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DAEC Plant Support Center
Operated by Nuclear Management Company, LLC

April 17, 2001
NG-01-0500

Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Station 0-P1-17
Washington, DC 20555-0001

Subject:	Duane Arnold Energy Center Docket No: 50-331 Op. License No: DPR-49 Submittal of Non-Proprietary Version of the Safety Analysis Report for Technical Specification Change Request (TSCR-042) – Extended Power Uprate (TAC # MB0543)
Reference:	NG-00-1900, "Technical Specification Change Request (TSCR-042): 'Extended Power Uprate'," dated November 16, 2000.
File:	A-117, SPF-189

Dear Sir(s):

On February 14, 2001, a conference call was held with the NRC Staff to discuss the submittal of a non-proprietary version of the Safety Analysis Report (NEDC-32980P) included with the referenced amendment request. As a result of this conference call, we agreed to submit both a non-proprietary version of the report, similar to that supplied on a previous extended power uprate application; and, to revise the docketed proprietary version to release some of the information identified as proprietary. The Attachment to this letter contains a non-proprietary version (NEDO-32980) of the Safety Analyis Report, which is suitable for public disclosure.

The revised version of the proprietary document (NEDC-32980P) is being prepared and will be submitted under separate cover, as soon as it is completed.

No new commitments are being made in this letter.

Please contact this office should you require additional information regarding this matter.

Sincerely yours,

Kenneth S. Putnam
Manager, Nuclear Licensing

A001

Attachment: Non-Proprietary General Electric Report, NEDO-32980, SAFETY ANALYSIS REPORT for DUANE ARNOLD ENERGY CENTER EXTENDED POWER UPRATE

cc: T. Browning
M. Wadley (w/o Attachment)
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Attachment to
NG-01-0500

Non-Proprietary

General Electric Report

NEDO-32980

SAFETY ANALYSIS REPORT for

DUANE ARNOLD ENERGY CENTER

EXTENDED POWER UPRATE

GE Nuclear Energy

NEDO-32980
Revision 0
DRF A22-00100-73
Class I
April 2001

Safety Analysis Report For Duane Arnold Energy Center Extended Power Uprate

E. D. Schrull

GE Nuclear Energy

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NEDO-32980
Revision 0
DRF A22-00100-73
Class I
April 2001

Safety Analysis Report For Duane Arnold Energy Center Extended Power Uprate

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Alliant Energy – IES Utilities

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
ARTS	APRM/RBM/Technical Specifications
ASME	American Society of Mechanical Engineers
AST	Alternative Source Terms
ATWS	Anticipated Transient Without Scram
AV	Allowable Value
B&PV	Boiler and Pressure Vessel
BHP	Brake Horsepower
BOP	Balance-of-Plant
BWR	Boiling Water Reactor
CACS	Containment Atmosphere Containment System
CAD	Containment Atmosphere Dilution
CAM	Containment Atmosphere Monitoring
CFD	Condensate Filter Demineralizer
CFR	Code of Federal Regulations
CO	Condensation Oscillation
COLR	Core Operating Limits Report
CRD	Control Rod Drive
CRDA	Control Rod Drop Accident
CS	Core Spray
DAEC	Duane Arnold Energy Center
DBA	Design Basis Accident
DC	Direct Current
DOR	Division of Operating Reactors
ECCS	Emergency Core Cooling System
EHC	Electro-hydraulic Control
EOC	End-of-Cycle

Term	Definition
EPRI	Electric Power Research Institute
EPU	Extended Power Uprate
EQ	Environmental Qualification
ESF	Engineered Safety Features
ESW	Emergency Service Water
FAC	Flow Assisted Corrosion
FPC	Fuel Pool Cooling
FPCCS	Fuel Pool Cooling and Cleanup System
FCS	Feedwater Control System
FHA	Fire Hazards Analysis
GDC	General Design Criterion
GE	General Electric Company
GL	Generic Letter
GSW	General Service Water
HELB	High Energy Line Break
HEPA	High Efficiency Particulate Adsorber
HPCI	High Pressure Coolant Injection
HVAC	Heating, Ventilating and Air Conditioning
HWC	Hydrogen Water Chemistry
ILBA	Instrument Line Break Accident
IPE	Individual Plant Evaluation
IPEEE	Individual Plant Examination – External Events
IRM	Intermediate Range Monitor
LCO	Limiting Condition for Operation
LLRPSF	Low Level Radwaste Processing Storage Facility
LOCA	Loss-of-Coolant Accident
LOFW	Loss of Feedwater Flow
LPCI	Low Pressure Coolant Injection
LPSP	Low Power Setpoint
LTP	Long Term Program
LTR	Licensing Topical Report
MCPR	Minimum Critical Power Ratio

Term	Definition
MELB	Moderate Energy Line Break
MELLLA	Maximum Extended Load Line Limit Analysis
MG	Motor-Generator
Mlb/hr	Million Pounds Per Hour
MOV	Motor Operated Valve
MSIV	Main Steam Isolation Valve
MSLBA	Main Steam Line Break Accident
MWe	Megawatt-electric
MWt	Megawatt-thermal
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
NWC	Normal Water Chemistry
OOS	Out of Service
ORTP	Original Rated Thermal Power
PCS	Pressure Control System
PCT	Peak Clad Temperature
PRA	Probabilistic Risk Assessment
PUSAR	Power Uprate Safety Analysis Report
QA	Quality Assurance
RBCCW	Reactor Building Closed Cooling Water
RBM	Rod Block Monitor
RCIC	Reactor Core Isolation Cooling
RCPB	Reactor Coolant Pressure Boundary
RG	Regulatory Guide
RHR	Residual Heat Removal
RHRSW	Residual Heat Removal Service Water
RIPD	Reactor Internal Pressure Difference
RPS	Reactor Protection System
RPT	Recirculation Pump Trip
RPV	Reactor Pressure Vessel

Term	Definition
RRS	Reactor Recirculation System
RTP	Rated Thermal Power
RWCU	Reactor Water Cleanup
RWE	Rod Withdrawal Error
RWM	Rod Worth Minimizer
SBO	Station Blackout
SFP	Spent Fuel Pool
SFU	Standby Filter Unit
SGTS	Standby Gas Treatment System
SJAE	Steam Jet Air Ejectors
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
SSC	Structures, Systems, and Components
STA	Spurious Trip Avoidance
SV	Safety Valve
TAF	Top of Active Fuel
TBV	Turbine Bypass Valve
TCV	Turbine Control Valve
TLO	Two (recirculation) Loop Operation
TSC	Technical Support Center
TSCR	Technical Specification Change Request
TSV	Turbine Stop Valve
UFSAR	Updated Final Safety Analysis Report
UHS	Ultimate Heat Sink

EXECUTIVE SUMMARY

This report summarizes the results of all significant safety evaluations performed that justify extending the licensed thermal power at the Duane Arnold Energy Center (DAEC) to 1912 MWt. This is the second phase of power uprate. The requested license power level is 20% above the Original Rated Thermal Power (ORTP) of 1593 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 DAEC 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. The environmental evaluation demonstrated that the EPU does not involve environmental effects that differ significantly from those previously evaluated for the presently authorized RTP level. Where environmental impacts differ from those previously evaluated, these effects have been shown to be insignificant and within regulatory environmental acceptance criteria. 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 Duane Arnold Energy Center (DAEC), 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 1912 MWt, which corresponds to 120% of the Original Rated Thermal Power (ORTP) level of 1593 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.

A glossary of terms is provided in Table 1-1.

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.

DAEC is currently licensed for a 100% RTP level of 1658 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 1912 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 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 ORTP and the steam flow from the vessel was increased to approximately 16.5% above the current 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. The environmental evaluation (Reference 2) demonstrated that the EPU does not involve environmental effects that differ significantly from those previously evaluated for the presently authorized RTP level. Where environmental impacts differ from those previously evaluated, these effects have been shown to be insignificant and within regulatory environmental acceptance criteria. 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)	1658	1912
Vessel Steam Flow (Mlb/hr)*	7.2	8.4
Full Power Core Flow Range		
Mlb/hr	42.6 to 49.0	48.5 to 49.0
% Rated	87 to 100	99 to 100
Dome Pressure (psia)	1040	No change
Dome Temperature (°F)	549.4	No change
Turbine Stop Valve Inlet Pressure (psia)	998	983
Full Power Feedwater		
Flow (Mlb/hr)	7.15	8.33
Temperature (°F)	424.0	431.4
Core Inlet Enthalpy (Btu/lb)**	528.6	526.9

* At normal feedwater heating

** At the 100% core flow condition

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

(1) MELLLA

(2) Single-loop Operation (SLO)

