire-related safe shutdown assessment and concludes that the effects of the increased decay heat on the ability of the required systems to achieve and maintain safe shutdown conditions have been adequately evaluated. NMPNS further concludes that the FPP will continue to meet the requirements of 10 CFR 50.48, Appendix R to 10 CFR Part 50, and GDCs 3 and 5 following implementation of the proposed EPU. Therefore, NMPNS finds the proposed EPU 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 NMPNS review for fission product control systems and structures covered the basis for developing the mathematical model for DBLOCA dose computations, the values of key parameters, the applicability of important modeling assumptions, and the functional capability of ventilation systems used to control fission product releases. The NMPNS review primarily focused on any adverse effects that the proposed EPU may have on the assumptions used in the analyses for control of fission products. The NRC's acceptance criteria are based on GDC41, 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.

Technical Evaluation

The assumptions regarding leakage and exhaust paths from the primary and secondary containments and other sources are described in detail in a separate license amendment which implemented the Alternative Source Term (AST) methodology for NMP2 (Reference 34). This amendment is based on 4067 MWt (corresponds to the EPU power level of 3988 MWt with a 2% ECCS evaluation uncertainty factor applied) and complies with RG 1.183.

Details of the radiological transport models used in the AST amendment are provided in References 34 & 35 and include changes from previous assumptions (i.e., pre-AST submittal assumptions) with regard to the fission product control systems and structures. These revised assumptions included values used for primary containment leakage to the secondary containment, primary containment bypass leakage directly to the environment, engineered safety feature leakage from the suppression pool to secondary containment, primary containment purge leakage, drawdown time, and SGTS flow. No credit was assumed for suppression pool scrubbing.

In addition, Section 4.5 of NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR), addresses the effect of EPU on the Standby Gas Treatment System. The results of this evaluation are described below.

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.

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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 (Reference 36).

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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.

Two bounding analyses have been performed in the CLTR to evaluate 1) systems that implement Alternate Source Term (AST) in accordance with RG 1.183 (Reference 37) and 2) systems that are committed to RG 1.3 (Reference 38) for fission product transport. [[

]] The parameters and their bounding values with a comparison to the NMP2 specific values are shown in Table 2.5-2.

The results of the 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. [[

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Along with the aforementioned charcoal ignition temperature (which is above the CLTR maximum component temperature and therefore acceptable), the fuel iodine inventory for the analyzed fuel load at EPU conditions is outside the applicable parameter of the CLTR bounding analysis (see Table 2.5-2). This is derived from and is representative of the accident source term for the NMP2 EPU conditions. That source term is used in the specific NMP2 SGTS analysis. The remaining parameters used in the CLTR bounding analysis for AST application are confirmed to bound the NMP2 plant-specific values. Therefore, with the exceptions mentioned above, the SGTS at NMP2 is confirmed to be consistent with the generic description provided in the CLTR.

Conclusion

NMPNS has reviewed the effects of the proposed EPU on fission product control systems and structures. NMPNS concludes that it has adequately accounted for the increase in fission products and changes in expected environmental conditions that would result from the proposed EPU. NMPNS further concludes that the fission product control systems and structures will continue to provide adequate fission product removal in post-accident environments following implementation of the proposed EPU. Based on this, NMPNS also concludes that the fission product control systems and structures will continue to meet the requirements of GDC41. Therefore, the NMPNS finds the proposed EPU 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) consists of two subsystems: (1) the "hogging" or startup system, which initially establishes main condenser vacuum, and (2) the Steam Jet Air Ejector System (SJAE), which maintains condenser vacuum once it has been established. The NMPNS review focused on modifications to the system that may affect gaseous radioactive material handling and release assumptions, and design features to preclude the possibility of an explosion (if the potential for explosive mixtures exists). The NRC's acceptance criteria for the MCES are based on (1) GDC60, insofar as it requires that the plant design include means to control the release of radioactive effluents; and (2) GDC64, 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.

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 SJAEs to operate satisfactorily with available dilution / motive steam flow; and

- Mechanical vacuum (hogging) pump capability to remove required non-condensable gases from the condenser at EPU start-up conditions (evacuation of the main condenser to bring it down to vacuum prior to operation at power).

The physical size of the primary condenser and evacuation time are the main factors in establishing the capabilities of the vacuum pumps. 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 were originally designed for operation at flows greater than those required at EPU conditions.

Conclusion

NMPNS has reviewed the assessment of required changes to the MCES and concludes that these changes have been adequately evaluated. NMPNS concludes that 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. NMPNS also concludes that the MCES will continue meet the requirements of GDCs 60 and 64. Therefore, NMPNS finds the proposed EPU 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. NMPNS reviewed changes to the turbine gland sealing system with respect to factors that may affect gaseous radioactive material handling (e.g., source of sealing steam, system interfaces, and potential leakage paths). The NRC's acceptance criteria for the turbine gland sealing system are based on (1) GDC60, insofar as it requires that the plant design include means to control the release of radioactive effluents; and (2) GDC64, 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.

Technical Evaluation

Taking into account the modification of the NMP2 main turbine to accept the increased steam flow at EPU operating conditions, and new springs for the steam seal regulators as a result of recent uprates, the evaluation of the turbine gland seal system demonstrated that no hardware changes are required to support operation at EPU conditions.

Conclusion

NMPNS has reviewed the assessment of the turbine gland sealing system and concludes that it has been adequately evaluated. NMPNS concludes that 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 GDCs 60 and 64. Therefore, NMPNS finds the proposed EPU acceptable with respect to the turbine gland sealing system.

2.5.2.4 Main Steam Isolation Valve Leakage Control System

Not applicable. NMP2 does not use a Main Steam Isolation Valve Leakage Control System. See Section 2.5.4.2 for a discussion of EPU impact on the holdup volume of the MSLs and Main Condenser.

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 NMPNS review for the proposed EPU focused on the effects of the proposed EPU on the capability of the system to provide adequate cooling to the spent fuel during all operating and accident conditions. The NRC's acceptance criteria for the spent fuel pool cooling and cleanup system are based on (1) GDC5, 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) GDC44, 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) GDC61, 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.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of Constant Pressure Power Uprates. Section 6.3 of the CLTR addresses the effect of Constant Pressure Power Uprate on the Fuel Pool. The results of this evaluation are described below.

The NMP2 Fuel Pool Cooling and Cleanup System (FPCCS) and the supporting systems are not interconnected with NMP1. EPU does not affect this separation, and these systems continue to support only NMP2.

The spent fuel cooling section of the FPCCS is classified as nuclear safety-related and is redundant. Cooling water to the FPCCS heat exchangers is provided by the reactor building closed loop cooling water (RBCCW) or service water systems. Additional FPCCS cooling is available from the RHR system. The service water system is redundant and classified as nuclear safety-related. EPU does not affect the alignments, availability or safety-related designations of these systems. EPU did not change the trains of cooling used to evaluate the effects of core offload.

The current thermal licensing basis for the NMP2 spent fuel pool (SFP) is to maintain the SFP bulk water temperature at or below 125°F under normal operating conditions, below 140°F for a normal full-core offload and below 150°F for emergency core offload. In addition the FPCCS is designed to maintain an average reactor coolant exit temperature below 150°F during plant refueling outages. Table 2.5-3 summarizes the SFP response to normal and full-core offloads.

EPU will increase the heat load on the FPCCS during and after refueling outages because of the increase in decay heat. The increased decay heat was calculated using the formulation and uncertainty factors from Branch Technical Position ASB 9-2 in NUREG-0800 (Reference 39), which yields conservative decay heat values relative to ANS 5.1-1979, which is also acceptable and approved for use by the NRC. This increased decay heat was then evaluated for batch and full core offloads. The evaluations credit the service water system for directly removing the decay heat from the FPCCS heat exchangers. For the months December through April, when refueling outages are expected to take place, the maximum expected service water temperature coincident with the maximum heat loads is approximately 50°F, while the heat removal by the FPCCS heat exchanger is conservatively based on a temperature of 52°F. The emergency full-core offload was also evaluated with the maximum service water temperature of 84°F. The results of the evaluations confirm that the additional heat load is within the FPCCS design basis capability, and does not adversely affect system components or functions. Maintaining the increase in decay heat for the full-core offload during a normal refueling outage within design limits is accomplished through existing administrative and procedural limitations which require cycle specific core offload evaluations prior to initiating the core offload. No changes are required to accommodate the emergency full-core offload. USAR Section 9.1.3 describes the FPCCS operation during normal refueling, normal full-core offload, and full-core emergency offload conditions. This section states that due to the time available for required operator actions following a faulted (i.e., line break) condition and the redundancy of the FPCCS system, FPCCS is assured for any single active or passive failure. Table 2.5-3 summarizes the response of the FPCCS to the three cases (Full Core Offload Following a Refueling Outage, Emergency Full Core Offload and Core Shuffle Following a Refueling Outage) described in USAR Section 9.1.3.2. The maximum temperatures with available cooling remain within the limits described in the USAR. The heating rate is sufficiently slow to allow operator actions to initiate a redundant cooling system. In the event of a complete loss of cooling to the pool, the boil off rates remain within the make-up capability. The FPCCS system remains capable of performing its required safety functions after EPU.

An evaluation of the capability of the FPCCS to maintain water clarity concludes that water clarity will not be affected by EPU.

The normal radiation levels around the SFP may increase slightly, primarily during fuel handling operation. Radiation levels in those areas of the plant, which are directly affected by the reactor core and spent fuel, increase by the percentage increase in the average power density of the fuel bundles. Therefore, for an EPU increase of 20%, the radiation dose rates increase by 20%.

The design of spent fuel pools is typically very conservative from the perspective of radiation exposure such that changes in the fuel inventory/bundle surface dose rate of 20% results in inconsequential changes in operating dose. Surveys of the dose rate in areas around the spent fuel pool at NMP2 show general dose rates less than 1 millirem per hour with some specific areas up to 2 millirems per hour. An increase of 20% would still result in general dose rates of approximately 1 millirem per hour with the highest rates up to 2.4 millirems per hour. Such changes will have little effect on plant operations or ALARA exposure. The current NMP2 radiation procedures and radiation monitoring program would detect any changes in radiation levels and initiate appropriate actions.

Conclusion

NMPNS has reviewed the assessment related to the spent fuel pool cooling and cleanup system and concludes that the effects of the proposed EPU on the spent fuel pool cooling function of the system have been adequately evaluated. Based on this review, NMPNS concludes that the spent fuel pool cooling and cleanup 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 GDCs 5, 44, and 61. Therefore, NMPNS finds the proposed EPU 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 (SWS) 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 NMPNS review covered the characteristics of the SWS components with respect to their functional performance as affected by adverse operational (i.e., water hammer) conditions, abnormal operational conditions, and accident conditions (e.g., a LOCA with the LOOP). The NMPNS review focused on the additional heat load that would result from the proposed EPU. The NRC's acceptance criteria are based on (1) GDC4, 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) GDC5, 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) GDC44, 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.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of Constant Pressure Power Uprates. Section 6.4 of the CLTR addresses the effect of Constant Pressure Power Uprate on the Ultimate Heat Sink. The results of this evaluation are described below.

The Service Water System provides once through cooling water from Lake Ontario to various safety-related and non-safety related plant systems and components. The SWS is designed to operate during normal, transient and post accident conditions. The safety-related portion of the Service Water System includes the pumps and (safety related to non-safety related) isolation valves along with the Division I and Division II headers in the Reactor and Control Buildings, Divisional supplies to the Emergency Diesel Generator Building, Service Water Pump Bays in the Screenwell Building and the Ultimate Heat Sink (UHS). The non-safety related portions of the Service Water System include the Turbine Building supply and return, Reactor Building Closed Cooling Water (RBCCW) heat exchangers and Reactor Building normal HVAC supply air cooling coil.

Safety-Related Loads

The safety-related portions of the Service Water System are designed to provide a reliable supply of cooling water during and following a DBA for the following essential equipment and systems:

- Residual Heat Removal System (RHS) Heat Exchangers
- Reactor Building Emergency Recirculation HVAC Units
- Control Building Unit Coolers
- Division I, II and HPCS Standby Diesel Generator Jacket Water Coolers
- Division I, II and HPCS Standby Diesel Generator Control Room Unit Coolers
- Division I, II and HPCS Switchgear Room Unit Coolers
- Control and Relay Room Chilled Water Chiller Condensers
- Reactor Building Unit Coolers
- Residual Heat Removal (RHR) Pumps Seal Coolers (safety-related backup to normal non-safety related RBCCW supply)
- Reactor Recirculation Pumps Motor, Seal and Bearing Oil Coolers Emergency Supply (SR for pressure boundary only)
- Spent Fuel Pool Cooling (SFPC) Water Heat Exchangers (safety-related backup to normal non-safety related RBCCW supply)
- Spent Fuel Pool (SFP) Emergency Fill
- Hydrogen Recombiners
- Control Room Chilled Water Emergency Supply

The evaluation of the systems performance is given in the following subsections.

Safety-Related Service Water Headers – Non-Power Dependent Heat Loads

The safety-related performance of the Service Water System during and following the most demanding design basis event, the LOCA, for the following equipment and systems is not dependent on RTP:

- Reactor Building Emergency Recirculation HVAC Units
- Control Building Unit Coolers
- Division I, II and HPCS Standby Diesel Generator Jacket Water Coolers
- Division I, II and HPCS Standby Diesel Generator Control Room Unit Coolers
- Division I, II and HPCS Switchgear Room Unit Coolers
- Control and Relay Room Chilled Water Chiller Condensers
- Reactor Building Unit Coolers
- Residual Heat Removal (RHR) Pumps Seal Coolers
- Reactor Recirculation Pumps Motor, Seal, and Bearing Oil Coolers
- Spent Fuel Pool (SFP) Emergency Fill
- Hydrogen Recombiners
- Control Room Chilled Water Emergency Supply
- Containment Spray Back-Up Supply

Safety-Related Service Water Headers – Power Dependent Heat Loads

The containment cooling analysis in Section 2.6.5 shows that the post-LOCA RHS 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 RHS and SWS. As discussed in Sections 2.6.1 and 2.6.5, the existing suppression pool structure and associated equipment have been reviewed for acceptability based on this increased post-LOCA suppression pool temperature. Therefore, the containment cooling analysis and equipment review demonstrate that the suppression pool temperature can be maintained within acceptable limits in the post-accident condition at EPU based on the existing capability of the SWS. The SWS has sufficient capacity at EPU to supply adequate cooling and makeup to the SFPC heat exchangers and SFP, respectively. In addition, the SWS has sufficient capacity to serve as a standby coolant supply for long term core and containment cooling as required for EPU conditions. The SWS flow rate is not changed.

Non-Safety Related Service Water Headers – Heat Loads

Four unit coolers are being added in the Turbine Building to remove increased heat loads from the condensate and condensate booster pumps. The unit coolers increase the normal Service Water System flow by approximately 1%. This increase is expected to have a negligible impact on the overall system flow distribution and heat removal capability.

Conclusion

A review related to the effects of the proposed EPU on the SWS concluded that these have been adequately evaluated for the increased heat loads on system performance that would result from the proposed EPU. It was concluded that the SWS will continue to be protected from the dynamic effects associated with flow instabilities and provide sufficient cooling for SSCs important to safety following implementation of the proposed EPU. Therefore, it has been determined that the SWS will continue to meet the requirements of GDCs 4, 5, and 44. Based on the above, the proposed EPU is found acceptable with respect to the SWS.

2.5.3.3 Reactor Auxiliary Cooling Water Systems

Regulatory Evaluation

The NMPNS review covered reactor auxiliary cooling water systems that are required for (1) safe shutdown during normal operations, anticipated operational occurrences, and mitigating the consequences of accident conditions, or (2) preventing the occurrence of an accident. These systems include closed-loop auxiliary cooling water systems for reactor system components, reactor shutdown equipment, ventilation equipment, and components of the ECCS. The NMPNS review covered the capability of the auxiliary cooling water systems to provide adequate cooling water to safety-related ECCS components and reactor auxiliary equipment for all planned operating conditions. Emphasis was placed on the cooling water systems for safety-related components (e.g., ECCS equipment, ventilation equipment, and reactor shutdown equipment). The NMPNS review focused on the additional heat load that would result from the proposed EPU. The NRC's acceptance criteria for the reactor auxiliary cooling water system are based on (1) GDC4, 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) GDC5, 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) GDC44, 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.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of Constant Pressure Power Uprates. Section 6.4 of the CLTR addresses the effect of Constant Pressure Power Uprate on the Water Systems. The results of this evaluation are described below.

2.5.3.3.1 Reactor Building Closed Cooling Water System

The RBCCW heat loads are mainly dependent on the reactor vessel temperature and/or flow rates in the systems cooled by the RBCCW. The change in vessel temperature is minimal and does not result in any significant increase in drywell cooling loads. The flow rates in the systems cooled by the RBCCW (e.g., Recirculation and WCS pumps cooling) do not change due to power uprate and therefore, are not affected by power uprate. The only significant increase in heat load due to EPU is an increase in Spent Fuel Pool Cooling heat load. Safety related cooling for the Spent Fuel Pool is provided by the SWS and not the RBCCW system. The normal heat load from the spent fuel pool is 16 MBTU/hr. This load would increase by less than 20% which is a small fraction of the RBCCW system design heat load of 73.6 MBTU/hr. This Spent Fuel Pool Cooling heat load occurs during refueling when other RBCCW loads are offline or significantly reduced. Therefore, the increase in Spent Fuel Pool Cooling heat load does not increase RBCCW system heat loads beyond system design. The operation of the remaining equipment cooled by the RBCCW (e.g., sample coolers and drain coolers) is not power-dependent and is not affected by power uprate. The RBCCW system contains sufficient redundancy in pumps and heat exchangers to ensure that adequate heat removal capability is available during normal operation. Sufficient heat removal capacity is available to accommodate the small increase in heat load due to EPU.

2.5.3.3.2 Turbine Building Closed Loop Cooling Water System

The supply temperature of the Turbine Building Closed Loop Cooling Water System (TBCCW) system is dependent on the heat rejected to the TBCCW system via components cooled by the system, as removed by the TBCCW heat exchangers and controlled by the system temperature control valve(s). Some heat loads on the TBCCW system are power-dependent and are increased by power uprate, such as those related to the power train pumps (condensate pumps, condensate booster pumps, heater drain pumps, reactor feed pumps) and Turbine Auxiliaries (generator hydrogen coolers, generator stator water coolers, generator leads coolers (bus duct cooling), and exciter alternator coolers). Examination of Table 2.5-4 shows the heat load of the total system is 109.6 MBTU/hr due to EPU. The system normal capacity is 129 MBTU/hr, which is unaffected by EPU. The increase in heat load of the TBCCW system can be accommodated by the margin in the system heat exchangers, and the system pumps have sufficient capacity to accommodate any minor flow increases from potential changes in localized flows to affected components, as required.

Conclusion

NMPNS has reviewed the assessment of the effects of the proposed EPU on the reactor auxiliary cooling water systems and concludes that these have been adequately evaluated for the increased heat loads from the proposed EPU on system performance. NMPNS concludes that the reactor auxiliary cooling water systems will continue to be protected from the dynamic effects associated with flow instabilities and provide sufficient cooling for SSCs important to safety following implementation of the proposed EPU. Therefore, NMPNS has determined that the reactor auxiliary cooling water systems will continue to meet the requirements of GDCs 4, 5, and 44. Based on the above, NMPNS finds the proposed EPU 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 NMPNS review focused on the effect that the proposed EPU has on the decay heat removal capability of the UHS. Additionally, the NMPNS review included evaluation of the design-basis UHS temperature limit determination to confirm that post-licensing data trends (e.g., air and water temperatures, humidity, wind speed, water volume) do not establish more severe conditions than previously assumed. The NRC's acceptance criteria for the UHS are based on (1) GDC5, 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; and (2) GDC44, 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.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of Constant Pressure Power Uprates. Section 6.4 of the CLTR addresses the effect of Constant Pressure Power Uprate on the Ultimate Heat Sink. The results of this evaluation are described below.

The UHS is Lake Ontario. It is designed to supply water at 84°F from the lake and to return water to the lake at a temperature less than the State Pollutant Discharge Elimination System (SPDES) limit of 110°F. The SPDES permit also limits the temperature differential between the discharge and suction to less than 30°F. As a result of operation at the EPU RTP level, the discharge temperature and temperature differential will increase due to higher heat loads. The normal operation discharge temperature is 98°F for CLTP and 100°F for EPU, a differential of plus 2°F. For the LOCA, the discharge temperature for CLTP and EPU is 126°F, no change. A review was performed to evaluate the increased UHS heat load for the EPU. The review concludes that the temperature of the discharge water and the differential temperature between the intake and discharge are within the limit set by the State of New York for normal and shutdown conditions. For LOCA conditions, the temperatures will exceed the SPDES limit, but the permit allows the limits to be exceeded under emergency conditions.

Conclusion

NMPNS has reviewed the information for addressing the effects that the proposed EPU would have on the UHS safety function, including the validation of the design-basis UHS temperature limit based on post-licensing data. NMPNS concludes 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 GDCs 5 and 44 following implementation of the proposed EPU. Therefore, NMPNS finds the proposed EPU acceptable with respect to the UHS.

2.5.4 Balance-of-Plant Systems

2.5.4.1 Main Steam

Regulatory Evaluation

The MSS transports steam from the NSSS to the power conversion system and various safety-related and non-safety related auxiliaries. The NMPNS review focused on the effects of the proposed EPU on the system's capability to transport steam to the power conversion system, provide heat sink capacity, supply steam to drive safety system pumps, and withstand adverse dynamic loads (e.g., water steam hammer resulting from rapid valve closure and relief valve fluid discharge loads). The NRC's acceptance criteria for the MSS are based on (1) GDC4, 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) GDC5, 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.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of Constant Pressure Power Uprates. Sections 3.4.1 and 3.7 of the CLTR address the effect of Constant Pressure Power Uprate on flow induced vibration in the main steam line and main steam line flow restrictors. The results of this evaluation are described below.

The heat balance for the EPU conditions is provided in Section 1.3. The heat balance shows the transport of steam to the power conversion equipment, the heat sink, and to steam driven components. Flow induced vibration and structural loading of the main steam system piping and supports is addressed in Section 2.2.2. Dynamic loading from water hammer is discussed below. SRV dynamic loads are discussed in Sections 2.2.2 and 2.2.3. The function and capability of the MSIVs are discussed in Section 2.2.2. SRV setpoint tolerance and FIV 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.

The RCPB system's capability to withstand adverse dynamic loads (e.g., water or steam hammer resulting from rapid valve closure) was evaluated. A summary of the results of the main steam and feedwater piping system evaluation, including pipe support loads, that contains the increased loading associated with EPU conditions (i.e., temperature, pressure, and flow, including the effects of feedwater and main steam flow induced transient loads at EPU conditions) along with a comparison to the code allowable limits is provided in Section 2.2.2, Pressure-Retaining Components and Component Supports. This section contains summary level tables that demonstrate the rigor of the analytical effort and provide an indication of the magnitude of the highest stress and load ratios found in the analysis.

SRV setpoint tolerance is independent of a CPPU. CPPU evaluations are performed using the existing SRV setpoint tolerance analytical limits as a basis. Actual historical in-service surveillance of SRV setpoint performance test results are monitored separately for compliance to the Technical Specifications and In-Service Testing program.

The in-service surveillance testing of the plant's SRVs have not shown a significant propensity for high setpoint drift greater than 3%.

An NMP2 monitoring program exists for SRV leakage. A monthly procedure is performed to trend SRV tail pipe temperatures. NMP2 has performed analyses and testing which investigated and addressed the potential for acoustic resonance due to the increased steam flow past the SRV standpipes, as well as other branch connections, and concluded that the onset of SRV standpipe vortex shedding acoustic resonance could be expected at main steam flow rates approximately 40% above the EPU 100% power steam flow rates. Therefore, SRV vibration resulting from acoustic resonance is not expected at the EPU operating conditions. For this reason, the existing SRV leakage monitoring instrumentation should be sufficient to detect any increased SRV leakage.

Increased main steam line (MSL) flow may affect vibration of the piping 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 initial plant operation at the higher steam flow rates. Attachment 10 to the EPU license amendment request contains details of the vibration monitoring program.

Therefore, the NMP2 FIV is confirmed to be consistent with the generic description provided in the CLTR.

Main Steam Line Flow Restrictors

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[illegible]	[illegible]	[illegible]]]

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.

The effects of EPU on the NMP2 MSL flow restrictors are confirmed to be consistent with the generic description described in the CLTR. The NMP2 restrictors were originally analyzed for these flow conditions and therefore the restrictors remain within the acceptable calculated differential pressure drop and choke flow limits under EPU conditions.

Conclusion

NMPNS has reviewed the assessment of the effects of the proposed EPU on the MSS and concludes that the effects of changes in plant conditions on the design of the MSS are adequately evaluated. NMPNS concludes that the MSS will maintain its ability to transport steam to the power conversion system, provide heat sink capacity, supply steam to steam-driven safety pumps, and withstand steam hammer. NMPNS further concludes that the MSS will continue to meet the requirements of GDCs 4 and 5. Therefore, NMPNS finds the proposed EPU acceptable with respect to the MSS.

2.5.4.2 Main Condenser

Regulatory Evaluation

The main condenser (MC) system is designed to condense and deaerate the exhaust steam from the main turbine and provide a heat sink for the turbine bypass system. NMP2 does not have a MSIV leakage control system. Therefore, the MC system also serves an accident mitigation function to act as a holdup volume for the plateout of fission products leaking through the MSIVs following core damage. The NMPNS review focused on the effects of the proposed EPU on the steam bypass capability with respect to load rejection assumptions, and on the ability of the MC system to withstand the blowdown effects of steam from the turbine bypass system. The NRC's acceptance criteria for the MC system are based on GDC60, insofar as it requires that the plant design include means to control the release of radioactive effluents.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of Constant Pressure Power Uprates. Section 7.2 of the CLTR addresses the effect of Constant Pressure Power Uprate on Condenser and Steam Jet Air Ejectors. The results of this evaluation are described below.

The main condenser, circulating water, and cooling tower 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.

The main condenser, circulating water, and cooling tower systems are not being modified for EPU operation. The performance of these systems was evaluated for EPU. This evaluation was based on a design duty over the actual range of circulating water inlet temperatures, and confirms that the condenser, circulating water system, cooling tower are adequate for EPU operation. The evaluation of the circulating water system at EPU conditions indicates sufficient system capacity to ensure that the plant maintains adequate condenser backpressure while meeting all environmental permit conditions related to the UHS and the plant cooling tower. Condenser hotwell temperature limitations may require load reductions at the upper range of the anticipated circulating water inlet temperatures.

EPU operation decreases the margin for the main condenser storage capacity from 7.75 minutes at CLTP to 6.55 minutes at EPU. Main condenser storage capacity remains greater than the required 5 minute holdup time for the decay of short-lived radioactive isotopes.

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

NMPNS has reviewed the assessment of the effects of the proposed EPU on the MC system and concludes that the effects of changes in plant conditions on the design of the MC system have been adequately addressed. NMPNS concludes that the MC system will continue to maintain its ability to withstand the blowdown effects of the steam from the Turbine Bypass System and thereby continue to meet GDC60 with respect to controlling releases of radioactive effluents. Therefore, NMPNS finds the proposed EPU acceptable with respect to the MC system.

2.5.4.3 Turbine Bypass

Regulatory Evaluation

The Turbine Bypass System is designed to discharge a stated percentage of rated main steam flow directly to the main condenser, bypassing the turbine. This steam bypass enables the plant to take step-load reductions up to the Turbine Bypass System (TBS) capacity without the reactor or turbine tripping. The system is also used during startup and shutdown to control reactor pressure. Because NMP2 does not have a MSIV leakage control system, the Turbine Bypass System also provides an accident mitigation function. A TBS, along with the MSS and MC system, is credited for mitigating the effects of MSIV leakage during a LOCA by the holdup and plateout of fission products. The NMPNS review for the Turbine Bypass System focused on the effects of the proposed EPU on the ability to bypass excessive steam flow during normal operations and AOOs. The NRC's acceptance criteria for the Turbine Bypass System are based on (1) GDC4, 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 Turbine Bypass System), and (2) GDC34, 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.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of Constant Pressure Power Uprates. Section 7.3 of the CLTR addresses the effect of Constant Pressure Power Uprate on the Turbine Bypass System. The results of this evaluation are described below.

The Turbine Steam Bypass System provides a means of accommodating excess steam generated during normal plant maneuvers and transients.

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The credited bypass capacity of 3,260,000 lbm/hr (unchanged from CLTP) is used as an input to the reload analysis process for the evaluation of limiting events that credit the Turbine Steam Bypass System (Section 2.8.5). [[

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Each of five bypass valves is designed to pass a steam flow of 836,000 lbm/hr (4,180,000 lbm/hr total). The bypass capacity at NMP2 remains adequate for normal operational flexibility at EPU rated thermal power.

Conclusion

NMPNS has reviewed the assessment of the effects of the proposed EPU on the Turbine Bypass System and concludes that the effects of changes in plant conditions on the design of the Turbine Bypass System have been adequately evaluated. NMPNS concludes that the Turbine Bypass System will continue to provide a means for accommodating excess steam generated during transients and shutting down the plant during normal operations. NMPNS concludes that the Turbine Bypass System will continue to meet GDCs 4 and 34. Therefore, NMPNS finds the proposed EPU acceptable with respect to the Turbine Bypass System.

2.5.4.4 Condensate and Feedwater

Regulatory Evaluation

The condensate and feedwater system (CFS) provides feedwater at a particular temperature, pressure, and flow rate to the reactor. The only part of the CFS classified as safety-related is the feedwater piping from the NSSS up to and including the outermost containment isolation valve. The NMPNS review focused on how the proposed EPU affects previous analyses and considerations with respect to the capability of the CFS to supply adequate feedwater during plant operation and shutdown, and isolate components, subsystems, and piping in order to preserve the system's safety function. The NRC's acceptance criteria for the CFS are based on (1) GDC4, 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) GDC5, 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) GDC44, 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.

Technical Evaluation

The FW and Condensate systems provide a reliable supply of FW at the temperature, pressure, quality, and flow rate as required by the reactor. Their performance has a major effect on plant availability and capability to operate at EPU conditions. For EPU, the FW and Condensate systems will meet their performance requirements with the following modifications to non-safety related equipment:

1. Feedwater Heater 6 Shell Side Safety Valve Replacement
2. Feedwater Heater 5 Shell Side Design Pressure Re-Rating
3. Feedwater Heater 6 Shell and Tube Side Design Pressure Re-Rating
4. Heater Drain Pump and Motor Replacement
5. Heater Drain Level Control Valve Replacement
6. Reactor Feedwater Pump Impeller Replacement and Modification
7. Reactor Feedwater Pump Motor Cable Replacement
8. Reactor Feedwater Pump Speed Increaser Replacement
9. Reactor Feedwater Pump Flow Control Valve Modifications
10. Feedwater System Design Pressure Re-Rating
11. Revise Recirculation Runback Logic to initiate on a Feedwater/Condensate Booster Pump Trip and increase Runback Rate to 9% per second
12. Feedwater System Setpoint Setdown

In addition to the condensate and feedwater system modifications listed above, the following equipment will be modified to support EPU operation:

1. Moisture Separator Reheater Shell Side Design Pressure Re-Rating
2. Moisture Separator Reheater Drain Receiver Design Pressure Re-Rating
3. Building Heating Intermediate Heat Exchanger Design Pressure Re-Rating

4. Scavenging Steam Relief Valve Replacement
5. Cross Around Relief Valve Replacement
6. Cross Around Piping Re-Rating

The following equipment is also being modified prior to EPU implementation. This equipment is being modified due to current material condition and these modifications are not an EPU required modification.

1. Replacement of Extraction Steam Expansion Bellows for the 1st through 4th Point Feedwater Heater Extraction Steam Lines on the 'B' and 'C' Lines
2. Replacement of the 3rd Point Feedwater Heaters

Normal Operation

System operating flows at EPU increase to approximately 118% of rated flow at the CLTP. The FW and Condensate system modifications assure acceptable performance with the new system operating conditions, provided that three condensate, three condensate booster, three heater drain, and two reactor feedwater pumps are in operation.

The FW heater design has been analyzed and verified to be acceptable for the higher FW heater flows, temperatures, and pressures for EPU. The FW heaters requiring re-rating will be re-rated prior to implementation of EPU. The performance of the FW heaters will be monitored during the EPU power ascension program.

The 3rd point feedwater heaters require replacement before the plant operates under EPU conditions. There is strong evidence that shows excessive wear and damage to the tube supports under current operating conditions. The damage has worsened as time progressed and if operating conditions were more severe, the detrimental effects on the tubes and tube supports would be greater.

Transient Operation

To account for FW demand transients, the FW system was evaluated and determined to have approximately 7% margin at EPU FW flow. For system operation with all system pumps available, the predicted operating parameters were acceptable and within the component capabilities. The post-condensate, condensate booster, and feed pump trip system capacity was evaluated. This evaluation confirmed that, with the modifications and changes listed above, these pumps will have the capability to supply the transient flow requirements. The post-condensate booster pump trip system capacity evaluation revealed that given the modifications listed above, insufficient capacity is available to maintain the required suction pressure on the reactor feed pump. Consequently, a condensate booster pump trip will require a reduction in plant power level. Therefore, reactor recirculation runback (RRRB) logic is being modified for EPU operation to initiate not only on a reactor feedwater pump trip but also on a condensate booster pump trip. This provides additional protection against multiple FW pumps tripping from a single or common initiating event. In conjunction with the RRRB logic changes, low suction pressure FW and Condensate Booster pump trips will be staggered. This will allow one FW pump to continue to operate, regain suction pressure and clear any low alarm signals which would otherwise trip the operating pump. These protection scheme changes provide equipment protection while reducing the likelihood of a single or common initiating event resulting in the loss of multiple pieces of equipment from service. The post-heater drain pump trip system capacity was also evaluated given the modifications listed above and found to be sufficient to meet the transient flow requirement. Long-term operation with a heater drain pump out of service will require a unit de-rate to maintain sufficient suction pressure on the reactor feed pumps, which is unchanged from CLTP conditions. A reactor feedwater pump flow control valve trim change is required to achieve the 7% margin at EPU FW flow. A setpoint setdown setting change mitigates the EPU impact on reactor vessel level response (see Attachment 7 to the EPU license amendment request for detailed large transient analysis).

Condensate Demineralizers

The effect of EPU on the condensate filter demineralizers (CFDs) was reviewed. The CFD system has been modified to support CFD full flow operation during backwashing and pre-coating without requiring a plant power reduction. The system experiences slightly higher loadings resulting in slightly reduced CFD run times. However, the reduced run times are acceptable (refer to Section 2.5.5 for the effect on the radwaste systems).

Conclusion

NMPNS has reviewed the assessment of the effects of the proposed EPU on the CFS and concludes that it has adequately accounted for the effects of changes in plant conditions on the design of the CFS. NMPNS concludes that the CFS will continue to maintain its ability to satisfy feedwater requirements for normal operation and shutdown, withstand water hammer, maintain isolation capability in order to preserve the system safety function, and not cause failure of safety-related SSCs. NMPNS further concludes that the CFS will continue to meet the requirements of GDCs 4, 5, and 44. Therefore, NMPNS finds the proposed EPU acceptable with respect to the CFS.

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 NMPNS review focused on the effects that the proposed EPU may have on (1) the design criteria of the gaseous waste management systems, (2) methods of treatment, (3) expected releases, (4) principal parameters used in calculating the releases of radioactive materials in gaseous effluents, and (5) design features for precluding the possibility of an explosion if the potential for explosive mixtures exists. The 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) GDC3, 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) GDC60, insofar as it requires that the plant design include means to control the release of radioactive effluents; (4) GDC61, insofar as it requires that systems that contain radioactivity be designed with appropriate confinement; and (5) 10 CFR 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.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of Constant Pressure Power Uprates. Section 8.2 of the CLTR addresses the effect of Constant Pressure Power Uprate on Gaseous Waste Management. The results of this evaluation are described below.

The CLTP design basis off-gas flow rate is 0.067 cfm/MWt. Based on the CLTR, this flow rate is very conservative. The normal operation off-gas flow rate is expected to increase by approximately 15% due to EPU. For EPU operation, the CLTP design basis is maintained and a plant specific evaluation was conducted. This evaluation verified that all structures, systems and components of the offgas system were acceptable for EPU operation.

The primary function of the Gaseous Waste Management (Offgas) System (GWMS) 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 GWMS involves the management of the condenser air removal system, gland seal exhaust and mechanical vacuum pump operation exhaust. Plant procedures exist to test for air infiltration (e.g., condenser) and repair as needed to maintain the Offgas System as functional.

The GWMS design criteria ensure that it will meet the plant licensing basis for controlling gaseous waste such that the total radiation exposure of persons in offsite areas will be within the applicable guideline values of 10 CFR 20.1302; 10 CFR 50, Appendix I; and 40 CFR 190. The plant gaseous waste licensing basis and the GWMS design criteria that support the licensing basis are unchanged by the EPU, and the plant will continue to satisfy this licensing basis under EPU operating conditions.

The GWMS methods of treatment for radiological releases consist of holdup and filtration to reduce the gaseous radioactivity that could be potentially released to offsite areas. These methods of treatment are applied to the condenser offgas system and turbine gland sealing system. The condenser offgas system radiological release rate is a function of fuel cladding performance, main condenser air inleakage, charcoal adsorber inlet dew point, and charcoal adsorber temperature. Because none of these parameters are significantly affected by the EPU, the capacity and capability of the condenser offgas holdup and filtration system to adequately perform its design function are unchanged by the EPU. The evaluation of the turbine gland sealing system is contained in Section 2.5.2.3. The evaluation confirmed that the turbine gland sealing system will continue to meet its design requirements under EPU conditions.

NMP2 has Technical Specification requirements 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 NMP2 to maintain the offgas limits. These procedures are not affected by EPU.

The design features for precluding the possibility of an explosion include (a) dilution to control the concentration of hydrogen (steam is used upstream of the recombiner and air is used downstream of the recombiner) and (b) catalytic recombination to remove the combustible gas. The combustible gas control component design requirements are determined by the quantity of radiolytic hydrogen and oxygen, which is expected to increase in proportion to the EPU power increase. Consequently, both the quantity of dilution steam upstream of the recombiner and minimum quantity of dilution air downstream of the recombiner will increase. The additional radiolytic hydrogen will also increase the catalytic recombiner temperature and offgas condenser heat load. These increases have been evaluated and it has been confirmed that sufficient margin remains in the NMP2 offgas system component design to ensure that the gaseous radwaste system will continue to satisfy the plant licensing basis.

Conclusion

NMPNS has reviewed the assessment of the gaseous waste management systems. NMPNS concludes that it has adequately accounted for the effects of the increase in fission product and amount of gaseous waste on the abilities of the systems to control releases of radioactive materials and preclude the possibility of an explosion, if the potential for explosive mixtures exists. NMPNS finds that the gaseous waste management systems will continue to meet their design functions following implementation of the EPU. NMPNS further concludes that the gaseous waste management systems will continue to meet the requirements of 10 CFR 20.1302; GDCs 3, 60, and 61; and 10 CFR 50, Appendix I, Sections II.B, II.C, and II.D. Therefore, NMPNS finds the proposed EPU acceptable with respect to the Gaseous Waste Management Systems.

2.5.5.2 Liquid Waste Management Systems

Regulatory Evaluation

The NMPNS review of liquid waste management systems focused on the effects that the proposed EPU may have on previous analyses and considerations related to the liquid waste management systems' design, design objectives, design criteria, methods of treatment, expected releases, and principal parameters used in calculating the releases of radioactive materials in liquid effluents. 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) GDC60, insofar as it requires that the plant design include means to control the release of radioactive effluents; (3) GDC61, insofar as it requires that systems that contain radioactivity be designed with appropriate confinement; and (4) 10 CFR 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.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of Constant Pressure Power Uprates. Section 8.1 of the CLTR addresses the effect of Constant Pressure Power Uprate on CLTR Liquid and Solid Waste Management. The results of this evaluation are described below.

The effect of EPU on the liquid waste management system (LWMS) is primarily a result of the increased load on the WCS and condensate demineralizers. Other increases in the LWMS load, such as increased leakage due to higher system pressures, are minimal. The increased demineralizer loads are expected to increase the volume of liquid waste processed by the LWMS due to EPU by less than 10%, which is not an appreciable increase when compared to the LWMS capacity. The existing confined liquid storage capacity can accommodate this small increase with no changes.

For the purpose of evaluating the radiological effects of EPU, it was assumed that the operational radiological sources increased by the EPU fraction, which is 20% relative to OLTP. There is enough margin between the actual operation sources and design basis sources to accommodate the 20% increase. Therefore, the current design basis sources remain bounding.

Since the liquid volume does not increase appreciably, and the radiological sources remain bounded by the existing design basis, the current design and operation of the LWMS will accommodate the effects of EPU with no changes, and the existing equipment and procedures that control releases to the environment will continue to ensure that releases remain within the applicable guideline values of 10 CFR 20.1302; 10 CFR 50, Appendix I; and 40 CFR 190.

Conclusion

NMPNS has reviewed the assessment of the liquid waste management systems. NMPNS concludes that it has adequately accounted for 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. NMPNS finds that the liquid waste management systems will continue to meet their design functions following implementation of the proposed EPU. NMPNS further concludes that the liquid waste management systems will continue to meet the requirements of 10 CFR 20.1302; GDCs 60 and 61; and 10 CFR 50, Appendix I, Sections II.A and II.D. Therefore, NMPNS finds the proposed EPU acceptable with respect to the liquid waste management systems.

2.5.5.3 Solid Waste Management Systems

Regulatory Evaluation

The NMPNS review of the solid waste management systems (SWMS) focused on the effects that the proposed EPU may have on previous analyses and considerations related to the design objectives in terms of expected volumes of waste to be processed and handled, the wet and dry types of waste to be processed, the activity and expected radionuclide distribution contained in the waste, equipment design capacities, and the principal parameters employed in the design of the SWMS. The NRC's acceptance criteria for the SWMS 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) GDC60, insofar as it requires that the plant design include means to control the release of radioactive effluents; (3) GDC63, insofar as it requires that systems be provided in waste handling areas to detect conditions that may result in excessive radiation levels; (4) GDC64, 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 71, which states requirements for radioactive material packaging.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of Constant Pressure Power Uprates. Section 8.1 of the CLTR addresses the effect of Constant Pressure Power Uprate on CLTR Liquid and Solid Waste Management. The results of this evaluation are described below.

The waste streams for the SWMS are (1) dry active waste, (2) spent ion exchange resin and filter sludge, and (3) evaporator concentrates. The EPU does not affect dry active waste so the volume and mix of dry active waste is unchanged. The effect of EPU on the SWMS is primarily a result of the increased load on the WCS and condensate demineralizers. The increased demineralizer loads are expected to increase the volumes of spent ion exchange resin and filter sludge (the resin is no longer regenerated and the evaporators have been taken out-of-service, therefore there are no evaporator concentrates). The installed pre-filters are expected to reduce the total quantity of spent ion exchange resins. However, no credit is taken for the pre-filters at the present time. The result is that the anticipated increase in solid radwaste volume is approximately 7%. Based on recent experience, there is enough margin between the actual solid radwaste volume and the design basis volume to accommodate this increase.

The EPU does not generate a new type of waste or create a new waste stream. Therefore, the types of radwaste that require shipment are unchanged.

For the purpose of evaluating the radiological effects of EPU, it was assumed that the operational radiological sources increased by the EPU fraction, which is 20% relative to OLTP. There is enough margin between the actual operation sources and design basis sources to accommodate the 20% increase. Therefore, the current design basis sources remain bounding.

Since the solid waste volume does not increase appreciably, and the radiological sources remain bounded by the existing design basis, the current design and operation of the SWMS will accommodate the effects of EPU with no changes, and the existing equipment and procedures that control waste shipments and releases to the environment will continue to ensure that releases remain within the applicable regulatory guidance.

Radiation effluent limits and monitoring requirements are independent of reactor thermal power, and therefore are not affected by EPU.

Conclusion

NMPNS has reviewed the assessment related to the solid waste management systems. NMPNS concludes that it has adequately accounted for the effects of the increase in fission product and amount of solid waste on the ability of the SWMS to process the waste. NMPNS finds that the solid waste management systems will continue to meet their design functions following implementation of the proposed EPU. NMPNS further concludes that it has demonstrated that the solid waste management systems will continue to meet the requirements of 10 CFR 20.1302; GDCs 60, 63, and 64; and 10 CFR 71. Therefore, NMPNS finds the proposed EPU acceptable with respect to the SWMS.

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 NMPNS review focused on increases in emergency diesel generator electrical demand and the resulting increase in the amount of fuel oil necessary for the system to perform its safety function. The NRC's acceptance criteria for the emergency diesel engine fuel oil storage and transfer system are based on (1) GDC4, 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) GDC5, 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) GDC17, insofar as it requires onsite power supplies to have sufficient independence and redundancy to perform their safety functions, assuming a single failure.

Technical Evaluation

The emergency loads are computed based on equipment nameplate data or brake horse power (BHP) with conservative demand factors applied. EPU conditions are achieved by utilizing existing equipment operating at or below the nameplate rating and within the calculated BHP for the required pump motors. In addition, USAR 9.5.4.2 defines the mission time as it relates to the diesel fuel oil system as each storage tank containing sufficient fuel oil for continuous operation of its respective diesel engine for 7 days. EPU does not impact this mission time. Therefore, under emergency conditions, the electrical supply and distribution components are adequate.

No increase in flow or pressure is required of any AC-powered ECCS equipment for EPU. Therefore, the amount of power required to perform safety-related functions (pump and valve loads) is not increased with EPU, and the current emergency power system remains adequate. The systems have sufficient capacity to support all required loads to achieve and maintain safe shutdown conditions and to operate the ECCS equipment following postulated accidents and transients.

Because the loads and mission times are not changed for EPU, no changes to the Emergency Diesel Engine Fuel Oil Storage and Transfer System are necessary.

Conclusion

The assessment has concluded that the emergency diesel generators loads and mission times are not changed for EPU, and as a result the amount of fuel oil required also remains unchanged under EPU. The existing NMP2 assessment of the amount of required fuel oil for the emergency diesel generators and its conclusions remain adequate under EPU conditions. NMPNS concludes 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 GDCs 4, 5, and 17. Therefore, NMPNS finds the proposed EPU 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 (LLHS) includes components and equipment used in handling new fuel at the receiving station. The NMPNS review covered the avoidance of criticality accidents, radioactivity releases resulting from damage to irradiated fuel, and unacceptable personnel radiation exposures. The NMPNS review focused on the effects of the new fuel on system performance and related analyses. The NRC's acceptance criteria for the LLHS are based on (1) GDC61, insofar as it requires systems containing radioactivity be designed with appropriate confinement and with suitable shielding for radiation protection; and (2) GDC62, insofar as it requires that criticality be prevented.

Technical Evaluation

NEDC-33004P-A, Revision 4, Constant Pressure Power Uprate, Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of Constant Pressure Power Uprates. Section 6.8 of the CLTR addresses the effect of Constant Pressure Power Uprate on plant systems that are not significantly affected. The results of this evaluation are described below.

This system is one of the plant systems that are considered systems not significantly affected per CLTR Section 6.8 (Reference 1) and it is addressed in Table 2.5-5.

The LLHS is based on as-built (CLTP) component specifications which are not changed by EPU (e.g., component dimensions) or which remain within as-built design capacity (e.g., fuel bundle weight). Thus, the LLHS meets the [[ ]] described by the NRC approved topical report.

Conclusion

NMPNS has reviewed the effects of the new fuel on the ability of the LLHS to avoid criticality accidents and concludes that the conclusion of the CLTR Section 6.8 has been correctly applied and there is no effect. Based on this review, NMPNS further concludes that the LLHS will continue to meet the requirements of GDCs 61 and 62 for radioactivity releases and prevention of criticality accidents. Therefore, NMPNS finds the proposed EPU acceptable with respect to the LLHS.

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.

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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 2.5-5, that are not addressed by this report are not significantly affected by EPU. The effect of EPU on the other systems at NMP2 was confirmed to be consistent with the CLTR.

Table 2.5-1 Appendix R Fire Event Evaluation Results

Parameter	CLTP [1,4]	EPU [1,4]	App. R Criteria
Peak Cladding Temperature (°F)	593 [2]	593 [2]	≤ 1500 °F
Maximum Operator Action Time to Open ADS valves (minute)	10.0	10.0	See Note 3
Peak Drywell Pressure (psig)	28.9	31.1	≤ 45
Peak Wetwell Airspace Pressure (psig)	25.3	27.4	≤ 45
Peak Drywell Temperature (°F)	271.9	274.7	≤ 340
Peak Wetwell Airspace Temperature (°F)	169.3	181.2	≤ 270
Suppression Pool Bulk Temperature (°F)	188.6	198.1	≤ 212

Notes:

1. Using SAFER/GESTR-LOCA and SHEX methodologies
2. Initial steady-state fuel temperature.
3. As illustrated by Figures 2.5-1 through 2.5-4, some core uncovery is anticipated, but sufficient cooling will be available to assure criterion of no fuel perforation (PCT < 1500 °F) is met. Analysis basis defines the maximum operator action time as the time it takes for the downcomer water level to reach top of active fuel (TAF). This timing exceeds 10 minutes for both CLTP and EPU conditions. As such the current maximum operator action time of 10 minutes continues to be supported and remains unchanged for EPU.
4. Reactor vessel pressure remains low enough to ensure no risk of reactor vessel overpressure.

Table 2.5-2 SGTS Iodine Removal Capacity Parameters

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Table 2.5-3 Spent Fuel Pool Parameters for EPU Off-Normal Operation After An Emergency Full–Core Offload

Conditions / Parameter	Limiting Core Offload	Limit
Configuration [1] Two trains of SFC are in service	Emergency Full Core Offload (764 Assemblies)	
Peak SFP Temperature (°F)	142.9	150
Batches (Assemblies) of old fuel in the SFP	9(3285)	
Empty Spaces in SFP	0	
Time to boil from loss of all cooling at peak temperature (hr)	3.6	
Boil off rate (gpm)	121	240
Heat transfer (MBTU/hr)	58.9	
Configuration [2] One train of SFC is in service	Full Core Offload Following a Refueling Outage (764 Assemblies)	
Peak SFP Temperature (°F)	139.1	140
Batches (Assemblies) of old fuel in the SFP	8(2920)	
Empty spaces in SFP	365	
Time to boil from loss of all cooling at peak temperature (hr)	5.1	
Boil off rate (gpm)	92	120
Heat transfer (MBTU/hr)	43.8	

Conditions / Parameter	**Limiting Core Offload**	**Limit**
Configuration[3] One train of SFC is in service	Core Shuffle Following a Refueling Outage (414 Assemblies)	
Peak SFP Temperature (ºF)	122.9	125
Batches (Assemblies) of old fuel in the SFP	8(2920)	
Empty spaces in SFP	715	
Time to boil from loss of all cooling at peak temperature (hr)	8.0	
Boil off rate (gpm)	74	
Heat transfer (MBTU/hr)	36.3	

Notes:

1. Assumes core offload begins 48 hours after reactor shutdown.
2. Assumes core offload begins 80 hours after reactor shutdown.
3. Assumes core offload begins 48 hours after reactor shutdown.

Table 2.5-4 EPU TBCCW Impact

Power Level	Total System Heat Load MBTU/hr	Total System Capacity MBTU/hr
CLTP	95.3	129
EPU	109.6	129

Table 2.5-5 Basis for Classification of No Significant Effect

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[[[illegible]	[illegible]	[illegible]	[illegible]	[illegible]

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[[[illegible]	[illegible]	[illegible]	[illegible]	[illegible]

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**Figure 2.5-1 Appendix R Evaluation Results
CLTP Fuel Heatup (SAFER) - Peak Cladding Temperature**

[[ ]]

**Figure 2.5-2 Appendix R Evaluation Results
CLTP Fuel Heatup (SAFER) - Water Level Outside the Shroud**

[[ ]]

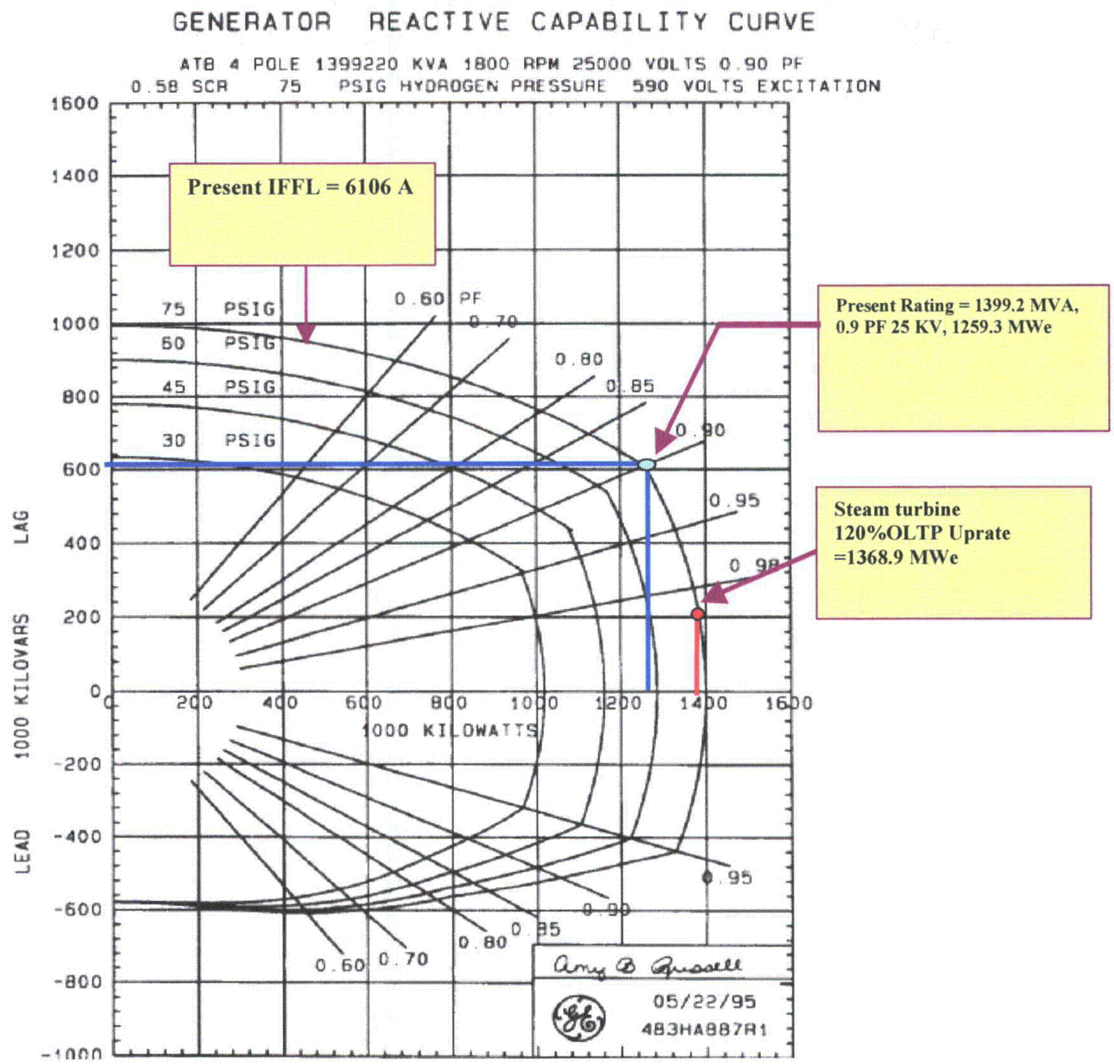
Figure 2.5-3 Appendix R Evaluation Results
EPU Fuel Heatup (SAFER) - Peak Cladding Temperature

[[ ]]

Figure 2.5-4 Appendix R Evaluation Results
EPU Fuel Heatup (SAFER) - Water Level Outside the Shroud

[[ ]]

Figure 2.5-5 Reactive Capability Curve



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 NMPNS review for the primary containment functional design covered (1) the temperature and pressure conditions in the drywell and wetwell due to a spectrum of postulated LOCAs, (2) the differential pressure across the operating deck for a spectrum of LOCAs, (3) suppression pool dynamic effects during a LOCA or following the actuation of one or more RCS safety/relief valves, (4) the consequences of a LOCA occurring within the containment (wetwell), (5) the capability of the containment to withstand the effects of steam bypassing the suppression pool, (6) the suppression pool temperature limit during RCS safety/relief valve operation, and (7) the analytical models used for containment analysis. The NRC's acceptance criteria for the primary containment functional design are based on (1) GDC4, 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) GDC16, 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) GDC50, 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) GDC13, 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) GDC64, 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.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of Constant Pressure Power Uprates. Section 4.1 of the CLTR addresses the effect of Constant Pressure Power Uprate on Primary Containment Functional Design. The results of this evaluation are described below.

