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

Dresden Nuclear Power Station, Units 2 and 3
Facility Operating License Nos. DPR-19 and DPR-25
NRC Docket Nos. 50-237 and 50-249

Quad Cities Nuclear Power Station, Units 1 and 2
Facility Operating License Nos. DPR-29 and DPR-30
NRC Docket Nos. 50-254 and 50-265

Subject: Safety Analysis Reports Supporting the License Amendment Request to Permit Uprated Power Operation

Reference: Letter from R. M. Krich (Commonwealth Edison Company) to U. S. NRC, "Request for License Amendment for Power Uprate Operation," dated December 27, 2000

In the referenced letter, Commonwealth Edison Company, now Exelon Generation Company (EGC), LLC, submitted a request for changes to the operating licenses and Technical Specifications for Dresden Nuclear Power Station (DNPS), Units 2 and 3, and Quad Cities Nuclear Power Station (QCNPS), Units 1 and 2, to allow operation with an extended power uprate. The referenced letter contained safety analysis reports supplied by General Electric (GE) Company supporting the proposed changes and requested that these safety analysis reports be withheld from public disclosure in accordance with 10 CFR 2.790(a)(4), "Public inspections, exemptions, requests for withholding." Attachments A and B to this letter provide non-proprietary versions of these safety analysis reports for DNPS and QCNPS, respectively.

Attachments C and D to this letter contain revised proprietary versions of the GE safety analysis reports for DNPS and QCNPS, respectively. These attachments contain proprietary information and we request that they be withheld from public disclosure in accordance with 10 CFR 2.790(a) (4). Attachment E provides the affidavit supporting the request for withholding Attachments C and D from public disclosure as required by 10 CFR 2.790(b)(1). These reports have been revised to alter the proprietary designations as described in Attachment E and also to revise certain technical information contained in the originally submitted reports. The revised technical information is indicated with revision bars. The revisions to the technical information have either previously been noted in correspondence with the NRC regarding this license amendment request or do not significantly affect the conclusions of the original safety analysis reports.

A001

Should you have any questions concerning this letter, please contact Mr. A. R. Haeger at (630) 657-2807.

Respectfully,

K. A. Ainger

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

Attachments:

Attachment A: GE Report NEDO-32962, Revision 1, "Safety Analysis Report for Dresden 2 & 3 Extended Power Uprate," August 2001 (Non-Proprietary)
Attachment B: GE Report NEDO-32961, Revision 1, "Safety Analysis Report for Quad Cities 1 & 2 Extended Power Uprate," August 2001 (Non-Proprietary)
Attachment C: GE Report NEDC-32962P, Revision 2, "Safety Analysis Report for Dresden 2 & 3 Extended Power Uprate," August 2001 (Proprietary)
Attachment D: GE Report NEDC-32961P, Revision 2, "Safety Analysis Report for Quad Cities 1 & 2 Extended Power Uprate," August 2001 (Proprietary)
Attachment E: GE Affidavit for Withholding NEDC-32961P and NEDC-32962P from Public Disclosure

cc: Regional Administrator - NRC Region III
NRC Senior Resident Inspector - Dresden Nuclear Power Station
NRC Senior Resident Inspector - Quad Cities Nuclear Power Station
Office of Nuclear Facility Safety - Illinois Department of Nuclear Safety

Attachment A
Safety Analysis Reports Supporting the License Amendment Request to Permit Uprated Power Operation
Dresden Nuclear Power Station, Units 2 and 3,
Quad Cities Nuclear Power Station, Units 1 and 2

GE Report NEDO-32962, Revision 1, "Safety Analysis Report for Dresden 2 & 3 Extended Power Uprate," August 2001 (Non-Proprietary)

GE Nuclear Energy
175 Curtner Ave., San Jose, CA 95125

NEDO-32962, Revision 1
DRF A22-00103-13
Class I
August 2001

SAFETY ANALYSIS REPORT
FOR
DRESDEN 2 & 3
EXTENDED POWER UPRATE

Prepared by: R. L. Hayes

Approved by:
L. W. King, Project Manager
General Electric Company

Approved by:
J. M. Nosko, Manager
Exelon Generation Company

IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT

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EXECUTIVE SUMMARY

This report summarizes the results of all significant safety evaluations performed that justify extending the licensed thermal power at Dresden Nuclear Power Station (DNPS) Units 2 and 3 to 2957, MWt. The requested license power level is approximately 117% of the current licensed rating of 2527 MWt.

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

A higher steam flow is achieved by increasing the reactor power along slightly revised rod and core flow control lines. A limited number of operating parameters are changed. Some setpoints are adjusted and instruments are recalibrated. Plant procedures are revised and tests similar to some of the original startup tests are performed. Modifications to some power generation equipment may be implemented over time, as needed.

Detailed evaluations of the reactor, engineered safety features, power conversion, emergency power, support systems, environmental issues, design basis accidents, and previous licensing evaluations were performed. This report demonstrates that DNPS can safely operate at the requested license power level of 2957 MWt. However, power generation modifications must be implemented in order to obtain the electrical power output associated with 100% of the EPU power level. Until these modifications are completed, the balance of plant may limit the electrical power output, which (in-turn) limits the operating thermal power level to less than the licensed power level.

The predominant plant licensing challenges have been reviewed, and it is concluded that this uprate can be accommodated (1) without a significant increase in the probability or consequences of an accident previously evaluated, (2) without creating the possibility of a new or different kind of accident from any accident previously evaluated, and (3) without exceeding any existing regulatory limits applicable to the plant which might cause a significant reduction in a margin of safety. Therefore, the requested EPU does not involve a significant hazards consideration.

1 OVERVIEW

1.1 Introduction

Uprating the power level of nuclear power plants can be done safely within certain plant-specific limits. Most GE BWR plants have the capability and margins for an uprating of 5 to 20% without major nuclear steam supply system (NSSS) hardware modifications. Many light water reactors have already been uprated worldwide. Over a thousand MWe have already been added by uprate in the United States. Several BWR plants are among those that have already been uprated. This evaluation justifies an EPU to 2957 MWt, corresponding to 117% of the current rated thermal power, for both DNPS Units 2 and 3. The original licensed thermal power is 2527 MWt.

The ARTS program is designed to increase plant operating efficiency by updating the thermal limits requirements. The APRM trip setdown (gain and setpoint) requirement is replaced by the ARTS power-dependent and flow-dependent thermal limits to reduce the need for manual setpoint adjustments and to allow more direct thermal limits administration. This change updates thermal limits administration, increases reliability, and provides better protection.

The ARTS-based thermal limits are specified for fuel protection during Anticipated Operational Occurrences (AOOs). The plant-specific portions of these generic ARTS limits were developed based on a representative core configuration.

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 at least 5% above the original rating. In addition, continuing improvements in the analytical techniques (e.g., computer codes) based on several decades of BWR safety technology, plant performance feedback, and improved fuel and core designs have resulted in a significant increase in the difference between the calculated safety analyses results and the licensing limits. The plant-specific uprate parameters are listed in Table 1-2.

Each unit is currently licensed at 2527 MWt, and most of the current safety analyses are based on this value. However, the ECCS-LOCA and Containment safety analyses are based on a power level of 1.02 times the licensed power level. The uprate power level included in this evaluation is a 17% (2957 MWt) thermal EPU of the currently licensed value. The EPU safety analyses are based on a power level of at least 1.02 times the EPU power level (1.02 x 2957 = 3016 MWt), except that some analyses are performed at 100% uprated power, because the Regulatory Guide 1.49 two percent power factor is already accounted for in the analysis methods.

The extended power uprate analysis basis assures that the power-dependent safety margin prescribed by the Code of Federal Regulations (CFR) will be maintained by meeting the appropriate regulatory criteria. NRC-accepted computer codes and calculational techniques are used to make the calculations that demonstrate meeting the stipulated criteria.

The major EPU analyses for Dresden and Quad Cities were performed using bounding parameters. This allows one evaluation to be performed that envelops all four units. The bounding value of each parameter was obtained by comparing the parameter across the four units and selecting the most limiting value. Therefore, the evaluation results in this report are conservative, and consequently, the actual operating values for any given unit may differ from the values shown herein.

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 without an increase in reactor operating pressure, (2) a corresponding increase in the feedwater system flow, (3) no increase in maximum core flow, and (4) reactor operation primarily along accepted rod/flow control lines. Plant-unique evaluations were based on a review of plant design and operating data to confirm excess design capabilities. The results of these evaluations are presented in the subsequent sections of this report.

1.3 EPU Plant Operating Conditions

The thermal hydraulic performance of a BWR reactor core is characterized by the total 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. They are determined by performing heat balance calculations for the reactor system at EPU conditions.

The EPU heat balance was determined such that the core thermal power is 117% of the current licensed core thermal power and the steam flow from the vessel was increased to approximately 119% of 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 power and 100% rated core flow.

Table 1-2 shows a summary of the reactor thermal-hydraulic parameters for the current rated condition and EPU conditions.

The UFSAR, core fuel reload evaluations, and/or the Technical Specifications currently include allowances for plant operation with the performance improvement features and the equipment out-of-service listed in Table 1-2. When limiting, the input parameters related to the performance improvement features or the equipment out-of-service have been included in the safety analyses for EPU. The use of these performance improvement features and allowing for equipment out-of-service is continued during EPU power operation. Where appropriate, the evaluations performed for uprate account for a 24 month fuel cycle length.

1.4 ARTS Power and Flow Dependent Limits

The ARTS improvements provide changes to the APRM system. An overview of the improvements is discussed below along with the identification of the evaluations necessary to support these improvements. The Technical Specifications (TS) change(s) associated with the ARTS improvements are provided in Table 11-1

The plant TS require that the flow-referenced APRM trips be lowered (setdown) when the core Maximum Total Peaking Factor (MTPF) exceeds the design Total Peaking Factor (TPF). The basis for this "APRM trip setdown" requirement originated under the previous Hench-Levy Minimum Critical Heat Flux Ratio (MCHFR) thermal limit criterion.

The change to the General Electric Thermal Analysis Basis critical power correlation, with its de-emphasis of local thermal hydraulic conditions, and the move to secondary reliance on flux scram for licensing basis anticipated operational occurrence (AOO) evaluations (for events terminated by anticipatory or direct scram) provides more effective and operationally acceptable alternatives to the setdown requirement. The ARTS program utilizes results of the AOO analyses to define initial condition operating thermal limits which conservatively ensure that all licensing criteria are satisfied without setdown of the flow-referenced APRM scram and rod block trips.

The objective of the APRM improvements is to justify removal of the APRM trip setdown requirement (APRM Gain and Setpoint TS). Two licensing areas, which can be affected by the elimination of the APRM Gain and Setpoint TS, are fuel thermal-mechanical integrity and loss-of-coolant accident (LOCA) analysis.

The (applicable) safety analyses used to evaluate the Operating Limit MCPR (OLMCPR), such that the SLMCPR is not violated, and to ensure that the fuel thermal-mechanical design bases are satisfied, are documented in Section 9.2. These analyses also establish the fuel type specific power- and flow-dependent limits for DNPS. The effect on the ECCS-LOCA response due to both the expansion of the power/flow map and the implementation of the ARTS improvement is discussed in Section 4.3.

The following changes result from the implementation of ARTS power and flow dependent limits:

1. Delete the requirement for setdown of the APRM scram and rod blocks.

2. Add new power-dependent MCPR adjustment factors, MCPR(P).

3. Replace the flow-dependent MCPR limits with the new flow-dependent MCPR adjustment factors, MCPR(F).

4. Add new power-dependent LHGR adjustment factors, LHGRFAC(P).

5. Add new flow-dependent LHGR adjustment factors, LHGRFAC(F).

6. Delete or modify affected TS and Bases.

1.5 Summary And Conclusions

The predominant plant licensing challenges have been reviewed to demonstrate how this uprate can be accommodated (1) without a significant increase in the probability or consequences of an accident previously evaluated, (2) without creating the possibility of a new or different kind of accident from any accident previously evaluated, and (3) without exceeding any existing regulatory limits or design allowable limits applicable to the plant which might cause a reduction in a margin of safety. The EPU described herein involves no significant hazard consideration.

Table 1-1

Glossary of Terms

Term	Definition
AC	Alternating current
ADS	Automatic Depressurization System
AL	Analytical Limit
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOO	Anticipated operating occurrences (moderate frequency transient events)
AP	Annulus pressurization
APCVS	Augmented Primary Containment Venting System
APRM	Average Power Range Monitor
ARO	All rods out
ARTS	APRM/RBM/Technical Specifications
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient Without Scram
AV	Allowable Value
BHP	Brake horse power
BOP	Balance-of-plant
BWR	Boiling Water Reactor
BWROG	BWR Owners Group
CAM	Containment Atmosphere Monitoring
CCSW	Containment Cooling Service Water
CCT	Critical Clearing Time
CD	Condensate demineralizers
CFR	Code of Federal Regulations
CGCS	Combustible Gas Control System
CO	Condensation oscillation
COLR	Core Operating Limits Report
CPD	Condensate polishing demineralizer
CPR	Critical power ratio
CRD	Control Rod Drive
CRDA	Control Rod Drop Accident
CRGT	Control Rod Guide Tube
CSC	Containment Spray Cooling
CST	Condensate Storage Tank
CS	Core Spray

DAR	Design Assessment Report
DBA	Design basis accident
DC	Direct current
DG	Diesel generator
DGCW	Diesel Generator Cooling Water
DNPS	Dresden Nuclear Power Station
DL	Discharge line
ECCS	Emergency Core Cooling System
EDG	Emergency diesel generators
EECW	Emergency Equipment Cooling Water
EESW	Emergency Equipment Service Water
EFPY	Effective full power years
EGC	Economic generation control
EHL	Emergency Heat Load
EHC	Electro-hydraulic control
ELLL	Extended Load Line Limit
ELTR	Extended power uprate licensing topical report
EOC	End of cycle
EOOS	Equipment out-of-service
EOP	Emergency Operating Procedure
EPP	Environmental Protection Plan
EPU	Extended power uprate
EQ	Environmental qualification
ER-OL	Environmental Report-Operating License stage
ESW	Emergency Service Water
FAC	Flow Accelerated Corrosion
FCS	Feedwater Control System
FCV	Flow Control Valve
FES	Final Environmental Statement
FFRO	Fast Flow Runout
FFWTR	Final Feedwater Temperature Reduction
FHA	Fuel Handling Accident
FWCF	Feedwater controller failure
FWHOOS	Feedwater heater(s) out-of-service
FPCC	Fuel Pool Cooling and Cleanup
FSAR	Final Safety Analysis Report
GE	General Electric Company
HD	Heater Drains

HX	Heat exchanger
HCU	Hydraulic Control Unit
HEI	Heat Exchange Institute
HELB	High Energy Line Break
HPCI	High Pressure Coolant Injection
HPSP	High power setpoint
HVAC	Heating Ventilating and Air Conditioning
IC	Isolation Condenser
ICA	Interim Corrective Actions
ICF	Increased Core Flow
IEB	Inspection and Enforcement Bulletin (original NRC title)
IEC	Information and Enforcement Circular (original NRC title)
IEEE	Institute of Electrical and Electronics Engineers
IEN	Inspection and Enforcement Notice (original NRC title)
IGSCC	Intergranular stress corrosion cracking
ILBA	Instrument Line Break Accident
IRM	Intermediate Range Monitor
JR	Jet reaction
LBB	Local Breaker Backup
LCO	Limiting Conditions for Operation
LCS	Leakage Control System
LERF	Large Early Release Frequency
LFA	Lead Fuel Assemblies
LHGR	Linear Heat Generation Rate
LHGRFAC(F)	Flow-dependent LHGR adjustment factor
LHGRFAC(P)	Power-dependent LHGR adjustment factor
LOCA	Loss-Of-Coolant Accident
LOFW	Loss of feedwater
LOOP	Loss of offsite power
LPCI	Low Pressure Coolant Injection
LPRM	Local Power Range Monitor
LPSP	Low Power Setpoint
LRNBP	Load Rejection with no Bypass
LTR	Licensing Topical Report
LUA	Lead use assembly
MAPLHGR	Maximum Average Planar Linear Heat Generation Rate
MBTU	Millions of BTUs
MCC	Motor Control Circuit/Center

MCHFR	Minimum Critical Heat Flux Ratio
MCPR	Minimum Critical Power Ratio
MCPR(F)	Flow-dependent MCPR adjustment factor
MCPR(P)	Power-dependent MCPR adjustment factor
MELB	Moderate Energy Line Break
MELLLA	Maximum Extended Load Line Limit Analysis
MEOD	Maximum Extended Operating Domain
MeV	Million Electron Volts
MG	Motor generator
MHC	Mechanical-Hydraulic Control
Mlb	Millions of pounds
MLHGR	Maximum Linear Heat Generation Rate
MOV	Motor operated valve
MSIV	Main Steam Isolation Valve
MS	Main steam
MSLB	Main steam line break
MSLBA	Main Steam line Break Accident
MSR	Moisture Separator Reheater
MTPF	Maximum Total Peaking Factor
MWt/MWth	Megawatt-thermal
MSL	Main steam line
MVA	Million Volt Amps
MWe	Megawatt-electric
NCAD	Nitrogen Containment Atmosphere Dilution
NCCW	Nuclear Closed Cooling Water
NPSH	Net positive suction head
NRC	Nuclear Regulatory Commission
NSSS	Nuclear steam supply system
NTSP	Nominal Trip Setpoint
NUREG	Nuclear Regulations
OFS	Orificed Fuel Support
OLMCPR	Operating Limit Minimum Critical Power Ratio
OLTP	Original Licensed Thermal Power
OOS	Out-of-service
PCS	Pressure Control System
PCT	Peak cladding temperature
PF	Power Factor
PRA	Probabilistic Risk Assessment

PSA	Probabilistic Safety Assessment
psi	Pounds per square inch
psia	Pounds per square inch - absolute
psid	Pounds per square inch - differential
psig	Pounds per square inch - gauge
PULD	Plant-Unique Load Definition
PWR	Pipe Whip Restraint
RBCCW	Reactor Building Closed Cooling Water
RBM	Rod Block Monitor
RCPB	Reactor Coolant Pressure Boundary
REM	Roentgen Equivalent Man (radiation dose measurement)
RFP	Reactor feed pump
RICSIL	Rapid Information Communication Service Information Letter
RIPD	Reactor internal pressure difference
RLB	Recirculation Line Break
RPCS	Rod Pattern Control System
RPS	Reactor Protection System
RPT	Recirculation Pump Trip
RPV	Reactor Pressure Vessel
RR	Reactor recirculation
RSLB	Recirculation system line break
RTP	Rated Thermal Power
RT_{NDT}	Reference temperature of nil-ductility transition
RV	Relief valve
RWCU	Reactor Water Cleanup
RWE	Rod Withdrawal Error
RWL	Rod Withdrawal Limiter
RWM	Rod Worth Minimizer
SAR	Safety Analysis Report
SBO	Station blackout
SCM	Steam condensing mode
SDC	Shutdown Cooling
SE	Safety Evaluation
SER	Safety Evaluation Report
SGTS	Standby Gas Treatment System
SIL	Services Information Letter
SJAE	Steam Jet Air Ejectors
SLCS	Standby Liquid Control System

SLMCPR	Safety Limit Minimum Critical Power Ratio
SLO	Single-loop operation
SORV	Stuck open relief valve
SPCM	Suppression pool cooling mode
SRM	Source Range Monitor
SRP	Standard Review Plan
SRV	Safety relief valve
SRVDL	Safety relief valve discharge line
SSV	Spring Safety Valve
SW	Service water
TAF	Top of active fuel
TB	Turbine bypass
TBCCW	Turbine Building Closed Cooling Water System
TCV	Turbine control valve
TFSP	Turbine first stage pressure
TG	Turbine generator
TGT	Turbine Generator Trip
TIP	Traversing In-Core Probe
TLO	Two (recirculation) loop operation
TPF	Total Peaking Factor
TPM	Thermal Power Monitor
TS	Technical Specification
TSV	Turbine Stop Valve
TTNBP	Turbine Trip – no Bypass
UFSAR	Updated Final Safety Analysis Report
UHS	Ultimate heat sink
USE	Upper shelf energy
VPF	Vane passing frequency
VWO	Valves wide open

Table 1-2

Current And Extended Uprate Plant Operating Conditions

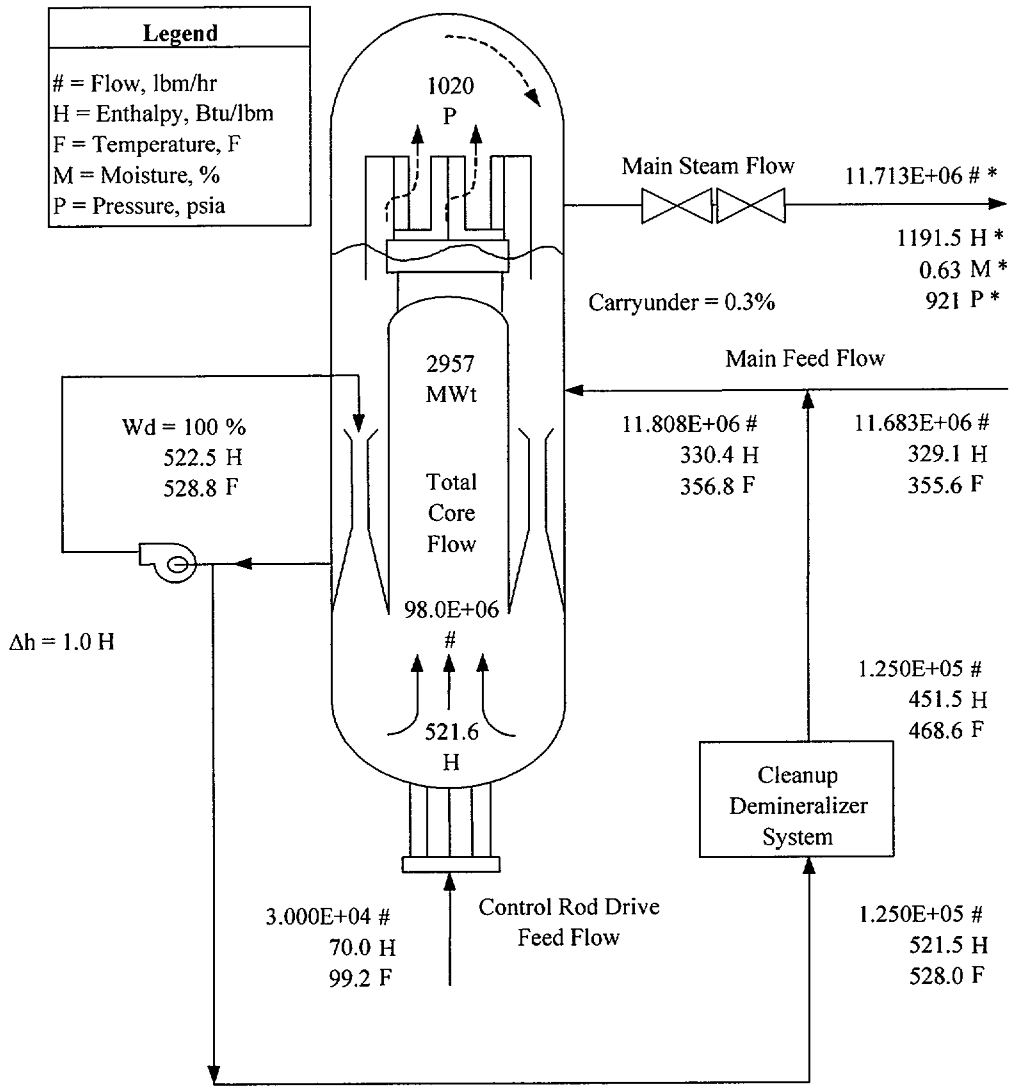
Parameter	**Current Rated Power Value**	**Extended Power Uprate Value**
Thermal Power (MWth)	2527	2957
Vessel Steam Flow (Mlb/hr) *	9.81	11.71
Full Power Core Flow Range		
Mlb/hr	85.3 to 98	93.4 to 105.8 **
% Rated	87 to 100	95.3 to 108 **
Dome Pressure (psig)	1005	No change
Dome Temperature (°F)	547.0	No change
Turbine Inlet Pressure (psig)	935.0	906.0
Full Power Feedwater		
Flow (Mlb/hr) *	9.78	11.68
Temperature Range (°F)	350 to 250	356 to 256
Core Inlet Enthalpy (Btu/lb) *	523.5	521.6

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

** To support projected plant enhancements that would allow for ICF operation, some analyses are based on 108% core flow.

Performance improvement features and/or equipment out-of-service included in EPU evaluations:

(1) Maximum Extended Load Line Limit Analysis (MELLLA)
(2) End-of-Cycle (EOC) Coastdown
(3) Single Loop Operation (SLO)
(4) Final Feedwater Temperature Reduction (FFWTR)
(5) Increased Core Flow (ICF)
(6) ARTS power and flow dependent limits



* Conditions at upstream side of TSV

Core Thermal Power	2957.0
Pump Heating	9.6
Cleanup Losses	-2.6
Other System Losses	-1.0
Turbine Cycle Use	2963.0 MWt

Figure 1-1. **Extended Power Uprate Heat Balance - Nominal**
(@ 100% Power and 100% Core Flow)

2 REACTOR CORE and FUEL PERFORMANCE

2.1 Fuel Design and Operation

EPU increases the average power density proportional to the power increase. However, this average power density is still within the current operating power density range of most other BWRs. EPU has some effects on operating flexibility, reactivity characteristics and energy requirements. The power distribution in the core is changed to achieve increased core power, while limiting the absolute power in any individual fuel bundle to within its allowable value.

At current or uprated conditions, all fuel and core design limits continue to be met by planned deployment of fuel enrichment and burnable poison. This is supplemented by core management control rod pattern and/or core flow adjustments. New fuel designs are not needed for EPU to ensure safety.

The subsequent reload core designs for operation at the EPU power level will ensure acceptable differences between the licensing limits and their corresponding operating values. Cycle-specific analyses will evaluate all fuel types in each reload core.

2.2 Thermal Limits Assessment

Operating thermal limits ensure that regulatory and/or safety limits are not exceeded for a range of postulated events (e.g., transients, LOCA). Cycle-specific core reload evaluations will evaluate the effects on any other fuel types that remain in the core. Both units have identical system geometry, reactor protection system configuration, mitigation functions, and similar thermal hydraulic and transient behavior characteristics. Cycle-specific core configurations, evaluated for each reload, confirm EPU capability, and establish or confirm cycle-specific limits, as is currently the practice.

Thermal limits management with ARTS power and flow dependent limits is described in Section 9.2.

2.3 Reactivity Characteristics

In the representative core evaluation, all minimum shutdown margin requirements apply to cold conditions ($\leq 212^{o}F$), and are maintained without change.

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. Technical Specifications cold shutdown margin requirements are not affected.

The uprated power/flow operating map (Figure 2-1) includes the operating domain changes for EPU power and the plant performance improvement features addressed in Section 1.3. The ARTS power and flow dependent limits analyses (Section 9.2) are in part based on Figure 2-1. The changes to the power/flow operating map are consistent with the previously NRC-approved generic descriptions. The maximum thermal operating power and maximum core flow shown on

Figure 2-1 correspond to the EPU power and the previously analyzed core flow range when rescaled so that EPU power is equal to 100% rated. The power/flow operating map changes incorporated into Figure 2-1 are consistent with the changes shown in Figure 5-1 of ELTR1.

For SLO, the maximum achievable power state point is assumed to be 70.2% uprated power (2076 MWth) at 55.1% flow (54 Mlb/hr).

2.4 Stability

DNPS is currently operating under the requirements of reactor stability Interim Corrective Actions (ICAs) and is in the process of implementing reactor stability Long-Term Solution Option III. However, EPU is scheduled to be implemented prior to arming the Option III solution (it is not considered to be fully implemented until the trip system is armed). Therefore, the effect of EPU is addressed on both the ICAs and on the stability Option III solution.

An evaluation determined the effect of EPU on core stability ICAs for EPU, to assure adequate level of protection against the occurrence of a thermal-hydraulic instability. The current instability exclusion region boundaries are unchanged with respect to absolute power level (MWt).

DNPS is implementing long term stability Option III. The Option III solution monitors Oscillation Power Range Monitor (OPRM) signals to determine when a reactor scram is required to terminate an instability event. The OPRM signal is evaluated by the Option III stability algorithms to determine when the signal is becoming sufficiently periodic and large to warrant a reactor scram to disrupt the oscillation.

ARTS power and flow dependent MCPR limits are used when confirming MCPR Safety Limit protection.

2.5 Reactivity Control

The 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. The CRD system has been generically evaluated. These generic evaluations conclude that the CRD systems for BWR/2-6 are acceptable for EPUs as high as 20% above the original licensed rated power. A confirmatory evaluation was performed for this EPU. The DNPS CRD system is consistent with the generic evaluations, and is acceptable for EPU.

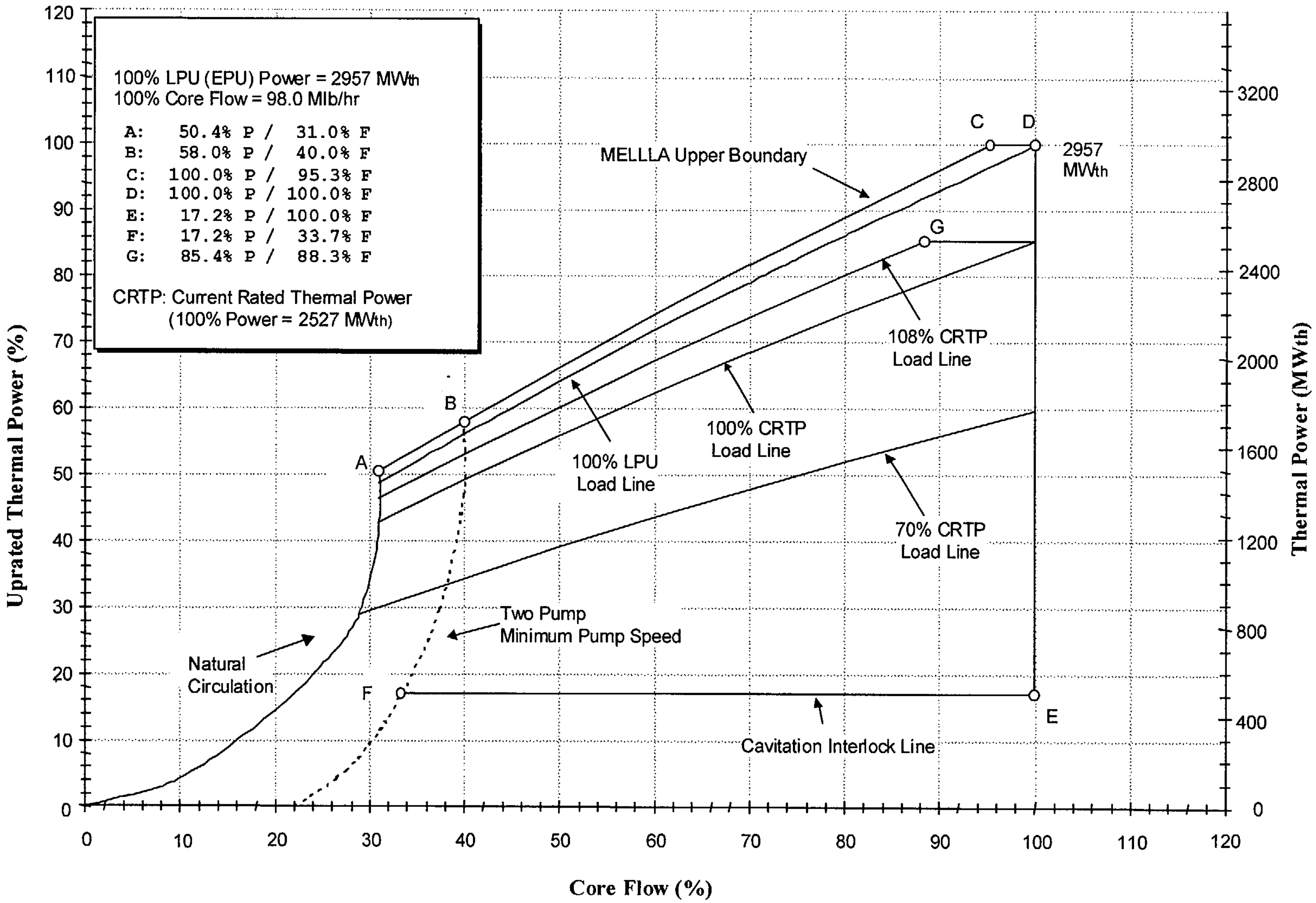


Figure 2-1. **Power/Flow Operating Map for EPU**

3 REACTOR COOLANT SYSTEM and CONNECTED SYSTEMS

3.1 Nuclear System Pressure Relief

The primary purpose of the nuclear system pressure relief is to prevent overpressurization of the nuclear system during abnormal operational transients. Each unit uses eight spring safety valves (SSVs), four relief valves (RVs) and a single safety/relief valve (SRV) together with the reactor scram function to provide this protection. The SSV, RV, and SRV setpoints are not changed with EPU.

The RVs were originally sized to prevent actuation of the SSVs by relieving the vessel pressure following a turbine stop valve closure coincident with failure of the turbine bypass system. However, with EPU, the RVs are not capable of preventing SSV actuation for an infrequent event such as a turbine trip without bypass. The RVs have the capacity to remove the generated steam and prevent SSV actuation for frequent events like the turbine trip with bypass. Therefore, the RV sizing basis changes with EPU.

SRV setpoint tolerance is independent of EPU. EPU evaluations are performed using the existing SRV setpoint tolerance analytical limits as a basis.

