ATTACHMENT 7

Safety Analysis Report for Browns Ferry Nuclear Plant Units 1, 2, and 3 Extended Power Uprate (Non-Proprietary)

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Non-Proprietary Information – Class I (Public)

SAFETY ANALYSIS REPORT

FOR

BROWNS FERRY NUCLEAR PLANT

UNITS 1, 2, AND 3

EXTENDED POWER UPRATE

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ACRONYMS AND ABBREVIATIONS

Term	Definition
AC	Alternating Current
ADHR	Auxiliary Decay Heat Removal
ADS	Automatic Depressurization System
AEC	Atomic Energy Commission
AHC	Access Hole Cover
AL	Analytical Limit
ALARA	As Low As Is Reasonably Achievable
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOI	Abnormal Operator Instruction
AOO	Anticipated Operational Occurrence (moderate frequency transient event)
AOV	Air-Operated Valve
AP	Annulus Pressurization
APRM	Average Power Range Monitor
AREVA	Areva Incorporated
ARI	Alternate Rod Insertion
ART	Adjusted Reference Temperature
ARTS	APRM/RBM/Technical Specifications
ASME	American Society of Mechanical Engineers
ASDC	Alternate Shutdown Cooling
AST	Alternate Source Term
ASTM	American Society for Testing and Materials
ATWS	Anticipated Transient Without Scram
AV	Allowable Value
AVZ	Above Vessel Zero
B-10	Boron 10

Term	Definition
BHP	Brake Horsepower
BFN	Browns Ferry Nuclear Plant
BIIT	Boron Injection Initiation Temperature
BOC	Beginning of Cycle
BOP	Balance-of-Plant
Browns Ferry	Browns Ferry Nuclear Plant (all units)
BTU	British Thermal Unit
B&W	Babcock and Wilcox
BWR	Boiling Water Reactor
BWROG	BWR Owners Group
BWRVIP	BWR Vessel and Internals Project
CAD	Containment Atmospheric Dilution
CAP	Containment Accident Pressure
CARV	Cross Around Relief Valve
CCW	Component Cooling Water
CDF	Core Damage Frequency
CF	Chemistry Factor
CFD	Condensate Filter Demineralizer
CFR	Code of Federal Regulations
CIC	Coolant Injection Cooling
CLTP	Current Licensed Thermal Power
CLTR	Constant Pressure Power Uprate Licensing Topical Report
CO	Condensation Oscillation
COLR	Core Operating Limits Report
CPPU	Constant Pressure Power Uprate
CR	Control Room
CRAVS	Control Room Area Ventilation System

Term	Definition
CRD	Control Rod Drive
CRDA	Control Rod Drop Accident
CRDH	Control Rod Drive Housing
CREVS	Control Room Emergency Ventilation System
CRGT	Control Rod Guide Tube
CRHZ	Control Room Habitability Zone
CS	Core Spray
CSBW	Cold Shutdown Boron Weight
CSC	Containment Spray Cooling
CSS	Containment Spray System
CST	Condensate Storage Tank
CUF	Cumulative Usage Factor
CWS	Circulating Water System
DBA	Design Basis Accident
DBA-LOCA	Design Basis Loss-of-Coolant Accident
DBE	Design Basis Event
DC	Direct Current
DFWCS	Digital Feedwater Control System
DHRP	Decay Heat Removal Pressure
DLO	Dual (Recirculation) Loop Operation
DOR	Division of Operating Reactors
DP	Differential Pressure (psid)
DW	Drywell
EAB	Exclusion Area Boundary
ECCS	Emergency Core Cooling System
ECCS-LOCA	Emergency Core Cooling System - Loss-of-Coolant Accident
EDG	Emergency Diesel Generator

Term	Definition
ECP	Electrochemical Potential
EECW	Emergency Equipment Cooling Water
EFDS	Equipment and Floor Drainage System
EFPY	Effective Full Power Years
EHC	Electro-Hydraulic-Control
EHPMP	Emergency High Pressure Makeup Pump
ELTR1	Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate Licensing Topical Report
ELTR2	Generic Evaluations of General Electric Boiling Water Reactor Extended Power Uprate Licensing Topical Report
EMA	Equivalent Margin Analysis
EOC	End of Cycle
EOI	Emergency Operating Instruction
EOL	End of Life
EOC-RPT	End of Cycle-Recirculation Pump Trip
EPRI	Electric Power Research Institute
EPU	Extended Power Uprate
EQ	Environmental Qualification
ESF	Engineered Safety Feature
ESFVS	Engineered Safety Feature Ventilation System
ESW	Emergency Service Water
FAC	Flow Accelerated Corrosion
FCF	Flow Correction Factor
FCV	Flow Control Valve
FFWTR	Final Feedwater Temperature Reduction
FHA	Fuel Handling Accident
FIV	Flow Induced Vibration

Term	Definition
FPC	Fuel Pool Cooling
FPCCS	Fuel Pool Cooling and Cleanup System
FPP	Fire Protection Program
FSS	Fire Safe Shutdown
FSTF	Full Scale Test Facility
ft.	Feet
FUSAR	Fuel Uprate Safety Analysis Report
FW	Feedwater
FWCF	Feedwater Controller Failure
FWH	Feedwater Heater
FWHOOS	Feedwater Heater Out-of-Service
FWLB	Feedwater Line Break
GDC	General Design Criteria
GE	General Electric
GEH	GE-Hitachi Nuclear Energy Americas LLC
GL	Generic Letter
GSU	Generator Step Up
HCTL	Heat Capacity Temperature Limit
HCVS	Hardened Containment Vent System
HDR	Header
HDWP	High Drywell Pressure
HELB	High Energy Line Break
HEPA	High Efficiency Particulate Adsorber
HP	High Pressure
hp	Horse Power
HPCI	High Pressure Coolant Injection
HSBW	Hot Shutdown Boron Weight

Term	Definition
HTR	Heater
HVAC	Heating Ventilating and Air Conditioning
HWC	Hydrogen Water Chemistry
HWL	High Water Level
HWWV	Hardened Wetwell Vent
HX	Heat Exchanger
I&C	Instrumentation and Control
IASCC	Irradiation-Assisted Stress Corrosion Cracking
IBA	Intermediate Break Accident
ICF	Increased Core Flow
ICGT	In Core Guide Tubes
ID	Identification
IE	Inspection and Enforcement
ICHGT	In Core Housing and Guide Tube
ICS	Integrated Computer System
IEEE	Institute of Electrical and Electronics Engineers
IGSCC	Intergranular Stress Corrosion Cracking
IORV	Inadvertent Opening of a Relief Valve
IPB	Isolated Phase Bus
IPEEE	Individual Plant Examination of External Events
IRM	Intermediate Range Monitor
ISFSI	Independent Spent Fuel Storage Installation
ISP	Integrated Surveillance Program
ISI	In-Service Inspection
IST	In-Service Testing
JIT	Just-in-Time
JOG	Joint Owner's Group

Term	Definition
JP	Jet Pump
JPSL	Jet Pump Sensing Line
kV	Kilovolt
LAR	License Amendment Request
LC	Level Control
LDI	Liquid Drop Impingement
LDR	Load Definition Report
LDS	Leak Detection System
LERF	Large Early Release Frequency
LFWH	Loss of Feedwater Heater
LOCA	Loss-of-Coolant Accident
LOFW	Loss of Feedwater
LOOP	Loss of Offsite Power
LP	Low Pressure
LPCI	Low Pressure Coolant Injection
LPRM	Local Power Range Monitor
LPSP	Low Power Setpoint
LPZ	Low Population Zone
LRPVP	Low Reactor Pressure Vessel Pressure
LSSS	Limiting Safety System Setting
LTP	Long-Term Program
LTR	Licensing Topical Report
LWL	Low Water Level
LWMS	Liquid Waste Management System
MCES	Main Condenser Evacuation System
MCR	Main Control Room
MCPR	Minimum Critical Power Ratio

Term	Definition
MELB	Moderate Energy Line Break
MELLL	Maximum Extended Load Line Limit
MELLLA	Maximum Extended Load Line Limit Analysis
MEQ	Mechanical Equipment Qualification
MeV	Million Electron Volts
MDRIR	Minimum Debris Retention Injection Rate
MEB	Mechanical Electrical Branch
Mlb or Mlbm	Millions of Pounds
MOV	Motor Operated Valve
MPS	Minimum Recirculation Pump Speed
MS	Main Steam
MSIV	Main Steam Isolation Valve
MSIVC	Main Steam Isolation Valve Closure
MSL	Main Steam Line
MSLB	Main Steam Line Break
MSLBA	Main Steam Line Break Accident
MSO	Multiple Spurious Operation
MSRV	Main Steam Relief Valve
MSRVDL	Main Steam Relief Valve Discharge Line
M&T	Measurement and Test
MVA	Million Volt Amps
MWe	Megawatts-Electric
MWt	Megawatt-Thermal
N/A	Not Applicable
NDE	Non-Destructive Examination
NEI	Nuclear Energy Institute
NFPA	National Fire Protection Association

Term	Definition
N-16	Nitrogen 16
NMCA	Noble Metal Chemical Addition
NobleChem™	Noble metal chemicals are added to coat metal surfaces as catalysts for HWC allowing IGSCC mitigation at lower hydrogen injection rates.
NPSH	Net Positive Suction Head
$NPSH_a$	Net Positive Suction Head Available
NPSHR	Net Positive Suction Head - Required
NQAM	Nuclear Quality Assurance Manual
NRC	Nuclear Regulatory Commission
NSI	Next Scheduled Inspection
NSSS	Nuclear Steam Supply System
NTSP	Nominal Trip Set Point
NUMAC	Nuclear Measurement and Control
NUMARC	Nuclear Management and Resources Council
NUREG	Nuclear Regulatory Commission Technical Report Designation
OC	Outside Primary Containment
ODCM	Offsite Dose Calculation Manual
OE	Operating Experience
OFS	Orificed Fuel Support
OI	Operator Instruction
OLMCPR	Operating Limit Minimum Critical Power Ratio
OLNC	On-Line NobleChem – Process to inject NobleChem™ with the plant on-line.
OLTP	Original Licensed Thermal Power
OOS	Out-of-Service
OSD	Original Steam Dryer
OSL	Site Environmental Dosimeter Stations

Term	Definition
ΔP	Differential Pressure - psi
PASS	Post-Accident Sampling Station
PC	Primary Containment
PCS	Pressure Control System
PCT	Peak Clad Temperature
PF	Power Factor
PLC	Programmable Logic Controller
PLUOOS	Power Load Unbalance OOS
PPT	Peak Pool Temperature
PRA	Probabilistic Risk Assessment
PRFO	Pressure Regulator Failure Open
PRNM	Power Range Neutron Monitoring
psi	Pounds per Square Inch
psia	Pounds per Square Inch - Absolute
psid	Pounds per Square Inch - Differential
psig	Pounds per Square Inch - Gauge
PSP	Pressure Suppression Pressure
P-T or P/T	Pressure-Temperature
PUAR	Plant Unique Analysis Report
PULD	Plant Unique Load Definition
QA/QC	Quality Assurance / Quality Control
RBCCW	Reactor Building Closed Cooling Water
RBM	Rod Block Monitor
RCIC	Reactor Core Isolation Cooling
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
RCW	Raw Cooling Water

Term	Definition
RDLB	Recirculation Discharge Line Break
RFP	Reactor Feedwater Pump
RFPT	Reactor Feedwater Pump Turbine
RFW	Reactor Feedwater
RG	Regulatory Guide
RHR	Residual Heat Removal
RHRSW	Residual Heat Removal Service Water
RIPD	Reactor Internal Pressure Difference
RMOV	Reactor Motor Operated Valve
RPT	Recirculation Pump Trip
RPTOOS	Recirculation Pump Trip Out of Service
RPV	Reactor Pressure Vessel
RRS	Reactor Recirculation System
RSD	Replacement Steam Dryer
RSLB	Recirculation Suction Line Break
RSW	Raw Service Water
RT_{NDT}	Reference Temperature of the Nil-Ductility Transition
RTP	Rated Thermal Power
RWCU	Reactor Water Cleanup
RWE	Rod Withdrawal Error
RWM	Rod Worth Minimizer
RX	Reactor
SAF	Single Active Failure
SAMG	Severe Accident Management Guideline
SBA	Small Break Accident
SBO	Station Blackout
SC	Safety Communication

Term	Definition
SCBA	Self-Contained-Breathing-Apparatus
SCW	Stator Cooling Water
SDC	Shutdown Cooling
SER	Safety Evaluation Report
SFA	Steam/Feedwater Application
SFIE	Steam Flow Induced Error
SFP	Spent Fuel Pool
SFPAVS	Spent Fuel Pool Area Ventilation System
SGTS	Standby Gas Treatment System
SHB	Shroud Head Bolts
SIF	Stress Intensification Factor
SIL	Services Information Letter
SJAE	Steam Jet Air Ejectors
SL	Service Level
SLC	Standby Liquid Control
SLCS	Standby Liquid Control System
SLO	Single-loop Operation
SNM	Susceptible Non-Modeled
SORV	Stuck Open Relief Valve
SP	Suppression Pool
SPC	Suppression Pool Cooling
SPDS	Safety Parameter Display System
SQN	Sequoyah Nuclear Plant
SR	Surveillance Requirement
SRLR	Supplemental Reload Licensing Report
SRM	Source Range Monitor
SRP	Standard Review Plan

Term	Definition
SRSS	Square Root of the Sum of the Squares
SRV	Safety Relief Valve
SRVDL	Safety Relief Valve Discharge Line
SSC	Systems, Structures and Components
SSP	Supplemental Surveillance Capsule Program
SSW	Sacrificial Shield Wall
STM	Steam
STP	Simulated Thermal Power
SW	Service Water
SWMS	Solid Waste Management System
TAF	Top of Active Fuel
TBS	Turbine Bypass System
TBVOOS	Turbine Bypass Valves OOS
TCV	Turbine Control Valve
TEDE	Total Effective Dose Equivalent
TFSP	Turbine First-Stage Pressure
T-G	Turbine-Generator
TID	Total Integrated Dose
TIP	Traversing In-core Probe
TLAA	Time Limiting Aging Analysis
TRM	Technical Requirement Manual
TS	Technical Specification
TSC	Technical Support Center
TSV	Turbine Stop Valve
TSVC	Turbine Stop Valve Closure
TT	Turbine Trip
TTNBP	Turbine Trip with no Steam Bypass Failure

Term	Definition
TVA	Tennessee Valley Authority
TW	The TW sequence is a severe accident sequence that is the result of an anticipated transient followed by a total loss of decay heat removal.
UFSAR	Updated Final Safety Analysis Report
UHS	Ultimate Heat Sink
UPS	Uninterruptible Power Supply
US	United States
USAS	USA Standard
USE	Upper Shelf Energy
USI	Unresolved Safety Issue
USNRC	United States Nuclear Regulatory Commission
USST	Unit Station Service Transformer
VFD	Variable Flow Drive
VPF	Vane Passing Frequency
VSL	Vessel
VWO	Valve Wide-Open
WB	Whole Body
WBN	Watts Bar Nuclear Plant
W_d	Drive Flow
WRA	Wear Rate Analysis
WW	Wetwell

EXECUTIVE SUMMARY

This Power Uprate Safety Analysis Report (PUSAR) summarizes the results of safety evaluations performed that justify uprating the licensed thermal power at Browns Ferry Nuclear Plant (Browns Ferry) Units 1, 2, and 3. The requested licensed power level is an increase to 3952 MWt from the current licensed reactor thermal power of 3458 MWt.

The PUSAR is presented in a format consistent with the template safety evaluation report (SER) contained in Section 3.2 of the US NRC, Office of Nuclear Reactor Regulation, Review Standard for Extended Power Uprates, RS-001, December 2003. The Regulatory Evaluations from the template SER have been modified to reflect the licensing basis of Browns Ferry.

GE-Hitachi Nuclear Energy Americas LLC (GEH) has previously developed and implemented a number of extended power uprates (EPUs) using the Nuclear Regulatory Commission (NRC) approved Licensing Topical Reports (LTRs), "Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate," NEDC-32424P-A, February 1999 (ELTR1) and "Generic Evaluations of General Electric Boiling Water Reactor Extended Power Uprate," NEDC-32523P-A, February 2000 (ELTR2). Based on extended power uprate (EPU) experience, GEH has developed an approach to uprate reactor power that maintains the current plant maximum normal operating reactor dome pressure. This approach is referred to as Constant Pressure Power Uprate and was approved by the NRC in the Licensing Topical Report (LTR) NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," hereafter referred to as the CLTR.

Some topics in the CLTR are directly fuel dependent because the fuel type affects the resulting evaluation or the consequences of transients or accidents. Because Browns Ferry Units 1, 2, and 3 will only contain Areva Incorporated, (AREVA) ATRIUM-10 and ATRIUM 10XM fuel types at the time of EPU implementation on the respective Browns Ferry units, the requested Browns Ferry EPU does not reference the CLTR as the basis for areas involving fuel-dependent topics, consistent with the NRC's Conditions and Limitations on the use of the CLTR. The fuel-dependent evaluations were performed by TVA or AREVA using NRC approved codes and methods. Due to the proprietary nature of the AREVA analyses, the fuel-dependent analyses in support of the requested EPU are contained in License Amendment Request (LAR) Attachment 8 (Fuel Uprate Safety Analysis Report) to the Browns Ferry EPU LAR. The safety evaluation sections in this report provide appropriate cross references to LAR Attachment 8 for fuel-related topics.

For evaluations independent of fuel type, this report provides a systematic application of the CLTR approach to systems, structures, components and evaluations, including the performance of plant-specific engineering assessments and confirmation of the applicability of the CLTR generic assessment required to support an EPU.

It is not the intent of this report to explicitly address all the details of the analyses and evaluations described herein. For example, only previously NRC-approved or industry-accepted methods were used for the analyses of accidents and transients, as referred to in the CLTR,

ELTR1, or ELTR2. Therefore, the safety analysis methods have been previously addressed, and thus, are not explicitly addressed in this report. Also, event and analysis descriptions that are already provided in other licensing reports or the Updated Final Safety Analysis Report (UFSAR) are not repeated within this report. This report, in conjunction with other Attachments to the EPU LAR, summarizes the significant evaluations needed to support a licensing amendment to allow for uprated power operation at Browns Ferry.

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

An increase in the electrical output of a BWR plant is accomplished primarily by generating and supplying higher steam flow to the turbine-generator. Browns Ferry, as currently 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 current licensed power level.

A higher steam flow is achieved by increasing the reactor power along specified control rod and core flow lines. A limited number of operating parameters are changed, some setpoints are adjusted and instruments are recalibrated. Plant procedures are revised and power ascension testing is performed to confirm the results of the safety analyses.

Detailed evaluations of the reactor, engineered safety features, power conversion, emergency power, support systems, and design basis accidents were performed. This report, in conjunction with the fuel dependent evaluations contained in LAR Attachment 8, demonstrate that Browns Ferry can safely operate at the requested EPU level. However, non-safety power generation modifications will be implemented in order to obtain the electrical power output associated with the uprate power. Until these modifications are completed, the non-safety related, balance of plant equipment may limit the electrical power output, which in turn may limit the operating thermal power level to less than the rated thermal power level. These modifications have been evaluated and they do not constitute a material alteration to the plant.

The evaluations and reviews were conducted in accordance with the CLTR and the criteria in ELTR1, ELTR2, or the TVA and AREVA codes and methods using NRC-approved or industry-accepted analysis methods. The results of these evaluations and reviews presented in this report are as follows:

- All fuel independent safety aspects of Browns Ferry that are affected by the increase in thermal power were evaluated (fuel dependent safety aspects are evaluated in LAR Attachment 8);
- No reliance on containment accident pressure is required to ensure adequate emergency core cooling system pump net positive suction head during accidents, abnormal operational transients or special (ATWS, SBO, Fire) events;
- Evaluations were performed using NRC-approved or industry-accepted analysis methods;

- Systems and components affected by EPU were reviewed to ensure there is no significant challenge to any safety system;
- No changes, which require compliance with more recent industry codes and standards, are being requested;
- No new design functions that require modifications are necessary for safety related systems, and any modification to non-safety related and/or power generation equipment will be implemented per 10 CFR 50.59; and
- The UFSAR will be updated for the EPU related changes, after EPU is implemented, per the requirements in 10 CFR 50.71(e).

1. INTRODUCTION

1.1 Report Approach

This Power Uprate Safety Analysis Report (PUSAR) summarizes the results of safety evaluations performed to justify uprating the licensed thermal power at Tennessee Valley Authority (TVA) Browns Ferry Nuclear Plant (Browns Ferry) Units 1, 2, and 3. The requested license power level is an increase to 3,952 MWt from the current licensed reactor thermal power (CLTP) of 3,458 MWt.

GE-Hitachi Nuclear Energy Americas LLC (GEH) has previously developed and implemented EPU at several nuclear power plants. Based on EPU experience, GEH has developed an approach to uprating reactor power that maintains the current plant maximum normal operating reactor dome pressure. This approach is referred to as Constant Pressure Power Uprate (CPPU) and is contained in the LTR NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," (Reference 1) hereafter referred to as the "CLTR." The NRC approved the CLTR in the staff SER contained in Reference 1 for BWR plants containing General Electric (GE) fuel types through GE14 and using GEH accident analysis methods.

Because Browns Ferry uses non-GE fuel, the CLTR is not applicable for fuel design dependent evaluations and transients. Analyses and evaluations performed in support of the generic dispositions in the CLTR are not applicable. Fuel dependent subjects are addressed in the complementary Fuel Uprate Safety Analysis Report (FUSAR) included as Attachment 8 to the License Amendment Request (LAR) for power uprate.

This evaluation justifies an EPU to 3,952 MWt, with no increase in reactor operating pressure, which corresponds to 120% of the original licensed thermal power (OLTP) for Browns Ferry. This report is presented in a format consistent with the template contained in Section 3.2 of the United States Nuclear Regulatory Commission (USNRC), Office of Nuclear Reactor Regulation, Review Standard for Extended Power Uprates, RS-001, December 2003 (Reference 2). The Regulatory Evaluations from the template have been modified to reflect the licensing basis of Browns Ferry.

1.1.1 Generic Assessments

Many of the component, system, and performance evaluations contained within this report have been generically evaluated in the CLTR and the "Generic Evaluations of General Electric Boiling Water Reactor Extended Power Uprate," (Reference 3) hereafter referred to as ELTR2, and found to be acceptable by the NRC. The plant-specific applicability of these generic assessments is identified and confirmed in the applicable sections of this report. Generic assessments are those safety evaluations that can be dispositioned for a group or all BWR plants by:

- A bounding analysis for the limiting conditions,
- Demonstrating that there is a negligible effect due to EPU, or

- Demonstrating that the required plant cycle-specific reload analyses are sufficient and appropriate for establishing the EPU licensing basis.

Bounding analyses may be based on either: (1) a demonstration that assessments provided in previous EPU LTRs that included a pressure increase (References 3 and 4) are bounding; or (2) on specific generic studies provided in the CLTR. For these bounding analyses, the current EPU experience is provided in the CLTR, ELTR1, and ELTR2, along with the basis and results of the assessment. For those EPU assessments having a negligible effect, the current EPU experience plus a phenomenological discussion of the basis for the assessment is provided in the CLTR. Assessments that are dependent on the fuel design were performed by others and are included in the complementary FUSAR and associated fuel-related reports in Attachments 8 through 38 of the EPU License Amendment Request (LAR).

Some of the safety evaluations affected by EPU are fuel cycle (reload) dependent. Reload dependent evaluations require that the reload fuel design, core loading pattern, and operational plan be established so that analyses can be performed to establish core operating limits. The reload analysis demonstrates that the core design for EPU meets the applicable NRC evaluation criteria and limits. Because of the lead-time required for the NRC review of this power uprate submittal, the Browns Ferry reload core design for the initial fuel cycle at uprated power are not established at the time of this submittal.

As discussed in Section 2.8.2, the EPU has a relatively small effect on core operating and safety limits. Therefore, the reload fuel design and core loading pattern dependent plant evaluations for EPU operations are performed with the reload analysis as part of the standard reload licensing process. Because Browns Ferry Units 1, 2, and 3 will only contain AREVA ATRIUM-10 and ATRIUM 10XM fuel types at the time of EPU implementation on the respective Browns Ferry units, the requested Browns Ferry EPU does not reference the CLTR as the basis for areas involving fuel-dependent topics, consistent with the NRC's Conditions and Limitations on the use of the CLTR. The fuel-dependent evaluations were performed by TVA or AREVA using NRC approved codes and methods. Due to the proprietary nature of the AREVA analyses, the fuel-dependent analyses in support of the requested EPU are contained in FUSAR Attachment 8 to the Browns Ferry EPU licensing amendment request (LAR).

No plant can implement a power uprate unless the appropriate reload core analysis is performed and all criteria and limits are satisfied. Otherwise, the plant would be in an unanalyzed condition. Based on current requirements, the reload analysis results are documented in the Supplemental Reload Licensing Report (SRLR), and the applicable core operating limits are documented in the plant-specific Core Operating Limits Report (COLR).

1.1.2 Plant-Specific Evaluation

Plant-specific evaluations are assessments of the principal evaluations that are not addressed by the generic assessments described in Section 1.1.1. The relative effect of EPU on the plant-specific evaluations and the methods used for their performance are provided. Where applicable, the assessment methodology is referenced. If a specific computer code is used, the name of this

computer code is provided in the section. Table 1-1 provides a summary of the computer codes used.

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

1.2 Purpose and Approach

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

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

1.2.1 Uprate Analysis Basis

Browns Ferry is currently licensed at the 100% CLTP level of 3,458 MWt. The EPU rated thermal power (RTP) level included in this evaluation is 120% of the OLTP. Plant-specific EPU parameters are listed in Table 1-2. The EPU safety analyses are based on a power level of 1.02 times the EPU power level unless the two percent power factor (PF) is already accounted for in the analysis methods consistent with the methodology described in Reference 5, or the 2% does not apply (e.g., Anticipated Transient Without Scram (ATWS) and SBO events).

1.2.2 Computer Codes

NRC-approved or industry-accepted computer codes and calculational techniques are used to demonstrate compliance with the applicable regulatory acceptance criteria. The codes used in the analyses for this report are provided in Table 1-1. Computer codes for the fuel dependent

analyses are specified in the FUSAR, Attachment 8 to the LAR. The application of these codes to the EPU analyses complies with the limitations, restrictions, and conditions specified in the approving NRC SER where applicable for each code. The limitations on use of these codes and methods as defined in the NRC staff position letter reprinted in ELTR1 and the NRC SER for ELTR2 were followed for this EPU analysis. Any exceptions to the use of the code or conditions of the applicable SERs are noted in Table 1-1. The application of the computer codes in Table 1-1 is consistent with the current Browns Ferry licensing basis except where noted.

1.2.3 Approach

The planned approach to achieving the higher power level consists of the change to the Browns Ferry licensing and design basis to increase the licensed power level to 3,952 MWt, consistent with the approach outlined in the CLTR, except as specifically noted, and with the approach outlined in ELTR1 for fuel-dependent evaluations. Consistent with the CLTR, the following plant-specific exclusions are exercised:

- No increase in maximum normal operating reactor dome pressure
- No increase to maximum licensed core flow
- No increase to currently licensed MELLLA upper boundary
- No change to source term methodology
- No new fuel product line introduction
- No change to fuel cycle length
- No additions to currently licensed operational enhancements

The plant-specific evaluations are based on a review of plant design and operating data, as applicable, to confirm excess design capabilities; and, if necessary, identify required modifications associated with EPU. All changes to the plant-licensing basis have been identified. For specified topics, generic analyses and evaluations in the CLTR, or ELTR1 and ELTR2 as applicable, demonstrate plant operability and safety. The dispositions in the CLTR are based on a 20% increase of OLTP, which is equal to the requested power uprate for Browns Ferry. For this increase in power, the conclusions of system/component acceptability stated in the CLTR and ELTR2 are bounding and have been confirmed for Browns Ferry. The scope and depth of the evaluation results provided herein are established based on the approach in the CLTR and ELTR2 and unique features of the plant. The results of the following evaluations are presented:

- **Reactor Core and Fuel Performance:** Assessments that are dependent on the fuel design were performed by others and are included in the complementary FUSAR included as Attachment 8 to the LAR for power uprate.
- **Reactor Coolant System (RCS) and Connected Systems:** Evaluations of the NSSS components and systems have been performed at EPU conditions. These evaluations confirm the acceptability of the effects of the higher power and the associated change in

process variables (i.e., increased steam and feedwater (FW) flows). Safety-related equipment performance is the primary focus, but key aspects of reactor operational capability are also included.

- **Engineered Safety Feature Systems:** The effects of EPU power operation on the Containment, emergency core cooling system (ECCS), Standby Gas Treatment System (SGTS) and other ESFs have been evaluated for key events. The evaluations include the containment responses during limiting Anticipated Operational Occurrences (AOOs) and special events, ECCS-LOCA, and safety relief valve (SRV) containment dynamic loads.
- **Control and Instrumentation:** The control and instrumentation signal ranges and ALs for setpoints have been evaluated to establish the effects of the changes in various process parameters such as power, neutron flux, steam flow and FW flow. As required, evaluations have been performed to determine the need for any Technical Specification (TS) allowable value (AV) changes for various functions (e.g., main steam line (MSL) high flow isolation setpoints).
- **Electrical Power and Auxiliary Systems:** Evaluations have been performed to establish the operational capability of the plant electrical power and distribution systems and auxiliary systems to ensure that they are capable of supporting safe plant operation at the EPU power level.
- **Power Conversion Systems:** Evaluations have been performed to establish the operational capability of various non-safety Balance-of-Plant (BOP) systems and components to ensure that they are capable of delivering the increased power output, and/or the modifications necessary to obtain full EPU power.
- **Radwaste Systems and Radiation Sources:** The liquid and gaseous waste management systems (GWMSs) have been evaluated at limiting conditions for EPU to show that applicable release limits continue to be met during operation at higher power. The radiological consequences have been evaluated for EPU to show that applicable regulations have been met for the EPU power conditions. This evaluation includes the effect of higher power level on source terms, on-site doses and off-site doses, during normal operation.
- **Reactor Safety Performance Evaluations:** Assessments that are dependent on the fuel design were performed by others and are included in the complementary FUSAR included as Attachment 8 to the LAR for power uprate.
- **Additional Aspects of EPU:** High-energy line break (HELB) and environmental qualification (EQ) evaluations have been performed at bounding conditions for EPU to show the continued operability of plant equipment under EPU conditions. The effects of EPU on the Browns Ferry Probabilistic Risk Assessment (PRA) have been analyzed to demonstrate that there are no new vulnerabilities to severe accidents.

1.3 EPU Plant Operating Conditions

1.3.1 Reactor Heat Balance

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

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

Figure 1-2 shows the EPU heat balance at 100% of EPU RTP and 100% rated core flow. Figure 1-3 shows the EPU heat balance at 102% of EPU RTP and 100% core flow with dome pressure at 1,070 psia.

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

1.3.2 Reactor Performance Improvement Features

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

1.4 Summary and Conclusions

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

The Browns Ferry licensing bases have been reviewed to demonstrate how this uprate can be accommodated without a significant increase in the probability or consequences of an accident previously evaluated, without creating the possibility of a new or different kind of accident from

any accident previously evaluated, and without exceeding any existing regulatory limits or design allowable limits applicable to the plant which might cause a reduction in a margin of safety. The EPU described herein involves no significant hazard consideration.

Table 1-1 Computer Codes Used For EPU

Task	Computer Code*	Version or Revision	NRC Approved	Comments
Nominal Reactor Heat Balance	ISCOR	09	Y(2)	NEDE-24011P Rev. 0 SER
Reactor Pressure Vessel (RPV) Fluence	TGBLA DORTG	06 01	Y N	 (8) (9)
Reactor Internal Pressure Differences (RIPDs)	ISCOR LAMB TRACG	09 07 02	Y(2) (3) Y	NEDE-24011P Rev. 0 SER NEDE-20566-P-A NEDE-32176P Rev. 0 NEDC-32177P Rev. 1 NRC TAC No. M90270
Reactor Vessel Integrity – Stress and Fatigue Evaluation	ANSYS FatiguePro	6.1 3.01	N N	(1) (1)
RPV Flow-Induced Vibration	ANSYS SAP4G07	6 07	N N	(1) NEDO-10909 (1)
Reactor Recirculation System	BILBO	04V	N/A	NEDE-23504, February 1977 (1)
Reactor Coolant Pressure Boundary Piping	TPIPE	Various	N	(7)
Piping Components Flow Induced Vibration	SAP4G07	07	N	GE NEDO-10909 (1)
Anticipated Transient Without Scram	ODYN STEMP PANACEA ISCOR	10 04 11 09	Y (5) Y(4) Y(2)	NEDE-24154P-A Supplement. 1, Vol. 4 NEDE-30130-P-A NEDE-24011P Rev. 0 SER
Containment System Response	SHEX M3CPT LAMB	06 05 08	Y Y (3)	(6) NEDO-10320, Apr. 1971 (NUREG-0661) NEDE-20566-P-A September 1986
Annulus Pressurization (AP)	ISCOR	09	Y(2)	NEDE-24011P Rev. 0 SER
Station Blackout	SHEX	06	Y	(6)
Fission Product Inventory	ORIGEN2	2.1	N	Isotope Generation and Depletion Code
MS Piping Analysis	TPIPE	16	N	Structural Analysis Program (7)

Table 1-1 Computer Codes Used For EPU (continued)

Task	Computer Code*	Version or Revision	NRC Approved	Comments
Plant Life Flow Accelerated Corrosion	CHECWORKS™ SFA	4.0	N	Industry supported software to assist the utility industry in planning and implementing inspection programs to prevent FAC failures.

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

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

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

(3) The LAMB code is approved for use in ECCS-LOCA applications (NEDE-20566-P-A and NEDO-20566A), but no approving SER exists for the use of LAMB in the evaluation of reactor internal pressure differences or containment system response. The use of LAMB for these applications is consistent with the model description of NEDE-20566-P-A (Reference 6).

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

(5) The STEMP code uses fundamental mass and energy conservation laws to calculate the suppression pool heatup. The use of STEMP was noted in NEDE-24222, "Assessment of BWR Mitigation of ATWS, Volume I and II (NUREG-0460 Alternate No. 3) December 1, 1979." The code has been used in ATWS applications since that time. It has also recently been accepted in the NRC review of NEDC-32868P, "GE14 Compliance with Amendment 22 of NEDE-24011-P-A (GESTAR)." There is no formal NRC review and approval of STEMP.

(6) The application of the methodology in the SHEX code to the containment response is approved by the NRC in the letter to G. L. Sozzi (GE) from A. Thadani (NRC), "Use of the SHEX Computer Program and

ANSI/ANS 5.1-1979 Decay Heat Source Term for Containment Long-Term Pressure and Temperature Analysis," July 13, 1993 (Reference 7).

(7) TPIPE is a linear elastic analysis of piping program used by TVA for analysis of the Main Steam (MS) and FW piping. TPIPE is not a safety analysis code that requires NRC approval. TVA validation and verification of the TPIPE program and related approval data is stored in TVA System ID 262127. The TPIPE program is described in the Browns Ferry UFSAR, Appendix C, Section C.3.7, and has been benchmarked against the NRC program EPIPE in accordance with the Standard Review Plan, NUREG-0800, Section 3.9.1.II and NUREG/CR-1677. TPIPE is TVA's program used for pipe analysis for all three units at Browns Ferry.

(8) The use of DORTG was approved by the NRC through the letter from H. N. Berkow (USNRC) to G. B. Stramback (GE), "Final Safety Evaluation Regarding Removal of Methodology Limitations for NEDC-32983P-A, General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluations (TAC No. MC3788)," November 17, 2005.

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

Table 1-2 OLTP, CLTP, and EPU Plant Operating Conditions

Parameter	OLTP Value[5]	CLTP Value[1]	EPU Value[4]
Thermal Power (MWt)	3,293	3,458	3,952
Vessel Steam Flow (Mlb/hr) [2]	13.370	14.153	16.440
Full Power Core Flow Range			
Mlb/hr	76.9 to 107.6	83.0 to 107.6	101.5 to 107.6
% Rated	75 to 105	81 to 105	99 to 105
Maximum Nominal Dome Pressure (psia)	1,020	1,050	1,050
Maximum Nominal Dome Temperature (°F)	547.0	550.5	550.5
Pressure at Upstream Side of Turbine Stop Valve (TSV) (psia)	960	1,000	983
Full Power FW			
Flow (Mlb/hr)	13.330	14.103	16.390
Temperature (°F)	377.0	381.7	394.5
Core Inlet Enthalpy (Btu/lb) [3]	521.6	524.7	523.2
Reactor Recirculation System (RRS) Outlet Design Temperature	575ºF	575ºF	575ºF
RRS Outlet Maximum Temperature	546ºF	550.5ºF [6]	550.5ºF
RRS Inlet Design Temperature	575ºF	575ºF	575ºF
RRS Inlet Maximum Temperature	546ºF	550.5ºF	550.5ºF
FW Nozzle Design Temperature	575ºF	575ºF	575ºF
FW Nozzle Maximum Temperature [7]	573ºF	573ºF	573ºF
Main Steam (MS) Nozzle Design Temperature	575ºF	575ºF	575ºF
MS Nozzle Maximum Temperature	546ºF	550.5ºF	550.5ºF
Core Spray (CS) Nozzle Design Temperature	575ºF	575ºF	575ºF
CS Nozzle Maximum Temperature [8]	546ºF	550.5ºF	550.5ºF

Notes:

1. Based on current reactor heat balance.
2. At normal FW heating.
3. At 100% core flow conditions.

4. Performance improvement features and/or equipment Out-of-Service (OOS) that are included in EPU evaluations:
    a. MELLLA
    b. Increased Core Flow (ICF)
    c. Single-loop Operation (SLO)
    d. Final Feedwater Temperature Reduction (FFWTR), 55°F Temperature Reduction
    e. APRM/RBM/Technical Specifications (ARTS)
    f. 3% Main Steam Relief Valve (MSRV) Setpoint Tolerance
    g. One MSRV OOS
    h. Turbine Bypass Valves OOS (TBVOOS)
    i. End-of-Cycle Recirculation Pump Trip (EOC RPT) OOS (RPTOOS)
    j. Feedwater Heaters Out-of-Service (FWHOOS), 55°F Temperature Reduction
    k. 24 Month Fuel Cycle
    l. Power Load Unbalance OOS (PLUOOS)
5. All nozzle maximum pressures are the same as the maximum normal dome pressure and the design pressure, 1,250 psig, remains unchanged from OLTP to EPU.
6. Maximum nominal dome temperature is 550.5ºF.
7. FW OLTP, CLTP and EPU maximum temperature values are based on loss of feedwater pumps for 102% rated thermal power conditions. During these times, nozzles will be filled with steam.
8. CS has no flow under normal operating conditions and the maximum temperature values correspond to the maximum nominal dome temperatures above.

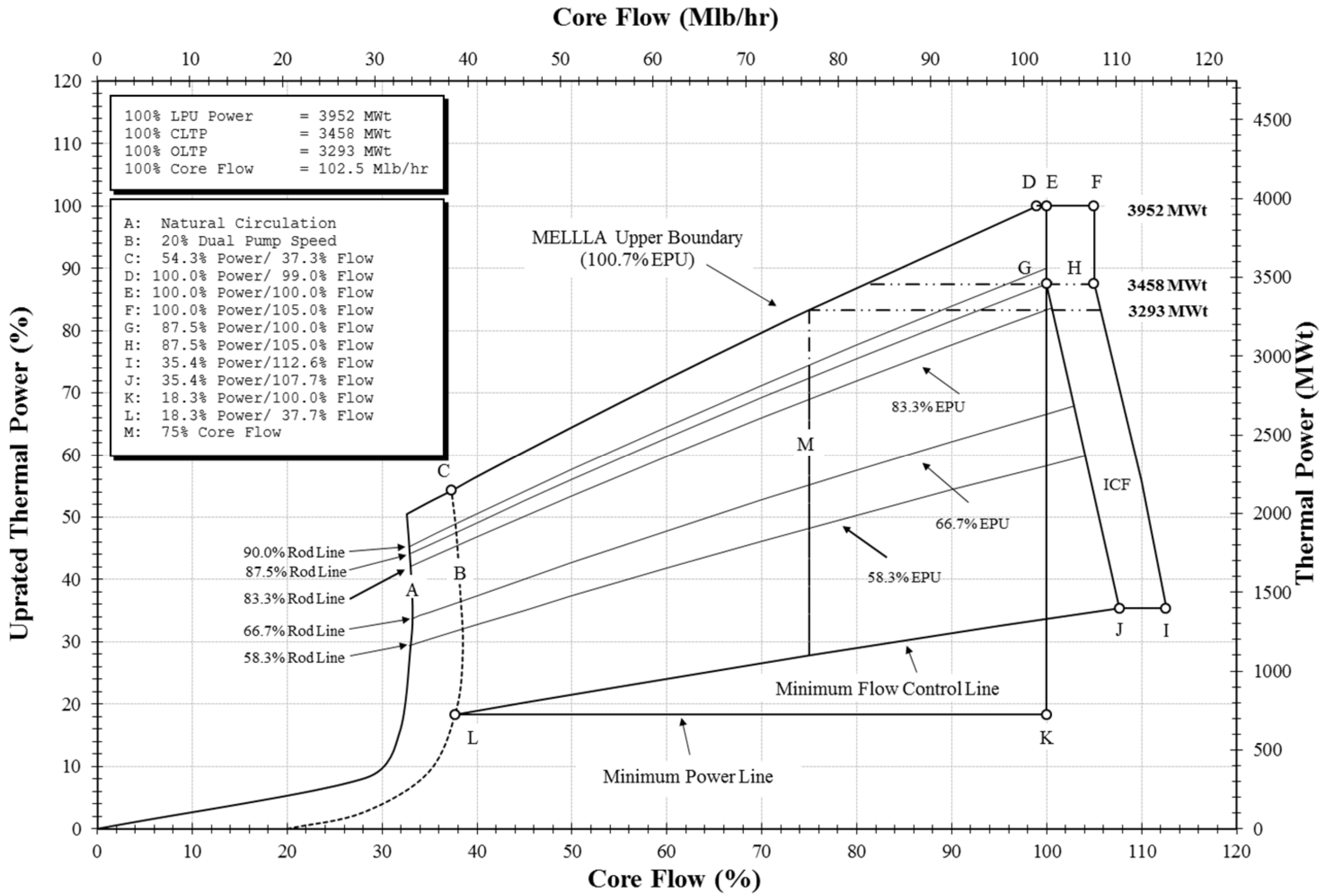
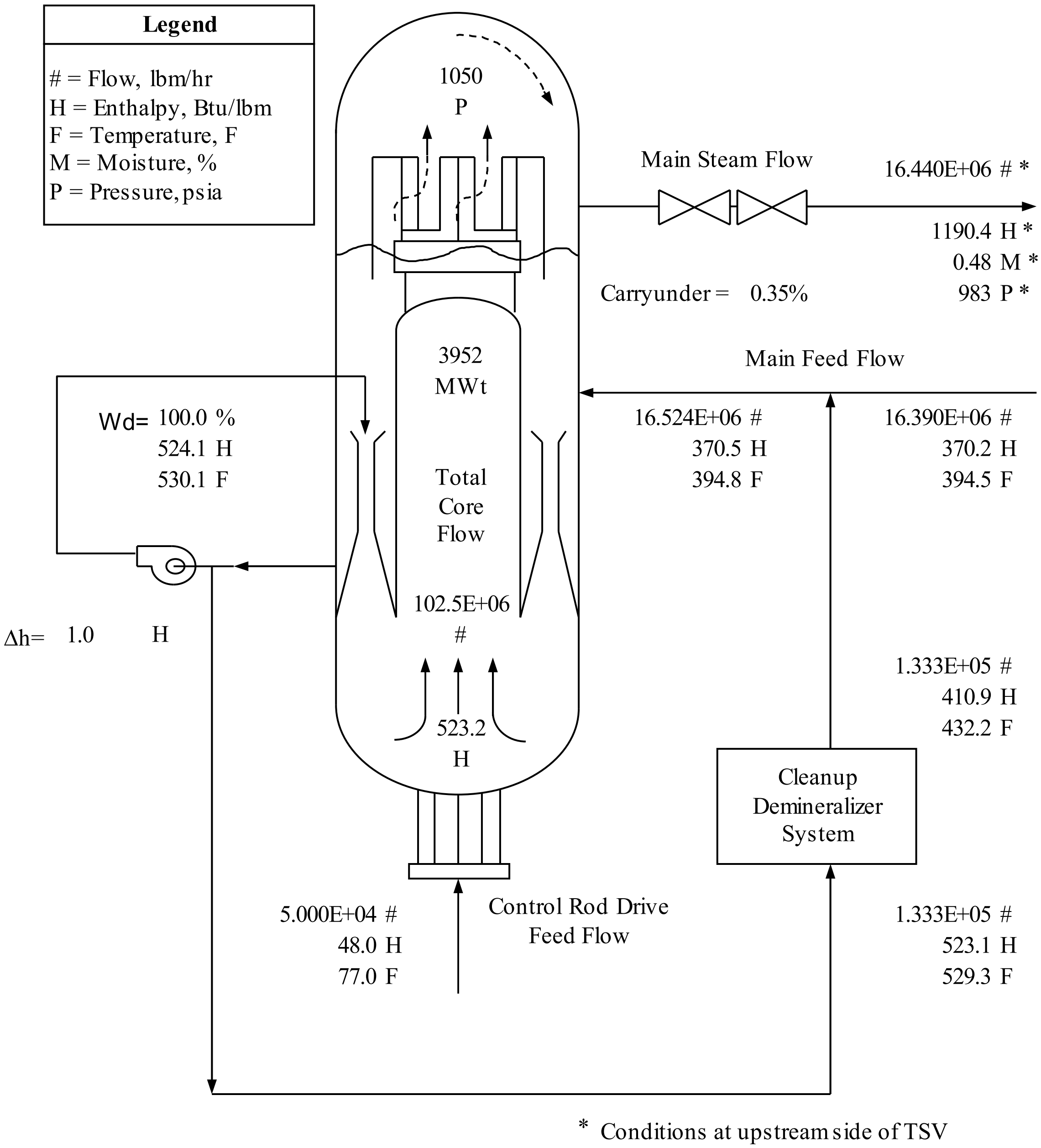


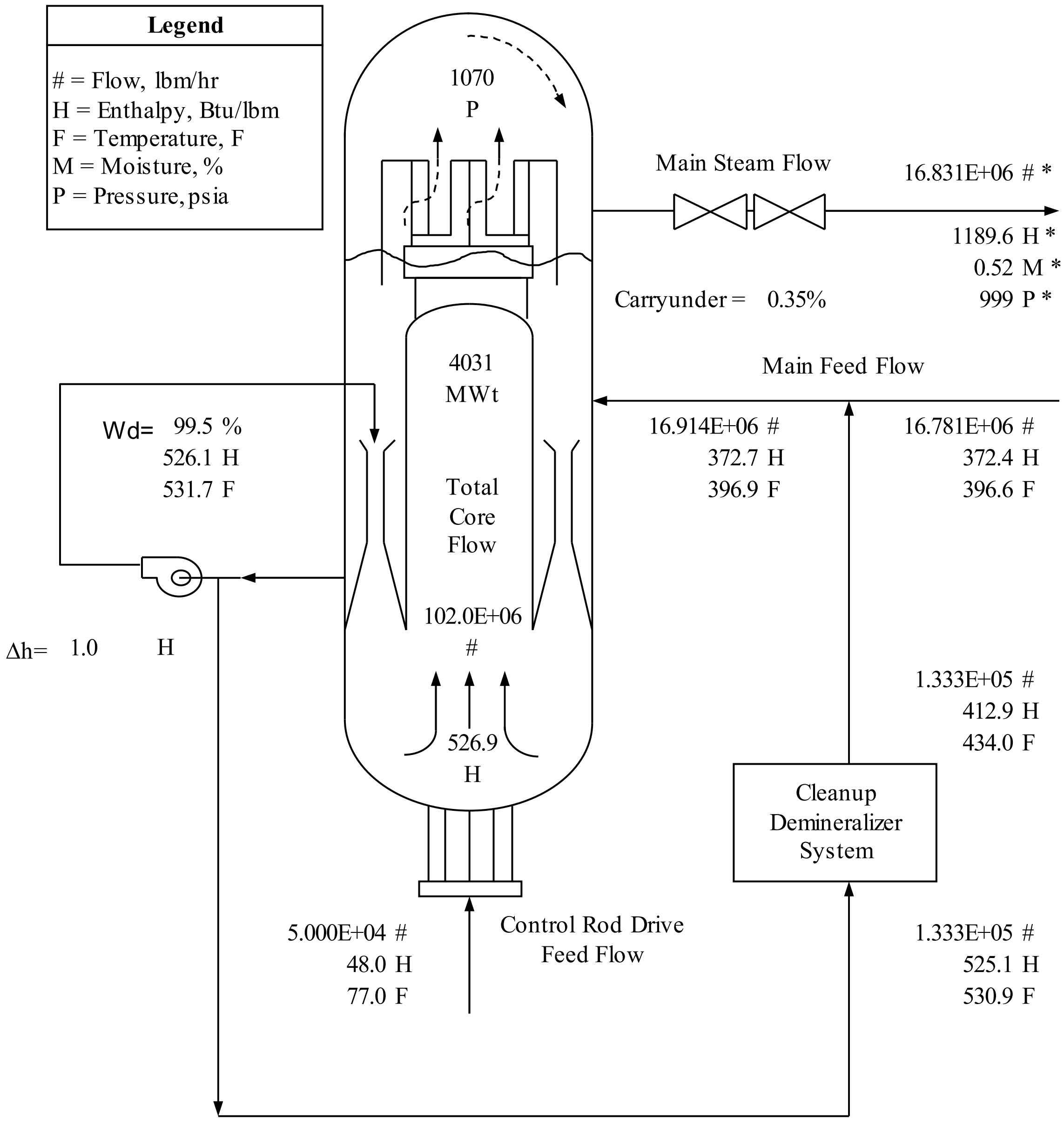
Figure 1-1 Power/Flow Operating Map for EPU



Core Thermal Power	3952.0
Pump Heating	10.6
Cleanup Losses	-4.4
Other System Losses	-1.1
Turbine Cycle Use	3957.1 MWt

Figure 1-2 EPU Heat Balance – Nominal

(@ 100% Power and 100% Core Flow)



* Conditions at upstream side of TSV

Core Thermal Power	4031.0
Pump Heating	10.4
Cleanup Losses	-4.4
Other System Losses	-1.1
Turbine Cycle Use	4035.9 MWt

Figure 1-3 EPU Heat Balance - Overpressure Protection Analysis

(@ 102% Power and 100% Core Flow)

2. SAFETY EVALUATION

2.1 Materials and Chemical Engineering

2.1.1 Reactor Vessel Material Surveillance Program

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

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

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

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

Browns Ferry Current Licensing Basis

The General Design Criteria (GDC) listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Browns Ferry principal design criteria predate these criteria. The Browns Ferry principal design criteria are listed in UFSAR Section 1.5, "Principal Design Criteria." In 1967, the Atomic Energy Commission (AEC) published for public comment a revised set of proposed General Design Criteria (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed GDC published in 1967, the Tennessee Valley Authority (TVA) performed a comparative evaluation of the design basis of Browns Ferry with the AEC proposed General Design Criteria of 1967. The Browns Ferry UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria set out in the July 1967 AEC release. For each group of criteria, there is a statement of TVA's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a table of references to locations in the Browns Ferry UFSAR where there is subject matter relating to the intent of that particular criteria.

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

Design Criteria (referred to here as “draft GDC”) is contained in Browns Ferry UFSAR Appendix A: draft GDCs-9 and 33. Final GDC-31 is applicable to Browns Ferry as described in “Browns Ferry Nuclear Plant (BFN), Unit 1- Application to Modify Technical Specification 3.4.9, ‘RCS Pressure and Temperature (P/T) Limits’ (BFN TS-484),” dated December 18, 2013 (Reference 8), “Browns Ferry Nuclear Plant (BFN), Unit 2 - Application to Modify Technical Specification 3.4.9, ‘RCS Pressure and Temperature (P/T) Limits’ (BFN TS-491),” dated June 19, 2014 (Reference 9), and “Browns Ferry Nuclear Plant, Unit 3 - Application to Modify Technical Specification 3.4.9, ‘RCS Pressure and Temperature (P/T) Limits’ (BFN TS-494),” dated January 27, 2015 (Reference 10).

The Reactor Vessel Material Surveillance Program is described in Browns Ferry UFSAR Section 4.2, “Reactor Vessel and Appurtenances Mechanical Design,” and the Bases to TS 3.4.9, “RCS Pressure and Temperature (P/T) Limits.”

In addition to the evaluations described in the Browns Ferry UFSAR, Browns Ferry’s systems and components were evaluated for license renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Browns Ferry License Renewal Safety Evaluation Report (SER), NUREG-1843, dated April 2006 (Reference 11). The license renewal evaluation associated with the Reactor Vessel Material Surveillance Program is documented in NUREG-1843, Section 3.0.3.2.19.

Technical Evaluation

The RPV fracture toughness evaluation process is described in Section 2.1.2. RPV embrittlement is caused by neutron exposure of the wall adjacent to the core including the regions above and below the core that experience fluence greater than or equal to 1×10^{17} n/cm^2. This region is defined as the beltline region. Operation at EPU conditions results in a higher neutron flux, which increases the integrated fluence over the period of plant life.

The surveillance program consists of three capsules for each unit. No capsules have been removed from the Browns Ferry Unit 1 vessel. Therefore, three capsules remain in the vessel, and have been there since plant startup. One capsule containing Charpy specimens was removed from the Browns Ferry Unit 2 vessel after 8.2 effective full power years (EFPY) of operation (end of Fuel Cycle 7), tested, reconstituted, and placed into the vessel during the Unit 2 Cycle 8 refueling outage. A second capsule was removed after 22.9 EFPY of operation (end of Fuel Cycle 16), tested, and analyzed. The remaining one of the three original capsules has been in the reactor vessel since plant startup. The first Browns Ferry Unit 3 capsule was removed from the vessel during the Fuel Cycle 8 outage, but was not tested. Browns Ferry Units 1, 2, and 3 are part of the BWR Vessel and Internals Project (BWRVIP) Integrated Surveillance Program (ISP) currently administered by Electric Power Research Institute (EPRI) and will comply with the withdrawal schedule specified for representative or surrogate surveillance capsules that now represent each unit. Therefore, the 10 CFR 50 Appendix H surveillance capsule schedule for the ISP governs. Implementation of EPU has no adverse effect on the BWRVIP withdrawal schedule.

The maximum normal operating dome pressure for EPU is unchanged from that for CLTP thermal power operation. Therefore, the hydrostatic and leakage test pressures are acceptable for EPU. Operation with EPU does not have an adverse effect on the reactor vessel fracture toughness because the Unit 1, 2, and 3 vessels remain in compliance with the regulatory requirements as demonstrated in Section 2.1.2.

Conclusion

TVA has evaluated the effects of the proposed EPU on the reactor vessel surveillance withdrawal schedule and has addressed changes in neutron fluence and their effects on the schedule. The evaluation indicates that the material surveillance program will continue to meet the requirements of 10 CFR Part 50, Appendix H, and 10 CFR 50.60, and will ensure continued compliance with draft GDCs-9 and 33, and final GDC-31 in this respect following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to the reactor vessel material surveillance program.

2.1.2 Pressure-Temperature Limits and Upper-Shelf Energy

Regulatory Evaluation

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

The NRC's acceptance criteria for P-T limits are based on (1) GDC-14, insofar as it requires that the RCPB be designed, fabricated, erected, and tested so as to have an extremely low probability of rapidly propagating fracture; (2) GDC-31, insofar as it requires that the RCPB be designed with margin sufficient to assure that, under specified conditions, it will behave in a non-brittle manner and the probability of a rapidly propagating fracture is minimized; (3) 10 CFR Part 50, Appendix G, which specifies fracture toughness requirements for ferritic components of the RCPB; and (4) 10 CFR 50.60, which requires compliance with the requirements of 10 CFR Part 50, Appendix G.

Specific NRC review criteria are contained in SRP Section 5.3.2 and other guidance provided in Matrix 1 of RS-001.

Browns Ferry Current Licensing Basis

The General Design Criteria (GDC) listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Browns Ferry Nuclear Plant (Browns Ferry) principal design criteria predate these criteria. The Browns Ferry principal design criteria are listed in UFSAR Section 1.5, "Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, the Tennessee Valley Authority (TVA) performed a comparative evaluation of the design basis of Browns Ferry with the AEC proposed General Design Criteria of 1967. The Browns Ferry UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this

comparative evaluation. This evaluation discusses each of the groups of criteria set out in the July 1967 AEC release. For each group of criteria, there is a statement of TVA's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a table of references to locations in the Browns Ferry UFSAR where there is subject matter relating to the intent of that particular criteria.

While Browns Ferry is not generally licensed to the final GDC or the 1967 AEC proposed General Design Criteria, a comparison of the final GDC to the applicable AEC proposed General Design Criteria can usually be made. For the final GDC listed in the Regulatory Evaluation above, the Browns Ferry comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Browns Ferry UFSAR Appendix A: draft GDC-9. Final GDC-31 is applicable to Browns Ferry as described in "Browns Ferry Nuclear Plant (BFN), Unit 1- Application to Modify Technical Specification 3.4.9, 'RCS Pressure and Temperature (P/T) Limits' (BFN TS-484)," dated December 18, 2013 (Reference 8), "Browns Ferry Nuclear Plant (BFN), Unit 2 - Application to Modify Technical Specification 3.4.9, 'RCS Pressure and Temperature (P/T) Limits' (BFN TS-491)," dated June 19, 2014 (Reference 9), and "Browns Ferry Nuclear Plant, Unit 3 - Application to Modify Technical Specification 3.4.9, 'RCS Pressure and Temperature (P/T) Limits' (BFN TS-494)," dated January 27, 2015 (Reference 10).

The Pressure-Temperature Limits and Upper Shelf Energy is described in Browns Ferry UFSAR Section 4.2, "Reactor Vessel and Appurtenances Mechanical Design," and the Bases to TS 3.4.9, "RCS Pressure and Temperature (P/T) Limits."

In addition to the evaluations described in the Browns Ferry UFSAR, Browns Ferry's systems and components were evaluated for license renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Browns Ferry License Renewal Safety Evaluation Report (SER), NUREG-1843, dated April 2006 (Reference 11). The license renewal evaluations associated with Pressure-Temperature Limits and Upper-Shelf Energy are documented in NUREG-1843, Sections 4.2.1 and 4.2.5.

<u>RCS Pressure and Temperature (P/T) Limits</u>

The Reactor Coolant System (RCS) Pressure and Temperature (P/T) Limits have been developed for EPU conditions and have been submitted to the NRC for approval as follows:

a. The Browns Ferry Unit 1 change was submitted to the NRC on December 18, 2013 and approved in License Amendment No. 287 on February 2, 2015.
b. The Browns Ferry Unit 2 change was submitted to the NRC on June 19, 2014 and approved in License Amendment No. 314 on June 2, 2015.
c. The current Browns Ferry Unit 3 P/T limits are based on EPU conditions and were approved by the NRC in License Amendment 247 on March 10, 2004. A revision to the

Browns Ferry Unit 3 P/T limits was submitted to the NRC on January 27, 2015, to address operation beyond the period of the original 40-year operating license and is currently under NRC review. These revised P/T limits have also been developed for EPU conditions.

Technical Evaluation

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

As explicitly stated in Section 3.2.1 of the CLTR, EPU may result in a higher operating neutron flux at the vessel wall, consequently increasing the integrated flux over time (neutron fluence). The neutron fluence is recalculated using the NRC-approved GEH neutron fluence methodology (Reference 12). This method is consistent with Regulatory Guide (RG) 1.190 (Reference 13) and utilizes a more representative fluence than previous methods. Browns Ferry meets all CLTR dispositions.

AREVA fuel will be used at Browns Ferry when EPU is implemented; however, the basis for the RPV flux is the GEH analysis using GE14 fuel. AREVA independently evaluates the bounding nature of the GEH results for the peak flux values for RPV inner diameter, and internals (shroud diameter, top guide, core plate) in FUSAR Section 2.1.2.

The topics addressed in this evaluation are:

Topic	CLTR Disposition	Browns Ferry Result
Fracture Toughness	Plant Specific	Meets CLTR Disposition

The revised fluence is used to evaluate the vessel against the requirements of 10 CFR 50, Appendix G. The results of these evaluations indicate that:

(a) The reduction in USE, using Equivalent Margin methods, demonstrates that there is an equivalent margin of safety against fracture for RPV materials such that it will remain qualified with respect to 10 CFR 50 Appendix G criterion for the design life of the vessel. The maximum decrease in USE for the beltline plate materials is 16% ([[ ]]) for Unit 2 at 48 EFPY. The maximum decrease in USE for the beltline weld materials is 33.5% ([[ ]]) for Unit 1 at 38 EFPY. These values are provided in Tables 2.1-1a through 2.1-1c.

(b) The beltline material Reference Temperature of Nil-Ductility Transition (RT_{NDT}) remains below 200°F. The N-16 water level instrumentation nozzle is included in the evaluation.

(c) The Technical Specification P-T curves were revised to incorporate the methodology of the GEH P-T curve LTR (Reference 14) and the ISP Browns Ferry Unit 2 second surveillance capsule results. The fracture toughness evaluation included the effects of the N-16 water level instrumentation nozzle that occurs within the beltline region. The hydro test pressure for EPU is the minimum nominal operating pressure.

(d) The end of life (EOL) shift is increased, and consequently, results in an increase in the Adjusted Reference Temperature (ART), which is the initial RT_{NDT} plus the shift. These values are provided in Tables 2.1-2a through 2.1-2c.

(e) The EOL beltline circumferential weld material mean RT_{NDT} remains bounded by the requirements of Generic Letter (GL) 98-05 (Reference 15), BWRVIP-05 (References 16 and 17), and BWRVIP 74-A (Reference 18). This comparison is provided in Table 2.1-3.

(f) GEH P-T limit curves include an adjustment for the column of water in a full RPV. The Browns Ferry EPU is a constant pressure power uprate, which, by definition, does not change the pressure from that considered for CLTP. The pressure head for Browns Ferry for a full vessel is 31.6 psig.

(g) ISP plate and weld materials have been considered in development of the beltline ART as defined in BWRVIP-135. In accordance with the guidance from BWRVIP-135 and the methodology provided in RG 1.99 Revision 2 (Reference 19), the surveillance materials are considered in the development of the P-T limit curves for Units 1 and 2, but are not considered in the development of the P-T limit curves for Unit 3.

(h) The generic pressure test P-T limit curve is based on dimensions cited in NEDC-33178P-A, Revision 1 (Reference 14). GEH P-T limit curves are considered acceptable for plant-specific application when it is demonstrated that the plant-specific dimensions are bounded by the generic dimensions, as is the case for Browns Ferry Units 1, 2, and 3.

(i) Ferritic piping within the RCPB has not been replaced since plant start-up.

Therefore, Browns Ferry meets all CLTR dispositions for fracture toughness.

Conclusion

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

2.1.3 Reactor Internal and Core Support Materials

Regulatory Evaluation

The reactor internals and core supports include structures, systems, and components (SSCs) that perform safety functions or whose failure could affect safety functions performed by other SSCs. These safety functions include reactivity monitoring and control, core cooling, and fission product confinement (within both the fuel cladding and the RCS).

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

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

Browns Ferry Current Licensing Basis

The General Design Criteria (GDC) listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Browns Ferry Nuclear Plant (Browns Ferry) principal design criteria predate these criteria. The Browns Ferry principal design criteria are listed in UFSAR Section 1.5, "Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, the Tennessee Valley Authority (TVA) performed a comparative evaluation of the design basis of Browns Ferry with the AEC proposed General Design Criteria of 1967. The Browns Ferry UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria set out in the July 1967 AEC release. For each group of criteria, there is a statement of TVA's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a table of references to locations in the Browns Ferry UFSAR where there is subject matter relating to the intent of that particular criteria.

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

The reactor internals and core supports are described in Browns Ferry UFSAR Section 3.3, "Reactor Vessel Internals Mechanical Design."

In addition to the evaluations described in the Browns Ferry UFSAR, Browns Ferry's systems and components were evaluated for license renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Browns Ferry License Renewal Safety Evaluation Report (SER), NUREG-1843, dated April 2006 (Reference 11). The license renewal evaluation associated with the Reactor Internal and Core Support Materials is documented in NUREG-1843, Section 2.3.1.2.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of EPUs. Section 10.7 of the CLTR addresses the effect of EPU on reactor internal and core support materials. The results of this evaluation are described below.

Browns Ferry meets all CLTR dispositions. The topics addressed in this evaluation are:

Topic	CLTR Disposition	Browns Ferry Result
Irradiated Assisted Stress Corrosion Cracking	Plant Specific	Meets CLTR Disposition

As explicitly stated in Section 10.7 of the CLTR, the increase in irradiation of the core internal components influences Irradiation-Assisted Stress Corrosion Cracking (IASCC). The longevity of most equipment is not affected by EPU. [[

]] A plant-specific analysis of IASCC is required for EPU.

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

Browns Ferry has a procedurally controlled program for the augmented NDE of selected RPV internal components in order to ensure their continued structural integrity. The inspection techniques utilized are primarily for the detection and characterization of service-induced, surface-connected planar discontinuities, such as Intergranular Stress Corrosion Cracking (IGSCC) and IASCC, in welds and in the adjacent base material. Browns Ferry belongs to the BWR Vessel and Internals Project (BWRVIP) organization and implementation of the procedurally controlled program is consistent with the BWRVIP issued documents. The inspection strategies recommended by the BWRVIP consider the effects of fluence on applicable components and are based on component configuration and field experience. The inspection program is modified for the inspection of the core plate bolts in accordance with Deviation Disposition No. DD-2011-01 (Reference 20). The inspection program is enhanced for additional inspections of the core plate beyond what is required by the BWRVIP.

Components selected for inspection include those that are identified as susceptible to in-service degradation and those where augmented examination is conducted for verification of structural

integrity. These components have been identified through the review of NRC Inspection and Enforcement Bulletins (IEBs), BWRVIP documents, and recommendations provided by General Electric Services Information Letters (GE SILs). The inspection program provides performance frequency for NDE and associated acceptance criteria. Components inspected include the following:

- Core Spray (CS) piping
- Core plate
- Core spray spargers
- Core shroud and core shroud support
- Jet pumps and associated components
- Top guide
- Lower plenum
- Vessel ID attachment welds
- Instrumentation penetrations
- Steam dryer drain channel welds
- FW spargers
- In-core flux monitoring guide tubes
- Control rod guide tubes

Inspected components are considered as being potentially susceptible to IASCC if the end-of-life fluence is in excess of 5×10^{20} n/cm^2 (E> 1 MeV). Three components have been identified as being potentially susceptible to IASCC, based upon the projected 54 EFPY fluence for Unit 1: (1) Top Guide, 2.06×10^{22} n/cm^2 (E> 1 MeV); (2) Shroud, 5.34×10^{21} n/cm^2 (E> 1 MeV); and (3) Core Plate, 7.33×10^{20} n/cm^2 (E> 1 MeV). Three components have been identified as being potentially susceptible to IASCC, based upon the projected 52 EFPY fluence for Units 2 and 3: (1) Top Guide, 1.98×10^{22} n/cm^2; (2) Shroud, 5.15×10^{21} n/cm^2; and (3) Core Plate, 7.07×10^{20} n/cm^2. The BWRVIP inspection recommendations that provide the scope, sample size, inspection method, and frequency of examination used to manage the effects of IASCC are as follows:

- Top Guide (BWRVIP-26-A and BWRVIP-183) (References 21 and 22)
- Shroud (BWRVIP-76) (Reference 23)
- Core Plate (BWRVIP-25) (Reference 24)

Continued implementation of the current procedure program assures the prompt identification of any degradation of reactor vessel internal components experienced during EPU operating conditions. To mitigate the potential for IGSCC and IASCC, Browns Ferry utilizes hydrogen

water chemistry and noble metals applications. Reactor vessel water chemistry conditions are also maintained consistent with the EPRI and established industry guidelines.

The service life of most equipment is not affected by EPU. The peak fluence increase experienced by the reactor internals does not represent a significant increase in the potential for IASCC. The current inspection strategy for the reactor internal components is expected to be adequate to manage any potential effects of EPU. No relevant indications have been observed during in the most recent grid beam inspection of Browns Ferry Units 1, 2, or 3.

Analysis of the core plate bolts was conducted as part of the Time Limiting Aging Analysis (TLAA) for the Browns Ferry license renewal, per Reference 25.

Therefore, Browns Ferry meets all CLTR dispositions for IASCC.

Conclusion

TVA has evaluated the effects of the proposed EPU on the integrity of reactor internal and core support materials. The evaluation indicates that the reactor internal and core support materials will continue to be acceptable and will continue to meet the requirements of draft GDC-1 and 10 CFR 50.55a. Therefore, the proposed EPU is acceptable with respect to reactor internal and core support materials.

2.1.4 Reactor Coolant Pressure Boundary Materials

Regulatory Evaluation

The RCPB defines the boundary of systems and components containing the high-pressure fluids produced in the reactor.

The NRC's acceptance criteria for RCPB materials are based on (1) 10 CFR 50.55a and GDC-1, insofar as they require that SSCs important to safety be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety functions to be performed; (2) GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents; (3) GDC-14, insofar as it requires that the RCPB be designed, fabricated, erected, and tested so as to have an extremely low probability of rapidly propagating fracture; (4) GDC-31, insofar as it requires that the RCPB be designed with margin sufficient to assure that, under specified conditions, it will behave in a non-brittle manner and the probability of a rapidly propagating fracture is minimized; and (5) 10 CFR Part 50, Appendix G, which specifies fracture toughness requirements for ferritic components of the RCPB.

Specific NRC review criteria are contained in SRP Section 5.2.3 and other guidance provided in Matrix 1 of RS-001. Additional review guidance for primary water stress-corrosion cracking of dissimilar metal welds and associated inspection programs is contained in GL 97-01, Information Notice 00-17, Bulletins 01-01, 02-01, and 02-02. Additional review guidance for thermal embrittlement of cast austenitic stainless steel components is contained in a letter from C. Grimes, NRC, to D. Walters, Nuclear Energy Institute, dated May 19, 2000.

Browns Ferry Current Licensing Basis

The General Design Criteria (GDC) listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Browns Ferry Nuclear Plant (Browns Ferry) principal design criteria predate these criteria. The Browns Ferry principal design criteria are listed in UFSAR Section 1.5, "Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, the Tennessee Valley Authority (TVA) performed a comparative evaluation of the design basis of Browns Ferry with the AEC proposed General Design Criteria of 1967. The Browns Ferry UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria set out in the July 1967 AEC release. For each group of criteria, there is a statement of TVA's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a table of references to locations in the Browns Ferry UFSAR where there is subject matter relating to the intent of that particular criteria.

While Browns Ferry is not generally licensed to the final GDC or the 1967 AEC proposed General Design Criteria, a comparison of the final GDC to the applicable AEC proposed General Design Criteria can usually be made. For the final GDC listed in the Regulatory Evaluation above, the Browns Ferry comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Browns Ferry UFSAR Appendix A: draft GDCs-1, 2, and 9. Final GDC-31 is applicable to Browns Ferry as described in "Browns Ferry Nuclear Plant (BFN), Unit 1- Application to Modify Technical Specification 3.4.9, 'RCS Pressure and Temperature (P/T) Limits' (BFN TS-484)," dated December 18, 2013 (Reference 8), "Browns Ferry Nuclear Plant (BFN), Unit 2 - Application to Modify Technical Specification 3.4.9, 'RCS Pressure and Temperature (P/T) Limits' (BFN TS-491)," dated June 19, 2014 (Reference 9), and "Browns Ferry Nuclear Plant, Unit 3 - Application to Modify Technical Specification 3.4.9, 'RCS Pressure and Temperature (P/T) Limits' (BFN TS-494)," dated January 27, 2015 (Reference 10).

The Reactor Coolant Pressure Boundary Materials is described in Browns Ferry UFSAR Sections 4.2, "Reactor Vessel and Appurtenances Mechanical Design," and 4.3, "Reactor Recirculation System."

In addition to the evaluations described in the Browns Ferry UFSAR, Browns Ferry's systems and components were evaluated for license renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Browns Ferry License Renewal Safety Evaluation Report (SER), NUREG-1843, dated April 2006 (Reference 11). The license renewal evaluation associated with the reactor coolant pressure boundary is documented in NUREG-1843, Sections 2.3.1 and 4.3.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of EPUs. Section10.7 of the CLTR addresses the effect of EPU on RCPB materials.

The temperature and flow increase experienced by the RCPB does not represent a significant increase in the potential for IGSCC. Other degradation mechanisms are addressed in other sections of this report. Fracture toughness of the vessel components is addressed in Section 2.1.2. Flow-Accelerated Corrosion (FAC) for the plant is addressed in Section 2.1.6. The structural evaluation of the RCPB piping is addressed in Section 2.2.2.2.1. Flow Induced Vibration (FIV) for the safety-related piping components is addressed in Section 2.2.2.1.3. Therefore, the current inspection strategy for the RCPB is adequate to manage any potential effects of EPU.

The Browns Ferry In-service Inspection (ISI) program for reactor coolant pressure boundary piping is in accordance with American Society of Mechanical Engineers (ASME) Section XI coupled with the augmented program for reactor coolant piping based on Generic Letter 88-01 (Reference 26), NUREG-0313 (Reference 27) and BWRVIP-75-A (Reference 28). The inspection techniques and NDE procedures utilized for ultrasonic examinations are qualified to the requirements of Appendix VIII of ASME Section XI (as implemented by the EPRI Performance Demonstration Initiative Program) for the detection and characterization of service-induced, surface-connected planar discontinuities, such as IGSCC.

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

The augmented inspection program is designed to detect potential degradation from IGSCC. For IGSCC to occur, three conditions must be present: (1) a susceptible material; (2) the presence of residual or applied tensile stress (such as from welding); and (3) aggressive environment. Operation at EPU conditions results in an insignificant change to temperature and flow conditions for portions of the RCPB piping and does not affect the other susceptibility factors associated with IGSCC. This is consistent with the conclusions presented in Section 3.6.1 of ELTR2.

The design of the RCPB piping and safe ends has been modified to reduce the amount of installed IGSCC susceptibility material. Table 2.1-9a lists the materials used in the Browns Ferry RCPB piping.

The RCPB weldments have been categorized and inspected in accordance with NUREG-0313 (Reference 27) and BWRVIP-75-A (Reference 28). Table 2.1-9b depicts the number of welds by category in each unit. The two Category G welds per unit are physically located inside containment penetrations, which prohibits direct physical examinations of the welds. Approval of alternative in-service inspection methods has been obtained and is being followed with respect to the Category G welds.

The nuclear industry has established that initiation and growth of IGSCC in stainless steel piping welds results from the combination of weld residual stress, an oxidizing environment, and a susceptible material. As described above, TVA has employed the use of IGSCC-resistant replacement material, applied weld stress improvement, and reduced the oxidizing environment with HWC. Operation at a higher power level will result in a slightly higher oxygen generation rate due to radiolysis of water; however, coolant chemistry will continue to be strictly controlled and maintained within specified limits. Implementation of EPU will not adversely affect the causative factors for IGSCC and, as such, the current established inspection and mitigation programs are adequate to support implementation of EPU.

Several IGSCC mitigation processes have been applied to Browns Ferry to reduce the RCPB components' susceptibility to IGSCC. Browns Ferry was designed, fabricated, and constructed with IGSCC addressed in most welds by one of three methods: (1) corrosion resistant materials; (2) solution heat treatment; or (3) clad with resistant materials. For the weldments where these three processes were not used, stress improvement processes were applied to reduce IGSCC susceptibility. Stress improvement processes and original construction processes used for IGSCC resistance are not affected by EPU. Also, Browns Ferry has implemented hydrogen water chemistry with noble metals, which reduces the potential for IGSCC of RCPB components.

In the reactor core, the bulk dissolved oxygen concentration depends on the amount of oxygen generated through radiolysis and the amount consumed in the recombination reaction with reactor water dissolved hydrogen. The rate of radiolytic generation is directly dependent on reactor power (neutron and gamma flux levels) and the reactor water dissolved hydrogen concentration is directly dependent on the rate of hydrogen injection to feedwater. As the rate of radiolytic generation of oxygen increases with higher EPU power levels, the hydrogen injection rate to feedwater will be increased proportionate to the increased feedwater flow rate. As such, the EPU feedwater hydrogen concentration will be the same as the CLTP feedwater hydrogen concentration. The EPU predicted hydrogen-to-feedwater injection rate increases by less than three scfm from CLTP, and the EPU predicted oxygen-to-offgas injection rate increases by less than 1.5 scfm from CLTP. These increases are well within the capacity of the existing Browns Ferry HWC systems. Monitoring of HWC system parameters will be performed under the existing site chemistry programs to ensure required injection rates for IGSCC mitigation at EPU conditions.

In addition to HWC, all Browns Ferry units have implemented the NobleChem™ (NMCA) process, with Browns Ferry Units 1 and 3 currently applying the annual On-line NobleChem™ (OLNC) injection process and Browns Ferry Unit 2 planning to transition from the Classic NMCA process to OLNC in 2015. The NobleChem™ processes are used in conjunction with HWC injection to feedwater to achieve IGSCC mitigation of reactor piping and internals at lower feedwater hydrogen addition rates than would be required for mitigation strategies that employ HWC only (e.g., Moderate HWC). Implementation of these programs at the Browns Ferry units

post-EPU will continue to be performed in accordance with the recommendations of the applicable EPRI BWRVIP guidelines and experience reports (References 29 through 32).

The primary parameters monitored for IGSCC mitigation at Browns Ferry are catalyst loading and Electrochemical Potential (ECP). The H_2:O_2 molar ratio (from the radiolysis/ECP model), hydrogen injection rate and reactor water oxygen concentration are secondary parameters monitored for IGSCC mitigation. These monitoring methods, currently employed or soon to be employed at all Browns Ferry units, will remain effective at EPU conditions.

Conclusion

TVA has evaluated the effects of the proposed EPU on the integrity of RCPB materials. The evaluation indicates that the RCPB materials will continue to be acceptable following implementation of the proposed EPU and will continue to meet the requirements of draft GDCs-1, 2, and 9, final GDC-31, 10 CFR Part 50, Appendix G, and 10 CFR 50.55a. Therefore, the proposed EPU is acceptable with respect to RCPB materials.

2.1.5 Protective Coating Systems (Paints) - Organic Materials

Regulatory Evaluation

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

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

Specific NRC review criteria are contained in SRP Section 6.1.2.

Browns Ferry Current Licensing Basis

The Browns Ferry current licensing basis regarding coatings is described in TVA letter to the NRC 120-day response to GL 98-04, dated November 10, 1998, "Browns Ferry Nuclear Plant (BFN), Sequoyah Nuclear Plant (SQN), and Watts Bar Nuclear Plant (WBN), 120-Day Response Generic Letter (GL) 98-04, 'Potential for Degradation of the ECCS and the Containment Spray System (CSS) After a Loss-of-Coolant Accident (LOCA) Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment,' Dated July 14, 1998." (Reference 33)

Technical Evaluation

The TVA protective coating program (Reference 34) 1) lists the coating systems approved for use in TVA nuclear plants; 2) separates them into service levels (SLs) designated SL-I, SL-II, SL-III; 3) provides the temperature and radiation qualification for SL-I; 4) defines the suitable application (i.e., dry, high humidity, immersion); and 5) defines proper surface preparation and application techniques. All SL-I coatings used at Browns Ferry are qualified for Design Basis

Accident (DBA) conditions of temperature, pressure, radiation and chemical effects which bound worst case conditions at EPU. SL-I coating is required in the primary containment.

Regulatory requirements such as 10 CFR 50 Appendix B, Regulatory Guide 1.54, GL 98-04, NUREG-1801, Information Notices, IE circulars, industry standards American Society for Testing and Materials (ASTM) D5144, ASTM D3843, ANSI 5.12, EPRI 1003102, and the TVA Nuclear Quality Assurance Manual (NQAM) are promulgated through Reference 34.

The Browns Ferry Service Level I coatings are subject to the requirements of Regulatory Guide 1.54 – 1973 (Reference 35), American National Standard Institute (ANSI) N101.2 – 1972 (Reference 36) and ANSI N101.4 – 1972 (Reference 37). The qualification testing for Service Level I coatings used for new applications or repair/replacement activities inside containment meets the applicable requirements contained in the standards and regulatory commitments listed above.

At EPU, the accumulated gamma dose for the DBA-LOCA is 1.5E8 Rad and is bounded by the SL-I coating qualification level of 1.0E9 Rad. At EPU, the peak drywell pressure and temperature for all LOCA events (See Table 2.6-1) are 50.9 psig (peak value for DBA-LOCA) and 336.9°F (steam line break LOCA), which are bounded by the SL-I coating qualification level of 70 psig pressure and 340°F temperature. The chemical constituency of the primary containment post-LOCA does not change as a result of EPU.

The Service Level I coatings approved for use at Browns Ferry and applied inside containment are listed in the table below:

Coating System	**Qualification Temperature (°F)**	**Qualification Dose (rads)**	**Qualification for Immersion**	**Application**
Valspar 78[(1)] AKA Vygard 78	340	$\geq 1 \times 10^9$	Yes	Unit 1 torus above water line tie in band Unit 2 and 3 torus
D6/Amercoat 90N	340	$\geq 1 \times 10^9$	No	Drywell liner
Amerlock 400 NT	340	$\geq 1 \times 10^9$	No	Drywell structural steel
UT-15	340	$\geq 1 \times 10^9$	Yes	Torus underwater repair
Kolor-Poxy 6548/7107	340	$\geq 1 \times 10^9$	Yes	Unit 1 torus immersion zone
Bio-Dur 561	340	$\geq 1 \times 10^9$	Yes	Torus underwater repair
Plasite [(1)]	340	$\geq 1 \times 10^9$	Yes	Unit 1 vapor space

Note:

1. Coating system has been discontinued and is no longer applied. However, the coating is still resident inside the primary containment.

The Service Level I protective coating systems used inside the containment were evaluated for their continued suitability for and stability under DBA-LOCA and HELB conditions, considering radiation, temperature, pressure, and chemical effects at EPU conditions. The Harsh Environmental Data drawings and supporting calculations currently include the effect of life extension and EPU.

Browns Ferry inspects the containment coating in accordance with plant procedures each refueling outage looking for failed or damaged coating. Coating conditions monitored by this program include checking for cracking, blistering, flaking, scaling, peeling, rust through, tiger striping, discoloration, embrittlement or mechanical damage. Any failed or damaged coating is remediated in accordance with plant procedures. The condition assessments and resulting repair, replacement, or removal activities ensure that the amount of coatings subject to detachment from the substrate during a LOCA is minimized to ensure post-accident operability of the ECCS suction strainers.

The inspection of the coating in the immersion zone of the torus is coordinated with the desludging of the torus which is frequency based. Inspection of the immersion zone has typically occurred every second or third outage.

Uncontrolled coatings are also identified and tracked to ensure the amount of uncontrolled coating which could contribute to ECCS strainer blockage is maintained below the established limit (157 ft^2) used in the design of the replacement ECCS suction strainers (Reference 38) installed in the Browns Ferry units. EPU does not change this limit. In accordance with Reference 34 requirements, Browns Ferry inspects SL-I coatings during each refueling outage. This inspection periodicity does not change with EPU.

Based on the conservative analysis summarized above, Browns Ferry has determined that reasonable assurance exists that when properly applied and maintained, the SL-I systems used in the primary containment will not detach under normal or accident conditions with the plant operating at EPU conditions.

Conclusion

TVA has evaluated the effects of the proposed EPU on the protective coatings. The evaluation indicates that the protective coatings will continue to be acceptable following implementation of the proposed EPU and will continue to meet the requirements of 10 CFR Part 50, Appendix B. Therefore, the proposed EPU is acceptable with respect to protective coatings.

2.1.6 Flow-Accelerated Corrosion

Regulatory Evaluation

FAC is a corrosion mechanism occurring in carbon steel components exposed to flowing single- or two-phase water. Components made from stainless steel are immune to FAC, and FAC is significantly reduced in components containing small amounts of chromium or molybdenum. The rates of material loss due to FAC depend on velocity of flow, fluid temperature, steam quality, oxygen content, and pH. During plant operation, control of these parameters is limited and the optimum conditions for minimizing FAC effects, in most cases, cannot be achieved. Loss of material by FAC will, therefore, occur.

Browns Ferry's FAC program is based on NUREG-1344, GL 89-08, and the guidelines in Electric Power Research Institute (EPRI) Report NSAC-202L-R4. It consists of predicting loss of material using the CHECWORKS™ computer code, and visual inspection and volumetric examination of the affected components. The NRC's acceptance criteria are based on the structural evaluation of the minimum acceptable wall thickness for the components undergoing degradation by FAC.

Browns Ferry Current Licensing Basis

The Browns Ferry program for addressing Flow Accelerated Corrosion is described in a TVA letter, dated July 19, 1989, "Response to Generic Letter 89-08 - Erosion/Corrosion-Induced Pipe Wall Thinning" (Reference 39). This response provided information regarding administrative controls, procedures, and engineering activities associated with this program.

In addition to the evaluations described in the Browns Ferry UFSAR, Browns Ferry's systems and components were evaluated for license renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Browns Ferry License Renewal Safety Evaluation Report (SER), NUREG-1843, dated April 2006 (Reference 11). The license renewal evaluation associated with the FAC program is documented in NUREG-1843, Section 3.0.3.2.9.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of EPUs. Section 10.7 of the CLTR addresses the effect of EPU on FAC.

Browns Ferry meets all CLTR dispositions. The results of this evaluation are described below.

Topic	CLTR Disposition	Browns Ferry Result
Flow Accelerated Corrosion	Plant Specific	Meets CLTR Disposition

The CLTR states that the increase in steam and FW flow rate as a result of EPU influence FAC. In order to monitor and control FAC, Browns Ferry maintains an effective FAC program. The EPU implementation at Browns Ferry will change a number of system water and steam flow rates, temperatures, and enthalpies, in turn changing dissolved oxygen concentration. All these factors affect FAC susceptibility status and FAC wear rates. As a result of EPU operating conditions, some lines will experience accelerated rates of FAC, while others will have reduced rates. It should be noted that no lines that were previously non-susceptible to FAC (as defined by the EPU heat balance) will become susceptible due to EPU operating conditions.

[[

]] The FAC program will not significantly change for EPU.

The FAC program at Browns Ferry is based on:

- NRC I&E Bulletin 87-01, "Thinning Pipe Walls in Nuclear Power Plants" (Reference 40)
- Generic Letter 89-08, "Erosion/Corrosion-Induced Pipe Wall Thinning" (Reference 39)
- EPRI NSAC-202L-R4, "Recommendations for an Effective Flow Accelerated Corrosion Program," Revision 4, November 2014 (Reference 41)

With regard to EPRI NSAC-202L-R4, the choice of the method for detecting and evaluating the effect of FAC on a component is dependent on the type of component and its history. Results of the evaluation reveal if the component will remain above minimum allowable wall thickness throughout the next operating cycle and what the predicted minimum wall thickness will be at the end of the operating cycle. Additionally, the evaluation shows the remaining service life of the component (based on the calculated minimum allowable wall thickness) and the Next Scheduled Inspection (NSI) outage. The NSI is an outage prior to the time that the component reaches minimum allowable wall thickness. Component wall thickness is analyzed using minimum wall thickness according to the Browns Ferry design methodology and acceptable only if it meets all the design requirements of Browns Ferry.

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

The FAC susceptible lines that do not meet the minimum requirements for modeling and analysis by CHECWORKS™ SFA are referred to as "Susceptible Non-Modeled" (SNM). This group is comprised of lines with unknown or widely varying operating conditions that prevent the

development of accurate predictive models. It includes bypass lines, recirculation lines, vent lines, high level dumps, and socket welded piping. Some small bore piping and piping susceptible to wall thinning mechanisms other than FAC are also included in this group. Selection of this piping for inspection is typically the result of industry experience, Browns Ferry experience, or engineering judgment.

One of the most import aspects of the Browns Ferry FAC program is the proper selection of locations for FAC inspection and subsequent replacement of degraded piping. This is accomplished using the following (detailed in Table 2.1-8):

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

The proposed EPU may affect the following aspects of the Browns Ferry FAC program.

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

These are evaluated as follows:

FAC System Susceptibility Evaluation

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

Wear Rates – CHECWORKS™ SFA Model Update for EPU

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

Tables 2.1-4a, 4b and 4c contains a listing of the CHECWORKS™ SFA run definitions (i.e., compilations of lines with similar operating conditions, water chemistry and usage

for analysis). A comparison of CLTP and EPU wear rate predictions identified changes for each unit.

- For Browns Ferry Unit 1 (Table 2.1-4a), 12 wear rate analysis runs were done: there was decrease of 30.2% to an increase of 16.74%. Of the run definitions for Unit 1, six had a decrease in the predicted wear rate while the remaining six run definitions exhibited an increase.
- For Browns Ferry Unit 2 (Table 2.1-4b), 13 wear rate analysis runs were done: there was decrease of 29.86% to an increase of 19.35%. Of the run definitions for Unit 2, six had a decrease in the predicted wear rate while the remaining seven run definitions exhibited an increase.
- For Browns Ferry Unit 3 (Table 2.1-4c), 13 wear rate analysis runs were done: there was decrease of 29.87% to an increase of 16.26%. Of the run definitions for Unit 3, seven had a decrease in the predicted wear rate while the remaining six run definitions exhibited an increase.

Based on a review of the changes in operating conditions, Browns Ferry found the resulting predicted wear rates to be consistent with EPU conditions.

Selection of Inspection and Replacement Locations

The current approach to select locations for FAC inspection does not change as a result of the EPU. However, there will be an increase in the number of FAC inspections performed on both CHECWORKS™ SFA-modeled and SNM piping over the next several refueling outages to ensure the effect of extended power uprate is understood. Inspections will be selected considering the changes in predicted wear rates, actual component thicknesses, operating time since last examination and design margin. This approach will ensure that FAC susceptible components are inspected or replaced prior to reaching code minimum wall thickness. Based on the EPU evaluation, no significant effect on the component replacement schedule is anticipated in the near term. The continued implementation of the existing Browns Ferry FAC program, updated appropriately to include EPU system parameters, will ensure that any required changes to the component inspection and replacement schedules are made prior to EPU implementation.

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

Benchmarking CHECWORKS™ SFA Predicted Component Thickness

Tables 2.1-5a, 5b and 5c presents a comparison of CHECWORKS™-predicted thicknesses to measured thicknesses for a sample component from each of the Wear Rate Analysis (WRA) run definitions. The selection process includes components with the highest predicted wear rates prior to EPU for each unit. The measured thicknesses were determined by ultrasonic testing non-destructive examination performed during the refueling outage as noted in the tables.

The tables show that, with the exception of two cases (one in Browns Ferry Unit 2 and the other in Browns Ferry Unit 3), the measured thickness from the inspection was greater than the predicted thickness, indicating that CHECWORKS™ SFA predictions are typically conservative.

Other than FAC, Browns Ferry also inspects certain components for degradation caused by Liquid Droplet Impingement (LDI). Indications that LDI may be present are valve leak-bys, or conditions (open valves, leaks) that cause the velocity of the two-phased mixture to increase dramatically. The FAC program also inspects for cavitation per system engineering requests.

The Browns Ferry FAC program adequately manages the effects on FAC due to EPU. Therefore, Browns Ferry meets all CLTR dispositions for FAC.

Conclusion

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

2.1.7 Reactor Water Cleanup System

Regulatory Evaluation

The Reactor Water Cleanup (RWCU) system provides a means for maintaining reactor water quality by filtration and ion exchange and a path for removal of reactor coolant when necessary. Portions of the RWCU system comprise the RCPB.

The NRC's acceptance criteria for the RWCU system are based on (1) GDC-14, insofar as it requires that the RCPB be designed, fabricated, erected, and tested so as to have an extremely low probability of rapidly propagating fracture; (2) GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents; and (3) GDC-61, insofar as it requires that systems that contain radioactivity be designed with appropriate confinement.

Specific NRC review criteria are contained in SRP Section 5.4.8.

Browns Ferry Current Licensing Basis

The General Design Criteria (GDC) listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Browns Ferry Nuclear Plant (Browns Ferry) principal design criteria predate these criteria. The Browns Ferry principal design criteria are listed in UFSAR Section 1.5, "Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, the Tennessee Valley Authority (TVA) performed a comparative evaluation of the design basis of Browns Ferry with the AEC proposed General Design Criteria of 1967. The Browns Ferry UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria set out in the

July 1967 AEC release. For each group of criteria, there is a statement of TVA's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a table of references to locations in the Browns Ferry UFSAR where there is subject matter relating to the intent of that particular criteria.

While Browns Ferry is not generally licensed to the final GDC or the 1967 AEC proposed General Design Criteria, a comparison of the final GDC to the applicable AEC proposed General Design Criteria can usually be made. For the final GDC listed in the Regulatory Evaluation above, the Browns Ferry comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Browns Ferry UFSAR Appendix A: draft GDCs-9, 34, 51, and 70.

The Reactor Water Cleanup System is described in Browns Ferry UFSAR Section 4.9, "Reactor Water Cleanup System."

In addition to the evaluations described in the Browns Ferry UFSAR, Browns Ferry's systems and components were evaluated for license renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Browns Ferry License Renewal Safety Evaluation Report (SER), NUREG-1843, dated April 2006 (Reference 11). The license renewal evaluation associated with the Reactor Water Cleanup System is documented in NUREG-1843, Section 2.3.3.21. Management of aging effects on the Reactor Water Cleanup System is documented in NUREG-1843, Section 3.0.3.2.15.

Technical Evaluation

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

The RWCU system is a normally operating system with no safety-related functions other than RCPB and containment isolation. 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 evaluation of the system performance of the Browns Ferry RWCU system under EPU conditions is presented below. The effects of EPU on the RWCU containment isolation function and valves are included in the containment isolation assessment in Sections 2.2.4 and 2.6.1.3.

Tables 2.1-6 and 2.1-7 contain the magnitude of changes in RWCU system operating conditions and a summary of the chemistry values. Browns Ferry meets all CLTR dispositions. The topics addressed in this evaluation are:

Topic	CLTR Disposition	Browns Ferry Result
System Performance	Plant Specific	Meets CLTR Disposition
Containment Isolation	Plant Specific	Addressed in Section 2.6.1.3

As explicitly stated in Section 3.11 of the CLTR, the RWCU system may be slightly affected by the increase in FW flow due to the power uprate.

RWCU system operation at the EPU RTP level slightly decreases the temperature within the RWCU system (from 530.5°F to 529.3°F). This system currently operates at flow rates consistent with the original design flow. The operating flow rates are not being changed for EPU. Table 2.1-6 provides the magnitude of changes in RWCU system operating conditions (e.g., a decrease in operating inlet temperature).

RWCU system flow is usually selected to be approximately 1% of FW system flow based on operational history. For the Browns Ferry EPU, the RWCU system was analyzed for flow at 133,300 lbm/hr. This flow rate is approximately 0.81% of EPU rated FW flow. The evaluation of RWCU performance for the Browns Ferry EPU considered water chemistry, heat exchanger performance, pump performance, flow control valve capability and filter / demineralizer performance. All aspects of performance were found to be within the design of the RWCU system at the analyzed flow for EPU conditions. The RWCU system analysis concludes that:

1. An increase in filter / demineralizer backwash frequency occurs, but this is within the capacity of the radwaste system.
2. The changes in operating system conditions result from a decrease in inlet temperature and a negligible increase in FW system operating pressure.
3. The RWCU system filter / demineralizer control valves will operate in the slightly more open position because of the negligible increase to the RWCU system discharge pressure.
4. No changes to instrumentation are required, and setpoint changes are not required due to the system process parameter changes.

Previous operating experience has shown that the increased FW flow results in increases in three key reactor coolant chemistry parameters. Table 2.1-7 provides a summary of the chemistry values and the evaluation results for each are presented below. These values use the maximum values for actual plant rolling averages for all three plants:

- Sulfates concentration – The current maximum average level of sulfates is 1.29 ppb for all three units. The expected reactor water sulfate level for EPU, considering the FW flow increase, is 1.50 ppb. This level is well below the administrative goal of 2.0 ppb and the action level of 5.0 ppb for sulfates.

- Chlorides concentration – The current maximum average level of chlorides is 0.33 ppb for all three units. The expected reactor water chloride level for EPU, considering the FW flow increase, is 0.38 ppb. This level is well below the administrative goal of 1.0 ppb and the action level of 5.0 ppb for chlorides.
- Reactor water conductivity – The calculated reactor water conductivity increases from 0.121 μS/cm to 0.132 μS/cm because of the increase in FW flow. This expected level is below the administrative goal for conductivity of 0.14 μS/cm and the action level of 0.30 μS/cm.

The effects of EPU on the RWCU system functional capability have been reviewed, and the system can perform adequately at EPU RTP with the CLTP RWCU system flow. As can be seen from Table 2.1-6, the changes in RWCU system operating conditions from CLTP to EPU are small. The Browns Ferry RWCU system has sufficient capacity to respond to the EPU conditions and maintain the chemistry parameters within administrative goals. Therefore, Browns Ferry meets all CLTR dispositions for system performance.

Conclusion

TVA has evaluated the effects of the proposed EPU on the RWCU system. The evaluation indicates that the RWCU system will continue to be acceptable following implementation of the proposed EPU and will continue to meet the requirements of the draft GDCs-9, 34, 51, and 70. Therefore, the proposed EPU is acceptable with respect to the RWCU system.

Table 2.1-1a Browns Ferry Unit 1 USE EMA – 60 Year Life (38 EFPY)

Equivalent Margin Analysis (EMA)
Plant Applicability Verification Form
for Browns Ferry Unit 1

60 Years (38 EFPY)

BWR/3-6 PLATE

Surveillance Plate USE:

%Cu	=	N/A	
1st Capsule Fluence	=	N/A n/cm^2	
1st Capsule Measured % Decrease	=	N/A	(Charpy Curves)
1st Capsule RG 1.99 Predicted % Decrease	=	N/A	(RG 1.99, Rev. 2, Figure 2)
Ratio of Measured to Predicted % Decrease	=	N/A	(RG 1.99, Rev. 2, Position 2.2)

Limiting Beltline Plate USE (Heat C2884-2):

%Cu	=	0.12	
38 EFPY 1/4T Fluence	=	1.09E+18 n/cm^2	
RG 1.99 Predicted % Decrease	=	13.0	(RG 1.99, Rev. 2, Figure 2)
Adjusted % Decrease	=	N/A	(RG 1.99, Rev. 2, Position 2.2)

13.0% $\leq$ [[ ]]

Therefore, vessel plates are bounded by Equivalent Margin Analysis

Equivalent Margin Analysis
Plant Applicability Verification Form
for Browns Ferry Unit 1

60 Years (38 EFPY)

BWR/2-6 WELD

Surveillance Weld USE <u>**406L44**</u>

%Cu	=	0.29	
1st Capsule Fluence	=	1.00E+18 n/cm^2	
2nd Capsule Fluence	=	1.83E+18 n/cm^2	
3rd Capsule Fluence	=	1.77E+18 n/cm^2	
4th Capsule Fluence	=	2.89E+18 n/cm^2	
5th Capsule Fluence	=	3.97E+17 n/cm^2	
6th Capsule Fluence	=	4.93E+17 n/cm^2	
1st Capsule Measured % Decrease	=	32.0	(Charpy Curves)
1st Capsule RG 1.99 Predicted % Decrease	=	26.0	(RG 1.99, Rev. 2, Figure 2)
2nd Capsule Measured % Decrease	=	33.0	(Charpy Curves)
2nd Capsule RG 1.99 Predicted % Decrease	=	29.5	(RG 1.99, Rev. 2, Figure 2)
3rd Capsule Measured % Decrease	=	36.5	(Charpy Curves)
3rd Capsule RG 1.99 Predicted % Decrease	=	29.5	(RG 1.99, Rev. 2, Figure 2)
4th Capsule Measured % Decrease	=	42.5	(Charpy Curves)
4th Capsule RG 1.99 Predicted % Decrease	=	32.5	(RG 1.99, Rev. 2, Figure 2)
5th Capsule Measured % Decrease	=	20.5	(Charpy Curves)
5th Capsule RG 1.99 Predicted % Decrease	=	21.0	(RG 1.99, Rev. 2, Figure 2)
6th Capsule Measured % Decrease	=	21.0	(Charpy Curves)
6th Capsule RG 1.99 Predicted % Decrease	=	22.0	(RG 1.99, Rev. 2, Figure 2)
Ratio of Measured to Predicted % Decrease (4th Capsule)	=	1.3	(RG 1.99, Rev. 2, Position 2.2)

Limiting Beltline Weld USE (406L44):

%Cu	=	0.29	
38 EFPY 1/4T Fluence	=	8.86E+17 n/cm^2	
RG 1.99 Predicted % Decrease	=	25	(RG 1.99, Rev. 2, Figure 2)
Adjusted % Decrease	=	33.5	(RG 1.99, Rev. 2, Position 2.2)

33.5% $\leq$ [[ ]]

Therefore, vessel welds are bounded by Equivalent Margin Analysis

Table 2.1-1b Browns Ferry Unit 2 USE EMA – 60-Year Life (48 EFPY)

Equivalent Margin Analysis
Plant Applicability Verification Form
for Browns Ferry Unit 2

60 Years (48 EFPY)

BWR/3-6 PLATE

Surveillance Plate USE[1] A0981-1

%Cu	=	0.14	
1st Capsule Fluence	=	2.40E+17 n/cm^2	
2nd Capsule Fluence	=	6.44E+17 n/cm^2	
1st Capsule Measured % Decrease	=	6	(Charpy Curves)
1st Capsule RG 1.99 Predicted % Decrease	=	9.5	(RG 1.99, Rev. 2, Figure 2)
2nd Capsule Measured % Decrease	=	-3.6 [1]	(Charpy Curves)
2nd Capsule RG 1.99 Predicted % Decrease	=	12	(RG 1.99, Rev. 2, Figure 2)
Ratio of Measured to Predicted % Decrease	=	<1	(RG 1.99, Rev. 2, Position 2.2)

Limiting Beltline Plate USE (Heat C2467-1):

%Cu	=	0.16	
48 EFPY 1/4T Fluence	=	1.34E+18 n/cm^2	
RG 1.99 Predicted % Decrease	=	16.0	(RG 1.99, Rev. 2, Figure 2)
Adjusted % Decrease	=	N/A	(RG 1.99, Rev. 2, Position 2.2)

16.0% $\leq$ [[ ]]

Therefore, vessel plates are bounded by Equivalent Margin Analysis

Note [1]: The 2nd capsule measured results demonstrated an increase in USE.

Equivalent Margin Analysis
Plant Applicability Verification Form
for Browns Ferry Unit 2

60 Years (48 EFPY)

BWR/2-6 WELD

Surveillance Weld USE BF2 ESW

%Cu	=	0.20	
1st Capsule Fluence	=	2.40E+17 n/cm^2	
2nd Capsule Fluence	=	6.44E+17 n/cm^2	
1st Capsule Measured % Decrease	=	5.9	(Charpy Curves)
1st Capsule RG 1.99 Predicted % Decrease	=	14.1	(RG 1.99, Rev. 2, Figure 2)
2nd Capsule Measured % Decrease	=	3.4	(Charpy Curves)
2nd Capsule RG 1.99 Predicted % Decrease	=	17.8	(RG 1.99, Rev. 2, Figure 2)
Ratio of Measured to Predicted % Decrease	=	<1	(RG 1.99, Rev. 2, Position 2.2)

Limiting Beltline Weld USE (ESW):

%Cu	=	0.24	
48 EFPY 1/4T Fluence	=	9.14E+17 n/cm^2	
RG 1.99 Predicted % Decrease	=	22.0	(RG 1.99, Rev. 2, Figure 2)
Adjusted % Decrease	=	N/A	(RG 1.99, Rev. 2, Position 2.2)

22.0% $\leq$ [[ ]]

Therefore, vessel welds are bounded by Equivalent Margin Analysis

Table 2.1-1c Browns Ferry Unit 3 USE EMA – 60-Year Life (54 EFPY)

Equivalent Margin Analysis
Plant Applicability Verification Form
for Browns Ferry Unit 3

60 Years (54 EFPY)

BWR/3-6 PLATE

Surveillance Plate USE:

%Cu	=	N/A	
1st Capsule Fluence	=	N/A n/cm^2	
1st Capsule Measured % Decrease	=	N/A	(Charpy Curves)
1st Capsule RG 1.99 Predicted % Decrease	=	N/A	(RG 1.99, Rev. 2, Figure 2)
Ratio of Measured to Predicted % Decrease	=	N/A	(RG 1.99, Rev. 2, Position 2.2)

Limiting Beltline Plate USE (Heat C3222-2):

%Cu	=	0.15	
54 EFPY 1/4T Fluence	=	1.25E+18 n/cm^2	
RG 1.99 Predicted % Decrease	=	15.0	(RG 1.99, Rev. 2, Figure 2)
Adjusted % Decrease	=	N/A	(RG 1.99, Rev. 2, Position 2.2)

15.0% $\leq$ [[ ]]

Therefore, vessel plates are bounded by Equivalent Margin Analysis

Equivalent Margin Analysis
Plant Applicability Verification Form
for Browns Ferry Unit 3

60 Years (54 EFPY)

BWR/2-6 WELD

Surveillance Weld USE:

%Cu	=	N/A	
1st Capsule Fluence	=	N/A n/cm^2	
1st Capsule Measured % Decrease	=	N/A	(Charpy Curves)
1st Capsule RG 1.99 Predicted % Decrease	=	N/A	(RG 1.99, Rev. 2, Figure 2)

Limiting Beltline Weld USE (ESW):

%Cu	=	0.24	
54 EFPY 1/4T Fluence	=	1.05E+18 n/cm^2	
RG 1.99 Predicted % Decrease	=	23.0	(RG 1.99, Rev. 2, Figure 2)
Adjusted % Decrease	=	N/A	(RG 1.99, Rev. 2, Position 2.2)

23.0% $\leq$ **[[ ]]**

Therefore, vessel welds are bounded by Equivalent Margin Analysis

Table 2.1-2a Browns Ferry Unit 1 Adjusted Reference Temperatures 60-Year License (38 EFPY)

Lower-Intermediate Plates

Thickness in inches = 6.125 — 38 EFPY Peak I.D. fluence = 1.58E+18 n/cm^2; 38 EFPY Peak 1/4 T fluence = 1.09E+18 n/cm^2

Lower Plates & Lower to Lower-Intermediate Girth Weld

Thickness in inches = 6.125 — 38 EFPY Peak I.D. fluence = 1.28E+18 n/cm^2; 38 EFPY Peak 1/4 T fluence = 8.86E+17 n/cm^2

Axial Welds

Thickness in inches = 6.125 — 38 EFPY Peak I.D. fluence = 1.58E+18 n/cm^2; 38 EFPY Peak 1/4 T fluence = 1.09E+18 n/cm^2

Water Level Instrumentation Nozzle

Thickness in inches = 6.125 — 38 EFPY Peak I.D. fluence = 4.77E+17 n/cm^2; 38 EFPY Peak 1/4 T fluence = 3.30E+17 n/cm^2

COMPONENT	HEAT OR HEAT/LOT	%Cu	%Ni	CF	Adjusted CF	Initial RT_{NDT} °F	1/4 T Fluence n/cm^2	38 EFPY 1/4T Δ RT_{NDT} °F	σ_I	σ_Δ	Margin °F	38 EFPY 1/4T Shift °F	38 EFPY 1/4T ART °F
PLATES:													
Lower Shell													
6-127-1	A0999-1	0.14	0.60	100		-20	8.86E+17	39	0	17	34	73	53
6-127-2	B5864-1	0.15	0.44	101		-20	8.86E+17	40	0	17	34	74	54
6-127-4	A1009-1	0.14	0.50	96		-10	8.86E+17	38	0	17	34	72	62
Lower-Intermediate Shell													
6-139-19	C2884-2	0.12	0.53	82		14	1.09E+18	36	0	17	34	70	84
6-139-20	C2868-2	0.09	0.48	58		30	1.09E+18	25	0	13	25	50	80
6-139-21	C2753-1	0.08	0.50	51		2	1.09E+18	22	0	11	22	44	46
WELDS:													
Axial Welds													
ESW	--	0.24	0.37	141		23.1	1.09E+18	61	13	28	62	123	146
Lower to Lower-Intermediate Girth Weld													
WF154	406L44	0.27	0.60	184		20	8.86E+17	72	10	28	59	132	152
BEST ESTIMATE CHEMISTRIES:													
None													
NOZZLES:													
N16 Water Level Instrumentation													
Forging	Inconel [1]	0.12	0.53	82		14	3.30E+17	19	0	9	19	38	52
Weld	Inconel												
INTEGRATED SURVEILLANCE PROGRAM:													
Plate [2]	A0981-1												
Weld [3, 4]	SSP-406144	0.29	0.69	205	[[ ]]	23.1	8.86E+17	110	10	28	59	170	193

Notes:

[1] The material properties used are those for the bounding adjacent shell plate from the lower-intermediate shell. The fluence considered is applicable at the nozzle location.

[2] The representative plate material is not the same heat number as the target plate; therefore the RG 1.99 Chemistry Factor (CF) is used. This information is not applicable to development of the P-T curves and is provided for information only.

[3] The initial RT_{NDT} is obtained from the limiting plant-specific plate and weld.

[4] The representative weld material is the same heat as the target weld; therefore, these results are considered in development of the P-T curves. Surveillance data from six (6) capsules are available. Scatter of this data exceeds credibility criteria. The fitted CF of [[ ]], based on the surveillance data, is higher than the RG 1.99 table CF of 204.95°F; therefore, the full margin term is applied. The CF is adjusted per RG 1.99 to be: (184°F/205°F) * [[ ]].

Table 2.1-2b Browns Ferry Unit 2 Adjusted Reference Temperatures 60-Year License (48 EFPY)

		Lower-Intermediate Plates	
Thickness =	6.125	inches	48 EFPY Peak I.D. fluence = 1.93E+18 n/cm²
			48 EFPY Peak 1/4 T fluence = 1.34E+18 n/cm²
		Lower Plates & Lower to Lower-Intermediate Girth Weld	
Thickness =	6.125	inches	48 EFPY Peak I.D. fluence = 1.56E+18 n/cm²
			48 EFPY Peak 1/4 T fluence = 1.08E+18 n/cm²
		Axial Welds	
Thickness =	6.125	inches	48 EFPY Peak I.D. fluence = 1.32E+18 n/cm²
			48 EFPY Peak 1/4 T fluence = 9.14E+17 n/cm²
		Water Level Instrumentation Nozzle	
Thickness =	6.125	inches	48 EFPY Peak I.D. fluence = 5.84E+17 n/cm²
			48 EFPY Peak 1/4 T fluence = 4.04E+17 n/cm²

COMPONENT	HEAT OR HEAT/LOT	%Cu	%Ni	CF	Adjusted CF	Initial RT_{NDT} °F	1/4 T Fluence n/cm²	48 EFPY 1/4 T ΔRT_{NDT} °F	σ_I	σ_Δ	Margin °F	48 EFPY 1/4 T Shift °F	48 EFPY 1/4 T ART °F
PLATES:													
Lower Shell													
6-127-14	C2467-2	0.16	0.52	112		-20	1.08E+18	48	0	17	34	82	62
6-127-15	C2463-1	0.17	0.48	117		-20	1.08E+18	51	0	17	34	85	65
6-127-17	C2460-2	0.13	0.51	88		0	1.08E+18	38	0	17	34	72	72
Lower-Intermediate Shell													
6-127-6	A0981-1	0.14	0.55	98		-10	1.34E+18	47	0	17	34	81	71
6-127-16	C2467-1	0.16	0.52	112		-10	1.34E+18	53	0	17	34	87	77
6-127-20	C2849-1	0.11	0.50	73		-10	1.34E+18	35	0	17	34	69	59
WELDS:													
Axial Welds													
ESW	--	0.24	0.37	141		23.1	9.14E+17	56	13	28	62	118	141
Lower to Lower-Intermediate Girth Weld													
	D55733	0.09	0.65	117		-40	1.08E+18	51	0	25	51	101	61
BEST ESTIMATE CHEMISTRIES:													
None													
NOZZLES:													
N16 Water Level Instrumentation													
Forging	Inconel [1]	0.16	0.52	112		-10	4.04E+17	29	0	15	29	58	48
Weld	Inconel												
INTEGRATED SURVEILLANCE PROGRAM:													
Plate [2, 3]	A0981-1	0.14	0.55	[[ ]]		-10	1.34E+18	68	0	8.5	17	85	75
Weld [4, 5]	BF2 ESW	0.20	0.33		[[ ]]	23.1	9.14E+17	114	13	14	38	152	175

Notes:

[1] The material properties used are those for the bounding adjacent shell plate from the lower-intermediate shell. The fluence considered is applicable at the nozzle location.

[2] The representative plate material is not the same heat as the target plate; therefore, the Position 1.1 RG 1.99 CF is used. This information is, however, applicable to development of the P-T curves because the same heat of material exists in the Unit 2 vessel lower-intermediate shell. The surveillance data is credible; therefore σ_Δ is reduced as permitted by RG 1.99.

[3] The initial RT_{NDT} is obtained from the plant-specific material in the lower-intermediate shell.

[4] The initial RT_{NDT} is obtained from the plant-specific material in the axial welds.

[5] The representative weld material is considered to be the same heat as the target weld; therefore, these results are considered in the development of the P-T curves. As this material is considered to be the same heat as the Unit 2 vessel, the adjusted CF is calculated per RG 1.99 to be (140.55°F/120.25°F) * [[ ]]. The surveillance data is credible; therefore σ_Δ is reduced as permitted by RG 1.99.

Table 2.1-2c Browns Ferry Unit 3 Adjusted Reference Temperatures 60-Year License (54 EFPY)

Lower-Intermediate Plates

Thickness =	6.125	inches	54 EFPY Peak I.D. fluence =	2.23E+18	n/cm^2
			54 EFPY Peak 1/4 T fluence =	1.54E+18	n/cm^2

Lower Plates & Lower to Lower-Intermediate Girth Weld

Thickness =	6.125	inches	54 EFPY Peak I.D. fluence =	1.80E+18	n/cm^2
			54 EFPY Peak 1/4 T fluence =	1.25E+18	n/cm^2

Axial Welds

Thickness =	6.125	inches	54 EFPY Peak I.D. fluence =	1.52E+18	n/cm^2
			54 EFPY Peak 1/4 T fluence =	1.05E+18	n/cm^2

Water Level Instrumentation Nozzle

Thickness =	6.125	inches	54 EFPY Peak I.D. fluence =	6.75E+17	n/cm^2
			54 EFPY Peak 1/4 T fluence =	4.67E+17	n/cm^2

COMPONENT	HEAT OR HEAT/LOT	%Cu	%Ni	CF	Initial RT_{NDT} °F	1/4 T Fluence n/cm^2	1/4T 54 EFPY Δ RT_{NDT} °F	σ_I	σ_Δ	Margin °F	1/4T 54 EFPY Shift °F	1/4T 54 EFPY ART °F
PLATES:												
Lower Shell												
6-145-4	C3222-2	0.15	0.52	106	10	1.25E+18	49	0	17	34	83	93
6-145-7	C3213-1	0.13	0.58	90	-20	1.25E+18	42	0	17	34	76	56
6-145-12	C3217-2	0.14	0.66	101.5	-4	1.25E+18	47	0	17	34	81	77
Lower-Intermediate Shell												
6-145-1	C3201-2	0.13	0.60	91	-20	1.54E+18	46	0	17	34	80	60
6-145-2	C3188-2	0.10	0.48	65	-20	1.54E+18	33	0	17	33	66	46
6-145-6	B7267-1	0.13	0.51	88	-20	1.54E+18	45	0	17	34	79	59
WELDS:												
Axial Welds												
ESW	--	0.24	0.37	141	23.1	1.05E+18	60	13	28	62	122	145
Lower to Lower-Intermediate Girth Weld												
	D55733	0.09	0.66	117	-40	1.25E+18	54	0	27	54	108	68
BEST ESTIMATE CHEMISTRIES:												
None												
NOZZLES:												
N16 Water Level Instrumentation												
Forging	Inconel [1]	0.13	0.6	91	-20	4.67E+17	26	0	13	26	51	31
Weld	Inconel											
INTEGRATED SURVEILLANCE PROGRAM:												
Plate [2]	A0981-1											
Weld [3]	BF2 ESW											

Notes:

[1] The material properties used are those for the bounding adjacent shell plate from the lower-intermediate shell. The fluence considered is applicable at the nozzle location.

[2] The representative plate material is not the same heat number as the target plate; therefore the RG 1.99 CF is used. This information is not applicable to development of the P-T curves and is provided for information only.

[3] The initial RT_{NDT} is obtained from the limiting plant-specific plate and weld.

[4] The representative weld material is not the same heat as the target weld; therefore, these results are not considered in development of the P-T curves.

Table 2.1-3 Effects of Irradiation on Browns Ferry RPV Circumferential Weld Properties

Parameter	B&W 64 EFPY[1]	Unit 1 38 EFPY	Unit 2 48 EFPY	Unit 3 54 EFPY
Cu%	0.31	0.27	0.09	0.09
Ni%	0.59	0.60	0.65	0.66
CF	196.7	184	117	117
Fluence at clad/weld interface (10^{19} n/cm^2)	0.19	0.128	0.156	0.18
ΔRT_{NDT} w/o margin (°F) (Note 2)	109.4	86	60	64
$RT_{NDT(U)}$ (°F)	20	20	-40	-40
Mean RT_{NDT} (°F)	129.4	106	20	24
P(F/E) NRC (Note 3)	4.83 x 10^{-4}	(Note 4)	(Note 4)	(Note 4)
P(F/E) BWRVIP (Note 3)	---	---	---	---

Notes:

[1] Data for the Babcock and Wilcox (B&W) group of plants was obtained from BWRVIP-05 and its SER.

[2] $\Delta RT_{NDT} = CF * f^{(0.28 - 0.10 \log f)}$ as defined in RG 1.99.

[3] P(F/E) means "Probability of a failure event".

[4] Although a conditional failure probability has not been calculated, the fact that the Browns Ferry values at the end of license are less than the 64 EFPY value provided by the NRC leads to the conclusion that the Browns Ferry RPV conditional failure probability is bounded by the NRC analysis, consistent with the requirements defined in GL 98-05.

Table 2.1-4a Browns Ferry Unit 1 Comparison of Key Parameters Influencing FAC Wear Rate

CLTP vs. EPU

CHECWORKS™ Wear Rate Analysis Run Definition Name	Temperature (°F)		Change (°F)	Velocity (ft/sec)		% Change	Oxygen (ppb)		% Change	Quality		% Change	Percent Change in Predicted Wear Rate Due to Power Uprate (See NOTE)
	CLTP 105%	EPU 120%		CLTP 105%	EPU 120%		CLTP 105%	EPU 120%		CLTP 105%	EPU 120%		
FWH 4 - FWH 3	242.3	248.8	6.5	13.841	16.104	16.35	52.7	52.7	0.00	0.0	0.0	N/A	16.27
FWH 3 - RFPs	301.5	309.3	7.8	14.269	16.625	16.51	52.7	52.7	0.00	0.0	0.0	N/A	-10.04
RFPs - FWH 2	301.5	309.3	7.8	16.094	18.751	16.51	52.7	52.7	0.00	0.0	0.0	N/A	-10.03
FWH 2 - FWH 1	334.4	343.5	9.1	13.261	15.47	16.66	52.7	52.7	0.00	0.0	0.0	N/A	7.04
FWH 1 - Rx	381.3	391.6	10.3	16.538	19.328	16.87	52.7	52.7	0.00	0.00	0.00	N/A	-9.67
EXT STM #1	390.7	403.0	12.3	19.751	21.991	11.34	6.0	7.5	24.1	0.892	0.888	-0.45	7.35
MSEP - FCVs	389.2	401.5	12.3	2.069	2.495	20.59	5.9	7.3	24.1	0.0	0.0	N/A	-29.95
FCVs - FWH 2	389.2	401.5	12.3	6.175	7.447	20.60	5.9	7.3	24.1	0.0	0.0	N/A	-30.2
FWH 1 - FWH 2	344.3	356.9	12.6	4.06	4.994	23.00	851.9	1009.1	18.46	0.0	0.0	N/A	16.74
FWH 2 - FWH 3	313.0	323.8	10.8	7.815	9.461	21.06	410.0	481.7	17.49	0.0	0.0	N/A	-10.24
FWH 3 - FWH 4	252.4	262.2	9.8	9.718	11.724	20.64	244.2	285.0	16.69	0.0	0.0	N/A	4.54
FWH 4 - FL TNK	196.5	204.8	8.3	3.871	4.64	19.87	80.2	92.4	15.19	0.0	0.0	N/A	10.21

Table 2.1-4b Browns Ferry Unit 2 Comparison of Key Parameters Influencing FAC Wear Rate

CLTP vs. EPU

CHECWORKS™ Wear Rate Analysis Run Definition Name	Temperature (°F)		Change (°F)	Velocity (ft/sec)		% Change	Oxygen (ppb)		% Change	Quality		% Change	Percent Change in Predicted Wear Rate Due to Power Uprate (See NOTE)
	CLTP 105%	EPU 120%		CLTP 105%	EPU 120%		CLTP 105%	EPU 120%		CLTP 105%	EPU 120%		
CON-FWH 3 to RFPs	301.5	309.3	7.8	14.269	16.625	16.51	60.2	60.2	0.00	0	0	N/A	-10.06
CON-FWH 4 to FWH 3	242.3	248.8	6.5	13.841	16.104	16.35	60.2	60.2	0.00	0	0	N/A	16.27
EX-Extraction #1	390.7	403	12.3	19.75	21.987	11.33	5.8	7.3	24.76	0.892	0.888	-0.45	9.59
EX-Extraction #2	345.7	356.6	10.9	25.169	31.583	25.48	2.7	3.3	22.42	0.967	0.965	-0.21	19.35
HDV-FWH 1 to FWH 2	344.3	356.9	12.6	4.06	4.994	23.00	821.9	973.6	18.46	0	0	N/A	14.95
HDV-FWH 2 to FWH 3	313	323.8	10.8	7.815	9.461	21.06	395.6	464.8	17.49	0	0	N/A	-10.34
HDV-FWH 3 to FWH 4	252.4	262.2	9.8	9.718	11.723	20.63	235.6	275.0	16.69	0	0	N/A	4.46
HDV-FWH 4 to FL TNK	196.6	204.9	8.3	3.871	4.639	19.84	77.4	89.1	15.19	0	0	N/A	11.05
HDV-MSEP to FCVs	389.2	401.5	12.3	2.069	2.496	20.64	5.7	7.1	24.71	0	0	N/A	-29.61
HDV-MSP FCV to FWH 2	389.2	401.5	12.3	6.176	7.447	20.58	5.7	7.1	24.71	0	0	N/A	-29.86
RFW-FWH 1 to Rx	381.9	391.6	10.3	16.938	19.796	16.87	60.2	60.2	0.00	0	0	N/A	-9.66
RFW-FWH 2 to FWH 1	334.4	343.5	9.1	13.261	15.47	16.66	60.2	60.2	0.00	0	0	N/A	7.03
RFW-RFPs to FWH 2	301.5	309.3	7.8	16.094	18.751	16.51	60.2	60.2	0.00	0	0	N/A	-10.05

Table 2.1-4c Browns Ferry Unit 3 Comparison of Key Parameters Influencing FAC Wear Rate

CLTP vs. EPU

CHECWORKS™ Wear Rate Analysis Run Definition Name	Temperature (°F)		Change (°F)	Velocity (ft/sec)		% Change	Oxygen (ppb)		% Change	Quality		% Change	Percent Change in Predicted Wear Rate Due to Power Uprate (See NOTE)
	CLTP 105%	EPU 120%		CLTP 105%	EPU 120%		CLTP 105%	EPU 120%		CLTP 105%	EPU 120%		
FWH 4 - FWH 3	242.3	248.8	6.5	13.841	16.104	16.35	58.5	58.5	0.00	0.0	0.0	N/A	16.26
FWH 3 - RFPs	301.5	309.3	7.8	14.269	16.625	16.51	58.5	58.5	0.00	0.0	0.0	N/A	-10.03
RFPs - FWH 2	301.5	309.3	7.8	16.094	18.751	16.51	58.5	58.5	0.00	0.0	0.0	N/A	-10.04
FWH 2 - FWH 1	334.4	343.5	9.1	13.261	15.47	16.66	58.5	58.5	0.00	0.0	0.0	N/A	7.03
FWH 1 - Rx	381.3	391.6	10.3	16.938	19.796	16.87	58.5	58.5	0.00	0.0	0.0	N/A	-9.68
EXT STM #1	390.7	403	12.3	19.75	21.987	11.33	5.7	7.1	24.73	0.892	0.888	-0.45	9.65
MSEP - FCVs	389.2	401.5	12.3	2.069	2.496	20.64	5.6	7.0	24.73	0.0	0.0	N/A	-29.62
FCVs - FWH 2	389.2	401.5	12.3	6.176	7.447	20.58	5.6	7.0	24.73	0.0	0.0	N/A	-29.87
FWH 1 - FWH 2	344.3	356.9	12.6	4.06	4.994	23.00	809.0	958.3	18.46	0.0	0.0	N/A	14.88
FWH A2 - FWH A3	313	323.8	10.8	7.815	9.461	21.06	389.4	457.5	17.49	0.0	0.0	N/A	-10.38
FWH B&C2 - B&C3	313	323.8	10.8	7.815	9.461	21.06	389.4	457.5	17.49	0.0	0.0	N/A	-10.39
FWH 3 - FWH 4	252.4	262.2	9.8	9.718	11.723	20.63	231.9	270.6	16.69	0.0	0.0	N/A	4.42
FWH 4 - FL TNK	196.6	204.9	8.3	3.871	4.639	19.84	76.2	87.7	15.19	0.0	0.0	N/A	11.41

Note:
The rate of Flow Accelerated Corrosion (FAC) is a complex process that is interdependent on many variables including temperature, velocity, oxygen concentration, and steam quality. Each variable affects the overall wear rate for a component differently. The algorithm in the CHECWORKS™ code has the ability to determine the overall effect on wear rate based on changes in each variable. The rate of FAC is related to the interaction of the parameters; thus, the primary reason for the predicted decrease under EPU conditions is associated with the change in operating temperature. The influence of temperature is represented by a bell curve. Flow accelerated corrosion rates increase as temperature increases up to approximately 300°F and then decrease as the temperature continues to increase beyond 300°F. The slopes of the bell curve are quite steep, which results in a relatively large decrease in wear rate based on a relatively small increase in temperature. The influence of velocity on the rate of flow accelerated corrosion is fairly linear. The slope of the velocity curve is relatively flat indicating that larger changes in velocity will have a lesser effect on rate of FAC degradation verses temperature. Evaluation of the entries in Tables 2-1.4a - c with negative changes in the predicted FAC wear rate indicates that the increase in temperature resulted in a larger overall reduction in the predicted wear rate than the corresponding increase from velocity. This results in a net reduction in the predicted wear rate.

