ENCLOSURE 15

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNITS 2 AND 3

PROPOSED TECHNICAL SPECIFICATIONS (TS) CHANGE TS - 418 - REQUEST FOR LICENSE AMENDMENT FOR EXTENDED POWER UPRATE OPERATION

NON-PROPRIETARY EMF-2982(NP) BROWNS FERRY UNITS 2 AND 3 SAFETY ANALYSIS REPORT FOR EXTENDED POWER UPRATE ATRIUM-10 FUEL SUPPLEMENT

See Attached:

EMF-2982(NP)
Revision 0

Browns Ferry Units 2 and 3
Safety Analysis Report for Extended Power Uprate
ATRIUM™-10 Fuel Supplement

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Framatome ANP, Inc.

EMF-2982(NP)
Revision 0

Browns Ferry Units 2 and 3
Safety Analysis Report for Extended Power Uprate
ATRIUM™-10 Fuel Supplement

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Nature of Changes

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1.	All	This is a new document.

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Executive Summary

Tennessee Valley Authority is implementing an extended power uprate (EPU) at Browns Ferry (BFN) Units 2 and 3 to increase the maximum power level from 3458 to 3952 MWt. General Electric report NEDC-33047P Revision 2, *Browns Ferry Units 2 and 3 Safety Analysis Report for Extended Power Uprate* (Reference 1), provides a summary of the analysis results for a GE14 equilibrium core. TVA's current plan is to go to the EPU power level using Framatome ANP, Inc.* (FANP) ATRIUM™-10† fuel. The report to follow is a summary of the results of the analysis addressing the impact of operation of BFN Units 2 and 3 at EPU conditions with ATRIUM-10 fuel.

The safety analyses performed to support EPU can be characterized as being fuel-related (demonstrating compliance with fuel design criteria and licensing requirements) or plant-related (demonstrating compliance with plant design criteria and licensing requirements). The plant-related safety analyses can be further classified as being either dependent or not dependent on fuel design parameters. Many of the Reference 1 EPU evaluations are not affected by a change in fuel design. This report provides results for fuel-related analyses and either analysis results or justification for the continued applicability of the Reference 1 evaluations for the plant-related analyses. All analyses are performed for a reference ATRIUM-10 equilibrium core. The analyses were performed using NRC-approved or industry-accepted analysis methods.

The results and conclusions presented below support the overall conclusion that Browns Ferry Units 2 and 3 can safely operate at EPU conditions with ATRIUM-10 fuel.

- There are no new potentially limiting events beyond those previously identified and there is no significant increase in the probability of a postulated event occurring.
- The safety aspects of the plant that are affected by EPU operation with ATRIUM-10 fuel were evaluated and no plant modifications are required to meet safety requirements.
- The results of the potentially limiting accident and transient analyses are reported and demonstrate that no existing regulatory limits are exceeded.
- Systems and components affected by the use of ATRIUM-10 fuel were reviewed to ensure there is no significant challenge to safety systems.
- Compliance with current plant environmental regulations has been maintained.

* Framatome ANP, Inc. is an AREVA and Siemens company.
† ATRIUM is a trademark of Framatome ANP.

1.0 Introduction

1.1 *Report Approach*

Tennessee Valley Authority (TVA) is implementing an extended power uprate (EPU) at Browns Ferry Units 2 and 3 to 3952 MWt, which is 120% of the original licensed thermal power (OLTP). Reference 1 presents results of safety evaluations performed that support the extended power uprate to 3952 MWt. The Reference 1 evaluations were performed by GE Nuclear Energy and assume a reference GE14 equilibrium core. ATRIUM-10 fuel will be used for operation at EPU conditions. As a result, analyses for ATRIUM-10 fuel have been performed at EPU conditions. This report summarizes the impact that operation with ATRIUM-10 fuel at EPU conditions has on the Reference 1 evaluations for Browns Ferry Units 2 and 3. The ATRIUM-10 EPU analyses follow the NRC-approved generic format and content described in References 2 and 3.

Table 1.1 presents a glossary of terms used in the report.

1.2 *Purpose and Approach*

The evaluations performed to assess the impact of utilizing ATRIUM-10 fuel demonstrate that the EPU power level increase can be accomplished within the applicable safety design criteria. Many of the Reference 1 safety evaluations and equipment assessments performed for the Browns Ferry EPU are unaffected by a change in fuel design. This Browns Ferry EPU report for ATRIUM-10 fuel follows a structure similar to Reference 1.

The safety analyses performed to support the extended power uprate can be characterized as being *fuel-related* (demonstrating compliance with fuel design criteria and licensing requirements) or *plant-related* (demonstrating compliance with plant design criteria and licensing requirements). Plant-related safety analyses can be further classified as being either dependent or not dependent on fuel design parameters (i.e., does the analysis depend significantly on any fuel-design-dependent input parameters). This report provides results for fuel-related analyses and either analysis results or justification for the continued applicability of plant-related analyses presented in Reference 1. All fuel-related analyses are performed for a reference ATRIUM-10 equilibrium core.

1.2.1 Uprate Analysis Basis

Reference 1 provides the evaluations to support operation at EPU conditions. The ATRIUM-10 EPU evaluations are based on the uprated power level of 3952 MWt. Table 1.2 presents plant-specific parameters used in the EPU analyses at rated conditions. For most of the EPU analyses, the 2% power factor discussed in Regulatory Guide 1.49 is accounted for in the analysis methods. Three exceptions are ASME overpressurization, loss of feedwater flow, and loss of cooling accident emergency core cooling system (LOCA-ECCS) analyses. These analyses are performed with the factor included in the analysis power level.

1.2.2 Computer Codes

The computer codes and calculation techniques used in the EPU analyses have been approved by the U.S. Nuclear Regulatory Commission (NRC) or are accepted throughout the nuclear industry. The computer codes used in the ATRIUM-10 EPU evaluations are listed in Table 1.3. The analyses are performed in accordance with the NRC limitations and restrictions included in the applicable Safety Evaluation Reports (SERs). Any exceptions are noted in Table 1.3.

1.2.3 Approach

The scope and depth of the evaluation results provided in this report are based on the processes used in Reference 1. Sections 2.0 – 11.0 provide evaluations of ATRIUM-10 fuel at EPU operation on the respective topics. The scope of the evaluations is summarized in the following sections:

Section 2.0 - Reactor Core and Fuel Performance. A representative ATRIUM-10 equilibrium fuel cycle operating at EPU conditions was used as the basis for the EPU analyses. Thermal limits, reactivity characteristics and stability are evaluated each cycle and will continue to be evaluated each cycle. The control rod reactivity control systems are not impacted by a change in fuel design.

Section 3.0 - Reactor Coolant and Connected Systems. The impact of BFN operation with ATRIUM-10 fuel on the Reference 1 evaluations of the NSSS components and systems has been assessed. Since the reactor operating pressure, core flow, steam flow, and feedwater flow are unchanged for operation with ATRIUM-10 fuel, the effects on the reactor coolant and connected systems are minor.

Section 4.0 - Engineered Safety Features. The effects of EPU operation with ATRIUM-10 fuel on the engineered safety features have been evaluated for key events. The evaluations show that the appropriate acceptance criteria are met with ATRIUM-10 fuel.

Section 5.0 - Control and Instrumentation. The instrumentation, control systems, and analytical limits for setpoints were evaluated for any impact due to a change in fuel design. The evaluations show that the use of ATRIUM-10 fuel will not impact the instrumentation and control systems. The Reference 1 analytical limits remain valid for ATRIUM-10 fuel.

Section 6.0 - Electrical Power and Auxiliary Systems. Evaluations of the electrical power and auxiliary systems show that operation with ATRIUM-10 fuel does not impact the capability of these systems to support safe plant operation at EPU conditions.

Section 7.0 - Power Conversion Systems. Fuel design does not affect the EPU thermal power, steam flow, or feedwater flow. As a result, the power conversion systems are not affected with the use of ATRIUM-10 fuel.

Section 8.0 - Radwaste Systems and Radiation Sources. Evaluations have been performed to ensure that the appropriate acceptance criteria are met with ATRIUM-10 fuel.

Section 9.0 - Reactor Safety Performance Evaluations. Analyses for ATRIUM-10 fuel were performed for the potentially limiting design basis events using limiting conditions for EPU. The use of ATRIUM-10 fuel does not introduce any new limiting events. The results of the ATRIUM-10 limiting accident and transient analyses show compliance with regulatory requirements.

Section 10.0 - Other Evaluations. EPU operation with ATRIUM-10 fuel does not impact the high energy line break and environmental qualification evaluations presented in Reference 1. The Reference 1 Individual Plant Evaluation also remains applicable.

Section 11.0 - Licensing Evaluations. With the exception of the emergency operating procedures which are updated on a cycle-specific basis, the Reference 1 licensing evaluations remain applicable for ATRIUM-10 fuel.

1.3 *Uprated Plant Operating Conditions*

The use of ATRIUM-10 fuel will have no impact on core operating conditions. However, for completeness, the operating conditions based on analyses performed by Framatome ANP (FANP) at EPU conditions are presented.

1.3.1 Reactor Heat Balance

The steady state thermal-hydraulic performance of a boiling-water reactor (BWR) reactor core can be characterized by the operating conditions - operating pressure, total core flow, and the coolant thermodynamic state. The operating conditions provide the initial and boundary conditions for the safety analyses that are necessary to show compliance with the appropriate acceptance criteria and are determined by performing heat balance calculations at EPU conditions.

The heat balance results for the EPU power level and 100% rated core flow are shown in Figure 1.1 and the results for 102% of uprated power and rated core flow are shown in Figure 1.2. Table 1.2 provides a summary of reactor thermal-hydraulic parameters for EPU conditions. Note that the pressure upstream of the turbine stop valve is higher than the Reference 1 value and is representative of the Browns Ferry Unit 1 configuration. The result is a lower steam line pressure drop which is conservative for pressurization events. A lower steam line pressure drop has no impact on the other operating conditions.

1.3.2 Reactor Performance Improvement Features

Several performance improvement features and equipment-out-of-service (EOOS) conditions are supported for current operation. A list of these features for operation at EPU conditions is presented in Table 1.2 and is the same as the list presented in Reference 1. While the features themselves are not fuel design dependent, they do impact analyses performed to establish or confirm the fuel design specific operating limits. The limiting analyses allowing for OOS features have been performed for ATRIUM-10 fuel at EPU conditions.

1.4 *Summary and Conclusions*

This evaluation demonstrates that operation of ATRIUM-10 fuel at BFN Units 2 and 3 at EPU conditions does not cause a significant increase in the probability or consequences of an accident previously evaluated. In addition, there are no new or different events or accidents that result from operating with ATRIUM-10 fuel. Limits can be established to support operation while continuing to meet the applicable plant regulatory and design limits. As a result, there is no reduction in the margin of safety.

Table 1.1 Glossary of Terms

AC	alternating current
ADHR	auxiliary decay heat removal
ADS	automatic depressurization system
AL	analytical limit
ALARA	as low as reasonably achievable
AOO	anticipated operational occurrence
APRM	average power range monitor
ARTS	APRM/RBM/Technical Specifications
ASME	American Society of Mechanical Engineers
AST	alternate source term
ATWS	anticipated transient without scram
ATWS-RPT	anticipated transient without scram recirculation pump trip
AVZ	above vessel zero
BFN	Browns Ferry Nuclear Plant
BOL	beginning of life
BOP	balance of plant
BWR	boiling water reactor
BWROG	BWR Owners Group
CAD	containment atmospheric dilution
CFR	Code of Federal Regulations
CLTP	current licensed thermal power (3458 MWt)
COLR	core operating limits report
CPR	critical power ratio
CRD	control rod drive
CRWE	control rod withdrawal error
DC	direct current
ECCS	emergency core cooling system
EHC	electro-hydraulic control
EOC	end of cycle
EOOS	equipment out-of-service
EPU	extended power uprate
FANP	Framatome ANP
FPCC	fuel pool cooling and cleanup
FWCF	feedwater controller failure – maximum demand
GL	generic letter
HPCI	high-pressure coolant injection
HVAC	heating ventilation and air conditioning

Table 1.1 Glossary of Terms *(Continued)*

ICA	interim corrective action
ICF	increased core flow
IRM	intermediate range monitor
LFWH	loss of feedwater heating
LHGR	linear heat generation rate
LOCA	loss-of-coolant accident
LOFW	loss of feedwater flow
LPCI	low-pressure coolant injection
LPCS	low-pressure core spray
LPRM	local power range monitor
LRNB	load rejection without bypass
MAPLHGR	maximum average planar linear heat generation rate
MCPR	minimum critical power ratio
MELLLA	maximum extended load line limit analysis
MSIV	main steam isolation valve
MSIVC	main steam isolation valve closure
MSIVF	main steam isolation valve closure with scram on high flux
MSRV	main steam relief valve
MWR	metal-water reaction
NPSH	net positive suction head
NRC	Nuclear Regulatory Commission, U.S.
NSSS	nuclear steam supply system
OLMCPR	operating limit minimum critical power ratio
OLTP	original licensed thermal power
OOS	out-of-service
OPRM	oscillation power range monitor
PCT	peak cladding temperature
PPM	parts per million
PRA	probabilistic risk assessment
PRNMS	power range neutron monitoring system
RBM	rod block monitor
RCIC	reactor core isolation cooling
RHR	residual heat removal
RPT	recirculation pump trip
RPTOOS	recirculation pump trip out-of-service
RPV	reactor pressure vessel
RTP	rated thermal power
RWCU	reactor water cleanup

Table 1.1 Glossary of Terms *(Continued)*

SBO	station blackout
scf	standard cubic feet
SER	safety evaluation report
SFP	spent fuel pool
SLCS	standby liquid control system
SLMCPR	safety limit minimum critical power ratio
SLO	single-loop operation
SRM	source range monitor
SSDS	safe shutdown system
TAF	top of active fuel
TBVOOS	turbine bypass valve OOS
TCV	turbine control valve
TS	Technical Specification
TSSS	Technical Specification scram speed
TSV	turbine stop valve
TTNB	turbine trip without bypass
TVA	Tennessee Valley Authority
UFSAR	Updated Final Safety Analysis Report
v/o	percent concentration by volume

Table 1.2 Browns Ferry EPU
Plant Operating Conditions for ATRIUM-10 Evaluations

Parameter	*Value*
Rated thermal power, MWt	3952
Rated core flow, Mlbm/hr	102.5
Rated steam flow,* Mlbm/hr	16.44
Rated power core flow range Mlbm/hr % rated	 101.5 to 107.6 99 to 105
Dome pressure, psia	1050
Dome temperature, °F	550.5
Pressure at upstream side of turbine stop valve (TSV), psia	988†
Rated power feedwater Flow, Mlbm/hr Temperature, °F	 16.39 394.5
Core inlet enthalpy,‡ Btu/lbm	523.2
EOOS / Operational Enhancements	
APRM / RBM / Technical Specifications (ARTS) - Maximum extended load line limit analysis (MELLLA) with power range neutron monitoring system (PRNMS)	
End-of-cycle (EOC) coastdown	
Single-loop operation	
Final feedwater temperature reduction	
Feedwater heater out-of-service	
1 main steam relief valve (MSRV) out-of-service (OOS)	
3% MSRV setpoint tolerance	
Increased core flow (ICF)	
EOC - recirculation pump trip OOS (RPTOOS)	
Turbine bypass valve OOS (TBVOOS)	
24-month fuel cycle	
Reactor Level 3 reduction	

* At normal feedwater heating.

† This value is higher than the Reference 1 value and is representative of the Browns Ferry Unit 1 configuration. The result is a lower steam line pressure drop which is conservative for pressurization events.

‡ At 100% core flow.

Table 1.3 Computer Codes Used for ATRIUM-10 EPU Analyses

Task	Computer Code	Version	NRC Approved	Comments
Reactor heat balance	HTBAL	UMAY94	*(1)*	----
Reactor core and fuel performance	CASMO-4 MICROBURN-B2	UFEB01a UJAN03	Y Y	EMF-2158(P)(A) Rev. 0 .... (Ref. 9) EMF-2158(P)(A) Rev. 0 .... (Ref. 9)
Safety limit MCPR	SAFLIM2	UOCT02	Y	ANF-524(P)(A) Rev. 2................ (Ref. 10)
Transient analysis	MICROBURN-B2 XCOBRA COTRANSA2 XCOBRA-T RODEX2	UJAN03 UAUG02 AAPR03 /UJUL03 UAUG02 UAPR02	Y Y *(2)* Y *(3)* Y *(3)* Y	EMF-2158(P)(A) Rev. 0 .... (Ref. 9) XN-NF-80-19(P)(A) Vol. 3 Rev. 2 .. (Ref. 8) ANF-913(P)(A) Vol. 1 Rev. 1 .... (Ref. 6) XN-NF-84-105(P)(A) Vol. 1 .... (Ref. 7) XN-NF-81-58(P)(A) Rev. 2.......... (Ref. 11)
Anticipated transient without scram – overpressurization	COTRANSA2	AAPR03	Y *(3)*	ANF-913(P)(A) Vol. 1 Rev. 1 .... (Ref. 6)
Appendix R - fire protection	RELAX HUXY RODEX2	UAUG02 UJAN01 UAPR02	*(4)* *(4)* *(4)*	EMF-2361(P)(A) Rev. 0 .... (Ref. 12) XN-CC-33(P)(A) Rev. 1............... (Ref. 13) XN-NF-81-58(P)(A) Rev. 2.......... (Ref. 11)
LOCA-ECCS	RELAX HUXY RODEX2	UAUG02 UJAN01 UAPR02	Y Y Y	EMF-2361(P)(A) Rev. 0 .... (Ref. 12) XN-CC-33(P)(A) Rev. 1............... (Ref. 13) XN-NF-81-58(P)(A) Rev. 2.......... (Ref. 11)
Reactor core stability	STAIF	UFEB02	Y	EMF-CC-074(P)(A) Vol. 4 Rev. 0... (Ref. 5)

NOTES:

1. HTBAL is not explicitly approved by the NRC but it is a stand-alone version of the heat balance routine included in the NRC-approved MICROBURN-B2 code documented in Reference 9.

2. The approval of XCOBRA is included in the approval of the THERMEX methodology in Reference 8.

3. The list of events for which COTRANSA2 and XCOBRA-T can be used was expanded in the clarification acceptance in Reference 14.

