PLANT SAFETY ANALYSIS

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14.0 PLANT SAFETY ANALYSIS

14.1 ANALYTICAL OBJECTIVE

The objective of the Plant Safety Analysis is to evaluate the ability of the plant to operate without undue hazard to the health and safety of the public.

Previous sections of this report provide the objective, design basis, and description of each major system and component. Systems that have unique requirements arising from considerations of nuclear safety are evaluated in the safety evaluation portions of those sections of the report. The safety evaluations consider the effects of failures within the system being investigated. Systems essential to safety are capable of performing their functions in adverse circumstances.

This chapter provides the analytical objective, design basis, and safety evaluation for the overall plant integrated systems. Limiting events which may be affected by reload core designs are evaluated and documented in reload licensing reports for each fuel cycle. Safety evaluations for specific reload fuel types are documented or referenced in either the Reload Licensing Report for the specific cycle or in the licensing topical report, "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A, and revisions thereto.

Safety analyses have also been performed to justify plant operating flexibility options such as operation in the Extended Load Line Limit (ELLL) Region, operation in the Increased Core Flow (ICF) Region and operation with Final Feedwater Temperature Reduction (FFWTR). Subsequent to these analyses, additional analyses have been performed[1], and plant performance improvements have been implemented on Units 1, 2, and 3 for operation in the Maximum Extended Load Line Limit (MELLL) Region. Analyses have also been performed for Extended Power Uprate (EPU) (120% OLTP).[2, 3] Results of these analyses are reconfirmed with each reload analyses as documented in the Reload Licensing Reports for a specific cycle.

Definitions for key terms used in this section are presented in Subsection 1.2, "Definitions."

It should be noted that AREVA Nuclear Reactor Operations was renamed “Framatome” in 2018. For the purposes of this chapter, “AREVA” should be considered synonymous with “Frameatome.”

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[1] NEDC-32422P, "Maximum Extended Load Line Limit and ARTS Improvement Program Analyses for Browns Ferry Nuclear Plant Unit 1, 2 and 3," GE Nuclear Energy, April 1995

[2] NEDC-33860P, “Safety Analysis Report for Browns Ferry Nuclear Plant Units 1, 2, and 3 Extended Power Uprate,” GE Hitachi Nuclear Energy, July 2015

[3] ANP-3403P, “Fuel Uprate Safety Analysis Report for Browns Ferry Nuclear Plant Units 1, 2, and 3,” AREVA Inc., June 2015

14.2 UNACCEPTABLE SAFETY RESULTS FOR ABNORMAL OPERATIONAL TRANSIENTS

1. The release of radioactive material to the environs to such an extent that the limits of 10 CFR 20 are exceeded.

2. Any fuel failure calculated as a result of the transient.

3. Nuclear system stress in excess of that allowed for transients by applicable industry codes.

14.3 UNACCEPTABLE SAFETY RESULTS FOR ACCIDENTS

1. Radioactive material release to such an extent that the guideline values of 10 CFR 50.67 would be exceeded.

2. Fuel Cladding temperature in excess of 2200°F for pipe breaks.

3. Nuclear system stresses in excess of that allowed for accidents by applicable industry codes.

4. Containment stresses in excess of that allowed for accidents by applicable industry codes when containment is required.

5. Overexposure to radiation of plant operation personnel in the control room.

6. Peak enthalpy of the fuel in excess of 280 cal/gm for the control rod drop accident.

14.4 APPROACH TO SAFETY ANALYSIS

14.4.1 General

The below probabilistic analysis discussion reflects capabilities at the time of the initial BFN design. The most informative approach to safety analysis is generally one based on probabilistic analysis. Such an approach allows precise statements of unacceptable safety results and permits categorization and evaluation of failures by relative probabilities. To satisfactorily effect such an approach, adequate data on component failure rates, failure modes, failure distributions, repair times, and repair time distributions are required. With the necessary data, models can be constructed and analyzed to reveal the realistic probabilities of events pertinent to nuclear safety. General Electric is currently compiling sources of data and developing the techniques of probabilistic analysis. Although probabilistic analysis currently provides much insight into the problems of safety, the technique has not matured sufficiently or gained the general acceptance necessary to permit it to be the major analysis tool.

Until the probability approach matures, two basic groups of events pertinent to safety (abnormal operational transients and accidents) will be investigated separately. The preclusion of unacceptable safety results requires that no damage to the fuel occurs and that no nuclear system process barrier damage results from any abnormal operational transient. Thus, analysis of this group of events evaluates the plant features that protect the first two radioactive material barriers. Analysis of the events in the second group (accidents) evaluates situations that require functioning of the engineered safeguards including containment. Tables 14.4-1 and 14.4-2 display the overall results of these analyses.

In considering the various abnormal operational transients and accidents, the full spectrum of conditions in which the core may exist is considered. This is accomplished by investigating the differing safety aspects of the six BWR operating states, as described in Appendix G. In general, only the most severe event of a given type is described in detail.

Since the preclusion of unacceptable safety results for abnormal operational transients requires that no fuel damage occur, the limiting abnormal operational transients are examined for each fuel cycle to ensure this requirement is met. Different transient methodologies have been employed for the Browns Ferry abnormal operational transient analyses. The analyses for the following events are not dependent on fuel type or reactor power level: Control Rod Removal Error during Refueling (Section 14.5.4.3), Fuel Assembly Insertion Error during Refueling (Section 14.5.4.4), and Loss of Habitability of the Control Room (Section 14.5.9). Therefore, the GE analysis methodology is valid for these events and provides conservatism by accounting for uncertainties in computed results and utilizing NRC-approved methods. All other accident analyses employ AREVA

NRC-approved codes and methodologies. The results and methodology for the currently applicable limiting transient and accident analyses are contained in UFSAR Appendix N.

14.4.2 Abnormal Operational Transients

Figure 14.4-1 shows (in block form) the general method of identifying and evaluating abnormal operational transients. Eight nuclear system parameter variations are listed as potential initiating causes of threats to the fuel and the nuclear system process barrier; the parameter variations are as follows:

a. Nuclear system pressure increase,

b. Reactor vessel water (moderator) temperature decrease,

c. Positive reactivity insertion,

d. Reactor vessel coolant inventory decrease,

e. Reactor core coolant flow decrease,

f. Reactor core coolant flow increase,

g. Core coolant temperature increase, and

h. Excess of coolant inventory.

These parameter variations, if uncontrolled, could result in excessive damage to the reactor fuel or damage to the nuclear system process barrier, or both. A nuclear system pressure increase threatens to rupture the nuclear system process barrier from internal pressure. A pressure increase also collapses the voids in the moderator, causing an insertion of positive reactivity that threatens fuel damage from overheating. A reactor vessel water (moderator) temperature decrease results in an insertion of positive reactivity as density increases. This could lead to fuel overheating. Positive reactivity insertions are possible from causes other than nuclear system pressure or moderator temperature changes; such reactivity insertions threaten fuel damage caused by overheating. Both a reactor vessel coolant inventory decrease and a reduction in the flow of coolant through the core threaten to overheat the fuel as the coolant becomes unable to adequately remove the heat generated in the core. An increase in coolant flow through the core reduces the void content of the moderator, resulting in an increased fission rate. If uncontrolled, excess of coolant inventory could result in excessive carryover.

These eight parameter variations include all of the effects within the nuclear system caused by abnormal operational transients that threaten the integrities of the reactor fuel or nuclear system process barrier. The variation of any one parameter may cause a change in another listed parameter; however, for analysis purposes, threats to barrier integrity are evaluated by groups according to the parameter variation originating the threat. For example, positive reactivity insertions resulting from sudden pressure increases are evaluated in the group of threats stemming from nuclear system pressure increases.

Abnormal operational transients are the results of single equipment failures or single operator errors that can be reasonably expected during any mode of plant operations. The following types of operational single failures and operator errors are identified:

a. The opening or closing of any single valve (a check valve is not assumed to close against normal flow),

b. The starting or stopping of any single component,

c. The malfunction or maloperation of any single control device,

d. Any single electrical failure, and

e. Any single operator error.

Operator error is defined as an active deviation from written operating procedures or nuclear plant standard operating practices. A single operator error is the set of actions which is a direct consequence of a single erroneous decision. The set of actions is limited as follows:

a. Those actions that could be performed by not more than one person,

b. Those actions that would have constituted a correct procedure had the initial decision been correct, and

c. Those actions that are subsequent to the initial operator error and have an effect on the designed operation of the plant, but are not necessarily directly related to the operator error.

Examples of single operator errors are as follows:

a. An increase in power above the established flow control power limits by control rod withdrawal in the specified sequences,

b. The selection and complete withdrawal of a single control rod out of sequence,

c. An incorrect calibration of an average power range monitor, and

d. Manual isolation of the main steam lines due to operator misinterpretation of an alarm or indication.

The five types of single errors or single malfunctions are applied to the various plant systems with a consideration for a variety of plant conditions to discover events that directly result in any of the listed undesired parameter variations. Once discovered, each event is evaluated for the threat it poses to the integrities of the radioactive material barriers. Generally, the most severe event of a group of similar events is described.

Two additional events are analyzed as special cases: (1) loss of habitability of the control room. This abnormal condition is postulated to demonstrate the capability to perform the operations required to maintain the plant in a safe condition from outside the control room, and (2) Inability to shut down the reactor with the control rods. This event is presented to justify the requirement for the Standby Liquid Control System and results in a normal shutdown using this system. Therefore, no further analysis or evaluation is required other than that presented in Subsection 3.8 ("Standby Liquid Control System").

14.4.3 Accidents

Figure 14.4-2 shows (in block form) the method of identifying and evaluating accidents. For analysis purposes, accidents are categorized as follows:

a. Accidents that result in radioactive material release from the fuel with the nuclear system process barrier, primary containment, and secondary containment initially intact,

b. Accidents that result in radioactive material release directly to the primary containment,

c. Accidents that result in radioactive material release directly to the secondary containment with the primary containment initially intact,

d. Accidents that result in radioactive material release directly to the secondary containment with the primary containment not intact, and

e. Accidents that result in radioactive material release outside the secondary containment.

Accidents are defined as hypothesized events that affect one or more of the radioactive material barriers and which are not expected during the course of plant operations. The accident types considered are as follows:

a. Mechanical failure of various components leading to the release of radioactive material from one or more barriers. The components referred to here are not components that act as radioactive material barriers. Examples of mechanical failures are breakage of the coupling between a control rod drive and the control rod, failure of a crane cable, and failure of a spring used to close an isolation valve.

b. Overheating of the fuel barrier. This includes overheating as a result of reactivity insertion or loss of cooling. Other radioactive material barriers are not considered susceptible to failure due to any potential overheating situation.

c. Arbitrary rupture of any single pipe up to and including complete severance of the largest pipe in the nuclear system process barrier. Such rupture is assumed only if the component to rupture is subjected to significant pressure.

The effects of the various accident types are investigated, with a consideration for a variety of plant conditions, to examine events that result in the release of radioactive material. The accidents resulting in potential radiation exposures greater than any other accident considered under the same general accident assumptions are designated design basis accidents and are described in detail.

To incorporate additional conservatism into the accident analyses, consideration is given to the effects of an additional, unrelated, unspecified fault. The fault is assumed to occur in a safety-related component or piece of equipment that is needed to respond to the initiating event in order to achieve the intended safety-related function. Such a fault is assumed to result in the maloperation of a device which is intended to mitigate the consequences of the accident. The assumed result of such an unspecified fault is restricted to such relatively common events as an electrical failure, instrument error, motor stall, breaker freeze-in, or valve maloperation. Highly improbable failures, such as pipe breaks, are not assumed to occur coincident with the assumed accident in the short term. The additional failures to be considered are in addition to failures caused by the accident itself.

In the analyses of the design basis accidents consideration for a variety of single additional failures is made by making analysis assumptions that are sufficiently conservative to include the range of effects from any single additional failure. Thus, there exists no single additional failure of the type to be considered that could worsen the computed radiological effects of the design basis accidents.

14.4.4 Barrier Damage Evaluations

14.4.4.1 Fuel Damage

Subsection 3.7 ("Thermal and Hydraulic Design") describes the various fuel failure mechanisms and establishes fuel damage limits for various plant conditions. Preclusion of unacceptable safety result 1 and 2, for Abnormal Operational Transients is determined by demonstrating that abnormal operational transients do not result in a minimum critical power ratio (MCPR) of less than 1.0. If MCPR does remain above 1.0, no fuel failures result from the transients, and thus the radioactivity released from the plant cannot be increased over the operating conditions existing prior to the transient. It should be noted that maintaining MCPR greater than 1.0 is a sufficient but not necessary condition to assure that no fuel damage occurs. (This is discussed in Subsection 3.7.)

For situations in which fuel damage is sustained, the extent of damage is determined by correlating fuel energy content, cladding temperature, fuel rod internal pressure, and cladding mechanical characteristics. These correlations are substantiated by fuel rod failure tests and are discussed in Subsection 3.7 and Section 6.

Preclusion of unacceptable safety result 2 for accidents is shown by demonstrating that fuel clad temperature remains below 2200°F. The selection of this temperature limitation is discussed in Section 6.

14.4.4.2 Nuclear System Process Barrier Damage

Preclusion of unacceptable safety result 3 for abnormal operational transients and unacceptable safety result 3 for accidents is assessed by comparing peak internal pressure with the overpressure transient allowed by the applicable industry code. The only significant areas of interest for internal pressure damage are the high-pressure portions of the nuclear system primary barrier: the reactor vessel and the high-pressure pipelines attached to the reactor vessel. The overpressure below which no damage can occur is taken as the lowest of pressure increases over design pressure allowed by either the ASME Code Section III for the reactor vessel or USAS B 31.1 Code for the high pressure nuclear system piping. The ASME Code Section III permits pressure transients up to 10 percent over design pressure (110 percent x 1250 psig = 1375 psig); USAS B 31.1 permits pressure transients up to 20 percent over the design pressure.

Thus, it can be concluded that the high-pressure portion of the nuclear system process barrier meets the design requirement if peak nuclear system pressure remains below 1375 psig.

An analysis performance measurement, which is discussed in Subsection 3.6 ("Nuclear Design"), is used to evaluate whether nuclear system process barrier damage occurs as a result of reactivity accidents. If peak fuel enthalpy remains below 280 calories per gram no nuclear system process barrier damage results from nuclear excursion accidents.

14.4.4.3 Containment Damage

Preclusion of unacceptable safety result 1 (for abnormal transients) and 4 (for accidents) requires that the primary and secondary containment retain their integrities for certain accident situations. Containment integrity is maintained as long as internal pressures remain below the maximum allowable values. The maximum allowable internal pressures are as follows:

Drywell (primary containment)	62 psig
Pressure Suppression Chamber (primary containment)	62 psig
Secondary Containment	2 inches H_2O

Damage to any of the radioactive material barriers as a result of accident-initiated fluid impingement and jet forces is considered in the other portions of the Safety Analysis Report where the mechanical design features of systems and components are described. Design basis accidents are used in determining the sizing and strength requirements of much of the essential nuclear system components. A comparison of the accidents considered in this section with those used in the mechanical design of equipment reveals that either the applicable accidents are the same or that the accident in this section results in less severe stresses than those assumed for mechanical design.

TABLE 14.4-1

(Sheet 1)

PLANT SAFETY ANALYSIS

SUMMARY OF ABNORMAL OPERATIONAL TRANSIENTS

Undesired Parameter Variation	Event Causing Transient	Scram Caused by
Nuclear system pressure increase	Generator trip without bypass	Turbine control valve fast closure
Nuclear system pressure increase	Turbine trip without bypass	Turbine stop valve closure
Nuclear system pressure increase	Main steam line isolation valve closure	Main steam line isolation valve closure
Nuclear system pressure increase	Loss of Condenser vacuum	Turbine stop valve closure
Nuclear system pressure increase	Bypass valve malfunction	Reactor vessel high pressure
Nuclear system pressure increase	Pressure regulator malfunction	Reactor vessel high pressure
Reactor water temperature decrease	Shutdown cooling malfunction decrease temperature	High Neutron flux
Reactor water temperature decrease	Loss of feedwater heater*	None
Reactor Water temperature decrease	Inadvertent pump start*	None
Positive reactivity insertion	Continuous rod withdrawal during power range operation*	None
Positive reactivity insertion	Continuous rod withdrawal during reactor startup*	High neutron flux
Positive reactivity insertion	Control rod removal error during refueling	High neutron flux
Positive reactivity insertion	Fuel assembly insertion error during refueling	High neutron flux
Coolant inventory decrease	Pressure regulator failure - open**	Main steam line isolation valve closure
Coolant inventory decrease	Open main steam relief valve**	
Coolant inventory decrease	Loss of feedwater flow	Reactor vessel low water level

*This transient results in no significant change in nuclear system pressure.

**This transient results in a depressurization.

TABLE 14.4-1

(Sheet 2)

PLANT SAFETY ANALYSIS

SUMMARY OF ABNORMAL OPERATIONAL TRANSIENTS

Undesired Parameter Variation	Event Causing Transient	Scram Caused by
Coolant inventory decrease	Loss of auxiliary power system	Loss of power to reactor protection
Core flow decrease	Recirculation flow control failure - decreasing flow**	None
Core flow decrease	Trip of one recirculation pump**	None
Core flow decrease	Trip of two recirculation pumps**	None
Core flow increase	Recirculation pump flow control failure increasing flow*	High neutron flux
Core flow increase	Startup of idle recirculation pump*	None
Excess of coolant inventory	Feedwater Controller failure-maximum demand	Turbine stop valve closure

*This transient results in no significant change in nuclear system pressure.

**This transient results in a depressurization.

TABLE 14.4-2

PLANT SAFETY ANALYSIS

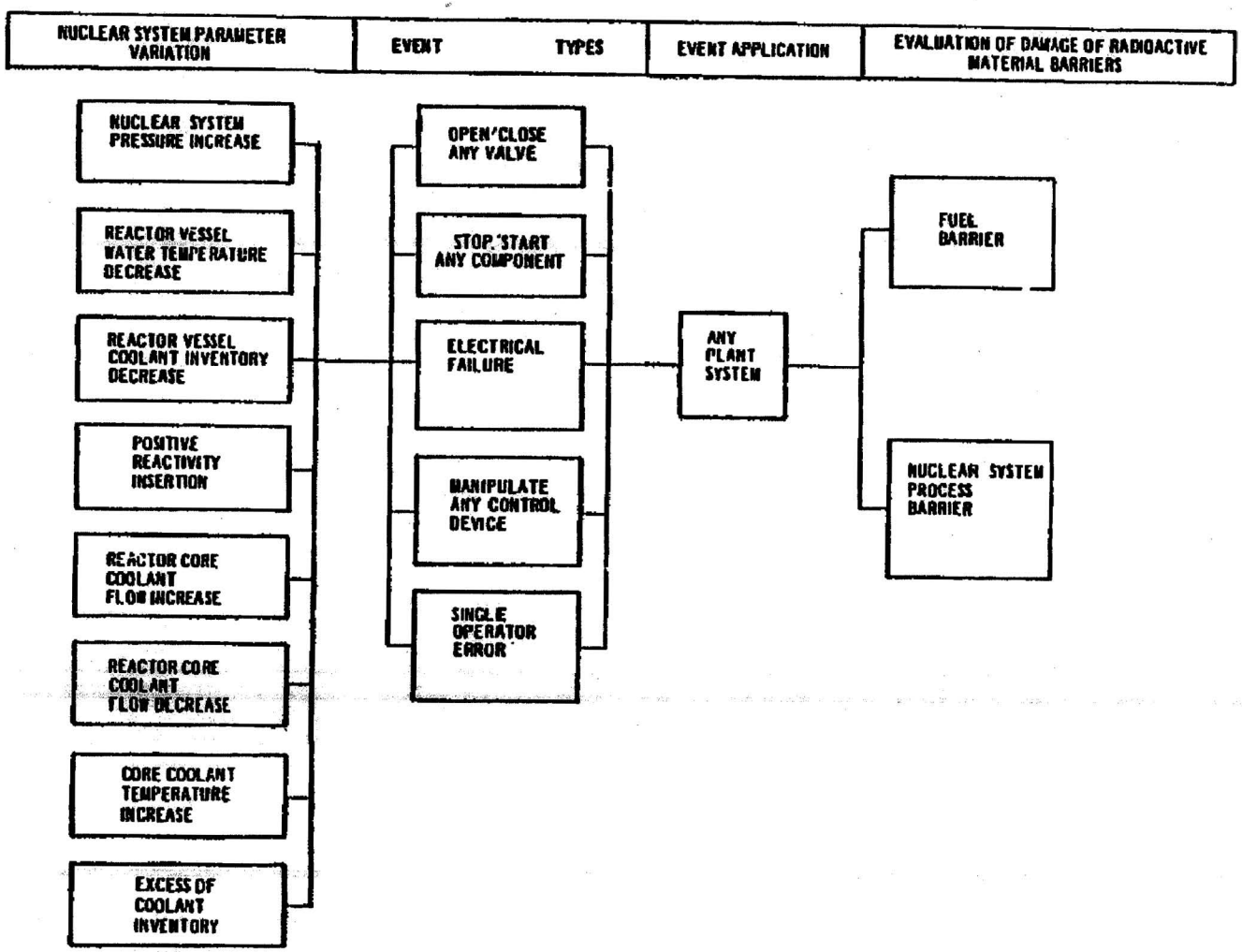
RESULTS OF DESIGN BASIS ACCIDENTS

Design Basis Accident	Percent of Core Reaching Cladding Temperature of 2200°F	Peak System Pressure
Rod Drop Accident	Not applicable***	<1375 psig
Loss of Coolant Accident	0	Not applicable*
Refueling Accident	0	Not applicable**
Main Steam Line Break Accident	0	Not applicable*

*This accident results in a depressurization.

**This accident occurs with the reactor vessel head off.

***Peak fuel enthalpy is less than 280 cal/gm.

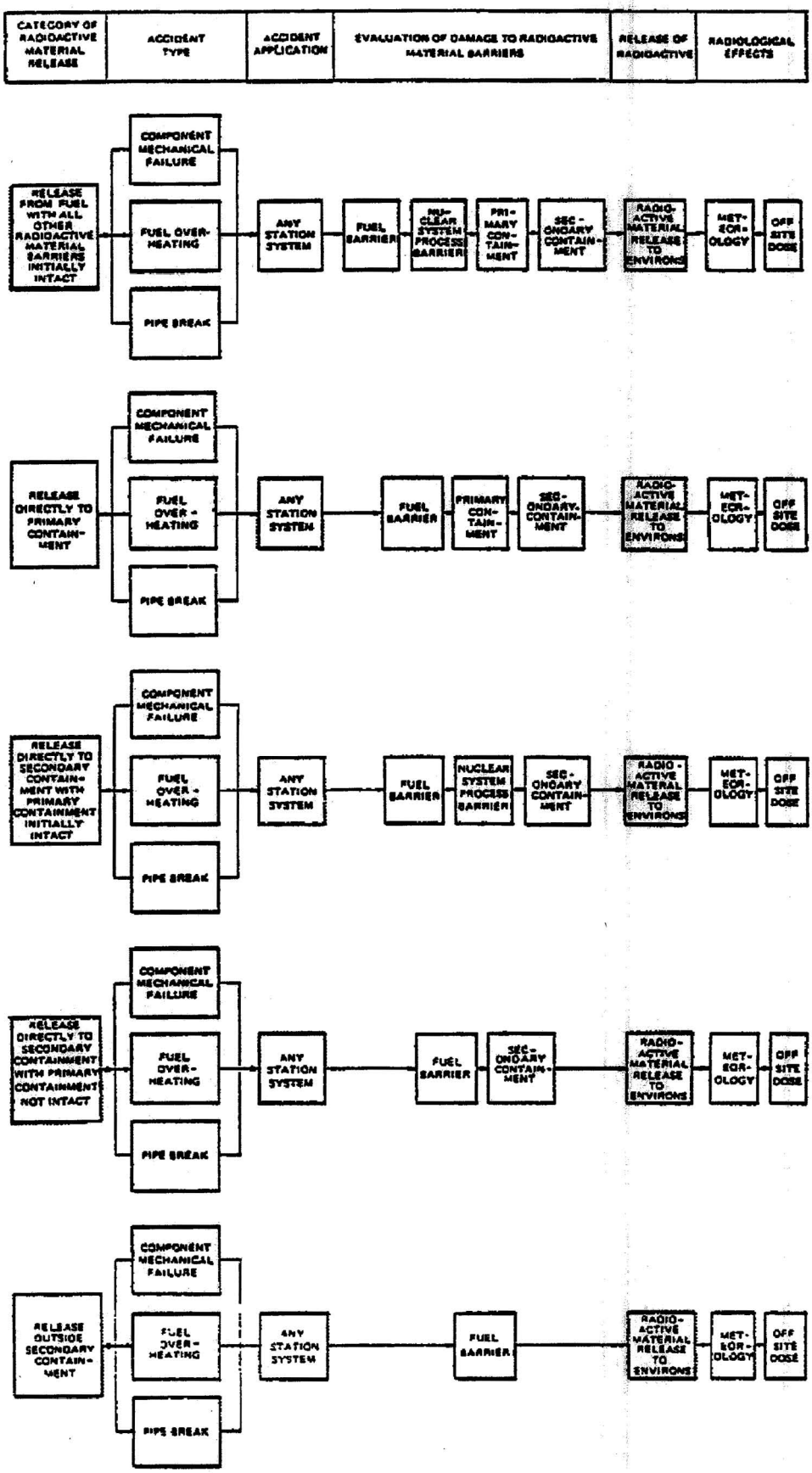


BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

Plant Safety Analysis—Method for Identifying and Evaluating Abnormal Operational Transients

FIGURE 14.4-1

AMENDMENT 16



AMENDMENT 16

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

Plant Safety Analysis
Method for Identifying
and Evaluating Accidents
FIGURE 14.4-2

14.5 ANALYSES OF ABNORMAL OPERATIONAL TRANSIENTS

14.5.1 Objective

This section contains general descriptions of abnormal operational transients analyzed for BFN Units 1, 2, and 3.

The results of these analyses may change with subsequent core reloads. The bounding transients are re-analyzed for each fuel reload and subsequent operating cycle to determine which is most limiting. Events for which a newer fuel reload specific analysis need to be performed are noted. These results can be found in the appropriate reload licensing document.

BFN Units 1, 2, and 3 have a similar system geometry, reactor protection system (RPS) configuration and mitigation functions (as described in earlier sections of the UFSAR). Additionally, BFN Units 1, 2, and 3 have similar thermal-hydraulic and transient behavior characteristics. Therefore, trends are expected to be the same for all units. Consequently, the transient analyses described in this chapter were performed for BFN Unit 3 and used as the representative unit to quantify trends for the other unit.

The analyses are based on the core loading characteristics of BFN 24-month fuel equilibrium cycle with AREVA ATRIUM-10XM. This is considered to be representative of future cycles, because specific fuel operating limits will continue to be calculated for each fuel cycle according to current reload practice. For the non-limiting transient events not re-analyzed on a reload specific basis, the 24-month cycle exposure assumption is also applicable to shorter fuel cycle length, since the fuel exposure variation has a negligible impact on the transient results and will not cause the severity trend to change significantly.

14.5.1.1 Transient Events Classification

The transient analyses that have been analyzed in this document are classified into seven categories of events. These seven categories and the transient events which they include are:

A) Events Resulting in a Nuclear System Pressure Increase:
   1. Generator Load Reject
   2. Loss of Condenser Vacuum
   3. Turbine Trip
   4. Turbine Bypass Valve Malfunction
   5. Main Steam Isolation Valve Closure
   6. Pressure Regulator Downscale Failure

B) Events Resulting in a Reactor Vessel Water Temperature Decrease:
   1. Loss of a Feedwater Heater
   2. Shutdown Cooling (Residual Heat Removal System) Malfunction-Decreasing Temperature
   3. Inadvertent Pump Start

C) Events Resulting in a Positive Reactivity Insertion:
   1. Continuous Control Rod Withdrawal During Power Range Operation
   2. Continuous Rod Withdrawal During Reactor Startup
   3. Control Rod Removal Error During Refueling
   4. Fuel Assembly Insertion Error During Refueling

D) Events Resulting in a Reactor Vessel Coolant Inventory Decrease:
   1. Pressure Regulator Failure Open
   2. Inadvertent Opening of a Main Steam Relief Valve
   3. Loss of Feedwater Flow
   4. Loss of Auxiliary Power

E) Events Resulting in a Core Coolant Flow Decrease:
   1. Recirculation Flow Control Failure - Decreasing Flow
   2. Trip of One Recirculation Pump
   3. Trip of Two Recirculation Pumps
   4. Recirculation Pump Seizure

F) Events Resulting in a Core Coolant Flow Increase:
   1. Recirculation Flow Controller Failure - Increasing Flow
   2. Startup of Idle Recirculation Pump

G) Events Resulting in Excess of Coolant Inventory:
   1. Feedwater Control Failure - Maximum demand

14.5.1.2 Transient Events Conditions for UFSAR Analysis

Computer Codes

NRC approved computer models have been used for the analysis of each event. The computer codes used in the different transient events analyses are AREVA codes unless otherwise noted, and are summarized as follows:

Transient Event Description	Computer Code Used for Analysis
Generator Trip (TCV Fast Closure) With Bypass Valves Failure	COTRANSA2 XCOBRA XCOBRA-T RODEX2 RODEX4
Load Rejection No Bypass/EOC-RPT-OOS	COTRANSA2 XCOBRA XCOBRA-T RODEX2 RODEX4
Bypass Valves Failure Following Turbine Trip, High Power	COTRANSA2 XCOBRA XCOBRA-T RODEX2 RODEX4
Closure of All Main Steam Line Isolation Valves	COTRANSA2
Closure of One Main Steam Line Isolation Valve	COTRANSA2
Loss of a Feedwater Heater	MICROBURN-B MICROBURN-B2 3-D
Inadvertent Pump Start	COTRANSA1/XCOBRA/XCOBRA-T
Continuous Rod Withdrawal During Power Range Operation	CASMO-4/MICROBURN-B2
Pressure Regulator Failure Open	COTRANSA2
Loss of Feedwater Flow	COTRANSA2
Recirculation Flow Control Failure-Decreasing Flow	CONTRANSA2/XCOBRA-T
Recirculation Pump Seizure	COTRANSA2 XCOBRA XCOBRA-T RODEX2 RODEX4
Recirculation Flow Control Failure-Increasing Flow	CASMO-4/MICROBURN-B2 XCOBRA
Startup of Idle Recirculation Loop	COTRANSA2 XCOBRA XCOBRA-T RODEX2 RODEX4
Feedwater Control Failure- Maximum Demand	COTRANSA2 XCOBRA XCOBRA-T RODEX2 RODEX4
Feedwater Control Failure- Maximum Demand/EOC-RPT-OOS	COTRANSA2 XCOBRA XCOBRA-T RODEX2 RODEX4
Feedwater Control Failure- Maximum Demand/TBP-OOS	COTRANSA2 XCOBRA XCOBRA-T RODEX2 RODEX4

The AREVA codes COTRANSA2, RODEX2 and CASMO-4/MICROBURN-B2 are the major codes used in the Overpressurization Analysis as described in the THERMEX methodology. COTRANSA2, reactor transient analysis code is currently being used for Browns Ferry reload analysis which includes Overpressurization analysis designs. The overpressurization was modeled by MSIV closure, TSV closure, and TCV closure (without bypass). The analyses were performed with the AREVA plant simulator code COTRANSA2 for 102% and 87.5% of 3952 MWt, and are performed on a cycle specific basis to demonstrate the adequacy of the pressure relief system. The plant response for transition cores will be similar to the plant response for a full core of ATRIUM 10XM fuel. Cycle-specific analyses reflecting the actual core configuration will be performed

as part of the reload licensing analysis. Core and fuel operating limits will be revised to ensure all licensing criteria are met.

For reload licensing analyses, the AREVA 3-D core simulator code MICROBURN-B2 is used for quasi steady-state analyses such as the RWE and LFWH. For the slow recirculation flow run-up event for setting $MCPR_f$ limits, the XCOBRA steady state core thermal hydraulics code is used. For fast transient (e.g., overpressurization) events, the one-dimensional kinetic thermal-hydraulic COTRANSA2 code is used for the reactor system analysis, with the XCOBRA/XCOBRA-T codes evaluating the initial and transient hot channel hydraulics and delta-CPR. All of these analyses are performed at the nominal reactor power conditions; the application methodology provides conservatism by accounting for uncertainties in the computed results.

Fuel pellet-to-cladding gap conductance values are evaluated using RODEX2. The ACE/ATRIUM 10XM critical power correlation is used to evaluate the thermal margin for the ATRIUM 10XM fuel. Safety Limit MCPR analyses use the SAFLIM3D code and associated methodology. RODEX4 is used to determine ATRIUM 10XM LHGRFACp multipliers.

The heat transfer coefficients used in the transient analysis are calculated using the NRC-approved code RODEX2 at each cycle exposure where analyses are required. The heat transfer coefficients are calculated for a core average value, used in COTRANSA2, and for a hot channel fuel model, used in XCOBRA-T.

Reload Analysis Scope:

The bounding transients are re-analyzed for each fuel reload and subsequent operating cycle to determine which is most limiting. The results of these specific analyses may change with subsequent core reloads. These results can be found in the appropriate reload licensing document. Events for which a cycle-specific reload analysis are performed are the following:

a. Generator Load Reject (TCV Fast Closure) with Turbine Bypass Valve Failure (LRNBP)
b. Turbine Bypass Valve Failure Following Turbine Trip, High Power (TTNBP)
c. Feedwater Controller Failure Maximum Demand (FWCF)
d. LFWH or Inadvertent Pump Start
e. Continuous Rod Withdrawal During Power Range Operation (RWE)
f. Pump Seizure during Single Loop Operation

If the thermal limits that result from any of the above events are clearly bounded by another event then that event is not analyzed.

14.5.1.3 Reactor Operating Domain

The power/flow map is shown in FSAR Chapter 3. It includes operation in the Maximum Extended Load Line Limit (MELLL) domain, which allows plant operation with core flow as low as 99 percent of rated at 3952 MWt. This boundary maintains the same maximum control rod load line as OLTP operation (i.e., 75 percent core flow at 3293 MWt) and is consistent with the generic guidelines provided in the GE Nuclear Energy Licensing Topical Report, “Constant Pressure Power Uprate,” NEDC-33004P-A, Revision 4, July 2003. The Increased Core Flow (ICF) domain is bounded by the constant recirculation pump speed line corresponding to 105 percent core flow at 100 percent rated power.

14.5.1.4 Reactor Heat Balance

The reactor heat balance defines the thermal-hydraulic parameter input and output within the vessel boundary at a selected core thermal power. These thermal-hydraulic parameters also initialize the conditions assumed for the plant safety analysis. The heat balance at 3952 MWt is shown in FSAR Chapter 1.

A computer program (ISCOR for GE analyses) is utilized to obtain heat balance parameters for operation at 100 and 102 percent power level (for events which require a 2 percent power uncertainty) and other power/flow points considered for transient analyses on the operating domain power/flow map.

14.5.1.5 Reactor Operating Flexibility Features

As previously stated, the BFN operating features include:

1) MELLL and average power range monitor (APRM)/rod block monitor (RBM) Technical Specification (ARTS) Improvements Program [NEDC-32433-P, "Maximum Extended Load Line Limit and ARTS Improvement Program Analyses for Browns Ferry Nuclear Plant Units 1, 2, and 3", April 1995].

2) ICF up to 105 percent of rated core flow individually or combined with Final Feedwater Temperature Reduction (FFWTR), corresponding to a 55 degrees F reduction in feedwater temperature at rated conditions. Additional analysis allows further reductions in feedwater temperature at low power with additional thermal limit penalties. It is not permissible to operate above the specified power with more than 55 degrees F reduction in feedwater temperature. [NEDO-22135, "Safety Review of Browns Ferry Nuclear Plant Unit No. 1 at Core Flow Conditions above Rated Flow During Cycle 5," October 1982; NEDO-22245, "Safety Review of Browns Ferry Nuclear Plant Unit No. 2 at Core Flow Conditions above Rated Flow During Cycle 5," October 1982; NEDO-22149, "Safety Review of Browns Ferry Nuclear Plant Unit No. 3 at Core Flow Conditions above Rated Flow During Cycle 5," June 1982; (b) Safety Review for Browns Ferry Unit 2 Cycle 7 Final Feedwater Temperature Reduction, NEDC-32356P, June 1994; Memo, J. M. Moose to A. W. Will, “Evaluation of Thermal Margin During Startup With Reduced Feedwater Temperature-Phase 2, Revision 1,” AWW:06:077R1, May 16, 2006].

3) Turbine Bypass Out-of-Service (TBP-OOS) [NEDC-32774P, Rev 1, "Safety Analyses for Browns Ferry NP Units 1, 2, and 3. Turbine Bypass and End-of-Cycle Recirculation Pump Trip Out-of-Service", September 1997.

4) End-of-Cycle Recirculation Pump Trip Out-of-Service (EOC-RPT-OOS) [NEDC-32774P, Rev 1, "Safety Analyses for Browns Ferry NP Units 1, 2, and 3. Turbine Bypass and End-of-Cycle Recirculation Pump Trip Out-of-Service", September 1997.

5) 24 Month Fuel Cycle.

6) Main Safety/Relief Valves Setpoint Tolerance Relaxation (±3 percent) and One Main Safety/Relief Valve Out-of-Service (1 MSRV-OOS).

7) Improved Standard Technical Specifications.

8) Limiting transients with PLU-OOS

These operating flexibility options have been included as part of the analyses assumptions for the BFN licensing analyses (NEDC-33860P, Revision 1, “Safety Analysis Report for Browns Ferry Nuclear Plant, Units 1, 2, and 3 Extended Power Uprate,” October 2016, and ANP-3403P, Revision 3, “Fuel Uprate Safety Analysis Report for Browns Ferry Units 1, 2, and 3,” December 2015. The EOC-RPT-OOS mode of operation eliminates the automatic Recirculation Pump Trip signal when Turbine Trip or Load Rejection occurs. As such, the core flow decreases at a slower rate following the recirculation pump trip due to the anticipated transient without scram (ATWS) High Pressure recirculation system trip, thus increasing the severity of the transient responses. This EOC-RPT-OOS option will only be analyzed for the limiting events, LRNBP, TTNBP, and FWCF. These limiting events bound the UFSAR events described in this section, even when it comes to applicability of these equipment OOS (EOOS) and setpoint relaxation options.

The Turbine Bypass Out-of-Service (TBP-OOS) contingency mode of operation produces a different evolution in the pressurization phases of the transients. The overpressurization is faster because the bypass system is not operable, thus the pressure setpoints are reached earlier. However, the positive reactivity insertion due to moderator void collapse is more severe; and this results in a higher delta-critical power ratio (delta-CPR) and, subsequently, a higher operating limit minimum critical power ratio (OLMCPR). The FWCF assumes that turbine bypass system is functional while other limiting transients do not. Consequently, this transient is strongly affected by TBP-OOS.

This option will only be analyzed for the FWCF transient which bounds the UFSAR events described in this section even when it comes to applicability of these EOOS and setpoint relaxation options.

The Main Steam Relief Valve (MSRV) Setpoint Tolerance Relaxation option is assumed for all the transients analyzed as long as the MSRV actuation (mainly for events resulting in a nuclear system pressure increase) results in a more severe transient response with these EOOS and setpoint relaxation options.

The off-rated power dependent MAPLHGR(P) and MCPR(P) limits and the off-rated flow dependent MAPLHGR(F) and MCPR(F) limits for these flexibility options are included in the Core Operating Limits Report (COLR). The current COLR for each BFN unit is included in Appendix B of the corresponding Technical Requirements Manual (TRM). To ensure fuel protection during postulated transients at off-rated power and flow conditions, these calculations include extensive transient analyses at various off-rated state points.

Off-rated power/flow conditions were assumed in the references mentioned above such that the entire power/flow map is bounded by the results obtained for the chosen conditions. Power/flow state points outside the power/flow map were analyzed in order to include extra conservatism in the calculations. Other operating flexibility features as listed above were also included in the transients analyses assumptions. Therefore, the transients analyzed in this chapter are protected by these off-rated limits (MAPLHGR(P), MCPR(P), MAPLHGR(F), and MCPR(F)) in the entire power flow map domain.

For reload analyses, off-rated thermal limits are calculated on a cycle-specific basis.

14.5.1.6 Transient Input Parameters

The range of system input parameters for transient analysis mainly consist of heat balance information, core characteristics, and reactor protection specifications. The inputs include the initial power and flow conditions, core pressure drop, void fraction, nuclear dynamic parameters (Doppler, void and scram reactivity coefficients), and plant operating configuration (such as scram speed, safety/relief valves setpoints, reactor scram setpoints, recirculation/feedwater pump trip).

Table 14.5-2 shows the analysis basis values of key parameters of BFN operation.

14.5.1.7 Transient Power/Flow/Exposure Conditions

The following rated thermal power and core flow conditions from the BFN power/flow map are selected as representative for the standard (STD), MELLL, and ICF regions:

1. 100P/99F (MELLL domain)
2. 100P/100F (standard domain)
3. 100P/105F (ICF domain)

The 100P is defined as 100 percent of rated power or 3952 MWt. The 100F is defined as 100 percent of rated core flow or 102.5 E6 lbm/hr; 99F and 105F are defined as 99 percent and 105 percent of rated core flow, respectively. As previously discussed, some transients are analyzed at 4031 MWt or 102 percent of rated power (i.e., 102P).

The UFSAR transient analyses have considered the full spectrum of core conditions from the beginning, middle, and end of the cycle (BOC, MOC, EOC), whichever is more limiting for the transient event under consideration. A bounding 24-month fuel cycle length is also included in the cycle exposure calculations.

14.5.2 Events Resulting in a Nuclear System Pressure Increase

Events that result directly in significant nuclear system pressure increases are those that result in a sudden reduction of steam flow while the reactor is operating at power. A survey of the plant systems has been made to identify events within each system that could result in the rapid reduction of steam flow. The following events were identified:

a. Generator Load Reject
b. Loss of Condenser Vacuum
c. Turbine Trip
d. Turbine Bypass Valve Malfunction
e. Closure of Main Steam Isolation Valve
f. Pressure Regulator Malfunction

14.5.2.1 Generator Load Reject (Turbine Control Valve [TCV] Fast Closure)

14.5.2.1.1 Transient Description

A loss of generator electrical load from high power conditions produces the following transient sequence:

a. Turbine-generator power/load unbalance circuitry operates the control valve fast acting solenoid valves to initiate turbine control valve (TCV) fast closure (minimum response time of TCV fast closure: 0.15 seconds),

b. Turbine control valve fast closure is sensed by the reactor protection system, which initiates a scram and simultaneous recirculation pump trip (for initial power levels above 26 percent rated),

c. The turbine bypass valves are opened simultaneously with turbine control valve closure, and reroute the vessel steam flow to the condenser.

d. Reactor vessel pressure rises to the MSRV setpoints, causing them to open for a short period of time.

e. The steam passed by the MSRVs is discharged into the suppression pool, and

f. The turbine bypass valve (TBV) system controls nuclear system pressure after the MSRVs close.

Below 26 percent of rated power, the TBV system will transfer steam around the turbine and thereby avoid reactor scram. This transient is not analyzed as it is bounded by the Generator Trip (TCV Fast Closure) With Turbine Bypass Valve Failure transient described in Section 14.5.2.2.

14.5.2.2 Generator Load Reject (TCV Fast Closure) with Turbine Bypass Valve Failure (LRNBP)

14.5.2.2.1 Transient Description

The most severe transient for a full-power generator trip occurs if the turbine bypass valves fail to operate. Although the TCV fast closure time is slightly longer than that of the turbine stop valves, the control valves are considered to be partially closed initially. This results in the generator trip steam supply shutoff being faster than the turbine stop valve steam shutoff.

A generator trip from high power conditions produces a transient sequence similar to the sequence described in Section 14.5.2.1 except the turbine bypass valves are assumed to remain closed. The LRNBP event is caused by the fast closure of all turbine control valves (TCVs) due to significant loss of electrical load on the generator. This will cause a sudden reduction in steam flow that results in significant vessel pressurization. The turbine bypass system is conservatively assumed to be inoperable for this event. A reactor scram signal is initiated by the TCVs closure.

The LRNBP event is identified as one of the most limiting abnormal operational transients for the BFN licensing analyses (assuming all equipment in service). Therefore, this event is analyzed to determine the operating limits and to verify the plant safety margins.

This abnormal operating transient is evaluated for each reload core to determine if this event could potentially alter the previous cycle MCPR operating limit. The analyses of this event for the most recent reload cycle is contained in the unit-specific and cycle-specific Reload Licensing Report.

14.5.2.2.2 Initial Conditions and Assumptions

The reload analysis described in this section was performed with the AREVA models/computer codes COTRANSA2, XCOBRA, XCOBRA-T, RODEX2, and RODEX4, at the limiting power/flow conditions at normal operation: 100 percent rated power (consistent with the current licensing methodology) and maximum core flow (ICF) conditions. For bounding purposes, normal feedwater temperature (as opposed to reduced feedwater temperature) is assumed since the reactor steam generation would be lower with a reduced feedwater temperature.

14.5.2.2.3 Interpretation of Transient Results

Figures 14.5-5a, b, and c show the plant-specific response to the generator load rejection without bypass at 100 percent rated power and 105 percent flow conditions. The neutron flux peaks at 333 percent of rated; the average heat flux peaks at 124 percent of rated. The peak pressure at the bottom of the vessel is 1321 psia which is well below the ASME upset code transients limit of 1375 psig while the peak steam line pressure is 1329 psia. The calculated delta-CPR at the stated conditions is 0.31 for AREVA fuel; this result is representative but not bounding for other fuel types.

At rated power, the delta-CPR for the LRNBP event is one of the most severe resulting from any other pressurization event. As power is reduced, the severity of the transient increases; but the fuel integrity is protected by the power-flow dependent thermal limits (see Section 14.5.8).

14.5.2.2.4 Generator Load Reject with Turbine Bypass Valve Failure with EOC-RPT-OOS

The EOC-RPT-OOS condition eliminates the automatic Recirculation Pump Trip signal when Load Rejection occurs increasing the severity of the transient response. At power levels below 26 percent of rated power (P_{bypass}), the RPT is always bypassed in conjunction with the scram on TSVs/TCVs closure. Therefore, these low power cases are not affected by the EOC-RPT-OOS condition.

Figures 14.5-6a, b, and c show the transient results for the 100 percent of rated power and 105 percent of rated core flow case. EOC exposure and normal feedwater temperature conditions have been assumed for this transient analysis.

The neutron flux peaks at 392 percent of rated, the average heat flux peaks at 130 percent of rated. The peak pressure at the bottom of the vessel is 1339 psia which is well below the ASME upset code transients limit of 1375 psig while the peak steam line pressure is 1304 psia. The calculated delta-CPR of this transient at the stated conditions is 0.38.

The penalty associated with EOC-RPT-OOS is about 0.07 in delta-CPR. At less than rated core flow, the penalty is smaller because of the relatively reduced beneficial effect of EOC-RPT.

Power-dependent LHGR(P) multipliers are used. Power and flow dependent MAPLHGR, LHGR, and MCPR limits are developed on a cycle-specific basis.

14.5.2.3 Loss of Condenser Vacuum (LCV)

14.5.2.3.1 Transient Description

This case is a severe abnormal operational transient resulting directly in a nuclear system pressure increase. It represents the events that would follow an assumed instantaneous loss of vacuum; main and feedwater turbines trip when their vacuum protection setpoints are reached, main turbine trip (TT) initiates reactor scram, recirculation pump trip (RPT), and turbine bypass opening. Later in the transient, the condenser vacuum is assumed to drop to the setpoints for closure of TBVs. This event is bounded by the TTNBP event.

14.5.2.4 Turbine Trip (TSV Closure)

14.5.2.4.1 Transient Description

A turbine trip is the result of a turbine or reactor system malfunction which results in a TSV fast closure (0.1 second closure time). This event represents a fast steam flow shutoff; and therefore, the potential for one of the most severe pressure-induced transients. Position switches on the stop valves provide the means of sensing the trip and initiating immediate reactor scram (for initial power levels above 26 percent of 3952 MWt). The bypass valves are opened by the control system upon a turbine trip. The bypass system regulates reactor pressure during reactor shutdown.

Although the TCV fast closure time is slightly longer (0.15 second) than that of the TSV (0.1 second), the control valves are considered to be partially closed initially. This results in the generator trip steam supply shutoff being faster than the turbine stop valve steam shutoff while the protection system response is almost the same for each case (see Section 14.5.2.1).

This event is bounded by the TTNBP event.

14.5.2.5 Turbine Bypass Valves Failure Following Turbine Trip, High Power (TTNBP)

14.5.2.5.1 Transient Description

This event is included to illustrate that a single failure could prevent the turbine bypass valves from opening in conjunction with a turbine trip.

The turbine trip with no bypass (TTNBP) event is similar to the LRNBP event. Even though the TTNBP has been shown to be bounded by the LRNBP, it is analyzed in the UFSAR for completeness.

14.5.2.5.2 Initial Conditions and Assumptions

The calculation of this transient has been performed with the AREVA models/computer codes COTRANSA2, XCOBRA, XCOBRA-T, RODEX2, and RODEX4, at the most limiting conditions: 100 percent of rated power, 105 percent of rated core flow, EOC exposure conditions, and normal feedwater temperature. The EOC exposure has been used because the top peaked axial power shape degrades the effectiveness of rod insertion during the reactor scram. The turbine bypass system is assumed to be inoperable.

14.5.2.5.3 Interpretation of Transient Results

Figures 14.5-9a, b, and c illustrate this transient. This transient evolves in a similar way to the TTBP event, although the bypass failure produces a more severe transient. Peak neutron flux reaches 332 percent of rated power while peak heat flux reaches 124 percent of rated power. Peak steam line pressure and peak vessel pressure reach 1329 and 1321 psia, respectively.

The results show a delta-CPR of 0.31 for this event. This trend is similar to that observed above for the LRNBP event. The TTNBP event is bounded by the LRNBP event. The TTNBP event is reanalyzed each cycle to evaluate the required core thermal operating limits (i.e., MCPR).

14.5.2.6 (Deleted)

14.5.2.7 Main Steam Isolation Valve (MSIV) Closure

Automatic circuitry or operator action can initiate closure of the main steam isolation valves. Position switches on the valves provide reactor scram if valve(s) in three or more main steam lines are less than 90 percent open, and the mode switch is in the Run position. However, protection system logic does permit the test closure of one valve without initiating scram from the position switches. Inadvertent closure of one or all of the isolation valves from reactor scrammed conditions (such as Appendix G) will produce no significant transient. Closures during plant heatup will be less severe than the maximum power cases (maximum stored and decay heat) which follow.

14.5.2.7.1 Closure of All Main Steam Isolation Valves

14.5.2.7.1.1 Transient Description

This transient represents the simultaneous isolation of all MSIVs while the reactor is operating at power. Reactor scram is initiated by the MSIVs position switches before the valves have traveled more than 10 percent from the initial open position. The closure of all MSIVs causes an abrupt pressure increase in the reactor vessel. The system pressure increase is mitigated by the actuation of the MSRVs.

The closure of all MSIVs event with direct scram failure (reactor scram on high neutron flux signal) is the design basis event to demonstrate compliance to the ASME vessel overpressure protection criteria (upset condition). The MSIVF (Flux Scram) is included in every cycle-specific reload licensing process to ensure that the ASME code allowable value for peak vessel pressure (1375 psig) is not exceeded.

14.5.2.7.1.2 Initial Conditions and Assumptions

This transient has been run with the CONTRANSA2 computer code at 102 percent power, 105 percent core flow, normal feedwater temperature, EOC exposure conditions, and 1 MSRV-OOS. The EOC exposure has been used because the top peaked axial power shape degrades the effectiveness of rod insertion during the reactor scram.

The MSIV closure event is analyzed with 12 out of 13 MSRVs in-service (with one of the MSRVs with lowest opening setpoint assumed out-of-service) and 3 percent setpoint tolerance. The reduced relief capacity also increases the severity of the reactor vessel pressure transient. The fastest MSIV closure curve has been considered for this analysis (3 second closure time) which represents the bounding closure characteristics.

14.5.2.7.1.3 Interpretation of Transient Results

Figures 14.5-11a, b, c, and d illustrate the transient results. Scram is initiated very early into the event, before any significant steam flow interruption occurs; therefore, no fuel center temperature or fuel surface heat flux peaks take place. A small neutron flux peak occurs near 0.5 seconds. All 12 operable MSRVs open when pressure reaches the lowest setpoint at about 4 seconds after the start of the isolation. They close sequentially as the stored heat is being dissipated and continues to intermittently discharge the decay heat. The fuel delta CPR resulting from this event is bounded by other more limiting pressurization event, such as the TTNBP event.

The calculated peak bottom vessel pressure is 1349 psig for BFN specific MSIV closure characteristics and is still below the 1375 psig ASME overpressure limit.

14.5.2.7.2 Closure of One Main Steam Isolation Valve

14.5.2.7.2.1 Transient description

Full closure of only one isolation valve without scram is permitted for testing purposes. Normal procedures for such a test will normally require an initial power reduction to less than or equal to 75 percent in order to avoid high flux or pressure scram or high steam flow isolation from the active steam lines. During the transient from full power, the steam flow disturbance may raise vessel pressure and reactor power resulting in a high neutron flux scram.

14.5.2.7.2.1 Initial Conditions and Assumptions

This transient has been analyzed with CONTRANSA2, XCOBRA, and XCOBRA-T at 100 percent of rated power, 105 percent of rated core flow (ICF conditions), and BOC exposure.

A value of 62.7 psid pressure drop in the steam line is assumed in the analysis. An increase in the steam line pressure drop has a small impact on the results and does not require a re-analysis of this event as long as this event remains a non limiting transient.

14.5.2.7.2.1 Interpretation of Transient Results

Figures 14.5-12a, b, and c illustrate this transient. The steam flow disturbance raises vessel pressure and reactor power; the peak neutron flux reaches 121 percent of rated. The peak surface heat flux reaches about 107 percent of rated. Peak steam line pressure (1082 psia) remains below the setting of the lowest MSRVs. Peak vessel pressure (1138 psia)

remains below the 1375 psig ASME overpressure limit. The peak fuel parameters are well below those from the limiting pressurization transient (LRNBP).

14.5.2.8 Pressure Regulator Failure

Approval to remove the pressure regulator downscale failure (PRDF) event as an abnormal operational transient was approved by license Amendment Nos. 244, 281, and 239 to Facility Operating Licenses Nos. DPR-33, DPR-52, and DPR-68 by NRC on April 4, 2003, based on the installation of a fault-tolerant electro-hydraulic turbine control system on Units 2 and 3, and a commitment to similarly modify Unit 1 prior to return to power operation. The reliability of the upgraded electro-hydraulic control system is such that a system failure that results in the simultaneous closure of all turbine control valves is not an anticipated failure and, hence, the PRDF transient no longer merits evaluation as an AOT.

14.5.3 Events Resulting in a Reactor Vessel Water Temperature Decrease

Events that result directly in a reactor vessel water temperature decrease are those that either increase the flow of cold water to the vessel or reduce the temperature of water being delivered to the vessel. The events that result in the most severe transients in this category are the following:

a. Loss of a Feedwater Heater
b. Shutdown Cooling (RHR) Malfunction - Decreasing Temperature
c. Inadvertent pump start

The most limiting conditions for these type of transients have been assumed, i.e. 100 percent of rated power and 99 percent of rated flow (MELLL conditions). Normal feedwater temperature is also assumed as the larger void coefficient produces a more severe transient.

14.5.3.1 Loss of Feedwater Heater (LFWH)

14.5.3.1.1 Transient Description

The purpose of evaluating this event is to determine the impact on the delta-CPR and on the fuel thermal and mechanical design limits. The LFWH event for BFN assumes a feedwater temperature reduction of 100 degrees F (from 394.8 degrees F to 294.8 degrees F).

The LFWH transient may be initiated by the accidental closure of the feedwater steam extraction shut-off valves or by bypassing feedwater around the feedwater heater. In either case, the feedwater temperature falls below its rated value; therefore, increasing the subcooling to the reactor core. The negative void reactivity coefficient results in an increase in core power, change in power distribution, and

decrease in bundle CPR. In the first case, a gradual subcooling transient is produced since there is stored heat in the heat exchanger. In the second case, a more abrupt subcooling transient is initiated due to the instantaneous removal of all feedwater heating. The maximum feedwater temperature loss (100 degrees F) due to a single equipment failure is the worst condition analyzed for BFN using this procedure.

Analyses performed for AREVA cores utilize the NRC approved methodology of ANF-1358(P)(A) Revision 3, “The Loss of Feedwater Heating Transient in Boiling Water Reactors.” This approved methodology is based on the use of MICROBURN reactor simulator code due to the quasi steady-state nature of the event.

For AREVA cores, the NRC approved methodology of ANF-1358(P)(A), “The Loss of Feedwater Heating Transient in Boiling Water Reactors,” is utilized to evaluate the LFWH event for each reload. This methodology uses results obtained from the NRC approved MICROBURN-B and MICROBURN-B2 3-D reactor simulator codes. The LFWH results are included in the Reload Analysis Report for each operating cycle. Previous cycle-specific AREVA analyses have been performed at 3458 MWt including fuel types through ATRIUM-10XM (also including ATRIUM-11 LTAs in the Unit 2 core). Analyses have also been performed supporting extended power uprate at 3952 MWt. Similar to the original GE analyses, the delta-CPR in all cases remains significantly nonlimiting. ANP-3403P Revision 3, “Fuel Uprate Safety Analysis Report for Browns Ferry Units 1, 2, and 3,” included as part of the EPU license amendment request, reported the following rated power results for the current ATRIUM 10XM fuel design.

Description	Power Level (MWt)	ΔCPR
LFWH – CLTP Rated Power	3458	0.14
LFWH – EPU Rated Power	3952	0.13

Therefore based upon the above, the LFWH is not a significant threat to fuel thermal margins; the Operating Limit CPR is established by other more limiting transients.

The average power range monitors provide an alarm to the operator at about 20 seconds after the cooler feedwater reaches the reactor vessel. Because nuclear system pressure remains essentially constant during this transient, the nuclear system process barrier is not threatened by high internal pressure. All fuel parameters remain bounded by the results of other limiting pressurization transients.

For AREVA cores, the approved methodology results in higher delta-CPRs for off-rated cases. Although the calculated percentage increase in power may be similar to the rated power case the higher CPR margin and lower initial void content in the core at offrated conditions results in the larger calculated change in CPR. This is addressed with power dependent delta-CPRs that are provided in the Reload Analysis Report each cycle. Even though the delta-CPRs increase with decreasing initial power, the LFWH event remains non-limiting versus the pressurization transients at off-rated power conditions.

14.5.3.2 Shutdown Cooling (RHR) Malfunction-Decreasing Temperature

A shutdown cooling malfunction leading to a moderator temperature decrease could result from misoperation of the controls for the Residual Heat Removal (RHR) system heat exchangers. The resulting temperature decrease causes a slow insertion of positive reactivity into the core. If the reactor were critical or near critical, a very slow reactor power increase could result. If no operator action were taken to control the power level, a high neutron flux reactor scram would terminate the transient without fuel damage and without any measurable nuclear system pressure increase.