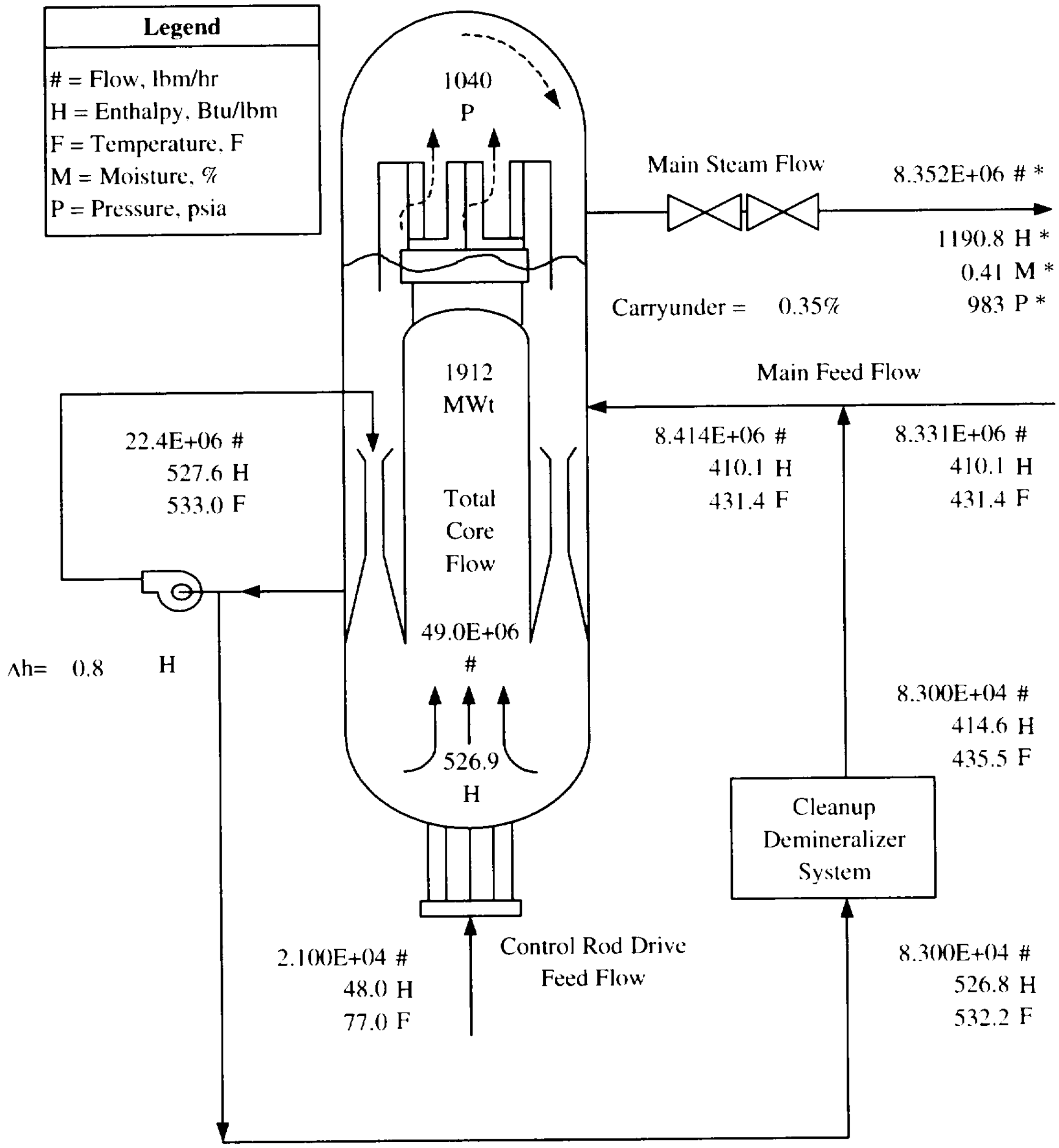
(3) One Automatic Depressurization System (ADS) valve OOS

(4) Average Power Range Monitor (APRM) / Rod Block Monitor (RBM) / Technical Specifications (ARTS)

(5) Turbine Bypass Valve (TBV) OOS

(6) Recirculation Pump Trip (RPT) OOS

(7) Main Steam Isolation Valve (MSIV) OOS



* Conditions at upstream side of TSV

Core Thermal Power	1912.0
Pump Heating	5.1
Cleanup Losses	-2.7
Assumed System Losses	-1.1
Turbine Cycle Use	1913.3 MWt

Figure 1-1. EPU Heat Balance - Nominal

2 REACTOR CORE AND FUEL PERFORMANCE

2.1 Fuel Design and Operation

At the ORTP or the EPU conditions, all fuel and core design limits continue to be met by planned employment 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 new fuel designs are 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 ($\leq$20°C) conditions, 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 Final Safety Analysis Report (UFSAR), 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

DAEC has implemented stability long-term solution Option I-D. It consists of an administratively controlled Exclusion Region and adequate Minimum Critical Power Ratio (MCPR) Safety Limit protection from the flow-biased APRM flux trip.

The Exclusion Region is affected by rated core power and operating conditions, is recalculated for the EPU plant/fuel cycle conditions, and is evaluated for continued applicability for each fuel cycle. The stability application Licensing Topical Report (LTR) has been submitted to the NRC (Reference 3) and will be reviewed on a schedule compatible with the DAEC Power Uprate Safety Analysis Report (PUSAR) submittal and review schedules. Adequate MCPR Safety Limit protection will be demonstrated for each fuel 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 all stresses and fatigue usage factors remain within their original design allowable values.

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

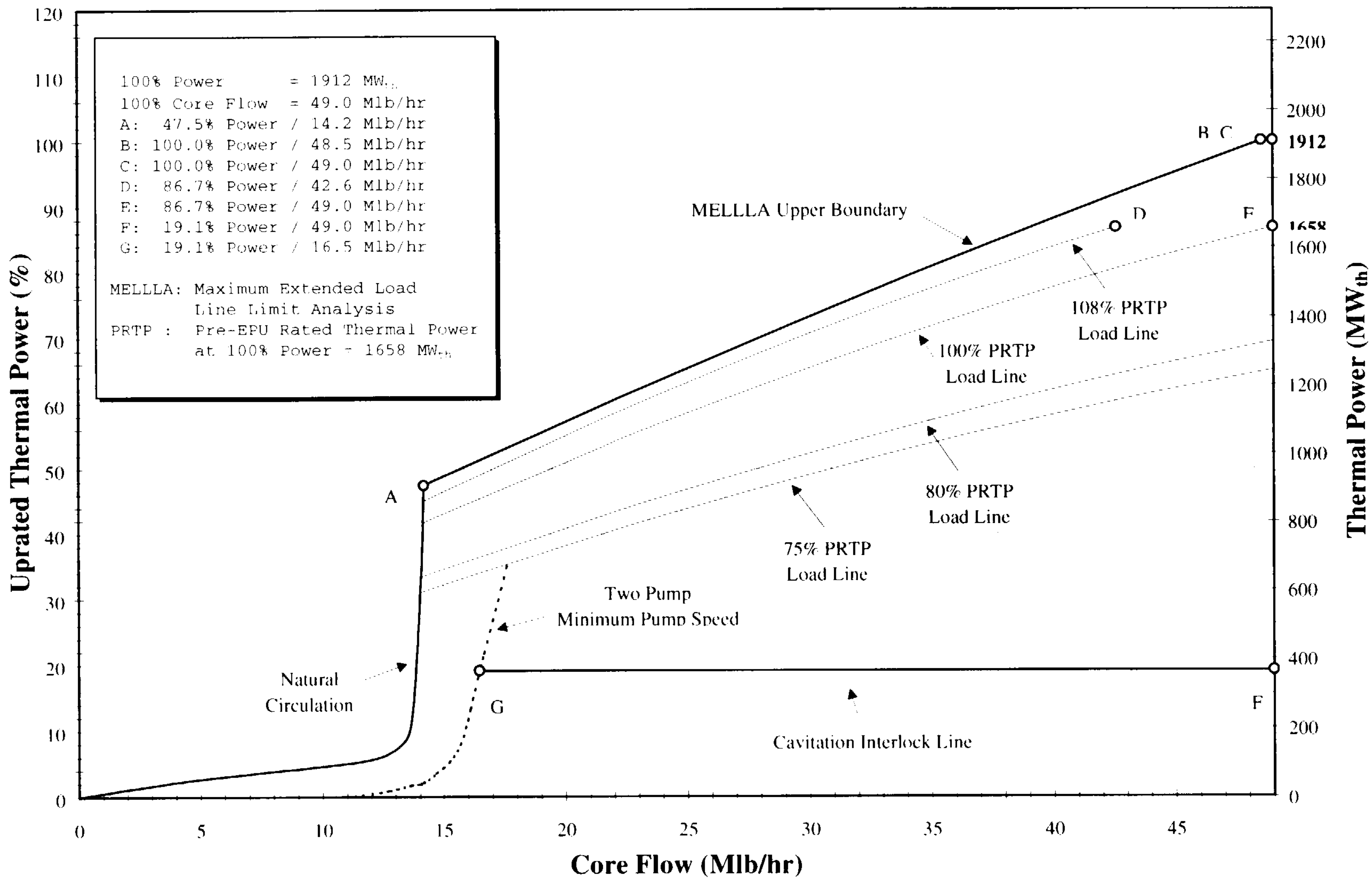


Figure 2-1 Power-Flow Operating Map For EPU

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 safety relief valves (SRVs) and safety valves (SVs) with scram provide this protection. The SRV and SV 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 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 1335 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 UFSAR 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 DAEC 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 1912 MWt and 100% 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 and pressure 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 Recirculation System components are made of stainless steel, and system flow increase due to the EPU is minor. Erosion/corrosion degradation of this system will not increase significantly under EPU conditions.