Table 2.1-5a Browns Ferry Unit 1 Components with Highest Predicted Wear Rate for Each Wear Rate Analysis Run Definition CHECWORKS™ SFA-Predicted Thickness vs. Measured Thickness

CHECWORKS™ Wear Rate Analysis Run Definition	Component Name	Component Type	Nominal Pipe Size (inches)	Nominal Thickness (inches)	Measured Thickness Tmeas (inches)	Predicted Thickness Tpred (inches)	Ratio of Measured Thickness to Predicted Thickness	Time of Inspection
FWH 4 - FWH 3	1CON11A-2RT	180° Return	18	0.438	0.452	0.308	1.47	REFUEL 6 - Restart
FWH 3 - RFPs	1CON12A-13E	Elbow	18	0.438	0.421	0.363	1.16	REFUEL 7
RFPs - FWH 2	1RFW1A - BFN	Orifice	18	0.861	0.771	0.602	1.28	REFUEL 10
FWH 2 - FWH 1	1RFW2A2-1N	Reducer	24.5	3.281	3.310	3.125	1.06	REFUEL 6 - Restart
FWH 1 - Rx	1RFW3B - 10FN	Orifice	24	1.219	1.149	0.812	1.41	REFUEL 10
EXT STM #1	1EX11 - 15T	Tee	30	0.375	0.440	0.257	1.71	REFUEL 10
MSEP - FCVs	1HDV9MSB2 - 10N	Exit Nozzle	4	0.237	0.538	0.144	3.75	REFUEL 8
FCVs - FWH 2	1HDV10MSA2 - 29O	Orifice	6	0.280	0.251	0.092	2.73	REFUEL 8
FWH 1 - FWH 2	1HDV1A1 - 3R	Reducer	8	0.322	0.277	0.045	6.19	REFUEL 8
FWH 2 - FWH 3	1HDV2A2 - 45R	Reducer	10	0.365	0.369	0.253	1.46	REFUEL 6 - Restart
FWH 3 - FWH 4	1HDV3A3 - 9R	Reducer	10	0.365	0.331	0.255	1.30	REFUEL 6 - Restart
FWH 4 - FL TNK	1HDV4B4 - 16R	Reducer	18	0.375	0.339	0.229	1.61	REFUEL 9

Table 2.1-5b Browns Ferry Unit 2 Components with Highest Predicted Wear Rate for Each Wear Rate Analysis Run Definition CHECWORKS™ SFA-Predicted Thickness vs. Measured Thickness

CHECWORKS™ Wear Rate Analysis Run Definition	Component Name	Component Type	Nominal Pipe Size (inches)	Nominal Thickness (inches)	Measured Thickness Tmeas (inches)	Predicted Thickness Tpred (inches)	Ratio of Measured Thickness to Predicted Thickness	Time of Inspection
CON-FWH 3 to RFPs	2CON12A -7T	Tee	18	0.438	0.618	0.312	1.98	REFUEL 10
CON-FWH 4 to FWH 3	2CON11A - 2R	180^0 Return	18	0.438	0.496	0.287	1.73	REFUEL 12
EX-Extraction #1	2EX11 - 15T	Tee	30	0.375	0.343	0.205	1.67	REFUEL 12
EX-Extraction #2	2EX32A - 20N	Reducer	12	0.500	0.330	0.477	**0.69**	REFUEL 12
HDV-FWH 1 to FWH 2	2HDV1A1- 3R	Reducer	8	0.322	0.310	0.107	2.91	REFUEL 11
HDV-FWH 2 to FWH 3	2HDV4A2 - 12R	Reducer	10	0.365	0.310	0.207	1.50	REFUEL 14
HDV-FWH 3 to FWH 4	2HDV6A3 - 9R	Reducer	10	0.365	0.334	0.269	1.24	REFUEL 10
HDV-FWH 4 to FL TNK	2HDV8A4 - 16A	Reducer	18	0.375	0.423	0.217	1.95	REFUEL 14
HDV-MSEP to FCVs	2HDV10MA2 - 30T	Tee	6	0.280	0.306	0.086	3.58	REFUEL 15
HDV-MSP FCV to FWH 2	2HDV2MSA1 - 19R	Reducer	6	0.280	0.269	0.086	3.12	REFUEL 10
RFW-FWH 1 to Rx	2RFW6B - 10FN	Orifice	24	1.219	1.154	0.950	1.21	REFUEL 11
RFW-FWH 2 to FWH 1	2RFW4C2 - 7N	Exit Nozzle	20	1.031	0.952	0.857	1.11	REFUEL 11
RFW-RFPs to FWH 2	2RFW1A - 8FN	Orifice	18	0.938	0.778	0.675	1.15	REFUEL 11

Table 2.1-5c Browns Ferry Unit 3 Components with Highest Predicted Wear Rate for Each Wear Rate Analysis Run Definition CHECWORKS™ SFA-Predicted Thickness vs. Measured Thickness

CHECWORKS™ Wear Rate Analysis Run Definition	Component Name	Component Type	Nominal Pipe Size (inches)	Nominal Thickness (inches)	Measured Thickness Tmeas (inches)	Predicted Thickness Tpred (inches)	Ratio of Measured Thickness to Predicted Thickness	Time of Inspection
FWH 4 - FWH 3	3CON11A -2RT	180^0 Return	18	0.438	0.501	0.305	1.64	REFUEL 12
FWH 3 - RFPs	3CON12A -10E	Elbow	18	0.438	0.551	0.348	1.58	REFUEL 14
RFPs - FWH 2	3RFW1C - 4E	Elbow	18	0.938	0.973	0.635	1.53	REFUEL 15
FWH 2 - FWH 1	3RFW2A2 - 1N	Reducer	24.5	3.281	3.340	2.736	1.22	REFUEL 10
FWH 1 - Rx	3RFW3A - 16FN	Orifice	24	1.219	1.136	0.799	1.42	REFUEL 11
EXT STM #1	3EX11 - 14T	Tee	30	0.375	0.396	0.316	1.26	REFUEL 10
MSEP - FCVs	3HDV7MSB1 - 17N	Exit Nozzle	5.75	0.962	0.504	0.828	**0.61**	REFUEL 10
FCVs - FWH 2	3HDV8MSA1 - 34R	Reducer	6	0.280	0.242	0.107	2.26	REFUEL 10
FWH 1 - FWH 2	3HDV1A1 - 3R	Reducer	8	0.322	0.305	0.104	2.92	REFUEL 11
FWH A2 - FWH A3	3HDV2A2 - 44R	Reducer	10	0.365	0.310	0.228	1.36	REFUEL 10
FWH B&C2 - B&C3	3HDV2B2 - 22T	Tee	10	0.365	0.405	0.294	1.38	REFUEL 9
FWH 3 - FWH 4	3HDV3A3 - 9R	Reducer	10	0.365	0.315	0.252	1.25	REFUEL 11
FWH 4 - FL TNK	3HDV4A4 - 16R	Reducer	18	0.375	0.407	0.238	1.71	REFUEL 13

Table 2.1-6 Comparison of RWCU System Operating Conditions

Parameter	Units	CLTP	EPU
Thermal Power	MWt	3,458	3,952
RWCU System Inlet Temperature	°F	530.5	529.3
RWCU System Inlet Pressure (RPV dome pressure, neglecting head)	psia	1,050	1,050
RWCU System Outlet Temperature	°F	433.5	432.2
RWCU System Flow	lbm/hr	133,300	133,300

Table 2.1-7 Comparisons of Chemistry Parameters for CLTP and EPU Cases

Item	Parameter	Units	CLTP Values	EPU Values
1	Maximum average sulfate concentration	ppb	1.29	1.50
2	Maximum average chloride concentration	ppb	0.33	0.38
3	Maximum average reactor water conductivity	μS/cm	0.121	0.132

Table 2.1-8 Selection Process Criteria for Components in the FAC Program

Selection Process Criteria	Description
CHECWORKS™ Model	The Browns Ferry FAC program selects components based on the results of the model's output (i.e., wear rate and remaining life). Components are selected from both lines that have not been inspected and from lines that have inspected components.
Susceptible Non-Modeled (SNM)	The Browns Ferry FAC program selects inspection components based on the susceptibility (highest trended wear rate and shortest remaining service life) of the non-modeled piping. A large amount of FAC susceptible piping cannot be modeled because of a lack of operating parameter data. This includes almost all of the small-bore piping. This also includes FW heater shells. Lines that are deemed highly susceptible and could have detrimental consequences if failure occurred are slated for inspection.
Industry Operating Experience	The Browns Ferry FAC program selects inspection components based on Operating Experiences (OEs) from the industry that are applicable to Browns Ferry. Periodically, OEs are reviewed for Browns Ferry applicability. If the event is applicable, suitable components are selected to address the issue.
Site Operating Experience (Internal)	The Browns Ferry FAC program selects inspection components based on site-specific events. Site Operating Experience also encompasses site specific information obtained from other site groups and other TVA sites. Periodically, the corrective action program is reviewed to discover if any situations had occurred that would be applicable to the program, (i.e., valve leak-bys, steam leaks, abnormal valve usage (open when should be closed)). The thermal performance report is also reviewed periodically to identify any applicable leaking valves whose piping may need to be inspected. Inspection components are also selected based on requests from system engineers or from design changes.
Inspection Trending / Re-Inspections	The Browns Ferry FAC program selects inspection components based on the NSI number. The component's NSI is based on the wear rate and the minimum allowable wall thickness. Components with a remaining life less than the time to the upcoming outage +1 full operating cycle are inspected.
Replacement Transition / Entrance Effects	Carbon steel components downstream of FAC-resistant material have shown to have higher wear rates. A sample of carbon steel components downstream of known resistant materials are considered for inspection.
Engineering Judgment	The Browns Ferry FAC program also selects inspection components based on engineering judgment using the criteria above. Engineering judgment is used when selecting inspection locations through industry and Browns Ferry operating experience.

Table 2.1-9a RCPB Piping and Safe End Materials of Construction

Nozzle Designation / System	Unit 1	Unit 2	Unit 3
N1x: Recirculation Suction	SA-182 F316NG	SA-376 GR 316	SA-376 GR 316
N2x: Recirculation Discharge	SA-182 F316NG	SA-182 F316NG	SA-182 F316NG
N3x: Main Steam	SA-105	SA-105	SA-105
N4x: Reactor Feedwater	SA-105	SA-105	SA-105
N5x: Core Spray	SA-182 F316NG	SA-182 F316NG	SA-182 F316NG
N6x: Rx Vessel Head Spray (Not Used)	A508 CL 2	A508 CL 2	A508 CL 2
N7: Rx Vessel Head Vent	A508 CL 2	A508 CL 2	A508 CL 2
N8x: Jet Pump Instrumentation	SA-182 F316NG	SA-182 F316NG	SA-182 F316NG
N9x: Control Rod Drive Return - Capped (Not Used)	SA336 F8	SA336 F8M	SA336 F8M
N10: SLC and Core Plate Differential	SA336 F8	SA336 F8M	SA336 F8M
N11x: Rx Vessel Level Instrumentation	SA336 F8	SA336 F8M	SA336 F8M
N12x: Rx Vessel Level Instrumentation	SA336 F8	SA336 F8M	SA336 F8M
N15: Rx Vessel Drain	SA-105	SA-105	SA-105

Table 2.1-9a RCPB Piping and Safe End Materials of Construction (continued)

Nozzle Designation / System	Unit 1	Unit 2	Unit 3
N16x: Rx Vessel Level Instrumentation	SA336 F8	SA336 F8M	SA336 F8M
CRD Drive Nozzles	SB-166 A333 Gr 1 A182 F316	SB-166 A333 Gr 1 A182 F316	SB-166 A333 Gr 1 A182 F316
Reactor Recirculation System (Note 1)	SA376 TP316NG SA403 WP 316NG SA182 F316NG SA182 F316L Cast A351 CF8 Cast A351 CF8M	A358 TP304 A376 TP304 SA403 WP316NG A182 F304 Cast A351 CF8 Cast A351 CF8M	A358 TP304 SA403 WP316NG SA376 TP316 SA182 F316 SA182 F316L Cast A351 CF8 Cast A351 CF8M
Main Steam System	A155 GR KC70 CL 1 A516 A106 GR B A234 GR B A105 GR II	A155 GR KC70 CL 1 A516 A106 GR B A234 GR B A420 GR WPLI A350 GR LFI	A155 GR KC70 CL 1 A516 A106 GR B A234 GR B A105 GR II
Reactor Feedwater System	A155 GR KC70 CL 1 A106 GR B A234 GFR WPB A333 GR 6 A420 GR WPL1 A105 GR II A234 GR B	A155 GR KC70 CL 1 A106 GR B A105 GR II A234 GR B	A155 GR KC70 CL 1 A234 GFR WPB A333 GR 6 A420 GR WPL1 A105 GR II A234 GR B

Table 2.1-9a RCPB Piping and Safe End Materials of Construction (continued)

Nozzle Designation / System	Unit 1	Unit 2	Unit 3
Core Spray System	SA333 GR 6 SA350 GR LF2 Cast A351 CF8	SA333 GR 6 A358 GR 304 SA350 GR LF2 SA420 GR WPL6 Cast A351 CF8	SA333 GR 6 A358 GR 304 SA350 GR LF2 SA420 GR WPL6 Cast A351 CF8
Rx Vessel Head Spray (Not Used)	A508 CL 2 SA105 GR II	A508 CL 2 SA105 GR II	A508 CL 2 SA105 GR II
Rx Vessel Head Vent	A/SA 106 Grade B A/SA 106 Grade C A508 CL 2 A/SA 105 SA105 GR II SA216 WCB A/SA 234 GR WPB	A312 TP304 A106 GR B A508 CL 2 SA105 GR II A182 F304 SA403 WP304 A234	A106 GR B A508 CL 2 SA105 GR II
Reactor Vessel Level Instrumentation	SA376 TP304 SA376 TP316 SA312 TP304 SA312 TP316 SA182 F316 SA182 F316L	A312 A376 TP304 A376 TP316 A182 F316	SA375 TP316 SA312 TP316 SA376 TP316 SA182 F316
SLC and Core Plate Differential Pressure	A312 TP304 A312 TP316 A376 TP304 A376 TP316 A 182 GR F304 A 182 GR F316	A312 TP304 A312 TP316 A376 TP304 A376 TP316 A 182 GR F304 A 182 GR F316	A312 TP304 A312 TP316 A376 TP304 A376 TP316 A 182 GR F304 A 182 GR F316

Table 2.1-9a RCPB Piping and Safe End Materials of Construction (continued)

Nozzle Designation / System	Unit 1	Unit 2	Unit 3
Jet Pump Instrumentation	SA376 TP316 SA312 TP316 SA182 F316 SA182 F316L	A376 TP304 A376 TP316 A 182 GR F304 A 182 GR F316	A376 TP304 A376 TP316 A 182 GR F304 A 182 GR F316
Reactor Vessel Drain	A/SA 376 TP304 SA-105 A/SA 182 F304 A/SA 182 F316 SA351 CF8M	A 376 TP304 SA105 A182 F304 A182 F316 SA351 CF8M	SA376 TP316 SA182 F316L SA182 F304 SA351 CF8M
CRD Return Line (Capped/Not Used)	SA 182 F316L	SA 182 F316L	SA 182 F316L

Note:

1. The Unit 1 Reactor Recirculation System (RRS) piping (pump suction and discharge piping, the ring header, the riser piping, and the inlet and outlet safe ends) has been replaced with ASME SA376 Type 316 NG stainless material, which is resistant to IGSCC. The recirculation suction and inlet safe ends are an improved crevice-free design. The replacement piping utilized an improved design which eliminated several piping welds. Additionally, the use of EPRI welding techniques (such as machine welding where practical and reduced energy input) and the application of a Mechanical Stress Improvement Process (MSIP) were utilized to reduce the potential for IGSCC. As a result of these efforts, all the Unit 1 RRS welds are Category A welds in accordance with NUREG-0313, Revision 2 classifications.

   The Unit 2 RRS pump suction and discharge piping and the bottom portion of the ten system risers are fabricated with ASTM A358 Type 304 stainless steel. The top portion of each riser including the riser elbow is ASME SA403 WP 316 NG. The recirculation inlet safe ends are ASME SA376 Type 316 NG. The recirculation inlet safe ends are an improved crevice-free design. To mitigate weld residual stresses in the Unit 2 RRS, the application of a MSIP or Induction Heat Stress Improvement (IHSI) were utilized on the accessible welds to reduce the potential for IGSCC.

   The Unit 3 RRS consists of suction and discharge piping fabricated with ASTM A358 Type 304 stainless steel. All piping downstream of the 28 inch recirculation discharge piping, including the cross, ring header, and the risers are ASME SA403 WP 316 NG stainless steel. The recirculation inlet safe ends are ASME SA376 Type 316 NG. The

recirculation inlet safe ends are an improved crevice-free design. The replacement piping utilized an improved design which eliminated several piping welds. To mitigate weld residual stresses in the Unit 3 RRS, the application of a MSIP or IHSI were utilized on the accessible welds to reduce the potential for IGSCC.

Table 2.1-9b Summary of RCPB Welds per Generic Letter 88-01/BWRVIP-75-A

IGSCC Weld Category	Unit 1	Unit 2	Unit 3
A	136	48	70
B	0	0	0
C	10	112	79
D	5	9	2
E	0	16	10
F	0	0	0
G	2	2	2

2.2 Mechanical and Civil Engineering

2.2.1 Pipe Rupture Locations and Associated Dynamic Effects

Regulatory Evaluation

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

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

Specific NRC review criteria are contained in SRP Section 3.6.2.

Browns Ferry Current Licensing Basis

The General Design Criteria (GDC) listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Browns Ferry Nuclear Plant (Browns Ferry) principal design criteria predate these criteria. The Browns Ferry principal design criteria are listed in UFSAR Section 1.5, "Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, the Tennessee Valley Authority (TVA) performed a comparative evaluation of the design basis of Browns Ferry with the AEC proposed General Design Criteria of 1967. The Browns Ferry UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria set out in the July 1967 AEC release. For each group of criteria, there is a statement of TVA's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a table of references to locations in the Browns Ferry UFSAR where there is subject matter relating to the intent of that particular criteria.

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

Browns Ferry Pipe Rupture Locations and Associated Dynamic Effects are described in Browns Ferry UFSAR Section 12.2, "Principal Structures and Foundations," and Appendix M, "Report on Pipe Failures Outside Containment in the Browns Ferry Nuclear Plant."

Technical Evaluation

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

Inside containment, the high-energy piping systems potentially affected by EPU are:

- Main steam
- Main steam drains
- Reactor Core Isolation Cooling (RCIC) steam line
- High Pressure Coolant Injection (HPCI) steam line
- FW
- Main steam safety relief valve piping (between the MSL and each SRV)
- Reactor pressure vessel head vent line

Main steam drains, RCIC steam, and HPCI steam line flow rates, pressures, and temperatures are unchanged from CLTP to EPU operating conditions. Therefore, the MS attached piping did not have additional break locations resulting from EPU operating conditions.

Outside containment, high-energy piping systems include:

- Main steam
- Feedwater
- HPCI (normally pressurized steam supply to turbine drive)
- RCIC (normally pressurized steam supply to turbine drive)
- RWCU

Of these, the only systems affected by EPU are main steam and feedwater. While main steam pressures and temperatures do not increase with EPU, the piping stress analysis of record was performed for EPU conditions to account for the increase in flows.

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

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

For EPU, high energy line breaks (HELBs) are evaluated for their effects on equipment qualification. Browns Ferry meets all CLTR dispositions. The topics addressed in this evaluation are:

Topic	CLTR Disposition	Browns Ferry Result
Steam Lines	Generic	Meets CLTR Disposition
Liquid Lines	Plant Specific	Meets CLTR Disposition

2.2.1.1 Steam Line Breaks

The CLTR states that there is no effect on steam line breaks because steam conditions at postulated break locations are unchanged. Therefore, EPU has no effect on the mass and energy releases from a HELB in a steam line.

A review of the heat balances produced for the Browns Ferry EPU confirmed that [[ ]]

The evaluation of the steam line HELB events in the Browns Ferry licensing basis meet all CLTR dispositions except as discussed below for the plant-specific MS Line intermediate break.

At Browns Ferry, the intermediate size steam line break is defined in the current licensing basis as the largest main steam line break (MSLB) that is not isolated by the MS line high flow sensors, which are set to limit flow below the analytical limit of 144% of rated MS flow. Because rated MS flow increases for EPU, the mass flow rate for the intermediate steam line break also increases. The CLTP design mass and energy releases for the intermediate size MSLB were evaluated and revised to account for the increase in the percent rated steam flow at EPU. This evaluation included the effects identified within the 10 CFR Part 21 Potentially Reportable Condition Notification: Error in Main Steam Line High Flow Calculational Methodology (Reference 42).

Therefore, the intermediate size MSLB remains bounded by the doubled-ended break in the main steam valve vault. The mass and energy releases for the doubled-ended break in the main steam valve vault at EPU are unchanged from the CLTP analyses.

2.2.1.2 Liquid Line Breaks

As stated in Section 10.1 of the CLTR, EPU may increase subcooling in the reactor vessel, which may lead to increased break flow rates for liquid line breaks. For Browns Ferry, the increase in vessel subcooling could affect the RWCU line break analysis. In addition, operation at EPU conditions requires an increase in the FW flow, which results in an increase in FW system pressure and a small increase in RWCU discharge pressure at the FW inlet. This increase in FW system pressure may lead to increased break flow rates for FW line breaks. For Browns Ferry, mass and energy releases for HELBs were re-evaluated at EPU conditions.

The plant-specific evaluation of liquid line breaks included the RWCU and FW systems as well as the effect of increased RWCU and FW operating pressure on pipe whip and jet impingement.

The results of the Browns Ferry evaluation of liquid line breaks outside containment are provided in Table 2.2-1.

2.2.1.2.1 RWCU Line Breaks

Operation at EPU involves an increase in the steam and feedwater flows, which results in a small increase in downcomer subcooling. This condition results in a small increase in the CLTP RWCU System mass flow rates. New mass and energy releases for RWCU line breaks have been analyzed for EPU conditions which include the effects of subcooling and the small increase in RWCU discharge pressure.

Structural effects of increased peak pressures were reviewed and found to be acceptable. The effects of increased peak calculated room temperatures in the Reactor Building resulting from the RWCU line breaks are addressed in the EPU EQ analysis. See Section 2.3.1 for EQ results.

2.2.1.2.2 Feedwater System Line Break

The FW System process conditions are changed for EPU to support the increased feedwater flows. The base RELAP5 feedwater system break models for CLTP were revised to account for EPU changes in the FW System process conditions including the changes to system temperatures and to the increased discharge head for the FW pumps. The EPU evaluation concludes that the associated changes in FW system line break mass and energy release will not challenge the bases for the current HELB analysis because the effects of the FW line break in the main steam valve vault are bounded by the effects of the postulated MSLB. Also, for the portion of the smaller RWCU piping attached to the FW piping in the main steam valve vault, mass and energy releases from breaks in the smaller RWCU piping are bounded by the FW line break mass and energy releases.

2.2.1.2.3 Pipe Whip and Jet Impingement

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

Inside containment, the only high-energy piping that experiences an increase in operating pressure due to EPU is in the FW and RRS systems. Outside containment, the only high-energy piping experiencing an increase in operating pressure due to EPU is the FW and RWCU system.

Increased FW fluid conditions associated with EPU will not affect the current HELB analysis in the main steam valve vault. The increase in FW fluid conditions are bounded by the MSLB, which remains unchanged from CLTP conditions. Pipe whip and jet impingement loads resulting from high energy pipe breaks are dependent on system pressure. The feedwater system operating pressure for the design of pipe whip and jet impingement is based on the feedwater system design pressure of 1,250 psig, which is higher than the EPU pressure of 1,200 psig. Therefore, EPU will have a negligible effect on FW pipe whip and jet impingement and will still be bounded by current design analyses.

The RRS will have a slight increase in operating pressure for EPU conditions. This slight increase in operating pressure remains bounded by the existing margin in the current analysis. Therefore, EPU will not affect the RRS pipe whip and jet impingement loads.

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

The adequacies of pipe stress and pipe support loads relative to pipe whip and jet impingement loads are evaluated in Section 2.2.2.

Therefore, Browns Ferry meets all CLTR dispositions for liquid line breaks.

<u>Conclusion</u>

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

2.2.2 Pressure-Retaining Components and Component Supports

<u>Regulatory Evaluation</u>

The NRC's acceptance criteria are based on (1) 10 CFR 50.55a and GDC-1, insofar as they require that SSCs important to safety be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety functions to be performed; (2) GDC-2, insofar as it requires that SSCs important to safety be designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions; (3) GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents; (4) GDC-14, insofar as it requires that the Reactor Coolant Pressure Boundary (RCPB) be designed, fabricated, erected, and tested so as to have an extremely low probability of rapidly propagating fracture; and (5) GDC-15, insofar as it requires that the RCS be designed with margin sufficient to ensure that the design conditions of the RCPB are not exceeded during any condition of normal operation.

Specific NRC review criteria are contained in SRP Sections 3.9.1, 3.9.2, 3.9.3, and 5.2.1.1 and other guidance provided in Matrix 2 of RS-001.

Browns Ferry Current Licensing Basis

The General Design Criteria (GDC) listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Browns Ferry Nuclear Plant (Browns Ferry) principal design criteria predate these criteria. The Browns Ferry principal design criteria are listed in UFSAR Section 1.5, "Principal Design Criteria." In 1967, the AEC published for public comment a

revised set of proposed General Design Criteria (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, the Tennessee Valley Authority (TVA) performed a comparative evaluation of the design basis of Browns Ferry with the AEC proposed General Design Criteria of 1967. The Browns Ferry UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria set out in the July 1967 AEC release. For each group of criteria, there is a statement of TVA's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a table of references to locations in the Browns Ferry UFSAR where there is subject matter relating to the intent of that particular criteria.

While Browns Ferry is not generally licensed to the final GDC or the 1967 AEC proposed General Design Criteria, a comparison of the final GDC to the applicable AEC proposed General Design Criteria can usually be made. For the final GDC listed in the Regulatory Evaluation above, the Browns Ferry comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Browns Ferry UFSAR Appendix A: draft GDCs-1, 2, 9, 33, 34, 40, and 42.

The Pressure Retaining Components and Component Supports are described in Browns Ferry UFSAR Section 3.3, "Reactor Vessel Internals Mechanical Design," and Chapter 12, "Structures and Shielding."

In addition to the evaluations described in the Browns Ferry UFSAR, Browns Ferry's systems and components were evaluated for license renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Browns Ferry License Renewal Safety Evaluation Report (SER), NUREG-1843, dated April 2006 (Reference 11). Systems and system components, programs used to manage aging effects, and time limited aging analyses are documented in NUREG-1843, Sections 2.3, 3.1, 3.5, and 4.3.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of EPUs. Section 3.2.2, 3.4 and 3.5 of the CLTR addresses the effect of EPU on Reactor Vessel Structural Evaluation, Flow-Induced Vibration and Piping Evaluation, respectively. The results of this evaluation are described below.

2.2.2.1 Flow-Induced Vibration (FIV)

The FIV evaluation addresses the influence of an increase in flow during EPU on RCPB piping and RCPB piping components.

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

Browns Ferry meets all CLTR dispositions. The topics addressed in this evaluation are:

Topic	CLTR Disposition	Browns Ferry Result
Structural Evaluation of Recirculation Piping	Generic	Meets CLTR Disposition
Structural Evaluation of Main Steam and FW Piping	Generic	Meets CLTR Disposition
Safety-Related Thermowells and Probes	Plant Specific	Meets CLTR Disposition
Structural Evaluation of Core Flow Dependent RPV Internals	Plant Specific	Meets CLTR Disposition

2.2.2.1.1 Structural Evaluation of Recirculation Piping

The CLTR states that there is no significant increase in the recirculation flow rate at EPU conditions.

The recirculation system drive flow remains unchanged at 17.7 Mlb/hr per loop at EPU conditions, resulting in no increase from CLTP to EPU operation. Consequently, the FIV levels of the RRS components are expected to remain essentially the same. Because RRS flow rates for EPU are essentially the same as previously experienced, no further evaluation or testing of the FIV levels of the RRS piping, branch piping (e.g., attached residual heat removal piping), or its suspension system is required.

The FIV effect on RRS piping inside containment at Browns Ferry meets all CLTR dispositions because the nominal reactor dome pressure remains the same and the RRS maximum drive flow does not increase.

2.2.2.1.2 Structural Evaluation of Main Steam and Feedwater Piping

The CLTR states that MS and FW flow rates increase due to the power uprate.

As a result of the increased flow rates and flow velocities, the MS and FW piping experience increased vibration levels, approximately proportional to the square of the flow velocities. Thus, for Browns Ferry, vibration levels may increase by up to 58.5% of OLTP. The ASME Code (NB-3622.3) and nuclear regulatory guidelines require some vibration test data be taken and evaluated for these high energy piping systems during initial operation at EPU conditions. Vibration data for the MS and FW piping inside containment will be acquired using remote

sensors, such as accelerometers and displacement probes as appropriate. A piping vibration startup test program, which meets the ASME code and regulatory requirements, will be performed.

Therefore, the assessment of the structural evaluation of MS and FW piping meets all CLTR dispositions. The EPU increase in MS and FW flow rates only affects the structural analysis of the MS and FW piping. The structural analyses of all other systems inside and outside of containment are not affected by the increase in MS and FW flow rates. FIV testing of the MS and FW piping system will be performed during EPU power ascension. Additional information related to the MS piping is provided in Section 2.5.4.1.1. The piping vibration monitoring program for Browns Ferry EPU is described in LAR Attachment 45. LAR Attachment 45 also provides projections of piping vibration levels at the EPU power level for piping systems both inside and outside containment.

2.2.2.1.3 Safety-Related Thermowells and Probes

As explicitly stated in Section 3.4 of the CLTR, MS and FW flow rates increase due to the power uprate. The CLTR requires a plant-specific evaluation of safety-related thermowells and probes in the MS and FW piping systems at EPU conditions.

Browns Ferry Units 1, 2 and 3 have no safety-related thermowells in the MS lines, but do have safety-related thermowells installed on the MSRV discharge pipes in MS system. There are no safety-related sample probes installed in the MS and FW systems at Browns Ferry.

The MS system flow increased from 3.54 Mlb/hr per line at CLTP to 4.11 Mlb/hr per line resulting in an increase of 16.2% during EPU operation. The FW system flow increased from 7.05 Mlb/hr per line at CLTP to 8.20 Mlb/hr per line resulting in an increase of 16.2% during EPU operation. The RRS thermowell evaluation was performed with a bounding ICF RRS flow rate of 18.81 Mlb/hr under EPU condition. The safety-related thermowells and probes in the MS, FW and RRS piping systems were evaluated and found to be adequate for EPU.

The methodology used to evaluate the FIV effects on piping components under EPU is described in Section 3.4.1 of the CLTR. This evaluation utilizes computer program SAP4G07 to develop dynamic finite element models of the MS/FW/RRS thermowells and sample probes.

Three-dimensional beam elements with six degrees of freedom are used to model the thermowell and sample probe and their sockolet/pipe weld. At the sockolet/pipe weld to the outer pipe wall, all six degrees of freedom are fixed. The masses of the thermowells or sample probes and their sockolet/pipe weld are lumped at the nodal points, which include both the structural mass and fluid mass displaced by the thermowells and sample probes. These added masses are used to account for the effects of fluid on the thermowells and sample probes vibration responses.

This evaluation of the piping components follows the FIV analysis guideline, as outlined in ASME code N-1300 (Reference 43). The resonance separation rule: $f_n/f_s<0.7$ or $f_n/f_s >1.3$ as established in Reference 43 (N-1324.1(d)) is used to determine if there exists an adequate

separation between the vortex shedding frequencies and the natural frequencies of the piping components.

When one of the natural frequencies is close to the vortex shedding frequency, the equation from Table N-1324.2(a)-1 of Reference 43 is used to calculate structural response of the piping components under lock-in condition. For off lock-in (no resonance) condition, the structural response was calculated using the standard methods (Reference 43, N-1324.2).

The safety-related thermowells and sample probes in the recirculation piping system were evaluated to be adequate for the RRS flow associated with ICF at EPU conditions.

The evaluation in accordance with ASME code, Section III, Division 1, Appendices, N-1300 (Reference 43) concludes that the safety-related thermowells and sample probes in MS, FW, and RRS systems at Browns Ferry Units 1, 2, and 3, remain structurally adequate to withstand the FIV effects under EPU conditions.

The maximum vibratory stress is calculated by using the square root of the sum of the squares (SRSS) of the responses in lift and drag directions with Stress Concentration Factor (SCF) of 2.0. The results of the analyses are presented below:

Item	Component	Maximum Stress under EPU (psi)	Allowable (psi)	Criteria f_n/f_s <0.7 OR f_n/f_s >1.3
1	MSRV Discharge Line Thermowell	6,075(1)	13,600	f_n/f_s<0.7
2	FW Thermowell	1,329	13,600	f_n/f_s>1.3
3	RRS Thermowell	3,020	13,600	f_n/f_s>1.3
4	RRS Sample Probe	142	13,600	f_n/f_s>>1.3

Note:
(1) Because the structural fundamental frequency of MS thermowell is lower than the vortex shedding frequency, the structural response of the thermowell at the lock-in condition is conservatively assumed and calculated by the equation from table N-1324.2(a)-1, Reference 43. This FIV stress bounds the current MS thermowell under EPU, which is expected to be less than 6,075 psi at the off lock-in condition.

Therefore, Browns Ferry meets all CLTR dispositions for safety-related thermowells and probes.

2.2.2.2 Piping Evaluation

2.2.2.2.1 Reactor Coolant Pressure Boundary Piping (Non-FIV) Evaluation

The RCPB system evaluation consists of a number of safety-related piping subsystems that move fluid through the reactor and other safety systems. The code of record for Browns Ferry safety-related piping, with the exception of the primary containment torus attached piping, is USA

Standard (USAS) B31.1.0 - 1967 (Reference 44). Because Reference 44 is incomplete with respect to plant operating conditions and code equations, the later ASME Section III code has been used in the development of load combinations and allowable stress criteria. Section III of the 1971 ASME Boiler and Pressure Code, including the Summer 1973 addenda, Subsection NC is used as guidance. However, analysis parameters, such as material allowable stresses, stress intensification factor (SIF) coefficient of thermal expansion, and elastic modulus are in accordance with USAS B31.1.0 - 1967. The Browns Ferry EPU piping evaluations are performed to these same codes of record without exception.

Browns Ferry meets all CLTR dispositions. The topics addressed in this evaluation are:

Topic	CLTR Disposition	Browns Ferry Result
Structural Evaluation for Unaffected Safety-Related Piping	Generic	Meets CLTR Disposition
Structural Evaluation for Affected Safety-Related Piping	Plant Specific	Meets CLTR Disposition

2.2.2.2.1.1 Structural Evaluation for Unaffected Safety-Related Piping

As stated in Section 3.5.1 of the CLTR, the flow, pressure, temperature, and mechanical loading for most of the RCPB piping systems do not increase for EPU. Consequently, there is no change in stress and fatigue evaluations.

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]] and therefore, Browns Ferry meets all CLTR dispositions for the structural evaluation for unaffected safety-related piping. Table 2.2-2 provides the justification for confirming the generic disposition for the above piping systems and segments.

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

Pipe Whip and Jet Impingement

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

Additionally, pipe stress calculations were revised to reflect EPU operating conditions for MS and FW. There are no increased pipe stress levels above the thresholds required for postulating HELBs, except at locations already evaluated for breaks. As a result, EPU conditions do not result in new HELB locations, nor affect existing HELB evaluations of pipe whip restraints and jet targets.

2.2.2.2.1.2 Structural Evaluation for Affected Safety-Related Piping

As stated in Section 3.5.1 of the CLTR, the FW and MS piping and associated branch piping up to the first anchor or support will experience an increase in the flow, pressure, and/or temperature, resulting in an increase in operating stress and fatigue. For all systems, the maximum stress levels were reviewed based on specific increases in temperature, pressure, and flow rate. EPU also increases the operating pipe support loads due to the above effects as well as increased fluid transient turbine stop valve closure (TSVC) loads that result from the increased steam flow rates.

The analyses of record for MS piping were revised to evaluate TSVC loads for all pipe nodes. These evaluations determined that the interface loads on snubbers, struts, guides, and flange connections at EPU conditions are within the design limits (capacities) of these components, and any required modifications have been completed. The analyses of record for FW piping inside containment were revised to account for the pressures and temperatures of EPU operating conditions. Therefore, design loads and stresses remain bounding for EPU. These evaluations determined that the interface loads on snubbers, struts, guides, and flange connections at EPU conditions are within the design limits (capacities) of these components and any required modifications have been completed.

For RCPB MS piping outside containment between the containment penetration and the outboard main steam isolation valve (MSIV), the TSVC fluid transient was evaluated in the revised piping stress analyses of record for EPU conditions. The MS analysis resulted in MS piping outside containment meeting all Code criteria. There are no required pipe support modifications for MS piping outside containment due to EPU. There are no pipe supports on the RCPB MS piping outside containment, between the containment penetration anchor and the outboard MSIV.

The FW system has been evaluated and found to meet the appropriate code criteria for EPU conditions, based on the design margins between calculated stresses and code limits in the current design. All piping stresses are below the code allowable values of the Browns Ferry analysis of record. The MS piping was analyzed for TSVC loads. Stresses in the MS and attached piping are below the code allowable values.

The pipe supports of the systems affected by EPU loading increases are reviewed to determine if there is sufficient margin to code acceptance criteria to accommodate the increased loadings. RCPB FW piping inside containment support loads are acceptable for EPU. The MS analysis resulted in MS piping inside containment meeting all Code criteria. EPU pipe stresses and support loads for RCPB FW piping outside containment are acceptable and meet all Code criteria. The MS analysis resulted in MS piping outside containment meeting all Code criteria.

Main Steam and Associated Piping System Evaluation

For Browns Ferry, an increase in flow and mechanical loads was evaluated on a plant-specific basis consistent with the methods specified in Appendix K of ELTR1. Plant-specific evaluations are required to demonstrate that the calculated stresses are less than the code allowable limits in accordance with the requirements of the applicable code of record in the existing design basis stress report.

The MS and associated branch piping inside containment and RCPB piping outside containment was evaluated to the USAS B31.1.0 - 1967 stress criteria (Reference 44), including the effects of EPU on piping stresses, piping support loads including the associated building structure, penetrations, piping interfaces with the RPV nozzles, flanges, and valves. Allowable stress values for MS piping inside containment and associated branch lines were taken from USAS B31.1.0 - 1967 (Reference 44). SRP Section 3.6.2, MEB 3-1 criteria is not a licensing commitment for Browns Ferry, but all pipe ruptures are postulated in accordance with current licensing basis.

Because the MS piping pressures and temperatures are not significantly affected by EPU, there is no effect on the analyses for these parameters. Seismic inertia loads, seismic building displacement loads, and MSRV discharge loads are not affected by EPU; thus, there is no effect on the analyses for these load cases. The increase in MS flow results in increased fluid transient loads from a TSVC transient. The TSVC loads bound the MSIV closure loads because the MSIV closure time is significantly longer than the TSV closure time. The analyses of record were revised for MS to include the TSVC transient.

The TSVC transient loading will increase due to the increase in the MS flow rate under EPU. Detailed and conservative modeling of this transient was performed to ensure that components, pipe stress, and support loads do not exceed their allowable code limits.

Pipe Stresses

A review of the increase in flow associated with EPU indicates that piping stress changes do not result in stress limits being exceeded for the MS system and attached branch piping or for RPV nozzles and containment penetrations. The revised design analyses have sufficient margin between calculated stresses and USAS B31.1.0 allowable limits (see Table 2.2-3a) to justify operation at EPU conditions. The pressure and temperature of the MS piping are unchanged for EPU.

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

The pipe stress analyses of record for RCPB MS piping outside containment were revised to evaluate the increased TSVC fluid transient loading with EPU. The revised analysis for the MS system outside containment demonstrates that the design has sufficient margin between calculated stresses and the allowable limits in the code of record, USAS B31.1.0 – 1967 (Reference 44) to justify operation at EPU conditions.

Pipe Supports

A review of the change in flow associated with EPU indicates that piping load changes do not result in load limits being exceeded for the MS piping system; therefore, the pipe supports for the MS piping system are adequate at EPU conditions. The current design analyses were updated for conditions representative of EPU operation in the MS piping system as applicable. No inside containment pipe support modifications are required for EPU.

Main Steam Isolation Valves

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

The MSIVs have been evaluated, as discussed in Section 4.7 of ELTR2, Supplement 1. The evaluation covers both the effects of the changes to the structural capability of the MSIV to meet pressure boundary requirements, and the potential effects of EPU-related changes to the safety functions of the MSIVs. The generic evaluation from ELTR2 is based on: (1) a 20% thermal power increase; (2) an increased operating dome pressure to 1,095 psia; (3) a reactor temperature increase to 556°F; and (4) steam and FW flow increases of about 24%. Table 1-2 provides the maximum nominal dome pressure and temperature as well as the changes in steam and FW flows. From these parameters, it can be determined that the evaluation from ELTR2 is applicable to Browns Ferry.

The MSIV has design features that ensure that MSIV closure time is maintained within the stroke time limits. The closing time of the MSIVs is controlled by the design of the hydraulic control valves and the function of the hydraulic damper. Therefore, the MSIV performance is

bounded by conclusions of the evaluation in Section 4.7 of ELTR2, and the Browns Ferry MSIVs are acceptable for EPU operation.

Feedwater System Evaluation

The pressure changes are insignificant for EPU and are bounded by those used in the analysis of record. The calculations of record were revised to reflect EPU operating temperatures (Table 2.2-5). The current licensing basis for the reactor FW system inside containment complies with the Browns Ferry code of record stress criteria (Reference 44) for the effect of thermal expansion displacement on the piping snubbers, hangers, and struts. Piping interfaces with RPV nozzles, penetrations, flanges, and valves also remain valid per current licensing basis. Note that the FW flow change of approximately 16.2% does not affect the reactor FW piping system loads and stresses because fluid transient and fatigue loads are not a part of the design basis in the original stress calculation. SRP Section 3.6.2, MEB 3-1 criteria is not a licensing commitment for Browns Ferry, but all pipe ruptures are postulated in accordance with the current licensing basis.

This discussion also applies to the FW system and associated branch piping outside containment. The FW piping design at EPU conditions was evaluated for compliance with the analysis code of record stress criteria (Reference 44).

Because the FW system piping operating temperatures increase slightly due to EPU, the effect of these parameters on the existing analyses was evaluated. Seismic inertia loads, and seismic building displacement loads are not affected by EPU; thus, there is no effect on the analyses for these load cases. Other external loading conditions are not changed by EPU. For the FW piping inside and outside containment, there is no FW system fluid transient analysis in the original or current design basis analysis so the increase in FW system flow has no effect on the current analysis.

Although fluid transient analyses are not required as part of the piping design basis, an analysis was performed that assessed the effect of transients on the FW piping at EPU conditions. The bounding transient considered was for a simultaneous trip of all three FW pump turbines that results in transient loading on piping as the FW pumps coast down. The results of this analysis showed that the FW piping is acceptable for FW fluid transients that occur at EPU conditions and that the FW piping design has sufficient margin between calculated stresses/loads and the allowable limits in the code of record (Table 2.2-3d).

Pipe Stresses

For FW piping inside containment, a review of the changes in operating pressure, temperature and flow associated with EPU indicates that piping stress changes do not result in stress limits being exceeded for the reactor FW piping system, for RPV nozzles, and at postulated pipe break locations (see Table 2.2-3b). The current Browns Ferry design analyses were revised for conditions representative of EPU operating modes in the FW piping system.

This discussion also pertains to the RCPB portion of the FW piping outside containment. A review of the increase in flow, operating pressure, and temperature associated with EPU indicates that piping load changes do not result in load limits being exceeded for the FW piping system and attached branch piping.

A review was also performed of postulated pipe break locations in accordance with the current licensing basis methodology. As a result of this review, no new postulated break locations were identified. The analysis for the FW system outside containment demonstrates that the design has sufficient margin between calculated stresses and the allowable limits in the code of record (Reference 44).

Pipe Supports

A review of the changes in operating pressure, temperature and flow associated with EPU indicates that piping load changes and thermal expansion displacements do not result in load limits being exceeded for the FW piping system; therefore, the pipe supports for the FW piping system are adequate at EPU conditions. The current design analyses were revised for conditions representative of EPU operating modes in the FW piping system.

Seismic inertia loads and seismic building displacement loads are not affected by EPU; thus, there is no effect on the analyses for these load cases.

The FW system piping outside containment was evaluated for the effects of EPU temperature increase on the piping design analyses. It was concluded that EPU does not have an adverse effect on FW pipe support design, and all loads were within limits.

Other Piping Evaluation

As previously noted, the nominal operating pressure and temperature of the reactor are not changed by EPU. Aside from MS and FW, no other system connected to the RCPB experiences a material increase in flow rate at EPU conditions. Only minor changes to fluid conditions are experienced by these systems due to higher steam flow from the reactor and the subsequent change in fluid conditions within the reactor. Additionally, piping dynamic loads due to MSRV discharge at EPU conditions are bounded by those used in the existing analyses.

These systems were previously evaluated for compliance with the code of record (Reference 44) stress criteria as required. Because none of these piping systems connected to the RCPB experience any significant change in operating conditions due to EPU, they are all acceptable as currently designed.

Therefore, Browns Ferry meets all CLTR dispositions for RCPB piping.

2.2.2.2.2 Balance-of-Plant (BOP) Piping Systems

The BOP piping systems evaluation consists of a number of piping subsystems that move fluid through systems outside the RCPB piping.

Browns Ferry meets all CLTR dispositions. The topics considered in this section are:

Topic	CLTR Disposition	Browns Ferry Result
Structural Evaluation for Unaffected Safety-Related Piping	Generic	Meets CLTR Disposition
Structural Evaluation for Unaffected Non-Safety Related Piping	Generic	Meets CLTR Disposition
Structural Evaluation for Affected Safety-Related Piping	Plant Specific	Meets CLTR Disposition
Structural Evaluation for Affected Non-Safety Related Piping	Plant Specific	Meets CLTR Disposition

2.2.2.2.2.1 Structural Evaluation for Unaffected BOP Piping

As stated in Section 3.5.2 of the CLTR, the flow, pressure, temperature, and mechanical loading for some BOP piping systems do not increase for EPU. Consequently, there is no change in stress and fatigue evaluations and these BOP piping systems meet all CLTR dispositions.

The following piping for BOP and NSSS outside containment only were confirmed to be unaffected by EPU conditions because either the flow, temperature, pressure, or other mechanical loads do not change in the system for EPU or the change is insignificant and has no effect on the piping system design:

- Auxiliary Steam Piping
- Circulating Water Piping
- Condensate Storage and Supply Piping
- Condenser Air Removal Piping
- CRD Piping
- Drywell (DW) Chilled Water Piping
- Fuel Pool Cooling (FPC) and Cleanup Piping
- Liquid Radwaste Piping
- Off Gas Piping
- Plant Chilled Water Piping
- Raw Service Water (RSW) Piping
- Raw Cooling Water (RCW) Piping
- Post-Accident Sampling Piping

- Process Sampling Piping
- RWCU Piping
- Standby Liquid Control (SLC) Piping (Outside Containment)
- Emergency Equipment Cooling Water (EECW) Piping
- Reactor Building Cooling Water (RBCCW) Piping
- Main Steam Relief Valve Drain Line (MSRVDL) Piping Beyond the First Anchor to the Quenchers
- RCIC Piping Outside Containment
- HPCI Piping Outside Containment

2.2.2.2.2.2 Structural Evaluation for Affected BOP Piping

As stated in Section 3.5.2 of the CLTR, the FW and MSL piping including the associated branch piping will experience an increase in the flow and/or temperature resulting in an increase in stress.

The Browns Ferry piping systems determined to be affected by EPU operation include:

- RHR Piping
- CS Piping Outside Containment
- MS Piping (Outside Containment)
- Extraction Steam Piping
- FW Piping (Outside Containment)
- Condensate Piping
- Moisture Separator Drains Piping
- FW Heater Vents and Drains Piping
- Cross Around Relief Valve (CARV) Discharge Piping
- Condensate Demineralizer Piping

For those systems with analyses, the maximum stress level analysis results were reviewed based on specific increases in temperature, pressure and flow rate (see Tables 2.2-4a through 2.2-4f and Table 2.2-5).

The code of record for Browns Ferry safety-related piping, with the exception of the primary containment torus attached piping, is USAS B31.1.0 - 1967 (Reference 44). Because Reference 44 is incomplete with respect to plant operating conditions and code equations, the later ASME Section III code has been used in the development of load combinations and allowable stress criteria. Section III of the 1971 ASME Boiler and Pressure Code, including the Summer 1973 addenda, Subsection NC is used as guidance. However, analysis parameters, such as material allowable stresses, SIF coefficient of thermal expansion, and elastic modulus are in

accordance with USAS B31.1.0 - 1967. The Browns Ferry code of record for primary containment torus attached piping is ASME Boiler and Pressure Code Section III, Division 1, Subsection NC, through the Summer 1977 addenda. The piping systems affected by EPU have been evaluated in accordance with these codes of record criteria for the EPU conditions based on the design margins between actual stresses and code limits in the original design. All piping stresses have been found to be below the code allowable limits of the present code of record.

A review was performed of postulated high energy pipe break locations in accordance with the requirements of the current licensing basis methodology. As a result of this review, no new postulated break locations were identified. Details regarding analyses pertaining to the dynamic effects of high-energy piping failures outside containment are provided in Section 2.2.1 and the environmental effects of piping failures outside containment are discussed in Section 2.5.1.3. Pipe failures in high energy non-nuclear safety and field routed piping not rigorously analyzed are postulated at all adverse locations with regards to systems, structures, and components that are important to safety. Although condensate piping is high energy per Browns Ferry licensing basis definitions, a break in the condensate system does not affect structures, systems, and components that are important to safety. Therefore, the condensate system has not been evaluated for postulated pipe failures.

Browns Ferry does not evaluate stratification in the piping evaluations of record and does not monitor for stratification at CLTP conditions. This disposition remains unchanged for EPU.

Main Steam and Associated Piping System Evaluation

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

Temperatures and pressures in the MS piping, including attached MS branch piping and turbine bypass piping, will not increase with EPU. Because MS piping pressures and temperatures do not increase with EPU, there was no effect on the analyses due to these parameters. The increase in MS flow results in increased transient forces from the TSV closure. The TSVC transient load is the only load increase for the MS piping and supports at EPU conditions.

For MS piping outside containment, a new pipe stress analysis was performed to evaluate the increased TSVC fluid transient loading with EPU. Detailed TSVC fluid transient forcing functions were developed and the piping stress analysis was evaluated at EPU conditions to determine loads at EPU due to the TSVC transient. The EPU MS analysis resulted in MS piping outside containment meeting all code of record criteria (Reference 44), and the pipe stress results are shown in Table 2.2-4a. With the implementation of support modifications as described in EPU LAR Attachment 47, the revised analysis for the MS system outside containment demonstrates that the design has sufficient margin between calculated stresses and the allowable limits in the Browns Ferry code of record to justify operation at EPU conditions.

Pipe Stresses

Reactor dome pressure and temperature remain unchanged for EPU. MS piping pressure and temperature at the TSV decrease slightly with increased friction losses at EPU. The results of the EPU stress analysis for MS outside containment demonstrate that the piping design has sufficient margin between calculated stresses and code allowable limits (see Table 2.2-4a) to justify operation at EPU conditions and that no modifications are necessary as a result of EPU.

Pipe Supports

Based on the MS pipe stress analysis, the pipe support loads, after the implementation of pipe support modifications discussed in LAR Attachment 47, remain within design load limits to Code allowables.

Therefore, the MS and associated piping meets all CLTR dispositions.

<u>Feedwater System Evaluation</u>

Operation at EPU conditions increases stresses on piping and piping system components due to slightly higher operating temperatures. Higher FW operating pressures result from the higher head loss associated with a higher FW flow rate. The increase in FW system operating pressure at EPU remains bounded by the FW system design pressure (1,250 psig) used in the current licensing basis stress calculations. The FW piping systems outside containment have been evaluated in accordance with the plant code of record criteria (Reference 44) for the EPU conditions based on the design margins between actual stresses and applicable code limits. All piping is below the code allowable of the present code of record (Reference 44). No new postulated pipe break locations were identified.

Pipe Stresses

Because the FW system piping pressures and temperatures increase slightly due to EPU, the effects of these parameters on the existing analyses were evaluated. Existing FW piping analyses are performed to design pressures and temperatures which remain bounding relative to EPU conditions. Seismic inertia loads, and seismic building displacement loads are not affected by EPU; thus, there is no effect on the analyses for these load cases. Other external loading conditions also are not changed by EPU. For the FW piping outside containment, there is no FW system fluid transient analysis in the existing design basis analysis, so the increase in FW system flow has no effect on the current analysis.

The FW temperature and pressure changes are insignificant relative to piping design for EPU. Also, fluid conditions in the FW piping design analyses bound the FW operating conditions at EPU. The flow change does not affect the FW piping design system because fluid transient loading is not a design load in the original or current stress analyses. Therefore, the current licensing basis for the FW system complies with the code of record for the effect of thermal expansion displacement on the piping snubbers, hangers, and struts. Piping interfaces with penetrations, flanges, and valves also remain valid per the current licensing basis.

Although fluid transient analyses are not required as part of the FW piping design basis, an analysis was performed that assessed the effect of transients on the FW piping at EPU conditions. The bounding transient considered was for a simultaneous trip of all three FW pump turbines that results in transient loading on piping as the FW pumps coast down. The results of this analysis showed that the BOP FW piping is acceptable for FW fluid transients that occur at EPU conditions and that the design has sufficient margin between calculated stresses/loads and the allowable limits in the code of record (Table 2.2-3d).

A review was also performed of postulated high energy pipe break locations in accordance with the requirements of the current licensing basis methodology. As a result of this review, no new postulated break locations were identified.

Based on existing margins available for the FW piping, it was concluded that EPU does not have an adverse effect on the FW piping design.

Pipe Supports

The FW system piping outside containment was evaluated for the effects of EPU operating pressure and temperature increase on the piping design analyses. There is no fluid transient analysis in the current FW design basis for Browns Ferry. Because the existing analyses bound the EPU conditions, it was concluded that EPU does not have an adverse effect on FW pipe support loads and design.

Therefore, the FW system meets all CLTR dispositions.

<u>Other Piping Evaluation</u>

Torus Attached Piping

The DBA-LOCA hydrodynamic loads, including the pool swell loads, condensation oscillation (CO) loads and chugging loads are not changed for Browns Ferry EPU. For EPU conditions, the DBA-LOCA containment response loads were evaluated and found to be unchanged by EPU (see Section 2.6.1.2.1) and thus, there are no resulting effects on the containment/torus attached piping and valves. The suppression pool temperature response for large and small break LOCA and other events is evaluated in Section 2.6 and is reported in Table 2.6-1. The peak suppression pool (SP) temperatures for these events at EPU are bounded by the current design analyses of the torus attached piping where the piping was analyzed at a conservatively high peak SP temperature of 187.3°F. With the implementation of support modifications to reinforce an existing pad at an ECCS ring header branch connection as described in EPU LAR Attachment 47, the bounding analysis for the ECCS ring header at a bounding peak SP temperature of 187.3°F demonstrates that the design has sufficient margin between calculated stresses and allowable limits in the Browns Ferry code of record to justify operation at EPU conditions. Other external loading conditions for the torus attached piping (e.g., seismic loads) are not affected by EPU. Additionally, piping dynamic loads due to MSRV discharge at EPU conditions (Section 2.6.1.2.2) are bounded by those used in the current Browns Ferry analyses of record.

The containment hydrodynamic load evaluation in Section 2.6.1.2.3 states that the SBA event thermal loads at EPU conditions (based on a SP temperature of 146°F) do not bound the thermal loads used in the Reference 45 load definition report (based on a SP temperature of 136°F), which provided thermal load input for subsequent use in the Reference 46 structural analysis of torus attached piping. Review of the Browns Ferry design calculations for the torus attached piping shows that the current design calculations conservatively used the Reference 45 thermal loads from the intermediate break accident (IBA) event (based on a SP temperature of 158°F) for load combinations that required hydrodynamic loading in the SBA and IBA analysis. The Reference 45 IBA event thermal load bounds both the EPU SBA and IBA thermal loads. Other SBA/IBA containment hydrodynamic loads/load combinations at EPU are either unchanged or bounded by the loads/load combinations used in the current design analyses of the torus attached piping.

For fire events (classified by TVA for piping design as an emergency condition service level) the EPU peak SP temperature (Section 2.5.1.4) is bounded by the current design analyses of the torus attached piping where the piping was analyzed at a conservative peak SP temperature of 223°F for fire events.

The load conditions for the torus attached piping are either unchanged for EPU or bounded by the loads used in the Browns Ferry current analysis of record. With the implementation of pipe support modifications to reinforce an existing pad at an ECCS ring header branch connection as discussed in LAR Attachment 47, the torus attached piping meets all code of record criteria at EPU conditions.

Other BOP Piping Systems

The piping and pipe supports of the other BOP systems affected by EPU loading increases were reviewed to determine if there is sufficient capacity margin to accommodate the increased loadings. This review shows that existing piping design analyses are performed to pressures and temperatures which bound EPU conditions for some systems. For others, the design analyses have sufficient margin between the calculated and Code allowable stress limits to accommodate the small increases in pressure or temperature with EPU. The evaluations (see Tables 2.2-4b through 2.2-4f and 2.2-5) demonstrate that for all systems, design margins for piping, supports and equipment nozzles are either unaffected by EPU or are adequate to accommodate the increased loads and movements resulting from EPU.

For BOP systems that do not require a detailed analysis, pipe routing and flexibility are considered to remain acceptable for EPU. These are non-safety-related BOP systems for which no piping or support analyses are documented. These are generally cold systems (< 200°F) where thermal stresses and displacements are not significant. For these systems, pipe routing and flexibility are considered to remain acceptable, as the pipe routing is not being changed with EPU, and any increase in operating temperature range with EPU is not significant.

2.2.2.3 Reactor Vessel

The RPV structure and support components form a pressure boundary to contain reactor coolant and moderator, and form a boundary against leakage of radioactive materials into the drywell. The RPV also provides structural support for the reactor core and internals.

The fatigue of plant-specific components is monitored for license renewal (Reference 11) and those components are listed in the table below. These components have been reviewed [[

]]

The high and low pressure seal leak detection nozzles have been reviewed and found acceptable for 60-year EPU conditions.

The effect of EPU was evaluated to ensure that the reactor vessel components continue to comply with the existing structural requirements of the ASME Boiler and Pressure Vessel Code. For the OLTP components under consideration, the ASME Boiler and Pressure Vessel Code, Section III, 1965 Code with Addenda to and including Summer 1965 (Units 1 and 2) and the ASME Boiler and Pressure Vessel Code, Section III, 1965 Code with Addenda to and including Summer 1966 (Unit 3) are applicable. These were used as the governing code and are considered the Code of Construction. However, if a component's design has been modified, the governing code for that component was the code used in the stress analysis of the modified component.

The following components that [[

]] were modified since the original construction are:

- FW Nozzle: This component was modified and the governing Code for the modification is the ASME Boiler and Pressure Vessel Code, Section III, 1974 Edition with Addenda to and including Summer 1976 (Units 1, 2 and 3).
- Recirculation Inlet Nozzle: This component was modified and the governing Code for the modification is the ASME Boiler and Pressure Vessel Code, Section III, 1980 Edition with Addenda to and including Winter 1981 (Units 1, 2 and 3).
- Recirculation Outlet Nozzle: This component was modified and the governing Code for the modification is the ASME Boiler and Pressure Vessel Code, Section III, 1980 Edition with Addenda to and including Winter 1981 (Units 1, 2 and 3).
- Core Spray Nozzle: This component was modified and the governing Code for the modification is the ASME Boiler and Pressure Vessel Code, Section III, 1974 Edition with Addenda to and including Summer 1976 (Units 1, 2 and 3).
- CRD Hydraulic System Return Nozzle Cap: This component was newly installed and the governing Code for the modification is the ASME Boiler and Pressure Vessel Code, Section III, 1974 Edition with Addenda to and including Summer 1976 (Units 1, 2 and 3).

- Jet Pump Instrumentation Seal Safe End: This component was modified and the governing Codes for the modification are the ASME Boiler and Pressure Vessel Code, Section III, 1980 Edition with Addenda to and including Winter 1981 (Units 1 and 2) and the ASME Boiler and Pressure Vessel Code, Section III, 1986 Edition (Unit 3).

New stresses are determined by scaling the "original" stresses based on the EPU conditions [[ ]]. The analyses were performed for the design, the normal and upset, and the emergency and faulted conditions. If there is an increase in jet reaction, pipe restraint or fuel lift loads, the changes are considered in the analysis of the components affected for normal, upset, emergency and faulted conditions. In all evaluations, all six stress components were considered; no simplified single stress methodology was employed.

Design Conditions

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

Normal and Upset Conditions

The reactor coolant temperature and flows (except: FW flow, recirculation flow, and main steam flow) at EPU conditions are only slightly changed from those at current rated conditions. Evaluations were performed at conditions that bound the change in operating conditions. The evaluation type is mainly reconciliation of the stresses and usage factors to reflect EPU conditions. A primary plus secondary stress analysis was performed showing EPU stresses still meet the requirements of the ASME Code, Section III. Lastly, the fatigue usage was evaluated for the limiting location of components with a [[ ]] The Browns Ferry fatigue analysis results for the limiting components are provided in Table 2.2-6. The plant-specific evaluations were performed with environmental fatigue using NUREG/CR-6909 (Reference 47) for the fatigue life correction factor to the ASME fatigue analyses. This was done to support Browns Ferry license renewal (Reference 11).

Emergency and Faulted Conditions

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

Therefore, reactor vessel meets all EPU dispositions.

<u>Conclusion</u>

TVA has evaluated the structural integrity of pressure-retaining components and their supports and has addressed the effects of the proposed EPU on these components and supports. The evaluation indicates that pressure-retaining components and their supports will continue to meet the requirements of 10 CFR 50.55a, draft GDCs-1, 2, 9, 33, 34, 40, and 42 following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to the structural integrity of the pressure-retaining components and their supports.

2.2.3 Reactor Pressure Vessel Internals and Core Supports

Regulatory Evaluation

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

The NRC's acceptance criteria are based on (1) 10 CFR 50.55a and GDC-1, insofar as they require that SSCs important to safety be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety functions to be performed; (2) GDC-2, insofar as it requires that SSCs important to safety be designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions; (3) GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents; and (4) GDC-10, insofar as it requires that the reactor core be designed with appropriate margin to assure that specified acceptable fuel design limits (SAFDLs) are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

Specific NRC review criteria are contained in SRP Sections 3.9.1, 3.9.2, 3.9.3, and 3.9.5 and other guidance provided in Matrix 2 of RS-001.

Browns Ferry Current Licensing Basis:

The General Design Criteria (GDC) listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Browns Ferry Nuclear Plant (Browns Ferry) principal design criteria predate these criteria. The Browns Ferry principal design criteria are listed in UFSAR Section 1.5, "Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, the Tennessee Valley Authority (TVA) performed a comparative evaluation of the design basis of Browns Ferry with the AEC proposed General Design Criteria of 1967. The Browns Ferry UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria set out in the July 1967 AEC release. For each group of criteria, there is a statement of TVA's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a table of references to locations in the Browns Ferry UFSAR where there is subject matter relating to the intent of that particular criteria.

While Browns Ferry is not generally licensed to the final GDC or the 1967 AEC proposed General Design Criteria, a comparison of the final GDC to the applicable AEC proposed General Design Criteria can usually be made. For the final GDC listed in the Regulatory Evaluation above, the Browns Ferry comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Browns Ferry UFSAR Appendix A: draft GDCs-1, 2, 40 and 42. Final GDC-10 is applicable to Browns Ferry as

described in "Browns Ferry Nuclear Plant, Unit 1 - Issuance of Amendments Regarding the Transition to Areva Fuel," dated July 3, 2012. (Reference 48)

The Reactor Pressure Vessel Internals and Core Supports are described in Browns Ferry UFSAR Sections 3.3, "Reactor Vessel Internals Mechanical Design," and 4.2, "Reactor Vessel and Appurtenances Mechanical Design."

In addition to the evaluations described in the Browns Ferry UFSAR, Browns Ferry's systems and components were evaluated for license renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Browns Ferry License Renewal Safety Evaluation Report (SER), NUREG-1843, dated April 2006 (Reference 11). The reactor internals and core support structural components evaluation for license renewal are discussed in NUREG-1843, Section 3.1.

<u>Technical Evaluation</u>

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of EPUs. Sections 3.3 and 3.4 of the CLTR address the effect of EPU on Reactor Vessel and Reactor Internals, respectively. The results of this evaluation are described below.

2.2.3.1 FIV Influence on Reactor Internal Components

The FIV evaluation of the RPV internals addresses the influence of an increase in flow during EPU. Browns Ferry meets all CLTR dispositions. The topics addressed in this evaluation are:

Topic	CLTR Disposition	Browns Ferry Result
Structural Evaluation of Core Flow Dependent RPV Internals	Generic	Meets CLTR Disposition
Structural Evaluation of Other RPV Internals	Plant Specific	Meets CLTR Disposition

2.2.3.1.1 Structural Evaluation of Core Flow Dependent RPV Internals

As stated in Section 3.4.2 of the CLTR, EPU causes an increase in reactor coolant quality and an increase in FW, steam, and recirculation pump drive flow.

[[ ]]

The core flow dependent RPV internal components (in-core guide tube and control rod guide tube) are confirmed to be consistent with the generic dispositions provided in the CLTR [[

]]

2.2.3.1.2 Structural Evaluation of Other RPV Internals

As stated in Section 3.4.2 of the CLTR, EPU causes an increase in reactor coolant quality and an increase in FW, steam, and recirculation pump drive flow.

The required RPV internals vibration assessment of the other RPV internals is described in the CLTR. EPU operation increases the steam production in the core, resulting in an increase in the core pressure drop. [[

]] The increase in power may increase the vibration level of reactor internals. Analyses were performed to evaluate the effects of FIV on the reactor internals at EPU conditions. This evaluation used a bounding reactor power of 102% of 3,952 MWt and 105% of rated core flow. [[

]] For components requiring an evaluation but not instrumented in Browns Ferry Unit 1, [[
]] The expected vibration levels for EPU were estimated by extrapolating the measured vibration data in Browns Ferry or similar plants and based on GEH BWR operating experience. These expected vibration levels were then compared with the established vibration acceptance criteria. The following components were evaluated:

a) Control Rod Guide Tube (CRGT)

b) In Core Guide Tubes (ICGT)

c) Feedwater Spargers

d) Jet Pumps (JP)

e) Jet Pump Sensing Lines (JPSL)

f) Shroud

g) Shroud Head and Separator Assembly

h) Core Spray Piping Line and Sparger

i) Fuel Assembly

j) Guide Rod

k) Shroud Head Bolts

l) Top Guide

m) Head Spare Instrument Nozzle

n) Top Head Instrument Nozzle

o) Top Head Vent Nozzle

p) Steam Line Nozzle

q) Water Level Instrument Nozzle

The results of the vibration evaluation show that continuous operation at a reactor power of 102% of 3,952 MWt and 105% of rated core flow does not result in any detrimental effects on the critical or safety-related reactor internal components shown above. Flow induced vibration of critical reactor internal components at EPU is predicted based on the available startup test data at [[ ]] Vibration amplitudes are also adjusted by a [[ ]] The extrapolated vibration amplitude response under EPU conditions is compared with the acceptance criterion in the percent criteria for each mode. The percentages of the criteria for all modes are cumulative as total percent criteria. [[ ]] The summary of the evaluation methods and results for the following components are:

Control Rod Guide Tubes (CRGT) and In-Core Guide Tubes (ICGT)

The vibration of the CRGT and ICGT in the lower plenum is a function of core flow. Because the maximum core flow under EPU remains unchanged from the OLTP or CLTP, the flow-induced vibrations of these components are not affected under EPU. Hence, there will be no increase in FIV stresses due to EPU. Maximum stresses during OLTP are well within the acceptance criteria and will remain about the same at EPU conditions.

FW Sparger

The FW sparger in Browns Ferry is of the improved triple thermal sleeve design. [[ ]] Therefore, the Browns Ferry FW sparger is acceptable under EPU conditions.

Jet Pumps

Results from strain gage measurements [[ ]]

Jet Pump Sensing Lines

Resonance of the recirculation pump Vane Passing Frequency (VPF) with the natural frequency of the JPSL is the cause of the JPSL stress. [[

]] The jet pump sensing lines remain acceptable for EPU conditions.

Shroud

For the shroud, the measured vibrations were extrapolated to the EPU conditions. Maximum stresses are less than 2,400 psi at OLTP and will remain well within acceptance criteria at EPU conditions. The calculated maximum stress is about 38% of the acceptance criteria or 3,800 psi at EPU conditions.

Shroud Head and Separator Assembly

For the shroud head and separator, [[

]]

Core Spray Piping and Sparger

[[

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

During EPU, the components in the core region and components such as the core spray line are primarily affected by the core flow. Components in the annulus region such as the jet pump are primarily affected by the recirculation pump drive flow and core flow. For EPU conditions at Browns Ferry, there is no change in the maximum licensed core flow as compared to the CLTP condition, resulting in negligible changes in FIV on the components in the annular and core regions. The core spray sparger is subjected to a very low flow velocity with no resonance. This indicates that FIV adequacy is assured during EPU operation.

Fuel Assembly

The EPU FIV effect on fuel assembly is contained in Attachments 24 and 26 (proprietary) and 25 and 27 (non-proprietary) of the EPU LAR.

Guide Rods

The guide rod is subjected to cross flow, and the procedure and criteria as established in ASME Code Section III N-1300 (Reference 43) is used. [[

]] Therefore, the guide rod at Browns Ferry is acceptable under FIV for EPU conditions.

Shroud Head Bolts

The shroud head bolt is subjected to cross flow, and the procedure and criteria as established in ASME Code, Section III, N-1300 (Reference 43) is used. [[

]]
Therefore, the shroud head bolt at Browns Ferry is acceptable under FIV for EPU conditions.

Top Guide

The core flow does not change under EPU, and the flow velocity around the top guide is lower than [[ ]] Therefore, the increase in FIV loads due to EPU conditions is minimal and will not adversely affect the structural adequacy of the top guide.

RPV Head Spare Instrument Nozzle

[[

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

RPV Top Head Instrument Nozzle

[[

]]
Thus, the stress in the nozzle due to FIV at EPU conditions is negligible.

RPV Top Head Vent Nozzle

[[

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

Steam Line Nozzle

[[

]] Thus, the stress in the nozzle due to FIV at EPU conditions is negligible.

Water Level Instrument Nozzle

The water level instrumentation nozzle connected to the steam leg is in the stagnant steam region, and is not affected by EPU, and the instrumentation nozzle connected to the reference leg of the narrow range reactor water level experiences similar changes as the lower part of the shroud guide rods or core spray pipe. The water level instrument nozzle with high fundamental natural frequency under low flow velocities assures that the FIV effects would be negligibly small.

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

The GEH criterion of 10,000 psi peak stress intensity is more conservative than the ASME allowable peak stress intensity of 13,600 psi for service cycles $\geq 10^{11}$.

Conservatively, the peak responses of the applicable modes are absolute summed.

The maximum vibration amplitude in each mode is used in the absolute sum process, whereas in reality the peak vibration amplitudes are unlikely to occur at the same time.

Therefore, it is concluded that the flow-induced vibrations for all evaluated components remain within the acceptance limits, and the FIV effects on reactor internal components meets all CLTR dispositions.

Steam Dryer

During EPU, the components in the upper zone of the reactor, such as the steam dryer, are mostly affected by the increased steam flow. As a result, the steam dryer can be significantly affected by EPU conditions.

The steam dryer is a non-safety-related components. Recent uprate experience indicates that FIV at EPU conditions may lead to high cycle fatigue failure of some dryer components. Failure of a dryer component does not represent a safety concern, but can result in a large economic effect. Quantitative analyses of the Browns Ferry steam dryers have been performed. The results showed that modifications to enhance structural integrity of the steam dryers would be needed for EPU conditions. Rather than modify the existing dryers, TVA has made a decision to replace the steam dryers. Attachment 40 of the Browns Ferry EPU license amendment request provides the analyses of the replacement steam dryers.

2.2.3.2 Reactor Internals

The RPV internals consist of the core support structure components and non-core support structure components. Browns Ferry meets all CLTR dispositions. The topics addressed in this evaluation are:

Topic	CLTR Disposition	Browns Ferry Result
Reactor Internal Pressure Differences	Plant Specific	Meets CLTR Disposition
Reactor Internals Structural Evaluation	Plant Specific	Meets CLTR Disposition
Steam Dryer Separator Performance	Plant Specific	Meets CLTR Disposition

2.2.3.2.1 Reactor Internal Pressure Differences

As stated in Section 3.3.1 of the CLTR, EPU results in higher pressure differences across the RPV internals due to higher core exit steam flow. The increase in core average power alone would result in higher core loads and Reactor Internals Pressure Differences (RIPDs) due to the higher core exit steam quality.

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

Tables 2.2-7 through 2.2-9 compare the RIPDs across the major reactor internal components during current and EPU operation in the Normal, Upset, and Faulted conditions, respectively.

The EPU reactor internal pressure difference (RIPD) calculations that are sensitive to fuel type are performed with a full core of GE13 fuel that includes the debris filter lower tie plate option. The RIPDs for GE13 fuel were demonstrated to be bounding for GE14 fuel as part of the GE14 new fuel introduction program. This is due to the higher flow resistance and resultant higher pressure drop of the GE13 fuel bundle. The RIPDs for GE13 fuel were also found to be bounding for both ATRIUM-10 and ATRIUM 10XM fuels except for the channel wall differential pressure (DP). The fuel channel RIPDs are discussed in Section 2.2.3.2.2. The EPU RIPDs are therefore applicable to GE13, GE14, ATRIUM-10, and ATRIUM 10XM fuel types.

The acoustic and flow-induced loads following a postulated recirculation line break were also evaluated using TRACG models (see Table 1-1). The methodology for determining the Browns Ferry acoustic and flow-induced loads at EPU rated thermal power is unchanged from that used

for current rated thermal power and is unaffected by the issue identified in GEH Safety Communication 12-20 (Reference 50). The acoustic and flow-induced loads associated with the extension of the MELLLA and ICF domain to include EPU operation are bounded by the acoustic and flow-induced loads associated with reduced feedwater temperature operation at the minimum pump speed point on the MELLLA line (Point "C" of Figure 1-1).

2.2.3.2.2 Reactor Internals Structural Evaluation (Non-FIV)

As stated in Section 3.3.2 of the CLTR, the typical loads considered in the EPU structural evaluation of the internals include: dead weight, RIPDs, seismic loads, thermal loads, flow loads, and acoustic and flow-induced loads due to a recirculation line break, consistent with the design basis. [[
]]

The RPV internals consist of the core support structure components and non-core support structure components. The RPV internals are not ASME Code components. However, the requirements of the ASME Code are used as guidelines in their design/analysis. The evaluations/stress reconciliation in support of EPU was performed consistent with the design basis analysis of the components. The reactor internal components evaluated are:

<u>Core Support Components</u>

- Shroud
- Shroud Support
- Core Plate
- Top Guide
- Control Rod Drive Housing (CRDH)
- Control Rod Guide Tube (CRGT)
- Orificed Fuel Support (OFS)
- Fuel Channel

<u>Non-Core Support Components</u>

- FW Sparger
- Jet Pumps
- Core Spray Line and Sparger
- Access Hole Cover (AHC)
- Shroud Head and Steam Separator Assembly
- In-Core Housing and Guide Tube (ICHGT)
- Vessel Head Cooling Spray Nozzle

- Jet Pump Instrument Penetration Seal
- Differential Pressure and Standby Liquid Control Line

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

The effects of the thermal-hydraulic changes due to EPU on the reactor internals were evaluated. All applicable loads and load combinations were considered consistent with the existing design basis analysis. These loads include the RIPDs (Section 2.2.3.2.1), dead weight, seismic loads, acoustic and flow induced loads, scram and thermal loads.

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

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

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

a) **Shroud:** The only shroud load affected by EPU is RIPD. Seismic and flow induced loads remain unaffected by EPU. Acoustic loads are bounded by the design basis values. The RIPDs show increases with respect to the Normal/Upset condition; whereas, for the Faulted condition, only the shroud head RIPD increases. Because buckling is the limiting stress condition for the shroud, the tensile (positive) stresses produced by pressure are conservatively neglected in the shroud buckling analysis, consistent with the current design basis analysis. Therefore, for EPU, there is no change to the previous results, and the stresses remain unchanged and within Code allowable values. Therefore, the shroud is structurally qualified for EPU.

b) **Shroud Support:** The only shroud support load affected by EPU is RIPD. Seismic loads and flow induced loads remain unchanged by EPU. Acoustic loads are bounded by the design basis values. The RIPDs show increases for the Normal/Upset service conditions; for the Faulted condition, only the shroud head RIPD increases. The effect of the change in EPU Normal/Upset condition RIPDs was assessed with respect to the design basis analysis

and the resultant stress was found to be within the allowable value. Therefore, the shroud support remains structurally qualified for EPU.

c) **Core Plate:** The only core plate load affected by EPU is RIPD. All other applicable loads (including dead weight and seismic) remain unaffected. Only the Normal/Upset condition RIPDs increase for EPU. The existing basis beam buckling loads and sliding were reconciled based on the changes in the pressure loads. The reconciled beam buckling and sliding results remain within allowable limits. Normal condition RIPD is bounded by the core plate plug design basis value. The minimum core plate plug design basis 14 EFPY design life based on IGSCC is still bounding for EPU conditions. Therefore, the core plate (including core plate plugs) remains structurally qualified for EPU.

d) **Top Guide:** The only top guide load affected by EPU is RIPD. All other applicable loads remain unaffected. The top guide EPU RIPDs are less than the previously qualified CLTP values for all service conditions. Therefore, the top guide remains unaffected by and structurally qualified for EPU.

e) **Control Rod Drive Housing (CRDH):** The CRDH (internal to the vessel) is subjected to the following primary loads: weight (guide tube + fuel), pressure, scram loads, seismic and the flow loads in the lower plenum. As a result of the EPU, the CRDH design pressure loads (vessel pressure), design scram loads, seismic loads and the flow in the lower plenum remain unaffected. The temperature change in the lower plenum is insignificant based on the results of the recirculation system analysis. Thus, there is no significant effect on the thermal stress conditions of the CRDH. Therefore, the CRDH remains unaffected and structurally qualified for EPU.

f) **Control Rod Guide Tube (CRGT):** Only the EPU Normal/Upset RIPDs increase relative to the previously qualified value. All other loads remain unaffected by EPU. Adequate CRGT lift margin exists in the EPU condition. The temperature and flow remain unchanged for EPU. The contribution of the increased RIPDs is small and the resulting stress for EPU is well within the stress allowable values. Therefore, the CRGT remains structurally qualified for EPU.

g) **Orificed Fuel Support (OFS):** The only OFS load affected by EPU is RIPD, which increases for the Normal/Upset conditions only. All other loads (dead weight and seismic) remain unaffected. The contribution of the increased RIPDs is small, and the resulting stress for EPU conditions is well within the stress allowable values. Therefore, the OFS remains structurally qualified for EPU.

h) **Fuel Channel:** The fuel channel RIPDs are within the design limits of the fuel for all service conditions. See LAR Attachments 24 and 26 for the fuel channel evaluations.

i) **Feedwater Sparger:** The only change as a result of EPU is the change in the feedwater flow and the temperature. All other applicable loads remain unaffected. Flow related loading is a minimal contributor to the primary stress in the feedwater sparger. The effect of increase in feedwater temperature due to EPU is bounded by the CLTP evaluation. The

change in the maximum flow, as documented in the reactor heat balance (Table 1-2), has an insignificant effect on the primary stress integrity of the component. Therefore, the feedwater sparger remains structurally qualified for EPU.

j) **Jet Pumps:** The predominant loads for jet pumps are: seismic loads, hydraulic flow loads, acoustic and flow induced loads. The change in the jet pump drive flow due to EPU is insignificant. The change in the flow temperature due to EPU is insignificant. Acoustic loads are bounded by the design basis values. The load conditions pertaining to the jet pump riser brace repair (Unit 3) remain unaffected by EPU. The existing repair design basis remains valid. The repair inspection interval is every other refueling outage, and the next scheduled inspection is during the Unit 3 Refueling Outage 18 (U3R18) in the Spring of 2018. Therefore, the jet pumps remain structurally qualified for EPU.

k) **Core Spray Line and Sparger:** The core spray system flow load and pressure remain unaffected. Because vessel pressure is unchanged due to EPU, the Faulted condition annulus downcomer load also remains unaffected. Seismic loads are unaffected by EPU. The thermal condition (< 2°F temperature difference) in the annulus remains practically unchanged. Therefore, the core spray line and the sparger remain structurally qualified for the EPU condition. Unrelated to EPU, the Unit 3 core spray line T-box and downcomer have been modified. However, because the applicable loads for the core spray system remain unaffected by EPU, the Unit 3 core spray line T-box and downcomer remain qualified in the as modified condition.

l) **Access Hole Cover (AHC):** The AHC experiences the same pressures as the shroud support plate and these RIPDs increase for Normal/Upset EPU conditions only. There is no significant change to the temperature and seismic loads due to EPU. The AHC location specific acoustic load is considered in the assessment. The effect of the EPU RIPDs and acoustic load is evaluated and reconciled with respect to the AHC design basis analysis. The design basis analysis remains valid for EPU conditions. Therefore, the AHC remains structurally qualified for EPU.

m) **Shroud Head and Steam Separator Assembly:** The only shroud head load affected by EPU is RIPD, which increases for all (Normal, Upset, and Faulted) conditions. Seismic and other dynamic loads are not affected by EPU. While the limiting factor for the assembly is the Shroud Head Bolt (SHB), the stress on the SHBs due to the increased shroud head RIPDs remains within the allowable for EPU. Thus, the shroud head and steam separator assembly remains qualified for EPU.

n) **In-Core Housing and Guide Tube (ICHGT)**: There is no change in the dead weight and seismic loads due to EPU. The temperature (< 2°F) and flow in the lower plenum remain essentially unchanged for EPU. The existing design basis remains acceptable. Thus, the ICHGT remains structurally qualified for EPU.

o) **Vessel Head Cooling Spray Nozzle**: The vessel head cooling spray nozzle is subject to dome pressure, seismic, and temperature effects. For EPU, there is no change in the nominal

dome pressure or temperature; seismic load also remains unchanged. Thus, the vessel head cooling spray nozzle remains structurally qualified for EPU. The vessel head cooling spray nozzle was capped and is no longer operational. The structural qualification of the vessel head cooling spray nozzle remains valid because there is no change to the loads when the nozzle is not in use.

p) **Jet Pump Instrument Penetration Seal:** The jet pump instrument penetration seal is not affected by EPU conditions because the vessel pressure and temperature remains essentially unchanged. EPU has no effect on seismic loading, which remains unchanged. Therefore, the jet pump instrument penetration seal remains structurally qualified for EPU conditions.

q) **Differential Pressure and Standby Liquid Control Line:** The core flow and the temperature are essentially unchanged for EPU conditions. Also, EPU has no effect on the existing seismic response of the differential pressure and standby liquid control line system. Therefore, the differential pressure and standby liquid control line remains structurally qualified for EPU.

2.2.3.2.3 Steam Dryer/Separator Performance

For Browns Ferry, the EPU performance of the steam dryer/separator was evaluated to ensure that the quality of steam leaving the reactor pressure vessel continues to meet existing operational criteria at EPU conditions. EPU results in an increase in saturated steam generated in the reactor core. For constant core flow, this in turn results in an increase in the separator inlet quality and dryer face velocity and a decrease in the water level inside the dryer skirt. These factors, in addition to the core radial power distribution, affect the steam dryer/separator performance.

The results of the evaluation demonstrated that Browns Ferry, using a representative equilibrium core design, has acceptable steam dryer/separator performance (i.e., moisture carryover $\leq$ 0.1 wt. %) at EPU conditions. Moisture carryover measurements are to be performed as part of the power ascension test plan as described in LAR Attachment 46.

Conclusion

TVA has evaluated the effects of the proposed EPU on the reactor internals and core supports. The evaluation indicates that the reactor internals and core supports will continue to meet the requirements of 10 CFR 50.55a, final GDC-10 and draft GDCs-1, 2, 40 and 42 following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to the design of the reactor internal and core supports.

2.2.4 Safety-Related Valves and Pumps

Regulatory Evaluation

The NRC's acceptance criteria are based on (1) GDC-1, insofar as it requires that SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed; (2) GDC-37, GDC-40, GDC-43, and GDC-46, insofar as they require that the ECCS, the containment heat

removal system, the containment atmospheric cleanup systems, and the cooling water system, respectively, be designed to permit appropriate periodic testing to ensure the leak-tight integrity and performance of their active components; (3) GDC-54, insofar as it requires that piping systems penetrating containment be designed with the capability to periodically test the operability of the isolation valves to determine if valve leakage is within acceptable limits; and (4) 10 CFR 50.55a(f), insofar as it requires that pumps and valves subject to that section must meet the inservice testing program requirements identified in that section.

Specific NRC review criteria are contained in SRP Sections 3.9.3 and 3.9.6 and other guidance provided in Matrix 2 of RS-001.

Browns Ferry Current Licensing Basis

The General Design Criteria (GDC) listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Browns Ferry Nuclear Plant (Browns Ferry) principal design criteria predate these criteria. The Browns Ferry principal design criteria are listed in UFSAR Section 1.5, "Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, the Tennessee Valley Authority (TVA) performed a comparative evaluation of the design basis of Browns Ferry with the AEC proposed General Design Criteria of 1967. The Browns Ferry UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria set out in the July 1967 AEC release. For each group of criteria, there is a statement of TVA's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a table of references to locations in the Browns Ferry UFSAR where there is subject matter relating to the intent of that particular criteria.

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

The inservice testing of safety-related valves and pumps is described in Browns Ferry UFSAR Section 6.6, "Inspection and Testing."

In addition to the evaluations described in the Browns Ferry UFSAR, Browns Ferry's systems and components were evaluated for license renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Browns Ferry License Renewal Safety Evaluation

Report (SER), NUREG-1843, dated April 2006 (Reference 11). The safety-related valves and pumps are addressed within NUREG-1843 under the systems that contain them.

Technical Evaluation

2.2.4.1 Background

In-Service Testing of Safety-Related Pumps and Valves

The In-service Testing (IST) of safety-related pumps and valves is addressed and documented in the Browns Ferry In-service Testing Program contained in Attachment 3, Part C "In-service Testing of Pumps and Valves" of TVA procedure NPG-SPP-09.1; "ASME Code and Augmented Programs." The Browns Ferry pump and valve in-service testing program, hereafter referred to as the IST program, meets the requirements of 10 CFR 50.55a(f).

The Browns Ferry Technical Specifications, Section 5.5.6, IST Program, states that this program provides controls for in-service testing of ASME Class 1, 2, and 3 pumps and valves and that the program shall include testing frequencies as specified in ASME OM Code, 2004 Edition through 2006 Addenda.

Containment Leakage Rate Testing Program

Containment leakage rate testing is addressed in UFSAR Section 5.2.5, and Browns Ferry TS Section 5.5.12. The Browns Ferry Containment Leakage Rate Testing Program implements testing requirements in accordance with 10 CFR 50, Appendix J, Option B, as modified by approved exemptions, and guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak Test Program," dated September 1995 (Reference 51). Tests that measure containment and isolation valve leak rates (Type A, B and C tests) are performed using the Technical Specification value for P_a. The containment design pressure is 56 psig.

From the containment analysis at EPU conditions, the peak containment pressure (P_a) is 49.1 psig. For Unit 2 and Unit 3 the containment peak pressure for EPU is lower than the CLTP Technical Specification P_a of 50.6 psig. For Unit 1 the P_a increases from the CLTP pressure of 48.5 psig to an EPU value of 49.1 psig. Therefore the leak rate testing requirements for containment and applicable isolation valves are affected by the proposed EPU. No components affected by the tests are transitioning from non-safety to safety as a result of EPU. Existing programmatic controls for the classification and maintenance of components and test values associated with Appendix J tests are sufficient to support EPU.

Pumps in the IST Program

The scope of the IST program is derived from the ASME OM Code, Subsection ISTB, "In-service Testing of Pumps in Light-Water Reactor Nuclear Power Plants." ASME Code Class boundaries and component safety functions are not affected by EPU and no system parameter changes are being introduced that will require revisions to the programs supporting 10 CFR 50.55a(f) requirements. Therefore, existing programmatic controls for the classification and maintenance of testing requirements associated with safety-related pumps are consistent with

the existing IST program and are sufficient to support EPU. Table 2.2-11 lists the systems with pumps in the IST program.

Valves in the IST Program

The scope of the IST program is derived from the ASME OM Code, Subsection ISTC, "In-service Testing of Valves in Light-Water Reactor Nuclear Power Plants." ASME Code Class boundaries and component safety functions are not affected by EPU and no system parameter changes are being introduced that will require revisions to the programs supporting 10 CFR 50.55a(f) requirements. Therefore, existing programmatic controls for the classification and maintenance of testing requirements associated with safety-related valves are consistent with the existing IST program and are sufficient to support EPU. Table 2.2-11 lists the systems with valves in the IST program.

Motor Operated Valve Program

The Browns Ferry Motor Operated Valve (MOV) Program implements the recommendations and requirements made in Generic Letter 89-10, "Safety-Related Motor Operated Valve Testing and Surveillance" (Reference 52). The scope of the program also includes the requirements of Generic Letter 96-05, "Verification of Design-Basis Capability of Safety-Related Motor Operated Valves" (Reference 53). Existing programmatic controls for periodic verification requirements associated with safety-related valves are consistent with the existing GL 89-10 and GL 96-05 programs and are sufficient to support EPU. The existing Browns Ferry calculations for GL 89-10 MOVs were reviewed and the review shows that the maximum ambient temperatures used in existing MOV calculations bound the maximum ambient temperatures for EPU with the exception of one Unit 3 motor operated valve, 3-FCV-75-53. (See Table 2.2-12). This temperature increase has no effect on the affected valve capability or margin. Other parameters such as valve differential pressure/line pressure, motor terminal voltage, pressure locking and thermal binding, and valve stroke time effect were evaluated and were found to be either unaffected by EPU or the EPU effect was bounded by the parameters used in the existing calculations of record. The peak containment pressure following a LOCA increases slightly due to EPU (less than 1.4 psig from the peak pressures used in the existing MOV calculations). MOVs that are required to operate during a LOCA were evaluated for the changes in peak containment pressure and were found to maintain positive thrust/torque margin against the thrust/torque required for the valves to perform their open or close function. No valves under the Browns Ferry GL 89-10 and GL 96-05 program require modification to support EPU implementation. Operation at EPU conditions does not affect the capability of the GL 89-10 MOVs to perform their design basis functions. Table 2.2-11 indicates the systems that contain GL 89-10 MOVs.

Air-Operated Valve Program

The TVA Air Operated Valve program (NETP-114) was evaluated for compliance with the Joint Owner's Group (JOG) air operated valve testing requirements and will continue to provide assurance that AOVs will be appropriately monitored and maintained during plant operations

under EPU conditions. Currently, no Browns Ferry air operated valves are classified as Category 1. Category 2 and 3 valves do not require design verification. No AOVs change to Category 1 as a result of EPU.

Generic Letter 95-07

GL 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves," August 17, 1995 (Reference 54) addresses the phenomena of pressure locking and thermal binding of safety-related power-operated gate valves. Pressure locking and thermal binding had been previously evaluated for all Browns Ferry safety-related gate valves. There were no air-operated or hydraulic-operated valves that were susceptible to pressure locking or thermal binding. The evaluation identified a number of MOVs that were susceptible to these conditions, which resulted in valve disc modifications and/or valve replacements. A review of the station commitments and modifications related to GL 95-07 indicates that EPU will not cause additional safety-related gate valves, formerly excluded by screening criteria, to be susceptible to pressure locking or thermal binding, and EPU will not affect the susceptibility of valves already modified to prevent these problems. Therefore, EPU has no effect on the potential for pressure locking or thermal binding of safety-related power-operated gate valves. Table 2.2-11 indicates the systems that contain GL 95-07 valves.

Lessons Learned

The Browns Ferry IST program, Containment Leak Rate program, MOV program and air operated valve (AOV) program utilize the Browns Ferry Corrective Action Program to evaluate and resolve non-conforming conditions identified during program performance. The purpose of the Browns Ferry Corrective Action Program is to stimulate and manage continuous improvement of station and organizational performance through identification, evaluation, correction and prevention of reoccurrence of unwanted and/or unexpected conditions, deviations, events, or issues that have the potential for affecting the safe, reliable, and efficient operation of Browns Ferry. Included in the program is recognition of any lessons learned or improvement opportunities identified from an assessment of missed opportunities to avoid the event/condition/issue. NPG SPP-22.300, Revision 2, is the administrative procedure that implements the requirements of the corrective action program that complies with 10 CFR 50, Appendix B, Criterion 16.

2.2.4.2 Description of Analyses and Evaluations

This section addresses the effect of EPU on the performance requirements of Browns Ferry safety-related components in the IST, Motor Operated Valve, and Air Operated Valve programs.

For the majority of the valves analyzed for EPU effects, there are minor effects on normal operating and DBA ambient temperatures. However, because EPU does not result in a significant change to the temperature assumptions used in the MOV calculations, the operation of the affected valves is not affected.

Because the drywell and wetwell normal and accident pressures are not significantly changed by EPU, the associated containment valves were not affected.

The calculations of design basis parameters for MOVs within the scope of GL 89-10 and GL 96-05 were reviewed to determine the effect of EPU on the valves. In most cases, the values of existing parameters bound the values expected under EPU conditions. In a few cases, there are slight increases above the current value. In those cases, the effect of the slight increase on the MOV has been evaluated in accordance with the requirements of the station MOV program and found to be within the design parameters.

Systems Not Significantly Affected by EPU:

The following systems contain pumps and/or power operated valves but are not significantly affected by EPU: Service Air, Instrument Air, Pneumatic Nitrogen, Hydrogen Supply, Carbon Dioxide supply, Containment Systems, Floor Drains, Sanitary Drains, Radioactive Drains, Sewer, Torus Drain, Miscellaneous Drains, Amertap, Suppression Pool Cleanup, Fuel Oil systems, Lube Oil systems, Process Steam and Aux Boilers, Monitoring and Sampling systems, Laundry, Sewage Treatment, Showers, Water Supply Systems, and Water Quality Systems. With no changes or effects to these systems due to EPU, the Browns Ferry program valves in these systems are not affected by EPU.

Nuclear Steam Supply Systems:

The Nuclear Steam Supply System (NSSS) systems include Core Spray (Section 2.8.5.6.2), High Pressure Coolant Injection (Section 2.8.5.6.2), Reactor Core Isolation Cooling (Section 2.8.4.3), Residual Heat Removal (Section 2.8.4.4), Reactor Water Cleanup (Section 2.1.7) and Standby Liquid Control (Section 2.8.4.5). Evaluations show that EPU has no effect on system operating pressures, flow rates, and pump head performance for Core Spray, High Pressure Coolant Injection, Reactor Core Isolation Cooling and the Reactor Water Cleanup systems. Changes in the Standby Liquid Control system are addressed in Section 2.8.4.5. ASME Code Class boundaries and component safety functions within these systems are not affected by EPU. Additionally, EPU does not introduce any system parameter changes that will require IST program revisions.

Changes in containment response affect certain program valves of the NSSS systems. The affected valves are evaluated in Section 2.2.4.3.

The individual NSSS systems are evaluated below.

Reactor Water Cleanup System

The Reactor Water Cleanup system is not changed as a result of EPU and is addressed in Section 2.1.7.

ASME Code Class boundaries and component safety functions within this system are not affected by EPU. Additionally, EPU does not introduce any system parameter changes that will require IST program revisions.

Standby Liquid Control System

The Standby Liquid Control system changes due to EPU are addressed in Section 2.8.4.5. For EPU, the maximum pump discharge pressure occurring during the limiting ATWS event is calculated at 1,201 psig. The program valves in the system were found to be acceptable for EPU.

ASME Code Class boundaries and component safety functions within this system are not affected by EPU. System discharge increases slightly for EPU but due to the fact that IST test pressures do not require changes, the IST program is unaffected.

Reactor Core Isolation Cooling

No changes are being made to the RCIC system as a result of EPU. This system is addressed in Section 2.8.4.3.

ASME Code Class boundaries and component safety functions within this system are not affected by EPU. Additionally, EPU does not introduce any system parameter changes that will require IST program revisions.

Residual Heat Removal System

No Residual Heat Removal system changes are being considered due to EPU. This system is addressed in Section 2.8.4.4.

ASME Code Class boundaries and component safety functions within this system are not affected by EPU. Additionally, EPU does not introduce any system parameter changes that will require IST program revisions.

High Pressure Coolant Injection System

No changes are being made to the HPCI system as a result of EPU. This system is addressed in Section 2.8.5.6.2.

ASME Code Class boundaries and component safety functions within this system are not affected by EPU. Additionally, EPU does not introduce any system parameter changes that will require IST program revisions.

Core Spray System

No changes are being made to the Core Spray system as a result of EPU. This system is addressed in Section 2.8.5.6.2.

ASME Code Class boundaries and component safety functions within this system are not affected by EPU. Additionally, EPU does not introduce any system parameter changes that will require IST program revisions.

Balance of Plant Systems:

Changes in system flow rates for EPU affect certain components in the balance of plant systems. The affected components are evaluated in Section 2.2.4.3.

ASME Code Class boundaries and component safety functions within this system are not affected by EPU. Additionally, EPU does not introduce any system parameter changes that will require IST program revisions.

Reactor Building Closed Loop Cooling System

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

ASME Code Class boundaries and component safety functions within this system are not affected by EPU. Additionally, EPU does not introduce any system parameter changes that will require IST program revisions.

Feedwater / Feedwater Pump Recirculation Systems

The FW system changes due to EPU are addressed in Section 2.5.4.4. The temperature, flow, and operating pressure will increase; however, the current design conditions bound the conditions at EPU. The Browns Ferry MOVs affected by EPU are evaluated in Section 2.2.4.3. The ability of the IST program check valves to perform their safety functions is not affected by EPU.

ASME Code Class boundaries and component safety functions within this system are not affected by EPU. Additionally, EPU does not introduce any system parameter changes that will require IST program revisions.

Main Steam System

The proposed EPU results in MS flow increase; however, the current component design bounds the conditions at EPU regarding pressure and temperature.

The Main Steam Safety Relief Valves are addressed in Section 2.8.4.2. The MSRV setpoints remain the same.

ASME Code Class boundaries and component safety functions within this system are not affected by EPU. Additionally, EPU does not introduce any system parameter changes that will require IST program revisions.

Reactor Recirculation System

The Reactor Recirculation system is addressed in Section 2.8.4.6.

ASME Code Class boundaries and component safety functions within this system are not affected by EPU. Additionally, EPU does not introduce any system parameter changes that will require IST program revisions.

Fuel Pool Cooling System

The Fuel Pool Cooling system is addressed in Section 2.5.3.1. Fuel Pool Cooling requirements are unaffected by EPU and additional heat limits are administratively controlled.

ASME Code Class boundaries and component safety functions within this system are not affected by EPU. Additionally, EPU does not introduce any system parameter changes that will require IST program revisions.

Control Rod Drive Hydraulic System

The Control Rod Drive Hydraulic system is not changed due to EPU and the system is addressed in Section 2.8.4.1. Although there is a slight pressure increase for operation at EPU, it is within the design capability of the system components and no temperature changes occur for EPU.

ASME Code Class boundaries and component safety functions within this system are not affected by EPU. Additionally, EPU does not introduce any system parameter changes that will require IST program revisions.

Raw Cooling Water System

The RCW system has a small increase in temperature over current operation for EPU but remains within design limits.

ASME Code Class boundaries and component safety functions within this system are not affected by EPU. Additionally, EPU does not introduce any system parameter changes that will require IST program revisions.

Emergency Equipment Cooling Water System

For the EECW system, the heat load increases are insignificant and flow demand, pump duty, and system pressure will not significantly change.

ASME Code Class boundaries and component safety functions within this system are not affected by EPU. Additionally, EPU does not introduce any system parameter changes that will require IST program revisions.

RHR Service Water System

The RHR shutdown cooling mode initiating pressure and temperature do not change with EPU. Therefore, there is no increase in the maximum Residual Heat Removal Service Water (RHRSW) system heat load when the RHR heat exchangers operate in the shutdown cooling mode during normal reactor shutdown.

ASME Code Class boundaries and component safety functions within this system are not affected by EPU. Additionally, EPU does not introduce any system parameter changes that will require IST program revisions.

2.2.4.3 Individual Component Evaluations

Valves Affected by Changes in Containment Response

Certain motor operated valves in the program are affected by changes in the containment response, see Section 2.6.1.1. These valves were evaluated and one was found to be affected by EPU conditions. The valve and its effect is presented in Table 2.2-12.

Conclusion

TVA has evaluated the effects of the proposed EPU on safety-related valves. The evaluation addressed the effects of the proposed EPU on its MOV programs related to GL 89-10 and GL 95-07. The evaluation indicates that safety-related valves will continue to meet the requirements of 10 CFR 50.55a(f) and draft GDCs-1, 38, 46, 47, 48, 57, 59, 60, 61, 63, 64, and 65 following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to safety-related valves.

2.2.5 Seismic and Dynamic Qualification of Mechanical and Electrical Equipment

Regulatory Evaluation

Mechanical and electrical equipment covered by this section includes equipment associated with systems that are essential to emergency reactor shutdown, containment isolation, reactor core cooling, and containment and reactor heat removal. Equipment associated with systems essential to preventing significant releases of radioactive materials to the environment are also covered by this section.

The NRC's acceptance criteria are based on (1) GDC-1, insofar as it requires that SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed; (2) GDC-30, insofar as it requires that components that are part of the RCPB be designed, fabricated, erected, and tested to the highest quality standards practical; (3) GDC-2, insofar as it requires that SSCs important to safety be designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions; (4) 10 CFR Part 100, Appendix A, which sets forth the principal seismic and geologic considerations for the evaluation of the suitability of plant design bases established in consideration of the seismic and geologic characteristics of the plant site; (5) GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents; (6) GDC-14, insofar as it requires that the RCPB be designed, fabricated, erected, and tested so as to have an extremely low probability of rapidly propagating fracture; and (7) 10 CFR Part 50, Appendix B, which sets quality assurance requirements for safety-related equipment.

Specific NRC review criteria are contained in SRP Section 3.10.

Browns Ferry Current Licensing Basis

The General Design Criteria (GDC) listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Browns Ferry Nuclear Plant (Browns Ferry) principal design criteria predate these criteria. The Browns Ferry principal design criteria are listed in UFSAR Section 1.5, "Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, the Tennessee Valley Authority (TVA) performed a comparative evaluation of the design basis

of Browns Ferry with the AEC proposed General Design Criteria of 1967. The Browns Ferry UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria set out in the July 1967 AEC release. For each group of criteria, there is a statement of TVA's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a table of references to locations in the Browns Ferry UFSAR where there is subject matter relating to the intent of that particular criteria.

While Browns Ferry is not generally licensed to the final GDC or the 1967 AEC proposed General Design Criteria, a comparison of the final GDC to the applicable AEC proposed General Design Criteria can usually be made. For the final GDC listed in the Regulatory Evaluation above, the Browns Ferry comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Browns Ferry UFSAR Appendix A: draft GDCs-1, 2, 9, 33, 34, 40 and 42.

The Seismic and Dynamic Qualification of Mechanical and Electrical Equipment is described in Browns Ferry UFSAR Section 12.2, "Principal Structures and Foundations."

Technical Evaluation

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

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

For protective (mechanical) devices located in an area designated as a harsh environment and which perform safety-related functions, the change in that environment resulting from a Design Basis Event (DBE) during EPU operation imposes no adverse effects on the performance of the mechanical equipment. The incremental increases in the environmental conditions due to radiation described for EPU operation do not result in reaching the environmental threshold levels where noticeable degradation may occur.

The internal process and external environmental changes associated with EPU will have no detrimental effect on the mechanical equipment's ability to perform safety-related functions in accordance with the original design basis of the plant.

Based on the above, mechanical equipment in Browns Ferry Units 1, 2 and 3, performing safety-related functions in a harsh environment do not experience detrimental effects when operating at 3,952 MWt core thermal power level. The incremental changes in the environmental conditions, mainly radiation, due to EPU operation do not affect the ability of the mechanical equipment to perform their intended safety functions.

The increase in radiation levels experienced by equipment during normal operation and accident conditions is expected to be proportional to the increase in power level. There is only a very small effect on pressure and temperature conditions due to the constant pressure assumption. Operation at EPU conditions increases the temperatures in areas near the FW lines due to the increased feedwater system operating temperature. However, the increases are expected to have little or no effect on the mechanical equipment materials.

The Browns Ferry design and licensing basis does not require a formal Mechanical Equipment Qualification (MEQ) program such as the EQ program for electrical equipment. Browns Ferry uses other existing programs to evaluate the qualification of mechanical components. The key elements are design control, procurement evaluations, testing/preventative maintenance and equipment monitoring. The design control program ensures that mechanical components are specified and procured for the environment in which they are intended to function. Periodic maintenance and testing are performed in accordance with plant and industry operating experience and vendor recommendations to ensure continued functionality.

The mechanical design of equipment/components (e.g., pumps, heat exchangers) in certain systems is affected by operation at EPU due to slightly increased temperatures (< 10%), and in some cases, flows (≤ 15%). However, experience has shown that the uprated operating conditions do not significantly affect the cumulative usage fatigue factors of mechanical components.

EPU effects on fluid induced loads due to postulated RRS pipe breaks (Section 2.6.1.2.1) inside containment remain bounded by the current analysis. The dynamic loading on the safety-related components (RHR containment spray headers and LPCI protection) are not affected by EPU conditions. The margin in the current analysis bounds the slight increase in operating pressure in the RRS due to EPU. The RCPB systems affected by EPU were evaluated within the piping assessments in Section 2.2.2.2.1. The piping systems affected by EPU remain bounded by the current piping analysis. Therefore, the piping and piping supports for the affected systems are adequately designed for EPU conditions.

Conclusion

TVA has evaluated the effects of the proposed EPU on the qualification of mechanical and electrical equipment and addressed the effects of the proposed EPU on this equipment. The evaluation indicates that the equipment will continue to meet the requirements of 10 CFR Part 100, Appendix A; 10 CFR Part 50, Appendix B; and draft GDCs-1, 2, 9, 33, 34, 40 and 42 following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to the qualification of the mechanical and electrical equipment.

Table 2.2-1 High Energy Line Break Outside Containment: Liquid Line Breaks

System or Break Location	EPU Effect on CLTP Mass Flow Release Rate
MS System, RCIC System, HPCI System	Unchanged
MS Line Intermediate Break	The total mass release increased by approximately 11%.
RWCU System	Mass flow rate increased by approximately 4.4% for RWCU breaks.
FW System (Double Ended Break)	The total mass release increased by approximately 12.5%.

Table 2.2-2 Reactor Coolant Pressure Boundary Structural Evaluation

System	Temperature		Pressure		Flow Rate		Mechanical Loading
	CLTP	EPU	CLTP	EPU	CLTP	EPU	
[[							
							]]

Table 2.2-3a Main Steam Pipe Stresses Due to EPU Conditions

Maximum Stress Summary: Unit 1 Line A

Service Level	Equation	Node	CLTP Stress (psi)	EPU Stress (psi)	EPU Allowable (psi)
Service Level A	Eq. 9U	150	Note 1	13,867	18,000
Service Level B	Eq. 9E	150	Note 1	17,110	27,000
Service Level C	Eq. 9E’	150	Note 1	13,867	22,500
Service Level D	Eq. 9E’’	150	Note 1	17,110	30,000
Sustained + Thermal	Eq. 9U+10	5	Note 1	27,040	52,500

Maximum Stress Summary: Unit 1 Line B

Service Level	Equation	Node	CLTP Stress (psi)	EPU Stress (psi)	EPU Allowable (psi)
Service Level A	Eq. 9U	S47	Note 1	16,113	18,000
Service Level B	Eq. 9E	S47	Note 1	25,478	27,000
Service Level C	Eq. 9E’	S47	Note 1	16,113	22,500
Service Level D	Eq. 9E’’	S47	Note 1	25,478	30,000
Sustained + Thermal	Eq. 9U+10	H2	Note 1	37,312	45,000

Maximum Stress Summary: Unit 1 Line C

Service Level	Equation	Node	CLTP Stress (psi)	EPU Stress (psi)	EPU Allowable (psi)
Service Level A	Eq. 9U	J22	Note 1	15,123	18,000
Service Level B	Eq. 9E	J22	Note 1	25,263	27,000
Service Level C	Eq. 9E’	J22	Note 1	15,123	22,500
Service Level D	Eq. 9E’’	J22	Note 1	25,263	30,000
Sustained + Thermal	Eq. 9U+10	252	Note 1	29,237	52,500

Table 2.2-3a Main Steam Pipe Stresses Due to EPU Conditions (continued)

Maximum Stress Summary: Unit 1 Line D

Service Level	Equation	Node	CLTP Stress (psi)	EPU Stress (psi)	EPU Allowable (psi)
Service Level A	Eq. 9U	J01	Note 1	15,819	18,000
Service Level B	Eq. 9E	J01	Note 1	26,001	27,000
Service Level C	Eq. 9E’	J01	Note 1	16,102	22,500
Service Level D	Eq. 9E’’	J01	Note 1	26,284	30,000
Sustained + Thermal	Eq. 9U+10	32	Note 1	24,767	52,500

Maximum Stress Summary: Unit 2 Line A

Service Level	Equation	Node	CLTP Stress (psi)	EPU Stress (psi)	EPU Allowable (psi)
Service Level A	Eq. 9U	150	Note 1	17,926	21,000
Service Level B	Eq. 9E	150	Note 1	20,921	31,500
Service Level C	Eq. 9E’	150	Note 1	19,919	26,250
Service Level D	Eq. 9E’’	150	Note 1	22,914	35,000
Sustained + Thermal	Eq. 9U+10	5	Note 1	33,656	52,500

Maximum Stress Summary: Unit 2 Line B

Service Level	Equation	Node	CLTP Stress (psi)	EPU Stress (psi)	EPU Allowable (psi)
Service Level A	Eq. 9U	H16	Note 1	14,199	18,000
Service Level B	Eq. 9E	H16	Note 1	17,756	27,000
Service Level C	Eq. 9E’	H16	Note 1	15,816	22,500
Service Level D	Eq. 9E’’	H16	Note 1	18,429	30,000
Sustained + Thermal	Eq. 9U+10	H3	Note 1	29,397	45,000

Table 2.2-3a Main Steam Pipe Stresses Due to EPU Conditions (continued)

Maximum Stress Summary: Unit 2 Line C

Service Level	Equation	Node	CLTP Stress (psi)	EPU Stress (psi)	EPU Allowable (psi)
Service Level A	Eq. 9U	RC-1	Note 1	16,010	18,000
Service Level B	Eq. 9E	RC-1	Note 1	16,286	27,000
Service Level C	Eq. 9E'	RC-1	Note 1	17,041	22,500
Service Level D	Eq. 9E''	RC-1	Note 1	17,317	30,000
Sustained + Thermal	Eq. 9U+10	252	Note 1	34,254	52,500

Maximum Stress Summary: Unit 2 Line D

Service Level	Equation	Node	CLTP Stress (psi)	EPU Stress (psi)	EPU Allowable (psi)
Service Level A	Eq. 9U	32	Note 1	18,910	21,000
Service Level B	Eq. 9E	55	Note 1	23,219	31,500
Service Level C	Eq. 9E'	55	Note 1	19,941	26,250
Service Level D	Eq. 9E''	55	Note 1	24,250	35,000
Sustained + Thermal	Eq. 9U+10	32	Note 1	30,971	52,500

Maximum Stress Summary: Unit 3 Line A

Service Level	Equation	Node	CLTP Stress (psi)	EPU Stress (psi)	EPU Allowable (psi)
Service Level A	Eq. 9U	150	Note 1	12,384	21,000
Service Level B	Eq. 9E	150	Note 1	15,110	31,500
Service Level C	Eq. 9E'	150	Note 1	13,703	26,250
Service Level D	Eq. 9E''	150	Note 1	16,429	35,000
Sustained + Thermal	Eq. 9U+10	39, 40	Note 1	28,977	52,500

Table 2.2-3a Main Steam Pipe Stresses Due to EPU Conditions (continued)

Maximum Stress Summary: Unit 3 Line B

Service Level	Equation	Node	CLTP Stress (psi)	EPU Stress (psi)	EPU Allowable (psi)
Service Level A	Eq. 9U	H16	Note 1	16,413	18,000
Service Level B	Eq. 9E	H16	Note 1	17,467	27,000
Service Level C	Eq. 9E'	H16	Note 1	18,070	22,500
Service Level D	Eq. 9E''	H16	Note 1	19,124	30,000
Sustained + Thermal	Eq. 9U+10	CENTR	Note 1	37,120	45,000

Maximum Stress Summary: Unit 3 Line C

Service Level	Equation	Node	CLTP Stress (psi)	EPU Stress (psi)	EPU Allowable (psi)
Service Level A	Eq. 9U	RC-1	Note 1	16,010	18,000
Service Level B	Eq. 9E	RC-1	Note 1	16,286	27,000
Service Level C	Eq. 9E'	RC-1	Note 1	18,558	22,500
Service Level D	Eq. 9E''	RC-1	Note 1	18,835	30,000
Sustained + Thermal	Eq. 9U+10	252	Note 1	34,254	52,500

Maximum Stress Summary: Unit 3 Line D

Service Level	Equation	Node	CLTP Stress (psi)	EPU Stress (psi)	EPU Allowable (psi)
Service Level A	Eq. 9U	55	Note 1	18,943	21,000
Service Level B	Eq. 9E	55	Note 1	23,290	31,500
Service Level C	Eq. 9E'	55	Note 1	21,492	26,250
Service Level D	Eq. 9E''	55	Note 1	25,839	35,000
Sustained + Thermal	Eq. 9U+10	36, 37	Note 1	37,284	52,500

Note:
1. The original EPU stress calculations for the BOP and safety-related (RCPB) piping for Browns Ferry units were completed in the 2002-2003 timeframe. Since completion, piping systems in both safety-related (RCPB) and BOP systems have been modified, using the EPU values. A significant effort to reconstitute CLTP stress values with the current analysis would be required. Given that the existing calculations (with EPU values) have determined that no code of record stress allowables have been exceeded indicates that the piping is acceptable for EPU conditions.

Table 2.2-3b Feedwater Pipe Stresses Due to EPU Conditions

Maximum Stress Summary: Unit 1 Line A

B31.1 Equation	Description	Node Joint	EPU Stress (psi)	Allowable (psi)	Stress Ratio
11	Primary + Secondary (Normal)	55B	35,053	37,500 (=SA+Sh)	0.935
9U+10	Primary + Secondary (Upset)	55BD	42,346	45,000 (=1.2*(SA+Sh))	0.941

Note: The original EPU stress calculations for the BOP and safety-related (RCPB) piping for Browns Ferry units were completed in the 2002-2003 timeframe. Since completion, piping systems in both safety-related (RCPB) and BOP systems have been modified, using the EPU values. A significant effort to reconstitute CLTP stress values with the current analysis would be required. Given that the existing calculations (with EPU values) have determined that no code of record stress allowables have been exceeded indicates that the piping is acceptable for EPU conditions.

Maximum Stress Summary: Unit 1 Line B

B31.1 Equation	Description	Node Joint	EPU Stress (psi)	Allowable (psi)	Stress Ratio
11	Primary + Secondary (Normal)	CENTR	33,226	37,500 (=SA+Sh)	0.886
9U+10	Primary + Secondary (Upset)	47	38,370	45,000 (=1.2*(SA+Sh))	0.853

Note: The original EPU stress calculations for the BOP and safety-related (RCPB) piping for Browns Ferry units were completed in the 2002-2003 timeframe. Since completion, piping systems in both safety-related (RCPB) and BOP systems have been modified, using the EPU values. A significant effort to reconstitute CLTP stress values with the current analysis would be required. Given that the existing calculations (with EPU values) have determined that no code of record stress allowables have been exceeded indicates that the piping is acceptable for EPU conditions.

Table 2.2-3b Feedwater Pipe Stresses Due to EPU Conditions (continued)

Maximum Stress Summary: Unit 2 Line A

B31.1 Equation	Description	Node Joint	EPU Stress (psi)	Allowable (psi)	Stress Ratio
11	Primary + Secondary (Normal)	SB20	29,807	30,000 (=SA+Sh)	0.994
9U+10	Primary + Secondary (Upset)	SB20	35,924	36,000 (=1.2*(SA+Sh))	0.998

Note: The original EPU stress calculations for the BOP and safety-related (RCPB) piping for Browns Ferry units were completed in the 2002-2003 timeframe. Since completion, piping systems in both safety-related (RCPB) and BOP systems have been modified, using the EPU values. A significant effort to reconstitute CLTP stress values with the current analysis would be required. Given that the existing calculations (with EPU values) have determined that no code of record stress allowables have been exceeded indicates that the piping is acceptable for EPU conditions.

Maximum Stress Summary: Unit 2 Line B

B31.1 Equation	Description	Node Joint	EPU Stress (psi)	Allowable (psi)	Stress Ratio
11	Primary + Secondary (Normal)	47	34,181	37,500 (=SA+Sh)	0.911
9U+10	Primary + Secondary (Upset)	47	39,324	45,000 (=1.2*(SA+Sh))	0.874

Note: The original EPU stress calculations for the BOP and safety-related (RCPB) piping for Browns Ferry units were completed in the 2002-2003 timeframe. Since completion, piping systems in both safety-related (RCPB) and BOP systems have been modified, using the EPU values. A significant effort to reconstitute CLTP stress values with the current analysis would be required. Given that the existing calculations (with EPU values) have determined that no code of record stress allowables have been exceeded indicates that the piping is acceptable for EPU conditions.

Table 2.2-3b Feedwater Pipe Stresses Due to EPU Conditions (continued)

Maximum Stress Summary: Unit 3 Line A

B31.1 Equation	Description	Node Joint	EPU Stress (psi)	Allowable (psi)	Stress Ratio
11	Primary + Secondary (Normal)	47, 48	37,463	37,500 (=SA+Sh)	0.999
9U+10	Primary + Secondary (Upset)	48	40,800	45,000 (=1.2*(SA+Sh))	0.907

Note: The original EPU stress calculations for the BOP and safety-related (RCPB) piping for Browns Ferry units were completed in the 2002-2003 timeframe. Since completion, piping systems in both safety-related (RCPB) and BOP systems have been modified, using the EPU values. A significant effort to reconstitute CLTP stress values with the current analysis would be required. Given that the existing calculations (with EPU values) have determined that no code of record stress allowables have been exceeded indicates that the piping is acceptable for EPU conditions.

Maximum Stress Summary: Unit 3 Line B

B31.1 Equation	Description	Node Joint	EPU Stress (psi)	Allowable (psi)	Stress Ratio
11	Primary + Secondary (Normal)	CENTR	31,736	37,500 (=SA+Sh)	0.846
9U+10	Primary + Secondary (Upset)	47	36,186	45,000 (=1.2*(SA+Sh))	0.804

Note: The original EPU stress calculations for the BOP and safety-related (RCPB) piping for Browns Ferry units were completed in the 2002-2003 timeframe. Since completion, piping systems in both safety-related (RCPB) and BOP systems have been modified, using the EPU values. A significant effort to reconstitute CLTP stress values with the current analysis would be required. Given that the existing calculations (with EPU values) have determined that no code of record stress allowables have been exceeded indicates that the piping is acceptable for EPU conditions.

Table 2.2-3c Feedwater Pipe Stresses Due to Feedwater Transient

	Feedwater Piping Inside Containment (Unit 1 Node 55B)
Allowable	27,000 psi
Max EPU Eqn. 9E Stress	25,751 psi
Existing EPU Eqn. 9E Stress Ratio	0.954
Max EPU Eqn. 9E Stress with RFPT Load	26,563 psi
EPU Eqn. 9E Stress Ratio with RFPT Load	0.984

Note: An evaluation was performed which evaluated the effects of a simultaneous three reactor feedwater pump turbine trip. This considered case is bounding and conservative. The evaluation was performed for a representative piping segment for FW piping inside containment and is applicable to Units 1, 2, and 3. The results are for the bounding node with the highest Eqn. 9U/9E/9F/9U+10 stress ratio which was Node 55B of Unit 1 for Eqn. 9E. The FW transient only affects those equation stresses.

Table 2.2-3d Feedwater and Condensate Pipe Stresses Due to Feedwater Transient

	Feedwater Piping Outside Containment (Node 65)	Condensate Piping (Node 200.1)
Allowable	22,500 psi	22,500 psi
Max EPU Stress (1)	17,133 psi	12,217 psi
Existing EPU Max Stress Ratio (1)	0.761	0.543
Max EPU Stress with RFPT Load	17,811 psi	12,274 psi
EPU Stress Ratio with RFPT Load	0.792	0.545

Note:

1. Stress corresponding to the maximum stress ratio of all Code equations. For feedwater piping outside containment and condensate piping, the maximum EPU stress ratio corresponds to Eqn. 10. Although Eqn. 10 is for thermal loadings, increasing the maximum stress ratio and confirming it is less than 1.0 is conservative.

Table 2.2-4a Main Steam System Piping (Outside Containment)

Maximum Stress Interactions for Main Steam Piping Outside Containment						
Service Level		CLTP Stress (psi)	EPU Stress (2) (psi)	Node	Allowable (psi)	Interaction Ratio
Eqn. 8	Sustained	Note 1	6,631	D90	15,000	0.442
Eqn. 9U	Occasional (Upset)	Note 1	17,563	L30A	18,000	0.976
Eqn. 10	Thermal Expansion	Note 1	16,688	B15	22,500	0.742
Eqn. 11	Sustained + Thermal Expansion	Note 1	21,821	B15	37,500	0.582

Note:

1. The original EPU stress calculations for the BOP and safety-related (RCPB) piping for Browns Ferry units were completed in the 2002-2003 time frame. Since completion, piping systems in both safety-related (RCPB) and BOP systems have been modified, using the EPU values. A significant effort to reconstitute CLTP stress values with the current analysis would be required. Given that the existing calculations (with EPU values) have determined that no code of record stress allowables have been exceeded indicates that the piping is acceptable for EPU conditions.

2. EPU stress from MS piping analysis; based on all four loops between the containment penetration anchor and the HP turbine inlet nozzles. Applies to Units 1, 2, and 3.

Table 2.2-4b Feedwater Piping

Maximum Stress for Feedwater Piping Outside Containment (1)						
Criteria Per ANSI B31.1		**Node**	**CLTP Stress (psi)**	**EPU Stress (psi)**	**Allowable (psi)**	**Interaction Ratio**
Eqn. 10	Thermal Expansion	65	Note 2	17,133	22,500	0.761

Notes:

1. Only the Equation 10 stresses increase due to the thermal increases associated with EPU. Applies to Units 1, 2, and 3.

2. The original EPU stress calculations for the BOP and safety-related (RCPB) piping for Browns Ferry units were completed in the 2002-2003 time frame. Since completion, piping systems in both safety-related (RCPB) and BOP systems have been modified, using the EPU values. A significant effort to reconstitute CLTP stress values with the current analysis would be required. Given that the existing calculations (with EPU values) have determined that no code of record stress allowables have been exceeded indicates that the piping is acceptable for EPU conditions.

Maximum pipe stress increase from CLTP analysis**:	
Temperature Expansion	5.4% *
Pressure	0% *
Fluid Transients	N/A *
Maximum pipe support loading increase (due to thermal expansion loading)**:	5.3% *

* The maximum increase in FW temperature range from CLTP to EPU is 5.4%. Pipe stresses remain within code allowables. Maximum increase in FW temperature for piping with rigid supports is 5.3%. There is no fluid transient loading in the current FW piping design basis.

** Bounding value of Units 1, 2, and 3.

Table 2.2-4c Condensate Piping

Maximum pipe stress increase from CLTP analysis**:	
Temperature Expansion	4.7%
Pressure	16.7% *
Fluid Transients	N/A *
Maximum pipe support loading increase (due to thermal expansion loading)**:	4.7%

* Condensate piping design pressure bounds EPU operating pressures. Pipe stress remains within code allowables. There is no fluid transient loading in the current condensate piping design basis.

** Bounding value of Units 1, 2, and 3.

Table 2.2-4d Extraction Steam Piping

Maximum pipe stress increase from CLTP analysis**:	
Temperature Expansion	6.5%
Pressure	13.3% *
Fluid Transients	N/A *
Maximum pipe support loading increase (due to thermal expansion loading)**:	4.6% *

* Extraction steam piping design pressure remains bounding for EPU. The maximum increase in support loading is for piping with rigid supports. There is no fluid transient loading in the current extraction steam piping design basis.

** Bounding value of Units 1, 2, and 3.

Table 2.2-4e FW Heater Drains and Vents Piping

Maximum pipe stress increase from CLTP analysis**:	
Temperature Expansion	9.4%
Pressure	0% *
Fluid Transients	N/A *
Maximum pipe support loading increase (due to thermal expansion loading)**:	0% *

* FW heater drains and vents piping design pressure remains bounding for EPU. Pipe stresses remain within code allowables. There is no fluid transient loading in the current FW heater drains and vents piping design basis. There are no rigid supports for this piping, but increased movement on spring supports is 0.1" or less.

** Bounding value of Units 1, 2, and 3

Table 2.2-4f Moisture Separator Vents and Drains Piping

Maximum pipe stress increase from CLTP analysis**:	
Temperature Expansion	4.24%
Pressure	0% *
Fluid Transients	N/A *
Maximum pipe support loading increase (due to thermal expansion loading)**:	4.24%

* Moisture separator vents and drains piping design pressure remains bounding for EPU. There is no fluid transient loading in the current moisture separator vents and drains piping design basis.

