AmerGen

An Exelon/British Energy Company

Clinton Power Station
R.R. 3 Box 228
Clinton, IL 61727-9351
Phone: 217 935-8881

RS-01-241

October 23, 2001

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

Clinton Power Station, Unit 1
Facility Operating License No. NPF-62
NRC Docket No. 50-461

Subject: Non-Proprietary Safety Analysis Report Supporting the License Amendment Request to Permit Uprated Power Operation at Clinton Power Station

Reference: Letter from J. M. Heffley (AmerGen Energy Company, LLC) to U. S. NRC, "Request for License Amendment for Extended Power Uprate Operation," dated June 18, 2001

In the referenced letter, AmerGen Energy Company, LLC (i.e., AmerGen) submitted a request for changes to the Facility Operating License No. NPF-62 and Appendix A to the Facility Operating License, Technical Specifications (TS), for Clinton Power Station (CPS) to allow operation at an uprated power level. The proposed changes in the referenced letter would allow CPS to operate at a power level of 3473 megawatts thermal (MWt). This represents an increase of approximately 20 percent rated core thermal power over the current 100 percent power level of 2894 MWt.

In support of the proposed changes, the referenced letter contained a safety analysis report supplied by General Electric (GE) Company and requested that this safety analysis report be withheld from public disclosure in accordance with 10 CFR 2.790(a)(4), "Public inspections, exemptions, requests for withholding." Following a request from the NRC Staff, we indicated that a non-proprietary version of the safety analysis report suitable for placement in the Public Document Room would be provided separately. Attachment A to this letter provides the non-proprietary version of the safety analysis report for CPS.

A001
Rec'd from NRC
01/31/02

Should you have any questions concerning this letter, please contact Mr. T. A. Byam at (630) 657-2804.

Respectfully,

K. A. Ainger

K. A. Ainger
Director – Licensing
Mid-West Regional Operating Group

Attachment:

Attachment A: GE Report NEDO-32989, Revision 0, "Safety Analysis Report for Clinton Power Station Extended Power Uprate," dated October 2001 (Non-Proprietary)

cc: Regional Administrator – NRC Region III
NRC Senior Resident Inspector – Clinton Power Station
Office of Nuclear Facility Safety – Illinois Department of Nuclear Safety

Attachment A

GE Report NEDO – 32989, Revision 0, "Safety Analysis Report For Clinton Power Station Extended Power Uprate," Dated October 2001 (Non-Proprietary)

GE Nuclear Energy

175 Curtner Ave., San Jose, CA 95125

NEDO-32989
Revision 0
DRF A22-00110-70
Class I
October 2001

Safety Analysis Report
For
Clinton Power Station
Extended Power Uprate

Prepared by: E. D. Schrull

Approved by: Erik Stromqvist

Erik Stromqvist, Project Manager
General Electric Company

Approved by: DC Spencer

Dale Spencer, Project Manager
Clinton Power Station

IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT

Please Read Carefully

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ACRONYMS AND ABBREVIATIONS

Term	Definition
AC	Alternating Current
ADS	Automatic Depressurization System
APRM	Average Power Range Monitor
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient Without Scram
B&PV	Boiler and Pressure Vessel
BHP	Brake Horsepower
BOP	Balance-of-plant
BWR	Boiling Water Reactor
CCW	Component Cooling Water
CFR	Code of Federal Regulations
COLR	Core Operating Limits Report
CPD	Condensate Polishing Demineralizer
CPR	Critical Power Ratio
CPS	Clinton Power Station
CRD	Control Rod Drive
CRDA	Control Rod Drop Accident
DBA	Design Basis Accident
DC	Direct Current
ECCS	Emergency Core Cooling System
EHC	Electro-hydraulic Control
EPP	Environmental Protection Plan
EPU	Extended Power Uprate
EQ	Environmental Qualification
ER/OL	Environmental Report, Operating License Stage
FAC	Flow Assisted Corrosion
FES	Final Environmental Statement
FFWTR	Final Feedwater Temperature Reduction

Term	Definition
FPCCS	Fuel Pool Cooling and Cleanup System
FCS	Feedwater Control System
FHA	Fuel Handling Accident
FWHOOS	Feedwater Heater Out-of-Service
GDC	General Design Criterion
GE	General Electric Company
GL	Generic Letter
HELB	High Energy Line Break
HEPA	High Efficiency Particulate Adsorber
HPCS	High Pressure Core Spray
HPSP	High Power Setpoint
HVAC	Heating, Ventilating and Air Conditioning
ICA	Interim Corrective Action
ICF	Increased Core Flow
IPE	Individual Plant Evaluation
IRM	Intermediate Range Monitor
LCO	Limiting Condition for Operation
LOCA	Loss-of-Coolant Accident
LOFW	Loss of Feedwater Flow
LPCI	Low Pressure Coolant Injection
LPCS	Low Pressure Core Spray
LPSP	Low Power Setpoint
MCPR	Minimum Critical Power Ratio
MELB	Moderate Energy Line Break
MELLLA	Maximum Extended Load Line Limit Analysis
MEOD	Maximum Extended Operating Domain
Mlb/hr	Million Pounds Per Hour
MOV	Motor Operated Valve
MSIV	Main Steam Isolation Valve
MSIVLCS	Main Steam Line Isolation Valve Leakage Control System
MSLBA	Main Steam Line Break Accident

Term	Definition
MWe	Megawatt-electric
MWt	Megawatt-thermal
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
OLMCPR	Operating Limit Minimum Critical Power Ratio
OLTP	Original Rated Thermal Power
OOS	Out of Service
OPRM	Oscillation Power Range Monitor
PCS	Pressure Control System
PCT	Peak Clad Temperature
PRA	Probabilistic Risk Assessment
PSA	Probabilistic Safety Assessment
QA	Quality Assurance
RBM	Rod Block Monitor
RCIC	Reactor Core Isolation Cooling
RCPB	Reactor Coolant Pressure Boundary
RG	Regulatory Guide
RHR	Residual Heat Removal
RIPD	Reactor Internal Pressure Difference
RPS	Reactor Protection System
RPT	Recirculation Pump Trip
RPV	Reactor Pressure Vessel
RRS	Reactor Recirculation System
RTP	Rated Thermal Power
RWCU	Reactor Water Cleanup
RWE	Rod Withdrawal Error
SBO	Station Blackout
SFP	Spent Fuel Pool
SGTS	Standby Gas Treatment System
SJAE	Steam Jet Air Ejectors

Term	Definition
SLCS	Standby Liquid Control System
SLMCPR	Safety Limit Minimum Critical Power Ratio
SLO	Single-loop Operation
SRM	Source Range Monitor
SRV	Safety/Relief Valve
SRVDL	Safety/Relief Valve Discharge Line
TAF	Top of the Active Fuel
TBCCW	Turbine Building Closed Cooling Water
TCV	Turbine Control Valve
TLO	Two (recirculation) Loop Operation
TSV	Turbine Stop Valve
USAR	Updated Safety Analysis Report
UHS	Ultimate Heat Sink

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EXECUTIVE SUMMARY

This report summarizes the results of all significant safety evaluations performed that justify extending the licensed thermal power at Clinton Power Station (CPS) to 3473 MWt. The requested license power level is approximately 20% above the Original Licensed Thermal Power (OLTP) of 2894 MWt.

Uprating the power level of nuclear power plants can be done safely within certain plant-specific limits and is a cost effective way to increase installed electrical generating capacity. An increase in electrical output of a General Electric (GE) Boiling Water Reactor (BWR) plant is accomplished primarily by generation and supply of higher steam flow to the turbine generator. The modified high-pressure turbines at CPS were designed to accommodate the increased steam flow at extended power uprate (EPU) conditions with adequate pressure control margin without increasing the maximum operating reactor vessel dome pressure.

Detailed evaluations of the reactor, engineered safety features, power conversion, emergency power, support systems, environmental issues, design basis accident analyses and previous licensing evaluations were performed.

This report supports the conclusion that this EPU 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.

1. INTRODUCTION AND SUMMARY

1.1 Introduction

Uprating the power level of nuclear power plants can be done safely within certain plant-specific limits. Most General Electric (GE) Boiling Water Reactor (BWR) plants, including the Clinton Power Station (CPS), have the capability and margins for a power uprate of up to 20% without major Nuclear Steam Supply System (NSSS) hardware modifications.

The evaluation presented in this report justifies an extended power uprate (EPU) to 3473 MWt, which corresponds to 120% of the Original Licensed Thermal Power (OLTP) level of 2894 MWt. The generic criteria, process, and scope of work required to provide sufficient information for use by the Nuclear Regulatory Commission (NRC) to grant approval to specific applications for increases in the authorized thermal power levels for GE BWRs are contained in ELTR1 (Reference 1). This report follows the NRC-approved generic process requirements contained in ELTR1.

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 above the original rating. In addition, continuing improvements in the analytical techniques and computer codes, plant performance feedback/operating experience, and implementations of improvements in fuel designs have resulted in a significant increase in the design and operating margins between the calculated safety analyses results and the licensing limits. These available differences in calculational results, combined with the as-designed excess equipment, system, and component capabilities (1) have allowed numerous BWRs to increase their thermal power ratings by 5% without any NSSS hardware modification, and (2) provide for power increases to 20% with some hardware modifications. These power increases involve no significant increase in the hazards presented by the plants as approved by the NRC at the original license stage.

CPS is currently licensed for a 100% RTP level of 2894 MWt. The safety analyses of design basis accidents (DBAs) and operational transients are based on a power level 102% above the proposed EPU power level of 3473 MWt, unless the 2% power factor is already accounted for in the analysis methods.

The EPU analysis basis ensures that the power-dependent safety margin prescribed by the Code of Federal Regulations (CFR) is maintained by meeting the appropriate regulatory criteria. Either NRC-approved or industry-accepted computer codes and calculational techniques are used to demonstrate meeting the applicable regulatory acceptance criteria.

The planned approach to achieving the higher power level consists of: (1) an increase in the core thermal power to create increased steam flow to the turbine, (2) a corresponding increase in the Feedwater system flow, (3) no increase in either maximum core flow or reactor dome pressure, and (4) reactor operation primarily along an extension of the standard Maximum Extended Load Line Limit Analysis (MELLLA) rod/flow control lines. Plant-unique evaluations were based on a review of plant design and operating data, as applicable, to confirm excess design capabilities, and, if necessary, identify any items which may require modifications associated with the EPU. For some items, bounding analyses and evaluations demonstrate plant operability and safety. The scope and depth of the evaluation results provided herein were established based on the generic BWR EPU guidelines and unique features of the plant. The results of the applicable evaluations presented in this report were found to be acceptable.

1.3 EPU Plant Operating Conditions

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

The EPU heat balance was determined such that the core thermal power is 120% of the OLTP and the steam flow from the vessel was increased to approximately 122% of the original value. The reactor heat balance is coordinated with the turbine heat balance. Figure 1-1 shows the EPU heat balance at 100% of EPU RTP and 100% rated core flow. Table 1-1 provides a summary of the reactor thermal-hydraulic parameters for the current rated condition and the EPU condition.