4. [

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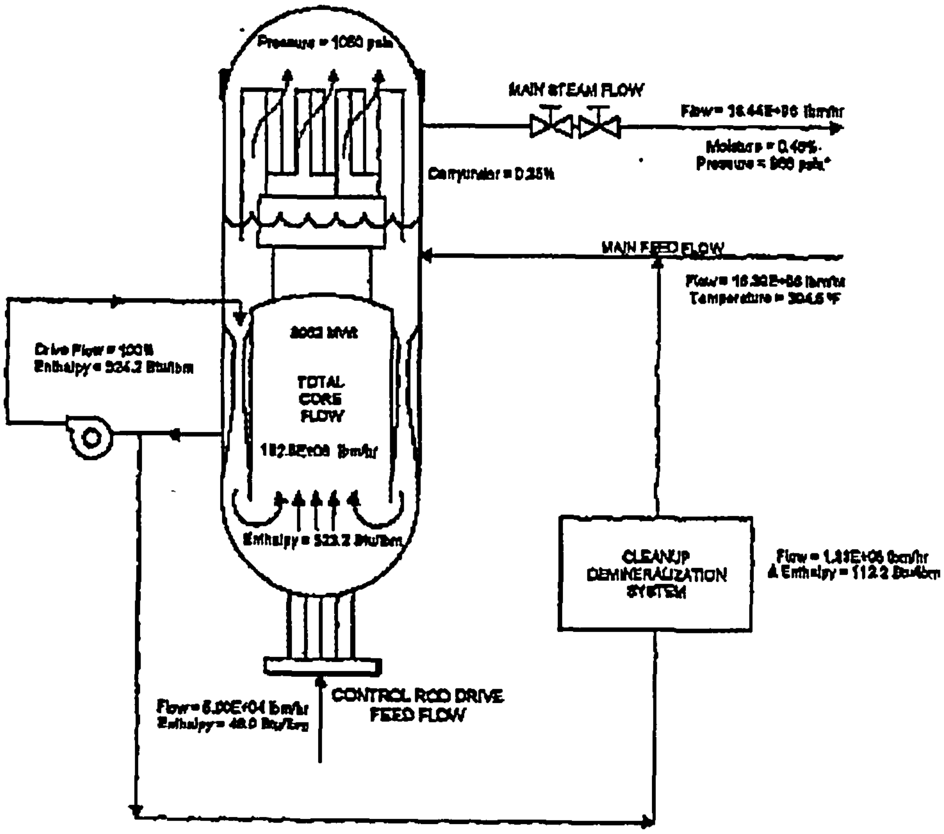
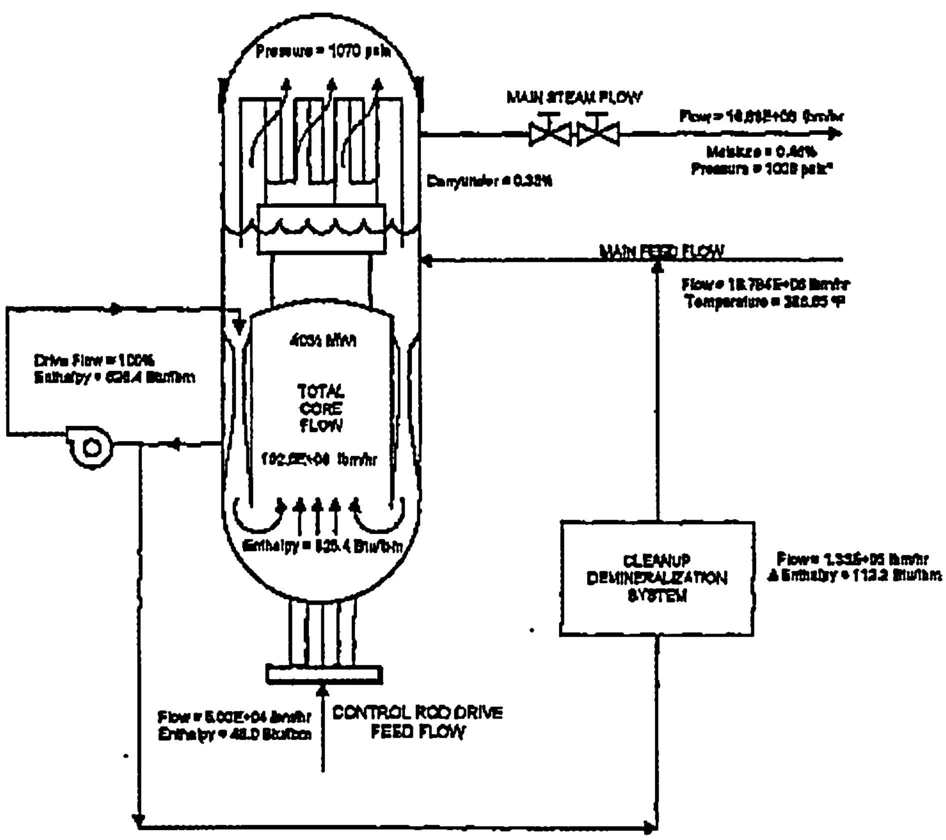


Core Thermal Power	3951.8
Pump Heating	10.8
Cleanup Losses	-4.4
Other System Losses	-1.1
Turbine Cycle Use	3956.7 MWt

*This value is higher than the Reference 1 value and is representative of the Browns Ferry Unit 1 configuration. This results in a lower steam line pressure drop which is conservative for pressurization events.

Figure 1.1 Browns Ferry Heat Balance for ATRIUM-10 EPU Analysis - Nominal (100% power and 100% core flow)



Core Thermal Power	4030.6
Pump Heating	10.8
Cleanup Losses	-4.4
Other System Losses	-1.1
Turbine Cycle Use	4038.3 MWt

* This value is higher than the Reference 1 value and is representative of the Browns Ferry Unit 1 configuration. This results in a lower steam line pressure drop which is conservative for pressurization events.

Figure 1.2 Browns Ferry Heat Balance for ATRIUM-10 EPU – Overpressure Protection Analysis (102% power and 100% core flow)

2.0 Reactor Core and Fuel Performance

2.1 *Fuel Design and Operation*

The effect of using ATRIUM-10 fuel on operation at EPU conditions is described below. A change in fuel design has no impact on the average bundle power. While the average bundle power for EPU remains 5.17 MW/bundle, the difference in the thermal-hydraulic, mechanical and neutronic characteristics of the ATRIUM-10 fuel necessitate changes to the lattice and core design in order to meet the EPU operating requirements. These changes include enrichment and burnable poison distribution, reload batch size, and loading patterns. The power distributions throughout the cycle are reviewed to ensure that the following operating limits are met.

- Minimum critical power ratio (MCPR). Ensures an acceptable low probability of fuel cladding failure resulting from the fuel experiencing boiling transition.
- Linear heat generation rate (LHGR). Ensures that fuel mechanical design bases are met.
- Maximum average planar linear heat generation rate (MAPLHGR). Ensures that peak cladding temperature and metal-water reaction criteria for the limiting loss-of-coolant accident (LOCA) are met.

The evaluations performed to assess the impact of using ATRIUM-10 fuel at EPU conditions assume a reference equilibrium core of ATRIUM-10 fuel consistent with the energy and cycle length requirements used in the Reference 1 evaluation. All fuel and core design limits continue to be met by the planned enrichment, burnable poison, and control rod positions. The methods used to perform the ATRIUM-10 fuel and core design analyses for EPU have been approved by the NRC and show that the ATRIUM-10 equilibrium core meets the EPU operating requirements and stays within the operating limits. All ATRIUM-10 reload core designs for operation at EPU conditions will take into account the operating limits discussed above (MCPR, LHGR, and MAPLHGR) to ensure acceptable design margins exist between the licensing limits and their corresponding operating values.

The NRC-approved exposure limits are not exceeded in the ATRIUM-10 equilibrium core design used in the EPU evaluations. Using ATRIUM-10 fuel may have some effects on the operating flexibility and reactivity characteristics but are accounted for in the design and licensing process. Appropriate radiation source terms have been determined which include ATRIUM-10 fuel at EPU conditions (See Sections 8 and 9.2).

2.1.1 Fuel Thermal Margin Monitoring Threshold

Since there is no change in the average bundle power with ATRIUM-10 fuel, there is no change to the thermal margin monitoring threshold.

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-10 fuel has on thermal limits. The evaluations were performed using an ATRIUM-10 equilibrium core. Consistent with the current practice, cycle-specific thermal limits are established or confirmed each reload based on the cycle-specific core configuration.

2.2.1 Safety Limit Minimum Critical Power Ratio

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). An analysis of the SLMCPR for the reference ATRIUM-10 equilibrium core was performed. The results support an SLMCPR of 1.08.

2.2.2 Operating Limit Minimum Critical Power Ratio

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. For the reference ATRIUM-10 equilibrium core, the OLMCPR for EPU RTP operation is shown in Table 9.2. 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 MCPR (ΔCPR) for the limiting anticipated operational occurrence (AOO) event and the SLMCPR. The impact that ATRIUM-10 fuel has on AOO events at EPU conditions is addressed in Section 9.1.

2.2.3 MAPLHGR and Maximum LHGR Operating Limits

LOCA-ECCS analyses are performed to demonstrate that the MAPLHGR limits provide the necessary protection. With FANP methods, MAPLHGR operating limits are established for a fuel type (e.g. ATRIUM-10 fuel) at a given plant. Analyses are performed each reload cycle to ensure that established MAPLHGR limits are applicable to the new fuel assembly design. The results presented in Section 4.3 show that the ATRIUM-10 MAPLHGR limits meet the regulatory limits. The ATRIUM-10 MAPLHGR limits are the same for CLTP and EPU conditions.

The 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. 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-10 fuel are not cycle-specific, the power- and flow-dependent LHGR multipliers are established each cycle since 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).

2.3 *Reactivity Characteristics*

Reload core design analyses are performed on a cycle-specific basis to ensure that required reactivity margins are maintained. Current Technical Specification (TS) requirements for cold shutdown margin are maintained at EPU conditions with ATRIUM-10 fuel by appropriate design of the burnable neutron absorber content and by judicious placement of fresh and irradiated assemblies. Operation with ATRIUM-10 fuel at EPU conditions does not change cold shutdown margin requirements. All current TS reactivity control requirements at the most reactive conditions are met and confirmed by cycle-specific analyses.

Cycle length and hot excess reactivity are maintained by appropriate selection of initial enrichment, fresh batch size, and burnable neutron absorber design. Sufficient design flexibility exists with the ATRIUM-10 fuel to accommodate operation at EPU conditions while maintaining adequate power distribution control.

2.3.1 <u>Power/Flow Operating Map</u>

The use of ATRIUM-10 fuel does not affect the power/flow operating map. Therefore, the Browns Ferry Units 2 and 3 EPU power/flow operating map presented in Reference 1 remains applicable.

2.4 *Stability*

Browns Ferry has installed a power range neutron monitoring system with oscillation power range monitors (OPRMs) to implement the BWR Owners Group (BWROG) Long-Term Stability Solution Option-III. This system is designed to provide for an automatic scram for the reactor when power oscillations above the system setpoint are detected. The Option-III trip is armed only when plant operation is within a defined region of the power/flow map. For EPU operation this Armed Region is defined with a flow boundary specified as 60% of rated flow. The power level boundary is defined as 30% of the original licensed thermal power scaled to the uprated power level. For the EPU of 120% of the original power, the power level boundary is therefore 25% (30%/1.20).

When the OPRM system is inoperable, the plant may use an alternate stability detect and suppress method. Current practice with the Option-III system is to use the stability interim corrective actions (ICAs) as the backup method. The ICAs include specific requirements for operator action as well as restrictions on operation in certain regions of the power/flow map. These ICA regions are validated on a cycle-specific basis using FANP's STAIF methodology (Reference 5) and expanded as necessary.

The OPRM system utilizes setpoints to ensure the SLMCPR is not exceeded during a postulated power oscillation. The OPRM setpoint is evaluated for each reload core (including transition cores) on a cycle-specific basis.

2.5 *Reactivity Control*

The control rod drive (CRD) system operation, control rod positioning and control rod cooling and system integrity are not dependent on fuel design.

3.0 Reactor Coolant and Connected Systems

3.1 *Nuclear System Pressure Relief*

Overpressurization analyses performed at EPU conditions for an ATRIUM-10 core demonstrate the adequacy of the pressure relief system. Compliance with the ASME pressure vessel code criteria is demonstrated for each reload core. The anticipated transient without scram (ATWS) vessel pressure evaluation is discussed in Section 9.3.1.

The fuel design does not affect the MSRV setpoints; therefore, there is no effect on the MSRV functionality (opening/closing).

3.1.1 MSRV Setpoint Tolerance

MSRV setpoint tolerance is independent of fuel design. The ATRIUM-10 EPU evaluations were performed using the existing MSRV setpoint tolerance analytical limit of 3%.

3.2 *Reactor Overpressure Protection Analysis*

The design pressure of the reactor vessel is not affected by fuel design and remains 1250 psig. Per the 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-10 equilibrium core were performed for the main steam isolation valve (MSIV) closure and turbine trip with turbine bypass failure events. The events were analyzed at 102% of EPU rated thermal power (RTP) and an initial dome pressure of 1055 psig, which is higher than the nominal dome pressure. The MSRV setpoints presented in Table 9.1 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 1343 psig. The corresponding calculated peak dome pressure is 1315 psig. 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-10 overpressure analyses are presented in Figures 3.1–3.13.

[

]

3.3 *Reactor Vessel and Internals*

Many of the RPV structure and support components are unaffected by a change in fuel design, and the Reference 1 analyses for EPU operation remain applicable for ATRIUM-10 fuel. A review of the effects on the reactor vessel and its internals due to operating with ATRIUM-10 fuel is presented below.

3.3.1 Reactor Vessel Fracture Toughness

[

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The maximum normal operating dome pressure is not dependent on fuel design; therefore, the hydrostatic and leakage test pressures are not affected by fuel design.

3.3.2 Reactor Vessel Structural Evaluation

The reactor vessel structural evaluation, including the vessel structural design considerations, the normal and upset conditions, and the emergency and faulted conditions, is not dependent on fuel design.

3.3.3 Reactor Internal Pressure Differences

The impact of a change in fuel design on the pressure differences and loads on the reactor internals was evaluated. The main fuel design items that potentially impact the reactor internal pressure differences and loads are the core pressure drop, the fuel channel geometry, and the weight of the fuel assembly. Thermal-hydraulic evaluations show that the ATRIUM-10 fuel has a lower pressure drop than the GE14 fuel design for a given power distribution and assembly flow. This result supports the conclusion that the reactor internal pressure differences that are impacted by the core pressure drop (i.e. core plate, guide tube, and shroud support) are acceptable. The reactor parameters that were evaluated due to a change in geometry of the fuel channel and/or assembly weight are the top guide pressure drop and fuel assembly lift. Pressure differences for the other reactor internal components are not affected by a change in fuel design.

[

] The increase in pressure drop is bound by the current licensed thermal power (CLTP) top guide pressure drop results presented in Tables 3-3 and 3-4 of Reference 1 for the normal and upset conditions. This shows that the pressure drop increase for the ATRIUM-10 fuel is acceptable for normal and upset conditions. The top guide pressure drop for the faulted condition is 3.1 psid, higher than the 2.8 psid value reported in Reference 1 Table 3-5. The impact of this higher pressure drop is discussed in Section 3.3.4.

The Browns Ferry ATRIUM-10 fuel channel analyses show that the maximum pressure load is acceptable. A pressure load limit analysis was performed on the fuel channel design to determine the allowable differential pressure while still satisfying the strength criteria. This limit was compared to the maximum calculated pressure load due to normal operation plus postulated accident loads.

An ATRIUM-10 fuel assembly liftoff evaluation was performed at EPU RTP and 105% core flow. Lower flow rates are bound by the 105% core flow result. The liftoff analysis shows that the net lift force for the ATRIUM-10 fuel during normal and upset conditions remains in the downward direction. The evaluation also concludes that under accident (faulted) conditions, the ATRIUM-10 assembly will remain engaged in the fuel support so the fuel lift criteria are met.

3.3.4 Reactor Internals Structural Evaluation

The impact of a change in fuel design on the structural adequacy of the reactor internals was evaluated. The maximum acoustic and flow-induced loads following a postulated recirculation line break are unaffected by a change in fuel design. Similar to the earlier pressure difference evaluation, the main fuel design items that potentially impact the reactor internals structural integrity are the core pressure drop, the fuel channel geometry, and the weight of the fuel assembly. The thermal-hydraulic evaluation shows that the ATRIUM-10 fuel has a lower pressure drop than the GE14 fuel design for a given power distribution and assembly flow. This result supports the conclusion that the reactor internal structural evaluations that are impacted by the core pressure drop (i.e. core plate, guide tube, shroud support, and access hole cover) are acceptable. The weight of the ATRIUM-10 assembly is essentially the same as the GE14 fuel assembly used in the Reference 1 analyses (ATRIUM-10 fuel weighs 0.1 lb or 0.02% less than GE14 fuel). Since there is no significant difference between the weights of the assemblies, the Reference 1 evaluation remains applicable for the reactor components that are affected by a change in the assembly weight: the CRD housing, control rod guide tubes, and orificed fuel support.

The larger top guide accident (faulted) pressure drop, discussed in Section 3.3.3, results in a higher upward force on the top guide. An evaluation of the loads on the top guide shows that the combination of top guide weight, seismic force, and hydraulic upward forces results in adequate margin to avoid top guide lift.

[

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3.3.5 Flow-Induced Vibration

[

] Therefore, there is no increase in the flow-induced vibration of the in-core guide tube with ATRIUM-10 fuel.

The flow-induced vibration evaluation for the ATRIUM-10 fuel channel has been shown to be acceptable. The use of ATRIUM-10 fuel does not impact the flow-induced vibration on the shroud, shroud head and separator, jet pumps, feedwater sparger, steam dryer, control rod guide tubes, and jet pump sensing lines.

3.3.6 Steam Separator and Dryer Performance

The performance of the steam separator and dryer is not dependent on fuel design.

3.4 *Reactor Recirculation System*

The core pressure drop for ATRIUM-10 fuel is lower than the pressure drop for the GE14 fuel. The result is a small decrease in the pump speed for a given core flow rate. The change in pressure drop does not affect the ability of the reactor recirculation pumps to meet the core flow requirements. Other aspects of the reactor recirculation system are not affected by a change in fuel design.

3.5 *Reactor Coolant Pressure Boundary Piping*

The reactor coolant pressure boundary piping evaluations are not dependent on fuel design.

3.6 *Main Steam Line Flow Restrictors*

The performance of the main steam line flow restrictors is not dependent on fuel design.

3.7 *Main Steam Isolation Valves*

The MSIV evaluation presented in Reference 1 remains applicable for ATRIUM-10 fuel.

