ATTACHMENT 9

ANP-3403NP, Fuel Uprate Safety Analysis Report for Browns Ferry Units 1, 2, and 3 (Non-Proprietary)

Fuel Uprate Safety Analysis Report for Browns Ferry Units 1, 2, and 3

ANP-3403NP
Revision 2

August 2015

AREVA Inc.

Nature of Changes

Item	Section(s) or Page(s)	Description and Justification
1	14	Added Technical Evaluation – Appendix R
2	24	Added Table 2.5-1. Renumbered existing table to Table 2.5-2
3	25	Added Figures 2.5-1 and 2.5-2. Renumbered existing figures to Figures 2.5-3 and 2.5-4
4	109	Adjust proprietary brackets
5	159	Added Reference 52 to the text
6	160	Updated footnote
7	165	Updated References 37, 38, and 39 to latest revisions
8	166	Added Reference 52

Contents

Tables

Figures

NOMENCLATURE

Abbreviation	Description
2PT	2 Recirculation Pump Trip
ABA	Amplitude Based Algorithm
AC	Alternating Current
ACE	AREVA's advanced critical power correlation
ADS	Automatic Depressurization System
AEC	Atomic Energy Commission
AL	Analytical Limit
ANS	American Nuclear Society
APLHGR	Average Planar LHGR
APRM	Average Power Range Monitor
AOO	Anticipated Operational Occurrence
AOT	Abnormal Operational Transient
ARI	All Rods In
ASME	American Society of Mechanical Engineers
AST	Alternative Source Term
ATWS	Anticipated Transient Without Scram
ATWS-PRFO	Anticipated Transient Without Scram – Pressure Regulator Failure Open
ATWS-RPT	Anticipated Transient Without Scram – Recirculation Pump Trip
AV	Analytical Value
AVZ	Above Vessel Zero
BFN	Browns Ferry Nuclear Plant
BLEU	Blended Low Enriched Uranium
BOC	Beginning Of Cycle
BOP	Balance of Plant
BPWS	Banked Position Withdrawal Sequence
BSP	Backup Stability Protection
BWR	Boiling Water Reactor

Abbreviation	Description
BWROG	Boiling Water Reactor Owners Group
CAD	Containment Atmosphere Dilution
CFR	Code of Federal Regulations
CGU	Commercial Grade Uranium
CLTP	Current Licensed Thermal Power (3458 MWt)
CMWR	Core-Wide Metal-Water Reaction
COLR	Core Operating Limits Report
CPR	Critical Power Ratio
CRD	Control Rod Drive
CRDA	Control Rod Drop Accident
CSA	Criticality Safety Analysis
DBA	Design Basis Accident
DIVOM	Delta over Initial MCPR Versus Oscillation Magnitude
DR	Decay Ratio
EAB	Exclusion Area Boundary
ECCS	Emergency Core Cooling System
EHC	Electro-Hydraulic Control
EOC	End Of Cycle
EOC-RPT-OOS	End Of Cycle Recirculation Pump Trip Out-Of-Service
EOOS	Equipment Out-Of-Service
EPSAG	Emergency Procedure and Severe Accident Guidelines
EPU	Extended Power Uprate
ESF	Engineered Safety Feature
ESFAS	Engineered Safety Feature Actuation System
FCDR	Fuel Cycle Design Report
FDLRX	Maximum Fuel Design Limit Ratio (ratio of LHGR to limit, same as MFLPD)
FFWTR	Final Feedwater Temperature Reduction
FHA	Fuel (and equipment) Handling Accident
FHOOS	Feedwater Heater Out-Of-Service
FSAR	Final Safety Analysis Report

Abbreviation	Description
FUSAR	Fuel Uprate Safety Analysis Report
FW	Feedwater
FWCF	Feedwater Controller Failure
GDC	General Design Criteria
GE / GNF	General Electric / Global Nuclear Fuel
GEH	General Electric Hitachi
GENE	General Electric Nuclear Energy
GRA	Growth Rate Algorithm
GSF	Generic Shape Function
HCOM	Hot Channel Oscillation Magnitude
HER	Hot Excess Reactivity
HFCL	High Flow Control Line
HPCI	High Pressure Coolant Injection
HSBW	Hot Shutdown Boron Worth
HTSP	High power Trip Set Point
I&C	Instrumentation and Control
ICA	Interim Corrective Actions
IHPS	Inadvertent HPCI Pump Start
IORV	Inadvertent Opening of a Main Steam Relief Valve
LAR	License Amendment Request
LFWH	Loss of Feedwater Heating
LHGR	Linear Heat Generation Rate
$LHGRFAC_f$	Flow-Dependent Linear Heat Generation Rate Multipliers
$LHGRFAC_p$	Power-Dependent Linear Heat Generation Rate Multipliers
LOCA	Loss Of Coolant Accident
LOFW	Loss of Feedwater Flow
LOOP	Loss of Offsite Power
LPCI	Low Pressure Coolant Injection
LPCS	Low Pressure Coolant Spray (aka core spray)
LPSP	Low Pressure Setpoint

Abbreviation	Description
LPSP AL	Low Power Set Point Analytical Limit
LPU	Licensed Power Uprate (3952 MWt)
LPZ	Low Population Zone
LRNB	Generator Load Rejection With No Bypass
LTS	Long Term Solution
MAPLHGR	Maximum Average Planar Linear Heat Generation Rate
MAPRAT	Maximum Average Planar Ratio (maximum ratio of APLHGR to limit)
MC	Main Condenser
MCPR	Minimum CPR
$MCPR_f$	Flow-Dependent MCPR
$MCPR_p$	Power-Dependent MCPR
MELLLA	Maximum Extended Load Line Limit Analysis
MFLCPR	Maximum Fraction of Limiting CPR
MFLPD	Maximum Fraction of Limiting Power Density (same as FDLRX)
MSBWP	Minimum Subcritical Banked Withdrawal Position
MSIV	Main Steam Isolation Valve
MSIVC	Main Steam Isolation Valve Closure
MSIVF	Main Steam Isolation Valve with Scram on High Flux
MSLB	Main Steam Line Break
MSRV	Main Steam Relief Valve
NCL	Natural Circulation Line
NFSV	New Fuel Storage Vault
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
OLMCPR	Operating Limit MCPR
OLTP	Original Licensed Thermal Power (3293 MWt)
OOS	Out-Of-Service
OPRM	Oscillation Power Range Monitor
PBDA	Period Based Detection Algorithm
PCT	Peak Cladding Temperature

Abbreviation	Description
PD	Power Density
PHE	Peak Hot Excess reactivity
PLFR	Part Length Fuel Rod
PLUOOS	Power Load Unbalance Out-of-Service
PRFO	Pressure Regulator Failure Open
psia	Pounds per Square Inch Absolute
psig	Pounds per Square Inch Gauge
PUREC	Power Uprate Representative Equilibrium Cycle
PUSAR	Power Uprate Safety Analysis Report
RBM	Rod Block Monitor
RCIC	Reactor Core Isolation Cooling
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
RIPD	Reactor Internal Pressure Differential
RHR	Residual Heat Removal
RPS	Reactor Protection System
RPT	Recirculation Pump Trip
RPTOOS	Recirculation Pump Trip Out-Of-Service
RPV	Reactor Pressure Vessel
RTP	Rated Thermal Power
RWE	Rod Withdrawal Error
RWM	Rod Worth Minimizer
SAFDL	Specified Acceptable Fuel Design Limits
SAG	Severe Accident Guidelines
SDM	Shutdown margin
scf	Standard Cubic Feet (1 atmosphere & 20°C)
SER	Safety Evaluation Report
SFSP	Spent Fuel Storage Pool
SFWPT	Single Feedwater Pump Trip
SIL	Service Information Letter

Abbreviation	Description
SLCS	Standby Liquid Control System
SLMCPR	Safety Limit MCPR
SLO	Single Loop Operation
SRP	Standard Review Plan
SRV	Safety Relief Valve
SS	Steady State
SSC	Systems, Structures and Components
TAF	Top of Active Fuel
TBS	Turbine Bypass System
TBVOOS	Turbine Bypass Valve Out-of-Service
TCV	Turbine Control Valve
TEDE	Total Effective Dose Equivalent
TLO	Two Loop Operation
TPU	Target Power Uprate (3952 MWt) – same as LPU
TS	Technical Specifications
TSSS	Technical Specifications Scram Speed
TTNB	Turbine Trip With No Bypass
TVA	Tennessee Valley Authority
UFSAR	Updated Final Safety Analysis Report
v/o	Volume Percent (wet basis)
VH	Void History

INTRODUCTION

Report Approach

This report summarizes the results of a subset of the safety evaluations performed to support the extended power uprate (EPU) license amendment request (LAR) for the Browns Ferry Nuclear Plant (BFN). The BFN EPU LAR requests an increase from the current licensed thermal power (CLTP) level of 3458 MWt to a new licensed power uprate (LPU) of 3952 MWt. The LPU is approximately 120% of the original licensed thermal power (OLTP) for BFN.

This report supplements the GEH supplied PUSAR (Attachment 6 of the EPU LAR). The PUSAR uses GEH GE14 fuel as the principal reference fuel type for the evaluation of the impact of EPU. However, the BFN units will utilize AREVA ATRIUM 10XM fuel, with the potential for some legacy ATRIUM 10 fuel, under EPU conditions. Therefore, this report is provided to supplement the PUSAR by addressing the effect of EPU conditions on the AREVA fuel in the BFN units. The subset of the safety analyses that are addressed in this report were identified by the Tennessee Valley Authority as being required to support the continued use of AREVA fuel designs in the BFN reactor cores in the EPU LAR.

This report follows the general format of RS-001 (Reference 1) for the sections being addressed. For example, where possible, the numbering of the sections included in this report is based upon the corresponding sections of RS-001. Because only selected portions of RS-001 are being addressed, the numbering of the individual sections of this report is not continuous (i.e. section numbers skip over sections of RS-001 that are not directly addressed).

Evaluation Basis

The safety evaluations documented in this report are based upon the continued use of the ATRIUM™ 10XM* fuel design. Where appropriate, evaluations for the existing ATRIUM-10 fuel design have also been included because the potential exists that some of these previously loaded assemblies may still be resident in a BFN reactor core that is operated at EPU conditions.

---

* ATRIUM is a trademark of AREVA Inc.

A number of the safety evaluations supporting EPU operation are cycle-specific because they are dependent upon the specific bundle and core designs. These are addressed in the reload licensing analyses performed for each cycle.

In this document, these cycle-specific safety evaluations are based upon an ATRIUM 10XM power uprate representative equilibrium cycle (PUREC) fuel cycle design (Reference 2, Attachment 16 of the EPU LAR).

Approved Methodologies

The safety evaluations utilize a series of NRC-approved AREVA methods which are summarized in a Licensing Compendium (Reference 3) which is included as Attachment 36 of the EPU LAR. Application of this approved methodology to EPU conditions remains within its approval basis and SER restrictions, as addressed in ANP-2860P Revision 2 (Reference 4). This methods applicability was extended to the ATRIUM 10XM design for BFN through Supplement 1P (Reference 5) which addressed CLTP conditions. This is further extended to the ATRIUM 10XM at EPU conditions in Supplement 2P (Reference 6) which is included as Attachment 34 of the EPU LAR.

2.1 MATERIALS AND CHEMICAL ENGINEERING

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 reactor coolant pressure boundary (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 BFN principal design criteria predate these criteria. The BFN 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 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 BFN 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 BFN 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 BFN comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as “draft GDC”) is contained in BFN UFSAR Appendix A: draft GDC-9. Final GDC-31 is applicable to BFN 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 48), “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 49), 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 50).

The Pressure-Temperature Limits and Upper Shelf Energy is described in the BFN 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 BFN UFSAR, BFN’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 BFN License Renewal Safety Evaluation Report (SER), NUREG-1843, dated April 2006 (Reference 7). 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.

RCS Pressure and Temperature (P/T) Limits

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 BFN 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 BFN 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 BFN 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 BFN 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

AREVA fuel will be used at BFN when EPU is implemented; however, the basis for the RPV flux is the GEH analysis using GE14 fuel. The AREVA evaluation of the bounding nature of the GEH results for the peak flux values for RPV inner diameter, and internals (shroud diameter, top guide, core plate) is as follows.

Exposure to neutron fast flux (E > 1 MeV) over time causes embrittlement of the reactor pressure vessel and internals. The ATRIUM™-10 fuel was the AREVA design first utilized in the BFN reactors. The ATRIUM-10 assembly impact on vessel and internals fluence originally was made by a stepwise dispositional argument starting from the GE13 fuel design which was the basis for vessel fluence calculations at that time. The transition from GE13 (9x9) to GE14 (10x10) was based upon the conclusion that the change in fuel type would not have a significant impact on the fluence effects. A similar conclusion was made to support the transition to the current ATRIUM-10 (10x10) fuel design. The transition to the ATRIUM 10XM design was similar in scope to the GE14 to ATRIUM-10 transition since they all are 10x10 lattice designs and therefore the same conclusions would apply.

The current BFN reactor vessel P/T curves are based upon a fluence calculation performed by GENE to support extended power uprate conditions using GE14 fuel with a batch size of 332 fresh bundles. AREVA also performed a calculation to determine the impact of BLEU fuel in these units using the previous ATRIUM-10 design. The lower reactivity of the BLEU fuel increased the batch size to 368 resulting in the loading of more high reactivity once-burnt assemblies on or near the periphery of the core. Both the GENE and AREVA calculations used an axial flux profile with essentially the same shape. In the latter calculation, AREVA was able to show that the vessel and internals flux for ATRIUM-10 BLEU fuel was bounded by the GENE analysis with significant margin, as shown below:

Comparison of Reactor Internals Fluence Evaluations

Point of Interest	Fast Neutron Fluxes ((n/cm2) / sec)	
	GENE	BLEU
Reactor Vessel	1.40 $\times 10^{9}$	7.96 $\times 10^{8}$
Shroud	3.14 $\times 10^{12}$	1.84 $\times 10^{12}$
Top Guide Grid	1.21 $\times 10^{13}$	1.02 $\times 10^{13}$
Core Plate *	4.31 $\times 10^{11}$	9.17 $\times 10^{11}$

* The BLEU calculation exhibits a higher flux value for the core plate because the assembly used did not have a natural-U bottom blanket (enriched U in the bottom zone). This is not consistent with past or planned ATRIUM-10 or ATRIUM 10XM designs.

The fluence that causes embrittlement at the vessel wall and adjacent internals is primarily caused by the peripheral assemblies. As noted previously, the AREVA analyses were based upon a BLEU core at EPU conditions which included a large fraction of once-burnt assemblies on the core periphery (~70%). Since the flux is proportional to the power generated, the presence of these once-burnt assemblies significantly increases the flux at the shroud and vessel walls. By comparison, the current loading patterns with BLEU fuel at 105% OLTP conditions result in primarily higher exposure twice-burnt (i.e., lower power) assemblies in the outer row. The introduction of the ATRIUM 10XM design will tend to decrease the size of the fresh batch fraction which would maintain or further decrease the outer shroud and vessel flux level. For example, the ATRIUM 10XM EPU Equilibrium Cycle is composed of 332 fresh assemblies with only higher exposure 3rd cycles bundles on the periphery. The ATRIUM 10XM EPU batch fraction is significantly lower that the previously evaluated ATRIUM-10 BLEU core loading and is the same as the previously evaluated GE14 loading even though the 10XM core contains some BLEU fuel. Consequently, the AREVA EPU based analysis for ATRIUM-10 BLEU fuel will remain bounding for the use of the ATRIUM 10XM design at EPU conditions. This conclusion with respect to ATRIUM 10XM remains valid even if the reload does not contain BLEU fuel.

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. The AREVA evaluation of reactor fluence for the ATRIUM 10XM design demonstrates the bounding nature of the GEH results for the peak flux values for RPV inner diameter, and internals at EPU conditions.

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 BFN principal design criteria predate these criteria. The BFN 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 General Design Criteria published in 1967, the Tennessee Valley Authority (TVA), performed a comparative evaluation of the design basis of BFN with the AEC proposed General Design Criteria of 1967. The BFN UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria sent 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 BFN UFSAR where there is subject matter relating to the intent of that particular criteria.

While BFN 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 BFN comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as “draft GDC”) is contained in BFN 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 BFN.

BFN instrumentation and control systems are described in BFN UFSAR Section 7, “Control and Instrumentation.”

BFN’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 BFN License Renewal Safety Evaluation Report (SER), NUREG-1843, dated April 2006 (Reference 7). The instrumentation and control systems were determined to be within the scope of license renewal and the components subject to age 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

2.4.1.1 Nuclear Steam Supply System Monitoring and Control Instrumentation

The nuclear steam supply system (NSSS) monitoring and control system operation is not dependent on fuel design.

2.4.1.1.1 Average Power Range Monitors

The average power range monitor (APRM) signals are rescaled as a result of EPU. Fuel design has no effect on the rescaling of the APRM response.

2.4.1.1.2 Local Power Range Monitors

The function of the local power range monitors is not dependent on fuel design.

2.4.1.1.3 Rod Block Monitor

The function of the rod block monitor is not dependent on fuel design.

2.4.1.1.4 Rod Worth Minimizer

The function of the rod worth minimizer is not dependent on fuel design.

2.4.1.2 BOP Monitoring and Control

No safety-related balance-of-plant (BOP) system setpoint changes are required for ATRIUM 10XM fuel.

2.4.1.2.1 Pressure Control System

The pressure control system, including the electro-hydraulic control (EHC) turbine control system, is not dependent on fuel design.

2.4.1.2.2 Turbine Steam Bypass System

The turbine steam bypass system is not dependent on fuel design.

The turbine bypass system is credited in the potentially limiting Feedwater Controller Failure Maximum Demand (FWCF) as discussed in Section 2.8.5. The bypass flow capacity is used in some anticipated operational occurrence (AOO) evaluations. If an event that established a core operating limit did credit the availability of the bypass system, the bypass flow would be used in the reload analysis to establish the core operating limit. The AOO events are discussed further in Section 2.8.5.

2.4.1.2.3 Feedwater Control System

The normal operation of the feedwater control system is not dependent on fuel design. The system response due to a failure of the feedwater control system is fuel design dependent, and is addressed in Section 2.8.5.1 (Feedwater Controller Failure Maximum Demand) and Section 2.8.5.2.3 (Loss of Feedwater Flow Event).

2.4.1.2.4 Leak Detection System

The leak detection system is not dependent on fuel design.

2.4.1.3 Technical Specification Instrument Setpoints

Safety analyses are performed to demonstrate the adequacy of analytical limit (AL) setpoints to ensure all licensing criteria are met. The AL setpoints developed for EPU conditions remain applicable for ATRIUM 10XM fuel.

2.4.1.3.1 Main Steam Line High Flow Isolation

The main steam line high flow isolation setpoint is not dependent on fuel design.

2.4.1.3.2 Turbine First-Stage Pressure Scram and Recirculation Pump Trip Bypass

Safety analyses take into account the revised setpoints. Therefore, the setpoints developed for EPU conditions remain applicable for ATRIUM 10XM fuel.

2.4.1.3.3 APRM Flow Biased Scram

The APRM neutron flux scram analytical limit is not dependent on fuel design.

The APRM flow-biased rod block and scram analytical lines are developed as a function of the recirculation loop drive flows. The APRM flow-biased setpoint function is not used in the transient analysis.

The impact of a control rod withdrawal error is limited by the rod block monitor (RBM) setpoint. RBM setpoints are determined based on cycle-specific rod withdrawal error (RWE) analyses. The RWE event is addressed in Section 2.8.5.4.2 “Uncontrolled Control Rod Assembly Withdrawal at Power.”

2.4.1.3.4 Rod Worth Minimizer Low Power Setpoint

The low pressure setpoint (LPSP) AL is maintained at the same value in terms of percent power (10% rated thermal power (RTP)) and the EPU has been evaluated on this basis. Below this setpoint, only withdrawals or insertions adhering to the cycle-specific control rod drop accident (CRDA) analysis and banked position withdrawal sequence (BPWS) are allowed, as described in Section 2.8.5.4.4.

2.4.1.3.5 Rod Block Monitor

The impact of EPU on the Rod Block Monitor (RBM) is the increase in power level. The severity of the 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. As discussed in Section 2.8.5.4.2, a cycle-specific analysis is performed to establish setpoints using NRC approved methodologies. A setpoint is chosen based upon the required OLMCPR. Because the cycle-specific analysis is performed at expected operating conditions and the setpoint is chosen based upon the required OLMCPR, the ability of the RBM system to protect the fuel SAFDLs will be maintained for EPU.

2.4.1.3.6 APRM Setdown in Startup Mode

The APRM setdown in startup mode is not dependent on fuel design.

2.4.1.3.7 Main Steam Line Low Pressure Isolation in the Run Mode

The low steam line pressure main steam isolation valve (MSIV) closure setpoint (RUN Mode) is not dependent on fuel design.

2.4.1.4 Changes to Instrumentation and Controls

EPU-related changes to instrumentation and controls (I&C) are not dependent on fuel design.

Conclusion

The effects of the proposed EPU with ATRIUM 10XM on the functional design of the reactor protection system (RPS), engineered safety feature actuation system (ESFAS), safe shutdown system, and control systems have been reviewed. It is concluded that the effects of the proposed EPU with ATRIUM 10XM on these systems have been adequately addressed and no changes are necessary due to AREVA fuel. It is further concluded that the systems will continue to meet the requirements of 10 CFR 50.55a(a)(1), 10 CFR 50.55a(h), and the current licensing basis. Therefore, the proposed EPU with ATRIUM 10XM is acceptable with respect to instrumentation and control.

2.5 PLANT SYSTEMS

2.5.1 Internal Hazards

2.5.1.4 Fire Protection

2.5.1.4.2 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 BFN principal design criteria predate these criteria. The BFN 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 General Design Criteria published in 1967, the Tennessee Valley Authority (TVA), performed a comparative evaluation of the design basis of BFN with the AEC proposed General Design Criteria of 1967. The BFN UFSAR, Appendix A, “Conformance to AEC Proposed General Design Criteria,” contains this comparative evaluation. This evaluation discusses each of the groups of criteria sent 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 BFN UFSAR where there is subject matter relating to the intent of that particular criteria.

While BFN 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 BFN comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as “draft GDC”) is contained in BFN UFSAR Appendix A: draft GDC-4. Final GDC-3 is applicable to BFN as described the BFN Fire Protection Report, Vol. 1, Rev. 20.

Fire Protection is described in BFN 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 BFN License Renewal Safety Evaluation Report (SER), NUREG-1843, dated April 2006 (Reference 7). 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, structural steel etc., 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.

Technical Evaluation – Appendix R

The limiting Appendix R fire event was analyzed assuming operation with AREVA fuel at EPU conditions. The Appendix R analyses were performed with LOCA Evaluation Models developed by AREVA and approved for reactor licensing analyses by the U.S. Nuclear Regulatory

Commission (NRC). The models and computer codes used by AREVA for LOCA analyses are collectively referred to as the EXEM BWR-2000 Evaluation Model. The EXEM BWR-2000 Evaluation Model and NRC approval are documented in References 8 through 12. The LOCA analysis codes are not explicitly approved for Appendix R analyses; however the physical phenomena, depressurization via the ADS, tracking the water level, and the fuel response (calculation of PCT) are similar to LOCA requirements.

The evaluation determined the effect of EPU on fuel cladding integrity and reactor vessel integrity as a result of the fire event.

The analyses were performed using plant parameters and plant geometry presented specifically for BFN Units 1, 2, and 3 in Reference 13. The plant parameters specified are based on EPU operation and the plant geometry includes any modifications necessary for EPU at BFN.

A complete Appendix R fire event analysis will address the following topics:

- Fire suppression and detection systems
- Operator response time
- Reactor vessel water level
- Suppression pool temperature
- Peak cladding temperature (PCT)

The topics of “fire suppression and detection systems” and “suppression pool temperature” are not fuel related and were not analyzed by AREVA. The behavior of reactor vessel water level in an Appendix R event is also not fuel related however, vessel water level was calculated in the analysis.

Three cases or scenarios of Appendix R events were analyzed. The cases differ primarily in the assumptions for main steam relief valve (MSRV) operation.

- Case 1 (base case) – 3 MSRVs are opened by operator action 25 minutes after accident initiation.
- Case 2 – Spurious signal opens 1 MSRV (Stuck Open Relief Valve (SORV)) for 10 minutes after accident initiation. 3 MSRVs are opened at 20 minutes.
- Case 3 –Spurious signal opens 1 MSRV (SORV) throughout the event. 3 MSRVs are opened at 20 minutes.

The primary parameter calculated in the cases was PCT. The results from these calculations are provided in Table 2.5-1, Figure 2.5-1, and Figure 2.5-2.

The Appendix R event blowdown phase was defined as the period after the start of the fire when mass is lost through the MSRVs. Blowdown was assumed to end when the ADS (MSRVs) decreased vessel pressure to the pressure where rated LPCS flow would have occurred if it were available. The vessel liquid level typically drops below the top of the active fuel during blowdown. One LPCI pump was available to recover the core during the refill period. Reflood was assumed to occur when sufficient entrained liquid mass flow to end the fuel heatup was calculated at the core hot node. PCT occurs at the time of hot node reflood. The system and hot channel calculations were continued past the time of hot node reflood to confirm reflood of the entire core and to calculate the vessel water level behavior after the time of hot node reflood.

The results of the Appendix R analysis for AREVA fuel at EPU conditions demonstrate that fuel cladding integrity and reactor vessel integrity will be maintained using the licensee specified and validated operator response times. Case 1 is the limiting case. The PCT is 1119°F and the peak vessel pressure is 1224 psia, both of which remain below the acceptance limits of 1500°F and 1375 psig.

Technical Evaluation – NFPA 805

The limiting NFPA 805 fire event was analyzed assuming operation with AREVA fuel at EPU conditions. The NFPA 805 analyses were performed with LOCA Evaluation Models developed by AREVA and approved for reactor licensing analyses by the U.S. Nuclear Regulatory Commission (NRC). The models and computer codes used by AREVA for LOCA analyses are collectively referred to as the EXEM BWR-2000 Evaluation Model. The EXEM BWR-2000 Evaluation Model and NRC approval are documented in References 8 through 12. The LOCA analysis codes are not explicitly approved for NFPA 805 analyses; however the physical phenomena, depressurization via the ADS, tracking the water level, and the fuel response (calculation of PCT) are similar to LOCA requirements.

The evaluation determined the effect of EPU on fuel cladding integrity and reactor vessel integrity as a result of the fire event.

The analyses were performed using plant parameters and plant geometry presented specifically for BFN Units 1, 2, and 3 in Reference 13. The plant parameters specified are based on EPU operation and the plant geometry includes any modifications necessary for EPU at BFN.

The new NFPA 805 calculations include three new considerations that were not included in Appendix R analyses.

1. Leakages from the reactor pressure vessel (RPV) and recirculation pump seals due to multiple spurious operation (MSO) of MSRVs caused by the fire event.

2. Cases with MSO of up to 13 main steam relief valves (MSRVs). Current Appendix R analyses only included cases with MSO of one MSRV.

3. The condensate system make up flow (CD/FW) will be modeled as the primary system to recover from the NFPA 805 fire event for one case. The Appendix R analyses only model the low pressure injection system (LPCI) as the emergency system used for core recovery.

The primary parameter calculated in the cases was PCT. The results from these calculations are provided in Table 2.5-2, Figure 2.5-3 and Figure 2.5-4.

The NFPA 805 event blowdown phase was defined as the period after the start of the fire when mass is lost through the MSRVs and reactor leakages. Blowdown was assumed to end when the ADS (MSRVs) decreased vessel pressure to the pressure where rated LPCS flow would have occurred if it were available. The vessel liquid level typically drops below the top of the active fuel during blowdown. One LPCI pump was available to recover the core during the refill period. Reflood was assumed to occur when sufficient entrained liquid mass flow to end the fuel heatup was calculated at the core hot node. PCT occurs at the time of hot node reflood. The system and hot channel calculations were continued past the time of hot node reflood to confirm reflood of the entire core and to calculate the vessel water level behavior after the time of hot node reflood.