14.5.3.3 Inadvertent Pump Start

14.5.3.3.1 Transient Description

Several systems are available for providing high pressure supplies of cold water to the vessel for normal or emergency functions. The control rod drive system and the makeup water system, normally in operation, can be postulated to fail in the high flow direction introducing the possibility of increased power due to higher core inlet subcooling. The same type of transient would be produced by inadvertent startup of either the reactor core isolation cooling (RCIC) or the high pressure core injection (HPCI) System. In all of these cases, the normal feedwater flow would be correspondingly reduced by the water level controls. A portion of the feedwater flow (at rated power condition) is replaced with a colder HPCI flow, and the net result is a mixed feedwater flow at a reduced temperature.

Since a single failure can only initiate one of the cold water systems, the system with the highest flow rate is usually analyzed. The severity of the resulting transient is highest for the largest of these abnormal events; for BFN, this is the inadvertent startup of the large, 5000 gpm, HPCI System.

This transient is evaluated to determine the MCPR response to a decrease in feedwater temperature due to the inadvertent startup of the HPCI system. This event is qualitatively reviewed as part of the reload licensing analysis to verify its non-limiting trend versus the cycle specific operating limits.

Since the startup of the steam-turbine driven pump takes approximately 25 seconds, the transient that occurs is very similar to the loss of feedwater heater transient described above. As in that case, the most threatening transient would occur where minimum initial fuel thermal margins exist during power operation.

14.5.3.3.2 Initial Conditions and Assumptions

This transient was analyzed by AREVA using the COTRANSA1/XCOBRA/XCOBRAT methodology. This analysis considered various cycle exposures and initial core flows.

As explained above, the inadvertent startup of the large, 5000 gpm, HPCI has been considered. The HPCI pump setpoint was conservatively assumed to be higher than this value to ensure that the actual HPCI setpoint is bounded. During the initiation and acceleration transient for the HPCI, the pump flow can overshoot the rated flow making the event more severe. The water temperature of the HPCI was assumed to be 40ºF.

The system was assumed to be in manual flow control, which results in higher flux, pressure and level peaks.

For an inadvertent HPCI start, the water level may rise to the L8 setpoint. All logic associated with this setpoint such as turbine, feedwater, HPCI trips, and RPT/ATWS options was considered.

14.5.3.3.3 Interpretation of Transient Results

The introduction of subcooled water due to the inadvertent HPCI startup causes an increase in reactor power, neutron and surface heat fluxes. Pressure and water level show a small increase. The power increase raises turbine steam flow slightly. The flux scram setpoint is not reached during this event.

The plant eventually reaches a steady state condition at an increased power level but with no significant threat to the fuel thermal margins. The AREVA analysis found the delta-CPR result for inadvertent HPCI startup was 0.13. Therefore, the event has considerable margin to the limiting generator load rejection without bypass and feedwater controller failure events. No fuel clad barrier damage results for the malfunction or inadvertent startup of HPCI or other auxiliary cold water supply systems.

14.5.4 Events Resulting in a Positive Reactivity Insertion

Events that result directly in positive reactivity insertions are the results of rod withdrawal errors and errors during refueling operations. The following events result in a positive reactivity insertion:

a. Continuous Rod Withdrawal During Power Range Operation
b. Continuous Rod Withdrawal During Reactor Startup
c. Control Rod Removal Error During Refueling
d. Fuel Assembly Insertion Error During Refueling

14.5.4.1 Continuous Rod Withdrawal During Power Range Operation

14.5.4.1.1 Transient Description

The RWE event is initiated by an operator erroneously selecting and continuously withdrawing a single high worth control rod.

Control rod withdrawal errors are considered over the entire power range from any normally expected rod pattern. The continuous withdrawal from any normal rod pattern of the maximum worth rod results in a very mild core transient. The system will stabilize at a higher power level with neither fuel damage nor nuclear system process barrier damage.

The limiting control rod withdrawal error during power range operation is examined each reload cycle. NRC approved methodology CASMO-4/MICROBURN-B2 is used for licensing analysis performed by AREVA. The result is presented in the Reload Licensing Report.

As part of the RBM system modification included in the ARTS Improvement program [NEDC-32433-P, "Maximum Extended Load Line Limit and ARTS Improvement Program Analyses for Browns Ferry Nuclear Plant Units 1, 2 and 3", April 1995 and "Browns Ferry Units 1, 2, and 3 Extended Power Uprate Task Report - Transient Analysis," FS1-0019595, Revision 1, May 2015, the fuel thermal-mechanical protection for a postulated RWE event is provided by the RBM power-dependent setpoints. The RWE event is re-analyzed every cycle to confirm the applicability of these ARTS generic limits and to verify the thermal-mechanical protection.

14.5.4.1.2 Initial Conditions and Assumptions

The core nuclear dynamic parameters are based on the cycle peak hot excess reactivity, and the control rod pattern used to simulate the RWE are assumed to be at nominal conditions. The analysis assumes the error rod is withdrawn

continuously from its initial position. During this event, the core average power increases until the event is terminated by a rod block signal.

14.5.4.1.3 Interpretation of Transient Results

A cycle specific RWE analysis is performed by AREVA. The RWE analysis is a bounding analysis that evaluates the withdrawal of maximum reactivity worth rods with conservative starting control rod patterns. The starting control rod patterns are conservatively selected to place the fuel near the fully inserted error rod at or near thermal limits. The analysis assumes that the reactor operator ignores the LPRM and RBM alarms and continues to withdraw the error rod until the motion is stopped by the RBM trip. The RBM trip setpoints for the cycle are selected to ensure that the RWE is not limiting compared to the limiting plant transients. The power dependent RMB trip setpoints are documented in the cycle specific COLR.

14.5.4.2 Continuous Rod Withdrawal during Reactor Startup

14.5.4.2.1 Transient Description

Control rod withdrawal errors are considered when the reactor is at a power level below the power range involving the startup range of the power/flow operating map. The most severe case occurs when the reactor is just critical at room temperature, and an out-of-sequence rod is continuously withdrawn. The rod worth minimizer would normally prevent withdrawal of such a rod. It is assumed that the Intermediate Range Neutron Monitoring (IRM) channels are in the worst conditions of allowed bypass. The scaling arrangement of the IRMs is such that for unbypassed IRM channels a scram signal is generated before the detected neutron flux has increased by more than a factor of ten. In addition, a high neutron flux scram is generated by the APRM at 13 percent and at 120 percent of rated power depending on the initial power level.

The original licensed thermal power (3293 MWt) UFSAR analysis was performed for a 2.5 percent delta-k control rod withdrawal at the rod drive speed of 0.3 ft/sec starting from an average moderator temperature of 82 degrees F.

The results of these analyses indicate a maximum fuel temperature well below the melting point of UO_2 and a maximum fuel clad temperature which is less than the normal operating temperature of the clad. The possible failure of the fuel clad due to strain was analyzed using the following conservative assumptions:

1 The total volume expansion of UO_2 is in the radial direction,
2. There is no thermal expansion of the fuel cladding, and
3. The fuel is assumed to be incompressible.

The results of these analyses indicate a maximum radial strain analogous to a radial expansion of 0.6 mils, which is much less than the postulated cladding damage limit of 1 percent plastic strain, which corresponds to 3.3 mils radial expansion.

Thus, no fuel damage will occur due to a continuous rod withdrawal during reactor startup.

The Continuous Rod Withdrawal during Reactor Startup transient does not need to be re-analyzed for 3458 MWt or 3952 MWt rated power, as the licensing basis criteria for fuel failure is that the fuel enthalpy must not exceed 170 cal/gm. At the above power levels, it is possible that a slightly higher fuel enthalpy above 60 cal/gm (reported in the previous analysis) can be reached due to the higher enrichment or other changes; but due to the considerable margin that exists to the 170 cal/gm limit, the result will be well below 170 cal/gm should a new analysis be performed. There existed several conservatisms in the original design basis analysis, such as:

1. The furthest possible distance between a control rod being withdrawn and a scram initiating IRM detector is used.
2. The rod shape function depicts the control rod being withdrawn at 0.3 ft/sec until the entire rod is withdrawn, but, in reality, the rod is withdrawn only 2.44 feet before the scram starts to reinsert the rod.
3. The RBM is assumed to fail to block the continuous withdrawal of an out-of-sequence rod.
4. No power flattening due to Doppler feedback is assumed.

Therefore, a re-analysis is not needed for the UFSAR at the rated power levels of 3458 or 3952 MWt.

14.5.4.3 Control Rod Removal Error During Refueling

The nuclear characteristics of the core ensure that the reactor is subcritical even in its most reactive condition with the most reactive control rod fully withdrawn during refueling.

When the mode switch is in Refuel, only one control rod can be withdrawn. Selection of a second rod initiates a rod block, thereby preventing the withdrawal of more than one rod at a time.

Therefore, the refueling interlocks will prevent any condition which could lead to inadvertent criticality due to a control rod withdrawal error during refueling when the mode switch is in the Refuel position.

In addition, the design of the control rod, incorporating the velocity limiter, does not physically permit the upward removal of the control rod without the simultaneous or

prior removal of the four adjacent fuel assemblies, thus, eliminating any hazardous condition.

14.5.4.4 Fuel Assembly Insertion Error During Refueling

The core is designed such that it can be made subcritical under the most reactive conditions with the strongest control rod fully withdrawn. Therefore, any single fuel assembly can be positioned in any available location without violating the shutdown criteria, providing all the control rods are fully inserted. The refueling interlocks require that all control rods must be fully inserted before a fuel bundle may be inserted into the core.

14.5.5 Events Resulting in a Reactor Vessel Coolant Inventory Decrease

Events that result directly in a decrease of reactor vessel coolant inventory are those that either restrict the normal flow of fluid into the vessel or increase the removal of fluid from the vessel. Four events have been considered in this category:

a. Pressure Regulator Failure Open
b. Inadvertent Opening of a MSRV
c. Loss of Feedwater Flow
d. Loss of Auxiliary Power

Normal feedwater temperature and minimum reactor water level have been assumed for these types of transients. The smaller initial water inventory in the vessel and the larger steam flow maximizes the inventory loss.

14.5.5.1 Pressure Regulator Failure Open

14.5.5.1.1 Transient Description

Should the pressure regulation function of the Turbine Control System fail in an open direction, the turbine admission valves can be fully opened with the turbine bypass valves partially or fully opened. This condition results in an initial decrease in the coolant inventory in the reactor vessel as the mass flow of steam leaving the vessel exceeds the mass flow of water entering the vessel. The total steam flow rate resulting from a pressure regulation malfunction is limited by the turbine controls to about 130 percent of rated flow.

The reactor water level swelling due to the decreasing reactor vessel pressure may reach the high level L8 setpoint initiating a TSV closure. Following this action, feedwater pumps trip, recirculation pumps trip, and reactor scram will take place. If L8 is not reached, the vessel depressurizes and the turbine header pressure may drop to the low pressure setpoint for reactor isolation (843 psig); the MSIVs will then close, and a reactor scram will be initiated.

There is no significant threat to the fuel thermal margins, but there is a small but rapid decrease in the saturated temperature to which the reactor system components are exposed, which might affect the hardware components.

14.5.5.1.2 Initial Conditions and Assumptions

Per Reference 2, AREVA has concluded that this event is non-limiting with respect to CPR limits. The analysis in Reference 2 evaluates, using the CONTRANSA2 code, if the LPIS for the MSIVs is adequate to support the low steam dome pressure safety limit being maintained during the time that the reactor is above 25% rated themal power during the PRFO event.

The Reference 2 analysis considers sensitivities to multiple power/flow statepoints, variations in feedwater temperature and dome pressure, MSIV closure times between 3.0 and 5.0 seconds, multiple cycle exposure values, multiple scram insertion speeds, and variations in core average gap conductance.

The physical LPIS setpoint for the MSIVs has been set to 843 psig, which will result in a less severe depressurization than the allowed value of 825 psig.

14.5.5.1.3 Interpretation of Transient Results

The analysis in Reference 2 yields the following results:

Initial conditions of low core flow are more conservative than high core flow.
Lower feedwater temperatures (feedwater heaters out-of-service) and the corresponding lower dome pressure values are conservative.
Slow MSIV closure time (5 seconds vs. 3 seconds) is conservative.
Minimum pressure during the PRFO event is relatively insensitive to cycle exposure.
Faster scram times provide a lower minimum steam dome pressure during the event.
High core average gap conductance provides a lower minimum steam dome pressure during the event.

Results from the Reference 2 case yielding the lowest steam dome pressure values are shown in Figure 14.5-15a and 14.5-15b.

Reference 2 concludes that the lowest pressure calculated for BFN does not change the low pressure safety limit value of 585 psig. The Reference 2 evaluation assumed a rated power level of 3458 MWt. The results of the evaluation demonstrated that the initial reactor power that produces the limiting minimum vessel pressure during the depressurization is an intermediate power level of 60% of the 3458 MWt value. Higher initial power levels were shown to produce a less limiting vessel pressure results. An evaluation of the event for Extended Power Uprate (EPU) (i.e., 3952 MWt) was performed in Reference 3. The evaluation for EPU

concluded that the limiting minimum pressure value results from intermediate initial power levels, having comparable margins to the low pressure safety limit as the Reference 2 non-EPU evaluation.

References:

1. GE 10 CFR Part 21 Communications SC05-03, “Potential to Exceed Low Pressure Technical Specification Safety Limit,” GE Nuclear, March 2005.

2. ANP-32459 Revision 1, “Browns Ferry Evaluation of PRFO Low Pressure Tecnical Specification Value,” AREVA NP, February 2014.

3. FS1-0019612, “Browns Ferry EPU FUSAR - Dispositions,” Revision 1, AREVA NP.

14.5.5.2 Inadvertent Opening of a MSRV (IORV)

14.5.5.2.1 Transient Description

The opening of a MSRV on the main steam line allows steam to be discharged into the primary containment. The sudden increase in the rate of steam flow leaving the reactor vessel causes the reactor vessel coolant (mass) inventory to decrease. The result is a mild depressurization transient. The turbine pressure regulator senses the pressure decrease and drops turbine flow to maintain pressure control. The reactor settles out at nearly the initial power.

The peak heat flux does not exceed the initial power and no specified acceptable fuel design limits are challenged. Therefore, this event is non-limiting relative to thermal operating limits.

Because pressure decreases throughout the transient, the nuclear system process barrier is not threatened by high internal pressure. The small amounts of radioactivity discharged with the steam are contained inside the primary containment; the situation is not significantly different, from a radiological viewpoint, from that normally encountered in cooling the plant using the relief valves to remove decay heat.

14.5.5.3 Loss of Feedwater Flow

A loss of feedwater flow results in a situation where the mass of steam leaving the reactor vessel exceeds the mass of water entering the vessel, resulting in a net decrease in the coolant inventory available to cool the core.

For this 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 to support operation at 3952 MWt. 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.

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 3952 MWt. 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.

This transient has been run by AREVA with the COTRANSA2 code. The following is the general sequence of events in the analysis. The reactor is assumed to be at 4031 MWt 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 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 analysis was ANS 5.1-1979 decay heat +10%, which bounds ANS 5.1-1979 + two sigma.

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 also show that the nominal Level 1 setpoint trip is avoided.

14.5.5.4 Loss of Auxiliary Power

Loss of auxiliary power is defined as an event which de-energizes all electrical buses that supply power to the unit auxiliary equipment such as recirculation, feedwater, and condenser circulating water pumps. The reactor is subjected to a complex sequence of events when the plant loses all auxiliary power. This can occur if all external grid connections are lost or if faults occur in the auxiliary power system itself causing, therefore, two types of transients: Loss of Auxiliary Power Transformers and Loss of Auxiliary Power Grids.

14.5.5.4.1 Transient Description

Estimates of the responses of the various reactor systems to loss of auxiliary power provided the following simulation sequence:

a. All pumps are tripped at 0 seconds. Normal coastdown times were used for the recirculation and feedwater pumps.

b. At 5 seconds, the reactor protection system instrumentation power is lost. This initiates closure of the MSIVs which also produces a scram signal.

The trip of the main condenser circulating water pumps causes the loss of the condenser vacuum. When vacuum protection setpoints are reached, turbine trip and closure of the TBVs take place.

For the Loss of Auxiliary Power Grids, the same sequence as above would be followed except that the reactor would also experience a generator load rejection and its associated scram at the beginning of the transient.

By about 20 seconds after the simulated loss of power, both transients look essentially identical. Pressure is cycling at approximately the lowest MSRV setpoint, and water level is dropping gradually until RCIC (or HPCI) operation restores water level control. The long-term water level transient is bounded by the Loss of Feedwater Flow long-term water level transient analyzed in Section 14.5.5.3. The delta-CPR and vessel pressure for these events are bounded by the LRNBP event.

14.5.6 Events Resulting in a Core Coolant Flow Decrease

Events that result directly in a core coolant flow decrease are those that affect the reactor recirculation system. Transients beginning from operating state F are the most severe since only in this state do power levels approach fuel thermal limits. The following events have been analyzed:

a. Recirculation Flow Control Failure-Decreasing Flow
b. Trip of One Recirculation Pump

c. Trip of Two Recirculation Pumps
d. Recirculation Pump Seizure

14.5.6.1 Recirculation Flow Control Failure - Decreasing Flow

14.5.6.1.1 Transient Description

VFD Speed Control:

Several varieties of recirculation flow control malfunctions can cause a decrease in core coolant flow. The manual runback controller could malfunction in such a way to continually command both VFDs to decelerate at the normal runback rate until both pumps are stopped. This event is less severe than the simultaneous tripping of both recirculation pumps as evaluated in paragraph 14.5.6.3.

The remaining recirculation flow controller malfunction is one in which a single flow controller fails and applies a braking action to a single recirculation pump. The pump speed reduction is slower than a recirculation pump seizure as evaluated in paragraph 14.5.6.4.

14.5.6.1.2 Initial Conditions and Assumptions

In support of AREVA's introduction of ATRIUM-10 fuel at 3458 MWt conditions, this event was analyzed with COTRANSA2/XCOBRAT for TLO conditions at 100 percent of rated power, 105 percent of rated core flow and at SLO conditions at 82 percent of rated power, 58 percent of rated core flow

14.5.6.1.3 Interpretation of Transient Results

VFD speed control:

For the AREVA analysis performed at 3458 MWt conditions, the peak neutron and heat fluxes do not increase above initial conditions. The calculated delta-CPR is 0.03, well below that for other types of transients analyzed; therefore, no impact on fuel integrity occurs.

For the AREVA analysis performed at SLO 3458 MWt conditions, the peak neutron and heat fluxes do not increase above initial conditions. The calculated delta-CPR is 0.19, well below that for other types of transients analyzed; therefore, no impact on fuel integrity occurs.

Since the peak neutron and heat fluxes do not increase above initial conditions and no impact on fuel integrity occurs, AREVA has qualitatively dispositioned this event as non limiting at 3952 MWt.

14.5.6.2 Trip of One Recirculation Pump

14.5.6.2.1 Transient Description

Normal trip of one VFD driven recirculation loop is accomplished through trip of the VFD or VFD supply breaker. Coastdown with only pump and motor inertia occurs. An abrupt reduction in core flow due to the trip of one of the recirculation pumps increases the core void fraction and, thereby, increases water level and reduces reactor power. This event is bounded by the TTNBP event.

14.5.6.3 Trip of Two Recirculation Pumps

14.5.6.3.1 Transient Description

VFD speed control:

This two-loop trip provides the evaluation of the fuel thermal margins maintained by the rotating inertia of the recirculation drive equipment. With the VFDs, all two recirculation pump trips will only have the pump and motor inertia during coastdown. Other than loss of auxiliary power covered in Section 14.5.5.4, loss of Raw Cooling Water (RCW) or an inadvertent RPT System trip could cause a trip to the power of both recirculation pumps. This event is bounded by the TTNBP event.

14.5.6.4 Recirculation Pump Seizure

14.5.6.4.1 Transient Description for Two Loop Operation

This case represents an assumed instantaneous seizure of the pump motor shaft of one recirculation pump. Flow through the affected loop is rapidly reduced due to the large hydraulic resistance introduced by the stopped rotor. This causes the core thermal power to decrease and reactor water level to swell. 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. This event is bounded by the LRNBP and FWCF events.

14.5.6.4.2 Transient Description for Single Loop Operation

This case represents an assumed instantaneous seizure of the pump motor shaft of the operating recirculating pump during single loop operation (SLO). Flow through the affected loop is rapidly reduced due to the large hydraulic resistance introduced by the stopped rotor. With both recirculation loops idle, the core transitions to natural convection cooling. This 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 SLO pump seizure is potentially more severe than the two loop case due to the complete loss of recirculation drive flow.

For AREVA designed fuel cycles, the SLO pump seizure transient is analyzed on a cycle specific basis to determine if this event could set the MCPR operating limit for SLO conditions. The results of this analysis for the most recent reload cycle are reflected in the unit-specific and cycle-specific reload safety analysis report.

14.5.7 Events Resulting in a Core Coolant Flow Increase

Events that result directly in a core coolant flow increase are those that affect the reactor recirculation system. The following events have been analyzed:

a. Recirculation Flow Control Failure - Increasing Flow
b. Startup of Idle Recirculation Pump

For both transients, no credit is conservatively taken for the APRM flow-biased flux scram occurrence.

14.5.7.1 Recirculation Flow Controller Failure - Increasing Flow

14.5.7.1.1 Transient Description

VFD speed control:

In this event, it is postulated that a single flow controller fails and signals the VFD to increase the pump speed. (The VFD controls are designed such that expected failures only affect one pump.) The maximum pump run-up rate is defined by using the maximum pump motor torque. The maximum pump motor torque is defined by the breakdown torque (maximum torque the motor develops under increasing load without abruptly loosing speed). The breakdown torque is applied to the pump, and the transient model determines the resultant pump run-up rate. The average run-up rate, for the first second, is 745 rpm/sec. At about 2.4 sec the pump speed reaches 1725 rpm. A pump trip is nominally designed to occur at the frequency (57.5Hz) associated with this speed. No credit is taken for this trip. The rapid increase in core inlet flow causes a large neutron flux peak which may exceed the high flux scram setpoint.

14.5.7.1.2 Initial Conditions and Assumptions

VFD speed control:

This transient was analyzed with CASMO-4/MICROBURN-B2/XCOBRA starting from the power level and flow corresponding to the lower end of the normal design flow

control range on the maximum control rod line when the reactor is initially at 66 percent of 3952 MWt and core flow is at 52 percent of rated.

One recirculation pump is driven with the physical maximum torque-breakdown torque. The high frequency pump trip is conservatively not credited. However, to assess the control system responses, a pump trip is simulated to occur at 3 seconds, which is after the time of MCPR.

14.5.7.1.3 Interpretation of Transient Results

VFD speed control:

Figures 14.5-24c through -24f illustrate this transient. At a time of one second, upward failure of the speed controller causes the VFD to increase the frequency at a rate such that the pump-motor operates at breakdown torque continuously. The resulting increase in core flow causes an increase in reactor power. No credit is conservatively taken for the APRM flow-flow biased flux scram. High flux scram setpoint is reached at 2.04 seconds. The rapid increase in core flow causes the void fraction to initially decrease and the water level to drop. As the system pressure decreases, following the reactor scram, the reactor water level rises but does not reach the high level L8 setpoint. Subsequently, the water level turns around but does not decrease to the low level L2 setpoint.

The changes in the nuclear system pressure are not significant with regard to overpressure. The pressure decreases over most of the transient. Peak steam line pressure reaches 1021 psia while peak vessel pressure reaches 1058 psia. Peak neutron flux reaches 211 percent of rated. The maximum heat flux is 89 percent of rated. The calculated delta-CPR is 0.13. Cycle-specific fuel thermal limits are determined for each reload.

14.5.7.2 Startup of Idle Recirculation Loop

14.5.7.2.1 Transient Description

The normal procedure for the startup of an idle recirculation loop requires the warm up of the idle drive loop water to within 50 degrees F of the active drive loop water by permitting the pressure head by the active jet pumps to cause reverse flow through the idle loop. This transient considers the failure wherein the loop drive water has been allowed to cool down to near ambient temperature, and the idle recirculation loop starts up without warming the drive loop water. The thermal-hydraulic perturbation will cause a spike in core thermal power.

14.5.7.2.2 Initial Conditions and Assumptions

VFD speed control:

The transient has been analyzed by AREVA with the COTRANSA2, XCOBRA, and XCOBRA-T codes. The following initial conditions were assumed:

a. One recirculation loop is idle and filled with cold water (100 degrees F minimum).

b. The active recirculation pump is operating at a speed that produces about 130 percent of normal rated jet pump diffuser flow in the active jet pumps.

c. The core is receiving 51.6 percent of its normal rated flow; the remainder of the coolant flows in the reverse direction through the inactive jet pumps.

d. The initial core power level is 75% of 3458 MWt. This power level is the highest anticipated power for single loop operation. No high flux scram is anticipated with the VFDs; therefore, the 75% power case is the limiting condition.

e. Startup acceleration rate is 150 rpm/sec.

f. Startup maximum pump speed is 400 rpm.

g. The idle recirculation pump suction valve is open, the pump discharge valve is closed.

h. No credit is given to the functionality of the APRM flow-biased flux scram. Only the high neutron flux scram is assumed in the analysis.

The loop startup transient sequence is:

a. The idle loop pump is started at 0 seconds, with a startup rate of 150 rpm/sec.

b. The pump reaches maximum speed of 400 rpm in less than 4 seconds.

c. The pump discharge valve is assumed to be fully open, coincident with the startup of the idle loop pump at 0 seconds. (Normal procedure would delay valve opening to separate the two portions of the flow

transient and make sure the idle loop water is properly mixed with the hot water in the vessel.)

14.5.7.2.3 Interpretation of Transient Results

VFD speed control:

Figures 14.5-25c through -25f illustrate this transient.

An early neutron flux peak, in response to the rapid core flow increase, of 89% of rated occurs at 4 seconds. The peak neutron flux of 93% of rated occurs at 38.3 seconds (time at which the cooler water is finally discharged from the idle loop). The corresponding peak heat flux is 93% of rated. Peak steam line and vessel pressures are 1030 and 1058 psia, occurring at 32.4 and 32.9 seconds, respectively. No damage occurs to the clad barrier. The power increase is terminated once all of the cooler water is discharged from the idle loop.

The event was most severe at BOC. There is significant MCPR margin and adequate LHGRFAC/MAPFAC margin.

The results show that this event is non-limiting for fuel thermal limits.

In addition, AREVA has dispositioned this event as non-limiting at 3952 MWt.

14.5.8 Events Resulting in Excess of Coolant Inventory

14.5.8.1 Feedwater Controller Failure Maximum Demand (FWCF)

14.5.8.1.1 Transient Description

An event which can cause directly an excess of coolant inventory is one in which makeup water flow is increased without changing other core parameters. The FWCF is the limiting event of the excess coolant inventory type. The FWCF to maximum demand is one of several potentially limiting events normally included in the cycle-specific reload licensing analyses to establish the MCPR operating limits. The analysis results for the FWCF to Maximum Demand event are present in the Reload Licensing Report for each cycle.

The FWCF event is a direct failure of a control device which results in the feedwater controller being forced to its upper limit, creating the maximum flow demand. Increases in feedwater flow result in increases in the core inlet subcooling and in the reactor water level. When the high water level setpoint is reached, the main turbine and feedwater pumps are tripped; and scram occurs due to the turbine stop valves closure.

14.5.8.1.2 Input Data and Assumptions

For the reload analysis, AREVA used the following models/computer codes: COTRANSA2, XCOBRA, XCOBRA-T, RODEX2, RODEX4.

The FWCF event was analyzed at 100% power (3952 MWt) and at 77.6% power (3067 MWt) as a typical off-rated operating condition. Since the ICF condition produces top peaked axial power shapes which degrade scram effectiveness, ICF was assumed for both power levels (e.g., 100P/105F and 77.6P/109F). Normal feedwater temperature was assumed for the rated power condition while reduced feedwater temperature was assumed for the off-rated power case for a maximum subcooling effect on the off-rated transient response. The EOC exposure was assumed to maximize the transient severity because the scram effectivity is reduced with the all-rods-out condition and the top peak power shape.

Normal feedwater temperature conditions, at 100 percent power, were found to be more limiting than reduced temperature conditions because of the large pressurization component of delta-CPR caused by the reduced steam line pressure drop. The large pressurization component of delta-CPR dominates over the subcooling component of delta-CPR; therefore, the case with larger steam flow was more severe.

The FWCF event assumed a feedwater flow runout of 22.79 million pounds-mass per hour at 1050 psia feedwater design pressure. The feedwater runout flow will be adjusted as needed for reload licensing analyses to reflect updated equipment performance information.

14.5.8.1.3 Interpretation of Transient Results

A plant-specific response of the BFN plant to a FWCF event is shown in Figures 14.5-28a, b, and c. The transient was initiated from 77.6P/109F as a typical off-rated operating state point. The feedwater pumps are assumed to accelerate to the maximum capability.

Sensed and actual water level increase during the initial part of the transient to about 38.1 inches. The high water level (L8) main turbine trip and feedwater turbine trip is initiated at 12.3 seconds preventing excessive carryover from damaging the turbines. The EOC-RPT is tripped simultaneously with the high reactor water level trip signals. A reactor scram occurs following the turbine trip event, limiting the neutron flux peak (298 percent of rated), surface heat flux peak (105 percent of rated), and fuel thermal transient excursion (delta-CPR = 0.43). Cycle-specific fuel thermal limits are determined for each reload. The power and flow dependent MAPLHGR, LHGR, and MCPR limits are developed on a cycle-specific bases.

The turbine bypass system opens to limit the pressure rise. The lower set relief valves open only momentarily and no excessive overpressure of the nuclear system process barrier occurs (peak steam line pressure 1206 psia). The bypass valves close later bringing the pressure in the vessel (peak vessel pressure 1223 psia) under control during reactor shutdown.

For 100P/105F, a peak neutron flux of 332 percent of rated and a peak heat flux of 129 percent of rated are reached. Peak steam line pressure reaches a value of 1265 psia while peak vessel pressure reaches a value of 1276 psia. No fuel damage occurs (delta-CPR = 0.34) with the application of the adequate operating limit CPR associated with this limiting transient.

At rated power, the delta-CPR resulting from the LRNBP and FWCF events is more severe than the delta-CPR resulting from any other pressurization events. As power is reduced to 87.5 percent of rated power or less, the delta-CPR resulting from the FWCF event continues to be higher than the delta-CPR resulting from any other pressurization event. For the FWCF, the power decrease results in a greater mismatch between runout and initial feedwater flow resulting in an increase in reactor subcooling and a more severe change in thermal limits during the event. Therefore, this transient along with the LRNBP defines the MCPR(P) and MAPLHGR(P) or LHGR(P) curves which protect the fuel integrity for low power.

For power operation below the P_{bypass}, the transient characteristics change due to the bypass of the direct scram on the closure of the TCVs or TSVs. The high neutron flux scram signal is conservatively bypassed, and the high pressure scram is delayed until the vessel pressure reaches this setpoint. The relatively large differences in delta-CPR between the LRNBP and FWCF which are seen between 87.5 percent and 26 percent rated are significantly reduced below P_{bypass}.

14.5.8.2 Feedwater Control Failure/Maximum Demand with EOC-RPT-OOS

Figures 14.5-30a, b, and c show the transient results for the 100 percent of rated power and 105 percent of rated core flow event. EOC exposure and normal feedwater temperature have been the conditions assumed for this transient analysis, the same as in the transient analysis with EOC-RPT in service described above.

The neutron flux peaks at 382 percent of rated, the average heat flux peaks at 134 percent of rated. The peak pressure at the bottom of the vessel is 1306 psia which is well below the ASME upset code limit transients limit of 1375 psig while the peak steam line pressure is 1275 psia. The calculated delta-CPR of this transient at the stated conditions is 0.32.

At off-rated power/flow conditions, such as the 77.6P/109F point, the delta-CPR is 0.40.

Cycle-specific RPT-OOs fuel thermal limits are determined for each reload.

14.5.8.3 Feedwater Control Failure/Maximum Demand with TBP-OOS

The Turbine Bypass Out-of-Service produces a different evolution in the limiting overpressurization transients. The overpressurization is faster because the bypass system is not operable, thus the pressure setpoints are reached earlier. The positive reactivity insertion due to moderator void collapse is more severe, and this results in a higher delta-CPR and, subsequently, a higher OLMCPR.

The FWCF event normally assumes that turbine bypass system is functional, and therefore, this transient is strongly affected by TBP-OOS.

Figures 14.5-3a, b, and c show the transient response at 100 percent of rated power and 105 percent of rated core flow. EOC exposure and normal feedwater temperature have been the conditions assumed for this transient analysis.

The neutron flux peaks at 368 percent of rated; the average heat flux peaks at 132 percent of rated. The peak pressure at the bottom of the vessel is 1323 psia which is well below the ASME upset code limit transients limit of 1375 psig while the peak steam line pressure is 1324 psia. The calculated delta-CPR of this transient at the stated conditions is 0.38. At rated power the impact on delta-CPR caused by TBP-OOS is approximately 0.04.

Cycle-specific TBP-OOS fuel thermal limits are determined for each reload.

14.5.9 Loss of Habitability of the Control Room

Loss of habitability of the control room is arbitrarily postulated as a special event to demonstrate the ability to safely shutdown the reactor from outside the control room. (For additional information, see Section 7.18 - Backup Control System.)

Table 14.5-1

(Deleted by Amendment 28)

Table 14.5-2
TRANSIENT ANALYSES INITIAL CONDITIONS

Parameter	AREVA Reload Analysis
Thermal Power, MWt	3952 (100%) / 4031 (102%)
Core Flow, Mlb/hr	102.5
Core Flow Range (% of current rated)	99-105
Vessel Steam Flow and FW flow, Mlb/hr	16.44
Analysis Dome Pressure, psia	1050
Analysis Turbine Pressure, psia	985
Feedwater Temperature, °F	394.8
Turbine Bypass Capacity	21.3% of rated vessel steam flow
Number of MSRVs	13
MSRV type	Target Rock
Opening response of relief functions	0.15 s
Opening delay of relief functions	0.45 s
MSRV Capacity per valve, lbm/hr (Based on 1090 psig setpoint)	870,000 (12 valves)
MSRV Setpoint, (number of valves @ psig) (+3% setpoint tolerance included)	4 @ 1174[ii] 4 @ 1184 4 @ 1194
MCPR Safety Limit	Cycle Specific
Recirculation Flow Control	VFD Flow Control
Core Average Gap Conductance (Btu/s-sq. ft -Deg F)	Case Dependent
High Neutron Flux Scram Setpoint	125.4% of rated power
High Pressure Scram Setpoint, psig	1101
High Pressure ATWS-RPT setpoint, psig	1177
Reactor L8 Water Level, in avz[iii]	588
Reactor L3 Water Level, in avz[iii]	518
Reactor L2 Water Level, in avz[iii]	448
Reactor L1 Water Level, in avz[iii]	372.5

i (deleted)
ii Considered only 3 out of 4 due to 1 MSRV-OOS
iii Above vessel zero