** Bounding value of Units 1, 2, and 3.

Table 2.2-5 BOP Piping System Evaluation

System	Temperature (°F)		Pressure (psig)		Flowrate (Mlb/hr)		Mechanical Loading
	CLTP	EPU	CLTP	EPU	CLTP	EPU	
Condensate Piping (3rd stage heater to RFP)	[[						
FW Piping (from RFP to RPV)							
MS Piping (max conditions at RPV)							
Extraction Steam Piping (High Pressure (HP) turbine exhaust to 1st stage heater)							
FW Heater Drains (1st stage heater to 2nd stage heater)							]]

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

	P + Q Stress[1] (ksi)			CUF				
Component	**CLTP (3,527 MWt)[2]**	**EPU (4,031 MWt)**	**Allowable (ASME Code Limit)**	**CLTP 40 years (3,527 MWt) [2]**	**CLTP 60 years (3,527 MWt)**	**EPU 40 / 60 years (4,031 MWt)**	**EPU with Environmental Fatigue U_{en} 60 years (4,031 MWt)**	**Allowable**
Feedwater Nozzle								
Unit 1:	37.3	57.7	69.9	0.984[3]			0.458/0.248[6]	1.0
Unit 2:	37.3	37.7	69.9	0.984[3]	Note 4	Note 5	0.444/0.240[6]	1.0
Unit 3:	37.3	37.7	69.9	0.984[3]			0.450/0.243[6]	1.0
Recirculation Inlet Nozzle								
Unit 1:	77.1/47.0 [7]	88.73/54.10 [7,17]	58.40	0.425[8]			0.212/0.981[9]	1.0
Unit 2:	77.1/47.0 [7]	88.73/54.10 [7,17]	58.40	0.425[8]	Note 4	Note 5	0.282/0.981[9]	1.0
Unit 3:	77.1/47.0 [7]	88.73/54.10 [7,17]	58.40	0.425[8]			0.279/0.981[9]	1.0

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

	P + Q Stress[1] (ksi)			CUF				
Component	**CLTP (3,527 MWt)[2]**	**EPU (4,031 MWt)**	**Allowable (ASME Code Limit)**	**CLTP (3,527 MWt) [2]**	**CLTP 60 years (3,527 MWt)**	**EPU 40/ 60 years (4,031 MWt)**	**EPU with Environmental Fatigue U_{en} 60 years (4,031 MWt)**	**Allowable**
Recirculation Outlet Nozzle								
Unit 1:	73.20	75.53	80.10	0.779[10]			0.034/0.015[11]	1.0
Unit 2:	73.20	75.53	80.10	0.779[10]	Note 4	Note 5	0.103/0.015[11]	1.0
Unit 3:	73.20	75.53	80.10	0.779[10]			0.102/0.015[11]	1.0
Core Spray Nozzle								
Unit 1:	42.50	44.37	52.46	0.073[10]			0.112/0.237[9]	1.0
Unit 2:	42.50	44.37	52.46	0.073[10]	Note 4	Note 5	0.345/0.237[9]	1.0
Unit 3:	42.50	44.37	52.46	0.073[10]			0.340/0.237[9]	1.0
CRD Hydraulic System Return Nozzle								
Unit 1:	68.0	70.99	80.0	0.363[10]		0.394/0.287 [12]		1.0
Unit 2:	68.0	70.99	80.0	0.363[10]	Note 4	0.363/0.287 [12]	NA	1.0
Unit 3:	68.0	70.99	80.0	0.363[10]		0.363/0.287 [12]		1.0

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

	P + Q Stress[1] (ksi)			CUF				
Component	**CLTP (3,527 MWt)[2]**	**EPU (4,031 MWt)**	**Allowable (ASME Code Limit)**	**CLTP (3,527 MWt) [2]**	**CLTP 60 years (3,527 MWt)**	**EPU 40/ 60 years (4,031 MWt)**	**EPU with Environmental Fatigue U_{en} 60 years (4,031 MWt)**	**Allowable**
2-inch Instrumentation Nozzle								
Unit 1:	80.20/0.03 [13]	83.73/0.048 [13]	69.9/1.0 [13]	0.06[10]				1.0
Unit 2:	80.20/0.03 [13]	83.73/0.048 [13]	69.9/1.0 [13]	0.06[10]	NA	Note 14	NA	1.0
Unit 3:	80.20/0.03 [13]	83.73/0.048 [13]	69.9/1.0 [13]	0.06[10]				1.0
Support Skirt								
Unit 1:	115.9/NA[7]	90.053/51.273 [7,17]	80.10/80.10	0.904		0.114/0.129 [12]		1.0
Unit 2:	115.9/NA[7]	90.053/51.273 [7,17]	80.10/80.10	0.904	Note 4	0.090/0.129 [12]	NA	1.0
Unit 3:	115.9/NA[7]	90.053/51.273 [7,17]	80.10/80.10	0.904		0.090/0.129 [12]		1.0

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

	P + Q Stress[1] (ksi)			CUF				
Component	**CLTP (3,527 MWt)[2]**	**EPU (4,031 MWt)**	**Allowable (ASME Code Limit)**	**CLTP (3,527 MWt) [2]**	**CLTP 60 years (3,527 MWt)**	**EPU 40/ 60 years (4,031 MWt)**	**EPU with Environmental Fatigue U_{en} 60 years (4,031 MWt)**	**Allowable**
Refueling Containment Skirt								
Unit 1:	86.70	87.87	88.00	0.328		0.283/0.304 [12]		1.0
Unit 2:	86.70	87.87	88.00	0.328	Note 4	0.348/0.304 [12]	NA	1.0
Unit 3:	86.70	87.87	88.00	0.328		0.348/0.304 [12]		1.0
Shroud Support								
Unit 1:	136.8/0.170 [13]	142.82/0.263 [13]	69.90/1.0 [13]	0.170				1.0
Unit 2:	136.8/0.170 [13]	142.82/0.263 [13]	69.90/1.0 [13]	0.170	NA	Note 14	NA	1.0
Unit 3:	136.8/0.170 [13]	142.82/0.263 [13]	69.90/1.0 [13]	0.170				1.0

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

	P + Q Stress[1] (ksi)			CUF				
Component	**CLTP (3,527 MWt)[2]**	**EPU (4,031 MWt)**	**Allowable (ASME Code Limit)**	**CLTP (3,527 MWt) [2]**	**CLTP 60 years (3,527 MWt)**	**EPU 40/ 60 years (4,031 MWt)**	**EPU with Environmental Fatigue U_{en} 60 years (4,031 MWt)**	**Allowable**
Main Closure Studs								
Unit 1:	103.3	103.3	110.1	0.762				1.0
Unit 2:	103.3	103.3	110.1	0.762	NA	Note 15	NA	1.0
Unit 3:	103.3	103.3	110.1	0.762				1.0
Vessel Shell								
Unit 1:	39.00	40.72	80.00	0.032			0.003	1.0
Unit 2:	39.00	40.72	80.00	0.032	Note 4	Note 5	0.010	1.0
Unit 3:	39.00	40.72	80.00	0.032			0.010	1.0

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

	P + Q Stress[1] (ksi)			CUF				
Component	**CLTP (3,527 MWt)[2]**	**EPU (4,031 MWt)**	**Allowable (ASME Code Limit)**	**CLTP (3,527 MWt) [2]**	**CLTP 60 years (3,527 MWt)**	**EPU 40/ 60 years (4,031 MWt)**	**EPU with Environmental Fatigue U_{en} 60 years (4,031 MWt)**	**Allowable**
CRD Penetration								
Unit 1:	73.00/ 0.005[13]	76.21/0.006 [13]	70.00/1.0 [13]	0.005				1.0
Unit 2:	114.0/ 0.093[13]	119.02/0.110 [13]	70.00/1.0 [13]	0.093	NA	Note 14	NA	1.0
Unit 3:	114.0/ 0.093[13]	119.02/0.110 [13]	70.00/1.0 [13]	0.093				1.0
Stabilizer Bracket								
Unit 1:	51.20	51.89	80.00	0.170				1.0
Unit 2:	51.20	51.89	80.00	0.170	NA	Note 14	NA	1.0
Unit 3:	51.20	51.89	80.00	0.170				1.0

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

	P + Q Stress[1] (ksi)			CUF				
Component	**CLTP (3,527 MWt)[2]**	**EPU (4,031 MWt)**	**Allowable (ASME Code Limit)**	**CLTP (3,527 MWt) [2]**	**CLTP 60 years (3,527 MWt)**	**EPU 40/ 60 years (4,031 MWt)**	**EPU with Environmental Fatigue U_{en} 60 years (4,031 MWt)**	**Allowable**
Jet Pump Instrumentation Seal								
Unit 1:	38.35	41.18	51.75					
Unit 2:	38.35	41.18	51.75	Note 16	NA	Note 16	NA	Note 16
Unit 3:	44.10	47.35	51.75					

Notes for Table 2.2-6:

1. Stress value for the limiting location of the component.
2. CLTP was conservatively evaluated at 102% of the stretch power uprate level (3,458 * 1.02 = 3,527 MWt).
3. These values are for the nozzle blend radius, which is the bounding location.
4. For 60-year license, EPU values are used.
5. Instead of EPU CUF, the environmentally assisted fatigue usage factors for 60 years are provided.
6. The first value is for the nozzle blend radius and the second value is for the safe end. These are the limiting locations evaluated for the FW nozzle.
7. Thermal bending included/Thermal bending removed.
8. The value is for the safe end which is the limiting location.
9. The first value is for the nozzle blend radius and the second value is for the safe end.
10. The value is for the limiting location of the nozzle.
11. The first value is for the nozzle blend radius and the second value is for the nozzle body cladding.
12. Value reported for 40-year license and 60-year license (40-year value / 60-year value).
13. P + Q value / Elastic-plastic CUF value
14. Fatigue evaluation is not performed as the generic disposition is applied.
15. Fatigue evaluation is not performed as the component specific disposition is applied. The fatigue value CUF < 1.0.
16. Fatigue evaluation is exempted by ASME Code Section III, NB-3222.4(d) 1980 Edition with Addenda to and including Winter 1981 (Units 1 and 2) and 1986 Edition (Unit 3).
17. P + Q without thermal bending is less than the ASME allowable. Therefore it meets ASME requirements. Note that the simplified elastic-plastic analysis is performed for fatigue evaluations according to ASME requirements.

Table 2.2-7 RIPDs for Normal Conditions

Parameter	CLTP [1] (psid)	EPU [2] (psid)
Shroud Support Ring and Lower Shroud	31.06	32.89
Core Plate and Guide Tube	22.84	24.40
Upper Shroud	8.23	8.55
Shroud Head	8.42	9.43
Shroud Head to Reactor Water Level (Irreversible [3])	10.8	12.24
Shroud Head to Reactor Water Level (Elevation [3])	1.07	0.94
Fuel Channel Wall (Core Average Power Bundle)	9.1	10.4
Fuel Channel Wall (Maximum Power Bundle)	11.67	13.31
Top Guide	0.61	0.61
Steam Dryer (OSD / RSD) [4]	0.33	0.42 / 0.42

Notes:

1. At 105% rated core flow with GE13 fuel.
2. At 105% rated core flow with GE13 fuel. The GE13 results are bounding for operation with GE14, ATRIUM-10, and ATRIUM 10XM.
3. Irreversible loss is the loss across the separators; the elevation loss or reversible head loss is the loss between the inside shroud (at the midpoint between the top of fuel and the shroud dome) and the exit of the separators.
4. OSD = Original Steam Dryer. RSD = Replacement Steam Dryer.

Table 2.2-8 RIPDs for Upset Conditions

Parameter	CLTP [1] (psid)	EPU [2] (psid)
Shroud Support Ring and Lower Shroud	33.46	35.29
Core Plate and Guide Tube	25.24	26.80
Upper Shroud	12.34	12.82
Shroud Head	12.63	14.14
Shroud Head to Reactor Water Level (Irreversible [3])	16.20	18.36
Shroud Head to Reactor Water Level (Elevation [3])	1.61	1.41
Fuel Channel Wall (Core Average Power Bundle)	12.0	13.3
Fuel Channel Wall (Maximum Power Bundle)	14.57	16.21
Top Guide	1.10	0.92
Steam Dryer (OSD / RSD) [4]	0.50	0.62 / 0.62

Notes:

1. At 105% rated core flow with GE13 fuel.
2. At 105% rated core flow with GE13 fuel. The GE13 results are bounding for operation with GE14, ATRIUM-10, and ATRIUM 10XM.
3. Irreversible loss is the loss across the separators; the elevation loss or reversible head loss is the loss between the inside shroud (at the midpoint between the top of fuel and the shroud dome) and the exit of the separators.
4. OSD = Original Steam Dryer. RSD = Replacement Steam Dryer.

Table 2.2-9 RIPDs for Faulted Conditions

Parameter	CLTP [1] (psid)	EPU [2] (psid)
Shroud Support Ring and Lower Shroud	52.0	51.0
Core Plate and Guide Tube	30.0	28.5
Upper Shroud	30.0	29.0
Shroud Head	30.0	29.5
Shroud Head to Reactor Water Level (Irreversible [3])	32.0	32.0
Shroud Head to Reactor Water Level (Elevation [3])	2.1	1.4
Fuel Channel Wall (Core Average Power Bundle)	12.9	14.0
Fuel Channel Wall (Maximum Power Bundle)	14.6	15.5
Top Guide	2.8	1.1
Steam Dryer (OSD / RSD) [4 5]	7.7 / 8.0	7.7 / 8.0

Notes:

1. At 105% rated core flow with GE13 fuel. The GE13 results are bounding for operation with GE14, ATRIUM-10, and ATRIUM 10XM.
2. Evaluations at these points considered both normal and reduced FW temperatures. The reduced FW temperature of 55ºF was used for EPU. The GE13 RIPD results are bounding for operation with GE14 fuel.
3. Irreversible loss is the loss across the separators; the elevation loss or reversible head loss is the loss between the inside shroud (at the midpoint between the top of fuel and the shroud dome) and the exit of the separators.
4. OSD = Original Steam Dryer. RSD = Replacement Steam Dryer.
5. Steam dryer loads are limiting at the cavitation interlock condition. The faulted condition steam dryer load is therefore unaffected by EPU implementation.

Table 2.2-10 Governing Stress Results for RPV Internal Components

	Component Location	Category/ Service Condition	Stress/Load Category	CLTP Basis Value	EPU Value	Allowable
1	Shroud	Normal/Upset	Buckling (psi)	2,176	2,176	7,720
		Faulted	Buckling (psi)	10,804	10,804	15,440
2	Shroud Support	Design Operating	Stress (psi)	24,500	30,062	34,950
		Faulted	Stress (psi)	66,000	66,000	69,900
3	Core Plate (including core plate plugs)	Normal/Upset	Buckling/Sliding Delta P (psid)	25.24	26.80	28.0
		Emerg./Fault.	Buckling/Sliding Delta P (psid)	32.0	32.0	42.0
4	Top Guide	Normal/Upset	Longest Beam Stress (psi)	23,754	23,754	25,388
		Emergency	Longest Beam Stress (psi)	33,674	33,674	38,081
		Faulted	Longest Beam Stress (psi)	33,674	33,674	50,775
5	Control Rod Drive Housing	Qualitative assessment. Not affected and remains qualified for EPU.				
6	Control Rod Guide Tube	Upset	Buckling (p/p_C)	0.24	0.255	0.40
		Emergency	Buckling (p/p_C)	0.304	0.304	0.60
		Faulted	Buckling (p/p_C)	0.304	0.304	0.80
7	Orificed Fuel Support	Upset	Stress (psi)	12,413	12,527	15,580
		Faulted	Stress (psi)	23,505	23,505	35,440
8	Fuel Channels	Qualified per Proprietary Fuel Design Basis.				

Table 2.2-10 Governing Stress Results for RPV Internal Components (continued)

	Component Location	Category/ Service Condition	Stress/Load Category	CLTP Basis Value	EPU Value	Allowable
9	Feedwater Sparger	Normal/Upset (Slotted Ring)	(Pm + Pb + Q – Therm. Bending) (psi)	70,800	70,910	76,500
		Normal/Upset (Header Pipe/Tee)	(Pm + Pb) (psi)	5,190	6,990	21,450
		Emergency (Header Pipe/Tee)	(Pm + Pb) (psi)	6,020	7,820	28,600
		Faulted (Header Pipe/Tee)	(Pm + Pb) (psi)	33,690	35,490	42,900
10	Jet Pump (including riser brace attachment repair – Unit-3)	Qualitative assessment. Not affected and remains qualified for EPU.				
11	Core Spray Line and Sparger (includes T-Box and downcomer repairs - Unit-3)	Qualitative assessment. Not affected and remains qualified for EPU.				
12	Access Hole Cover	Normal/Upset	(Pm + Pb) (psi)	6,756	7,093	34,950
		Emergency/ Faulted	Qualitative assessment. Remains qualified for EPU.			
13	Shroud Head and Steam Separator Assembly	Normal/Upset	(Pm + Pb) (psi)	33,993	34,489	34,950
		Emergency	(Pm + Pb) (psi)	31,348	34,671	52,425
		Faulted	(Pm + Pb) (psi)	41,432	41,758	69,900
14	In-Core Housing and Guide Tube	Qualitative assessment. Not affected and remains qualified for EPU.				

Table 2.2-10 Governing Stress Results for RPV Internal Components (continued)

	Component Location	Category/ Service Condition	Stress/Load Category	CLTP Basis Value	EPU Value	Allowable
15	Vessel Head Cooling Spray Nozzle	Qualitative assessment. Not affected and remains qualified for EPU.				
16	Jet Pump Instrument Penetration Seal	Qualitative assessment. Not affected and remains qualified for EPU.				
17	Differential Pressure and Standby Liquid Control Line	Qualitative assessment. Not affected and remains qualified for EPU.				

Table 2.2-11 Systems with Pumps and Valves in the IST Program

System	System Number	IST Pumps	IST Valves	GL 89-10 GL 96-05 Valves	GL 95-07 Valves	System Affected by EPU
Main Steam	001	NA	X	X	NA	X
Feedwater	003	NA	X	NA	NA	X
Heater Drains and Vents	006	NA	X	NA	NA	NA
Boiler Drains and Vents and Blowdown	010	NA	X	NA	NA	NA
Auxiliary Boiler System	012	NA	X	NA	NA	NA
RHR Service Water	023	X	X	X	NA	NA
Raw Water Cooling	024	NA	X	NA	NA	NA
Reactor Water Sampling	043	NA	X	NA	NA	NA
Raw Water Chemical Treatment System	050	NA	X	NA	NA	NA
Standby Liquid Control	063	X	X	NA	NA	
Primary Containment	064	NA	X	NA	NA	X
Emergency Equipment Cooling Water	067	NA	X	NA	NA	NA
Reactor Recirculation	068	NA	X	X	NA	NA
Reactor Water Cleanup	069	NA	X	X	NA	X

Table 2.2-11 Systems with Pumps and Valves in the IST Program (continued)

System	System Number	IST Pumps	IST Valves	GL 89-10 GL 96-05 Valves	GL 95-07 Valves	System Affected by EPU
Reactor Building Closed Cooling Water	070	NA	X	X	NA	NA
RCIC	071	X	X	X	X	NA
HPCI	073	X	X	X	X	NA
RHR	074	X	X	X	X	NA
CS	075	X	X	X	X	NA
Radwaste	077	NA	X	NA	NA	NA
Fuel Pool Cooling	078	NA	X	NA	NA	NA
Control Rod Drive	085	NA	X	NA	NA	NA

Note: Cells with NA indicate that the system has no components in the respective program.

Table 2.2-12 EPU Effects to Browns Ferry Program Valves

Valve ID	Valve Function	Maximum Differential Pressure Change, psi	Ambient Temp Change	EPU Effect
3-FCV-75-53	Core Spray Inboard Injection Valve	--	+ 5°F	No effect on valve capability or margin.

2.3 Electrical Engineering

2.3.1 Environmental Qualification of Electrical Equipment

Regulatory Evaluation

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

The NRC's acceptance criteria for EQ of electrical equipment are based on 10 CFR 50.49, which sets forth requirements for the qualification of electrical equipment important to safety that is located in a harsh environment.

Specific NRC review criteria are contained in SRP Section 3.11.

Browns Ferry Current Licensing Basis

The Browns Ferry program for environmental qualification of electrical equipment is described in Browns Ferry UFSAR Section 8.9, "Safety Systems Independence Criteria and Bases for Electrical Cable Installation."

In addition to the evaluations described in the Browns Ferry UFSAR, Browns Ferry's environmental qualification of electrical equipment was evaluated for license renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Browns Ferry License Renewal Safety Evaluation Report (SER), NUREG-1843, dated April 2006 (Reference 11). The environmental qualification of electrical equipment for license renewal is discussed in NUREG-1843, Sections 2.6.1.4 and 4.4.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of EPUs. Section 10.3.1 of the CLTR addresses the effect of EPU on the Environmental Qualification of Electrical Equipment. The results of this evaluation are described below.

Browns Ferry meets all CLTR dispositions. The topics addressed in this evaluation are:

Topic	CLTR Disposition	Browns Ferry Result
Electrical Equipment	Plant Specific	Meets CLTR Disposition

The CLTR states that the increase in power level increases the radiation levels experienced by equipment during normal operation and accident conditions. Because of the constant pressure assumption, there is only a very small effect on pressure and temperature conditions.

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

For areas affected by EPU operating conditions, associated safety-related electrical equipment was reviewed consistent with the requirements of NUREG-0588 Category II, Division of Operating Reactors (DOR) Guidelines of IE Bulletin No.79-01B or 10 CFR 50.49 (or NUREG-0588 Category I) and NRC Regulatory Guide 1.89 to ensure the existing qualification for the normal and accident conditions expected in the areas where the devices are located remains adequate. The review focused on the effect of environmental changes due to EPU. The DOR Guidelines of IE Bulletin No. 79-01B and 10 CFR 50.49 acceptance criteria were used in making this determination. Margin evaluation complies with the recommendations of IEEE 323-1971 and IEEE 323-1974 or is qualitatively justified based on separate data that establish material capabilities.

The EQ Program equipment qualification basis was evaluated using the expected changes to existing normal and accident radiation doses when operating at the EPU increased reactor power level. Table 2.3-1 summarizes the changes to the EQ environmental parameters due to EPU and changes due to remodeling Standby Gas Treatment Flow in all Reactor Buildings. The normal and post-LOCA radiation dose value changes are based on an EPU scaling factor and applied to plant areas inside and outside primary containment. The EPU scaling factor is for gamma and beta contributors applied over the post-accident time intervals up to 100 days. Once the effect on post-EPU radiation dose value was determined, all equipment was evaluated and found to remain qualified for post-EPU parameters with respect to radiation. This includes equipment with sufficient life to demonstrate radiation qualification through the end of plant life (60 years) or with designated qualified life of less than 60 years. Limited life components are addressed within the Browns Ferry EQ program as warranted.

EQ file updates will be completed as required prior to EPU implementation per TVA procedure NPG-SPP-09.2**,** “Equipment Environmental Qualification (EQ) Program.” Post-EPU EQ compliance is not contingent upon any plant modifications, replacement of equipment, or other compensatory measure.

No new EQ electrical components are being added to the 10 CFR 50.49 program due to EPU.

Inside Primary Containment

EQ for safety-related electrical equipment located inside containment is based on steam line break (SLB) and/or DBA-LOCA conditions along with the temperature, pressure, humidity, and radiation consequences, and includes the normal operating environments expected to exist during plant operation. These changes will occur over the 100-day profile, necessitating the evaluation of the new requirements based on the series of individual component type tests available to the EQ Program. The higher Drywell EPU accident temperature profile has been determined to be

acceptable because there is sufficient positive margin in the CLTP Accident Degradation Evaluation calculations to ensure the equipment would still function as required at the higher temperatures. The current Drywell EQ CLTP temperature profile and revised EPU EQ temperature profile are shown in Figure 2.3-1. These profiles were developed by combining the bounding curves for both the SLB and LOCA DBAs. These profiles are also the worst case EQ enveloping profiles for all plant EQ locations in the primary and secondary containments. The EPU profile exceeds the existing Drywell profile over the entire 100 days. The EPU temperature peak occurs at the same time as the CLTP peak, it is slower to degrade until the RPV depressurizes, and the 100 day final temperature is slightly higher than the CLTP ending temperature. The peak accident pressure will increase from 50.64 psig (65.04 psia) to 50.9 psig (65.3 psia). All components were re-evaluated with respect to the EPU peak pressure of 50.9 psig and are documented in Table 2.3-2, which demonstrates that the component qualification limits bound the postulated EPU accident pressure with sufficient margin. The inside containment normal pressure and normal and accident humidity conditions will not change as a result of EPU. The maximum normal and maximum abnormal temperatures will increase by 0.12°F, which will be rounded to 1°F. This small temperature increase will have a minor effect on thermal qualified life and will be addressed as part of normal EQ maintenance replacement. There are no EQ components in the Wetwell (Torus); therefore, an accident curve is not provided for the Wetwell.

The current radiation levels under normal plant conditions were conservatively evaluated to increase in proportion to the increase in reactor thermal power. The total integrated dose (TID) levels generally will increase by < 16% above CLTP levels inside primary containment, see Table 2.3-4 for comparisons. The total integrated doses (normal plus accident) for EPU conditions were evaluated and determined not to exceed the radiation doses for most of the equipment located inside primary containment. Only some cables and solenoids have a decreased qualified life due to EPU implementation.

Table 2.3-3 provides a comparison of each type of component qualification dose with the EPU EQ TID for relevant plant locations. It was determined that some Drywell cables do not have sufficient radiation qualification to meet or exceed the EPU total integrated dose (normal + accident) for 60 years of operation. These cables, as a result of the EPU dose values, will require maintenance replacement once their accumulated normal dose equals the qualification dose minus the accident dose. The normal EQ maintenance activity will replace those cables which have a limited qualified life due to radiation aging.

The increased radiation doses will result in a reduction of the radiation life for some solenoids located inside primary containment. However, the qualified life based on thermal aging is shorter than that of the radiation life for these solenoids. Therefore, the component qualified life will not be reduced due to the increased radiation doses at EPU. Normal EQ maintenance activities will replace those components which have a limited qualified life due to thermal aging.

Humidity during LOCA events inside primary containment typically reaches saturation (100% RH and condensing) early in the event progression and remains saturated for most if not

all of the analyzed period. Because of this characteristic, humidity is typically not graphed. The Browns Ferry EQ program assumes saturated conditions for the duration of the LOCA/SLB event; therefore EPU has no effect on the EQ qualification to humidity.

Outside Primary Containment

The HELB analysis for the secondary containment evaluates numerous break locations and sizes occurring in the HPCI, RCIC, RWCU and MS/FW systems. The effect of operating at EPU conditions was evaluated based on the mass and energy releases and the analytical bases used in Browns Ferry HELB analyses. It was determined that the RWCU HELB will change peak temperatures in some EQ rooms. However, the increased temperatures at EPU conditions remain bounded by the temperatures used for the equipment qualification. See Table 2.3-5 for the effected EQ room descriptions. Table 2.3-5 provides a listing of the EQ components affected by the increase in RWCU HELB peak accident temperatures and their qualification limit. Figure 2.3-2 depicts the bounding HELB temperature EPU and CLTP profiles, which is for the Steam Tunnel. The Steam Tunnel HELB temperature profile is not affected by EPU.

Unlike the Drywell, the EQ Program does not assume saturated conditions for the duration for the HELB events outside primary containment. The only HELB event which will change due to EPU is the RWCU line break. For reactor building areas where the relative humidity reaches 100% following an RWCU line break, the period for 100% relative humidity conditions will increase to 4 hours. This increased period at 100% relative humidity does not exceed the post-accident conditions for which the affected safety-related electrical equipment is qualified.

The Steam Tunnel (EQ Room 7) bulk temperature at EPU conditions will increase by 0.37°F and is conservatively rounded up to a 1°F increase for the EPU evaluation. This temperature increase has a minor effect on thermal qualified life and is addressed as part of normal EQ maintenance replacement. The pressure and humidity conditions in the Steam Tunnel do not change due to EPU. The pressures, ambient temperatures, and humidity conditions for all other EQ locations outside primary containment remain unchanged by EPU.

The current radiation levels under normal plant conditions were conservatively evaluated to increase in proportion to the increase in reactor thermal power at EPU; see Table 2.3-4 for a listing of the CLTP and EPU values. The TID levels will generally increase by less than 20% above CLTP levels outside primary containment. A few areas will increase greater than 20% due to EPU dose rate increases. The qualification basis for the EQ program equipment was evaluated based on the revised EPU normal and accident radiation dose values, except where EPU component (location)-specific dose values are applied. See Table 2.3-5 for identification of each type of component, along with the EPU EQ TID for relevant plant locations and qualification dose. EQ equipment was evaluated and most of the EQ equipment was found to remain fully qualified for post-EPU parameters with respect to radiation. Some of the components, which were determined to not have sufficient radiation qualification to meet or exceed the EPU total integrated dose (normal + accident) for 60 years of operation, will require replacement. The normal EQ maintenance activity will replace those components which have a limited qualified life due to thermal aging, mechanical aging or radiation aging.

Accident temperature, pressure, and humidity environments used for qualification of equipment outside primary containment result from an MSLB, or other HELBs, whichever is limiting for each plant area. The HELB pressure profiles for CLTP conditions were determined to be bounding for EPU conditions. The peak HELB temperatures at EPU rated thermal power are bounded by the values used for equipment qualification at CLTP conditions.

Conclusion

TVA has evaluated the effects of the proposed EPU on the environmental conditions for the qualification of electrical equipment. The evaluation indicates that the electrical equipment will continue to meet the relevant requirements of 10 CFR 50.49 following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to the EQ of electrical equipment.

2.3.2 Offsite Power System

Regulatory Evaluation

The offsite power system includes two or more physically independent circuits capable of operating independently of the onsite standby power sources.

The NRC's acceptance criteria for offsite power systems are based on GDC-17.

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

Browns Ferry Current Licensing Basis

The Browns Ferry offsite power system is described in Browns Ferry UFSAR Section 8.3, "Transmission System." Final GDC-17 is applicable to Browns Ferry as described in UFSAR Section 8.3.

Browns Ferry's Offsite Power System was evaluated for license renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Browns Ferry License Renewal Safety Evaluation Report (SER), NUREG-1843, dated April 2006 (Reference 11). The offsite power system is discussed in the NUREG-1843, Sections 2.5.1 and 3.6.

Technical Evaluation

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

Browns Ferry meets all CLTR dispositions. The topics addressed in this evaluation are:

Topic	CLTR Disposition	Browns Ferry Result
AC Power (Degraded Voltage)	Plant Specific	Meets CLTR Disposition
AC Power (Normal Operation)	Plant Specific	Meets CLTR Disposition

2.3.2.1 AC Power (Degraded Voltage)

As explicitly stated in Section 6.1 of the CLTR, the increase in thermal power translates to an increased electrical output from the station. Changes in electrical requirements to support normal plant operation are not safety-related. The increased power from the main generator will have no adverse effect on the transmission system's ability to supply loads required for safe shutdown.

The Browns Ferry Unit 1, Unit 2, and Unit 3 main generators are each connected to a set of three single phase main generator step-up transformers. The 500 kV and 161 kV switchyards consist of the buswork, disconnect switches, circuit breakers, and the associated control and protection systems. Browns Ferry has two sources of offsite power from the 500 kV and 161 kV transmission network. The offsite power circuits from the transmission network to the safety-related Division I (4.16 kV shutdown boards A and B) and Division II (4.16 kV shutdown boards C and D) for Units 1 and 2 are as follows:

- From the 500 kV switchyard through Unit Station Service Transformer (USST) 1B to a 4.16 kV unit board. That unit board feeds 4.16 kV shutdown bus 1 or 2, which then feed two of the Unit 1 and 2 4.16 kV shutdown boards (A and B or C and D).
- From the 500 kV switchyard through USST 2B to a 4.16 kV unit board. That unit board feeds 4.16 kV shutdown bus 1 or 2, which then feed two of the Unit 1 and 2 4.16 kV shutdown boards (A and B or C and D).
- From the 161 kV transmission network, through Common Station Service Transformer (CSST) A to Start bus 1A or 1B, to a 4.16 kV unit board, to 4.16 kV shutdown bus 1 or 2, which then feeds two of the Unit 1 and 2 4.16 kV shutdown boards (A and B or C and D).

The offsite power circuits from the transmission network to the safety-related Division I (4.16 kV shutdown boards 3EA and 3EB) and Division II (4.16 kV shutdown boards 3EC and 3ED) for Unit 3 are as follows:

- From the 500 kV switchyard through USST 3B to 4.16 kV unit board 3A and/or 3B. Each unit board feeds two of the Unit 3 4.16 kV shutdown boards (3EA and 3EB or 3EC and 3ED).

- From the 161 kV transmission network, through CSST A to Start bus 1A or 1B, then to a 4.16 kV unit board. That unit board feeds two of the Unit 3 4.16 kV shutdown boards (3EA and 3EB or 3EC and 3ED).
- From the 161 kV transmission network, through CSST B to Start bus 1A or 1B, then to a 4.16 kV unit board. That unit board feeds two of the Unit 3 4.16 kV shutdown boards (3EA and 3EB or 3EC and 3ED).

Multiple Browns Ferry units can utilize the 161 kV offsite power circuit simultaneously. However, once a load from one Browns Ferry unit is connected to a 161 kV offsite power circuit (via the Start buses), Browns Ferry operating procedures require disabling the automatic transfer of selected 4.16 kV unit boards and/or 4.16 kV common boards on the other Browns Ferry units to the 161 kV circuits. This is to prevent overloading of the CSSTs. Upon a loss of the normal 500 kV offsite circuit, the emergency diesel generators would supply the associated safety-related loads in both divisions needed to mitigate the immediate consequences of an accident or analyzed operational transient. Therefore, the 161 kV circuit CSSTs can still be credited as a qualified alternate offsite circuit for multiple units. However, access to the 161 kV circuit will require a delayed manual transfer when operators can manually control the loads on the 4.16 kV Start buses to support long term post-accident or transient recovery and shutdown. This description of the Browns Ferry power distribution system is unchanged by EPU.

The protective relaying schemes are designed to protect the equipment from electrical faults. Electrical ratings and margins associated with major components of the offsite power system are given in Table 2.3-6. The review of the protective relaying for the main generator determined that no changes are required for operation at EPU. The transmission system stability study modeled electrical ratings and margins associated with major components of the offsite power system. The transmission system stability study documents the load flow analysis of the off-site power supply and provides input to site calculations.

A transmission system stability study has been performed, considering the increase in electrical output, to demonstrate conformance to 10 CFR 50 Appendix A, General Design Criteria (GDC) 17. Details of this study are provided in LAR Attachment 43 “Transmission System Stability Evaluation” to the EPU license amendment request. The analysis shows the limiting pre-event outages for the 2015 (CLTP) and 2019 (post-EPU) peak cases. The pre-event outages include the loss of TVA’s largest generating unit, loss of a Browns Ferry unit, and loss of each of the transmission lines within the TVA grid. The loss of 500 kV-161 kV transformer banks bounds the effect to the system of loss of any large loads on the grid. Browns Ferry offsite power is adequate to operate loads required for safe shutdown and will preclude the inadvertent separation from the offsite supply. Therefore, the offsite power at degraded voltage meets all CLTR dispositions.

The grid stability analysis evaluated the effect of the EPU on the off-site power transmission system (i.e., the grid) and the ability to meet the minimum required voltage levels to the on-site power system of each Browns Ferry Unit.

The effect of EPU on the on-site power distribution system, including the degraded voltage protection system, was reviewed separately in plant calculations using the Browns Ferry analytical electrical system computer mode.

Degraded voltage protection is provided for the 4 kV safety-related shutdown boards. Each board has three upper degraded voltage relays and three lower degraded voltage relays that sense each of the three phase-to-phase voltages on the shutdown board potential transformer secondaries. A third set of three relays detect loss of power.

If two of the three upper degraded voltage relays sense a shutdown board voltage above the setpoint (4400 V) for more than five seconds, the time delay relays will actuate and give annunciation. The annunciation will alert the operators to reduce board voltage.

If two of the three loss of voltage relays sense a shutdown board voltage below the setpoint (2870 V) for more than 1.5 seconds, the diesel generator starts. If the condition exists for 5 seconds, the relays will initiate load shedding and 4 kV shutdown board power isolation for diesel generator breaker closure.

If two of the three relays lower degraded voltage relays sense a shutdown board voltage below the setpoint (3920V) for about 0.3 seconds, the time delay relays will begin timing. Due to inaccuracy of the relay, dropout may occur at any voltage between 3940 V and 3899 V. The analysis was performed at the degraded voltage lower boundary of 3900 V to ensure all connected safety-related loads and boards remain within their rated operating voltage ranges. The relays reset when voltage recovers greater than 3962 V. Due to inaccuracy of the relay, reset may occur at any voltage between 3941 V and 3983 V. In the analysis, to ensure the relays reset, the voltage must recover to at least 3983 V. If a degraded voltage exists for approximately 4 seconds, the diesel generator will start. If the relays are actuated and the voltage recovers within 5.95 seconds, the relays will reset and the board will not transfer to the diesels. If a degraded voltage exists for greater than 5.95 seconds, the relays will initiate load shedding and 4 kV shutdown board power isolation for diesel generator breaker closure. To ensure margin, the analysis uses 5.6 seconds for resetting the degraded voltage relays.

The results of the analysis indicate that no changes are required to the degraded voltage setpoints as a result of EPU.

2.3.2.2 AC Power (Normal Operation)

TVA owns both the transmission system and Browns Ferry Units 1, 2, and 3. The Off-site Power System is part of the transmission system. The operation and maintenance of the Off-site Power System is under the control of the Transmission Power Systems organization under TVA. The operation and maintenance of the On-site Power systems are under the control of Browns Ferry. The On-site power system ends at the high side outputs of the main power transformers, the common station service transformers, and the cooling tower transformers.

As explicitly stated in Section 6.1 of the CLTR, the increase in thermal power translates to an increased electrical output from the station. For the off-site power supply, other than the main

generator, the equipment is adequate for operation with the uprated electrical output. Changes in electrical requirements to support normal plant operation are not safety-related.

The existing off-site electrical equipment was determined to not need reinforcements for operation with the uprated electrical output and increased electrical loading. The increased power from the generator will have no adverse effect on the transmission system's ability to supply loads required for safe shutdown.

The review of the transmission system stability study concluded the following:

- A transmission system stability study has been performed, considering the increase in electrical output, to demonstrate conformance to GDC 17 (10 CFR 50 Appendix A). Details of this study are provided in the Attachment 43 "Transmission System Stability Evaluation" to the EPU license amendment request. Browns Ferry offsite power voltages resulting from loss of TVA's largest generating unit, loss of a Browns Ferry unit, and loss of each of the transmission lines within the TVA grid, are adequate to operate loads required for safe shutdown and will preclude the inadvertent separation from the offsite supply.
- The transmission system stability study determined that there was no significant effect to the ability of the grid to supply sufficient shutdown power after the power uprate.
- Reactive load capabilities are stated in Attachment 43 "Transmission System Stability Evaluation" to the EPU license amendment request. In addition, the Interconnect System Impact Study being performed in accordance with the TVA Large Generator Interconnect Procedure will address reactive load in detail.
- The Generator Step-up (GSU) transformer rating is 1,500 MVA. The main generator ratings are 1,330 MVA for Unit 1 and 1,332 MVA for Unit 2 and Unit 3.
- The 500 kV switchyard components (i.e., bus, breakers, switches, transformers, and lines) are adequate for increased generator output associated with EPU.

The Unit 1, Unit 2, and Unit 3 GSU transformers have been replaced and upgraded to support the increase in generator output. The maximum rating of the rewound generator is 1,330 MVA for Unit 1 and 1,332 MVA for Units 2 and 3, which is less than the generator step-up transformers rating of 1,500 MVA @ 65°C. The amount of power the generator sends through the GSU is equal to the generator output minus the house loads (that are tapped off the iso-phase bus through the USST before going through the GSU) and the transformer losses. As a result, under normal operations the transformers have substantial margin.

The Unit 1, Unit 2 and Unit 3 isolated phase bus (IPB) duct work, cooling coils and fans have been modified to increase the continuous current rating to provide for operation at EPU output.

The transmission system stability study did not identify any required upgrades for the 500 kV switchyard components associated with operation at the EPU electrical output. The model used in the study included components in the switchyard, GSU, and the main generator. The components performance are unaffected by operation at EPU conditions during normal operation and meets all CLTR dispositions.

Conclusion

TVA has evaluated the effects of the proposed EPU on the offsite power system. The evaluation indicates that the offsite power system will continue to meet the requirements of the final GDC-17 following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to the offsite power system.

2.3.3 AC Onsite Power System

Regulatory Evaluation

The alternating current (AC) onsite power system includes those standby power sources, distribution systems, and auxiliary supporting systems provided to supply power to safety-related equipment.

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

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

Browns Ferry Current Licensing Basis

The Browns Ferry onsite AC power system is described in Browns Ferry UFSAR Section 8.4, "Normal Auxiliary Power System" and Section 8.5, "Standby AC Power Supply and Distribution." Final GDC-17 is applicable to Browns Ferry as described in UFSAR Section 8.3.

Browns Ferry's Onsite AC Power System was evaluated for license renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Browns Ferry License Renewal Safety Evaluation Report (SER), NUREG-1843, dated April 2006 (Reference 11). The onsite AC power system is discussed in NUREG-1843, Sections 2.5.1 and 3.6.

Technical Evaluation

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

The Browns Ferry AC on-site power distribution system consists of transformers, buses, and switchgear. AC power to the distribution system is provided from the transmission system, Transmission Switchyard, and from onsite Diesel Generators.

The AC onsite power system consists of equipment and systems required to provide AC power to safety-related and non-safety-related loads as long as offsite power is available. This includes 500 kV transformers, 161 kV transformers, 22 kV transformers, 4.16 kV transformers, 4.16 kV switchgears, 480 V transformers, 480 V load centers and motor control centers, 208/120V

distribution panels, and Uninterruptible Power Supply (UPS) systems. The AC onsite power system provides standby power for safety-related unit functions from onsite standby diesel generators.

The AC onsite power distribution system loads were reviewed under both normal and abnormal operating scenarios. In both cases, loads are computed based on equipment nameplate data or brake horsepower (BHP), with conservative demand factors applied. These loads are used as inputs for the computation of load, voltage drop and short circuit current values which were modeled in a commercially available electrical analysis software package. The significant changes in electrical load demand are associated with increasing the size of the condensate pumps and condensate booster pumps to restore hydraulic margin. The Browns Ferry review covered the AC power components with respect to their functional performance as affected by various configurations and loading conditions including full operation and unit trip with LOCA. The Browns Ferry review focused on the additional electric load that would result from the proposed EPU. Sufficient margin is available so that no electrical distribution system modifications are required.

There are no changes to the emergency diesel generator loads or load sequencing for EPU. Therefore the fuel oil requirements do not change and the existing supply is adequate.

Browns Ferry meets all CLTR dispositions. The topics addressed in this evaluation are:

Topic	CLTR Disposition	Browns Ferry Result
AC Power (Degraded Voltage)	Plant Specific	Meets CLTR Disposition
AC Power (Normal Operation)	Plant Specific	Meets CLTR Disposition

2.3.3.1 AC Onsite Power (Degraded Voltage)

As explicitly stated in Section 6.1 of the CLTR, the increase in thermal power translates to an increased electrical output from the station.

Operation at the EPU power level is achieved in both normal and emergency conditions by operating equipment at or below the nameplate ratings. The EPU load flow and voltage drop analyses conservatively assume the transient electrical loading conditions that could exist upon a trip of either a condensate or condensate booster pump (CBP). Table 2.3-7 provides a summary of the loading changes to the onsite power analysis model due to EPU operation. The electrical system can tolerate the CBP overload condition for the short duration of the transient.

Browns Ferry operation at EPU RTP includes upgraded condensate pumps and CBPs that deliver higher head, which improves operating margins. These modifications are complete except for the Unit 3 CBPs, which will be upgraded prior to Unit 3 EPU implementation. Larger reactor

recirculation pump motors have been installed to provide the ability to reliably achieve ICF conditions (105% RCF) at both CLTP and EPU RTP. The larger condensate pump motors (1,250 hp), CBP motors (3,000 hp) and reactor recirculation pump motors (8,657 hp) do not change the conclusions of the current Browns Ferry degraded voltage analysis. The analysis encompasses the safety-related 4.16 kV buses and is independent of voltage profiles for the balance of plant buses.

Therefore, AC onsite power at degraded voltage meets all CLTR dispositions.

2.3.3.2 AC Onsite Power (Normal Operation)

The existing protective relay settings are adequate; coordination is maintained between the pump motor breakers and the 4.16 kV and 480 V switchgear main feeder breakers. The existing protective relay settings for pump motors are based on the motor's nameplate rating. The proposed loading of the buses with the larger condensate pump motors and CBP motors was evaluated and determined to be acceptable with retained margin. Detailed design of the replacement condensate pumps and condensate booster pumps addressed the revised relay settings to maintain coordination and ensure adequate cable sizing.

The analytical electrical system computer model developed for Browns Ferry updated the main power transformer size to reflect the recent change of main power transformers and the proposed changes to the main generators and condensate pumps.

Load flow, voltage drop and short circuit current evaluations were performed to verify the adequacy of the AC on-site power system for the proposed changes. Analyzed EPU BHP loads as discussed above are within the electrical distribution equipment capabilities (i.e., unit station service transformers, common station service transformers, cooling tower transformers and buses). The running and starting voltages for motors are within the acceptable values.

Therefore, AC onsite power during normal operation meets all CLTR dispositions.

<u>Conclusion</u>

TVA has evaluated the effects of the proposed EPU on the AC onsite power system including the effects of the proposed EPU on the system's functional design. The evaluation indicates that the AC onsite power system will continue to meet the requirements of final GDC-17 following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to the AC onsite power system.

2.3.4 DC Onsite Power System

<u>Regulatory Evaluation</u>

The direct current (DC) onsite power system includes the DC power sources and their distribution and auxiliary supporting systems that are provided to supply motive or control power to safety-related equipment.

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

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

Browns Ferry Current Licensing Basis

The General Design Criteria (GDC) listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Browns Ferry Nuclear Plant (Browns Ferry) principal design criteria predate these criteria. The Browns Ferry principal design criteria are listed in UFSAR Section 1.5, "Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, the Tennessee Valley Authority (TVA) performed a comparative evaluation of the design basis of Browns Ferry with the AEC proposed General Design Criteria of 1967. The Browns Ferry UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria set out in the July 1967 AEC release. For each group of criteria, there is a statement of TVA's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a table of references to locations in the Browns Ferry UFSAR where there is subject matter relating to the intent of that particular criteria.

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

The Browns Ferry onsite DC power system is described in Browns Ferry UFSAR Section 8.8, "Auxiliary DC Power Supply and Distribution."

Browns Ferry's onsite DC power system was evaluated for license renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Browns Ferry License Renewal Safety Evaluation Report (SER), NUREG-1843, dated April 2006 (Reference 11). The onsite DC power system was determined to be within the scope of license renewal and the components subject to aging management review are evaluated on a plant wide basis as commodities. The onsite DC power supplies are described in NUREG-1843, Section 2.5.1. The electrical commodity groups are described in NUREG-1843, Section 2.5.1, and aging management for electrical commodities is described in NUREG-1843, Section 3.6.

Technical Evaluation

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

The Browns Ferry direct current (DC) power distribution system provides control and motive power for various systems/components within the plant. The results of the battery sizing calculation for the LOCA/LOOP analysis scenario show that the existing batteries have adequate voltage at the end of the duty cycle. It also shows all required DC devices are within their design voltage range.

Browns Ferry meets all CLTR dispositions. The topics addressed in this evaluation are:

Topic	CLTR Disposition	Browns Ferry Result
DC Power Requirements	Plant Specific	Meets CLTR Disposition

As stated in Section 6.2 of the CLTR, DC loads are not significantly increased as a result of power uprate.

The DC power system provides DC power to instrumentation, controls and motive force power required for equipment required to operate during accident conditions. Equipment includes safety-related switchgear, DC motor operated valves, HPCI turbine auxiliary oil pumps, HPCI gland seal condenser condensate pumps, RCIC gland seal vacuum tank condensate pumps, RCIC gland seal vacuum pumps, emergency lighting, and 120V inverters. These DC loads are not affected by EPU and there are no Class 1E DC Power load changes required for EPU implementation.

Therefore this analysis concludes that the DC power system is adequate to support the EPU power increase.

Conclusion

TVA has evaluated the effects of the proposed EPU on the onsite DC power system and has accounted for the effects of the proposed EPU on the system's functional design. The evaluation indicates that the DC onsite power system will continue to meet the requirements of draft GDCs-24 and 39 following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to the DC onsite power system.

2.3.5 Station Blackout

Regulatory Evaluation

Station blackout (SBO) refers to a complete loss of AC electric power to the essential and nonessential switchgear buses in a nuclear power plant. SBO involves the Loss of Offsite Power (LOOP) concurrent with a turbine trip and failure of the onsite emergency AC power system. SBO does not include the loss of available AC power to buses fed by station batteries through inverters or the loss of power from "alternate AC sources".

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

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

Browns Ferry Current Licensing Basis

The licensing basis for station blackout is described in Browns Ferry UFSAR Section 8.10, "Station Blackout."

Station blackout coping equipment was evaluated for license renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Browns Ferry License Renewal Safety Evaluation Report (SER), NUREG-1843, dated April 2006 (Reference 11). Station blackout is discussed in NUREG-1843, Section 2.1.3. The station blackout coping equipment was determined to be within the scope of license renewal and the components subject to aging management review are evaluated on a plant wide basis as commodities. The electrical commodity groups are described in NUREG-1843, Section 2.5.1, and aging management for electrical commodities is described in NUREG-1843, Section 3.6.

Technical Evaluation

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

Browns Ferry meets all CLTR dispositions. The topic addressed in this evaluation is:

Topic	CLTR Disposition	Browns Ferry Result
Station Blackout	Plant Specific	Meets CLTR Disposition

The CLTR states that the plant responses to and coping abilities for an SBO event are affected slightly by operation at the power uprate level, due to the increase in the decay heat.

SBO was re-evaluated using the guidelines of Nuclear Management and Resources Council (NUMARC) 87-00, "Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors" (Reference 55), and Regulatory Guide 1.155, Station Blackout (Reference 56). [[
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The major characteristics that affect the ability to cope with a SBO event are identified in NUMARC 87-00 Revision 1 as:

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

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

NUMARC 87-00 Revision 1 (Section 7) provides two methods for conducting the assessment. The second method, the Alternate AC Approach, is used in the Browns Ferry SBO assessment. This method uses equipment that is capable of being electrically isolated from the preferred off-site and emergency on-site AC power sources. Alternate AC approach would entail a short period of time in an AC Independent state (up to one hour) while operators initiate power from the backup source. Browns Ferry SBO assessment assumes only one unit in station blackout with the other two units available to supply Alternate AC to the blacked-out unit. Use of Alternate AC power is limited to providing the required cooling systems to certain areas (control room, control bay, and electrical board rooms).

The Alternate AC Approach is the method for calculating the coping period where the plant uses equipment that is capable of being electrically isolated from the preferred off-site and emergency on-site AC power sources.

The four-hour coping duration criteria for Alternate AC Approach plants applies to Browns Ferry. Thus, Browns Ferry must meet the SBO requirements for at least four hours.

Condensate Inventory for Decay Heat Removal

Analyses have shown that the Browns Ferry condensate inventory is adequate to meet the SBO coping requirement for EPU conditions. The current CST inventory reserve (135,000 gallons) for RCIC and HPCI use ensures that adequate water volume is available to remove decay heat, depressurize the reactor and maintain reactor vessel level above the top of active fuel (approximately 114,000 gallons required at EPU conditions) during the coping period.

Class 1E Battery Capacity

There are no changes to the systems and equipment used to respond to a SBO and no change in the required coping time for operation at EPU conditions. The capacity of the existing Unit 1, Unit 2, and Unit 3 Class 1E batteries have adequate capacity and voltage to support the loads required to cope with a SBO event for a period of four hours. The calculated short circuit current is within the allowable limit of all protective devices to support a SBO event for a period of four hours. There are no changes to the design capacities of the Class 1E batteries required for EPU implementation.

Evaluation of the Browns Ferry Class 1E Battery Capacity has shown that Browns Ferry has adequate battery capacity to support decay heat removal during a SBO for the required coping duration. The battery capacity analysis of record is conservative in that it includes an assumption in the model that various HPCI System loads, which are relatively large, operate for long periods during the SBO mitigation sequence. The CLTP mitigation sequence includes a single and relatively short HPCI cycle, and the resulting HPCI loads are bounded by the analysis. EPU does not significantly increase the HPCI loading, and similar to CLTP, only one relatively short HPCI cycle (approximately 7 minutes) is predicted by the EPU containment analysis analytical model (SHEX) model for SBO mitigation. Similarly, the number of required RCIC cycles in the CLTP and EPU mitigation sequence as predicted by the model is well below the RCIC initiations assumed in the analysis of record. Given the above, the battery capacity remains adequate to support operation of the required coping equipment operation after EPU.

Compressed Gas Capacity

The EPU SBO evaluation has shown that the Browns Ferry air operated safety relief valves (MSRVs) required for decay heat removal have sufficient compressed gas capacity for the required automatic and manual operation during the SBO event for EPU conditions. Simulation of SBO at EPU conditions, using the GEH SHEX code (See Table 1-1), indicates 74 total MSRV cycles are required as compared to the compressed gas inventory capable of thousands of cycles. Sufficient capacity remains to perform emergency RPV depressurization in case it is required. Therefore, adequate compressed gas capacity exists to support the MSRV actuations because the maximum number of MSRV valve operations is less than the capacity of the pneumatic supply. Timing of the operator action to cross-tie drywell control air to the containment atmospheric dilution (CAD) system is discussed in Section 2.11.1.2.1.

Effects of Loss of Ventilation

The effect of loss of ventilation in dominant areas of concern containing equipment necessary to achieve and maintain safe shutdown during a station blackout is evaluated for SBO.

Areas containing equipment necessary to cope with an SBO event were evaluated for the effect of loss of ventilation due to an SBO. The evaluation shows that equipment operability is maintained because the SBO environment is milder than the existing design and qualification bases.

These areas for Browns Ferry included:

1. Control Building Rooms
2. Reactor Building Shutdown Board Rooms/Electrical Board Rooms
3. Drywell
4. RCIC Room
5. HPCI Room
6. Main Steam Tunnel
7. Torus Room

The EPU SBO evaluation used the "Alternate AC" power source approach (Section 2.3) and the methodology of Section 2.7, "Effects of the loss of ventilation methodology," of NUMARC 87-00, Revision 1 (Reference 55). The evaluation shows that equipment operability is maintained because the SBO environment is milder than the existing design and qualification bases, as summarized below:

The drywell evaluation using SHEX determined that the maximum EPU temperature compared to the maximum CLTP temperature does not change and is bounded by the existing design and qualification bases.

Outside the drywell, the SBO loss-of-ventilation evaluation for the Control Building Rooms, Reactor Building Shutdown Board Rooms/Electrical Board Rooms, RCIC Room, HPCI Room, Main Steam Tunnel, Reactor Building General Floor Area, and Tours Room determined that, compared to CLTP, equipment operability is maintained because the SBO environment is milder than the existing design and qualification bases.

Containment Isolation

Containment isolation capability is not adversely affected by the SBO event for EPU as the SBO environment conditions do not change significantly after EPU and containment isolation is not adversely affected by the SBO for EPU.

SBO Containment Response Analysis

Key inputs for the Browns Ferry EPU SBO evaluation are contained in Table 2.3-8a. The SBO sequence of events is provided in Table 2.3-8b. The plant response to and coping capabilities for an SBO event are affected slightly by operation at EPU due to the increase in initial power level and decay heat. There are no changes to the systems and equipment used to respond to an SBO event, nor is the required coping time of four hours changed. The SBO event calculations for CLTP and EPU conditions are performed using the NRC-approved SHEX computer program and nominal ANSI/ANS 5.1-1979 decay heat source term at 100% equilibrium power for containment long-term pressure and temperature analysis. The energy contribution from metal-water reaction in the core is not modeled as the core does not uncover during the event and metal-water reaction would not occur (Reference 7).

The battery capacity remains adequate to support RCIC and HPCI operation after EPU. Adequate compressed gas capacity exists to support all SBO mitigation equipment requirements.

The SBO evaluation at EPU conditions shows a need for an additional 13% over CLTP of Condensate Storage Tank water for RCIC and HPCI use to ensure that adequate water volume is available to remove decay heat, depressurize the reactor, and maintain reactor vessel level above the top of active fuel. This increases the total Condensate Storage Tank volume required to approximately 114,000 gallons, which is well within the current Condensate Storage Tank inventory reserve of < 135,000 gallons.

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

Key Containment Parameters Comparison

Parameter	Units	CLTP	EPU	Design Limit
Peak Drywell Pressure	psia	41.4	43.4	< 70.0
Peak SP (Torus) Temperature	ºF	194.1	203.7	< 281.0
Peak Drywell Temperature	°F	276	276	< 281.0

The containment response comparison is based on a scenario that provides conservative containment parameters designed to result in a more severe containment response.

Based on the above evaluations, Browns Ferry continues to meet the requirements of 10 CFR 50.63 after the EPU. Therefore, SBO meets all CLTR dispositions.

<u>Conclusion</u>

TVA has evaluated the effects of the proposed EPU on the plant's ability to cope with and recover from an SBO event for the period of time established in the plant's licensing basis. The evaluation indicates that the plant will continue to meet the requirements of 10 CFR 50.63 following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to SBO.

Table 2.3-1 Summary of EPU Effect on EQ DBA Environmental Parameters

Environmental Parameter	Effect of EPU	
	Inside Primary Containment	Outside Primary Containment
Temperature, Maximum Normal and Maximum Abnormal[1]	1°F increase for all Drywell Elevations.	There will be no change to EQ locations, except for the Steam Tunnel, which will increase by 1°F.
Temperature, Accident[1]	Peak temperature increases to 336.9°F	HELB- There will be no changes in the MS/FW, RCIC and HPCI peaks. Between 5 to 20°F increase in RWCU peaks in Rooms 6B, 8, 9A, 9D, 12, 13, 16, 19 and 20. There will be a decrease in EQ Room 18. (See Table 2.3-4 for Room descriptions). LOCA – All but a few EQ locations will fractionally increase in temperature. The Torus Room and NE Pump Room will increase by 6.3°F and 1.2°F, respectively.
Pressure	Peak pressure increases to 65.3 psia.	There will be no change.
Humidity	No change	RWCU HELB time at 100% Relative Humidity will increase to four hours.
Containment Spray	No change	Not applicable.
Submergence	No change	There will be no change.
Radiation	TID will increase in all areas.	The TID will increase in all areas except the Stack, RWCU Backwash Receiving Tank Room and Cleanup Demineralizer Valve Room, which will decrease. (Note: there are no EQ components in the Stack),

Note:

1. Current component qualification testing bounds the temperature increases.

Table 2.3-2 Evaluation of Pressure Qualification of EQ Components in the Drywell

Component	Qualification Pressure (psig)	Margin[1] (%)
Cable - Rockbestos Company [Type PXJ / PXMJ]	117.2	130
Cable - Eaton Cable [Type MS]	72	41
Cable - Brand Rex [Type PXJ / PXMJ]	113	122
Cable - Okonite Company [Type PX / PXJ / PXMJ]	112	120
Cable - Rockbestos Company [Type PXJ / PXMJ]	107	110
Cable - Rockbestos Company [Type MS]	117.2	130
Cable - Rockbestos Company [Coaxial Cable]	133	161
Cable - Okonite Company [Type PXJ / PXMJ]	112	120
Conduit Seal - CON/AX Corp. ECSA	75	47
Connector - CON/AX Coaxial Connector	80.4	57
Temperature Element - Weed SP611-1A-A-3-C-2-75-4D4-2, 1B1D/612D-1A-C-6-C-17-00	75	47
Special Measure Transmitter - TEC VFMS2273A, 504A, 160-2 and 2273-C2	82	61
Limit Switch - N/AMCO EA-740 (QTR-180)	127	149
MOV - Limitorque AC/IPC SMB-000, 00, 0, 2, and SB-3	105	106
EPA - Conax 7504-10001-03, -04, -05 and 7FO2-10000-01	74.5/80/55	46/57/8
EPA - GE 100 Series (TVA ID Nos. EA and EF)	104/103	104/102
EPA - GE Canister 238X600RH	63	23
Solenoid - ASCO NP Series, 206 Series	113.3	122
Solenoid - Automatic Valve C-5497	65	27
Solenoid - Target Rock 1/2 SMS-S-02-5	68.2	33
Splices - Raychem WCSF-N Series	66	29
Splices - Raychem NPkV, NPKC, NPKP, NPKS, NMCK, NCBK, NESK	132/66	159/29
Raychem WCSF-N Series, NPkV, NPKC, NPKP, NPKS, NMCK, NCBK, NESK	120	135

Note:

1. Margin is calculated based on gauge pressure relative to the EPU peak Drywell pressure of 50.9 psig (65.3 psia).

Table 2.3-3 Evaluation of Radiation Qualification of EQ Components

Component	Rooms (Note 1)	EPU TID [rads] (Note 2)	Qualification Dose [rads]
Cable - American Insulated Wire [Type PXJ / PXMJ]	OC	1.11E+07	1.70E+07
Cable - American Insulated Wire [Type PXJ / PXMJ	OC	1.70E+07	9.01E+07
Cable - Anaconda Power and Control [Type PXMJ]	OC	1.86E+07	1.80E+08
Cable - Rockbestos Company [Type PXJ / PXMJ]	PC/OC	2.15E+08[3]	1.80E+08
Cable - Continental (Anaconda) [Type MS]	OC	1.11E+07	2.22E+07
Cable - Anaconda Cable Co. [Type PJJ]	OC	1.85E+07	2.40E+07
Cable - Brand-Rex Company [Type PN / PNJ]	OC	1.62E+07	2.40E+07
Cable - Brand Rex Company [Type MS]	OC	1.85E+07	1.89E+08
Cable - Eaton Cable [Type MS]	PC/OC	2.29E+08[3]	1.80E+08
Cable - Essex International, Inc. [Type CPJ / CPJJ]	OC	1.62E+07	1.70E+07
Cable - Essex Cable [Type PXMJ]	OC	1.70E+07	1.70E+07
Cable - Essex Group, Inc. [Type PXJ / PXMJ]	OC	1.70E+07	1.85E+08
Cable - General Cable Corporation [Type PNJ]	OC	1.12E+07	2.40E+07
Cable - General Cable Corporation [Type CPSJ]	OC	1.12E+07	1.70E+07
Cable - General Cable Corporation [Type CPJ]	OC	1.12E+07	1.70E+07
Cable - Okonite Company [Type PXJ]	OC	1.85E+07	1.81E+08
Cable - Phelps Dodge Cable [Type CPJ]	OC	1.62E+07	1.70E+07
Cable - Triangle / PWC Inc. [Type PN / PNJ]	OC	4.32E+08[3]	2.40E+07
Cable - Triangle / PWC Inc. [Type CPJ / CPJJ]	OC	1.12E+07	1.70E+07
Cable - Triangle / PWC Inc. [Type CPSJ]	OC	1.12E+07	1.70E+07
Cable - Rockbestos KXL-780, Firewall III [Type MS]	OC	1.11E+07	2.00E+08
Cable - Rome Cable [Type CPJ / CPJJ]	OC	1.62E+07	1.70E+07
Cable - Rome (Cyprus) [Type PJJ]	OC	5.19E+07	6.75E+07
Cable - Simplex Wire and Cable Co.[Type CPJ]	OC	1.12E+07	1.7E+07
Cable - Sumitomo Electric Industries, Ltd. [Type CPJJ]	OC	1.62E+07	2.23E+07
Cable - Tamaqua - Products Cable Corp. [Type PNJ]	OC	1.70E+07	2.4E+07
Cable - Times Wire and Cable [Type MS]	OC	1.11E+07	6.76E+07
Cable - Brand Rex [Type PXJ / PXMJ]	PC/OC	2.29E+08[3]	1.8E+08

Table 2.3-3 Evaluation of Radiation Qualification of EQ Components (continued)

Component	Rooms (Note 1)	EPU TID [rads] (Note 2)	Qualification Dose [rads]
Cable - Simplex Wire & Cable Co.[Type CPSJ]	OC	1.11E+07	1.7E+07
Cable - Triangle / PWC Inc. [Type PJJ]	OC	1.62E+07	5.85E+07
Cable - Okonite Company [Type PX / PXJ / PXMJ]	PC/OC	2.29E+08[3]	1.81E+08
Cable - Rockbestos Company [Type SIS]	OC	1.58E+08	1.8E+08
Cable - Okonite Company [Type EPSJ]	OC	1.57E+08	1.8E+08
Cable - Rockbestos Company [Type PXJ / PXMJ]	PC/OC	2.04E+08[3]	1.80E+08
Cable - Rockbestos Company [Type MS]	PC/OC	2.29E+08[3]	1.80E+08
Cable - Rockbestos Company [Type Coaxial Cable]	PC/OC	2.29E+08[3]	1.84E+08
Cable - Okonite Company [Type PXJ / PXMJ]	PC/OC	1.35E+08	1.81E+08
Cable - Anaconda Cable [Type MS]	OC	1.11E+07	1.80E+08
Cable - ITT Surprenant Cable [Type MS]	OC	7.44E+06	1.80E+08
Cable - American Insulated Wire [Type PXMJ]	OC	1.70E+07	7.5E+07
Cable - Essex Cable [Type PXMJ]	OC	1.70E+07	7.81E+07
Conduit Seal - CON/AX Corp. ECSA	PC/OC	1.82E+08	2.25E+08
Conduit Seal - Rosemount Inc. 353C	OC	1.62E+07	1.11E+08
Conduit Seal - EGS Quick Disconnect	OC	1.85E+07	2.00E+08
Connector - CON/AX Coaxial Connector	PC/OC	2.08E+08	2.25E+08
Handswitch - Cutler-Hammer 1025OT	OC	5.19E+07	8.00E+07
Handswitch - General Electric CR2940	OC	1.62E+07	1.80E+07
Flow Switch - SOR 103AS	OC	9.30E+06	3.30E+07
Flow Switch - SOR 141 Series	OC	9.30E+06	3.00E+07
Flow/Level Switch - Fluid Components Inc. FR72-45A, FR72-4HTR-DLL, FR72-1R	OC	5.19E+07	5.81E+07
Level Switch - Magnetrol 291 Series	OC	3.18E+06	6.76E+06
Level Switch - Magnetrol 402 Series	OC	1.34E+06	1.98E+08
Pressure Switch - SOR 5N/6N/12N	OC	8.48E+06	3.00E+07
Pressure Switch - SOR Test Report 9058-102	OC	1.12E+07	3.30E+07
Temperature Element - Weed SP611-1A-A-3-C-2-75-4D4-2, 1B1D/612D-1A-C-6-C-17-00	PC/OC	2.29E+08	2.73E+08

Table 2.3-3 Evaluation of Radiation Qualification of EQ Components (continued)

Component	Rooms (Note 1)	EPU TID [rads] (Note 2)	Qualification Dose [rads]
Temperature Switch – EGS (Fenwal) 01-170230-090, 01-170020-090	OC	9.12E+06	5.05E+07
Temperature Switch - SOR 201, 205	OC	1.43E+07	3.30E+07
Temperature Switch - SOR Test Report 9058-102	OC	9.30E+06	3.45E+07
Special Measure Transmitter - TEC VFMS2273A, 504A, 160-2 and 2273-C2	PC	1.63E+08	2.22E+08
Limit Switch - N/AMCO EA-740	0C	1.70E+07	2.04E+08
Limit Switch - N/AMCO EA-180	OC	1.12E+07	2.04E+08
Limit Switch - Honeywell/Microswitch OP-AR/OPD-AR/OPD-AR-30	OC	8.95E+06	1.1E+07
Limit Switch - N/AMCO EA-740 (QTR-180)	PC/OC	2.01E+08	2.04E+08
Motors - GE 5K6348XC23A and 5K6336XC198A	OC	1.02E+06	1.46E+07
Motor - Reliance TEFC-XT Type P, Random Wound Motors	OC	9.30E+06	2.2E+08
Motor - Reliance 4160VAC	OC	9.30E+06	2.0 E+08
MOV - Limitorque AC/IPC SMB-000, 00, 0, 2, and SB-3 (MOVs components have various qualification doses and TIDs. Listed in order are Fiberite, Phenolic, Motor Insulation and Wiring and Splices)	PC	1.97E+08 1.93E+08 2.33E+08[3] 1.88E+08[3]	2.27E+08 2.11E+08 2.04E+08 1.80E+08
MOV - Limitorque AC/OPC SMB-000- Thru SMB-5T	OC	1.85E+07	2.04E+08
MOV - Limitorque DC/OPC SMB-000, 00, 0, 2, 3, 4T and SB-0 (MOVs motor and all other components have a separate qualification doses and TIDs.)	OC	1.05E+07(motors)[3] 1.17E+07	1.0E+7 (motors) 2.11 E+08
EPA - Conax 7504-10001-03, -04, -05 and 7FO2-10000-01	PC/OC	1.0E+08	1.0E+08
EPA - GE 100 Series (Penetration Seals and Pigtails have separate qualification doses and TIDs)	PC/OC	4.94E+07 7.70E+07	5.0E+07 1.0E+08
EPA - GE Canister 238X600RH	PC/OC	7.70E+07	8.3E+07
HVAC - Ellis-Watts ACH275.LC39	OC	6.95E+05 8.37E+05	1.36E+07 3.12E+07
Damper Motor Actuators - Raymond Controls Sure 24-10-4	OC	2.87E+06	3.3E+06

Table 2.3-3 Evaluation of Radiation Qualification of EQ Components (continued)

Component	Rooms (Note 1)	EPU TID [rads] (Note 2)	Qualification Dose [rads]
Damper Motor Actuators - Raymond Controls Sure 25-10-4CW	OC	5.25E+06	6.10E+06
Solenoid - VALCOR V526-529-2	OC	1.85E+07	1.69E+08
Solenoid - ASCO NP Series, 206 Series	PC/OC	1.92E+08	2.01E+08
Solenoid - Automatic Valve C-5497	PC/OC	1.36E+08[3]	2.55E+07
Solenoid - Target Rock 81NN, 92Z	OC	1.70E+07	3.35E+07
Solenoid - Target Rock 1/2 SMS-S-02-5	PC	1.45E+08[3]	1.0E+07
Solenoid – AVCO Scram Solenoid Pilot Valve	0C	9.21E+04	2.49E+05
Splices - Raychem WCSF-N Series	PC/OC	2.29E+08[3]	2.0E+08
Splices - Raychem NPkV, NPKC, NPKP, NPKS, NMCK, NCBK, NESK	PC/OC	2.11E+08	2.20E+08
Splices - Raychem NMCK8/NHVT	OC	1.11E+07	5.0E+07
Raychem WCSF-N Series, NPkV, NPKC, NPKP, NPKS, NMCK, NCBK, NESK	PC/OC	2.04E+08[3]	1.951E+08
Raychem Nuclear High Voltage (5/8 kV) Splices	OC	1.12E+07	2.15E+08
Terminal Block - General Electric CR151A, CR151B, EB-5, EB-25	OC	1.57E+08	2.27E+08
Transformer - BBC VPE	OC	6.76E+05	8.7E+05
Pressure Transmitter - Rosemount 1153 B	OC	1.05E+07	2.62E+07
Pressure Transmitter - Rosemount 1153D/1154/115 SERIES	OC	1.79E+07	5.19E+07
Pressure Transmitter - Gould PD3200-100 Series, 400, PDH3200-030	OC	1.07E+06	5.55E+07
Pressure Transmitter – Weed DTN2010	OC	8.49E+06	1.10E+07

Notes:

1) "PC" indicates primary containment. "OC" indicates Outside Primary Containment.

2) The EPU TID is the sum of the normal dose, accident dose and no margin because the Browns Ferry radiation parameters were calculated using methods of Appendix D to NUREG 0588 Revision 1, per Section 1.4 of NUREG 0588 Revision 1, a 10% margin is not required.

3) The component has a limited life of less than 60 years due to EPU radiation and will be replaced periodically as part of the normal EQ maintenance program.

Table 2.3-4 Normal Maximum and Total Radiation Requirements for Rooms at Browns Ferry

Area		Normal Operating Dose			DBA Dose[1] (Gamma + Beta)		Total Integrated Dose[1] (Normal + Accident)		
Room	Description	40 Years	60 years[2]	60 yr +EPU	LOCA	LOCA + EPU	40 Years	60 years	60 yr + EPU
		(RADS)	(RADS)	(RADS)	(RADS)	(RADS)	(RADS)	(RADS)	(RADS)
0	Drywell El. 549.92' to 585.0'	6.2E+07	9.3E+07	1.31E+08	2.47E+09	2.83E+09	2.54E+09	2.57E+09	2.97E+09
0	Drywell El. 585.0' to 617.0'	6.2E+07	9.3E+07	1.31E+08	2.46E+09	2.83E+09	2.53E+09	2.56E+09	2.97E+09
0	Drywell El. 617.0' to 639.0'	6.2E+07	9.3E+07	1.31E+08	2.44E+09	2.80E+09	2.51E+09	2.54E+09	2.94E+09
0	Drywell El. 639.0' and above	6.2E+07	9.3E+07	1.31E+08	2.44E+09	2.80E+09	2.51E+09	2.54E+09	2.94E+09
00	Wetwell	2.34E+06	3.51E+06	4.01E+06	2.43E+09	2.80E+09	2.44E+09	2.44E+09	2.81E+09
1	RX. Bldg. EL. 519.0' HPCI Room	1.4E+04	2.1E+04	2.52E+04	1.67E+06	2.21E+06	1.69E+06	1.70E+06	2.24E+06
2	RX. Bldg. EL. 519.0' Southwest Pump Room	1.8E+05	2.7E+05	3.15E+05	8.07E+06	8.91E+06	8.25E+06	8.34E+06	9.23E+06
3	RX. Bldg. EL. 519.0' Northwest Pump Room	1.5E+04	2.25E+04	2.70E+04	8.07E+06	8.91E+06	8.09E+06	8.10E+06	8.94E+06
4	RX. Bldg. EL. 519.0' Northeast Pump Room	3.9E+04	5.85E+04	6.93E+04	8.07E+06	8.91E+06	8.11E+06	8.13E+06	8.98E+06

Table 2.3-4 Normal Maximum and Total Radiation Requirements for Rooms at Browns Ferry (continued)

Area		Normal Operating Dose			DBA Dose[1] (Gamma + Beta)		Total Integrated Dose[1] (Normal + Accident)		
Room	Description	40 Years	60 years[2]	60 yr +EPU	LOCA	LOCA + EPU	40 Years	60 years	60 yr + EPU
5	RX. Bldg. EL. 519.0’ Southeast Pump Room	2.1E+05	3.15+05	3.78E+05	8.07E+06	8.91E+06	8.28E+06	8.39E+06	9.29E+06
6A-D	RX. Bldg. EL. 519.0’ Torus Room	7.01E+05	1.06E+06	1.26E+6	1.33E+7	1.50E+07	1.40E+07	1.44E+07	1.63E+07
7	RX. Bldg. EL. 565.0’ Main Steam Tunnel	8.1E+06	1.22E+07	1.21E+07	5.37E+06	6.35E+06	1.35E+07	1.76E+07	1.85E+07
8	RX. Bldg. EL. 565.0’ General Floor Area	7.0E+05	1.05E+06	1.26E+06	8.77E+06	9.79E+06	9.47E+06	9.82E+06	1.11E+07
9A	RX. Bldg. EL. 593.0’ RHR Heat Exchanger Rooms	4.2E+05	6.30E+05	7.58E+05	5.97E+06	6.68E+06	6.39E+06	6.60E+06	7.44E+06
9B	RX. Bldg. EL. 593.0’ General Area (Southwest Quadrant)	4.21E+05	6.32E+05	1.80E+06	5.97E+06	6.68E+06	6.40E+06	6.61E+06	7.48E+06
9C	RX. Bldg. EL. 593.0’ General Area (Northwest Quadrant)	4.2E+05	6.30E+05	7.58E+05	5.97E+06	6.68E+06	6.39E+06	6.60E+06	7.44E+06

Table 2.3-4 Normal Maximum and Total Radiation Requirements for Rooms at Browns Ferry (continued)

Area		Normal Operating Dose			DBA Dose[1] (Gamma + Beta)		Total Integrated Dose[1] (Normal + Accident)		
Room	**Description**	**40 Years**	**60 years[2]**	**60 yr +EPU**	**LOCA**	**LOCA + EPU**	**40 Years**	**60 years**	**60 yr + EPU**
5	RX. Bldg. EL. 519.0' Southeast Pump Room	2.1E+05	3.15+05	3.78E+05	8.07E+06	8.91E+06	8.28E+06	8.39E+06	9.29E+06
9D	RX. Bldg. EL. 593.0' General Area (Northeast Quadrant)	4.2E+05	6.30E+05	7.58E+05	5.97E+06	6.68E+06	6.39E+06	6.60E+06	7.47E+06
9E	RX. Bldg. EL. 593.0' General Area (Southeast Quadrant)	4.21E+05	6.32E+05	1.80E+06	5.97E+06	6.68E+06	6.40E+06	6.61E+06	8.48E+06
10	RX. Bldg. EL. 593.0' RWCU Pump Rooms	9.2E+05	1.38E+06	1.58E+06	2.89E+05	6.60E+05	1.21E+06	1.67E+06	2.24E+06
11	RX. Bldg. EL. 593.0' Nonregenerative Heat Exchanger Room	6.1E+06	9.15E+06	1.04E+07	7.9E+05	1.22E+06	6.89E+06	9.94E+06	1.17E+07
12	RX. Bldg. EL. 621.25' General Floor Area	4.2E+05	6.3E+05	7.56E+05	3.47E+06	3.99E+06	3.89E+06	4.10E+06	4.75E+06

Table 2.3-4 Normal Maximum and Total Radiation Requirements for Rooms at Browns Ferry (continued)

Area		Normal Operating Dose			DBA Dose[1] (Gamma + Beta)		Total Integrated Dose[1] (Normal + Accident)		
Room	Description	40 Years	60 years[2]	60 yr +EPU	LOCA	LOCA + EPU	40 Years	60 years	60 yr + EPU
13	RX. Bldg. EL. 639.0' General Floor (South) Area	2.6E+04	3.9E+04	4.73E+04	2.89E+05	6.60E+05	3.15E+05	3.28E+05	7.08E+05
14	RX. Bldg. EL. 639.0' General Floor (North) Area	7.0E+04	1.05E+05	1.26E+05	2.89E+05	6.60E+05	3.59E+06	3.94E+05	7.86E+05
15	RX. Bldg. EL. 664.0' Refueling Floor	1.1E+04	1.65E+04	2.23E+04	2.89E+05	6.60E+05	3.00E+06	3.06E+05	6.83E+05
16	RX. Bldg. EL. 593' RWCU Backwash Receiving Tank Room	3.01E+08	4.52E+08	4.31E+08	2.89E+05	6.60E+05	3.02E+08	4.53E+08	4.32E+08
17A and B	RX. Bldg. EL. 639.0' RWCU Demineralizers A and B	Note 3	Note 3	Note 3	2.89E+05	6.60E+05	2.89E+05	2.89E+05	6.60E+05
18	RX. Bldg. EL. 621.25' Cleanup Demineralizer Valve Room	3.0E+08	4.50E+08	4.50E+08	2.89E+05	6.60E+05	3.01E+08	4.51E+08	4.51E+08

Table 2.3-4 Normal Maximum and Total Radiation Requirements for Rooms at Browns Ferry (continued)

Area		Normal Operating Dose			DBA Dose[1] (Gamma + Beta)		Total Integrated Dose[1] (Normal + Accident)		
Room	**Description**	**40 Years**	**60 years[2]**	**60 yr +EPU**	**LOCA**	**LOCA + EPU**	**40 Years**	**60 years**	**60 yr + EPU**
19	RX. Bldg. EL. 565.0' Drywell Access Area	7.4E+04	1.11E+05	1.26E+05	1.63E+07	1.84E+07	1.64E+07	1.65E+07	1.86E+07
20	RX. Bldg. EL. 565.0' Traversing In-core Probe (TIP) Room	1.2E+07	1.80E+07	2.02E+07	4.8E+05	8.70E+05	1.25E+07	1.85E+07	2.11E+07
22	RX. Bldg. EL. 565.0' SGTS Building General Spaces	1.0E+03	1.50E+03	1.50E+03	1.43E+08	1.57E+08	1.44E+08	1.44E+08	1.58E+08
23	Stack Area	8.8E+03	1.32E+04	1.65E+04	8.45E+05	7.14E+05	8.54E+05	8.58E+05	7.31E+05

Notes:

General Note - Unit 2 values used as representative for Units 1, 2, and 3. All three Drywells have the same source term and the Drywell doses are the same due to being identical in terms of operating power, Reactor Pressure Vessels and sacrificial shield concrete density. The measured normal dose rates in the Unit 2 and 3 portions of the Reactor Building and Standby Gas Treatment Building are the same and Unit 1 was confirmed to be less than Units 2 and 3. The Reactor Building gamma accident doses are the same for all three Units and the Beta accident doses are slightly higher on Unit 1 for some areas, but insignificant to the normal plus accident gamma doses.

1. The CLTP doses do not reflect the changes made to the EPU doses as a result of flow changes for SGTS.

2. 60 year normal dose calculated by multiplying the 40 year dose by 1.5.

3. Normal dose not calculated because there is no class 1E equipment located in this room.

Table 2.3-5 RWCU LOCA/HELB Temperature Evaluation Outside Containment

Room Number[2]	Room Description	Component	Qualification Limit (°F)[1]	Peak EPU Accident Temperature (°F)
6B	RX. Bldg. EL. 519.0 Torus Room	MOV –DC Limitorque Outside Primary Containment	340°F	135°F
8	RX. Bldg. EL. 565.0 General Floor Area	Limit switches (N/AMCO, Honeywell)	308°F	175°F
8	RX. Bldg. EL. 565.0 General Floor Area	Penetrations (Conax, GE)	340°F	175°F
8	RX. Bldg. EL. 565.0 General Floor Area	Level Switches (SOR. FCI)	203°F	180°F
8	RX. Bldg. EL. 565.0 General Floor Area	Connector (Conax)	445°F	180°F
9D	RX. Bldg. EL. 593.0 General Area (Northwest Quadrant)	MOV – AC Limitorque Outside Primary Containment	250°F	180°F
9D	RX. Bldg. EL. 593.0 General Area (Northwest Quadrant)	Raychem Splices	358°F	180°F
9D	RX. Bldg. EL. 593.0 General Area (Northwest Quadrant)	Transmitters (Rosemount, Weed, Gould)	180°F	180°F
9D	RX. Bldg. EL. 593.0 General Area (Northwest Quadrant)	Solenoids (Valcor, ASCO, AVCO, Target Rock)	535°F	439°F (includes heat rise)
9D	RX. Bldg. EL. 593.0 General Area (Northwest Quadrant)	Temperature Elements (Weed, Fenwal, SOR)	350°F	180°F
9D	RX. Bldg. EL. 593.0 General Area (Northwest Quadrant)	Conduit Seals (Conax, Rosemount, EGS)	375°F	180°F
9D	RX. Bldg. EL. 593.0 General Area (Northwest Quadrant)	Hand Switches (Cutler-Hammer, GE)	330°F	180°F
9D	RX. Bldg. EL. 593.0 General Area (Northwest Quadrant)	Pressure Switches (SOR)	227°F	180°F
9D	RX. Bldg. EL. 593.0 General Area (Northwest Quadrant)	Terminal Block (GE)	350°F	180°F
9D	RX. Bldg. EL. 593.0 General Area (Northwest Quadrant)	HVAC (Ellis-Watts)	215°F	180°F

Table 2.3-5 RWCU LOCA/HELB Temperature Evaluation Outside Containment (continued)

Room Number[2]	Room Description	Component	Qualification Limit (°F)[1]	Peak EPU Accident Temperature (°F)
9D	RX. Bldg. EL. 593.0 General Area (Northwest Quadrant)	Flow Switch (SOR)	325°F	180°F
12	RX. Bldg. EL. 621.25 General Floor Area	Transformer (Brown Boveri)	447°F	380°F (includes heat rise)
16	RX. Bldg. EL. 593 RWCU Backwash Receiving Tank Room [Unit 2]	Cable –Various Vendors	205°F[3]	215°F (for ~60 seconds)

Notes:

1. Lowest qualification peak LOCA/HELB temperature for the component type.
2. Worst case EQ Location (Room)
3. DOR cable is worst case limited to 205°F long term peak temperature but may exceed 205°F for approximately 10 minutes as long it is under a maximum temperature of 300°F (TVA Letter to the NRC Dated 05/11/1989 (Docket no. 50-250)).

Table 2.3-6 Offsite Electrical Equipment Ratings and Margins

Component	Component Rating	CLTP Duty	CLTP Margin (%)	EPU Duty	EPU Margin (%)
Main Generator (MVA Capability /power factor)	1,330/0.95 U1 1,332/0.93 U2 1,332/0.93 U3	1,280/0.90 U1 1,280/0.90 U2 1,280/0.90 U3	3.8 U1 3.9 U2 3.9 U3	1,330/0.95 U1 1,332/0.93 U2 1,332/0.93 U3	0.0 0.0 0.0
Isolated Phase Bus Maximum Continuous Current (Amps)	36,740 U1 36,796 U2 (Note 1) 36,796 U3	35,359 U1 35,359 U2 35,359 U3	3.8 U1 3.9 U2 3.9 U3	36,740 U1 36,796 U2 36,796 U3	0.0 0.0 0.0
Main Generator Step-Up Transformers (MVA)	1,500 U1 1,500 U2 1,500 U3	1,150 U1 1,150 U2 1,150 U3 (Note 2)	23.3 U1 23.3 U2 23.3 U3	1,280 U1 1,282 U2 1,282 U3 (Note 3)	14.7 14.5 14.5
Unit Station Service Transformers (MVA)	24/32/40 MVA OA/FA/FOA @55°C			27.34 U1 27.61 U2 22.94 U3 (Note 4)	31.65 30.98 42.65
Unit Station Service Transformers (MVA)	24/32 MVA OA/FA @55°C			19.91 U1 18.20 U2 20.41 U3 (Note 4)	37.78 43.13 36.22
Common Station Service Transformers (A and B) (MVA)	21.9/29.2/36.5 MVA OA/FA/FOA @55°C			36.15 36.15 (Note 5)	0.96 0.96

Notes:

1. The Unit 2 isophase bus forced cooling system has the capability to remove the heat generated from 36,796 amps, which is based on an original rating of 36,740 amps with 45,000 scfm of cooling flow, and testing which demonstrated an increased cooling capability based on a cooling flow of 55,000 scfm with two fans operating.
2. Based on maximum historical transformer loading data.
3. Based on maximum generator output minus auxiliary loads of 50MVA.
4. Based on normal loading.
5. Based on maximum shutdown loading.

Table 2.3-7 Electrical Distribution System Load Changes

Motor Description	Nameplate hp	Required BHP EPU	Max. Analyzed BHP EPU
Condensate Pumps (U1)	1,250	1,025 (1)	1,212.5 (2)
Condensate Pumps (U2)	1,250	1,025 (1)	1,212.5 (2)
Condensate Pumps (U3)	1,250	1,025 (1)	1,212.5 (2)
Condensate Booster Pumps (U1)	3,000	2,470.5 (1)	3,720 (2)
Condensate Booster Pumps (U2)	3,000	2,470.5 (1)	3,720 (2)
Condensate Booster Pumps (U3)	3,000	2,470.5 (1)	3,720 (2)
Reactor Recirculation Pumps (U1)	8,657	8,657 (3)	8,657 (3)
Reactor Recirculation Pumps (U2)	8,657	8,657 (3)	8,657 (3)
Reactor Recirculation Pumps (U3)	8,657	8,657 (3)	8,657 (3)

Notes:

1. Normal operation at EPU RTP with three condensate and three condensate booster pumps in service.
2. Maximum transient load assuming either a trip of one condensate pump or one condensate booster pump.
3. Electrical analyses at EPU conditions assume reactor recirculation pump motors operate at nameplate hp.

Table 2.3-8a Key Inputs for Browns Ferry Station Blackout

Parameter[1]	Value
Initial Reactor Power	3,952 MWt
Initial Reactor Pressure	1,055 psia
Decay Heat	ANS/ANSI 5.1 1979 standard consistent with recommendations of GEH SIL 636
Initial Suppression Pool Temperature	95°F
Initial Suppression Pool Volume Low Water Level (LWL)	122,940 ft^3
Initial Wetwell Pressure	14.4 psia
Initial Drywell Temperature	150°F
Initial Drywell Pressure	15.5 psia
Initial Drywell free airspace volume	171,000 ft^3
Initial Wetwell free airspace volume	135,000 ft^3
Initial wetwell (WW) airspace temperature	95°F
CST Water Temperature	130°F
CST Inventory	135,000 gallons available
Initial Drywell Relative Humidity	20%
Initial Wetwell Relative Humidity	100%
RHR Heat exchanger K factor (per heat exchanger)	265 BTU/Sec-°F
RHR pump flow rate (per pump)	6500 GPM
RHR service water flow rate to RHR heat exchangers	4000 gpm
RHR service water temperature	95°F
Leakage rate from primary containment	2% of containment air mass per day
Containment heat sinks modeled	Yes

(1) RPV volume, related masses, and wetwell to drywell vacuum breakers are provided in Table 2.6-2a.
(2) Containment heat sinks are modeled in the SBO evaluation using the EPU values shown in Table 2.6-6a.

Table 2.3-8b Browns Ferry Station Blackout Sequence of Events

Browns Ferry Station Blackout Sequence of Events for EPU	
Time (sec)	**Description**
~0	Loss of Offsite Power Reactor scram MSIV start to close Loss of Feedwater RCIC available to maintain reactor water level HPCI available to maintain reactor water level
~4.0	MSIV closed
~5	FW flow stops
~10 to 256	MSRVs open (relief mode)
267	Begin HPCI Injection (high drywell pressure)
641	End HPCI Injection (Level 8)
1,200	Begin manual MSRV operation for RPV pressure control
2,683	Begin RCIC Injection (Level 2)
7,320	Manual MSRV cooldown complete.
11,194	End RCIC Injection (Level 8)
14,400	Offsite power restored (end of coping period) Containment cooling initiated with 2 RHR pumps/2 RHR heat exchangers in SP cooling mode (RHR flow of 6500gpm/pump; RHR heat exchanger K-factor of 265 BTU/Sec-°F)

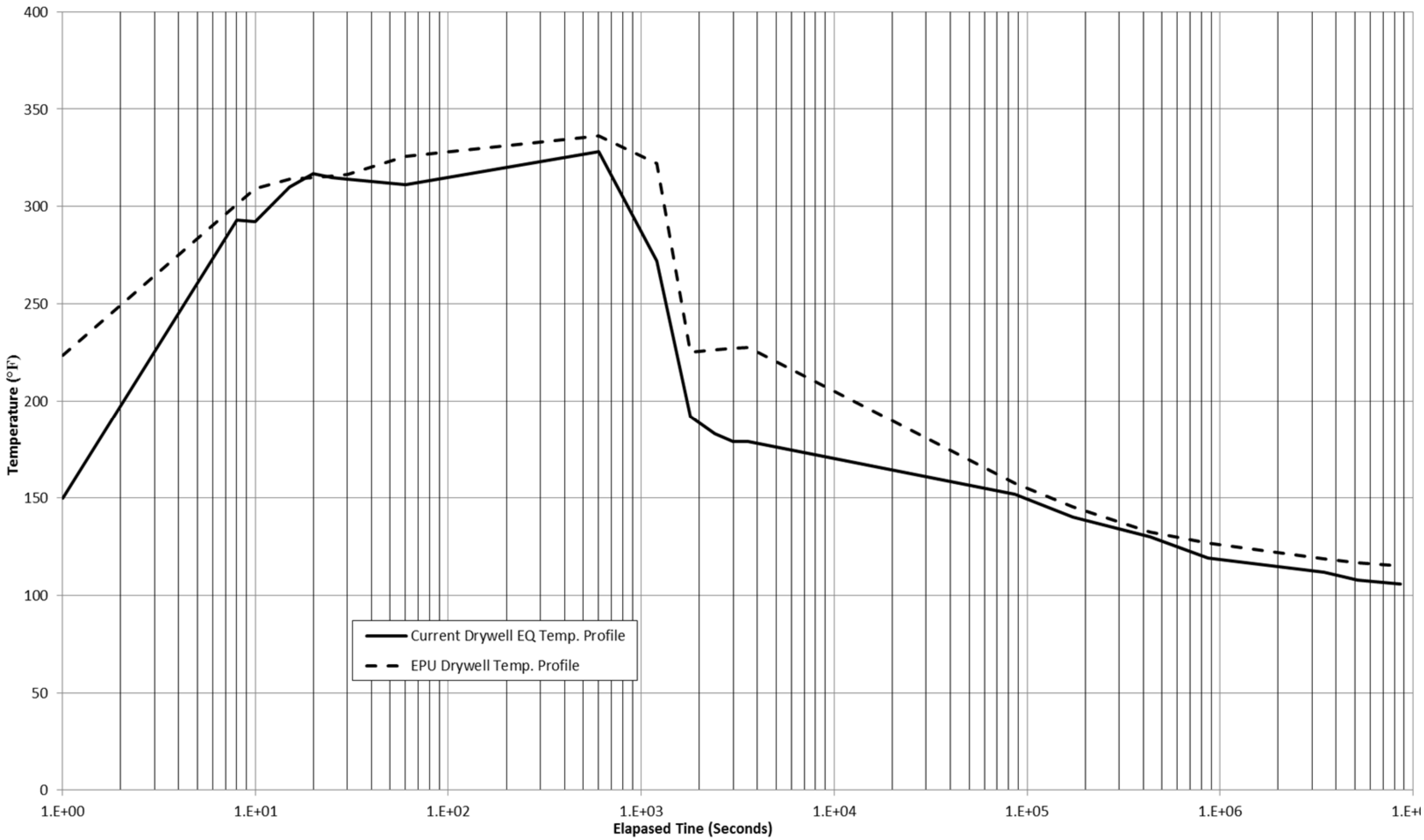

Figure 2.3-1 Worst Case Drywell Temperature Profile

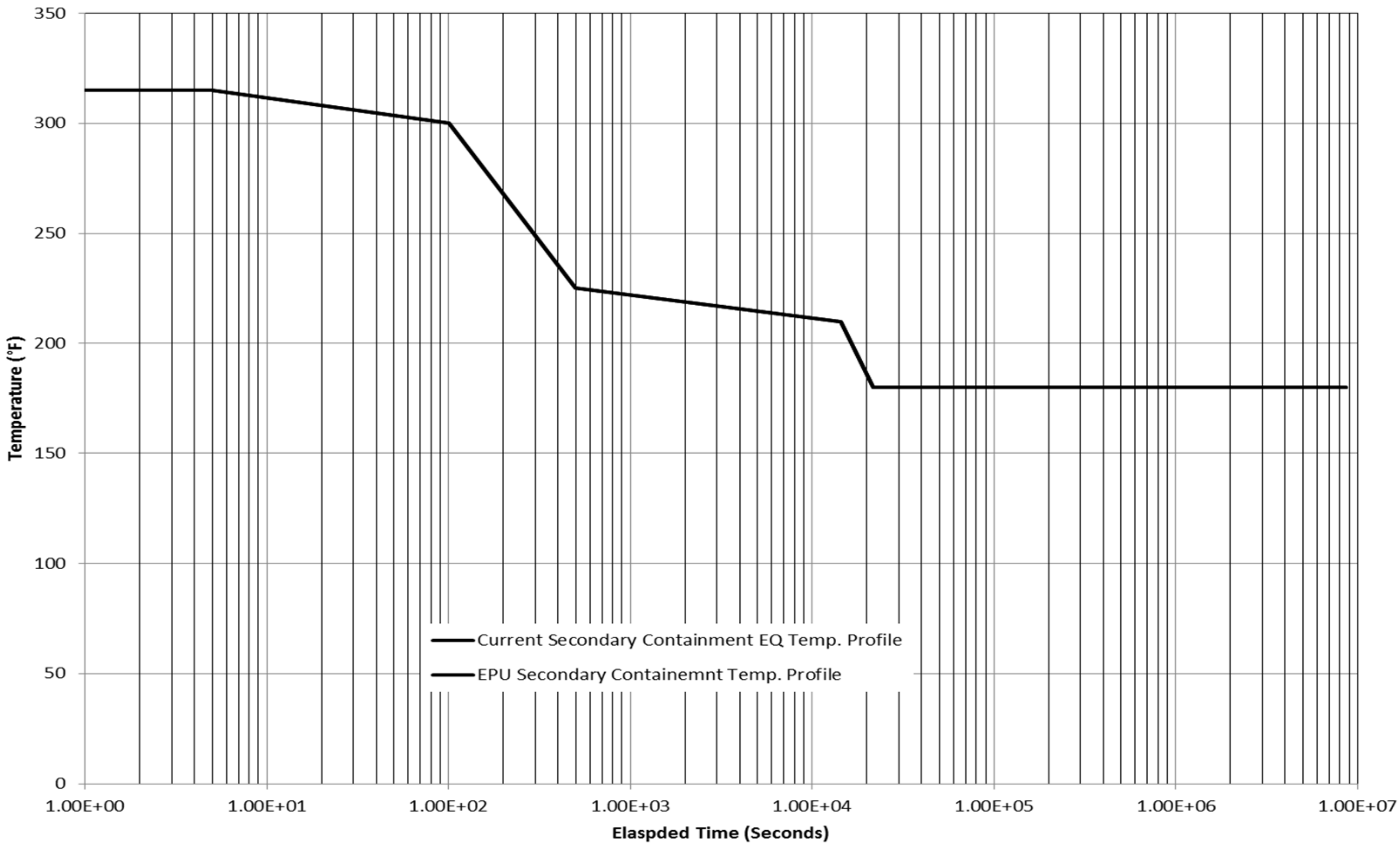

Figure 2.3-2 Worst Case Secondary Containment EQ Temperature Profile

(Note the bounding CLTP and EPU temperature profiles are the same)

2.4 Instrumentation and Controls

2.4.1 Reactor Protection, Safety Features Actuation, and Control Systems

Regulatory Evaluation

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

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

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

Browns Ferry Current Licensing Basis

The General Design Criteria (GDC) listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Browns Ferry Nuclear Plant (Browns Ferry) principal design criteria predate these criteria. The Browns Ferry principal design criteria are listed in UFSAR Section 1.5, "Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, the Tennessee Valley Authority (TVA) performed a comparative evaluation of the design basis of Browns Ferry with the AEC proposed General Design Criteria of 1967. The Browns Ferry UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria set out in the July 1967 AEC release. For each group of criteria, there is a statement of TVA's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a table of references to locations in the Browns Ferry UFSAR where there is subject matter relating to the intent of that particular criteria.

While Browns Ferry is not generally licensed to the final GDC or the 1967 AEC proposed General Design Criteria, a comparison of the final GDC to the applicable AEC proposed General Design Criteria can usually be made. For the final GDC listed in the Regulatory Evaluation above, the Browns Ferry comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Browns Ferry UFSAR Appendix A: draft GDCs-1, 12, 13, 14, 15, 19, 20, 22, 23, 25, 26, 40, and 42. Final GDC-19 is applicable to Browns Ferry.

Browns Ferry instrumentation and control systems are described in Browns Ferry UFSAR Section 7, "Control and Instrumentation."

Browns Ferry's instrumentation and control systems were evaluated for license renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Browns Ferry License Renewal Safety Evaluation Report (SER), NUREG-1843, dated April 2006 (Reference 11). The instrumentation and control systems were determined to be within the scope of license renewal and the components subject to aging management review are evaluated on a plant wide basis as commodities. The electrical commodity groups are described in NUREG-1843, Section 2.5, and aging management for electrical commodities is described in NUREG-1843, Section 3.6.

Technical Evaluation

The setpoint calculation methodology, safety limit-related Limiting Safety System Setting (LSSS) determination, and instrument setpoint controls are discussed in this section.

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

2.4.1.1 Nuclear Steam Supply System Monitoring and Control Instrumentation

As stated in Section 5.1 of the CLTR, the instruments and controls used to monitor and directly interact with or control reactor parameters are usually within the NSSS. Changes in process variables and their effects on instrument performance and setpoints were evaluated for EPU operation to determine any related changes. Process variable changes are implemented through changes in normal plant operating procedures. TSs address instrument AVs and/or setpoints for those NSSS sensed variables that initiate protective actions. The effects of EPU on TS instrument functions are addressed in Section 2.4.1.3.

The EPU affects the performance of the Neutron Monitoring System. These performance effects are associated with the Average Power Range Monitors (APRMs), Local Power Range Monitors (LPRMs), Intermediate Range Monitors (IRMs), and Source Range Monitors (SRMs).

Browns Ferry meets all CLTR dispositions. The topics addressed in this evaluation are:

Topic	CLTR Disposition	Browns Ferry Result
APRM, IRM, SRM	Generic	Meets CLTR Disposition
Local Power Range Monitors	Generic	Meets CLTR Disposition
Rod Block Monitor	Generic	Meets CLTR Disposition
Rod Worth Minimizer	Generic	Meets CLTR Disposition

2.4.1.1.1 Average Power Range, Intermediate Range and Source Range Monitors

The CLTR states that at rated power, the increase in power level increases the average flux in the core and at the in-core detectors.

The Average Power Range Monitor (APRM power signals are calibrated to read 100% at the new licensed power (i.e., EPU RTP). The Intermediate Range Monitors (IRM)s provide full overlap with the APRMs.

The APRM, IRM, and source range monitor (SRM) systems installed at Browns Ferry are in accordance with the requirements established by the GEH design specifications. The specifications provide confirmation that the APRM, IRM, and SRM systems meet all CLTR dispositions.

2.4.1.1.2 Local Power Range Monitors

The CLTR states that at rated power, the increase in power level increases the flux at the LPRMs.

The average flux experienced by the detectors increases due to the average power increase in the core. The maximum flux experienced by an Local Power Range Monitor (LPRM) remains approximately the same because the peak bundle power does not increase.

Due to the increase in neutron flux experienced by the LPRMs and traversing incore probes (TIPs), the neutronic life of the LPRM detectors may be reduced and radiation levels of the TIPs may be increased. LPRMs are designed as replaceable components. The LPRM accuracy at the increased flux is within specified limits, and LPRM lifetime is an operational consideration that is handled by routine replacement. TIPs are stored in shielded rooms. A small increase in radiation levels is accommodated by the radiation protection program for normal plant operation.

Reliability of LPRM instrumentation and accurate prediction of in-bundle pin powers typically requires operation with bypass voids lower than 5% at nominal conditions. LAR Attachment 34 concludes that bypass voiding does not exceed 5% for any LPRMs.

The LPRMs installed at Browns Ferry are in accordance with the requirements established by the GEH design specifications. The specifications provide confirmation that the LPRMs meet all CLTR dispositions.

2.4.1.1.3 Rod Block Monitor

The CLTR states that the increase in power level at the same APRM reference level results in increased flux at the LPRMs that are used as inputs to the rod block monitor (RBM).

The RBM instrumentation is referenced to an APRM channel. Because the APRM has been rescaled, there is only a small effect on the RBM performance due to the LPRM performance at the higher average local flux. The change in performance does not have a significant effect on the overall RBM performance.

The RBMs installed at Browns Ferry are in accordance with the requirements established by the GEH design specifications. The specifications provide confirmation that the RBMs meet all CLTR dispositions.

2.4.1.1.4 Rod Worth Minimizer

The assessment of the RWM is provided in FUSAR Section 2.4.1.1.4.

2.4.1.2 BOP Monitoring and Control

As stated in Section 5.2 of the CLTR, operation of the plant at EPU conditions has minimal effect on the BOP system instrumentation and control devices. Based on EPU operating conditions for the power conversion and auxiliary systems, most process control valves and instrumentation have sufficient range/adjustment capability for use at the EPU conditions. However, some (non-safety) modifications may be needed to the power conversion systems to obtain EPU RTP.

Browns Ferry meets all CLTR dispositions. The topics considered in this section are:

Topic	CLTR Disposition	Browns Ferry Result
Pressure Control System (PCS)	Generic	Meets CLTR Disposition
Turbine Steam Bypass System (Normal Operation)	Generic	Meets CLTR Disposition
Turbine Steam Bypass System (Safety Analysis)	Generic	Meets CLTR Disposition

Topic	CLTR Disposition	Browns Ferry Result
FW Control System (Normal Operation)	Generic	Meets CLTR Disposition
FW Control System (Safety Analysis)	Generic	Meets CLTR Disposition
Leak Detection System (LDS)	Generic	Meets CLTR Disposition

2.4.1.2.1 Pressure Control System

The CLTR states that the increase in power level increases the steam flow to the turbine.

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

The PCS at Browns Ferry meets all CLTR dispositions.

2.4.1.2.2 Turbine Steam Bypass System

The CLTR states that the bypass system capacity, in terms of mass flow, is not changed for EPU. As a result, the increase in power level and resulting increase in steam flow to the turbine effectively reduces the bypass system capacity in terms of percent steam flow. The turbine bypass system is not essential for turbine operation and is not credited in any limiting events analyses as discussed in Section 2.8.5.

The Turbine Steam Bypass System is a normal operating system that is used to bypass excessive steam flow. This system is non-safety-related. The flow capacity of the bypass system, 3.5 Mlbm/hr, is not changed. [[

]] The AOO events are discussed further in Section 2.8.5.

The Turbine Steam Bypass System at Browns Ferry meets all CLTR dispositions.

2.4.1.2.3 Feedwater Control System

The CLTR states that the increase in power results in an increase in FW flow.

The FW Control System is a normally operating system to control and maintain the reactor vessel water level. EPU results in an increase in FW flow. FW control operational testing is

included in the EPU implementation plan as described in Section 2.12 to ensure that the FW response is acceptable. Failure of this system is evaluated in the reload analysis for each reload core with the FW controller failure-maximum demand event. An Loss of Feedwater (LOFW) event can be caused by downscale failure of the controls. The LOFW is discussed in Section 2.8.

The FW Control System at Browns Ferry meets all CLTR dispositions [[

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2.4.1.2.4 Leak Detection System

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

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- **Main Steam Tunnel in the Reactor Building and in the Turbine Building:** The increased FW temperature results in a negligible increase (< 1°F) in the MS tunnel temperature. The LDS temperature and differential temperature setpoints remain unchanged for EPU. As a result, the MS tunnel temperature setpoint is conservative because it slightly increases leak detection sensitivity and is not changed.
- **Drywell:** The normal operating drywell area temperature experiences a negligible change for EPU conditions; therefore, the DW LDS is not affected.
- **RWCU:** There is no significant change to the RWCU system temperature and pressure and no change to the RWCU system flow; therefore, the RWCU LDS is not affected.
- **RCIC:** There is no increase in the system temperature, pressure, or flow; therefore, the RCIC LDS is not affected.
- **RHR Shutdown Cooling Mode:** There is no increase in the RHR Shutdown Cooling Mode temperature or pressure; therefore, the RHR system LDS is not affected.
- **HPCI**: There is no increase in the system temperature, pressure, or flow; therefore, the HPCI LDS is not affected.

The flow-based LDS is not affected by EPU, with the exception of MSL high flow. MSL high flow is discussed in Section 2.4.1.3.

The LDS at Browns Ferry meets all CLTR dispositions [[

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2.4.1.3 Technical Specification Instrument Setpoints

As stated in Section 5.3 of the CLTR, Allowable Values (AVs) and/or Nominal Trip Setpoints (setpoints) are those sensed variables which initiate protective actions and are generally associated with the safety analysis. AVs are highly dependent on the results of the safety

analysis. The safety analysis generally establishes the ALs. The AVs and other instrument setpoints include consideration of measurement uncertainties and are derived from the ALs. The settings are selected with sufficient margin to minimize inadvertent initiation of the protective action, while assuring that adequate operating margin is maintained between the system settings and the actual limits. There is typically substantial margin in the safety analysis process that should be considered in establishing the setpoint process used to establish the Technical Specification AVs and other setpoints.

Increases in the core thermal power and steam flow affect some instrument setpoints. These setpoints are adjusted to maintain comparable differences between system settings and actual limits, and reviewed to ensure that adequate operational flexibility and necessary safety functions are maintained at the EPU RTP level. Where the power increase results in new instruments being employed, an appropriate setpoint calculation is performed and TS and/or Technical Requirement Manual (TRM) changes are implemented, as required. [[

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Browns Ferry has elected not to use the simplified methodology and has applied the existing GE methodology, Reference 57, for the APRM and RBM setpoint functions and the existing TVA methodology, Reference 58, for all other setpoint functions to the Technical Specification instrument setpoints. The TVA setpoint methodology (Reference 58) has been accepted by the NRC for setpoint calculations as referenced in the NRC safety evaluation report for Browns Ferry technical specification change, TS-453 (Reference 59). The GE setpoint methodology (Reference 57) used in the setpoint calculations for the neutron monitoring system functions affected by EPU (e.g., APRM and RBM) was also used during the initial licensing and subsequent NRC approval for installation of the power range neutron monitoring (PRNM) system for each respective Browns Ferry unit. All Technical Specification instruments were evaluated for effects from EPU. This evaluation included a review of environmental (i.e., radiation and temperature) effects, process (i.e., measured parameter) effects and analytical (i.e., AL and margins) effects on the subject instruments.

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

The setpoint calculation methodology for the Browns Ferry EPU is not per the generic disposition of the CLTR because TVA has elected not to use the simplified methodology stated in the CLTR. The setpoint value for each topic addressed in this section meet all CLTR dispositions. The topics considered in this section are:

Topic	CLTR Disposition	Browns Ferry Result
Main Steam Line High Flow Isolation – Setpoint Calculation Methodology	Generic	Full setpoint calculation performed with Reference 58 methodology.
Main Steam Line High Flow Isolation – Setpoint Value	Plant Specific	Meets CLTR Disposition
Turbine First-Stage Pressure Scram and Recirculation Pump Trip Bypass - Setpoint Calculation Methodology	Generic	Full setpoint calculation performed with Reference 58 methodology due to HP turbine replacement.
Turbine First-Stage Pressure Scram and Recirculation Pump Trip Bypass - Setpoint Value	Plant Specific	AL revised using guidelines of Section F.4.2.3 of Reference 4 (ELTR1).
APRM Flow Biased Scram - Setpoint Calculation Methodology	Generic	Full setpoint calculation performed with Reference 57 methodology.
APRM Flow Biased Scram – Setpoint Value	Plant Specific	Meets CLTR Disposition
Rod Worth Minimizer Low Power Setpoint - Setpoint Calculation Methodology	Generic	Full setpoint calculation performed with Reference 58 methodology.
Rod Worth Minimizer Low Power Setpoint – Setpoint Value	Plant Specific	Meets CLTR Disposition
Rod Block Monitor	Generic	Meets CLTR Disposition

Topic	CLTR Disposition	Browns Ferry Result
APRM Setdown in Startup Mode – Setpoint Calculation Methodology	Generic	Full setpoint calculation performed with Reference 57 methodology.
APRM Setdown in Startup Mode – Setpoint Value	Plant Specific	Meets CLTR Disposition

2.4.1.3.1 Main Steam Line High Flow Isolation

The CLTR states that the effect on the Main Steam Line High Flow Isolation due to EPU is increased reactor power level and steam flow.

The MSL high flow isolation setpoint is used to initiate the isolation of the Group 1 primary containment isolation valves. The only safety analysis event that credits this trip is the MSLB accident. For this accident, there are diverse trips from high area temperature and high area differential temperature in the main steam tunnel. For Browns Ferry, there is sufficient margin to choke flow, so the AL for EPU is unchanged from the current percent of rated steam flow (144% rated steam flow) in each MSL. No new instrumentation is required (the existing instrumentation has the required upper range limit and calibration span the instrument loops need to accommodate the new setpoint). A new setpoint was calculated using the Reference 60 methodology and an AV change is required to change the differential pressure at the allowable steam flow.

The MSL AL to choke flow margin calculation was performed in accordance with the methodology specified in GEH Services Information Letter (SIL) No. 438 Revision 2 (Reference 61) and communicated to utilities in GEH 10 CFR Part 21 Safety Communication (SC) 12-18 Revision 2 (Reference 62). The Browns Ferry plant-specific EPU MSL flow element choke flow to MSL high flow AL margin evaluation incorporates the resolution of the GEH 10 CFR Part 21 issue (Reference 42).

Therefore, the Main Steam Line High Flow Isolation setpoint meets all CLTR plant-specific dispositions.

2.4.1.3.2 Turbine First-Stage Pressure Scram and Recirculation Pump Trip Bypass

The CLTR states that the effect on the turbine first stage pressure (TFSP) Scram and RPT Bypass Permissive due to EPU is increased reactor power level and a potential change to TFSP. EPU results in an increased power level, and the HP turbine modifications result in a change to the relationship of TFSP to reactor power level. The TFSP setpoint is used to reduce scrams and RPTs at low power levels where the turbine bypass system (TBS) is effective for TTs and generator load rejections. In the safety analysis, this trip bypass only applies to events at low power levels that result in a turbine trip (TT) or load rejection. Based on the guidelines in

Section F.4.2.3 of Reference 4 (ELTR1), the TFSP Scram and RPT Bypass Permissive AL in percent RTP is reduced from 30% at CLTP to 26% at EPU. [[

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]] Therefore, a new setpoint is calculated using the TVA methodology per Reference 60, and [[ ]] The AV (in psig) for Browns Ferry is revised prior to EPU implementation.

To assure that the new value is appropriate, an EPU plant ascension startup test or normal plant surveillance is performed to validate that the actual plant interlock is cleared consistent with the safety analysis. EPU startup testing is described in Section 2.12.

Therefore, the TFSP Scram and RPT Bypass Permissive meet all CLTR plant specific dispositions.

2.4.1.3.3 APRM Flow Biased Simulated Thermal Power – High Scram

This function is referred to in the Browns Ferry TSs as the APRM Flow-Biased Simulated Thermal Power (STP) – High function. The CLTR states that the effect on the APRM Flow Biased Scram due to EPU is increased reactor power level. APRM Simulated Thermal Power – High function provides protection against transients where Thermal Power increases slowly and protects the fuel cladding integrity by ensuring that the minimum critical power ratio (MCPR) safety limit is not exceeded. This operating limit for the operating domain is established to provide a pre-emptive scram and to prevent a gross violation of the licensed domain.

The Browns Ferry AL for this function is being revised based on the methodology outlined in the CLTR. Therefore, a new setpoint was calculated using the TVA methodology per Reference 58, and [[ ]] The AV (in %RTP) for Browns Ferry will be revised prior to EPU implementation. The clamped AL will retain its value in percent power. Therefore, APRM Flow-Biased Scram at Browns Ferry meets all of the CLTR dispositions.

2.4.1.3.4 Rod Worth Minimizer Low Power Setpoint

The AL in terms of percent RTP does not change, and [[

]] Browns Ferry Technical Specifications do not define an AV for this setpoint function.

The CLTR states that the effect on the RWM Low Power Setpoint (LPSP) due to EPU is increased reactor power level and increased FW flow. For the RWM LPSP instrument function at Browns Ferry, the measurement parameter is main steam and feedwater flow.

The RWM LPSP is used to bypass the rod pattern constraints established for the Control Rod Drop Accident (CRDA) at greater than a pre-established low power level.

Therefore, the RWM LPSP calculation methodology meets all CLTR dispositions.

The LPSP AL is maintained at the same value in terms of percent power (10% RTP) and the EPU has been evaluated on this basis. Below this setpoint, only banked position mode withdrawals or insertions are allowed. Therefore, the RWM LPSP meets all CLTR plant specific dispositions.

The LPSP measurement parameter is main steam flow and feedwater flow. The existing steam flow and feed flow measurement transmitters have sufficient range for EPU rated steam flow and feed flow. A new setpoint in terms of rated steam flow and feedwater flow was calculated using the Reference 58 methodology.

2.4.1.3.5 Rod Block Monitor

The generic disposition of the Rod Block Monitor in the CLTR states that the effect on the Rod Block Monitor due to EPU is increased reactor power level.

Consistent with the generic disposition discussed above, the severity of a rod withdrawal error (RWE) during power operation event is dependent upon the RBM rod block setpoint. This setpoint is only applicable to the control rod withdrawal error. [[

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2.4.1.3.6 APRM Setdown in Startup Mode

This function is referred to in the Browns Ferry TSs as the APRM Neutron Flux – High, Setdown function. The CLTR states that the effect on the APRM Setdown in Startup Mode due to EPU is a reduced TS safety limit for reduced pressure or low core flow conditions.

No specific safety analyses take direct credit for this function. It indirectly ensures that reactor power does not exceed 23% RTP before the Mode Switch is placed in "RUN." The APRM setdown in the startup mode provides margin to the safety limit. The value for the TS safety limit for reduced pressure or low core flow condition is established to satisfy the fuel thermal limits monitoring requirements.

The Browns Ferry AL for this function will change due to the EPU based on the methodology outlined in the CLTR. Therefore, a new setpoint was calculated using the TVA methodology per Reference 58, and the TS applicable condition for fuel thermal limits monitoring requirements in % RTP has been changed. The AV (in % RTP) for Browns Ferry will be revised prior to EPU

implementation. Therefore, APRM Setdown in Startup Mode at Browns Ferry meets the CLTR disposition.

2.4.1.3.7 Main Steam Line Low Pressure Isolation in the Run Mode

The PCS (see Section 2.4.1.2.1) pressure setpoint does not change in a power uprate. However, the steam line pressure near the turbine, where this sensor is located, is expected to change. The margin assessment confirmed that the remaining margin at EPU conditions will not impose any new constraints on the performance of surveillances with the existing setpoint. The margin assessment is performed as an operational screening check to ascertain the potential of normal turbine surveillances (individual stop and control valve full stroke closure at power) to cause pressure drops that could actuate the trip instrumentation. GE SIL 130 provides the criterion applied. SIL 130 provided a basis for reducing the margin down to a minimum of 100 psi to avoid spurious steam line isolations in the event of plant scrams. The margin for the projected uprate conditions is 125.3 psid. The MSL low pressure isolation AL, AV and nominal trip setpoint (NTSP) are unchanged for EPU.

2.4.1.4 Changes to Instrumentation and Controls

In the CLTR SER, the staff requested that the plant-specific submittal address all EPU-related changes to instrumentation, such as scaling changes, changes to upgrade obsolescent instruments, and changes to the control philosophy. Table 2.4-2 provides this information.

The instrument modifications described in Table 2.4-2 that have not been completed to date will be completed prior to EPU operation.

<u>Conclusion</u>

TVA has evaluated the effects of the proposed EPU on the functional design of the reactor trip system, safe shutdown system, and control systems. The evaluation indicates that Browns Ferry will continue to meet the requirements of 10 CFR 50.55a(a)(1), 10 CFR 50.55a(h), final GDC-19, and draft GDCs-1, 12, 13, 14, 15, 19, 20, 22, 23, 25, 26, 40, and 42. Therefore, the proposed EPU is acceptable with respect to instrumentation and controls.

Table 2.4-1 Technical Specification Setpoint Information

	Values	
Parameter	**Current**	**EPU**
APRM Calibration Basis (MWt)	3458	3952
APRM High Flux Scram AL (% RTP)	125.4	No Change[2]
APRM Simulated Thermal Power – High		
DLO AL[1] (% RTP)	0.66W + 68.0 [3]	0.55W + 67.5 [3]
SLO AL[1] (% RTP)	0.66(W-ΔW) + 68.0 [3,4]	0.55(W -ΔW) + 67.5 [3,4]
Clamp AL[1] (% RTP)	122	No Change[2]
APRM Rod Block AL		
DLO Flow Biased[(1)] (% RTP)	0.66W + 64	0.55W + 63.5
SLO Flow Biased[(1)] (% RTP)	0.66(W_d-ΔW) + 64	0.55(W_d -ΔW) + 63.5
Clamp AL (% RTP)	118	No Change
APRM Neutron Flux – High Setdown (% RTP)		
Scram AL	25	23
Rod Block (APRM Upscale (Startup) AL (% RTP))	15	13
Rod Block Monitor ALs		
Low Power Setpoint (Enable) (% RTP)	30	No Change
Intermediate Power Setpoint (% RTP)	65	No Change
High Power Setpoint (% RTP)	85	No Change
Low Trip Setpoint (% Reference Level)	121.0	No Change [5]
Intermediate Trip Setpoint (% Reference Level)	116.0	No Change [5]
High Trip Setpoint (% Reference Level)	111.0	No Change [5]

Table 2.4-1 Technical Specification Setpoint Information (continued)

Parameter	Values	
	Current	**EPU**
RWM LPSP AL (% RTP)	10	No Change[6]
Main Steam Line High Flow Isolation (% rated steam flow) AL	144	144
Turbine First-Stage Pressure Scram and Recirculation Pump Trip Bypass (% RTP) AL	30	26
Reactor Vessel Water Level – Low, Level 3 Scram (inches Above Vessel Zero (AVZ)) AL	518	No Change[7]
Main Steam Line Low Pressure Isolation (in RUN Mode) Allowable Value Trip Setpoint, psig	$\geq$ 825	No Change[8]

Notes:

1. No credit is taken in any safety analysis for the flow referenced setpoints.
2. The EPU APRM Neutron Flux - High Scram, APRM Simulated Thermal Power – High Clamp and APRM Neutron Flux – High Setdown remain the same in terms of percent rated thermal power.
3. W is the Recirculation Drive Flow in percent of Rated flow. ΔW is the difference between the dual loop operation (DLO) and SLO drive flow at the same core flow. The current value of ΔW is 10% and is not changed.
4. The ALs for SLO operation are unchanged in terms of MWt.
5. The cycle-specific reload analysis is used to determine any change in the rod block trip setpoint. The RBM trip setpoints listed are based on an OLMCPR of 1.25. The trip setpoints corresponding to other OLMCPR values also would remain the same for EPU.
6. The EPU RWM LPSP remains the same in terms of percent rated thermal power.
7. The AL, AV and NTSP are not changed for EPU for this setpoint function. EPU satisfies the issue with Steam Flow Induced Error (SFIE; also called "Bernoulli error") in the case that the steam dryer skirt becomes uncovered for a loss of feedwater flow transient, per the related safety communication SC04-14 (Reference 63).
8. The MSL Low Pressure Isolation (in RUN Mode) Actual Trip Setpoint, is 843 psig and is unchanged for EPU.

Table 2.4-2 Changes to Instrumentation and Controls

Parameter	EPU Change[5]	Implementation Status[3]		
		U1	U2	U3
MSL High Flow	Rescale loops and revise setpoints for AL of 144% rated steam flow (PDT/PDIS-1-13A-D,-25A-D,-36A-D,-50A-D)[1]	N	N	N
MSL Flow	Rescale loops (FT-1-13,-25,-36,-50)[2]	Y	Y	N
Turbine 1st Stage Pressure (Scram Bypass Permissive)	Recalibrate for AL of 26.25% RTP (PT-1-81A/B,-91A/B)	N	N	N
Turbine 1st Stage Pressure	Rescale loop (PT-1-81)[2]	Y	Y	N
Turbine Exhaust Intermediate Pressure	Rescale loop (PT-1-100)	Y	Y	N
APRM Flow Biased STP Scram	Recalibrate (APRM-92-1,-2,-3,-4)	N	N	N
APRM Flow Biased STP Rod Block	Recalibrate	N	N	N
APRM Scram Setdown	Recalibrate	N	N	N
APRM Rod Block Setdown	Recalibrate	N	N	N
RBM Power Dependent Setpoints	Recalibrate per cycle-specific reload analysis	N	N	N
RWM LPSP	No change for EPU	-	-	-
Digital FW Controls Software	Update RWM alarm/enable setpoints and DFWCS parameters[4]	N	N	N
RFW Line A/B Flow	Rescale loops (FT-3-78A,-78B)[2]	Y	Y	N
RFW Pump A/B/C Suction Pressure	Replace pressure gauge with new range (PI-2-121,-122,-123)	Y	Y	N
RFW Pump A/B/C Low Suction Pressure Alarm/Trip	Recalibrate for new switch setpoints (PS-2-121A/B,-122A/B,-123A/B)	Y	Y	N
RFPT LP Steam Inlet Flow	Rescale loops (FT-1-117, -120)	Y	Y	N
RFPT Low Condenser Vacuum Trip	Added pressure switches and revise trip logic to 2-out-of-3 (PS-3-200A/B/C,-201A/B/C,-202A/B/C)	Y	Y	Y
RFP Discharge Flow	Rescale loops (FT-3-20,-13,-6)[2]	Y	Y	N
RFP Seal Injection DP	Replace DP controller and revise setpoints (PDC-3-80,-82,-84,-86,-88,-90)	Y	Y	N
RRS Jet Pump Head	Recalibrate loop (PDT/PDI-3-51)	Y	Y	Y
RCW Flow from RRS VFD HX A/B	Rescale loops (FIT-24-182,-187)	Y	Y	Y
RRS Pump A/B Winding Temperature	Revise high alarm setpoints (TA-68-58,-84)	Y	Y	Y
RRS VFD A/B Protective Relays	Revise protective relay setting (MMR, DFR)	Y	Y	Y

Table 2.4-2 Changes to Instrumentation and Controls (continued)

Parameter	EPU Change[5]	Implementation Status[3]		
		U1	U2	U3
RRS Pump Motor Controls Software	Revise upper power runback steam flow setpoint (RRMST:UPWRSPT)	N	N	N
SPE Bypass Line Flow	Rescale flow indicator loop (FT/FI-2-42)	Y	NA	NA
SJAE A/B Trip and Standby Auto Start	Modified logic to remove trip on low condenser vacuum and eliminate auto start of standby SJAE (PS-2-5B,-8B)	Y	Y	Y
SJAE A/B Condensate Pressure	Revise pressure switch setpoints to prevent inadvertent SJAE isolation (PS-2-34,-40)	Y	Y	N
SJAE Steam Supply Stage I, II, III Pressure	Revise low steam supply pressure isolation switch setpoints (PS-1-150, -152, -166, -167)	Y	Y	Y
Condensate Pump Discharge Header (HDR) Flow	Replace flow transmitter, controller, recorder and rescale loops (FT/FC-2-29 and XR-2-26)[8]	Y	Y	N
Condensate Pump Motor Current	Replace motor ammeters in Main Control Room (MCR) (EI-2-26,-21,-15) and at 4 kV unit board (EI-2-26/8, -21/7, -15/5)[7] with increased range	Y	Y	Y
Condensate Pump Breakers	Revise breaker relay trip settings (U-8, U-7, U-5)[9]	Y	Y	Y
Condensate Pump Motor Stator/Bearing Temperature	Revise high temperature alarm setpoints (TE-2-25A-H and J, -20A-H and J, -14A-H and J)	Y	Y	Y
Condensate Booster Pump Motor Current	Replace ammeters in MCR (EI-2-56,-62,-68) and at 4 kV unit board (II-2-56,-62,-68) with increased range	Y	Y	N
Condensate Booster Pump Breakers	Revise breaker relay trip settings (U-9, U-8, U-6)	Y	Y	N
Condensate to RFW Pump A/B/C Pressure	Replace pressure indicator with increased range (PI-2-81,-93,-106)	Y	-	-
Condenser A/B/C CCW Outlet Flow	Rescale and add outlet flow signals to ICS (FIT-27-156,-157,-158,-159,-160,-161)	Y	Y	Y
Condenser A/B/C CCW Temperature	Add inlet/outlet temperature signals to ICS (TE-27-33B,-36B-41B,-44B,-49B,-52B, -57B, -60B,-65B,-68B,-73B,-76B)	Y	Y	Y
Condenser A/B/C CCW Pressure	Replace transmitters and rescale inputs to ICS (PT-2-1,-5,-8)	Y	Y	Y

Table 2.4-2 Changes to Instrumentation and Controls (continued)

Parameter	EPU Change[5]	Implementation Status[3]		
		U1	U2	U3
Condenser A/B/C Low Vacuum Turbine Trip, Bypass Trip, and Alarm	Replace vacuum switches with pressure transmitters that provide input to the electro-hydraulic control (EHC) system (PS-47-72A/B,-73A/B,-74A/B,-75A/B/C, -125A/B/C)	N	N	N
Condenser A/B/C Low Vacuum - Digital EHC System Software Update	Revise turbine control software settings to support replacement of low condenser vacuum pressure switches with transmitters	N	N	N
No. 1, 2 Extraction HDR Pressure	Rescale loops (PT-5-3,-36)	Y	Y	Y
No. 2, 4 Extraction Steam Pressure (from LP Turbine A/B/C to Heater (HTR) A4/B4/C4)	Replace pressure gauges with increased range (PI-5-33B,-34,-36,-46,-47,-49,-59,-60,-62)	Y	Y	Y
FWH - A1/B1/C1 Outlet Temperature	Rescale ICS inputs (TE-3-44,-37,-30)	Y	Y	N
FWH - A1/B1/C1 Shell Pressure	Rescale loops (PT/PI-5-6,-10,-14)	Y	Y	N
FWH - A2/B2/C2 Shell Pressure	Rescale loops (PT/PI-5-18,-22,-26)	Y	Y	N
FWH - A1/B1/C1, A2/B2/C2, A3/B3/C3 Level	Replace obsolete transmitters (LT-6-1A/B,-4A/B, -7A/B,-19A/B,-22A/B,-25A/B,-37A/B,-40A/B,-43A/B)	Y	Y	Y
FWH - A1/B1/C1, A2/B2/C2, A3/B3/C3 Level	Rescale transmitter loops (LT-6-1A/B,-4A/B, -7A/B,-19A/B,-22A/B,-25A/B,-37A/B,-40A/B,-43A/B)	Y	Y	N
FWH - A1/B1/C1, A2/B2/C2, A3/B3/C3 Level	Replace obsolete FW HTR Level Indicating Controllers (LIC-6-1,-4,-7,-19,-22,-25,-37,-40,-43)	Y	NA	NA
FWH - A1/B1/C1, A2/B2/C2, A3/B3/C3 Level	Replace obsolete level switches and recalibrate (LS-6-1A/B,-4A/B, -7A/B,-19A/B,-22A/B,-25A/B,-37A/B,-40A/B,-43A/B)	Y	NA	NA
Moisture Separator Level Control (LC) Reservoir Drain Flow	Remove trip function of switches (FIS-6-56A/B, -57A/B,-58A/B) and retain as flow indicators with increased span (FI-6-56A/B,-57A/B,-58A/B)[6]	Y	Y	Y
Condensate Demineralizer vessel (VSL) Flow	Replace transmitters and rescale loops (FIT-2-208A thru H and J), add 10th VSL flow channel (FIT-2-208K)	Y	Y	Y
Condensate Demineralizer VSL DP	Replace transmitters, rescale loops (PDIT-2-205A thru H and J), add 10th VSL DP channel (PDIT-2-205K) and revise Programmable Logic Controller (PLC) high DP setpoint (PDSH-2-130SP)	Y	Y	Y

Table 2.4-2 Changes to Instrumentation and Controls (continued)

Parameter	EPU Change[5]	Implementation Status[3]		
		U1	U2	U2
Condensate Demineralizer VSL Resin Trap DP	Replace transmitters and rescale loops (PDIT-2-207A thru H and J) add 10th VSL flow channel (PDIT-2-207K)	Y	Y	Y
Condensate Demineralizer Inlet/Outlet Pressure	Add pressure gauges for 10th demineralizer (PI-2-205AK, -BK)	Y	Y	Y
Condensate Injection Flow	Add new flow indicator (FI-2-7043)	Y	Y	Y
Generator Hydrogen Pressure	Replace obsolete low and high pressure control/alarm switches and revise setpoints (PS-35-18A,-19,-18B) - revise control setpoints (PCV-35-5A,-5B,-39)	N	N	N
Generator Protection	Revise main generator over-excitation relay setpoint (J1K)	N	N	N
Generator Stator Cooling Water (SCW) Flow	Rescale transmitter (FIT-35-65) and revise low alarm setpoint (FA-35-65)	Y	N	Y
Generator SCW Inlet Flow	Revise low flow turbine runback/trip setpoints (FS-35-65A/B/C)	Y	N	Y
Generator SCW Inlet Pressure	Revise high/low alarm setpoints (PA-35-90A/B)	Y	N	Y
Generator SCW DP	Revise high/low/low low alarm setpoints (PDA-35-91A/B/C)	Y	N	Y
Generator SCW Outlet Temperature	Revise high temperature turbine runback/trip and alarm setpoints (TS-35-71A/B/C, TIS-35-72)	Y	N	Y
Generator SCW Cooler Discharge Pressure	Revise controller/control valve setpoints (PC/PCV-35-55)	Y	N	Y
Isophase Bus Duct Phase A/B/C Temperature at Generator	Revise high temperature alarm setpoints (TS-262-3A1,-3A2,-3B1,-3B2,-3C1,-3C2)	Y	Y	Y
Isophase Bus Duct Phase A/B/C Temperature at Main Bus	Revise high temperature alarm setpoints (TS-262-3A3,-3B3,-3C3)	Y	Y	Y
Digital EHC System Software	Revise turbine control software settings for electrical overspeed setpoint, intermediate pressure, power load unbalance, turbine first stage pressure, and MWe control	N	N	N
Offgas Condenser Cooling Water	Replace obsolete temperature sensor and rescale indicator (TE/TI-2-256)	Y	Y	Y

Table 2.4-2 Changes to Instrumentation and Controls (continued)

Parameter	EPU Change[5]	Implementation Status[3]		
		U1	U2	U3
Hydrogen Water Chemistry	Replace and rescale HWC oxygen to condensate flow indicator (FI-4-9)	Y	Y	N
Hydrogen Water Chemistry PLC	Install updated software in HWC PLC (PLC-4-40) for H_2 and O_2 injection rates at EPU	Y	Y	N
MCR Recorders	Replace obsolete recorders - reactor vessel level/total FW flow (XR-3-53), main steam flow (FR-46-5)	Y	Y	Y
MCR Recorders	Rescale recorders - reactor vessel level/total FW flow (XR-3-53), main steam flow (FR-46-5)	Y	Y	N

Notes:

1. Requires change to the differential pressure setpoint value.
2. Includes associated software updates to FWCS and RFPT Woodward Governor controls.
3. Implementation Status:
   "Y" (Yes)
   "N" (No) - The modification will be installed prior to implementing EPU on the respective Browns Ferry Unit.
4. Software updates to FWCS.
5. All loops rescaled to EPU values include corresponding rescaling of Integrated Computer System (ICS) inputs where applicable.
6. For Unit 1, the flow indicating switches were removed and replaced with flow indicating transmitters (FIT-6-56A/B,-57A/B,-58A/B) and rescaled for EPU.
7. For Unit 1, the component number for Condensate Pump 1C is EI-2-15/6.
8. Unit 1 controller is FIC-2-29 and Unit 3 recorder is FR-2-29.
9. Unit 1 Pump 1C Breaker relay is U-6.