The results of the NFPA 805 analysis for AREVA fuel at EPU conditions demonstrate that fuel cladding integrity and reactor vessel integrity will be maintained using the licensee specified and validated operator response times. The most limiting analyzed result, that remains below the 1500^{o}F acceptance limit, is from the case with 11 MSRVs open due to MSO at the start of the event, and remaining open throughout the event, with LPCI on due to operator action at 20 minutes. The PCT is 1330^{o}F and the peak vessel pressure is 1097 psia, both of which remain below the acceptance limits of 1500^{o}F and 1375 psig.

Conclusion

BFN 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.

2.5.4 Balance-of-Plant Systems

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 BFN principal design criteria predate these criteria. The BFN 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 General Design Criteria published in 1967, the Tennessee Valley Authority (TVA), performed a comparative evaluation of the design basis of BFN with the AEC proposed General Design Criteria of 1967. The BFN UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria sent 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 BFN UFSAR where there is subject matter relating to the intent of that particular criteria.

While BFN 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 BFN comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as “draft GDC”) is contained in BFN UFSAR Appendix A: draft GDC-70.

The main condenser system is described in BFN 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 BFN License Renewal Safety Evaluation Report (SER), NUREG-1843, dated April 2006 (Reference 7). 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

The condenser is not dependent on fuel design.

The absolute value of lbm/hr of the steam bypassed to the main condenser (MC) during AOO transients evaluated for EPU with bypass operation is not increased for EPU. Because the absolute value of bypass flow is unchanged with EPU, the conditions within the condenser during AOO transients remain unchanged with EPU.

Conclusion

The effects of the proposed EPU with ATRIUM 10XM fuel on the MC system have been considered. It is concluded that the MC system will continue to maintain its ability to withstand the blowdown effects of the steam from the turbine bypass system (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 MC system.

2.5.4.3 Turbine Bypass

Regulatory Evaluation

The Turbine Bypass System (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. A 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 BFN principal design criteria predate these criteria. The BFN 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 General Design Criteria published in 1967, the Tennessee Valley Authority (TVA), performed a comparative evaluation of the design basis of BFN with the AEC proposed General Design Criteria of 1967. The BFN UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria sent 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 BFN UFSAR where there is subject matter relating to the intent of that particular criteria.

While BFN 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 BFN comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in BFN UFSAR Appendix A: draft GDCs-40 and 42. There is no draft GDC directly associated with final GDC-34.

The TBS is described in BFN 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 BFN License Renewal Safety Evaluation Report (SER), NUREG-1843, dated April 2006 (Reference 7). 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

The steam bypass capacity is used as an input to the reload analysis process for the evaluation of AOO events that credit the Turbine Steam Bypass System. The bypass flow capacity is used in some AOO evaluations. If an event that established a core operating limit credits the availability of the bypass system, the bypass flow is used in the reload analysis to establish the core operating limit. AOO events that credit the TBS utilize a capacity that has been adjusted to a lower fraction of rated steam flow. The AOO events are discussed further in Section 2.8.5. The use of ATRIUM 10XM fuel will not affect the steam bypass capacity requirements.

Conclusion

The effects of the proposed EPU with ATRIUM 10XM on the TBS have been reviewed. It is concluded that the proposed EPU with ATRIUM 10XM will continue to meet the current TBS licensing basis. Therefore, the proposed EPU with ATRIUM 10XM is acceptable with respect to the TBS.

Table 2.5-1 Browns Ferry Appendix R Fire Event Results

CASE	ECCS	PCT (°F)	Peak Pressure at Bottom of Vessel (psia)
Appendix R Case 1 3 MSRVs opened at 25 minutes	1 LCPI operational	1119	1224
Appendix R Case 2 3 MSRVs opened at 20 minutes with 1 MSRV Open for 10 Minutes	1 LCPI operational	1000	1214
Appendix R Case 3 3 MSRVs opened at 20 minutes with 1 MSRV Open Throughout	1 LCPI operational	828	1214

Table 2.5-2 Browns Ferry NFPA 805 Fire Event Results

CASE	ECCS	PCT (°F)	Peak Pressure at Bottom of Vessel (psia)
MSO of 11 MSRVs	1 LPCI on at 20 Minutes	1330	1097

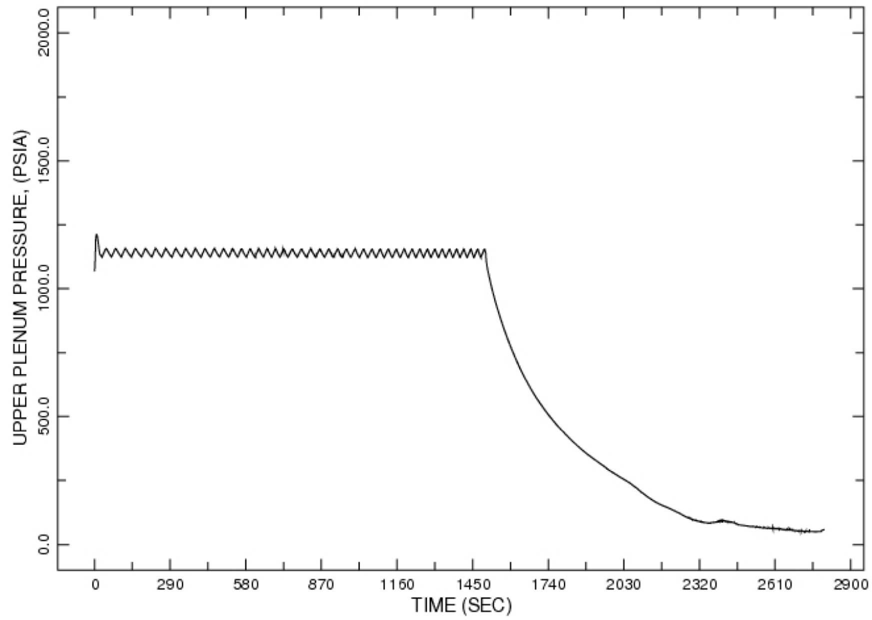

Figure 2.5-1 Appendix R Upper Plenum Pressure
Case 1 - 3 MSRVs Opened at 25 Minutes

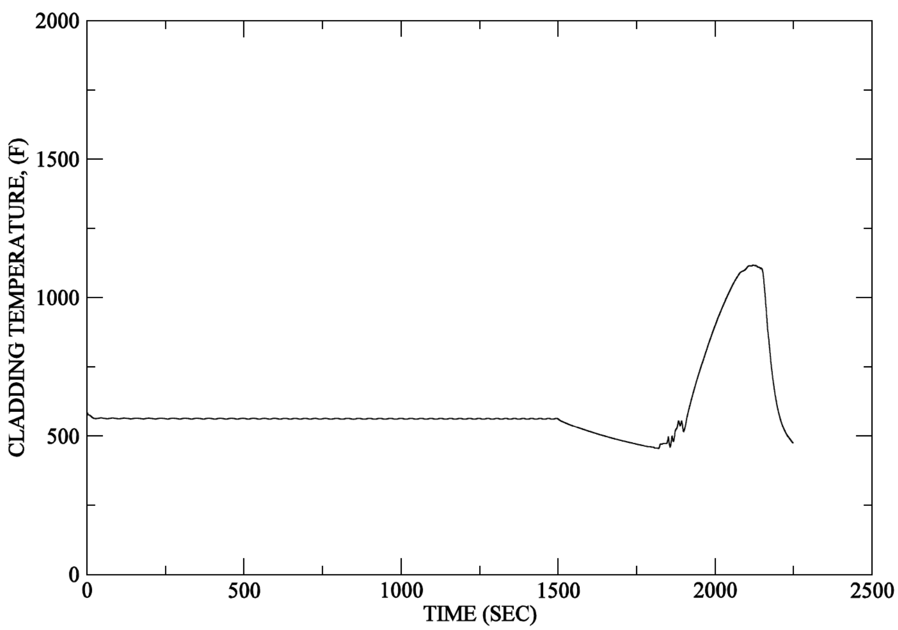

Figure 2.5-2 Appendix R PCT Rod Surface Temperature
Case 1 - 3 MSRVs Opened at 25 Minutes

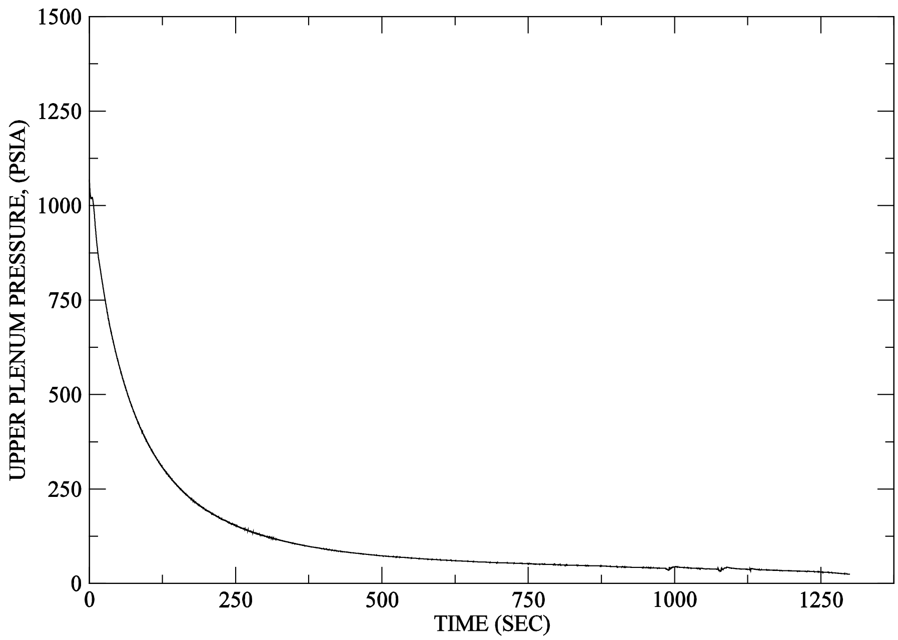

Figure 2.5-3 NFPA 805 Upper Plenum Pressure 11 MSRVs Open

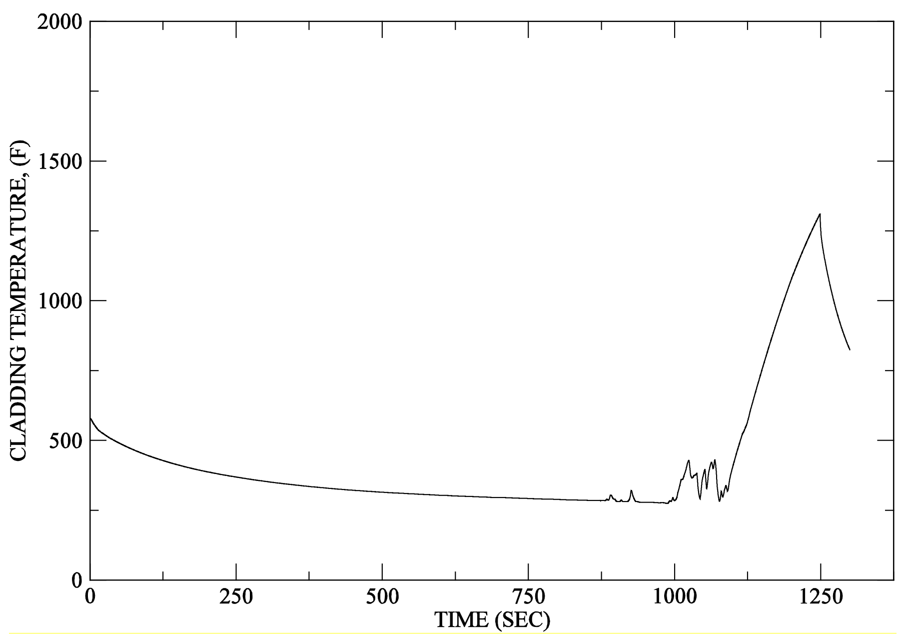

Figure 2.5-4 NFPA 805 PCT Rod Surface Temperature 11 MSRVs Open

2.6 REACTOR SYSTEMS

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 BFN principal design criteria predate these criteria. The BFN 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 General Design Criteria published in 1967, the Tennessee Valley

Authority (TVA), performed a comparative evaluation of the design basis of BFN with the AEC proposed General Design Criteria of 1967. The BFN UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria sent 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 BFN UFSAR where there is subject matter relating to the intent of that particular criteria.

While BFN 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 BFN comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in BFN 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 BFN UFSAR Section 5.2.6, "Combustible Gas Control in Primary Containment." BFN's containment is inerted during power operation.

Technical Evaluation

The combustible gas control system is designed to ensure an inert atmosphere in the drywell and wetwell is maintained after a postulated LOCA. This is accomplished by injecting nitrogen into the drywell and wetwell to keep the oxygen concentration below 5% by volume. Cladding mass affects the amount of metal-water reaction and consequently the nitrogen injection requirements. **[**

] The results of the evaluation for **[** **]** , at EPU conditions, show that the required CAD system start time for the design basis case is 30 hours, compared to 37 hours under CLTP conditions. This start time does not impact the ability of the operators to appropriately respond to a LOCA. Figure 2.6-1 shows the results of the integrated hydrogen production rates from radiolysis and metal-water reactions. The drywell and wetwell uncontrolled hydrogen and oxygen concentrations are presented in Figure 2.6-2. The drywell pressure response assuming no venting is presented in Figure 2.6-3 and the CAD system nitrogen volume requirements are presented in Figure 2.6-4.

Technical Specification (TS) 3.6.3.1 "Containment Atmospheric Dilution (CAD) system, requires that 2500 gallons of liquid nitrogen (191,000 scf) be stored in each of two tanks to meet the CAD system inerting requirements. As a result of increased production rate of radiolytic gas following EPU, the required 7-day volume of nitrogen to satisfy TS 3.6.3.1 increases to 2615 gallons (200,000 scf) from 2108 gallons (161,200 scf) under CLTP conditions, which exceeds the available 2500 gallons (191,000 scf) supply required by TS 3.6.3.1. Analysis of the containment pressure buildup as a result of continuing CAD operation, under EPU conditions, shows that the containment repressurization limit of 30 psig is reached 15 days post-LOCA, compared to 18 days under CLTP conditions.

Conclusion

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.

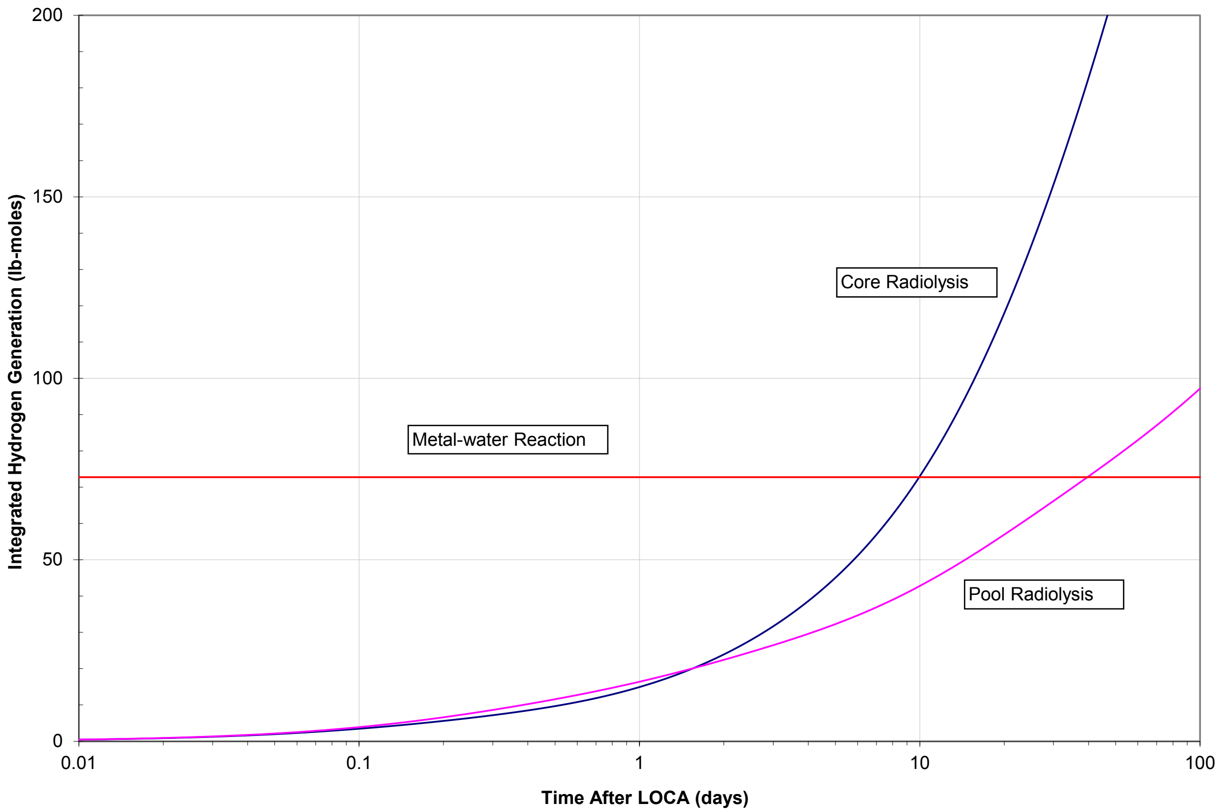

Figure 2.6-1 Time-integrated Containment Hydrogen Generation

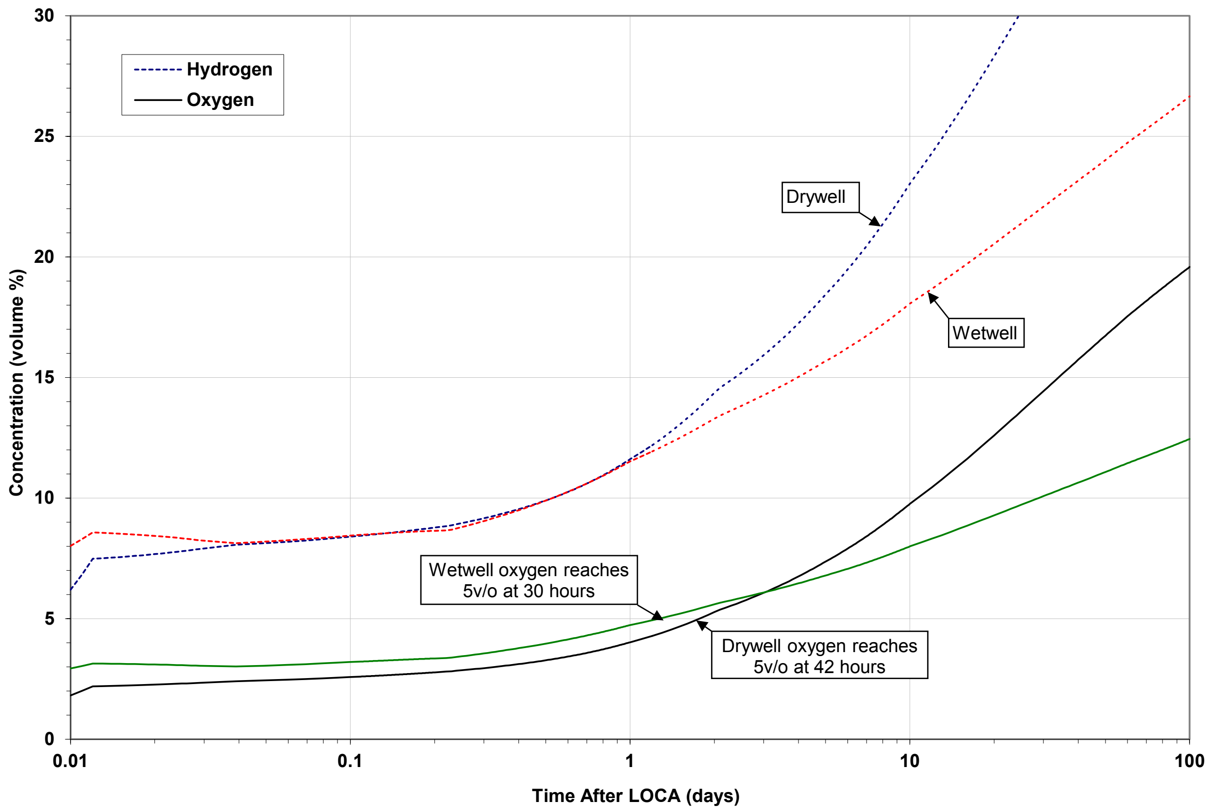

Figure 2.6-2 Uncontrolled H2 and O2 Concentrations in Drywell and Wetwell

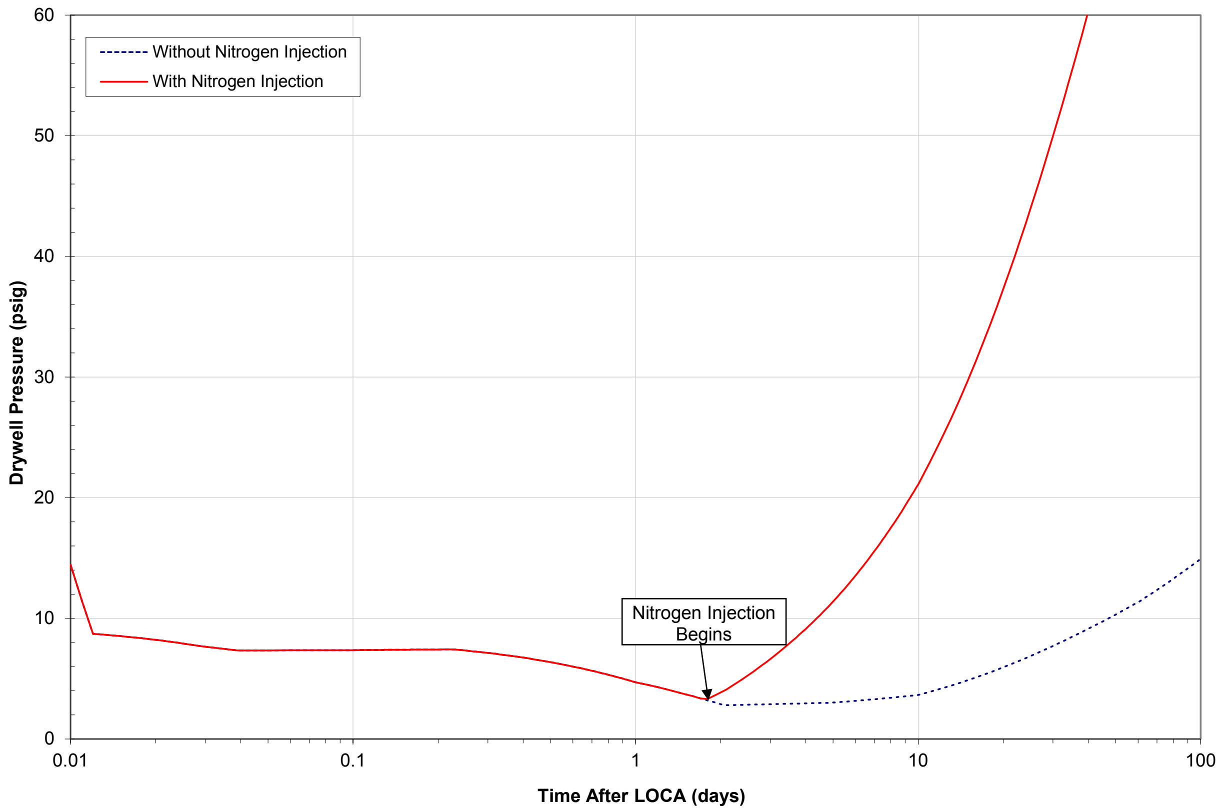

Figure 2.6-3 Drywell Pressure Response to CAD Operation without Venting

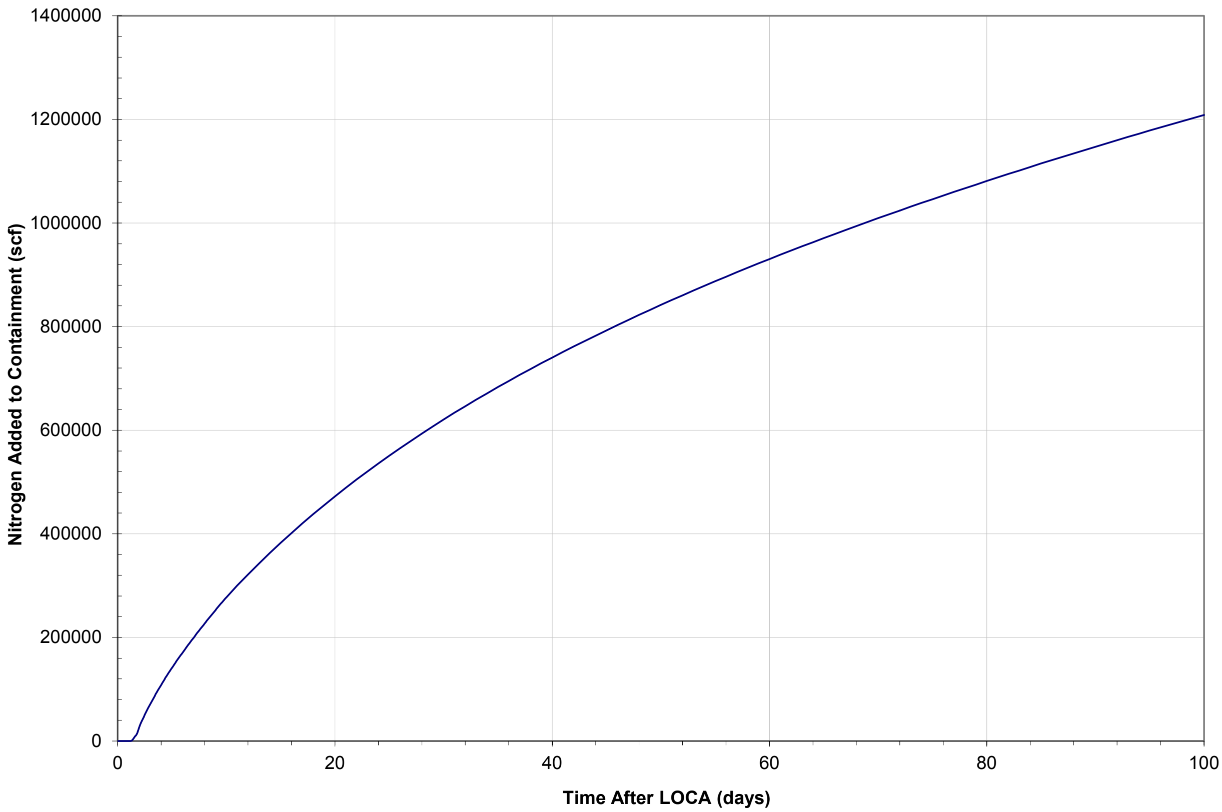

Figure 2.6-4 CAD System Nitrogen Volume Requirement

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 emergency core cooling system (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 BFN as described in "NRC Issuance of Amendments Regarding the Transition to AREVA Fuel," dated July 3, 2012 (Reference 14).

The Fuel System Design is described in BFN UFSAR Chapter 3, "Reactor."

Technical Evaluation

All three of the BFN units are in the process of transitioning to AREVA ATRIUM™ 10XM fuel from the previous ATRIUM-10 design. The required changes to the BFN Technical Specifications were previously reviewed and approved by the NRC (Reference 15). The initial transition began with Unit 2 Cycle 19 in the spring of 2015. Only AREVA fuel types will be resident in the BFN cores during EPU implementation. Even though all units will have transitioned to the ATRIUM 10XM fuel design prior to EPU implementation, the EPU LAR submittal is based in part on the use of evaluations utilizing the GEH/GNF GE14 fuel design. Consequently, additional EPU evaluations have been performed that assume a representative equilibrium core of ATRIUM 10XM fuel. As noted earlier, the ATRIUM 10XM design will be resident in the BFN cores and furthermore will be the dominant fuel type by the time EPU is implemented.

At the CLTP as well as at the EPU RTP conditions, all fuel design limits will be met through fuel bundle and core design combined with plant operational strategies. Revised loading patterns, changes in enrichment and gadolinia loading, and larger batch sizes will be used as necessary to provide the required operating flexibility and to maintain fuel cycle length. The representative ATRIUM 10XM equilibrium core demonstrates the performance of this fuel design in a BFN EPU core loading, including meeting all criteria in regard to ensuring that the underlying fuel design limits are protected.