BFN-19

Figures 14.5-1, 14.5-2, 14.5-3, and 14.5-4

Deleted by Amendment 19

Figure 14.5-5

Deleted by Amendment 28

Figure 14.5-5a

EOC LRNB at 100P/105F with TSSS
Key Parameters

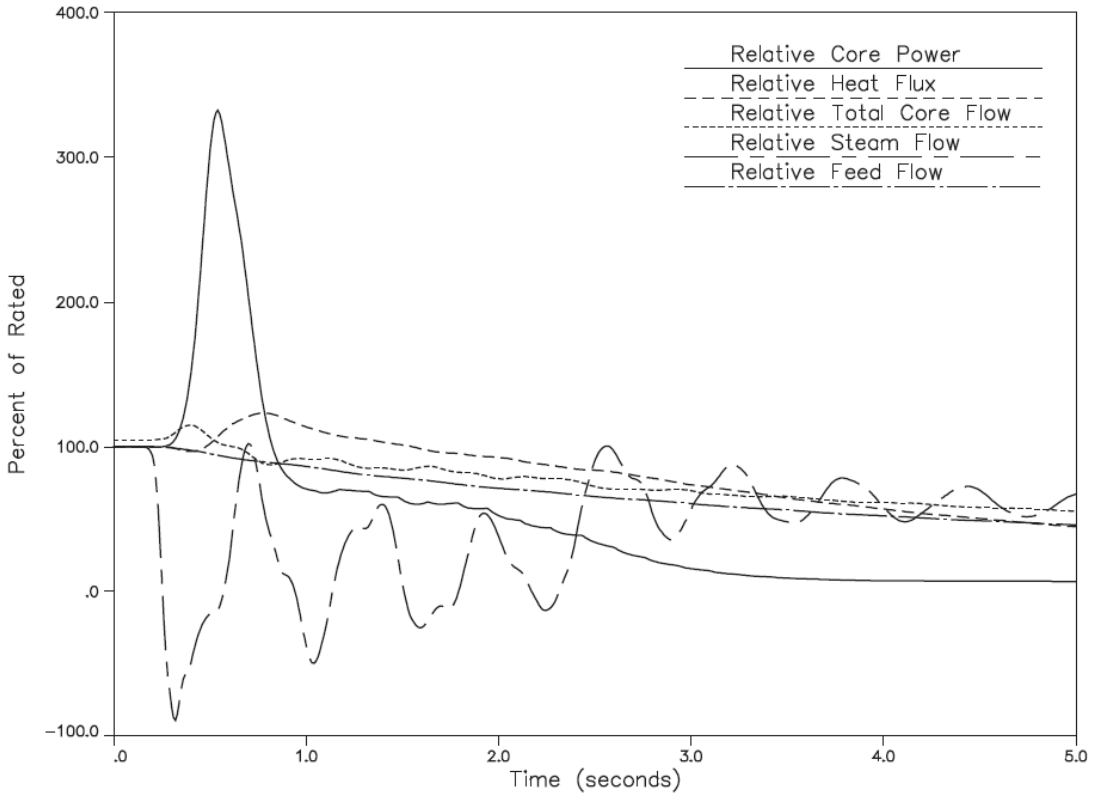


Figure 14.5-5b

EOC LRNB at 100P/105F with TSSS
Sensed Water Level

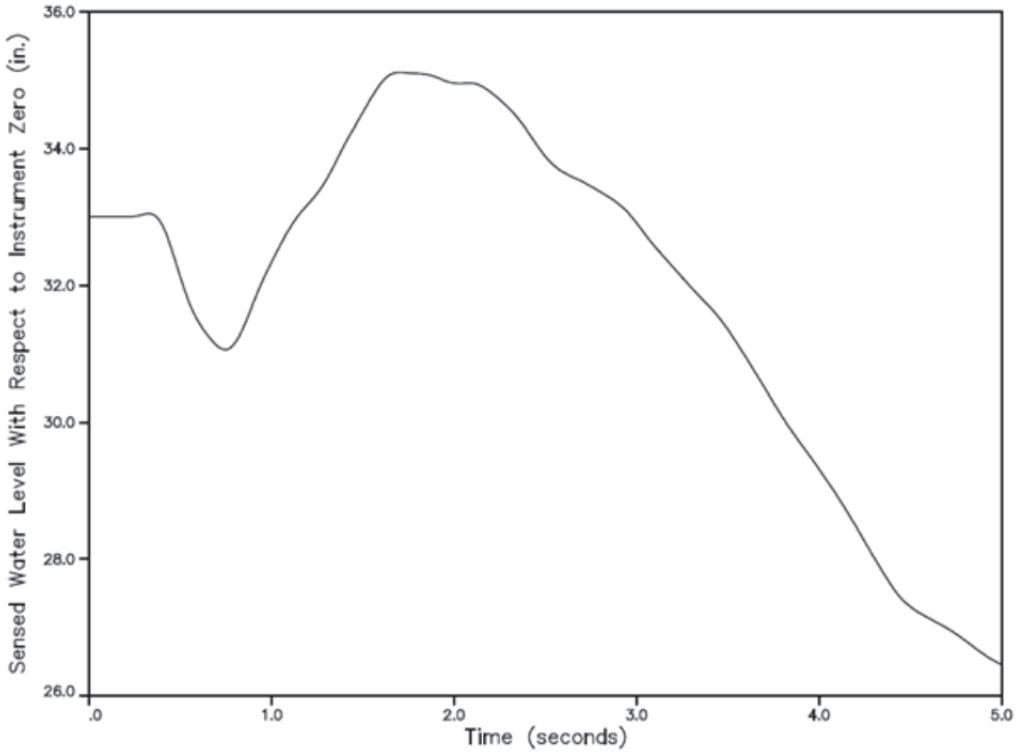


Figure 14.5-5c

EOC LRNB at 100P/105F with TSSS
Vessel Pressures

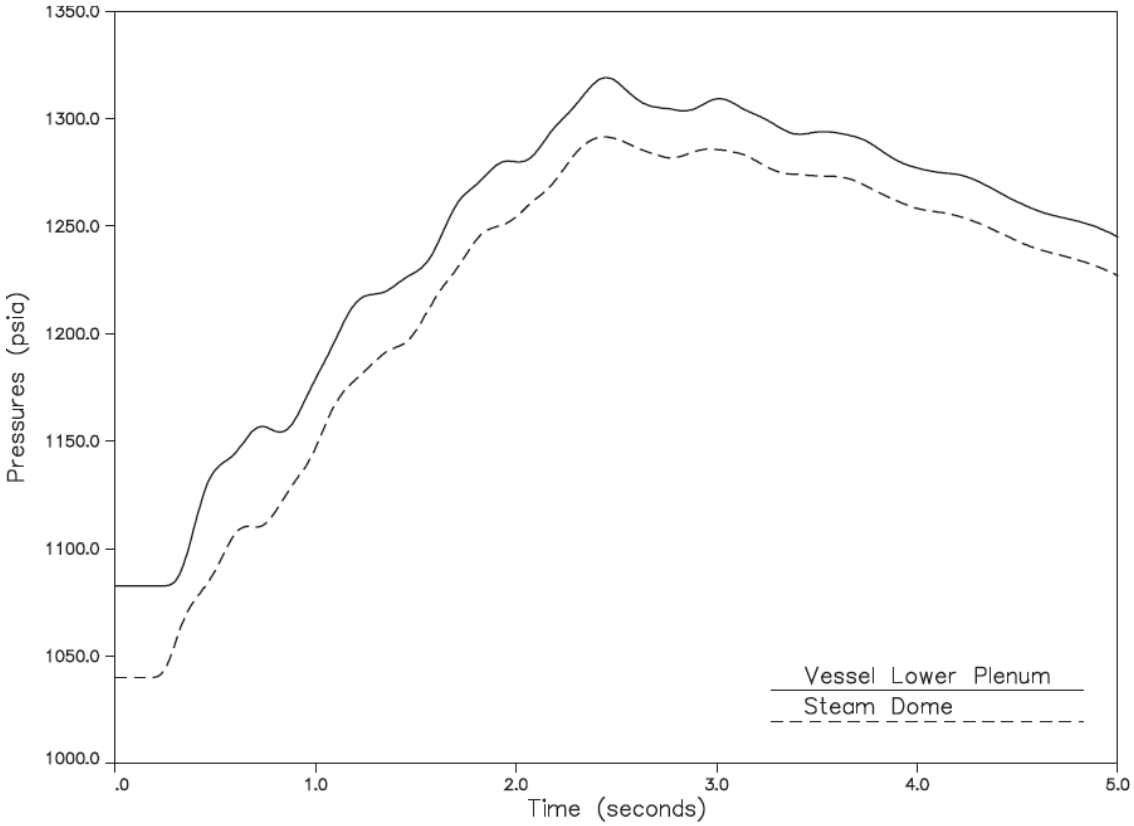


Figure 14.5-6

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Figure 14.5-6a

LRNB with EOC-RPT-OOS at 100P/105F with TSSS
Key Parameters

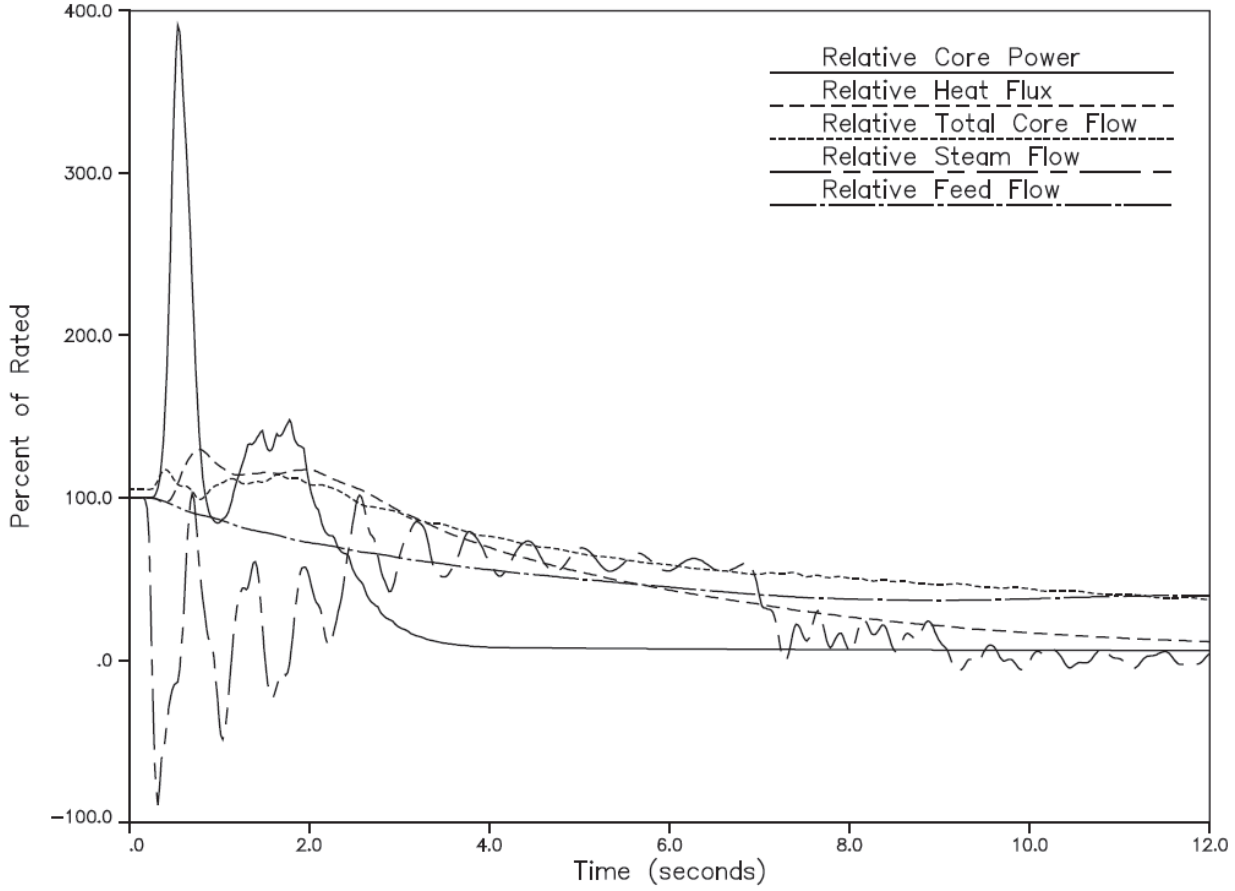

Figure 14.5-6b

LRNB with EOC-RPT-OOS at 100P/105F with TSSS
Sensed Water Level

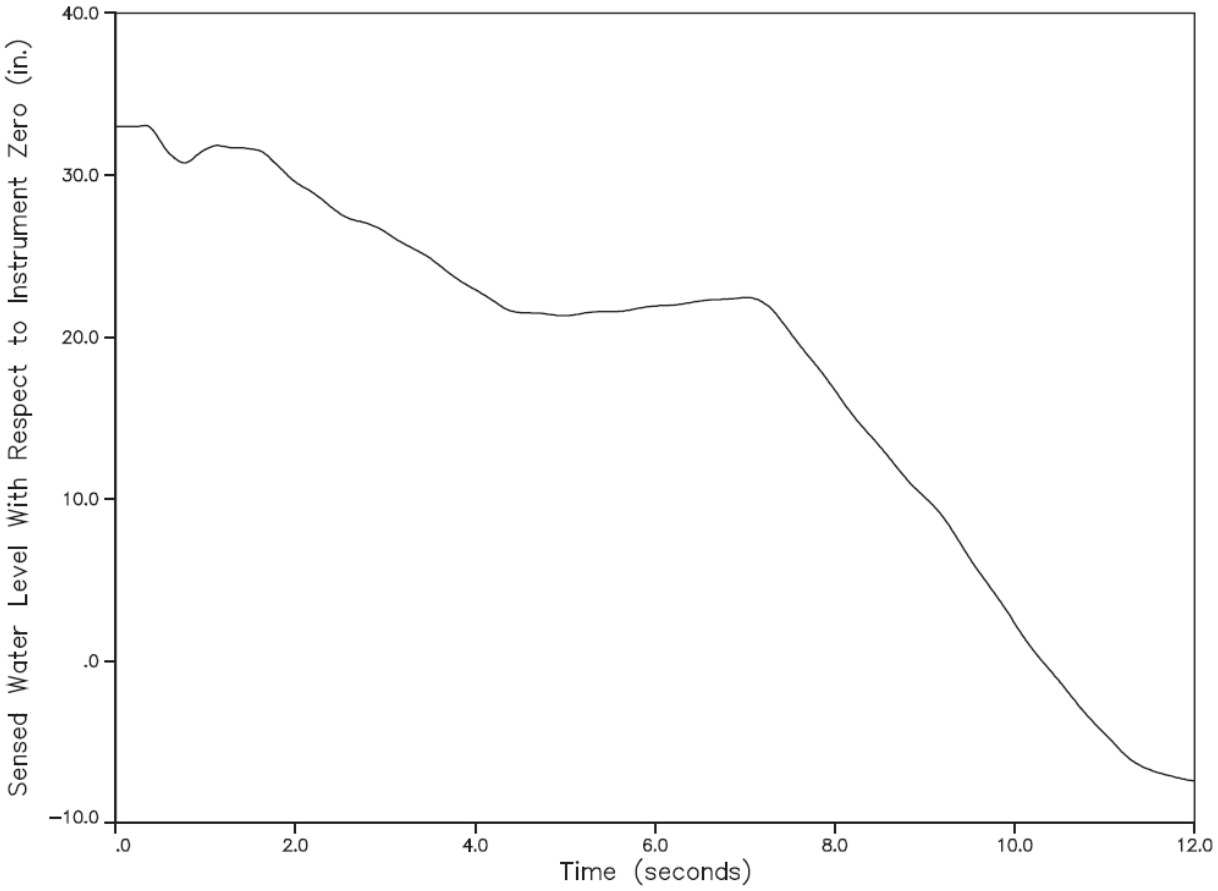

Figure 14.5-6c

LRNB with EOC-RPT-OOS at 100P/105F with TSSS
Vessel Pressures

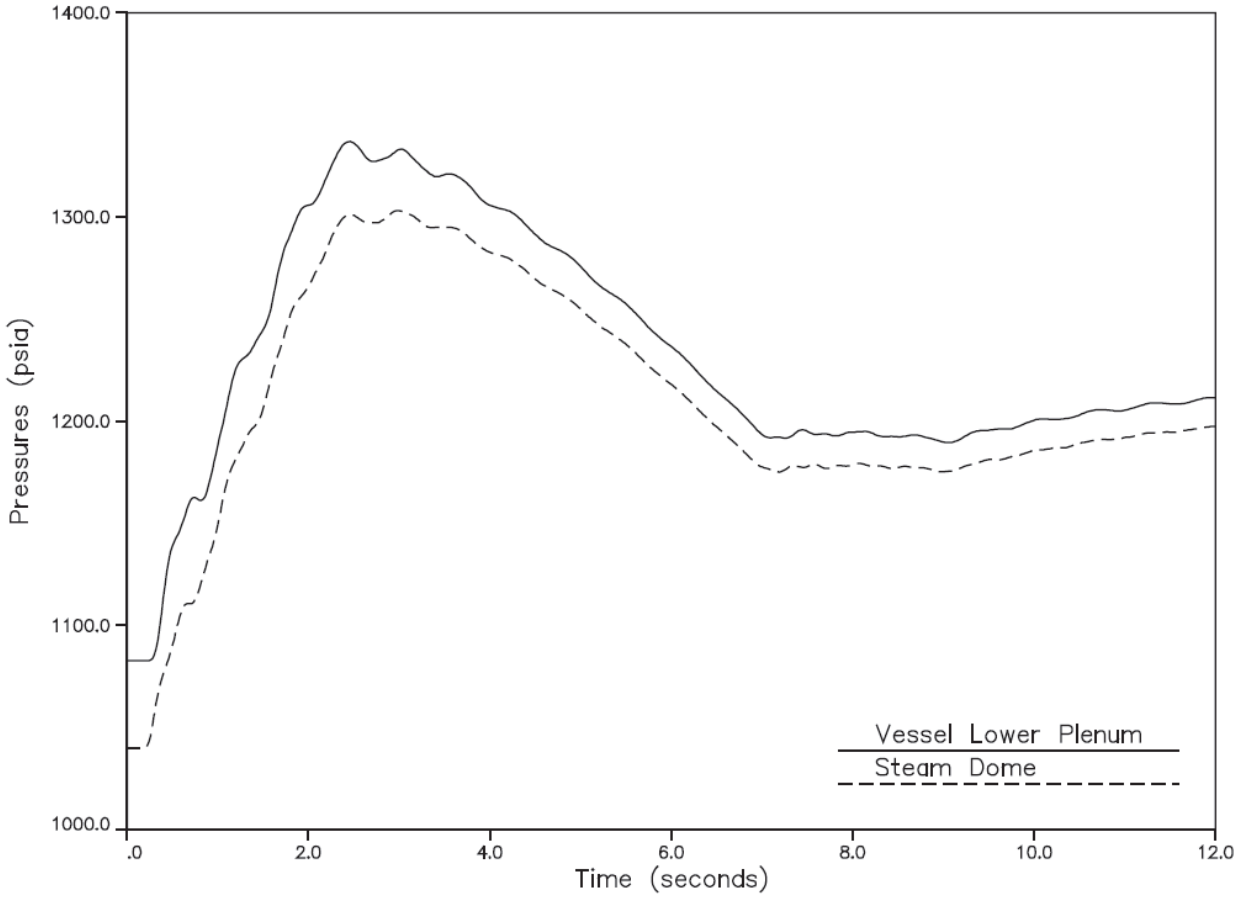

Figure 14.5-7a

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Figure 14.5-7b

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Figure 14.5-8

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Figure 14.5-9

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Figure 14.5-9a

TTNB at 100P/105F with TSSS
Key Parameters

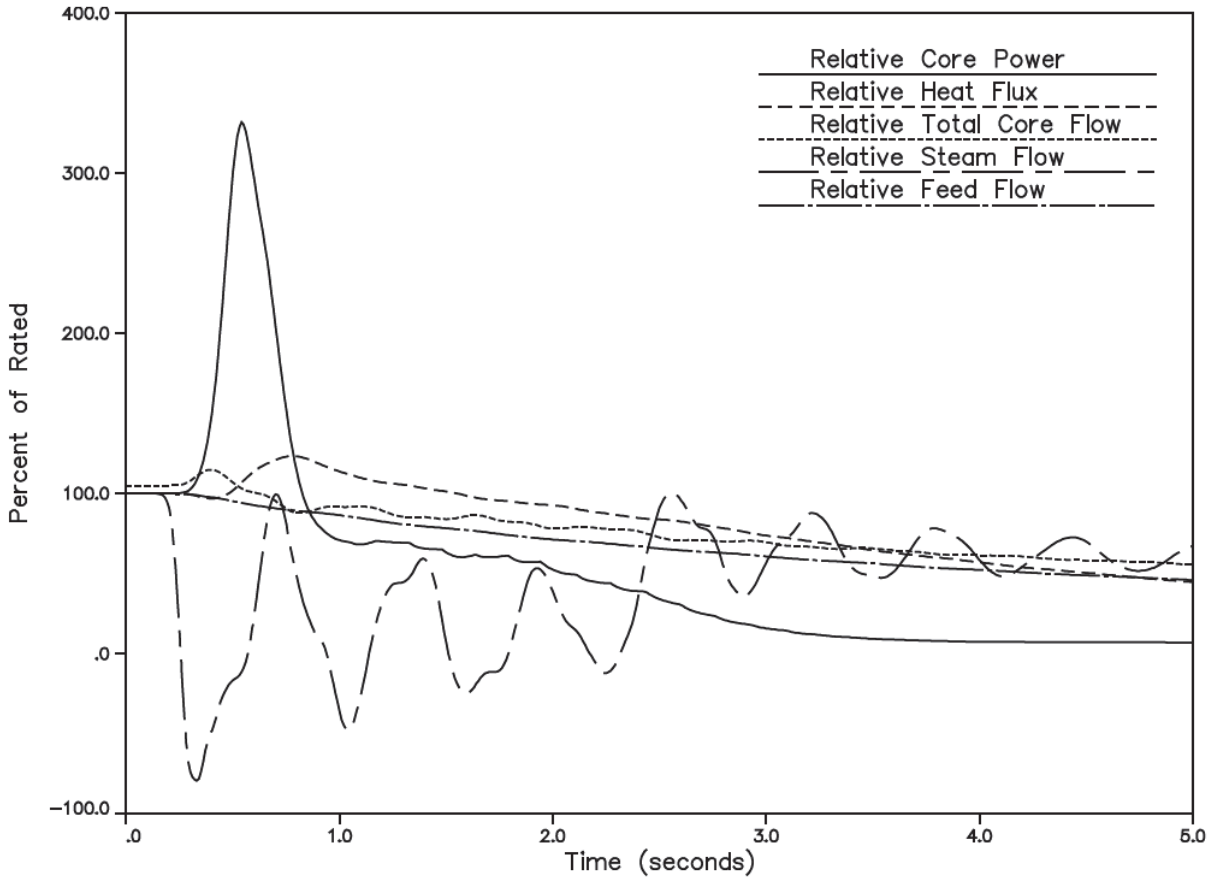

Figure 14.5-9b

TTNB at 100P/105F with TSSS
Sensed Water Level

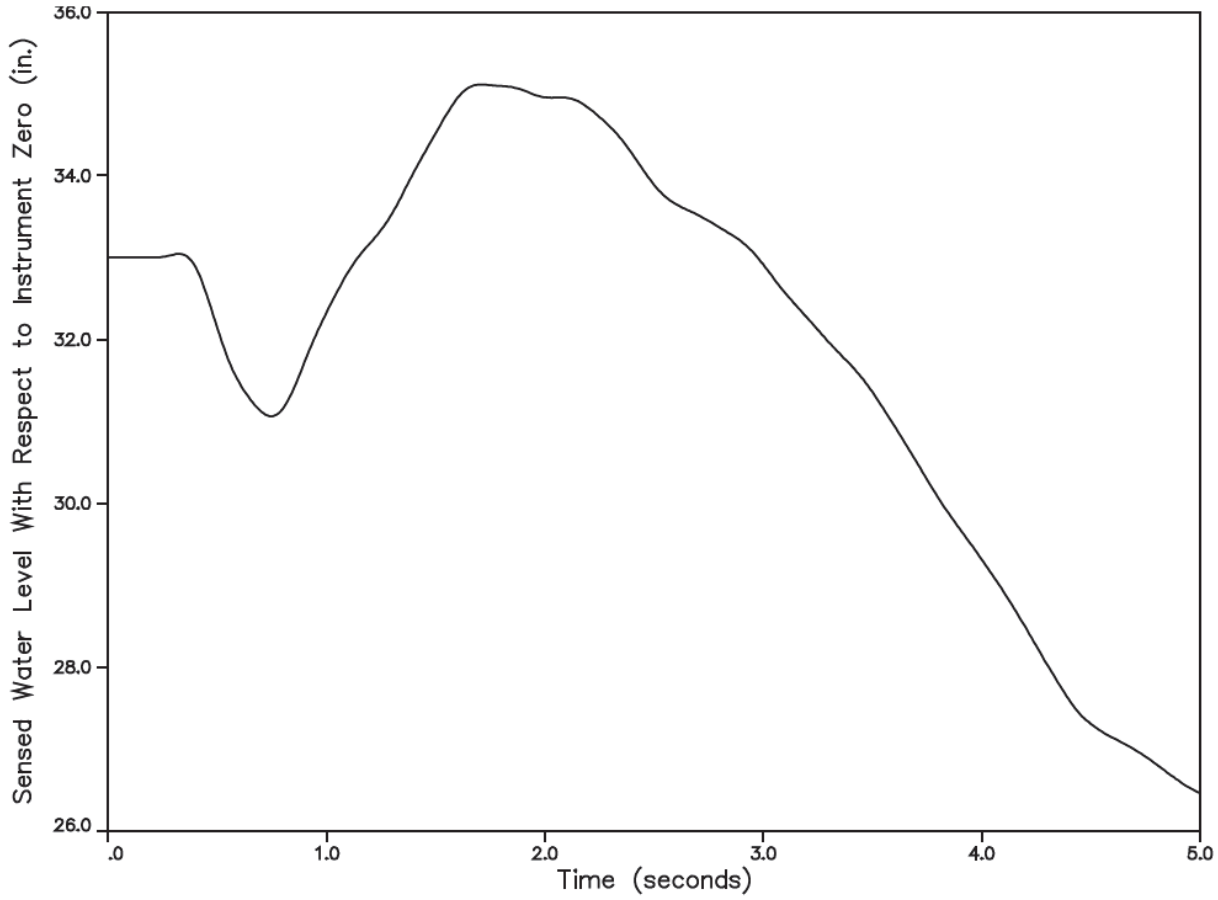

Figure 14.5-9c

TTNB at 100P/105F with TSSS
Vessel Pressures

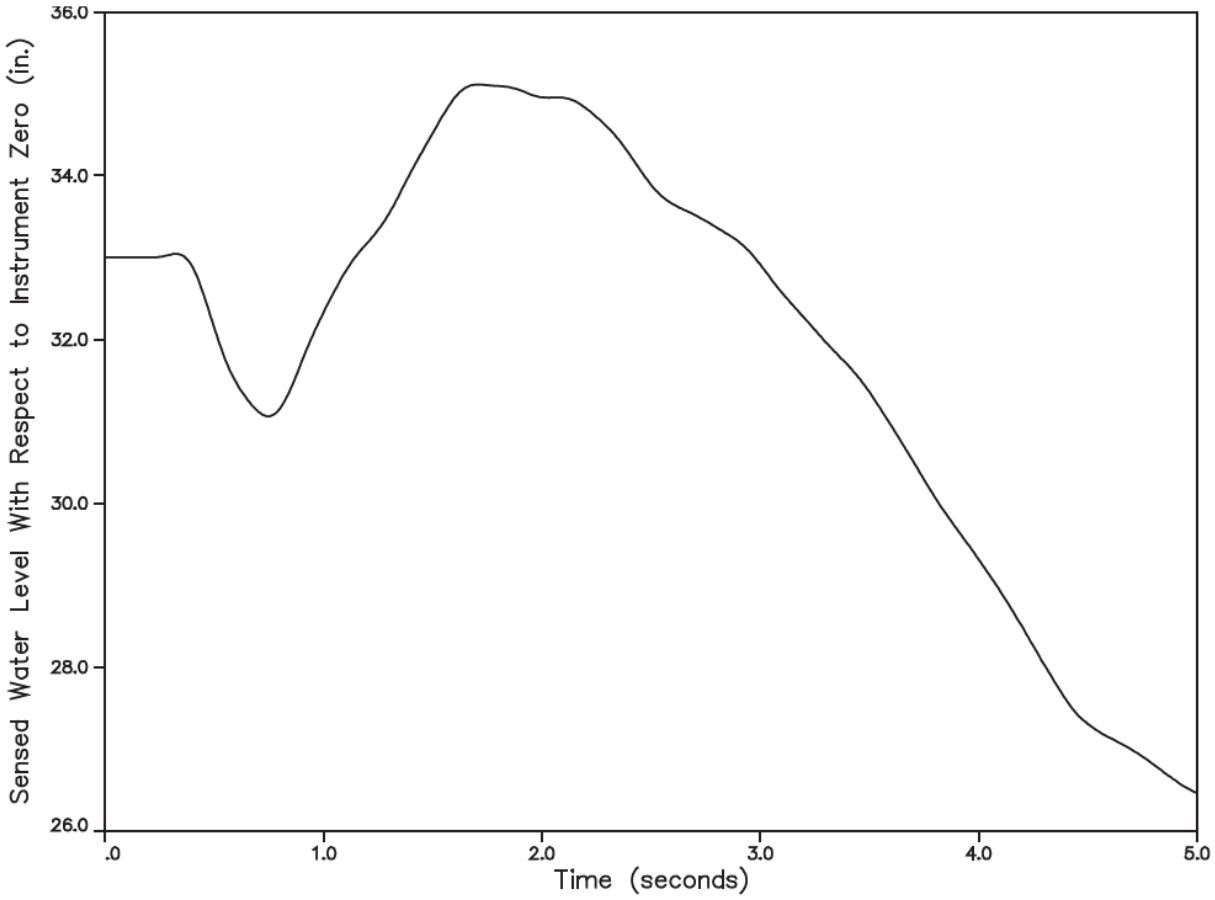

Figure 14.5-10a

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Figure 14.5-10b

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Figure 14.5-11

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Figure 14.5-11a

MSIV Closure Overpressurization Event at 102P/105F with TSSS
Key Parameters

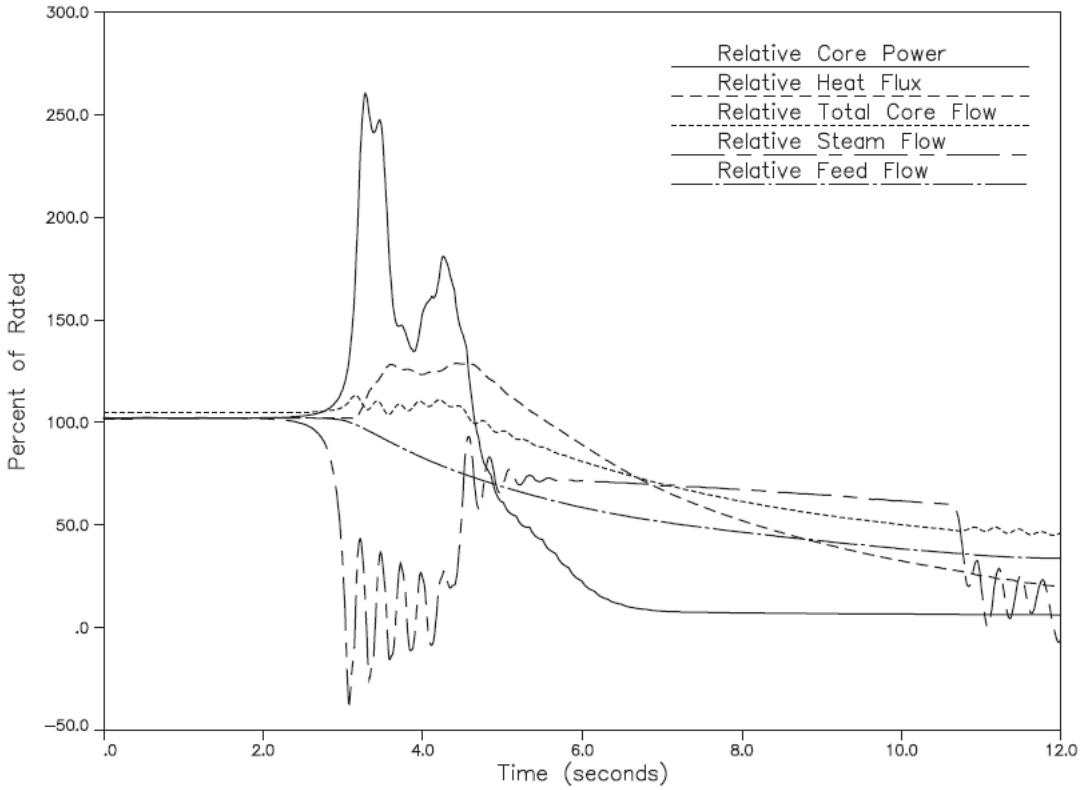


Figure 14.5-11b

MSIV Closure Overpressurization Event at 102P/105F with TSSS Sensed Water Level

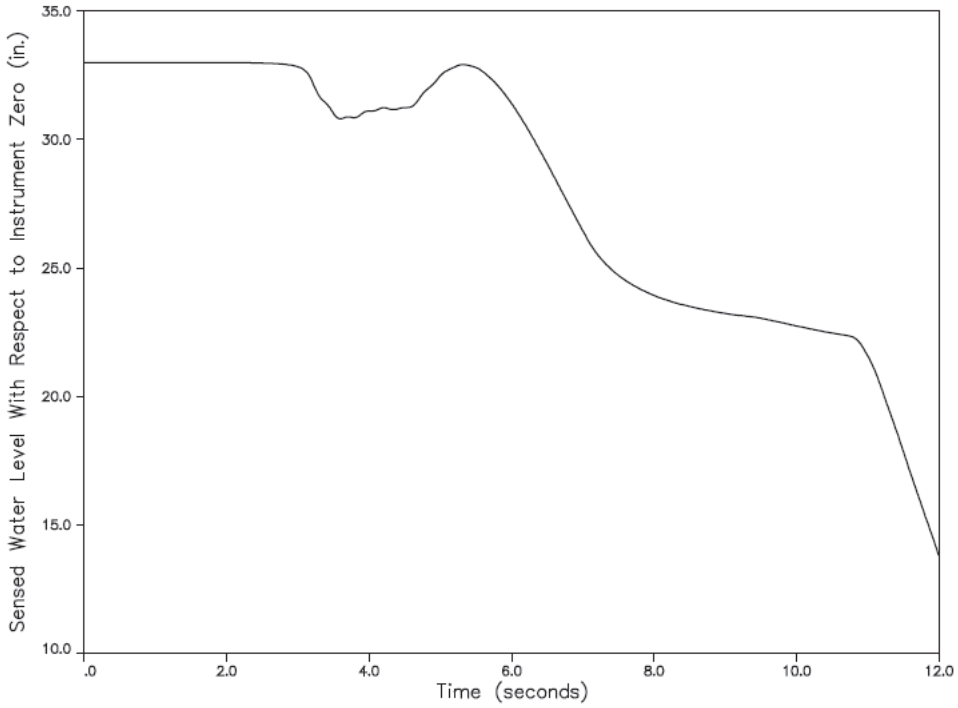
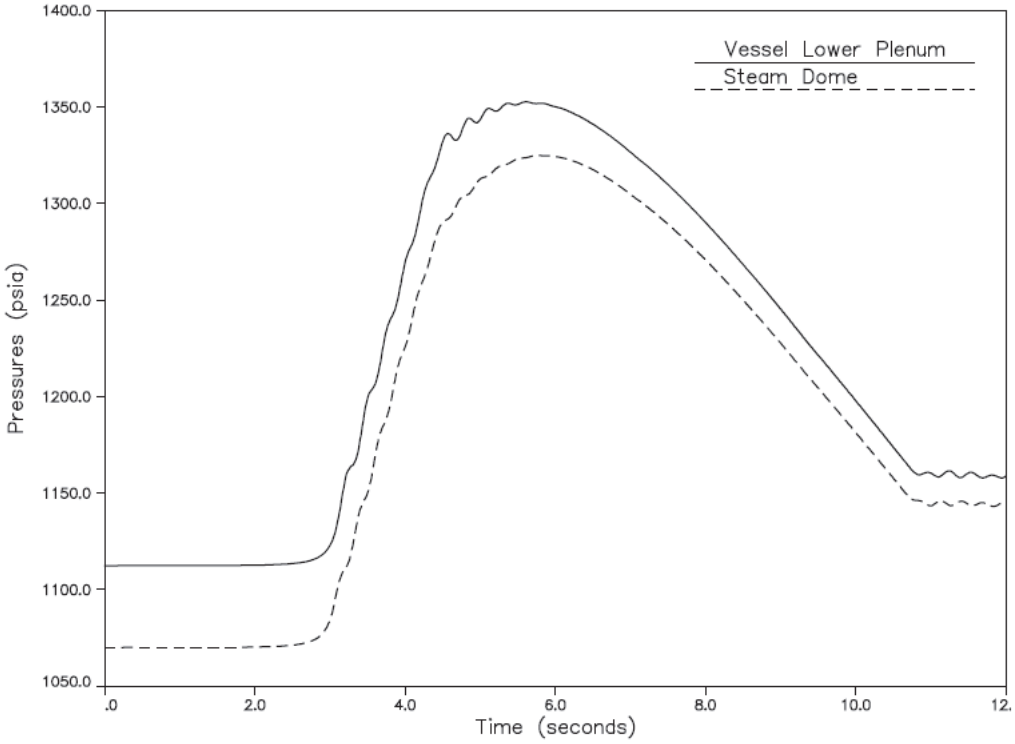


Figure 14.5-11c

MSIV Closure Overpressurization Event at 102P/105F with TSSS
Vessel Pressures*



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

Figure 14.5-11d

MSIV Closure Overpressurization Event at 102P/105F with TSSS Safety/Relief Valve Flow Rates

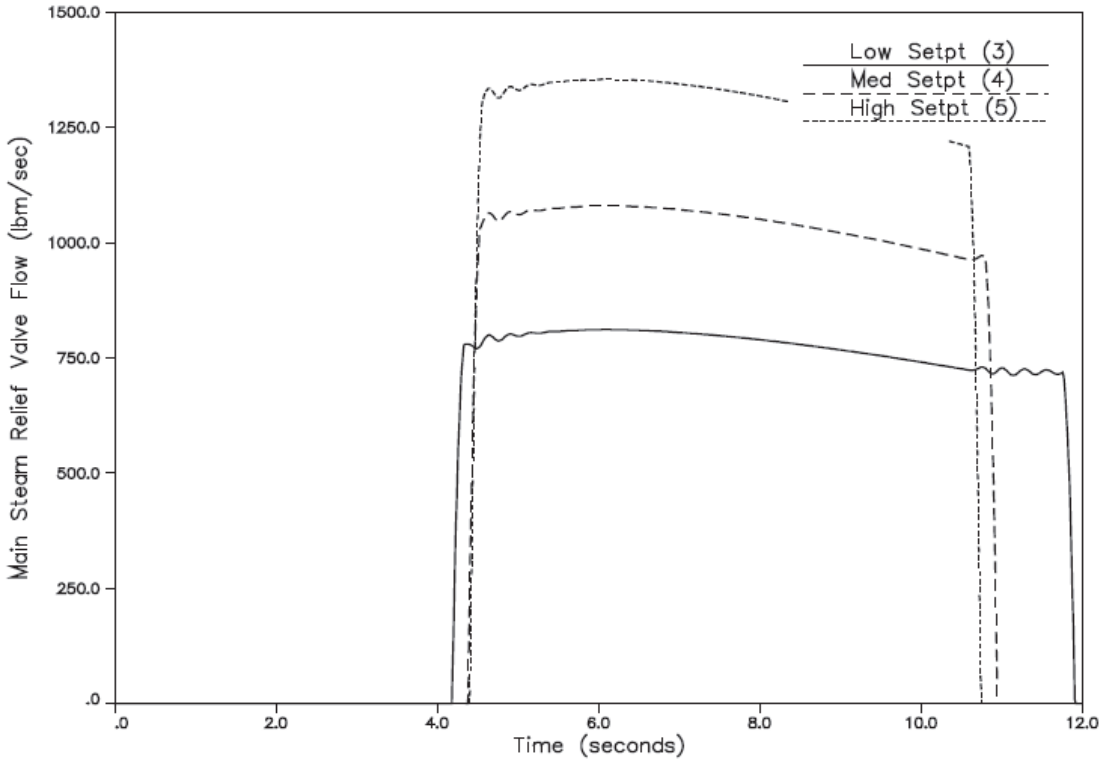


Figure 14.5-12a

One MSIV Closure at 100P/105F with TSSS
Key Parameters

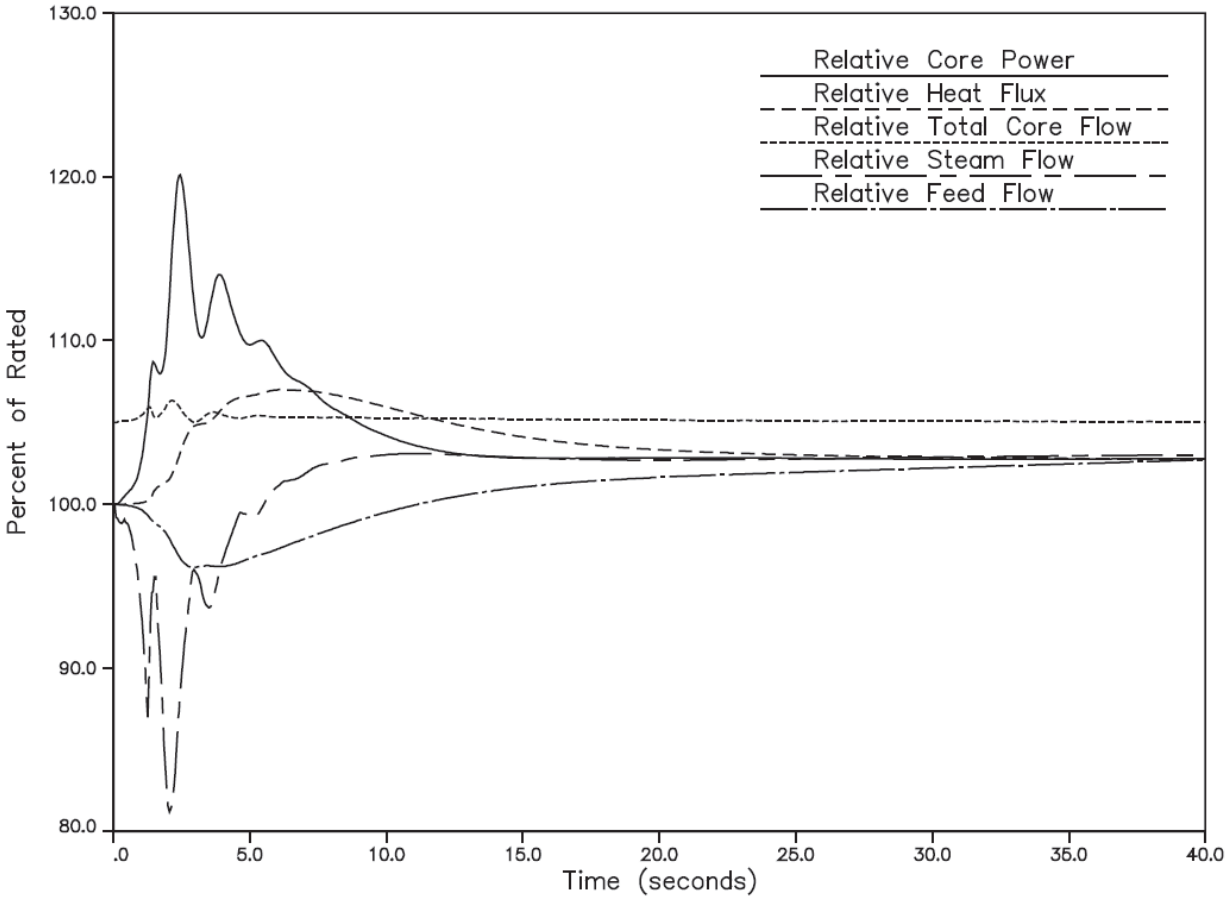


Figure 14.5-12b

One MSIV Closure at 100P/105F with TSSS
Sensed Water Level

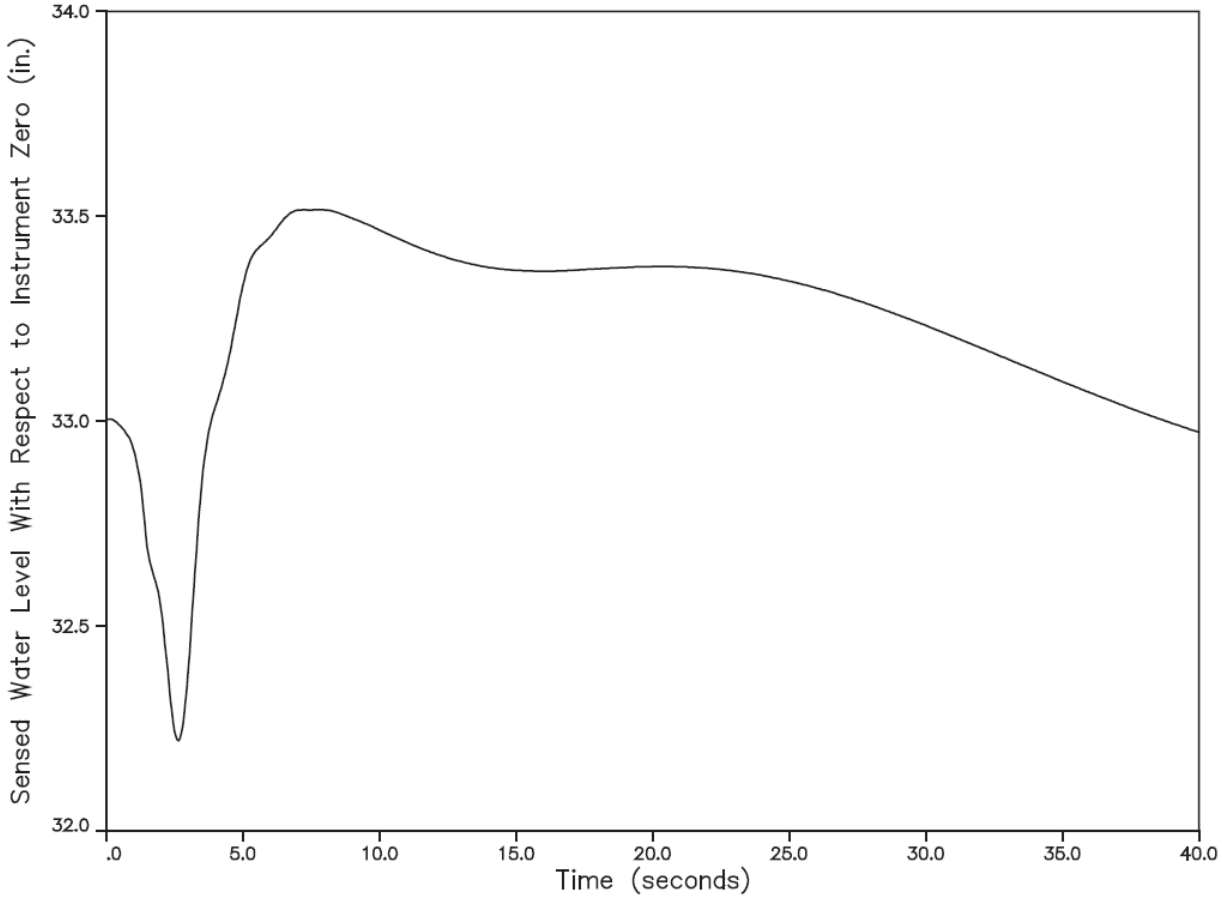


Figure 14.5-12c

One MSIV Closure at 100P/105F with TSSS
Vessel Pressures

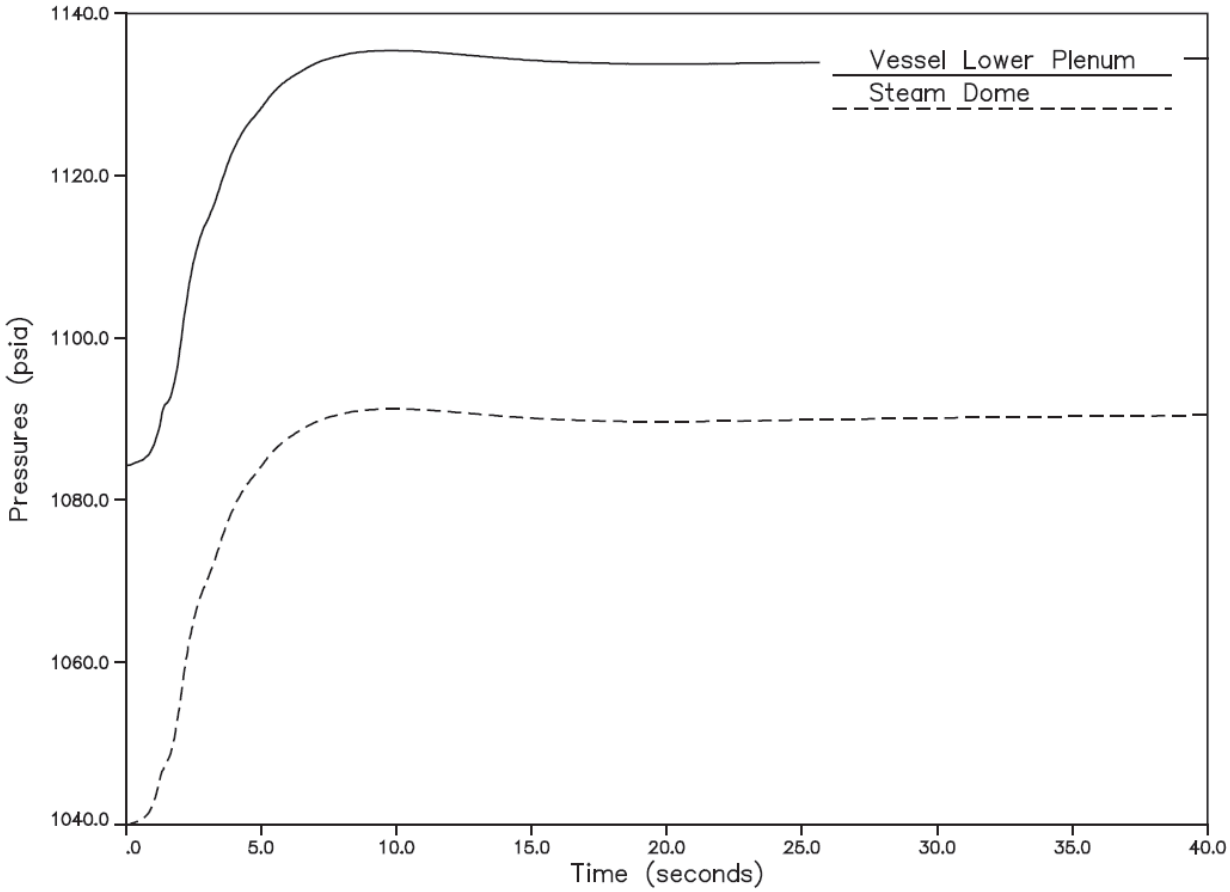


Figure 14.5-13a

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Figure 14.5-13b

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Figure 14.5-14a

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Figure 14.5-14b

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Figure 14.5-15a

Pressure Regulator Failure Open
2075 MWt / 35F

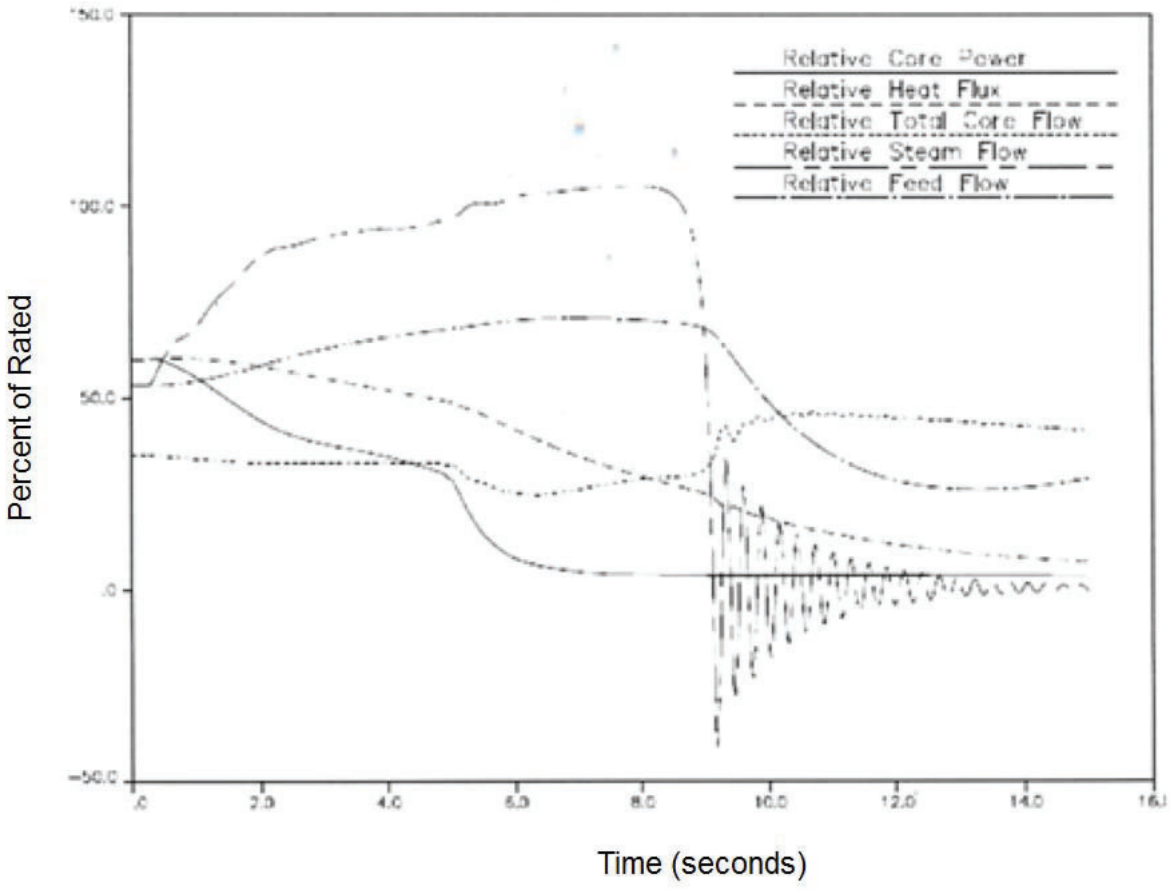

Figure 14.5-15b

Pressure Regulator Failure Open
2075 MWt / 35F

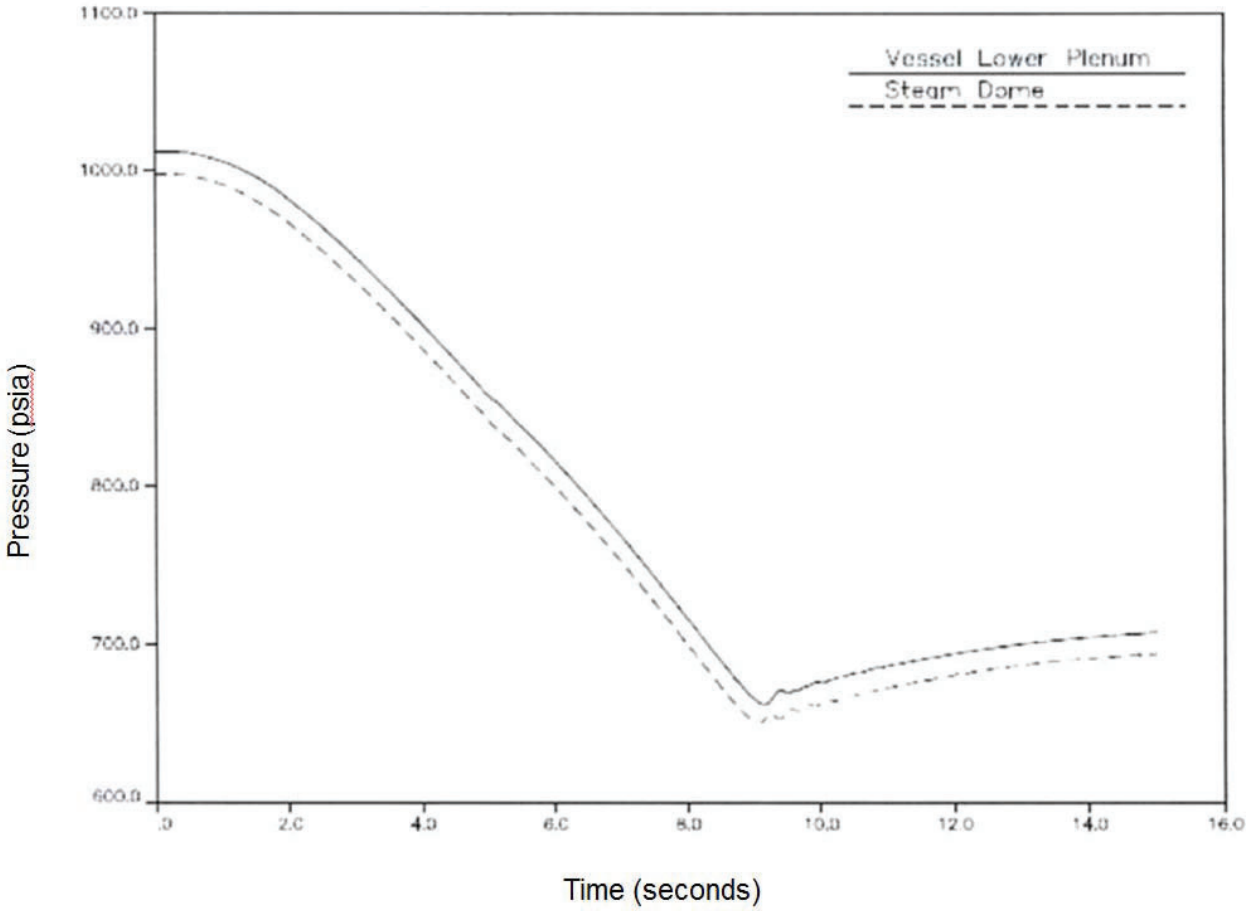

Figure 14.5-16a

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Figure 14.5-16b

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Figure 14.5-17a

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Figure 14.5-17b

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Figure 14.5-17c

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Figure 14.5-18a

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Figure 14.5-18b

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Figure 14.5-19a

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Figure 14.5-19b

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Figure 14.5-20a

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Figure 14.5-20b

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Figure 14.5-21a

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Figure 14.5-21b

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Figure 14.5-22a

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Figure 14.5-22b

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Figure 14.5-22c

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Figure 14.5-22d

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Figure 14.5-22e

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Figure 14.5-22f

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Figure 14.5-23a

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Figure 14.5-23b

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Figure 14.5-24a

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Figure 14.5-24b

Deleted by Amendment 28

Figure 14.5-24c

Fast Recirculation Pump Runup at 66P/52F
Key Parameters

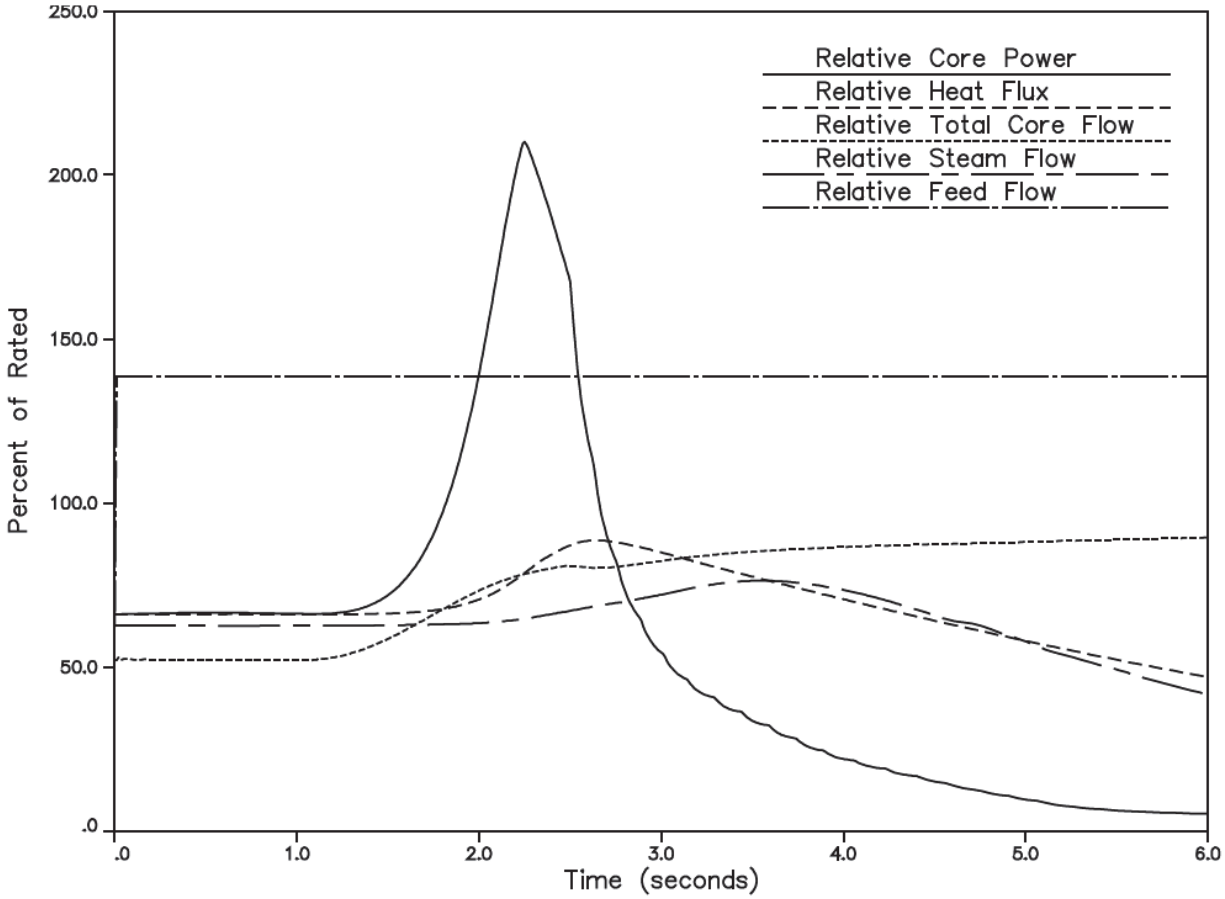

Figure 14.5-24d

Fast Recirculation Pump Runup at 66P/52F
Sensed Water Level

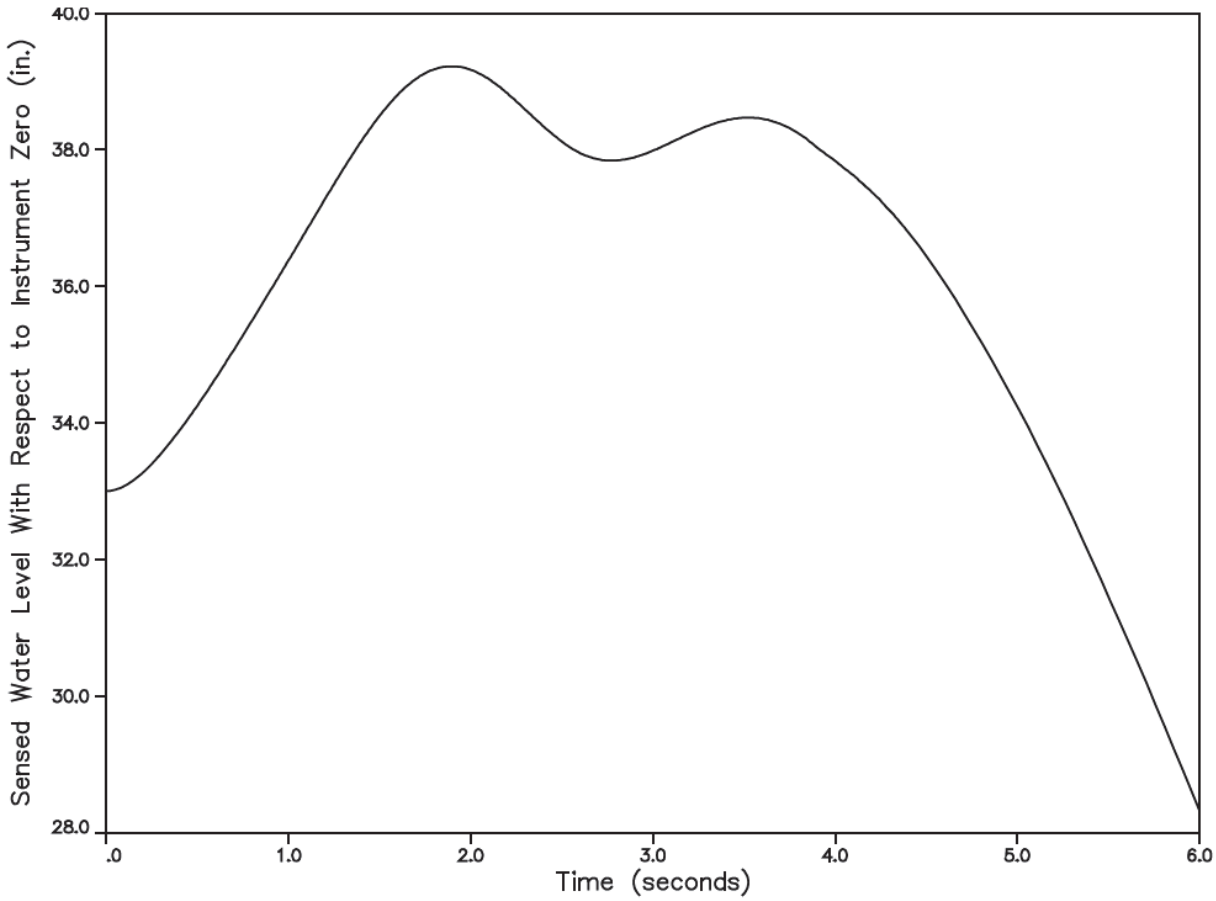

Figure 14.5-24e

Fast Recirculation Pump Runup at 66P/52F
Vessel Pressures

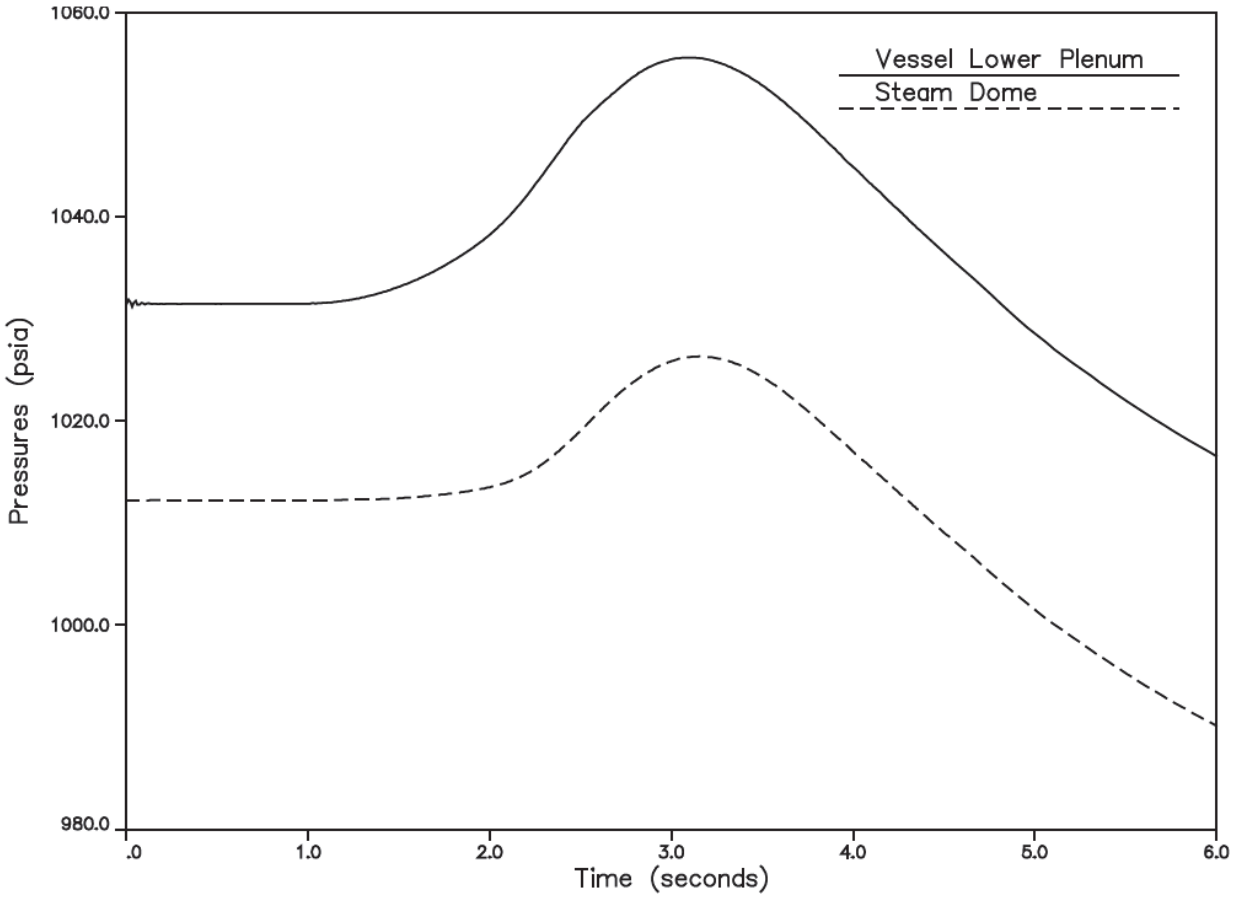

Figure 14.5-24f

Fast Recirculation Pump Runup at 66P/52F
Recirculation Pump Speed

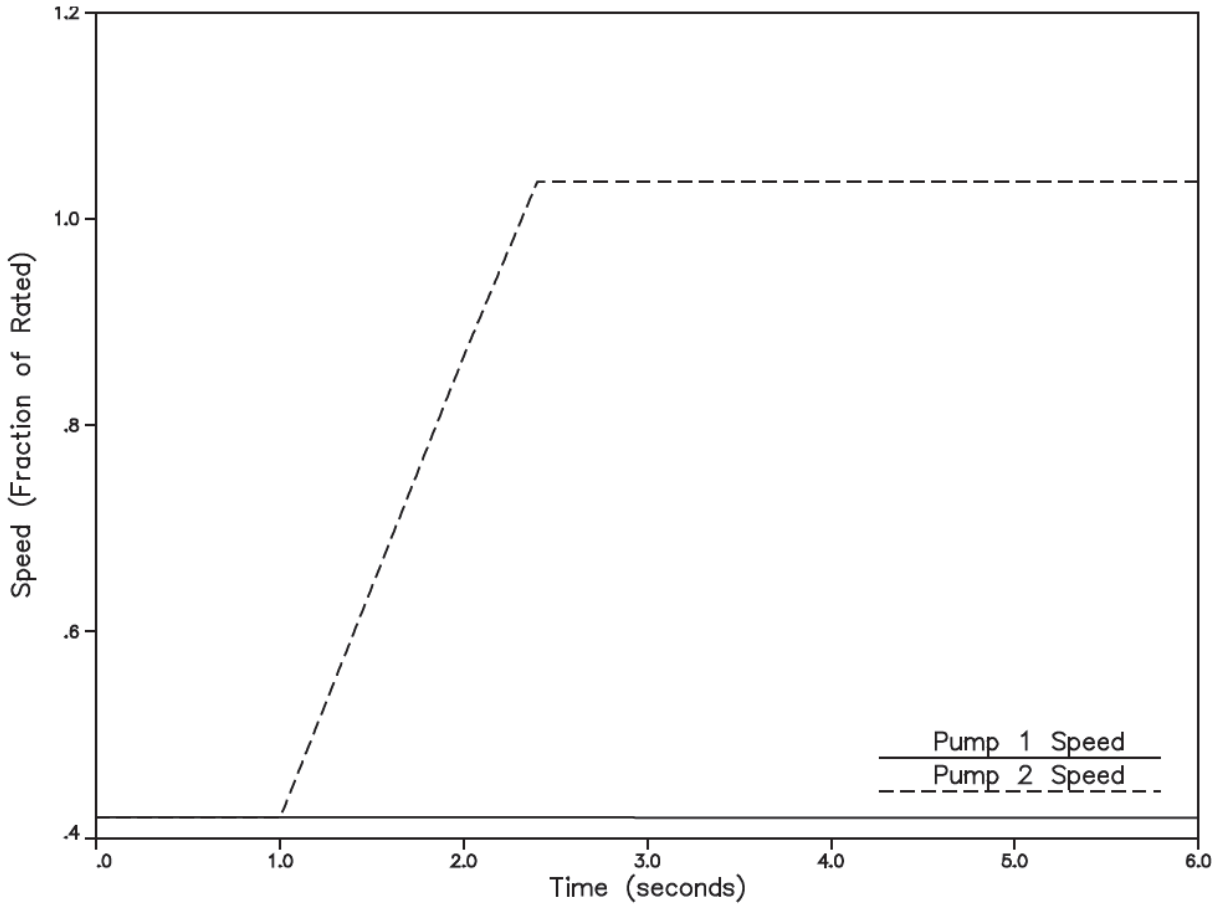

Figure 14.5-25a

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Figure 14.5-25b

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Figure 14.5-25c

105% OLTP Idle Loop Startup at 75P/52F
Key Parameters

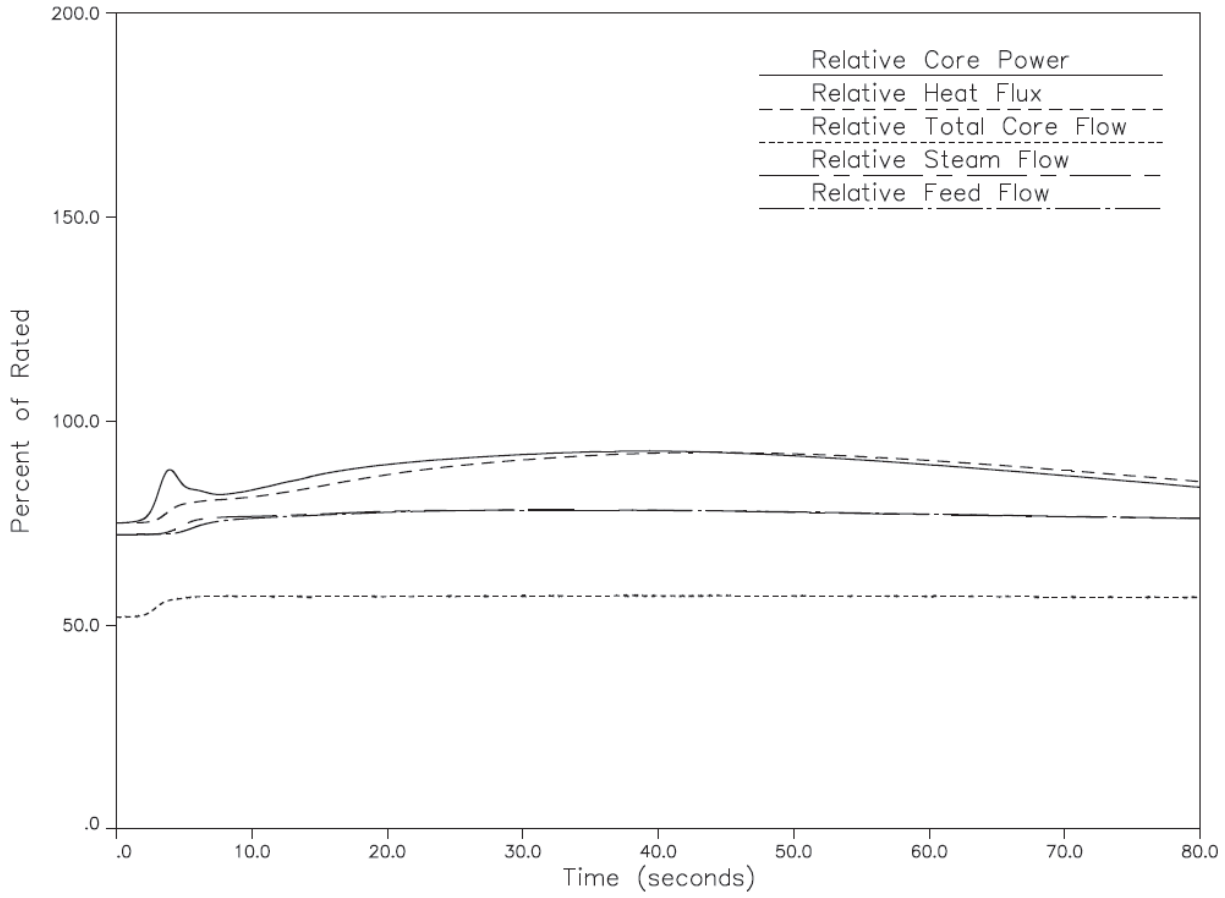


Figure 14.5-25d

105% OLTP Idle Loop Startup at 75P/52F
Sensed Water Level

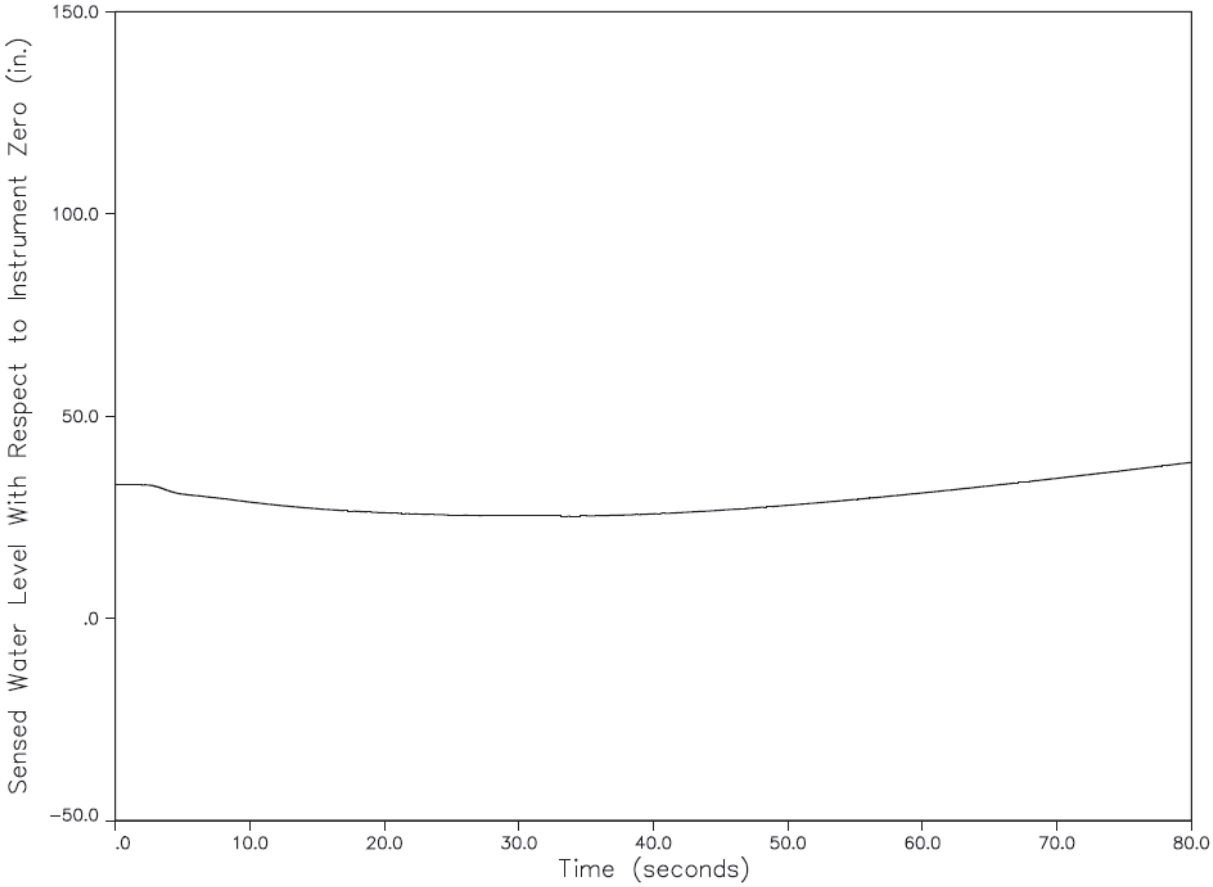
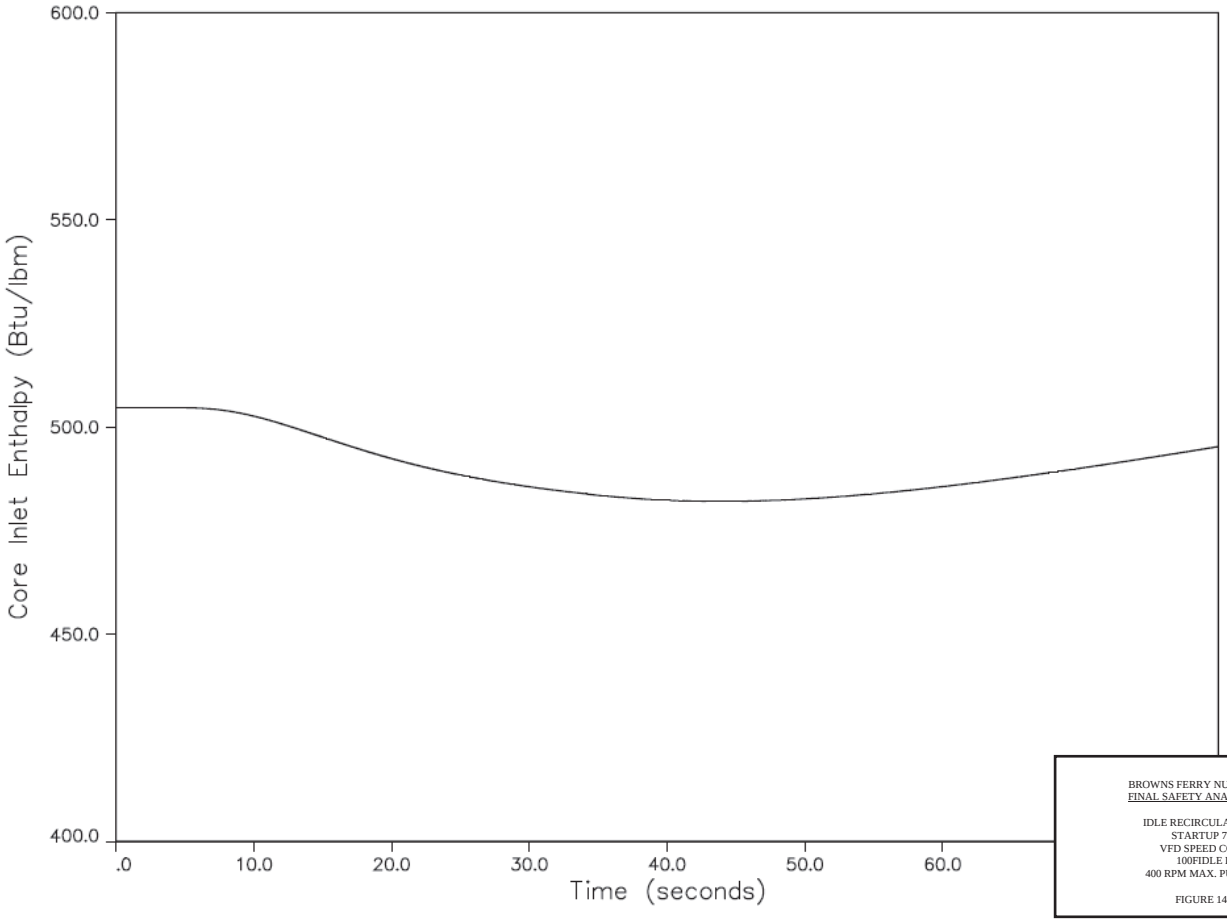


