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 2 and 3 for operation in the Maximum Extended Load Line Limit (MELLL) Region. 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."

[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

14.2 <u>UNACCEPTABLE SAFETY RESULTS FOR ABNORMAL OPERATIONAL TRANSIENTS</u>

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 current GE analysis methodology initiates events from the licensed power (3458 MWt) and provides conservatism by accounting for uncertainties in computed results and utilizing NRC approved best estimate methods. The FANP NRC-approved methodology applied to BF3 Cycle 12 and later cores is a deterministic approach utilizing conservative inputs as appropriate to allow for plant condition and modeling uncertainties. The latest GE analysis and results are described in NEDC-32484P and GE-NE-B13-01866-21. These analyses are based upon an assumed power of 3458 MWt (nominal) and 3527 MWt (Appendix K). These values reflect uprated power and are also conservatively above the Browns Ferry pre-uprated power of 3293 MWt. The non-limiting abnormal operational transients are not updated for each refueling cycle. For GE analyzed reload cores, the Reload Licensing Report, NEDE-24011-P-A, NEDC-32484P, and GE-NE-B13-01866-21 should be consulted for the currently applicable analysis results and methodology. For reload cores with FANP ATRIUM-10 fuel, the Reload Licensing Report should be consulted for the currently applicable analysis results and methodology. For non-power uprated conditions, the results of the analyses at 3293 MWt previously contained in Sections 14.5 and 14.6 have been relocated to Sections 14.10 and 14.11, respectively.

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.

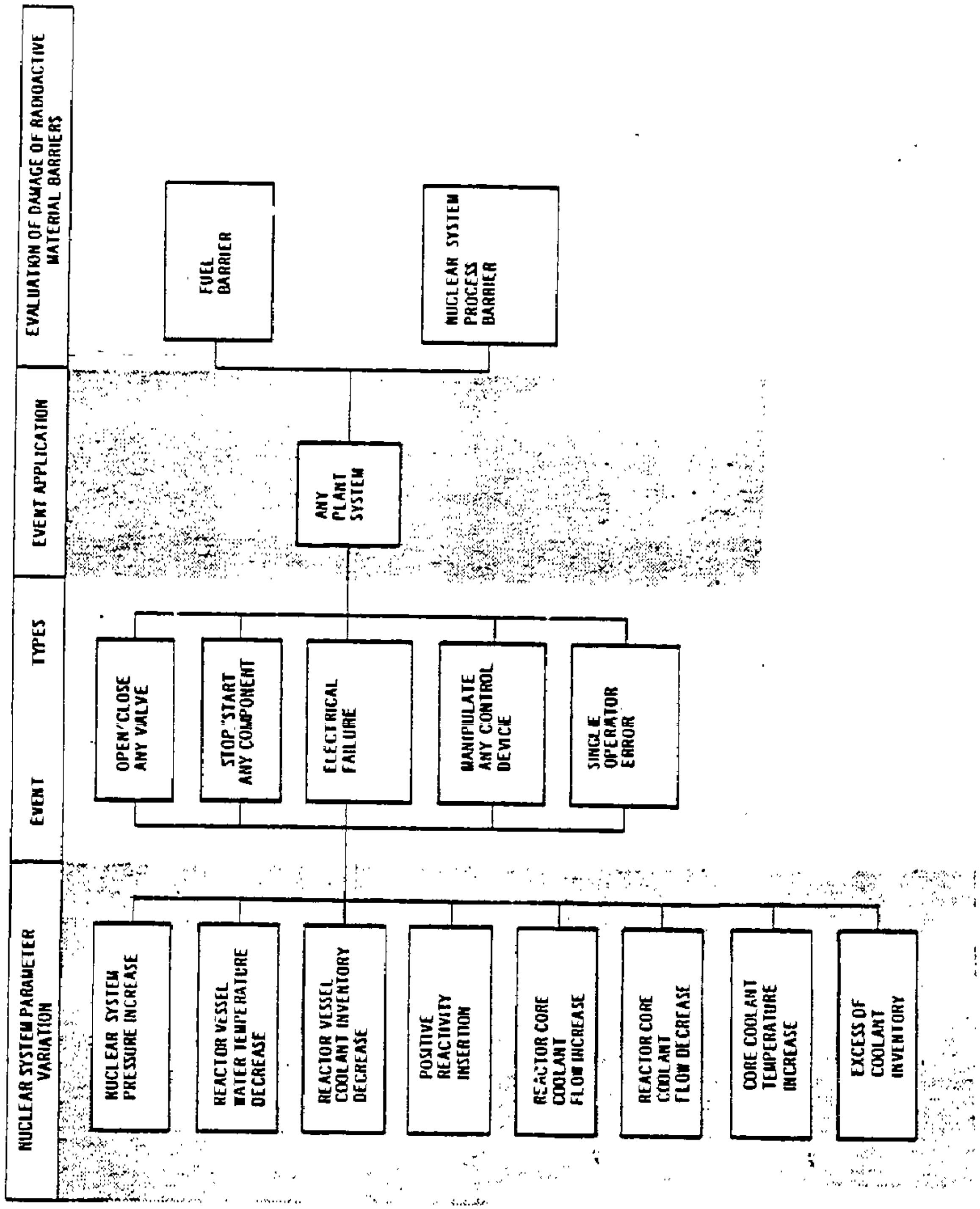
An analysis performance measurement, which is discussed in Subsection 3.6 ("Nuclear Design"), is used to evaluate whether nuclear system process barrier damage occurs as a result of reactivity accidents. If peak fuel enthalpy remains below 280 calories per gram no nuclear system process barrier damage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

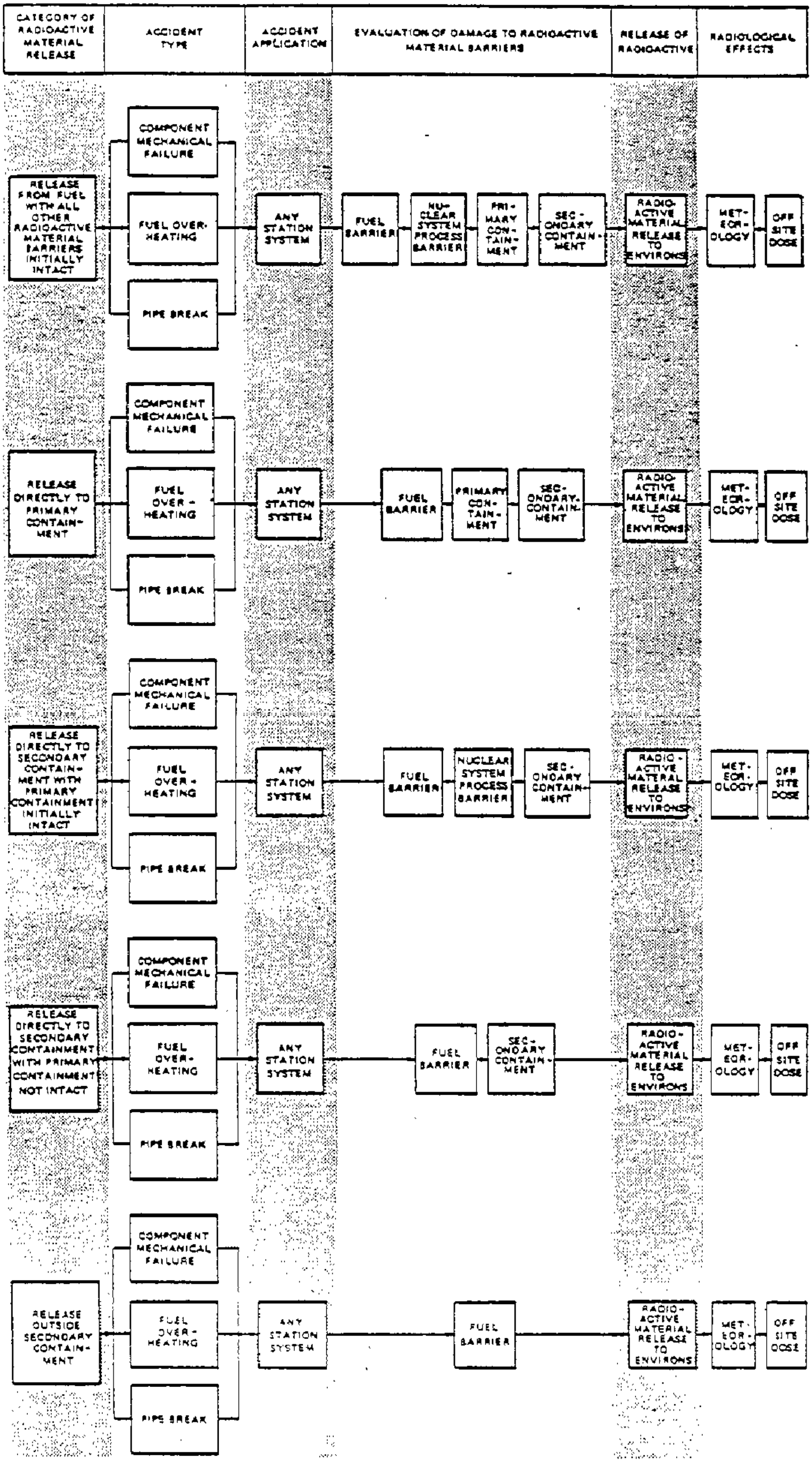
Damage to any of the radioactive material barriers as a result of accident-initiated fluid impingement and jet forces is considered in the other portions of the Safety Analysis Report where the mechanical design features of systems and components are described. Design basis accidents are used in determining 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.



AMENDMENT 16

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

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

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

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.

14.5 ANALYSES OF ABNORMAL OPERATIONAL TRANSIENTS - UPRATED

14.5.1 Objective

This section contains general descriptions of abnormal operational transients analyzed for BFN Units 2 and 3 at uprated conditions. The similar results at pre-uprated conditions can be found in Section 14.10.

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 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 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 GE analyses are based on the core loading characteristics of BFN 24-month fuel equilibrium cycle with GE13 fuel which bounds GE9 and GE11 fuel designs, also present in the BFN core. This is considered to be representative of future cycles, including GE14 and Framatome-ANP ATRIUM-10 fuel, 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

GE Analysis Computer Codes

NRC-approved computer models have been used for the analysis of each event, consistent with the analyses guidelines established in "Generic Evaluation of General Electric Boiling Water Reactor Power Uprate Licensing Topical Report", NEDC-31984P, July 1991. The computer codes used in the different transient events analyses are summarized as follows:

Transient Event Description	**GE Computer Code Used for Analysis**
Generator Trip (TCV Fast Closure) With Bypass Valves Failure	1-D ODYN Model
Load Rejection No Bypass/EOC-RPT-OOS	1-D ODYN Model
Loss of Condenser Vacuum	1-D ODYN Model
Turbine Stop Valve Closure/Turbine Trip	1-D ODYN Model
Bypass Valves Failure Following Turbine Trip, High Power	1-D ODYN Model
Bypass Valves Failure Following Turbine Trip, Low Power	1-D ODYN Model
Closure of All Main Steam Line Isolation Valves	1-D ODYN Model
Closure of One Main Steam Line Isolation Valve	1-D ODYN Model
Loss of a Feedwater Heater	REDY Point Model /PANACEA
Inadvertent Pump Start	REDY Point Model
Continuous Rod Withdrawal During Power Range Operation	PANACEA
Continuous Rod Removal Error During Refueling	PANACEA
Fuel Assembly Insertion Error During Refueling	None
Pressure Regulator Failure Open	REDY Point Model
Inadvertent Opening of a Main Steam Relief Valve	REDY Point Model
Loss of Feedwater Flow - short term	REDY Point Model
Loss of Feedwater Flow - long term	SAFER Model
Loss of Auxiliary Power Transformers	REDY Point Model
Loss of Auxiliary All Grid Connections	1-D ODYN Model
Recirculation Flow Control Failure-Decreasing Flow	REDY Point Model
Trip of One Recirculation Pump	REDY Point Model
Trip of Two Recirculation Pumps	M/G Set: REDY Point Model VFD: 1-D ODYN Model
Recirculation Pump Seizure	REDY Point Model

Transient Event Description	GE Computer Code Used for Analysis
Recirculation Flow Control Failure-Increasing Flow	M/G Set: REDY Point Model VFD: 1-D ODYN Model
Startup of Idle Recirculation Loop	M/G Set: REDY Point Model VFD: 1-D ODYN Model
Feedwater Control Failure- Maximum Demand	1-D ODYN Model
Feedwater Control Failure- Maximum Demand/EOC-RPT-OOS	1-D ODYN Model
Feedwater Control Failure- Maximum Demand/TBP-OOS	1-D ODYN Model

The overpressurization transients and other events such as the Loss of Offsite Power due to loss of all connection grids and the Feedwater Controller Failure - Maximum Demand have been analyzed using the one-dimensional kinetic thermal-hydraulic ODYN computer code. The ISCOR code models the core thermal-hydraulics, and TASC models the single hot channel with the boundary conditions provided by ODYN and ISCOR. The TACLE driver runs simultaneously ODYN, ISCOR, and TASC.

Reactivity insertion transients, such as the Rod Withdrawal Error (RWE) transient, are analyzed with the 3-D core simulator PANACEA in order to eliminate over-conservatisms associated with the REDY point-kinetics model and to provide a more realistic simulation of this quasi steady-state transient. The Loss of Feedwater Heater (LFWH) transient is analyzed with PANACEA as well as with REDY assuming that the initial and final reactor condition are steady-state. The limiting loss of inventory transient, Loss of Feedwater Flow, is run with the SAFER code for the analysis of the long-term inventory.

The following transients have been reanalyzed for Recirculation Pump VFDs using the one-dimensional kinetic thermal hydraulic ODYN computer code: Trip of Two Recirculation Pumps, Recirculation Flow Control Failure - Increasing Flow, and Startup of Idle Recirculation Loop.

All other transients described in this document have been analyzed using the point model kinetic thermal-hydraulic REDY computer code. ISCOR and TASC are run together by the DCPRF driver with the boundary conditions provided by REDY. The events have been analyzed at the selected power/flow state points as shown in Table 14.5-1. These are the most limiting rated power/flow state points.

The ODYN and PANACEA computer models assume 3458 MWt as the initial core thermal power, except where noted. The Loss of Condenser Vacuum, Turbine Stop Valve Closure/Turbine Trip, Closure of all and one main steam line isolation valve(s) (MSIV), and the Loss of all Auxiliary Power Grids transients are analyzed with the ODYN code at 3527 MWt or 102 percent of rated power because of the lack of specific statistical adders for power uncertainty. The REDY point model assumes 3527 MWt for its initial conditions. This 2 percent power increase is conservatively set because, unlike ODYN (except in the cases above described) and PANACEA, the REDY evaluation does not introduce a statistical adder for power uncertainty. All of the reload analysis events which are limiting from the viewpoint of fuel thermal margin do include the statistical consideration for power level (and other uncertainties as described in GESTAR, NEDE-24011-P.A.)

Framatome ANP Analysis Computer Codes

For Framatome ANP reload licensing analyses, the 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 ΔCPR. 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.

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)

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 at the uprated condition 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 81 percent of rated at 3458 MWt. This boundary maintains the same maximum control rod load line as pre-uprate operation (i.e., 75 percent core flow at the pre-uprate 3293 MWt condition) and is consistent with the generic guidelines provided in "General Guidelines for General Electric Boiling Water Reactor Power Uprate," NEDC-31897P-1, June 1991. 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 3458 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°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°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) ATRIUM-10, GE14, GE13 and earlier GE fuel designs

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

8) Improved Standard Technical Specifications.

9) Limiting transients with PLU-OOS

These operating flexibility options, with the exception of PLU-OOS, have been included as part of the analyses assumptions for the BFN Power Uprate licensing analyses (Reference NEDC-32751P, "Power Uprate Safety Analysis for BFNP Units 2 and 3," September 1997). The EOC-RPT-OOS contingency 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 (ΔCPR) 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.

Under ARTS/MELLL conditions a new set of power and flow dependent CPR and maximum average planar linear heat generation rate (MAPLHGR) limits have been calculated in 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 "Power Uprate Evaluation Task Report for BFNP Units 1, 2 & 3 Transient Analysis", GE-NE-B13-01866-05, August 1997. Modified flow dependent MCPR corrections were later supplied in GE/GNF letter 262-00-021-01, “TVA Unit 3 Cycle 10 MCPR(F) Limits”, which will continue to apply to all follow-on Browns Ferry Unit 1, 2, or 3 cores that contain only GE11 and later fuel designs. The power dependent MAPLHGR(P) and MCPR(P) limits and the flow dependent MAPLHGR(F) and MCPR(F) limits 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 FANP reload analyses, off-rated thermal limits are calculated by FANP 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/81F (MELLL domain)
2. 100P/100F (standard domain)
3. 100P/105F (ICF domain)

The 100P is defined as 100 percent of rated power or 3458 MWt. The 100F is defined as 100 percent of rated core flow or 102.5 E6 lbm/hr; 81F and 105F are defined as 81 percent and 105 percent of rated core flow, respectively. As previously discussed, some transients are analyzed at 3527 MWt or 102 percent of rated power (i.e., 102P).

Table 14.5-1 lists the Power/Flow operating conditions for each transient analyzed in this chapter.

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 30 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 30 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

For GE reload analyses, the analysis described in this section was performed with the ODYN computer code 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. The EOC all-rods-out exposure is assumed to conservatively bound the control rod insertion effectiveness at any other cycle exposure. For FANP reload analyses, the FANP computer codes and analysis methodology described in Section 3.7.7.1.2 “MCPR Operating Limit Calculation Procedure” are used.

14.5.2.2.3 Interpretation of Transient Results

Figure 14.5-5 shows 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 568 percent of initial; the average heat flux peaks at 125 percent of its initial value. The peak pressure at the bottom of the vessel is 1283 psia which is well below the ASME upset code transients limit of 1375 psig while the peak steam line pressure is 1245 psia. The calculated ΔCPR at the stated conditions is 0.19 for GE13 fuel; this result is representative but not bounding for other GE fuel types.

At rated power, the ΔCPR 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 30 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.

Figure 14.5-6 shows 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 same as in the transient analysis with TBV in service described above.

The neutron flux peaks at 674 percent of initial, the average heat flux peaks at 130 percent of its initial value. The peak pressure at the bottom of the vessel is 1293 psia which is well below the ASME upset code transients limit of 1375 psig while the peak steam line pressure is 1248 psia. The calculated ΔCPR of this transient at the stated conditions is 0.23.

The penalty associated with EOC-RPT-OOS is about 0.04 in ΔCPR. At less than rated core flow, the penalty is smaller because of the relatively reduced beneficial effect of EOC-RPT.

The impact of the EOC-RPT-OOS on the transient fuel protection at off-rated power/flow conditions has been addressed with the appropriate revision to the

ARTS-based power-dependent MCPR and MAPLHGR limits, as required [NEDC-32774P, Rev 1, "Safety Analyses for Browns Ferry Nuclear Plant Units 1, 2 and 3 Turbine Bypass and End-of-Cycle Recirculation Pump Trip Out-of-Service", September 1997].

For FANP reload fuels, power-dependent LHGR(P) limits are used instead of MAPLHGR(P) limits. For FANP reload analyses, the 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 (19 in Hg), 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.

14.5.2.3.2 Initial Conditions and Assumptions

Because it is an overpressurization transient, it has been analyzed with the ODYN code. Two cases have been analyzed: ICF (105 percent of rated core flow) and MELLL (81 percent of rated core flow), both at 102 percent of rated power. The EOC exposure has been used because the top peaked axial power shape degrades the effectiveness of rod insertion during the reactor scram. Normal feedwater temperature is assumed in this analysis to maximize the vessel steam flow during the transient.

The turbine bypass system opens from turbine stop valve (TSV) closure and closes at 5 seconds due to loss of condenser vacuum signal (at 7 inches Hg, assuming rate of loss of vacuum -2 inches Hg/second to conservatively give only 5 seconds of bypass flow).

The feedwater system trips at time 0 with a 5 seconds coastdown. This transient is modeled with a MSIV closure initiation at 5 seconds as the vacuum protection setpoint is reached. (Although BFN does not have the MSIV closure signal on low vacuum protection setpoint, this assumption has no impact on the transient responses. The key transient parameters such as peak neutron heat flux, peak surface heat flux, and peak vessel pressure occurs prior to 4 seconds and, thus, are not affected by the MSIV closure action at 5 seconds.) One MSRV is assumed out of service with the resulting relief capacity of 73.8 percent of rated steam flow used in the analysis.

14.5.2.3.3 Interpretation of Transient Results

Figures 14.5-7a and b illustrate this transient at ICF conditions which results in the most severe response. Peak neutron flux reaches 435 percent of rated; however, the fuel surface heat flux reaches 121 percent of its initial value. The relief valves open fully to limit the pressure rise, then sequentially reclose as the stored energy is dissipated. The peak nuclear system pressure at the bottom of the vessel (1243 psia) is also well below the nuclear process barrier transient pressure limit of 1375 psig.

This transient is equivalent to a turbine trip with bypass operable event. Therefore, this transient is bounded by the Turbine Trip No Bypass and Load Rejection No Bypass events; and no damage to the fuel results from this transient.

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 30 percent). 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).

14.5.2.4.2 Initial Conditions and Assumptions

The calculation of this transient has been performed with the ODYN computer code at the most limiting conditions: 100 percent of rated power, 105 percent of rated core flow, EOC exposure conditions, and normal feedwater temperature. The turbine bypass system is assumed to be operable.

14.5.2.4.3 Interpretation of Transient Results

Figure 14.5.8 illustrates this transient. The reactor scrams very early in the transient along with the fast opening of the TBV. A recirculation pump trip (RPT) is initiated upon the turbine trip. The reactor pressure rises to the MSRV setpoints causing them to open for a short period. The TBV system continues operating throughout the transient.

The fuel thermal transient is mild relative to the limiting events. Peak neutron flux reaches 447 percent of the rated power; the fuel surface heat flux reaches 122 percent of its initial value. The ΔCPR resulting from this event is bounded by the TTNBP transient. No damage to the fuel results from the transient. Peak pressure in the bottom of the vessel (1246 psia) and at the steam lines is below the ASME code limits for the nuclear process barrier (1375 psig).

Turbine trips from lower initial power levels decrease in severity because the vessel pressurization transient is milder with the reduced vessel steam flow rate. At core power less than 30 percent of rated, the turbine bypass system is capable of handling the vessel steam flow from a turbine trip event; and thus, the reactor scram signal from a turbine stop valve closure is bypassed.

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 ODYN computer code 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

Figure 14.5-9 illustrates 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 564 percent of rated power while peak heat flux reaches 125 percent of rated power. Peak steam line pressure and peak vessel pressure reach 1243 and 1281 psia, respectively.

The results show a ΔCPR of 0.19 for this event. This trend is similar to that observed above for the LRNBP event. The TTNBP event is bounded by the LRNBP event.

14.5.2.6 Bypass Valves Failure Following Turbine Trip, Low Power

14.5.2.6.1 Transient Description

This abnormal operational transient is of interest because it is initiated at the highest power for which turbine stop valve closure and turbine control valve fast closure scrams and Recirculation Pump Trip (RPT) is automatically bypassed by an interlock with a turbine load signal. The highest power level to bypass reactor scram is about 30 percent of rated power. Reactor scram is initiated by high dome pressure.

14.5.2.6.2 Initial Conditions and Assumptions

The calculation of this transient has been performed with the ODYN computer code at 30 percent of rated power and 50 percent of rated core flow, EOC exposure, and normal feedwater temperature.

14.5.2.6.3 Interpretation of Transient Results

Figures 14.5-10a and b illustrate this transient. Reactor scram is initiated at about 5.5 seconds by high vessel pressure (peak vessel pressure: 1217 psia). Peak neutron flux reaches 61 percent of rated while peak heat flux reaches 41 percent of rated. No bypass flow is assumed; however, a portion of the MSRVs open to relieve the pressure transient. The peak steam line pressure (1199 psia) remains below the ASME code limits. No fuel damage occurs since fuel integrity is protected by the application of the ARTS based power-flow dependent MCPR and MAPLHGR limits [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 "Power Uprate Evaluation Task Report for BFNP Units 1, 2 & 3 Transient Analysis", GE-NE-B13-01866-05, August 1997].

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 (Operating State D) 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 ODYN 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

Figure 14.5-11 illustrates 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 1234 psia 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 ODYN at 102 percent of rated power, 105 percent of rated core flow (ICF conditions), and EOC exposure. The exposure used has been EOC because the top peaked axial power shape degrades the effectiveness of rod insertion during the reactor scram.

A typical value of 60 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 and b illustrate this transient. The steam flow disturbance raises vessel pressure and reactor power causing a high neutron flux scram at about 4 seconds; the peak neutron flux reaches 131 percent of rated. The peak surface heat flux reaches about 110 percent of rated. Peak steam line pressure (1113 psia) remains below the setting of the lowest MSRVs. Peak vessel pressure (1155 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 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. 102 percent of rated power and 81 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 ΔCPR and on the fuel thermal and mechanical design limits. The LFWH event for BFN assumes a feedwater temperature reduction of 100° F (from 382° F to 282° 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° F) due to a single equipment failure is the worst condition analyzed for BFN using this procedure.

14.5.3.1.2 Initial Conditions and Assumptions

This transient was analyzed for thermal-hydraulic dynamic description purposes with REDY and for ΔCPR calculations with PANACEA 3-D reactor simulator code due to the quasi steady-state nature of the LFWH transient.

The LFWH analysis was conducted at two different conditions, 105 percent of rated core flow (ICF) and 81 percent of rated core flow (MELLL), in order to ascertain the most limiting condition for the transient results. Both codes, REDY and PANACEA, result in the same limiting conditions, 81 percent of rated flow and BOC exposure. The analysis was performed at 102 percent of rated power with REDY and 100 percent of rated power with PANACEA.

14.5.3.1.3 Interpretation of Transient Results

Figures 14.5-13a and b illustrate this transient with the recirculation control system in the manual flow control mode analyzed at the BOC exposure for the MELLL condition as it resulted in the most severe transient. The introduction of subcooled water into the reactor causes reactor power to slowly increase, neutron flux responds immediately, and surface heat flux lags behind. The power increase raises the turbine steam flow but does not reach the high neutron flux scram setpoint.

The plant eventually reaches a steady state condition at an increased power level. Because of the nature of the LFWH event, it results in a slow, monotonic increase in reactor power and surface heat flux. Because of the quasi steady-state nature of the LFWH event, the core thermal margins can be evaluated by analysis of the beginning and end-points of the event with a qualified steady-state 3-D reactor simulator code. For the LFWH event, PANACEA 3-D reactor simulator code showed a peak neutron and surface heat flux of 117%, and ΔCPR of 0.11. (For FANP analyses, the MICROBURN-B2 3-D reactor simulator code is used to calculate LFWH results which are included in the Reload Licensing Analysis

Report.) Therefore, 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.

This transient is less severe from lower power levels for two main reasons: (1) lower initial power levels will have initial fuel parameter values less limiting than the values assumed here, and (2) the magnitude of the power rise decreases with the initial power condition. Therefore, transients from other reactor operating states or lower power levels within Operating State F will be less severe.

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 (operating states B or D), 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 (maximum power within reactor Operating State F).

14.5.3.3.2 Initial Conditions and Assumptions

This transient was analyzed with REDY at 102 percent of licensed power condition and two limiting power/flow state points: increased core flow (105 percent of rated) and MELLL (81 percent of rated flow), both at BOC exposure condition.

As explained above, the inadvertent startup of the large, 5000 gpm, HPCI System has been considered. During the initiation and acceleration transient for the HPCI, the pump flow can overshoot the rated flow making the event more severe. An overshoot of 20 percent was used in this transient. Also, for conservatism purpose, a 10 percent margin was added to the HPCI flow rate. The water temperature of the HPCI was assumed to be 40° F with an enthalpy of 11.0 Btu/lb.

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

Figures 14.5-14a and b illustrate this transient with the recirculation control system in the manual flow control mode and BOC/MELLL conditions. 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 analysis at the BOC/MELLL condition is more limiting than the BOC/ICF condition with regards to the fuel transient ΔCPR requirement as shown in the table below.

Inadvertent Pump Start Results

	PEAK NEUTRON FLUX (%)	PEAK HEAT FLUX (%)	- DELTA CPR
BOC/MELLL	116.3	114.7	0.11
BOC/ICF	116.7	114.5	0.11

The plant eventually reaches a steady state condition at an increased power level but with no significant threat to the fuel thermal margins. This transient results to be less severe than the LFWH transient as anticipated, because its effect on feedwater temperature produces a change of less than 80°F compared to the 100°F change previously analyzed for the LFWH transient. Therefore, and according to GESTAR II (NEDE-24011-P.A.), the reload cycle-specific analysis only includes the LFWH event. No fuel clad barrier damage results for the malfunction or inadvertent startup of any of these auxiliary cold water supply systems.

A similar inadvertent HPCI startup analysis was performed by FANP using COTRANSA2/XCOBRA/XCOBRAT methodology.[1] This analysis also considered the effect of asymmetric HPCI flow distribution entering the reactor pressure vessel. The analysis considered various cycle exposures and initial core flows. The ΔCPR result was 0.02 more limiting than the case with symmetric HPCI injection, and the change in peak LHGR increased by less than 3%. Therefore, the results of the HPCI injection event continue to be much less limiting than results for the generator load rejection without bypass transient (Section 14.5.2.2).

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

[1] Reference: T. A. Galioto to J. F. Lemons, “Disposition of Browns Ferry Inadvertent HPCI Event with Asymmetric Flow Distribution”, TAG:04:035, February 20, 2004.

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 (approximately 0.2 percent Δk) 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. The methodology in NEDE-24011-P-A is used for licensing analysis performed by GE. NRC approved methodology is used for licensing analysis performed by FANP. 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 "Power Uprate Evaluation Task Report for BFNP Units 1, 2 & 3 Transient Analysis", GE-NE-B13-01866-05, August 1997], 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 when the licensing analysis is performed by GE. For licensing analysis performed by FANP, the thermal-mechanical protection is verified in the cycle specific RWE analysis.

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

For licensing analysis performed by GE, a specific RWE analysis has been performed based on a bounding 24-month GE13 equilibrium cycle core design. The analysis included a statistical evaluation of a range of control rod withdrawal errors conditions such that the rod with the maximum possible error worth can be determined. This type of error could only be achieved by deliberate operator action or by numerous operator errors during rod pattern manipulation prior to the selection and complete withdrawal of the rod. Abnormal indications and APRM alarms would warn the operator as he approaches this abnormal rod pattern. The power-dependent RBM setpoint stops the rod withdrawal. Neither nuclear system process barrier damage nor fuel damage occur as long as the OLMCPR is established by more limiting transients, a condition which is verified every cycle specific evaluation. The results are shown in the following table.