The evaluation of the thermal limits used to protect these fuel design limits are discussed in detail later in this document. These limits are evaluated for each core reload using NRC approved methodologies which are already approved for use in Section 5.6.5 of the BFN Technical Specifications. Application of this methodology to EPU conditions remains within its approval basis and SER restrictions, as addressed in ANP-2860P Revision 2 (Reference 4). This methods applicability was extended to the ATRIUM 10XM design for BFN through Supplement 1P (Reference 5) which addressed CLTP conditions. This is further extended to the ATRIUM 10XM at EPU conditions in Supplement 2P (Reference 6) which is included as Attachment 34 of the EPU LAR. Therefore, because the fuel design limits are evaluated in accordance with approved methodology for each core reload, the assessment of the BFN ATRIUM 10XM fuel product line design at EPU conditions is acceptable.

Decay Heat

Reference 51 evaluated the effect of ATRIUM 10 fuel on decay heat for Browns Ferry EPU power conditions and concluded that, for the same fuel cycle parameters, the decay heat results are unaffected by the fuel design. From this evaluation, there are two conclusions that can be made. One, variations in lattice design for a specific fuel assembly have an insignificant effect on the decay heat results and two, the primary input parameters affecting the results are the fission fractions for the U235, U238, and Pu239 isotopes. The following table presents the range of fission fractions for these isotopes from the GE14 and ATRIUM-10 lattices used in the Reference 51 evaluation. The table also includes the fission fractions for the ATRIUM 10XM lattices that were evaluated.

Comparison of Fission Fractions for Various Fuel Designs

Fuel Design	^{235}U Fission Fraction	^{238}U Fission Fraction	^{239}Pu Fission Fraction
GE14	0.4821 - 0.5363	0.0699 - 0.0997	0.3938 - 0.4182
ATRIUM-10	0.4902 - 0.5367	0.0710 - 0.0961	0.3923 - 0.4137
ATRIUM 10XM	0.4635 - 0.5547	0.0703 - 0.1020	0.3750 - 0.4346

Analyses were performed with the following input parameters:

- EOC core average exposure = 40 GWd/MTU
- Irradiation time interval = 4 years
- GE14 bundle average enrichment = 4.021 wt%
- ATRIUM-10 bundle average enrichment = 4.27 wt%
- ATRIUM 10XM bundle average enrichment = 4.257 wt%
- EPU power level = 3952 MW_{th}

Reference 51 analyzed eight ATRIUM-10 lattice / void fraction combinations and sixteen GE14 lattice / void fraction combinations having fission fractions in the ranges provided in the table above. A separate analysis considered 22 ATRIUM 10XM lattice/void fraction combinations having fission fractions in the ranges provided in this table. From these evaluations it is concluded that even with these varying fission fractions, decay heat results were not significantly impacted. A comparison of limiting GE14 results and limiting ATRIUM-10 results showed that the average absolute deviation in unadjusted/uncorrected shutdown power fraction was less than 0.1%. A comparison of limiting ATRIUM 10XM results and ATRIUM-10 results showed that the average absolute deviation in unadjusted/uncorrected shutdown power fraction was less than 0.3%. For the introduction of the ATRIUM 10XM fuel, the fission fractions are comparable to the ones used in the Reference 51 assessment. The fuel design differences in fission fractions are considerably less than the range of fission fractions evaluated. Therefore, it can be concluded that the decay heat is insignificantly impacted by the ATRIUM 10XM fuel design.

In addition, conservative assumptions were applied in the Reference 51 analysis to ensure the final results are bounding with respect to transition cores and equilibrium cores. Decay heat results were generated based on ANSI/ANS-5.1-1979 Standard with an added conservatism corresponding to two-sigma (2σ) uncertainty, equal to 6%. Since decay heat power is not significantly impacted by the ATRIUM 10XM fuel design, it can be concluded that the uncertainty in decay heat power is not significantly impacted because the uncertainty is primarily a function of the decay heat itself (Section 3.4 of ANSI/ANS-5.1-1979). The data was prepared with an allowance for miscellaneous Actinides and Activation Products consistent with the recommendations of SIL 636.

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 BFN principal design criteria predate these criteria. The BFN 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 General Design Criteria published in 1967, the Tennessee Valley Authority (TVA), performed a comparative evaluation of the design basis of BFN with the AEC proposed General Design Criteria of 1967. The BFN UFSAR, Appendix A, “Conformance to AEC Proposed General Design Criteria,” contains this comparative evaluation. This evaluation discusses each of the groups of criteria sent 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 BFN UFSAR where there is subject matter relating to the intent of that particular criteria.

While BFN 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 BFN comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as “draft GDC”) is contained in BFN 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 BFN as described in “NRC Issuance of Amendments Regarding the Transition to AREVA Fuel,” dated July 3, 2012 (Reference 14).

Nuclear design is described in BFN UFSAR Chapter 3, “Reactor.”

Technical Evaluation

2.8.2.1 Core Operation

EPU increases the average power density proportional to the power increase and has some effects on operating flexibility, reactivity characteristics and energy requirements. The additional energy requirements for EPU are met by an increase in bundle enrichment, an increase in the reload fuel batch size, and/or changes in fuel loading pattern to maintain the desired plant operating cycle length.

All three of the BFN units are in the process of transitioning to AREVA ATRIUM™ 10XM fuel from the previous ATRIUM-10 design. The initial transition began with Unit 2 Cycle 19 in the spring of 2015 and only AREVA fuel types will be used through EPU implementation. Even though all units will have transitioned to the ATRIUM 10XM fuel design prior to EPU implementation, this submittal is based in part on the use of evaluations utilizing the GEH/GNF GE14 fuel design. Consequently, additional EPU evaluations have been performed that assume a representative equilibrium core of ATRIUM 10XM fuel. As noted earlier, the ATRIUM 10XM design will be resident in the BFN cores and furthermore will be the dominant fuel type by the time EPU is implemented.

At the CLTP as well as at the EPU RTP conditions, all fuel design limits will be met through fuel bundle and core design combined with plant operational strategies. Revised loading patterns, changes in enrichment and gadolinia loading, and larger batch sizes will be used as necessary to provide the required operating flexibility and to maintain fuel cycle length. The representative ATRIUM 10XM equilibrium core demonstrates the performance of this fuel design in a BFN EPU core loading, including meeting all criteria in regard to ensuring that the underlying fuel design limits are protected.

The evaluation of the thermal limits used to protect these fuel design limits are discussed in detail later in this document. These limits are evaluated for each core reload using NRC approved methodologies which are already approved for use in Section 5.6.5 of the BFN Technical Specifications. Application of this methodology to EPU conditions remains within its approval basis and SER restrictions, as addressed in ANP-2860P Revision 2 (Reference 4). This methods applicability was extended to the ATRIUM 10XM design for BFN through

Supplement 1P (Reference 5) which addressed CLTP conditions. This is further extended to the ATRIUM 10XM at EPU conditions in Supplement 2P (Reference 6) which is included as Attachment 34 of the EPU LAR. Therefore, because the fuel design limits are evaluated in accordance with approved methodology for each core reload, the assessment of the use of ATRIUM 10XM fuel product line design for EPU operation at BFN is acceptable.

2.8.2.1.1 Core Design

BFN is currently licensed with an average bundle power of 4.53 MWt/bundle (CLTP/764 or 3458 MWt/764). The average bundle power for EPU is 5.17 MWt/bundle (LPU/764 or 3952 MWt/764). The EPU average bundle power is comparable to the range of other operating BWRs.

The peak bundle power is controlled by adherence to the licensing thermal limits and does not significantly change with power uprate. The increased power level does increase the core average bundle power with a corresponding flattening of the radial power distribution across the core. The additional energy requirements for power uprate are met by an increase in bundle enrichment, an increase in the reload fuel batch size, and/or changes in fuel loading pattern to maintain the desired plant operating cycle length. The power distribution in the core is changed to achieve increased core power, while limiting the minimum critical power ratio (MCPR), maximum linear heat generation rate (MLHGR), and maximum average planar linear heat generation rate (MAPLHGR) in any individual fuel bundle to be within limits as defined in the COLR.

The reactor core design power distribution represents a thermal operating state at design conditions. It includes allowances for the combined effects on the fuel heat flux and temperature of the gross and local power density distributions, control rod pattern, and reactor power level adjustments during plant operation. NRC-approved core design methods were used to analyze core performance at the EPU RTP level. Detailed fuel cycle calculations of a representative equilibrium core design for BFN (Reference 2, Attachment 16 of the EPU LAR) demonstrate the feasibility of EPU RTP operation while maintaining fuel design limits, as summarized in the following table:

Topic	Licensing Requirement	Design Margin*	BFN ATRIUM 10XM Result
Maximum Fraction of Limiting CPR (MFLCPR)	< 1.0	< 0.93	0.928
Fuel Design Limit Ratio (FDLRX or MFLPD)	< 1.0	< 0.89	0.879
Maximum Average Planar LHGR Ratio (MAPRAT)	< 1.0	< 0.89	0.879

Thermal-hydraulic design and operating limits ensure an acceptably low probability of boiling transition-induced fuel cladding failure occurring in the core, even for the most severe postulated operational transients. As needed, limits are also placed on fuel APLHGR and/or fuel rod LHGRs in order to meet both PCT limits for the limiting LOCA and fuel mechanical design bases.

The subsequent reload core designs for operation at the EPU RTP level will take into account the above limits to ensure acceptable differences between the licensing limits and their corresponding operating values. EPU will result in a small change in fuel burnup, the amount of fuel to be used, and isotopic concentrations of the radionuclides in the irradiated fuel relative to the original level of burnup. NRC-approved limits for burnup on the AREVA 10XM fuel design are not exceeded. For an example CLTP condition, the BFN Unit 3 EOC 18 peak bundle discharge exposure is predicted to be **[ ]** GWd/MTU. The corresponding EPU ATRIUM 10XM equilibrium cycle peak bundle discharge exposure is predicted to be **[ ]** GWd/MTU. ATRIUM 10XM fuel is required to have bundle discharge exposure **[ ]** GWd/MTU. This is in compliance with the fuel dependent limitations on discharge burnup. Also, due to the higher steady-state operating power associated with the EPU, the short-term curie content of the reactor fuel increases. The BFN ATRIUM 10XM EPU equilibrium cycle average fresh bundle

* Design margin is primarily specified to ensure adequate operational flexibility during cycle operation. The margin required for the EPU core is similar to that specified for BFN cores operating at CLTP conditions. The design goals for a given cycle are reviewed between AREVA and TVA as part of the design process and may be adjusted depending on core tracking trends.

enrichment is predicted to be **[ ]**. The corresponding CLTP Unit 3 Cycle 18 average fresh bundle enrichment is expected to be **[ ]**. There is no maximum licensed bundle enrichment for AREVA fuel designs but the bundle enrichment is constrained by the maximum pellet enrichment of **[ ]**. The effects of higher power operation on radiation sources and DBA doses are discussed in Section 2.9.

Therefore, because the core design is established in accordance with approved methodology for each core reload, the assessment of this topic for BFN is acceptable.

2.8.2.1.2 Fuel Thermal Margin Monitoring Threshold

The percent power level above which fuel thermal margin monitoring is required may change with EPU. The original plant operating licenses set this monitoring threshold at a typical value of 25% of rated thermal power. For the highest power density reactors at original licensed thermal power (OLTP) conditions, this monitoring threshold was equivalent to bundle powers up to 1.2 MWt. **[**

]

For EPU, the fuel thermal margin monitoring threshold is scaled down, if necessary, to ensure that monitoring is initiated prior to exceeding an average bundle power of 1.2 MWt. For BFN, the current licensed thermal power (CLTP) of 3458 MWt maintains a bundle power below this threshold at 25% of rated power (0.25*3458 MWt/764 bundles = 1.1 MWt/bundle). However, the threshold must be adjusted to meet this average bundle power criterion for EPU conditions.

100 * (1.2 MWt/bundle * 764 bundles / 3952 MWt) = 23% of EPU RTP

A change in the fuel thermal monitoring threshold also requires a corresponding change to the Technical Specifications (TS) reactor core safety limit for reduced pressure or low core flow.

The basis for not monitoring thermal limits below this threshold is the large margin to thermal limits as described in the TS Bases. Therefore, with these large margins, there are no transients that have limiting consequences when initiated from the 0 – 23 percent power range.

2.8.2.2 Thermal Limits Assessment

Assurance that regulatory limits are not exceeded during postulated anticipated operational occurrences and accidents is accomplished by applying operating limits on the fuel. This section discusses the impact ATRIUM 10XM fuel has on thermal limits. The evaluations were performed using an equilibrium core of ATRIUM 10XM fuel. Consistent with the current practice, cycle-specific thermal limits are established or confirmed each reload based on the cycle-specific core configuration.

2.8.2.2.1 Safety Limit MCPR

The safety limit minimum critical power ratio (SLMCPR) can be affected by a new fuel design due to changes in the power and flow distributions. In addition, differences in fuel-related uncertainties will impact the SLMCPR results. The SLMCPR analysis reflects the actual core loading and is performed for each reload core (including transition cores). Therefore, because the SLMCPR is established in accordance with approved methodology for each core reload, the assessment of this topic for BFN is acceptable.

2.8.2.2.2 MCPR Operating Limit

The operating limit minimum critical power ratio (OLMCPR) is determined each cycle based on the results of the reload transient analyses. The OLMCPR for a given fuel design is dependent on its critical power performance. The OLMCPR is determined based on analyses reflecting actual core loading (including transition cores).

The OLMCPR is established to protect the sum of the change in critical power ratio (ΔCPR) for the limiting AOO event and the SLMCPR. The impact that ATRIUM 10XM fuel has on AOO events at EPU conditions is addressed in Section 2.8.5.

2.8.2.2.3 MAPLHGR Limit

The MAPLHGR Operating Limit ensures that the plant does not exceed regulatory limits established in 10 CFR 50.46 or by the fuel design limits. The MAPLHGR Operating Limit is determined by analyzing the limiting LOCA for the plant.

The ECCS performance is addressed in Section 2.8.5.6.2, and uses a reference equilibrium core of ATRIUM 10XM fuel for EPU. Compared to CLTP, this evaluation shows that no change in the MAPLHGR limit is required for EPU for Single Loop Operation (SLO) or Two Loop Operation (TLO) recirculation system operation.

Therefore, because the MAPLHGR Operating Limit is established in accordance with approved methodology for each core reload, the assessment of this topic for BFN is acceptable.

2.8.2.2.4 LHGR Operating Limit

The linear heat generation rate (LHGR) limits ensure that the plant does not exceed the thermal-mechanical design limits. LHGR limits are fuel type dependent and apply regardless of power level, and thus are not affected by EPU. The LHGR limits are evaluated each reload cycle to ensure that the established LHGR limits are applicable to the new fuel assembly design. To support operation at off-rated conditions, power- and flow-dependent multipliers are applied to the LHGR limits to ensure that the fuel meets the thermal-mechanical limits during anticipated operational occurrences. While the LHGR limits for ATRIUM 10XM fuel are not cycle-specific, the power- and flow-dependent LHGR multipliers are established each cycle because they are affected by the core response during a transient. The LHGR operating limits and the power- and flow-dependent multipliers are documented in the cycle-specific core operating limits report (COLR).

2.8.2.2.5 Power and Flow Dependent Limits

A flow-dependent multiplier is applied to the LHGR thermal limits when the plant is operating at less than 100% core flow. Flow-dependent MCPR limits ($MCPR_f$) are also established. The flow dependent limits are based on the results of the slow recirculation flow increase analysis. The flow-dependent limits are established or confirmed each cycle and are based on a conservative flow runup path.

The LHGR thermal limits are also modified by a power-dependent multiplier when the unit is operating at less than 100% power. Power-dependent MCPR ($MCPR_p$) limits are also established to support operation at off-rated conditions. These power-dependent limits are based on the results of the off-rated transient analyses performed each cycle.

2.8.2.3 Reactivity Characteristics

The higher core energy requirement of EPU has the potential to impact the hot excess reactivity and reduce operating shutdown margin. The following table provides a comparison of the ATRIUM 10XM equilibrium core to a core design for current rated power conditions.

Topic	CLTP	EPU	Exposure Basis
Maximum Hot Excess Reactivity (%Δk/k)	[ ]	[ ]	nominal
Minimum Cold Shutdown Margin (%Δk/k)	[ ]	[ ]	short

The differences in this table show this potential trend; however, the changes are within the range of those found in normal cycle to cycle variations. These cycle to cycle variations are driven primarily by changes in the previous cycle operational history as well as by the bundle designs, i.e., enrichment and gadolinia loadings.

Hot excess reactivity is a parameter of interest to operation for two reasons: 1) the magnitude determines the required rod density, and 2) the rate of change of reactivity may determine when adjustments to this rod density are required to compensate. Both the core hot excess magnitude and reactivity swing (i.e., flatness of the hot excess reactivity curve) are controlled during the cycle bundle and core design process.

The hot excess reactivity magnitude is controlled to ensure that enough rod density is available to compensate for unexpected variations in the core reactivity while maintaining the ability to control the margin to the licensed thermal limits. For example, an excessively high magnitude for hot excess reactivity may result in too many rods inserted to effectively control power peaking thereby affecting thermal limits during operation. On the opposite extreme, a very low hot excess reactivity could potentially result in a condition at which full power could not be achieved if the core reactivity was lower than predicted.

The hot excess reactivity rate of change, or reactivity swing, is of interest to operation because large changes may require additional rod pattern adjustments and/or sequence exchanges. An

EPU core is more susceptible to this because the extension of the high flow control line to the higher power level reduces the available flow window that can be used to compensate for these reactivity changes at rated power conditions. To help offset this reduction in the size of the flow window, it is desirable to design EPU cores with a flatter hot excess curve than was typically required for CLTP core designs. This was accomplished in the BFN representative ATRIUM 10XM equilibrium core design as illustrated in Figure 2.8-1. The control rod patterns established for this EPU representative core minimize the rod pattern adjustments between sequence exchanges as illustrated in Appendix A of ANP-3342P (Reference 2, Attachment 16 of the EPU LAR) demonstrating no significant adverse impact on BFN operation.

Each operating cycle is designed to meet the shutdown margin requirements specified by Technical Specification 3.1.1 to ensure that the core can be made subcritical with the highest reactivity worth control rod fully withdrawn (and the remaining blades inserted) at the most reactive condition throughout the cycle. This requirement is included in the generic fuel design criteria in Section 5.4 of ANF-89-98(P)(A) (Reference 16). Section 5.4 of Reference 17 indicates that compliance with this requirement is maintained with the performance of cycle-specific calculations. Furthermore, compliance with this Technical Specification requirement is also verified with a shutdown margin demonstration performed during the initial startup of each cycle.

The calculations involved in the cycle-specific analysis of the shutdown margin use the NRC approved CASMO-4 / MICROBURN-B2 methodology, EMF-2158(P)(A) (Reference 18). Application of this methodology to EPU conditions remains within its approval basis and SER restrictions, as addressed in ANP-2860P Revision 2 (Reference 4). This methods applicability was extended to the ATRIUM 10XM design for BFN through Supplement 1P (Reference 5) which addressed CLTP conditions. This is further extended to the ATRIUM 10XM at EPU conditions in Supplement 2P (Reference 6) which is included as Attachment 34 of the EPU LAR. The acceptability of the AREVA approved methodology for use in calculating the cold shutdown margin for BFN is specifically addressed in Section 7.1 of ANP-2860P Rev. 2, Section 5.1 of Supplement 1P, and Section 7.1 of Supplement 2P.

BFN Technical Specification 3.1.1 establish a cold shutdown margin requirement of 0.38 %Δk/k by reference to the applicable unit and cycle specific COLR. In Section 7.1 of References 4 and 6, operating experience with EPU reloads within the US was provided that shows the CASMO-4 / MICROBURN-B2 shutdown margin predictions remain consistent with the previous experience base. Furthermore, it is shown that significant margin exists to the existing criterion due to the practice of using a design target significantly larger than the criterion combined with small observed variations in the cold critical target from cycle to cycle. Therefore, it is concluded based upon observed results and the margin available that the acceptance criterion referred to in the BFN Technical Specifications remains valid for operation at EPU conditions.

Based on the previous discussions and the demonstration provided with the representative ATRIUM 10XM equilibrium core in ANP-3342P (Reference 2, Attachment 16 of the EPU LAR) it has been shown that the required hot excess reactivity and shutdown margin can be achieved for EPU through appropriate fuel and core design. Because plant reactivity margins are established in accordance with approved methodology for each core reload and the Technical Specification acceptance criterion remains supported, the assessment of these topics for EPU at BFN is acceptable.

2.8.2.4 Applicability of AREVA Methods to Extended Power Uprate

The applicability of AREVA approved methodology to EPU conditions is addressed separately in ANP-2860P Revision 2 (Reference 4). This methods applicability was extended to the ATRIUM 10XM design for BFN through Supplement 1P (Reference 5) which addressed CLTP conditions. This is further extended to the ATRIUM 10XM at EPU conditions in Supplement 2P (Reference 6) which is included as Attachment 34 of the EPU LAR. The topics related to the methodology used in the nuclear design and analysis includes the following:

- Neutronic Methods (Chapter 7 of References 4 and 6, Chapter 5 of Reference 5)
- Shutdown Margin including ability to predict cold eigenvalues (Section 7.1 of References 4 and 6, Section 5.1 of Reference 5)
- LHGR Monitoring of Advanced Fuel Designs including impact of LPRM modeling (Section 7.2 of References 4 and 6)
- Bypass Boiling evaluation for EPU conditions (Section 7.5 of Reference 4 and Section 7.3 of Reference 6)
- Normal Operation including impact on key parameters such as axial powers and void fractions (Section 7.4 of Reference 6)

Conclusion

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 BFN principal design criteria predate these criteria. The BFN 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 General Design Criteria published in 1967, the Tennessee Valley Authority (TVA), performed a comparative evaluation of the design basis of BFN with the AEC proposed General Design Criteria of 1967. The BFN UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria sent 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 BFN UFSAR where there is subject matter relating to the intent of that particular criteria.

While BFN 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 BFN comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as “draft GDC”) is contained in BFN UFSAR Appendix A: draft GDC-7. Final GDC-10 is applicable to BFN as described in “NRC Issuance of Amendments Regarding the Transition to AREVA Fuel,” dated July 3, 2012.

The Thermal and Hydraulic Design is described in BFN UFSAR Section 3.7, “Thermal and Hydraulic Design.” Power oscillations are addressed in BFN UFSAR Appendix N, “Reload Licensing Report.”

Technical Evaluation

Thermal-hydraulic stability protection during operation of BFN is provided by an Oscillation Power Range Monitoring (OPRM) system implementing the BWROG Long Term Detect and Suppress Option III solution (Reference 19). During periods of time when the hardware is not considered to be operable, protection is provided by the BWROG Backup Stability Protection (BSP) criteria provided in Reference 20.

The impact of EPU is addressed in this evaluation for the following stability related topics:

- Option III OPRM Trip-Enabled Region and Setpoint
- Option III Hot Channel Oscillation Magnitude
- Backup Stability Protection
- ATWS with Core Instability

Additionally, this evaluation addresses the impact of EPU on the hydraulic loads on the core and RCS components during normal operation and DBA conditions.

2.8.3.1 Option III

2.8.3.1.1 Option III – OPRM Trip Enabled Region and Trip Setpoint

The Option III setpoint may be affected by EPU operating conditions. The OPRM trip-enabled region will be rescaled with EPU.

Option III is a detect-and-suppress solution that combines closely spaced LPRM detectors into "cells" to effectively detect any mode of reactor instability. Plants implementing Option III must demonstrate that the Option III trip setpoint is adequate to provide SLMCPR protection for anticipated reactor instability.

BFN has adopted the stability Option III Long Term Solution (LTS) (Reference 19). Option III evaluations are core reload dependent and are performed for each fuel cycle reload. No changes to the BFN currently licensed Option III stability solution hardware and software algorithms are required for EPU operation. In the event that the OPRM system is declared inoperable, BFN will continue to operate under the BWR Owners Group (BWROG) Guidelines for Backup Stability Protection (BSP) as described in Reference 20. After the EPU is implemented, cycle-specific setpoints and BSP regions will continue to be determined and documented in the cycle-specific Reload Safety Analysis Report.

The Option III solution combines closely spaced LPRM detectors into "cells" to effectively detect either core-wide or regional modes of reactor instability. These cells are termed OPRM cells and are configured to provide local area coverage with multiple channels. The BFN Option III hardware combines the LPRM signals and evaluates the cell signals with instability detection algorithms. The Period Based Detection Algorithm (PBDA) is the only algorithm credited in the Option III licensing basis. Two defense-in-depth algorithms, referred to as the Amplitude Based Algorithm (ABA) and the Growth Rate Based Algorithm (GRA), offer a higher degree of assurance that fuel failure will not occur as a consequence of stability related oscillations.

The OPRM trip is armed only when plant operation is within the OPRM trip-enabled region. The current BFN OPRM trip-enabled region is defined as the region on the power/flow map with power ≥ 25% of RTP (thermal limit monitoring threshold) and core flow < 60% of rated core flow. For EPU, the BFN OPRM trip-enabled region is rescaled to maintain the same relative power and flow boundaries. Because the rated core flow is not changed, the 60% core flow boundary is not rescaled. The 25% rated power (CLTP) boundary is the current thermal limit monitoring threshold. This value is rescaled to 23% of EPU rated power to match the rescaled thermal limit monitoring threshold.

The BFN OPRM trip-enabled region is shown in Figure 2.8-2. The BSP evaluation described in Section 2.8.3.1.3 shows the current Option III trip-enabled region is adequate for EPU. The adequacy of the OPRM trip-enabled region will be confirmed for each fuel reload.

Stability Option III provides SLMCPR protection by generating a reactor scram if a reactor instability that exceeds the specified OPRM trip setpoints is detected. The OPRM setpoints are determined per an NRC-approved methodology (References 19 and 22). The actual reactor simulator supporting the two recirculation pump trip (2PT) setpoint calculation is the NRC approved MICROBURN-B2 (Reference 18) code.

The Option III stability reload licensing basis calculates the limiting OLMCPRs required to protect the SLMCPR for both steady-state and transient stability events as specified in the Option III methodology. These OLMCPRs are calculated for a range of OPRM setpoints for EPU operation.

Selection of appropriate OPRM trip setpoints is then based upon the OLMCPRs required to provide adequate SLMCPR protection. This determination relies on the DIVOM curve to determine the OPRM setpoints that protect the SLMCPR during an anticipated instability event. The DIVOM slope was developed based on a RAMONA5-FA evaluation in accordance with the BWROG Regional Mode DIVOM Guideline (References 21 and 22).

2.8.3.1.2 Option III – Hot Channel Oscillation Magnitude (HCOM)

The Option III setpoint may be affected by EPU operating conditions. The OPRM trip-enabled region will be rescaled with EPU.

The Option III automatic scram is provided by the OPRM system. The generic analyses for the Option III hot channel oscillation magnitude and the OPRM hardware were designed to be independent of core power.

Although the Option III solution requires cycle-specific evaluations, a demonstration analysis was performed based on the equilibrium ATRIUM 10XM reference EPU core design. The cycle-specific DIVOM calculation supports the generic slope (0.45, Figure 4-13 of Reference 19) in the OPRM setpoint calculation, as illustrated in Figure 2.8-3. Table 2.8-1 provides the results of the OPRM setpoint calculation for the ATRIUM 10XM reference EPU core. These results are based upon a SLMCPR of 1.06. Assuming a power-dependent OLMCPR of 1.40 at rated power and a flow-dependent OLMCPR at 45% rated flow of 1.51, an OPRM amplitude setpoint of 1.15 may be used without stability setting the OLMCPR. The actual setpoint will be established at the time of each fuel reload based on the cycle-specific core design.