Figure 14.5-25e

105% OLTP Idle Loop Startup at 75P/52F
Core Inlet Enthalpy



BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

IDLE RECIRCULATION LOOP
STARTUP 75P/52F
VFD SPEED CONTROL
100FIDLE LOOP
400 RPM MAX. PUMP SPEED

FIGURE 14.5-25e

Figure 14.5-25f

105% OLTP Idle Loop Startup at 75P/52F
Loop Flows

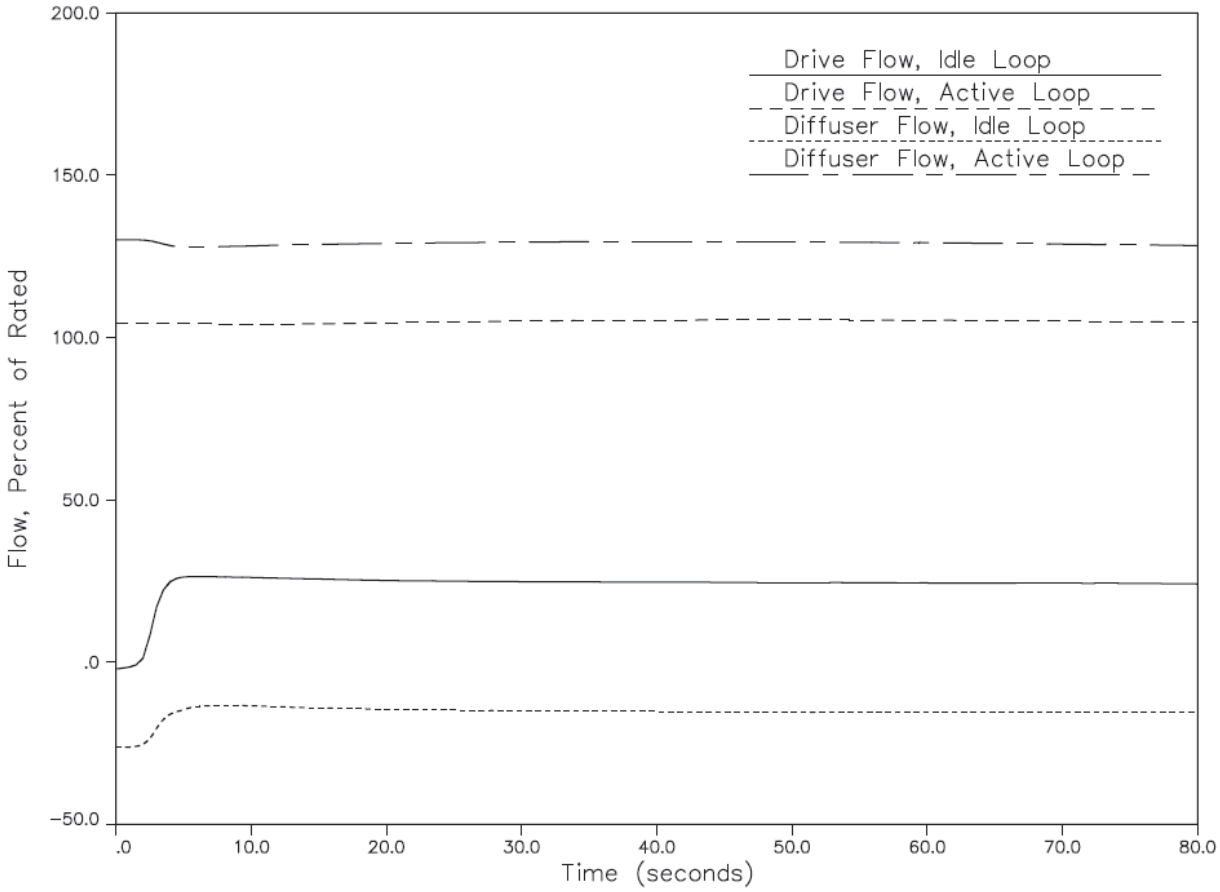


Figure 14.5-26a

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Figure 14.5-26b

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Figure 14.5-27a

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Figure 14.5-27b

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Figure 14.5-28

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Figure 14.5-28a

FWCF at 77.6P/109F with TSSS
Key Parameters

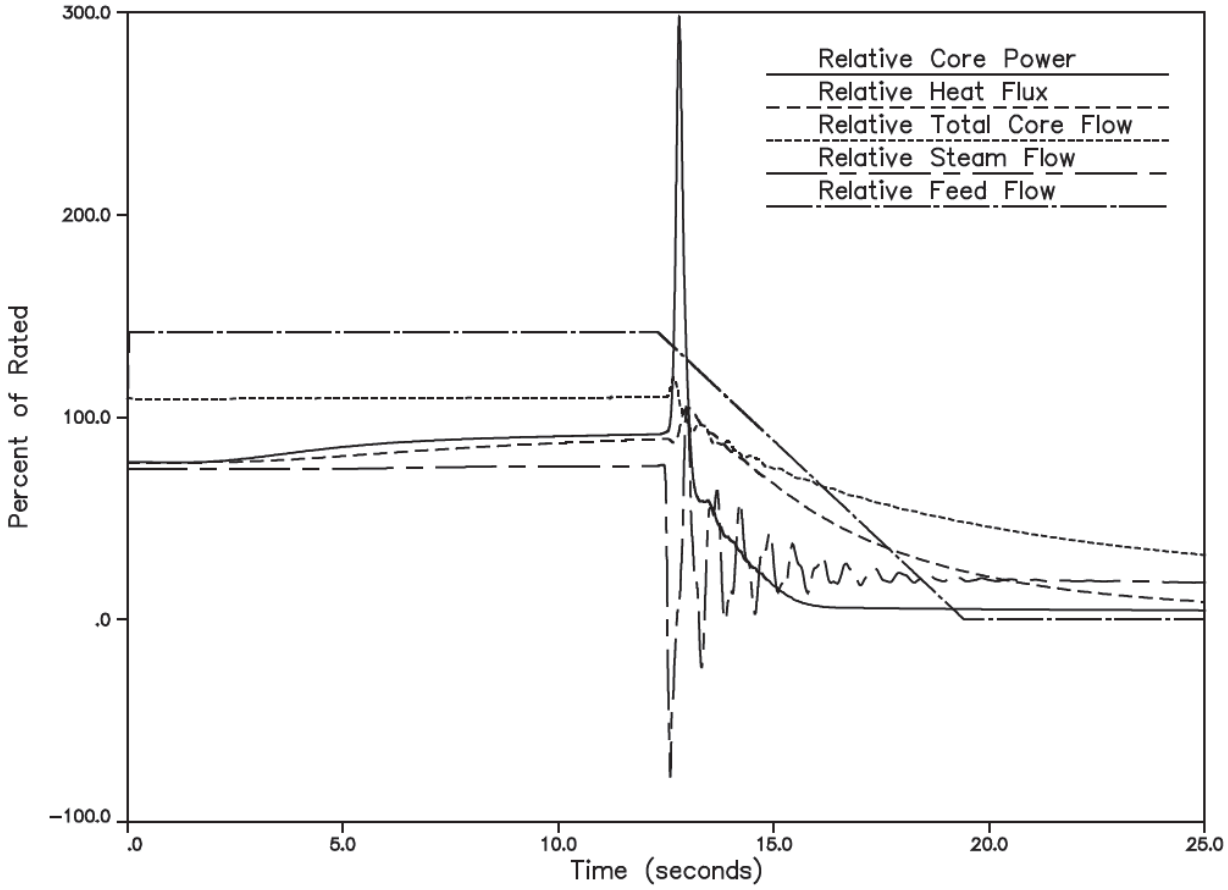

Figure 14.5-28b

FWCF at 77.6P/109F with TSSS
Sensed Water Level

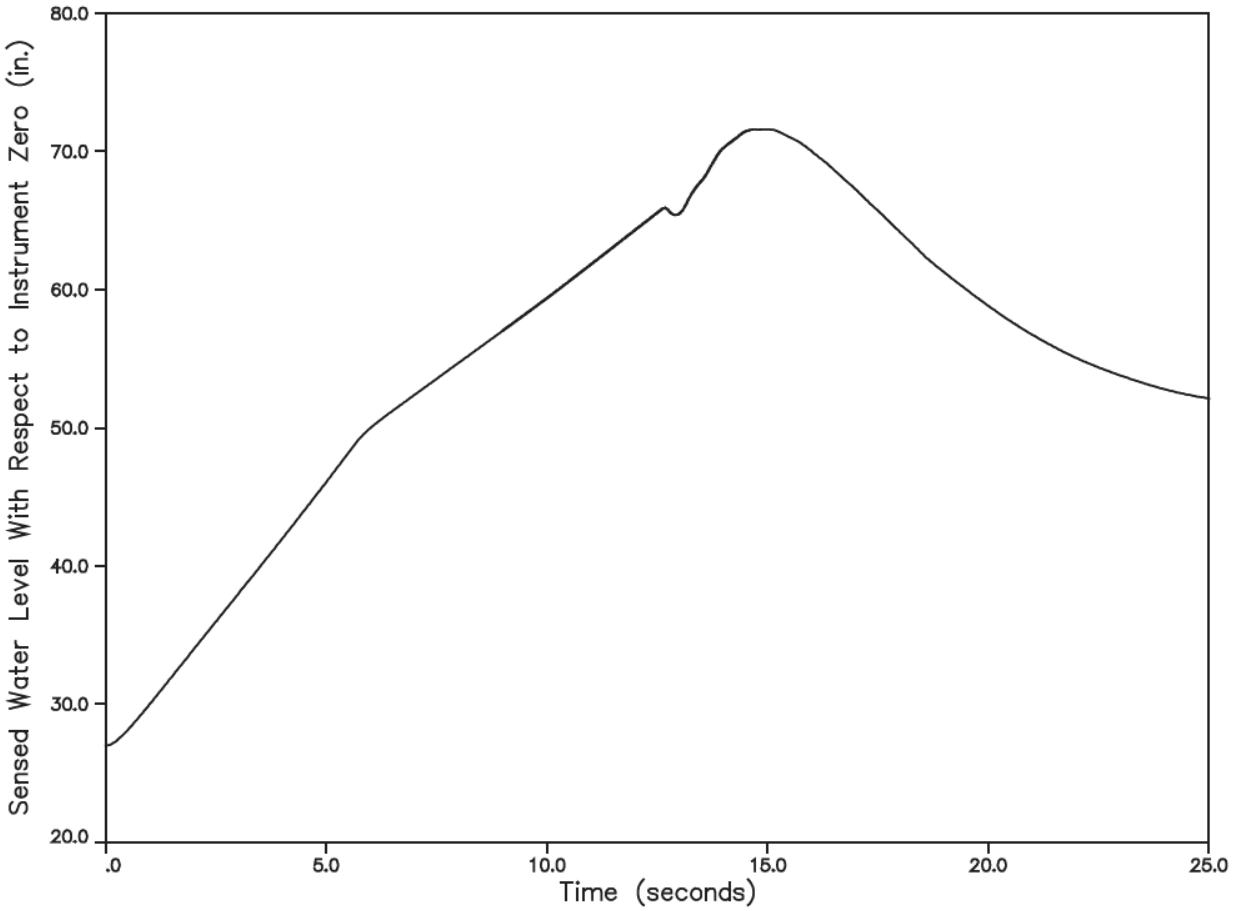

Figure 14.5-28c

FWCF at 77.6P/109F with TSSS
Vessel Pressures

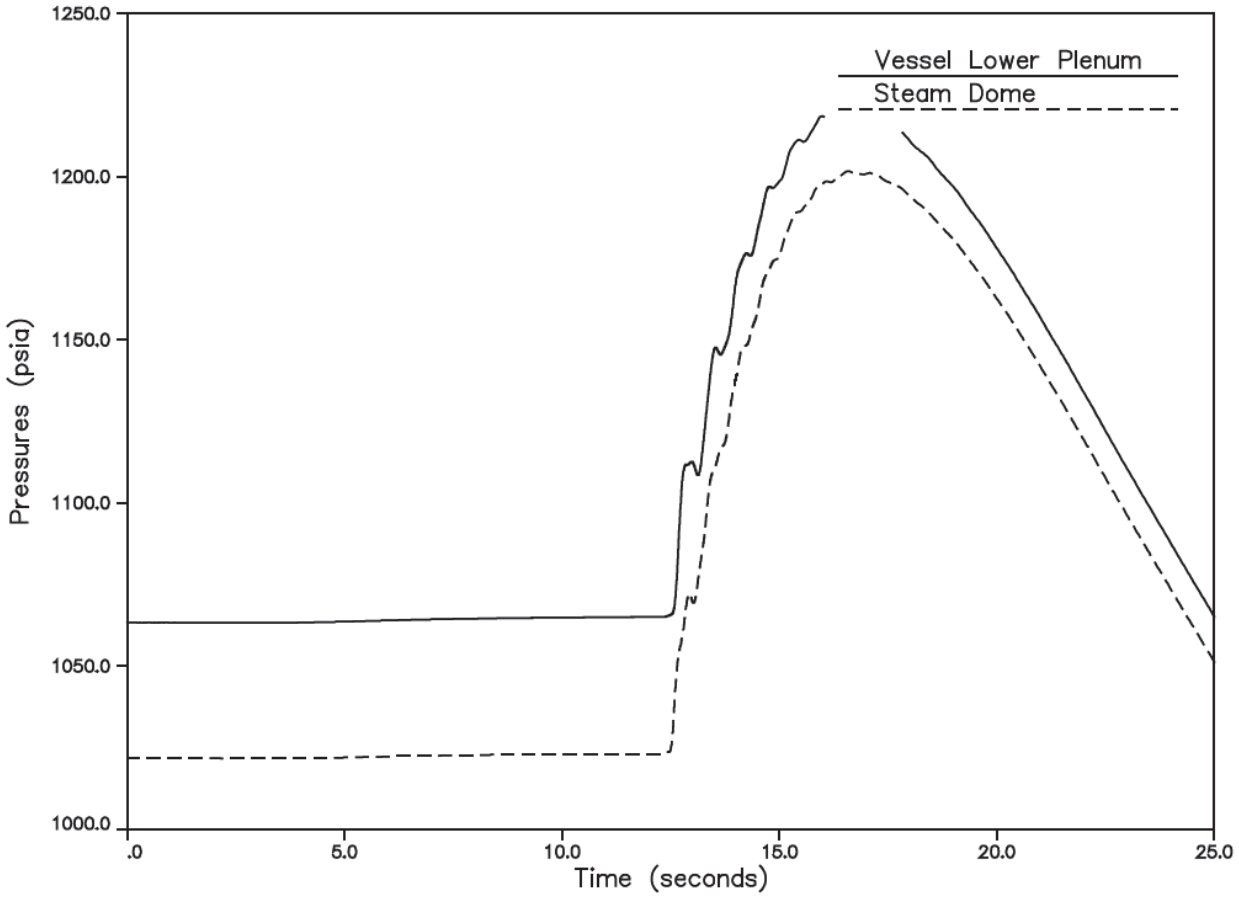

Figure 14.5-29

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Figure 14.5-30

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Figure 14.5-30a

FWCF with EOC-RPT-OOS at 100P/105F with TSSS
Key Parameters

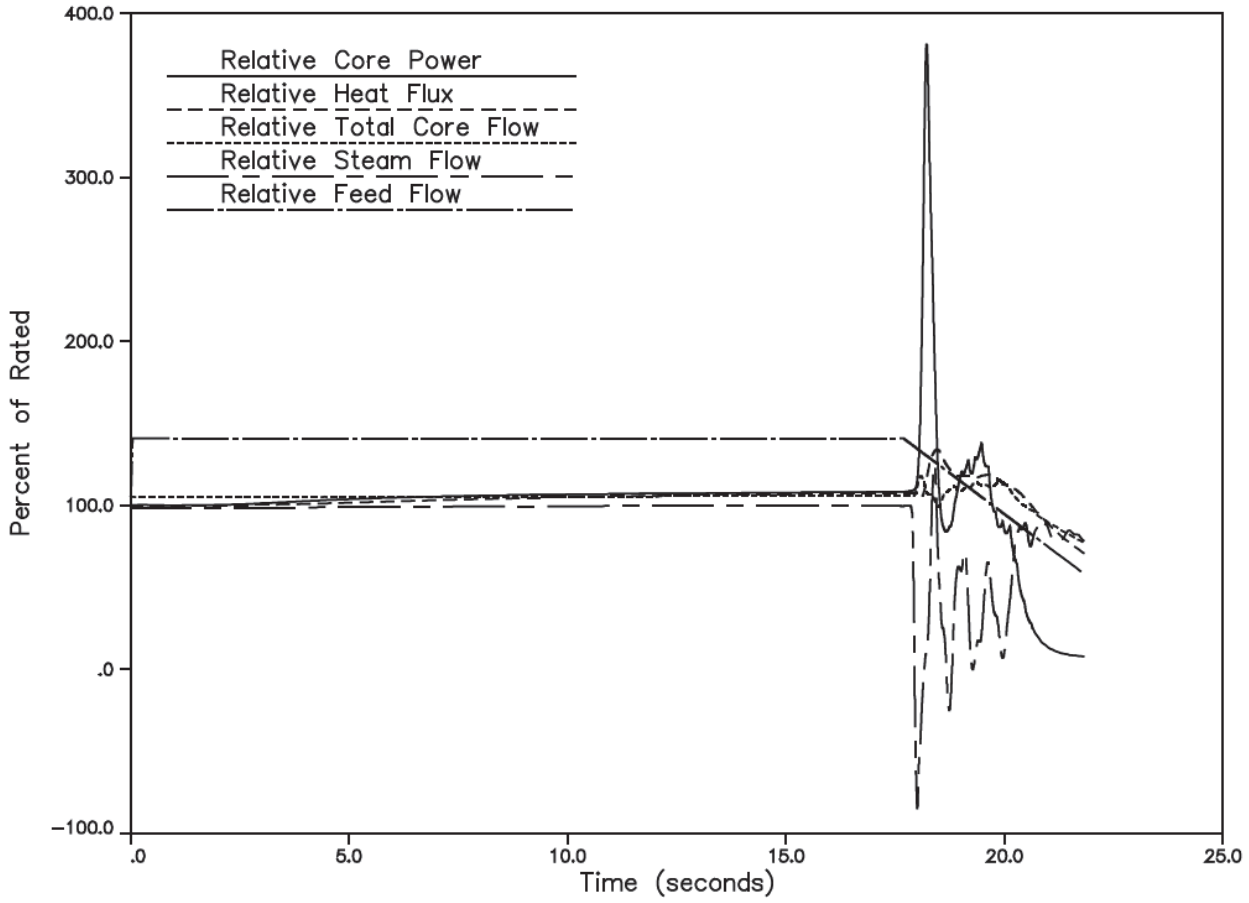

Figure 14.5-30b

FWCF with EOC-RPT-OOS at 100P/105F with TSSS
Sensed Water Level

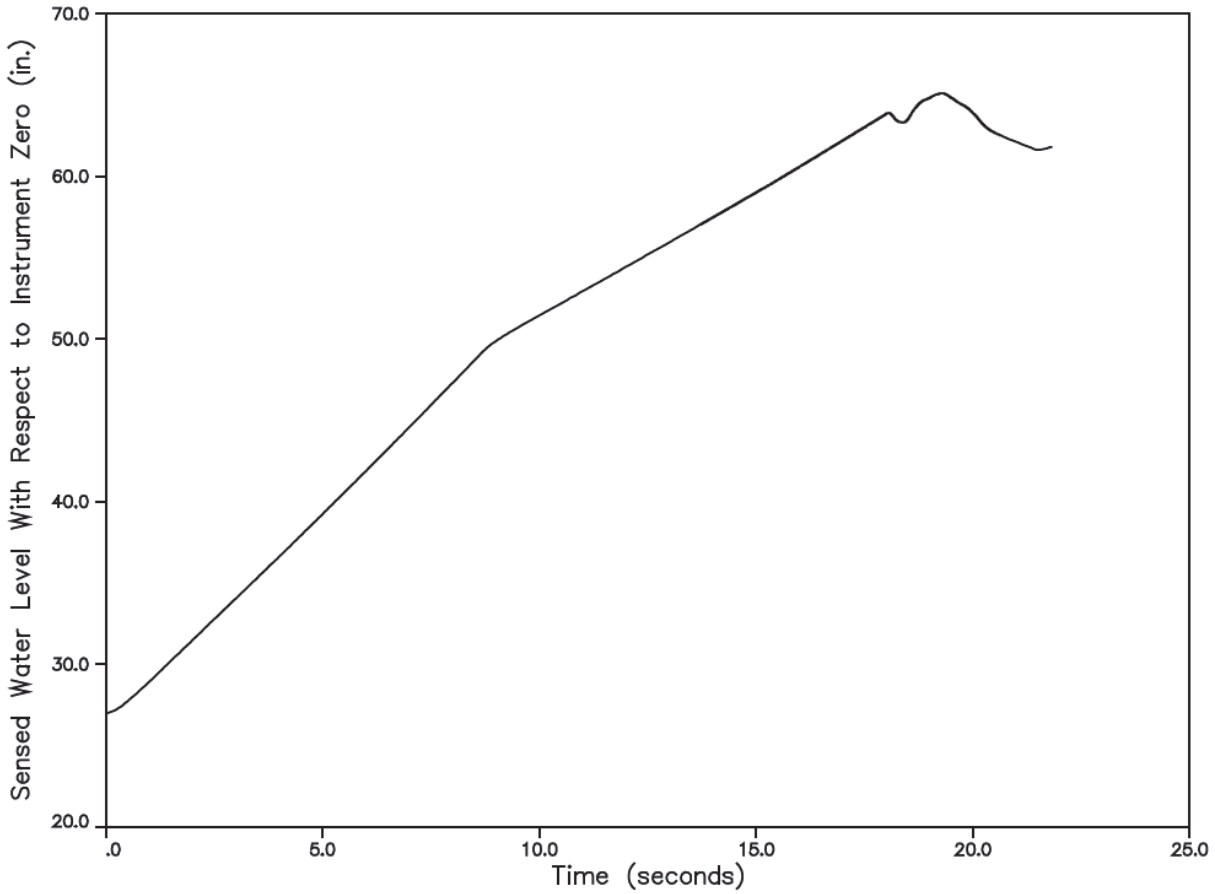

Figure 14.5-30c

FWCF with EOC-RPT-OOS at 100P/105F with TSSS
Vessel Pressures

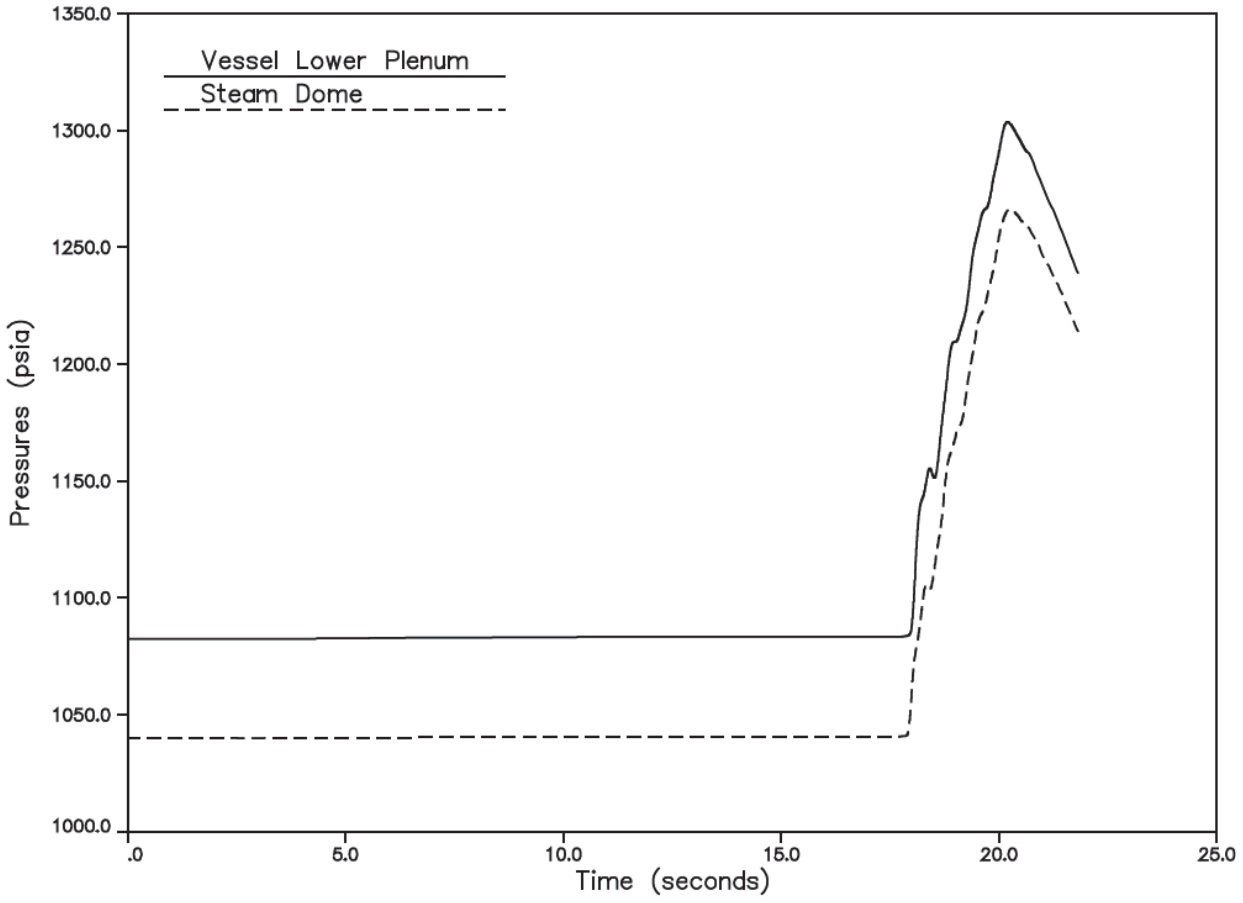

Figure 14.5-31

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Figure 14.5-31a

FWCF with TBVOOS at 100P/105F with TSSS
Key Parameters

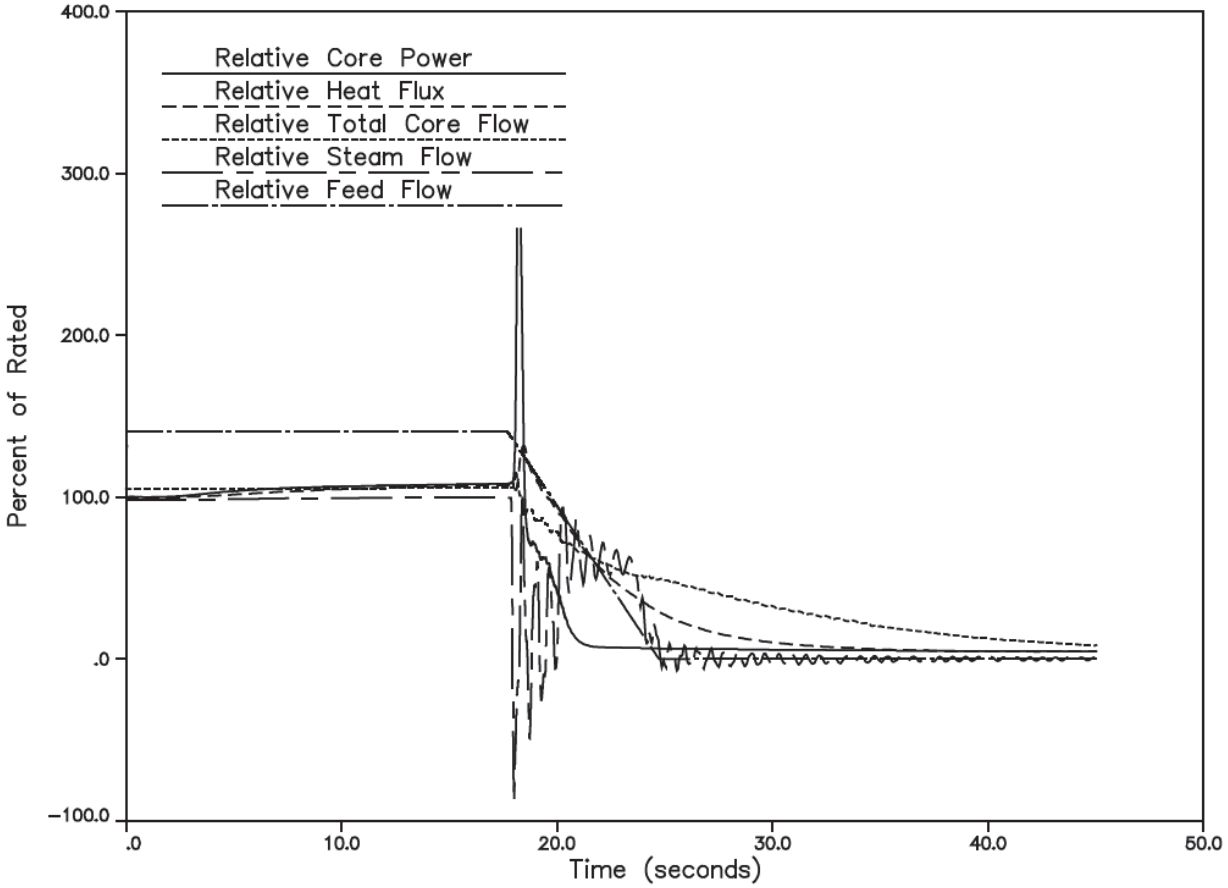


Figure 14.5-31b

FWCF with TBVOOS at 100P/105F with TSSS
Sensed Water Level

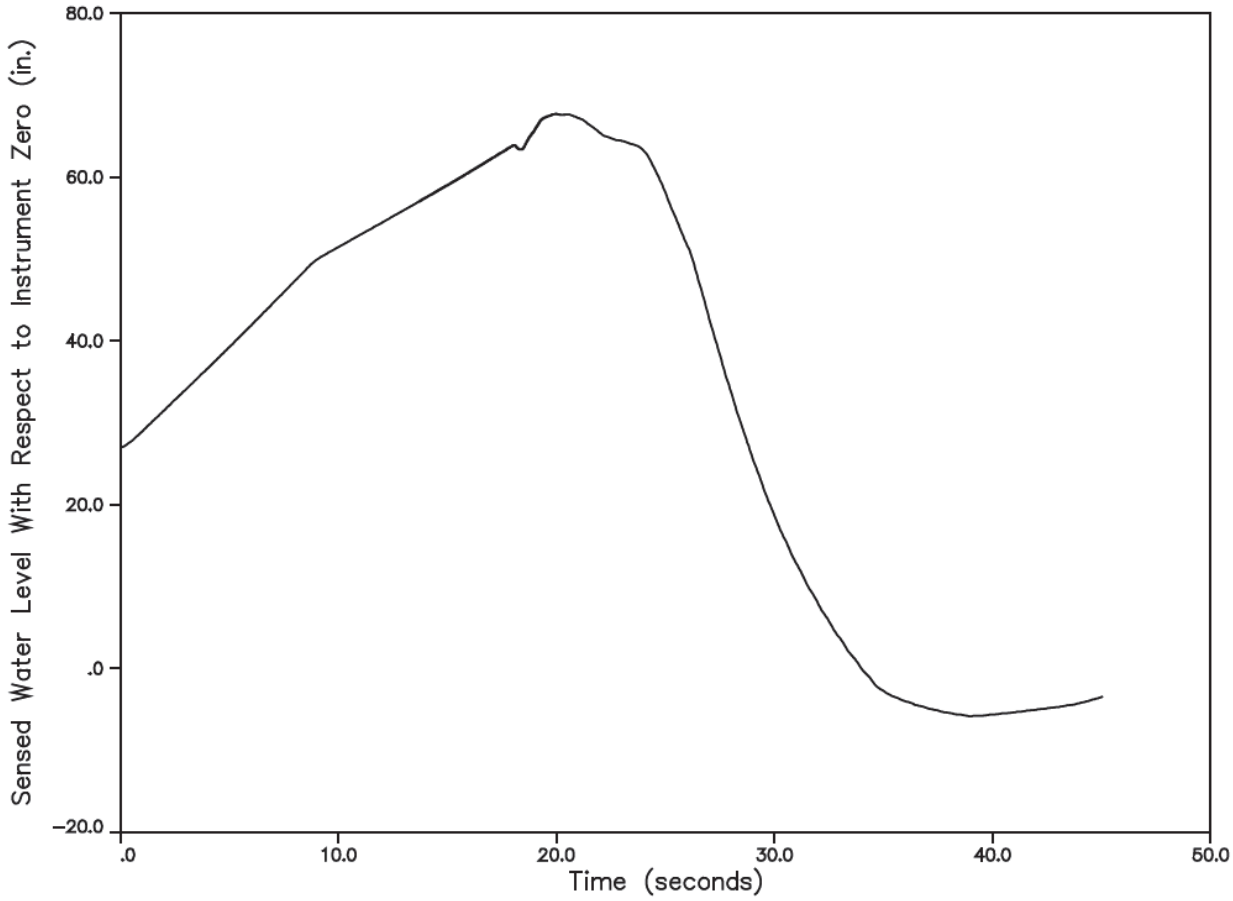
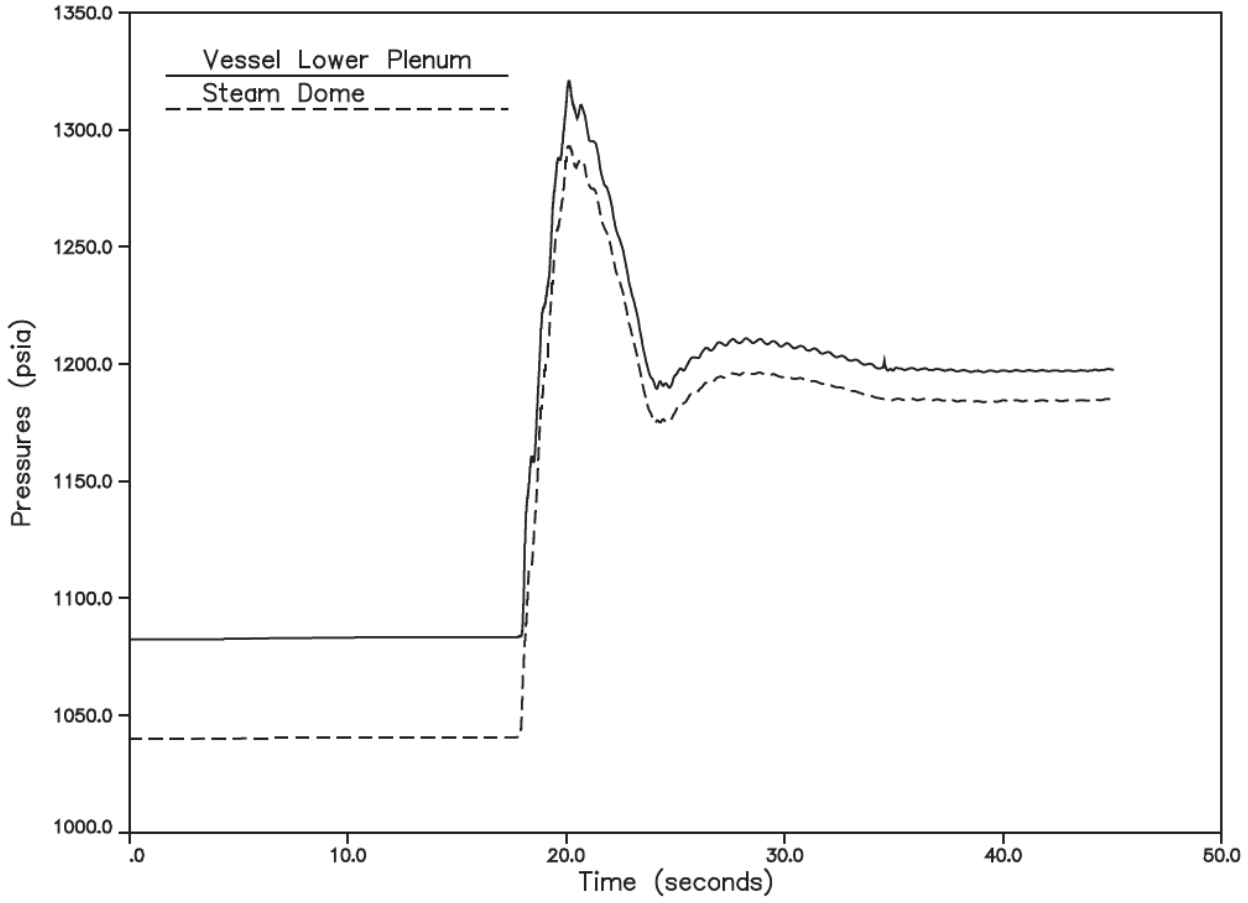


Figure 14.5-31c

FWCF with TBVOOS at 100P/105F with TSSS
Vessel Pressures



14.6 ANALYSIS OF DESIGN BASIS ACCIDENTS

This section contains general descriptions of the evaluation of design basis accidents for BFN Units 1, 2, and 3.

14.6.1 Introduction

The methods described in Subsection 14.4 for identifying and evaluating accidents have resulted in the establishment of design basis accidents for the various accident categories as follows:

	Accident Category	Design Basis Accident
a.	Accidents that result in radioactive material release from the fuel with the nuclear system process barrier, primary containment, and secondary containment initially intact.	Rod drop accident (single control rod)
b.	Accidents that result in radioactive material release directly to the primary containment.	Loss-of-coolant accident (rupture of one recirculation loop).
c.	Accidents that result in radioactive material release directly to the secondary containment with the primary containment initially intact.	Accidents in this category are less severe than those in categories "d" and "e", below.
d.	Accidents that result in radioactive material release directly to the secondary containment with the primary containment not intact.	Refueling accident (fuel assembly drops on spent fuel during refueling).
e.	Accidents that result in radioactive material releases outside the secondary containment.	Steam line break accident (main steam line breaks outside of secondary containment).

An investigation of accident possibilities reveals that accidents in category "c" are less severe than those in categories "d" and "e". There are two varieties of

accidents in category "c": (1) failures of the nuclear system process barrier inside the secondary containment, and (2) failures involving fuel that is located outside the primary containment but inside the secondary containment. Under the accident selection rules described in Subsection 14.4, a main steam line break inside the reactor building is the most severe accident of the first variety; but this accident results in a radioactivity release to the environs no greater than that resulting from the main steam line break outside the secondary containment. Similarly, the most severe accident of the second variety is the dropping of a fuel assembly during refueling. Because the consequences of accidents in category "c" are less severe than those resulting from similar accidents in other categories, the accidents in category "c" are not described.

14.6.2 Control Rod Drop Accident (CRDA)

The accidents that result in releases of radioactive material from the fuel with the nuclear system process barrier, primary containment, and secondary containment initially intact are the results of various failures of the Control Rod Drive System. Examples of such failures are collet finger failures in one control rod drive mechanism, a control drive system pressure regulator malfunction, and a control rod drive mechanism ball check valve failure. None of the single failures associated with the control rods or the control rod system results in a greater release of radioactive material from the fuel than the release that results when a single control rod drops out of the core after being disconnected from its drive and after the drive has been retracted to the fully withdrawn position. Thus, this control rod drop accident is established as the design basis accident for the category of accidents resulting in radioactive material release from the fuel with all other barriers initially intact. A highly improbable combination of actual events would be required for the design basis control rod drop accident to occur. The actual events required are as follows:

a. Failure of the rod-to-drive coupling. The design of the coupling itself reduces the probability of separation. Tests conducted under both simulated reactor conditions and the conditions more extreme than those expected in reactor service have shown that the coupling does not separate, even after thousands of scram cycles. Tests also show that the coupling does not separate when subjected to forces 30 times greater than that which can be achieved by normal control rod drive operation.

b. Sticking of the control rod in its fully inserted position as the drive is withdrawn. The control rods are designed to minimize the probability of sticking in the core. The control rod blades, which are equipped with rollers or spacer pads at the top of the control rod blade and rollers at the bottom that make contact with the channel walls, travel in gaps between the fuel assemblies with approximately 1/2-inch total clearance. Control rods of similar design, now in use in operating reactors, have exhibited no tendency to stick in the core due to distortion or swelling of the blade.

c. Full withdrawal of the control rod drive.

d. Failure of the operator to notice the lack of response of neutron monitoring channels as the rod drive is withdrawn.

e. Failure of the operator to verify rod coupling. The control rod bottoms on a seal preventing the control rod drive from being withdrawn at the overtravel position. Attempting to withdraw a control rod drive to the overtravel position provides a method for verifying rod coupling: this verification is required whenever neutron monitoring equipment response does not indicate that the rod is following the drive.

The CRDA is a limiting event that is impacted by core and fuel design, and thus it must be considered for each reload cycle. An improved Rod Worth Minimizer incorporating a "Banked Position Withdrawal Sequence" (BPWS) has been developed which greatly reduces the maximum control rod worth that could occur during an CRDA such that in all cases the peak fuel enthalpy is much less than the acceptance criteria of 280 cal/gm. A bounding generic evaluation[1] of the CRDA for all BWRs and fuel designs has been performed by GE for plants utilizing the BPWS. The cycle specific CRDA results are provided in the Reload Licensing Analysis Report.

The BPWS is a methodology to minimize potential enthalpy deposition within the context of generic analyses. BPWS maintains incremental rod worths to relatively low values.

BPWS is effective on a generic basis for all production line reactors and all fuel designs currently in use for initial, reload, and equilibrium core designs.

14.6.2.1 (Deleted)

14.6.2.2 CRDA Analysis and Results for AREVA Licensed Reload Cores

The AREVA analytical methods, assumptions, and conditions for evaluating the excursion aspects of the control rod drop accident have been reviewed and approved by the NRC. Analyses are performed assuming BPWS rules or equivalent are in force to limit dropped rod worths to reasonable values.

The AREVA cycle specific application of the generic CRDA methodology shows that peak deposited enthalpies do not exceed 280 cal/g. For AREVA methods, the most limiting condition to experience a CRDA occurs in the hot standby state. The reload fuel vendors' CRDA methodology conservatively assumes an adiabatic boundary

[1] NEDE 24011-P-A, GESTAR II

condition at the pellet-gap interface and no direct moderator heating. This prevents heat transfer from the fuel rod to the coolant, thus the deposited enthalpy is equivalent to the energy produced in the fuel. Doppler feedback limits the excursion until the rods are fully inserted.

The core at the time of rod drop accident is assumed to contain no xenon, to be in a hot-startup condition, and to have the control rods in a sequence consistent with BPWS rules or equivalent. For conservatism, eight rods are assumed to be inoperable and remain in the fully inserted position. The location of the inoperable rods are chosen to conservatively increase the worth of the dropped rod. Since the maximum incremental rod worth is maintained at very low values (by BPWS rules or equivalent), the postulated CRDA does not result in peak enthalpies in excess of 280 calories per gram.

The radiological evaluations are based on the assumed failure of 850 fuel rods of a GE fuel type which bound the radiological releases for all fuel rod types in the current core. In the AREVA analysis, rods with deposited enthalpies exceeding 170 cal/g are assumed to fail. If the number of rods exceeding the failure threshold is shown to be below 850 dose equivalent rods, it is concluded that the current radiological evaluation remains applicable.

The results of the peak enthalpy calculation for the current reload cycle are presented in the Reload Licensing Analysis Report. These results demonstrate that the maximum incremental rod worth is below the worth required to result in a CRDA which would exceed 280 cal/g peak fuel enthalpy and that the fuel failures predicted (if any) are fewer than those assumed in the radiological evaluation of record. The conclusion is that the 280 cal/g threshold is protected and that the radiological evaluation accounting for 850 failed fuel rods remains applicable for AREVA fuel.

14.6.2.3 Fuel Damage

Fuel rod damage estimates were initially based upon the UO_2 vapor pressure data of Ackerman[6] and interpretation of all the available SPERT, TREAT, KIWI, and PULSTAR test results which show that the immediate fuel rod rupture threshold is about 425 cal/g. Two especially applicable sets of data come from the PULSTAR[7] and ANL-TREAT[8] tests. The PULSTAR tests, which used UO_2 pellets of six percent enrichment with Zr-2 cladding, achieved maximum fuel enthalpies of about 200 cal/g with a minimum period of 2.83 milliseconds. The coolant flow was by natural

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6 Ackerman, R. J., Gilles, W. P., and Thorn, R. J.: "High Temperature Vapor Pressure of UO_2," Journal of Chemical Physics, December 1956, Vol. 25, No. 6.

7 MacPhee, J., and Lumb, R. F.: "Summary Report, PULSTAR Pulse Tests-II," WNY-020, February 1965.

8 Baker, L., Jr., and Tevebaugh, A. D.: "Chemical Engineering Division Report, January-June 1964, Section V - Reactor Safety," ANL-6900.

convection. Film boiling occurred, and there were local clad bulges; however, fuel pin integrity was maintained, and there were no abnormal pressure rises.

The two ANL-TREAT tests used Zircaloy clad UO_2 pins with energy inputs of 280 and 450 calories per gram, respectively.

	Test 1	Test 2
Input Energy (cal/g)	280	450
Final Mean Particle Diameter (mils)	60	30
Pressure Rise Rate (psi/sec)	30	60

The ultimate degree of fuel fragmentation and dispersal of the two cases is not significantly different; however, the pressure rise rate in the higher energy test is increased by a factor of 20. This strongly implies that the dispersion rate in the higher energy test was significantly higher than that of the lower energy test. This leads to the logical conclusion that although a high degree of fragmentation occurs for fuel in the 200 to 300 calories per gram range, the breakup and dispersal into the water is gradual and pressure rise rates are very modest. On the other hand, for fuel above the 400 calories per gram range, the breakup and dispersal is prompt; and much larger pressure rise rates are probable.

Based on the analysis of the above referenced data, it was estimated that 170 cal/g is the threshold for eventual fuel cladding perforation. Fuel melting is estimated to occur in the 220 to 280 cal/g range, and a minimum of 425 cal/g is required to cause immediate rupture of the fuel rods due to UO_2 vapor pressures.

14.6.2.4 BPWS Analysis for GE Analyzed Reload Cores

The accident is analyzed for both the startup range and the power range. The cold startup state will refer to a critical reactor with fuel and moderator temperatures of 20°C, a reactor pressure of one atmosphere, and an initial power fraction of 10^{-8} of rated power level. The hot startup conditions will be defined as a critical reactor at operating pressure, saturated temperature, and initial power fractions of 10^{-6} of rated. Hot standby will be used to define a reactor which is producing sufficient power to maintain all electrical systems without the aid of auxiliary power. This is usually in the 5 to 10% power range. From these definitions, it is obvious that the cold startup and hot startup states will be in the startup range; and that the hot standby case will be in the power range.

For the generic BPWS analysis, the fuel designs considered included a single enrichment design with uniform axial gadolinium (Type 1 fuel), a single enrichment design with axially distributed gadolinium (Type 2 fuel), and a mixed enriched, three radial region design (Type 3 fuel). Then the incremental control rod worths were calculated for the Type 1, Type 2, and Type 3 fuel designs for 368, 560, and 748 bundles size cores. These size cores were utilized to represent cores of the general small, medium and large sizes. The highest incremental control rod worth encountered for any of these fuel designs and core sizes was calculated as the beginning of the equilibrium cycle with Type 3 fuel in a 748 bundle size core. This incremental reactivity worth was 0.0083 Δk.

A design basis control rod drop accident with a control rod worth of 0.0083 Δk would result in a peak fuel enthalpy of 135 Cal/g. Since the calculated incremental control rod worth for all other conditions analyzed is less than 0.0083Δk, it follows that the resultant peak full enthalpy due to a design basis control rod accident within the constraints of the BPWS will be less than or equal to 135 Cal/g which is less than both the 170 cal/g and 280 cal/g criteria discussed above.

14.6.2.5 Fission Product Release From Fuel

The following assumptions were used in the initial calculation of fission product activity release from the fuel.

a. Eight hundred fifty fuel rods fail, per General Electric (GE) Licensing Topical Report, NEDO-31400A.

b. The reactor has been operating at design power (with a 1.02 uncertainty factor) with an average fuel burn-up of 35 to 39 GWd/MT prior to the accident. This assumption results in equilibrium concentration of fission products in the fuel. The rods that have failed are assumed to have operated at a power peaking factor of 1.5[9].

c. Of the rods that fail, 0.77% of the fuel melts, per NEDO-31400A. The following percentages of radioactive material are released to the reactor coolant from the failed fuel rods[9]:

[9] Regulatory Guide 1.183 and NUREG-0800, Section 15.4.9.

Radionuclide Group	Non-Melted Rods	Melted Rods
Noble Gases	10%	100%
Iodine	10%	50%
Other Halogens	5%	30%
Alkali Metals	12%	25%
Tellerium Group	0%	5%
Barium, Strontium	0%	2%
Noble Metals	0%	0.25%
Cerium Group	0%	0.05%
Lanthanum Group	0%	0.02%

14.6.2.6 Fission Product Transport

The following assumptions were used in calculating the amounts of fission product activity transported from the reactor vessel to the main condenser (initial core):

a. Of the radioactive material released from the fuel, 100% of the noble gases, 10% of the iodines, and 1% of the remaining radionuclides are assumed to reach the turbines and condensers[9].

b. Activity is assumed to be released from core instantaneously to the condenser.

14.6.2.7 Fission Product Release to Environs

The following assumptions and initial conditions were used in the calculation of fission product activity released to the environs:

a. On reaching the condenser, 100% of noble gases, 10% of iodines, and 1% of the particulate radionuclides are available for release to the environment. Radioactive decay during holdup in the low pressure turbine and condenser is assumed.

b. The accident is assumed to occur while condenser vacuum is being maintained with the mechanical vacuum pump (MVP). During normal operation, vacuum is maintained with the steam-jet-air ejector, the discharge, from which, is through a holdup (time delay) and filter system. The assumed

operation of the mechanical vacuum pump results in the discharge of the condenser activity directly to the environment via the elevated release point but without the benefits of holdup (decay) or filtration beyond the condenser.

c. All of the noble gas activity transferred to the condenser is assumed to be airborne in the condenser. The halogen and particulate activity transferred to the condenser experiences the removal effects of the condensate as described above.

d. The rate at which the condenser activity is discharged to the environment is dependent upon the free volume of the turbine and condenser and the discharge rate of the mechanical vacuum pump. The numerical values appropriate to these parameters are 187,000 ft^3 (low pressure turbine volume plus condenser free volume) and 1,850 cfm mechanical vacuum pump discharge rate.

e. A continuous ground level release of 20 cfm occurs at the base of the stack. The 20 cfm leakage mixes within the rooms at the base of the stack (34,560 ft^3, 50% of 69,120 ft^3 because of incomplete mixing).

f. Atmospheric dispersion coefficients, X/Q, for elevated releases under fumigation conditions, elevated releases under normal atmospheric conditions and ground level releases at the base of the stack are used. X/Q values applicable to the time periods, distances, and geometric relationships (offsite and control room) are shown in Table 14.6-8. Control room X/Q values for the base of the stack releases are calculated using the computer code ARCON96. For sites, such as BFN, with control room ventilation intakes that are close to the base of tall stacks, ARCON96 underpredicts the X/Q values for top of stack releases; therefore, top of stack releases to the control room intakes are evaluated using the methods of Regulatory Guides 1.145 and 1.111.

g. The maximum control room X/Q for the top and bottom of the stack releases is used for each time period. The effective X/Q is a factor of two less than the values listed because of the dual air intake configuration of the control bay ventilation (i.e., one intake is not contaminated).

Based upon these conditions, the fission product release rate to the environment is shown in Table 14.6-1.

14.6.2.8 Radiological Effects

The BFN analysis for the CRDA consists of two potential release paths; condenser leakage at 1% per day into the turbine building or through SJAE and offgas system as analyzed by the NEDO-31400A, and the MVP discharge as analyzed in accordance with Regulatory Guide 1.183. The “worst-case” radiological exposure

resulting from the activity discharged from a CRDA and a Regulatory Guide 1.183 source term would be from the MVP release path. The resulting control room dose is less than the 10 CFR 50.67 limit of 5 Rem TEDE. The EAB and LPZ doses from the MVP are well below the Regulatory Guide 1.183 reference values of 6.3 REM TEDE.

The dominant contributor to dose for the CRDA is Iodine 131 (I-131). Table 14.6-1 shows the I-131 activity in four locations (main condenser, stack room, control room, and environment) for the full 30 days of the dose calculation described above. This is an output of the RADTRAD computer code (NUREG/CR-6604) used for the CRDA dose analysis. Radioactive decay is considered in all locations except the environment (i.e., the environment represents a summation of all activity released). The environmental release totals approximately 10 percent of the activity initially reaching the main condenser. The main condenser is depleted of 95% of the activity by about five hours. This is consistent with an 1850 cfm exhaust rate and a 187,000 ft^3 volume (i.e., a release rate of about 0.6 volumes per hour).

14.6.3 Loss of Coolant Accident (LOCA)

Accidents that could result in release of radioactive material directly into the primary containment are the results of postulated nuclear system pipe breaks inside the drywell. All possibilities for pipe break sizes and locations have been investigated including the severance of small pipe lines, the main steam lines upstream and downstream of the flow restrictors, and the recirculation loop pipelines. The most severe nuclear system effects and the greatest release of radioactive material to the primary containment results from a complete circumferential break of one of the recirculation loop pipelines. This accident is established as the design basis loss of coolant accident.

ECCS cooling performance must be calculated in accordance with an acceptable evaluation model and must be calculated for a number of postulated loss-of-coolant accidents of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated loss-of-coolant accidents are calculated. For peak cladding temperatures and limiting break sizes, see Section 6.5.3.1.

LOCA models used for reload fuel analyses are described in EMF-2361(P)(A)[12] and ANP-3377P[13]. Plant specific information on models used and results of the LOCA analysis for the current operating cycle is given in a separate document prepared in conjunction with the reload licensing amendments. Additional information on the sequence of events during a LOCA and the response of the primary containment during a LOCA is given in ANP-3377P[13] and NEDO-10320[14].

14.6.3.1 Initial Conditions and Assumptions

The analysis of this accident is performed using the following assumptions:

a. The reactor is operating at the most severe condition at the time the recirculation pipe breaks, which maximizes the parameter of interest: primary containment response, fission product release, or Core Standby Cooling System requirements.

b. A complete loss of normal AC power occurs simultaneously with the pipe break. This additional condition results in the longest delay time for the Engineered Safeguards.

c. The recirculation loop pipeline is considered to be instantly severed. This results in the most rapid coolant loss and depressurization with coolant discharged from both ends of the break.

d. One active single failure within the plant is postulated to occur concurrent with the pipe break.

e. A seismic event is neither postulated to occur concurrently with the LOCA nor as a initiator of the pipe break.

14.6.3.2 Nuclear System Depressurization and Core Heatup

In Section 6, "Core Standby Cooling Systems," the initial phases of the loss of coolant accident are described and evaluated. Included in that description are the rapid depressurization of the nuclear system, the operating sequences of the Core Standby Cooling Systems, and the heatup of the fuel.

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[12] EMP-2361(P)(A), Revision 0, EXEM BWR-2000 ECCS Evaluation Model, Framatome ANP Inc., May 2001 as supplemented by the site-specific approval in NRC safety evaluation dated April 27, 2012, February 15, 2013, and July 31, 2014.

[13] ANP-3377P, Revision 3, Browns Ferry Units 1, 2, and 3 LOCA Break Spectrum Analysis for ATRIUM 10XM Fuel (EPU), AREVA NP, Inc., August 2015.

[14] The General Electric Pressure Suppression Containment Analytical Model, NEDO-10320.

14.6.3.3 Primary Containment Response

For Units 1, 2, and 3 primary containment response, refer to Section 14.12.

14.6.3.4 Fission Products Released to Primary Containment

The following assumptions and initial conditions were used in calculating the amounts of fission products released from the nuclear system to the drywell:

a. Source terms based on the ORIGEN computer code with a 1.02 multiplier per Regulatory Guide 1.183.

b. The reactor has been operating at design power (3952 MWt) for a 24 month fuel cycle. The average fuel burnup is 35 to 39 GWd/MT prior to the accident.

c. The radionuclides considered include those identified as being potentially important contributors to TEDE in NUREG/CR-6604.

d. The core inventory release fractions, timing, and chemical form are those specified in Regulatory Guide 1.183. Table 14.6-7 gives the bounding core inventory of each isotope.

14.6.3.5 Fission Product Release From Primary Containment

Fission products are released from the primary containment to the secondary containment via primary containment penetration leakage at the Technical Specification leakage limit. Primary containment atmosphere is released via main steam isolation valve leakage to the high and low pressure turbines and the condenser. Primary containment atmosphere is released directly to the Standby Gas Treatment System during operation of the Containment Atmospheric Dilution (CAD) System. Primary containment atmosphere is released above the Units 1, 2, and 3 Reactor Buildings via leakage of the hardened containment venting system isolation valves. The Emergency Core Cooling Systems (ECCS) leak into the secondary containment. The following assumptions were used in calculating the amounts of fission products released from the primary containment:

a. The primary containment minimum free volume (drywell and wetwell) is 278,400 ft^3. The drywell volume is 159,000 ft^3 and the torus gas space volume is 119,400 ft^3. The drywell torus gas space volumes are treated as separate volumes until after the activity release to the containment is complete and then these volumes are assumed to be well mixed. The activity release is entirely to the drywell.

b. The primary to secondary containment leak rate was taken as two percent volume per day (232 cfh).

c. The four main steam lines are assumed to leak a total of 150 scfh which is the Technical Specification limit.

d. The containment vent system flow path operates for a period of 24 hours at a flow rate of 139 cfm at 10 days, 20 days, and 29 days post-accident. This flow is filtered via the SGTS filters.

e. The hardened containment vent isolation valves leak a total of 10 scfh to the independent release points above the Unit 1, 2, and 3 Reactor Buildings. Release associated with leakage from the hardened containment vent isolation valves is assumed to begin at 11 hours.

f. Twenty gpm ECCS leakage into secondary containment in accordance with NUREG-0800, Section 15.6.5, Appendix B.

g. No credit is taken for spray removal in the containment.

h. Natural removal rates for particulates in the drywell are based on the correlations of NUREG-CR-6604. For elemental iodine, the natural removal coefficients for removal of plateout are based on the expressions of SRP 6.5.2.

i. For the purpose of suppression pool pH control, the accident is assumed to be a recirculation line break.

Additionally, an analysis evaluated the suppression pool pH in the event of a DBA LOCA involving fuel damage. The objective of the analysis was to demonstrate that the suppression pool pH remains at or above 7.0; thus, ensuring that the particulate iodine (Cesium Iodide - CsI) deposited into the suppression pool during this event does not re-evolve and become airborne as elemental iodine.

The calculation methodology was based on the approach outlined in NUREG-1465 and NUREG/CR-5950. Specifically, credit was taken for sodium pentaborate solution addition to the suppression pool water as a result of SLCS operation.

The initial effects on suppression pool pH come from rapid fission product transport and formation of cesium compound, which would result in increasing the suppression pool pH. As radiolytic production of nitric acid and hydrochloric acid proceeds and these acids are transported to the suppression pool over the first days of the event, the suppression pool water would become more acidic. The buffering

effect of SLCS injection within several hours is sufficient to offset the effects of these acids that are transported to the pool. Sufficient sodium pentaborate solution is available to maintain the suppression pool pH at or above 7.0 for 30 days post-accident.