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 existing design pressure and temperature for the MSIVs are the same as the reactor coolant pressure boundary, and will bound the maximum operating pressure and temperature under EPU conditions.

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 nor is the radioactive content of the reactor water significantly increased. 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, ANSI B31.7 and ANSI B31.1 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/NC/ND, ANSI B31.7, or ANSI B31.1 Power Piping 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 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 insignificant. 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." Evaluations have confirmed that the EPU has no significant effect on flow-induced erosion/corrosion. 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 UFSAR 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 well below the wetwell structural design temperature.
- The calculated wetwell airspace temperature remains well below the wetwell airspace design temperature.
- 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 recirculation suction line break DBA-LOCA analyses. The LOCA dynamic loads with the EPU include pool swell, condensation oscillation (CO) and chugging. For a Mark I plant like DAEC, the vent thrust loads were also evaluated.

The results from the containment analyses performed for the dynamic (Mark I) 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 vent thrust loads with the EPU are calculated to be up to 5% higher than the plant-specific values calculated during the Mark I Containment Long Term Program (LTP). However, the calculated stress ratio and displacement ratio at the EPU conditions remain less than one. Therefore, the increased vent thrust loads are acceptable.

The local pool temperature limit for SRV discharge is specified in NUREG-0783 because of concerns resulting from unstable condensation observed at high pool temperatures in plants without quenchers. Reference 4 provides justification for elimination of this limit for plants with quenchers on the SRV discharge lines. Reference 4 is contingent upon the SRV quenchers being located above the pump suction strainers. For DAEC, certain portions of the SRV quenchers are located below the suction strainers. However, the SRV quenchers and RHR and core spray (CS) pump suction strainers are located in different sections (i.e., bays) in the torus and are widely separated from each other horizontally. Based on this horizontal separation, Reference 4 is applicable to DAEC and the 200°F limit for local suppression pool temperature is not governing.

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, which can affect the SRV loads that might be introduced by the EPU, 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.

All motor-operated valves (MOVs) included in the Generic Letter (GL) 89-10 Program were evaluated for the effects of the EPU. GL 89-10 MOV Program-required calculation revisions, switch setting adjustments, and/or modifications to ensure satisfactory performance will be completed prior to EPU operation.

DAEC installed a hardened wetwell vent system in response to GL 89–16. The design of the existing hardened wetwell vent continues to be acceptable for preventing containment overpressure at the EPU conditions.

Finally, there is no effect on the DAEC's response to GL 96-06 considering the EPU post-accident conditions.

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 Coolant Injection (HPCI) 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 HPCI 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 evaluation of the LPCI System 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 CS System is automatically initiated in the event of a LOCA. When operating in conjunction with other ECCS, the CS System is required to provide adequate core cooling for all LOCA events. The increase in decay heat due to the EPU could increase the calculated PCT following a postulated LOCA by a small amount. The evaluation of the CS System indicates that its existing 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 CS 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 Control Room and Technical Support Center Habitability

The ability of the air cleanup systems in the control room and technical support center to process intake atmosphere is unaffected by the EPU. The charcoal filter bed removal efficiency for radioiodine is unaffected by EPU. As a result of application of alternative source terms (AST) derived from NUREG-1465 (See Section 9.2), the calculated post-DBA-LOCA total iodine loading on the Control Building standby filter units (SFUs) decreases to 8.25E-06 mg/gm of charcoal at the EPU conditions. In addition, the calculated post-DBA-LOCA total iodine loading on the Technical Support Center SFUs decreases to 7.54E-06 mg/gm of charcoal at the EPU conditions. Both of these results are well below the Regulatory Guide 1.52 value of 2.5 mg/gm of charcoal. Therefore, the systems contain sufficient charcoal to ensure iodine removal efficiencies greater than 90%.

4.4 Standby Gas Treatment System

The Standby Gas Treatment System (SGTS) is designed to minimize offsite and control room doses during venting and purging of the primary and secondary containment atmosphere under accident or abnormal conditions. The capacity of the SGTS was selected to maintain the secondary containment at a slight negative pressure. This capability is not affected by the EPU.

The charcoal filter bed removal efficiency for radioiodine is unaffected by the EPU. As a result of application of AST derived from NUREG-1465 (see Section 9.2), the post-DBA-LOCA (the limiting event) total iodine loading is 0.003 mg/gm of charcoal at the EPU conditions, which is well below the Regulatory Guide 1.52 value of 2.5 mg/gm. The system therefore contains sufficient charcoal to ensure iodine removal efficiencies greater than the current design requirement of 99%.

4.5 Post-LOCA Combustible Gas Control

Post-LOCA combustible gas control is provided by the Containment Atmosphere Control System (CACS). The CACS consists of three subsystems: the primary containment purge system, the primary containment nitrogen inerting system, and the Containment Atmosphere Dilution (CAD) System. The CAD System is an engineered safety function (ESF), which includes the Containment Atmosphere Monitoring (CAM) System. The CACS is designed to maintain the post-LOCA containment atmosphere below hydrogen flammability limits by controlling the concentration of oxygen to not exceed 5% by volume. The time required to reach the 5% oxygen limit following the LOCA, conservatively assuming zero containment leakage, decreases from 3.5 days for current RTP to 2.3 days for EPU RTP. This reduction in the time

required for CAD System initiation does not affect the ability of the operators to respond. Therefore, the CACS retains its capability of meeting its design basis function of controlling oxygen concentration following the postulated DBA-LOCA.

Evaluation of the nitrogen requirements to maintain the containment atmosphere below the 5% flammability limit for seven days post-LOCA shows that the minimum stored volume increases from 50,000 scf for current RTP to 67,000 scf for EPU RTP. The CAD nitrogen storage system has sufficient capacity to accommodate this increase in minimum storage volume.

The CAM System is currently required to be operating post-LOCA. The system sample lines are heat traced to ensure accurate readings by preventing condensation in the sample lines whenever the containment atmosphere is significantly higher than the temperature of the sample lines. The existing heat tracing is effective up to 200°F. However, containment atmosphere temperatures have increased, such that the existing heat tracing will no longer be effective and the resulting H_2 and O_2 indications will not be within the required accuracy. Because of operating temperature limits on the radiation monitors that also use these sample lines, it is not practical to increase the temperature setting on the existing heat tracing to bound the new containment temperatures. As stated previously, the time required to reach the 5% oxygen limit following a hypothetical LOCA is 2.3 days at EPU conditions. The EPU containment analysis shows that the post-LOCA containment temperatures for the DBA would return below 200°F approximately 22 hours after the event begins. Therefore, by establishing an administrative requirement to have H_2 and O_2 monitoring in operation within 24 hours following the event, it would be possible to monitor and assess the H_2 and O_2 trends for at least 24 hours prior to the 2.3 day mark when the CAD System would be required to inject nitrogen to maintain the containment below the 5% O_2 limit.

5 INSTRUMENTATION AND CONTROL

5.1 Nuclear Steam Supply System

This EPU involves no increase in reactor pressure, and the pressure-dependent setpoints do not require modification. However, increases in core thermal power and steam flow affect some instrument setpoints.

The APRM power signals will be rescaled to the 1912 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 Rod Worth Minimizer (RWM) does not perform a safety-related function. The function of the RWM is to support the operator by enforcing rod patterns until reactor power has reached appropriate levels. No adjustment to operational settings are required as the inputs used by the RWM for the Low Power Setpoint (LPSP) are currently bypassed, such that the RWM is operational at all power levels.

The determination of instrument setpoints is based on plant operating experience, conservative licensing analyses, and/or (limiting design/operating values. Each setpoint is selected with sufficient margin between the actual trip setting and the value used in the safety analysis (analytical limit) to allow for instrument accuracy, calibration, and drift. Sufficient margin is provided wherever possible between the actual trip setting and the normal operating limit to preclude inadvertent initiation of the protective action, i.e., spurious trip avoidance (STA).

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

- Reactor vessel high-pressure scram
- Anticipated transient without scram recirculation pump trip (ATWS-RPT) high pressure trip
- SRV and SV setpoints
- Main steam high flow isolation

- The APRM high power scram analytical limit remains unchanged, however, the flow-biased scram analytical limit is changed as identified below.
- The Rod Block Monitor (RBM) System has three upscale trip levels, which are based on three thermal power level ranges. These power levels in terms of percent of rated thermal power are not changed.
- Main steam line high radiation isolation
- Low steam line pressure MSIV closure
- Reactor water level instruments
- Main steam line tunnel high temperature isolations
- Low condenser vacuum MSIV isolation
- RCIC steam line high flow isolation
- HPCI steam line high flow isolation

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 turbine stop valve closure and turbine control valve fast closure scram bypass analytical limit is reduced by the ratio of the power increase. However, the new analytical limit does not change in terms of absolute power.

5.2 Balance-Of-Plant

Operation of the plant at the EPU RTP level has minimal effect on the BOP system instrumentation and control devices. Any required changes will be performed prior to operation at the 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 normal operating range. The PCS consists of the pressure regulation system, turbine control valve system and steam bypass valve system. The main turbine speed/load control function is

performed by the main turbine-generator electro-hydraulic control (EHC) system. The steam bypass valve pressure control function is performed by the turbine steam bypass control system. With some (non-safety) power conversion system modifications, the current design has sufficient pressure control range to control system disturbances at the EPU conditions. Thus, the existing main turbine-generator EHC and the steam bypass control system are adequate for the EPU conditions. Specific EHC and steam bypass control system tests will be performed during the power ascension phase.