Rod Withdrawal Error Results

Power Range (%)	Rod Block Monitor Setpoint (%)	Maximum $\Delta CPR/ICPR_{95/95}$*	ARTS generic $\Delta CPR/ICPR_{95/95}$**	Calculated MCPR Operating Limit***
85-100	108	0.14	0.13	1.28
70-85	112	0.17	0.19	1.31
30-70	118	0.17	0.28	1.32

* Evaluated at the 95% probability and 95% confidence level

** From 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

*** OLMCPR = SLMCPR/(1- $\Delta CPR/ICPR_{95/95}$)
Where ICPR = Initial CPR & SLMCPR = Safety Limit MCPR

FANP Licensing Analysis

For licensing analysis performed by FANP, a cycle specific RWE analysis is performed. 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 15 percent and at 120 percent of rated power depending on the initial power level.

The pre-uprate UFSAR analysis was performed for a 2.5 percent Δk control rod withdrawal at the rod drive speed of 3 in./sec starting from an average moderator temperature of 82° 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 uprated conditions, as the licensing basis criteria for fuel failure is that the fuel enthalpy must not exceed 170 cal/gm. At the uprated power, 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 uprated conditions.

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

This transient was analyzed with REDY. The worst case has been analyzed for maximum initial power: 102 percent of rated power and 100 percent of rated core flow, BOC exposure.

Conservative scram, void and Doppler reactivity multipliers for power decrease have been used during the initial blowdown portion of the transient.

The most severe case is assumed to be the one resulting from a pressure regulation malfunction in which a steam flow demand capable of fully opening the turbine control and bypass valves occurs. The Maximum Combined Flow Limiter (MCFL) has been conservatively set to the maximum value of 150 percent to ensure the full opening of the turbine control and bypass valves.

The low steam line pressure isolation setpoint has been set to 843 psig to allow margin to the plant limit of 825 psig.

A conservatively long 5 second closure time is assumed for the MSIV closure with initiation of a reactor isolation signal to avoid rapid cooldown.

14.5.5.1.3 Interpretation of Transient Results

Figures 14.5-15a and b illustrate this transient at BOC exposure. As a consequence of the decreasing turbine inlet pressure, vessel dome pressure decreases resulting in bulk fluid volume increase which produces level swells. In addition, the resulting increase in core void volume causes reactor power to decrease due to negative void feedback.

The depressurization causes a rapid rise in reactor vessel water level up to the high level setpoint (L8) at 3 seconds. This initiates a turbine trip: main turbine stop valve closure, feedwater and recirculation pumps trip, and reactor scram due to the TSV closure. A plant-specific turbine trip response occurs, but it is milder than other limiting transients since power has begun to drop due to the depressurization and because the bypass system is already open (due to the failed pressure regulation). Therefore, vessel pressure increase is mild, and there is no relief valves actuation.

The MSIVs automatically close when pressure at the turbine decreases below 843 psig at 46 seconds.

The peak neutron flux and fuel surface heat flux do not exceed the initial power. No fuel damage occurs.

The low pressure isolation stops the saturation temperature drop about 30° F below the initial conditions, thus, preventing too large a thermal gradient to occur in the metal hardware components.

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.

14.5.5.2.2 Initial Conditions and Assumptions

This transient was analyzed with REDY. The transient begins with the system at nominal operating conditions with 102 percent of rated power and one of the relief valves open, remaining open throughout the transient. Two cases have been run for this transient: 100 percent of rated core flow and BOC exposure conditions and 105 percent of rated core flow and EOC exposure conditions. This event is not sensitive to initial core flow; therefore, the 100F and 105F conditions are chosen to reflect the most likely operating state at BOC and EOC. The worst case is BOC and rated core flow conditions. The capacity assumed for the opening of the relief valve is 6.15 percent of rated nuclear system steam flow.

14.5.5.2.3 Interpretation of Transient Results

Figures 14.5-16a and b illustrate this transient with BOC exposure and rated core flow. The inadvertent opening of one of the relief valves on the main steam line produces 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 neutron flux and fuel surface heat flux do not exceed the initial power. No fuel damage results from the transient. 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

14.5.5.3.1 Transient Description

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.

This transient has been analyzed with the transient model REDY for the initial portion of the event. In order to evaluate the water level behavior, a long term evaluation has been performed with the LOCA computer model SAFER.

As part of the short-term REDY analysis, the feedwater control system failures or feedwater pump trips can lead to partial or complete loss of feedwater flow. Following the trip of all feedwater pumps, feedwater system inertia results in a 5 second feedwater flow decrease before flow is completely stopped. The decrease in feedwater flow produces a slight decrease in core pressure drop and in core inlet subcooling, both of which increase core void fraction. This condition results in an initial reactor power decrease and reduces the reactor vessel water level drop for the first few seconds of the transient. The water level continues to decrease until a low level scram is initiated at L3. Decay and stored heat continue to create steam and the level continues to drop. When the wide range (WR) sensed level reaches the L2 setpoint, the recirculation pumps trip, and the RCIC system is actuated. No credit is taken for HPCI actuation, and RCIC maintains adequate water inventory. The MSIVs are closed if wide range sensed level reaches the L1 setpoint.

The limiting parameter to be considered in this event is the water level which is calculated as part of the long-term analysis. The design criteria for this event is that downcomer water level must remain above L1 (approximately at the top of active fuel, (TAF) elevation).

The loss of feedwater flow event is evaluated to confirm that this event remains non-limiting at rated conditions and to assess whether the water level can be sufficiently maintained by the RCIC system without initiation of the low pressure emergency core cooling systems. This event does not pose any direct threat to the fuel in terms of a power increase from the initial conditions. The fuel will be protected provided the water level inside the shroud does not drop below the TAF.

14.5.5.3.2 Initial Conditions and Assumptions

This evaluation has been performed taking into account the lowering of the MSIV reactor water level set point [NEDE-30012, December 1982]. This long-term evaluation was performed using the Appendix K evaluation models with the following conservative assumptions:

a. Conservative decay heat values (ANS-5.1-1979 +10 percent) are used to maximize heat addition to the vessel, MSRV challenges, and inventory loss.

b. This transient is most severe from high power conditions, because the rate of level decrease is greatest and the amount of stored and decay heat to be dissipated is highest. Therefore, the analysis was performed at the 102 percent of rated power condition and 100 percent of rated core flow. This event is not sensitive to initial core flow, and the 100 percent core flow condition is chosen as typical representation of the plant operating condition. Conditions are assumed such that all cycle exposures are bounded.

c. Water level was considered to be at normal level, since this transient is relatively insensitive to changes in initial water level above L3.

d. The feedwater pumps are assumed to coast down in one second. This is also consistent with the Appendix K loss of coolant accident (LOCA) analysis.

e. Only RCIC will initiate at Level 2. Since the HPCI injection rate is about 10 times that of RCIC, this assumption provides the most severe challenge to the reactor core cooling.

f. The RCIC system was initiated with a 30 second time delay after WR level had reached the L2 setpoint.

g. The RCIC flow enthalpy (temperature) was considered to be equal to the feedwater flow enthalpy for the first 2 minutes of the transient. This accounts for the warmer feedwater flow entering the vessel before the colder RCIC flow can actually reach the vessel.

h. For the short term calculation, recirculation pumps were tripped off when WR level reached the L2 setpoint. However, for the long term calculation the pump trip is assumed to occur at L4 in order to simulate the pump runback and minimize the vessel water inventory during the transient.

i. The avoidance of the low level L1 setpoint is treated as an operational criteria; therefore, no MSIV closure takes place.

The major change from earlier analytical approach for this transient is that the main steam lines are no longer isolated with the startup of RCIC when the reactor water level reaches the reactor water level 2 setpoint.

14.5.5.3.3 Interpretation of Transient Results

Figures 14.5-17a and b illustrate the short-term transient at BOC and rated core flow conditions. From the REDY analysis, feedwater pump system inertia results in a 5 second feedwater flow decrease before flow is completely stopped. The decrease in feedwater flow produces a slight pressure drop and a decrease in core inlet subcooling, both of which increase core void fraction, reducing reactor power initially and helps moderate the decrease in actual reactor vessel water level for the first few seconds of the transient. The water level continues to decrease until a low level scram is initiated at L3 at 7 seconds. Decay and stored heat continues to create steam and the level continues to drop. When the WR sensed level reaches the L2 setpoint (20 seconds), the recirculation pumps trip and the RCIC system is actuated with a 30 second time delay. This maintains adequate water inventory. The MSIVs remain open and the main condenser remain as a heat sink. Pressure in the reactor vessel decreases gradually with the power reduction so that no threat is posed for the nuclear system process barrier. The vessel pressure reaches the turbine pressure and remains at this value.

As shown in Figure 14.5-17c (long-term), the feedwater flow coastdown occurred within one minute of initiation; and RCIC alone is still capable of maintaining adequate core coverage with the MSIVs open. RCIC also maintains reactor water level above the MSIV water level isolation setpoint; therefore, the MSIVs remain open and the main condenser remains as a heat sink. Reactor pressure is maintained by the pressure control system and the turbine bypass valve. Pressure suppression pool heatup is not a concern since there is no actuation of the MSRVs.

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.

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 M/G sets are assumed to coast down to the point that RPS instrumentation power is lost. This initiates closure of the MSIVs which also produces a scram signal.

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.

14.5.5.4.1 Loss of All Auxiliary Power Transformers

14.5.5.4.1.1 Transient Description

All pumps are tripped initially. Normal coastdown times are considered for the recirculation (5 seconds with M/G sets and 3.5 seconds with VFD) and feedwater pumps. The protection system M/G sets are assumed to coastdown very early in the transient (at 5 seconds) to the point where scram and main steam isolation occurs. 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.

14.5.5.4.1.2 Initial Conditions and Assumptions

This transient was analyzed with REDY at 102 percent of rated power, 100 percent of rated core flow, and BOC exposure. Conservative scram, void and Doppler reactivity multipliers for power decrease are used.

BFN has relay-type circuitry (RTS) which will generate an independent reactor scram and MSIV closure signal due to loss of power to the scram and MSIV solenoids. Both signals are assumed to actuate at 5 seconds after the loss of offsite power.

Loss of main condenser circulating water pumps causes the condenser vacuum to drop to the turbine trip setting by about 6 seconds. Since the MSIVs close at 5 seconds, the turbine trip and bypass closure have no effect on the transient. The bounding 3 seconds MSIV closure time is assumed.

14.5.5.4.1.3 Interpretation of Transient Results

Figures 14.5-18a and b illustrate this transient with BOC exposure conditions. The initial portion of the transient is very similar to the loss of all feedwater described above except for the recirculation pump trip. Initiation of scram, isolation valve closure, and turbine trip all occur between 5 to 6 seconds and the transient changes to that of an isolation event. Bypass operation lasts for about 2 seconds until MSIV total closure, after which the MSRVs open and close at their respective pressure setpoints as the remainder of the stored heat is dissipated. Both peak neutron flux and peak heat flux reach only 102 percent of rated, their initial values. Peak vessel pressure reaches 1217 psia, and peak steam line pressure reaches 1200 psia; therefore, the ASME reactor pressure limit is not challenged. With one of the lowest opening setpoint MSRV assumed OOS, the MSRVs reopened and reclosed as the generated heat drops down into the decay heat characteristic. This pressure relief cycle continues with slower frequency and shorter relief discharges as the decay heat drops off up to the time the Residual Heat Removal system, in the shutdown cooling mode, can dissipate the heat. Sensed level does not drop to the RCIC, HPCI, and MSIV isolation initiation setpoints during the analyzed time.

14.5.5.4.2 Loss of All Auxiliary Power Grids

14.5.5.4.2.1 Transient Description

An alternate transient results if complete connection with the external grid is lost. 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.

14.5.5.4.2.2 Initial Conditions and Assumptions

This transient has been run with the ODYN computer code because of the initial pressurization at 102 percent of rated power and 105 percent of rated core flow. The EOC exposure has been used because the top peaked axial power shape degrades the effectiveness of rod insertion during the reactor scram.

Turbine bypass valves open at 0.1 seconds due to turbine/generator trip and function per design until forced close after reaching the loss of condenser vacuum setpoint. BFN has relay-type RTS circuitry which will generate an independent reactor scram and MSIV closure signal due to loss of power to the scram and MSIV solenoids. This condition will occur about 5 seconds after the loss of offsite power. The bounding 3 seconds MSIV closure time is assumed.

14.5.5.4.2.3 Interpretation of Transient Results

Figures 14.5-19a and b show the results obtained for this transient. At 0.0 seconds, a full load rejection takes place with its associated scram. Recirculation pumps, condenser circulatory water pumps, and feedwater pumps trip off following the loss of all grid connections. Turbine bypass valves open at 0.1 seconds due to turbine/generator trip and remain available until closed due to reaching the condenser vacuum setpoint at 8 seconds. MSIV closure is completed at 5 seconds after the loss of offsite power. MSRVs open and close at their respective pressure setpoints. WR sensed level does not drop to the RCIC, HPCI, and MSIV isolation initiation setpoints during the transient event.

Peak vessel pressure reaches 1240 psia and peak steam line pressure reaches 1213 psia; peak heat flux reaches 118 percent of rated and peak neutron flux reaches 373 percent of rated. These results are bounded by other limiting pressurization events such as 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

M/G Set speed control:

Several varieties of recirculation flow control malfunctions can cause a decrease in core coolant flow. The master controller could malfunction in such a way that a zero speed signal is generated for both recirculation pumps. The recirculation flow control system is provided with a speed demand limiter which is set so that this situation cannot be more severe than the simultaneous tripping of both recirculation pump M/G set drive motors. A simultaneous trip of both recirculation pump M/G set drive motors is evaluated in paragraph 14.5.6.3.

The remaining recirculation flow controller malfunction is one in which a failure in a M/G set speed controller could cause the variable speed converter to move at its maximum speed in the direction of zero pump speed and flow. This transient is similar but less severe than the trip of one recirculation pump. The pump speed reduction is slower than simply opening a field breaker so that the decrease in fuel thermal margins is less severe. A trip of one recirculation pump is evaluated in paragraph 14.5.6.2.

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

This transient was analyzed with REDY at 102 percent of rated power, 100 percent of rated core flow, and BOC fuel exposure conditions.

Conservative scram, void and Doppler reactivity multipliers for power decrease were used. In loss of recirculation flow transients, less negative void reactivity feedback is more severe since it results in a slower power decay.

The most severe case is the failure of one of the M/G set speed controllers since the speed controller rate limits are adjusted to keep the master flow controller failure less severe.

For M/G set recirculation flow control plants, fluid coupler velocity is assumed to decrease at a maximum rate of 25 percent/second for the speed decrease in one loop case.

For VFD recirculation flow control, it can be conservatively assumed that the pump shaft seizes as in a recirculation pump seizure event. For FANP reload analyses, a more realistic analysis is made that considers the maximum potential braking torque in decreasing flow for the VFD controller to determine the recirculation pump speed deceleration.

14.5.6.1.3 Interpretation of Transient Results

M/G Set speed control:

Figures 14.5-20a and b illustrate this transient. Downward failure of the speed controller causes the coupler to de-couple at its maximum rate. The resulting decrease in flow causes a decrease in reactor power. As core flow is reduced, the void fraction increases, causing a reactor water level swell. Diffuser flow of the tripped pump reverses at about 9 seconds while diffuser flow of the active pump reaches to approximately 160 percent.

This transient did not reach the narrow range high level setpoint L8 nor the wide range low level setpoint L2.

The change in MCPR is negligible; therefore, no impact on fuel integrity occurs. Neutron and heat flux and vessel and steam line pressure do not exceed their initial values.

VFD speed control:

The results of the VFD controller failure - decreasing flow transient were the same as for a recirculation pump seizure because it was analyzed as a shaft seizure and the near instantaneous stoppage of the pump. The peak neutron and heat fluxes do not increase above initial conditions. The calculated ΔCPR is 0.10, well below that for other types of transients analyzed; therefore, no impact on fuel integrity occurs.

14.5.6.2 Trip of One Recirculation Pump

14.5.6.2.1 Transient Description

Normal trip of one MG set driven recirculation loop is accomplished through the drive motor breaker. However, a worse coastdown transient occurs if the generator field excitation breaker is opened, separating the pump and its motor from the inertia of the motor/generator set. 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. This condition is assumed for this calculation.

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. The fuel surface heat flux decreases at a slower rate than the flow due to the inherent time constants of the fuel, thus momentarily reducing thermal margins. Should the flow decrease too rapidly, fuel is threatened with a momentary high power/low flow situation. This transient is, therefore, evaluated to ensure adequate thermal margins.

14.5.6.2.2 Initial Conditions and Assumptions

This transient was analyzed with REDY at 102 percent of rated power, 100 percent of rated core flow, and BOC fuel exposure conditions. This event is not sensitive to initial core flow and exposure, thus the 100% core flow and BOC condition is chosen as typical representation of the plant operating condition.

In loss of recirculation flow transients, the less negative void reactivity feedback is more severe since it results in a slower power decay. Therefore, conservative scram, void and Doppler reactivity multipliers are used.

The recirculation pump-motor shaft inertia time constants are at minimum values, because it will result in a more severe transient due to the sharper decrease in core flow.

14.5.6.2.3 Interpretation of Transient Results

Figures 14.5-21a and b illustrate this transient. The condition assumed for this calculation is the opening of the generator field excitation breaker, separating the pump and its motor from the inertia of the M/G set. Diffuser flow on the tripped side reverses at about 2 seconds; however, M-ratio in the active jet pumps increases greatly producing about 165 percent of normal diffuser flow. Neither the high level setpoint L8 or low level setpoint L2 are reached during the transient event.

The change in MCPR is small, bounded by the Recirculation Pump Seizure event (Section 14.5.6.4); therefore, no impact on fuel integrity occurs. Neutron and heat flux and vessel and steam line pressure do not exceed their initial values.

14.5.6.3 Trip of Two Recirculation Pumps

14.5.6.3.1 Transient Description

M/G Set speed control:

This two-loop trip provides the evaluation of the fuel thermal margins maintained by the rotating inertia of the recirculation drive equipment. A single failure of the Recirculation Pump Trip (RPT) System logic can trip both pumps, which removes the pump motors from the inertia of the MG sets.

An abrupt reduction in core flow due to the trip of both recirculation pumps increases the core void fraction and, thereby, increases water level and reduces reactor power. The fuel surface heat flux decreases at a slower rate than the flow due to the inherent time constants of the fuel, thus momentarily reducing thermal margins. Should the flow decrease too rapidly, fuel is threatened with a momentary high power/low flow situation. This transient provides the evaluation of the fuel thermal margins maintained by the minimum design rotating inertia of the recirculation pump motors.

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.

14.5.6.3.2 Initial Conditions and Assumptions

M/G Set speed control:

This transient was analyzed with REDY at 102 percent of rated power, 100 percent core flow, and BOC fuel exposure. This event is not sensitive to initial core flow and exposure; the 100% core flow and BOC condition is chosen as typical representation of the plant operating condition.

In loss of recirculation flow transients, the less negative void reactivity feedback is more severe since it results in a slower power decay. Therefore, conservative void and Doppler coefficients are used in this analysis.

The assumed transient was the trip of both recirculation pump motors. The recirculation pump-motor shaft inertia time constants are assumed to be at their minimum values, because it will result in a more severe transient due to the sharper decrease in core flow.

VFD speed control:

This transient was analyzed with ODYN. The initial conditions are 100% power and flow and BOC fuel exposure. The recirculation pump-motor shaft inertia time constants are assumed to be at their minimum values. Because this event is a power decrease event, with no impact on fuel thermal margins, it is sufficient to use representative conditions.

14.5.6.3.3 Interpretation of Transient Results

M/G Set speed control:

Figures 14.5-22a and b illustrate this transient resulting from the trip of both M/G set drive motors with the minimum design rotating inertia and with BOC exposure conditions, which result in the smallest margin to the safety limit (slower power decay vs. less core inlet flow). The core inlet flow decreases rapidly due to the trip of both recirculation drive motors. The reactor water level swells due to this rapid flow reduction but did not reach the high level L8 setpoint. The vessel pressure increase did not affect significantly the nuclear system process barriers. The low level L2 setpoint was not reached in this transient.

The neutron flux and surface heat flux, as well as vessel and steam line pressure, did not increase over the initial conditions. Fuel thermal margin reached the worst condition near 2.0 seconds; however, the change in MCPR is small, bounded by the Recirculation Pump Seizure event (see Section 14.5.6.4); therefore, no impact on fuel integrity occurs. No scram is initiated directly by the simultaneous M/G set motor trip, and the power settles out at part-load, natural circulation conditions. The trip of both RPT breakers, which separates the MG set inertia from the recirculation pumps, has also been analyzed as part of the variable frequency drive modifications and has shown to have similar results. The results show that this more severe flow transient event, due to the change in pump coast down time constant does not result in any challenges to core thermal limits.

VFD speed control:

An abrupt reduction in core flow, due to the trip of both recirculation pumps, increases the core void fraction and thereby, reduces reactor power and increases water level. The water level change, during the event, is not sufficient to reach either L8 (high level) or L2 (low level). The neutron flux, surface heat flux, steam line pressure and vessel pressure do not increase over the initial conditions. There is no reduction in fuel thermal martins. No scram is initiated directly by the RPT and the power settles out at part-load, natural circulation conditions. Figures 14.5-22c through -22f illustrate this transient.

14.5.6.4 Recirculation Pump Seizure*

14.5.6.4.1 Transient Description

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.

14.5.6.4.2 Initial Conditions and Assumptions

This transient was analyzed with REDY at 102 percent of rated power and 100 percent of rated core flow, at BOC fuel exposure.

In loss of recirculation flow transients, a less negative void reactivity feedback is more severe, since it results in a slower power decay and, thus, results in the smallest margin to the fuel safety limit. Therefore, conservative scram, void and Doppler reactivity multipliers for power decrease are used.

The recirculation pump-motor shaft inertia time constants are at their minimum values, because it will result in a more severe transient due to the sharper decrease in core flow.

14.5.6.4.3 Interpretation of Transient Results

Figures 14.5-23a and b illustrate this transient. The drive flow in the seized loop decreases rapidly due to the large pressure loss introduced by the stopped rotor. Core coolant flow reaches its minimum value at about 1.5 seconds. The reactor water level swells due to this rapid flow reduction, reaching the high water level L8 setpoint at about 3 seconds. This causes a turbine trip (turbine stop valve closure) and reactor scram, feedwater pumps trip, and trip of the remaining recirculation pump. The resulting increase in peak vessel pressure (1152 psia) and peak steam line pressure (1137 psia) do not significantly affect the nuclear system process barriers. The low level L2 setpoint is not reached in this transient.

The peak neutron and heat fluxes did not increase above the initial conditions. The calculated ΔCPR is 0.10, well below that for other types of transients analyzed; therefore, no impact on fuel integrity occurs.

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* This event has been reclassified as an accident (see GESTAR II, NEDE-24011-P-A-US)

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

M/G Set speed control:

Several possibilities exist for an unplanned increase in core coolant flow resulting from a recirculation flow control system malfunction. Failure of the master controller can result in a speed increase for both recirculation pumps. The reasonably expected failure modes of the control system can result in a single M/G set fluid coupler demand step changing to maximum or both M/G set fluid coupler demands ramping to maximum at a rate that would be bounded by the worst case single M/G set fluid coupler failure discussed below. The most severe case of increasing coolant flow results when the M/G set fluid coupler for one recirculation pump attempts to achieve full speed at maximum acceleration. The rapid increase in core inlet flow causes a large neutron flux peak which may reach the trip setting and scram the reactor.

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

M/G Set speed control:

This transient was analyzed with REDY 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 reactor power is initially at 75 percent of rated and core flow is at 52 percent of rated.

The M/G set fluid coupler for one recirculation pump attempts to achieve full speed at maximum acceleration. Fluid coupler velocity was assumed to increase at a maximum rate of 25 percent/seconds.

VFD speed control:

This transient was analyzed with ODYN 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 75 percent of rated 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

M/G Set speed control:

Figures 14.5-24a and b illustrate this transient. Upward failure of the speed controller causes the coupler to increase speed at its maximum rate. The resulting increase in flow causes an increase in reactor power. Scram is initiated at 2 seconds due to the high flux setpoint (peak neutron flux: 189 percent of rated). No credit is conservatively taken for the APRM flow-biased flux scram occurrence. The rapid increase in power causes the void fraction to initially drop and the water level to decrease. As the system pressure decreases following the reactor scram signal, the reactor water level rises but does not reach the high level L8 setpoint. The water level, subsequently, turns around but does not decrease to the L2 setpoint.

The changes in nuclear system pressure are not significant with regard to overpressure. The pressure decreases over most of the transient. Peak steam line pressure reaches 1045 psia while peak vessel pressure reaches 1078 psia. The transient fuel surface heat flux reaches 95 percent of rated heat flux. The maximum core flow runout is not changed with power uprate. In addition, the slope of the maximum rod line is unchanged. Therefore, the expected change in ΔCPR at the power uprate condition for this event will remain well within the margins provided by the ARTS-based off-rated flow dependent thermal limits [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 "Power Uprate Evaluation Task Report for BFNP Units 1, 2 & 3 Transient Analysis", GE-NE-B13-01866-05, August 1997].

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 1.9 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 1017 psia while peak vessel pressure reaches 1053 psia. Peak neutron flux reaches 181 percent of rated at 2.1 seconds. The maximum heat flux is 94% of rated at 2.4 seconds. The calculated ΔCPR is 0.14 (at 2.7 seconds), less than that for the bounding transients (TTNBP, LRNBP, and FWCF). Considering that additional margin is provided by the ARTS-based off-rated power and flow dependent thermal limits (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), it is clear that there is substantial margin between this event and the bounding transients, i.e., no violation of fuel integrity occurs.

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° 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

M/G Set speed control:

This transient has been analyzed with REDY. The following bounding initial conditions were assumed:

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

b. The active recirculation pump is operating at a speed that produces about 150 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 up through the inactive jet pumps,

d. Two different reactor power levels have been analyzed; 75 and 30 percent of rated power, both at BOC exposure condition. The latter is the highest initial power for which a high neutron flux scram is not initiated. Normal procedures require startup of an idle loop at a much lower power. If transient is initiated from higher power, reactor scram on high neutron flux will occur; and the results will be less severe. Cases at high power are documented for a complete transient description at the different combinations of core power and pump fluid coupler position.

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

f. The most limiting case is when the idle pump fluid coupler is at a setting which approximates 50 percent of generator-speed demand, corresponding the speed required to provide pump breakaway torque.

g. 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 drive motor breaker is closed at 0 seconds,

b. The motor reaches near synchronous speed quickly while the generator approaches full speed in approximately 5 seconds,

c. Next the generator field breaker is closed, loading the generator and applying starting torque to the pump motor. Generator speed decreases as the generator tries to start the stopped rotor of the pumps. Pump breakaway is modeled to occur at 8 seconds. Speed demand is programmed back to a predetermined speed setting, and

d. The pump discharge valve is opened as soon as its interlock with the drive motor breaker is cleared. (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 reactor vessel.) A nonlinear 30-second valve opening characteristic is used.

Three cases have been run for this transient:

Case (1) with coupler position (19 percent) providing an approximate generator speed demand of 50 percent and with initial 75 percent of rated power.

Case (2) with the minimum coupler position (11 percent) that avoids direct high flux scram and with initial 75 percent of rated power.

Case (3) with coupler position (19 percent) providing an approximate generator speed demand of 50 percent and initial power of 30 percent of rated, the highest initial power for which a high neutron flux scram is not initiated.

VFD speed control:

The transient has been analyzed with ODYN. The following initial conditions were assumed:

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

b. The active recirculation pump is operating at a speed that produces about 150 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 rated. 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. A 30% power case is not required.

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 opened, coincident with the startup of the idle loop pump at 0 seconds. A nonlinear 30 second valve opening characteristic is used (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

M/G Set speed control:

Figures 14.5-25a and b, 14.5-26a and b, and 14.5-27a and b illustrate this transient. Shortly after the pump begins to move, a surge in flow from the active diffusers gives the core inlet flow a sharp rise. The M/G set starts at 0 seconds simultaneously with the 30 second opening of the discharge valve. The motor reaches near synchronous speed quickly while the generator approaches full speed in 5 seconds. At 5 seconds, the speed control is reduced to 1 percent to correspond to the closing of the field breaker and the torque loading of the generator.

The generator does not respond until breakaway torque is applied to the pump motor at 8 seconds. Generator speed drops off rapidly and pump speed accelerates until they equalize and then coast down to a predetermined speed setting.

Case (1): The rapid acceleration in pump speed coupled with partially open discharge valves increases drive flow/core flow and results in a large neutron flux spike. A short duration neutron flux peak of 247 percent of rated is reached at about 11 seconds with a corresponding peak heat flux of 104 percent of rated. This transient excursion results in a reactor scram on high neutron flux signal. Peak steam line pressure reached 1054 psia while peak vessel pressure reached 1088 psia. No damage occurs to the fuel clad barrier.

The flow of cold water in the vessel causes fluctuation in level and, subsequently, feedwater flow. The influx of cold water in conjunction with flux scram suppresses steam flow at about 20 seconds. The drive flow peaks at about 55 percent and then coasts down to establish a flow corresponding to the 20 percent speed reference. Jet pump flow in the idle loop peaks at about 20 percent but turns negative at about 17 seconds as the developed head is overcome by the head in the active loop and the diffuser flow is reversed.

Case (2): A short duration neutron flux peak of about 116 percent is produced with a corresponding peak heat flux of 93 percent of rated. No high flux scram occurs, and no damage occurs to the fuel clad barrier. Peak steam line pressure reached 1054 psia while peak vessel pressure reached 1086 psia. The cold loop water mixes in the bulk water region and its temperature does not significantly affect the core transient.

Case (3): This is the case (with a generator speed demand of 50 percent) with the highest initial power (30 percent of rated) for which high neutron flux scram is not initiated. Peak neutron flux reached 117 percent of rated while peak heat flux reached 53 percent of rated. Peak steam line pressure reached 1016 psia while peak vessel pressure reached 1040 psia. No damage occurs to the fuel clad barriers. For a 50 percent generator speed demand if the transient is initiated from higher power, scram will occur and the results will be less severe.

No fuel damage occurs at low power/flow conditions fuel integrity due to the application of MCPR(P), MCPR(F), MAPLHGR(P), and MAPLHGR(F) limit curves.

The results show that this event remains non-limiting for fuel thermal margins requirements even in the most severe case (Case (3)).

VFD speed control:

Figures 14.5-25c through -25f illustrate this transient. While the pump quickly reaches maximum speed (in less than 4 seconds), the loop flow increases slowly mirroring the flow area of the discharge valve. Between 8 and 10 seconds, the discharge valve flow area increases rapidly. After 10 seconds, the discharge valve offers little hydraulic resistance and is essentially full open. In response to the discharge valve flow area increase, the loop flow increases rapidly between 8 and 10 seconds. The core flow and neutron flux likewise shows surges in the same time frame. The high neutron flux scram setpoint is not reached. Subsequent to this early power surge, driven by flow change, the power continues increasing slowly as the cooler water of the idle loop makes its way into the core. The power increase is terminated once all of the cooler water is discharged from the idle loop.