3.2 Reactor Overpressure Protection Analysis

The design pressure of the reactor vessel and reactor coolant pressure boundary (RCPB) remains at 1250 psig. The acceptance limit for pressurization events is the ASME code allowable peak pressure of 1375 psig (110% of design value). The limiting pressurization events are conservatively analyzed, and assume that the events initiate at a reactor dome pressure of 1005 psig and one SRV out-of-service (OOS). The peak calculated RPV pressure remains below the 1375 psig ASME limit, and the maximum calculated dome pressure remains below the Technical Specification 1345 psig Safety Limit. Therefore, there is no decrease in margin of safety.

3.3 Reactor Vessel and Internals

Comprehensive reviews have assessed the effects of increased power conditions on the reactor vessel and its internals. These reviews and associated analyses show continued compliance with the original design and licensing criteria for the reactor vessel and internals.

RPV embrittlement is caused by neutron exposure of the wall adjacent to the core (the "beltline" region). EPU operation may result in a higher neutron flux, which may increase the integrated fluence at the RPV wall over the period of plant license. Because the pre-EPU fluence value bounds the fluence calculated for EPU, the pre-EPU fluence value is used for the EPU evaluations, which demonstrate that the vessels comply with regulatory requirements, and operation with EPU does not have an adverse effect on the reactor vessel fracture toughness.

The effect of the EPU was evaluated to ensure that the reactor vessel components continue to comply with the existing structural requirements of the ASME Boiler and Pressure Vessel Code.

For the components under consideration, Section III, Nuclear Vessels 1965 Edition is the code of construction.

However, if a component underwent a design modification, the governing code for that component was the code used in the stress analysis of the modified component. Typically, new stresses are determined by scaling the "original" stresses, based on EPU conditions (pressure, temperature and flow). The analyses were performed for the design, the normal and upset, and the emergency and faulted conditions. If there is an increase in annulus pressurization, jet reaction, pipe restraint or fuel lift loads, the changes are considered in the analysis of the components affected for upset, emergency and faulted conditions.

The increase in core average power results in higher core loads and reactor internal pressure differences (RIPDs) due to the higher core exit steam flow. The recalculated core loads and RIPD for EPU increase relative to the previous RIPD analyses because of the increase in the thermal power and the consideration of a new core configuration of GE14 fuel. The RIPDs were calculated for normal steady-state operation, upset and faulted conditions for all major reactor internal components, and determined to be acceptable.

A reactor internals structural evaluation of the key reactor internal components was performed to assess the structural integrity for the load changes associated with EPU. This evaluation demonstrates that the structural integrity of the core support and non-core support structure reactor internal components is maintained in the EPU operating condition, consistent with the design basis. However, additional engineering evaluations will be performed to determine if the jet pump riser brace will be susceptible to vibration from the recirculation pump vane passing frequency (VPF). The evaluations will determine if modifications are required to alter the natural frequency of the jet pump braces.

The results of an EPU vibration evaluation show that operation up to 2957 MWt and 100% of rated core flow is possible without any detrimental effects on the safety-related reactor internal components.

Other than structural integrity, the steam separators and dryer do not perform a safety-related function. A plant-specific performance evaluation determined that the steam separators and dryer are capable of performing their operational design function at the increased power level. However, EPU conditions result in an increase in saturated steam generated in the reactor core. For constant core flow, this in turn results in an increase in the separator inlet quality and dryer face velocity and a decrease in the water level inside the dryer skirt, all of which affect the steam separator-dryer performance. The results of the evaluation demonstrate that the steam separator-dryer performance remains acceptable up to some portion of extended power prior to any substantive hardware modification. To reduce the moisture content, hardware modifications are required. These modifications will be completed before EPU implementation. Steam moisture content will be monitored during initial extended power startup testing to determine an acceptable operational moisture content.

3.4 Reactor Recirculation System

The evaluation of the reactor recirculation system performance at EPU conditions determined that adequate core flow can be maintained. Therefore, EPU power operation is within the capability of the reactor recirculation system.

3.5 Reactor Coolant Pressure Boundary Piping

Operation at EPU changes the conditions experienced by the reactor coolant pressure boundary (RCPB). The effects of EPU have been evaluated using the present code(s) of record. In addition to changes in mechanical loading, piping thickness values of carbon steel components can be affected by flow-accelerated corrosion (erosion/corrosion). Erosion/corrosion rates may be influenced by EPU changes in fluid velocity, temperature, and moisture content within carbon steel piping water systems. For systems with an increase in flow rates, vibration can also be induced or aggravated.

The Reactor Recirculation (RR) system was evaluated for compliance with the ANSI B31.1 and applicable criteria for the effects of thermal expansion. A review of the slight change in temperature associated with EPU indicates that RR piping load changes do not result in load limits being exceeded for the RR piping system or for interfacing RPV nozzles, penetrations, flanges or valves. No new postulated pipe break location was identified. The piping load changes do not result in any load limit being exceeding the load limit for any piping snubber, hanger, strut or pipe whip restraint.

The RR system components are made of stainless steel, and system flow does not increase for EPU. Therefore, erosion/corrosion concerns are not applicable to this system.

The Main Steam (MS) and Feedwater (FW) systems experiences increases in flow by approximately 20% due to EPU. The MS and FW piping systems (inside containment) were evaluated for the increases in related loads. The piping load changes do not result in load limits being exceeded for the MS or FW piping system or for interfacing RPV nozzles, penetrations, flanges or valves. No new postulated pipe break location is identified.

The MS piping was evaluated using conservative bounding increases for the effects of load increases related to higher flow rates on supporting snubbers, hangers, struts and pipe whip restraints. This review indicates that the original design analyses do not in every location include sufficient margin to accommodate the higher loads. More detailed analyses demonstrate that the design is adequate for operation at EPU conditions. Minor modifications to pipe support components or support structures are required and will be completed prior to EPU implementation.

The FW piping system was evaluated for the effects of the system condition changes on the supporting snubbers, hangers and struts. This review indicates that the existing design is adequate for EPU conditions and that piping load changes do not result in the load limit of any supporting member being exceeded.

Because piping thickness values of MS & FW carbon steel piping can be affected by flow-accelerated corrosion (erosion/corrosion), and because flow-accelerated corrosion is affected by changes in fluid velocities, temperatures and moisture content, flow-accelerated corrosion effects were evaluated for the carbon steel piping applications within the RCPB.

The integrity of high energy piping systems is assured by proper design in accordance with the applicable Codes and Standards. A consideration in assuring proper design and maintaining system operation within the design is the allowable piping thickness values. The plant has an established program for monitoring pipe wall thinning in single-phase high energy carbon steel piping. The effects of EPU will be incorporated into the existing program.

The adequacy of the other RCPB piping designs for operation at EPU conditions has been evaluated. The nominal operating pressure and temperature of the reactor are not changed by EPU. Aside from MS and FW, no other system connected to the RCPB experiences an increased flow rate at EPU conditions. Only minor changes to fluid conditions will be experienced by these systems due to higher steam flow from the reactor and the subsequent change in fluid conditions within the reactor. Additionally, dynamic piping loads for RV and SRV at EPU conditions are bounded by those used in the existing analyses. These effects have been evaluated for the RCPB portion of the RPV bottom head drain line, RPV head vent line, Isolation Condenser piping, Shutdown Cooling piping, LPCI/Containment Cooling, Core Spray, High Pressure Coolant Injection piping, RV+SRV discharge piping and Reactor Water Cleanup piping, as required.

These other systems were evaluated for compliance with the ANSI B31.1 or ASME Code stress criteria (as applicable). Since none of these piping systems experience any significant change in operating conditions, they are all acceptable as currently designed.

Of these other systems, only the RWCU system has load changes significant enough to require evaluation. The effects of thermal expansion displacements on the supporting snubbers, hangers and struts were reviewed and determined not to result in any load limit being exceeded. Therefore, the existing design is adequate for EPU.

These other systems were evaluated during the development of the plant's flow-accelerated corrosion program, to determine their susceptibility to the affects of flow accelerated corrosion. EPU only slightly changes the inlet temperature to the RWCU system, and does not change any operating parameter of the other RCPB systems listed above. Therefore, the flow accelerated corrosion potential within any of these systems is not expected to change.

The safety-related Main Steam (MS) piping and the safety-related Feedwater (FW) piping will have increased flow rates and flow velocities in order to accommodate EPU. The MS and FW piping will experience increased vibration levels. Other piping systems are not affected. The ASME code requires some vibration test data be taken and evaluated per the nuclear regulatory guidelines for these high energy piping systems, when initially operated at EPU conditions. Vibration data for the MS and FW piping inside containment must be acquired using remote sensors. A piping vibration startup test program that meets the ASME code, in accordance with the regulatory guidelines, will be performed. This program is outlined in Section 10.4.

3.6 Main Steam Line Flow Restrictors

EPU has no adverse effect on the main steam line flow restrictor function. The effects of EPU on main steam line flow restrictor safety and design bases, as identified in UFSAR Section 5.4.4, were evaluated and found to be acceptable.

3.7 Main Steam Isolation Valves

The Main Steam Isolation Valves (MSIVs) are part of the RCPB, and perform the safety function of steam line isolation during certain abnormal events. The MSIVs have been generically evaluated. The generic evaluation covers both the effects of the changes to the structural capability of the MSIV to meet pressure boundary requirements, and the potential effects of EPU related changes to the safety functions of the MSIVs. The conditions for DNPS 2 & 3 are bounded by those in the generic analysis. Technical Specification timing requirements will continue to be met. Therefore, EPU conditions are bounded by the conclusions of the generic evaluation, and the MSIVs are acceptable for EPU operation.

3.8 Isolation Condenser

The Isolation Condenser (IC) system provides core cooling in the event of a transient where the reactor pressure vessel is isolated from the main condenser concurrent with the loss of all feedwater flow. The limiting acceptance criterion for the loss of feedwater flow transient event is to provide adequate core cooling during the transient by maintaining sufficient water level inside the core shroud to ensure that the top of active fuel remains covered throughout the event.

Operation of the IC system at EPU conditions does not have any effect on the availability or the reliability of the system, and does not invalidate any of the original design pressures or temperatures for the system components.

The IC system has been evaluated for the loss of feedwater flow transient (LOFW) event. The evaluation was performed with a reactor vessel high pressure initiation time delay of 15 seconds maximum. Changing the time delay from the current maximum of 17 seconds to a maximum of 15 seconds ensures that the isolation condenser will initiate to remove decay heat for this transient. The LOFW analysis results demonstrate that the limiting acceptance and operational criteria for reactor vessel water level will continue to be met. Therefore, the IC system is acceptable for EPU operation.

3.9 LPCI/Containment Cooling and Shutdown Cooling Systems

The LPCI/CC system is designed to restore and maintain the coolant inventory in the reactor vessel, the SDC system provides primary system decay heat removal following reactor shutdown for post accident conditions. The LPCI/Containment Cooling System is designed to operate in the Low Pressure Coolant Injection (LPCI) mode, Suppression Pool Cooling (SPC) mode, and Containment Spray Cooling (CSC) mode. The SDC System is designed to provide Shutdown Cooling (SDC) or Fuel Pool Cooling (FPC) assist heat removal. The LPCI mode is discussed in Subsection 4.2. The effects of EPU on the remaining modes are discussed in the following subsections.

The operational objective for normal shutdown is to reduce the bulk reactor temperature to 140°F in approximately 24 hours, using three SDC heat exchanger loops. The cool down time for EPU meets this operational objective.

During normal plant operation, the function of the SPC mode is to maintain the pool temperature below the TS limit. Following abnormal events, the SPC mode controls the long-term pool temperature so that the containment design temperature is not exceeded. This objective is met with EPU, because the containment analysis (Section 4.1) confirms that the pool temperature remains below its design limit.

The CSC mode provides suppression pool water to spray headers in the drywell and suppression chamber to reduce containment pressure and temperature during post-accident conditions. EPU increases the containment spray temperature. This increase has no effect on the calculated peak values of drywell pressure, drywell temperature and suppression chamber pressure, because these parameters reach peak values prior to actuation of the containment spray.

FPC assist uses the SDC System heat removal capacity, to provide supplemental fuel pool cooling in the event that the fuel pool heat load exceeds the heat removal capability of the Fuel Pool Cooling and Cleanup (FPCC) System. This mode can be operated to maintain the Fuel Pool temperature within acceptable limits. As discussed in Section 6.3, the increase in fuel pool heat load due to EPU does not exceed the heat removal capacity of this operational mode.

3.10 Reactor Water Cleanup System

The Reactor Water Cleanup (RWCU) system operation at the EPU slightly decreases the temperature and maintains the same pressure within the RWCU System. This system is designed to remove solid and dissolved impurities from recirculated reactor coolant, thereby reducing the concentration of radioactive and corrosive species in the reactor coolant. The system is capable of performing this function at the EPU level.

The RWCU System can perform adequately during EPU with original system flow. EPU results in a slight increase in the reactor water conductivity because of the increase in feedwater flow. However, the present reactor water conductivity limits are expected to be met.

3.11 Balance-Of-Plant Piping Evaluation

This section addresses the adequacy of non-RCPB balance-of-plant (BOP) piping design for operation at EPU conditions. Large bore and small bore safety-related and nonsafety-related piping and supports not addressed in Section 3.5 were evaluated for acceptability at EPU conditions. The system conditions changed by EPU, which have the potential to affect the various piping systems, are primarily due to:

- Increases in flow in the MS, FW and other systems forming part of the turbine cycle.

- Increases in temperature and pressure in portions of the MS, extraction steam, heater drain and cross-around steam piping resulting from the high pressure turbine rotor replacement, which effectively opens the steam flow path.
- Increases in pressure in portions of the FW system resulting from higher FW flow rates.
- Increased temperature of the post-LOCA Torus, which affects all connected piping.

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 NC/ND or ANSI B31.1 Power Piping Code equations. The original codes of record and code allowables, as referenced in the appropriate calculations, were used.

The Design Basis Accident (DBA)-LOCA dynamic loads including the pool swell, vent thrust, condensation oscillation (CO) and chugging loads as well as RV and SRV discharge loads were originally defined and evaluated. The design of structures attached to the Torus shell, such as piping system, vent penetrations and valves include these design basis DBA-LOCA and RV/SRV hydrodynamic loads. These hydrodynamic loads are not increased by EPU conditions, and there is no resulting impact on the design of structures attached to the Torus shell.

Operation at EPU conditions increases stresses on piping and piping system components due to higher operating temperature, pressure and/or flow rate internal to the pipes. For all systems, the maximum stress levels results were reviewed based upon conservative bounding criteria developed from system-specific increases in temperature, pressure and/or flow rate. These piping systems were evaluated to determine if sufficient margins exist in the original design analyses to accommodate the increased stresses due to EPU. Some MS and Torus attached piping was found not to have sufficient margin in the original design analyses to justify its acceptability at the bounding EPU loading conditions. More detailed analyses were performed that demonstrate the adequacy of the existing piping design for EPU conditions. In some cases, piping modifications are required to bring the piping within Code allowable stress limits. These modifications will be completed prior to implementation of EPU. No new postulated pipe break location was identified during this review.

Loads on pipe supports increase due to the same EPU conditions that increase piping stresses. However, when combining these increases with the loads that are not affected by EPU, such as seismic and deadweight, the overall combined support load increases are generally insignificant except for MS and Torus attached piping.

The supports for piping systems with increased stresses at EPU conditions were evaluated to determine if sufficient margins exist between bounding EPU stresses and Code limits in the existing design to accommodate the EPU changes. Some supports were found not to have sufficient margin in the original design/analyses to justify acceptability at EPU conditions. In these cases, more detailed analyses were performed that demonstrate the adequacy of the existing pipe support design for EPU conditions. In some cases, modifications of the supports, structural attachments or supporting steel are required to meet Code allowable stress limits. These modifications will be completed prior to implementation of EPU.

The integrity of high energy piping systems is assured by proper design in accordance with the applicable codes and standards. A consideration in assuring proper design and maintaining system operation within the design is the allowable piping thickness values. Because piping thickness values of carbon steel components can be affected by flow accelerated corrosion (erosion/corrosion), the plant has an established program for monitoring pipe wall thinning in single phase and two-phase high energy carbon steel piping. The effects of EPU will be incorporated into the existing plant pipe monitoring program. This program ensures that EPU effects on high energy piping systems potentially susceptible to pipe wall thinning due to flow accelerated corrosion will be addressed.

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. Short-term and long-term containment analyses results are reported in the UFSAR. The short-term analysis is primarily directed at determining the drywell pressure response during the initial blowdown of the reactor vessel inventory to the containment following a large break inside the drywell. The long-term analysis is primarily directed at the suppression pool temperature response, considering the decay heat addition to the suppression pool. The EPU containment analysis demonstrates that the containment and drywell pressure and temperature responses remain within design allowables.

The LOCA containment dynamic loads include pool swell, condensation oscillation (CO), chugging, and vent thrust loads. Evaluation of the LOCA dynamic loads for EPU is primarily based on the short-term DBA-LOCA pressure and temperature response analysis. The DBA-LOCA pressure and temperature response analyses provide the calculated values of the controlling parameters for the dynamic loads throughout the blowdown. The key parameters are drywell and wetwell pressure, vent flow rates and suppression pool temperature. The DBA-LOCA dynamic loads for EPU remain bounded by the existing load definitions.

The RV plus SRV discharge loads include RV+SRV discharge line (DL) loads, suppression pool boundary pressure loads and drag loads on submerged structures. The RV/SRV discharge loads are evaluated for two different actuation phases: initial actuation and re-actuation. For EPU, the RV/SRV discharge loads due to initial actuation and re-actuation remain bounded by the existing load definitions.

Because this EPU does not include a reactor operating pressure increase, the changes in actual asymmetrical loads on the vessel, attached piping and biological shield wall, due to a postulated pipe break in the annulus between the reactor vessel and biological shield wall are minor. The biological shield wall and component designs remain adequate, because there is sufficient pressure margin available.

The capability of the containment isolation valves to perform their isolation function during normal operations and under engineered safety features actuation conditions has been determined to be acceptable, except as addressed below.

All motor-operated valves (MOVs) used as containment or high energy line break (HELB) isolation valves will be reviewed for the effects of EPU conditions, including potential locking and thermal binding (GL 95-07). If specific valves require calculation revisions, actuator adjustments and/or physical changes to ensure satisfactory performance, then these upgrades and any other field adjustments or modifications will be performed prior to EPU operation.

The plant's past response to Generic Letter 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions," was reviewed for EPU post accident conditions. The results of existing evaluation and the past responses remain valid for the EPU.

4.2 Emergency Core Cooling Systems

The High Pressure Coolant Injection (HPCI) system performance has been generically evaluated for a reactor operating pressure increase. Because there is no pressure increase for this EPU, HPCI operating conditions and performance are not affected, and the generic evaluation is bounding. Therefore, the HPCI system is acceptable for EPU.

The Low Pressure Coolant Injection (LPCI) mode of the LPCI/CC System is automatically initiated in the event of a LOCA. The increase in decay heat due to EPU could increase the calculated peak cladding temperature (PCT) following a postulated LOCA by a small amount. The ECCS performance evaluation presented in Section 4.3 demonstrates that the existing LPCI mode performance capability, in conjunction with the other ECCS, is adequate to meet the post-LOCA core cooling requirement for EPU conditions. The LPCI equipment required to perform the LPCI function are within the existing equipment capabilities, except that the cooling water temperature for the LPCI/CC pump motor upper bearing could exceed the current design value. The LPCI/CC pump motor upper bearing oil uses cooling water from the pump discharge, which is at an increased temperature due to the higher suppression pool water temperature. Therefore, the bearing will be re-qualified for higher temperature, or a modification to ensure adequate bearing cooling will be done prior to the EPU implementation.

The Core Spray (CS) system is automatically initiated in the event of a LOCA. The increase in decay heat due to EPU could increase the calculated PCT following a postulated LOCA by a small amount. The ECCS performance evaluation presented in Section 4.3 indicates that the existing CS system performance capability, in conjunction with the other ECCS, is adequate to meet the post-LOCA core cooling requirement for EPU conditions. The CS equipment required to perform the CS injection function are within the existing equipment capabilities, except that the cooling water temperature for the CS pump motor upper bearing would exceed the current design value. The CS pump motor upper bearing oil uses cooling water from the pump discharge, which is at an increased temperature due to higher suppression pool water temperature. Therefore, the bearing will be re-qualified for higher temperature, or a modification to ensure bearing operation will be done prior to the EPU implementation.

The Automatic Depressurization System (ADS) is required to reduce reactor pressure following a small break LOCA. The ADS initiation logic and ADS valve control are adequate for EPU conditions. The ability to initiate ADS on appropriate signals is not affected by EPU. To achieve the required flow capacity for EPU conditions, five ADS valves must be operable. Prior to EPU, only four ADS valves were required to be operable.

The ECCS NPSH requirements were evaluated for EPU conditions based on the pressure and temperature conditions determined by the containment analysis (Section 4.1), flow requirements based on the containment and LOCA analyses (Section 4.3) and flow losses, including suction strainer losses, determined using the same methodology previously reviewed by the NRC.

Additional credit for containment overpressure is required because the suppression pool temperature increases at a faster rate and peaks at a higher value compared to the pre-EPU conditions during a loss of coolant accident (LOCA). Because vapor pressure increases as the suppression pool temperature increases, the net positive suction head available (NPSHa) for each ECCS pump is reduced. To offset this reduction in NPSHa, more overpressure credit is required. More overpressure is also available, since the containment and suppression pool pressures also increase at a faster rate and peak at a higher value than before EPU.

Existing plant emergency operating procedures include cautions concerning exceeding ECCS pump NPSH limits. The procedures also contain ECCS pump curves of pump flow versus torus pressure and temperature conditions. The same cautions and NPSH curves are included in the emergency operating procedures that control use of containment sprays. Thus, the operators have sufficient procedural direction to control both ECCS pump flow and containment pressure within limits.

The requested overpressure credit was based on the methodology previously approved for DNPS in a 1997 license amendment regarding containment overpressure. This methodology followed the original design basis of one ECCS suction strainer completely blocked, with the remaining three strainers in clean condition. The head loss across the three clean strainers was assumed to be the same as the head loss for the original suction strainers, although those strainers were subsequently replaced with higher capacity strainers. Thus, the assumed head loss is slightly higher than the actual head loss expected with the new strainers. This assumption maintains consistency with the basis for approval of the 1997 amendment.

NPSH calculations have been performed for EPU conditions with the strainer head loss assumptions described above for two short term and two long term flow conditions. The limiting short term ECCS flow case is all four LPCI pumps and both core spray pumps operating at maximum flow conditions. The limiting long term ECCS flow rate is the same as in the 1997 calculations that formed the basis of the currently approved overpressure credit. This limiting flow rate is 19,000 gallons per minute (gpm) distributed as follows: two core spray pumps operating at 4,500 gpm each, one LPCI pump at 5,000 gpm, and two more LPCI pumps at 2,500 gpm each. This flow case is significantly more than the minimum long term flow of 9,750 gpm required to maintain adequate core and containment cooling after EPU. The minimum flow case of one core spray pump operating at 4,750 gpm and one LPCI pump operating at 5,000 gpm is the other case analyzed in the calculations.

In the short term, there is a period from approximately 290 seconds to 600 seconds during which some ECCS pump cavitation can occur, since the available NPSH is less than the required NPSH. This period is after the time at which the peak cladding temperature (PCT) has been reached at approximately 240 seconds. Prior to 290 seconds, the credited overpressure ensures that adequate NPSH is available to meet the core cooling requirements assumed in the PCT calculations. After 600 seconds, ECCS pump throttling restores adequate NPSH. Pump cavitation for the brief time from 290 seconds to 600 seconds is not of concern due to short duration of the cavitation.

4.3 Emergency Core Cooling System Performance

The Emergency Core Cooling Systems (ECCS) are designed to provide protection against hypothetical loss-of-coolant accidents (LOCA) caused by ruptures in the primary system piping. The ECCS performance analysis models satisfy the requirements of 10 CFR 50.46 and 10 CFR 50 Appendix K. The limiting break was analyzed using both nominal and Appendix K assumptions at pre-uprate and at 117% thermal power uprate to assess the impact of EPU. The largest difference between EPU and pre-EPU was less than 10°F for the limiting break PCT. Therefore, the increase in reactor power due to EPU has a negligible impact on the Licensing Basis PCTs, the local oxidation, the hydrogen generation, the coolable geometry, and the long-term cooling.

In the small break LOCA events for which HPCI is assumed to fail, it is assumed that the ADS has the four RVs and the one SRV functioning.

Consistent with the implementation of ARTS power and flow dependent limits, no credit for the APRM setdown was taken in determining the effects of operating within the EPU power/flow map.

4.4 Main Control Room Atmosphere Control System

The increase in heat gain to the control room as a result of EPU for both normal and emergency modes is insignificant. The iodine loading on the control room filters remains a small fraction of the allowable limit of total Iodine (radioactive plus stable) per gram of activated carbon, identified in Regulatory Guide 1.52. Therefore, the control room iodine filter efficiency is not affected by EPU.

4.5 Standby Gas Treatment System

The capacity of the SGTS was selected to provide a negative differential pressure between secondary containment and the outside air of at least 0.25-inch of water. This capability is not affected by EPU. The charcoal filter bed design removal efficiency of 95% for radioiodine is unaffected by EPU.

The amount of cooling airflow needed to limit the adsorber temperature increases, due to fission product decay heating, from 48 cfm to 74 cfm, which is well below the available design flow of 300 cfm. No other SGTS parameter is affected by EPU.

4.6 Post-LOCA Combustible Gas Control

The post-LOCA combustible gas control system (CGCS) consists of the primary containment inerting system, the Nitrogen Containment Atmosphere Dilution (NCAD) system, the Containment Atmosphere Monitoring (CAM) system, and the Augmented Primary Containment Venting System (APCVS). The CGCS 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. Only the post-LOCA production of hydrogen and oxygen by radiolysis, which increases in proportion to power level, is directly impacted by EPU. The hydrogen contribution from metal-water reaction of fuel cladding is not affected by the EPU but is affected by fuel design. Therefore, the analysis considers the impact of GE14 fuel introduction on metal-water hydrogen production.

The analysis shows that the increases in metal-water reaction and post-LOCA radiolytic hydrogen and oxygen production do not impact the ability of the system to maintain containment oxygen at or below the 5% flammability limit, using Regulatory Guide 1.7 assumptions. The time required to reach the 5% oxygen limit following the LOCA, based on 1% per day containment leakage, decreases from 25 hours for pre-EPU conditions to 19 hours for EPU. This reduction in required initiation time does not affect the ability of the operators to respond. Therefore, the CGCS retains its capability of meeting its design basis function of controlling oxygen concentration following the postulated DBA LOCA. GE14 fuel bounds the legacy fuel.

Evaluation of the nitrogen requirements to maintain the containment atmosphere at or below the 5% flammability limit for 7 days post-LOCA shows that the minimum stored volume requirement is 141,000 scf. The NCAD nitrogen storage system, with a minimum volume of 200,000 scf, therefore has sufficient capacity to accommodate 7 days of post-LOCA operation. Analysis of the containment pressure buildup as a result of continuing NCAD operation shows that the containment operating pressure limit of 31 psig (50% of the design pressure) is not exceeded until 32 days after the LOCA. Therefore, the minimum 30-day acceptance limit (to reach 50% of the design pressure) is met for EPU.

5 INSTRUMENTATION and CONTROL

5.1 NSSS Monitoring and Control Systems

The instruments and controls that directly interact with or control the reactor are usually considered within the Nuclear Steam Supply System (NSSS). The NSSS process variables, instrument setpoints and Regulatory Guide 1.97 instrumentation that could be affected by EPU were evaluated. As part of EPU implementation, both the ComEd and General Electric setpoint methodologies are used to generate the allowable values and (nominal trip) setpoints related to the analytical limit changes for EPU.

Changes in process variables and their effects on instrument setpoints were evaluated for EPU operation to determine any related changes. Process variable changes are implemented through changes in plant procedures.

Increases in the core thermal power and steam flow affect some instrument setpoints, as described in Section 5.3. These setpoints are adjusted to maintain comparable differences between system settings and actual limits, and reviewed to assure that adequate operational flexibility and necessary safety functions are maintained at the EPU power level.

For EPU, the average power range monitor (APRM) power signals are adjusted to the EPU power such that the indications read 100% at the new licensed power.

EPU has little effect on the intermediate range monitor (IRM) overlap with the source range monitors (SRM) 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 elimination of the APRM gain and setpoint requirement (due to ARTS power and flow dependent limits) is described in Sections 1.4 and 9.2.

EPU slightly reduces the neutronic life of the LPRM detectors and radiation levels of the TIPs, but the change is expected to be very small.

The Rod Block Monitor (RBM) initiates a control rod block if local power exceeds a preset limit around a selected rod during withdrawal. The RBM is required to be operable when the reactor is at ≥ 30% of current rated power. This applicability value does not change for EPU.

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. Adjustment to the calibration value is needed to maintain the setpoint for EPU.

5.2 BOP Monitoring and Control Systems

Operation of the plant at the EPU power level has minimal effect on the balance-of-plant (BOP) system instrumentation and control devices. Based on EPU operating conditions for the power conversion and auxiliary systems, process control valves and instrumentation have sufficient range/adjustment capability for use at the expected EPU conditions, except as noted in the sections that address each BOP system. However, some modifications may be needed to the power conversion systems to obtain full EPU.

The pressure control system (PCS) provides fast and stable responses to system disturbances related to steam pressure and 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 increased steam flow for EPU along with a change to the turbine high pressure rotor requires the Turbine Control Valves (TCV) to operate under different conditions. The flow capacity of the TCVs and other characteristics after modifications to the high pressure turbine rotor require evaluations to assure that all requirements regarding interaction between the T-G and the NSSS have been addressed.

Specific EHC and steam bypass control system tests will be performed during the power ascension phase. These tests are summarized in Section 10.4.

The turbine EHC system was reviewed for the increase in core thermal power and the associated increase in rated steam flow. For EPU conditions, a second steam line resonance compensator (SLRC) card will be installed to attenuate third harmonic resonance. In addition, TCV Diode Function Generator tuning for the redesigned conditions will be required. The control systems are expected to perform normally for EPU operation.

Modifications to the TCVs may be required for the uprated throttle conditions. Confirmation testing will be performed during power ascension (see Section 10.4).

The feedwater control system is used to maintain water level control in the reactor. The capacity of the feedwater pumps is adequate to support the EPU, and this will be demonstrated by startup testing. The basic capacity requirement for adequate reactor water level control is approximately 105% of the operating point flow rate. The feedwater system has capacity in excess of the 105% of the EPU rated feedwater flow required for transient operation with three feedwater pumps operating. With adjustments in feedwater and steam flow instrument spans and feedwater pump runout protection, the control system is capable of accessing as much of the flow as needed. Therefore, the capacity is sufficient for acceptable control.

The control system is adjusted to provide acceptable operating response on the basis of unit behavior. It has been set up successfully to cover the current power range using startup and periodic testing. For EPU, no change in the operating water level is required. The feedwater flow control system device settings have the sufficient adjustment ranges to ensure satisfactory operation. This will be confirmed by performing unit tests during the power ascension to EPU conditions (Section 10.4).

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 EPU. Each of the systems (listed below) where leak detection could potentially be affected by EPU, was evaluated, and no leak detection related change is required.

- Main Steam Tunnel Temperature Based Leak Detection
- RWCU System Temperature Based Leak Detection
- IC System Temperature Based Leak Detection
- HPCI System Temperature Based Leak Detection
- Non-Temperature Based Leak Detection

5.3 Instrument Setpoints

Instrument setpoints in the Technical Specifications (TS) are established using approved setpoint methodologies. 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 also provided between the actual trip setting and the normal operating limit to preclude inadvertent initiation of the protective action.

The following paragraphs discuss those instrument setpoint analytical limits that are potentially affected by EPU. Plant setpoints (derived from the EPU analytical limits) ensure timely actuation of the necessary safety functions while avoiding spurious trips during EPU operation.

- Because no pressure increase is associated with this EPU, the scram analytical limit (AL) on reactor high pressure is unchanged.
- The current ATWS-RPT high pressure setpoint was included in the ATWS evaluation discussed in Section 9.3. This evaluation concludes that the current ATWS-RPT high pressure setpoint is acceptable for EPU.
- Because there is no increase in reactor operating dome pressure, the setpoints for the SSVs, SRV and RVs are not increased.
- The current Main Steam Line (MSL) High Flow Isolation AL of 120% of rated steam flow is increased, to improve operating margin and reduce unnecessary MSL isolations. Because of the differences in the MSL flow restrictor sizings, the ALs differ between the units. The Unit 2 MSL design permits its AL to be raised to 125% of the EPU steam flow. The Unit 3 MSL design permits its AL to be raised to 140% of the EPU steam flow. This provides additional operating margin without reducing safety margin. Each unit's instrumentation will be recalibrated for its higher steam flow condition, and thus, the Technical Specifications Allowable Values are being changed accordingly. These changes help ensure that sufficient pressure differences to the trip setpoints exist to allow for normal plant testing of the MSIVs and turbine stop and control valves.

- New analytical limits (ALs) of the flow biased APRM scrams and rod blocks are developed for EPU, similar to those shown in Figure 5-1 of ELTR1. The ALs for the APRM Flow Biased Scram, APRM Rod Block, and RBM Setpoints form the basis for the EPU/MELLLA setpoints, including the minimum core flow allowable at EPU power. The EPU application of the flow biased RBM (non-ARTS) is to maintain the same AL values, which is the same basis as for the Fixed (Non-Flow Biased) High APRM Scram. The APRM Scram and Rod Block are clamped at their maximum power values based on a core flow of 95.3%. The Rod Block Monitor is clamped at its maximum power value based on 100% core flow. The MELLLA AL for the fixed (clamped) APRM scram for two recirculation loop operation remains the same but the AL for single recirculation loop operation (SLO) is changed to be the same as the AL for TLO.