14.6.3.6 Fission Product Release to Environs

Secondary Containment Releases

The fission product activity in the secondary containment at any time (t) is a function of the leakage rate from the primary containment, the volumetric discharge rate from the secondary containment and radioactive decay. During normal power operation, the secondary containment ventilation rate is 75 air changes per day; however, the normal ventilation system is turned off and the Standby Gas Treatment System (SGTS) is initiated as a result of low reactor water level, high drywell pressure, or high radiation in the Reactor Building. Any fission product removal effects in the secondary containment such as plateout are neglected. The fission product activity released to the environs is dependent upon the fission product inventory airborne in the secondary containment, the volumetric flow from the secondary containment, and the efficiency of the various components of the SGTS.

The following assumptions were used to calculate the fission product activity released to the environment from the secondary containment:

a. The primary containment atmosphere leakage to secondary containment mixes instantaneously and uniformly within the secondary containment.

b. The effective mixing volume of the secondary containment is 1,311,209 ft^3.

c. The SGTS removes fission products from secondary containment. If only two of the SGTS trains are in operation (i.e., SGTS flow of 16,200 cfm), a short period exists at the start of the accident during which the secondary containment becomes pressurized relative to the outside environment. However, negative pressure would be re-established in secondary containment prior to fission product release times specified by Regulatory Guide 1.183. Once the secondary containment pressure is reduced below atmospheric pressure, all releases from secondary containment to the environment are through the SGTS filters via the plant stack. If all three trains of SGTS are in operation (i.e., SGTS flow of 24,750 cfm), all releases to the environment from secondary containment are through the SGTS filters via the plant stack. The case with three trains in operation is the limiting condition.

d. The containment vent system flow path operates for a period of 24 hours at a flow rate of 139 cfm at 10 days, 20 days, and 29 days post-accident. This flow is filtered via the SGTS filters.

e. The ECCS systems leak reactor coolant directly to the secondary containment. The maximum water temperature is less than 212°F. The volume available for mixing is 1.31E5 ft^3. Ten percent of the iodine in the ECCS leakage is assumed to become airborne.

f. Filter efficiency for the SGTS was taken as 90 percent for organic and 0% inorganic (elemental) iodine.

g. Release to the environment from the plant stack is composed of two flow paths. A continuous ground level release of 20 cfm occurs at the base of the stack. This flow results from SGTS leakage through the backdraft dampers in the base of the stack. Subsection 5.3.3, "Secondary Containment System Description" describes the backdraft dampers. The 20 cfm leakage mixes uniformly within the rooms at the base of the stack (50% of the room volume of 69,120 ft^3). The remaining SGTS flow exits the stack at a height of 183 meters above ground elevation.

h. Fumigation conditions exist for 30 minutes when the post-accident control room accumulated dose rate is the maximum.

i. Atmospheric dispersion coefficients, X/Q, for elevated releases under fumigation conditions, elevated releases under normal atmospheric conditions and ground level releases at the base of the stack are used. X/Q values applicable to the time periods, distances, and geometric relationships (offsite and control room) are shown in Table 14.6-8. Control room X/Q values for the base of the stack releases are calculated using the computer code ARCON96. For sites, such as BFN, with control room ventilation intakes that are close to the base of tall stacks, ARCON96 underpredicts the X/Q values for top of stack releases; therefore, top of stack releases to the control room intakes are evaluated using the methods of Regulatory Guides 1.145 and 1.111.

j. The maximum control room X/Q for the top and bottom of the stack releases is used for each time period. Note that the effective X/Q is a factor of two less than the values listed because of the dual air intake configuration of the control bay ventilation.

k. The hardened containment vent isolation valves leak a total of 10 scfh to independent release points above the Unit 1, 2 and 3 Reactor Buildings with a delay of 11 hours for leakage to reach the release point. A bounding control room X/Q is used for each time period for this release path.

Main Steam Isolation Valve Leakage Releases

The leakage from primary containment via the MSIVs is transferred 1) to the main turbine (high pressure and low pressure) via the four steam lines and 2) to the condenser via the alternate leakage treatment (ALT) flow path formed by the steam line drain. The leakage from the turbine and condenser migrates to the turbine deck and subsequently is exhausted to the atmosphere via the turbine building roof vents with no credit for hold-up or removal in the Turbine Building. The path takes advantage of the large volume of the main steam lines and the condenser to hold up and plate out fission products in the MSIV leakage effluent. The following assumptions were used to calculate the fission product activity released to the environment from the turbine building:

a. The four main steam lines are assumed to leak a total of 150 scfh which is the Technical Specification limit. The direct leakage path to the turbines processes only 0.5% of the total leakage. The remainder goes to the condenser via the ALT flow path. The main steam piping from the outermost isolation valve up to the turbine stop valve, the bypass/drain piping to the main condenser and the main condenser will retain their structural integrity during and following a safe-shutdown earthquake (SSE).

b. Aerosol and elemental iodine removal due to sedimentation is credited in the main steam lines and in the main condenser. Aerosol settling velocities for sedimentation are determined for the steam lines and the main condenser per the AEB 98-03 distribution. Settling velocities are based on removal coefficients for the different volumes considering prior volume sedimentation removal. Elemental iodine removal in the steam lines utilizes the Bixler model of NUREG/CR-6604. The elemental iodine removal rate in the condenser is conservatively assumed to be the same as that for particulate.

c. The free volume of the low pressure turbines is 51,000 ft^3 and the effective volume of the condenser is 122,400 ft^3 (90% of the total condenser volume).

d. No credit is taken for holdup in the turbine building.

e. Ground level atmospheric dispersion coefficients, X/Q, for releases from the turbine building roof exhaust applicable to the time periods, distances, and geometric relationships (offsite and control room) are shown in Table 14.6-8. Control room X/Q values are calculated using the computer code ARCON96.

14.6.3.7 Radiological Effects

The LOCA provides the most severe radiological releases to the primary and secondary containments and, thus, serves as the bounding design basis accident in determining post-accident offsite and control room personnel doses.

Offsite Doses

Offsite doses of interest resulting from the activity released to the environment as a consequence of the loss of coolant accident are the maximum 2-hour TEDE for the exclusion area boundary (EAB) (1,465 meters), and the corresponding 30-day TEDE at the low population zone (LPZ) boundary (3,200 meters).

The offsite doses are calculated using the RADTRAD code (NUREG/CR-6604). RADTRAD is a radiological consequence analysis code used to model plan control volumes for radionuclide transport and removal and account for atmospheric dispersion of offsite and control room locations by use of appropriate X/Qs.

The largest calculated total offsite dose is well within the 10 CFR 50.67 limit.

Control Room

The control room doses are calculated using RADTRAD (NUREG/CR-6604). The model accounts for the atmospheric dispersion to the dual control room intakes by use of appropriate X/Qs and models the control bay habitability zone with no credit taken for the Control Room Emergency Ventilation System (CREVS) filters (i.e., 6717 cfm of unfiltered inleakage into the Control Room), occupancy times, breathing rates in accordance with Regulatory Guide 1.183 and calculates the TEDE. Atmospheric dispersion coefficients are based on release point, geometric relationship of the release point, and receptor and atmospheric conditions based on site specific meteorological data. The model accounts for the control room geometry (210,000 ft^3).

The direct gamma dose contribution from the piping inside secondary containment and the secondary containment atmosphere are included. One section of core spray piping in each unit is routed just outside the common Control Building/Reactor Building wall. This piping will be carrying suppression pool water in the event of a LOCA.

All of these exposure mechanisms (unfiltered pressurization flow, unfiltered inleakage, and direct dose) are combined to produce a total control room dose for the duration of the accident. Since CREVS has dual air intakes placed on opposite sides of the control building and can function with a single active failure in the inlet isolation system, in accordance with NUREG-0800, the control room dose is divided by a factor of 2 to account for dilution effects. The 30 day integrated post-accident doses in the control room are within the limits of 5 REM TEDE as specified in 10 CFR 50.67.

14.6.4 Refueling Accident

The current safety evaluation for the Refueling Accident is contained in the licensing topical report for nuclear fuel, "General Electric Standard Application For Reactor Fuel," NEDE-24011-P-A, and subsequent revisions thereto. Accidents that result in the release of radioactive materials directly to the secondary containment are events that can occur when the primary containment is open. A survey of the various plant conditions that could exist when the primary containment is open reveals that the greatest potential for the release of radioactive material exists when the primary containment head and reactor vessel head have been removed. With the primary containment open and the reactor vessel head off, radioactive material released as a result of fuel failure is available for transport directly to the reactor building.

Various mechanisms for fuel failure under this condition have been investigated. Refueling Interlocks will prevent any condition which could lead to inadvertent criticality due to control rod withdrawal error during refueling operations when the mode switch is in the Refuel position. The Reactor Protection System is capable of initiating a reactor scram in time to prevent fuel damage for errors or malfunctions occurring during deliberate criticality tests with the reactor vessel head off. The possibility of mechanically damaging the fuel has been investigated.

The design basis accident for this case is one in which one fuel assembly is assumed to fall onto the top of the reactor core.

The discussion in Subsections 14.6.4.1 and 14.6.4.2 provides the analyses for the dropping of a 7 x 7 assembly and a 8 x 8 assembly. The analyses for all current General Electric product line fuel bundle designs are contained in supplements to NEDE-24011-P-A. The NEDE evaluates each new fuel design against the 7x7 fuel design for the original core load. The 7x7 fuel handling accident resulted in 111 failed fuel rods. Evaluations of other fuel types have been performed as a comparison of the fuel damage to the 7x7 fuel design. The activity release for these other fuel types is bounded by the GE 7x7 case. The historical and current calculated doses are much less than the regulatory guidelines.

The refueling accident results documented in this section are applicable for fuel cycles containing an initial reload of new AREVA fuel, including the use of blended, low-enriched uranium (BLEU). The AREVA fuel load chain is different from GE assembly designs because the load is distributed through the center water channel rather than through the rods.

However, the failure mechanisms for the AREVA assemblies will produce similar number of rod failures as in the GE14 design. No other aspect of utilizing the AREVA fuel affects the current analysis; therefore, the current analysis remains applicable.

14.6.4.1 Assumptions

1. The fuel assembly is dropped from the maximum height allowed by the fuel handling equipment.

2. The entire amount of potential energy, referenced to the top of the reactor core, is available for application to the fuel assemblies involved in the accident. This assumption neglects the dissipation of some of the mechanical energy of the falling fuel assembly in the water above the reactor core and requires the complete detachment of the assembly from the fuel hoisting equipment. This is only possible if the fuel assembly handle, the fuel grapple, or the grapple cable breaks.

3. None of the energy associated with the dropped fuel assembly is absorbed by the fuel material (uranium dioxide).

14.6.4.2 Fuel Damage

Dropping a fuel assembly onto the reactor core from the maximum height allowed by the refueling equipment, less than 30 feet, results in an impact velocity of 40 ft/sec. The kinetic energy acquired by the falling fuel assembly is approximately 17,000 ft-lb for a 7 x 7 fuel bundle and approximately 18,150 ft-lb for a 8 x 8 fuel bundle. This energy is dissipated in one or more impacts. The first impact is expected to dissipate most of the energy and cause the largest number of cladding failures. To estimate the expected number of failed fuel rods in each impact, an energy approach has been used.

The fuel assembly is expected to impact on the reactor core at a small angle from the vertical possibly inducing a bending mode of failure on the fuel rods of the dropped assembly. Fuel rods are expected to absorb little energy prior to failure due to bending if it is assumed that each fuel rod resists the imposed bending load by two equal, opposite concentrated forces. Actual bending tests with concentrated point loads show that each fuel rod absorbs about 1 ft-lb prior to cladding failure. For rods which fail due to gross compression distortion, each rod is expected to absorb about 250 ft-lbs before cladding failure (this is based on 1 percent uniform plastic deformation of the rods). The energy of the dropped assembly is absorbed by the fuel, cladding, and other core structure. A fuel assembly consists of about 72 percent fuel, 11 percent cladding, and 17 percent other structural material by weight. Thus, the assumption that no energy is absorbed by the fuel material inserts considerable conservatism into the mass-energy calculations that follow.

The energy absorption on successive impacts is estimated by consideration of a plastic impact. Conservation of momentum under a plastic impact show that the fractional kinetic energy absorbed during impact is

$$1 - \frac{M_1}{M_1 + M_2}$$

where M_1 is the impacting mass and M_2 is the struck mass. Based on the fuel geometry within the reactor core, four fuel assemblies are struck by the impacting assembly. The fractional energy loss on the first impact is about 80 percent.

The second impact is expected to be less direct. The broad side of the dropped assembly impacts approximately 24 more fuel assemblies so that after the second impact only 135 ft-lbs (about 1 percent of the original kinetic energy) is available for a third impact. Because a single fuel rod is capable of absorbing 250 ft-lb in compression before cladding failure, it is unlikely that any fuel rods fail on a third impact.

If the dropped fuel assembly strikes only one or two fuel assemblies on the first impact, the energy absorption by the core support structure results in about the same energy dissipation on the first impact as in the case where four fuel assemblies are struck. The energy relations on the second and third impacts remain about the same as in the original case. Thus, the calculated energy dissipation is as following:

First impact	80 percent
Second impact	19 percent
Third impact	1 percent (no cladding failures)

The first impact dissipates 0.80 x 17,000 or 13,600 ft-lbs of energy for a 7 x 7 fuel bundle and 0.80 x 18,150 or 14,500 ft-lbs of energy for a 8 x 8 fuel bundle. It is assumed that 50 percent of this energy is absorbed by the dropped fuel assembly and that the remaining 50 percent is absorbed by the struck fuel assemblies. Because the fuel rods of the dropped fuel assembly are susceptible to the bending mode of failure, and because 1 ft-lb of energy is sufficient to cause cladding failure due to bending, all 49 (7 x 7 fuel bundle) or 62 (8 x 8 fuel bundle) rods of the dropped fuel assembly are assumed to fail. Because the 8 tie rods of each struck fuel assembly are more susceptible to bending failure than the other 41 rods, it is assumed that they fail upon the first impact. Thus 4 x 8 = 32 tie rods (total in four assemblies) are assumed to fail.

Because the remaining fuel rods of the struck assemblies are held rigidly in place, they are susceptible only to the compression mode of failure. To cause cladding

failure of one fuel rod due to compression, 250 ft-lbs of energy is required. To cause failure of all the remaining rods of the four struck assemblies, 250 x 41 x 4 or 41,000 ft-lbs for the 7 x 7 fuel or 250 x 54 x 4 or 54,000 ft-lbs for the 8 x 8 fuel of energy would have to be absorbed in cladding alone. Thus, it is clear that not all the remaining fuel rods of the struck assemblies can fail on the first impact. The number of fuel rod failures due to compression is computed as follows:

7 x 7 fuel
$$\frac{0.5 \times 13{,}600 \times \left(\frac{11}{11 + 17}\right)}{250} = 11$$

8 x 8 fuel
$$\frac{0.5 \times 14{,}500 \times \left(\frac{11}{11 + 17}\right)}{250} = 12$$

Thus, during the first impact, the fuel rod failures are as follows:

		7 x 7	8 x 8	
Dropped assembly	-	49	62	rods (bending)
Struck assemblies	-	32	32	tie rods (bending)
Struck assemblies	-	11	12	rods (compression)
		92	106	failed rods

Because of the less severe nature of the second impact and the distorted shape of the dropped fuel assembly, it is assumed that in only 2 of the 24 struck assemblies are the tie rods subjected to bending failure. Thus, 2 x 8 = 16 tie rods are assumed to fail. The number of fuel rod failures due to compression on the second impact is computed as follows:

7 x 7
$$\frac{\frac{0.19}{2} \times 17{,}000 \times \frac{11}{11 + 17}}{250} = 3$$

8 x 8
$$\frac{\frac{0.19}{2} \times 18{,}150 \times \frac{11}{11 + 17}}{250} = 3$$

Thus, during the second impact the fuel rod failures are as follows:

Struck assemblies	-	16	tie rods (bending)
Struck assemblies	-	3	rods (compression)
		19	failed rods

The total number of failed rods resulting from the accident is as follows:

	7 x 7	8 x 8	
First impact	92	106	rods
Second impact	19	19	rods
Third impact	0	0	rods
	111	125	failed rods (total)

14.6.4.3 Fission Product Release From Fuel

The radiological dose consequences resulting from a refueling accident have been evaluated using Alternative Source Terms (AST) in accordance with 10 CFR 50.67 and the guidelines of Regulatory Guide 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors.”

Fission product release estimates for the accident are based on the following assumptions:

a. The reactor has been operating at design power (3952 MWt) for 24 month fuel cycle. The average fuel burnup is 35 to 39 GWd/MT prior to the accident. The 24-hour decay time allows time for the reactor to be shut down, the nuclear system depressurized, the reactor vessel head removed, and the reactor vessel upper internals removed. It is not expected that these evolutions could be accomplished in less than 24 hours.

b. The activity in the fuel bundle is determined using the ORIGEN code at a core power of 4031 MWt modified with a power peaking factor of 1.5 and Regulatory Guide 1.183 power factor of 1.02 with a decay of 24 hours.

c. One hundred eleven fuel rods are assumed to fail. This was the conclusion of the analysis of mechanical damage to the fuel based on the GE 7x7 fuel design.

14.6.4.4 Fission Product Release to Secondary Containment

The following assumptions were used to calculate the fission product release to the secondary containment (per Regulatory Guide 1.183):

a. Fraction of Fuel Rod Inventory Released (infinite decontamination for nuclides other than iodine and noble gases):

Noble Gases (Except Kr 85)	5 percent
Kr 85	10 percent
Iodines (Except I-131)	5 percent
I-131	8 percent

b. Iodine Decontamination Factor in Reactor Cavity Pool Water — 200 elemental and organic

c. Iodine Species — 99.85% elemental, 0.15% organic

14.6.4.5 Fission Product Release to Environs

The following assumptions and initial conditions are used in calculating the dose existing at the exclusion area boundary, at the low population zone, and the control room operators due to fission product release.

a. The release is assumed to be an instantaneous ground level release to the environment with no holdup time in secondary containment. Accordingly, no credit is taken for filtering by the standby gas treatment system and no credit is taken for an elevated release at the main stack.

b. No credit is taken for isolation of the control room nor for any filtering by the control room emergency ventilation system.

c. The X/Q for the control room is reduced by 50% to reflect the credit for dual control room air intakes as allowed by Standard Review Plan Section 6.4.

d. Control Room Free Volume - 210,000 ft^3

The design basis fuel handling accident assumes that during the refueling period a fuel bundle is dropped into the reactor cavity pool. The dropped fuel bundle strikes additional bundles in the reactor core fracturing 111 fuel pins (assuming GE 7x7 fuel design). The inventory described above will be released from the fractured fuel rods. A decontamination factor of 200 for elemental and organic is applicable for iodine released at depth under water. The radioactive releases to the air space above the pool are released instantaneously to the atmosphere with no holdup in secondary containment and no filtering by the Standby Gas Treatment System. The assumptions used to evaluate the fuel handling design basis accident event are

defined in Nuclear Regulatory Commission Regulatory Guide 1.183. Further guidance is contained in the Standard Review Plans in NUREG-800, Section 15.0.1. The total activity released is greater for a fuel handling accident in the reactor cavity pool than for an accident in the fuel storage pool. Normally, the number of fuel rods fractured in a drop into the reactor vessel pool is slightly larger than the number of rods fractured in a drop into the storage pool. This provides a bigger source for the vessel event.

The fuel handling accident was evaluated using RADTRAD computer programs as described in Section 14.6.3.7. The X/Q values based on the refueling vents from 0-2 hours were used in computing the dose consequences of this release.

14.6.4.6 Radiological Effects

The radiological exposures following the refueling accident have been evaluated in the control room, at the site boundary, and at the LPZ boundary. The calculated dose assumes that all of the activity is exhausted instantaneously through a roof vent; with no credit for holdup time nor filtering by SGTS.

Boundary dose resulting from design basis accident events has been judged by comparing the dose to the 10 CFR 50.67, “Accident Source Term,” limits. This regulation uses radiation doses of 25 Rem TEDE for doses to the public and 5 Rem TEDE for the control room as guides under accident conditions. In the Standard Review Plan, NUREG-800, the limits for doses to the public are reduced by 25 percent to 6.3 Rem TEDE. The calculated doses are much less than the guidelines (< 6.3 Rem TEDE for EAB and LPZ and < 5 Rem TEDE for the control room).

14.6.5 Main Steam Line Break Accident

Accidents that result in the release of radioactive materials outside the secondary containment are the results of postulated breaches in the nuclear system process barrier. The design basis accident is a complete severance of one main steam line outside the secondary containment. Figure 14.6-7 shows the break location. The analysis of the accident is described in three parts as follows:

a. Nuclear System Transient Effects

This includes analysis of the changes in nuclear system parameters pertinent to fuel performance and the determination of fuel damage.

b. Radioactive Material Release

This includes determination of the quantity and type of radioactive material released through the pipe break and to the environs.

c. Radiological Effects

This portion determines the dose effects of the accident to control room and offsite persons.

14.6.5.1 Nuclear System Transient Effects

14.6.5.1.1 Assumptions

The following assumptions are used in evaluating response of nuclear system parameters to the steam line break accident outside the secondary containment:

a. The reactor is operating at the power associated with maximum mass release.

b. Reactor vessel water level is normal for initial power level assumed at the time the break occurs.

c. Nuclear system pressure is normal for the initial power level.

d. The steam pipeline is assumed to be instantly severed by a circumferential break. The break is physically arranged so that the coolant discharge through the break is unobstructed. These assumptions result in the most severe depressurization rate of the nuclear system.

e. For the purpose of fuel performance, the main steam isolation valves are assumed to be closed 5.5 seconds after the break. This assumption is based on the 0.5 second time required for the development of the automatic isolation signal (high differential pressure across the main steam line flow restrictor) and the 5 second closure time for the valves.

For the purpose of radiological dose calculations, the main steam isolation valves are assumed to be closed at 5.5 seconds after the break. Faster main steam isolation valve closure could reduce the mass loss until finally some other process line break would become controlling. However, the resulting radiological dose for this break would be less than the main steam line break with a five second valve closure. Thus, the postulated main steam line break outside the primary containment with a five second isolation valve closure results in maximum calculated radiological dose and is, therefore, the design basis accident.

f. The mass flow rate through the upstream side of the break is assumed to be not affected by isolation valve closure until the isolation valves are closed far enough to establish limiting critical flow at the valve location. After limiting critical flow is established at the isolation valve, the mass flow is assumed to decrease linearly as the valve is closed.

g. The mass flow rate through the downstream side of the break is assumed to be not affected by the closure of the isolation valves in the unbroken steam lines until those valves are far enough closed to establish limiting critical flow at the valves. After limiting critical flow is established at the isolation valve positions, the mass flow is assumed to decrease linearly as the valves close.

h. Feedwater flow is assumed to decrease linearly to zero over the first five seconds to account for the slowing down of the turbine-driven feed pumps in response to the rise in reactor vessel water level.

i. A loss of auxiliary AC power is assumed to occur simultaneous with the break. This results in the immediate loss of power to the recirculation pumps. Recirculation flow is assumed to coast down with a three second time constant.

14.6.5.1.2 Sequence of Events

The sequence of events following the postulated main steam line break is as follows:

The steam flow through both ends of the break increases to the value limited by critical flow considerations. The flow from the upstream side of the break is limited initially by the main steam line flow restrictor. The flow from the downstream side of the break is limited initially by the downstream break area. The decrease in steam pressure at the turbine inlet initiates closure of the main steam isolation valves within about 200 milliseconds after the break occurs (see Subsection 7.3 "Primary Containment Isolation System"). Also, main steam isolation valve closure signals are generated as the differential pressures across the main steam line flow restrictors increase above isolation setpoints. The instruments sensing flow restrictor differential pressures generate isolation signals within about 500 milliseconds after the break occurs.

A reactor scram is initiated as the main steam isolation valves begin to close (see Subsection 7.2, "Reactor Protection System"). In addition to the scram initiated from main steam isolation valve closure, voids generated in the moderator during depressurization contribute significant negative reactivity to the core even before the scram is complete. Because the main steam line flow restrictors are sized for the main steam line break accident, reactor vessel water level remains above the top of the fuel throughout the transient.

14.6.5.1.3 Coolant Loss and Reactor Vessel Water Level

The mass release during a main steamline break outside containment was analyzed at full power and hot standby conditions. At full power, the initial steam flow rate through the break is approximately 7300 lb/sec, while the steam generation rate is

almost 4000 lb/sec. The break flow-steam generation mismatch causes a depressurization of the reactor vessel. The formation of bubbles in the reactor vessel water causes a rapid rise in the water level. The analytical model used to calculate level rise predicts a rate of rise of about 6 feet/second. Thus, the water level reaches the vessel steam nozzles at 4 to 5 seconds after the break.

At hot standby, the initial break flow is almost 6600 lb/sec as shown in Figure 14.6-8; but the steam generation rate is about 27 lb/sec. The rise in reactor water level is much faster and reaches the vessel steam nozzles in about one second after the break. From that time on, a two-phase mixture is discharged from the break. The two-phase flow rates are determined by vessel pressure and mixture enthalpy.[16] Due to the longer duration of two-phase break flow, the hot standby conditions result in much more liquid flowing through the break than at full power such that the total mass release is about 70% greater at hot standby than at full power.

As shown in Figure 14.6-8, two-phase flow is discharged through the break at an almost constant rate until late in the transient. This is the result of not taking credit for the effect of valve closure on flow rate until isolation valves are far enough closed to establish critical flow at the valve locations. The slight decrease in discharge flow rate is caused by depressurization inside the reactor vessel. The linear decrease in discharge flow rate at the end of the transient is the result of the assumption regarding the effect of valve closure on flow rate after critical flow is established at the valve location.

The following total masses of steam and liquid are discharged through the break prior to a 5.5 second isolation valve closure:

Steam	11,975 pounds
Liquid	42,215 pounds

The evaluation of fuel performance used a bounding time of 10.5 seconds for closure of the main steam isolation valves. Analysis of fuel conditions reveals that no fuel rod perforations due to high temperature occur during the depressurization, even with the conservative assumptions regarding the operation of the recirculation and feedwater systems. MCHFR remains above 1.0 at all times during the transient. MCHFR has been replaced by a similar fuel thermal parameter called MCPR (Minimum Critical Power Ratio). No fuel rod failures due to mechanical loading during the depressurization occur because the differential pressures resulting from the transient do not exceed the designed mechanical strength of the core assembly.

[16] Moody, F. J.,: "Two Phase Vessel Blowdown from Pipes," Journal of Heat Transfer, ASME Vol. 88, August 1966, page 285.

After the main steam isolation valves close, depressurization stops and natural convection is established through the reactor core. Even if the event is initiated from full power (which has a much lower mass release) with a delayed main isolation valve closure, no fuel cladding perforation occurs even if the stored thermal energy in the fuel were simply redistributed while natural convection is being established; cladding temperature would be about 1000°F, well below the temperatures at which cladding can fail. Thus, it is concluded that even for a 10.5 second main steam isolation valve closure, fuel rod perforations due to high temperature do not occur. For shorter valve closure times, the accident is less severe. After the main steam isolation valves are closed, the reactor can be cooled by operation of any of the normal or standby cooling systems. The MCHFR never drops below 1.0, the core is always cooled by very effective nucleate boiling. Transient limits for nonstandard test or demonstration fuel bundles are given in Appendix N.

14.6.5.2 Radioactive Material Release

14.6.5.2.1 Assumptions

The following assumptions are used in the calculation of the quantity and types of radioactive material released from the nuclear system process barrier outside the secondary containment:

a. The amounts of steam and liquid discharged are as calculated from the analysis of the nuclear system transient.

b. The concentrations of biologically significant radionuclides contained in the coolant discharged as liquid (which subsequently flashes to steam) and the coolant discharged as steam are based on the ANSI/ANS-18.1-1984, "Radioactive Source Term for Normal Operation of Light Water Reactors" methodology. The halogens considered are I-131, I-132, I-133, I-134, and I-135. The values obtained by the ANSI/ANS-18.1 evaluation are then scaled to represent a dose equivalent I-131 concentration of 32 μCi/gm which is greater than the 26 μCi/gm maximum Technical Specification limit and 10 times the equilibrium value for continued full power operation allowed by Technical Specifications.

c. The concentration of noble gases leaving the reactor vessel at the time of the accident are based on the ANSI/ANS-18.1 concentrations with an appropriate scaling based on NEDO-10871, "Technical Derivation of BWR 1971 Design Basis Radioactive Material Source Terms".

d. It is assumed that the main steam isolation valves are fully closed at 5.5 seconds after the pipe break occurs. This allows 500 milliseconds for the generation of the automatic isolation signal and 5 seconds for the valves to close. The valves and valve control circuitry are designed to provide main steam line isolation in no more than 5.5 seconds. The actual closure time setting for the isolation valves is less than 5 seconds.

e. Due to the short half-life of nitrogen-16 the radiological effects from this isotope are of no major concern and are not considered in the analysis.

f. Atmospheric dispersion coefficients, X/Q, for ground level releases from the turbine building exhaust are used. X/Q values applicable to the time periods, distances and geometric relationships (offsite and control room) are shown in Table 14.6-8. Control room X/Q values are calculated using the computer code ARCON96.

g. All of the activity released from the reactor vessel to the Turbine Building is conservatively assumed to escape to the environment.

14.6.5.2.2 Fission Product Release From Break

Using the above assumptions, the following amounts of radioactive materials are released from the nuclear system process barrier:

Noble gases	1.342×10^3 Ci
Iodine 131	5.254×10^1 Ci
Iodine 132	4.737×10^2 Ci
Iodine 133	3.533×10^2 Ci
Iodine 134	8.549×10^2 Ci
Iodine 135	5.031×10^2 Ci

The above releases take into account the total amount of liquid released as well as the liquid converted to steam during the accident.

14.6.5.3 Radiological Effects

The control room dose is divided by 2 because of the dilution effect of the dual air intake configuration of the control bay ventilation. Shine due to radioisotopes in the Turbine Building is also accounted for in the total control room operator dose. The shine is not divided by 2. The control room operator doses due to a MSLB are less than the 10 CFR 50.67 limit of 5 Rem TEDE. The offsite doses are less than the 10 CFR 50.67 limit of 25 Rem TEDE for the maximum Technical Specification reactor coolant (32 μCi/gm I-131 equivalent). Also, the offsite doses are less than 10% of the 10 CFR 50.67 limits for the maximum equilibrium reactor coolant (3.2 μCi /gm). It is concluded that no danger to the health and safety of the public results as a consequence of this accident.

Table 14.6-1
Sheet 1 of 2

Iodine-131 Activity (Ci) by Location as Function of Time for CRDA

Time - hrs	Main Cond	Stack Rm	Control Rm	Environment
0	2.99E+04	0.00E+00	0.00E+00	0.00E+00
0.4	2.35E+04	3.39E+00	2.32E-02	6.27E+02
0.5	2.22E+04	4.11E+00	2.57E-02	7.60E+02
0.8	1.8TE+04	6.03E+00	1.45E-02	1.12E+03
1.1	1.55E+04	7.62E+00	8.20E-03	1.42E+03
1.4	1.29E+04	8.93E+00	4.65E-03	1.67E+03
1.7	1.08E+04	1.00E+01	2.65E-03	1.88E+03
2	9.05E+03	1.09E+01	1.52E-03	2.06E+03
2.3	7.56E+03	1.16E+01	8.60E-04	2.20E+03
2.6	6.32E+03	1.22E+01	4.89E-04	2.33E+03
2.9	5.29E+03	1.27E+01	2.80E-04	2.43E+03
3.2	4.42E+03	1.31E+01	1.62E-04	2.51E+03
3.5	3.69E+03	1.34E+01	9.49E-05	2.59E+03
3.8	3.09E+03	1.36E+01	5.65E-05	2.65E+03
4.1	2.58E+03	1.38E+01	3.44E-05	2.70E+03
4.4	2.16E+03	1.40E+01	2.16E-05	2.74E+03
4.7	1.80E+03	1.41E+01	1.40E-05	2.77E+03
5	1.51E+03	1.41E+01	9.44E-06	2.80E+03
5.3	1.26E+03	1.42E+01	6.62E-06	2.83E+03
5.6	1.05E+03	1.42E+01	4.82E-06	2.85E+03
5.9	8.81E+02	1.42E+01	3.62E-06	2.86E+03
6.2	7.37E+02	1.42E+01	2.80E-06	2.88E+03
6.5	6.16E+02	1.42E+01	2.22E-06	2.89E+03
6.8	5.15E+02	1.41E+01	1.78E-06	2.90E+03
7.1	4.30E+02	1.41E+01	1.45E-06	2.91E+03
7.4	3.60E+02	1.40E+01	1.19E-06	2.92E+03
7.7	3.01E+02	1.40E+01	9.81E-07	2.92E+03
8	2.51E+02	1.39E+01	8.13E-07	2.93E+03
8.3	2.10E+02	1.39E+01	5.80E-07	2.93E+03

Table 14.6-1
Sheet 2 of 2

Iodine-131 Activity (Ci) by Location as Function of Time for CRDA

Time - hrs	Main Cond	Stack Rm	Control Rm	Environment
8.6	1.76E+02	1.38E+01	4.29E-07	2.93E+03
8.9	1.47E+02	1.37E+01	3.28E-07	2.94E+03
9.2	1.23E+02	1.36E+01	2.56E-07	2.94E+03
9.5	1.03E+02	1.36E+01	2.04E-07	2.94E+03
9.8	8.58E+01	1.35E+01	1.65E-07	2.94E+03
10.1	7.17E+01	1.34E+01	1.35E-07	2.94E+03
10.4	6.00E+01	1.33E+01	1.11E-07	2.94E+03
24	1.78E-02	1.01E+01	0	2.95E+03
96	0	2.22E+00	0	2.95E+03
720	0	0	0	2.95E+03

TABLE 14.6-2

(Deleted by Amendment 19)

TABLE 14.6-3

(Deleted)

Table 14.6-4

(Deleted)

Table 14.6-5

(Deleted)

Table 14.6-6

(Deleted)

Table 14.6-7

BOUNDING CORE INVENTORY

Isotope	Ci/MWt t=o	Ci/MWt t=24 hr	Isotope	Ci/MWt t=o	Ci/MWt t=24 hr
CO58	1.430E+02	1.416E+02	XE131M	3.544E+02	3.487E+-02
CO60	1.425E+02	1.424E+02	TE132	3.829E+04	3.089E+04
KR83M	3.432E+03	1.387E+01	I132	3.885E+04	3.184E+04
KR85	3.601E+02	3.601E+02	I133	5.534E+04	2.559E+04
KR85M	7.329E+03	1.811E+02	XE133	5.504E+04	5.303E+04
RB86	6.372E+01	6.141E+01	XE133M	1.734E+03	1.562E+03
KR87	1.446E+04	3.051E-02	I134	6.141E+04	1.450E-03
KR88	2.009E+04	5.743E+01	CS134	5.703E+03	5.697E+03
KR89	2.521E+04	0.000E+00	I135	5.250E+04	4.189E+03
SR89	2.786E+04	2.748E+04	XE135	1.971E+04	1.429E+04
SR90	3.165E+03	3.165E+03	XE135M	1.135E+04	6.823E+02
Y90	3.283E+03	3.273E+03	CS136	1.941E+03	1.841E+03
SR91	3.487E+04	6.103E+03	XE137	5.023E+04	0.000E+00
Y91	3.583E+04	3.564E+04	CS137	4.037E+03	4.037E+03
SR92	3.677E+04	7.922E+01	BA137M	3.829E+03	3.810E+03
Y92	3.696E+04	1.168E+03	XE138	4.757E+04	1.172E-26
Y93	4.147E+04	8.084E+03	BA139	4.930E+04	4.170E-01
ZR95	4.880E+04	4.822E+04	BA140	4.909E+04	4.644E+04
NB95	4.897E+04	4.897E+04	LA140	5.231E+04	5.079E+04
ZR97	4.953E+04	1.851E+04	LA141	4.498E+04	7.085E+02
MO99	5.088E+04	3.956E+04	CE141	4.535E+04	4.463E+04
TC99M	4.454E+04	3.772E+04	LA142	4.397E+04	1.035E+00
RU103	4.094E+04	4.018E+04	CE143	4.245E+04	2.597E+04
RU105	2.710E+04	6.615E+02	PR143	4.113E+04	4.075E+04
RH105	2.559E+04	1.840E+04	CE144	3.810E+04	3.810E+04
RU106	1.488E+04	1.486E+04	ND147	1.806E+04	1.698E+04
SB127	2.796E+03	2.369E+03	NP239	5.201E+05	3.902E+05
TE127	2.773E+03	2.580E+03	PU238	2.805E+02	2.805E+02
TE127M	3.721E+02	3.719E+02	PU239	1.234E+01	1.238E+01
SB129	8.457E+03	1.952E+02	PU240	1.730E+01	1.730E+01
TE129	8.326E+03	1.236E+03	PU241	4.450E+03	4.448E+03
TE129M	1.615E+03	1.590E+03	AM241	5.449E+00	5.470E+00
TE131M	5.155E+03	2.976E+03	CM242	1.234E+03	1.234E+03
I131	2.669E+04	2.481E+04	CM244	5.697E+01	5.697E+01

Table 14.6-8

(Sheet 1)

VALUES FOR X/Q FOR ACCIDENT DOSE CALCULATIONS

Time Period	Control Room (sec/m^3)		Site Boundary (sec/m^3)	LPZ Boundary (sec/m^3)
Top of Stack Releases (LOCA & CRDA)	U1 Intake	Unit 3 Intake		
0-0.5 hrs*	3.40E-5	3.02E-5	2.35E-5	1.26E-5
0.5-2 hrs	9.08E-13	1.41E-7	1.19E-6	1.13E-6
2-8 hrs	3.41E-13	4.50E-8		5.75E-7
8-24 hrs	2.09E-13	2.54E-8		4.10E-7
1-4 days	7.21E-14	7.36E-9		1.97E-7
4-30 days	1.57E-14	1.24E-9		6.88E-8
Base of Stack Releases (LOCA & CRDA)				
0-2 hrs	2.00E-4	8.60E-5	2.62E-4	1.31E-4
2-8 hrs	1.28E-4	6.46E-5		6.61E-5
8-24 hrs	5.72E-5	2.80E-5		4.69E-5
1-4 days	4.05E-5	2.00E-5		2.23E-5
4-30 days	3.09E-5	1.53E-5		7.96E-6

Table 14.6-8

(Sheet 2)

VALUES FOR X/Q FOR ACCIDENT DOSE CALCULATIONS

Time Period	Control Room (sec/m^3)		Site Boundary (sec/m^3)	LPZ Boundary (sec/m^3)
Refueling Vent Releases (FHA and LOCA) ****	U1 Intake	Unit 3 Intake		
0-2 hrs	4.60E-4	**	2.62E-4	1.31E-4
2-8 hrs	Not Used	**		Not Used
8-24 hrs	1.57E-4	**		4.69E-5
1-4 days	1.12E-4	**		2.23E-5
4-30 days	7.90E-5	**		7.96E-6

**** Control room 0-2 hr for FHA only; values 8 hr+ for control room and LPZ Boundary used as surrogate for HCVS release

Turbine Building Exhaust Release (MSLB - EAB/LPZ; Post-LOCA MSIV Leakage - Unit 1 Only)				
0-2 hrs	3.22E-4	**	2.62E-4	1.31E-4
2-8 hrs	2.77E-4	**		6.61E-5
8-24 hrs	1.31E-4	**		4.69E-5
1-4 days	7.91E-5	**		2.23E-5
4-30 days	6.10E-5	**		7.96E-6

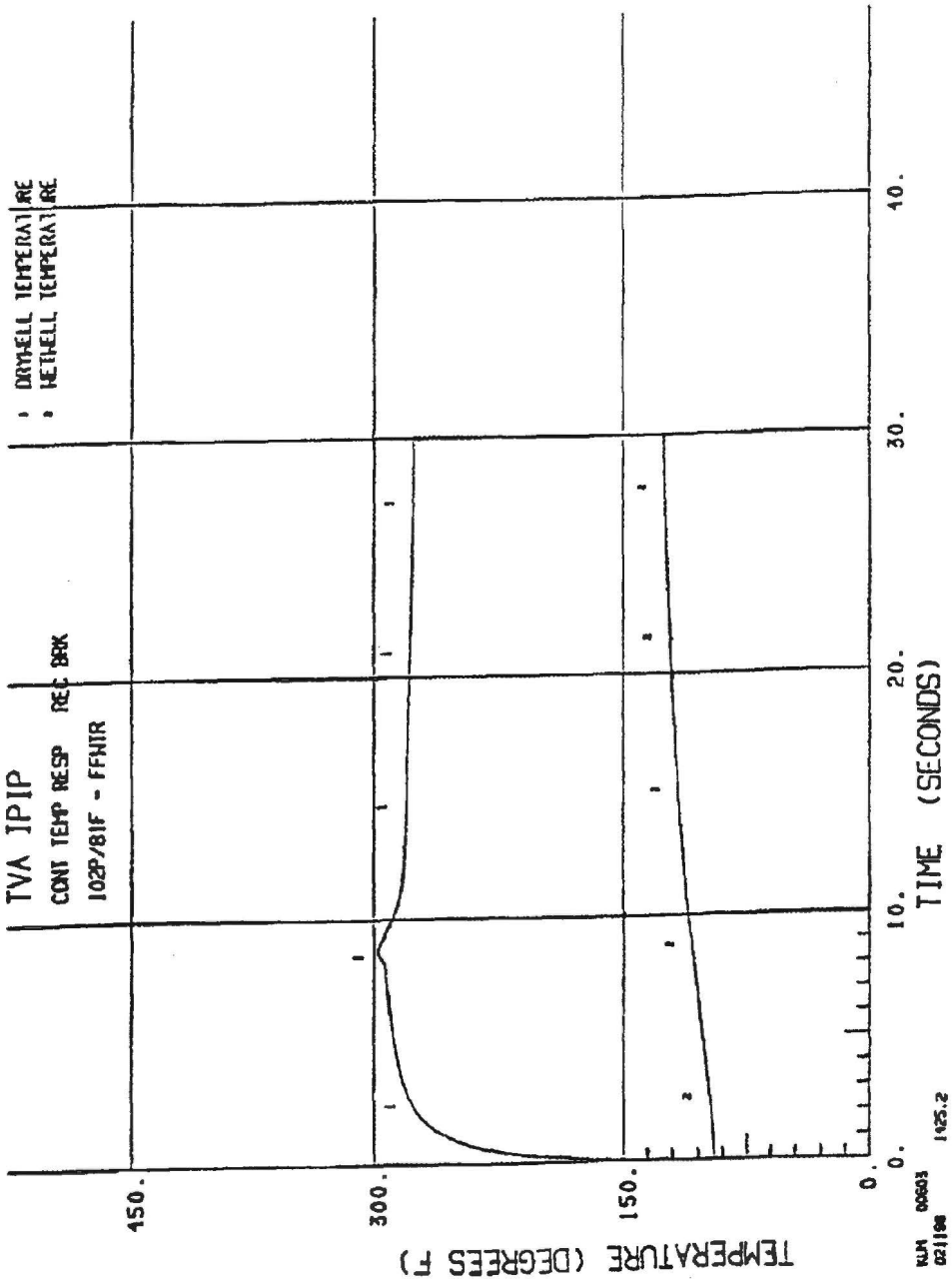
**Bounded by the Unit 1 Intake

Turbine Building Roof Ventilators Release (Post LOCA MSIV Leakage Units 2/3 Only)				
0-2 hrs	***	2.17E-4	2.62E-4	1.31E-4
2-8 hrs	***	1.64E-4		6.61E-5
8-24 hrs	***	7.89E-5		4.69E-5
1-4 days	***	4.33E-5		2.23E-5
4-30 days	***	3.35E-5		7.96E-6

***Bounded by the Unit 3 Intake

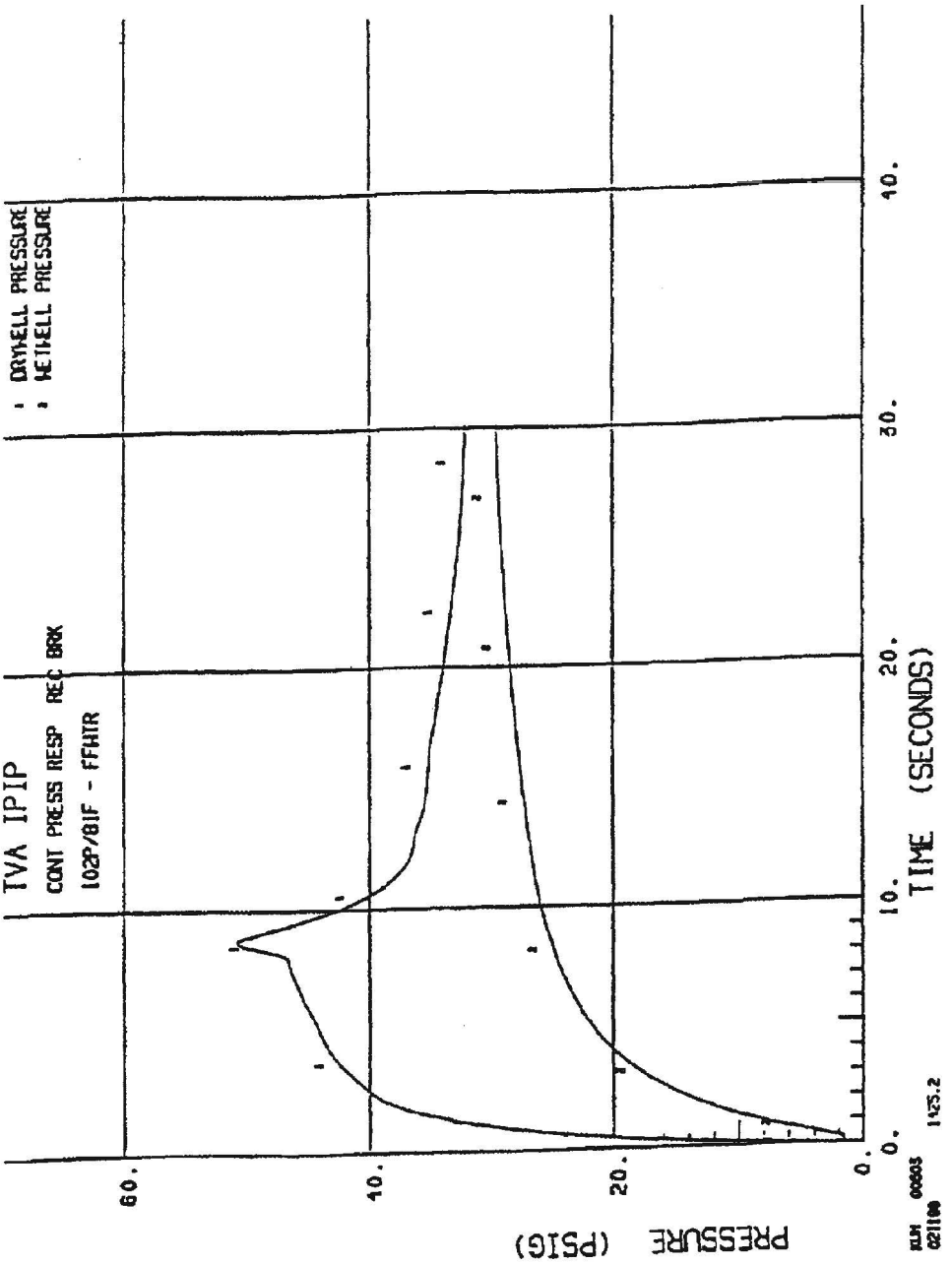
Table 14.6-9

(Deleted by Amendment 19)



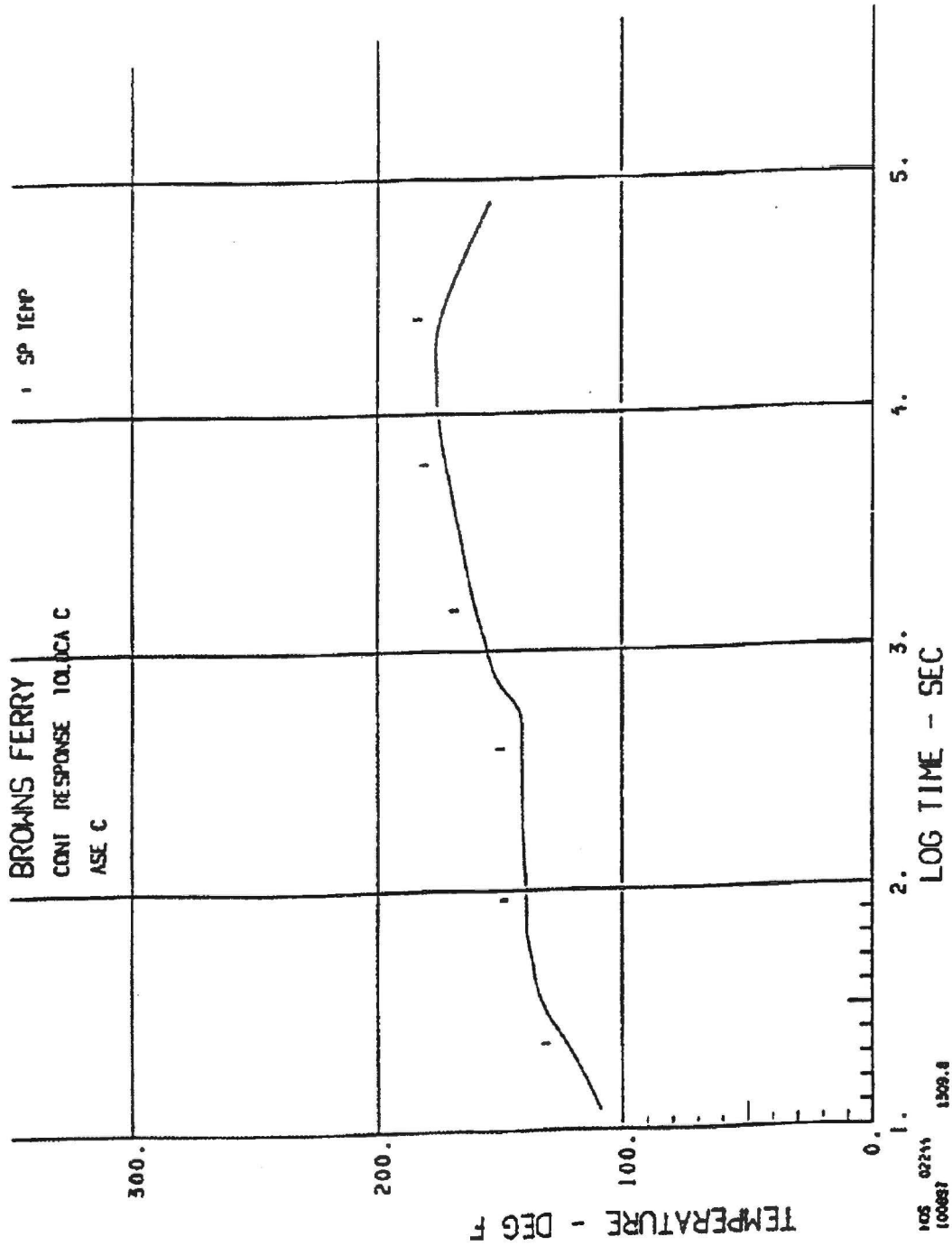
AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT
DBA-LOCA SHORT-TERM CONTAINMENT TEMPERATURE RESPONSE (102% OF UPRATED POWER, 81% CF) FIGURE 14.6-1



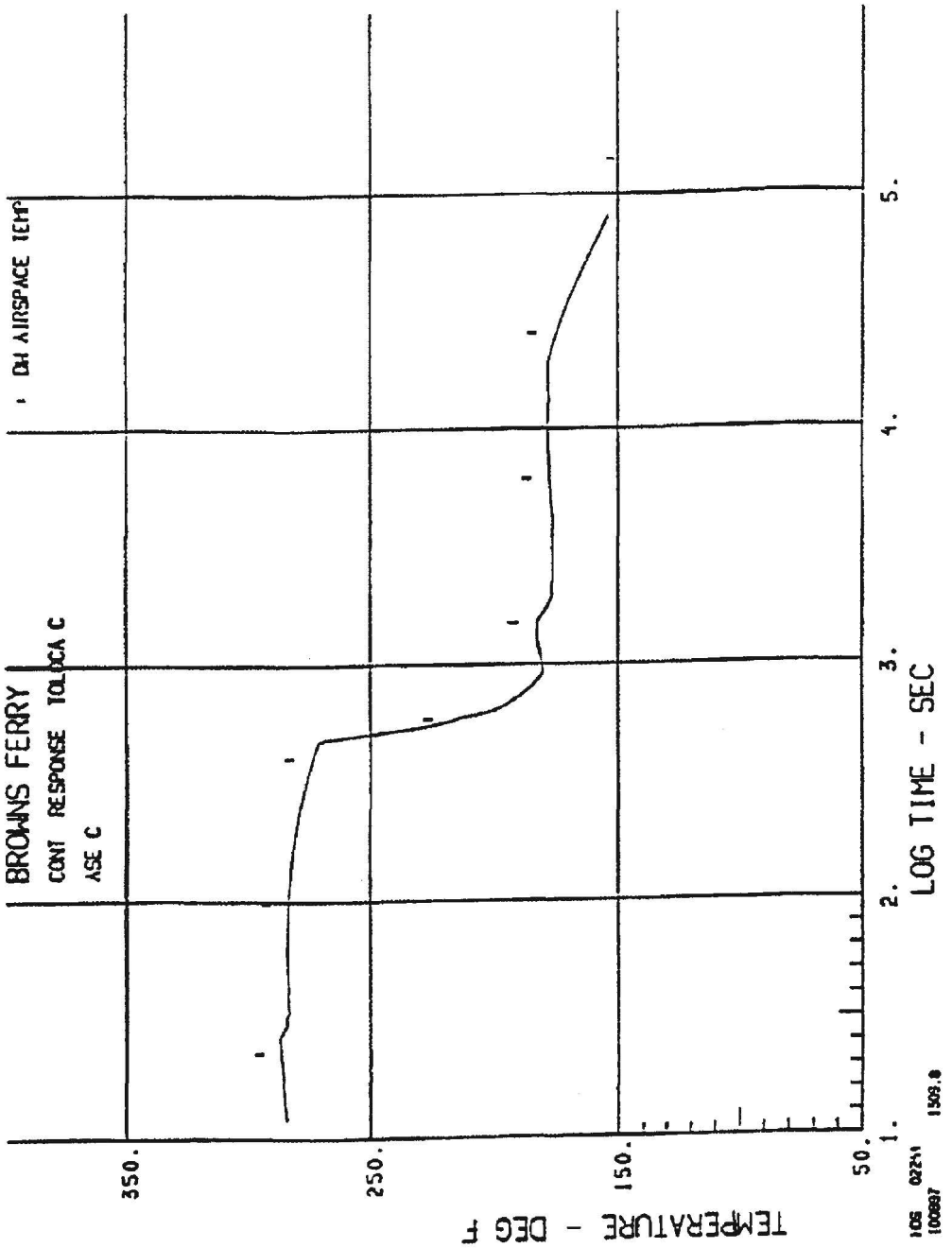
AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT
DBA-LOCA SHORT-TERM CONTAINMENT PRESSURE RESPONSE (102% OF UPRATED POWER, 81% CF) FIGURE 14.6-2



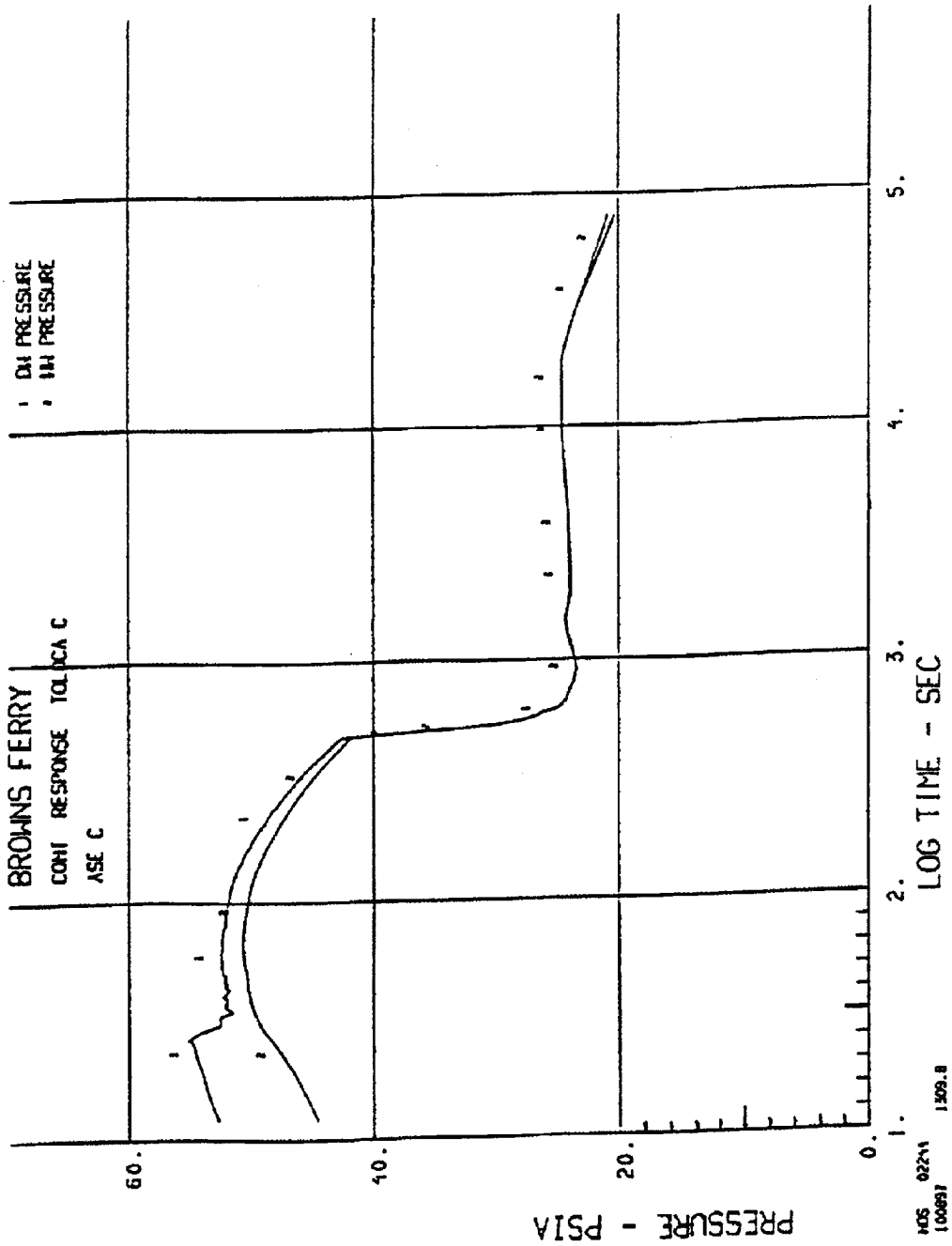
AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT
DBA-LOCA LONG-TERM WETWELL TEMPERATURE RESPONSE (102% OF UPRATE POWER, 100% CF) FIGURE 14.6-3



AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT
DBA-LOCA LONG-TERM DRYWELL TEMPERATURE RESPONSE (102% OF UPRATED POWER, 100% CF) FIGURE 14.6-4