An early neutron flux peak, in response to the rapid core flow increase, of 87% of rated occurs at 9 seconds. The peak neutron flux of 93% of rated occurs at 75 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 1017 and 1045 psia, both occurring at about 76 seconds. No damage occurs to the clad barrier.

Application of the MCPR(P), MCPR(F), MAPFAC(P), and MAPFAC(F) limit curves assures no fuel damage occurs from events originated from low power/flow conditions.

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

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 GE reload analyses, the ODYN model was used to simulate this transient event, consistent with the current reload licensing methodology [General Electric Company, "General Electric Standard Application for Reactor Fuel", NEDE-24011-P-A-13, August 1996, and the US Supplement, NEDE-24011-P-A-13-US, August 1996]. For FANP reload analyses, the FANP computer codes and analysis methodology described in Section 3.7.7.1.2 “MCPR Operating Limit Calculation Procedure” are used.

The FWCF event was analyzed at 100% power and at 75% rated power 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 75F/108F). 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 ΔCPR caused by the reduced steam line pressure drop. The large pressurization component of ΔCPR dominates over the subcooling component of ΔCPR; therefore, the case with larger steam flow was more severe.

The FWCF event assumed a feedwater flow runout of 124.2 percent flow at 1060 psia feedwater design pressure. Consistent with this event analytical design basis, the feedwater flow runout capacity included approximately 5 percent additional margin for conservatism. 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 14.5-28 and 14.5-29. The transient was initiated from 75P/108F as a typical off-rated operating state point. This low power high flow condition produces a more severe steam/feed flow mismatch and level transient (Figure 14.5-28) than at high power high flow condition, as shown in Figure 14.5-29. The feedwater pumps are assumed to accelerate to their maximum capability.

From Figure 14.5-28, sensed and actual water level increase during the initial part of the transient at about 3.0 inches/second. The high water level (L8) main turbine trip and feedwater turbine trip is initiated at 10 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 (283 percent of rated), surface heat flux peak (98 percent of rated), and fuel thermal transient excursion (ΔCPR = 0.24). The application of the ARTS-based MCPR(P), MCPR(F), MAPLHGR(P), and MAPLHGR(F) curves ensure the fuel integrity at off-rated power/flow conditions [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 "Power Uprate Evaluation Task Report for BFNP Units 1, 2 & 3 Transient Analysis", GE-NE-B13-01866-05, August 1997]. For FANP reload fuels, LHGR(P) and LHGR(F) limits are used instead of MAPLHGR(P) and MAPLHGR(F) limits. For FANP reload analyses, the power and flow dependent MAPLHGR, LHGR and MCPR limits are developed on a cycle-specific basis.

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 1140 psia). The bypass valves close later bringing the pressure in the vessel (peak vessel pressure 1165 psia) under control during reactor shutdown.

In Figure 14.5-29, for 100 percent rated power and ICF, a peak neutron flux of 475 percent of rated and a peak heat flux of 127 percent of rated are reached. Peak steam line pressure reaches a value of 1217 psia while peak vessel pressure reaches a value of 1250 psia. No fuel damage occurs (ΔCPR = 0.19) with the application of the adequate operating limit CPR associated with this limiting transient.

At rated power, the ΔCPR resulting from the LRNBP and FWCF events is more severe than the ΔCPR resulting from any other pressurization events. As power is reduced to 75 percent of rated power or less, the ΔCPR resulting from the FWCF event is higher than the one from the LRNBP 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) (for FANP fuel) 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 75 percent and 30 percent rated are significantly reduced below P_{bypass}.

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

EOC-RPT-OOS eliminates the automatic Recirculation Pump Trip signal when Turbine Trip occurs increasing the severity of the transient responses.

Figure 14.5-30 shows 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 570 percent of initial, the average heat flux peaks at 132 percent of its initial value. The peak pressure at the bottom of the vessel is 1260 psia which is well below the ASME upset code limit transients limit of 1375 psig while the peak steam line pressure is 1219 psia. The calculated ΔCPR of this transient at the stated conditions is 0.23.

The penalty of EOC-RPT-OOS is around 0.04 in ΔCPR. At off-rated power/flow conditions, such as the 75P/52F point, the penalty is smaller because of the relatively reduced beneficial effect of EOC-RPT.

The impact of the EOC-RPT-OOS on the transient fuel protection at off-rated power/flow conditions has been addressed with the appropriate revision to the ARTS-based power-dependent MCPR and MAPLHGR limits, as required [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]. For FANP reload analyses, cycle-specific RPT-OOS fuel thermal limits are determined.

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 ΔCPR 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.

Figure 14.5-31 shows 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 628 percent of initial; the average heat flux peaks at 134 percent of its initial value. The peak pressure at the bottom of the vessel is 1280 psia which is well below the ASME upset code limit transients limit of 1375 psig while the peak steam line pressure is 1248 psia. The calculated ΔCPR of this transient at the stated conditions is 0.24. At rated power the impact on ΔCPR caused by TBP-OOS is approximately 0.05.

The impact of the TBP-OOS on the transient fuel protection at off-rated power/flow conditions has been addressed with the appropriate revision to the ARTS-based power-dependent MCPR and MAPLHGR limits, as required [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]. For FANP reload analyses, cycle-specific TBP-OOS fuel thermal limits are determined.

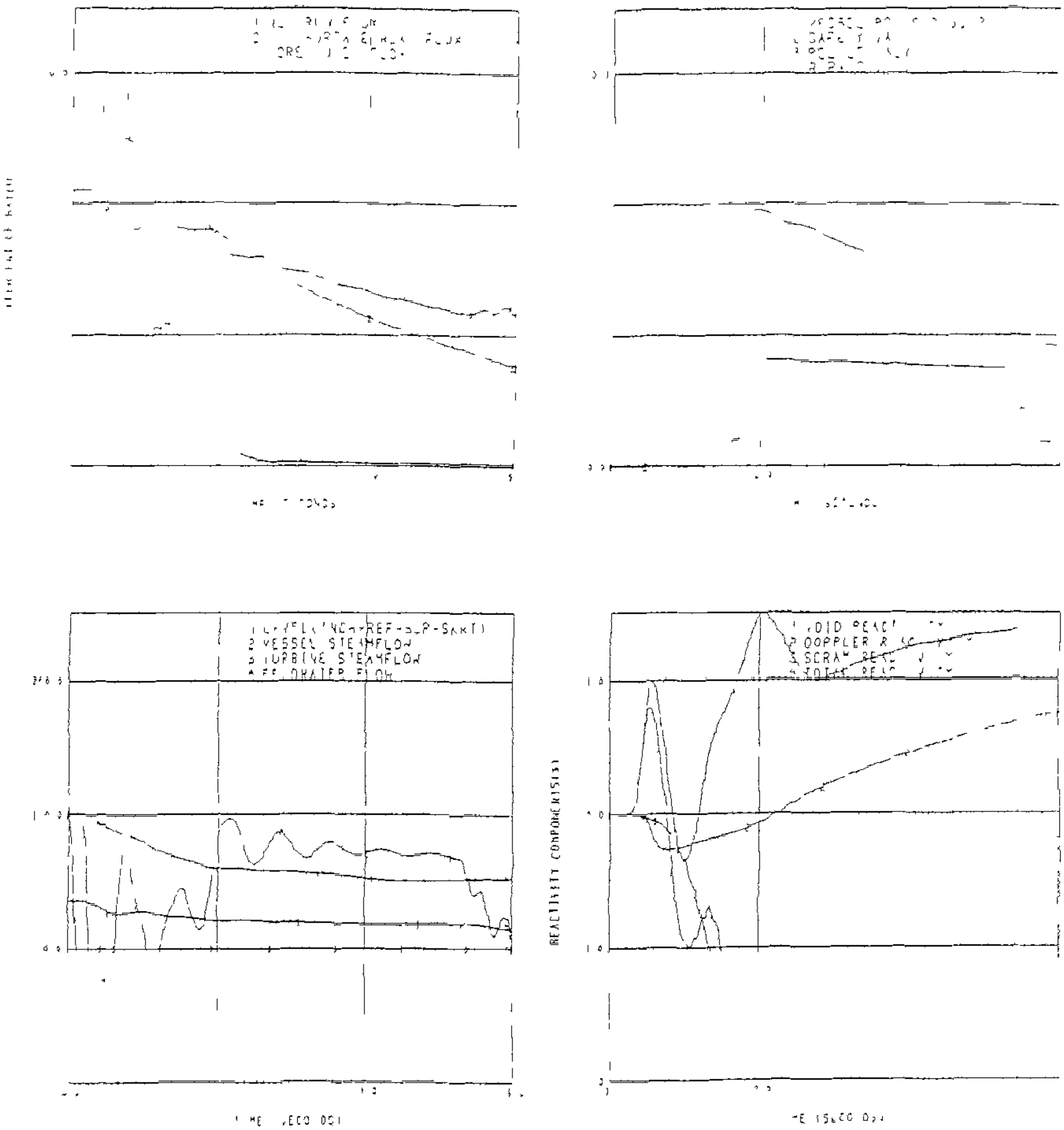
14.5.9 Loss of Habitability of the Control Room

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.)

BFN-19

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

(Deleted by Amendment 19)

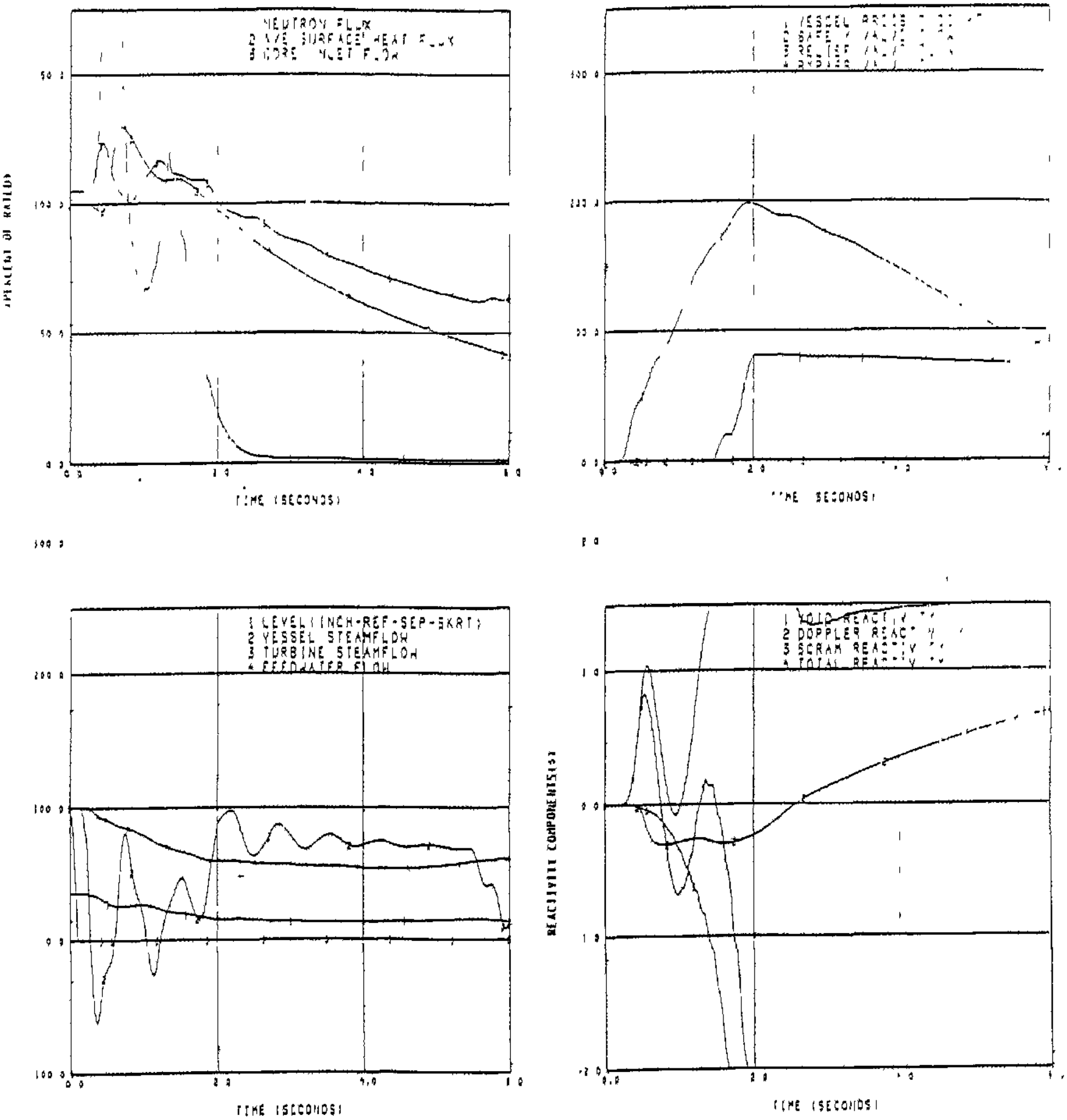


AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

GERERATOR TRIP (TCV FAST CLOSURE)
WITH BYPASS VALVE FAILURE 100P/105F

FIGURE 14.5-5

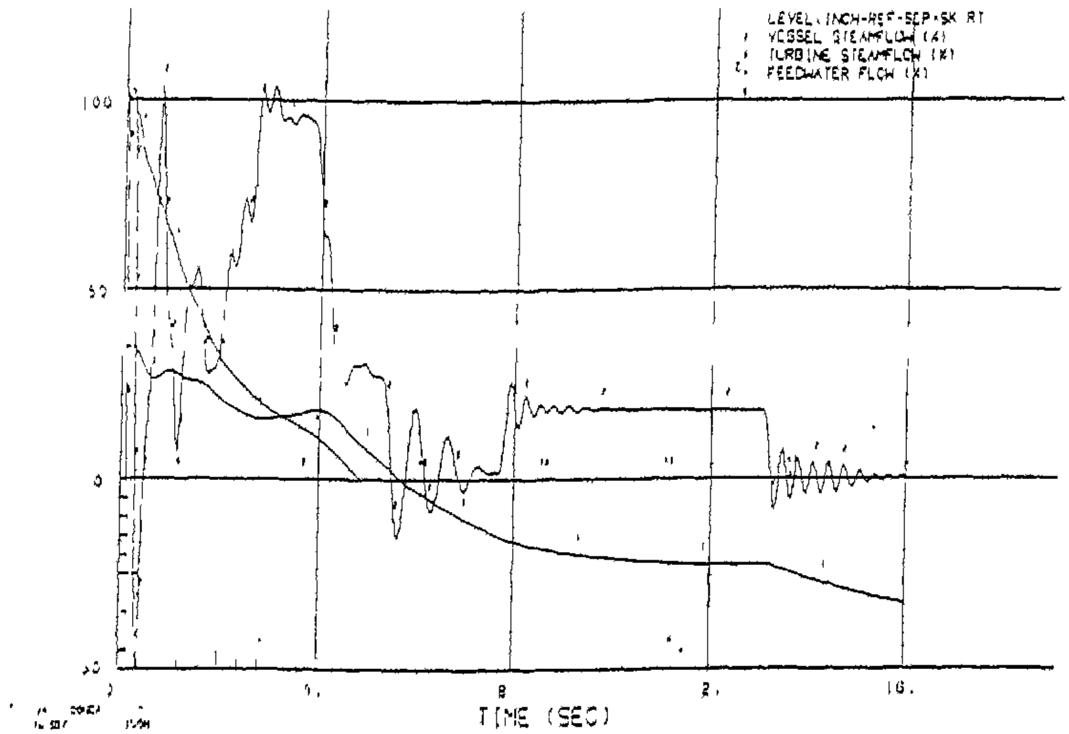


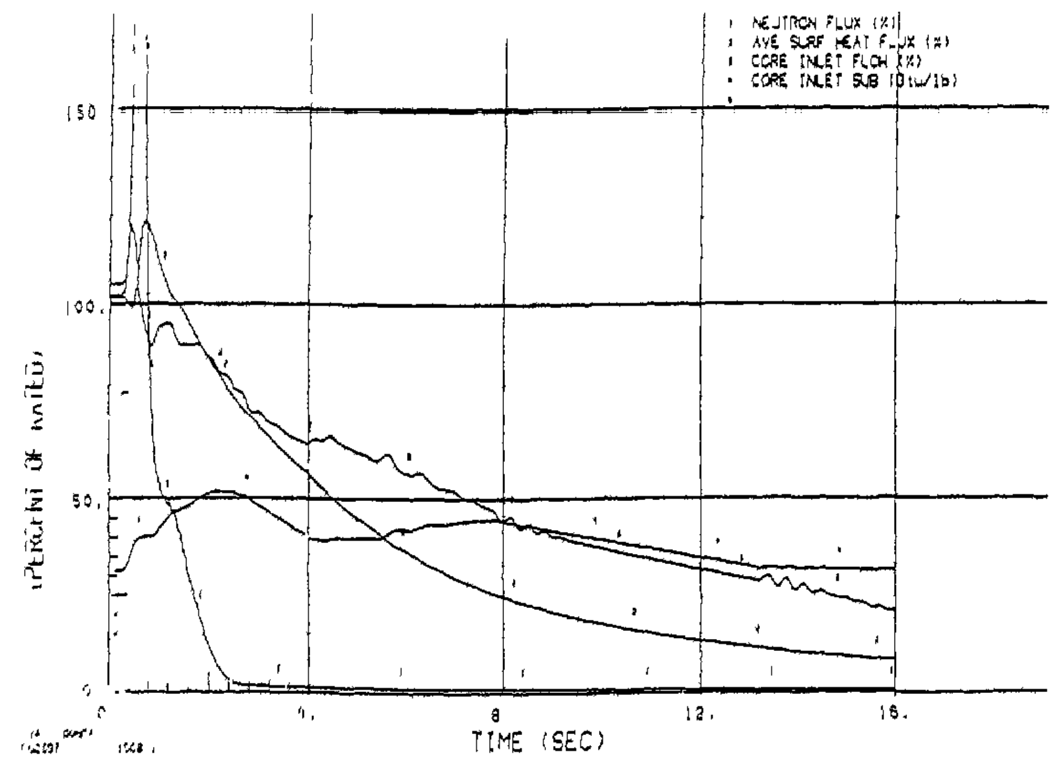
AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

LOAD REJECTION NO BYPASS/EOC-RPT-OOS
100P/105F

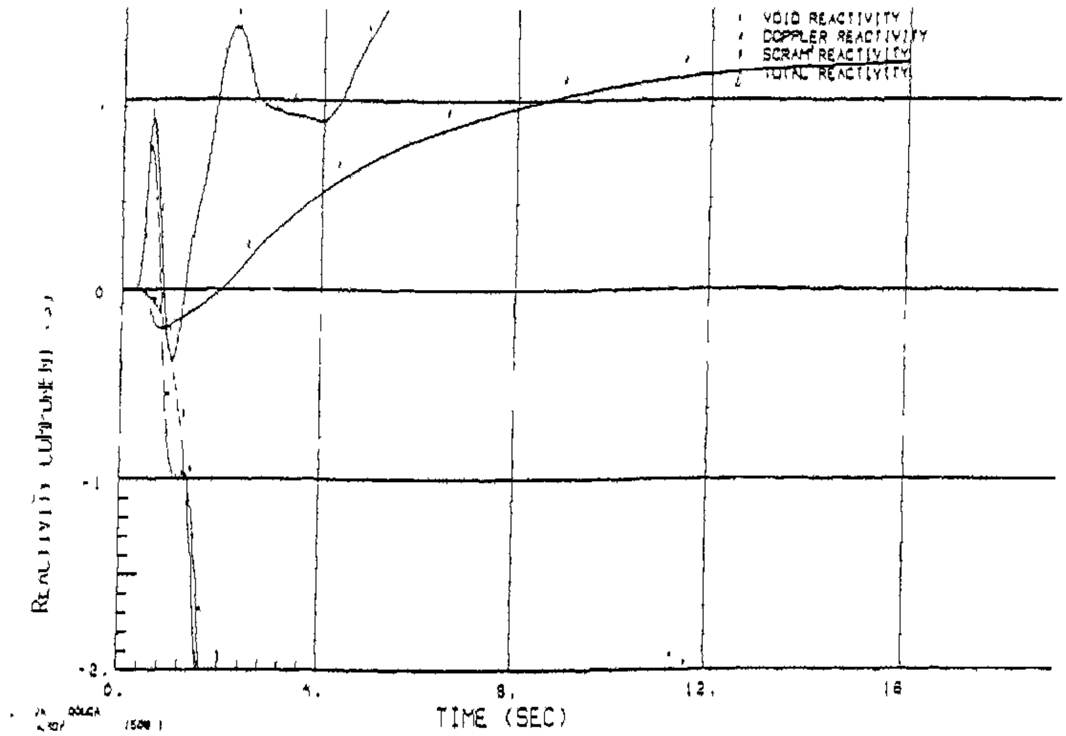
FIGURE 14.5-6

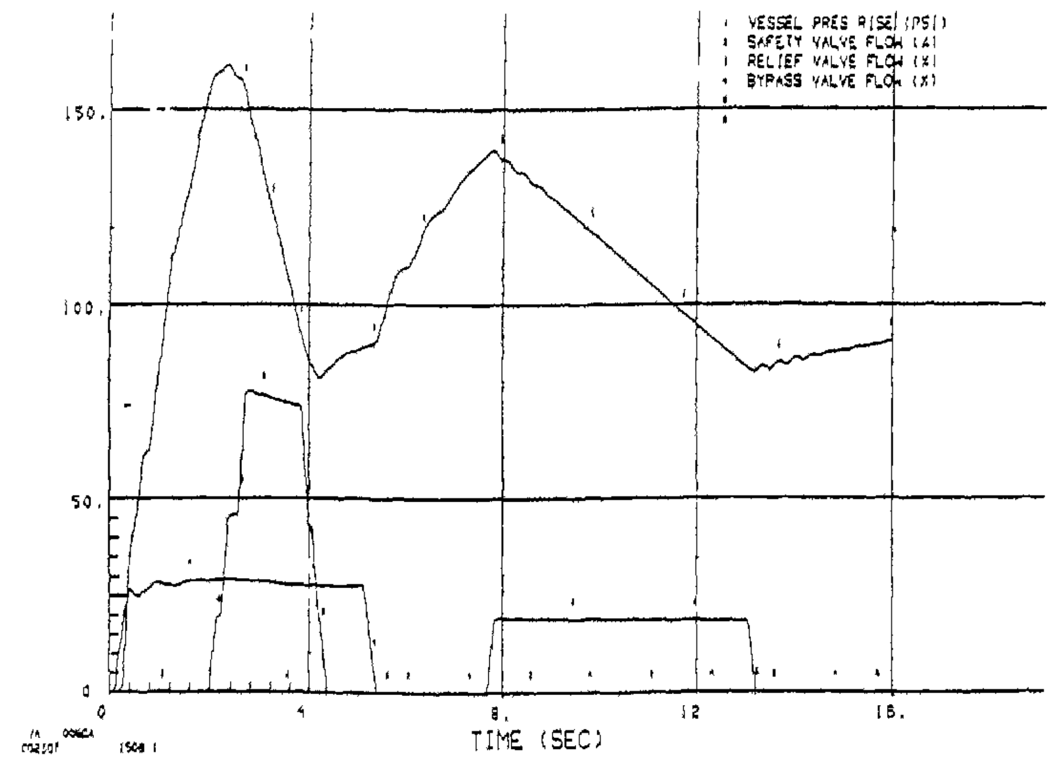




AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT
LOSS OF CONDENSER VACUUM 102P/105F FIGURE 14.5-7a