2.5 Plant Systems

2.5.1 Internal Hazards

2.5.1.1 Flooding

2.5.1.1.1 Flood Protection

Regulatory Evaluation

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

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

Specific NRC review criteria are contained in SRP Section 3.4.1.

Browns Ferry Current Licensing Basis

The General Design Criteria (GDC) listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Browns Ferry Nuclear Plant (Browns Ferry) principal design criteria predate these criteria. The Browns Ferry principal design criteria are listed in UFSAR Section 1.5, "Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, the Tennessee Valley Authority (TVA) performed a comparative evaluation of the design basis of Browns Ferry with the AEC proposed General Design Criteria of 1967. The Browns Ferry UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria set out in the July 1967 AEC release. For each group of criteria, there is a statement of TVA's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a table of references to locations in the Browns Ferry UFSAR where there is subject matter relating to the intent of that particular criteria.

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

Browns Ferry internal flooding hazards are described in Browns Ferry UFSAR Section 10.16.4.6 "Evaluation for Flooding due to Failure of Low Energy Piping Systems Outside Primary Containment."

Browns Ferry's internal flooding hazards were evaluated for license renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Browns Ferry

License Renewal Safety Evaluation Report (SER), NUREG-1843, dated April 2006 (Reference 11). The license renewal evaluation associated with the flood protection barriers is documented in NUREG-1843, Section 2.4. During plant license renewal evaluations, tanks, and pipes which were not already in scope pursuant to 10 CFR 54.4(a)(1) or (a)(3) were evaluated to ensure they were not "non-safety equipment whose failure could affect a safety function" (Criterion (a)(2)). Components that met the inclusion criteria were evaluated within the system that contained them. Additionally, civil features whose function was to control, abate, or minimize the effects of flooding were identified and evaluated within the structure that contained them.

Technical Evaluation

2.5.1.1.1.1 High Energy Line Break

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

High Energy Line Breaks (HELBs) are evaluated for their effects on equipment qualification.

Browns Ferry meets all CLTR dispositions. The topics addressed in this evaluation are:

Topic	CLTR Disposition	Browns Ferry Result
Liquid Lines	Plant Specific	Meets CLTR Disposition

As stated in Section 10.1 of the CLTR, EPU may increase subcooling in the reactor vessel, which may lead to increased mass and energy release rates for liquid line breaks for RWCU only.

Components and/or equipment required for safe shutdown of the reactor were evaluated for the effect of flooding from breaks and cracks in high-energy lines. The evaluations verified that the plant can be safely shut down, assuming a concurrent single active failure in systems necessary to mitigate the consequences of the postulated component failure. Systems that are affected by EPU are FW and RWCU. The CLTP mass and energy releases for feedwater line breaks are affected by EPU implementation due to the changes in the feedwater system including increased feedwater flow rate and modifications to the condensate, condensate booster and feedwater pumps. At EPU, the RWCU system will operate at a lower enthalpy. Plant flooding due to internal piping failures in these systems was evaluated for changes due to EPU. Plant flooding is conservatively evaluated based on the most limiting event, which is the FW line break. In this event, the entire hotwell volume is being released in the main steam valve vault and main steam tunnel, and then drains to the Reactor Building. Because no changes are made to the existing hotwell inventory, draining systems, and flood barriers, the flood levels in the Reactor Building

due to a FW break are unchanged. RWCU line break flood level increase is due to the RWCU operating at a lower enthalpy, which will result in an increase in the critical crack flow by 4.41%. The change in the RWCU flood level remains bounded by the FW line break flood level. The remaining systems evaluated are not affected by EPU and remain bounded by the current flooding analyses. Internal flooding due to postulated failures in piping systems is not affected by EPU. Therefore, Browns Ferry meets all CLTR dispositions for liquid lines.

2.5.1.1.1.2 Moderate Energy Line Break

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

The EPU effect on Moderate Energy Line Break (MELB) spray and subcompartment temperature is addressed in Section 2.5.1.3.2. This section discusses the EPU effect on flooding levels.

Browns Ferry addresses the concern of moderate energy line breaks through various initiatives including:

- Probabilistic Risk Assessment – Internal Flooding Analysis
- Seismic Interaction Piping and Other Components Based on the Unresolved Safety Issues (USI) A-17, "System Interactions in Nuclear Power Plants" and USI A-46, "Seismic Qualifications of Equipment in Operating Plants."

While Browns Ferry was not originally licensed using the Standard Review Plan Section 3.6.1, the licensing and design basis includes specific conditions for MELBs for effects on safety-related equipment.

MELBs are evaluated for their effects on equipment used for safe shutdown.

Browns Ferry meets all CLTR dispositions. The topics addressed in this evaluation for MELB are:

Topic	CLTR Disposition	Browns Ferry Result
Flooding	Generic	Meets CLTR Disposition

The CLTR states that EPU results in no change in the inventory contained in moderate energy lines.

The flow rates and/or the system inventories of analyzed moderate energy piping systems do not increase for EPU. System design limits (design pressure) used as input to the MELB flooding analyses are not changed by EPU. EPU does not affect the ability of the plant to cope with

effects of spray from MELBs. EPU does not introduce new MELB locations and does not introduce or move safety-related equipment.

EPU will not affect the normal operating water levels and pressure of the suppression pool (torus), the flood seals between the Reactor and Turbine Building, flood barriers (curbs) of ECCS rooms, and flood level detection equipment in the lower Reactor Building elevation. The Intake Pumping Station and location of safety-related equipment within the structure will not change for EPU.

Sources of flooding and protection measures in the Circulating Water System (CWS) are not affected by EPU. The CWS is located in the Turbine Building, which is sealed from the Reactor Building to an elevation 572.5 feet, and the time for operator action remains valid, as EPU does not increase the design pressure for the CWS.

Therefore, MELB internal flooding meets all CLTR dispositions.

Conclusion

TVA has evaluated the effects of the proposed EPU on internal flooding hazards. The evaluation indicates that SSCs important to safety will continue to be protected from flooding and will continue to meet the requirements of draft GDC-2 following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to flood protection.

2.5.1.1.2 Equipment and Floor Drains

Regulatory Evaluation

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

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

Specific NRC review criteria are contained in SRP Section 9.3.3.

Browns Ferry Current Licensing Basis

The General Design Criteria (GDC) listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Browns Ferry Nuclear Plant (Browns Ferry) principal design criteria predate these criteria. The Browns Ferry principal design criteria are listed in UFSAR Section 1.5, "Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967,

the Tennessee Valley Authority (TVA) performed a comparative evaluation of the design basis of Browns Ferry with the AEC proposed General Design Criteria of 1967. The Browns Ferry UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria set out in the July 1967 AEC release. For each group of criteria, there is a statement of TVA's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a table of references to locations in the Browns Ferry UFSAR where there is subject matter relating to the intent of that particular criteria.

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

The equipment and floor drains are described in Browns Ferry UFSAR Section 10.16, "Equipment and Floor Drainage Systems."

Browns Ferry's equipment and floor drain systems were evaluated for license renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Browns Ferry License Renewal Safety Evaluation Report (SER), NUREG-1843, dated April 2006 (Reference 11). During plant license renewal evaluations, tanks and pipes which were not already in scope pursuant to 10 CFR 54.4(a)(1) or (a)(3) were evaluated to ensure they were not "non-safety equipment whose failure could affect a safety function" (Criterion (a)(2)). Components that met the inclusion criteria were evaluated within the system that contained them. Additionally, civil features whose function was to control, abate, or minimize the effects of flooding were identified and evaluated within the structure that contained them.

Technical Evaluation

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

Browns Ferry meets all CLTR dispositions. The topics addressed in this evaluation are:

Topic	CLTR Disposition	Browns Ferry Result
Waste Volumes	Plant Specific	Meets CLTR Disposition

The CLTR states that power uprate does not affect the floor drain collector subsystem and the waste collector subsystem operation or equipment performance. The floor drain collector subsystem and the waste collector (equipment drain) subsystem both receive periodic inputs from a variety of sources. Neither subsystem is expected to experience a large increase in the total volume of liquid and solid waste due to operation at the EPU condition. The design of the Browns Ferry equipment and floor drains inside and outside of containment has been evaluated to ensure any EPU-related liquid radwaste increases can be processed. Browns Ferry has sufficient capacity to handle added liquid increases expected (i.e., it can collect and process the drain fluids). Therefore, EPU does not affect system operation or equipment performance and meets all CLTR dispositions.

Conclusion

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

2.5.1.1.3 Circulating Water System

Regulatory Evaluation

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

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

Specific NRC review criteria are contained in SRP Section 10.4.5.

Browns Ferry Current Licensing Basis

The General Design Criteria (GDC) listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Browns Ferry Nuclear Plant (Browns Ferry) principal design criteria predate these criteria. The Browns Ferry principal design criteria are listed in UFSAR Section 1.5, "Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, the Tennessee Valley Authority (TVA) performed a comparative evaluation of the design basis of Browns Ferry with the AEC proposed General Design Criteria of 1967. The Browns Ferry UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria set out in the July 1967 AEC release. For each group of criteria, there is a statement of TVA's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with

the intent of the group of criteria. Following a restatement of each of the proposed criteria is a table of references to locations in the Browns Ferry UFSAR where there is subject matter relating to the intent of that particular criteria.

While Browns Ferry is not generally licensed to the final GDC or the 1967 AEC proposed General Design Criteria, a comparison of the final GDC to the applicable AEC proposed General Design Criteria can usually be made. For the final GDC listed in the Regulatory Evaluation above, the Browns Ferry comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Browns Ferry UFSAR Appendix A. No draft GDCs directly apply to the Circulating Water System.

The Circulating Water System is described in Browns Ferry UFSAR Section 11.6, "Condenser Circulating Water System."

The Browns Ferry circulating water system was evaluated for license renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Browns Ferry License Renewal Safety Evaluation Report (SER), NUREG-1843, dated April 2006 (Reference 11). The license renewal evaluation associated with the Circulating Water System is documented in NUREG-1843, Section 2.3.4.6. Management of aging effects on the Circulating Water System is documented in NUREG-1843, Section 3.3.

Technical Evaluation

The circulating water system is not being modified for EPU operation. The performance of the system was evaluated for EPU based on the original design capacity of the CWS and the cooling tower system over the actual range of circulating water inlet temperatures, and confirms that the circulating water system and heat sink are adequate for EPU operation. The evaluation of the CWS at EPU power indicates sufficient system capacity to ensure that the plant maintains adequate condenser backpressure. However, condenser backpressure limitations may require load reductions at the upper range of the anticipated circulating water inlet temperatures.

Conclusion

There are no EPU related modifications to the CWS. Performance was analyzed with respect to EPU power levels. Condenser backpressure limitations may require load reductions at the upper range of the anticipated circulating water inlet temperatures. The effect of EPU on the flooding analyses is addressed in Section 2.5.1.1.1.

2.5.1.2 Missile Protection

2.5.1.2.1 Internally Generated Missiles

Regulatory Evaluation

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

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

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

Browns Ferry Current Licensing Basis

The General Design Criteria (GDC) listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Browns Ferry Nuclear Plant (Browns Ferry) principal design criteria predate these criteria. The Browns Ferry principal design criteria are listed in UFSAR Section 1.5, "Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, the Tennessee Valley Authority (TVA) performed a comparative evaluation of the design basis of Browns Ferry with the AEC proposed General Design Criteria of 1967. The Browns Ferry UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria set out in the July 1967 AEC release. For each group of criteria, there is a statement of TVA's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a table of references to locations in the Browns Ferry UFSAR where there is subject matter relating to the intent of that particular criteria.

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

The missile protection for internally generated missiles is described in Browns Ferry UFSAR Sections 5.2.4.6, "Missile and Pipe Whip Prevention," and 11.2.2, "Power Generation Design Basis."

Browns Ferry's systems and components were evaluated for license renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Browns Ferry License Renewal Safety Evaluation Report (SER), NUREG-1843, dated April 2006 (Reference 11). The equipment and components credited with mitigating the effect of missiles are documented in NUREG-1843, Section 2.4, and the programs credited with managing that equipment aging are documented in NUREG-1843, Section 3.5.

<u>Technical Evaluation</u>

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

effects of EPUs. Section 7.1 of the CLTR addresses the effect of EPU on the turbine generator. The results of this evaluation regarding turbine missiles are described below.

Browns Ferry meets all CLTR dispositions. The topics addressed in this evaluation are:

Topic	CLTR Disposition	Browns Ferry Result
Turbine-Generator Missile Avoidance	Plant Specific	Meets CLTR Disposition

As explicitly stated in Section 7.1 of the CLTR, the increase in steam flow can change the previous missile avoidance and protection analysis.

The Browns Ferry design inherently provides missile protection for the safety-related SSCs, and important-to-safety non-safety related SSCs, by orienting the main and RFP turbines perpendicular to the control bay, Reactor Building and other structures containing safety-related and important-to-safety systems and components. This configuration ensures, in the unlikely event of a turbine failure, any missiles escaping the turbine shell are ejected away from the control bay, Reactor Building and other structures containing safety-related and important-to-safety systems and components. All three units have "favorably oriented" turbines as defined by RG 1.115 (Reference 64).

Unit 1 has replaced the LP turbine rotors with monoblock integral rotors. The new rotors are not susceptible to low speed rotor failure. A specific missile generation study is not required for the Unit 1 turbine. Integral (monoblock) rotors are not considered a source for potential missile generation for EPU for Unit 1. For Browns Ferry Units 2 and 3, specific calculations have been performed to determine the probability of a turbine missile. The worst case probability based on inspection frequency and turbine valve testing is 3.3×10^{-5}/year. When this turbine missile probability is applied using the NRC approved methodology (Reference 64) for calculating turbine missile damage probability, the resultant probability is 3.3×10^{-8}/year, which is below the acceptance criteria of 1×10^{-7}/year. The probability of a missile as the result of a runaway overspeed event is acceptable for EPU.

See Section 2.5.1.2.2 for additional information on the main turbine.

Transients which affect the feedwater pumps and turbines will be limited by the protective features of the feedwater control system. Therefore, there is no adverse effect associated with transients on the feedwater system.

Because the extended power uprate is at a constant pressure, there is no increase in the operating pressure of other auxiliary systems located in the Reactor Building for either the normally operating systems or the standby ECCS required to mitigate the consequences of abnormal transients or accidents. There is no change in the potential for generation of missiles or the energy of analyzed missiles in either safety-related systems or non-safety related systems in the proximity of safety-related SSCs. Therefore, the missile analyses remain valid.

The spent fuel pool system is located in the reinforced concrete Reactor Building. There is no large normally operating rotating equipment adjacent to the spent fuel pool. Dynamic effects and missiles that might result from plant equipment failures in the vicinity of the spent fuel pool have not changed with respect the plant's current design.

The review criterion specified in Matrix 5 of RS-001 is applicable to EPUs that result in substantially higher system pressures or changes in existing system configuration. Pressure does increase in the condensate and feedwater systems. However, the areas of increased pressure are not in the vicinity of SSCs important to safety as defined by RG 1.115 Appendix A. The Browns Ferry EPU does not create any condition resulting in an increase in probability of the generation of internal missiles. In addition, the Browns Ferry EPU does not entail any changes in equipment configurations that could change the effect of internally generated missiles on important-to-safety equipment. Therefore, internally generated missiles meet all CLTR dispositions.

Conclusion

TVA has evaluated changes in system pressures, configurations, and equipment rotational speeds necessary to support the proposed EPU. The evaluation indicates that SSCs important to safety will continue to be protected from the effects of internally generated missiles in accordance with draft GDC-40. Therefore, the proposed EPU is acceptable with respect to the protection of SSCs important to safety from internally generated missiles.

2.5.1.2.2 Turbine Generator

Regulatory Evaluation

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

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

Specific NRC review criteria are contained in SRP Section 10.2.

Browns Ferry Current Licensing Basis

The General Design Criteria (GDC) listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Browns Ferry Nuclear Plant (Browns Ferry) principal design criteria predate these criteria. The Browns Ferry principal design criteria are listed in UFSAR Section 1.5, "Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, the Tennessee Valley Authority (TVA) performed a comparative evaluation of the design basis

of Browns Ferry with the AEC proposed General Design Criteria of 1967. The Browns Ferry UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria set out in the July 1967 AEC release. For each group of criteria, there is a statement of TVA's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a table of references to locations in the Browns Ferry UFSAR where there is subject matter relating to the intent of that particular criteria.

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

The turbine generator is described in Browns Ferry UFSAR Sections 11.2, "Turbine-Generator," and 7.11, "Pressure Regulator and Turbine-Generator Control."

Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Browns Ferry License Renewal Safety Evaluation Report (SER), NUREG-1843, dated April 2006 (Reference 11). The license renewal evaluation associated with the turbine generator is documented in NUREG-1843, Section 2.3.4. Management of aging effects on the turbine generator is documented in NUREG-1843, Section 3.4.

Technical Evaluation

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

The topics addressed in this evaluation are:

Topic	CLTR Disposition	Browns Ferry Result
Turbine-Generator Performance	Plant Specific	Meets CLTR Disposition

The turbine-generator converts the thermal energy in the steam into electrical energy. The increase in thermal energy and steam flow from the reactor is translated to an increased electrical output from the station by the turbine-generator. The increase in steam flow can also change the previous missile avoidance and protection analysis (See Section 2.5.1.2.1).

The turbine-generator is required for normal plant operation and is not safety-related. Experience with previous power uprate applications indicates that turbine and generator modifications (e.g., turbine rotating element modification) are required to support power uprate. These modifications are required to support normal operation and are non-safety related. The turbine-generator overspeed protection systems were evaluated to ensure that adequate protection is provided for EPU conditions.

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

At CLTP and at a reactor dome pressure of 1,050 psia, the main turbines operate with a current rated throttle steam flow of 14.153 Mlbm/hr at a throttle pressure of 1,000 psia. The generators are rated at 1,280 MVA at a power factor of 0.9.

At EPU RTP and at a reactor dome pressure of 1,050 psia, the main turbines will operate with a rated throttle steam flow of 16.44 Mlbm/hr at a throttle pressure of 983 psia. The original Browns Ferry main generators were rewound in anticipation of uprating the power. The reactive capability curves are shown in Figures 2.5-2a (Unit 1) and 2.5-2b (Units 2 and 3). The current main generators are rated as follows for EPU:

- Unit 1: 1,330 MVA at a 0.95 power factor
- Units 2 and 3: 1,332 MVA at a 0.93 power factor

The existing HP turbine for each Browns Ferry unit is not capable of passing the required EPU steam flow rate and will be replaced prior to EPU. The new HP turbine section has been designed with an effective throttle flow margin of 5 percent above the required EPU throttle flow. The design point of the new HP turbine included the flow margin in order to ensure that the HP turbine will pass the rated throttle flow, as well as to allow for reactor pressure control. Therefore, the Valves Wide Open (VWO) condition refers to the turbine supply steam flow with additional margin over rated condition when adjusted for the lower inlet pressure associated with higher flow. For operation at EPU, the high pressure turbine has been re-designed with replacement diaphragms, buckets, and a new rotor, for at least the minimum target throttle flow margin, to increase the flow passing capability.

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

As part of the EPU on Unit 1, the original shrunk-on Low Pressure (LP) rotors were replaced with rotors of monoblock (integral) design. The HP rotors will also be replaced with rotors of monoblock design. Per CLTR Section 7.1, "The only safety related evaluation is the plant specific turbine-generator missile avoidance and protection analysis. The entrapped energy following a turbine trip or load rejection increases slightly for CPPU. Relative to the turbine generator missile protection analysis, many power plants have replaced high pressure and low pressure shrunk-on rotors with an integral rotor without shrunk-on wheels. These integral rotors are not considered a source for potential missile generation for CPPU for the slight increase in entrapped energy; therefore, a plant specific analysis is not required."

As part of the EPU on Units 2 and 3, modifications and inspections have been performed to the shrunk-on wheels for the LP turbine rotors to reduce the probability of LP turbine rotor blade failure and ejection. The HP turbine rotors will also be replaced with rotors of monoblock design. A specific missile generation study was performed. See Section 2.5.1.2.1 for the turbine missile evaluation.

The turbine 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 potential overspeed capability. The entrapped energy increases slightly for EPU conditions. Appendix A of the CLTR states that although the power uprate slightly increases the energy trapped in the turbine following a load rejection, the turbine overspeed would remain within design limits.

The turbine overspeed scenario considered is the emergency case where the EHC controls and the control and intercept valves fail to respond to the initial overspeed due to a load rejection event. For this scenario, the unit rapidly accelerates to the overspeed trip setpoint, thereby trip-closing the main and intermediate stop valves. The operating condition analyzed was the maximum power, valves wide open case, with low backpressure. This approach accounts for the two basic contributors to peak overspeed due to a load rejection event: 1) the energy due to entrapped (or entrained) steam within the steam path and inlet piping downstream of the main and intermediate steam valves; and 2) what is termed "valve lag overspeed," which takes into account the energy contributed by new steam entering the machine during the response time of the control and trip systems, and during the actual closing time of these valves. The overspeed trip setpoint is established such that the resulting peak speed will not exceed the 120% emergency overspeed limit due to overshoot. This ensures that the turbine is protected in an overspeed event. The turbine and turbine control system design changes for EPU have not yet been installed and the specific control setpoints have not been established. The setpoints will be adjusted to ensure that the turbine will not exceed 120% of rated speed due to overshoot. Equipment important to safety associated with the plant is protected from main turbine missiles by physical barriers and favorable alignment. Additionally, the independent spent fuel storage installation has been evaluated and determined acceptable with regard to plant generated main turbine missiles using the EPU turbine failure probability analyses as input. The effect of EPU is

offset by ensuring that the turbine speed will not exceed 120% of rated during an overspeed event.

Conclusion

TVA has evaluated the effects of the proposed EPU on the turbine generator. The evaluation indicates that the turbine generator will continue to provide adequate turbine overspeed protection to minimize the probability of generating turbine missiles and will continue to meet the requirements of draft GDC-40 following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to the turbine generator.

2.5.1.3 Pipe Failures

Regulatory Evaluation

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

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

Specific NRC review criteria are contained in SRP Section 3.6.1.

Browns Ferry Current Licensing Basis

The General Design Criteria (GDC) listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Browns Ferry Nuclear Plant (Browns Ferry) principal design criteria predate these criteria. The Browns Ferry principal design criteria are listed in UFSAR Section 1.5, "Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, the Tennessee Valley Authority (TVA) performed a comparative evaluation of the design basis of Browns Ferry with the AEC proposed General Design Criteria of 1967. The Browns Ferry UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria set out in the July 1967 AEC release. For each group of criteria, there is a statement of TVA's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a table of references to locations in the Browns Ferry UFSAR where there is subject matter relating to the intent of that particular criteria.

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

Design Criteria (referred to here as "draft GDC") is contained in Browns Ferry UFSAR Appendix A: draft GDC-40.

Piping failures outside containment are described in Browns Ferry UFSAR Appendix M, "Report on Pipe Failures Outside Containment in the Browns Ferry Nuclear Plant."

Technical Evaluation

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

2.5.1.3.1 High Energy Piping Outside Containment

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

Pipe break criteria were evaluated based on the requirements of Appendix M of the UFSAR, which is based on current licensing basis requirements. The combinations of stresses were evaluated to meet the requirement of pipe break criteria. Based on these criteria, no new postulated pipe break locations were identified.

Existing high-energy line break locations outside containment that are affected by EPU are identified in Section 2.2.1 with the effects summarized in Table 2.2-1.

Browns Ferry meets all CLTR dispositions. The topics addressed in this evaluation are:

Topic	CLTR Disposition	Browns Ferry Result
Steam Lines	Generic	Meets CLTR Disposition
Liquid Lines	Plant Specific	Meets CLTR Disposition

2.5.1.3.1.1 Steam Lines

The effect of EPU on HELB mass and energy release rates for steam lines outside containment is documented in Section 2.2.1.1. Section 2.2.1.1 concludes that the generic CLTR disposition for high-energy line breaks in steam lines is applicable and that EPU has no effect on HELB mass and energy release rates for steam lines outside containment. The Browns Ferry design basis for steam line breaks also includes a plant-specific MS Line intermediate break for which the effects of EPU are documented in Section 2.2.1.1.

The CLTR states that there is no effect on steam line breaks because steam conditions at the postulated break conditions are unchanged.

EPU has no effect on the steam pressure or enthalpy at the postulated break locations. Therefore, EPU has no effect on the mass and energy releases from a HELB in a steam line.

Therefore, the Browns Ferry steam lines meet all CLTR dispositions.

2.5.1.3.1.2 Liquid Lines

The effect of EPU on HELB mass and energy release rates for liquid lines outside containment is documented in Section 2.2.1.2. The evaluations document energy release rates for the RWCU and FW systems. Section 2.2.1.2.1 documents the plant specific HELB evaluation of RWCU line breaks, and Section 2.2.1.2.2 documents the plant specific HELB evaluation for FW line breaks. The effects of EPU operation on the feedwater line break (FWLB) pipe whip, jet impingement, jet reaction and flooding analyses are addressed in Sections 2.2.1.2 and 2.5.1.1.

The CLTR states that EPU may increase subcooling in the reactor vessel, which may lead to increased break flow rates for liquid line breaks. EPU conditions may result in an increase in the mass and energy release for liquid line breaks. Therefore, liquid line breaks are evaluated for EPU, and the evaluations include EPU effects on subcompartment pressures and temperatures, pipe whip and jet impingement, and flooding.

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

RWCU mass and energy release rates and their effect on environmental conditions (compartment pressures and temperatures) were re-analyzed for both CLTP and EPU (for Units 2 and 3). Unit 1 was evaluated for EPU only. The CLTP mass and energy release rates (Units 2 and 3) for FW line breaks are negligibly affected by EPU. However, the effects of a FW system line break on main steam valve vault peak pressures and temperatures will continue to be bounded by a main steam line break in the main steam valve vault.

Therefore, the Browns Ferry Liquid Lines meet all CLTR dispositions.

2.5.1.3.2 Moderate Energy Piping Outside Containment

As stated in Section 2.5.1.1, system design limits (design pressure) used as input to the MELB flooding analyses and are not changed by EPU. Because the Browns Ferry MELB mass releases and environmental conditions (pressures and temperatures) are not affected by the EPU, there is no adverse effect on post-MELB control room habitability or on access to areas important to safe control of post-accident operations.

Topic	CLTR Disposition	Browns Ferry Result
Flooding	Generic	Meets CLTR Disposition

The CLTR states that EPU results in no change in the inventory contained in moderate energy lines.

Therefore, flooding meets all CLTR dispositions.

2.5.1.3.3 Environmental Conditions

All EQ equipment outside primary containment remains above the flood levels resulting from postulated pipe breaks and therefore are not subject to submergence.

Conclusion

TVA has evaluated the changes that are necessary for the proposed EPU. The evaluation indicates that SSCs important to safety will continue to be protected from the dynamic effects of postulated piping failures in fluid systems outside containment and will continue to meet the requirements of draft GDC-40 following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to protection against postulated piping failures in fluid systems outside containment.

2.5.1.4 Fire Protection

Regulatory Evaluation

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

The NRC's acceptance criteria for the FPP are based on (1) 10 CFR 50.48 and associated Appendix R to 10 CFR Part 50, insofar as they require the development of an FPP to ensure, among other things, the capability to safely shut down the plant; (2) GDC-3, insofar as it requires that (a) SSCs important to safety be designed and located to minimize the probability and effect of fires, (b) noncombustible and heat resistant materials be used, and (c) fire detection and fighting systems be provided and designed to minimize the adverse effects of fires on SSCs important to safety; (3) GDC-5, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions.

Specific NRC review criteria are contained in SRP Section 9.5.1.1, as supplemented by the guidance provided in Attachment 1 to Matrix 5 of Section 2.1 of RS-001.

Browns Ferry Current Licensing Basis

The General Design Criteria (GDC) listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Browns Ferry Nuclear Plant (Browns Ferry) principal design criteria predate these criteria. The Browns Ferry principal design criteria are listed in UFSAR Section 1.5, "Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, the Tennessee Valley Authority (TVA) performed a comparative evaluation of the design basis of Browns Ferry with the AEC proposed General Design Criteria of 1967. The Browns Ferry

UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria set out in the July 1967 AEC release. For each group of criteria, there is a statement of TVA's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a table of references to locations in the Browns Ferry UFSAR where there is subject matter relating to the intent of that particular criteria.

While Browns Ferry is not generally licensed to the final GDC or the 1967 AEC proposed General Design Criteria, a comparison of the final GDC to the applicable AEC proposed General Design Criteria can usually be made. For the final GDC listed in the Regulatory Evaluation above, the Browns Ferry comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Browns Ferry UFSAR Appendix A: draft GDC-4. Final GDC-3 is applicable to Browns Ferry as described in the Browns Ferry Fire Protection Report, Volume 1, Revision 20.

Fire Protection is described in Browns Ferry UFSAR Section 10.11, "Fire Protection Systems" and the Fire Protection Report.

Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Browns Ferry License Renewal Safety Evaluation Report (SER), NUREG-1843, dated April 2006 (Reference 11). The fire protection systems are documented in NUREG-1843, Section 2.3.3.6. Fire barrier materials are addressed as a commodity group, while walls, floors, doors, and structural steel are evaluated within the building that contains them. Components credited with achieving safe shutdown following a fire are evaluated within the system that contains them. Management of aging effects on the fire protection systems is documented in NUREG-1843, Section 3.3.

By letter dated March 27, 2013 (Reference 65), TVA submitted a LAR to transition Browns Ferry's licensing basis to the National Fire Protection Association (NFPA) 805 standard in accordance with 10 CFR 50.48(c). The current licensing basis per 10 CFR 50.48(b) will be superseded. The transition to NFPA 805 is currently under NRC staff review, and TVA anticipates its approval prior to implementing EPU operation. Accordingly, the fire protection analysis described in this section is based on NFPA 805 implementation. Although TVA fully expects approval of the transition to NFPA for Browns Ferry, Appendix A to this PUSAR provides an EPU evaluation under the current fire protection program in accordance with 10 CFR 50.48(b) and 10 CFR 50, Appendix R requirements.

Technical Evaluation

2.5.1.4.1 Fire Protection Program

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

effects of EPUs. Section 6.7 of the CLTR addresses the effect of EPU on the fire protection program. The results of this evaluation are described below.

As explicitly stated in Section 6.7 of the CLTR, [[

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Therefore, the reactor and containment responses and operator actions will be evaluated on a plant-specific basis for EPU.

This section addresses the effect of EPU on the fire protection program, fire suppression and detection systems, and reactor and containment system responses to postulated fire events.

Once the NFPA 805 (Reference 66) fire protection transition is implemented, Browns Ferry will meet all CLTR dispositions. The topics addressed in this evaluation are:

Topic	**CLTR Disposition**	**Browns Ferry Result**
Fire Suppression and Detection Systems	Plant Specific	Meets CLTR Disposition
Operator Response Time	Plant Specific	Meets CLTR Disposition
Peak Cladding Temperature	Plant Specific	Meets CLTR Disposition
Vessel Water Level	Plant Specific	Meets CLTR Disposition
Suppression Pool Temperature	Plant Specific	Meets CLTR Disposition

The higher decay heat associated with EPU results in higher heat input into the suppression pool which, without mitigation, will result in higher suppression pool temperatures. The higher decay heat may also result in lower vessel water levels or higher Peak Cladding Temperatures (PCTs), depending on the plant-specific analysis basis. As a result of these effects, fire suppression and detection systems, operator response time, peak clad temperature (PCT), and suppression pool temperature need to be addressed.

Tennessee Valley Authority (TVA) is implementing the Nuclear Energy Institute methodology NEI 04-02, Revision 2, "Guidance for Implementing a Risk-informed, Performance-based Fire Protection Program Under 10 CFR 50.48(c)" (NEI 04-02) (Reference 67), to transition Browns

Ferry Nuclear Plant (BFN) Units 1, 2 and 3 from its current fire protection licensing basis to the new requirements as outlined in NFPA 805. The BFN NFPA 805 Fire Safe Shutdown Analysis consists of a deterministic analysis and a performance based analysis. The deterministic analysis (NFPA 805 Section 4.2.3) identifies and evaluates one success path for each fire area to meet the nuclear safety performance criteria of Section 1.5. Section 2.5.1.4.2, Fire Event, addresses the deterministic analysis. For instances where the nuclear safety performance criteria are not met, a performance based analysis (NFPA 805 Section 4.2.4) is performed to demonstrate that risk is acceptable and that defense in depth and safety margin are maintained. The performance based analysis is addressed in LAR Attachment 44.

Safe Shutdown Systems, equipment, and compensatory measures will be sufficient to support EPU. EPU is found to not affect the elements of the fire protection program related to: (1) fire suppression and detection systems, (2) fire zones/areas, (3) fire barriers, and (4) fire protection responsibilities of plant personnel. Administrative controls, associated with fire protection in the Technical Specifications, the Fire Protection Report, and the Nuclear QA Plan, will be adequate for EPU conditions.

As a risk reduction action in the NFPA 805 Transition LAR (Reference 65), a non-safety Emergency High Pressure Makeup Pump (EHPMP) will be installed. The EHPMP will supply water to the RPV from the CST. Injecting with this pump will also add additional volume to the suppression pool. The EHPMP will not be credited for Containment Accident Pressure elimination; however, the analysis shows that a net positive suction head improvement for the safe shutdown pumps would be realized. (See Section 2.6.5.2) Other EPU modifications will be assessed and assured not to adversely affect the ability to achieve and maintain the fuel in a safe and stable condition in the event of a fire.

Original NFP 805 analyses were performed at EPU conditions and therefore operator action times cannot be compared to CLTP conditions. To ensure that PCT remains less than the acceptance criterion in the most limiting scenario, one LPCI pump must be manually aligned for injection within 20 minutes. The EPU requires no new operator actions for fire safe shutdown of the plant and there are no actions required inside the primary containment.

The reactor and containment responses to the postulated fire events at EPU conditions are described in Section 2.5.1.4.2. The results show that for the limiting thermal-hydraulic cases, peak fuel cladding temperature, vessel water level, and suppression pool temperature meet the acceptance criteria and there is sufficient time for the operators to perform the necessary actions to meet the NFPA 805 requirement to achieve and maintain the fuel in a safe and stable condition in the event of a fire.

Therefore, once the NFPA 805 fire protection transition is implemented, Browns Ferry will meet all CLTR dispositions.

2.5.1.4.2 Fire Event

The limiting NFPA 805 fire events were analyzed under EPU conditions. The fuel heat-up analysis was performed using the NRC accepted AREVA LOCA methodology

(RELAX/HUXY). The containment analysis was performed using the GEH SHEX model. These analyses determined the effect of EPU on fuel cladding integrity, reactor vessel integrity, and containment integrity as a result of the fire event. The two bounding cases described below are identified as "Case 1" and "Case 4." See Tables 2.5-1, 2.5-2, and 2.5-3 for the inputs and results of the fire event analyses.

Case 1: The bounding safe shutdown case for PCT has Multiple Spurious Operation (MSO) of 11 of the 13 MSRVs which depressurize the reactor, and one RHR pump aligned in the LPCI/ASDC mode at 20 minutes. The analysis shows that the calculated PCT of 1,330°F is acceptable from a deterministic perspective (< 1,500°F) (See FUSAR Section 2.5.1.4).

Case 4 (See Figure 2.5-1): The bounding safe shutdown case for peak suppression pool temperature has reactor depressurization beginning at 25 minutes using three MSRVs. As the reactor is depressurized, condensate pumps replenish reactor inventory until hotwell inventory is depleted. After condensate is secured, one RHR pump is aligned into LPCI/ASDC mode. One RHRSW pump is initiated at two hours. Peak SP temperature reaches 208.0°F and this meets the containment integrity acceptance criteria of < 281°F and the torus attached piping limit of <223°F (See Section 2.2.2.2.2.2). Analyses show that containment accident pressure credit is not required to ensure adequate pump net positive suction head (NPSH) to mitigate a fire event (see Section 2.6.5.2 and LAR Attachment 39).

The results of Case 4, and the evaluations in Section 2.6.5.2, FUSAR Section 2.5.1.4, and LAR Attachment 39, demonstrate that the peak fuel cladding temperature, vessel water level, and suppression pool temperature meet the acceptance criteria and the time available for the operators to perform the necessary actions is sufficient. Therefore, EPU has no adverse effect on the ability of the systems and personnel to mitigate the effects of a fire event and satisfies the requirement of achieving and maintaining the fuel in a safe and stable condition in the event of a fire.

Conclusion

TVA has evaluated fire-related safe shutdown requirements and has accounted for the effects of the increased decay heat on the ability of the required systems to achieve and maintain safe shutdown conditions. The evaluation indicates that the FPP will continue to meet the requirements of 10 CFR 50.48, final GDC-3, and draft GDC-4 following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to fire protection.

2.5.2 Fission Product Control

2.5.2.1 Fission Product Control Systems and Structures

Regulatory Evaluation

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

Specific NRC review criteria are contained in SRP Section 6.5.3.

Browns Ferry Current Licensing Basis

The General Design Criteria (GDC) listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Browns Ferry Nuclear Plant (Browns Ferry) principal design criteria predate these criteria. The Browns Ferry principal design criteria are listed in UFSAR Section 1.5, "Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, the Tennessee Valley Authority (TVA) performed a comparative evaluation of the design basis of Browns Ferry with the AEC proposed General Design Criteria of 1967. The Browns Ferry UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria set out in the July 1967 AEC release. For each group of criteria, there is a statement of TVA's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a table of references to locations in the Browns Ferry UFSAR where there is subject matter relating to the intent of that particular criteria.

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

The Standby Gas Treatment System is described in Browns Ferry UFSAR Section 5.3.3.7, "Standby Gas Treatment System."

Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Browns Ferry License Renewal Safety Evaluation Report (SER), NUREG-1843, dated April 2006 (Reference 11). The license renewal evaluation associated with the Standby Gas Treatment System is documented in NUREG-1843, Section 2.3.2.2. Management of aging effects on the Standby Gas Treatment System is documented in NUREG-1843, Section 3.2.

<u>Technical Evaluation</u>

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of EPUs. Section 4.5 of the CLTR addresses the effect of EPU on the SGTS.

The assumptions regarding leakage and exhaust paths from the primary and secondary containments and other sources are as described in Alternative Source Term (AST) methodology for Browns Ferry (Reference 68).

Browns Ferry meets all CLTR dispositions. Therefore, a plant-specific evaluation is not required. The topic addressed in this evaluation is:

Topic	CLTR Disposition	Browns Ferry Result
Flow Capacity	Generic	Meets CLTR Disposition
Iodine Removal Capability	Generic	Meets CLTR Disposition

The CLTR states that the core inventory of iodine and subsequent loading on the SGTS filters or charcoal adsorbers are affected by EPU.

The SGTS is designed to maintain secondary containment at a negative pressure and to provide an elevated release path for the removal of fission products potentially present during abnormal conditions. By preventing the ground level release of airborne particulates and halogens, the SGTS limits off-site dose following a postulated DBA. The flow capacity of the SGTS and its ability to maintain a negative pressure in the secondary containment are discussed in Section 2.6.6.

At Browns Ferry, neither the SGTS component design nor the filter materials are being altered due to the EPU. The total (radioactive plus stable) post-LOCA iodine loading on the charcoal adsorbers increases proportionally with the increase in core iodine inventory, which increases with core thermal power. However, and as accepted by the CLTR, sufficient charcoal mass is present so that the post-LOCA iodine loading on the charcoal remains does not increase decay heating such that operation is challenged or there is a threat of charcoal ignition.

Browns Ferry is not committed to RG-1.52 with respect to iodine loading onto SGTS charcoal. As is stated in Reference 1, “[[

]]”

Two bounding analyses have been performed in the CLTR to evaluate decay heating in the SGTS for: 1) plants that implement AST in accordance with RG 1.183 (Reference 68), and 2) plants committed to RG 1.3 (Reference 71) for fission product transport. From Reference 1, "[[

]]" The parameters and their bounding values, with a comparison to the Browns Ferry specific values, are shown in Table 2.5-4.

As seen in Table 2.5-4, the Browns Ferry SGTS design is bounded with respect to the applicable parameters.

[[

]] No credit is taken for charcoal adsorption for any DBA. Credit is taken for high efficiency particulate adsorber (HEPA) filter removal of 90% of the particulate activity in the DBA-LOCA analysis (Reference 72).

[[

]]. Browns Ferry credits 2 (out of 3) trains operating during the accident period, therefore, the values presented are accepted as the maximum heating and iodine loading for one of the two trains. The Browns Ferry SGTS utilizes a low flow cooling system to assure no desorption of radionuclides in the case of increased decay heating.

While decay heat from fission products accumulated within the system filters and charcoal adsorbers increases with the increase in thermal power, the low flow cooling sub-system of the SGTS will still continue to protect the system from desorption should there be a loss of a system fan.

The parameters used in the CLTR bounding analysis for AST application are confirmed to bound the Browns Ferry plant-specific values. Therefore the Browns Ferry SGTS design and operation under EPU conditions is consistent with the overall CLTR disposition for the SGTS (that the ability of the SGTS to remove fission products is not adversely affected by EPU) and satisfies applicable regulatory guidance.

Conclusion

TVA has evaluated the effects of the proposed EPU on the Standby Gas Treatment System. The evaluation indicates that the system will continue to provide adequate fission product removal in post-accident environments following implementation of the proposed EPU and will continue to meet the requirements of draft GDC-70. Therefore, the proposed EPU is acceptable with respect to the fission product control systems and structures.

2.5.2.2 Main Condenser Evacuation System

Regulatory Evaluation

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

The NRC's acceptance criteria for the MCES are based on (1) GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents; and (2) GDC-64, insofar as it requires that means be provided for monitoring effluent discharge paths and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences and postulated accidents.

Specific NRC review criteria are contained in SRP Section 10.4.2.

Browns Ferry Current Licensing Basis

The General Design Criteria (GDC) listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Browns Ferry Nuclear Plant (Browns Ferry) principal design criteria predate these criteria. The Browns Ferry principal design criteria are listed in UFSAR Section 1.5, "Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, the Tennessee Valley Authority (TVA) performed a comparative evaluation of the design basis of Browns Ferry with the AEC proposed General Design Criteria of 1967. The Browns Ferry UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria set out in the July 1967 AEC release. For each group of criteria, there is a statement of TVA's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a table of references to locations in the Browns Ferry UFSAR where there is subject matter relating to the intent of that particular criteria.

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

The Main Condenser Evacuation System is described in Browns Ferry UFSAR Section 11.4, "Main Condenser Gas Removal and Turbine Sealing System."

Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Browns Ferry License Renewal Safety Evaluation Report (SER), NUREG-1843, dated April 2006

(Reference 11). The license renewal evaluation associated with the Main Condenser Evacuation System is documented in NUREG-1843, Section 2.3.4. Management of aging effects on the Main Condenser Evacuation System is documented in NUREG-1843, Section 3.4.2.

Technical Evaluation

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

Browns Ferry meets all CLTR dispositions. The topics addressed in this evaluation are:

Topic	CLTR Disposition	Browns Ferry Result
Condenser and SJAE	Plant Specific	Meets CLTR Disposition

The CLTR states that the increase in steam flow increases the heat removal requirement for the condenser. The additional power level increases the non-condensable gases generated by the reactor.

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

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

- Non-condensable gas flow capacity of the SJAE system;
- Capability of the SJAEs to operate satisfactorily with available dilution / motive steam flow; and
- Mechanical vacuum (hogging) pump capability to remove required non-condensable gases from the condenser at EPU start-up conditions

The capacity of the SJAEs is adequate because they were originally designed for operation at flows greater than those required at EPU conditions. Therefore, the main condenser evacuation system design bases for Browns Ferry are unchanged for EPU.

Conclusion

There are no EPU related changes to the MCES and the MCES will continue to maintain its ability to control and provide monitoring for releases of radioactive materials to the environment following implementation of the proposed EPU. The MCES will continue to meet the requirements of draft GDCs-17 and 70. Therefore, the proposed EPU is acceptable with respect to the MCES.

2.5.2.3 Turbine Gland Sealing System

Regulatory Evaluation

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

The NRC's acceptance criteria for the turbine gland sealing system are based on (1) GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents; and (2) GDC-64, insofar as it requires that means be provided for monitoring effluent discharge paths and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences and postulated accidents.

Specific NRC review criteria are contained in SRP Section 10.4.3.

Browns Ferry Current Licensing Basis

The General Design Criteria (GDC) listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Browns Ferry Nuclear Plant (Browns Ferry) principal design criteria predate these criteria. The Browns Ferry principal design criteria are listed in UFSAR Section 1.5, "Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, the Tennessee Valley Authority (TVA) performed a comparative evaluation of the design basis of Browns Ferry with the AEC proposed General Design Criteria of 1967. The Browns Ferry UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria set out in the July 1967 AEC release. For each group of criteria, there is a statement of TVA's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a table of references to locations in the Browns Ferry UFSAR where there is subject matter relating to the intent of that particular criteria.

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

The Turbine Gland Sealing System is described in Browns Ferry UFSAR Section 11.4, "Main Condenser Gas Removal and Turbine Sealing System."

Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Browns Ferry License Renewal Safety Evaluation Report (SER), NUREG-1843, dated April 2006 (Reference 11). The license renewal evaluation associated with the Turbine Gland Sealing

System is documented in NUREG-1843, Section 2.3.4. Management of aging effects on the Turbine Gland Sealing System is documented in NUREG-1843, Section 3.4.2.

Technical Evaluation

Each turbine sealing system includes a steam seal regulator with the necessary valves to maintain a constant positive pressure in the steam seal supply header and a single steam-packing exhauster condenser equipped with two full-capacity blowers to prevent steam leakage at the turbine shaft seals. The turbine sealing system prevents the leakage of steam into the Turbine Building and also prevents the leakage of air into the main condenser. During normal power operations, a pressure regulator valve and two seal steam header unloader valves maintain the seal steam header pressure at approximately 4 psig. To regulate the seal steam header pressure, the unloader valves divert excess seal steam to the main condenser. For EPU, larger unloader valves (8” to 10”) and associated piping are being installed to provide additional capability to maintain the seal steam header pressure at approximately 4 psig. EPU conditions will not affect the capability of the turbine sealing system to contain activated nitrogen and limit exposure to radiation.

Conclusion

TVA has evaluated the effects of the proposed EPU on the turbine gland sealing system. This evaluation indicated that the turbine gland sealing system will continue to meet the performance requirements following modification of the non-safety related seal steam header unloader valves and associated piping. After the modifications are implemented, the turbine gland sealing system will continue to maintain its ability to control and provide monitoring for releases of radioactive materials to the environment consistent with draft GDCs-17 and 70. Therefore, the proposed EPU is acceptable with respect to the turbine gland sealing system.

2.5.2.4 Main Steam Isolation Valve Leakage Control System

Regulatory Evaluation

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

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

Specific NRC review criteria are contained in SRP Section 6.7.

Browns Ferry Current Licensing Basis

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

Technical Evaluation

Not applicable.

Conclusion

Not applicable.

2.5.3 Component Cooling and Decay Heat Removal

2.5.3.1 Spent Fuel Pool Cooling and Cleanup System

Regulatory Evaluation

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

The NRC's acceptance criteria for the spent fuel pool cooling and cleanup system are based on (1) GDC-5, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions; (2) GDC-44, insofar as it requires that a system with the capability to transfer heat loads from safety-related SSCs to a heat sink under both normal operating and accident conditions be provided; and (3) GDC-61, insofar as it requires that fuel storage systems be designed with RHR capability reflecting the importance to safety of decay heat removal, and measures to prevent a significant loss of fuel storage coolant inventory under accident conditions.

Specific NRC review criteria are contained in SRP Section 9.1.3, as supplemented by the guidance provided in Attachment 2 to Matrix 5 of Section 2.1 of RS-001.

Browns Ferry Current Licensing Basis

The General Design Criteria (GDC) listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Browns Ferry Nuclear Plant (Browns Ferry) principal design criteria predate these criteria. The Browns Ferry principal design criteria are listed in UFSAR Section 1.5, "Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, the Tennessee Valley Authority (TVA) performed a comparative evaluation of the design basis of Browns Ferry with the AEC proposed General Design Criteria of 1967. The Browns Ferry UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria set out in the July 1967 AEC release. For each group of criteria, there is a statement of TVA's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a table of references to locations in the Browns Ferry UFSAR where there is subject matter relating to the intent of that particular criteria.

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

The Spent Fuel Pool Cooling and Cleanup System is described in Browns Ferry UFSAR Section 10.5, "Fuel Pool Cooling and Cleanup System."

Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Browns Ferry License Renewal Safety Evaluation Report (SER), NUREG-1843, dated April 2006 (Reference 11). The license renewal evaluation associated with the Spent Fuel Pool Cooling and Cleanup System is documented in NUREG-1843, Section 2.3.3.26. Management of aging effects on the Spent Fuel Pool Cooling and Cleanup System is documented in NUREG-1843, Section 3.3.2.

Technical Evaluation

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

Browns Ferry meets all CLTR dispositions. The topics addressed in this evaluation are:

Topic	CLTR Disposition	Browns Ferry Result
Fuel Pool Cooling (Normal Core Offload and Full Core Offload)	Plant Specific	Meets CLTR Disposition
Crud Activity and Corrosion Products	Plant Specific	Meets CLTR Disposition
Radiation Levels	Plant Specific	Meets CLTR Disposition
Fuel Racks	Generic	Meets CLTR Disposition

2.5.3.1.1 Fuel Pool Cooling (Normal and Full Core Offload)

As stated in Section 6.3.1 of the CLTR, for the same time after shutdown, the spent fuel pool heat load increases due to the decay heat generation as a result of EPU.

The spent fuel cooling section of the Fuel Pool Cooling and Cleanup System (FPCCS) consists of two trains of pumps and heat exchangers and two trains of the non-safety Auxiliary Decay Heat Removal (ADHR) system. The RHR safety-related system supplemental fuel pool cooling mode may be used to augment the capacity of the FPCCS when the ADHR system is unavailable.

The Browns Ferry Spent Fuel Pool (SFP) bulk water temperature must be maintained below the licensing limit of 150°F. The temperature requirement assures operator comfort (an operational requirement), and provides ample margin against an inventory loss in the fuel pool due to evaporation or boiling. The limiting condition is a full core discharge with all remaining storage locations filled with used fuel from prior discharges. EPU does not affect the alignments, availability or safety-related designations of these systems. EPU did not change the trains of cooling used to evaluate the effects of core offload.

EPU will increase the decay heat load 14.29% for fuel being offloaded from the reactor. This will result in a small overall increase in the heat load on the FPCCS during and after refueling outages because of the increase in decay heat. The decay heat for the EPU was calculated using the formulation and uncertainty factors from ANSI/ANS-5.1-1979 with two-sigma uncertainty and a correction for miscellaneous actinides and activation products. The use of ANSI/ANS-5.1-1979 has been endorsed in NUREG-0800 Section 9.2.5, Revision 3 (Reference 73).

The effect of this heat load on the SFP temperature was then evaluated for bounding full core offloads added to a bounding SFP heat load from previously offloaded batches. The evaluation of the full core offload credits one loop of the FPCCS and one loop of the ADHR system for directly removing the decay heat from the SFP. The result of this conservative evaluation shows that, using the single loop of FPCCS and ADHR alone, the SFP temperature can be maintained below 150°F.

EPU does not affect the heat removal capability of the FPCCS, the ADHR system, or the supplemental fuel pool cooling mode of the RHR system. EPU results in slightly higher core decay heat loads during refueling. Each reload affects the decay heat generation in the SFP after a batch discharge of fuel from the reactor. The full core offload heat load in the SFP reaches a maximum immediately after the full core discharge. Plant procedures limit the rate of heat addition to the fuel pool based on calculated operational heat load limits and available heat removal systems. Operational considerations for these procedural limits include delaying initial fuel movements into the pool and/or limitations on the rate of transfer of the fuel to the pool.

The SFP normal makeup source is from the Seismic Category II condensate storage system with a capacity of 100 gpm and is not affected by EPU and remains adequate for EPU conditions.

Browns Ferry has two Seismic Category I emergency makeup sources, the RHR/RHR service water crosstie and the emergency equipment cooling water system; each has a makeup capability of at least 150 gpm.

Existing plant instrumentation and procedures provide adequate indications and direction for monitoring and controlling SFP temperature and level during normal batch offloads and the unexpected case of the limiting full core offload. Symptom based operating procedures exist to provide mitigation strategies including placing additional cooling trains or systems in service, stopping fuel movement, and initiating make-up if necessary. The symptom based entry conditions and mitigation strategies for these procedures do not require changes for EPU.

A normal batch offload (approximately 332 fuel bundles) is assumed for outage planning with the additional assumptions in either case (batch or full core) of only one of two trains of FPCCS and only one of two trains of ADHR available, 24-month fuel cycle, and ANSI/ANS 5.1-1979 + 2σ.

2.5.3.1.2 Crud Activity and Corrosion Products

Section 6.3 of the CLTR requires a plant-specific evaluation for the fuel pool crud activity and corrosion products.

As stated in Section 6.3.2 of the CLTR, crud activity and corrosion products associated with spent fuel can increase slightly due to power uprate. The amount of crud activity and pool quality are operational considerations and are unrelated to safety. An evaluation of the capability of the FPCCS to maintain water clarity concludes that water clarity will not be affected by EPU. Therefore, the crud activity and corrosion products meet all CLTR dispositions.

2.5.3.1.3 Radiation Levels

As stated in Section 6.3.3 of the CLTR, the normal radiation levels around the SFP may increase slightly, primarily during fuel handling operations. Radiation levels in those areas of the plant, which are directly affected by the reactor core and spent fuel, increase by the percentage increase in the average power density of the fuel bundles. Therefore, for an EPU increase of 14.29%, the radiation dose rates increase by 14.29%. The radiation level around the SFP is an operational consideration and is unrelated to safety.

EPU will increase the core thermal power by up to 14.29% from 3,458 MWt to 3,952 MWt. The radiation levels in the spent fuel are therefore assumed to increase by 14.29% due to EPU. This increase is acceptable as compared to worst case area dose limits.

The design of spent fuel pools is typically very conservative from the perspective of radiation exposure such that changes in the fuel inventory/bundle surface dose rate of 14.29% results in inconsequential changes in operating dose. The current Browns Ferry radiation procedures and radiation monitoring program would detect any changes in radiation levels and initiate appropriate actions. Therefore, the radiation levels around the SFP meet all CLTR dispositions.

2.5.3.1.4 Fuel Racks

The fuel racks at Browns Ferry are generically addressed in the Section 6.3.4 of the CLTR.

The increase in decay heat from EPU results in a higher heat load in the fuel pool during long-term storage. The fuel racks are designed for higher temperatures (212ºF) than the licensing limit of 150ºF.

The fuel racks at Browns Ferry are confirmed to be consistent with the generic description provided in the CLTR because the fuel racks design temperature is greater than the licensing limit.

Conclusion

TVA has evaluated the spent fuel pool cooling and cleanup system and accounted for the effects of the proposed EPU on the spent fuel pool cooling function of the system. The evaluation concludes that the system will continue to provide sufficient cooling capability to cool the spent fuel pool following implementation of the proposed EPU and will continue to meet the requirements of draft GDCs-4, 67, and 69. Therefore, the proposed EPU is acceptable with respect to the spent fuel pool cooling and cleanup system.

2.5.3.2 Station Service Water Systems

Regulatory Evaluation

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

The NRC’s acceptance criteria are based on (1) GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, including flow instabilities and loads (e.g., water hammer), maintenance, testing, and postulated accidents; (2) GDC-5, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions; and (3) GDC-44, insofar as it requires that a system with the capability to transfer heat loads from safety-related SSCs to a heat sink under both normal operating and accident conditions be provided.

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

Browns Ferry Current Licensing Basis

The General Design Criteria (GDC) listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Browns Ferry Nuclear Plant (Browns Ferry) principal design criteria predate these criteria. The Browns Ferry principal design criteria are listed in UFSAR Section 1.5, “Principal Design Criteria.” In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (Federal Register 32FR10213, July 11, 1967).

Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, the Tennessee Valley Authority (TVA) performed a comparative evaluation of the design basis of Browns Ferry with the AEC proposed General Design Criteria of 1967. The Browns Ferry UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria set out in the July 1967 AEC release. For each group of criteria, there is a statement of TVA's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a table of references to locations in the Browns Ferry UFSAR where there is subject matter relating to the intent of that particular criteria.

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

The Browns Ferry design includes three open loop cooling water systems. The Raw Cooling Water System supplies water to the Reactor and Turbine Buildings for cooling. The Plant Service Water System is described in Browns Ferry UFSAR Section 10.7, "Raw Cooling Water System."

The Residual Heat Removal Service Water System is provided to remove the heat rejected by the residual heat removal system during normal shutdown and accident operations. In addition this system provides a source of water for the Emergency Equipment Cooling Water System. The Residual Heat Removal Service Water System is described in Browns Ferry UFSAR Section 10.9, "RHR Service Water System."

The Emergency Equipment Cooling Water System is provided to remove the heat rejected by the equipment that must operate under accident conditions. The Emergency Equipment Cooling Water System is described in Browns Ferry UFSAR Section 10.10, "Emergency Equipment Cooling Water System."

Browns Ferry's current licensing basis regarding GL 89-13 is discussed in TVA's response to the NRC by letter dated March 16, 1990, "Response to Generic Letter 89-13 Service Water Problems Affecting Safety-Related Equipment." Browns Ferry's current licensing basis regarding GL 96-06 is discussed in TVA's response to the NRC, "Browns Ferry Revision 1- Response to Generic Letter 96-06 - Assurance of Equipment Operability and Containment Integrity During Design Basis Accident Conditions," dated October 23, 1997.

Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Browns Ferry License Renewal Safety Evaluation Report (SER), NUREG-1843, dated April 2006 (Reference 11). The license renewal evaluation associated with the Emergency Service Water

System and Residual Heat Removal Service Water System is documented in NUREG-1843, Section 2.3.3.3. The license renewal evaluation associated with the Plant Service Water System is documented in NUREG-1843, Section 2.3.3.5. Management of aging effects on the Emergency Equipment Cooling Water System, Residual Heat Removal Service Water System, and Plant Service Water System is documented in NUREG-1843, Section 3.3.2.

Technical Evaluation

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

The EECW system includes pumps, valves, piping and instrumentation to provide cooling water from the Ultimate Heat Sink to safety-related plant equipment and backup cooling water to non-essential plant equipment. The EECW system is safety-related and is designed to operate during design basis events. The EECW System provides backup cooling flow to the RCW System. The EECW supply valves to the RCW System automatically isolate on low EECW header pressure to guarantee adequate flow to the essential components.

The non-safety-related RCW system provides screened and chemically treated once through cooling water to various non-safety-related plant systems, components, and space coolers. The RCW system may also be operated during loss of power conditions only when standby diesel-generated power reserve margin is available. The RCW system includes pumps, valves, piping and instrumentation that provide cooling water to various non-safety-related systems and components, including the turbine-associated equipment heat exchangers and RBCCW heat exchangers.

The non-safety-related RSW system supplies river water for yard-watering, cooling for plant equipment which the RCW system may not conveniently serve, and to function as a keep-fill system for the raw water Fire Protection System.

The RHRSW system pumps and associated piping and valves are safety-related and provide cooling water from the Ultimate Heat Sink to the RHR heat exchangers during normal shutdown, flood conditions, and during post-accident conditions (LOCA).

Browns Ferry meets all CLTR dispositions. The topics addressed in this evaluation are:

Topic	CLTR Disposition	Browns Ferry Result
Water Systems Performance (Safety-Related)	Plant Specific	Meets CLTR Disposition
Water Systems Performance (Normal Operation)	Plant Specific	Meets CLTR Disposition
Suppression Pool Cooling (RHR Service Operation)	Plant Specific	Meets CLTR Disposition

2.5.3.2.1 Water System Performance (Safety-Related)

As explicitly stated in Section 6.4 of the CLTR, EPU results in a greater decay heat rate which increases the safety-related water systems cooling requirement during accident conditions. The performance of safety-related service water systems during and immediately following the most limiting design basis event, the LOCA, is not significantly affected by reactor power. For DBA-LOCA conditions, the RHRSW heat loads will increase slightly due to an increase in maximum suppression pool temperature from 172.1°F to 179.0°F for EPU. For normal shutdown, the maximum RHRSW heat loads will not increase for EPU because the associated pressure and temperature process conditions for normal shutdown cooling are not changing from CLTP to EPU.

The safety-related portions of the RHRSW and Emergency Equipment Cooling Water Systems are designed to provide a reliable supply of cooling water during and following a DBA, design basis flood, or loss of offsite power conditions, for the following essential equipment and systems:

Services which have increased heat loads with EPU:

- RHR Heat Exchangers
- RBCCW Heat Exchangers*
- RHR Pumps Room Coolers
- CS Pump Room Coolers

Services for which heat loads are not dependent on RTP:

- Emergency Diesel Generator (EDG) Heat Exchangers (Jacket Water, Air, and Lube Oil Coolers)
- Standby Coolant Supply System (Emergency RHRSW cross-connect to RHR system to provide reactor core or primary containment cooling if RHR is lost)
- Supplemental Cooling to SFP

- Makeup flow to the SFP*
- Unit 3 Electric Board Room air conditioning unit
- Unit 3 Control Bay Chillers
- Unit 3 Shutdown Board Room Chillers
- Control Air Compressors*
- Unit 1/2 Emergency Condensing Unit*

*Denotes non-essential load

The increase in heat load to the RHR Pump Room Coolers and CS Room Coolers is a result of a post-LOCA increase in room temperature in each area. This increase in room temperature will slightly increase the EECW discharge temperature but will not be significant as the room temperatures increase is negligible.

The increase in heat load to the RBCCW Heat Exchangers results in a negligible temperature increase.

Control Air Compressors and RBCCW heat exchangers are normally serviced by non-safety-related RCW. EECW provides backup water in the case of a RCW failure. These loads isolate on low EECW header pressure to ensure flow to the essential EECW loads.

The EECW system flow rates, and thereby flow velocities, remain unchanged due to EPU.

The EECW and RHRSW systems were evaluated for changes due to EPU and are adequate as currently designed. Therefore, the EECW and RHRSW meet all CLTR dispositions.

2.5.3.2.2 Water System Performance (Normal Operation)

As stated in Section 6.4 of the CLTR, EPU results in an increased heat load during normal operation.

The increased non-safety related RCW system heat loads at EPU are due primarily to the increase in heat loads from the isolated phase bus duct heat exchanger(s), certain turbine building pump area coolers and condensate booster pump motor cooler(s). Plant modifications to rerate the main generator have been implemented to accommodate EPU. The main generator stator cooling water and hydrogen cooler heat loads for the uprated main generator (See PUSAR Section 2.5.1.2.2) are bounded by heat loads for the original generator rating. This is because the uprated generator reactive power output, the primary contributor to stator cooling and hydrogen cooler heat load, is constrained to be less than the reactive power output of the original generator rating. Additionally, the Isophase Bus Duct modifications increased the RCW flow. The RCW system is capable of providing the additional flow. With these increased heat loads, the RCW system discharge temperature increases approximately 0.1°F at EPU RTP. Therefore, the RCW system is expected to meet the requirements of the system with respect to heat loads and flow due to EPU because the RCW system temperature increase at EPU is negligible.

There are no power dependent loads on the RSW system, and therefore there are no heat load increases due to EPU.

Therefore, RCW and RSW performance during normal operation meets all CLTR dispositions.

2.5.3.2.3 Suppression Pool Cooling (RHR Service Water Operation)

As stated in Section 6.4 of the CLTR, EPU results in a greater decay heat rate.

The containment cooling analysis in Section 2.6.5 shows that the post-LOCA RHR heat load increases due in part to an increase in reactor decay heat. The post-LOCA containment and suppression pool responses have been calculated based on an energy balance between the post-LOCA heat loads and the heat removal capacity of the RHR and RHRSW. The containment cooling analysis and equipment review demonstrate that the suppression pool temperature can be maintained within acceptable limits in the post-accident condition at EPU based on the existing capability of the RHRSW system. The EPU post-accident containment system response results in an increase in the maximum Suppression Pool temperature from 172.1°F to 179°F. The containment cooling analysis results in a total heat load rejected to the RHRSW system due to post-accident suppression pool cooling of 74.2 MBtu/hr/in-service RHR Heat Exchanger. The maximum RHRSW fluid outlet temperature during suppression pool cooling from the RHR heat exchangers will increase to 133.4°F, which remains below the 150°F design temperature for the RHRSW discharge piping. The RHRSW system transfers heat to the ultimate heat sink (UHS), which is addressed in Section 2.5.3.4. Therefore, Suppression Pool Cooling meets all CLTR dispositions.

<u>Conclusion</u>

TVA has evaluated the effects of the proposed EPU on the station service water system including any increased heat loads on system performance that would result from the proposed EPU. The evaluation indicates that the station service water systems will continue to provide sufficient cooling for SSCs important to safety following implementation of the proposed EPU. Therefore, the station service water systems will continue to meet the requirements of draft GDCs-4, 40 and 42. Additionally, the Browns Ferry GL 89-13 Program (i.e., scope, maintenance, and testing) to manage and monitor raw water cooling systems and the Browns Ferry GL 96-06 Program to ensure equipment operability and containment integrity during design basis accident conditions, are not affected by the proposed EPU. Based on the above, the proposed EPU is acceptable with respect to the station service water systems.

2.5.3.3 Reactor Auxiliary Cooling Water Systems

<u>Regulatory Evaluation</u>

These systems include closed-loop auxiliary cooling water systems for reactor system components, reactor shutdown equipment, ventilation equipment, and components of the ECCS.

The NRC's acceptance criteria for the reactor auxiliary cooling water system are based on (1) GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal

operation including flow instabilities and attendant loads (i.e., water hammer), maintenance, testing, and postulated accidents; (2) GDC-5, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions; and (3) GDC-44, insofar as it requires that a system with the capability to transfer heat loads from safety-related SSCs to a heat sink under both normal operating and accident conditions be provided.

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

Browns Ferry Current Licensing Basis

The General Design Criteria (GDC) listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Browns Ferry Nuclear Plant (Browns Ferry) principal design criteria predate these criteria. The Browns Ferry principal design criteria are listed in UFSAR Section 1.5, “Principal Design Criteria.” In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, the Tennessee Valley Authority (TVA) performed a comparative evaluation of the design basis of Browns Ferry with the AEC proposed General Design Criteria of 1967. The Browns Ferry UFSAR, Appendix A, “Conformance to AEC Proposed General Design Criteria,” contains this comparative evaluation. This evaluation discusses each of the groups of criteria set out in the July 1967 AEC release. For each group of criteria, there is a statement of TVA’s understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a table of references to locations in the Browns Ferry UFSAR where there is subject matter relating to the intent of that particular criteria.

While Browns Ferry is not generally licensed to the final GDC or the 1967 AEC proposed General Design Criteria, a comparison of the final GDC to the applicable AEC proposed General Design Criteria can usually be made. For the final GDC listed in the Regulatory Evaluation above, with the exception of final GDC-44, the Browns Ferry comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as “draft GDC”) is contained in Browns Ferry UFSAR Appendix A: draft GDCs-4, 40 and 42.

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

The Reactor Building Closed Cooling Water System is described in Browns Ferry UFSAR Section 10.6, “Reactor Building Closed Cooling Water System.”

Browns Ferry’s current licensing basis regarding GL 89-13 is discussed in TVA’s response to the NRC by letter dated March 16, 1990, “Response to Generic Letter 89-13 Service Water Problems Affecting Safety-Related Equipment.” Browns Ferry’s current licensing basis regarding GL 96-06 is discussed in TVA’s response to the NRC, “Browns Ferry Revision 1-

Response to Generic Letter 96-06 - Assurance of Equipment Operability and Containment Integrity During Design Basis Accident Conditions," dated October 23, 1997.

Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Browns Ferry License Renewal Safety Evaluation Report (SER), NUREG-1843, dated April 2006 (Reference 11). The license renewal evaluation associated with the Reactor Building Closed Cooling Water System is documented in NUREG-1843, Section 2.3.3.22. Management of the effects of aging on the RBCCW system is documented in NUREG-1843, Section 3.3.2.

Technical Evaluation

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

Browns Ferry meets all CLTR dispositions. The topics addressed in this evaluation are:

Topic	CLTR Disposition	Browns Ferry Result
Water Systems Performance (Non-Safety Related)	Plant Specific	Meets CLTR Disposition

The non-safety-related Reactor Auxiliary Cooling Water system includes the RBCCW system. The safety-related and normal operation water systems are evaluated in Section 2.5.3.2, Station Service Water Systems.

Reactor Building Closed Cooling Water System

EPU increases the heat loads on the RBCCW system, but due to an overly conservative analysis that was performed for CLTP, the computed heat load for EPU is decreased. The RBCCW heat loads are mainly dependent on the reactor vessel temperature and/or flow rates in the systems cooled by the RBCCW. The flow rates in the RBCCW system do not change due to EPU. The only component heat load increase at EPU conditions is an estimated 6.5% increase in Reactor Recirculation Pump and motor heat load. The remaining heat loads remain the same or decrease due to excessive conservatism in the CLTP heat load analysis. There are negligible changes to system operating temperatures and pressures as a result of EPU. There are no changes to RBCCW System operation. The RBCCW system contains sufficient redundancy in pumps and heat exchangers to ensure that adequate heat removal capability is available during normal operation.

Therefore, RBCCW meets all CLTR dispositions.

Conclusion

TVA has evaluated the effects of the proposed EPU on the reactor auxiliary cooling water systems including any increased heat loads from the proposed EPU on system performance. The

evaluation indicates that the reactor auxiliary cooling water systems will continue to provide sufficient cooling for SSCs important to safety following implementation of the proposed EPU. Therefore, the reactor auxiliary cooling water systems will continue to meet the requirements of draft GDCs-4, 40 and 42. Additionally, the Browns Ferry GL 89-13 Program (i.e., scope, maintenance, and testing) to manage and monitor raw water cooling systems and the Browns Ferry GL 96-06 Program to ensure equipment operability and containment integrity during design basis accident conditions, are not affected by the proposed EPU. Based on the above, the proposed EPU is acceptable with respect to the reactor auxiliary cooling water systems.

2.5.3.4 Ultimate Heat Sink

Regulatory Evaluation

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

The NRC's acceptance criteria for the UHS are based on (1) GDC-5, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions; and (2) GDC-44, insofar as it requires that a system with the capability to transfer heat loads from safety-related SSCs to a heat sink under both normal operating and accident conditions be provided.

Specific NRC review criteria are contained in SRP Section 9.2.5.

Browns Ferry Current Licensing Basis

The General Design Criteria (GDC) listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Browns Ferry Nuclear Plant (Browns Ferry) principal design criteria predate these criteria. The Browns Ferry principal design criteria are listed in UFSAR Section 1.5, "Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, the Tennessee Valley Authority (TVA) performed a comparative evaluation of the design basis of Browns Ferry with the AEC proposed General Design Criteria of 1967. The Browns Ferry UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria set out in the July 1967 AEC release. For each group of criteria, there is a statement of TVA's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a table of references to locations in the Browns Ferry UFSAR where there is subject matter relating to the intent of that particular criteria.

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

above, with the exception of final GDC-44, the Browns Ferry comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Browns Ferry UFSAR Appendix A: draft GDC-4. There is no draft GDC directly associated with final GDC-44.

The Wheeler Reservoir/Tennessee River serves as the ultimate heat sink for the plant. The ultimate heat sink temperature limit is described in Browns Ferry UFSAR Section 14.6.3.3.2.3, "Long-Term Response."

Technical Evaluation

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

Browns Ferry meets all CLTR dispositions. The topics addressed in this evaluation are:

Topic	CLTR Disposition	Browns Ferry Result
Ultimate Heat Sink	Plant Specific	Meets CLTR Disposition

The Browns Ferry UHS is the Wheeler Reservoir/Tennessee River. The maximum allowable supply temperature from the UHS is 95°F, which is governed by the limits established in TS 3.7.2. The UHS temperature limit is not affected by EPU. EPU will have no effect on the UHS as a source of cooling water for EECW and RHRSW systems, which dissipate reactor decay heat and essential cooling loads during normal or emergency reactor shutdowns.

The Browns Ferry design includes an UHS which provides heat removal capability for safe reactor shutdown in the event of the site related natural phenomena and failures of man-made structures associated with the safety evaluation of the UHS. The UHS safety function is to provide sufficient cooling water to support 1 accident unit and 2 units in shutdown for at least 30 days. The UHS must remain capable of withstanding the following events without loss of safety function: the most severe single natural phenomena expected at the site, the site-related event (e.g., transportation accident, river diversion), reasonable combinations of less severe natural phenomena and/or site related event, or a single failure of a manmade structure.

The EECW and RHRSW systems use the UHS to provide cooling water during accident and shutdown. They are capable of meeting their requirements at EPU with this heat sink.

As explicitly stated in Section 6.4 of the CLTR, EPU results in increased heat load during normal operation and a greater decay heat rate, which increases the safety-related water systems cooling requirements during accident conditions.

The RHR heat exchanger heat load increase, along with other smaller increases discussed in Section 2.5.3.2 must be accommodated by the UHS at EPU. The UHS is operated so that the present limits (e.g., UHS maximum temperature, minimum Wheeler Reservoir level, and minimum Tennessee River flow rate) are not changed or exceeded as a result of EPU.

The UHS was evaluated for its capability to handle the increased EPU heat load. The evaluation demonstrates that UHS can maintain the cooling water supplied within the design basis minimum water level and minimum flow rate. EPU has no effect on the UHS design function. Therefore, UHS meets all CLTR dispositions.

Conclusion

The effects that the proposed EPU would have on the UHS safety function have been reviewed. The proposed EPU will not compromise the design basis safety function of the UHS. The UHS will continue to satisfy the requirements of the current licensing basis following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to the UHS.

2.5.4 Balance-of-Plant Systems

2.5.4.1 Main Steam

Regulatory Evaluation

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

The NRC's acceptance criteria for the MS are based on (1) GDC-4, insofar as it requires that SSCs important to safety be protected against dynamic effects, including the effects of missiles, pipe whip, and jet impingement forces associated with pipe breaks; and (2) GDC-5, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions.

Specific NRC review criteria are contained in SRP Section 10.3.

Browns Ferry Current Licensing Basis

The General Design Criteria (GDC) listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Browns Ferry Nuclear Plant (Browns Ferry) principal design criteria predate these criteria. The Browns Ferry principal design criteria are listed in UFSAR Section 1.5, "Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, the Tennessee Valley Authority (TVA) performed a comparative evaluation of the design basis of Browns Ferry with the AEC proposed General Design Criteria of 1967. The Browns Ferry UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria set out in the July 1967 AEC release. For each group of criteria, there is a statement of TVA's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with

the intent of the group of criteria. Following a restatement of each of the proposed criteria is a table of references to locations in the Browns Ferry UFSAR where there is subject matter relating to the intent of that particular criteria.

While Browns Ferry is not generally licensed to the final GDC or the 1967 AEC proposed General Design Criteria, a comparison of the final GDC to the applicable AEC proposed General Design Criteria can usually be made. For the final GDC listed in the Regulatory Evaluation above, the Browns Ferry comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as “draft GDC”) is contained in Browns Ferry UFSAR Appendix A: draft GDCs-4 and 40.

Main steam piping is discussed in several UFSAR sections including Chapter 4, “Reactor Coolant System.”

Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Browns Ferry License Renewal Safety Evaluation Report (SER), NUREG-1843, dated April 2006 (Reference 11). The license renewal evaluation associated with the main steam piping is documented in NUREG-1843, Section 2.3.4. Management of the effects of aging on the main steam piping is documented in NUREG-1843, Section 3.4.2.

Technical Evaluation

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

2.5.4.1.1 Structural Evaluation of Main Steam Piping

NEDC-33004P-A, Revision 4, “Constant Pressure Power Uprate,” Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of EPUs. Section 3.4.1 of the CLTR addresses the effect of EPU on flow induced vibration in the MSL. The results of this evaluation are described below.

Browns Ferry meets all CLTR dispositions. The topics addressed in this evaluation are:

Topic	CLTR Disposition	Browns Ferry Result
Structural Evaluation of Main Steam Piping	Generic	Meets CLTR Disposition

The CLTR states that because the MS piping pressures and temperatures are not affected by EPU, there is no effect on the analyses for these parameters. Seismic inertia loads, seismic

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

The capability of the MS piping to withstand dynamic loads at EPU conditions was evaluated. A summary of the results of the MS piping system evaluation that contains the increased loading associated with EPU conditions (i.e., temperature, pressure, and flow, including the effects of the MS flow induced transient loads at EPU conditions) along with a comparison to the code allowable limits is provided in Section 2.2.2.

SRV setpoint tolerance is independent of an EPU. Browns Ferry transient analyses conservatively bound the existing SRV setpoint tolerance ALs. Actual historical in-service surveillance of SRV setpoint performance test results are monitored separately for compliance to the TSs and In-Service Testing program.

Browns Ferry has an ongoing evaluation program to resolve problems resulting in SRV surveillance testing exceeding the 3% tolerance.

Increased MSL flow may affect vibration of the piping during normal operation. The vibration frequency, extent, and magnitude depend upon plant-specific parameters, valve locations, the valve design, and piping support arrangements. The effects of EPU on Flow-Induced Vibration (FIV) of the piping will be assessed by vibration testing during initial plant operation at the higher steam flow rates. This topic is addressed in Section 2.2.2.1.2. Attachment 45 to the EPU license amendment request contains details of the vibration monitoring program.

FIV may increase incidents of SRV leakage. Browns Ferry currently has procedures and installed instrumentation in place to detect and take actions concerning SRV seat leakage. These procedures and installed instrumentation are considered acceptable to monitor for SRV seat leakage at EPU rated steam flow conditions. TVA has conducted drywell vibration studies directly related to SRV standpipes and branch connections and the effects of acoustic resonance. This has resulted in installation of acoustic vibration suppressors. This is to ensure that SRV vibration resulting from acoustic resonance is not expected at EPU operating conditions.

Therefore, the structural evaluation of MS piping meets all CLTR dispositions.

2.5.4.1.2 Main Steam Line Flow Restrictors

NEDC-33004P-A, Revision 4, “Constant Pressure Power Uprate,” Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of EPUs. Section 3.7 of the CLTR addresses the effect of EPU on the MSL flow restrictors. The results of this evaluation are described below.

Browns Ferry meets all CLTR dispositions. The topics addressed in this evaluation are:

Topic	CLTR Disposition	Browns Ferry Result
Structural Integrity	Generic	Meets CLTR Disposition

The CLTR states that at uprated power, the flow restrictors are required to pass a higher flow rate, which will result in an increased pressure drop.

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

After a postulated steam line break outside containment, the fluid flow in the broken steam line increases until it is limited by the MSL flow restrictor. [[

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The Browns Ferry restrictors were originally analyzed for these flow conditions and therefore the restrictors remain within the acceptable calculated differential pressure drop and choke flow limits under EPU conditions. Therefore, the flow restrictors meet all CLTR dispositions.

Conclusion

TVA has evaluated the effects of the proposed EPU on MS including the effects of changes in plant conditions on the design of MS. The evaluation indicates that the system will continue to meet the requirements of draft GDCs-4 and 40. Therefore, the proposed EPU is acceptable with respect to MS.

2.5.4.2 Main Condenser

Regulatory Evaluation

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

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

Specific NRC review criteria are contained in SRP Section 10.4.1.

Browns Ferry Current Licensing Basis

The General Design Criteria (GDC) listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Browns Ferry Nuclear Plant (Browns Ferry) principal design criteria predate these criteria. The Browns Ferry principal design criteria are listed in UFSAR Section 1.5, "Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, the Tennessee Valley Authority (TVA) performed a comparative evaluation of the design basis of Browns Ferry with the AEC proposed General Design Criteria of 1967. The Browns Ferry UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria set out in the July 1967 AEC release. For each group of criteria, there is a statement of TVA's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a table of references to locations in the Browns Ferry UFSAR where there is subject matter relating to the intent of that particular criteria.

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

The main condenser system is described in Browns Ferry UFSAR Section 11.3, "Main Condenser System."

Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Browns Ferry License Renewal Safety Evaluation Report (SER), NUREG-1843, dated April 2006 (Reference 11). The license renewal evaluation associated with the main condenser system is documented in NUREG-1843, Section 2.3.4. The management of the effects of aging on the main condenser system is documented in NUREG-1843, Section 3.4.2.

Technical Evaluation

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

Browns Ferry meets all CLTR dispositions. The topics addressed in this evaluation are:

Topic	CLTR Disposition	Browns Ferry Result
Condenser and SJAE	Plant Specific	Meets CLTR Disposition

As stated in the CLTR, the increase in steam flow increases the heat removal requirement for the condenser. The additional power level increases the non-condensable gases generated by the reactor.

The main condenser is designed to reject heat to the circulating water system and thereby maintain adequately low condenser pressure as recommended by the turbine vendor. Maintaining adequately low condenser pressure assures the efficient operation of the turbine-generator and minimizes wear on the turbine last stage blades.

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

The main condenser is not being modified for EPU operation. The performance of the condenser was evaluated for EPU. This evaluation was based on a design duty over the actual range of circulating water inlet temperatures, and confirms that the condenser backpressure remains below the high alarm setpoint, and the turbine trip setpoint during normal operation. Condenser backpressure limitations may require load reductions at the upper range of the anticipated circulating water inlet temperatures.

Main condenser storage capacity has been evaluated for hotwell retention time and found to be acceptable for EPU operation. The holdup time for the decay of short-lived radioisotopes (primarily N-16) remains a conservative decay time and is acceptable for EPU operation.

The absolute value in lbm/hr of the steam bypassed to the main condenser during a load rejection event is not increased for EPU as discussed in FUSAR Section 2.5.4.2.

Therefore, the Condenser and Steam Jet Air Ejectors for Browns Ferry meet all CLTR dispositions.

Conclusion

TVA has considered the effects of the proposed EPU with ATRIUM 10XM fuel on the main condenser system. It is concluded that the main condenser system will continue to maintain its ability to withstand the blowdown effects of the steam from the TBS and thereby continue to meet the current licensing basis with respect to controlling releases of radioactive effluents. Therefore, the proposed EPU with ATRIUM 10XM fuel is acceptable with respect to the main condenser system.

2.5.4.3 Turbine Bypass

Regulatory Evaluation

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

The NRC's acceptance criteria for the TBS are based on (1) GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents (including pipe breaks or malfunctions of the TBS), and (2) GDC-34, insofar as it requires that a RHR system be provided to transfer fission product decay heat and other residual heat from the reactor core at a rate such that SAFDLs and the design conditions of the RCPB are not exceeded.

Specific NRC review criteria are contained in SRP Section 10.4.4.

Browns Ferry Current Licensing Basis

The General Design Criteria (GDC) listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Browns Ferry Nuclear Plant (Browns Ferry) principal design criteria predate these criteria. The Browns Ferry principal design criteria are listed in UFSAR Section 1.5, "Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, the Tennessee Valley Authority (TVA) performed a comparative evaluation of the design basis of Browns Ferry with the AEC proposed General Design Criteria of 1967. The Browns Ferry UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria set out in the July 1967 AEC release. For each group of criteria, there is a statement of TVA's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a table of references to locations in the Browns Ferry UFSAR where there is subject matter relating to the intent of that particular criteria.

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

Appendix A: draft GDCs-40 and 42. There is no draft GDC directly associated with final GDC-34.

The TBS is described in Browns Ferry UFSAR Section 11.5, "Turbine Bypass System."

Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Browns Ferry License Renewal Safety Evaluation Report (SER), NUREG-1843, dated April 2006 (Reference 11). The TBS is included in the discussion of the license renewal evaluation for the Main Steam System. That discussion can be found in NUREG-1843, Section 2.3.4. Management of aging effects on the Main Steam System is documented in NUREG-1843, Section 3.4.2.

Technical Evaluation

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

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

Browns Ferry meets all CLTR dispositions. The topics addressed in this evaluation are:

Topic	CLTR Disposition	Browns Ferry Result
Turbine Steam Bypass (Safety Analysis)	Generic	Meets CLTR Disposition

The CLTR states that the increase in steam flow reduces the relative capacity of the Turbine Steam Bypass System.

See FUSAR Section 2.5.4.3 for the TBS safety analysis effect.

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

The turbine bypass valves are rated for a total steam flow capacity of not less than 25% of the rated reactor steam flow, or 3.5 Mlbm/hr. Each of nine bypass valves is designed to pass a steam flow of 389,000 lbm/hr and does not change at EPU RTP. At EPU conditions, rated reactor steam flow is 16.44 Mlbm/hr, resulting in a bypass capacity of 21.3% of EPU rated steam flow. The bypass capacity at Browns Ferry remains adequate for normal operational flexibility at EPU RTP.

The bypass capacity is used as an input to the reload analysis process for the evaluation of transient events that credit the Turbine Steam Bypass System.

Therefore, the Browns Ferry steam bypass capacity used in the turbine steam bypass safety analysis meets all CLTR dispositions.

Conclusion

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

2.5.4.4 Condensate and Feedwater

Regulatory Evaluation

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

The NRC's acceptance criteria for the condensate and feedwater system are based on (1) GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation including possible fluid flow instabilities (e.g., water hammer), maintenance, testing, and postulated accidents; (2) GDC-5, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions; and (3) GDC-44, insofar as it requires that a system with the capability to transfer heat loads from safety-related SSCs to a heat sink under both normal operating and accident conditions be provided, and that the system be provided with suitable isolation capabilities to assure the safety function can be accomplished with electric power available from only the onsite system or only the offsite system, assuming a single failure.

Specific NRC review criteria are contained in SRP Section 10.4.7.

Browns Ferry Current Licensing Basis

The General Design Criteria (GDC) listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Browns Ferry Nuclear Plant (Browns Ferry) principal design criteria predate these criteria. The Browns Ferry principal design criteria are listed in UFSAR Section 1.5, "Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, the Tennessee Valley Authority (TVA) performed a comparative evaluation of the design basis of Browns Ferry with the AEC proposed General Design Criteria of 1967. The Browns Ferry UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria set out in the July 1967 AEC release. For each group of criteria, there is a statement of TVA's understanding

of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a table of references to locations in the Browns Ferry UFSAR where there is subject matter relating to the intent of that particular criteria.

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

The condensate and feedwater system is described in Browns Ferry UFSAR Section 11.8, "Condensate and Reactor Feedwater Systems." The condensate demineralizer system is described in Browns Ferry UFSAR Section 11.7, "Condensate Filter-Demineralizer System."

Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Browns Ferry License Renewal Safety Evaluation Report (SER), NUREG-1843, dated April 2006 (Reference 11). The license renewal evaluation associated with the condensate and feedwater system is documented in NUREG-1843, Section 2.3.4. The management of the effects of aging on the condensate and feedwater system is documented in NUREG-1843, Section 3.4.2.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of EPUs. Section 7.4 of the CLTR addresses the effect of EPU on the Condensate and FW Systems. The results of this evaluation are described below.

Browns Ferry meets all CLTR dispositions. The topics addressed in this evaluation are:

Topic	CLTR Disposition	Browns Ferry Result
FW and Condensate Systems	Plant Specific	Meets CLTR Disposition

The CLTR states that the increase in power level increases the FW requirements of the reactor.

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

Normal Operation

System operating flows at EPU increase approximately 16% of rated flow at the CLTP. The condensate and FW systems will be modified to ensure acceptable performance with the new system operating conditions. See LAR Attachment 47 for modifications description.

Transient Operation

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

The FW system post-feed pump trip capacity was evaluated to confirm that with the modifications to the FW and condensate system configurations, the capability to supply the transient flow requirements is maintained or increased. A transient analysis was performed to determine the reactor level response following a single FW pump trip. The results of the analysis in FUSAR Section 2.8.5.2.3.2 show that the system response is adequate during EPU conditions.

Condensate Demineralizers

The condensate filter demineralizers (CFD) are acceptable for EPU. The system experiences slightly higher loadings resulting in slightly reduced CFD run times. However, the reduced run times are acceptable (refer to Section 2.5.5 for the effects on the radwaste systems).

Therefore, the FW and condensate systems meet all CLTR dispositions.

Conclusion

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

2.5.5 Waste Management Systems

2.5.5.1 Gaseous Waste Management Systems

Regulatory Evaluation

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

The NRC's acceptance criteria for gaseous waste management systems are based on (1) 10 CFR 20.1302, insofar as it provides for demonstrating that annual average concentrations of radioactive materials released at the boundary of the unrestricted area do not exceed specified values; (2) GDC-3, insofar as it requires that (a) SSCs important to safety be designed and located to minimize the probability and effect of fires, (b) noncombustible and heat resistant materials be used, and (c) fire detection and fighting systems be provided and designed to minimize the adverse effects of fires on SSCs important to safety; (3) GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents; (4) GDC-61, insofar as it requires that systems that contain radioactivity be designed with appropriate confinement; and (5) 10 CFR Part 50 Appendix I, Sections II.B, II.C, and II.D, which set numerical guides for design objectives and limiting conditions for operation to meet the "as low as is reasonably achievable" (ALARA) criterion.

Specific NRC review criteria are contained in SRP Section 11.3.

Browns Ferry Current Licensing Basis

The General Design Criteria (GDC) listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Browns Ferry Nuclear Plant (Browns Ferry) principal design criteria predate these criteria. The Browns Ferry principal design criteria are listed in UFSAR Section 1.5, "Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, the Tennessee Valley Authority (TVA) performed a comparative evaluation of the design basis of Browns Ferry with the AEC proposed General Design Criteria of 1967. The Browns Ferry UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria set out in the July 1967 AEC release. For each group of criteria, there is a statement of TVA's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a table of references to locations in the Browns Ferry UFSAR where there is subject matter relating to the intent of that particular criteria.

While Browns Ferry is not generally licensed to the final GDC or the 1967 AEC proposed General Design Criteria, a comparison of the final GDC to the applicable AEC proposed General Design Criteria can usually be made. For the final GDC listed in the Regulatory Evaluation above, the Browns Ferry comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Browns Ferry UFSAR Appendix A: draft GDCs-69 and 70. Final GDC-3 is applicable to Browns Ferry as described in the Browns Ferry Fire Protection Report, Volume 1, Revision 20.

The gaseous waste management system is described in Browns Ferry UFSAR Section 9.4, "Gaseous Radwaste System" and Section 9.5, "Gaseous Radwaste System (Modified)."

Technical Evaluation

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

Browns Ferry meets all CLTR dispositions. The topics addressed in this evaluation are:

Topic	CLTR Disposition	Browns Ferry Result
Offsite Release Rate	Generic	Meets CLTR Disposition
Recombiner Performance	Generic	Meets CLTR Disposition

2.5.5.1.1 Offsite Release Rate

The CLTR states that under EPU conditions, offgas system functions other than the recombiner and related components, [[ ]]

The Browns Ferry site-specific CLTP design basis radiolytic gas production rate, 0.070 cfm/MWt, is greater than or equal to [[

]].

These are constant rates. As these rates are proportional to reactor power in each unit, the radiolytic gas flow rate is expected to increase in proportion to the change in power, approximately 20% under EPU conditions as compared to OLTP. Because the actual radiolytic gas flow rate at EPU conditions is within the design basis (radiolytic gas) flow rate at OLTP, the design basis production value is acceptable at EPU conditions. As such, the OLTP design basis is maintained at EPU conditions and an evaluation was conducted. This evaluation verified that all structures, systems and components of the offgas system were acceptable for EPU operation.

The primary function of the gaseous waste management system is to process and control the release of gaseous radioactive effluents to the site environs so that the total radiation exposure of persons in offsite areas is within the guideline values of 10 CFR 50 Appendix I.

The offgas system radiological release rate is administratively controlled to remain within existing site release rate limits and is a function of fuel cladding performance, main condenser air in-leakage, charcoal adsorber inlet dew point, and charcoal adsorber temperature. The Browns Ferry TS require administrative controls (i.e., Radioactive Effluent Controls Program) to limit radioactive gas releases to the environment. These controls require plant procedures for addressing fuel cladding failure or high activity in offgas. Such procedures are not affected by

EPU. Further information regarding the production of noble gases at EPU conditions is found in Section 2.9.1.2.

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

The gaseous waste management system methods of treatment for radiological releases from the offgas system consist of holdup and filtration to reduce the gaseous radioactivity that could be potentially released to offsite areas. The capacity and capability of the offgas holdup and filtration system to adequately perform its design function are unchanged by EPU.

The gaseous waste management systems are designed to meet the requirements of 10 CFR 20 and 10 CFR 50 Appendix I in accordance with the Offsite Dose Calculation Manual (ODCM), Reference 74. Browns Ferry compliance with the dose limits to the public of 10 CFR 20 and 10 CFR 50 Appendix I is described in Section 2.10.1.2.4 (Table 2.10-2).

The offsite release rate at Browns Ferry meets all CLTR dispositions.

2.5.5.1.2 Recombiner Performance

The CLTR states that under EPU conditions, core radiolysis increases linearly with reactor thermal power, thus increasing the heat load on the offgas recombiner and related components.

The design features for precluding the possibility of an explosion include: (a) dilution to control the concentration of hydrogen; and (b) catalytic recombination to remove the combustible gas. The gaseous waste management system at Browns Ferry is consistent with GEH design specifications for radiolytic flow rate, and the Browns Ferry-specific value for radiolytic gas production rate is 0.045 cfm/MWt, which is well below the Browns Ferry site specific design value of 0.070 cfm/MWt (130ºF and 1 atm.). Therefore, the recombiner and condenser, as well as downstream system components, are designed to handle the increase in thermal power of the EPU. The gaseous waste management system component design requirements are determined by the quantity of radiolytic hydrogen and oxygen, which is expected to increase in proportion to the EPU power increase. The additional radiolytic hydrogen will also increase the catalytic recombiner temperature and offgas condenser heat load. These increases have been evaluated and it has been confirmed that sufficient margin remains in the Browns Ferry offgas system component design to ensure that the system will continue to satisfy the plant licensing basis.

The recombiner performance at Browns Ferry meets all CLTR dispositions.

Conclusion

TVA has evaluated the gaseous waste management systems and the increase in fission product and amount of gaseous waste on the abilities of the system to control releases of radioactive

materials and preclude the possibility of an explosion if the potential for explosive mixtures exists. The evaluation indicates that the gaseous waste management systems will continue to meet their design functions following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to the gaseous waste management systems.

2.5.5.2 Liquid Waste Management Systems

Regulatory Evaluation

The NRC's acceptance criteria for the liquid waste management systems are based on (1) 10 CFR 20.1302, insofar as it provides for demonstrating that annual average concentrations of radioactive materials released at the boundary of the unrestricted area do not exceed specified values; (2) GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents; (3) GDC-61, insofar as it requires that systems that contain radioactivity be designed with appropriate confinement; and (4) 10 CFR Part 50 Appendix I, Sections II.A and II.D, which set numerical guides for dose design objectives and limiting conditions for operation to meet the ALARA criterion.

Specific NRC review criteria are contained in SRP Section 11.2.

Browns Ferry Current Licensing Basis

The General Design Criteria (GDC) listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Browns Ferry Nuclear Plant (Browns Ferry) principal design criteria predate these criteria. The Browns Ferry principal design criteria are listed in UFSAR Section 1.5, "Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, the Tennessee Valley Authority (TVA) performed a comparative evaluation of the design basis of Browns Ferry with the AEC proposed General Design Criteria of 1967. The Browns Ferry UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria set out in the July 1967 AEC release. For each group of criteria, there is a statement of TVA's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a table of references to locations in the Browns Ferry UFSAR where there is subject matter relating to the intent of that particular criteria.

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

The liquid waste management system is described in Browns Ferry UFSAR Section 9.2, "Liquid Radwaste System."

Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Browns Ferry License Renewal Safety Evaluation Report (SER), NUREG-1843, dated April 2006 (Reference 11). The license renewal evaluation associated with the liquid waste management system is documented in NUREG-1843, Section 2.3.3.25. Management of aging effects on the liquid waste management system is documented in NUREG-1843, Section 3.3.2.

Technical Evaluation

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

As stated in Section 8.1 of the CLTR, the Liquid Radwaste System collects, monitors, processes, stores and returns processed radioactive waste to the plant for reuse or for discharge.

Browns Ferry meets all CLTR dispositions. The topics addressed in this evaluation are:

Topic	CLTR Disposition	Browns Ferry Result
Waste Volumes	Plant Specific	Meets CLTR Disposition
Coolant Fission and Corrosion Product Levels	Plant Specific	Meets CLTR Disposition

2.5.5.2.1 Waste Volumes

The CLTR states that increased power levels and steam flow result in the generation of slightly higher levels of liquid radwaste (about 3.44% for Browns Ferry). The largest sources of liquid waste are from the backwash of condensate and RWCU filter-demineralizers. Other increases in the liquid waste management system (LWMS) loads are minimal. The effect of EPU on the LWMS is primarily a result of the increased load on condensate filter/demineralizers. Similarly, the RWCU filter-demineralizer requires more frequent backwashes due to slightly higher levels of activation and fission products.

Because the RWCU flow rate will remain the same as CLTP, but an increase in contaminate concentration is projected, the RWCU system is projected to experience a slight increase in filter demineralizer backwash frequency. The current capacity of the LWMS can accommodate this small increase.

Because the liquid volume does not increase appreciably for EPU, the current design and operation of the LWMS will accommodate the effects of EPU with no changes. The offsite concentration of liquid effluents at CLTP and EPU conditions meets 10 CFR 20 and the dose from liquid effluents meets 10 CFR 50 Appendix I, as shown in Table 2.10-2. The existing equipment and procedures that control releases to the environment will continue to ensure that releases remain within the applicable guideline values of 10 CFR 20.1302, 10 CFR 50 Appendix I, and 40 CFR 190. Browns Ferry compliance with these dose limits to the public is described in Section 2.10.1.2.4 (Table 2.10-2).

Therefore, the waste volumes meet all CLTR dispositions.

2.5.5.2.2 Coolant Fission and Corrosion Product Levels

The CLTR states that increased power levels and steam flow result in the generation of slightly higher levels of coolant concentrations of fission and corrosion products.

The coolant activation and corrosion products are slightly increased as a result of EPU as discussed in Section 8.4 of the CLTR.

Per the AST submittal, a calculation of activated corrosion and fission products in the reactor coolant was performed in accordance with ANSI/ANS-18.1-1984, "Radioactive Source Term for Normal Operation of Light Water Reactors" (Reference 75). Input parameters that change as a result of EPU conditions include core power, weight of water in the reactor vessel, condensate demineralizer flow rate, and steam flow rate.

The determination of activated corrosion products in the reactor coolant was performed the same way for all three units.

The current design and operation of the LWMS will accommodate the effects of the EPU with no changes. The existing equipment and procedures that control releases to the environment will continue to ensure that releases remain within the applicable guideline values of 10 CFR 20.1302, 10 CFR 50 Appendix I, and 40 CFR 190.

Conclusion

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

2.5.5.3 Solid Waste Management Systems

Regulatory Evaluation

The NRC's acceptance criteria for the solid waste management system are based on (1) 10 CFR 20.1302, insofar as it provides for demonstrating that annual average concentrations of radioactive materials released at the boundary of the unrestricted area do not exceed specified

values; (2) GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents; (3) GDC-63, insofar as it requires that systems be provided in waste handling areas to detect conditions that may result in excessive radiation levels; (4) GDC-64, insofar as it requires that means be provided for monitoring effluent discharge paths and the plant environs for radioactivity that may be released from normal operations, including AOOs, and postulated accidents; and (5) 10 CFR Part 71, which states requirements for radioactive material packaging.

Specific NRC review criteria are contained in SRP Section 11.4.

Browns Ferry Current Licensing Basis

The General Design Criteria (GDC) listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Browns Ferry Nuclear Plant (Browns Ferry) principal design criteria predate these criteria. The Browns Ferry principal design criteria are listed in UFSAR Section 1.5, "Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, the Tennessee Valley Authority (TVA) performed a comparative evaluation of the design basis of Browns Ferry with the AEC proposed General Design Criteria of 1967. The Browns Ferry UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria set out in the July 1967 AEC release. For each group of criteria, there is a statement of TVA's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a table of references to locations in the Browns Ferry UFSAR where there is subject matter relating to the intent of that particular criteria.

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

The solid waste management system is described in Browns Ferry UFSAR Section 9.3, "Solid Radwaste System."

Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Browns Ferry license renewal Safety Evaluation Report (SER), NUREG-1843, dated April 2006 (Reference 11). The license renewal evaluation associated with the solid waste management system is documented in NUREG-1843, Section 2.3.3.25. Management of aging effects on the solid waste management system is documented in NUREG-1843, Section 3.3.2.

Technical Evaluation

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

The Solid Radwaste System collects, monitors, processes, and stores processed radioactive waste prior to offsite disposal. Browns Ferry meets all CLTR dispositions. The topics considered in this section are:

Topic	CLTR Disposition	Browns Ferry Result
Coolant Fission and Corrosion Product Levels	Plant Specific	Meets CLTR Disposition
Waste Volumes	Plant Specific	Meets CLTR Disposition

2.5.5.3.1 Coolant Fission and Corrosion Product Levels

EPU does not change the types of solid radwaste which are generated, or add a new type of solid radwaste, as there are no new inputs being added to the radwaste system, and the radwaste system will not be modified as part of the EPU. The primary source of solid radwaste is in the form of spent resins. However, the resin is replaced based on pressure drop across the demineralizer and conductivity design criteria prior to exceeding radiological criteria. Therefore, any increase in the primary coolant activity will not significantly increase the activity of the spent resin.

The existing equipment and procedures that control waste shipments and releases to the environment will continue to ensure that releases remain within the applicable regulatory guidance.

2.5.5.3.2 Waste Volumes

The CLTR states that increased power levels and steam flow result in the generation of slightly higher levels of liquid and solid radwaste.

The effect of EPU on the Solid Waste Management System (SWMS) is primarily a result of the increased load on condensate filter/demineralizers. The result is that the increase in solid radwaste volume is conservatively considered as up to 15%. Based on previous EPU experience from other plants, there is enough margin between the actual solid radwaste volume and design basis volume to accommodate this increase. The EPU projected usage of the Browns Ferry SWMS process capacity will be approximately 50% of the installed processing capacity.

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

Because the solid volume does not increase appreciably, the current design and operation of the SWMS will accommodate the effects of EPU with no changes, and the existing equipment and procedures that control waste shipments and releases to the environment will continue to ensure that releases remain within the applicable regulatory guidance. Therefore, the waste volumes meet all CLTR dispositions.

Conclusion

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

2.5.6 Additional Considerations

2.5.6.1 Emergency Diesel Engine Fuel Oil Storage and Transfer System

Regulatory Evaluation

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

The NRC's acceptance criteria for the emergency diesel engine fuel oil storage and transfer system are based on (1) GDC-4, insofar as it requires that SSCs important to safety be protected against dynamic effects, including missiles, pipe whip, and jet impingement forces associated with pipe breaks; (2) GDC-5, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions; and (3) GDC-17, insofar as it requires onsite power supplies to have sufficient independence and redundancy to perform their safety functions, assuming a single failure.

Specific NRC review criteria are contained in SRP Section 9.5.4.

Browns Ferry Current Licensing Basis

The General Design Criteria (GDC) listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Browns Ferry Nuclear Plant (Browns Ferry) principal design criteria predate these criteria. The Browns Ferry principal design criteria are listed in UFSAR Section 1.5, "Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, the Tennessee Valley Authority (TVA) performed a comparative evaluation of the design basis of Browns Ferry with the AEC proposed General Design Criteria of 1967. The Browns Ferry

UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria set out in the July 1967 AEC release. For each group of criteria, there is a statement of TVA's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a table of references to locations in the Browns Ferry UFSAR where there is subject matter relating to the intent of that particular criteria.

While Browns Ferry is not generally licensed to the final GDC or the 1967 AEC proposed General Design Criteria, a comparison of the final GDC to the applicable AEC proposed General Design Criteria can usually be made. For the final GDC listed in the Regulatory Evaluation above, the Browns Ferry comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Browns Ferry UFSAR Appendix A: draft GDCs-4 and 40. Final GDC-17 is applicable to Browns Ferry as described in UFSAR Section 8.3.

The Diesel Engine Fuel Oil Storage and Transfer capability is described in Browns Ferry UFSAR Section 8.5, "Standby AC Power Supply and Distribution."

Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Browns Ferry License Renewal Safety Evaluation Report (SER), NUREG-1843, dated April 2006 (Reference 11). The license renewal evaluation associated with the Diesel Engine Fuel Oil Storage and Transfer capability is documented in NUREG-1843, Section 2.3.3.2. Management of aging effects on the Diesel Engine Fuel Oil Storage and Transfer capability is documented in NUREG-1843, Section 3.3.2.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of EPUs. Section 6.8 of the CLTR addresses the effect of EPU on other systems not addressed in the CLTR. It concludes that systems not specifically addressed in the CLTR are not significantly affected by the power uprate. The Emergency Diesel Engine Fuel Oil Storage and Transfer System is not addressed in the CLTR, and this disposition applies to Browns Ferry.

There are no changes to the EDG loads for EPU. EPU conditions are achieved by utilizing existing equipment operating at or below the nameplate rating and within the calculated BHP for the required pump motors. No increase in electrical equipment demand on the EDG's is expected as a result of EPU. Therefore, under emergency conditions, the electrical supply and distribution components are adequate.

No increase in flow or pressure is required of any AC powered ECCS equipment. Therefore, the amount of power required to perform safety-related functions (pump and valve loads) is not increased with EPU, and the current emergency power system remains adequate. The systems

have sufficient capacity to support all required loads to achieve and maintain safe shutdown conditions and to operate the ECCS equipment following postulated accidents and transients.

Because the loads and mission times are not changed for EPU, no changes to the emergency diesel engine fuel oil storage and transfer system are necessary.

Conclusion

TVA has evaluated the required fuel oil for the emergency diesel generators and the effects of any increased electrical demand on fuel oil consumption. The evaluation indicates that the fuel oil storage and transfer system will continue to provide an adequate amount of fuel oil to allow the diesel generators to meet the onsite power requirements of final GDC-17 and draft GDCs-4 and 40. Therefore, the proposed EPU is acceptable with respect to the fuel oil storage and transfer system.

2.5.6.2 Light Load Handling System (Related to Refueling)

Regulatory Evaluation

The light load handling system includes components and equipment used in handling new fuel at the receiving station and the loading of spent fuel into shipping casks.

The NRC's acceptance criteria for the light load handling system are based on (1) GDC-61, insofar as it requires that systems that contain radioactivity be designed with appropriate confinement and with suitable shielding for radiation protection; and (2) GDC-62, insofar as it requires that criticality be prevented.

Specific NRC review criteria are contained in SRP Section 9.1.4.

Browns Ferry Current Licensing Basis

The General Design Criteria (GDC) listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Browns Ferry Nuclear Plant (Browns Ferry) principal design criteria predate these criteria. The Browns Ferry principal design criteria are listed in UFSAR Section 1.5, "Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, the Tennessee Valley Authority (TVA) performed a comparative evaluation of the design basis of Browns Ferry with the AEC proposed General Design Criteria of 1967. The Browns Ferry UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria set out in the July 1967 AEC release. For each group of criteria, there is a statement of TVA's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a table of references to locations in the Browns Ferry UFSAR where there is subject matter relating to the intent of that particular criteria.

While Browns Ferry is not generally licensed to the final GDC or the 1967 AEC proposed General Design Criteria, a comparison of the final GDC to the applicable AEC proposed General Design Criteria can usually be made. For the final GDC listed in the Regulatory Evaluation above, the Browns Ferry comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Browns Ferry UFSAR Appendix A: draft GDCs-66, 68, and 69.

The light load handling system is described in Browns Ferry UFSAR Section 10.3, "Spent Fuel Storage."

Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Browns Ferry License Renewal Safety Evaluation Report (SER), NUREG-1843, dated April 2006 (Reference 11). The license renewal evaluation associated with the light load handling system is documented in NUREG-1843, Section 3.0.3.2.13.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of EPUs. Section 6.8 of the CLTR addresses the evaluation of the effect of the EPU on several plant systems that were not addressed elsewhere in that report. The Light Load Handling System (related to Fuel Handling and Storage System) is one of the systems so evaluated (see Table 2.5-5, Item 18). CLTR Section 6.8 is supported by ELTR1 (Reference 4), Section 5.12 and Appendix J, also previously approved by the NRC for use as guidelines for EPUs. The results of this evaluation are described below.

Browns Ferry meets all CLTR dispositions. The topics addressed in this evaluation are:

Topic	CLTR Disposition	Browns Ferry Result
Fuel Handling and Storage System	Generic	Meets CLTR Disposition

The EPU has been found to not have any significant effect on the Fuel Handling and Storage System. The Fuel Handling and Storage System meets the CLTR disposition.

Conclusion

Implementing EPU does not require introducing any new fuel designs. Therefore, the fuel handling analysis is not affected by EPU. An evaluation of the light load handling system for the proposed EPU is not required. The proposed EPU is acceptable with respect to the light load handling system.

2.5.7 Additional Review Areas (Plant Systems)

NEDC-33004P-A, Revision 4, Constant Pressure Power Uprate, Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the

effects of EPUs. Section 6.8 of the CLTR addresses the evaluation of the effect of the EPU on several plant systems that were not addressed elsewhere in that report. The systems included in this evaluation are listed in Table 2.5-5. CLTR Section 6.8 is supported by ELTR1 (Reference 4), Section 5.12 and Appendix J, also previously approved by the NRC for use as guidelines for EPUs. The results of this evaluation are described below.

Browns Ferry meets all CLTR dispositions. The topics addressed in this evaluation are:

Topic	CLTR Disposition	Browns Ferry Result
Other Systems	Generic	Meets CLTR Disposition

The EPU has been found to not have any significant effect on the systems listed in Table 2.5-5.

The assessment of other systems meets the CLTR disposition.

Table 2.5-1 NFPA 805 Fire Event Key Inputs

Input Parameters	Values
Reactor Thermal Power	3,952 MWt
RPV Dome Pressure	1,055 psia
Decay Heat	ANS 5.1-1979 without 2σ uncertainty adder and with GEH SIL 636 recommendations
Initial Suppression Pool Liquid Volume	124,200 ft^3
Initial Suppression Pool and Wetwell Airspace Temperature	95°F
Initial Wetwell Pressure	14.4 psia
Initial Drywell Pressure	15.5 psia
Initial Drywell Temperature	150°F
Initial Wetwell Relative Humidity	100%
Initial Drywell Relative Humidity	20%
Drywell and Wetwell and Pool Heat Sinks Modeled	Yes
Drywell Heat Load Modeled	Yes
RHR Service Water Temperature	92°F
RHR Heat Exchanger "K" Factor per Loop	307 Btu/sec-°F
Number of RHR Loops Available	1
Number of RHR Pumps in One RHR Loop	1
ASDC RHR Flow Rate	7,500 gpm
Condensate Available for Injection	90,000 gallons

Table 2.5-2 NFPA 805 Case 4 (EPU) Fire Event Evaluation Results

Item	Parameters	Values
1	Peak DW Pressure (psia)	24.1 ~21,570 seconds
2	Peak DW Temperature (ºF)	276.5 ~1,500 seconds
3	Peak WW Airspace Pressure (psia)	24.6 ~21,570 seconds
4	Peak WW Airspace Temperature (ºF)	210.2 ~50,390 seconds
5	Peak Pool Temperature (ºF)	208.0 ~18,280 seconds

Table 2.5-3 NFPA 805 Case 4 (EPU) Sequence of Events

Approximate Elapsed Time	Events
0 seconds	• Reactor scram occurs. • Main Steam Isolation Valves (MSIVs) start to close. • Feedwater pump is tripped. • Drywell coolers are tripped. • Condensate system continues to operate.
3.5 seconds	MSIVs are fully closed. After isolation, MSRVs automatically start to open and close to maintain RPV pressure.
25 minutes	Begin rapid depressurization using three MSRVs. RPV makeup is supplied by the condensate system.
~ 40 minutes	Condensate inventory available for injection is depleted. Operators secure condensate flow and initiate ASDC using 7,500 gpm of RHR flow in the LPCI mode.
2 hours	RHR heat exchanger is placed into service.
72 hours	Event is terminated.

Table 2.5-4 SGTS Iodine Removal Capacity Parameters

Parameter	Generic Input Criteria	Browns Ferry-Specific Value
[[		
		]]

Notes:

1. The value provided for this parameter is based on the configuration of the Browns Ferry SGTS. Two trains of that system will be the minimum number that will operate in a DBA scenario. As such, the fuel iodine inventory value for a single Browns Ferry unit is split evenly between the two. Also the flowrate provided is for a single train.
2. Actual MSIV leakage is 100 scfh. It is not routed to the SGTS.

Results of the CLTR AST evaluation are applicable to Browns Ferry and show that the maximum charcoal loading, [[ ]] well below the 2.5 mg/gm maximum value in RG 1.52 (although Browns Ferry is not committed to that regulatory guide for iodine loading to the charcoal). The maximum component temperature is approximately 168ºF with normal flow conditions and 500ºF under conditions of a failed fan with minimum cooling flow, well below the 625ºF charcoal ignition temperature.

The parameters used in the CLTR analysis for AST application are confirmed to bound the Browns Ferry SGTS plant specific values. Therefore, the SGTS at Browns Ferry is confirmed to be consistent with the generic description provided in the CLTR.

Table 2.5-5 Basis for Classification of No Significant Effect

Item	System Type	Browns Ferry System Name (Number)	Functional Description	Basis for Classification
[[				

Table 2.5-5 Basis for Classification of No Significant Effect (continued)

Item	System Type	Browns Ferry System Name (Number)	Functional Description	Basis for Classification

Table 2.5-5 Basis for Classification of No Significant Effect (continued)

Item	System Type	Browns Ferry System Name (Number)	Functional Description	Basis for Classification

Table 2.5-5 Basis for Classification of No Significant Effect (continued)

Item	System Type	Browns Ferry System Name (Number)	Functional Description	Basis for Classification

Table 2.5-5 Basis for Classification of No Significant Effect (continued)

Item	System Type	Browns Ferry System Name (Number)	Functional Description	Basis for Classification

Table 2.5-5 Basis for Classification of No Significant Effect (continued)

Item	System Type	Browns Ferry System Name (Number)	Functional Description	Basis for Classification

Table 2.5-5 Basis for Classification of No Significant Effect (continued)

Item	System Type	Browns Ferry System Name (Number)	Functional Description	Basis for Classification

Table 2.5-5 Basis for Classification of No Significant Effect (continued)

Item	System Type	Browns Ferry System Name (Number)	Functional Description	Basis for Classification

Table 2.5-5 Basis for Classification of No Significant Effect (continued)

Item	System Type	Browns Ferry System Name (Number)	Functional Description	Basis for Classification

Table 2.5-5 Basis for Classification of No Significant Effect (continued)

Item	System Type	Browns Ferry System Name (Number)	Functional Description	Basis for Classification
				]]

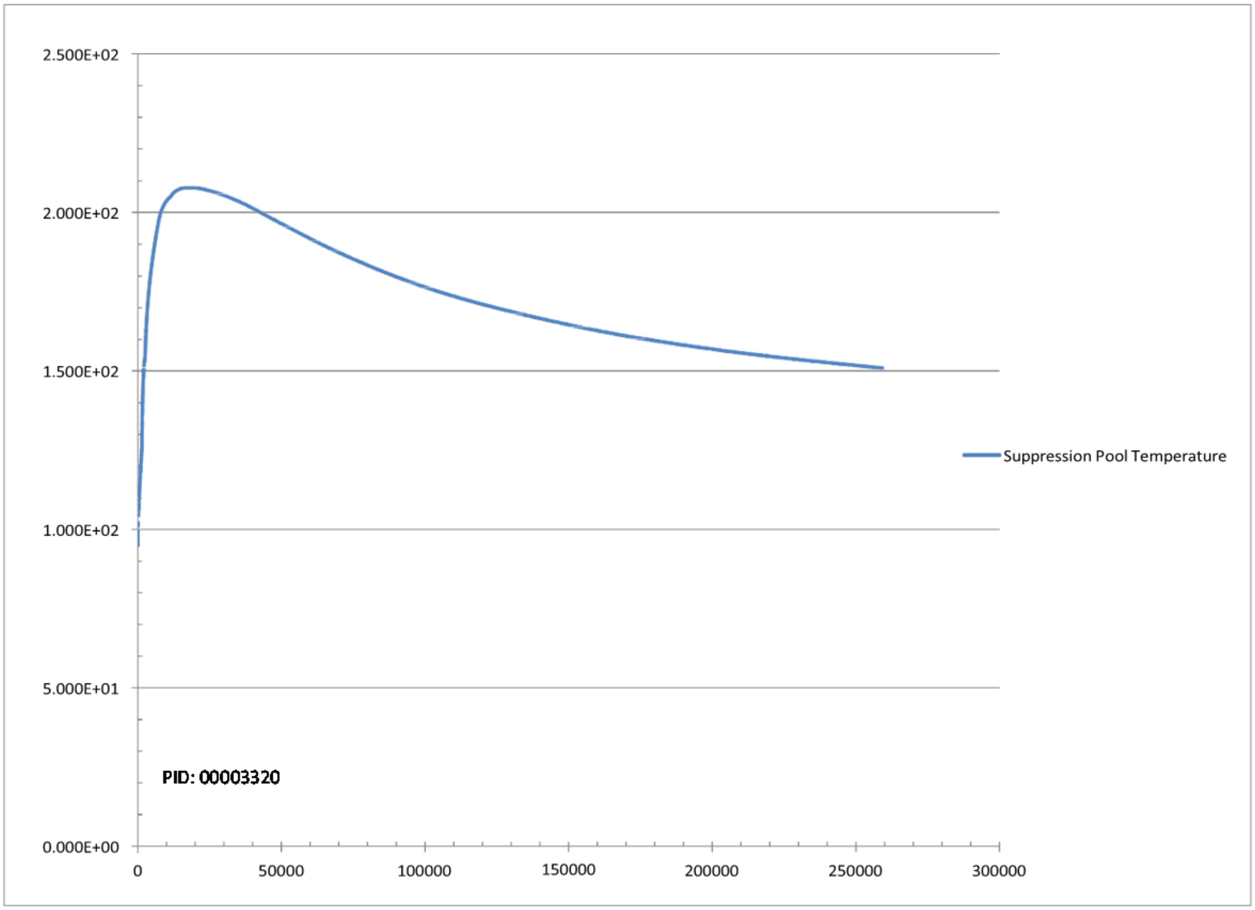

Figure 2.5-1 NFPA 805 Case 4 (EPU) Fire Event Suppression Pool Temperature

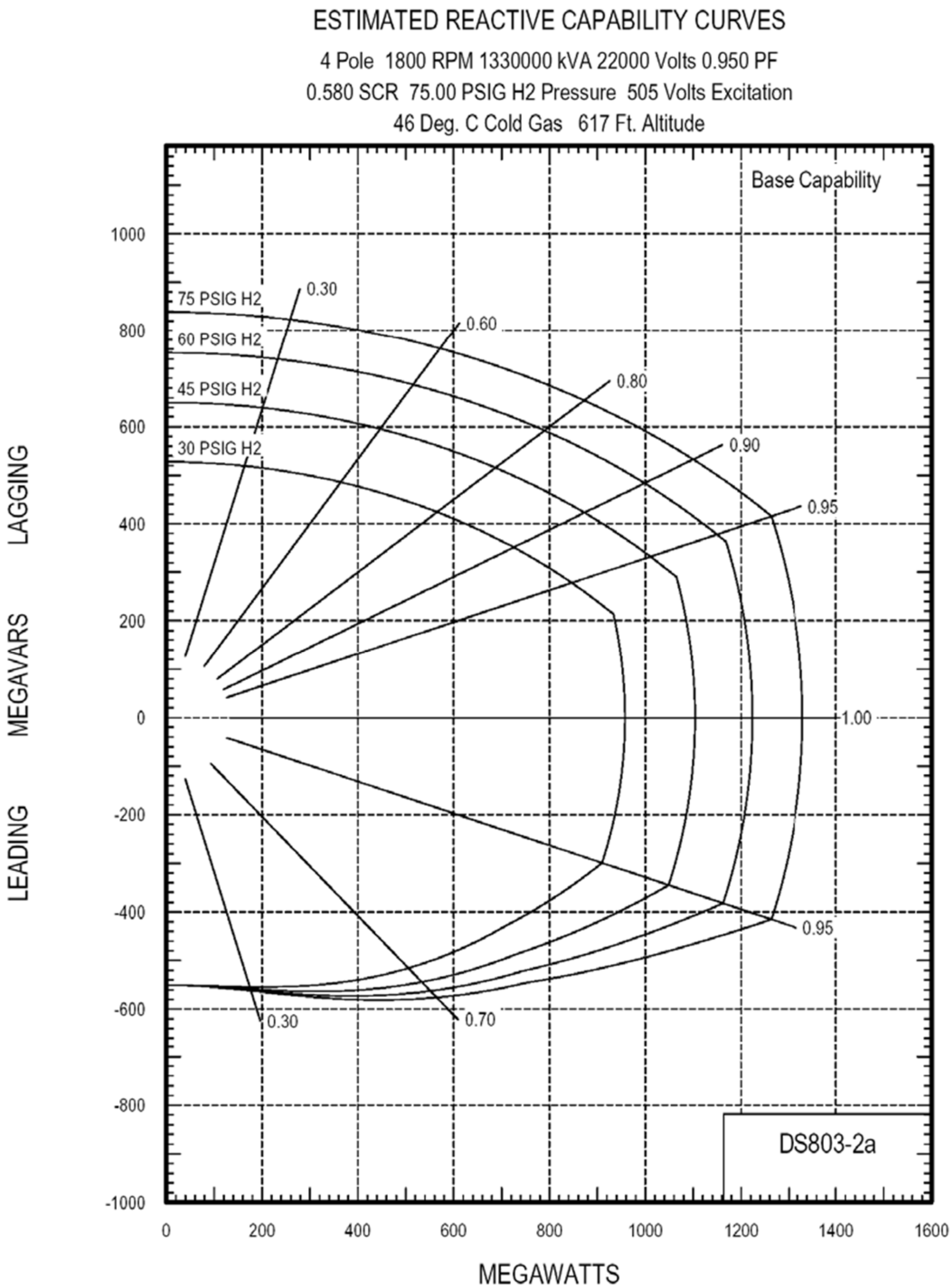

Figure 2.5-2a Browns Ferry Unit 1 Generator Reactive Capability Curve

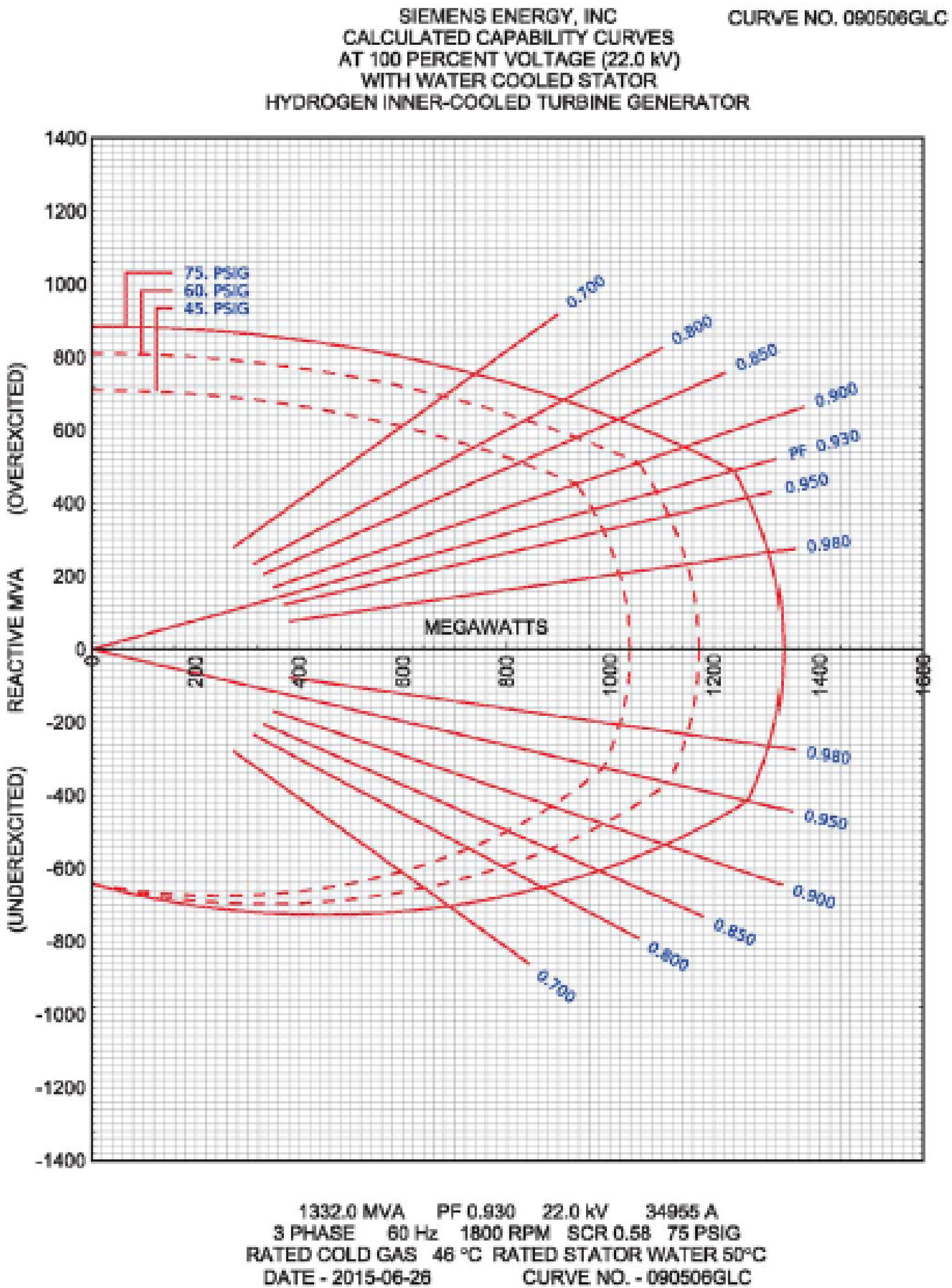

Figure 2.5-2b Browns Ferry Units 2 and 3 Generator Reactive Capability Curve

2.6 Containment Review Considerations

2.6.1 Primary Containment Functional Design

Regulatory Evaluation

The containment encloses the reactor system and is the final barrier against the release of significant amounts of radioactive fission products in the event of an accident.