The turbine EHC system was reviewed for the increase in core thermal power and the associated increase in rated steam flow and implementation of partial-arc admission. For the EPU conditions, new diode function generator circuit boards and steam line resonance compensators will be installed. The control systems are expected to perform normally for EPU RTP operation. No modifications to the turbine control valves or the turbine bypass valves are required for operation at the EPU throttle conditions. Confirmation testing will be performed during power ascension.

The Feedwater Control System (FCS) is used to maintain water level control in the reactor. With modifications to some nonsafety-related system components, the FCS will be capable of controlling feedwater flow as needed to maintain RPV level. The current controller adjustments are expected to be satisfactory for EPU; however, this will be confirmed by performing unit tests during the power ascension to EPU conditions.

The instrument setpoints associated with primary system leak detection have been evaluated with respect to the slightly 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 reviewed, and determined to be adequate for operation with the EPU-related electrical output, as shown in Table 6-1. The review concluded the following.

- With minor modification, the isolated phase bus duct is adequate for the maximum output at both rated and 95% generator output voltage.

- Modifications to the existing main transformers will be necessary to accommodate full EPU power output from the main generator. The associated switchyard components (rated for maximum transformer output) are adequate for the EPU-related transformer output.

- A grid stability analysis was performed for 1790 MWt, considering the increase in electrical output, to demonstrate conformance to General Design Criterion (GDC) 17 (10 CFR 50, Appendix A). At 1790 MWt, there is no significant affect on grid stability or reliability. This analysis will be re-performed when the main transformers are replaced to confirm that there is no affect on grid stability or reliability at 1912 MWt. There are no modifications associated with the EPU, which would increase electrical loads beyond the off-site equipment capability, or which would revise the logic of the distribution systems. There are also no new instabilities as a result of the EPU and further, the EPU increases the dampening of the existing oscillations for some transients.

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 with conservative demand factors applied. 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.

Station loads under emergency operation/distribution conditions (emergency diesel generators) are based on equipment nameplate data. Operation at the EPU RTP level is achieved by utilizing

existing equipment operating at or below the nameplate rating and within the calculated brake horsepower (BHP) for the stated pumps; therefore, under emergency conditions the electrical supply and distribution components are adequate.

No increase in flow or pressure is required of any AC-powered ECCS equipment for the EPU. Therefore, the amount of power required to perform safety-related functions (pump and valve loads) is not increased with the EPU, and the current emergency power system remains adequate. 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 design basis loading, nor revise any control logic; therefore the DC power distribution system is 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 both the increased decay heat generated by the fuel operated at the EPU RTP level and the larger fuel batches discharged to the SFP to implement the 24-month operating cycle. Both normal and emergency off-load scenarios, as described in the UFSAR, were analyzed. This analysis has been submitted to the NRC in a license amendment request under separate cover letter (Reference 5).

Prior to the bulk fuel pool temperature exceeding the 150°F design temperature of the FPCCS, its operation is secured and the RHR System is placed in service in the Fuel Pool Cooling (FPC) assist mode at approximately 120°F. The results of the new analysis show that while there will be a slight extension in the amount of time that the RHR System will need to be in service in its FPC assist mode after EPU implementation, adequate SFP cooling remains assured.

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 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. This conclusion is based on the following considerations.

The safety-related service water systems [Emergency Service Water (ESW) and RHR Service Water (RHRSW) Systems] are designed to provide a reliable supply of cooling water during and following a DBA for the following essential equipment and systems:

RHR heat exchangers;

Emergency diesel-generator;

RHR pump seal coolers (not safety-related);

RHR and CS pump room cooling unit;

HPCI room cooling unit;

RCIC room cooling unit;

Control building chiller;

Core Spray pump motor cooler;

RHR Service Water pump motor coolers;

Heating Ventilating and Air Conditioning (HVAC) instrument air compressors;

Spent fuel pool as emergency makeup water (if needed); and

Containment flooding (if needed).

Evaluations show that the implementation of the EPU does not require a change to the safety-related service water systems.

Regarding the nonsafety-related heat loads, the heat rejected to the General Service Water (GSW) System via the Reactor Building Closed Cooling Water (RBCCW) System and other auxiliary heat

loads increases from the EPU due to an increase in main generator losses rejected to the stator water coolers, hydrogen coolers and exciter coolers in addition to increased bus cooler heat loads. 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. The service water discharge is not to the ultimate heat sink (UHS) but to the Circulating Water System that is cooled by the cooling towers during normal operation. Therefore, the GSW 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 cooling towers are adequate for EPU operation.

Due to the EPU, increased RWCU flow through the RWCU Heat Exchangers increases the heat load on the RBCCW System. Additional fuel pool heat load may require the use of two Fuel Pool Heat Exchangers following a refueling outage increasing the heat load to the RBCCW System, which in turn, may require the operation of all three pumps and heat exchangers. All other heat loads in equipment cooled by the RBCCW do not increase significantly because either the temperature and flow rates for the associated heat loads remain within their original design values or are not power-dependent (e.g., sample coolers) and thus, are not affected by the EPU. The operation of the remaining equipment cooled by the RBCCW is not power-dependent and is not affected by EPU.

The UHS for the DAEC is the Cedar River. As a result of operation at the EPU RTP level, the post-LOCA UHS water temperature increases. Following a LOCA, the RHRSW and ESW heat load is normally rejected back to the plant cooling towers. If the cooling towers are not in operation the total RHRSW and ESW heat load is rejected to the Cedar River via the dilution structure.

A review was performed to evaluate the increased UHS heat load for the EPU. The review concludes that the existing UHS System provides a sufficient quantity of water, using the River Water Supply pumps, at a temperature of 95°F (design temperature) or less following a design basis LOCA. A comparison of the state discharge limits (to the air and ultimate heat sink) to the bounding analysis discharges for the EPU demonstrates that the plant remains within the state discharge limit, during EPU operation.

The environmental effects of EPU operation are controlled at the same levels as for the original analyses.

6.5 Standby Liquid Control System

The operating capability of the Standby Liquid Control System (SLCS) is unaffected by the EPU. However, a new fuel design combined with an extension in the fuel cycle operating time requires an increase in the minimum reactor boron concentration from 600 ppm to 660 ppm. The increase in reactor boron concentration is achieved through an increase in the volume of the stored neutron absorber solution. A Technical Specification Change Request (TSCR) requesting this increase in solution volume requirement has been submitted under separate cover (Reference 6). Implementation of the EPU has no adverse effect on the ability of the SLCS to mitigate an ATWS.

As part of the adoption of the AST (Section 9.2), the SLCS is utilized to buffer the pH of the suppression pool in the event of an accident leading to significant core damage and fission product release (i.e., iodine).

6.6 Power-Dependent Heating Ventilation and Air Conditioning

The HVAC systems consist mainly of heating, cooling supply, exhaust and recirculation units in the reactor building, drywell, and turbine building. EPU operation is expected to result in slightly higher process temperatures and a small increase in the heat load due to higher electrical currents in some motors and cables.

The affected areas are the feedwater heater bay and condenser areas in the turbine building. Most other areas are unaffected by the EPU because the process temperatures remain relatively constant. Heat loads in the drywell increase slightly due to increases in the recirculation pump motor horsepower and the feedwater process temperature. The maximum temperature increase is 1.3°F. The condensate, feedwater pump motors and reactor recirculation motor-generator (MG) set motor horsepower increases result in a temperature increase that is less than 2°F. In the turbine building, the temperature at the feedwater heaters increases by less than 5°F due to the EPU.

Based on a review of design basis calculations and environmental qualification design temperatures, the above increases are within the excess design capability available. Therefore, the design and operation of the HVAC is not adversely affected by 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 DAEC 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 in compliance with the requirements of 10 CFR 50 Appendix R assuming EPU conditions. The results of the Appendix R 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 Appendix R event. Therefore, the EPU has no adverse effect on the ability of the systems and personnel to mitigate the effects of an Appendix R fire event, and satisfies the requirements of Appendix R with respect to achieving and maintaining safe shutdown in the event of a fire.