1.4 Summary And Conclusions

This report supports the conclusion that this EPU 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

Current and EPU Plant Operating Conditions

Parameter	Current RTP Value	EPU RTP Value
Thermal Power (MWt)	2894	3473
Vessel Steam Flow (Mlb/hr) *	12.454	15.153
Full Power Core Flow Range		
Mlb/hr	63.4 to 90.4	83.7 to 90.4
% Rated	75 to 107	99 to 107
Dome Pressure (psig)	1025	No change
Dome Temperature (°F)	549.4	No change
Turbine Inlet Pressure (psia)	982	954
Full Power Feedwater		
Flow (Mlb/hr) *	12.427	15.126
Temperature Range (°F)	370 to 420	380 to 430
Core Inlet Enthalpy (Btu/lb)**	527.8	525.5

* At design feedwater heating

** At design feedwater heating and 100% core flow condition

Performance improvement features and/or equipment out-of-service (OOS) included in the EPU evaluations are:

(1) Maximum Extended Operating Domain (MEOD), includes MELLLA and Increased Core Flow (ICF)

(2) Feedwater Heater OOS (FWHOOS)

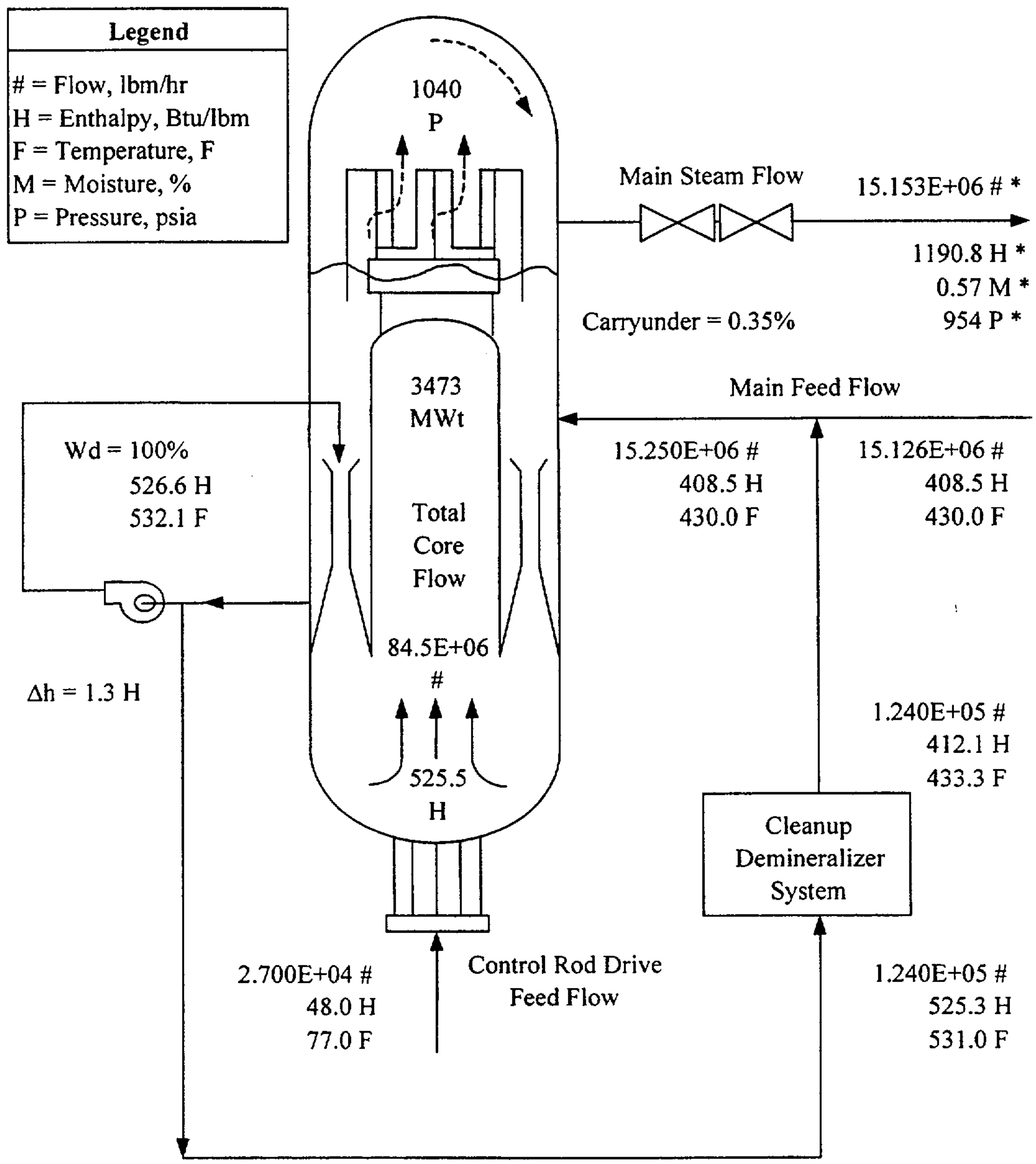
(3) Single-loop Operation (SLO)

(4) Final Feedwater Temperature Reduction (FFWTR)

(5) Two Safety Relief Valves (SRVs) OOS

(6) One Automatic Depressurization System (ADS) valve OOS

(7) 3% SRV setpoint tolerance



* Conditions at upstream side of TSV

Core Thermal Power	3473.0
Pump Heating	9.1
Cleanup Losses	-4.1
Other System Losses	-1.1
Turbine Cycle Use	3476.9 MWt

Figure 1-1

EPU Heat Balance – Nominal

2. REACTOR CORE AND FUEL PERFORMANCE

2.1 Fuel Design and Operation

At the OLTP or the EPU conditions, all fuel and core design limits continue to be met by planned deployment of fuel enrichment and burnable poison, and supplemented by core management control rod pattern and/or core flow adjustments. Revised loading patterns, larger batch sizes, and potentially new fuel designs may be used to provide additional operating flexibility and maintain fuel cycle length.

2.2 Thermal Limits Assessment

Operating limits ensure that regulatory and/or safety limits are not exceeded for a range of postulated events [e.g., transients, loss-of-coolant accidents (LOCA)]. Cycle-specific core configurations, evaluated for each reload, confirm EPU RTP capability and establish or confirm cycle-specific limits, as is currently the practice. The evaluation of thermal limits for the EPU core shows that the current thermal margin design limits can be maintained.

2.3 Reactivity Characteristics

All minimum shutdown margin requirements apply to cold conditions (< 200°F), and are maintained without change. The Technical Specifications cold shutdown margin requirements are not affected. Operation at higher power could reduce the hot excess reactivity during the cycle. This loss of reactivity does not affect safety, and is not expected to significantly affect the ability to manage the power distribution through the cycle to achieve the target power level.

The EPU power-flow operating map (Figure 2-1) includes the operating domain changes for the EPU and the plant performance improvement features currently allowed for in the Updated Safety Analysis Report (USAR), core fuel reload evaluations, and/or the Technical Specifications. The maximum thermal operating power and maximum core flow shown on Figure 2-1, correspond to the EPU RTP. Figure 2-1 shows the current maximum licensed rod line and the proposed maximum rod line for EPU on an absolute power basis.

2.4 Stability

CPS is currently operating under the requirements of reactor stability Interim Corrective Actions (ICAs) and is implementing reactor stability Long-Term Solution Option III. The Oscillation Power Range Monitor (OPRM) system is scheduled to be armed in a future cycle (it is not considered to be fully implemented until the trip system is armed). Because the ICAs are used as a backup solution when the OPRM system fails, the effect of EPU is addressed on both the ICAs (Reference 2) and on the stability Option III solution (Reference 3).

To ensure an adequate level of protection against the occurrence of a thermal-hydraulic instability, the ICAs instability exclusion region boundaries are unchanged with respect to absolute power level (MWt).

The Option III solution monitors OPRM signals to determine when a reactor scram is required. The OPRM system may only cause a scram when plant operation is in the Option III Trip Enabled Region. The Trip Enabled Region was modified for EPU operation to maintain the pre-EPU absolute power and flow coordinates. The stability-based Operating Limit Minimum Critical Power Ratio (OLMCPR) associated with the OPRM setpoint assures that the Critical Power Ratio (CPR) safety limit is not violated following an instability event. Adequate safety limit MCPR protection is demonstrated for each reload cycle.

2.5 Reactivity Control

The Control Rod Drive (CRD) system introduces changes in core reactivity by positioning neutron absorbing control rods within the reactor. It is also required to scram the reactor by rapidly inserting withdrawn rods into the core.

Because there is no increase in the vessel operating pressure, CRD scram performance and CRD mechanism structural and functional integrity are not affected by the EPU.

The components of the CRD mechanism, which form part of the primary pressure boundary, have been designed in accordance with the ASME Boiler and Pressure Vessel (B&PV) Code, Section III. The EPU engineering analyses show that limiting CRD component stresses are within the allowable stress criteria and that the current fatigue analysis is valid.

Based on the above, the CRD system is acceptable for the EPU.

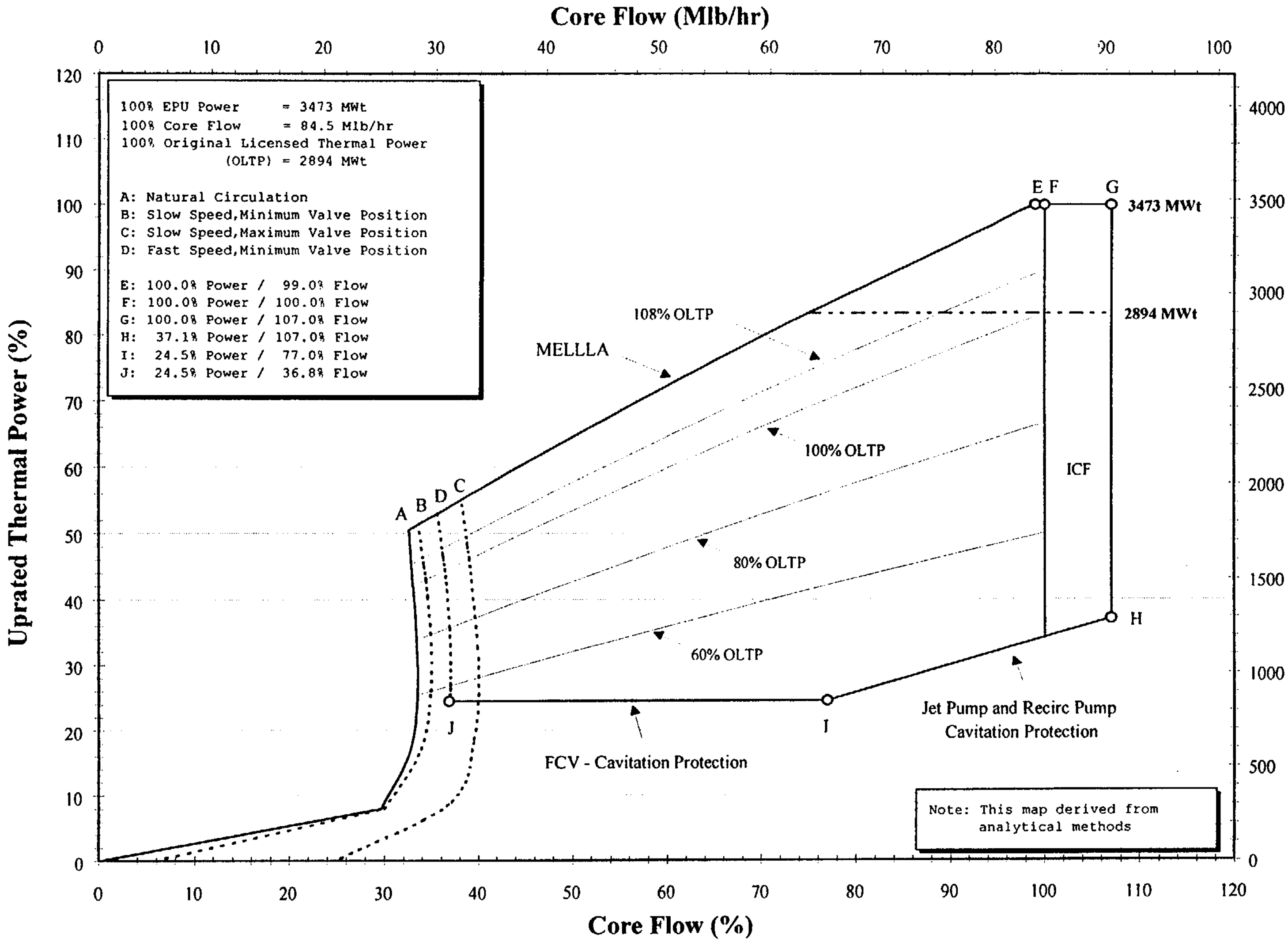


Figure 2-1
Power-Flow Operating Map For EPU

3. REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

3.1 Nuclear System Pressure Relief

The purpose of the nuclear system pressure relief is to prevent overpressurization of the nuclear system during abnormal operational transients. The plant SRVs with scram provide this protection. The SRV setpoints are not changed with the EPU, because the maximum operating dome pressure is not changed.

3.2 Reactor Overpressure Protection

The design pressure of the reactor vessel and reactor pressure coolant boundary 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 limiting pressurization event remains the MSIV closure with flux scram. Starting from 102% of EPU RTP conditions, the peak calculated reactor pressure vessel (RPV) pressure remains below the 1375 psig ASME limit and reactor steam dome pressure remains below the Technical Specification 1325 psig Safety Limit. Therefore, there is no decrease in margin of safety.