- The RBM instrument setpoints are determined on a fuel cycle-specific basis and will be modified (as needed) when EPU is first implemented.

- The purpose of the Low Steam Line Pressure MSIV Closure (RUN Mode) trip is to initiate MSIV closure on low steam line pressure when the reactor is in the RUN mode. This setpoint is not changed for the EPU.

- The reactor water level trip values used in the safety analyses do not require changing as a result of EPU. However, the reactor low water level scram AL is being reduced, to provide additional operating margin (i.e., prevent unnecessary scrams) for a reactor recirculation runback on a loss of a reactor feedwater pump from EPU conditions. The revised low water level scram AL is used in the applicable EPU safety analyses (i.e., transient and ECCS-LOCA). Also, the primary containment and RWCU isolation trips initiate from the same reactor low water level as used for the scram trip. Therefore, the allowable values (AVs) used for the primary containment and RWCU isolations must be revised to remain consistent with the scram function.

- At EPU conditions, the increase in steam tunnel ambient temperature is not significant, and thus, no change to the MSL Tunnel High Temperature Isolation setpoint is required.

- With the increased heat input due to EPU, the condenser backpressure rises. The plant has a nominal alarm for condenser low vacuum at 24.5 inches Hg and a nominal scram at 22.5 inches Hg. To maintain adequate operating margin between the alarm and the scram, the alarm setpoint, nominal scram setpoint and associated AV will be adjusted. The AL for this function is unchanged.

- The TSV Closure and TCV Fast Closure Scram Bypass AL expressed as a percent of rated thermal power is reduced by the ratio of the power increase. The new AL does not change with respect to absolute thermal power and steam flow, and thus, there is no effect on the transient response. A high pressure turbine rotor modification changes the relationship between turbine first stage pressure and steam flow such that the scram bypass AL in psig must change to assure that the scram bypass does not occur above the desired core thermal power and turbine steam flow point.

- For EPU, the Rod Worth Minimizer low power setpoint (LPSP) remains 10% of RTP. This is conservative, because it requires enforcement of rod pattern controls to a higher absolute power level.

- The pressure control system (PCS) is discussed in Section 5.2. The pressure setpoint, pressure regulator gain, main steam line pressure drop, turbine stop valve inlet pressure and turbine-generator required load setpoint are related to each other and to reactor dome pressure. The reactor dome pressure is not changed for EPU. However, the increased steam flow results in a somewhat greater steam line pressure loss. Therefore, the steam bypass control system pressure regulator operational setpoint must be adjusted to achieve the desired reactor pressure. Due to small differences in plant parameters, the optimal pressure regulator setpoint may slightly differ between the units. Specific EHC and steam bypass control system tests will be performed during the initial power ascension following any T-G modifications needed to implement EPU. These tests are summarized in Section 10.4.

- The current value of the feedwater flow setpoint for recirculation cavitation protection is unchanged in terms of absolute feedwater flow rate. However, the relative setpoint, as it appears on the power/flow map, is reduced slightly to account for the EPU.

- For EPU, the AL for the Isolation Condenser (IC) steam/condensate line high flow indications remain based on 300% of the maximum system flow.

- For EPU, the AL for the HPCI steam line high flow isolation remains based on 300% of the maximum rated steam flow to the HPCI turbine.

6 ELECTRICAL POWER and AUXILIARY SYSTEMS

6.1 AC Power

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

- The isolated phase bus duct is adequate for both rated voltage and low voltage current output. The adequacy of the bus duct cooling system is being evaluated, and any necessary changes to the system will be made prior to EPU implementation.
- The main transformers and the associated switchyard components are adequate for the uprated output.
- A grid stability analysis determined that there is no significant effect on grid stability or reliability. There is no modification associated with EPU that would increase electrical loads beyond those levels previously included, or revise the control logic of the distribution systems.

Station loads under normal operation/distribution conditions are computed based on equipment nameplate data and calculated brake horsepower with actual diversity factors applied. The only identifiable change in electrical load demand is associated with condensate and booster pumps, reactor recirculation pumps, reactor feedwater pumps, and condensate demineralizers. The increased flow due to EPU conditions requires energizing the installed spare (third) reactor feedwater pump, energizing the installed spare (fourth) condensate and booster pump, and the increase of the operating point for the two reactor recirculation pumps. These additional loads when evaluated by design basis calculations result in acceptable operation of the electrical auxiliary system during normal startup and operation with two auxiliary transformers in service.

Operation at EPU conditions on a single transformer exceeds the non-safety 4160V switchgear short circuit rating, the transformer winding rating, and the bus duct rating. Also, in the event of a fast transfer to single transformer operation at EPU conditions, the same situation will exist. To address these potential operational problems, Dresden will institute a procedurally controlled load shedding scheme to be implemented following a fast transfer. This approach will be confirmed by thermal analysis or an engineering evaluation to address the overload conditions for the auxiliary transformers, the bus duct, and related connections. To address the potential operational problem due to the switchgear overduty condition, a test to upgrade the switchgear and breakers to a higher momentary current rating will be performed and a time delay of about 6 cycles on the short circuit interrupting will be implemented. A review of the 4160V bus and auxiliary transformer overcurrent relay setpoints will also be performed to ensure proper settings for operation at EPU conditions.

No increase in flow or pressure is required of any AC-powered ECCS equipment for EPU. Therefore, the amount of power required to perform safety-related functions (pump and valve loads) is not increased with EPU and the current emergency diesel generator 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 DC Power

The direct current (DC) loading requirements in the UFSAR were reviewed, and no reactor power dependent loads were identified that affected the DC Power System design. System loads were computed based on equipment nameplate data. Operation at the EPU level does not increase any loads beyond nameplate rating or revise any control logic; therefore, the DC power distribution system is adequate. To restore the margin at the reactor building DC panels, the amperage capacity the main feed cables to these panels will be increased.

6.3 Fuel Pool

The effects of EPU on fuel pool cooling, crud and corrosion products in the fuel pool, radiation levels and structural adequacy of the fuel racks are small and within the design limits of the affected systems and components.

EPU increases the spent fuel pool heat load. The adequacy of the FPCCS is determined by evaluating the ability of the system to maintain the temperature of the fuel pool. The fuel pool temperature is analyzed by calculating the decay heat load following a normal batch discharge or full core discharge, with other spaces filled as a result of fuel discharges from normal refueling outages. Analyses performed include the use of SDC system heat exchanger in the Fuel Pool Assist Mode as well as the FPCCS heat exchangers, for maintaining the fuel pool temperatures below the required limit for EPU conditions.

The EPU analysis assumes a 24 month fuel cycle as the basis. Each fuel cycle affects the decay heat generation in the spent fuel discharged from the reactor. This evaluation considers the expected heat load in the spent fuel storage pool at the uprated conditions, and analyses confirm the capability of the fuel pool cooling system to maintain adequate fuel pool cooling.

Crud activity and corrosion products associated with spent fuel may increase slightly due to EPU. However, the increase is shown to be insignificant, and fuel pool water quality is maintained by the fuel pool cleanup system.

The normal radiation levels around the pool can increase slightly primarily during fuel handling operation. This increase is acceptable and does not significantly increase the operational doses to personnel or equipment.

The fuel racks are designed for higher temperatures than are anticipated from EPU. There is no effect on the design of the fuel racks, because the original design fuel pool temperatures are not exceeded. Therefore, the racks are acceptable for the higher local decay heat loads.

6.4 Water Systems

The environmental effects of EPU are controlled at the same levels as for the original analyses. That is, none of the present limits for plant environmental releases are increased as a consequence of EPU. If the plant releases challenge environmental limits then plant operation is managed such that the existing limits would not be violated with uprate.

The safety-related service water systems are designed to provide reliable supplies of water for the following essential equipment and systems:

Containment cooling heat exchangers;

Diesel generator cooling water (DGCW) heat exchangers;

DGCW pump motors;

Control room emergency ventilation system refrigeration condensing unit;

LPCI room coolers;

HPCI room cooler;

Containment cooling service water (CCSW) pump vault coolers; and

CCSW keep fill, alternate water supply.

The safety-related performance of the CCSW and DGCW service water systems during and following the most demanding design basis event, the LOCA with LOOP, has been reviewed and found acceptable. The containment cooling analysis in Section 4.1 assumes that the post LOCA containment cooling does not change. The increased heat load is within the existing capacity of the CCSW system.

The temperature of service water discharge results from the heat rejected to the service water system via the closed cooling water systems and other auxiliary heat loads. The major service water heat load increases from EPU reflect an increase in main generator losses rejected to the stator water coolers and hydrogen coolers in addition to increased Turbine Building Closed Cooling Water (TBCCW) and Reactor Building Closed Cooling Water (RBCCW) heat loads.

The increased heat loads result in a slight increase in the temperature of the service water discharged to the Circulating Water System.

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

EPU operation increases the heat rejected to the condenser, and therefore, reduces the difference between the operating pressure and the required minimum condenser vacuum. If condenser pressures approach the backpressure limitation, then reactor thermal power must be reduced to maintain adequate condenser vacuum, thereby limiting generator output.

A comparison of state discharge limits to the current discharges and bounding analysis discharges for EPU demonstrates that the plant remains within the state discharge limit during operation at EPU. Regardless, if needed to accommodate extremes in ambient conditions, plant operations (e.g., temporary plant de-rating) will ensure that state discharge limits are not exceeded.

The heat loads on the RBCCW system do not increase significantly by EPU because they depend mainly on either vessel temperature or flow rates in the systems cooled by the RBCCW. The RBCCW system contains sufficient redundancy in pumps and heat exchangers to ensure that adequate heat removal capability is always available. Therefore, sufficient heat removal capacity is available to accommodate the increase in heat load due to EPU.

The heat loads, which are increased by EPU, on the Turbine Building Closed Cooling Water (TBCCW) system include the Bus Duct Coolers, the added heat from the operation of the fourth Condensate/Condensate Booster Pump and the added heat from the operation of the third Reactor Feed Pump. The remaining TBCCW heat loads are not strongly dependent upon reactor power and do not increase significantly. The additional heat loads can be removed by the TBCCW system with a minimal increase in TBCCW temperature, which will have negligible effect on the equipment cooled by the TBCCW and is therefore deemed acceptable.

The normal heat sink is the river via the intake and discharge canals. However, in the event of a loss of the downstream dam, the water trapped in the intake canal becomes the ultimate heat sink (UHS). In this event, make-up water addition is required to the intake canal for decay heat removal at EPU conditions. This make-up activity is currently required for present plant operations. Sufficient time is available to replenish the water in the intake canal following a loss of the dam to adequately remove the decay heat at EPU to maintain shutdown conditions.

6.5 Standby Liquid Control System

The ability of the SLCS boron solution to achieve and maintain safe shutdown is not affected by EPU. SLCS shutdown capability (in terms of required boron concentration) is reevaluated for each fuel reload.

The ATWS performance evaluation (Section 9.4) shows that EPU has no adverse effect on the ability of the SLCS to mitigate an ATWS.

6.6 Power-Dependent Heating Ventilation and Air Conditioning

The heating ventilation and air conditioning (HVAC) systems consist mainly of heating or cooling supply, exhaust and recirculation units in the turbine building, reactor building and the drywell. The EPU is expected to result in a small increase in the heat loads caused by slightly higher process temperatures and higher electrical currents in some motors and cables.

The affected areas are the steam tunnel, ECCS pump rooms, and drywell in the reactor building; the feedwater heater bay and condenser area, feedwater pumps, condensate/condensate booster pumps and the MG set areas in the turbine building. Other areas are unaffected by the EPU because the process temperatures remain relatively constant.

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

6.7 Fire Protection

Operation of the plant at the EPU power 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 the plant modification and fire protection programs. The safe shutdown systems and equipment used to achieve and maintain cold shutdown conditions do not change, and are adequate for EPU conditions. The operator actions required to mitigate the consequences of a fire are not affected. Therefore, the fire protection systems and analyses are not affected by EPU.

The reactor and containment responses to the postulated 10 CFR 50 Appendix R fire event at EPU conditions were re-analyzed, and show that the fuel PCT, reactor pressure, and containment pressures and temperatures are below the acceptance limits. This plant-specific evaluation demonstrates 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 EPU demonstrate that fuel cladding, RPV and containment integrities are maintained. Therefore, EPU has no adverse effect on the ability 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

Systems with No Impact:

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

- Automatic Dispatch
- Cathodic Prot
- Communications
- Control Rod Velocity Limitors
- Counting/Decon RM HVAC Blower/Fan
- CR HVAC
- Crane & Hoists
- CRD Support Shootout Steel
- Elevators
- Fuel/ Refueling – Tools & Equipment
- Guard House HVAC SPLY / Fan
- HRSS HVAC

- Lab and Counting Rm Equip
- Lighting and Receptacles
- Misc (non-power generation) Systems
- Miscellaneous HVAC
- Out BLDG HVAC
- Oxygen Injection
- Radwaste Facility HVAC
- Screen Wash
- Service Air/ Emergency Breathing Air
- Service BLDG A/C HTG HVAC
- Sewage Treatment
- Site
- Turbine Building & RW Air Part Sample
- UPS/ Batt and CMPTR Room HVAC
- Vent DMPR / Equipment
- Waste Water Treatment
- Work Execution Center HVAC

Systems with Insignificant Impact:

Some systems are affected in a very minor way by operation of the plant at the EPU level. For the following systems, the effects of EPU are insignificant to the design or operation of the system and equipment:

- Area Radiation Monitoring (alarm setpoints may be adjusted slightly based on area dose rate changes)
- Aux Electric Equipment Room ventilation
- Battery Room ventilation
- Clean Demin/ Makeup Demin
- Control Rod Blades
- Control Room Panels
- Corrosion Test Loop
- Diesel Oil
- EDG/SBO Vent Fans
- FW Pump motor HVAC system

- High Radiation Sampling
- HVAC & U1 Gaseous Monitoring
- Hydrogen & Zinc Addition
- Hypochlorite
- Instrument Air
- Local Panels and Racks
- N2 Inerting & DW O2 Sample
- Off Gas Air Part Sample
- Reactor Protection System
- Test Equipment
- Well Water

Table 6-1

Uprated Plant Electrical Characteristics

Main Generator Electrical Design Parameters [1]	Data	
	D 2 Value	D 3 Value
Generator Rating (MVA)	960	960
Gross Generator Output (MWe)	912	912
Rated (KV)	18	18
Power Factor	0.95	0.95
Current Output (Amps)[2]	30792	30792
Isolated Phase Bus Duct Rating:		
Main Section (Amps)	33,000	33,000
Branch Section (Amps)	2,000	2,000
Main Transformers Rating (MVA)	985	952
Transformer Output (MVA)	940	940

Notes:

1. Main Generator MVA ratings for EPU were evaluated and found acceptable.
2. The current output is calculated using Gross Generator Output (960MVA)

7 POWER CONVERSION SYSTEMS

The power conversion systems were originally designed to utilize the energy available from the nuclear steam supply system and were designed to accept the system and equipment flows resulting from continuous operation at 9,754,965 lb/hr of design steam flow. However, the structural capabilities of the power conversion systems allow for steam flows greater than the (9,754,965 lb/hr) design steam flow, to EPU conditions, with modifications to the high pressure turbine and to some nonsafety-related equipment.

7.1 Turbine-Generator

With uprate the expected generator output is 912 MWe at 0.95 power factor which is within the capability of the generator.

Steam specification calculations were performed to determine the uprated turbine steam path conditions. From the thermodynamic models, turbine and generator stationary and rotating components were evaluated for increased loadings, pressure drops, thrusts, stresses, overspeed capability and other design considerations to ensure that design limits are not exceeded and that plant operation remains acceptable at the EPU condition. In addition, valves, control systems and other support systems were evaluated. The evaluations show that the modifications to the high pressure turbine and some nonsafety-related equipment should ensure satisfactory operation at EPU conditions.

EPU has a negligible effect on HP rotor strength properties and mechanical parameters. The replacement EPU HP rotor consists of an integral rotor, without shrunk-on wheels. The new integral HP turbine rotor is not considered a source for potential missile generation, and therefore, a HP turbine rotor missile probability analysis is not required.

An evaluation of the LP rotors is being performed. The results of this evaluation will be used to determine if changes are required.

The overspeed calculation compares the entrapped steam energy contained within the turbine and the associated piping, after the stop valves trip, and the sensitivity of the rotor train for the capability of overspeeding. The entrapped energy increases slightly for EPU conditions. However, there is sufficient design margin in the current overspeed trip settings to provide protection for a turbine trip, such that should a change in the overspeed settings be necessary, it can be accommodated.

7.2 Condenser and Steam Jet Air Ejectors

The condenser was calculated for performance at EPU conditions based on a cold water temperature at 93°F and current water system flow. An additional analysis for EPU conditions also determined the condenser backpressure would be below its Hg abs design limit.

Both condenser hotwell capacities and level instrumentation are adequate for EPU conditions. Condenser tube staking is planned for the main condensers, which provides adequate protection against tube vibration damage at EPU conditions.

The design of the condenser air removal system is not adversely affected by EPU. The physical size of the primary condenser and the evacuation time are the main factors in establishing the capabilities of the vacuum pumps. These parameters do not change. Because the flow rate does not change, there is no change to the two minute holdup time in the mechanical vacuum pump discharge line. Planned steam dilution modifications of the condenser air removal system to address existing performance issues will provide adequate capacity for EPU conditions.

7.3 Turbine Steam Bypass

The turbine bypass valves were initially rated for a total steam flow capacity of not less than 40% of the original rated reactor steam flow of 9.81 Mlb/hr. Each of 9 bypass valves is designed to pass a steam flow of 0.436 Mlb/hr for a total bypass capacity of 3.92 Mlb/hr. At EPU conditions, rated reactor steam flow is 11.71 Mlb/hr, resulting in a bypass capacity of 33.5%, which is adequate for EPU. All of the transient analyses involving bypass capacity remain valid because the assumed bypass flow is not changed for EPU.

7.4 Feedwater and Condensate Systems

The feedwater and condensate systems do not perform a system level safety-related function. They are designed to provide a reliable supply of feedwater at the temperature, pressure, quality and flow rate as required by the reactor. Their performance has a major effect on plant availability and capability to operate at the uprated condition. For EPU, the feedwater and condensate systems will meet their performance criteria with modifications to some nonsafety-related equipment and changes in operating line-up

Modifications, such as recirculation runback, and alteration of operating system line-up to some nonsafety-related equipment in the feedwater and condensate systems are necessary to attain full licensed EPU thermal power. The current power level requires operation of three of the four condensate/condensate booster pumps and two of the three feedwater pumps. At EPU conditions, operation of all four condensate/condensate booster pumps and all three feedwater pumps is required.

Normal Operation:

The condensate and feedwater systems were originally designed for 105% rated steam flows. Operation at the EPU level does not significantly affect the operating conditions of these systems. As flow through individual pumps increases, the discharge pressure at the condensate and condensate booster pumps decreases due to the pump head characteristics at increased flows. During steady-state conditions, the condensate and feedwater systems have adequate NPSH for all of the pumps to operate without cavitation in the uprated conditions.

The existing feedwater design pressure and temperature requirements are adequate.

Transient Operation:

To account for feedwater demand transients, the feedwater system was evaluated to ensure that a minimum of 5% margin above the EPU feedwater flow was available. This is the same criterion applied to the original design. For system operation with all system pumps available, the predicted operating parameters were acceptable and within the component capabilities.

The plant will implement a reactor recirculation runback feature, to ensure scram avoidance during EPU conditions. A transient analysis was performed to determine the feedwater capacity available following a single feedwater pump trip and subsequent recirculation system runback. The results of the analysis show that the system response is dependent on the runback rate, rather than the feedwater system capability to avoid a scram during the short-term portion of the transient.

Condensate Demineralizer System:

The effect of EPU on the Condensate Demineralizer System was reviewed. The addition of a pre-filtration system allows the plant to continue efficient operation following EPU implementation. The pre-filtration system decreases the burden on the demineralizers, and the system is adequate for EPU operation. The time interval between backwashing (as a system) should increase with the pre-filtration system. Section 8 addresses the effects on the radwaste systems.

8 RADWASTE SYSTEMS AND RADIATION SOURCES

8.1 Liquid Waste Management

The liquid radwaste system collects, monitors, processes, stores and returns processed radioactive waste to the plant for reuse or for discharge. The concentration of activated corrosion products in liquid wastes is expected to increase proportionally to the EPU. The volume of liquid wastes is not expected to increase appreciably. The volume of condensate resin generated is expected to increase proportionally to the EPU, due to increased temperature and flow in the condensate system.

An evaluation concludes that the requirements of 10 CFR 20 and 10 CFR 50, Appendix I will continue to be met. Therefore, EPU does not have an adverse effect on the processing of liquid radwaste, and there are no significant environmental effects.

8.2 Gaseous Waste Management

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

Air inleakage evacuated from the main condenser contains non-condensable radioactive gas, normally consisting of activation gases (principally N-16, O-19 and N-13) and fission product radioactive noble gases. These non-condensable gases, along with the non-radioactive air inleakage, are continuously removed from the main condensers by the steam jet air ejectors (SJAEs), which discharge into the offgas system. This process stream represents the major source of radioactive gas (greater than all other sources combined) exiting the primary system.

The activity of airborne effluents released through building vents does not significantly increase with EPU. The expected gaseous effluents are within limits for original power operation. There are no significant environmental effects due to EPU.

Offgas System:

The radioactive releases from the offgas system are conservatively estimated to increase proportionally to the EPU. This estimate is conservative because it is based on the assumption of a non-negligible amount of fuel leakage due to defects. Because the current and expected fuel defect rates are extremely small, the actual offgas release rate may not increase. EPU increases reactor condensate temperature, which increases the offgas condenser effluent temperature, thus requiring setpoints changes to downstream non-safety temperature instruments.

8.3 Radiation Sources in 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.

8.4 Radiation Sources in Reactor Coolant

Radiation sources in the coolant are primarily a function of fuel defects, power level, and operation of the water cleanup systems. It is expected that some increase in fission product activity in reactor coolant will be seen. Using the formula in ANSI/ANS 18.1-1999, "Radiological Source Term for Normal Operation for Light Water Reactors," the increase would result in a calculated 12% increase in concentration. Even with this increase, the reactor coolant activity levels will be fractional parts of the design basis coolant concentrations.Therefore, EPU should essentially have no adverse effect on day to day operation of the plant.

Hydrogen Water Chemistry (HWC) increases the concentration of N-16 in the steam relative to the concentration with Normal Water Chemistry (NWC). The plant is treated by the NobleChem™ process, which significantly reduces the needed hydrogen injection rate compared to the HWC rate without NobleChem™. Therefore, NobleChem™ significantly reduces the N-16 increase normally associated with HWC. The net effect of NobleChem™ on N-16 concentration more than compensates for any potential increase in N-16 caused by EPU.

8.5 Radiation Levels

For EPU, normal operation radiation levels are expected to increase by no more than the percentage increase in power level. For conservatism, many aspects of the plant were originally designed for higher-than-expected radiation sources. Thus, the increase in radiation levels does not affect radiation zoning or shielding in the various areas of the plant because it is offset by conservatism in the original design, source terms used and analytical techniques.

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

The change in core inventory resulting from EPU is expected to increase post-accident radiation levels by no more than the percentage increase in power level. The estimated increase in radiation levels at EPU conditions does not significantly affect the post-accident radiation zoning or shielding assessment presented in the UFSAR, because the estimated increase in dose rate levels is offset by the conservatism in the analytical techniques utilized to develop the original dose rates. EPU has no effect on the habitability of the Technical Support Center or Emergency Operations Facility.

8.6 Normal Operation Off-Site Doses

For EPU, the normal operation activity in the reactor coolant is expected to increase by approximately the same percentage as that of the uprate, i.e., 17%. Examination of the normal radiological effluent doses reported for the last 5 years (1995 – 1999) indicate that the current releases are a small fraction of the 10 CFR 50 Appendix I guidelines. Thus, the dose effect of EPU continues to be a small fraction of the 10 CFR 50 Appendix I guidelines, and remains within the limits of 10 CFR 20.

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 investigated according to the type of initiating event per Regulatory Guide 1.70, Chapter 15. The generic guidelines identify the analytical methods, the operating conditions that are to be assumed, and the criteria that are to be applied. The additional analyses for ARTS power and flow dependent limits are addressed in Section 9.2.

The analytical results for a representative core show that the overall capability of the design meets all transient safety criteria for EPU operation.

The cycle specific SLMCPRs for both two recirculation loop and single recirculation loop operations will be supplied in the Core Operating Limit Reports (COLRs).

The severity of transients at less than EPU RTP are not significantly affected by EPU, because of the protection provided by the ARTS power and flow dependent limits.

The Loss of Feedwater Flow (LOFW) transient was analyzed for EPU. The sequences of events do not require any new operator actions or shorter operator response times. Therefore, the operator actions for a LOFW transient do not significantly change for EPU.

9.2 Transient Analysis For ARTS Power and Flow Dependent Limits

The core-wide AOOs were analyzed to support the EPU (which includes the MELLLA domain) and the incorporation of the ARTS power and flow dependent limits program. To support the implementation of the ARTS power and flow dependent limits program, these analyses determine the off-rated power- and flow-dependent MCPR and LHGR curves associated with the removal of the APRM gain and setpoint requirement. These evaluations also include consideration from the ECCS-LOCA analysis (Section 4.3).

Transient analyses at a variety of power and flow conditions were performed during original development of the ARTS improvement program. These evaluations are applicable for operation in the MELLLA region. The analyses were utilized to study the trend of transient severity without the APRM gain and setpoint.

Generic power-dependent MCPR and LHGR limits (in terms of multipliers on the plant's rated operating limits) were developed for use in the low power range. The applicability of these generic limits is verified for plant-specific application during the initial ARTS application for that plant. Plant-specific analyses of limiting transients confirm the applicability of the generic power-dependent limits. Cycle specific limits may also be used for any part of the range. A comparison of these plant-specific calculated values with the generic power-dependent MCPR limits (MCPR(P) limits) verifies the applicability of the generic limits to DNPS.

In the absence of the APRM gain and setpoint requirement, power-dependent LHGR limits, expressed in terms of a LHGR multiplier, LHGRFAC(P), are substituted to assure adherence to the fuel thermal-mechanical design bases. The power-dependent LHGRFAC(P) limits were generated using the same database as used to determine the MCPR multiplier (K(P)). Similar to the MCPR(P) limits, plant-specific transient analyses were performed to demonstrate the applicability of the generic LHGRFAC(P) limits.

The transient and initial condition selection, as well as the approach taken to confirm and develop the appropriate plant-specific LHGRFAC(P) limits, is identical to that described in the above discussion for MCPR(P).

Flow-dependent MCPR limits, MCPR(F), ensure that the Safety Limit MCPR (SLMCPR) is not violated during recirculation flow increase events. To verify the applicability of the generic flow-dependent MCPR limits, recirculation flow runout events were performed at a typical mid-cycle exposure condition. These flow runout events were simulated along a rod line which bounds the maximum licensed rod line to the maximum core flow runout values at 108% core flow condition. The ARTS-based MCPR(F) limit is specified as an absolute value and is generic and cycle-independent.

Flow-dependent LHGR limits, LHGRFAC(F), ensure adherence to all fuel thermal-mechanical design bases in the event of slow recirculation flow runout event. The same transients events used to support the MCPR(F) operating limits were analyzed generically, and the resulting overpowers were statistically evaluated as a function of the initial and maximum core flow. From the bounding overpowers, the LHGRFAC(F) limits were derived such that, during these events, the peak transient linear heat generation rate would not exceed fuel mechanical limits. The flow-dependent LHGR limits are generic, cycle-independent and are specified in terms of multipliers, LHGRFAC(F), to be applied to the rated LHGR values.

At any given power/flow state (P,F), all four limits are determined: MCPR(P), LHGRFAC(P), MCPR(F) and LHGRFAC(F). The most limiting MCPR and the most limiting LHGR [maximum of MCPR(P) and MCPR(F) and minimum of LHGRFAC(P) and LHGRFAC(F)] are the governing limits.

The results of the analyses documented above can be utilized to determine the plant-specific OLMCPRs.

9.3 Design Basis Accidents

For EPU, the power dependent plant-specific radiological assessments reported in the UFSAR are re-evaluated at 102% of the EPU RTP level. The plant-specific radiological analyses were performed based on EPU conditions for selected postulated accidents. The events reanalyzed were the Loss-of-Coolant Accident (LOCA), the Fuel Handling Accident (FHA), and the Control Rod Drop Accident (CRDA). The resulting doses from these accidents are provided in Tables 9-

1, 9-2 and 9-3, and demonstrate that the plant continues to meet the applicable regulatory guideline exposures values.

9.4 Special Events

For EPU, the plant-specific ATWS analysis was performed, and the results ensure that the following ATWS acceptance criteria are met:

1. Peak vessel bottom pressure less than ASME Service Level C limit of 1500 psig.
2. Peak clad temperature within the 10 CFR 50.46 limit of 2200°F.
3. Peak clad oxidation within the requirements of 10 CFR 50.46.
4. Peak suppression pool temperature shall not exceed 202°F (bounding post-accident suppression pool temperature).
5. Peak containment pressure shall not exceed 62 psig (peak allowable design pressure).

Therefore, the plant response to an ATWS event at EPU is acceptable.

The DNPS station blackout (SBO) evaluation was performed using the guidelines of NUMARC 87-00 except where USNRC Regulatory Guide 1.155 takes precedence. The plant responses to and coping capabilities for an SBO event are not affected by operation at the EPU level, because the increase in the decay heat for EPU is absorbed by the operation of the Isolation Condenser. There is no change to the systems and equipment used to respond to an SBO nor is the required coping time changed. Therefore, the plant continues to meet the requirements of 10 CFR 50.63 after EPU implementation.

Table 9-1

LOCA Radiological Consequences

Location	UFSAR	Current	EPU	Limit
Exclusion Area:				
Whole Body Dose, rem	2(1)	2	2.3	≤ 25
Thyroid Dose, rem	37(1)	37	46	≤ 300
Low Population Zone:				
Whole Body Dose, rem	1(1)	1	1.2	≤ 25
Thyroid Dose, rem	230(1)	230	290	≤ 300
Control Room:				
Whole Body Dose, rem	0.208(2)	0.424	0.505	≤ 5
Thyroid Dose, rem	22.64(2)	22.96	29.6	≤ 30
Beta Dose, rem	2.14(2)	9.70	11.5	≤ 30

Notes:

1. UFSAR Sect.15.6.5.5.1, NRC analysis.
2. UFSAR Sect.15.6.5.5.3, Table 15.6-10.

Table 9-2

CRDA Radiological Consequences

Location	UFSAR	Current	EPU	Limit
Scenario 1: NRC Standard Review Plan 15.4.9 Approach				
Exclusion Area:				
Whole Body Dose, rem	(1)	0.47(2)	0.58	≤ 6.25
Thyroid Dose, rem	(1)	12.3(2)	16	≤ 75
Low Population Zone:				
Whole Body Dose, rem	(1)	3.8E-2(2)	4.6E-2	≤ 6.25
Thyroid Dose, rem	(1)	0.83(2)	1.1	≤ 75
Scenario 2: Release Via the Augmented Offgas System				
Exclusion Area:				
Whole Body Dose, rem	Not reported	0.71(2)	0.86	≤ 6.25
Thyroid Dose, rem	in UFSAR	11.4(2)	14.8	≤ 75
Low Population Zone:				
Whole Body Dose, rem	Not reported	7.6E-2(2)	9.1E-2	≤ 6.25
Thyroid Dose, rem	in UFSAR	0.60(2)	0.78	≤ 75

Notes:

1. Independent analysis performed by NRC indicates that the resulting radiological consequences are less than the acceptance criteria given in SRP 15.4.9.
2. Doses developed to support a proposed license amendment request to delete the scram and isolation function of the Main Steam Line Radiation Monitor as described in a letter from R. M. Krich (ComEd) to U.S. NRC, "Request for an Amendment to Technical Specifications For Elimination of Main Steam Line Radiation Monitor Isolation and Scram Functions," dated December 30, 1999 (Doses documented in UFSAR change package # 98015, 6/00).

Table 9-3

FHA Radiological Consequences

Location	UFSAR	Current	EPU	Limit
Fuel Handling Accident (Single fuel bundle and handling equipment dropped)				
Offsite:				
Whole Body Dose, rem	3.74E-3(1)	(2)	(2)	≤ 25
Thyroid Dose, rem	1.33E-3(1)	(2)	(2)	≤ 300
Exclusion Area:				
Whole Body Dose, rem	(1)	0.156(3)	0.183	≤ 6.25
Thyroid Dose, rem	(1)	3.05(3)	3.84	≤ 75
Low Population Zone:				
Whole Body Dose, rem	(1)	2.03E-2(3)	2.38E-2	≤ 6.25
Thyroid Dose, rem	(1)	0.362(3)	0.456	≤ 75
Control Room:				
Whole Body Dose, rem	Not reported in UFSAR	1.32E-2(3)	1.54E-2	≤ 5
Thyroid Dose, rem		8.09(3)	10.2	≤ 30
Beta Dose, rem		0.491(3)	0.575	≤ 30

Notes:

(1) UFSAR Table 15.7-8 lists doses as a function of distance and meteorological condition. The values are at 1/2 mile under unstable 2 mph wind speed meteorological condition and represent the worst case values reported.

(2) Not evaluated as it is considered historical information.

(3) Doses developed to support proposed conversion to Improved Technical Specifications (ITS) as described in a letter from R. M. Krich (ComEd) to U.S. NRC, "Request for Technical Specifications Changes for Dresden, Units 2 and 3, LaSalle County Station, Units 1 and 2, and Quad Cities Nuclear Power Station, Units 1 and 2, to Implement Improved Standard Technical Specifications," dated March 3, 2000.