6.8 Systems Not Impacted By EPU

The following systems are not affected by operation of the plant at the EPU condition:

Administration Building HVAC
Administration Building Sumps
Administration Building Elevator
Annunciators (Except Fire Protection)
Annunciators (Fire Protection)
Auxiliary Heating System Boiler
Badging Center HVAC
Breathing Air
Buildings and Structures
Cathodic and Freeze Protection
Control Building/Standby Gas Treatment Instrument Air System "A"
Control Building/Standby Gas Treatment Instrument Air System "B"
Chemical Labs and Equipment

CO_2 Fire Protection

Computer Room Halon Fire Protection

Containment Isolation Monitoring

Control Building Chiller and Auxiliaries "A"

Control Building Chiller and Auxiliaries "B"

Drywell Cranes and Material Handling

Data Acquisition Center HVAC

Decontamination Facilities

Domestic Water

Doors

Drywell Personnel Hatch

Electrical Manhole Sump Pump

Emergency Lights

Emergency Lights for Safe Shutdown Path

External and Internal Phone

Fire Barriers

Fire Detection

Fire Protection

Fuel Handling Systems

HVAC, Intake Structure

Inactive Solid Radwaste (Refuse)

Instrument AC Control Power

Instrument Shop and Equipment

Intake and Traveling Screens

Intrusion Alarms and Monitors

Lighting Panel Power Supply

Low Level Radwaste Processing Storage Facility (LLRPSF) Area Sumps

LLRPSF Crane

LLRPSF HVAC

Lube Oil Transfer and Storage

Machine Shop and Offgas Building HVAC

Meteorological System

Miscellaneous Cranes

Nitrogen System
Offgas Building Sumps
P. A. System/Fire and Evacuation
Personnel Access Hatch
Radio Communications
Radwaste Building HVAC
Radwaste Building Sumps
Radwaste Sumps
Reactor and Radwaste Building Sampling
Reactor Building Access Control
Reactor Building Crane and Elevator
Reactor Manual Control
Reactor Protection System
Sanitary Drains
Screen Wash
Security Building HVAC
Security Fence
Seismic Monitoring
Service Air
Smoke Detection
Sound Powered Phones
Storm Drains
Sumps, Reactor Building
Tools
Torus Vacuum Breakers
Technical Support Center (TSC) HVAC
TSC Standby Power
Turbine Building Access Control
Turbine Building Crane
Turbine Building Sampling

Some DAEC systems are affected to a very small extent by operation of the plant at the EPU condition. For these systems, the effects are insignificant to the design or operation of the system

and equipment:

- Chlorination and Acid Feed
- Computers
- HVAC, Pumphouse
- HVAC, Standby Diesel Generator
- Hydrogen Water Chemistry System
- Instrument Air
- Makeup Demineralizer
- Panels
- Pleasant Creek Pump Station and Valves
- Sumps, Drywell
- Turbine Building Sumps
- Zinc Injection System

Table 6-1

EPU Plant Electrical Characteristics

Gross Generator Output (MWe)	677
Rated Voltage (KV)	22
Power Factor	0.95
Generator Rated Output (MVA)	715
Current Output (A)	18770
Isolated Phase Bus Duct Rating:	
Main Section (A)	20000
Auxiliary Transformer Section (A)	1200
Main Transformers Rating (MVA) (Note 1)	≥715,225

Notes:

1. Final nameplate rating to be determined.

7 POWER CONVERSION SYSTEMS

The power conversion systems were originally designed to utilize the energy available from the NSSS and were 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 rated steam flow.

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%; however, the turbine currently operates with a flow margin of approximately 4%. At the EPU conditions, turbine operation requires an increase in throttle flow to approximately 122% of original rated. To maintain the GE standard flow margin of 3%, the high-pressure turbine will be modified to increase its flow passing capability.

A rotor missile analysis was performed at the EPU conditions based on the NRC approved methodology in NUREG-1048, which applies to units with GE monoblock rotors. Based on the calculated results of control system failure, which is on the order of 10^{-8} per year, the missile failure probability is acceptable.

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 potential decrease in the life of the condenser tubes was assessed against the effects of the increased shell side flows. Existing life-cycle management programs of non-destructive

examinations and periodic tube inspections will monitor and address condenser tube life reduction.

- 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 mechanical vacuum pump. 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 ORTP flows.

7.3 Turbine Steam Bypass

The turbine bypass valves were initially rated for a total steam flow capacity of not less than 25% of the original rated reactor steam flow, or 1.71 Mlb/hr. Each of two bypass valves is designed to pass a steam flow of 856,000 lbm/hr for a total bypass capacity of 1.71 Mlb/hr. At the EPU RTP level, rated reactor steam flow is 8.35 Mlb/hr, resulting in a bypass capacity of 20.6%, which was used in the transient analysis. Thus, bypass capacity remains adequate for the EPU.

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 a major affect on plant availability and capability to operate at the EPU condition.

Some modifications to nonsafety-related equipment in the Feedwater and Condensate Systems are necessary to attain full EPU core thermal power. The equipment modifications may include resizing the feedwater and condensate pump impellers, pump motors, and motor/pump couplings, and possible modifications of the feedwater flow regulation valves. Implementation of these modifications will follow the 10 CFR 50.59 process. In addition, a review of these modifications will be conducted to confirm that they do not constitute a material alteration.

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.

The potential for life cycle reduction of the feedwater heaters was assessed against the effects of the flows, pressures, and temperature at the EPU conditions. Based on this assessment, it may be necessary to modify the 3A/B, 4A/B, and 5A/B feedwater heaters to accommodate the higher flow associated with EPU operation. Existing life-cycle management programs of non-destructive examinations and periodic tube inspections will monitor and address feedwater heater life reduction.

A transient analysis was performed to determine the effects of a single feedwater pump trip with subsequent recirculation system runback. The results of the analysis indicate that similar to the pre-EPU conditions, the remaining feedwater pump is not sufficient to maintain level to prevent a reactor low water level scram. The potential exists that the Level 2 setpoint may also be exceeded.

The effect of the EPU on the condensate filter demineralizers (CFDs) was reviewed. In summary, the increased system flow rate due to EPU will require minor non-safety modifications to the flow monitoring equipment while the increased system pressure may require additional analysis and/or modifications.

8 RADWASTE SYSTEMS AND RADIATION SOURCES

8.1 Liquid Waste Management

Based on a review of plant operating effluent reports, revised Appendix I evaluations, and the slight increase expected from EPU, it is concluded that the requirements of 10 CFR 20 and 10 CFR 50, Appendix I will 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.

The non-condensable gases (which primarily consist of N-13, N-16, O-19 and various noble gases) are continuously removed from the main condensers by the 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 steaming 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 well within limits following implementation of the EPU.

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 actual DAEC plant operating data under normal water chemistry conditions, the radiolytic H_2 flow rate increases, but remains well within the design capacity of the system. For hydrogen water chemistry operation with the maximum feedwater H_2 addition rates anticipated at DAEC, the net H_2 flow rate to the Offgas System actually decreases due to the

reduction in radiolysis. Hence, normal water chemistry operation is the bounding condition for Offgas System operation at the EPU conditions.

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 no greater than the increase in power.

For post-operation evaluations, two sets of source data are applied. The first is the gamma-ray source, which is used in shielding calculations for the core and for individual fuel bundles. This set of source terms 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 fission product inventories were developed and used in the evaluation of design basis accidents.

8.4 Radiation Sources In The Coolant

Radiation sources in the coolant are primarily a function of fuel defects, power level, and operation of the RWCU System. Because EPU is not expected to significantly affect the fuel defect rate, the coolant activity level should increase (at most) proportional to power. The design basis concentrations for N-16 activity in the reactor water were changed from 50 μCi/g to 60 μCi/g. However, while the magnitude of the source production increases in proportion to power, the concentration in the steam remains nearly constant.

Hydrogen Water Chemistry (HWC) results in a large increase in the concentration of N-16 in the steam relative to the concentration with Normal Water Chemistry (NWC). This is due to the fraction of the total N-16 production that is in volatile chemical form being significantly increased. As a result of increased carryover of N-16 in the steam, the concentration levels in the reactor water should decrease slightly. Consequently, the N-16 levels in the reactor water due to the EPU would increase less than in the case with NWC. The effect of the EPU was based on the current design basis source terms.

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 basis 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, the increase could be slightly higher. 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. In addition, the plant has established successful cobalt reduction, zinc injection, HWC, and noble metal chemical addition programs, which result in a decrease in post-operation radiation levels and/or reduced repairs required in radiation areas.

The change in core inventory resulting from the EPU and application of the AST does not affect the plant's design basis post-accident radiation levels. 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. The post-accident habitability of the Technical Support Center was also evaluated and demonstrated to remain within regulatory dose limits.

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 Technical Specification limits implement the guidelines of 10 CFR 50, Appendix I. A review of the normal radiological effluent doses shows that at original power, the doses are a small fraction of the doses allowed by Technical Specification limits. 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 UFSAR 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 evaluated according to the type of initiating event per Regulatory Guide 1.70, Chapter 15.

Most of the transient events are analyzed at the full EPU RTP and maximum allowed core flow operating point on the power-flow map. Analytical results demonstrate the capability of the design to meet all transient safety criteria for EPU RTP conditions.

The cycle-specific SLMCPRs will be supplied in the Core Operating Limit Reports (COLRs). The original 25% of RTP Technical Specification Safety Limit, transient analysis based Limiting Conditions for Operation (LCOs) and Surveillance Requirements (SRs) were originally based on generic analyses. The 25% RTP value should be reduced by 1/1.153 to 21.7%. Therefore, the LCO and SR thresholds for thermal power and the Safety Limit after uprating remain at the same value of 414.5 MWt as before the EPU (i.e., 25% of the current RTP), or 21.7% of the new RTP.

The Loss of Feedwater Flow (LOFW) transient assuming an additional single failure (loss of HPCI) was analyzed for the EPU. During this low probability event, reactor water level is automatically maintained above the top of the active fuel (TAF) by the RCIC System, without any operator action and the safety criterion is met. Operator action is needed to inhibit the ADS actuation in the near term and for long-term plant shutdown. After water level is restored, the operator would manually control water level, reduce reactor pressure, and initiate RHR shutdown cooling.