3.3 Reactor Vessel And Internals

Evaluations of the reactor vessel and vessel internals concluded that the corresponding peak vessel loads and fluence conditions resulting from this EPU were within the existing design bases of these structures.

The estimated fluence for EPU conditions was conservatively increased above the USAR end-of-life value. Therefore, the higher fluence was used to evaluate the vessel against the requirements of 10 CFR 50 Appendix G. The vessel remains in compliance with the regulatory requirements during EPU conditions.

With regards to structural integrity, because there are no changes in the design conditions due to the EPU, the design stresses are unchanged and the ASME Code requirements applicable to CPS are still met. Because there are only minor changes from current rated conditions (temperature and flow), the analysis results for normal, upset, emergency, and faulted conditions show that all components meet their ASME Code requirements.

The increase in core average power results in higher core loads and reactor internal pressure differences (RIPDs) due to the higher core exit steam quality. The RIPDs were re-calculated for normal steady-state operation, upset, and faulted conditions for all major reactor internal components and determined to be acceptable.

The results of a vibration evaluation show that operation up to 3473 MWt and 107% of rated core flow is possible without any detrimental effects on the safety-related reactor internal components.

The expected performance of the steam separators and dryer was evaluated to ensure that the quality of the steam leaving the reactor pressure vessel continues to meet existing operational criteria at the EPU conditions. The results of the evaluation demonstrate that the steam separator-dryer performance remains acceptable at the EPU conditions.

3.4 Reactor Recirculation System

An evaluation of the Reactor Recirculation System (RRS) performance concluded that the existing design margin of the RRS is well within the slight changes in system temperature, pressure, and flow resulting from EPU.

3.5 Reactor Coolant Pressure Boundary Piping

The effects of EPU were evaluated for the reactor coolant piping systems which are part of the primary reactor coolant pressure boundary (RCPB) and which could be affected by an EPU-related increase in flow or operating temperature. These evaluations concluded that EPU does not have an adverse effect on the primary piping systems design. The slight increase in temperature associated with the EPU that affects piping and piping support loads does not result in load limits being exceeded.

The RRS components are made of stainless steel, and system flow increase due to the EPU is minor. Therefore, erosion/corrosion concerns are not applicable to this system.

The Main Steam and associated piping systems and Feedwater system piping are made of carbon steel, which can be affected by flow-accelerated corrosion (erosion/corrosion). The integrity of high energy piping systems is assured by proper design in accordance with the applicable Codes and Standards. The plant has an established program for monitoring pipe wall thinning in single-phase and two-phase high-energy carbon steel piping. Other RCPB piping systems [Reactor Core Isolation Cooling (RCIC) system, RPV head vent and bottom head drain, Reactor Water Cleanup (RWCU) system, and portions of the Residual Heat Removal (RHR) system] affected by flow-accelerated corrosion (FAC) are also included in this program.

EPU operation results in some changes to parameters affecting flow-induced erosion/corrosion in those systems associated with the turbine cycle (e.g., Condensate, Feedwater, Main Steam). The evaluation of and inspection for flow-induced erosion/corrosion in Balance-of-Plant (BOP) piping systems that is affected by FAC is addressed by compliance with NRC Generic Letter 89-08, "Erosion/Corrosion in Piping." EPU evaluations have confirmed that the EPU has no significant effect on flow-induced erosion/corrosion.

3.6 Main Steam Line Flow Restrictors

An evaluation of the main steam line flow restrictors concluded that the existing design margin of the flow restrictors is well within the slight changes in conditions resulting from EPU.

3.7 Main Steam Isolation Valves

The MSIVs are part of the RCPB and must be able to close within specific limits at all design and operating conditions upon receipt of a closure signal. The MSIVs have been evaluated and are acceptable for EPU operation.

3.8 Reactor Core Isolation Cooling System

The RCIC system provides core cooling in the event of a transient where the RPV is isolated from the main condenser concurrent with the loss of all feedwater flow. For EPU, the reactor dome pressure and the SRV setpoints remain unchanged. Consequently, there is no change to the RCIC high-pressure injection process parameters and no change to the overspeed trip margins.

3.9 Residual Heat Removal System

The RHR system is designed to restore and maintain the coolant inventory in the reactor vessel and to remove sensible and decay heat from the primary system and containment following reactor shutdown for both normal and post accident conditions. Evaluations indicate that the implementation of EPU does not prevent any of the RHR modes from performing their intended functions.

3.10 Reactor Water Cleanup System

The RWCU 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. Operation of the plant at the EPU RTP level does not increase the temperature or the pressure within the RWCU system. EPU results in a slight increase in the reactor water conductivity because of the increase in feedwater flow. However, the reactor water conductivity limits will be met. Therefore, implementation of the EPU does not prevent the system from performing its intended function.

3.11 Balance-of-Plant Piping

This section addresses the adequacy of the BOP piping design outside the RCPB for operation at the EPU conditions.

Large bore and small bore ASME Section III, Class 1, 2, and 3 piping and supports not addressed in Section 3.5 were evaluated for acceptability at the EPU conditions, and shown to be adequate as currently designed. The evaluation of the BOP piping and supports was performed in a manner similar to the evaluation of RCPB piping systems and supports (Section 3.5), using applicable ASME Section III, Subsections NB-3600/NC-3600 Code equations. The original Codes of record (as referenced in the appropriate calculations), Code allowable and analytical techniques were used, and no new assumptions were introduced.

Operation at the proposed EPU conditions increases pipe stresses due to slightly higher operating temperatures 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 and were found to meet the appropriate code criteria for the EPU conditions.

Operation at EPU conditions causes a slight increase in the pipe support loadings due to increases in the temperature of the affected piping systems. However, when considering the loading combination with other loads that are not affected by EPU, such as seismic and deadweight, the overall combined support load increase is small. There is adequate margin between the original design stresses and code limits of the supports to accommodate the load increase within the appropriate code criteria. Therefore, the design of the BOP piping systems is adequate to accommodate the EPU.

EPU operation results in some changes to parameters affecting flow-induced erosion/corrosion in those systems associated with the turbine cycle (e.g., Condensate, Feedwater, Main Steam). The evaluation of and inspection for flow-induced erosion/corrosion in BOP piping systems is addressed by compliance with NRC Generic Letter 89-08, "Erosion/Corrosion in Piping." The affected systems are currently monitored by the plant Erosion/Corrosion Program. Continued monitoring of the systems provides a high level of confidence in the integrity of potentially susceptible high energy piping systems. Appropriate changes to piping inspection frequency will be implemented to ensure adequate margin exists for those systems with changing process conditions. This program provides assurance that the EPU has no adverse effect on high energy piping systems potentially susceptible to pipe wall thinning due to erosion/corrosion.

4. ENGINEERED SAFETY FEATURES

4.1 Containment System Performance

The USAR provides the containment responses to various postulated accidents that validate the design basis for the containment. Operation during EPU changes some of the conditions for the containment analyses. The containment pressure and temperature responses have been reanalyzed to demonstrate the plant's capability to operate with the EPU. The results of the analyses are as follows:

- The calculated peak bulk suppression pool temperature remains below the wetwell structural design temperature.
- Peak calculated containment temperature is below the design structural limit.
- The calculated drywell pressure remains well below the containment design pressure.
- The effect of EPU on net positive suction head (NPSH) for pumps taking suction from the suppression pool was evaluated. Calculations show that adequate NPSH is assured under EPU conditions.

The LOCA containment dynamic loads analysis for the EPU is based primarily on the short-term main steam line break and recirculation line break LOCA analyses. The LOCA dynamic loads with the EPU include pool swell, condensation oscillation (CO) and chugging.

The results from the containment analyses performed for the dynamic loads evaluations indicates that the short-term containment response conditions are within the range of test conditions used to define the pool swell and CO loads for the plant. The containment response conditions with the EPU for times beyond the initial blowdown period (when chugging would occur) are within the conditions used to define the chugging loads. Therefore, the pool swell, CO, and chugging loads are not affected by the EPU.

The SRV air-clearing loads include SRV discharge line (SRVDL) loads, suppression pool boundary pressure loads and drag loads on submerged structures. For the first SRV actuations following an event involving RPV pressurization, the only parameter change that can affect the SRV loads, is an increase in SRV opening setpoint pressure. Because the EPU does not increase the SRV opening setpoints, there is no effect on the loads from the first SRV actuation.

The effect of EPU on subsequent actuation loads due to changes in the SRVDL water level and time between actuations was also evaluated. The EPU has an insignificant affect on the loads from subsequent SRV actuations.

The system designs for containment isolation are not affected by the EPU. The capability of the actuation devices to perform with the higher flow and temperature during normal operations and under post-accident conditions has been determined to be acceptable.

The motor-operated (MOV) requirements in the USAR were reviewed, and no changes to the functional requirements of the Generic Letter (GL) 89-10 MOVs are identified as a result of operating at the EPU RTP level. The operability of MOVs is documented as part of the CPS GL 89-10 MOV program. If specific valves require calculation revisions, actuator adjustments and/or physical changes to ensure satisfactory performance, they are upgraded, adjusted, or modified, as necessary.

4.2 Emergency Core Cooling Systems

The Emergency Core Cooling Systems (ECCS) are designed to provide protection against hypothetical LOCAs caused by ruptures in the primary system piping. The functional capability of each system was determined to be acceptable for the EPU.

The High Pressure Core Spray (HPCS) system has been evaluated for its design basis requirement to provide coolant flow to the reactor to prevent excessive fuel peak clad temperatures (PCT) following small breaks, and its function of fulfilling the objectives of the RCIC system in response to a transient event. The evaluation of the HPCS system concludes that it is acceptable for operation during the EPU.

The Low Pressure Coolant Injection (LPCI) mode of the RHR system is automatically initiated in the event of a LOCA. When operating in conjunction with other ECCS, the LPCI mode is required to provide adequate core cooling for all LOCA events. The increase in decay heat due to the EPU could increase the calculated PCT following a postulated LOCA by a small amount. The ECCS performance evaluation indicates that the existing LPCI mode performance capability, in conjunction with the other ECCS, is adequate to meet the post-LOCA core cooling requirement for the EPU conditions.

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. The increase in decay heat due to the EPU could increase the calculated PCT following a postulated LOCA by a small amount. The ECCS performance evaluation indicates that the existing LPCS performance capability, in conjunction with the other ECCS, is adequate to meet the post-LOCA core cooling requirement for the EPU conditions.

The ADS uses SRVs to reduce reactor pressure following a small break LOCA, when it is assumed that the high pressure ECCS has failed. This function allows LPCI and LPCS to inject coolant into the vessel. The ADS initiation logic is not affected and is adequate for the EPU conditions.

Therefore, the ECCS performance under all LOCA conditions, and their analysis models, satisfy the requirements of 10 CFR 50.46 and 10 CFR 50 Appendix K.

4.3 Main Control Room Atmosphere Control System

The control room HVAC system is designed to maintain a habitable environment and to ensure the operability of all the components in the control room under all the station operating and accident conditions. The system is designed to maintain a positive pressure within the control room envelope with respect to the adjacent areas to preclude infiltration of unconditioned air, during all the operating modes except when the system is in recirculation mode or when the system is in the maximum outside air purge mode. The performance of the Main Control Room Atmosphere Control System is not affected as a result of the proposed EPU uprate. The makeup air filter trains are capable of removing 99.95% of all particulate matter larger than 0.3 microns and no less than 99% of all forms of iodine. As a result of the EPU, the outside air iodine concentration increases by up to 20%. The amount of charcoal in the makeup air train is more than adequate to handle the additional iodine loading and the additional decay heat as a result of radionuclides deposited is insignificant. The revised control room doses are bounded by the current licensing basis.

4.4 Standby Gas Treatment System

By limiting the release of air-borne particulates and halogens, the Standby Gas Treatment System (SGTS) is designed to control off-site dose rates following a postulated design basis accident. The design flow capacity of the system was selected to maintain the secondary containment at the required negative pressure to prevent exfiltration of air from the reactor building. This capability is unaffected by the EPU.