3.8 *Reactor Core Isolation Cooling*

The reactor core isolation cooling (RCIC) system is designed to maintain sufficient water level to ensure adequate cooling after a loss of feedwater flow event, in conjunction with a reactor isolation event. The RCIC system must inject sufficient makeup water so that the water level remains above the top of active fuel (TAF). There is also an operational requirement that the RCIC system be able to restore the reactor water level while avoiding initiation of the automatic depressurization system (ADS) timer and MSIV closure activation on low-low-low water level (Level 1). Loss of feedwater flow analyses were performed without reactor isolation since this scenario presents a greater challenge to the system's ability to maintain the water level above the Level 1 setpoint. The impact of using ATRIUM-10 fuel at Browns Ferry Units 2 and 3 on this event is addressed in Section 9.1.3. The results of the evaluation indicate that the water level does not drop below the TAF. Results also show that the RCIC system meets the operational requirement that it maintain the water level above the Level 1 setpoint. The plant response for transition cores will be similar. Therefore, these conclusions are also applicable for transition cores of GE14 and ATRIUM-10 fuel.

[

] Other RCIC system parameters such as pressure and NPSH requirements are not dependent on fuel design.

3.9 *Residual Heat Removal System*

The primary design parameters for the residual heat removal (RHR) system are the decay heat in the core and amount of reactor heat discharged into the containment during a LOCA. [

] Reactor power is independent of fuel design and use of ATRIUM-10 fuel will have a negligible impact on the vessel water inventory.

The RHR system is designed to operate in several different modes, including assisting the fuel pool cooling and cleanup (FPCC) system in removing heat from the fuel pool to support off-loading of the entire core. As discussed in Section 6.3, the fuel pool heat load does not exceed the combined heat removal capacities of the RHR and FPCC systems.

The use of ATRIUM-10 fuel has no impact on the other RHR system modes of operation.

3.10 *Reactor Water Cleanup System*

The reactor water cleanup (RWCU) system operation is not dependent on fuel design.

3.11 *Balance-of-Plant Piping Evaluation*

The balance-of-plant piping systems evaluation is not dependent on fuel design.

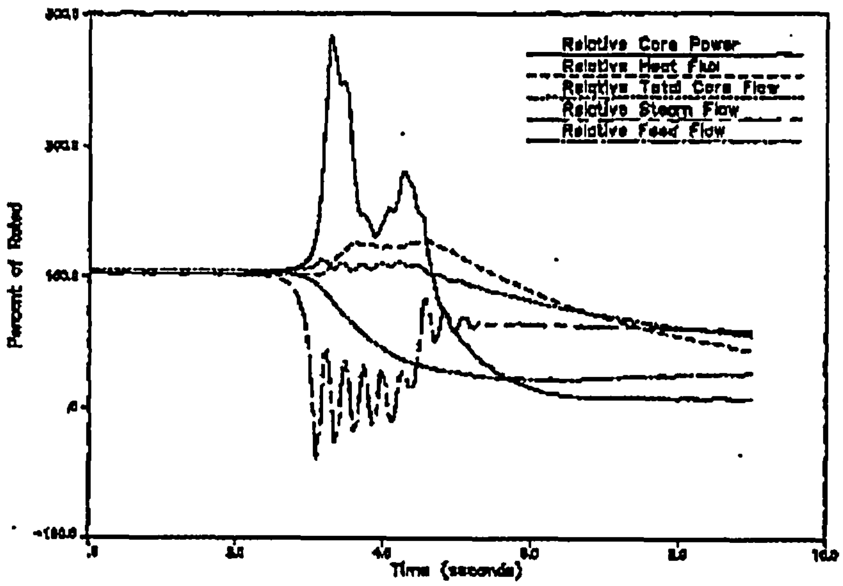

Figure 3.1 MSIV Closure With Flux Scram at 102%P/105%F – Power, Heat Flux, and Flows

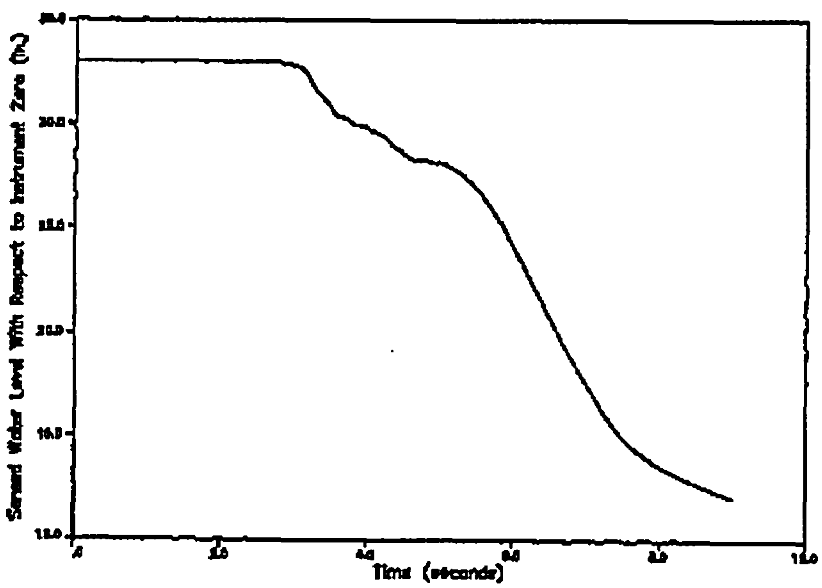

Figure 3.2 MSIV Closure With Flux Scram at 102%P/105%F – Downcomer Water Level

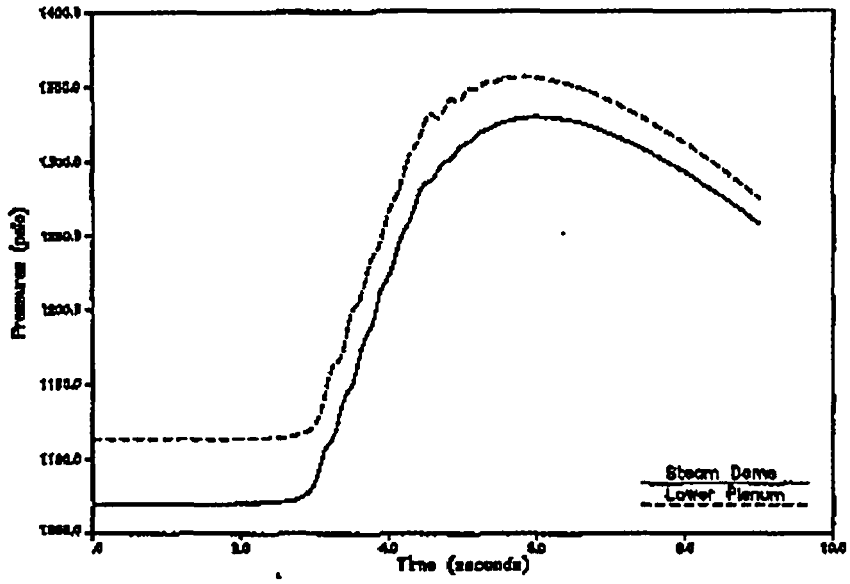

Figure 3.3 MSIV Closure With Flux Scram at 102%P/105%F – Pressures

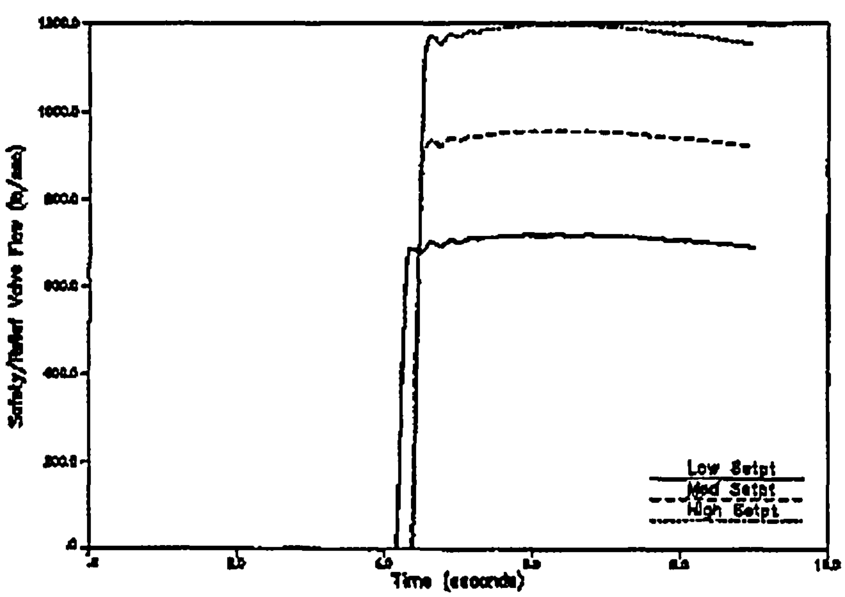

Figure 3.4 MSIV Closure With Flux Scram at 102%P/105%F – MSRV Flows

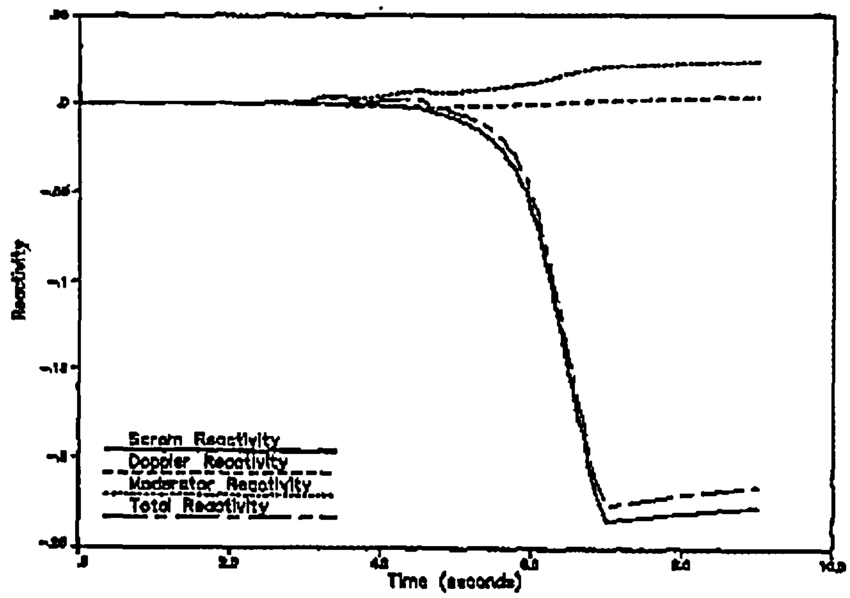

Figure 3.5 MSIV Closure With Flux Scram at 102%P/105%F – Reactivities

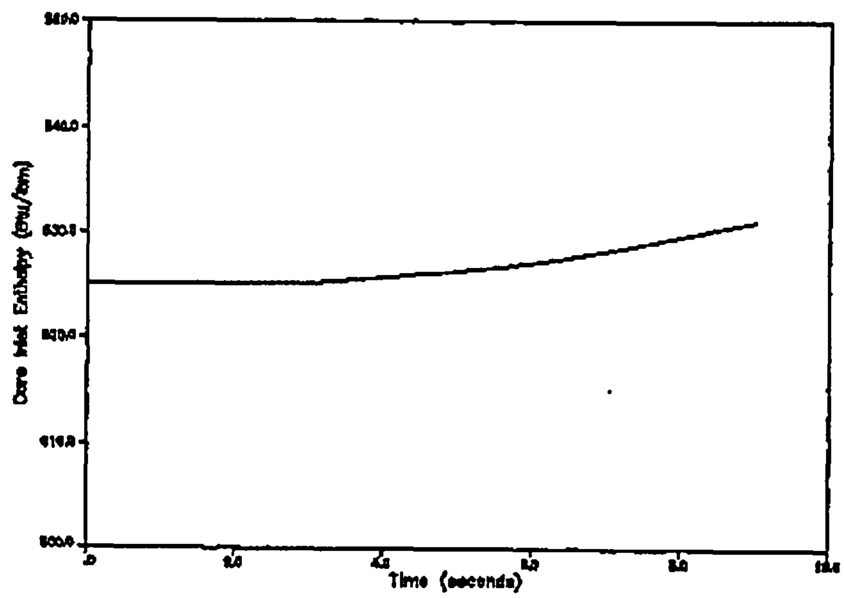

Figure 3.6 MSIV Closure With Flux Scram at 102%P/105%F – Core Inlet Enthalpy

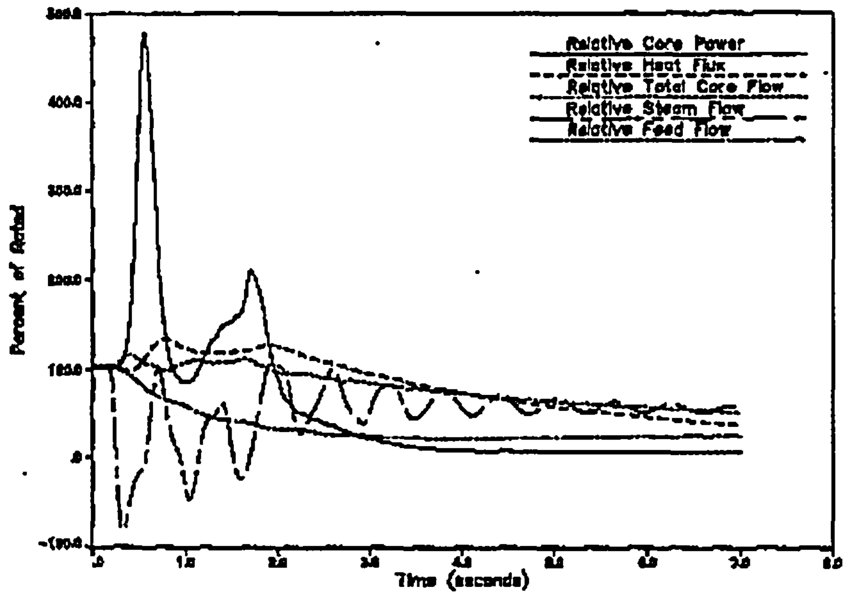

Figure 3.7 Turbine Trip With Bypass Failure and Flux Scram at 102%P/105%F – Power, Heat Flux, and Flows

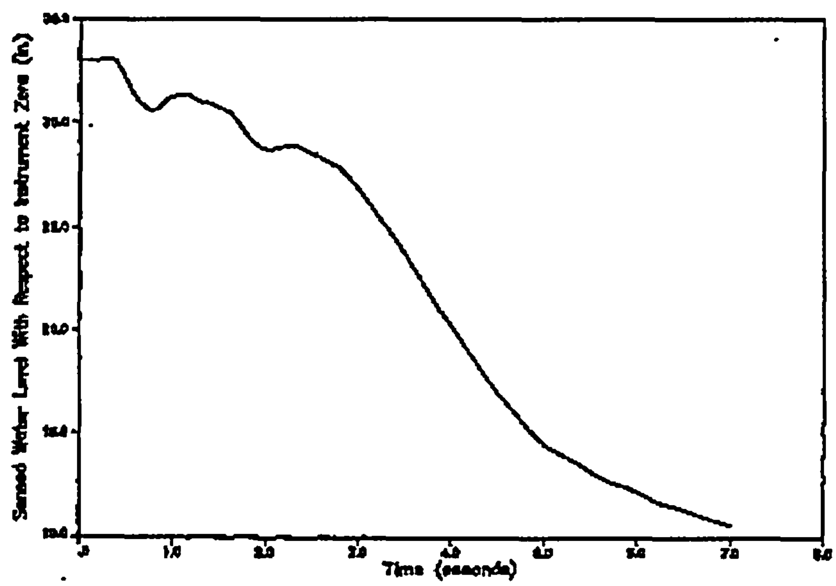

Figure 3.8 Turbine Trip With Bypass Failure and Flux Scram at 102%P/105%F – Downcomer Water Level

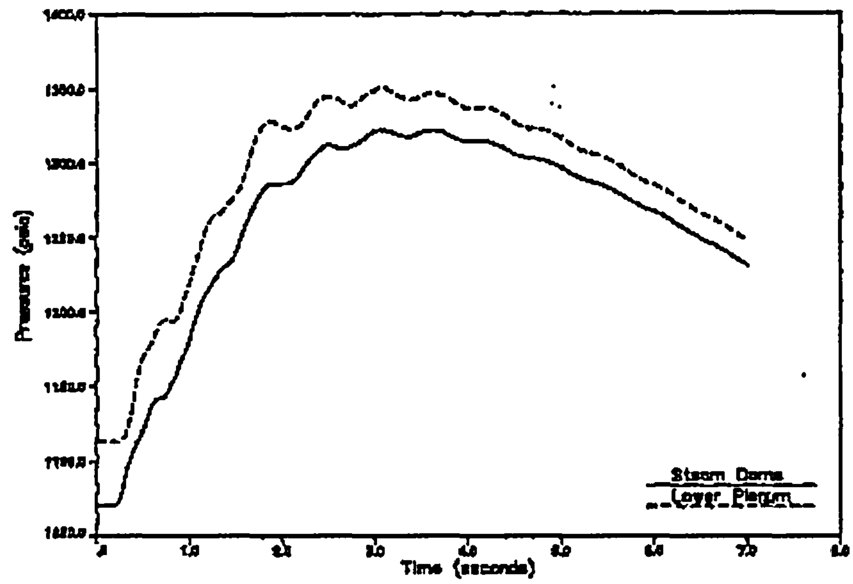

Figure 3.9 Turbine Trip With Bypass Failure and Flux Scram at 102%P/105%F – Pressures

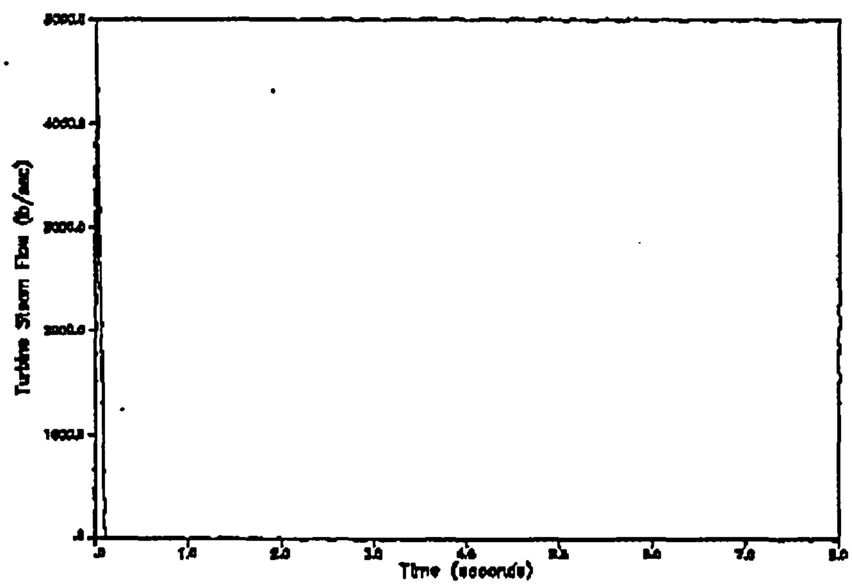

Figure 3.10 Turbine Trip With Bypass Failure and Flux Scram at 102%P/105%F – Turbine Steam Flow