9.2 Design Basis Accidents

The radiological consequences of the plant design basis LOCA have been reviewed for impact of the EPU and the implementation of the NUREG-1465 Alternative Source Term as described in Reference 7. The results are within the guidelines of 10 CFR 50.67 and GDC 19 of 10 CFR 50, Appendix A. Other accidents (non-LOCA) analyzed in the UFSAR have similarly been reviewed and remain below their regulatory limits. A TSCR for the implementation of the AST has been submitted under a separate cover letter (Reference 8).

The events analyzed were the LOCA, the Main Steam Line Break Accident (MSLBA), the Fuel Handling Accident (FHA), and the Control Rod Drop Accident (CRDA). The plant-specific results for the EPU shown in Tables 9-1 through 9-4.

9.3 Special Events

A DAEC-specific 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.

Plant response to and coping capabilities for a station blackout (SBO) event are affected slightly by operation at the EPU RTP level, due to an increase in the decay heat. In addition, the initial conditions and assumptions for SBO under the EPU conditions have been revised to be consistent with NUMARC 87-00 and Regulatory Guide 1.155. 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 / AST	Limit
Exclusion Area: Dose (rem TEDE)	0.25	≤ 25
Low Population Zone: Dose (rem TEDE)	0.60	≤ 25
Control Room: Dose (rem TEDE)	4.15	≤ 5
Technical Support Center: Dose (rem TEDE)	4.40	≤ 5

Table 9-2

MSLBA Radiological Consequences

Location	**EPU / AST**	**Limit**
Case 1: Iodine concentration in coolant = 2 μCi/gm dose-equivalent I-131		
Exclusion Area: Dose (rem TEDE)	0.67	≤ 25
Low Population Zone: Dose (rem TEDE)	0.19	≤ 25
Control Room: Dose (rem TEDE)	2.61	≤ 5
Case 2: Iodine concentration in coolant = 0.2 μCi/gm dose-equivalent I-131		
Exclusion Area: Dose (rem TEDE)	6.7E-02	≤ 2.5
Low Population Zone: Dose (rem TEDE)	1.9E-02	≤ 2.5
Control Room: Dose (rem TEDE)	0.26	≤ 5

Table 9-3

FHA Radiological Consequences

Location	EPU / AST	Limit
Exclusion Area: Dose (rem TEDE)	0.94	≤ 6.25
Low Population Zone: Dose (rem TEDE)	0.23	≤ 6.25
Control Room: Dose (rem TEDE)	3.16	≤ 5
Technical Support Center: Dose (rem TEDE)	2.83	≤ 5

Table 9-4

CRDA Radiological Consequences

Location	**EPU / AST**	**Limit**
Exclusion Area: Dose (rem TEDE)	0.06	≤ 6.25
Low Population Zone: Dose (rem TEDE)	0.04	≤ 6.25

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). Evaluation of these piping systems determined that there is no change in postulated break locations.

At the EPU RTP level, HELBs outside the primary containment can cause the pressure and temperature profiles to increase. Implementation of the EPU can cause changes in HELB mass and energy releases due to RPV pressure changes or changes in the fluid conditions within the system piping. In piping with subcooled liquid, an increase in the subcooling due to EPU can result in higher break flows. However, the result of the decrease in enthalpy due to subcooling offsets the slight increase in blowdown rate. The net effect is either no change or a small decrease in the peak temperature in the area outside containment and no change to the peak pressure in the area outside containment. Therefore, the 104.1% ORTP analysis is bounding for the EPU RTP condition. The relative humidity change is negligible.

The HELB analysis evaluation was made for all systems evaluated in the UFSAR. The evaluation shows that the affected building and rooms 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 impact on feedwater line break flooding for the EPU, because flooding for a feedwater line break is bounded by flooding for a main steam line break.

A Moderate Energy Line Break (MELB) break analysis is not within the DAEC licensing basis. Therefore, MELB is not applicable to DAEC for the EPU.

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 devices are located remain adequate. Margins were originally applied to the environmental parameters in accordance with Division of Operating Reactors (DOR) or IEEE 323-1974 guidelines, as applicable; these margins continue to be met for the EPU.

The implementation of the EPU includes the use of an AST based on NUREG-1465 as the DAEC design and licensing basis for evaluating offsite and control room doses. However, the evaluation of the effect of the EPU on Equipment Qualification (EQ) is based on the interim guidance given in SECY-99-240 where the continued use of the TID-14844 post-accident source term release is considered acceptable for evaluating proposed plant modifications on previously-analyzed integrated component doses, regardless of the accident source term used to evaluate offsite and control room doses.

EQ for safety-related electrical equipment located inside the containment is based on MSLBA and/or DBA LOCA conditions and their resultant temperature, pressure, humidity, and radiation consequences, and includes the environment expected to exist during normal plant operation.

Increases in the normal operating temperature do not affect the EQ of the equipment but need to be evaluated as potentially reducing the qualified life of the equipment. For the first hour after the postulated steam line breaks, the temperature profiles using EPU RTP conditions are bounded by the current (104.1% ORTP) Drywell temperature profile. After the first hour, the temperature profiles using EPU RTP conditions exceed the current temperature profile. In accordance with the DAEC EQ program requirements, the equipment inside the containment will be requalified or upgraded to the new temperature profiles as part of the implementation of the EPU.

Similarly, the calculated peak Drywell pressure using EPU RTP conditions is not bounded by the current (104.1% ORTP) peak pressure. However, a comparison of the demonstrated qualification pressure of each component in the Drywell to the higher peak pressure confirmed that the higher peak pressure has no affect on environmental qualification.

The current 40 year normal radiation dose inside the containment was evaluated to increase approximately 20%, i.e., proportional to the increase in EPU RTP. However, the accident dose was evaluated to decrease approximately 18%, due to a change in calculational methods. While both the current and EPU calculations are based on the TID-14844 source term, the EPU

evaluation used a more-detailed integration method for the post-accident dose than that used in the current evaluation. The analysis demonstrates that the increase in the normal drywell dose is offset by the decrease in the accident dose for the EPU.

The current EQ for equipment inside the containment is based on a 100% humidity environment. This is not changed for the EPU.

Accident temperature, pressure, and humidity environments used for qualification of equipment outside containment result from a main steam line break in the steam tunnel, or other high energy line breaks, whichever is limiting for each plant area. The accident temperature, pressure and humidity conditions resulting from the main steam line break or HELB do not change for the EPU.

The EPU normal and accident doses are either bounded by the 104.1% ORTP doses or the increases are insignificant and do not exceed the minimum value for a harsh environment, with two exceptions. The normal dose in the RWCU Heat Exchange Room and the RWCU Pump Room increases for the EPU. However, the EPU total dose (i.e., normal and accident) in these rooms is bounded by the current dose level used to qualify all components potentially affected by this increase. Therefore, these components continue to be qualified for the EPU.

10.3 Mechanical Component Design Qualification

The design and qualification of safety-related mechanical components is part of the DAEC's design control process under the 10 CFR 50, Appendix B, Quality Assurance (QA) Program, as described in Chapter 17.2 of the UFSAR. Appropriate design control is also applied to nonsafety-related and BOP equipment, commensurate with good engineering practices, under the DAEC QA Program. These same design controls were applied to the EPU.

Mechanical systems, structures, and components (SSCs) that could be affected by the changes associated with the EPU, such as changes in system pressures, temperatures, and flowrates, were evaluated. SSCs potentially affected by these changes were either found to be acceptable under the new conditions, will be monitored more frequently as part of an existing program (e.g., In-Service Testing/Inspection, flow-assisted corrosion), or will be modified as part of the implementation of the EPU. All such modifications will be conducted under the DAEC QA Program.

10.4 Required Testing

The following testing will be performed at the time of implementation of EPU:

Surveillance testing will be performed on the instrumentation that requires recalibration for EPU.

Steady-state data will be taken at points from 90% up to the previous RTP, so that system performance parameters can be projected for the EPU RTP before the previous power rating is exceeded.

Power increases beyond the previous RTP will be made in increments of $\leq$5%. Steady-state operating data will be taken and evaluated at each step.

Control system checks 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.

The same performance criteria shall be used as in the original power ascension tests, unless they have been replaced by updated criteria since the initial test program.

The piping vibration levels of the Main Steam Line System piping and the Feedwater System piping within containment will be monitored during initial plant operation at the new EPU operating conditions. These piping systems must be monitored for vibration because the mass flow rates and pressure levels in these piping systems will increase noticeably during EPU operations. The mass flow rates in these systems will increase in proportion to the EPU power level increase. The startup vibration test program will show that these piping systems are vibrating at acceptable levels during initial plant operation at the EPU conditions.

The DAEC containment leakrate testing program is required by 10 CFR 50 Appendix J and is described within UFSAR Section 6.2 and Technical Specification 5.5.12. This test program periodically pressurizes the containment (Type A test), the containment penetrations (Type B test), and the containment isolation valves (except for the MSIVs) and test boundaries (Type C tests) to the calculated peak containment pressure (P_a), and compares the resulting leakage to the allowable leakage limit (L_a). P_a increases to 45.7 psig for the EPU. Therefore, Technical Specification 5.5.12 is being revised to reflect this new calculated peak containment pressure value for the EPU.