The NRC's acceptance criteria for the primary containment functional design are based on (1) GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, and that such SSCs be protected against dynamic effects; (2) GDC-16, insofar as it requires that reactor containment be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment; (3) GDC-50, insofar as it requires that the containment and its associated heat removal systems be designed so that the containment structure can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated temperature and pressure conditions resulting from any LOCA; (4) GDC-13, insofar as it requires that instrumentation be provided to monitor variables and systems over their anticipated ranges for normal operation and for accident conditions, as appropriate, to assure adequate safety; and (5) GDC-64, insofar as it requires that means be provided to monitor the reactor containment atmosphere for radioactivity that may be released from normal operations and from postulated accidents.

Specific NRC review criteria are contained in SRP Section 6.2.1.1.C.

Browns Ferry Current Licensing Basis

The General Design Criteria (GDC) listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Browns Ferry Nuclear Plant (Browns Ferry) principal design criteria predate these criteria. The Browns Ferry principal design criteria are listed in UFSAR Section 1.5, "Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, the Tennessee Valley Authority (TVA) performed a comparative evaluation of the design basis of Browns Ferry with the AEC proposed General Design Criteria of 1967. The Browns Ferry UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria set out in the July 1967 AEC release. For each group of criteria, there is a statement of TVA's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a table of references to locations in the Browns Ferry UFSAR where there is subject matter relating to the intent of that particular criteria.

While Browns Ferry is not generally licensed to the final GDC or the 1967 AEC proposed General Design Criteria, a comparison of the final GDC to the applicable AEC proposed General Design Criteria can usually be made. For the final GDC listed in the Regulatory Evaluation above, the Browns Ferry comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as “draft GDC”) is contained in Browns Ferry UFSAR Appendix A: draft GDCs-10, 12, 17, 40, 42, and 49.

The primary containment is described in Browns Ferry UFSAR Sections 5.2, “Primary Containment System” and 7.3, “Primary Containment Isolation System.”

Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Browns Ferry License Renewal Safety Evaluation Report (SER), NUREG-1843, dated April 2006 (Reference 11). The license renewal evaluation associated with the primary containment is documented in NUREG-1843, Sections 2.3.2.1.1 and 2.4.1.1. Management of aging effects on the primary containment is documented in NUREG-1843, Sections 3.2.2 and 3.5.2.

Technical Evaluation

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

The Browns Ferry UFSAR provides the containment responses to various postulated accidents that validate the design basis for the containment. EPU operation changes some of the conditions for the containment analyses. For example, the short-term DBA-LOCA containment response during the blowdown is governed by the blowdown flow rate. This blowdown flow rate is dependent on the reactor initial thermal-hydraulic conditions, such as vessel pressure and the mass and energy of the vessel fluid inventory, which change slightly with EPU. Also, the long-term heat-up of the suppression pool following a LOCA or a transient is governed by the ability of the RHR to remove decay heat. Because the decay heat depends on the initial reactor power level, the long-term containment response is affected by EPU. The containment response was reanalyzed to demonstrate the plant's capability to operate with a rated power increase to 3,952 MWt. The key plant parameters used to model and analyze the plant response at EPU are provided in Table 2.6-2a.

The analyses of containment pressure and temperature responses, as described in Section 2.6.1.1, were performed at a power level of 102% of EPU RTP in accordance with ELTR1 using GEH codes and models. The M3CPT code was used to model the short-term containment pressure and temperature response. The modeling used in the M3CPT analyses is described in References 45 and 76. References 45 and 76 describe the basic containment analytical models used in GEH codes. Reference 6 describes the more detailed RPV model (LAMB) used for determining the vessel break flow in the containment analyses for EPU.

The LAMB code models the recirculation loop as a separate pressure node. It also allows for inclusion of flashing in the pipe and vessel during the blowdown and flow choking at the jet pump nozzles when the conditions warrant. The use of the LAMB blowdown flow in M3CPT was identified in ELTR1 by reference to the LAMB code qualification in Reference 6.

The SHEX code was used to model the long-term containment pressure and temperature response. The key models in SHEX are based on models described in References 45, 76 and 77. The GEH containment analysis methodologies have been applied to all BWR power uprate projects performed by GEH and accepted by the NRC.

The Browns Ferry original long-term containment analyses did not credit passive heat sinks in the drywell, wetwell airspace, and suppression pool. This conservative assumption was identified to the NRC as Assumption 6 of Attachment 1 to the March 12, 1993 GE letter referenced in Reference 7. Long-term containment analyses performed for Browns Ferry EPU now includes credit for these passive heat sinks. This is herein identified as a change in methodology. [[

]] (Assumption 8 of the same GE letter).

The effects of EPU on the containment dynamic loads due to a LOCA or MSRV discharge have also been evaluated as described in Section 2.6.1.2. The containment hydrodynamic loads have been defined generically for Mark I plants as part of the Mark I Containment Long-Term Program (LTP) (Reference 78) and approved by the NRC in Reference 79. The Browns Ferry plant-specific dynamic loads were defined in References 46 and 80, using the NRC approved methods of Reference 78. The evaluation of the LOCA containment dynamic loads is based primarily on the results of the short-term analysis described in Section 2.6.1.2.

The MSRV discharge load evaluation is based on no changes in the MSRV opening setpoints for EPU.

The metal-water reaction energy versus time relationship is calculated using the method described in USNRC Regulatory Guide 1.7 (Reference 81) as a normalized value (fraction of reactor thermal power). All of the energy from the metal-water reaction is assumed transferred to the reactor coolant in the first 120 seconds into the LOCA. The metal-water reaction energy represents a very small fraction of the total shutdown energy transferred to the coolant.

Browns Ferry uses the Mark I containment design. Per the discussion in Section 4.1 of the NRC Safety Evaluation (SE) for the CLTR (Reference 1), benchmarking cases, originally stipulated in Reference 4 and Reference 7, using SHEX are not required for Mark I and Mark III containment analyses. To quote: "The NRC has performed independent confirmatory analyses on extended uprates for both Mark I and Mark III containment designs and found the results consistent with SHEX results. Therefore, the confirmatory calculations with SHEX (benchmarking with current licensing basis assumptions – pre-uprate) for plant specific modeling are not required for extended power uprates for Mark I and Mark III containment designs." Therefore, following the

NRC safety evaluation of the CLTR (Reference 1), confirmatory benchmarking cases of SHEX are not required and were not performed for Browns Ferry EPU.

Browns Ferry meets all CLTR dispositions. The topics addressed in this evaluation are:

Topic	CLTR Disposition	Browns Ferry Result
Pool Temperature Response	Plant Specific	Meets CLTR Disposition
Wetwell Pressure	Plant Specific	Meets CLTR Disposition
Drywell Temperature	Plant Specific	Meets CLTR Disposition
Drywell Pressure	Plant Specific	Meets CLTR Disposition
Containment Dynamic Loads	Plant Specific	Meets CLTR Disposition
Containment Isolation	Plant Specific	Meets CLTR Disposition
Motor-Operated Valves	Plant Specific	Meets CLTR Disposition
Hardened Wetwell Vent System	Plant Specific	Meets CLTR Disposition
Equipment Operability	Plant Specific	Meets CLTR Disposition

2.6.1.1 Containment Pressure and Temperature Response

The CLTR states that the suppression pool temperature increases as a result of the higher decay heat associated with EPU. As a result of this, the suppression pool temperature response, wetwell pressure, drywell temperature, and drywell pressure need to be addressed. Short-term and long-term containment analysis results are reported in the UFSAR. The short-term analysis is directed primarily at determining the drywell pressure response during the initial blowdown of the reactor vessel inventory to the containment following a large break inside the drywell. Short-term containment response analyses were performed for the limiting DBA-LOCA that assumes a double-ended guillotine break of a recirculation suction line (RSLB) to demonstrate that EPU does not result in exceeding the containment design limits.

The long-term analysis is directed primarily at the suppression pool temperature response, considering the decay heat addition to the suppression pool. The RSLB DBA-LOCA, the double-ended guillotine break of a recirculation discharge line (RDLB) and small steam break LOCAs were reanalyzed for EPU. Peak values of the containment pressure and temperature responses to these LOCA events are given in Table 2.6-1. The effect of local suppression pool temperatures during MSRV discharges was addressed in accordance with the NUREG-0783 (Reference 82) criteria. Peak suppression pool temperatures resulting from the postulated ATWS, Station Blackout, and Fire events are given in Table 2.6-3.

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

2.6.1.1.1 Long-Term Suppression Pool Temperature Response

2.6.1.1.1.1 Bulk Pool Temperature

The long-term bulk pool temperature response for EPU was evaluated for the limiting DBA-LOCA in Section 14.6.3.3 (Case C) of the UFSAR. This DBA-LOCA is an instantaneous guillotine break of the RSLB. For Browns Ferry EPU, RDLB LOCA and small break LOCAs were also analyzed at EPU conditions

Per GE Safety Communication SC 06-01 (Reference 83), the potential was identified that a single failure that eliminated only the RHR heat exchanger could prove more limiting than the typically analyzed scenario of the single failure of an entire AC electrical power source. The Browns Ferry RHR system is configured with two loops of RHR, with each loop having its own separate injection point to the reactor pressure vessel, and with each loop having its own separate return to the suppression pool. Each loop is comprised of two RHR pumps with each pump having its own separate heat exchanger on its discharge. The current licensing basis analysis (Reference 84) for the short-term (first 10 minutes after the accident) evaluation of the RSLB assumed a Single Active Failure (SAF) where only two of the four RHR pumps were available. In order to address the issue identified in SC 06-01 (Reference 83), the RSLB EPU analysis assumes that all four RHR pumps are running in the short-term phase of the RSLB DBA-LOCA. This assumption will maximize the ECCS pump heat addition to the suppression pool and thereby maximize the suppression pool temperature. The RDLB analysis for CLTP conservatively assumed that all four RHR pumps are running in the short-term phase of the RSLB DBA-LOCA. The EPU analysis also conservatively assumes all four RHR pumps are running in the short-term phase of the RSLB DBA-LOCA. Therefore, the issue identified in SC 06-01(Reference 83) is addressed in the EPU analysis

The acceptability of ECCS pump NPSH based on the containment analysis suppression pool temperature response is demonstrated in Section 2.6.5.2. RHR and core spray pumps can be throttled to decrease required NPSH from the required NPSH at pump run-out flow conditions, provided containment cooling requirements are satisfied.

The analysis of the RSLB DBA-LOCA was performed at 102% of EPU RTP. The time-dependent SP temperature response is presented in Figure 2.6-1 and the calculated peak values

for LOCA bulk pool temperature for the CLTP and the EPU RTP case are compared in Table 2.6-1. The EPU analyses were performed using a decay heat table based on ANS/ANSI 5.1-1979 with 2-sigma adders with additional actinides and activation products per GE SIL 636 (Reference 85). No modifications were made to this standard.

The containment system response to the accident is divided into two analysis phases. The first phase, hereafter referred to as the short-term phase covers the period up to 10 minutes after the accident initiation. During the short-term phase, no operator action is credited in the analysis. The second phase, hereafter referred to as the long-term phase covers the period after 10 minutes following the accident initiation. During the long-term phase, operator actions such as those to reduce electrical loading on the emergency diesel generators and to re-align portions of the ECCS from core cooling mode to containment cooling mode are credited. The RSLB DBA-LOCA analysis assumes that offsite power is lost concurrently with the accident initiation and that offsite power is not available during the accident analysis period. Separate RSLB analysis cases are run with initial conditions to either maximize or minimize the containment drywell and wetwell pressure response while maximizing the suppression pool temperature response in order to determine the sensitivity of the peak suppression pool temperature response to perturbed initial conditions. No containment leakage is assumed except for the RSLB cases with initial conditions to minimize the containment drywell and wetwell pressure response while maximizing the suppression pool temperature response, for which containment leakage (2% per day) and the leakage from MSIVs (150 scfh for all steam lines) are considered. In addition, the containment responses to various modes of containment cooling are evaluated. These three RHR cooling modes are (1) Coolant Injection Cooling (CIC), where RHR flow is cooled by the RHR heat exchanger before being discharged into the reactor vessel; (2) Containment Spray Cooling (CSC), where RHR flow is cooled by the RHR heat exchanger and then discharged to the containment via the DW spray and wetwell spray headers; and (3) Suppression Pool Cooling (SPC), where RHR flow is cooled by the RHR heat exchanger and then discharged back to the suppression pool.

A complete LOOP is assumed to occur concurrent with the accident initiation. If a worst-case SAF such as failure of one emergency electrical power source (emergency diesel generator or loss of a 4 kV shutdown board) is assumed concurrent with the accident, then less than the full complement of low pressure ECCS pumps (four RHR pumps and four CS pumps) would be available during the short-term phase of the accident. However, if no SAF is assumed, then the full complement of ECCS pumps would be available. The initial condition of no SAF during the short-term phase is limiting for the determination of ECCS pump NPSH during the accident because of the Browns Ferry ECCS pump suction configuration where each ECCS pump does not have a dedicated ECCS suction strainer and piping suction directly from the suppression pool (torus). For each Browns Ferry unit, there are four ECCS suction strainers installed in the torus. The torus water volume then communicates to the ECCS pump suctions via a torus ring header located below the torus. This configuration result in higher ECCS piping head loss when there are multiple ECCS pumps running. In addition, a larger number of running ECCS pumps will lead to higher pump heat addition to the suppression pool. Conformance with GEH SC 06-01

(Reference 83) is made by assuming all low pressure ECCS pumps start during the short-term phase of the accident.

All ECCS pumps are assumed to be available for the first 600 seconds after accident initiation. No RHRSW flow is assumed to the RHR heat exchangers and there is no heat removal from the RHR heat exchangers during the short-term phase. RPV liquid is discharged from the break into the drywell causing rapid vessel depressurization and a rapid increase in the drywell pressure and temperature. For the first 600 seconds following the accident, four RHR pumps in LPCI mode (with two RHR pumps injecting liquid into the intact recirculation loop and the other two RHR pumps into the broken recirculation loop) and four CS pumps are used to cool the core. For the RSLB DBA-LOCA, the RHR flow into the broken recirculation loop will be directed to the RPV and RHR flow will not go into runout flow because the RHR injection point is between the RPV and the closed reactor recirculation discharge valve (the reactor recirculation discharge valve in each reactor recirculation loop receives an automatic closure signal during a LOCA). HPCI is assumed available and will start on either high DW pressure or low RPV level. However, HPCI will isolate on low steam pressure. The ECCS injection of suppression pool water, along with the assumed addition of feedwater, produces a recovery of the reactor water level. This allows water heated by decay heat and vessel sensible energy to be discharged into the drywell, and subsequently into the suppression pool.

If the accident were to occur on either Unit 1 or 2 and a worst-case SAF such as failure of one emergency electrical power source (emergency diesel generator or loss of a 4 kV shutdown board) is assumed concurrent with the accident, then less than the full complement of low pressure ECCS pumps (four RHR pumps and four CS pumps) would be available during the long-term phase of the accident. Assuming that one RHR pump is required for shutdown of the non-accident unit, only two RHR pumps and two RHR heat exchangers are assumed available for long-term containment cooling in the accident unit.

After 600 seconds, operator actions are credited. One loop of CS with two CS pumps continues to be available for RPV water makeup. One loop of CS with two pumps is secured because two CS pumps can supply adequate long-term core cooling. One loop of RHR with two pumps is secured, and another loop of RHR with two pumps is switched to a RHR mode of containment cooling with its associated RHRSW flow activated for two heat exchangers. Three RHR cooling modes are investigated: (1) Coolant Injection Cooling (CIC) where RHR in LPCI mode with flow from the suppression pool is cooled by the RHR heat exchanger before being discharged into the reactor vessel; (2) CSC where RHR flow from the suppression pool is discharged as drywell and wetwell sprays; and (3) SPC where the RHR flow from the suppression pool is cooled by the RHR heat exchanger before being discharged back into the suppression pool. The heat exchanger K-value and RHR pump flow rate are presented in Table 2.6-2a. Initial conditions (initial DW pressure, initial wetwell pressure and initial DW temperature) were also perturbed to both maximize and minimize the peak containment pressure and thereby investigate the effect on peak suppression pool temperature. The resulting calculated peak bulk SP

temperature for RSLB DBA-LOCA at 10 minutes after the accident initiation is 152.8°F and the peak bulk SP temperature for RSLB DBA-LOCA is 179.0°F.

The containment response during the first 10 minutes following the accident initiation for a RDLB-LOCA was calculated using Browns Ferry specific inputs to maximize suppression pool temperature and minimize containment pressure, similar to the RSLB DBA-LOCA analysis. The key parameter differences between the RDLB and the RSLB during the short-term phase of the accident are: (1) the break area (4.2 ft^2 for the RSLB versus 1.94 ft^2 for the RDLB) and, (2) the RHR flow rate and RHR injection path into the broken recirculation loop. For the RDLB, the RHR flow into the broken recirculation loop discharges directly to the drywell and the RHR flow into the broken loop is assumed at runout conditions (11,000 gpm per RHR pump for the RDLB versus 9,000 gpm per RHR pump for the RSLB). The resulting calculated peak bulk SP temperature for the RDLB at 10 minutes after the accident initiation is 152.0°F. The suppression pool temperature and corresponding wetwell pressure for the RDLB analyses are used in the evaluation of the available NPSH for the CS and the RHR pumps. The results of that evaluation are provided in Section 2.6.5.2.

The suppression pool temperature response was also calculated for the spectrum of small steam line break LOCAs as evaluated for the drywell temperature response. The most limiting bulk suppression pool temperature response to the small steam line break LOCA was found to occur for the smallest break size evaluated, a 0.01 ft^2 break, which produced a peak bulk suppression pool temperature of 182.7°F (See Section 2.6.5.1).

The suppression pool temperature response was also calculated for the shutdown of a non-accident unit. The time-dependent SP temperature response is presented in Figure 2.6-1a. The peak bulk suppression pool temperature for this case is 185.1°F (See Section 2.6.5.1).

Based on the analysis and limit values shown in Table 2.6-1, the peak bulk pool temperature for the LOCA events at EPU RTP is acceptable from a structural design standpoint. With calculated peak bulk suppression pool temperatures below the design limit, small break LOCAs and non-LOCA events with EPU are also acceptable from a structural design standpoint

2.6.1.1.1.2 Local Pool Temperature with MSRV Discharge

The local pool temperature limit for MSRV discharge was specified in NUREG-0783 (Reference 82) because of concerns resulting from unstable condensation observed at high pool temperatures in plants without quenchers. Quencher devices such as the T-quenchers used in the Browns Ferry units mitigate these loads. The peak local suppression pool temperature at Browns Ferry has been evaluated for EPU, with the same scenario assumptions as evaluated in the original analysis of Reference 86, and meets the NUREG-0783 criteria. This evaluation demonstrated a minimum subcooling of approximately 20°F locally at the quencher. This meets the acceptance criteria included in NUREG-0783 and also ensures that the exiting quencher steam is condensed before posing a steam ingestion potential for any ECCS pump suction. Therefore, the peak local suppression pool temperature at Browns Ferry remains acceptable at EPU conditions.

Containment pressure response and temperature response as a result of [[ ]] were evaluated and found to be acceptable. Therefore, the containment pressure and temperature response meets all CLTR dispositions.

2.6.1.1.2 Steam Bypass Capability

Containment response is based on maintaining the pressure suppression function by limiting the leakage from the drywell to the suppression pool due to leakage between the drywell and wetwell airspace. In the event that excessive bypass leakage was to occur, over-pressurization of the primary containment could occur. The acceptance criterion for Mark I plants such as Browns Ferry, with regard to steam bypass leakage, is that the measured leakage is not greater than the leakage that would result from a one inch diameter opening. This maximum bypass leakage is confirmed by plant tests as directed in Browns Ferry Technical Specification Surveillance Requirement (SR) 3.6.1.1.2. The current steam bypass effective area capability, $A/\sqrt{K}$, which was established from Browns Ferry analysis, is 0.18 ft^2. This 0.18 ft^2 effective area is approximately 54 times greater than the effective area of a one-inch opening (a one-inch plate orifice has an $A/\sqrt{K}$ of ~0.0033 ft^2).

The steam bypass analysis was performed at an initial power level of 102% of EPU RTP. At EPU conditions, the steam bypass analyses were performed by assuming a spectrum of steam line breaks and by crediting containment heat sinks. Mechanistic energy and mass transfer between the suppression pool and airspace more realistically model the physical phenomenon for steam bypass conditions. In the current licensing basis analysis, operator action to initiate containment spray and thereby mitigate containment over pressurization is assumed to occur 10 minutes after the wetwell pressure reaches 35 psig. In order to address the possible interruption of containment cooling due to receipt of a LOCA signal caused by high drywell pressure concurrent with low RPV pressure, the EPU analysis conservatively assumed, for all break sizes except the smallest analyzed break size of 0.01 ft^2, a 20 minute delay for the initiation of containment sprays after the wetwell pressure reaches 35 psig. For the smallest break size of 0.01 ft^2, the RPV depressurization rate is sufficiently small that operators can inhibit the containment cooling interruption caused by high drywell pressure concurrent with low RPV pressure. The EPU analysis for the 0.01 ft^2 break assumed the operators initiate containment spray 10 minutes after the wetwell pressure reaches 35 psig. The EPU evaluation shows that the peak containment pressure remains below the containment design pressure with no change in the current steam bypass effective area capability (0.18 ft^2) with an initial DW temperature of 130ºF. EPU requires no change to the existing Browns Ferry TS SR 3.6.1.1.2 which is to detect flow paths between the drywell and wetwell whose total capacity is equal to or greater than the capacity of a one-inch diameter plate orifice (a one-inch plate orifice has an $A/\sqrt{K}$ of ~0.0033 ft^2).

2.6.1.2 Containment Dynamic Loads

The CLTR states that the suppression pool temperature increases as a result of the higher decay heat associated with EPU. As a result, containment dynamic loads are addressed in the following sections.

2.6.1.2.1 Loss-of-Coolant Accident Loads

The LOCA containment dynamic loads analysis for EPU is primarily based on the short-term RSLB LOCA analyses and compliance with generic criteria developed through testing programs. The analyses were performed as described in Section 2.6.1.1 with break flows calculated using a more detailed RPV model (Reference 6). The NRC approved use of this model for the EPU containment evaluations in Reference 4. These analyses also provide calculated values for the controlling parameters for the dynamic loads throughout the blowdown. The key parameters are drywell and wetwell pressures, vent flow rates and suppression pool temperature. The LOCA dynamic loads considered in the EPU evaluations include pool swell, CO and chugging. For Mark I plants like Browns Ferry, the vent thrust loads were also evaluated.

The results of the EPU pool swell evaluation confirmed that the current pool swell load definition remains bounding. The containment response conditions for EPU are within the range of test conditions used to define CO loads for the plant. The containment response conditions for EPU are within the conditions used to define the chugging loads. The vent thrust loads at EPU conditions were calculated to be less than the plant-specific values calculated during the Mark I containment LTP.

The Mark I containment program Load Definition Report (LDR) Table 4.5.1-1 (Reference 78) defines the onset and duration times for chugging based on break size. For intermediate break sizes, chugging ends at 900 seconds after onset of chugging, 905 seconds after the break; for small break sizes, chugging ends at 900 seconds after onset of chugging, 1,200 seconds after the break. Discussion of the chugging duration time is provided in Sections 2.2, 2.3, and 4.4.1.1 of the LDR (Reference 78). For the load definition, chugging is assumed to end when reactor pressure is reduced to or below the drywell pressure, essentially stopping break flow and therefore vent steam flow. This vessel depressurization for the IBA and SBA events is due to initiation of the Automatic Depressurization System (ADS). The load definition of the LDR does not include any credit for operation of containment (drywell) sprays. However, Emergency Operating Procedures (EOPs) for Browns Ferry include direction to initiate drywell sprays when wetwell pressure exceeds 12.0 psig. Containment analyses performed for Browns Ferry EPU have shown that wetwell pressure will exceed this drywell spray initiation pressure of 12.0 psig by 600 seconds following initiation of the event if conditions for chugging are present. Initiation of drywell sprays will rapidly reduce drywell pressure and stop chugging. However, as reported in GEH SC 11-10 (Reference 87), for plants like Browns Ferry, where a LOCA signal is initiated on concurrent high DW pressure plus low RPV pressure, DW spray initiation could be delayed up to 1,200 seconds after initiation of an IBA or SBA LOCA. Therefore the chugging duration could be extended to a maximum of 1,200 seconds, which exceeds the duration times identified in Reference 78 for Mark I plants and Reference 46 for Browns Ferry.

The effect of the chugging duration extension to 1,200 seconds was evaluated for Browns Ferry. From the Browns Ferry plant unique analysis report (PUAR (Reference 46)), the limiting fatigue usage factors for containment components are listed below:

Component	Allowable Fatigue Usage	DBA Usage Factor	SBA/IBA Usage Factor
Drywell Structure			
General Shell and Cradle	1.0	0.051	0.096
Penetration X-204C	1.0	0.020	0.103
Containment Vent System			
Downcomer/Vent Header Intersection	1.0	0.559	0.610
Downcomer/Tie Bar Intersection	1.0	0.107	0.353
Torus Bellow Intersection	1.0	0.000	0.000

The highest fatigue usage factor from the above table for the SBA/IBA is 0.610. This 0.610 fatigue usage factor is the sum of the fatigue usage factor due to MSRV actuation (0.392) and the fatigue usage factor due to chugging (0.218). The number of chugging cycles is factored to consider a 900 second duration. To account for the 1,200 second chugging duration, the fatigue usage factor due to chugging can be linearly extrapolated: 0.218 * (1,200/900) = 0.291. Adding this chugging fatigue usage factor to the MSRV fatigue usage factor of 0.392 provides a revised fatigue usage factor of 0.683, which remains below the allowable fatigue usage factor of 1.0. All other components remain well below the allowable fatigue usage factor even if all fatigue usage is conservatively attributed only to chugging and the PUAR value identified in the above table is factored by 1,200/900.

2.6.1.2.2 Safety Relief Valve Loads

The MSRV loads include MSRVDL loads, suppression pool boundary pressure loads, and drag loads on submerged structures. The MSRV opening setpoint pressure, the initial water leg in the MSRVDL, the MSRVDL geometry, and the suppression pool geometry influence these loads. The MSRV loads were evaluated for two different actuation phases: initial actuation and subsequent actuation.

For the initial MSRV actuation following an event involving RPV pressurization, the only parameter change potentially introduced by EPU, which can affect the MSRV loads definition, is an increase in MSRV opening setpoint pressure. However, the changes proposed for EPU do not include an increase in the MSRV opening setpoint pressure.

The load definition for subsequent MSRV actuations is not affected by EPU because the MSRVDL reflood height used for Browns Ferry is the maximum reflood height (Reference 46) that is not affected by the time between MSRV closing and MSRV reopening. The maximum reflood height is controlled by the MSRVDL geometry and the MSRVDL vacuum breaker capacity. Because all these parameters, including the MSRV setpoints, do not change, loads due to subsequent MSRV actuations are not affected by EPU.

Therefore, EPU does not affect the MSRV loads or load definitions.

2.6.1.2.3 LOCA Pressure and Temperature Loads

The Reference 80 plant unique load definition report (PULD) provided LOCA-induced pressure and temperature results from DBA-LOCA (DBA), IBA, and SBA events as an input for subsequent use in the Reference 46 structural analysis. The IBA and SBA events were re-evaluated at 102% EPU RTP using initial conditions and assumptions consistent with the Reference 80 analysis. The results of the Browns Ferry EPU analysis show that all DW and WW pressure and temperatures at EPU conditions are bounded by the values of Reference 80 with the exception of the peak WW and SP temperature for the SBA. At EPU conditions, the SBA peak WW and SP temperature is 146°F, which does not bound the Reference 80 result of 136°F. [[

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The evaluation of WW and SP piping that have the SBA temperature as a structural load combination input is contained in Section 2.2.2.2.2.2 (Other Piping Evaluation).

Given the fact that current containment dynamic load evaluations remain bounding and applicable for plant operation at EPU conditions, and that the current MSRV load definition is still applicable, all CLTR dispositions are met.

2.6.1.3 Containment Isolation

The CLTR states that the suppression pool temperature increases as a result of the higher decay heat associated with EPU. However, the system designs for containment isolation are not affected by EPU. The capabilities of isolation actuation devices to perform during normal operations and under post-accident conditions have been determined to be acceptable. Therefore, the Browns Ferry containment isolation capabilities are not adversely affected by the EPU and all CLTR dispositions are met.

2.6.1.4 Generic Letter 89-16 Hardened Wetwell Vent

In response to GL 89-16, Browns Ferry installed a Hardened Wetwell Vent (HWWV) to mitigate the pressure increase during a TW severe accident sequence. The vent capacity is currently sized to prevent the containment pressure from exceeding the primary containment pressure limit with constant heat input equal to 1% of 3,458 MWt or 34.58 MWt. The Browns Ferry HWWV design for Unit 1, which is functionally the same for Units 2 and 3 with respect to the 1% vent capacity, was approved by Amendment No. 269 to Renewed License No. DPR-33 (Reference 89). At

EPU conditions, the existing vent capacity will be reduced to 0.88% of rated thermal power. This capacity will be restored to 1% of rated EPU power as discussed below.

In response to EA-13-109, "Issuance of Order to Modify Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation Under Severe Accident Conditions" (Reference 90), Browns Ferry will modify the hardened containment vent system (HCVS) as described in the Browns Ferry letter to the NRC dated August 28, 2014 (Reference 91) and approved by NRC letter dated December 23, 2014 (Reference 92).

The consideration of EPU conditions and the 1% vent capacity requirement is consistent with NEI 13-02, "Industry Guidance for Compliance with Order EA-13-109," Revision 0 (Reference 93). NRC endorsed NEI 13-02 by JLD-ISG-2013-02, Revision 0, "Compliance with Order EA-13-109, Order Modifying Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation Under Severe Accident Conditions," Interim Staff Guidance, November 14, 2013 (Reference 94). The Browns Ferry letter details the modification schedule for the HCVS for full compliance with Phase 1 of EA-13-109. The HCVS modifications for each Browns Ferry unit will be completed prior to implementing EPU at that unit.

2.6.1.5 Generic Letter 96-06

GL 96-06 identified potential problems with equipment operability and containment integrity during design-basis accident conditions as a result of: (1) water hammer and/or two-phase flow conditions in cooling water systems serving the containment air coolers; and (2) thermally induced over-pressurization of isolated piping sections in containment.

The Browns Ferry response to GL 96-06 stated that in the event of a LOCA or MSLB with a coincident LOOP, the RBCCW pumps will be load shed at the start of the event. The first RBCCW pump will then be given a start signal 40 seconds after it was shed. Based on computed results, voiding in the RBCCW drywell atmosphere cooling coils will not occur for at least 60 seconds while the RBCCW pumps are stopped. As a RBCCW pump will have been restarted prior to voiding occurring, water hammer and/or two-phase flow in the RBCCW system are not a concern. EPU is not altering the RBCCW system serving the drywell atmosphere cooling coils.

The Browns Ferry analysis of record at CLTP shows that the recirculation line break DBA-LOCA event causes boiling in the drywell cooling coils sooner than for steam line breaks at CLTP conditions. A comparison of the drywell temperature profiles during a DBA-LOCA for CLTP and EPU conditions shows a minor difference in temperature (less than 2°F) and lasting only seconds. Based on the minor differences in the drywell temperature profiles and the margin between the RBCCW pump start (40 seconds) and calculated time to boil (61 seconds), it is concluded that voiding will not occur under EPU conditions prior to the restart of a RBCCW pump.

For steam line break LOCAs, including the MSLB, the DW temperature at EPU conditions (335.2°F as shown on Note 6 of Table 2.6-1) is bounded by the CLTP peak DW temperature of 336°F used in the analysis of record (using the same initial DW temperature of 150°F at both CLTP and EPU). The peak DW temperature at EPU using a conservatively low initial DW

temperature (70°F) to maximize the containment pressure response is 336.9°F (see Table 2.6-1), which slightly exceeds the peak temperature value used in the analysis of record. The slightly higher peak temperature will result in a slight reduction in the time to boil and a subsequent reduction in margin between the time to boil and the time that the RBCCW pumps restart and return flow to the drywell atmosphere cooling coils. This reduction in margin is considered negligible and does not affect the conclusion that voiding will not occur under EPU conditions prior to the restart of the RBCCW pump.

The Browns Ferry response to GL 96-06 included four primary containment penetrations and process lines that were identified as being susceptible to thermal pressurization:

1) Demineralized water system,

2) Drywell floor drain sump discharge,

3) Drywell equipment drain sump discharge, and

4) Reactor water sampling system.

The demineralized water system is acceptable at EPU as controls are in place to ensure the header is drained prior to power operation each cycle. The current analysis of record calculation, using an assumed constant DW temperature of 336°F following a LOCA showed that 2.06 gallons of water would have to be drained from the system prior to power operation in order to prevent system over-pressurization following a LOCA. A slightly higher assumed EPU DW temperature of 336.9°F will increase the drain requirement to 2.07 gallons, which is negligible.

The DW floor and DW equipment drain sump discharge lines are acceptable as a 0.06-inch (1/16-inch diameter) orifice has been drilled in each discharge check valve. This orifice ensures adequate leakage to prevent over-pressurization due to thermal expansion. The current analysis of record calculation, using an assumed constant DW temperature of 336°F following a LOCA, showed that an orifice diameter of 0.052-inches was sufficient to relieve the flow associated with thermal expansion in the discharge lines. A slightly higher assumed EPU DW temperature of 336.9°F will negligibly increase the flow requirements, and the existing orifices are adequately sized to pass this flow at EPU.

The thermally induced over-pressurization of the reactor sampling system is calculated at CLTP (using a constant DW temperature of 336°F following a LOCA) to reach approximately 2,546 psig which will lift the inboard globe isolation valve disc off its seat and relieve the pressure back to the reactor vessel. The pressure associated with the seat lift is well within the design pressures of the reactor sampling equipment in the drywell subject to the thermal over-pressurization. The increase in drywell temperature to 336.9°F at EPU will result in a negligible reduction in margin for the prevention of over-pressurization in the reactor water sampling system. The pressure associated with the seat lift at EPU remains well within the design pressures of the reactor sampling equipment in the drywell subject to the thermal over-pressurization.

The list of penetrations susceptible to thermal over-pressurization during design-basis accident conditions does not change at EPU conditions. A review of the EPU process conditions was performed for other systems (e.g., main steam, RWCU, RHR, and SLC) that could be potentially susceptible to thermal over-pressurization. The review concluded that EPU does not result in any changes to either the physical configuration or process conditions of the systems that would change the current Browns Ferry disposition of these systems as acceptable for thermal over-pressurization. EPU is not adding any new containment penetrations or performing any physical/procedural changes to the penetration configuration, or the process lines that pass through them; therefore, the Browns Ferry analysis of thermally induced over-pressurization of isolated piping sections in containment remains valid.

Therefore, the existing Browns Ferry response to GL 96-06 remains valid for EPU and all CLTR dispositions are met.

Conclusion

TVA has evaluated the containment temperature and pressure transient and accounted for the increase of mass and energy resulting from the proposed EPU. The evaluation indicates that containment systems will continue to provide sufficient pressure and temperature mitigation capability to ensure that containment integrity is maintained. The evaluation further indicates that containment systems and instrumentation will continue to be adequate for monitoring containment parameters and release of radioactivity during normal and accident conditions and the containment and associated systems will continue to meet the requirements of draft GDCs-10, 12, 17, 40, 42, and 49 following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to primary containment functional design.

2.6.2 Subcompartment Analyses

Regulatory Evaluation

A subcompartment is defined as any fully or partially enclosed volume within the primary containment that houses high-energy piping and would limit the flow of fluid to the main containment volume in the event of a postulated pipe rupture within the volume.

The NRC's acceptance criteria for subcompartment analyses are based on (1) GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, and that such SSCs be protected against dynamic effects; and (2) GDC-50, insofar as it requires that containment subcompartments be designed with sufficient margin to prevent fracture of the structure due to the calculated pressure differential conditions across the walls of the subcompartments.

Specific NRC review criteria are contained in SRP Section 6.2.1.2.

Browns Ferry Current Licensing Basis

The General Design Criteria (GDC) listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Browns Ferry Nuclear Plant (Browns Ferry) principal design

criteria predate these criteria. The Browns Ferry principal design criteria are listed in UFSAR Section 1.5, "Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, the Tennessee Valley Authority (TVA) performed a comparative evaluation of the design basis of Browns Ferry with the AEC proposed General Design Criteria of 1967. The Browns Ferry UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria set out in the July 1967 AEC release. For each group of criteria, there is a statement of TVA's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a table of references to locations in the Browns Ferry UFSAR where there is subject matter relating to the intent of that particular criteria.

While Browns Ferry is not generally licensed to the final GDC or the 1967 AEC proposed General Design Criteria, a comparison of the final GDC to the applicable AEC proposed General Design Criteria can usually be made. For the final GDC listed in the Regulatory Evaluation above, the Browns Ferry comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Browns Ferry UFSAR Appendix A: draft GDCs-40, 42, and 49.

The primary containment is described in Browns Ferry UFSAR Sections 5.2, "Primary Containment System" and 12.2, "Principal Structures and Foundations."

Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Browns Ferry License Renewal Safety Evaluation Report (SER), NUREG-1843, dated April 2006 (Reference 11). The license renewal evaluation associated with the primary containment is documented in NUREG-1843, Sections 2.3.2.1.1 and 2.4.1.1. Management of aging effects on the primary containment is documented in NUREG-1843, Sections 3.2.2 and 3.5.2.

Technical Evaluation

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

Browns Ferry meets all CLTR dispositions. The topics addressed in this evaluation are:

Topic	CLTR Disposition	Browns Ferry Result
Liquid Lines	Plant Specific	Meets CLTR Disposition

As stated in Section 10.1 of the CLTR, EPU may increase subcooling in the reactor vessel, which may lead to increased break flow rates for liquid line breaks.

An annular structure is located inside the drywell around the RPV in order to provide thermal and radiation shielding, and is called the Sacrificial Shield Wall (SSW). The SSW is designed to withstand the differential pressure that would develop across the wall as a result of a high pressure pipe break within the annulus (i.e., between the RPV and the SSW).

For Browns Ferry, pipes with nominal diameters of four inches or smaller are the only reactor coolant lines investigated, because the reactor vessel safe-end welds for these nozzles are located within the sacrificial shield area. The minimum wall thickness for the various piping systems occurs at the safe-end joint to the piping. All other sections from this joint back to the reactor vessel have thicker wall sections and, therefore, have lower stresses. The largest line which has the safe end located in the annulus is the 4-inch jet pump instrument line.

The EPU assessment is an extension of the prior 5% power uprate assessment. The Moody slip critical flow model is used to generate critical mass flux values based on downcomer conditions corresponding to OLTP, Current Licensed Thermal Power (CLTP), EPU thermal power and the operating condition with maximum downcomer subcooling (minimum recirculation pump speed with feedwater temperature reduction). Shield wall differential pressure estimates are developed by scaling the OLTP analysis shield wall differential pressure based on the following assumptions:

1. Non-condensable gases that are initially in the annulus are effectively vented by the time of peak annulus to drywell vent differential pressure.
2. The break fluid flashes to a two phase homogeneous mixture with 100% water entrainment. The mixture saturation pressure is equal to the annulus pressure, and the mixture enthalpy is equal to the break fluid enthalpy.
3. The drywell pressure is assumed to remain constant during the event and to be identical for all operating conditions. Evaluations were performed for a set of drywell pressures ranging from 14.4 psia to 17.0 psia to ensure conservative differential pressure estimates. Note that the results of the shield wall differential pressure calculations demonstrate that the shield wall differential pressure estimates for a given condition vary by less than 1% for this range of initial drywell pressures.
4. Based on the low differential pressures, the annulus to drywell vent does not experience critical flow during the event. The annulus pressure is therefore equal to the sum of the vent differential pressure and the assumed drywell pressure.
5. Flow through the annulus to drywell vent is assumed to be isentropic.

Scaling is performed with a modified version of Darcy's formula that accounts for second order effects of compressible vent flow. The following equation is used to estimate the shield wall annulus to drywell vent differential pressure:

$$DP_2 = DP_1 * (G_{c,2}^2/G_{c,1}^2) * (\rho_1/\rho_2)\ (Y_1^2/Y_2^2)$$

Where:

DP_2 is the SSW annulus to drywell vent differential pressure for the new operating condition.

DP_1 is the OLTP condition shield wall differential pressure (2.0 psid).

$G_{c,2}$ is the Moody Slip critical mass flux based on the pressure and enthalpy of the fluid in the downcomer for the new operating condition.

$G_{c,1}$ is the Moody Slip critical mass flux based on the pressure and enthalpy of the fluid in the downcomer for the OLTP operating condition.

ρ_1 is the density of the annulus mixture for the OLTP condition.

ρ_2 is the density of the annulus mixture for the new operating condition.

Y_1 is vapor expansion factor applicable to the OLTP condition.

Y_2 is vapor expansion factor applicable to the new operating condition.

The vapor expansion factors (Y_1 and Y_2) are calculated as the ratio of the downstream density over the upstream density based on the assumption that the saturated two-phase mixture undergoes an isentropic expansion from the annulus pressure to the drywell pressure. The annulus pressure and annulus mixture enthalpy are used to determine the annulus mixture specific entropy. The downstream condition properties are established based on the assumption that the downstream fluid is a saturated two-phase mixture with a specific entropy equal to the upstream mixture specific entropy and a pressure equal to the assumed drywell pressure.

The shield wall differential pressure is calculated by adding a conservative estimate of the static head of the two phase mixture in the annulus to the calculated annulus to drywell vent differential pressure estimate.

The annulus pressure load on the biological shield wall due to a postulated break in a 4-inch jet pump instrument line nozzle is evaluated at EPU conditions. The CLTP annulus pressure load (2.4 psid), documented in UFSAR Section 12.2.2.6, remains bounding compared to the 102% EPU annulus pressure load of 2.3 psid for normal feedwater temperature operation. For reduced feedwater temperature operation at 102% EPU power, the annulus pressure load is 2.5 psid. For the limiting minimum pump speed, reduced feedwater temperature operating condition, the annulus pressure load is 3.6 psid. The results of the EPU evaluation, which addresses the effects of both EPU and operation at the limiting off-rated condition along the MELLLA operating domain upper boundary (Minimum Recirculation Pump Speed (MPS) point with feedwater temperature reduction), indicate that the SSW pressure difference design limit is not exceeded.

Parameter	102% CLTP MELLLA NFWT	102% EPU Rated Flow NFWT	102% EPU Rated Flow RFWT	55.4% EPU[1] MELLLA Line RFWT	Design Limit (psid)
Critical Mass Flux (lbm/sec-ft^2)	10,091.9	9,803.6	10,244.7	12,485.8	N/A
Maximum SSW DP (psid)	2.4	2.3	2.5	3.6	19

Note:

1. 102% of the power level at the intersection of the MELLLA line and the minimum pump speed line (1.02 * 54.3%).

Subcompartment Pressurization Evaluation

As discussed earlier, the differential pressure loading on the SSW is not significantly affected by EPU. For normal feedwater temperature operation, EPU implementation will result in a small decrease in SSW differential pressures. Increased downcomer subcooling associated with reduced feedwater temperature operation at EPU results in a small increase in SSW differential pressures. Reduced feedwater temperature operation at the intersection of the MELLLA line and the minimum pump speed line, which is not affected by EPU implementation, produces the highest downcomer subcooling, and limiting SSW differential pressures. The peak SSW differential pressures resulting from the limiting 4-inch jet pump instrument line break at CLTP and at EPU conditions remain below the SSW differential pressure design limit.

Therefore, the subcompartment analyses meet all CLTR dispositions.

Conclusion

TVA has evaluated the change in predicted pressurization resulting from the increased mass and energy release. The evaluation indicates that containment SSCs important to safety will continue to be protected from the dynamic effects resulting from pipe breaks and that the subcompartments will continue to have sufficient margins to prevent failure of the structure due to a pressure difference across the walls at design basis conditions following implementation of the proposed EPU. Based on this, the plant will continue to meet draft GDCs-40, 42, and 49 for the proposed EPU. Therefore, the proposed EPU is acceptable with respect to subcompartment analyses.

2.6.3 Mass and Energy Release

2.6.3.1 Mass and Energy Release Analysis for Postulated Loss of Coolant

Regulatory Evaluation

The release of high-energy fluid into containment from pipe breaks could challenge the structural integrity of the containment, including subcompartments and systems within the containment.

The NRC's acceptance criteria for mass and energy release analyses for postulated LOCAs are based on (1) GDC-50, insofar as it requires that sufficient conservatism be provided in the mass and energy release analysis to assure that containment design margin is maintained; and (2) 10 CFR Part 50, Appendix K, insofar as it identifies sources of energy during a LOCA.

Specific NRC review criteria are contained in SRP Section 6.2.1.3.

Browns Ferry Current Licensing Basis

The General Design Criteria (GDC) listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Browns Ferry Nuclear Plant (Browns Ferry) principal design criteria predate these criteria. The Browns Ferry principal design criteria are listed in UFSAR Section 1.5, "Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, the Tennessee Valley Authority (TVA) performed a comparative evaluation of the design basis of Browns Ferry with the AEC proposed General Design Criteria of 1967. The Browns Ferry UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria set out in the July 1967 AEC release. For each group of criteria, there is a statement of TVA's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a table of references to locations in the Browns Ferry UFSAR where there is subject matter relating to the intent of that particular criteria.

While Browns Ferry is not generally licensed to the final GDC or the 1967 AEC proposed General Design Criteria, a comparison of the final GDC to the applicable AEC proposed General Design Criteria can usually be made. For the final GDC listed in the Regulatory Evaluation above, the Browns Ferry comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Browns Ferry UFSAR Appendix A: draft GDC-49.

The mass and energy release analysis is described in Browns Ferry UFSAR Section 14.6.3.3, "Primary Containment Response."

Technical Evaluation

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

effects of EPUs. Section 4.1 of the CLTR addresses the effect of EPU on Containment System Performance. The results of this evaluation are described below.

Browns Ferry meets all CLTR dispositions. The topics addressed in this evaluation are:

Topic	CLTR Disposition	Browns Ferry Result
Drywell Temperature	Plant-Specific	Meets CLTR Disposition
Drywell Pressure	Plant-Specific	Meets CLTR Disposition

2.6.3.1.1 Drywell Temperature

The CLTR states that the suppression pool temperature increases as a result of the higher decay heat associated with EPU. The assumption of constant pressure minimizes the effect on other aspects of the containment evaluation. The bounding drywell temperature occurs during a break of a steam line. A spectrum of steam line break sizes have been evaluated to ensure a bounding drywell environmental qualification temperature profile is established. The time-dependent DW temperature responses are presented in Figure 2.6-9. The analysis has been performed in accordance with NUREG-0588 (Reference 95), and the most limiting drywell temperature from this analysis occurs for the break size of 0.25 ft^2 at initial drywell temperature of 70ºF, and is shown in Table 2.6-1. Although the drywell environment may see temperatures approaching 340°F, the containment structure, including the drywell shell, has a design temperature limit of 281°F, also shown in Table 2.6-1. The most limiting temperature for the drywell shell has been analyzed to be less than 281°F and therefore remains bounded by the containment design temperature.

Predicted peak Drywell (DW) shell temperatures reported in Table 2.6-1 are obtained from steam line break analysis with the SHEX code which models [[

]] This results in the peak DW wall temperature of 280.8°F for 0.25 ft^2 break. The maximum predicted DW shell temperatures occur at the beginning of the event prior to the initiation of DW sprays. This is because the maximum DW atmosphere temperature and DW pressure conditions occur during this early period with maximum heat transfer to the DW shell. DW pressure and temperature, together with DW shell temperature, decrease when DW spray is initiated.

The maximum DW airspace and DW shell temperature occurs early in the event with the assumption that a High DW Pressure (HDWP)/Low RPV Pressure (LRPVP) LOCA signal occurs in the accident unit 10 minutes after the accident initiation. This LOCA signal at

10 minutes will result in a further 10 minute delay in the initiation of DW spray in the accident unit (20 minute total delay for DW spray initiation).

The wetwell gas space peak temperature response was calculated assuming a heat and mass transfer model between suppression pool and wetwell gas space that is calculated mechanistically. Table 2.6-1 shows the calculated peak wetwell gas space temperature for the DBA-LOCA of 174°F for EPU. The wetwell gas temperatures are bounded by the wetwell design temperature of 281°F.

2.6.3.1.2 Short-Term Containment Pressure Response

The CLTR states that the suppression pool temperature increases as a result of the higher decay heat associated with EPU. The assumption of constant pressure minimizes the effect on other aspects of the containment evaluation. The short-term containment response analysis was performed for the limiting DBA-LOCA that assumes a double-ended guillotine break of a recirculation suction line to demonstrate that EPU does not result in exceeding the containment design limits. The short-term containment pressure response analysis of a main steam line break is not performed at EPU because it is bounded by the limiting DBA-LOCA. The short-term analysis covers the blowdown period during which the maximum drywell pressure and wetwell pressure occur. The analysis was performed at 102% of EPU RTP level. The time-dependent results of the limiting short-term analysis are presented in Figures 2.6-3 through 2.6-8 and are summarized in Table 2.6-1. Table 2.6-1 also includes comparisons of the pressure values calculated for EPU to the design pressures and to pressure values from previous calculations based on the current power. Table 2.6-5 shows the comparisons of inputs between the Browns Ferry current short term design basis and the analysis at EPU. The maximum calculated containment pressure for EPU remains within the containment design pressure value of 56 psig, and thus, is acceptable and all CLTR dispositions are met.

The short-term analysis was performed at EPU conditions for three different initial containment conditions. The Design case (D) considers the most limiting initial containment conditions of 70°F in the drywell and 2.6 psig in the drywell and 1.5 psig in the wetwell. The Bounding case (B) considers initial containment conditions of 130°F in the drywell and the same pressures in the drywell and wetwell as that for design case, bounding normal operation. A Reference case (R) is also evaluated that assumes initial conditions of 150°F in the drywell and 2.6 psig in the drywell and 1.5 psig in the wetwell – initial conditions used in the Browns Ferry power rerate analysis (Reference 84). The Design case (D) and Bounding case (B) were also performed at CLTP conditions to provide comparison for evaluating the effect of operation at EPU conditions.

Lower initial drywell temperature increases the initial drywell non-condensable gas mass that is vented to the wetwell after the LOCA. This increased mass results in a higher wetwell pressure response and a higher peak drywell pressure.

The use of the Design (D) cases initial drywell temperature is to provide the most conservative hypothesized initial conditions in order to demonstrate that a DBA-LOCA initiated at the Browns Ferry EPU power level will not challenge the Browns Ferry containment design pressure

of 56 psig. The Design Case initial temperature of 70°F is well below the lowest drywell initial temperature that can be achieved with Browns Ferry operating at power and is therefore very conservative for demonstrating the maximum Browns Ferry containment pressure response at EPU conditions.

The initial drywell temperature for the Bounding (B) cases was developed with a conservative historical statistical basis, which also achieve a conservative prediction of the containment pressure response due to a DBA-LOCA at the EPU power level. The containment pressure response determined from the DBA-LOCA using conservative initial conditions is then used to determine a conservative value of 'Pa' for 10 CFR Part 50 Appendix J leakage rate testing. The Bounding Case initial temperature of 130°F represents a lower statistical bound of the 5-year historical normal drywell operating temperature during power operation of the Browns Ferry units.

Conclusion

TVA has evaluated the mass and energy release and accounted for the sources of energy identified in 10 CFR Part 50, Appendix K. Based on this, the mass and energy release analysis meets the requirements in the current licensing basis for ensuring that the analysis is conservative. Therefore, the proposed EPU is acceptable with respect to the mass and energy release for a postulated LOCA.

2.6.4 Combustible Gas Control in Containment

Regulatory Evaluation

Following a LOCA, hydrogen and oxygen may accumulate inside the containment due to chemical reactions between the fuel rod cladding and steam, corrosion of aluminum and other materials, and radiolytic decomposition of water. If excessive hydrogen is generated, it may form a combustible mixture in the containment atmosphere.

The NRC's acceptance criteria for combustible gas control in containment are based on (1) 10 CFR 50.44, insofar as it requires that plants be provided with the capability for controlling combustible gas concentrations in the containment atmosphere; (2) GDC-5, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions; (3) GDC-41, insofar as it requires that systems be provided to control the concentration of hydrogen or oxygen that may be released into the reactor containment following postulated accidents to ensure that containment integrity is maintained; (4) GDC-42, insofar as it requires that systems required by GDC-41 be designed to permit appropriate periodic inspection; and (5) GDC-43, insofar as it requires that systems required by GDC-41 be designed to permit appropriate periodic testing. Additional requirements based on 10 CFR 50.44 for control of combustible gas apply to plants with a Mark III type of containment that do not rely on an inerted atmosphere to control hydrogen inside the containment.

Specific NRC review criteria are contained in SRP Section 6.2.5.

Browns Ferry Current Licensing Basis

The General Design Criteria (GDC) listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Browns Ferry Nuclear Plant (Browns Ferry) principal design criteria predate these criteria. The Browns Ferry principal design criteria are listed in UFSAR Section 1.5, "Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, the Tennessee Valley Authority (TVA) performed a comparative evaluation of the design basis of Browns Ferry with the AEC proposed General Design Criteria of 1967. The Browns Ferry UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria set out in the July 1967 AEC release. For each group of criteria, there is a statement of TVA's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a table of references to locations in the Browns Ferry UFSAR where there is subject matter relating to the intent of that particular criteria.

While Browns Ferry is not generally licensed to the final GDC or the 1967 AEC proposed General Design Criteria, a comparison of the final GDC to the applicable AEC proposed General Design Criteria can usually be made. For the final GDC listed in the Regulatory Evaluation above, with the exception of final GDC-41, the Browns Ferry comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Browns Ferry UFSAR Appendix A: draft GDC-4. There are no draft GDCs directly associated with final GDCs-41, 42, and 43.

Combustible gas control in containment is described in Browns Ferry UFSAR Section 5.2.6, "Combustible Gas Control in Primary Containment." Browns Ferry's containment is inerted during power operation.

<u>Technical Evaluation</u>

See FUSAR Section 2.6.4.

<u>Conclusion</u>

The containment combustible gas control system was reviewed and it was found that the effects of the proposed EPU have been adequately addressed. An increase to the liquid nitrogen minimum storage volume specified in TS 3.6.3.1, which ensures a 7-day supply, is required so the system will continue to have sufficient capability following the implementation of the proposed EPU. Refer to the EPU LAR Enclosure and Attachments 2 and 3 for the proposed TS change. The containment combustible gas control system will continue to meet the requirements of the current licensing basis, as well as 10 CFR 50.44. Therefore, the proposed EPU is acceptable with respect to combustible gas control in containment.

2.6.5 Containment Heat Removal

Regulatory Evaluation

Fan cooler systems, spray systems, and RHR systems are provided to remove heat from the containment atmosphere and from the water in the containment wetwell.

The NRC's acceptance criteria for containment heat removal are based on GDC-38, insofar as it requires that a containment heat removal system be provided, and that its function shall be to rapidly reduce the containment pressure and temperature following a LOCA and maintain them at acceptably low levels.

Specific NRC review criteria are contained in SRP Section 6.2.2, as supplemented by Regulatory Guide (RG) 1.82.

Browns Ferry Current Licensing Basis

The General Design Criteria (GDC) listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Browns Ferry Nuclear Plant (Browns Ferry) principal design criteria predate these criteria. The Browns Ferry principal design criteria are listed in UFSAR Section 1.5, "Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, the Tennessee Valley Authority (TVA) performed a comparative evaluation of the design basis of Browns Ferry with the AEC proposed General Design Criteria of 1967. The Browns Ferry UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria set out in the July 1967 AEC release. For each group of criteria, there is a statement of TVA's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a table of references to locations in the Browns Ferry UFSAR where there is subject matter relating to the intent of that particular criteria.

While Browns Ferry is not generally licensed to the final GDC or the 1967 AEC proposed General Design Criteria, a comparison of the final GDC to the applicable AEC proposed General Design Criteria can usually be made. For the final GDC listed in the Regulatory Evaluation above, the Browns Ferry comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Browns Ferry UFSAR Appendix A: draft GDCs-41 and 52.

The containment heat removal systems are described in Browns Ferry UFSAR Sections 5.2, "Primary Containment System," and 4.8, "Residual Heat Removal System (RHRS)."

Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Browns Ferry License Renewal Safety Evaluation Report (SER), NUREG-1843, dated April 2006 (Reference 11). The license renewal evaluation associated with containment heat removal is

documented in NUREG-1843, Section 2.3.2. Management of aging effects on the primary containment is documented in NUREG-1843, Section 3.2.2.

Technical Evaluation

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

Browns Ferry meets all CLTR dispositions. The topics addressed in this evaluation are:

Topic	CLTR Disposition	Browns Ferry Result
Pool Temperature Response	Plant Specific	Meets CLTR Disposition
ECCS Net Positive Suction Head (NPSH)	Plant Specific	Meets CLTR Disposition

2.6.5.1 Pool Temperature Response

Section 4.2.6 of the CLTR states that the suppression pool temperature increases as a result of the higher decay heat associated with EPU. The long-term bulk pool temperature response for EPU is evaluated for the limiting RSLB DBA-LOCA in Section 14.6.3.3 (Case C of UFSAR Section 5.2.4.3) of the UFSAR. For the Browns Ferry EPU, a guillotine break of a Recirculation Discharge Line (RDLB) and small pipe breaks were also analyzed at EPU conditions. The most severe, highest, peak bulk suppression pool temperature for all LOCA break types and sizes is provided in Table 2.6-1.

Suppression Pool Temperature Response – RSLB DBA-LOCA

The analysis of the RSLB DBA-LOCA was performed at 102% of EPU RTP. The calculated SP temperature response is presented in Figure 2.6-1, the DW and WW temperature responses are presented in Figure 2.6-2, and the peak values for LOCA bulk pool temperature for the CLTP and the EPU RTP case are compared in Table 2.6-1. The EPU analysis was performed using a decay heat table based on ANS/ANSI 5.1-1979 with 2-sigma adders with additional actinides and activation products per GE SIL 636 (Reference 85). No modifications were made to this standard.

The containment system response to the accident is divided into two analysis phases. The first phase, hereafter referred to as the short-term phase covers the period up to 10 minutes after the accident initiation. During the short-term phase, no operator action is credited in the analysis. The second phase, hereafter referred to as the long-term phase covers the period after 10 minutes following the accident initiation. During the long-term phase, operator actions such as those to reduce electrical loading on the emergency diesel generators and to re-align portions of the

ECCS from core cooling mode to containment cooling mode are credited. The RSLB DBA-LOCA analysis assumes that offsite power is lost concurrently with the accident initiation and that offsite power is not available during the accident analysis period. Separate RSLB analysis cases are run for EPU with initial conditions to either maximize or minimize the containment drywell and wetwell pressure response while maximizing the suppression pool temperature response in order to determine the sensitivity of the peak suppression pool temperature response to perturbed initial conditions. No containment leakage is assumed except for the RSLB cases with initial conditions to minimize the containment drywell and wetwell pressure response while maximizing the suppression pool temperature response, for which containment leakage (2% per day) and the leakage from MSIVs (150 scfh for all steam lines) are considered. In addition, the containment responses to various modes of containment cooling are evaluated. These three RHR cooling modes are: (1) Coolant Injection Cooling (CIC), where RHR flow is cooled by the RHR heat exchanger before being discharged into the reactor vessel; (2) CSC, where RHR flow is cooled by the RHR heat exchanger and then discharged to the containment via the DW spray and wetwell spray headers; and (3) SPC, where RHR flow is cooled by the RHR heat exchanger and then discharged back to the suppression pool.

A complete LOOP is assumed to occur concurrent with the accident initiation. If a worst-case SAF such as failure of one emergency electrical power source (emergency diesel generator or loss of a 4 kV shutdown board) is assumed concurrent with the accident, then less than the full complement of low pressure ECCS pumps (four RHR pumps and four CS pumps) would be available during the short-term phase of the accident. However, if no SAF is assumed, then the full complement of ECCS pumps would be available. The initial condition of no SAF during the short-term phase is limiting for the determination of ECCS pump NPSH during the accident because of the Browns Ferry ECCS pump suction configuration where each ECCS pump does not have a dedicated ECCS suction strainer and piping suction directly from the suppression pool (torus). For each Browns Ferry unit, there are four ECCS suction strainers installed in the torus. The torus water volume then communicates to the ECCS pump suctions via a torus ring header located below the torus. This configuration result in higher ECCS piping head loss when there are multiple ECCS pumps running. In addition, a larger number of running ECCS pumps will lead to higher pump heat addition to the suppression pool. Conformance with GEH SC 06-01 (Reference 83) is made by assuming all low pressure ECCS pumps start during the short-term phase of the accident.

All ECCS pumps are assumed to be available for the first 600 seconds after accident initiation. No RHRSW flow is assumed to the RHR heat exchangers and there is no heat removal from the RHR heat exchangers during the short-term phase. RPV liquid is discharged from the break into the drywell causing rapid vessel depressurization and a rapid increase in the drywell pressure and temperature. For the first 600 seconds following the accident, four RHR pumps in LPCI mode (with two RHR pumps injecting liquid into the intact recirculation loop at a flow rate of 9,000 gpm per RHR pump and the other two RHR pumps into the broken recirculation loop at a flow rate of 9,000 gpm per RHR pump) and four CS pumps, each with flow rate of 3,550 gpm, are used to cool the core. For the RSLB DBA-LOCA, the RHR flow into the broken

recirculation loop will be directed to the RPV and RHR flow will not go into runout flow because the RHR injection point is between the RPV and the closed reactor recirculation discharge valve (the reactor recirculation discharge valve in each reactor recirculation loop receives an automatic closure signal during a LOCA). HPCI is assumed available and will start on either high DW pressure or low RPV level. However, HPCI will isolate on low steam pressure. The ECCS injection of suppression pool water, along with the assumed addition of feedwater produces a recovery of the reactor water level. This allows water heated by decay heat and vessel sensible energy to be discharged into the drywell, and subsequently into the suppression pool.

If the accident were to occur on either Unit 1 or 2 and a worst-case SAF such as failure of one emergency electrical power source (emergency diesel generator or loss of a 4 kV shutdown board) is assumed concurrent with the accident, then less than the full complement of low pressure ECCS pumps (four RHR pumps and four CS pumps) would be available during the long-term phase of the accident. Assuming that one RHR pump is required for shutdown of the non-accident unit, only two RHR pumps and two RHR heat exchangers are assumed available for long-term containment cooling in the accident unit.

After 600 seconds, operator actions are credited. One loop of CS with two CS pumps continues to be available for RPV water makeup. One loop of CS with two pumps is secured because two CS pumps can supply adequate long-term core cooling after accident initiation. The CS pump flow is 3125 gpm for each of the two CS pumps in the remaining in-service CS loop. The throttling of CS flow is not a new operator action for EPU. One loop of RHR with two pumps is secured, and another loop of RHR with two pumps is switched to a RHR mode of containment cooling with its associated RHRSW flow activated for two heat exchangers. A conservatively low RHRSW flow value of 4,000 gpm to each in-service RHR heat exchanger is assumed in the analysis. The analysis assumes operator action to throttle RHR flow to 6,500 gpm per RHR pump. The throttling of RHR flow is not a new operator action for EPU. Three RHR cooling modes are investigated: (1) CIC where RHR in LPCI mode with flow from the suppression pool is cooled by the RHR heat exchanger before being discharged into the reactor vessel; (2) CSC where RHR flow from the suppression pool is discharged as drywell and wetwell sprays; and (3) SPC where the RHR flow from the suppression pool is cooled by the RHR heat exchanger before being discharged back into the suppression pool. The heat exchanger K-value and RHR pump flow rate are presented in Table 2.6-2a. Initial conditions (initial DW pressure, initial wetwell pressure, initial DW relative humidity and initial DW temperature) are also perturbed in separate analysis cases to both maximize and minimize the peak containment pressure and thereby investigate the effect on peak suppression pool temperature. The resulting calculated peak bulk SP temperature for RSLB DBA-LOCA at 10 minutes after the accident initiation is 152.8°F and the peak bulk SP temperature for RSLB DBA-LOCA is 179.0°F.

The possible effect of containment cooling interruption on the accident unit due to concurrent shutdown and cooldown of the non-accident units was also investigated. Prior to depressurizing the non-accident unit below the RPV pressure that would result in initiating a LOCA signal due

to high drywell pressure (if high DW pressure exists) in conjunction with low RPV pressure, the operators recognize that a LOCA signal could occur and therefore inhibit the generation of a LOCA signal from the non-accident unit. This LOCA signal inhibit is performed by the operation of permanently installed hand switches in the Browns Ferry main control room and is a normal action per Browns Ferry procedures to prevent the generation of a false LOCA signal that could result in interruption of containment cooling on both the accident and the non-accident unit. Therefore, there are no additional containment cooling interruptions on the accident unit due to interaction from the non-accident units.

Suppression Pool Temperature Response –RDLB LOCA

The containment response during the first 10 minutes following the accident initiation for a RDLB was calculated using Browns Ferry specific inputs to maximize suppression pool temperature and minimize containment pressure, similar to the RSLB -LOCA analysis. The key parameter differences between the RDLB and the RSLB during the short-term phase of the accident are: 1) the break area (4.2 ft^2 for the RSLB versus 1.94 ft^2 for the RDLB), and 2) the RHR flow rate and RHR injection path into the broken recirculation loop.

For the RDLB-LOCA, all ECCS pumps are assumed to be available for the first 600 seconds. No RHRSW flow is assumed to the RHR heat exchangers and there is no heat removal from the RHR heat exchangers during the short-term phase. RPV liquid is discharged from the break into the drywell causing rapid vessel depressurization and a rapid increase in the drywell pressure and temperature. For the first 600 seconds following the accident, four RHR pumps in LPCI mode (with two RHR pumps injecting liquid into the intact recirculation loop at a flow rate of 9,000 gpm per RHR pump and the other two RHR pumps into the broken recirculation loop at a flow rate of 11,000 gpm per RHR pump) and four CS pumps, each with flow rate of 3,550 gpm, are used to cool the core. For the RDLB LOCA, the RHR flow into the broken recirculation loop discharges directly to the drywell and the RHR flow into the broken loop is assumed at runout conditions. HPCI is assumed available and will start on either high DW pressure or low RPV level. However, HPCI will isolate on low steam pressure. The ECCS injection of suppression pool water, along with the assumed addition of feedwater produces a recovery of the reactor water level. This allows water heated by decay heat and vessel sensible energy to be discharged into the drywell, and subsequently into the suppression pool. The resulting calculated peak bulk SP temperature for the RDLB at 10 minutes after the accident initiation is 152.0°F.

Suppression Pool Temperature Response –Small Steam Break LOCA

For the Browns Ferry small break LOCA, a spectrum of small steam line breaks was evaluated. Initial reactor conditions are consistent with operation at 102% of EPU RTP, and the same decay heat, relaxation and metal-water reaction energies are assumed as is used for the large DBA-LOCA analysis. Consistent with the large DBA-LOCA assumptions, a complete LOOP is assumed. A worst-case single failure is also assumed for this analysis to minimize the available quantity of containment cooling. This single failure is either the failure to start an EDG or the loss of a 4 kV shutdown board. The single failure assumption will result in no more than three CS pumps and three RHR pumps automatically starting on either low RPV level or HDWP

concurrent with LRPVP. For cases where HPCI is assumed available, HPCI will automatically start on either HDWP or on low RPV level.

Cases with HPCI (high pressure ECCS) available and with no HPCI available are evaluated to determine the effect of the availability of high pressure ECCS on the limiting peak pool temperature and the limiting drywell temperature. The condensate storage tank is assumed unavailable during the accident and HPCI pump suction is assumed available only from the suppression pool. For Browns Ferry, HPCI is qualified only for water temperatures up to 140°F. If HPCI is conservatively assumed available, HPCI will provide reactor inventory makeup until the reactor pressure decreases below the HPCI isolation pressure, after which low-pressure ECCS provides reactor inventory makeup. If HPCI is not available, ADS would be used to rapidly reduce reactor pressure to allow low-pressure ECCS to provide vessel makeup. Such use of ADS results in a faster heatup of the suppression pool. With reactor pressure at the time of peak pool temperature the same, the total (integrated) sensible heat addition to the suppression pool remains the same, but the total (integrated) decay heat to the pool at the time of peak suppression pool temperature is less for the fast pool heatup. In addition, the heat removed from the pool is greater for the faster pool heatup. Thus, a faster pool heatup will result in a lower peak suppression pool temperature. For this reason, the assumption of crediting the HPCI as available until it isolates on low steam pressure is conservative for the determination of a peak suppression pool temperature response.

Automatic starting of ECCS pumps will occur in accordance with their start logic and timing for electrical loading. Automatic start of CS and RHR will result in reactor vessel inventory makeup provided by three CS pumps and three RHR pumps in LPCI mode. Operators initiate depressurization of the RPV at 100°F/ hour when the suppression pool temperature reaches 120°F. At no sooner than 10 minutes after the start of the accident, operators will stop the third RHR pump and third CS pump. When containment conditions permit (drywell and wetwell pressures and drywell temperatures), operators will either re-align or start two RHR pumps in containment spray mode (two RHR pumps at 6,500 gpm each with two RHR heat exchangers with a K-factor of 265 BTU/sec-ºF per heat exchanger) and one CS loop (two CS pumps with maximum flow of 3,125 gpm/pump). For breaks greater than 0.01 ft^2, drywell and wetwell spray initiation is delayed by up to 1,200 seconds (20 minutes) to address concerns related to ECCS interruption caused by a subsequent LOCA signal activated on HDWP concurrent with LRPVP. For the smallest break (0.01 ft^2), the late LOCA signal will occur much later in the event and the operator would inhibit the late LOCA signal and the additional drywell spray delay will not occur. This LOCA signal inhibit is performed by the operation of permanently installed hand switches in the Browns Ferry main control room which is a normal action per Browns Ferry procedures to prevent the generation of a LOCA signal that could result in interruption of containment cooling when core cooling has already been confirmed. RPV depressurization is terminated when RPV pressure reaches 50 psig. Because Browns Ferry is a hot shutdown plant, entry into alternate shutdown cooling for entering cold shutdown is not required. Operators will maintain the plant at this pressure until shutdown cooling can be restored. The resulting calculated peak bulk SP temperature for a steam line break is 182.7°F, which occurs for the 0.01 ft^2 break. Figure 2.6-10 shows the suppression pool

temperature response for the limiting break size of 0.01 ft^2. Table 2.6-6 shows the comparisons of inputs between the Browns Ferry current long term design basis for small steam line breaks and the analysis at EPU.

The peak suppression pool temperature of 182.7°F is for the case where HPCI is assumed available and the initial DW temperature is 70°F. The sensitivity of this peak suppression pool temperature due to initial DW temperature was investigated by setting the initial DW temperature to 150°F. The resulting calculated peak bulk SP temperature for a 0.01 ft^2 steam line break with initial DW temperature of 150°F is 182.7°F, which demonstrates the insensitivity of initial DW temperature on the peak suppression pool temperature. The peak SP temperature for the limiting 0.01 ft^2 steam line break where HPCI was assumed unavailable was 181.5°F, which demonstrates that the assumption of HPCI availability during the event is conservative.

The possible effect of containment cooling interruption on the accident unit due to concurrent shutdown and cooldown of the non-accident units was also investigated. Prior to depressurizing the non-accident unit below the RPV pressure that would result in initiating a LOCA signal due to high drywell pressure (if high DW pressure exists) in conjunction with low RPV pressure, the operators recognize that a LOCA signal could occur and therefore inhibit the generation of a LOCA signal from the non-accident unit. This LOCA signal inhibit is performed by the operation of permanently installed hand switches in the Browns Ferry main control room which is a normal action per Browns Ferry procedures to prevent the generation of a false LOCA signal that could result in interruption of containment cooling on both the accident and the non-accident units. Therefore, there are no additional containment cooling interruptions on the accident unit due to interaction from the non-accident units.

Suppression Pool Temperature Response – Non-Accident Units

The suppression pool temperature response for the non-accident units was also evaluated. For the purpose of this evaluation, the term "non-accident unit" refers to the Browns Ferry unit that is both not experiencing a LOCA and has the minimum containment cooling equipment available. This evaluation is applicable to either of the following conditions: 1) a LOOP for all three Browns Ferry units (with no LOCA), or 2) a LOCA on any one unit concurrent with a simultaneous LOOP for the remaining two units. The bounding condition has been evaluated for two scenarios that either assume the CST is available or assume the CST is not available.

For Units 1 and 2, there are four 4 kV shutdown boards (4 kV shutdown board A, B, C and D) shared between the two units. Each 4 kV shutdown board is supplied during a LOOP by a safety-related EDG. Power distribution to 480V shutdown boards and 480V Reactor Motor Operated Valve (RMOV) boards is redundant in that each 480V board can be supplied power from two of the Unit 1 and 2 shared 4 kV shutdown boards. For Unit 3, there are four dedicated 4 kV shutdown boards (4 kV shutdown board 3EA, 3EB, 3EC and 3ED). Each 4 kV shutdown board for Unit 3 is supplied during a LOOP by a safety-related EDG. Power distribution to Unit 3 480V shutdown boards and 480V RMOV boards is redundant in that each 480V board can be supplied power from two of the 4 kV shutdown boards designated for Unit 3. In addition, the 4 kV electrical distribution system at Browns Ferry allows a Unit 3 EDG to either power a

de-energized Unit 1 and 2 4 kV shutdown board or to operate in parallel with a Unit 1 and 2 EDG for powering a Unit 1 and 2 4 kV shutdown board. Figure 2.6-1c is an illustration of the power distribution scheme at Browns Ferry to the 480V RMOV board level.

Conservatively assuming that the DBA-LOCA occurs concurrently with a LOOP, reactor isolation and scram will occur on the non-accident units. Concurrent with the LOOP, the worst case single failure for containment cooling is the loss of a 4 kV shutdown board (A, B, C or D) shared between Units 1 and 2. This single failure is more severe than loss of an EDG alone because it prevents repowering the lost (de-energized) 4 kV shutdown board from one of the Unit 3 EDGs. For this assumed electrical power failure, only three RHR pumps would be available for either core or containment cooling between Units 1 and 2. The LOCA analysis assumes that two of these RHR pumps would be used for long-term containment cooling in the accident unit.

Paralleling of a Unit 3 EDG with the EDG supplying power to the non-accident unit (so that two EDGs are supplying power to one Unit 1 and 2 4 kV shutdown board) is not assumed. Therefore, EDG power limitations are conservatively assumed that allow the starting and alignment of only one RHR pump and one RHR heat exchanger for containment cooling on the non-accident unit.

The loss of the 4 kV shutdown board may also result in loss of the normally aligned power to the 480V shutdown board and the 480V RMOV board that supplies power to the RHR Shutdown Cooling (SDC) isolation valves for the non-accident unit. However, the Browns Ferry electrical system configuration is such that there are redundant means of re-powering the 480V shutdown boards and the 480V RMOV boards that supply power to both additional DW coolers and the RHR SDC isolation valves. Therefore, there is no loss of the ability to place RHR SDC into service due to electrical power limitations. Drywell cooling is initially lost for the non-accident unit because the LOOP signal in conjunction with a LOCA signal on the accident unit causes the loads to be stripped from the 4 kV shutdown boards and then re-sequenced on as the EDGs repower the 4 kV shutdown boards. Within 90 seconds after the LOOP, a minimum of four drywell coolers are automatically re-started and are available for DW cooling in the non-accident unit. Operators are able to manually restart DW coolers and restore power to RHR SDC isolation valves later in the event using the redundant power sources mentioned above.

The capability of the non-accident unit to achieve cold shutdown was analyzed at 102% of EPU RTP and ANS/ANSI 5.1-1979 with 2-sigma adders decay heat. The decay heat model includes additional actinides and activation products per GE SIL 636 (Reference 85). This analysis includes the assumption of reactor shutdown initiated by a loss of offsite power (for all three Browns Ferry units) with concurrent loss of a 4 kV shutdown board shared between Units 1 and 2. Two scenarios are evaluated. Scenario 1 assumes that the Condensate Storage Tank (CST) volume is available and HPCI provides high pressure inventory makeup to the RPV with HPCI pump suction from the CST. Scenario 2 assumes that the CST volume is not available and HPCI provides high pressure inventory makeup to the RPV with HPCI pump suction from the suppression pool. Initial conditions and key input parameters for the non-accident unit containment response evaluation are shown in Table 2.6-2b.

Scenario 1 - CST Available

The event is initiated by LOOP. The LOOP causes a reactor scram, a containment isolation signal due to loss of power to the Nuclear Steam Supply System (NSSS) isolation relays, tripping of the FW pumps and loss of power to the DW coolers. The MSIVs are assumed to be fully closed at 3.5 seconds after event initiation. In the analysis, the FW temperature is initially at or above 337°F (saturation temperature is at 100 psig). Following the closure of the MSIVs, the FW is assumed to flash to steam and then is injected into the vessel. The FW mass entering the vessel after closure of the MSIVs is conservatively assumed to come into thermal equilibrium with the downstream FW piping as the FW travels toward the vessel. FW injection into the vessel is assumed to resume when the RPV pressure is reduced to below 220 psig which ensures that all hot FW at a temperature equal to and greater than 337ºF is injected into the vessel before the suppression pool temperature peaks. This assumption is conservative because the timing results in the FW enthalpy addition occurring late in the event when the SP temperature is high and will therefore result in a more conservative (higher) SP temperature response.

The MSRVs will automatically cycle to control RPV pressure. At 90 seconds into the event, four drywell coolers will have automatically restarted. The HPCI pump will automatically start on low RPV level with HPCI pump suction from the CST. At ten minutes after reactor shutdown, the operators align one loop of RHR (one RHR pump, one RHR heat exchanger and RHRSW cooling flow of 4500 gpm to the RHR heat exchanger) in suppression pool cooling mode with a flow rate of 9700 gpm. At approximately 20 minutes after the start of the event, operators are assumed to restart an additional four DW coolers to provide additional cooling to the non-accident unit drywell and restore power to NSSS isolation relays (Browns Ferry operators can perform this action within the assumed action time). When the non-accident unit SP temperature reaches 110°F, but no sooner than ten minutes after reactor shutdown, the operators commence manual reactor depressurization and reactor cooldown at a rate of 100°F/hr. Prior to depressurizing the non-accident unit below the RPV pressure that would result in initiating a LOCA signal due to high drywell pressure (if high DW pressure exists) in conjunction with low RPV pressure, the operators recognize that a LOCA signal could occur and therefore inhibit the generation of a LOCA signal from the non-accident unit. This LOCA signal inhibit is performed by the operation of permanently installed hand switches in the Browns Ferry main control room and is a normal action per Browns Ferry procedures to prevent the generation of a false LOCA signal that could result in interruption of containment cooling on the non-accident unit. HPCI is assumed to isolate on low RPV pressure when RPV pressure decreases to 150 psig. A single core spray pump is started to provide RPV inventory makeup after HPCI is no longer available. Further depressurization of the RPV to 100 psig is accomplished by opening MSRVs.

When RPV pressure reaches 100 psig, the analysis assumes that the operators will maintain the RPV at this pressure. Operators stop the RHR pump in suppression pool cooling and begin transitioning RHR to SDC mode. The transition to place SDC in operation is assumed to take 20 minutes (Browns Ferry operators confirmed that this assumed operator action time can be

achieved). During this 20 minute transition period from RHR in suppression pool cooling to SDC, there is no cooling of the suppression pool. Cooldown of the RPV to cold shutdown conditions on the non-accident unit is accomplished with SDC. Cold shutdown is achieved when bulk reactor liquid water temperature is less than or equal to 212°F. The peak bulk suppression pool cooling temperature for this analysis at EPU conditions is 185.1°F. The resulting time-dependent bulk suppression pool temperature response is presented in Figure 2.6-1a.

Scenario 2 - CST Not Available

The event is initiated by LOOP. The LOOP causes a reactor scram, a containment isolation signal due to loss of power to the Nuclear Steam Supply System (NSSS) isolation relays, tripping of the FW pumps and loss of power to the DW coolers. The MSIVs are assumed to be fully closed at 3.5 seconds after event initiation. In the analysis, the FW temperature is initially at or above 337°F (saturation temperature is at 100 psig). Following the closure of the MSIVs, the FW is assumed to flash to steam and then is injected into the vessel. The FW mass entering the vessel after closure of the MSIVs is conservatively assumed to come into thermal equilibrium with the downstream FW piping as the FW travels toward the vessel. FW injection into the vessel is assumed to resume when the RPV pressure is reduced to below 220 psig which ensures that all hot FW at a temperature equal to and greater than 337ºF is injected into the vessel before the suppression pool temperature peaks. This assumption is conservative because the timing results in the FW enthalpy addition occurring late in the event when the SP temperature is high and will therefore result in a more conservative (higher) SP temperature response.

The MSRVs will automatically cycle to control RPV pressure. At 90 seconds into the event, four drywell coolers will have automatically restarted. The HPCI pump will automatically start on low RPV level with HPCI pump suction from the SP. The CST volume is assumed to not be available, consistent with the assumptions used for the containment system response for a LOCA. HPCI provides reactor inventory makeup until SP temperature reaches 140°F. If the SP temperature reaches 140°F, HPCI is secured because HPCI availability cannot be assured with a SP temperature greater than 140°F. At ten minutes after reactor shutdown, the operators align one loop of RHR (one RHR pump, one RHR heat exchanger and RHRSW cooling flow of 4500 gpm to the RHR heat exchanger) in suppression pool cooling mode with a flow rate of 9700 gpm. At approximately 20 minutes after the start of the event, operators are assumed to restart an additional four DW coolers to provide additional cooling to the non-accident unit drywell and restore power to NSSS isolation relays (Browns Ferry operators can perform this action within the assumed action time). When the non-accident unit SP temperature reaches 110°F, but no sooner than ten minutes after reactor shutdown, the operators commence manual reactor depressurization and reactor cooldown at a rate of 100°F/hr. Prior to depressurizing the non-accident unit below the RPV pressure that would result in initiating a LOCA signal due to high drywell pressure (if high DW pressure exists) in conjunction with low RPV pressure, the operators recognize that a LOCA signal could occur and therefore inhibit the generation of a LOCA signal from the non-accident unit. This LOCA signal inhibit is performed by the operation of permanently installed hand switches in the Browns Ferry main control room and is a

normal action per Browns Ferry procedures to prevent the generation of a false LOCA signal that could result in interruption of containment cooling on the non-accident unit. When the SP temperature reaches 140°F, the analysis assumes that HPCI is secured. A single core spray pump is started to provide RPV inventory makeup after HPCI is no longer available. Further depressurization of the RPV to 100 psig is accomplished by opening MSRVs.

When the RPV pressure reaches 100 psig, the analysis assumes that the operators will maintain the RPV at this pressure. Operators stop the RHR pump in suppression pool cooling and begin transitioning RHR to SDC mode. The transition to place SDC in operation is assumed to take 20 minutes (Browns Ferry operators confirmed that this assumed operator action time can be achieved). During this 20 minute transition period from RHR in suppression pool cooling to SDC, there is no forced cooling of the suppression pool from RHR. Cooldown of the RPV to cold shutdown conditions on the non-accident unit is accomplished with SDC. Cold shutdown is achieved when bulk reactor liquid water temperature is less than or equal to 212°F. The peak bulk suppression pool cooling temperature for this analysis at EPU conditions is 180.0°F. The resulting time-dependent bulk suppression pool temperature response is presented in Figure 2.6-1b.

2.6.5.2 ECCS Net Positive Suction Head

Section 4.2.6 of the CLTR states that EPU rated thermal power operation increases the reactor decay heat, which increases the heat addition to the suppression pool during a large break LOCA, small break LOCA, shutdown of the Browns Ferry non-accident units following LOOP and accident in one Browns Ferry unit, loss of RHR SDC, Stuck Open Relief Valve (SORV) with RPV isolation, fire, SBO, and ATWS events. During these accidents, transients and special events, the RHR and CS pumps operate as necessary to provide the required core and containment cooling. Adequate NPSH margin is demonstrated during these events to assure essential pump operation. ECCS pump NPSH margin is evaluated for the limiting conditions during these accidents, transients and special events. The limiting NPSH conditions depend on the pump flow rates, suction strainer debris loading (for debris generating events), pump suction piping frictional losses, suppression pool (torus) level, and suppression pool temperature. A maximum torus pressure of 14.4 psia is assumed in the NPSH analyses. No Containment Accident Pressure (CAP) is used for calculating Net Positive Suction Head Available (NPSHa) in any event for Browns Ferry EPU. EPU evaluations of ECCS pump NPSH are consistent with RG 1.82 (Reference 96) and recent NRC CAP guidance (Reference 97).

Events resulting in fluid discharge into the drywell (through LOCA break flow or through operation of containment sprays) can “holdup” a portion of the torus inventory on the drywell floor and in the suppression system vent headers resulting in a “drawdown” of the torus water level. Consistent with the guidance of RG 1.82 (Reference 96), the NPSH analyses of these events (RSLB DBA-LOCA, RDLB LOCA, and small break LOCA) conservatively adjust the torus water level by subtracting this holdup volume from the torus water volume to reduce the inventory (torus water level) used in computing the NPSH static head term. For these events, drawdown includes the suppression pool / RPV break flow inventory that is held up in the

drywell and in the pressure-suppression system vent headers. A holdup volume of 8,304 ft^3 was used in the applicable NPSH analyses. The holdup volume was calculated based on the geometry of Browns Ferry drywell and suppression system vent headers. This holdup volume is more conservative (larger) than that used in previous Browns Ferry NPSH evaluations because previous NPSH evaluations did not include the water volume held up in the suppression system vent headers. This holdup volume is more conservative (larger) than that used in previous Browns Ferry NPSH evaluations because previous NPSH evaluations did not include the water volume held up in the suppression system vent headers.

Torus water level can also change from the initial torus water level due to RPV inventory loss (in the case of a RCPB leak or break) or due to RCIC/HPCI operation (RPV makeup) when the RCIC/HPCI pump suction is aligned from the CST. The Browns Ferry NPSH evaluations use the transient suppression pool (torus) water volume determined from the containment response safety analysis for the respective analysis event and converts the suppression pool water volume to a water level in feet above the respective ECCS pump suction centerline.

NPSHa is determined from the following equation for pumps with the suction source surface elevation above the pump elevation, such as the ECCS pumps used at Browns Ferry:

$$NPSHa = h_a + h_s - h_f - h_{vap}$$

where:

h_a = atmospheric pressure above suppression pool (torus), feet

h_s = water static height (suppression pool surface elevation - pump suction centerline elevation), feet

h_f = suction pipe and strainer frictional head loss at the respective pump flow, feet

h_{vap} = vapor pressure at the respective suppression pool temperature, feet

Consistent with Reference 97, the ECCS pump Net Positive Suction Head - Required (NPSHR) used in EPU NPSH margin evaluations contains a 21% uncertainty for the large and small break LOCA events, and 0% uncertainty for other events. The NPSHR including uncertainty (effective NPSHR or $NPSHR_{eff}$), is determined as follows:

$$NPSHR_{eff} = (1 + uncertainty)NPSHR_{3\%}$$

$NPSHR_{3\%}$ is the RHR and core spray pump NPSHR based on a 3% reduction in pump head during testing ($NPSHR_{3\%}$). The $NPSHR_{3\%}$ values will vary depending on the pump flow rate and are determined from the vendor supplied pump curves. $NPSHR_{eff}$ is used in the discussion of RHR and core spray pump NPSH margins. Table 2.6-4 provides the $NPSHR_{3\%}$ and $NPSHR_{eff}$ values used in each event evaluation for NPSH margin.

The current ECCS NPSH evaluation for Browns Ferry Units 1, 2, and 3 is contained in the Browns Ferry UFSAR (Section 6.5.5 for DBA-LOCA, Section 7.19.5 for ATWS, Section 8.10.2

for SBO) and in the Browns Ferry fire protection report for a fire event. The ECCS NPSH evaluations that support these UFSAR sections and the fire protection report predate the issuance of Reference 97. Therefore, only NPSHR without accounting for uncertainty was used in all current ECCS NPSH evaluations. The NPSHR values used in the current ECCS NPSH analyses were based on transient NPSHR curves provided by the pump vendor. The transient NPSHR curves are a function of pump flow rate and operating duration to preclude pump damage for 8,000 hours post-transient event.

A generic assessment of the 21% uncertainty term applicable to the Browns Ferry CS pumps (Sulzer Model 12x16x14.5 CVDS) was submitted to the NRC by Reference 98. A generic and plant-specific assessment of the uncertainty term for the Browns Ferry RHR pumps (Sulzer Model 18x24x28 CVIC) was submitted to the NRC by Reference 99. References 98 and 99 determined the Browns Ferry CS and RHR uncertainty to be within a generic value of 21%. The Reference 99 report conclusion states, "The results reported are representative for the specific evaluated application, the Browns Ferry CVIC RHR pumps, and provide assurance that use of a generic 21% $NPSH_{3\%}$ uncertainty for similar pumps in similar service conditions is a reasonable and bounding value." A similar statement is provided in the Reference 98 CVDS pump report. Based on the conclusions of References 98 and 99, use of a generic 21% uncertainty for the Browns Ferry RHR and CS pumps is reasonable and bounding were a plant-specific uncertainty assessment to be performed and used in the EPU NPSH margin assessment.

The NPSH margins were calculated assuming system flow rates that meet or exceed ECCS pump operational requirements for the event. A listing of the safety analysis ECCS pump flow rate and the ECCS pump flow rate used in the corresponding NPSH evaluation are shown in Table 2.6-4. Consistent with Reference 97, the ECCS pump flows used in the NPSH analysis were conservatively increased by a factor of at least $1/\sqrt{0.97}$ (1.015) to account for the reduction in pump flow rate associated with a 3% reduction in pump total developed head.

Per Sections 6.3.3 and 6.6.8 of Reference 97, the zone of maximum erosion rate should be considered to lie between NPSH margin ratios of 1.2 to 1.6 and the NRC selected a time limit of 100 hours for the time permitted in the maximum erosion zones. The RHR and core spray pump operating times with NPSH margin ratios less than 1.6 are shown in the applicable transient NPSHa figures. The margin ratio is defined by $NPSHa/NPSHR_{3\%}$. If this ratio is less than 1.6, the operating time spent below 1.6 has been determined and is shown in Table 2.6-4.

Consideration of ECCS suction strainer debris loading within the NPSH evaluations at EPU conditions is consistent with the Browns Ferry CLTP analysis of record for the large break DBA-LOCA (RSLB) event. ECCS suction strainer debris loading is only considered for LOCA (RSLB DBA-LOCA, RDLB LOCA and small break LOCA) events. For all other events, ECCS suction strainer debris loading is not assumed because high energy fluid release from the RPV and subsequent suppression pool heatup is caused only by piped MSRV discharge to the suppression pool (torus) exiting through T-quenchers below the suppression pool water level. Browns Ferry does not have any unpiped spring safety valves discharging directly into the

drywell. Table 2.6-4 indicates how ECCS suction strainer debris loading is considered for each of the analyzed events.

The Browns Ferry ECCS configuration includes an ECCS ring header circumscribing the suppression chamber (torus) with connecting piping to four inlet penetrations through the torus wall into the suppression pool. Inside the suppression pool, each connecting line is fitted with a flanged surface for mating to the ECCS strainer flanges. The ECCS ring header supplies the suction piping of the RHR, CS, HPCI and RCIC systems.

Because the ECCS ring header and the connecting piping to the ECCS strainers is common to the suction of all of the ECCS pumps, the flow and pressure distribution for the ring header and strainers is different for varying system demands. Therefore, to determine individual ECCS pump suction piping and strainer friction losses (the h_f term in the NPSHa equation), the entire torus, torus ring header and ECCS suction piping network was modeled and hydraulic analyses were performed. For each event evaluated for ECCS NPSH, separate hydraulic analyses were performed and included all possible pump combinations in order to determine the suction piping and strainer friction loss term for each running ECCS pump. The largest calculated RHR and CS pump suction piping and strainer friction loss was used in the determination of NPSHa for a given event. Table 2.6-4 shows the largest RHR and CS pump suction piping and strainer friction loss term for each of the analyzed events.

Large Break LOCA Short-Term Phase ECCS NPSH

As discussed in Sections 2.6.1.1.1.1 and 2.6.5.1, the suppression pool temperature response was evaluated at 102% EPU RTP for an instantaneous double-ended guillotine break of a recirculation suction line and an instantaneous double-ended guillotine break of a recirculation discharge line. The RSLB is the DBA-LOCA for Browns Ferry. Following a DBA-LOCA (RSLB) or large break LOCA (RDLB), the RHR and CS pumps operate to provide the required core and containment cooling. The NPSH evaluation conservatively assumes operation of the RHR and CS pumps at flow rates exceeding the values assumed in the safety analysis. The RHR and CS pumps are assumed to operate at $1/\sqrt{0.97}$ (1.015) times the flow rate credited in the safety analysis. The NPSH margins for the RHR and CS pumps were evaluated for the limiting conditions following a DBA-LOCA (RSLB) or large break LOCA (RDLB). The limiting NPSH conditions depend on the pump flow rate, debris loading on the suction strainers, pipe frictional losses, suppression pool level, and suppression pool temperature. The RSLB results in a higher suppression pool temperature at the end of the short-term (10 minute) analysis period than does the RDLB. At the end of the short-term analysis period, the suppression pool temperature for the RSLB DBA-LOCA is 152.8°F and the suppression pool temperature for the RDLB LOCA is 152.0°F. The RSLB DBA-LOCA results in a slightly higher suppression pool level at the end of the short-term analysis period than does the RDLB due to the higher break flow rate associated with the RSLB DBA-LOCA. However, the higher combined ECCS flow associated with the RDLB LOCA results in a significantly higher pump suction piping and ECCS strainer friction loss term in the NPSHa equation. The combined effect of the RDLB higher flow rates through the ECCS suction strainers and ring header, the lower peak suppression pool temperature, and

the lower suppression pool level (less RPV inventory blown down into the suppression pool from the smaller break) is that the short-term RDLB LOCA was more limiting with respect to NPSHa than the short-term RSLB DBA-LOCA. The single-most dominant of these effects was the higher suction piping friction losses associated with the higher flows through the ECCS ring header for the RDLB LOCA as compared to the RSLB DBA-LOCA. NPSHa and NPSH margin is only calculated for the CS pumps and the RHR pumps that are required for core cooling during the short term phase of the accident. The safety evaluation flow rates for the required RHR pumps and CS pumps are 9,000 gpm and 3,550 gpm, respectively. The ECCS NPSH evaluation flow rates for the required RHR and CS pumps are 9,138 gpm and 3,604 gpm, respectively. Suppression pool (torus) drawdown of 8,304 ft^3 due to break flow during the event is assumed in the NPSH evaluation.

The maximum suppression pool temperature, NPSHa, NPSH margin, and the operating time with NPSH margin ratio < 1.6 are listed in Table 2.6-4. The pump flow rates used in the ECCS NPSH evaluation are conservatively higher than those used in the safety analysis that provides the suppression pool temperature response. Time-history plots of the NPSHa are provided for the RHR and CS pumps in Figure 2.6-11a and Figure 2.6-12a, respectively.

Adequate NPSH at the NPSH evaluation flow rate demonstrates that the pump will deliver at least the required flow rate, even if the pump has insufficient NPSHa to operate at the higher run-out flow rate. That is, even if the pump is not able to deliver the run-out flow rate due to insufficient NPSH at run-out, but does have positive NPSH margin at the flow rate assumed in the applicable safety analysis, then the actual delivered pump flow rate will be between the safety analysis flow rate (where there is positive NPSH margin) and the pump run-out flow rate (where there is negative NPSH margin). Therefore, by demonstrating that there is positive NPSH margin in the short-term LOCA NPSH analysis for the required/credited pumps, it is assured that the actual flow rate that will be delivered is sufficient to satisfy the safety analysis.

As stated above, the actual delivered pump flow rate will be between the safety analysis flow rate (where there is positive NPSH margin) and the pump run-out flow rate (where there may be negative NPSH margin). Because it is assumed that the operators take actions to control the RHR and CS pumps at ten minutes (see following subsection for evaluation of Large Break LOCA Long-Term Phase ECCS NPSH), this condition, should it occur, would exist for no more than ten minutes.

During this ten minute period it is prudent to address two aspects of pump operation at these conditions: (1) whether the pump(s) could actually be operating with less than $NPSHR_{3\%}$; and (2) whether the pump(s) could sustain any damage during this ten minute period. The SECY-11-0014 guidance (Reference 97) addresses these two concerns and the Boiling Water Reactors Owners Group (BWROG) provided in-depth assessments in two BWROG reports for the Browns Ferry RHR pumps: “Pump Operation at Reduced NPSHa Conditions” (Reference 100) and “BWROG CVIC Report Task 4, Operation in Maximum Erosion Rate Zone” (Reference 101).

References 100 and 101 address the potential damage that could occur while operating in the maximum erosion rate zone and have quantitatively determined that the pumps could be operated in such conditions for a time period far in excess of the ten minutes immediately following a large break LOCA. Furthermore, the reports conclude that it reasonable to expect that a short period of low NPSHa operation, such as that occurring for ten minutes immediately following a large break LOCA, will not adversely affect the operation of the Browns Ferry pumps for a long-term large break LOCA mission.

As an alternative to the preceding discussion concerning possible pump degradation during short term operation of the Browns Ferry ECCS pumps with potential negative NPSH margin, additional hydraulic analyses were performed to determine the RHR and CS pump run-out flows based upon the intersection of the RHR/CS pump curves and the corresponding system resistance curves during the RDLB LOCA. These pump run-out flow rates are as follows:

- RHR pump run-out flow rate is 9,842 gpm to the intact RRS loop for the RDLB LOCA. For the RDLB LOCA, two RHR pumps are assumed in the NPSH evaluation to operate at this flow rate.
- RHR pump run-out flow rate to the broken RRS loop is 10,945 gpm for the RDLB LOCA. For the RDLB LOCA, two RHR pumps are assumed in the NPSH evaluation to operate at this flow rate.
- CS pump run-out flow rate is 3,830 gpm. For the RDLB LOCA, four CS pumps are assumed in the NPSH evaluation to operate at this flow rate.

The ECCS pump run-out flow rates were then used as inputs to hydraulic analyses of the suppression pool, torus ring header, ECCS suction strainers and ECCS pump suction piping to calculate the RHR and CS pump suction piping and strainer friction loss term (h_f in the NPSHa equation) for the required ECCS pumps. The limiting (highest) suction piping and strainer friction head loss term for the RHR and CS pumps is for the RDLB LOCA case. ECCS NPSH evaluations were performed with these higher suction piping and strainer friction loss terms. Based upon the vendor supplied ECCS pump curves, the increased RHR and CS pump flows of this supplemental evaluation result in increased NPSH required ($NPSHR_{3\%}$).

The Browns Ferry RHR pumps discharging into a broken loop with NPSHa less than the NPSH required during the short-term phase of the RDLB-LOCA event were evaluated for effect on subsequent pump operation during the long-term RDLB-LOCA mission by the pump vendor in Reference 100. Reference 100 concluded that a short period of low NPSHa operation would not adversely affect the operation of the Browns Ferry RHR pumps for a long-term RDLB-LOCA mission. Additionally, the pump impeller service life while operating in the maximum erosion rate zone was also evaluated by the pump vendor in Reference 101. In Reference 101, it was concluded that the impeller integrity is assured for the long-term LOCA (RSLB or RDLB) mission with the pump operated at any flow rate within the operating range, including the short-term pump run-out flow rate during a RDLB-LOCA event. Based on the evaluations of References 100 and 101, it is concluded that the Browns Ferry RHR pumps discharging into a

broken loop during the short-term phase of the RDLB-LOCA will be available for use later during the long-term RDLB-LOCA mission, such as suppression pool cooling or containment spray, with no adverse effect on the pump performance or reliability.

The $NPSHR_{eff}$, NPSHa, NPSH margin, and the operating time with NPSH margin ratio < 1.6 for this supplemental evaluation are listed in Table 2.6-4a. The results of this supplemental ECCS NPSH evaluation for the short-term phase of a large break LOCA demonstrate that there exists positive NPSH margin, without reliance on CAP, for the RHR and CS pumps that will continue to operate during the long-term phase of the Browns Ferry large break LOCA.

Large Break LOCA Long-Term Phase ECCS NPSH

As discussed in Sections 2.6.1.1.1.1 and 2.6.5.1, operators will take action following a RSLB DBA-LOCA or RDLB LOCA to secure CS and RHR pumps not needed for core or containment cooling, throttle CS flow for core cooling, and will transfer RHR from core-cooling (LPCI) mode to containment cooling mode. The large break LOCA containment analysis results presented in Sections 2.6.1.1.1.1 and 2.6.5.1 demonstrate that the suppression pool temperature response at the end of the short-term phase of the accident is 0.8°F higher for the RSLB DBA-LOCA; therefore, only the RSLB DBA-LOCA was evaluated for long-term suppression pool temperature response (greater than 10 minutes following accident initiation) because the RDLB LOCA suppression pool temperature response would be bounded by the results for the RSLB DBA-LOCA. The long-term containment response for the RSLB DBA-LOCA was performed at 102% of EPU RTP. For both the RSLB DBA-LOCA and the RDLB-LOCA, the long-term phase number of running CS and RHR pumps is identical, the number of RHR heat exchangers in service for containment heat removal is identical, the CS pump flow rates are identical, and the RHR pump flows are identical. Therefore, only the RSLB DBA-LOCA long-term phase is evaluated for ECCS NPSH because the RSLB DBA-LOCA results will bound the results for the RDLB LOCA. The NPSH evaluation assumes operation of the RHR and CS pumps at flow rates exceeding the values assumed in the safety analysis: 6,600 gpm for the RHR pumps and 3,173 gpm for the CS pumps. The RHR and CS pumps are assumed to conservatively operate at $1/\sqrt{0.97}$ (1.015) times the flow rate credited in the safety analysis. The NPSH margins for the RHR and CS pumps were evaluated for the limiting conditions following a RSLB DBA-LOCA. ECCS suction strainer debris loading consistent with the Browns Ferry analysis of record is assumed. Suppression pool (torus) drawdown of 8,304 ft^3 due to break flow and operation of containment sprays during the event is assumed in the NPSH evaluation. The limiting NPSH conditions depend on the pump flow rate, debris loading on the suction strainers, pipe frictional losses, suppression pool level, and suppression pool temperature. The maximum suppression pool temperature, NPSHa, NPSH margin, and the operating time with NPSH margin ratio < 1.6 are listed in Table 2.6-4. The pump flow rates used in the ECCS NPSH evaluation are conservatively higher than those used in the safety analysis that provides the suppression pool temperature response. Time-history plots of the NPSHa are provided for the RHR and CS pumps in Figure 2.6-11b and Figure 2.6-12b, respectively.

Small Break LOCA ECCS NPSH

As discussed in Section 2.6.5.1, the Browns Ferry suppression pool temperature response was evaluated at 102% of EPU RTP for a spectrum of small breaks (0.01 ft^2, 0.05 ft^2, 0.10 ft^2, 0.25 ft^2, 0.50 ft^2 and 1.0 ft^2). Except for the two smallest breaks, 0.01 ft^2 and 0.05 ft^2, the suppression pool temperature response results in lower pool temperatures than for the DBA-LOCA. For all small break cases, during the first 10 minutes of the event the peak suppression pool temperature is at least 20°F lower than for the DBA-LOCA (RSLB). In addition, during the first 10 minutes of the small break LOCA, the RHR and CS pumps are either not operating or are operating at minimum flow. Therefore, the RHR and CS NPSH margins for all events except for the long- term response of the 0.01 ft^2 and 0.05 ft^2 break cases are bounded by the results of the DBA-LOCA analysis. Of these two small breaks, the 0.01 ft^2 break results in the highest suppression pool temperature and will therefore result in the lowest RHR and CS pump NPSH margins.

The NPSH margins for the ECCS pumps were evaluated for the limiting conditions following a 0.01 ft^2 small break. For the small break event NPSH analysis, the RHR pumps are assumed to operate in the NPSH evaluation at 6,600 gpm, and the CS pumps are assumed to operate in the NPSH evaluation at 3,173 gpm. ECCS suction strainer debris loading equal to that assumed in the large break LOCA is conservatively assumed in the NPSH evaluation. Suppression pool (torus) drawdown of 8,304 ft^3 due to break flow and operation of containment sprays during the event is assumed in the NPSH evaluation. The limiting NPSH conditions depend on the pump flow rates, suction strainers and pipe frictional losses, suppression pool level and suppression pool temperature.

Because a small steam line break (accident event) is neither a DBA-LOCA nor a special event, the value of an appropriate uncertainty term is in the range of 0% - 21%, where 21% is the generic value selected for the Browns Ferry DBA-LOCA. As stated in Section 2.6.5.1, the suppression pool temperature response was evaluated for cases where HPCI was assumed not available and where HPCI was conservatively assumed available with suction from the suppression pool for the entire duration of the small break event. Because HPCI is only qualified for a suction temperature of up to 140°F, the assumption of HPCI available with suction from the suppression pool during the entire event is not realistic. The peak suppression pool temperature with HPCI unavailable was calculated as 181.5°F and with HPCI available, taking suction from the suppression pool, is 182.7°F. Where the core remains covered (HPCI available), fuel integrity is not challenged, and the resulting peak suppression pool temperature is 182.7°F, an appropriate uncertainty for this small line break is 0%.

To conservatively demonstrate the NPSH margin for all small breaks, the generic 21% uncertainty was initially applied to the smallest size break where HPCI was assumed available, 0.01 ft^2, which produced the highest peak suppression pool temperature, 182.7°F. The resulting NPSH margin using the generic 21% uncertainty for the limiting (core spray) pump, 0.1 ft., while conservative, is small.

A larger margin is demonstrated if a smaller, more appropriate uncertainty is justified and used, or if the NPSH margin assessment with 21% uncertainty is more appropriately applied to an event where HPCI is not available. Because the difference in vapor pressure from 182.7°F to 181.5°F is 0.5 feet, an improvement in NPSH margin of 0.5 feet results (without consideration for the other contributors to NPSH margin).

Consequently, these two different cases were examined with the more appropriate uncertainties applied: zero-percent uncertainty for the HPCI available case, where fuel would not be uncovered, and 21% uncertainty for the HPCI not available case, where RPV depressurization and use of the low pressure systems would be required (higher potential for fuel to be uncovered). For the HPCI available case with 0% uncertainty, the SP volume is 134,500 cubic feet when the peak SP temperature of 182.7°F is reached at 14,794 seconds. The corresponding CS pump NPSH margin is 4.3 feet. For the no-HPCI case with 21% uncertainty, the SP volume is 134,400 cubic feet when the peak SP temperature of 181.5°F is reached at 12,822 seconds. The corresponding NPSH margin for the no HPCI case is 0.6 feet.

The maximum suppression pool temperature, NPSHa and NPSH margin are listed in Table 2.6-4. The pump flow rates used in the ECCS NPSH evaluation are conservatively higher than those used in the safety analysis that provides the suppression pool temperature response. Time-history plots of the NPSHa for the RHR and CS pumps are provided in Figure 2.6-13a and Figure 2.6-13b, respectively.

Loss of RHR SDC ECCS NPSH

The Browns Ferry loss of RHR SDC event NPSH evaluation analyzed the mode of achieving cold shutdown where RHR provides suppression pool cooling and CS provides reactor cooling. The suppression pool transient bulk temperature response to this event was performed as part of the evaluation to determine conformance with NUREG-0783 (see Section 2.6.1.1.1.2.) The containment response analysis was performed at 102% EPU RTP.

HPCI is also assumed to operate during this event, with an assumed primary water suction source from the suppression pool, to provide RPV inventory make-up with reactor pressure above the HPCI isolation pressure. The assumption of HPCI operation is conservative for the determination of peak suppression pool temperature. However, HPCI pump suction from the suppression pool is limited to suppression pool temperatures below 140°F, the maximum allowed temperature at Browns Ferry for HPCI operation. HPCI can be secured prior to the suppression pool temperature reaching 140°F with no effect on the ability to ensure core cooling. The HPCI pump NPSH margin at 140°F suppression pool temperature is 15.8 feet with an assumed HPCI flow rate of 5,000 gpm.

For the NPSH analysis of the loss of RHR SDC event, the RHR pumps are assumed to operate at 6,600 gpm and the CS pumps are assumed to operate at 3,173 gpm. Zero percent uncertainty is applied in the determination of $NPSHR_{eff}$ because this is a non-design basis event. ECCS suction strainer debris loading and "holdup" volume are not assumed in the NPSH evaluation because there is no pipe break or operation of containment sprays during the event.

The maximum suppression pool temperature, NPSH margin, and the operating time < 1.6 margin ratio are listed in Table 2.6-4. The pump flow rates used in the ECCS NPSH evaluation are conservatively higher than those used in the safety analysis that provides the suppression pool temperature response. Time-history plots of the NPSHa are provided for the RHR and CS pumps in Figure 2.6-14a and Figure 2.6-14b, respectively.

The suppression pool temperature response analysis for the loss of RHR SDC event is also applicable for a small liquid break LOCA wherein the suppression pool cooling mode is used in lieu of the containment spray cooling mode. Because the suppression pool peak temperature response for this event is bounded by the suppression pool temperature response for the small break LOCA, the NPSHa and NPSH margin for the small liquid break LOCA is also bounded by the RHR pump and CS pump NPSHa and NPSHa margins reported in Table 2.6-4 for the small break LOCA.

Stuck Open Relief Valve (SORV) with RPV Isolation ECCS NPSH

The suppression pool bulk temperature response due to a SORV with RPV isolation event was evaluated at 102% EPU RTP. The suppression pool transient bulk temperature response to this event was performed as part of the evaluation to determine conformance with NUREG-0783 (see Section 2.6.1.1.1.2.) For this event, RHR operates in suppression pool cooling mode and CS operates to provide RPV coolant inventory makeup at low reactor pressure.

HPCI is also assumed to operate during this event, with an assumed primary water suction source from the suppression pool, to provide RPV inventory make-up with reactor pressure above the HPCI isolation pressure. The assumption of HPCI operation is conservative for the determination of peak suppression pool temperature. However, HPCI pump suction from the suppression pool is limited to suppression pool temperature below 140°F, the maximum allowed temperature at Browns Ferry for HPCI operation. HPCI can be secured prior to suppression pool temperature reaching 140°F with no effect on the ability to ensure core cooling. The HPCI pump NPSH margin at 140°F suppression pool temperature is 15.8 feet with an assumed HPCI flow rate of 5000 gpm.

For the SORV event NPSH analysis, the RHR pumps are assumed to operate at 6,600 gpm and the CS pumps operate at 3,173 gpm. Zero percent uncertainty is applied in the determination of $NPSHR_{eff}$ because this is a non-design basis event. ECCS suction strainer debris loading and "holdup" volume are not assumed in the NPSH evaluation because there is no pipe break or operation of containment sprays during the event. The maximum suppression pool temperature, NPSHa, NPSH margin, and the operating time with a margin ratio less than 1.6 are listed in Table 2.6-4. The pump flow rates used in the ECCS NPSH evaluation are conservatively higher than those used in the safety analysis that provides the suppression pool temperature response. Time-history plots of the NPSHa are provided for the RHR and CS pumps in Figure 2.6-15a and Figure 2.6-15b, respectively.

Fire Event ECCS NPSH

In the containment response analysis for a Browns Ferry fire event as described in Section 2.5.1.4.2, a single RHR pump is the only ECCS pump assumed to operate in order to achieve fire event safe shutdown. The containment response to a fire event was performed at EPU RTP. In the ECCS NPSH evaluation, a RHR flow rate of 7,615 gpm is used. For Browns Ferry, the limiting fire event scenario terminates following initiation of Alternate Shutdown Cooling (ASDC) and safe and stable conditions are achieved. Zero percent uncertainty is applied in the determination of $NPSHR_{eff}$ for this special event. ECCS suction strainer debris loading and "holdup" volume are not included in the NPSH evaluation because there is no assumption of a pipe break or operation of containment sprays during the event. A nominal initial suppression pool level was assumed, which is consistent with the NRC guidance contained in Reference 97.

The maximum suppression pool temperature, NPSHa, NPSH margin, and the operating time with a margin ratio less than 1.6 are listed in Table 2.6-4. The RHR pump flow rate used in the ECCS NPSH evaluation are conservatively higher than those used in the safety analysis that provides the suppression pool temperature response. A time-history plot of the NPSHa for the limiting fire event is provided in Figure 2.6-16. This case demonstrates positive NPSH margin and thus, CAP credit is not required. However, the small margin prompted a further sensitivity case to show increased margin.

The sensitivity case involved an analysis where a postulated 1,000 hp electric-driven Emergency High Pressure Makeup Pump (EHPMP) could be used as defense-in-depth to inject water from the CST through the FW piping and into the RPV while the RHR pump was operating in ASDC mode. This effectively provides a means of pumping CST inventory through the RPV and into the torus to increase the suppression pool mass, providing more mass to accept the heat input from the RPV while at the same time increasing the suppression pool level which would increase the RHR pump NPSHa. This case used the Browns Ferry TS value of 95°F for RHRSW temperature. This case also used the EPU design RHR heat exchanger K-value of 287 BTU/sec-°F. Further details concerning the determination of the RHR heat exchanger K-value are contained in LAR Attachment 39.

The results from the sensitivity case described above are provided in Table 2.6-4b. The sensitivity case is provided for comparison purposes only. The improvement in the NPSH margin by using the EHPMP is 2.9 feet compared to the results contained in Table 2.6-4 for the fire event.

Station Blackout ECCS NPSH

The Browns Ferry SBO event described in Section 2.3.5 postulates that on-site and off-site power are lost for the entire four hour coping duration. The containment response to SBO was performed at EPU RTP. Core cooling is maintained with high pressure injection systems (HPCI and/or RCIC) taking suction from the CST and excess reactor steam is vented to the suppression pool using MSRVs. At the end of the four hour coping period, RHR pumps are operated in

suppression pool cooling mode. NPSH concerns for the SBO event are related to the suppression pool level, pump suction strainer and suction piping friction losses, and peak suppression pool temperature at the end of the four hour coping period when suppression pool cooling is initiated. For the SBO event, the only ECCS pumps operating with suction from the suppression pool are the RHR pumps. The assumed RHR pump flow for the SBO NPSH evaluation is 6,600 gpm. Zero percent uncertainty is applied in the determination of $NPSHR_{eff}$ for this special event. ECCS suction strainer debris loading and "holdup" volume are not included in the NPSH evaluation because there is no assumption of a pipe break or operation of containment sprays during the event. The maximum suppression pool temperature, NPSHa, NPSH margin, and the operating time with a margin ratio less than 1.6 for the SBO scenario is listed in Table 2.6-4. The pump flow rate used in the ECCS NPSH evaluation is conservatively higher than that used in the safety analysis that provides the suppression pool temperature response. The HPCI pumps are also credited for the SBO event, operating for a maximum of 30 minutes with suction from the CST only. A time-history plot of the NPSHa is provided in Figure 2.6-17.

ATWS ECCS NPSH

As discussed in Section 2.8.5.7, the limiting event with respect to peak suppression pool temperature is the ATWS-LOOP event (two RHR pumps / heat exchangers) which results in a peak suppression pool temperature of 173.3°F at EPU RTP. The most limiting non-LOOP (four RHR pumps / heat exchangers) ATWS event is main stem isolation valve closure (MSIVC) EOC, which experiences a peak suppression pool temperature of 171.7°F. The ATWS events were analyzed at EPU RTP. Similar to the previous discussion concerning the effect of total pump flow on ECCS pump suction piping and strainer friction loss (RSLB DBA-LOCA versus RDLB LOCA short-term discussion), when the combined transient effects of suppression pool temperature, suppression pool level and ECCS pump suction losses are considered, the ATWS events resulting in the least NPSH margin is the non-LOOP event (MSIVC EOC) where the pump suction piping losses exceed the gain from a lower peak pool temperature. The NPSH margin is 14.2 feet for this non-LOOP event whereas the NPSH margin for the LOOP event is 15.5 feet. Consequently, with no CAP credit, there is substantial NPSH margin for all the ATWS events.

For the ATWS event, the only ECCS pumps operating from the suppression pool are the RHR pumps. HPCI supplies makeup to the RPV with suction from the CST. The CS pumps are not credited for the ATWS event. The assumed RHR pump flow for the ATWS event NPSH analysis is 6,600 gpm. Zero percent uncertainty is applied in the determination of $NPSHR_{eff}$ for this special event. ECCS suction strainer debris loading and "holdup" volume are not included in the NPSH evaluation because there is no assumption of a pipe break or operation of containment sprays during the event. The Browns Ferry RPV pressure relief system uses only MSRVs that are piped to discharge headers (T-quenchers) below the torus water level. There are no un-piped spring safety valves that discharge directly into the drywell and could contribute to ECCS suction strainer debris loading during an ATWS event. The maximum suppression pool temperature, NPSHa, NPSH margin, and the operating time < 1.6 margin ratio for the ATWS

event are listed in Table 2.6-4. The pump flow rates used in the ECCS NPSH evaluation are conservatively higher than those used in the safety analysis that provides the suppression pool temperature response. A time-history plot of the NPSHa is provided in Figure 2.6-18.

Shutdown of the Non-Accident Unit Following LOOP and Accident in the Accident Unit ECCS NPSH

The suppression pool temperature response during shutdown and cooldown of the non-accident Browns Ferry units during an accident, including DBA-LOCA (on the accident unit) concurrent with loss of offsite power and loss of a 4 kV shutdown board is discussed in Section 2.6.5.1. This event results in the non-accident Browns Ferry unit entering into shutdown cooling mode in order to achieve cold shutdown conditions. Evaluation of the shutdown of the non-accident unit was performed at 102% EPU RTP. For the non-accident unit safe shutdown NPSH analysis, a single RHR pump in the non-accident unit is assumed to operate at 10,000 gpm, which is conservatively higher than the RHR pump flow rate (9,700 gpm) assumed in the safety analysis that provides the suppression pool temperature response. For the non-accident unit NPSH analysis CS is assumed to operate at 3,173 gpm to provide RPV inventory makeup when HPCI is not available. HPCI is assumed available for part of this event. HPCI can be operated either with suction from the CST or HPCI operation can be secured prior to the suppression pool temperature reaching the 140°F qualification limit for HPCI. Zero percent uncertainty is applied in the determination of $NPSHR_{eff}$ because this is a non-design basis event. ECCS suction strainer debris loading and "holdup" volume are not assumed in the NPSH evaluation because there is no pipe break or operation of containment sprays during the event. The maximum suppression pool temperature, NPSH margin, and the operating time with a margin ratio less than 1.6 is listed inTable 2.6-4. Time history plots of NPSHa for the RHR and CS pumps are provided in Figure 2.6-19a and Figure 2.6-19b, respectively.

ECCS NPSH Summary

EPU RTP operation increases the reactor decay heat, which increases the heat addition to the suppression pool following a DBA-LOCA or other events. The peak suppression pool temperature for the analyzed accidents and transients is within the design capability of the ECCS pumps. Adequate NPSHa is demonstrated, and no credit for CAP is needed. The ECCS pump operating time with a margin ratio less than 1.6 is much less than 100 hours for any event.

The debris generated and transported following a LOCA that can cause ECCS strainer head loss includes fiber, reflective metal insulation, qualified coatings, dirt/dust, rust flakes, sludge, and unqualified coatings. The ECCS suction strainers are passive, stacked-disc strainers, which were designed, manufactured and tested by GE Nuclear Energy. The ECCS strainer design debris load, which was used as an input to the strainer design, is documented Reference 38. The quantity and characterization of the strainer debris loading is based on the methodology in Reference 102. The Browns Ferry design basis ECCS suction strainer debris loading was evaluated and is not affected by EPU.

The ECCS pumps have been analyzed for plant-specific conditions and have sufficient NPSH margin to perform satisfactorily under all accident and transient conditions. Therefore, all CLTR dispositions are met for ECCS pump NPSH at EPU conditions for Browns Ferry.

Conclusion

TVA has evaluated the containment heat removal systems and addressed the effects of the proposed EPU. The evaluation indicates that the systems will continue to meet their operational criteria with respect to rapidly reducing the containment pressure and temperature following a LOCA and maintaining them at acceptably low levels. Therefore, the proposed EPU is acceptable with respect to containment heat removal systems.

2.6.6 Secondary Containment Functional Design

Regulatory Evaluation

The secondary containment structure and supporting systems of dual containment plants are provided to collect and process radioactive material that may leak from the primary containment following an accident. The supporting systems maintain a negative pressure within the secondary containment and process this leakage.

The NRC's acceptance criteria for secondary containment functional design are based on (1) GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, and be protected from dynamic effects (e.g., the effects of missiles, pipe whipping, and discharging fluids) that may result from equipment failures; and (2) GDC-16, insofar as it requires that reactor containment and associated systems be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment.

Specific NRC review criteria are contained in SRP Section 6.2.3.

Browns Ferry Current Licensing Basis

The General Design Criteria (GDC) listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Browns Ferry Nuclear Plant (Browns Ferry) principal design criteria predate these criteria. The Browns Ferry principal design criteria are listed in UFSAR Section 1.5, "Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, the Tennessee Valley Authority (TVA) performed a comparative evaluation of the design basis of Browns Ferry with the AEC proposed General Design Criteria of 1967. The Browns Ferry UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria set out in the July 1967 AEC release. For each group of criteria, there is a statement of TVA's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a

table of references to locations in the Browns Ferry UFSAR where there is subject matter relating to the intent of that particular criteria.

While Browns Ferry is not generally licensed to the final GDC or the 1967 AEC proposed General Design Criteria, a comparison of the final GDC to the applicable AEC proposed General Design Criteria can usually be made. For the final GDC listed in the Regulatory Evaluation above, the Browns Ferry comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Browns Ferry UFSAR Appendix A: draft GDCs-10, 40, and 42.

The secondary containment systems are described in Browns Ferry UFSAR Sections 5.3, "Secondary Containment System," and 12.2, "Principal Structures and Foundations."

Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Browns Ferry License Renewal Safety Evaluation Report (SER), NUREG-1843, dated April 2006 (Reference 11). The license renewal evaluation associated with the secondary containment is documented in NUREG-1843, Sections 2.3.2.1 and 2.4.1.1. Management of aging effects on the secondary containment is documented in NUREG-1843, Sections 3.2.2 and 3.5.2.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of EPUs. Section 4.5 of the CLTR addresses the effect of EPU on the SGTS.

The SGTS is designed to maintain secondary containment at a negative pressure and to provide an elevated release path for the exhaust air for removal of fission products potentially present during abnormal conditions. By minimizing ground level release and providing for an elevated release point for the airborne particulates and halogens, the SGTS limits off-site dose following a postulated DBA. The SGTS fission product control and removal function evaluation is described in Section 2.5.2.1. Generic bounding analyses have been performed with results located in Section 4.5 of the CLTR. The results of this evaluation are given below.

Browns Ferry meets all CLTR dispositions. The topics addressed in this evaluation are:

Topic	CLTR Disposition	Browns Ferry Result
Flow Capacity	Generic	Meets CLTR Disposition

The CLTR states that the core inventory of iodine and subsequent loading on the SGTS filter or charcoal adsorbers are affected by EPU.

The design flow capacity of the SGTS was selected to maintain the secondary containment at the required negative pressure to minimize the potential for exfiltration of air from the reactor

building. [[

]] Therefore, Browns Ferry HEPA filters are satisfactory for EPU operation.

The secondary containment structure, openings, and pathways and drawdown time are unaffected by EPU. Because the maximum dome pressure is also not changed for EPU, there is no effect on the ability of secondary containment to contain mass and energy released to it. There is no increase in mass and energy released to secondary containment for EPU. The secondary containment temperature and pressure are not evaluated further in the CLTR because there is no effect as a result of EPU. Therefore, the evaluation of the SGTS ability to maintain secondary containment at a negative pressure and contain radionuclides is adequate for this topic. Therefore, the flow capacity meets all CLTR dispositions.

Conclusion

TVA has evaluated the ability of the secondary containment to provide an essentially leak-tight barrier against uncontrolled release of radioactivity to the environment. The evaluation indicates that the secondary containment and associated systems will continue to provide an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment following implementation of the proposed EPU. Based on this, the secondary containment and associated systems will continue to meet the requirements of draft GDCs-10, 40, and 42. Therefore, the proposed EPU is acceptable with respect to secondary containment functional design.

Table 2.6-1 Browns Ferry Containment Performance Results

Parameter	CLTP	CLTP –with EPU Model (1)	EPU –with EPU Model	Design Limit
Peak Drywell Pressure (psig) (2, 7)	50.6 (11) - U2, U3 48.5 (12) - U1	50.6 (D) 48.8 (B) 48.2 (R)(13)	50.9 (D) 49.1 (B) 48.5 (R)	56.0
Peak Drywell Temperature (°F)(3)	297 (8, 11) -U2, U3 295.2 (8,12) - U1	336.6 (6)	336.9 (6)	281 (4)
Peak Bulk Pool Temperature (°F)	177 (10, 11) - U2, U3 187.3 (10,12) - U1	172.1 (5)	179.0 (5)	281 (4)
Peak Wetwell Pressure (psig) (5)	36.3 (9) - U2, U3 30.5 (12) - U1	29.8	30.2	56.0
Peak Wetwell Temperature (°F) (5)	N/A	167	174	281 (4)

Notes:

1. Containment analyses performed for the EPU use methods that are similar to the methods used for the CLTP analyses. Analyses performed at 102% of 3458 MWt. The analysis at CLTP with the EPU Model uses the plant inputs defined for the EPU model including improved RHR heat exchanger performance (as discussed in Section 2.6.1.1.1).
2. Most limiting values obtained from the short-term analysis.
3. Most limiting values of drywell atmosphere temperature obtained from the long-term steam line break analysis performed for environmental qualification of equipment in the drywell.
4. Temperature limit is the design temperature for the containment vessel (shell). Maximum calculated drywell shell temperature is 280.8°F, which does not exceed the drywell shell design limit temperature of 281°F.
5. Peak values for long-term DBA-LOCA analysis. A peak bulk suppression pool temperature of 185.1°F was calculated for the non-accident unit shutdown for EPU. A peak bulk suppression pool temperature of 182.7°F at EPU was calculated for the small steam line break LOCA analysis performed for maximum drywell temperature for environmental qualification, and is therefore the most limiting peak bulk suppression pool temperature for all LOCA break sizes. The heat exchanger K value is 265 BTU/sec-ºF and the RHRSW temperature of 95ºF was used.
6. Refer to Figure 2.6-9 for SHEX output at EPU. This peak value occurs for the break size of 0.25 ft^2 with initial drywell temperature of 70ºF. For EPU, the peak drywell airspace temperature is 335.2ºF for an initial drywell temperature of 150ºF, compared to 336.9ºF for an initial drywell temperature of 70ºF.
7. Three cases are reported, Design (D), Bounding (B), and Reference (R). The Design case assumes an initial drywell temperature of 70°F. The Bounding case assumes an initial drywell temperature of 130°F - conditions corresponding to the lower bound Browns Ferry normal operating DW temperature and conditions corresponding to the upper bound Browns Ferry normal operating containment pressure. The Reference cases assumes an initial drywell temperature of 150°F - the same initial drywell temperature used in the Reference 84 (Unit 2, Unit 3) and Reference 103 (Unit 1) analyses. These cases were performed at CLTP and EPU reactor conditions in order to provide a comparison for the effect of operation at EPU conditions. All cases were run at 102% power and 105% flow. CLTP for Units 2 and 3 used NFWT. CLTP for Unit 1 used FWTR. CLTP with EPU model and EPU used FWTR.