The NMP2 USAR provides the containment responses to various postulated accidents that validate the design basis for the containment. EPU operation changes some of the conditions for the containment analyses. For example, the Short-Term DBLOCA containment response during the blowdown is governed by the blowdown flow rate. This blowdown flow rate is dependent on the reactor initial thermal-hydraulic conditions, such as vessel pressure and the mass and energy of the vessel fluid inventory, which change slightly with EPU. Also, the long-term heat-up of the suppression pool following a LOCA or a transient is governed by the ability of the RHS 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 response was reanalyzed to demonstrate the plant's capability to operate with a rated power increase to 3988 MWt. The key plant parameters used to model and analyze the plant response at EPU are provided in Table 2.6-2.

The analyses of containment pressure and temperature responses, as described in Section 2.6.1.1, were performed in accordance with RG 1.49 and ELTR1 using GEH codes and models. The M3CPT code was used to model the Short-Term containment pressure and temperature response. The modeling used in the M3CPT analyses is described in References 40 and 41. References 40 and 41 describe the basic containment analytical models used in GEH codes. Reference 42 describes the more detailed RPV model (LAMB) used for determining the vessel break flow in the containment analyses for EPU.

The LAMB code models the recirculation loop as a separate pressure node. It also allows for inclusion of flashing in the pipe and vessel during the blowdown and flow choking at the jet pump nozzles when the conditions warrant. The use of the LAMB blowdown flow in M3CPT was identified in ELTR1 by reference to the LAMB code qualification in Reference 42.

The SHEX code was used to model the Long-Term containment pressure and temperature response. The key models in SHEX are based on models described in Reference 41. The GEH containment analysis methodologies have been applied to all BWR power uprate projects performed by GEH and accepted by the NRC.

The effects of EPU on the containment dynamic loads due to a LOCA or SRV discharge have also been evaluated as described in Section 2.6.1.2. The containment hydrodynamic loads have been defined generically for Mark II plants as part of the Mark II containment program and are described in detail in the Dynamic Forcing Function Report (DFFR) (Reference 43). The DFFR loads were reviewed and approved by the NRC in NUREG-0487 (Reference 44) and NUREG-0808 (Reference 45). The specific application of these loads to NMP2 is described in Section 6A.4 of the NMP2 USAR. The evaluation of the LOCA containment dynamic loads is based primarily on the results of the Short-Term analysis described in Section 2.6.1.2.

The SRV discharge load evaluation would normally consider any increases in the SRV opening setpoints for EPU. Because EPU does not change the SRV setpoints, the pressure related SRV loads do not change.

2.6.1.1 Containment Pressure and Temperature Response

Short-Term and Long-Term containment analysis results are reported in the USAR. The Short-Term analysis is directed primarily at determining the containment pressure response during the initial blowdown of the reactor vessel inventory to the containment following a large break inside the drywell. Due to the NMP2 containment configuration with respect to crucial parameters such as downcomer vent area to break area, the NMP2 initial peak dry well (DW) pressure excursion during the initial blowdown period of a DBLOCA may not produce the most limiting containment pressure conditions. As a result, the potential that the DW pressure will exceed this initial spike at some time later in the transient when the entire content of the DW air has been transferred into the wet well (WW) was examined. Therefore, in addition to the typically analyzed suite of Short-Term cases run for a limited duration of about 40 seconds, additional Extended-Short-Term cases were analyzed that include a longer transient run extended to over 250 seconds of transient time to ensure that the peak pressure condition is analyzed.

The Long-Term analyses are directed primarily at the pool temperature response, considering the decay heat addition to the suppression pool. The DBLOCA and Alternate Shutdown Cooling Event (ASDC) were both reanalyzed for EPU. Peak values of the containment pressure and temperature responses to the DBLOCA are given in Table 2.6-1, however, it should be noted that the ASDC results are limiting for peak suppression pool and wetwell temperature. The impact of local suppression pool temperatures during SRV discharges was addressed in accordance with the NUREG-0783 (Reference 51) criteria.

The current steam bypass analysis assumes an initiation time for containment sprays of 30 minutes. Use of this initiation time at EPU conditions results in a primary containment pressure increase that is within the primary containment design; however, the margin to the design pressure is significantly reduced. The current licensing basis for the initiation of containment sprays for similar events provided in USAR 6.2.1.1.3, Design Evaluation - Assumptions for Long-Term Cooling, and in the Alternative Radiological Source Term Safety Evaluation (Reference 34) is 20 minutes. For this reason, the primary containment steam bypass analysis for EPU was performed assuming a containment spray initiation time of 20 minutes and resulted in the restoration of the margin to the design pressure.

The effect of EPU on the events which yield the limiting containment pressure and temperature response is provided below.

2.6.1.1.1 Long-Term Suppression Pool Temperature Response

(a) Bulk Pool Temperature

The Long-Term bulk pool temperature response for EPU is evaluated for the limiting DBLOCA in Section 6.2 of the USAR and the limiting Alternate Shutdown activity in Section 15.2 of the USAR. Per GE Safety communication SC06-01 (Reference 52), the potential was identified that a single failure that eliminated only the RHR heat exchanger could prove more limiting than the typically analyzed scenario of the single failure of the entire RHR train. This scenario was previously analyzed for NMP2 at CLTP and reported as the limiting scenario in Reference 53. The maximum bulk suppression pool (SP) temperature allowed for these events is the ECCS net positive suction head (NPSH) pump limit of 212°F.

The analysis of the DBLOCA was performed at 102% of EPU RTP. The time dependent SP and WW temperature response is presented in Figures 2.6-4 and 2.6-5 and the calculated peak values for LOCA bulk pool temperature for the current Analysis of Record and the EPU RTP case are compared in Table 2.6-1. The EPU analyses were performed using an NMP2-specific decay heat table based on ANS/ANSI 5.1-1979 with 2-sigma adders with additional actinides and activation products per GE SIL 636 (Reference 54), which is more realistic than the generic decay heat table that was used for the Reference 53 analysis. The analysis assumed the single failure of one of the two RHR heat exchangers. The EPU analysis also used more realistic RHR heat exchanger performance assumptions to limit the predicted suppression pool temperatures to their current values. The resulting calculated peak bulk suppression pool temperature is 207°F. This temperature is well within the ECCS NPSH pump limit of 212°F.

The highest bulk pool temperature response from a non-LOCA event results from an Alternate Shutdown Cooling event as demonstrated by the recent reanalysis reported in Reference 53. This event was also analyzed at 102% of EPU RTP using more realistic RHR heat exchanger performance assumptions to limit the predicted suppression pool temperatures to their current values. The limiting alternate shutdown activity assumes reactor isolation with availability of one RHR heat exchanger. The resulting time-dependent SP and WW temperature response is presented in Figure 2.6-6 and the peak bulk pool temperature at 102% of EPU RTP is 210°F which also is within the ECCS NPSH pump limit of 212°F.

(b) Suppression Pool Temperature with SRV Discharge

The suppression pool temperature limit for SRV discharge was originally specified in NUREG-0783, because of concerns resulting from unstable condensation observed at high pool temperatures in plants without quenchers. Quencher devices such as the T-quenchers used in NMP2 mitigate these loads. The peak local suppression pool temperature at NMP2 has been evaluated for EPU and meets the NUREG-0783 criteria. This evaluation demonstrated a minimum subcooling of approximately 25°F locally at the quencher. This ensures that the exiting quencher steam is condensed before posing an ECCS steam ingestion potential. Therefore, the peak local suppression pool temperature at NMP2 is acceptable for EPU conditions.

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 primarily based on the Short-Term RSLB LOCA analyses and compliance with generic criteria developed through testing programs. The analyses were performed as described in Section 2.6.1.1 with break flows calculated using a more detailed RPV model (Reference 42). The NRC approved use of this model for the EPU containment evaluations in Reference 2. These analyses also provide calculated values for the controlling parameters for the dynamic loads throughout the blowdown. The key parameters are drywell and wetwell pressures, vent flow rates and suppression pool temperature. The LOCA dynamic loads considered in the EPU evaluations include pool swell, condensation oscillation and chugging.

The results of EPU pool swell evaluations confirmed that the current pool swell load definition remains bounding. The containment response conditions for EPU are within the range of test conditions used to define condensation oscillation loads for the plant. The containment response conditions for EPU are within the conditions used to define the chugging loads. Therefore, the LOCA dynamic loads are not affected by EPU.

2.6.1.2.2 Safety Relief Valve Loads

NMP2 is equipped with Kraftwerks Union T-quencher devices. The methodology to calculate the SRV air-clearing loads employs pressure signatures derived from the Kraftwerks Union T-quencher test program to account for a design reactor pressure of ~1250 psig (Reference 55). This design basis reactor pressure (~1250 psig) for the NMP2 SRV loads definition is greater than the maximum safety setpoint of 1241 psig established for the EPU. The EPU will not impact the NMP2 design basis SRV air-clearing loads definition. Therefore, it is concluded that the current SRV loads definition is still applicable to the EPU.

2.6.1.3 Generic Letter 96-06

The NMP2 response to GL 96-06 was based on the post-accident containment pressure and temperature response for CLTP conditions. As the containment analysis presented within this section demonstrates, the CLTP conditions are not significantly impacted by EPU conditions. Therefore, the existing NMP2 response remains valid for EPU.

Conclusion

NMPNS has reviewed the containment temperature and pressure transient and concludes that it has adequately accounted for the increase of mass and energy resulting from the proposed EPU. NMPNS further concludes that containment systems will continue to provide sufficient pressure and temperature mitigation capability to ensure that containment integrity is maintained. NMPNS also concludes that containment systems and instrumentation will continue to be adequate for monitoring containment parameters and release of radioactivity during normal and accident conditions and the containment and associated systems will continue to meet the requirements of GDCs 4, 13, 16, 50, and 64 following implementation of the proposed EPU. Therefore, NMPNS finds the proposed EPU 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 NMPNS review for subcompartment analyses covered the determination of the design differential pressure values for containment subcompartments. The NMPNS review focused on the effects of the increase in mass and energy release into the containment due to operation at EPU conditions, and the resulting increase in pressurization. The NRC's acceptance criteria for subcompartment analyses are based on (1) GDC4, 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) GDC50, 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.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of Constant Pressure Power Uprates. Section 10.1 of the CLTR addresses the effect of Constant Pressure Power Uprate on Subcompartment Analyses. The results of this evaluation are described below.

An annular structure of reinforced concrete is located inside the drywell around the Reactor Pressure Vessel (RPV) in order to provide thermal and radiation shielding, and is called the Biological Shield Wall (BSW). The BSW is designed to withstand the differential pressure that would develop across the wall as a result of a high pressure pipe break within in the annulus (i.e., between the RPV and the BSW). A pipe break in this region results in a combination of four dynamic loads, collectively referred to as Annulus Pressurization (AP) Loads. These four dynamic loads consist of 1) the asymmetric pressurization of the annular area between the BSW and RPV, 2) the jet reaction resulting from the break flow through the vessel nozzle, 3) the jet impingement on the vessel of the break flow from the broken pipe, and 4) the impact load absorbed by the pipe whip restraint. These loads are a function of the break size, location, fluid thermal-hydraulic conditions, and the annular vent area to the rest of the drywell.

The second subcompartment region analyzed is the volume between the RPV head and the drywell head. Because of the limited vent area between the drywell head compartment and the drywell, a pipe break in either region can cause a differential pressure load across the drywell head refueling bulkhead plate.

AP Load Evaluation

The mass and energy release rate profiles used in developing the asymmetric loads at MELLLA boundary line were calculated using the methods from NEDO-24548, "Annulus Pressurization Load Adequacy Evaluation" (Reference 56). The large pipe break mass and energy release rates at OLTP/CLTP and EPU conditions were recalculated using the NEDO-24548 methodology. During the review of the impact of EPU conditions on the AP load, several non-conservative assumptions were discovered related to the original design basis for mass energy release, the dynamic structural response for off-rated conditions, and the limiting mass energy release. In order to reconcile these non-conservative assumptions, a review of the AP pressure time histories was performed for the large piping segment breaks within the annulus for effects including the structural dynamic response of the reactor vessel, reactor vessel internals, attachments to the vessel, and attachments to the bio-shield wall. The breaks included the Reactor Recirculation discharge, FW, and LPCI for several power flow points along the MELLLA boundary from minimum flow through maximum EPU power. This review determined that the impact of EPU operation on the resulting structural forces and accelerations was an increase in the range of 0% to 8% when compared to those for OLTP/CLTP operation. However, the result of the combined changes of the non-conservative assumptions and EPU is an increase in the AP load structural forces and accelerations in the range of 0% to 45% for most of the components and structures evaluated, with the increases for a few components in the range of 63% to 133%.

The results from the updated dynamic analyses, including impacts from both EPU and the non-conservative assumptions, were compared against those used as input to the component structural analyses of record. The effect of the increase in AP loads on the total component stresses is reduced when the AP loads are combined with the SSE seismic loads by the square root of the sum of the squares in the faulted load combination. The SSE seismic loads in the load combination are not affected by EPU. The results of these evaluations show that all reactor vessel and internals, and associated vessel attachments and supports remain within design basis faulted allowable limits. The revised AP dynamic load is included in the evaluations discussed in Sections 2.2.2 and 2.2.3.

The containment SSCs important to safety will continue to be protected from the dynamic effects resulting from pipe breaks and that the subcompartments will continue to have sufficient margins to prevent fracture of the structure due to pressure difference across the walls following implementation of the proposed EPU.

Subcompartment Pressurization Evaluation

The pressure loading on the drywell head refueling bulkhead plate due to a postulated break in the RCIC head spray line in the drywell head subcompartment is not affected by EPU because the steam dome pressure is constant with the CLTP. However, the postulated recirculation suction line break in the drywell affects the upward pressure loading on the bulkhead plate. The fluid enthalpy at the break location is not significantly affected (less than 1%), while the break location pressure is essentially the same as at CLTP. Consequently, the mass and energy release from the break is not significantly affected. The upward pressure loading on the plate has more than 10% margin to the allowable stresses. Therefore, the drywell head refueling bulkhead plate design remains adequate.

As discussed earlier, the differential pressure loading on the BSW is not significantly affected by the EPU. The peak BSW asymmetric pressure load resulting from the limiting recirculation pump discharge line break at CLTP and at EPU conditions remains below the BSW design differential pressure. The original BSW design used conservative asymmetric and uniform pressure loads.

Conclusion

NMPNS has reviewed the change in predicted AP Loads resulting from the increased mass and energy release due to EPU. NMPNS concludes that containment SSCs important to safety will continue to be protected from the dynamic effects resulting from pipe breaks and that the subcompartments will continue to have sufficient margins to prevent fracture of the structure due to pressure difference across the walls following implementation of the proposed EPU. NMPNS concludes that the plant will continue to meet GDCs 4 and 50 for the proposed EPU. Therefore, NMPNS finds the proposed EPU acceptable with respect to subcompartment analyses.

2.6.3 Mass and Energy Release

2.6.3.1 Mass and Energy Release Analysis for Postulated Loss of Coolant

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 NMPNS review covered the energy sources that are available for release to the containment and the mass and energy release rate calculations for the initial blowdown phase of the accident. The NRC's acceptance criteria for mass and energy release analyses for postulated LOCAs are based on (1) GDC50, 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 50, Appendix K, insofar as it identifies sources of energy during a LOCA.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of Constant Pressure Power Uprates. Section 4.1 of the CLTR addresses the effect of Constant Pressure Power Uprate on Containment System Response. The results of this evaluation are described below.

2.6.3.1.1 Containment Gas Temperature Response

The drywell design temperature (340°F) has been determined based on a bounding analysis of the superheated gas temperature which can be caused by a blowdown of steam to the drywell during a small break LOCA. This analysis conservatively determined a bounding combination of vessel pressure and drywell pressure that produces a maximum calculated drywell temperature. The USAR reported that expansion of reactor steam under these conditions will result in a calculated peak drywell temperature of 325.8°F. These bounding conditions, which are described in Section 6.2.1.1.3 of the USAR, are derived independent of the initial reactor power. Therefore, EPU has no effect on the peak drywell temperature.

The wetwell gas space peak temperature response was calculated assuming a heat transfer model that promotes thermal equilibrium between the pool and wetwell gas space. Table 2.6-1 shows the calculated bulk pool temperature of 207°F for the DBLOCA at EPU. The wetwell gas space temperature is also 207°F for EPU. The wetwell gas temperatures are less than the wetwell design temperature of 270°F.

2.6.3.1.2 Short-Term Containment Pressure Response

Short-Term containment response analyses were performed for the limiting DBLOCA that 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 pressure, wetwell pressure and differential pressure between the drywell and wetwell occur. These analyses were performed at 102% of

EPU RTP level. For NMP2 the Extended-Short-Term analysis discussed in Section 2.6.1.1 remained limiting for peak pressure at EPU. The time-dependent results of the limiting Extended-Short-Term analyses are presented in Figures 2.6-7 and 2.6-8 and are summarized in Table 2.6-1. Table 2.6-1 also includes comparisons of the pressure values calculated for EPU to the design pressures and to pressure values from previous calculations based on the current power. The maximum calculated containment pressure for EPU remains within the design value, and thus, is acceptable. The peak calculated drywell-to-wetwell pressure also remains within its design value.

Conclusion

NMPNS has reviewed the mass and energy release and concludes that it has adequately addressed the effects of the proposed EPU and appropriately accounts for the sources of energy identified in 10 CFR 50, Appendix K. Based on this, NMPNS finds that the mass and energy release analysis meets the requirements in GDC50 for ensuring that the analysis is conservative. Therefore, NMPNS finds the proposed EPU 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 NMPNS review covered (1) the production and accumulation of combustible gases, (2) the capability to prevent high concentrations of combustible gases in local areas, (3) the capability to monitor combustible gas concentrations, and (4) the capability to reduce combustible gas concentrations. The NMPNS review primarily focused on any effect that the proposed EPU may have on hydrogen release assumptions, and how increases in hydrogen release are mitigated. 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) GDC5, 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) GDC41, 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) GDC42, insofar as it requires that systems required by GDC41 be designed to permit appropriate periodic inspection; and (5) GDC43, insofar as it requires that systems required by GDC41 be designed to permit appropriate periodic testing.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of Constant Pressure Power Uprates. Section 4.7 of the CLTR addresses the effect of

Constant Pressure Power Uprate on the Post-LOCA Combustible Gas Control System. The results of this evaluation are described below.

The Combustible Gas Control System is designed to maintain the post-LOCA concentration of oxygen or hydrogen in the containment atmosphere below the lower flammability limit.

Following a LOCA, the Combustible Gas Control System (CGCS) is designed to maintain the hydrogen and oxygen concentrations of the drywell and wetwell atmospheres below 5 percent by volume. The post-LOCA production of hydrogen and oxygen by radiolysis increases proportionally with increased power level. Hydrogen produced by corrosion is dependent on local containment temperatures. As such, there is a direct rise in corrosion-produced hydrogen with any rise in containment temperatures due to EPU. Metal-water reaction is only affected by fuel design and is not dependent on changes due to post-LOCA system operation after EPU implementation. Refer to Figure 2.6-1 for hydrogen production profile. Based on starting the recombiners at either a 3.4% hydrogen concentration or a 3.6% oxygen concentration and accounting for a one and one half hour warm-up time, the required start time for the drywell recombiner is 32.6 hours (Figures 2.6-2 and 2.6-3 - Note the divergence of the controlled/uncontrolled concentration profiles in the cited figures due to recombiner start).

Although the EPU evaluation is in accordance with RG 1.7, several evaluation parameters differ from the CLTP analysis. These changes included initial drywell and wetwell temperatures and humidity, a shorter recombiner warmup time (1.5 hours instead of 6 hours), lower recombiner initiation limits (3.4% H_2 and 3.6% O_2 instead of 4% H_2 and 4.5% O_2), a change in fuel design, and a smaller fuel cladding mass for the determination of hydrogen produced by the metal-water reaction. As a result, the operator action time for starting the drywell recombiner (32.6 hours) cannot be directly compared to the CLTP evaluation time (43.5 hours). However, there is still ample time for the operators to react to this indication (rising hydrogen/oxygen levels) and start the recombiners to assure that hydrogen is maintained below the flammability limit. Therefore, the CGCS system continues to meet its design function post-LOCA after EPU implementation.

EPU has no effect on component maximum operating temperature. The operating temperature of the recombiners is dependent only on the containment hydrogen and oxygen concentration when the recombiners are in operation. Because the system start time is procedurally controlled to limit the containment hydrogen or oxygen concentration to the lower flammability limit, maximum operating temperature is not influenced by EPU.

Along with the mitigating system, CGCS, RG 1.7 requires that, "Each boiling water reactor...should have capability to (a) measure the hydrogen concentration in the containment, (b) mix the atmosphere in the containment." These systems (hydrogen analyzers and the mixing system) operate by either a fully active or a number of active and passive means, respectively. None of the operating parameters associated with these two required capabilities are adversely affected by the implementation of EPU. Therefore these systems will continue in their ability to be able to support post-LOCA combustible gas generation mitigation if called upon to do so.

Conclusion

NMPNS has reviewed the assessment related to combustible gas and concludes that the plant will continue to have sufficient capabilities consistent with the requirements in 10 CFR 50.44 and GDCs 5, 41, 42, and 43 as discussed above. Therefore, NMPNS finds the proposed EPU acceptable with respect to combustible gas control in containment.

2.6.5 Containment Heat Removal

Regulatory Evaluation

Fan cooler systems, spray systems, and residual heat removal (RHR) systems are provided to remove heat from the containment atmosphere and from the water in the containment wetwell. The NMPNS review in this area focused on (1) the effects of the proposed EPU on the analyses of the available net positive suction head (NPSH) to the containment heat removal system pumps and (2) the analyses of the heat removal capabilities of the RHR heat exchangers. The NRC's acceptance criteria for containment heat removal are based on GDC38, 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.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of Constant Pressure Power Uprates. Sections 4.1 and 4.2 of the CLTR address the effect of Constant Pressure Power Uprate on Containment Heat Removal. The results of this evaluation are described below.

Long-Term Suppression Pool Temperature Response

The long-term bulk pool temperature response for EPU is evaluated for the limiting DBLOCA in Section 6.2 of the USAR and the limiting Alternate Shutdown activity in Section 15.2 of the USAR. The maximum bulk suppression pool temperature allowed for these events is 212°F.

The analysis of the DBLOCA was performed at 102% of EPU RTP. The calculated time dependent SP and WW temperature response is presented in Figures 2.6-4 & 2.6-5 and the peak values for LOCA bulk pool temperature, for the current Analysis of Record (AOR) and the EPU RTP case are compared in Table 2.6-1. The EPU analyses were performed using a NMP2-specific decay heat table based on ANS/ANSI 5.1-1979 with 2-sigma adders. The analysis assumed the single failure of one of the two RHR heat exchangers. The EPU analysis also used RHR heat exchanger performance that is supported by RHR heat exchanger performance testing. Table 2.6-3 provides the "K" values and related details for: 1) clean heat exchanger; 2) most recent testing data; 3) EPU analyses; and 4) original design. NMP2 ensures that the actual heat removal capability remains above the required or analyzed value through heat exchanger thermal performance testing which is conducted just prior to planned periodic RHR heat exchanger cleaning. The performance testing acceptance criteria has been conservatively established based on the EPU analyses and reflects a maximum acceptable total fouling resistance that is less than the value assumed in the analyses (see Table 2.6-3). The resulting calculated peak bulk suppression pool temperature is 207°F. This temperature is well within the ECCS NPSH pump limit of 212°F.

The highest bulk pool temperature response from a non-LOCA event results from an Alternate Shutdown Cooling event as demonstrated by the recent reanalysis reported in Reference 53. This event was also analyzed at 102% of EPU RTP and ANS/ANSI 5.1-1979 with 2-sigma adders decay heat. The heat exchanger performance used in the analysis is shown in Table 2.6-3. The limiting alternate shutdown activity assumes reactor isolation with availability of one RHR heat exchanger. The resulting time-dependent SP and WW temperature response is presented in Figure 2.6-6 and the peak bulk pool temperature at 102% of EPU RTP is 210°F (same as current plant configuration), which also is within the design limit of 212°F.

ECCS Net Positive Suction Head

Following a LOCA, the RHR, LPCS and HPCS 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 DBLOCA using design inputs from current calculations. The limiting NPSH conditions depend on the pump flow rates, debris loading on the suction strainers, pipe frictional losses, suppression pool level and suppression pool temperature. No changes to any of these parameters result from the implementation of EPU.

The assumptions in the ECCS NPSH calculations are not dependent on the EPU RTP, but have been updated since the NMP2 response to NRC GL 97-04 (Reference 58). EPU RTP operation increases the reactor decay heat, which increases the heat addition to the suppression pool following a LOCA; however, the analyzed suppression pool water temperature and containment pressure remain essentially unchanged because of original analysis conservatisms as described in Section 2.6.1.1.1. The suppression pool temperature continues to be within the design limit of 212°F used in the NPSH calculations.

The methodology used by NMP2 to determine the amount of debris generated and transported to the strainers is based on NEDO-32686-A, the BWROG Utility Resolution Guidance for ECCS Suction Strainer Blockage (Reference 59). The assumption used for protective coatings, specifically inorganic zinc with epoxy topcoat, was 85 lbm (NEDO-32686, Section 3.2.2.2.2.1.1). This is a bounding value and is not affected by EPU.

Organic materials were assessed as unqualified coatings (i.e., carbon-based paint chips) via testing of the ECCS strainer design under simulated LOCA conditions at Alden Research Labs. Quantities of paint chips and fiber debris were included in the test program. During the testing, paint chips added to the pool did not contribute to the head loss due to post-LOCA debris for the strainer approach velocities and suppression pool turbulence conditions calculated for NMP2. Strainer approach velocities are not affected by EPU. The results of the containment analysis at

EPU conditions were within the conditions used to define the chugging loads. Therefore, suppression pool turbulence is not affected.

The above results support the above conclusion that the debris loading on the suction strainers and the methodology used to calculate available ECCS NPSH for EPU are the same as the pre-EPU conditions.

Conclusion

NMPNS has reviewed the containment heat removal systems and concludes that the effects of the proposed EPU have been adequately addressed. NMPNS finds that the systems will continue to meet GDC38 with respect to rapidly reducing the containment pressure and temperature following a LOCA and maintaining them at acceptably low levels. Therefore, NMPNS finds the proposed EPU acceptable with respect to containment heat removal systems.

2.6.6 Secondary Containment Functional Design

Regulatory Evaluation

The secondary containment structure and supporting systems are provided to collect and process radioactive material that may leak from the primary containment following an accident. The supporting systems maintain a negative pressure within the secondary containment and process this leakage. The NMPNS review covered (1) analyses of the pressure and temperature response of the secondary containment following accidents within the primary and secondary containments; (2) analyses of the effects of openings in the secondary containment on the capability of the depressurization and filtration system to establish a negative pressure in a prescribed time; (3) analyses of any primary containment leakage paths that bypass the secondary containment; (4) analyses of the pressure response of the secondary containment resulting from inadvertent depressurization of the primary containment when there is vacuum relief from the secondary containment; and (5) the acceptability of the mass and energy release data used in the analysis. The NMPNS review primarily focused on the effects that the proposed EPU may have on the pressure and temperature response and drawdown time of the secondary containment, and the effect this may have on offsite dose. The NRC's acceptance criteria for secondary containment functional design are based on (1) GDC4, 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) GDC16, 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.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of Constant Pressure Power Uprates. Section 4.5 of the CLTR addresses the effect of

Constant Pressure Power Uprate on the Standby Gas Treatment System. The results of this evaluation are described below.

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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The Secondary Containment structure, openings, pathways, and drawdown time are unaffected by EPU. Therefore, the ability of SGTS to maintain secondary containment at a negative pressure and contain radionuclides does not change as a result of EPU.

The effect of EPU on pipe failures outside containment, including any impact on mass and energy releases, is discussed in Section 2.5.1.3.

Conclusion

NMPNS has reviewed the secondary containment pressure and temperature transient and the ability of the secondary containment to provide an essentially leak-tight barrier against uncontrolled release of radioactivity to the environment. NMPNS concludes that there is no significant increase of mass and energy that would result from the proposed EPU and further concludes that the secondary containment and associated systems will continue to provide an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment following implementation of the proposed EPU. NMPNS concludes that the secondary containment and associated systems will continue to meet the requirements of GDCs 4 and 16. Therefore, NMPNS finds the proposed EPU acceptable with respect to secondary containment functional design.

Table 2.6-1 NMP2 Containment Performance Results

Parameter	DBLOCA CLTP from AOR	DBLOCA CLTP with EPU Model[1]	DBLOCA EPU with EPU Model	Limit
Peak Drywell Pressure (psia)[4]	51.5 (Reference 60)	51.9	52.9	59.7
Peak Drywell Temperature (°F)[3]	276.0	279.4	279.7[6]	340
Peak Bulk Pool Temperature (°F)[5]	205 (Reference 53)	196	207	212
Peak Wetwell Pressure (psia)[4]	46.5 (Reference 60)	46.2	47.1	59.7
Peak Wetwell Temperature (°F)[5]	205 (Reference 53)	196	207	270
Peak DW to WW delta-P (psi)[2]	18.6 (Reference 61)	18.31	18.6	25

Notes:

1. Containment analyses performed for the EPU use methods and assumptions that are unchanged from the methods and assumptions used for the AOR at CLTP. The analysis at CLTP with the EPU Model uses the plant inputs defined for the EPU model including improved RHR heat exchanger performance (as discussed in Section 2.6.1.1.1).
2. Most limiting values obtained from Short-Term Analysis
3. The DW peak temperature limit (340°F) has been determined based on a bounding analysis of the superheated gas temperature which can be caused by a blowdown of steam to the drywell during a small break LOCA as discussed in Section 2.6.3.1.1.
4. Most limiting Peak Pressure Values from Extended-Short-Term Analysis
5. Peak temperature values from Long-Term DBLOCA analysis. A peak temperature of 210°F was calculated for the ASDC event for EPU.
6. Refer to Figure 2.6-5 for SHEX output.

Table 2.6-2 Long-Term Containment Response Key Analysis Input Values

No.	Parameter	Unit	Analysis Value
1.	Reactor		
a.	Initial power level		
	1. 102% current rated power	MWt	3536
	2. 102% uprated power	MWt	4068
b.	Initial vessel dome pressure		
	1. At 102% current rated power	psia	1055
	2. At 102% uprated power	psia	1055
c.	Decay heat model		
	1. Short-term DBLOCA		ANS 5-1971 + 20%
	2. Long-term		ANS 5.1-1979 + 2σ
d.	Vessel volumes		
	1. Total vessel free volume	ft^3	21134
	2. Liquid vessel volume	ft^3	11799
e.	Vessel related masses (used in long-term calculation)		
	1. Liquid mass per recirc. loop	lbm	17669
	2. Liquid mass in the HPCS piping between the RPV nozzle and first normally closed valve	lbm	682
	3. Liquid mass in the RCIC piping between the RPV nozzle and first normally closed valve	lbm	0 - included in other data
	4. Liquid mass in the LPCI/RHR piping between the RPV and the first normally closed valve	lbm	5208
	5. Liquid mass in the core spray (CS) piping between the RPV nozzle and the first normally closed valve	lbm	620
	6. Steam mass in the main steam line to the first isolation valve. • One inner steam line • One outer steam line	lbm	 990 990

No	Parameter	Unit	Analysis Value
f.	Time at which MSIVs start to close Fully closed	sec	0.5 3.5
2.	**Drywell/Vent System**		
a.	Total drywell free volume (including vent system)	ft^3	306200
b.	Initial drywell pressure (range)	psig	-0.5 to 0.75
c.	Initial drywell temperature (range)	°F	105 to 150
d.	Initial drywell relative humidity (range)	%	20 to 100
e.	Number of downcomers		121
f.	Downcomer Submergence		
	1. Low water level (LWL)	ft	9.5
	2. High water level (HWL)	ft	11.0
g.	Loss coefficient for vent system including entrance and exit losses (based on vent exit flow area)		2.36
h.	Downcomer internal diameter	ft	1.9375
3.	**Wetwell/Suppression Pool**		
a.	Initial suppression pool volume (including water in vents)		
	1. Low water level (LWL)	ft^3	145200
	2. High water level (HWL)	ft^3	154400
b.	Initial suppression pool temperature (max)	°F	90
c.	Initial wetwell airspace volume		
	1. Low water level (LWL)	ft^3	199800
	2. High water level (HWL)	ft^3	190600
d.	Initial wetwell/containment airspace pressure (max)	psig	0.75
e.	Initial wetwell/containment airspace temperature	°F	70-122
f.	Initial wetwell/containment airspace relative humidity	%	100
4.	**LPCI/LPCS/HPCS**		
a.	LPCI runout flow rate	gpm	6660
b.	LPCI pump shutoff head	psid	210
c.	LPCI pump horse power	hp	1000

No	Parameter	Unit	Analysis Value
d.	LPCS runout flow rate	gpm	6600
e.	LPCS pump shutoff head	psid	289
f.	LPCS pump horse power	hp	1500
g.	HPCS runout flow rate	gpm	6250
h.	HPCS pump shutoff head	psid	1080
i.	HPCS pump horse power	hp	3050
5.	**RHR**		
a.	Heat exchanger K-value	BTU/sec-°F/HX	See Table 2.6-3
b.	Service water temperature	°F	84
c.	Drywell spray flow rate (1 RHR pump)	gpm	6457
d.	Wetwell spray flow rate (1 RHR pump)	gpm	413
e.	RHR flow rate in pool cooling mode (1 RHR pump)	gpm	7450
6.	**Wetwell-to-Drywell Vacuum Breakers**		
a.	Pressure difference between wetwell and drywell for vacuum breakers to be fully open – value provided is for each of the 2 VB's in series.	psid	0.5
b.	Number of vacuum breaker assemblies (multiple valve systems)		4
c.	Flow area of each vacuum breaker assembly at which loss coefficient is given below	ft^2	2.948
d.	Total loss coefficient of each vacuum breaker assembly		12.2

Table 2.6-3 Heat Exchanger Performance

RHR Heat Exchanger Condition	RHR Heat Exchanger K-value (BTU/sec-°F/HX)	Total Fouling Resistance (hr-ft²-°F/BTU/HX)	Total Fouling Resistance (% of Design)
Clean	384	0.000000	0
Heat Exchanger Testing Data	363	0.000192	12
Heat Exchanger Testing Acceptance Criteria	-	0.001204	73
Analyzed EPU - ASDC	270[1]	0.001357	82
Analyzed EPU - DBLOCA	265[1]	0.001433	87
Design	260	0.001649	100

Note 1: Long term analysis for LOCA and ASDC were used to establish the heat exchanger value needed to obtain acceptable results. The plant capability to achieve this performance has been confirmed.

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Figure 2.6-1 Time Integrated Containment Hydrogen Generation at EPU

[[ ]]

Figure 2.6-2 Containment Hydrogen

[[ ]]

Figure 2.6-3 Containment Oxygen

[[ ]]

Figure 2.6-4 EPU SP Temperature Response to DBLOCA – SCO06-01 Case

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Figure 2.6-5 EPU DW Airspace Temperature Response to DBLOCA – SCO06-01 Case

Note: The SHEX output shows a short duration DW temperature excursion to about 310°F within the first 20 to 40 seconds of the LOCA. This temperature excursion is attributed to the break flow modeling within the SHEX code, which assumes steam-only break flow when the water level in the vessel falls below the break elevation and the break is uncovered. This drives the drywell to a superheated condition, which lasts for approximately 20 seconds, until the water level recovers to the break elevation and liquid break flow is restored. However, two-phase flow occurs for this condition, not steam-only break flow, which would preclude the drywell temperature excursion indicated in this figure. Therefore, the drywell temperature excursion shown in this figure between 20 and 40 seconds should not be treated as a real phenomenon. Note that two-phase break flow is modeled by the M3CPT code, which establishes the basis for peak DBLOCA drywell pressure and temperature and which do not predict the drywell temperature excursion shown here.

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Figure 2.6-6 EPU Suppression Pool Temperature Response to ASDC Event

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Figure 2.6-7 EPU Extended Short-Term DBLOCA Containment Pressure Response

Note: Plots are from the 102% Power / 100% Flow case with extended Feedwater delivery

[[ ]]

Figure 2.6-8 EPU Extended-Short-Term DBLOCA Containment Temperature Response

Note: Plots are from the 102% Power / 100% Flow case with extended Feedwater delivery

2.7 Habitability, Filtration, and Ventilation

2.7.1 Control Room Habitability System

Regulatory Evaluation

NMPNS reviewed the control room habitability system and control building layout and structures to ensure that plant operators are adequately protected from the effects of accidental releases of toxic and radioactive gases. A further objective of the NMPNS review was to ensure that the control room can be maintained as the backup center from which technical support center personnel can safely operate in the case of an accident. The NMPNS review focused on the effects of the proposed EPU on radiation doses, toxic gas concentrations, and estimates of dispersion of airborne contamination. The NRC's acceptance criteria for the control room habitability system are based on (1) GDC4, 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) GDC19, 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 total effective dose equivalent (TEDE) for the duration of the accident.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of Constant Pressure Power Uprates. Section 4.4 of the CLTR addresses the effect of Constant Pressure Power Uprate on the MCR Atmosphere Control System. The results of this evaluation are described below.

The radiological effect of EPU on the Control Room Emergency Filtration System (CREF) is due to an increase in the iodine released during an accident. NMP2 has implemented the AST methodology which affects the design basis accident iodine release model (Reference 34). The analyses that supported the AST implementation were performed for 102% of EPU power level (i.e., 4067 MWt), and thus incorporate the increased EPU iodine release as well as the effects of the AST iodine release model. These analyses included the radiological consequences of the DBAs documented in Chapter 15 of the NMP2 USAR that potentially result in the most significant control room exposures (i.e., LOCA, MSLB, fuel handling accident (FHA) and CRDA). In all cases the control room doses were within regulatory limits.

The CREF system functions during a DBA to provide charcoal filtered outside air for personnel ventilation and pressurization of the control room envelope. Redundant radiation detectors are provided at the outside air intakes to automatically divert the outside air to the CREF system. With no change to the detection and controls, the operation of the CREF system is not impacted. The effect of EPU on control room doses was also included in the analyses that supported implementation of the AST methodology. The results of the AST analyses confirmed that the current iodine release and subsequent load on the CREF filters, which is based on the RG 1.3 model, remains bounding.

These analyses included the radiological consequences of the DBAs documented in Chapter 15 of the NMP2 USAR that potentially result in the most significant control room exposures (i.e., LOCA, MSLB, FHA and CRDA). In all cases the control room doses were within regulatory limits.

The quantities and locations of gases and hazardous chemicals that could affect the control room are unaffected by EPU. Therefore, EPU has no effect on the design basis potential toxic gas concentrations that are mitigated by the control room habitability system, and the current analyses remain bounding.

Conclusion

NMPNS has reviewed 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. NMPNS concludes that there is no increase in toxic gases that would result from the proposed EPU. There are no changes to the CRAV and CREF system configurations or system parameters as a result of EPU. NMPNS further concludes that the control room habitability system will continue to provide the required protection following implementation of the proposed EPU. Based on this, NMPNS concludes that the control room habitability system will continue to meet the requirements of GDCs 4 and 19. Therefore, NMPNS concludes that 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) for the fuel-handling building, control room, and areas containing ESF components. The NMPNS review focused on the effects of the proposed EPU on system functional design, environmental design, and provisions to preclude temperatures in the adsorber section from exceeding design limits. The NRC's acceptance criteria for ESF atmosphere cleanup systems are based on (1) GDC19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) total effective dose equivalent (TEDE), as defined in § 50.2, for the duration of the accident; (2) GDC41, 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) GDC61, insofar as it requires that systems that may contain radioactivity be designed to assure adequate safety under normal and postulated accident conditions; and (4) GDC64, 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.

Technical Evaluation

Control Room habitability and modeling are described in detail in a separate license amendment which implemented the Alternative Source Term (AST) methodology for NMP2 (Reference 34). This amendment is based on 4067 MWt (corresponds to the EPU power level of 3988 MWt with a 2% ECCS evaluation uncertainty factor applied) and complies with RG 1.183. This amendment along with its associated license amendment request (Reference 35) concluded that the control room post-accident doses will not exceed the values specified in 10 CFR 50.67 following AST implementation. It was also determined that continued compliance with NUREG-0737, Item II.B.2, will be maintained and that vital areas remain accessible post-accident.

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of Constant Pressure Power Uprates. Section 4.5 of the CLTR addresses the effect of Constant Pressure Power Uprate on the Standby Gas Treatment System. The results of this evaluation are described below.

The ESF atmosphere cleanup system at NMP2 is the Standby Gas Treatment System (SGTS).

The 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.

Two bounding analyses have been performed in the CLTR to evaluate 1) systems that implement AST in accordance with RG 1.183 and 2) systems that are committed to RG 1.3 for fission product transport. These evaluations are bounding for Mark I, II, & III containment designs and are highly conservative, because the power level from the largest BWR uprated to 120% of original licensed thermal power is combined with the BWR having the smallest containment volume. These bounding analyses envelop all GEH BWR SGTS operating conditions.

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]], the parameters used in the CLTR bounding analysis for AST application are confirmed to bound the NMP2 plant-specific values. Therefore, the SGTS at NMP2 is confirmed to be consistent with the generic description provided in the CLTR.

Details regarding the SGTS evaluation based on post-LOCA operation after EPU implementation are described in Section 2.5.2.

Conclusion

NMPNS has reviewed the assessment of the effects of the proposed EPU on the ESF atmosphere cleanup system, SGTS. NMPNS concludes that the assessment adequately accounted for the increase of fission products and changes in expected environmental conditions that would result from the proposed EPU, and further conclude that the SGTS will continue to provide adequate fission product removal in post accident environments for the ESF atmospheric cleanup following implementation of the proposed EPU. SGTS meets the generic dispositions described by the NRC approved CLTR. Based on this, NMPNS concludes that the requirements of GDCs 19, 41, 61, and 64 will continue to be met for the ESF atmospheric cleanup by the SGTS. Therefore, NMPNS finds the proposed EPU acceptable with respect to ESF atmospheric cleanup.

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 NMPNS review of the CRAVS focused on the effects that the proposed EPU will have on the functional performance of safety-related portions of the system. The NMPNS review included the effects of radiation, combustion, and other toxic products; and the expected environmental conditions in areas served by the CRAVS. The NRC's acceptance criteria for the CRAVS are based on (1) GDC4, 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) GDC19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) total effective dose equivalent (TEDE), as defined in 10 CFR 50.2, for the duration of the accident; and (3) GDC60, insofar as it requires that the plant design include means to control the release of radioactive effluents.

Technical Evaluation

The heating, ventilating 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 evaluation for EPU. The Control Room Area Ventilation System (CRAVS) maintains temperature and humidity conditions suitable for personnel comfort and for equipment reliable operation inside the control room envelope, which includes MCR, the Relay Room, and HVAC Equipment Room with Control Room Emergency Filtration (CREF) Units. The CRAVS also maintains the control room envelope at positive pressure to inhibit air infiltration. Heat loads for the control room area envelope include boundary transmission, lighting and equipment such as 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. There is no increase in toxic gases release that may result from EPU. The control of the concentration of airborne radioactive material in the control room envelope during AOOs and after postulated accidents is accomplished by the CREF system described in Section 2.7.1 in conjunction with the CRAVS. There is no change to the CRAV and CREF system configurations or system parameters as a result of EPU.

Conclusion

NMPNS has reviewed the assessment of the effects of the proposed EPU on the ability of the CRAVS to provide a controlled environment for the comfort and safety of control room personnel and to support the operability of control room components. NMPNS concludes that NMP2 has adequately accounted for the increase of toxic and radioactive gases that would result from a DBA under the conditions of the proposed EPU, and associated changes to parameters affecting environmental conditions for control room personnel and equipment. Accordingly, NMPNS concludes that the CRAVS will continue to provide an acceptable control room environment for safe operation of the plant following implementation of the proposed EPU. NMPNS also concludes that the system will continue to suitably control the release of gaseous radioactive effluents to the environment. Based on this, NMPNS concludes that the CRAVS will continue to meet the requirements of GDCs 4, 19, and 60. Therefore, NMPNS finds the proposed EPU acceptable with respect to the CRAVS.

2.7.4 Spent Fuel Pool Area Ventilation System

The NMP2 design does not include a separate spent fuel pool area ventilation system. The Reactor Building Ventilation System provides normal ventilation to the Spent Fuel Pool Area and is described in Section 2.7.5. The Standby Gas Treatment System functions to control radionuclide inventory in the Spent Fuel Pool Area, and its EPU evaluation is described in Sections 2.5.2.1 and 2.6.6. The Engineered Safety Feature Ventilation System is described in Section 2.7.6 and provides ventilation cooling to this area when the secondary containment is isolated during some AOOs.

2.7.5 Auxiliary and Radwaste Area and Turbine Areas Ventilation Systems

Regulatory Evaluation

The function of the drywell ventilation system (DVS), radwaste area ventilation system (RAVS) and the turbine area ventilation system (TAVS) is to maintain ventilation in the drywell, 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 NMPNS review focused on the effects of the proposed EPU on the functional performance of the safety-related portions of these systems. The NRC's acceptance criteria for the DVS, RAVS and TAVS are based on GDC60, insofar as it requires that the plant design include means to control the release of radioactive effluents.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of Constant Pressure Power Uprates. Section 6.6 of the CLTR addresses the effect of

Constant Pressure Power Uprate on CLTR Power Dependent Heating, Ventilation and Air Conditioning. The results of this evaluation are described below.

The Reactor Building ventilation system, Radwaste Building ventilation system, and the Turbine Building ventilation system evaluated in the CLTR are only those that are power dependent. The power dependent heating ventilation and air conditioning (HVAC) systems consist mainly of heating, cooling supply, exhaust, and recirculation units in the reactor, radwaste and turbine building.

The reactor, radwaste and the turbine building ventilation systems function to control concentration of airborne radioactive material in their service areas during normal operation and provide a means for movement of air from clean to progressively greater potentially contaminate areas prior to exhaust. Also these systems maintain the building at slightly negative pressure with respect to the outdoors to prevent unmonitored release due to air exfilration.

The reactor building ventilation system also supports three additional functions; 1) an ESF HVAC function that is described in Section 2.7.6 2) a drywell cooling function, and 3) a primary containment purge function.

Drywell ventilation is non-safety related, and consists of unit coolers within the drywell separated from the reactor building. This system provides an environment that ensures equipment performance within required temperature limits. EPU results in slightly higher process temperatures and small increases in the heat load due to higher electrical currents in some motors and cables.

The primary containment purge system in conjunction with the reactor building ventilation system and the Standby Gas Treatment System are used to control primary containment atmospheric gas concentrations and atmospheric pressure during normal operation.

Except as described in Section 2.7.6, these systems do not function to control the concentration of airborne radioactive material in these areas during AOOs, and after postulated accidents.

The affected areas are the steam tunnel in the turbine building, the drywell, feedwater heater bay, heater drain pumps, feedwater pumps and condensate/condensate booster pump areas of the turbine building. Other areas (including radwaste area) are unaffected by the EPU because the process temperatures are bounded by the pre-EPU analysis.

The heat load in the turbine building steam tunnel increases due to the increase in the feedwater temperature. The steam tunnel area coolers and main ventilation direct supply air to this area are capable of removing the heat load change.

The increase in heat load due to increased power requirements of the feedwater pump motors is within the capability of the pump area coolers. Similarly, the increase in heat load due to increase in heater drain pump motors is within the capability of the pump coolers. The increase heat load due to increased power requirements of the condensate and condensate booster pump motors is addressed by the addition of supplemental coolers.

In the drywell the increase in feedwater process temperature and slight increase in the recirculation pump motor horsepower are within the capability of the area coolers. Based on a review of design basis calculations and proposed modifications, the design of the HVAC for temperature, pressure and gas concentration control is adequate for the EPU.

None of the areas in the reactor building and auxiliary bays are affected by the EPU because the process temperatures remain relatively constant including the areas associated with the spent fuel pool and cooling system.

In summary, the building/areas where the heat gain will increase due to EPU are provided in Table 2.7-1, along with the planned modification for those cooling systems with no margin.

Conclusion

NMPNS has reviewed the assessment of the effects of the proposed EPU on the DVS, RAVS, and TAVS. NMPNS concludes that it has adequately accounted for the effects of the proposed EPU on the capability of these systems to maintain ventilation in the drywell, radwaste equipment areas, and in the turbine area, permit personnel access (except drywell), control the concentration of airborne radioactive material in these areas, and control release of gaseous radioactive effluents to the environment. Based on this, NMPNS concludes that the DVS, RAVS and TAVS will continue to meet the requirements of GDC60. Therefore, NMPNS finds the proposed EPU acceptable with respect to the DVS, RAVS and the TAVS.

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 NMPNS review for the ESFVS focused on the effects of the proposed EPU on the functional performance of the safety-related portions of the system. The NMPNS review also covered (1) the ability of the ESF equipment in the areas being serviced by the ventilation system to function under degraded ESFVS performance; (2) the capability of the ESFVS to circulate sufficient air to prevent accumulation of flammable or explosive gas or fuel vapor mixtures from components (e.g., storage batteries and stored fuel); and (3) the capability of the ESFVS to control airborne particulate material (dust) accumulation. The NRC's acceptance criteria for the ESFVS are based on (1) GDC4, 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) GDC17, insofar as it requires onsite and offsite electric power systems be provided to permit functioning of SSCs important to safety; and (3) GDC60, insofar as it requires that the plant design include means to control the release of radioactive effluents.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of Constant Pressure Power Uprates. Section 6.6 of the CLTR addresses the effect of Constant Pressure Power Uprate on CLTR Power Dependent Heating, Ventilation and Air Conditioning. The results of this evaluation are described below.

The ESF HVAC system is part of the Reactor Building Ventilation System consisting of local area cooling and recirculation units within the reactor building and auxiliary bays. EPU results in slightly higher process temperatures. This portion of the ESF HVAC system functions to control the concentration of airborne radioactive material in these areas during AOOs and after postulated accidents. The control of the concentration of airborne radioactive material in the secondary containment after postulated accidents is accomplished using the Standby Gas Treatment System described in Sections 2.5.2 and 2.6.6 in conjunction with these portions of the reactor building ventilation system.

The primary containment purge system, in conjunction with the reactor building ventilation system and the Standby Gas Treatment System, is used to control primary containment atmospheric gas concentrations and atmospheric pressure following postulated accidents in addition to the combustible gas control system described in Section 2.6.4.

None of the areas in the reactor building and auxiliary bays are affected by the EPU because the process temperatures remain relatively constant following postulated accidents.

Conclusion

NMPNS has reviewed the evaluation of the effects of the proposed EPU on the ESFVS. NMPNS has determined that the evaluation adequately accounted for the effects of the proposed EPU on the ability of the ESFVS to provide a suitable and controlled environment for ESF components following implementation of the proposed EPU. Based on this, NMPNS concludes that the ESFVS will continue to meet the requirements of GDCs 4, 17 and 60. Therefore, NMPNS finds the proposed EPU acceptable with respect to the ESFVS.

Table 2.7-1 EPU Impact on Building Heat Load

Building/Areas	System/Equipment	Current Load (BTU/h)	EPU Load (BTU/h)	Existing Cooling Cap (BTU/h)	Margin in %	Mod
Drywell	DVS Unit Coolers	4,107,672	4,255,388	4,317,000	1.5	No
Steam Tunnel	TAV Unit Coolers 2VT-UC210A/B	322,100	329,902	417,500	27	No
FW Pump Area 34	TAV Unit Coolers 2VT-UC210A-F	563,903	527,683	590,000	11.8	No
HD Pump Area 31	TAV Unit Coolers 2VT-UC201A&B	509,925	550,207	640,000	16	No
HD Pump Area 32	TAV Unit Coolers 2VT-UC202A&B	520,578	557,427	640,000	15	No
HD Pump Area 33	TAV Unit Coolers 2VT-UC203A&B	494,256	536,884	640,000	19	No
CD Pump Area 28	TAV Unit Coolers 2VT-UC208A&B	773,791	678,621	390,000	(-) 43	Yes. Note 1
CDB Pump Area 28	TAV Unit Coolers 2VT-UC207A&B	835,737	1,110,765	640,000	(-) 42	Yes. Note 1

Note:

1. Modification is planned to add Unit Coolers.

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. NMPNS reviewed the fuel system to ensure that (1) the fuel system is not damaged as a result of normal operation and AOOs, (2) fuel system damage is never so severe as to prevent control rod insertion when it is required, (3) the number of fuel rod failures is not underestimated for postulated accidents, and (4) coolability is always maintained. The NMPNS review covered fuel system damage mechanisms, limiting values for important parameters, and performance of the fuel system during normal operation, AOOs, and postulated accidents. The NRC's acceptance criteria are based on (1) 10 CFR 50.46, insofar as it establishes standards for the calculation of ECCS performance and acceptance criteria for that calculated performance; (2) GDC10, 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) GDC27, 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) GDC35, 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.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of Constant Pressure Power Uprates. Section 2.1 of the CLTR addresses the effect of Constant Pressure Power Uprate on the fuel system design. The results of this evaluation are described below.

The effect of EPU on the fuel design for NMP2 is described below.

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The CLTR requirement for confirmation of the generic applicability of the fuel product line design topic is that the CLTP core at NMP2 consists only of GE fuel types. NMP2 transitioned to GE14 fuel in Cycle 10 and will continue to use only GE 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 and the generic evaluation in the CLTR is acceptable.

Conclusion

NMPNS has reviewed the analyses related to the effects of the proposed EPU on the fuel system design of the fuel assemblies, control systems, and reactor core. NMPNS concludes that it has adequately accounted for the effects of the proposed EPU on the fuel system and demonstrated that (1) the fuel system will not be damaged as a result of normal operation and AOOs, (2) the fuel system damage will never be so severe as to prevent control rod insertion when it is required, (3) the number of fuel rod failures will not be underestimated for postulated accidents, and (4) coolability will always be maintained. Based on this, NMPNS concludes that the fuel system and associated analyses will continue to meet the requirements of 10 CFR 50.46, GDC10, GDC27, and GDC35 following implementation of the proposed EPU. Therefore, NMPNS finds the proposed EPU acceptable with respect to the fuel system design.