The OPRM system is dependent upon combining a number of individual LPRM detector readings and is therefore potentially susceptible to conditions in the core that can affect the accuracy of the LPRM readings. These readings could be impacted if significant voiding were to occur in the bypass region, i.e. the region in which the LPRM strings are located. Instability events are most likely to occur at low flow conditions. EPU involves extending the existing MELLLA High Flow Control Line (HFCL) to reach the higher power condition but does not affect the absolute power and flow conditions in the area of most interest for instability events. Consequently, EPU does not adversely impact the ability of the OPRM instrumentation to function versus operation at current rated power conditions. The impact of bypass voiding at BFN on the OPRM system is addressed in Section 2.1 of Reference 4. The conclusion of this evaluation was that the steady state and dynamic effects of bypass boiling on lowering the sensitivity of individual LPRM detectors cause **[**

] OPRM signals used for comparison with the OPRM amplitude setpoint.

2.8.3.1.3 BSP Evaluation

BFN implements BSP (Reference 20) as the stability licensing basis should the Option III OPRM system be declared inoperable. The BSP evolved from the stability ICAs (Reference 23), which restrict plant operation in the high power, low core flow region of the BWR power/flow operating map. The ICAs provide guidance that reduces the likelihood of an instability event by limiting the period of operation in regions of the power/flow map most susceptible to thermal hydraulic instability.

For EPU conditions the BSP regions will continue to be the stability licensing basis for BFN (Reference 20) if the Option III OPRM system is declared inoperable. The BSP consists of two regions (I-Scram and II-Controlled Entry). These replaced the previously used ICAs that were composed of three regions (I-Scram, II-Exit, and III-Controlled Entry). The standard ICA region endpoints on the High Flow Control Line (HFCL) and on the Natural Circulation Line (NCL) define the base BSP region endpoints on the HFCL and on the NCL. The bounding plant- and cycle- specific BSP region endpoints must enclose the corresponding base BSP region endpoints on the HFCL and the NCL. If a calculated BSP region endpoint is located inside the corresponding base BSP region endpoint, the corresponding base BSP region endpoint must replace it. That is, the selected points will result in the largest, or most conservative, region sizes. The proposed BSP Scram and Controlled Entry region boundaries may be constructed by connecting the corresponding bounding endpoints on the HFCL and the NCL using the Generic Shape Function (GSF) (Reference 20).

The AREVA EPU equilibrium cycle demonstration analysis was used to determine the STAIF (Reference 24) calculated BSP endpoints for nominal feedwater temperature and minimum (or reduced) feedwater temperature as shown in Table 2.8-2 and Table 2.8-3, respectively. The power/flow statepoints provided in these tables represent the base minimal region boundaries except where noted. The limiting endpoints between the calculated and the Base BSP endpoints are used along with the GSF to construct the BSP regions for nominal feedwater temperature and minimum or reduced feedwater temperature as shown in Figure 2.8-4 and Figure 2.8-5, respectively.

2.8.3.2 ATWS with Core Instability

ATWS with core instability event occurs at natural circulation following a RPT. Therefore, it is initiated at approximately the same power level as a result of EPU operation because the MELLLA upper boundary is not increased. The core design necessary to achieve EPU operations may affect the susceptibility to coupled thermal-hydraulic/neutronic core oscillations at the natural circulation condition, but would not significantly affect the event progression. This is discussed in Section 2.2 of Reference 4.

Several factors affect the response of an ATWS instability event, including operating power and flow conditions and core design. The limiting ATWS core instability evaluation presented in References 25 and 26 was performed for an assumed plant initially operating at OLTP and the MELLLA minimum flow point.

EPU allows plants to increase their operating thermal power but does not allow an increase in control rod line. Given this, the plant will arrive at approximately the same power at natural circulation for both OLTP and EPU power levels.

Feedwater Heater Out-Of-Service (FHOOS) and Final Feedwater Temperature Reduction (FFWTR) are operational flexibility options that allow continued operation with reduced feedwater temperature.

Initial operating conditions of FHOOS and FFWTR do not significantly affect the ATWS instability response reported in References 25 and 26. The limiting ATWS evaluation assumes that all feedwater heating is lost during the event and the injected feedwater temperature approaches the lowest achievable main condenser hot well temperature. This decrease in feedwater temperature drives an increase in core power and an overall more severe event. Initial power is unchanged for both the FHOOS and FFWTR conditions - the additional reactivity associated with the reduced feedwater temperature is typically offset with control rods, as needed. For both the FHOOS and FFWTR conditions, an ATWS event analysis would be initiated from the same limiting power/flow statepoint assumed for the normal feedwater temperature case and transition to essentially the same natural circulation statepoint prior to the onset of power oscillations. Because the initial statepoint and the natural circulation statepoint are both the same for FHOOS conditions, and the change in feedwater temperature is less when starting from FHOOS and FFWTR conditions, an ATWS stability event initiated from FHOOS and FFWTR conditions will be bound by a normal feedwater temperature case.

Operator actions will mitigate an ATWS instability event. The actions contained in References 25 and 26 bound the entire BWR fleet and are applicable to BFN. The conclusion of Reference 26 and the associated NRC SER that the analyzed operator actions effectively mitigate an ATWS instability event are applicable to the operating conditions expected for EPU at BFN.

2.8.3.3 Hydraulic Loading

Operation with a mechanically different fuel design can affect the pressure differences across reactor internal components due to differences in thermal hydraulic characteristics. An evaluation of Reactor Internals Pressure Differentials (RIPD) was performed by GEH for both ATRIUM-10 and ATRIUM 10XM. The evaluation shows AREVA fuel designs are either bounded, or essentially the same as, the GE13 which has been the basis of BFN RIPD analyses. Consequently, ATRIUM-10 and ATRIUM 10XM RIPD inputs to the reactor internal structural integrity evaluation do not produce unacceptable consequences at EPU operating conditions.

<u>Conclusion</u>

The effects of the proposed EPU on the thermal and hydraulic design of the core and the RCS have been reviewed. It is concluded that the review has adequately accounted for the effects of the proposed EPU on the thermal and hydraulic design and demonstrated that the design (1) has been accomplished using acceptable analytical methods, (2) is a proven design, (3) provides acceptable margins of safety from conditions that would lead to fuel damage during normal reactor operation and AOOs, and (4) is not susceptible to thermal-hydraulic instability. It is further concluded that the effects of the proposed EPU on the hydraulic loads on the core and RCS components have been adequately accounted for. Based on this, the thermal and hydraulic design 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 thermal and hydraulic design.

2.8.4 Emergency Systems

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 BFN principal design criteria predate these criteria. The BFN 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 General Design Criteria published in 1967, the Tennessee Valley Authority (TVA), performed a comparative evaluation of the design basis of BFN with the AEC proposed General Design Criteria of 1967. The BFN UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria sent 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 BFN UFSAR where there is subject matter relating to the intent of that particular criteria.

While BFN 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 BFN comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as “draft GDC”) is contained in BFN 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 BFN 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 BFN License Renewal Safety Evaluation Report (SER), NUREG-1843, dated April 2006 (Reference 7). 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 3.2.2.

Technical Evaluation

The design pressure of the reactor vessel is not affected by fuel design and remains 1250 psig. Per the American Society of Mechanical Engineers (ASME) code, the acceptance limit for pressurization events is 110% of the design pressure, or 1375 psig, for the reactor vessel. Overpressurization analyses using an ATRIUM 10XM equilibrium core were performed using the NRC approved code COTRANSA2 (Reference 27) for the MSIV closure and turbine trip with turbine bypass failure events. The events were analyzed at 102% of EPU RTP and an initial dome pressure of 1055 psig, which is higher than the nominal dome pressure. The main steam relief valve (MSRV) setpoints provided in Reference 28 were used with 1 MSRV (with the lowest setpoint) assumed out of service. No credit was taken for the MSIV or turbine stop valve position scram.

The results show that the MSIV closure with scram on high flux (MSIVF) is the limiting overpressure event. The calculated peak vessel pressure at the bottom of the vessel is 1349 psig. The corresponding calculated peak dome pressure is 1320 psig. The peak pressure values include adjustments to address the Nuclear Regulatory Commission (NRC) concerns associated with the void-quality correlation, exposure-dependent thermal conductivity, and Doppler effects. The results remain below the 1375-psig ASME peak vessel limit and the 1325-psig dome pressure safety limit presented in the Technical Specifications. The results of the limiting ATRIUM 10XM overpressure analyses are presented in Figure 2.8-6 through Figure 2.8-9.

The adequacy of the pressure relief system is also demonstrated by the overpressure protection evaluation and by the anticipated transient without scram (ATWS) evaluation (Section 2.8.5.7) for each reload core at EPU conditions.

For BFN, no safety/relief valve (SRV) setpoint increase is needed because there is no change in the dome pressure or simmer margin. Therefore, there is no effect on valve functionality (opening/closing).

[

]

Conclusion

The effects of the proposed EPU with ATRIUM 10XM on the overpressure protection capability of the plant during power operation have been reviewed. The results of that review demonstrate that: (1) pressurization events and overpressure protection features adequately account for the effects of the proposed EPU; and (2) the plant will continue to have sufficient pressure relief capacity to ensure that pressure limits are not exceeded. Based on this, it is concluded that the overpressure protection features will continue to meet the current licensing basis following implementation of the proposed EPU. Therefore, the proposed EPU with ATRIUM 10XM is acceptable with respect to overpressure protection during power operation.

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 affect 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 BFN principal design criteria predate these criteria. The BFN 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 General Design Criteria published in 1967, the Tennessee Valley Authority (TVA), performed a comparative evaluation of the design basis of BFN with the AEC proposed General Design Criteria of 1967. The BFN UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria sent 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 BFN UFSAR where there is subject matter relating to the intent of that particular criteria.

While BFN 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 BFN comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as “draft GDC”) is contained in BFN 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 BFN 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 BFN License Renewal Safety Evaluation Report (SER), NUREG-1843, dated April 2006 (Reference 7). The license renewal evaluation associated with the SLCS System is documented in NUREG-1843, Section 2.3.3.18. Management of aging effects on the SLCS is documented in NUREG-1843, Section 3.3.2.

Technical Evaluation

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 impact of EPU is addressed in this evaluation for the following standby liquid control system related topics:

- Core Shutdown Margin
- System Performance and Hardware
- Suppression Pool Temperature Following Limiting ATWS Event

2.8.4.5.1 Core Shutdown Margin

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 directly affected by EPU. However, core loading changes can affect the core reactivity which in turn can impact SLCS system performance. This would include core loading changes required for operation at EPU conditions. For this reason, SLCS shutdown capability (in terms of the required reactor boron concentration) is reevaluated for each fuel reload.

The calculations involved in the cycle-specific analysis of the SLCS shutdown margin use the NRC approved CASMO-4 / MICROBURN-B2 methodology, EMF-2158(P)(A) (Reference 18). Application of this methodology to EPU conditions remains within its approval basis and SER restrictions, as addressed in ANP-2860P Revision 2 (Reference 4). This methods applicability was extended to the ATRIUM 10XM design for BFN through Supplement 1P (Reference 5) which addressed CLTP conditions. This is further extended to the ATRIUM 10XM at EPU conditions in Supplement 2P (Reference 6) which is included as Attachment 34 of the EPU LAR.

A demonstration evaluation of the SLCS shutdown margin has been performed using the ATRIUM 10XM EPU equilibrium core documented in ANP-3342P (Reference 2), included as Attachment 16 of the EPU LAR. The SLCS system boron concentration has been changed to 720 ppm natural Boron equivalent for EPU. The results using this boron concentration demonstrate minimum SLCS shutdown margin for the equilibrium core of 3.03 %Δk/k assuming a short EOC N-1, as shown in Tables 2.1 and 3.4 of ANP-3342P. This minimum corresponds to a short EOC cycle N-1 shutdown and occurs at BOC for cycle N. **[**

]

Therefore, the SLCS system meets the required shutdown margin capability for EPU conditions.

2.8.4.5.2 System Performance and Hardware

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 MSRVs operating in the safety relief mode. At BFN, the nominal reactor dome pressure and the MSRV setpoints are unchanged for EPU. Consequently, the capability of the BFN SLCS to provide its backup shutdown function is not affected by EPU. The SLCS is not dependent upon any other MSRV operating modes.

Based on the results of the BFN EPU ATWS analysis, the maximum reactor lower plenum pressure following the limiting ATWS event reaches 1201 psig (1216 psia) during the time the SLCS is analyzed to be in operation (see PUSAR Section 2.8.4.5.2). This result demonstrates that there is acceptable margin to the SLCS relief valve setpoint.

2.8.4.5.3 Suppression Pool Temperature Following ATWS Event

As shown in Section 2.8.5.7.2, differences in fuel design as it pertains to ATRIUM 10XM and GE14 fuel will not significantly impact the suppression pool temperature response following the ATWS event at EPU conditions.

Conclusion

The effects of the proposed EPU on the SLCS have been reviewed and it was found the SLCS adequately accounts for the EPU with ATRIUM 10XM. It was demonstrated that the system will continue to provide the function of reactivity control independent of the CRD system following implementation of the proposed EPU. Based on this, the SLCS will continue to meet the requirements of 10 CFR 50.62(c)(4) and the current licensing basis, following implementation of the proposed EPU. Therefore the proposed EPU with ATRIUM 10XM is acceptable with respect to the SLCS.

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 condition 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 BFN principal design criteria predate these criteria. The BFN 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 General Design Criteria published in 1967, the Tennessee Valley

Authority (TVA), performed a comparative evaluation of the design basis of BFN with the AEC proposed General Design Criteria of 1967. The BFN UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria sent 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 BFN UFSAR where there is subject matter relating to the intent of that particular criteria.

While BFN 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 BFN comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in BFN 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 BFN UFSAR Section 14.5.3.1, "Loss of Feedwater Heater (LFWH)." The analysis of a feedwater controller failure with maximum demand is described in BFN UFSAR Section 14.5.8.1, "Feedwater Controller Failure Maximum Demand (FWCF)." The analysis of an inadvertent opening of a Main Steam Relief Valve is described in BFN UFSAR Section 14.5.5.2, "Inadvertent Opening of an MSRV (IORV)."

Technical Evaluation

The Decrease in Feedwater Temperature limiting event (LFWH with manual flow control) and the Increase in Feedwater Flow limiting event (FWCF) are confirmed to be within the BFN reload evaluation scope.

The FWCF and LFWH are performed with the NRC-approved methods described in References 12, 27, 29, and Reference 30 respectively. The transient evaluation initial conditions are provided in Table 2.8-4, and the results of the EPU evaluations are reported in Table 2.8-5, Table 2.8-6, and Table 2.8-7. The results of the limiting FWCF event are presented in Figure 2.8-10 through Figure 2.8-12.

The pressure regulator failure open (PRFO) transient event causes no significant threat to the fuel thermal margins. The peak heat flux and fuel surface heat flux do not exceed the initial power and no fuel damage occurs. Therefore, this event is non-limiting relative to anticipated operational transient (AOT) thermal operating limits. No transient analysis required for EPU.

The inadvertent opening of a MSRV (IORV) results in a mild depressurization event. The peak heat flux does not exceed the initial power and no SAFDLs are challenged. Therefore, this event is non-limiting relative to AOT thermal operating limits.

Conclusion

The analyses of the excess heat removal events described above have been reviewed to ensure they have adequately accounted for operation of the plant at the proposed power level with ATRIUM 10XM fuel and were performed using acceptable analytical models. Based on these analyses, it has been demonstrated that the reactor protection and safety systems will continue to ensure that the SAFDLs and RCPB pressure limits will not be exceeded as a result of these events. Based on this, it is concluded that the plant will continue to meet the requirements of the current licensing basis, following implementation of the proposed EPU with ATRIUM 10XM. Therefore, the proposed EPU with ATRIUM 10XM is acceptable with respect to the events stated.

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 condition 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 BFN principal design criteria predate these criteria. The BFN 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 General Design Criteria published in 1967, the Tennessee Valley Authority (TVA), performed a comparative evaluation of the design basis of BFN with the AEC proposed General Design Criteria of 1967. The BFN UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation

discusses each of the groups of criteria sent 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 BFN UFSAR where there is subject matter relating to the intent of that particular criteria.

While BFN 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 BFN comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in BFN 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 BFN 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 BFN 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 BFN 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 BFN UFSAR.

Technical Evaluation

The Loss of External Load limiting event (Generator Load Rejection with Steam Bypass Failure (LRNB)) and the Turbine Trip limiting event (Trip with Steam Bypass Failure (TTNB)) are confirmed to be within the BFN reload evaluation scope.

The LRNB and TTNB are performed with the NRC-approved methods described in References 12, 27, and 29. The transient evaluation initial conditions are provided in Table 2.8-4, and the results of the EPU evaluations are reported in Table 2.8-5, Table 2.8-6, and Table 2.8-7. The results of the limiting LRNB event are presented in Figure 2.8-13 through Figure 2.8-15.

Closure of the MSIV causes a pressurization and subsequent power increase. Though it is generally bound by the LRNB, the MSIV closure will be performed at EPU conditions. The transient evaluation initial conditions are provided in Table 2.8-4, and the results of the EPU evaluations are reported in Table 2.8-5 and Table 2.8-6.

The Steam Pressure Regulator Failure Closed event was eliminated as an AOT by the installation of a digital fault-tolerant main turbine electro-hydraulic control system. Therefore this event is not required for EPU.

The Loss of Condenser Vacuum event is equivalent to a turbine trip with bypass operable event and is therefore bound by the TTNB event.

Conclusion

The analyses of the decrease in heat removal (i.e., an increase in reactor pressure) events described above have been reviewed to ensure they have adequately accounted for operation of the plant at the proposed power level with ATRIUM 10XM fuel and were performed using acceptable analytical models. The results of those analyses demonstrate that the reactor protection and safety systems will continue to ensure that the SAFDLs and the RCPB pressure limits will not be exceeded as a result of these events. Based on this, it is concluded that the plant will continue to meet the requirements of the current licensing basis, following implementation of the proposed EPU with ATRIUM 10XM. Therefore, TVA finds the proposed EPU with ATRIUM 10XM acceptable with respect to the events stated.

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 condition 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 BFN principal design criteria predate these criteria. The BFN 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 General Design Criteria published in 1967, the Tennessee Valley Authority (TVA), performed a comparative evaluation of the design basis of BFN with the AEC proposed General Design Criteria of 1967. The BFN UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria sent 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 BFN UFSAR where there is subject matter relating to the intent of that particular criteria.

While BFN 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 BFN comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in BFN 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 BFN UFSAR Section 14.5.5.4, "Loss of Auxiliary Power."

Technical Evaluation

Loss of auxiliary power can occur if all external grid connections are lost or if faults occur in the auxiliary power system itself causing two types of transients: Loss of Auxiliary Power Transformers and Loss of Auxiliary Power Grids. Operation at EPU conditions will not affect the key parameters of this event – MSRV pressure setpoint, pump trip setpoints, MSIV trip setpoint, pump coastdown rates, scram setpoints, etc. Therefore, the long-term water level response of Loss of Auxiliary Power is bounded by the Loss of Feedwater Flow long-term water level transient. The ΔCPR and vessel pressure for the Loss of Auxiliary Power event is bounded by the LRNB event.

Therefore this event is not required for EPU.

Conclusion

The loss of non-emergency alternating current (AC) power to station auxiliaries event described above has been evaluated to ensure it has adequately accounted for operation of the plant at the proposed power level with ATRIUM 10XM. The results of that evaluation demonstrate that the reactor protection and safety systems will continue to ensure that the SAFDLs and the RCPB pressure limits will not be exceeded as a result of this event. Based on this, TVA concludes that the plant will continue to meet the requirements of the current licensing basis, following implementation of the proposed EPU with ATRIUM 10XM. Therefore, TVA finds the proposed EPU with ATRIUM 10XM acceptable with respect to the events stated.

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 condition 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 BFN principal design criteria predate these criteria. The BFN 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 General Design Criteria published in 1967, the Tennessee Valley Authority (TVA), performed a comparative evaluation of the design basis of BFN with the AEC proposed General Design Criteria of 1967. The BFN UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria sent 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 BFN UFSAR where there is subject matter relating to the intent of that particular criteria.

While BFN 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 BFN comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in BFN 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 BFN UFSAR Section 14.5.5.3, "Loss of Feedwater Flow."

<u>Technical Evaluation</u>

2.8.5.2.3.1 Loss of Feedwater Flow Event

For the loss of feedwater flow (LOFW) event, adequate transient core cooling is provided by maintaining the water level inside the core shroud above the top of active fuel (TAF). A loss of all feedwater flow was performed for BFN at EPU conditions. This analysis assumed failure of the high pressure coolant injection (HPCI) system and used only the reactor core isolation cooling (RCIC) system to restore the reactor water level.

Because of the extra decay heat from EPU, slightly more time is required for the automatic systems to restore water level. Operator action is only needed for long-term plant shutdown. The results of the LOFW analysis for BFN show that the minimum water level is 66 inches above the TAF at EPU conditions. After the water level is restored, the operator manually controls the water level, reduces reactor pressure, and initiates residual heat removal (RHR) shutdown cooling. This sequence of events does not require any new operator actions or shorter operator response times.

The following is the general sequence of events in the analysis. The reactor is assumed to be at 102% of the EPU power level when the LOFW occurs. The initial level in the model is conservatively set at the low-level scram setpoint and reactor feedwater is instantaneously isolated at event initiation. Scram is initiated at the start of the event. When the level decreases to the low-low level setpoint, the RCIC system is initiated. The RCIC flow to the vessel begins at 141 seconds into the event, minimum level is reached at 1007 seconds and level is recovered after that point. Only RCIC flow is credited to recover the reactor water level. There are no additional failures assumed beyond the failure of the HPCI system.

The only other key analysis assumption for the LOFW analysis was the assumed decay heat level of ANS 5.1-1979 with a two-sigma uncertainty. The assumed decay heat level for the EPU analysis was ANS 5.1-1979 decay heat +10%, which bounds ANS 5.1-1979 + two sigma. Thus, the key analytical assumptions are the same or conservative relative to the current licensing basis.

This LOFW analysis is performed to demonstrate acceptable RCIC system performance. The design basis criterion for the RCIC system is confirmed by demonstrating that it is capable of maintaining the water level inside the shroud above the TAF during the LOFW transient. The minimum level is maintained at least 66 inches above the TAF, thereby demonstrating acceptable RCIC system performance. There are no applicable equipment out of service assumptions for this transient.

An operational requirement is that the RCIC system restores the reactor water level while avoiding automatic depressurization system (ADS) timer initiation and MSIV closure activation functions associated with the low-low-low reactor water level setpoint (Level 1). This requirement is intended to avoid unnecessary initiations of safety systems. This requirement is not a safety-related function. The results of the LOFW analysis for BFN show that the nominal Level 1 setpoint trip is avoided.

Therefore, the LOFW event meets all design criteria.

2.8.5.2.3.2 Loss of One Feedwater Pump

Higher reactor power and decay heat results in a lower reactor water level for loss of water level events.

In order to avoid unnecessary reactor scrams, an operational requirement is applied that the water level remains above the low level setpoint (Level 3) during a single feedwater pump trip (SFWPT) event. The SFWPT analysis results for a full core of ATRIUM 10XM fuel at EPU conditions show that no scram occurs since the minimum water level is 10 inches above the Level 3 setpoint.

Conclusion

The analysis of the Loss of Normal Feedwater Flow event has been reviewed to ensure it adequately accounted for operation of the plant at the proposed power level with ATRIUM 10XM and was performed using acceptable analytical models. The results of that analysis demonstrate that the reactor protection and safety systems will continue to ensure that the SAFDLs and the RCPB pressure limits will not be exceeded as a result of this event. Based on this, it is concluded that the plant will continue to meet the requirements of the current licensing basis, following implementation of the proposed EPU with ATRIUM 10XM. Therefore, the proposed EPU with ATRIUM 10XM is acceptable with respect to the events stated.

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 condition 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 BFN principal design criteria predate these criteria. The BFN 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 General Design Criteria published in 1967, the Tennessee Valley Authority (TVA), performed a comparative evaluation of the design basis of BFN with the AEC proposed General Design Criteria of 1967. The BFN UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation

discusses each of the groups of criteria sent 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 BFN UFSAR where there is subject matter relating to the intent of that particular criteria.

While BFN 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 BFN comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in BFN 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 BFN UFSAR Section 14.5.6, "Events Resulting in Core Coolant Flow Decrease."

Technical Evaluation

The recirculation pump trip (RPT) event of two pumps, also known as 2PT, has no impact on fuel integrity. Neutron flux, surface heat flux, vessel pressure, and steam line pressure do not exceed their initial values. This event is bound by TTNB. Therefore, analysis of this event is not required for EPU.

The RPT event of a single pump has no impact on fuel integrity. Neutron flux, surface heat flux, vessel pressure, and steam line pressure do not exceed their initial values. This event is bound by TTNB. Therefore, analysis of this event is not required for EPU.

The consequences of the recirculation flow controller failure (decreasing flow) are less severe than RPT and pump seizure events. Therefore, analysis of this event is not required for EPU.

Conclusion

The decrease in reactor coolant flow events have been evaluated to ensure they have adequately accounted for operation of the plant at the proposed power level with ATRIUM 10XM. The results of that evaluation demonstrate that the reactor protection and safety systems will continue to ensure that the SAFDLs and the RCPB pressure limits will not be exceeded as a result of this event. Based on this, TVA concludes that the plant will continue to meet the requirements of the current licensing basis, following implementation of the proposed EPU with ATRIUM 10XM. Therefore, the proposed EPU with ATRIUM 10XM is acceptable with respect to the events stated.

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 BFN principal design criteria predate these criteria. The BFN 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 General Design Criteria published in 1967, the Tennessee Valley Authority (TVA), performed a comparative evaluation of the design basis of BFN with the AEC proposed General Design Criteria of 1967. The BFN UFSAR, Appendix A, “Conformance to AEC Proposed General Design Criteria,” contains this comparative evaluation. This evaluation discusses each of the groups of criteria sent 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 BFN UFSAR where there is subject matter relating to the intent of that particular criteria.

While BFN 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 BFN comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as “draft GDC”) is contained in BFN UFSAR Appendix A: draft GDCs-32, 33, 34, and 35. Final GDC-27 is applicable to BFN as described in “NRC Issuance of Amendments Regarding the Transition to Areva Fuel,” dated 7/3/2012.

The analysis of a one recirculation pump seizure accident is described in BFN UFSAR Section 14.5.6.4 “Recirculation Pump Seizure.”

Technical Evaluation

The pump seizure event causes a ΔCPR of 0.10 or less (Reference 31), which is well below the results for the limiting rated power AOO events. Therefore, the results for the pump seizure event in two loop operation (TLO) are conservatively bounded by the limiting rated power AOO events. Analysis of this event is not required for EPU.

The pump shaft break event is bound by the pump seizure event. Therefore analysis of this event is not required for EPU.

Conclusion

The sudden decrease in core coolant flow events have been evaluated to ensure they have adequately accounted for operation of the plant at the proposed power level with ATRIUM 10XM. The results of that evaluation demonstrate that the reactor protection and safety systems will continue to ensure that the SAFDLs and the RCPB pressure limits will not be exceeded as a result of this event. Based on this, it is concluded that the plant will continue to meet the requirements of the current licensing basis, following implementation of the proposed EPU with ATRIUM 10XM. Therefore, the proposed EPU with ATRIUM 10XM is acceptable with respect to the events stated.

2.8.5.4 Reactivity and Power Distribution Anomalies

2.8.5.4.1 Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition

Regulatory Evaluation

An uncontrolled control rod assembly withdrawal from subcritical or low power startup conditions may be caused by a malfunction of the reactor control or rod control systems. This withdrawal will uncontrollably add positive reactivity to the reactor core, resulting in a power excursion

The 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 BFN principal design criteria predate these criteria. The BFN 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 General Design Criteria published in 1967, the Tennessee Valley Authority (TVA), performed a comparative evaluation of the design basis of BFN with the AEC proposed General Design Criteria of 1967. The BFN UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria sent 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 BFN UFSAR where there is subject matter relating to the intent of that particular criteria.

While BFN 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 BFN comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in BFN UFSAR Appendix A: draft GDCs-14, 15, and 31. Final GDC-10 is applicable to BFN as described in "NRC Issuance of Amendments Regarding the Transition to AREVA Fuel," dated 7/3/2012.