8. This value is for a recirculation line liquid break. The Unit 2 and Unit 3 peak drywell airspace temperature for a steam line break is 336ºF (Reference 104). The Unit 1 peak drywell airspace temperature for a steam line break is 335.4ºF (Reference 105).
9. Values are from UFSAR Figure 14.6-5 or Figure 3-4 from Reference 104 for DBA-LOCA.
10. For Unit 2 and Unit 3, the heat exchanger K value is 223 BTU/sec-ºF and the RHRSW temperature of 92ºF was used. For Unit 1, the heat exchanger K value is 223 BTU/sec-ºF and the RHRSW temperature of 95ºF was used.
11. Value for Unit 2 and Unit 3 is from Reference 84. The Unit 2 and Unit 3 DBA-LOCA analysis was performed at 102% of 3458 MWt with an initial DW temperature of 150°F.
12. Value for Unit 1 is from Reference 103. The Unit 1 DBA-LOCA analysis was performed at 102% of 3952 MWt with an initial DW temperature of 150°F.
13. The reduction in peak pressure for Browns Ferry Units 2 and 3 is due to the selection of bounding mass and energy release data points for input into the GEH M3CPT code that more closely match the GEH LAMB code break flow output as compared to the selection used for the Browns Ferry Unit 2 and 3 analysis for power uprate supporting CLTP (Reference 84). This technique, which is consistent with the current analysis used for Unit 1, results in lower mass and energy release to the drywell, which produces a lower peak drywell pressure and temperature at the same power level. The Unit 2 and 3 CLTP peak drywell pressure value of 50.6 psig is reduced to 48.2 psig using this technique with all other containment analysis inputs and modeling (LAMB code using Moody Slip critical break flow model input into M3CPT) unchanged.

Table 2.6-2a Containment Response Key Analysis Input Values

No	Parameter	Unit	Analysis Value
1.	**Reactor**		
a.	Initial power level		
	1. 102% current rated power	MWt	3,527
	2. 102% uprated power	MWt	4,031
b.	Feedwater temperature		
	1. Normal Feedwater Temperature (FWT) at 102% CLTP	°F	381.7
	2. Normal FWT at 102% EPU	°F	394.5
	3. Reduced FWT at 102% EPU	°F	339.8
c.	Initial vessel dome pressure		
	1. At 102% current rated power	psia	1,055
	2. At 102% uprated power	psia	1,055
d.	Decay heat model		
	1. Short-term DBA-LOCA		ANS 5 + 20%
	2. Long-term		ANS 5.1 + 2σ
e.	Vessel volumes		
	1. Total vessel free volume	ft^3	20,682
	2. Liquid vessel volume	ft^3	11,790
f.	Vessel related masses (used in long-term calculation)		
	1. Liquid mass in recirculation loops	lbm	63,560
	2. Liquid mass in the HPCI piping between the RPV nozzle and first normally closed valve	lbm	8,621
	3. Liquid mass in the RCIC piping between the RPV nozzle and first normally closed valve	lbm	1,245
	4. Liquid mass in the RHR piping between the RPV and the first normally closed valve	lbm	9,535
	5. Liquid mass in the CS piping between the RPV nozzle and the first normal closed valve.	lbm	2,622

Table 2.6-2a Containment Response Key Analysis Input Values (continued)

No	Parameter	Unit	Analysis Value
g.	Time at which MSIVs start to close Fully closed	sec	0.5 3.5
2.	**Drywell/Vent System**		
a.	Total drywell free volume (including vent system)	ft^3	159,000 to 171,000 (Note 1)
b.	Initial drywell pressure (range)	psia	15.5 to 17.0 (Note 5)
c.	Initial drywell temperature	°F	70 (Design) 130 (Bounding) 150 (Reference)
d.	Initial drywell relative humidity (range)	%	20 to 100 (Note 6)
e.	Elevation of downcomer exit from bottom of suppression pool	ft	11.5
f.	Downcomer Submergence		
	1. LWL	ft	2.92
	2. High water level (HWL)	ft	3.83
g.	Loss coefficient for vent system including entrance and exit losses (based on vent exit flow area)		5.32 (Note 2)
h.	Downcomer internal diameter	ft	1.958
3.	**Wetwell/Suppression Pool**		
a.	Initial suppression pool volume (including water in vents)		
	1. LWL	ft^3	122,940 (Note 7)
	2. HWL	ft^3	131,400
b.	Initial suppression pool temperature	°F	95 (Note 8)
c.	Total suppression chamber volume excluding the volume occupied by the vent system		
	1. LWL	ft^3	135,000 (Note 3)
	2. HWL	ft^3	119,400

Table 2.6-2a Containment Response Key Analysis Input Values (continued)

No	Parameter	Unit	Analysis Value
d.	Initial wetwell/containment airspace pressure (range)	psia	14.4 to 15.9 (Note 9)
e.	Initial wetwell/containment airspace temperature	°F	95
f.	Initial wetwell/containment airspace relative humidity	%	100
4.	**RHR**		
a.	Heat exchanger K-value	BTU/sec-°F/HX	265
b.	Service water temperature	°F	95
c.	Drywell spray flow rate (two RHR pumps)	gpm	12,350
d.	Wetwell spray flow rate (two RHR pumps)	gpm	650
e.	RHR flow rate in suppression pool cooling mode (two RHR pumps)	gpm	13,000
5.	**Wetwell-to-Drywell Vacuum Breakers**		
a.	Pressure difference between wetwell and drywell for vacuum breakers to be fully open	psid	0.5 (Note 4)
b.	Number of vacuum breaker assemblies	6	
c.	Flow area of each vacuum breaker assembly at which loss coefficient is given below	ft^2	1.41
d.	Total loss coefficient of each vacuum breaker assembly		0.45

Notes:

1 Vent thrust loads and LOCA analyses to minimize the containment pressure are calculated assuming a minimum DW volume of 159,000 ft^3.

2 Plant specific vent system pressure loss coefficients were developed during the Mark I containment long-term program in Table 4.1.1-2 of Reference 78. This value is Browns Ferry specific and used for containment analysis.

3 This value is used for containment long-term analyses.

4 For LOCA analyses that minimize the containment pressure response, the pressure difference between the wetwell and drywell for the vacuum breakers to be fully open of 0.05 psid is conservatively used.

5 For short and long term containment analyses designed to maximize the containment pressure response, an initial DW pressure of 17.0 psia is used. For long term analyses designed to minimize the containment pressure response, an initial DW pressure of 15.5 psia is used.

6 For analyses designed to minimize the containment pressure response an initial DW RH value of 100% RH is used. For analyses designed to maximize the containment pressure response, an initial DW RH value of 20% RH is used. A lower initial DW RH value maximizes the mass of non-condensible gas in the DW and

therefore leads to a higher containment pressure response. The suppression pool temperature response is not sensitive to the initial DW RH parameter.

7 All containment analyses designed to maximize the suppression pool temperature response assume an initial low suppression pool volume corresponding to LWL.

8 TS maximum value for normal operation.

9 Initial value is DW initial pressure value minus 1.1 psi for DW-WW pressure control per Browns Ferry TS 3.6.2.6.

Table 2.6-2b Non-Accident Unit Containment Response Key Analysis Input Values

Number	Parameter	Unit	Value
1	**Reactor**		
A	Initial power level 102% uprated power	MWt	4,031
B	Initial FW temperature at 102% uprated power (102% of 3,952 MWt)	°F	396.6
C	Initial vessel dome pressure at 102% uprated power	psia	1,055
D	Decay heat model	N/A	1979 ANS 5.1 + 2σ
E	Vessel volumes		
1	Total vessel free volume	ft^3	20,682
2	Vessel liquid volume - subcooled	ft^3	7,926
3	Vessel liquid volume - saturated	ft^3	3,864
F	Vessel related masses		
1	Liquid mass in main steam lines to the inboard isolation valve	lbm	0
2	Liquid mass in one recirculation loop	lbm	31,780
3	Liquid mass in the HPCI piping between the RPV nozzle and first normally closed valve	lbm	8,621
4	Liquid mass in RHR/LPCI shutdown piping between the RPV nozzle and first normally closed valve	lbm	9,535
5	Liquid mass in the RCIC piping between the RPV nozzle and first normally closed valve	lbm	1,245
6	Liquid mass in the CS piping between the RPV nozzle and the first normal closed valve.	lbm	2,622

Table 2.6-2b Non-Accident Unit Containment Response Key Analysis Input Values (continued)

Number	Parameter	Unit	Value
G	MSIV Closure		
1	Time at which MSIVs start to close	sec	0.5
2	Time at which MSIVs become fully closed	sec	3.5
2	**Drywell**		
A	Total drywell airspace volume	ft^3	171,000
B	Initial drywell pressure	psia	15.5
C	Initial drywell temperature	°F	150
D	Initial drywell relative humidity	%	20
3	**Wetwell/Suppression Pool**		
A	Initial suppression pool volume LWL	ft^3	122,940
B	Initial suppression pool temperature	°F	95
C	Initial wetwell airspace free volume - LWL in suppression pool	ft^3	135,000
D	Initial wetwell airspace pressure	psia	14.4
E	Initial wetwell airspace temperature	°F	95
F	Initial wetwell airspace relative humidity	%	100
4	**RHR**		
A	Heat exchanger K-value	BTU/sec-°F	302
B	Service water temperature	°F	95
C	RHR flow rate in suppression pool cooling mode	gpm	9,700

Table 2.6-2b Non-Accident Unit Containment Response Key Analysis Input Values (continued)

Number	Parameter	Unit	Value
D	RHR flow rate in SDC mode	gpm	9,700
E	Number of RHR loops for cooling (one RHR loop is one RHR pump and one RHR heat exchanger)	N/A	1
5	**RHR Service Water**		
A	RHRSW flowrate through one RHR heat exchanger	gpm	4,500
6	**Condensate Storage Tank**		
A	Condensate storage tank volume available for RPV inventory makeup	ft^3	135,000
B	Condensate storage tank temperature	°F	130
7	**Drywell Air Cooler**		
A	Heat removal capability of each DW air cooler	BTU/hour	636,000

Table 2.6-3 Browns Ferry Peak Suppression Pool Temperatures for Postulated ATWS, Station Blackout, and NFPA 805 Events

Event	Peak Suppression Pool Temperature
Limiting ATWS (Loss of Off-Site Power)	173.3ºF
Station Blackout	203.7ºF
NFPA 805 Fire	208.0ºF

Table 2.6-4 ECCS Pump EPU NPSH Summary

Event		RSLB Long Term	RDLB Short Term		Small Steam Break	Loss of Shutdown Cooling	SORV	Non-Accident Unit Shutdown	SBO	ATWS	Fire Event
	Event Type	Design Basis Accident	Accident		Accident	Abnormal Operational Transient	Abnormal Operational Transient	Abnormal Operational Transient	Special Event	Special Event	Special Event
Parameter	**Units**										
Number of operating RHR pumps	NA	2	2	2	2	2	2	1	4[(1)]	4	1
RHR pump flow rate in safety analysis	(gpm)	6,500	11,000	9,000	6,500	6,500	6,500	9,700	6,500	6,500	7,500
RHR pump flow in NPSH analysis	(gpm)	6,600	11,169	9,138	6,600	6,600	6,600	10,000	6,600	6,600	7,615
Number of operating CS pumps	NA	2	4		2	1	1	1	N/A	N/A	N/A
CS pump flow rate in safety analysis	(gpm)	3,125	3,550		3,125	3,125	3,125	3,125	N/A	N/A	N/A
CS pump flow in NPSH analysis	(gpm)	3,173	3,604		3,173	3,173	3,173	3,173	N/A	N/A	N/A
Total flow rate in ring header for NPSH discussion	(gpm)	19,546	55,030		19,546	16,373	16,373	13,173	26,400	26,400	7,615
Suction strainer debris loading assumed?	NA	Yes	Yes		Yes	No	No	No	No	No	No

Note:

1. The SBO analysis sequence of events shown in Table 2.3-4b states that two RHR pumps and two RHR heat exchangers are placed in service at the end of the four hour coping period. The SBO NPSH evaluation assumed a more limiting case where four RHR pumps are placed in service at the end of the four hour coping period. The configuration of four running RHR pumps results in a higher suction piping and strainer friction loss term and a more limiting NPSH margin determination than for a two running RHR pump configuration.

Table 2.6-4 ECCS Pump EPU NPSH Summary (continued)

RHR Pump NPSH Summary

Event		RSLB Long Term	RDLB Short Term	Small Steam Break	Loss of Shutdown Cooling	SORV	Non-Accident Unit Shutdown	SBO	ATWS	Fire Event
	Event Type	Design Basis Accident	Accident	Accident	Abnormal Operational Transient	Abnormal Operational Transient	Abnormal Operational Transient	Special Event	Special Event	Special Event
Parameter	Units									
Peak SP Temperature (PPT)	(°F)	179.0	152.0	181.5	178.3	161.8	185.1	203.7	171.7	208.0
h_a, atmospheric pressure above SP	(feet)	34.2	33.9	34.3	34.2	34.0	34.3	34.5	34.1	34.6
RHR h_s, water static height (1)	(feet)	14.7	14.1	14.4	14.9	14.5	15.9	16.1	15.4	15.3
RHR h_f, suction pipe and strainer friction loss	(feet)	2.8	12.4	2.8	2.5	2.5	2.3	3.6	3.6	1.31
h_{vap}, vapor pressure @ PPT	(feet)	17.5	9.2	18.5	17.2	11.7	20.0	29.8	14.7	32.6
RHR pump available NPSH ($NPSHa = h_a + h_s - h_f - h_{vap}$)	(feet)	28.7	26.4	27.4	29.4	34.3	27.9	17.2	31.2	16.04
RHR pump required NPSH ($NPSHR_{3\%}$)	(feet)	17.0	18.0	17.0	17.0	17.0	21.0	17.0	17.0	16.0
RHR pump NPSH uncertainty	(%)	21	21	21	0	0	0	0	0	0
RHR pump $NPSHR_{eff}$ ($\{1+NPSH_{uncertainty}\} \times NPSHR_{3\%}$)	(feet)	20.6	21.8	20.6	17.0	17.0	21.0	17.0	17.0	16.0
RHR pump NPSH margin ($NPSHa - NPSHR_{eff}$)	(feet)	8.1	4.7	6.8	12.4	17.3	6.9	0.2	14.2	0.04
RHR pump minimum NPSH ratio ($NPSHa/NPSHR_{3\%}$)	NA	1.7	1.5	1.6	1.7	2.0	1.3	1.0	1.8	1.0
Time RHR pump NPSH ratio < 1.6	(hours)	0	< 1	0	0	0	< 1	< 3	0	< 16

Note:

1. The water static height is the difference between the SP level and the RHR pump suction centerline elevation. The Browns Ferry SP (torus) zero elevation is at a plant elevation of 521.5 feet. The RHR pump suction centerline elevation is at a plant elevation of 521.6 feet. These values are applicable to all three Browns Ferry units.

Table 2.6-4 ECCS Pump EPU NPSH Summary (continued)

CS Pump NPSH Summary

Event		RSLB Long Term	RDLB Short Term	Small Steam Break	Loss of Shutdown Cooling	SORV	Non-Accident Unit Shutdown	SBO Note 1	ATWS Note 1	Fire Event Note 1
	Event Type	Design Basis Accident	Accident	Accident	Abnormal Operational Transient	Abnormal Operational Transient	Abnormal Operational Transient	Special Event	Special Event	Special Event
Parameter	**Units**									
Peak SP Temperature (PPT)	(°F)	179.0	152.0	181.5	178.3	161.8	185.1	N/A	N/A	N/A
h_a, Atmospheric pressure above SP	(feet)	34.2	33.9	34.3	34.2	34.0	34.3	N/A	N/A	N/A
CS h_s, Water static height (2)	(feet)	15.0	14.4	14.7	15.2	14.8	16.2	N/A	N/A	N/A
CS h_f, suction pipe and strainer friction loss	(feet)	5.7	12.3	5.7	1.9	1.9	1.9	N/A	N/A	N/A
h_{vap}, vapor pressure @ PPT	(feet)	17.5	9.2	18.5	17.2	11.7	20.0	N/A	N/A	N/A
CS pump available NPSH ($NPSHa = h_a + h_s - h_f - h_{vap}$)	(feet)	26.1	26.8	24.8	30.4	35.2	28.6	N/A	N/A	N/A
CS pump required NPSH ($NPSHR_{3\%}$)	(feet)	20.0	20.0	20.0	20.0	20.0	20.0	N/A	N/A	N/A
CS pump NPSH uncertainty	(%)	21	21	21	0	0	0	N/A	N/A	N/A
CS pump $NPSHR_{eff}$ ($\{1+NPSH_{uncertainty}\}$ x $NPSHR_{3\%}$)	(feet)	24.2	24.2	24.2	20.0	20.0	20.0	N/A	N/A	N/A
CS pump NPSH margin ($NPSHa - NPSHR_{eff}$)	(feet)	1.9	2.6	0.6	10.4	15.2	8.6	N/A	N/A	N/A
CS pump minimum NPSH ratio ($NPSHa/NPSHR_{3\%}$)	NA	1.3	1.3	1.2	1.5	1.76	1.4	N/A	N/A	N/A
Time CS pump NPSH ratio < 1.6	(hours)	< 18	< 1	< 16	< 1	0	< 1	N/A	N/A	N/A

Notes:

1. Core spray pumps do not operate during these events.
2. The water static height is the difference between the SP level and the CS pump suction centerline elevation. The Browns Ferry SP (torus) zero elevation is at a plant elevation of 521.5 feet. The CS pump suction centerline elevation is at a plant elevation of 521.3 feet. These values are applicable to all three Browns Ferry units.

Table 2.6-4a ECCS Pump EPU NPSH Summary - Supplemental Evaluation

Event		RDLB Short Term	
Parameter	**Units**		
Number of operating RHR pumps	NA	2	2
RHR pump flow in NPSH analysis	(gpm)	10,945	9,842
Number of operating CS pumps	NA	4	
CS pump flow in NPSH analysis	(gpm)	3,830	
Total flow rate in ring header for NPSH discussion	(gpm)	56,894	
Suction strainer debris loading assumed?	NA	Yes	
RHR Pump Evaluation			
Peak SP Temperature (PPT)	(°F)	152.0	
h_a, Atmospheric pressure above SP	(feet)	33.9	
RHR h_s, Water static height	(feet)	14.1	
RHR h_f, suction pipe & strainer friction loss	(feet)	13.5	
h_{vap}, vapor pressure @ PPT	(feet)	9.2	
RHR pump Available NPSH ($NPSHa = h_a + h_s - h_f - h_{vap}$)	(feet)	25.3	
RHR pump Required NPSH ($NPSHR_{3\%}$)	(feet)	20.0	
RHR pump NPSH uncertainty	(%)	21	
RHR pump $NPSHR_{eff}$ ($\{1+NPSH_{uncertainty}\}$ x $NPSHR_{3\%}$)	(feet)	24.2	
RHR pump NPSH margin ($NPSHa - NPSHR_{eff}$)	(feet)	1.1	
RHR pump minimum NPSH ratio ($NPSHa/NPSHR_{3\%}$)	NA	1.3	
Time RHR pump NPSH ratio < 1.6	(hours)	< 1	

Table 2.6-4a ECCS Pump EPU NPSH Summary - Supplemental Evaluation (Continued)

Event		RDLB Short Term
Parameter	**Units**	
CS Pump Evaluation		
Peak SP Temperature (PPT)	(°F)	152.0
h_a, Atmospheric pressure above SP	(feet)	33.9
CS h_s, Water static height	(feet)	14.4
CS h_f, suction pipe & strainer friction loss	(feet)	13.6
h_{vap}, vapor pressure @ PPT	(feet)	9.2
CS pump Available NPSH (NPSHa = $h_a + h_s - h_f - h_{vap}$)	(feet)	25.5
CS pump Required NPSH ($NPSHR_{3\%}$)	(feet)	21.0
CS pump NPSH uncertainty	(%)	21
CS pump $NPSHR_{eff}$ ($\{1+NPSH_{uncertainty}\}$ x $NPSHR_{3\%}$)	(feet)	25.4
CS pump NPSH margin (NPSHa - $NPSHR_{eff}$)	(feet)	0.1
CS pump minimum NPSH ratio (NPSHa/$NPSHR_{3\%}$)	NA	1.2
Time CS pump NPSH ratio < 1.6	(hours)	< 1

Table 2.6-4b Fire Event ECCS Pump Sensitivity Cases

Event		Fire Event with HPMU Pump
	Case Type	Sensitivity
	Event Type	Special Event
Parameter	**Units**	
INPUTS for Containment Analysis		
RHR heat exchanger K-value	(BTU/sec-°F)	287
Initial torus volume (nominal initial SP level)	(ft^3)	125,400
RHRSW temperature	(°F)	95
OUTPUT from Containment Analysis		
Peak SP Temperature (PPT)	(°F)	206.2
NPSH ANALYSIS		
Number of operating RHR pumps	NA	1
RHR pump flow rate in safety analysis	(gpm)	7,500
RHR pump flow in NPSH analysis	(gpm)	7,615
Number of operating CS pumps	NA	0
CS pump flow rate in safety analysis	(gpm)	N/A
CS pump flow in NPSH analysis	(gpm)	N/A
Total flow rate in ring header for NPSH discussion	(gpm)	7,615
Suction strainer debris loading assumed	NA	No
Peak SP Temperature (PPT)	(°F)	206.2
h_a, Atmospheric pressure above SP	(feet)	34.6
RHR h_s, Water static height	(feet)	17.0
RHR h_f, suction pipe and strainer friction loss	(feet)	1.3
h_{vap}, vapor pressure @ PPT	(feet)	31.4
RHR pump Available NPSH ($NPSH_a = h_a + h_s - h_f - h_{vap}$)	(feet)	18.9
RHR pump Required NPSH ($NPSHR_{3\%}$)	(feet)	16.0
RHR pump NPSH uncertainty	(%)	0
RHR pump $NPSHR_{eff}$ ($\{1+NPSH_{uncertainty}\}$ x $NPSHR_{3\%}$)	(feet)	16.0
RHR pump NPSH margin (NPSHa - $NPSHR_{eff}$)	(feet)	2.9

Table 2.6-5 Input Comparisons for Containment Short-Term Analysis between CLTP and EPU

Parameter	Current Design Analysis Input Value(3)	EPU Analysis Input Value(3)
RSLB critical flow model	[[	
Break flow area		]]
Decay heat model	1971 ANS 5 + 20%	1971 ANS 5 + 20%
Percentage of initial reactor thermal power	102%	102%
Initial reactor pressure	1,053 psia	1,055 psia
Initial containment pressure	2.6 psig Drywell 1.5 psig Wetwell	2.6 psig Drywell 1.5 psig Wetwell
Initial containment temperature	150ºF Drywell 95ºF Wetwell	70ºF, 130ºF and 150ºF Drywell 95ºF Wetwell
Initial containment relative humidity	100% Wetwell 20% Drywell	100% Wetwell 20% Drywell
Initial suppression pool level	High Water Level	High Water Level
Initial suppression pool temperature	95ºF	95ºF
Initial downcomer submergence height	3.83 ft	3.83 ft
Downcomer pressure loss coefficient	5.32 (non-dimensional units) (Note 1)	5.32 (non-dimensional units) (Note 1)
Drywell holdup volume	N/A	N/A
Time from scram at which MSIV starts to close	0.5 seconds	0.5 seconds

Table 2.6-5 Input Comparisons for Containment Short-Term Analysis between CLTP and EPU (continued)

Parameter	Current Design Analysis Input Value (3)	EPU Analysis Input Value (3)
MSIV closure time	3 seconds	3 seconds
Time from scram at which FW isolation valve starts to close	[[	
FW valve closure time		]]
FW temperature	381.7ºF ([[ ]].) (Note 2)	394.5ºF ([[ ]].) (Note 2)
Drywell free volume	171,000 ft³	171,000 ft³
Wetwell free gas space volume	119,400 ft³	119,400 ft³

Notes:

1. Plant specific vent system pressure loss coefficients were developed during the Mark I containment long-term program in Table 4.1.1-2 of Reference 78. This value is Browns Ferry specific and is used for containment analysis.
2. The feedwater temperatures are based on reactor heat balances at 100% CLTP or 100% EPU and 100% rated core flow.
3. The larger drywell volume of 171,000 ft^3 (compared to the minimum DW volume of 159,000 ft^3. See table 2.6-2a) results in a larger initial drywell non-condensable gas mass and more non-condensable gas transferred to the wetwell during a LOCA. This maximizes the wetwell and drywell pressure and is conservative.

Table 2.6-6 Input Comparisons Between CLTP and EPU for Limiting Long Term SP Temperature Analysis

Parameter	Current Design Analysis Input Value	EPU Analysis Input Value	Justification for Differences Between Current and EPU Analysis Input Value if EPU Value is Less Conservative
SSLB critical flow model	[[		N/A
Break flow area		]]	More break sizes are used for EPU.
Decay heat model	ANS 5.1-1979 + 2σ	ANS 5.1-1979 + 2σ	N/A
Percentage of initial reactor thermal power	102%	102%	N/A
Initial containment pressure	2.6 psig	2.6 psig	N/A

Table 2.6-6 Input Comparisons Between CLTP and EPU for Limiting Long Term SP Temperature Analysis (continued)

Parameter	Current Design Analysis Input Value	EPU Analysis Input Value	Justification for Differences Between Current and EPU Analysis Input Value if EPU Value is Less Conservative
Initial containment pressure	2.6 psig	2.6 psig	N/A
Initial containment temperature	[[		]]
Initial containment relative humidity	20% Drywell 100% Wetwell	20% Drywell 100% Wetwell	N/A

Table 2.6-6 Input Comparisons Between CLTP and EPU for Limiting Long Term SP Temperature Analysis (continued)

Parameter	Current Design Analysis Input Value	EPU Analysis Input Value	Justification for Differences Between Current and EPU Analysis Input Value if EPU Value is Less Conservative
Initial suppression pool level/SP volume	Low water level corresponding SP liquid volume of 121,500 ft^3 (with zero drywell to torus differential pressure)	Low water level corresponding SP liquid volume of 122,940 ft^3 (with drywell to torus differential pressure)	EPU value is determined by accounting for the operating DW-to-Wetwell operating differential pressure. The current design analysis value assumed SP volume without DW/WW pressure control of 1.1 psid between the wetwell and drywell in service. The EPU analysis assumed DW/WW pressure control in service. Browns Ferry Technical Specifications require the DW/WW pressure control to be in service.
Initial suppression pool temperature	95ºF	95ºF	N/A
Initial downcomer submergence height	2.92 ft (Note 1)	2.92 ft (Note 1)	N/A
Downcomer pressure loss coefficient	5.32 (non-dimensional units)	5.32 (non-dimensional units)	N/A

Table 2.6-6 Input Comparisons Between CLTP and EPU for Limiting Long Term SP Temperature Analysis (continued)

Parameter	Current Design Analysis Input Value	EPU Analysis Input Value	Justification for Differences Between Current and EPU Analysis Input Value if EPU Value is Less Conservative
Drywell holdup liquid volume	[[ ]]	3,823 ft^3	For EPU drywell holdup volume is realistically modeled.
Time from scram at which MSIV starts to close	0.5 seconds	0.5 seconds	N/A
MSIV closure time	3 seconds	3 seconds	N/A
Time from scram at which FW valve starts to close	Note 2	Note 2	NA
FW valve closure time	[[ ]].(Note 2)	[[ ]] (Note 2)	[[ ]] See Note 2.

Table 2.6-6 Input Comparisons Between CLTP and EPU for Limiting Long Term SP Temperature Analysis (continued)

Parameter	Current Design Analysis Input Value	EPU Analysis Input Value	Justification for Differences Between Current and EPU Analysis Input Value if EPU Value is Less Conservative
FWT	381.7ºF	394.5ºF	The feedwater temperatures are based on reactor heat balances at 100% CLTP and 100% EPU at 100% rated core flow. Note: [[ ]]
Drywell free volume	159,000 ft³	171,000 ft³	[[ ]]
Wetwell free gas space volume	129,300 ft³	135,000 ft³	The wetwell free gas volumes correspond to the initial SP water volumes. The previously calculated wetwell free gas volume was incorrect. The EPU value is the correct value.

Table 2.6-6 Input Comparisons Between CLTP and EPU for Limiting Long Term SP Temperature Analysis (continued)

Parameter	Current Design Analysis Input Value	EPU Analysis Input Value	Justification for Differences Between Current and EPU Analysis Input Value if EPU Value is Less Conservative
RHR heat exchanger K-value	223 BTU/sec-ºF for one RHR pump with flow rate of 6,500 gpm and one service water (SW) pump with flow rate of 4,000 gpm.	265 BTU/sec-ºF for one RHR pump with flow rate of 6,500 gpm and one SW pump with flow rate of 4,000 gpm.	EPU K-factor values based on Browns Ferry specific heat exchanger testing. Further justification is provided in the ECCS NPSH evaluation.
RHR operating mode (spray or SP cooling)	Note 3	Note 3	[[ ]]
RHR heat exchanger hot side flow	6,500 gpm	6,500 gpm	N/A
RHR heat exchanger cold side flow	4,000 gpm (Note 4)	4,000 gpm (Note 4)	N/A
Total core spray flow to reactor	7,100 gpm	7,100 gpm when time < 600 seconds 6,250 gpm when time > 600 seconds	The core spray flow rate after 600 seconds in the EPU analysis is consistent with the ECCS LOCA analysis for fuel response.

Table 2.6-6 Input Comparisons Between CLTP and EPU for Limiting Long Term SP Temperature Analysis (continued)

Parameter	Current Design Analysis Input Value	EPU Analysis Input Value	Justification for Differences Between Current and EPU Analysis Input Value if EPU Value is Less Conservative
Total HPCI flow to the reactor	4,500 gpm	4,500 gpm [[ ]]	[[ ]]
Thermal conductor shape (for example wall, hollow cylinder, or rod)	Note 5	Note 5	Note 5
Thermal conductor material of each shape	Note 5	Note 5	Note 5
Thermal conductor heat transfer area of each shape	Note 5	Note 5	Note 5

Table 2.6-6 Input Comparisons Between CLTP and EPU for Limiting Long Term SP Temperature Analysis (continued)

Parameter	Current Design Analysis Input Value	EPU Analysis Input Value	Justification for Differences Between Current and EPU Analysis Input Value if EPU Value is Less Conservative
Thermal conductor heat transfer coefficient for each shape	Note 5	Note 5	Note 5

Notes:

1. These values are for zero DW-to-WW pressure. With ~1.1 psid between DW and WW, these values are slightly larger (~ 3.0 ft).
2. [[

   ]]
3. [[

   ]]
4. [[

   ]] For current design basis analyses, service water is 92ºF. For EPU, the service water temperature is 95ºF.
5. [[

]]

Table 2.6-6a Heat Sink Input Comparisons Between CLTP and EPU for Limiting Long Term SP Temperature Analysis

Current Design Analysis Heat Sinks

Drywell Heat Sink Parameter	Node 1
[[	
	]]

Wetwell Airspace Heat Sink Parameter	Node 1
[[	
	]]

Suppression Pool Heat Sink Parameter	Node 1
[[	
	]]

Table 2.6-6a Heat Sink Input Comparisons Between CLTP and EPU for Limiting Long Term SP Temperature Analysis (continued)

EPU Analysis Heat Sinks

Drywell Heat Sink Parameter	Node 1	Node 2	Node 3	Node 4	Node 5
[[					
					]]

Wetwell Airspace Heat Sink Parameter	Node 1	Node 2
[[		
		]]

Suppression Pool Heat Sink Parameter	Node 1
[[	
	]]

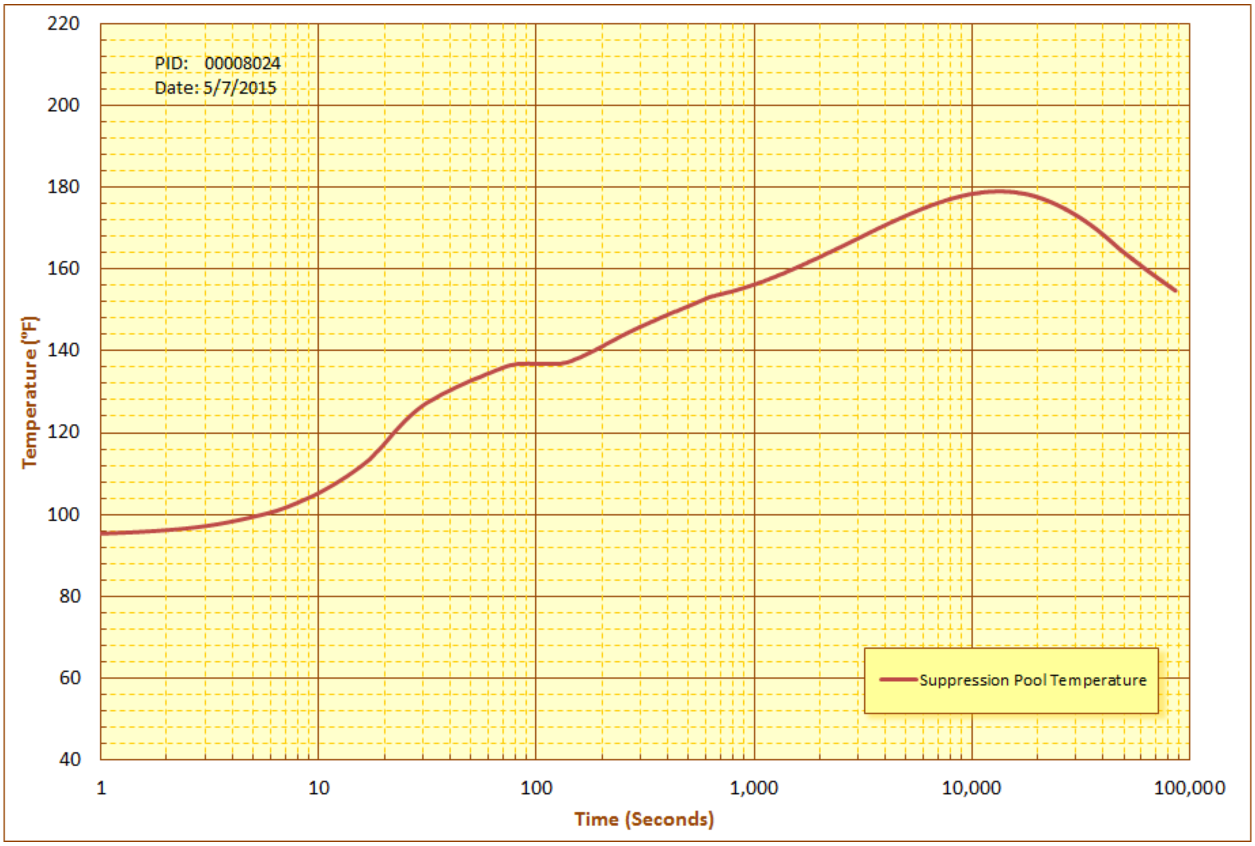


Figure 2.6-1 EPU Suppression Pool Temperature Response to RSLB DBA-LOCA (CIC)

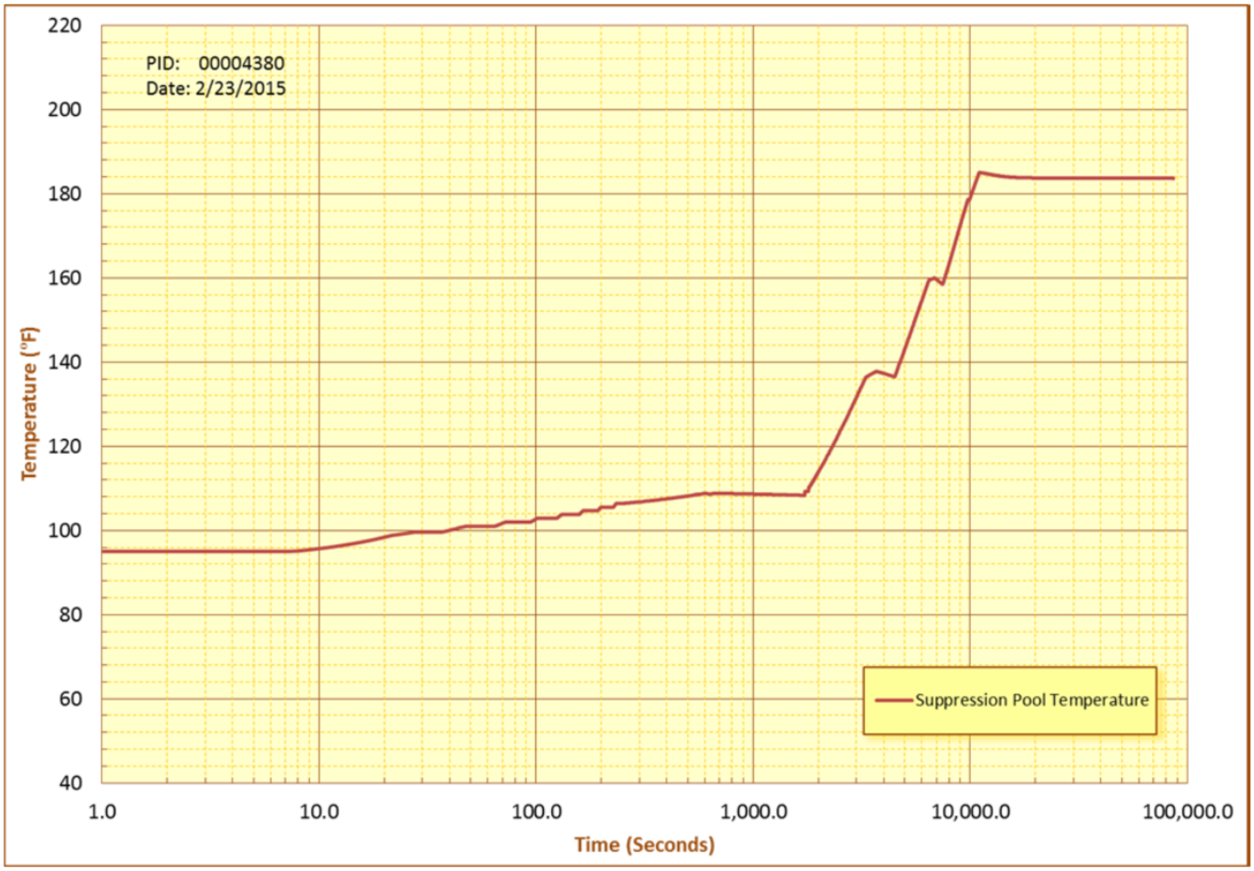


Figure 2.6-1a EPU SP Temperature Response of Non-Accident Unit Shutdown - CST Available

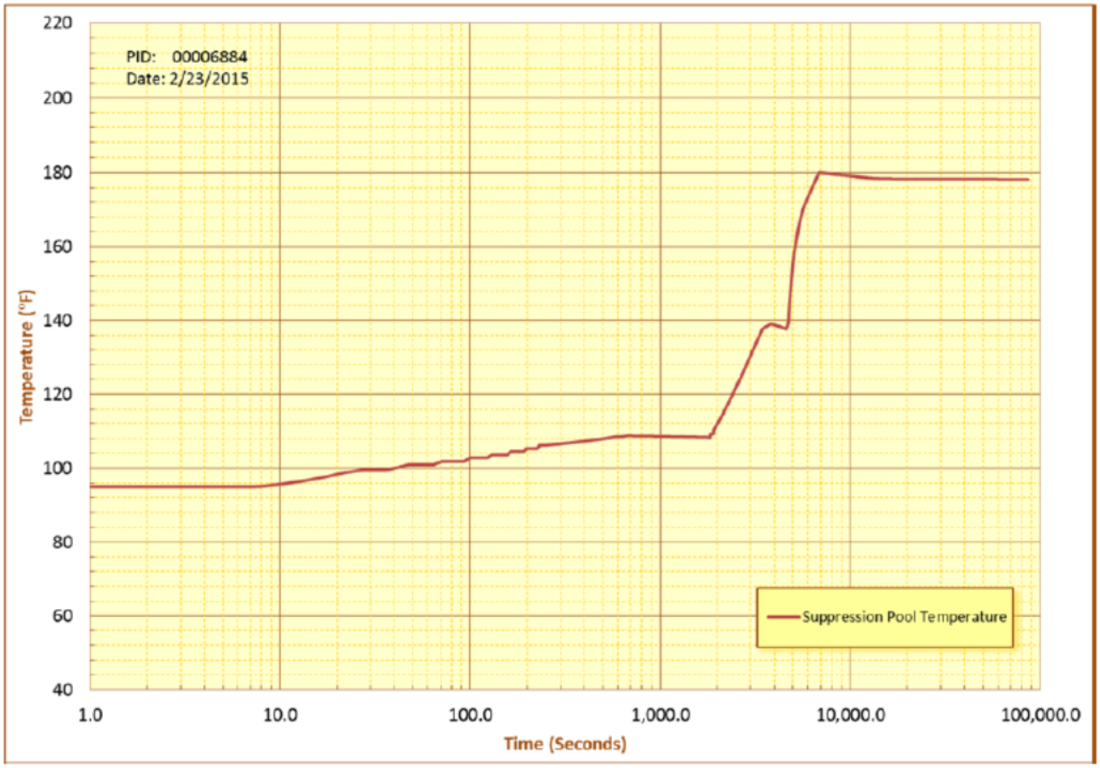
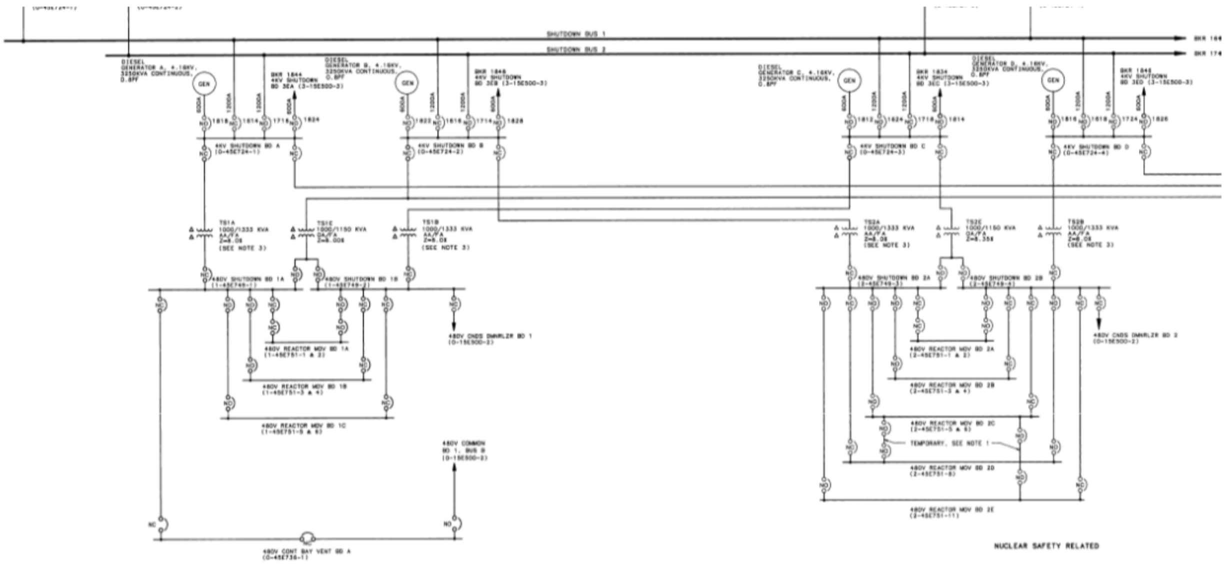
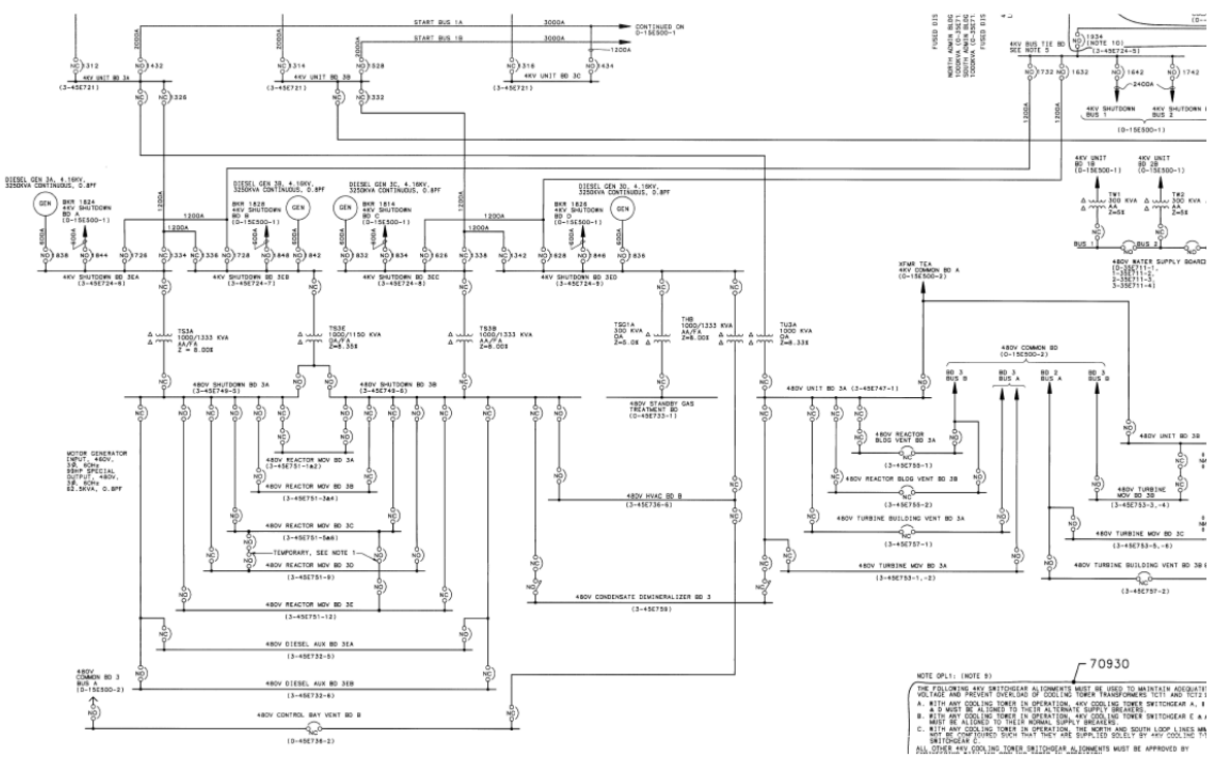


Figure 2.6-1b EPU SP Temperature Response of Non-Accident Unit Shutdown - CST Not Available



Unit 1 and Unit 2 4 kV Shutdown, 480V Shutdown, and 480V RMOV Boards



Unit 3 4 kV Shutdown, 480V Shutdown, and 480V RMOV Boards

Figure 2.6-1c Illustration of Browns Ferry 4 kV Distribution System to 480V RMOV Board Level

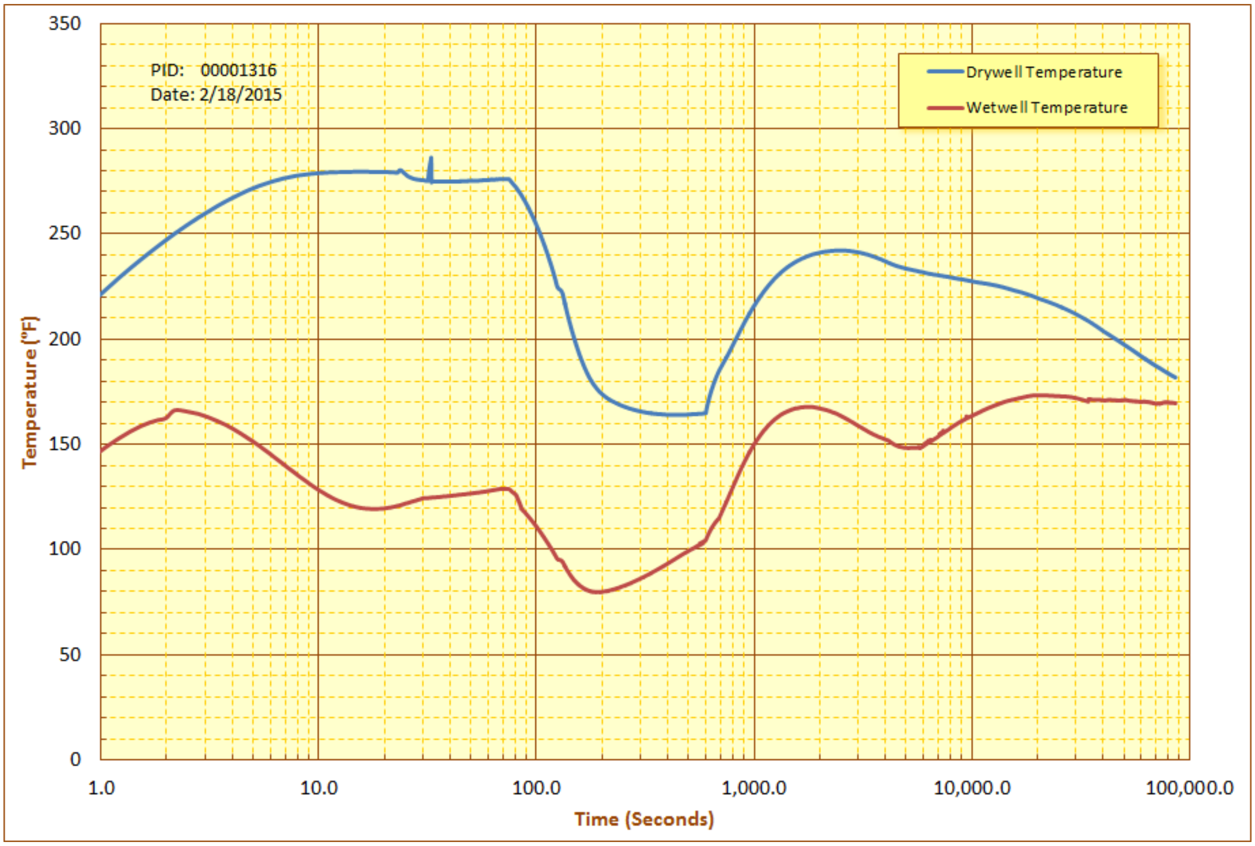


Figure 2.6-2 EPU DW and WW Temperature Response to RSLB DBA-LOCA (SPC)

Note: This figure is generated from the calculations designed to maximize containment pressure and SP temperature. This figure shows a short duration DW temperature excursion within the first 30 to approximately 40 seconds of the LOCA. [[

]]

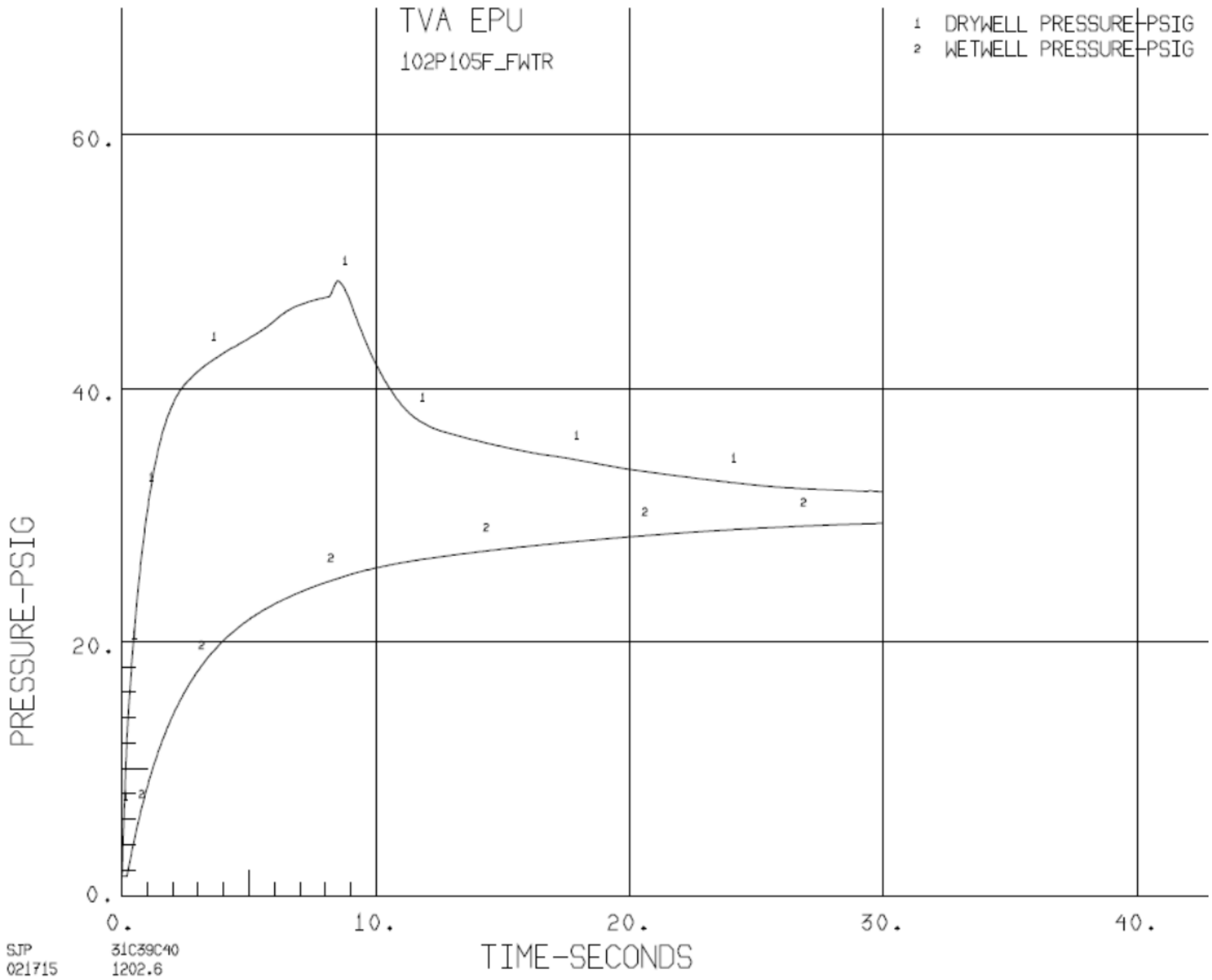


Figure 2.6-3 EPU Short-Term RSLB DBA-LOCA Containment Pressure Response (Reference Condition: Initial DW Temperature=150ºF)

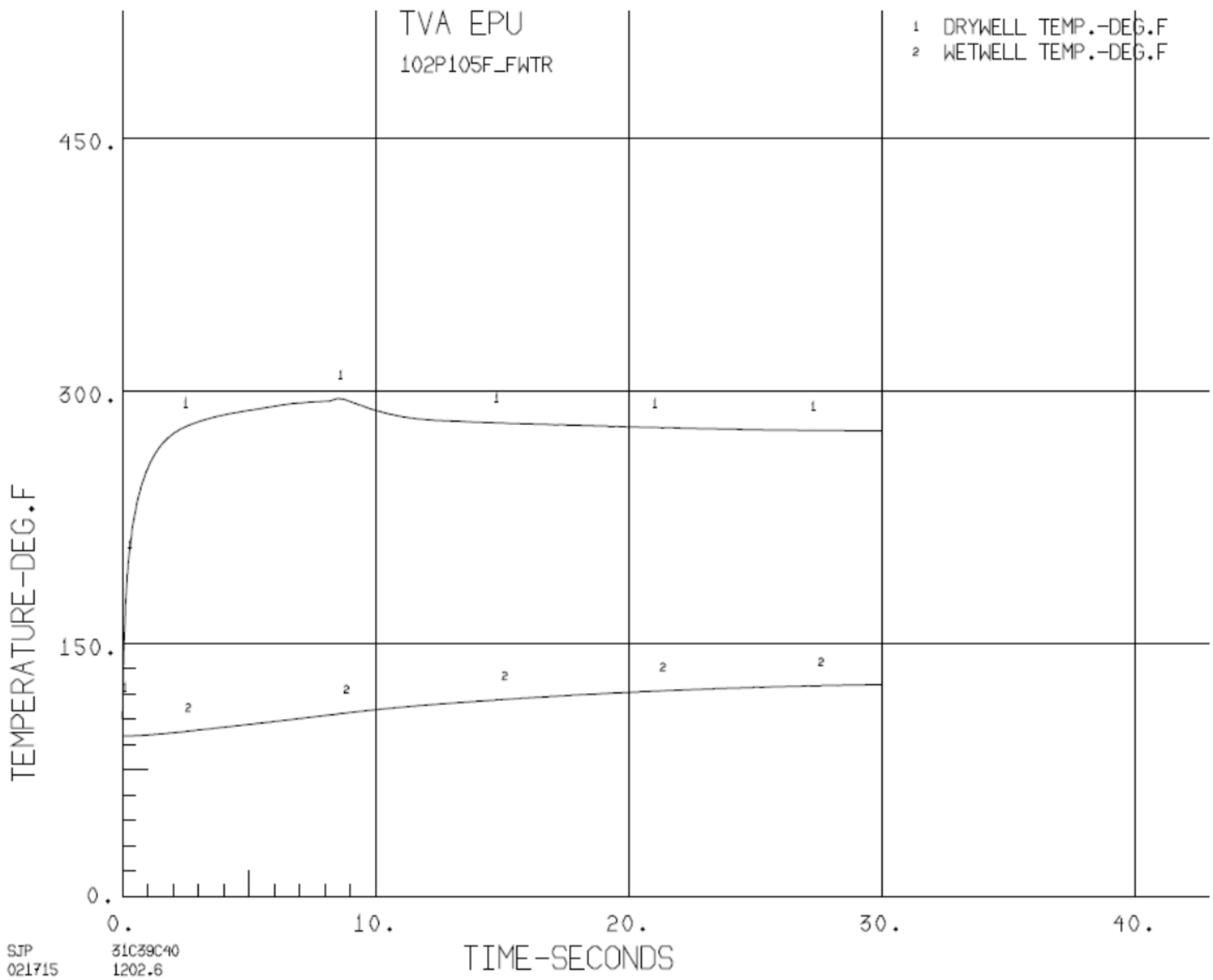


Figure 2.6-4 EPU Short-Term RSLB DBA-LOCA Containment Temperature Response (Reference Condition: Initial DW Temperature =150ºF)

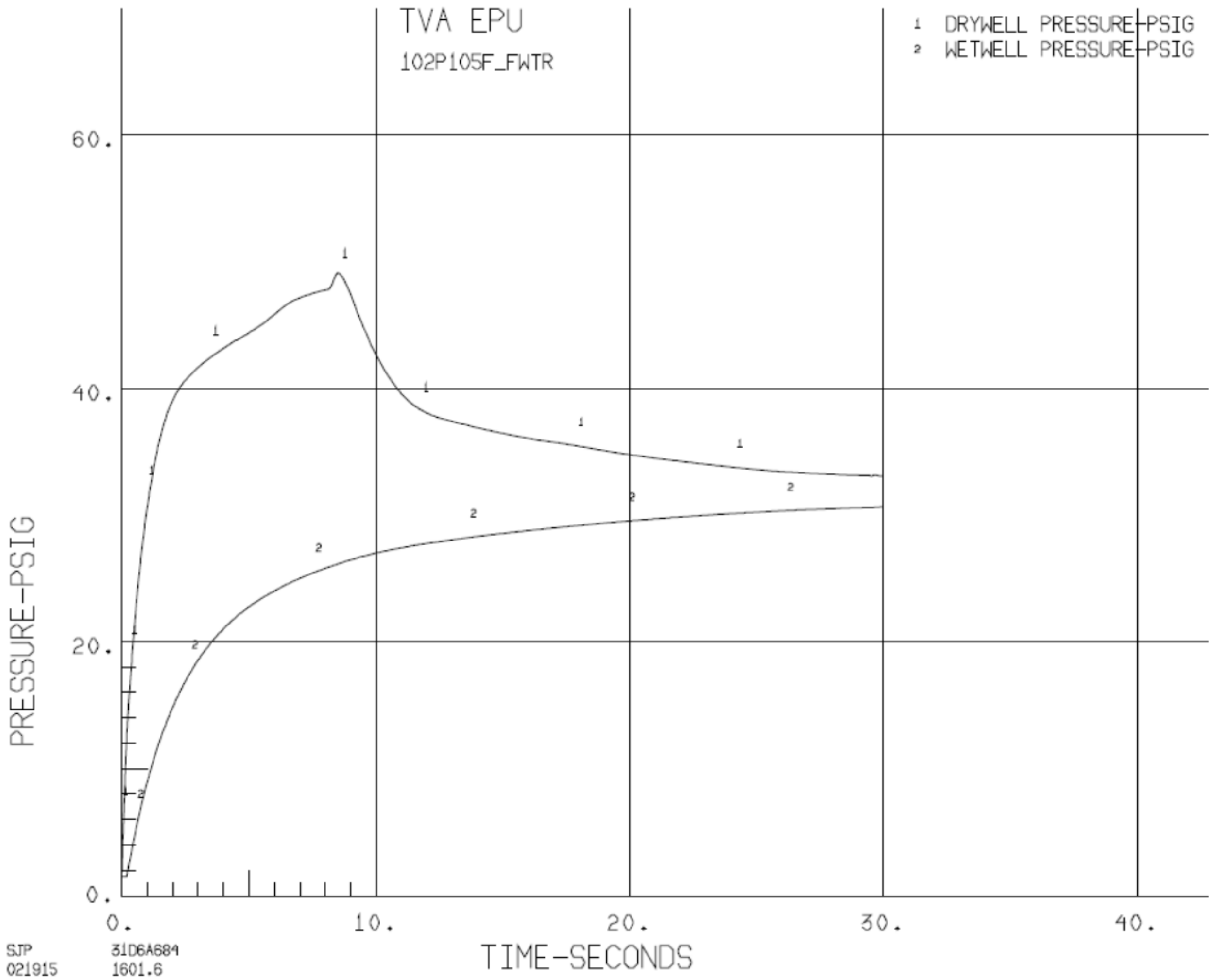


Figure 2.6-5 EPU Short-Term RSLB **DBA-LOCA Containment Pressure Response (Bounding Condition: Initial DW Temperature =130ºF)**

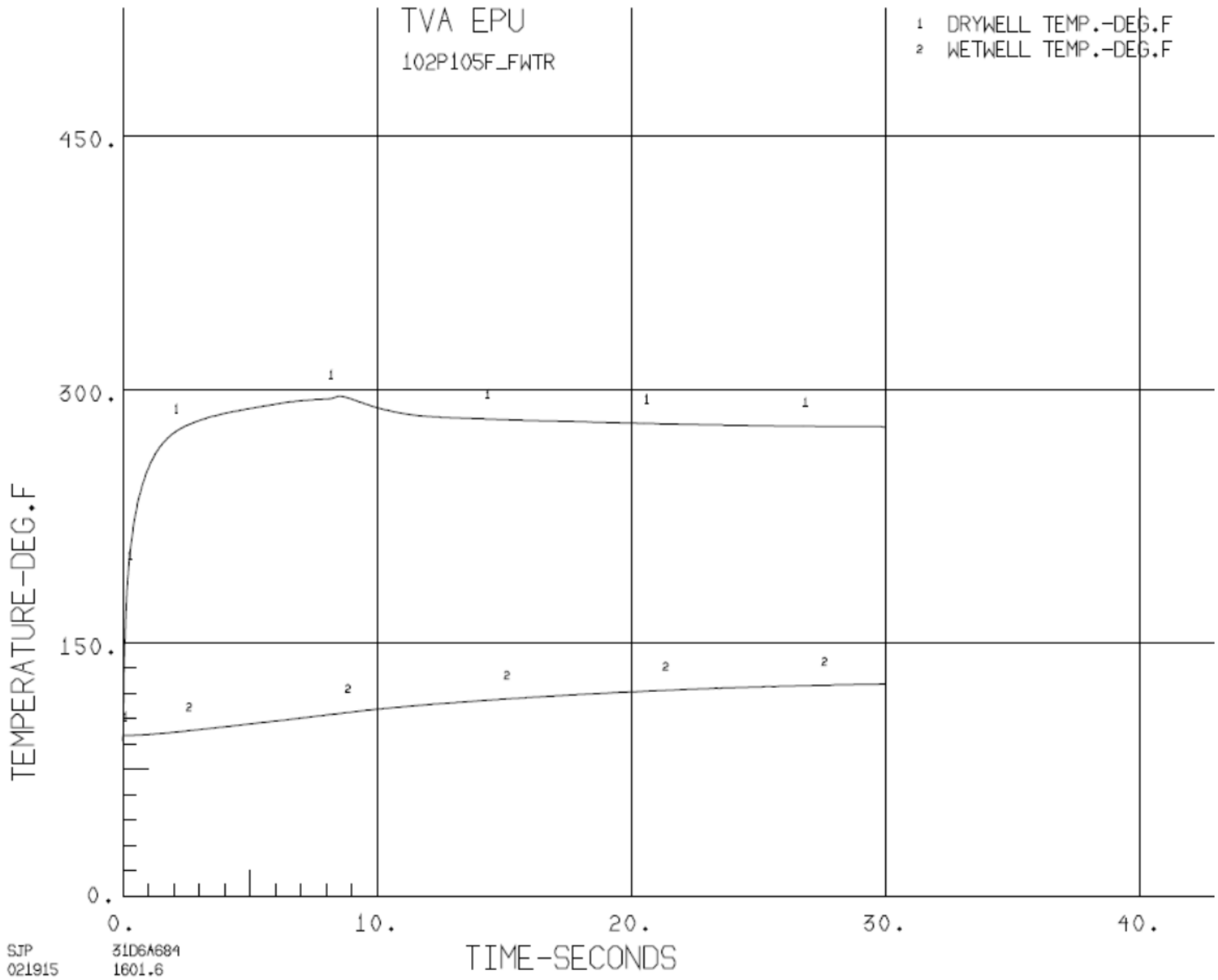


Figure 2.6-6 EPU Short-Term RSLB DBA-LOCA Containment Temperature Response (Bounding Condition: Initial DW Temperature =130ºF)

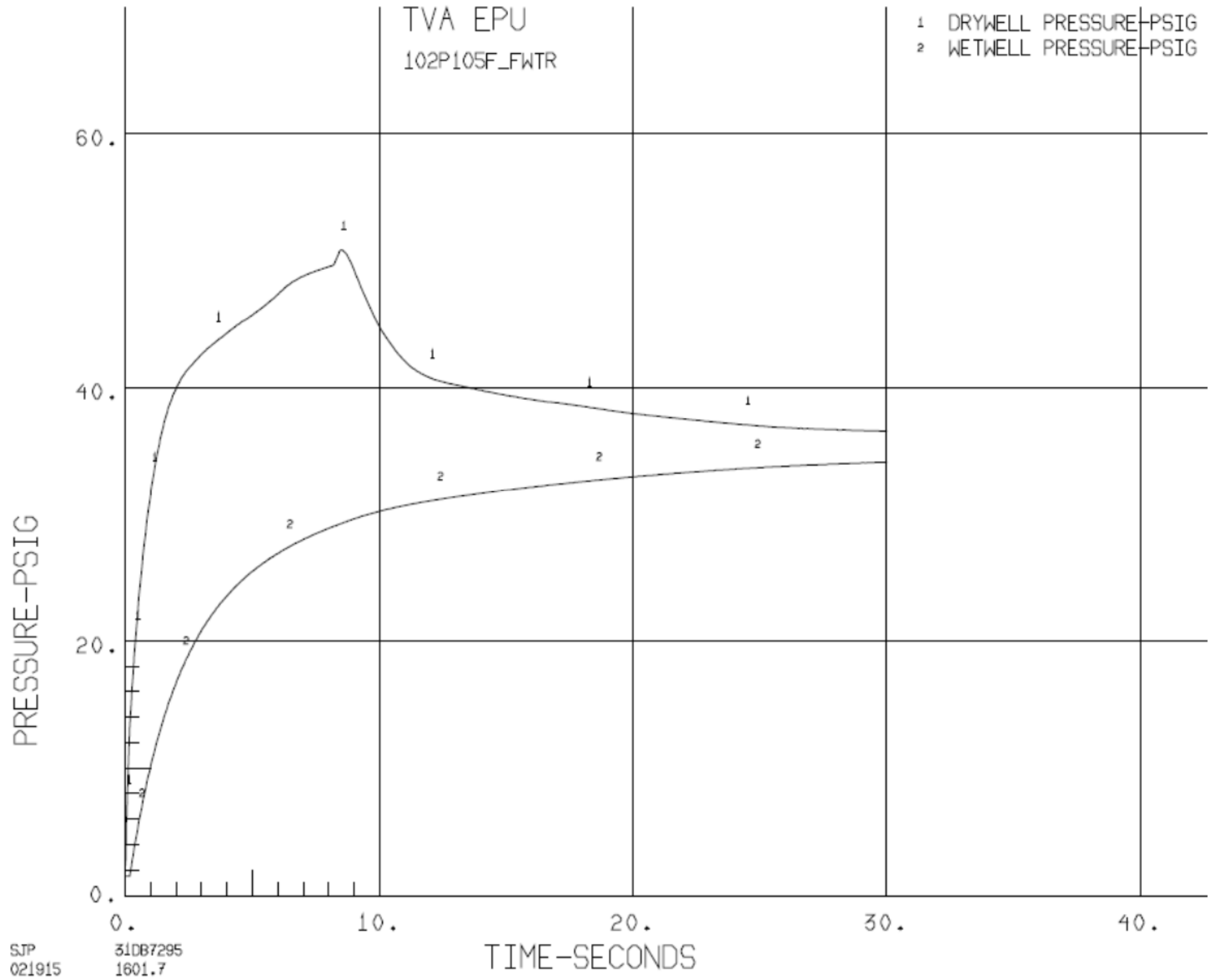


Figure 2.6-7 EPU Short-Term RSLB DBA-LOCA Containment Pressure Response (Design Condition: Initial DW Temperature =70ºF)

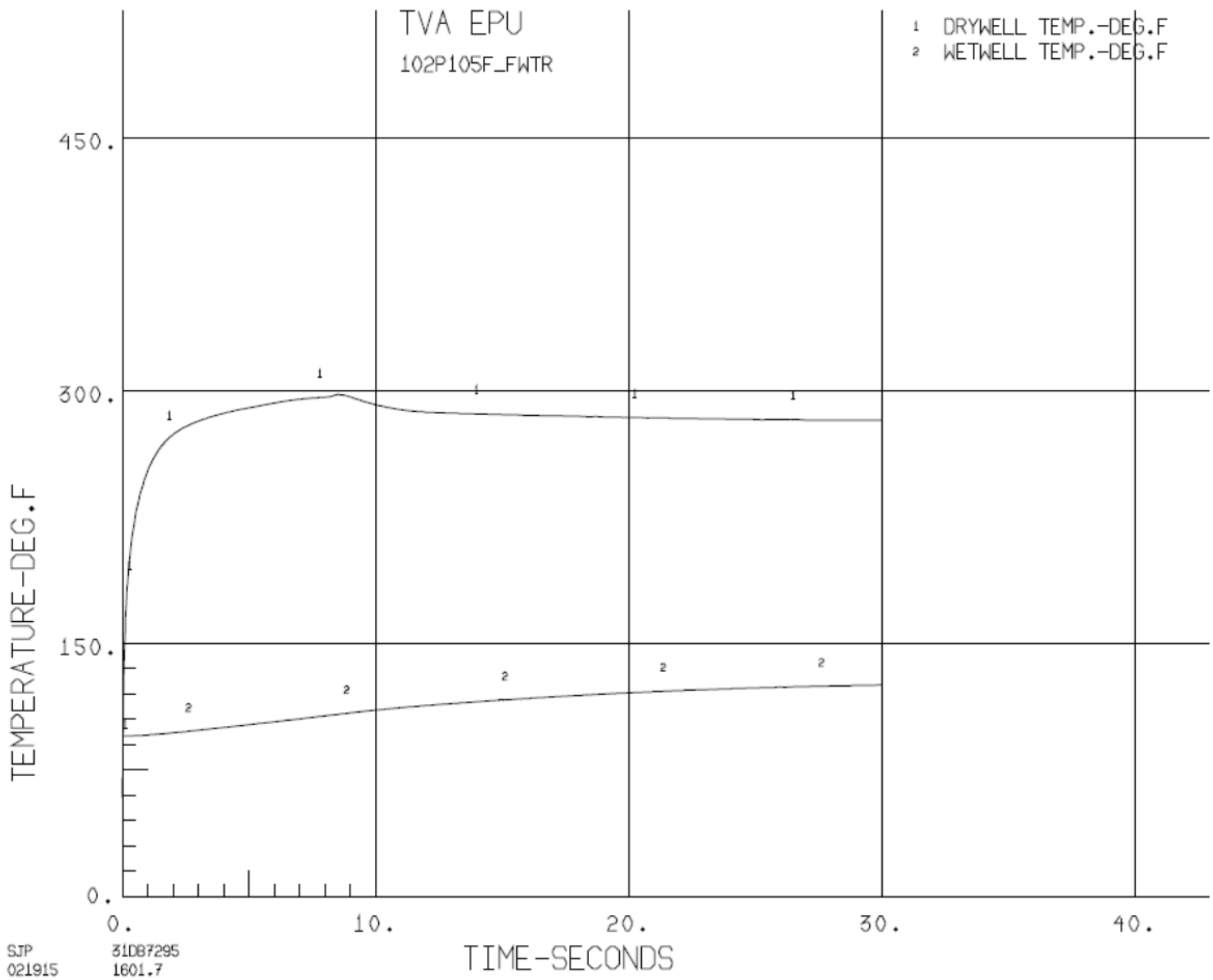


Figure 2.6-8 EPU Short-Term RSLB DBA-LOCA Containment Temperature Response (Design Condition: Initial DW Temperature =70ºF)

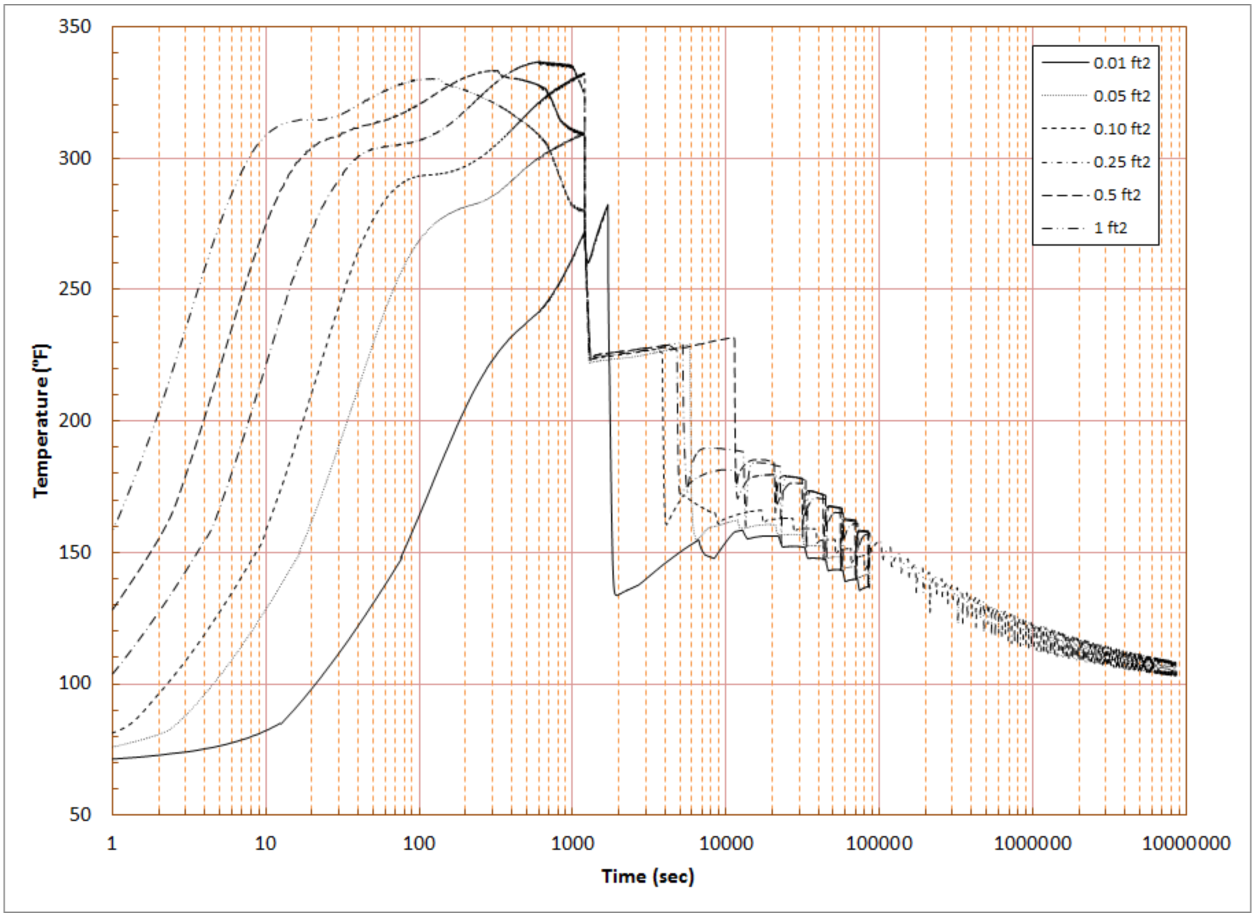


Figure 2.6-9 EPU Long-Term Small Steam Line Break LOCA Drywell Temperature Response

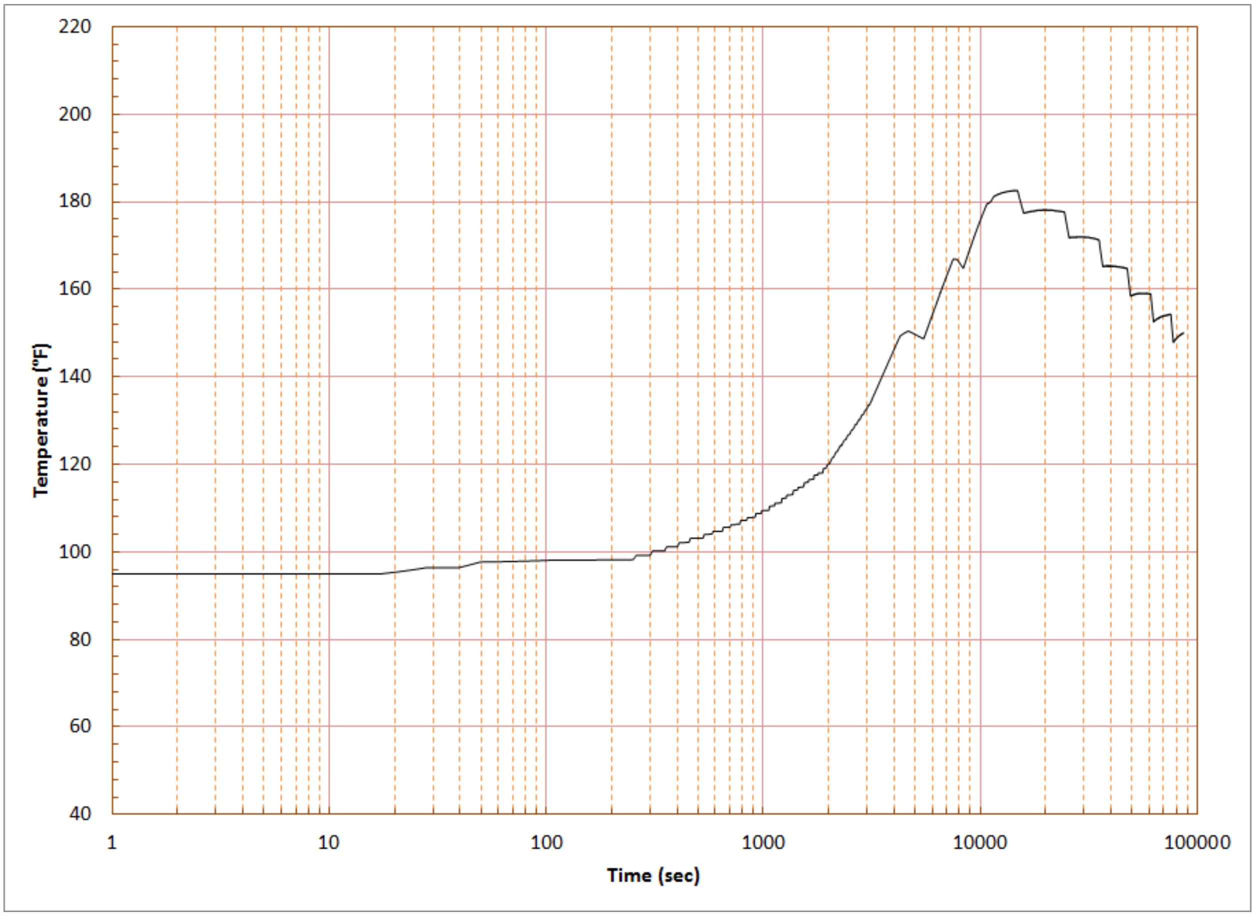


Figure 2.6-10 EPU Long-Term Small Steam Break LOCA Suppression Pool Temperature Response – 0.01 ft^2 Break with HPCI Available

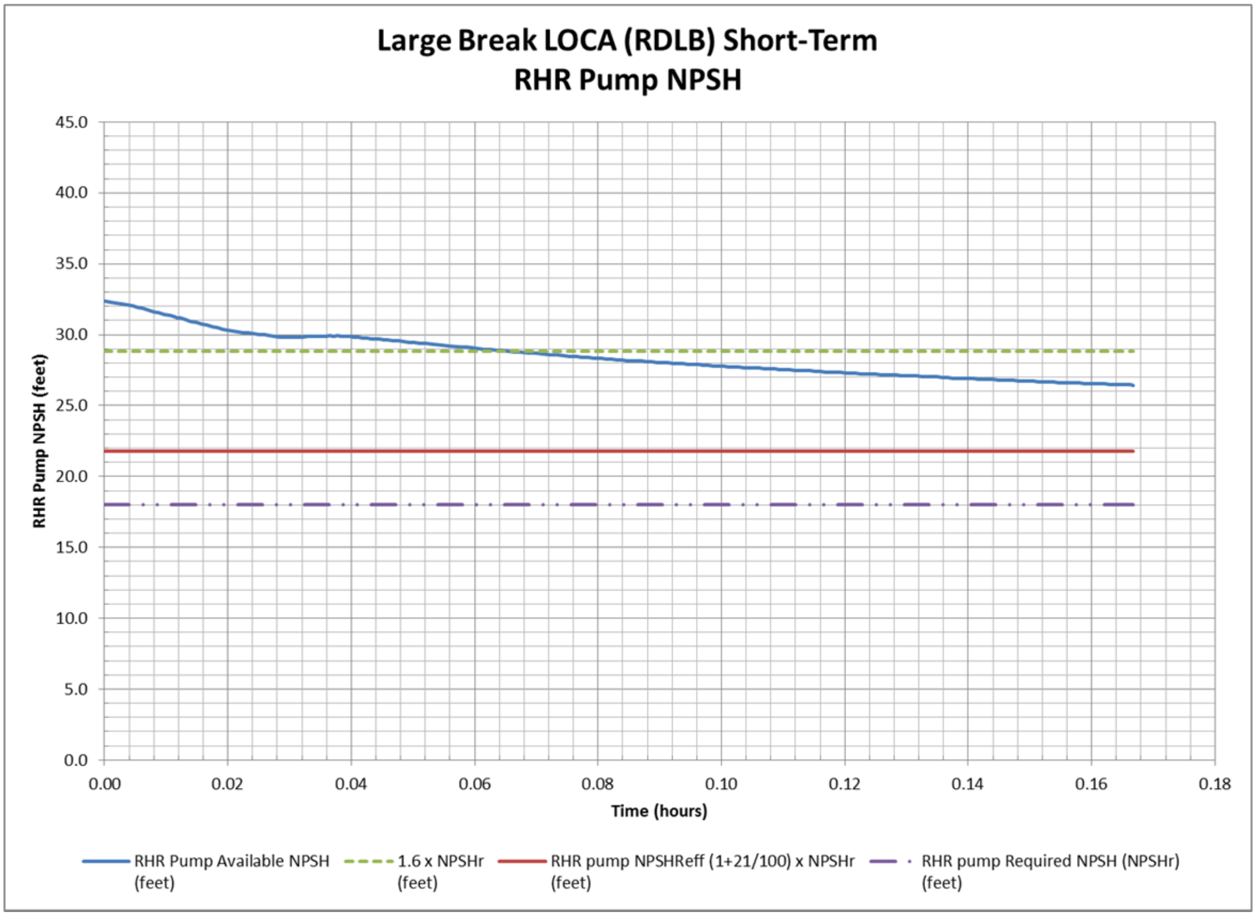


Figure 2.6-11a Large Break-LOCA Short Term RHR NPSH versus Time

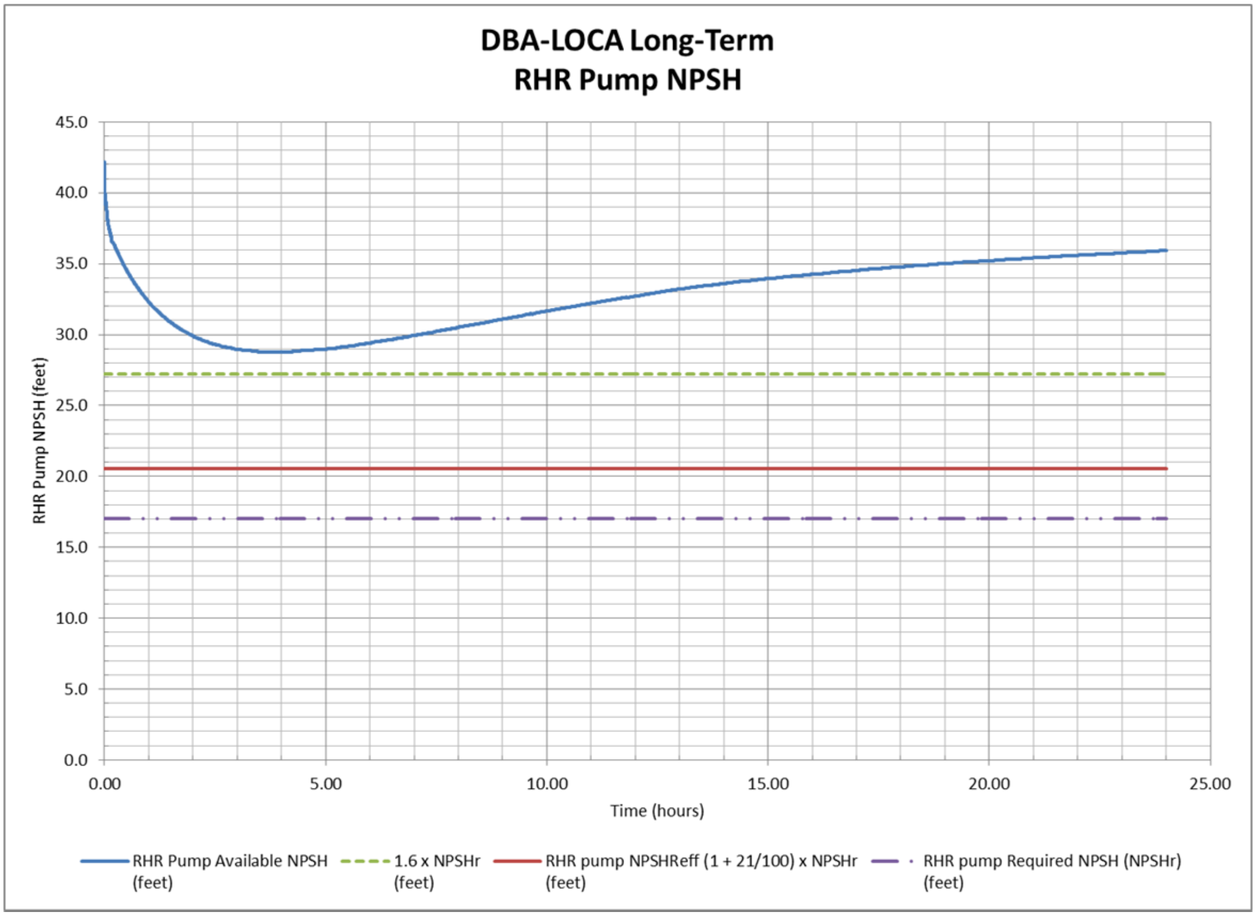


Figure 2.6-11b DBA-LOCA Long Term RHR NPSH versus Time

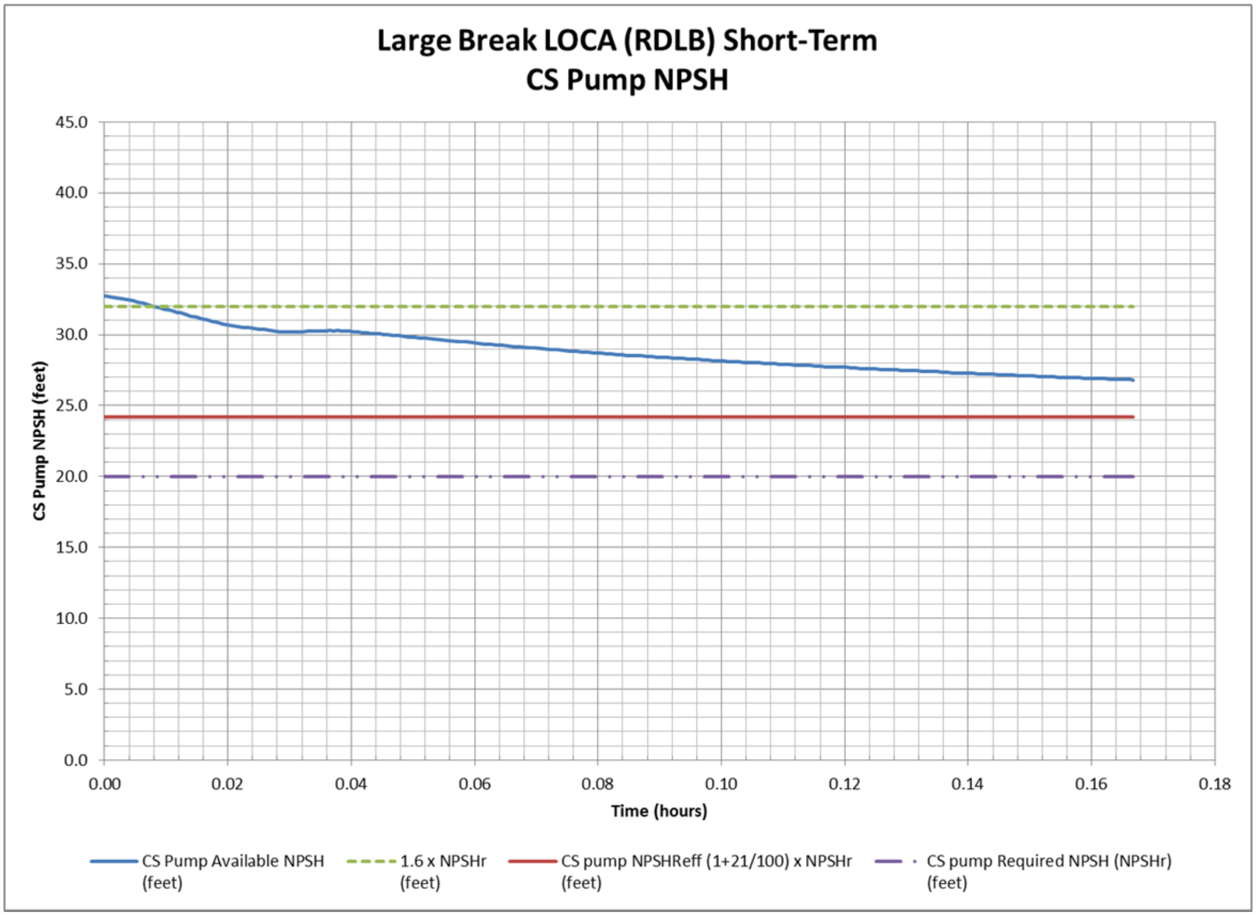


Figure 2.6-12a DBA-LOCA Short Term CS NPSH versus Time

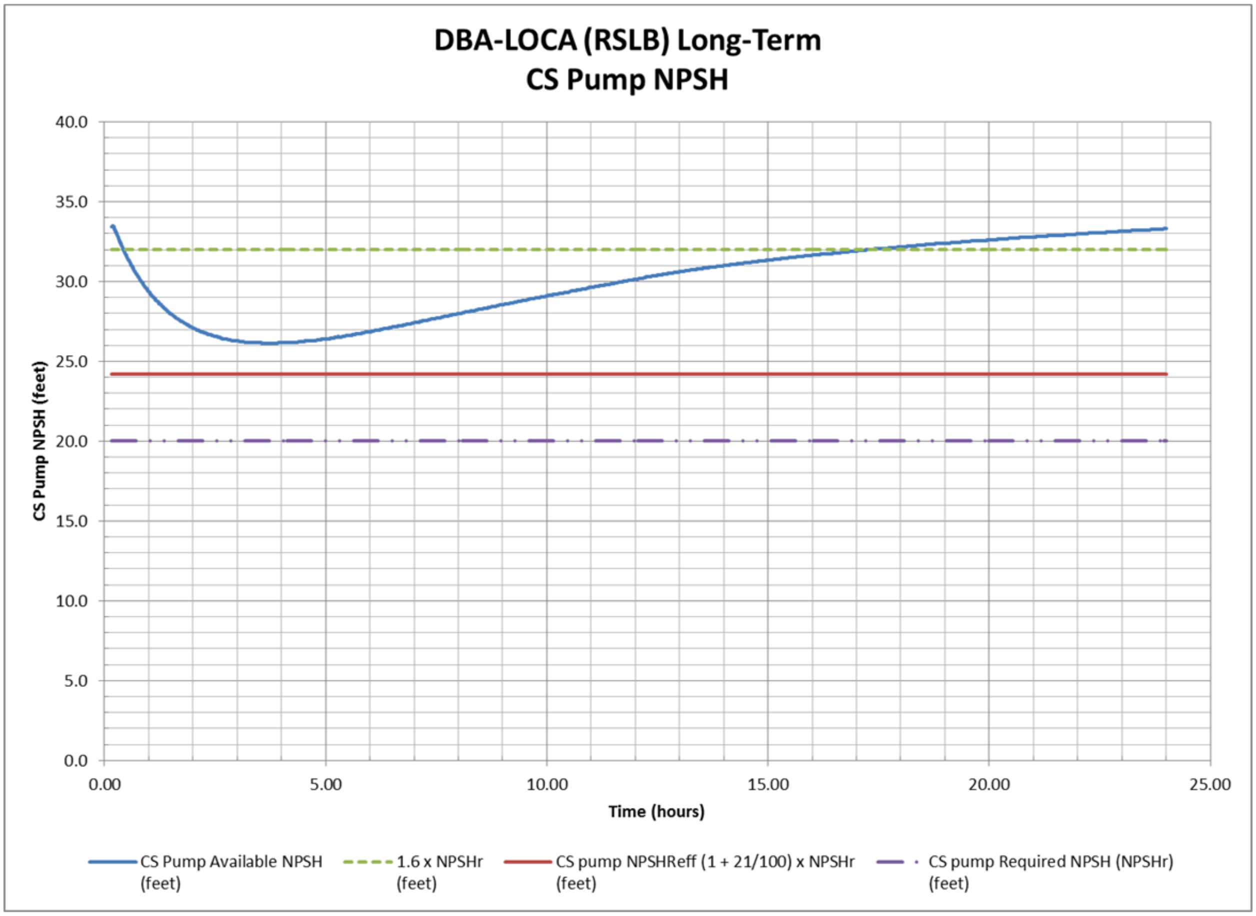


Figure 2.6-12b DBA-LOCA Long Term CS NPSH versus Time

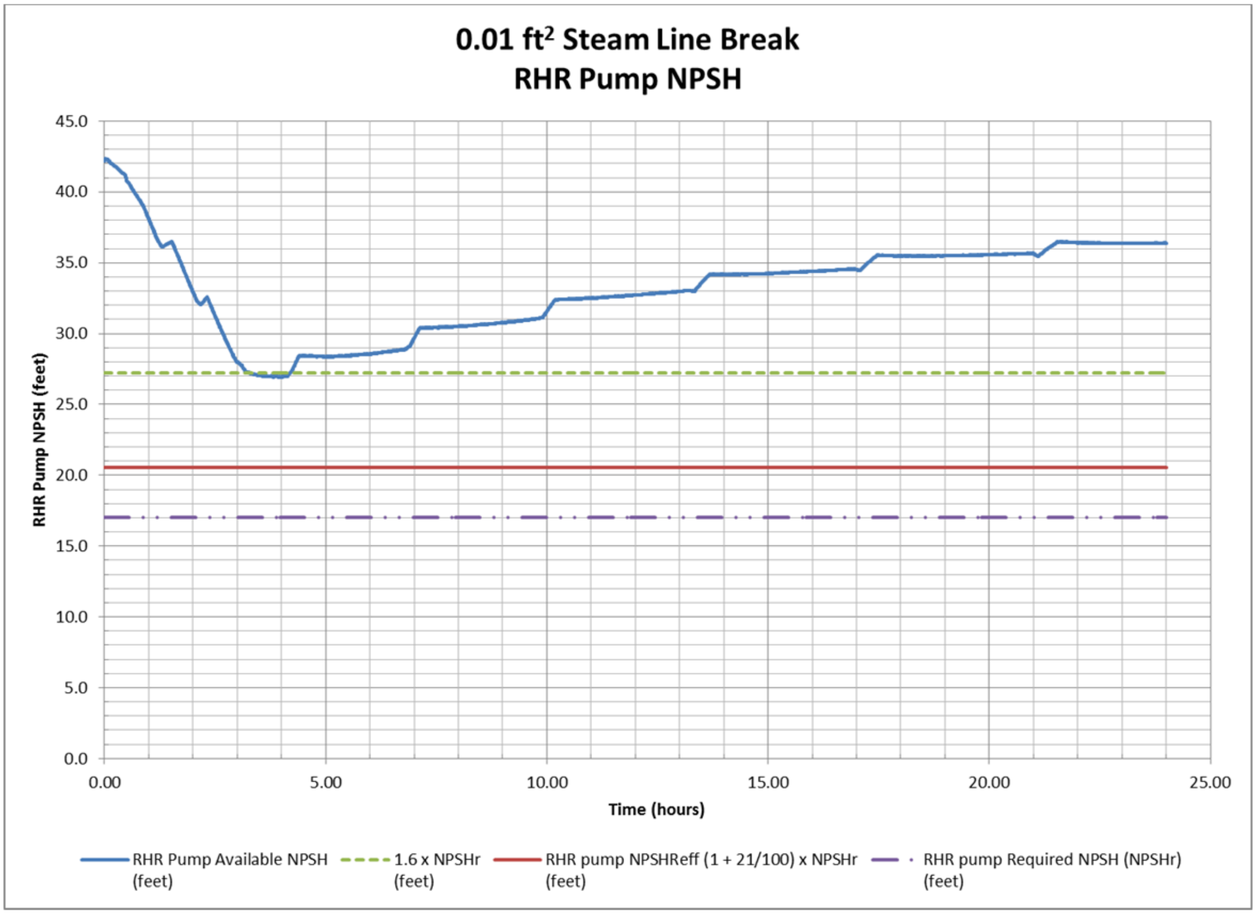


Figure 2.6-13a Small Break LOCA RHR NPSH versus Time

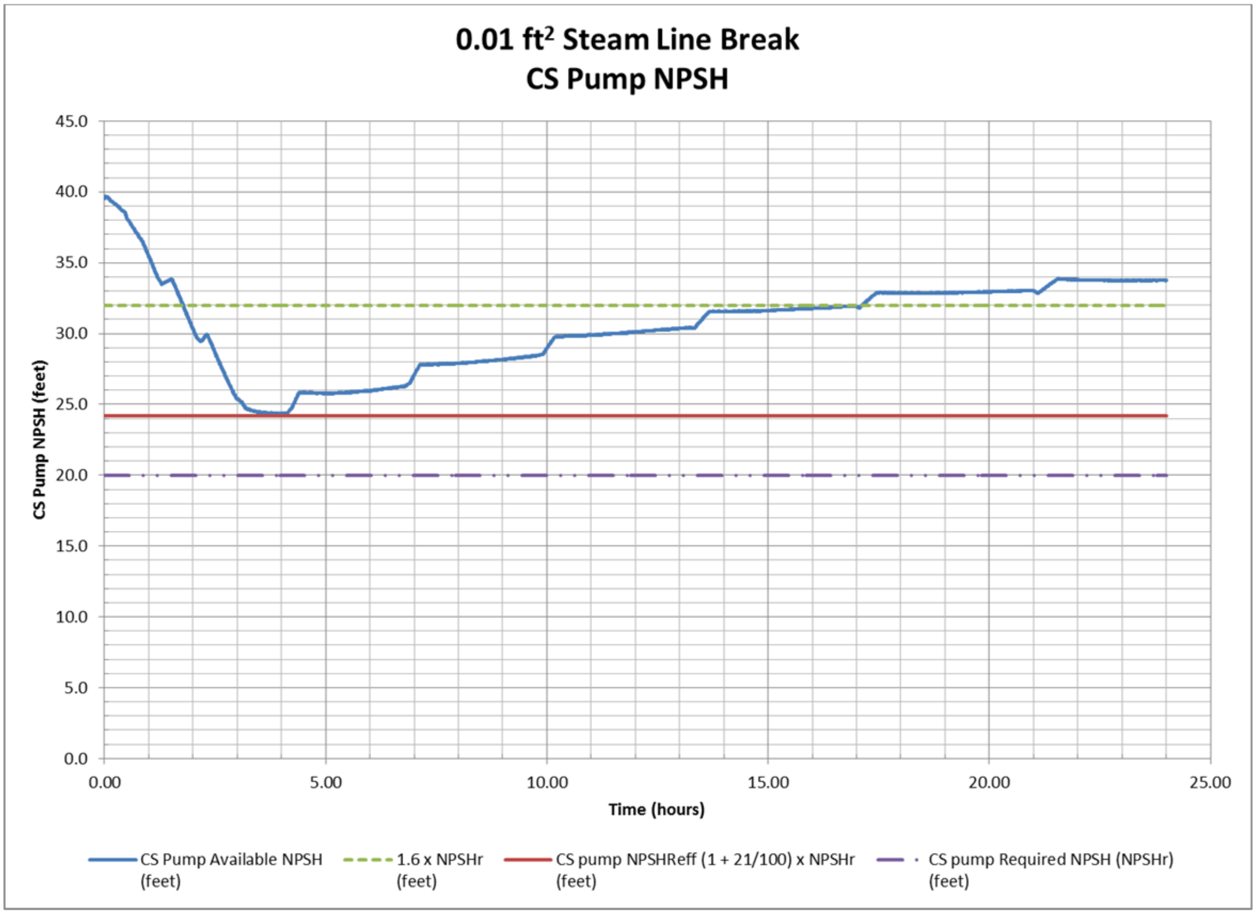


Figure 2.6-13b Small Break LOCA CS NPSH versus Time

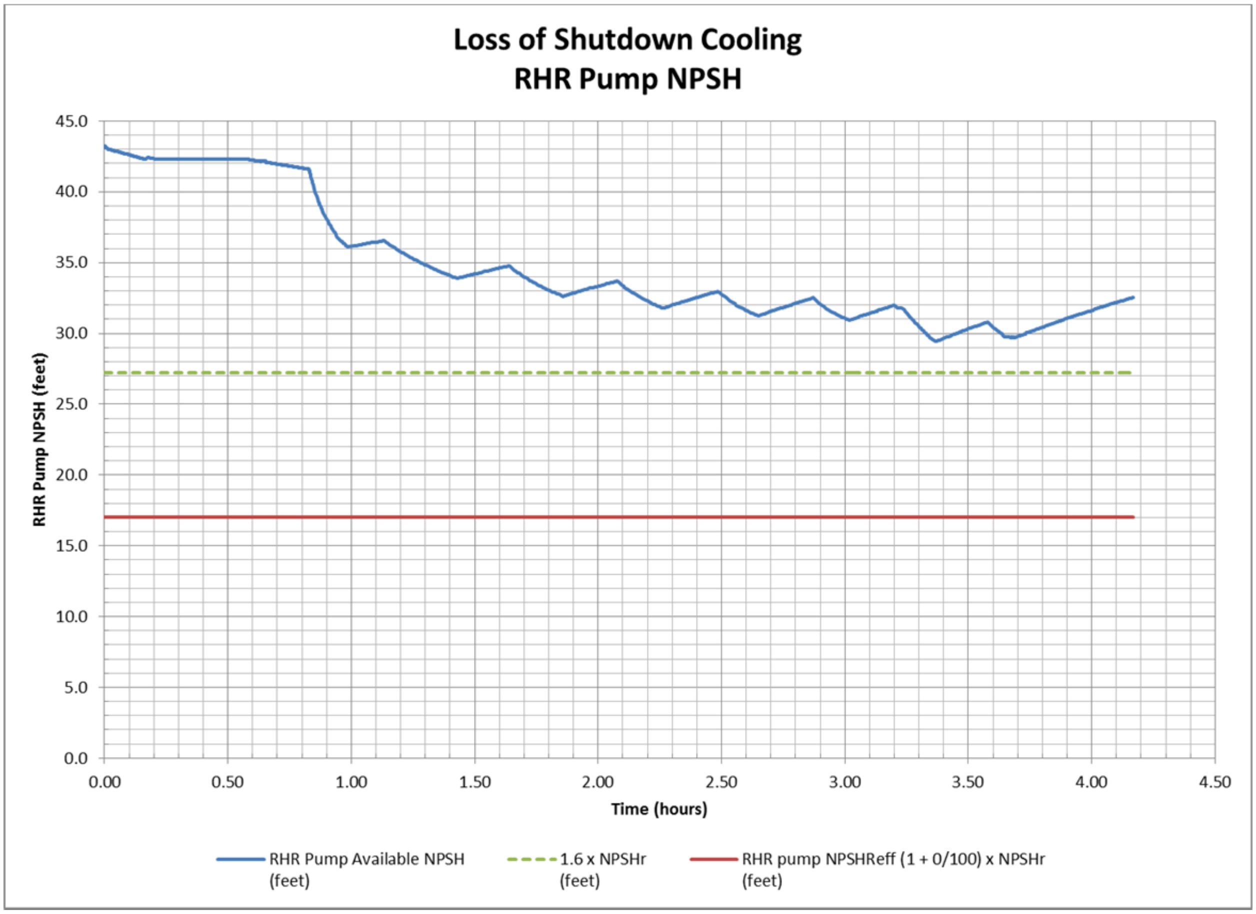


Figure 2.6-14a Loss of RHR SDC - RHR NPSH versus Time

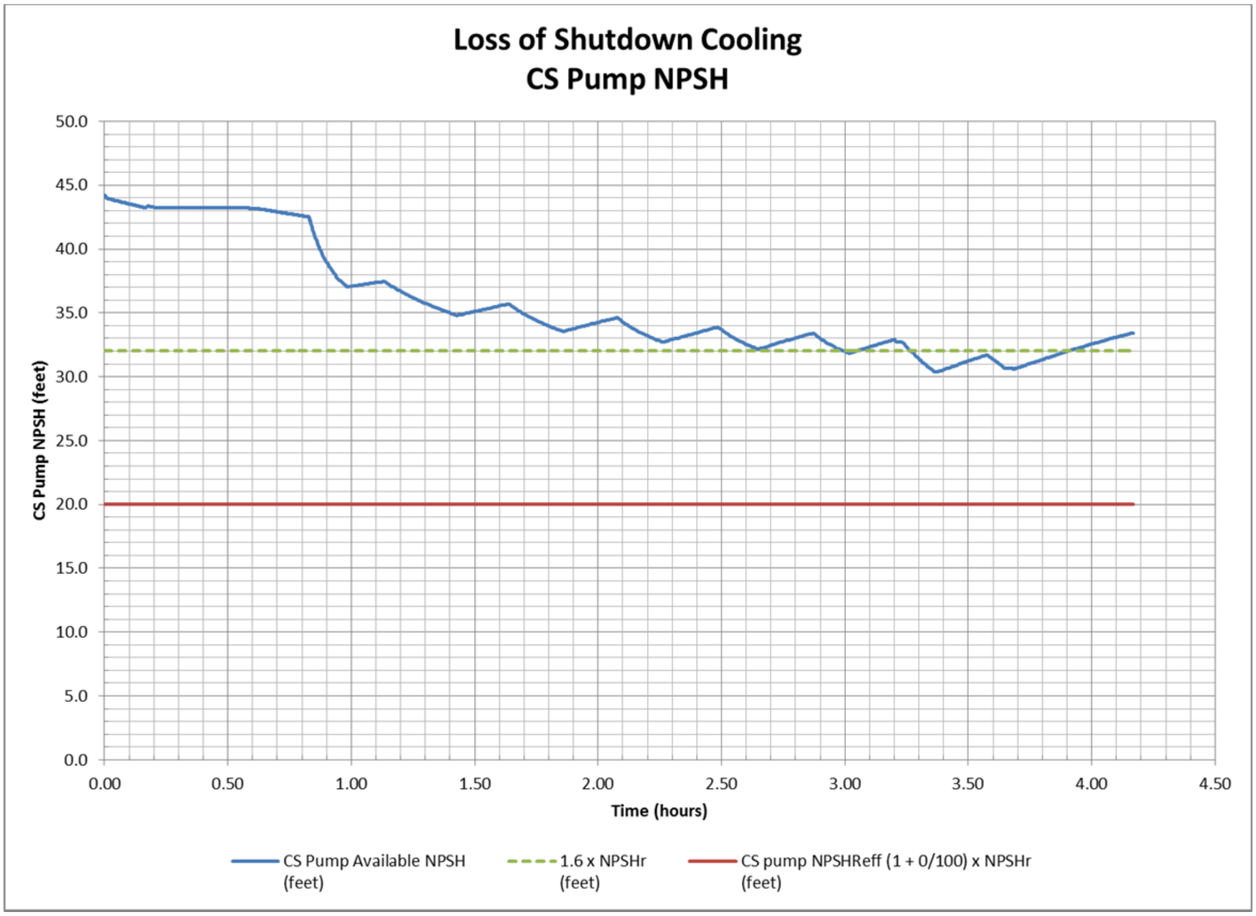


Figure 2.6-14b Loss of RHR SDC - CS NPSH versus Time

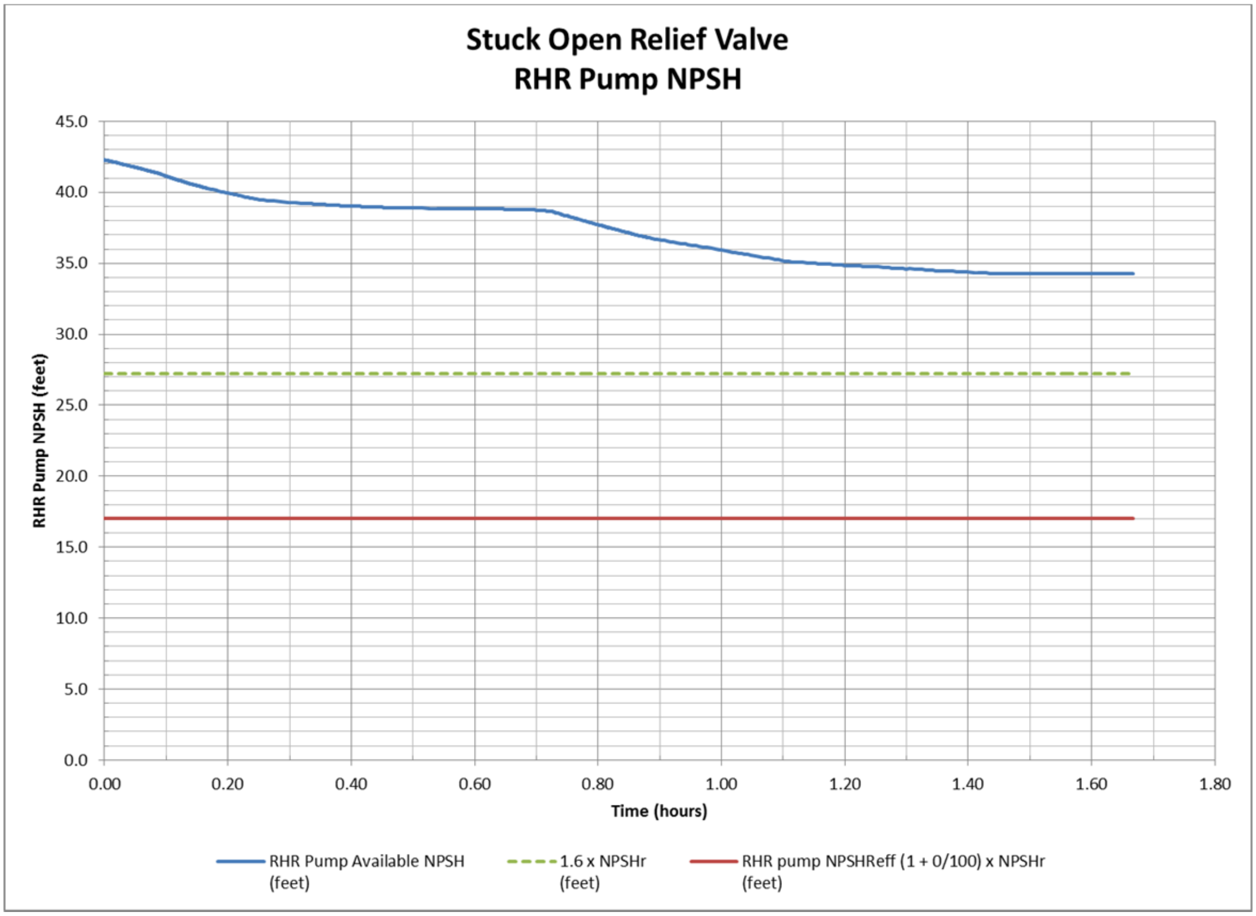


Figure 2.6-15a SORV with RPV Isolation Event - RHR NPSH versus Time

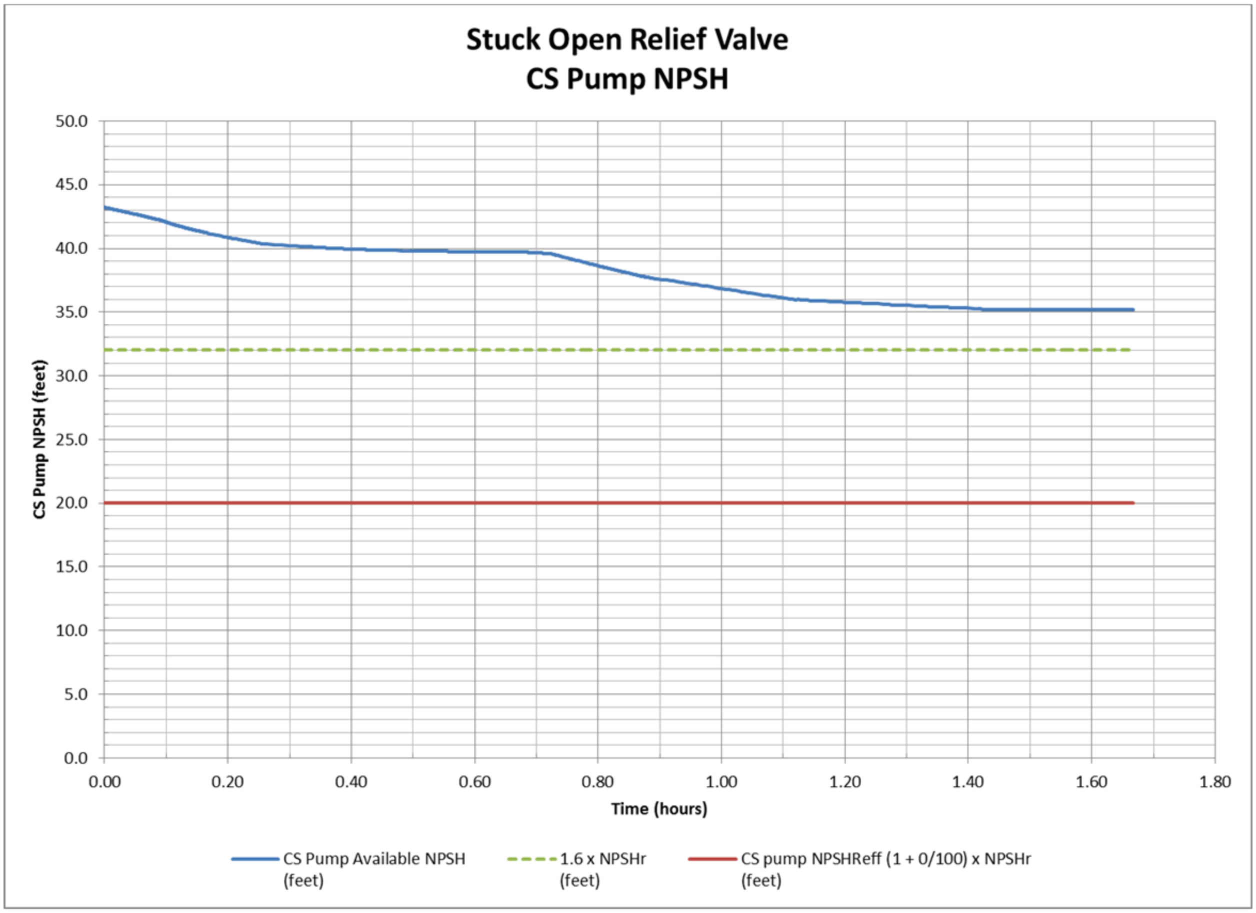


Figure 2.6-15b SORV with RPV Isolation Event - CS NPSH versus Time

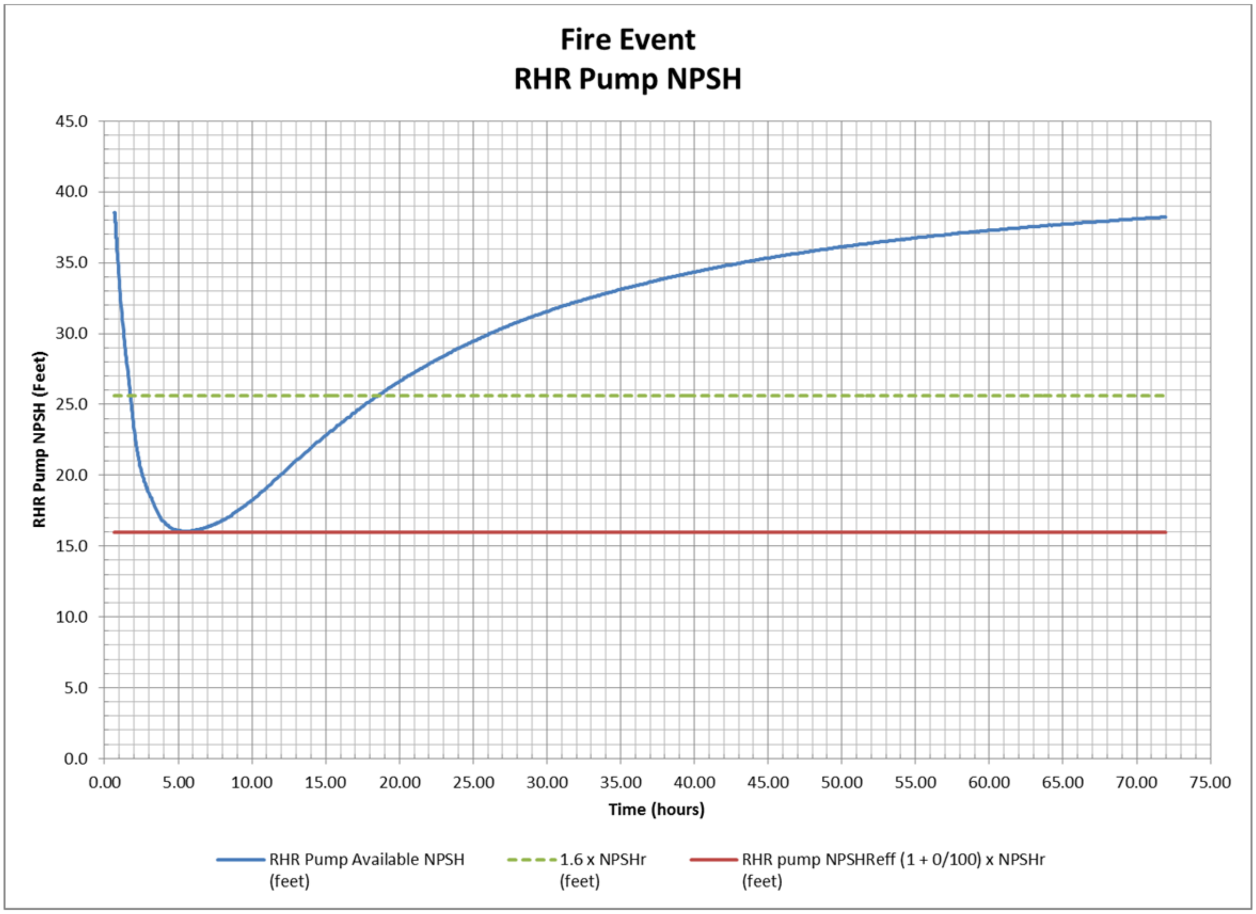
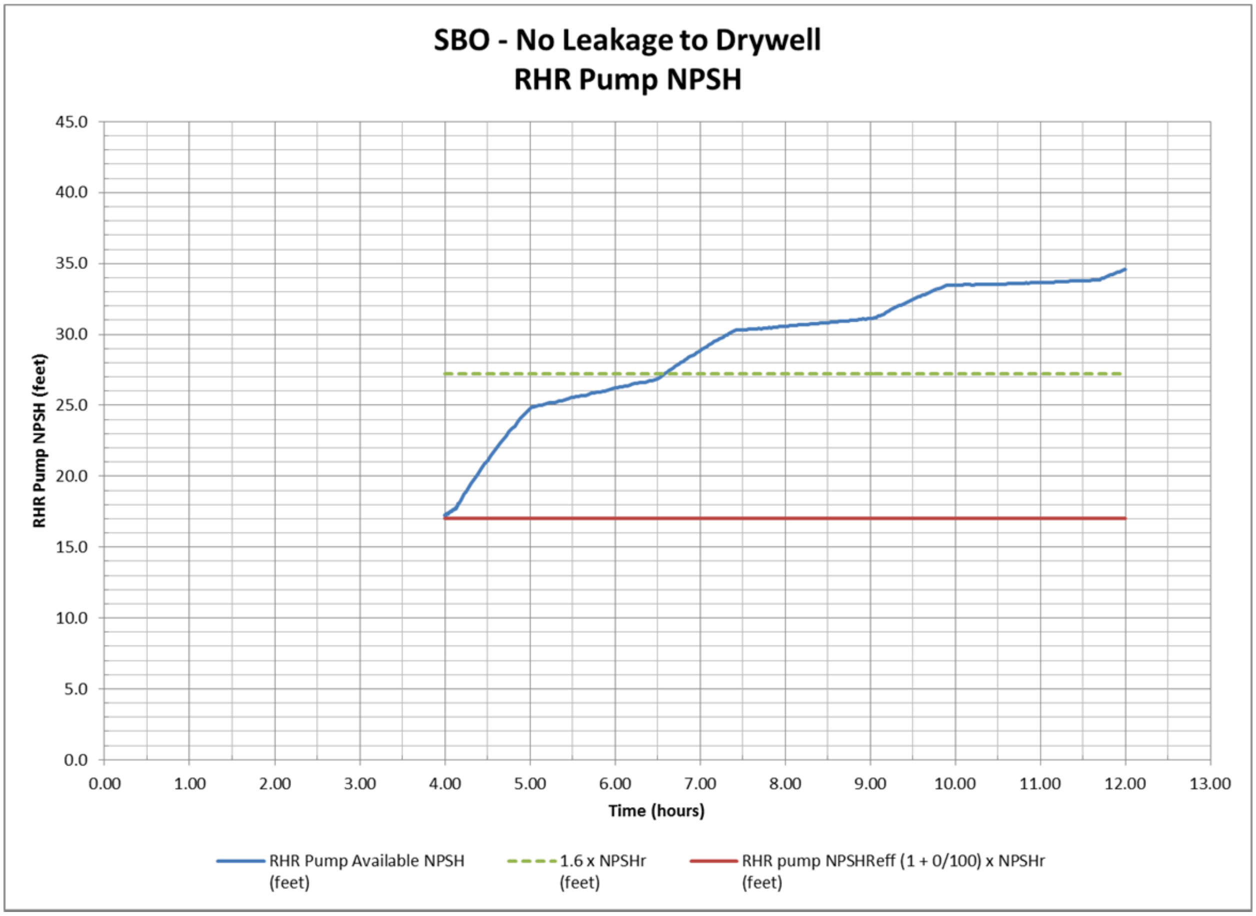


Figure 2.6-16 Fire Event - RHR NPSH versus Time



Note: During the SBO coping period (the first four hours of the event), there is no RHR pump in service. Therefore, RHR pump NPSH is not calculated for the SBO coping period.