The charcoal filter bed removal efficiency for radioiodine is also not affected by the EPU. The total post-LOCA iodine loading on the filters increases proportionally with the increase in core iodine inventory. Under the EPU conditions, the post-LOCA iodine loading increases from 1.3 to 1.6 mg of total iodine per gram of charcoal, but remains well below the original design capacity of the filter and below that allowed by Regulatory Guide 1.52. Therefore, the SGTS is unaffected by the EPU and retains its capability of meeting its design basis requirement for mitigation of offsite doses following a postulated design basis accident.

4.5 Main Steam Isolation Valve Positive Leakage Control System

The main steam line isolation valve leakage control system (MSIVLCS) controls the release of fission products that leak through the MSIVs following a LOCA. The leakage is directed to bleed lines aided by blowers, which maintain the pressure between the inboard and outboard isolation valves and between the outboard isolation valves and the downstream shutoff valves slightly negative with respect to atmosphere. The bleed lines pass the leakage into the SGTS. This capability is unaffected by EPU.

4.6 Post-LOCA Combustible Gas Control

The Combustible Gas Control system is designed to maintain the hydrogen concentrations of the drywell and containment atmospheres below the lower flammability limit following a hypothetical LOCA. The post-LOCA production of hydrogen and oxygen by radiolysis increases proportionally with power level. The increase in radiolysis due to the EPU has a minor impact on the time available to start the system before reaching procedurally controlled limits, but does not impact the ability of the system to maintain hydrogen below the lower flammability limit. The minimum required start time for the containment mixer decreases from 2.2 hours to 2.0 hours. The minimum required start time for the recombiner decreases from 28.1 hours to 22.6 hours. As a result, the maximum containment hydrogen concentration increases from 3.6% to 3.8%, occurring 523 hours (21.8 days) following the LOCA.

The EPU has no impact on recombiner maximum operating temperature. The maximum operating temperature of the recombiners is dependent only on the maximum containment hydrogen concentration when the recombiners are in operation. The containment maximum hydrogen concentration is procedurally controlled to remain below the Regulatory Guide 1.7 flammability limit, and therefore is not influenced by the EPU.

5. INSTRUMENTATION AND CONTROL

5.1 Nuclear Steam Supply System

The NSSS process variables and instrument setpoints that could be affected by the EPU were evaluated. Increases in core thermal power and steam flow affect some instrument setpoints.

The Average Power Range Monitor (APRM) power signals will be rescaled to the 3473 MWt power level, such that the indications read 100% at the new licensed power level.

EPU has little effect on the intermediate range monitor (IRM) overlap with the source range monitors (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. No change is needed in the APRM downscale setting.

The determination of instrument allowable values and setpoints is based on plant operating experience and the conservative analytical limits used in specific licensing safety analyses. The settings are selected with sufficient margin to preclude inadvertent initiation of the protective action during operation.

The following instrument ALs remain unchanged due to implementation of the EPU:

- Reactor vessel high-pressure scram
- Anticipated transient without scram recirculation pump trip high pressure trip
- SRV setpoints
- The APRM high power scram AL remains unchanged, however, the flow-biased scram AL is changed as identified below.
- The Control Rod Block Pattern Control Rod Withdrawal Limiter (RWL) has a low power setpoint (LPSP) and a high power setpoint (HPSP). For EPU, the HPSP AL is kept the same in terms of percent of RTP. The LPSP AL is changed as discussed below.
- Main steam line turbine building high temperature isolation
- Low steam line pressure MSIV closure
- Reactor water level instruments

- Main steam line tunnel high temperature isolation
- Low condenser vacuum MSIV isolation
- RCIC steam line high flow isolation
- Feedwater flow setpoint for recirculation cavitation protection

The following instrument analytical limits are changed due to implementation of the EPU:

- The APRM flow-biased high power scram is redefined to reflect the change in the maximum allowable load line region.
- The Control Rod Block Pattern Control RWL LPSP has lower and upper bounding ALs. For EPU, the LPSP lower and upper bounding ALs are rescaled to maintain the original absolute thermal power basis.
- The turbine stop valve closure and turbine control valve fast closure scram bypass AL (%RTP) is reduced by the ratio of the power increase. However, the new AL does not change in terms of absolute power.
- The main steam line (MSL) high flow isolation AL will be reduced. At the EPU power level, the MSL flow restrictors limit (choke) the steam flow below the current trip setpoint. To ensure the MSL high flow isolation function, the trip value must be less than the choked MSL steam flow value, and thus, the MSL high flow isolation AL must be reduced.

5.2 Balance-of-Plant

Operation of the plant at the EPU RTP level has minimal effect on the balance-of-plant (BOP) system instrumentation and control devices. Based on the EPU operating conditions for the power conversion and auxiliary systems, most process control values and instrumentation have sufficient range and adjustment capability for use at EPU conditions. Some (non-safety) modifications to the power conversion systems may be needed to obtain 100% EPU RTP.

The Pressure Control System (PCS) provides fast and stable response to system disturbances related to pressure and steam flow changes so that reactor pressure is controlled within its allowed high and low values. The PCS consists of the turbine-generator electro-hydraulic control (EHC) system and the steam bypass pressure control system (SBPCS). For the EPU, no modifications are required to the SBPCS and the only setpoint change required is applying a new pressure regulator setpoint. Other adjustments are limited to "fine tuning" of the control settings, as may be required to operate optimally at the increased power level.

The turbine EHC system was reviewed for the increase in core thermal power and the associated increase in rated steam flow. New turbine control valve (TCV) Diode Function Generator tuning and updating of the characteristic TCV tuning parameter curve are necessary for the control systems to perform normally at EPU conditions. No modification to the turbine control valves or the turbine bypass valves is required for operation at the EPU throttle conditions. Confirmation testing is also be performed during power ascension.

The Feedwater Control System (FCS) is used to maintain water level control in the reactor. For the EPU no modifications are required to the FCS. The current controller adjustments are expected to be satisfactory for EPU; however, the device settings are confirmed by performing unit tests during the power ascension to EPU conditions.

The instrument setpoints associated with system leak detection have been evaluated with respect to the higher operating steam flow and feedwater temperature for the EPU. There is no significant effect on any leak detection system due to the EPU.

6. ELECTRICAL POWER AND AUXILIARY SYSTEMS

6.1 Alternating Current Power

The existing off-site electrical equipment was determined to be adequate for operation with the EPU-related electrical output, as shown in Table 6-1. The review concluded the following:

- The need to modify the isolated phase bus duct and associated coolers to support EPU-related generator output.
- The need to replace the main power transformers to support EPU-related generator output.
- The need to replace, modify and uprate the 345KV switchyard components including circuit breakers, disconnect switches and current transformers (evaluated for generator output, minus main power transformer losses, without unit auxiliary transformer loads) to meet EPU continuous current requirements.

A grid stability analysis has been performed on the 345kV network for a net output of 1120MW (Gross MW minus auxiliary power usage) to demonstrate conformance to General Design Criteria (GDC) 17 (10 CFR 50, Appendix A). EPU has an effect on grid stability and reliability. Modifications to the electrical supply and distribution system are planned to maintain GDC 17 compliance considering the effects of EPU-related power operation.

The onsite power distribution system consists of transformers, buses, and switchgear. Alternating Current (AC) power to the distribution system is provided from the transmission system or from onsite diesel generators. Station batteries provide Direct Current (DC) power to the distribution system.

Station loads under normal operation/distribution conditions are computed based on equipment nameplate data or brake horsepower (BHP). Operation at the EPU RTP level is achieved by utilizing new or existing equipment operating within its design capability; therefore, under normal conditions, the electrical supply and distribution components (switchgear, motor control centers, cables, etc.) are adequate.

Portions of the non-class 1E auxiliary power distribution equipment have presently identified under-voltage, overload and short circuit overduty conditions under certain unusual plant analyzed conditions. These conditions are exacerbated with the additional EPU non-safety loading. Modifications to address these issues will be implemented as appropriate.

Station loads under emergency operation/distribution conditions (emergency diesel generators) are based on equipment running BHP data. No load increases were identified for operation at the EPU RTP; therefore, under emergency conditions the electrical supply and distribution components are adequate.

No increase in the electrical demand is required of any AC-powered ECCS equipment for the EPU. The safety related buses running voltages are adequate for EPU operation. However, due to the increase of the electrical demand on the reserve auxiliary transformer, there is a reduction of the ECCS equipment block motor start voltages beyond the OLTP design limits. Further analyses will be performed to address the lower start voltages to ensure that the 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.

6.2 Direct Current Power

Operation at the EPU RTP level does not increase any loads beyond nameplate rating or revise any control logic; therefore the DC power distribution system remains adequate.

6.3 Fuel Pool Cooling

An evaluation of the Fuel Pool Cooling and Cleanup System (FPCCS) was performed to determine its ability to handle the higher heat load in the spent fuel pool (SFP) after EPU implementation. The heat load is increased due to the increased decay heat generated by the fuel operated at the EPU RTP level. FPCCS heat removal capability is adequate to remove the EPU decay heat loads, and it would not result in a delay in removing RHR system from service (i.e., the outage day the FPCCS can maintain the Spent Fuel Pool temperature below 150°F such that the Fuel Pool Assist mode of the RHR is not required). The FPCCS heat exchangers are sufficient to remove the decay heat during normal refueling and under full core off-load conditions.

The normal radiation levels around the SFP may increase slightly, primarily during fuel handling operations. This increase is acceptable and does not significantly increase the operational doses to personnel or equipment. There is no effect on the design of the SFP storage racks, because the original SFP design temperature is not exceeded.

6.4 Water Systems

Evaluations of the service water systems were performed to determine the effect of the EPU on these systems. The results of these evaluations concluded that the safety-related and nonsafety-related service water system capabilities are adequate, and the environmental effects of EPU are controlled at the same level as is presently in place. That is, the plant operation is managed such that none of the present limits (e.g., maximum allowed cooling water discharge temperature) is increased for EPU. This conclusion is based on the following considerations.

The safety-related shutdown service water (SX) system provides reliable supplies of cooling water during and following a DBA for the following essential equipment:

RHR heat exchangers;

RHR pump seal coolers;

SX pump motor coolers;

Diesel generator heat exchangers;

Drywell chillers; (Note: The drywell chillers are isolated during a LOCA, but remain in service for a LOOP event.)

FPCCS heat exchangers;

FPCCS pump motor exchangers;

Control room HVAC chillers;

SGTS exhaust HI-Range radiation monitor cooler;

SGTS room coil cabinets;

Hydrogen recombiner room coil cabinets;

SX pump room coil cabinets;

Combustible Gas Control System (CGCS) room cooling coil cabinets;

Division switchgear heat removal condensing units;

Inverter room coil cabinets;

Division IV inverter room coil cabinet;

ECCS LPCS pump room coil cabinet;

ECCS RHR pump room coil cabinets;

ECCS RHR heat exchanger room coil cabinets;

ECCS RCIC pump room coil cabinet;

ECCS HPCS pump room coil cabinets;

Main Steam Isolation Valve (MSIV) leakage room coil cabinet;

MSIV leakage outboard room coil cabinet; and

SX strainer backwash.

Evaluations show that the implementation of the EPU does not require modifications to the SX system.

Regarding the nonsafety-related heat loads, the heat rejected to the service water system via the closed cooling water systems and other auxiliary heat loads increases from the EPU due to an increase in main generator losses rejected to the stator water coolers and hydrogen coolers and the Turbine Building Closed Cooling Water (TBCCW) system. The increase in service water heat loads from these sources due to EPU operation is projected to be approximately proportional to the EPU itself.

For normal operation, the maximum service water temperatures occur during peak summer months. An EPU discharge temperature was estimated assuming both realistic conditions and very conservative bounding conditions. Comparing the current plant and state limited discharge temperatures compared to EPU conditions demonstrates that the service water system is adequate for the EPU conditions.

Performance of the main condenser was evaluated for EPU. This evaluation was based on a design duty over the actual yearly range of circulating water inlet temperatures, and confirms that the condenser, circulating water system and heat sink are adequate for EPU operation. The state thermal discharge limits were compared to bounding analysis discharges for EPU and demonstrate that the plant remains within the state limits during operation at EPU conditions.

The heat loads on the Component Cooling Water (CCW) system do not increase significantly due to EPU because they depend mainly on either reactor vessel water temperature or flow rates in the systems cooled by the CCW. The change in reactor vessel water temperature is minimal. The only increases in heat loads due to EPU are: the operation of the Reactor Recirculation pumps at a higher power level and an increase in the Fuel Pool Coolers heat load. Therefore, the CCW system can accommodate the increased heat load due to EPU.