The analysis of a rod withdrawal error transient is described in BFN UFSAR Section 14.5.4, "Events Resulting in a Positive Reactivity Insertion."

Technical Evaluation

The evaluation of the Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition event for EPU is a comparison of the expected maximum increase in peak fuel enthalpy and the acceptance criterion of 170 cal/gram. The BFN ATRIUM 10XM EPU PUREC core is described in ANP-3342P (Reference 2) included as Attachment 16 of the EPU LAR. This core consists of only AREVA ATRIUM 10XM assemblies and the EPU is limited to ≤ 120% of OLTP. There is no change to the reactor manual control system or control rod hydraulic control units for EPU. The RWM installed provides the same level of protection of AREVA fuel following EPU as at CLTP provided the power increase is ≤ 120% of OLTP, and BPWS is used at power levels below the lower LPSP AL. No change in peak fuel enthalpy is expected due to EPU because an RWE is a localized low-power event. If the peak fuel rod enthalpy is conservatively assumed to increase by a factor of 1.2, the RWE peak fuel enthalpy at EPU will be 72 cal/gram. This enthalpy is well below the acceptance criterion of 170 cal/gram.

For BFN OLTP, the low power RWE was evaluated and the maximum deposited enthalpy was determined to be 60 cal/gram. AREVA performed a calculation similar to the original analysis for an EPU core and confirmed that 60 cal/gram bounded the AREVA fuel. A full rod withdrawal was assumed and the trip set point was not evaluated. The evaluation used a rod withdrawal rate of 3.6 in/sec and did not credit reactor scram.

The actual evaluation is independent of the event initialization method. The analysis supports an uncontrolled assembly withdrawal from the subcritical conditions which may be caused by a malfunction of the reactor control or rod control system or operator error.

Conclusion

The analysis of the uncontrolled control rod assembly withdrawal from a subcritical or low power startup condition was reviewed to ensure it has adequately accounted for the changes in core design necessary for operation of the plant at the proposed power level. The evaluation approach is consistent with that described in the CLTR. Analyses has been performed for BFN to confirm that fuel design limits and RCPB pressure limits will not be exceeded under EPU conditions. Based on this, the plant 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 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 RPS 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 BFN principal design criteria predate these criteria. The BFN 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 General Design Criteria published in 1967, the Tennessee Valley Authority (TVA), performed a comparative evaluation of the design basis of BFN with the AEC proposed General Design Criteria of 1967. The BFN UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria sent 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 BFN UFSAR where there is subject matter relating to the intent of that particular criteria.

While BFN 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 BFN comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as “draft GDC”) is contained in BFN UFSAR Appendix A: draft GDCs-14, 15, and 31. Final GDC-10 is applicable to BFN as described in “NRC Issuance of Amendments Regarding the Transition to AREVA Fuel,” dated 7/3/2012.

The analysis of a rod withdrawal error transient is described in BFN UFSAR Section 14.5.4, “Events Resulting in a Positive Reactivity Insertion.”

Technical Evaluation

The Uncontrolled Control Rod Assembly Withdrawal at power (RWE) event is confirmed to be within the BFN reload evaluation scope. The RWE analytical process is described in the NRC approved licensing topical report XN-NF-80-19(P)(A) (Reference 32). The analysis is performed using the approved MICROBURN-B2 reactor simulator code (Reference 18) and approved CPR correlations, such as the ACE CPR correlation for the ATRIUM 10XM fuel design (References 33 and 34). The corresponding thermal-mechanical basis of AREVA fuel is analyzed using the approved RODEX4 methodology described in Reference 35.

These methodologies are utilized to perform cycle specific evaluations which account for the changes in core design or operating strategy. This includes any changes required for operation at EPU conditions. Application of this methodology to EPU conditions remains within its approval basis and SER restrictions, as addressed in ANP-2860P Revision 2 (Reference 4). This methods applicability was extended to the ATRIUM 10XM design for BFN through Supplement 1P (Reference 5) which addressed CLTP conditions. This is further extended to the ATRIUM 10XM at EPU conditions in Supplement 2P (Reference 6) which is included as Attachment 34 of the EPU LAR.

A cycle specific evaluation is demonstrated on a representative EPU core. The event was analyzed at EPU conditions and resulted in an unblocked ΔCPR of 0.27. The ΔCPR versus RBM setpoint is provided in Table 2.8-8 and the corresponding RBM operability requirements are provided in Table 2.8-9. This category of transient is included with other potentially limiting transients in the determination of the required OLMCPR; thus, it is acceptable.

The actual evaluation is independent of the event initialization method. The analysis supports an uncontrolled assembly withdrawal at power which may be caused by a malfunction of the reactor control or rod control system or operator error.

Conclusion

A specific reload analyses has been performed to confirm that fuel design limits and RCPB pressure limits will not be exceeded under EPU conditions. Based on this, the plant 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 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 condition 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 BFN principal design criteria predate these criteria. The BFN 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 General Design Criteria published in 1967, the Tennessee Valley

Authority (TVA), performed a comparative evaluation of the design basis of BFN with the AEC proposed General Design Criteria of 1967. The BFN UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria sent 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 BFN UFSAR where there is subject matter relating to the intent of that particular criteria.

While BFN 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 BFN comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in BFN 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 BFN as described in "NRC Issuance of Amendments Regarding the Transition to Areva Fuel," dated July 3, 2012.

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 BFN Sections 14.5.7.1, "Recirculation Flow Controller Failure – Increasing Flow" and 14.5.7.2, "Startup of Idle Recirculation Loop."

Technical Evaluation

The fast recirculation pump flow runup event involves an unplanned increase in core coolant flow resulting from a control system malfunction. The rapid increase in core inlet flow causes a large neutron flux peak which may scram the reactor. Generally this event is non-limiting compared to other AOO events. The event was analyzed to verify it is non-limiting. The statepoint 66% RTP and 52% rated core flow was analyzed. The event initiates at 1.0 seconds with an average runup rate of 745 rpm for the first second. Then it reaches 1725 rpm at approximately 2.4 seconds. The results of the EPU evaluation are provided in Table 2.8-5 and Table 2.8-6.

The Slow Flow Runup event was analyzed to determine the flow-dependent operating limits ($MCPR_f$ and $LHGRFAC_f$). Limiting power/ flow conditions for manual flow control and the TLO SLMCPR were used. Analyses support TLO core flow run-up capability of 107% for operation up to 105% rated core flow. The results of the EPU evaluation are provided in Table 2.8-10.

Per NUREG–0800 Section 15.4.4-15.4.5, “Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature, and Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate,” Revision 2, March 2007 (Reference 36), for a BWR, “Startup of a Recirculation Loop at an Incorrect Temperature event” is called the “Startup of an Idle Recirculation Pump” event.

The Startup of an Idle Recirculation Loop event is non-limiting compared to other AOO events. Therefore this event is not required for EPU.

Conclusion

The analyses of the increase in core flow events described above have been reviewed to ensure they have adequately accounted for operation of the plant at the proposed power level with ATRIUM 10XM and were performed using acceptable analytical models. The results of those analyses demonstrate that the reactor protection and safety systems will continue to ensure that the SAFDLs and the RCPB pressure limits will not be exceeded as a result of these events. Based on this, it is concluded that the plant will continue to meet the requirements of the current licensing basis, following implementation of the proposed EPU with ATRIUM 10XM. Therefore, the proposed EPU with ATRIUM 10XM is acceptable with respect to the events stated.

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 BFN principal design criteria predate these criteria. The BFN 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 General Design Criteria published in 1967, the Tennessee Valley Authority (TVA), performed a comparative evaluation of the design basis of BFN with the AEC proposed General Design Criteria of 1967. The BFN UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria sent 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 BFN UFSAR where there is subject matter relating to the intent of that particular criteria.

While BFN 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 BFN comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in BFN UFSAR Appendix A: draft GDC-32.

The analysis of a control rod drop accident is described in BFN UFSAR Section 14.6.2, “Control Rod Drop Accident (CRDA).”

Technical Evaluation

The spectrum of CRDAs does not change with EPU. The evaluation of a CRDA for the BFN EPU is a comparison of the expected maximum increase in peak fuel enthalpy with the acceptance criterion of 280 cal/gram and comparison of the number of rods failed to that used in radiological assessment. For BFN, the dose consequence criterions are confirmed by verifying that the number of fuel rod failures does not exceed 850. This CRDA evaluation is performed on a cycle specific basis using AREVA methodology.

The CRDA analytical process is generically described in the NRC approved licensing topical report XN-NF-80-19(P)(A) (Reference 32). A cycle specific analysis using the approved MICROBURN-B2 reactor simulator code (Reference 18) results in the deposited fuel enthalpy for the analyzed rod withdrawal sequence. This cycle specific evaluation accounts for the changes in core design or operating strategy such as those required for operation at EPU conditions. Application of AREVA methodology to EPU conditions remains within its approval basis and SER restrictions, as addressed in ANP-2860P Revision 2 (Reference 4). This methods applicability was extended to the ATRIUM 10XM design for BFN through Supplement 1P (Reference 5) which addressed CLTP conditions. This is further extended to the ATRIUM 10XM at EPU conditions in Supplement 2P (Reference 6) which is included as Attachment 34 of the EPU LAR.

The cycle to cycle results of the CRDA may vary. However, each cycle is specifically evaluated based on the actual core loading and projected operation. The impact of EPU is reflected in the fuel bundle design and operating history which are directly included in the cycle specific evaluation of the CRDA. Control Rod Sequencing at BFN for CLTP and EPU follows the BPWS. There is no change to the reactor manual control system or control rod hydraulic control units for EPU. The RWM installed at BFN provides the same level of protection for the fuel following EPU as at CLTP provided that BPWS is used at power levels below the LPSP AL. The evaluation of this event for the BFN EPU on a cycle specific bases requires that deposited enthalpy is below the acceptance criteria of 280 cal/gm.

The results from the evaluation of the representative EPU core show that the deposited enthalpy is well below the acceptance criteria.

Constraint	Criteria	Value	Discussion
Peak Fuel Enthalpy	<230 cal/gram	142.0	280 cal/g is the current licensing limit. Meeting the lower 230 cal/g criteria ensures that the licensing limit is met.
Peak Fuel Enthalpy	<170 cal/gram	142.0	Fuel Failure Threshold
Number of failed Rods	<850	0	No rods exceed the 170 cal/g failure threshold since peak deposited enthalpy is below this limit.

Conclusion

The CRDA has been evaluated to account for operation of the plant at the proposed power level. The evaluation is consistent with the approach described in the CLTR. A plant specific reload analyses has been performed for BFN to confirm that fuel design limits and RCPB pressure limits will not be exceeded under EPU conditions. Based on this, the plant will continue to meet the requirements of the current licensing basis 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 overpressurization of the RCS. Alternatively, a power level decrease and depressurization may result. Reactor protection and safety systems are actuated to mitigate these events.

The 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 BFN principal design criteria predate these criteria. The BFN 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 General Design Criteria published in 1967, the Tennessee Valley Authority (TVA), performed a comparative evaluation of the design basis of BFN with the AEC proposed General Design Criteria of 1967. The BFN UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation

discusses each of the groups of criteria sent 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 BFN UFSAR where there is subject matter relating to the intent of that particular criteria.

While BFN 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 BFN comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in BFN UFSAR Appendix A: draft GDC-29. There is no draft GDC directly applicable to the final GDC-15. Final GDC-10 is applicable to BFN as described in "NRC Issuance of Amendments Regarding the Transition to Areva Fuel," dated July 3, 2012.

The analysis of an event that involves inadvertent operation of ECCS and increases reactor coolant inventory is described in BFN UFSAR Section 14.5.3, "Events Resulting in a Reactor Vessel Water Temperature Decrease."

Technical Evaluation

The inadvertent HPCI pump start (IHPS) event involves an increase in the vessel inventory due to the inadvertent injection of HPCI flow into the core which may threaten thermal margins. Generally the results are non-limiting compared to other AOO events and therefore, is not analyzed on a reload specific basis. IHPS was analyzed at EPU conditions to ensure it remains non-limiting using the conditions provided in Table 2.8-4.

For EPU conditions, the calculation confirms high water level (Level 8) trip of the main turbine does not occur, thereby ensuring the event remains non-limiting. The results are provided in Table 2.8-5 and Table 2.8-6.

Conclusion

The analyses of the inadvertent operation of emergency core cooling system (ECCS) or malfunction that increases reactor coolant inventory have been reviewed to ensure they have adequately accounted for operation of the plant at the proposed power level with ATRIUM 10XM and were performed using acceptable analytical models. The results of those analyses demonstrate that the reactor protection and safety systems will continue to ensure that the SAFDLs and the RCPB pressure limits will not be exceeded as a result of these events. Based on this, it is concluded that the plant will continue to meet the requirements of the current licensing basis, following implementation of the proposed EPU with ATRIUM 10XM. Therefore, the proposed EPU with ATRIUM 10XM is acceptable with respect to the events stated.

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 BFN principal design criteria predate these criteria. The BFN 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 General Design Criteria published in 1967, the Tennessee Valley

Authority (TVA), performed a comparative evaluation of the design basis of BFN with the AEC proposed General Design Criteria of 1967. The BFN UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria sent 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 BFN UFSAR where there is subject matter relating to the intent of that particular criteria.

While BFN 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 BFN comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in BFN UFSAR Appendix A: draft GDC-29. There is no draft GDC directly applicable to the final GDC-15. Final GDC-10 is applicable to BFN as described in "NRC Issuance of Amendments Regarding the Transition to Areva Fuel," dated July 3, 2012.

The analysis of an event that results in an inadvertent opening of a pressure relief valve is described in BFN UFSAR Section 14.5.5.2, "Inadvertent Opening of a MSRV (IORV)."

Technical Evaluation

The IORV event results in a mild depressurization. The peak heat flux does not exceed the initial power and no SAFDLs are challenged. Therefore, this event is non-limiting relative to AOT thermal operating limits and is not required for EPU.

Conclusion

The inadvertent opening of a pressure relief valve events have been evaluated to ensure they have adequately accounted for operation of the plant at the proposed power level with ATRIUM 10XM. The results of that evaluation demonstrate that the reactor protection and safety systems will continue to ensure that the SAFDLs and the RCPB pressure limits will not be exceeded as a result of this event. Based on this, it is concluded that the plant will continue to meet the requirements of the current licensing basis, following implementation of the proposed

EPU with ATRIUM 10XM. Therefore, the proposed EPU with ATRIUM 10XM is acceptable with respect to the events stated.

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 BFN principal design criteria predate these criteria. The BFN 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 General Design Criteria published in 1967, the Tennessee Valley

Authority (TVA), performed a comparative evaluation of the design basis of BFN with the AEC proposed General Design Criteria of 1967. The BFN UFSAR, Appendix A, “Conformance to AEC Proposed General Design Criteria,” contains this comparative evaluation. This evaluation discusses each of the groups of criteria sent 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 BFN UFSAR where there is subject matter relating to the intent of that particular criteria.

While BFN 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 BFN comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as “draft GDC”) is contained in BFN UFSAR Appendix A: draft GDCs-40 and 42. Final GDCs-27 and 35 are applicable to BFN as described in “NRC Issuance of Amendments Regarding the Transition to AREVA Fuel,” dated July 3, 2012.

The analysis of a loss-of-coolant accident is described in BFN 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 BFN License Renewal Safety Evaluation Report (SER), NUREG-1843, dated April 2006 (Reference 7). The license renewal evaluations associated with HPCI, Core Spray, RHR, and ADS are located in NUREG-1843 section 2.3.2.

Technical Evaluation

The purpose of the LOCA-ECCS analysis is to specify the maximum average planar linear heat generation rate (MAPLHGR) limit versus exposure for ATRIUM 10XM and ATRIUM-10 fuel designs and to demonstrate that the MAPLHGR limit is adequate to ensure that the LOCA-ECCS criteria in 10 CFR 50.46 are satisfied for operation at or below the limit for EPU operation. The results of a loss-of-coolant accident (LOCA) break spectrum analysis for BFN at EPU conditions are documented in this report.

The EPU break spectrum report (Reference 37) and MAPLHGR analysis reports for ATRIUM 10XM (Reference 38) and ATRIUM-10 (Reference 39) fuel designs have been submitted by AREVA that document detailed ECCS-LOCA analysis. No ECCS changes are required to meet LOCA analysis acceptance criteria.

The purpose of the break spectrum analysis is to identify the parameters that result in the highest calculated PCT during a postulated LOCA. The LOCA parameters identified in the break spectrum report include the following.

- Break location
- Break type (double-ended guillotine (DEG) or split)
- Break size
- Limiting ECCS single failure
- Axial power shape (top- or mid-peaked)

The purpose of the MAPLHGR report is to document that the following fuel dependent 10 CFR 50.46 criteria are satisfied and establish the fuel dependent MAPLHGR operating limit.

- Licensing PCT
- Maximum local cladding oxidation
- Core-wide metal water reaction

2.8.5.6.2.1 High Pressure Coolant Injection System

The main purpose of the high-pressure coolant injection (HPCI) system is to provide makeup water to the reactor vessel during a small break LOCA that does not rapidly depressurize the reactor vessel. The modeled characteristics of the HPCI system are detailed in Table 4.4 of Reference 37. The adequacy of the HPCI system is demonstrated by the acceptable results of the evaluation. The HPCI system performance requirements are not impacted by EPU.

2.8.5.6.2.2 Low Pressure Coolant Injection System

The LPCI mode of the RHR system provides a source of cooling water during a LOCA. The modeled characteristics of the LPCI system are detailed in Table 4.5 of Reference 37. The adequacy of the LPCI system is demonstrated by the acceptable results of the evaluation. The LPCI system performance requirements are not impacted by EPU.

2.8.5.6.2.3 Low Pressure Core Spray System

The low pressure core spray (LPCS) provides a source of cooling water during a LOCA and the potential for cooling flow downward from the upper plenum to the top of the fuel assembly. The modeled characteristics of the LPCS system are detailed in Table 4.6 of Reference 37. LPCS down cooling is not credited until the time of rated spray. After the time of rated spray, the convective heat transfer coefficients specified in Appendix K are justified for AREVA fuel types as noted in Reference 40. The adequacy of the LPCS system is demonstrated by the acceptable results of the evaluation. The LPCS system performance requirements are not impacted by EPU.

2.8.5.6.2.4 Automatic Depressurization System

The automatic depressurization system (ADS) reduces pressure during a small break LOCA resulting in an earlier initiation of low pressure ECCS. The modeled characteristics of the ADS system are detailed in Table 4.7 of Reference 37. The adequacy of the ADS system is demonstrated by the acceptable results of the evaluation. The ADS system performance requirements are not impacted by EPU.

2.8.5.6.2.5 Emergency Core Cooling Performance

The analyses were performed with LOCA Evaluation Models developed by AREVA and approved for reactor licensing analyses by the U.S. Nuclear Regulatory Commission (NRC). The models and computer codes used by AREVA for LOCA analyses are collectively referred to as the EXEM BWR-2000 Evaluation Model. The EXEM BWR-2000 Evaluation Model and NRC approval are documented in References 8 - 12 and 40. **[**

]

The BFN ECCS is designed to provide protection against postulated LOCAs caused by ruptures in the primary system piping. The ECCS performance characteristics are not changed for EPU. The effects of EPU on ECCS-LOCA response is evaluated on a plant-specific basis. ECCS-LOCA performance analyses demonstrate that the 10 CFR 50.46 requirements continue to be met at the EPU rated thermal power conditions.

The basic break spectrum response is not affected by EPU. For a BWR, a LOCA may occur over a wide spectrum of break locations and sizes. Responses to the break vary significantly over the break spectrum. The largest possible break is a double-ended rupture of a recirculation pipe; however, this is not necessarily the most severe challenge to the emergency core cooling system (ECCS). A double-ended rupture of a main steam line causes the most rapid primary system depressurization, but because of other phenomena, steam line breaks are seldom limiting with respect to the criteria of 10 CFR 50.46. Special analysis considerations are required when the break is postulated to occur in a pipe that is used as the injection path for an ECCS (e.g. core spray line). Although these breaks are relatively small, their existence disables the function of an ECCS. In addition to break location dependence, different break sizes in the same pipe produce quite different event responses, and the largest break area is not necessarily the most severe challenge to the event acceptance criteria. Because of these complexities, an analysis covering the full range of break sizes and locations was required.

For SLO, a multiplier is applied to the Two-Loop MAPLHGR Operation limits. The SLO multiplier is established such that the PCT for SLO is less than the limiting PCT for two-loop operation.

[

] At EPU power condition, the MELLLA core flow extends to approximately 99.0 % of rated core flow. Therefore, the EPU analysis results at rated power and flow are applied to the MELLLA condition. Also, the effect of ICF on PCT is acceptable with EPU. Thus the SLO, MELLLA, and ICF domain remain valid with EPU.

2.8.5.6.2.5.1 Large Break Peak Clad Temperature – Break spectrum (ECCS-LOCA)

The BFN break spectrum response is determined by the ECCS network design that is common to all BWRs. The BFN reactors are small break limited as determined by AREVA's EXEM BWR-2000 evaluation model (Reference 8). This trend in the break spectrum analysis was not impacted by EPU.

2.8.5.6.2.5.2 Small Break Peak Clad Temperature – Limiting Case (ECCS-LOCA)

The PCT for the limiting LOCA is determined primarily by the hot bundle power, which is unchanged with EPU. In the BFN analysis, the hot bundle is assumed to be operating at the thermal limits (MCPR, MAPLHGR, and LHGR); these limits are not changed for EPU.

A complete analysis for a given break size starts with the specification of fuel parameters using RODEX2 (Reference 9). RODEX2 is used to determine the initial stored energy for both the blowdown analysis (RELAX system and hot channel) and the heatup analysis (HUXY). The RODEX2 code was approved by the NRC in the early 1980s. At that time, thermal conductivity degradation with burnup was not well characterized by irradiation or post-irradiation testing. As a result, fuel codes at that time did not account for thermal conductivity degradation (TCD). The newer RODEX4 code (Reference 35) explicitly incorporates the impact of TCD with exposure. RODEX4 calculations were performed with and without the models which account for TCD. The differences in the RODEX4 results were used to increase the stored energy calculated by RODEX2 prior to their input to HUXY.

The HUXY code (Reference 10) is used to perform heatup calculations for the entire LOCA transient and provides PCT and local clad oxidation at the axial plane of interest. The heat generated by metal-water reaction (MWR) is included in the HUXY analysis. HUXY is used to calculate the thermal response of each fuel rod in one axial plane of the hot channel assembly. These calculations consider thermal-mechanical interactions within the fuel rod. The clad swelling and rupture models from NUREG-0630 (Reference 11) are incorporated into HUXY. The HUXY code complies with the 10 CFR 50 Appendix K criteria for LOCA Evaluation Models.

[

] are used in the HUXY analysis.

Calculations assumed an initial core power of 102% of 3952 MWt as per NRC requirements. 3952 MWt corresponds to 120% of the original licensed thermal power (OLTP) and is referred to as extended power uprate (EPU). The initial flow conditions supports EPU power operation on the low end of the MELLLA domain (99% of nominal core flow) to the high end of the ICF domain (105% of nominal core flow). Initial flow and void distributions as well as the reactor power distributions (Figures 4.7 and 4.8 of Reference 37) are determined with the XCOBRA (Reference 12) steady state computer code.

The LOCA analysis was performed using plant parameters and plant geometry presented specifically for BFN in Reference 13. The plant parameters specified are based on EPU operation and the plant geometry includes any modifications necessary for EPU at BFN.

Based on the results of the break spectrum analysis (Reference 37), the limiting break characteristics for AREVA fuel are as follows.

Limiting LOCA
Break Characteristics

Location	recirculation discharge pipe
Type / size	split / 0.23 ft^2
Single failure	battery (DC) power, board A
Axial power shape	top-peaked
Initial State	102% power / **[ ]**

For the limiting break, ADS and LPCS are the only emergency core cooling systems available as noted in Table 5.1 of Reference 37.

The fuel performance results are provided as follows.

Parameter	ATRIUM 10XM	ATRIUM-10
Exposure (GWd/MTU)	0.0	0.0
Peak cladding temperature (°F)	2008	2086
Local cladding oxidation (max %)	1.90	2.51
Total hydrogen generated (% of total hydrogen possible)	<1.0	<1.0

The fresh reload fuel in an EPU core is ATRIUM 10XM fuel. The evaluation also supports previously exposed ATRIUM-10 assemblies that may be included in a transition cycle with co-resident ATRIUM 10XM fuel type operating at EPU conditions.

The SLO evaluation continues to support a SLO MAPHLGR multiplier for 0.85 to bound the results of the TLO evaluation as demonstrated in Section 8.0 of Reference 37. The SLO MAPHLGR multiplier is not impacted by EPU.

Non-recirculation line breaks (e.g. ECCS line breaks) were demonstrated to be well below the 10 CFR 50.46 PCT limit of 2200 (°F) (Section 5.3 of Reference 37).

The effect of EPU on the calculated PCT is acceptable as long as the impact of the results on the Licensing Basis PCT remains below the 10 CFR 50.46 limits. The current TS values for ECCS initiation are bounded by the analysis; no changes to these values were required for EPU. Plant-specific analyses demonstrate that there is sufficient ADS capacity, with six ADS valves in service and none out of service, at EPU conditions, to remain below these limits.

2.8.5.6.2.5.3 Local Cladding Oxidation (ECCS-LOCA)

The effect of EPU on the calculated local cladding oxidation is acceptable as long as the impact of the results on the Licensing Basis local cladding oxidation remains below the 10 CFR 50.46 limits. The EPU evaluation has shown that the local cladding oxidation is sufficiently below the 10 CFR 50.46 requirement.

2.8.5.6.2.5.4 Core-Wide Metal-Water Reaction (ECCS-LOCA)

The effect of EPU on the calculated Core-Wide Metal-Water Reaction (CMWR) is acceptable as long as the impact of the results on the Licensing Basis CMWR remains below the 10 CFR 50.46 limits. The EPU evaluation has shown that the CMWR satisfies the 10 CFR 50.46 requirement.

2.8.5.6.2.5.5 Coolable Geometry (ECCS-LOCA)

EPU has no effect on the coolable geometry (ECCS-LOCA). Conformance with coolable geometry requirements is demonstrated by conformance with the 2200(°F) Licensing Basis PCT limit, local cladding oxidation limit of 17%, and total hydrogen generation limit of 1% of the total; therefore, the 10 CFR 50.46 requirement is met.

2.8.5.6.2.5.6 Long-Term Cooling (ECCS-LOCA)

Long-term coolability addresses the issue of reflooding the core and maintaining a water level adequate to cool the core and remove decay heat for an extended time period following a LOCA. For non-recirculation line breaks, the core can be reflooded to the top of the active fuel and be adequately cooled indefinitely. For recirculation line breaks, the core will initially remain covered following reflood due to the static head provided by the water filling the jet pumps to a level of approximately two-thirds core height. Eventually, the heat flux in the core will not be adequate to maintain a two-phase water level over the entire length of the core. Beyond this time, the upper third of the core will remain wetted and adequately cooled by core spray. Maintaining water level at two-thirds core height with one core spray system operating is sufficient to maintain long-term coolability as demonstrated by the NSSS vendor (Reference 43).

Conclusion

BFN has evaluated the LOCA events and the ECCS. The evaluation concludes that operation of the plant at the proposed power level is acceptable. In addition, BFN 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 the plant will continue to meet the requirements of the current licensing basis, 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.
- each BWR have equipment to trip the reactor coolant recirculation pumps automatically under conditions indicative of an ATWS.

Review NRC guidance is provided in Matrix 8 of RS-001.

Browns Ferry Current Licensing Basis

The analysis of anticipated transients without scram is described in BFN UFSAR Section 7.19, “Anticipated Transient without Scram.”

Technical Evaluation

The anticipated transient without scram (ATWS) overpressure evaluation includes consideration of the most limiting RPV overpressure case. In addition, the criteria for peak vessel bottom pressure less than ASME Service Level C Limit of 1500 psig was evaluated.