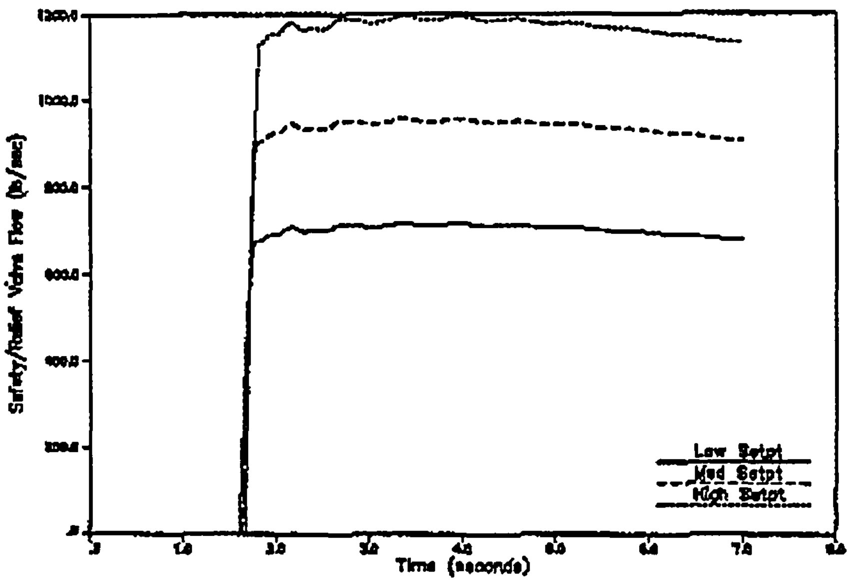

Figure 3.11 Turbine Trip With Bypass Failure and Flux Scram at 102%P/105%F – MSRV Flows

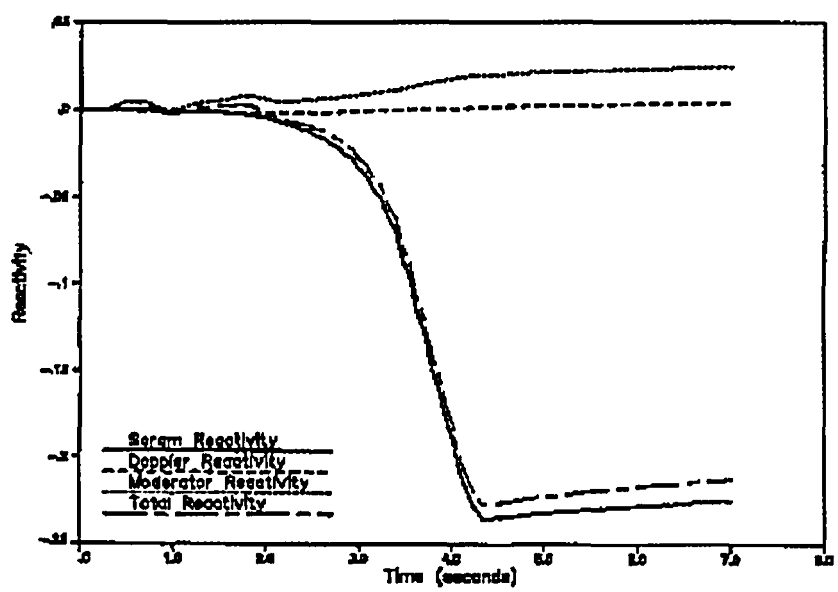

Figure 3.12 Turbine Trip With Bypass Failure and Flux Scram at 102%P/105%F – Reactivities

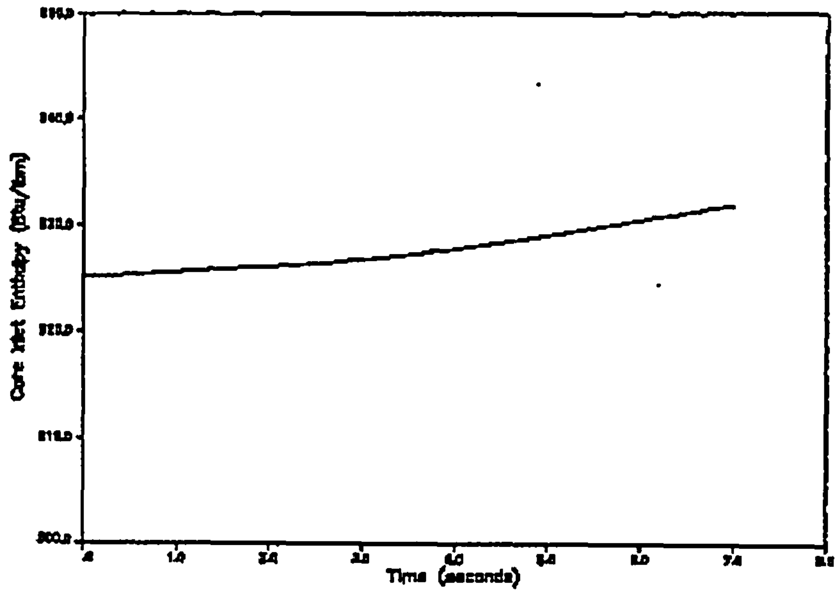

Figure 3.13 Turbine Trip With Bypass Failure and Flux Scram at 102%P/105%F – Core Inlet Enthalpy

4.0 Engineered Safety Features

4.1 *Containment System Performance*

4.1.1 Containment Pressure and Temperature Response

Short-term and long-term containment response for EPU operation is evaluated in Reference 1. The short-term analysis is directed at evaluating the drywell response during the initial blowdown phase of a LOCA, which is governed by the blowdown flow rate. The mass and energy flow out of the reactor vessel during the early part of a LOCA is dependent on the reactor operating condition (pressure, fluid temperature, core flow, and thermal power). [

]

[

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4.1.1.1 Long-Term Suppression Pool Temperature Response

[

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4.1.1.2 Short-Term Gas Temperature Response

Analyses of the superheated gas temperature reached during the steam blowdown phase of a LOCA form the basis of the short-term gas temperature limit. [

]

4.1.1.3 Short-Term Containment Pressure Response

The short-term containment response covers the blowdown phase of the LOCA and the subsequent impact on the drywell and wetwell pressures. [

]

4.1.2 Containment Dynamic Loads

4.1.2.1 Loss-of-Coolant Accident Loads

The short-term LOCA analyses form the basis of the Reference 1 LOCA containment dynamic loads analysis. [

]

4.1.2.2 Main Steam Relief Valve Loads

The MSRV loads are not dependent on fuel design.

4.1.2.3 Subcompartment Pressurization

The analysis of subcompartment pressurization is not dependent on fuel design.

4.1.3 Containment Isolation

The system designs for containment isolation are not dependent on fuel design.

4.1.4 Generic Letter 89-10 Program

The motor operated valve evaluations performed to assess the functional requirements of generic letter (GL) 89-10 are not dependent on fuel design.

4.1.5 Generic Letter 89-16

[

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4.1.6 Generic Letter 95-07

The evaluation for issues associated with GL 95-07 is not dependent on fuel design.

4.1.7 Generic Letter 96-06

The EPU evaluation for issues relating to GL 96-06 was performed using the containment LOCA response results. [

]

4.2 *Emergency Core Cooling Systems*

Each component of the ECCS is discussed in the following subsections, and the limiting LOCA-ECCS results are summarized in Section 4.3.

4.2.1 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. Operation of the HPCI system is not dependent on fuel design. The evaluation presented in Section 4.3 demonstrates that the HPCI system, in conjunction with the other ECCS components, is adequate to meet the 10 CFR 50.46 criteria.

4.2.2 Low-Pressure Coolant Injection

The low-pressure coolant injection (LPCI) mode of the RHR system provides a source of cooling water during a LOCA. Operation of the LPCI system is not dependent on fuel design. The evaluation presented in Section 4.3 demonstrates that the LPCI system, in conjunction with the other ECCS components, is adequate to meet the 10 CFR 50.46 acceptance criteria.

4.2.3 Core Spray System

The low-pressure core spray (LPCS) system provides a source of cooling water during a LOCA. Operation of the LPCS system is not dependent on fuel design. The evaluation presented in Section 4.3 demonstrates that the LPCS system, in conjunction with the other ECCS components, is adequate to meet the 10 CFR 50.46 criteria.

4.2.4 Automatic Depressurization System

The ADS reduces reactor pressure during a small break LOCA. The ability to initiate and operate the ADS is not dependent on fuel design. The evaluation presented in Section 4.3 demonstrates that the ADS, in conjunction with the other ECCS components, is adequate to meet the 10 CFR 50.46 criteria.

4.2.5 ECCS Net Positive Suction Head

The limiting net positive suction head (NPSH) conditions for the ECCS are dependent on the total pump flow rates, debris loading on the suction strainers, and suppression pool temperature. While the pump flow rates and design loading are not dependent on fuel design, the suppression pool temperature is dependent on the decay heat. [

]

4.3 *Emergency Core Cooling System Performance*

The Browns Ferry ECCS performance evaluation for ATRIUM-10 fuel is summarized in this section. A Browns Ferry break spectrum analysis for ATRIUM-10 fuel was performed using FANP methodology. Based on the results of the break spectrum analysis, the limiting break characteristics for ATRIUM-10 fuel at Browns Ferry are as follows:

- Break size/geometry – 0.5 ft^2 split break.
- Break location – recirculation discharge line.
- Single failure – DC power failure (battery failure).
- Axial power shape – mid-peaked.

For the limiting break, ADS and LPCS are the only emergency core cooling subsystems available. The analyses support operation at EPU power (3952 MWt) over a range of core flow rates including the ICF and MELLLA regions (105% to 99% of rated core flow). A multiplier of 1.02 is applied to the reactor power in the analyses, as discussed in Section 1.2.1.

The licensing basis peak clad temperature (PCT) for ATRIUM-10 fuel at EPU RTP is 1990°F, which is well within the Appendix K 2200°F limit. The maximum local cladding oxidation is 1.79% (< 17%), and the total hydrogen generation is < 1.0% of the total possible. Table 4.1 presents the limiting LOCA analysis results for ATRIUM-10 fuel. The MAPLHGR limit analyses show that the limiting exposure for Browns Ferry ATRIUM-10 fuel is beginning of life (BOL).

Calculations were also performed to support single-loop operation (SLO) with ATRIUM-10 fuel with a multiplier applied to reduce the MAPLHGR limit. The SLO analyses show that the PCT is well below the Appendix K limit of 2200°F.

4.4 *Main Control Room Atmosphere Control System*

The Reference 1 discussion remains applicable for ATRIUM-10 fuel since the TVA alternative source term (AST) submittal (Reference 4) includes ATRIUM-10 fuel characteristics.

4.5 *Standby Gas Treatment System*

The Reference 1 discussion remains applicable for ATRIUM-10 fuel since the TVA AST submittal (Reference 4) includes ATRIUM-10 fuel characteristics.

4.6 *Main Steam Isolation Valve Leakage Control System*

Browns Ferry does not use a main steam isolation valve leakage control system.

4.7 *Post-LOCA Combustible Gas Control*

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. Reference 1 presents the results of the evaluation performed to support EPU. Cladding mass affects the amount of metal-water reaction and consequently the nitrogen injection requirements. [

] The results of the evaluation for ATRIUM-10 fuel show that the required CAD system start time for the design basis case is 30 hours, slightly less than the 32-hour start time reported in Reference 1. This shorter start time does not impact the ability of the operators to appropriately respond to a LOCA. Figure 4.1 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 4.2. The drywell pressure response assuming no venting is presented in Figure 4.3 and the CAD system nitrogen requirements are presented in Figure 4.4.

The Technical Specifications require that 2500 gallons of liquid nitrogen (191,000 scf) be stored in each of two tanks to meet the CAD system inerting requirements. As discussed in Reference 1, since additional liquid nitrogen can be delivered in one day or less, demonstrating that the TS storage requirement can provide the site requirements for four days is sufficient. The results of the ATRIUM-10 CAD system evaluation indicate that a volume of 109,000 scf of nitrogen in needed. While this is greater than the required four-day supply of 104,834 scf reported in Reference 1, it is less than the available 191,000 scf supply required by the Technical Specifications.

Table 4.1 Limiting LOCA Results for ATRIUM-10 EPU Operation

Parameter	Value
Peak cladding temperature, °F	1990
Local cladding oxidation, max %	1.79
Total hydrogen generated, % of total hydrogen possible	< 1.0

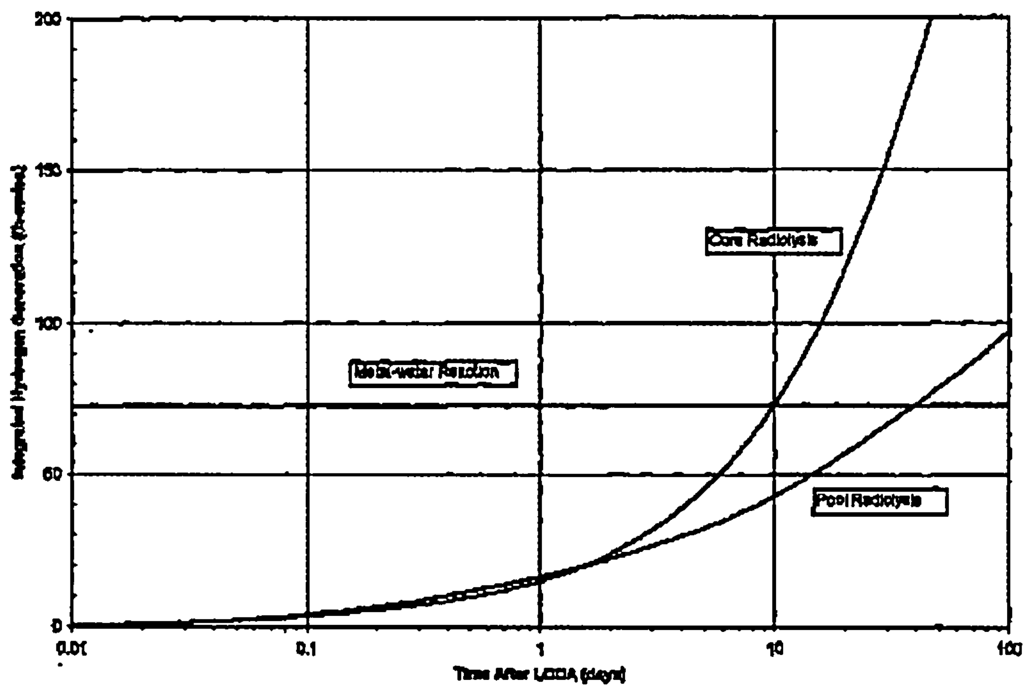

Figure 4.1 Browns Ferry Time-Integrated Containment Hydrogen Generation

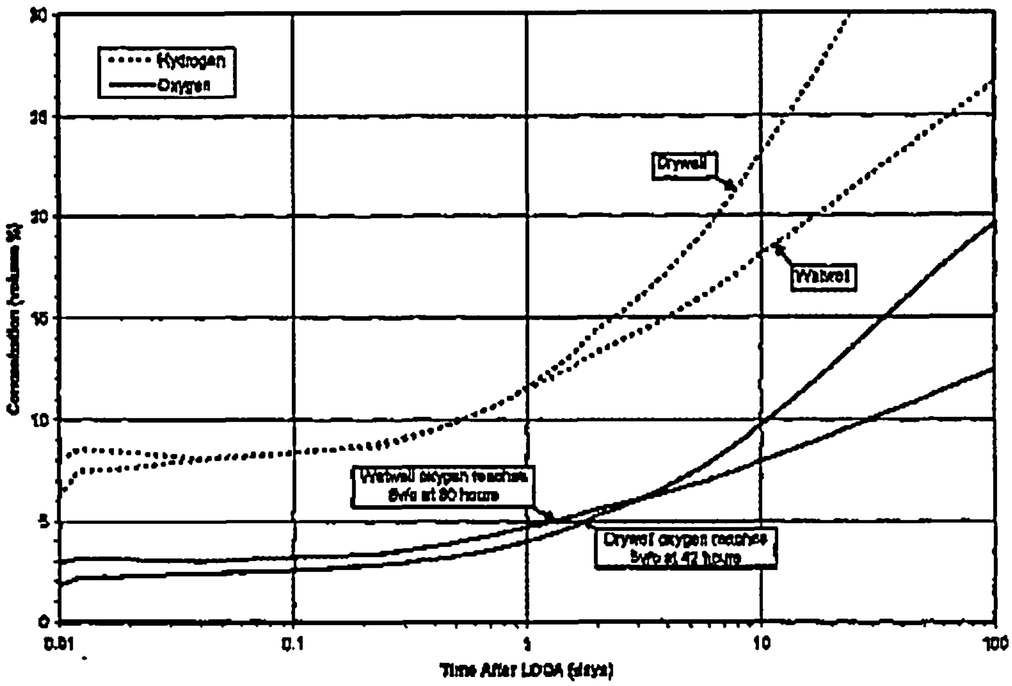

Figure 4.2 Browns Ferry Uncontrolled H_2 and O_2 Concentrations in Drywell and Wetwell