The power-dependent heat loads on the TBCCW system increased by the EPU, are those related to the operation of the bus duct cooler, exciter coolers, and spare condensate and condensate booster pumps. The TBCCW flow requirements are expected to increase as a result of modifications to the exciter and bus duct coolers. Subsequent modifications to the TBCCW system may be required to meet the flow requirements for EPU. The TBCCW, like the CCW, contains sufficient capacity to assure that adequate heat removal capability is always available. Therefore, sufficient cooling capacity for EPU operation is available.

The Ultimate Heat Sink (UHS) for the CPS consists of a submerged pond within Lake Clinton formed by the submerged dam across the North Fork of Salt Creek channel approximately 1 mile from the Circulating Water Screen House. The ultimate heat sink was originally designed for two 991 MWe units. Therefore, the UHS provides a sufficient quantity of water at a temperature less than the design temperature following a design basis LOCA for the single CPS Unit operating at the EPU RTP level.

The environmental effects of EPU operation are controlled at the same levels as are presently in place.

6.5 Standby Liquid Control System

The Standby Liquid Control System (SLCS) is designed to shut down the reactor from rated power condition to cold shutdown in the postulated situation that all or some of the control rods cannot be inserted. The ability of the SLCS to achieve and maintain safe shutdown is not affected by the EPU. The use of a new fuel design, combined with the specified fuel cycle operating time, does not require an increase in the current minimum reactor boron concentration of 660 ppm.

The performance of the SLCS during a postulated ATWS is evaluated in Section 9.3.1 for a representative core design at the EPU power level. The evaluation shows that the EPU has no adverse effect on the ability of the SLCS to mitigate an ATWS.

6.6 Power-Dependent Heating Ventilation and Air Conditioning

The HVAC systems consist mainly of heating, cooling supply, exhaust and recirculation units in the turbine building, containment building and the drywell, auxiliary building, fuel handling building, control building, and the radwaste building. EPU results in a small increase in the heat load caused by slightly higher process temperatures and higher electrical currents in some motors and cables.

The affected areas are the steam tunnel in the containment and the auxiliary buildings, the drywell in the containment building, the feedwater heater bay, condenser, steam driven feedwater pumps, condensate/condensate booster pump areas of the turbine building, and the fuel pool cooling areas in fuel handling building. Other areas are unaffected by the EPU because the process temperatures and electrical loads remain relatively constant.

The heat load in the containment and auxiliary building steam tunnels increases due to the increase in the feedwater process temperature. The increased heat load is within the margin of the steam tunnel area coolers. In the drywell the increase in feedwater process temperature and the slight increase in the recirculation pump motor horsepower are within the margins in the system capacity.

In the turbine building, the maximum temperature increase in the feedwater heater bay and condenser areas is less than 2°F due to the increase in the feedwater process temperatures. The increase heat load due to increased power requirements of the condensate and condensate booster pump motors is within the margin of the pump area coolers. The increase in temperature of the steam supplying the steam driven feedwater pumps increases the heat load on the area coolers, but the heat load remains within the area cooler margins.

In the fuel building, the increase in heat load due to a slight fuel pool cooling process temperature increase is within the margin of the area coolers.

Based on a review of design basis calculations and environmental qualification design temperatures, the design of the HVAC is adequate for the EPU.

6.7 Fire Protection

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

A plant-specific evaluation was performed to demonstrate safe shutdown capability assuming EPU conditions. The results of the evaluation for the EPU demonstrate that fuel cladding integrity, RPV integrity and containment integrity are maintained, and that sufficient time is available for the operator to perform the necessary actions. No changes are required in the equipment required for safe shutdown for the postulated fire events. Therefore, the EPU has no adverse effect on the ability of the systems and personnel to mitigate the effects of a fire event, and the requirement to achieve and maintain safe shutdown in the event of a fire is satisfied.

6.8 Systems Not Impacted By EPU

The following systems are not affected by operation of the plant at the EPU condition or the replacement of the Main Turbine High and Low Pressure rotors:

1. Fuel Handling and Transfer System and Fuel Support System
2. Service Air System
3. Reactor Protection System
4. Breathing Air System
5. Process Sampling/PASS
6. Auxiliary Boiler System
7. Auxiliary Steam
8. Annunciators
9. Generator Purge
10. Miscellaneous Drywell (Maintenance Rule LLRT, ILRT)
11. Oil Transfer
12. Special Doors
13. Suppression Pool Cleanup & Transfer
14. Miscellaneous Personnel Contamination Monitors
15. Spectral Analysis, ALARA, and Dosimetry
16. Screen Wash
17. Diesel Fuel Oil and Diesel Generator Room Ventilation
18. Miscellaneous Secondary Containment Loop Seals
19. Illinois Department of Nuclear Safety Equipment

Some CPS systems are affected to a small extent by operation of the plant at the EPU condition or the replacement of the Main Turbine High and Low Pressure rotors. For these systems, the effects are insignificant to the design or operation of the system and equipment:

1. Corrosion Control and Heat Exchangers
2. Hydrogen Injection (SCC Mitigation)
   (Not Currently Operating To Be Implemented During Changes for EPU)
3. Instrument Air
4. Loose Parts Monitoring
5. Auxiliary Control Room Panels
6. Power Generation Control Complex (PGCC)
7. Local Instrument Panels
8. Main Control Room Panels
9. NSPS Self Test System
10. Switchgear Heat Removal
11. Raw Water (Lake or Well)
12. Make-Up Demineralizers
13. Buffer System - Data Link Computer
14. Special Test Equipment

Table 6-1

EPU Plant Electrical Characteristics

Guaranteed Generator Output (MWe)	1138.5
Rated Voltage (KV)	22
Power Factor	0.9
Guaranteed Generator Output (MVA)	1265
Current Output (kA)	33.2
Isolated Phase Bus Duct Rating:	
Main Section (kA)	33.72
Branch Section (kA)	19.47
Main Transformers Rating (MVA)	1425
EPU Transformer Output (MVA)	1184 (Note 1)

Note:

1. Generator at maximum output MVA, and no loading on unit auxiliary transformers.

7. POWER CONVERSION SYSTEMS

The power conversion systems were originally designed to accept the system and equipment flows resulting from continuous operation at 105% of rated steam flow. However, the structural capabilities of the power conversion systems allow for steam flows greater than 105% of original/current rated steam flow with modifications to some non-safety-related equipment.

7.1 Turbine-Generator

The turbine and generator was originally designed with a maximum flow-passing capability and generator output in excess of rated conditions to ensure that the original rated steam-passing capability and generator output is achieved. This excess design capacity ensured 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 5%. New high pressure (HP) and low pressure (LP) turbine designs are scheduled to be installed for the EPU and the moisture separator upgraded for the uprated turbine conditions. The flow margin remains at 5%.

A mechanical review of the new turbine rotors was conducted to evaluate steady state, vibrational and upset stress conditions that are affected by the EPU steam conditions and loadings. EPU will have a negligible effect on HP and LP rotor strength properties and mechanical parameters. The HP and LP turbine rotors are not considered a source for potential missile generation, and therefore, neither a HP nor a LP turbine rotor missile probability analysis is required.

7.2 Condenser And Steam Jet Air Ejectors

The performance of the main condensers was evaluated for EPU with the following conclusions:

- Both condenser hotwell capacities and level instrumentation are adequate for the EPU condition. The effect of increased steam flows on condenser tube vibration will be evaluated and additional tube staking will be performed if required to support operation at EPU conditions.

- The design of the condenser air removal system is not adversely affected and no modification to the system is required. 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. The design capacity of the steam jet air ejectors (SJAEs) is not affected because they were originally designed for operation at significantly greater than warranted flows.

7.3 Turbine Steam Bypass

The turbine bypass valves were initially rated for a total steam flow capacity of not less than 35% of the original rated reactor steam flow, or 12.454 Mlb/hr. Each of six bypass valves is designed to pass a steam flow of 726,425 lbm/hr for a total bypass capacity of 4.359 Mlb/hr. At the EPU RTP level, rated reactor steam flow is 15.153 Mlb/hr, resulting in a bypass capacity of 28.8%. The steam bypass system is a normal operating system and nonsafety-related. Even though the bypass capacity as a function of percent uprated steam flow is reduced, the actual steam bypass capacity is unchanged, and is used as an input to transient analyses for the evaluation of events that credit the turbine bypass system availability. Since the EPU transient analysis results are acceptable, the turbine bypass capacity is adequate for EPU operation.

7.4 Feedwater And Condensate Systems

The feedwater and condensate systems are designed to provide a reliable supply of feedwater at the temperature, pressure, quality, and flow rate as required by the reactor. However, these systems do not perform a system level safety-related function. Therefore, these systems are not safety-related. Their performance does, however, have an affect on plant availability and capability to operate at the EPU condition.

A review of these systems has identified that specific equipment changes, including a modification to the feed pump turbines and the addition of an auto-start feature to the standby motor driven feedwater pump following a trip of an operating turbine driven feedwater pump, may be necessary. Modifications associated with the EPU are reviewed in accordance with plant procedures to ensure compliance with 10 CFR 50.59.

During steady-state conditions, the feedwater and condensate systems will have adequate NPSH for all of the pumps to operate without cavitation in the EPU conditions.

To account for feedwater demand transients, the feedwater system will be evaluated to ensure that a minimum of 5% margin above the EPU feedwater flow is available. The current feedwater and condensate system configuration is capable of supplying the transient flow requirements up to approximately 112% of rated feedwater flow at OLTP. Future system modifications will be made so that for operation with all system pumps available, the system will meet the operational performance criteria up to the EPU power level.

The effect of the EPU on the condensate polishing demineralizers (CPDs) was reviewed. The CPD will experience slightly higher loadings from operation at the EPU power level which result in slightly reduced run times. However, the reduced run times are acceptable. Because a spare unit is utilized when cleaning is required, reduced run times (more frequent cleaning) of polisher units does not affect condensate demineralizer capacity.

8. RADWASTE SYSTEMS AND RADIATION SOURCES

8.1 Liquid Waste Management

Based on a review of plant operating effluent reports and the slight increase expected from EPU, it is concluded that the requirements of 10 CFR 20 and 10 CFR 50, Appendix I are expected to be met. Therefore, EPU does not have an adverse effect on the processing of liquid radwaste and there are no significant environmental effects.

8.2 Gaseous Waste Management

The Gaseous Waste Management Systems collect, control, process, store, and dispose of gaseous radioactive waste generated during normal operation and abnormal operational occurrences. The gaseous waste management systems include the offgas system and various building ventilation systems. The systems are designed to meet the requirements of 10 CFR 20 and 10 CFR 50, Appendix I.

Non-condensable radioactive gas from the main condenser, along with air inleakage, normally contains activation gases (principally N-16, O-19 and N-13) and fission product radioactive noble gases. This is the major source of radioactive gas (greater than all other sources combined). These non-condensable gases, along with non-radioactive air, are continuously removed from the main condenser by the steam jet air ejectors (SJAEs), which discharge into the Offgas System.

Building ventilation systems control airborne radioactive gases by using a combination of devices such as High Efficiency Particulate Air (HEPA) filters and charcoal filters. The ventilation system radiation monitors signal automatic isolation dampers, and supply and exhaust fans are controlled to maintain negative air pressure, where required, to limit migration of gases. The activity of airborne effluents released through building vents is not expected to increase significantly with the EPU. The concentration of coolant activation products is expected to remain unchanged because the linear increase in production of these products is offset by the linear increase in the steam generation rate. The release limit is an administratively controlled variable, and is not a function of core power. The gaseous effluents are well within limits at original power operation and remain within limits following EPU implementation.

Core radiolysis (i.e., formation of H_2 and O_2) increases linearly with core power, thus increasing the heat load on the recombiner and related components. Based on a heat balance for the offgas recombiner using current plant operating data for full power operations under normal water chemistry conditions, the radiolytic H_2 flow rate increases, but remains well within the design capacity of the system.

8.3 Radiation Sources In The Reactor Core

During power operation, the radiation sources in the core 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 released per unit of reactor power. Therefore, the increase in the operating source term is proportional to the increase in power.