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. For the IORV event, the reactor vessel is not pressurized from reactor isolation; therefore, this event is non-limiting. For the LOOP event, the fast opening of the bypass valves will reduce the pressure wave created by the reactor isolation; therefore, this event is non-limiting. The only two cases that need to be further analyzed include

(1) Main Steam Isolation Valve Closure (MSIVC) and (2) PRFO. These events have been analyzed and the results are presented below in Sections 2.8.5.7.1 through 2.8.5.7.3.

The EPU ATWS analysis is performed with the NRC-approved methods described in Reference 27. The key inputs to the ATWS analysis are provided in Table 2.8-11. The results of the analysis are provided in Table 2.8-12.

The ATWS mitigation requirements defined in 10 CFR 50.62 are shown to be in compliance in PUSAR (Attachment 6 of the EPU LAR) Section 2.8.5.7.

The results of the ATWS analysis meet the above ATWS acceptance criteria. Therefore, the BFN response to an ATWS event at EPU is acceptable. The potential for thermal-hydraulic instability in conjunction with ATWS events is evaluated in Section 2.8.3.2.

2.8.5.7.1 ATWS (Overpressure)

The overpressure evaluation includes a review of the results of the analyses of ATWS events to identify the most limiting RPV overpressure conditions. Two events, MSIVC and PRFO, were further analyzed for BFN. The higher steam flow will result in higher peak vessel pressures.

The key inputs to the BFN ATWS overpressurization analysis are provided in Table 2.8-11. The results of the MSIVC and PRFO ATWS events are provided in Figure 2.8-16 through Figure 2.8-23.

The limiting ATWS event with respect to RPV overpressure for BFN is PRFO. The PRFO event, prior to SLCS initiation, produces the highest peak lower plenum pressure (1469 psia). The peak pressure value includes adjustments to address the NRC concerns associated with the void-quality correlation, exposure-dependent thermal conductivity, and Doppler effects. The results show that the ATWS overpressurization criteria are met for EPU conditions with one lowest setpoint MSRV out of service.

2.8.5.7.2 ATWS (Suppression Pool Temperature)

Fuel design differences may impact the power and pressure excursion experienced during the ATWS event. This in turn may impact the amount of steam discharged to the suppression pool and containment. **[**

]

[The void coefficient determines what power level the reactor will decrease following the recirculation pump trip. The more negative the void coefficient, the greater decrease in power. The boron worth determines how fast power will decrease after the SLCS is initiated.

A comparison of ATRIUM 10XM to GE14 results for the void and boron worth coefficients is provided in Table 2.8-15. This evaluation assumed the same burnup distributions so only the fuel designs are different. The results show that the reactivity coefficients of these fuel types are similar. For both BOC and EOC exposures the GE14 fuel has a slightly more negative void coefficient and the ATRIUM 10XM fuel has a slightly higher boron worth.

In order to investigate the impact of the void coefficient, the ATWS event was extended to cover 240 seconds after event initiation. This extends the event to the time when boron begins to enter the core. No operator actions were assumed during AREVA's analysis of this event. Not modeling the water level reduction (per Emergency Operating Instructions) keeps the power level artificially high which will maximize the difference in steam discharged to the containment. This simulation allows for a direct comparison between fuel types of steam mass discharged to containment.

Three cores were simulated: a full EPU core of ATRIUM 10XM, a full EPU core of ATRIUM-10 and an EPU core containing one reload of ATRIUM-10 fuel with the remaining fuel being GE14. The results of these analyses are shown in Table 2.8-16. A comparison of results shows that the ATRIUM-10 core discharges the lowest amount of steam to containment, discharging between 1.5% and 2.5% less steam than the ATRIUM 10XM core. Comparing the ATRIUM 10XM and the ATRIUM-10/GE14 core, one can see that the steam discharge at BOC is nearly identical. At EOC, the ATRIUM 10XM core discharges approximately 0.7% more steam than the ATRIUM-10/GE14.

The impact of this higher steam discharge to the suppression pool early in the event would result in an increase of less than 1 degree F in the suppression pool temperature prior to SLC initiation and prior to initiation of RHR in suppression pool cooling mode. This temperature increase would be ameliorated by 1) higher suppression pool heat rejection capability of the RHR system when it is placed in service (a higher pool temperature results in higher suppression pool heat removal rate by the RHR heat exchangers) and 2) the ATRIUM 10XM core reaching hot shutdown conditions earlier due to the higher boron worth. In conclusion,]

[

]

2.8.5.7.3 ATWS (Peak Cladding Temperature)

For ATWS events, the acceptance criteria for PCT and local cladding oxidation for ECCS, defined in 10 CFR 50.46, are adopted to ensure an ATWS event does not impede core cooling. Coolable core geometry is assured by meeting the 2200°F PCT and the 17% local cladding oxidation acceptance criteria stated in 10 CFR 50.46.

There is no core uncovery associated with the ATWS event, hence the PCT and local cladding oxidation results will be bounded by LOCA (Section 2.8.5.6.2). Therefore, the PCT and local cladding oxidation for the BFN ATWS events is qualitatively evaluated to demonstrate compliance with the acceptance criteria of 10 CFR 50.46.

Conclusion

The analysis of the ATWS event has been reviewed to ensure it has adequately accounted for the effects of the use of ATRIUM 10XM during the proposed EPU. The analysis demonstrates that ARI, SLCS, and RPT systems have been installed and that they will continue to meet the requirements of 10 CFR 50.62 and the analysis acceptance criteria following implementation of the proposed EPU with ATRIUM 10XM. Therefore the proposed EPU with ATRIUM 10XM 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 BFN principal design criteria predate these criteria. The BFN 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 General Design Criteria published in 1967, the Tennessee Valley Authority (TVA), performed a comparative evaluation of the design basis of BFN with the AEC proposed General Design Criteria of 1967. The BFN UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria sent 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 BFN UFSAR where there is subject matter relating to the intent of that particular criteria.

While BFN 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 BFN comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as “draft GDC”) is contained in BFN UFSAR Appendix A: draft GDC-66.

New Fuel Storage is described in BFN 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 BFN License Renewal Safety Evaluation Report (SER), NUREG-1843, dated April 2006 (Reference 7). 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

Nuclear reactor plants include facilities for the storage of new fuel. BFN has a new fuel storage facility (also referred to as the new fuel storage vault) for each unit. However, at BFN this facility is not used and the new fuel is placed directly into the spent fuel storage pool following receipt inspection (BFN UFSAR Section 10.2.5). 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 sub-critical 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 BFN principal design criteria predate these criteria. The BFN 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 General Design Criteria published in 1967, the Tennessee Valley Authority (TVA), performed a comparative evaluation of the design basis of BFN with the AEC proposed General Design Criteria of 1967. The BFN UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria sent 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 BFN UFSAR where there is subject matter relating to the intent of that particular criteria.

While BFN 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 BFN comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as “draft GDC”) is contained in BFN UFSAR Appendix A: draft GDCs-40 and 66.

Spent Fuel Storage is described in BFN 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 BFN License Renewal Safety Evaluation Report (SER), NUREG-1843, dated April 2006 (Reference 7). 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.

<u>Technical Evaluation</u>

ATRIUM 10XM is the fuel design that will be loaded for EPU operation at BFN. A small number of exposed ATRIUM-10 assemblies may also be co-resident in the initial EPU operating cycles. The spent fuel storage pool criticality safety analyses for these fuel designs are provided in ANP-3160P (Reference 44) and ANP-2945P (Reference 45), respectively.

ATRIUM 10XM fuel loaded into the BFN reactors must meet the criticality storage requirements provided in Table 2.1 of ANP-3160P. The primary constraints are a series of enrichment and Gadolinia loading requirements for the enriched lattices within the assembly. If any of these primary constraints cannot be met then a secondary reactivity constraint must be met. Specifically, all enriched lattices with an assembly must not exceed a maximum in-rack k-infinity of **[ ]** using the CASMO-4 lattice physics code (Reference 18) or it cannot be stored in the BFN spent fuel storage pools**.** Cycle specific confirmation that these SFSP criticality storage constraints have been met is completed during the bundle and core design phase. The following table provides a summary of the compliance for the ATRIUM 10XM bundles in the EPU equilibrium cycle reference core (Reference 2, Attachment 16 of the EPU LAR).

[

]

The ATRIUM 10XM criticality safety analysis (CSA) documented in ANP-3160P is compliant with the NRC interim staff guidance document for SFSP criticality safety analyses, DSS-ISG-2010-01 Revision 0 (Reference 46). This compliance is documented in Table 3.1 of the Reference 44 report. The ATRIUM-10 Reference 45 report also meets the intent of this staff guidance document. As part of meeting the requirements of DSS-ISG-2010-01, a number of sensitivities were performed and are documented in Section 6 of both CSA reports. The sensitivity studies performed for the ATRIUM 10XM CSA and the ATRIUM-10 CSA are used in the evaluation below to disposition the continued application of the current CSA for operation at extended power uprate conditions.

Impact of Change in Power Density

Operation of the core at a higher power level represents an increase in power density during depletion when compared to CLTP conditions. The impact of power density used for fuel depletion is addressed for the ATRIUM 10XM fuel design in ANP-3160P (Assumption 4 of Section 6.5, Reference 44). A similar evaluation was also performed for the ATRIUM-10 design in Reference 45. Table 6.5 of both referenced reports documents the impact of ±50% changes in the depletion power density which bounds the change to EPU conditions of < 15% of current rated power. These evaluations show that increasing power density results in a very small reduction in the in-rack k-infinity. The following table summarizes this impact assuming a 40% void depletion history.

[

]

Based upon the above comparison, the EPU impact of a change in the power level on the fuel depletion is bound by the value assumed in the original SFSP criticality safety analyses for both the ATRIUM 10XM and ATRIUM-10 fuel designs.

Impact of Changes in Void Fraction

The peak power seen by the fuel in an EPU core remains constrained by thermal limits such that the primary impact is on radial power flattening across the core (i.e., more bundles operating near the limits). This radial power flattening corresponds to an increase in the average void fraction that will be seen by the fuel during EPU operation.

The impact of the void history used for fuel depletion was addressed by the original Reference 44 and 45 CSAs. In these analyses, a sensitivity analysis was performed to determine the impact of void history on the calculated peak in-rack reactivity for the reference bounding lattices for each fuel design. The CSA was then performed based upon the limiting void history condition (i.e., higher in-rack k-infinity). This is illustrated in Figure 6.5 in each of the corresponding CSAs. Figure 2.8-24 provides a summary of this sensitivity for the ATRIUM 10XM and ATRIUM-10 fuel designs, respectively.

Impact of Changes in Controlled Depletion

The control rod density in operation is a function of the amount of hot excess reactivity throughout the operating cycle. Higher hot excess reactivity requires more control rod density to compensate and lower hot excess reactivity requires less control rod density. As discussed previously in Section 2.8.2, “Nuclear Design”, extended power uprate conditions have a potential tendency to reduce the hot excess reactivity, however; the observed change is within normal cycle to cycle variations seen for current rated conditions.

Table 6.6 of both ANP-3160P and ANP-2945P demonstrate that both the ATRIUM 10XM and ATRIUM-10 fuel designs are most limiting when depleted without a control blade present (i.e., uncontrolled depletion). This is the standard depletion condition used in both criticality evaluations. The following table provides a summary of the impact of controlled depletion assuming 100% power density and a 40% void history. Other power density and void combinations show similar behavior.

[

]

In summary, the existing criticality safety analyses are based upon uncontrolled depletion which is the most reactive condition and EPU does not significantly impact expected rod density during operation. Therefore, the current analyses documented in ANP-3160P and ANP-2945P will continue to remain applicable for assemblies that operate at EPU conditions in regard to control depletion history.

Impact on Fuel Temperature

The peak fuel temperature is not expected to change significantly in EPU operation since this is largely constrained by the steady-state LHGR limits. However, due to radial power flattening a larger fraction of the core will be operating at a higher power level potentially increasing the average fuel temperature.

Tables 6.4 of both ANP-3160P (Reference 44) and ANP-2945P (Reference 45) demonstrate that the in-rack k-infinity is insensitive to changes in the fuel temperature assumed in the depletion calculations. The following table provides a summary of these results for the ATRIUM 10XM and ATRIUM-10 reference bounding lattices assuming a 40% void history. Other void histories show similar behavior.

Impact of Depletion Fuel Temperature at 40% VH

Fuel Design	Lattice Type	-100 °F	Nominal Temperature	+100 °F	Comments
ATRIUM 10XM (Reference Bounding)	Top	0.8795	0.8795	0.8795	Table 6.4, Ref. 44
	Bottom	0.8781	0.8781	0.8782	
ATRIUM-10 (Reference Bounding)	Top	0.8812	0.8813	0.8813	Table 6.4, Ref. 45
	Bottom	0.8801	0.8801	0.8802	

In summary, the table above shows insignificant impact on the in-rack k-infinity for sensitivity calculations of ±100 °F in the assumed fuel temperature. Therefore, the analyses documented in ANP-3160P and ANP-2945P will continue to remain applicable for assemblies that operate at EPU conditions in regard to the assumed fuel depletion temperature.

Impact on Axial Power Shape

The criticality evaluations in ANP-3160P and ANP-2945P make no assumptions about the depletion axial power shape. Instead each fuel lattice is evaluated at the exposure and void history condition that produces its lifetime maximum reactivity. While extended power uprate operation is not expected to change the axial power shape beyond the variations normally seen during operation, the existing treatment bounds any perturbation that might result due to operation in the EPU operating domain.

Impact of BLEU

The impact of using BLEU was explicitly addressed in Section 6.4 of ANP-3160P and Section 6.5 of ANP-2945P. The CSAs in both evaluations are based upon the use of commercial grade uranium, that is the allowable in-rack reactivity's and corresponding enrichment and Gadolinia loading requirements were established without crediting the impact of the U236 contained in BLEU fuel. The U236 content of BLEU fuel acts as a neutron absorber and reduces the lattice reactivity when compared to an equivalent lattice composed of commercial grade uranium. This is illustrated in Figure 2.8-25 which compares in-rack reactivity for the ATRIUM 10XM reference bounding lattices with and without BLEU fuel. Therefore, the use of commercial grade uranium in the SFSP criticality safety analyses is conservative and BLEU fuel meeting the storage criteria may be stored in the BFN SFSP fuel storage racks. EPU operation does not change this conclusion.

Neutron Efficacy and Potential Degradation

The BFN spent fuel pool contains high density storage racks utilizing Boral™ plates sandwiched between inner and outer surfaces of stainless steel. The Boral core is made of a central segment composed of a dispersion of boron carbide in aluminum. The central core is clad on both sides with aluminum. The stainless steel container tubes are closure welded with vent holes to prevent the buildup of hydrogen gas. The Boral is modeled using the design minimum Boron-10 areal density of 0.013 g/cm^2. No attempt is made to credit as-built Boron-10 content. This applies to all of the storage racks in each of the spent fuel pools for BFN.

In a water environment, neutron scattering ensures that neutrons approach the Boral from a full range of incident angles. This minimizes the potential for neutron streaming and reduces the significance of self-shielding.

Boral material has not demonstrated a significant degradation potential similar to that seen with racks containing other materials such as Boraflex™. However, under certain conditions, corrosion gases can be trapped within a Boral plate and the aluminum cladding can be deformed to create blisters on the surface of the plate. These blister regions exclude water and can therefore affect the neutron moderation of the Boral storage rack. The Reference 44 ATRIUM 10XM criticality safety analysis applies a uniform void region as a conservative model for this potential blistering condition.

Impact of Spent Fuel Pool Temperature

Operation at extended power uprate conditions will increase the decay heat of the fuel discharged to the spent fuel pool due to the higher power operation prior to shutdown. This increase in power generated per bundle and the larger discharge batch size will tend to increase the heat load of the spent fuel pool with a potential corresponding increase in SFSP water temperature. The ATRIUM 10XM CSA in ANP-3160P documents that calculations were performed at various SFSP water temperatures. These calculations confirmed that the lower temperature bound of **[ ]** was limiting. A similar calculation was also performed for the ATRIUM-10 CSA in ANP-2945P with the same results. Consequently, the CSAs in References 44 and 45 will not be adversely impacted by any potential increase in SFSP water temperature due to EPU operation.

Conclusion

BFN 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 above combined with the evaluation in PUSAR Section 2.5.3.1 provided in Attachment 6 of the EPU LAR 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, BFN concludes that the spent fuel storage facilities 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 spent fuel storage.

Table 2.8-1 Option III Setpoints, EPU ATRIUM 10XM Equilibrium Cycle

OPRM Amplitude Setpoint	OLMCPR(SS)	OLMCPR(2PT)
1.05	1.15	1.11
1.06	1.17	1.12
1.07	1.19	1.14
1.08	1.20	1.16
1.09	1.22	1.18
1.10	1.24	1.20
1.11	1.26	1.22
1.12	1.28	1.24
1.13	1.30	1.26
1.14	1.33	1.28
1.15	1.35	1.30
Acceptance Criteria	**Off-rated OLMCPR at 45% Flow**	**Rated Power OLMCPR**

Table 2.8-2 BSP Region Intercepts, Nominal Feedwater Temperature

Region Boundary Intercept	EPU Power (% rated)	Core Flow (% rated)	Core/Global Decay Ratio	Regional Decay Ratio	Channel Decay Ratio
REGION 1: SCRAM Region					
1A (HFCL)	56.55	40	0.733	0.472	0.137
1B (NCL)	39.34	29	0.654	0.506	0.205
REGION 2: CONTROLLED ENTRY Region					
2A (HFCL)	64.50	50	0.698	0.423	0.089
2B (NCL)	27.54	29	0.526	0.420	0.068

Table 2.8-3 BSP Region Intercepts, Reduced Feedwater Temperature

Region Boundary Intercept	EPU Power (% rated)	Core Flow (% rated)	Core (Global) Decay Ratio	Regional Decay Ratio	Channel Decay Ratio
REGION 1: SCRAM Region					
1A (HFCL)	56.55	40	0.830	0.606	0.154
1B (NCL)*	37.00	29	0.837	0.596	0.180
REGION 2: CONTROLLED ENTRY Region					
2A (HFCL)	64.50	50	0.749	0.478	0.089
2B (NCL)	27.54	29	0.500	0.389	0.064

* The scram region boundary on the natural circulation line for reduced temperature operation (Point 1B of Table 2.8-3 above) has been modified from the base minimal region to meet decay ratio acceptance criteria.

Table 2.8-4 Parameters Used for Transient Analysis

Parameter	CLTP	EPU
Rated Thermal Power (MWt)	3458	3952
Analysis Power (% Rated)	100 / 102 (Note 1)	100 / 102 (Note 1)
Analysis Dome Pressure (psia)	1050 / 1070 (Note 1)	1050 / 1070 (Note 1)
Rated Core Flow (Mlbm/hr)	102.5	102.5
Rated Core Flow Range (% Rated)	81.0 – 105.0	99.0 – 105.0
Normal Feedwater Temperature (°F)	382.2	394.8
Change in Feedwater Temperature (Maximum) (worst single failure of Feedwater heaters) (ΔT °F)	100	100
Number of MSRVs assumed in the analysis	12	12

Notes:

1. Minimum critical power ratio (MCPR) analyses are performed at 100% of rated core power and a dome pressure of 1050 psia. ASME Overpressure analysis is performed at 102% of rated core power and a dome pressure of 1070 psia.

Table 2.8-5 Base Case Transient Results

AOO Event	ΔCPR*	$LHGRFAC_p$
FWCF		
100% Power	0.34	1.00
87.5% Power	0.39	1.00
LRNB		
100% Power	0.31	1.00
87.5% Power	0.32	1.00
TTNB		
100% Power	0.31	1.00
87.5% Power	0.31	1.00
MSIV Closure (All)		
100% Power	0.16	1.00
87.5% Power	0.15	1.00
MSIV Closure (Single)		
100% Power	0.11	1.00
87.5% Power	0.09	1.00
IHPS†		
100% Power	0.13	1.00
87.5% Power	0.16	1.00
Fast Recirculation Runup		
66% P / 52% F	0.13	1.00
RWE		
100% Power	0.27	1.00
85% Power	0.28	1.00
65% Power	0.36	1.00
40% Power	0.58	1.00

* $MCPR_p$ is calculated as the sum of the ΔCPR and SLMCPR.

† High water level (Level 8) trip of the main turbine does not occur.

Table 2.8-6 Transient Analysis MCPR Results

AOO Transient	CLTP (ΔCPR)	EPU (ΔCPR)
Load Rejection with Bypass Failure	0.31	0.31
Turbine Trip with Bypass Failure	0.31	0.31
Feedwater Controller Failure Max Demand	0.37	0.34
Inadvertent HPCI	Note 1	0.13
Loss of Feedwater Heating	0.14	0.13
Rod Withdrawal Error	0.31 (Note 2)	0.27 (Note 2)
Slow Recirculation Increase	$MCPR_f$	$MCPR_f$
Fast Recirculation Increase	Note 1	0.13
MSIV Closure All Valves	Note 1	0.16
MSIV Closure 1 Valve	Note 1	0.11

Notes:

1. Event not analyzed at CLTP as they were previously dispositioned as non-limiting. The EPU evaluation confirms the event is non-limiting
2. Results presented for the unblocked condition.

Table 2.8-7 Equipment Out-Of-Service (EOOS) Transient Results

EOOS	AOO Event	ΔCPR*	$LHGRFAC_p$
Recirculation pump trip out-of-service (RPTOOS)	**FWCF**		
	100% Power	0.32	1.00
	87.5% Power	0.36	1.00
	LRNB		
	100% Power	0.38	1.00
	87.5% Power	0.35	1.00
Feedwater heaters out-of-service (FHOOS)	**FWCF**		
	100% Power	0.37	1.00
	87.5% Power	0.42	1.00
	LRNB		
	100% Power	0.31	1.00
	87.5% Power	0.31	1.00
RPTOOS and FHOOS	**FWCF**		
	100% Power	0.34	1.00
	87.5% Power	0.38	1.00
	LRNB		
	100% Power	0.30	1.00
	87.5% Power	0.28	1.00
Power load unbalance out-of-service (PLUOOS)	**LRNB**		
	100% Power	0.32	1.00
	87.5% Power	0.35	1.00
PLUOOS and RPTOOS	**LRNB**		
	100% Power	0.39	1.00
	87.5% Power	0.37	1.00
PLUOOS and FHOOS	**LRNB**		
	100% Power	0.32	1.00
	87.5% Power	0.33	1.00
PLUOOS and RPTOOS and FHOOS	**LRNB**		
	100% Power	0.31	1.00
	87.5% Power	0.31	1.00
Turbine bypass valves out-of-service (TBVOOS)	**FWCF**		
	100% Power	0.38	1.00
	87.5% Power	0.43	1.00

* $MCPR_p$ is calculated as the sum of the ΔCPR and SLMCPR.

Table 2.8-7 Equipment Out-of-Service (EOOS) Transient Results *(Continued)*

EOOS	AOO Event	ΔCPR*	$LHGRFAC_p$
RPTOOS and TBVOOS	**FWCF**		
	100% Power	0.42	1.00
	87.5% Power	0.43	0.97
TBVOOS and FHOOS	**FWCF**		
	100% Power	0.40	1.00
	87.5% Power	0.45	1.00
RPTOOS and TBVOOS and FHOOS	**FWCF**		
	100% Power	0.38	1.00
	87.5% Power	0.41	1.00

* $MCPR_p$ is calculated as the sum of the ΔCPR and SLMCPR.

Table 2.8-8 RWE ΔCPR versus Setpoint

Analytical RBM Setpoint (%)	ΔCPR	OLMCPR*
1.07	0.23	1.29
1.11	0.25	1.31
1.14	0.27	1.33
1.17	0.27	1.33

Table 2.8-9 BFN Equilibrium Cycle RBM Operability Requirements

Thermal Power (%Rated)	MCPR	Comment
≥ 27 % and < 90%	<1.64	Dual Loop operation
	<1.68	Single loop operation
≥90%	<1.34	Dual Loop operation

* Based on a MCPR Safety Limit of 1.06.

Table 2.8-10 $MCPR_f$ and $LHGRFAC_f$ Results

Core Flow (% of rated)	$MCPR_f$	$LHGRFAC_f$
30.0	1.61	0.63
77.1	--	1.00
78.0	1.28	--
107.0	1.28	1.00

Table 2.8-11 BFN Key Inputs for ATWS Analysis

Input Variable	EPU	87.5% of EPU RTP	CLTP*
Reactor power (MWt)	3952	3458	3458
Reactor dome pressure (psia)	1050	1039.8	1050
Each MSRV capacity at 103% of 1090 psig (lbm/hr)	870,000	870,000	870,000
High pressure ATWS-RPT setpoint (psig)	1177	1177	1177
Number of MSRVs	13	13	13
Number of MSRVs OOS	1	1	1

Table 2.8-12 BFN Results for ATWS Analysis

Acceptance Criteria	EPU	87.5% of EPU RTP	CLTP*
Peak vessel bottom pressure (psig)†	1469	1375	1404

Table 2.8-13 MSIVC ATWS Sequence of Events

Item	Event	Event Time (sec)
1	MSIV Isolation Initiated	0.00
2	MSIVs Fully Closed	4.00
3	Peak Neutron Flux	4.50
4	High Pressure ATWS Setpoint	4.97
5	Opening of the First Relief Valve	5.42
6	Recirculation Pumps Trip	5.47
7	Peak Heat Flux	5.92
8	Peak Vessel Pressure	11.21

* Based on a represented BFE CLTP core consisting of 1/3 ATRIUM 10XM and 2/3 ATRIUM-10.

† The peak pressure results include adjustments to address the NRC concerns associated with the void-quality correlation, exposure-dependent thermal conductivity, and Doppler effects.