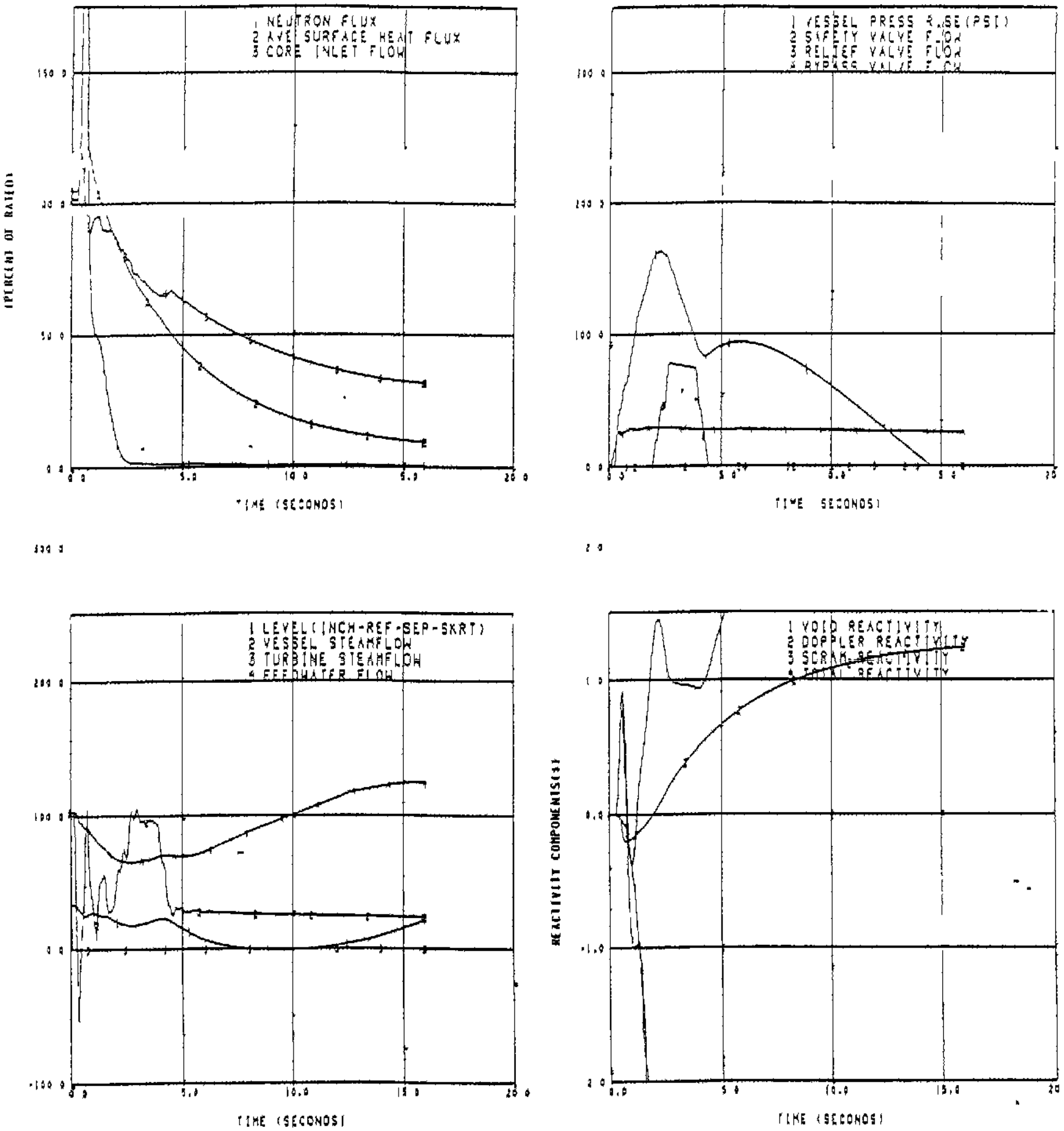


AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

LOSS OF CONDENSER VACUUM
102P/105F

FIGURE 14 5-7b

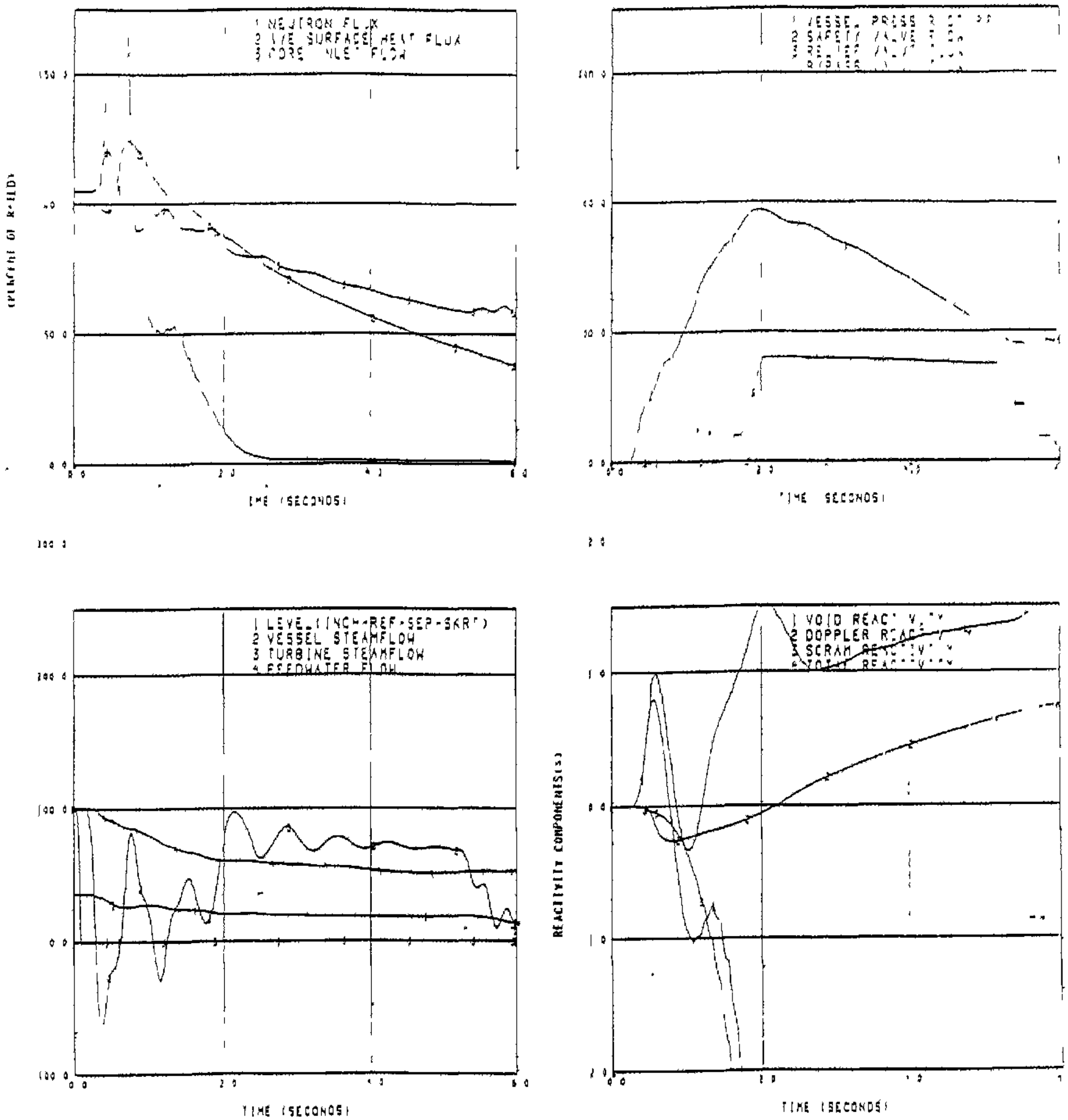


AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

TURBINE STOP VALVE
CLOSURE/TURBINE TRIP
102P/105F

FIGURE 14.5-8

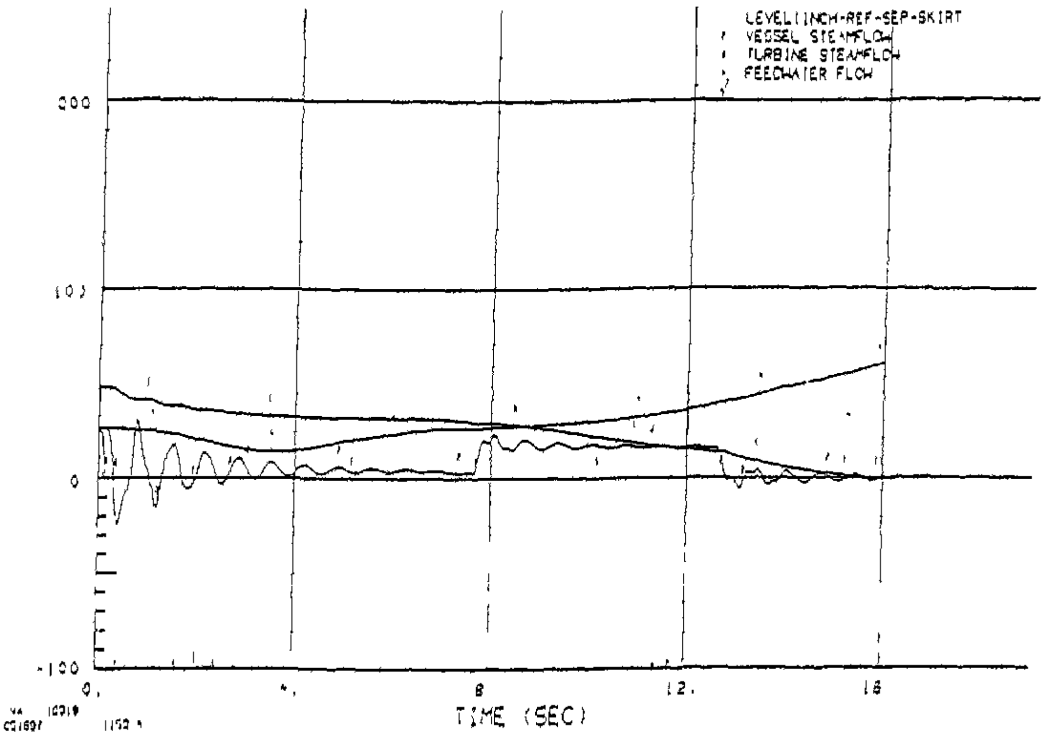


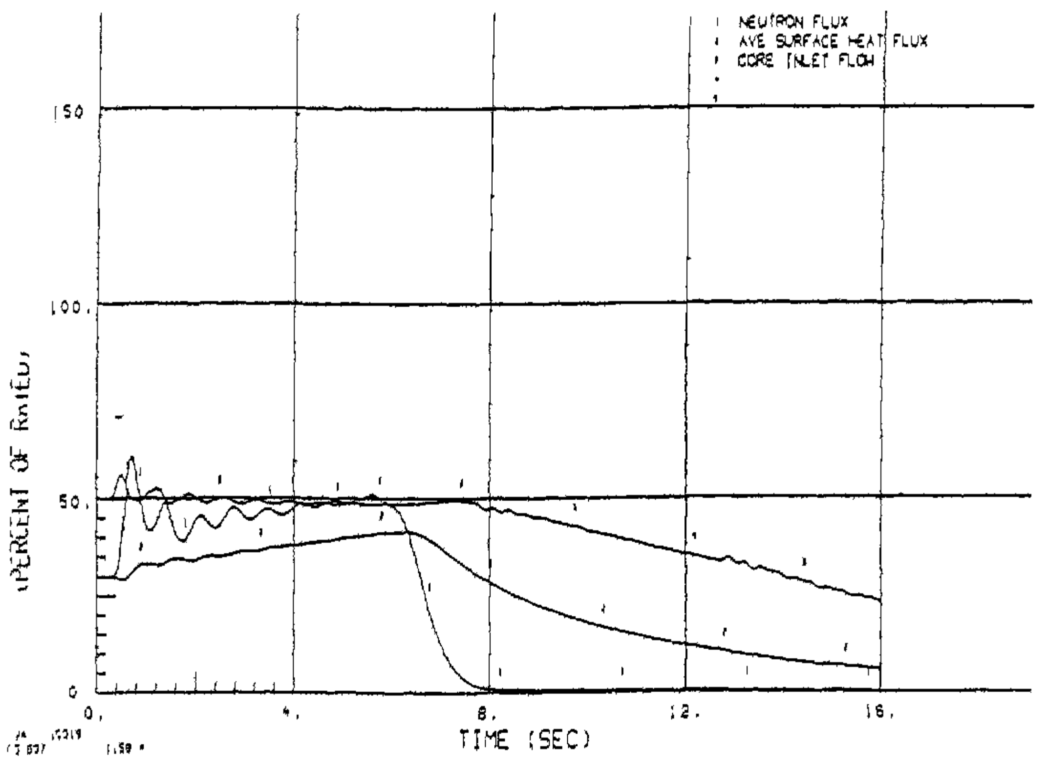
AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

BYPASS FAILURE FOLLOWING TURBINE TRIP
HIGH POWER 100P/105F

FIGURE 14 5-9





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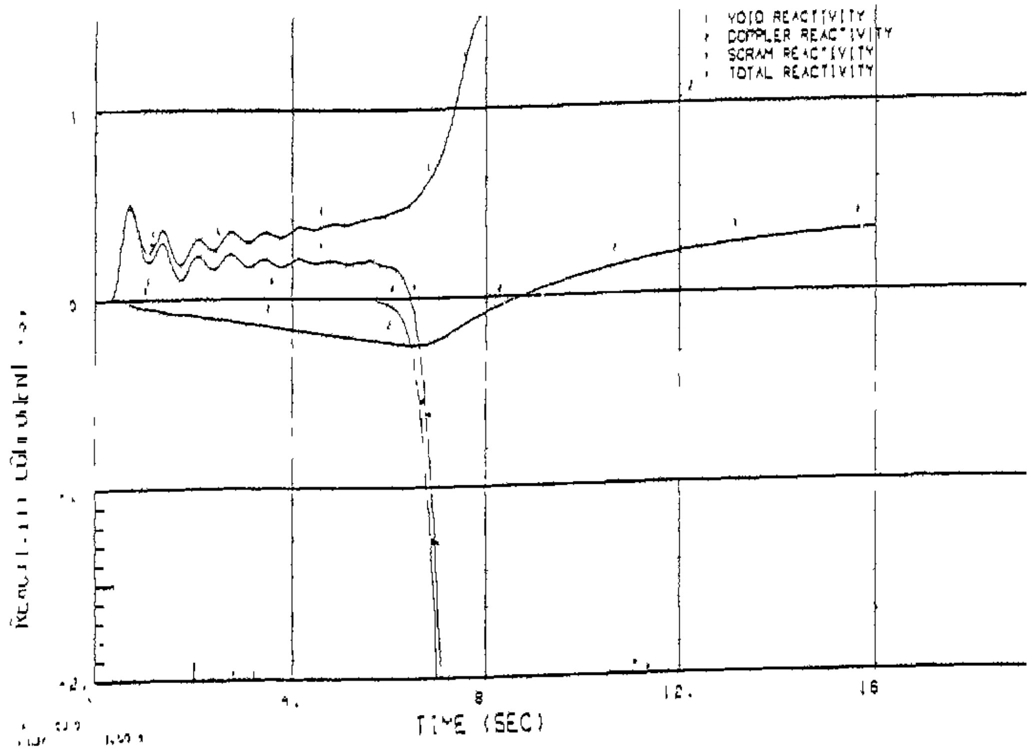
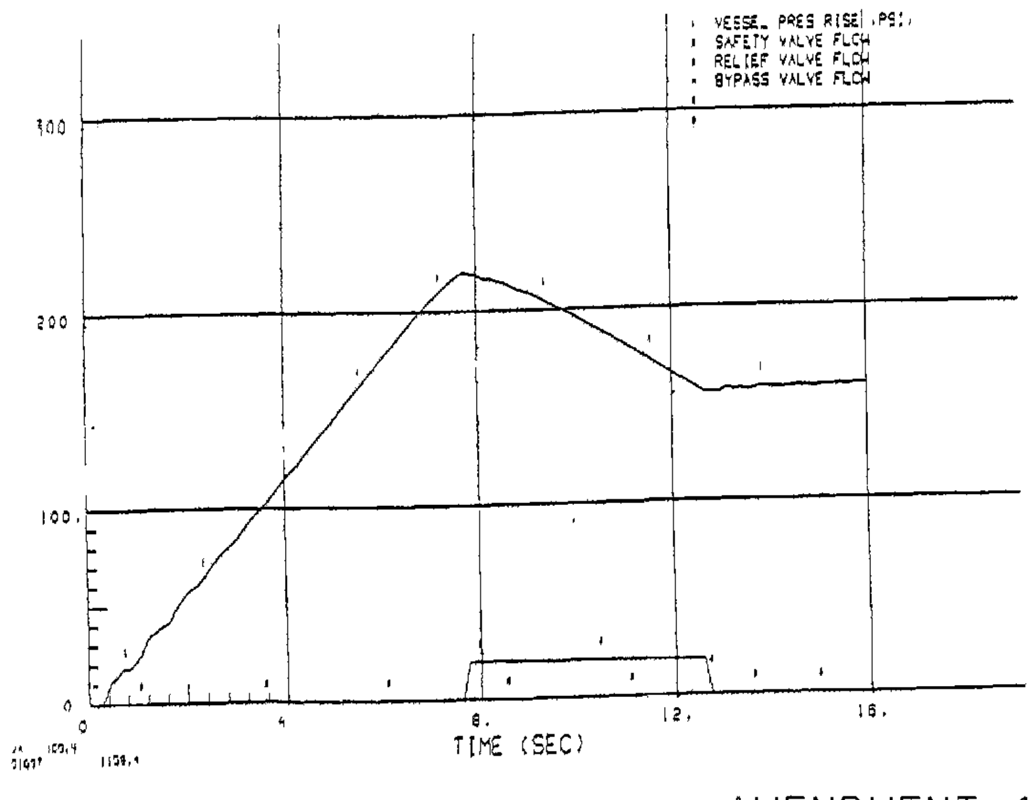
BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

BYPASS FAILURE FOLLOWING TURBINE TRIP
LOW POWER 30P/50F

FIGURE 14.5-10a



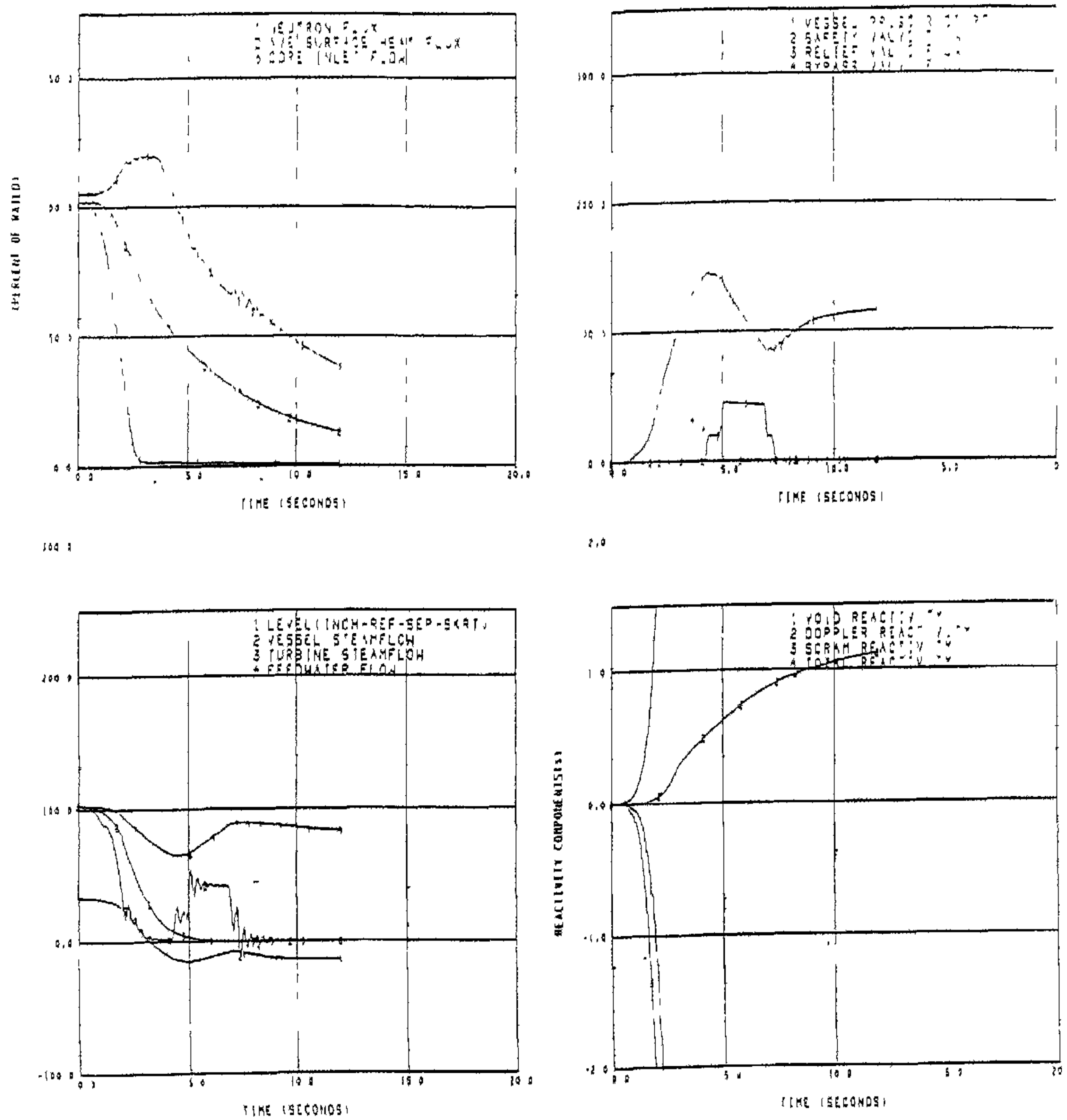


AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

BYPASS FAILURE FOLLOWING TURBINE TRIP
LOW POWER 30P/50F

FIGURE 14.5-10b

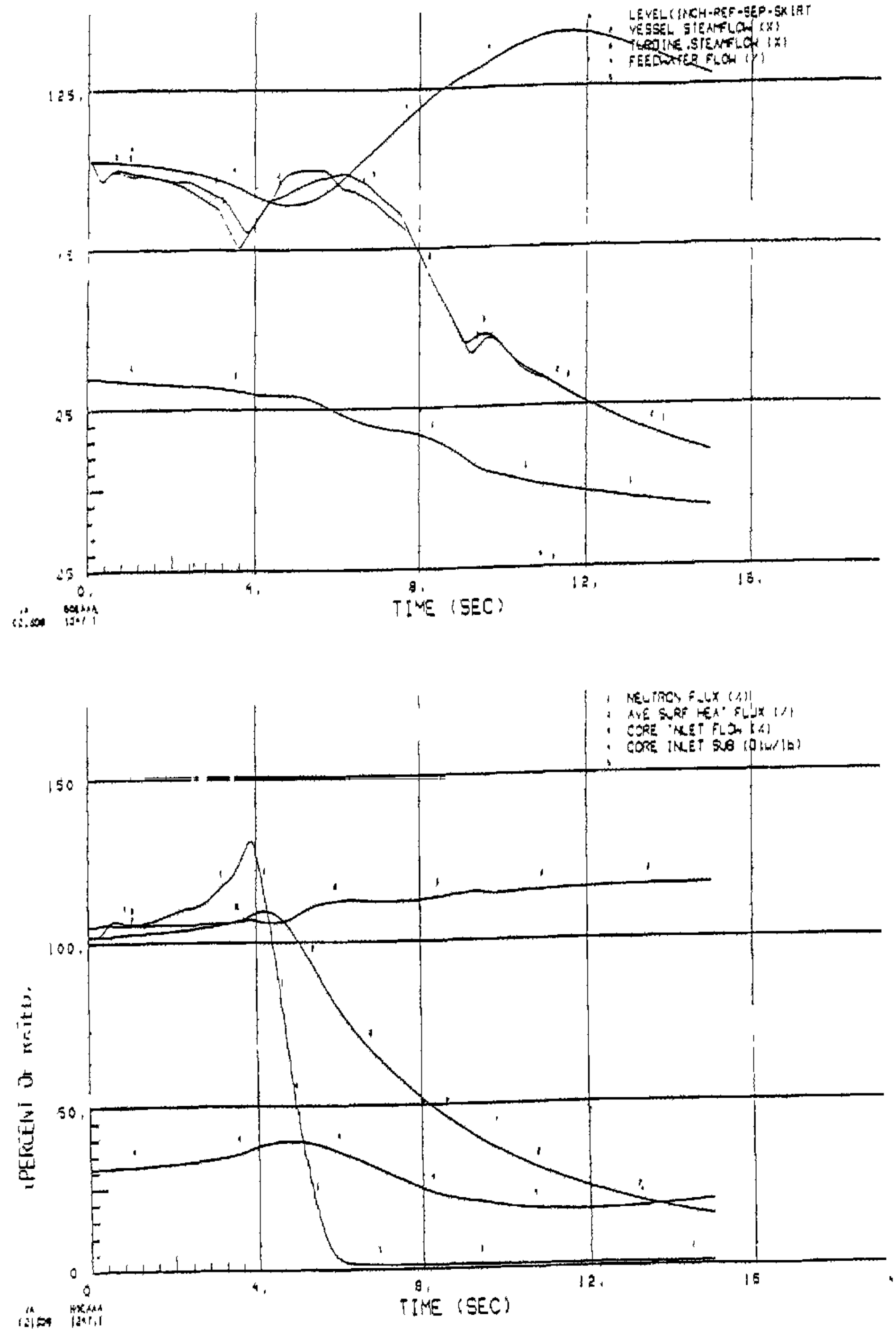


AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

CLOSURE OF ALL MAIN STEAM LINE
ISOLATION VALVES 102P/105F

FIGURE 14.5-11

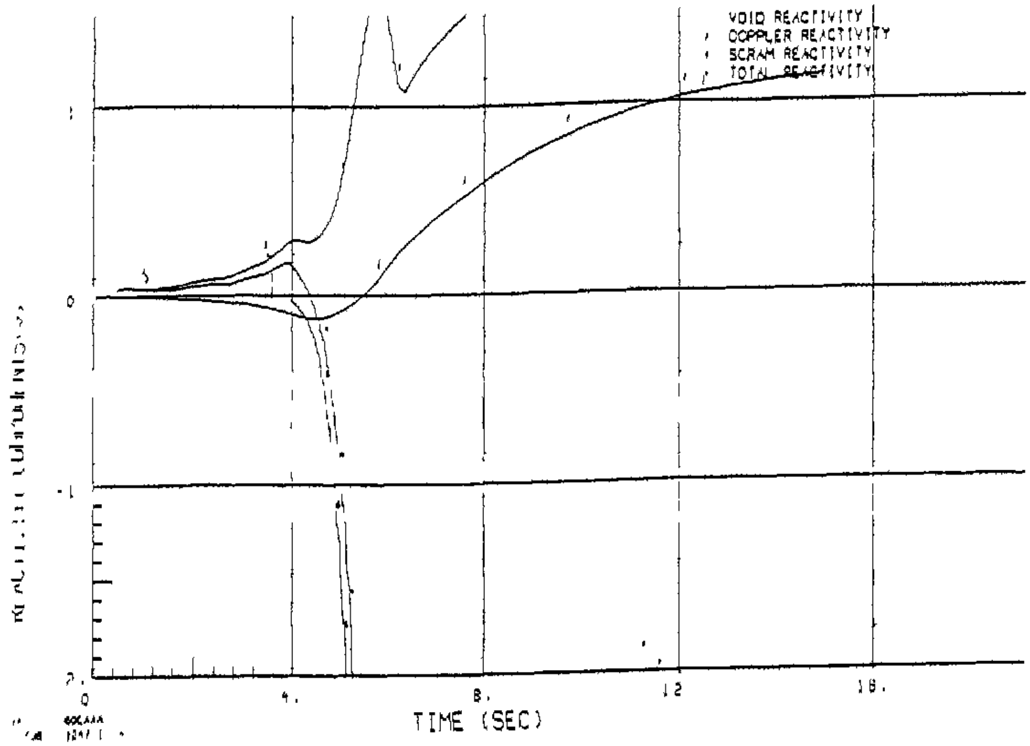


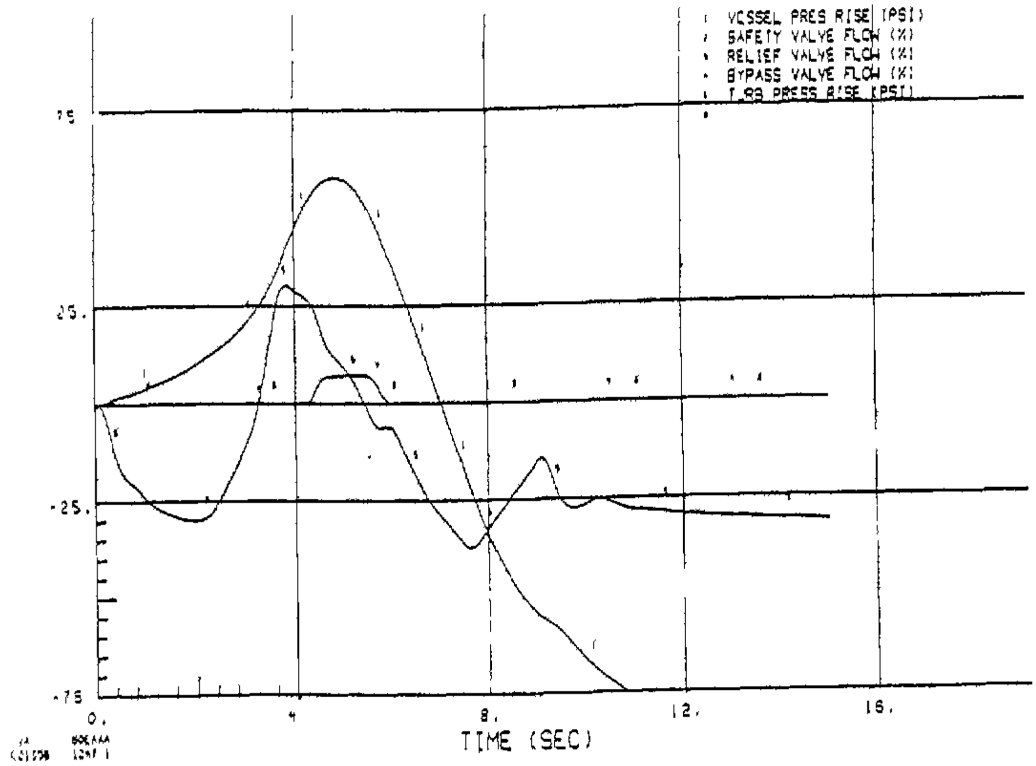
AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

CLOSURE OF ONE MAIN STEAM LINE
ISOLATION VALVE 102P/105F

FIGURE 14.5-12a



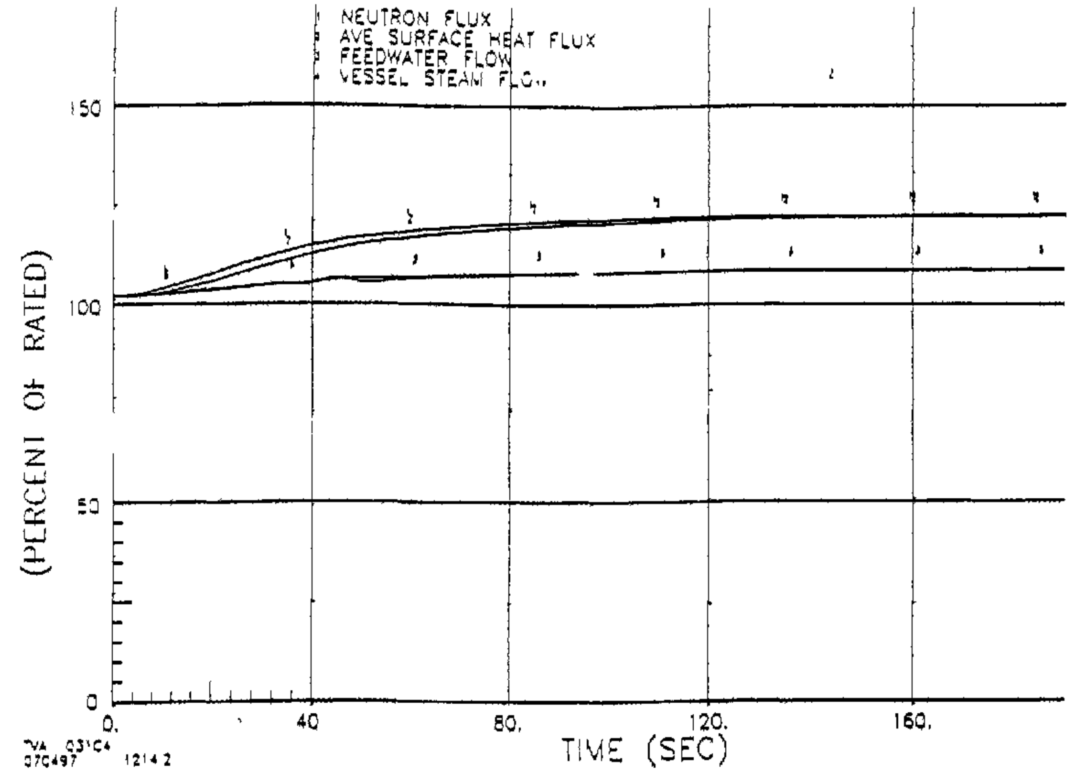


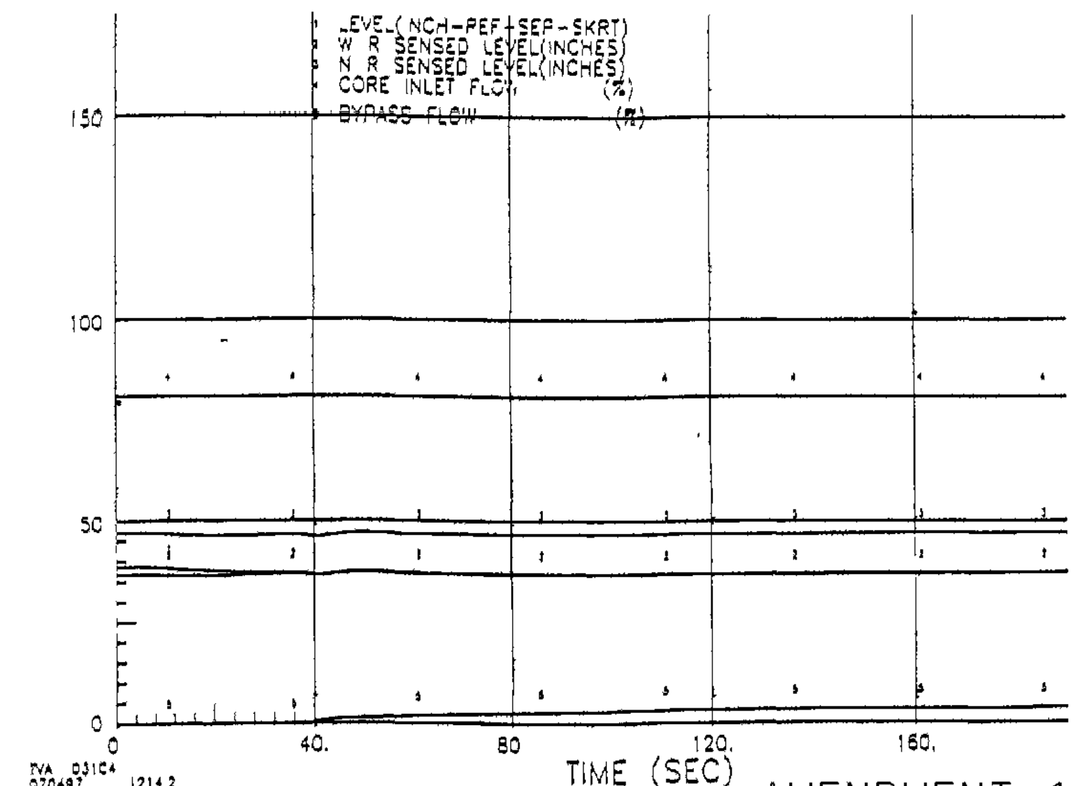
AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