For post-operation evaluations, two forms of source data are applied. The first is the core gamma-ray source, which is used in shielding calculations for the core and for individual fuel bundles. This source term increases in proportion to reactor power. The second is used for post-accident evaluations, which are performed in compliance with regulatory guidance that applies different release and transport assumptions to different fission products. Plant-specific or bounding fission product inventories are used in the evaluation of the affected design basis accidents.

8.4 Radiation Sources In The Coolant

During reactor operation, the coolant passing through the core region becomes radioactive as a result of nuclear reactions. The coolant activation is the dominant source in the turbine building and in the lower regions of the drywell. Because these sources are produced by interactions in the core region, their rates of production are proportional to power. As a result, the activation products, observed in the reactor water and steam, increase in approximate proportion to the increase in thermal power. The activation products in the steam are bounded by the existing design basis concentrations.

The reactor coolant contains activated corrosion products, which are the result of metallic materials entering the water and being activated in the reactor region. Under the EPU conditions, the corrosion product concentrations are not expected to exceed the design basis concentrations. Therefore, no change is required in the design basis activated corrosion product concentrations for the EPU.

8.5 Radiation Levels

For the EPU, normal operation radiation levels increase slightly. For conservatism, many aspects of the plant were originally designed for higher-than-expected radiation sources. Thus, the increase in radiation levels does not affect radiation zoning or shielding in the various areas of the plant, because it is offset by conservatism in the original design, source terms used, and analytical techniques.

Post-operation radiation levels in most areas of the plant are expected to increase by no more than the percentage increase in power level. In a few areas, near the reactor water piping and liquid radwaste equipment, the increase could be slightly higher. Regardless, individual worker exposures should be maintained within acceptable limits by the site ALARA program, which controls access to radiation areas. Procedural controls compensate for increased radiation levels.

The change in core inventory resulting from the EPU is expected to increase post-accident radiation levels by no more than the percentage increase in power level. A review of areas requiring post-accident occupancy (per NUREG-0737 Item II.B) concluded that access needed for accident mitigation is not significantly affected by the EPU.

For the EPU, normal operation gaseous activity levels increase slightly. The increase in activity levels is generally proportional to the percentage increase in core thermal power. The EPU does not involve significant increases in the offsite dose from noble gases, airborne particulates, iodine, tritium or liquid effluents. In addition, radiation from shine is not a significant exposure pathway. Present offsite radiation levels are a negligible portion of background radiation. Therefore, the normal offsite doses are not significantly affected by operation at the EPU RTP level and remain below the limits of 10 CFR 20 and 10 CFR 50, Appendix I.

9. REACTOR SAFETY PERFORMANCE EVALUATIONS

9.1 Reactor Transients

The USAR evaluates the effects of a wide range of potential plant transients. Disturbances to the plant caused by a malfunction, a single equipment failure, or an operator error are investigated according to the type of initiating event per Regulatory Guide 1.70, Chapter 15. The transient events that are analyzed at the EPU RTP and maximum core flow rate operating point on the power-flow map demonstrate the capability of the design to meet all safety criteria for EPU RTP conditions.

The cycle-specific OLMCPRs are supplied in the Core Operating Limit Reports (COLRs). The historical 25% of RTP value for the Technical Specification Safety Limit, some thermal limits monitoring Limiting Conditions for Operation (LCOs) thresholds, and some Surveillance Requirements (SRs) thresholds is based on generic analyses (evaluated up to ~50% of original RTP) applicable to the plant with highest average bundle power (the BWR6) for all of the BWR product lines. As a result of the EPU, the Safety Limit percent RTP basis, some thermal limits monitoring LCOs, and some SR percent RTP thresholds are reduced to 21.6% RTP.

The Loss of Feedwater Flow (LOFW) transient was analyzed for EPU. During a LOFW transient and assuming an additional single failure (Loss of RCIC or HPCS), reactor water level is automatically maintained above the top of the active fuel (TAF) by the RCIC or the HPCS System, without any operator action. Because of the additional decay heat from the 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 water level is restored, the operator manually controls the water level, reduces reactor pressure, and initiates RHR shutdown cooling. These sequences do not require any new operator actions or shorter response times. Therefore, the operator actions for a LOFW transient do not significantly change for the EPU.

9.2 Design Basis Accidents

The doses resulting from the accidents analyzed are compared with the applicable dose limits in Tables 9-1 through 9-4. The plant-specific results for the EPU remain below established regulatory limits.

9.3 Special Events

An ATWS analysis for the EPU condition was performed resulting in the peak vessel pressure, peak clad temperature, peak clad oxidation, peak suppression pool temperature, and peak containment pressure meeting the acceptance criteria. Therefore, the plant response to an ATWS event during EPU operation is acceptable.

The plant responses to and coping capabilities for a station blackout (SBO) event are affected slightly by operation at the EPU RTP level, due to the increase in the decay heat. There are no changes to the systems or equipment used to respond to an SBO, nor is the required coping time changed. The plant continues to meet the requirements of 10 CFR 50.63 after the EPU.

Table 9-1

LOCA Radiological Consequences

Location	**EPU**	**Limit**
Exclusion Area:		
Whole Body Dose (rem)	13.5	≤ 25
Thyroid Dose (rem)	267	≤ 300
Low Population Zone:		
Whole Body Dose (rem)	4.5	≤ 25
Thyroid Dose (rem)	102	≤ 300
Control Room:		
Whole Body Dose (rem)	3.5	≤ 5
Thyroid Dose (rem)	29	≤ 30
Beta Dose (rem)	26	≤ 30

Table 9-2

MSLBA Radiological Consequences

Location	**EPU**	**Limit**
Exclusion Area:		
Whole Body Dose (rem)	0.008	≤ 2.5
Thyroid Dose (rem)	0.45	≤ 30
Low Population Zone:		
Whole Body Dose (rem)	0.0019	≤ 2.5
Thyroid Dose (rem)	0.11	≤ 30

Table 9-3

FHA Radiological Consequences

Location	EPU	Limit
Exclusion Area:		
Whole Body Dose (rem)	0.31	≤ 6.25
Thyroid Dose (rem)	0.32	≤ 75
Low Population Zone:		
Whole Body Dose (rem)	0.066	≤ 6.25
Thyroid Dose (rem)	0.070	≤ 75

Table 9-4

CRDA Radiological Consequences

Location	EPU	Limit
Exclusion Area:		
Whole Body Dose (rem)	0.023	≤ 6.25
Thyroid Dose (rem)	0.19	≤ 75
Low Population Zone:		
Whole Body Dose (rem)	0.0073	≤ 6.25
Thyroid Dose (rem)	0.22	≤ 75

10. ADDITIONAL ASPECTS OF EXTENDED POWER UPRATE

10.1 High Energy Line Break

Operation at the EPU RTP level requires an increase in the steam and feedwater flows, which results in a slight increase in downcomer subcooling. This, in turn, results in a small increase in the mass and energy release rates following high-energy line breaks (HELBs). The postulated break locations remain the same because the piping configuration does not change due to EPU.

The HELB analysis evaluation was made for all systems evaluated in the USAR. The evaluation shows that the affected building and cubicles that support the safety-related function are designed to withstand the resulting pressure and thermal loading following an HELB. The equipment and systems that support a safety-related function are also qualified for the environmental conditions imposed upon them.

Because there is no pressure increase, pipe whip and jet impingement loads do not significantly change. Existing calculations supporting the dispositions of potential targets of pipe whip and jet impingement from the postulated HELBs have been reviewed, and determined to be adequate for the safe shutdown effects in the EPU RTP condition. Existing pipe whip restraints and jet impingement shields, and their supporting structures are also adequate for the EPU conditions.

There is no affect on the plant internal flooding analysis or safe shutdown analysis due to EPU.

The Moderate Energy Line Break (MELB) analysis is addressed in Section 3.6 of the USAR. Operation at the EPU RTP level does not require an increase in the reactor vessel pressure during full power operation. None of the plant flooding zones contains piping with a potential MELB location affected by the reactor operating conditions changed for the EPU. Therefore, the MELB analysis for the plant is not affected.

10.2 Equipment Qualification

The safety-related electrical equipment was reviewed to assure that the existing qualification for the normal and accident conditions expected in the area where the equipment is located remain adequate. Margins in accordance with IEEE 323 were originally applied to the environmental parameters, and no change is needed for the EPU.

Environmental qualification (EQ) for safety-related electrical equipment located inside the containment is based on DBA conditions, main steam line break or LOCA, and their resultant temperature, pressure, humidity, and radiation consequences, and includes the environments expected to exist during normal plant operation. The current accident conditions for temperature and pressure are modified for the EPU RTP conditions. The current radiation levels under normal plant conditions do not increase except in the vicinity of the reactor. Radiation levels under accident conditions were conservatively evaluated to increase 17%.

Accident temperature, pressure, and humidity environments used for qualification of equipment outside containment result from a main steam line break in the pipe tunnel, or other high energy line breaks, whichever is limiting for each plant area. The accident temperature, pressure and humidity conditions resulting from a LOCA do not change with the power level, but some of the HELB pressure profiles increase by a small amount. The normal temperature, pressure, and humidity conditions do not change as a result of the EPU. The normal radiation levels were evaluated, and bound the EPU level.

An EQ review for the EPU conditions identified some equipment located both within the containment and outside the containment that could potentially be affected by the higher accident temperature and radiation levels. The qualification of this equipment is resolved by reanalysis, by refined radiation calculations (location specific), by slightly reducing qualified life, or by performing additional tests/analyses to support qualification.

10.3 Mechanical Component Design Qualification

Operation at the EPU RTP level increases the normal process temperature slightly. The accident radiation level and the normal radiation level also increase slightly due to the EPU, and were evaluated as discussed in Section 10.2. Reevaluation of the safety-related mechanical equipment with non-metallic components identified some equipment potentially affected by the EPU temperature and radiation conditions. The qualification of this equipment is resolved by reanalysis, by refined radiation calculations (location specific), by slightly reduced qualified life, or by performing additional tests/analyses to support qualification.

The mechanical design of equipment/components (valves, pumps, snubbers, etc.) in certain systems is affected by operation at the EPU RTP level due to slightly increased accident temperatures and radiation levels, and in some cases, flow. The revised operating conditions do not significantly affect the cumulative usage fatigue factors of mechanical components. Therefore, the mechanical components and component supports are adequately designed for the EPU conditions.

10.4 Required Testing

The following testing will be performed during the initial power ascension steps for EPU:

- Surveillance testing will be performed on the instrumentation that requires re-calibration for EPU in addition to the testing performed according to the plant Technical Specifications schedule.
- Steady-state data will be taken during power ascension beginning at 90% OLTP power and continuing at each EPU power increase increment. This data will allow system performance parameters to be projected through the EPU power ascension.
- Power increases beyond the previous rating will be made along an established flow control/rod line in increments of ≤ 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.
- Control system tests will be performed for the feedwater/reactor water level controls and pressure controls. These operational tests will be made at the at the appropriate plant conditions for that test and at each power increment above the previous rated power condition, to show acceptable adjustments and operational capability.

Original performance criteria and modified performance criteria updated since the initial test program are utilized for supporting EPU power ascension testing.

CPS does not intend to perform testing regarding initiating an automatic scram from high power. The operating history of the plant has shown previous transient events from full power to be within expected peak limiting values. The transient analysis performed for this EPU demonstrates that all safety criteria are met and that this uprate does not cause any previous non-limiting events to become limiting. Performing such testing given the available information is considered non-conservative and a challenge to reactor safety. Should any future large transients occur, plant procedures require verification that the plant responded in accordance with expected responses with respect to the USAR. Existing plant event data recorders are capable of confirming the actual versus expected response.

The piping vibration levels of the Main Steam system piping and the Feedwater system piping will be monitored during initial plant operation at the EPU conditions. These piping systems will be monitored for vibration because the mass flow rates in these piping systems increase during EPU operations. The mass flow rates in these systems increase approximately in proportion to the EPU power level increase. The startup vibration test program is expected to show that these piping systems are vibrating at acceptable levels during initial plant operation at the EPU conditions.