10 ADDITIONAL ASPECTS OF EPU

10.1 High Energy Line Break

Operation at the EPU level requires an increase in the steam and feedwater flows. This, in turn, results in a small increase in the mass and energy release rates following high energy line breaks. Evaluation of these piping systems determined that there is no change in postulated break locations.

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

At the EPU RTP level, the mass and energy releases for high energy line breaks outside the primary containment can increase, potentially causing the sub-compartment pressure and temperature profiles to increase. The relative humidity change is negligible. In most cases, the increase in the blowdown rate is small and the resulting profiles are generally bounded by the existing profiles due to the conservatism in the current HELB analyses. The HELBs evaluated are the:

- Main Steam System Line Break;
- Feedwater System Line Break;
- ECCS Line Breaks;
- IC System Line Break;
- RWCU System Line Break; and
- Instrument Line Break.

Pipe Whip and Jet Impingement:

The following addresses the effects of jet impingement from high energy lines, as addressed in UFSAR Appendices 3A and 3B, and pipe whip restraint and pipe break criteria, as addressed in UFSAR Section 3.6.

Existing calculations supporting the dispositions of potential targets of pipe whip and jet impingement from the postulated HELBs were reviewed for the effect of EPU. The review shows that higher loads/piping stresses in main steam and small changes in pressure in the Feedwater line have been evaluated for jet impingement loads and found to be acceptable. For the remaining high energy systems, existing pipe whip, and jet impingement loadings remain bounding for EPU.

Therefore, the existing pipe whip restraints and jet impingement shields, and their supporting structures are adequate for EPU.

Internal Flooding from HELB:

The HELB analysis evaluation for flooding in the main steam tunnel due to a Main Steam and Feedwater pipe break assumes flooding of the entire below grade volume. This analysis approach is conservative and remains bounding for EPU.

The critical parameter affecting the high energy HPCI or Isolation Condenser steam line break analysis relative to EPU is an increase in reactor vessel dome pressure. However, dome pressure does not increase for EPU. Therefore, there is no increase in the blowdown rate, and the previous HELB flooding analyses in the reactor building are bounding for the EPU.

The original analysis for an RWCU HELB in the reactor building determined that the bounding case for maximum total mass release, which could cause flooding, was not the critical flow from the broken pipe, but a case where the flow was not sufficient to initiate automatic break isolation. Therefore, even with a slightly increased RWCU blowdown rate due to EPU, the original analysis is still the bounding case.

10.2 Moderate Energy Line Break

The design basis for Moderate Energy Line Break (MELB) protection features at DNPS is based on system parameters unchanged by EPU. Therefore, MELB is not affected by EPU for DNPS.

10.3 Environmental Qualification

The safety-related electrical equipment environmental qualification documentation was reviewed to assure the existing qualification for the normal and accident conditions expected in the area where the devices are located remain adequate. Conservatisms in accordance with the original qualification program were applied to the environmental parameters and no change is needed for EPU.

The changes (radiation, pressure, temperature and humidity, as applicable) to the environmental conditions of affected safety-related equipment inside and outside containment were evaluated. This evaluation of equipment qualification for EPU conditions identified some equipment potentially affected by EPU conditions. The qualification of this equipment was resolved by refined radiation calculations, by the use of new test data, by evaluating the operational requirements, or by replacement with qualified equipment.

10.4 Required Testing

Compared to the initial startup program, and consistent with the NRC-approved generic EPU guideline, EPU requires only limited subset of the original startup test program. As applicable to this plant's design, testing for EPU is consistent with the generic guideline.

The same performance criteria will be used as in the original power ascension tests, unless they have been replaced by updated criteria since the initial test program. Because neither steam pressure or recirculation flow have been changed for the EPU program, testing of system performance affected by these parameters is not necessary. Vibration testing of the main steam and feedwater piping is necessary, because of the magnitude of the increase in steam and feedwater flows and the limited piping vibration data available from the initial startup.

Steam separator-dryer performance will be monitored during power ascension testing. The test will quantify the moisture carryover from the steam separator-dryer to determine acceptable operational values. Data will be collected and evaluated at pre-EPU 100% power and at each incremental power increase during power ascension.

A summary report of the EPU program will be submitted to the NRC after the completion of the EPU test program. When applicable, the results from the EPU test program will be used to revise the operator training program to more accurately reflect the effects of EPU.

Recirculation Pump Testing:

Vibration testing of the recirculation pumps is not required because there is no change in the maximum core flow for the EPU condition.

10 CFR 50 Appendix J Testing:

The plant 10 CFR 50 Appendix J test program is required by the Technical Specifications and is described in UFSAR Section 6.2. This test program periodically pressurizes the containment (Type A test), the containment penetrations (Type B test), and the containment isolation valves and test boundary (Type C tests) to the calculated peak containment pressure (P_a), and measures leakage. For EPU, P_a changes to 43.9 psig. Therefore, the 10 CFR 50 Appendix J test program will be revised to reflect this calculated peak containment pressure value.

Main Steam Line and Feedwater Piping Flow Induced Vibration Testing:

The piping vibration levels of two large piping systems within containment for each plant will be monitored during initial plant operation at the new EPU operating conditions. The startup vibration test program performed for each unit is expected to show that these piping systems are vibrating at acceptable levels during EPU conditions. The two piping systems that are affected by an EPU that must be monitored for vibrations for each plant are the Main Steam Line system piping and the Feedwater system piping. These two piping systems will be monitored for vibration, because the mass flow rates in these piping systems will increase noticeably during EPU operations. As part of the piping vibration test program, a Test Specification, Test Plan and Procedure, Preliminary Test Report and Final Test Report will be prepared, to properly direct and document each phase of this test program, which will be performed for each unit.

10.5 Individual Plant Evaluation

The plant uses a probabilistic risk/safety assessment (PRA/PSA) to comply with the Individual Plant Evaluation (IPE) requirement. Consistent with Section 5.11.11 of ELTR1 (Reference 1), the plant-specific PRA/PSA was assessed (reviewed) for the effect of EPU. This review concludes that EPU does not introduce any new vulnerability, and thus, EPU has negligible impact on plant risk. The increase in the current Core Damage Frequency (CDF) of 2.61E-06/yr due to EPU implementation is conservatively estimated as 2.4E-7/yr (9% of the current CDF value). The increase in the Large Early Release Frequency (LERF) of 1.44 E-06/yr due to EPU implementation is conservatively estimated as 1.4E-07/yr (10% of the current LERF value). The increase is due to shortened operator response times for certain scenarios, and to a conservatively assumed turbine trip initiating event frequency increase.

10.6 Operator Training and Human Factors

Before EPU operation is initiated, training required to operate the plant at EPU conditions will be provided. 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 EPU conditions, operator actions for transients, accident and special events do not change, because EPU does not change any of the automatic plant safety functions or the nature of the response. However, some of the assumed operator response times are slightly reduced. Training on these scenarios and the changes in response times will be provided.

Data obtained during startup testing will be incorporated into additional training as needed. The classroom training will cover various aspects of EPU including changes to parameters, setpoints, scales, procedures, systems and startup test procedures. The classroom training will be combined with simulator training. The simulator training will include, as a minimum, a demonstration of transients that show the greatest change in plant response at EPU power 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 (i.e., Equipment Qualification, Flow Accelerated Corrosion) to assess age-related component changes. Equipment qualification is addressed in Section 10.3, and flow accelerated corrosion is addressed in Sections 3.5 and 3.11. 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 Other Applicable Requirements

The analysis, design, and implementation of EPU was reviewed for compliance with the current plant licensing basis acceptance criteria and for compliance with new regulatory requirements and operating experience in the nuclear industry. Generic reviews of the BWR EPU program for compliance with regulatory requirements and industry communications were performed, and these reviews identified the issues that are generically evaluated and issues to be evaluated on a plant-unique basis. The applicable plant-unique evaluations have been performed for the subjects addressed below.

All of the issues from the following subjects are either generically evaluated or are evaluated on a plant-specific basis as part of the EPU program. These evaluations conclude that every issue (1) is not affected by EPU, (2) is already incorporated into the generic EPU program, or (3) is bounded by the plant-specific EPU evaluations. The NRC and industry communications evaluated cover the subjects listed below.

Code of Federal Regulations (CFRs)

NRC TMI Action Items

Action Items (Formerly Unresolved Safety Issues)

NRC Regulatory Guides

NRC Generic Letters

NRC Bulletins

NRC Information Notices

NRC Circulars

INPO Significant Operating Reports (applicable to EPU)

GE Services Information Letters

GE Rapid Information Communication Service Information Letters

Other plant-unique items whose previous evaluations could be affected by operation at the EPU level are being reviewed. These are (1) the NRC and industry communications discussed above, (2) the safety evaluations for work in progress and not yet integrated into the plant design, (3) the temporary modifications that could have been reviewed prior to the EPU and still exist after EPU implementation, and (4) the plant emergency operating procedures (EOPs). These items will be reviewed for possible effect by the EPU, and will be found to be either acceptable for EPU, or will be revised to reflect EPU conditions.

11.2 Impact on Technical Specifications

Implementation of EPU with ARTS power and flow dependent limits requires revision of a number of the Technical Specifications (TS). Table 11-1 contains a list of TS items that are changed to implement EPU and ARTS power and flow dependent limits. A brief description of the nature of each change is also provided. The evaluations summarized in this report provide the justifications for these TS changes.

11.3 Environmental Assessment

ARTS power and flow dependent limits are not related to any plant release, and thus, have no environmental impact.

The environmental effects of EPU will be controlled at the same levels as for the current analyses. None of the present limits for plant environmental releases, such as ultimate heat sink temperature or plant vent radiological limits, will be increased as a consequence of EPU. The environment assessment concludes the effects of EPU will be insignificant, because the normal effluents and doses will remain well within 40 CFR 190, 10 CFR 20 and 10 CFR 50, Appendix I limits.

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 DNPS ARTS power and flow dependent limits program is effectively the same program as the Partial ARTS program implemented at the LaSalle County Station units. The LaSalle program is documented in Reference 1, and was approved in Reference 2.

All significant safety analyses and evaluations have been performed, and their results justify an extended power uprate (EPU) of 17% to 2957 MWt.

The ARTS power and flow dependent limits program has the specific objectives of increasing plant operating efficiency, and updating thermal limits requirements and administration. The analyses summarized herein provide the analytical basis for the following changes associated with the ARTS power and flow dependent limits program:

- Implementation of power- and flow-dependent fuel thermal limits to support elimination of the APRM gain and setpoint requirements.
- Maintaining the RBM operability requirements in terms of the measurable core thermal limit performance parameter, MCPR.

11.4.1.1 Modification Summary

An increase in electrical output of a BWR plant is accomplished primarily by generation and supply of higher steam flow for the turbine generator. Most BWR plants, as currently licensed, have an as-designed equipment and system capability to accommodate steam flow rates at least 5% above the original rating. In addition, continuing improvements in the analytical techniques (computer codes and data) based on several decades of BWR safety technology, plant performance feedback, 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 the capability to increase their thermal power ratings of between 5 and 10% without major nuclear steam supply system (NSSS) hardware modifications, and to provide for power increases to 20% with limited hardware modifications, with no significant increase in the hazards presented by the plant as approved by the NRC at the original license stage.

The plan for achieving higher power is to modestly expand the power flow map and increase core flow along standard Maximum Extended Load Line Limit Analysis (MELLLA) flow control lines. However, there is no increase in the maximum recirculation flow limit or operating pressure over the pre-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.

The ARTS improvements provide changes to the APRM system. The reactor limits, instrument setpoints, operability requirement and Technical Specification changes associated with the ARTS improvements are provided in Table 11-1.

The objective of the APRM improvements is to justify removal of the APRM gain and setpoint (trip setdown) requirement. Two licensing areas, which can be impacted by the elimination of the gain and setpoint requirement, are fuel thermal-mechanical integrity and ECCS-LOCA performance.

The following criteria ensure the satisfaction of the applicable licensing requirements, and were applied to demonstrate the acceptability of elimination of the APRM gain and setpoint requirement:

- The Safety Limit MCPR shall not be violated as a result of any AOOs.
- All fuel thermal-mechanical design bases shall remain within the licensing limits described in the GE generic fuel licensing report.
- Peak cladding temperature and maximum cladding oxidation fraction following a LOCA shall remain within the limits defined in 10 CFR 50.46.

The safety analyses used to evaluate the Operating Limit MCPR (OLMCPR), such that the SLMCPR will not be violated and to ensure that the fuel thermal-mechanical design bases are satisfied, are documented in Section 9.2. These analyses also establish the fuel type specific power- and flow-dependent MCPR and LHGR curves for DNPS. The effect on the ECCS-LOCA response due to both the expansion of the power/flow map and the implementation of the ARTS improvement is discussed in Section 4.3.

The following changes result from the ARTS power and flow dependent limits improvement program:

1. Delete the requirement for setdown of the APRM scram and rod blocks.
2. Add new power-dependent MCPR adjustment factors, MCPR(P).
3. Replace K_F with the new flow-dependent MCPR adjustment factors, MCPR(F).
4. Add new power-dependent LHGR adjustment factors, LHGRFAC(P).
5. Add new flow-dependent LHGR adjustment factors, LHGRFAC(F).
6. Delete or modify affected Technical Specifications and Bases.

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 Uprate Analysis Basis

DNPS is currently licensed for a 100% power level of 2527 MWt. The current safety analysis basis assumes that the reactor had been operating continuously at the licensed power level, except for the ECCS-LOCA and short-term containment analyses, which were performed at 102% of licensed thermal power. The EPU increases the rated thermal power (RTP) by 17% of the originally licensed value. The EPU with ARTS power and flow dependent limits safety analyses are based on a power level of at least 1.02 times the EPU power level, except that some analyses are performed at 100% rated power, because the Regulatory Guide 1.49 2% power factor is already accounted for in the analysis methods.

11.4.2.2 Margins

The above EPU analysis basis ensures that the power dependent margins prescribed by the Code of Federal Regulations (CFR) are maintained by meeting the appropriate regulatory criteria. Similarly, design margins specified by application of the American Society of Mechanical Engineers (ASME) design rules are maintained, as are other margin-ensuring criteria used to judge the acceptability of the plant. Environmental margins are maintained by not increasing any of the present limits for releases, such as ultimate heat sink maximum temperature or plant vent radiological limits.

11.4.2.3 Fuel Thermal Limits

No change is required in the basic fuel design to achieve the EPU power level, implement ARTS power and flow dependent limits improvements or to meet the plant licensing limits. No increase in allowable peak bundle power is requested for EPU. The current fuel operating limits will still be met at the EPU power level. 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. No new fuel design is required for EPU with ARTS power and flow dependent limits.

11.4.2.4 Makeup Water Sources

The Boiling Water Reactor 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. Consequently, 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 at least three feedwater and four condensate system pumps, the low pressure emergency core cooling system (LPCI & CS) pumps, the high pressure emergency core cooling system (HPCI) pump, the Standby Liquid Control (SLC) pumps, and the Control Rod Drive (CRD) pumps. Many of these diverse water supply means are redundant in equipment and also redundant in systems (e.g., there are several pumps and complete redundant piping systems).

EPU with ARTS power and flow dependent limits 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 Emergency Core Cooling Systems (ECCS) during loss-of-coolant-accidents.

EPU results in a 17% 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.

11.4.2.5 Design Basis Accidents

Design Basis Accidents (DBAs) are very low probability 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 pipe break sizes in the largest recirculation, steam, and feedwater lines, a postulated break in one of the ECCS lines, and the most limiting small lines. This break range bounds the full spectrum of large and small, high and low energy line breaks; and the success of plant systems to mitigate the accidents, while accommodating a single active equipment failure in addition to the postulated LOCA. Several of the most significant licensing assessments are or will be made using these LOCA ground rules. These assessments are:

- Challenges to Fuel (ECCS-LOCA performance evaluation) in accordance with the rules and criteria of 10 CFR 50.46 and Appendix K wherein the predominant criterion is the fuel peak cladding 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 (cooling) pool temperature for long-term cooling in accordance with 10 CFR 50 Appendix A Criterion 38.
- DBA Radiological Consequences calculated and compared to the criteria of 10 CFR 100.

11.4.2.6 Challenges to Fuel

The ECCS-LOCA performance evaluation (see Section 4.3) was conducted through application of the 10 CFR 50 Appendix K evaluation models, and demonstrates that EPU does not significantly affect the ECCS-LOCA performance evaluation results. The LOCA evaluations with the equilibrium cycle core of GE14 fuel demonstrate compliance with the ECCS acceptance criteria. The licensing safety margin will not be affected by EPU. The slightly (< 10°F) increased PCTs for EPU are insignificant. Therefore, the ECCS safety margin will not be affected by EPU.

The ARTS power and flow dependent limits do not affect ECCS-LOCA performance evaluation.

11.4.2.7 Challenges to the Containment

The effect of EPU on the peak values for containment pressure and temperature confirms the suitability of the plant for operation at EPU power. Also, the effect of 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 EPU are within the range of conditions used to define the current dynamic loads, current safety criteria are met and no further structural analysis is required. The change in short-term containment response is negligible. Because there will be more residual heat with 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.

ARTS power and flow dependent limits do not affect the Containment analysis.

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 atmospheric dispersion factors and the dose exposure pathways do not change. Therefore, the only factor, which could influence the magnitude of the consequences, is the quantity of activity released to the environment. This quantity is a product of the activity released from the core or reactor coolant and the transport mechanisms between the source region and the effluent release point. The transport mechanisms between the source region and the effluent release point are unchanged by EPU.

For EPU, the events evaluated are the Loss-of-Coolant-Accident (LOCA), the Main Steam Line Break Accident (MSLBA) outside containment, the Fuel Handling Accident (FHA), the Control Rod Drop Accident (CRDA), the Instrument Line Break (ILB) and the Offgas Treatment System Component Failure.

The EPU will not change the radiological consequences of a MSLBA outside containment, since the mass and energy releases following a MSLBA remain unaffected by EPU, and the activity released is based on primary coolant at Technical Specification levels, which is also unaffected by EPU.

The EPU will not change the radiological consequences of an ILB outside containment since the reactor coolant mass release used in the current analysis envelopes the post-EPU conditions, and the activity released is based on primary coolant at Technical Specification levels which is unaffected by EPU.

The EPU will not change the radiological consequences of an Offgas Treatment System Component Failure since a conservative source term was used in the original analysis.

For the remaining DBAs, the primary parameter of importance is the activity released from the fuel. Because the mechanism of fuel failure is not influenced by EPU, the only parameter of importance is the actual inventory of fission products in the fuel rod. The only parameters affecting fuel inventory are the increase in thermal power, and to some extent, the cycle length.

The DBA that has historically been limiting from a radiological viewpoint is the LOCA, for which USNRC Regulatory Guide 1.3, or its equivalent, has been applied. For this accident, it is assumed that 100% of the noble gases and 50% of the iodines in the core are released to the primary containment. These release fractions are not influenced by EPU or cycle length. The LOCA dose consequences remain below regulatory guidelines.

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

ARTS power and flow dependent limits do not affect any radiological analysis, and thus, the consequences of all accidents are not affected.

11.4.2.9 Transient Analyses

The effects of plant transients were evaluated (in Section 9.1) 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 are primarily evaluated against the Safety Limit Minimum Critical Power Ratio (SLMCPR). The most limiting transient is slightly more severe when initiated from the EPU RTP level, and results in a slightly larger change in MCPR than that initiated from the current power level. The Operating Limit MCPR is increased appropriately to assure that the SLMCPR is not infringed upon, if any transient is initiated from the EPU RTP level. Plus, the limiting transients are analyzed for each specific fuel cycle. Licensing acceptance criteria are not exceeded. Therefore, the margin of safety is not affected by EPU.

Use of the ARTS related power and flow dependent MCPR limits ensures that the SLMCPR will not be exceeded.

11.4.2.10 Combined Effects

EPU analyses use fuel designed to current NRC-approved criteria and operated within NRC-approved limits to produce more power in the reactor, and thus, increases steam flow to the turbine. NRC-approved design criteria are used to assure equipment mechanical performance at EPU conditions. Scram frequency is minimized by small adjustments to reactor instrumentation. These adjustments are attributed to the small changes in the reactor operating conditions. DBAs are hypothesized to evaluate challenges to the fuel, containment and off-site dose limits. These challenges have been evaluated separately in accordance with extremely conservative regulatory procedures such that the separate effects are more severe than any combined effects. The off-site dose evaluation, in accordance with Regulatory Guide 1.3 and SRP-15.6.5, calculates more severe DBA radiological consequences than the combined effects of the hypothetical LOCA producing the greatest challenge to the fuel and/or containment. That is, the DBA producing the highest PCT and/or containment pressure, does not damage the large amounts of fuel assumed in the off-site dose evaluation. Therefore, the combined effects of the most severe hypothetical LOCA are conservatively bounded by the off-site dose evaluation.

11.4.2.11 Non-LOCA Radiological Release Accidents

All of the other radiological releases discussed in Regulatory Guide 1.70 UFSAR Chapters 11 and 15 are either unchanged because they are not power-dependent, or increase at most by the amount of the EPU. The dose consequences for all of the radiological release accident events are bounded by the "Design Basis Radiological Consequences" events discussed above.

11.4.2.12 Equipment Qualification

Plant Equipment and Instrumentation has been evaluated against the criteria appropriate for EPU. Significant groups/types of the equipment have been justified for EPU by generic evaluations. Some of the qualification testing/justification at the current power level was done at more severe conditions than the minimum required. In some cases, the qualification envelope did not change significantly due to EPU. A process has been developed to ensure qualification of the equipment whose current qualification does not already bound 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 EPU in a manner comparable to that for safety-related NSSS systems/equipment. Generic and plant-specific evaluations justify EPU operation for BOP systems/equipment. Modifications (e.g., turbine modifications) will be made (via 10 CFR 50.59) where needed to fully implement EPU.

11.4.2.14 Environmental Consequences

Except for particulate matter emissions from the cooling towers, the environmental effects of EPU can be controlled below the same limits as for the current power level. The particulate matter limits in the state operating permit will be increased to implement EPU. None of the present ultimate heat sink temperature or plant vent radiological release limits are increased as a result of EPU.

11.4.2.15 Technical Specifications Changes

The Technical Specifications (TS) ensure that plant and system performance parameters are maintained within the values assumed in the safety analyses. That is, the TS parameters (setpoints, 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 TS changes justified by the safety analyses summarized in these reports are listed in Table 11-1. Proper account is taken of inaccuracies introduced by instrument accuracy and calibration accuracy. This assures that the actual plant responses will be less severe than those represented by the safety analysis. Similarly, the TS address equipment operability (availability) and put limits on equipment out-of-service (not available for use) times such that the actual plant can be expected to have at least the complement of equipment available to mitigate abnormal plant events assumed in the safety analyses. Because the safety analyses for EPU with ARTS power and flow dependent limits show that the results are acceptable within regulatory limits, public health and safety is confirmed. TS changes consistent with the EPU power level and the ARTS power and flow dependent limits improvements are made in accordance with methodology already approved for the plant and continue to provide a comparable level of protection as TS 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 DNPS 117% extended power uprate (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.

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

As summarized below, the increase in power level with ARTS power and flow dependent limits improvements discussed herein will not significantly increase the probability or consequences of an accident previously evaluated.

The probability (frequency of occurrence) of Design Basis Accidents occurring is not affected by the increased power level or by the ARTS power and flow dependent limits, because plant equipment still complies with the applicable regulatory and design basis criteria. An evaluation of the BWR probabilistic risk assessments concludes that the calculated core damage frequencies do not significantly change due to EPU or ARTS power and flow dependent limits. Scram setpoints (i.e., equipment settings that initiate automatic plant shutdowns) are established such that there is no significant increase in scram frequency due to uprate. No new challenges to safety-related equipment result from EPU or ARTS power and flow dependent limits.

Radiological release events (accidents) have been evaluated, and shown to meet the guidelines of 10 CFR 100. Therefore, the changes in consequences of hypothetical accidents are in all cases insignificant. The EPU accident evaluation results do not exceed any of their NRC-approved acceptance limits. The spectrum of hypothetical accidents and transients has been investigated, and are shown to meet the plant's currently licensed regulatory criteria. In the area of core design, for example, the fuel operating limits such as Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) and Safety Limit Minimum Critical Power Ratio (SLMCPR) are still met, and fuel reload analyses will show that plant transients meet the criteria accepted by the NRC as specified in NEDO-24011, "GESTAR II." Challenges to fuel (ECCS performance) are evaluated, and shown to still meet the criteria of 10 CFR 50.46 and Appendix K.

ARTS power and flow dependent limits do not affect a radiological analysis result from any postulated accident, nor does it affect the containment analysis.

Challenges to the containment have been evaluated, and the containment and its associated cooling systems continue to meet 10 CFR 50 Appendix A Criterion 38, Long Term Cooling, and Criterion 50, Containment.

Radiological release events (accidents) have been evaluated, and shown to meet the guidelines of 10 CFR 100.

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

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

Equipment that could be affected by EPU or ARTS power and flow dependent limits has been evaluated. No new operating mode, safety-related equipment lineup, accident scenario or equipment failure mode is involved with EPU. The full spectrum of accident considerations, defined in Regulatory Guide 1.70, has been evaluated, and no new or different kind of accident has been identified. EPU and ARTS power and flow dependent limits use already developed technologies, and apply them within the capabilities of already existing plant equipment in accordance with presently existing regulatory criteria. Industry experience with ARTS and BWRs with higher power levels than described herein have not identified any new power dependent or ARTS related accident.

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.

EPU only affects design and operational margins. Challenges to the fuel, reactor coolant pressure boundary, and containment were reanalyzed for EPU conditions. The fuel integrity is maintained by meeting existing design and regulatory limits. The calculated loads of all affected structures, systems and components, including the reactor coolant pressure boundary, remain within design allowables for all design basis event categories. The containment performance analysis demonstrates that the containment remains within all of its design limits following the most severe design basis accident.

The use of ARTS power and flow dependent limits improvements ensures that the plant does not exceed any fuel thermal limit, and thus, the margin of safety is not affected.

Because the plant reactions to transients and hypothetical accidents do not result in exceeding the presently approved NRC acceptance limits, EPU with ARTS power and flow dependent limits does not involve a significant reduction in a margin of safety.

Conclusions:

An EPU to 117% of original rated power with ARTS power and flow dependent limits has been investigated. The method for achieving higher power is to slightly increase some plant operating parameters. The plant licensing challenges have been evaluated and demonstrate how this uprate with ARTS power and flow dependent limits 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 power uprate of the amount described herein and ARTS power and flow dependent limits do not involve a Significant Hazards Consideration.

Table 11-1

Technical Specifications Affected By EPU With ARTS

TS Location	Description of Change
1.1 Definitions	Delete the definition of Fuel Design Limiting Ratio For Centerline Melt (FDLRC), because this definition is no longer applicable with the implementation of the ARTS related changes, discussed in Sections 1.4 and 9.2. Revise the value of Rated Thermal Power (RTP) definition to EPU power level (2957 MWt) shown in Table 1-2.
3.2.4	Delete TS 3.2.4 (entirely), as the APRM Gain and Setpoint requirement are superseded by the ARTS related changes, discussed in Sections 1.4 and 9.2.
SR 3.3.1.1.2	Delete reference to LCO 3.2.4, because TS 3.2.4 is deleted due to ARTS changes.
SR 3.3.1.1.14, Table 3.3.1.1-1 Functions 8 and 9	Reduce the TSV-Closure and TCV Fast Closure scram bypass power level by the ratio of the power increase (1/1.17), from 45% RTP to 38.5%.
3.3.1.1 Required Action E.1	Revised action %RTP value to be consistent with the RPS %RTP Bypass value, i.e., from 45% RTP to 38.5% RTP, to maintain the same absolute thermal power value.
Table 3.3.1.1-1 Function 2.b.	Revise the APRM Flow Biased scram equations for two and single recirculation loop operation, consistent with the discussion in Section 5.3. Revise the allowable value for the APRM TLO clamped scram from 120% RTP to 122% RTP, based on Reference 4.
Table 3.3.1.1-1 Function 2.c	Revise the allowable value for the APRM fixed neutron flux – high from 120% RTP to 122% RTP, based on Reference 4.
Table 3.3.1.1-1 Function 4.	As discussed in Section 5.3, revise the Reactor Vessel Water Level – Low scram Allowable Value from ≥ 10.24 inches to ≥ 2.24 inches, based on the revised analytical limit.
Table 3.3.1.1-1 Function 10.	As discussed in Section 5.3, revise the Turbine Condenser Vacuum – Low scram Allowable Value from ≥ 21.15 inches Hg vacuum to ≥ 21.4 inches Hg vacuum.
Table 3.3.5.1-1, Function 1.e	Revise the allowable value for the core spray pump start time delay relay from ≤ 13.8 seconds to ≤ 11.0 seconds, based on Reference 5.

TS Location	Description of Change
Table 3.3.5.1-1, Function 2.e	Revise the allowable value for the low pressure coolant injection pump start time delay relay – pumps B and D from $\leq$ 8.8 seconds to $\leq$ 5.5 seconds, based on Reference 5.
Table 3.3.6.1-1, Function 1.a	Revise the allowable value for the Reactor Vessel water level – low low function from $\leq$ -56.77 inches to $\leq$ -56.34 inches, based on Reference 6.
Table 3.3.6.1-1 Function 2.a.	As discussed in Section 5.3, revise the Reactor Vessel Water Level – Low primary containment isolation Allowable Value from $\geq$ 10.24 inches to $\geq$ 2.24 inches, based on the revised analytical limit.
Table 3.3.6.1-1 Function 5.b.	As discussed in Section 5.3, revise the Reactor Vessel Water Level – Low Reactor Water Cleanup system isolation Allowable Value from $\geq$ 10.24 inches to $\geq$ 2.24 inches, based on the revised analytical limit.
Table 3.3.6.1-1 Function 1.d.	As discussed in Section 5.3, revise the Main Steam Line Flow – High isolation Allowable Value from $\leq$ 160.6 psid to $\leq$ 259.2 psid for Unit 2, and from $\leq$ 117.1 psid to $\leq$ 252.6 psid for Unit 3, based on the revised analytical limits.
SR 3.3.5.2.2	Consistent with the transient analysis, revise the Isolation Condenser time delay from 17 seconds to 15 seconds.
SR 3.4.3.1	Because the number of valves that can perform the spring safety function includes the one SRV, the associated surveillance should include the SRV. Therefore, to the safety valve listing add a row showing "1" valve with a setpoint of "1135 ± 11.3."
3.5.1	To be consistent with the ECCS-LOCA analysis (Section 4.3), the number of operable relief function valves is increased from four relief valves to four relief valves and one safety/relief valve (SRV).
(New) SR 3.5.1.12	To ensure the operability of the relief function of the Target Rock SRV, add a new surveillance that states "Verify ADS pneumatic supply header pressure is $\geq$ 80 psig." This surveillance to be performed every 31 days. This is based on Reference 5.
5.5.12	Based the containment performance analysis addressed in Section 4.1, revise the "Pa" value to be equal to the peak calculated containment pressure of 43.9 psig, as discussed in Section 10.4.
5.6.5, Item a.4	Delete Item a.4, because it is based on TS 3.2.4, which is deleted due to ARTS related changes.

12 References

1. GE Nuclear Energy, "Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate," (ELTR1), Licensing Topical Report NEDO-32424, Class I (Non-proprietary), April 1995.

2. Letter from Gary G. Benes (Nuclear Licensing Administrator, Commonwealth Edition) to William T. Russell (Director, USNRC), "LaSalle County Nuclear Power Station Units 1 and 2 Application for Amendment Request to Facility Operating Licenses NPF-11 and NPF-18, Appendix A, Technical Specifications Partial ARTS Implementation NRC Docket Nos. 50-373 and 50-374," June 9, 1994.

3. Letter from William D. Reckley (Project Manager, USNRC) to D. L. Farrar (Manager, Commonwealth Edition Company), "Issuance of Amendments (TAC Nos. M89631 and M89632)," April 13, 1995.

4. Letter from R. M. Krich (Exelon Generation Company) to U. S. NRC, "Supplement to Request for License Amendment for Power Uprate Operation," dated April 13, 2001

5. Letter from R. M. Krich (Exelon Generation Company) to U. S. NRC, "Supplement to GE14 Fuel License Amendment Request," dated August 13, 2001

6. Letter from K. A. Ainger (Exelon Generation Company) to U. S. NRC, "Supplement to Request for License Amendment for Power Uprate Operation," dated August 29, 2001

Attachment B
Safety Analysis Reports Supporting the License Amendment Request to Permit Uprated Power Operation
Dresden Nuclear Power Station, Units 2 and 3,
Quad Cities Nuclear Power Station, Units 1 and 2

GE Report NEDO-32961, Revision 1, "Safety Analysis Report for Quad Cities 1 & 2 Extended Power Uprate," August 2001 (Non-Proprietary)

175 Curtner Ave., San Jose, CA 95125

GE Nuclear Energy

NEDO-32961, Revision 1
DRF A22-00103-13
Class I
August 2001

SAFETY ANALYSIS REPORT FOR QUAD CITIES 1 & 2 EXTENDED POWER UPRATE

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IMPORTANT NOTICE REGARDING

CONTENTS OF THIS REPORT

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EXECUTIVE SUMMARY

This report summarizes the results of all significant safety evaluations performed that justify extending the licensed thermal power at Quad Cities Units 1 and 2 to 2957, MWt. The requested license power level is approximately 117.8% of the current licensed rating of 2511 MWt.

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

A higher steam flow is achieved by increasing the reactor power along slightly revised rod and core flow control lines. A limited number of operating parameters are changed. Some setpoints are adjusted and instruments are recalibrated. Plant procedures are revised and tests similar to some of the original startup tests are performed. Modifications to some power generation equipment may be implemented over time, as needed.