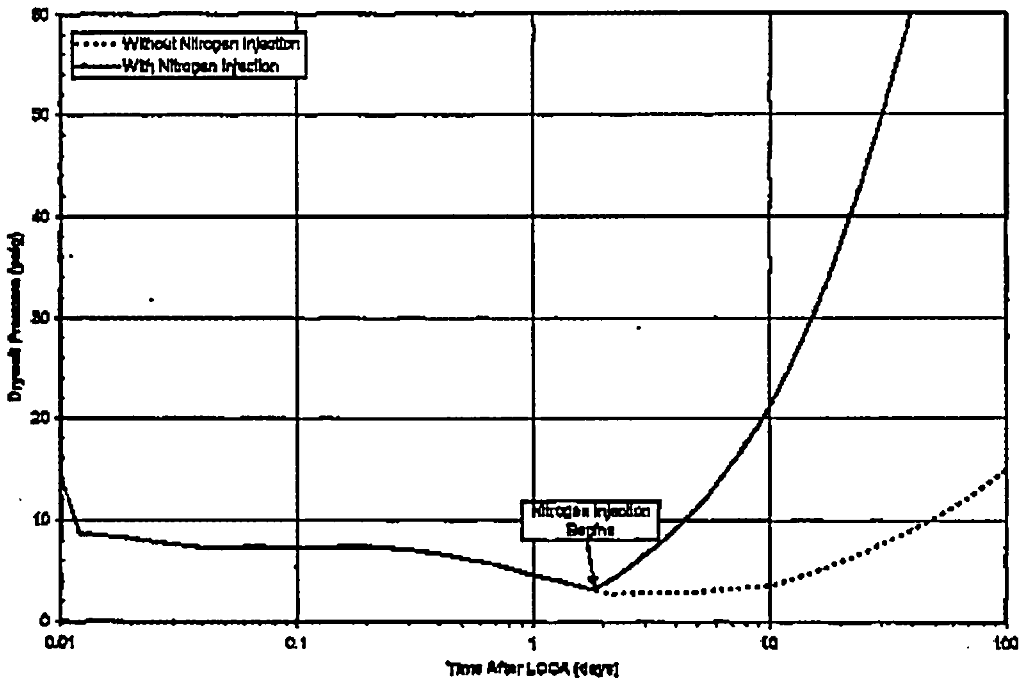

Figure 4.3 Browns Ferry Drywell Pressure Response to CAD Operation Without Venting

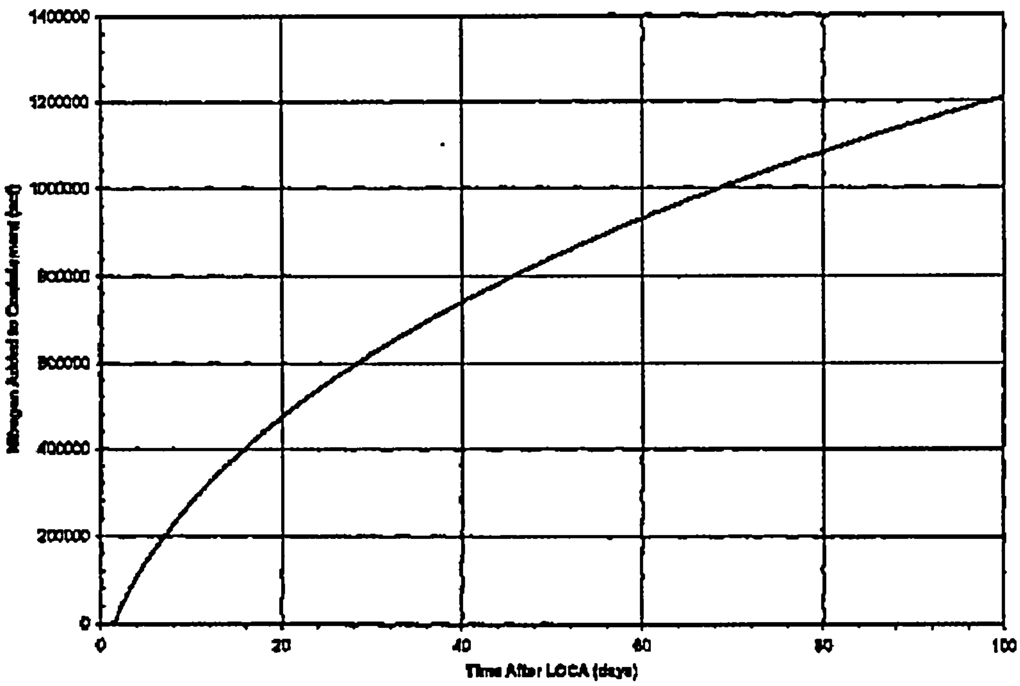

**Figure 4.4 Browns Ferry CAD System
Nitrogen Volume Requirement**

5.0 Instrumentation and Control

5.1 *NSSS Monitoring and Control Systems*

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

5.1.1 Control Systems Evaluation

Increases in core thermal power affect some instrument setpoints as discussed in Section 5.3. However, these increases are not dependent on fuel design.

5.1.2 Neutron Monitoring System

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. Operation with ATRIUM-10 fuel has no effect on the intermediate range monitors (IRMs) and source range monitors (SRMs).

5.1.3 Rod Worth Minimizer

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

5.2 *BOP Monitoring and Control Systems*

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

5.2.1 Pressure Control System

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

5.2.2 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 discussed in Section 9.1 (Feedwater Controller Failure Maximum Demand) and Section 9.1.3 (loss of feedwater flow event).

5.2.3 Leak Detection System

The leak detection system is not dependent on fuel design.

5.3 *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 in Reference 1 for EPU conditions remain applicable for ATRIUM-10 fuel as discussed below.

5.3.1 High-Pressure Scram

The high-pressure scram analytical limit is not dependent on fuel design.

5.3.2 High-Pressure ATWS Recirculation Pump Trip

The anticipated transient without scram recirculation pump trip (ATWS-RPT) is a significant factor in the ATWS peak vessel pressure events. The ATWS-RPT is initiated by high vessel dome pressure and/or low reactor water level. For the ATWS events, the low reactor water level ATWS-RPT does not affect the limiting results. The low reactor water level setpoint is not affected by EPU. The current ATWS-RPT high-pressure setpoint was included in the ATWS evaluation discussed in Section 9.3.1. Since this evaluation demonstrates that the peak vessel pressure remains below the applicable limit, the current ATWS-RPT high-pressure setpoint is acceptable for EPU operation with ATRIUM-10 fuel.

5.3.3 Main Steam Relief Valve

The MSRV setpoints are not dependent on fuel design. The current values were used in the overpressure protection and transient analyses discussed in Sections 3.2 and 9.1.

5.3.4 Main Steam High Flow Isolation

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

5.3.5 Neutron Monitoring System

The average power range monitor (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. Thermal-hydraulic analyses have indicated that the pressure drop through an ATRIUM-10 assembly is slightly lower than the pressure drop through a GE14 assembly. [

] No changes to the APRM flow-biased rod block and scram analytical limits are needed.

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 control rod withdrawal error (CRWE) analyses. A CRWE analysis for an equilibrium cycle with ATRIUM-10 fuel at EPU conditions was performed, and the results show that CLTP (3458 MWt) setpoints remain applicable.

5.3.6 Main Steam Line High Radiation Scram

Browns Ferry does not have a main steam line radiation level scram.

5.3.7 Low Steam Line Pressure MSIV Closure (RUN Mode)

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

5.3.8 Reactor Water Level Instruments

The reactor water level instruments are not dependent on fuel design.

5.3.9 Main Steam Tunnel High Temperature Isolation

The main steam tunnel high temperature isolation setpoint is not dependent on fuel design.

5.3.10 Low Condenser Vacuum

Browns Ferry does not have a low condenser vacuum MSIV isolation or scram trip.

5.3.11 TSV Closure and TCV Fast Closure Scram Bypass

The turbine stop valve (TSV) closure and turbine control valve (TCV) fast closure scram bypass is not dependent on fuel design.

5.3.12 Rod Worth Minimizer

The rod worth minimizer low-power setpoint is not dependent on fuel design.

5.3.13 Pressure Regulator

The pressure regulator setpoints are not dependent on fuel design.

5.3.14 Feedwater Flow Setpoint for Recirculation Cavitation Protection

The feedwater flow setpoint for recirculation cavitation protection is not dependent on fuel design.

5.3.15 RCIC Steam Line High-Flow Isolation

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

5.3.16 HPCI Steam Line High-Flow Isolation

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

6.0 Electrical Power and Auxiliary Systems

6.1 *AC Power*

6.1.1 Off-Site Power System

The off-site electrical equipment and grid stability are not dependent on fuel design.

6.1.2 On-Site Power Distribution System

The electrical load demands are not dependent on fuel design.

6.2 *DC Power*

The DC power distribution system is not dependent on fuel design.

6.3 *Fuel Pool*

The fuel pool systems are made up of several components and systems including: storage pools, fuel racks, the fuel pool cooling and cleanup (FPCC) system, and the auxiliary decay heat removal (ADHR) system. These components and systems are used to keep the temperature of the fuel pool below a specified limit by removing the decay heat from irradiated fuel stored in the pool.

6.3.1 Fuel Pool Cooling

The Browns Ferry spent fuel pool (SFP) bulk water temperature must be maintained below the licensing limit of 150°F. [

]

6.3.2 Crud Activity and Corrosion Products

[

]

6.3.3 Radiation Levels

[

] Any small change in exposure would be mitigated through the Browns Ferry ALARA program.

6.3.4 Fuel Racks

The fuel rack design temperature is not dependent on fuel design.

6.4 *Water Systems*

The performance of the safety and non-safety service water systems are not dependent on fuel design.

6.5 *Standby Liquid Control System*

The standby liquid control system (SLCS) is designed to render and maintain the core to a subcritical condition in the event that insufficient control rod insertion results when a scram signal is received. The minimum required boron quantity for EPU was set in Reference 1 to ensure an adequate reactor boron concentration. FANP has evaluated the ability of the SLCS to render and maintain an ATRIUM-10 equilibrium core, initially at EPU conditions, to a cold subcritical shutdown condition. This evaluation shows that an equilibrium core designed for EPU conditions and composed entirely of ATRIUM-10 fuel will be subcritical with the Reference 1 boron concentration.

[

] This is confirmed with cycle-specific analyses reflecting the actual core configuration (including transition cores) as part of the reload licensing process.

During the first pressure peak of the ATWS vessel pressure evaluation, the lower plenum pressure increases above the opening setpoint of the SLCS discharge relief valves. After the initial pressure increase, the lower plenum pressure drops below the closure setpoint of the SLCS discharge relief valves and remains below the opening setpoint for the remainder of the event. Therefore, the SLCS pump discharge relief valves would close, and remain closed, if the SLCS is started before the lower plenum pressure recovers from the first transient peak. As a result, the reactor pressure response during an ATWS with ATRIUM-10 fuel will not impact the ability of the SLCS to perform its function.

6.6 *Power-Dependent HVAC*

The heating ventilation and air conditioning (HVAC) systems are not dependent on fuel design.

6.7 *Fire Protection*

Section 6.7 of Reference 1 presented an evaluation of the effect of EPU on the fire protection program, fire suppression and detection systems, and reactor and containment system responses to postulated 10 CFR 50 Appendix R fire events. There are no fuel-related parameters used in the evaluation of the fire suppression and detection systems. The containment response to an Appendix R fire event is determined by the rate of steam flow through the MSRVs and the ADS during the event. This flow is dependent on the initial reactor operating power and pressure, but not on fuel related parameters. Therefore, the containment response is not fuel related. The response of the reactor core to an Appendix R event is fuel related. Section 6.7.1 provides the Appendix R analysis results for ATRIUM-10 fuel.

6.7.1 10 CFR 50 Appendix R Fire Event

The limiting Appendix R fire event was analyzed assuming operation with ATRIUM-10 fuel at EPU conditions. [

]

The postulated Appendix R fire event using the minimum safe shutdown system (SSDS) was analyzed for the three cases described below:

Case 1. No spurious operation of plant equipment occurs and the operator initiates three MSRVs 25 minutes into the event.

Case 2. One MSRV opens immediately due to a spurious signal generated as a result of the fire. The MSRV is closed 10 minutes into the event by operator action. The operator initiates three MSRVs 20 minutes into the event.

Case 3. One MSRV opens immediately and remains open. The operator initiates three MSRVs 20 minutes into the event.

The analyses assume one LPCI pump is available to mitigate the Appendix R event. LPCI flow is modeled to begin at 319.5 psig. The recirculation line isolation valve is assumed to always remain open, which reduces the LPCI flow to the core.

The results of the Appendix R analysis for ATRIUM-10 fuel at EPU conditions demonstrate that fuel cladding integrity and reactor vessel integrity are maintained and that sufficient time is available for the reactor operator to accomplish the necessary actions. For ATRIUM-10 fuel, Case 1 is the limiting case. The PCT is 1238°F and the peak reactor pressure is 1224 psia.

Spurious operation of the HPCI system was also evaluated. The purpose of the evaluation was to determine the time the water level would reach the elevation of the main steam lines if operation of the HPCI system is assumed to start at the beginning of an Appendix R event. The analysis results show that the time the reactor water level would reach the main steam line elevation is greater than 5 minutes. A change to the procedures is necessary to ensure that HPCI system isolation occurs within 5 minutes.

[

]

6.8 *Systems Not Impacted by Extended Power Uprate*

6.8.1 Systems With No Impact

Operation with ATRIUM-10 fuel will not impact any of the systems listed in Table 6-6 of Reference 1.

6.8.2 Systems With Insignificant Impact

Operation with ATRIUM-10 fuel will not impact the systems listed in Table 6-7 of Reference 1.

7.0 Power Conversion Systems

7.1 *Turbine-Generator*

The turbine-generator design criteria are not dependent on fuel design.

7.2 *Condenser and Steam Jet Air Ejectors*

The condenser and steam jet air ejectors are not dependent on fuel design.

7.3 *Turbine Steam Bypass*

The steam bypass capacity is not dependent on fuel design.

7.4 *Feedwater and Condensate Systems*

7.4.1 Normal Operation

Normal operation of the feedwater and condensate systems is independent of the fuel design.

7.4.2 Transient Operation

The operation of the feedwater and condensate systems with all system pumps available is not dependent on fuel design. The system capacity after a single feedwater pump trip was analyzed with ATRIUM-10 fuel to show that the system maintains the capability to supply adequate flow (Section 9.1.3).

7.4.3 Condensate Demineralizers

The condensate filter demineralizers are not dependent on fuel design.

8.0 Radwaste and Radiation Sources

8.1 *Liquid and Solid Waste Management*

Liquid and solid waste management at EPU conditions are not impacted by fuel design.

8.2 *Gaseous Waste Management*

Gaseous waste management at EPU conditions is not impacted by fuel design.

8.3 *Radiation Sources in the Reactor Core*

8.3.1 Normal Operation

During normal operation, the fission rate directly impacts the radiation sources. The fission rate is directly related to the core power level and thus is not impacted by fuel design.

8.3.2 Normal Post-Operation

The accumulated fission products are the primary source of post-operation radiation sources. The fission product inventories are roughly proportional to core thermal power and vary little with fuel design. The Reference 1 evaluations included appropriate source terms applicable to ATRIUM-10 fuel. For post-operation accidents, TVA is implementing alternate source term (AST) methodology and submitted the AST implementation for NRC review. For the AST analysis, source terms were determined based on inputs that included ATRIUM-10 fuel operated at EPU conditions.

8.4 *Radiation Sources in Reactor Coolant*

The evaluations from Reference 1 remain applicable to ATRIUM-10 fuel.

8.5 *Radiation Levels*

8.5.1 Normal Operation

[

]

8.5.2 Normal Post-Operation

The Reference 1 evaluations included the consideration of, and are applicable to, ATRIUM-10 fuel.

8.5.3 Post-Accident

The 100-day post-accident radiation doses evaluated in Reference 1 were calculated with consideration of ATRIUM-10 fuel. Plant specific analyses for NUREG-0737, Item II.B.2, post-accident mission doses evaluated in Reference 1 were also calculated with consideration of ATRIUM-10 fuel. Therefore, the Reference 1 dose calculations are applicable to ATRIUM-10 fuel.

8.6 Normal Operation Off-Site Doses

The Reference 1 evaluations remain applicable to ATRIUM-10 fuel.

9.0 Reactor Safety Performance Evaluations

9.1 *Reactor Transients*

The UFSAR indicates that abnormal operational transients, also referred to as AOOs, are the result of single equipment failures or single operator errors that can be reasonably expected to occur during any mode of operation. The events are categorized based on the potential initiating cause of the threat to the fuel and reactor system. Analyses for the potentially limiting transient events have been performed to assess the impact of operation with ATRIUM-10 fuel at EPU conditions. The analyses are based on an equilibrium core of ATRIUM-10 fuel operating at full EPU thermal power and are discussed below. The transient analysis results show that the ATRIUM-10 fuel has the capability to meet all the transient analysis licensing criteria at EPU conditions.

Table 9.1 presents key reactor operating parameters used in the transient analyses. Pressurization transients (LRNB, TTNB, FWCF, MSIVC) were analyzed using the approved transient analysis methodology documented in References 6–8. The fast recirculation flow increase and inadvertent HPCI actuation events were also analyzed using the transient methodology. The loss of feedwater flow event was analyzed using COTRANSA2, the system transient analysis code (Reference 6). The other nonpressurization events used the MICROBURN-B2 methodology (CRWE, LFWH) or the XCOBRA methodology (slow recirculation flow increase). The SLMCPR of 1.08 (Section 2.2.1) presented in Table 9.1 was used to calculate OLMCPR values. For all events, one lowest setpoint MSRV was assumed to be out-of-service, and a +3% MSRV setpoint tolerance was applied.

The results of transient analyses at full EPU RTP with Technical Specification scram speeds (TSSS) are presented in Table 9.2. Figures 9.1–9.28 present the response of several system parameters for the limiting pressurization transient analyses.

The application of power- and flow-dependent limits at off-rated power and flow conditions is not affected by a change in fuel design. The actual limits are established using cycle-specific analysis results to ensure that adequate protection is provided.

The threshold reduction to 23% RTP for thermal limits monitoring and some surveillance requirements discussed in Reference 1 is not dependent on fuel design.

[

] Cycle-specific analyses reflecting the actual core configuration (including transition cores) will be performed as part of the reload licensing analysis. Core and fuel operating limits will be revised if necessary to ensure that all licensing criteria are satisfied.

9.1.1 Fuel Thermal Margin Events

Operation with ATRIUM-10 fuel does not have a significant impact on the relative nature of the limiting AOO events. While the fuel design can impact the magnitude of the event result, the potentially limiting events remain the same. This is consistent with transition reload core analyses for different power level and power density plants. The potentially limiting events are consistent with those identified in Reference 1. The rated power OLMCPR necessary to support operation of the reference ATRIUM-10 equilibrium core with TSSS is 1.38.

9.1.2 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.

9.1.3 Loss of Feedwater Flow Event

During a loss of feedwater flow (LOFW) event, the water level decreases due to the mismatch between steam flow and feedwater flow. Makeup water is needed to maintain adequate core cooling and keep the water level above the top of active fuel. An LOFW analysis was performed for a full core of ATRIUM-10 fuel at EPU RTP assuming that the RCIC system is the only system available to restore the reactor water level. Results of the LOFW analysis show that the minimum water level inside the shroud is 71 inches above the top of active fuel (TAF), thereby ensuring adequate cooling of the core.