The CPS containment leakrate testing program is required by 10 CFR 50 Appendix J and is described within USAR Section 6.2. This test program periodically pressurizes the containment (Type A test), the containment penetrations (Type B test), and the containment isolation valves and test boundary (Type C tests) to the calculated peak containment pressure (P_a), and measures leakage. As a result of EPU analysis, the calculated peak containment pressure is reduced to 7 psig. The current value of P_a used for containment testing is 9 psig. This value of P_a bounds the peak containment pressure calculated for EPU for all tested containment penetrations with the exception of check valves.

ANS 56.8-1994 specifies that P_a be the peak, calculated pressure, and that the test pressure for check valves not exceed this value by more than 10%. For penetrations protected by check valves, increased closing force is applied by testing at 9 psig versus 7 psig. CPS intends to continue testing with the current value of 9 psig on valves with metal components in the seat and disk. The disk rests in the seat and this small extra pressure would not change the seat to disk interface or affect the measured leakage. CPS has two Feedwater check valves with resilient seats. These valves are designed to seal at low pressure, and the higher test pressure may improve the sealing ability of these valves. CPS will test these valves as required, using the new calculated peak containment pressure of 7 psig. The test pressure would be kept at or returned to 9 psig if these valves were modified to non-resilient seats, or if further analysis showed no change in leakage for the 9 psig test pressure.

10.5 Individual Plant Evaluation

The plant uses a probabilistic risk/safety assessment (PRA/PSA) to comply with the Individual Plant Evaluation (IPE) requirement. A plant-specific PRA/PSA will be assessed for the effect of the EPU. The assessment and any necessary changes will be completed as required to support operation of the plant at a higher power level.

10.6 Operator Training And Human Factors

Additional training is required to operate the plant in the EPU condition. As changes to the plant are identified and processed, an evaluation to determine the specific changes to the operator training program is performed. This evaluation includes the effect on the plant simulator.

For EPU conditions, operator responses to transient, accident and special events are not significantly affected. The EPU does not change any of the automatic safety functions. After the applicable automatic responses have initiated, the follow-on operator actions (e.g., maintaining safe shutdown, core cooling, and containment cooling) for plant safety do not change for the EPU.

Training required to operate the plant following the EPU will be conducted prior to restart of the unit at the EPU conditions. Data obtained during startup testing will be incorporated into additional training as needed. The classroom training will cover various aspects of the EPU including changes to parameters, setpoints, scales, procedures, systems and startup test procedures. The classroom training will be combined with simulator training and will include, as a minimum, a demonstration of transients that show the greatest change in plant response at the EPU RTP compared to current power. Simulator changes and fidelity revalidation will be performed in accordance with ANSI/ANS 3.5-1985.

10.7 Plant Life

The longevity of most equipment is not affected by the EPU. Various programs are implemented to monitor the aging of plant components, including Equipment Qualification, Flow Accelerated Corrosion, and Inservice Inspection. These programs were reviewed and do not significantly change for the EPU. In addition, the Maintenance Rule provides oversight for the other mechanical and electrical components, important to plant safety, to guard against age-related degradation.

11. LICENSING EVALUATIONS

11.1 Evaluation Of Other Applicable Licensing Requirements

The analysis, design, and implementation of EPU were reviewed for compliance with the current plant licensing basis and for compliance with new regulatory requirements and operating experience in the nuclear industry. Plant unique evaluations have been performed for the subjects addressed below.

All of the issues raised by the following sources were evaluated on a plant-specific basis as part of the EPU program. These evaluations conclude that every issue is either: (1) not affected by the EPU, (2) already incorporated into the generic EPU program, or (3) bounded by the plant-specific EPU evaluations.

Code of Federal Regulations (CFR)
NRC TMI Action Items
NRC Action Items (Formerly Unresolved Safety Issues) and New Generic Issues
NRC Regulatory Guides
NRC Generic Letters
NRC Bulletins
NRC Information Notices
NRC Circulars
INPO Significant Operating Experience Reports (applicable to the EPU)
GE Services Information Letters
GE Rapid Information Communication Service Information Letters

Plant-unique items whose previous evaluations could be affected by operation at the EPU Rated Thermal Power (RTP) level have been or are being reviewed. These are (1) the NRC and Industry communications discussed above, (2) the safety evaluations for work in progress and not yet integrated into the plant design, (3) the temporary modifications that could have been reviewed prior to the EPU and still exist after EPU implementation, and (4) the plant emergency operating procedures (EOPs). These items have been reviewed for possible effect by the EPU and were found to be acceptable for the EPU, or updated to account for the effects of EPU. Administrative controls will be put in place to ensure compliance is maintained during the implementation period of EPU.

11.2 Impact On Technical Specifications

Implementation of the EPU requires revision of a number of the Technical Specifications. Table 11-1 contains a list of Technical Specification items, which are changed to implement the EPU.

11.3 Environmental Assessment

Environmental evaluations were performed, and determined that the EPU would not involve an unreviewed environmental issue and would meet the criteria established for a categorical exclusion to not require an environmental review.

The environmental effects of EPU will be controlled at the same levels as for the current analyses. That is, none of the present limits for plant environmental releases, such as effluent discharge temperature or plant vent radiological limits, will be increased as a consequence of EPU. Section 6 shows how plant operation would be managed for a plant already on heat sink limits such that the existing limits would not be violated with the EPU. In this example, the plant would take advantage of EPU when the weather was such that the waste heat could be accommodated without exceeding limits, and the plant power would be reduced as necessary to not violate the ultimate heat sink temperature limit. This management scheme is appropriate at both the current and the EPU RTP level should unusual environmental conditions develop which need to be accommodated by the plant. A comparable management scheme is intended for all such environmental limits with which the plant is presently required to comply.

Non-radioactive environmental discharges increase very slightly due to EPU RTP level. Liquid discharges may be slightly warmer and/or have small increases in dissolved and suspended solids. There is essentially no change in the non-radiological atmospheric releases.

The non-radiological environmental effects of the proposed EPU were reviewed based on the information submitted in the Environmental Report, Operating License Stage (ER/OL); the NRC Final Environmental Statement (FES); and the requirements of the Environmental Protection Plan (EPP). Based on this review, it is concluded that the proposed EPU will have no significant effect on the non-radiological elements of concern and the plant will be operated in an environmentally acceptable manner as established by the FES. Existing Federal, State and local regulatory permits presently in effect will accommodate the EPU without modification. The makeup water sources requirements are not increased beyond the present EPP. Effects to air, water, and land resources will be essentially non-existent.

The evaluation of effects of EPU on radiological effluents or offsite doses, as evaluated in the ER/OL and the FES, is summarized in Section 8. There may be very slight increases in the radionuclides released to the environment through gaseous and liquid effluents, but in quantities well within design limits. The quantity (number) of spent fuel bundles discharged per refueling cycle will not be significantly affected by the EPU, however, the short-term radioactivity level will be slightly higher, but still below the previously established limits. The evaluation concludes that these effects of EPU will be insignificant, because the normal effluents and doses will remain well within 10 CFR 20 and 10 CFR 50, Appendix I limits.

The proposed EPU does not require a change to the Environmental Protection Plan or constitute an unreviewed environmental question because it does not involve:

(a) A significant increase in any adverse environmental effect previously evaluated in the final statement, environmental effect appraisals, or in any decisions of the Atomic Safety and Licensing Board; or

(b) A significant change in effluents or power level; or

(c) A matter not previously reviewed and evaluated in the documents specified in Item (a), which may have a significant adverse environmental effect.

The evaluations also establish that the EPU qualifies for a categorical exclusion not requiring an environmental review in accordance with 10 CFR 51.22(c)(9) because it does not:

(a) Involve a significant hazard,

(b) Result in a significant increase in the amounts of any effluents that may be released offsite; or

(c) Result in a significant increase in individual or cumulative occupational radiation exposure.

11.4 Significant Hazards Consideration Assessment

11.4.1 Introduction

Uprating the power level of nuclear power plants can be done safely within certain plant-specific limits, and is an extremely cost effective way to increase the installed electricity generating capacity. Several light water reactors have already been uprated world wide, including numerous boiling water reactors (BWRs) in the United States, Switzerland and Spain.

All significant safety analyses and evaluations have been performed, and their results justify uprating the licensed thermal power at the Clinton Power Station (CPS) by 20% to 3473 MWt.

11.4.1.1 Modification Summary

An increase in electrical output of a BWR plant is accomplished primarily by generation and supply of higher steam flow for the turbine generator. Most BWR plants, as currently 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 and data) 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 margins between calculated safety analysis results and the licensing limits. These available safety analysis differences, combined with the excess as-designed equipment, system and component capabilities, provide BWR plants with the capability to increase their thermal power ratings between 5 and 10% without major nuclear steam supply system (NSSS) hardware modifications, and to provide for power increases to 20% with limited non-safety hardware modifications, with no significant increase in the hazards presented by the plant as approved by the Nuclear Regulatory Commission (NRC) at the original license stage.

The plan for achieving higher power is to modestly extend the power/flow map by increasing core flow along existing flow control lines. However, there is no increase in the maximum core flow limit or operating pressure over the pre-extended power uprate (EPU) values. For EPU operation, the plant can readily be modified to have adequate control over inlet pressure conditions at the turbine, to account for the pressure drop through the steam lines at higher flow and to provide sufficient pressure control and turbine flow capability.

11.4.2 Discussions of Issues Being Evaluated

Plant performance and responses to hypothetical accidents and transients have been evaluated for an EPU license amendment. This safety assessment summarizes the safety significant plant reactions to events analyzed for licensing the plant, and the potential effects on various margins of safety, and thereby concludes that no significant hazards consideration will be involved.

11.4.2.1 EPU Analysis Basis

CPS is currently licensed for a 100% power level of 2894 MWt [100% of the Original Licensed Thermal Power (OLTP)]. The original safety analysis basis assumed that the reactor had been operating continuously at a power level at least 1.02 times the OLTP. For example, the original ECCS-LOCA and transients analyses are based on 104.2% of OLTP. The EPU RTP level included in this evaluation is a 120% thermal EPU (3473 MWt) of the OLTP value. Therefore, this EPU increases power less than 16% over the value used in the original ECCS-LOCA and transient safety analyses. The EPU safety analyses are based on a power level of at least 1.02 times the EPU RTP level, except that some analyses are performed at 100% RTP, because the 2% power factor from Regulatory Guide 1.49 is already accounted for in the analysis methods.

11.4.2.2 Margins

The above EPU analysis basis ensures that the power dependent margins prescribed by the Code of Federal Regulations (CFRs) are maintained by meeting the appropriate regulatory criteria. NRC-approved or industry-accepted computer codes and calculational techniques were used to perform the calculations that demonstrate meeting the acceptance criteria. Similarly, design margins specified by application of the ASME design rules are maintained, as are other margin-ensuring criteria used to judge the acceptability of the plant. Environmental margins are maintained by not increasing any of the present limits for releases, such as ultimate heat sink temperature or plant vent radiological limits.

11.4.2.3 Fuel Thermal Limits

No change is required in the basic fuel design to achieve the EPU RTP level or to meet the plant licensing limits. No increase in allowable peak bundle power is requested for the EPU. Analyses for each fuel reload will continue to meet the criteria accepted by the NRC as specified in NEDO-24011 (GESTAR II) or otherwise approved in the Technical Specifications. In addition, future fuel designs will meet acceptance criteria approved by the NRC.

11.4.2.4 Makeup Water Sources

The BWR design concept includes a variety of ways to pump water into the reactor vessel to deal with all types of events. There are numerous safety-related and nonsafety-related cooling water sources. The safety-related cooling water sources alone would maintain core integrity by providing adequate cooling water. There are high and low pressure, high and low volume, safety and non-safety grade means of delivering water to the vessel. These means include the feedwater and condensate system pumps, the low pressure emergency core cooling systems (ECCS) (Low Pressure Coolant Injection (LPCI) and Low Pressure Core Spray (LPCS)) pumps, the high pressure ECCS (High Pressure Core Spray (HPCS)) pump, the Reactor Core Isolation Cooling (RCIC) pump/turbine, the Standby Liquid Control (SLC) pumps, and the Control Rod Drive (CRD) pumps.

The EPU does not result in an increase or decrease in the available water sources, nor does it change the selection of those assumed to function in the safety analyses. NRC-approved methods were used for analyzing the performance of the ECCS during Loss-of-Coolant Accidents (LOCAs).

The EPU results in a 20% increase in decay heat, and thus, the core cooling time to reach cold shutdown increases. This is not a safety concern, and the existing cooling capacity can bring the plant to cold shutdown within an acceptable time span.