Detailed evaluations of the reactor, engineered safety features, power conversion, emergency power, support systems, environmental issues, design basis accidents, and previous licensing evaluations were performed. This report demonstrates that Quad Cities can safely operate at the requested license power level of 2957 MWt. However, power generation modifications must be implemented in order to obtain the electrical power output associated with 100% of the EPU power level. Until these modifications are completed, the balance of plant may limit the electrical power output, which (in-turn) limits the operating thermal power level to less than the licensed power level.

The predominant plant licensing challenges have been reviewed, and it is concluded that this uprate can be accommodated (1) without a significant increase in the probability or consequences of an accident previously evaluated, (2) without creating the possibility of a new or different kind of accident from any accident previously evaluated, and (3) without exceeding any existing regulatory limits applicable to the plant which might cause a significant reduction in a margin of safety. Therefore, the requested EPU does not involve a significant hazards consideration.

1 OVERVIEW

1.1 Introduction

Uprating the power level of nuclear power plants can be done safely within certain plant-specific limits. Most GE BWR plants have the capability and margins for an uprating of 5 to 20% without major nuclear steam supply system (NSSS) hardware modifications. Many light water reactors have already been uprated worldwide. Over a thousand MWe have already been added by uprate in the United States. Several BWR plants are among those that have already been uprated. This evaluation justifies an EPU to 2957 MWt, corresponding to 117.8% of the current rated thermal power, for both Quad Cities Units 1 and 2. The original licensed thermal power is 2511 MWt.

The ARTS program is designed to increase plant operating efficiency by updating the thermal limits requirements. The APRM trip setdown (gain and setpoint) requirement is replaced by the ARTS power-dependent and flow-dependent thermal limits to reduce the need for manual setpoint adjustments and to allow more direct thermal limits administration. This change updates thermal limits administration, increases reliability, and provides better protection.

The ARTS-based thermal limits are specified for fuel protection during Anticipated Operational Occurrences (AOOs). The plant-specific portions of these generic ARTS limits were developed based on a representative core configuration.

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 at least 5% above the original rating. In addition, continuing improvements in the analytical techniques (e.g., computer codes) based on several decades of BWR safety technology, plant performance feedback, and improved fuel and core designs have resulted in a significant increase in the difference between the calculated safety analyses results and the licensing limits. The plant-specific uprate parameters are listed in Table 1-2.

Each unit is currently licensed at 2511 MWt, and most of the current safety analyses are based on this value. However, the ECCS-LOCA and Containment safety analyses are based on a power level of 1.02 times the licensed power level. The uprate power level included in this evaluation is a 17.8% (2957 MWt) thermal EPU of the currently licensed value. The EPU safety analyses are based on a power level of at least 1.02 times the EPU power level (1.02 x 2957 = 3016 MWt), except that some analyses are performed at 100% uprated power, because the Regulatory Guide 1.49 two percent power factor is already accounted for in the analysis methods.

The extended power uprate analysis basis assures that the power-dependent safety margin prescribed by the Code of Federal Regulations (CFR) will be maintained by meeting the appropriate regulatory criteria. NRC-accepted computer codes and calculational techniques are used to make the calculations that demonstrate meeting the stipulated criteria.

The major EPU analyses for Quad Cities and Dresden were performed using bounding parameters. This allows one evaluation to be performed that envelops all four units. The bounding value of each parameter was obtained by comparing the parameter across the four units and selecting the most limiting value. Therefore, the evaluation results in this report are conservative, and consequently, the actual operating values for any given unit may differ from the values shown herein.

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 without an increase in reactor operating pressure, (2) a corresponding increase in the feedwater system flow, (3) no increase in maximum core flow, and (4) reactor operation primarily along accepted rod/flow control lines. Plant-unique evaluations were based on a review of plant design and operating data to confirm excess design capabilities. The results of these evaluations are presented in the subsequent sections of this report.

1.3 EPU Plant Operating Conditions

The thermal hydraulic performance of a BWR reactor core is characterized by the total 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. They are determined by performing heat balance calculations for the reactor system at EPU conditions.

The EPU heat balance was determined such that the core thermal power is 117.8% of the current licensed core thermal power and the steam flow from the vessel was increased to approximately 120% of 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 power and 100% rated core flow.

Table 1-2 shows a summary of the reactor thermal-hydraulic parameters for the current rated condition and EPU conditions.

The UFSAR, core fuel reload evaluations, and/or the Technical Specifications currently include allowances for plant operation with the performance improvement features and the equipment out-of-service listed in Table 1-2. When limiting, the input parameters related to the performance improvement features or the equipment out-of-service have been included in the safety analyses for EPU. The use of these performance improvement features and allowing for equipment out-of-service is continued during EPU power operation. Where appropriate, the evaluations performed for uprate account for a 24 month fuel cycle length.

1.4 ARTS Power and Flow Dependent Limits

The ARTS improvements provide changes to the APRM system. An overview of the improvements are discussed below along with the identification of the evaluations necessary to support these improvements. The Technical Specifications (TS) change(s) associated with the ARTS improvements are provided in Table 11-1

The plant TS require that the flow-referenced APRM trips be lowered (setdown) when the core Maximum Total Peaking Factor (MTPF) exceeds the design Total Peaking Factor (TPF). The basis for this "APRM trip setdown" requirement originated under the previous Hench-Levy Minimum Critical Heat Flux Ratio (MCHFR) thermal limit criterion.

The change to the General Electric Thermal Analysis Basis critical power correlation, with its de-emphasis of local thermal hydraulic conditions, and the move to secondary reliance on flux scram for licensing basis anticipated operational occurrence (AOO) evaluations (for events terminated by anticipatory or direct scram) provides more effective and operationally acceptable alternatives to the setdown requirement. The ARTS program utilizes results of the AOO analyses to define initial condition operating thermal limits which conservatively ensure that all licensing criteria are satisfied without setdown of the flow-referenced APRM scram and rod block trips.

The objective of the APRM improvements is to justify removal of the APRM trip setdown requirement (APRM Gain and Setpoint TS). Two licensing areas, which can be affected by the elimination of the APRM Gain and Setpoint TS, are fuel thermal-mechanical integrity and loss-of-coolant accident (LOCA) analysis.

The (applicable) safety analyses used to evaluate the Operating Limit MCPR (OLMCPR), such that the SLMCPR is not violated, and to ensure that the fuel thermal-mechanical design bases are satisfied, are documented in Section 9.2. These analyses also establish the fuel type specific power- and flow-dependent limits for Quad Cities. The effect on the ECCS-LOCA response due to both the expansion of the power/flow map and the implementation of the ARTS improvement is discussed in Section 4.3.

The following changes result from the implementation of ARTS power and flow dependent limits:

1. Delete the requirement for setdown of the APRM scram and rod blocks.

2. Add new power-dependent MCPR adjustment factors, MCPR(P).

3. Replace the flow-dependent MCPR limits with the new flow-dependent MCPR adjustment factors, MCPR(F).

4. Add new power-dependent LHGR adjustment factors, LHGRFAC(P).

5. Add new flow-dependent LHGR adjustment factors, LHGRFAC(F).

6. Delete or modify affected TS and Bases.

1.5 Summary And Conclusions

The predominant plant licensing challenges have been reviewed to demonstrate how this uprate can be accommodated (1) without a significant increase in the probability or consequences of an accident previously evaluated, (2) without creating the possibility of a new or different kind of accident from any accident previously evaluated, and (3) without exceeding any existing regulatory limits or design allowable limits applicable to the plant which might cause a reduction in a margin of safety. The EPU described herein involves no significant hazard consideration.

Table 1-1

Glossary of Terms

Term	Definition
AC	Alternating current
ADS	Automatic Depressurization System
ADHR	Alternate Decay Heat Removal
AL	Analytical Limit
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOO	Anticipated operating occurrences (moderate frequency transient events)
AP	Annulus pressurization
APCVS	Augmented Primary Containment Venting System
APRM	Average Power Range Monitor
ARO	All rods out
ARTS	APRM/RBM/Technical Specifications
ASME	American Society of Mechanical Engineers
AST	Alternative Source Term
ATWS	Anticipated Transient Without Scram
AV	Allowable Value
BHP	Brake horse power
BOP	Balance-of-plant
BWR	Boiling Water Reactor
BWROG	BWR Owners Group
CAM	Containment Atmosphere Monitoring
CCT	Critical Clearing Time
CD	Condensate demineralizers
CDF	Core Damage Frequency
CFR	Code of Federal Regulations
CGCS	Combustible Gas Control System
CO	Condensation oscillation
COLR	Core Operating Limits Report
CPD	Condensate polishing demineralizer
CPR	Critical power ratio
CRD	Control Rod Drive
CRDA	Control Rod Drop Accident
CRGT	Control Rod Guide Tube
CSC	Containment Spray Cooling

CST	Condensate Storage Tank
CS	Core Spray
DAR	Design Assessment Report
DBA	Design basis accident
DC	Direct current
DG	Diesel generator
DGCW	Diesel Generator Cooling Water
DL	Discharge line
ECCS	Emergency Core Cooling System
EDG	Emergency diesel generators
EECW	Emergency Equipment Cooling Water
EESW	Emergency Equipment Service Water
EFPY	Effective full power years
EHL	Emergency Heat Load
EHC	Electro-hydraulic control
EGC	Economic generation control
ELLL	Extended Load Line Limit
EOC	End of cycle
EOOS	Equipment out-of-service
ELTR	Extended power uprate licensing topical report
EOP	Emergency Operating Procedure
EPP	Environmental Protection Plan
EPU	Extended power uprate
EQ	Environmental qualification
ER-OL	Environmental Report-Operating License stage
ESW	Emergency Service Water
FAC	Flow Accelerated Corrosion
FCS	Feedwater Control System
FCV	Flow Control Valve
FES	Final Environmental Statement
FFRO	Fast Flow Runout
FFWTR	Final Feedwater Temperature Reduction
FHA	Fuel Handling Accident
FWCF	Feedwater controller failure
FWHOOS	Feedwater heater(s) out-of-service
FPCC	Fuel Pool Cooling and Cleanup
FSAR	Final Safety Analysis Report
GE	General Electric Company

HD	Heater Drains
HX	Heat exchanger
HCU	Hydraulic Control Unit
HEI	Heat Exchange Institute
HELB	High Energy Line Break
HPCI	High Pressure Coolant Injection
HPSP	High power setpoint
HVAC	Heating Ventilating and Air Conditioning
ICA	Interim Corrective Actions
ICF	Increased Core Flow
IEB	Inspection and Enforcement Bulletin (original NRC title)
IEC	Information and Enforcement Circular (original NRC title)
IEEE	Institute of Electrical and Electronics Engineers
IEN	Inspection and Enforcement Notice (original NRC title)
IGSCC	Intergranular stress corrosion cracking
ILBA	Instrument Line Break Accident
IRM	Intermediate Range Monitor
JR	Jet reaction
LBB	Local Breaker Backup
LCO	Limiting Conditions for Operation
LCS	Leakage Control System
LERF	Large Early Release Frequency
LFA	Lead Fuel Assemblies
LHGR	Linear Heat Generation Rate
LHGRFAC(F)	Flow-dependent LHGR adjustment factor
LHGRFAC(P)	Power-dependent LHGR adjustment factor
LOCA	Loss-Of-Coolant Accident
LOFW	Loss of feedwater
LOOP	Loss of offsite power
LPCI	Low Pressure Coolant Injection
LPRM	Local Power Range Monitor
LPSP	Low Power Setpoint
LRNBP	Load Rejection with no Bypass
LTR	Licensing Topical Report
LUA	Lead use assembly
MAPLHGR	Maximum Average Planar Linear Heat Generation Rate
MBTU	Millions of BTUs
MCC	Motor Control Circuit/Center

MCPR	Minimum Critical Power Ratio
MCPR(F)	Flow-dependent MCPR adjustment factor
MCPR(P)	Power-dependent MCPR adjustment factor
MCHFR	Minimum Critical Heat Flux Ratio
MELB	Moderate Energy Line Break
MELLLA	Maximum Extended Load Line Limit Analysis
MEOD	Maximum Extended Operating Domain
MeV	Million Electron Volts
MG	Motor generator
MHC	Mechanical-Hydraulic Control
Mlb	Millions of pounds
MLHGR	Maximum Linear Heat Generation Rate
MOV	Motor operated valve
MSIV	Main Steam Isolation Valve
MS	Main steam
MSLB	Main steam line break
MSLBA	Main Steam Line Break Accident
MSR	Moisture Separator Reheater
MTPF	Maximum Total Peaking Factor
MWt/MWth	Megawatt-thermal
MSL	Main steam line
MVA	Million Volt Amps
MWe	Megawatt-electric
NCAD	Nitrogen Containment Atmosphere Dilution
NCCW	Nuclear Closed Cooling Water
NPSH	Net positive suction head
NRC	Nuclear Regulatory Commission
NSSS	Nuclear steam supply system
NTSP	Nominal Trip Setpoint
NUREG	Nuclear Regulations
OFS	Orificed fuel support
OLMCPR	Operating Limit Minimum Critical Power Ratio
OLTP	Original Licensed Thermal Power
OOS	Out-of-service
PCS	Pressure Control System
PCT	Peak cladding temperature
PF	Power Factor
PRA	Probabilistic Risk Assessment

PSA	Probabilistic Safety Assessment
psi	Pounds per square inch
psia	Pounds per square inch - absolute
psid	Pounds per square inch - differential
psig	Pounds per square inch - gauge
PULD	Plant-Unique Load Definition
PWR	Pipe Whip Restraint
OCNPS	Quad Cities Nuclear Power Station
RBCCW	Reactor Building Closed Cooling Water
RBM	Rod Block Monitor
RCIC	Reactor Core Isolation Cooling
RCPB	Reactor Coolant Pressure Boundary
REM	Roentgen Equivalent Man (radiation dose measurement)
RFP	Reactor feed pump
RHR	Residual Heat Removal
RHRSW	Residual Heat Removal Service Water
RICSIL	Rapid Information Communication Service Information Letter
RIPD	Reactor internal pressure difference
RLB	Recirculation Line Break
RPCS	Rod Pattern Control System
RPS	Reactor Protection System
RPT	Recirculation Pump Trip
RPV	Reactor Pressure Vessel
RR	Reactor recirculation
RSLB	Recirculation system line break
RTP	Rated Thermal Power
RT_{NDT}	Reference temperature of nil-ductility transition
RV	Relief valve
RWCU	Reactor Water Cleanup
RWE	Rod Withdrawal Error
RWL	Rod Withdrawal Limiter
RWM	Rod Worth Minimizer
SAR	Safety Analysis Report
SBO	Station blackout
SCM	Steam condensing mode
SDC	Shutdown Cooling
SE	Safety Evaluation
SER	Safety Evaluation Report

SGTS	Standby Gas Treatment System
SIL	Services Information Letter
SJAE	Steam Jet Air Ejectors
SLCS	Standby Liquid Control System
SLMCPR	Safety Limit Minimum Critical Power Ratio
SLO	Single-loop operation
SORV	Stuck open relief valve
SPCM	Suppression pool cooling mode
SRM	Source Range Monitor
SRP	Standard Review Plan
SRV	Safety relief valve
SRVDL	Safety relief valve discharge line
SSV	Spring Safety Valve
SW	Service water
TAF	Top of active fuel
TB	Turbine bypass
TBCCW	Turbine Building Closed Cooling Water System
TCV	Turbine control valve
TFSP	Turbine first stage pressure
TG	Turbine generator
TGT	Turbine Generator Trip
TIP	Traversing In-Core Probe
TLO	Two (recirculation) loop operation
TPF	Total Peaking Factor
TPM	Thermal Power Monitor
TS	Technical Specifications
TSV	Turbine Stop Valve
TTNBP	Turbine Trip – no Bypass
UFSAR	Updated Final Safety Analysis Report
UHS	Ultimate heat sink
USE	Upper shelf energy
VWO	Valves wide open

Table 1-2

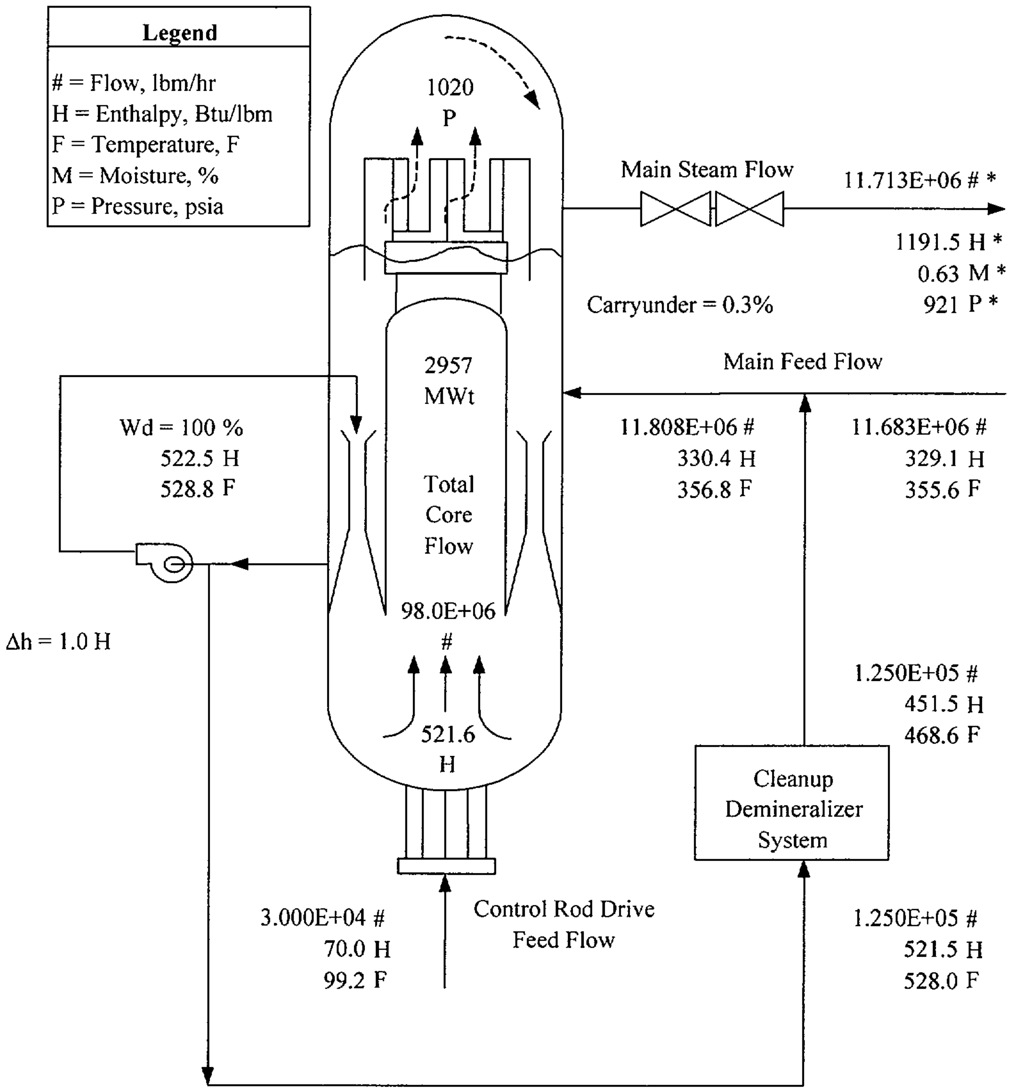
Current And Extended Uprate Plant Operating Conditions

Parameter	**Current Rated Power Value**	**Extended Power Uprate Value**
Thermal Power (MWth)	2511	2957
Vessel Steam Flow (Mlb/hr) *	9.76	11.71
Full Power Core Flow Range		
Mlb/hr	85.3 to 105.8	94.4 to 105.8
% Rated	87 to 108	95.3 to 108
Dome Pressure (psig)	1005	No change
Dome Temperature (°F)	547.0	No change
Turbine Inlet Pressure (psig)	939.0	906.0
Full Power Feedwater		
Flow (Mlb/hr) *	9.73	11.68
Temperature Range (°F)	340 to 240	356 to 256
Core Inlet Enthalpy (Btu/lb) *	523.7	521.6

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

Performance improvement features and/or equipment out-of-service included in EPU evaluations:

(1) Maximum Extended Load Line Limit Analysis (MELLLA)
(2) End-of-Cycle (EOC) Coastdown
(3) Single Loop Operation (SLO)
(4) Final Feedwater Temperature Reduction (FFWTR)
(5) Increased Core Flow (ICF)
(6) ARTS power and flow dependent limits



* Conditions at upstream side of TSV

Core Thermal Power	2957.0
Pump Heating	9.6
Cleanup Losses	-2.6
Other System Losses	-1.0
Turbine Cycle Use	2963.0 MWt

Figure 1-1. **Extended Power Uprate Heat Balance - Nominal**
(@ 100% Power and 100% Core Flow)

2 REACTOR CORE and FUEL PERFORMANCE

2.1 Fuel Design and Operation

EPU increases the average power density proportional to the power increase. However, this average power density is still within the current operating power density range of most other BWRs. EPU has some effects on operating flexibility, reactivity characteristics and energy requirements. The power distribution in the core is changed to achieve increased core power, while limiting the absolute power in any individual fuel bundle to within its allowable value.

At current or uprated conditions, all fuel and core design limits continue to be met by planned deployment of fuel enrichment and burnable poison. This is supplemented by core management control rod pattern and/or core flow adjustments. New fuel designs are not needed for EPU to ensure safety.

The subsequent reload core designs for operation at the EPU power level will ensure acceptable differences between the licensing limits and their corresponding operating values. Cycle-specific analyses will evaluate all fuel types in each reload core.

2.2 Thermal Limits Assessment

Operating thermal limits ensure that regulatory and/or safety limits are not exceeded for a range of postulated events (e.g., transients, LOCA). Cycle-specific core reload evaluations will evaluate the effects on any other fuel types that remain in the core. Both units have identical system geometry, reactor protection system configuration, mitigation functions, and similar thermal hydraulic and transient behavior characteristics. Cycle-specific core configurations, evaluated for each reload, confirm EPU capability, and establish or confirm cycle-specific limits, as is currently the practice.

Thermal limits management with ARTS power and flow dependent limits is described in Section 9.2.

2.3 Reactivity Characteristics

In the representative core evaluation, all minimum shutdown margin requirements apply to cold conditions ($\leq 212^{\circ}$F), and are maintained without change.

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. Technical Specifications cold shutdown margin requirements are not affected.

The uprated power/flow operating map (Figure 2-1) includes the operating domain changes for EPU power and the plant performance improvement features addressed in Section 1.3. The ARTS power and flow dependent limits analyses (Section 9.2) are in part based on Figure 2-1. The changes to the power/flow operating map are consistent with the previously NRC-approved generic descriptions. The maximum thermal operating power and maximum core flow shown on

Figure 2-1 correspond to the EPU power and the previously analyzed core flow range when rescaled so that EPU power is equal to 100% rated. The power/flow operating map changes incorporated into Figure 2-1 are consistent with the changes shown in Figure 5-1 of ELTR1.

For SLO, the maximum achievable power state point is assumed to be 70.2% uprated power (2076 MWth) at 55.1% flow (54 Mlb/hr).

2.4 Stability

Quad Cities is currently operating under the requirements of reactor stability Interim Corrective Actions (ICAs) and is in the process of implementing reactor stability Long-Term Solution Option III. However, EPU is scheduled to be implemented prior to arming the Option III solution (it is not considered to be fully implemented until the trip system is armed). Therefore, the effect of EPU is addressed on both the ICAs and on the stability Option III solution.

An evaluation determined the effect of EPU on core stability ICAs for EPU, to assure adequate level of protection against the occurrence of a thermal-hydraulic instability. The instability exclusion region boundaries are unchanged with respect to absolute power level (MWt).

Quad Cities is implementing long term stability Option III. The Option III solution monitors Oscillation Power Range Monitor (OPRM) signals to determine when a reactor scram is required to terminate an instability event. The OPRM signal is evaluated by the Option III stability algorithms to determine when the signal is becoming sufficiently periodic and large to warrant a reactor scram to disrupt the oscillation.

ARTS power and flow dependent MCPR limits are used when confirming MCPR Safety Limit protection.

2.5 Reactivity Control

The 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. The CRD system has been generically evaluated. These generic evaluations conclude that the CRD systems for BWR/2-6 are acceptable for EPUs as high as 20% above the original licensed rated power. A confirmatory evaluation was performed for this EPU. The Quad Cities CRD system is consistent with the generic evaluations, and is acceptable for EPU.

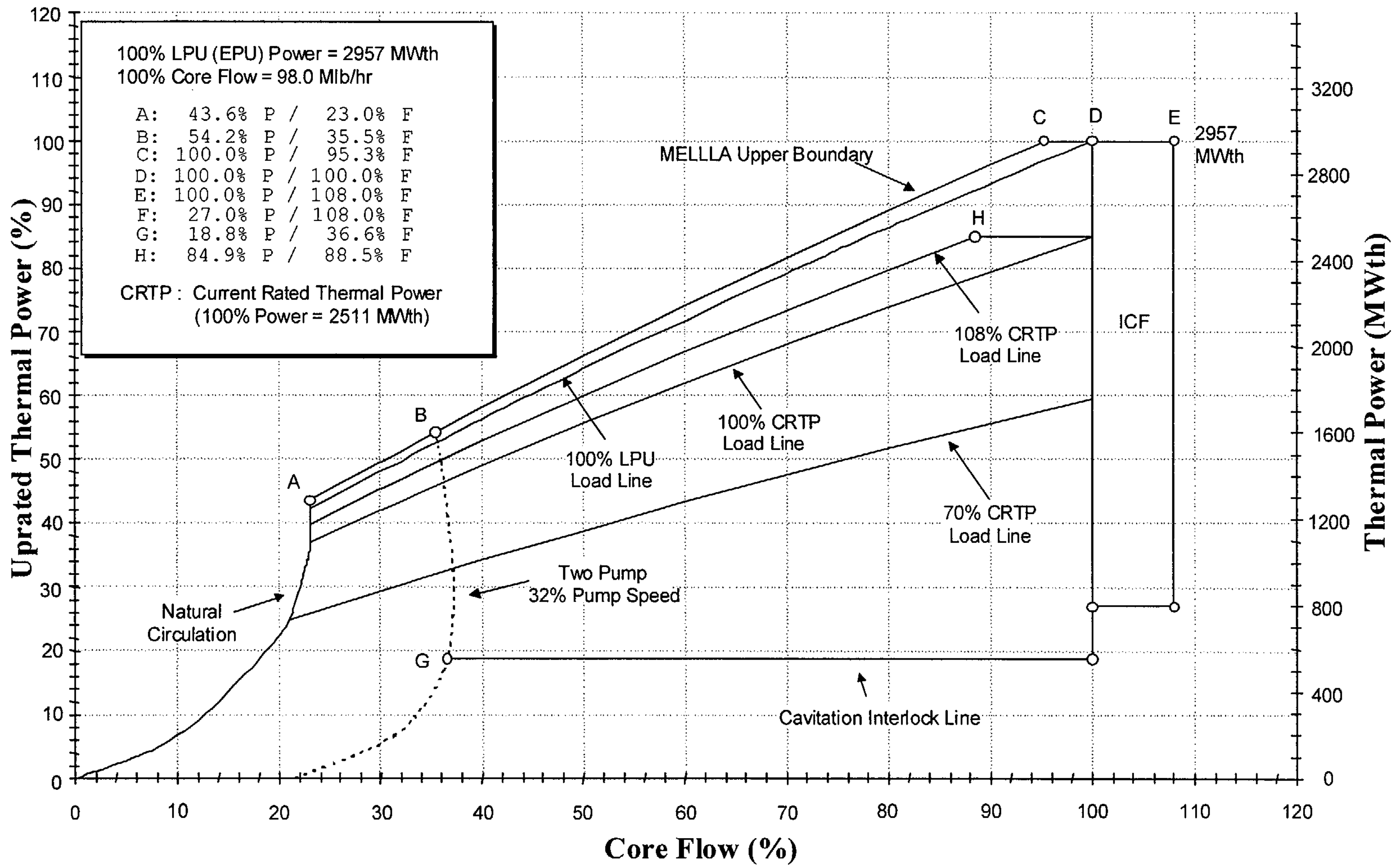


Figure 2-1. Power/Flow Operating Map for EPU

3 REACTOR COOLANT SYSTEM and CONNECTED SYSTEMS

3.1 Nuclear System Pressure Relief

The primary purpose of the nuclear system pressure relief is to prevent overpressurization of the nuclear system during abnormal operational transients. Each unit uses eight spring safety valves (SSVs), four relief valves (RVs) and a single safety/relief valve (SRV) together with the reactor scram function to provide this protection. The SSV, RV, and SRV setpoints are not changed with EPU.

The RVs were originally sized to prevent actuation of the SSVs by relieving the vessel pressure following a turbine stop valve closure coincident with failure of the turbine bypass system. However, with EPU, the RVs are not capable of preventing SSV actuation for an infrequent event such as a turbine trip without bypass. The RVs have the capacity to remove the generated steam and prevent SSV actuation for frequent events like the turbine trip with bypass. Therefore, the RV sizing basis changes with EPU.

SRV setpoint tolerance is independent of EPU. EPU evaluations are performed using the existing SRV setpoint tolerance analytical limits as a basis.

3.2 Reactor Overpressure Protection Analysis

The design pressure of the reactor vessel and reactor coolant pressure boundary (RCPB) remains at 1250 psig. The acceptance limit for pressurization events is the ASME code allowable peak pressure of 1375 psig (110% of design value). The limiting pressurization events are conservatively analyzed, and assume that the events initiate at a reactor dome pressure of 1005 psig and one SRV out-of-service (OOS). The peak calculated RPV pressure remains below the 1375 psig ASME limit, and the maximum calculated dome pressure remains below the Technical Specification 1345 psig Safety Limit. Therefore, there is no decrease in margin of safety.

3.3 Reactor Vessel and Internals

Comprehensive reviews have assessed the effects of increased power conditions on the reactor vessel and its internals. These reviews and associated analyses show continued compliance with the original design and licensing criteria for the reactor vessel and internals.

RPV embrittlement is caused by neutron exposure of the wall adjacent to the core (the "beltline" region). EPU operation may result in a higher neutron flux, which may increase the integrated fluence at the RPV wall over the period of plant license. Because the pre-EPU fluence value bounds the fluence calculated for EPU, the pre-EPU fluence value is used for the EPU evaluations, which demonstrate that the vessels comply with regulatory requirements, and operation with EPU does not have an adverse effect on the reactor vessel fracture toughness.

The effect of the EPU was evaluated to ensure that the reactor vessel components continue to comply with the existing structural requirements of the ASME Boiler and Pressure Vessel Code.

For the components under consideration, Section III, Nuclear Vessels 1965 Edition is the code of construction.

However, if a component underwent a design modification, the governing code for that component was the code used in the stress analysis of the modified component. Typically, new stresses are determined by scaling the "original" stresses, based on EPU conditions (pressure, temperature and flow). The analyses were performed for the design, the normal and upset, and the emergency and faulted conditions. If there is an increase in annulus pressurization, jet reaction, pipe restraint or fuel lift loads, the changes are considered in the analysis of the components affected for upset, emergency and faulted conditions.

The increase in core average power results in higher core loads and reactor internal pressure differences (RIPDs) due to the higher core exit steam flow. The recalculated core loads and RIPD for EPU increase relative to the previous RIPD analyses because of the increase in the thermal power and the consideration of a new core configuration of GE14 fuel. The RIPDs were calculated for normal steady-state operation, upset and faulted conditions for all major reactor internal components, and determined to be acceptable.

A reactor internals structural evaluation of the key reactor internal components was performed to assess the structural integrity for the load changes associated with EPU. This evaluation demonstrates that the structural integrity of the core support and non-core support structure reactor internal components is maintained in the EPU operating condition, consistent with the design basis.

The results of an EPU vibration evaluation show that operation up to 2957 MWt and 108% of rated core flow is possible without any detrimental effects on the safety-related reactor internal components.

Other than structural integrity, the steam separators and dryer do not perform a safety-related function. A plant-specific performance evaluation determined that the steam separators and dryer are capable of performing their operational design function at the increased power level. However, EPU conditions result in an increase in saturated steam generated in the reactor core. For constant core flow, this in turn results in an increase in the separator inlet quality and dryer face velocity and a decrease in the water level inside the dryer skirt, all of which affect the steam separator-dryer performance. The results of the evaluation demonstrate that the steam separator-dryer performance remains acceptable up to some portion of extended power prior to any substantive hardware modification. To reduce the moisture content, hardware modifications are required. These modifications will be completed before EPU implementation. Steam moisture content will be monitored during initial EPU startup testing to determine an acceptable operational moisture content.

3.4 Reactor Recirculation System

The evaluation of the reactor recirculation system performance at EPU conditions determined that adequate core flow can be maintained. Therefore, EPU power operation is within the capability of the reactor recirculation system.

3.5 Reactor Coolant Pressure Boundary Piping

Operation at EPU changes the conditions experienced by the reactor coolant pressure boundary (RCPB). The effects of EPU have been evaluated using the present code(s) of record. In addition to changes in mechanical loading, piping thickness values of carbon steel components can be affected by flow-accelerated corrosion (erosion/corrosion). Erosion/corrosion rates may be influenced by EPU changes in fluid velocity, temperature, and moisture content within carbon steel piping water systems. For systems with an increase in flow rates, vibration can also be induced or aggravated.

The Reactor Recirculation (RR) system evaluated for compliance with the ANSI B31.1 and applicable criteria for the effects of thermal expansion. A review of the slight change in temperature associated with EPU indicates that RR piping load changes do not result in load limits being exceeded for the RR piping system or for interfacing RPV nozzles, penetrations, flanges or valves. No new postulated pipe break location was identified. The piping load changes do not result in any load limit being exceeding the load limit for any piping snubber, hanger, strut or pipe whip restraint.

The RR system components are made of stainless steel, and system flow does not increase for EPU. Therefore, erosion/corrosion concerns are not applicable to this system.