2.8.2 Nuclear Design

Regulatory Evaluation

NMPNS reviewed the nuclear design of the fuel assemblies, control systems, and reactor core to ensure that fuel design limits will not be exceeded during normal operation and anticipated operational transients, and that the effects of postulated reactivity accidents will not cause significant damage to the RCPB or impair the capability to cool the core. The NMPNS review covered core power distribution, reactivity coefficients, reactivity control requirements and control provisions, control rod patterns and reactivity worths, criticality, burnup, and vessel irradiation. The NRC's acceptance criteria are based on (1) GDC10, 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) GDC11, 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) GDC12, 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) GDC13, 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) GDC20, 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) GDC25, 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) GDC26, 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) GDC27, 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) GDC28, 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.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of Constant Pressure Power Uprates. Sections 2.1, 2.2, and 2.3 of the CLTR address the effect of Constant Pressure Power Uprate on the nuclear design. The results of this evaluation are described below.

The effect of EPU on the fuel design for NMP2 is described below.

2.8.2.1 Core Design

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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 limits as defined in the COLR.

2.8.2.2 Fuel thermal margin monitoring threshold

In general, 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 NMP2, at an EPU power level of 25%, the average bundle power is greater than 1.2 MWt.

For EPU, as specified in the CLTR, the fuel thermal margin monitoring threshold is scaled down, if necessary, 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 NMP2, the fuel thermal monitoring threshold is established at 23% of EPU RTP (25%*(1.2 / (0.25 * 3988 MWt / 764 bundles)) = 23%). A change in the fuel thermal monitoring threshold also requires a corresponding 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 NMP2 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 $\leq$ 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 for single loop operation will normally be 0.01 or 0.02 greater than the SLMCPR for two loop operation. A 0.02 value shall be added to the calculated cycle-specific SLMCPR value for both the single-loop and two-loop SLMCPR as required by Reference 6.

2.8.2.3.2 MCPR Operating Limit

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EPU operating conditions have only a small effect on the MCPR Operating Limit (OLMCPR). 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. Therefore, an EPU assessment is not required because the reload core at the increased power will be analyzed and the applicable MCPR Operating Limit will be determined.

The MCPR Operating Limit at NMP2 is confirmed to be consistent with the generic description provided in the CLTR because the MCPR Operating Limit will be evaluated for the uprated reload core prior to EPU implementation.

2.8.2.3.3 MAPLHGR and LHGR Operating Limits

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EPU operating conditions do not usually affect the MAPLHGR or Maximum LHGR Operating Limits. The MAPLHGR and Maximum LHGR Operating Limits ensure that the plant does not exceed regulatory limits established in 10 CFR 50.46 or by the fuel design limits. The MAPLHGR Operating Limit is determined by analyzing the limiting 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. The reload analysis confirms that the MAPLHGR and Maximum LHGR Operating Limits for each reload fuel bundle design remain acceptable.

The MAPLHGR and Maximum LHGR Operating Limits at NMP2 are confirmed to be consistent with the generic description provided in the CLTR because the MAPLHGR and Maximum LHGR Operating Limits will be evaluated for the uprated reload core prior to EPU implementation.

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, and is also applicable for an 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. Because plant shutdown and reactivity margins must meet NRC approved limits established in Reference 4 on a cycle specific basis and are evaluated for each plant reload core, separate hot excess reactivity and shutdown margin analyses are not required for EPU.

The NMP2 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 Additional Topics from GEH Licensing Topical Report NEDC-33173P, "Applicability of GE Methods to Expanded Operating Domains,"

GEH Licensing Topical Report NEDC-33173P, "Applicability of GE Methods to Expanded Operating Domains," (Reference 6) was approved by the NRC in January 2008. Section 9.0 of the NRC Safety Evaluation contains several limitations and conditions that require additional topics to be addressed in EPU applications. The additional topics related to Nuclear Design are provided below.

2.8.2.5.1 Steady-State 5 Percent Bypass Voiding Evaluations

Limitation and Condition 9.17 of Reference 6 requires the bypass voiding to be evaluated on a cycle-specific basis to confirm that the void fraction remains below 5 percent. Limitation and Condition 9.17 is applicable to EPU conditions consistent with Reference 62.

The best-estimate means of determining 4-channel bypass void fraction is with TRACG. TRACG was applied in response to Methods RAI 14 (Reference 63). TRACG is capable of accurately modeling bypass heating and cross flow.

A conservative approach (ISCOR) was discussed in References 64 and 65. ISCOR conservatively calculates hot bypass channel voiding using its direct moderator-heating model and providing no credit for cross flow while applying additional conservatism with bounding 4-bundle peaking. The use of ISCOR is a more simplified and efficient process to implement compared to the use of TRACG and typically demonstrates margin to the 5% bypass void fraction requirement at the LPRM D Level.

For NMP2 reload core prior to EPU implementation, a calculation will be performed with the conservative ISCOR process at licensed EPU core power and minimum core flow (e.g. 120 % CLTR, 99% flow). The purpose of the calculation is to confirm that the bypass void fraction remains below 5 percent at all LPRM levels when operating at steady-state conditions within the licensed operating domain consistent with Reference 62.

If the resulting bypass void fraction is found to exceed the 5% requirement, it is acceptable to relax the conservative ISCOR input assumptions as long as the overall approach can be demonstrated to remain conservative relative to TRACG. It is also acceptable to perform a cycle-specific TRACG analysis with consideration of assumptions that will tend to maximize bypass void fraction (e.g. bypass flow and 4-bundle peaking).

The highest calculated bypass voiding at any LPRM level will be provided with the plant-specific SRLR.

For the representative cycle core used in EPU evaluation the steady-state 5% bypass voiding evaluation is provided in the following table:

% of Rated Core Power (EPU)	% of Rated Core Flow (EPU)	Hot Channel Void Fraction in Bypass Region at Instrumentation D Level
100%	100	0.026
100%	99	0.027

2.8.2.5.2 Power-to-Flow Ratio

Limitation and Condition 9.3 requires plant-specific EPU applications to confirm that the core thermal power to core flow ratio will not exceed 50 MWt/Mlbm/hr at the low flow point at rated power (e.g., EPU: 100% Power / 99% Flow) in the allowed operating domain. The core thermal power to total core flow ratio is reported in the following table:

% of Rated Core Power (EPU)	% of Rated Core Flow (EPU)	Power-to-Flow Ratio (MWt/Mlbm/hr)
100	99%	37.13

The 120% OLTP power (i.e. the EPU power) is 3988 MWt and the 99% rated flow is 107.42 Mlbm/hr. As seen from the table, the core thermal power to total core flow ratio does not exceed 50 MWt/Mlbm/hr. This power to flow ratio analysis is consistent with GEH letter, Implementation of Methods limitations – NEDC-33173P (TAC No. MD0277)," September 18, 2008 (Reference 62).

The power-flow map is independent of fuel design and does not change cycle to cycle. Therefore, the power to flow ratio for NMP2's future EPU cycles will also remain below 50 MWt/Mlbm/hr.

2.8.2.5.3 R-Factor

Limitation and Condition 9.6 requires the plant specific R-factor calculation at a bundle level be consistent with lattice axial void conditions expected for the hot channel operating state.

The GE14 bundle R-factors generated for this task are consistent with GNF standard design procedures which use an axial void profile shape with 50% average in-channel voids. This is consistent with lattice axial void conditions expected for the hot channel operating state.

Figure 2.8-19 shows bundle average void fractions corresponding to hot channels with low critical power ratios (MCPRs) from the NMP2 EPU core. The figure demonstrates that the generic R-factor profile, with an average void fraction of 0.50, is representative of the MCPR-limiting void conditions predicted by PANAC11.

2.8.2.5.4 Plant-Specific Application

Limitation and Condition 9.24 requires plant-specific EPU applications to provide a prediction of key parameters for cycle exposures for operation at EPU. The following parameters: (1)

Maximum Bundle Power; (2) Flow for Peak Bundle Power; (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 NMP2 data are plotted with the available EPU experience base as required by Limitation and Condition 9.24.

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 beginning of cycle (BOC), middle of cycle (MOC), and EOC. Because the minimum margins to specific limits occur at exposures other than the traditional BOC, MOC, and EOC, the data are provided at these other exposures as applicable (Figures 2.8-17 and 2.8-18). The bundle power is dimensionless. To obtain the bundle power in MWt, multiply each value by the average bundle power of 5.2199. The average bundle power is equal to 3988/764, where 3988 MWt is the EPU RTP and 764 is the total number of bundles in the core.

2.8.2.5.5 Application of 10 Weight Percent Gd

Limitation and Condition 9.13 requires application of 10 weight percent Gd to EPU applications.

For NMP2, the maximum burnable poison concentration used is 8.0 Weight Percent Gd_2O_3; therefore, Limitation and Condition 9.13 is not applicable.

Conclusion

NMPNS has reviewed the analyses related to the effect of the proposed EPU on the nuclear design of the fuel assemblies, control systems, and reactor core. NMPNS concludes that it has adequately accounted for the effects of the proposed EPU on the nuclear design and has demonstrated that the fuel design limits will not be exceeded during normal or anticipated operational transients, and that the effects of postulated reactivity accidents will not cause significant damage to the RCPB or impair the capability to cool the core. Based on this evaluation and in coordination with the reviews of the fuel system design, thermal and hydraulic design, and transient and accident analyses, NMPNS concludes that the nuclear design of the fuel assemblies, control systems, and reactor core will continue to meet the applicable requirements of GDCs 10, 11, 12, 13, 20, 25, 26, 27, and 28. Therefore, NMPNS finds the proposed EPU acceptable with respect to the nuclear design.

2.8.3 Thermal and Hydraulic Design

Regulatory Evaluation

NMPNS reviewed the thermal and hydraulic design of the core and the RCS to confirm that the design (1) has been accomplished using acceptable analytical methods, (2) is equivalent to or a justified extrapolation from proven designs, (3) provides acceptable margins of safety from conditions which would lead to fuel damage during normal reactor operation and AOOs, and (4) is not susceptible to thermal-hydraulic instability. The review also covered hydraulic loads on the core and RCS components during normal operation and DBA conditions and core thermal-hydraulic stability under normal operation and anticipated transients without scram (ATWS) events. The acceptance criteria are based on (1) GDC10, 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) GDC12, insofar as it requires that the reactor core and associated coolant, control, and protection systems be designed to assure that power oscillations, which can result in conditions exceeding SAFDLs, are not possible or can reliably and readily be detected and suppressed.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of Constant Pressure Power Uprates. Section 2.4.4 of the CLTR addresses the effect of Constant Pressure Power Uprate on Plants with Enhanced Option III. Section 9.3.3 of the CLTR addresses the effect of Constant Pressure Power Uprate on ATWS with Core Instability. The results of this evaluation are described below.

Section 3.2 of ELTR1 documents interim corrective actions and four long-term stability options. NMP2 has adopted Option III (Reference 66). Option III evaluations are core reload dependent and are performed for each reload fuel cycle. The NMP2 Option III hardware has been installed and connected to the Reactor Protection System. In the event that the Oscillation Power Range Monitor (OPRM) system is declared inoperable, NMP2 will operate under the BWROG Guidelines for Backup Stability Protection (BSP) as described in Reference 67. After the EPU is implemented, cycle specific setpoints and BSP regions will be determined and documented in the same Supplemental Reload Licensing Report (SRLR).

2.8.3.1 Plants with Enhanced Option III

The Option III solution combines closely spaced LPRM detectors into "cells" to effectively detect either core-wide or regional modes of reactor instability. These cells are termed OPRM cells and are configured to provide local area coverage with multiple channels. The NMP2 Option III hardware combines the LPRM signals and evaluates 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 (GRA), offer a high degree of assurance that fuel failure will not occur as a consequence of stability related oscillations.

The OPRM trip is armed only when plant operation is within the OPRM trip-enabled region. The OPRM trip-enabled region is generically defined as the region on the power/flow map with power ≥ 30% of RTP and core flow ≤ 60% of rated core flow. For EPU, the NMP2 OPRM 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% CLTP boundary is rescaled to the 26% EPU thermal power limit using the CLTP/EPU ratio.

The NMP2 OPRM trip-enabled region is shown in Figure 2.8-20. 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 trip-enabled region will be confirmed for each reload.

Stability Option III provides SLMCPR protection by generating a reactor scram if a reactor instability that exceeds the specified trip setpoint is detected. The OPRM setpoint is determined per an NRC approved methodology (References 66 and 68).

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 Delta critical power ratio (CPR) over Initial CPR Versus Oscillation Magnitude (DIVOM) curve (Delta Critical Power Ratio Over Initial Critical Power Ratio 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 Mode DIVOM Guideline (Reference 68).

The generic analyses for the Option III hot channel oscillation magnitude and the OPRM hardware were designed to be independent of core power. [[

]] (Reference 69).

Although the Option III solution requires cycle-specific evaluations, a demonstration analysis was performed based on an equilibrium GE14 core using nominal core simulator wrapups at limiting conditions. For this analysis, a DIVOM slope of 0.45 was used. As shown in Table 2.8-1, with an assumed SLMCPR of 1.09, rated OLMCPR of 1.39 and an off-rated OLMCPR (at 45% Flow) of 1.60, an OPRM setpoint of 1.14 is the highest setpoint that may be used without stability setting the OLMCPR. The actual setpoint will be established in accordance with NMP2 Technical Specifications.

Consistent with Limitation 9.18 per Reference 6, the OPRM system will incorporate a 5% calibration error on the OPRM setpoints to address the bypass voiding uncertainty at low-flow conditions. Since NMP2 utilizes the Option III stability solution, the APRM calibration error required by Limitation 9.18 is not applicable (see Section 6.2 of Reference 6). This calibration error has been included in the OPRM Amplitude Setpoints shown in Table 2.8-1. This is consistent with Reference 62, the OPRM setpoints are based on an OLMCPR that does not incorporate the OLMCPR adder (0.01 adder based on the void fraction uncertainty).

2.8.3.2 BSP Evaluation

NMP2 implements Backup Stability Protection (BSP) (Reference 67) as the stability licensing basis if the Option III OPRM system is declared inoperable. The BSP evolved from the stability interim corrective actions (ICAs) (Reference 70), 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.

If the Option III OPRM system is declared inoperable, implementation of the associated BSP regions will constitute the stability licensing basis for NMP2 (Reference 67). 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) in Reference 70. The standard ICA region endpoints on the high flow control line (HFCL) and on the Natural Circulation Line (NCL) define the base BSP region endpoints on the HFCL and on the NCL. The bounding plant-and-cycle-specific BSP region endpoints must enclose the corresponding base BSP region endpoints on the HFCL and the NCL. If a calculated BSP region endpoint is located inside the corresponding base BSP region endpoint, the corresponding base BSP region endpoint must replace it. Therefore, the selected points will result in the largest, or most conservative, region sizes. The proposed BSP Scram and Controlled Entry region boundaries may be constructed by connecting the corresponding bounding endpoints 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 2.8-2. 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 2.8-21.

2.8.3.3 ATWS with Core Instability

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[illegible]	[illegible]	[illegible]
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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 71 and 72 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 the pre-EPU 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. Reference 72 and the associated NRC SER concluded that the analyzed operator actions effectively mitigate an ATWS instability event and these conclusions are applicable to the operating conditions expected for EPU at NMP2.

The EPU effect on ATWS with core instability at NMP2 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

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

2.8.4 Emergency Systems

2.8.4.1 Functional Design of Control Rod Drive System

Regulatory Evaluation

The NMPNS review covered the functional performance of the control rod drive (CRD) system to confirm that the system can effect a safe shutdown, respond within acceptable limits during AOOs, and prevent or mitigate the consequences of postulated accidents. The review also covered the CRD system cooling system to ensure that it will continue to meet its design requirements. The NRC's acceptance criteria are based on (1) GDC4, 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) GDC23, insofar as it requires that the protection system be designed to fail into a safe state; (3) GDC25, 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) GDC26, 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) GDC27, 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) GDC28, 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) GDC29, 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.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of Constant Pressure Power Uprates. Section 2.5 of the CLTR addresses the effect of Constant Pressure Power Uprate on the functional design of the CRD system. The results of this evaluation are described below.

The 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 NMP2 ARI system is not affected by EPU because it has no thermal power dependency. The topics addressed in this evaluation for NMP2 are:

2.8.4.1.1 Control Rod Scram

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For pre-BWR/6 plants, the scram times are decreased by the transient pressure response, and therefore the effect of EPU is bounded by the current response. 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. Because the nominal reactor dome pressure for EPU does not change, the scram time performance relative to current plant operation is the same. Therefore, the current Technical Specification scram requirements are applicable. For pre-BWR/6 plants, the generic scram times for American Society of Mechanical Engineers (ASME) overpressure protection and critical power ratio pressurization transient analyses are not adversely affected by the reactor transient pressure and, therefore, remain valid.

The CRD system control rod scram at NMP2 is confirmed to be consistent with the generic description provided in the CLTR for pre-BWR/6 plants because the NMP2 is a BWR/5 plant.

2.8.4.1.2 Control Rod Drive Positioning and Cooling

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	]]	Reactor dome pressure is unchanged for EPU. [[ ]]
[[	]]	Reactor dome pressure is unchanged for EPU. [[

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There is a slight increase in the pressure above the core plate (approximately 1 psi for EPU), and the automatic operation of the system flow control valve maintains the required drive water pressure and cooling water flow rate. Therefore, the CRD positioning and cooling functions are not affected. The CRD cooling and normal CRD positioning functions are operational considerations, not safety-related functions, and are not affected by EPU operating conditions.

Plant operating data has confirmed that the CRD system flow control valve operating position has sufficient operating margin. Therefore, the CRD system drive positioning and cooling at NMP2 is confirmed to be consistent with the generic description provided in the CLTR.

2.8.4.1.3 Control Rod Drive Integrity Assessment

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The postulated abnormal operating condition for the CRD design assumes a failure of the CRD system pressure-regulating valve that applies the maximum pump discharge pressure to the CRD mechanism internal components. This postulated abnormal pressure bounds the ASME reactor overpressure limit. The reactor operating conditions for an EPU do not affect the CRD pump discharge pressure. Therefore, the stresses for the limiting CRD component do not change. Other mechanical loadings are addressed in Sections 2.2.2 and 2.2.3 of this report.

The CRD system integrity at NMP2 is confirmed to be consistent with the generic description provided in the CLTR because the reactor overpressure limit is not exceeded.

Conclusion

NMPNS has reviewed the analyses related to the effects of the proposed EPU on the functional design of the CRD system. NMPNS concludes that they have adequately accounted for the effects of the proposed EPU on the CRD system and has demonstrated that the system's ability to effect 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. NMPNS further concludes that sufficient cooling exists to ensure the system's design bases will continue to be met upon implementation of the proposed EPU. Based on this, NMPNS concludes that the fuel system and associated analyses will continue to meet the requirements of GDCs 4, 23, 25, 26, 27, 28, and 29, and 10 CFR 50.62(c)(3) following implementation of the proposed EPU. Therefore, NMPNS finds the proposed EPU acceptable with respect to the functional design of the CRD system.

2.8.4.2 Overpressure Protection During Power Operation

Regulatory Evaluation

Relief and safety valves and the reactor protection system provide overpressure protection for the RCPB during power operation. The NMPNS review covered relief and safety valves on the main steam lines and piping from these valves to the suppression pool. The NRC's acceptance criteria are based on (1) GDC15, 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) GDC31, insofar as it requires that the RCPB be designed with sufficient margin to assure that it behaves in a nonbrittle manner and that the probability of rapidly propagating fracture is minimized.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of Constant Pressure Power Uprates. Section 3.1 of the CLTR addresses the effect of Constant Pressure Power Uprate on Nuclear System Pressure Relief/Overpressure Protection. The results of this evaluation are described below.

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).

For NMP2, 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 (TT) 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 analyzed 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 MSIVF 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 1050 psia (which is higher than the nominal EPU dome pressure), and two 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 1316 psig. The corresponding calculated maximum reactor dome pressure is 1286 psig. The peak calculated RPV pressure remains below the 1375 psig ASME limit, and the maximum calculated dome pressure remains below the Technical Specification 1325 psig Safety Limit. Therefore, the results are acceptable and within the applicable limits. The results of EPU overpressure protection analysis for the NMP2 MSIVF event are consistent with the generic analysis in ELTR2. The NMP2 response to the MSIVF event is provided as Figure 2.8-22.

The MSIVF event is performed using the NRC approved code ODYN (Reference 73) (see Table 1-1).

Conclusion

NMPNS has reviewed the analyses related to the effects of the proposed EPU on the overpressure protection capability of the plant during power operation. NMPNS concludes that it has (1) adequately accounted for the effects of the proposed EPU on pressurization events and overpressure protection features and (2) demonstrated that the plant will continue to have sufficient pressure relief capacity to ensure that pressure limits are not exceeded.. Based on this, NMPNS concludes that the overpressure protection features will continue to meet GDCs 15 and 31 following implementation of the proposed EPU. Therefore, NMPNS finds the proposed EPU 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 NMPNS review covered the effect of the proposed EPU on the functional capability of the system. The NRC's acceptance criteria are based on (1) GDC4, insofar as it requires that SSCs important to safety be protected against dynamic effects; (2) GDC5, 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) GDC29, 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) GDC33, 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) GDC34, 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) GDC54, 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.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of Constant Pressure Power Uprates. Section 3.9 of the CLTR addresses the effect of Constant Pressure Power Uprate on the Reactor Core Isolation Cooling System. The results of this evaluation are described below.

The RCIC system evaluation for EPU at NMP2 addressed the following topics:

- System performance and hardware
- Net positive suction head
- Adequate core cooling for limiting LOFW events (Addressed in Section 2.8.5.2.3)
- Inventory makeup - Operational (Reactor Water) Level 1 avoidance (Addressed in Section 2.8.5.2.3)

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The RCIC system is required to maintain sufficient water inventory in the reactor to permit adequate core cooling following a reactor vessel isolation event accompanied by loss of flow from the FW system. The system design injection rate must be sufficient for compliance with the system limiting criteria to maintain the reactor water level above top of active fuel (TAF) at EPU conditions. The RCIC system is designed to pump water into the reactor vessel over a wide range of operating pressures. The results of the NMP2 plant-specific evaluation indicate adequate water level margin above TAF at EPU conditions. Thus, the RCIC injection rate is adequate to meet this design basis event.

An operational requirement is that the RCIC system can restore the reactor water level while avoiding Automatic Depressurization System (ADS) timer initiation and MSIV closure activation functions associated with the low-low-low reactor water level setpoint (Level 1). This requirement is intended to avoid unnecessary initiations of safety systems. The results of the NMP2 plant-specific evaluation indicates that the RCIC system is capable of maintaining the water level outside the shroud above nominal Level 1 setpoint through a limiting LOFW event at EPU conditions. Thus, the RCIC injection rate is adequate to meet the requirements for inventory makeup. The reactor system response to a LOFW transient with RCIC is discussed in Section 2.8.5.2.3.

For EPU, there is no change to the normal reactor operating pressure and the SRV setpoints remain the same. There is no change to the maximum specified reactor pressure for RCIC system operation, no changes to the RCIC system performance parameters, and no effect on the maximum reactor pressure postulated to be present during system startup. Therefore, no changes are required to meet the performance requirements for the RCIC system or to limit the maximum startup transient speed peak.

The NPSH available for the RCIC pump does not change because there are no physical changes to the pump suction configuration, and no changes to the system flow rate or minimum atmospheric pressure in the suppression chamber or condensate storage tank (CST). EPU does not affect the capability to transfer the RCIC pump suction on high suppression pool level or low CST level from its normal alignment, the CST, to the suppression pool, and does not change the existing requirements for the transfer. For ATWS and fire protection, operation of the RCIC system at suppression pool temperatures greater than the operational limit may be accomplished by using the dedicated CST volume as the source of water. Therefore, the specified operational temperature limit for the process water does not change with the EPU. The NPSH required by the RCIC pump does not change because there is no change to the maximum rated pump speed or the required pump flow rate. The effect of EPU on the operation of the RCIC system during SBO events is discussed in Section 2.3.5

The RCIC system at NMP2 is confirmed to be consistent with the generic description provided in the CLTR. No RCIC system power dependent functions or operating requirements (flows, pressure, temperature, and NPSH) are added or changed from the original design or licensing bases.

Conclusion

NMPNS has reviewed the analyses related to the effects of the proposed EPU on the ability of the RCIC system to provide decay heat removal following an isolation of main feedwater event and a station blackout event and the ability of the system to provide makeup to the core following a small break in the RCPB. NMPNS concludes that it has adequately accounted for the effects of the proposed EPU on these events and demonstrated that the RCIC system will continue to provide sufficient decay heat removal and makeup for these events following implementation of the proposed EPU. Based on this, NMPNS concludes that the RCIC system will continue to meet the requirements of GDCs 4, 5, 29, 33, 34 and 54, and 10 CFR 50.63 following implementation of the proposed EPU. Therefore, NMPNS finds the proposed EPU 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 a low pressure system, which takes over the shutdown cooling function when the RCS temperature is reduced. The NMPNS review covered the effect of the proposed EPU on the functional capability of the RHR system to cool the RCS following shutdown and provide decay heat removal. The NRC's acceptance criteria are based on (1) GDC4, insofar as it requires that SSCs important to safety be protected against dynamic effects; (2) GDC5, 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) GDC34, which specifies requirements for an RHR system.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of Constant Pressure Power Uprates. Section 3.10 of the CLTR addresses the effect of Constant Pressure Power Uprate on the RHR system. The results of this evaluation are described below.

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 NMP2, 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, Fuel Pool Cooling (FPC) (Supplemental Spent Fuel Pool Cooling) assist and Steam Condensing mode (SCM).

The LPCI mode, as it relates to the LOCA response, is discussed in Section 2.8.5.6.2, which concludes that 10 CFR 50.46 limits are met at EPU conditions.

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).

Suppression pool temperatures for all EPU events remain within the design limits. Therefore, the suppression pool temperature during a postulated LOCA at EPU conditions does not change the capabilities of RHR system equipment to perform the LPCI, SPC and CSC functions. Containment pressures for EPU events increased slightly above the CLTP analyzed pressures, but remained below the existing Technical Specification peak containment internal pressure, Pa. The slight increase in the predicted containment pressure during a postulated LOCA at EPU conditions (1.4 psi per Table 2.6-1) remains within the equipment design parameters and thus does not adversely affect the hardware capabilities of RHR system equipment to perform the LPCI, SPC and CSC functions.

The FPC Assist (Spent Fuel Pool Cooling) mode, using existing RHR system 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 Cleanup (FPCC) system. The adequacy of fuel pool cooling, including use of the Spent Fuel Pool Cooling mode, is addressed in Section 2.5.3.1.

The effects of EPU on the remaining modes are discussed in the following subsections.

Shutdown Cooling Mode

The Technical Specifications require Mode 4 (≤ 200°F) be achieved in 36 hours under certain conditions when a LCO is not met. For EPU, the shutdown cooling analysis shows that the reactor can be cooled to 212°F in 14.7 hours at EPU conditions. This is an increase from 9.75 hours at CLTP conditions, but considerably less than the 36-hour requirement of the Technical Specifications.

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 in service, is increased from 14.3 hours at CLTP conditions to approximately 18.5 hours at EPU conditions. This calculated normal reactor shutdown time satisfies the USAR Section 5.4.7 time criterion of 20 hours, which was selected based on engineering judgment to ensure that SDC operation impact on a normal reactor shutdown schedule is minimized. This SDC design criterion was used as one of the bases for sizing the RHR system 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 impact outage schedules. This impact 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.

Steam Condensing Mode

SCM is not a safety-related mode and is not routinely used. The SCM was designed to maintain the reactor in a hot standby condition when the reactor is isolated from the main condenser so that an equipment malfunction can be corrected. The objective of the SCM is to permit a timely return to power operation when the reactor is no longer isolated. This mode was originally designed so that (1) all of the steam from the reactor could be condensed in both of the RHR system heat exchangers (HXs) at 30 minutes following a reactor scram, and (2) one RHR system HX is capable of condensing all of the steam generated in the reactor at 90 minutes following the reactor scram. The increased decay heat due to EPU increases the RHR system heat exchanger heat load duty. The effect of EPU extends condensing times and is only an operational consideration and is not a safety concern.

Conclusion

NMPNS has reviewed the analyses related to the effects of the proposed EPU on the RHR system. NMPNS concludes that it has adequately accounted for the effects of the proposed EPU on the system and demonstrated that the RHR system will maintain its ability to cool the RCS following shutdown and provide decay heat removal. Based on this, NMPNS concludes that the RHR system will continue to meet the requirements of GDCs 4, 5, and 34 following implementation of the proposed EPU. Therefore, NMPNS finds the proposed EPU 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 functions by injecting a boron solution into the reactor to effect shutdown. The NMPNS review covered the effect of the proposed EPU on the functional capability of the system to deliver the required amount of boron solution into the reactor. The NRC's acceptance criteria are based on (1) GDC26, 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) GDC27, 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. As stated in NMP2 Technical Specification Bases B 3.1.7, Standby Liquid Control System, the SLCS is manually initiated from the main control room, as directed by the emergency operating procedures, if the operator believes the reactor cannot be shut down, or kept shut down, with the control rods. The SLCS can also be automatically initiated as required by 10 CFR 50.62.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of Constant Pressure Power Uprates. Section 6.5 of the CLTR addresses the effect of Constant Pressure Power Uprate on SLCS. The results of this evaluation are described below.

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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		]] NMP2 uses only GE fuel through GE14. The SLCS shutdown capability is reevaluated for each reload core.

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 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 780 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 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 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 upper plenum pressure following the limiting ATWS event reaches 1221.3 psig (1236 psia) during the time the SLCS is analyzed to be in operation. Consequently, there is a corresponding increase in the maximum pump discharge pressure to 1326.4 psig and a 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 re-close 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 re-close 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 relief valve setpoint margin is 31.6 psi. This margin is based on a SLCS pump relief valve setpoint of 1358 psig (1400 psig minus 3% tolerance). 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. There are no timer setting changes for EPU for NMP2, and the ATWS analysis confirms acceptable results.

Conclusion

NMPNS has reviewed the analyses related to the effects of the proposed EPU on the SLCS and concludes that it has adequately accounted for the effects of the proposed EPU on the system and demonstrated that the system will continue to provide the function of reactivity control independent of the control rod system following implementation of the proposed EPU. Based on this, NMPNS concludes that the SLCS will continue to meet the requirements of GDCs 26 and 27, and 10 CFR 50.62(c)(4) following implementation of the proposed EPU. Therefore, NMPNS finds the proposed EPU acceptable with respect to the SLCS.

2.8.4.6 Reactor Recirculation System Performance

The Reactor Recirculation System (RRS) performance is addressed in Section 3.6 of the CLTR.

The EPU power condition is accomplished by operating along extensions of current rod lines on the power/flow map with no increase in the maximum core flow. The core reload analyses are performed with the most conservative allowable core flow. The evaluation of the RRS performance at EPU power determines that adequate core flow can be maintained.

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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The CLTR shows that recirculation pump NPSH at full EPU power does not significantly increase the NPSH required or significantly reduce the NPSH margin.

The NMP2 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, and the flow mismatch limits are not affected because a detailed ECCS evaluation was not required.

SLO is limited to off-rated conditions and is not affected by EPU.

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2.8.5 Accident and Transient Analyses

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 NMPNS review consisted of confirmation of the applicability of the [[ ]] in the NRC approved NEDC-33004P-A, "Constant Pressure Power Uprate" for the specific NMP2 application. The NRC's acceptance criteria are based on (1) GDC10, 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) GDC15, 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) GDC20, 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) GDC26, 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.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of Constant Pressure Power Uprates. Section 9 of the CLTR addresses the effect of Constant Pressure Power Uprate on sources for excessive heat removal events. The results of this evaluation are described below.

Section 9 of the CLTR and the response to CLTR NRC RAI Set 9 Number 14 RXSB 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 NMP2 reload evaluation scope. [[

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The Increase in Steam Flow event ([[ ]]) is [[ ]] within the reload evaluation scope. [[ ]]

The Inadvertent Opening of a Safety Relief Valve event is [[ ]]

Conclusion

NMPNS reviewed the CLTR [[ ]] for the excessive heat removal events and concludes that it applies to NMP2. Applicability of this generic disposition to NMP2 demonstrates that the reactor protection and safety systems will continue to ensure that the SAFDLs and the RCPB pressure limits will not be exceeded as a result of this event. Based on this, NMPNS concludes that NMP2 will continue to meet the requirements of GDCs 10, 15, 20, and 26 following implementation of the proposed EPU. Therefore, NMPNS finds the proposed EPU acceptable with respect to the excessive heat removal events.

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 NMPNS review consisted of confirmation of the applicability of the [[ ]] in the NRC approved NEDC-33004P-A, "Constant Pressure Power Uprate" for the specific NMP2 application. The NRC's acceptance criteria are based on (1) GDC10, 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) GDC15, 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) GDC26, 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.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of Constant Pressure Power Uprates. Section 9 of the CLTR addresses the effect of Constant Pressure Power Uprate on sources for decrease in heat removal events. The results of this evaluation are described below.

Section 9 of the CLTR and the response to CLTR NRC RAI Set 9 Number 14 RXSB 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 NMP2 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 NMP2 reload evaluation scope.

A Pressure Regulator Failure Downscale (PRFD) for all GEH BWRs results in a very mild pressure change where the backup regulator takes control without a scram. The small pressure and power changes make this a non-limiting event.

Conclusion

NMPNS reviewed the CLTR [[ ]] for the decrease in heat removal events and concludes that this applies to NMP2. Applicability of this [[ ]] to NMP2 demonstrates that the reactor protection and safety systems will continue to ensure that the SAFDLs and the RCPB pressure limits will not be exceeded as a result of this event. Based on this, NMPNS concludes that NMP2 will continue to meet the requirements of GDCs 10, 15, and 26 following implementation of the proposed EPU. Therefore, NMPNS finds the proposed EPU acceptable with respect to the decrease in heat removal events.

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 coastdown 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 NMPNS review consisted of confirmation of the applicability of the [[ ]] in the NRC approved NEDC-33004P-A, "Constant Pressure Power Uprate" for the specific NMP2 application. The NRC's acceptance criteria are based on (1) GDC10, 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) GDC15, 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) GDC26, 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.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of Constant Pressure Power Uprates. Section 9 of the CLTR addresses the effect of Constant Pressure Power Uprate on Loss of Non-Emergency AC Power to the Station Auxiliaries. The results of this evaluation are described below.

Section 9 of the CLTR and the response to CLTR NRC RAI Set 9 Number 14 RXSB 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

NMPNS reviewed the CLTR [[ ]] for the loss of nonemergency AC power to station auxiliaries event and concludes that it applies to NMP2. Applicability of this [[ ]] to NMP2 demonstrates that the reactor protection and safety systems will continue to ensure that the SAFDLs and the RCPB pressure limits will not be exceeded as a result of this event. Based on this, NMP2 will continue to meet the requirements of GDCs 10, 15, and 26 following implementation of the proposed EPU. Therefore, NMPNS finds the proposed EPU acceptable with respect to the loss of nonemergency 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 NMPNS review covered (1) the sequence of events, (2) the analytical model used for analyses, (3) the values of parameters used in the analytical model, and (4) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC10, 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) GDC15, insofar as it requires that the RCS and its associated auxiliary systems be designed with margin sufficient to ensure that the design conditions of the RCPB are not exceeded during any condition of normal operation; and (3) GDC26, 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.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of Constant Pressure Power Uprates. Section 9.1.3 of the CLTR addresses the effect of Constant Pressure Power Uprate on Loss of Water Level Events. The results of this evaluation are described below.

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 NMP2 at EPU conditions. This analysis assumed failure of the HPCS 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. 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.

As discussed in Section 2.8.4.3, an operational requirement is that the RCIC system restores the reactor water level while avoiding ADS timer initiation and MSIV closure activation functions associated with the low-low-low reactor water level setpoint (Level 1). This requirement is intended to avoid unnecessary initiations of safety systems. This requirement is not a safety-related function. The results of the LOFW analysis for NMP2 show that the nominal Level 1 setpoint trip is avoided.

As described in the PUSAR, Table 1-1, for Transient Analysis, the modeling tool used is the SAFER04 model, which is the same model used in the ECCS LOCA analysis. The general sequence of events in the analysis is as follows. The reactor is assumed to be at 102% of the EPU power level when the LOFW occurs. The initial level in the model is conservatively set at the low-level scram setpoint and reactor feedwater is instantaneously isolated at event initiation. Scram is initiated at the start of the event. When the level decreases to the low-low level setpoint, the RCIC system is initiated. The RCIC flow to the vessel begins at 68 seconds into the event, minimum level is reached at 1007 seconds and level is recovered after that point. Only RCIC flow is credited to recover the reactor water level. There are no additional failures assumed beyond the failure of the HPCS system. The only other key analysis assumption for the LOFW analysis, discussed in Section 9.1.3 of NEDC-33004P-A, was the assumed decay heat level of ANS 5.1-1979 with a two-sigma uncertainty. The assumed decay heat level for the EPU analysis bounds ANS 5.1-1979 + two sigma. Thus, the key analytical assumptions are the same or conservative relative to the current licensing basis. This LOFW analysis is performed to demonstrate acceptable RCIC system performance. The design basis criterion for the RCIC system is confirmed by demonstrating that it is capable of maintaining the water level inside the shroud above the top of active fuel during the LOFW transient. The minimum level (see Figure 2.8-23) is maintained at least 153 inches above the top of active fuel, thereby demonstrating acceptable RCIC system performance. There are no applicable equipment out of service assumptions for this transient.

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[illegible]	[illegible]	[illegible]
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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 NMP2 EPU. [[

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Conclusion

NMPNS has reviewed the analyses of the Loss of Normal Feedwater Flow event and concludes that analyses have adequately accounted for operation of NMP2 at the proposed power level and were performed using acceptable analytical models. NMPNS further concludes that the reactor protection and safety systems will continue to ensure that the SAFDLs and the RCPB pressure limits will not be exceeded as a result of the loss of normal feedwater flow. Based on this, NMPNS concludes that NMP2 will continue to meet the requirements of GDCs 10, 15, and 26 following implementation of the proposed EPU. Therefore, NMPNS finds the proposed EPU acceptable with respect to the Loss of Normal Feedwater Flow event.

2.8.5.3 Decrease in Reactor Coolant System Flow

2.8.5.3.1 Loss of Forced Reactor Coolant Flow

Regulatory Evaluation

A decrease in reactor coolant flow occurring while the plant is at power could result in a degradation of core heat transfer. An increase in fuel temperature and accompanying fuel damage could then result if SAFDLs are exceeded during the transient. Reactor protection and safety systems are actuated to mitigate the transient. The NMPNS review consisted of confirmation of the applicability of the [[ ]] in the NRC approved NEDC-33004P-A, "Constant Pressure Power Uprate" for the specific NMP2 application. The NRC's acceptance criteria are based on (1) GDC10, 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) GDC15, 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) GDC26, 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.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of Constant Pressure Power Uprates. Section 9 of the CLTR addresses the effect of Constant Pressure Power Uprate on Loss of Forced Reactor Coolant Flow. The results of this evaluation are described below.

Section 9 of the CLTR and the response to CLTR NRC RAI Set 9 Number 14 RXSB 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 [[

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Conclusion

NMPNS reviewed the CLTR [[ ]] for the Loss of Forced Reactor Coolant Flow event and concludes that it applies to NMP2. Applicability of this [[ ]] to NMP2 demonstrates that the reactor protection and safety systems will continue to ensure that the SAFDLs and the RCPB pressure limits will not be exceeded as a result of this event. Based on this, NMP2 will continue to meet the requirements of GDCs 10, 15, and 26 following implementation of the proposed EPU. Therefore, NMPNS finds the proposed EPU acceptable with respect to the 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 that 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 NMPNS review consisted of confirmation of the applicability of the [[ ]] in the NRC approved NEDC-33004P-A, "Constant Pressure Power Uprate" for the specific NMP2 application. The NRC's acceptance criteria are based on (1) GDC27, 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) GDC28, 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) GDC31, insofar as it requires that the RCPB be designed with margin sufficient to assure that, under specified conditions, it will behave in a nonbrittle manner and the probability of a rapidly propagating fracture is minimized.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of Constant Pressure Power Uprates. Section 9 of the CLTR addresses the effect of Constant Pressure Power Uprate on Reactor Recirculation Pump Rotor Seizure and Reactor Recirculation Pump Shaft Break. The results of this evaluation are described below.

Section 9 of the CLTR and the response to CLTR NRC RAI Set 9 Number 14 RXSB 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, [[

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Conclusion

NMPNS reviewed the CLTR [[ ]] for the Reactor Recirculation Pump Rotor Seizure and Reactor Recirculation Pump Shaft Break events and concludes that it applies to NMP2. Applicability of this [[ ]] to NMP2 demonstrates that the reactor protection and safety systems will continue to ensure that the SAFDLs and the RCPB pressure limits will not be exceeded as a result of this event. Based on this, NMPNS concludes that NMP2 will continue to meet the requirements of GDCs 27, 28, and 31 following implementation of the proposed EPU. Therefore, NMPNS finds the proposed EPU acceptable with respect to the Reactor Recirculation Pump Rotor Seizure and Reactor Recirculation Pump Shaft Break events.

2.8.5.4 Reactivity and Power Distribution Anomalies

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

Regulatory Evaluation

An uncontrolled control rod assembly withdrawal from subcritical or low power startup conditions may be caused by a malfunction of the reactor control or rod control systems. This withdrawal will uncontrollably add positive reactivity to the reactor core, resulting in a power excursion. The NMPNS review covered (1) the description of the causes of the transient and the transient itself, (2) the initial conditions, (3) the values of reactor parameters used in the analysis, (4) the analytical methods and computer codes used, and (5) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC10, 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) GDC20, 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) GDC25, 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.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of Constant Pressure Power Uprates. Section 5.1.2 of the CLTR addresses the effect of Constant Pressure Power Uprate on Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition. The results of this evaluation are described below.

The evaluation of the Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition event for NMP2 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 Rod Withdrawal Error (RWE) analysis for NMP2 is based on Reference 74. The NMP2 EPU core consists only of GE fuel assemblies and the EPU is limited to 120% of OLTP. There is also no change to the NMP2 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

NMPNS has reviewed the analyses of the uncontrolled control rod assembly withdrawal from a subcritical or low power startup condition and concludes that analyses have adequately accounted for the changes in core design necessary for operation of NMP2 at the proposed power level. NMPNS also concludes that the analyses were performed using acceptable analytical models. NMPNS further concludes that it has demonstrated that the reactor protection and safety systems will continue to ensure the SAFDLs are not exceeded. Based on this, NMPNS concludes that NMP2 will continue to meet the requirements of GDCs 10, 20, and 25 following implementation of the proposed EPU. Therefore, NMPNS finds the proposed EPU acceptable with respect to the uncontrolled control rod assembly withdrawal from a subcritical or low power startup condition.

2.8.5.4.2 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 NMPNS review consisted of confirmation of the applicability of the [[ ]] in the NRC approved NEDC-33004P-A, "Constant Pressure Power Uprate" for the specific NMP2 application. The NRC's acceptance criteria are based on (1) GDC10, 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) GDC20, 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) GDC25, 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.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of Constant Pressure Power Uprates. Section 9 of the CLTR addresses the effect of Constant Pressure Power Uprate on Uncontrolled Control Rod Assembly Withdrawal at Power. The results of this evaluation are described below.

Section 9 of the CLTR and the response to CLTR NRC RAI Set 9 Number 14 RXSB 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 NMP2 reload evaluation scope. [[

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Conclusion

NMPNS reviewed the CLTR [[ ]] for the Uncontrolled Control Rod Assembly Withdrawal at Power event and concludes that it applies to NMP2. Applicability of this [[ ]] to NMP2 demonstrates that the reactor protection and safety systems will continue to ensure that the SAFDLs and the RCPB pressure limits will not be exceeded as a result of this event. Based on this, NMPNS concludes that NMP2 will continue to meet the requirements of GDCs 10, 20, and 25 following implementation of the proposed EPU. Therefore, NMPNS finds the proposed EPU acceptable with respect to the Uncontrolled Control Rod Assembly Withdrawal at Power 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 NMPNS review consisted of confirmation of the applicability of the [[ ]] in the NRC approved NEDC-33004P-A, "Constant Pressure Power Uprate" for the specific NMP2 application. The NRC's acceptance criteria are based on (1) GDC10, 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) GDC20, 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) GDC15, 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) GDC28, 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) GDC26, 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..

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of Constant Pressure Power Uprates. Section 9 of the CLTR addresses the effect of the Startup of an Idle Recirculation Pump and the Failure of the Recirculation Flow Controller events. The results of this evaluation are described below.

Section 9 of the CLTR and the response to CLTR NRC RAI Set 9 Number 14 RXSB 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 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 [[

]]

The Startup of an Idle Recirculation Pump event is [[ ]] ARTS was approved for NMP2 by Reference 61.

Conclusion

NMPNS reviewed the CLTR [[ ]] for the Startup of an Idle Recirculation Pump and the Failure of the Recirculation Flow Controller events and concludes that it applies to NMP2. Applicability of this [[ ]] to NMP2 demonstrates that the reactor protection and safety systems will continue to ensure that the SAFDLs and the RCPB pressure limits will not be exceeded as a result of this event. Based on this, NMPNS concludes that NMP2 will continue to meet the requirements of GDCs 10, 15, 20, 26, and 28 following implementation of the proposed EPU. Therefore, NMPNS finds the proposed EPU acceptable with respect to the Startup of an Idle Recirculation Pump and the Failure of the Recirculation Flow Controller events.

2.8.5.4.4 Spectrum of Rod Drop Accidents

Regulatory Evaluation

NMPNS evaluated the consequences of a control rod drop accident in the area of reactor physics. The NMPNS review covered the occurrences that lead to the accident, safety features designed to limit the amount of reactivity available and the rate at which reactivity can be added to the core, the analytical model used for analyses, and the results of the analyses. The NRC's acceptance criteria are based on GDC28, 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.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of Constant Pressure Power Uprates. Section 5.1.2 and 5.3.4 of the CLTR addresses the effect of Constant Pressure Power Uprate on Rod Worth Minimizer. The results of this evaluation are described below.

The spectrum of CRDAs does not change with EPU. The evaluation of a CRDA for the NMP2 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 NMP2 is based on Reference 75. The NMP2 EPU core consists only of GE fuel assemblies and the EPU is limited to 120% of OLTP. Control Rod Sequencing at NMP2 for CLTP and EPU follows the BPWS. There is also no change to the NMP2 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. 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

NMPNS has reviewed the analyses of the rod drop accident and concludes that analyses have adequately accounted for operation of NMP2 at the proposed power level and were performed using acceptable analytical models. NMPNS further concludes that it has demonstrated that appropriate reactor protection and safety systems will prevent postulated reactivity accidents that could (1) result in damage to the RCPB greater than limited local yielding, or (2) cause sufficient damage that would significantly impair the capability to cool the core. Based on this, NMPNS concludes that NMP2 will continue to meet the requirements of GDC28 following implementation of EPU. Therefore, NMPNS finds the proposed EPU acceptable with respect to the rod drop accident.

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 NMPNS review consisted of confirmation of the applicability of the [[ ]] in the NRC approved NEDC-33004P-A, "Constant Pressure Power Uprate" for the specific NMP2 application. The NRC's acceptance criteria are based on (1) GDC10, 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) GDC15, 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) GDC26, 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.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of Constant Pressure Power Uprates. Section 9 of the CLTR addresses the effect of Constant Pressure Power Uprate on Inadvertent Operation of ECCS or Malfunction that Increases Reactor Coolant Inventory. The results of this evaluation are described below.

Section 9 of the CLTR and the response to CLTR NRC RAI Set 9 Number 14 RXSB 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 NMP2 reload evaluation scope. [[

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Conclusion

NMPNS reviewed the CLTR [[ ]] for the Inadvertent Operation of ECCS or Malfunction that Increases Reactor Coolant Inventory category event and concludes that it applies to NMP2. Applicability of this [[ ]] to NMP2 demonstrates that the reactor protection and safety systems will continue to ensure that the SAFDLs and the RCPB pressure limits will not be exceeded as a result of this event. Based on this, NMPNS concludes that NMP2 will continue to meet the requirements of GDCs 10, 15, and 26 following implementation of the proposed EPU. Therefore, NMPNS finds the proposed EPU acceptable with respect to the Inadvertent Operation of ECCS or Malfunction that Increases Reactor Coolant Inventory.

2.8.5.6 Decrease in Reactor Coolant Inventory

2.8.5.6.1 Inadvertent Opening of a Pressure Relief Valve

Regulatory Evaluation

The inadvertent opening of a pressure relief valve results in a reactor coolant inventory decrease and a decrease in RCS pressure. The pressure relief valve discharges into the suppression pool. Normally there is no reactor trip. The pressure regulator senses the RCS pressure decrease and partially closes the turbine control valves (TCVs) to stabilize the reactor at a lower pressure. The reactor power settles out at nearly the initial power level. The coolant inventory is maintained by the feedwater control system using water from the condensate storage tank via the condenser hotwell. The NMPNS review consisted of confirmation of the applicability of the [[ ]] in the NRC approved NEDC-33004P-A, "Constant Pressure Power Uprate" for the specific NMP2 application. The NRC's acceptance criteria are based on (1) GDC10, 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) GDC15, 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) GDC26, 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.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of Constant Pressure Power Uprates. Section 3.1 and Section 9 of the CLTR address the effect of Constant Pressure Power Uprate on an Inadvertent Opening of a Pressure Relief Valve. The results of this evaluation are described below.

Section 9 and Section 3.1 of the CLTR and the response to CLTR NRC RAI Set 9 Number 14 RXSB 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 [[

]]

Conclusion

NMPNS reviewed the CLTR [[ ]] for the Inadvertent Opening of a Pressure Relief Valve event and concludes that it applies to NMP2. Applicability of this [[ ]] to NMP2 demonstrates that the reactor protection and safety systems will continue to ensure that the SAFDLs and the RCPB pressure limits will not be exceeded as a result of this event. Based on this, NMP2 will continue to meet the requirements of GDCs 10, 15, and 26 following implementation of the proposed EPU. Therefore, NMPNS finds the proposed EPU acceptable with respect to 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 NMPNS review covered (1) NMPNS's determination of break locations and break sizes; (2) postulated initial conditions; (3) the sequence of events; (4) the analytical model used for analyses, and calculations of the reactor power, pressure, flow, and temperature transients; (5) calculations of peak cladding temperature, total oxidation of the cladding, total hydrogen generation, changes in core geometry, and long-term cooling; (6) functional and operational characteristics of the reactor protection and ECCS systems; and (7) operator actions. The NRC's acceptance criteria are based on (1) 10 CFR 50.46, insofar as it establishes standards for the calculation of ECCS performance and acceptance criteria for that calculated performance; (2) 10 CFR 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) GDC4, 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) GDC27, 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) GDC35, 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.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of Constant Pressure Power Uprates. Sections 4.2.2, 4.2.3, 4.2.4, 4.2.5, and 4.3 of the CLTR address the effect of Constant Pressure Power Uprate on the Emergency Core Cooling System and Loss-of-Coolant Accidents. The results of this evaluation are described below.

The NMP2 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.

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 Core Spray

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The High Pressure Core Spray (HPCS) system is designed to spray water into the reactor vessel over a wide range of operating pressures and was evaluated in Section 4.3 of ELTR2. The HPCS System provides reactor vessel coolant inventory makeup in the event of a small break LOCA that does not immediately depressurize the reactor vessel and helps to depressurize the reactor vessel. This system also provides spray cooling for long-term core cooling after a LOCA.

The HPCS System also serves as a backup to the RCIC System to provide makeup water in the event of an LOFW flow transient, as described in Section 2.8.5.2.3. Because the HPCS injection results in RPV depressurization, and there is no change to the range of pressures over which HPCS is required for injection, the HPCS System adequately meets the safety requirement following an LOFW event.

There is no change to the maximum specified reactor pressure for HPCS System operation and no change in the HPCS System performance parameters. The maximum injection pressure for the HPCS System is conservatively based on the upper AL for the lowest available group of SRVs. Because the SRV settings remain the same for EPU, the HPCS System operating conditions and operating functions also remain the same. Therefore, there is no change in the original design pressures or temperatures for the system components. EPU does not change the power required by the pump or the power required by the HPCS diesel generator unit.

Because the maximum normal operating pressure and the SRV setpoints do not change for EPU, the HPCS System performance requirements do not change. Therefore, the HPCS System at NMP2 is confirmed to be consistent with the generic description provided in the CLTR.

Low Pressure Core Spray

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The Low Pressure Core Spray (LPCS) system is automatically initiated in the event of a LOCA. When operating in conjunction with other ECCS, the LPCS system is required to provide adequate core cooling for all LOCA events. There is no change in the reactor pressures at which the LPCS is required.

The LPCS system sprays water into the reactor vessel after it is depressurized. The primary purpose of the LPCS 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 LPCS 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 LPCS system. The generic core spray distribution assessment provided in ELTR2, Section 3.3, continues to be valid for EPU.

The LPCS system at NMP2 is confirmed to be consistent with the generic description provided in the CLTR because the system functions are not changed and the core cooling capacity is adequate.

Low Pressure Coolant Injection

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[illegible]	[illegible]	[illegible]
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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.

The LPCI mode at NMP2 is confirmed to be consistent with the generic description provided in the CLTR because there is no change in the reactor pressures at which the LPCI mode of the RHR System is required and the core cooling capacity is adequate.

Automatic Depressurization System

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[illegible]	[illegible]	[illegible]
		]]

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 LPCS and LPCI to inject coolant into the reactor vessel. 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. The ADS at NMP2 is confirmed to be consistent with the generic description provided in the CLTR because the SRV setpoints and functions remain the same, the ADS timers are not changed and the small break LOCA event mitigation is acceptable.

Emergency Core Cooling System Performance

The NMP2 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 rated thermal power conditions.

[[

[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 recirculation suction line break and the worst small break under the HPCS-Diesel Generator failure scenario. Consistent with Limitation and Condition 9.7 of IMLTR SE (Reference 6), [[

]] 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 large break PCT, and therefore large break LOCA has 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 large break 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 [[ ]] of the local oxidation and core-wide metal-water reaction are confirmed by the change in NMP2 Licensing Basis PCT due to EPU resulting in a Licensing Basis PCT which is sufficiently below the 10 CFR 50.46 limit of 2200°F. Coolable geometry and long-term cooling have been dispositioned generically for BWRs. These [[ ]]s are not affected by EPU.

In addition to the large break LOCA analysis, the small break LOCA response was analyzed. The increased decay heat associated with EPU results in a longer ADS blowdown and a higher PCT for the small break LOCA. Previous analysis (Reference 76) demonstrates that NMP2 is a small break Appendix K PCT limited plant. The effect of EPU on the calculated small break PCT is acceptable as long as the impact of the results on the Licensing Basis PCT remains below the 10 CFR 50.46 limits. The current Technical Specification values for ECCS initiation were used for the analysis; no changes to these values were required for EPU. Plant-specific analyses demonstrate that there is sufficient ADS capacity, with six ADS valves in service and one out-of-service, at EPU conditions, to remain below these limits. Key input parameters to the SAFER/GESTR LOCA evaluation model are provided in Table 2.8-3. Input parameters are selected as nominal or representative values. For Appendix K calculations, select inputs are chosen so as to set a bounding condition or to assure conservatism.