Table 2.8-14 PRFO ATWS Sequence of Events

Item	Event	Event Time (sec)
1	TCVs and Bypass Valves Start Open	0.00
2	MSIV Closure Initiated by Low Steam Line Pressure	6.02
3	MSIVs Fully Closed	10.02
4	High Pressure ATWS Setpoint	13.39
5	Opening of the First Relief Valve	13.79
6	Peak Neutron Flux	13.87
7	Recirculation Pumps Trip	13.89
8	Peak Heat Flux	14.29
9	Peak Vessel Pressure	19.70

Table 2.8-15 [ ]

[ ]

Table 2.8-16 [ ]

[ ]

[

]

Figure 2.8-1 Comparison of Hot Excess Reactivity for EPU and CLTP Conditions (nominal EOC N-1)

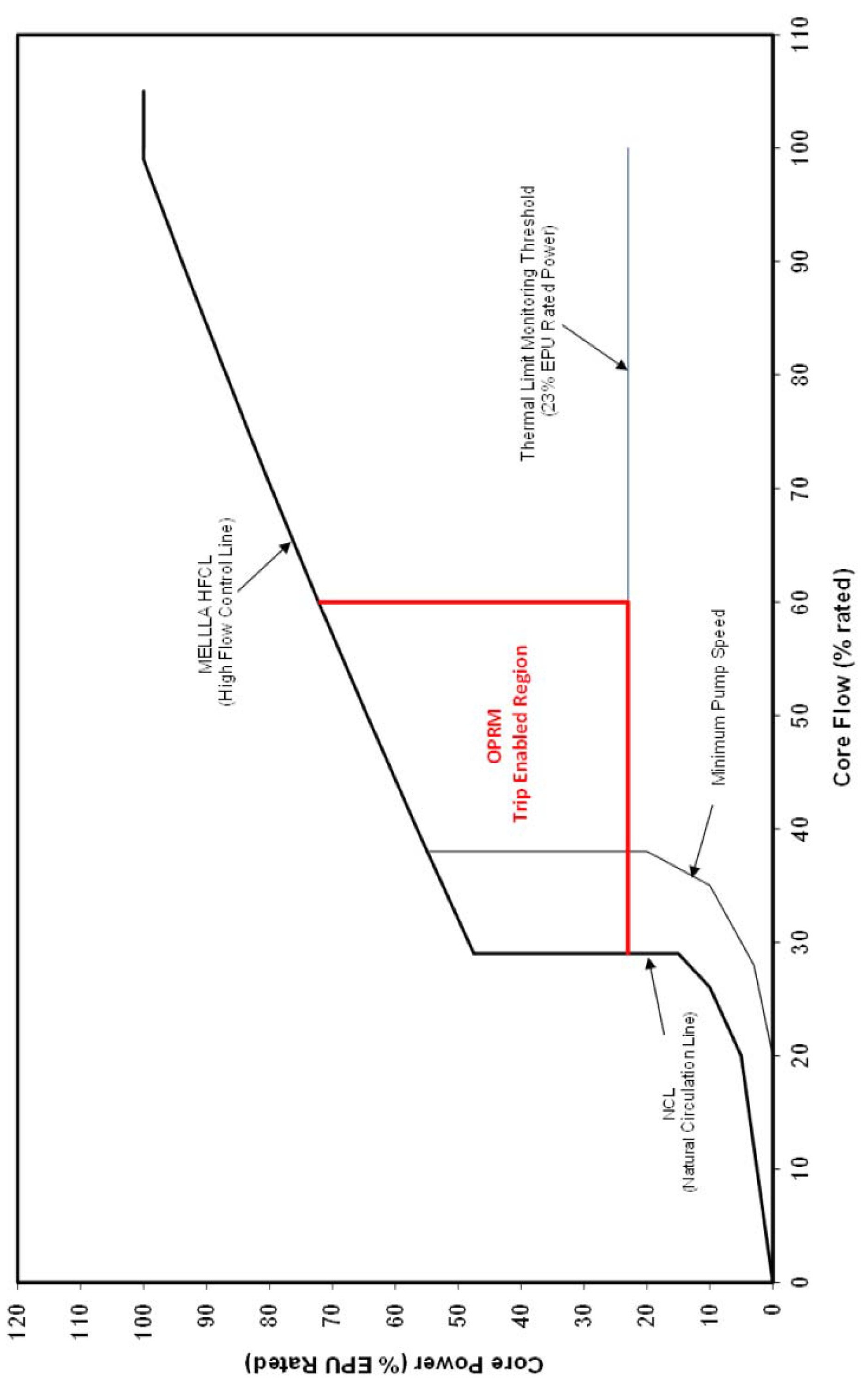

Figure 2.8-2 OPRM (Option III) Trip-Enabled Region

[

]

Figure 2.8-3 Limiting Cycle Specific DIVOM Results for ATRIUM 10XM EPU Reference Cycle

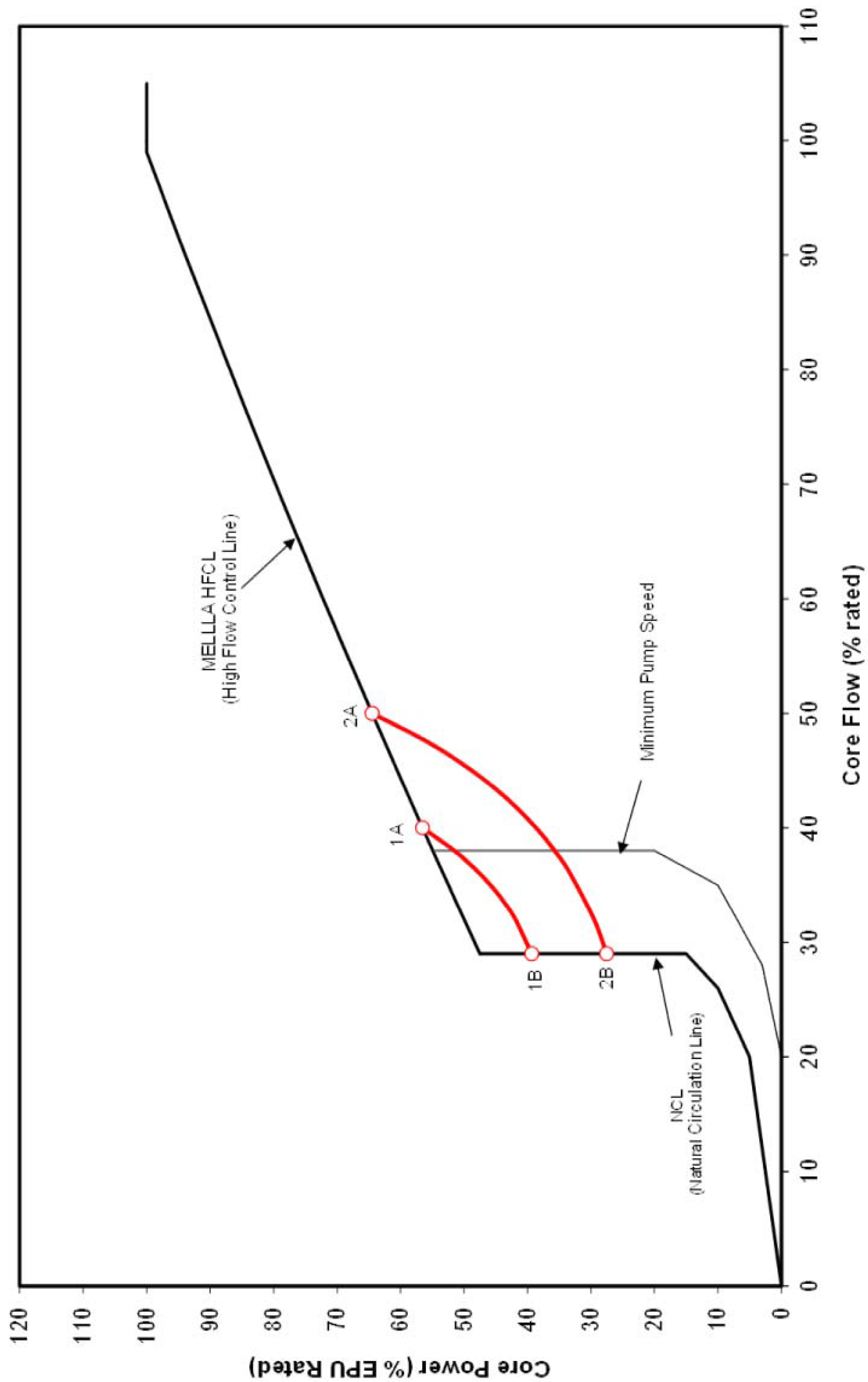

Figure 2.8-4 Backup Stability Regions, Nominal Feedwater Temperature

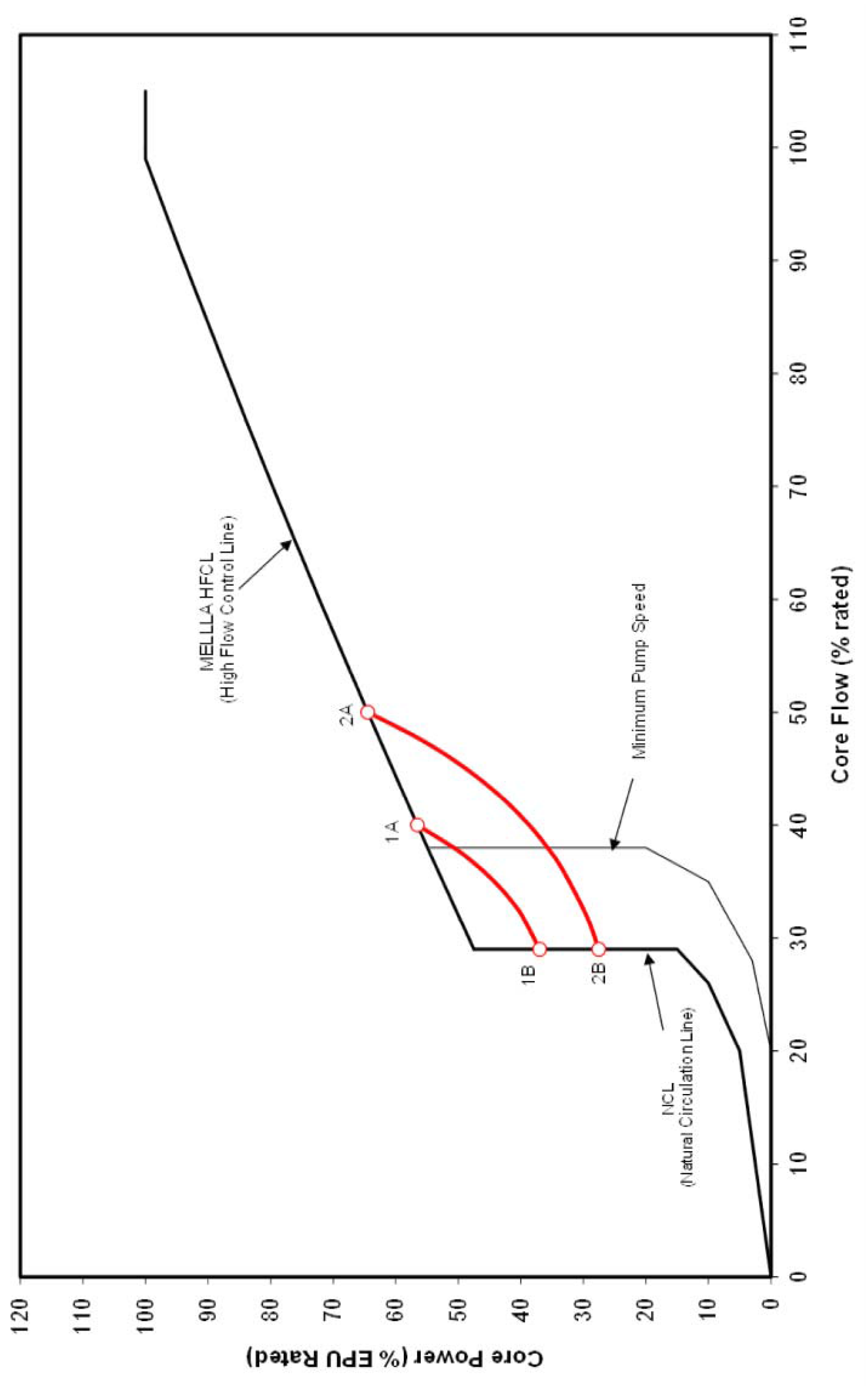

Figure 2.8-5 Backup Stability Regions, Reduced Feedwater Temperature

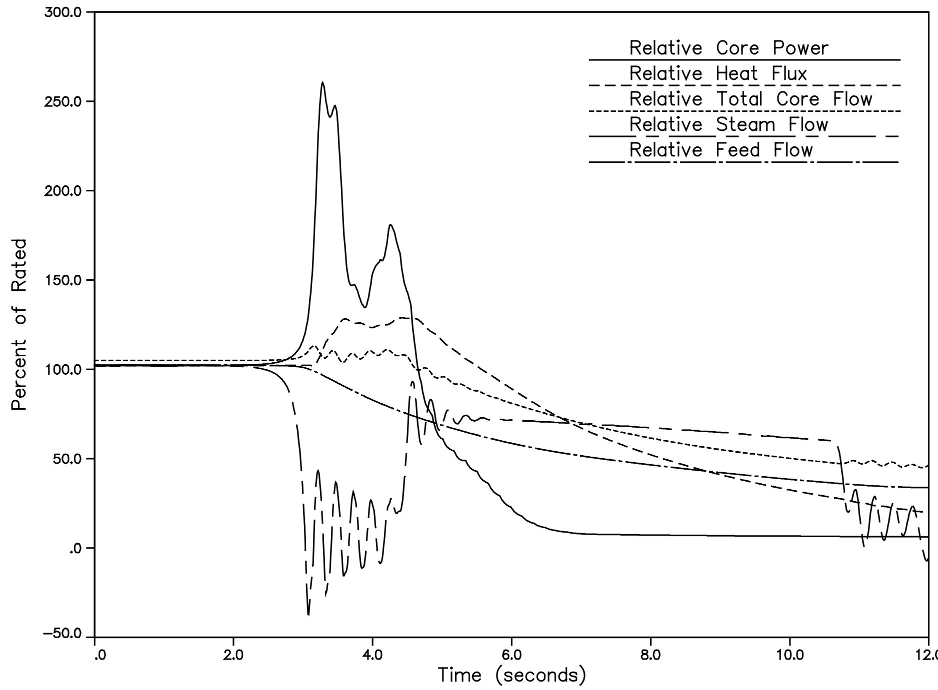

Figure 2.8-6 MSIV Closure Overpressurization Event at 102P/105F – Key Parameters

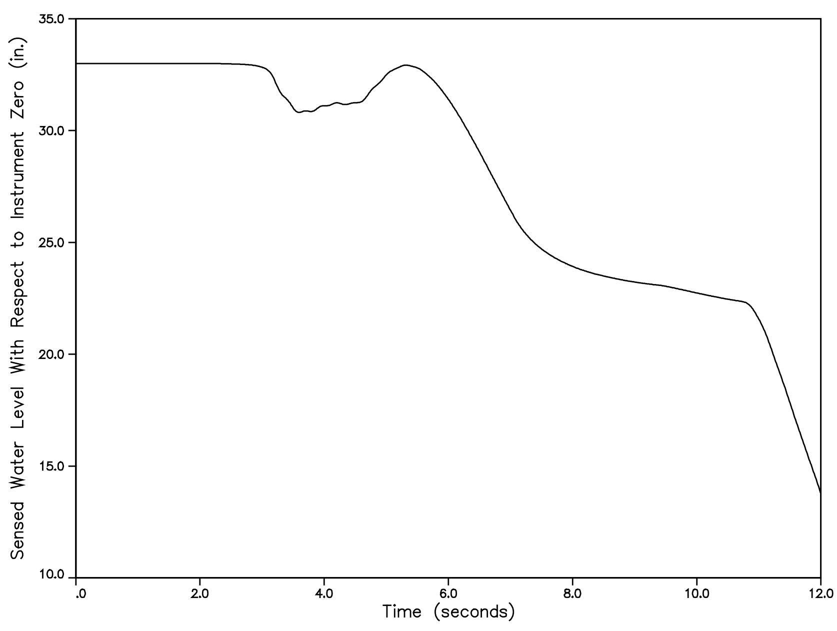

Figure 2.8-7 MSIV Closure Overpressurization Event at 102P/105F – Sensed Water Level

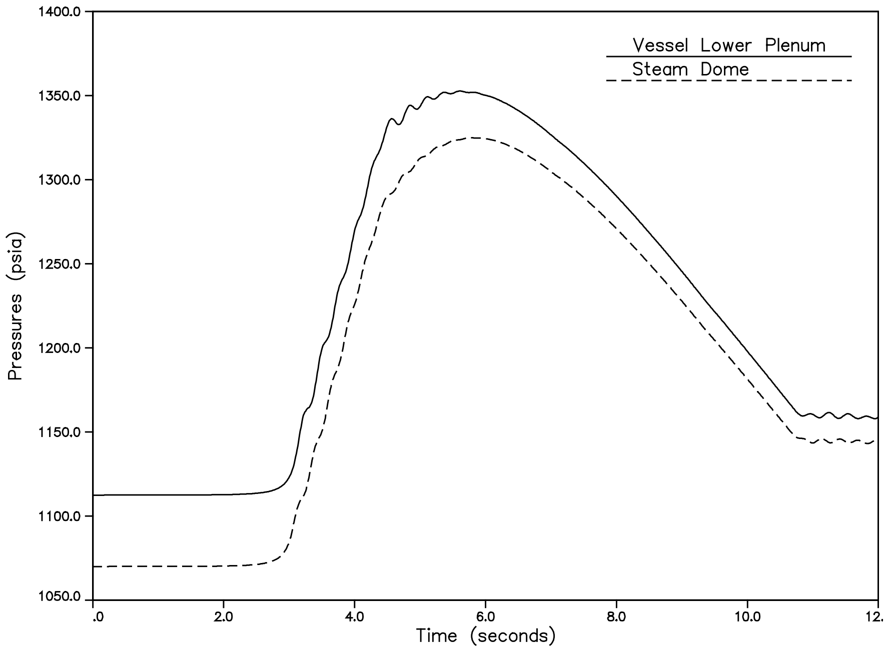

Figure 2.8-8 MSIV Closure Overpressurization Event at 102P/105F – Vessel Pressures*

* The pressures presented in this figure do not include the adjustments associated with NRC concerns with the void-quality correlation, exposure-dependent thermal conductivity, and Doppler effects.

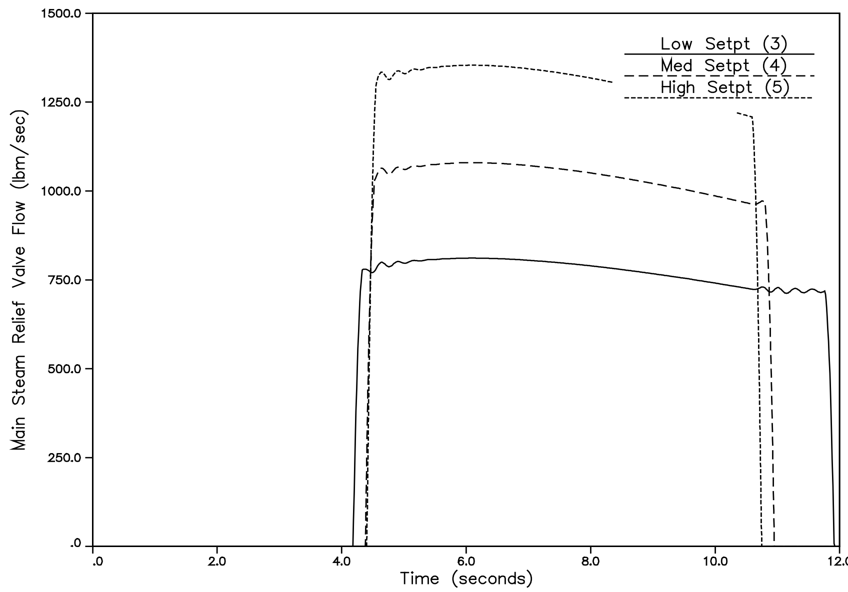

Figure 2.8-9 MSIV Closure Overpressurization Event at 102P/105F – Safety/Relief Valve Flow Rates

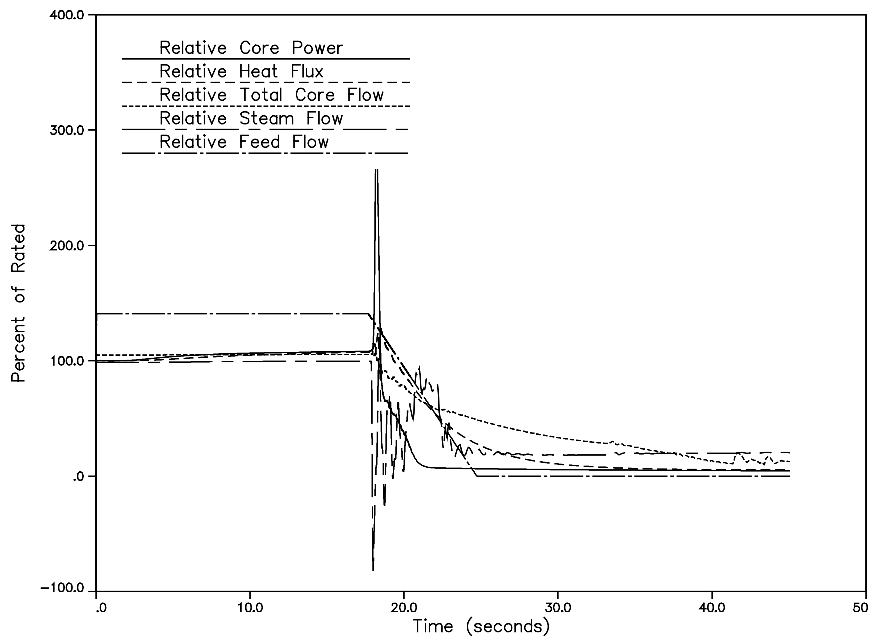

Figure 2.8-10 EOC FWCF at 100P/105F – TSSS Key Parameters

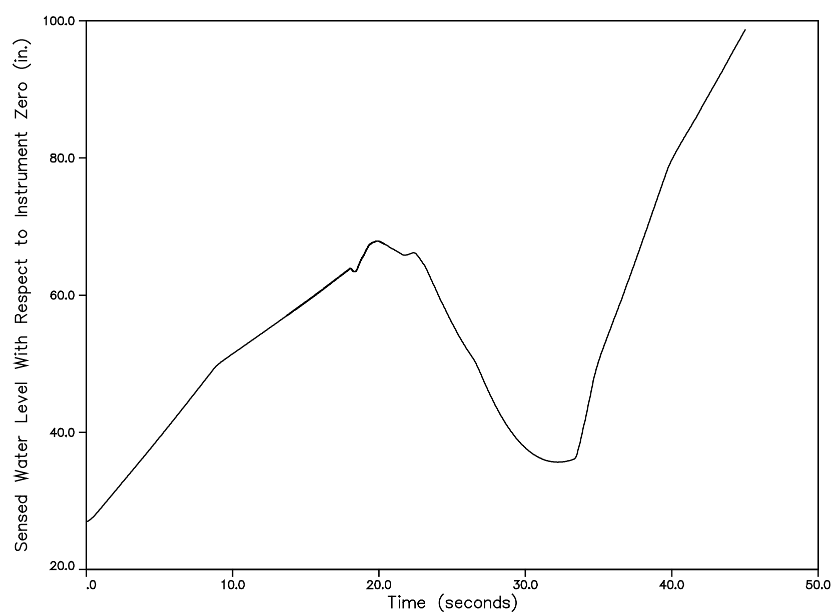

Figure 2.8-11 EOC FWCF at 100P/105F – TSSS Sensed Water Level

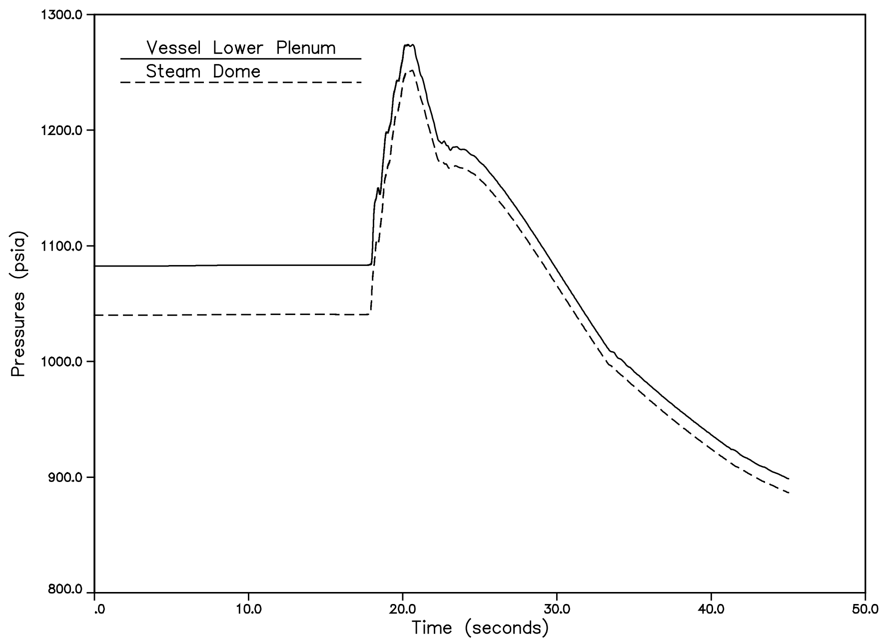

Figure 2.8-12 EOC FWCF at 100P/105F – TSSS Vessel Pressures

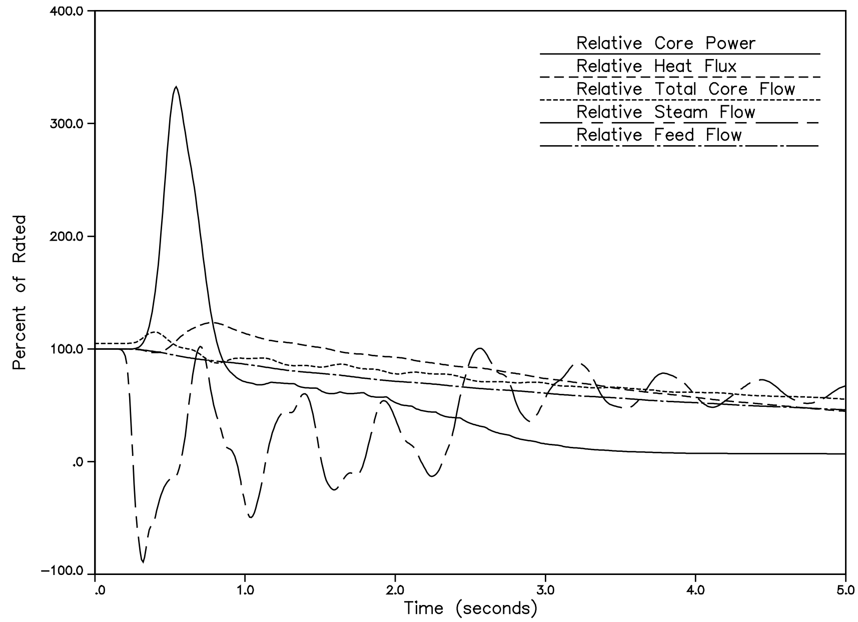

Figure 2.8-13 EOC LRNB at 100P/105F – TSSS Key Parameters

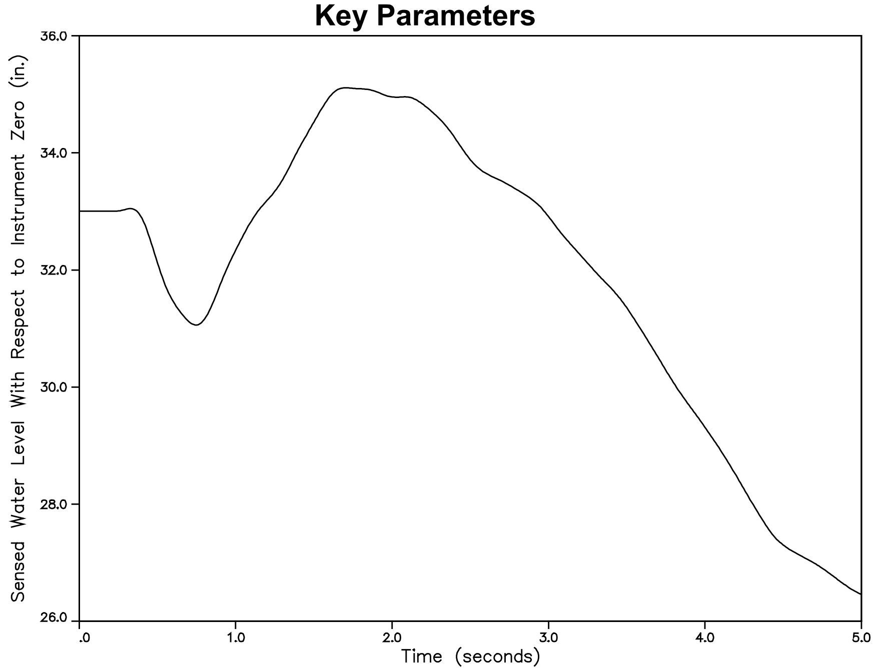

Figure 2.8-14 EOC LRNB at 100P/105F – TSSS Sensed Water Level

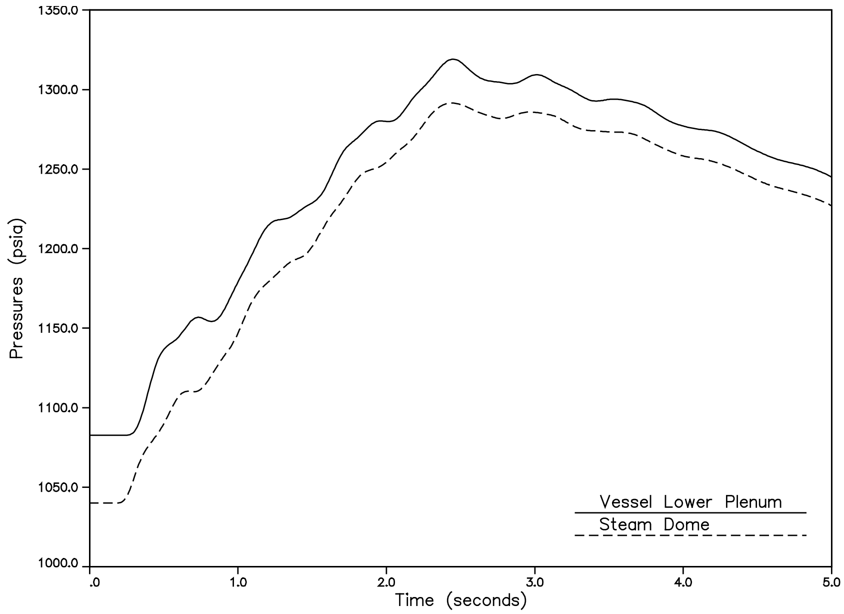

Figure 2.8-15 EOC LRNB at 100P/105F – TSSS Vessel Pressures

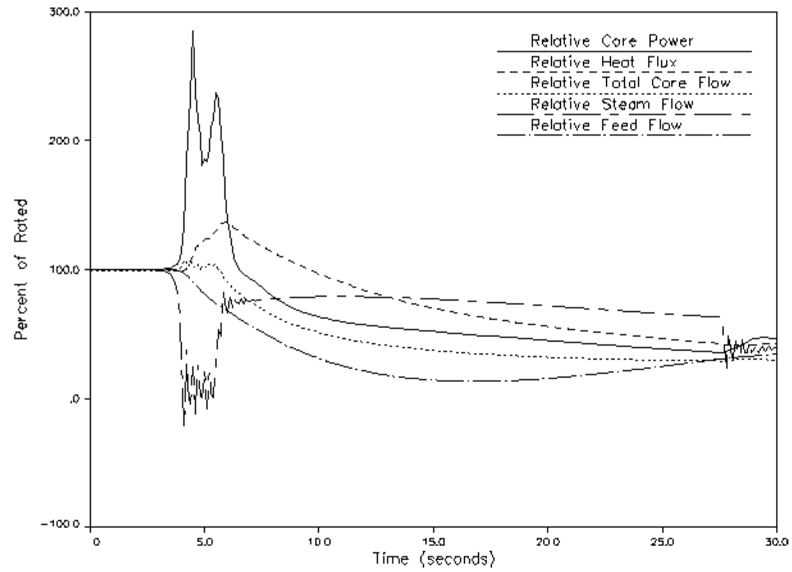

Figure 2.8-16 MSIVC ATWS Overpressurization Event at 100P/99F – Key Parameters

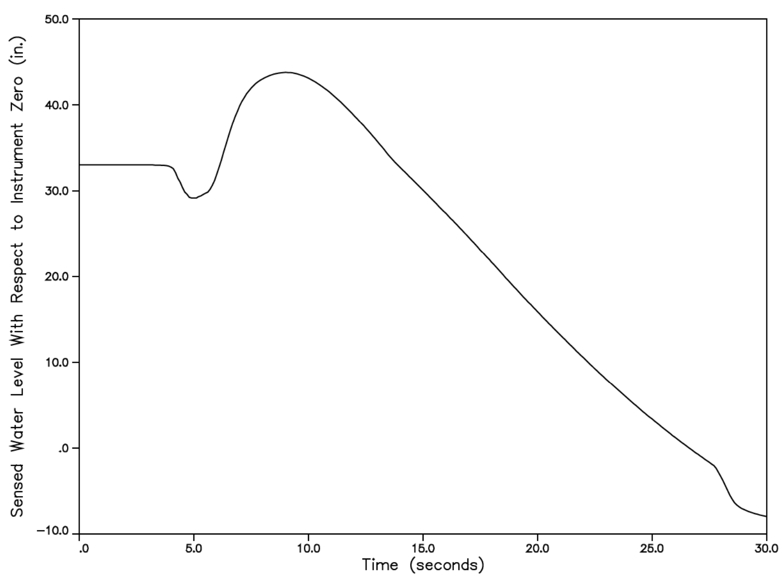

Figure 2.8-17 MSIVC ATWS Overpressurization Event at 100P/99F – Sensed Water Level