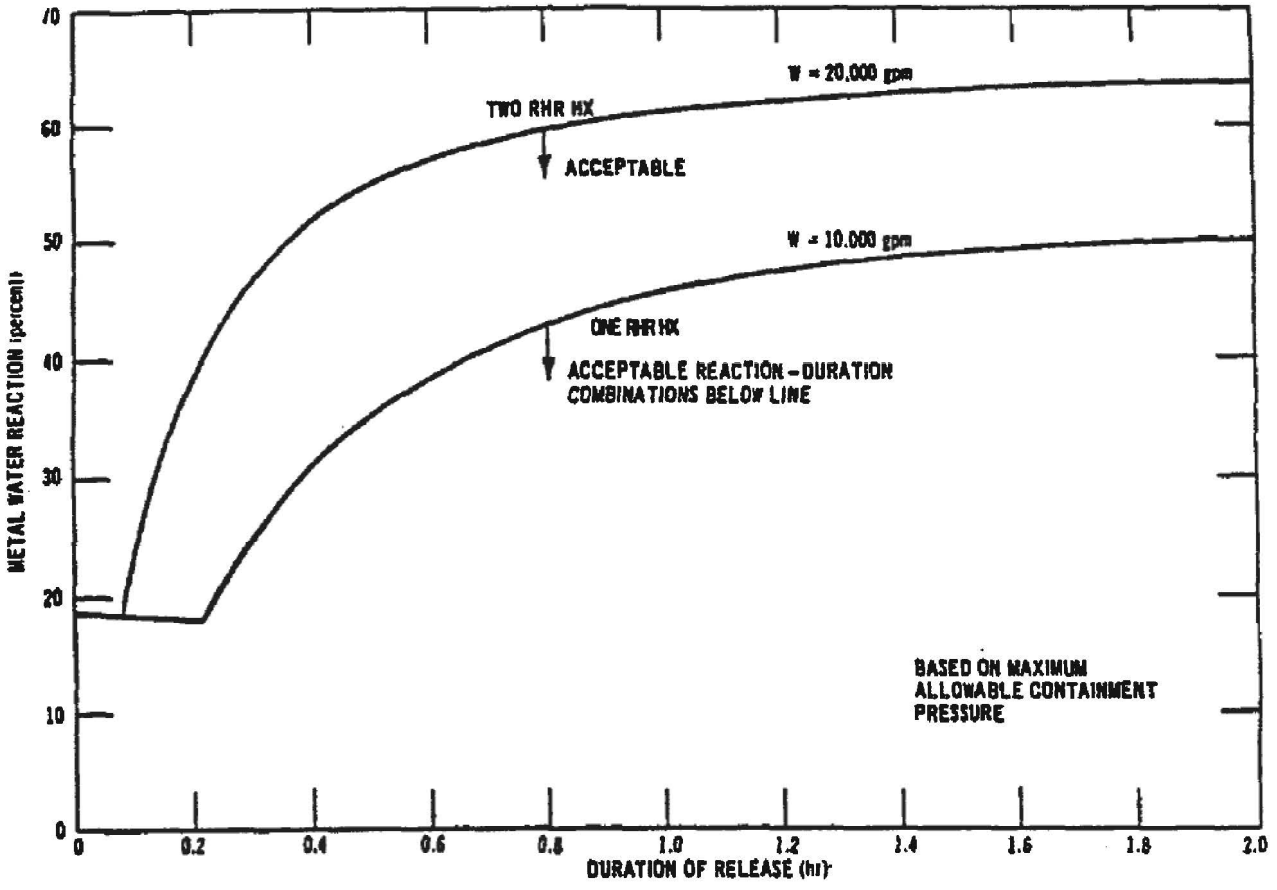


AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

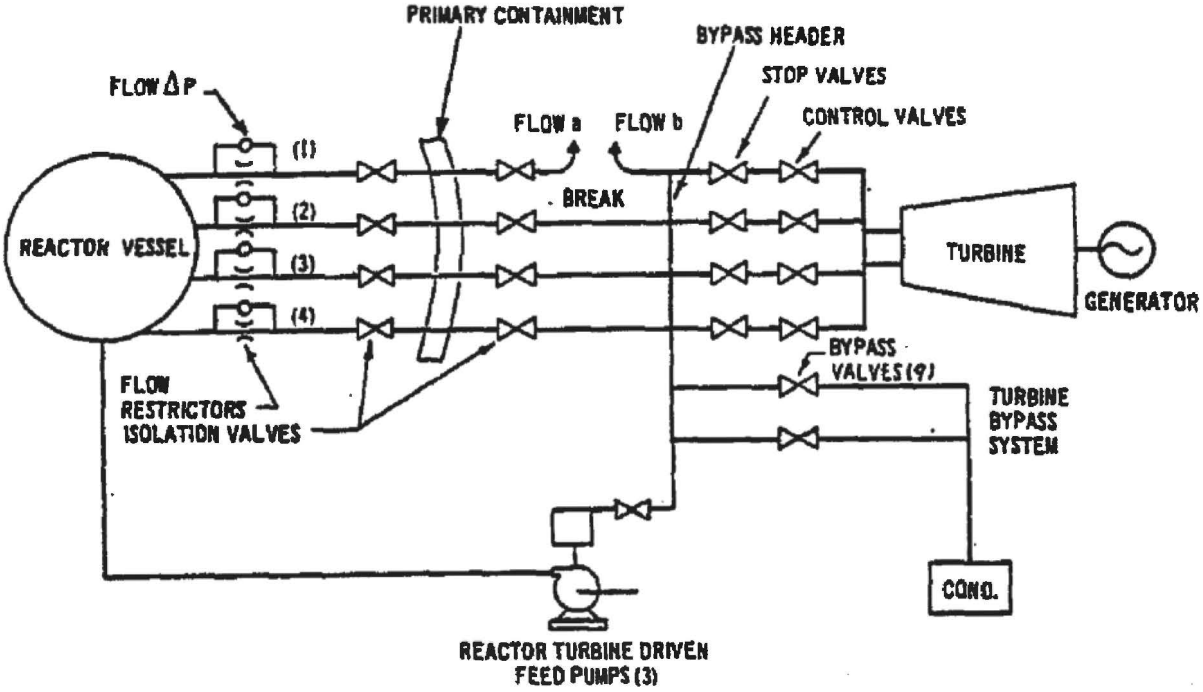
DBA-LOCA LONG-TERM PRESSURE RESPONSE
(102% OF UPRATED POWER, 100% CF)

FIGURE 14.6-5



AMENDMENT 17

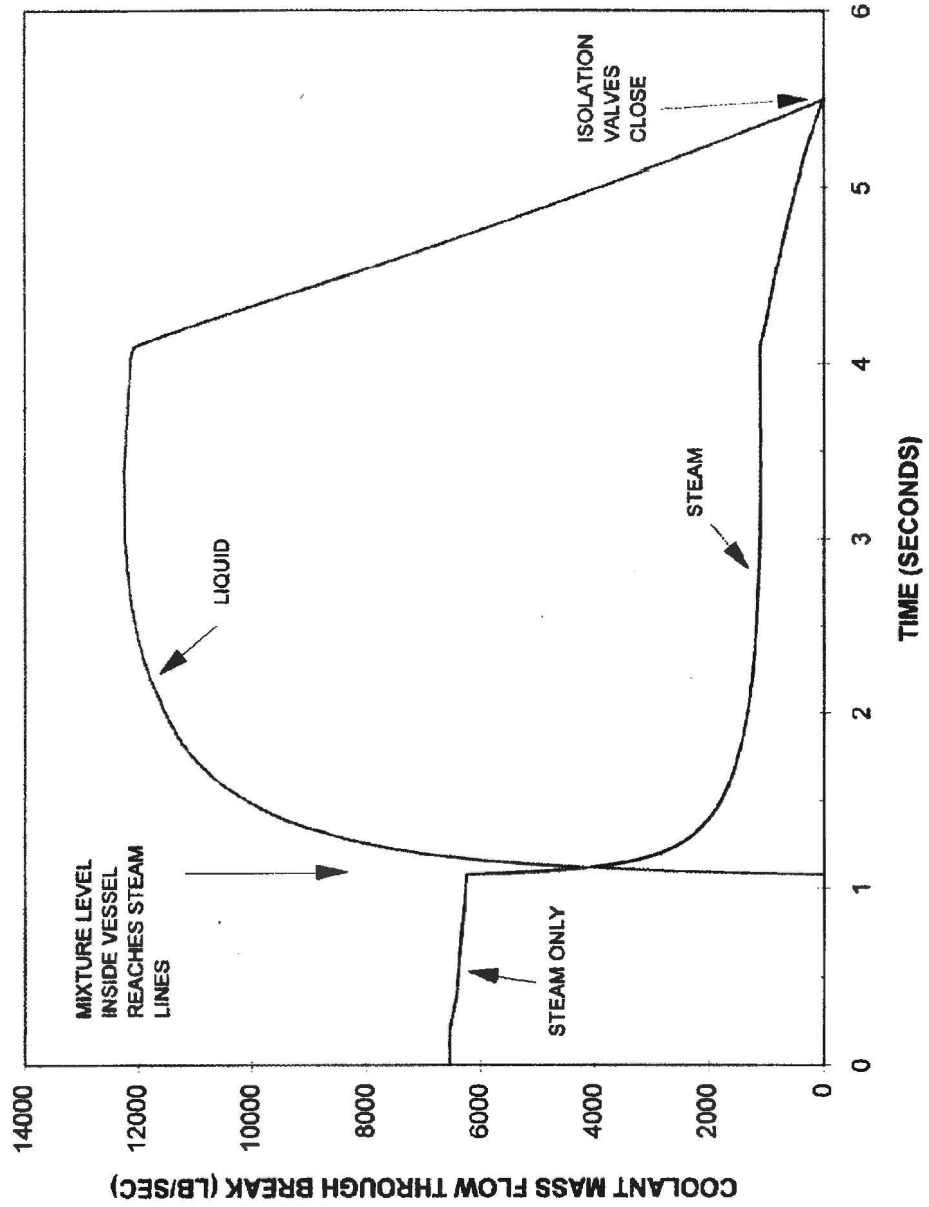
BROWNS FERRY NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT
Loss-of-Coolant Accident, Primary Containment Capability for Metal-Water Reaction FIGURE 14.6-6



AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT
Main Steamline Break Accident Break Location FIGURE 14.6-15

FIGURE 14.6-7

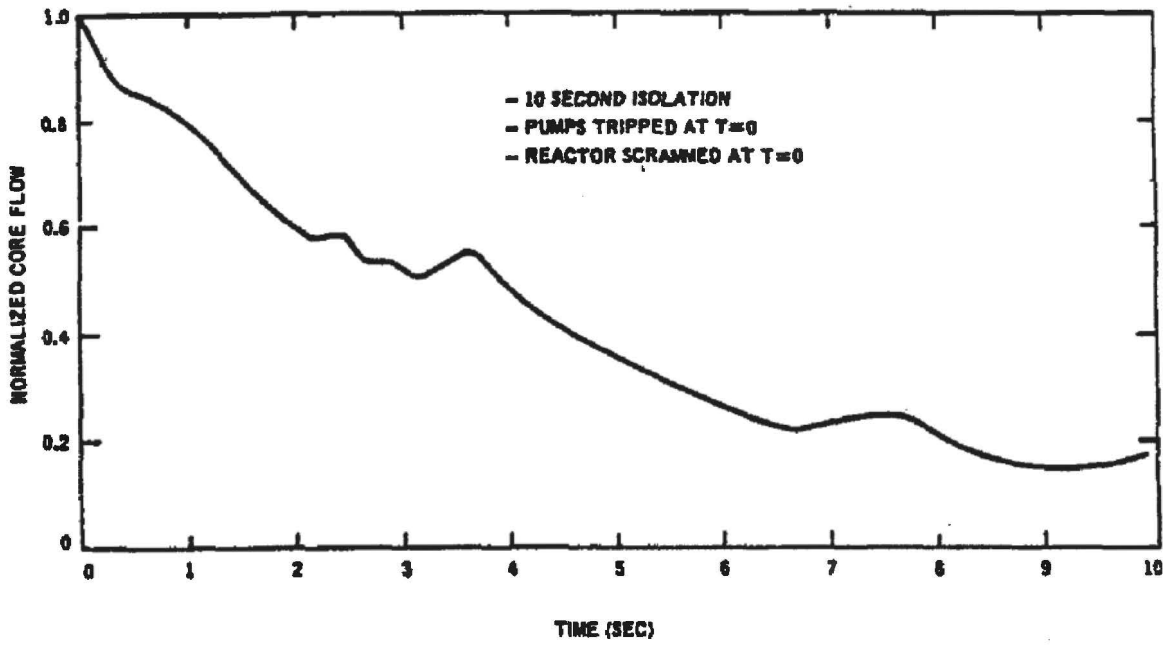


AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

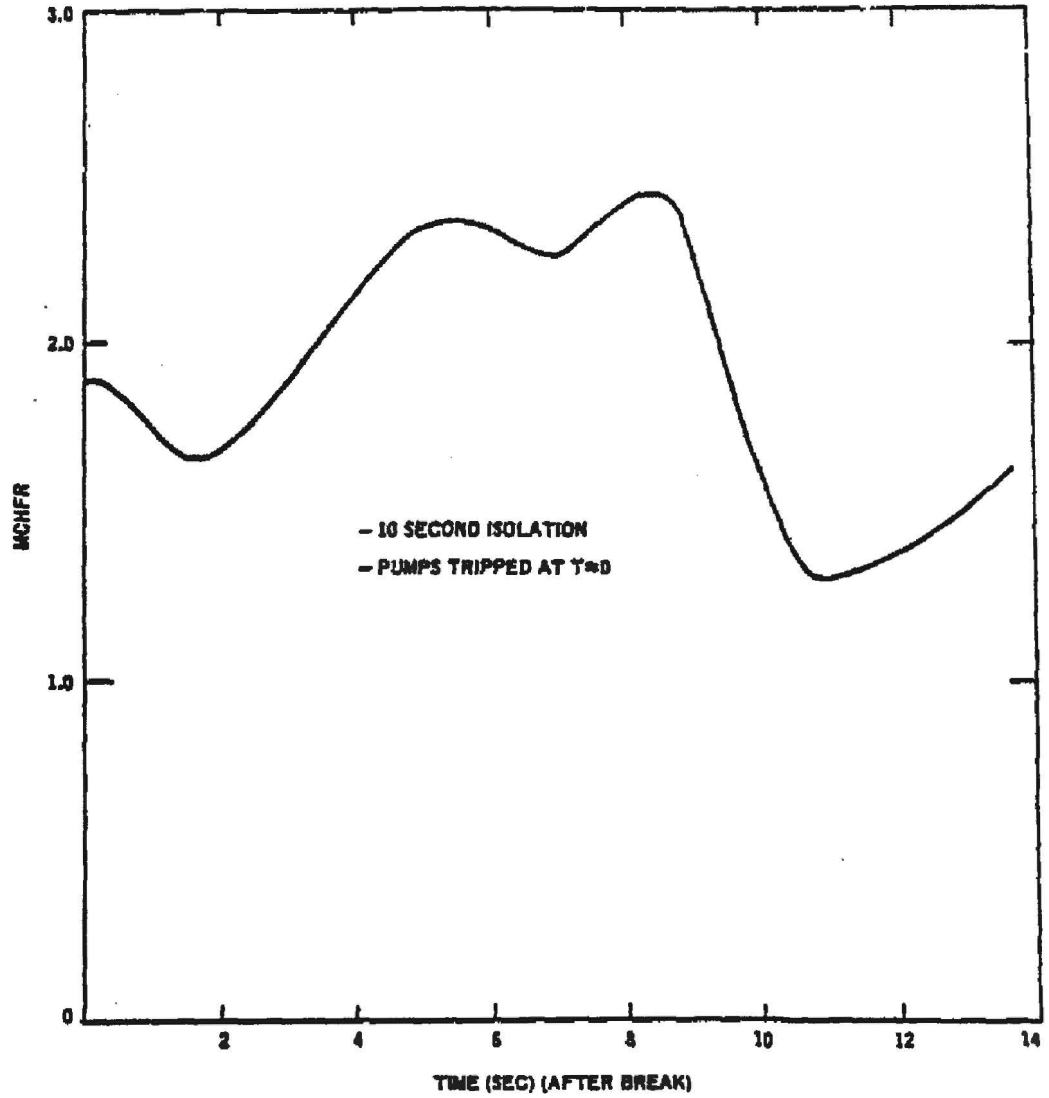
MAIN STEAMLINE BREAK ACCIDENT
MASS OF COOLANT LOST THROUGH BREAK
(HOT STANDBY CONDITIONS)

FIGURE 14.6-8



AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT
Main Steamline Break Accident Normalized Core Inlet Flow FIGURE 14.6-9



AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT
Main Steamline Break Accident Minimum Critical Heat Flux Ratio FIGURE 14.6-10

BFN-26

Figure 14.6-11

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BFN-26

Figure 14.6-12

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Figure 14.6-13

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Figure 14.6-14

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Figure 14.6-15

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Figure 14.6-16

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Figure 14.6-17

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BFN-26

Figure 14.6-18

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14.7 CONCLUSIONS

Because the spectrum of abnormal operational transients has been approached and analyzed by a method that included the various combinations of plant problems and operating conditions, general conclusions regarding the plant's behavior in response to operational problems can be made. Because none of the abnormal operational transients results in any fuel parameter exceeding its limiting value (no fuel damage), it can be concluded that unacceptable safety result l and 2 are precluded. Because peak nuclear system pressure does not exceed 1375 psig as a result of any abnormal operational transient, it can be concluded that unacceptable safety result 3 for abnormal operational transients is precluded.

The broad approach to and methodical categorization of accidents leading to unplanned releases of radioactive material from the fuel barrier and the nuclear system process barrier also justify general conclusions. A comparison of each of the design basis accident analyses with the unacceptable safety results for accidents show that items 1, 3, 4, 5 and 6 are satisfied. In Section 6 ("Core Standby Cooling System"), it is shown that in no portion of the core does the cladding attain a temperature of 2200°F for any loss of coolant accident. Thus, unacceptable safety result 2 for accidents is precluded.

14.8 ANALYTICAL METHODS

This section contains historical information for the initial operating cycle of Browns Ferry Units 1, 2, and 3. The current methodology is discussed in Section 14.6.

14.8.1 Nuclear Excursion Analysis

14.8.1.1 Introduction

Although extensive preventative measures in the forms of equipment design and procedural controls are taken to avoid nuclear excursions, such an event is assumed as a design basis accident. A continued effort is made in the area of analytical methods to assure that nuclear excursion calculations reflect the state of the art in the field. This section outlines only the broader aspects of the subject. Greater detail is available in technical literature.[1]

14.8.1.2 Description

There are many ways of inserting reactivity into a large-core boiling water reactor. However, most of them result in a relatively slow rate of reactivity insertion and therefore pose no threat to the system. The one category of reactivity additions that must be considered in evaluating large nuclear excursions is that associated with the control rod system. It appears, at this time, that the rapid removal of a high-worth control rod is the only way of obtaining a high enough rate of reactivity insertion to result in a potentially significant excursion.

The rapid removal of a high-worth rod results in a high local reactivity in a small region of the core. For large, loosely coupled cores, this would result in a highly peaked power distribution and subsequent shutdown mechanisms. Significant shifts in the spatial power generation would occur during the course of the excursion; therefore, the method of analysis must be capable of properly accounting for any possible effects of the power distribution shifts. This is an effect which is not significant in small cores.

With this background in mind, it is now possible to categorize nuclear excursions in water-moderated, oxide cores. The categorization criterion that seems most definitive is one based on the principal shutdown mechanisms that come into play. This method is particularly useful here because for fuel such as that in the current General Electric product line reactors, the principal shutdown mechanisms have a direct relationship to both the consequences of the excursion and the

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1 Wood, J. E.: "Analysis Methods of Hypothetical Super-Prompt Critical Reactivity Transients in Large Power Reactors," General Electric Company, Atomic Power Equipment Department, April 1968 (APED-5448).

applicable method of analysis. With respect to the energy densities presented, the following reference points are used:

Enthalpy = 0 cal/gm at ambient temperature,
Enthalpy = 220 cal/gm at incipient melting of UO_2,
Enthalpy = 280 cal/gm at fully-molten UO_2, and
Enthalpy = 425 cal/gm when UO_2 vapor pressure is 1000 psi.

Table 14.8-1 describes the three categories of nuclear excursions, assuming a very low initial power level. As shown in Table 14.8-1, there is some overlap in the three ranges of excursions. The indicated numbers for reactivity insertion rate, minimum period, and peak energy density are nominal values and will vary somewhat from one reactor to another.

In the low reactivity insertion rate range, the reactor is barely prompt critical, and the energy that is stored in the fuel as a result of the nuclear burst is built up at a relatively slow rate. As a result, there may be a significant amount of heat transfer out of the fuel during the burst, and the negative moderator coefficient as well as the U-238 Doppler effect contributes to the shutdown mechanisms. In the medium range, the period is much shorter, and there is very little heat transfer out of the fuel during the burst. In this case, the principal shutdown mechanism is the Doppler effect. Finally, in the high range, there exists the possibility of core disassembly during the burst, due to high internal pressure causing prompt failure of fuel rods. This results in a significant contribution toward shutdown of the excursion.

In terms of consequences, the low range is limited to no fuel cladding damage, or at worst, a small amount of burnout. This poses no threat to nuclear system integrity; therefore, from a safety viewpoint, only the medium and high ranges are considered. The design basis rod drop accident is in the medium range, well below the range where core disassembly is possible.

14.8.2 Reactor Vessel Depressurization Analysis

This section contains descriptions of the analytical methods utilized to analyze accidents for the initial operating cycle. The bounding analysis has been reanalyzed by NEDC-32484P, Revision 1 and its associated references. The following original information is retained in this section for historical purposes.

14.8.2.1 Introduction

The analytical methods used to calculate the energy and mass release rates issuing from a reactor vessel during rapid depressurization are described in this section. Conservation of mass and energy equations are written for a constant-volume system containing saturated steam and liquid in thermodynamic equilibrium to determine the thermodynamic state in the vessel. Mass flow rates into and out of

the vessel are then used to find the rate of change of system pressure and mass inventory.

14.8.2.2 Theoretical Development

The mathematical formulation for the depressurization of the reactor vessel can be derived by considering the conservation of mass and energy in the constant-volume system during rapid depressurization as shown in the control volume sketch below. If the mass flow rates are known it is possible to develop expressions of the rate of change of mass, energy, and pressure within the system.

CONTROL VOLUME, V

	P(t)	
	M(t)	
$\Sigma_j W_j(t)$	X(t)	
	h(t)	$\Sigma_i W_i(t)$

14.8.2.2.1 Mass Balance

The volume of the control system is comprised of saturated liquid and saturated vapor in equilibrium:

$$V = M_f v_f + M_g v_g = \text{constant}, \qquad (14.1)$$

where:

V = Total volume of the system (i.e., the reactor vessel)
v = Specific volume, and
M = Mass.

(The subscripts f and g refer to the liquid and vapor phases, respectively.)

Since the total mass in the system is simply

$$M = M_f + M_g \qquad (14.2)$$

then the steam quality by weight, is given as,

$$X = \frac{M_g}{M} \qquad (14.3)$$

14.8.2.2.2 Mass Rate of Change in Vessel

From continuity the rate of change of vapor mass in the system is equal to the net inflow of vapor plus the rate at which liquid is flashed to vapor due to depressurization. Hence,

$$\frac{dM_g}{dt} = \sum_j w_{g_j} - \sum_i w_{g_i} + W_{fg} \qquad (14.4)$$

where:

w = mass flow rate
W_{fg} = net flashing rate.

(The subscript j corresponds to inflow while i refers to the outflow from the vessel evaluated at the thermodynamic conditions within the system.) Similarly, the rate of change of liquid mass in the vessel is

$$\frac{dM_f}{dt} = \sum_j w_{f_j} - \sum_i w_{f_i} - W_{fg} \qquad (14.5)$$

14.8.2.2.3 Rate of Change of Energy in Vessel

The rate of change of energy in the system can be expressed from the First Law of Thermodynamics:

(Net energy inflow) - (net energy outflow) = (rate of change of internal energy)

$$\left(\dot{q} + \sum_j w_f h_f + \sum_j w_g h_g \right) - \left(\sum_i w_f h_f + \sum_i w_g h_g \right) =$$

$$\frac{d}{dt} \left(M_f h_f + M_g h_g - VP \right) \qquad (14.6)$$

where:

h = Enthalpy,
P = Saturated pressure in the system, and
•q = Heat transfer rate to the fluid from the surroundings (solids).

The right hand side of Equation (14.6) can be expanded; using the chain rule, to yield

(Rate of change of internal energy)

$$= \left[M_g \frac{dh_g}{dP} + M_f \frac{dh_f}{dP} \right] \frac{dP}{dt} + h_g \frac{dM_g}{dt} + h_f \frac{dm_f}{dt} - V \frac{dP}{dt} \quad (14.6a)$$

14.8.2.2.4 Flashing Rate in Vessel

After substituting Equations (14.4), (14.5), and (14.6a) into Equation (14.6), the expression for the net flashing rate is:

$$W_{f_g} = \frac{1}{h_{f_g}} \dot{q} + \sum_j w_g h_g - \left(\sum_j w_g \right) h_g + \sum_j w_f h_f$$

$$- \left(\sum_j w_f \right) h_f - \left[M_g \frac{dh_g}{dP} + M_f \frac{dh_f}{dP} - V \right] \frac{dP}{dt} \quad (14.7)$$

14.8.2.2.5 Vessel Depressurization Rate

In order to arrive at an expression for depressurization rate, we start by differentiating Equation (14.1), realizing that for a fixed total system volume dV/dt = 0; then,

$$M_g \frac{dv_g}{dt} + v_g \frac{dM_g}{dt} + M_f \frac{dv_f}{dt} + v_f \frac{dM_f}{dt} = 0 \quad (14.8)$$

Now expanding this by means of the chain rule we obtain:

$$v_g \frac{dM_g}{dt} + v_f \frac{dM_f}{dt} + \left(M_f \frac{dv_f}{dP} + M_g \frac{dv_g}{dP} \right) \frac{dP}{dt} = 0 \quad (14.9)$$

With expressions for dM_g/dt and dM_f/dt as given in Equations (14.4) and (14.5), Equation (14.9) can be written:

$$v_g \left[\sum_j^{w_g} - \sum_i^{w_g + W_{fg}} \right] \qquad (14.10)$$

$$+ \; v_f \left[\sum_j^{w_f} - \sum_i^{w_f - W_{fg}} \right] + \left[M_f \frac{dv_f}{DP} + M_g \frac{dv_g}{DP} \right] \frac{dP}{dt} = 0$$

After substituting Equation (14.7) into Equation (14.10) and rearranging, the following expression for depressurization rate is obtained:

$$\frac{dP}{dt} = - \left[\frac{f_1 (P) + f_2 (P)}{f_3 (P)} \right] \qquad (14.11)$$

where:

$$f_1(P) = v_f \left(\sum_j w_f - \sum_i w_f \right) + v_g \left(\sum_j w_g - \sum_i w_g \right)$$

$$f_2 (P) =$$

$$\frac{v_{fg}}{h_{fg}} \left[\dot{q} + \sum_j w_f h_f - \left(\sum_j w_f \right) h_f + \sum_j w_g h_g - \left(\sum_j w_g \right) h_g \right]$$

$$f_3 (P) = M_g \left[\frac{dv_g}{dP} - \left(\frac{v_{fg}}{h_{fg}} \right) \frac{dh_g}{dP} \right]$$

$$+ M_f \left[\frac{dv_f}{dP} - \left(\frac{v_{fg}}{h_{fg}} \right) \frac{dh_f}{dP} \right] + \left(\frac{v_{fg}}{h_{fg}} \right) \frac{V}{J}$$

J = 778 ft lbs (enthalpy)/Btu

14.8.2.2.6 Mass Flow Rates

The mass flow rates entering the reactor vessel during the blowdown are treated as functions of time and are independent of the internal thermodynamic conditions in the vessel. These flow rates can be liquid or vapor or some combination of the two. The outlet flow rate can be calculated from one of two flow models: critical flow as a function of the control volume stagnation properties P_o and h_o, or supercritical flow as a function of the pressure difference $P_o - P_{sink}$ (sink refers to the pressure outside the vessel).

Critical flow is flow which is "choked" at some point where the Mach number is unity in the line through which depressurization is taking place. Critical or maximum flow (both single-phase and two-phase) persists when the ratio of driving pressure (vessel pressure) to sink pressure (drywell) is greater than approximately two. The critical flow analysis of F. J. Moody[2] is used to determine the flow rate for critical flow conditions.

For the instantaneous values of pressure, P, enthalpy, h, and friction coefficient, f L/d, a three-variable interpolation is performed using Moody's results to find the critical mass velocity:

$$G_c = G\left(P, h, \overline{fL} / d\right) \qquad (14.12)$$

The mass flow rate is now calculated from

$$w_c = AG_c, \qquad (14.13)$$

where:

A = minimum flow area in the line.

Supercritical flow will exist prior to the formation of bubbles in a liquid flow and establishment of two-phase critical flow, or when the source pressure is low so that the ratio of $P_o / P_{sink} < 2$.

Supercritical mass velocity is calculated from:

2 Moody, F. J.: "Maximum Two-Phase Vessel Blowdown from Pipes," General Electric Company, Atomic Power Equipment Department, April 1965 (APED-4827).

$$G_{sc} = \frac{2g(P_o - P_{sink})}{v_f\ (1.4 + \overline{fL}\ /_d\ )^2_\Phi} \qquad (14.14)$$

where:

Φ = Martinelli-Nelson two-phase multiplier.[3]

The mass flow rate is:

$$W_{SC} = A\ \ G_{SC} \qquad (14.15)$$

14.8.2.3 <u>Numerical Solution</u>

If a function of time and its time derivatives are known at time t_1, then the value of the function at time $t_1 + \Delta t$ can be obtained from a Taylor series expansion. The first three terms of the series are:

$$f(t_1 + \Delta t) = f(t_1) + \frac{\Delta t}{1!} f'(t_1) + \frac{\Delta\ t^2}{2!} f''\ (t_1) + \ .\ .\ .\ , \qquad (14.16)$$

where:

$$f'(t_1) = \frac{df}{dt} \text{ at } t = t_1$$

$$f''(t_1) = \frac{d^2\ f}{dt^2} \text{ at } t = t_1 \text{ and}$$

$$\Delta t = \text{Size of time step.}$$

Integration - If the term involving the second derivative is negligible, the Euler forward integration method is obtained

$$f(t_1 + \Delta t) = f(t_1) + \Delta t\ f'(t_1) \qquad (14.17)$$

Time Step - A variable time step based on an accuracy criterion has been used in the integration method. The error made in one extrapolation of the Euler method

3 Martenelli, R. C. and Nelson, D. B., “Prediction of Pressure Drop During Forced-Circulation Boiling Water,” Trans. ASME, Vol. 70, 1948, p. 695.

can be approximated by the third term of Taylor's series given by Equation (14.16); i.e.,

$$e \approx \frac{\Delta t^2}{2!} f''(t_1) \qquad (14.18)$$

An exact equation for the second time derivative can be approximated by the rate of change of the first derivative; i.e.,

$$f''(t_1) = \frac{f'(t_1 + \Delta t) - f'(t_1)}{\Delta t} \qquad (14.19)$$

After substituting Equations (14.10) into (14.18), an approximation of the error made in one time step can be calculated:

$$e \approx \frac{\Delta t}{2} \mid f'1(t_1 + \Delta t) - f'(t_1) \qquad (14.20)$$

If the magnitude of this error is within the error criterion, then the time step is doubled for the next calculation. If $|e| > \varepsilon$, then the time step is halved and the previous calculations are repeated.

Calculations - Equations (14.4), (14.5), and (14.11) are programmed for machine calculation using the numerical methods described above.

14.8.3 Reactor Core Heatup Analysis

This section contains descriptions of the analytical methods utilized to analyze accidents for the initial operating cycle. The bounding analysis has been reanalyzed by NEDC-32484P, Revision 1 and its associated references. The following original information is retained in this section for historical purposes.

14.8.3.1 Introduction

The analytical method used to calculate the reactor core thermal transient following a loss-of-coolant accident is described in this section. The fuel temperature, cladding temperature, channel temperature, and amount of metal-water reaction are calculated as functions of time from the start of the accident. In this analysis the power of decaying fission products, the chemical energy released by metal-water

reactions, and the stored heat in the fuel, cladding, and other metal in the core are included as heat sources.

The fuel rods are classified such that those with similar power levels and fuel bundle locations are analyzed as a group. A one-dimensional heat balance is then written for each type of fuel rod. Heat is transferred from the surface of the fuel rods by convection to the water, steam or hydrogen formed in the metal-water reaction. In addition, thermal radiation between fuel rods and from the rods to the channel is accounted for in the overall heat balance.

14.8.3.2 Theoretical Development

A typical fuel rod consists of uranium dioxide fuel with a Zircaloy cladding. An initial core fuel bundle consists of 49 fuel rods, grouped together to form a square array which is surrounded by a metal channel. The fuel rods are divided into four radial temperature zones for the numerical calculations as shown in Figure 14.8-1. The cladding, on the other hand, is described by the average cladding temperature, with an outer surface temperature computed from the average temperature. The channel (Figure 14.8-1) is considered to be at a uniform temperature radially. The fuel rods within the channel are divided into four representative zones to describe the spatial variation of power generation. The entire reactor core is made up of several hundred fuel bundles and channels. To describe the radial variations of power generation, the core is divided into five radial zones. The fuel rods and channels are divided into five axial regions. Axial conduction between regions is neglected. Each channel is considered to be isolated from the rest of the core so that interactions between adjacent channels is neglected.

14.8.3.2.1 Heat Sources.

The energy generated by delayed neutrons and decaying fission products is assumed to be uniform within a fuel rod and to have the same radial and axial variation within the core as the steady-state power distribution. The chemical energy released by the metal-water reaction is described by the parabolic rate law given by Baker[4] , where the rate of change of the metal oxide thickness is written as

$$\frac{d\delta}{dt} = \frac{K}{\delta} \exp\left(D / T_c\right) \qquad (14.21)$$

where:

K = Rate coefficient,

4 Baker, L. J, and Avins, R. O.: "Analyzing the Effects of a Zirconium-Water Reaction," Nucleonics, 23(7), 70-74 (July 1965).

T_c = Cladding temperature,
D = Activation coefficient, and
δ = Oxide thickness

The heat generation rate and hydrogen release rate are proportional to the rate of change of oxide generated. The chemical heat liberated is given as follows:

$$\frac{dQ_c}{dt} = \frac{d\delta}{dt} \Delta H \rho_c A_s \qquad (14.22)$$

where:

H = Heat of reaction,
ρ_c = Density of metal, and
A_s = Exposed surface area of oxide.

The mass rate of hydrogen generated is

$$\frac{dW_H}{dt} = 2 \frac{d\delta}{dt} \rho_c A_s \frac{N_{H_2}}{N_{METAL}} \qquad (14.23)$$

where:

W_H = Mass of hydrogen generated and
N = Molecular weight.

The above reaction rate considers that there is an unlimited source of saturated steam available for the reaction. The empirical reaction constants, K and D, are based upon experimental data obtained under conditions where the metal and water are at the same temperature. Therefore, for Equation (14.21) to be correct the water must be heated to the cladding temperature. The energy required to heat this water is deducted from the total chemical energy added to the system.

14.8.3.2.2 Conduction Heat Transfer

The heatup analysis considers only radial conduction of heat from the fuel to the cladding surface. Axial conduction along the fuel rods or to support structures is neglected. Resistance to heat flow through the fuel-cladding gap is taken into account.

14.8.3.2.3 Convection Heat Transfer

Heat is transferred from the cladding and channel to the surrounding fluid by thermal radiation and convection. During the blowdown a convection heat transfer coefficient must be calculated. The water level is calculated from the mass inventory in the reactor vessel during the blowdown. If an axial node is covered with water or steam water mixtures, the heat transfer coefficient for that node is obtained from the Jens-Lottes correlation for boiling heat transfer:

$$h_B = \frac{e^{P/900}}{1.9} (Q_s)^{0.75} \qquad (14.24)$$

where:

P = Reactor pressure
Q_S = Surface heat flux

Equation (14.24) is used to describe the heat transfer coefficient if the calculated water level is above the center of the node. When water level drops below the center of the node, it is treated as being completely uncovered and the convective heat transfer rate diminishes to zero.

14.8.3.2.4 Radiation

Thermal radiation between fuel rods and the fuel channel box is permitted if they are not covered with water. To simplify calculations, the fuel rods are grouped into four groups. Figure 14.8-1 shows the channel configuration. Group 1 rods exchange radiation with Groups 2, 3, and 4 rods and the channel. Group 2 rods exchange radiation with Groups 1, 3, and 4 rods and the channel. Group 3 rods exchange radiation with Groups 1, 2, and 4 rods and the channels. Finally, Group 4 rods exchange radiation only with Groups 1, 2, and 3 rods. Radiation view factors are also calculated for each group of rods. The view factors together with the emissivity and relative areas are converted to radiation coefficients used in the Stephan-Boltzman equation for obtaining the radiant heat transfer.

14.8.3.3 Method of Solution

The fuel, cladding, and channel temperature are calculated at each time step by considering the aforementioned energy consideration. All temperatures are integrated using a simple Euler forward difference method:

$$\Phi(t + \Delta T) = \Phi(t) + \frac{d\Phi(t)}{dt} \Delta t \qquad (14.25)$$

All physical properties are considered constant with temperature and time. The model utilizes the calculated histories of pressure, water level, and heat transfer coefficients. The sink temperature for all convective heat transfer calculations is determined by the saturation temperature at the given pressure.

14.8.4 Containment Response Analysis

This section contains descriptions of the analytical methods utilized to analyze accidents for the initial operating cycle. The bounding long-term pressure suppression pool analysis has been reanalyzed by NEDC-32484P, Revision 2, GE-NE-B13-01866-4, Revision 2 and their associated references. The following original information for the long-term pressure suppression pool analysis is retained in this section for historical purposes.

14.8.4.1 Short Term Containment Response

The analytical model used to evaluate the short term response of a pressure suppression containment to a loss-of-coolant accident consists of five submodels, i.e.,

1. Reactor vessel model,

2. Drywell model,

3. Vent clearing model,

4. Vent flow model, and

5. Pressure Suppression chamber model.

These submodels are described in detail in the topical report "The General Electric Pressure Suppression Containment Analytical Model," NEDO-10320, April 1971. Included in the report are all the assumptions used in the model as well as descriptions of experimental verification and a discussion of the degree of conservatism inherent in the calculated results.

14.8.4.2 Long Term Containment Pressure Response

The preceding analytical model is used to calculate the containment transient during the reactor vessel depressurization and during the containment depressurization

which follows the vessel transient. Once the depressurization is over (about 600 seconds after the accident), a considerably simplified model can be used. The key assumptions employed in the simplified model are:

a. Drywell and pressure suppression chamber, both saturated and at the same total pressure,

b. An energy balance is performed to determine the temperature of the emergency core cooling flow as it drains by gravity back into the pressure suppression chamber. The drywell is conservatively assumed to be 5°F hotter than the water draining back into the pressure suppression pool,

c. The pressure suppression chamber air temperature is taken equal to the pool temperature which is determined from an energy balance on the pool mass, and

d. No credit is taken for heat losses from the primary containment.

Since no mass is being added to the pressure suppression pool, the pool temperature can be calculated based on the following energy balance:

$$\dot{T}_s = \frac{H_D \dot{m}_{D_o} - h_s \dot{m}_{s_o} - \dot{q}_{H_x}}{M_{W_s}} \quad (14.26)$$

where:

h_D = enthalpy of water leaving drywell
$\dot{m}_{D_o}$ = flow rate out of drywell
h_s = enthalpy of water in pressure suppression chamber
$\dot{m}_{s_o}$ = flow rate out of pressure suppression chamber
$\dot{q}_{H_x}$ = heat removal rate of heat exchanger
M_{w_s} = mass of water in presssure suppression chamber.

Assuming no storage in drywell

$$\dot{m}_{D_o} = \dot{m}_{s_o} = \dot{m}_{CSCS}$$

And since the only heat source is the core decay heat, we have:
Therefore,

$$(h_D - h_s)\ \dot{m}_{CSCS} = \dot{q}_D \quad (14.27)$$

$$\dot{T}_s = \frac{\dot{q}_D - \dot{q}_{H_x}}{M_{w_s}} \quad (14.28)$$

which can be integrated to give T as a function of time. At any point in time the drywell temperature is given by:

$$T_D = T_s + \frac{\dot{q}_D}{\dot{m}_{CSCS}} + 5^\circ\ F \quad (14.29)$$

With the pressure suppression chamber and drywell temperatures known and their total pressures assumed equal, it is now possible to solve for the total pressure

$$P_D = P_S$$

$$P_{a_D} + P_{v_D} = P_{a_S} + P_{v_S}$$

$$\frac{M_{a_D}\ R_{T_D}}{V_D} + P_{v_D} = \frac{M_{a_s}\ R_{T_s}}{V_{v_S}} + P_{v_s} \quad (14.30)$$

The total mass, M_T, can be determined from a mass balance on the primary containment:

$$\dot{M}_T = \dot{M}_{a_D} + \dot{M}_{a_s} = \dot{m}_i - \dot{m}_{LEAK} \quad (14.31)$$

where:

- $\dot{m}_i$ = all noncondensible flow into containment, e.g., hydrogen from metal-water reaction, and
- $\dot{m}_{LEAK}$ = leakage from primary containment.

Therefore, at any time, M_T, is known, and

$$M_T = M_{as} + M_{ad} \qquad (14.32)$$

The two equations (14.30 & 14.32) can be solved for the two unknowns (M_{a_s} and M_{a_d}) and the pressure determined.

The leakage rate from the primary containment is determined from the following relationship:

$$\dot{m}_{LEAK} = L_T \left[\frac{1 - \left(\frac{1}{p}\right)^2}{1 - \left(\frac{1}{P_T}\right)^2} \right]^{1/2} \qquad (14.33)$$

where:

L_T = Leak rate at test pressure
P_T = Test pressure in absolute atmospheres
P = Containment pressure in absolute atmospheres

The above equations are solved simultaneously on a step-by-step basis to obtain the long-term pressure transient of the primary containment.

14.8.5 Analytical Methods for Evaluating Radiological Effects

14.8.5.1 Introduction

This section describes the analytical techniques used to calculate the radiological exposures for design basis accidents. The descriptions following are retained for historical purposes only. The current descriptions for the radiological effects of design basis accidents are given in Sections 14.6 and 14.11.

Methods for evaluating external exposures from airborne fission products and internal exposures from inhalation of airborne radioactive materials are given.

The first portion of the analysis concerns the meteorological considerations that describe the dissemination of the radioactive material as it emanates from the source and spreads through the atmosphere. The second portion of the analysis describes the radiological effects on man as a result of the dispersed radioactive materials.

The radiological effects of the design basis accidents are evaluated at various discrete distances from the plant. The nearest distance is approximately the site boundary with other distances given to illustrate the change of the radiological effects with distance.

Since airborne materials are released via an elevated release point, the effects at short distances for any diffusion condition are usually much less for all modes of exposure except from the passing cloud. At these short distances, the plume has not yet reached ground level so that exposure from inhalation is small. The passing cloud effect, however, remains nearly constant due to essentially line-source geometry of the elevated plume.

14.8.5.2 Meteorological Diffusion Evaluation Methods

14.8.5.2.1 General

Six points in the atmospheric diffusion spectrum are used to evaluate the radiological effects of secondary containment leakage via the elevated release point. These points represent the meteorological conditions which could exist at the site.

The atmospheric diffusion methods are the same as those reported in the Journal of Applied Meteorology.[5]

14.8.5.2.2 Height of Release

Discharge from the secondary containment to the atmosphere emanates from the elevated release point. The effective height of release is the sum of the release point height plus any effluent rise due to momentum or buoyancy. For most of the design basis accidents, the additional effects of momentum and buoyancy are negligible, so that the effective release height is equal to the elevated release point height (183 meters). While buoyancy effects are significant for the steam line break accident, the conservative assumption is made that the release height is equal to the top of the turbine building.

14.8.5.2.3 Diffusion Conditions

An important parameter used in the atmosphere diffusion calculation is the measure of wind direction persistence and variability of direction. This parameter is the product of the standard deviation of the horizontal wind direction fluctuations, $\sigma \varnothing$ and the average wind velocity $\bar{u}$. Combined with the assumed stability condition, specification of $\sigma \varnothing \bar{u}$ permits calculation of air concentrations at various distances from the source.

A conservative value of 0.1 radian-meters/second is used for this parameter to describe the horizontal spreading of the plume for 1 meter/ second wind speed conditions. A value of 1.0 radian-meter/second is typical for a 5 meter/second condition. These values are typical for a one hour period. A choice of wind direction persistence (number of continuous hours) of 24 hours is used for poor diffusion conditions. This period is conservative when used with $\sigma \varnothing \bar{u}$ of 0.1 radian-meter/ second and 1 meter/second wind speed, based upon the U.S. Weather Bureau Data shown in Table 14.8-2.[6] Table 14.8-2 shows that wind persistency of periods as long as 24 hours occurs only about 0.1 percent of the time or less at the sites listed. The sites include flat terrain, coastal and lake shore sites and some valley locations. For a wind speed of 5 meters/second, a value of 1.0 radian-meters/second corresponds to a $\sigma \varnothing$ of 0.20 radians which is similar to the value of 0.1 radians for the 1 meter/second case. Thus, about the same amount of

5 Fuguay, J. J., Simpson, C. L., and Hinds, W. T. "Prediction of Enviornmental Exposures from Sources Near the Ground Based on Hanford Experimental Data." Journal of Applied Meteorology, Vo. 3 No. 6, December 1964.

6 Pack, D. H., Angell, J. K., Van Der Hoven, I., and Slade, D. H., USWB, "Recent Developments in the Application of Meteorology to Reactor Safety," presented at the 1964 Geneva Conference, paper number A/CONF/28/P/714.

wind variability is considered and the conservative 24-hour persistence assumption is applicable to both cases.

14.8.5.2.4 Applied Meteorology

The diffusion and wind direction persistence conditions and breathing rates used for the design basis accident calculations are given in Table 14.8-3.

14.8.5.2.5 Cloud Dispersion Calculations

The dispersion of the released effluent is described by the Gaussian Diffusion Equation given below.

$$X / Q_o = \frac{f_d}{2p\, s_y\, s_z\, \bar{u}}\, e - \frac{1}{2}\left(\frac{y^2}{s_y^2} + \frac{z^2}{s_x^2}\right) \quad (14.34)$$

where:

X/Q_o	=	integrated air concentration (X) per unit activity release (Q_o)
Y	=	distance from centerline crosswind (since plume centerline is used, Y = o)
z	=	height of plume above ground
f_d	=	cloud depletion factor (halogens only) see paragraph 14.8.5.2.6
σ_y	=	Horizontal diffusion coefficient
σ_z	=	Vertical diffusion coefficient

σ_y and σ_z are defined as follows:

$$\sigma^2_y = At - A\alpha + A\alpha e^{-t/a} \text{ (See footnote 5)} \quad (14.35)$$

where:

$A = 13 + 232.5\,(\sigma\,\theta\,\bar{u})$

$\alpha = \dfrac{A}{2\ (\sigma\theta\,\bar{u})^2}$

t = time after release and is = $x/\bar{u}$, where x is downwind distance.

The vertical cloud growth, as defined by the standard deviation of width σz is given by

$$\sigma_x^2 = a\left(1 - e^{-k^2 t^2}\right) + bt \quad \text{(stable case) (See footnote 5) (14.36)}$$

$$\sigma_z^2 = \frac{C_z^2 x^{(2-n)}}{2} \quad \text{(neutral / unstable case) (See footnote 7) (14.37)}$$

The values of the constants in Equations (14.36) and (14.37) are given below.

Stability	Wind Speed (M/sec)	a (M^2)	b (M^2/sec)	k^2 (Sec^{-2})
VS	1	3.4×10^{1}	2.5×10^{-2}	8.8×10^{-4}
U	1	-	-	-
U	5	-	-	-
N	1	-	-	-
N	5	-	-	-
MS	1	9.7×10^{1}	3.3 x 10-$^{-1}$	2.5×10^{-4}

Stability	Wind Speed (M/sec)	c_z ($M^{n/2}$)	n -
VS	1	-	-
U	1	3.0×10^{-1}	2.0×10^{-1}
U	5	2.6×10^{-1}	2.0×10^{-1}
N	1	1.5×10^{-1}	2.5×10^{-1}
N	5	1.2×10^{-1}	2.5×10^{-1}
MS	1	-	-

The conventional "reflection" factor of 2 usually applied for releases from ground-level is not included. For the passing cloud dose, which is primarily a

gamma dose, the entire cloud volume is integrated as an "infinite" number of point sources to plus and minus infinity in the z-direction ignoring interception by the ground, so that the entire cloud volume is included. Inhalation doses are a function of concentration at the ground and subject to "reflection" effects if they exist. Since the materials of interest in inhalation effects deposit on the ground, "perfect" reflection will not occur, but rather the cloud will expand distorting the Gaussian mass distribution resulting in at most a small increase in concentration. In addition, no account is taken of the better diffusion near the ground compared to the stack exit elevation used. In any event, an increase by a factor of less than 2 but perhaps more than 1 may be a result of this "reflection" effect. A factor of 1.0 is used in this analysis.

No distinction in the choice of the diffusion parameter $\sigma\varnothing\bar{u}$ is made between the first two-hour period and the total accident dose calculations. This is inconsistent because larger values of this parameter are obviously appropriate for the longer time period. That is, the values used, as discussed in paragraph 14.8.5.2.3, are for one-hour periods, and thus are somewhat conservative when applied to the two-hour period dose calculation and are markedly conservative for the total accident calculation. Lack of data at this time for the longer time period does not permit more precise estimates to be made.

14.8.5.2.6 <u>Cloud Depletion and Ground Deposition</u>

The fallout concentrations of radioactive materials are determined on the basis of particle settling by eddy diffusion only, since settling by gravity is expected to be negligible in this case.

The extent of halogen and solid fission product deposition on the ground is a function of the apparent deposition velocity, which, in turn, is considered to be function of the diffusion condition and wind speed. Deposition velocities used in this evaluation are given below.[7]

Meteorology	Wind Velocity (M/sec)	Deposition Velocity (cm/sec) Noble Gases	Halogens
Very stable	1	0	0.24
Moderately stable	1	0	0.34
Unstable	1	0	0.80
Unstable	5	0	4.00
Neutral	1	0	0.46
Neutral	5	0	2.30

These values of deposition velocity are used in the calculation of the cloud depletion term f_d.

$$f_d = \exp\left[- \frac{v_g}{u_o} \sqrt{\frac{2}{p}} \frac{u_o}{u_h} \int_o^t \frac{u_h \exp\left(\frac{-z^2}{2s^{2z}}\right)}{s_z} dt \right] \quad (14.38)$$

where:

f_d = Cloud depletion factor due to fallout
V_g = Deposition velocity of isotope in question (cm/sec)
u_o = Wind speed at ground level (cm/sec)
u_h = Wind speed at height of release (cm/sec)
σz = Vertical diffusion coefficient (cm)

14.8.5.2.7 Air Concentration Calculation

Using the equations developed above, the integrated air concentration from a release of 1 curie of activity is calculated in curie-seconds per cubic meter. These data are given in Tables 14.8-4 and 14.8-5 for the specified release heights and meteorological conditions.

7 Watson, E. C. and Gamertsfelder, C. C., "Environmental Radioactive Contamination as a Factor in Nuclear Plant Siting Criteria," HW-SA-2809, February 14, 1963.

14.8.5.3 Radiological Effects Calculation

The radiological doses of primary consideration are inhalation and cloud gamma. While the deposition gamma dose may be important from a decontamination viewpoint, it is of minor importance in evaluating the radiological consequences of a design basis accident and is, therefore, insignificant in this analysis.

The downwind radiological effects, such as cloud gamma and inhalation exposure, are a function principally of the integrated air concentration at any point. Calculation of this integrated concentration has been described in the preceding paragraphs. This paragraph describes the conversion of air concentration to radiation dose.

14.8.5.3.1 Passing Cloud Dose

The ground level whole body cloud gamma dose which is received from airborne radioactive materials is determined by summing the dose contribution from each incremental volume of air containing fission product activity. The dose from a point in space to a receptor located at coordinates X, Y, and Z is determined as follows:

$$D_g = \sum_{K=1}^{m} C_1 C_K f_K \int_{-\infty}^{\infty} \int_{-\infty}^{\infty} \int_{-\infty}^{\infty} X G_K \, dxdydz \quad (14.39)$$

D_g = Gamma dose at the receptor point (rem)
C_1 = Conversion factor (3.7×10^{10} disintegrations/sec-curie)
X = Integrated air concentration (curie-sec/m^3)
f_K = The number of photons of the Kth isotope released per disintegration (photons/dis)
C_K = Flux to dose conversion factor. $\frac{\text{rem-m}^2\text{ - sec}}{\text{Sec-y}}$
G_K = Dose attenuation kernel which is defined as follows

Where

$G_K = Be^{-uT} / 4\pi T^2$

Where

B = Buildup factor = 1 + KuT

$$K = \frac{u - u_a}{ua} \quad (14.40)$$

Where u is the total absorption coefficient and u_a is energy absorption coefficient (m^{-1})

T = Distance from the source to the detector position and is equal to

$$\sqrt{x_1^2 + y_1^2 + z_1^2} \quad (14.41)$$

14.8.5.3.2 Inhalation Dose

The inhalation dose is an internal exposure which is received as a consequence of inhaling airborne radioactive fission products. Depending upon the isotopes inhaled there may be one or more organs which are affected.

The total activity inhaled during the inhalation period is

$$Q \text{ dep.} = \chi_i B_r \quad (14.42)$$

where:

χ_i = Time integral of the air concentration previously defined in paragraph 14.8.5.2.7 (curie-sec/m^3)

B_r = Breathing rate (m^3/sec)

When the above equation is multiplied by an appropriate conversion factor C_i (rem/sec-curie inhaled), a dose rate in the organ is obtained. The total dose resulting from inhalation of a mixture of fission products is

$$D_i = \sum_{i=1}^{N} \int_o^t c_i B_r C_i e^{-\lambda_i t} dt \quad (14.43)$$

Where

D_i = Total inhalation dose (rad)

λ_i = Effective decay constant of the ith isotope in the organ of reference (sec^{-1})

The summation sign indicates that all isotopes contributing to the organ dose are added together to obtain the total inhalation dose.

The conversion factor C_i is applicable to the isotope of the organ of interest and is calculated by the use of the following mathematical model[8]

$$C_i = \frac{C_1 f_a E C_3}{MC_2} \quad (14.44)$$

C_i = Activity to dose conversion factor (rad/sec-curie inhaled)
C_1 = 1.6×10^{-6} ergs/MeV
f_a = Fraction of inhaled material reaching the organ of reference
E = The effective energy absorbed per disintegration MeV/dis)
C_3 = 3.7×10^{10} dis/sec-curie
M = Mass of the organ (gms)
C_2 = 100 ergs/gm-rad

Therefore:

$$C_i = \frac{(1.6 \times 10^{-6})(f_a)(E)(3.7 \times 10^{10})}{(M)(100)}$$

$$= \frac{5.92 \times 10^2 f_a E}{M} \text{ (rad / sec curie inhaled)} \quad (14.45)$$

Upon integration of Equation (14.43), the total inhalation dose is

$$D_T = \sum_{i=1}^{N} \frac{c_i B_r C_i (1 - e^{-l_i t})}{l_i} \quad (14.46)$$

8 Morgan, K.A., Snyder, W. S. Auxier, J. A., "Report of the ICRP Committee II on Permissible Dose for Internal Radiation (1959)" Health Physics, Vol. 3 (1960).

If T is large compared to λ, Equation (14.46) can be simplified to

$$D_T = \sum_{i=1}^{N} \frac{c_i B_r C_i}{l_i} = \sum_{i=1}^{N} \frac{c_i B_r C_i T_i}{0.693} \quad (14.47)$$

Where:

T_i = Effective half life of the ith isotope and is equal to

$$T_i = \frac{T_b T_r}{T_b + T_r}$$

T_b = Biological half life (sec)
T_r = Radioactive half life (sec)

If the effective half life is defined in terms of days and is combined with the conversion factor C'_i Equation (14.47) can be expressed follows

$$D_T = \sum_{i=1}^{N} c_i B_r C'_i \quad (14.48)$$

Where

$$C'_i = \frac{8.64 \times 10^4}{6.93 \times 10^{-1}} T_i C_i = 1.25 \times 10^5 C_i T_i \text{ (rad / curie inhaled)} \quad (14.49)$$

For the thyroid gland the dose conversion factor is

$$C'_i = \frac{(1.25 \times 10^5)(5.92 \times 10^2)(f_a) ET_i}{M}$$

$$= \frac{7.40 \times 10^7 f_a ET_i}{20} = 3.7 \times 10^6 f_a ET_i \quad (14.50)$$

The numerical values which are used in Equation (14.45), as well as the dose conversion factor, C'_i, are given in Table 14.8-6.

Table 14.8-1

CHARACTERISTICS OF NUCLEAR EXCURSIONS
WATER-MODERATED OXIDE CORES

Range	Reactivity Insertion Rate ($/sec)	Minimum Period (ms)	Peak Energy Density (cal/gm)	Principal Shutdown Mechanisms
Low	<2.5	>4	<120	Doppler Effect Moderator Effects
Medium	2-25	7-2	100-425	Doppler Effect
High	>20	<3	>380	Doppler Effect Core Disassembly

Table 14.8-2

DOSE COMPUTATIONAL METHODS WIND DIRECTION PERSISTENCE

Frequency of Duration in Hours

(One Sector - 22 1/2°)

Station	Direction*	50%	10%	1%	0.1%	Longest No. Hours	Longest No. Hours** in any Direction	
Augusta, Georgia	W	2	3	8	13	18	W	18
Birmingham, Alabama	S	2	4	9	16	16	SSE	20
Chicago, Illinois	SSW	2	5	12	21	22	NSE	25
Little Rock, Arkansas	SSW	2	4	9	17	28	SSE	28
Phoenix, Arizona	E	2	3	6	9	12	E	12
Rochester, New York	WSW	2	6	13	23	28	WSW	28
Salt Lake City, Utah	SSE	2	4	7	13	15	S	17
San Diego, California	NW	2	6	12	16	17	WSW	33
Tampa, Florida	ENE	2	3	7	13	14	SSW	18
Yakima, Washington	W	2	5	8	14	17	WNW	19

*Direction examined is the one showing greatest frequency of persistent winds.

**Longest number of hours observed may not be same direction as direction showing most frequency of persistent winds.

Table 14.8-3

METEOROLOGY APPLICABLE TO DESIGN BASIS ACCIDENTS

Time After Accident	Diffusion Conditions Investigated		Wind Variance During Indicated Time Period	Breathing Rate M^3/sec
	Stability Category*	$\sigma \varnothing$ u		
0-8 hrs	VS-1, MS-1, N-1, N-5, U-1, U-5	0.1 for $\bar{u}$ = 1.0 1.0 for $\bar{u}$ = 5.0	None (centerline concentration)	3.47×10^{-4}
8-24 hrs	VS-1, MS-1 N-1, N-5 U-1, U-5	0.1 for $\bar{u}$ = 1.0 1.0 for $\bar{u}$ = 5.0	None (centerline concentration)	1.75×10^{-4}
>24 hrs	VS-1, MS-1 N-1, N-5 U-1, U-5	0.1 for $\bar{u}$ = 1.0 1.0 for $\bar{u}$ = 5.0	Wind assumed to blow in 22.5° sector 1/4 of the time	2.32×10^{-4}

*VS denotes very stable meteorological conditions.