CLOSURE OF ONE MAIN STEAM LINE
ISOLATION VALVE 102P/105F

FIGURE 14.5-12b



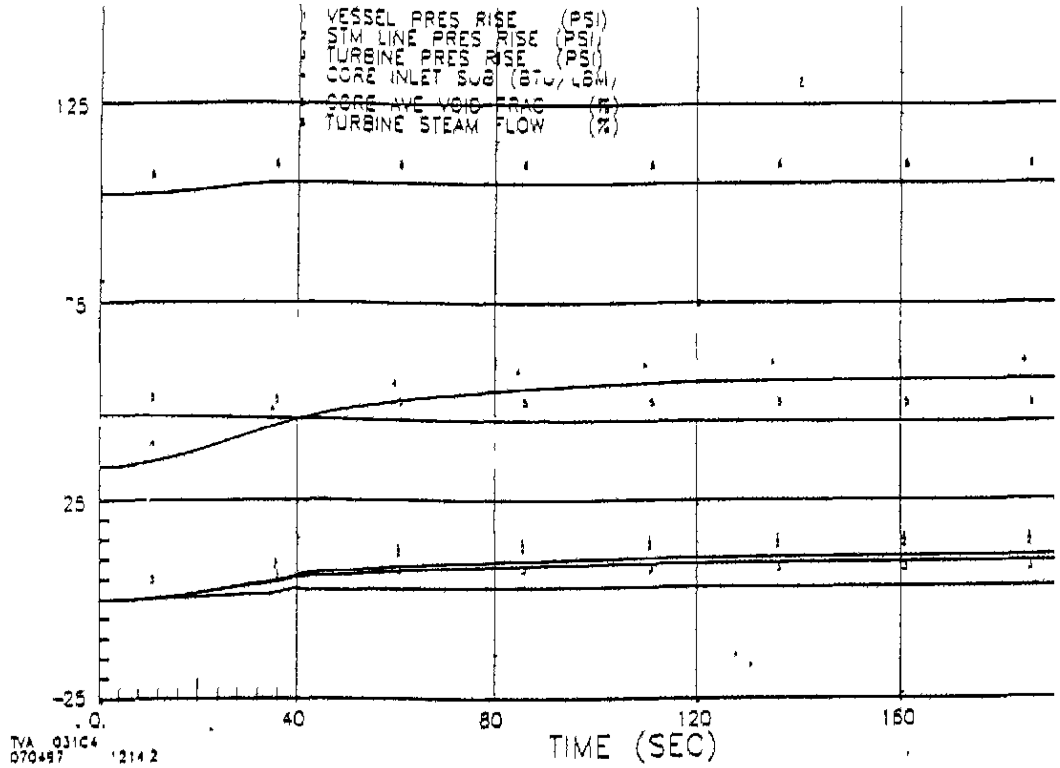


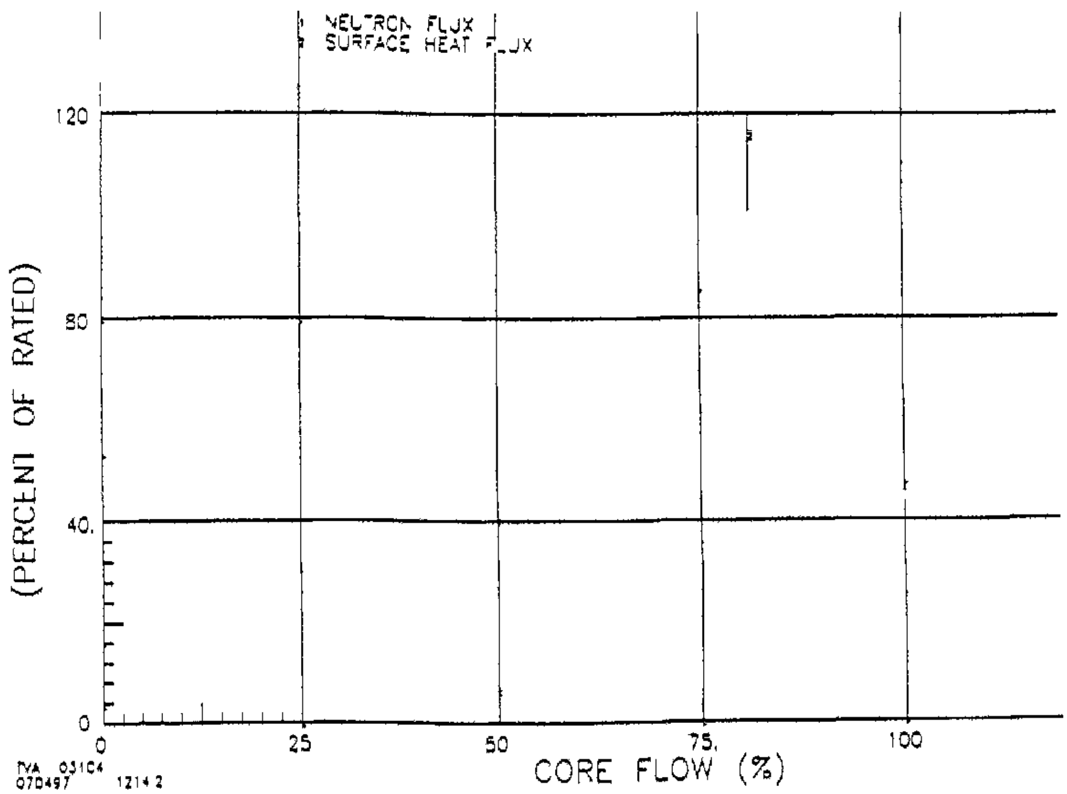
AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

LOSS OF FEEDWATER HEATER
102P/81F

FIGURE 14.5-13a



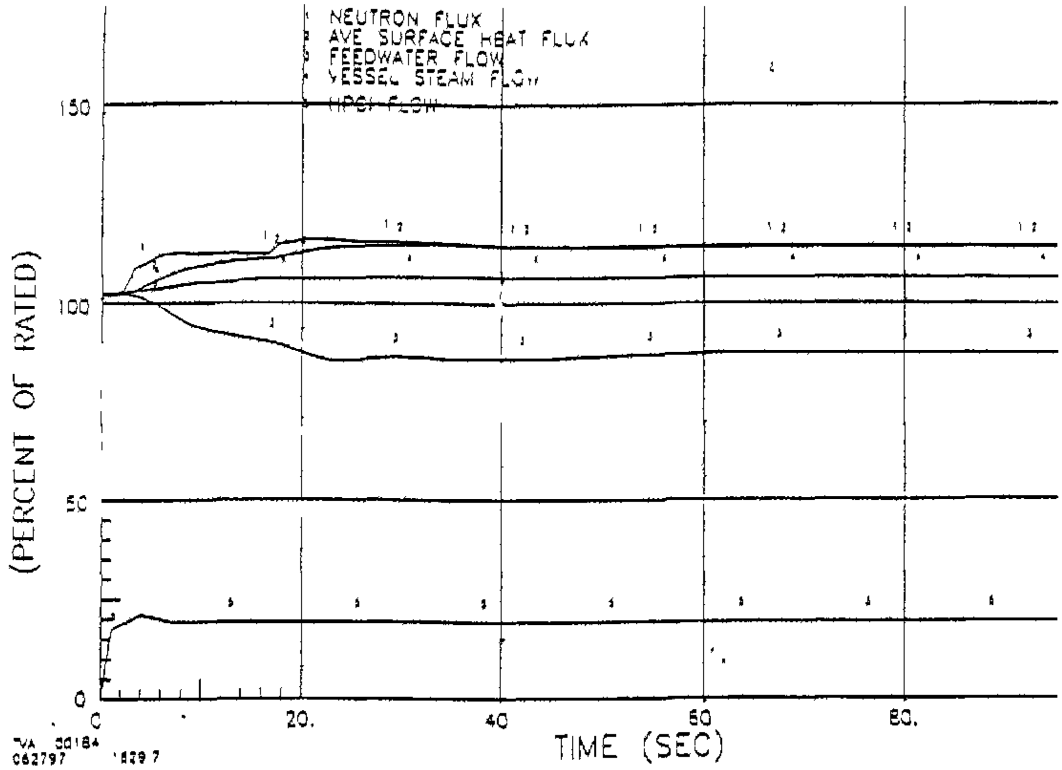


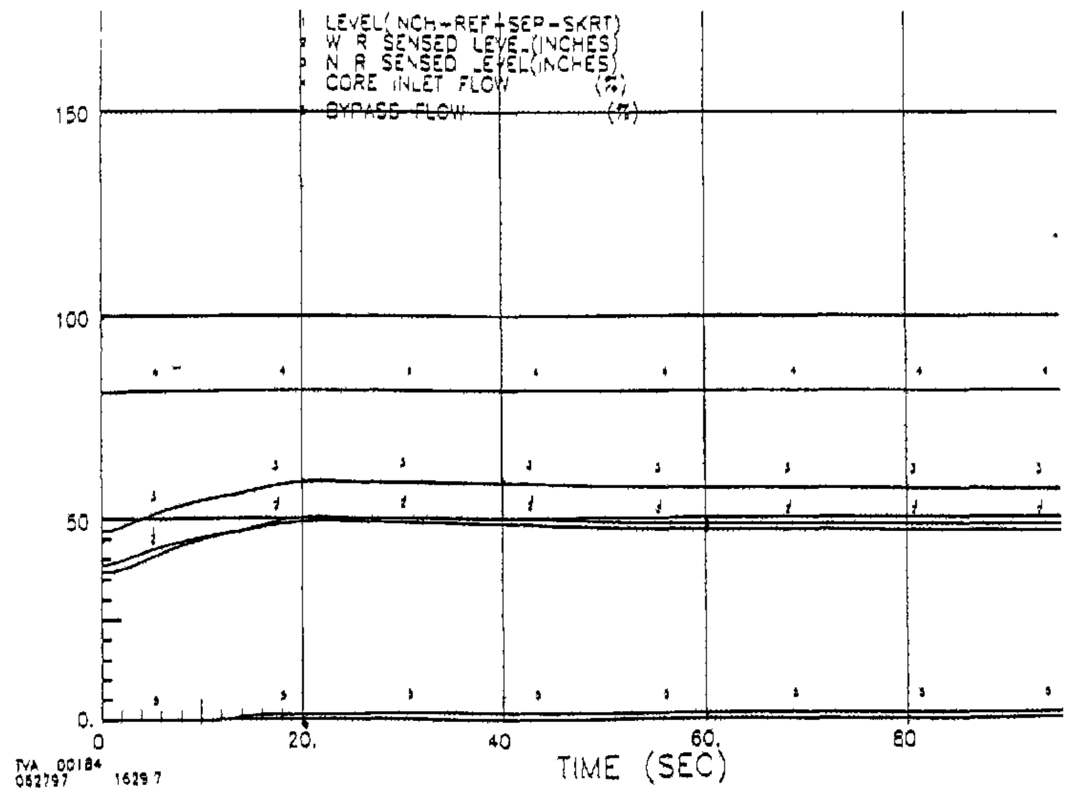
AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

LOSS OF FEEDWATER HEATER
102P/81F

FIGURE 14.5-13b



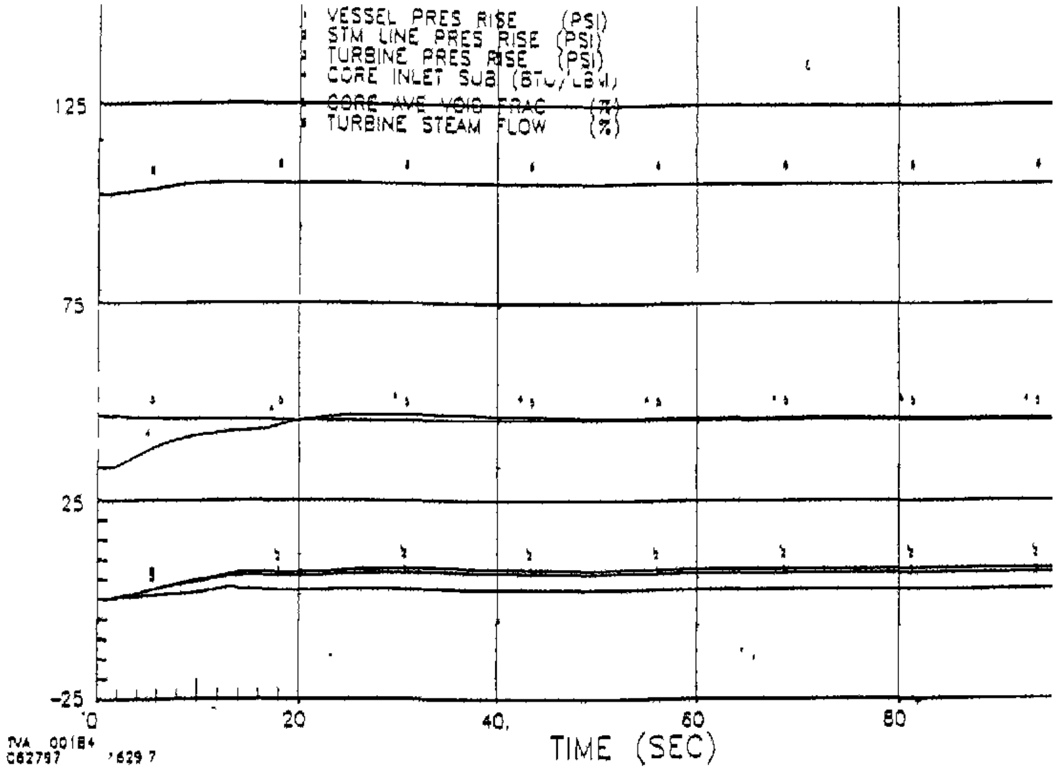


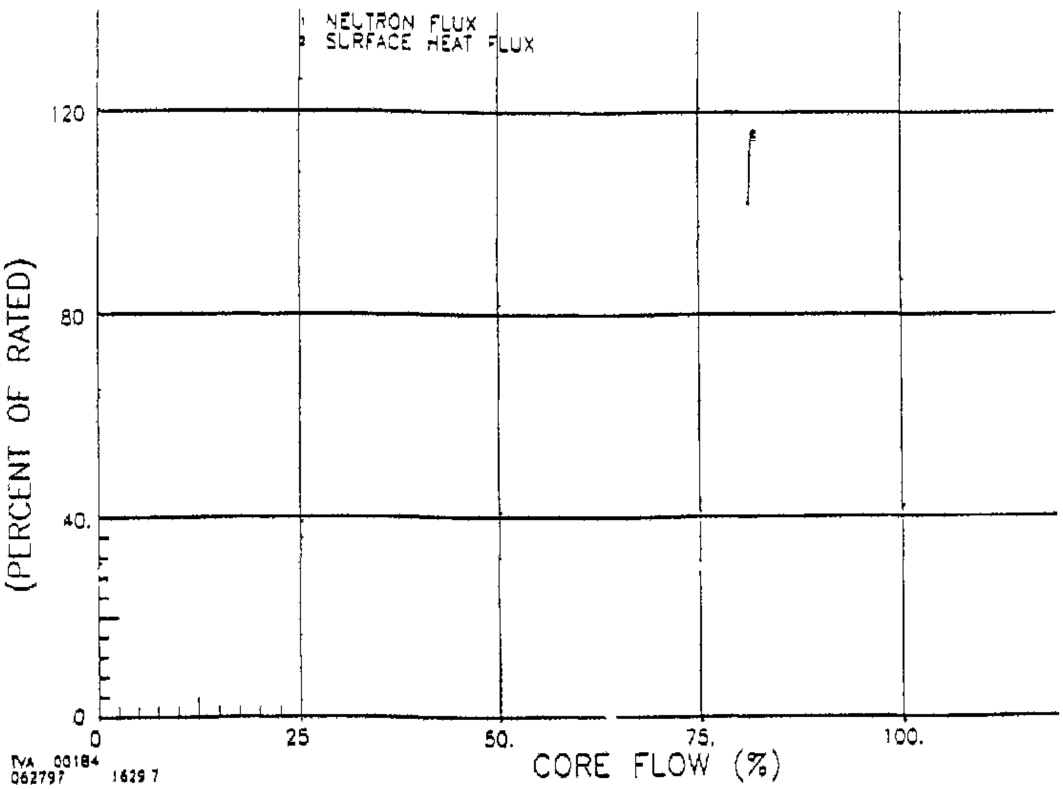
AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

INADVERTENT PUMP START
102P/81F

FIGURE 14.5-14a



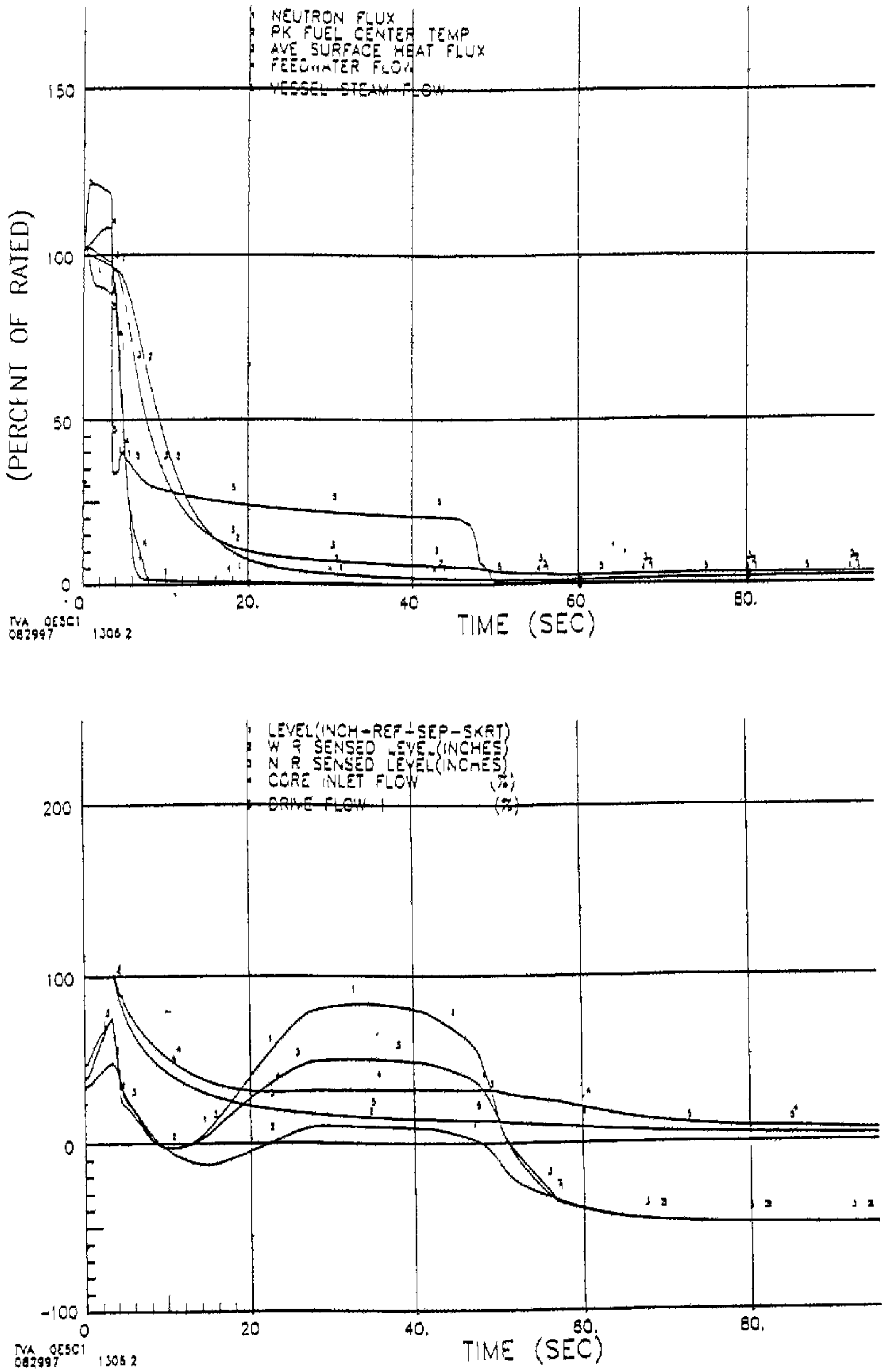


AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

INADVERTENT PUMP START
102P/81F

FIGURE 14.5-14b

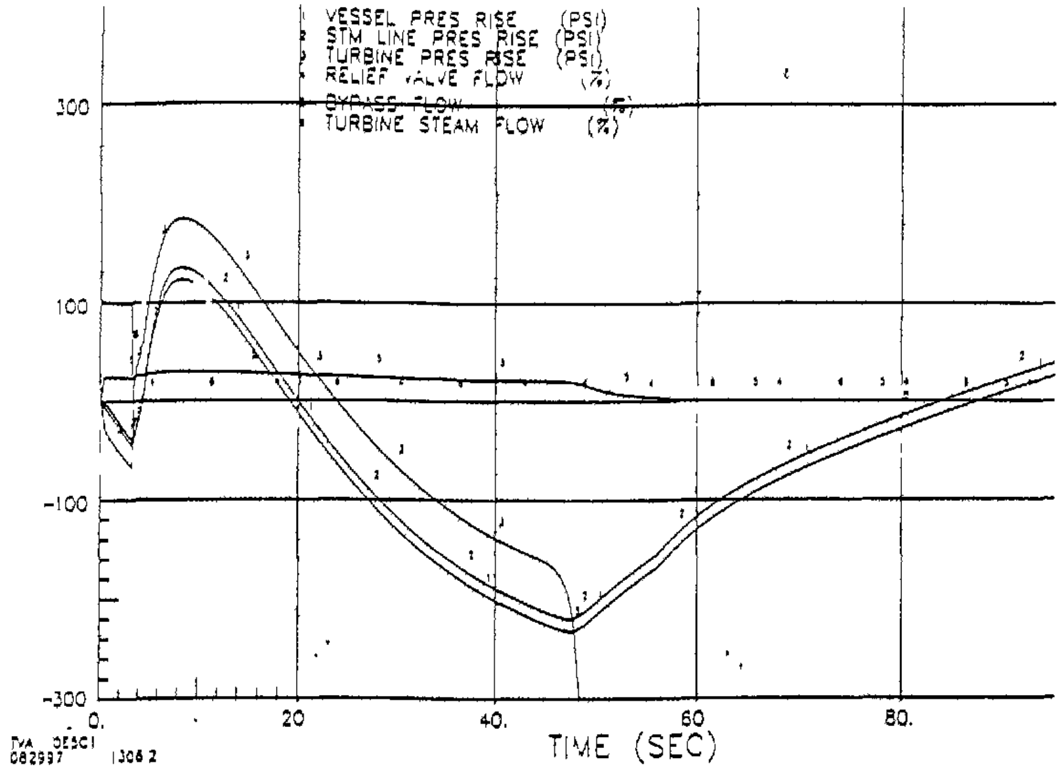


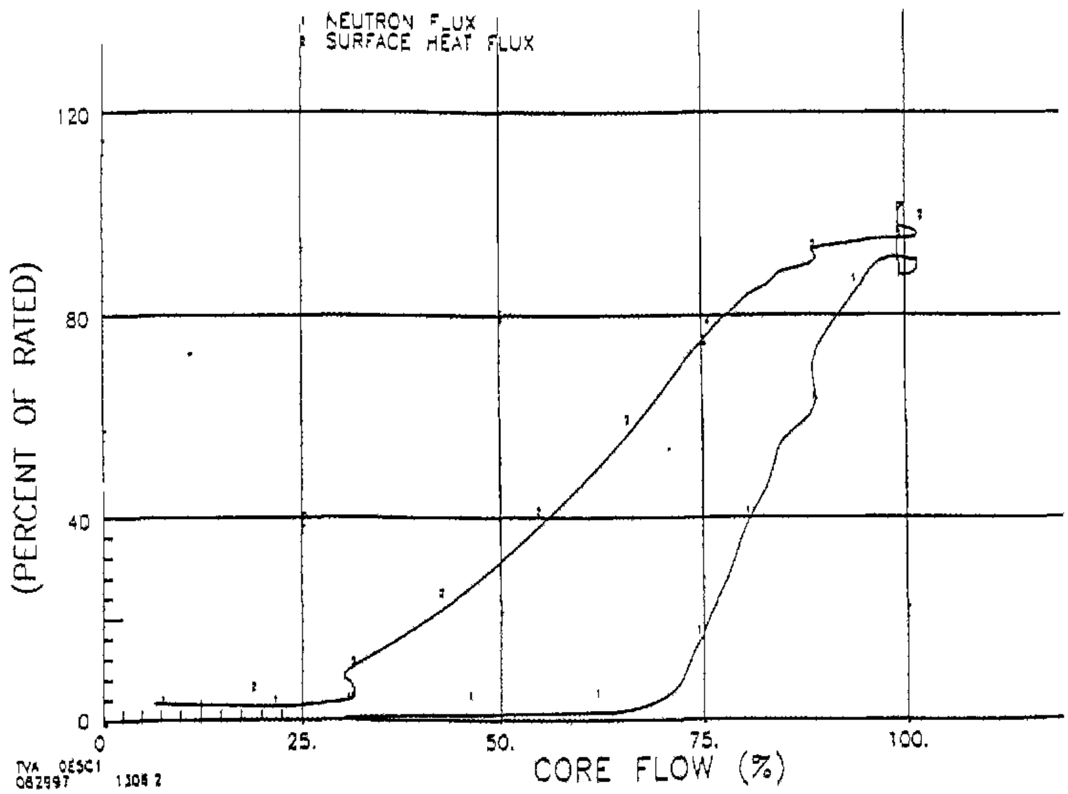
AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

PRESSURE REGULATOR FAILURE OPEN
102P/100F

FIGURE 14.5-15a



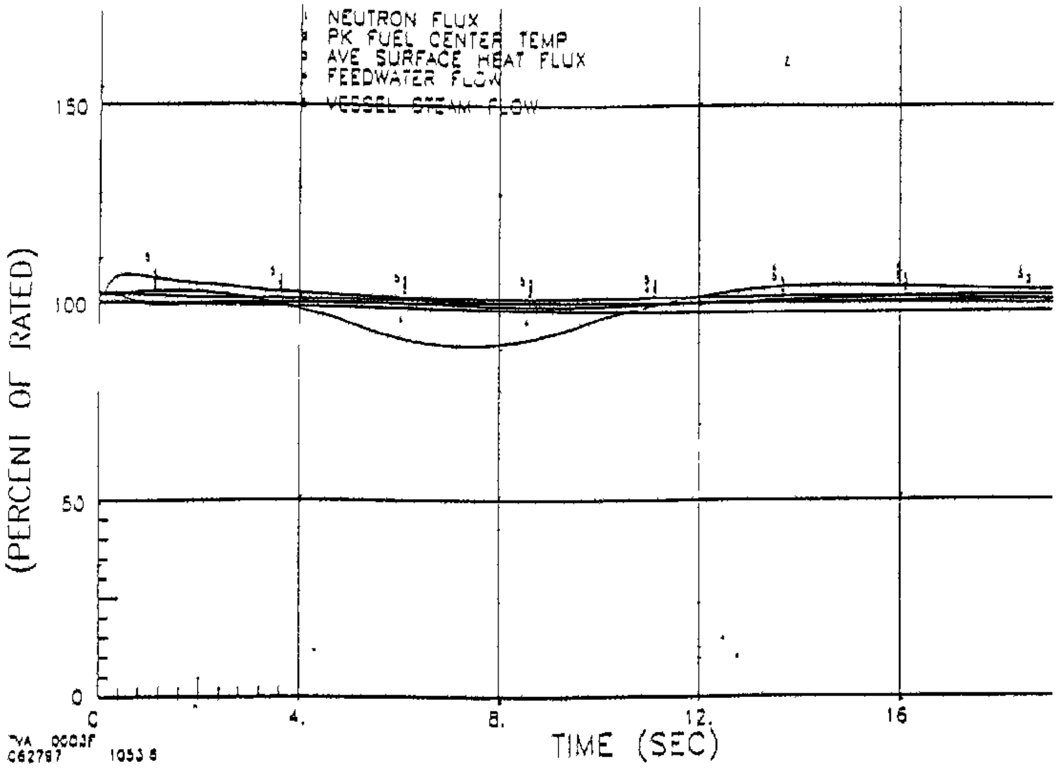


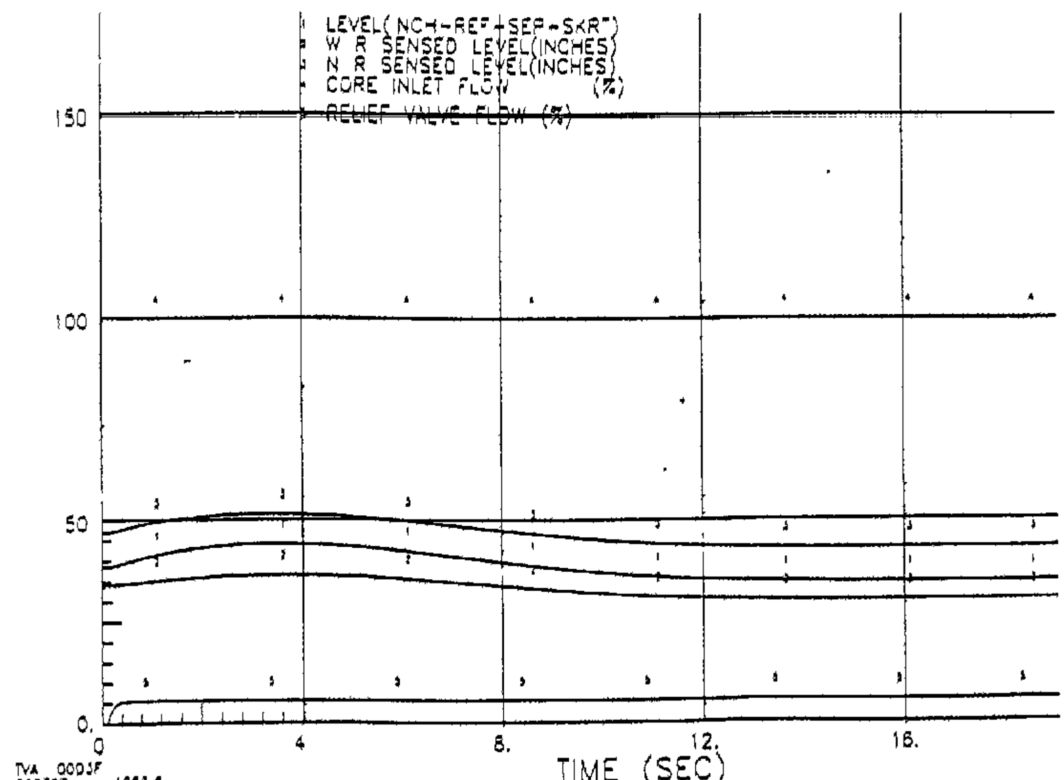
AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