In addition to the requirement that the reactor water level remain above TAF, an operational requirement is applied that the water level remains above the low-low-low water level setpoint (Level 1). This ensures that the ADS timer and MSIV closure trip are not activated. While this requirement is not a safety function, meeting it avoids unnecessary actuation of the safety systems. The LOFW analysis results show that the water level remains above the Level 1 setpoint.

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-10 fuel at EPU conditions show that no scram occurs since the minimum water level is 16 inches above the Level 3 setpoint.

9.2 *Design Basis Accidents*

Plant specific radiological dose consequence analyses have been performed for the DBAs utilizing AST in accordance with 10 CFR 50.67. These analyses were performed with consideration of, and are applicable to, ATRIUM-10 fuel at Browns Ferry Units 2 and 3.

9.3 *Special Events*

9.3.1 Anticipated Transient Without Scram

The anticipated transient without scram (ATWS) overpressure evaluation includes consideration of the most limiting RPV overpressure case. The analyses presented in Reference 1 were performed to verify three acceptance criteria:

- Peak vessel bottom pressure less than ASME Service Level C limit of 1500 psig.
- Peak suppression pool temperature less than 281°F (wetwell shell design temperature).
- Peak containment pressure less than 56 psig (drywell design pressure).

Analyses were performed for a full core of ATRIUM-10 fuel at EPU conditions to evaluate compliance with the ATWS peak vessel pressure criteria. [

] As a result, the containment pressure limit will not be violated with ATRIUM-10 fuel.

[

] Since there is no core uncovery associated with the ATWS event, these two criteria will be bounded by the LOCA PCT and local cladding oxidation.

The key inputs to the Browns Ferry ATWS overpressurization analysis are provided in Table 9.3. Results of the analysis are provided in Table 9.4. The results show that the ATWS overpressurization criteria are met for EPU conditions with one lowest setpoint MSRV out of service.

9.3.1.1 ATWS With Core Instability

The summary presented in Section 9.3.1.1 of Reference 1 indicates that for an unmitigated case, a small fraction of the core experiences locally high peak clad temperature (dryout) and some fuel damage cannot be precluded. For the mitigated case (reduction of reactor water level to reduce core inlet subcooling and direct injection of boron in the presence of power oscillations) extended dryout was not expected. The NRC has concluded based on the BWROG submittals and their own work (Technical Evaluation Report for Reference 15) that the presence of large amplitude power oscillations does not alter the consequences of an ATWS event significantly enough to warrant a change in the current ATWS Rule. Furthermore, even though a large uncertainty is associated with the calculated power oscillation amplitude, the conclusion would remain valid even if the amplitude was in error by an order of magnitude.

Operation with ATRIUM-10 fuel results in small changes to parameters important to determining the reactor stability. [

] Based on these observations, the impact of operation with ATRIUM-10 is considered small compared to the large uncertainties associated with the event analysis.

9.3.2 Station Blackout

The plant response and coping capabilities for a station blackout (SBO) were evaluated in Reference 1. [

]

Table 9.1 Browns Ferry Parameters Used for ATRIUM-10 EPU Transient Analyses

Rated thermal power, MWt	3952
Analysis power, % rated	100*
Analysis dome pressure, psia	1050
Analysis turbine pressure, psia	988
Rated vessel steam flow, Mlbm/hr	16.44
Analysis steam flow, % rated	100
Rated core flow, Mlbm/hr	102.5
Rated power core flow range, % rated	99–105
Analysis core flow, % rated	99–105
Normal feedwater temperature, °F	394.5
Steam bypass capacity, % rated steam flow	21.69
Reactor low-level water level 3 scram (in AVZ)	518
Safety limit MCPR	1.08
Number of MSRVs assumed in analysis Low bank Mid-bank High bank	 3† 4 5
MSRV setpoint, psig Low bank Mid-bank High bank	 1135 + 3% 1145 + 3% 1155 + 3%

* The only events analyzed at a different power level (102%) are the loss of feedwater and ASME overpressurization events.

† One lowest setpoint MSRV is assumed OOS.

**Table 9.2 Browns Ferry ATRIUM-10
Transient Analysis Results***

Event	Peak Neutron Flux, % of rated EPU	Peak Heat Flux, % of rated EPU	ΔCPR	MCPR Operating Limit
Generator load rejection with bypass failure	285	118	0.30	1.38
Turbine trip with bypass failure	298	119	0.30	1.38
Feedwater controller failure max demand	288	122	0.30	1.38
Pressure regulator downscale failure	†	†	†	†
Loss of feedwater heating	‡	‡	0.10	1.18
Inadvertent HPCI actuation	114	113	0.15	1.23
Rod withdrawal error	‡	‡	0.20§	1.28
Slow recirculation increase	NA	NA	NA	$MCPR_f$
Fast recirculation increase	203	91	0.12	1.20
Generator load rejection with bypass	264	115	0.26	1.34
MSIV closure – all valves	142	101	0.10	**
MSIV closure – 1 valve	117	107	0.09	**
Loss of feedwater flow††	102	102	NA	NA
Loss of 1 feedwater pump††	100	100	NA	NA

* All analyses performed with TSSS insertion times.
† Not calculated based on UFSAR 14.5.2.8.
‡ Peak neutron flux and peak heat flux are not reported for the slow transients.
§ With rod block monitor setpoint of 111%.
** Bounded by the generator load rejection with bypass failure.
†† These events were performed assuming a conservatively low core flow of 85% of rated.

Table 9.3 Key Inputs for ATRIUM-10 EPU ATWS Analysis

Reactor power, MWt	3952
Reactor dome pressure, psia	1050
MSRV capacity, Mlbm/hr	11.31
High-pressure ATWS-RPT, psig	1177
Number of MSRVs out-of-service (OOS)	1

Table 9.4 ATRIUM-10 EPU ATWS Overpressurization Analysis Results

Peak vessel bottom pressure, psig	1483

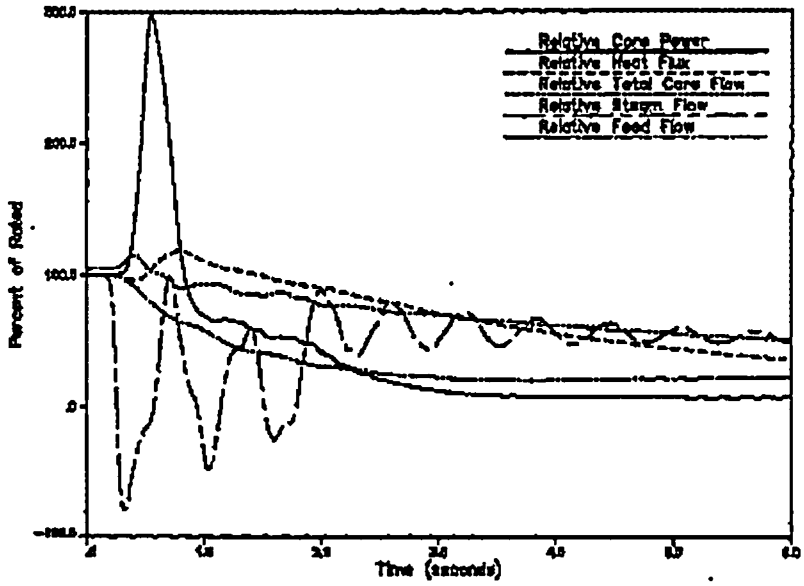

Figure 9.1 Turbine Trip With Bypass Failure at 100%P/105%F – Power, Heat Flux, and Flows

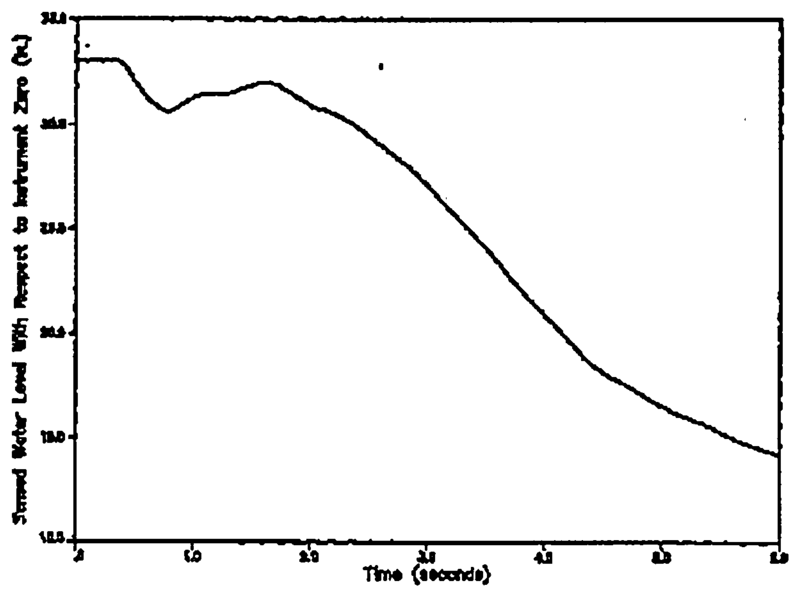

Figure 9.2 Turbine Trip With Bypass Failure at 100%P/105%F – Downcomer Water Level

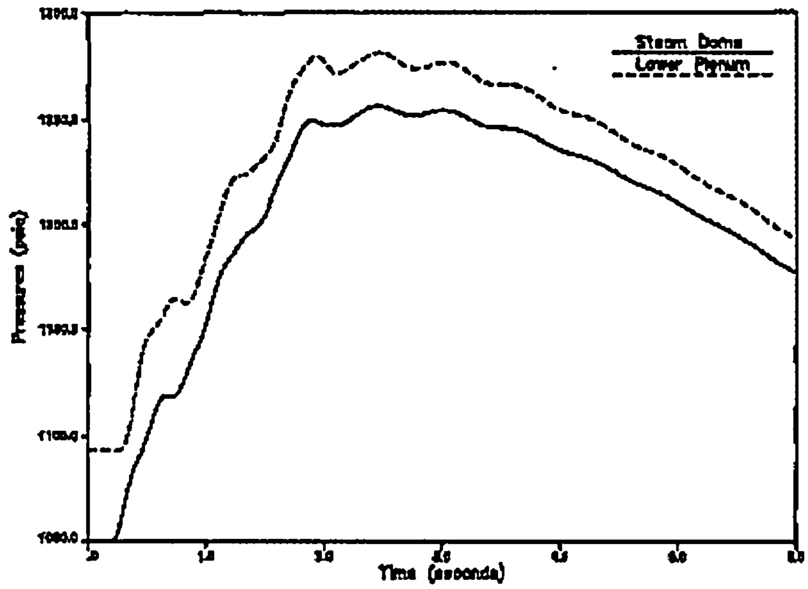

Figure 9.3 Turbine Trip With Bypass Failure at 100%P/105%F – Pressures

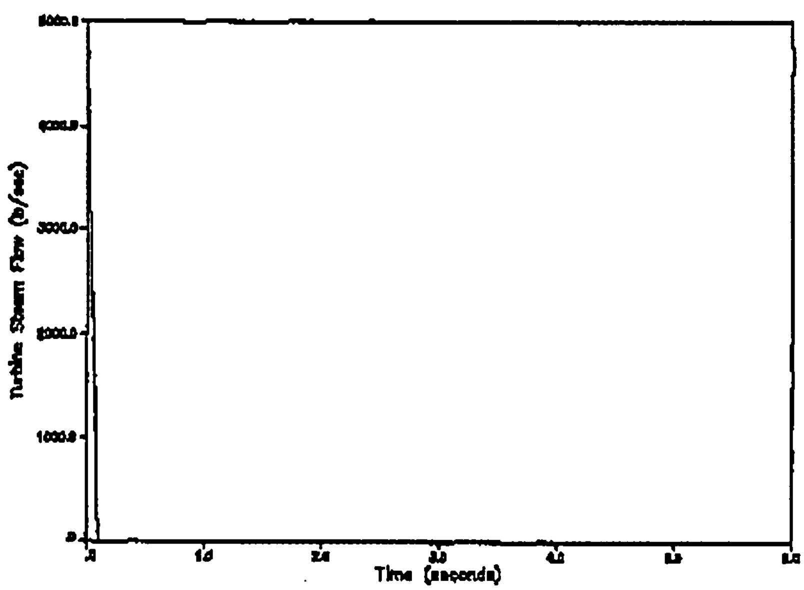

Figure 9.4 Turbine Trip With Bypass Failure at 100%P/105%F – Turbine Steam Flow

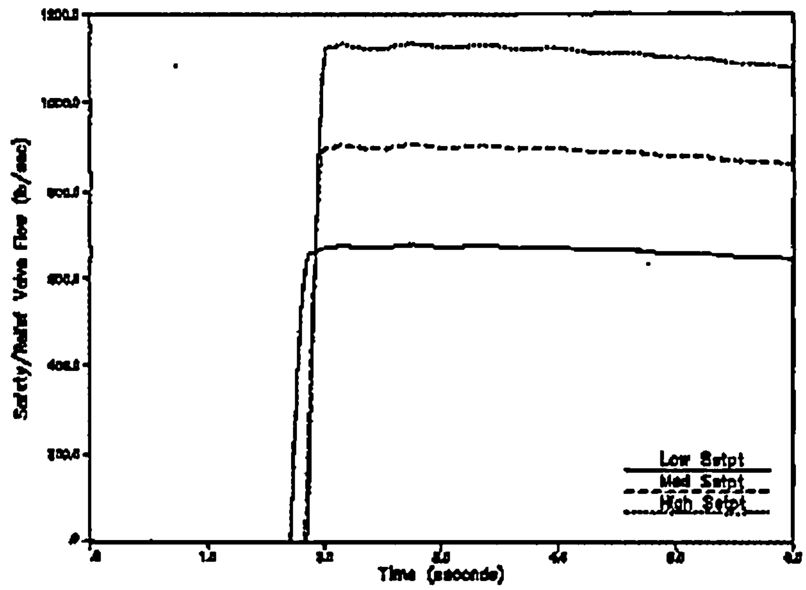

Figure 9.5 Turbine Trip With Bypass Failure at 100%P/105%F – MSRV Flows

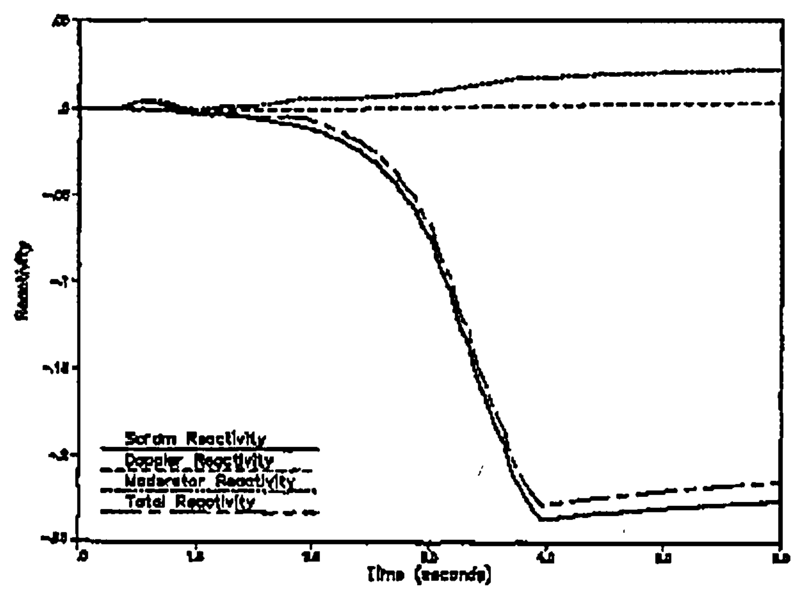

Figure 9.6 Turbine Trip With Bypass Failure at 100%P/105%F – Reactivities

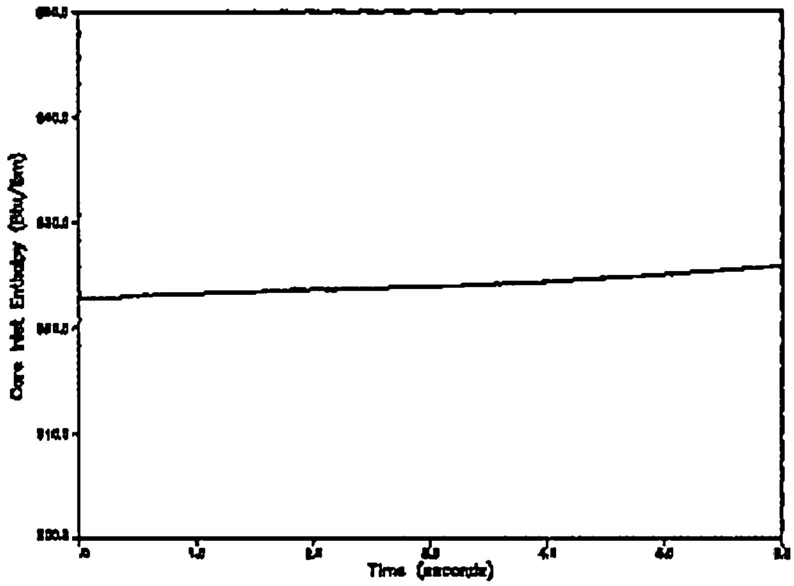

Figure 9.7 Turbine Trip With Bypass Failure at 100%P/105%F – Core Inlet Enthalpy

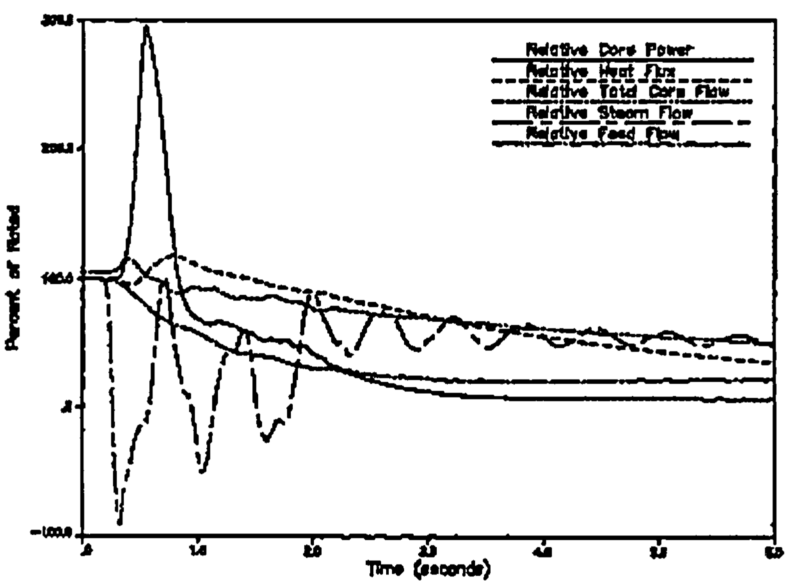

Figure 9.8 Generator Load Rejection With Bypass Failure at 100%P/105%F – Power, Heat Flux, and Flows