MS - moderately stable, N-neutral, and U - unstable meteorological conditions. 1 and 5 denotes wind speed in meters/second.

Table 14.8-4

CALCULATED AIR CONCENTRATION FOR 183 METER RELEASE HEIGHT

Distance (meters)	Activity of Interest	VS-1	MS-1	N-1	N-5	U-1	U-5
			(Curie-sec/m^3/curie released) Meteorological Conditions				
1,400	Noble Gases	0	5.3×10^{-18}	2.1×10^{-7}	2.3×10^{-9}	4.0×10^{-6}	4.3×10^{-7}
	Halogens	0	5.3×10^{-18}	2.1×10^{-7}	2.3×10^{-9}	3.9×10^{-6}	4.2×10^{-7}
3,000	Noble Gases	0	4.2×10^{-12}	1.7×10^{-6}	1.5×10^{-7}	1.9×10^{-6}	2.9×10^{-7}
	Halogens	0	4.2×10^{-12}	1.7×10^{-6}	1.5×10^{-7}	1.9×10^{-6}	2.8×10^{-7}
8,000	Noble Gases	1.8×10^{-36}	1.4×10^{-8}	9.8×10^{-7}	1.7×10^{-7}	4.7×10^{-7}	8.4×10^{-8}
	Halogens	1.8×10^{-36}	1.4×10^{-8}	9.4×10^{-7}	1.6×10^{-7}	4.4×10^{-7}	7.9×10^{-8}
16,000	Noble Gases	1.9×10^{-22}	1.3×10^{-7}	4.1×10^{-7}	8.2×10^{-8}	1.7×10^{-7}	3.2×10^{-8}
	Halogens	1.9×10^{-22}	1.3×10^{-7}	3.9×10^{-7}	7.6×10^{-8}	1.6×10^{-7}	2.9×10^{-8}

Symbols refer to stability and wind speed, i.e., VS, MS, N, U, means very stable, moderately stable, neutral and unstable respectively and 1 and 5 means 1 meter/sec and 5 meters/sec, respectively. The diffusion parameter $\Delta \varnothing$ u assumed is 0.1 radian-meter/sec for the 1 meter/sec cases and 1.0 radian-meter/sec for the 5 meter/sec cases.

Table 14.8-5

CALCULATED AIR CONCENTRATION FOR 183 METER RELEASE HEIGHT

Distance (meters)	Activity of Interest	VS-1	MS-1	N-1	N-5	U-1	U-5
			(Curie-sec/m^3/curie released) Meteorological Conditions				
1,400	Noble Gases	3.9×10^{-5}	7.2×10^{-5}	3.9×10^{-5}	1.1×10^{-5}	7.5×10^{-6}	2.0×10^{-6}
	Halogens	3.7×10^{-5}	7.0×10^{-5}	3.7×10^{-5}	1.1×10^{-5}	6.9×10^{-6}	1.8×10^{-6}
3,000	Noble Gases	1.1×10^{-5}	4.2×10^{-5}	1.1×10^{-5}	3.5×10^{-6}	1.9×10^{-6}	5.2×10^{-7}
	Halogens	1.0×10^{-5}	3.8×10^{-5}	1.0×10^{-5}	3.1×10^{-6}	1.7×10^{-6}	4.6×10^{-7}
8,000	Noble Gases	2.1×10^{-6}	1.5×10^{-5}	2.1×10^{-6}	6.5×10^{-7}	3.3×10^{-7}	8.9×10^{-8}
	Halogens	1.8×10^{-6}	1.2×10^{-5}	1.8×10^{-6}	5.6×10^{-7}	2.9×10^{-7}	7.5×10^{-8}
16,000	Noble Gases	6.2×10^{-7}	6.8×10^{-6}	6.2×10^{-7}	1.9×10^{-7}	9.6×10^{-8}	2.5×10^{-8}
	Halogens	5.2×10^{-7}	4.7×10^{-6}	5.2×10^{-7}	1.6×10^{-7}	8.0×10^{-8}	2.0×10^{-8}

Symbols refer to stability and wind speed, i.e., VS, MS, N, U, means very stable, moderately stable, neutral and stable respectively and 1 and 5 means 1 meter/sec and 5 meters/sec, respectively. The diffusion parameter Δ 8,45 u assumed is 0.1 radian-meter/sec for the 1 meter/sec cases and 1.0 radian-meter/sec for the 5 meter/sec cases.

Table 14.8-6

THYROID DOSE CONVERSION FACTORS

Isotope	Effective 1/2 Life (Days)	f_a	E (Mev/dis)	C'_i Rad/ curie inhaled)
I-131	$7.6 \times 10^{\circ}$	2.3×10^{-1}	2.3×10^{-1}	1.48×10^{6}
I-132	9.7×10^{-2}	2.3×10^{-1}	6.5×10^{-1}	5.65×10^{4}
I-133	8.7×10^{-1}	2.3×10^{-1}	5.4×10^{-1}	4.21×10^{5}
I-134	3.6×10^{-2}	2.3×10^{-1}	8.2×10^{-1}	2.64×10^{4}
I-135	2.8×10^{-1}	2.3×10^{-1}	5.2×10^{-1}	1.30×10^{5}

FUEL ROD DETAIL

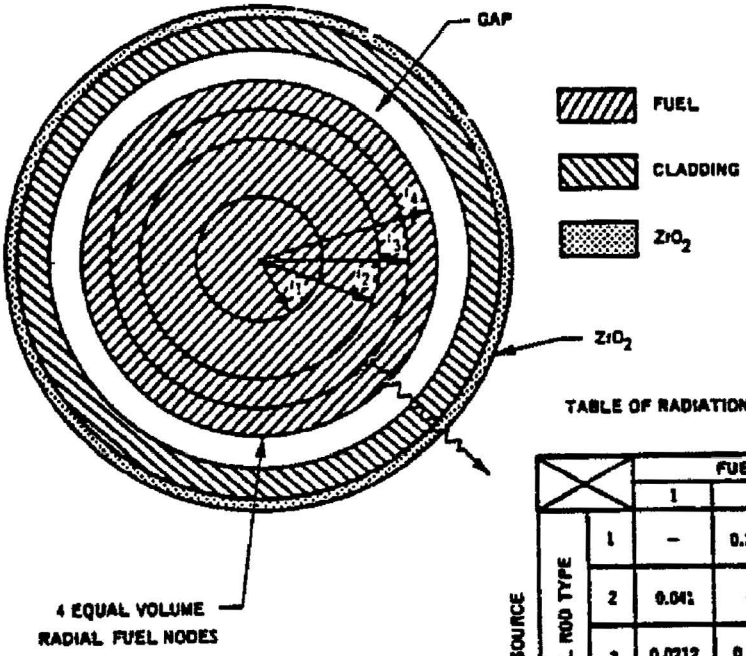
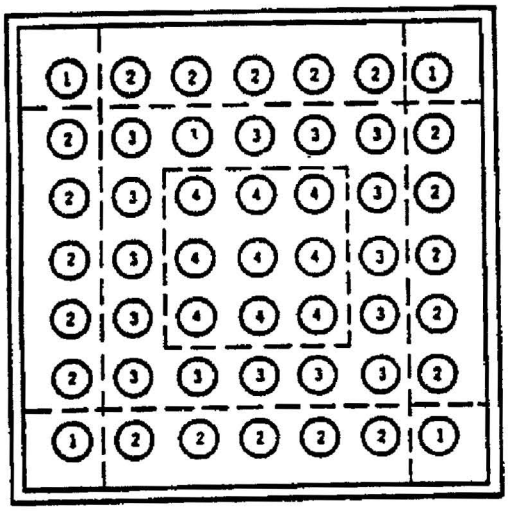


TABLE OF RADIATION HEAT TRANSFER COEFFICIENTS

SOURCE		RECEPTOR: FUEL ROD TYPE 1	2	3	4	CHANNEL SIDE WALL
FUEL ROD TYPE	1	–	0.205	0.0649	0.007	0.2914
	2	0.041	–	0.177	0.0287	0.1784
	3	0.0212	0.221	–	0.150	0.0235
	4	0.003	0.0637	0.268	–	NIL
CHANNEL SIDE WALL		0.0975	0.299	0.031	NIL	–

CHANNEL CROSS SECTION



FUEL BUNDLE DETAIL

CHANNEL		
ROD TYPE	No. OF RODS	ROD POWER FACTOR
1	4	1.24
2	20	1.09
3	16	0.91
4	9	0.85

AMENDMENT 16

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

Fuel Rod and Fuel Bundle Details
FIGURE 14.8-1

14.9 (Deleted)

TABLE 14.9-1

(Deleted)

Table 14.9-2

(Sheet 1)

(Deleted)

Table 14.9-2

(Sheet 2)

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.

Table 14.9-2

(Sheet 3)

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Table 14.9-2

(Sheet 4)

(Deleted)

14.10 (Deleted)

Figures 14.10-1 through 14.10-20

Deleted by Amendment 28

14.11 (Deleted)

Table 14.11-1 through Table 14.11-9

(Deleted)

Figure 14.11-1 through Figure 14.11-18

(Deleted)

14.12 ANALYSIS OF THE PRIMARY CONTAINMENT RESPONSE

This section describes the analytical bases and results supporting BFN Units 1, 2, and 3 operation at the rated thermal power (RTP) level of 3952 MWt.

BFN Units 1, 2, and 3 use the Mark I primary containment design. The main function of the Mark I containment design is to accommodate pressure and temperature conditions within the drywell resulting from a LOCA or a reactor blowdown through the MSRV discharge piping and, thereby, to limit the release of fission products to values which will ensure off-site dose rates below the 10 CFR 50.67 limits. In the event of a pipe break in the drywell, water and/or steam from the RPV are discharged into the drywell. The resulting increase in the drywell pressure forces the water and steam, along with non- condensable gases initially existing in the drywell, through the vents which connect the drywell to the suppression pool. During a reactor blowdown through the MSRVs, the steam is directly discharged into the suppression pool. The reactor blowdown flow rate is dependent on the reactor initial thermal-hydraulic conditions, such as vessel dome pressure and the mass and energy of the fluid inventory in the RPV.

The long-term heatup of the suppression pool following a LOCA is governed by the capability of the RHR System to remove decay heat which is transferred from the RPV to the suppression pool.

The Primary Containment system requirements are:

Design Pressure	$\leq$ 56 psig
Design Temperature	$\leq$ 281°F

14.12.1 Methodology

The analyses of containment pressure and temperature responses for design basis accident were performed at a power level of 102% of RTP in accordance with NEDC-32424P-A, Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate (ELTR1) (Reference 1) using GE-Hitachi (GEH) codes and models. The M3CPT code was used to model the short-term containment pressure and temperature response. The modeling used in the M3CPT analyses is described in References 2 and 3. References 2 and 3 describe the basic containment analytical models used in GEH codes. Reference 4 describes the more detailed RPV model (LAMB) used for determining the vessel break flow in the containment analyses.

The LAMB code models the recirculation loop as a separate pressure node. It also allows for inclusion of flashing in the pipe and vessel during the blowdown and flow choking at the jet pump nozzles when the conditions warrant. The use of the LAMB blowdown flow in M3CPT was identified in ELTR1 by reference to the LAMB code qualification in Reference 4.

The SHEX code was used to model the long-term containment pressure and temperature response. The results from the SHEX long-term containment temperature response (and the results from the ODYN/STEMP containment temperature response for the limiting ATWS event) were passed as inputs to the ECCS pump NPSH evaluations for each event. These NPSH evaluations demonstrated that ECCS pump NPSHA was more than adequate and quantified the resulting NPSH margins without crediting Containment Accident Pressure (CAP). The key models in SHEX are based on models described in Reference 3. The GEH containment analysis methodologies have been applied to all BWR power uprate projects performed by GEH and accepted by the NRC.

Original long-term containment analyses did not credit passive heat sinks in the drywell, torus airspace, and suppression pool. This conservative assumption was identified to the NRC as Assumption 6 of Attachment 1 to the March 12, 1993 GE letter referenced in Reference 5. Long-term containment analyses performed for BFN Units 1, 2, and 3 now includes credit for these passive heat sinks. This was identified as a change in methodology in Reference 6. These long-term containment analyses continue to conservatively neglect any heat loss from the containment through the containment walls to the reactor building or environs (Assumption 8 of the same GE letter, Reference 5).

The metal-water reaction energy versus time relationship is calculated using the method described in USNRC RG 1.7 (Reference 7) as a normalized value (fraction of reactor thermal power). All of the energy from the metal-water reaction is assumed transferred to the reactor coolant in the first 120 seconds into the LOCA. The metal-water reaction energy represents a very small fraction of the total shutdown energy transferred to the coolant.

A summary of the codes, version and NRC approval is provided below:

Computer Code	Version	NRC Approved	Comments
SHEX	06	Yes	Note 2
M3CPT	05	Yes	NEDO-10320, Apr. 1971 (NUREG-0661)
LAMB	08	Note 1	NEDE-20566-P-A September 1986

Note 1: The LAMB code is approved for use in ECCS-LOCA applications (NEDE-20566-P-A), but no approving NRC safety analysis report exists for the use of LAMB in the evaluation of Reactor Internal Pressure Differences (RIPDs) or containment system response. The use of LAMB for these applications is consistent with the model description of NEDE-20566-P-A.

Note 2: The application of the methodology in the SHEX code to the containment response is approved by the NRC in the letter to G. L. Sozzi (GE) from A. Thadani (NRC), "Use of the SHEX Computer Program and ANSI/ANS 5.1-1979 Decay Heat Source Term for Containment Long-Term Pressure and Temperature Analysis," July 13, 1993 (Reference 5).

14.12.2 Short-Term Containment Pressure Response

The short-term containment response analysis was performed for the limiting DBA LOCA that assumes a double ended guillotine break of a recirculation suction line to demonstrate that operation at RTP (3952 MWt) does not result in exceeding the containment design limits. The short-term analysis covers the blowdown period during which the maximum drywell pressure and suppression chamber (torus) pressure occur. The analysis was performed at 102% of RTP (4031 MWt). The time-dependent results of the limiting short-term analysis are presented in Figures 14.12-1 through 14.12-6 and are summarized in Table 14.12-6. The maximum calculated containment pressure remains within the design value.

The short-term analysis was performed for three different initial containment conditions. The Design case (D) considers the most limiting initial containment conditions of 70°F in the drywell and 2.6 psig in the drywell and 1.5 psig in the wetwell (torus). The Bounding case (B) considers initial containment conditions of 130°F in the drywell and 2.6 psig in the drywell and 1.5 psig in the wetwell, bounding normal operation. A Reference case (R) is also evaluated that assumes initial

conditions of 150°F in the drywell and 2.6 psig in the drywell and 1.5 psig in the wetwell. It is conservative to use 130ºF as the initial drywell gas temperature instead of 150ºF because it leads to a higher peak drywell temperature in the analysis use for equipment qualification evaluation. A lower initial drywell gas temperature in the analysis results in more initial non-condensable gas in the drywell which leads to a higher pressure and accordingly higher peak temperature for the small steam line break analysis. The Design case (D) and Bounding case (B) were also performed at 3458 MWt (105% Power Uprate) conditions to provide comparison for evaluating the impact of operation at RTP (3952 MWt) conditions. The key parameters used to model and analyze the plant response at RTP are provided in Table 14.12-1.

The use of the Design (D) cases initial drywell temperature is to provide the most conservative hypothesized initial conditions in order to demonstrate that a DBA LOCA will not challenge the BFN containment design pressure of 56 psig. The Design Case initial temperature of 70°F is well below the lowest drywell initial temperature that can be achieved with BFN operating at power and is therefore very conservative for demonstrating the maximum BFN containment pressure response.

The initial drywell temperature for the Bounding (B) cases was developed with a conservative historical statistical basis, which also achieve a conservative prediction of the containment pressure response due to a DBA LOCA. The containment pressure response determined from the DBA LOCA using conservative initial conditions is then used to determine a conservative value of ‘Pa’ for 10 CFR Part 50 Appendix J leakage rate testing. The Bounding Case initial temperature of 130°F represents a lower statistical bound of the 5-year historical normal drywell operating temperature during power operation of BFN Units 1, 2, and 3.

14.12.3 Drywell and Wetwell Temperature

The bounding drywell temperature occurs during a break of a small steam line (SSLB). A spectrum of steam line break sizes have been evaluated to ensure a bounding drywell environmental qualification temperature profile is established (Figure 14.12-7.a). The analysis was performed in accordance with NUREG-0588 (Reference 9), and the most limiting drywell temperature from this analysis is shown in Table 14.12-6. Although the drywell environment may see temperatures approaching 337°F, the most limiting temperature for the drywell shell has been analyzed to be within the design temperature of 281°F. The peak drywell wall temperature from the complete spectrum of analyzed break sizes is 280.8 F. The maximum predicted drywell shell temperatures occur at the beginning of the event prior to the initiation of drywell sprays. This is because the maximum drywell atmosphere temperature and drywell pressure conditions occur during this early period with maximum heat transfer to the drywell shell. Drywell pressure and

temperature, together with drywell shell temperature, decrease when drywell spray is initiated. The drywell pressure response to SSLB is shown in Figure 14.12.7.b.

The maximum DW airspace and DW shell temperature occurs early in the event with the assumption that a High Drywell Pressure (HDWP)/Low RPV Pressure (LRPVP) LOCA signal occurs in the accident unit 10 minutes after the accident initiation. This LOCA signal at 10 minutes will result in a further 10 minute delay in the initiation of drywell spray in the accident unit (20 minute total delay for drywell spray initiation).

The torus gas space peak temperature response was calculated assuming a heat and mass transfer model between the pool and torus gas space that is calculated mechanistically. Table 14.12-6 shows the calculated peak torus gas space temperature for the DBLOCA of 174°F. The torus gas temperatures are bounded by the torus design temperature of 281°F.

14.12.4 Long-Term Bulk Suppression Pool Temperature Response-Design Basis Accidents

The long-term bulk pool temperature response at RTP is evaluated for the limiting design basis LOCA (DBA-LOCA) originally analyzed (i.e., one core spray loop and one RHR loop with two RHR pumps, two heat exchangers and two RHR service water pumps with containment spray). This DBA-LOCA is an instantaneous guillotine break of the recirculation loop suction line (RSLB). A guillotine break of a Recirculation Discharge Line (RDLB), small break LOCAs and special events (Fire event, Station Blackout, and ATWS) were also analyzed at RTP conditions.

The GE Safety communication SC 06-01 (Reference 10) identified the potential that a single failure that eliminated only the RHR heat exchanger could prove more limiting than the typically analyzed scenario of the single failure of an entire AC electrical power source. The Browns Ferry RHR system is configured with two loops of RHR, with each loop having its own separate injection point to the reactor pressure vessel, and with each loop having its own separate return to the suppression pool. Each loop is comprised of two RHR pumps with each pump having its own separate heat exchanger on its discharge. The current licensing basis analysis (Reference 8) for the short-term (first 10 minutes after the accident) evaluation of the RSLB assumed a Single Active Failure (SAF) where only two of the four RHR pumps were available. In order to address the issue identified in SC 06-01, the RSLB RTP analysis assumes that all four RHR pumps are running in the short-term phase of the RSLB DBA-LOCA. This assumption will maximize the ECCS pump heat addition to the suppression pool and thereby maximize the suppression pool temperature. The RDLB analysis for a RTP of 3458 MWt conservatively assumed that all four RHR pumps are running in the short-term

phase of the RSLB DBA-LOCA. The 3952 MWt RTP analysis also conservatively assumes all four RHR pumps are running in the short-term phase of the RSLB DBA-LOCA. Therefore, the issue identified in SC 06-01 is addressed in the 3952 MWt RTP analysis.

14.12.4.1 Suppression Pool Temperature Response – RSLB DBA-LOCA

The analysis of the RSLB DBA-LOCA was performed at 102% RTP (4031 MWt). The calculated suppression pool temperature response is presented in Figure 14.12-8, the drywell and wetwell temperature responses are presented in Figure 14.12-9, and the peak value for LOCA bulk pool temperature is shown in Table 14.12-6. The RTP analysis was performed using a decay heat table based on ANS/ANSI 5.1-1979 with 2-sigma adders with additional actinides and activation products per GE SIL 636 (Reference 11). No modifications were made to this standard.

The containment system response to the accident is divided into two analysis phases. The first phase, hereafter referred to as the short-term phase covers the period up to 10 minutes after the accident initiation. During the short-term phase, no operator action is credited in the analysis. The second phase, hereafter referred to as the long-term phase covers the period after 10 minutes following the accident initiation. During the long-term phase, operator actions such as those to reduce electrical loading on the emergency diesel generators and to re-align portions of the ECCS from core cooling mode to containment cooling mode are credited. The RSLB DBA-LOCA analysis assumes that offsite power is lost concurrently with the accident initiation and that offsite power is not available during the accident analysis period. Separate RSLB analysis cases are run for EPU with initial conditions to either maximize or minimize the containment drywell and wetwell pressure response while maximizing the suppression pool temperature response in order to determine the sensitivity of the peak suppression pool temperature response to perturbed initial conditions. No containment leakage is assumed except for the RSLB cases with initial conditions to minimize the containment drywell and wetwell pressure response while maximizing the suppression pool temperature response, for which containment leakage (2% per day) and the leakage from MSIVs (150 scfh for all steam lines) are considered. In addition, the containment responses to various modes of containment cooling are evaluated. These three RHR cooling modes are: (1) Coolant Injection Cooling (CIC), where RHR flow is cooled by the RHR heat exchanger before being discharged into the reactor vessel; (2) Containment Spray Cooling (CSC), where RHR flow is cooled by the RHR heat exchanger and then discharged to the containment via the DW spray and wetwell spray headers; and (3) Suppression Pool Cooling (SPC), where RHR flow is cooled by the RHR heat exchanger and then discharged back to the suppression pool.

A complete Loss of Offsite Power (LOOP) is assumed to occur concurrent with the accident initiation. If a worst-case SAF such as failure of one emergency electrical power source (emergency diesel generator or loss of a 4KV shutdown board) is assumed concurrent with the accident, then less than the full complement of low pressure ECCS pumps (four RHR pumps and four CS pumps) would be available during the short-term phase of the accident. However, if no SAF is assumed, then the full complement of ECCS pumps would be available. The initial condition of no SAF during the short-term phase is limiting for the determination of ECCS pump NPSH during the accident because of the Browns Ferry ECCS pump suction configuration where each ECCS pump does not have a dedicated ECCS suction strainer and piping suction directly from the suppression pool (torus). For each Browns Ferry unit, there are four ECCS suction strainers installed in the torus. The torus water volume then communicates to the ECCS pump suctions via a torus ring header located below the torus. This configuration result in higher ECCS piping head loss when there are multiple ECCS pumps running. In addition, a larger number of running ECCS pumps will lead to higher pump heat addition to the suppression pool. Conformance with GEH SC 06-01 is made by assuming all low pressure ECCS pumps start during the short-term phase of the accident.

All ECCS pumps are assumed to be available for the first 600 seconds after accident initiation. No RHRSW flow is assumed to the RHR heat exchangers and there is no heat removal from the RHR heat exchangers during the short-term phase. RPV liquid is discharged from the break into the drywell causing rapid vessel depressurization and a rapid increase in the drywell pressure and temperature. For the first 600 seconds following the accident, four RHR pumps in LPCI mode (with two RHR pumps injecting liquid into the intact recirculation loop at a flow rate of 9,000 gpm per RHR pump and the other two RHR pumps into the broken recirculation loop at a flow rate of 9,000 gpm per RHR pump) and four CS pumps, each with flow rate of 3,550 gpm, are used to cool the core. For the RSLB DBA-LOCA, the RHR flow into the broken recirculation loop will be directed to the RPV and RHR flow will not go into runout flow because the RHR injection point is between the RPV and the closed reactor recirculation discharge valve (the reactor recirculation discharge valve in each reactor recirculation loop receives an automatic closure signal during a LOCA). HPCI is assumed available and will start on either high DW pressure or low RPV level. However, HPCI will isolate on low steam pressure. The ECCS injection of suppression pool water, along with the assumed addition of feedwater produces a recovery of the reactor water level. This allows water heated by decay heat and vessel sensible energy to be discharged into the drywell, and subsequently into the suppression pool.

If the accident were to occur on either Unit 1 or 2 and a worst-case SAF such as failure of one emergency electrical power source (emergency diesel generator or loss of a 4KV shutdown board) is assumed concurrent with the accident, then less than the full complement of low pressure ECCS pumps (four RHR pumps and four CS

pumps) would be available during the long-term phase of the accident. Assuming that one RHR pump is required for shutdown of the non-accident unit, only two RHR pumps and two RHR heat exchangers are assumed available for long-term containment cooling in the accident unit.

After 600 seconds, operator actions are credited. One loop of CS with two CS pumps continues to be available for RPV water makeup. One loop of CS with two pumps is secured because two CS pumps can supply adequate long-term core cooling after accident initiation. The CS pump flow rate for the remaining CS loop is assumed in the analysis as 3,125 gpm per loop. The throttling of CS flow is not a new operator action for EPU. One loop of RHR with two pumps is secured, and another loop of RHR with two pumps is switched to a RHR mode of containment cooling with its associated RHRSW flow activated for two heat exchangers. A conservatively low RHRSW flow value of 4,000 gpm to each in-service RHR heat exchanger is assumed in the analysis. The analysis assumes operator action to throttle RHR flow to 6,500 gpm per RHR pump. Three RHR cooling modes are investigated: (1) Coolant Injection Cooling (CIC) where RHR in LPCI mode with flow from the suppression pool is cooled by the RHR heat exchanger before being discharged into the reactor vessel; (2) Containment Spray Cooling (CSC) where RHR flow from the suppression pool is discharged as drywell and wetwell sprays; and (3) Suppression Pool Cooling (SPC) where the RHR flow from the suppression pool is cooled by the RHR heat exchanger before being discharged back into the suppression pool. The heat exchanger K-value and RHR pump flow rate are presented in Table 14.12-1. Initial conditions (initial DW pressure, initial wetwell pressure and initial DW temperature) are also perturbed in separate analysis cases to both maximize and minimize the peak containment pressure and thereby investigate the effect on peak suppression pool temperature. The resulting calculated peak bulk SP temperature for RSLB DBA-LOCA at 10 minutes after the accident initiation is 152.8°F and the peak bulk SP temperature for RSLB DBA-LOCA is 179.0°F.

The possible effect of containment cooling interruption on the accident unit due to concurrent shutdown and cooldown of the non-accident units was also investigated. Prior to depressurizing the non-accident unit below the RPV pressure that would result in initiating a LOCA signal due to high drywell pressure (if high DW pressure exists) in conjunction with low RPV pressure, the operators recognize that a LOCA signal could occur and therefore inhibit the generation of a LOCA signal from the non-accident unit. This LOCA signal inhibit is performed by the operation of permanently installed hand switches in the Browns Ferry main control room and is a normal action per Browns Ferry procedures to prevent the generation of a false LOCA signal that could result in interruption of containment cooling on both the accident and the non-accident unit. Therefore, there are no additional containment cooling interruptions on the accident unit due to interaction from the non-accident units.

14.12.4.2 Suppression Pool Temperature Response – Recirculation Discharge Line Break

The containment response during the first 10 minutes following the accident initiation for a RDLB was calculated using Browns Ferry specific inputs to maximize suppression pool temperature and minimize containment pressure, similar to the RSLB DBA-LOCA analysis. The key parameter differences between the RDLB and the RSLB during the short-term phase of the accident are: 1) the break area (4.2 ft2 for the RSLB versus 1.94 ft2 for the RDLB), and 2) the RHR flow rate and RHR injection path into the broken recirculation loop.

For the RDLB LOCA, all ECCS pumps are assumed to be available for the first 600 seconds. No RHRSW flow is assumed to the RHR heat exchangers and there is no heat removal from the RHR heat exchangers during the short-term phase. RPV liquid is discharged from the break into the drywell causing rapid vessel depressurization and a rapid increase in the drywell pressure and temperature. For the first 600 seconds following the accident, four RHR pumps in LPCI mode (with two RHR pumps injecting liquid into the intact recirculation loop at a flow rate of 9,000 gpm per RHR pump and the other two RHR pumps into the broken recirculation loop at a flow rate of 11,000 gpm per RHR pump) and four CS pumps, each with flow rate of 3,550 gpm, are used to cool the core. For the RDLB LOCA, the RHR flow into the broken recirculation loop discharges directly to the drywell and the RHR flow into the broken loop is assumed at runout conditions. HPCI is assumed available and will start on either high DW pressure or low RPV level. However, HPCI will isolate on low steam pressure. The ECCS injection of suppression pool water, along with the assumed addition of feedwater produces a recovery of the reactor water level. This allows water heated by decay heat and vessel sensible energy to be discharged into the drywell, and subsequently into the suppression pool. The resulting calculated peak bulk SP temperature for the RDLB at 10 minutes after the accident initiation is 152.0°F.

14.12.4.3 Suppression Pool Temperature Response – Small Steam Break LOCA

For the Browns Ferry small break LOCA, a spectrum of small steam line breaks was evaluated. Initial reactor conditions are consistent with operation at 102% of 3952 MWt RTP, and the same decay heat, relaxation and metal-water reaction energies are assumed as is used for the large DBA-LOCA analysis. Consistent with the large DBA-LOCA assumptions, a complete LOOP is assumed. A worst-case single failure is also assumed for this analysis to minimize the available quantity of containment cooling. This single failure is either the failure to start an EDG or the loss of a 4KV shutdown board. The single failure assumption will result in no more than three CS pumps and three RHR pumps automatically starting on either low RPV level or High DW Pressure (HDWP) concurrent with Low RPV Pressure (LRPVP). For cases

where HPCI is assumed available, HPCI will automatically start on either HDWP or on low RPV level.

Cases with HPCI (high pressure ECCS) available and with no HPCI available are evaluated to determine the effect of the availability of high pressure ECCS on the limiting peak pool temperature and the limiting drywell temperature. The condensate storage tank is assumed unavailable during the accident and HPCI pump suction is assumed available only from the suppression pool. For Browns Ferry, HPCI is qualified only for water temperatures up to 140°F. If HPCI is conservatively assumed available, HPCI will provide reactor inventory makeup until the reactor pressure decreases below the HPCI isolation pressure, after which low-pressure ECCS provides reactor inventory makeup. If HPCI is not available, ADS would be used to rapidly reduce reactor pressure to allow low-pressure ECCS to provide vessel makeup. Such use of ADS results in a faster heatup of the suppression pool. With reactor pressure at the time of peak pool temperature the same, the total (integrated) sensible heat addition to the suppression pool remains the same, but the total (integrated) decay heat to the pool at the time of peak suppression pool temperature is less for the fast pool heatup. In addition, the heat removed from the pool is greater for the faster pool heatup. Thus, a faster pool heatup will result in a lower peak suppression pool temperature. For this reason, the assumption of crediting the HPCI as available until it isolates on low steam pressure is conservative for the determination of a peak suppression pool temperature response.

Automatic starting of ECCS pumps will occur in accordance with their start logic and timing for electrical loading. Automatic start of CS and RHR will result in reactor vessel inventory makeup provided by three CS pumps and three RHR pumps in LPCI mode. Operators initiate depressurization of the RPV at 100°F/ hour when the suppression pool temperature reaches 120°F. At no sooner than 10 minutes after the start of the accident, operators will stop the third RHR pump and third CS pump. When containment conditions permit (drywell and wetwell pressures and drywell temperatures), operators will either re-align or start two RHR pumps in containment spray mode (two RHR pumps at 6,500 gpm each with two RHR heat exchangers with a K-factor of 265 BTU/sec-ºF per heat exchanger) and one CS loop (two CS pumps with maximum flow of 3,125 gpm/pump). For breaks greater than 0.01 ft^2, drywell and wetwell spray initiation is assumed delayed by up to 1,200 seconds (20 minutes) to address concerns related to ECCS interruption caused by a subsequent LOCA signal activated on HDWP concurrent with LRPVP. For the smallest break (0.01 ft^2), the late LOCA signal will occur much later in the event and the operator would inhibit the late LOCA signal and the additional drywell spray delay will not occur. This LOCA signal inhibit is performed by the operation of permanently installed hand switches in the Browns Ferry main control room which is a normal action per Browns Ferry procedures to prevent the generation of a LOCA signal that could result in interruption of containment cooling when core cooling has already

been confirmed. RPV depressurization is terminated when RPV pressure reaches 50 psig. Because Browns Ferry is a hot shutdown plant, entry into alternate shutdown cooling for entering cold shutdown is not required. Operators will maintain the plant at this pressure until Normal Shutdown Cooling (NSDC) can be restored. The resulting calculated peak bulk SP temperature for a steam line break is 182.7°F, which occurs for the 0.01 ft^2 break. Figure 14.12-10 shows the suppression pool temperature response for the limiting break size of 0.01 ft^2.

The peak suppression pool temperature of 182.7°F is for the case where HPCI is assumed available and the initial DW temperature is 70°F. The sensitivity of this peak suppression pool temperature due to initial DW temperature was investigated by setting the initial DW temperature to 150°F. The resulting calculated peak bulk SP temperature for a 0.01 ft^2 steam line break with initial DW temperature of 150°F is 182.7°F, which demonstrates the insensitivity of initial DW temperature on the peak suppression pool temperature. The peak SP temperature for the limiting 0.01 ft^2 steam line break where HPCI was assumed unavailable was 181.5°F, which demonstrates that the assumption of HPCI availability during the event is conservative.

The possible effect of containment cooling interruption on the accident unit due to concurrent shutdown and cooldown of the non-accident units was also investigated. Prior to depressurizing the non-accident unit below the RPV pressure that would result in initiating a LOCA signal due to high drywell pressure (if high DW pressure exists) in conjunction with low RPV pressure, the operators recognize that a LOCA signal could occur and therefore inhibit the generation of a LOCA signal from the non-accident unit. This LOCA signal inhibit is performed by the operation of permanently installed hand switches in the Browns Ferry main control room which is a normal action per Browns Ferry procedures to prevent the generation of a false LOCA signal that could result in interruption of containment cooling on both the accident and the non-accident units. Therefore, there are no additional containment cooling interruptions on the accident unit due to interaction from the non-accident units.

The peak suppression pool temperature for small liquid line breaks is bounded by the containment analysis results for small steam line breaks. Because the break flow is an isenthalpic process, the steam line breaks result in superheated steam discharged to the drywell atmosphere, while the liquid line breaks result in two-phase break flow into the drywell atmosphere. Therefore, the drywell temperature response for the small liquid line breaks is less severe than for the small steam line breaks. The emergency operating instruction entry point for drywell spray is either not reached for the liquid line breaks or the duration of drywell spray operation is much less for the liquid line breaks than for the steam line breaks. Because the non-operation or limited operation of drywell sprays for the liquid line breaks allows a greater holdup of the sensible heat in the drywell, less heat is discharged to the

suppression pool for the liquid line breaks and the peak suppression pool temperature is consequently less for a given liquid line break size than for the same size steam line break.

14.12.4.4 Suppression Pool Temperature Response – Non-Accident Units

Evaluation of the containment response for the non-accident units was also evaluated based on the conservative assumption that the RSLB DBA-LOCA occurs concurrently with a LOOP, resulting in reactor isolation and scram on the non-accident unit. The term “non-accident unit” refers to the Browns Ferry unit that is both not experiencing a LOCA and has the minimum containment cooling equipment available. This evaluation is applicable to either of the following conditions: 1) a LOOP for all three Browns Ferry units (with no LOCA), or 2) a LOCA on any one unit concurrent with a simultaneous LOOP for the remaining two units. The bounding condition has been evaluated for two scenarios that either assume the CST is available or assume the CST is not available.

For Units 1 and 2, there are four 4kV shutdown boards (4kV shutdown board A, B, C and D) shared between the two units. Each 4kV shutdown board is supplied during a LOOP by a safety-related emergency diesel generator (EDG). Power distribution to 480V shutdown boards and 480V Reactor Motor Operated Valve (RMOV) boards is redundant in that each 480V board can be supplied power from two of the Unit 1 and 2 shared 4kV shutdown boards. For Unit 3, there are four dedicated 4kV shutdown boards (4kV shutdown board 3EA, 3EB, 3EC and 3ED). Each 4kV shutdown board for Unit 3 is supplied during a LOOP by a safety-related EDG. Power distribution to Unit 3 480V shutdown boards and 480V RMOV boards is redundant in that each 480V board can be supplied power from two of the 4kV shutdown boards designated for Unit 3. In addition, the 4kV electrical distribution system at Browns Ferry allows a Unit 3 EDG to either power a de-energized Unit 1 and 2 4kV shutdown board or to operate in parallel with a Unit 1 and 2 EDG for powering a Unit 1 and 2 4kV shutdown board.

Conservatively assuming that the DBA-LOCA occurs concurrently with a LOOP, reactor isolation and scram will occur on the non-accident units. Concurrent with the LOOP, the worst case single failure for containment cooling is the loss of a 4kV shutdown board (A, B, C or D) shared between Units 1 and 2. This single failure is more severe than loss of an EDG alone because it prevents repowering the lost (de-energized) 4kV shutdown board from one of the Unit 3 EDGs. For this assumed electrical power failure, only three RHR pumps would be available for either core or containment cooling between Units 1 and 2. The LOCA analysis assumes that two of these RHR pumps would be used for long-term containment cooling in the accident unit.

Paralleling of a Unit 3 EDG with the EDG supplying power to the non-accident unit (so that two EDGs are supplying power to one Unit 1 and 2 4kV shutdown board) is not assumed. Therefore, EDG power limitations are conservatively assumed that allow the starting and alignment of only one RHR pump and one RHR heat exchanger for containment cooling on the non-accident unit.

The loss of the 4kV shutdown board may also result in loss of the normally aligned power to the 480V shutdown board and the 480V RMOV board that supplies power to the RHR Shutdown Cooling (SDC) isolation valves for the non-accident unit. However, the Browns Ferry electrical system configuration is such that there are redundant means of re-powering the 480V shutdown boards and the 480V RMOV boards that supply power to both additional DW coolers and the RHR SDC isolation valves. Therefore, there is no loss of the ability to place RHR SDC into service due to electrical power limitations. Drywell cooling is initially lost for the non-accident unit because the LOOP signal in conjunction with a LOCA signal on the accident unit causes the loads to be stripped from the 4kV shutdown boards and then re-sequenced on as the EDGs repower the 4kV shutdown boards. Within 90 seconds after the LOOP, a minimum of four drywell coolers are automatically re-started and are available for DW cooling in the non-accident unit. Operators are able to manually restart DW coolers and restore power to RHR SDC isolation valves later in the event using the redundant power sources mentioned above.

The capability of the non-accident unit to achieve cold shutdown was analyzed at 102% of 3952 MWt RTP and ANS/ANSI 5.1-1979 with 2-sigma adders decay heat. The decay heat model includes additional actinides and activation products per GE SIL 636. The analysis includes the assumption of reactor shutdown initiated by a loss of offsite power (for all three Browns Ferry units) with concurrent loss of a 4kV shutdown board shared between Units 1 and 2. Two scenarios are evaluated. Scenario 1 assumes that the Condensate Storage Tank (CST) volume is available and HPCI provides high pressure inventory makeup to the RPV with HPCI pump suction from the CST. Scenario 2 assumes that the CST volume is not available and HPCI provides high pressure inventory makeup to the RPV with HPCI pump suction from the suppression pool. Initial conditions and key input parameters for the non-accident unit containment response evaluation are shown in Table 14.12-2.

Scenario 1 - CST Available

The event is initiated by LOOP. The LOOP causes a reactor scram, a containment isolation signal due to loss of power to the Nuclear Steam Supply System (NSSS) isolation relays, tripping of the feedwater pumps and loss of power to the drywell coolers. The MSIVs are assumed to be fully closed at 3.5 seconds after event initiation. In the analysis, the FW temperature is initially at or above 337°F (saturation temperature is at 100 psig). Following the closure of the MSIVs, the

feedwater is assumed to flash to steam and then is injected into the reactor vessel. The feedwater mass entering the vessel after closure of the MSIVs is conservatively assumed to come into thermal equilibrium with the downstream feedwater piping as the feedwater travels toward the vessel. Feedwater injection into the vessel is assumed to resume when the RPV pressure is reduced to below 220 psig which ensures that all hot feedwater at a temperature equal to and greater than 337ºF is injected into the vessel before the suppression pool temperature peaks. This assumption is conservative because the timing results in the FW enthalpy addition occurring late in the event when the suppression pool temperature is high and will therefore result in a more conservative (higher) suppression pool temperature response.

The MSRVs will automatically cycle to control RPV pressure. At 90 seconds into the event, four drywell coolers will have automatically restarted. The HPCI pump will automatically start on low RPV level with HPCI pump suction from the CST. At ten minutes after reactor shutdown, the operators align one loop of RHR (one RHR pump, one RHR heat exchanger and RHRSW cooling flow of 4500 gpm to the RHR heat exchanger) in suppression pool cooling mode with a flow rate of 9700 gpm. At approximately 20 minutes after the start of the event, operators are assumed to restart an additional four drywell coolers to provide additional cooling to the non-accident unit drywell and restore power to NSSS isolation relays. When the non-accident unit suppression pool temperature reaches 110°F, but no sooner than ten minutes after reactor shutdown, the operators commence manual reactor depressurization and reactor cooldown at a rate of 100°F/hr. Prior to depressurizing the non-accident unit below the RPV pressure that would result in initiating a LOCA signal due to high drywell pressure (if high drywell pressure exists) in conjunction with low RPV pressure, the operators recognize that a LOCA signal could occur and therefore inhibit the generation of a LOCA signal from the non-accident unit. This LOCA signal inhibit is performed by the operation of permanently installed hand switches in the Browns Ferry main control room and is a normal action per Browns Ferry procedures to prevent the generation of a false LOCA signal that could result in interruption of containment cooling on the non-accident unit. HPCI is assumed to isolate on low RPV pressure when RPV pressure decreases to 150 psig. A single core spray pump is started to provide RPV inventory makeup after HPCI is no longer available. Further depressurization of the RPV to 100 psig is accomplished by opening MSRVs.

When RPV pressure reaches 100 psig, the analysis assumes that the operators will maintain the RPV at this pressure. Operators stop the RHR pump in suppression pool cooling and begin transitioning RHR to shutdown cooling mode. The transition to place shutdown cooling in operation is assumed to take 20 minutes. During this 20 minute transition period from RHR in suppression pool cooling to shutdown cooling there is no cooling of the suppression pool. Cooldown of the RPV to cold

shutdown conditions on the non-accident unit is accomplished with SDC. Cold shutdown is achieved when bulk reactor liquid water temperature is less than or equal to 212°F. The peak bulk suppression pool cooling temperature is 185.1°F. The resulting time-dependent bulk suppression pool temperature response is presented in Figure 14.12-11.

Scenario 2 - CST Not Available

The event is initiated by LOOP. The LOOP causes a reactor scram, a containment isolation signal due to loss of power to the NSSS isolation relays, tripping of the feedwater pumps and loss of power to the drywell coolers. The MSIVs are assumed to be fully closed at 3.5 seconds after event initiation. In the analysis, the feedwater temperature is initially at or above 337°F (saturation temperature is at 100 psig). Following the closure of the MSIVs, feedwater is assumed to flash to steam and then is injected into the vessel. The feedwater mass entering the RPV after closure of the MSIVs is conservatively assumed to come into thermal equilibrium with the downstream feedwater piping as the feedwater travels toward the reactor vessel. Feedwater injection into the vessel is assumed to resume when the RPV pressure is reduced to below 220 psig which ensures that all hot FW at a temperature equal to and greater than 337ºF is injected into the vessel before the suppression pool temperature peaks. This assumption is conservative because the timing results in the FW enthalpy addition occurring late in the event when the suppression pool temperature is high and will therefore result in a more conservative (higher) suppression pool temperature response.

The MSRVs will automatically cycle to control RPV pressure. At 90 seconds into the event, four drywell coolers will have automatically restarted. The HPCI pump will automatically start on low RPV level with HPCI pump suction from the suppression pool. The CST volume is assumed to not be available, consistent with the assumptions used for the containment system response for a LOCA. HPCI provides reactor inventory makeup until suppression pool temperature reaches 140°F. If the suppression pool temperature reaches 140°F, HPCI is secured because HPCI availability cannot be assured with a SP temperature greater than 140°F. At ten minutes after reactor shutdown, the operators align one loop of RHR (one RHR pump, one RHR heat exchanger and RHRSW cooling flow of 4500 gpm to the RHR heat exchanger) in suppression pool cooling mode with a flow rate of 9700 gpm. At approximately 20 minutes after the start of the event, operators are assumed to restart an additional four drywell coolers to provide additional cooling to the non-accident unit drywell and restore power to NSSS isolation relays. When the non-accident unit suppression pool temperature reaches 110°F, but no sooner than ten minutes after reactor shutdown, the operators commence manual reactor depressurization and reactor cooldown at a rate of 100°F/hr. Prior to depressurizing the non-accident unit below the RPV pressure that would result in initiating a LOCA

signal due to high drywell pressure (if high drywell pressure exists) in conjunction with low RPV pressure, the operators recognize that a LOCA signal could occur and therefore inhibit the generation of a LOCA signal from the non-accident unit. This LOCA signal inhibit is performed by the operation of permanently installed hand switches in the Browns Ferry main control room and is a normal action per Browns Ferry procedures to prevent the generation of a false LOCA signal that could result in interruption of containment cooling on the non-accident unit. When the SP temperature reaches 140°F, the analysis assumes that HPCI is secured. A single core spray pump is started to provide RPV inventory makeup after HPCI is no longer available. Further depressurization of the RPV to 100 psig is accomplished by opening MSRVs.

When the RPV pressure reaches 100 psig, the analysis assumes that the operators will maintain the RPV at this pressure. Operators stop the RHR pump in suppression pool cooling and begin transitioning RHR to SDC mode. The transition to place SDC in operation is assumed to take 20 minutes (Browns Ferry operators confirmed that this assumed operator action time can be achieved). During this 20 minute transition period from RHR in suppression pool cooling to SDC, there is no forced cooling of the suppression pool from RHR. Cooldown of the RPV to cold shutdown conditions on the non-accident unit is accomplished with SDC. Cold shutdown is achieved when bulk reactor liquid water temperature is less than or equal to 212°F. The peak bulk suppression pool cooling temperature is 180.0°F.

14.12.5 Long-Term Bulk Suppression Pool Temperature Response-Special Events

14.12.5.1 Station Blackout

The containment response to a station blackout (SBO) was evaluated at 100% RTP (3952 MWt). The evaluation used the NRC approved method of NUMARC 87-00 (Reference 12) and NRC Regulatory Guide 1.155 (Reference 13). BFN is evaluated as an “Alternate AC Approach” plant per Reference 12. Alternate AC approach would entail a short period of time in an AC Independent state (up to one hour) while operators initiate power from the backup source. The BFN SBO assessment assumes only one unit in station blackout with the other two units available to supply Alternate AC to blacked-out unit. Use of Alternate AC power is limited to providing the required cooling systems to certain areas (control room, control bay, and electrical board rooms). The SBO coping duration for BFN is four hours. The containment response was analyzed using the SHEX analysis code as was utilized for the design basis accident responses. In accordance with Reference 12 and 13 methodologies, the containment response was evaluated with consideration of both zero reactor coolant system leakage into the primary containment and 61 gpm

reactor coolant system leakage into the primary containment. Conservative Technical Specification limits were used as inputs to the containment response analyses. The specific values are shown in Table 14.12-3.

The SBO scenario is based on a LOOP which causes turbine trip and reactor scram on all three BFN Units. One BFN unit suffers a SBO and HPCI and RCIC are the only available sources of makeup to maintain reactor water level on the SBO unit. HPCI and RCIC start on low reactor level and take suction from the condensate storage tank (CST). Automatic and manual actuation of safety relief valves (MSRVs) provides reactor pressure control. Operators begin manual control of MSRVs to control pressure and cooldown rate at approximately 20 minutes into the SBO event. One hour after the SBO event initiation, alternate AC is available from non - SBO units to provide HVAC loads (control room, control bay and electrical board rooms) for the remainder of the SBO coping duration. The SBO coping duration ends at four hours and containment cooling capability is restored. The peak bulk suppression pool temperature for this analysis at 100% RTP conditions is 203.7°F and the peak drywell pressure is 43.4 psia. The resulting time-dependent bulk suppression pool temperature response is presented in Figure 14.12-12.

In addition to the containment response, CST inventory, battery capacity, compressed gas capacity, the effects of loss of ventilation, and containment isolation capability were evaluated to verify the response to a SBO.

Condensate Inventory for Decay Heat Removal

Analyses have shown that the BFN condensate inventory is adequate to meet the SBO coping requirement for RTP conditions. The current CST inventory reserve for RCIC and HPCI use ensures that adequate water volume is available to remove decay heat, depressurize the reactor and maintain reactor vessel level above the top of active fuel (TAF) during the coping period.

Class 1E Battery Capacity

Evaluation of the BFN Class 1E Battery Capacity has shown that BFN has adequate battery capacity to support decay heat removal during a SBO for the required coping duration. The battery capacity remains adequate to support required coping equipment operation at RTP.

Compressed Gas Capacity

BFN meets the requirement for compressed gas capacity. An evaluation has shown that the BFN air operated MSRVs required for decay heat removal have sufficient compressed gas capacity for the required automatic and manual operation during the

SBO event at RTP conditions. Sufficient capacity remains to perform emergency RPV depressurization in case it is required. Adequate compressed gas capacity exists to support the MSRV actuations because the maximum number of MSRV valve operations is less than the capacity of the pneumatic supply.

Effects of Loss of Ventilation

Areas containing equipment necessary to cope with an SBO event were evaluated for the effect of loss of ventilation due to an SBO. The evaluation shows that equipment operability is maintained because the SBO environment is milder than the existing design and qualification bases.

Containment Isolation

The containment isolation capability was shown to be within the existing design and qualification basis.

14.12.5.2 Fire Event

The limiting NFPA 805 fire events were analyzed at RTP conditions. The fuel heat-up analysis was performed using the NRC approved, AREVA LOCA Methodology (RELAX/HUXY). The containment analysis was performed using the GEH SHEX model. These analyses determined the affect of EPU on fuel cladding integrity and containment integrity as a result of the fire event. The two bounding cases described below are identified as “Case 1” and “Case 4.” Key inputs to the fire event analyses are shown in Table 14.12-4.

Case 1 : The bounding safe shutdown case for PCT has Multiple Spurious Operation (MSO) of 11 of the 13 MSRVs, which depressurize the reactor, and one RHR pump aligned in the LPCI/ASDC mode at 20 minutes, The analysis shows that the calculated 1330°F PCT is acceptable from a deterministic perspective (<1500°F).

Case 4: The bounding safe shutdown case for peak suppression pool temperature has reactor depressurization beginning at 25 minutes using three MSRVs. As the reactor is depressurized, condensate pumps replenish reactor inventory until hotwell inventory is depleted. After condensate is secured, one RHR pump is aligned into LPCI/ASDC mode. One RHRSW pump is initiated at 2 hours. Peak SP temperature reaches 207.7°F and this meets the containment integrity acceptance criteria of <281°F. The peak suppression pool temperature is less than qualification temperature of the torus attached piping. The resulting time-dependent bulk suppression pool temperature response is presented in Figure 14.12-13.

14.12.5.3 Anticipated Transients Without Scram (ATWS)

The BFN ATWS 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, Main Steam Isolation Valve Closure (MSIVC), Pressure Regulator Failure – Open (PRFO), Loss of Off-Site Power (LOOP), and Inadvertent Opening of a Relief Valve (IORV). Consistent with the event selection disposition contained in Section L.3.3 of Reference 1 (ELTR-1), these four events were analyzed at RTP for the containment system response (suppression pool temperature and containment pressure) from ATWS.

The EPU ATWS analyses for the containment response were performed using the NRC approved code ODYN, to determine the heat addition to the suppression pool from MSRV flow, and STEMP, to determine the suppression pool heatup due to energy input from the MSRVs. The key inputs and limiting results for the containment response to ATWS events are presented in Tables 14.12-5 and 14.12-6. The limiting ATWS event with respect to containment response for Browns Ferry is LOOP where the peak suppression pool temperature is 173.3°F. The resulting time-dependent bulk suppression pool temperature response is presented in Figure 14.12-14. The limiting ATWS events with respect to ECCS pump net positive suction head are the MSIVC and PRFO events where the peak suppression pool temperature is 171.8°F. The peak suppression pool temperature and containment pressure results are well below the containment design temperature and pressure. Therefore, the Browns Ferry EPU ATWS analysis for the containment response complies with the acceptance criteria of 10CFR50.62.

REFERENCES

1 NEDC-32424P-A, “Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate,” February 1999 (ELTR 1).

2 GE Nuclear Energy, NEDO-10320P, “The General Electric Pressure Suppression Containment Analytical Model,” April 1971.

3 GE Nuclear Energy, NEDO-20533, “The General Electric Mark III Pressure Suppression Containment System Analytical Model,” June 1974, and NEDO-20533, Supplement 1, September 1975.

4 GE Nuclear Energy, NEDE-20566-P-A, “General Electric Model for LOCA Analysis in Accordance with 10CFR50 Appendix K,” September 1986.

5 Ashok Thadani (NRC) to Gary L. Sozzi (GE), “Use of the SHEX Computer Program and ANSI/ANS 5.1-1979 Decay Heat Source Term for Containment Long-Term Pressure and Temperature Analysis.”

6 NEDC-33860P, “Safety Analysis Report for Browns Ferry Nuclear Plant Units 1, 2, and 3 Extended Power Uprate,” Rev.0, September 2015.

7 Regulatory Guide 1.7, “Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident,” Revision 2, November 1978.

8 GE Nuclear Energy, NEDC-32751P, “Power Uprate Safety Analysis for the Browns Ferry Nuclear Plant Units 2 and 3,” September 1997.

9 NUREG-0588,”Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment,” Revision 1, July 1981.

10 GE Nuclear Energy Safety Communication, SC06-01, “Worst Single Failure for Suppression Pool Temperature Analysis,” January 19, 2006.

11 GE Nuclear Energy Service Information Letter No. 636, “Additional Terms Included in Reactor Decay Heat Calculations,” Revision 1, June 2001.

12 “Guidelines and Technical bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors”, MUMARC 87-00 Rev.1 August 1988.

13 Regulatory Guide 1.155, “Station Blackout,” August 1988.

14 NEDC-32523P-A, “Generic Evaluation of General Electric Boiling Water Reactor Extended Power Uprate,” February 2000 (ELTR 2).

Table 14.12-1
DBA Containment Response Key Analysis Input Values
(Page 1 of 2)

Parameter	Unit	Value
Reactor		
Initial Power Level (102% RTP)	MWt	4,031
Normal Feedwater Temperature at 102% of 3952 MWt	°F	394.5
Reduced Feedwater Temperature at 102% of 3952 MWt	°F	339.8
RPV Dome Pressure	psia	1,055
Decay Heat Model- Short Term DBA LOCA		ANS 5 1971+20%
Decay Heat Model-Long Term		1979 ANS 5.1+2σ
RPV Free Volume	ft^3	20,682
RPV Liquid Volume - subcooled	ft^3	11,790
RPV Liquid Volume - saturated	ft^3	3,864
RPV Related Masses for Long Term Calculation		
Liquid mass in recirculation loops	lbm	63,560
Liquid mass in the HPCI piping between the RPV nozzle and first normally closed valve	lbm	8,621
Liquid mass in the RCIC piping between the RPV nozzle and first normally closed valve	lbm	1,245
Liquid mass in the RHR piping between the RPV and the first normally closed valve	lbm	9,535
Liquid mass in the core spray piping between the RPV nozzle and the first normal closed valve.	lbm	2,622
MSIV closure initiation	second	0.5
MSIV full closure	second	3.5
Drywell Vent System		
Drywell free volume (including vent system)	ft^3	159,000 to 171,000 (Note 1)
Initial drywell pressure	psia	15.1 to 17.0
Initial drywell Temperature (D=design, B=bounding, R=reference)	°F	70 (D) 130 (B) 150 (R)
Initial drywell relative humidity (range)	%	20 to 100
Downcomer submergence- Low water level	ft	2.92

Table 14.12-1
DBA Containment Response Key Analysis Input Values
(Page 2 of 2)

Parameter	Unit	Value
Downcomer submergence- High water level	ft	3.83
Loss coefficient for vent system including entrance and exit losses	real	5.32
Downcomer internal diameter	ft	1.958
Wetwell/Suppression Pool		
Initial Suppression pool volume including water in vents - low water level	ft^3	122,940
Initial Suppression pool volume including water in vents - high water level	ft^3	131,400
Initial pool temperature - maximum	°F	95
Wetwell air volume excluding volume occupied by the vent system - Low water level	ft^3	135,000 (Note 2)
Wetwell air volume excluding volume occupied by the vent system - High water level	ft^3	119,400
Initial wetwell/containment air space pressure (range)	psia	14.4 to 15.9
Initial wetwell/containment airspace temperature (maximum)	°F	95
Initial wetwell/containment air space relative humidity	%	100
RHR		
K value (single HX)	BTU/sec°F	265
RHR service water temperature	°F	95
Drywell spray flow rate (2 RHR pumps)	gpm	12,350
Wetwell spray flow rate (2 RHR pump)	gpm	650
RHR flow rate in SPC mode (2 RHR pumps)	gpm	13,000
Wetwell to Drywell Vacuum Breakers		
Pressure difference between wetwell and drywell for vacuum breakers to be fully open	psid	0.5 (Note 3)
Number of vacuum breaker assemblies		6
Flow area of each vacuum breaker assembly at which loss coefficient is given below	ft^2	1.41
Loss coefficient or each vacuum breaker assembly	Real	0.45

Notes:
(1) Vent thrust loads and LOCA analyses to minimize the containment pressure are calculated assuming a minimum DW volume of 159,000 ft^3.
(2) This value is used for containment long-term analyses.
(3) For LOCA analyses that minimize the containment pressure response, the pressure difference between the wetwell and drywell for the vacuum breakers to be fully open of 0.05 psid is conservatively used.