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. At EPU power condition, the MELLLA core flow is approximately 99% of rated core flow. Therefore, the EPU analysis results at rated power and flow are applied to the MELLLA condition. Also, the effect of increased core flow (ICF) on PCT is negligible with EPU. Thus the SLO, MELLLA, and ICF domain remain valid with 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 EPU Licensing Basis PCT for GE14 fuel is less than 1540°F, which represents a 60°F increase from the pre-EPU Licensing Basis PCT of less than 1480°F evaluated at pre-EPU power and rated core flow. Restrictions imposed by the NRC on Upper Bound PCT have been removed for NMP2. The Upper Bound PCT has been shown bounded by the Licensing Basis PCT, consistent with the previous evaluation (Reference 77), and need not be recalculated for EPU implementation. The results of these analyses are provided in Table 2.8-4.

Conclusion

NMPNS concludes that analyses have adequately accounted for operation of NMP2 at the proposed power level and that the analyses were performed using acceptable analytical models. NMPNS further concludes that it has demonstrated that the reactor protection system and the ECCS will continue to ensure that the peak cladding temperature, total oxidation of the cladding, total hydrogen generation, and changes in core geometry, and long-term cooling will remain within acceptable limits. Based on this, NMPNS concludes that NMP2 will continue to meet the requirements of GDCs 4, 27, 35, 10 CFR 50.46, and 10 CFR 50 Appendix K following implementation of the proposed EPU. Therefore, NMPNS finds the proposed EPU acceptable with respect to the LOCA.

2.8.5.7 Anticipated Transients Without Scram

Regulatory Evaluation

ATWS is defined as an AOO followed by the failure of the reactor portion of the protection system specified in GDC20. The regulation at 10 CFR 50.62 requires that:

- Each BWR has 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 has 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 SLCS initiation must be automatic and must be designed to perform its function in a reliable manner for plants granted a construction permit after July 26, 1984, and for plants granted a construction permit prior to July 26, 1984, that have already been designed and built to include this feature.
- Each BWR has equipment to trip the reactor coolant recirculation pumps automatically under conditions indicative of an ATWS.

The NMPNS review was conducted to ensure that (1) the above requirements are met, (2) sufficient margin is available in the setpoint for the SLCS pump discharge relief valve such that

SLCS operability is not affected by the proposed EPU, and (3) operator actions specified in NMP2's Emergency Operating Procedures are consistent with the generic emergency procedure guidelines/severe accident guidelines (EPGs/SAGs), insofar as they apply to NMP2 design. In addition, NMPNS reviewed the ATWS analysis to ensure that (1) the peak vessel bottom pressure is less than the ASME Service Level C limit of 1500 psig; (2) the peak clad temperature is within the 10 CFR 50.46 limit of 2200 °F; (3) the peak suppression pool temperature is less than the design limit; and (4) the peak containment pressure is less than the containment design pressure. NMPNS also evaluated the potential for thermal-hydraulic instability in conjunction with ATWS events using the methods and criteria approved by the NRC staff. For this analysis, NMPNS reviewed the limiting event determination, the sequence of events, the analytical model and its applicability, the values of parameters used in the analytical model, and the results of the analyses.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of Constant Pressure Power Uprates. Section 9.3.1 of the CLTR addresses the effect of Constant Pressure Power Uprate on ATWS. The results of this evaluation are described below.

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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 LOOP cases.

For NMP2, the LOOP does not result in a reduction in the RHR pool cooling capability relative to the MSIVC and PRFO cases. With the same RHR pool cooling capability, the containment response for the MSIVC and PRFO cases bound the LOOP case. Therefore, the ATWS event selection for NMP2 is confirmed to be consistent with the generic description provided in the CLTR.

NMP2 meets the ATWS mitigation requirements defined in 10 CFR 50.62:

- Installation of an ARI system;
- Boron injection equivalent to 86 gpm; and
- Installation of automatic Recirculation Pump Trip (RPT) logic (i.e., ATWS-RPT).

The 86 gpm boron injection equivalency requirement of 10 CFR 50.62 is satisfied via the following relationship:

$$(Q/86) \times (M251/M) \times (C/13) \times (E/19.8) > 1$$

where:

Q = Expected standby liquid control system (SLCS) flow rate (gpm)

M251/M = Mass of water in a 251-inch diameter reactor vessel (lbs) / mass of water in the reactor vessel and recirculation system at hot rated condition (lbs)

C = Sodium pentaborate solution concentration (weight percent)

E = Boron-10 isotope enrichment (19.8% of natural boron)

For NMP2,

Q = 82.4 gpm

M251/M = 1 (since NMP2 has a 251-inch diameter reactor vessel)

C = 13.6 wt %

E = 25.0 %

Therefore, the 86 gpm equivalency requirement is satisfied as follows:

$$(Q/86) \times (M251/M) \times (C/13) \times (E/19.8) > 1$$

$$(82.4/86) \times (1) \times (13.6/13) \times (25.0/19.8) = 1.266 > 1$$

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 190°F; and
- Peak containment pressure less than 45 psig.

The limiting events for the acceptance criteria discussed above are the PRFO event and the MSIVC event.

The MSIVC, PRFO, and LOOP sequence of events are given in Tables 2.8-7, 2.8-8, and 2.8-9, respectively.

A plant specific ATWS analysis was performed for CLTP and for EPU RTP to demonstrate the effect of EPU on the ATWS acceptance criteria. There are no changes to the assumed operator actions for the EPU ATWS analysis. The key inputs to the ATWS analysis are provided in Table 2.8-5. The results of the analysis are provided in Table 2.8-6.

The EPU ATWS analysis is performed using the NRC approved code ODYN (see Table 1-1).

The results of the ATWS analysis meet the above ATWS acceptance criteria. Therefore, the NMP2 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 1600°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. Therefore, the PCT and local cladding oxidation criteria are generically addressed to demonstrate compliance with the ATWS Rule for EPU.

As discussed in Section 2.8.4.5, Standby Liquid Control System, sufficient margin is available in the setpoint for the SLCS pump discharge relief value. The ATWS PCT for NMP2 is confirmed to be consistent with the generic description provided in the CLTR.

The potential for thermal-hydraulic instability in conjunction with ATWS events is evaluated in Section 2.8.3.3.

Conclusion

NMPNS has reviewed the information related to ATWS and concludes that it has adequately accounted for the effects of the proposed EPU on ATWS. NMPNS concludes that it has demonstrated that ARI, SLCS, and recirculation pump trip systems have been installed and that they will continue to meet the requirements of 10 CFR 50.62 and the analysis acceptance criteria following implementation of the proposed EPU. Therefore, NMPNS finds the proposed EPU acceptable with respect to ATWS.

2.8.5.8 Fuel Thermal Margin Events

The 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 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. Limitations and conditions 9.4 and 9.19 of Reference 6 that require a 0.02 SLMCPR adder and the 0.01 OLMCPR adder are applicable to NMP2 EPU and will be applied to EPU core designs through the RLA process.

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, which includes the Reference 6 adders. Therefore, plant-specific fuel thermal margin event evaluation is not required for EPU.

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The thermal margin event analysis at NMP2 is confirmed to be consistent with the generic description provided in the CLTR because it is evaluated for the EPU reload core prior to EPU implementation.

Consistent with Limitation and Condition 9.9 and 9.10 of Reference 6, 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 21.2%. The minimum calculated margin to the cladding strain criterion was 20.0%. 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 consistent with Limitation and Condition 9.10 of Reference 6.

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 NMP2 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. Therefore, there is no change in these power-dependent limits due to steady-state power uprate.

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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The power and flow dependent limits at NMP2 are confirmed to be consistent with the generic description provided in the CLTR because they are evaluated for the EPU reload core prior to EPU implementation.

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 NMPNS review covered the ability of the storage facilities to maintain the new fuel in a subcritical array during all credible storage conditions. The review focused on the effect of changes in fuel design on the analyses for the new fuel storage facilities. The NRC's acceptance criteria are based on GDC62, insofar as it requires the prevention of criticality in fuel storage systems by physical systems or processes, preferably utilizing geometrically safe configurations.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of Constant Pressure Power Uprates. Section 6.3.4 of the CLTR addresses the effect of Constant Pressure Power Uprate on Fuel Racks. The results of this evaluation are described below.

The additional energy requirements for EPU are met by an increase in bundle enrichment, an increase in the fuel reload batch size, and/or changes in the fuel loading pattern. With the exception of the increased enrichment, there are no changes in the fuel design associated with EPU.

The increased enrichment will be balanced by changes in the gadolinia loading to ensure that fuel reactivity remains within existing limits because plant shutdown and reactivity margins must continue to meet existing NRC approved limits. The increased reload batch size can be accommodated by the existing new and spent fuel storage facilities, so that no changes are required for the new fuel storage facility. The fuel reload pattern has no effect on the new fuel storage requirements.

GE14 new fuel assemblies were analyzed in GEH Low-Density Fuel Storage (LDFS) racks. New fuel bundle geometries were assumed with uniform lattice enrichments of 5.0w% ^{235}U using both full and part length fuel rods and including representative placement and numbers of Gadolinia rods.

The new fuel criticality analyses include conservative assumptions relative to enrichment, in order to maximize reactivity (Table 2.8-10), while maintaining the fuel storage criticality safety limits in the racks. Operation at EPU will not change these assumptions.

Therefore, the new fuel storage requirements are bounded by the current licensing basis, and no changes are required to ensure that new fuel can be maintained in a subcritical array during all credible storage conditions.

Conclusion

NMPNS has reviewed the analyses related to the effect of the new fuel on the analyses for the new fuel storage facilities and concludes that the new fuel storage facilities will continue to meet the requirements of GDC62 following implementation of the proposed EPU. Therefore, NMPNS finds the proposed EPU 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 or dry fuel storage casks. The NMPNS review covered the effect of the proposed EPU on the criticality analysis. The NRC's acceptance criteria are based on (1) GDC4, 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) GDC62, insofar as it requires that criticality in the fuel storage systems be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of Constant Pressure Power Uprates. Section 6.3.4 of the CLTR addresses the effect of

Constant Pressure Power Uprate on Fuel Racks. The results of this evaluation are described below.

The spent fuel pool continues to rely on a supplemental neutron absorber (poison) to maintain subcriticality. NMP2 has Boral fuel storage racks which uses the original poison.

As discussed in Section 2.8.6.1, the increased enrichment that will support EPU will be balanced by changes in the gadolinia loading to ensure that fuel reactivity remains within existing limits because plant shutdown and reactivity margins must continue to meet existing NRC approved limits. Therefore, EPU has no effect on the criticality analyses, and the current analyses bound the EPU fuel that will be stored in the spent fuel pool.

As discussed in Section 2.5.3.1, EPU will increase the heat load on the fuel pool cooling system during and after refueling outages because of the increase in decay heat. Although there is an increase in the fuel pool heat load due to higher decay heat, the pool temperature continues to remain below the design temperature for all offload scenarios. The temperature will also remain below the operating limit for normal operating conditions.

The spent fuel storage racks at NMP2 were analyzed with the GE14 fuel design. All storage rack conditions were analyzed assuming the GE14 fuel lattice geometry with uniform enrichment distributions, explicit burnable absorbers and fission products for the spent fuel racks.

The spent fuel storage criticality analyses include conservative assumptions relative to enrichment, exposure, and void history in order to maximize reactivity (Table 2.8-11), while maintaining the fuel storage criticality safety limits in the racks. Operation at EPU will not change these assumptions. The maximum reactivity summary for the Boral spent fuel storage is provided in Table 2.8-12.

In addition, there is no impact on the spent fuel storage racks from the increased EPU heat load.

Conclusion

NMPNS has reviewed the analyses related to the effects of the proposed EPU on the spent fuel storage capability and concludes that it has adequately accounted for the effects of the proposed EPU on the spent fuel rack temperature and criticality analyses. NMPNS also 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, NMPNS concludes that the spent fuel storage facilities will continue to meet the requirements of GDCs 4 and 62 following implementation of the proposed EPU. Therefore, NMPNS finds the proposed EPU acceptable with respect to spent fuel storage.

Table 2.8-1 Option III Setpoint Demonstration

OPRM Amplitude Setpoint	OLMCPR(SS)	OLMCPR(2PT)
1.04	1.181	1.016
1.05	1.200	1.032
1.06	1.220	1.049
1.07	1.240	1.067
1.08	1.261	1.085
1.09	1.283	1.104
1.10	1.304	1.122
1.11	1.326	1.141
1.12	1.349	1.161
1.13	1.373	1.181
1.14	1.398	1.202
Acceptance Criteria	Off-rated OLMCPR at 45% Flow Estimated to be 1.60	Rated Power OLMCPR Estimated to be 1.39

Table 2.8-2 ODYSY Decay Ratios at BSP Region Boundary Endpoints

Point*	Power (%)	Flow (%)	Core Decay Ratio	Channel Decay Ratio
Controlled Entry (Region II), NCL Runs				
B2	33.36	29.30	0.782	0.374
B2-ICA	28.68	28.90	<0.782	
Scram (Region I), NCL Runs				
B1	46.07	29.50	0.793	0.395
B1-ICA	41.40	29.50	<0.793	
Controlled Entry (Region II), HFCL Runs				
A2	55.13	38.25	0.795	0.383
A2-ICA	64.51	50.00	<0.795	
Scram (Region I), HFCL Runs				
A1	46.63	27.93	0.795	0.461
A1-ICA	56.61	40.00	<0.795	

* 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 2.8-3 Key Input Parameters for SAFER/GESTR LOCA Evaluation

Item	Parameter**	Unit	Nominal Value	Appendix K Value
1	Current Licensed Thermal Power (CLTP)	MWt	3467	3536
2	Extended Power Uprate (EPU)	MWt	3988	4068
3	Vessel Steam Dome Pressure	psia	1055*	1055
4	Rated Core Flow	Mlb/hr	108.5	108.5
5	Maximum Recirculation Suction Line Break Area	ft^2	3.131	3.131
6	GE14 Number of Fuel Rods per Bundle	NA	92	92
7	GE14 PLHGR (Nominal/App. K)	KW/ft	12.8	13.4
8	GE14 Worst Pellet Exposure for ECCS Evaluation	MWd/MTU	16000	16000
9	Single Failure Input	NA	HPCSDG	HPCSDG
10	Limiting Large / Small Break Location	NA	Recirculation Suction Line	Recirculation Suction Line

* Nominal analysis was performed with the conservative 10 CFR 50 Appendix K vessel steam dome pressure of 1055 psia.

** Main Steam and Feedwater flow rates for the 100% EPU power/100% flow case with 1055 psia dome pressure are 17.657 and 17.604 Mlb/hr respectively as determined by ISCOR.

Table 2.8-4 ECCS Conformance Results

Parameter	CLTP	EPU	10 CFR 50.46 Limit
Method	SAFER/GESTR	SAFER/GESTR	
Thermal Power (MWt)	3467	3988	
Licensing Basis PCT (°F)	< 1480	< 1540	< 2200
Cladding Oxidation (% Original Clad Thickness)	< 0.2	< 0.3	< 17
Hydrogen Generation, Core-Wide Metal-Water Reaction (%)	< 0.1	< 0.1	< 1.0
Coolable Geometry	Acceptable	Acceptable	PCT < 2200 °F, and Local Oxidation <17%
Core Long Term Cooling	Acceptable	Acceptable	Core flooded to TAF OR Core flooded to jet pump suction elevation and at least one core spray system is operating at rated flow.
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Table 2.8-5 NMP2 Key Inputs for ATWS Analysis

Input Variable	CLTP	EPU
Reactor power (MWt)	3467	3988
Reactor dome pressure (psia)	1035	1035
Each SRV capacity at 1145 psig (Mlbm/hr)	0.890	0.890
High pressure ATWS-RPT (psig)	1095	1095
Number of SRVs Out-of-service (OOS)	2	2

Table 2.8-6 NMP2 Results for ATWS Analysis

Acceptance Criteria	CLTP [1,2]	EPU [1]
Peak vessel bottom pressure (psig)	1290	1372
Peak suppression pool temperature (°F)	155	163
Peak containment pressure (psig)	< 6.0	7.0

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.

Table 2.8-7 MSIVC Sequence of Events

Item	Event	CLTP BOC Event Time (sec)	EPU BOC Event Time (sec)	CLTP EOC Event Time (sec)	EPU EOC Event Time (sec)
1	MSIV Isolation Initiated	0.0	0.0	0.0	0.0
2	MSIVs Fully Closed	4.0	4.0	4.0	4.0
3	High Pressure ATWS Setpoint	4.0	3.9	4.0	3.9
4	Peak Neutron Flux	4.1	4.0	4.0	4.0
5	Recirculation Pumps Trip	4.5	4.4	4.5	4.4
6	Opening of the First Relief Valve	4.8	4.6	4.7	4.6
7	Peak Heat Flux	5.2	5.0	5.2	5.0
8	Peak Vessel Pressure	6.3	8.5	5.9	7.8
9	Feedwater Reduction Initiated	38.0	38.0	38.0	38.0
10	BIIT Reached	68.0	59.0	69.0	60.0
11	SLCS Pumps Start	124	124	124	124
12	Hot Shutdown Achieved (Neutron Flux Remains < 0.1%)	485	476	530	513
13	RHR Cooling Initiated	1080	1080	1080	1080
14	Peak Suppression Pool Temperature	14561	16486	14221	16278

Table 2.8-8 PRFO Sequence of Events

Item	Event	CLTP BOC Event Time (sec)	EPU BOC Event Time (sec)	CLTP EOC Event Time (sec)	EPU EOC Event Time (sec)
1	TCV and Bypass Valves Start Open	0.1	0.2	0.1	0.2
2	MSIV Closure Initiated by Low Steamline Pressure	18.1	15.2	17.1	14.2
3	MSIVs Fully Closed	22.1	19.2	21.1	18.2
4	Peak Neutron Flux	23.5	20.4	22.9	19.7
5	High Pressure ATWS Setpoint	25.9	22.5	25.2	21.5
6	Recirculation Pumps Trip	26.5	23.0	25.7	22.0
7	Opening of the First Relief Valve	26.7	23.2	25.9	22.1
8	Peak Heat Flux	26.9	23.6	25.9	22.3
9	Peak Vessel Pressure	27.8	26.6	27.1	25.5
10	Feedwater Reduction Initiated	59.5	56.0	59.5	56.0
11	BIIT Reached	89.0	82.0	90.0	81.0
12	SLCS Pumps Start	146	143	145	141
13	Hot Shutdown Achieved (Neutron Flux Remains < 0.1%)	509	490	547	520
14	RHR Cooling Initiated	1080	1080	1080	1080
15	Peak Suppression Pool Temperature	14469	16684	14456	16481

Table 2.8-9 LOOP Sequence of Events

Item	Event	CLTP BOC Event Time (sec)	EPU BOC Event Time (sec)	CLTP EOC Event Time (sec)	EPU EOC Event Time (sec)
1	Loss of Auxiliary Power Transformers	0.0	0.0	0.0	0.0
2	Recirculation Pumps Trip	0.0	0.0	0.0	0.0
3	Feedwater/Condensate Pumps Trip	0.0	0.0	0.0	0.0
4	Peak Heat Flux	0.0	0.1	0.0	0.1
5	Turbine Trip due to Loss of Main Condenser Vacuum	8.0	8.0	8.0	8.0
6	Turbine Bypass Valves Start Open due to Turbine Trip	8.1	8.2	8.1	8.2
7	Peak Neutron Flux	8.7	8.7	8.8	8.8
8	High Pressure ATWS Setpoint	10.3	10.2	10.3	10.2
9	Opening of the First Relief Valve (SRVs Operate in Safety Mode after First Relief Valve Cycle)	11.8	11.6	11.8	11.5
10	MSIV Isolation Initiated	28.0	28.0	28.0	28.0
11	Turbine Bypass Valves (if open) Tripped to Close	28.0	28.0	28.0	28.0
12	MSIVs Fully Closed	32.0	32.0	32.0	32.0
13	Peak Vessel Pressure	55.1	85.0	52.7	80.8
14	SLCS Pumps Start	130	130	130	130
15	Hot Shutdown Achieved (Neutron Flux Remains < 0.1%)	527	550	573	594

Table 2.8-10 New Fuel Storage Rack Design Parameters

Parameter	CLTP	EPU
K_{eff} (Subcritical multiplication factor)	< 0.90 (For normal conditions)	No change
K_{eff} (Subcritical multiplication factor)	< 0.95 (For abnormal conditions)	No change
U^{235} (Lattice average enrichment)	5.0w%	No change
k_{∞} (In-core peak eigenvalue limit)	1.3392	No change

Table 2.8-11 Spent Fuel Storage Rack Design Parameters

Parameter	CLTP	EPU
K_{eff} (Subcritical multiplication factor)	< 0.95 (For normal & abnormal conditions)	No change
U^{235} (Lattice average enrichment)	4.90w%	No change
Analysis performed at Void History	0%, 40% and 70% (0% voids results highest reactivity)	No change
Exposure	10 GWd/ST at 0% Voids	No change
k_∞ (In-core peak eigenvalue limit)	1.3392	No Change

Table 2.8-12 NMP2 Spent Fuel Storage Maximum Reactivity Summary

Parameter	Boral Spent Fuel Storage
K_{nom} (Reference rack eigenvalue)	0.9273
Monte Carlo calculational bias	0.0031
Degradation penalty	0.0
$K_{(nom+bias)}$	0.9304
Uncertainties	0.0109
K_{max} (Maximum reactivity)	0.9413

Figure 2.8-1 Power of Peak Bundle versus Cycle Exposure

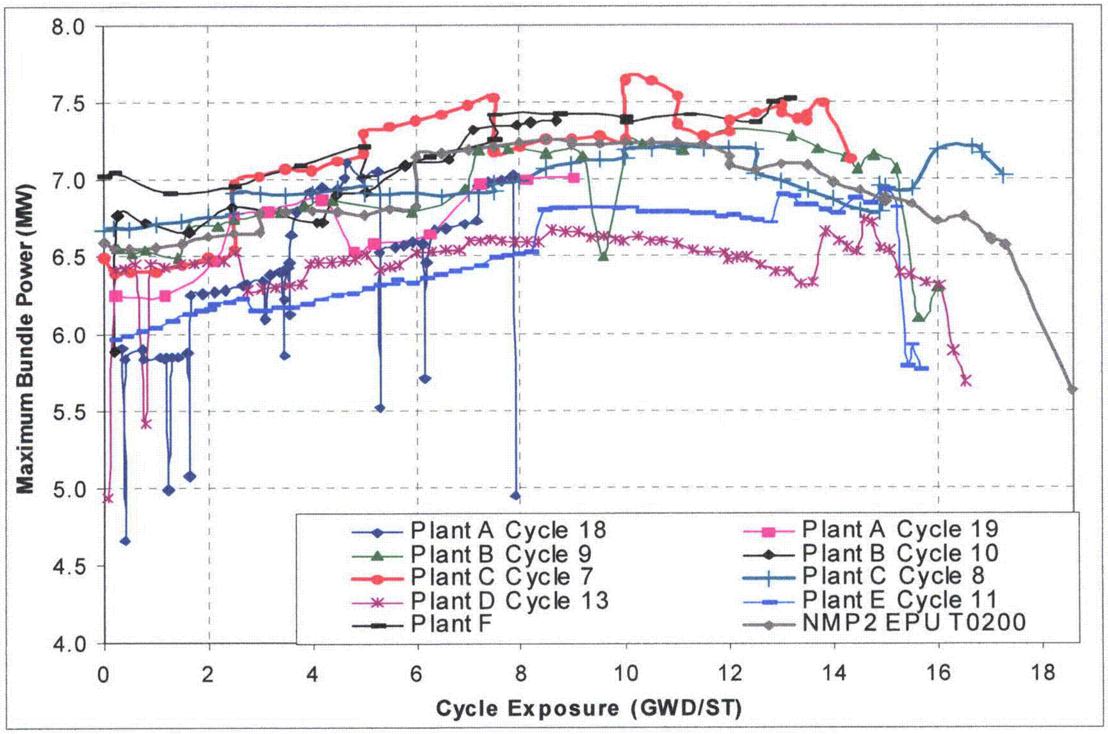


Figure 2.8-2 Coolant Flow for Peak Bundle versus Cycle Exposure

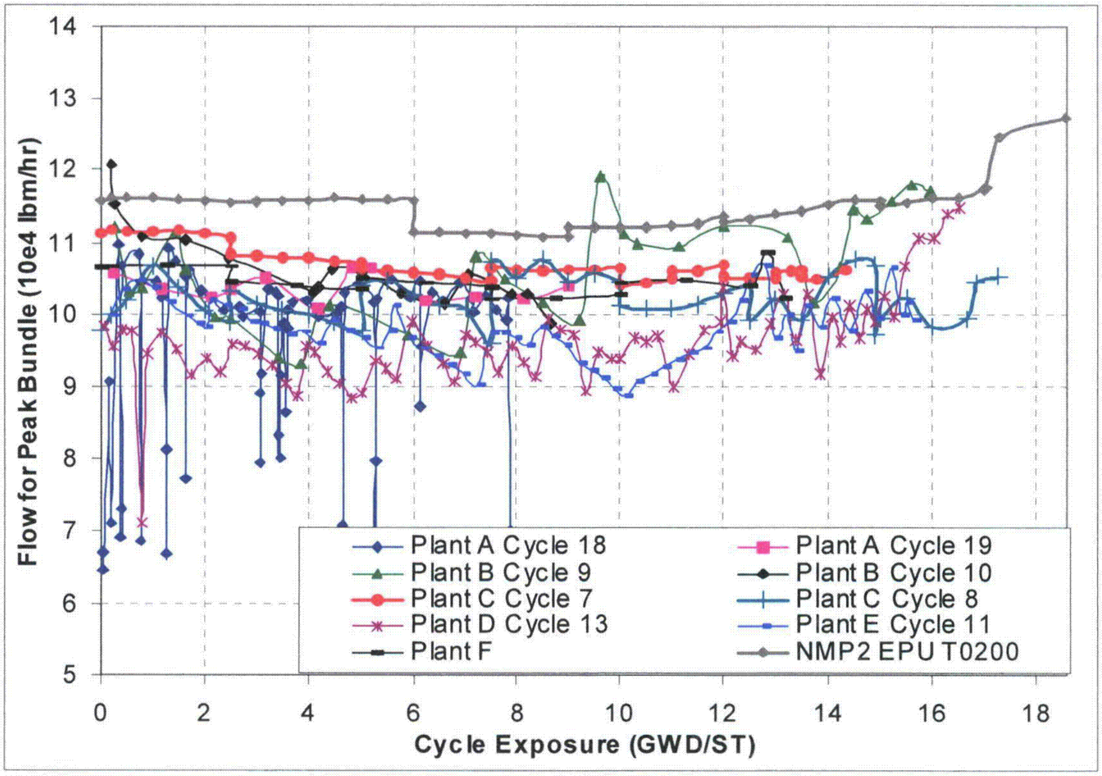


Figure 2.8-3 Exit Void Fraction for Peak Power Bundle versus Cycle Exposure

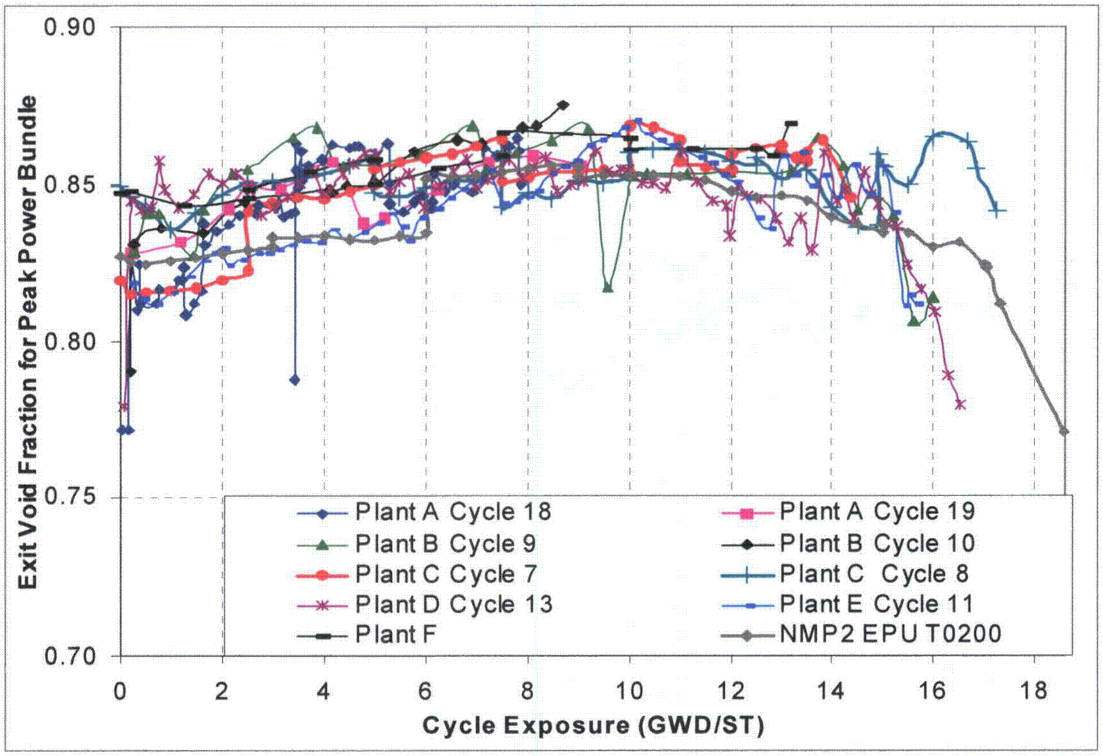


Figure 2.8-4 Maximum Channel Exit Void Fraction versus Cycle Exposure

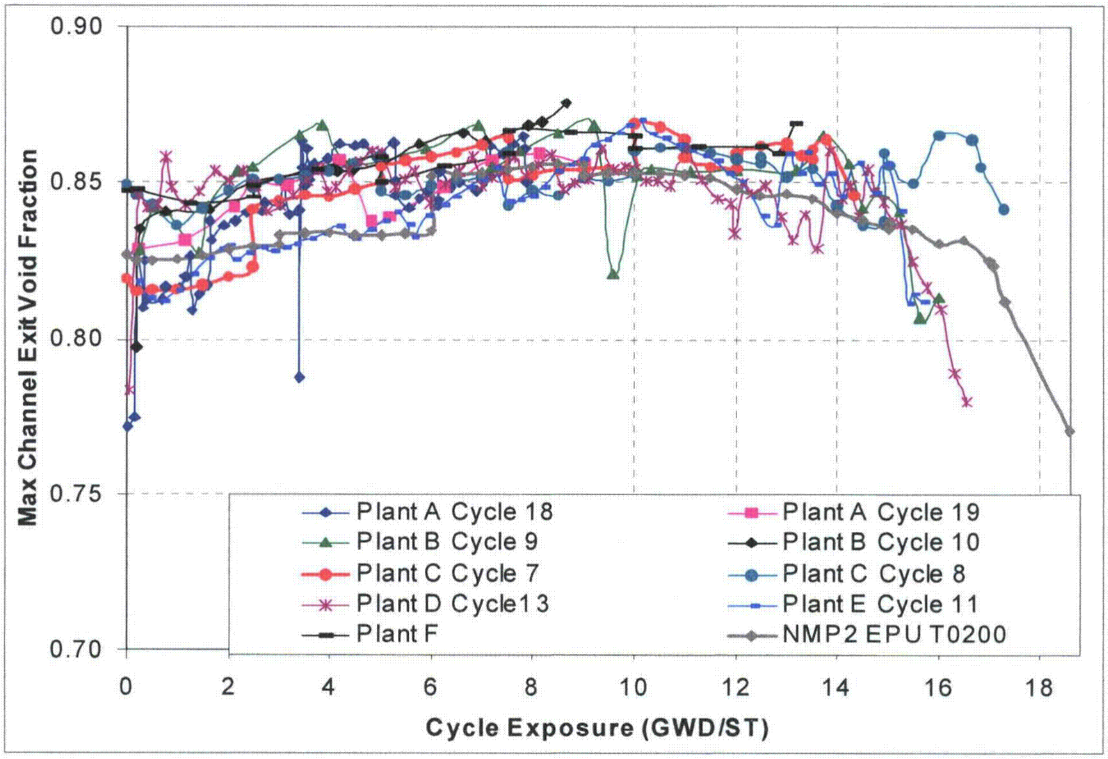


Figure 2.8-5 Core Average Exit Void Fraction versus Cycle Exposure

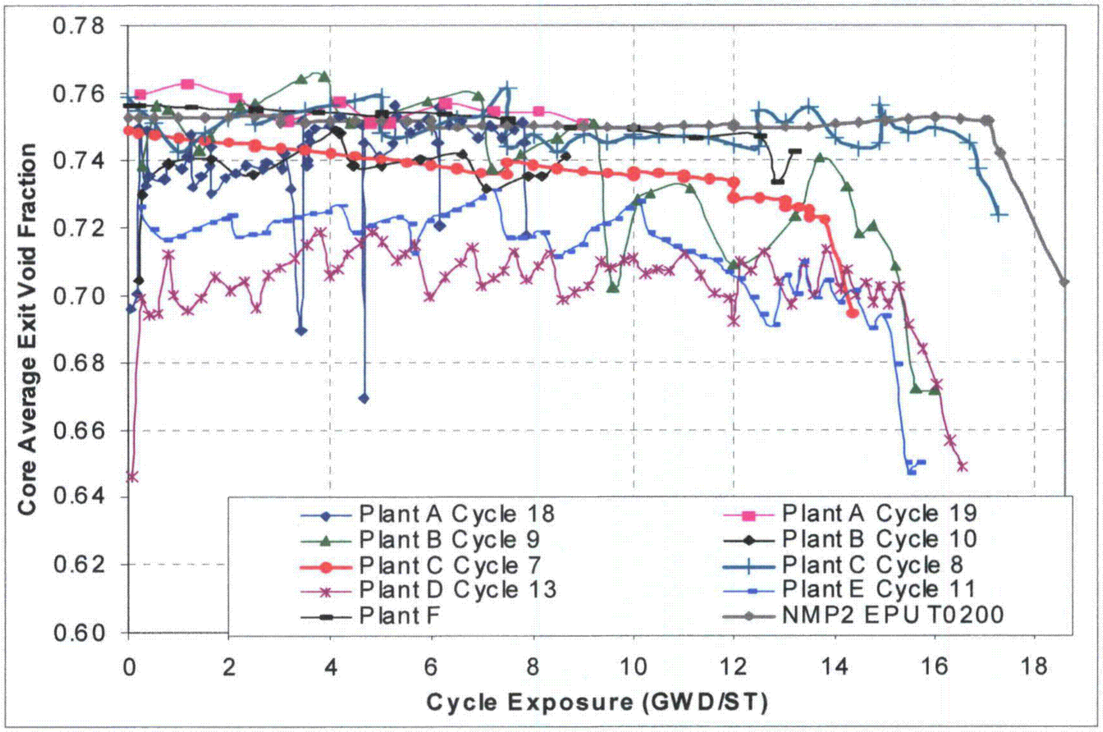


Figure 2.8-6 Peak LHGR versus Cycle Exposure

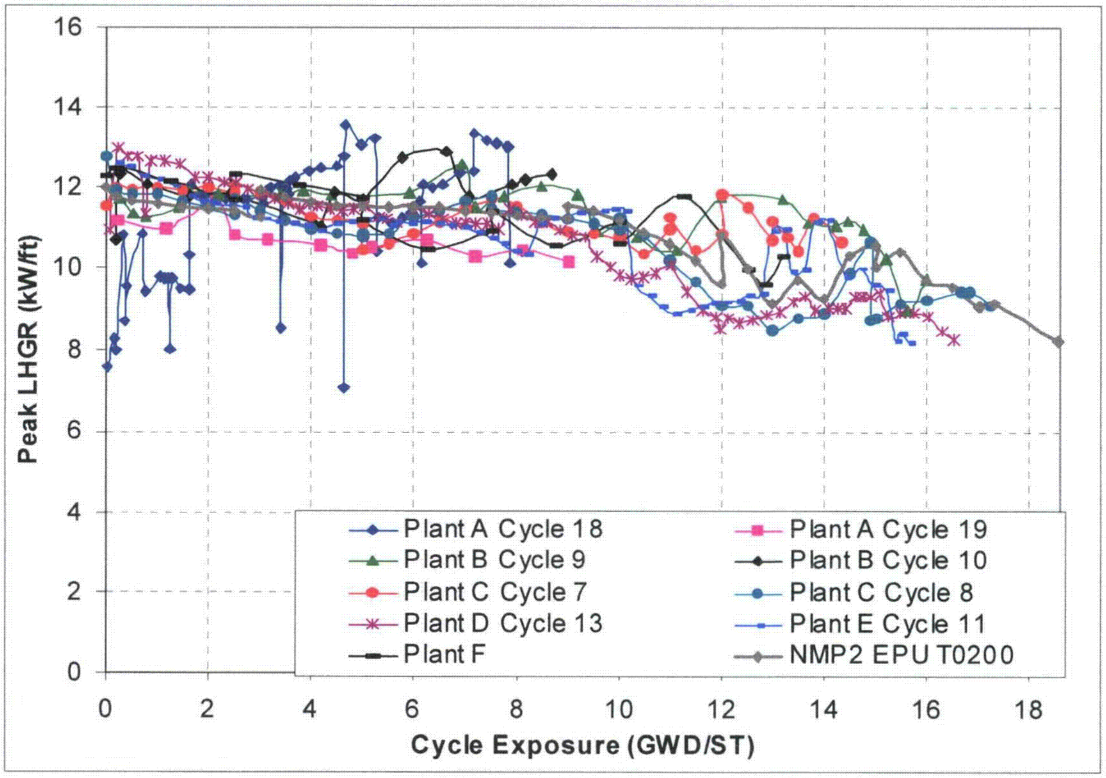


Figure 2.8-7 Peak Nodal Exposures

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	EQ -120%	51.174
NMP2 EPU T0200		51.760

Figure 2.8-8 Dimensionless Bundle Power at BOC (200 MWd/ST)

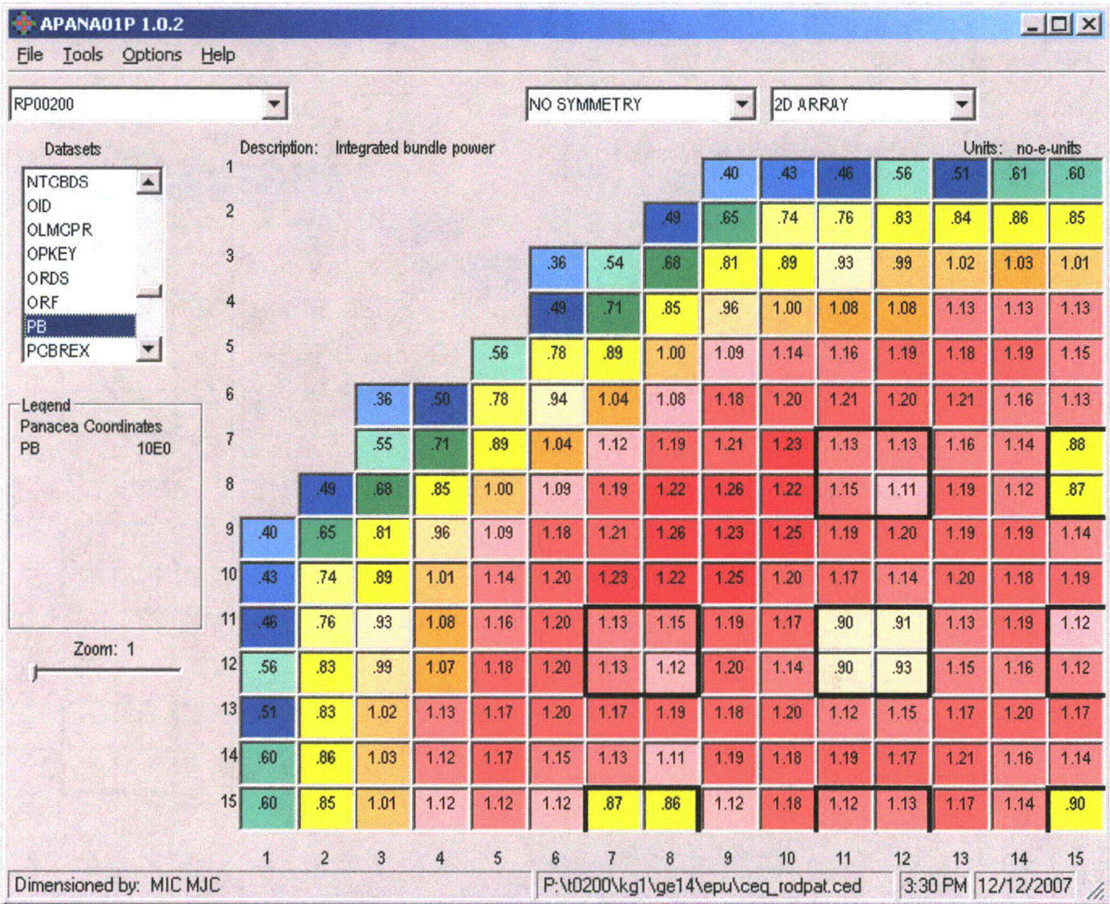


Figure 2.8-9 Dimensionless Bundle Power at MOC (10000 MWd/ST)

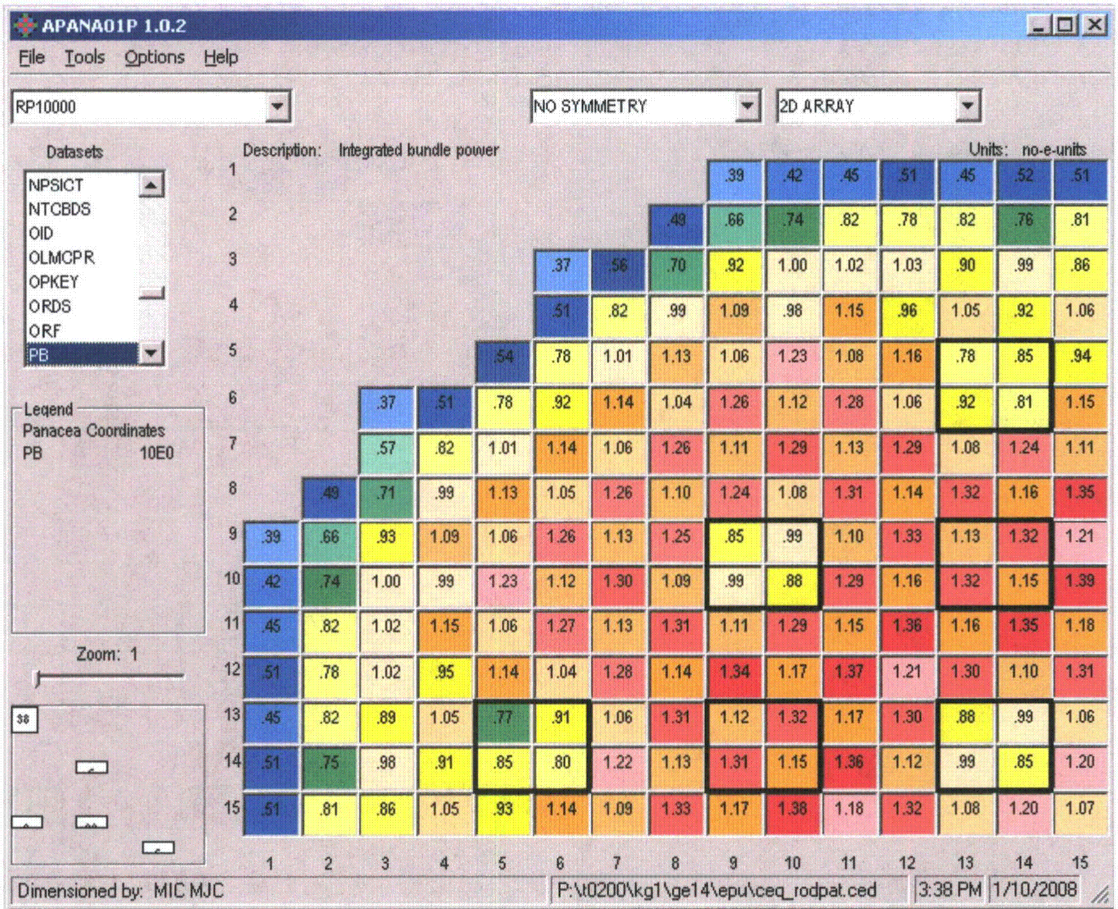


Figure 2.8-10 Dimensionless Bundle Power at EOC (18577 MWd/ST)

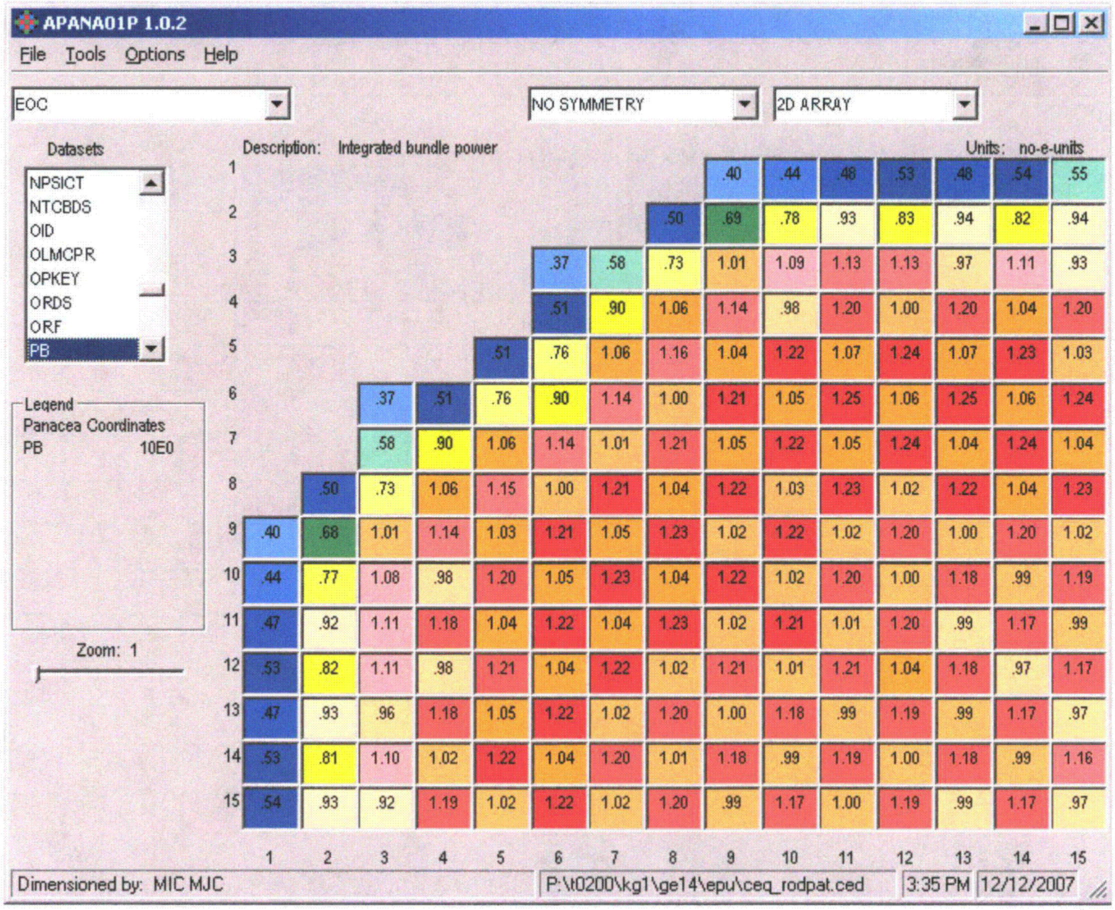
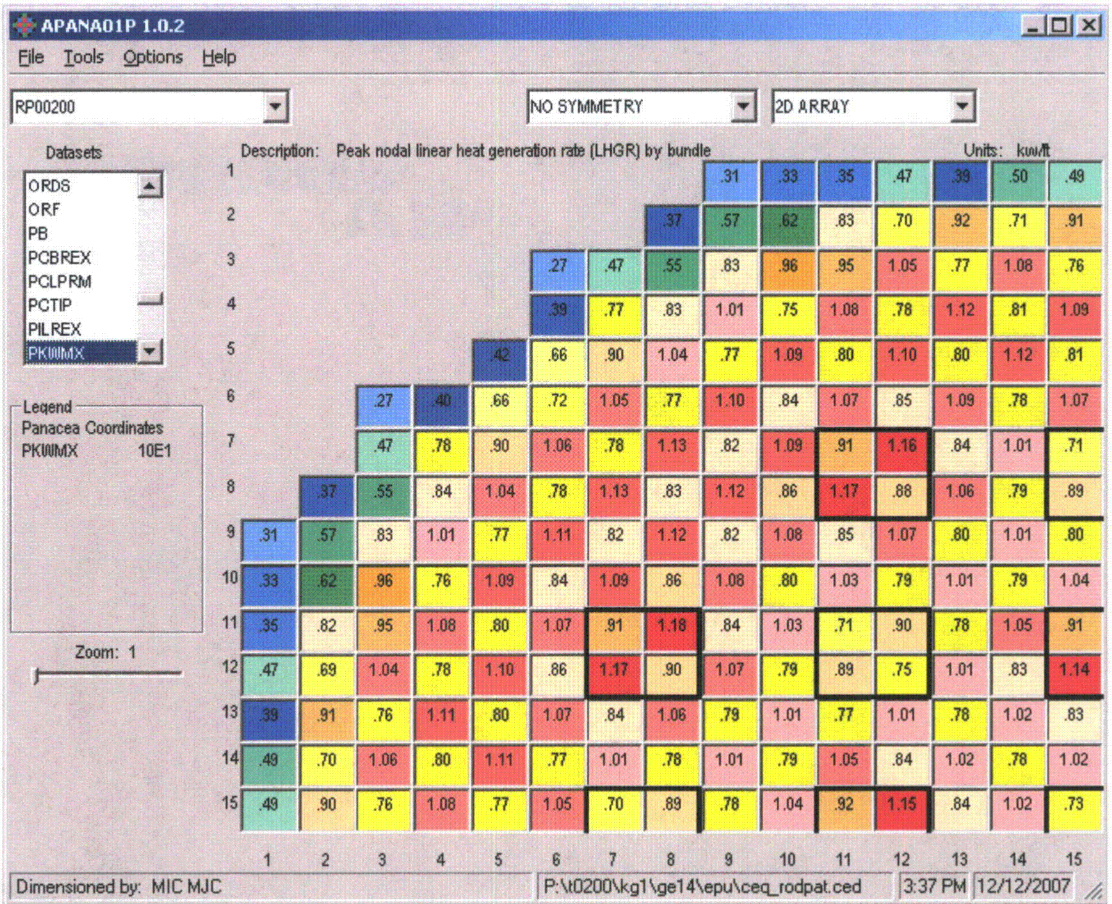
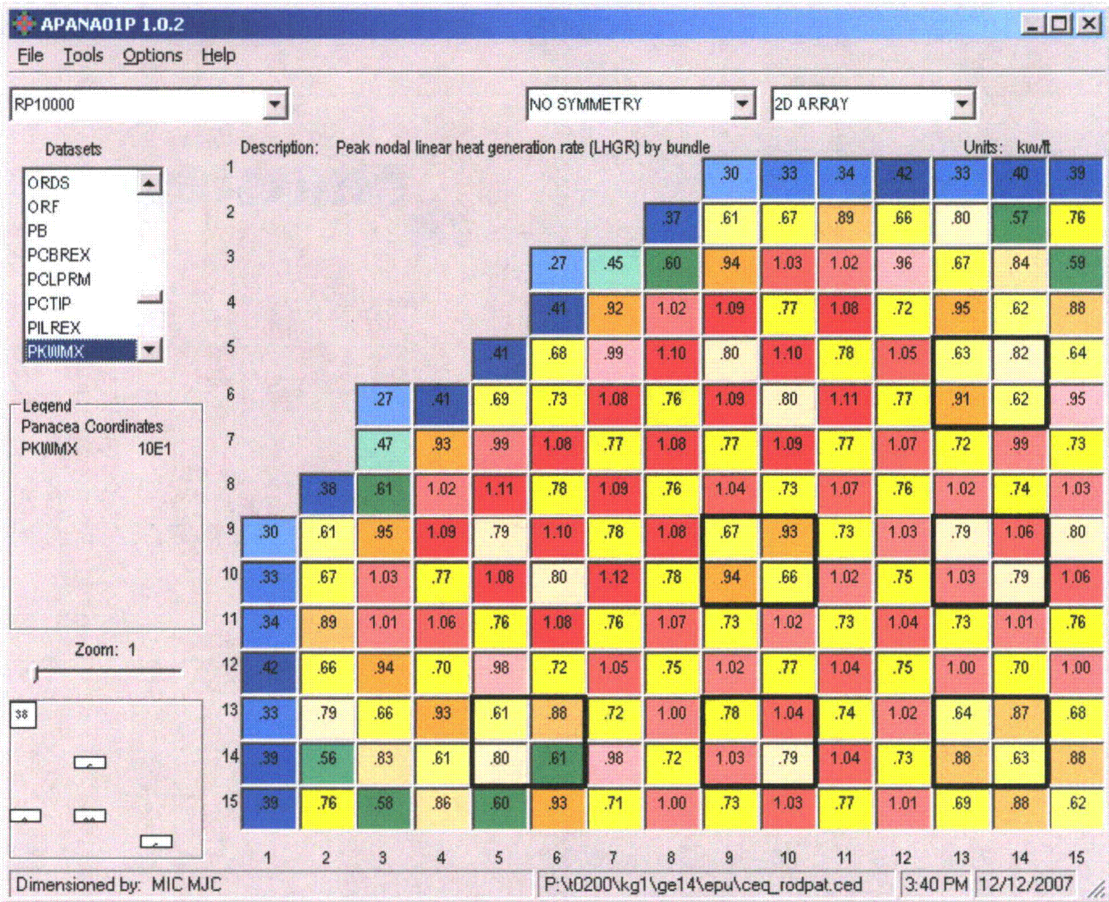


Figure 2.8-11 Bundle Operating LHGR (KW/ft) at BOC (200 MWd/ST)