The Main Steam (MS) and Feedwater (FW) systems experiences increases in flow by approximately 20% due to EPU. The MS and FW piping systems (inside containment) were evaluated for the increases in related loads. The piping load changes do not result in load limits being exceeded for the MS or FW piping system or for interfacing RPV nozzles, penetrations, flanges or valves. The original piping design has sufficient design margin to justify adequacy at EPU conditions. No new postulated pipe break location is identified.

The MS piping was evaluated using conservative bounding increases for the effects of load increases related to higher flow rates on supporting snubbers, hangers, struts and pipe whip restraints. This review indicates that the original design analyses do not in every location include sufficient margin to accommodate the higher loads. More detailed analyses demonstrate that the design is adequate for operation at EPU conditions. Minor modifications to pipe support components or support structures are required and will be completed prior to EPU implementation.

The FW piping system was evaluated for the effects of the system condition changes on the supporting snubbers, hangers and struts. This review indicates that the existing design is adequate for EPU conditions and that piping load changes do not result in the load limit of any supporting member being exceeded.

Because piping thickness values of MS & FW carbon steel piping can be affected by flow-accelerated corrosion (erosion/corrosion), and because flow-accelerated corrosion is affected by changes in fluid velocities, temperatures and moisture content, flow-accelerated corrosion effects were evaluated for the carbon steel piping applications within the RCPB.

The integrity of high energy piping systems is assured by proper design in accordance with the applicable Codes and Standards. A consideration in assuring proper design and maintaining system operation within the design is the allowable piping thickness values. The plant has an established program for monitoring pipe wall thinning in single-phase high energy carbon steel piping. The effects of EPU will be incorporated into the existing program.

The adequacy of the other RCPB piping designs for operation at EPU conditions has been evaluated. The nominal operating pressure and temperature of the reactor are not changed by EPU. Aside from MS and FW, no other system connected to the RCPB experiences an increased flow rate at EPU conditions. Only minor changes to fluid conditions will be experienced by these systems due to higher steam flow from the reactor and the subsequent change in fluid conditions within the reactor. Additionally, dynamic piping loads for RV and SRV at EPU conditions are bounded by those used in the existing analyses. These effects have been evaluated for the RCPB portion of the RPV bottom head drain line, RPV head vent line, Reactor Core Isolation Cooling piping, Residual Heat Removal piping, Core Spray, High Pressure Coolant Injection piping, RV+SRV discharge piping and Reactor Water Cleanup piping, as required.

These other systems were evaluated for compliance with the ANSI B31.1 or ASME Code stress criteria (as applicable). Since none of these piping systems experience any significant change in operating conditions, they are all acceptable as currently designed.

Of these other systems, only the RWCU system has load changes significant enough to require evaluation. The effects of thermal expansion displacements on the supporting snubbers, hangers and struts were reviewed and determined not to result in any load limit being exceeded. Therefore, the existing design is adequate for EPU.

These other systems were evaluated during the development of the plant's flow-accelerated corrosion program, to determine their susceptibility to the affects of flow accelerated corrosion. EPU only slightly changes the inlet temperature to the RWCU system, and does not change any operating parameter of the other RCPB systems listed above. Therefore, the flow accelerated corrosion potential within any of these systems is not expected to change.

The safety-related MS piping and the safety-related FW piping will have increased flow rates and flow velocities in order to accommodate EPU. The MS and FW piping will experience increased vibration levels. Other piping systems are not affected. The ASME code requires some vibration test data be taken and evaluated per the nuclear regulatory guidelines for these high energy piping systems, when initially operated at EPU conditions. Vibration data for the MS and FW piping inside containment must be acquired using remote sensors. A piping vibration startup test program that meets the ASME code, in accordance with the regulatory guidelines, will be performed. This program is outlined in Section 10.4.

3.6 Main Steam Line Flow Restrictors

EPU has no adverse effect on the main steam line flow restrictor function. The effects of EPU on main steam line flow restrictor safety and design bases, as identified in UFSAR Section 5.4.4, were evaluated and found to be acceptable.

3.7 Main Steam Isolation Valves

The Main Steam Isolation Valves (MSIVs) are part of the RCPB, and perform the safety function of steam line isolation during certain abnormal events. The MSIVs have been generically evaluated. The generic evaluation covers both the effects of the changes to the structural capability of the MSIV to meet pressure boundary requirements, and the potential effects of EPU related changes to the safety functions of the MSIVs. The conditions for Quad Cities 1 & 2 are bounded by those in the generic analysis. Technical Specification timing requirements will continue to be met. Therefore, EPU conditions are bounded by the conclusions of the generic evaluation, and the MSIVs are acceptable for EPU operation.

3.8 Reactor Core Isolation Cooling

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, when the RPV pressure is greater than the maximum allowable for initiation of a low pressure core cooling system. For EPU, there is no change to the RCIC high pressure injection process parameters. The calculated minimum required RCIC injection rate at EPU conditions remains below the specified system design flow rate. Consequently, RCIC turbine operation with the EPU does not result in any changes to the startup transients or system reliability. The EPU does not decrease the NPSH available for the RCIC pump or change the NPSH required above the specified design value. Surveillance testing and the infrequent demands for system injection for the EPU, occur at the same pre-EPU reactor operating pressures. As a result, there is no change to the existing system and component reliability rates

The RCIC system has been evaluated for loss of feedwater transient event. This evaluation was performed consistent with the guidelines specified in ELTR1. The results demonstrate that acceptance criterion will continue to be met. Therefore, the RCIC system is acceptable for EPU.

3.9 Residual Heat Removal Systems

The RHR System is designed to restore and maintain the coolant inventory in the reactor vessel and provide primary system decay heat removal following reactor shutdown for both normal and post accident conditions. The RHR System is designed to operate in the Low Pressure Coolant Injection (LPCI) mode, Shutdown Cooling (SDC) mode, Suppression Pool Cooling (SPC) mode, Containment Spray Cooling (CSC) mode, and Fuel Pool Cooling (FPC) assist. The LPCI mode is discussed in Subsection 4.2. The effects of EPU on the remaining modes are discussed in the following subsections.

For EPU, the SDC mode operational objective was evaluated using two RHR loops. The resultant cooldown time for EPU meets its operational objective.

During normal plant operation, the function of the SPC mode is to maintain the pool temperature below the Technical Specification limit. Following abnormal events, the SPC mode controls the long-term pool temperature so that the containment design temperature is not exceeded. This requirement is met with EPU, because the containment analysis (Section 4.1) confirms that the pool temperature remains below its design limit.

The CSC mode provides suppression pool water to spray headers in the drywell and suppression chambers to reduce containment pressure and temperature during post-accident conditions. EPU increases the containment spray temperature. This increase has no effect on the calculated peak values of drywell pressure, drywell temperature and suppression chamber pressure, because these parameters reach peak values prior to actuation of the containment spray.

FPC assist uses the RHR heat removal capacity, to provide supplemental fuel pool cooling in the event that the fuel pool heat load exceeds the heat removal capability of the Fuel Pool Cooling and Cleanup (FPCC) system. This mode can be operated to maintain the Fuel Pool temperature within acceptable limits. As discussed in Section 6.3, the increase in fuel pool heat load due to EPU does not exceed the heat removal capacity of this RHR mode.

3.10 Reactor Water Cleanup System

The Reactor Water Cleanup (RWCU) system operation at the EPU slightly decreases the temperature and maintains the same pressure within the RWCU System. This system is designed to remove solid and dissolved impurities from recirculated reactor coolant, thereby reducing the concentration of radioactive and corrosive species in the reactor coolant. The system is capable of performing this function at the EPU level.

The RWCU system can perform adequately during EPU with original RWCU system flow. EPU results in a slight increase in the reactor water conductivity because of the increase in feedwater flow. However, the present reactor water conductivity limits are expected to be met.

3.11 Balance-Of-Plant Piping Evaluation

This section addresses the adequacy of non-RCPB balance-of-plant (BOP) piping design for operation at EPU conditions. Large bore and small bore safety-related and nonsafety-related piping and supports not addressed in Section 3.5 were evaluated for acceptability at EPU conditions. The system conditions changed by EPU, which have the potential to affect the various piping systems, are primarily due to:

- Increases in flow in the MS, FW and other systems forming part of the turbine cycle.
- Increases in temperature and pressure in portions of the MS, extraction steam, heater drain and cross-around steam piping resulting from the high pressure turbine rotor replacement, which effectively opens the steam flow path.

- Increases in pressure in portions of the FW system resulting from higher FW flow rates.
- Increased temperature of the post-LOCA Torus, which affects all connected piping.

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 NC/ND or ANSI B31.1 Power Piping Code equations. The original codes of record and code allowables, as referenced in the appropriate calculations, were used.

The Design Basis Accident (DBA)-LOCA dynamic loads including the pool swell, vent thrust, condensation oscillation (CO) and chugging loads as well as RV and SRV discharge loads were originally defined and evaluated. The design of structures attached to the Torus shell, such as piping system, vent penetrations and valves include these design basis DBA-LOCA and RV/SRV hydrodynamic loads. These hydrodynamic loads are not increased by EPU conditions, and there is no resulting impact on the design of structures attached to the Torus shell.

Operation at EPU conditions increases stresses on piping and piping system components due to higher operating temperature, pressure and/or flow rate internal to the pipes. For all systems, the maximum stress levels results were reviewed based upon conservative bounding criteria developed from system-specific increases in temperature, pressure and/or flow rate. These piping systems were evaluated to determine if sufficient margins exist in the original design analyses to accommodate the increased stresses due to EPU. Some MS and Torus attached piping was found not to have sufficient margin in the original design analyses to justify its acceptability at the bounding EPU loading conditions. More detailed analyses were performed, which demonstrate the adequacy of the existing piping design for EPU conditions. In some cases, piping modifications are required to bring the piping within Code allowable stress limits. These modifications will be completed prior to implementation of EPU. No new postulated pipe break location was identified during this review.

Loads on pipe supports increase due to the same EPU conditions that increase piping stresses. However, when combining these increases with the loads that are not affected by EPU, such as seismic and deadweight, the overall combined support load increases are generally insignificant except for MS and Torus attached piping.

The supports for piping systems with increased stresses at EPU conditions were evaluated to determine if sufficient margins exist between bounding EPU stresses and Code limits in the existing design to accommodate the EPU changes. Some supports were found not to have sufficient margin in the original design/analyses to justify acceptability at EPU conditions. In these cases, more detailed analyses were performed that demonstrate the adequacy of the existing pipe support design for EPU conditions. In some cases, modifications of the supports, structural attachments or supporting steel are required to meet Code allowable stress limits. These modifications will be completed prior to implementation of EPU.

The integrity of high energy piping systems is assured by proper design in accordance with the applicable codes and standards. A consideration in assuring proper design and maintaining system operation within the design is the allowable piping thickness values. Because piping thickness values of carbon steel components can be affected by flow accelerated corrosion (erosion/corrosion), the plant has an established program for monitoring pipe wall thinning in single phase and two-phase high energy carbon steel piping. The effects of EPU will be incorporated into the existing plant pipe monitoring program. This program ensures that EPU effects on high energy piping systems potentially susceptible to pipe wall thinning due to flow accelerated corrosion will be addressed.

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. Short-term and long-term containment analyses results are reported in the UFSAR. The short-term analysis is primarily directed at determining the drywell pressure response during the initial blowdown of the reactor vessel inventory to the containment following a large break inside the drywell. The long-term analysis is primarily directed at the suppression pool temperature response, considering the decay heat addition to the suppression pool. The EPU containment analysis demonstrates that the containment and drywell pressure and temperature responses remain within design allowables.

The LOCA containment dynamic loads include pool swell, condensation oscillation (CO), chugging, and vent thrust loads. Evaluation of the LOCA dynamic loads for EPU is primarily based on the short-term DBA-LOCA pressure and temperature response analysis. The DBA-LOCA pressure and temperature response analyses provide the calculated values of the controlling parameters for the dynamic loads throughout the blowdown. The key parameters are drywell and wetwell pressure, vent flow rates and suppression pool temperature. The DBA-LOCA dynamic loads for EPU remain bounded by the existing load definition

The RV plus SRV discharge loads include RV+SRV discharge line (DL) loads, suppression pool boundary pressure loads and drag loads on submerged structures. The RV/SRV discharge loads are evaluated for two different actuation phases: initial actuation and re-actuation. For EPU, the RV/SRV discharge loads due to initial actuation and the re-actuation remain bounded by the existing load definitions.

Because this EPU does not include a reactor operating pressure increase, the changes in actual asymmetrical loads on the vessel, attached piping and biological shield wall, due to a postulated pipe break in the annulus between the reactor vessel and biological shield wall are minor. The biological shield wall and component designs remain adequate, because there is sufficient pressure margin available.

The capability of the containment isolation valves to perform their isolation function during normal operations and under engineered safety features actuation conditions has been determined to be acceptable, except as addressed below.

All motor-operated valves (MOVs) used as containment or high energy line break (HELB) isolation valves will be reviewed for the effects of EPU conditions, including potential locking and thermal binding (GL 95-07). If specific valves require calculation revisions, actuator adjustments and/or physical changes to ensure satisfactory performance, then these upgrades and any other field adjustments or modifications will be performed prior to EPU operation.

The plant's past response to Generic Letter 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions," was reviewed for EPU post accident conditions. The results of existing evaluation and the past responses remain valid for the EPU.

4.2 Emergency Core Cooling Systems

HPCI performance has been generically evaluated for a reactor operating pressure increase. Because there is no pressure increase for this EPU, HPCI operating conditions and performance are not affected, and the generic evaluation is bounding. Therefore, the HPCI system is acceptable for EPU.

The Low Pressure Coolant Injection (LPCI) mode of the RHR system is automatically initiated in the event of a LOCA. The increase in decay heat due to EPU could increase the calculated peak cladding temperature (PCT) following a postulated LOCA by a small amount. The ECCS performance evaluation presented in Section 4.3 demonstrates that the existing LPCI mode performance capability, in conjunction with the other ECCS, is adequate to meet the post-LOCA core cooling requirement for EPU conditions. The RHR equipment required to perform the LPCI function are within the existing equipment capabilities.

The Core Spray (CS) system is automatically initiated in the event of a LOCA. The increase in decay heat due to EPU could increase the calculated PCT following a postulated LOCA by a small amount. The ECCS performance evaluation presented in Section 4.3 indicates that the existing CS system performance capability, in conjunction with the other ECCS, is adequate to meet the post-LOCA core cooling requirement for EPU conditions. The CS equipment required to perform the CS injection function are within the existing equipment capabilities.

The Automatic Depressurization System (ADS) is required to reduce reactor pressure following a small break LOCA. The ADS initiation logic and ADS valve control are adequate for EPU conditions. The ability to initiate ADS on appropriate signals is not affected by EPU. To achieve the required flow capacity for EPU conditions, five ADS valves must be operable. Prior to EPU, only four ADS valves were required to be operable.

The ECCS NPSH requirements were evaluated for EPU conditions based on the pressure and temperature conditions determined by the containment analysis (Section 4.1), flow requirements based on the containment and LOCA analyses (Section 4.3) and flow losses, including suction strainer losses, determined using the same methodology previously reviewed by the NRC.

Calculations show that the minimum available NPSH margin for the Core Spray and RHR pumps is not reduced during the short-term or long-term period following a DBA-LOCA. As with the original design analysis, the NPSH calculation does take credit for the wetwell airspace pressure during both short-term and long-term periods. The credit taken for wetwell airspace pressure is adjusted for EPU conditions. This adjustment maintains the same (or greater) margin between the credited pressure profile and the analytical profile and the same (or greater) margin between the credited pressure profile and the pressure required for operation of each pump.

The available NPSH and required NPSH for the HPCI pump are not changed for the EPU, since the system configuration and design temperature do not change.

4.3 Emergency Core Cooling System Performance

The Emergency Core Cooling Systems (ECCS) are designed to provide protection against hypothetical loss-of-coolant accidents (LOCA) caused by ruptures in the primary system piping. ECCS performance and analysis models satisfy the requirements of 10 CFR 50.46 and 10 CFR 50 Appendix K. The limiting break was analyzed using both nominal and Appendix K assumptions at pre-uprate and at 117.8% thermal power uprate to assess the impact of EPU. The largest difference between EPU and pre-EPU was less than 10°F for the limiting break PCT. Therefore, the increase in reactor power due to EPU has a negligible impact on the Licensing Basis PCTs, the local oxidation, the hydrogen generation, the coolable geometry, and the long-term cooling.

In the small break LOCA events for which HPCI is assumed to fail, it is assumed that the ADS has the four RVs and the one SRV functioning.

Consistent with the implementation of ARTS power and flow dependent limits, no credit for the APRM setdown was taken in determining the effects of operating within the EPU power/flow map.

4.4 Main Control Room Atmosphere Control System

The increase in heat gain to the control room as a result of EPU for both normal and emergency modes is insignificant. The iodine loading on the control room filters remains a small fraction of the allowable limit of total Iodine (radioactive plus stable) per gram of activated carbon, identified in Regulatory Guide 1.52. Therefore, the control room iodine filter efficiency is not affected by EPU.

4.5 Standby Gas Treatment System

The capacity of the SGTS was selected to provide a negative differential pressure between secondary containment and the outside air of at least 0.25-inch of water. This capability is not affected by EPU. The charcoal filter bed design removal efficiency for radioiodine is unaffected by EPU.

The amount of cooling airflow needed to limit the adsorber temperature increases, due to fission product decay heating, from 48 cfm to 74 cfm, which is well below the available design flow of 300 cfm. No other SGTS parameter is affected by EPU.

4.6 Post-LOCA Combustible Gas Control

The post-LOCA combustible gas control system (CGCS) consists of the primary containment inerting system, the Nitrogen Containment Atmosphere Dilution (NCAD) system, the Containment Atmosphere Monitoring (CAM) system, and the Augmented Primary Containment Venting System (APCVS). The CGCS 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. Only the post-LOCA production of hydrogen and oxygen by radiolysis, which increases in proportion to power level, is directly impacted by EPU. The hydrogen contribution from metal-water reaction of fuel cladding is not affected by the EPU but is affected by fuel design. Therefore, the analysis considers the impact of GE14 fuel introduction on metal-water hydrogen production.

The analysis shows that the increases in metal-water reaction and post-LOCA radiolytic hydrogen and oxygen production do not impact the ability of the system to maintain containment oxygen at or below the 5% flammability limit, using Regulatory Guide 1.7 assumptions. The time required to reach the 5% oxygen limit following the LOCA, based on 1% per day containment leakage, decreases from 25 hours for pre-EPU conditions to 19 hours for EPU. This reduction in required initiation time does not affect the ability of the operators to respond. Therefore, the CGCS retains its capability of meeting its design basis function of controlling oxygen concentration following the postulated DBA LOCA. GE14 fuel bounds the legacy fuel.

Evaluation of the nitrogen requirements to maintain the containment atmosphere at or below the 5% flammability limit for 7 days post-LOCA shows that the minimum stored volume requirement is 141,000 scf. The NCAD nitrogen storage system, with a minimum volume of 200,000 scf, therefore has sufficient capacity to accommodate 7 days of post-LOCA operation. Analysis of the containment pressure buildup as a result of continuing NCAD operation shows that the containment operating pressure limit of 31 psig (50% of the design pressure) is not exceeded until 32 days after the LOCA. Therefore, the minimum 30-day acceptance limit (to reach 50% of the design pressure) is met for EPU.

5 INSTRUMENTATION and CONTROL

5.1 NSSS Monitoring and Control Systems

The instruments and controls that directly interact with or control the reactor are usually considered within the Nuclear Steam Supply System (NSSS). The NSSS process variables, instrument setpoints and Regulatory Guide 1.97 instrumentation that could be affected by EPU were evaluated. As part of EPU implementation, both the ComEd and General Electric setpoint methodologies are used to generate the allowable values and (nominal trip) setpoints related to the analytical limit changes for EPU.

The following summarizes the results of the NSSS evaluations.

Changes in process variables and their effects on instrument setpoints were evaluated for EPU operation to determine any related changes. Process variable changes are implemented through changes in plant procedures.

Increases in the core thermal power and steam flow affect some instrument setpoints, as described in Section 5.3. These setpoints are adjusted to maintain comparable differences between system settings and actual limits, and reviewed to assure that adequate operational flexibility and necessary safety functions are maintained at the EPU power level.

For EPU, the average power range monitor (APRM) power signals are adjusted to the EPU power such that the indications read 100% at the new licensed power.

EPU has little effect on the intermediate range monitor (IRM) overlap with the source range monitors (SRM) 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 elimination of the APRM gain and setpoint requirement (due to ARTS power and flow dependent limits) is described in Sections 1.4 and 9.2.

EPU slightly reduces the neutronic life of the LPRM detectors and radiation levels of the TIPs, but the change is expected to be very small.

The Rod Block Monitor (RBM) initiates a control rod block if local power exceeds a preset limit around a selected rod during withdrawal. The RBM is required to be operable when the reactor is at ≥ 30% of current rated power. This applicability value does not change for EPU.

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. Adjustment to the calibration value is needed to maintain the setpoint for EPU.

5.2 BOP Monitoring and Control Systems

Operation of the plant at the EPU power level has minimal effect on the balance-of-plant (BOP) system instrumentation and control devices. Based on EPU operating conditions for the power conversion and auxiliary systems, process control valves and instrumentation have sufficient range/adjustment capability for use at the expected EPU conditions, except as noted in the sections that address each BOP system. However, some modifications may be needed to the power conversion systems to obtain full EPU.

The pressure control system (PCS) provides fast and stable responses to system disturbances related to steam pressure and 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 increased steam flow for EPU along with a change to the turbine high pressure rotor requires the Turbine Control Valves (TCV) to operate under different conditions. The flow capacity of the TCVs and other characteristics after modifications to the high pressure turbine rotor require evaluations to assure that all requirements regarding interaction between the T-G and the NSSS have been addressed.

Specific EHC and steam bypass control system tests will be performed during the power ascension phase. These tests are summarized in Section 10.4.

The turbine EHC system was reviewed for the increase in core thermal power and the associated increase in rated steam flow. For EPU conditions, a second steam line resonance compensator (SLRC) card will be installed to attenuate third harmonic resonance. In addition, TCV Diode Function Generator tuning for the redesigned conditions will be required. The control systems are expected to perform normally for EPU operation.

Modifications to the TCVs may be required for the uprated throttle conditions. Confirmation testing will be performed during power ascension (see Section 10.4).

The feedwater control system is used to maintain water level control in the reactor. The capacity of the feedwater pumps is adequate to support the EPU, and this will be demonstrated by startup testing. The basic capacity requirement for adequate reactor water level control is approximately 105% of the operating point flow rate. The feedwater system has capacity in excess of the 105% of the EPU rated feedwater flow required for transient operation with three feedwater pumps operating. With adjustments in feedwater and steam flow instrument spans and feedwater pump runout protection, the control system is capable of accessing as much of the flow as needed. Therefore, the capacity is sufficient for acceptable control.

The control system is adjusted to provide acceptable operating response on the basis of unit behavior. It has been set up successfully to cover the current power range using startup and periodic testing. For EPU, no change in the operating water level is required. The feedwater flow control system device settings have the sufficient adjustment ranges to ensure satisfactory operation. This will be confirmed by performing unit tests during the power ascension to EPU conditions (Section 10.4).

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 EPU. Each of the systems (listed below) where leak detection could potentially be affected by EPU, was evaluated, and no leak detection related change is required.

- Main Steam Tunnel Temperature Based Leak Detection
- RWCU System Temperature Based Leak Detection
- RCIC System Temperature Based Leak Detection
- HPCI System Temperature Based Leak Detection
- Non-Temperature Based Leak Detection

5.3 Instrument Setpoints

Instrument setpoints in the Technical Specifications (TS) are established using approved setpoint methodologies. 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 also provided between the actual trip setting and the normal operating limit to preclude inadvertent initiation of the protective action.

The following paragraphs discuss those instrument setpoint analytical limits that are potentially affected by EPU. Plant setpoints (derived from the EPU analytical limits) ensure timely actuation of the necessary safety functions while avoiding spurious trips during EPU operation.

- Because no pressure increase is associated with this EPU, the scram analytical limit (AL) on reactor high pressure is unchanged.

- The current ATWS-RPT high pressure setpoint was included in the ATWS evaluation discussed in Section 9.3. This evaluation concludes that the current ATWS-RPT high pressure setpoint is acceptable for EPU.

- Because there is no increase in reactor operating dome pressure, the setpoints for the SSVs, SRV and RVs are not increased.

- The Main Steam Line (MSL) High Flow Isolation AL remains at 140% of EPU rated steam flow. The instrumentation will be replaced with a higher range differential pressure instrument and recalibrated for the higher steam flow condition. This ensures that sufficient difference to the trip setpoint exists to allow for normal plant testing of the MSIVs and turbine stop and control valves.

- New ALs of the flow biased APRM scrams and rod blocks are developed for EPU, similar to those shown in Figure 5-1 of ELTR1. The ALs for the APRM Flow Biased Scram, APRM Rod Block, and RBM Setpoints form the basis for the EPU/MELLLA setpoints, including the minimum core flow allowable at EPU power. The EPU application of the flow biased RBM (non-ARTS) is to maintain the same AL values, which is the same basis as for the Fixed (Non-Flow Biased) High APRM Scram. The APRM Scram and Rod Block are clamped at their maximum power values based on a core flow of 95.3%. The Rod Block Monitor is clamped at its maximum power value based on 100% core flow. The MELLLA AL for the fixed (clamped) APRM scram for two recirculation loop operation remains the same but the AL for single recirculation loop operation (SLO) is changed to be the same as the AL for TLO.

- The RBM instrument setpoints are determined on a fuel cycle-specific basis and will be modified (as needed) when EPU is first implemented.

- The purpose of the Low Steam Line Pressure MSIV Closure (RUN Mode) trip is to initiate MSIV closure on low steam line pressure when the reactor is in the RUN mode. This setpoint is not changed for the EPU.

- The reactor water level trip values used in the safety analyses do not require changing due to EPU. However, the reactor low water level scram AL is being reduced, to provide additional operating margin (i.e., prevent unnecessary scrams) for a reactor recirculation runback on a loss of a reactor feedwater pump from EPU conditions. The revised low water level scram AL is used in the applicable EPU safety analyses (i.e., transient and ECCS-LOCA). Also, the primary containment, RWCU, RHR Shutdown Cooling System, secondary containment, and Control Room Emergency Ventilation (CREV) system isolation trips initiate from the same reactor low water level as used for the scram trip. Therefore, the allowable values (AVs) used for the primary containment, RWCU, RHR Shutdown Cooling System, secondary containment, and Control Room Emergency Ventilation (CREV) system isolations must be revised to remain consistent with the scram function.

- At EPU conditions, the increase in steam tunnel ambient temperature is not significant, and thus, no change to the MSL Tunnel High Temperature Isolation setpoint is required.

- With the increased heat input due to EPU, the condenser backpressure rises. The plant has a nominal alarm for condenser low vacuum at 25 inches Hg and a nominal scram at 23 inches Hg. To maintain adequate operating margin between the alarm and the scram, the alarm setpoint, nominal scram setpoint and associated AV will be adjusted. The AL for this function is unchanged.

- The TSV Closure and TCV Fast Closure Scram Bypass AL expressed as a percent of rated thermal power is reduced by the ratio of the power increase. The new AL does not change with respect to absolute thermal power and steam flow, and thus, there is no effect on the transient response. A high pressure turbine rotor modification changes the relationship between turbine first stage pressure and steam flow such that the scram bypass AL in psig must change to assure that the scram bypass does not occur above the desired core thermal power and turbine steam flow point.

- For EPU, the Rod Worth Minimizer low power setpoint (LPSP) remains 10% of RTP. This is conservative, because it requires enforcement of rod pattern controls to a higher absolute power level.
- The pressure control system (PCS) is discussed in Section 5.2. The pressure setpoint, pressure regulator gain, main steam line pressure drop, turbine stop valve inlet pressure and turbine-generator required load setpoint are related to each other and to reactor dome pressure. The reactor dome pressure is not changed for EPU. However, the increased steam flow results in a somewhat greater steam line pressure loss. Therefore, the steam bypass control system pressure regulator operational setpoint must be adjusted to achieve the desired reactor pressure. Due to small differences in plant parameters, the optimal pressure regulator setpoint may slightly differ between the units. Specific EHC and steam bypass control system tests will be performed during the initial power ascension following any T-G modifications needed to implement EPU. These tests are summarized in Section 10.4.
- The current value of the feedwater flow setpoint for recirculation cavitation protection is unchanged in terms of absolute feedwater flow rate. However, the relative setpoint, as it appears on the power/flow map, is reduced slightly to account for the EPU.
- For EPU, the AL for the RCIC steam line high flow isolation remains based on 300% of the maximum rated steam flow to the RCIC turbine.
- For EPU, the AL for the HPCI steam line high flow isolation remains based on 300% of the maximum rated steam flow to the HPCI turbine.

6 ELECTRICAL POWER and AUXILIARY SYSTEMS

6.1 AC Power

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

- The isolated phase bus duct is adequate for both rated voltage and low voltage current output.
- The main transformers and the associated switchyard components are adequate for the uprated output.
- A grid stability analysis determined that there is no significant effect on grid stability or reliability. There is no modification associated with EPU that would increase electrical loads beyond those levels previously included, or revise the control logic of the distribution systems.

Station loads under normal operation/distribution conditions are computed based on equipment nameplate data and calculated brake horsepower with actual diversity factors applied. The only identifiable change in electrical load demand is associated with condensate and booster pumps, reactor recirculation pumps, reactor feedwater pumps, and condensate demineralizers. The increased flow due to EPU conditions requires energizing the installed spare (third) reactor feedwater pump, energizing the installed spare (fourth) condensate and booster pump, and the increase of the operating point for the two reactor recirculation pumps. These additional loads when evaluated by design basis calculations result in acceptable operation of the electrical auxiliary system during normal startup and operation with two auxiliary transformers in service.

Operation at EPU conditions on a single transformer exceeds the non-safety 4160V switchgear short circuit rating, the transformer winding rating, and the bus duct rating. Also, in the event of a fast transfer to single transformer operation at EPU conditions, the same situation will exist. To address these potential operational problems, Quad Cities will institute a procedurally controlled load shedding scheme to be implemented following a fast transfer. This approach will be confirmed by thermal analysis or an engineering evaluation to address the overload conditions for the auxiliary transformers, the bus duct, and related connections. To address the potential operational problem due to the switchgear overduty condition, a test to upgrade the switchgear and breakers to a higher momentary current rating will be performed and a time delay of about 6 cycles on the short circuit interrupting will be implemented. A review of the 4160V bus and auxiliary transformer overcurrent relay setpoints will also be performed to ensure proper settings for operation at EPU conditions.

No increase in flow or pressure is required of any AC-powered ECCS equipment for EPU. Therefore, the amount of power required to perform safety-related functions (pump and valve loads) is not increased with EPU and the current emergency diesel generator 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 DC Power

The direct current (DC) loading requirements in the UFSAR were reviewed, and no reactor power dependent loads were identified that affected the DC Power System design. System loads were computed based on equipment nameplate data. Operation at the EPU level does not increase any loads beyond nameplate rating or revise any control logic; therefore, the DC power distribution system is adequate.

6.3 Fuel Pool

The effects of EPU on fuel pool cooling, crud and corrosion products in the fuel pool, radiation levels and structural adequacy of the fuel racks are small and within the design limits of the affected systems and components.

EPU increases the spent fuel pool heat load. The adequacy of the FPCCS is determined by evaluating the ability of the system to maintain the temperature of the fuel pool. The fuel pool temperature is analyzed by calculating the decay heat load following a normal batch discharge or full core discharge, with other spaces filled as a result of fuel discharges from normal refueling outages. The results of the analyses show that the maximum heat load in the spent fuel storage pool to be less than the heat removal capability of the fuel pool cooling heat exchangers, and the peak fuel pool temperature remains below its limit.

Crud activity and corrosion products associated with spent fuel may increase slightly due to EPU. However, the increase is shown to be insignificant, and fuel pool water quality is maintained by the fuel pool cleanup system.

The normal radiation levels around the pool can increase slightly primarily during fuel handling operation. This increase is acceptable and does not significantly increase the operational doses to personnel or equipment.

The fuel racks are designed for higher temperatures than are anticipated from EPU. There is no effect on the design of the fuel racks, because the original design fuel pool temperatures are not exceeded. Therefore, the racks are acceptable for the higher local decay heat loads.

6.4 Water Systems

The environmental effects of EPU are controlled at the same levels as for the original analyses. That is, none of the present limits for plant environmental releases are increased as a consequence of EPU. If the plant releases challenge environmental limits then plant operation is managed such that the existing limits would not be violated with EPU.

The safety-related service water systems are designed to provide reliable supplies of water for the following essential equipment and systems:

Residual heat removal heat exchangers;

Diesel generator cooling water (DGCW) heat exchangers;

Control room emergency ventilation system refrigeration condensing unit;

RHR pump motor coolers;

RHR pump seal coolers;

HPCI room cooler;

Residual heat removal service water (RHRSW) pump cubical coolers;

DGCW pump cubical coolers;

Core spray room coolers;

RHR heat exchanger room coolers; and

Spent fuel pool, if needed, as emergency makeup.

The safety-related performance of the RHRSW and DGCW service water systems during and following the most demanding design basis event, the LOCA with LOOP, has been reviewed and found acceptable. The containment cooling analysis in Section 4.1 assumes the post LOCA containment cooling capacity does not change. The increased heat load is within the existing capacity of the RHRSW System.

The temperature of service water discharge results from the heat rejected to the service water system via the closed cooling water systems and other auxiliary heat loads. The major service water heat load increases from EPU reflect an increase in main generator losses rejected to the stator water coolers and hydrogen coolers in addition to increased Turbine Building Closed Cooling Water (TBCCW) and Reactor Building Closed Cooling Water (RBCCW) heat loads.