Figure 2.6-17 SBO Event - RHR NPSH versus Time

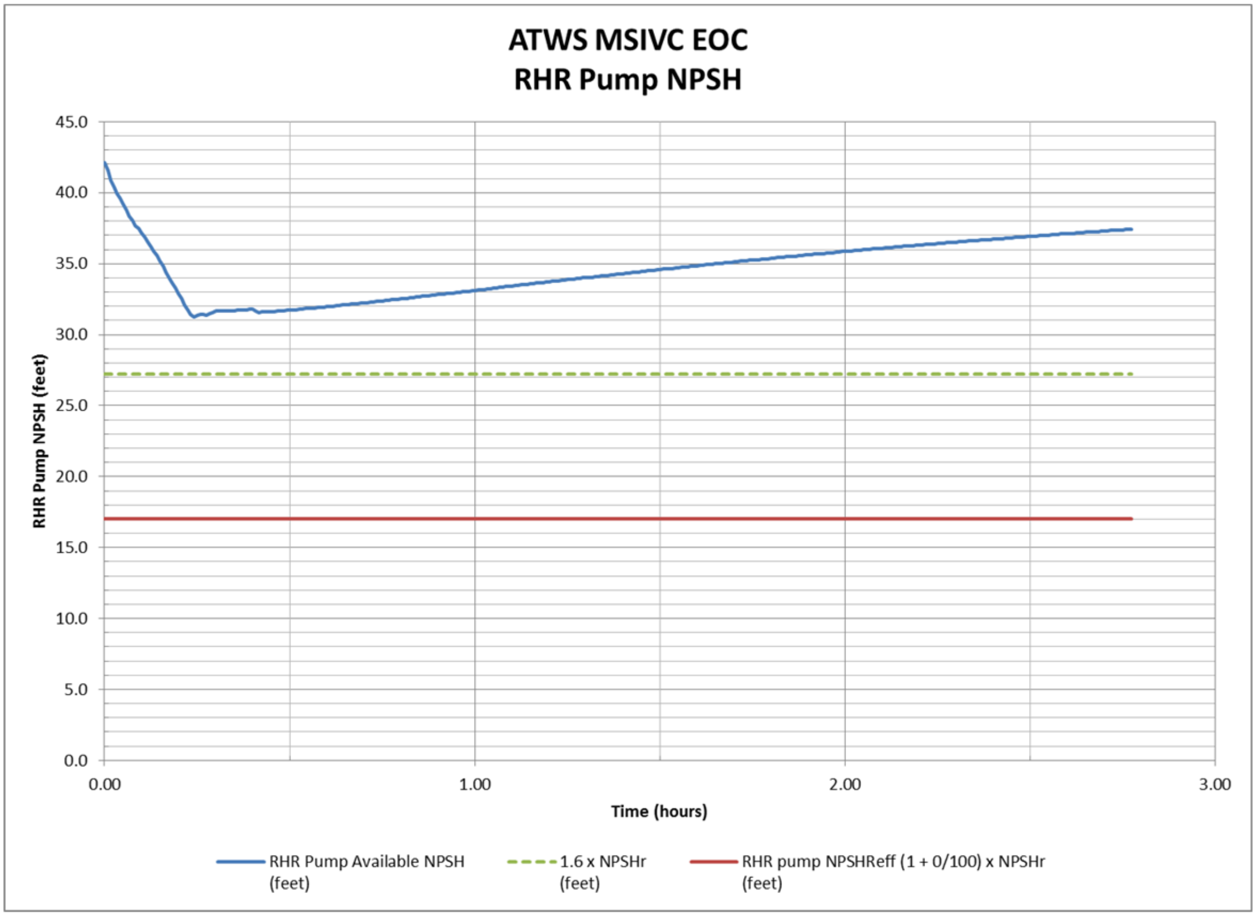
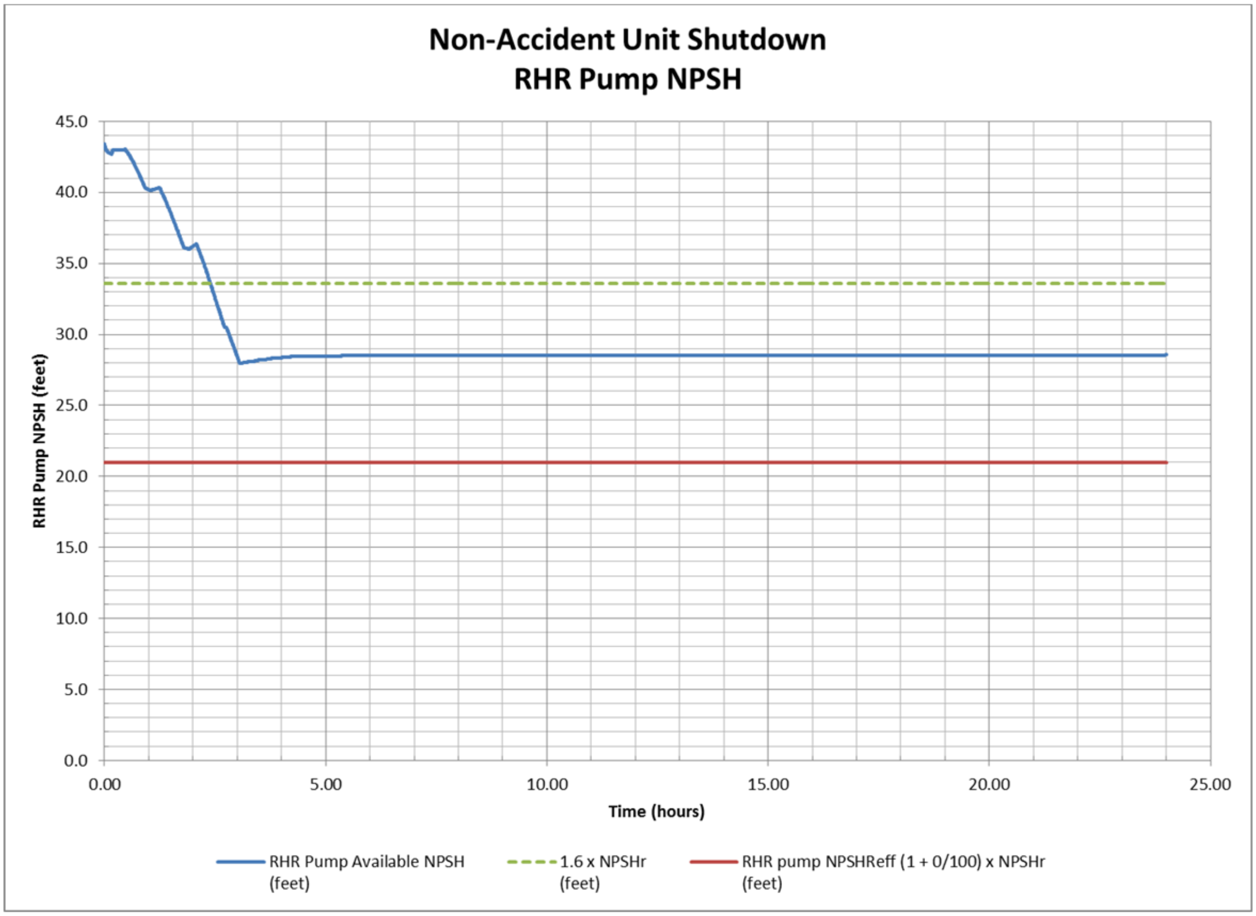
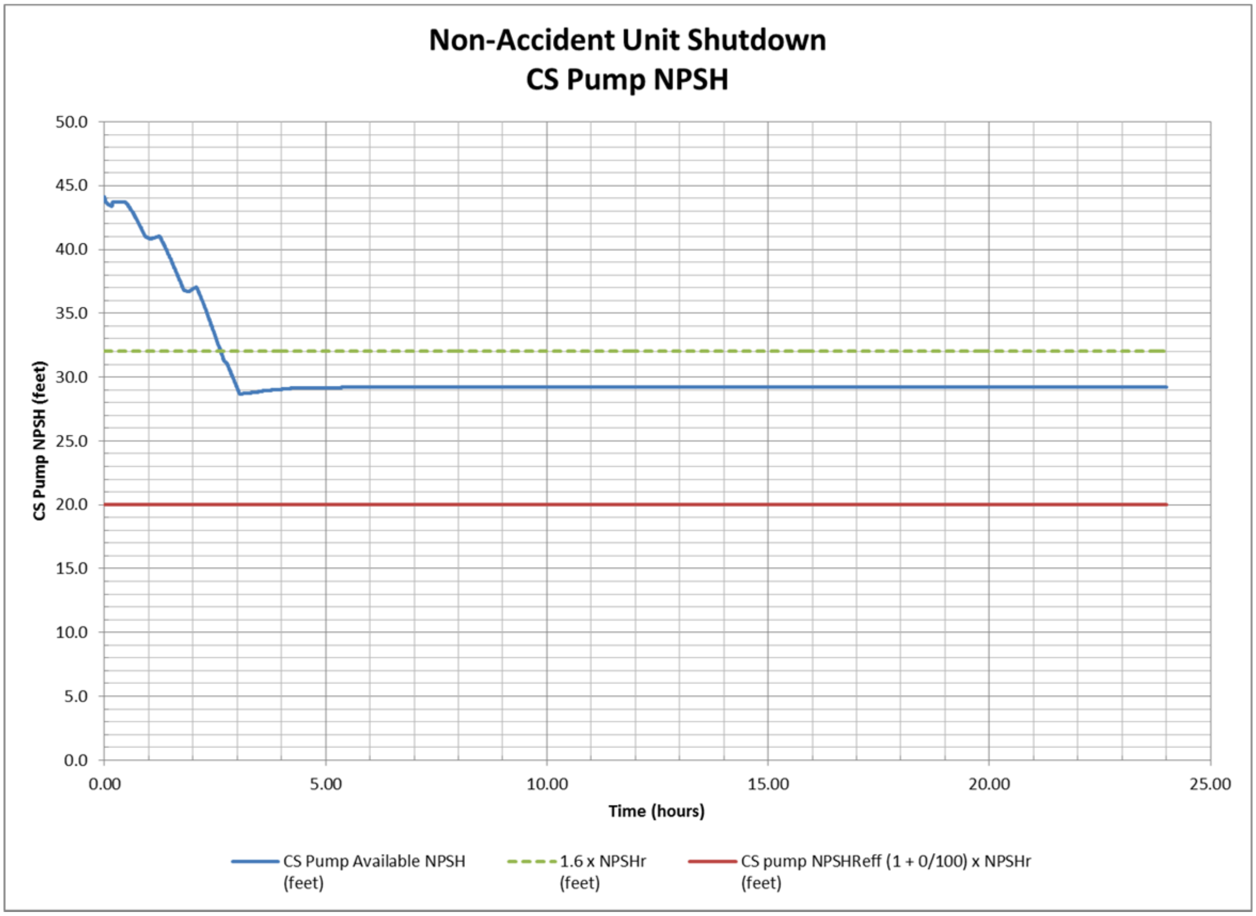


Figure 2.6-18 ATWS - RHR NPSH versus Time



Note: RHR pump suction from the suppression pool is secured at 2.72 hours when transition to shutdown cooling is started. RHR SDC is initiated at 3.06 hours.

Figure 2.6-19a Shutdown of the Non-Accident Unit - RHR NPSH versus Time



Note: CS pump is stopped at 3.06 hours when RHR SDC is initiated

Figure 2.6-19b Shutdown of the Non-Accident Unit – CS NPSH versus Time

2.7 Habitability, Filtration, and Ventilation

2.7.1 Control Room Habitability System

Regulatory Evaluation

The NRC's acceptance criteria for the control room habitability system are based on (1) final GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with postulated accidents, including the effects of the release of toxic gases; and (2) GDC-19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent, to any part of the body, for the duration of the accident.

Specific NRC review criteria are contained in SRP Section 6.4 and other guidance provided in Matrix 7 of RS-001.

Browns Ferry Current Licensing Basis

The General Design Criteria (GDC) listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Browns Ferry Nuclear Plant (Browns Ferry) principal design criteria predate these criteria. The Browns Ferry principal design criteria are listed in UFSAR Section 1.5, "Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, the Tennessee Valley Authority (TVA) performed a comparative evaluation of the design basis of Browns Ferry with the AEC proposed General Design Criteria of 1967. The Browns Ferry UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria set out in the July 1967 AEC release. For each group of criteria, there is a statement of TVA's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a table of references to locations in the Browns Ferry UFSAR where there is subject matter relating to the intent of that particular criteria.

While Browns Ferry is not generally licensed to the final GDC or the 1967 AEC proposed General Design Criteria, a comparison of the final GDC to the applicable AEC proposed General Design Criteria can usually be made. For the final GDC listed in the Regulatory Evaluation above, the Browns Ferry comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Browns Ferry UFSAR Appendix A: draft GDC-40. Final GDC-19 is applicable to Browns Ferry as described in "Browns Ferry Nuclear Plant, Units 1, 2, and 3 - Issuance of Amendments Regarding Full-Scope Implementation of Alternative Source Term," dated September 27, 2004 (Reference 72).

The Control Room Habitability System is described in Browns Ferry UFSAR Section 10.12.5.3, "Control Building."

Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Browns Ferry License Renewal Safety Evaluation Report (SER), NUREG-1843, dated April 2006 (Reference 11). The license renewal evaluation associated with the Control Room Habitability System is documented in NUREG-1843, Section 2.3.3.9. Management of aging effects on the Control Room Habitability System is documented in NUREG-1843, Section 3.3.2.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of EPUs. Section 4.4 of the CLTR addresses the effect of EPU on Main Control Room Atmosphere Control System.

The main Control Room Emergency Ventilation System (CREVS) functions during a DBA or an AOO to provide filtered air for personnel ventilation and pressurization of the control room envelope. Redundant radiation detectors are provided at the outside air intakes to automatically initiate emergency flow through filtration units. With no change to detection and controls, the operation of the control room HVAC system is not affected.

Browns Ferry meets all CLTR dispositions. The topic addressed in this evaluation is:

Topic	CLTR Disposition	Browns Ferry Result
Iodine Intake	Plant Specific	Meets CLTR Disposition

The CLTR states that EPU increases the radioisotopes seen by the control room atmosphere control system following an accident.

The radiological effect of EPU on the CREVS is an increase in the particulates, including particulate iodine, released during an accident. Browns Ferry has implemented the AST methodology, which affects the DBA iodine model. The AST analyses were performed at 102% of the EPU power level (i.e., 4,031 MWt), and thus incorporate the increased EPU iodine release as well as the effects of the AST iodine release model. These analyses included the radiological consequences of the DBAs currently documented in Section 14.6 of the Browns Ferry Units 1, 2, and 3 UFSAR that potentially result in the most significant control room exposures. In all cases, the control room doses at EPU conditions are within the regulatory limits, as shown in Tables 2.9-6 to 2.9-11.

The quantities and locations of gases and hazardous chemicals that could affect control room habitability are unaffected by EPU. Therefore, EPU has no effect on the potential toxic gas concentrations in the main control room.

Conclusion

TVA has evaluated the effects of the proposed EPU on the ability of the control room habitability system to protect plant operators against the effects of accidental releases of toxic and radioactive gases and any increase of toxic and radioactive gases that would result from the proposed EPU. The evaluation indicates that the control room habitability system will continue to provide the required protection following implementation of the proposed EPU. Based on this, the control room habitability system will continue to meet the requirements of draft GDC-40 and final GDC-19 and 10 CFR 50.67. Therefore, the proposed EPU is acceptable with respect to the control room habitability system.

2.7.2 Engineered Safety Feature Atmosphere Cleanup

Regulatory Evaluation

ESF atmosphere cleanup systems are designed for fission product removal in post-accident environments. These systems generally include primary systems (e.g., in-containment recirculation) and secondary systems (e.g., standby gas treatment systems and emergency or post-accident air cleaning systems) for the Fuel-Handling Building, control room, Shield Building, and areas containing ESF components.

The NRC's acceptance criteria for ESF atmosphere cleanup systems are based on (1) GDC-19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent, to any part of the body, for the duration of the accident; (2) GDC-41, insofar as it requires that systems to control fission products released into the reactor containment be provided to reduce the concentration and quality of fission products released to the environment following postulated accidents; (3) GDC-61, insofar as it requires that systems that may contain radioactivity be designed to assure adequate safety under normal and postulated accident conditions; and (4) GDC-64, insofar as it requires that means be provided for monitoring effluent discharge paths and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences (AOOs), and postulated accidents.

Specific NRC review criteria are contained in SRP Section 6.5.1.

Browns Ferry Current Licensing Basis

The General Design Criteria (GDC) listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Browns Ferry Nuclear Plant (Browns Ferry) principal design criteria predate these criteria. The Browns Ferry principal design criteria are listed in UFSAR Section 1.5, "Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (Federal Register 32FR10213, July 11, 1967).

Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, the Tennessee Valley Authority (TVA) performed a comparative evaluation of the design basis of Browns Ferry with the AEC proposed General Design Criteria of 1967. The Browns Ferry UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria set out in the July 1967 AEC release. For each group of criteria, there is a statement of TVA's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a table of references to locations in the Browns Ferry UFSAR where there is subject matter relating to the intent of that particular criteria.

While Browns Ferry is not generally licensed to the final GDC or the 1967 AEC proposed General Design Criteria, a comparison of the final GDC to the applicable AEC proposed General Design Criteria can usually be made. For the final GDC listed in the Regulatory Evaluation above, the Browns Ferry comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Browns Ferry UFSAR Appendix A: draft GDCs-17 and 70. There is no draft GDC directly associated with final GDC-41. Final GDC-19 is applicable to Browns Ferry as described in "Browns Ferry Nuclear Plant, Units 1, 2, and 3 - Issuance of Amendments Regarding Full-Scope Implementation of Alternative Source Term," dated September 27, 2004 (Reference 72).

The ESF atmosphere cleanup system at Browns Ferry is the Standby Gas Treatment System. The Standby Gas Treatment System is described in Browns Ferry UFSAR Section 5.3.3.7, "Standby Gas Treatment System."

Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Browns Ferry License Renewal Safety Evaluation Report (SER), NUREG-1843, dated April 2006 (Reference 11). The license renewal evaluation associated with the Standby Gas Treatment System is documented in NUREG-1843, Section 2.3.2.2. The management of the effects of aging on the Standby Gas Treatment System is documented in NUREG-1843, Section 3.2.2.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of EPUs. Section 4.5 of the CLTR addresses the effect of EPU on the SGTS. The results of this evaluation are described below.

One of the two ESF atmosphere cleanup systems at Browns Ferry is the Control Room Ventilation System. The acceptability of this system under EPU conditions is addressed in Section 2.7.1. The second ESF atmosphere cleanup system is the SGTS.

The SGTS is designed to maintain secondary containment at a negative pressure and to provide an elevated release path for the exhaust air for removal of fission products potentially present during abnormal conditions. By providing for an elevated release path for airborne particulates

and halogens, the SGTS limits off-site dose following a postulated DBA. The effect of a EPU on the performance of the SGTS was evaluated in the CLTR based on two bounding analyses. CLTR dispositions regarding the flow capacity and iodine removal capability of the SGTS have been addressed in Sections 2.6.6 and 2.5.2.1, respectively. No credit is taken for charcoal adsorption for any DBA. Credit is taken for HEPA filter removal of 90% of the particulate activity in the DBA-LOCA analysis (Reference 72).

Details regarding the SGTS evaluation based on post-LOCA operation after EPU implementation are described in Section 2.5.2. The SGTS at Browns Ferry meets all CLTR dispositions.

Conclusion

TVA has evaluated the effects of the proposed EPU on the ESF atmosphere cleanup systems and accounted for any increase of fission products and changes in expected environmental conditions that would result from the proposed EPU. The evaluation indicates that the ESF atmosphere cleanup systems will continue to provide adequate fission product removal in post-accident environments following implementation of the proposed EPU. The ESF atmosphere cleanup systems will continue to meet the requirements of draft GDCs-17 and 70, final GDC-19, and 10 CFR 50.67. Therefore, the proposed EPU is acceptable with respect to the ESF atmosphere cleanup systems.

2.7.3 Control Room Area Ventilation System

Regulatory Evaluation

The function of the Control Room Area Ventilation System (CRAVS) is to provide a controlled environment for the comfort and safety of control room personnel and to support the operability of control room components during normal operation, AOOs, and DBA conditions.

The NRC's acceptance criteria for the CRAVS are based on (1) GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents; (2) GDC-19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident; and (3) GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents.

Specific NRC review criteria are contained in SRP Section 9.4.1.

Browns Ferry Current Licensing Basis

The General Design Criteria (GDC) listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Browns Ferry Nuclear Plant (Browns Ferry) principal design criteria predate these criteria. The Browns Ferry principal design criteria are listed in UFSAR Section 1.5, "Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (Federal Register 32FR10213, July 11, 1967).

Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, the Tennessee Valley Authority (TVA) performed a comparative evaluation of the design basis of Browns Ferry with the AEC proposed General Design Criteria of 1967. The Browns Ferry UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria set out in the July 1967 AEC release. For each group of criteria, there is a statement of TVA's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a table of references to locations in the Browns Ferry UFSAR where there is subject matter relating to the intent of that particular criteria.

While Browns Ferry is not generally licensed to the final GDC or the 1967 AEC proposed General Design Criteria, a comparison of the final GDC to the applicable AEC proposed General Design Criteria can usually be made. For the final GDC listed in the Regulatory Evaluation above, the Browns Ferry comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Browns Ferry UFSAR Appendix A: draft GDCs-40 and 70. Final GDC-19 is applicable to Browns Ferry as described in "Browns Ferry Nuclear Plant, Units 1, 2, and 3 - Issuance of Amendments Regarding Full-Scope Implementation of Alternative Source Term," dated September 27, 2004 (Reference 72).

The control room area ventilation system is described in Browns Ferry UFSAR Section 10.12.5.3, "Control Building."

Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Browns Ferry License Renewal Safety Evaluation Report (SER), NUREG-1843, dated April 2006 (Reference 11). The license renewal evaluation associated with the control room area ventilation system is documented in NUREG-1843, Section 2.3.3.9. Management of aging effects on the control room area ventilation system is documented in NUREG-1843, Section 3.3.2.

Technical Evaluation

The heating, ventilating and air conditioning (HVAC) systems discussed in the CLTR are only those that have power dependent heat loads. Power dependent HVAC systems require a plant-specific evaluation for EPU. The Control Room HVAC System maintains temperature and humidity conditions suitable for personnel comfort and for equipment reliable operation inside the control room envelope. The Control Room HVAC System also maintains the control room envelope at positive pressure to inhibit air infiltration (see Section 2.7.1).

There is no increase in toxic or asphyxiant gas release that may result from EPU. The control of the concentration of airborne radioactive material in the control room envelope during AOOs and after postulated accidents is accomplished by the Control Room HVAC system described in Section 2.7.1.

Heat loads for the control room area envelope include boundary transmission, lighting, and equipment such as control room panels. These heat loads are not affected by the slightly higher

process temperatures that may result from EPU, thus they are not power dependent. EPU does not add any electrical or electronic equipment to the control room. EPU may add some amperage for control and indication signals, but the resulting changes in temperature are considered negligible. The conductance of heat through the building structure to the control room is expected to increase only slightly. The heat load increase is expected to be insignificant in comparison with the total control room heat load. Therefore, the control room temperature increase is expected to be insignificant as a result of EPU implementation. Table 2.7-1 shows the effect of EPU on the Ventilation Systems.

There is no change to the Control Room HVAC System configuration or system parameters as a result of EPU.

Conclusion

TVA has evaluated the effects of the proposed EPU on the ability of the CRAVS to provide a controlled environment for the comfort and safety of control room personnel and to support the operability of control room components. The evaluation indicates that the CRAVS will continue to provide an acceptable control room environment for safe operation of the plant following implementation of the proposed EPU and will continue to meet the requirements of draft GDCs-40 and 70, final GDC-19, and 10 CFR 50.67. Therefore, the proposed EPU is acceptable with respect to the CRAVS.

2.7.4 Spent Fuel Pool Area Ventilation System

Regulatory Evaluation

The function of the Spent Fuel Pool Area Ventilation System (SFPAVS) is to maintain ventilation in the spent fuel pool equipment areas, permit personnel access, and control airborne radioactivity in the area during normal operation, AOOs, and following postulated fuel handling accidents.

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

Specific NRC review criteria are contained in SRP Section 9.4.2.

Browns Ferry Current Licensing Basis

The Browns Ferry design does not include a separate spent fuel pool area ventilation system. Ventilation in this area is provided by the Reactor Building HVAC system under normal conditions. When required, the Standby Gas Treatment System maintains ventilation for this area.

Technical Evaluation

The Browns Ferry design does not include a separate spent fuel pool area ventilation system. As described above, during normal power operation the Reactor Building ventilation system

provides ventilation from the Refueling Floor (i.e., the Spent Fuel Pool Area). The SGTS performs this function during abnormal plant operations (accident conditions) and its EPU evaluation is described in Sections 2.5.2.1 and 2.6.6.

Conclusion

Not applicable. The Browns Ferry design does not include a separate spent fuel pool area ventilation system.

2.7.5 Reactor, Turbine, and Radwaste Building Ventilation Systems

Regulatory Evaluation

The function of the Reactor, Turbine, and Radwaste Building Ventilation Systems is to maintain ventilation in the reactor, turbine, and radwaste buildings, permit personnel access, and control the concentration of airborne radioactive material in these areas during normal operation, during AOOs, and after postulated accidents.

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

Specific NRC review criteria are contained in SRP Sections 9.4.3 and 9.4.4.

Browns Ferry Current Licensing Basis

The General Design Criteria (GDC) listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Browns Ferry Nuclear Plant (Browns Ferry) principal design criteria predate these criteria. The Browns Ferry principal design criteria are listed in UFSAR Section 1.5, "Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, the Tennessee Valley Authority (TVA) performed a comparative evaluation of the design basis of Browns Ferry with the AEC proposed General Design Criteria of 1967. The Browns Ferry UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria set out in the July 1967 AEC release. For each group of criteria, there is a statement of TVA's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a table of references to locations in the Browns Ferry UFSAR where there is subject matter relating to the intent of that particular criteria.

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

The Reactor, Turbine, and Radwaste Building Ventilation Systems are described in Browns Ferry UFSAR Section 10.12, "Heating Ventilating and Air-Conditioning Systems."

Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Browns Ferry License Renewal Safety Evaluation Report (SER), NUREG-1843, dated April 2006 (Reference 11). The license renewal evaluation associated with the auxiliary and radwaste area ventilation system is documented in NUREG-1843, Section 2.3.3.8. The management of the effects of aging on the auxiliary and radwaste area ventilation system is documented in NUREG-1843, Section 3.3.2.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of EPUs. Section 6.6 of the CLTR addresses the effect of EPU on CLTR Power Dependent Heating, Ventilation and Air Conditioning. The results of this evaluation are described below.

Browns Ferry meets all CLTR dispositions. The topics addressed in this evaluation are:

Topic	CLTR Disposition	Browns Ferry Result
Power Dependent HVAC Performance	Plant-Specific	Meets CLTR Disposition

The CLTR states that EPU results in slightly higher process temperatures and electrical loads on the HVAC system.

The Turbine Building, Reactor Building, drywell, and Radwaste Building ventilation systems evaluated in the CLTR are only those that are power dependent. The power dependent heating ventilation and air conditioning (HVAC) systems consist mainly of heating, cooling supply, exhaust, and recirculation units in the Turbine Building, Reactor Building, Radwaste Building, and the drywell. The control of the concentration of airborne radioactive material in the Reactor Building is controlled by the Reactor Building Ventilation system during normal operation. During AOOs and after postulated accidents, control of the concentration of airborne radioactive material in the reactor building is controlled by use of the Standby Gas Treatment System described in Sections 2.5.2 and 2.6.6. Monitoring of the Radwaste Building exhaust, and the Turbine Building exhaust, is not affected by EPU; additionally, monitoring of the Turbine Gland Sealing System and the Mechanical Vacuum Pump System are not affected by EPU.

At Browns Ferry, the normal operating EPU process temperatures affecting the normal HVAC loads increase slightly from CLTP values. However, the increases in temperatures and heat loads will not have a significant effect on the HVAC system.

Currently, no modifications are planned to any HVAC or atmospheric clean-up system and there is no significant EPU effect on HVAC systems during normal operation or accident conditions.

During normal operation, Main Steam Tunnel temperatures increase slightly due to an increase in FW temperature. However, the increase in temperature will be less than 0.5°F. Any heat load increases in the drywell are not considered significant, and are within existing system margin. The HVAC systems serving the remaining areas served by these HVAC systems are unaffected by the EPU because the process temperatures and equipment heat loads are not power dependent and not affected by EPU. Table 2.7-1 shows the effect of EPU on the Ventilation Systems.

The reactor recirculation system variable flow drive (VFD) installation, a modification that replaced the original reactor recirculation system motor-generator sets, resulted in a reduced heat load on the Reactor Building HVAC system. The maximum temperatures calculated are considered within the daily anticipated temperature fluctuation.

Based on a review of design basis calculations, the design of the HVAC is adequate for EPU. Therefore, the power dependent HVAC performance meets all CLTR dispositions.

Conclusion

TVA has evaluated the effects of the proposed EPU on the power dependent HVAC systems that serve the Reactor, Turbine, and Radwaste Buildings. Therefore, the proposed EPU is acceptable with respect to HVAC system operation in the Turbine Building, Reactor Building, and drywell.

2.7.6 Engineered Safety Feature Ventilation System

Regulatory Evaluation

The function of the Engineered Safety Feature Ventilation System (ESFVS) is to provide a suitable and controlled environment for ESF components following certain anticipated transients and DBAs.

The NRC's acceptance criteria for the ESFVS are based on (1) GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents; (2) GDC-17, insofar as it requires onsite and offsite electric power systems be provided to permit functioning of SSCs important to safety; and (3) GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents.

Specific NRC review criteria are contained in SRP Section 9.4.5.

Browns Ferry Current Licensing Basis

The General Design Criteria (GDC) listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Browns Ferry Nuclear Plant (Browns Ferry) principal design criteria predate these criteria. The Browns Ferry principal design criteria are listed in UFSAR Section 1.5, "Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967,

the Tennessee Valley Authority (TVA) performed a comparative evaluation of the design basis of Browns Ferry with the AEC proposed General Design Criteria of 1967. The Browns Ferry UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria set out in the July 1967 AEC release. For each group of criteria, there is a statement of TVA's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a table of references to locations in the Browns Ferry UFSAR where there is subject matter relating to the intent of that particular criteria.

While Browns Ferry is not generally licensed to the final GDC or the 1967 AEC proposed General Design Criteria, a comparison of the final GDC to the applicable AEC proposed General Design Criteria can usually be made. For the final GDC listed in the Regulatory Evaluation above, the Browns Ferry comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Browns Ferry UFSAR Appendix A: draft GDCs-40, 42, and 70. Final GDC-17 is applicable to Browns Ferry as described in UFSAR Section 8.3.

The engineered safety feature ventilation system is discussed in Browns Ferry UFSAR Section 10.12, "Heating, Ventilating and Air-Conditioning Systems," and UFSAR Section 5.3.3.6.2, "Equipment Area Cooling."

Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Browns Ferry License Renewal Safety Evaluation Report (SER), NUREG-1843, dated April 2006 (Reference 11). The license renewal evaluation associated with the engineered safety feature ventilation system is documented in NUREG-1843, Section 2.3.3.9. Management of aging effects on the engineered safety feature ventilation system is documented in NUREG-1843, Section 3.3.2.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of EPUs. Section 6.6 of the CLTR addresses the effect of EPU on CLTR Power Dependent Heating, Ventilation and Air Conditioning (HVAC). The results of this evaluation are described below.

Browns Ferry meets all CLTR dispositions. The topics addressed in this evaluation are:

Topic	CLTR Disposition	Browns Ferry Result
Power Dependent HVAC Performance	Plant-Specific	Meets CLTR Disposition

The CLTR states that slightly higher process temperatures and electrical loads occur as a result of EPU.

The ESF HVAC systems consist mainly of heating, cooling supply, exhaust, and recirculation units serving the Electric Board Room and Battery Rooms for Unit 3, the Standby Diesel Generator Rooms, and the ECCS Pump Rooms (RHR, HPCI, CS, and RCIC). These systems do not function to control the concentration of airborne radioactive material in these areas during normal operation, during AOOs, and after postulated accidents. The control of the concentration of airborne radioactive material in the secondary containment during normal operation, during AOOs, and after postulated accidents is accomplished using the Standby Gas Treatment System described in Sections 2.5.2 and 2.6.6.

During normal operation, the HVAC systems serving these areas are unaffected by the EPU because the process temperatures remain bounded by CLTP conditions.

The design basis post-LOCA Reactor Building temperatures will not increase with EPU. Increase in heat loads and temperatures in the ECCS pump rooms are negligible. Additionally, there are no major equipment modifications in the Electric Board Room and Battery Rooms, and therefore, design heat loads in these rooms will not change with EPU. The Diesel Generator remains below rated capacity and there is no electrical loading or process temperature change in this area. Therefore, there is no increase in design basis heat load for this area. Table 2.7-1 shows the effect of EPU on the Ventilation Systems.

Therefore, the power dependent HVAC performance meets all CLTR dispositions.

Conclusion

TVA has evaluated the effects of the proposed EPU on the ESFVS and the effects of the proposed EPU on the ability of the ESFVS to provide a suitable environment for ESF components. The evaluation indicates that the ESFVS will continue to assure a suitable environment for the ESF components following implementation of the proposed EPU and will continue to meet the requirements of draft GDCs-40, 42 and 70 and final GDC-17. Therefore, the proposed EPU is acceptable with respect to the ESFVS.

Table 2.7-1 EPU Effect on Ventilation Systems

System	EPU Effect
Turbine Building Ventilation System	Increases in process temperatures results in slight temperature increase. The turbine building is not an EQ zone. The design of the Turbine Building HVAC system is adequate to handle the increase in heat load.
Reactor Building Ventilation System	EPU does not result in significant temperature increases in areas of the Reactor Building. The expected increase in the Main Steam Tunnel is < 0.5°F, which is not significant. The temperature of the General Floor Area at El 639 will increase to a peak of 128.7°F for the most limiting Reactor Building room. The design of the HVAC system is adequate for EPU.
Drywell Ventilation System	EPU will not result in a significant increase in drywell heat load or area temperature increases (< 0.5°F). The drywell HVAC system is adequate to handle the small increase in heat load.
Radwaste Building Ventilation System	Negligible effect due to EPU.
Ventilation Systems for Miscellaneous Rooms and Buildings	Core Spray Pump room temperature will increase to a bounding 118.2°F. RHR Pump room temperature will increase to a bounding 131.3°F. The RHR heat exchanger rooms temperature will increase to a bounding 131.0°F. The bounding temperature is the Browns Ferry Unit 1, Unit 2, or Unit 3 highest temperature prediction for the respective room.
Control Room HVAC	Negligible effect due to EPU. No process temperature changes in the Control Room/Control Building.
Emergency Ventilating Systems	Negligible effect due to EPU. Some electrical operational loads may increase slightly, but will stay below design loads.

2.8 Reactor Systems

2.8.1 Fuel System Design

Regulatory Evaluation

The fuel system consists of arrays of fuel rods, burnable poison rods, spacer grids and springs, end plates, channel boxes, and reactivity control rods.

The NRC's acceptance criteria are based on (1) 10 CFR 50.46, insofar as it establishes standards for the calculation of ECCS performance and acceptance criteria for that calculated performance; (2) GDC-10, insofar as it requires that the reactor core be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs; (3) GDC-27, insofar as it requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to assure the capability to cool the core is maintained; and (4) GDC-35, insofar as it requires that a system to provide abundant emergency core cooling be provided to transfer heat from the reactor core following any LOCA.

Specific NRC review criteria are contained in SRP Section 4.2 and other guidance provided in Matrix 8 of RS-001.

Browns Ferry Current Licensing Basis

Final GDCs-10, 27 and 35 are applicable to Browns Ferry as described in "Browns Ferry Nuclear Plant, Unit 1 - Issuance of Amendments Regarding the Transition to Areva Fuel," dated July 3, 2012. (Reference 48)

The Fuel System Design is described in Browns Ferry UFSAR Chapter 3, "Reactor."

Technical Evaluation

See FUSAR Section 2.8.1.

Conclusion

The effects of the proposed EPU on the fuel system design of the fuel assemblies, control systems, and reactor core have been reviewed. The review has adequately accounted for the effects of the proposed EPU on the fuel system and demonstrated that: (1) the fuel system will not be damaged as a result of normal operation and AOOs; (2) the fuel system damage will never be so severe as to prevent control rod insertion when it is required; (3) the number of fuel rod failures will not be underestimated for postulated accidents; and (4) the fuel is adequately cooled during all operational modes. Based on this, it is concluded that the fuel system and associated analyses will continue to meet the requirements of 10 CFR 50.46 and the current licensing basis following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to the fuel system design.

2.8.2 Nuclear Design

Regulatory Evaluation

The NRC's acceptance criteria are based on (1) GDC-10, insofar as it requires that the reactor core be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs; (2) GDC-11, insofar as it requires that the reactor core be designed so that the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity; (3) GDC-12, insofar as it requires that the reactor core be designed to assure that power oscillations, which can result in conditions exceeding SAFDLs, are not possible or can be reliably and readily detected and suppressed; (4) GDC-13, insofar as it requires that instrumentation and controls be provided to monitor variables and systems affecting the fission process over anticipated ranges for normal operation, AOOs and accident conditions, and to maintain the variables and systems within prescribed operating ranges; (5) GDC-20, insofar as it requires that the protection system be designed to initiate the reactivity control systems automatically to assure that acceptable fuel design limits are not exceeded as a result of AOOs and to automatically initiate operation of systems and components important to safety under accident conditions; (6) GDC-25, insofar as it requires that the protection system be designed to assure that SAFDLs are not exceeded for any single malfunction of the reactivity control systems; (7) GDC-26, insofar as it requires that two independent reactivity control systems be provided, with both systems capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes; (8) GDC-27, insofar as it requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to assure the capability to cool the core is maintained; and (9) GDC-28, insofar as it requires that the reactivity control systems be designed to assure that the effects of postulated reactivity accidents can neither result in damage to the RCPB greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals so as to significantly impair the capability to cool the core.

Specific NRC review criteria are contained in SRP Section 4.3 and other guidance provided in Matrix 8 of RS-001.

Browns Ferry Current Licensing Basis

The General Design Criteria (GDC) listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Browns Ferry Nuclear Plant (Browns Ferry) principal design criteria predate these criteria. The Browns Ferry principal design criteria are listed in UFSAR Section 1.5, "Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, the Tennessee Valley Authority (TVA) performed a comparative evaluation of the design basis of Browns Ferry with the AEC proposed General Design Criteria of 1967. The Browns Ferry UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation.

This evaluation discusses each of the groups of criteria set out in the July 1967 AEC release. For each group of criteria, there is a statement of TVA's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a table of references to locations in the Browns Ferry UFSAR where there is subject matter relating to the intent of that particular criteria.

While Browns Ferry is not generally licensed to the final GDC or the 1967 AEC proposed General Design Criteria, a comparison of the final GDC to the applicable AEC proposed General Design Criteria can usually be made. For the final GDC listed in the Regulatory Evaluation above, the Browns Ferry comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Browns Ferry UFSAR Appendix A: draft GDCs-7, 8, 12, 13, 14, 15, 27, 28, 29, 30, 31, and 32. Final GDCs-10 and 27 are applicable to Browns Ferry as described in "Browns Ferry Nuclear Plant, Unit 1 - Issuance of Amendments Regarding the Transition to Areva Fuel," dated July 3, 2012. (Reference 48)

Nuclear design is described in Browns Ferry UFSAR Chapter 3, "Reactor."

<u>Technical Evaluation</u>

2.8.2.1 Core Operation

See FUSAR Section 2.8.2.1.

2.8.2.1.1 Fuel Thermal Margin Monitoring Threshold

See FUSAR Section 2.8.2.1.1.

2.8.2.2 Thermal Limits Assessment

See FUSAR Section 2.8.2.2.

2.8.2.3 Reactivity Characteristics

See FUSAR Section 2.8.2.3.

<u>Conclusion</u>

The effects of the proposed EPU on the nuclear design of the fuel assemblies, control systems, and reactor core have been reviewed. It has been concluded that the review has adequately accounted for the effects of the proposed EPU on the nuclear design and has demonstrated the fuel design limits will not be exceeded during normal or anticipated operational transients, and the effects of postulated reactivity accidents will not cause significant damage to the RCPB or impair the capability to cool the core. Based on this evaluation and in coordination with the reviews of the fuel system design, thermal and hydraulic design, and transient and accident analyses, it is concluded that the nuclear design of the fuel assemblies, control systems, and reactor core will continue to meet the applicable requirements of the current licensing basis. Therefore, the proposed EPU is acceptable with respect to the nuclear design.

2.8.3 Thermal and Hydraulic Design

Regulatory Evaluation

The NRC's acceptance criteria are based on (1) GDC-10, insofar as it requires that the reactor core be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs; and (2) GDC-12, insofar as it requires that the reactor core and associated coolant, control, and protection systems be designed to assure that power oscillations, which can result in conditions exceeding SAFDLs, are not possible or can reliably and readily be detected and suppressed.

Specific NRC review criteria are contained in SRP Section 4.4 and other guidance provided in Matrix 8 of RS-001.

Browns Ferry Current Licensing Basis

The General Design Criteria (GDC) listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Browns Ferry Nuclear Plant (Browns Ferry) principal design criteria predate these criteria. The Browns Ferry principal design criteria are listed in UFSAR Section 1.5, "Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, the Tennessee Valley Authority (TVA) performed a comparative evaluation of the design basis of Browns Ferry with the AEC proposed General Design Criteria of 1967. The Browns Ferry UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria set out in the July 1967 AEC release. For each group of criteria, there is a statement of TVA's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a table of references to locations in the Browns Ferry UFSAR where there is subject matter relating to the intent of that particular criteria.

While Browns Ferry is not generally licensed to the final GDC or the 1967 AEC proposed General Design Criteria, a comparison of the final GDC to the applicable AEC proposed General Design Criteria can usually be made. For the final GDC listed in the Regulatory Evaluation above, the Browns Ferry comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Browns Ferry UFSAR Appendix A: draft GDC-7. Final GDC-10 is applicable to Browns Ferry as described in "Browns Ferry Nuclear Plant, Unit 1 - Issuance of Amendments Regarding the Transition to Areva Fuel," dated July 3, 2012. (Reference 48)

The Thermal and Hydraulic Design is described in Browns Ferry UFSAR Section 3.7, "Thermal and Hydraulic Design." Power oscillations are addressed in Browns Ferry UFSAR Appendix N, "Reload Licensing Report."

Technical Evaluation

See FUSAR Section 2.8.3.

Conclusion

TVA has evaluated the effects of the proposed EPU on the thermal and hydraulic design of the core and the RCS. The evaluation indicates that the thermal and hydraulic design will continue to meet the requirements of final GDC-10 and draft GDC-7 following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to thermal and hydraulic design.

2.8.4 Emergency Systems

2.8.4.1 Functional Design of Control Rod Drive System

Regulatory Evaluation

The NRC's acceptance criteria are based on (1) GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents; (2) GDC-23, insofar as it requires that the protection system be designed to fail into a safe state; (3) GDC-25, insofar as it requires that the protection system be designed to assure that SAFDLs are not exceeded for any single malfunction of the reactivity control systems; (4) GDC-26, insofar as it requires that two independent reactivity control systems be provided, with both systems capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes; (5) GDC-27, insofar as it requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to assure the capability to cool the core is maintained; (6) GDC-28, insofar as it requires that the reactivity control systems be designed to assure that the effects of postulated reactivity accidents can neither result in damage to the RCPB greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals so as to significantly impair the capability to cool the core; (7) GDC-29, insofar as it requires that the protection and reactivity control systems be designed to assure an extremely high probability of accomplishing their safety functions in event of AOOs; and (8) 10 CFR 50.62(c)(3), insofar as it requires that all BWRs have an alternate rod injection (ARI) system diverse from the reactor trip system, and that the ARI system have redundant scram air header exhaust valves.

Specific NRC review criteria are contained in SRP Section 4.6.

Browns Ferry Current Licensing Basis

The General Design Criteria (GDC) listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Browns Ferry Nuclear Plant (Browns Ferry) principal design criteria predate these criteria. The Browns Ferry principal design criteria are listed in UFSAR Section 1.5, "Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, the Tennessee Valley Authority (TVA) performed a comparative evaluation of the design basis of Browns Ferry with the AEC proposed General Design Criteria of 1967. The Browns Ferry UFSAR, Appendix A,

"Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria set out in the July 1967 AEC release. For each group of criteria, there is a statement of TVA's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a table of references to locations in the Browns Ferry UFSAR where there is subject matter relating to the intent of that particular criteria.

While Browns Ferry is not generally licensed to the final GDC or the 1967 AEC proposed General Design Criteria, a comparison of the final GDC to the applicable AEC proposed General Design Criteria can usually be made. For the final GDC listed in the Regulatory Evaluation above, with the exception of final GDC-27 and GDC-29, the Browns Ferry comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Browns Ferry UFSAR Appendix A: draft GDCs-26, 27, 28, 29, 30, 31, 32, 40, and 42.

The design of the Control Rod Drive system is described in Browns Ferry UFSAR Section 3.4, "Reactivity Control Mechanical Design."

Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Browns Ferry License Renewal Safety Evaluation Report (SER), NUREG-1843, dated April 2006 (Reference 11). The license renewal evaluation associated with the Control Rod Drive System is documented in NUREG-1843, Section 2.3.3.29. Management of aging effects on the Control Rod Drive System is documented in NUREG-1843, Section 3.3.2.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of EPUs. Section 2.5 of the CLTR addresses the effect of EPU on the functional design of the CRD system. The results of this evaluation are described below.

As stated in Section 2.5 of the CLTR, the CRD system is used to control core reactivity by positioning neutron absorbing control rods within the reactor and to scram the reactor by rapidly inserting withdrawn control rods into the core.

Browns Ferry meets all CLTR dispositions. The topics addressed in this evaluation are:

Topic	CLTR Disposition	Browns Ferry Result
Scram Time Response	Generic	Meets CLTR Disposition
CRD Positioning	Generic	Meets CLTR Disposition

Topic	CLTR Disposition	Browns Ferry Result
CRD Cooling	Generic	Meets CLTR Disposition
CRD Integrity	Generic	Meets CLTR Disposition

All Browns Ferry units use BWR/6 control rod drives modified for use in pre-BWR/6 plants. The BWR/6 control rod drives are acceptable replacements for BWR/4 control rod drives for all performance parameters and do not affect CLTR dispositions.

2.8.4.1.1 Scram Time Response

The CLTR states that for pre-BWR/6 plants, the scram times are decreased by the transient pressure response, and therefore the effect of EPU is bounded by the current response.

At normal operating conditions, the CRD hydraulic control unit accumulator supplies the initial scram pressure and, as the scram continues, the reactor becomes the primary source of pressure to complete the scram. Because the normal reactor dome pressure for EPU does not change, the scram time performance relative to current plant operation is the same. Therefore, pre-BWR/6 plants will retain their current TS scram requirement.

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The CRD system control rod scram at Browns Ferry is confirmed to be consistent with the generic description provided in the CLTR for pre-BWR/6 plants because Browns Ferry is a BWR/4 plant.

2.8.4.1.2 Control Rod Drive Positioning and Cooling

As stated in Section 2.5 of the CLTR, the increase in reactor power at the EPU operating condition results in [[ ]] from the CRD system to the CRDs during normal plant operation.

EPU is evaluated on the basis of operation at the same dome pressure but higher core power and steam flow. [[ ]]

2.8.4.1.2.1 Control Rod Drive Positioning

The CLTR states that, with reactor dome pressure unchanged, there is [[ ]], and the automatic operation of the system flow control valve maintains the required drive water pressure. Therefore, the CRD

positioning function is not affected. The normal CRD positioning function is an operational consideration, not a safety-related function, and is not affected by EPU operating conditions.

For Browns Ferry, plant operating data has confirmed that [[

]]

Therefore, the CRD system drive positioning meets all CLTR dispositions.

2.8.4.1.2.2 Control Rod Drive Cooling

The CLTR states that, with reactor dome pressure unchanged, there is [[ ]], and the automatic operation of the system flow control valve maintains the required cooling water flow rate. Therefore, the CRD cooling function is not affected. The CRD cooling function is an operational consideration, not a safety-related function, and is not affected by EPU operating conditions.

For Browns Ferry, plant operating data has confirmed that [[

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Therefore, the CRD system drive cooling meets all CLTR dispositions.

2.8.4.1.3 Control Rod Drive Integrity Assessment

The CLTR states that [[ ]] on CRD integrity. The transient pressures due to uprated power may create higher pressure loadings.

The postulated abnormal operating condition for the CRD design assumes a failure of the CRD system pressure-regulating valve that applies the maximum pump discharge pressure to the CRD mechanism internal components. [[

]] Other mechanical loadings are [[ ]] addressed in Sections 2.2.2 and 2.2.3.

Therefore, the CRD system integrity meets all CLTR dispositions.

Conclusion

TVA has evaluated the effects of the proposed EPU on the CRD system. The evaluation indicates that the system's ability to effect a safe shutdown, respond within acceptable limits, and prevent or mitigate the consequences of postulated accidents will be maintained following the implementation of the proposed EPU. Based on this, TVA concludes that the fuel system and associated analyses will continue to meet the requirements of draft GDCs-26, 27, 28, 29, 30, 31, 32, 40, and 42 and 10 CFR 50.62(c)(3) following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to the functional design of the CRD system.

2.8.4.2 Overpressure Protection During Power Operation

Regulatory Evaluation

Relief and safety valves and the reactor protection system provide overpressure protection for the RCPB during power operation.

The NRC's acceptance criteria are based on (1) GDC-15, insofar as it requires that the RCS and associated auxiliary, control, and protection systems be designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including AOOs; and (2) GDC-31, insofar as it requires that the RCPB be designed with sufficient margin to assure that it behaves in a non-brittle manner and that the probability of rapidly propagating fracture is minimized.

Specific NRC review criteria are contained in SRP Section 5.2.2.

Browns Ferry Current Licensing Basis

The General Design Criteria (GDC) listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Browns Ferry Nuclear Plant (Browns Ferry) principal design criteria predate these criteria. The Browns Ferry principal design criteria are listed in UFSAR Section 1.5, "Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, the Tennessee Valley Authority (TVA) performed a comparative evaluation of the design basis of Browns Ferry with the AEC proposed General Design Criteria of 1967. The Browns Ferry UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria set out in the July 1967 AEC release. For each group of criteria, there is a statement of TVA's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a table of references to locations in the Browns Ferry UFSAR where there is subject matter relating to the intent of that particular criteria.

While Browns Ferry is not generally licensed to the final GDC or the 1967 AEC proposed General Design Criteria, a comparison of the final GDC to the applicable AEC proposed General Design Criteria can usually be made. For the final GDC listed in the Regulatory Evaluation above, with the exception of final GDC-15, the Browns Ferry comparative evaluation of the comparable 1967 AEC

proposed General Design Criteria (referred to here as "draft GDC") is contained in Browns Ferry UFSAR Appendix A: draft GDCs-9, 33, 34, and 35. There is no draft GDC directly associated with final GDC-15.

Overpressure protection during power operation is described in Browns Ferry UFSAR Section 4.4, "Nuclear System Pressure Relief System."

Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Browns Ferry License Renewal Safety Evaluation Report (SER), NUREG-1843, dated April 2006 (Reference 11). The license renewal evaluation associated with overpressure protection is documented in NUREG-1843, Section 2.3.1.3. Management of aging effects on overpressure protection is documented in NUREG-1843, Section 3.1.2.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of EPUs. Section 3.1 of the CLTR addresses the effect of EPU on nuclear system pressure relief/overpressure protection. The results of this evaluation are described below.

As stated in Section 3.1 of the CLTR, the system operating pressure does not change but the steam flow rate increases. The increased steam flow rate associated with uprated power may increase steam line vibration. The increased core steam generation also causes an increase in the pressurization during some transient events.

Browns Ferry meets all CLTR dispositions. The topics addressed in this evaluation are:

Topic	CLTR Disposition	Browns Ferry Result
Overpressure Capacity	Plant Specific	Meets CLTR Disposition

The CLTR states that the increased core steam generation causes an increase in the pressurization during some transient events.

The nuclear system pressure relief system prevents over-pressurization of the nuclear system during AOOs, the plant ASME upset overpressure protection event, and postulated ATWS events. The plant SRVs, along with other functions, provide this protection. An evaluation was performed in order to confirm the adequacy of the pressure relief system for EPU conditions.

The SRV discharge lines were designed and configured so that the discharge backpressure at the valve outlet is not greater than 40% of the inlet pressure. The valves were designed to achieve sonic (choked) flow conditions through the valve up to this backpressure ratio to provide flow independence to the discharge piping losses and backpressure. The backpressure to inlet pressure

ratio is a function of discharge line geometry, which will not change with EPU. Therefore, SRV capacity will not be affected by the EPU discharge line backpressure.

See FUSAR Section 2.8.4.2 for details of the overpressure capacity evaluation.

Conclusion

TVA has evaluated the effects of the proposed EPU on the overpressure protection capability of the plant during power operation. The evaluation indicates that the plant will continue to have sufficient pressure relief capacity to ensure that pressure limits are not exceeded. Based on this, TVA concludes that the overpressure protection features will continue to meet draft GDCs-9, 33, 34, and 35 following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to overpressure protection during power operation.

2.8.4.3 Reactor Core Isolation Cooling System

Regulatory Evaluation

The RCIC system serves as a standby source of cooling water to provide a limited decay heat removal capability whenever the main feedwater system is isolated from the reactor vessel. In addition, the RCIC system may provide decay heat removal necessary for coping with a station blackout. The water supply for the RCIC system comes from the condensate storage tank, with a secondary supply from the suppression pool.

The NRC's acceptance criteria are based on (1) GDC-4, insofar as it requires that SSCs important to safety be protected against dynamic effects; (2) GDC-5, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be demonstrated that sharing will not impair its ability to perform its safety function; (3) GDC-29, insofar as it requires that the protection and reactivity control systems be designed to assure an extremely high probability of accomplishing their safety functions in event of AOOs; (4) GDC-33, insofar as it requires that a system to provide reactor coolant makeup for protection against small breaks in the RCPB be provided so the fuel design limits are not exceeded; (5) GDC-34, insofar as it requires that a residual heat removal system be provided to transfer fission product decay heat and other residual heat from the reactor core at a rate such that SAFDLs and the design conditions of the RCPB are not exceeded; (6) GDC-54, insofar as it requires that piping systems penetrating containment be designed with the capability to periodically test the operability of the isolation valves to determine if valve leakage is within acceptable limits; and (7) 10 CFR 50.63, insofar as it requires that the plant withstand and recover from an SBO of a specified duration.

Specific NRC review criteria are contained in SRP Section 5.4.6.

Browns Ferry Current Licensing Basis

The General Design Criteria (GDC) listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Browns Ferry Nuclear Plant (Browns Ferry) principal design criteria predate these criteria. The Browns Ferry principal design criteria are listed in UFSAR Section 1.5, "Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (Federal Register 32FR10213, July 11, 1967). Although not

explicitly licensed to the AEC proposed General Design Criteria published in 1967, the Tennessee Valley Authority (TVA) performed a comparative evaluation of the design basis of Browns Ferry with the AEC proposed General Design Criteria of 1967. The Browns Ferry UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria set out in the July 1967 AEC release. For each group of criteria, there is a statement of TVA's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a table of references to locations in the Browns Ferry UFSAR where there is subject matter relating to the intent of that particular criteria.

While Browns Ferry is not generally licensed to the final GDC or the 1967 AEC proposed General Design Criteria, a comparison of the final GDC to the applicable AEC proposed General Design Criteria can usually be made. For the final GDC listed in the Regulatory Evaluation above, with the exception of final GDC-29 and final GDC-34, the Browns Ferry comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Browns Ferry UFSAR Appendix A: draft GDCs-4, 40, 51, and 57. There is no draft GDC directly applicable to final GDC-29 or final GDC-34.

The Reactor Core Isolation Cooling System is described in Browns Ferry UFSAR Section 4.7, "Reactor Core Isolation Cooling System."

Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Browns Ferry License Renewal Safety Evaluation Report (SER), NUREG-1843, dated April 2006 (Reference 11). The license renewal evaluation associated with the RCIC system is documented in NUREG-1843, Section 2.3.3.23. Management of aging effects on the RCIC system is documented in NUREG-1843, Section 3.3.2.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of EPUs. Section 3.9 of the CLTR addresses the effect of EPU on the reactor core isolation cooling system. The results of this evaluation are described below.

The RCIC system evaluation for EPU at Browns Ferry addressed the following topics:

- System performance and hardware
- Net positive suction head
- Adequate core cooling for limiting LOFW events (Addressed in Section 2.8.5.2.3)
- Inventory makeup - Operational Level 1 avoidance (Addressed in Section 2.8.5.2.3)

Browns Ferry meets all CLTR dispositions. The topics addressed in this evaluation are:

Topic	CLTR Disposition	Browns Ferry Result
System Performance and Hardware (RCIC)	Generic	Meets CLTR Disposition
Net Positive Suction Head (RCIC)	Generic	Meets CLTR Disposition

2.8.4.3.1 System Performance and Hardware

The CLTR states that there is no effect on RCIC system performance and hardware due to EPU.

The RCIC system is required to maintain sufficient water inventory in the reactor to permit adequate core cooling following a reactor vessel isolation event accompanied by loss of flow from the FW system. The system design injection rate must be sufficient for compliance with the system limiting criteria to maintain the reactor water level above Top of Active Fuel (TAF) at EPU conditions. The RCIC system is designed to pump water into the reactor vessel over a wide range of operating pressures. The results of the Browns Ferry plant-specific evaluation indicate adequate water level margin above TAF at EPU conditions. Thus, the RCIC injection rate is adequate to meet this design basis event.

An operational requirement is that the RCIC system can restore the reactor water level while avoiding ADS timer initiation and MSIV closure activation functions associated with the low-low-low reactor water level setpoint (Level 1). This requirement is intended to avoid unnecessary initiations of safety systems. The results of the Browns Ferry plant-specific evaluation indicate that the RCIC system is capable of maintaining the water level outside the shroud above the Level 1 setpoint through a limiting LOFW event at EPU conditions. Thus, the RCIC injection rate is adequate to meet the requirements for inventory makeup. The reactor system response to a LOFW transient with RCIC is discussed in Section 2.8.5.2.3.

For EPU, there is no change to the normal reactor operating dome pressure (1,050 psia for both CLTP and EPU conditions), and the SRV setpoints remain the same. There is no change to the maximum specified reactor pressure for RCIC system operation, [[

]] The Browns Ferry RCIC pump is adequate to support EPU.
[[
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The system performance and hardware for RCIC meets all CLTR dispositions.

2.8.4.3.2 Net Positive Suction Head

The CLTR states that there is no effect on RCIC Net Positive Suction Head (NPSH) due to EPU.

The Browns Ferry minimum NPSH available for the Browns Ferry RCIC pump does not change because there are no physical changes to the pump suction configuration, and no changes to the

system flow rate or minimum atmospheric pressure in the suppression pool or Condensate Storage Tank (CST). EPU does not affect the capability to transfer the RCIC pump suction on high suppression pool level or low CST level from its normal alignment, the CST, to the suppression pool. EPU does not affect the capability to transfer the RCIC pump suction on high suppression pool temperature to the CST and does not change the existing requirements for the transfer. Therefore, the specified operational temperature limit for the process water does not change with EPU. Because Browns Ferry is not changing the RCIC pump or its operating parameters, the required NPSH does not change.

The effect of EPU on the operation of the RCIC system during SBO events is discussed in Section 2.3.5. The effect of EPU on the operation of the RCIC system during a Fire Event is discussed in Section 2.5.1.4.

Maximum pump speed and maximum pump injection flow are unchanged for EPU and there is no change in maximum normal operating dome pressure (1,050 psia at CLTP and EPU conditions). The SRV setpoints remain the same. No RCIC system power dependent functions or operating requirements (flows, pressure, temperature, and NPSH) are added or changed from the original design or licensing bases.

The RCIC NPSH at Browns Ferry meets all CLTR dispositions.

Conclusion

TVA has evaluated the effects of the proposed EPU on the ability of the RCIC system to provide decay heat removal following an isolation of main feedwater event. The evaluation indicates that the RCIC system will continue to provide sufficient decay heat removal and makeup for this event following implementation of the proposed EPU. Based on this, TVA concludes that the RCIC system will continue to meet the requirements draft GDCs-4, 40, 51, and 57 following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to the RCIC system.

2.8.4.4 Residual Heat Removal System

Regulatory Evaluation

The RHR system is used to cool down the RCS following shutdown. The RHR system is typically a low-pressure system that takes over the shutdown cooling function when the RCS temperature is reduced.

The NRC's acceptance criteria are based on (1) GDC-4, insofar as it requires that SSCs important to safety be protected against dynamic effects; (2) GDC-5, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions; and (3) GDC-34, which specifies requirements for an RHR system.

Specific NRC review criteria are contained in SRP Section 5.4.7 and other guidance provided in Matrix 8 of RS-001.

Browns Ferry Current Licensing Basis

The General Design Criteria (GDC) listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Browns Ferry Nuclear Plant (Browns Ferry) principal design criteria predate these criteria. The Browns Ferry principal design criteria are listed in UFSAR Section 1.5, "Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, the Tennessee Valley Authority (TVA) performed a comparative evaluation of the design basis of Browns Ferry with the AEC proposed General Design Criteria of 1967. The Browns Ferry UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria set out in the July 1967 AEC release. For each group of criteria, there is a statement of TVA's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a table of references to locations in the Browns Ferry UFSAR where there is subject matter relating to the intent of that particular criteria.

While Browns Ferry is not generally licensed to the final GDC or the 1967 AEC proposed General Design Criteria, a comparison of the final GDC to the applicable AEC proposed General Design Criteria can usually be made. For the final GDC listed in the Regulatory Evaluation above, with the exception of final GDC-34, the Browns Ferry comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Browns Ferry UFSAR Appendix A: draft GDCs-4, 40, and 42. There is no draft GDC directly associated with final GDC-34.

The Residual Heat Removal system is described in Browns Ferry UFSAR Section 4.8, "Residual Heat Removal System (RHRS)."

Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Browns Ferry License Renewal Safety Evaluation Report (SER), NUREG-1843, dated April 2006 (Reference 11). The license renewal evaluation associated with the RHR system is documented in NUREG-1843, Section 2.3.2.4. Management of aging effects on the RHR system is documented in NUREG-1843, Section 3.2.2.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of EPUs. Section 3.10 of the CLTR addresses the effect of EPU on the RHR system. The results of this evaluation are described below.

As explicitly stated in Section 3.10 of the CLTR, the RHR system is designed to restore and maintain the reactor coolant inventory following a LOCA and remove reactor decay heat following reactor shutdown for normal, transient, and accident conditions. The EPU effect on the RHR

system is a result of the higher decay heat in the core corresponding to the uprated power and the increased amount of reactor heat discharged into the containment during a LOCA.

For Browns Ferry, the RHR system is designed to operate in the LPCI mode, SDC mode, SPC mode, CSC mode, standby cooling mode and FPC assist (supplemental spent fuel pool cooling). Browns Ferry meets all CLTR dispositions. The topics addressed in this evaluation are:

Topic	CLTR Disposition	Browns Ferry Result
LPCI Mode	Generic	Addressed in Section 2.8.5.6.2
Suppression Pool and Containment Spray Cooling Modes	Plant Specific	Addressed in Section 2.6.5
Shutdown Cooling Mode	Plant Specific	Meets CLTR Disposition
Fuel Pool Cooling Assist	Plant Specific	Addressed in Section 2.5.3.1
Standby Cooling Mode	Plant Specific	Addressed in Section 2.8.4.4.5

2.8.4.4.1 LPCI Mode

The CLTR states that there is no change in the reactor pressures at which the LPCI mode of RHR is required. The LPCI mode, as it relates to the LOCA response, is discussed in Section 2.8.5.6.2, which concludes that 10 CFR 50.46 limits are met at EPU conditions. The LPCI system at Browns Ferry meets all CLTR dispositions.

2.8.4.4.2 Suppression Pool and Containment Spray Cooling

The CLTR states that the suppression pool temperature increases as a result of the higher decay heat associated with EPU. The SPC mode is manually initiated following isolation transients and a postulated LOCA to maintain the containment pressure and suppression pool temperature within design limits. The CSC mode reduces drywell pressure, drywell temperature, and suppression chamber pressure following an accident. The adequacy of these operating modes is demonstrated by the containment analysis (Section 2.6.5).

Suppression pool temperatures for evaluated design basis EPU events remain within the design limits. Therefore, the suppression pool temperature during a postulated LOCA at EPU conditions does not change the capabilities of RHR system equipment to perform the SPC and CSC functions. Containment pressures for these EPU events increased slightly above the CLTP analyzed pressures, but remained below the existing peak containment internal pressure limit. The slight increase in the predicted containment pressure during a postulated LOCA at EPU conditions (See Table 2.6-1)

remains within the equipment design parameters and thus does not adversely affect the hardware capabilities of RHR system equipment to perform the SPC and CSC functions. Therefore, the Suppression Pool and Containment Spray Cooling modes meet all CLTR dispositions.

2.8.4.4.3 Shutdown Cooling Mode

The CLTR states that a longer time is required for reactor cool down as a result of the higher decay heat associated with EPU. The SDC mode is designed to remove the sensible and decay heat from the reactor primary system during a normal reactor shutdown. This non-safety operational mode allows the reactor to be cooled down within a certain time objective, so that the SDC mode of operation will not become critical path during refueling operations. EPU increases the reactor decay heat, which requires a longer time for cooling down the reactor. The SDC analysis for EPU determined that the time needed for cooling the reactor to 125°F during normal reactor shutdown, with two RHR pumps and associated heat exchangers in service, is increased to approximately 34 hours at EPU conditions from approximately 24 hours at CLTP. The increase in the normal reactor shutdown time for EPU indicates that a normal reactor shutdown may take longer, which could affect outage schedules. This may have an effect on plant availability, but has no effect on plant safety or the design operating margins and therefore, requires no change to the RHR system. Therefore, the SDC mode meets all CLTR dispositions.

2.8.4.4.4 Fuel Pool Cooling Assist

The CLTR states that the spent fuel pool heat load increases due to the decay heat generation as a result of the EPU. The FPC assist (supplemental spent fuel pool cooling) mode, using existing RHR system heat removal capacity, provides supplemental fuel pool cooling capability in the event that the fuel pool heat load exceeds the heat removal capability of the Fuel Pool Cooling and Cleanup (FPCC) system. The adequacy of fuel pool cooling, including use of the spent fuel pool cooling mode, is discussed in Section 2.5.3.1, which concludes that EPU does not affect this system. Therefore, the FPC assist mode meets all CLTR dispositions.

2.8.4.4.5 Standby Cooling Mode

The RHR interunit crossties and the RHR service water standby coolant supply connection provide a long-term reactor core and primary containment cooling capability. These capabilities provide added long-term redundancy to the emergency core and containment cooling systems by the ability to utilize the adjacent unit's RHR equipment. These non-safety-related functions are provided with remote/manual isolation valves normally aligned in the closed position.

These functions are not affected by EPU because there is no change to the performance requirements for other emergency core and containment cooling systems.

Conclusion

TVA has evaluated the effects of the proposed EPU on the RHR system. The evaluation indicates that the RHR system will maintain its ability to cool the RCS following shutdown and provide decay heat removal. Based on this, TVA concludes that the RHR system will continue to meet the

requirements of the current licensing basis following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to the RHR system.

2.8.4.5 Standby Liquid Control System

Regulatory Evaluation

The Standby Liquid Control System (SLCS) provides backup capability for reactivity control independent of the control rod system. The SLCS injects a boron solution into the reactor to effect shutdown.

The NRC's acceptance criteria are based on (1) GDC-26, insofar as it requires that two independent reactivity control systems of different design principles be provided, and that one of the systems be capable of holding the reactor subcritical in the cold condition; (2) GDC-27, insofar as it requires that the reactivity control systems have a combined capability, in conjunction with poison addition by the ECCS, to reliably control reactivity changes under postulated accident conditions; and (3) 10 CFR 50.62(c)(4), insofar as it requires that the SLCS be capable of reliably injecting a borated water solution into the reactor pressure vessel at a boron concentration, boron enrichment, and flow rate that provides a set level of reactivity control.

Specific NRC review criteria are contained in SRP Section 9.3.5 and other guidance provided in Matrix 8 of RS-001.

Browns Ferry Current Licensing Basis

The General Design Criteria (GDC) listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Browns Ferry Nuclear Plant (Browns Ferry) principal design criteria predate these criteria. The Browns Ferry principal design criteria are listed in UFSAR Section 1.5, "Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, the Tennessee Valley Authority (TVA) performed a comparative evaluation of the design basis of Browns Ferry with the AEC proposed General Design Criteria of 1967. The Browns Ferry UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria set out in the July 1967 AEC release. For each group of criteria, there is a statement of TVA's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a table of references to locations in the Browns Ferry UFSAR where there is subject matter relating to the intent of that particular criteria.

While Browns Ferry is not generally licensed to the final GDC or the 1967 AEC proposed General Design Criteria, a comparison of the final GDC to the applicable AEC proposed General Design Criteria can usually be made. For the final GDC listed in the Regulatory Evaluation above, with the exception of final GDC-27, the Browns Ferry comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Browns Ferry

UFSAR Appendix A: draft GDCs-27, 28, 29, and 30. There is no draft GDC directly associated with final GDC-27.

The SLCS is described in Browns Ferry UFSAR Section 3.8, "Standby Liquid Control System."

Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Browns Ferry License Renewal Safety Evaluation Report (SER), NUREG-1843, dated April 2006 (Reference 11). The license renewal evaluation associated with SLCS is documented in NUREG-1843, Section 2.3.3.18. Management of aging effects on SLCS is documented in NUREG-1843, Section 3.3.2.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of EPUs. Section 6.5 of the CLTR addresses the effect of EPU on SLCS. The results of this evaluation are described below.

The SLCS is designed to shut down the reactor from rated power conditions to cold shutdown in the postulated situation that some or all of the control rods cannot be inserted. This manually operated system pumps a highly enriched sodium pentaborate solution into the vessel, to provide neutron absorption and achieve a subcritical reactor condition. SLCS is designed to inject over a wide range of reactor operating pressures.

The SLCS is also credited in the radiological dose analysis for a LOCA to provide a buffering agent (sodium pentaborate) to the suppression pool water. The use of a buffering agent is needed to ensure that the suppression pool pH remains above 7.0 under worst case conditions for 30 days following a LOCA.

Browns Ferry meets all CLTR dispositions. The topics addressed in this evaluation are:

Topic	CLTR Disposition	Browns Ferry Result
Core Shutdown Margin	Generic	Meets CLTR Disposition
System Performance and Hardware	Plant Specific	Meets CLTR Disposition
Suppression Pool Temperature Following Limiting ATWS Event	Plant Specific	Meets CLTR Disposition

2.8.4.5.1 Core Shutdown Margin

Section 6.5 of the CLTR states that the ability of the SLCS boron solution to achieve and maintain safe shutdown is not a direct function of core thermal power, and therefore, is not affected by EPU.

SLCS shutdown capability (in terms of the required reactor boron concentration) is reevaluated for each fuel reload. The cold boron shutdown concentration of 720 ppm natural boron for Unit 1 does not change for EPU. The cold boron shutdown concentration of 660 ppm natural boron for Units 2 and 3 changes to 720 ppm for EPU. No changes are necessary to the solution volume / concentration or to the boron-10 enrichment for EPU to achieve the required reactor boron concentration for cold shutdown conditions for Unit 1. Because of the increase in cold boron shutdown concentration for Units 2 and 3, the minimum weight of boron-10 to be injected to achieve cold shutdown conditions changes for Units 2 and 3 for EPU from 186 lbs to 203 lbs. The boron-10 enrichment, for all units, is changing to increase the boron injection rate for the ATWS analysis from 63.1 atom-% to 94 atom-%.

Therefore, the SLCS shutdown margin capability meets all CLTR dispositions.

2.8.4.5.2 System Performance and Hardware

As stated in Section 6.5 of the CLTR, the effect of EPU on system performance and hardware is increased heat load and potential increase in transient reactor pressure. The SLCS is designed for injection at a maximum reactor pressure equal to the upper AV for the lowest group of SRVs operating in the safety relief mode. At Browns Ferry, the nominal reactor dome pressure and the SRV setpoints are unchanged for EPU. Consequently, the capability of the Browns Ferry SLCS to provide its backup shutdown function is not affected by EPU. The SLCS is not dependent upon any other SRV operating modes.

Based on the results of the Browns Ferry EPU ATWS analysis, the maximum reactor lower plenum pressure following the limiting ATWS event reaches 1,201 psig (1,216 psia) during the time the SLCS is analyzed to be in operation. For EPU, the maximum SLCS pump discharge pressure is 1,295 psig and the operating pressure margin for the pump discharge relief valves remains acceptable. Consideration was given to system flow, head losses for full injection, and cyclic pressure pulsations due to the positive displacement pump operation in determining the pressure margin to the opening set point for the pump discharge relief valves. The relief valve setpoint margin is 33 psi for EPU. This margin is based on a SLCS pump relief valve nominal setpoint of 1,425 psig. The pump discharge relief valves are periodically tested to confirm the setpoint. The operation of the pump discharge system was analyzed to confirm that the loss of flow through an open relief valve would not compromise the required boron injection function (due to an early SLCS initiation). The evaluation compared the open/close setpoint of the pump discharge relief valves with the calculated maximum SLCS pump discharge pressure expected during the most limiting ATWS transient. It was confirmed that the SLCS relief valves would close prior to analyzed initiation if system initiation were to occur prior to the reactor pressure recovering from the initial transient peak. Therefore, the current SLCS process parameters associated with the minimum boron injection rate are not changed.

The SLCS ATWS performance is evaluated in Section 2.8.5.7 for a representative core design for EPU. The evaluation confirmed acceptable results and demonstrates that EPU has no adverse effect on the ability of the SLCS to mitigate an ATWS. Therefore, Browns Ferry SLCS performance and hardware meet all CLTR dispositions.

2.8.4.5.3 Suppression Pool Temperature Following an ATWS Event

The boron injection rate requirement, specifically the isotopic enrichment of boron-10, for maintaining the peak suppression pool water temperature limits following the limiting ATWS event with SLCS injection is increased for EPU. The suppression pool temperature following an ATWS event at Browns Ferry was determined on a plant-specific basis consistent with Section L.3.3 of ELTR1 (Reference 4).

A higher boron-10 injection rate will allow a more rapid shutdown of the reactor after an ATWS event, resulting in a lower integrated heat addition to the suppression pool and a subsequent lower suppression pool temperature, with the benefit of gaining margin for ECCS pump net positive suction head.

2.8.4.5.4 Suppression Pool pH Control

Suppression pool pH control following a LOCA is not affected by EPU and no changes are required to the minimum sodium pentaborate solution concentration and volume.

Conclusion

TVA has evaluated the effects of the proposed EPU on the SLCS. The evaluation indicates that the system will continue to provide the function of reactivity control independent of the control rod system following implementation of the proposed EPU. Based on this, TVA concludes that the SLCS will continue to meet the requirements of draft GDCs-27, 28, 29, and 30 and 10 CFR 50.62(c)(4) following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to the SLCS.

2.8.4.6 Reactor Recirculation System Performance

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of EPUs. Section 3.6 of the CLTR addresses the effect of EPU on the Reactor Recirculation System (RRS). The results of this evaluation are described below. RRS performance is not specifically addressed in NRC "Review Standard for Extended Power Uprates," RS-001.

The EPU power condition is accomplished by operating along extensions of current rod lines on the power/flow map with no increase in the maximum core flow. The core reload analyses are performed with the most conservative allowable core flow. The evaluation of the RRS performance at EPU power determines that adequate core flow can be maintained.

The cavitation protection interlock remains the same in terms of absolute flow rates. This interlock is based on subcooling in the external recirculation loop and thus is a function of absolute FW flow

rate and FW temperature at less than full thermal power operating conditions. Therefore, the interlock is not changed by EPU.

Browns Ferry meets all CLTR dispositions. The topics addressed in this evaluation are:

Topic	CLTR Disposition	Browns Ferry Result
Net Positive Suction Head (NPSH)	Generic	Meets CLTR Disposition
Flow Mismatch	Generic	Meets CLTR Disposition
Single-Loop Operation	Generic	Meets CLTR Disposition

2.8.4.6.1 Net Positive Suction Head

The CLTR states that increased voids in the core during normal uprated power operation requires a slight increase in the recirculation drive flow to achieve the same core flow.

The CLTR shows that recirculation pump NPSH at full EPU power does not significantly increase the NPSH required or significantly reduce the NPSH margin. The maximum design core flow of 107.6 Mlbm/hr (105%) rated core flow is unchanged at EPU. Based on past uprate analyses, the NPSH required at full power does not significantly increase or reduce the NPSH margin because the required change in recirculation flow is small.

Therefore, the effects of EPU on NPSH meets all CLTR dispositions.

2.8.4.6.2 Flow Mismatch

The Browns Ferry recirculation loop jet pump flow mismatch TS limits do not change because these limits are based on rated core flow, which is not affected by EPU, and the flow mismatch limits are not affected because a detailed ECCS evaluation was not required for Browns Ferry at EPU conditions by the EPU LOCA evaluation.

Therefore, the effect of EPU on flow mismatch meets all CLTR dispositions.

2.8.4.6.3 Single-loop Operation

The CLTR states that increased voids in the core during normal uprated power operation requires a slight increase in the recirculation drive flow to achieve the same core flow.

Single-Loop Operation (SLO) is limited to off-rated conditions and is not affected by EPU. SLO operation at Browns Ferry is restricted to a reactor power of less than or equal to 1,729 MWt (43.75%) RTP, and core flow is limited to less than or equal to 51.25 Mlbm/hr (50%) RCF. The

power limit for SLO stays the same, requiring a proportional change in the rated percent power at the uprate power level.

Therefore, the effects of EPU on single-loop operation meet all CLTR dispositions.

2.8.5 Accident and Transient Analyses

2.8.5.1 Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Main Steam Relief or Safety Valve

Regulatory Evaluation

Excessive heat removal causes a decrease in moderator temperature which increases core reactivity and can lead to a power level increase and a decrease in shutdown margin. Any unplanned power level increase may result in fuel damage or excessive reactor system pressure. Reactor protection and safety systems are actuated to mitigate the transient.

The NRC's acceptance criteria are based on (1) GDC-10, insofar as it requires that the RCS be designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations including AOOs; (2) GDC-15, insofar as it requires that the RCS and its associated auxiliary systems be designed with margin sufficient to ensure that the design conditions of the RCPB are not exceeded during any condition of normal operation; (3) GDC-20, insofar as it requires that the reactor protection system be designed to initiate automatically the operation of appropriate systems, including the reactivity control systems, to ensure that SAFDLs are not exceeded during any condition of normal operation, including AOOs; and (4) GDC-26, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operation, including AOOs, SAFDLs are not exceeded.

Specific NRC review criteria are contained in SRP Section 15.1.1-4 and other guidance provided in Matrix 8 of RS-001.

Browns Ferry Current Licensing Basis

The General Design Criteria (GDC) listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Browns Ferry Nuclear Plant (Browns Ferry) principal design criteria predate these criteria. The Browns Ferry principal design criteria are listed in UFSAR Section 1.5, "Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, the Tennessee Valley Authority (TVA) performed a comparative evaluation of the design basis of Browns Ferry with the AEC proposed General Design Criteria of 1967. The Browns Ferry UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria set out in the July 1967 AEC release. For each group of criteria, there is a statement of TVA's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria.

Following a restatement of each of the proposed criteria is a table of references to locations in the Browns Ferry UFSAR where there is subject matter relating to the intent of that particular criteria.

While Browns Ferry is not generally licensed to the final GDC or the 1967 AEC proposed General Design Criteria, a comparison of the final GDC to the applicable AEC proposed General Design Criteria can usually be made. For the final GDC listed in the Regulatory Evaluation above, with the exception of final GDC-15, the Browns Ferry comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Browns Ferry UFSAR Appendix A: draft GDCs-6, 14, 15, and 29. There is no draft GDC directly associated with final GDC-15.

The analysis of a loss of feedwater heating transient is described in Browns Ferry UFSAR Section 14.5.3.1, "Loss of Feedwater Heater (LFWH)." The analysis of a feedwater controller failure with maximum demand is described in Browns Ferry UFSAR Section 14.5.8.1, "Feedwater Controller Failure (FWCF)." The analysis of an inadvertent opening of a Main Steam Relief Valve is described in Browns Ferry UFSAR Section 14.5.5.2, "Inadvertent Opening of an MSRV (IORV)."

Technical Evaluation

See FUSAR Section 2.8.5.1.

Conclusion

TVA has performed plant-specific reload analyses to confirm that fuel design limits and RCPB pressure limits will not be exceeded under EPU conditions. The CLTR requires that approved analytical methods be used for the EPU core reload analysis. Based on this, Browns Ferry will continue to meet the requirements of draft GDCs-6, 14, 15, and 29 following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to a decrease in reactor water temperature event.

2.8.5.2 Decrease in Heat Removal by the Secondary System

2.8.5.2.1 Loss of External Load; Turbine Trip; Loss of Condenser Vacuum; Closure of Main Steam Isolation Valve; and Steam Pressure Regulator Failure (Closed)

Regulatory Evaluation

A number of initiating events may result in unplanned decreases in heat removal by the secondary system. These events result in a sudden reduction in steam flow and, consequently, result in pressurization events. Reactor protection and safety systems are actuated to mitigate the transient.

The NRC's acceptance criteria are based on (1) GDC-10, insofar as it requires that the RCS be designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations, including AOOs; (2) GDC-15, insofar as it requires that the RCS and its associated auxiliary systems be designed with margin sufficient to ensure that the design conditions of the RCPB are not exceeded during any condition of normal operation; and (3) GDC-26, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate

of reactivity changes to ensure that under conditions of normal operation, including AOOs, SAFDLs are not exceeded.

Specific NRC review criteria are contained in SRP Section 15.2.1-5 and other guidance provided in Matrix 8 of RS-001.

Browns Ferry Current Licensing Basis

The General Design Criteria (GDC) listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Browns Ferry Nuclear Plant (Browns Ferry) principal design criteria predate these criteria. The Browns Ferry principal design criteria are listed in UFSAR Section 1.5, "Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, the Tennessee Valley Authority (TVA) performed a comparative evaluation of the design basis of Browns Ferry with the AEC proposed General Design Criteria of 1967. The Browns Ferry UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria set out in the July 1967 AEC release. For each group of criteria, there is a statement of TVA's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a table of references to locations in the Browns Ferry UFSAR where there is subject matter relating to the intent of that particular criteria.

While Browns Ferry is not generally licensed to the final GDC or the 1967 AEC proposed General Design Criteria, a comparison of the final GDC to the applicable AEC proposed General Design Criteria can usually be made. For the final GDC listed in the Regulatory Evaluation above, with the exception of final GDC-15, the Browns Ferry comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Browns Ferry UFSAR Appendix A: draft GDCs-6 and 29. There is no draft GDC directly associated with final GDC-15.

The analysis of a generator load rejection is described in Browns Ferry UFSAR Section 14.5.2.2.4 "Generator Load Reject with Turbine Bypass Valve Failure with EOC-RPT-OOS." The analysis of a turbine trip without bypass is described in Browns Ferry UFSAR Section 14.5.2.5 "Turbine Bypass Valves Failure Following Turbine Trip, High Power (TTNBP)." The analysis of an MSIV closure event is described in Browns Ferry UFSAR Section 14.5.2.7 "Main Steam Isolation Valve (MSIV) Closure." The pressure regulator downscale failure is no longer evaluated as an abnormal operating transient per Section 14.5.2.8 of the Browns Ferry UFSAR.

Technical Evaluation

See FUSAR Section 2.8.5.2.1.

Conclusion

TVA has performed plant-specific reload analyses to confirm that fuel design limits and RCPB pressure limits will not be exceeded under EPU conditions. The CLTR requires that approved

analytical methods be used for the EPU core reload analysis. Based on this, Browns Ferry will continue to meet the requirements of draft GDCs-6 and 29 following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to an increase in reactor pressure event.

2.8.5.2.2 Loss of Non-Emergency AC Power to the Station Auxiliaries

Regulatory Evaluation

The loss of non-emergency AC power is assumed to result in the loss of all power to the station auxiliaries and the simultaneous tripping of all reactor coolant circulation pumps. This causes a flow coast down as well as a decrease in heat removal by the secondary system, a turbine trip, an increase in pressure and temperature of the coolant, and a reactor trip. Reactor protection and safety systems are actuated to mitigate the transient.

The NRC's acceptance criteria are based on (1) GDC-10, insofar as it requires that the RCS be designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations, including AOOs; (2) GDC-15, insofar as it requires that the RCS and its associated auxiliary systems be designed with margin sufficient to ensure that the design conditions of the RCPB are not exceeded during any condition of normal operation; and (3) GDC-26, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operation, including AOOs, SAFDLs are not exceeded.

Specific NRC review criteria are contained in SRP Section 15.2.6 and other guidance provided in Matrix 8 of RS-001.

Browns Ferry Current Licensing Basis

The General Design Criteria (GDC) listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Browns Ferry Nuclear Plant (Browns Ferry) principal design criteria predate these criteria. The Browns Ferry principal design criteria are listed in UFSAR Section 1.5, "Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, the Tennessee Valley Authority (TVA) performed a comparative evaluation of the design basis of Browns Ferry with the AEC proposed General Design Criteria of 1967. The Browns Ferry UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria set out in the July 1967 AEC release. For each group of criteria, there is a statement of TVA's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a table of references to locations in the Browns Ferry UFSAR where there is subject matter relating to the intent of that particular criteria.

While Browns Ferry is not generally licensed to the final GDC or the 1967 AEC proposed General Design Criteria, a comparison of the final GDC to the applicable AEC proposed General Design

Criteria can usually be made. For the final GDC listed in the Regulatory Evaluation above, with the exception of final GDC-15, the Browns Ferry comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Browns Ferry UFSAR Appendix A: draft GDCs-6 and 29. There is no draft GDC directly associated with final GDC-15.

The analysis for loss of non-emergency AC power to the station auxiliaries is described in Browns Ferry UFSAR Section 14.5.5.4, "Loss of Auxiliary Power."

Technical Evaluation

See FUSAR Section 2.8.5.2.2.

Conclusion

TVA has performed plant-specific reload analyses to confirm that fuel design limits and RCPB pressure limits will not be exceeded under EPU conditions. The CLTR requires that approved analytical methods be used for the EPU core reload analysis. Based on this, Browns Ferry will continue to meet the requirements of draft GDCs-6 and 29 following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to a loss of non-emergency AC power to station auxiliaries event.

2.8.5.2.3 Loss of Normal Feedwater Flow

Regulatory Evaluation

A loss of normal feedwater flow could occur from pump failures, valve malfunctions, or a LOOP. Loss of feedwater flow results in an increase in reactor coolant temperature and pressure which eventually requires a reactor trip to prevent fuel damage. Decay heat must be transferred from fuel following a loss of normal feedwater flow. Reactor protection and safety systems are actuated to provide this function and mitigate other aspects of the transient.

The NRC's acceptance criteria are based on (1) GDC-10, insofar as it requires that the RCS be designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations, including AOOs; (2) GDC-15, insofar as it requires that the RCS and its associated auxiliary systems be designed with margin sufficient to ensure that the design conditions of the RCPB are not exceeded during any condition of normal operation; and (3) GDC-26, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operation, including AOOs, SAFDLs are not exceeded.

Specific NRC review criteria are contained in SRP Section 15.2.7 and other guidance provided in Matrix 8 of RS-001.

Browns Ferry Current Licensing Basis

The General Design Criteria (GDC) listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Browns Ferry Nuclear Plant (Browns Ferry) principal design criteria predate these criteria. The Browns Ferry principal design criteria are listed in UFSAR Section 1.5,

"Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, the Tennessee Valley Authority (TVA) performed a comparative evaluation of the design basis of Browns Ferry with the AEC proposed General Design Criteria of 1967. The Browns Ferry UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria set out in the July 1967 AEC release. For each group of criteria, there is a statement of TVA's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a table of references to locations in the Browns Ferry UFSAR where there is subject matter relating to the intent of that particular criteria.

While Browns Ferry is not generally licensed to the final GDC or the 1967 AEC proposed General Design Criteria, a comparison of the final GDC to the applicable AEC proposed General Design Criteria can usually be made. For the final GDC listed in the Regulatory Evaluation above, with the exception of final GDC-15, the Browns Ferry comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Browns Ferry UFSAR Appendix A: draft GDCs-6 and 29. There is no draft GDC directly associated with final GDC-15.

The analysis of the loss of normal feedwater flow transient is described in Browns Ferry UFSAR Section 14.5.5.3, "Loss of Feedwater Flow."

Technical Evaluation

See FUSAR Section 2.8.5.2.3.

Conclusion

TVA has evaluated the loss of normal feedwater flow event and accounted for operation of the plant at the proposed power level using acceptable analytical models. Browns Ferry is consistent with the approach described in the CLTR. TVA has performed plant-specific reload analyses to confirm that fuel design limits and RCPB pressure limits will not be exceeded under EPU conditions. Based on this, Browns Ferry will continue to meet the requirements of draft GDCs-6 and 29 following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to a loss of normal feedwater flow event.

2.8.5.3 Decrease in Reactor Coolant System Flow

2.8.5.3.1 Loss of Forced Reactor Coolant Flow

Regulatory Evaluation

A decrease in reactor coolant flow occurring while the plant is at power could result in a degradation of core heat transfer. An increase in fuel temperature and accompanying fuel damage could then result if SAFDLs are exceeded during the transient. Reactor protection and safety systems are actuated to mitigate the transient.

The NRC's acceptance criteria are based on (1) GDC-10, insofar as it requires that the RCS be designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations, including AOOs; (2) GDC-15, insofar as it requires that the RCS and its associated auxiliary systems be designed with margin sufficient to ensure that the design conditions of the RCPB are not exceeded during any condition of normal operation; and (3) GDC-26, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operation, including AOOs, SAFDLs are not exceeded.

Specific NRC review criteria are contained in SRP Section 15.3.1-2 and other guidance provided in Matrix 8 of RS-001.

Browns Ferry Current Licensing Basis

The General Design Criteria (GDC) listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Browns Ferry Nuclear Plant (Browns Ferry) principal design criteria predate these criteria. The Browns Ferry principal design criteria are listed in UFSAR Section 1.5, "Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, the Tennessee Valley Authority (TVA) performed a comparative evaluation of the design basis of Browns Ferry with the AEC proposed General Design Criteria of 1967. The Browns Ferry UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria set out in the July 1967 AEC release. For each group of criteria, there is a statement of TVA's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a table of references to locations in the Browns Ferry UFSAR where there is subject matter relating to the intent of that particular criteria.

While Browns Ferry is not generally licensed to the final GDC or the 1967 AEC proposed General Design Criteria, a comparison of the final GDC to the applicable AEC proposed General Design Criteria can usually be made. For the final GDC listed in the Regulatory Evaluation above, with the exception of final GDC-15, the Browns Ferry comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Browns Ferry UFSAR Appendix A: draft GDCs-6 and 29. There is no draft GDC directly associated with final GDC-15.

The analysis of loss of forced reactor coolant flow is described in Browns Ferry UFSAR Section 14.5.6, "Events Resulting in Core Coolant Flow Decrease."

<u>Technical Evaluation</u>

See FUSAR Section 2.8.5.3.1.

Conclusion

TVA has performed plant-specific reload analyses to confirm that fuel design limits and RCPB pressure limits will not be exceeded under EPU conditions. Based on this, Browns Ferry will continue to meet the requirements of draft GDCs-6 and 29 following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to a loss of forced reactor coolant flow event.

2.8.5.3.2 Reactor Recirculation Pump Rotor Seizure and Reactor Recirculation Pump Shaft Break

Regulatory Evaluation

The events postulated are an instantaneous seizure of the rotor or break of the shaft of a reactor recirculation pump. Flow through the affected loop is rapidly reduced, leading to a reactor and turbine trip. The sudden decrease in core coolant flow while the reactor is at power results in a degradation of core heat transfer which could result in fuel damage. The initial rate of reduction of coolant flow is greater for the rotor seizure event. However, the shaft break event permits a greater reverse flow through the affected loop later during the transient and, therefore, results in a lower core flow rate at that time. In either case, reactor protection and safety systems are actuated to mitigate the transient.

The NRC's acceptance criteria are based on (1) GDC-27, insofar as it requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to assure the capability to cool the core is maintained; (2) GDC-28, insofar as it requires that the reactivity control systems be designed to assure that the effects of postulated reactivity accidents can neither result in damage to the RCPB greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals so as to significantly impair the capability to cool the core; and (3) GDC-31, insofar as it requires that the RCPB be designed with margin sufficient to assure that, under specified conditions, it will behave in a non-brittle manner and the probability of a rapidly propagating fracture is minimized.

Specific NRC review criteria are contained in SRP Section 15.3.3-4 and other guidance provided in Matrix 8 of RS-001.

Browns Ferry Current Licensing Basis

The General Design Criteria (GDC) listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Browns Ferry Nuclear Plant (Browns Ferry) principal design criteria predate these criteria. The Browns Ferry principal design criteria are listed in UFSAR Section 1.5, "Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, the Tennessee Valley Authority (TVA) performed a comparative evaluation of the design basis of Browns Ferry with the AEC proposed General Design Criteria of 1967. The Browns Ferry UFSAR, Appendix A,

"Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria set out in the July 1967 AEC release. For each group of criteria, there is a statement of TVA's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a table of references to locations in the Browns Ferry UFSAR where there is subject matter relating to the intent of that particular criteria.

While Browns Ferry is not generally licensed to the final GDC or the 1967 AEC proposed General Design Criteria, a comparison of the final GDC to the applicable AEC proposed General Design Criteria can usually be made. For the final GDC listed in the Regulatory Evaluation above, with the exception of final GDC-27, the Browns Ferry comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Browns Ferry UFSAR Appendix A: draft GDCs-32, 33, 34, and 35. Final GDC-27 is applicable to Browns Ferry as described in "Browns Ferry Nuclear Plant, Unit 1 - Issuance of Amendments Regarding the Transition to Areva Fuel," dated July 3, 2012. (Reference 48)

The analysis of a one recirculation pump seizure accident is described in Browns Ferry UFSAR Section 14.5.6.4 "Recirculation Pump Seizure."

<u>Technical Evaluation</u>

See FUSAR Section 2.8.5.3.2.

<u>Conclusion</u>

TVA has performed plant-specific reload analyses to confirm that fuel design limits and RCPB pressure limits will not be exceeded under EPU conditions. Based on this, Browns Ferry will continue to meet the requirements of draft GDCs-32, 33, 34, and 35 and final GDC-27 following implementation of the proposed EPU. Therefore, the proposed is EPU acceptable with respect to a sudden recirculation pump rotor seizure and reactor recirculation pump shaft break event.

2.8.5.4 Reactivity and Power Distribution Anomalies

2.8.5.4.1 Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition

<u>Regulatory Evaluation</u>

An uncontrolled control rod assembly withdrawal from subcritical or low power startup conditions may be caused by a malfunction of the reactor control or rod control systems. This withdrawal will uncontrollably add positive reactivity to the reactor core, resulting in a power excursion.

The NRC's acceptance criteria are based on (1) GDC-10, insofar as it requires that the RCS be designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations, including AOOs; (2) GDC-20, insofar as it requires that the reactor protection system be designed to initiate automatically the operation of appropriate systems, including the reactivity control systems, to ensure that SAFDLs are not exceeded as a result of AOOs; and (3) GDC-25,

insofar as it requires that the protection system be designed to assure that SAFDLs are not exceeded for any single malfunction of the reactivity control systems.

Specific NRC review criteria are contained in SRP Section 15.4.1 and other guidance provided in Matrix 8 of RS-001.

Browns Ferry Current Licensing Basis

The General Design Criteria (GDC) listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Browns Ferry Nuclear Plant (Browns Ferry) principal design criteria predate these criteria. The Browns Ferry principal design criteria are listed in UFSAR Section 1.5, "Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, the Tennessee Valley Authority (TVA) performed a comparative evaluation of the design basis of Browns Ferry with the AEC proposed General Design Criteria of 1967. The Browns Ferry UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria set out in the July 1967 AEC release. For each group of criteria, there is a statement of TVA's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a table of references to locations in the Browns Ferry UFSAR where there is subject matter relating to the intent of that particular criteria.

While Browns Ferry is not generally licensed to the final GDC or the 1967 AEC proposed General Design Criteria, a comparison of the final GDC to the applicable AEC proposed General Design Criteria can usually be made. For the final GDC listed in the Regulatory Evaluation above, the Browns Ferry comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Browns Ferry UFSAR Appendix A: draft GDCs-14, 15, and 31. Final GDC-10 is applicable to Browns Ferry as described in "Browns Ferry Nuclear Plant, Unit 1 - Issuance of Amendments Regarding the Transition to Areva Fuel," dated July 3, 2012. (Reference 48)

The analysis of a rod withdrawal error transient is described in Browns Ferry UFSAR Section 14.5.4, "Events Resulting in a Positive Reactivity Insertion."

<u>Technical Evaluation</u>

See FUSAR Section 2.8.5.4.1.

<u>Conclusion</u>

TVA has evaluated the uncontrolled control rod assembly withdrawal from a subcritical or low power startup condition and accounted for the core design changes necessary for operation of the plant at the proposed power level. Browns Ferry is consistent with the approach described in the CLTR. TVA has performed plant-specific reload analyses to confirm that fuel design limits and RCPB pressure limits will not be exceeded under EPU conditions. Based on this, Browns Ferry will continue to meet the requirements of draft GDCs-14, 15, and 31 and final GDC-10 following

implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to an uncontrolled control rod assembly withdrawal from a subcritical or low power startup condition event.

2.8.5.4.2 Uncontrolled Control Rod Assembly Withdrawal at Power

Regulatory Evaluation

An uncontrolled control rod assembly withdrawal at power may be caused by a malfunction of the reactor control or rod control systems. This withdrawal will uncontrollably add positive reactivity to the reactor core, resulting in a power excursion.

The NRC's acceptance criteria are based on (1) GDC-10, insofar as it requires that the RCS be designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations, including AOOs; (2) GDC-20, insofar as it requires that the reactor protection system be designed to initiate automatically the operation of appropriate systems, including the reactivity control systems, to ensure that SAFDLs are not exceeded as a result of AOOs; and (3) GDC-25, insofar as it requires that the protection system be designed to assure that SAFDLs are not exceeded for any single malfunction of the reactivity control systems.

Specific NRC review criteria are contained in SRP Section 15.4.2 and other guidance provided in Matrix 8 of RS-001.

Browns Ferry Current Licensing Basis

The General Design Criteria (GDC) listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Browns Ferry Nuclear Plant (Browns Ferry) principal design criteria predate these criteria. The Browns Ferry principal design criteria are listed in UFSAR Section 1.5, "Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, the Tennessee Valley Authority (TVA) performed a comparative evaluation of the design basis of Browns Ferry with the AEC proposed General Design Criteria of 1967. The Browns Ferry UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria set out in the July 1967 AEC release. For each group of criteria, there is a statement of TVA's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a table of references to locations in the Browns Ferry UFSAR where there is subject matter relating to the intent of that particular criteria.

While Browns Ferry is not generally licensed to the final GDC or the 1967 AEC proposed General Design Criteria, a comparison of the final GDC to the applicable AEC proposed General Design Criteria can usually be made. For the final GDC listed in the Regulatory Evaluation above, the Browns Ferry comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Browns Ferry UFSAR Appendix A: draft GDCs-14, 15, and 31. Final GDC-10 is applicable to Browns Ferry as described in "Browns Ferry

Nuclear Plant, Unit 1 - Issuance of Amendments Regarding the Transition to Areva Fuel," dated July 3, 2012. (Reference 48)

The analysis of a rod withdrawal error transient is described in Browns Ferry UFSAR Section 14.5.4, "Events Resulting in a Positive Reactivity Insertion."

Technical Evaluation

See FUSAR Section 2.8.5.4.2.

Conclusion

TVA has performed plant-specific reload analyses to confirm that fuel design limits and RCPB pressure limits will not be exceeded under EPU conditions. Based on this, Browns Ferry will continue to meet the requirements of draft GDCs-14, 15, and 31 and final GDC-10 following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to a continuous rod withdrawal during power range operation event.

2.8.5.4.3 Startup of a Recirculation Loop at an Incorrect Temperature and Flow Controller Malfunction Causing an Increase in Core Flow Rate

Regulatory Evaluation

A startup of an inactive loop transient may result in either an increased core flow or the introduction of cooler water into the core. This event causes an increase in core reactivity due to decreased moderator temperature and core void fraction.

The NRC's acceptance criteria are based on (1) GDC-10, insofar as it requires that the RCS be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs; (2) GDC-20, insofar as it requires that the protection system be designed to initiate automatically the operation of appropriate systems to ensure that SAFDLs are not exceeded as a result of operational occurrences; (3) GDC-15, insofar as it requires that the RCS and its associated auxiliary systems be designed with margin sufficient to ensure that the design conditions of the RCPB are not exceeded during AOOs; (4) GDC-28, insofar as it requires that the reactivity control systems be designed to assure that the effects of postulated reactivity accidents can neither result in damage to the RCPB greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals so as to significantly impair the capability to cool the core; and (5) GDC-26, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operation, including AOOs, SAFDLs are not exceeded.

Specific NRC review criteria are contained in SRP Section 15.4.4-5 and other guidance provided in Matrix 8 of RS-001.

Browns Ferry Current Licensing Basis

The General Design Criteria (GDC) listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Browns Ferry Nuclear Plant (Browns Ferry) principal design criteria predate these criteria. The Browns Ferry principal design criteria are listed in UFSAR Section 1.5,

"Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, the Tennessee Valley Authority (TVA) performed a comparative evaluation of the design basis of Browns Ferry with the AEC proposed General Design Criteria of 1967. The Browns Ferry UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria set out in the July 1967 AEC release. For each group of criteria, there is a statement of TVA's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a table of references to locations in the Browns Ferry UFSAR where there is subject matter relating to the intent of that particular criteria.

While Browns Ferry is not generally licensed to the final GDC or the 1967 AEC proposed General Design Criteria, a comparison of the final GDC to the applicable AEC proposed General Design Criteria can usually be made. For the final GDC listed in the Regulatory Evaluation above, with the exception of final GDC-15, the Browns Ferry comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Browns Ferry UFSAR Appendix A: draft GDCs-14, 15, 29, and 32. There is no draft GDC directly applicable to the final GDC-15. Final GDC-10 is applicable to Browns Ferry as described in "Browns Ferry Nuclear Plant, Unit 1 - Issuance of Amendments Regarding the Transition to Areva Fuel," dated July 3, 2012. (Reference 48)

The analysis of startup of a recirculation loop at an incorrect temperature and flow controller malfunction causing an increase in core flow rate is described in Browns Ferry UFSAR Sections 14.5.7.1, "Recirculation Flow Controller Failure – Increasing Flow" and 14.5.7.2, "Startup of Idle Recirculation Loop."

Technical Evaluation

See FUSAR Section 2.8.5.4.3.

Conclusion

TVA is consistent with the approach described in the CLTR. TVA has performed plant-specific reload analyses to confirm that fuel design limits and RCPB pressure limits will not be exceeded under EPU conditions. Based on this, Browns Ferry will continue to meet the requirements of the draft GDCs-14, 15, 29, and 32 and final GDC-10 following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to startup of an idle recirculation pump or recirculation flow controller failure event.

2.8.5.4.4 Spectrum of Rod Drop Accidents

Regulatory Evaluation

The NRC's acceptance criteria are based on GDC-28, insofar as it requires that the reactivity control systems be designed to assure that the effects of postulated reactivity accidents can neither result in damage to the RCPB greater than limited local yielding, nor disturb the core, its support

structures, or other reactor vessel internals so as to significantly impair the capability to cool the core.

Specific NRC review criteria are contained in SRP Section 15.4.9 and other guidance provided in Matrix 8 of RS-001.

Browns Ferry Current Licensing Basis

The General Design Criteria (GDC) listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Browns Ferry Nuclear Plant (Browns Ferry) principal design criteria predate these criteria. The Browns Ferry principal design criteria are listed in UFSAR Section 1.5, "Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, the Tennessee Valley Authority (TVA) performed a comparative evaluation of the design basis of Browns Ferry with the AEC proposed General Design Criteria of 1967. The Browns Ferry UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria set out in the July 1967 AEC release. For each group of criteria, there is a statement of TVA's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a table of references to locations in the Browns Ferry UFSAR where there is subject matter relating to the intent of that particular criteria.

While Browns Ferry is not generally licensed to the final GDC or the 1967 AEC proposed General Design Criteria, a comparison of the final GDC to the applicable AEC proposed General Design Criteria can usually be made. For the final GDC listed in the Regulatory Evaluation above, the Browns Ferry comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Browns Ferry UFSAR Appendix A: draft GDC-32.

The analysis of a control rod drop accident is described in Browns Ferry UFSAR Section 14.6.2, "Control Rod Drop Accident (CRDA)."

<u>Technical Evaluation</u>

See FUSAR Section 2.8.5.4.4.

<u>Conclusion</u>

TVA has evaluated the CRDA and accounted for operation of the plant at the proposed power level. Browns Ferry is consistent with the approach described in the CLTR. TVA has performed plant-specific reload analyses to confirm that fuel design limits and RCPB pressure limits will not be exceeded under EPU conditions. Based on this, Browns Ferry will continue to meet the requirements of draft GDC-32 following implementation of the EPU. Therefore, the proposed EPU is acceptable with respect to a CRDA.

2.8.5.5 Inadvertent Operation of ECCS or Malfunction that Increases Reactor Coolant Inventory

Regulatory Evaluation

Equipment malfunctions; operator errors and abnormal occurrences could cause unplanned increases in reactor coolant inventory. Depending on the temperature of the injected water and the response of the automatic control systems, a power level increase may result and, without adequate controls, could lead to fuel damage or over-pressurization of the RCS. Alternatively, a power level decrease and depressurization may result. Reactor protection and safety systems are actuated to mitigate these events.

The NRC's acceptance criteria are based on (1) GDC-10, insofar as it requires that the RCS be designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations, including AOOs; (2) GDC-15, insofar as it requires that the RCS and its associated auxiliary systems be designed with margin sufficient to ensure that the design conditions of the RCPB are not exceeded during AOOs; and (3) GDC-26, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operation, including AOOs, SAFDLs are not exceeded.

Specific NRC review criteria are contained in SRP Section 15.5.1-2 and other guidance provided in Matrix 8 of RS-001.

Browns Ferry Current Licensing Basis

The General Design Criteria (GDC) listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Browns Ferry Nuclear Plant (Browns Ferry) principal design criteria predate these criteria. The Browns Ferry principal design criteria are listed in UFSAR Section 1.5, "Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, the Tennessee Valley Authority (TVA) performed a comparative evaluation of the design basis of Browns Ferry with the AEC proposed General Design Criteria of 1967. The Browns Ferry UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria set out in the July 1967 AEC release. For each group of criteria, there is a statement of TVA's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a table of references to locations in the Browns Ferry UFSAR where there is subject matter relating to the intent of that particular criteria.

While Browns Ferry is not generally licensed to the final GDC or the 1967 AEC proposed General Design Criteria, a comparison of the final GDC to the applicable AEC proposed General Design Criteria can usually be made. For the final GDC listed in the Regulatory Evaluation above, with the exception of final GDC-15, the Browns Ferry comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Browns Ferry UFSAR Appendix A: draft GDC-29. There is no draft GDC directly applicable to the final

GDC-15. Final GDC-10 is applicable to Browns Ferry as described in "Browns Ferry Nuclear Plant, Unit 1 - Issuance of Amendments Regarding the Transition to Areva Fuel," dated July 3, 2012. (Reference 48)

The analysis of an event that increases reactor coolant inventory is described in Browns Ferry UFSAR Section 14.5.8, "Events Resulting in Excess of Coolant Inventory."

Technical Evaluation

See FUSAR Section 2.8.5.5.

Conclusion

TVA is consistent with the approach described in the CLTR. TVA has performed plant-specific reload analyses to confirm that fuel design limits and RCPB pressure limits will not be exceeded under EPU conditions. Based on this, Browns Ferry will continue to meet the requirements of draft GDC-29 and final GDC-10 following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to an inadvertent operation of the ECCS or a malfunction that increases reactor coolant inventory.

2.8.5.6 Decrease in Reactor Coolant Inventory

2.8.5.6.1 Inadvertent Opening of a Pressure Relief Valve

Regulatory Evaluation

The inadvertent opening of a pressure relief valve results in a reactor coolant inventory decrease and a decrease in RCS pressure. The pressure relief valve discharges into the suppression pool. Normally there is no reactor trip. The pressure regulator senses the RCS pressure decrease and partially closes the turbine control valves (TCVs) to stabilize the reactor at a lower pressure. The reactor power settles out at nearly the initial power level. The feedwater control system maintains the coolant inventory using water from the condensate storage tank via the condenser hotwell.

The NRC's acceptance criteria are based on (1) GDC-10, insofar as it requires that the RCS be designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations, including AOOs; (2) GDC-15, insofar as it requires that the RCS and its associated auxiliary systems be designed with margin sufficient to ensure that the design conditions of the RCPB are not exceeded during AOOs; and (3) GDC-26, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operation, including AOOs, SAFDLs are not exceeded.

Specific NRC review criteria are contained in SRP Section 15.6.1 and other guidance provided in Matrix 8 of RS-001.

Browns Ferry Current Licensing Basis

The General Design Criteria (GDC) listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Browns Ferry Nuclear Plant (Browns Ferry) principal design criteria predate these criteria. The Browns Ferry principal design criteria are listed in UFSAR Section 1.5,

"Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, the Tennessee Valley Authority (TVA) performed a comparative evaluation of the design basis of Browns Ferry with the AEC proposed General Design Criteria of 1967. The Browns Ferry UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria set out in the July 1967 AEC release. For each group of criteria, there is a statement of TVA's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a table of references to locations in the Browns Ferry UFSAR where there is subject matter relating to the intent of that particular criteria.

While Browns Ferry is not generally licensed to the final GDC or the 1967 AEC proposed General Design Criteria, a comparison of the final GDC to the applicable AEC proposed General Design Criteria can usually be made. For the final GDC listed in the Regulatory Evaluation above, with the exception of final GDC-15, the Browns Ferry comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Browns Ferry UFSAR Appendix A: draft GDC-29. There is no draft GDC directly applicable to the final GDC-15. Final GDC-10 is applicable to Browns Ferry as described in "Browns Ferry Nuclear Plant, Unit 1 - Issuance of Amendments Regarding the Transition to Areva Fuel," dated July 3, 2012. (Reference 48)

The analysis of an event that results in an inadvertent opening of a pressure relief valve is described in Browns Ferry UFSAR Section 14.5.5.2, "Inadvertent Opening of a MSRV (IORV)."

Technical Evaluation

See FUSAR Section 2.8.5.6.1.

Conclusion

TVA is consistent with the approach described in the CLTR. TVA has performed plant-specific reload analyses to confirm that fuel design limits and RCPB pressure limits will not be exceeded under EPU conditions. Based on this, Browns Ferry will continue to meet the requirements of draft GDC-29 and final GDC-10 following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to an inadvertent opening of a pressure relief valve event.

2.8.5.6.2 Emergency Core Cooling System and Loss-of-Coolant Accidents

Regulatory Evaluation

LOCAs are postulated accidents that would result in the loss of reactor coolant from piping breaks in the RCPB at a rate in excess of the capability of the normal reactor coolant makeup system to replenish it. Loss of significant quantities of reactor coolant would prevent heat removal from the reactor core, unless the water is replenished. The reactor protection and ECCS systems are provided to mitigate these accidents.

The NRC's acceptance criteria are based on (1) 10 CFR 50.46, insofar as it establishes standards for the calculation of ECCS performance and acceptance criteria for that calculated performance; (2) 10 CFR Part 50, Appendix K, insofar as it establishes required and acceptable features of evaluation models for heat removal by the ECCS after the blowdown phase of a LOCA; (3) GDC-4, insofar as it requires that SSCs important to safety be protected against dynamic effects associated with flow instabilities and loads such as those resulting from water hammer; (4) GDC-27, insofar as it requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to assure the capability to cool the core is maintained; and (5) GDC-35, insofar as it requires that a system to provide abundant emergency core cooling be provided to transfer heat from the reactor core following any LOCA at a rate so that fuel clad damage that could interfere with continued effective core cooling will be prevented.

Specific NRC review criteria are contained in SRP Sections 6.3 and 15.6.5 and other guidance provided in Matrix 8 of RS-001.

Browns Ferry Current Licensing Basis

The General Design Criteria (GDC) listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Browns Ferry Nuclear Plant (Browns Ferry) principal design criteria predate these criteria. The Browns Ferry principal design criteria are listed in UFSAR Section 1.5, "Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, the Tennessee Valley Authority (TVA) performed a comparative evaluation of the design basis of Browns Ferry with the AEC proposed General Design Criteria of 1967. The Browns Ferry UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria set out in the July 1967 AEC release. For each group of criteria, there is a statement of TVA's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a table of references to locations in the Browns Ferry UFSAR where there is subject matter relating to the intent of that particular criteria.

While Browns Ferry is not generally licensed to the final GDC or the 1967 AEC proposed General Design Criteria, a comparison of the final GDC to the applicable AEC proposed General Design Criteria can usually be made. For the final GDC listed in the Regulatory Evaluation above, the Browns Ferry comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Browns Ferry UFSAR Appendix A: draft GDCs-40 and 42. Final GDCs-27 and 35 are applicable to Browns Ferry as described in "Browns Ferry Nuclear Plant, Unit 1 - Issuance of Amendments Regarding the Transition to Areva Fuel," dated July 3, 2012. (Reference 48)

The analysis of a loss-of-coolant accident is described in Browns Ferry UFSAR Section 14.6.3, "Loss of Coolant Accident (LOCA)."

Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Browns Ferry License Renewal Safety Evaluation Report (SER), NUREG-1843, dated April 2006 (Reference 11). The license renewal evaluations associated with HPCI, CS, RHR, and ADS are located in NUREG-1843 Section 2.3.2.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of EPUs. Sections 4.2.2, 4.2.3, 4.2.4, 4.2.5, and 4.3 of the CLTR address the effect of EPU on the ECCS and LOCAs. The results of this evaluation are described below.

The ECCS includes the HPCI system, the CS system, the LPCI mode of the RHR system, and the ADS.

See FUSAR Section 2.8.5.6.2 for further discussion of ECCS performance.

Each ECCS is discussed in the following sections. The effect on the functional capability of each system due to EPU is addressed. [[
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Browns Ferry meets all CLTR dispositions. The topics addressed in this evaluation are:

Topic	CLTR Disposition	Browns Ferry Result
High Pressure Coolant Injection	Generic	Meets CLTR Disposition
Core Spray	Generic	Meets CLTR Disposition
Low Pressure Coolant Injection System	Generic	Meets CLTR Disposition
Automatic Depressurization	Generic	Meets CLTR Disposition

2.8.5.6.2.1 High Pressure Coolant Injection

The CLTR states that there is no change to the normal reactor operating pressure or the SRV setpoints.

The generic disposition of HPCI in the CLTR states that the increase in decay heat changes the response of the reactor water level following a small break LOCA or a loss of FW transient event. There is no change to the normal reactor operating pressure or the SRV setpoints.

The HPCI system is designed to pump water into the reactor vessel over a wide range of operating pressures. The primary purpose of the HPCI system is to maintain reactor vessel coolant inventory in the event of a small break LOCA that does not immediately depressurize the reactor vessel. In this event, the HPCI system maintains reactor water level and helps depressurize the reactor vessel. The adequacy of the HPCI system is demonstrated in the ECCS performance discussion at the end of this section.

Consistent with the generic disposition discussed above, for EPU, there is no change to the maximum nominal reactor operating pressure of 1,050 psia, and the SRV ALs remain the same. [[

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Because the maximum normal operating pressure and the SRV setpoints do not change for EPU, the HPCI system performance requirements do not change. Therefore, the HPCI system at Browns Ferry is confirmed to be consistent with the generic description provided in the CLTR and thus no further evaluation is required.

NPSH requirements are discussed in Section 2.6.5.2

2.8.5.6.2.2 Core Spray

The CLTR states that there is no change in the reactor pressures at which the CS function is required.

The CS system is automatically initiated in the event of a LOCA. When operating in conjunction with other ECCS, the CS system is required to provide adequate core cooling for all LOCA events. There is no change in the reactor pressures at which the CS system is required.

The CS system sprays water into the reactor vessel after it is depressurized. The primary purpose of the CS system is to provide reactor vessel coolant inventory makeup for a large break LOCA and for any small break LOCA after the reactor vessel has depressurized. It also provides long-term core cooling in the event of a LOCA. The CS system meets all applicable safety criteria for EPU.

The slight change in the system operating condition, such as peak suppression pool temperature and pressure, due to EPU for a postulated LOCA does not affect the hardware capabilities of the CS system. Core spray distribution is not directly credited in the short-term cooling LOCA analyses. This is consistent with ECCS evaluation models specified in Appendix K to 10 CFR 50. Therefore, the convective heat transfer coefficients used during the short-term spray cooling period are the conservative values specified in Appendix K.

The CS system at Browns Ferry meets all CLTR dispositions because the system functions are not changed and the core cooling capacity is adequate.

2.8.5.6.2.3 Low Pressure Coolant Injection

The CLTR states that there is no change in the reactor pressures at which the LPCI mode of RHR is required.

The LPCI mode of the RHR system is automatically initiated in the event of a LOCA. The primary purpose of the LPCI mode is to help maintain reactor vessel coolant inventory for a large break LOCA and for any small break LOCA after the reactor vessel has depressurized. The LPCI operating requirements are not affected by EPU, and the ECCS performance evaluation demonstrates the adequacy of the LPCI mode core cooling performance.

The LPCI mode at Browns Ferry meets all CLTR dispositions.

2.8.5.6.2.4 Automatic Depressurization System

The CLTR states that EPU does not change the conditions at which the ADS must function.

The ADS uses SRVs to reduce the reactor pressure following a small break LOCA when it is assumed that the high-pressure systems have failed. This allows the CS and LPCI to inject coolant into the reactor vessel. EPU does not change the conditions at which the ADS must function. The ADS initiation logic and valve control are not affected by EPU conditions.

The adequacy of the ADS is demonstrated by the performance evaluation discussed in Section 2.8.5.6.2.5. The ADS at Browns Ferry meets all CLTR dispositions because the SRV setpoints and functions remain the same, the ADS timers are not changed and the small break LOCA event mitigation is acceptable.

2.8.5.6.2.5 Emergency Core Cooling System Performance

See FUSAR Section 2.8.5.6.2.5.

Conclusion

TVA has evaluated the LOCA events and the ECCS. The evaluation concludes that operation of Browns Ferry at the proposed power level is acceptable In addition, TVA has performed cycle specific reload analyses to confirm that the peak cladding temperature, total oxidation of the cladding, total hydrogen generation, and changes in core geometry and long-term cooling will remain within acceptable limits. Based on this, the evaluation concludes that Browns Ferry will continue to meet the requirements of draft GDCs-40 and 42, final GDCs-27 and 35, and 10 CFR 50.46 following implementation of the proposed EPU, and is, therefore, acceptable.

2.8.5.7 Anticipated Transients Without Scram

Regulatory Evaluation

ATWS is defined as an AOO followed by the failure of the reactor portion of the protection system specified in GDC-20. The regulation at 10 CFR 50.62 requires that:

- each BWR have an ARI system that is designed to perform its function in a reliable manner and be independent (from the existing reactor trip system) from sensor output to the final actuation device.

- each BWR have a standby liquid control system (SLCS) with the capability of injecting into the reactor vessel a borated water solution with reactivity control at least equivalent to the control obtained by injecting 86 gpm of a 13 weight-percent sodium pentaborate decahydrate solution at the natural boron-10 isotope abundance into a 251-inch inside diameter reactor vessel. The system initiation must be automatic.
- each BWR have equipment to trip the reactor coolant recirculation pumps automatically under conditions indicative of an ATWS.

NRC review guidance is provided in Matrix 8 of RS-001.

Browns Ferry Current Licensing Basis

The analysis of anticipated transients without scram is described in Browns Ferry UFSAR Section 7.19, “Anticipated Transient without Scram.”

Technical Evaluation

NEDC-33004P-A, Revision 4, “Constant Pressure Power Uprate,” Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of CPPUs. Section 9.3.1 of the CLTR addresses the effect of CPPUs on ATWS.

Analysis of ATWS events is required for CLTP and for EPU RTP to ensure that the following ATWS acceptance criteria are met:

- Maintain containment integrity (i.e., maximum containment pressure and temperature less than the design pressure (56 psig) and temperature (281°F) of the containment structure).
- Maintain reactor vessel integrity (i.e., peak vessel bottom pressure less than the ASME Service Level C limit of 1,500 psig).
- Maintain coolable core geometry (Coolable core geometry is assured by meeting the 2,200°F peak cladding temperature and the 17% local cladding oxidation acceptance criteria of 10 CFR 50.46).

This evaluation reviewed the results of the ATWS analyses considering the limiting cases for RPV overpressure and for suppression pool temperature / containment pressure. Previous evaluations considered four ATWS events: MSIVC, Pressure Regulator Failure – Open (PRFO), LOOP, and Inadvertent Opening of a Relief Valve (IORV). Consistent with the event selection disposition contained in Section L.3.3 of Reference 4 (ELTR1), these four events are analyzed for the containment system response (suppression pool temperature and containment pressure) from ATWS. The results are presented below in Sections 2.8.5.7.1 through 2.8.5.7.3.

The EPU ATWS analyses for the containment response are performed using the NRC approved code ODYN, to determine the heat addition to the suppression pool from MSRV flow, and STEMP, to determine the suppression pool heatup due to energy input from the MSRVs (see Table 1-1).

The ODYN code was previously used in the ATWS analysis for Browns Ferry Unit 1 at a CLTP of 3,458 MWt (Reference 89). The GEH REDY code was used in the ATWS analysis in support of the Browns Ferry Unit 2 and 3 transitions to the currently licensed MELLLA operating domain

(Reference 106) and in support of the Browns Ferry Unit 2 and 3 CLTP of 3,458 MWt (Reference 84). The ODYN code has been used for BWR EPU ATWS analyses since the approval of References 3 and 4.

The STEMP code was previously used for the analysis of ATWS containment response in support of the Browns Ferry Unit 2 and 3 transitions to the currently licensed MELLLA operating domain (Reference 106) and in support of the Browns Ferry Unit 2 and 3 CLTP of 3,458 MWt (Reference 84). The GEH SHEX code was used for the ATWS containment analysis for Browns Ferry Unit 1 at a CLTP of 3,458 MWt (Reference 89). The SHEX code was used for Browns Ferry Unit 1 to respond to NRC questions concerning containment accident pressure, which could not be modeled by the STEMP code. Because containment accident pressure is not required for EPU, the EPU ATWS containment analyses utilize the STEMP code for all three Browns Ferry units.

STEMP calculates the temperature rise of the suppression pool due to MSRV discharge. [[

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The key inputs to the ATWS analysis for the containment response are provided in Table 2.8-1. The results of the analysis are provided in Table 2.8-2 and discussed below.

The results of the ATWS analysis meet the above ATWS acceptance criteria for the containment response. Therefore, the Browns Ferry containment response to an ATWS event at EPU is acceptable. The potential for thermal-hydraulic instability in conjunction with ATWS events is evaluated in FUSAR Section 2.8.3.2.

Browns Ferry meets the ATWS mitigation requirements defined in 10 CFR 50.62: installation of an ARI system; SLCS boron injection equivalent to 86 gpm of 13 weight percent natural boron; and installation of automatic RPT logic (i.e., ATWS-RPT). The plant-specific ATWS analysis takes credit for the ATWS-RPT and SLCS. However, ARI is not credited.

The 86 gpm boron injection equivalency requirement of 10 CFR 50.62 is satisfied via the following relationship:

$$(Q/86) \times (M251/M) \times (C/13) \times (E/19.8) \geq 1$$

where:

Q = Expected Standby Liquid Control System (SLCS) flow rate (gpm)

M251/M = Mass of water in a 251-inch diameter reactor vessel and recirculation system (lbs) / mass of water in the reactor vessel and recirculation system at hot rated condition (lbs)

C = Sodium pentaborate solution concentration (weight percent)

E = Boron-10 isotope enrichment (atom-percent)

For Browns Ferry at EPU conditions,

Q = 50.0 gpm

M251/M = 1 (each Browns Ferry unit has a 251-inch diameter reactor vessel)

C = 8.7 %

E = 94.0 %

Therefore, the 86 gpm equivalency requirement is satisfied as follows:

$$(Q/86) \times (M251/M) \times (C/13) \times (E/19.8) \geq 1$$

$$(50.0/86) \times (1) \times (8.7/13) \times (94.0/19.8) = 1.847 \geq 1$$

There are no new operator actions and no changes to the currently assumed operator actions for the EPU ATWS analysis.

2.8.5.7.1 ATWS (Overpressure)

The Browns Ferry ATWS RPV overpressure evaluation is presented in FUSAR Section 2.8.5.7.1.

The MSIVC event produces the highest peak lower plenum pressure at the time of SLCS initiation (1,216 psia). This lower plenum pressure result is evaluated in Section 2.8.4.5.2 for effects on SLCS.

2.8.5.7.2 ATWS (Suppression Pool Temperature)

The higher power and decay heat at EPU RTP will result in higher suppression pool temperatures. The increased core power and reactor steam flow rates, in conjunction with the MSRV capacity and response times, could affect the capability of the SLCS to mitigate the consequences of an ATWS event.

The suppression pool temperature evaluation includes a review of the results of the analyses of ATWS events to identify the most limiting containment response. Four events, MSIVC, PRFO, LOOP and IORV, were further analyzed for Browns Ferry. The ATWS event selection for Browns Ferry EPU is consistent with the four events specified in Section L.3.3 of Reference 4.

The key inputs and limiting results for the containment response to ATWS events are presented in Tables 2.8-1 and 2.8-2. The MSIVC, PRFO, LOOP and IORV sequence of events are given in Tables 2.8-3 through 2.8-6, respectively. The transient responses to these events are presented in Figures 2.8-1 through 2.8-18. The limiting ATWS event with respect to containment response for Browns Ferry is LOOP. The peak suppression pool temperature and containment pressure results are well below the containment design temperature and pressure. Therefore, the Browns Ferry EPU ATWS analysis for the containment response complies with the acceptance criteria of 10 CFR 50.62.

2.8.5.7.3 ATWS (Peak Cladding Temperature)

The Browns Ferry ATWS peak cladding temperature evaluation is presented in FUSAR Section 2.8.5.7.3.

Conclusion

TVA has evaluated ATWS and accounted for the effects of the proposed EPU on ATWS. The evaluation confirmed that ARI, SLCS, and recirculating pump trip systems will continue to meet the requirements of 10 CFR 50.62. Therefore, the proposed EPU is acceptable with respect to ATWS.

2.8.6 Fuel Storage

2.8.6.1 New Fuel Storage

Regulatory Evaluation

Nuclear reactor plants include facilities for the storage of new fuel. The quantity of new fuel to be stored varies from plant to plant, depending upon the specific design of the plant and the individual refueling needs.

The NRC's acceptance criteria are based on GDC-62, insofar as it requires the prevention of criticality in fuel storage systems by physical systems or processes, preferably utilizing geometrically safe configurations.

Specific NRC review criteria are contained in SRP Sections 9.1.1 and 9.1.2.

Browns Ferry Current Licensing Basis

The General Design Criteria (GDC) listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Browns Ferry Nuclear Plant (Browns Ferry) principal design criteria predate these criteria. The Browns Ferry principal design criteria are listed in UFSAR Section 1.5, "Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, the Tennessee Valley Authority (TVA) performed a comparative evaluation of the design basis of Browns Ferry with the AEC proposed General Design Criteria of 1967. The Browns Ferry UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria set out in the July 1967 AEC release. For each group of criteria, there is a statement of TVA's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a table of references to locations in the Browns Ferry UFSAR where there is subject matter relating to the intent of that particular criteria.

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

Criteria (referred to here as "draft GDC") is contained in Browns Ferry UFSAR Appendix A: draft GDC-66.

New Fuel Storage is described in Browns Ferry UFSAR Section 10.2, "New Fuel Storage."

Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Browns Ferry License Renewal Safety Evaluation Report (SER), NUREG-1843, dated April 2006 (Reference 11). The license renewal evaluation associated with the fuel storage is documented in NUREG-1843, Section 2.3.3.27. Management of aging effects on fuel storage is documented in NUREG-1843, Section 3.3.2.

Technical Evaluation

The new fuel storage facility (also referred to as the new fuel storage vault) at Browns Ferry is not used because new fuel is placed directly into the spent fuel storage pool following receipt inspection. Consequently, the effect of EPU on the new fuel storage facility has not been evaluated.

Conclusion

Not applicable.

2.8.6.2 Spent Fuel Storage

Regulatory Evaluation

Nuclear reactor plants include storage facilities for the wet storage of spent fuel assemblies. The safety function of the spent fuel pool and storage racks is to maintain the spent fuel assemblies in a safe and subcritical array during all credible storage conditions and to provide a safe means of loading the assemblies into shipping casks.

The NRC's acceptance criteria are based on (1) GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents; and (2) GDC-62, insofar as it requires that criticality in the fuel storage systems be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

Specific NRC review criteria are contained in SRP Sections 9.1.1 and 9.1.2.

Browns Ferry Current Licensing Basis

The General Design Criteria (GDC) listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Browns Ferry Nuclear Plant (Browns Ferry) principal design criteria predate these criteria. The Browns Ferry principal design criteria are listed in UFSAR Section 1.5, "Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, the Tennessee Valley Authority (TVA) performed a comparative evaluation of the design basis of Browns Ferry with the AEC proposed General Design Criteria of 1967. The Browns Ferry UFSAR, Appendix A,

"Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria set out in the July 1967 AEC release. For each group of criteria, there is a statement of TVA's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a table of references to locations in the Browns Ferry UFSAR where there is subject matter relating to the intent of that particular criteria.

While Browns Ferry is not generally licensed to the final GDC or the 1967 AEC proposed General Design Criteria, a comparison of the final GDC to the applicable AEC proposed General Design Criteria can usually be made. For the final GDC listed in the Regulatory Evaluation above, the Browns Ferry comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Browns Ferry UFSAR Appendix A: draft GDCs-40 and 66.