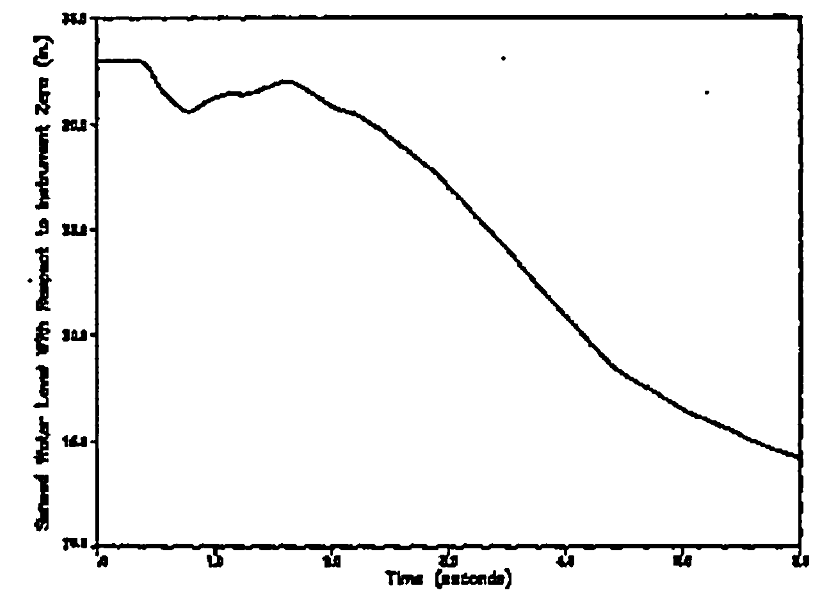

Figure 9.9 Generator Load Rejection With Bypass Failure at 100%P/105%F – Downcomer Water Level

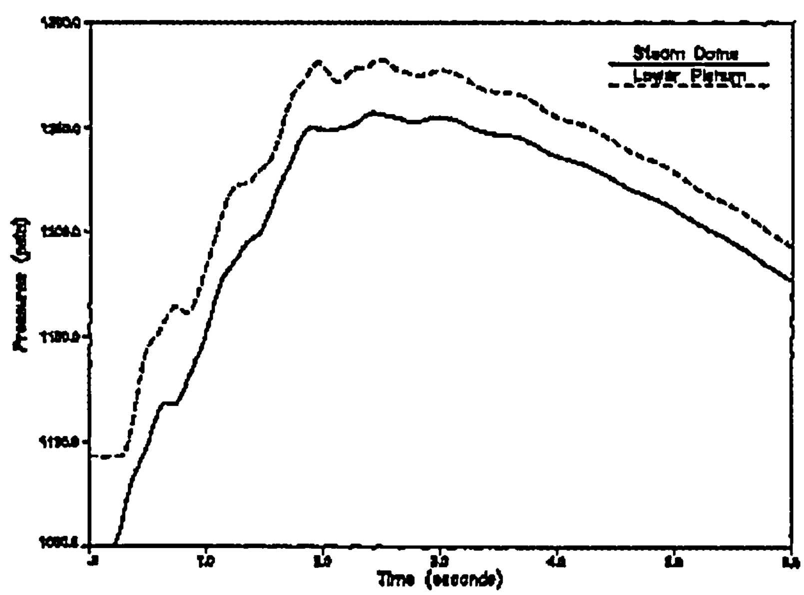

Figure 9.10 Generator Load Rejection With Bypass Failure at 100%P/105%F – Pressures

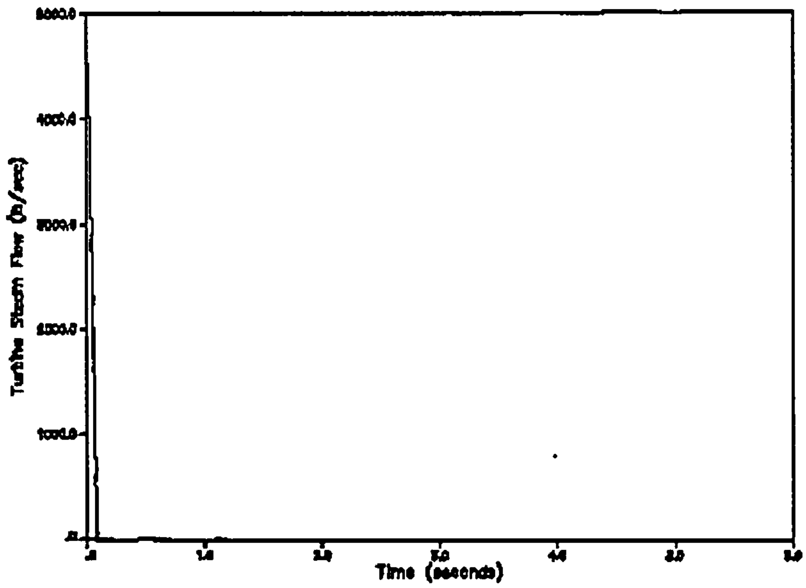

Figure 9.11 Generator Load Rejection With Bypass Failure at 100%P/105%F – Turbine Steam Flow

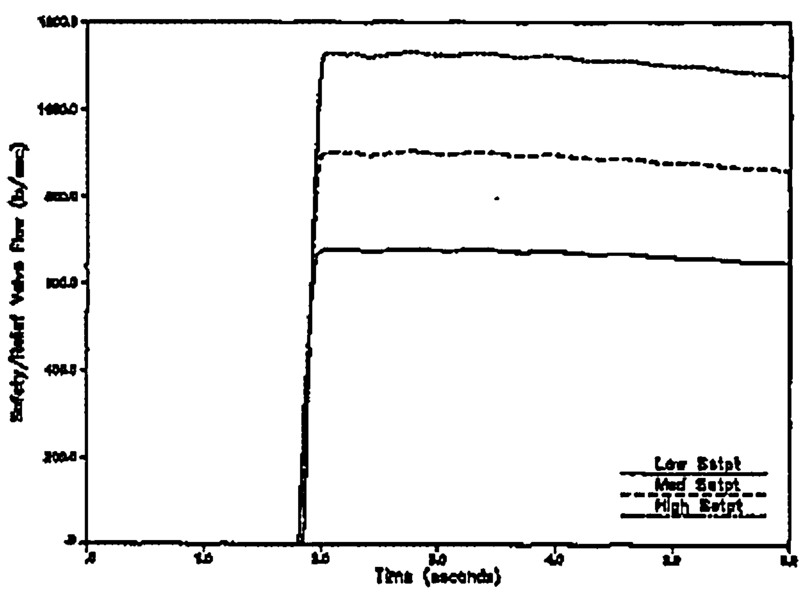

Figure 9.12 Generator Load Rejection With Bypass Failure at 100%P/105%F – MSRV Flows

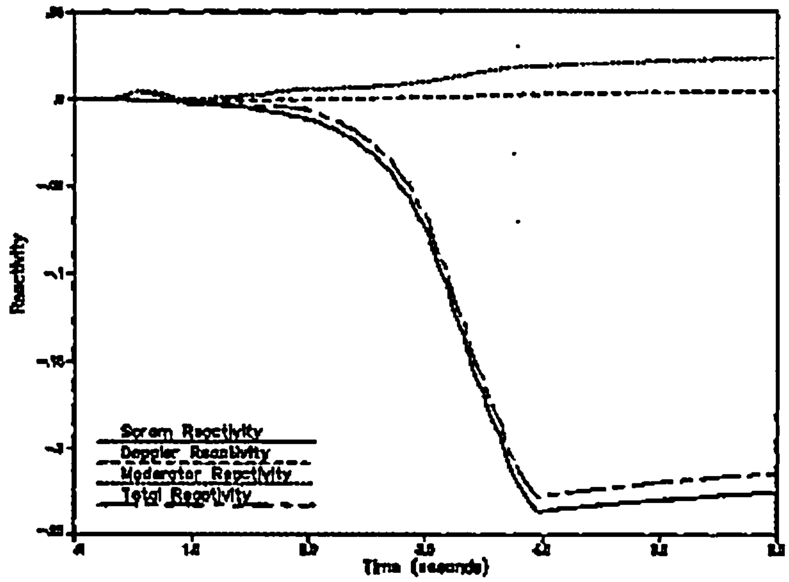

Figure 9.13 Generator Load Rejection With Bypass Failure at 100%P/105%F – Reactivities

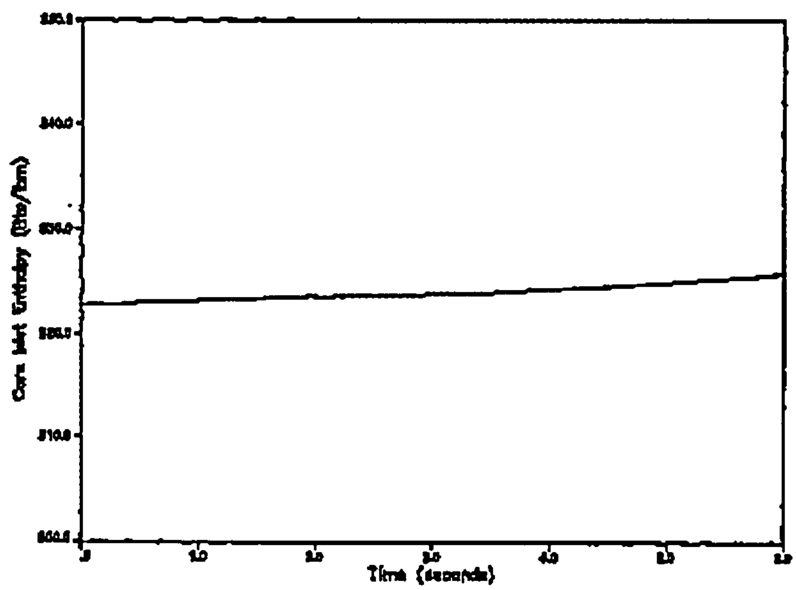

Figure 9.14 Generator Load Rejection With Bypass Failure at 100%P/105%F – Core Inlet Enthalpy

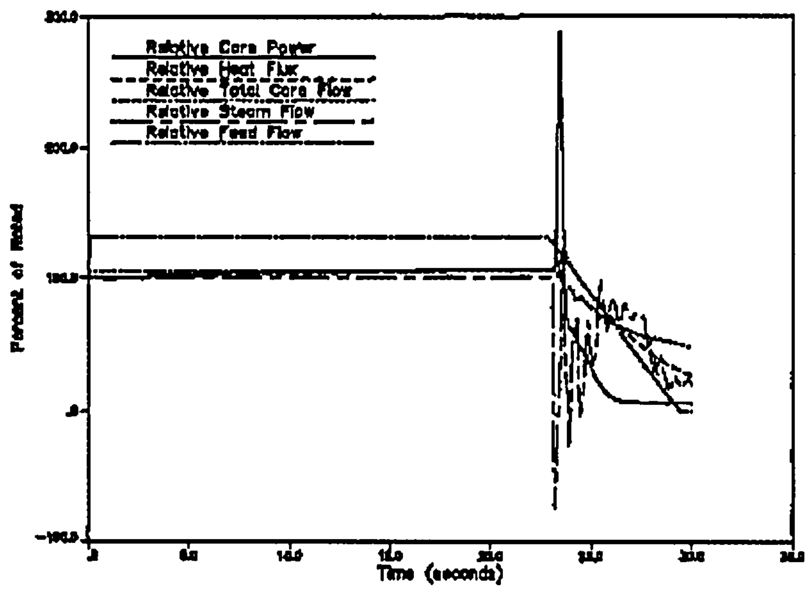

Figure 9.15 Feedwater Controller Failure Maximum Demand at 100%P/105%F – Power, Heat Flux, and Flows

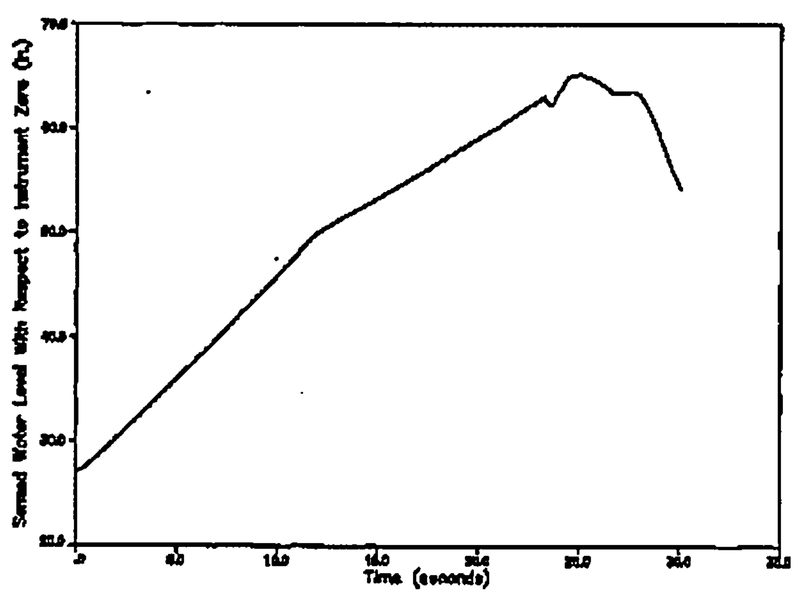

Figure 9.16 Feedwater Controller Failure Maximum Demand at 100%P/105%F – Downcomer Water Level

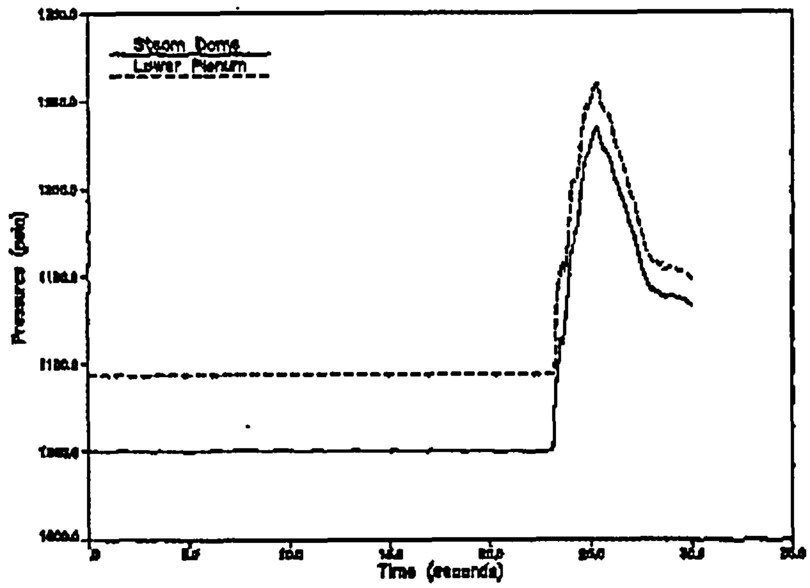

Figure 9.17 Feedwater Controller Failure Maximum Demand at 100%P/105%F – Pressures

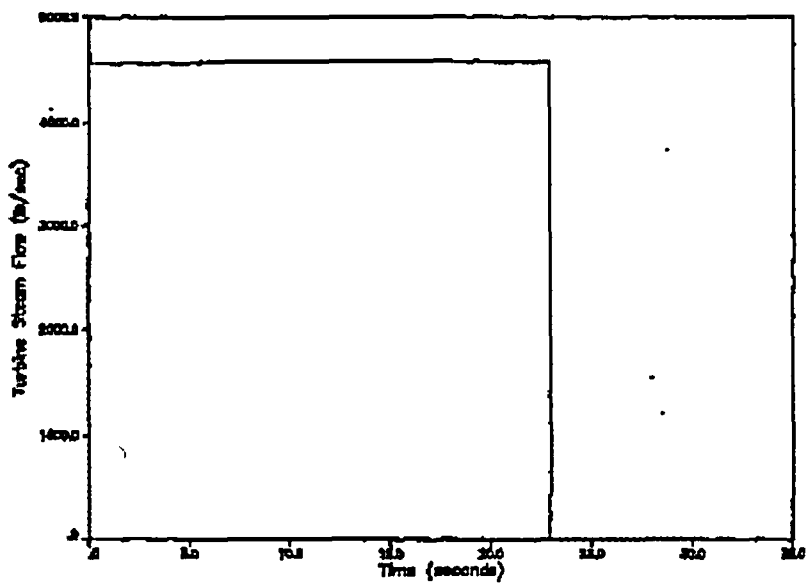

Figure 9.18 Feedwater Controller Failure Maximum Demand at 100%P/105%F – Turbine Steam Flow

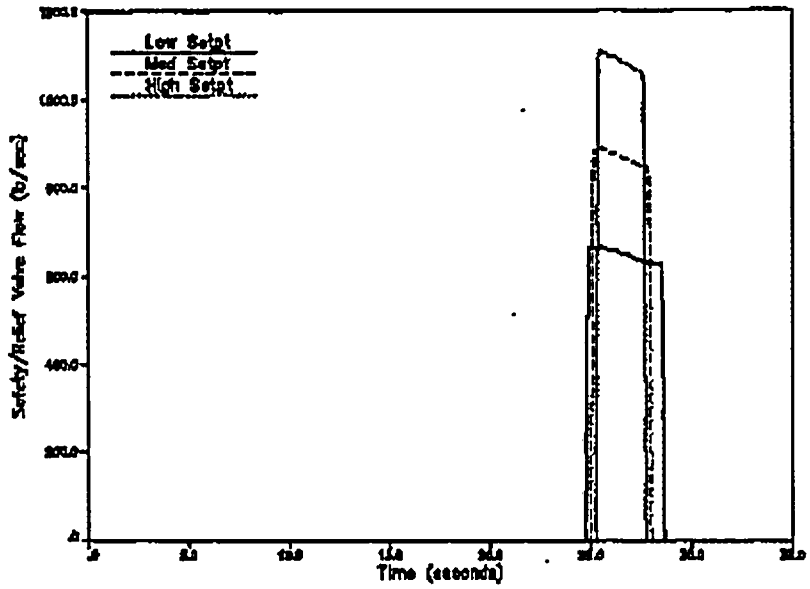

Figure 9.19 Feedwater Controller Failure Maximum Demand at 100%P/105%F – MSRV Flows