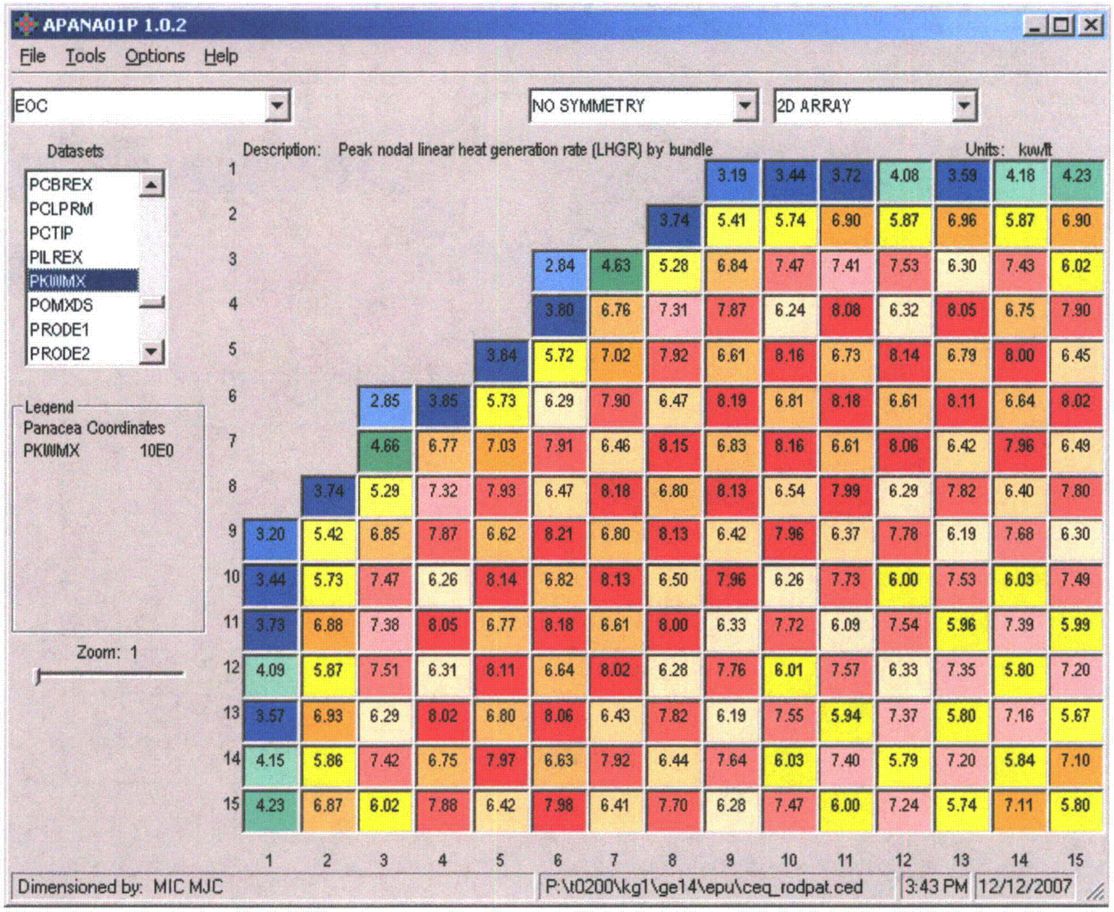
	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1									.31	.33	.35	.47	.39	.50	.49
2								.37	.57	.62	.83	.70	.92	.71	.91
3						.27	.47	.55	.83	.96	.95	1.05	.77	1.08	.76
4						.39	.77	.83	1.01	.75	1.08	.78	1.12	.81	1.09
5					.42	.66	.90	1.04	.77	1.09	.80	1.10	.80	1.12	.81
6			.27	.40	.66	.72	1.05	.77	1.10	.84	1.07	.85	1.09	.78	1.07
7			.47	.78	.90	1.06	.78	1.13	.82	1.09	.91	1.16	.84	1.01	.71
8		.37	.55	.84	1.04	.78	1.13	.83	1.12	.86	1.17	.88	1.06	.79	.89
9	.31	.57	.83	1.01	.77	1.11	.82	1.12	.82	1.08	.85	1.07	.80	1.01	.80
10	.33	.62	.96	.76	1.09	.84	1.09	.86	1.08	.80	1.03	.79	1.01	.79	1.04
11	.35	.82	.95	1.08	.80	1.07	.91	1.18	.84	1.03	.71	.90	.78	1.05	.91
12	.47	.69	1.04	.78	1.10	.86	1.17	.90	1.07	.79	.89	.75	1.01	.83	1.14
13	.39	.91	.76	1.11	.80	1.07	.84	1.06	.79	1.01	.77	1.01	.78	1.02	.83
14	.49	.70	1.06	.80	1.11	.77	1.01	.78	1.01	.79	1.05	.84	1.02	.78	1.02
15	.49	.90	.76	1.08	.77	1.05	.70	.89	.78	1.04	.92	1.15	.84	1.02	.73

Figure 2.8-12 Bundle Operating LHGR (KW/ft) at MOC (10000 MWd/ST)



	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1									.30	.33	.34	.42	.33	.40	.39
2								.37	.61	.67	.89	.66	.80	.57	.76
3						.27	.45	.60	.94	1.03	1.02	.96	.67	.84	.59
4						.41	.92	1.02	1.09	.77	1.08	.72	.95	.62	.88
5					.41	.68	.99	1.10	.80	1.10	.78	1.05	.63	.82	.64
6			.27	.41	.69	.73	1.08	.76	1.09	.80	1.11	.77	.91	.62	.95
7			.47	.93	.99	1.08	.77	1.08	.77	1.09	.77	1.07	.72	.99	.73
8		.38	.61	1.02	1.11	.78	1.09	.76	1.04	.73	1.07	.76	1.02	.74	1.03
9	.30	.61	.95	1.09	.79	1.10	.78	1.08	.67	.93	.73	1.03	.79	1.06	.80
10	.33	.67	1.03	.77	1.08	.80	1.12	.78	.94	.66	1.02	.75	1.03	.79	1.06
11	.34	.89	1.01	1.06	.76	1.08	.76	1.07	.73	1.02	.73	1.04	.73	1.01	.76
12	.42	.66	.94	.70	.98	.72	1.05	.75	1.02	.77	1.04	.75	1.00	.70	1.00
13	.33	.79	.66	.93	.61	.88	.72	1.00	.78	1.04	.74	1.02	.64	.87	.68
14	.39	.56	.83	.61	.80	.61	.98	.72	1.03	.79	1.04	.73	.88	.63	.88
15	.39	.76	.58	.86	.60	.93	.71	1.00	.73	1.03	.77	1.01	.69	.88	.62

Figure 2.8-13 Bundle Operating LHGR (KW/ft) at EOC (18577 MWd/ST)



	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1									3.19	3.44	3.72	4.08	3.59	4.18	4.23
2								3.74	5.41	5.74	6.90	5.87	6.96	5.87	6.90
3						2.84	4.63	5.28	6.84	7.47	7.41	7.53	6.30	7.43	6.02
4						3.80	6.76	7.31	7.87	6.24	8.08	6.32	8.05	6.75	7.90
5					3.84	5.72	7.02	7.92	6.61	8.16	6.73	8.14	6.79	8.00	6.45
6			2.85	3.85	5.73	6.29	7.90	6.47	8.19	6.81	8.18	6.61	8.11	6.64	8.02
7			4.66	6.77	7.03	7.91	6.46	8.15	6.83	8.16	6.61	8.06	6.42	7.96	6.49
8		3.74	5.29	7.32	7.93	6.47	8.18	6.80	8.13	6.54	7.99	6.29	7.82	6.40	7.80
9	3.20	5.42	6.85	7.87	6.62	8.21	6.80	8.13	6.42	7.96	6.37	7.78	6.19	7.68	6.30
10	3.44	5.73	7.47	6.26	8.14	6.82	8.13	6.50	7.96	6.26	7.73	6.00	7.53	6.03	7.49
11	3.73	6.88	7.38	8.05	6.77	8.18	6.61	8.00	6.33	7.72	6.09	7.54	5.96	7.39	5.99
12	4.09	5.87	7.51	6.31	8.11	6.64	8.02	6.28	7.76	6.01	7.57	6.33	7.35	5.80	7.20
13	3.57	6.93	6.29	8.02	6.80	8.06	6.43	7.82	6.19	7.55	5.94	7.37	5.80	7.16	5.67
14	4.15	5.86	7.42	6.75	7.97	6.63	7.92	6.44	7.64	6.03	7.40	5.79	7.20	5.84	7.10
15	4.23	6.87	6.02	7.88	6.42	7.98	6.41	7.70	6.28	7.47	6.00	7.24	5.74	7.11	5.80

Figure 2.8-14 Bundle Operating MCPR at BOC (200 MWd/ST)

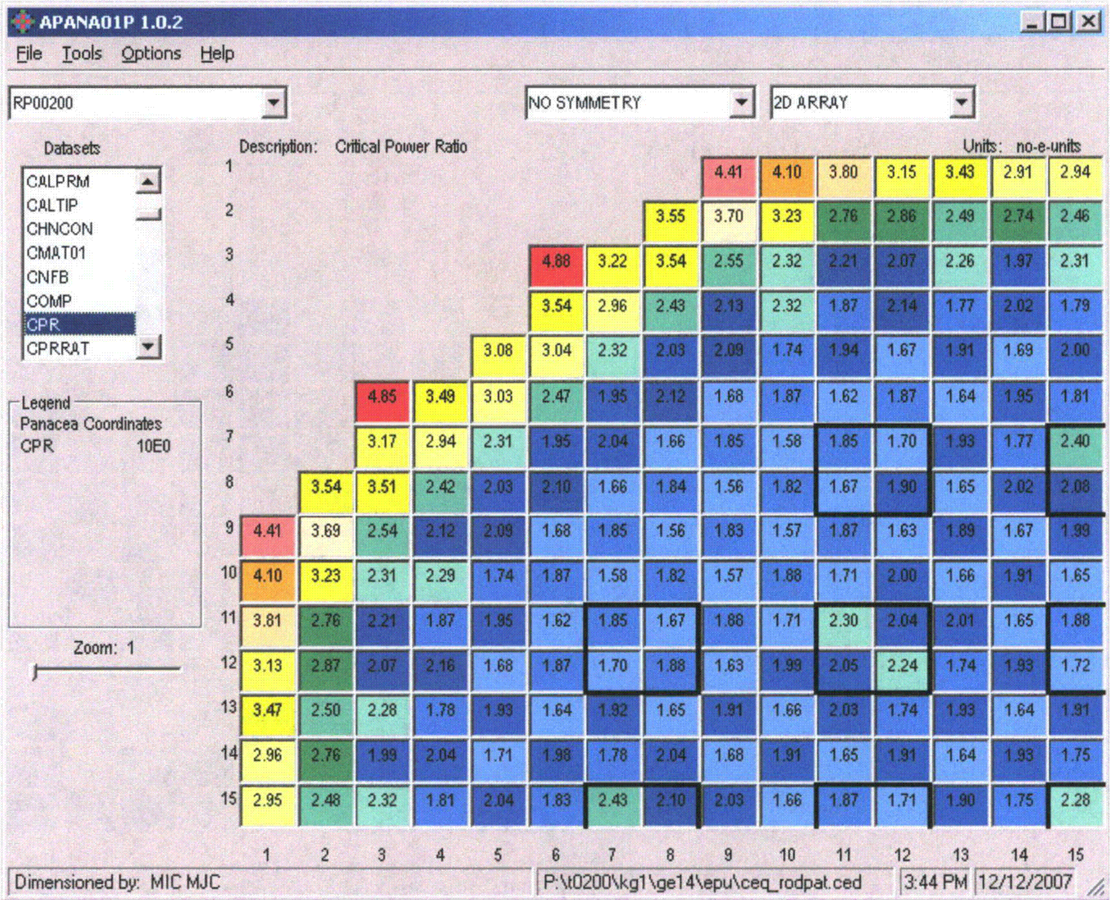


Figure 2.8-15 Bundle Operating MCPR at MOC (10000 MWd/ST)

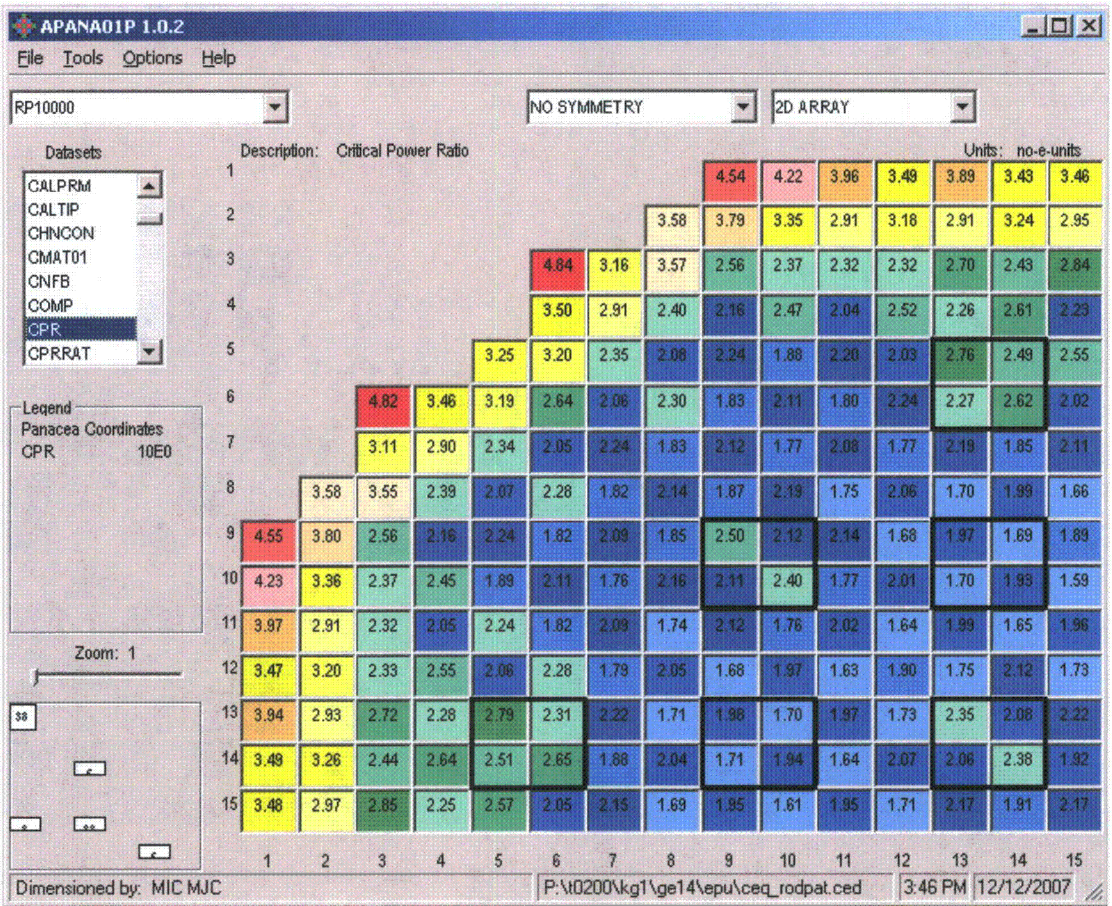
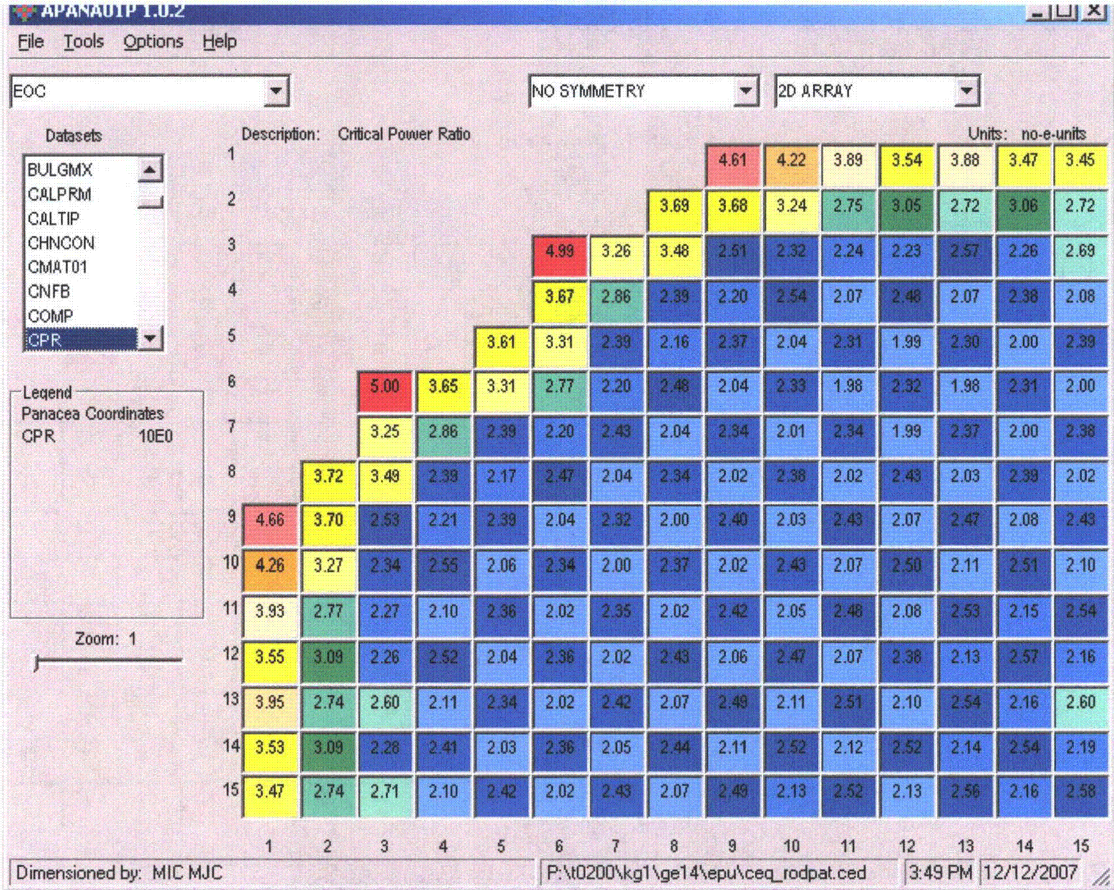


Figure 2.8-16 Bundle Operating MCPR at EOC (18577 MWd/ST)



	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1									4.61	4.22	3.89	3.54	3.88	3.47	3.45
2								3.69	3.68	3.24	2.75	3.05	2.72	3.06	2.72
3						4.99	3.26	3.48	2.51	2.32	2.24	2.23	2.57	2.26	2.69
4						3.67	2.86	2.39	2.20	2.54	2.07	2.48	2.07	2.38	2.08
5					3.61	3.31	2.39	2.16	2.37	2.04	2.31	1.99	2.30	2.00	2.39
6			5.00	3.65	3.31	2.77	2.20	2.48	2.04	2.33	1.98	2.32	1.98	2.31	2.00
7			3.25	2.86	2.39	2.20	2.43	2.04	2.34	2.01	2.34	1.99	2.37	2.00	2.38
8		3.72	3.49	2.39	2.17	2.47	2.04	2.34	2.02	2.38	2.02	2.43	2.03	2.39	2.02
9	4.66	3.70	2.53	2.21	2.39	2.04	2.32	2.00	2.40	2.03	2.43	2.07	2.47	2.08	2.43
10	4.26	3.27	2.34	2.55	2.06	2.34	2.00	2.37	2.02	2.43	2.07	2.50	2.11	2.51	2.10
11	3.93	2.77	2.27	2.10	2.36	2.02	2.35	2.02	2.42	2.05	2.48	2.08	2.53	2.15	2.54
12	3.55	3.09	2.26	2.52	2.04	2.36	2.02	2.43	2.06	2.47	2.07	2.38	2.13	2.57	2.16
13	3.95	2.74	2.60	2.11	2.34	2.02	2.42	2.07	2.49	2.11	2.51	2.10	2.54	2.16	2.60
14	3.53	3.09	2.28	2.41	2.03	2.36	2.05	2.44	2.11	2.52	2.12	2.52	2.14	2.54	2.19
15	3.47	2.74	2.71	2.10	2.42	2.02	2.43	2.07	2.49	2.13	2.52	2.13	2.56	2.16	2.58

Figure 2.8-17 Bundle Operating LHGR (KW/ft) at 00 MWd/ST (peak MFLPD point)

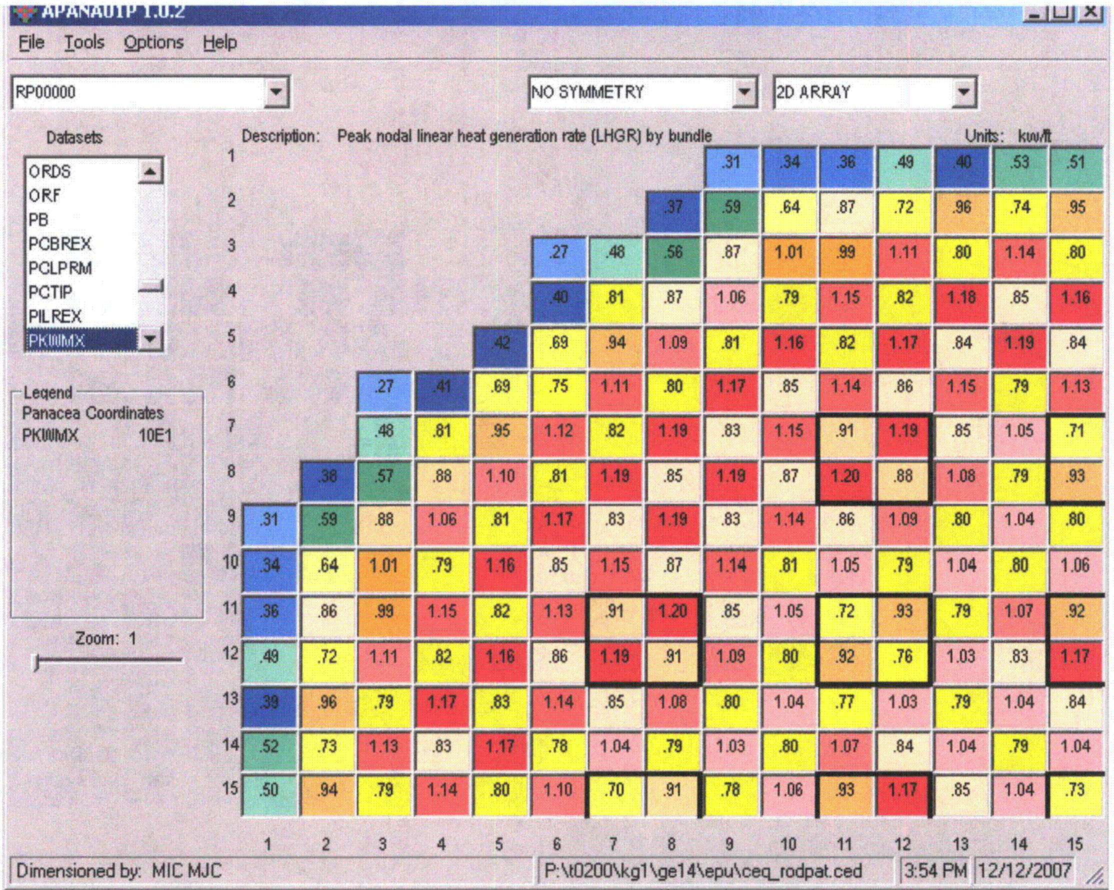


Figure 2.8-18 Bundle Operating MCPR at 6000 MWd/ST (peak MFLCPR point)

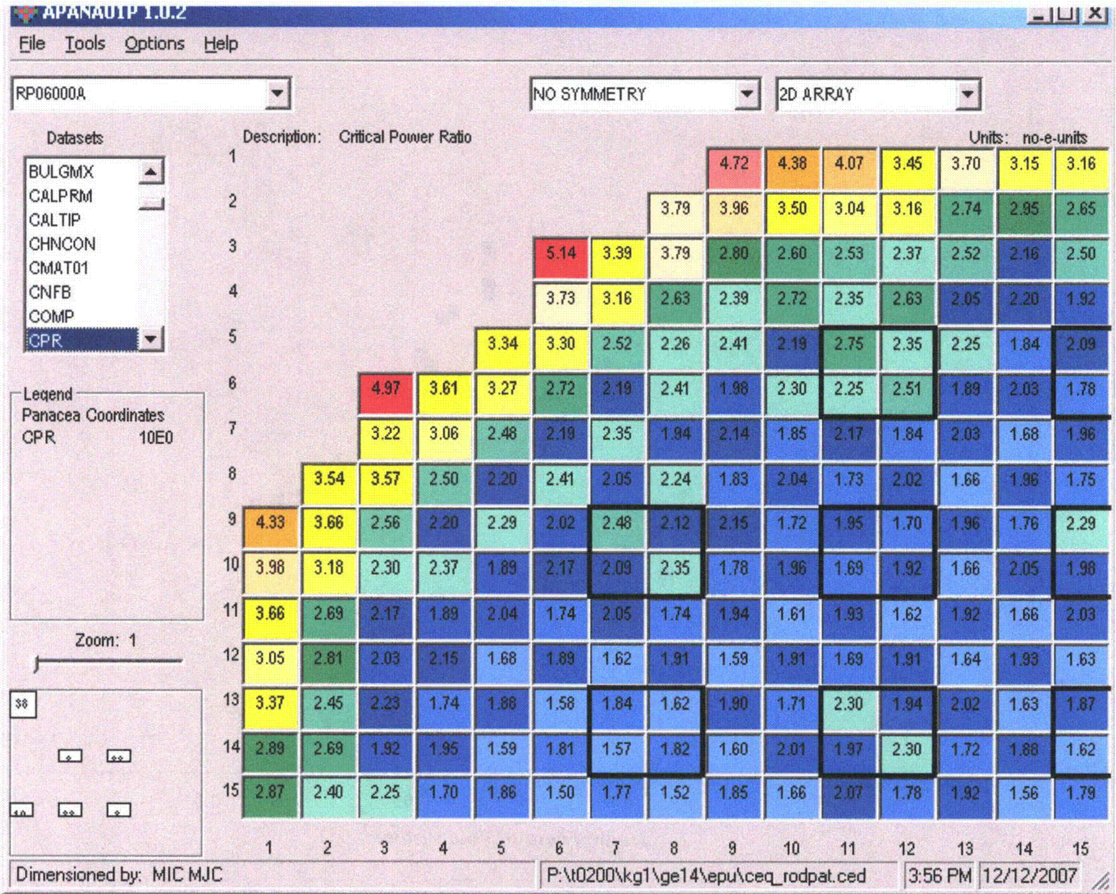


Figure 2.8-19 Bundle Average Void Fractions for Bundles Having Low CPRs

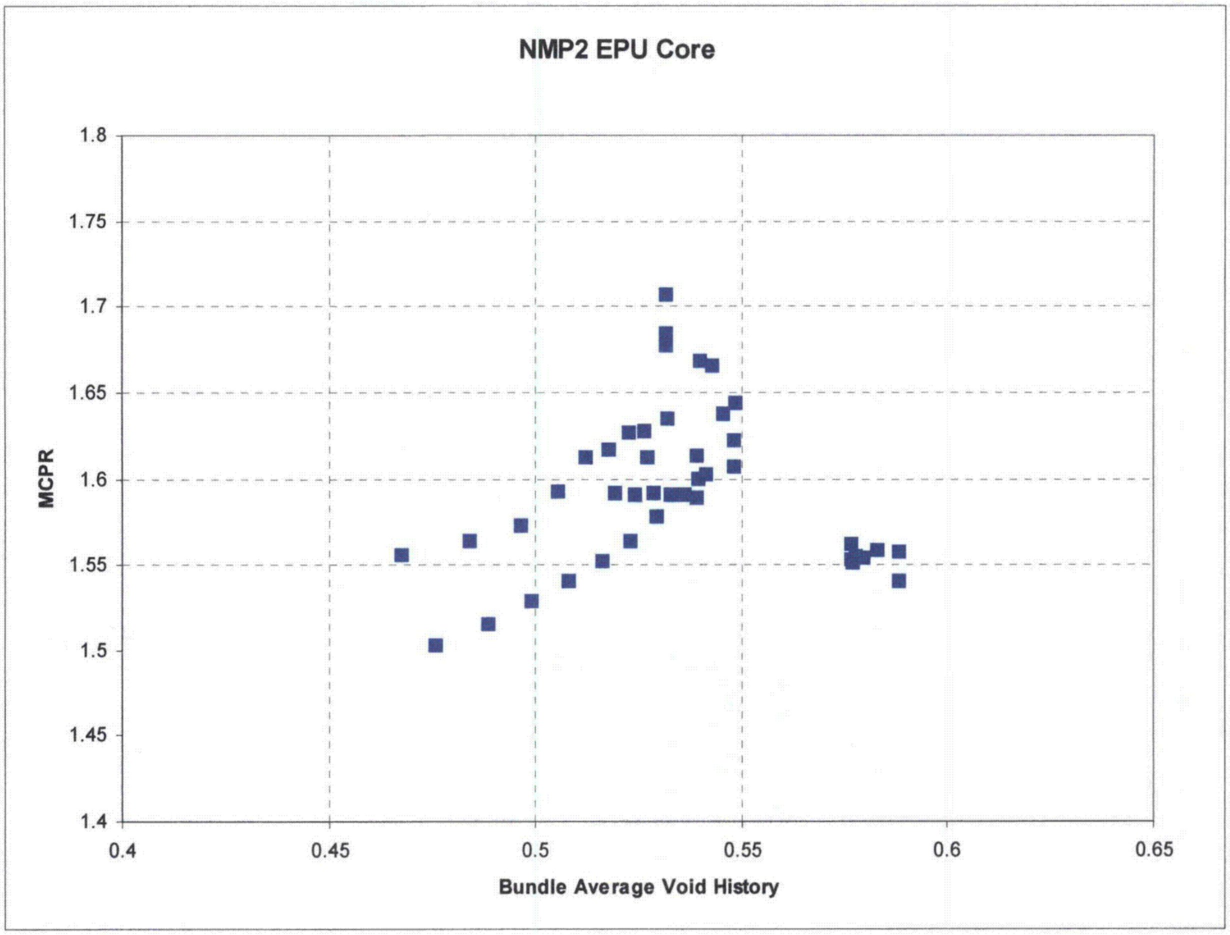


Figure 2.8-20 Illustration of OPRM Trip-Enabled Region

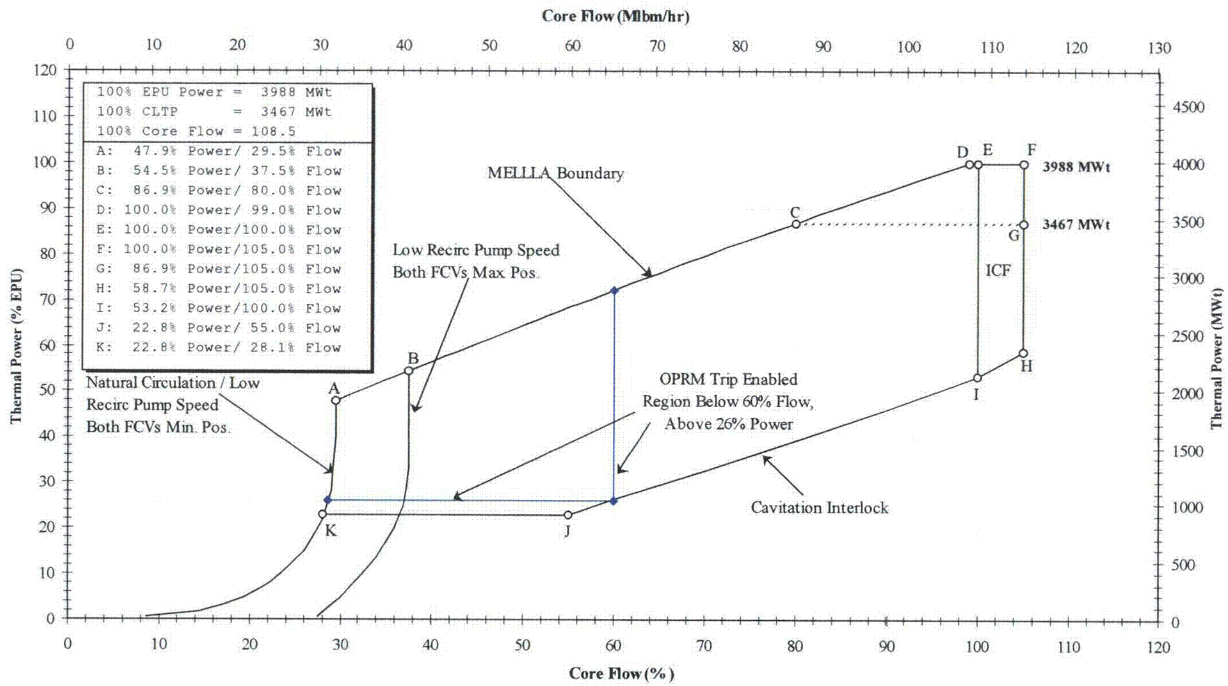


Figure 2.8-21 Proposed BSP Regions for Equilibrium GE14 Core

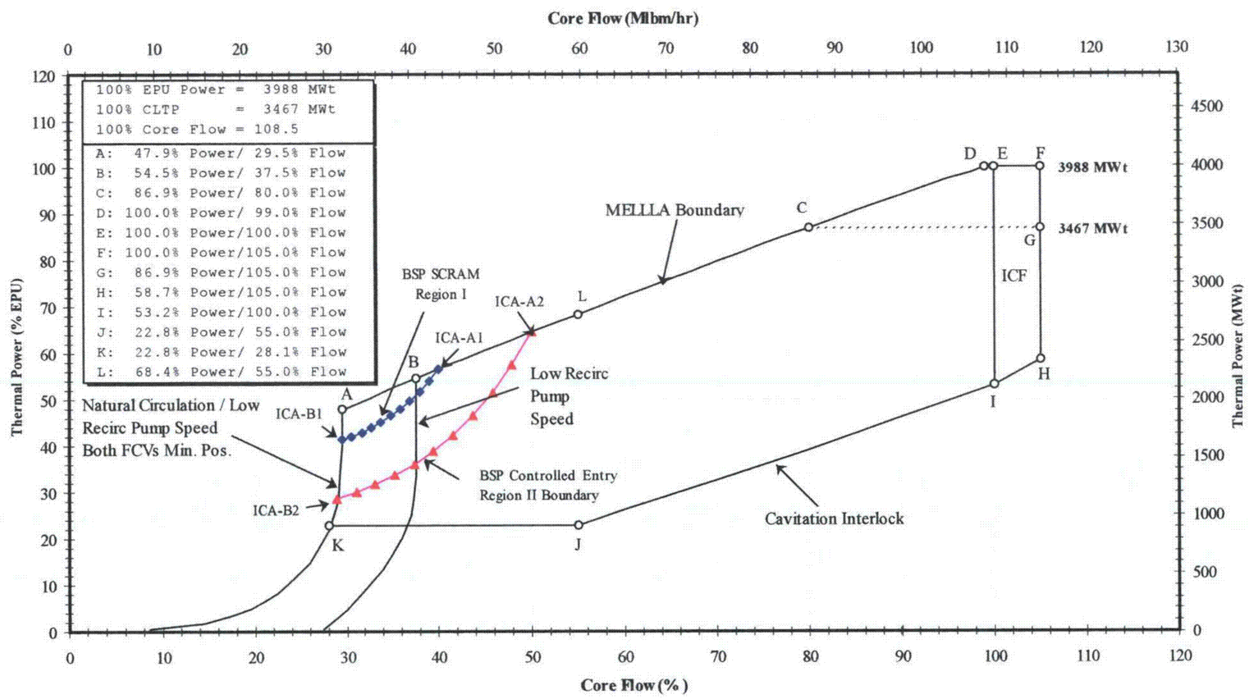


Figure 2.8-22 Response to MSIV Closure with Flux Scram

(102% EPU power, 105% core flow, and 1050 psia initial dome pressure)

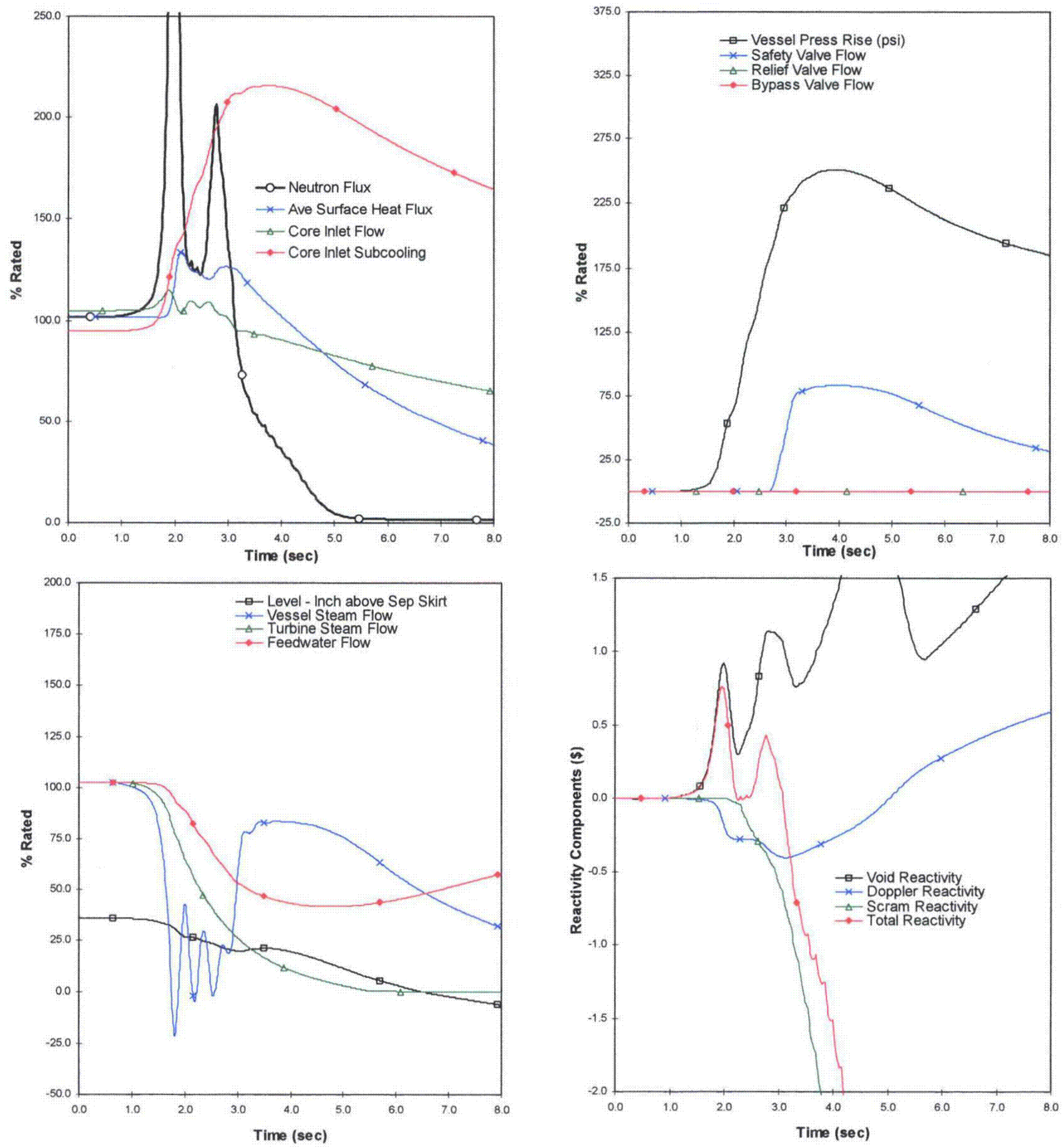
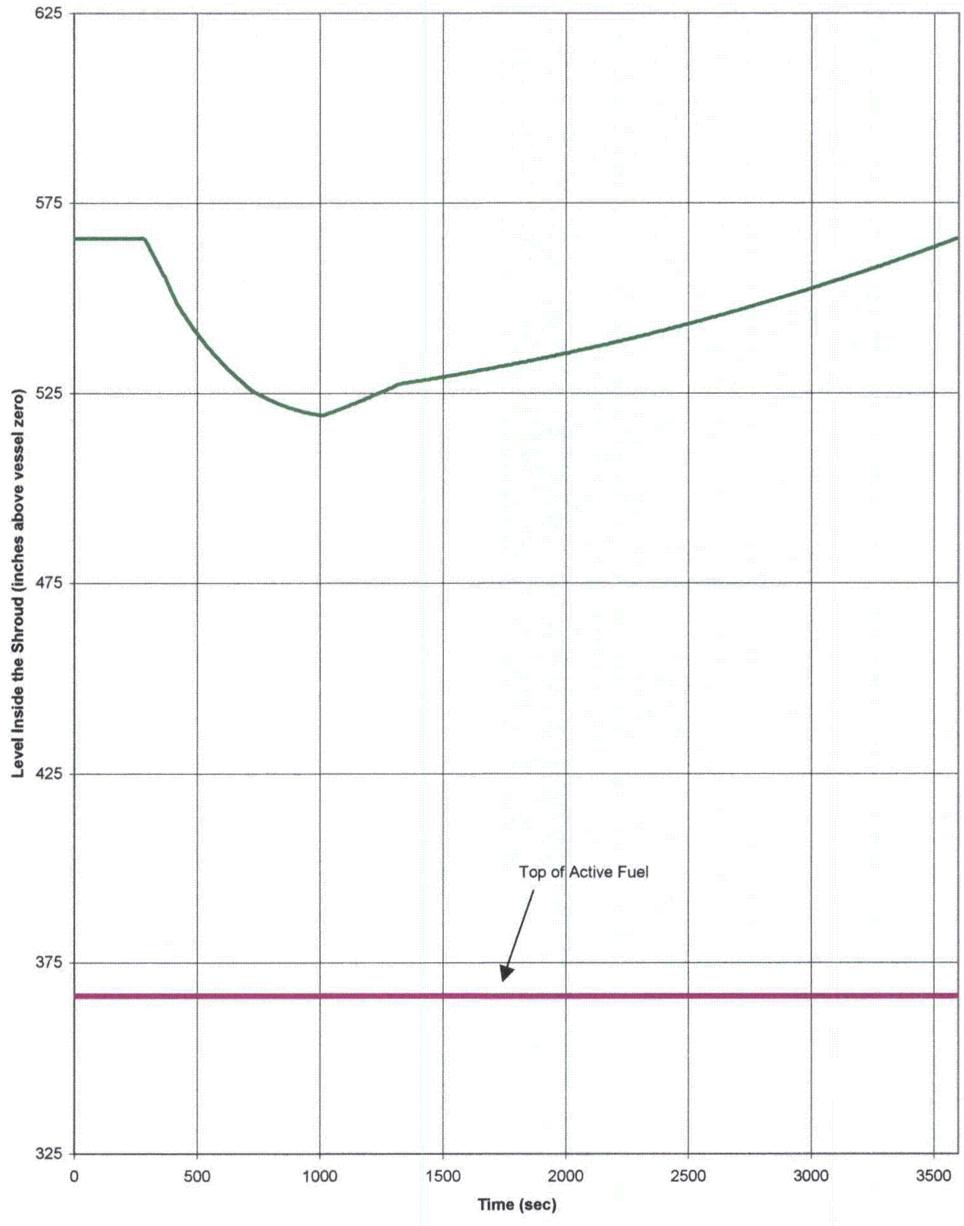


Figure 2.8-23 Loss of Feedwater Flow – Level Inside the Shroud



2.9 Source Terms and Radiological Consequences Analyses

2.9.1 Source Terms for Radwaste Systems Analyses

Regulatory Evaluation

NMPNS reviewed the radioactive source term associated with EPU to ensure the adequacy of the sources of radioactivity used by NMP2 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 NMPNS review included the parameters used to determine (1) the concentration of each radionuclide in the reactor coolant, (2) the fraction of fission product activity released to the reactor coolant, (3) concentrations of all radionuclides other than fission products in the reactor coolant, (4) leakage rates and associated fluid activity of all potentially radioactive water and steam systems, and (5) potential sources of radioactive materials in effluents that are not considered in the plant's USAR related to liquid waste management systems and gaseous waste management systems. The NRC's acceptance criteria for source terms are based on (1) 10 CFR 20, insofar as it establishes requirements for radioactivity in liquid and gaseous effluents released to unrestricted areas; (2) 10 CFR 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) GDC60, insofar as it requires that the plant design include means to control the release of radioactive effluents.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of Constant Pressure Power Uprates. Sections 8.3 and 8.4 of the CLTR address the effect of Constant Pressure Power Uprate on the Radiation Sources in the Reactor Core and in the Reactor Coolant. The results of this evaluation are described below.

Radiation sources in the reactor coolant at NMP2 include activation products, activated corrosion products, and fission products. Tables 2.9-1 through 2.9-6 contain the activity levels, concentrations, and release rates for these radiation sources.

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 activation of the water in the core region is in approximate proportion to the increase in thermal power. However, while the magnitude of the source production increases in proportion to power, the concentration in the steam remains nearly constant. This is because the increase in activation production is balanced by the increase in steam flow. The margin in the NMP2 plant design basis for reactor coolant activation concentrations significantly exceeds potential increases due to EPU. Therefore, no change is required in the activation design basis reactor coolant concentrations for 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 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.

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 0.037 Curies/sec, within the original design basis of 0.35 Curies/sec. Therefore, no change is required in the design basis for offgas activity for 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. EPU fission product activity levels in the reactor water remain a fraction (< 12%) of the design basis fission product activity.

The total activated corrosion product activity was calculated to be less than 28% of design basis levels. The sum of the activated corrosion product activity and the fission product activity remains a small fraction (< 12% for water, <15% for steam) of the total design basis activity. Therefore, the activated corrosion product and fission product activities design bases for NMP2 are unchanged for EPU.

For EPU, normal radiation sources are expected to increase slightly. Shielding aspects of the plant were conservatively designed for total normal radiation sources. Thus, the increase in radiation sources does not affect radiation zoning or shielding and plant radiation area procedural controls will compensate for increased normal radiation sources.

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.

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 calculated inventories are approximately proportional to core thermal power. Consequently, for EPU, the inventories of those radionuclides, which reached or approached equilibrium, are expected to increase in proportion to the thermal power increase. The inventories of the very long-lived radionuclides, which did not approach equilibrium, are both power and exposure dependent. They are expected to increase proportionally with power if the fuel irradiation time remains within the current basis. Thus, the long-lived radionuclides are expected to increase proportionally to power. The radionuclide inventories are calculated in terms of Curies per megawatt of reactor thermal power at various times after shutdown.

The results of the NMP2 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

NMPNS has reviewed the radioactive source term associated with the proposed EPU and concludes that the proposed parameters and resultant composition and quantity of radionuclides are appropriate for the evaluation of the radioactive waste management systems. NMPNS further concludes that the proposed radioactive source term meets the requirements of 10 CFR Part 20, 10 CFR 50, Appendix I, and GDC60. Therefore, NMPNS finds the proposed EPU acceptable with respect to source terms.

2.9.2 Radiological Consequences Analyses Using Alternative Source Terms

Regulatory Evaluation

NMPNS reviewed the DBA radiological consequences analyses. The radiological consequences analyses reviewed are the LOCA, fuel handling accident (FHA), control rod drop accident (CRDA), and main steamline break (MSLB). The review for each accident analysis included (1) the sequence of events; and (2) models, assumptions, and values of parameter inputs used for the calculation of the total effective dose equivalent (TEDE). 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) GDC19, 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.

Technical Evaluation

The impact of the proposed EPU on the radiological consequences of the LOCA, FHA, CRDA, and MSLB was evaluated by the NRC in a separate license amendment, Reference 34, which approved a full-scope implementation of an Alternative Source Term (AST) that complies with the guidance given in RG 1.183 and NRC Standard Review Plan 15.01. This amendment is based on 4067 MWt (corresponds to the EPU power level of 3988 MWt with a 2% ECCS evaluation uncertainty factor applied). The calculated AST dose analyses show that the dose criteria of 10 CFR 50.67 are met for the EPU power level. This analysis was subsequently validated for the EPU reactor operating domain to confirm that the AST input parameters remained bounding.

Conclusion

NMP2 has evaluated the approved license amendment containing the Alternative Source Term accident analyses performed in support of the proposed EPU and concludes that it adequately accounts for the effects of the proposed EPU. NMP2 further concludes that the plant site and the dose-mitigating ESFs remain acceptable with respect to the radiological consequences of postulated DBAs since, as set forth above, the calculated total effective dose equivalent (TEDE) at the exclusion area boundary (EAB), at the low population zone outer boundary, and in the control room meet the exposure guideline values specified in 10 CFR 50.67 and GDC19, as well as applicable acceptance criteria denoted in SRP Section 15.0.1. Therefore, the NMP2 has determined that the Alternative Source Term License Amendment is acceptable with respect to the radiological consequences of DBAs following an EPU.

Table 2.9-1 Total Activity Levels

Item	Parameter	Unit	Calculated EPU Value	Expected CLTP Value	Design Basis Value	Design Basis to EPU Comparison
1	Activity concentrations of principal radionuclides in fluid streams for normal operation	N/A	Table 2.9-2	--	--	Table 2.9-2 contains the EPU calculated radionuclide concentrations in reactor water and steam for NMP2.
2	Total fission product activity concentration in reactor water	μCi/g	1.6E-01	1.7E-01	1.5E+00	EPU value is bounded with adequate margin by design basis. See Table 2.9-4.
3	Total fission product activity concentration in reactor steam	μCi/g	2.5E-03	2.4E-03	1.7E-02	EPU value is bounded with adequate margin by design basis. See Table 2.9-4.
4	Total non-coolant activation product activity concentration in reactor water	μCi/g	2.3E-02	4.4E-02	8.5E-02	EPU value is bounded with adequate margin by design basis. See Table 2.9-5.
5	Total non-coolant activation product activity concentration in reactor steam	μCi/g	2.3E-05	4.4E-05	8.5E-05	EPU value is bounded with adequate margin by design basis. See Table 2.9-5.
6	Total fission product offgas concentrations	μCi/sec after 30 min	3.7E+04	1.0E+05	3.5E+05	EPU value is bounded with adequate margin by design basis. See Table 2.9-3.

Note: **It should be noted that some of the calculated EPU total activity levels are lower than the CLTP expected total activity levels from the USAR. This can be explained by a change in calculation methodologies. The CLTP values were calculated using the GALE code (NUREG-0016) which was issued in 1976. The EPU analysis used the NRC approved ANSI/ANS-18.1-1999 standard method, which has been updated recently (1999).**

Table 2.9-2 Activity Concentrations of Principal Radionuclides in Fluid Streams for Normal EPU Operation

	Coolant	Steam			Coolant	Steam
	µCi/g	µCi/g			µCi/g	µCi/g
Class 1				**Class 6**		
Kr-83m		5.9E-04		Fe-59	1.9E-05	1.9E-08
Kr-85m		1.0E-03		Co-58	6.2E-05	6.2E-08
Kr-85		4.0E-06		Co-60	1.2E-04	1.2E-07
Kr-87		3.3E-03		Ni-63	6.2E-07	6.2E-10
Kr-88		3.3E-03		Cu-64	1.9E-03	1.9E-06
Kr-89		2.1E-02		Zn-65	6.2E-05	6.2E-08
Xe-131m		3.3E-06		Sr-89	6.2E-05	6.2E-08
Xe-133m		4.9E-05		Sr-90	4.4E-06	4.4E-09
Xe-133		1.4E-03		Y-90	4.4E-06	4.4E-09
Xe-135m		4.4E-03		Sr-91	2.6E-03	2.6E-06
Xe-135		3.8E-03		Sr-92	6.7E-03	6.7E-06
Xe-137		2.6E-02		Y-91	2.5E-05	2.5E-08
Xe-138		1.5E-02		Y-92	4.0E-03	4.0E-06
Class 2				Y-93	2.6E-03	2.6E-06
I-131	3.3E-03	6.5E-05		Zr-95	5.0E-06	5.0E-09
I-132	2.7E-02	5.3E-04		Nb-95	5.0E-06	5.0E-09
I-133	2.2E-02	4.3E-04		Mo-99	1.3E-03	1.3E-06
I-134	4.4E-02	8.9E-04		Tc-99m	1.3E-03	1.3E-06
I-135	3.0E-02	6.0E-04		Ru-103	1.2E-05	1.2E-08
Class 3				Rh-103m	1.2E-05	1.2E-08
Rb-89	3.6E-03	3.6E-06		Ru-106	1.9E-06	1.9E-09
Cs-134	1.7E-05	1.7E-08		Rh-106	1.9E-06	1.9E-09
Cs-136	1.1E-05	1.1E-08		Ag-110m	6.2E-07	6.2E-10
Cs-137	4.5E-05	4.5E-08		Te-129m	2.5E-05	2.5E-08
Cs-138	7.2E-03	7.2E-06		Te-131m	6.3E-05	6.3E-08
Ba-137m	4.5E-05	4.5E-08		Te-132	6.3E-06	6.3E-09
Class 4				Ba-140	2.5E-04	2.5E-07
N-16	6.0E+01	5.0E+01		La-140	2.5E-04	2.5E-07
Class 5				Ce-141	1.9E-05	1.9E-08
H-3	1.0E-02	1.0E-02		Ce-144	1.9E-06	1.9E-09
Class 6				Pr-144	1.9E-06	1.9E-09
Na-24	1.3E-03	1.3E-06		W-187	1.9E-04	1.9E-07
P-32	2.5E-05	2.5E-08		Np-239	5.0E-03	5.0E-06
Cr-51	1.9E-03	1.9E-06				
Mn-54	2.2E-05	2.2E-08				
Mn-56	1.7E-02	1.7E-05				
Fe-55	6.2E-04	6.2E-07				

Table 2.9-3 EPU Noble Gas Radionuclide Source Term and Design Basis Comparison

Isotope	Steam, Valid for t=30 min		
	EPU noble gas source term	Expected CLTP noble gas source term (USAR Table 11.1-3)	Design Basis noble gas source term (USAR Table 11.3-2)
Class 1	**μCi/sec**	**μCi/sec**	**μCi/sec**
Kr-83m	1.1E+03	2.8E+03	
Kr-85m	2.1E+03	5.7E+03	
Kr-85	8.9E+00	2.0E+01	
Kr-87	5.6E+03	1.5E+04	
Kr-88	6.5E+03	1.8E+04	
Kr-89	6.5E+01	1.8E+02	
Xe-131m	7.3E+00	1.5E+01	
Xe-133m	1.1E+02	2.9E+02	
Xe-133	3.1E+03	8.2E+03	
Xe-135m	2.5E+03	6.7E+03	
Xe-135	8.1E+03	2.1E+04	
Xe-137	2.6E+02	6.7E+02	
Xe-138	7.7E+03	2.1E+04	
Total	**3.7E+04**	**1.0E+05**	**3.5E+05**

Table 2.9-4 EPU Fission Product Reactor Water and Steam Comparisons to Expected and Design Basis Values

Isotope	EPU Analysis Values µCi/gm	CLTP Expected Values µCi/gm	Design Basis Values µCi/gm	EPU Analysis Values µCi/gm	CLTP Expected Values µCi/gm	Design Basis Values µCi/gm
	Reactor Water			Steam		
I-131	3.3E-03	1.7E-03	1.3E-02	6.5E-05	2.7E-05	2.6E-04
I-132	2.7E-02	3.1E-02	2.2E-01	5.3E-04	4.7E-04	3.3E-03
I-133	2.2E-02	2.3E-02	1.6E-01	4.3E-04	3.7E-04	2.6E-03
I-134	4.4E-02	5.7E-02	4.0E-01	8.9E-04	1.1E-03	7.8E-03
I-135	3.0E-02	2.4E-02	1.7E-01	6.0E-04	3.9E-04	2.8E-03
Rb-89	3.6E-03	3.1E-03	2.2E-02	3.6E-06	3.1E-06	2.2E-05
Cs-134	1.7E-05	1.2E-05	8.5E-05	1.7E-08	1.2E-08	8.5E-08
Cs-136	1.1E-05	7.8E-06	5.5E-05	1.1E-08	7.8E-09	5.5E-08
Cs-137	4.5E-05	3.1E-05	2.2E-04	4.5E-08	3.1E-08	2.2E-07
Cs-138	7.2E-03	6.0E-03	1.6E-01	7.2E-06	6.0E-06	1.6E-04
Ba-137m	4.5E-05	N/A	N/A	4.5E-08	N/A	N/A
Sr-89	6.2E-05	3.9E-05	1.2E-03	6.2E-08	3.9E-08	1.2E-06
Sr-90	4.4E-06	2.8E-06	8.8E-05	4.4E-09	2.8E-09	8.8E-08
Y-90	4.4E-06	N/A	N/A	4.4E-09	N/A	N/A
Sr-91	2.6E-03	1.7E-03	3.2E-02	2.6E-06	1.7E-06	3.2E-05
Sr-92	6.7E-03	4.8E-03	6.6E-02	6.7E-06	4.8E-06	6.6E-05
Y-91	2.5E-05	1.6E-05	1.1E-04	2.5E-08	1.6E-08	1.1E-07
Y-92	4.0E-03	2.8E-03	2.0E-02	4.0E-06	2.8E-06	2.0E-05
Y-93	2.6E-03	1.7E-03	1.2E-02	2.6E-06	1.7E-06	1.2E-05
Zr-95	5.0E-06	3.2E-06	2.3E-05	5.0E-09	3.2E-09	2.3E-08
Nb-95	5.0E-06	3.2E-06	2.3E-05	5.0E-09	3.2E-09	2.3E-08
Mo-99	1.3E-03	8.0E-04	8.9E-03	1.3E-06	8.0E-07	8.9E-06
Tc-99m	1.3E-03	8.8E-03	6.2E-02	1.3E-06	8.8E-06	6.2E-05
Ru-103	1.2E-05	7.9E-06	5.6E-05	1.2E-08	7.9E-09	5.6E-08
Rh-103m	1.2E-05	N/A	N/A	1.2E-08	N/A	N/A
Ru-106	1.9E-06	1.2E-06	8.5E-06	1.9E-09	1.2E-09	8.5E-09
Rh-106	1.9E-06	N/A	N/A	1.9E-09	N/A	N/A
Te-129m	2.5E-05	1.6E-05	1.3E-04	2.5E-08	1.6E-08	1.3E-07
Te-131m	6.3E-05	4.0E-05	2.8E-04	6.3E-08	4.0E-08	2.8E-07
Te-132	6.3E-06	4.0E-06	5.6E-03	6.3E-09	4.0E-09	5.6E-06
Ba-140	2.5E-04	N/A	N/A	2.5E-07	N/A	N/A
La-140	2.5E-04	1.6E-04	3.4E-03	2.5E-07	1.6E-07	3.4E-06
Ce-141	1.9E-05	1.2E-05	8.5E-05	1.9E-08	1.2E-08	8.5E-08
Ce-144	1.9E-06	1.2E-06	1.3E-05	1.9E-09	1.2E-09	1.3E-08
Pr-144	1.9E-06	N/A	N/A	1.9E-09	N/A	N/A
Np-239	5.0E-03	3.2E-03	9.5E-02	5.0E-06	3.2E-06	9.5E-05
Total	**1.6E-01**	**1.7E-01**	**1.5E+00**	**2.5E-03**	**2.4E-03**	**1.7E-02**

Table 2.9-5 EPU Activated Corrosion Product Reactor Water and Steam Comparisons to Expected and Design Basis Values

Isotope	EPU Analysis Values μCi/gm	CLTP Expected Values μCi/gm	Design Basis Values μCi/gm	EPU Analysis Values μCi/gm	CLTP Expected Values μCi/gm	Design Basis Values μCi/gm
	Reactor Water			Steam		
Na-24	1.3E-03	4.1E-03	4.1E-03	1.3E-06	4.1E-06	4.1E-06
P-32	2.5E-05	7.9E-05	7.9E-05	2.5E-08	7.9E-08	7.9E-08
Cr-51	1.9E-03	2.4E-03	2.4E-03	1.9E-06	2.4E-06	2.4E-06
Mn-54	2.2E-05	2.8E-05	4.4E-05	2.2E-08	2.8E-08	4.4E-08
Mn-56	1.7E-02	2.4E-02	5.5E-02	1.7E-05	2.4E-05	5.5E-05
Fe-55	6.2E-04	3.9E-04	3.9E-04	6.2E-07	3.9E-07	3.9E-07
Fe-59	1.9E-05	1.2E-05	8.8E-05	1.9E-08	1.2E-08	8.8E-08
Co-58	6.2E-05	7.9E-05	5.5E-03	6.2E-08	7.9E-08	5.5E-06
Co-60	1.2E-04	1.6E-04	5.5E-04	1.2E-07	1.6E-07	5.5E-07
Ni-63	6.2E-07	3.9E-07	3.9E-07	6.2E-10	3.9E-10	3.9E-10
Cu-64	1.9E-03	1.3E-02	1.3E-02	1.9E-06	1.3E-05	1.3E-05
Zn-65	6.2E-05	7.9E-05	7.9E-05	6.2E-08	7.9E-08	7.9E-08
Ag-110m	5.0E-07	3.9E-07	6.6E-05	5.0E-10	3.9E-10	6.6E-08
W-187	1.5E-04	1.2E-04	3.3E-03	1.5E-07	1.2E-07	3.3E-06
Total	**2.3E-02**	**4.4E-02**	**8.5E-02**	**2.3E-05**	**4.4E-05**	**8.5E-05**

Table 2.9-6 EPU Coolant Activation Product Reactor Water and Steam Comparisons to Expected and Design Basis Values

Isotope	EPU Analysis Values µCi/gm	CLTP Expected Values µCi/gm	Design Basis Values µCi/gm	EPU Analysis Values µCi/gm	CLTP Expected Values µCi/gm	Design Basis Values µCi/gm
	Reactor Water			Steam		
N-16	6.0E+01	6.0E+01	1.0E+02	5.0E+01	5.0E+01	1.0E+02

2.10 Health Physics

2.10.1 Occupational and Public Radiation Doses

Regulatory Evaluation

NMPNS has taken the necessary steps to ensure that any dose increases will be maintained as low as is reasonably achievable. The NMPNS review included an evaluation of any increases in radiation sources and how this may affect plant area dose rates, plant radiation zones, and plant area accessibility. NMPNS evaluated how personnel doses needed to access plant vital areas following an accident are affected. NMPNS considered the effects of the proposed EPU on Nitrogen-16 levels in the plant and any effects this increase may have on radiation doses outside the plant and at the site boundary from skyshine. NMPNS also considered the effects of the proposed EPU on plant effluent levels and any effect this increase may have on radiation doses at the site boundary. The NRC's acceptance criteria for occupational and public radiation doses are based on 10 CFR 20 and GDC19.