10.5 Individual Plant Evaluation

The DAEC developed Level 1 and Level 2 Probabilistic Risk Assessment (PRA) models as part of the Individual Plant Examination (IPE) and Individual Plant Examination – External Events (IPEEE) in response to Generic Letter 88-20 and its supplements. The DAEC has maintained these PRA models to conform to plant configuration and operating procedure changes subsequent to the original development, i.e., it is a "living PRA." The PRA is utilized for assessing hardware changes to the facility and operating and maintenance practices for impact on overall plant risk. It has also been used for screening of components and development of reliability goals in accordance with the Maintenance Rule (i.e., 10 CFR 50.65). Because of its on-going use as a decision-making tool, the DAEC PRA has been through a peer review as part of the BWR Owners' Group PRA certification program.

Changes due to EPU implementation were evaluated for impact on the PRA models in the following key areas:

- Initiating Event Frequency
- Component Reliability
- System Success Criteria
- Operator Response

The evaluation concluded that EPU implementation does not change initiating event frequencies or component reliability assumed in the current PRA. Further, while some plant parameters are affected by EPU implementation, these changes were within the existing margin of the current system success criteria in the PRA and revisions were not required in order to satisfy the overall safety success criteria. Finally, operator response time was slightly decreased for some events. The Electric Power Research Institute (EPRI) guidelines, "PSA Applications Guide," (Reference 9), were used to determine the "significance" of EPU implementation on the DAEC PRA. Per the EPRI Guide, implementation of the DAEC EPU is not a risk significant change to the facility.

10.6 Operator Training And Human Factors

Additional training required to operate the plant in the EPU condition is expected to be minimal. The changes to the plant have been identified and the operator training program is being evaluated to determine the specific changes required for operator training. This evaluation includes the effect on the plant simulator.

For the EPU conditions, operator responses to transient, accident and special events are expected to be only slightly 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 operation 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 as appropriate. The simulator training as a minimum will include 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. There are various plant programs (EQ, FAC) that deal with age-related components. 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 Information Notices

NRC Circulars

INPO Significant Operating Experience Reports

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 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, and (3) the plant emergency operating procedures (EOPs). These items have been reviewed for possible effect by the EPU, and were found to be either acceptable for the EPU, or were revised to reflect the EPU conditions.

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

The EPU does not involve any significant impacts to the environment. There are no new significant environmental hazards in addition to those previously evaluated. The environmental impacts and adverse effects identified by the Staff for DAEC operation at 1658 MWt in the Summary and Conclusions Section of the Final Environmental Statement (Reference 10) continue to bound plant operation at the EPU conditions. The proposed changes do not, individually or cumulatively, affect the human environment. There is no significant change in the types or amounts of plant effluents. The EPU does not involve significant increases in individual or cumulative occupational radiation exposure.

The effect of the EPU on the environment does not prevent continued compliance with any DAEC environmental permit. None of the license conditions for environmental protection will be changed for the EPU. No effluent limits will be exceeded, and the present large margins to these limits will not be significantly changed. The EPU does not involve an increase in the discharge of hazardous substances, contaminants, or pollutants and does not involve the use of any new hazardous substances, contaminants, or pollutants.

The EPU does not involve any changes to air quality or water quality. The EPU does not result in any changes to land usage and has an insignificant effect on groundwater and surface water usage. The amount of water withdrawn and consumed from the Cedar River remains within that previously evaluated. The slight increase in discharge canal temperature has an insignificant effect on river temperature and will not result in any changes to aquatic biota. The EPU will not involve new or different discharges of contaminants and does not involve changes to any bioaccumulation effects for aquatic organisms. The quality of drinking water is not affected.

The EPU does not involve any changes to wildlife habitat and does not result in any significant changes to aquatic or terrestrial biota. There are no deleterious effects on the diversity of biological systems or the sustainability of species due to the EPU. The EPU does not involve any additional changes to the stability and integrity of ecosystems. The EPU does not affect the previous conclusions on impingement or entrainment. The EPU does not affect DAEC's compliance with Sections 316(a) or 316(b) of the Federal Water Pollution Control Act.

The EPU does not significantly change any doses to the public from radiological effluents, and offsite doses will continue to be well within regulatory limits. In the Safety Evaluation for the DAEC (Reference 11), the Staff concluded that the release of radioactive material in liquid and gaseous effluents from the DAEC will meet the requirements of 10 CFR 50 for keeping such effluent levels to unrestricted areas as low as practicable and will result in doses that are a small percentage of the 10 CFR 20 limits (Reference 11, Section 11.8). The Staff based this conclusion on assumptions for effluent releases that bound releases expected for the EPU. Occupational dose will be maintained well within regulatory limits, and changes in radiation levels will not significantly increase the dose to the DAEC work force. For accident dose, the methodology for certain design basis accidents was updated. This methodology is consistent with previously approved Staff methods, and the resultant dose is well within the applicable regulatory limits. The EPU does not involve significant increases in the probability or consequences of previously evaluated environmental accidents.

This detailed environmental evaluation has been submitted under separate cover (Reference 12) and demonstrates that in most cases the EPU does not involve any environmental impacts that are different from those previously evaluated for the current power level. Where environmental impacts which differ from those previously evaluated have been identified, these impacts have been shown to be well within regulatory environmental acceptance criteria.

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.

The significant safety analyses and evaluations have been performed, and their results justify uprating the licensed thermal power at the Duane Arnold Energy Center (DAEC) by 15.3% to 1912 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 and computer codes based on several decades of BWR safety technology, plant performance feedback, operating experience, and improved fuel and core designs have resulted in a significant increase in the design and operating 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 expand the operating envelope on the power/flow map through implementation of Maximum Extended Load Line Limit Analysis (MELLLA). 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 already has or can readily be modified to have adequate control over inlet pressure conditions at the turbine, to account for the larger 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

DAEC is currently licensed for a 100% RTP level of 1658 MWt (104.1% of the Original Rated Thermal Power (ORTP) of 1593 MWt). The current accident analyses were generally performed at 102% of the current RTP level, in accordance with Regulatory Guide (RG) 1.49. Some analyses were performed at 100% EPU RTP, because the 2% power factor of RG 1.49 is already accounted for in the analysis methodology or the additional margin is not required (e.g., Anticipated Transient Without Scram). The EPU RTP level (1912 MWt) included in this evaluation is a 15.3% thermal power increase from the current RTP. Similar to current analyses, the EPU safety analyses are based on a power level of ≥1.02 times the EPU RTP level, except where it is accounted for in the analysis methodology or is not required.

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 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. Plus, 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 Core Spray (CS)) pumps, the high pressure ECCS (High Pressure Coolant Injection (HPCI)) pump, the Reactor Core Isolation Cooling (RCIC) pump, 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 15.3% increase in decay heat, and thus, the core cooling time to reach cold shutdown requires more time. This is not a safety concern, and the existing cooling capacity can bring the plant to cold shutdown within an acceptable time span (Section 3.9.1).

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, while accommodating a single active equipment failure in addition to the postulated LOCA. This break range bounds the full spectrum of large and small, high and low energy line breaks. 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 50.67 and General Design Criterion (GDC) 19.

11.4.2.6 Challenges to Fuel

The ECCS are described in Section 6.3 of the plant Updated Final Safety Analysis Report (UFSAR). 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 power 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 UFSAR 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 dose exposure pathways do not change for the EPU and Alternative Source Term (AST)

implementation, except for consideration of a short positive pressure period in the secondary containment that results in leakage to the environment until a negative pressure is re-established. The atmospheric dispersion factors have been revised for AST implementation. The quantity of fission products 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 reanalyzed, and the instrument line break accident (ILBA) was reviewed.

For the MSLBA, the quantity of activity in the primary coolant and in the offgas used in the evaluation of this postulated event is based on Technical Specification limits, which are revised for the EPU/AST. However, the plant original MSLBA analysis was not based on the Technical Specifications limits. Thus, the EPU/AST updated MSLBA analysis is impacted by the implementation of the AST, which includes the new Technical Specifications limit activity and the revised atmospheric dispersion factors.

For the ILBA, the only transport mechanism influenced by the EPU is the quantity of coolant mass discharged to the environment. The ILBA is not a limiting event. For the ILBA, increased mass loss would only occur if the operating pressure were increased. However, the requested EPU does not need or include an increase in operating pressure, and thus, the consequences of an ILBA do not change.

For the remaining DBAs (i.e., CRDA, LOCA, and FHA), the only parameter of importance is the activity released from the fuel. Because the mechanics of fission product release from the fuel are 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 Draft Regulatory Guide 1081 has been applied. For this accident, the BWR AST release fractions specified in Draft Regulatory Guide 1081 are assumed. These release fractions are not influenced by the EPU. The radiological consequences from the updated EPU/AST LOCA DBA, as shown in Section 9, remain 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 50.67 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 was calculated to assure that the SLMCPR is not infringed upon, if any transient is initiated from the EPU RTP level. Plus, 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 maintained 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 specified by Regulatory Guide 1.183 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 evaluations conforming to Regulatory Guide 1.183.