PRESSURE REGULATOR FAILURE OPEN
102P/100F

FIGURE 14.5-15b



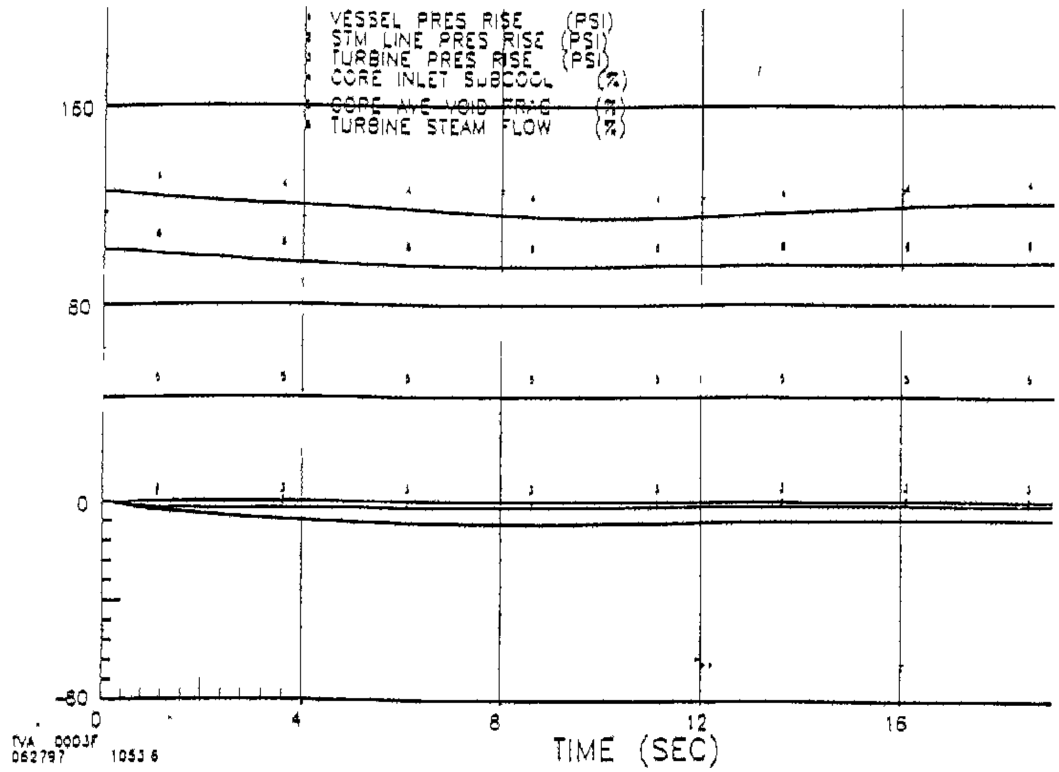


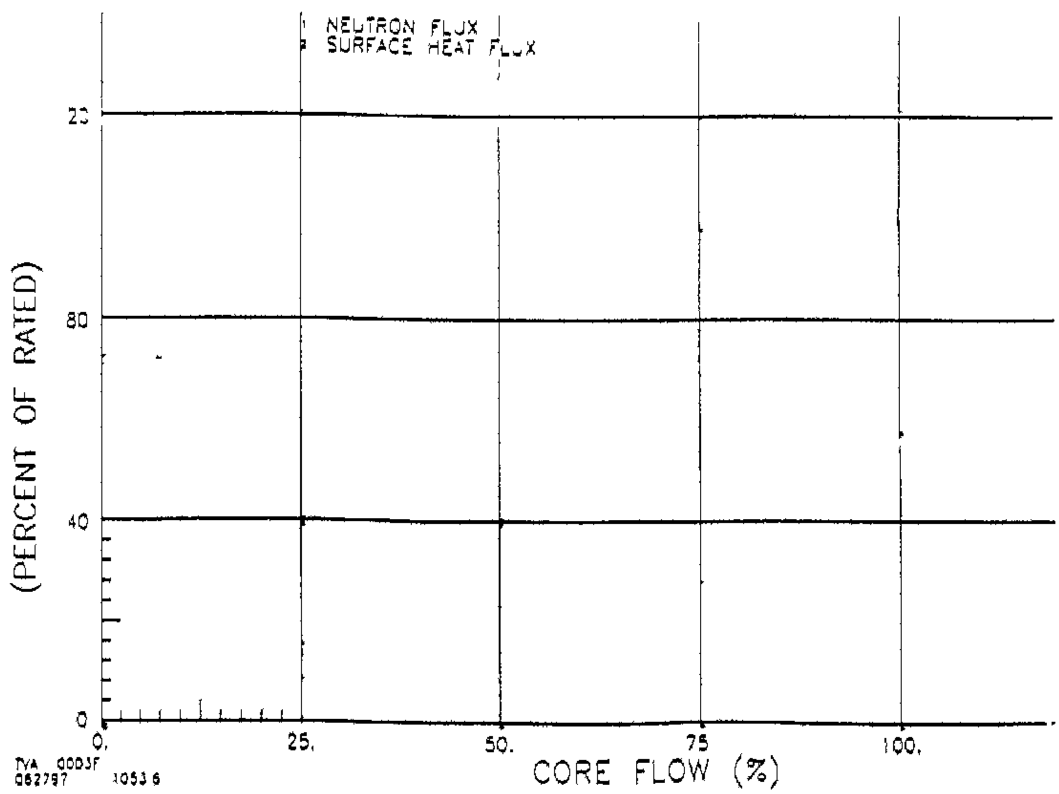
AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

INADVERTENT OPENING OF A RELIEF VALVE
102P/100F

FIGURE 14.5-16a



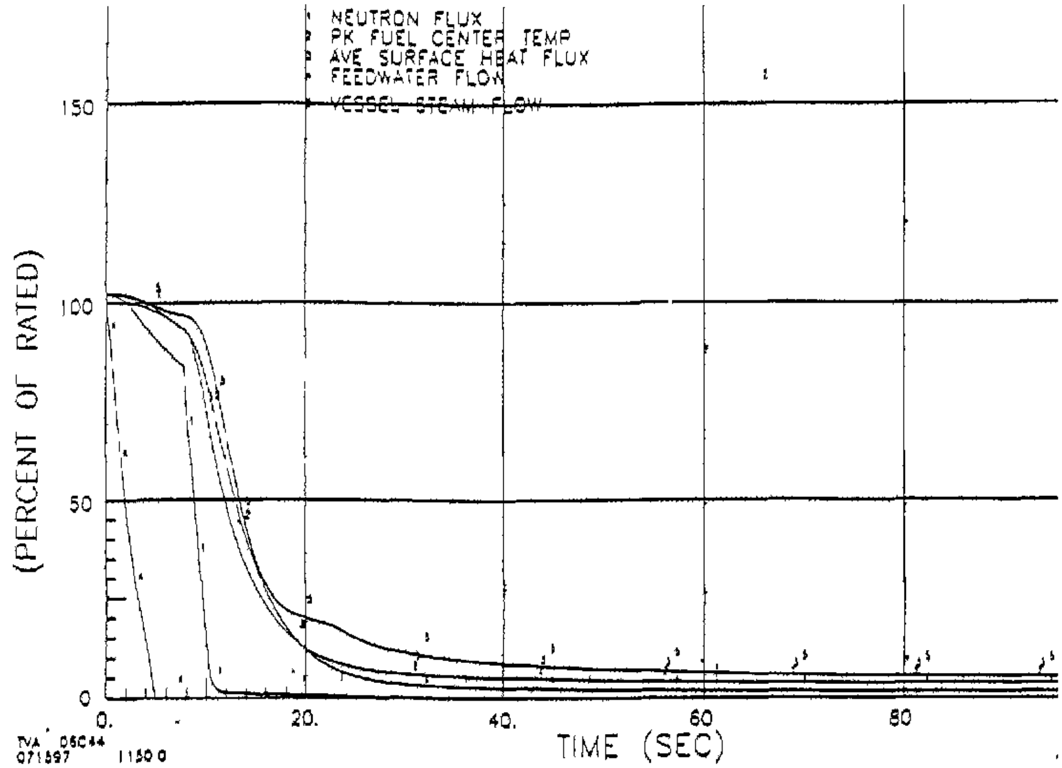


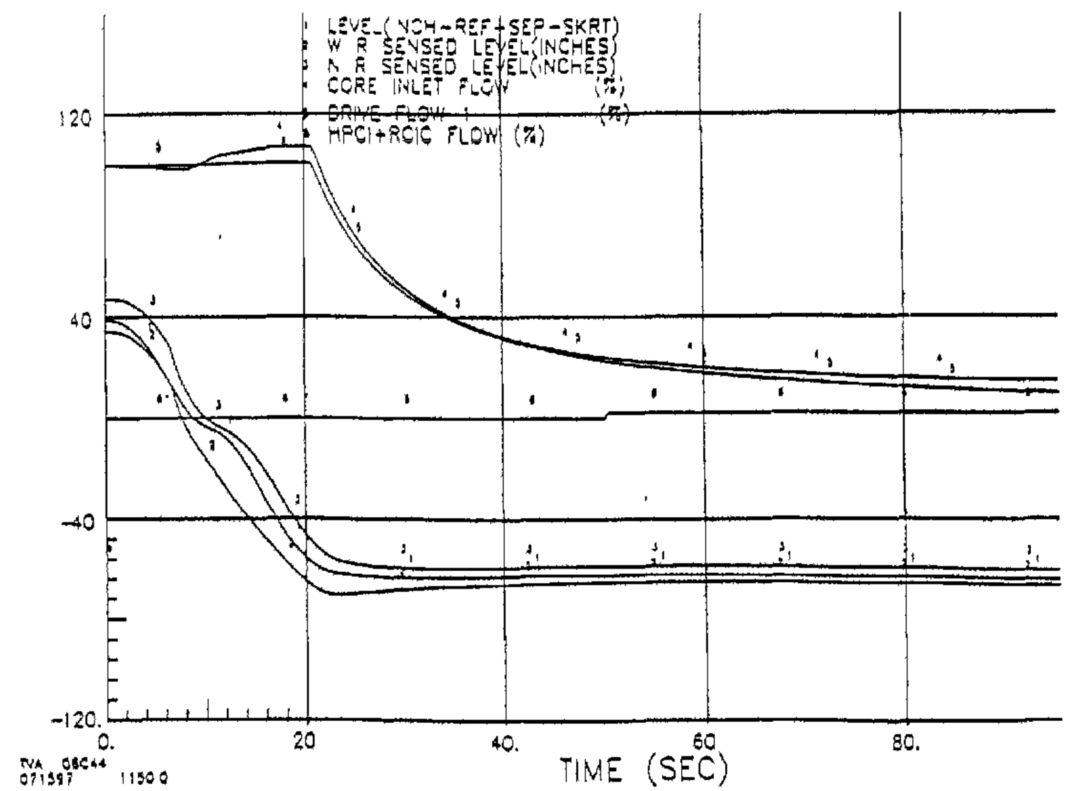
AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

INADVERTENT OPENING OF A RELIEF VALVE
102P/100F

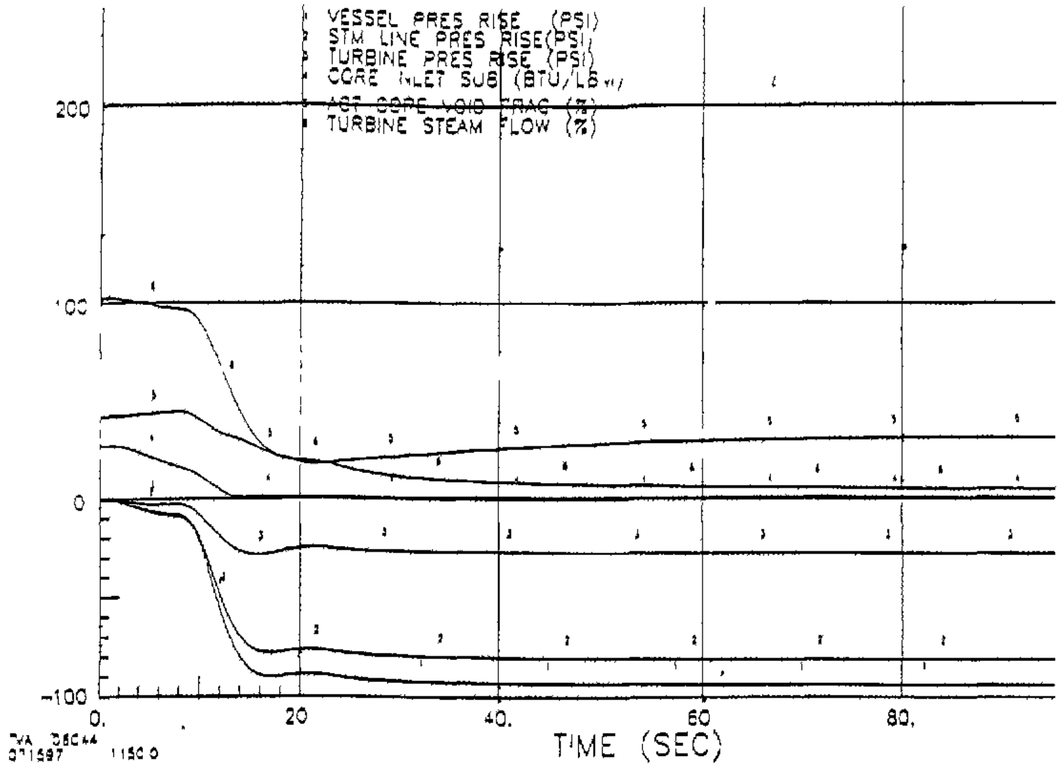
FIGURE 14.5-16b

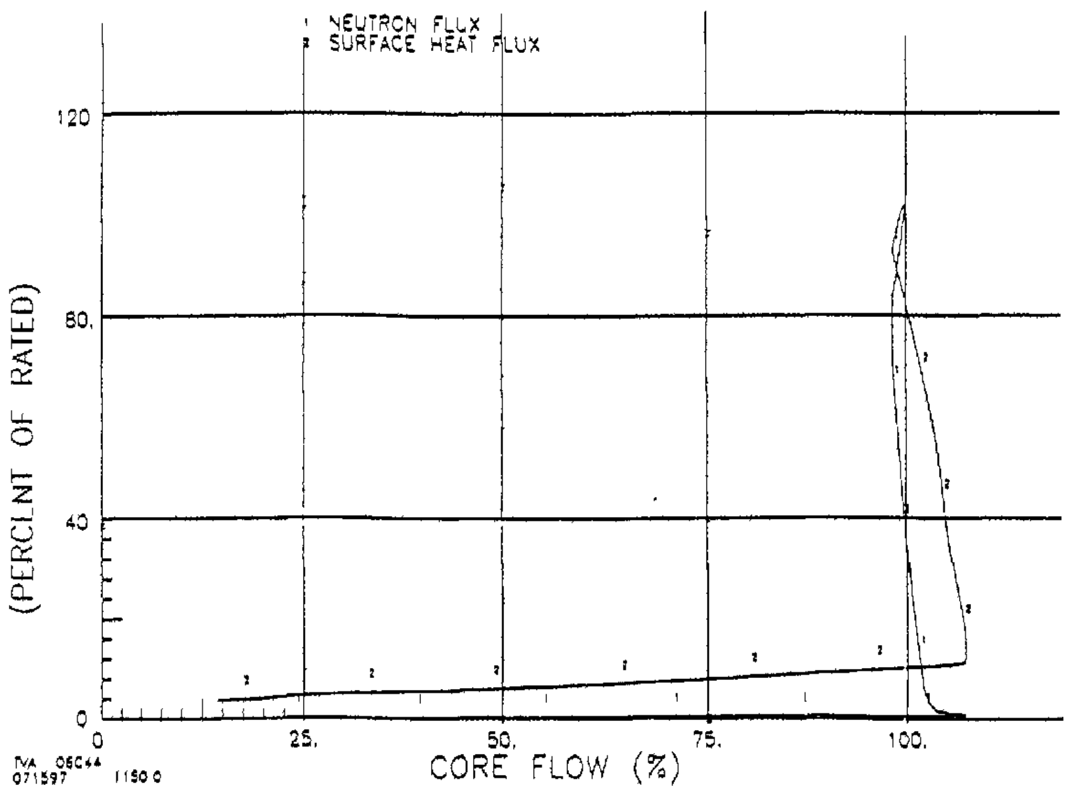




AMENDMENT 17

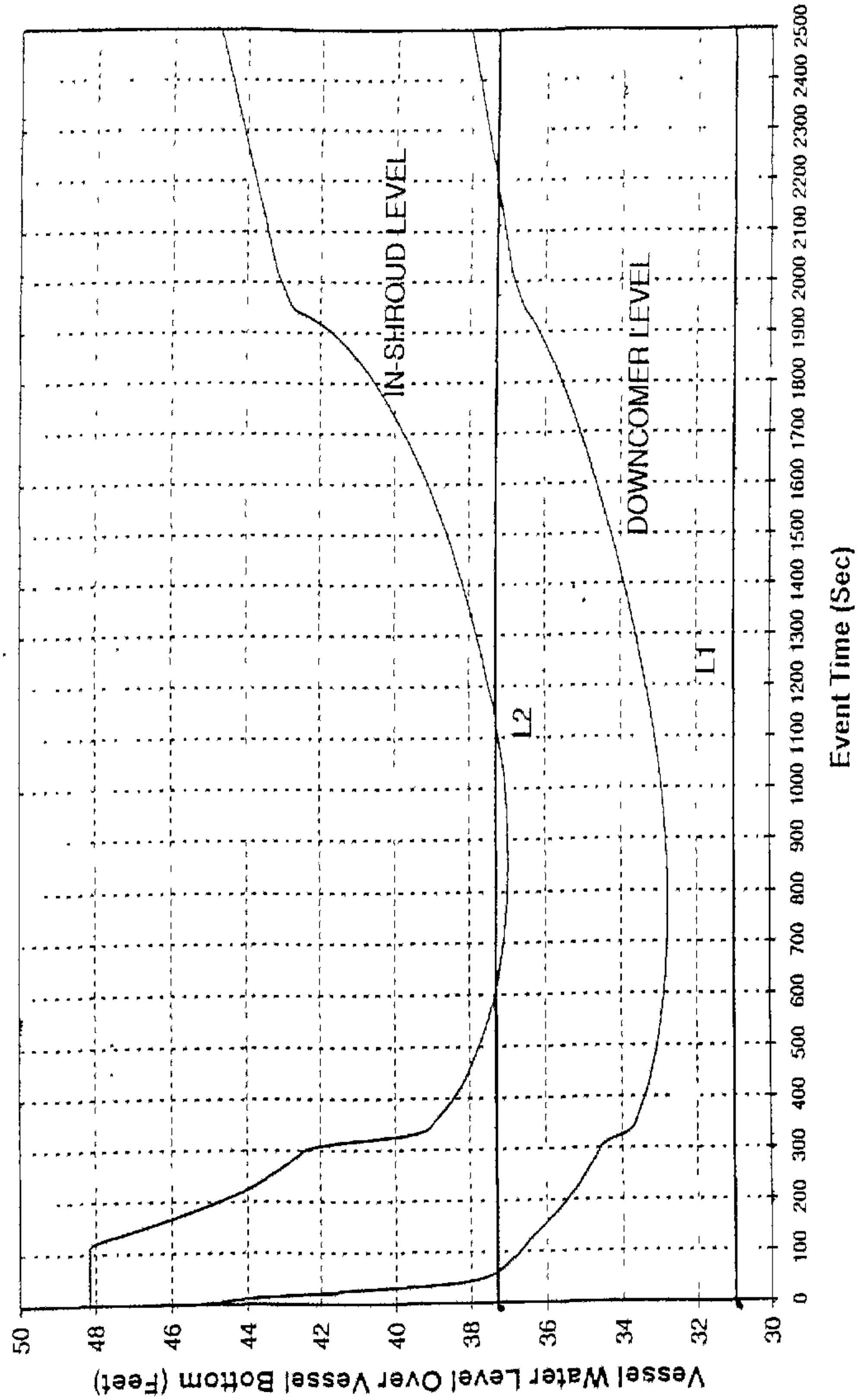
BROWNS FERRY NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT
LOSS OF FEEDWATER FLOW SHORT TERM 102P/100F FIGURE 14.5-17a





AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT
LOSS OF FEEDWATER FLOW SHORT TERM 102P/100F
FIGURE 14.5-17b

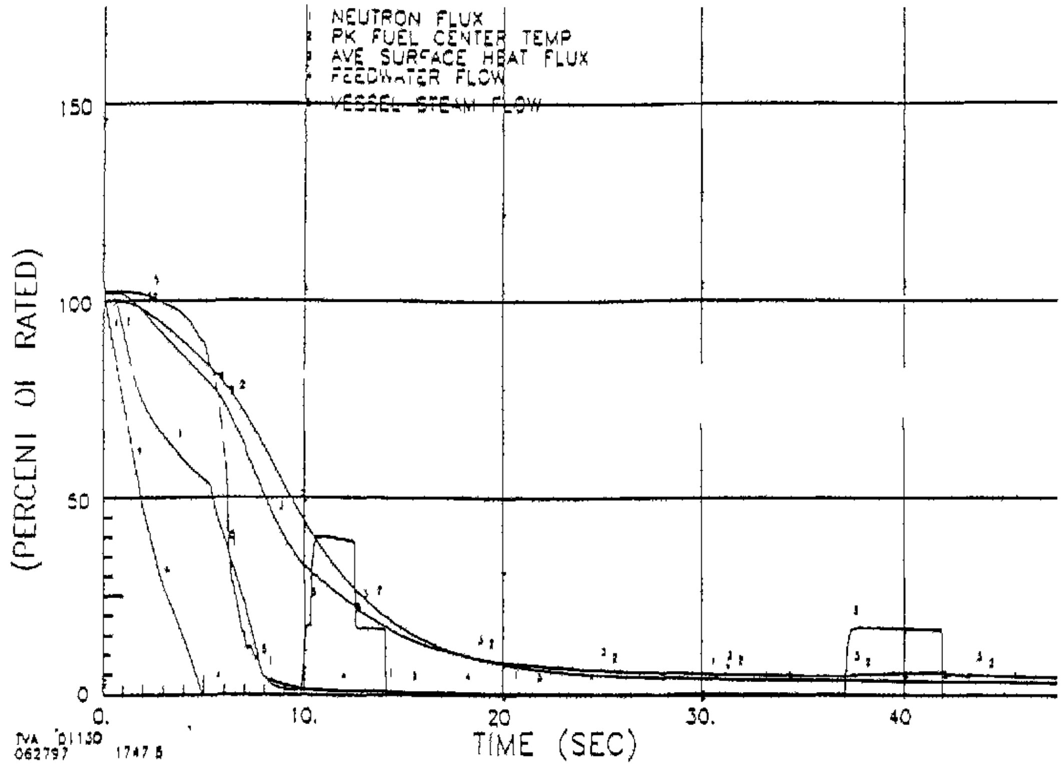


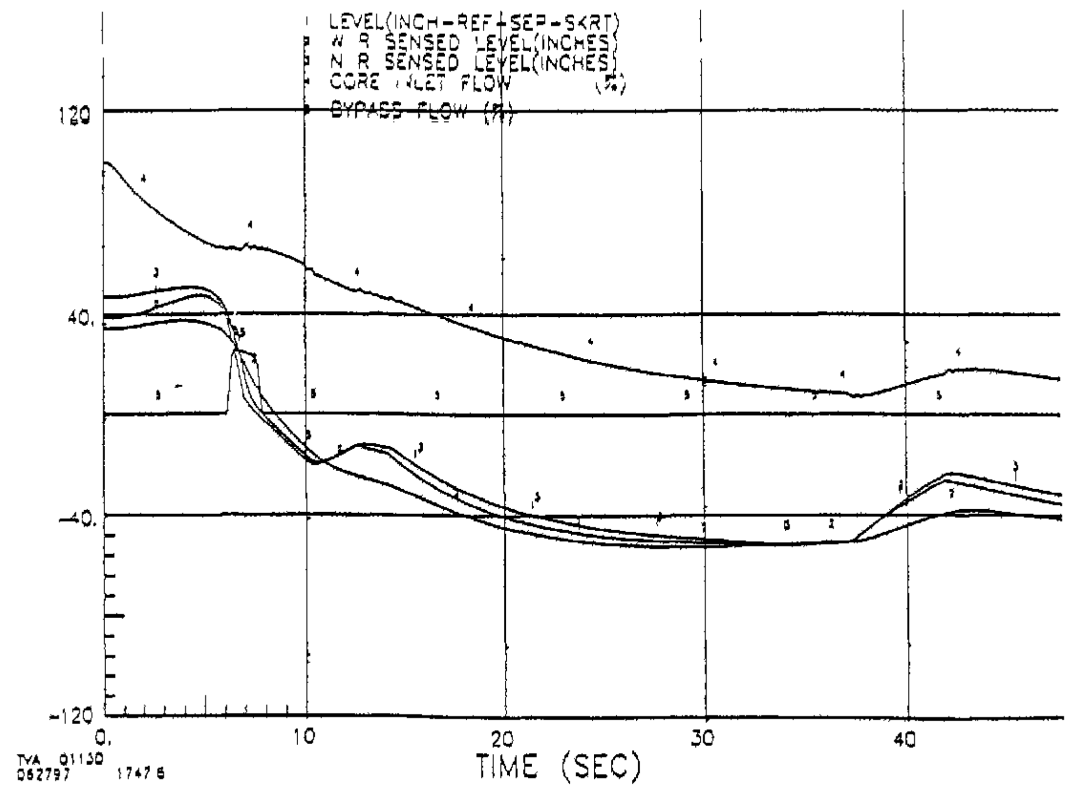
AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

LOSS OF FEEDWATER FLOW
LONG TERM 102P/100F

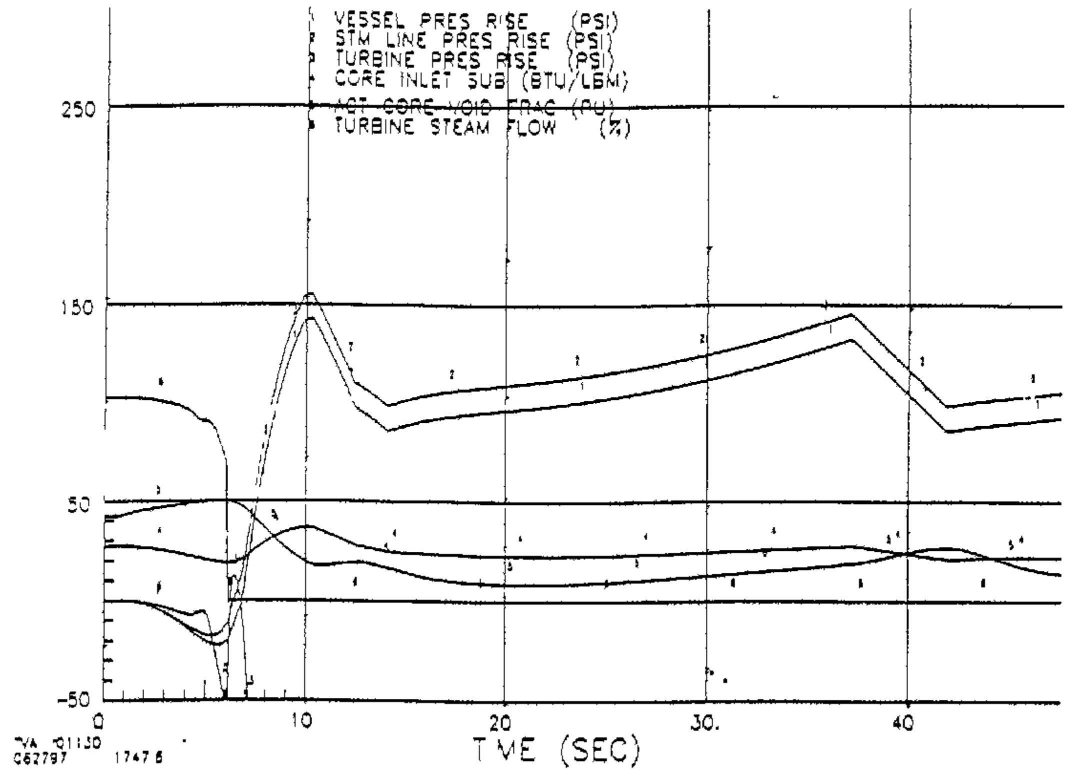
FIGURE 14 5-17c

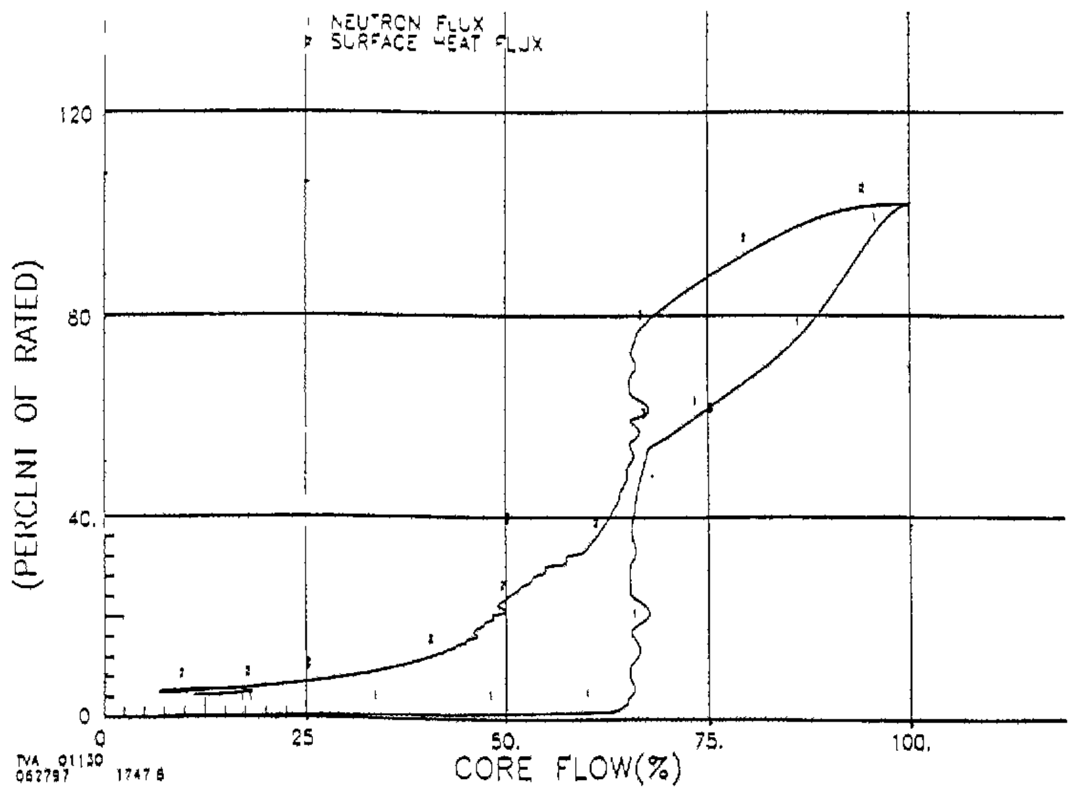




AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT
LOSS OF AUXILIARY POWER TRANSFORMERS 102P/100F
FIGURE 14.5-18a



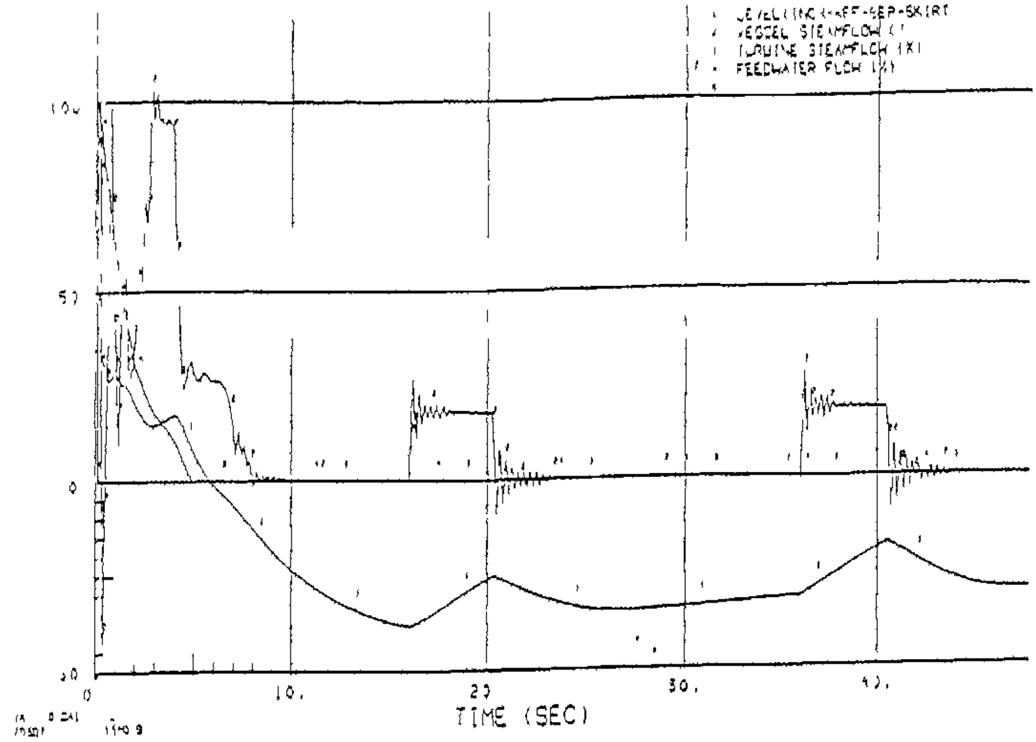


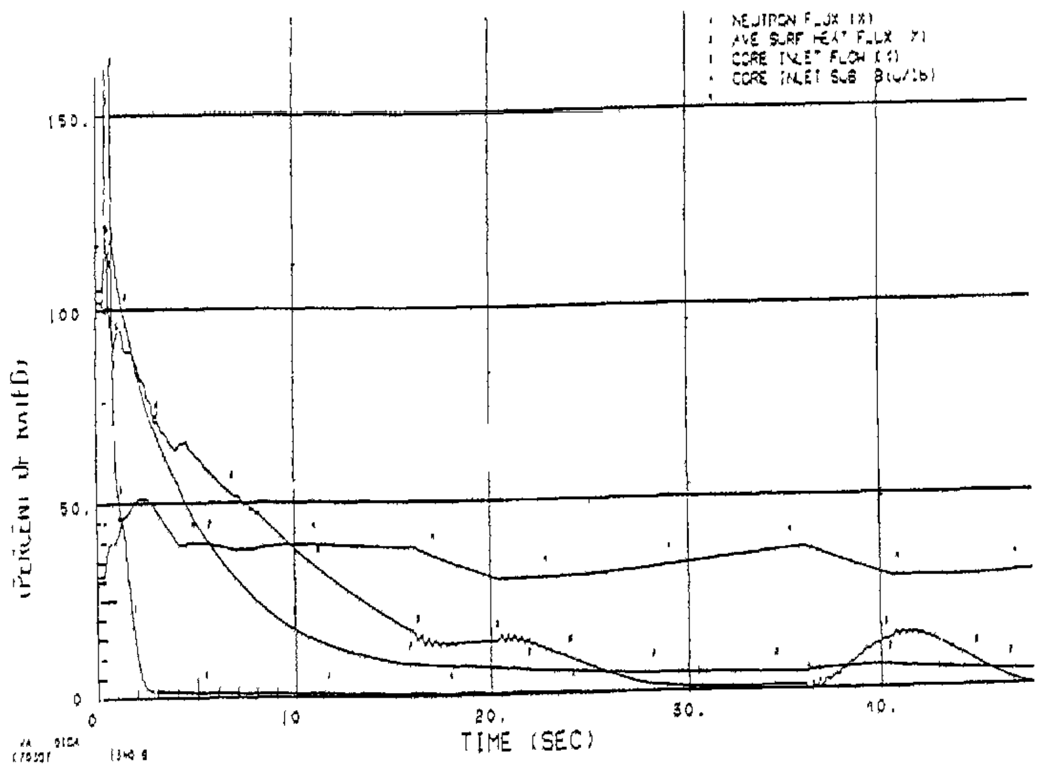
AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