The increased heat loads result in a slight increase in the temperature of the service water discharged to the Circulating Water System.

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

EPU operation increases the heat rejected to the condenser, and therefore, reduces the difference between the operating pressure and the required minimum condenser vacuum. If condenser pressures approach the backpressure limitation, then reactor thermal power must be reduced to maintain adequate condenser vacuum, thereby limiting generator output.

A comparison of state discharge limits to the current discharges and bounding analysis discharges for EPU demonstrates that the plant remains within the state discharge limit during operation at EPU. Regardless, if needed to accommodate extremes in ambient conditions, plant operations (e.g., temporary plant de-rating) will ensure that state discharge limits are not exceeded.

The heat loads on the RBCCW system do not increase significantly by EPU because they depend mainly on either vessel temperature or flow rates in the systems cooled by the RBCCW. The RBCCW system contains sufficient redundancy in pumps and heat exchangers to ensure that adequate heat removal capability is always available. Therefore, sufficient heat removal capacity is available to accommodate the increase in heat load due to EPU.

The heat loads, which are increased by EPU, on the TBCCW system include the Bus Duct Coolers, the added heat from the operation of the fourth Condensate/Condensate Booster Pump and the added heat from the operation of the third Reactor Feed Pump. The remaining TBCCW heat loads are not strongly dependent upon reactor power and do not increase significantly. The additional heat loads can be removed by the TBCCW system with a minimal increase in TBCCW temperature, which will have negligible effect on the equipment cooled by the TBCCW and is therefore deemed acceptable.

The normal heat sink is the river via the intake and discharge canals. However, in the event of a loss of the downstream dam, the water trapped in the intake canal becomes the ultimate heat sink (UHS). In this event, make-up water addition is required to the intake canal for decay heat removal at EPU conditions. This make-up activity is currently required for present plant operations. Sufficient time is available to replenish the water in the intake canal following a loss of the dam to adequately remove the decay heat at EPU to maintain shutdown conditions.

6.5 Standby Liquid Control System

The ability of the SLCS boron solution to achieve and maintain safe shutdown is not affected by EPU. SLCS shutdown capability (in terms of required boron concentration) is reevaluated for each fuel reload.

The ATWS performance evaluation (Section 9.4) shows that EPU has no adverse effect on the ability of the SLCS to mitigate an ATWS.

6.6 Power-Dependent Heating Ventilation and Air Conditioning

The heating ventilation and air conditioning (HVAC) systems consist mainly of heating or cooling supply, exhaust and recirculation units in the turbine building, reactor building and the drywell. The EPU is expected to result in a small increase in the heat loads caused by slightly higher process temperatures and higher electrical currents in some motors and cables.

The affected areas are the steam tunnel, ECCS pump rooms, and drywell in the reactor building; the feedwater heater bay and condenser area, feedwater pumps, condensate/condensate booster pumps and the MG set areas in the turbine building. Other areas are unaffected by the EPU because the process temperatures remain relatively constant.

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

6.7 Fire Protection

Operation of the plant at the EPU power 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 the plant modification and fire protection programs. The safe shutdown systems and equipment used to achieve and maintain cold shutdown conditions do not change, and are adequate for EPU conditions. The operator actions required to mitigate the consequences of a fire are not affected. Therefore, the fire protection systems and analyses are not affected by EPU.

The reactor and containment responses to the postulated 10 CFR 50 Appendix R fire event at EPU conditions were re-analyzed, and show that the fuel PCT, reactor pressure, and containment pressures and temperatures are below the acceptance limits. This plant-specific evaluation demonstrates 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 EPU demonstrate that fuel cladding, RPV and containment integrities are maintained. Therefore, EPU has no adverse effect on the ability 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

Systems with No Impact:

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

- Cathodic Protection
- Control Rod Velocity Limiters
- Control Room HVAC- Normal
- Counting Room HVAC Blower/Fan
- Crane & Hoists
- CRD Support Shootout Steel
- Economic Generation Control
- Elevators
- Fish Hatchery
- Gas Monitor HVAC Exhaust Fans
- Generator Auto Dispatch
- Guard House
- Industrial Security
- Lighting, Receptacle, Dist. Panel
- Miscellaneous non-power generation systems
- Miscellaneous Buildings
- Miscellaneous HVAC
- MISC Outside Work

- Natural Gas
- Nuclear Fuel Handling Equipment
- Out Building HVAC
- Public Address
- Radwaste Facility HVAC
- Refuel Bridge (Tools, Serv)
- River Screen House
- RPS Motor Generator Sets HVAC
- Service Air
- Service Bldg A/C Heating HVAC
- Service Building
- Spare Parts
- Startup Equipment
- Station Heating
- Turb RW Part Sample
- UPS/ Batt and CMPTR Room HVAC
- Vent DMPR / Equipment
- Visitors Center HVAC
- Waste Water Treatment

Systems with Insignificant Impact:

Some systems are affected in a very minor way by operation of the plant at the EPU level. For the following systems, the effects of EPU are insignificant to the design or operation of the system and equipment:

- Area Radiation Monitor (alarm setpoints may be adjusted slightly based on area dose rate changes)
- Control Rod Blades
- DG Cooling Water
- Diesel Fuel Oil
- DW, N2, O2, Analyzer
- EDG Vent Fans
- FW PMP MTR HVAC system
- High Rad Sample

- Hydrogen Addition
- Hypochlorite
- Instrument Air & DW Pneumatic
- Local Panels & Racks
- Main Control Room Panels
- Make-up Demineralizer
- Reactor Protection
- Test Instruments
- Well Water

Table 6-1

Uprated Plant Electrical Characteristics

Main Generator Electrical Design Parameters [1]	Data	
	QC 1 Value	**QC 2 Value**
Generator Rating (MVA)	960	960
Gross Generator Output (MWe)	912	912
Rated (KV)	18	18
Power Factor	0.95	0.95
Current Output (Amps)[2]	30792	30792
Isolated Phase Bus Duct Rating:		
Main Section (Amps)	33,000	33,000
Branch Section (Amps)	2,000	2,000
Main Transformers Rating (MVA)	985	952
Transformer Output (MVA)	940	940

Notes:

1. Main Generator MVA ratings for EPU were evaluated and found acceptable.
2. The current output is calculated using Gross Generator Output (960MVA)

7 POWER CONVERSION SYSTEMS

The power conversion systems were originally designed to utilize the energy available from the nuclear steam supply system and were designed to accept the system and equipment flows resulting from continuous operation at 9,754,965 lb/hr of design steam flow. However, the structural capabilities of the power conversion systems allow for steam flows greater than the (9,754,965 lb/hr) design steam flow, to EPU conditions, with modifications to the high pressure turbine and to some nonsafety-related equipment.

7.1 Turbine-Generator

With uprate the expected generator output is 912 MWe at 0.95 power factor which is within the capability of the generator.

Steam specification calculations were performed to determine the uprated turbine steam path conditions. From the thermodynamic models, turbine and generator stationary and rotating components were evaluated for increased loadings, pressure drops, thrusts, stresses, overspeed capability and other design considerations to ensure that design limits are not exceeded and that plant operation remains acceptable at the EPU condition. In addition, valves, control systems and other support systems were evaluated. The evaluations show that the modifications to the high pressure turbine and some nonsafety-related equipment should ensure satisfactory operation at EPU conditions.

EPU has a negligible effect on HP rotor strength properties and mechanical parameters. The replacement EPU HP rotor consists of an integral rotor, without shrunk-on wheels. The new integral HP turbine rotor is not considered a source for potential missile generation, and therefore, a HP turbine rotor missile probability analysis is not required.

An evaluation of the LP rotors is being performed. The results of this evaluation will be used to determine if changes are required.

The overspeed calculation compares the entrapped steam energy contained within the turbine and the associated piping, after the stop valves trip, and the sensitivity of the rotor train for the capability of overspeeding. The entrapped energy increases slightly for EPU conditions. However, there is sufficient design margin in the current overspeed trip settings to provide protection for a turbine trip, such that should a change in the overspeed settings be necessary, it can be accommodated.

7.2 Condenser and Steam Jet Air Ejectors

The condenser was calculated for performance at EPU conditions based on a cold water temperature at 86°F and current water system flow. An additional analysis for EPU conditions also determined the condenser backpressure would be below its Hg abs design limit.

Both condenser hotwell capacities and level instrumentation are adequate for EPU conditions. Condenser tube staking is planned for the main condensers, which provides adequate protection against tube vibration damage at EPU conditions.

The design of the condenser air removal system is not adversely affected by EPU. The physical size of the primary condenser and the evacuation time are the main factors in establishing the capabilities of the vacuum pumps. These parameters do not change. Because the flow rate does not change, there is no change to the two minute holdup time in the mechanical vacuum pump discharge line. Planned steam dilution modifications of the condenser air removal system to address existing performance issues will provide adequate capacity for EPU conditions.

7.3 Turbine Steam Bypass

The turbine bypass valves were initially rated for a total steam flow capacity of not less than 40% of the original rated reactor steam flow of 9.76 Mlb/hr. Each of 9 bypass valves is designed to pass a steam flow of 0.433 Mlb/hr for a total bypass capacity of 3.90 Mlb/hr. At EPU conditions, rated reactor steam flow is 11.71 Mlb/hr, resulting in a bypass capacity of 33.3%, which is adequate for EPU. All of the transient analyses involving bypass capacity remain valid because the assumed bypass flow is not changed for EPU.

7.4 Feedwater and Condensate Systems

The feedwater and condensate systems do not perform a system level safety-related function. They are designed to provide a reliable supply of feedwater at the temperature, pressure, quality and flow rate as required by the reactor. Their performance has a major effect on plant availability and capability to operate at EPU conditions. For EPU, the feedwater and condensate systems will meet their performance criteria with modifications to some nonsafety-related equipment and changes in operating line-up

Modifications, such as recirculation runback, and alteration of operating system line-up to some nonsafety-related equipment in the feedwater and condensate systems are necessary to attain full licensed EPU thermal power. The current power level requires operation of three of the four condensate/condensate booster pumps and two of the three feedwater pumps. At EPU conditions, operation of all four condensate/condensate booster pumps and all three feedwater pumps is required.

Normal Operation:

The condensate and feedwater systems were originally designed for 105% rated steam flows. Operation at the EPU level does not significantly affect the operating conditions of these systems. As flow through individual pumps increases, the discharge pressure at the condensate and condensate booster pumps decreases due to the pump head characteristics at increased flows. During steady-state conditions, the condensate and feedwater systems have adequate NPSH for all of the pumps to operate without cavitation in the uprated conditions.

The existing feedwater design pressure and temperature requirements are adequate.

Transient Operation:

To account for feedwater demand transients, the feedwater system was evaluated to ensure that a minimum of 5% margin above the EPU feedwater flow was available. This is the same criterion applied to the original design. For system operation with all system pumps available, the predicted operating parameters were acceptable and within the component capabilities.

The plant will implement a reactor recirculation runback feature, to ensure scram avoidance during EPU conditions. A transient analysis was performed to determine the feedwater capacity available following a single feedwater pump trip and subsequent recirculation system runback. The results of the analysis show that the system response is dependent on the runback rate, rather than the feedwater system capability to avoid a scram during the short-term portion of the transient.

Condensate Demineralizer System:

The effect of EPU on the Condensate Demineralizer System was reviewed. The system is adequate for uprate operation with the addition of another demineralizer unit. The demineralizer operational flow is maintained, but with a slight increase in burden on the units, and thus, the time interval between backwashing (as a system) decreases. Section 8 addresses the effects on the radwaste systems.

8 RADWASTE SYSTEMS AND RADIATION SOURCES

8.1 Liquid Waste Management

The liquid radwaste system collects, monitors, processes, stores and returns processed radioactive waste to the plant for reuse or for discharge. The concentration of activated corrosion products in liquid wastes is expected to increase proportionally to the EPU. The volume of liquid wastes is not expected to increase appreciably. The volume of condensate resin generated is expected to increase proportionally to the EPU, due to increased temperature and flow in the condensate system.

An evaluation concludes that the requirements of 10 CFR 20 and 10 CFR 50, Appendix I will continue to be met. Therefore, EPU does not have an adverse effect on the processing of liquid radwaste, and there are no significant environmental effects.

8.2 Gaseous Waste Management

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

Air inleakage evacuated from the main condenser contains non-condensable radioactive gas, normally consisting of activation gases (principally N-16, O-19 and N-13) and fission product radioactive noble gases. These non-condensable gases, along with the non-radioactive air inleakage, are continuously removed from the main condensers by the steam jet air ejectors (SJAEs), which discharge into the offgas system. This process stream represents the major source of radioactive gas (greater than all other sources combined) exiting the primary system.

The activity of airborne effluents released through building vents does not significantly increase with EPU. The expected gaseous effluents are within limits for original power operation. There are no significant environmental effects due to EPU.

Offgas System:

The radioactive releases from the offgas system are conservatively estimated to increase proportionally to the EPU. This estimate is conservative because it is based on the assumption of a non-negligible amount of fuel leakage due to defects. Because the current and expected fuel defect rates are extremely small, the actual offgas release rate may not increase. EPU increases reactor condensate temperature, which increases the offgas condenser effluent temperature, thus requiring setpoints changes to downstream non-safety temperature instruments.

8.3 Radiation Sources in 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.

8.4 Radiation Sources in Reactor Coolant

Radiation sources in the coolant are primarily a function of fuel defects, power level, and operation of the water cleanup systems. It is expected that some increase in fission product activity in reactor coolant will be seen. Using the formula in ANSI/ANS 18.1-1999, "Radiological Source Term for Normal Operation for Light Water Reactors," the increase would result in a calculated 12% increase in concentration. Even with this increase, the reactor coolant activity levels will be fractional parts of the design basis coolant concentrations. Therefore, EPU should essentially have no adverse effect on day to day operation of the plant.

Hydrogen Water Chemistry (HWC) increases the concentration of N-16 in the steam relative to the concentration with Normal Water Chemistry (NWC). The plant is treated by the NobleChem™ process, which significantly reduces the needed hydrogen injection rate compared to the HWC rate without NobleChem™. Therefore, NobleChem™ significantly reduces the N-16 increase normally associated with HWC. The net effect of NobleChem™ on N-16 concentration more than compensates for any potential increase in N-16 caused by EPU.

8.5 Radiation Levels

For EPU, normal operation radiation levels are expected to increase by no more than the percentage increase in power level. For conservatism, many aspects of the plant were originally designed for higher-than-expected radiation sources. Thus, the increase in radiation levels does not affect radiation zoning or shielding in the various areas of the plant because it is offset by conservatism in the original design, source terms used and analytical techniques.

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

The change in core inventory resulting from EPU is expected to increase post-accident radiation levels by no more than the percentage increase in power level. The estimated increase in radiation levels at EPU conditions does not significantly affect the post-accident radiation zoning or shielding assessment presented in the UFSAR, because the estimated increase in dose rate levels is offset by the conservatism in the analytical techniques utilized to develop the original dose rates. EPU has no effect on the habitability of the Technical Support Center or Emergency Operations Facility.

8.6 Normal Operation Off-Site Doses

For EPU, the normal operation activity in the reactor coolant is expected to increase by approximately the same percentage as that of the uprate, i.e., 18%. Examination of the normal radiological effluent doses reported for the last 5 years (1995 – 1999) indicate that the current releases are a small fraction of the 10 CFR 50 Appendix I guidelines. Thus, the dose effect of EPU continues to be a small fraction of the 10 CFR 50 Appendix I guidelines, and remains within the limits of 10 CFR 20.

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 investigated according to the type of initiating event per Regulatory Guide 1.70, Chapter 15. The generic guidelines identify the analytical methods, the operating conditions that are to be assumed, and the criteria that are to be applied. The additional analyses for ARTS power and flow dependent limits are addressed in Section 9.2.

The EPU analysis uses the (NRC approved) GEMINI transient analysis methods discussed in Appendix E of ELTR1 (Reference 1). The results for a representative core show that the overall capability of the design meets all transient safety criteria for EPU operation.

The cycle specific SLMCPRs for both two recirculation loop and single recirculation loop operations will be supplied in the Core Operating Limit Reports (COLRs).

The severity of transients at less than rated power are not significantly affected by EPU, because of the protection provided by the ARTS power and flow dependent limits.

The Loss of Feedwater Flow (LOFW) transient was analyzed for EPU. The sequences of events do not require any new operator actions or shorter operator response times. Therefore, the operator actions for a LOFW transient do not significantly change for EPU.

9.2 Transient Analysis For ARTS Power and Flow Dependent Limits

The core-wide AOOs were analyzed to support the EPU (which includes the MELLLA domain) and the incorporation of the ARTS power and flow dependent limits program. To support the implementation of the ARTS power and flow dependent limits program, these analyses determine the off-rated power- and flow-dependent MCPR and LHGR curves associated with the removal of the APRM gain and setpoint requirement. These evaluations also include consideration from the ECCS-LOCA analysis (Section 4.3).

Transient analyses at a variety of power and flow conditions were performed during original development of the ARTS improvement program. These evaluations are applicable for operation in the MELLLA region. The analyses were utilized to study the trend of transient severity without the APRM gain and setpoint.

Generic power-dependent MCPR and LHGR limits (in terms of multipliers on the plant's rated operating limits) were developed for use in the low power range. The applicability of these generic limits is verified for plant-specific application during the initial ARTS application for that plant. Plant-specific analyses of limiting transients confirm the applicability of the generic power-dependent limits. Cycle specific limits may also be used for any part of the range. A comparison of these plant-specific calculated values with the generic power-dependent MCPR limits (MCPR(P) limits) verifies the applicability of the generic limits to Quad Cities.

In the absence of the APRM gain and setpoint requirement, power-dependent LHGR limits, expressed in terms of a LHGR multiplier, LHGRFAC(P), are substituted to assure adherence to the fuel thermal-mechanical design bases. The power-dependent LHGRFAC(P) limits were generated using the same database as used to determine the MCPR multiplier (K(P)). Similar to the MCPR(P) limits, plant-specific transient analyses were performed to demonstrate the applicability of the generic LHGRFAC(P) limits.

The transient and initial condition selection, as well as the approach taken to confirm and develop the appropriate plant-specific LHGRFAC(P) limits, is identical to that described in the above discussion for MCPR(P).

Flow-dependent MCPR limits, MCPR(F), ensure that the Safety Limit MCPR (SLMCPR) is not violated during recirculation flow increase events. To verify the applicability of the generic flow-dependent MCPR limits, recirculation flow runout events were performed at a typical mid-cycle exposure condition. These flow runout events were simulated along a rod line which bounds the maximum licensed rod line to the maximum core flow runout values at 108% core flow condition. The ARTS-based MCPR(F) limit is specified as an absolute value and is generic and cycle-independent.

Flow-dependent LHGR limits, LHGRFAC(F), ensure adherence to all fuel thermal-mechanical design bases in the event of slow recirculation flow runout event. The same transients events used to support the MCPR(F) operating limits were analyzed generically, and the resulting overpowers were statistically evaluated as a function of the initial and maximum core flow. From the bounding overpowers, the LHGRFAC(F) limits were derived such that, during these events, the peak transient linear heat generation rate would not exceed fuel mechanical limits. The flow-dependent LHGR limits are generic, cycle-independent and are specified in terms of multipliers, LHGRFAC(F), to be applied to the rated LHGR values.

At any given power/flow state (P,F), all four limits are determined: MCPR(P), LHGRFAC(P), MCPR(F) and LHGRFAC(F). The most limiting MCPR and the most limiting LHGR [maximum of MCPR(P) and MCPR(F) and minimum of LHGRFAC(P) and LHGRFAC(F)] are the governing limits.

The results of the analyses documented above can be utilized to determine the plant-specific OLMCPRs.

9.3 Design Basis Accidents

For EPU, the power dependent plant-specific radiological assessments reported in the UFSAR are re-evaluated at 102% of the EPU RTP level. The plant-specific radiological analyses were performed based on EPU conditions for selected postulated accidents. The events reanalyzed were the Loss-of-Coolant Accident (LOCA), the Fuel Handling Accident (FHA), and the Control

Rod Drop Accident (CRDA). The resulting doses from these accidents are provided in Tables 9-1, 9-2 and 9-3, and demonstrate that the plant continues to meet the applicable regulatory guideline exposures values.

9.4 Special Events

For EPU, the plant-specific ATWS analysis was performed, and the results ensure that the following ATWS acceptance criteria are met:

1. Peak vessel bottom pressure less than ASME Service Level C limit of 1500 psig.
2. Peak clad temperature within the 10 CFR 50.46 limit of 2200°F.
3. Peak clad oxidation within the requirements of 10 CFR 50.46.
4. Peak suppression pool temperature shall not exceed 202°F (bounding post-accident suppression pool temperature).
5. Peak containment pressure shall not exceed 62 psig (peak allowable design pressure).

Therefore, the plant response to an ATWS event at EPU is acceptable.

The Quad Cities station blackout (SBO) was performed using the guidelines of NUMARC 87-00, except where USNRC Regulatory Guide 1.155 takes precedence. The plant responses to and coping capabilities for an SBO event are affected slightly by operation at the EPU level, because of the increase in the decay heat for EPU. However, since decay heat is effectively controlled by the use of RCIC and the RVs during the one hour period without AC cooling, while the SBO diesels are loaded to restore power, containment parameters are never challenged. At EPU power there is no change to the systems and equipment used to respond to an SBO nor is the required coping time changed. Therefore, the plant continues to meet the requirements of 10 CFR 50.63 after EPU implementation.

Table 9-1

LOCA Radiological Consequences

Location	UFSAR	Current	EPU	Limit
Offsite:				
Whole Body Dose, rem	5.3E-4(1)	(2)	(2)	≤ 25
Thyroid Dose, rem	1.3E-4(1)	(2)	(2)	≤ 300
Exclusion Area:				
Whole Body Dose, rem	5(3)	5	6	≤ 25
Thyroid Dose, rem	120(3)	120	152	≤ 300
Low Population Zone:				
Whole Body Dose, rem	< 5(3)	<5	< 6	≤ 25
Thyroid Dose, rem	< 120(3)	< 120	< 152	≤ 300
Control Room:				
Whole Body Dose, rem	0.118(4)	0.314	0.377	≤ 5
Thyroid Dose, rem	21.88(4)	22.75	29.6	≤ 30
Beta Dose, rem	1.23(4)	8.71	10.5	≤ 30

Notes:

(1) UFSAR Sect.15.6.5.5.1, Table 15.6-6 (original analysis). This table lists doses as a function of distance and meteorological condition. The doses listed above are at ¼ mile under unstable 2 mph wind speed meteorological condition and represent the worst case values reported.

(2) Not evaluated as it is considered historical information.

(3) UFSAR Sect.15.6.5.5.1, AEC analysis, 1% per day primary containment leak rate.

(4) UFSAR Sect.15.6.5.5.3, Table 15.6-8.

Table 9-2

CRDA Radiological Consequences

Location	UFSAR	Current	EPU	Limit
Offsite:				
Whole Body Dose, rem	1.2E-2(1)	(2)	(2)	≤ 25
Thyroid Dose, rem	1.2E-3(1)	(2)	(2)	≤ 300
Exclusion Area:				
Whole Body Dose, rem	(1)	2.86(3)	3.41	≤ 6.25
Thyroid Dose, rem	(1)	9.43(3)	12.1	≤ 75
Low Population Zone:				
Whole Body Dose, rem	(1)	0.507(3)	0.602	≤ 6.25
Thyroid Dose, rem	(1)	1.04(3)	1.33	≤ 75
Control Room:				
Whole Body Dose, rem	Not	0.224(3)	0.266	≤ 5
Thyroid Dose, rem	reported in	21.8(3)	28.0	≤ 30
Beta Dose, rem	UFSAR	4.53(3)	5.35	≤ 30

Notes:

(1) UFSAR Table 15.4-2 lists doses as a function of distance and meteorological condition. The doses reported above are at ¼ mile under unstable 2 mph wind speed meteorological condition and represent the worst case values reported.

(2) Not evaluated as it is considered historical information.

(3) Doses developed to support a proposed license amendment request to delete the scram and isolation function of the Main Steam Line Radiation Monitor as described in a letter from R.M. Krich (ComEd) to U.S. NRC, "Request for an Amendment to Technical Specifications For Elimination of Main Steam Line Radiation Monitor Isolation and Scram Functions," dated December 30, 1999

Table 9-3

FHA Radiological Consequences

Location	UFSAR	Current	EPU	Limit
Fuel Handling Accident (Single fuel bundle and handling equipment dropped)				
Offsite:				
Whole Body Dose, rem	5.9E-3(1)	(2)	(2)	≤ 25
Thyroid Dose, rem	4.1E-3(1)	(2)	(2)	≤ 300
Exclusion Area:				
Whole Body Dose, rem	(1)	0.358(3)	0.422	≤ 6.25
Thyroid Dose, rem	(1)	9.92(3)	12.6	≤ 75
Low Population Zone:				
Whole Body Dose, rem	(1)	3.8E-2(3)	4.48E-2	≤ 6.25
Thyroid Dose, rem	(1)	0.687(3)	0.873	≤ 75
Control Room:				
Whole Body Dose, rem	Not	1.20E-2(3)	1.40E-2	≤ 5
Thyroid Dose, rem	reported in	7.66(3)	9.73	≤ 30
Beta Dose, rem	UFSAR	0.462(3)	0.545	≤ 30

Notes:

(1) UFSAR Table 15.7-3 lists doses as a function of distance and meteorological condition. The doses reported above are at ¼ mile under unstable 2 mph wind speed meteorological condition and represent the worst case values reported.

(2) Not evaluated as it is considered historical information.

(3) Doses developed to support proposed conversion to Improved Technical Specifications (ITS) as described in a letter from R.M. Krich (ComEd) to U.S. NRC, "Request for Technical Specifications Changes for Dresden Nuclear Power Station, Units 2 and 3, LaSalle County Station, Units 1 and 2, and Quad Cities Nuclear Power Station, Units 1 and 2, to Implement Improved Standard Technical Specifications," dated March 3, 2000

10 ADDITIONAL ASPECTS OF EPU

10.1 High Energy Line Break

Operation at the EPU level requires an increase in the steam and feedwater flows. This, in turn, results in a small increase in the mass and energy release rates following high energy line breaks. Evaluation of these piping systems determined that there is no change in postulated break locations.

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

At the EPU RTP level, the mass and energy releases for high energy line breaks outside the primary containment can increase, potentially causing the sub-compartment pressure and temperature profiles to increase. The relative humidity change is negligible. In most cases, the increase in the blowdown rate is small and the resulting profiles are generally bounded by the existing profiles due to the conservatism in the current HELB analyses. The HELBs evaluated are the:

- Main Steam System Line Break;
- Feedwater System Line Break;
- ECCS Line Breaks;
- RCIC System Line Break;
- RWCU System Line Break; and
- Instrument Line Break.

Pipe Whip and Jet Impingement:

The following addresses the effects of jet impingement from high energy lines, as addressed in UFSAR Section 3.6.

Existing calculations supporting the dispositions of potential targets of pipe whip and jet impingement from the postulated HELBs were reviewed for the effect of EPU. The review shows that higher loads/piping stresses in main steam and small changes in pressure in the Feedwater line have been evaluated for jet impingement loads and found to be acceptable. For the remaining high energy systems, existing pipe whip, and jet impingement loadings remain bounding for the EPU.

Therefore, the existing pipe whip restraints and jet impingement shields, and their supporting structures are adequate for EPU.

Internal Flooding from HELB:

The HELB analysis evaluation for flooding in the main steam tunnel due to a Main Steam or Feedwater pipe break assumes flooding of the entire below grade volume. This analysis approach is conservative and remains bounding for EPU.

10.2 Moderate Energy Line Break

The design basis for Moderate Energy Line Break (MELB) protection features at Quad Cities is based on system parameters unchanged by EPU. Therefore, MELB is not affected by EPU for Quad Cities.

10.3 Environmental Qualification

The safety-related electrical equipment environmental qualification documentation was reviewed to assure the existing qualification for the normal and accident conditions expected in the area where the devices are located remain adequate. Conservatisms in accordance with the original qualification program were applied to the environmental parameters and no change is needed for EPU.

The changes (radiation, pressure, temperature and humidity, as applicable) to the environmental conditions of affected safety-related equipment inside and outside containment were evaluated. This evaluation of equipment qualification for EPU conditions identified some equipment potentially affected by EPU conditions. The qualification of this equipment was resolved by refined radiation calculations or by the use of new test data.

10.4 Required Testing

Compared to the initial startup program, and consistent with the NRC-approved generic EPU guideline, EPU requires only limited subset of the original startup test program. As applicable to this plant's design, testing for EPU is consistent with the generic guideline.

The same performance criteria will be used as in the original power ascension tests, unless they have been replaced by updated criteria since the initial test program. Because neither steam pressure or recirculation flow have been changed for the uprate program, testing of system performance affected by these parameters is not necessary. Vibration testing of the main steam and feedwater piping is necessary, because of the magnitude of the increase in steam and feedwater flows and the limited piping vibration data available from the initial startup.

Steam separator-dryer performance will be monitored during power ascension testing. The test will quantify the moisture carryover from the steam separator-dryer to determine acceptable operational values. Data will be collected and evaluated at pre-uprate 100% power and at each incremental power increase during power ascension.

A summary report of the EPU program will be submitted to the NRC after the completion of the uprate test program. When applicable, the results from the uprate test program will be used to revise the operator training program to more accurately reflect the effects of EPU.

Recirculation Pump Testing:

Vibration testing of the recirculation pumps is not required because there is no change in the maximum core flow for the EPU condition.

10 CFR 50 Appendix J Testing:

The plant 10 CFR 50 Appendix J test program is required by the Technical Specifications and is described in UFSAR Section 6.2. This test program periodically pressurizes the containment (Type A test), the containment penetrations (Type B test), and the containment isolation valves and test boundary (Type C tests) to the calculated peak containment pressure (P_a), and measures leakage. For EPU, P_a changes to 43.9 psig. Therefore, the 10 CFR 50 Appendix J test program will be revised to reflect this calculated peak containment pressure value.

Main Steam Line and Feedwater Piping Flow Induced Vibration Testing:

The piping vibration levels of two large piping systems within containment for each plant will be monitored during initial plant operation at the new EPU operating conditions. The startup vibration test program performed for each unit is expected to show that these piping systems are vibrating at acceptable levels during EPU conditions. The two piping systems that are affected by an EPU that must be monitored for vibrations for each plant are the Main Steam Line system piping and the Feedwater system piping. These two piping systems will be monitored for vibration, because the mass flow rates in these piping systems will increase noticeably during EPU operations. As part of the piping vibration test program, a Test Specification, Test Plan and Procedure, Preliminary Test Report and Final Test Report will be prepared, to properly direct and document each phase of this test program, which will be performed for each unit.

10.5 Individual Plant Evaluation

The plant uses a probabilistic risk/safety assessment (PRA/PSA) to comply with the Individual Plant Evaluation (IPE) requirement. Consistent with Section 5.11.11 of ELTR1 (Reference 1), the plant-specific PRA/PSA was assessed (reviewed) for the effect of EPU. This review concludes that EPU does not introduce any new vulnerability, and thus, EPU has negligible impact on plant risk. The increase in the current Core Damage Frequency (CDF) of 4.61E-06/yr due to EPU implementation is conservatively estimated as 2.4E-7/yr (5% of the current CDF value). The increase in the Large Early Release Frequency (LERF) of 3.3E-06/yr due to EPU implementation is conservatively estimated as 1.3E-07/yr (4% of the current LERF value). The increase is due to shortened operator response times for certain scenarios, and to a change from one to two relief valves needed for emergency depressurization scenarios.

10.6 Operator Training and Human Factors

Before EPU operation is initiated, training required to operate the plant at EPU conditions will be provided. 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 EPU conditions, operator actions for transients, accident and special events do not change, because EPU does not change any of the automatic plant safety functions or the nature of the response. However, some of the assumed operator response times are slightly reduced. Training on these scenarios and the changes in response times will be provided.

Data obtained during startup testing will be incorporated into additional training as needed. The classroom training will cover various aspects of EPU including changes to parameters, setpoints, scales, procedures, systems and startup test procedures. The classroom training will be combined with simulator training. The simulator training will include, as a minimum, a demonstration of transients that show the greatest change in plant response at EPU power 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 (i.e., Equipment Qualification, Flow Accelerated Corrosion) to assess age-related component changes. Equipment qualification is addressed in Section 10.3, and flow accelerated corrosion is addressed in Sections 3.5 and 3.11. 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 Other Applicable Requirements

The analysis, design, and implementation of EPU was reviewed for compliance with the current plant licensing basis acceptance criteria and for compliance with new regulatory requirements and operating experience in the nuclear industry. Generic reviews of the BWR EPU program for compliance with regulatory requirements and industry communications were performed, and these reviews identified the issues that are generically evaluated and issues to be evaluated on a plant-unique basis. The applicable plant-unique evaluations have been performed for the subjects addressed below.

All of the issues from the following subjects are either generically evaluated or are evaluated on a plant-specific basis as part of the EPU program. These evaluations conclude that every issue (1) is not affected by EPU, (2) is already incorporated into the generic EPU program, or (3) is bounded by the plant-specific EPU evaluations. The NRC and industry communications evaluated cover the subjects listed below.

Code of Federal Regulations (CFRs)

NRC TMI Action Items

Action Items (Formerly Unresolved Safety Issues)

NRC Regulatory Guides

NRC Generic Letters

NRC Bulletins

NRC Information Notices

NRC Circulars

INPO Significant Operating Reports (applicable to EPU)

GE Services Information Letters

GE Rapid Information Communication Service Information Letters

Other plant-unique items whose previous evaluations could be affected by operation at the EPU level are being reviewed. These are (1) the NRC and industry communications discussed above, (2) the safety evaluations for work in progress and not yet integrated into the plant design, (3) the temporary modifications that could have been reviewed prior to the EPU and still exist after EPU implementation, and (4) the plant emergency operating procedures (EOPs). These items will be reviewed for possible effect by the EPU, and will be found to be either acceptable for EPU, or will be revised to reflect EPU conditions.