11.4.2.11 Non-LOCA Radiological Release Accidents

All of the limiting non-LOCA events discussed in Regulatory Guide 1.70 Chapter 15 have been updated for impact from the EPU/AST. The dose consequences for all of the non-LOCA radiological release accident events are shown in Section 9 to remain below regulatory limits.

11.4.2.12 Equipment Qualification

Plant equipment and instrumentation has 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. The existing DAEC Environmental Qualification program will ensure qualification of the equipment whose current qualification is not bounded at 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, non-safety service water, 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 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 uncertainties, using the GE setpoint methodology, in the development of the Technical Specification Allowable Values (AVs). The AVs provide margin to assure that the Analytical Limits used in the safety analyses

are not exceeded, accounting for these instrument and calibration uncertainties. The actual in-plant settings are chosen to provide margin to both the AV and operational limit for spurious trip avoidance and account for instrument drift between calibrations.

Similarly, the Technical Specifications address equipment Operability (available to perform it safety analysis function) and place appropriate restrictions on the time the equipment may remain out-of-service (not capable of performing its required function) to minimize overall plant risk, while allowing a reasonable time for repair.

Because the safety analyses for the EPU show that the results are within regulatory limits, public health and safety is assured. Thus, the proposed Technical Specification changes to implement the EPU continue to provide a comparable level of protection as those previously issued by the NRC.

11.4.3 Assessment Against 10 CFR 50.92 Criteria

10 CFR 50.91(a) states "At the time a licensee requests an amendment, it must provide to the Commission ... its analysis about the issue of no significant hazards consideration using the standards in § 50.92." The following provides this analysis for the DAEC EPU. The conclusions are based on the evaluations provided in this report, and are summarized as appropriate to the following safety considerations in accordance with 10 CFR 50.92.

In addition, a typographical error, inadvertently introduced during the conversion to the Improved Technical Specifications (Amendment #223), is being corrected in this application.

1) **Will the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

The increase in power level discussed herein will not significantly increase the probability or consequences of an accident previously evaluated.

The probability (frequency of occurrence) of Design Basis Accidents is not affected by the increased power level, because the plant still complies with the regulatory and design basis criteria established for plant equipment (e.g., ASME code, IEEE standards, ANSI standards). Instrument setpoints (equipment settings that initiate automatic plant trips) and equipment operating margins are established such that there is no expected increase in transient event frequency due to the EPU. Currently established inspection/monitoring and testing programs

are being revised, where needed, as a result of changes in plant conditions due to EPU implementation (e.g., erosion/corrosion monitoring in BOP piping).

The consequences of previously analyzed accidents, which could occur from EPU implementation, are not significantly increased. The spectrum of hypothetical accidents and transients has been investigated, and are shown to meet the appropriate regulatory criteria. The EPU accident evaluation results do not exceed any of their NRC-approved acceptance limits.

Challenges to the major fission product barriers: fuel cladding, reactor coolant pressure boundary, and containment have all been evaluated.

Challenges to the fuel cladding from abnormal transients and accidents have been analyzed and appropriate limits established (e.g., Maximum Average Planar Linear Heat Generation Rate [MAPLHGR] and SLMCPR) to ensure that fuel cladding integrity will be maintained under EPU conditions.

Challenges to the Reactor Coolant Pressure Boundary were evaluated under EPU conditions (pressure, temperature, flow and radiation) and found to meet their acceptance criteria for allowable stresses and overpressure margin.

Challenges to the containment under postulated EPU accident conditions have been evaluated, and the containment and its associated cooling systems continue to demonstrate margin to their design basis pressure and temperature limits.

Radiological release events (accidents) have been evaluated, and shown to meet the regulatory limits of 10 CFR 50.67 and 10 CFR Part 50, Appendix A, General Design Criterion 19.

Based on the above, it is concluded that implementation of the EPU will not significantly increase either the probability or consequences of any previously evaluated accident.

Correction of a typographical error in the Technical Specifications cannot increase the probability or consequences of any previously analyzed accident.

2) Will the change create the possibility of a new or different kind of accident from any accident previously evaluated?

As summarized below, EPU implementation will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The full spectrum of accident considerations, defined in the UFSAR, has been evaluated, and no new or different kind of accident has been identified. GE has designed BWRs of higher power levels than the EPU RTP level and no new power-dependent accidents have been identified. The EPU uses already developed technology (i.e., codes, standards, and methods), and applies it in accordance with existing regulatory criteria. Plant equipment is not operated in a significantly different way than currently and all required safety margins are maintained. Consequently, no new failure modes are introduced by EPU implementation.

Correction of a typographical error in the Technical Specifications cannot introduce the possibility of any new or different accidents than those previously analyzed.

3) Will the change involve a significant reduction in a margin of safety?

As summarized below, this change will not involve a significant reduction in a margin of safety.

The challenges to affected structures, systems and components have been evaluated and will remain within their acceptance criteria for all design basis events. No NRC acceptance criterion is exceeded. The margins of safety currently designed into the plant are not affected by the EPU. EPU is achieved by utilizing the extra design and operational margin built into the plant over and above that required for public health and safety. Because the plant response to transients and hypothetical accidents does not result in exceeding any NRC acceptance limits, in particular, offsite radiological dose, EPU implementation does not involve a significant reduction in a margin of safety.

Correction of a typographical error in the Technical Specifications cannot reduce the margin of safety.

Conclusions:

An EPU to 120% of original RTP has been investigated. The method for achieving higher power is to increase some plant operating parameters. The challenges to plant systems, structures, and components have been evaluated and demonstrate how 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 presently existing regulatory limits or acceptance criteria applicable to the plant which might cause a reduction in a margin of safety.

Having arrived at negative declarations with regards to the criteria of 10 CFR 50.92, this assessment concludes that the EPU of the amount described herein does not involve a Significant Hazards Consideration.

Table 11-1

Technical Specifications and Bases Affected by EPU

Technical Specification Item	Description of Change
Section 1.1, Definitions	Revised the definition of RATED THERMAL POWER to be the EPU maximum licensed power level of 1912 MWt.
Safety Limit (SL) 2.1.1.1	Revised the SL for fuel cladding integrity at low core flow and reactor pressure from the current 25% RTP to 21.7% RTP (25% x 1658/1912).
Limiting Condition for Operation (LCO) 3.2.1: - Applicability - Req'd Action B.1 - Surveillance Requirement (SR) 3.2.1.1 LCO 3.2.2: - LCO Applicability - Req'd Action B.1 SR 3.2.2.1	Revised the percentage of RTP value related to thermal limits monitoring from 25% RTP to 21.7% RTP.
LCO 3.3.1.1: - SR 3.3.1.1.2	Revised the percentage of RTP value related to deferral of the SR until 12 hours after reaching 25% RTP during plant startup, from 25% RTP value to 21.7%. The RTP value being changed is contained in the SR and the associated NOTE.
LCO 3.3.1.1: - Required Action E.1 - SR 3.3.1.1.16 - Table 3.3.1.1-1 Functions 8 and 9	Revised the percentage of RTP value corresponding to the power level where the direct Reactor Protection System (RPS) trips, i.e., scram, on Turbine Stop Valve (TSV) or Turbine Control Valve (TCV) fast closure are automatically bypassed from 30% RTP to 26% RTP.
LCO 3.3.4.1: - Applicability - Req'd Action C.2 - SR 3.3.4.1.4	Revised the percentage of RTP value corresponding to the power level where the End-of-Cycle Recirculation Pump Trip (EOC-RPT) on TSV or TCV fast closure are automatically bypassed from 30% RTP to 26% RTP.
LCO 3.3.1.1: - Table 3.3.1.1-1 Function 2b	Replaced the current AVs for the Two-loop Operation (TLO) Average Power Range Monitor (APRM) Flow-Biased, High RPS trip with the equation for the AV to implement the MELLLA. A new footnote (c) is being added to define the term "W" used in the AV equation.
LCO 3.3.1.1: - Table 3.3.1.1-1:	Replaced the current AVs for the Single-loop Operation (SLO) APRM Flow Biased – High RPS trip with the equation for the AV to

Technical Specification Item	Description of Change
Footnote (b)	implement the MELLLA. The new footnote (c) identified above is used to define the term "W" used in the AV equation.
LCO 3.4.1: - SR 3.4.1.1 a & b	Revised the percentage of RTP value corresponding to the power level where a recirculation pump speed mismatch surveillance is performed from 80% RTP to 69.4% RTP.
LCO 3.4.2: - SR 3.4.2.1	Revised the percentage of RTP value contained in the NOTE corresponding to the power level where the evaluation of jet pump performance can be deferred for up to 24 hours from 25% RTP to 21.7% RTP.
LCO 3.6.3.1: - SR 3.6.3.1.1	Revised the volume requirement for nitrogen storage for the Containment Atmospheric Dilution (CAD) System from 50,000 scf to 67,000 scf.
LCO 3.7.7: - Applicability - Required Action B.1	Revised the percentage of RTP value where the Main Turbine Bypass Valve System is required to be OPERABLE from 25% RTP to 21.7% RTP.
Section 5.5.12, Primary Containment Leakage Testing Program	Revised the peak calculated containment pressure (P_a) from 43 psig to 45.7 psig.

12 REFERENCES

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