LOSS OF AUXILIARY POWER TRANSFORMERS
102P/100F

FIGURE 14 5-18b



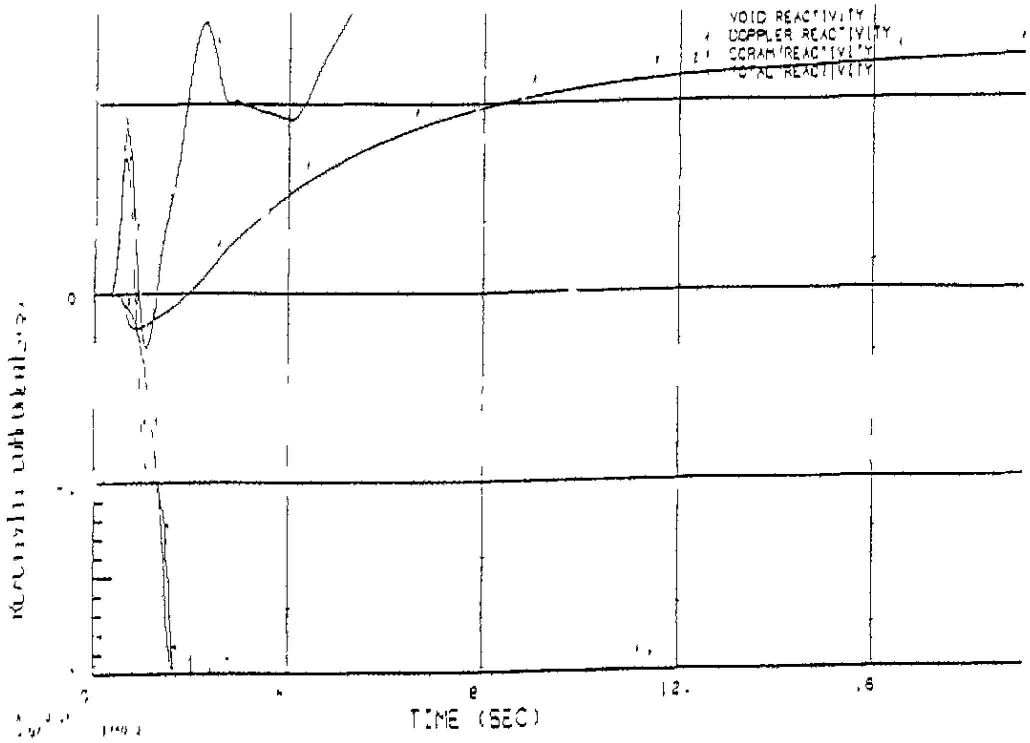


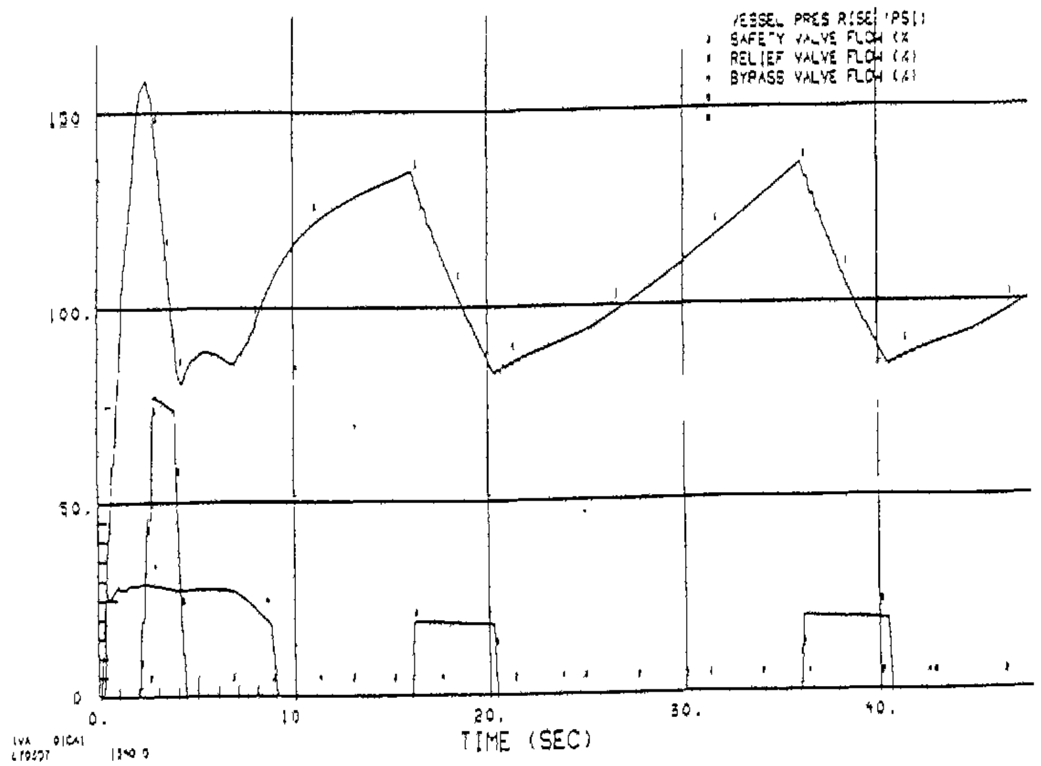
AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

LOSS OF AUXILIARY POWER
ALL GRID CONNECTIONS
102P/105F

FIGURE 14 5-19a



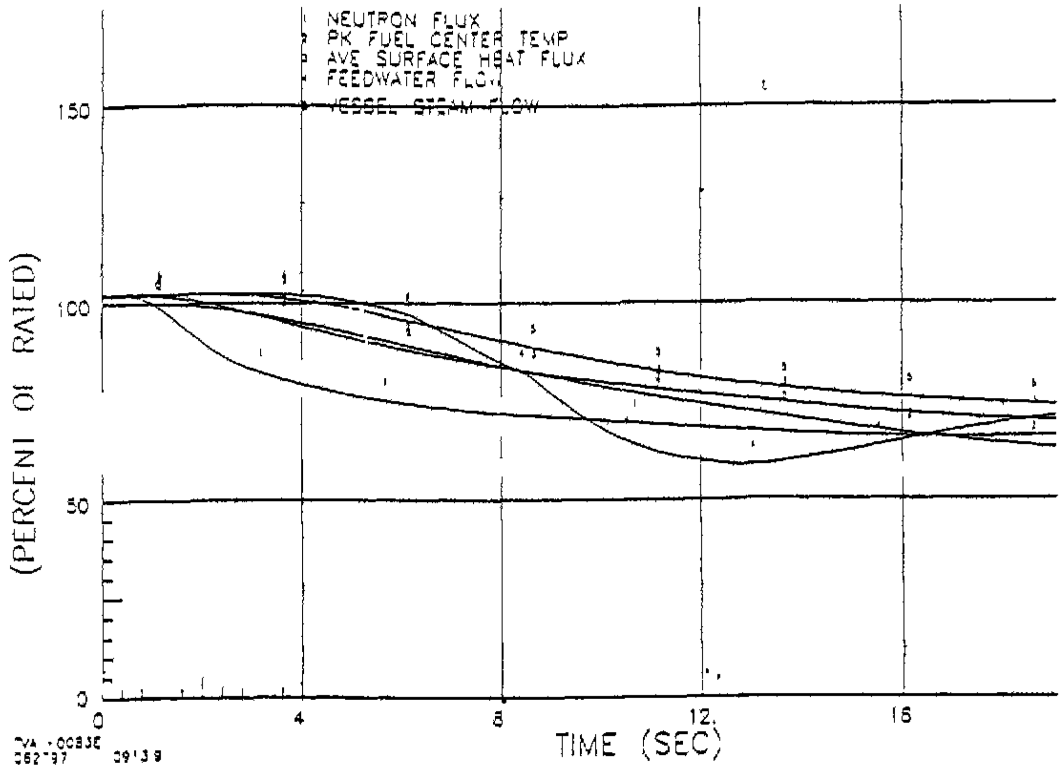


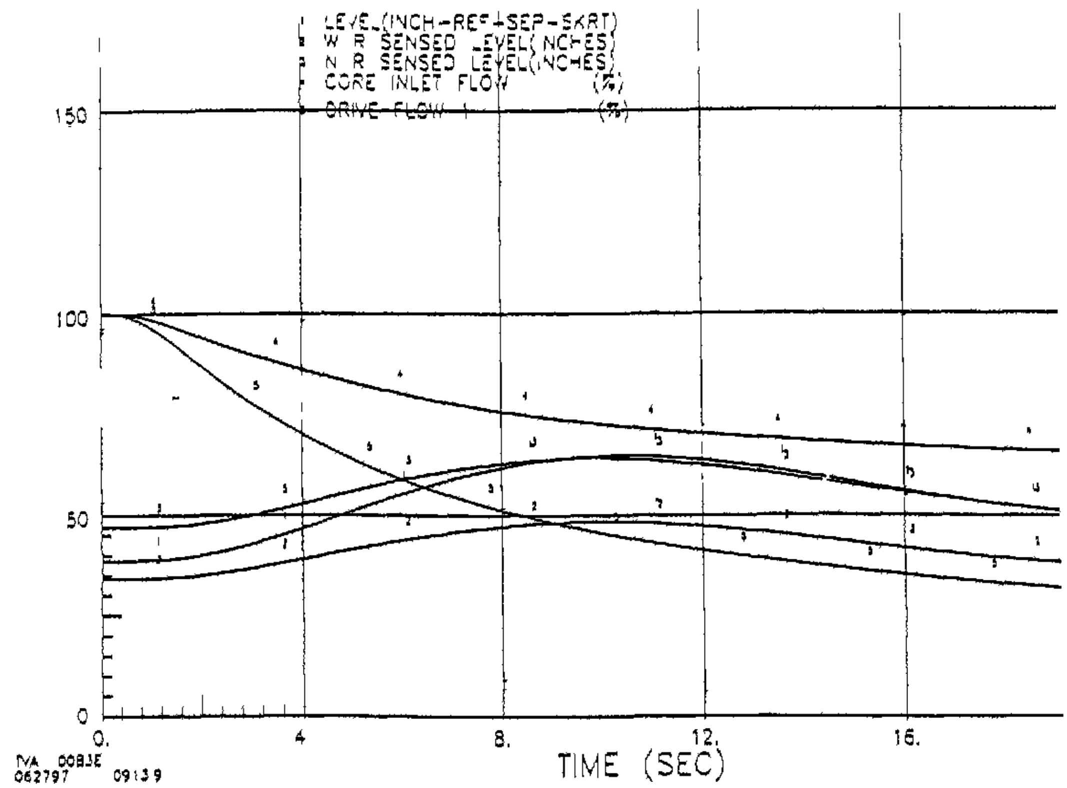
AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

LOSS OF AUXILIARY POWER
ALL GRID CONNECTIONS
102P/105F

FIGURE 14.5-19b



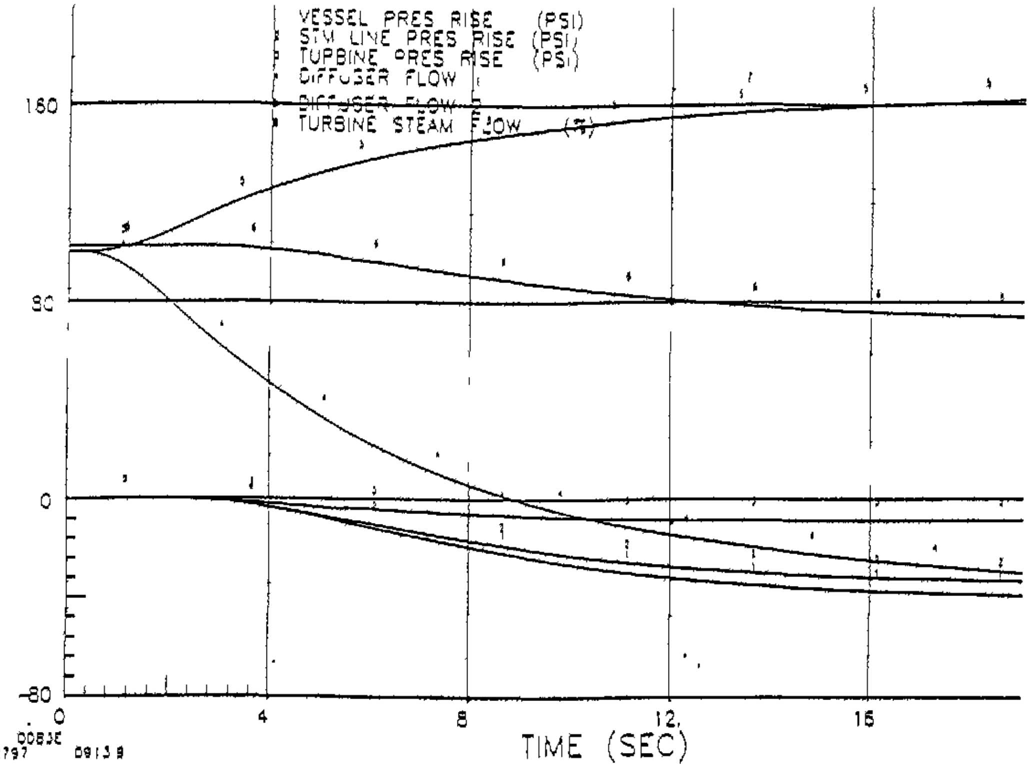


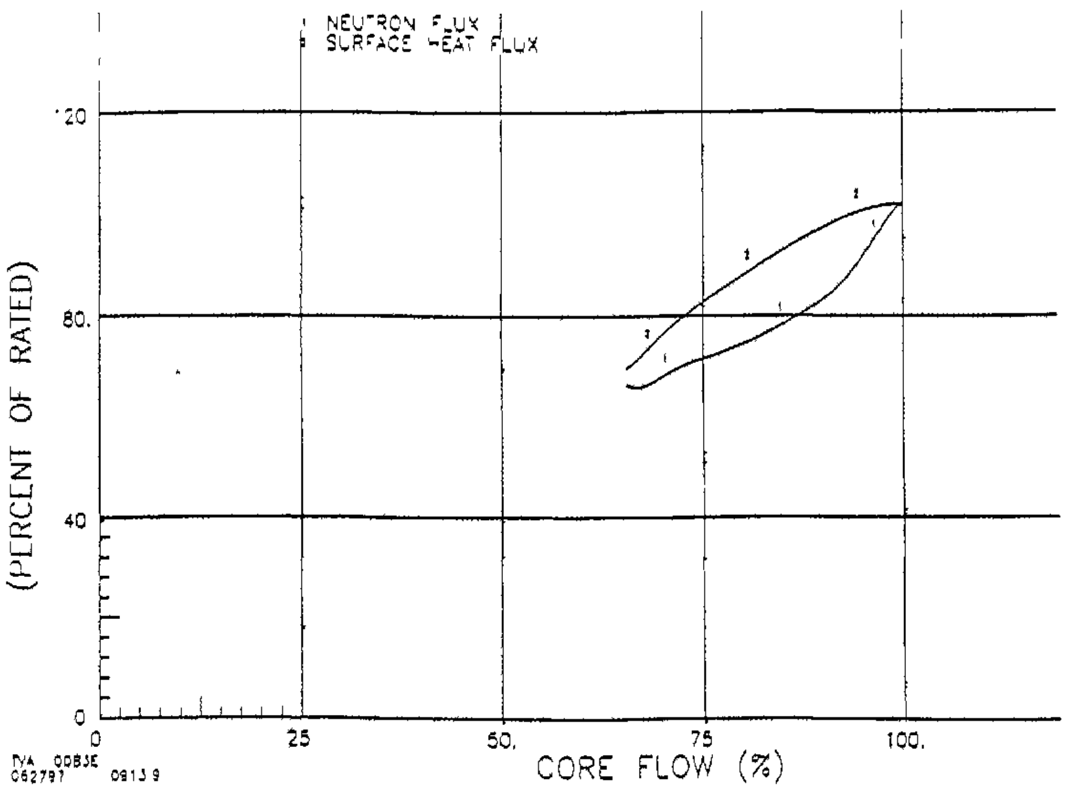
AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

RECIRCULATION FLOW CONTROL
FAILURE-DECREASING FLOW
102P/100F

FIGURE 14.5-20a



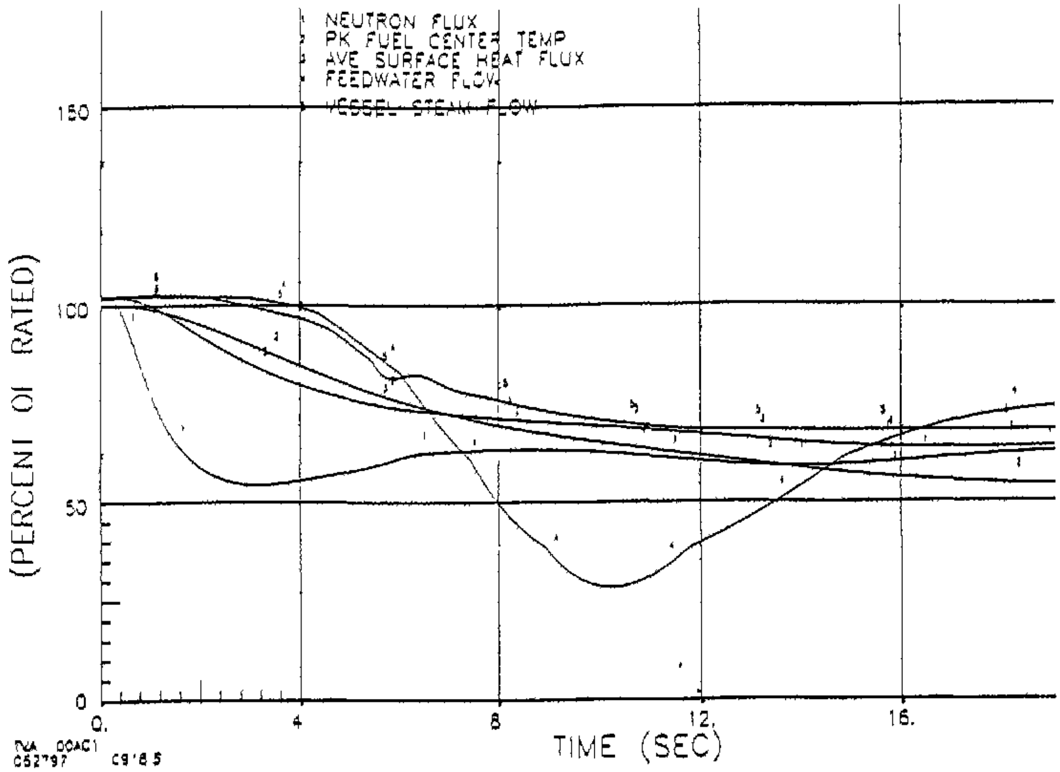


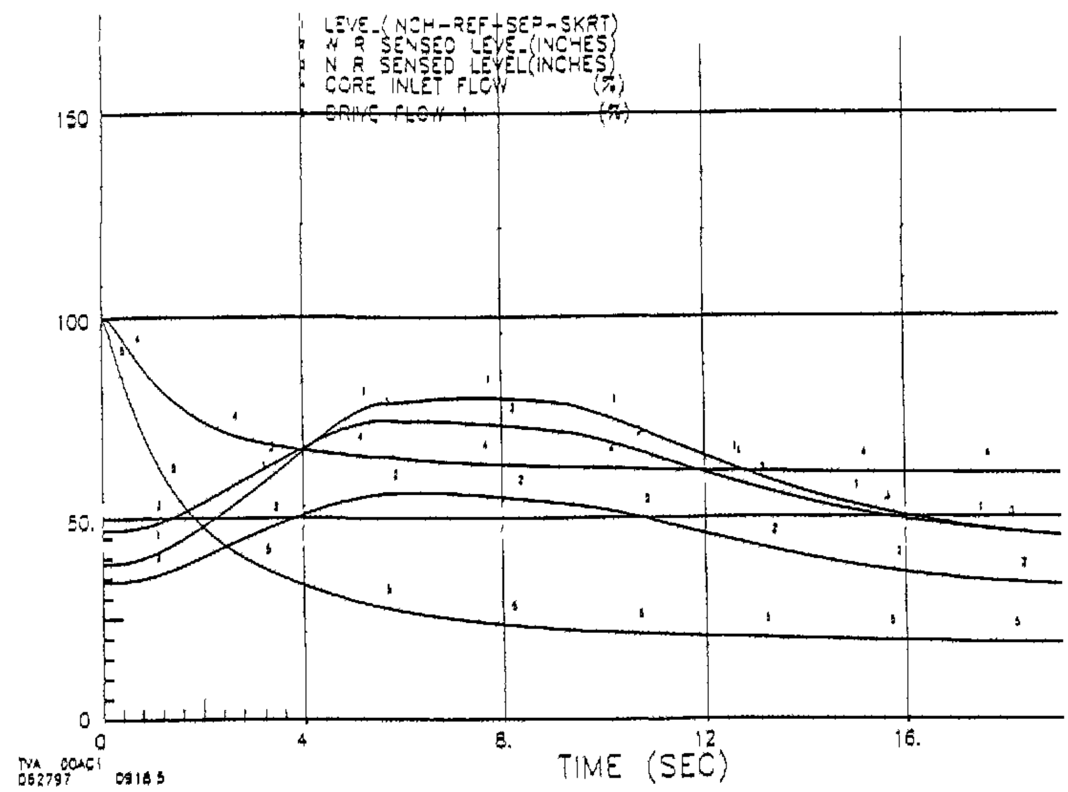
AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

RECIRCULATION FLOW CONTROL
FAILURE-DECREASING FLOW
102P/100F

FIGURE 14.5-20b



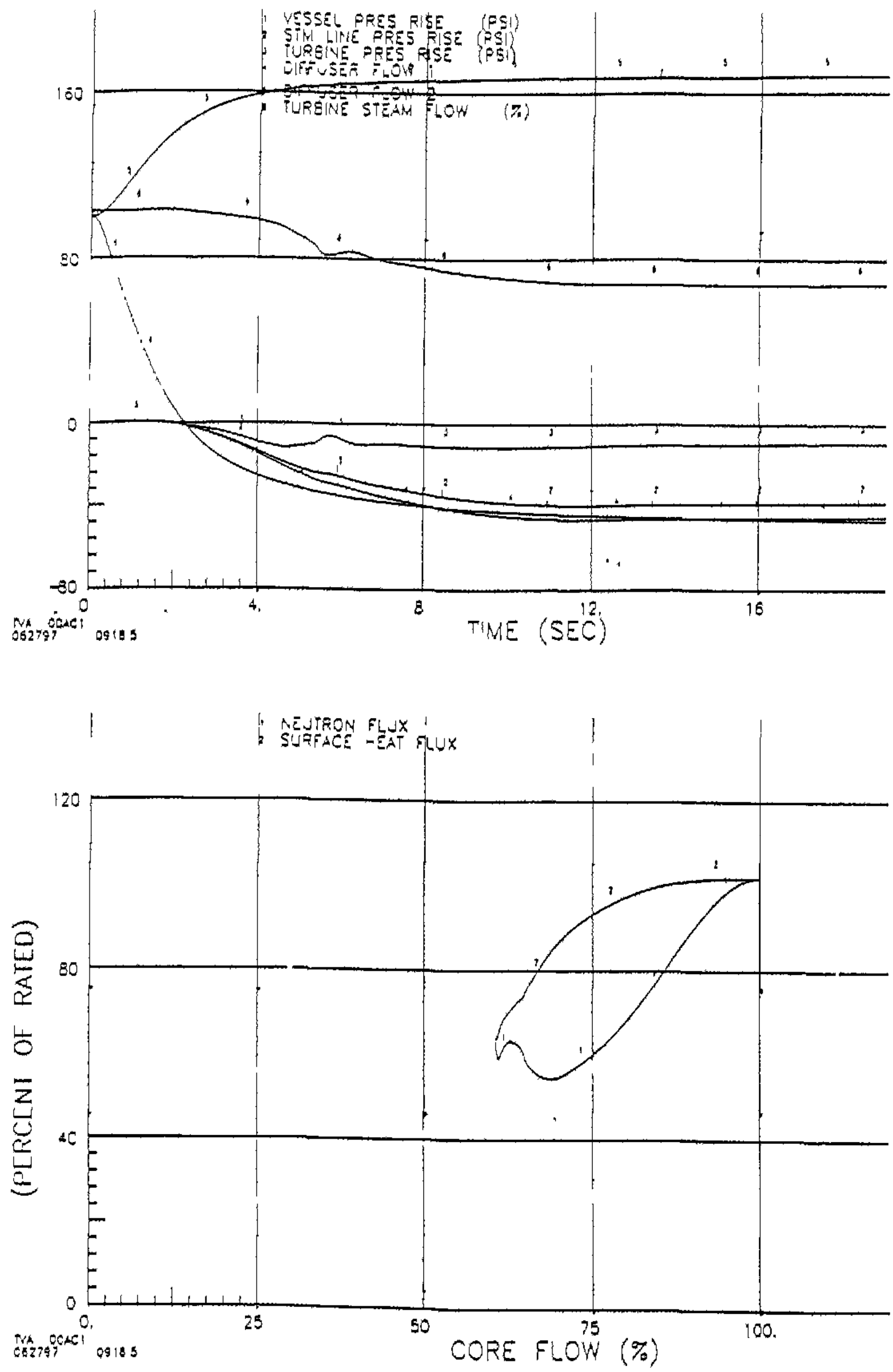


AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

ONE RECIRCULATION PUMP TRIP
102P/100F

FIGURE 14.5-21a

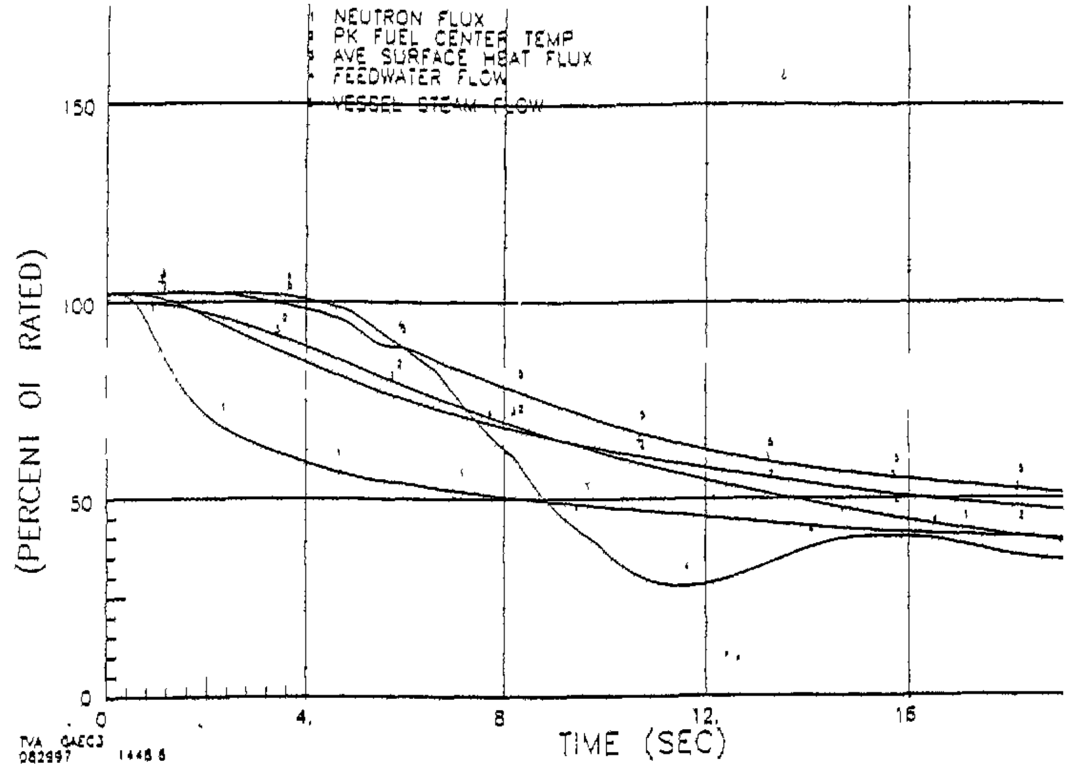


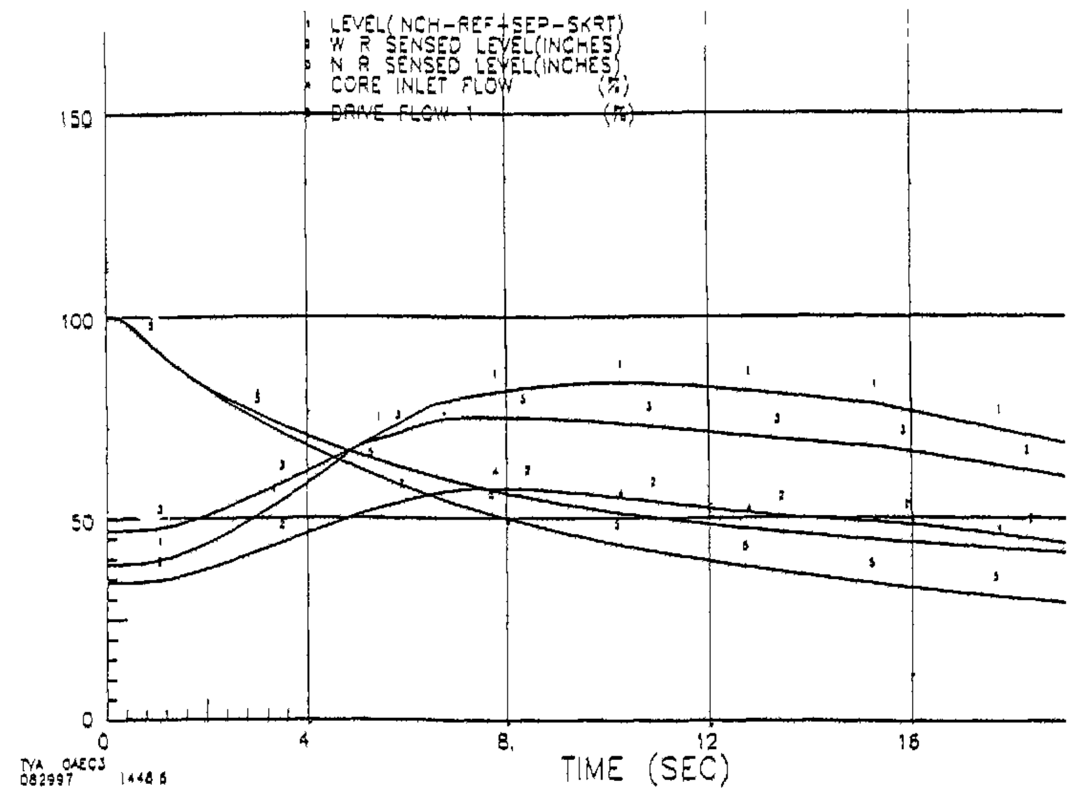
AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