Spent Fuel Storage is described in Browns Ferry UFSAR Section 10.3, "Spent Fuel Storage."

Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Browns Ferry License Renewal Safety Evaluation Report (SER), NUREG-1843, dated April 2006 (Reference 11). The license renewal evaluation associated with the fuel storage is documented in NUREG-1843, Section 2.3.3.27. Management of aging effects on the fuel storage is documented in NUREG-1843, Section 3.3.2.

Technical Evaluation

See FUSAR Section 2.8.6.2.

Conclusion

TVA has evaluated the effects of the proposed EPU on the spent fuel storage capability and accounted for the effects of the proposed EPU. The evaluation concludes that the spent fuel pool design will continue to ensure an acceptably low temperature and an acceptable degree of subcriticality following implementation of the proposed EPU. Based on this, TVA concludes that the Browns Ferry spent fuel storage facilities will continue to meet the requirements of draft GDCs-40 and 66 following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to spent fuel storage.

Table 2.8-1 Browns Ferry Key Inputs for EPU ATWS Analysis

Input Variable	CLTP	EPU	Basis
Reactor Power (MWt)	3,458	3,952	Rated Value
Analyzed Power (MWt)	3,293	3,952	Bounding initial condition at EPU. OLTP value used maximizes the effect of EPU.(1)
Analyzed Core Flow (Mlbm/hr / % Rated)	76.9/ 75	101.475/ 99	Bounding initial condition on Maximum Extended Load Line Limit (MELLL) upper boundary. Results in highest power level after recirculation pump trip.
Reactor Dome Pressure (psig)	1,035	1,035	Rated Value
MSIV Closure Time (seconds)	4.0	4.0	Nominal Value
High Pressure ATWS-RPT Setpoint (psig)	1,177.0	1,177.0	Analytical Limit more conservative than Technical Specification AV of 1,175 psig.
MSL Low Pressure Isolation Setpoint (psig)	825	825	Technical Specification AV
RCIC Flow Rate (gpm)	540	600	Technical Specification value for EPU. Nominal Value for CLTP.
HPCI Flow Rate (gpm)	4,500	5,000	Technical Specification value for EPU. Nominal Value for CLTP.
Number of MSRVs / MSRVs OOS	13 / 1	13 / 1	Plant Configuration. Design Unchanged
Number of MSRVs OOS	1	1	Plant Configuration. Design Unchanged
Each MSRV Capacity at 1,090 psig (Mlbm/hr)	0.87	0.87	Plant Configuration. Design Unchanged
SRV Analytical Opening Setpoints (psig)	1,179 to 1,199	1,174 to 1,194	Note 2
SLCS Injection Location	Lower Plenum	Lower Plenum	Plant Configuration. Design Unchanged
SLCS Injection Rate (gpm)	39.0	50.0	Note 3
Number of SLCS Pumps Credited for Injection	1	1	Plant Configuration. Design Unchanged
Boron-10 Enrichment (Atom %)	68.1	94.0	Note 4

Table 2.8-1 Browns Ferry Key Inputs for EPU ATWS Analysis (continued)

Input Variable	CLTP	EPU	Basis
Sodium Pentaborate Concentration (% by Weight)	8.0	8.7	Note 5
SLCS Liquid Transport Time (seconds)	60.0	28.5	Note 6
Initial Suppression Pool Liquid Volume (ft^3)	123,000	122,940	Nominal value. EPU value conservative.
Initial Suppression Pool Temperature (°F)	95	95	Technical Specification
RHR Heat Exchanger Effectiveness Per Loop (BTU/sec-°F)	223	277	Note 7
Number of RHR Suppression Pool Cooling Loops (All Events Except Loss of Offsite Power Event)	2	4	Plant Configuration. Note 8
Number of RHR Suppression Pool Cooling Loops During a Loss of Offsite Power Event	2	2	Plant Configuration. Note 9
RHR Startup Delay (seconds after T = 0)	660	660	Note 10
RHR Service Water Temperature (°F)	95	95	Technical Specification
Decay Heat Correlation	May-Witt	May-Witt	Note 11
Steam Extraction Points for Feedwater Heaters	Note 12	Note 12	Plant Configuration
Main Turbine Bypass Valve Capacity (Mlbm/hr)	3.5	3.5	Plant Configuration

Notes :

(1) To maximize the effect of EPU, a baseline is established at the OLTP level, assuming the current licensed equipment performance assumptions and plant parameters.

(2) In the ODYN analysis methodology, the MSRV setpoints for the ATWS analysis are statistically spread around the upper analytical limit. The EPU values are consistent with the values used for the RPV ATWS analysis contained in the FUSAR.

(3) The CLTP analysis used the current Browns Ferry Technical Specification value for SLCS flow rate. The EPU analysis uses a nominal SLCS pump flow rate. The EPU value is more conservative (lower) than the plant action level for SLCS testing specified in Browns Ferry Procedure SR 3.1.7.7 - Standby Liquid Control System Functional Test.

(4) As part of the Browns Ferry EPU implementation, TVA will perform a plant modification to increase the SLCS storage tank B-10 enrichment to 96 atom-%. 94 atom-% enrichment is credited in the EPU analysis.

(5) The CLTP analysis used the Technical Specification minimum concentration value. The EPU analysis uses a nominal concentration value that remains below the maximum allowed concentration stated in Browns Ferry Technical Specification SR 3.1.7.4.

(6) The CLTP analysis used a conservative generic transport time. The EPU analysis used a Browns Ferry plant specific value that considers the actual pipe lengths from the SLCS storage tank to the RPV lower plenum, the SLCS pipe internal diameters, and the EPU SLCS flow rate of 50 gpm. The calculated transport time was multiplied by a factor of 1.2 for additional conservatism.

(7) The EPU heat exchanger effectiveness assumes a RHR flow rate of 6,500 gpm and RHR SW flow rate of 4,000 gpm through each in-service RHR heat exchanger for events that assume operation of 4 RHR loops (see Note 8 below). The heat exchange fouling resistance, used to calculate the RHR heat exchanger effectiveness value used in the ATWS analysis, is a nominal value that is supported by Browns Ferry plant specific RHR heat exchanger testing. Details concerning the fouling resistance used and the determination of the RHR heat exchanger K-value for the ATWS analysis are presented in Browns Ferry EPU LAR Attachment 39.

The EPU heat exchanger effectiveness assumes a RHR flow rate of 6500 gpm and RHR SW flow rate of 4500 gpm through each in-service RHR heat exchanger for the event that assumes operation of 2 RHR loops (see Note 9 below). The heat exchange fouling resistance, used to calculate the RHR heat exchanger effectiveness value used in the ATWS analysis, is a conservative value that is supported by Browns Ferry plant specific RHR heat exchanger testing. Details concerning the fouling resistance used and the determination of the RHR heat exchanger K-value for the ATWS analysis are presented in the Browns Ferry EPU LAR Attachment 39.

(8) The RHR suppression pool cooling configuration does not change for EPU. An RHR loop is defined as one RHR pump, one RHR heat exchanger and RHR SW flow of 4,000 gpm through the RHR heat exchanger. For ATWS events other than LOOP, the plant operators would be directed by plant EOIs to maximize suppression pool cooling. Because there is no concurrent event on the non-ATWS unit, four RHR loops are assumed available for suppression pool cooling in the ATWS unit.

(9) The RHR suppression pool cooling configuration does not change for EPU. An RHR loop is defined as one RHR pump, one RHR heat exchanger and RHR SW flow of 4,500 gpm through the RHR heat exchanger. For the LOOP ATWS event, operators will be directed by EOIs to maximize suppression cooling. Because there is also a LOOP (without ATWS) on the remaining two Browns Ferry units, only two RHR loops are assumed available for suppression pool cooling on the ATWS unit.

(10) The RHR startup delay time assumes no operator action for containment cooling for the first 10 minutes of the event with an additional 60 seconds for RHR to reach full effectiveness.

(11) The May-Witt decay heat correlation is used in the suppression pool temperature calculation following reactor shutdown. The May-Witt decay heat correlation yields a conservative pool heat-up compared to the 1979 ANS 5.1 + 2σ curve.

(12) The steam extraction points for feedwater heaters are downstream of the MSIVs, such that FW heating is lost following MSIV isolation. The specific extraction points are as follows:

- HP turbine exhaust to FW heater number 1 (highest pressure FW heater)
- LP turbine stage 7 to FW heater number 2
- LP turbine stage 8 to FW heater number 3
- LP turbine stage 10 to FW heater number 4
- LP turbine stage 12 to FW heater number 5 (lowest pressure FW heater)

Table 2.8-2 Browns Ferry Containment Results for ATWS Analysis

MSIVC Event

Acceptance Criteria	Acceptance Criteria	EPU Result
Peak Suppression Pool Temperature (°F)	281.0	171.7
Peak Containment Pressure (psig)	56.0	8.0

PRFO Event

Acceptance Criteria	Acceptance Criteria	EPU Result
Peak Suppression Pool Temperature (°F)	281.0	171.6
Peak Containment Pressure (psig)	56.0	8.0

LOOP Event

Acceptance Criteria	Acceptance Criteria	EPU Result
Peak Suppression Pool Temperature (°F)	281.0	173.3
Peak Containment Pressure (psig)	56.0	8.7

IORV Event

Acceptance Criteria	Acceptance Criteria	EPU Result
Peak Suppression Pool Temperature (°F)	281.0	142
Peak Containment Pressure (psig)	56.0	4.0

Table 2.8-3 MSIVC Sequence of Events

Event Response	EPU BOC Event Time (sec)	EPU EOC Event Time (sec)
MSIV Isolation Initiated	0.0	0.0
High Pressure ATWS Setpoint	4.0	4.0
MSIVs Fully Closed	4.0	4.0
Peak Neutron Flux	4.0	4.0
Opening of the First Relief Valve	3.9	3.8
Recirculation Pumps Trip	4.5	4.5
Peak Heat Flux	4.8	4.7
Peak Vessel Pressure	10.5	10.4
Feedwater Reduction Initiated	30.0	30.0
BIIT Reached	38.0	38.0
SLCS Pumps Start (1)	124.0	124.0
RHR Cooling Initiated	660	660
RPV Water Level Increased after Hot Shutdown Boron Weight Injected	809-1,009	809-1,009
Peak Suppression Pool Temperature	874	868
Hot Shutdown Achieved (Neutron Flux Below 0.1% for More Than 100 seconds)	1,050	1,022

Note:

1. SLCS injection is the later time of either: 1) two minutes after the high-pressure recirculation pump trip or 2) when the suppression pool temperature reaches the Boron Injection Initiation Temperature (BIIT). For Browns Ferry, there is no automatic actuation of the SLCS. Actuation of the SLCS occurs by operator manipulation of key-lock switches on the main control room front panel.

Table 2.8-4 PRFO Sequence of Events

Event Response	EPU BOC Event Time (sec)	EPU EOC Event Time (sec)
Turbine Control Valves (TCV) and Bypass Valves Start Open	0.1	0.1
MSIV Closure Initiated by Low Steam Line Pressure	15.7	14.9
MSIVs Fully Closed	19.7	18.9
Peak Neutron Flux	21.8	19.4
Opening of the First Relief Valve	21.6	20.8
High Pressure ATWS Setpoint	21.8	21.0
Recirculation Pumps Trip	22.2	21.6
Peak Heat Flux	22.5	21.7
Peak Vessel Pressure	28.6	27.7
Feedwater Reduction Initiated	46.0	46.0
BIIT Reached	56.0	55.0
SLCS Pumps Start (1)	141.8	141.0
RHR Cooling Initiated	660	660
RPV Water Level Increased after Hot Shutdown Boron Weight Injected	827-1,027	827-1,027
Peak Suppression Pool Temperature	889	889
Hot Shutdown Achieved (Neutron Flux Below 0.1% for More Than 100 seconds)	1,077	1,067

Note:

1. SLCS injection is the later time of either 1) two minutes after the high-pressure recirculation pump trip or 2) when the suppression pool temperature reaches the BIIT. For Browns Ferry, there is no automatic actuation of the SLCS. Actuation of the SLCS occurs by operator manipulation of key-lock switches on the main control room front panel.

Table 2.8-5 LOOP Sequence of Events

Event Response	EPU EOC Event Time (sec)
Main Turbine Generator Tripped	0.0
Recirculation Pumps Trip	0.0
Feedwater Pump Coastdown Initiated due to Tripping on LOOP of Motor Driven Condensate and Condensate Booster Pumps	0.0
Peak Neutron Flux	0.5
Peak Heat Flux	0.6
Opening of the First Relief Valve	0.9
High Pressure ATWS Setpoint	1.1
MSIV Isolation Initiates	2.0
MSIVs Fully Closed	6.0
Peak Vessel Pressure	7.1
BIIT Reached	40.0
SLCS Pumps Start (1)	121.0
RHR Cooling Initiated	660
RPV Water Level Increased after Hot Shutdown Boron Weight Injected	806-1,006
Hot Shutdown Achieved (Neutron Flux Below 0.1% for More Than 100 seconds)	1,236
Peak Suppression Pool Temperature	9,192

Note:

1. SLCS injection is the later time of either: 1) two minutes after the high-pressure recirculation pump trip or 2) when the suppression pool temperature reaches the BIIT. For Browns Ferry, there is no automatic actuation of the SLCS. Actuation of the SLCS occurs by operator manipulation of key-lock switches on the main control room front panel.

Table 2.8-6 IORV Sequence of Events

Event Response	EPU EOC Event Time (sec)
Inadvertent Opening of One Relief Valve	0.0
Peak Neutron Flux	0.0
Peak Vessel Pressure	0.0
Peak Heat Flux	0.2
BIIT Reached	434
Recirculation Pumps Tripped	434
Feedwater Reduction Initiated	434
SLCS Pumps Start (1)	434
RHR Cooling Initiated	660
RPV Water Level Increased after Hot Shutdown Boron Weight Injected	1,119-1,319
Hot Shutdown Achieved (Neutron Flux Below 0.1% for More Than 100 seconds)	1,273
Peak Suppression Pool Temperature	5,379

Note:

1. SLCS injection is the later time of either: 1) two minutes after the high-pressure recirculation pump trip or 2) when the suppression pool temperature reaches the BIIT. For Browns Ferry, there is no automatic actuation of the SLCS. Actuation of the SLCS occurs by operator manipulation of key-lock switches on the main control room front panel.

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Figure 2.8-1 EPU MELLLA BOC MSIVC

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Figure 2.8-2 EPU MELLLA BOC MSIVC

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Figure 2.8-3 EPU MELLLA BOC MSIVC

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Figure 2.8-4 EPU MELLLA BOC PRFO

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Figure 2.8-5 EPU MELLLA BOC PRFO

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Figure 2.8-6 EPU MELLLA BOC PRFO

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Figure 2.8-7 EPU MELLLA EOC MSIVC

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Figure 2.8-8 EPU MELLLA EOC MSIVC

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Figure 2.8-9 EPU MELLLA EOC MSIVC

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Figure 2.8-10 EPU MELLLA EOC PRFO

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Figure 2.8-11 EPU MELLLA EOC PRFO

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Figure 2.8-12 EPU MELLLA EOC PRFO

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Figure 2.8-13 EPU MELLLA EOC LOOP

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Figure 2.8-14 EPU MELLLA EOC LOOP

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Figure 2.8-15 EPU MELLLA EOC LOOP

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Figure 2.8-16 EPU MELLLA EOC IORV

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Figure 2.8-17 EPU MELLLA EOC IORV

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Figure 2.8-18 EPU MELLLA EOC IORV

2.9 Source Terms and Radiological Consequences Analyses

2.9.1 Source Terms for Radwaste Systems Analyses

Regulatory Evaluation

TVA reviewed the radioactive source term associated with EPUs to ensure the adequacy of the sources of radioactivity used by TVA as input to calculations to verify that the radioactive waste management systems have adequate capacity for the treatment of radioactive liquid and gaseous wastes.

The NRC's acceptance criteria for source terms are based on (1) 10 CFR Part 20, insofar as it establishes requirements for radioactivity in liquid and gaseous effluents released to unrestricted areas; (2) 10 CFR Part 50 Appendix I, insofar as it establishes numerical guides for design objectives and limiting conditions for operation to meet the "as low as is reasonably achievable" criterion; and (3) GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents.

Specific NRC review criteria are contained in SRP Section 11.1.

Browns Ferry Current Licensing Basis

The General Design Criteria (GDC) listed in RS-001 are those currently specified in 10 CFR 50, Appendix A. The applicable Browns Ferry Nuclear Plant (Browns Ferry) principal design criteria predate these criteria. The Browns Ferry principal design criteria are listed in UFSAR Section 1.5, "Principal Design Criteria." In 1967, the AEC published for public comment a revised set of proposed General Design Criteria (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, the Tennessee Valley Authority (TVA) performed a comparative evaluation of the design basis of Browns Ferry with the AEC proposed General Design Criteria of 1967. The Browns Ferry UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria set out in the July 1967 AEC release. For each group of criteria, there is a statement of TVA's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a table of references to locations in the Browns Ferry UFSAR where there is subject matter relating to the intent of that particular criteria.

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

The radioactive waste systems are described in Browns Ferry UFSAR Chapter 9, "Radioactive Waste Control Systems."

Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Browns Ferry License Renewal Safety Evaluation Report (SER), NUREG-1843, dated April 2006 (Reference 11). The license renewal evaluation associated with the solid and liquid radioactive waste systems are documented in NUREG-1843, Section 2.3.3.25. Management of aging effects on the solid and liquid radioactive waste systems is documented in NUREG-1843, Section 3.3.2.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of EPUs. Sections 8.3 and 8.4 of the CLTR address the effect of EPU on the Radiation Sources in the Reactor Core and in the Reactor Coolant. The results of this evaluation are described below.

2.9.1.1 Radiation Sources in the Reactor Core

See FUSAR Section 2.9.1.1.

2.9.1.2 Radiation Sources in Reactor Coolant

For coolant activation products, the typical margin in the plant design basis for reactor coolant concentrations significantly exceeds the potential increases due to power uprate and needs to be verified. Also, because the transport time from core exit to downstream points will decrease with increased flow from EPU, the resultant dose rates in the MSLs, turbines, and condenser area will increase roughly proportional to power uprate. In the case of activated corrosion products and fission products, plant-specific analysis is required by the CLTR to verify that the corrosion product concentrations do not exceed the design basis concentrations.

Tables 2.9-1 through 2.9-5 contain the activity levels, concentrations, and release rates for these radiation sources for Browns Ferry. Fission and activation product concentrations for a moisture carryover fraction of 0.1 wt% are reported for EPU conditions. The topics addressed in this evaluation are:

Topic	CLTR Disposition	Browns Ferry Result
Coolant Activation Products	Plant-Specific	Table 2.9-2
Activated Corrosion Products and Fission Products	Plant-Specific	Tables 2.9-1 through 2.9-5

2.9.1.2.1 Coolant Activation Products

The CLTR, Section 8.4.1, states that increases in reactor power will increase the activity of activation products found in reactor coolant. During reactor operation, the coolant passing through the core region becomes radioactive as a result of nuclear reactions. The coolant activation, especially N^{-16} activity, is the dominant source in the turbine building and in the lower regions of the drywell. The activation of the water in the core region is in approximate proportion to the increase in thermal power. [[

]] The margin in the Browns Ferry plant design basis for reactor coolant activation concentrations significantly exceeds potential increases due to EPU. Therefore, no change is required in the activation design basis reactor coolant concentrations for EPU and all CLTR dispositions are met for coolant activation products.

2.9.1.2.2 Activated Corrosion Products and Fission Products

The CLTR, Section 8.4.1, states that increases in reactor power will increase the activity of corrosion products and fission products found in reactor coolant. The reactor coolant contains activated corrosion products, which are the result of metallic materials entering the water and being activated in the reactor region. Under EPU conditions, the FW flow increases with power and the activation rate in the reactor region increases with power. The net result is an increase in the activated corrosion product present in the coolant.

Fission products in the reactor coolant are separable into the products in the steam and the products in the reactor water. The activity in the steam consists of noble gases released from the core plus carryover activity from the reactor water. This activity is the noble gas offgas that is included in the plant design. The calculated offgas concentrations at 30 minutes of decay for EPU are 3.6E+04 μCi/sec, within the original design basis of 3.5E+05 μCi/sec, per Table 2.9-4. Therefore, no change is required in the design basis for offgas activity for EPU.

The fission product activity in the reactor water, like the activity in the steam, is the result of minute releases from the fuel rods. The isotopes used for the comparison of design basis reactor water concentration to EPU reactor water concentrations are those isotopes that are common to both the Browns Ferry design basis and ANSI/ANS-18.1-1984. EPU fission product activity levels in the reactor water remain a fraction (2%) of the design basis fission product activity, per Table 2.9-5.

The total activated corrosion product activity is approximately 5% higher than the original design basis activity as a consequence of EPU. However, the sum of the activated corrosion product activity and the fission product activity remains a small fraction (3%) of the total design basis activity. Therefore, the activated corrosion product and fission product activities design bases for Browns Ferry are unchanged for EPU.

For EPU, normal radiation sources are expected to increase slightly as shown in Table 2.9-3. Shielding aspects of the plant were conservatively designed for a more limiting design basis

source term. Thus, the increase in radiation sources does not affect radiation zoning or shielding and plant radiation area procedural controls will compensate for increased normal radiation sources. Therefore, activated corrosion and fission products meet all CLTR dispositions.

Conclusion

TVA has evaluated the radioactive source term for radwaste systems associated with the proposed EPU. The evaluation concludes that the proposed radioactive source term meets the requirements of 10 CFR Part 20, 10 CFR Part 50 Appendix I, and draft GDC-70. Therefore, the proposed EPU is acceptable with respect to source terms.

2.9.2 Radiological Consequences Analyses Using Alternative Source Terms

Regulatory Evaluation

The NRC's acceptance criteria for radiological consequences analyses using an alternative source term are based on (1) 10 CFR 50.67, insofar as it sets standards for radiological consequences of a postulated accident; and (2) GDC-19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem Total Effective Dose Equivalent (TEDE), as defined in 10 CFR 50.2, for the duration of the accident.

Specific NRC review criteria are contained in SRP Section 15.0.1.

Browns Ferry Current Licensing Basis

Final GDC-19 is applicable to Browns Ferry as described in "Browns Ferry Nuclear Plant, Units 1, 2, and 3 - Issuance of Amendments Regarding Full-Scope Implementation of Alternative Source Term," dated September 27, 2004 (Reference 72).

Radiological consequences associated with potential Browns Ferry accidents are addressed in UFSAR Section 14.6, "Analysis of Design Basis Accidents - Uprated."

Technical Evaluation

In accordance with the current licensing basis documented in Browns Ferry UFSAR Section 14.6, dose consequences are evaluated for the following events.

- UFSAR Section 14.6.3 – Loss-of-Coolant Accident (LOCA)
- UFSAR Section 14.6.4 – Refueling Accident
- UFSAR Section 14.6.2 – Control Rod Drop Accident (CRDA)
- UFSAR Section 14.6.5 – Main Steam Line Break Accident (MSLBA)

The magnitude of radiological consequences of a DBA is proportional to the quantity of radioactivity released to the environment. This quantity is a function of the fission products released from the core as well as the transport mechanism between the core and the release point.

The effect of the proposed EPU on the radiological consequences of the LOCA, fuel handling accident (FHA), CRDA, and the MSLBA is based on an assessment of the effect of EPU changes

on the dose consequence analyses that were evaluated by the NRC in the SER for the Browns Ferry AST License Amendments 251, 290, and 249 (Reference 72), which approved a full-scope implementation of an AST that complies with the guidance given in RG 1.183 (Reference 68) and 10 CFR 50.67. The referenced amendments are based on 3,952 MWt (corresponding to the EPU power level). The EPU DBA analyses are performed for 102% of the EPU power level of 3,952 MWt, which is 4,031 MWt.

The LOCA, FHA, CRDA, and MSLBA were assessed for the EPU reactor operating domain (i.e., the EPU core power level with ECCS evaluation uncertainty factor applied) to confirm that the EPU doses remained within regulatory limits.

Loss-of-Coolant Accident

The post-LOCA doses at the Exclusion Area Boundary (EAB), Low Population Zone (LPZ), Control Room (CR), and Technical Support Center (TSC) were analyzed for EPU conditions. The analysis was performed based on plant operation at 102% of the EPU power level of 3,952 MWt. The EPU core inventory was used. The analysis methods were not changed from those used in Reference 72. All dose significant design inputs and assumptions are the same as those in Reference 72.

In addition, the suppression pool pH was evaluated to ensure that the pH would remain above 7 for the duration of the LOCA. This analysis assumes that sodium pentaborate is injected via the Standby Liquid Control System (SLCS) within two hours of the onset of the LOCA. It was shown that the suppression pool pH remains above 7 for the duration of the LOCA (30 days). The quantity of the sodium pentaborate required to maintain the suppression pool pH above 7 for the duration of the LOCA is provided in Technical Specification SR 3.1.7. This analysis ensures that the particulate form of iodine (CsI) would be retained in the suppression pool water and not re-evolve and become airborne as elemental iodine.

The EPU post-LOCA EAB, LPZ, CR, and TSC doses were determined to be within the applicable regulatory limits. The results and regulatory criteria are summarized in Table 2.9-6, along with the estimated CLTP results.

Refueling Accident

The post-FHA EAB**,** LPZ, and CR doses were analyzed for EPU conditions. The analysis was performed based on plant operation at 102% of the EPU power level of 3,952 MWt. The EPU core inventory was used. The analysis methods and results were not changed from those used in Reference 72.

The EPU post-FHA EAB and CR doses were determined to be within the applicable regulatory limits. The results and regulatory criteria are summarized in Table 2.9-7, along with the estimated CLTP results.

Control Rod Drop Accident (CRDA)

The post-CRDA EAB, LPZ, and CR doses were analyzed for EPU conditions. The analysis was performed based on plant operation at the EPU power level of 3,952 MWt. The EPU core

inventory was used. The analysis methods were not changed from those used in Reference 72. The updated design inputs were confirmed to remain applicable or bounded for EPU conditions.

The EPU post-CRDA EAB, LPZ, and CR doses were determined to be within the applicable regulatory limits. The results and regulatory criteria are summarized in Table 2.9-8, along with the estimated CLTP results.

Main Steam Line Break Accident (MSLBA)

As described in UFSAR Section 14.6.5, accidents that result in the release of radioactive materials outside the secondary containment are the result of postulated breaches in the nuclear system process barrier. The design basis accident is a complete severing of one main steam line outside the secondary containment. For the purpose of radiological dose calculations, the main steam isolation valves are assumed to be closed at 5.5 seconds after the break. The postulated main steam line break outside the primary containment with a five second isolation valve closure results in the maximum calculated radiological dose and is, therefore, the design basis accident.

The MSLB accident is analyzed based on plant operation at 102% of the EPU power level of 3,952 MWt and assuming a continuous release (from the Turbine Building within two hours) and an instantaneous "puff" release with all of the inventory of the break released to the environment. A concentration of 32 μCi/gm I-131 is conservatively used to bound the Technical Specification limit of 26 μCi/gm for iodine spiking conditions. This source term is not affected by EPU.

The EPU post-accident doses for the MSLBA were determined to be within the applicable regulatory limits as described in UFSAR Section 14.6.5.3. The results and regulatory criteria are summarized in Tables 2.9-9 and 2.9-10.

<u>Post-LOCA Vital Area Mission Doses</u>

An additional review of the doses associated with access to vital areas was conducted to determine the effect of EPU. The times required for transit to and work in vital areas are not changed with EPU.

Vital areas are defined in NUREG-0737, Item II.B.2, as those areas "which will or may require occupancy to permit an operator to aid in the mitigation of or recovery from an accident." Compliance to NUREG-0737, Item II.B.2, assures the shielding adequacy necessary to reduce the Whole Body (WB) dose (i.e., external dose) to an operator to perform the vital function in a given mission time to less than the allowable limit of 5 rem TEDE dose.

For other personnel, the post-accident shielding dose and shielding adequacy at EPU conditions is evaluated in Section 2.10.1.

<u>Post-LOCA Vital Areas Requiring Continuous Occupancies</u>

<u>Control Room (CR)</u>

The post-accident CR dose contributions from various radioactive sources are analyzed and listed in Tables 2.9-6 through 2.9-10, along with the estimated CLTP results. Table 2.9-11 lists

the post-LOCA dose for locations requiring continuous occupancy, along with the estimated CLTP results.

Technical Support Center (TSC)

The post-LOCA TSC dose contributions from various radioactive sources are listed in Table 2.9-11. The TSC is at the same location as the Control Room, within the Control Room Habitability Zone (CRHZ); thus, the same atmospheric dispersion factors were used to calculate the dose at the TSC receptor.

Post-LOCA Vital Areas Requiring Infrequent Occupancies

The vital areas requiring infrequent occupancies to perform the required vital functions at EPU conditions are listed in Table 2.9-12, including the resulting doses along with the estimated CLTP results. The radiation exposures to vital areas are calculated using the occupancy times determined based on time-motion studies. The applicable plant procedures take complete control of the radiation exposure during vital functions by providing the radiation protection coverage to perform radiation surveys, and determining occupancy and radiation protection requirements before the vital functions are performed to maintain the resulting whole body exposure to ALARA and within the guideline value. For mission doses, it is assumed that the operators will use Self-Contained-Breathing-Apparatus (SCBA) equipment prior to performance of vital functions, except for the restart of control bay chillers.

Conclusion

TVA has evaluated and accounted for the effects of the proposed EPU on the accident analyses. The evaluation concludes that the plant site and the dose-mitigating ESFs remain acceptable with respect to the radiological consequences of postulated DBAs because, as set forth above, the calculated TEDE at the EAB, at the LPZ outer boundary, and in the control room meet the exposure guideline values specified in 10 CFR 50.67 and the current licensing basis. Therefore, the proposed EPU is acceptable with respect to radiological consequences of DBAs.

Table 2.9-1 Total Activity Levels

Item	Parameter	Unit	Calculated EPU Value	Design Basis Value	EPU to Design Basis Value Comparison
1	Activity concentrations of principal radionuclides in fluid streams for normal operation	N/A	Table 2.9-2		Table 2.9-2 contains the EPU calculated radionuclide concentrations in reactor water and steam.
2	Total fission product offgas source term	μCi/sec, t = 30 min	3.6E+04	3.5E+05	Design basis value is bounding for EPU
3	Total fission product activity concentration in reactor water	μCi/g	1.089E-01	5.73E+00	Design basis value is bounding for EPU
4	Total activated corrosion product activity concentration in reactor water	μCi/g	6.647E-02	6.359E-02	Design basis value is not bounding for EPU(1)
5	Total Fission Products and Activated Corrosion Products (Sum of Items 3 and 4)	μCi/g	1.753E-01	5.794E+00	Design basis value for the sum of activated corrosion products and the fission products activity is bounding for EPU.

(1) Although EPU value is not bounded by the design basis value, the total activated corrosion product activity concentration is much smaller than the fission product concentration. The sum of the activated corrosion product activity and the fission product activity remains a small fraction (3%) of the total design basis activity in reactor water.

Table 2.9-2 Activity Concentrations of Principal Radionuclides in Fluid Streams for EPU

Isotope	EPU x 1.02 Reactor Water µCi/g	EPU x 1.02 Reactor Steam µCi/g
Class 1: Noble Gases		
Kr-83m		5.900E-04
Kr-85m		1.000E-03
Kr-85		4.000E-06
Kr-87		3.300E-03
Kr-88		3.300E-03
Kr-89		2.100E-02
Xe-131m		3.300E-06
Xe-133m		4.900E-05
Xe-133		1.400E-03
Xe-135m		4.400E-03
Xe-135		3.800E-03
Xe-137		2.600E-02
Xe-138		1.500E-02
Class 2: Halogens		
I-131	2.451E-03	3.676E-05
I-132	2.210E-02	3.315E-04
I-133	1.648E-02	2.527E-04
I-134	3.988E-02	6.029E-04
I-135	2.347E-02	3.520E-04
Class 3: Cesium, Rubidium		
Rb-89	3.972E-03	3.972E-06
Cs-134	3.061E-05	3.061E-08
Cs-136	2.033E-05	2.033E-08
Cs-137	8.162E-05	8.162E-08
Cs-138	8.053E-03	8.053E-06

Table 2.9-2 Activity Concentrations of Principal Radionuclides in Fluid Streams for EPU (continued)

Isotope	EPU x 1.02 Reactor Water	EPU x 1.02 Reactor Steam
Class 4: Water Activation Products		
N-16	6.000E+01	5.000E+01
Class 5: Tritium		
H-3	1.000E-02	1.000E-02
Class 6a: Activated Corrosion Products		
Na-24	1.076E-02	1.076E-05
P-32	2.261E-04	2.261E-07
Cr-51	6.793E-03	6.793E-06
Mn-54	7.935E-05	7.935E-08
Mn-56	4.734E-02	4.734E-05
Fe-55	1.134E-03	1.134E-06
Fe-59	3.398E-05	3.398E-08
Co-58	2.266E-04	2.266E-07
Co-60	4.535E-04	4.535E-07
Ni-63	1.134E-06	1.134E-09
Cu-64	3.203E-02	3.203E-05
Zn-65	2.267E-04	2.267E-07
W-187	3.286E-04	3.286E-07
Class 6b: Other Fission Products and Actinide		
Sr-89	1.133E-04	1.133E-07
Sr-90	7.936E-06	7.936E-09
Y-90	7.936E-06	7.936E-09
Sr-91	4.203E-03	4.203E-06
Sr-92	9.512E-03	9.512E-06
Y-91	4.532E-05	4.532E-08

Table 2.9-2 Activity Concentrations of Principal Radionuclides in Fluid Streams for EPU (continued)

Isotope	EPU x 1.02 Reactor Water	EPU x 1.02 Reactor Steam
Y-92	5.847E-03	5.847E-06
Y-93	4.221E-03	4.221E-06
Zr-95	9.064E-06	9.064E-09
Nb-95	9.064E-06	9.064E-09
Mo-99	2.237E-03	2.237E-06
Tc-99m	2.237E-03	2.237E-06
Ru-103	2.265E-05	2.265E-08
Rh-103m	2.265E-05	2.265E-08
Ru-106	3.401E-06	3.401E-09
Rh-106	3.401E-06	3.401E-09
Ag-110m	1.134E-06	1.134E-09
Te-129m	4.530E-05	4.530E-08
Te-131m	1.102E-04	1.102E-07
Te-132	1.121E-05	1.121E-08
Ba-137m	8.162E-05	8.162E-08
Ba-140	4.521E-04	4.521E-07
La-140	4.521E-04	4.521E-07
Ce-141	3.397E-05	3.397E-08
Ce-144	3.401E-06	3.401E-09
Pr-144	3.401E-06	3.401E-09
Np-239	8.930E-03	8.930E-06

Table 2.9-3 Comparison of Normal Operation (CLTP) and EPU Activation and Fission Products

Isotopes	Half Life	CLTP (μCi/g)	EPU x 1.02 (μCi/g)	EPUx1.02/CLTP
Class 1: Noble Gases (in Reactor Steam)				
Kr-83m	1.83 hr	5.900E-04	5.900E-04	1.00
Kr-85m	4.48 hr	1.000E-03	1.000E-03	1.00
Kr-85	10.7 y	4.000E-06	4.000E-06	1.00
Kr-87	76 min	3.300E-03	3.300E-03	1.00
Kr-88	2.84 hr	3.300E-03	3.300E-03	1.00
Kr-89	3.18 min	2.100E-02	2.100E-02	1.00
Xe-131m	11.77 d	3.300E-06	3.300E-06	1.00
Xe-133m	2.19 d	4.900E-05	4.900E-05	1.00
Xe-133	5.25 d	1.400E-03	1.400E-03	1.00
Xe-135m	15.6 min	4.400E-03	4.400E-03	1.00
Xe-135	9.1 hr	3.800E-03	3.800E-03	1.00
Xe-137	3.82 min	2.600E-02	2.600E-02	1.00
Xe-138	14.1 min	1.500E-02	1.500E-02	1.00
Class 2 : Halogens (in Reactor Water)				
I-131	8.04 d	2.359E-03	2.451E-03	1.04
I-132	2.28 hr	2.096E-02	2.210E-02	1.05
1-133	20.9 hr	1.583E-02	1.648E-02	1.04
1-134	52.6 min	3.743E-02	3.988E-02	1.07
1-135	6.61 hr	2.245E-02	2.347E-02	1.05
Class 3: Cesium, Rubidium (in Reactor Water)				
Rb-89	15.2 min	3.651E-03	3.972E-03	1.09
Cs-134	2.062 y	2.721E-05	3.061E-05	1.12
Cs-136	13.1 d	1.808E-05	2.033E-05	1.12

Table 2.9-3 Comparison of Normal Operation (CLTP) and EPU Activation and Fission Products (continued)

Isotopes	Half Life	CLTP (μCi/g)	EPU x 1.02 (μCi/g)	EPUx1.02/CLTP
Cs-137	30.17 y	7.255E-05	8.162E-05	1.13
Cs-138	32.2 min	7.391E-03	8.053E-03	1.09
Class 6a: **Activated Corrosion Products (in Reactor Water)**				
Na-24	15.02 hr	9.621E-03	1.076E-02	1.12
P-32	14.28 d	2.011E-04	2.261E-04	1.12
Cr-51	27.7 d	6.039E-03	6.793E-03	1.12
Mn-54	312 d	7.053E-05	7.935E-05	1.13
Mn-56	2.579 hr	4.288E-02	4.734E-02	1.10
Fe-55	2.7 y	1.008E-03	1.134E-03	1.13
Fe-59	44.6 d	3.021E-05	3.398E-05	1.12
Co-58	70.8 d	2.014E-04	2.266E-04	1.13
Co-60	5.271 y	4.031E-04	4.535E-04	1.13
Ni-63	100 y	1.008E-06	1.134E-06	1.13
Cu-64	12.7 hr	2.866E-02	3.203E-02	1.12
Zn-65	244.1 d	2.015E-04	2.267E-04	1.13
W-187	23.9 hr	2.932E-04	3.286E-04	1.12
Class 6b: **Other Fission Products and Actinide (in Reactor Water)**				
Sr-89	50.5 d	1.007E-04	1.133E-04	1.13
Sr-90	28.8 y	7.054E-06	7.936E-06	1.13
Y-90	64.1 hr	7.054E-06	7.936E-06	1.13
Sr-91	9.5 hr	3.767E-03	4.203E-03	1.12
Sr-92	2.71 hr	8.611E-03	9.512E-03	1.10
Y-91	58.5 d	4.029E-05	4.532E-05	1.12
Y-92	3.54 hr	5.281E-03	5.847E-03	1.11

Table 2.9-3 Comparison of Normal Operation (CLTP) and EPU Activation and Fission Products (continued)

Isotopes	Half Life	CLTP (μCi/g)	EPU x 1.02 (μCi/g)	EPUx1.02/CLTP
Y-93	10.2 hr	3.781E-03	4.221E-03	1.12
Zr-95	64 d	8.057E-06	9.064E-06	1.12
Nb-95	35 d	8.057E-06	9.064E-06	1.12
Mo-99	66.02 hr	1.992E-03	2.237E-03	1.12
Tc-99m	6.02 hr	1.992E-03	2.237E-03	1.12
Ru-103	39.4 d	2.014E-05	2.265E-05	1.12
Rh-103m	56.1 min	2.014E-05	2.265E-05	1.12
Ru-106	367 d	3.023E-06	3.401E-06	1.13
Rh-106	29.8 sec	3.023E-06	3.401E-06	1.13
Ag-110m	252 d	1.008E-06	1.134E-06	1.13
Te-129m	33.5 d	4.027E-05	4.530E-05	1.12
Te-131m	30 hr	9.829E-05	1.102E-04	1.12
Te-132	78 hr	9.976E-06	1.121E-05	1.12
Ba-137m	2.551 min	7.255E-05	8.162E-05	1.13
Ba-140	12.79 d	4.020E-04	4.521E-04	1.12
La-140	40.3 hr	4.020E-04	4.521E-04	1.12
Ce-141	32.5 d	3.020E-05	3.397E-05	1.12
Ce-144	284 d	3.023E-06	3.401E-06	1.13
Pr-144	17.3 min	3.023E-06	3.401E-06	1.13
Np-239	2.35 d	7.952E-03	8.930E-03	1.12

Table 2.9-4 Comparison of Design Basis to EPU Noble Gas Radionuclide Source Terms

Offgas Fission Product Concentration Valid for t= 30 minutes		
Isotope	Design Basis (μCi/sec)	EPU x 1.02 (μCi/sec)
Kr-83m	8.9E+03	1.0E+03
Kr-85m	2.0E+04	2.0E+03
Kr-85	2.6E+01	8.5E+00
Kr-87	5.5E+04	5.3E+03
Kr-88	6.1E+04	6.2E+03
Kr-89	9.2E+02	6.4E+01
Xe-131m	5.3E+01	7.0E+00
Xe-133m	6.6E+02	1.0E+02
Xe-133	1.8E+04	3.0E+03
Xe-135m	2.8E+04	2.5E+03
Xe-135	6.1E+04	7.8E+03
Xe-137	3.0E+03	2.4E+02
Xe-138	9.3E+04	7.3E+03
Total	3.5E+05	3.6E+04

Table 2.9-5 Comparison of Design Basis to EPU Radiation Sources

Isotopes	Half Life	Design Basis* Reactor Water (μCi/ml)	EPU x 1.02 Reactor Water (μCi/g)	EPUx1.02/Design Basis
Fission Products				
I-131	8.04 d	1.700E-01	2.451E-03	0.01
I-132	2.28 hr	1.020E+00	2.210E-02	0.02
I-133	20.9 hr	1.040E+00	1.648E-02	0.02
I-134	52.6 min	1.470E+00	3.988E-02	0.03
I-135	6.61 hr	1.300E+00	2.347E-02	0.02
Tc-99m	6.02 hr	1.300E-01	2.237E-03	0.02
Mo-99	66.02 hr	6.000E-01	2.237E-03	0.00
Ag-110m	252 d	6.300E-05	1.134E-06	0.02
Subtotal		5.730E+00	1.089E-01	0.02
Activated Corrosion Products				
Na-24	15.02 hr	2.000E-03	1.076E-02	5.38
P-32	14.28 d	2.000E-05	2.261E-04	11.31
Mn-56	2.579 hr	5.200E-02	4.734E-02	0.91
Co-58	70.8 d	5.400E-03	2.266E-04	0.04
Co-60	5.271 y	5.400E-04	4.535E-04	0.84
Fe-59	44.6 d	8.020E-05	3.398E-05	0.42
Mn-54	312 d	4.200E-05	7.935E-05	1.89
Cr-51	27.7 d	5.000E-04	6.793E-03	13.59
W-187	23.9 hr	3.010E-03	3.286E-04	0.11
Zn-65	244.1 d	2.000E-06	2.267E-04	113.35
Subtotal		6.359E-02	6.647E-02	1.05
Fission Products and Activated Corrosion Products				
Total		5.794E+00	1.753E-01	0.03

* A reactor water density of 1 g/cc is conservatively assumed.

Table 2.9-6 LOCA Radiological Consequences

	TEDE Dose (rem)			
	Receptor Location			
	CR	EAB	LPZ	TSC
Estimated Dose CLTP	0.81	0.34	1.87	0.81
Calculated Dose EPU	1.94	1.71	2.38	1.94
Allowable TEDE Limit	5.0	25.0	25.0	5.0

Table 2.9-7 FHA Radiological Consequences

	TEDE Dose (rem)		
	Receptor Location		
	CR	EAB	LPZ
Estimated Dose CLTP	0.57	1.33	0.67
Calculated Dose EPU	0.54	0.86	0.43
Allowable TEDE Limit	5.0	6.3	6.3

Table 2.9-8 CRDA Radiological Consequences

	TEDE Dose (rem)		
	Receptor Location		
	CR	EAB	LPZ
Estimated Dose CLTP	0.23	1.99	1.14
Calculated Dose EPU	0.26	1.17	0.70
Allowable TEDE Limit	5.0	6.3	6.3

Table 2.9-9 MSLB Pre-Incident Iodine Spike Radiological Consequences

	TEDE Dose (rem)		
	Receptor Location		
	CR	EAB	LPZ
Estimated Dose CLTP	0.41	1.30	0.65
Calculated Dose EPU	0.41	1.30	0.65
Allowable TEDE Limit	5.0	25	25

Table 2.9-10 MSLB Equilibrium Iodine Concentration Radiological Consequences

	TEDE Dose (rem)		
	Receptor Location		
	CR	EAB	LPZ
Estimated Dose CLTP	0.041	0.130	0.065
Calculated Dose EPU	0.041	0.130	0.065
Allowable TEDE Limit	5.0	2.5	2.5

Table 2.9-11 Post-LOCA Vital Areas Requiring Continuous Occupancies

Areas Requiring Continuous Occupancy	30-Day Dose TEDE Dose (rem)	
	Estimated CLTP	EPU
Control Room	1.62	1.94
Technical Support Center	1.62	1.94

Table 2.9-12 Post-LOCA Mission Doses

Access Route	Mission Time (hr)	Projected Total Mission Dose	
		CLTP (rem)	EPU (rem)
Control Room/TSC to Restart Control Bay Chillers Following an Accident [a]			
MSLB [b]	0.2394	0.192	0.25
LOCA	0.2394	0.183	0.238
Control Room/TSC to Post-Accident Sampling Station [c,d]			
Whole Body (5 rem):			
Unit 1	9.88	5.15 rem	6.59 rem
Unit 2	10.15	10.7 rem	13.7 rem
Unit 3	10.43	10.9 rem	14.0 rem
Extremities (50 rem):			
Unit 1	9.88	54.4 rem	69.6 rem
Unit 2	10.15	60.1 rem	76.9 rem
Unit 3	10.43	60.2 rem	77.1 rem
Control Room /TSC to Manually Realign HVAC Equipment following a LOCA	1	0.05 rem	0.06 rem

Doses dependent upon accessibility during the early phases of the accident; use of SCBA is assumed, except for restart of control bay chiller.

(a) Maximum ratio of core inventories of 1.30.

(b) MSLB bounds the CRDA mission dose.

(c) Even though the sum of the whole body doses exceeds 5 rem and the sum of the extremities exceeds 50 rem, no single sample exceeds that respective value. It should be noted that these doses can be distributed over a number of individuals.

(d) Details regarding the Post-Accident Sampling Station (PASS) mission doses are provided in Table 2.9-13.

Table 2.9-13 Post-Accident Sampling Mission Dose Summary CLTP versus EPU

Sample	CLTP			EPU		
	Unit 1	Unit 2	Unit 3	Unit 1	Unit 2	Unit 3
Whole Body Mission Doses (rem)						
Degassed RCS Sample	1.92	3.56	3.61	2.46	4.56	4.62
Small RCS Sample	1.08	2.22	2.26	1.38	2.84	2.89
Dissolved Gas Sample	1.35	3.02	3.07	1.77	3.87	3.93
Containment Air Sample	0.73	1.87	1.91	0.94	2.39	2.44
Total [a,b]	5.15	10.67	10.85	6.59	13.7	14.0
Extremity Dose (rem)						
Degassed RCS Sample	27.20	28.80	28.90	34.80	36.90	37.00
Small RCS Sample	1.33	2.47	2.51	1.70	3.16	3.21
Dissolved Gas Sample	10.30	12.00	12.00	13.20	15.40	15.40
Containment Air Sample	15.60	16.80	16.80	20.00	21.50	21.50
Total [a]	54.43	60.07	60.21	69.6	76.9	77.1

(a) Doses dependent upon accessibility during the early phases of the accident; use of SCBA is assumed.

(b) Even though the sum of the whole body doses exceeds 5 rem and the sum of the extremities exceeds 50 rem, no single sample exceeds that respective value. It should be noted that these doses can be distributed over a number of individuals.

2.10 Health Physics

2.10.1 Occupational and Public Radiation Doses

Regulatory Evaluation

The NRC's acceptance criteria for occupational and public radiation doses are based on 10 CFR Part 20 and GDC-19.

Specific NRC review criteria are contained in SRP Sections 12.3 and 12.4 and other guidance provided in Matrix 10 of RS-001.

Browns Ferry Current Licensing Basis

Final GDC-19 is applicable to Browns Ferry as described in "Browns Ferry Nuclear Plant, Units 1, 2, and 3 - Issuance of Amendments Regarding Full-Scope Implementation of Alternative Source Term," dated September 27, 2004 (Reference 72).

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of EPUs. Sections 8.5 and 8.6 of the CLTR address the effect of EPU on the radiation sources in the reactor core and in the reactor coolant. The results of this evaluation are described below.

2.10.1.1 Increases in Radiation Sources

The proposed EPU power level of 3,952 MWt is a 20% increase over the OLTP level of 3,293 MWt. All reported percentage increases throughout Section 2.10.1.1 are relative to OLTP unless otherwise specified.

The Browns Ferry EPU corresponds to an increase in the rated thermal power of 20% in the OLTP level of 3,293 MWt, or an increase of approximately 14% over the current licensed thermal power (CLTP) level of 3,458 MWt. The production rate of radiation from the fission process itself (neutron and gamma) and radioactive material (either fission or activation products) in the reactor core are directly dependent on the neutron flux and power level of the reactor. Therefore, these source terms are expected to increase proportionately with the increase in the rated thermal power of 20% both directly in the reactor core and in the primary coolant.

For the majority of the doses estimates, the effects of the EPU on doses were evaluated by assuming a proportional increase of 20% in the fission product inventory, and the resulting reactor coolant concentrations. An increase in the activation products of 20% was also assumed. The concentration of noble gases and other volatile radioisotopes in the main steam are expected to remain unchanged. The increase in the activity is expected to be offset by the increase in the steam flow rate. Although the concentration remains the same, the steam flow rate results in proportionate increase in the rate that these radioisotopes are introduced into the Main Condenser and Off Gas systems.

For the very short lived activities, most significantly the N-16, the decreased transit time (and decay time) in the main steam line, and the increased mass flow of the steam results in a larger increase in these activities in the major turbine components. In general, the dose changes due to N-16 in the equipment above grade are a significant contributor to dose due to skyshine offsite. This consists of equipment such as the main steam lines, turbines, extraction steam, and the upper portions of the moisture separators, and the piping between the moisture separators and LP turbine. It is assumed that N-16 activity and the resultant dose rate increase proportionately with the rated thermal power (i.e., 1.20 with Hydrogen Water Chemistry (HWC) effect excluded). From operational experience with Unit 2 and 3, the effect with HWC included is 1.32 from CLTP to EPU. No new high radiation zones are created around equipment in the turbine building due to the increase in the N-16 activity. CLTP radiation levels around turbine building equipment with HWC and Noble Metal addition result in acceptable radiation levels. Therefore, an increase of 1.32 in the radiation levels for EPU conditions from CLTP due to increased N-16 activity is assumed to remain acceptable; however, they may exceed the original design criteria of 1 mrem/hr in the turbine building general area. For areas in which the dose rate does exceed the original design criteria, ALARA principles will be utilized to minimize personnel exposure.

As stated above, the EPU results in an increase in steam flow which results in higher N-16 and other activation products in the turbines. The increased flow rate and velocity, which result in shorter travel times to the turbine and less radioactive decay in transit, lead to higher radiation levels in and around the turbines and offsite skyshine dose. With the consideration of shielding provided for steam piping, turbines, and condensers and distance from the sources, skyshine contribution to offsite normal radiation dose is expected to be negligible. Any discernible increase in radiation as a result of N-16 would be measured on the site environmental dosimeter stations (Optically-Simulated Luminescence dosimetry, or OSLs). Therefore, it is unlikely that the increase in N-16 source term due to the EPU will result in any measurable offsite dose to the public.

The concentration of non-volatile fission products, actinides, and corrosion and wear products in the reactor coolant are expected to increase proportionately with the power increase. However, the increase steam flow and moisture carryover in the steam may result in an increase in the activity to the balance of plant. In support of the EPU, new steam dryers will be installed to accommodate the increase in steam flow at the uprated power. This will ensure that the moisture carryover will remain very low. With the installation of the new steam dryers and increased steam flow, the rate at which the activity is introduced to the balance of plant will increase proportionately to the steam flow. It is expected that the relative contribution of these non-volatile fission products will be a small increase to the dose rates around the balance of plant system during power operations.

In summary, EPU is assumed to increase the reactor generated radiation sources by up to 20% compared to the OLTP, with a factor up to 32% based on CLTP possible for N-16 contributions and HWC in the BOP systems containing turbine building equipment.

Fission product activity in the reactor water is the result of a combination of tramp uranium and small releases from the fuel rods. With an assumed increase of 20% above OLTP, fission product concentrations in the reactor coolant remain bounded by the OLTP design basis sources used for shielding design**.**

Fission product activity in the reactor steam is a result of noble gas releases and halogen and moisture carry-over from the reactor coolant. With an assumed increase of 20% above OLTP, the fission products in the condenser offgas, and the offgas release rates after 30 minutes of decay, remain well below the OLTP design basis of 0.1 Ci per second design in the UFSAR.

2.10.1.2 Occupational and Onsite Radiation Exposures

The CLTR topics regarding occupational and onsite radiation exposures are listed below. Browns Ferry meets all CLTR dispositions. The topics addressed in this evaluation are:

Topic	CLTR Disposition	Browns Ferry Result
Normal Operational Radiation Levels	Plant-Specific	Meets CLTR Disposition
Post-Operation Radiation Levels	Plant-Specific	Meets CLTR Disposition
Post-Accident Radiation Levels	Plant-Specific	Meets CLTR Disposition

2.10.1.2.1 Occupational and Onsite Radiation Exposures

As documented in NUREG-0713, normal occupational doses are well within the 5 rem TEDE limit during normal plant operation as shown in Table 2.10-1. Based on a 20% increase in the power and the resultant radiation field, it is conservatively assumed that the occupational dose will increase proportionately with the power (an increase of 32% is assumed in the turbine building due to the effect of N-16 and HWC). Therefore, assuming a 32% increase in the radiation field, occupational doses are expected to remain well within the annual dose limit of 5 rem TEDE in accordance with 10 CFR 20.1201. These doses will further be controlled via ALARA principles.

The doses calculated for accident conditions are not expected to exceed an increase of 20% as a result of the increase in fission products at EPU conditions The shielding must also be sufficient to provide control room habitability per GDC 19 and operator access to vital areas per NUREG-0737, “Clarification of TMI (Three Mile Island) Action Plan Requirements, “Item II.B.2 during the accident (see Control Room Habitability Requirements in Section 2.7.1) Included in areas that require continuous access is the control room and the technical support center. Required missions (during accident conditions) to the reactor building refueling floor/spent fuel pool

makeup and emergency diesel generator building due to the increased source term due to the EPU does not adversely affect the accessibility to the vital areas or the mission doses to perform required actions during postulated accidents. Conditions within the control room during an accident are provided in Section 2.9.2.

Per the CLTR, plants employing the HWC often exceed the original basis for shielding in the turbine building and offsite and are licensed under empirical analysis for the operation with HWC. TVA has performed analyses to evaluate the effect of HWC and noble metal addition to the radiation levels in the plant. As such, personnel exposures will be maintained within acceptable limits via the Browns Ferry ALARA/Radiation Program. Procedural controls will compensate for increased radiation levels in the plant due to the EPU. Per the CLTR, plants with zinc injection, HWC, and noble metal addition show a decrease in post-operation radiation levels and/or reduced repairs required in radiation areas.

Reactor Building Radiation Levels

The CLTR states that normal operational radiation levels will increase slightly as a result of the EPU. Additionally, improvements in performance with respect to lowering occupational doses (e.g., use of ALARA principles), as documented in NUREG-0713, exceed anticipated increases in radiation levels as a result of EPU effects.

In primary containment, radiation levels near the reactor vessel are assumed to increase 20%. However, the primary containment is inaccessible during operation, and because of the margin in the shielding around the reactor vessel and drywell, the increase of 20% above the OLTP will not measurably increase occupational doses during power operation.

The radiation sources in the core (i.e., the fission product inventory and actinides) are conservatively assumed to increase in proportion to the increase in the power. This includes the core inventory, reactor coolant activity, and the steam activity. The reactor vessel is surrounded by containment and shielding. Therefore, a 20% increase in the power and hence the conservative assumption of a comparable increase in the source term will not adversely affect the occupational dose during normal operation. Similarly, the radiation shielding provided in the balance of plant (i.e., around radioactive waste systems, main steam lines, and the main turbine) is conservatively designed to minimize the effect of the increased source terms on the occupational dose in the normally occupied areas of the plant during normal operations. Therefore, the radiation zones designations for normally occupied areas of the plant will remain acceptable based on the current shielding designs.

SFP Radiation Levels

Radiation levels due to the spent fuel are assumed to increase 20% above the OLTP. Radiation exposures in accessible areas adjacent to the sides and the bottom of the spent fuel pool are expected to be within the allowable dose rate limit of the existing radiation zone designation. Expected increases in areas surrounding the SFP will occur primarily during the course of fuel handling activities. The dose rates adjusted for post-EPU conditions, including above the water surface on the bridge at the closest operator distance from freshly discharged fuel will remain

acceptable. Radiation levels around the spent fuel pool area are assumed to increase proportionately with the increase in the power up to 20%. However, there is sufficient margin in the current design analysis to bound a 20% increase in direct radiation due to the EPU, see Table 2.10-1a.

Outside Containment Radiation Levels

Outside primary containment, radiation shielding was specified using the OLTP design basis radiation sources. For these areas, the actual operating sources, not the design basis sources actually increase. The normal operation coolant source term is based on a conservative failed fuel fraction and the post-accident coolant source term is based on TID-14844 (Reference 107) which bounds the AST source term. These two source terms combined are used to assess the radiation levels in the plant and hence the total integrated dose over a 60 year life for equipment qualification. In other words, the design basis source is used to develop the shield design basis on a conservative failed fuel fraction, which for the normal operation dose was originally assumed to be derived from a significantly greater failed fuel fraction than expected during normal operation (ANS/ANSI-18.1-1984). For accident conditions, the activity released per the TID-14844 guidance exceeds the activity released per the Regulatory Guide 1.183 (Reference 68) by more than 20%. Therefore, these original bounding design values will inherently bound the EPU conditions.

The design basis source term was also used to establish the shielding design for the BOP areas (e.g., turbine building, condenser, and offgas system) which as justified above bounds the expected increase in the normal operating source terms. Therefore, the shield design has sufficient margin in the Browns Ferry design to maintain occupational doses at acceptable levels at EPU conditions. At EPU conditions, the increase in BOP areas is primarily due to the increase in N-16 activity in the steam which is estimated to increase by a factor of 1.32 (with HWC effect included) at EPU conditions. Thus, the dose rate is expected to increase by a factor of 1.32 in affected areas. Based on operational data, the resultant dose rates may exceed the original design value of 1 mrem/hour; however, through ALARA principles personnel exposure is expected to be acceptable with respect to occupational dose. The total activity associated with condensate cleanup system may be greater, but the activity is distributed among the filters and demineralizers. Therefore, the increase in dose rate is expected to be less than 20% above the OLTP. In summary, based on the design basis source term used for the shield design, there is sufficient margin in the Browns Ferry design to ensure the shielding is adequate to maintain occupational and onsite dose ALARA.

Hydrogen Water Chemistry and Noble Metal Chemical Addition Effect

As stated above, it is assumed that N-16 activity and the resultant dose rate is assumed to increase proportionately with the rated thermal power (i.e., by a factor of 1.20 with HWC effect excluded and by a factor of 1.32 with HWC effect included). Browns Ferry implemented HWC for Unit 1 in Cycle 11 and in Unit 3 in Cycle 9 outages. At which point, the dose rates

utilized in design calculations were conservatively increased by a factor of 5 to account for HWC. No other additional increases in the N-16 doses are expected.

The HWC effect has been evaluated and is reflected in Table 2.10-1a to Table 2.10-1c and Table 2.10-2.

In summary, the effect from the combination of Hydrogen Water Chemistry and Noble Metal (Noble Chem^{TM}) water chemistry on radiation levels has been reviewed at EPU conditions. This review has been used to evaluate the integrated dose to various instruments for equipment qualification. The dose rates were conservatively increased by a factor of five using a TID-14844 source term. Therefore, there is sufficient margin to allow a 20% increase in direct radiation due to the EPU.

Radiation Protection Features

Browns Ferry was designed with sufficient shielding (see discussion above) to ensure occupational and offsite exposures will remain ALARA at EPU conditions. In addition, procedural controls will compensate for increased radiation levels, and good housekeeping practices will minimize the potential for the spread of contamination within the plant.

Annual Occupation Exposure

The normal occupational doses are well within the 5 rem TEDE limit during normal plant operation. Based on a 20% increase in the power and the resultant radiation field, it is conservatively assumed that the occupational dose will increase proportionately with the power. Therefore, the occupational dose is expected to remain well within the annual occupational dose limit of 5 rem in accordance with 10 CFR 20. Therefore, ALARA principles will continue to be met.

Control Room Habitability

Operating at EPU conditions will result in an increased core inventory of radioactive material that is available for release during postulated accident conditions. The shielding must also be sufficient to provide control room habitability per GDC 19 and operator access to vital areas per NUREG-0737, "Clarification of TMI (Three Mile Island) Action Plan Requirements," Item II.B.2 during the accident. (see Control Room Habitability Requirements in Section 2.7.1). Included in areas that require continuous access is the control room and the technical support center. Infrequent missions to the reactor building refueling floor/spent fuel pool makeup, and emergency diesel generator building are also assumed. The increased source term due to the EPU does not adversely affect the accessibility to the vital areas or the mission doses to perform required actions during postulated accidents. Conditions within the control room during an accident are provided in Section 2.9.2.

Radiation Zone Review

Per the CLTR, plants employing HWC often exceed the original basis for shielding in the turbine building and offsite and are licensed under empirical analysis for the operation with HWC. TVA has performed analyses to evaluate the effect of HWC and noble metal addition to the radiation

levels in the plant. Personnel exposures will be maintained within acceptable limits via the Browns Ferry ALARA/Radiation Program. Procedural controls will compensate for increased radiation levels in the plant due to the EPU. Per the CLTR, plants with zinc injection, HWC, and noble metal addition show a decrease in post-operation radiation levels and/or reduced repairs required in radiation areas.

The radiation shielding provided in the balance of plant (i.e., around radioactive waste systems, main steam lines, and the main turbine) is conservatively designed to minimize the effect of the increased source terms on the occupational dose in the normally occupied areas of the plant during normal operations. Therefore, the radiation zones designations for normally occupied areas of the plant will remain acceptable based on the current shielding designs.

Radiation Monitoring Setpoints

Radiation monitor setpoints will be evaluated during EPU power ascension. Radiation monitor setpoints are procedurally controlled and may be adjusted as background radiation levels change for EPU conditions.

2.10.1.2.2 Post-Operation Radiation Levels

The normal operating doses specified for Browns Ferry are generally based on dose rate measurements at various locations during plant operation at OLTP conditions. The normal doses specified for OLTP conditions were increased by 20% for these areas. Browns Ferry has sufficient margin to accommodate a 20% increase (See Tables 2.10-1a through Table 2.10-1c) in the observed dose rate. Individual worker exposures can be maintained within acceptable limits by controlling access to radiation areas using the site ALARA program. Procedural controls compensate for increased radiation levels. In addition, Browns Ferry has previously implemented zinc injection and noble metal chemical addition to limit the increase in normal radiation doses from the implementation of hydrogen water chemistry.

Maintenance of Components

Normal radiation doses will increase by approximately 14% due to EPU compared to the CLTP. This increase in the radiation levels will not affect the ability of the plant programs to manage component service life for non-metallic parts that are outside of the environmental qualification program.

2.10.1.2.3 Post-Accident Radiation Levels

The increased post-accident radiation levels have no adverse effect on safety-related plant equipment. A plant-specific analysis for NUREG-0737, Item II.B.2, post-accident mission doses has been performed.

Operating at EPU conditions will result in an increased core inventory of radioactive material that is available for release during postulated accident conditions. The shielding must also be sufficient to provide for operator access to vital areas per NUREG-0737, "Clarification of TMI (Three Mile Island) Action Plan Requirements," Item II.B.2 during the accident. Included in areas that require continuous access are the control room and the technical support center (See

Section 2.7.1 of this report). Infrequent missions to the reactor building refueling floor/spent fuel pool makeup, and emergency diesel generator building are also assumed. The increased source term due to the EPU does not adversely affect the accessibility to the vital areas or the mission doses to perform required actions during postulated accidents. Mission doses are discussed in further detail in Section 2.9.2.

Vital Area Accessibility

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 RWCU and liquid radwaste equipment, the increase could be slightly higher. However, sufficient margin exist in the analyses such that personnel can perform any action necessary to mitigate or recover from an accident without exceeding GDC-19.

Maintenance of Components

Post-accident radiation doses will increase by approximately 14% due to EPU compared to the CLTP. This increase in the post-accident radiation levels will not affect the ability of plant programs to manage component service life for non-metallic parts that are outside of the environmental qualification program.

2.10.1.2.4 Public and Offsite Radiation Exposures

Browns Ferry meets all CLTR dispositions. The CLTR topics regarding public and offsite radiation exposures are listed below.

Topic	CLTR Disposition	Browns Ferry Result
Off-Site Plant Gaseous Emissions	Plant-Specific	Meets CLTR Disposition
Plant Skyshine from the Turbine	Plant-Specific	Meets CLTR Disposition

The CLTR states that for EPU, normal operation gaseous activity levels increase slightly, while the level of N-16 in the turbine increases in proportion to the rated steam flow.

The normal offsite doses are not significantly affected by operation at EPU conditions and will remain below the limits of 10 CFR 20, 10 CFR 50 Appendix I, and 40 CFR 190. The primary sources of normal offsite doses are: (1) airborne releases from the offgas system; and (2) gamma shine from the plant turbines. The normal operation offsite radiation exposures are the result of liquid and gaseous effluents and direct radiation or shine from onsite sources such as spent fuel stored onsite, solid waste, and shine from the N-16 source in the main steam lines and turbines.

The increase in activity levels is assumed to be proportional to the percentage increase in core thermal power. Technical Specification Section 5.5.1, "Offsite Dose Calculation Manual

(ODCM)" (Reference 74), implements the guidelines of 10 CFR 50 Appendix I. EPU does not involve significant increases in the offsite dose from noble gases, airborne particulates, iodine, tritium, or liquid effluents. Present offsite radiation levels form a negligible portion of background radiation. Therefore, the normal offsite doses are not significantly affected by operation at EPU and remain below the limits of 10 CFR 20, 10 CFR 50 Appendix I, and 40 CFR 190.

Browns Ferry implemented zinc injection and noble metal chemical addition to limit the increase in normal radiation doses from the implementation of HWC. The EPU results in an increase in steam flow which results in higher levels of N-16 and other activation products in the turbines. The increased flow rate and velocity, which result in shorter travel times to the turbine and less radioactive decay in transit, lead to higher radiation levels in and around the turbines and skyshine. With the consideration of shielding provided for steam piping, turbines, and condensers and distance from the sources, skyshine contribution to offsite normal radiation dose is expected to be negligible. Any discernible increase in radiation would be measured on the site environmental dosimeter stations. Therefore, it is expected that the increase in N-16 source term due to EPU will not result in any measurable dose to the public. The total dose to a member of the public includes effluent doses (liquid and gaseous); however, these exposure pathways are negligible in comparison to the direct radiation doses.

Direct radiation from dry fuel storage, in the Independent Spent Fuel Storage Installation (ISFSI), and Condensate Storage Tank (CST) are contributors to offsite dose due to shine. The maximum design dose from direct radiation from the ISFSI is 13.8 mrem/yr (whole body) and from the CST is 7.23 mrem/yr (whole body) at EPU conditions, respectively. In contrast, the dose contribution from shine from the Turbine Building due to N-16 is negligible. The dose from gaseous effluents from ISFSI leakage from four casks is 0.742 mrem/yr. The design annual dose from plant gaseous effluents is 1.47 mrem/yr (whole body) when processing Browns Ferry design data with BWR-GALE (Reference 108) and ODCM codes. The design liquid effluent contribution is 0.042 mrem/yr (whole body) when processing Browns Ferry design data with BWR GALE and ODCM codes. Therefore, the maximum annual dose to a member of the public from direct radiation and plant effluents are estimated to be 23.3 mrem whole body (See Table 2.10-2). Thus, the maximum annual dose to a member of the public from all dose contributors is within the 40 CFR 190 dose limit of 25 mrem/yr whole body.

Currently, the Browns Ferry offsite dose limits are well within 10 CFR 20, 10 CFR 50, Appendix I and 40 CFR 190 limits for dose to the public, as indicated in Table 2.10-2. The environmental monitoring program that is in place will continue to ensure that the offsite doses remain well within regulatory limits and will provide an indication should the doses increase above measured background levels.

Based on environmental monitoring data, the projected annual doses from plant effluents at EPU conditions are well within the 25 mrem dose limit of 40 CFR 190 for an offsite member of the public. The limiting dose receptors for members of the public would be those onsite (e.g., food vendors) because their work locations are nearer to the Turbine Building, the ISFSI, or the CST.

The maximum annual dose to vendors would not likely exceed 18 mrem under EPU and NobleChem™ water chemistry and reduced hydrogen injection conditions on all three units. Consequently, the 10 CFR 20.1301 annual dose limit of 100 mrem for a member of the public onsite would not be exceeded. Therefore, the projected annual dose to an onsite member of the public will be well within the dose limit.

2.10.1.3 Operational Radiation Protection Program

The increased production of non-volatile fission products and actinides, and corrosion and wear products in the reactor coolant may result in higher plate-out on the surfaces of the reactor coolant system, and in low flow areas in the reactor systems; as well as, an increase in dose rates from contained sources, surface contamination, and airborne radioactivity. These sources could result in increased occupational exposure during maintenance activities of these systems. However, the radiation protection program and ALARA practices will minimize the effect of increased radiation sources due to increased plate-out in the reactor system to personnel exposure and public exposure. The Radiation Program and ALARA practices include pre-job briefs, use of supplemental shielding (e.g., lead blankets), pre-job decontamination, and contamination control practices will ensure occupational doses will be maintained ALARA. To ensure individual worker exposure is maintained ALARA, access will be controlled to radiation areas.

The normal occupational doses are currently well within the 5 rem TEDE limit during normal plant operation. Based on a 20% increase in the power and the resultant radiation field, it is conservatively assumed that the occupational dose will increase proportionately with the power. Therefore, the occupational dose is expected to remain well within the annual occupational dose limit of 5 rem in accordance with 10 CFR 20. Therefore, ALARA principles will continue to be met.

In summary, the current operational radiation protection programs will control and/or compensate for potential increases in the radiation levels in the plant from contained sources, surface contamination, and airborne activity.

Conclusion

TVA has evaluated the effects of the proposed EPU on radiation source terms and plant radiation levels. The evaluation concludes that any increases in radiation doses will be maintained as low as is reasonably achievable. The evaluation further concludes that the proposed EPU meets the requirements of 10 CFR 20, 10 CFR 50 Appendix I, 40 CFR 190, and final GDC-19. Therefore, the proposed EPU is acceptable with respect to radiation protection and ensuring that occupational radiation exposures will be maintained as low as is reasonably achievable.

Table 2.10-1 Browns Ferry Average Occupational Dose for CLTP and EPU

Year	CLTP Average Dose (rem)	CLTP Collective Dose/MW-yr (person-rem)	EPU Average Dose (rem)	Dose Limit (rem)
2008	0.18	0.17	0.24	5
2009	0.16	0.12	0.21	5
2010	0.20	0.20	0.26	5
2011	0.14	0.10	0.18	5
2012	0.15	0.16	0.20	5

Table 2.10-1a Current and Anticipated Measured Radiation Dose in Selected Areas of the Reactor Building for 60 Year Normal Operation

Zone Number	Description	Plant Elevation	CLTP Dose (rads)	Scaling Factor(1)	EPU Dose(1) (rads)
0	Drywell with Permall Shield	N/A	8.85E+07	1.2	1.06E+08
0	Drywell without Permall Shield	N/A	1.10E+08	1.2	1.31E+08
00 (U2)	Wetwell (Torus)	N/A	3.35E+06	1.2	4.01E+06
00 (U3)	Wetwell (Torus)	N/A	5.37E+06	1.2	6.44E+06
1	HPCI Pump Room	519	2.10E+04	1.2	2.52E+04
2	Southwest RHR Pump Room	519	2.63E+05	1.2	3.15E+05
3	NW RCIC & Core Spray Room	519	2.25E+04	1.2	2.70E+04
4(U2)	NE Core Spray Room	519	5.78E+04	1.2	6.93E+04
4(U3)	NE Core Spray Room	519	2.70E+05	1.2	3.24E+05
5	SE RHR Pump Room	519	3.15E+05	1.2	3.78E+05
6(U2)	Pressure Suppression Chamber Room	519	1.05E+06	1.2	1.26E+06
6(U3)	Pressure Suppression Chamber Room	519	1.68E+06	1.2	2.02E+06
7	Main Steam Valve Vault	565	1.21E+07	1.0	1.21E+07
8	General Floor Area - 565	565	1.05E+06	1.2	1.26E+06
9A	RHR HX Rooms	593	6.32E+05	1.2	7.58E+05
(U1) 9B (U2)	General Floor Area - 593 Southwest	593	1.50E+06	1.2	1.80E+06
9B (U3)	General Floor Area - 593 Southwest	593	6.32E+05	1.2	7.58E+05
9C	General Floor Area - 593 NW	593	6.32E+05	1.2	7.58E+05
9D	General Floor Area - 593 NE	593	6.32E+05	1.2	7.58E+05
(U1) 9E (U2)	General Floor Area - 593 SE	593	1.50E+06	1.2	1.80E+06
9E(U3)	General Floor Area - 593 SE	593	6.32E+05	1.2	7.58E+05
10	RWCU Pump Room	593	1.31E+06	1.2	1.58E+06
11	RWCU Nonregenerative HX Room	593	8.70E+06	1.2	1.04E+07
12	General Floor Area - 621	621	6.30E+05	1.2	7.56E+05
13	General Floor Area - South	639	3.95E+04	1.2	4.73E+04
14	General Floor Area - North	639	1.05E+05	1.2	1.26E+05
15	Refueling Floor Area	664	2.23E+04	1.0	2.23E+04
16	RWCU Backwash Tank Room	593	4.31E+08	1.0	4.31E+08
17A	RWCU Drain Room A	639	N/A	N/A	N/A
17B	RWCU Drain Room A	639	N/A	N/A	N/A
18	RWCU Demineralizer Valve Room	621	4.50E+08	1.0	4.50E+08

Table 2.10-1a Current and Anticipated Measured Radiation Dose in Selected Areas of the Reactor Building for 60 Year Normal Operation (continued)

Zone Number	Description	Plant Elevation	CLTP Dose (rads)	Scaling Factor(1)	EPU Dose(1) (rads)
19	Drywell Access Area	565	1.05E+05	1.2	1.26E+05
20	TIP System Room	565	1.68E+07	1.2	2.02E+07
21	TIP Drive Room	565	N/A	N/A	N/A
22	SGTS Area	565	1.50E+03	1.0	1.50E+03
23	SGTS Stack	568 to 597.5	1.32E+04	1.2	1.58E+04

Notes

(1) This table contains a representative set of rooms across various floors and measured radiation levels at CLTP and EPU conditions.

Table 2.10-1b Current and Anticipated Measured Radiation Dose in Selected Areas of the Drywell

Elevation in The Drywell	CLTP Dose (rads)	Scaling Factor(1)	EPU Dose(2) (rads)
Normal Operating Dose in the Drywell			
Elevation Zone 549.2 to 639			
Neutron Dose	9.1E+04	1.2	1.1E+05
Gamma Dose	5.9E+07	1.2	7.1E+07
Total Dose	5.9E+07	N/A	7.1E+07
Post-Accident Gamma Dose in the Drywell			
Elevation Zone 549.2 to 585			
Airborne Gamma			
1 hour	3.3E+06	1.02	3.4E+06
1 Day	1.9E+07	1.0	1.9E+07
100 Days	4.3E+07	1.12	4.8E+07
Piping			
1 hour	1.8E+05	1.24	2.2E+05
1 Day	1.2E+06	1.22	1.5E+06
100 Days	6.4E+06	1.05	6.7E+06
Water on Floor			
1 hour	1.2E+05	1.24	1.5E+05
1 Day	8.3E+05	1.22	1.0E+06
100 Days	4.2E+06	1.05	4.4E+06
Plateout Gamma			
1 hour	2.6E+05	1.21	3.1E+05
1 Day	2.9E+06	1.22	3.5E+06
100 Days	8.7E+06	1.19	1.0E+07
Total Gamma			
1 hour	3.9E+06	N/A	4.1E+06
1 Day	2.4E+07	N/A	2.5E+07
100 Days	6.2E+07	N/A	7.0E+07

Table 2.10-1b Current and Anticipated Measured Radiation Dose in Selected Areas of the Drywell (continued)

Elevation in The Drywell	CLTP Dose (rads)	Scaling Factor(1)	EPU Dose(2) (rads)
Post-Accident Gamma Doses in the Drywell			
Elevation Zone 585 to 617			
Airborne Gamma			
1 Hour	3.2E+06	1.02	3.3E+06
1 Day	1.8E+07	1.0	1.8E+07
100 Days	4.2E+07	1.12	4.7E+07
Piping			
1 Hour	1.3E+05	1.24	1.6E+05
1 Day	9.4E+05	1.22	1.1E+06
100 Days	4.8E+06	1.05	5.0E+06
Water on Floor			
1 Hour	1.4E+04	1.24	1.7E+04
1 Day	9.6E+04	1.22	1.2E+05
100 Days	4.9E+05	1.05	5.1E+05
Plateout Gamma			
1 Hour	2.6E+05	1.21	3.1E+05
1 Day	2.9E+06	1.22	3.5E+06
100 Days	8.7E+06	1.19	1.0E+07
Total Gamma			
1 Hour	3.6E+06	N/A	3.8E+06
1 Day	2.2E+07	N/A	2.3E+07
100 Days	5.6E+07	N/A	6.3E+07

Table 2.10-1b Current and Anticipated Measured Radiation Dose in Selected Areas of the Drywell (continued)

Elevation in The Drywell	CLTP Dose (rads)	Scaling Factor[1]	EPU Dose[2] (rads)
Post-Accident Gamma Doses in the Drywell			
Elevation Zone 617 to 639			
Airborne Gamma			
1 Hour	1.8E+06	1.02	1.8E+06
1 Day	9.9E+06	1.0	9.9E+06
100 Days	2.2E+07	1.12	2.5E+07
Piping			
1 Hour	8.8E+04	1.24	1.1E+05
1 Day	6.2E+05	1.22	7.6E+05
100 Days	3.2E+06	1.05	3.4E+06
Water on Floor			
1 Hour	4.2E+03	1.24	5.2E+03
1 Day	3.0E+04	1.22	3.7E+04
100 Days	1.5E+05	1.05	1.6E+05
Plateout Gamma			
1 Hour	2.6E+05	1.21	3.1E+05
1 Day	2.9E+06	1.22	3.5E+06
100 Days	8.7E+06	1.19	1.0E+07
Total Gamma			
1 Hour	2.2E+06	N/A	2.3E+06
1 Day	1.3E+07	N/A	1.4E+07
100 Days	3.4E+07	N/A	3.9E+07

Table 2.10-1b Current and Anticipated Measured Radiation Dose in Selected Areas of the Drywell (continued)

Elevation in The Drywell	CLTP Dose (rads)	Scaling Factor[(1)]	EPU Dose[(2)] (rads)
Post-Accident Gamma Doses in the Drywell			
Elevation Zone 639 to 655			
Airborne Gamma			
1 Hour	1.7E+06	1.02	1.7E+06
1 Day	9.8E+06	1	9.8E+06
100 Days	2.2E+07	1.12	2.5E+07
Piping			
1 Hour	8.8E+04	1.24	1.1E+05
1 Day	6.2E+05	1.22	7.6E+05
100 Days	3.2E+06	1.05	3.4E+06
Water on Floor			
1 Hour	2.4E+03	1.24	3.0E+03
1 Day	1.7E+04	1.22	2.1E+04
100 Days	8.6E+04	1.05	9.0E+04
Plateout Gamma			
1 Hour	2.6E+05	1.21	3.1E+05
1 Day	2.9E+06	1.22	3.5E+06
100 Days	8.7E+06	1.19	1.0E+07
Total Gamma			
1 Hour	2.1E+06	N/A	2.2E+06
1 Day	1.3E+07	N/A	1.4E+07
100 Days	3.4E+07	N/A	3.8E+07

Notes:

(1) The post-accident scalars provided in this table are the maximum values based on the ratio of the integrated gamma doses for each type of fuel at EPU conditions divided by the integrated gamma doses at OLTP conditions based on the TID-14844 source.

(2) This table contains a representative set of rooms across various floors and measured radiation levels at CLTP and EPU conditions.

Table 2.10-1c Current and Anticipated Measured Radiation Dose in Selected Areas of the Turbine Building

Normal Operating Doses in the Turbine Building(1)			
Panel / Location	**CLTP(2) Dose mRads**	**Scaling Factor**	**EPU Dose mRads**
2-25-95	1.32E+06	1.2	1.58E+06
2-25-97	1.32E+06	1.2	1.58E+06
2-25-100A	1.32E+06	1.2	1.58E+06
2-25-101B	1.68E+06	1.2	2.02E+06
2-25-107	1.32E+06	1.2	1.58E+06
2-25-108	1.32E+06	1.2	1.58E+06
2-25-109	1.32E+06	1.2	1.58E+06
2-25-110	1.32E+06	1.2	1.58E+06
2-25-111	1.32E+06	1.2	1.58E+06
2-25-112	1.32E+06	1.2	1.58E+06
2-25-113A	1.32E+06	1.2	1.58E+06
2-25-113B	1.32E+06	1.2	1.58E+06
2-25-113C	1.32E+06	1.2	1.58E+06
2-25-115A	2.63E+06	1.2	3.15E+06
2-25-115B	1.32E+06	1.2	1.58E+06
2-25-115C	1.32E+06	1.2	1.58E+06
2-25-121	1.32E+06	1.2	1.58E+06
2-25-122	1.32E+06	1.2	1.58E+06
2-25-126	1.32E+06	1.2	1.58E+06
2-25-127	1.32E+06	1.2	1.58E+06
2-25-152	1.58E+06	1.2	1.90E+06
Battery Board 4	1.32E+06	1.2	1.58E+06
Battery Board 5	1.32E+06	1.2	1.58E+06
Battery Board 6	1.32E+06	1.2	1.58E+06
Turbine MOV Board 2A	1.32E+06	1.2	1.58E+06
Turbine MOV Board 2C	1.32E+06	1.2	1.58E+06

Table 2.10-1c Current and Anticipated Measured Radiation Dose in Selected Areas of the Turbine Building (continued)

Normal Operating Doses in the Turbine Building(1)			
Panel / Location	**CLTP(2) Dose mRads**	**Scaling Factor**	**EPU Dose mRads**
Battery Charger 0-CHGA-248-005	1.32E+06	1.2	1.58E+06
Battery Charger 0-CHGA-248-006	1.32E+06	1.2	1.58E+06
RCW Pump 2A	1.32E+06	1.2	1.58E+06
RCW Pump 2B	1.32E+06	1.2	1.58E+06
RCW Pump 2C	1.32E+06	1.2	1.58E+06
General Area Unit 2 Generator	1.32E+06	1.2	1.58E+06
Unit 2 Stator Cooling Unit	1.32E+06	1.2	1.58E+06
2-25-114	1.32E+06	1.2	1.58E+06
2-25-116A	1.32E+06	1.2	1.58E+06
2-25-116B	1.32E+06	1.2	1.58E+06
2-25-116C	1.32E+06	1.2	1.58E+06
2-25-150	1.32E+06	1.2	1.58E+06
2-25-151A	1.32E+06	1.2	1.58E+06
2-25-151B	1.32E+06	1.2	1.58E+06
2-25-151C	1.32E+06	1.2	1.58E+06
2-25-151D	1.32E+06	1.2	1.58E+06
2-25-151E	1.32E+06	1.2	1.58E+06
2-25-151F	1.32E+06	1.2	1.58E+06
2-25-151G	1.32E+06	1.2	1.58E+06
2-25-151H	1.32E+06	1.2	1.58E+06
2-25-151J	1.32E+06	1.2	1.58E+06
2-25-275	1.32E+06	1.2	1.58E+06
2-25-278	1.32E+06	1.2	1.58E+06
2-925-682	1.32E+06	1.2	1.58E+06
U2 Excitation Board	1.32E+06	1.2	1.58E+06
U1 Control Air Dryer Area	1.32E+06	1.2	1.58E+06
U2 Control Air Dryer Area	1.32E+06	1.2	1.58E+06

Table 2.10-1c Current and Anticipated Measured Radiation Dose in Selected Areas of the Turbine Building (continued)

Normal Operating Doses in the Turbine Building[1]			
Panel / Location	**CLTP[2] Dose mRads**	**Scaling Factor**	**EPU Dose mRads**
Post-Accident Gamma Doses in the Turbine Building			
The turbine building does not serve as a primary or secondary containment nor does it contain any equipment for safe shutdown; therefore, the Post-Accident Gamma Dose is not applicable.			

Notes

(1) This table contains a representative set of either panels or rooms across various floors and measured radiation levels at CLTP and EPU conditions.

(2) CLTP values are estimated based on 60-year normal dose, including EPU upscaling.

Table 2.10-2 Design Basis and Reported Annual Dose to Members of the Public

10 CFR 50 Appendix I Dose Analysis					
Type of Dose	**3,458 MWt (CLTP)**	**4,031 MWt (102% EPU[1])**	**Actual Plant Effluent Releases (Average From 2009-2013)**	**Projected EPU (1.20 x Average Actual)**	**10 CFR 50, Appendix I Dose Acceptance Criteria**
Liquid Effluents					
Maximum dose to total body from all pathways (mrem/yr)	4.20E-02	4.20E-02	1.40E-02	1.68E-02	9 (3 / Unit)
Maximum dose to any organ from all pathways (mrem/yr)	1.74E-01	1.89E-01	2.04E-02	2.45E-02	30 (10 / Unit)
Gaseous Effluents					
Gamma dose in air from noble gases (mrad/yr)	1.34E+00	1.36E+00	2.28E-06	2.74E-06	30 (10 / Unit)
Beta dose in air from noble gases (mrad/yr)	9.42E-01	9.54E-01	1.54E-06	1.85E-06	60 (20 / Unit)
Skin dose in air (mrem/yr)	2.45E+00	2.46E+00	2.52E-03	3.02E-03	45 (15 / Unit)
Total Body dose (mrem/yr)	1.46E+00	1.47E+00	2.15E-03	2.58E-03	15 (5 / Unit)
Maximum dose to organ – Thyroid (mrem/yr)	6.09E+00	6.24E+00	9.83E-02	1.18E-01	45 (15 / Unit)

Table 2.10-2 Design Basis and Reported Annual Dose to Members of the Public (continued)

<table>
<tr><th colspan="4">10 CFR 50 Appendix I Dose Analysis</th></tr>
<tr><th>Calculation Methodology</th><th>GALE-BG, BL ODCM Codes</th><th>ANSI/ANS-18.1-1984 BWR-GALE (Liquid and Gaseous) ODCM Codes Regulatory Guide 1.109, Revision 1 (Reference 60)</th><th>N/A</th></tr>
<tr><th colspan="4">10 CFR 20, Appendix B, Table 2 Liquid and Gaseous Effluent Concentration Analysis</th></tr>
<tr><td colspan="2">Liquid Effluents</td><td colspan="2">0.03(2)</td></tr>
<tr><td colspan="2">Gaseous Effluents</td><td colspan="2">0.42(2)</td></tr>
<tr><th colspan="4">Offsite Direct Dose Contributions</th></tr>
<tr><td colspan="2">Direct Radiation from ISFSI(3)</td><td colspan="2">13.8 mrem/yr</td></tr>
<tr><td colspan="2">Direct Radiation from CST(3)</td><td colspan="2">7.23 mrem/yr</td></tr>
<tr><td colspan="2">N-16 Shine</td><td colspan="2">Negligible</td></tr>
<tr><th colspan="4">40 CFR 190 Site Evaluation – 25 mrem/yr Limit</th></tr>
<tr><td colspan="2">Gaseous Effluent Releases for 3 Units</td><td colspan="2">1.47 mrem/yr (whole body)</td></tr>
<tr><td colspan="2">Gaseous Effluents from Leakage from ISFSI(4)</td><td colspan="2">0.742 mrem/yr (whole body)</td></tr>
<tr><td colspan="2">Liquid Effluent Releases for 3 Units</td><td colspan="2">0.042 mrem/yr (whole body)</td></tr>
<tr><td colspan="2">Total</td><td colspan="2">23.3 mrem/yr (whole body)</td></tr>
</table>

Note:

1. 4031 MWt is 102% of the EPU power level, 3952 MWt, and used in radiological calculations for conservative dose determination.
2. Sum of the fractions of allowed radionuclide concentrations in 10 CFR 20, Appendix B, Table 2; based on off-gas release rate of 100μCi/s/MWt.
3. Dose taken at a distance of 400 meters from the cask array, and based on extremely conservative photon spectrum and conservative estimate of time for an individual to be at the river location.
4. ISFSI leakage is from the four casks nearest to the river location.

2.11 Human Performance

2.11.1 Human Factors

Regulatory Evaluation

The area of human factors deals with programs, procedures, training, and plant design features related to operator performance during normal and accident conditions.

The NRC's acceptance criteria for human factors are based on GDC-19, 10 CFR 50.120, 10 CFR Part 55, and the guidance in GL 82-33.

Specific NRC review criteria are contained in SRP Sections 13.2.1, 13.2.2, 13.5.2.1, and 18.0.

Browns Ferry Current Licensing Basis

Final GDC-19 is applicable to Browns Ferry as described in "Browns Ferry Nuclear Plant, Units 1, 2, and 3 - Issuance of Amendments Regarding Full-Scope Implementation of Alternative Source Term," dated September 27, 2004 (Reference 72).

Technical Evaluation

Human factors engineering is an essential tool that helps ensure plant operators can effectively and safely operate the facility under normal, abnormal, and emergency conditions. When a plant change is initiated per the design configuration change process, an impact review is required by Operations and Training. The Operations review looks for procedures requiring revision and any training desired. Training reviews all plant modifications and procedure changes (except editorial changes) and assesses them using the systematic approach to training. Results of these reviews, including training requirements and physical/modeling changes needed for the simulator, are incorporated into the engineering change package and tracked to completion by the design change process. Additionally, design changes associated with the main control room receive a Human Factors Engineering review.

2.11.1.1 Changes in Emergency and Abnormal Operating Procedures

The changes in Emergency Operating Instructions (EOIs) and the Severe Accident Management Guidelines (SAMGs) reflect the change in power level but will not be changed in a manner that involves a change in accident mitigation philosophy.

The following Emergency Operating Instruction (EOI) curves and values have been identified as being affected:

- Heat Capacity Temperature Limit (HCTL) curve - The HCTL curve will be revised as a result of the increase in decay heat rejected to the SP. The change is not significant (approximately 1°F).
- Pressure Suppression Pressure (PSP) curve - The PSP curve will be revised as a result of the increase in decay heat rejected to the SP. The change is not significant (approximately 0.2 psig).
- Minimum Debris Retention Injection Rate (MDRIR) curve - The MDRIR curve will be revised as a result of the increase in decay heat loading. The injection flow will increase

by approximately 15% of the CLTP injection flow but will have minimal effect on the plant operators.

- Hot Shutdown Boron Weight (HSBW) and Cold Shutdown Boron Weight (CSBW) - The percentage of tank level that must be injected will change due to the increase in boron enrichment.
- Decay Heat Removal Pressure (DHRP) - The DHRP will be affected by the increase in decay heat loading. The pressure will increase by approximately 12 psig but will have minimal effect on the plant operators.

The following EOIs and Severe Accident Management Guidelines (SAMG)s are planned to be revised as a result of EPU (The modifications mentioned can be found in EPU LAR Attachment 47.):

- EOI-1 FLOWCHART and Bases, RPV CONTROL, are affected by the installation of the Emergency High Pressure Makeup Pump (EHPMP) modification.
- EOI-1A FLOWCHART and Bases, ATWS RPV CONTROL, are affected by changes to the Cold and Hot Shutdown Boron Weights and the installation of the EHPMP modification.
- EOI-2 FLOWCHART and Bases, PRIMARY CONTAINMENT CONTROL, are affected by the changes to the Heat Capacity Temperature Limit and the Pressure Suppression Pressure.
- EOI-5, CURVES AND CAUTIONS, is affected by changes to the Heat Capacity Temperature Limit and the Pressure Suppression Pressure.
- C-2 FLOWCHART and Bases, EMERGENCY RPV DEPRESSURIZATION, are affected by the changes to the Decay Heat Removal Pressure.
- C-2A FLOWCHART and Bases, ATWS EMERGENCY RPV DEPRESSURIZATION, are affected by the changes in the Cold Shutdown Boron Weight and the Decay Heat Removal Pressure.
- C-4 FLOWCHART and Bases, RPV FLOODING, are affected by changes to the Decay Heat Removal Pressure and by installation of the EHPMP modification.
- C-4A FLOWCHART and Bases, ATWS RPV FLOODING, are affected by changes to the Decay Heat Removal Pressure and by installation of the EHPMP modification.
- "Emergency High Pressure Makeup Pump," will be developed to allow operators to make-up to the RPV from the CST with a newly installed non-safety-related pump. (Future new procedure to be developed as part of implementation of NFPA 805.)
- "Hardened Wetwell Vent," will be developed to allow operators to vent the suppression pool air space through a hardened vent pipe that exhausts above the Reactor Building roof. (Future procedures will be developed to implement the requirements for the mitigation of beyond-design basis events.)
- SAMG-1, PRIMARY CONTAINMENT FLOODING, is affected by changes to the Pressure Suppression Pressure, the Minimum Debris Retention Injection Rate, and by installation of the EHPMP modification.

Abnormal Operating Procedures (AOPs) at Browns Ferry are defined as Abnormal Operator Instructions (AOIs), Annunciator Response Procedures (ARPs), Fire Safe Shutdown (FSSs), and select sections of Operator Instructions (OIs).

- AOI-3-1, "Loss of Reactor Feedwater or Reactor Water Level High/Low," will be revised to no longer require operators to immediately lower reactor power level to 80%, by reducing Recirculation Pump flow, in order to avoid a low reactor water level scram. The Condensate/Condensate Booster/Feedwater pumps were all changed from 1/3 capacity to 1/2 capacity pumps. The loss of certain single pumps or pump combinations, at EPU conditions, can require a power reduction of as much as 7% to re-establish desired NPSH ratios or to reduce the Condensate Booster Pump horsepower (still within its service factor) back to within its nameplate rating.
- AOI 47-3, "Loss of Condenser Vacuum," was revised to reflect the modification that replaced condenser low vacuum pressure switches with pressure transmitters. These transmitters provide Turbine Trip and Bypass Valve Trip inputs to the Electro-Hydraulic Control System and provide condenser vacuum indication in the CR via an existing recorder. Now the trips and operator indication will originate from the same transmitter and are therefore aligned.
- AOI-57-1A, "Loss of Offsite Power (161 and 500 kV)/Station Blackout (SBO)," will be revised for the following:
  1) To incorporate a time sensitive operator action:
     a. Crosstie of the Containment Atmospheric Dilution system to the Drywell Control Air System approximately 2 hours into the SBO scenario.
  2) To incorporate changes to the SBO HCTL and SBO Pressure Suppression Pressure curves for the SBO scenario.
- AOI 57-1E, "Grid Instability," will be revised to take into account the uprated Turbine/Generator output capability for each of the units:

  Unit 1: 1,330 MWe at 0.95 Power Factor

  Units 2 and 3: 1,332 MWe at 0.93 Power Factor
- OI-47, "Turbine Generator System," required revision to reflect that the rewound Main Generator will be required to be tripped within 60 seconds of a loss of Stator Cooling Water. The previous time requirement was 70 seconds.

EOIs and AOPs will also be rescaled as required to reflect the power uprate.

2.11.1.2 Changes to Operator Actions Sensitive to Power Uprate

Most abnormal events result in automatic plant shutdown (scram). Some abnormal events result in SRV actuation, ADS actuation and/or automatic ECCS actuation. All analyzed events result in safety-related SSCs remaining within their design limits. EPU does not change any automatic safety function. Changes to subsequent operator actions are as follows.

2.11.1.2.1 Changes for Design Basis Accidents (DBA) and Events

An EHPMP will be installed as an additional option for operators to provide water to the RPV in the EOI and SAMG procedures.

AOI-57-1A, "Loss of Offsite Power (161 and 500 kV)/Station Blackout (SBO)," will be revised to incorporate a time sensitive action as a result of EPU:

1) Crosstie of the Containment Atmospheric Dilution system to the Drywell Control Air System approximately 2 hours into the SBO scenario.

This time sensitive operator action is a simple task, requires a small time duration to perform (< 10 minutes), is performed in the control room (CR), and will easily be able to be successfully performed within the 2 hour required timeframe. As such, time validation of this action is not necessary.

2.11.1.2.2 Fire Safe Shutdown (FSS) Events

The purpose of the FSS procedures is to provide supplemental instruction, including the manual recovery actions needed, in conjunction with the EOIs should an EOI initiating condition exist, to ensure safe shutdown of Unit 1, Unit 2 or Unit 3, or all three units if conditions require, in the event of a disabling fire event. Existing plant procedures are written to support an Appendix R fire at EPU conditions. The NFPA 805 FSS procedures will allow the plant to use all available equipment until proven unreliable with the exception of annunciators. Credited equipment is equipment which has the highest probability to remain functional in the fire event. Use of the credited, or preferred, equipment is desired in that it is least likely to be affected due to a fire related event.

Attachment 47 of the EPU LAR provides a listing and discussion of the modifications planned for EPU. The effect of these modifications on the Browns Ferry Fire Protection Program will be evaluated, in accordance with TVA's configuration change process, prior to EPU implementation. Per the process, these modifications will be evaluated to assure the changes do not affect the approved Fire Protection Program and will not adversely affect the ability to achieve and maintain safe shutdown in accordance with the current Browns Ferry license conditions and procedures.

As the FSSs are symptom based, the implementation of EPU does not change how the FSSs will be implemented or executed. The EHPMP will be available as an additional option for operators to provide water to the RPV. No operator actions need to be performed more quickly as a result of EPU implementation.

2.11.1.2.3 Anticipated Transient Without Scram Event

An EHPMP is not credited but will be available as an additional option for operators to provide water to the RPV. There will also be changes to the cold and hot shutdown boron weights and the decay heat removal pressure. No operator actions need to be performed more quickly as a result of EPU implementation.

2.11.1.2.4 Conclusion

The changes to Browns Ferry operator actions, as a result of the EPU, are small in number. There is only one time sensitive operator action. This action is a simple task, requires a small time duration to perform (< 10 minutes), is performed in the control room and will easily be able to be successfully performed within the two hour required timeframe. The changes to operator actions will be reflected in the procedures and the operators will receive appropriate classroom and/or simulator training prior to EPU implementation. There are no new or revised operator workarounds as a result of EPU.

2.11.1.3 Changes to Control Room Controls, Displays and Alarms

Changes to the CR are prepared in accordance with the plant design change process. Under this process, a Human Factors engineering review is performed for changes associated with the Browns Ferry CR. The change process also requires a review by Operations and Training personnel. Results of these reviews, including simulator effect and training requirements, are incorporated into the engineering change package and tracked to completion by the design change process.

The following changes have been/will be made to the CR Controls, Displays and / or Alarms resulting from EPU:

- Controls will be installed in the Control Room that will allow the operators to start an EHPMP that will take suction from the CST and discharge to the RPV.
- The Turbine First Stage Pressure Scram Bypass setpoint and associated alarm will be changed.
- Condenser pressure transmitters have been installed and the Turbine Trip and Turbine-Bypass Trip signal and alarms originate from these instruments as does the control room indication for the operators.
- Changes will be made to the Rod Worth Minimizer to reflect EPU conditions. A new runback for the Recirculation Pumps to 75% speed will occur on a scram signal to prevent water level from reaching the Level 2 setpoint.
- Removed the SJAE auto-start capability and replaced the HS-150/152 three-position switches with two-position switches.
- The control switches for the former Moisture Separator Drain Pumps that were removed now operate Moisture Separator Isolation Valves. A seventh control switch allows condensate to be injected into the Moisture Separator Drain Line.
- Controls for an additional Bus Duct Cooler Fan have been installed.

TS instruments for instrument and control systems are affected by EPU as described in the Enclosure to the EPU LAR and Attachments 2 and 3.

2.11.1.3.1 Conclusion

The changes to Browns Ferry CR interfaces as a result of the EPU do not significantly affect operator human performance. Operator training for changes to CR interfaces, alarms, and

indications will be accomplished in accordance with the plant training and simulator program as described in Section 2.11.1.5.

2.11.1.4 Changes to the Safety Parameter Display System (SPDS)

The purpose of the Browns Ferry Safety Parameter Display System (SPDS) is to continuously display information from which plant safety status can be readily and reliably assessed. The principal function of the SPDS is to aid CR personnel during abnormal and emergency conditions in determining the safety status of the plant and in assessing whether abnormal conditions warrant corrective action by operators to avoid a degraded core.

The following changes will be made to the SPDS as a result of the Browns Ferry EPU:

- HCTL curve: The HCTL curve will be revised as a result of the additional decay heat rejected to the suppression pool.
- PSP curve: The PSP curve will be revised as a result of the increase in decay heat rejected to the suppression pool.

2.11.1.4.1 Conclusion

The changes to the Browns Ferry SPDS as a result of the EPU do not significantly affect operator actions and mitigation strategies. The changes will be made in accordance with the configuration change process and the operators will receive appropriate classroom and/or simulator training prior to implementation.

2.11.1.5 Changes to the Operator Training Program and the Control Room Simulator

Training of Operations personnel will occur on all EPU modifications necessary to support unit operation at EPU conditions. The operator training is presented in the classroom and on the simulator. The major EPU change for the CR operators involves the installation of higher capacity condensate/condensate booster/feedwater pumps (see EPU LAR Attachment 47).

Licensed and non-licensed operator training will be provided prior to the cycle implementing the changes and will focus on plant modifications, procedure changes, startup test procedures, and other aspects of EPU including changes to parameters, setpoints, scales, and systems. The applicable lesson plans will be revised to reflect changes as a result of the EPU. Simulator training during this phase will also include training on performance effects of new modifications; this will support the power ascension plan. The training includes evaluation tools such as written exams, simulator evaluations, and task performance tools as deemed appropriate. Successful completion of training is verified, as required by plant procedures, as part of the turnover of the modification to operations. Prior to startup, following the refueling outage for EPU, the operators will be given classroom and simulator Just-In-Time (JIT) training to cover last minute training items and perform startup training and startup testing evolutions on the simulator.

Browns Ferry has two control room simulators. One of the simulators is a duplicate of the Unit 2 CR and the other is a duplicate of the Unit 3 CR. The simulators are modified whenever modifications, affecting simulator fidelity, are installed in the plant. Unit 1 is very similar to

Units 2 and 3 and the differences for Unit 1 are covered with classroom training and/or with equipment mock-ups.

Installation of the EPU changes to the simulator are performed in accordance with ANSI/ANS-3.5 1985, "Nuclear Power Plant Simulators for Use in Operator Training" (Reference 109). The simulator changes will include hardware changes for new and modified CR Instrumentation and Control (I&C), software updates for modeling changes due to EPU (i.e., Condensate/Condensate Booster/Feedwater pump modifications), setpoint changes, and re-tuning of the core physics model for cycle specific data. The simulator process computer will be updated for EPU modifications.

Operating data will be collected during EPU implementation and start-up testing. This data will be compared to simulator data as required by ANSI/ANS-3.5 1985. Additionally, simulator acceptance testing will also be conducted to benchmark the simulator performance based on design and engineering analysis data.

Lessons learned from power ascension testing and operation at EPU conditions will be fed back into the training process to update the training material and processes as required.

Conclusion

TVA has evaluated the changes to operator actions, human-system interfaces, procedures, and training required for the proposed EPU and (1) appropriately accounted for the effects of the proposed EPU on the available time for operator actions and (2) taken appropriate actions to ensure that operator performance is not adversely affected by the proposed EPU. The evaluation concludes that Browns Ferry will continue to meet the requirements of final GDC-19, 10 CFR 50.120, and 10 CFR Part 55 following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to the human factors aspects of the required system changes.

2.12 Power Ascension and Testing Plan

2.12.1 Approach to EPU Power Level and Test Plan

Regulatory Evaluation

The purpose of the EPU test program is to demonstrate that SSCs will perform satisfactorily in service at the proposed EPU power level. The test program also provides additional assurance that the plant will continue to operate in accordance with design criteria at EPU conditions.

The NRC's acceptance criteria for the proposed EPU test program are based on 10 CFR Part 50, Appendix B, Criterion XI, which requires establishment of a test program to demonstrate that SSCs will perform satisfactorily in service.

Specific NRC review criteria are contained in SRP Section 14.2.1.

Browns Ferry Current Licensing Basis

Browns Ferry UFSAR, Section 13.4, "Pre-Operational Test Program," provides an overview of the initial power ascension test program.

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of EPUs. Section 10.4 of the CLTR addresses the testing required for the initial power ascension following the implementation of EPU. The results of this evaluation are described below.

Testing is required for the initial power ascension during implementation of EPU. A standard set of tests is established for the initial power ascension steps of EPU, which supplement the normal TS testing requirements. The EPU testing program at Browns Ferry is based on the Browns Ferry specific initial EPU power ascension and TSs. 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.

2.12.1.1 Testing Program

Browns Ferry meets all CLTR dispositions. The topics addressed in this section are:

Topic	CLTR Disposition	Browns Ferry Result
Testing Program	Generic	Meets CLTR Disposition

The CLTR states that the increase in power level changes plant and system performance.

Based on the analyses and GEH BWR experience with uprated plants, a standard set of tests has been established for the initial power ascension steps of EPU. Testing will be done in

accordance with the TS surveillance requirements on instrumentation that is re-calibrated for EPU conditions. These tests supplement the normal TS testing requirements.

Overlap between the IRM and APRM will be assured.

Steady-state data will be taken at points from 90% up to 100% of the CLTP RTP, so that system performance parameters can be projected for EPU power before the CLTP RTP is exceeded.

EPU power increases above the 100% CLTP RTP will be made along an established flow control/rod line in increments of equal to or less than 5% power. Steady-state operating data, including fuel thermal margin, will be taken and evaluated at each step. Routine measurements of reactor and system pressures, flows, and vibration will be evaluated from each measurement point, prior to the next power increment. Radiation measurements will be made at selected power levels to ensure the protection of personnel.

Control system tests will be performed for the reactor FW/reactor water level controls and pressure controls. These operational tests will be made at the appropriate plant conditions for that test at each of the power increments, to show acceptable adjustments and operational capability. Testing will be done to confirm the power level near the turbine first-stage scram bypass setpoint.

Details on the FIV monitoring program are provided in Attachment 45 to the EPU license amendment request.

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

]] Vibrational testing is addressed in Attachment 45 to the EPU license amendment request.

The EPU testing program at Browns Ferry, which is based on the specific testing required for the Browns Ferry initial EPU power ascension, supplemented by normal TS testing, meets all CLTR dispositions. Attachment 46 to the EPU license amendment request contains details of the testing program. The Browns Ferry power ascension testing program will provide management oversight and control to assure Browns Ferry can operate safely at the EPU licensed power level. Management review and approval of test results at each power level will be provided prior to increasing power to the next level.

2.12.1.2 Transient Tests and Modifications

Large transient testing is normally performed on new plants because experience does not exist to confirm a plant's operation and response to events. However, these tests are not normally performed for plant modifications following initial startup because of a rigorous design control process, well-established QA/QC and maintenance programs including component and system level post-modification testing and extensive experience with general behavior of unmodified equipment. When major modifications are made to the plant, large transient testing may be

needed to confirm that modifications were correctly implemented. However, such testing should only be imposed if it is deemed necessary to demonstrate safe operation of the plant.

Browns Ferry does not intend to perform large transient testing as part of EPU implementation. The justification for not performing large transient testing is provided in EPU LAR Attachment 46. This justification will confirm: a) all plant modifications have been evaluated and implemented properly, and b) integrated plant performance and transient operation is consistent with the completed analyses. Transient experience at high powers at operating BWR plants has shown a close correlation of the plant transient data to the evaluated events. The operating history of Browns Ferry demonstrates previous transient events from full power are within expected peak limiting values. The transient analysis performed for the Browns Ferry EPU demonstrates all safety criteria are met and this uprate does not cause any previous non-limiting events to become limiting. [[

]] on a plant-specific basis. Some instrument setpoints were changed. The instrument setpoints that were changed (see Table 2.4-1) do not contribute to the response to large transient events. [[

]] Should any future large transients occur, Browns Ferry procedures require verification the actual plant response is in accordance with the predicted response. Existing plant event data recorders are capable of acquiring the necessary data to confirm the actual versus expected response.

Further, [[

]] In addition, the limiting transient analyses are included as part of the each cycle's reload analysis.

Refer to Attachment 46 in the LAR for additional detail about the EPU testing program.

Conclusion

TVA has provided the EPU test program, including plans for the initial approach to the proposed maximum licensed thermal power level and transient testing necessary to demonstrate that plant equipment will perform satisfactorily at the proposed increased maximum licensed thermal power level. The proposed EPU test program provides adequate assurance that Browns Ferry will operate in accordance with design criteria and that SSCs affected by the proposed EPU, or modified to support the proposed EPU, will perform satisfactorily in service. Therefore, the proposed EPU test program is acceptable.

2.13 Risk Evaluation

2.13.1 Risk Evaluation of EPU

Regulatory Evaluation

TVA conducted a risk evaluation to: (1) demonstrate the risks associated with the proposed EPU are acceptable; and (2) determine if "special circumstances" are created by the proposed EPU. As described in Appendix D of SRP Section 19.2, special circumstances are present if any issue would potentially rebut the presumption of adequate protection provided by Browns Ferry to meet the deterministic requirements and regulations. TVA's review covered the effect of the proposed EPU on core damage frequency (CDF) and large early release frequency (LERF) for the plant due to changes in the risks associated with internal events, external events, and shutdown operations. The NRC's risk acceptability guidelines are contained in RG 1.174 (Reference 110). In addition, TVA's review covered the quality of the risk analyses used by Browns Ferry to support the application for the proposed EPU. This included a review of Browns Ferry's actions to address issues or weaknesses that have been raised in previous industry reviews of the probabilistic risk assessment (PRA) models, various self-assessments, and in a recent peer review that was performed in accordance with the combined ASME/ANS PRA Standard (Reference 111).

Technical Evaluation

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of EPUs. Section 10.5 of the CLTR addresses the effect of EPU on CLTR Individual Plant Evaluation. The results of this evaluation are described below.

The Browns Ferry PRA analysis covers both internal and external events. The Browns Ferry PRA is used to compare CLTP and post-EPU plant design and operation. A combination of quantitative and qualitative methods is used to assess the potential risk effects of EPU from internal and external events hazards. The details and results from this assessment are included as Attachment 44 to the Browns Ferry EPU LAR. The scope of Attachment 44 includes the complete risk contribution associated with EPU at Browns Ferry. The evaluation in Attachment 44 addresses initiating event frequency, component reliability, success criteria, operator response, external events, shutdown risk, and PRA quality in detail. Risk effects due to internal events, internal flooding, and internal fires are quantitatively assessed using the Browns Ferry PRA Revision 6 model of record. External events are evaluated by assessing the effect of EPU on the qualitative analyses of the Browns Ferry Individual Plant Examination of External Events (IPEEE) submittal. The effects on shutdown risk contributions are evaluated on a qualitative basis. The results are consistent with the CLTR description and analysis of this topic.

The effect of EPU on internal initiating events and internal fire events risk have been assessed by reviewing the changes in plant design and operations resulting from EPU. The changes have been mapped to appropriate elements of the PRA and modified as needed to estimate the risk

effect (CDF and LERF) of the post-EPU plant. The EPU is estimated to increase the total CDF and LERF as follows:

Unit	CDF_{CLTP}	CDF_{EPU}	ΔCDF	$LERF_{CLTP}$	$LERF_{EPU}$	ΔLERF
1	5.91E-05	6.08E-05	1.69E-06	7.96E-06	8.73E-06	7.74E-07
2	5.96E-05	6.14E-05	1.74E-06	7.99E-06	8.65E-06	6.63E-07
3	6.47E-05	6.64E-05	1.67E-06	7.18E-06	7.72E-06	5.45E-07

Using the risk acceptance guidelines established in NRC Regulatory Guide 1.174 (Reference 110) and the calculated results from the Level 1 and 2 PRA, the best estimate for the Browns Ferry CDF and LERF risk increase due to the EPU for Unit 1, 2 and 3 is in Region II (i.e., "small" risk changes). Additionally, based on the information available for external event effects, it is estimated that the incorporation of these contributors would not change this conclusion.

The sensitivity cases performed in this analysis, which included internal initiating events and internal fire events, also showed that delta CDF and delta LERF remain within Region II of the risk acceptance guidelines described in Regulatory Guide 1.174 (Reference 110).

Conclusion

TVA has evaluated the risk implications associated with the implementation of the proposed EPU. The evaluation indicates that the risks associated with the proposed EPU are acceptable and do not create the "special circumstances" described in Appendix D of SRP Section 19.2. Therefore, the risk implications of the proposed EPU are acceptable.

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APPENDIX A

EPU fire event evaluation based on the current fire protection program in accordance with 10 CFR 50.48(b) and 10 CFR 50, Appendix R requirements.

2.5.1.4 Fire Protection

Technical Evaluation

2.5.1.4.1 Fire Protection Program

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

As explicitly stated in Section 6.7 of the CLTR, [[

]] Therefore, the reactor and containment responses and operator actions were evaluated on a plant-specific basis for EPU.

This section addresses the effect of EPU on the FPP, fire suppression and detection systems, and reactor and containment system responses to postulated 10 CFR 50 Appendix R fire events. Browns Ferry meets all CLTR dispositions.[(1)] The topics addressed in this evaluation are:

Topic	CLTR Disposition	Browns Ferry Result
Fire Suppression and Detection Systems	Plant Specific	Meets CLTR Disposition
Operator Response Time	Plant Specific	Meets CLTR Disposition[(1)]
Peak Cladding Temperature	Plant Specific	Meets CLTR Disposition[(1)]
Vessel Water Level	Plant Specific	Meets CLTR Disposition[(1)]
Suppression Pool Temperature	Plant Specific	Meets CLTR Disposition[(1)]

(1) The Browns Ferry Fire Protection Program has been analyzed at EPU conditions. These analyses show that Browns Ferry meets all CLTR dispositions. However, Browns Ferry currently is not in compliance with 10 CFR 50 Appendix R and is presently employing compensatory measures, allowed by discretionary enforcement, while the plant transitions to NFPA 805 fire protection requirements (Reference 65).

The higher decay heat associated with EPU may reduce the time available for the operator to perform the actions necessary to achieve and maintain cold shutdown conditions. The higher decay heat also may result in higher suppression pool temperatures, in lower vessel water levels or higher peak cladding temperatures (PCTs), depending on the plant-specific analysis basis. A plant-specific evaluation was performed to demonstrate safe shutdown capability in compliance with the requirements of 10 CFR 50 Appendix R assuming EPU conditions. The evaluation determined the effect of EPU on fuel cladding integrity, reactor vessel integrity, and containment integrity as a result of the fire event.

[[

]] The EPU requires no new operator actions for fire safe shutdown of the plant and there are no operator manual actions required inside the primary containment.

The effect of EPU on pump NPSH during a fire event is discussed in Section 2.6.5.2.

Browns Ferry does not take credit in any safety analysis for the fire protection system other than for fire protection activities. Procedural guidance is provided under Emergency Operator Instructions (EOIs), Severe Accident Management Guidelines (SAMGs), and Safe Shutdown Instructions (SSIs), for utilizing fire protection system pumps to supply water to the reactor, spent fuel storage pools, the drywell, or the suppression chamber, if necessary. However, this use of the non-safety related fire protection system is not credited in any safety analysis, and EPU operation will not require any changes to these procedures regarding the utilization of the fire protection system.

The reactor and containment responses to the postulated 10 CFR 50 Appendix R fire events at EPU conditions are provided in Section 2.5.1.4.2. The results show that the peak fuel cladding temperature, reactor water level, and suppression pool temperature are within the acceptance limits. There is an analytical reduction of five minutes from CLTP conditions for the operators to perform the necessary actions to achieve and maintain safe shutdown conditions; however the actual time, procedurally stipulated for this action, remains unchanged. Cold shutdown is achieved well within the 72 hours required by Appendix R.

Therefore, with consideration of the information contained in footnote (1) from the table under Section 2.5.1.4.1, the Fire Protection Program at Browns Ferry meets all CLTR dispositions.

2.5.1.4.2 Fire Event

The limiting Appendix R fire events were analyzed under EPU conditions. The fuel heatup analysis was performed using the NRC approved AREVA LOCA Methodology (RELAX/HUXY). The containment analysis was performed using the NRC approved GEH SHEX model (Reference 7). These evaluations were used to determine the effect of EPU on fuel cladding integrity, reactor vessel integrity, and containment integrity as a result of the fire event.

The four shutdown methods defined in the Browns Ferry Fire Protection Report are as follows:

Case 1: No spurious operation of plant equipment occurs and the operator opens three MSRVs at 25 minutes into the event.

Case 2: One MSRV opens immediately due to a spurious opening signal generated as a result of the fire. The MSRV is reclosed at 10 minutes into the event due to the operator action. The operator opens three MSRVs at 20 minutes into the event.

Case 3: One MSRV opens immediately as in Case 2, but remains open throughout the event. The operator opens three MSRVs at 20 minutes into the event.

Case 4: MSRVs are used to maintain a controlled reactor cooldown with HPCI providing reactor makeup inventory. As reactor pressure decreases, HPCI will trip and make-up inventory is provided by RHR in LPCI alignment while continuing in SPC mode. To achieve cold shutdown, RHR is realigned at 12 hours into SDC mode.

The bounding PCT for Browns Ferry is seen in shutdown method Case 1 with one RHR pump in LPCI mode. The peak PCT is 1119°F which is less than the 1500°F acceptance criteria. (See FUSAR Section 2.5.1.4.2).

The highest peak suppression pool temperature (SPT) for Browns Ferry occurs in shutdown method Case 1. This case follows the assumption, stated in the Browns Ferry Fire Protection Report, that offsite power is assumed to be unavailable during a fire event. While performing analyses for Browns Ferry's NFPA 805 transition, it was learned that a slight revision to Case 1, called Case Max SPT, bounds Case 1 with respect to effect on containment parameters. Case Max SPT is outlined below.

Case Max SPT (See Tables 2.5-1, 2.5-2, 2.5-3 and Figure 2.5-1)

As part of the NFPA 805 transition, a variation of Case 1 (identified as Case Max SPT) was generated to examine the scenario involving no loss of offsite power and the continued injection of condensate into the RPV until the hotwell inventory is exhausted. Case Max SPT bounds Case 1 with respect to effect on containment parameters. Identical to Case 1 except instead of having one RHR pump aligned in the LPCI mode at 20 minutes, a condensate pump is allowed to maintain vessel inventory until the hotwell contents are exhausted (approximately 40 minutes). Then one RHR pump is aligned in the LPCI mode.

If offsite power were available, which is possible for fires in some areas, then it would be possible for a condensate pump to inject the hotwell volume to the RPV via the condensate/feedwater system. This inventory, approximately 90,000 gallons, would get heated by the piping and reactor core and eventually relieved to the suppression pool through the MSRVs when the RHR pump injects into the vessel in the LPCI/ASDC mode. The peak SPT for Case Max SPT is 208°F and this meets the containment integrity acceptance criteria of <281°F and the torus attached piping limit of <223°F (See Section 2.2.2.2.2.2).

This is the worst-case scenario for peak SPT and is used as an input in the analysis for available NPSH for the safe shutdown system pumps. Analyses show that containment accident pressure credit is not required to ensure adequate pump NPSH to mitigate a fire event. (See Section 2.6.5.2 and EPU LAR Attachment 39.)

The results of Case Max SPT, the evaluations in Section 2.6.5.2, FUSAR Section 2.5.1.4.2 and EPU LAR Attachment 39, demonstrate that the peak fuel cladding temperature, vessel water level, and

suppression pool temperature meet the acceptance criteria and the actual time stipulated for the operators to perform the necessary actions is unchanged from CLTP conditions. With the maximum suppression pool temperature of 208°F using the RHR heat exchanger K-value (307 Btu/sec-°F) in EPU LAR Attachment 39, the time to reach cold shutdown is within the 72 hours required by Appendix R. Therefore, EPU has no adverse effect on the ability of the systems and personnel to mitigate the effects of a fire event and satisfies the requirement of achieving and maintaining safe shutdown conditions in the event of a fire.

Table 2.5-1 Appendix R Fire Case Max SPT (EPU) Fire Event Key Inputs

Input Parameters	Values
Reactor Thermal Power	3952 MWt
RPV Dome Pressure	1055 psia
Decay Heat	ANS 5.1-1979 without 2σ uncertainty adder and with GEH SIL 636 recommendations
Initial Suppression Pool Liquid Volume	124,200 ft^3
Initial Suppression Pool and Wetwell Airspace Temperature	95 °F
Initial Wetwell Pressure	14.4 psia
Initial Drywell Pressure	15.5 psia
Initial Drywell Temperature	150 °F
Initial Wetwell Relative Humidity	100%
Initial Drywell Relative Humidity	20%
Drywell and Wetwell and Pool Heat Sinks Modeled	Yes
Drywell Heat Load Modeled	Yes
RHR Service Water Temperature	92 °F
RHR Heat Exchanger "K" Factor per Loop	307 Btu/sec-°F
Number of RHR Loops Available	1
Number of RHR Pump in one RHR Loop	1
ASDC RHR Flow Rate	7,500 gpm
Condensate Available for Injection	90,000 gallons

Table 2.5-2 Appendix R Fire Event Evaluation Results

Parameter	CLTP (105% of OLTP)	EPU Shutdown Method Case 1 (120% of OLTP)	Appendix R Criteria
Peak Fuel Cladding Temperature (°F)	Note 1	1119 Case 1	1500
Maximum Reactor Pressure at Vessel Bottom Head (psig)	Note 1	1224 Case 1	1375
Maximum Drywell Pressure (psig)	Note 1	15 Case 4	56
Maximum Drywell Temperature (°F)	Note 1	<281[2] Case 4	281
Bulk Suppression Pool Temperature (°F)	Note 1	208 Case Max SPT	Note 3

Notes:

1. Formal calculations for the Appendix R Fire Event were not performed at CLTP conditions.
2. Assuming no drywell cooling is available, the drywell air temperature would exceed 281°F at 18.5 hours. In the implementing SSIs for this safe shutdown pathway, RHR shutdown cooling is entered at 12 hours, which will result in a rapid decrease in reactor pressure, water saturation temperature, and vessel temperature. Therefore, the primary containment wall temperature will not be exceeded using this fire safe shutdown pathway. Entry into ASDC mode at 12 hours would have a similar, but more immediate effect in reducing drywell temperature because water at suppression chamber temperature is pumped to the reactor.
3. The bulk suppression pool temperature must be low enough to assure adequate suppression capability during reactor depressurization and to assure adequate net positive suction head for the systems using the suppression pool as a water source.

Table 2.5-3 Appendix R Fire Case Max SPT (EPU) Sequence of Events

Approximate Elapsed Time	Events
0 seconds	• Reactor scram occurs. • Main Steam Isolation Valves (MSIVs) start to close. • Feedwater pump is tripped. • Drywell coolers are tripped. • Condensate system continues to operate.
3.5 seconds	MSIVs are fully closed. After isolation, MSRVs automatically start to open and close to maintain RPV pressure.
25 min	Begin rapid depressurization using 3 MSRVs. RPV makeup is supplied by the Condensate system.
~ 40 min	Condensate inventory available for injection is depleted. Operators secure condensate flow and initiate ASDC using 7,500 gpm of RHR flow in the LPCI mode.
2 hours	RHR heat exchanger is placed into service.
72 hours	Event is terminated.

Table 2.6-3 Browns Ferry Peak Suppression Pool Temperature for Postulated ATWS, Station Blackout, and Appendix R Fire Events

Event	Peak Suppression Pool Temperature
Limiting ATWS (Loss of Offsite Power)	173.3°F
Station Blackout	203.7°F
Appendix R Fire	208.0°F

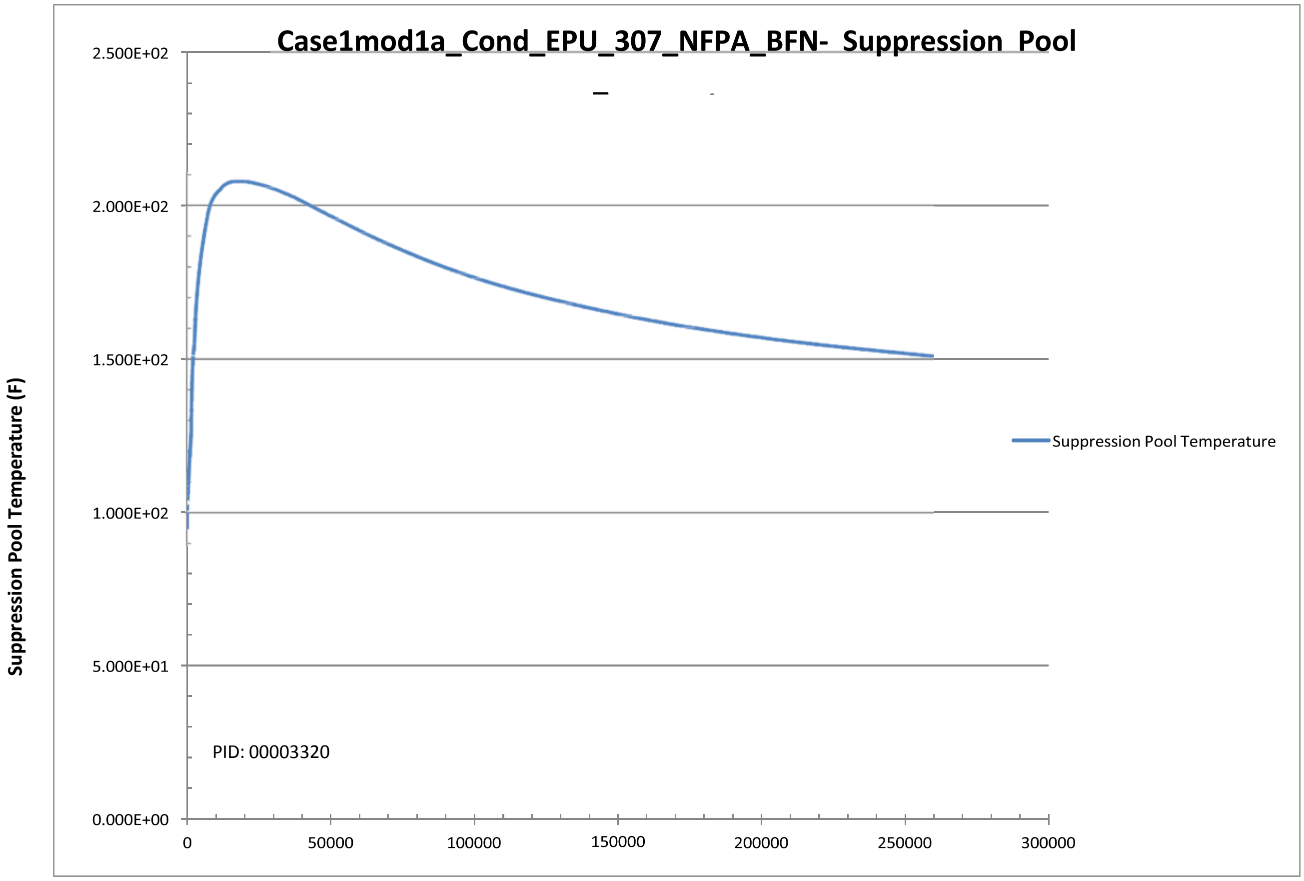

Figure 2.5-1 Appendix R Case Max SPT (EPU) Fire Event Suppression Pool Temperature