Table 14.12-2
Non-Accident Unit Containment Response Key Analysis Input Values
(Page 1 of 2)

Parameter	Unit	Value
Reactor		
Initial power level 102% RTP	MWt	4,031
Initial FW temperature at 102% RTP (102% of 3,951MWt)	°F	396.6
Initial vessel dome pressure at 102% RTP	psia	1,055
Decay heat model	N/A	1979 ANS 5.1 + 2σ
Vessel volumes		
Total vessel free volume	ft^3	20,682
Vessel liquid volume - subcooled	ft^3	7,926
Vessel liquid volume - saturated	ft^3	3,864
Vessel related masses		
Liquid mass in main steam lines to the inboard isolation valve	lbm	0
Liquid mass in one recirculation loop	lbm	31,780
Liquid mass in the HPCI piping between the RPV nozzle and first normally closed valve	lbm	8,621
Liquid mass in RHR/LPCI shutdown piping between the RPV nozzle and first normally closed valve	lbm	9,535
Liquid mass in the RCIC piping between the RPV nozzle and first normally closed valve	lbm	1,245
Liquid mass in the CS piping between the RPV nozzle and the first normal closed valve	lbm	2,622
MSIV Closure		
Time at which MSIVs start to close	sec	0.5
Time at which MSIVs become fully closed	sec	3.5
Drywell		
Total drywell airspace volume	ft^3	171,000
Initial drywell pressure	psia	15.5
Initial drywell temperature	°F	150
Initial drywell relative humidity	%	20
Wetwell/Suppression Pool		
Initial suppression pool volume low water level (LWL)	ft^3	122,940
Initial suppression pool temperature	°F	95
Initial wetwell airspace free volume - LWL in suppression pool	ft^3	135,000
Initial wetwell airspace pressure	psia	14.4
Initial wetwell airspace temperature	°F	95
Initial wetwell airspace relative humidity	%	100

Table 14.12-2
Non-Accident Unit Containment Response Key Analysis Input Values
(Page 2 of 2)

Parameter	Unit	Value
RHR		
Heat exchanger K-value	BTU/sec-°F	302
Service water temperature	°F	95
RHR flow rate in suppression pool cooling mode	gpm	9,700
RHR flow rate in SDC mode	gpm	9,700
Number of RHR loops for cooling (one RHR loop is one RHR pump and one RHR heat exchanger)	N/A	1
RHR Service Water		
RHRSW flowrate through one RHR heat exchanger	gpm	4,500
Condensate Storage Tank		
Condensate storage tank volume available for RPV inventory makeup	gallon	135,000
Condensate storage tank temperature	°F	130
Drywell Air Cooler		
Heat removal capability of each drywell air cooler	BTU/hour	636,000

Table 14.12-3
Station Blackout Containment Response Key Analysis Input Values

Parameter	Value
Initial Reactor Power	3,952 MWt
Initial Reactor Pressure	1,055 psia
Decay Heat	ANS/ANSI 5.1 1979 standard consistent with recommendations of GEH SIL 636
Initial Suppression Pool Temperature	95°F
Initial Suppression Pool Volume (LWL)	122,940 ft^3
Initial Wetwell Pressure	14.4 psia
Initial Drywell Temperature	150°F
Initial Drywell Pressure	15.5 psia
Initial Drywell free airspace volume	171,000 ft^3
Initial Wetwell free airspace volume	135,000 ft^3
Initial WW airspace temperature	95°F
CST Water Temperature	130°F
CST Inventory	135,000 gallons available
Initial Drywell Relative Humidity	20%
Initial Wetwell Relative Humidity	100%
RHR Heat exchanger K factor (per heat exchanger)	265 BTU/Sec-°F
RHR pump flow rate (per pump)	6,500 gpm
RHR service water flow rate to RHR heat exchangers	4,000 gpm
RHR service water temperature	95°F
Leakage rate from primary containment	2% of containment air mass per day
Containment heat sinks modeled	Yes

Table 14.12-4
Fire Event Containment Response Key Analysis Input Values

Input Parameters	Values
Reactor Thermal Power	3,952 MWt
RPV Dome Pressure	1,055 psia
Decay Heat	ANS 5.1-1979 without 2σ uncertainty adder and with GEH SIL 636 recommendations
Initial Suppression Pool Liquid Volume	122,940 ft^3 (Note 1)
Initial Suppression Pool and Wetwell Airspace Temperature	92 °F (Note 2)
Initial Wetwell Pressure	14.4 psia
Initial Drywell Pressure	15.5 psia
Initial Drywell Temperature	150 °F
Initial Wetwell Relative Humidity	100%
Initial Drywell Relative Humidity	20%
Drywell and Wetwell and Pool Heat Sinks Modeled	Yes
Drywell Heat Load Modeled	Yes
RHR Service Water Temperature	88 °F (Note 2)
RHR Heat Exchanger "K" Factor per Loop	290 Btu/sec-°F (Note 3)
Number of RHR Loops Available	1
Number of RHR Pump in one RHR Loop	1
ASDC RHR Flow Rate	7,500 gpm
Condensate available for injection	90,000 gallons

Notes

(1) Suppression Pool Volume corresponding to Browns Ferry Technical Specification low suppression pool water level with differential pressure control in service.

(2) Nominal values based on Browns Ferry plant data over seven year period from January 2008 through December 2014. Data analysis for this parameter shows that Browns Ferry operates at least 95% of time below this value.

(3) RHR heat exchanger "K" factor based on RHR flow of 7,500 gpm, RHRSW flow of 4,500 gpm, RHRSW temperature of 88°F and conservative RHR heat exchanger fouling resistance.

Table 14.12-5
ATWS Event Containment Response Key Analysis Input Values
(Page 1 of 2)

Input Variable	Value
Reactor power (MWt)	3,952
Analyzed Power (MWt)	3,952
Analyzed Core Flow (Mlbm/hr / % Rated)	101.475 / 99
Reactor dome pressure (psig)	1,035
MSIV Closure Time (seconds)	4.0
High pressure ATWS-RPT setpoint (psig)	1,177.0
MSL low pressure isolation setpoint (psig)	825
RCIC flow rate (gpm)	600
HPCI flow rate (gpm)	5,000
Number of MSRVs / MSRVs Out-of-service (OOS)	13 / 1
Number of MSRVs Out-of-service (OOS)	1
Each MSRV capacity at 1090 psig (Mlbm/hr)	0.87
MSRV Analytical Opening Setpoints (psig)	1,174 to 1,194 Note 1
SLCS Injection Location	Lower Plenum
SLCS Injection Rate (gpm)	50.0
Number of SLCS pumps credited for injection	1
Boron-10 Enrichment (Atom %)	94.0
Sodium Pentaborate Concentration (% by Weight)	8.7
SLCS Liquid Transport Time (seconds)	28.5
Initial Suppression Pool Liquid Volume (ft^3)	122,940 Note 7
Initial Suppression Pool Temperature (°F)	95
RHR Heat Exchanger Effectiveness Per Loop (BTU/sec-°F)	259 / 277 Note 2
Number of RHR Suppression Pool Cooling Loops (all events except Loss of Offsite Power Event)	4 Note 3
Number of RHR Suppression Pool Cooling Loops During a Loss of Offsite Power Event	2 Note 4

Table 14.12-5
ATWS Event Containment Response Key Analysis Input Values
(Page 2 of 2)

Input Variable	Value
RHR startup delay (seconds after T = 0)	660 Note 5
RHR Service Water Temperature (°F)	95
Decay Heat Correlation	May-Witt Note 6

Notes:

(1) In the ODYN analysis methodology, the MSRV setpoints for the ATWS analysis are statistically spread around the upper analytical limit.

(2) The heat exchanger effectiveness of 259 BTU/sec-°F assumes a RHR flow rate of 6500 gpm and RHR SW flow rate of 3800 gpm through each in-service RHR heat exchanger for events that assume operation of 4 RHR loops (see Note 3 below).
The EPU heat exchanger effectiveness of 277 BTU/sec-°F assumes a RHR flow rate of 6500 gpm and RHR SW flow rate of 4500 gpm through each in-service RHR heat exchanger for the event that assumes operation of 2 RHR loops (see Note 4 below).

(3) An RHR loop is defined as one RHR pump, one RHR heat exchanger and RHR SW flow of 3800 gpm through the RHR heat exchanger. For ATWS events other than LOOP, the plant operators would be directed by BFN Emergency Operating Instructions to maximize suppression pool cooling. Since there is no concurrent event on the non-ATWS unit, four RHR loops are assumed available for suppression pool cooling in the ATWS unit.

(4) An RHR loop is defined as one RHR pump, one RHR heat exchanger and RHR SW flow of 4500 gpm through the RHR heat exchanger. For the LOOP ATWS event, operators will be directed by BFN Emergency Operating Instructions to maximize suppression cooling. Since there is also a LOOP (without ATWS) on the remaining two Browns Ferry Units, only two RHR loops are assumed available for suppression pool cooling on the ATWS unit.

(5) The RHR startup delay time assumes no operator action for containment cooling for the first 10-minutes of the event with an additional 60 seconds for RHR to reach full effectiveness.

(6) The May-Witt decay heat correlation is used in the suppression pool temperature calculation following reactor shutdown. The May-Witt decay heat correlation yields a conservative pool heat-up compared to the 1979 ANS 5.1 + 2σ curve.

(7) The value of 122, 940 ft^3 for the initial Suppression Pool Liquid Volume does not include the volumes of the ECCS ring header and the RHR piping for each running RHR pump. These two volumes were included in the assumptions for the ATWS containment analysis and result in a total suppression pool liquid volume of 127,000 ft^3.

Table 14.12-6
Containment Response Results

Parameter	ST DBA LOCA	LT DBA LOCA	LT SSLB(2)	Non-Accident Unit	SBO (3)	Fire Event	ATWS
Peak Drywell Pressure (psig)	50.9 (D) 49.1 (B) 48.5 (R)	22.5	49.6 psia	N/A	43.4 psia	24.2 psia	8.7
Peak Drywell Airspace Temperature (°F)	297.5 (D) 295.8 (B) 295.2 (R)	287	336.9	N/A	275.2	276.3	N/A
Peak Bulk Suppression Pool Temperature (°F)	152.8(1)	179.0	182.7	185.1	203.7	207.7	173.3
Peak Torus (Wetwell) Pressure (psia)	N/A	30.2	48.0	N/A	N/A	24.6	N/A
Peak Torus (Wetwell) Air Temperature (°F)	N/A	174	NA	N/A	N/A	209.0	N/A

N/A denotes a noncritical parameter for this analysis.

Notes:

(1) Peak Suppression Pool Temperature for RSLB DBA-LOCA at 10 minutes after LOCA initiation. For the RDLB LOCA, the peak suppression pool temperature at 10 minutes after LOCA initiation is 152.0°F.

(2) Peak values from spectrum of SSLB analyses.

(3) Peak values from 0 gpm and 61 gpm reactor coolant system leakage cases

Figure 14.12-1

Short-Term RSLB DBA LOCA Containment Pressure Response Design Case

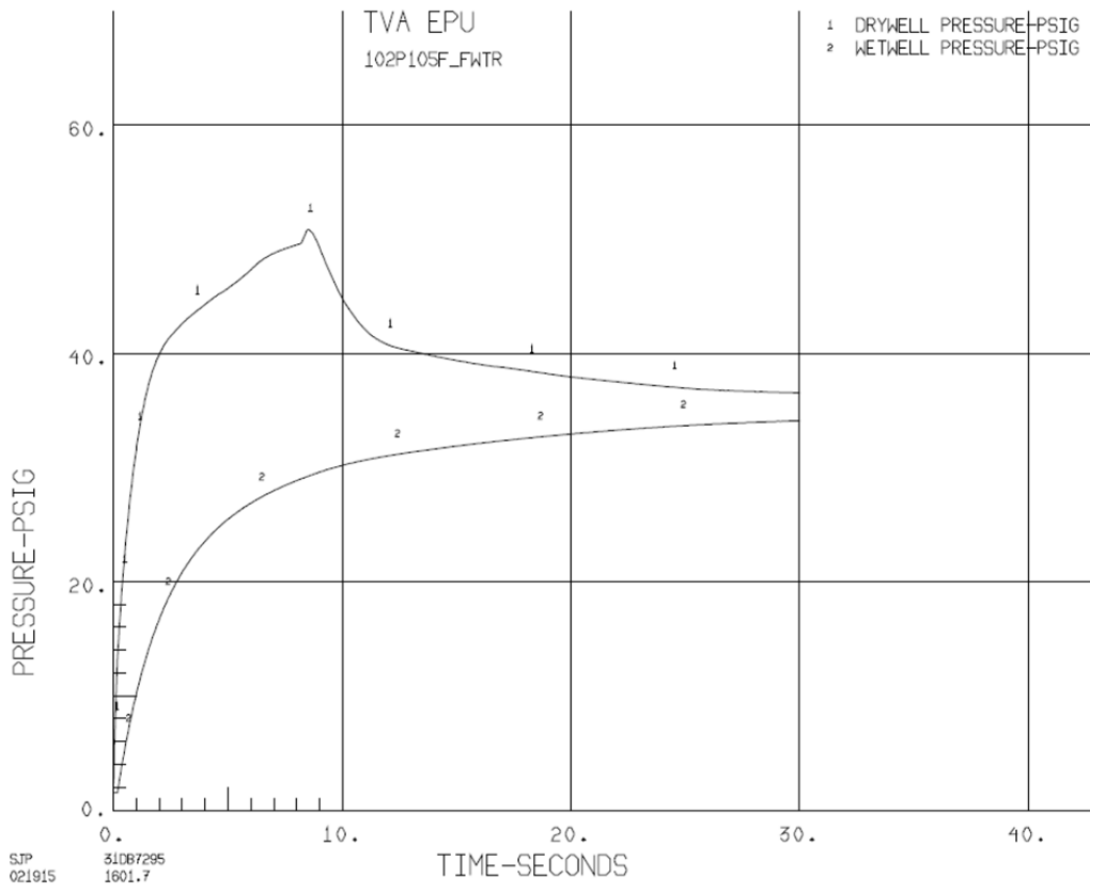

Figure 14.12-2

Short-Term RSLB DBA LOCA Containment Temperature Response Design Case

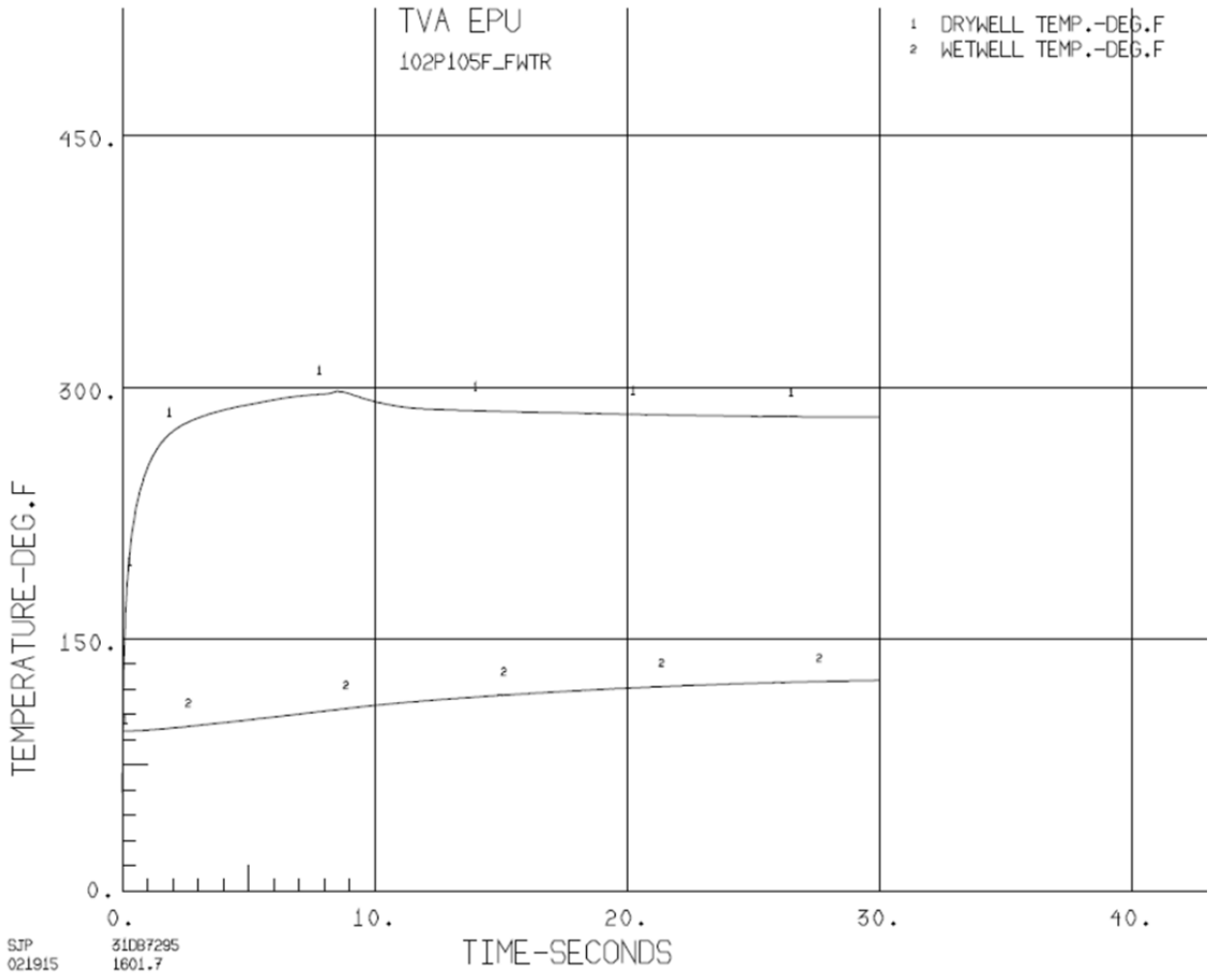

Figure 14.12-3

Short-Term RSLB DBA LOCA Containment Pressure
Response Bounding Case

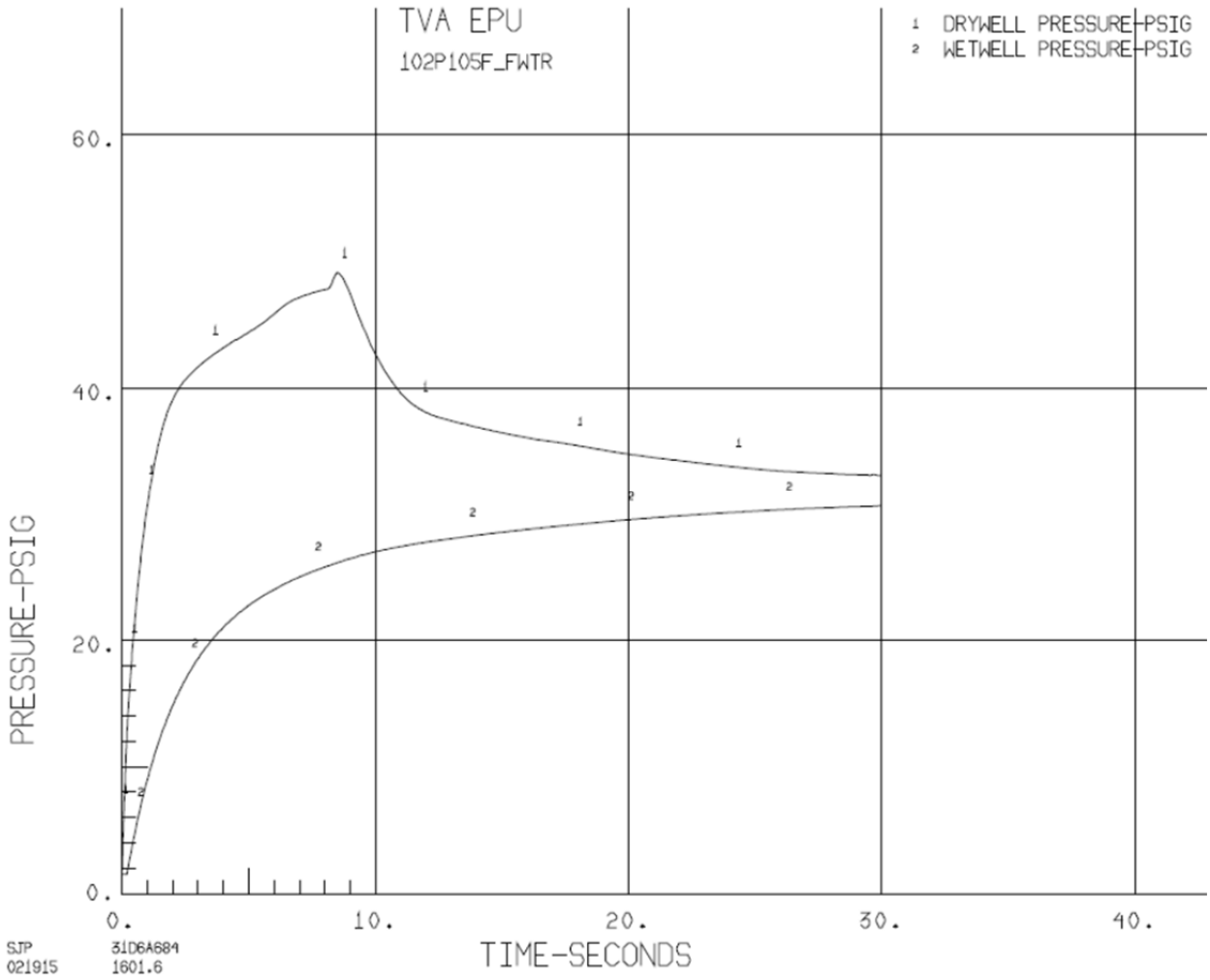

Figure 14.12-4

Short-Term RSLB DBA LOCA Containment Temperature
Response Bounding Case

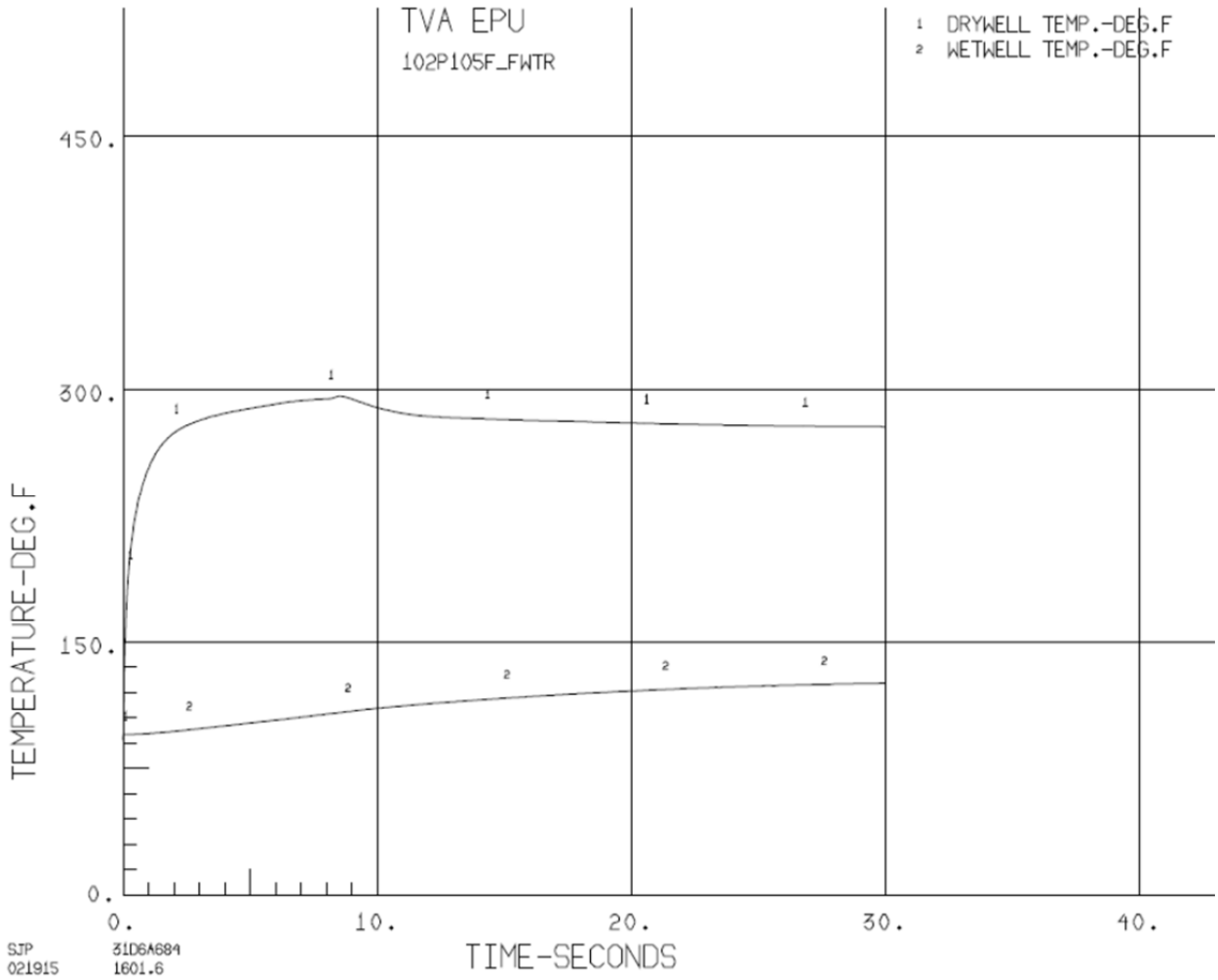

Figure 14.12-5

Short-Term RSLB DBA LOCA Containment Pressure
Response Reference Case

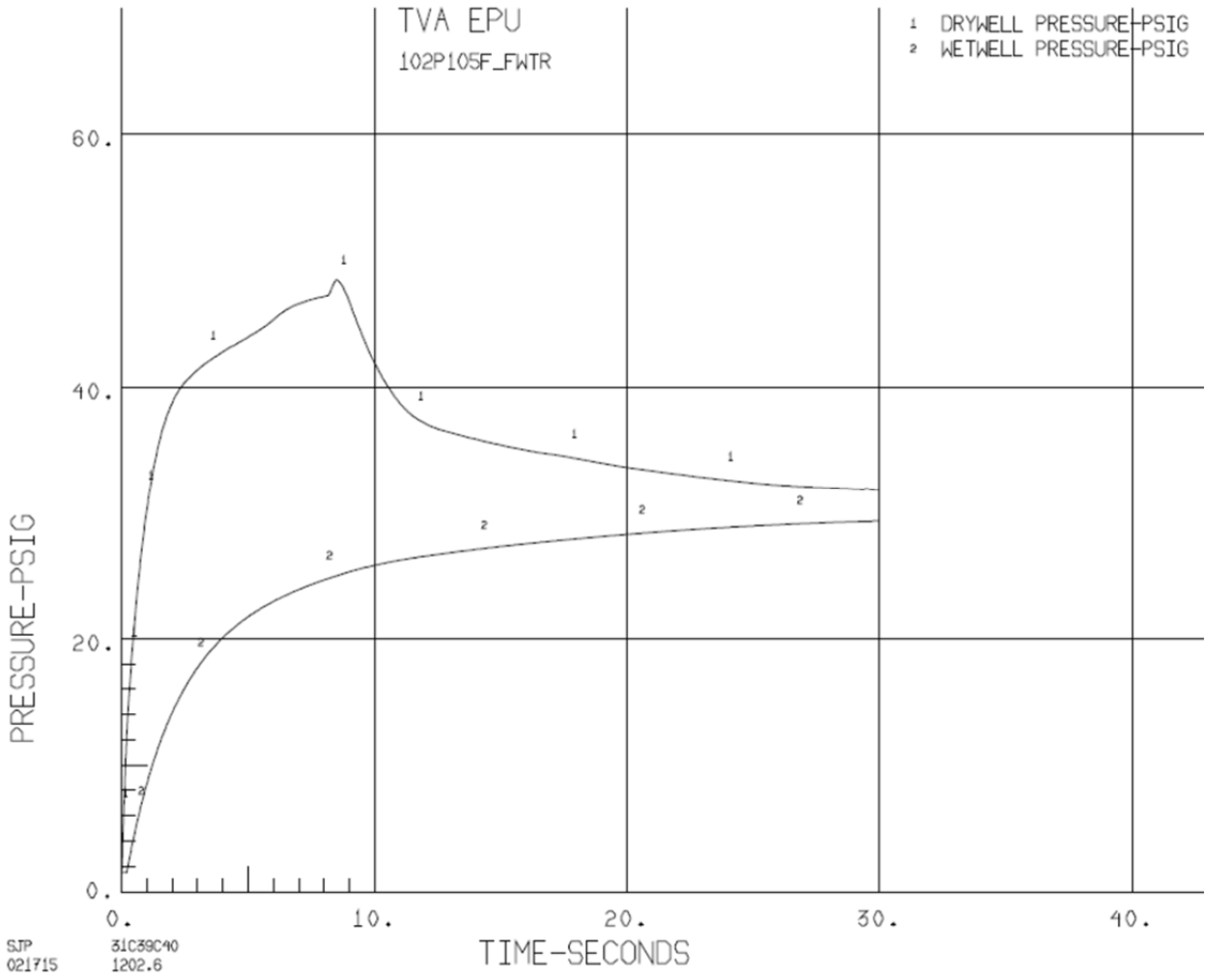

Figure 14.12-6

Short-Term RSLB DBA LOCA Containment Temperature Response Reference Case

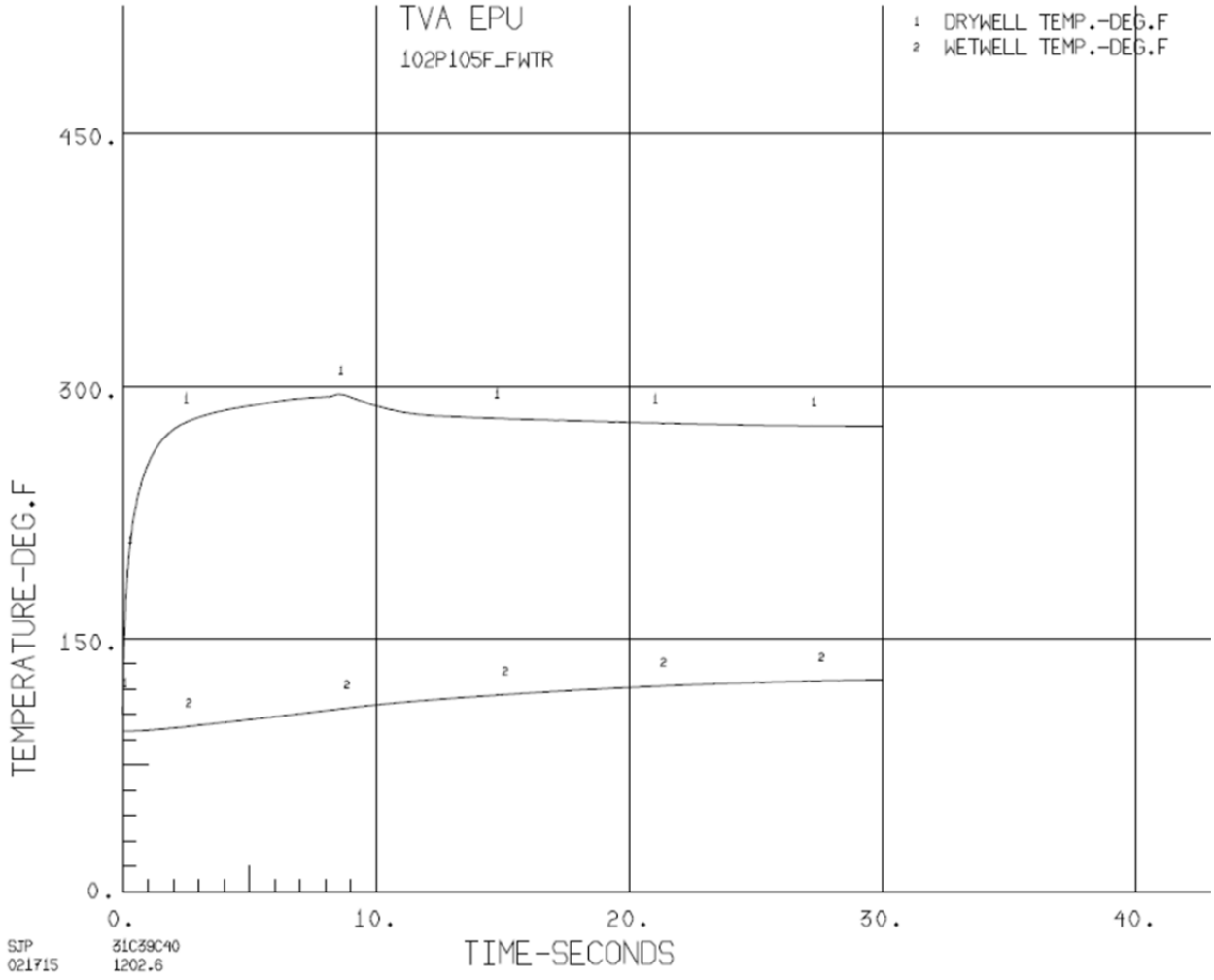

Figure 14.12-7a

Drywell Temperature Response - Small Steam Line Break LOCA

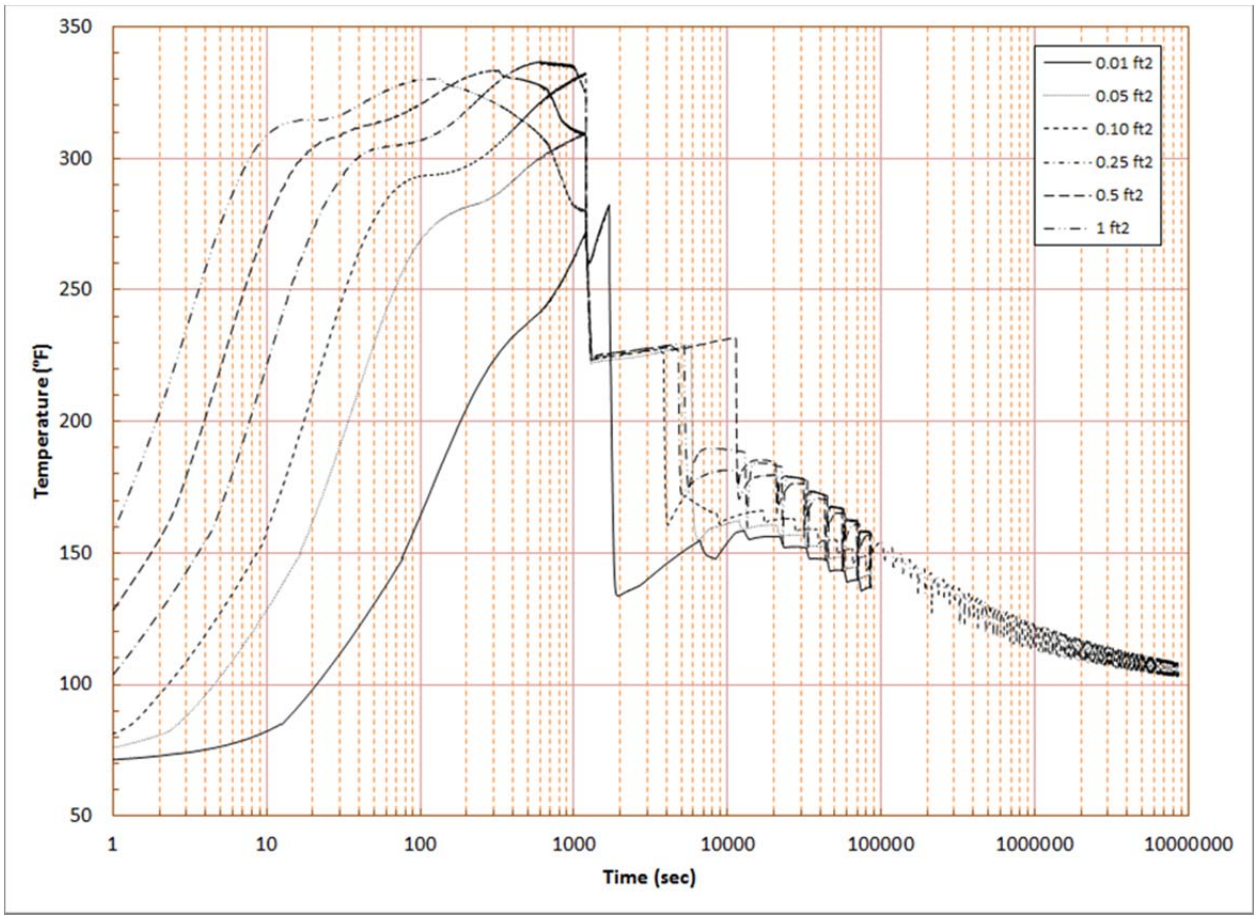

Figure 14.12-7b

Drywell Pressure Response - Small Steam Line Break LOCA

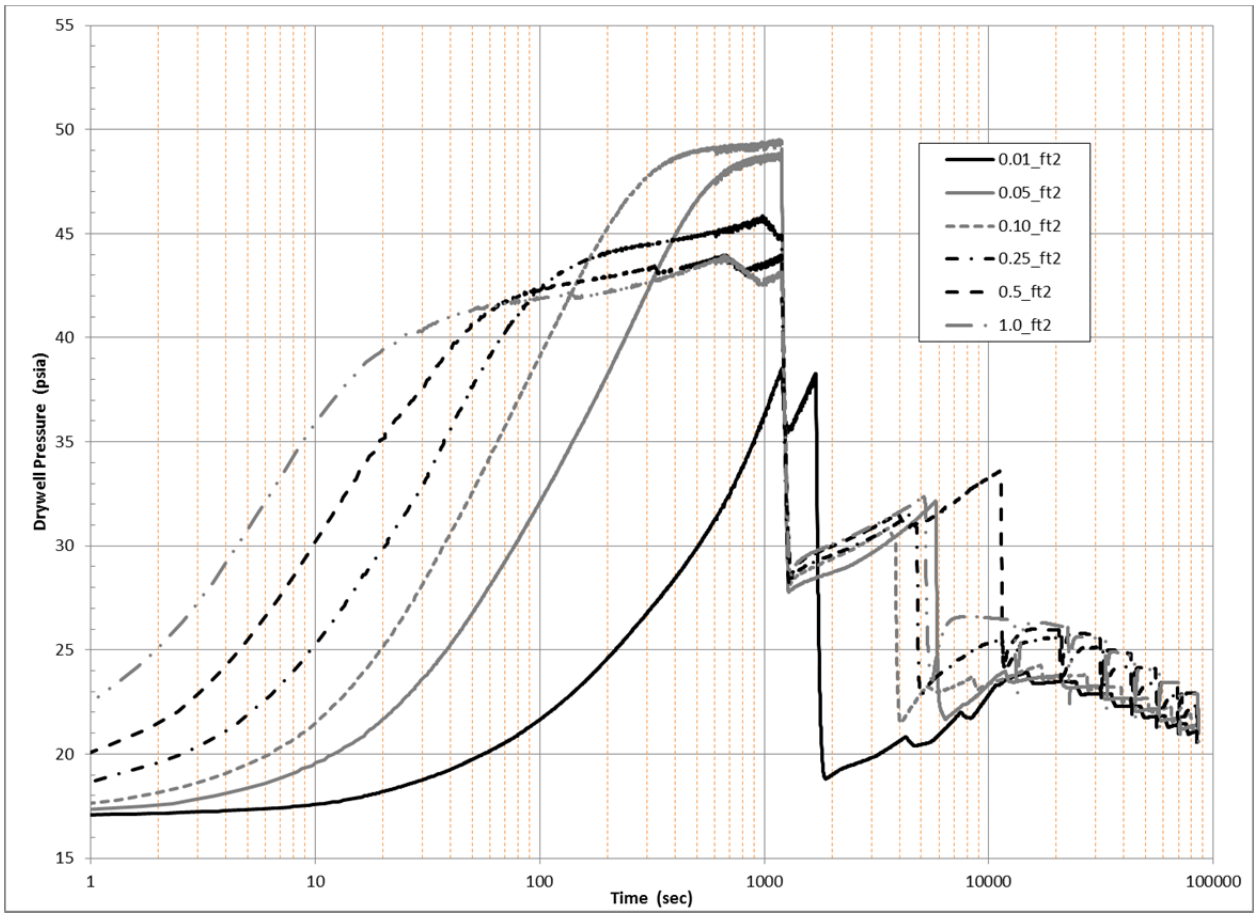

Figure 14.12-8

Suppression Pool Temperature Response - RSLB DBA LOCA

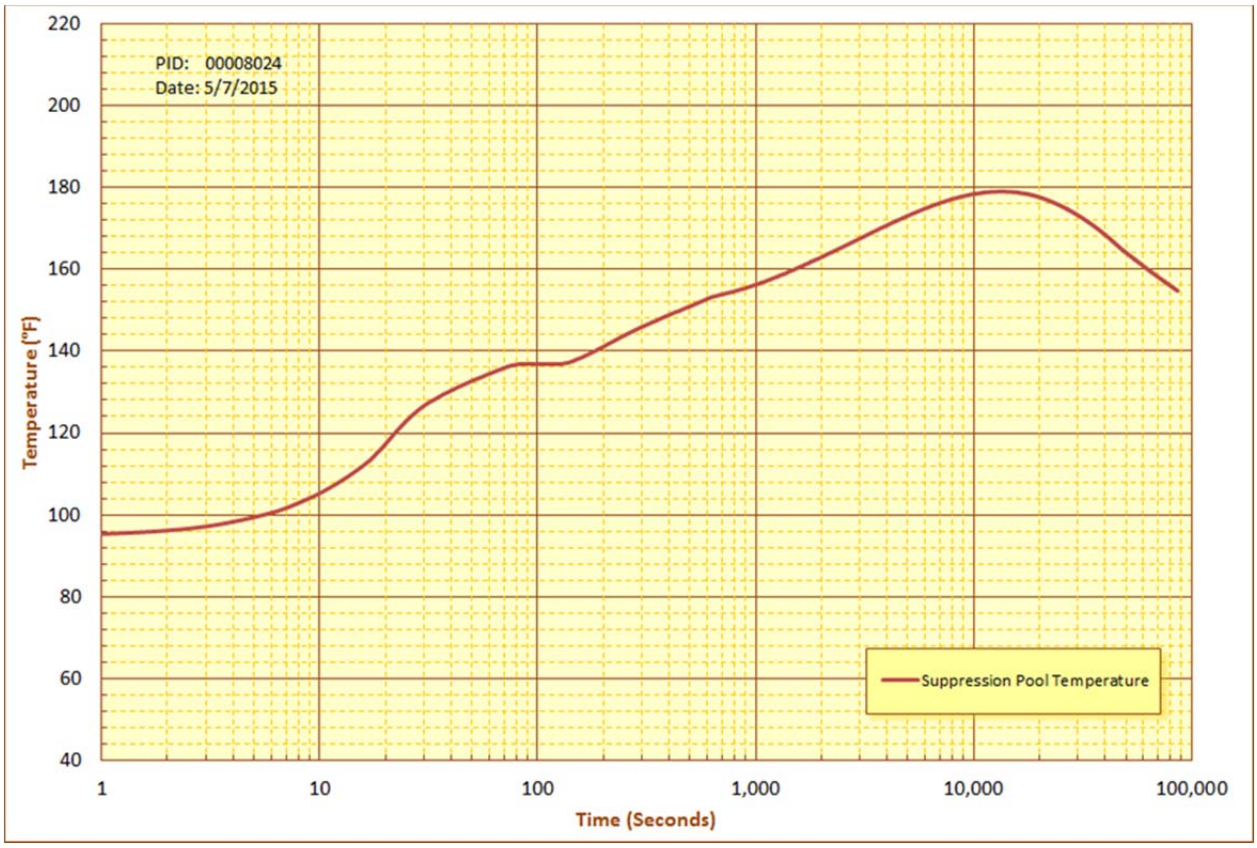

Figure 14.12-9

Drywell and Wetwell Airspace Temperature Response - RSLB DBA LOCA

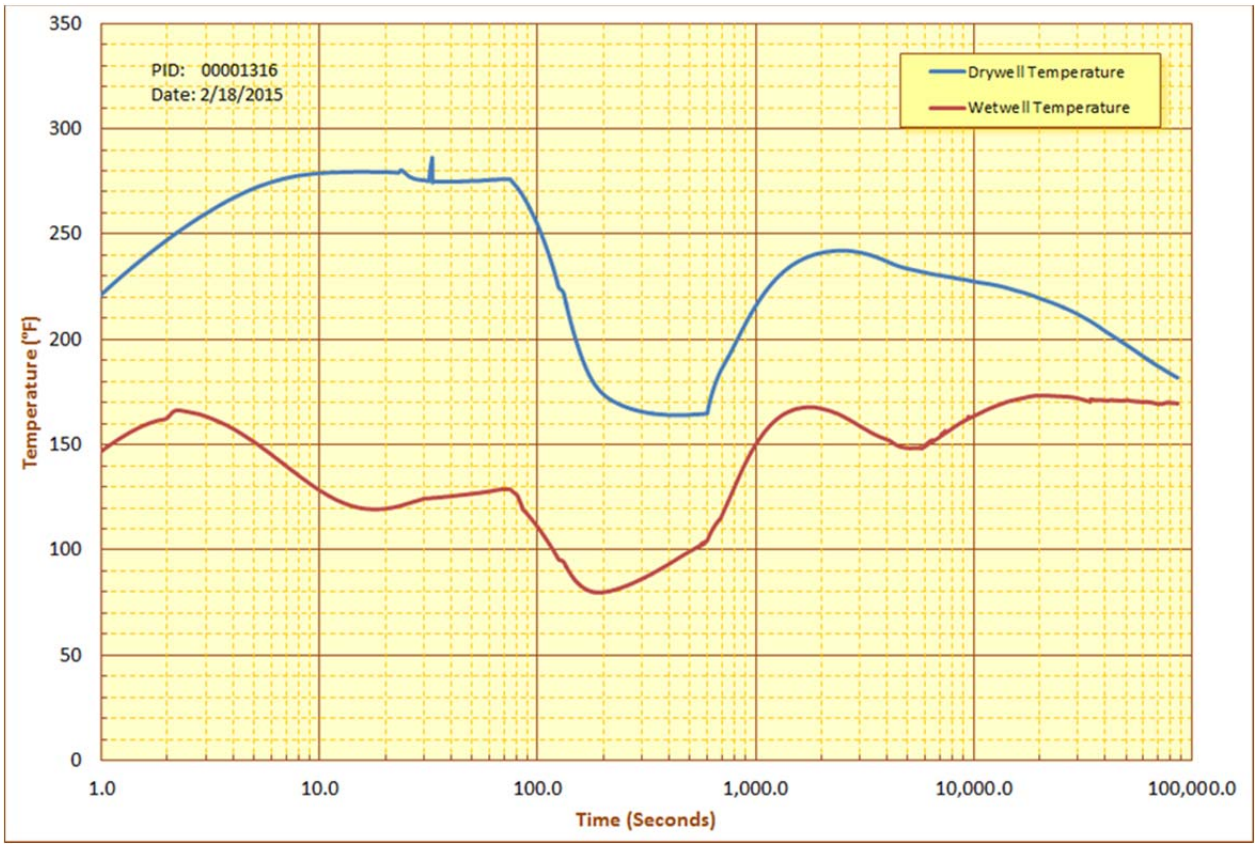

Figure 14.12-10

Long-Term Small Steam Line Break LOCA Suppression Pool Temperature Response - 0.01 ft^2 Break with HPCI Available

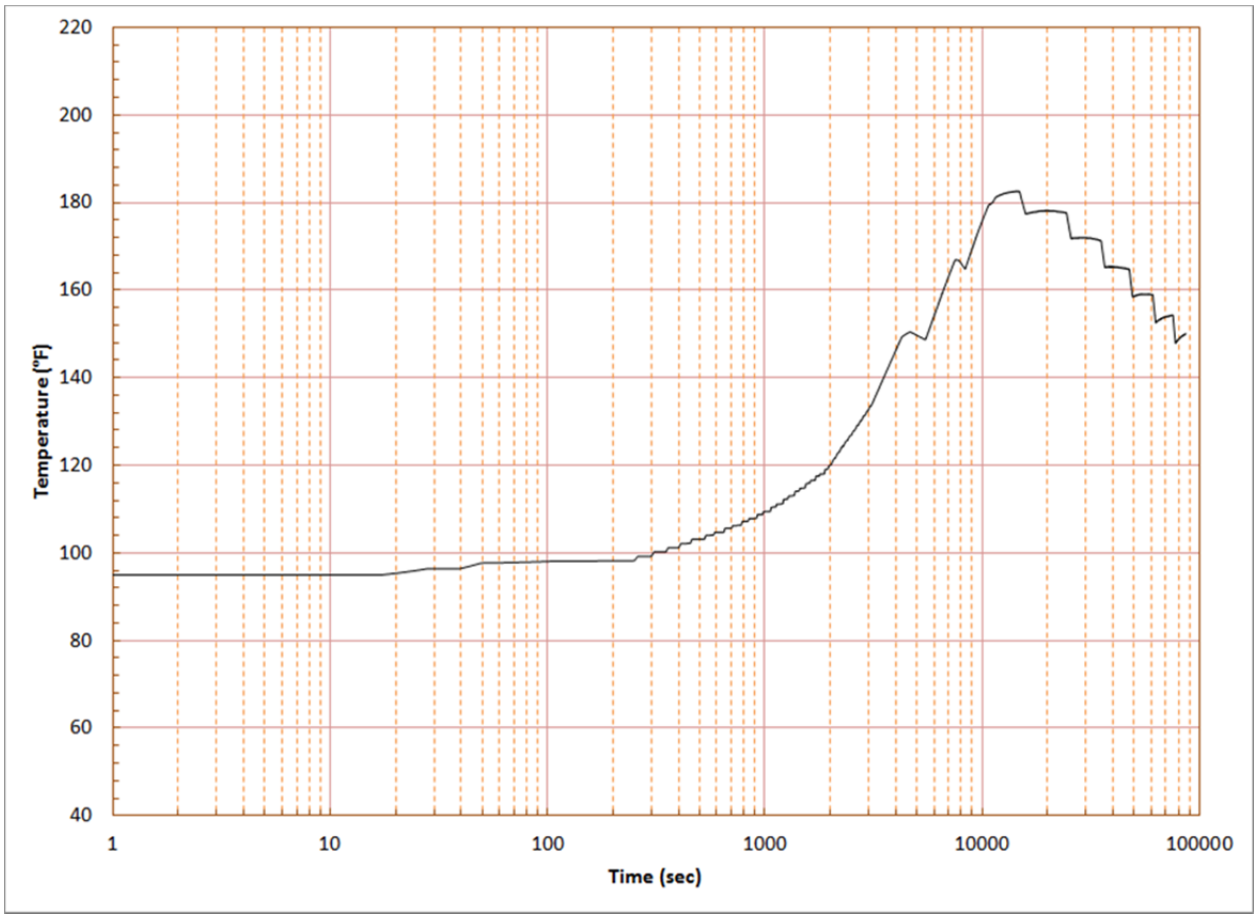

Figure 14.12-11

Suppression Pool Temperature Response
Non-Accident Unit Shutdown
(CST Available)

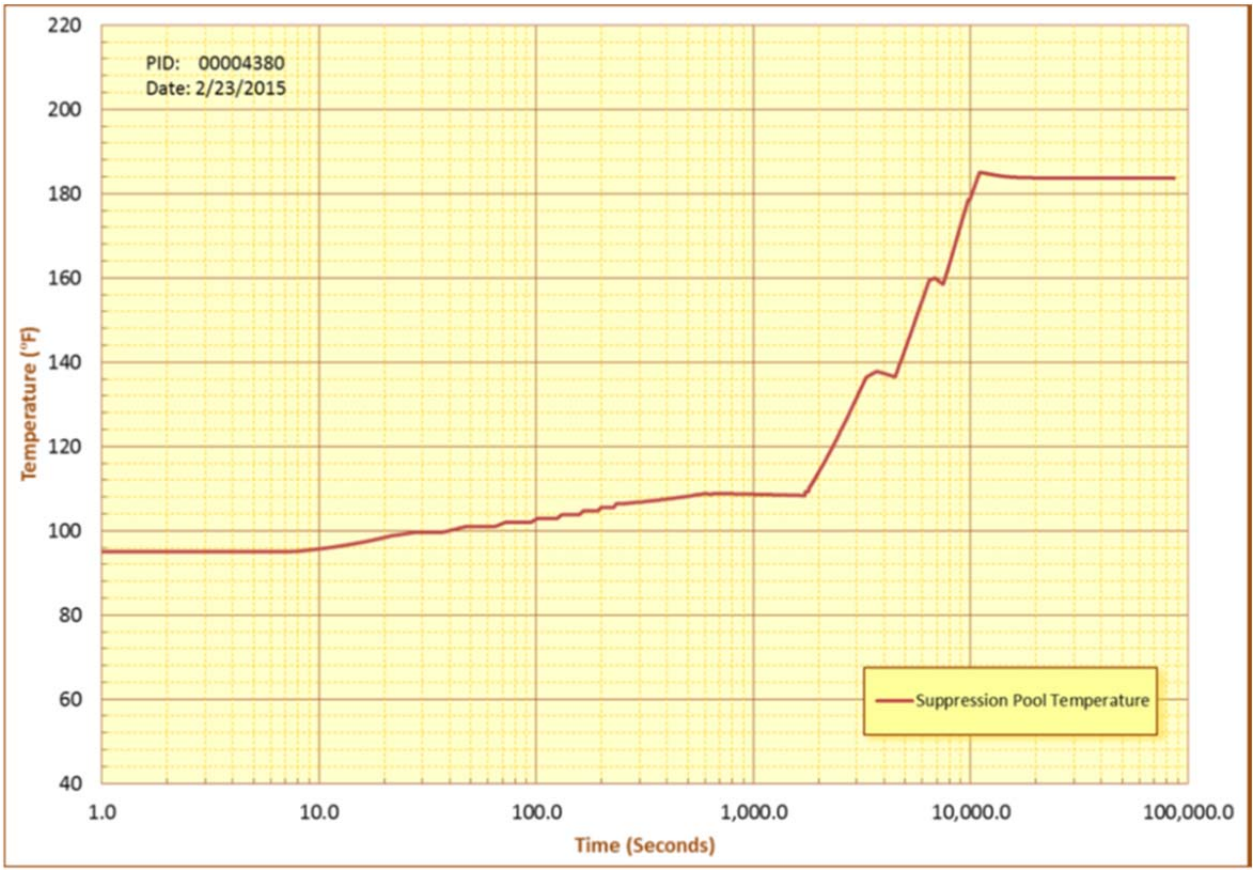

Figure 14.12-12

Station Blackout Suppression Pool Temperature Response

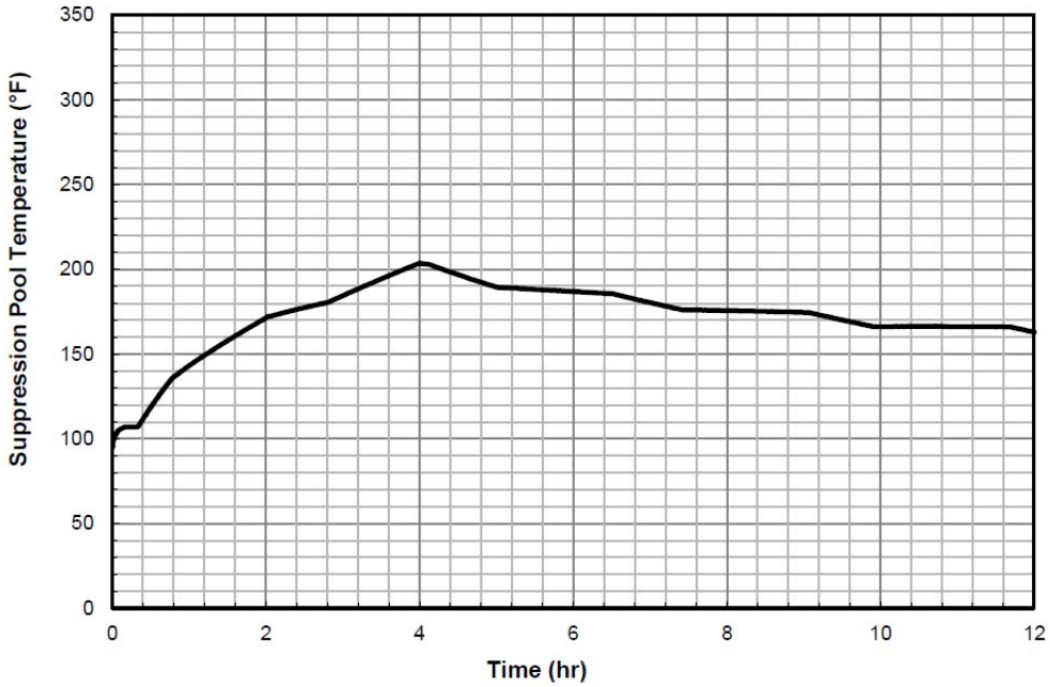

Figure 14.12-13

Fire Event Suppression Pool Temperature Response

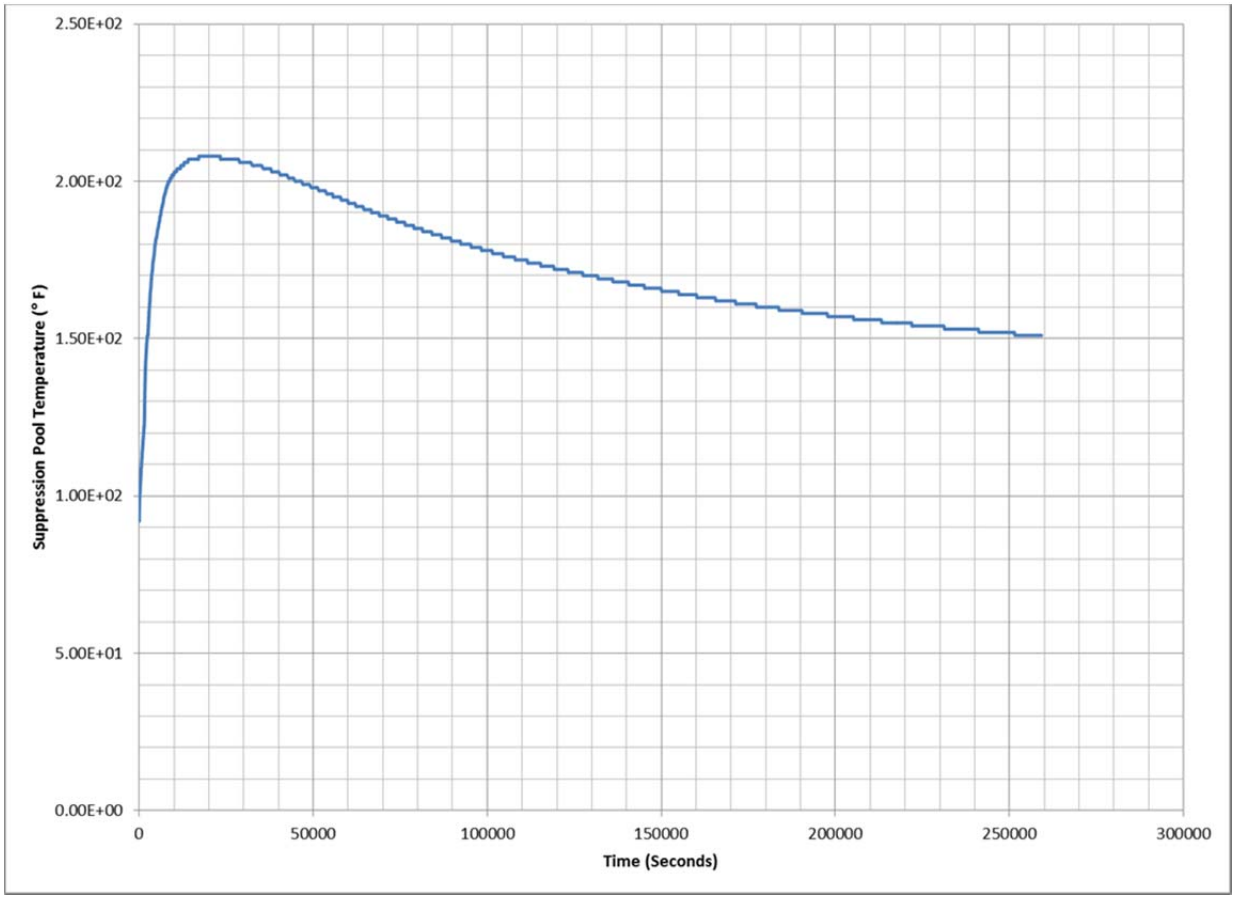

Figure 14.12-14

ATWS Suppression Pool Temperature and Containment Pressure Response - LOOP