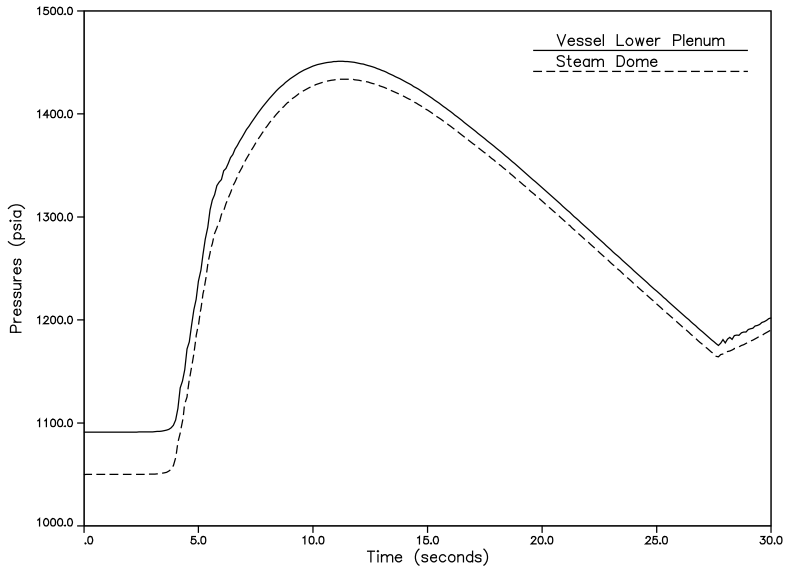

Figure 2.8-18 MSIVC ATWS Overpressurization Event at 100P/99F – Vessel Pressures*

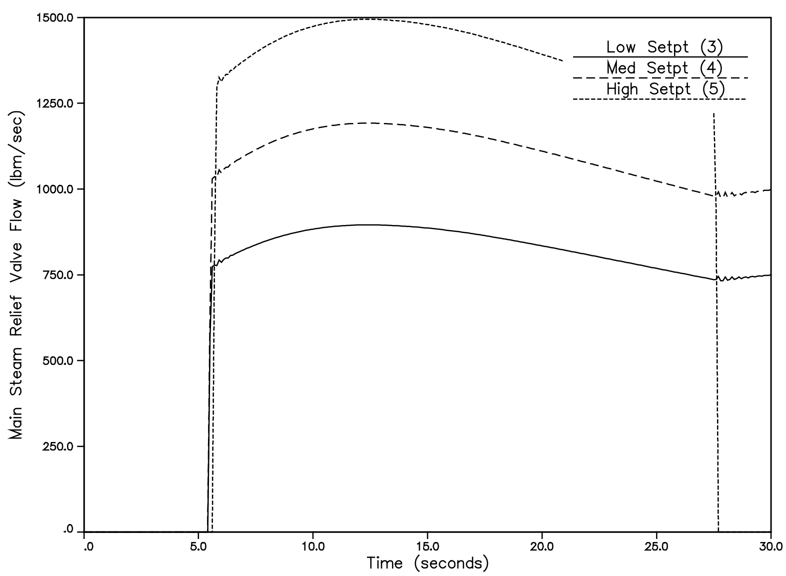

Figure 2.8-19 MSIVC ATWS Overpressurization Event at 100P/99F – Safety/Relief Valve Flow Rates

* The peak pressure results do not include adjustments to address the NRC concerns associated with the void-quality correlation, exposure-dependent thermal conductivity, and Doppler effects.

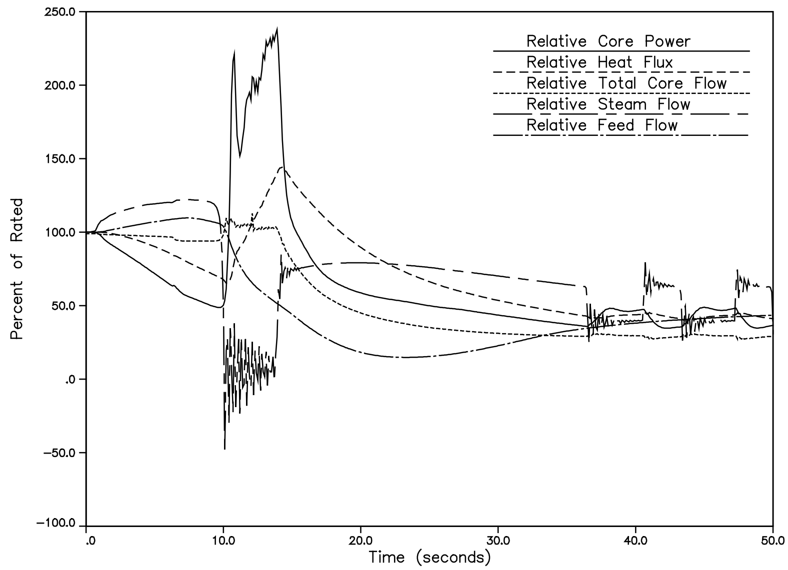

Figure 2.8-20 PRFO ATWS Overpressurization Event at 100P/99F – Key Parameters

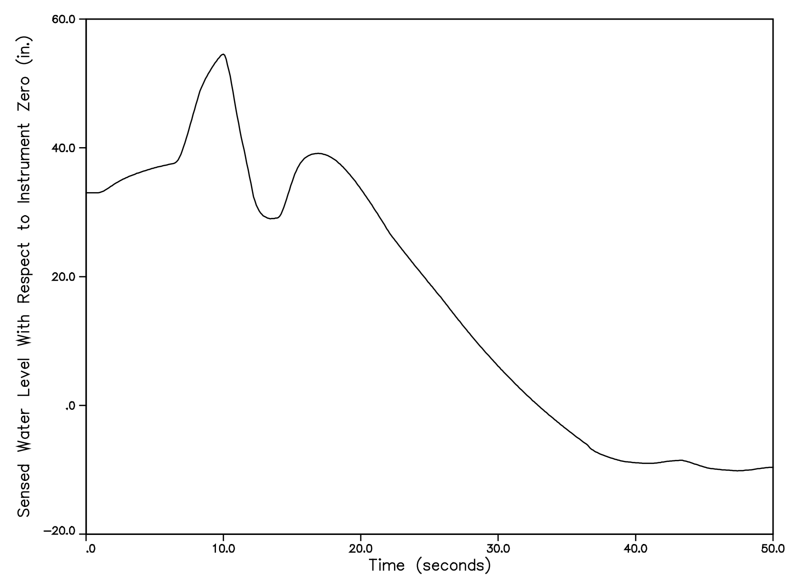

Figure 2.8-21 PRFO ATWS Overpressurization Event at 100P/99F – Sensed Water Level

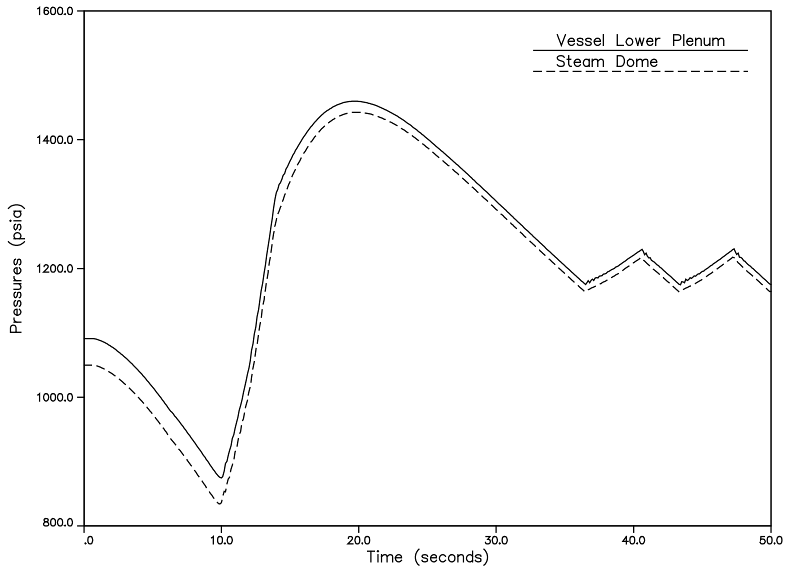

Figure 2.8-22 PRFO ATWS Overpressurization Event at 100P/99F – Vessel Pressures*

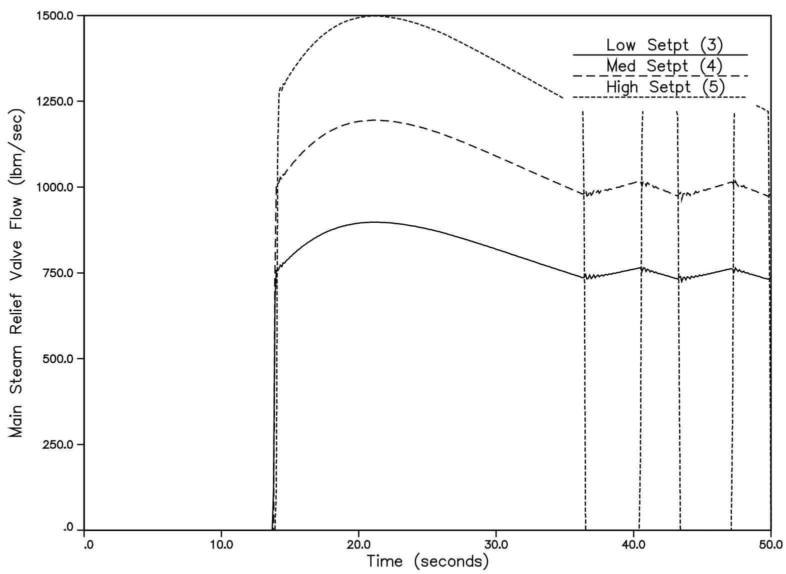

Figure 2.8-23 PRFO ATWS Overpressurization Event at 100P/99F – Safety/Relief Valve Flow Rates

* The peak pressure results do not include adjustments to address the NRC concerns associated with the void-quality correlation, exposure-dependent thermal conductivity, and Doppler effects.

[

]

Figure 2.8-24 Impact on Void History Depletion on In-Rack K-infinity

[

]

Figure 2.8-25 BLEU versus Commercial Grade Uranium Reactivity Comparison

2.9 SOURCE TERMS AND RADIOLOGICAL CONSEQUENCES ANALYSES

2.9.1 Source Terms for Radwaste Systems Analyses

Regulatory Evaluation

The NRC staff reviewed the radioactive source term associated with EPUs to ensure the adequacy of the sources of radioactivity used by the licensee 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 BFN principal design criteria predate these criteria. The BFN 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 General Design Criteria published in 1967, the Tennessee Valley Authority (TVA), performed a comparative evaluation of the design basis of BFN with the AEC proposed General Design Criteria of 1967. The BFN UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria sent 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 BFN UFSAR where there is subject matter relating to the intent of that particular criteria.

While BFN 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 BFN comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in BFN UFSAR Appendix A: draft GDC-70.

The radioactive waste systems are described in BFN 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 BFN License Renewal Safety Evaluation Report (SER), NUREG-1843, dated April 2006 (Reference 7). 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

The following Technical Evaluation addresses only the fuel specific portions of the preceding Regulatory Evaluation and Current Licensing Basis sections.

2.9.1.1 Radiation Sources in the Reactor Core

The radiation sources in the core are directly related to the fission rate during power operation. These sources include radiation from the fission process, accumulated fission products and neutron reactions as a secondary result of fission.

For the EPU at BFN, the radiological evaluation has been performed at EPU conditions following methods and assumptions defined in Regulatory Guide 1.183 (Reference 47).

Because these analyses were performed at a power level of 4031 MWt (102 percent of 3952 MWt), the radiological source terms are applicable up to a rated thermal power of 3952 MWt.

The approved AST (Reference 52) is based upon isotopic source terms that bound the following core loadings:

- GNF provided GE14 fuel containing commercial grade uranium (CGU)
- AREVA provided ATRIUM-10 fuel containing CGU
- AREVA provided ATRIUM-10 fuel containing blended low enriched uranium (BLEU)
- Any combination of the above fuel designs

BFN has since begun loading the AREVA ATRIUM 10XM fuel design which will be the fresh fuel design loaded into the proposed EPU operating cycles. Isotopic source term analyses for this new fuel design were performed at EPU conditions for both CGU and BLEU loadings. TVA confirmed that the dose consequences from the approved AST licensing basis remained bounding for doses calculated using the ATRIUM 10XM source terms (Reference 15).

Therefore, the current AST licensing bases, including the use of the ATRIUM 10XM fuel design, remain valid for the proposed EPU.

Furthermore, the ATRIUM 10XM EPU equilibrium cycle design (Reference 2, Attachment 16 of the EPU LAR) has been reviewed to ensure that BFN cores designed for EPU operation will remain within the source term calculation basis. This review is summarized in the following table.

Key Parameter	Criteria*	Actual Value	Detailed Results
Core Power (MWt)	≤ 3952	3952	Meets Criteria.
Average Enrichment (%)	[ ]	[ ]	Meets Criteria.
Peak Bundle Average Exposure (GWd/MTU)	[ ]	[ ]	Meets Criteria.
AST Core Average Exposure Limit (GWd/MTU)	[ ]	[ ]	Meets Criteria.
Maximum rod average LHGR for rods with burnups exceeding 54 GWd/MTU. (kw/ft)	≤ 6.3	≤ 6.3	Meets Criteria (from footnote 11 of Regulatory Guide 1.183).

The post-operation radiation sources in the core are primarily the result of accumulated fission products. Two separate forms of post-operation source data are normally applied. The first of these is the core gamma-ray source. The total gamma energy source increases in proportion to reactor power, i.e. the increase in gamma energy source is directly proportional to the power level increase.

The second set of post-operation source data consists primarily of nuclide activity inventories for fission products in the fuel. These data are needed for post-accident and SFP evaluations, which are performed in compliance with regulatory guidance that applies different release and transport assumptions to different fission products. The core fission product inventories for these evaluations are based on an assumed fuel irradiation time, which develops "equilibrium" activities in the fuel.

* The criteria provided in the table above are based upon the inputs assumed for the BFN AST (Reference 52), with the exception of the core average exposure limit. The core average exposure limit presented is the value assumed for the ATRIUM 10XM fuel type supporting the Reference 15 License Amendment Request.

The radionuclide inventories of the full core at EOC are calculated in terms of Curies per MWt. The actual core inventory of radionuclides for the AST is a bounding core inventory which has been generated to support the radiological accident dose consequence evaluations. This inventory bounds reloads of GE-14, ATRIUM-10, and ATRIUM 10XM fuel.

Conclusion

The radioactive source term associated with the proposed EPU has been reviewed. The current licensing basis source term is based upon EPU conditions and it has been found to remain bounding for the fuel designs that may be in a BFN EPU core. Therefore, the proposed EPU is acceptable with respect to the source term.

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

BFN UFSAR, Section 13.5, "Startup and Power Test Program," provides an overview of the initial power ascension test program.

Technical Evaluation

The use of ATRIUM 10XM fuel will not impose any additional testing requirements for extended power uprate (EPU) operation.

Conclusion

The EPU test program was reviewed, including plans for the initial approach to the proposed maximum licensed thermal power level, transient testing necessary to demonstrate that plant equipment will perform satisfactorily at the proposed increased maximum licensed thermal power level, and the test program's conformance with applicable regulations. It is concluded that the use of ATRIUM 10XM fuel will not impose any additional testing requirements for EPU operation.

REFERENCE

1. RS-001 Revision 0, *Review Standard for Extended Power Uprates*, US NRC, December 2003.
2. ANP-3342P Revision 1, *Browns Ferry EPU (120% OLTP) Equilibrium Fuel Cycle Design*, AREVA Inc., August 2015.
3. ANP-2637 Revision 6, *Boiling Water Reactor Licensing Methodology Compendium*, October 2014.
4. ANP-2860P Revision 2, *Browns Ferry Unit 1 – Summary of Responses to Request for Additional Information,* AREVA NP, October 2009.
5. Supplement 1P Revision 0 to ANP-2860P Revision 2, *Browns Ferry Unit 1 – Summary of Responses to Request for Additional Information, Extension for ATRIUM 10XM,* AREVA NP, November 2012.
6. Supplement 2P Revision 1 to ANP-2860P Revision 2, *Browns Ferry Unit 1 – Summary of Responses to Request for Additional Information, Extension for Use of ATRIUM 10XM Fuel for Extended Power Uprate,* AREVA Inc., August 2015.
7. BFN License Renewal Safety Evaluation Report (SER), NUREG-1843, dated April 2006.
8. EMF-2361(P)(A) Revision 0, *EXEM BWR-2000 ECCS Evaluation Model,* Framatome ANP, May 2001, as supplemented by the site-specific approval in NRC safety evaluations dated April 27, 2012 (for Unit 1), February 15, 2013 (for Units 2 and 3), and July 31, 2014.
9. XN-NF-81-58(P)(A) Revision 2 and Supplements 1 and 2, *RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model,* Exxon Nuclear Company, March 1984.
10. XN-CC-33(A) Revision 1, *HUXY: A Generalized Multirod Heatup Code with 10 CFR 50 Appendix K Heatup Option Users Manual*, Exxon Nuclear Company, November 1975.
11. XN-NF-82-07(P)(A) Revision 1, *Exxon Nuclear Company ECCS Cladding Swelling and Rupture Model*, Exxon Nuclear Company, November 1982.
12. XN-NF-80-19(P)(A) Volume 3 Revision 2, *Exxon Nuclear Methodology for Boiling Water Reactors, THERMEX: Thermal Limits Methodology Summary Description*, Exxon Nuclear Company, January 1987.
13. ANP-3325P Revision 2, *Browns Ferry Units 1, 2, and 3 LOCA Parameters Document (EPU),* AREVA Inc., January 2015.
14. Letter US NRC (E. Brown) to TVA (J. Shea), *BROWNS FERRY NUCLEAR PLANT, UNIT 1 – ISSUANCE OF AMENDMENTS REGARDING THE TRANSITION TO AREVA FUEL (TAC NO. ME3775) (TS-473)*, dated July 3, 2012 (Accession No. ML12129A149).
15. Letter US NRC to TVA, *Browns Ferry Nuclear Plant, Units 1, 2, and 3- Issuance of Amendment Regarding Technical Specification (TS) Change TS-478 Addition of Analytical Methodologies to TS 5.6.5 and Revision of TS 2.1.1.2 for Unit 2*, July 31, 2014. (ML14113A286)

16. ANF-89-98(P)(A) Revision 1 and Supplement 1, *Generic Mechanical Design Criteria for BWR Fuel Designs*, Advanced Nuclear Fuels, May 1995.

17. ANP-2899P Revision 0, *Fuel Design Evaluation for ATRIUM™ 10XM BWR Reload Fuel*, AREVA NP, April 2010.

18. EMF-2158(P)(A) Revision 0, *Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICROBURN-B2,* Siemens Power Corporation, October 1999.

19. NEDO-32465-A, BWR Owners Group Licensing Topical Report, *Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications*, August 1996.

20. OG02-0119-260, *Backup Stability Protection (BSP) for Inoperable Option III Solution*, July 17, 2002.

21. GE-NE-0000-0028-9714-R1, *Plant-Specific Regional Mode DIVOM Procedure Guideline,* GE Nuclear Energy, June 2005.

22. BAW-10255PA Revision 2, *Cycle-Specific DIVOM Methodology Using the RAMONA5-FA Code*, AREVA NP, May 2008.

23. BWROG-94078, "BWR Owners' Group Guidelines for Stability Interim Corrective Action," June 6, 1994.

24. EMF-CC-074(P)(A) Volume 4 Revision 0, *BWR Stability Analysis – Assessment of STAIF with Input from MICROBURN-B2*, Siemens Power Corporation, August 2000.

25. NEDO-32047-A, *ATWS Rule Issues Relative to BWR Core Thermal-Hydraulic Stability*, GE Nuclear Energy, June 1995.

26. NEDO-32164, *Mitigation of BWR Core Thermal-Hydraulic Instabilities in ATWS*, GE Nuclear Energy, December 1992.

27. ANF-913(P)(A) Volume 1 Revision 1 and Volume 1 Supplements 2, 3 and 4, *COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses*, Advanced Nuclear Fuels Corporation, August 1990.

28. ANP-3355P Revision 1, *Browns Ferry Units 1, 2, and 3 Plant Parameters Document (EPU),* AREVA Inc., March 2015.

29. XN-NF-84-105(P)(A) Volume 1 and Volume 1 Supplements 1 and 2, *XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis*, Exxon Nuclear Company, February 1987.

30. ANF-1358(P)(A) Revision 3, *The Loss of Feedwater Heating Transient in Boiling Water Reactors*, Framatome ANP, September 2005.

31. Letter WB Bird (TVA) to DB McBurney (AREVA), “Transmittal of Recirculation Pump Seizure Information for Two Loop Operation,” February 20, 2015. (38-9236411-000)

32. XN-NF-80-19(P)(A) Volume 1 and Supplements 1 and 2, *Exxon Nuclear Methodology for Boiling Water Reactors – Neutronics Methods for Design and Analysis*, Exxon Nuclear Company, March 1983.

33. ANP-10298PA Revision 0, *ACE/ATRIUM 10XM Critical Power Correlation*, AREVA NP, March 2010.

34. ANP-3140(P) Revision 0, *Browns Ferry Units 1, 2, and 3 Improved K-factor Model for ACE/ATRIUM 10XM Critical Power Correlation*, AREVA NP, August 2012.

35. BAW-10247PA Revision 0, *Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors*, AREVA NP, February 2008.

36. NUREG-0800, U.S. Nuclear Regulatory Commission, Standard Review Plan, Section 15.4.4-15.4.5, "Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature, and Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate," Revision 2, March 2007.

37. ANP-3377P Revision 3, *Browns Ferry Units 1, 2, and 3 LOCA Break Spectrum Analysis for ATRIUM 10XM Fuel (EPU),* August 2015.

38. ANP-3378P Revision 3, *Browns Ferry Units 1, 2, and 3 LOCA-ECCS Analysis MAPLHGR Limits for ATRIUM 10XM Fuel (EPU),* August 2015.

39. ANP-3384P Revision 3, *Browns Ferry Units 1, 2, and 3 LOCA-ECCS Analysis MAPLHGR Limits for ATRIUM-10 Fuel (EPU)*, August 2015.

40. EMF-2292(P)(A) Revision 0, *ATRIUM™-10: Appendix K Spray Heat Transfer Coefficients*, Siemens Power Corporation, September 2000.

41. Letter, P. Salas (AREVA) to Document Control Desk, U.S. Nuclear Regulatory Commission, *Proprietary Viewgraphs and Meeting Summary for Closed Meeting on Application of the EXEM BWR-2000 ECCS Evaluation Methodology*, NRC:11:096, September 22, 2011.

42. Letter, T.J. McGinty (NRC) to P. Salas (AREVA), *Response to AREVA NP, Inc. (AREVA) Proposed Analysis Approach for Its EXEM Boiling Water Reactor (BWR)-2000 Emergency Core Cooling System (ECCS) Evaluation Model*, July 5, 2012.

43. NEDO-20566A*, Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR Appendix K,* General Electric Company, September 1986.

44. ANP-3160(P) Revision 0, *Browns Ferry Nuclear Plant Units 1, 2, and 3 Spent Fuel Storage Pool Criticality Safety Analysis for ATRIUM™ 10XM Fuel*, AREVA NP, October 2012.

45. ANP-2945(P) Revision 1, *Browns Ferry Nuclear Plant Units 1, 2, and 3 Spent Fuel Storage Pool Criticality Safety Analysis*" AREVA NP, July 2011.

46. Final Division of Safety Systems Interim Staff Guidance, DSS-ISG-2010-01 Revision 0, *Staff Guidance Regarding The Nuclear Safety Analysis for Spent Fuel Pools*, Nuclear Regulatory Commission (ADAMS Accession Number ML110620086).

47. Regulatory Guide 1.183, *Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors*, July 2000.

48. Letter TVA to US NRC, *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. (ML13358A067)

49. Letter TVA to US NRC, *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. (ML14175A307)

50. Letter TVA to US NRC, *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. (ML15040A698)

51. Letter from T. A. Galioto to J. F. Lemons, *Browns Ferry GE14 Decay Heat Comparison with ATRIUM-10 Decay Heat Results for Extended Power Uprate*, January 30, 2003. (TAG:03:009 T1.2-JO2BFE114_TAG-03-009-000)

52. Letter US NRC to TVA, *Browns Ferry Nuclear Plant – Issuance of Amendments Regarding Full Scope Implementation of Alternative Source Term*, September 27, 2004 (ML042730028).