Technical Evaluation

Increases in Radiation Sources

The EPU maximum authorized power level of 3988 MWt is a 20% increase over the OLTP level of 3323 MWt.

The radiation sources that are affected by EPU are the sources generated inside the reactor during power operation. The production rate of reactor generated sources is approximately proportional to the core power level because the fission rate and neutron flux are proportional to the power level. Therefore, EPU is assumed to increase the production rate of reactor generated sources by approximately 20% from the OLTP production rates. Specifically, an increase of approximately 20% (from the OLTP values) is assumed for the production rate of fission products, the production rate of activation products (due to interaction of the neutron flux with the reactor water and water borne corrosion products), and radiation from the fission process itself (neutron and gamma).

The effects of EPU on doses were evaluated by assuming that the increase in fuel fission product inventory is proportional to EPU, (i.e., 20%) and that the increase in reactor water fission and activation product concentrations is proportional to the EPU increased production rate (i.e., 20%).

While the increased production rate of reactor water activation products (e.g., N-16) is approximately proportional to the increased power, the concentration in the steam at the reactor nozzle remains nearly constant because the increase in activation production is balanced by the increase in steam flow. However, because of the increased steam flow, the transit times to turbine building equipment are reduced, which reduces the decay time for short–lived isotopes such as N-16. Also, the steam pressure is greater in some components. As a result, even though the N-16 concentration in main steam does not change appreciably, the total inventory of N-16 does increase in turbine building equipment such as the turbines, cross-over piping, and condenser. The N-16 inventory increases exponentially with the reduction in transit time and directly with the pressure. The limiting increase in the N-16 inventory in the major turbine system components is 30%. This is due to an increase of about 10% due to the reduced transit time and 18% due to an increase in pressure (and density), i.e. 1.10 x 1.18 = 1.3.

The moisture carryover with EPU is expected to be bounded by 0.1%. This is a factor 2 greater than the OLTP moisture carryover of 0.05%. The principal effect of this will be to change the distribution of the radionuclide activity in the waste streams. Specifically, a small fraction of the activity that would be accumulated on the reactor water cleanup system resin will be accumulated on the condensate cleanup media (i.e., filters and resin). The radionuclide inventory on the condensate cleanup media will remain a small fraction of the design basis values.

In summary, EPU is assumed to increase the reactor generated radiation sources by up to 20% and the N-16 sources will increase by up to 30%. However, as discussed below, because of the margin in the OLTP design basis sources for reactor water and steam at the reactor nozzle, the corresponding EPU sources continue to be bounded by the OLTP design basis.

The fission product activity in the reactor water is the result of a combination of tramp uranium and minute releases from the fuel rods. With an assumed increase of 20%, the fission product concentrations in reactor water remain less than 1% of the OLTP design basis sources, and the OLTP design basis sources remain bounding.

The fission product activity in reactor steam is a result of carry-over from the reactor water. With an assumed increase of 20%, the fission products in the condenser offgas, and the offgas release rates after 30 minutes of decay, remain well below the OLTP design basis of 0.35 Ci/sec after 30 minutes of decay. Furthermore, the actual releases to the environment are much less than the "Expected Releases," in the USAR. For example, for the year 2004, the releases were less than 1% of the "Expected Releases" in the USAR.

Occupational and Onsite Radiation Exposures

Inside containment, the radiation levels near the reactor vessel are assumed to increase by 20%. However, the reactor vessel is inaccessible during operation, and because of the margin in the shielding around the reactor vessel, an increase of 20% will not measurably increase occupational doses during power operation.

The radiation levels due to spent fuel are assumed to increase by 20%.

The N-16 dose is expected to increase in proportion to the N-16 inventory in the components. This increase is due to a combination of reduced transit times from the reactor vessel nozzle to the component and the increase in pressure in some components. The N-16 dose rate is expected to increase by approximately 30%. However, the areas with a significant N-16 inventory are heavily shielded and not routinely occupied, and the N-16 is only present during operation.

Outside containment, radiation shielding was specified using the OLTP design basis radiation sources. Outside containment, it is the actual operating sources, not design basis sources, which may increase. The actual OLTP N-16 concentration at the reactor nozzle is approximately 50% of the OLTP design basis. With this margin, and the increased steam flow rates and increased pressure for EPU, the dose rates in areas of the plant affected by N-16 may increase by about 30%. Other fission and activation product concentrations, and the resultant dose rates, may increase up to 20% under EPU operating conditions. The total activity associated with the condensate cleanup system may be greater, but the activity is distributed among the prefilters and demineralizers, so the increase in dose rate is expected to be less than 20%. There is sufficient margin in the NMP2 design to ensure that shielding is adequate to maintain occupational and onsite doses ALARA. Based on the 2006 Edition of NUREG 0713 (Reference 78), the total occupational dose between 1996 and 2006 ranged from 230 to 517 person-rem, so the annual occupational doses can vary by more than 50%. For example, for the year 2004, the average dose per exposed worker at Nine Mile Point (Units 1 and 2 together) was 250 mrem, which is 5% of the limit allowed by 10 CFR 20.1201. Annual cumulative occupational radiation exposure may increase by as much as 20% due to EPU, which is well within the historical variation in station annual cumulative exposure. Individual worker exposure will continue to be maintained within acceptable limits by the ALARA program, which controls access to radiation areas.

Hydrogen Water Chemistry and NobleChem processes have been implemented at NMP2. No additional increase in the N-16 doses, other than the anticipated 30% attributed to the changes in conditions associated with EPU, is expected.

The existing radiation zoning design (e.g., the maximum designed dose rates for each area of the plant) for areas outside the N-16 affected areas will not change as a result of the increased dose rates associated with EPU. A review was performed to identify areas where the design basis radiation doses could result in a change in the radiation dose zone designation as a result of EPU (Table 2.10-1). Based on this review, it was concluded that no changes in the radiation zone designations will need to be made as a result of EPU.

The effect of EPU on access to plant vital areas following an accident (Item II.b.2 of NUREG-0737) was evaluated based on existing OLTP analyses that utilize TID-14844 (Reference 79) source term methods. The evaluation determined that the existing OLTP analyses, when conservatively scaled to uprated conditions, remain compliant with the requirements of GDC19 of Appendix A to 10 CFR 50.

An additional review of the doses associated with access to vital areas was conducted to determine the effect of EPU. The times required for transit to and work in vital access areas are not changed with EPU. The operator doses are expected to increase by up to 20%. After evaluating this increase, it was concluded that all of the doses are within the limits of GDC19.

In summary, analyses and measurements have confirmed that operation under EPU conditions will have a negligible effect on onsite and occupational radiation exposure.

Public and Offsite Radiation Exposures

The primary sources of normal offsite doses at NMP1 and NMP2 are airborne releases, liquid effluent releases from the radwaste system, and gamma skyshine from N-16 in the plant turbines and some unshielded main steam piping.

Implementation of EPU could increase the components of offsite doses due to releases of airborne and liquid radioactivity by up to 20%. The component of the offsite dose due to N-16 skyshine could increase by 30% (this is based on the increase in pressure and velocity and is described in the "Increases in Radiation Sources" section above). Both NMP1 and NMP2 contribute to the offsite dose. Therefore, the increase in the total offsite dose will be less than the NMP2 assumed increases of 20% and 30%. The design basis normal offsite doses to members of the public based on the current power level and EPU are shown in Table 2.10-2.

Currently, the offsite doses from operation of NMP1 and NMP2 are well below regulatory guidelines as indicated in Table 2.10-2. The environmental monitoring program that is in place will continue to ensure that the offsite doses are well within regulatory limits and will provide indication should the doses increase above measured background levels.

In summary, analyses and measurements have confirmed that operation under EPU conditions will have a negligible effect on public and offsite radiation exposure.

Operational Radiation Protection Program

The increased production rates of fission and activation products could increase dose rates from contained sources, surface contamination, and airborne radioactivity. The current operational programs (pre-job briefings, use of supplemental shielding, pre-job decontamination, contamination control practices, etc.) will continue to ensure that, with these increases, the occupational doses will continue to remain ALARA.

Individual worker exposure will continue to be maintained within acceptable limits by the radiation protection program, which controls access to radiation areas.

In summary, the current operational radiation protection programs are capable of controlling, and compensating for, the potential increases in contained sources and surface contamination.

Conclusion

NMPNS has reviewed the effects of the proposed EPU on radiation source terms and plant radiation levels. NMPNS concludes that necessary steps have been taken to ensure that any increases in radiation doses will be maintained as low as reasonably achievable. NMPNS further concludes that the proposed EPU meets the requirements of 10 CFR 20 and GDC19. Therefore, NMPNS finds the proposed EPU acceptable with respect to radiation protection and ensuring that occupational radiation exposures will be maintained as low as reasonably achievable.

Table 2.10-1 Current and Anticipated Measured Radiation Fields in Selected Areas

Area Description	**Radiation Zone** (mrem/hr)	**Measured Survey Results** **(CLTP)** (mrem/hr)	**Anticipated Survey Results** **(EPU)** (mrem/hr)
Turbine Building 306' Open Areas	Not identified	<0.2 to 6	<0.2 to 7
Turbine Building 306' LP Turbine	I (≥100)	10 to 260	13 to 340
Turbine Building 306' East Side Open Areas	Not identified	<0.2 to 8	<0.2 to 10
Turbine Building 306' "A" MSR & "B" MSR	I (≥100)	15 to 400	20 to 520
Turbine Building 277' Condenser Bay	I (≥100)	20 to 250	26 to 325
Turbine Building 277' "A" West SJAE & "B" East SJAE	I (≥100)	4 to 1200	5 to 1560
Turbine Building 250' South Condenser Bay	I (≥100)	1 to 48	1.3 to 62
Turbine Building 250' East Condenser Bay	I (≥100)	10 to 800	13 to 1040
Turbine Building 250' - 277' "C" Feedwater Heaters	I (≥100)	0.4 to 400	0.5 to 520
Turbine Building 250' Open Areas North Hall	Not identified	1 to 15	1.3 to 20
Reactor Building 340' MSIV Room	I (≥100)	1 to 1200	1.2 to 1440
Reactor Building 215' Open Areas	Not identified	0.2 to 30 with hot spots up to 300	0.2 to 36 with hot spots up to 360

Area Description	Radiation Zone (mrem/hr)	Measured Survey Results (CLTP) (mrem/hr)	Anticipated Survey Results (EPU) (mrem/hr)
Reactor Building 198' "A" RHR HX Room	I (≥100)	5 to 80 (Shutdown)	6 to 96 (Shutdown)
Reactor Building 198' "B" RHR HX Room	I (≥100)	6 to 200	7 to 240
Reactor Building 175' RCIC Room	I to IV (≤5 to ≥100) Depends on mode of operation	<0.2 to 1.5	<0.2 to 1.8
Reactor Building 175' North Aux Bay	I (≥100)	<0.2 to 300	<0.2 to 360
Reactor Building 175' South Aux Bay	I (≥100)	<0.2 to 25	<0.2 to 30

Table 2.10-2 Design Basis and Reported Annual Dose to Members of the Public

Design Basis Doses		
Dose Category	**Site Boundary Dose (mrem/yr)**	
	NMP2 (Current)	**NMP2 EPU**
Solidified Waste	7.92	9.50
N-16 Turbine Building (Skyshine)	6.2	8.06
Effluent	0.61	0.73
NMP1 & JA Fitzpatrick Plants	4.04	4.04
TOTAL	18.77	22.33
INCREASE		3.56
% INCREASE		19%
The EPU dose is less than the 25 mrem/year allowed by 40 CFR 190		
2004 Reported Offsite Doses		
Dose Receptor	**NMP2 (Current)**	**NMP2 EPU**
Whole Body	0.0201	0.0261
Skin	0.112	0.134
Maximum Organ	0.180	0.216
The offsite dose with EPU is expected to be less than 1 mrem/year		
The effects of EPU are based on a 30% increase for N-16 (skyshine) doses and a 20% increase for other dose contributions.		

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 NMPNS human factors evaluation was conducted to ensure that operator performance is not adversely affected as a result of system changes made to implement the proposed EPU. The NMPNS review covered changes to operator actions, human-system interfaces, and procedures and training needed for the proposed EPU. The NRC's acceptance criteria for human factors are based on GDC19, 10 CFR 50.120, 10 CFR 55, and the guidance in GL 82-33.

Technical Evaluation

2.11.1.1 Changes in Emergency and Abnormal Operating Procedures

The changes to the Emergency Operating Procedures (EOPs) reflect the change in power level, but will not be adjusted in a manner that involves a change in accident mitigation philosophy. Those variables that are also used in Severe Accident Procedures will be updated. No changes to accident mitigation philosophy will be made.

The following EOP Parameters have been identified as being impacted:

- Heat Capacity Temperature Limit - The EPU will result in additional heat being added to the suppression pool during certain accident scenarios. The Heat Capacity Temperature Limit (HCTL) curve will be revised as a result of the increase in decay heat rejected to the suppression pool. The change is not significant (~ 1 degree).

- Pressure Suppression Pressure - The Pressure Suppression Pressure Curve will be revised as a result of the increase in reactor power and in decay heat loading. The change is not significant (~0.1 psi).

- Cold Shutdown Boron Weight - The Cold Shutdown Boron Weight will be revised as a result of the increase in the equilibrium core design for EPU by ~ 18%. The Hot Shutdown Boron Weight is expected to be impacted by an equivalent amount. Upon cycle specific analysis these values will be confirmed. The technical specification value does not change as it is conservative compared to the values used in the EOPs.

The planned changes to abnormal operating procedures, called Special Operating Procedures (SOPs) at NMPNS are outlined below.

- The SOPs listed below will be revised to rescale action points associated with reactor power. These variables will be adjusted to address the new full power value but the event mitigation philosophy will not be changed. Impacted procedures include: N2-SOP-6 Feedwater Failures, N2-SOP-8, Unplanned Power Changes, N2-SOP-9, Loss of Condenser

Vacuum, N2-SOP-21, Turbine Trip, N2-SOP-23, EHC Pressure Regulator Failure, N2-SOP-29, Sudden Reduction in Core Flow, N2-SOP-34, Stuck Open Safety Relief Valve, and N2-SOP-101D, Rapid Power Reduction.

- N2-SOP-6, Feedwater Failures will be revised as part of the modification process. The Reactor Recirculation Runback Logic will be modified to initiate the runback immediately upon a feedwater pump trip.

- N2-SOP-9, Loss of Condenser Vacuum will be revised. Turbine Back Pressure Alarm Limit for EPU full power operation will be slightly less restrictive and will be incorporated in the Loss of Vacuum SOP. The alarm set point will be changed to allow operation closer to the trip set point.

- N2-SOP-10, Main Condenser Tube Rupture / Condensate High Conductivity has been revised. Steps were added to verify closure of the condensate demineralizer bypass valve installed as part of EPU modifications. This is not a new action. There are now two demineralizer bypass flow paths. This step closes a second demineralizer bypass valve which is in parallel with the existing bypass valve. The EPU bypass valve only bypasses the demineralizers. The original bypass valve bypasses the demineralizers and the condensate pre-filters. The demineralizer bypass installed for EPU is intended to be used in situations when a heater drain pump is out-of-service. Typically, demineralizer bypass will be isolated. This modification has already been installed, but will not be used until EPU is implemented.

- N2-SOP-31R, Refuel Operations Alternate Shutdown Cooling and N2-SOP-38, Loss of Spent Fuel Pool Cooling have decay heat curves, heat up rates and temperature related data sheets that will be revised to reflect the new EPU values.

2.11.1.2 Changes to Operator Actions Sensitive to Power Uprate

Most abnormal events result in automatic plant shutdown (scram). Some abnormal events result in Safety Relief Valve Actuation, Automatic Depressurization System (ADS) Actuation and/or automatic Emergency Core Cooling System (ECCS) actuation. All analyzed events result in safety-related systems, structures and components (SSCs) remaining within design limits. EPU does not change any automatic safety function. The subsequent operator action for maintaining core cooling, containment cooling and safe shutdown for plant safety remains unchanged.

There are no new credited operator actions required as a result of EPU. The analysis for EPU credits existing manual actions following the same time limits currently credited for the CLTP limit except as noted below.

Several operator actions are evaluated in the license basis for NMP2. The following actions were considered during the EPU technical evaluation:

- Combustible Gas Control in Containment assumes the following operator actions. Operators actuate containment sprays within 30 minutes after the LOCA (USAR 6.2.5.1).

Operators initiate the hydrogen recombiners in 32.6 hours. The hydrogen recombiner initiation time is reduced by 10.9 hours. CLTP initiation time is 43.5 hours. This time reduction is due to a change in the analyzed value of percent of hydrogen and oxygen concentrations in the containment at which the recombiner system is started following the accident. For CLTP, the system initiation gas concentration values are 4% hydrogen and 4.5% oxygen. (USAR 6.2.5.2.2) For EPU, the system initiation gas concentration values are 3.4% hydrogen and 3.6% oxygen. Because of this change in gas concentration value at which the system is initiated, a direct comparison of the CLTP and post EPU results is not considered relevant. Operating procedure system initiation values do not change for EPU, only the engineering evaluation is changed.

- The ATWS analysis assumes operator action in 120 seconds to initiate the Standby Liquid Control System and 1080 seconds to initiate RHS Suppression Pool Cooling. These times do not change for EPU. NMP2 plant-specific ATWS analysis takes credit for Operator action to inhibit ADS. (USAR 15.8.4) This does not change for EPU.

- Long Term DBLOCA assumes operators initiate containment cooling 30 minutes from initiation of the event. This time does not change for EPU. (USAR Table 6.2-52) (Appendix 6C Humphrey Concerns)

- For Steam Bypass, containment cooling is initiated in 20 minutes. This is a reduction of ten minutes from CLTP conditions for this event. However, the current licensing basis for the initiation of containment sprays for similar events provided in USAR 6.2.1.1.3, Design Evaluation - Assumptions for Long-Term Cooling, and in the Alternative Radiological Source Term Safety Evaluation (Reference 34) is 20 minutes. For this reason, this time reduction has no impact on actual Operator response time.

- For the AST evaluation, containment cooling spray mode of RHR is initiated in 20 minutes. This time does not change for EPU.

- For a station blackout (SBO), actions to establish reactor water level control and reactor pressure control are initiated 2 minutes into the event. This time does not change for EPU. The other 15, 30, 60 and 120 minute action times to defeat RCIC trips, reduce control room heat loads, reduce RCIC room heat load and reduce DC battery loads remain unchanged for EPU as well. The HCTL curve for SBO changes but the change is not significant (~ 1 degree).

- For the Control Room Evacuation procedures, there is no change in action time and no new required actions. The CLTP action time to initiate a blow down from the remote shutdown panel is within 10 minutes. (USAR 9B.8.2.4) Operators have demonstrated that the actions for the control room evacuation can be performed within this time frame. The analysis was revised for EPU and takes credit for a peak fuel clad temperature of less than 1500 °F instead of requiring RPV water level to remain above TAF. With this change in acceptance criteria, the maximum operator action time to initiate a depressurization from the remote shutdown panel is 13.4 minutes. The containment temperature analysis was performed with a nominal operator action time of 10 minutes. The EPU operator action time credited in the analysis will remain at 10 minutes since operators have demonstrated that this performance criterion can be satisfied. The change in acceptance criteria demonstrates adequate margin to respond to this event effectively at EPU conditions.

- RHR shutdown cooling mode is assumed to be unavailable following all transients. Operator action is assumed to establish alternate shutdown cooling 1 hour after the reactor pressure permissive of 150 psia is reached. (USAR 6A.10.2.3.2 Power Uprate Analysis (1995 Stretch Uprate)). This time does not change for EPU conditions.

- Instrument Line Break into Secondary Containment assumes operators commence a reactor cool down in 30 minutes (NMP2 Calculation ES-114). This time does not change for EPU (USAR 15.6.2).

- The time interval from loss of fuel pool cooling to fuel pool boiling (time to boil) is decreased under EPU conditions. For the full core offload case, the time to boil is decreased from approximately 5.5 hours to approximately 5.1 hours. For normal operation, SFP system is assumed to be re-aligned in 3 hours (USAR 9.1.3), and for station blackout the coping time is 4 hours. Therefore there are no changes from CLTP to EPU.

- Credit is taken for Operator action to isolate a leak 30 minutes after detection to mitigate flooding of an area in the plant (USAR Appendix 3C Section 3C.5, Compartment Flooding as a Result of Breaks or Cracks). For flooding in the Main Steam Tunnel (MST) operator action is required as soon as possible and within 100 minutes. This does not change for EPU conditions.

- The Control Room Special Filter Train system design is such that both trains of HVC special filter trains will simultaneously autostart on a LOCA signal or supply air radiation monitor trip signal. Both fans will continue to run until manually secured. One of the two operating fans is required to be shutdown within 20 minutes of initiation on a valid LOCA/Hi radiation signal. This action is required to comply with Engineering analysis of Control Room dose during accident conditions. That fan once placed in standby will autostart if the running fan trips. (N2-OP-53A) This time does not change for EPU.

- Operators ensure less contaminated outside air intake path is selected during a LOCA within 8 Hours (Increased MSIV Leakage Amendment) (N2-OP-53A) This time does not change for EPU.

- Operator action is credited to manual isolate reactor water clean up system on line break in the main steam tunnel in 1 hour to limit the release. This time does not change for EPU. (Calc S14-MST-A47-U2-37)

The Emergency Operating Procedures (EOPs) are symptom based procedures. They are used for a wide range of accidents and events that might challenge operators. These procedures, as written, do not have specific time constraints or time limits associated with their execution. They address plant symptoms/parameters independent of cause which allows for comprehensive actions to mitigate fuel damage and to prevent radioactive release. EPU conditions will result in greater decay heat loads. The actual EOP actions performed by operators are not changed. Those actions that remove decay heat will be influenced. The NMP2 simulator will be modified for EPU changes. Decay heat changes and their effects on EOP execution have been reviewed and appropriate training will be provided to the operators. USAR action times that would be executed under the guidance of EOPs are included in the evaluation of this section. The EPU analysis shows there are no significant changes to the operator action times for Emergency Operating Procedures.

The Special Operating Procedures are typically event based procedures. USAR action times for events such as station blackout and control room evacuation (Appendix R fire) have been evaluated and do not change for EPU. In other cases, the procedures are designed so that the severity of the event dictates the time available for the response, ranging from immediate operator actions to more long range response. As the previous discussion demonstrates, there are no significant changes to the operator action times for Special Operating Procedures.

2.11.1.3 Changes to Control Room Controls, Displays and Alarms

The impact to the control room instruments and controls is minimal. There are no changes to these systems/controls that will affect the operator's ability to interpret, read or respond to the information provided by the updated systems/controls. The plant process computer system operation is not affected by EPU.

The changes to the control room are prepared in accordance with the plant design change process. Under this process, a Human Factors engineering review is performed for changes associated with the NMP2 control room. The change process also requires an impact review by Operations and Training personnel. Results of these reviews are incorporated into the Engineering Change Package. Training requirements, including simulator impact and implementation requirements are also identified and tracked to completion by the design change process.

As outlined in Section 2.4.1.3, Technical Specification Instrument Setpoints for the following instrument and control systems are impacted:

- Alarm and trip values for APRMs are being revised to reflect the changes associated with the EPU rated thermal power level increase.
- For MSL High Flow Group 1 Isolations, the analytical trip value remains the same in terms of percent. Alarm and trip values for MSL High Flow Group 1 Isolations are being revised to reflect the changes associated with the EPU rated thermal power level increase and steam flow increase.
- The alarm and trip value for the Turbine First Stage Pressure Scram Bypass Permissive is being revised to reflect the changes associated with the EPU rated thermal power level increase. The absolute thermal power associated with the Turbine First Stage Pressure Scram Bypass Permissive (Pbypass) remains unchanged. The specific first stage pressure associated with this power will change.

- The Rod Worth Minimizer and the Rod Block Monitor setpoints remain at the same value in terms of percent. The absolute power values will change accordingly.

- Reactor Vessel Water Level –

  - The Low Level (L3) Scram Analytical Limit changes to account for potential instrument inaccuracies. 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, the 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 the Reactor Vessel Water Level – Low, Level 3 Scram is conservatively reduced to show that water level remains above the TAF for a loss of feedwater (LOFW) event. The Technical Specification Allowable Value and the nominal trip set point remain unchanged. It is anticipated that there will be a real effect associated with water level rate of change on a LOFW event. The dryer differential pressure and the margin to dryer skirt uncovery increases. Once the dryer skirt is uncovered, the downcomer level will lower at a higher rate as differential pressure is equalized.

  - The feedwater level control system will be modified to address the increase in feedwater pump speed to maintain margin to the level 8 trips post scram.

- MSIV Closure – MSL Low Pressure (RUN Mode) Technical Specification Allowable Value is unchanged. The pressure transmitters will be recalibrated.

- For EPU, the OPRM trip – enable region is rescaled to maintain the same absolute power/flow region boundaries. The 30% CLTP boundary is rescaled to 26% EPU thermal power limit. The flow boundary is unchanged for EPU.

The following instrument and control systems are impacted:

- Reactor feedwater flow and steam flow control room indicating meters and recorders will be modified to increase the usable range.

- Heater drain pump flow control room indicators will be modified to increase the usable range.

- The main feedwater pump ammeters will be replaced to increase the usable range.

- The reactor recirculation runback logic and the associated alarm response will be revised to initiate a runback upon feedwater pump trip and a condensate booster pump trip.

- Main Turbine 1st Stage Steam Flow recorder indication will be rescaled.

- Controls and indication will be added for 2 new feedwater pump breakers supplying the A pump from 2NPS-SWG003 and the B pump from 2NPS-SWG001.
- The Load Set and Load Meters on the EHC panel will be replaced.
- The Moisture Separator Reheater Pressure indicators will be rescaled.
- A control switch and position indication lights have been added for a condensate demineralizer bypass valve.
- A computer point will be added to calculate main generator MVA.
- An alarm will be added for Hotwell temperature.
- The electro-hydraulic control system will be recalibrated to support the high pressure turbine replacement.
- The maximum expected condenser backpressure of 5.67 in-HgA is above the current alarm setpoint of 5.4 in-HgA (USAR Section 10.4.1.3), but below the turbine trip setpoint of 7.9 in-HgA (USAR section 10.4.1.3). The alarm setpoint will be increased to provide margin for EPU operation.

Table 2.11-1 shows those instrument loops with actions impacted by EPU that automatically operate equipment if set points are met (if not previously discussed in this section). Table 2.4-2 of this submittal has a complete list of instrument changes.

The 3D Monicore Computer will be updated as part of the Cycle Specific Reload Analysis Modification.

2.11.1.4 Changes on the Safety Parameter Display System (SPDS)

The purpose of the NMP2 SPDS is to continuously display information from which plant safety status can be readily and reliably assessed. The principal function of the SPDS is to aid control room personnel during abnormal and emergency conditions in determining the safety status of the plant and in assessing whether abnormal conditions warrant corrective action by Operators to avoid a degraded core.

SPDS equipment is not being modified for the EPU. The information presented on the SPDS displays and the method of presentation remains unchanged for EPU.

The SPDS system also provides procedure based display concepts to support execution of the NMP2 EOPs. In conjunction with changes required to the EOPs for EPU operation, the following SDPS EOP curves will be revised:

- Pressure Suppression Pressure (PSP)
- Heat Capacity Temperature Limit (HCTL)

2.11.1.5 Changes to the Operator Training Program and the Control Room Simulator

The Operations Training Group develops required training on modifications installed that affect the plant operation in accordance with industry standards. The operator training is typically presented in the classroom and on the simulator, as appropriate. The training will focus on the plant modifications, procedure changes, start up and test requirements and other aspects of EPU at NMP2. Training demonstrations will be conducted highlighting the changes that impact EOPs and SOPs. As determined by the training analysis process, appropriate classroom, simulator and in plant training will be conducted prior to power escalation or as required to operate modified systems on start up.

The simulator will be modified to maintain the required fidelity in accordance with site procedures and ANSI/ANS 3.5 - 1998. The simulator changes will include hardware changes for new and modified instrumentation and controls, software updates for modeling EPU changes and re-tuning of the core physics model for cycle specific data. Simulator performance will be validated using design analysis data and start up and test data from the EPU project and implementation program.

The operator training will be scheduled to start in the 3rd quarter of 2011 and continue through the 1st quarter of 2012. Details of the training will be developed through the modification process and the training development process. Detailed schedules will be developed in accordance with NMPNS training procedures.

Conclusion

NMPNS has evaluated the changes to operator actions, human-system interfaces, procedures and training required for the proposed EPU and concludes that the effects of the proposed EPU on the available time for operator actions have been appropriately accounted for and appropriate actions have been taken to ensure that operator performance is not adversely affected by the proposed EPU. NMPNS further concludes that the requirements of GDC19, 10 CFR 50.120, and 10 CFR 55 will continue to be met following implementation of the proposed EPU.

Table 2.11-1 Instruments Loops with Actions Impacted by EPU

Device	Description	Setpoint
2CNM-FSL68A/B/C	RFP Suction Flow Low – Controls associated FWP min flow valves	Recalibrate <5,400 gpm
2CNM-PS39A/B/C	CBP 2A/2B/2C Suction Pressure Lo-Lo – Trips associated CBP	Recalibrate <38 psig
2CNM-PS42A/B/C	CBP 2A/2B/2C Suction Pressure Low – Starts CP, Blocks associated CBP start and trips pump after TD 10 sec.	Recalibrate <48 psig
2CNM-PS73A/B/C	RFP 1A/1B/1C Suction Pressure Lo-Lo - Trips associated FWP after TD 18 sec	Recalibrate <190 psig
2CNM-PS74A/B/C	RFP 1A/1B/1C Suction Pressure Low – Starts CBP, trips associated FWP after TD 45 sec. Also engages flow limiter logic if coincident with turbine trip.	Recalibrate <210 psig
2CRS-PT102	Crossaround Steam Pressure – Turbine Power input signal to PLU turbine control valve closure signal	Recalibrate < 40% CLTP
2CRS-PT103	Reheater Pressure – Input to the Moisture Separator Reheater pressure control valves	Recalibrate
2DSR-PT78A/B 2DSR- PSH78A/B	Scavenging Steam Pressure – Closes Scavenging Steam Isolation Valves on High pressure.	Recalibrate >365 psig
2ESS-PS110/115	5th point heaters extraction steam supply header pressure low – Controls building heat steam supply valves.	Recalibrate <105 psig >115 psig
2ESS-PS112/116	4th point heaters extraction steam supply header pressure low – Controls turbine sealing steam reboiler steam supply valves.	Recalibrate <50 psig >57 psig
2FWS-FT1A/1B	Feedwater Flow to Reactor – RCS Pump Downshift	Recalibrate
2FWS-PDI1A/1B	Feedwater Flow Differential Pressure for H2 flow control module – Power Flow Mode	Recalibrate
2HDL-PS50A/B/C	4th point HDP 1A/B/C Suction Press Low – Trips associated HDP and closes min flow valve	Recalibrate <5 psig
2HDL-FT35A/B/C	4th point HDP Recirculation control	Recalibrate
2HRS-PS107/108	MSR Cross around pressure (< 10% load) – Opens Moisture Separator Reheater Drain Tank Emergency Drain Valves	Recalibrate <2.0 psig/ <20 psig

Device	Description	Setpoint
2MSS-PT104/2MSS-PSH104/2MSS-PSL104	Turbine 1st stage pressure 85% load, 30% load – Power Input Permissive to Open the Condensate Demineralize/Prefilter Bypass Valve and the Feedwater Heater String Bypass Valve (85% load) and the Main Steam Auxiliary Drain Control permissive (30% load)	Recalibrate >603 psia <213 psia
2MSS-PT103/2MSS-PSL103/2MSS-PSLL103	Turb 1st stage pressure 15% load, 5% load – Main Steam Auxiliary Drain Control permissive (15% & 5% load)	Recalibrate <106/<35 psia
2MSS-PT22A/B	Reheater Pressure – Input to the Moisture Separator Reheater pressure control valves	Replace
2MSS-PT143/144	Main Steam Inlet Header Pressure- PT143/144 provide signal (secondary/primary) to EHC pressure control unit (high value gate) for TCV AND BYP VLV Control.	Recalibrate
2MSS-PT101/2MSS-PSL101	Main Steam Inlet Header Pressure – Auxiliary Steam header pressure control for auxiliary steam supply to the main turbine sealing steam clean steam reboilers.	Replace < 200 psig

2.12 Power Ascension and Testing Plan

2.12.1 Approach to EPU Power Level and Test Plan

Regulatory Evaluation

The purpose of EPU test program is to demonstrate that SSCs will perform satisfactorily in service at the proposed EPU power level. The test program also provides additional assurance that the plant will continue to operate in accordance with design criteria at EPU conditions. The NMPNS review included an evaluation of: (1) plans for the initial approach to the proposed maximum licensed thermal power level, including verification of adequate plant performance, (2) transient testing necessary to demonstrate that plant equipment will perform satisfactorily at the proposed increased maximum licensed thermal power level, and (3) the test program's conformance with applicable regulations. The NRC's acceptance criteria for the proposed EPU test program are based on 10 CFR Part 50, Appendix B, Criterion XI, which requires establishment of a test program to demonstrate that SSCs will perform satisfactorily in service.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of Constant Pressure Power Uprates. Section 10.4 of the CLTR addresses the testing required for the initial power ascension following the implementation of CPPU. The results of this evaluation are described below.

Testing is required for the initial power ascension following the implementation of EPU. The topics addressed in this section are:

[[

		]]

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 done in accordance with the Technical Specification Surveillance Requirements on instrumentation that is re-calibrated for EPU conditions. Overlap between the IRM and APRM will be assured.

Testing will be done to confirm the power level near the turbine first-stage scram bypass setpoint.

Steady-state data will be taken at points from 90% up to 100% of the CLTP RTP, so that system performance parameters can be projected for EPU power before the CLTP RTP is exceeded.

EPU power increases above the 100% CLTP RTP 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 and pressure controls. 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.

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. [[

]] Vibrational testing is addressed in Attachment 10 to the EPU license amendment request.

The EPU testing program at NMP2, which is based on the specific testing required for the NMP2 initial EPU power ascension, supplemented by normal Technical Specification testing, is confirmed to be consistent with the generic description provided in the CLTR. Attachment 7 to the EPU license amendment request contains details of the testing program.

NMP2 does [[ ]] large transient testing as part of EPU implementation. The justification for [[ ]] large transient testing is provided as Attachment 7 to the EPU license amendment request.

Further, [[

]] In addition, the limiting transient analyses are included as part of the reload licensing analysis.

Conclusion

NMPNS has reviewed 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, and the test program's conformance with applicable regulations. NMPNS concludes that 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. Further, NMPNS finds that there is reasonable assurance that the EPU testing program satisfies the requirements of 10 CFR Part 50, Appendix B, Criterion XI. Therefore, NMPNS finds the proposed EPU test program acceptable.

2.13 Risk Evaluation

2.13.1 Risk Evaluation of EPU

Regulatory Evaluation

A risk evaluation has been conducted 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 plant to meet the deterministic requirements and regulations. This evaluation covered the impact of the proposed EPU on core damage frequency (CDF) and large early release frequency (LERF) for the plant due to changes in the risks associated with internal events, external events, and shutdown operations. In addition, the evaluation covered the quality of the risk analyses used to support the application for the proposed EPU. This included a review of the actions to address issues or weaknesses that have been raised in previous NRC staff reviews of the individual plant examinations (IPE), individual plant examination of external events (IPEEE), or by an industry peer review. The NRC's risk acceptability guidelines are contained in RG 1.174 (Reference 80).

Technical Evaluation

The NMP2 probabilistic risk assessment (PRA) covers both internal and external events. This plant specific PRA is used to compare pre-EPU and post-EPU plant design and operation. A combination of quantitative and qualitative methods is used to assess the potential risk impacts of EPU. The following subsections provide the risk evaluation.

2.13.1.1 Probabilistic Risk Assessment Quality

The NMP2 PRA was originally submitted to the NRC in 1992 as the IPE submittal in response to GL 88-20 (Reference 81), which addressed internal events and internal flooding events. The NRC issued the Staff Evaluation in 1994. The NRC concluded that NMP2's process was capable of identifying the most likely severe accidents and no significant impacts on the PRA were identified. Although this overall conclusion was positive, the NRC contractor report attached to the SE did identify potential weaknesses that were considered in PRA updates and are addressed in the PRA documentation.

A one-week extensive on-site review of the PRA was conducted as part of the BWROG Peer Review Certification Program in 1997. The weaknesses identified from this review were associated with improving guidance, documentation, the cleanup of crediting procedures not in place, and pre-initiator human reliability analysis (HRA). Subsequent to the BWROG review, the NMP2 PRA model was updated to address comments generated from these reviews. Overall, the certification provided high technical marks on the PRA and there were no comments that significantly impacted PRA results.

The NMP2 PRA is currently in the process of a major update to conform to the ASME PRA Standard and RG 1.200 (Reference 82). Thus, the current PRA used for this EPU application predates the present PRA update effort to meet ASME Capability Category II PRA quality by 2009. However, the NMP2 internal events PRA has been reviewed by the NRC (IPE) and peer reviewed by the BWROG peer review certification process (the predecessor to the NEI sponsored PRA Peer Review, NEI 00-02 (Reference 83)). The certification process was developed to establish a method of assuring the technical quality of the PRA for the spectrum of its potential uses. The certification team conducted an intensive one-week effort to evaluate the scope, comprehensiveness, completeness, and fidelity of the NMP2 PRA, as well as the process to be utilized in both performing PRA applications and utilizing PRA results. This interaction focused on the understanding and procedures associated with such applications to verify that the assumptions, limitations, and results of the PRA are communicated and understood. The results of the certification review were used as input into the NMP2 PRA update process, as well as programmatic changes. The NMP2 grades from the evaluation were very good (supporting Grade 3 applications) and there were no importance level A elements identified. This is important and necessary to address: 1) the technical adequacy of the PRA, 2) the quality of the PRA and 3) the quality of the PRA update process. However, this extensive review identified numerous insights relative to improving the quality of the NMP2 PRA. These improvements have been addressed in the present PRA. The peer review findings are summarized in Tables 2.13-1 and 2.13-2.

The IPEEE, which addressed external events in response to Supplement 4 of GL 88-20 (Reference 81), was submitted to the NRC in 1995. The NRC issued the Staff Evaluation in 1998. Again, the overall conclusions were positive but the NRC contractor report attached did identify potential weaknesses that were considered in PRA updates and are addressed in the PRA documentation. In addition, the present PRA model is an integrated model containing both internal and external events.

Based on the above, the NMP2 PRA is of sufficient quality and scope for this application. The modeling is detailed; including a comprehensive set of initiating events (transients, LOCAs, and support system failures) including internal flood, fires and seismic, system modeling, human reliability analysis and common cause evaluations.

2.13.1.2 Internal Events

The risk impacts from internal events resulting from EPU have been assessed by reviewing the changes in plant design and operations resulting from EPU. The changes have been mapped to appropriate elements and the PRA has been modified as needed to estimate the risk impact (CDF and LERF) of the post-EPU plant. As a result of EPU, CDF was calculated to increase by 5.1% (7.4E-7/year increase) and LERF was estimated to increase by 2.1% (1.2E-8/year increase). If the sensitivity analyses risk estimates are included, CDF was calculated to increase by 6.5% (9.3E-7/year increase) and LERF was estimated to increase by 5% (2.8E-8/year increase).

These assessments included evaluations of EPU impacts on the following areas, as described in subsections below:

- Initiating event frequency

- Component reliability
- Success criteria
- Operator response

2.13.1.2.1 Internal Initiating Event Frequencies

The evaluation of EPU changes indicates no new initiating event or increased frequencies of existing initiators is anticipated. A partial loss of feedwater initiating event (PLOF) is considered as a new initiating event in the ongoing PRA update and is therefore considered in this evaluation as described below for transients. A number of changes to the balance of plant (BOP) equipment are being made to restore or improve design and operating margins as described below.

Modification Review

Modifications required to support EPU were reviewed and determined not to result in new initiating events or accidents. In addition, no measurable decrease in equipment reliability or increase in the frequency of plant challenges was found. This is based on a review of plant modifications and engineering judgment based on knowledge of the PRA models (see Table 2-13-3). There are no plans to operate equipment beyond design ratings. In a few cases, design ratings were analyzed and increased to ensure margins in design are not significantly impacted. Although experience indicates that major changes to equipment can increase equipment unavailability in the short-term due to break-in ("bathtub curve"), this impact cannot be easily quantified and steady state conditions are expected to be equivalent or better than current plant performance. No changes to the PRA model were identified.

The following provides a summary of key modifications required to support EPU:

- Replace high pressure turbine
- Feedwater pump impeller and speed changer
- Replace heater drain pump
- Replace low pressure turbine cross around piping relief valve
- Install additional HVAC coolers in the condensate and condensate booster pump areas
- Replace atmospheric exhaust hood diaphragms
- Replace feedwater motor cable
- Feedwater system modifications will modify pump protection schemes to preclude multiple pumps from tripping simultaneously in response to single or common initiating event. These changes will minimize margin reductions for condensate system pump trips

and maintain margin for feedwater system pump trips for a partial loss of feedwater event and preclude a loss of feedwater event.

Procedure Changes

Section 2.11 describes changes to emergency and abnormal operating procedures. These changes have a minor effect; no changes to the PRA are identified. Adjustments to procedures including operating procedures and emergency procedures will be made in accordance with EPU operating conditions and updated analyses. Although symptom-based procedures generally do not require change, certain parameter thresholds and curves are dependent on power and decay heat levels. Those that play a role in the PRA and may require adjustment for EPU include the following:

- Boron Injection Initiation Temperature (BIIT) [no change to the 110°F initiation]
- Heat Capacity Temperature Limit (HCTL) [minor change]
- Primary Containment Pressure Limit (PCPL) [no change]
- Pressure Suppression Pressure Limit (PSP) [minor change]

Such adjustments will not change accident mitigation modeling in the PRA and will be minor such that timing and human reliability are not affected (see Success Criteria and Operator Response Sections 2.13.1.2.3 and 2.13.1.2.4, respectively). No major changes in procedures have been identified that impact the PRA.

The minimum RPV level following a control room evacuation is impacted by EPU because increased decay heat would reduce operator response time available. N2-SOP-78 "Control Room Evacuation" requires operator action within 9 minutes of an event to maintain water level above top of active fuel (TAF). For the control room evacuation procedures there is no change in the actions time and no new required actions. The CLTP action time to initiate a blow down from the remote shutdown panel is within 10 minutes (USAR 9B.8.2.4). Operators have demonstrated that the actions for the control room evacuation can be performed within this time frame. The analysis was revised for EPU and takes credit for a peak fuel clad temperature of <1500°F. instead of requiring RPV water level to remain above top of active fuel (TAF). With this change in acceptance criteria, the maximum operator action time to initiate depressurization from the remote shutdown panel is 13.4 minutes. The containment temperature analysis was performed with a nominal operator action time of 10 minutes. Although the post-EPU conditions lead to a RPV water level reaching TAF in a shorter time frame, the impact on fuel temperature is small and does not approach the 1500°F limit. This is within the PRA success criteria, is not a significant impact, and is evaluated in the HRA (operator action ZHRA3).

Set Point Changes

Key actuation signal set points such as RPV dome pressure, RPV level and SRV pressure are not expected to change. No changes to the PRA model as a result of set point changes have been identified.

As shown in Table 2.13-3, certain instrument replacement or modification may be necessary. Many instruments will require recalibration and or set point changes; a few will be replaced in the main steam systems and minimal tuning of control loops is expected during startup. None of these changes impact the PRA.

Plant Operating Conditions

The key plant operation modifications to be made in support of EPU are:

- Increased reactor thermal power from 3467 MWt (current licensed thermal power, CLTP) to 3988 MWt (EPU). The original licensed thermal power (OLTP) was 3323 MWt.
- Corresponding change in feedwater flow (increase <20%)
- Operation with ARTS/MELLLA

RPV operating pressure, temperatures and core flows are essentially unchanged. The feedwater flow will increase to support EPU. It is anticipated that the long-term initiating event frequency is unchanged and no change is being made to the PRA as a result of EPU.

The MELLLA refers to a region of the power/flow map where the plant is licensed to operate at a higher rod line without increasing recirculation flow. The MELLLA curve has been implemented to allow an acceptable core flow range at higher power levels associated with EPU. This operation does not significantly impact transient initiating event frequencies. Although no quantifiable changes in initiating frequency can be identified at this time, the frequency of a PLOF reactor scram on level was identified as the most likely impact and this initiating event is not presently included in the PRA model. As a result, a sensitivity case regarding this event is quantified in the Risk Impact Calculations Section.

Transients

No changes are being made to the number of normally operating pumps and equipment in the balance of plant, thus no direct impact on transient initiating frequencies is anticipated. However, the following were identified as potential changes to be evaluated:

- Modifications to the secondary plant (e.g., feedwater and recirculation runback) are being implemented to support EPU. This is expected to improve or restore reliability and flexibility with feedwater operation and have no measurable effect on the initiating event frequency. The present (pre-EPU) operating configuration is 3/3/2 (3 condensate, 3 booster and 2 feed pumps), which will not change post-EPU. However, post-EPU the ability to operate at full power in a 2/2/2 configuration will be lost. The primary impact on the PRA is the potential increase in initiating events associated with the systems, which has been addressed by modifications described below.

- NMP2 feedwater system design provides the ability to respond to the loss of a single feedwater pump (PLOF). Recirculation system flow automatically runs back to reduce reactor power to within the capacity of a single feedwater pump without a low level scram occurring. Since the existing PRA does not model this initiating event, and there is a concern that PLOF could result in a reactor scram or at least the probability of such an event may be increased by EPU, the potential risk impact of increasing PLOF is evaluated with the NMP2 PRA (see the Risk Impact Calculations Section) as a sensitivity case.

- Also, the loss of a condensate booster pump could cause cavitation at the main feed pump suction. As a result, modifications will ensure that a condensate booster pump trip will initiate recirculation system runback with an increased runback rate of 9% per second to prevent a low level scram and loss of all feedwater (the likelihood of scram is assumed unchanged by EPU). This is also evaluated in the Risk Impact Calculations Section.

- The main condenser operating back-pressure margin (loss of vacuum trip) has been decreased, but the margin was determined to be acceptable. Any minor potential increase in this initiating event cannot be identified at this time. A sensitivity case is described in the Risk Impact Calculations Section with regard to potential risk impact if the loss of condenser (LOC) initiating event increased.

- The main generator step up (GSU) transformers were found to have reduced margins. Although no increase in the turbine trip initiating event is anticipated, a sensitivity case is described in the Risk Impact Calculations Section with regard to potential risk impact if the turbine trip (TT) initiating event increased.

Minor changes in the electrical area are required such as feedwater cable, relay setting, etc. Protective relay settings and increased bus duct cooling evaluations ensure that margins are maintained in the electrical systems. No significant change to electrical reliability was identified with EPU. No significant changes to support systems are planned in support of EPU, thus no impact on support system initiating event frequencies are postulated.

<u>LOCAs</u>

No changes to RPV operating pressure, inspection frequencies, or primary water chemistry are planned in support of EPU. Thus, no impact on LOCA frequencies can be postulated. Although the reactor coolant pressure boundary is unlikely to be affected and PRA LOCA frequencies will be unchanged, the sensitivity of assuming an increase in LOCA frequencies is evaluated in the Risk Impact Calculations Section.

SORV

Although the potential for an inadvertent open SRV (IORV) as an initiating event is not increased (reactor pressure and SRV set points are unchanged), the potential for a transient induced stuck open SRV (SORV) could increase since the number of post-trip SRV challenges during isolation events is expected to increase. Thus, an increased frequency of SORV is included in the Risk Impact Calculations Section.

Internal Floods

The methodology used in calculating initiating event frequencies for internal floods is based on the number of pipe segments and/or lengths of pipe. Major piping changes are not required for EPU except in the secondary plant in the Turbine Building, and these changes will restore and improve reliability. The addition of four unit coolers in the Turbine Building will add limited small diameter service water piping, which is insignificant in comparison to existing large diameter piping that already exists in the Turbine Building. Thus, the internal flooding frequencies remain the same.

Based on Section 2.1.6, the flow accelerated corrosion rates are not expected to be significant and both the flow accelerated corrosion and the temporary vibration monitoring programs post-EPU reduce any potential increased risk. The feedwater increased flow rate was considered as a potential impact, but feedwater breaks outside containment have been screened from the PRA as initiating events because of low risk; a minor increase in their frequency would not change this conclusion.

2.13.1.2.2 Component Reliability

EPU will not significantly impact the reliability of equipment. Hardware changes in support of EPU may be characterized as replacement with enhanced like components or upgrades to existing components. No significant effect on the long-term average failure rates (initiating events and equipment reliability) due to replacement/modification of components is anticipated. If any degradation were to occur as a result of EPU, existing plant monitoring programs would address any such issues.

No planned operational modifications as part of EPU involve operating equipment beyond design ratings.

2.13.1.2.3 Success Criteria

There are margins associated with systems required to support success criteria in the NMP2 PRA, which has been confirmed by EPU calculations and MAAP runs for the PRA. In addition, EPU does not change the plant configuration or operation in a manner that results in new accident sequences or changes in existing accident sequence modeling. The only impact identified is the timing of accident sequence progression due to increased power levels and decay heat post-trip (see the Operator Actions Section regarding potential impact on operator reliability) and the possible increase in SORV probability. The following, and Table 2.13-4, summarizes key MAAP scenarios evaluated to investigate potential impacts of EPU on PRA success criteria and timing (human reliability):

- Given a turbine trip with feedwater and the main condenser available, transient analysis predicts that SRVs will open although plant-specific experience post-EPU will have to confirm this result. Transient analysis for pre-EPU conditions also indicated minor SRV lifting, which is not consistent with actual plant trips. Even if it is determined that SRV opening will occur, it would be a minor impact and the sensitivity of a SORV is included for risk evaluation in the Risk Impact Calculations Section.