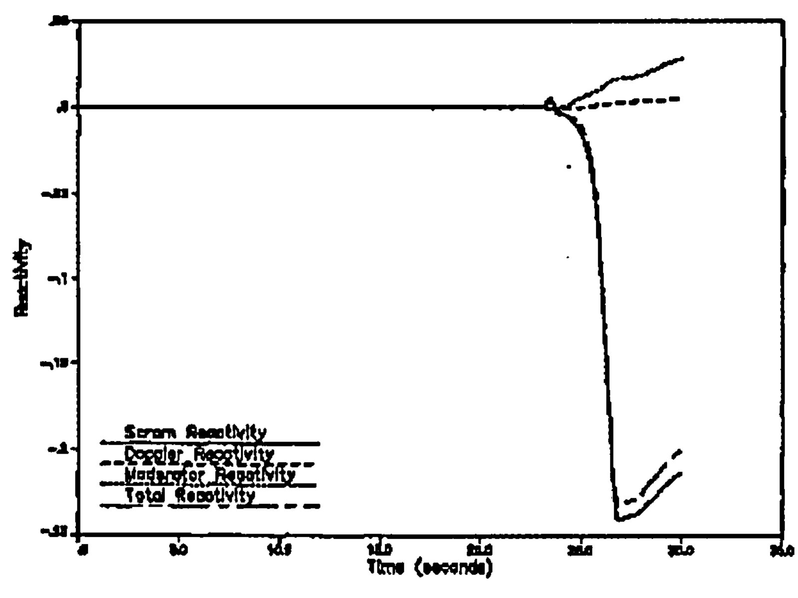

Figure 9.20 Feedwater Controller Failure Maximum Demand at 100%P/105%F – Reactivities

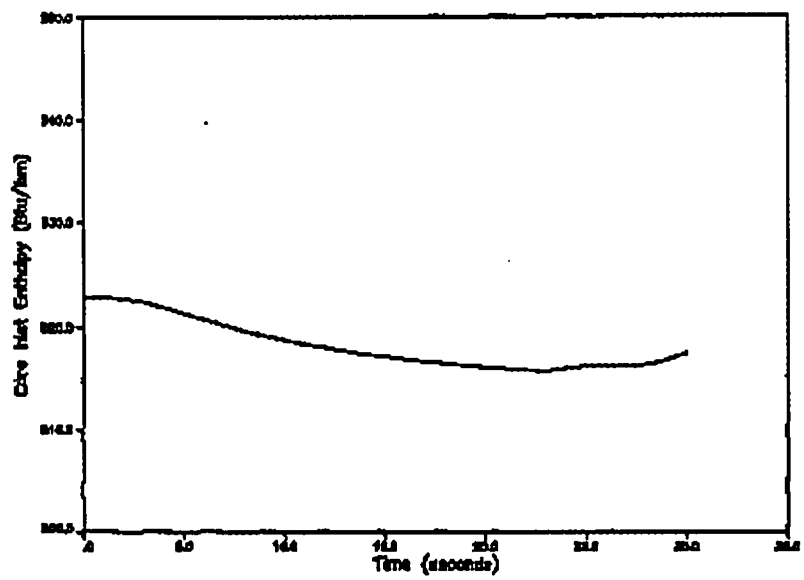

Figure 9.21 Feedwater Controller Failure Maximum Demand at 100%P/105%F – Core Inlet Enthalpy

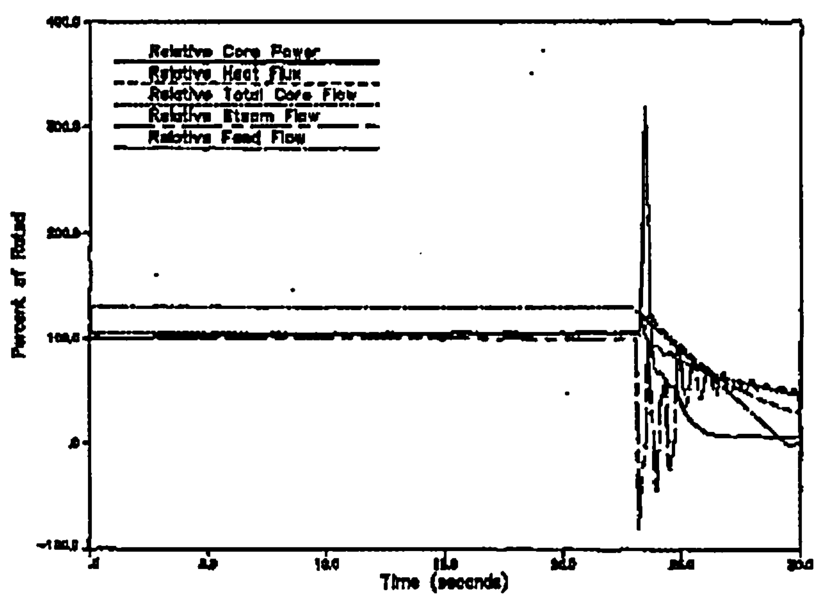

Figure 9.22 Feedwater Controller Failure Maximum Demand With Bypass Failure at 100%P/105%F – Power, Heat Flux, and Flows

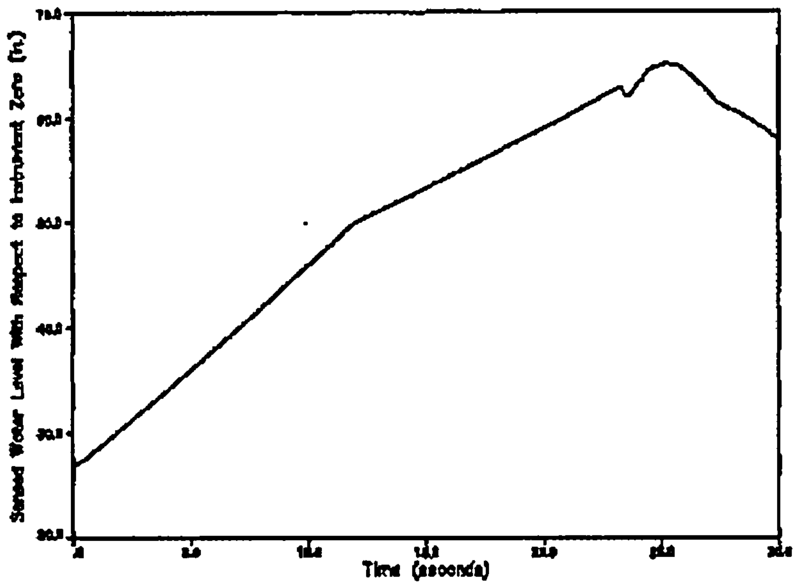

Figure 9.23 Feedwater Controller Failure Maximum Demand With Bypass Failure at 100%P/105%F – Downcomer Water Level

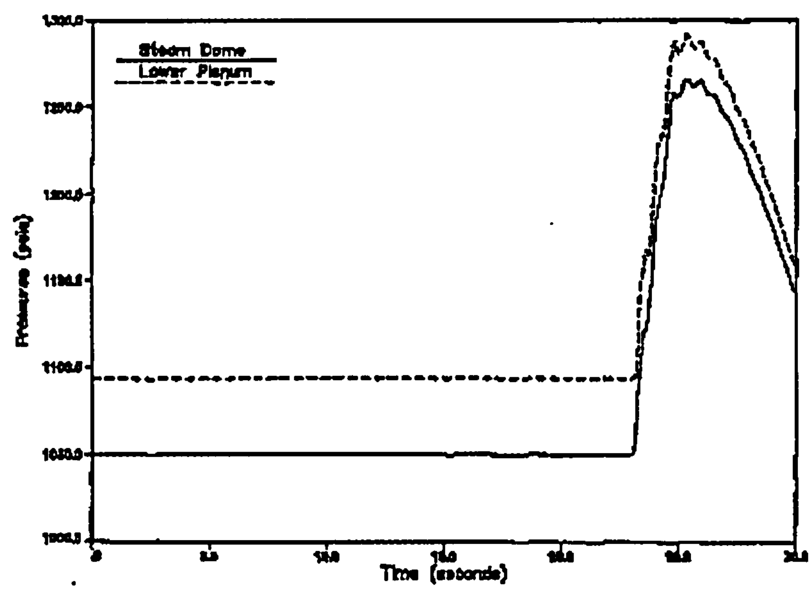

Figure 9.24 Feedwater Controller Failure Maximum Demand With Bypass Failure at 100%P/105%F – Pressures

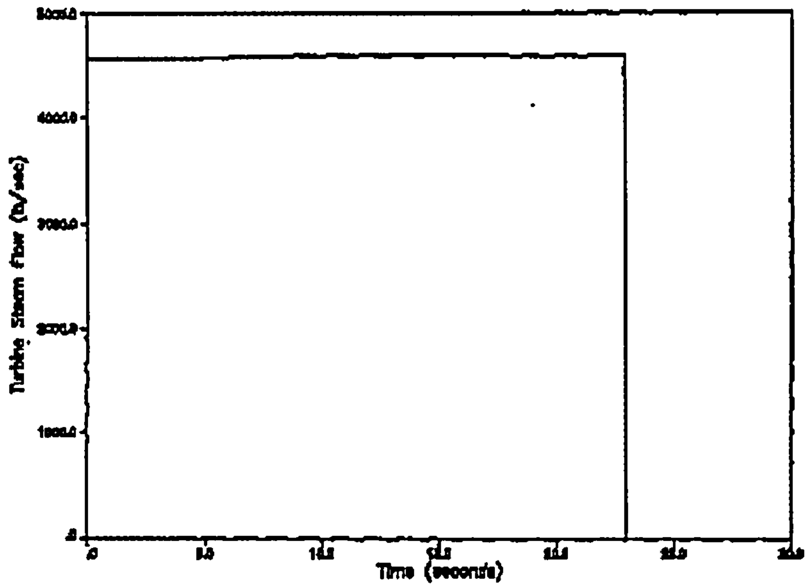

Figure 9.25 Feedwater Controller Failure Maximum Demand With Bypass Failure at 100%P/105%F – Turbine Steam Flow

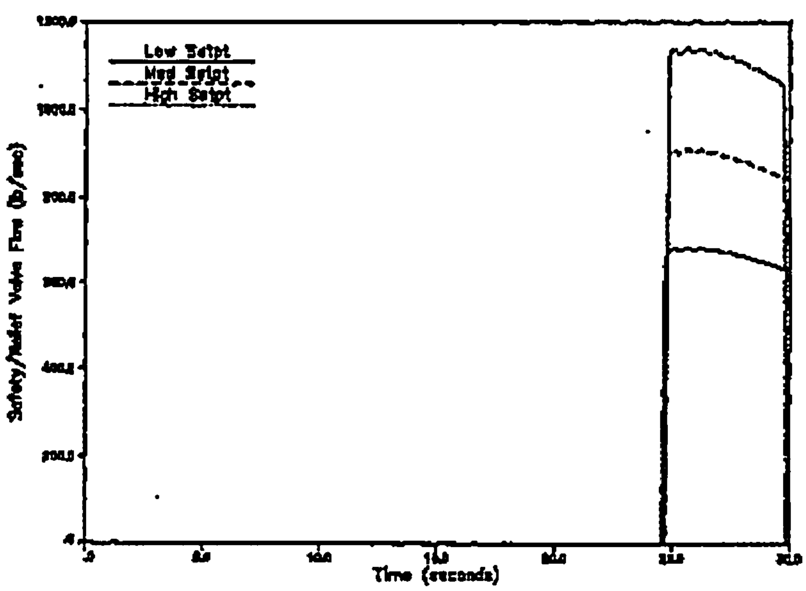

Figure 9.26 Feedwater Controller Failure Maximum Demand With Bypass Failure at 100%P/105%F – MSRV Flows

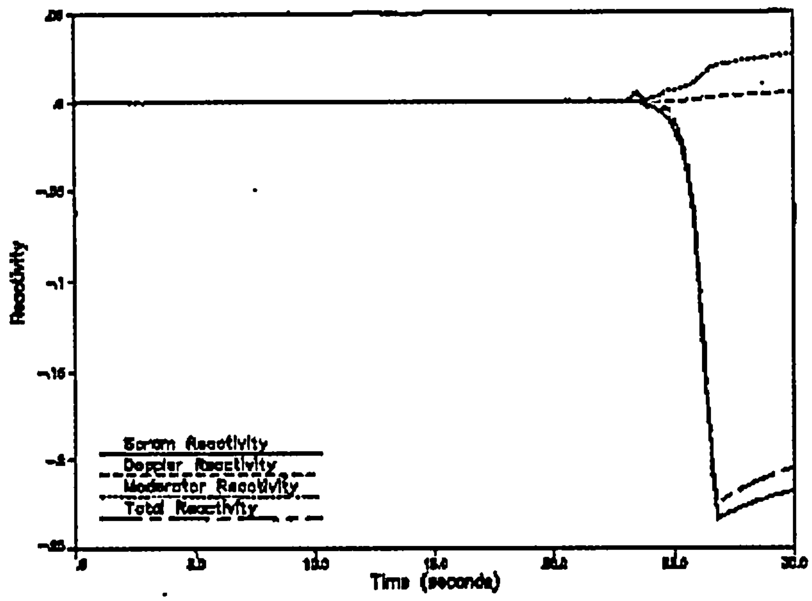

Figure 9.27 Feedwater Controller Failure Maximum Demand With Bypass Failure at 100%P/105%F – Reactivities

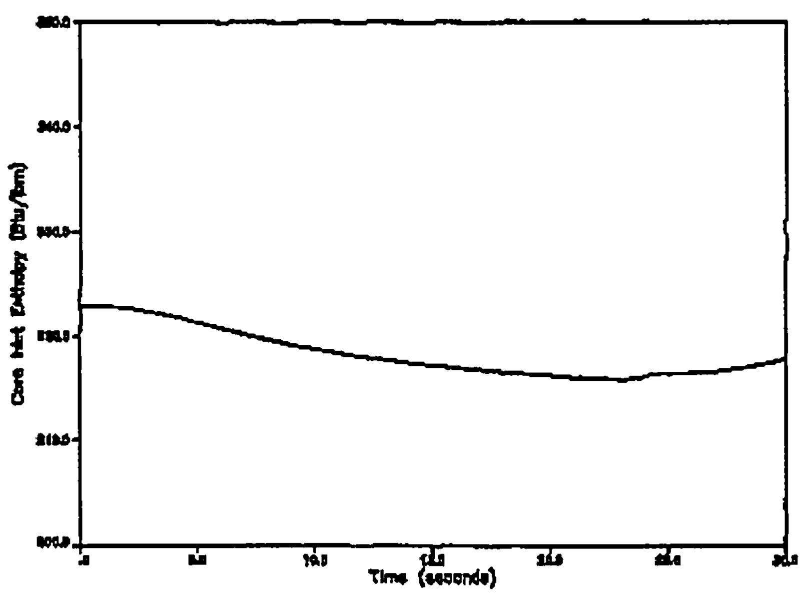

Figure 9.28 Feedwater Controller Failure Maximum Demand With Bypass Failure at 100%P/105%F – Core Inlet Enthalpy

10.0 Other Evaluations

10.1 *High Energy Line Break*

High energy line breaks are evaluated for their effects on equipment qualification. The mass and energy release rates following a high energy line break are not dependent on fuel design.

10.2 *Moderate Energy Line Break*

Moderate energy line breaks are evaluated for their effects on equipment qualification. The flooding evaluations are not dependent on fuel design.

10.3 *Environmental Qualification*

[

] Operation with ATRIUM-10 fuel does not affect the temperature, pressure or humidity consequences presented in the Reference 1 equipment qualification evaluations.

10.4 *Testing*

The use of ATRIUM-10 fuel will not impose any additional testing requirements for EPU operation.

10.4.1 Recirculation Pump Testing

There are no additional requirements for vibration testing of the recirculation pumps due to the use of ATRIUM-10 fuel at EPU conditions.

10.4.2 10 CFR 50 Appendix J Testing

The containment pressurization testing is not dependent on fuel design.

10.4.3 Main Steam Line, Feedwater, and Reactor Recirculation Piping Flow-Induced Vibration Testing

The vibration testing of the main steam line, feedwater, and reactor recirculation piping are not fuel design dependent.

10.5 *Individual Plant Evaluation*

Probabilistic risk assessments (PRAs) are performed to evaluate the risk of plant operations. The core damage frequency and large early release frequency are not affected by fuel design.

10.5.1 Initiating Event Frequency

The frequencies for the initiating events are not dependent on fuel design.

10.5.2 Component and System Reliability

The component and system reliability does not depend on fuel design.

10.5.3 Operator Response

The operator response capabilities following an initiating event are not affected by fuel design.

10.5.4 Success Criteria

The success criteria defined in the individual BFN PRA models are used to protect the reactor fuel; however, the criteria themselves are not dependent on fuel design.

10.5.5 External Events

There are no additional vulnerabilities to external events due to the use of ATRIUM-10 fuel at EPU conditions.

10.5.6 Shutdown Risk

[

] Therefore, operation with ATRIUM-10 fuel will not affect the shutdown risk.

10.5.7 PRA Quality

The PRA quality is not dependent on fuel design.

10.6 *Operator Training and Human Factors*

Reference 1 Section 10.6 discusses additional training needs for EPU plant operation. As noted in Section 6.7.1, a change to the procedures and subsequent operator training are needed to require HPCI system isolation during an Appendix R event within 5 minutes. No other additional training requirements will be necessary for the use of ATRIUM-10 fuel at EPU conditions aside from the currently established cycle-specific core training and fuel specific training requirements. The ATRIUM-10 fuel will be included in TVA's cycle-specific training. Necessary simulator changes to appropriately model the ATRIUM-10 fuel will be made prior to operation.

10.7 *Plant Life*

[

] Continued implementation of the plant inspection program ensures that any degradation of applicable components that occurs during operation is promptly identified.

11.0 Licensing Evaluations

11.1 *Other Applicable Requirements*

11.1.1 NRC and Industry Communications

The issues arising from NRC and industry communications are not dependent on fuel design.

11.1.2 Plant-Unique Items

The only plant-unique item potentially affected by the use of ATRIUM-10 fuel is the emergency operating procedures. These procedures will be revised prior to operation with ATRIUM-10 fuel at EPU conditions.

Abnormal operating procedures are event based, and are not impacted by fuel design.

12.0 References

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4. "Browns Ferry Nuclear Plant (BFN) – Units 1, 2 and 3 – License Amendment – Alternate Source Term," TVA-BFN-TS-405, July 31, 2002.

5. 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.

6. 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.

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8. 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.

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13. XN-CC-33(P)(A) Revision 1, *HUXY: A Generalized Multirod Heatup Code with 10 CFR 50 Appendix K Heatup Option Users Manual*, Exxon Nuclear Company, November 1975.

14. Letter, S. Richards (NRC) to J. F. Mallay (FANP), "Siemens Power Corporation RE: Request for Concurrence on Safety Evaluation Report Clarifications (TAC No. MA6160)," May 31, 2000.

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

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