11.2 Impact on Technical Specifications

Implementation of EPU with ARTS power and flow dependent limits requires revision of a number of the Technical Specifications (TS). Table 11-1 contains a list of TS items that are changed to implement EPU and ARTS power and flow dependent limits. A brief description of the nature of each change is also provided. The evaluations summarized in this report provide the justifications for these TS changes.

11.3 Environmental Assessment

ARTS power and flow dependent limits are not related to any plant release, and thus, have no environmental impact.

The environmental effects of EPU will be controlled at the same levels as for the current analyses. None of the present limits for plant environmental releases, such as ultimate heat sink temperature or plant vent radiological limits, will be increased as a consequence of EPU. The environment assessment concludes the effects of EPU will be insignificant, because the normal effluents and doses will remain well within 40 CFR 190, 10 CFR 20 and 10 CFR 50, Appendix I limits.

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 Quad Cities ARTS power and flow dependent limits program is effectively the same program as the Partial ARTS program implemented at the LaSalle County Station units. The LaSalle program is documented in Reference 2, and was approved in Reference 3.

All significant safety analyses and evaluations have been performed, and their results justify an extended power uprate (EPU) of 17.8% to 2957 MWt.

The ARTS power and flow dependent limits program has the specific objectives of increasing plant operating efficiency, and updating thermal limits requirements and administration. The analyses summarized herein provide the analytical basis for the following changes associated with the ARTS power and flow dependent limits program:

- Implementation of power- and flow-dependent fuel thermal limits to support elimination of the APRM gain and setpoint requirements.
- Maintaining the RBM operability requirements in terms of the measurable core thermal limit performance parameter, MCPR.

11.4.1.1 Modification Summary

An increase in electrical output of a BWR plant is accomplished primarily by generation and supply of higher steam flow for the turbine generator. Most BWR plants, as currently licensed, have an as-designed equipment and system capability to accommodate steam flow rates at least 5% above the original rating. In addition, continuing improvements in the analytical techniques (computer codes and data) based on several decades of BWR safety technology, plant performance feedback, 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 the capability to increase their thermal power ratings of between 5 and 10% without major nuclear steam supply system (NSSS) hardware modifications, and to provide for power increases to 20% with limited hardware modifications, with no significant increase in the hazards presented by the plant as approved by the NRC at the original license stage.

The plan for achieving higher power is to modestly expand the power flow map and increase core flow along standard Maximum Extended Load Line Limit Analysis (MELLLA) flow control lines. However, there is no increase in the maximum recirculation flow limit or operating pressure over the pre-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.

The ARTS improvements provide changes to the APRM system. The reactor limits, instrument setpoints, operability requirement and Technical Specification changes associated with the ARTS improvements are provided in Table 11-1.

The objective of the APRM improvements is to justify removal of the APRM gain and setpoint (trip setdown) requirement. Two licensing areas, which can be impacted by the elimination of the gain and setpoint requirement, are fuel thermal-mechanical integrity and ECCS-LOCA performance.

The following criteria ensure the satisfaction of the applicable licensing requirements, and were applied to demonstrate the acceptability of elimination of the APRM gain and setpoint requirement:

- The Safety Limit MCPR shall not be violated as a result of any AOOs.
- All fuel thermal-mechanical design bases shall remain within the licensing limits described in the GE generic fuel licensing report.
- Peak cladding temperature and maximum cladding oxidation fraction following a LOCA shall remain within the limits defined in 10 CFR 50.46.

The safety analyses used to evaluate the Operating Limit MCPR (OLMCPR), such that the SLMCPR will not be violated and to ensure that the fuel thermal-mechanical design bases are satisfied, are documented in Section 9.2. These analyses also establish the fuel type specific power- and flow-dependent MCPR and LHGR curves for Quad Cities. The effect on the ECCS-LOCA response due to both the expansion of the power/flow map and the implementation of the ARTS improvement is discussed in Section 4.3.

The following changes result from the ARTS power and flow dependent limits improvement program:

1. Delete the requirement for setdown of the APRM scram and rod blocks.
2. Add new power-dependent MCPR adjustment factors, MCPR(P).
3. Replace K_F with the new flow-dependent MCPR adjustment factors, MCPR(F).
4. Add new power-dependent LHGR adjustment factors, LHGRFAC(P).
5. Add new flow-dependent LHGR adjustment factors, LHGRFAC(F).
6. Delete or modify affected Technical Specifications and Bases.

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 Uprate Analysis Basis

Quad Cities is currently licensed for a 100% power level of 2511 MWt. The current safety analysis basis assumes that the reactor had been operating continuously at the licensed power level, except for the ECCS-LOCA and short-term containment analyses, which were performed at 102% of licensed thermal power. The EPU increases the rated thermal power (RTP) by 17.8% of the originally licensed value. The EPU with ARTS power and flow dependent limits safety analyses are based on a power level of at least 1.02 times the EPU power level, except that some analyses are performed at 100% rated power, because the Regulatory Guide 1.49 2% power factor is already accounted for in the analysis methods.

11.4.2.2 Margins

The above EPU analysis basis ensures that the power dependent margins prescribed by the Code of Federal Regulations (CFR) are maintained by meeting the appropriate regulatory criteria. Similarly, design margins specified by application of the American Society of Mechanical Engineers (ASME) design rules are maintained, as are other margin-ensuring criteria used to judge the acceptability of the plant. Environmental margins are maintained by not increasing any of the present limits for releases, such as ultimate heat sink maximum temperature or plant vent radiological limits.

11.4.2.3 Fuel Thermal Limits

No change is required in the basic fuel design to achieve the EPU power level, implement ARTS power and flow dependent limits improvements or to meet the plant licensing limits. No increase in allowable peak bundle power is requested for EPU. The current fuel operating limits will still be met at the EPU power level. 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. No new fuel design is required for EPU with ARTS power and flow dependent limits.

11.4.2.4 Makeup Water Sources

The Boiling Water Reactor 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. Consequently, 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 at least three feedwater and four condensate system pumps, the low pressure emergency core cooling system (LPCI & CS) pumps, the high pressure emergency core cooling system (HPCI) pump, the Reactor Core Isolation Cooling (RCIC) pump/turbine, the Standby Liquid Control (SLC) pumps, and the Control Rod Drive (CRD) pumps. Many of these diverse water supply means are redundant in equipment and also redundant in systems (e.g., there are several pumps and complete redundant piping systems).

EPU with ARTS power and flow dependent limits 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 Emergency Core Cooling Systems (ECCS) during loss-of-coolant-accidents.

EPU results in a 17.8% 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.

11.4.2.5 Design Basis Accidents

Design Basis Accidents (DBAs) are very low probability 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 pipe break sizes in the largest recirculation, steam, and feedwater lines, a postulated break in one of the ECCS lines, and the most limiting small lines. This break range bounds the full spectrum of large and small, high and low energy line breaks; and the success of plant systems to mitigate the accidents, while accommodating a single active equipment failure in addition to the postulated LOCA. Several of the most significant licensing assessments are or will be made using these LOCA ground rules. These assessments are:

- Challenges to Fuel (ECCS-LOCA performance evaluation) in accordance with the rules and criteria of 10 CFR 50.46 and Appendix K wherein the predominant criterion is the fuel peak cladding 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 (cooling) pool temperature for long-term cooling in accordance with 10 CFR 50 Appendix A Criterion 38.
- DBA Radiological Consequences (calculated and compared to the criteria of 10 CFR 100.

11.4.2.6 Challenges to Fuel

The ECCS-LOCA performance evaluation (see Section 4.3) was conducted through application of the 10 CFR 50 Appendix K evaluation models, and demonstrates that EPU does not significantly affect the ECCS-LOCA performance evaluation results. The LOCA evaluations with the equilibrium cycle core of GE14 fuel demonstrate compliance with the ECCS acceptance criteria. The licensing safety margin will not be affected by EPU. The slightly (< 10°F) increased PCTs for EPU are insignificant. Therefore, the ECCS safety margin will not be affected by EPU.

The ARTS power and flow dependent limits do not affect ECCS-LOCA performance evaluation.

11.4.2.7 Challenges to the Containment

The effect of EPU on the peak values for containment pressure and temperature confirms the suitability of the plant for operation at EPU power. Also, the effect of 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 EPU are within the range of conditions used to define the current dynamic loads, current safety criteria are met and no further structural analysis is required. The change in short-term containment response is negligible. Because there will be more residual heat with 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.

ARTS power and flow dependent limits do not affect the Containment analysis.

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 atmospheric dispersion factors and the dose exposure pathways do not change. Therefore, the only factor, which could influence the magnitude of the consequences, is the quantity of activity released to the environment. This quantity is a product of the activity released from the core or reactor coolant and the transport mechanisms between the source region and the effluent release point. The transport mechanisms between the source region and the effluent release point are unchanged by EPU.

For EPU, the events evaluated are the Loss-of-Coolant-Accident (LOCA), the Main Steam Line Break Accident (MSLBA) outside containment, the Fuel Handling Accident (FHA), the Control Rod Drop Accident (CRDA), the Instrument Line Break (ILB) and the Offgas Treatment System Component Failure.

The EPU will not change the radiological consequences of a MSLBA outside containment, since the mass and energy releases following a MSLBA remain unaffected by uprate, and the activity released is based on primary coolant at Technical Specification levels, which is also unaffected by EPU.

The EPU will not change the radiological consequences of an ILB outside containment since the reactor coolant mass release used in the current analysis envelopes the post-EPU conditions, and the activity released is based on primary coolant at Technical Specification levels which is unaffected by EPU.

The EPU will not change the radiological consequences of an Offgas Treatment System Component Failure since a conservative source term was used in the original analysis.

For the remaining DBAs, the primary parameter of importance is the activity released from the fuel. Because the mechanism of fuel failure is not influenced by EPU, the only parameter of importance is the actual inventory of fission products in the fuel rod. The only parameters affecting fuel inventory are the increase in thermal power, and to some extent, the cycle length.

The DBA that has historically been limiting from a radiological viewpoint is the LOCA, for which USNRC Regulatory Guide 1.3, or its equivalent, has been applied. For this accident, it is assumed that 100% of the noble gases and 50% of the iodines in the core are released to the primary containment. These release fractions are not influenced by EPU or cycle length. The LOCA dose consequences remain below regulatory guidelines.

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

ARTS power and flow dependent limits do not affect any radiological analysis, and thus, the consequences of all accidents are not affected.

11.4.2.9 Transient Analyses

The effects of plant transients are evaluated (in Section 9.1) 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 are primarily evaluated against the Safety Limit Minimum Critical Power Ratio (SLMCPR). The most limiting transient is slightly more severe when initiated from the EPU RTP level, and results in a slightly larger change in MCPR than that initiated from the current power level. The Operating Limit MCPR is increased appropriately to assure that the SLMCPR is not infringed upon, if any transient is initiated from the EPU RTP level. Plus, the limiting transients are analyzed for each specific fuel cycle. Licensing acceptance criteria are not exceeded. Therefore, the margin of safety is not affected by EPU.

Use of the ARTS related power and flow dependent MCPR limits ensures that the SLMCPR will not be exceeded.

11.4.2.10 Combined Effects

EPU analyses use fuel designed to current NRC-approved criteria and operated within NRC-approved limits to produce more power in the reactor, and thus, increases steam flow to the turbine. NRC-approved design criteria are used to assure equipment mechanical performance at EPU conditions. Scram frequency is minimized by small adjustments to reactor instrumentation. These adjustments are attributed to the small changes in the reactor operating conditions. DBAs are hypothesized to evaluate challenges to the fuel, containment and off-site dose limits. These challenges have been evaluated separately in accordance with extremely conservative regulatory procedures such that the separate effects are more severe than any combined effects. The off-site dose evaluation, in accordance with Regulatory Guide 1.3 and SRP-15.6.5, calculates more severe DBA radiological consequences than the combined effects of the hypothetical LOCA producing the greatest challenge to the fuel and/or containment. That is, the DBA producing the highest PCT and/or containment pressure, does not damage the large amounts of fuel assumed in the off-site dose evaluation. Therefore, the combined effects of the most severe hypothetical LOCA are conservatively bounded by the off-site dose evaluation.

11.4.2.11 Non-LOCA Radiological Release Accidents

All of the other radiological releases discussed in Regulatory Guide 1.70 UFSAR Chapters 11 and 15 are either unchanged because they are not power-dependent, or increase at most by the amount of the EPU. The dose consequences for all of the radiological release accident events are bounded by the "Design Basis Radiological Consequences" events discussed above.

11.4.2.12 Equipment Qualification

Plant Equipment and Instrumentation has been evaluated against the criteria appropriate for EPU. Significant groups/types of the equipment have been justified for EPU by generic evaluations. Some of the qualification testing/justification at the current power level was done at more severe conditions than the minimum required. In some cases, the qualification envelope did not change significantly due to EPU. A process has been developed to ensure qualification of the equipment whose current qualification does not already bound 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 EPU in a manner comparable to that for safety-related NSSS systems/equipment. Generic and plant-specific evaluations justify EPU operation for BOP systems/equipment. Modifications (e.g., turbine modifications) will be made (via 10 CFR 50.59) where needed to fully implement EPU.

11.4.2.14 Environmental Consequences

The environmental effects of EPU can be controlled below the same permitted limits as for the current power level. Monitoring of river temperatures will occur at higher river flows to demonstrate compliance with the current state thermal discharge limits. None of the present ultimate heat sink temperature or plant vent radiological release limits are increased as a results of EPU.

11.4.2.15 Technical Specifications Changes

The Technical Specifications (TS) ensure that plant and system performance parameters are maintained within the values assumed in the safety analyses. That is, the TS parameters (setpoints, 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 TS changes justified by the safety analyses summarized in these reports are listed in Table 11-1. Proper account is taken of inaccuracies introduced by instrument accuracy and calibration accuracy. This assures that the actual plant responses will be less severe than those represented by the safety analysis. Similarly, the TS address equipment operability (availability) and put limits on equipment out-of-service (not available for use) times such that the actual plant can be expected to have at least the complement of equipment available to mitigate abnormal plant events assumed in the safety analyses. Because the safety analyses for EPU with ARTS power and flow dependent limits show that the results are acceptable within regulatory limits, public health and safety is confirmed. TS changes consistent with the EPU power level and the ARTS power and flow dependent limits improvements are made in accordance with methodology already approved for the plant and continue to provide a comparable level of protection as TS 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 Quad Cities 117.8% extended power uprate (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.

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

As summarized below, the increase in power level with ARTS power and flow dependent limits improvements discussed herein will not significantly increase the probability or consequences of an accident previously evaluated.

The probability (frequency of occurrence) of Design Basis Accidents occurring is not affected by the increased power level or by the ARTS power and flow dependent limits, because plant equipment still complies with the applicable regulatory and design basis criteria. An evaluation of the BWR probabilistic risk assessments concludes that the calculated core damage frequencies do not significantly change due to EPU or ARTS power and flow dependent limits. Scram setpoints (i.e., equipment settings that initiate automatic plant shutdowns) are established such that there is no significant increase in scram frequency due to uprate. No new challenges to safety-related equipment result from EPU or ARTS power and flow dependent limits.

Radiological release events (accidents) have been evaluated, and shown to meet the guidelines of 10 CFR 100. Therefore, the changes in consequences of hypothetical accidents are in all cases insignificant. The EPU accident evaluation results do not exceed any of their NRC-approved acceptance limits. The spectrum of hypothetical accidents and transients has been investigated, and are shown to meet the plant's currently licensed regulatory criteria. In the area of core design, for example, the fuel operating limits such as Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) and Safety Limit Minimum Critical Power Ratio (SLMCPR) are still met, and fuel reload analyses will show that plant transients meet the criteria accepted by the NRC as specified in NEDO-24011, "GESTAR II." Challenges to fuel (ECCS performance) are evaluated, and shown to still meet the criteria of 10 CFR 50.46 and Appendix K.

ARTS power and flow dependent limits do not affect a radiological analysis result from any postulated accident, nor does it affect the containment analysis.

Challenges to the containment have been evaluated, and the containment and its associated cooling systems continue to meet 10 CFR 50 Appendix A Criterion 38, Long Term Cooling, and Criterion 50, Containment.

Radiological release events (accidents) have been evaluated, and shown to meet the guidelines of 10 CFR 100.

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

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

Equipment that could be affected by EPU or ARTS power and flow dependent limits has been evaluated. No new operating mode, safety-related equipment lineup, accident scenario or equipment failure mode is involved with EPU. The full spectrum of accident considerations, defined in Regulatory Guide 1.70, has been evaluated, and no new or different kind of accident has been identified. EPU and ARTS power and flow dependent limits use already developed technologies, and apply them within the capabilities of already existing plant equipment in accordance with presently existing regulatory criteria. Industry experience with ARTS and BWRs with higher power levels than described herein have not identified any new power dependent or ARTS related accident.

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.

EPU only affects design and operational margins. Challenges to the fuel, reactor coolant pressure boundary, and containment were reanalyzed for EPU conditions. The fuel integrity is maintained by meeting existing design and regulatory limits. The calculated loads of all affected structures, systems and components, including the reactor coolant pressure boundary, remain within design allowables for all design basis event categories. The containment performance analysis demonstrates that the containment remains within all of its design limits following the most severe design basis accident.

The use of ARTS power and flow dependent limits improvements ensures that the plant does not exceed any fuel thermal limit, and thus, the margin of safety is not affected.

Because the plant reactions to transients and hypothetical accidents do not result in exceeding the presently approved NRC acceptance limits, EPU with ARTS power and flow dependent limits does not involve a significant reduction in a margin of safety.

Conclusions:

An EPU to 117.8% of original rated power with ARTS power and flow dependent limits has been investigated. The method for achieving higher power is to slightly increase some plant operating parameters. The plant licensing challenges have been evaluated and demonstrate how this uprate with ARTS power and flow dependent limits 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 power uprate of the amount described herein and ARTS power and flow dependent limits do not involve a Significant Hazards Consideration.

Table 11-1

Technical Specifications Affected By EPU With ARTS

TS Location	Description of Change
1.1 Definitions	Delete the definition of Fuel Design Limiting Ratio For Centerline Melt (FDLRC), because this definition is no longer applicable with the implementation of the ARTS related changes, discussed in Sections 1.4 and 9.2. Revise the value of Rated Thermal Power (RTP) definition to EPU power level (2957 MWt) shown in Table 1-2.
3.2.4	Delete TS 3.2.4 (entirely), as the APRM Gain and Setpoint requirement are superseded by the ARTS related changes, discussed in Sections 1.4 and 9.2.
SR 3.3.1.1.2	Delete reference to LCO 3.2.4, because TS 3.2.4 is deleted due to ARTS changes.
SR 3.3.1.1.13, Table 3.3.1.1-1 Functions 8 and 9	Reduce the RPS TSV-Closure and TCV Fast Closure scram bypass power level from 45% RTP to 38.5% RTP, to maintain approximately the same absolute thermal power value.
3.3.1.1 Required Action E.1	Revise action %RTP value to be consistent with the RPS %RTP Bypass value from 45% RTP to 38.5% RTP, to maintain approximately the same absolute thermal power value.
Table 3.3.1.1-1 Function 2.b.	Revise the APRM Flow Biased scram equations for two and single recirculation loop operation, consistent with the discussion in Section 5.3. Revise the allowable value for the APRM TLO clamped scram from 120% RTP to 122% RTP, based on Reference 4.
Table 3.3.1.1-1 Function 2.c	Revise the allowable value for the APRM fixed neutron flux – high from 120% RTP to 122% RTP, based on Reference 4.
Table 3.3.1.1-1 Function 4.	As discussed in Section 5.3, revise the Reactor Vessel Water Level – Low scram Allowable Value from ≥ 11.8 inches to ≥ 3.8 inches, based on the revised analytical limit.
Table 3.3.1.1-1 Function 10.	As discussed in Section 5.3, revise the Turbine Condenser Vacuum – Low scram Allowable Value from ≥ 21.8 inches Hg vacuum to ≥ 21.4 inches Hg vacuum.
Table 3.3.6.1-1 Function 2a.	As discussed in Section 5.3, revise the Reactor Vessel Water Level – Low primary containment isolation Allowable Value from ≥ 11.8 inches to ≥ 3.8 inches, based on the revised analytical limit.

TS Location	Description of Change
Table 3.3.6.1-1 Function 5b.	As discussed in Section 5.3, revise the Reactor Vessel Water Level – Low Reactor Water Cleanup system isolation Allowable Value from ≥ 11.8 inches to ≥ 3.8 inches, based on the revised analytical limit.
Table 3.3.6.1-1 Function 6b.	As discussed in Section 5.3, revise the Reactor Vessel Water Level – Low RHR Shutdown Cooling system isolation Allowable Value from ≥ 11.8 inches to ≥ 3.8 inches, based on the revised analytical limit provided in Table 5-1.
Table 3.3.6.1-1 Function 1.d.	As discussed in Section 5.3, revise the Main Steam Line Flow – High Main Steam Line Isolation Allowable Value from < 138% rated steam flow to < 254.3 psid.
Table 3.3.6.2-1 Function 1.	As discussed in Section 5.3, revise the Reactor Vessel Water Level – Low secondary containment system isolation Allowable Value from ≥ 11.8 inches to ≥ 3.8 inches, based on the revised analytical limit provided in Table 5-1.
Table 3.3.7.1-1 Function 1.	As discussed in Section 5.3, revise the Reactor Vessel Water Level – Low Control Room Emergency Ventilation (CREV) system isolation Allowable Value from ≥ 11.8 inches to ≥ 3.8 inches, based on the revised analytical limit provided in Table 5-1.
Table 3.3.7.1-1 Function 3.	As discussed in Section 5.3, revise the Main Steam Line Flow – High Control Room Emergency Ventilation (CREV) system isolation Allowable Value from < 138% rated steam flow to < 254.3 psid.
3.5.1	To be consistent with the ECCS-LOCA analysis (Section 4.3), the number of operable relief function valves is increased from four relief valves to four relief valves and one safety/relief valve (SRV).
(New) SR 3.5.1.12	To ensure the operability of the relief function of the Target Rock SRV, add a new surveillance that states "Verify ADS pneumatic supply header pressure is ≥ 80 psig." This surveillance to be performed every 31 days. This is based on Reference 5.
5.5.12	Based the containment performance analysis addressed in Section 4.1, revise the "Pa" value to be equal to the peak calculated containment pressure of 43.9 psig, as discussed in Section 10.4.
5.6.5, Item a.4	Delete Item a.4, because it is based on TS 3.2.4, which is deleted due to ARTS related changes.

12 References

1. GE Nuclear Energy, "Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate," (ELTR1), Licensing Topical Report NEDO-32424, Class I (Non-proprietary), April 1995.

2. Letter from Gary G. Benes (Nuclear Licensing Administrator, Commonwealth Edition) to William T. Russell (Director, USNRC), "LaSalle County Nuclear Power Station Units 1 and 2 Application for Amendment Request to Facility Operating Licenses NPF-11 and NPF-18, Appendix A, Technical Specifications Partial ARTS Implementation NRC Docket Nos. 50-373 and 50-374," June 9, 1994.

3. Letter from William D. Reckley (Project Manager, USNRC) to D. L. Farrar (Manager, Commonwealth Edition Company), "Issuance of Amendments (TAC Nos. M89631 and M89632)," April 13, 1995.

4. Letter from R. M. Krich (Exelon Generation Company) to U. S. NRC, "Supplement to Request for License Amendment for Power Uprate Operation," dated April 13, 2001.

5. Letter from R. M. Krich (Exelon Generation Company) to U. S. NRC, "Supplement to GE14 Fuel License Amendment Request," dated August 13, 2001.

Attachment E
Safety Analysis Reports Supporting the License Amendment Request to Permit Uprated Power Operation
Dresden Nuclear Power Station, Units 2 and 3,
Quad Cities Nuclear Power Station, Units 1 and 2

GE Affidavit for Withholding NEDC-32961P and NEDC-32962P from Public Disclosure

General Electric Company

AFFIDAVIT

I, **David J. Robare,** being duly sworn, depose and state as follows:

(1) I am Technical Projects Manager, Technical Services, General Electric Company ("GE") and have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld is contained in the GE proprietary report NEDC-32962P, DRF A22-00103-13, *Safety Analysis Report for Dresden 2 & 3 Extended Power Uprate*, Revision 2, Class III (GE Proprietary Information), dated August 2001. This document, taken as a whole, constitutes a proprietary compilation of information, some of it also independently proprietary, prepared by the General Electric Company. The independently proprietary elements are identified by bars marked in the left margin adjacent to the specific material.

(3) In making this application for withholding of proprietary information of which it is the owner, GE relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC Sec. 552(b)(4), and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10 CFR 9.17(a)(4), 2.790(a)(4), and 2.790(d)(1) for "trade secrets and commercial or financial information obtained from a person and privileged or confidential" (Exemption 4). The material for which exemption from disclosure is here sought is all "confidential commercial information", and some portions also qualify under the narrower definition of "trade secret", within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975F2d871 (DC Cir. 1992), and Public Citizen Health Research Group v. FDA, 704F2d1280 (DC Cir. 1983).

(4) Some examples of categories of information which fit into the definition of proprietary information are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by General Electric's competitors without license from General Electric constitutes a competitive economic advantage over other companies;

b. Information which, if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product;

c. Information which reveals cost or price information, production capacities, budget levels, or commercial strategies of General Electric, its customers, or its suppliers;

d. Information which reveals aspects of past, present, or future General Electric customer-funded development plans and programs, of potential commercial value to General Electric;

e. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection.

Both the compilation as a whole and the marked independently proprietary elements incorporated in that compilation are considered proprietary for the reason described in items (4)a. and (4)b., above.

(5) The information sought to be withheld is being submitted to NRC in confidence. That information (both the entire body of information in the form compiled in this document, and the marked individual proprietary elements) is of a sort customarily held in confidence by GE, and has, to the best of my knowledge, consistently been held in confidence by GE, has not been publicly disclosed, and is not available in public sources. All disclosures to third parties including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence. Its initial designation as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in paragraphs (6) and (7) following.

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge. Access to such documents within GE is limited on a "need to know" basis.

(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist or other equivalent authority, by the manager of the cognizant marketing function (or his delegate), and by the Legal Operation, for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside GE are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary agreements.

(8) The information identified by bars in the margin is classified as proprietary because it contains detailed results and conclusions from these evaluations, utilizing analytical models and methods, including computer codes, which GE has developed, obtained NRC approval of, and applied to perform evaluations of transient and accident events in the GE Boiling Water Reactor ("BWR"). The development and approval of these system, component, and thermal hydraulic models and computer codes was achieved at a significant cost to GE, on the order of several million dollars.

The remainder of the information identified in paragraph (2), above, is classified as proprietary because it constitutes a confidential compilation of information, including detailed results of analytical models, methods, and processes, including computer codes, and conclusions from these applications, which represent, as a whole, an integrated process or approach which GE has developed, obtained NRC approval of, and applied to perform evaluations of the safety-significant changes necessary to demonstrate the regulatory acceptability of a given increase in licensed power output for a GE BWR. The development and approval of this overall approach was achieved at a significant additional cost to GE, in excess of a million dollars, over and above the very large cost of developing the underlying individual proprietary analyses.

To effect a change to the licensing basis of a plant requires a thorough evaluation of the impact of the change on all postulated accident and transient events, and all other regulatory requirements and commitments included in the plant's FSAR. The analytical process to perform and document these evaluations for a proposed power uprate was developed at a substantial investment in GE resources and expertise. The results from these evaluations identify those BWR systems and components, and those postulated events, which are impacted by the changes required to accommodate operation at increased power levels, and, just as importantly, those which are <u>not</u> so impacted, and the technical justification for not considering the latter in changing the licensing basis. The scope thus determined forms the basis for GE's offerings to support utilities in both performing analyses and providing licensing consulting services. Clearly, the scope and magnitude of effort of any attempt by a competitor to effect a similar licensing change can be narrowed considerably based upon these results. Having invested in the initial evaluations and developed the solution strategy and process described in the subject document GE derives an important competitive advantage in selling and performing these services. However, the mere knowledge of the impact on each system and component reveals the process, and provides a guide to the solution strategy.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to GE's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of GE's comprehensive BWR technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive

physical database and analytical methodology and includes development of the expertise to determine and apply the appropriate evaluation process. In addition, the technology base includes the value derived from providing analyses done with NRC-approved methods, including justifications for not including certain analyses in applications to change the licensing basis.

GE's competitive advantage will be lost if its competitors are able to use the results of the GE experience to avoid fruitless avenues, or to normalize or verify their own process, or to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions. In particular, the specific areas addressed by any document and submittal to support a change in the safety or licensing bases of the plant will clearly reveal those areas where detailed evaluations must be performed and specific analyses revised, and also, by omission, reveal those areas not so affected.

While some of the underlying analyses, and some of the gross structure of the process, may at various times have been publicly revealed, enough of both the analyses and the detailed structural framework of the process have been held in confidence that this information, in this compiled form, continues to have great competitive value to GE. This value would be lost if the information as a whole, in the context and level of detail provided in the subject GE document, were to be disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources, including that required to determine the areas that are not affected by a power uprate and are therefore blind alleys, would unfairly provide competitors with a windfall, and deprive GE of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing its analytical process.

STATE OF CALIFORNIA)
) ss:
COUNTY OF SANTA CLARA)

David J. Robare, being duly sworn, deposes and says:

That he has read the foregoing affidavit and the matters stated therein are true and correct to the best of his knowledge, information, and belief.

Executed at San Jose, California, this 30TH day of AUGUST 2001.

David J. Robare
General Electric Company

Subscribed and sworn before me this 30th day of August 2001.

TERRY J. MORGAN
Commission # 1304914
Notary Public - California
Santa Clara County
My Comm. Expires May 18, 2005

Notary Public, State of California

General Electric Company

AFFIDAVIT

I, **David J. Robare,** being duly sworn, depose and state as follows:

(1) I am Technical Projects Manager, Technical Services, General Electric Company ("GE") and have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld is contained in the GE proprietary report NEDC-32961P, DRF A22-00103-13, *Safety Analysis Report for Quad Cities 1 & 2 Extended Power Uprate*, Revision 2, Class III (GE Proprietary Information), dated August 2001. This document, taken as a whole, constitutes a proprietary compilation of information, some of it also independently proprietary, prepared by the General Electric Company. The independently proprietary elements are identified by bars marked in the left margin adjacent to the specific material.

(3) In making this application for withholding of proprietary information of which it is the owner, GE relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC Sec. 552(b)(4), and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10 CFR 9.17(a)(4), 2.790(a)(4), and 2.790(d)(1) for "trade secrets and commercial or financial information obtained from a person and privileged or confidential" (Exemption 4). The material for which exemption from disclosure is here sought is all "confidential commercial information", and some portions also qualify under the narrower definition of "trade secret", within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975F2d871 (DC Cir. 1992), and Public Citizen Health Research Group v. FDA, 704F2d1280 (DC Cir. 1983).

(4) Some examples of categories of information which fit into the definition of proprietary information are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by General Electric's competitors without license from General Electric constitutes a competitive economic advantage over other companies;

b. Information which, if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product;

c. Information which reveals cost or price information, production capacities, budget levels, or commercial strategies of General Electric, its customers, or its suppliers;

d. Information which reveals aspects of past, present, or future General Electric customer-funded development plans and programs, of potential commercial value to General Electric;

e. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection.

Both the compilation as a whole and the marked independently proprietary elements incorporated in that compilation are considered proprietary for the reason described in items (4)a. and (4)b., above.

(5) The information sought to be withheld is being submitted to NRC in confidence. That information (both the entire body of information in the form compiled in this document, and the marked individual proprietary elements) is of a sort customarily held in confidence by GE, and has, to the best of my knowledge, consistently been held in confidence by GE, has not been publicly disclosed, and is not available in public sources. All disclosures to third parties including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence. Its initial designation as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in paragraphs (6) and (7) following.

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge. Access to such documents within GE is limited on a "need to know" basis.

(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist or other equivalent authority, by the manager of the cognizant marketing function (or his delegate), and by the Legal Operation, for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside GE are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary agreements.

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The remainder of the information identified in paragraph (2), above, is classified as proprietary because it constitutes a confidential compilation of information, including detailed results of analytical models, methods, and processes, including computer codes, and conclusions from these applications, which represent, as a whole, an integrated process or approach which GE has developed, obtained NRC approval of, and applied to perform evaluations of the safety-significant changes necessary to demonstrate the regulatory acceptability of a given increase in licensed power output for a GE BWR. The development and approval of this overall approach was achieved at a significant additional cost to GE, in excess of a million dollars, over and above the very large cost of developing the underlying individual proprietary analyses.

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physical database and analytical methodology and includes development of the expertise to determine and apply the appropriate evaluation process. In addition, the technology base includes the value derived from providing analyses done with NRC-approved methods, including justifications for not including certain analyses in applications to change the licensing basis.

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While some of the underlying analyses, and some of the gross structure of the process, may at various times have been publicly revealed, enough of both the analyses and the detailed structural framework of the process have been held in confidence that this information, in this compiled form, continues to have great competitive value to GE. This value would be lost if the information as a whole, in the context and level of detail provided in the subject GE document, were to be disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources, including that required to determine the areas that are not affected by a power uprate and are therefore blind alleys, would unfairly provide competitors with a windfall, and deprive GE of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing its analytical process.

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David J. Robare, being duly sworn, deposes and says:

That he has read the foregoing affidavit and the matters stated therein are true and correct to the best of his knowledge, information, and belief.

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David J. Robare
General Electric Company

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Notary Public, State of California