- The number of SRV cycles (U2CNT1) given all MSIVs close as an initiating event resulted in an increased number of SRV challenges. Specifically, MAAP analysis determined that 94 SRV cycles were expected in the first hour of a MSIV closure type scenario under pre-EPU conditions and 104 cycles were expected in the first hour under post-EPU conditions. The sensitivity of a transient induced stuck open SRV (SORV) is included in the Risk Impact Calculations Section.

- Feedwater success for the PRA required 24 hour mission time with CST makeup to the condenser hotwell (U2CNT1) was checked; this success criteria is not affected by EPU.

- The timing for key containment conditions that could result in RPV blow down as well as containment venting or even containment failure has changed (U2CNT1), but the times are so long that this does not significantly impact human reliability or success criteria.

- The timing to blow down at -39 inches (or lower) per EOP-RPV given loss of high pressure injection (U2LOF2) is an important operator action. This scenario assumes all injection is lost at time zero, which is conservative. The change in time is assessed in the Operator Actions Section with respect to the potential impact on operator reliability (operator action basic event MSSZODMSSOP1OO01).

- Given blow down success at -39 inches, one LPCI pump with 2 SRVS is a success (U2LOF2). Service water crosstie to RHR (U2LOF3) is not successful for both pre-EPU and post-EPU calculations (core damage occurs, but core is retained in vessel for both cases). However, the flow rates used are judged to be conservative and this alignment is believed to be a success in the PRA (an open item that is being addressed in the PRA update). The Service water crosstie to RHR is considered a success for long-term injection in both pre-EPU and post-EPU conditions.

- During a station blackout, the timing to containment conditions that could impact the availability of RCIC is not significantly affected (U2LOF4). The first condition to affect RCIC is the heat capacity temperature limit (HCTL), which changes from more than 10 hours to more than 8 hours. Since AC power recovery is required within 8 hours in the SBO model, this has no impact on the model or risk calculations.

- During a station blackout, the shortest time to recover AC power (LOOP recovery) is the time to core damage without RCIC success (U2LOF5) which is between 40 and 50 minutes. A difference of 5 to 10 minutes has no measurable impact on AC power recovery (this is within the uncertainty of AC power recovery).

- The number of SRVs challenged during an ATWS could not be reliably determined, but GEH analysis indicates margin even with 2 SRVs assumed unavailable. Thus, the probability of over pressure due to failure of several SRVs is still dominated by common cause failure of the SRVs, which is unchanged in the PRA.

- The timing to key conditions that impact operator actions for ATWS (U2AT1, U2AT2 and U2AT3) does not significantly affect modeling. At NMP2, standby liquid control (SLC) is automatically actuated; therefore a critically important operator action at most plants is not required in the NMP2 model. Other automatic actions such as recirculation pump trip (RPT) and feedwater runback ensure that initial power is reduced quickly while SLC is being injected. For other actions, which are not automatic such as aligning suppression pool cooling (pool temperature exceeds 90°F) and ADS inhibit (Level 1), the timing is immediate for both pre-EPU and post-EPU and the influence of EPU is judged minor.

RPV Inventory Control

Feedwater, RCIC, HPCS, LPCS, and LPCI have adequate margins such that success criteria will remain the same post-EPU. CRD alone cannot provide inventory control immediately after a plant trip and as a result is not credited for early injection success. This will remain the case post EPU. The same is true for the firewater crosstie to RHR.

RPV Pressure Control

Operating pressure will not be increased as a result of EPU, but pressure following a plant trip with isolation of the main condenser or an ATWS will increase slightly. However, the number of SRVs expected to lift is not expected to change significantly and the impact on PRA success criteria and the resulting failure probability is insignificant because the failure probability is dominated by common cause failure probability of several SRVs to open. GEH analysis with 2 SRVs out of service (unavailable) for ATWS indicates margins.

The number of SRVs required to successfully depressurize the reactor when low pressure injection systems are required is not changed as indicated in MAAP cases U2LOF2 and U2AT3.

Containment Heat Removal

The timing of these scenarios is long term such that the difference in timing will not impact human reliability calculations (MAAP Case U2CNT1). Use of main condenser, RHR and containment venting in the PRA is unchanged by EPU.

ATWS

The timing of human response is relatively quick as described above and evaluated in the HRA descriptions; the difference in timing is not judged to significantly influence human reliability as described in the Operator Actions Section.

LOOP Recovery

The important influences on ability and timing to recover LOOP such as during a station blackout include battery life, availability of injection sources, etc. The time to boil-off reactor inventory given failure of these coping systems in the long-term is a secondary effect that has no measurable impact. The most important timing would be the case where RCIC and HPCS fail at time zero (U2LOF5), but the PRA model does not credit firewater makeup if RCIC does not work for at least 2 hours, thus any change in timing will not have a quantitative impact on PRA results for this sequence.

Level 2 and 3 Models

The Level 2 PRA framework, functional fault trees, and Level 2 basic event failure probabilities remain unchanged in the transition from pre-EPU to EPU.

There are small changes in the accident progression timing. The timing to RPV breach will change slightly due to increased decay heat. For example, a total loss of injection scenario (U2LOF1) indicates that the timing to vessel breach changes from approximately 3.1 hours to 2.5 hours. This is insignificant in the Level 2 model because in either case, the timing as it affects the determination of LERF remains the same. Other modeling details are much more important to establishing LERF conditions. The time to containment venting would also be slightly shorter, but these are long-term sequences that are not considered to be "early" releases.

Although radiological source terms would be higher, the definition of LERF in the NMP2 PRA is based on fractional releases which do not change. The NMP2 PRA does not include a Level 3 model and this is not required to be evaluated for EPU.

2.13.1.2.4 Operator Actions

The NMP2 PRA, like other plants, is dependent on operator actions to successfully mitigate accidents. The success of these actions depends on performance shaping factors. The performance shaping factor that is principally influenced by EPU is the time available to detect, diagnose and perform required actions. The higher power level and resulting decay heat levels post-trip reduces the time available for some operator actions in the PRA. To quantify the potential effect, calculations using the MAAP computer code were performed.

Operator actions in the NMP2 PRA were reviewed to determine which may be affected by reduced timing (Table 2.13-5). The following provides a summary of key actions:

- RPV blow down when high pressure injection systems are unavailable is important as the time to reach the top of active fuel and blow down is faster. The following basic events apply and are identified as risk important in Table 2.13-6:

  - MSSZODMSSOP1OO01 (Emergency Depressurize – Transient/SLOCA)
  - MSSZODMSSOP1OO02 (Emergency Depressurize – MLOCA)
  - HRAZHRBOPSRICS02 (Emergency Depressurization – Control Room Fire)
  - HRAZHRBOPSRICF03 (Emergency Depressurization at RSP– Control Room Fire)

  There is also an operator action in the ISLOCA model (basic event MSSZODMSSOP1OO05), which is not a risk significant action. Similarly, an operator action in the ATWS model is not risk significant.

- Operator actions associated with preventing a plant trip were considered potentially important, but judged not to be affected by EPU as summarized below and in Table 2.13-6:

  - IERZIECBFLOODX01 (Recover From Flood in Control Building)
  - IERZIEKABXRECV02 (Prevent Trip After Partial 115 kV Loss)

- The PRA models recovery from partial loss of offsite power initiating events (basic event ENSZKROPERSWAP01), which was considered important because the limiting time for this action would be the case where high pressure injection fails similar to RPV blow down above. This action was also identified as risk important in Table 2.13-6.

- Service water control during transients that affect the system was considered and is identified in Table 2.13-6 (basic event SWPZSAHHSWSOOO01), but this human error probability (HEP) was treated as an immediate response (alarm response) and not considered highly sensitive to any reduced operator response time (e.g., TBCCW heat up due to higher generator heat loads) since the service water, RBCLC and TBCCW systems have margins.

- ATWS response is considered potentially important and was evaluated as summarized below:

  - Time available for the operators to disable the RPV Level 1 MSIV isolation signal is short, and as a result this operator action is set to guaranteed failure in the PRA. This would still be the case for EPU.
  - The time available to restore feedwater after automatic runback (basic event FWSZFWXXXXXXXX01) already has a high HEP (0.5), but is included in the EPU change in risk assessment (see Table 2.13-7) because of its importance.

- Blow down to allow low pressure injection (basic event ZODA=0.16) would be affected by the timing, but this is low importance and is not considered in the risk impact assessment.

- ADS inhibit is an immediate response and considered important to ATWS response (basic events MSSZAIOPACT1OO01 and MSSZAIOPACT2OO02). These are relatively reliable actions, are not judged to be significantly impacted by EPU and are not risk significant. The timing is also essentially the same for pre-EPU and post-EPU.

- Controlling level and power (basic events RPSZCHXXXXXXXX01 and RPSZCHXXXXXXXX02) is important to ATWS response. The second operator action is set to guaranteed failure in the PRA and is not considered further. The first operator action is included in the EPU risk impact assessment.

Those operator actions in the NMP2 PRA with a CDF Fussel-Vesely (FV) > 0.01 or risk achievement worth (RAW) > 2 are identified and evaluated in Table 2.13-6 to determine those that could potentially be affected by EPU. Several of these were also identified as having a LERF FV > 0.01 or RAW > 2. At the bottom of the list, two additional actions were identified as having LERF FV > 0.01. Those with a potential to be impacted based on decreased timing as a result of EPU are highlighted in the table. These operator actions were evaluated further to assess the potential impact of EPU on human reliability. Table 2.13-7 summarizes this evaluation.

MAAP thermal hydraulic calculations were performed as summarized in Table 2.13-4 to estimate the changes in timing for operator actions. These timings were used as input to the HRA analysis and the results are summarized in Table 2.13-7.

No significant changes to the control room or the plant are being made that would impact the PRA human reliability analysis. Displays and scales could be affected, but these will not impact the quantification of any human reliability analysis in the PRA.

2.13.1.3 External Events

Although the frequency of external events is not affected by EPU, the following sections review potential impacts on their mitigation (seismic, fire and other external events).

2.13.1.3.1 Seismic Events

The frequency of earthquakes is not dependent on reactor power or operation. Thus, no impact seismic initiating event frequency is postulated. Seismic initiating events from the IPEEE are incorporated into the NMP2 PRA model, thus any impacts on these accident sequences are addressed quantitatively in the Risk Impact Calculations Section with regard to the potential risk increase due to decreases in human reliability. The PRA models a separate operator action to blow down RPV given a seismic event and failure of high pressure injection (ZOD05), but this action is not risk significant. As a result, the risk impact of EPU is minor and not evaluated.

2.13.1.3.2 Fires

The frequency of fires is not dependent on reactor power or operation. Thus, no impact on fire initiating event frequency is postulated. Fire initiating events from the IPEEE are incorporated into the NMP2 PRA model, thus any impacts on these accident sequences are addressed quantitatively in the Risk Impact Calculations Section with regard to the potential risk increase due to decreases in human reliability.

2.13.1.3.3 Other External Events

In addition to seismic and internal fires, the IPEEE analyzed other external hazards such as high winds, tornadoes, transportation and nearby facility events. The analysis of these events was based on a review to the NRC Standard Review Plan requirements and these were screened due to low risk. Since EPU has no impact on compliance with these requirements, the risk from these events is acceptable.

2.13.1.4 Shutdown Operations

The effect of EPU on shutdown risk is similar to the effect on the at-power PRA. Based on insights from the at-power PRA impact assessment, the areas of review appropriate for shutdown risk are initiating events, success criteria and human reliability analysis. Based on a review of the potential impacts on these areas, EPU has an insignificant impact on shutdown risk. The outage risk management process at NMP2 describes the risk management process, defense-in-depth and protective functions versus shutdown states, which are unchanged by EPU.

The following qualitative discussion applies to shutdown conditions during Hot Shutdown (Mode 3), Cold Shutdown (Mode 4), and Refueling (Mode 5). The EPU risk impact during transitions such as at-power (Mode 1) to Hot Shutdown and Startup (Mode 2) to at-power is judged to be subsumed by the at-power PRA. This is a consistent industry assumption based on limited time in these transitions, similar success criteria, reduced power levels and RG 1.174 (Reference 80) which states that not all aspects of risk need to be addressed for every application.

Although an integrated shutdown risk model is not included within the NMP2 PRA model, shutdown risk modeling has been performed in the past to support outage risk management. These evaluations were reviewed and considered with regard to the qualitative discussions below.

<u>Initiating Events</u>

No new initiating events or increased potential for initiating events during shutdown have been identified due to EPU. Shutdown initiating events include ways to disrupt decay heat removal including loss of RCS inventory (inadvertent drain down, LOCA) and the decay heat removal systems (shutdown cooling mode of RHR or its support systems). None of these were found to be affected by EPU.

Success Criteria

No impact on previous identified success criteria during shutdown has been identified due to EPU. However, the following minor impacts were identified:

- The increased decay heat levels associated with EPU delays the time after shutdown when alternate heat removal systems can be used to remove decay heat. However, this delay has minimal impact on success.
- The boil-down time is longer after the reactor is shutdown in comparison to the at-power PRA. The time to core damage at CLTP is approximately 9.6 hours versus 8.3 hours for EPU at one day into the outage. The time exceeds 24 hours after the refueling cavity is flooded.
- Because of low decay heat in comparison to at-power, the EPU has a minor impact on inventory makeup requirements during a loss of decay heat removal event. The only pumping system that could be sensitive to decay heat increases is CRD pump inventory makeup via the CRD seals. However, this pump does not presently have sufficient makeup by itself until several hours after shutdown. The EPU would extend this time slightly with minimal impact on risk.
- The impact on other success criteria such as suppression pool cooling capacity, blow down loads, RPV over pressure, etc. is negligible due to lower pressures and decay heat levels during shutdown.

Human Reliability

The primary impact of EPU on risk during shutdown is the decrease in allowable operator action times in responding to loss of decay heat removal events. As described previously (success criteria) the reduction in times to core damage is on the order of 14% (9.6 hours to 8.3 hours) 1 day after shutdown. This change would not result in a significant increase in human error probabilities for most operator actions using current human reliability analysis techniques. Such small changes in time to already lengthy allowable operator response times results in negligible changes in calculated human error probabilities.

2.13.1.5 Risk Impact Calculations

The pre-EPU CDF and LERF from the NMP2 PRA are as follows:

U2BASER3 CDF = 1.9E-5/year

U2BASER3 LERF = 7.2E-7/year

U2BASER3 CDF with Pre-EPU Updated HEP = 1.44E-5/year (used for delta CDF)

U2BASER3 LERF with Pre-EPU Updated HEP = 5.67E-7/year (used for delta LERF)

Since the NMP2 PRA is undergoing a major update, which includes a complete update to the HRA analysis, the baseline CDF and LERF were calculated with the updated HEPs in Table 2.13-7. The resulting CDF and LERF provided above were then used as the baseline for calculating the HRA delta CDF and LERF below. This was required because it would not be productive to attempt estimating post EPU HEPs using outdated analyses and these HEPs were being re-evaluated in support of the ongoing PRA update.

The following summarizes the risk impact assessment results:

Impact	Δ CDF	Δ LERF	Comment
Human Reliability (HRA)	7.3E-7	1.2E-8	New Risk
Partial Loss of Feedwater	4.7E-8	4.7E-9	Sensitivity
Loss of Condenser	1.4E-8	1.3E-9	Sensitivity
Turbine Trip	1.3E-7	9.7E-9	Sensitivity
SORV	8.4E-9	1.0E-10	New Risk
LOCA	2.6E-9	8.3E-11	Sensitivity
Total New Risk	**7.4E-7**	**1.2E-8**	**New Risk**
Total with Sensitivity	**9.3E-7**	**2.8E-8**	**New Risk + Sensitivity**

The above increase in risk meets the acceptance guidelines described in RG 1.174 (Reference 80), which states that an increase in CDF in the range of 1E-6 to 1E-5 will be considered when it can be reasonably shown that the total CDF is less than 1E-4. Since total CDF is less than 1E-4 and the quantified new risk is less than 1E-6, the quantified impact of EPU is acceptable.

The NMP2 PRA is currently undergoing a major PRA update in 2008 to comply with ASME PRA Standard and RG 1.200 (Reference 82). Thus, as described previously, the HRA portion of the risk assessment included updated HEPs for both pre-EPU and post-EPU.

Operator Reliability

The impact of increased decay heat and reduced time available for operator actions was evaluated. Because the human error probabilities in the current PRA, based on the original IPE, are outdated and are being updated, the impact on CDF and LERF associated with reduced times for these actions was calculated on a consistent basis as follows:

- The current HEPs as described in Table 2.13-7 were updated to current methodology
- The HEPs for EPU were estimated using the same consistent technique (Table 2.13-7)
- The change in CDF and LERF for these two cases was calculated

The resulting change in CDF and LERF are summarized in the above table.

Partial Loss of Feedwater (PLOF)

PLOF is presently not included as an initiating event in the NMP2 PRA because these events are unlikely and total loss of feedwater, which is modeled, subsumes this event. Even though EPU evaluations concluded that the likelihood of scram is not increased, it was concluded that this initiating event should be added to the PRA and used to perform sensitivity analyses on this initiating event. Accordingly, this initiating event was first added to the model to provide a baseline for estimating the increase in risk associated with a potential increased probability of reactor scram given that a single feedwater pump fails during normal power operation. This analysis shows that the additional risk in the PRA is small.

In addition, the loss of a condensate booster pump (CBP) was evaluated with regard to its impact on loss of feedwater initiating events in the PRA. A review of plant experience was performed to determine whether there have been any CBP failures during power operation. Applicable events would provide a basis for estimating the frequency of initiating operator and plant response to runback recirculation and reducing power to prevent a reactor scram. The results of the review indicate that no CBP losses occurred at power during the past 10 years. Thus, it can be concluded that the frequency of CBP loss is <0.1/year and loss of a main feed pump is more likely. Thus, it was concluded that the conservative sensitivity includes the contribution from CBP loss. The following failures have to occur in order for CBP loss to cause a scram:

- Recirculation runback logic will ensure that trip of a Condensate Booster Pump will automatically initiate a recirculation flow control valve runback at high power to prevent a scram.
- Staggered feedwater pump suction trips will ensure that Reactor Feed Pumps will not trip simultaneously. A trip of a feedwater pump will automatically initiate recirculation flow control valve runback to prevent a scram (the likelihood of scram is assumed to not be changed by EPU)

The above failures, in combination with a low initiating frequency (<0.1/year) for loss of a CBP, ensure that both partial and total loss of feedwater initiating events are unlikely and restored to pre-EPU levels.

Loss of Condenser (LOC)

Because condenser back pressure margins are affected by EPU, the frequency of LOC could increase slightly. The initiating event frequency task for the PRA update will not increase the frequency of LOC based on EPU; however, the potential sensitivity of an increase was evaluated.

Initiating Event (IE)	Current PRA			Sensitivity
	IE Frequency	CDF	LERF	
LOC – Loss of Condenser	0.14	1.4E-7	1.3E-8	If it is assumed that LOC increases by 10%, the change in CDF would be 1.4E-8, which would be a negligible increase in risk.

Turbine Trip (TT)

Because the generator step up transformer margin is affected by EPU, the frequency of TT could increase slightly. The initiating event frequency task for the PRA update will not increase the frequency of TT based on EPU; however, the potential sensitivity of an increase was evaluated.

Initiating Event (IE)	Current PRA			Sensitivity
	IE Frequency	CDF	LERF	
TT – Turbine Trip	1.5	1.3E-6	9.7E-8	If it is assumed that TT increases by 10%, the change in CDF would be ~1E-7, which would be <1% change in risk.

Stuck Open SRV (SORV)

SORV (3.9E-3) and the inadvertent opening of an SRV (1.6E-2) are combined in initiating event IORV in the NMP2 PRA (0.02/yr total). Because the number of SRV challenges increase due to EPU power increase, the frequency of IORV increases slightly. Per MAAP run U1CNT1, 104 SRV cycles are expected over the first hour of MSIV-closure type scenario response given post EPU conditions. For pre-EPU conditions, 94 such cycles are expected (MAAP U1CNT1p). This represents a 10% increase in expected demands. While the SORV initiator is determined by the number of demands in the first several minutes, the increase in the number of challenges over one hour is deemed an appropriate measure of the relative change in the challenge. The potential sensitivity of an increase in SORV was evaluated.

Initiating Event (IE)	Current PRA			Sensitivity
	IE Frequency	CDF	LERF	
IORV – Inadvertent Open SRV	0.02	4.2E-7	5.1E-9	If SORV increases by 10% (0.004/yr to 0.0044/yr) (IORV changes from 0.02 to 0.0204), the change in CDF would be 8.4E-9 and the change in LERF is 1.0E-10.

Loss of Coolant Accident (LOCA)

Because of increased flow rates it is assumed that increased reactor energy could result in LOCA frequency increases. This is judged to be a conservative assumption based on Section 2.1.6. The total LOCA initiating event frequency includes small (8E-3 [5E-4 in update]), medium (3E-3 [1E-4 in update]) and large (7E-4 [7E-6 in update]) breaks. The initiating event frequency task for the PRA update will not increase the frequency of LOCA based on EPU; however, the potential sensitivity of an increase was evaluated. In addition, the LOCA frequencies are being updated in the ongoing PRA update to be current with the most recent industry data from NUREG/CR-6928 (Reference 84), which will reduce all three initiating event frequencies by more than an order of magnitude. Thus, this sensitivity includes these more realistic LOCA frequencies.

Initiating Event (IE)	Current PRA with updated IE			Sensitivity
	IE Frequency	CDF	LERF	
LOCA – Loss of Coolant	6.1E-4	2.6E-8	8.3E-10	If it is assumed that LOCA increases by 10%, the change in CDF would be less than 1E-8 a negligible risk change.

Conclusion

NMPNS has reviewed the assessment of the risk implications associated with the implementation of the proposed EPU and concludes that it has adequately modeled and/or addressed the potential effects associated with the implementation of the proposed EPU. NMPNS further concludes that the results of the risk analysis indicate 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.

No new impacts are identified for initiating event frequencies, component reliability, or success criteria, but impacts are expected for a limited number of operator actions because of a decrease in available operator response times resulting from the increase in decay heat associated with EPU. Risk increases, including sensitivity analyses associated with unlikely initiating event changes, are small and within the acceptance guidelines of RG 1.174 (Reference 80).

Therefore, NMPNS finds the risk implications of the proposed EPU acceptable.

Table 2.13-1 NRC Review Comments Summary

Note: The NRC SERs for the NMP2 IPE and IPEEE were reviewed and specific comments were identified and assigned as individual items for the NMP2 PRA update. Provided in the table below is a listing of each comment, along with the NMPNS PRA team response/resolution.

NRC Comments on IPE and IPEEE		
Item[1]	**Comments**	**Response/Disposition**
IPE-Letter Pages 2 & 3	Description of IPE results and unique features.	Description is reasonably accurate for the IPE. A description of the present PRA results will be different. For example, the statement "No credit is taken for recovery over 20 to 30 hour containment failure" is no longer true in the PRA; improvements in recovery have been incorporated.
IPE-Letter Page 2	Niagra Mohawk Power Corporation (NMPC) developing procedures to prevent RCIC trip under loss of service water.	This has not been incorporated into procedures, but is an obvious action (required within 2 hours in SBO procedure SOP-1) and TSC guidance (monitoring area temperatures, etc.) is expected to identify this obvious action. Per the NMP2 Station Blackout Bases Document, RCIC room heatup calculations assume the door is closed, but it is open to allow lower room temperatures.
IPE Section 6.2 identifies NMPC plans credited in IPE. If not implemented NMPC should revise IPE to reflect as-built, as-operated. Need not submit to NRC, but retain records for future.	To install valves in SGTS to increase reliability of containment venting.	Subsequent to the IPE, this valve installation modification was cancelled as not being cost-beneficial. As a result, training and procedure changes were pursued to assure that human reliability credited in IPE is reasonable. Several drills and training sessions addressed this aspect of the EOPs, including the last resort option of SGTS Bldg blowout. The latest EOPs have removed stops in the procedure; now the operators continue in the PC-P leg of procedure N2-EOP-PC and anticipate containment venting alignment. In addition, TSC guidance and resources will improve the obvious need to anticipate this alignment and provide resources. Although all these improvements in procedures, training, resources, and etc. are judged to support or improve the HRA value, the IPE values are still being used until a re-evaluation of this HRA is performed.
	Develop procedures to enhance Aux Bay room cooling during loss of service water.	No procedure was developed and credit for operator actions has been removed from the IPE model. Gothic calculation (SAS-PRA2-S-RHS-CALC, June 1998) shows that the limiting room (RHR B room) is marginal and realistically does not require room cooling. Still, the PRA model conservatively fails RHR rooms A and B if room cooling fails. The LPCS and LPCI C rooms clearly do not require room cooling and this dependency is no longer included. The RHR A and B failures although conservative do not impact the PRA results.

NRC Comments on IPE and IPEEE		
Item[1]	**Comments**	**Response/Disposition**
	Enhance SBO procedures.	SBO procedures were not available at the time of the IPE. Subsequent to the IPE, SBO procedures (SOP-1, 2 and 3) were developed and support the IPE assumptions.
	Provide additional internal flood guidance.	Additional guidance includes opening doors from outside that will remove water from the building. Guidance to isolate the flood is also included (see Alarm Response Procedure N2-ARP-01, Rev. 00 pages 1305-1307)
	Improve test & maintenance procedures to reduce the likelihood of ISLOCA.	Procedure change has not been made nor is it judged necessary. The PRA (Section 5.3.3) now incorporates the low probability of a MOV being opened during testing & maintenance without procedure improvements; this contributor is an insignificant risk contributor. On-line risk monitoring ensures that the unlikely coincident activities needed to initiate this event are identified in advance (e.g., PRA model should be conservative).
IPEEE-SE Page 2	0.5g HCLPF for 24 hrs does not meet EPRI SMA guidance.	Clarification: The 0.5g HCLPF is for 72 hours (see comments on TE below).
IPEEE-SE Page 6	Vulnerability definition not provided.	The fact that no vulnerability definition is provided does not provide a problem for NRC since the risk is obviously acceptable based on the NMP evaluation. See PRA Section 10 relative to risk management.
IPEEE-SE Page 6	Plant improvements needed.	<u>Seismic mounting of rack, cabinet and hoist assembly</u> The plant modifications for the seismic mounting described have been made (IPEEE page 7-2). <u>CR Fire</u> EOP-RPV is now retained at the remote shutdown panels. The control room fire risk in the PRA is judged to be conservative and is not dominating. There are no plans to add explicit TSC guidance or additional training.
IPEEE-TE Page vii	No freeze date.	This comment refers to a data freeze date beyond which additional data would not be considered. A date for data analysis for this PRA was implemented; however, other aspects of the PRA were allowed to change as appropriate to final sign-off.
IPEEE-TE Page ix Page 30	Tornado screening incomplete.	No action to be taken. NRC's analysis also shows that risk from high winds is low and can be screened.

NRC Comments on IPE and IPEEE		
Item[1]	**Comments**	**Response/Disposition**
IPEEE-TE Page ix, xii Pages 31-34, 44	External flood bounding analyses appear flawed and incomplete.	TE agrees that external flooding can be screened based on SRP compliance, but disagrees with NMP simplistic bounding argument. It is very difficult to estimate the risk from floods and there are numerous combinations of events that must be considered. It is NMP's position that a detailed analysis, considering plant procedures and timing, would lead to a low risk on the order of 1E-6/yr. Since there is very little that can be done cost effectively to reduce this risk further, no additional analyses are planned.
IPEEE-TE Pages x, xii Pages 7, 9, 10, 11, 41, 43	0.5g HCLPF for 24 hrs and not meeting EPRI SMA guideline for success reliability.	Clarification: The 0.5g HCLPF is for 72 hours. Only when using success path reliability guidelines of EPRI SMA does a 0.23 HCLPF result unless we credit equipment not in analysis scope. EPRI SMA is only guidance and justification for deviating is provided by the PRA analysis. This was shown to be non-risk significant by NMP and TE seems to agree. Also, note that the NMP PRA success criteria are for 24 hours not 72 hours, including external events.
IPEEE-TE Page x, xii Page 2, 24, 28, 44	Additional equipment failures due to smoke and combustibles not adequately addressed.	NMP does not know of any analyses to address this issue. If new analyses become available NMP will consider this further.
IPEEE-TE Page x Pages 24, 28, 44	No fire barrier failure rates in analysis, cross zone fire analysis.	Because of limited combustibles, limited active barriers, reliable detection and suppression, the screening and analysis is judged conservative. Scenarios where fire barriers failed were judged to be very low risk contributors. NMPC agrees that documentation of these judgments could be improved. The risk ranking of fire barriers will likely require this analysis improvement.
IPEEE-TE Page x Page 31	GI-103: No details of re-evaluation in submittal.	The FSAR re-evaluation was not repeated in submittal and there is no plan to do this as it adds no value.
IPEEE-TE Page xi Page 45	Plant improvements identified during walk down.	The storage rack near the RCIC motor-operated valves has been secured (IPEEE page 7-2).
IPEEE-TE Page xii Pages 2, 26, 43	Operator error rates for control room fires are highly optimistic, etc.	The most reliable operator action is used for only those fire scenarios where the control room remains habitable and equipment needed for immediate plant control is operating successfully. Also see response to IPEEE RAI II.1.
IPEEE-TE Page xii Pages 2, 19, 43	Heat release rate for cabinet fire not representative.	No action to be taken as it does not appear to impact the analysis conclusions.

NRC Comments on IPE and IPEEE		
Item[1]	**Comments**	**Response/Disposition**
IPEEE-TE Pages 2, 27	Seismic fires due to weakly anchored cabinets not addressed.	There are no known weakly anchored electrical cabinets at NMP2.
IPEEE-TE Page 7	Stuck open SRV and Large LOCA not addressed.	A stuck open SRV with RCIC success guarantees successful RPV isolation (nitrogen is not needed) and allows low pressure injection success. Therefore, the stuck open SRV event improves the number and reliability of success paths and is an insignificant risk contributor. Also, medium and large LOCAs due to pipe breaks are incorporated in the 0.5g HCLPF fragility in the PRA model.
IPEEE-TE Page 8	SLC seismic capacity.	The RPS system is very reliable with significant redundancy built into the function. Because of this, RRCS and SLC need not be "safety related" nor "seismic Category I" under the Regulations. The 0.5g HCLPF fragility in the PRA model incorporates RPS seismic failure. The frequency of seismic initiator and failure of RPS (non-seismic) during seismic initiating event is low in the PRA. Given this low risk and dependency on the operators in the ATWS model, no RRCS or SLC seismic evaluations are needed.
IPEEE-TE Page 9	HEP of 0.01 for depressurization equates to unreliability of all low pressure injection.	Depressurization is redundant to RCIC and HPCS for the 0.23 HCLPF success paths. This is included in the PRA.
IPEEE-TE Page 9	SBO procedure modification needed relative to depressurization and minimizing depletion of nitrogen.	EOPs address how to conserve nitrogen, specifically, EOP-RPV and EOP-C3. SOP-1 and SOP-2 have specific actions on how to conserve battery power. Separate criteria are given for blackout in lieu of the normal HCTL limits in EOP-6 Section 29.
IPEEE-TE Page 11	Consideration of human actions in the SMA not entirely in keeping with SMA guidance.	Compliance with SMA is believed to be in the IPEEE. The TE states that seismic PRA fully considered human actions and suggests safety significance is low.
IPEEE-TE Page 13	Consideration of piping degradation (e.g., wear) and impact on seismic flooding risk not included.	The 0.5g HCLPF fragility in the PRA incorporates this risk. The probability of degradation below this seismic capacity is negligible.
IPEEE-TE Page 27	No dependency matrix was provided and plant unique phenomena were not addressed.	NMP response to NRC questions provided IPE dependency matrix. No other important or unique dependencies or phenomena were identified.
IPEEE-TE Page 35,36	Approach to identifying other external events was not comprehensive.	NMP did consider other external hazards listed in the PRA procedures guide. This was not documented because it was not requested by the IPEEE scope.

NRC Comments on IPE and IPEEE		
Item[1]	**Comments**	**Response/Disposition**
IPEEE-TE Page 36	Little detail provided on systems interactions.	NMP believes that the present effort is reasonable.
IPEEE-TE Page 36	No specific information was provided concerning smoke impact on fire fighting effectiveness.	Smoke can affect fire fighting effectiveness and this is considered in training, etc.
IPEEE-TE Page 43	Seismic hazard assessment was truncated at 1.02g.	This will not impact the results, but will be considered in a future update.

[1] SE = Staff Evaluation (Enclosure 1 to NRC letter); TE = Technical Evaluation (Enclosure 2 to NRC letter)

Table 2.13-2 Significant PRA Certification Findings and Observations (F&O)

Element / Sub-Element	PRA Certification F&O	Level of Significance	Risk Impact - Response/Resolution
Element: MU Sub-elements: 4, 9, 13, 14	These are Sub-elements which will not be complete until the first and subsequent update cycles are complete. Grades assigned are contingent upon follow-through by the PSA and associated groups.	B	NEG-CA-001 Rev. 4 is currently being used.
Element: MU Sub-element: 5	In addressing plant specific failure events during the PS update, the UPS event which occurred at NMP should be included in the basic events	B	As part of the PRA update, all initiating events at NMP2, including the UPS event were evaluated and included in the PRA (Section 5.3.1). The impact of the UPS event of 8/13/91 was basically a loss of feedwater subsequent to a plant trip. Based on this event alone, the unavailability of feedwater is presently judged to be optimistic because it does not account for this event. However, the loss of feedwater initiating event increased from 0.05 (IPE) to 0.14 in the PRA update, which is judged to reasonably capture this event. The unavailability of feedwater, given it was not the initiator, was not increased because of the initiator frequency and the fact that measures have taken to preclude the UPS event from recurring.

Element / Sub-Element	PRA Certification F&O	Level of Significance	Risk Impact - Response/Resolution
Element: MU Sub-element: 6	There is in place a good system of archiving the PSA model and other related documents. This system should be well documented to insure that this information is assessable in the event of discontinuity in program management or other upset.	B	Tier 1 and 2 documentation for the PRA update are available both in hard copy with signatures and electronically. The documentation also summarizes changes from the original IPE. Background documentation (IPE and IPEEE and supporting information) is archived in files and on CD ROM.
Element: IE Sub-element: 7	1. Should probably examine possible inclusion of BOC (Break Outside Containment) as an initiator in light of its potential contribution to early-high release, not just CDF. 2. Also might consider multiple stuck open relief valves, as an initiator. 3. Should also examine the assumption of not analyzing sequences subsequent to a manual shutdown or manual scram. While these are usually "controlled" shutdowns, systems and operators are still challenged. In some cases, a manual scram would not be "completely controlled" depending on the need for the scram.	B	1. PRA update Section 5.3.3 was improved to explain why the frequency of Core Damage and LERF are low. 2. The frequency of multiple SRVs opening is on the same order of magnitude as Large LOCA with less severe challenges to the containment. PRA update Section 5.3.3 addresses this subject. 3. Manual shutdown events are now explicitly modeled in the PRA update (see PRA Section 5.3.1).

Element / Sub-Element	PRA Certification F&O	Level of Significance	Risk Impact - Response/Resolution
Element: IE Sub-element: 8	The scope of LERs and shutdown history is described; events are shown in Tables A-3 and A-4 (Tier 2). However, it is not clear why the transformer/UPS event of 8/13/91 was not included in the initiator data base.	B	The UPS event occurred after the IPE cutoff date. However, the event is explicitly included in the PRA update (see Section 5.3.1).

Element / Sub-Element	PRA Certification F&O	Level of Significance	Risk Impact - Response/Resolution
Element: IE Sub-element: 13	<u>System 26 (P. 3.2.1.26-1 of IPE)</u> The LOOP frequency and the recovery are intimately tied together. The NUREG-1032 recovery curves can be applied each on its specific frequencies. However, it appears that the NUREG-1032 weighted recovery curve was used and applied to the LOOP frequency which was based solely on grid and plant centered data. This appears to be optimistic relative to the NUREG-1032 assertions relative to severe weather because the magnitude and sample size of the plant specific data does not preclude a non-negligible weather component estimated after the guidance stipulated in NUREG-1032. It is advocated by the Certification team that the data only supports updating the plant centered data from 0.087/yr to 0.04/yr. Therefore, the weighted average of recovery should be recalculated coupling the new IE frequency which should include a 0.01 frequency for severe weather with the corresponding NUREG-1032 recovery curves.	B	Based on the PRA update, LOOP frequency increased from 0.04 to 0.11 based on plant specific data. Then, NUREG-1032 is used for recovery and includes weather events. The writeup was also improved during the PRA update (PRA Section 4.2.26 and 3.1)

Element / Sub-Element	PRA Certification F&O	Level of Significance	Risk Impact - Response/Resolution
Element: IE Sub-element: 16	LOOP frequency development should not preclude non-negligible severe weather component. Its 1 in 100 year value can't be precluded based on a short generating history. It should be added and included more appropriately in the recovery value.	B	LOOP recovery and use of NUREG-1032 has always properly accounted for severe weather. This was rectified during the PRA update and write-up was improved.
Element: AS Sub-element: 5	The evaluation of accident sequence response using RCIC can be strongly influenced by the plant specific feature at NMP-2 of the Dikkers SRVs. The Dikkers SRVs have characteristics associated with them that result in RPV depressurization to very low pressures when the EOP direction is followed to open all ADS SRVs. Following the emergency depressurization directions results in the RPV pressure reduced to well below the pressure required for RCIC operation whether or not the low pressure trip is bypassed. This effect is to make RCIC unavailable whenever emergency depressurization is directed by the EOPs.	B	SBO procedure N2-SOP-01 Rev. 4 Cautions the Operators that "operating with RPV pressure less than 200 psig can jeopardize RCIC availability." Also, most recent EOPs (1/1/99) provide new direction (EOP-6, Attachment 29) so that depressurization does not necessarily make RCIC unavailable. Also, MAAP calculations indicate that it takes at least 4-6 hours without containment heat removal (per EOPs and operator training, RPV pressure is maintained below HCTL and other containment limits) before eventual emergency depressurization may occur. Since the SBO analysis ends at 8 hours this is not an important issue.

Element / Sub-Element	PRA Certification F&O	Level of Significance	Risk Impact - Response/Resolution
Element: AS Sub-element: 5	Based on Calculation EC-129, the Division I battery 2BYS*BAT2A Type NCN-35 is able to supply loads during SBO for six hours. In this calculation, the loads not required during SBO event are assumed to be shed within two hours. The current assumption is that the station battery could last for 10 hours. Therefore, it is recommended that the event tree analyses for the SBO scenario be revised.	B	The SBO model has been revised as part of the PRA update. Recovery is now only allowed out to 8 hours given successful DC load shedding (based on latest analysis). This has a minor impact because there was very little credit in the original analysis beyond 8 hours anyway (See Section 3.2.1.3).
Element: AS Sub-element: 6	SBO There is a revised SBO evaluation for NMP2 from GE which indicates that there are a number of new constraints on the ability to cope with an SBO. These include reduced battery life, requirements to depressurize within 4 hours, and higher RCIC room temperatures. These considerations are judged to adversely impact the SBO accident sequence evaluation in the PSA.	B	The latest GE analysis and procedures were reviewed and considered in the PRA update and the SBO model was revised extensively (see previous observation). The SBO risk was reduced due to modeling changes (mostly due to changes in procedures to use HPCS to supply Div I or II AC). The latest procedures and training incorporated insights from the IPE. Since LOOP frequency has increased based on plant specific data, the overall effect of the update was not a reduced risk.

Element / Sub-Element	PRA Certification F&O	Level of Significance	Risk Impact - Response/Resolution
Element: AS Sub-element: 11	ROOM COOLING The treatment of room cooling for RHR and LPCS operability is described in a confusing and conflicting manner in the IPE documentation. For example, the room cooling requirement for RHR is not clearly delineated in the dependency matrix and the method of room cooling treatment for loss of service water cases is highly dependent in the model on the operating action to open doors. This is not currently proceduralized and therefore should not be credited. There may also be calculations with GOTHIC that could justify not requiring an active room cooling system. These issues need to be clarified to ensure that system importances for applications are accurately reflected.	B	Both the confusing documentation and the modeling of room cooling have been corrected in the PRA update. Gothic calculations (SAS-PRA2-S-RHS-CALC, June 1998) show that the limiting room (RHR B room) is marginal and realistically does not require room cooling. Still, the PRA model conservatively fails RHR rooms A and B if room cooling fails. The LPCS and LPCI C rooms clearly do not require room cooling and this dependency is no longer included. The RHR A & B failures although conservative do not impact the PRA results.

Element / Sub-Element	PRA Certification F&O	Level of Significance	Risk Impact - Response/Resolution
Element: AS Sub-element: 7	DFP The diesel fire pump alignment under SBO when essential lighting has been shed appears to be a difficult process. There is questionable evidence that the alignment can be performed and the LPCI valve opened under SBO conditions.	B	The present model only allows a 0.5 probability of success (0.2 operator action failure). The most recent EOPs ensure that the DFP will be aligned early (level below scram set point and stops in the EOPs have been removed); the operators will not wait. SBO model only allows DFP success if RCIC was successful for 2 hours. The operators practice the physical alignment and the LPCI MOVs are accessible. This has not been practiced in a SBO condition where the operators have to use flashlights. However, given that this would be done by sending operations personnel out in pairs, the above EOP changes, and timing in the SBO model, a 0.2 probability of failure is judged reasonable if not conservative. We may pursue taking more credit for the operator in the future. A separate open item was whether the DFP can protect the core 2 hours after event initiation (0.3 failure probability). Preliminary MAAP calculations indicate that a diesel fire pump with 1 of 2 injection paths is marginal. Therefore, the 0.5 S1 failure probability may not be conservative, but is still considered reasonable given our present state of knowledge. This will be considered further relative to risk management and future updates.

Element / Sub-Element	PRA Certification F&O	Level of Significance	Risk Impact - Response/Resolution
Element: AS Sub-element: 10	TRACEABILITY The traceability of individual elements of the model are difficult in some cases. The miscalibration of the low pressure permissives on the LPCI AND LPCS lines are identified as possible pre-initiator HEPs, but their basic event is: • not identified in the IPE discussion of the HEP • does not have a calculation to support the quantification of the HEP referenced in the IPE • is not included in the fault tree for the low pressure injection systems • is not included in the cutsets for the respective top events • is included in the ECCS initiation logic	B	The PRA update improves the documentation. Section 5.2 identifies HRA event ZEC01 and basic event ISCZECMISCALIB01 as included in top event ECV which models common cause failure of all ECCS low pressure injection paths. When ECV fails, all low pressure injection paths fail in the PRA model.

Element / Sub-Element	PRA Certification F&O	Level of Significance	Risk Impact - Response/Resolution
Element: AS Sub-element: 5	<u>SBO</u> The accident sequence evaluation for SBO needs to be re-evaluated based on the revised SBO report which substantially shortened the available time for coping from 19 hours used in the Rev 1 of the SBO calculation and in turn used as a basis for the IPE to 4 hours in Rev 2 of the SBO report. This is believed to have a major impact on the quantification of dominant core damage sequences. Because a realistic assessment of the Rev 1 results was used in the original IPE model, the quantified impact of Rev 2 is expected to be not large. This shows good judgment in the use of the original optimistic Rev 1 SBO report results.	B	The PRA update includes a major revision to the SBO model. As discussed in response to other observations, the model only goes to 8 hours (e.g., no credit is given to recovery beyond 8 hours) and is based on latest analysis and procedures. There was very little credit in the original model beyond 8 hours. Refer to PRA Section 3.2.1.1.
Element: AS Sub-element: 13	The impact of load shedding assumptions on the PSA should be re-evaluated and their results documented.	B	The PRA update includes a major revision to the SBO model based on latest procedures and analysis. Although the HRA has not been redone, the procedures are consistent and in some instances exceed the IPE assumptions. The model conservatively assumes core damage occurs early at 2 hours if load shedding fails.
Element: AS Sub-element: 10	<u>DEPENDENCIES AND LOW PRESSURE PERMISSIVE</u>	B	In the PRA update, miscalibration is included in top event ECV. Failure of ECV fails LPCS and LPCI injection paths. Success criteria will not allow low

Element / Sub-Element	PRA Certification F&O	Level of Significance	Risk Impact - Response/Resolution
	The HRA discussion identifies the injection valve low pressure permissive as a potential CCF probability due to miscalibration. However, in the low pressure injection systems there is no identification of this CCF failure mode. This failure mode does not appear to be discussed in any other IPE section. The basic event appears to have been put in the E1, E2, E3 ECCS initiation logic which is assumed to be able to be backed up by manual actuation if auto initiation fails. IA and IB and LS are not affected by this failure if ME is successful. There may also be some additional HEP that could be included to address the question of locally opening the injection valve and bypassing the low pressure permissive by turning the valve hand wheel. No HRA is performed to support this action. There does not appear to have been a clear definition of what the HEP was, where it was calculated, or what logic model it applies to. The impact is judged to be small but it cannot be readily confirmed because the		pressure makeup through these paths from any source. See PRA Sections 4.2.4 and 3.2.1 event tree rules for SUP4, TR1, etc. No detailed calculation has been developed for this miscalibration HRA; the evaluation is described in PRA Section 5.2. See also the response to Element HR, Sub-element 6.

Element / Sub-Element	PRA Certification F&O	Level of Significance	Risk Impact - Response/Resolution
	dependencies associated with the failure of this permissive could adversely impact LPCI, LPCS, SW X-TIE, AND THE DIESEL FIRE PUMP. Ensure the HEP for the low RPV pressure permissive is: • described in the LP injection systems • quantified in a calculation • treated among "tops" so that the dependency is accurately reflected		

Element / Sub-Element	PRA Certification F&O	Level of Significance	Risk Impact - Response/Resolution
Element:AS Sub-element: 5	CONTAINMENT VENT The EOPs and the EOP-6 specific attachment for venting taken together represent the written basis for operator response to challenges to high containment pressure. However, the vent that is allowed by these is assessed in the HRA to have a 1.0 failure probability. Despite this, the model appears to use a more optimistic HEP that was developed including assumptions regarding procedural modifications. Recent emergency drill experience indicates that the operating staff in conjunction with the TSC could decide under certain conditions to vent the containment without requiring the extensive alignment of the "hard piped" system.	B	Major improvements have occurred since the IPE. Latest improved EOPs have removed stops such that operators will not wait for the high pressure condition that requires venting. This was confirmed with Operations. In other words, with the knowledge of venting alignment difficulties and the improved EOPs, there is a high likelihood of success. Containment venting has been addressed in drills and training and as part of the SAM process. Present EOPs and supporting procedures were found to provide adequate flexibility and to address support states. The SAM process and TSC guidance will also help. The present analysis (same as IPE) is judged reasonable to conservative.
Element: DA Sub-element: 7	If there is a sufficient experience base, recommend replacement of maintenance unavailability data with plant specific data.	B	Current plant specific maintenance unavailability is being used in the PRA update (Section 5.1).

Element / Sub-Element	PRA Certification F&O	Level of Significance	Risk Impact - Response/Resolution
Element: DA Sub-element: 8	Numerical results for common cause failure of SRVs to depressurize appear to be quite low.	B	NRC/INEL common cause data parameters are used in the PRA update and judged to be reasonable if not conservative.

Element / Sub-Element	PRA Certification F&O	Level of Significance	Risk Impact - Response/Resolution
Element: DA Sub-element: 9	SRV/SOLENOID CCF The SRV data and the associated solenoids can be expected to have a CCF term or terms. The NMP2 model has an extensive degree of CCF terms. The IPE currently uses ~ 1E-6 as the CCF for all valves and 1E-5 for the sum of all multiple hardware failures of a CCF nature. This may be optimistic. However, a simplistic CCF approach using generic SRV data results in estimating the CCF probability at 4E-4. This estimate should be checked against the design and possibility of a common cause failure.	B	NRC/INEL common cause data parameters are used in the PRA update for SRVs, SOVs, check valves, and are judged to be reasonable, if not conservative. The simplistic model and values suggested above do not apply at NMP2; detailed common cause modeling is utilized. Global common cause (easiest comparison to simplistic approach) in the NMP model is ~2E-5 for all SRVs and ~2E-5 for all SOVs. Thus, the simplistic approach appears to be conservative by an order of magnitude. The depressurization function using the SRVs (which includes the ADS SRVs) was also evaluated using operating experience data. There have been four precursor events that were identified in the data. These are the following: • All SRVs and solenoids at a BWR located in the northeast U.S. were mistakenly covered with insulation during an outage. These solenoids were subjected to temperatures well above their qualification temperature for an entire cycle. Several of the valves failed to open on their subsequent bench test. This precursor is judged to affect all SRV solenoids and may cause all SRV solenoids to fail. (Not all SRVs were failed in the precursor).

Element / Sub-Element	PRA Certification F&O	Level of Significance	Risk Impact - Response/Resolution
			• An event was reported in January 1980 at a BWR, that hard-seat check valves were installed in the pneumatic supply to the accumulators of the ADS SRVs. These check valves allow back leakage which could bleed down the accumulators and make ADS inoperable for some events. This event was treated as rectified and is not counted in the data because of the action required by IE Bulletin 80-01 dated January 11, 1980. • In August of 1993 at a midwest BWR, 1 of the 6 SRVs failed its remote actuation test at the test facilities. The failure was due to a failed diaphragm in the air operator, stemming from improper application of insulation. Although the other valves were discovered to be improperly insulated, they passed their remote actuation tests. This could be considered a potential common cause failure. • A recent error at another midwest U.S. BWR reinforces the need to accurately account for the common cause failure. Three faulty solenoid control valves made, tested, and certified by Target Rock Corporation were found. • According to an NRC preliminary notification, workers disassembling one of the bad valves found that "internal corrosion had caused binding of the

Element / Sub-Element	PRA Certification F&O	Level of Significance	Risk Impact - Response/Resolution
			solenoid operating mechanism." Target Rock indicated that, in a possible deviation from procedures, the valves may have not been dried properly following hydrostatic testing at the Target Rock facility. All events were common cause failures, but they did not result in failure of <u>all</u> ADS valves. Using these four precursor events the following calculation was performed: • One additional failure is assumed. • The estimated time unavailable before discovery = 6 months, i.e., the event is considered to have caused an unavailability of 6 months for those valves that failed.
Element: DA Sub-element: 15	The probability of a SORV conditional on its need to open for various transient initiators is not modeled. Transients with SORV are terminated and believed to be accounted for in the IORV/Small LOCA tree. This is adequate if the initiating event frequency for IORV adequately includes the SORV conditional probability which may change	B	The IORV initiator was recalculated based on plant specific data (see Sections 5.3.1 and 5.3.3).

Element / Sub-Element	PRA Certification F&O	Level of Significance	Risk Impact - Response/Resolution
	for sensitivity studies, applications, and updated transient data.		
Element: DA Sub-element: 15	The value of and the rationale for the diesel mission time is not documented. The only source of the value was a RISKMAN file. This is a fairly highly visibility and controversial PSA issue.	B	Systems analysis Tier 1 (Section 4.2.6) and Tier 2 identify the fact that diesel mission time is 6 hours. The basis is that the SBO model only goes to 8 hours and recovery time depends on when the diesel fails (e.g., time to core uncovery after 6 hours of EDG success is much longer). Since these conservatisms are not accounted for in the SBO model, 6 hours was chosen as a reasonable, but conservative time.
Element: DA Sub-element: 15	<u>RPS</u> (duplicate of SY-19) The scram system description and the basis for the point estimate calculation for mechanical and electrical common cause failure are incomplete. NUREG-0460 is referenced for the estimated failure probabilities, but this document does not justify the 4.3E-6 mechanical common cause failure probability. The basis for the cited value requires that the scram air header have a low pressure scram signal as input to the RPS. The system description does not define this and therefore the cited conditional probabilities do not apply.	B	The reference used in the NMP2 PRA is outdated and the uncertainties are large for the electrical and mechanical SCRAM failure probabilities. INEEL/EXT-98-00670, October 1998, "General Electric Reactor Protection System Unavailability, 1998 – 1995 (draft 2)" suggests an unavailability estimate of 3.8E-6/year. The present NMP2 analysis is judged to be conservative.

Element / Sub-Element	PRA Certification F&O	Level of Significance	Risk Impact - Response/Resolution
Element: HR Sub-element: 5	The IPE does not provide any real insight into a systematic process being followed to conclude that pre-initiator HIs could be assumed to be subsumed into maintenance unavailabilities.	B	Pre-initiators were assessed for each system during the IPE and PRA update. They were not assumed to be subsumed into maintenance unavailabilities. The revised evaluation is documented in the Systems Analysis and included in the PRA model summarized in Table 5.2-1.
Element: HR Sub-element: 6	The source and analysis behind the selection of 1.0E-5 for common cause mis-calibration of instrumentation is not adequate. A more complete explanation and /or analysis should be provided in the update of the IPE.	B	The likelihood of miscalibration is low as documented in Section 5.2 of the PRA update. The ~1E-5 value is similar to NUREG results.
Element: HR Sub-element: 10	The tier 2 HRA document appears to be missing the following: HHU21—Stop RPV depressurization before RCIC stalls. This HEP is not evaluated in the HRA document even though it references another HEP. It uses a value of 1E-2 as the failure probability even though there is no procedure to deal with the Dikkers SRV effect of allowing depressurization to below the RCIC operability point of 50 to 60 psig.	B	This action is no longer included in the model because depressurization is not expected to occur during the SBO time window of 8 hours. Also, the procedure (N2-SOP-01) cautions operators with regard to depressurizing too low and the latest EOPs contain guidance relative to not having to depressurize with RCIC running (EOP-6 Attachment 29). See also the response to Element AS, Sub-element 5.

Element / Sub-Element	PRA Certification F&O	Level of Significance	Risk Impact - Response/Resolution
Element: HR Sub-element: 10	<u>SBO</u> HEPs for SBO may need to be re-evaluated using the directions in SOP-01 and SOP-02. These directions may alter the assessed HEPs. Neither SOP-01 nor SOP-02 specify pre-alignment of the DFP for injection prior to reducing the essential lighting. This is judged to result in a substantial degradation in DFP successful alignment probability.	B	The operators are in the EOPs, which address the DFP, as well as SOPs during SBO. The latest EOPs ensure that DFP will be aligned early without hesitation. This has been confirmed with Operations. In addition, DFP alignments are likely to be accomplished before reducing essential lighting loads (DC load shed). The original HRA analysis is considered conservative. (See also the response to the Element AS, Sub-element 7).
Element: HR Sub-element: 11	<u>DFP</u> The HRA appears to be performed assuming that the power to MOV 24A is available to support opening it during the assumed alignment for RPV injection. EOP-6 Attachment 6 does not identify how the valves are to be opened or the difficulty involved in opening the valves under different conditions such as SBO or loss of service water. The HRA apparently assumes the following optimistic assumptions regarding DFP alignment under SBO conditions: • no load shed of essential lighting which is specified in SOP 01	B	Power is not assumed available during SBO. An operator must open MOV24A locally. All valves can be turned by the crew. Confirmed with Operations that if the valve fails to open or can not be opened due to no AC power, it is understood that it will be opened locally. See also the responses to Element AS, Sub-element 7, and Element HR, Sub-element 10.