11.4.2.5 Design Basis Accidents

Design Basis Accidents (DBAs) are very low probability hypothetical events whose characteristics and consequences are used in the design of the plant, so that the plant can mitigate their consequences to within acceptable regulatory limits. For BWR licensing evaluations, capability is demonstrated for coping with the range of hypothetical pipe break sizes in the largest recirculation, steam, and feedwater lines, a postulated break in one of the ECCS lines, and the most limiting small lines. This break range bounds the full spectrum of large and small, high and low energy line breaks; and the success of plant systems to mitigate the accidents, while accommodating a single active equipment failure in addition to the postulated LOCA. Several of the most significant licensing assessments are made using these LOCA ground rules. These assessments are:

- Challenges to Fuel (ECCS Performance Analyses) in accordance with the rules and criteria of 10 CFR 50.46 and Appendix K wherein the predominant criterion is the fuel peak clad temperature (PCT).

- Challenges to the Containment wherein the primary criteria of merit are the maximum containment pressure calculated during the course of the LOCA and maximum suppression pool temperature for long-term cooling in accordance with 10 CFR 50 Appendix A Criterion 38.

- DBA Radiological Consequences calculated and compared to the criteria of 10 CFR 100.

11.4.2.6 Challenges to Fuel

The ECCS are described in Section 6.3 of the plant Updated Safety Analysis Report (USAR). The ECCS Performance Evaluation was conducted through application of the 10 CFR 50 Appendix K evaluation models, and demonstrates the continued conformance to the acceptance criteria of 10 CFR 50.46. As mentioned above, a complete spectrum of pipe breaks was investigated from the largest recirculation line down to the most limiting small line break. The licensing safety margin is not affected by the EPU. The increased PCT consequences for the EPU are insignificant compared to the large amount by which the results are below the regulatory criteria. Therefore, the ECCS acceptance criteria continue to be satisfied.

11.4.2.7 Challenges to the Containment

The effect of the EPU on the peak values for containment pressure and temperature confirms the suitability of the plant for operation at the EPU RTP level. Also, the effects of the EPU on the conditions that affect the containment dynamic loads are determined, and the plant is judged satisfactory for EPU RTP operation. Where plant conditions with the EPU are within the range of conditions used to define the current dynamic loads, current safety criteria are met and no further structural analyses is required. The change in short-term containment response is negligible. Because there will be more residual heat with the EPU, the containment long-term response slightly increases. However, containment pressures and temperatures remain below their design limits following any design basis accident, and thus, the containment and its cooling systems are judged to be satisfactory for EPU operation.

11.4.2.8 Design Basis Accident Radiological Consequences

The USAR provides the radiological consequences for each DBA. The magnitude of the potential consequences is dependent upon the quantity of fission products released to the environment, the atmospheric dispersion factors and the dose exposure pathways. The atmospheric dispersion factors and the dose exposure pathways do not change for the EPU. Therefore, the only factor, which could influence the magnitude of the consequences, is the quantity of activity released to the environment. This quantity is a product of the activity released from the core and the transport mechanisms between the core and the effluent release point.

For the EPU, the Control Rod Drop Accident (CRDA), LOCA, Fuel Handling Accident (FHA) and the Main Steam Line Break Accident (MSLBA) were reevaluated.

For the MSLBA, the quantity of activity in the primary coolant used in the evaluation of this postulated event is unaffected by the EPU. The activity in the primary coolant is based on the design basis coolant activity, which remains unchanged for the EPU.

For the remaining DBAs (i.e., CRDA, LOCA, and FHA), the only parameter of importance is the activity released from the fuel. Because the mechanism of fuel failure is not influenced by the EPU, the only parameter of importance is the actual inventory of fission products in the fuel rod. If the only parameter affecting fuel is an increase in thermal power, then the increase in the quantity of fission products can be assumed to be proportional to the increase in power.

The DBA, which has historically been limiting from a radiological viewpoint, is the LOCA, for which USNRC Regulatory Guide 1.3, or its equivalent, has been applied. For this accident, it is assumed that 100% of the noble gases and 50% of the iodines in the core are released to the primary containment. These release fractions are not influenced by the EPU. The radiological consequences of the LOCA were reanalyzed and the increase in these consequences as a result of EPU remains below regulatory guidelines. The EPU LOCA evaluation results include the 2% power uncertainty factor from Regulatory Guide 1.49.

The results of all radiological analyses remain below the 10 CFR 100 guideline values. Therefore, radiological safety margins will be maintained.

11.4.2.9 Transient Analyses

The effects of plant transients were evaluated by investigating a number of disturbances of process variables and malfunctions or failures of equipment according to a scheme of postulating initiating events. These events were primarily evaluated against the Safety Limit Minimum Critical Power Ratio (SLMCPR). The Operating Limit MCPR is increased appropriately to assure that the SLMCPR is not infringed upon, if any transient is initiated from the EPU RTP level. In addition, the limiting transients are analyzed for each specific fuel cycle. Licensing acceptance criteria are not exceeded.

11.4.2.10 Combined Effects

The EPU analyses use fuel designed to current NRC-approved criteria and are operated within NRC-approved limits to produce more power in the reactor, and thus, increase steam flow to the turbine. NRC-approved design criteria are used to assure equipment mechanical performance at the EPU conditions. Scram frequency is minimized by small adjustments to reactor instrumentation. These adjustments are attributed to the small changes in the reactor operating conditions. DBAs are hypothesized to evaluate challenges to the fuel, containment and off-site dose limits. These challenges have been evaluated separately in accordance with extremely conservative regulatory procedures such that the separate effects are more severe than any combined effects. The off-site dose evaluation in accordance with Regulatory Guide 1.3 and

SRP-15.6.5 provides a more severe DBA radiological consequences scenario than the combined effects of the hypothetical LOCA, which produces the greatest challenge to the fuel and/or containment. That is, the DBA, which produces the highest PCT and/or containment pressure, does not damage large amounts of fuel, and thus, the source terms and resulting doses would be much smaller than those postulated in conformance with Regulatory Guide 1.3 evaluations.

11.4.2.11 Non-LOCA Radiological Release Accidents

All of the other radiological releases discussed in Regulatory Guide 1.70 USAR Chapters 11 and 15 are either unchanged because they are not power-dependent, or increase at most by the amount of EPU. Most of the radiological assessments presented in the USAR for the current power level were based on ≥ 102% of the current power level, thus the assessment for these events at the EPU is usually only a 20% increase in the calculated dose. The dose consequences for all of the non-LOCA radiological release accident events is bounded by the "Design Basis Radiological Consequences" events discussed above.

11.4.2.12 Equipment Qualification

Plant equipment and instrumentation have been evaluated against the criteria appropriate for the EPU. Significant groups/types of equipment have been justified for the EPU by generic evaluations. Some of the qualification testing/justification at the current RTP level was done at more severe conditions than the minimum required. In some cases, the qualification envelope did not change significantly due to the EPU. A process has been developed to ensure qualification of the equipment whose current qualification does not already bound the EPU conditions.

11.4.2.13 Balance-of-Plant

Balance-of-plant (BOP) systems/equipment used to perform safety-related and normal operation functions have been reviewed for the EPU in a manner comparable to that for safety-related NSSS systems/equipment. This includes, but was not necessarily limited to, all or portions of the Main Steam, Feedwater, Turbine, Condenser, Condensate, Essential and Non-essential Service Water, Emergency Diesel Generator, BOP piping, and support systems. Significant groups/types of BOP equipment/systems are justified for the EPU by generic evaluations. Plant-specific evaluations justify EPU operation for BOP systems/equipment that are not generically justified.

11.4.2.14 Environmental Consequences

The environmental effects of the EPU will be controlled below the same limits as for the current power level. That is, none of the present environmental release limits, such as ultimate heat sink temperature and plant vent radiological release limits, will be increased as a result of the EPU.

11.4.2.15 Technical Specification Changes

The Technical Specifications ensure that plant process variables and system performance parameters are maintained within the values assumed in the safety analyses. That is, the Technical Specification parameters (process variables, Allowable Values, operating limits, etc.) are selected such that the actual equipment is maintained equal to or more conservative than the assumptions used in the safety analyses. The Technical Specification changes justified by the safety analyses summarized in this report are listed in Table 11-1.

Proper account is taken of inaccuracies introduced by instrument drift, instrument accuracy, and calibration accuracy. For example, to assure conservatism in a high reactor pressure safety analysis event, the high reactor pressure trips are set lower in the Technical Specifications than those used in the safety analysis. This assures that the actual plant responses will be less severe than those represented by the safety analysis. Similarly, the Technical Specifications address equipment operability (availability) and put limits on equipment out-of-service (not available for use) times such that the actual plant can be expected to have at least the complement of equipment available to mitigate abnormal plant events assumed in the safety analyses. Because the safety analyses for the EPU show that the results are acceptable within regulatory limits, public health and safety is confirmed. Technical Specification changes consistent with the EPU RTP level are made in accordance with methodology already approved for the plant and continue to provide a comparable level of protection as Technical Specifications previously issued by the NRC.

11.4.3 Assessment Against 10 CFR 50.92 Criteria

The assessment of significant hazards consideration is included in the licensee submittal.

Table 11-1

Technical Specifications and Bases Affected by EPU

TS Location	Description of Change
1.1 Definitions	Revise value of rated thermal power definition to EPU level.
2.1.1.1, App. 3.2.1, 3.2.1 Action B.1, SR 3.2.1.1, App. 3.2.2, 3.2.2 Action B.1, SR 3.2.2.1, App. 3.2.3, 3.2.3 Action B.1, SR 3.2.3.1, 3.3.1.1.Action F.1, SR 3.3.1.1.2, Table 3.3.1.1-1 Function 5, SR 3.4.3.1, App. 3.7.6, 3.7.6 Action B.1	The Safety Limit % RTP basis and the thermal limits monitoring LCO, Applicability, Actions and SR % RTP thresholds are reduced from 25% to 21.6%.
3.1.3 Condition D, App 3.1.6, SR 3.3.2.1.4, SR 3.3.2.1.5, Table 3.3.2.1-1 footnote c	Reduce the thermal power applicability by the ratio of the power increase (2894/3473), from 20% RTP to 16.7% RTP.
Figure 3.1.7-1	Replace the sodium pentaborate solution (curve) figure, to be consistent with the ATWS analysis in Section 9.3.1. The change is associated with changing the minimum allowable solution concentration to 10.8%.
3.3.1.1 Action E.1, SR 3.3.1.1.16, Table 3.3.1.1-1 Functions 9 and 10	Reduce the turbine stop valve (TSV) Closure and TCV Fast Closure scram bypass power level by the ratio of the power increase (2894/3473), from 40% RTP to 33.3% RTP.
Table 3.3.1.1-1 Function 2.b.	Revise the APRM Flow Biased scram equations for two and single recirculation loop operation.
SR 3.3.2.1.2, SR 3.3.2.1.5, Table 3.3.2.1-1 note (b)	Reduce the % RTP values associated with the LPSP by the ratio of the power increase (2894/3473), from 35% RTP to 29.2% RTP.
App. 3.3.4.1, 3.3.4.1 Action D.2, SR 3.3.4.1.4	Reduce the applicability % RTP values by the ratio of the power increase (2894/3473), from 40% RTP to 33.3% RTP.
Table 3.3.6.1-1 Function 1.c.	Revise the Main Steam Line Flow – High allowable value.
LCO 3.4.1 B.1, 3.4.1 Condition and Action E	To maintain the same absolute thermal power basis for single recirculation loop operation, reduce the % RTP values by the ratio of the power increase (2894/3473), from 70% RTP to 58 % RTP.
Figure 3.4.1-1	Revise the thermal power (% rated) axis by the ratio of power increase (2894/3473).
SR 3.4.11.8, SR 3.4.11.9	Reduce power level by the ratio of the power increase (2894/3473), from 30% RTP to 25% RTP.

12. REFERENCES

1. GE Nuclear Energy, "Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate," Licensing Topical Report NEDO-32424, Class I (Non-proprietary), April 1995.

2. BWROG-94078, "BWR Owner's Group Guidelines for Stability Interim Corrective Action," June 1994.

3. GE Nuclear Energy, "Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications," NEDO-32465-A, August 1996.