Element / Sub-Element	PRA Certification F&O	Level of Significance	Risk Impact - Response/Resolution
	• all valves are accessible, but no information provided to justify this • all valves can be turned by the crew, but no information provided • sufficient crew is on-site to carry out the actions • power is available to MOV 24A These are all judged to be optimistic, and the assumption that power is available to MOV 24A is clearly incorrect in the way the DFP is used in the PSA model.		
Element: HR Sub-element: 12	<u>FW FLOW CONTROL DURING ATWS</u> Re-establishing feedwater between 25 sec after feedwater runback ("lockout" time) and 83 sec when Level 1 is passed isolating the condenser hotwell due to MSIV closure appears to be given too much credit at 0.5.	B	Re-establishing feedwater does not have to occur in the time frame suggested and it was judged that there was some chance. NMP does not believe in using 1.0 when there is an opportunity for success (based on HRA and interviews). We judge that the 0.5 value is appropriate.

Element / Sub-Element	PRA Certification F&O	Level of Significance	Risk Impact - Response/Resolution
Element: HR Sub-element: 16	The HRA analyst used a "cause based" analysis procedure (EPRI-TR-100259) for developing HHA1. This is a stress related event and the EPRI procedure is judged not to be effective in differentiating between stress and non-stress sequences. Therefore this HEP may be lower than the sequence can justify.	B	There are several hours to perform this local action when time permits. Even the TSC could perform the action. NMP consider the value to be conservative.
Element: HR Sub-element: 16	LPCI/LPCS FLOW CONTROL UNDER ATWS EOP-6 Throttle ECCS Attachment 3 This appears difficult to implement and is not the procedure evaluated as part of the HRA for this action.	B	This action is performed after emergency depressurization and the EOPs utilize LPCI A and B as the preferred ECCS trains. Throttling is available in the control room from these trains, which makes the task much easier than having to apply EOP-6 Attachment 3. Even if EOP-6 Attachment 3 is needed, it is straight forward and is performed in the control building. A re-evaluated HEP is judged unnecessary at this time.

Element / Sub-Element	PRA Certification F&O	Level of Significance	Risk Impact - Response/Resolution
Element: HR Sub-element: 23	HHMA1: MA & MB (Loss of SW) This action is to open LPCI room doors to assure room cooling. The HRA assumes a procedure is in place. However, a procedure could not be identified—neither EOP-6 nor SOP-01 specify opening LPCI doors or MCC doors for room cooling. The HRA assumes a procedure exists and uses a value of 0.1 conditional failure probability (90% success).	B	Operator action has been removed from the model. Loss of room cooling fails RHR A & B with no credit for operators. This is conservative based on a Gothic calculation.
Element: HR Sub-element: 28	The HEP, HHU-21, is an action identified in Table 3.3.3-1 as "Stop depressurization before RCIC stalls." There is no EOP for this action: therefore, the analysis (per the Table "see HHOA1") is not a valid analysis since the timing, stress and steps to perform are not identified.	B	This has been removed from the model because the updated model stops at 8 hours before containment conditions becomes an issue. Also, see response to Element AS, Sub-element 5, and Element HR, Sub-element 10.

Element / Sub-Element	PRA Certification F&O	Level of Significance	Risk Impact - Response/Resolution
Element: DE Sub-element: 4	It appears that the dependency matrix was constructed with plant design basis in mind, rather than the realistic (as modeled) basis for the PRA. This may be somewhat confusion for future users Example: Noted dependency of RHR on normal AC, TBCCW and Service Water for pump seal cooling. System discussion notes assumption that seal cooling is not needed. Component Block Description tables (in system portion of the report) are good in that they define failure mode, initial state, actuated state, support system and state on loss of support. The matrix should relate to this better.	B	The dependency matrix was intended to address all dependencies that the engineers could identify during the PRA development without requiring consideration as to whether they were needed in the model. Note that seal cooling during shutdown cooling is a dependency but shutdown cooling has not yet been added to the PRA. The Systems Analysis (PRA Section 4.2) identifies the dependencies that are modeled and why some may not be modeled.
Element: DE Sub-element: 5	It is not apparent that pre-accident human actions are incorporated in the modeling (common cause miscalibration or failure to restore from maintenance).	B	This was considered again during the systems analysis task during the PRA update. It is better documented in the systems analysis. Several misalignment pre-initiator events were added to the model. Nothing significant was found or added to the model.

Element / Sub-Element	PRA Certification F&O	Level of Significance	Risk Impact - Response/Resolution
Element: DE Sub-element: 9	The evaluations are simplistic, e.g., room heat-up evaluations, zebra muscles, etc. Each evaluation should be supported by quantitative analysis where appropriate rather than being a qualitative evaluation.	B	This was considered during the PRA update with minor changes made (failure over a 24 hour mission time with the design and programs is not judged likely). No cost-benefit justification for further quantitative analysis could be made given the present modeling, including common, etc. Loss of lake intake to service water was added to the PRA as an initiating event (LKX) to provide additional completeness.
Element: DE Sub-element: 9	The flooding screening criteria that states floods which do not cause initiating events and impact an important system should be eliminated. Such criteria are very difficult to justify. A broader set of floods should be considered.	B	Section was clarified during the PRA update to say that generally these types of failure are required in order to be important, which NMP still believes. The original write-up implied that this was a basis for modeling. Note that there are still some floods that were screened out that could be modeled in the future; this will be considered as a future update.

Element / Sub-Element	PRA Certification F&O	Level of Significance	Risk Impact - Response/Resolution
Element: QU Sub-element: 8	The following assumptions are made in the analysis: • In a loss of Div. I Emergency AC power event, it is assumed that the Div. II charger would not be able to maintain the load. • During an SBO event, if RCIC is successful for the first 2 hours, there is a probability (0.1, assumption) that the operator would improperly depressurize the vessel and cause the unavailability of RCIC. Although the procedure (SBO-6) reminds the operating staff to use caution, no guidance is provided. • Discussion with a shift supervisor during the certification peer review indicated that if directed to emergency depressurize by the EOPs, RCIC availability would not be a reason to stop the depressurization.	B	• The charger is credited in the PRA update as suggested. • The 0.1 probability event has been removed in the PRA update as not likely during the first 4 to 8 hours. • EOP and SOP procedure changes improve the depressurization concern. See responses to Element AS, Sub-element 5, and Element HR, Sub-element 10.

Element / Sub-Element	PRA Certification F&O	Level of Significance	Risk Impact - Response/Resolution
Element: QU Sub-element: 18	There are several operator actions that are credited in the analysis but the procedure guidance is either not in place or not clear. For example: • venting the containment • opening the doors to provide room cooling • depressurization of the vessel when makeup is provided by RCIC during SBO.	B	The room heat-up calculation has been completed and model revised accordingly. The updated model is based on procedures, training, and operator interviews. Note that there are some actions that are not explicit in the procedures, but are obvious and confirmed by interviews with operations and training. For example, if an MOV does not open or close, the operators would send someone locally (e.g., HA01).

Element / Sub-Element	PRA Certification F&O	Level of Significance	Risk Impact - Response/Resolution
Element: SY Sub-element: 7	Injection system piping "keep filled" systems are not modeled because they are not considered to cause failure if not functional. The treatment of the keep fill system is a strong potential variable identified among different plants regarding its treatment in the PSA. The treatment varies from: • Not included in the model to • Included in the model, and if unavailable, causes the system to be unavailable (i.e., operators would not use the system if injection pipe known not to be full) This variation is extremely different. There can be some plant specific design or procedural differences that affect this treatment.	B	The systems analysis documentation (e.g., PRA Section 4.2.1.11) was improved to explain why explicit modeling of the keep fill system is not required.
Element: SY Sub-element: 7	LOOP load shedding diesel start sequence and reloading not modeled.	B	This was considered during the PRA update. Failure of diesel generator load sequencing is assumed to be included in the basic events for EDG start, MOV supply operation, and circuit breaker demand. The failure of the load sequencing is considered a small contributor in comparison to the other failure modes.

Element / Sub-Element	PRA Certification F&O	Level of Significance	Risk Impact - Response/Resolution
Element: SY Sub-element: 8	IPE documentation indicates that mis-calibration of ECCS pressure permissive is modeled. Such an event could not be identified in the fault trees for E1, E2, A1 or 1B.	B	This has always been in the IPE and the PRA update as basic event ISCZECMISCALIB01.
Element: SY Sub-element: 8	IPE indicates that there is potential for human induced common cause failure for SLCB (failure to restore). It is assessed to be 3E-3 (or 3E-4 after some procedure changes). However, fault tree SL includes events "Valves Misaligned after Testing—Operator Error" and "Isolation Valve Misaligned After Quarterly Testing." Only the first shows up in the SL cutsets and then with a probability of 1E-5.	B	The documentation and fault tree has been revised in the PRA update. A single event is used to represent unavailability of SLCS due to misalignment.
Element: SY Sub-element: 10	There is no common cause event for ECCS suction (suppression pool) strainer plugging.	B	Top event ST has been added to the PRA which models common cause ECCS Suppression Pool suction strainer plugging.

Element / Sub-Element	PRA Certification F&O	Level of Significance	Risk Impact - Response/Resolution
Element: SY Sub-elements: 10 & 12	Support system requirements appear to be accounted for in the model but the supporting documentation is confusing and not clear in some cases. Example: HVAC requirements for RHR pumps. Indicates that pumps would fail with loss of cooling (~ 5 hrs.) but do not model because loss of HVAC to MCC area is more restrictive because it fails two injection paths. Discussion for MCC area coolers said that cooling would not be a problem until 9 hours (and then only if RHR and LPCS had not started by then). Therefore, it was not important. This implies that HVAC for MCC areas is not modeled when it actually is.	B	Documentation has been clarified and the modeling revisions were made as part of PRA update.
Element: SY Sub-element: 17	RCIC may have temperature trips on high Main Steam Tunnel and RHR room temperature. These trips do not appear to be modeled in the RCIC system analysis. These trips need to be included in the RCIC model to account for common failures causing both MSIV closure and RCIC failure. A plant-specific room heatup calculation should be performed to insure that this is not a special initiator.	B	High area temperature trips (RHR A and B rooms and RCIC) have been added to the RCIC model in the PRA update.

Element / Sub-Element	PRA Certification F&O	Level of Significance	Risk Impact - Response/Resolution
Element: SY Sub-element: 26	SRV DESCRIPTION The description of the SRV capability and its characteristics are not provided. The Dikkers characteristics of importance to include in the description are the following: • nitrogen pressure required to open SRVs under different containment conditions • lowest RPV pressure that emergency depressurization will bring the RPV to • leakage characteristics of the nitrogen supply • duration of the nitrogen supply • operator actions necessary to provide SRV capability • accident response • qualification temperatures and pressures of the SRV and solenoids • treatment of relief valves on the pneumatic lines	B	The original IPE and the PRA correctly account for the Dikkers SRVs. In fact, the potential for depressurizing all the way to ~0 psi was a concern identified in the original IPE as part of the SBO analysis. The EOPs now address this potential cause for making RCIC unnecessarily unavailable. PRA Section 4.2.1.13 discusses the model, timing of nitrogen supply, etc. It was not deemed necessary to have a "Dikkers SRV" discussion, but this may be considered if necessary for specific applications. Relief valves on pneumatic lines have been neglected as insignificant contributors.

Element / Sub-Element	PRA Certification F&O	Level of Significance	Risk Impact - Response/Resolution
Element: SY Sub-element: 26	DIESEL FIRE PUMP The flow rate and the pressure capability of the DFP for RPV injection would be useful. Specifically, a calculation that identifies whether the DFP can provide adequate core cooling and under what containment and RPV conditions.	B	Preliminary calculations and MAAP analysis have been conducted which indicate that DFP is marginal in protecting the core. The 0.5 probability of success once thought to be conservative is considered reasonable until further analyses are conducted.
Element: SY Sub-element: 26	Identify the low pressure permissive logic and its configuration for all low pressure injection valves. Also define how this low pressure permissive is included in the evaluation of the service water cross tie injection and the DFP injection evaluation. Specifically, is the low pressure permissive miscalibration failure mode included in all injection modes using SW and the DFP.	B	Non-SBO Model Common cause miscalibration is now in top event ECV in the PRA update. Failure of ECV guarantees failure of all ECCS injection paths including top events IA and IB (RHR A and B injection paths). If IA and IB fail (e.g., due to ECV), then fire water and service water crosstie are also failed in the model since they depend on IA and IB. This is all documented in the RISKMAN PRA model. SBO-Model ECV is neglected as an insignificant contributor to DFP failure (ECV failure << DFP).

Element / Sub-Element	PRA Certification F&O	Level of Significance	Risk Impact - Response/Resolution
Element: TH Subselement: 4	Success Criteria related items that could use better documentation or model changes in the update include the following: • room cooling treatment for RHR and MCC rooms • DFP alignment success probability when performed under SBO conditions involving load shedding of all essential lighting • RCIC and DFP success given revised GE SBO report • RCIC success following Emergency Depressurization • Depressurization requirement for Medium LOCA with RCIC initially available (conservative assumption)	B	• Room cooling treatment has been clarified and model changed. • DFP credit has not been changed. Preliminary analysis indicates that the model is reasonable. • SBO analysis and model have been updated per the latest GE report. • Emergency depressurization does not occur with RCIC success in SBO for at least 4-6 hours. • Model revised such that MLOCA and RCIC success lead to emergency depressurization success.

Element / Sub-Element	PRA Certification F&O	Level of Significance	Risk Impact - Response/Resolution
Element: TH Sub-element: 10	<u>ROOM HEAT-UP</u> There is an effective discussion of the room heatup calculations that addresses various rooms in the plant relative to room cooling requirements. The dependency matrices and the documented discussion relative to system capability under loss of room cooling may not always be consistent. In addition, there may be more recent information to support more realistic modeling of the system capability under loss of room cooling.	B	Documentation, calculations and models; and the event tree models have been updated. See response to Element IE, Sub-element 3.
Element: TH Sub-element: 12	There is very little discussion of the thermal hydraulic calculations that are used in the various aspects of the model.	B	MAAP models have been updated as well as the Tier 1 and 2 documentation.

Table 2.13-3 Plant Modification Considerations

Modification	PRA Impact
ARTS/MELLLA implementation	Potential for PLOF increase evaluated
Bypass around condensate demineralizers	Reliability improvement, installed
Capacitor banks for grid stability	Evaluated and not required, no impact on reliability
Re-rate various BOP valves, piping and heaters	Mostly calculations and engineering, no reliability impact
Simulator upgrades for 2010 RFO	Ensure consistency with plant – no impact on PRA
HP turbine replacement	Yes – needed to support EPU, restores reliability
Steam dryer modification	Restores or improves reliability
Piping vibration monitoring	Temporary modification for monitoring and ensuring no impact
Feedwater pump impeller and speed increase replacement	Yes – Speed increase to support EPU, impeller replacement (5 vane to 7 vane) to increase flow and restore or improve reliability
Reciculation runback logic changes	Yes – restores or improves reliability to prevent loss of feed transients
Heater drain pump replacement	Yes - restores or improves reliability
Balance of plant piping support replacement	Restores or improves reliability
Replace LP turbine cross around piping relief valves	Yes - restores or improves reliability
Install additional HVAC coolers	Yes - restores or improves reliability (turbine building)
Instrument replacement and modification	As necessary, restores or improves reliability, mostly rescaling anticipated
Replace heater safety valves	As necessary, restores or improves reliability
Change switchyard protective relay string	Evaluated and not required, no impact on reliability
Replace atmospheric exhaust hood diaphragms	Yes – restores or improves reliability
Voltage regulator/power stabilizer	Not required, no PRA impact
Feedwater motor cable replacement	Yes – restores or improves reliability
Simulator upgrades for 2012 RFO	Ensure consistency with plant – no impact on PRA
Increase bus duct cooling	Yes - restores or improves reliability
Equipment qualification	No impact – concerns are minor and long term radiation issues
Improve main transformer cooling	Yes – restores or improves reliability

Table 2.13-4 MAAP Summary

ID (1)	**Scenario Description & Objective**	**Pre-EPU Results**	**Post-EPU Results**
U2CNT1	MSIVs close with feedwater available and no containment heat removal – demonstrate feedwater success for 24 hours from the CST and note timing to containment failure	32.4 Hrs to SRV Reclose Feedwater success for 24 hours Time to HCTL = 2.9 Hrs Time to PSPL = 14.6 Hrs Time to PCPL = 22.8 Hrs Time to DTL (340°F) = 37.1 Hrs Time to Cont Failure = 41.5 Hrs	26.2 Hrs to SRV Reclose Feedwater success for 24 hours Time to HCTL = 2.3 Hrs Time to PSPL = 11.8 Hrs Time to PCPL = 18.5 Hrs Time to DTL = 30.5 Hrs Time to Cont Failure = 33.4 Hrs
U2LOF1	Total loss of feedwater without injection, MSIVs open – evaluate timing to core damage	Time to Level 1 = 4 Min Time to -14 inches (TAF) = 11 Min Time to -39 inches = 16 Min Time to core damage = 50 Min Time to vessel breach = 3.1 Hrs	Time to Level 1 = 3 Min Time to -14 inches (TAF) = 8 Min Time to -39 inches = 13 Min Time to core damage = 45 Min Time to vessel breach = 2.5 Hrs
U2LOF2	Total loss of feedwater without injection, MSIVs close, open 2 SRVs at -39 inches and allow only one LPCI – evaluate timing to blow down and LPCI success	Time to Level 1 = 16 Min Time to -14 inches (TAF) = 23 Min Time to -39 inches = 29 Min 2 SRV + 1 LPCI is a success	Time to Level 1 = 13 Min Time to -14 inches (TAF) = 18 Min Time to -39 inches = 23 Min 2 SRV + 1 LPCI is a success
U2LOF3	Total loss of feedwater without CRD, RCIC, HPCS. Open 2 SRVs at -39 inches and allow only service water crosstie to RHR	See U2LOF2 for timing to -39 inches Core damage at 55 Min with 2 SRV + RHRSW (in vessel core recovery)	See U2LOF2 for timing to -39 inches Core damage at 48 Min with 2 SRV + RHRSW (in vessel core recovery)
U2LOF4	Station blackout with RCIC success – evaluate containment conditions that can impact RCIC operability	Time to HCTL = 10 Hrs Time to PSPL = 16.6 Hrs Time to 50 psig (back press) = 26.1 Hrs	Time to HCTL = 8.1 Hrs Time to PSP = 13.4 Hrs Time to 50 psig (back press) = 21.1 Hrs

ID (1)	Scenario Description & Objective	Pre-EPU Results	Post-EPU Results
U2LOF5	Station blackout with RCIC failure, blow down with all 7 ADS SRVs at TAF – evaluated time to core damage (shortest time for LOOP recovery)	Time to core damage = 47 Min	Time to core damage = 41 Min
U2MLOCAW2	3-inch Water LOCA, no high pressure injection, allow only 1 LPCI train Determine timing to -39 inches	Time to -39 inches = 7 Min Time to Core 1/3 covered = 13 Min Time to Core Damage = 32 Min	Time to -39 inches = 6 Min Time to Core 1/3 covered = 11 Min Time to Core Damage = 31 Min
U2AT1	Turbine trip ATWS with auto SLC, RPT, and Feedwater runback success. No RCIC and CRD (not credited in model) and no SPC, but MSIVs open. At TAF, control RPV level at 0.5 to 1 ft above TAF with feedwater – evaluate timing to containment conditions	Time to Level 1 = 2 Min Containment conditions that would require RPV blow down were not exceeded (power reduced below turbine bypass)	Time to Level 1 = 2 Min Containment conditions that would require RPV blow down were not exceeded (power reduced below turbine bypass)
U2AT2A	MSIVs close ATWS with auto SLC, RPT, and Feedwater runback success. No RCIC and CRD (not credited in model). At TAF, control RPV level at 0.5 to 1 ft above TAF with feedwater – evaluate HCTL prevention with both trains of SPC	Time to Level 1 = 2 Min To SP Temp > 90°F = 1 Min Time to HCTL = 34 Min	Time to Level 1 = 2 Min To SP Temp > 90°F = 1 Min Time to HCTL = 9 Min

ID (1)	Scenario Description & Objective	Pre-EPU Results	Post-EPU Results
U2AT3	Total loss of feedwater ATWS with auto SLC, RPT, and Feedwater runback success. No RCIC and no CRD (not needed). Open 3 SRVs at -39 inches and allow 1 LPCI pump injection – evaluate timing and 3 SRV success	Time to Level 1 = 1 Min Time to TAF = 1 Min Time to -39 inches = 1 Min 3 SRVs and LPCI is a success	Time to Level 1 = 1 Min Time to TAF = 1 Min Time to -39 inches = 1 Min 3 SRVs and LPCI is a success
U2INJ1	Total loss of feedwater, MSIVs closed, no HPI, depressurize at -39 inches with 3 SRVs, Inject with LPI, fail injection at 24 hours.	Time to Core Damage = 33.6 Hours	Time to Core Damage = 32.3 Hours

Notes:

(1) MAAP Case IDs are for the post-EPU runs. The pre-EPU cases have a "p" added at the end of the ID
(2) Core Damage based on PRA Update is 1800°F for 10 minutes

Table 2.13-5 Operator Action Review

Basic Event	Top Event & Scenario Type	Operator Action	Mean HEP	EPU Screening
ZAI01	AI – ATWS, RCIC success	Inhibit ADS	5.6E-3	ADS inhibit is an immediate response and considered important to ATWS response. These are relatively reliable actions, are not judged to be significantly impacted by EPU and are not risk significant.
ZAI02	AI - ATWS, RCIC unavailable	Inhibit ADS	5.6E-3	ADS inhibit is an immediate response and considered important to ATWS response. These are relatively reliable actions, are not judged to be significantly impacted by EPU and are not risk significant.
ZAS01	AS - support model, all initiators	Align standby air system pre-filter	0.1	This action relates to response following a support system malfunction. This system is not impacted by EPU and its PRA response remains unchanged. The timeframe for action is mainly driven by the time available prior to air system depressurization (i.e., capacity of receiver tank) and not by EPU conditions. However, this action indirectly supports the RPV inventory control function in that instrument air is required for feedwater system operation. EPU can compress the time that feedwater can be unavailable during transient response and in that regard EPU could increase the importance of this action. However, the focus of this action relates solely to operation of the instrument air system, which is not significantly affected by EPU.

Basic Event	Top Event & Scenario Type	Operator Action	Mean HEP	EPU Screening
ZAS02	AS - support model, all initiators	Align standby air system dryer bypass	0.1	This action relates to response following a support system malfunction. This system is not impacted by EPU and its PRA response remains unchanged. The timeframe for action is mainly driven by the time available prior to air system depressurization (i.e., capacity of receiver tank) and not by EPU conditions. However, this action indirectly supports the RPV inventory control function in that instrument air is required for feedwater system operation. EPU can compress the time that feedwater can be unavailable during transient response and in that regard EPU could increase the importance of this action. However, the focus of this action relates solely to operation of the instrument air system, which is not significantly affected by EPU.
ZAS03	AS - support model, all initiators	Align standby air system dryer	0.1	This action relates to response following a support system malfunction. This system is not impacted by EPU and its PRA response remains unchanged. The timeframe for action is mainly driven by the time available prior to air system depressurization (i.e., capacity of receiver tank) and not by EPU conditions. However, this action indirectly supports the RPV inventory control function in that instrument air is required for feedwater system operation. EPU can compress the time that feedwater can be unavailable during transient response and in that regard EPU could increase the importance of this action. However, the focus of this action relates solely to operation of the instrument air system, which is not significantly affected by EPU.

Basic Event	Top Event & Scenario Type	Operator Action	Mean HEP	EPU Screening
ZAS04	AS - support model, all initiators	Align standby air system valve train	0.1	This action relates to response following a support system malfunction. This system is not impacted by EPU and its PRA response remains unchanged. The timeframe for action is mainly driven by the time available prior to air system depressurization (i.e., capacity of receiver tank) and not by EPU conditions. However, this action indirectly supports the RPV inventory control function in that instrument air is required for feedwater system operation. EPU can compress the time that feedwater can be unavailable during transient response and in that regard EPU could increase the importance of this action. However, the focus of this action relates solely to operation of the instrument air system, which is not significantly affected by EPU.
ZAS05	AS - support model, all initiators	Align standby air system after filter	0.1	This action relates to response following a support system malfunction. This system is not impacted by EPU and its PRA response remains unchanged. The timeframe for action is mainly driven by the time available prior to air system depressurization (i.e., capacity of receiver tank) and not by EPU conditions. However, this action indirectly supports the RPV inventory control function in that instrument air is required for feedwater system operation. EPU can compress the time that feedwater can be unavailable during transient response and in that regard EPU could increase the importance of this action. However, the focus of this action relates solely to operation of the instrument air system, which is not significantly affected by EPU.

Basic Event	Top Event & Scenario Type	Operator Action	Mean HEP	EPU Screening
ZAS06	AS - support model, all initiators	Align standby RBCLC sub-loop	0.1	This action relates to response following a support system malfunction. This system is not impacted by EPU and its PRA response remains unchanged. The timeframe for action is mainly driven by the time available prior to air system depressurization (i.e., capacity of receiver tank) and not by EPU conditions. However, this action indirectly supports the RPV inventory control function in that instrument air is required for feedwater system operation. EPU can compress the time that feedwater can be unavailable during transient response and in that regard EPU could increase the importance of this action. However, the focus of this action relates solely to operation of the instrument air system, which is not significantly affected by EPU.
ZCF01	CF - loss of containment pressure control	Injection with external sources after containment failure	1E-2	This action is undertaken very late in accident response and involves controlling injection over the long-term. Enhanced decay heat loads related to EPU will have an inconsequential impact on timing for and content of tasks related to this action.
ZCH01	CH - ATWS, depressurized RPV	Terminate and prevent low pressure injection to prevent flushing out boron	4.6E-2	EPU power and decay heat increase can impact this HEP.
ZCH02	CH - ATWS, RPV not depressurized	Restart HPCS	1.0	This action is not credited in the PRA.
ZCI01	CI - loss of containment pressure control, injection from external sources stopped. (115 kV AC Avail)	Start HPCS from suppression pool	0.177	This action is undertaken very late in accident response and involves controlling injection over the long-term. Enhanced decay heat loads related to EPU will have an inconsequential impact on timing for and content of tasks related to this action.
ZCI02	CI - loss of containment pressure control, injection from external sources stopped. (115 kV AC Unavail)	Start HPCS from suppression pool	0.286	This action is undertaken very late in accident response and involves controlling injection over the long-term. Enhanced decay heat loads related to EPU will have an inconsequential impact on timing for and content of tasks related to this action.

Basic Event	Top Event & Scenario Type	Operator Action	Mean HEP	EPU Screening
ZCN01	*CN - Scram, MISVs don't* close	*Mode switch in shutdown to* prevent MSIV closure	3E-4	*This action is not impacted in terms* of content by EPU. Required manipulations remain unchanged. The time required for this action is driven chiefly by the timeframe established for cooldown as the MSIVs will not automatically close, with regard to mode switch position, until the RPV is depressurized. The timeframe for cooldown is not expected to change significantly during post EPU conditions. Also, this is an immediate, post SCRAM operator action that is completed within the first minute of a transient. EPU conditions will not significantly alter the performance of this action.
ZCN02	CN - MSIV closure during transient	Reopen MSIV to provide path to condenser	7E-2	This action is not impacted in terms of content by EPU. Required manipulations remain unchanged. The time required for this action is driven chiefly by the timeframe that the condenser vacuum can be maintained with the MSIVs closed. This is not expected to be significantly impacted by EPU.
ZCN03	CN - MSIV ATWS event	Reopen MSIV to provide path to condenser	1.0	This action is not credited in the PRA.
ZCR01	CR - Control room fire, major impact	Operator stays in control room	0.5	Whether operators stay in the control room or not depends on the fire not EPU change. Timing of the scenario can be impacted by EPU but this contribution is minor compared to the significance of a control room fire.
ZCR02	CR - Control room fire, minor BOP impact	Operator stays in control room	0.1	Whether operators stay in the control room or not depends on the fire not EPU change. Timing of the scenario can be impacted by EPU but this contribution is minor compared to the significance of a control room fire.
ZCRD1	CRD – Control Rod Drive	Operator Fails to Align Standby CRD Train	0.1	EPU can compress the time that high pressure injection can be unavailable during transient response and in that regard EPU could increase the importance of this action. However, early in event response CRD is a minimal benefit because of its low flowrate. The benefit of CRD is over the longer term when decay heat loads have reduced and containment challenges are considered. EPU will have an *inconsequential impact* of actions and functions in the longer-term phase of severe accident response.
ZCV01	CV - Loss of RHR, all support systems available	Vent containment	6E-3	This action is required many hours after plant trip and will not be impacted.

Basic Event	Top Event & Scenario Type	Operator Action	Mean HEP	EPU Screening
ZCV02	CV - Loss of RHR, no instrument air or AC	Vent containment	1.4E-2	This action is required many hours after plant trip and will not be impacted.
ZCV03	CV - Loss of RHR, no instrument nitrogen	Vent containment	8.7E-3	This action is required many hours after plant trip and will not be impacted.
ZCV04	CV – Operator Fails to Vent During SBO	Vent Containment	0.5	This action is required many hours after plant trip and will not be impacted.
ZD101	D1 - support model, all initiators	Align standby charger	1.0	This action relates to response following a support system malfunction. This system is not impacted by EPU and its PRA response remains unchanged. The timeframe for action is mainly driven by the capacity of the batteries and EPU will not cause a significant impact.
ZEC01	ECV - Test and maintenance	Miscalibration of ECCS actuation	1E-5	This is a pre-initiator calibration activity that is not impacted by EPU.
ZFW01	FW - ATWS, feedwater runback	Restore motor driven feedwater before MSIVs go closed on low level	0.5	EPU power and decay heat increase can impact this HEP
ZFW02	FW - Small LOCA	Restore motor driven feedwater	5.5E-3	This action involves restoration of feedwater during a small LOCA following a postulated high RPV level trip. In such a condition, decay heat, which can be enhanced by EPU, is not a significant concern compared to the inventory loss of the LOCA. Thus, EPU has a minimal impact on this action.
ZHA01	HA & HB - Failure of RHR HE valve	Manually open valve	1E-2	This action involves responding to equipment failure related to residual heat removal and suppression pool cooling. This action is required only after several hours and increased decay heat loads from EPU have a minor impact on this action.
ZHR01	HRA - Fire in control room, recovery in CR with minor damage	Inventory control and long term heat removal	1E-3	Applies to cases where operators remain in the control room and RCIC or HPCS is successful, therefore the timing is not demanding and impact is judged minimal.
ZHR02	HRA - Fire in control room, recovery in CR with loss of RCIC & HPCS or recovery from RSP within procedures	Inventory control and long term heat removal	1E-2	Most demanding time is HPCS and RCIC unavailable with operators in control room, which could be impacted by EPU.

Basic Event	Top Event & Scenario Type	Operator Action	Mean HEP	EPU Screening
ZHR03	HRA - Fire in control room, recovery in CR with loss of RCIC, HPCS & support systems or recovery from RSP with RCIC unavailable	Inventory control and long term heat removal	0.04	Most demanding time is HPCS and RCIC unavailable with operators at remote shutdown, which could be impacted by EPU
XHS01	HS/HPCS - misalignment	2CSH*HCV120 misaligned	1E-3	This is a pre-initiator action not impacted by EPU.
ZHS01	HS: HPCS - Station Blackout, RCIC failed	Align HPCS EDG to Div 1 or 2 Bus (RCIC Failed)	0.02	Since timing will be based on HPCS operation until Level 8 or high EDG temperature, timing is approximately 1 hr (less time critical). This action has a low importance in the PRA.
ZHS02	HS: HPCS - Station Blackout, RCIC success or Early CSD, DC Load Shed=S	Align HPCS EDG to Div 1 or 2 Bus	0.01	With RCIC success, timing is driven by battery capacity and EPU is a minimal impact.
ZHS03	HS: RCIC Success or Plant S/D, DC Load Shed Failed	Align HPCS EDG to Div 1 or 2 Bus	3.5E-3	With RCIC success, timing is driven by battery capacity and EPU is a minimal impact. Even with load shed failed, the action is not driven by EPU impacts given RCIC in operation.
ZHS04	HS: RCIC Success or Plant S/D, DC Load Shed Success	Align HPCS EDG to Div 1 or 2 Bus	1.1E-3	With RCIC success, timing is driven by battery capacity and EPU is a minimal impact.
XIA01	IA – misalignment	2RHS*HCV53A misalign	1E-3	This is a pre-initiator action not impacted by EPU.
XIB01	IB – misalignment	2RHS*HCV53B misalign	1E-3	This is a pre-initiator action not impacted by EPU.
ZIC01	IC - loss of RHR room cooling	Bypass RCIC trip on high area temp	0.1	This action involves response to loss of room cooling. Room heatup is not expected to be impacted by EPU and the scope and timing of this action will not be significantly impacted.
ZIC02	IC - level 8 trip failure	Prevent overfill	0.8	This action is essentially not credited in the PRA.
ZIC03	IC - Loss of room cooling	Prevent (open door) or recover from RCIC isolation on high room temperature	0.1	This action involves response to loss of room cooling. Room heatup is not expected to be impacted by EPU and the scope and timing of this action will not be significantly impacted.
ZIE01	IER - Initiating event recovery	Isolate FLCB initiator before flooding all AC power in control building	1E-3	Timing is dependent on the flood rate not EPU change
ZIE02	IER - Initiating event recovery	Prevent plant trip given partial LOOP	0.1	This is an immediate operator action to prevent a plant trip given a partial LOOP; HEP not sensitive to EPU changes due to TBCCW/RBCLC margins.

Basic Event	Top Event & Scenario Type	Operator Action	Mean HEP	EPU Screening
ZIL01	IL - ATWS, depressurization required	Override RCIC low pressure trip	0.57	This task is challenged after the RPV is depressurized. EPU conditions could compress the time for this action but the impact is minor.
ZIS01	Level II IS - Station Blackout	Locally isolate outside MOVs	0.1	This is a Level 2 PRA action that involves manual containment isolation in a station blackout. Failure probability for this action is driven by radiation challenges and resource commitments. EPU primarily impacts timing related to this action and this action is not particularly time-sensitive.
ZKR01	KR - Loss of offsite source A	Cross-connect remaining source to bus A	2E-2	The timing limiting case is in support of RPV injection therefore; impact would be similar to emergency depressurization (ZOD).
ZKR03	KR - Loss of offsite source A or B	Cross-tie 115Kv	1E-2	The timing limiting case is in support of RPV injection therefore; impact would be similar to emergency depressurization (ZOD).
XLC01	LC – misalignment	2RHS*HCV53C misalign	1E-3	This is a pre-initiator action not impacted by EPU.
XLS01	LS – misalignment	2CSL*HCV117 misalign	1E-3	This is a pre-initiator action not impacted by EPU.
ZME01	ME - Transient or small LOCA, auto-initiation of ECCS fails	Manually initiate ECCS	0.1	This action applies to low pressure injection systems if automatic action fails. This is a simple, high profile action whose timeframe will be compressed by EPU. However, because the action is not time-challenged, the impact of EPU is minor.
ZMO01	MO – ATWS	Bypass MSIV closure signal	1.0	Not credited in the PRA.
ZMS01	MS – ATWS	Put mode switch in shutdown	3E-4	This action is not impacted in terms of content by EPU. Required manipulations remain unchanged. The time required for this action is driven chiefly by the timeframe established for cooldown as the MSIVs will not automatically close, with regard to mode switch position, until the RPV is depressurized. The timeframe for cooldown is not expected to change significantly during post EPU conditions. Also, this is an immediate, post SCRAM operator action that is completed within the first minute of a transient. EPU conditions will not significantly alter the performance of this action.

Basic Event	Top Event & Scenario Type	Operator Action	Mean HEP	EPU Screening
ZN201	N2 - support model, all initiators	Align standby vaporizer	0.1	This action relates to response following a support system malfunction. This system is not impacted by EPU and its PRA response remains unchanged. The timeframe for action is mainly driven by the time available prior to system depressurization (i.e., capacity of receiver tank) and not by EPU conditions. This action does not support the RPV inventory control function and EPU driven time-compression is not a factor.
ZN202	N2 - support model, all initiators	Align standby trim heater	0.1	This action relates to response following a support system malfunction. This system is not impacted by EPU and its PRA response remains unchanged. The timeframe for action is mainly driven by the time available prior to system depressurization (i.e., capacity of receiver tank) and not by EPU conditions. This action does not support the RPV inventory control function and EPU driven time-compression is not a factor.
ZN203	N2 - support model, all initiators	Align valve bypass	0.1	This action relates to response following a support system malfunction. This system is not impacted by EPU and its PRA response remains unchanged. The timeframe for action is mainly driven by the time available prior to system depressurization (i.e., capacity of receiver tank) and not by EPU conditions. This action does not support the RPV inventory control function and EPU driven time-compression is not a factor.
ZN204	N2 - Failure of normal nitrogen supply	Valve in high pressure nitrogen system	3.6E-3	This action relates to response following a support system malfunction. This system is not impacted by EPU and its PRA response remains unchanged. The timeframe for action is mainly driven by the time available prior to system depressurization (i.e., capacity of receiver tank) and not by EPU conditions. This action does not support the RPV inventory control function and EPU driven time-compression is not a factor.
ZUS03	NA/NB/OB - support model, all initiators	Swap power supply	1.0	The timing limiting case is in support the RPV injection therefore; impact would be similar to emergency depressurization (ZOD). However, this action is not credited.

Basic Event	Top Event & Scenario Type	Operator Action	Mean HEP	EPU Screening
ZNR01	NR - support model, all initiators	Cross-tie 115Kv to NSR bus	0.1	The timing limiting case is in support the RPV injection therefore; impact would be similar to emergency depressurization (ZOD). This action has a low importance in the PRA.
ZOA01	OA - Station Blackout	Bypass RCIC isolations (15 minutes)	3.3E-3	The timing limiting case is in support the RPV injection therefore; impact would be similar to emergency depressurization (ZOD). This action has a low importance in the PRA.
ZOA02	OA - Station Blackout	Shed DC loads (2 hours)	3.3E-3	This action is driven by battery capacity and is not impacted by EPU conditions.
ZOA03	OA - Station Blackout	Open RCIC room door (2 hours)	3.3E-3	EPU is not expected to impact RCIC room heatup in SBO
ZOB01	OB - support model, all initiators	Recover swing bus	0.1	The timing limiting case is in support the RPV injection therefore; impact would be similar to emergency depressurization (ZOD). This action has a low importance in the PRA.
ZOCV1	OCV: SBO	Operators recover nitrogen supply to inside Containment Vent AOVs by re-alignment of SOV supply power	1E-2	Containment venting is a long-term function that is not significantly impacted by EPU conditions.
ZOD01	OD – Transient, small LOCA or SBO, loss of high pressure injection	Manually emergency depressurize	1.1E-3	EPU power and decay heat increase can impact this HEP
ZOD02	OD - Medium LOCA, loss of high pressure injection	Manually emergency depressurize	3E-3	EPU power and decay heat increase can impact this HEP
ZOD05	OD - ISLOCA and Seismic, loss of high pressure injection	Manually emergency depressurize	1E-2	EPU power and decay heat increase can impact this HEP
ZODA	OD - ATWS, loss of high pressure injection	Manually emergency depressurize	0.16	EPU power and decay heat increase can impact this HEP
ZOH01	OH - Transient/LOCA	Align containment heat removal	1E-5	Containment heat removal is a long-term function that is not significantly impacted by EPU conditions.
ZOH02	OH – ATWS	Align containment heat removal	9.6E-3	Containment heat removal is a long-term function that is not significantly impacted by EPU conditions.
ZOPCH1	OPCH: SBO	Align portable charger w/in 30 min	0.9	This is not credited in the PRA.
ZOPCH2	OPCH: SBO	Align portable charger w/in 2 hours	1E-2	This is not credited in the PRA.

Basic Event	Top Event & Scenario Type	Operator Action	Mean HEP	EPU Screening
ZOSVL1	OSVL: SBO	Operators recover nitrogen supply to inside SRV accumulators by re-alignment of SOV supply power	1E-2	This is a recovery useful only over the long-term function and is not significantly impacted by EPU conditions.
ZOV01	OV – Small/Medium LOCA, vapor suppression system fails	Initiate containment sprays	4E-2	EPU power and decay heat increase can impact this HEP. This action has a low importance in the PRA.
ZRW01	RW - support model, all initiators	Align standby RBCLC heat exchanger	0.05	This action relates to response following a support system malfunction. This system is not impacted by EPU and its PRA response remains unchanged. This action indirectly supports the RPV inventory control function in that RBCLC is required for CRD system operation. However, CRD is useful only over the long-term (see above) and EPU is not expected to be a significant impact.
ZS101	S1 - Station Blackout (short term)	Align fire water pump for injection	0.2	Not credited in the PRA.
ZS102	S1 - Station Blackout (Longer term)	Align fire water pump for injection	5E-2	This action applies in the longer-term such that EPU impacts are minor.
XSL01	SL - Test and maintenance	Misalign SLS valves after maintenance	3E-3	Pre-initiator action not affected by EPU.
ZSV01	SV – long term N2 makeup	Locally align N2 to SRVs	0.1	This action applies in the longer-term such that EPU impacts are minor.
ZSW01	SW - Loss of all ECCS	Align service water pump to LPCI discharge for low pressure injection	4E-2	This action could be impacted by the compressed boiloff time related to EPU. However, this applies only following massive coincident equipment failures such that it is not a significant PRA action.
ZSA01	SWA, SWB, SWAB - Service water transients due to loss of one or more pumps, cross-tie inadvertently closes, or loss of support system	Start pump E of F when required, isolate RBCLC/TBCCW loads, control service water flow to prevent low flow or runout, and restart tripped pumps.	2E-4	This action is driven by the time to heat component cooling systems. EPU is expected to have a minor impact on this action.
ZTW01	TW - support model, all initiators	Align standby TBCCW heat exchanger	5E-2	This action is driven by the time to heat component cooling systems. EPU is expected to have a minor impact on this action.

Basic Event	Top Event & Scenario Type	Operator Action	Mean HEP	EPU Screening
ZTW02	TW - support model, all initiators	Open alternate TBCCW valve	0.1	This action is driven by the time to heat component cooling systems. EPU is expected to have a minor impact on this action.
ZUS03	US1/US3 - support model, all initiators	Align backup power supply	1.0	This action is driven by the time to heat component cooling systems. EPU is expected to have a minor impact on this action.
CZHUPOOL-24X	Level II CZ & CE - pre-accident	Operating with normal pool level	1E-5	This is a Level 2 post core melt action that applies in the longer term and is not expected to be significant impacted by EPU.
CZHUCOINJ00X	Level II CZ & CE - core melt progression	Restores cooling injection after control rods are melted	1E-4	This is a Level 2 post core melt action that applies in the longer term and is not expected to be significant impacted by EPU.
CTHUPHSLC01X	Level II CZ & CE - core melt progression	Inject SLS with boron for low water level	1.0	This is a Level 2 post core melt action that applies in the longer term and is not expected to be significant impacted by EPU.
FDHUDWVP-00X	Level II FD & FB - containment flood scenario	RPV vent - align equipment	0.1	This is a Level 2 post core melt action that applies in the longer term and is not expected to be significant impacted by EPU.
CPSZCLXXXUNI	Level II FD & FB - containment flood scenario	Unisolate containment flood instruments	2E-2	This is a Level 2 post core melt action that applies in the longer term and is not expected to be significant impacted by EPU.
FDHUSUSFL00X	Level II FD & FB - containment flood scenario	Drywell vent, suspends flooding to erroneous indications	1.0	This is a Level 2 post core melt action that applies in the longer term and is not expected to be significant impacted by EPU.
FDHUCCAL-00X	Level II FD & FB - containment flood scenario	Drywell vent, recognizes MPCWLL	1E-3	This is a Level 2 post core melt action that applies in the longer term and is not expected to be significant impacted by EPU.
MSSZRVACTION	Level II FD & FB - containment flood scenario	RPV vent - override isolation signals	Note 1	This is a Level 2 post core melt action that applies in the longer term and is not expected to be significant impacted by EPU.
MSSZRVACTION1 X02	Level II FD & FB - containment flood scenario	RPV vent - engage breaker in N. Aux bay for 2MSS*MOV112	Note 1	This is a Level 2 post core melt action that applies in the longer term and is not expected to be significant impacted by EPU.
ZFB01 ZFD01	Level II FD & FB - containment flood scenario	Drywell vent	0.1	This is a Level 2 post core melt action that applies in the longer term and is not expected to be significant impacted by EPU.
FIHUNDEOP00X	Level II FI & FC - core not contained in-vessel	Flood containment	0.1	This is a Level 2 post core melt action that applies in the longer term and is not expected to be significant impacted by EPU.
FIHUCLVNT--X	Level II FI & FC - core not contained in-vessel	Flood containment - close wetwell vent	1E-2	This is a Level 2 post core melt action that applies in the longer term and is not expected to be significant impacted by EPU.
GVHUH202--X	Level II GV - de-inerted containment	Staff checks H2/O2 indication	Note 1	This is a Level 2 post core melt action that applies in the longer term and is not expected to be significant impacted by EPU.
GVHUSAMPLE	Level II GV - de-inerted containment	Staff confirms grab sample	1.0	This is a Level 2 post core melt action that applies in the longer term and is not expected to be significant impacted by EPU.
GVHUBP253--X & GVHUBP254--X	Level II GV - de-inerted containment	Start analyzer pump A/B	Note 1	This is a Level 2 post core melt action that applies in the longer term and is not expected to be significant impacted by EPU.

Basic Event	Top Event & Scenario Type	Operator Action	Mean HEP	EPU Screening
FDHUDWVP-00X	Level II GV	Align drywell venting	Note 1	This is a Level 2 post core melt action that applies in the longer term and is not expected to be significant impacted by EPU.
ZGV01 (GVHUVENT-24X)	Level II GV - de-inerted containment	Initiate purge - combustible gas control	1E-2	This is a Level 2 post core melt action that applies in the longer term and is not expected to be significant impacted by EPU.
HRHURHRCL00X	Level II HR - Class I, Class IIIB, Class IIIC	Initiate suppression pool cooling	1E-6	This is a Level 2 post core melt action that applies in the longer term and is not expected to be significant impacted by EPU.
RXHUEXCI-00X	Level II IR & RX - Class I, IIIB, IIIC	Intervenes and terminates injection	1E-4	This is a Level 2 post core melt action that applies in the longer term and is not expected to be significant impacted by EPU.
RXHUHWINJ24X	Level II IR & RX - All Classes	Makeup to condenser hotwell	0.1	This is a Level 2 post core melt action that applies in the longer term and is not expected to be significant impacted by EPU.
MUPHNOSWT00X	Level II MU - maintain makeup	Switch injection to alternate outside RB	1.0	This is a Level 2 post core melt action that applies in the longer term and is not expected to be significant impacted by EPU.
OIHUB4RPV00X	Level II OI & OP - Class IA, IC	Emergency depressurize during in-vessel core degradation	0.13	This is a Level 2 action that is similar to the Level 1 Emergency depressurization action. This EPU impact will be similar to that noted for the related Level 1 action.
OIHUNODPR00X	Level II OI & OP - Class ID	Maintain already successful depressurization	1E-3	This is a Level 2 post core melt action that applies in the longer term and is not expected to be significant impacted by EPU.
RNHUNOIS00X	Level II RB & RM - RB effectiveness	Several human actions associated with submerging break in RB, tree set to 1.0	1.0	This is a Level 2 post core melt action that applies in the longer term and is not expected to be significant impacted by EPU.
TDNUPROCD24X	Level II TD & TR - core melt progression	Procedure precludes use of drywell spray	1.0	This is a Level 2 post core melt action that applies in the longer term and is not expected to be significant impacted by EPU.
TDHURSWDN24X	Level II TD & TR - core melt progression	Align RHRSW to drywell spray	1.0	This is a Level 2 post core melt action that applies in the longer term and is not expected to be significant impacted by EPU.
TDHUHPREC24X	Level II TD & TR - long term recovery	Recover high pressure injection	0.9	This is a Level 2 post core melt action that applies in the longer term and is not expected to be significant impacted by EPU.
TDHULPREC24X	Level II TD & TR - long term recovery	Recover low pressure injection	0.9	This is a Level 2 post core melt action that applies in the longer term and is not expected to be significant impacted by EPU.
TDHURHRSW24X	Level II TD & TR - core melt progression	Provide makeup to RPV using RHRSW	0.1	This is a Level 2 post core melt action that applies in the longer term and is not expected to be significant impacted by EPU.
VCNUNDREC00X	Level II VC - severe accident overpressure	Recognizes symptoms of severe accident	Note 1	This is a Level 2 post core melt action that applies in the longer term and is not expected to be significant impacted by EPU.
VCHUNDEP00X	Level II VC - severe accident overpressure	Interprets EOP instructions	Note 1	This is a Level 2 post core melt action that applies in the longer term and is not expected to be significant impacted by EPU.
VCHUINHES00X	Level II VC - severe accident overpressure	Does not hesitate	Note 1	This is a Level 2 post core melt action that applies in the longer term and is not expected to be significant impacted by EPU.

Basic Event	Top Event & Scenario Type	Operator Action	Mean HEP	EPU Screening
ZVC01 (VCHUNDIMP00X)	Level II VC - severe accident overpressure	Align containment vent	1E-2	This is a Level 2 post core melt action that applies in the longer term and is not expected to be significant impacted by EPU.

Note 1: The basic events included in Level 2 documentation. However, it was not used in the quantification of accident sequences.

Table 2.13-6 Operator Action Screening

Basic Event	Description	FV	HEP Timing Evaluation
MSSZODMSSOP1OO01	Emergency Depressurization - Tran	0.32	EPU power and decay heat increase can impact this HEP
CPSZCVXXXXXXXX02	Containment Vent - Air Lost	0.07	This action is required many hours after plant trip and will not be impacted
ICSZMEECCSACTX01	Manual ECCS Actuation	0.05	This action applies to low pressure injection systems if automatic action fails
IERZIECBFLOODX01	Recover From Flood in Control Bldg	0.05	Timing is dependent on the flood rate not EPU change
MSSZODMSSOP1OO02	Emergency Depressurization - MLOCA	0.03	EPU power and decay heat increase can impact this HEP
A3_OPSEDG	Start EDG	0.03	This is an immediate recovery action not highly sensitive to EPU changes
ENSZKROPERSWAP01	Crosstie 115kv After Partial Loss	0.02	The timing limiting case is in support the RPV injection therefore; impact would be similar to emergency depressurization above.
HRAZHRBOPSRICS02	Recover From Fire in Control Room	0.02	Most demanding time is HPCS and RCIC unavailable with operators in control room (similar to emergency depressurization above)
H_OPSRCIC	Start RCIC pump	0.02	This is an immediate recovery action not highly sensitive to EPU changes
SWPZSAHHSWSOOO01	Service Water Control	0.02	This is an immediate response to partial service water failures; HEP not sensitive to EPU changes as TBCCW/RBCLC margins significant
IERZIEKABXRECV02	Prevent Trip After Partial 115 kV Loss	0.02	This is an immediate operator action to prevent a plant trip given a partial LOOP; HEP not sensitive to EPU changes due to TBCCW/RBCLC margins
HRAZHRBOPSRICF03	Recover From Fire in Control Room	0.02	Most demanding time is HPCS and RCIC unavailable with operators at remote shutdown (similar to emergency depressurization above)
ZCRZCROPACTCR202	Operators Stay in CR - Fire in CR BOP	0.02	Whether operators stay in the control room or not depends on the fire not EPU change
FWSZFWXXXXXXXX01	Restore Feedwater - ATWS	0.02	EPU power and decay heat increase can impact this HEP
ICSZICHBYPTEMP01	Prevent RCIC Trips - High Area Temp	0.01	SOP for SBO requires action in 15 min, but actual heat up to trip set point is longer and EPU does not affect RCIC room heat up
H_OPSHPCS	Start HPCS pump	0.01	This is an immediate recovery action not highly sensitive to EPU changes
CPSZCVXXXXXXXX01	Containment Vent - All Support Avail	0.01	This action is required many hours after plant trip and will not be impacted
RPSZCHXXXXXXXX01	Control Level & Power - ATWS	0.01	EPU power and decay heat increase can impact this HEP
ICSZICOPENDOOR03	Prevent RCIC Trips - High Area Temp	0.01	SOP for SBO requires RCIC room door to be open in 2 hrs. EPU change will not impact this HEP.

Basic Event	Description	FV	HEP Timing Evaluation
HRAZHRBOPFIRE001	Recover From Fire in Control Room	0.01	Applies to cases where operators remain in the control room and RCIC or HPCS is successful, therefore the timing is not demanding.
DERZISXXXXXXXX01	Containment Isolation	LERF	Timing to core damage and isolation is longer and not significantly impacted
RPSZCHXXXXXXXX02	Control Level & Power - ATWS	LERF	This HEP is set to guaranteed failure in PRA

Table 2.13-7 Operator Action Evaluation

Basic Event	Description	MAAP Case (Table 2.13-4)	Pre-EPU		Post-EPU	
			Time	HEP	Time	HEP
MSSZODMSSOP1OO01	Emergency Depressurization – Transient/SLOCA (1)	U2LOF1	40 Min	2.8E-4	35 Min	3.2E-4 (2)
ICSZMEECCSACTX01	Manual ECCS Actuation	U2LOF1	13 Min	0.1 (3)	11 Min	0.1 (3)
MSSZODMSSOP1OO02	Emergency Depressurization – MLOCA (1)	U2MLOCAW2	13 Min	1.8E-3	11 Min	2.1E-3 (2)
ENSZKROPERSWAP01	Crosstie 115kv After Partial Loss (5)	U2LOF1	40 Min	4.7E-2	35 Min	5.4E-2 (2)
HRAZHRBOPSRICS02	Recover From Fire in Control Room (6)	U2LOF1	40 Min	2.0E-3	35 Min	2.3E-3 (2)
HRAZHRBOPSRICF03	Recover From Fire in Control Room at RSP (6)	U2LOF1	40 Min	1.2E-2	35 Min	4.4E-2
FWSZFWXXXXXXXX01	Restore Feedwater – ATWS (7)	U2AT3	2.5 Min	0.41	2.5 Min	0.41 (4)
RPSZCHXXXXXXXX01	Control Level & Power – ATWS	U2AT3	6 Min	3.3E-2	6 Min	3.3E-2 (4)
DERZISXXXXXXXX01	Fail to Isolate Containment – Level 2 LERF	U2LOF1	50 Min	8.2E-2	45 Min	8.4E-2

Notes:

(1) Timing is conservatively based on no high pressure injection makeup at start of event.

(2) This post EPU HEP was identical to the pre EPU HEP as the action was required in the same discrete time step. The post EPU HEP was increased by 15% to evaluate the sensitivity of reducing the time within that time step.

(3) This HEP is less than 1E-2 for both pre and post EPU. The 0.1 value has been used in the baseline PRA because of equipment dependencies where auto actuation failure may be non-recoverable. Since this is not affected by EPU there is no change in risk.

(4) There was no change in HEP thus no change in risk

(5) Top event KR split fractions were calculated and used to update master frequency files

(6) Several scenarios and system dependencies are affected, but the timing is conservatively based on loss of high pressure injection.

(7) Time to Level 1 is used as restoring feedwater before Level 1 allows operators to bypass MSIV Level 1 trip (not presently credited in the PRA) if this action is successful. Note that time to -39 inched (blow down required) could be used given present model.