ONE RECIRCULATION PUMP TRIP
102P/100F

FIGURE 14.5-21b



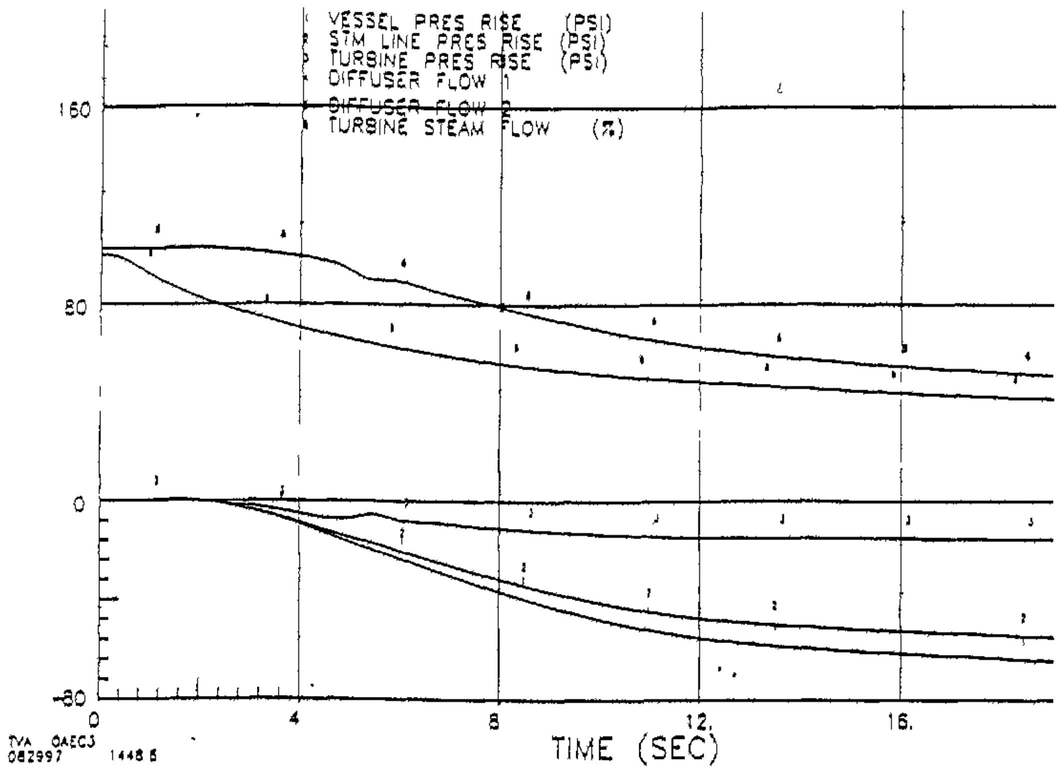


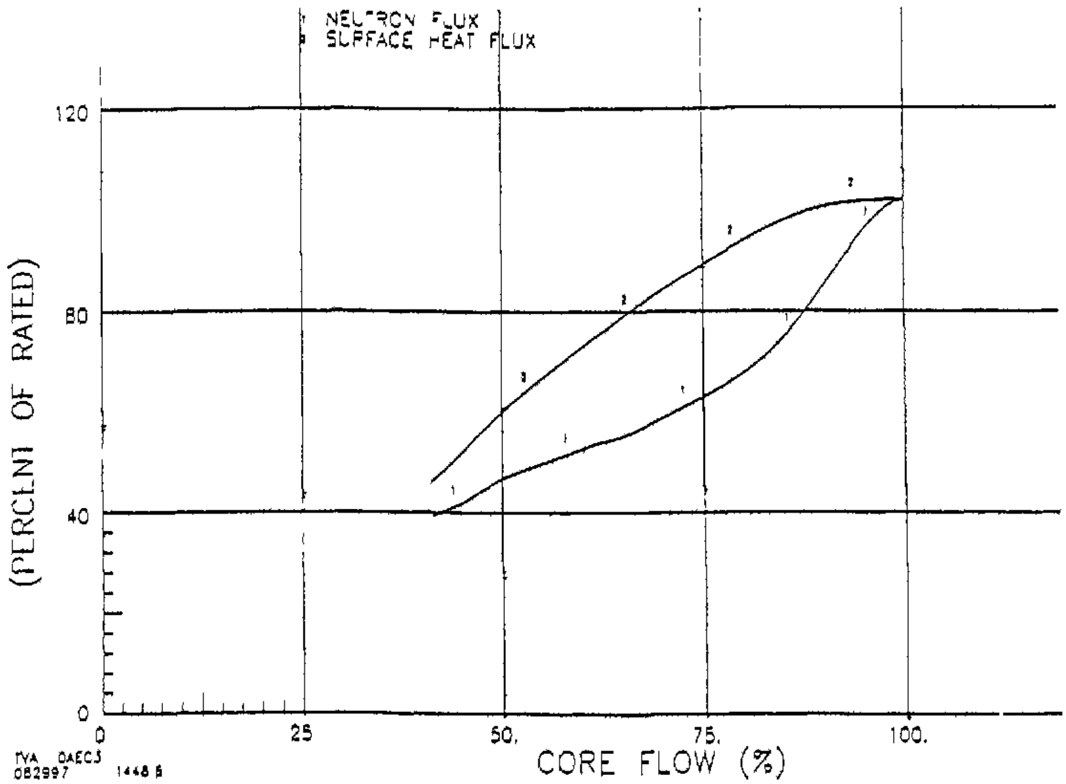
AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

TWO RECIRCULATION M/G SET
DRIVE MOTORS TRIP
102P/100F

FIGURE 14.5-22a



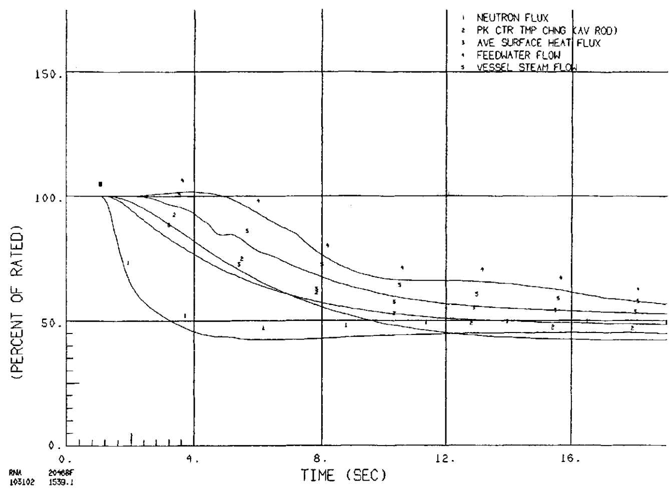


AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

TWO RECIRCULATION M/G SET
DRIVE MOTORS TRIP
102P/100F

FIGURE 14.5-22b

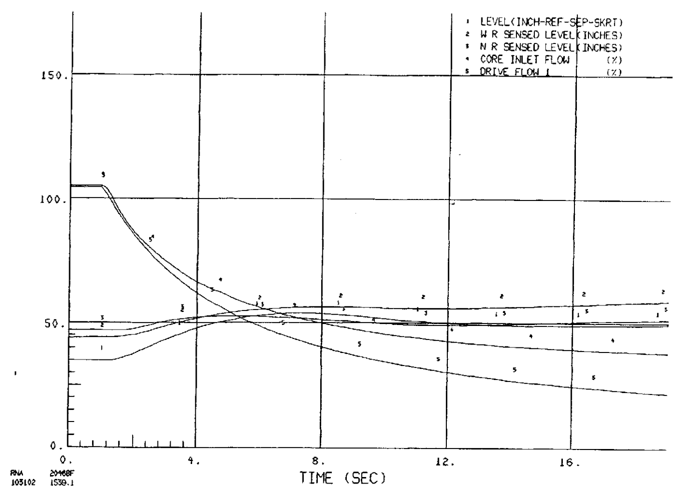


ADMENDMENT 20

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

TWO RECIRCULATION PUMP TRIP
WITH VFDS
100P/100F

FIGURE 14.5-22c

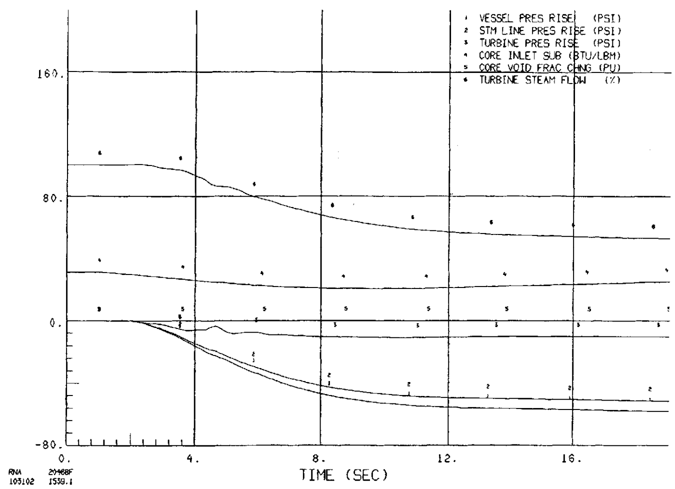


ADMENDMENT 20

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

TWO RECIRCULATION PUMP TRIP
WITH VFDS
100P/100F

FIGURE 14.5-22d

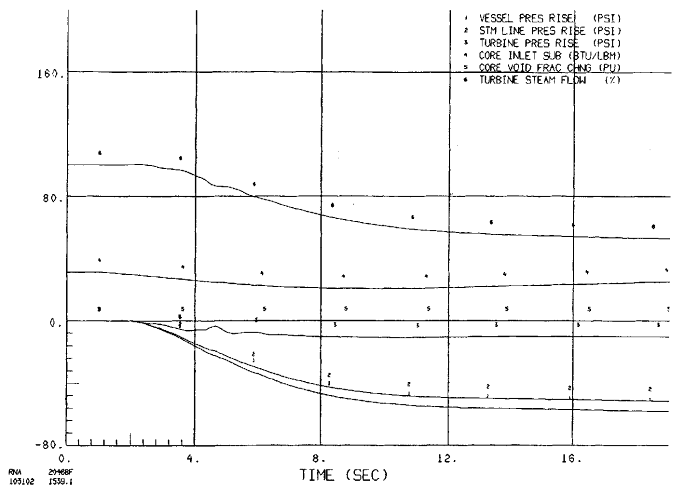


ADMENDMENT 20

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

TWO RECIRCULATION PUMP TRIP
WITH VFDS
100P/100F

FIGURE 14.5-22e

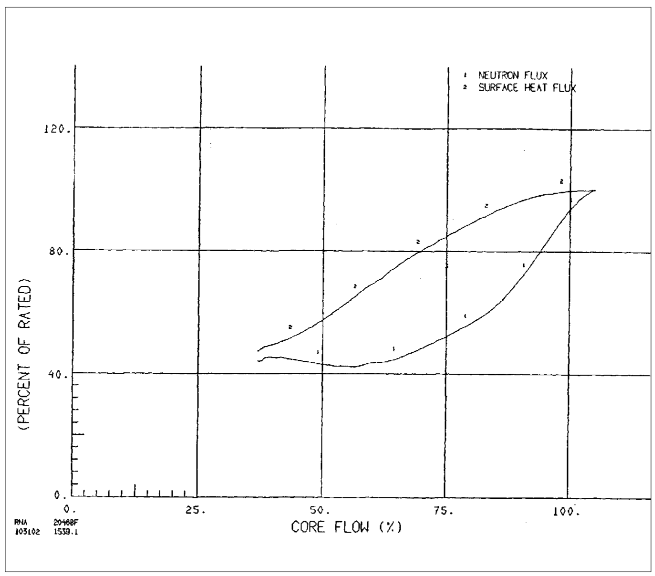


ADMENDMENT 20

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

TWO RECIRCULATION PUMP TRIP
WITH VFDS
100P/100F

FIGURE 14.5-22e

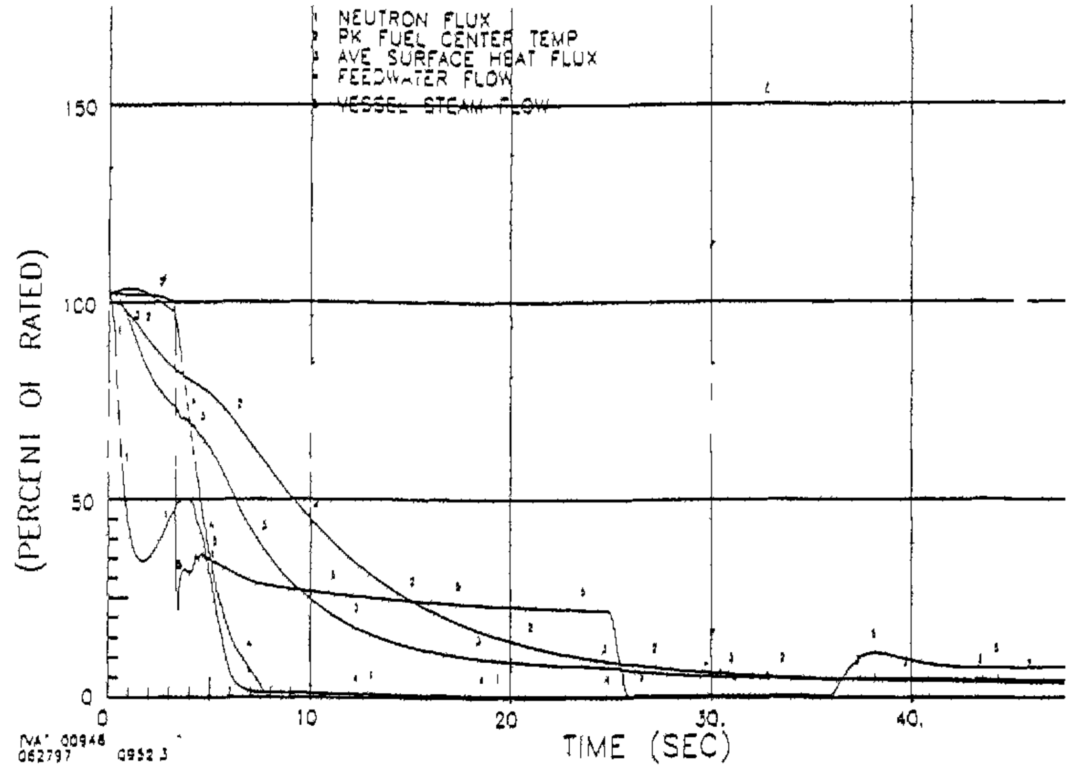


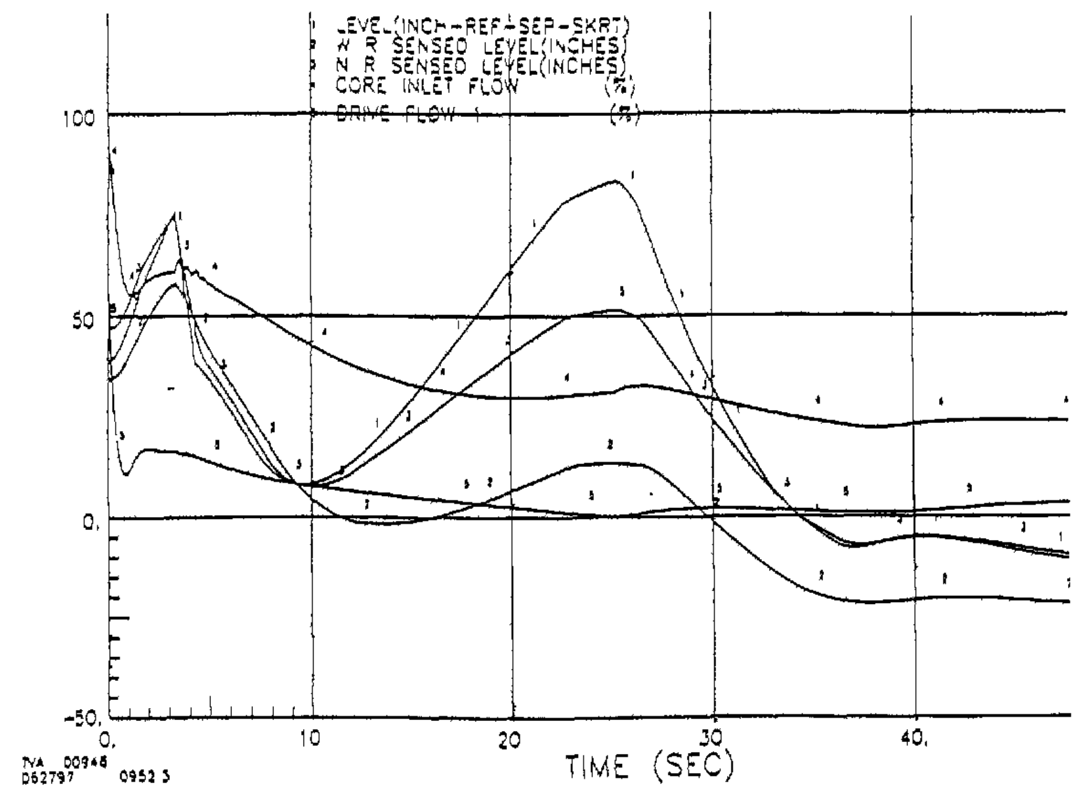
ADMENDMENT 20

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

TWO RECIRCULATION PUMP TRIP
WITH VFDS
100P/100F

FIGURE 14.5-22f



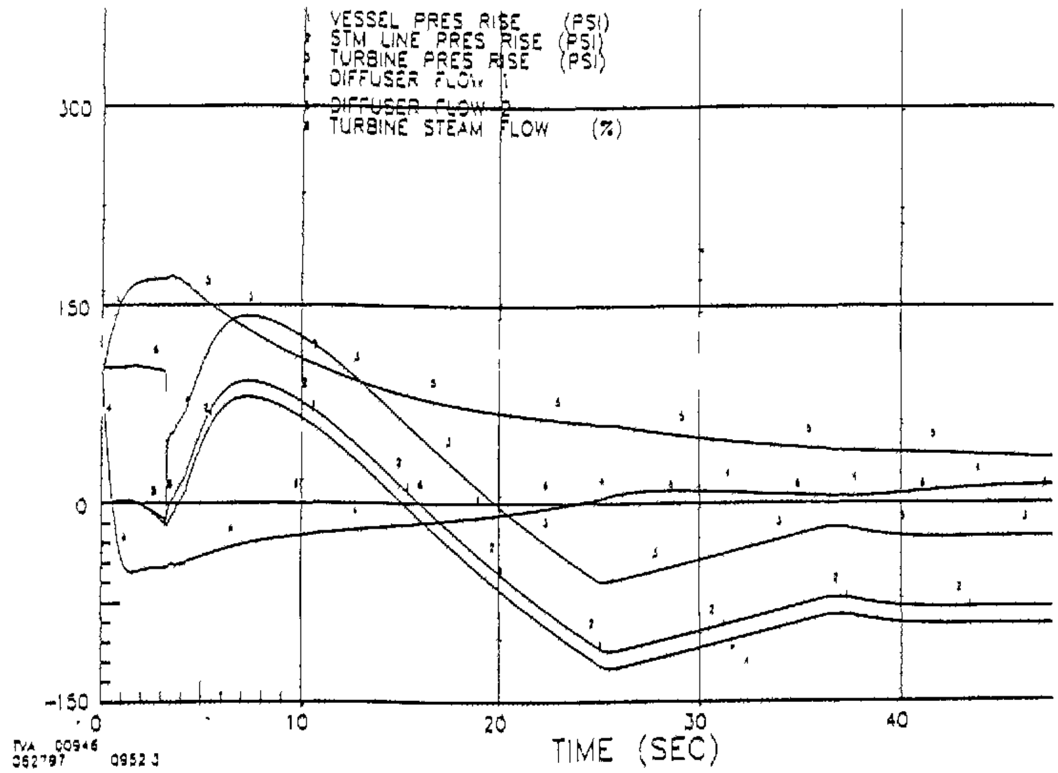


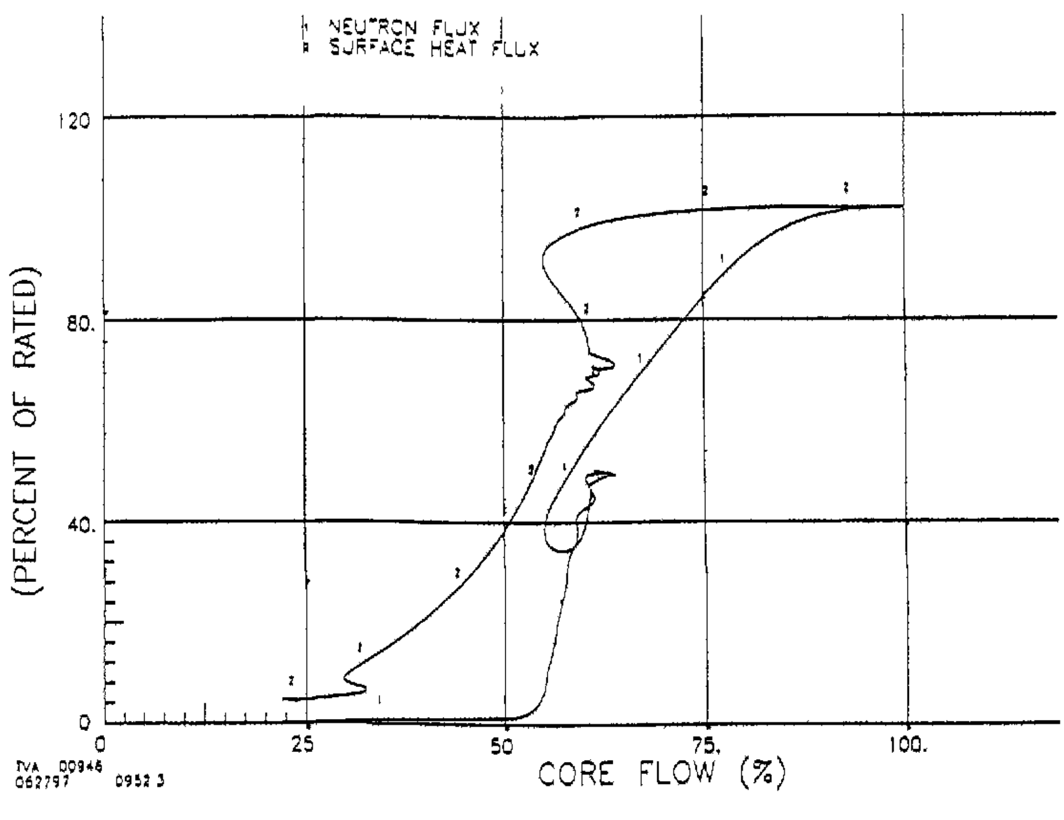
AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

ONE RECIRCULATION PUMP SEIZURE
102P/100F

FIGURE 14.5-23a



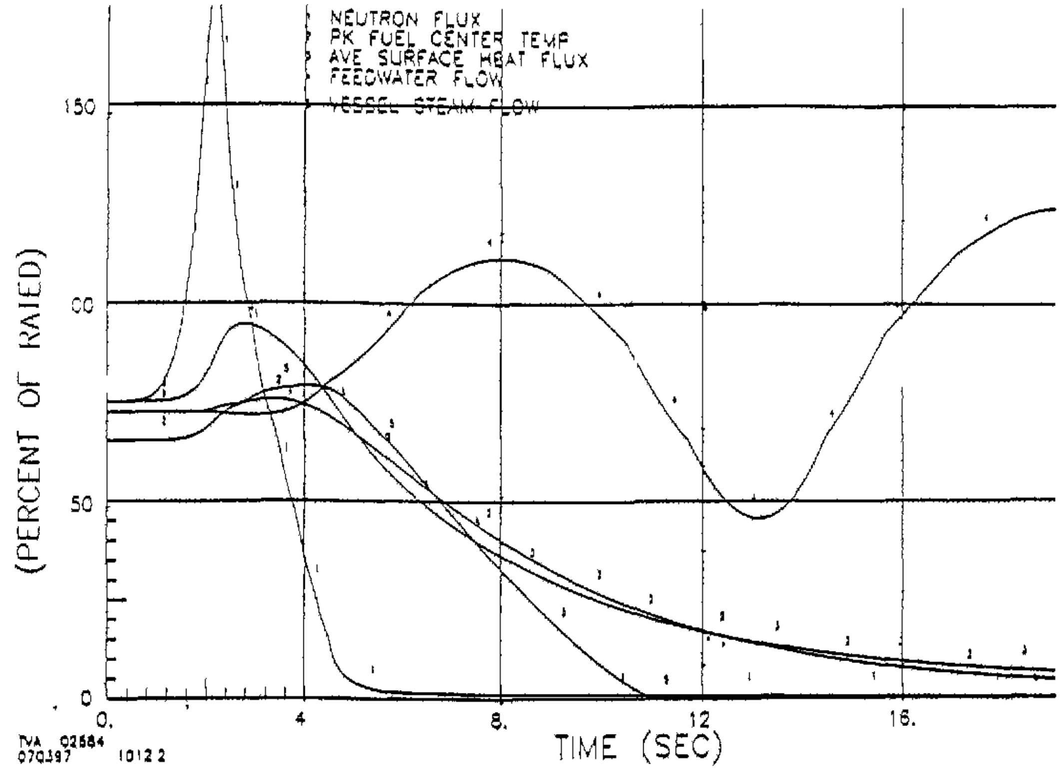


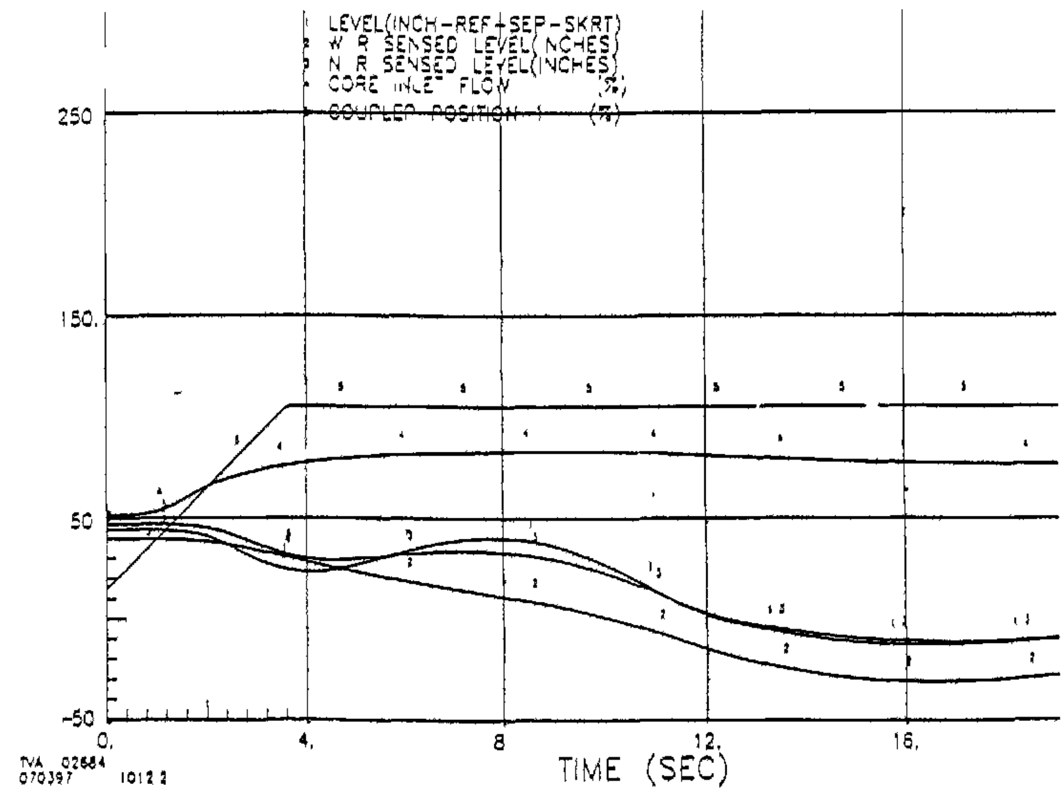
AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

ONE RECIRCULATION PUMP SEIZURE
102P/100F

FIGURE 14.5-23b



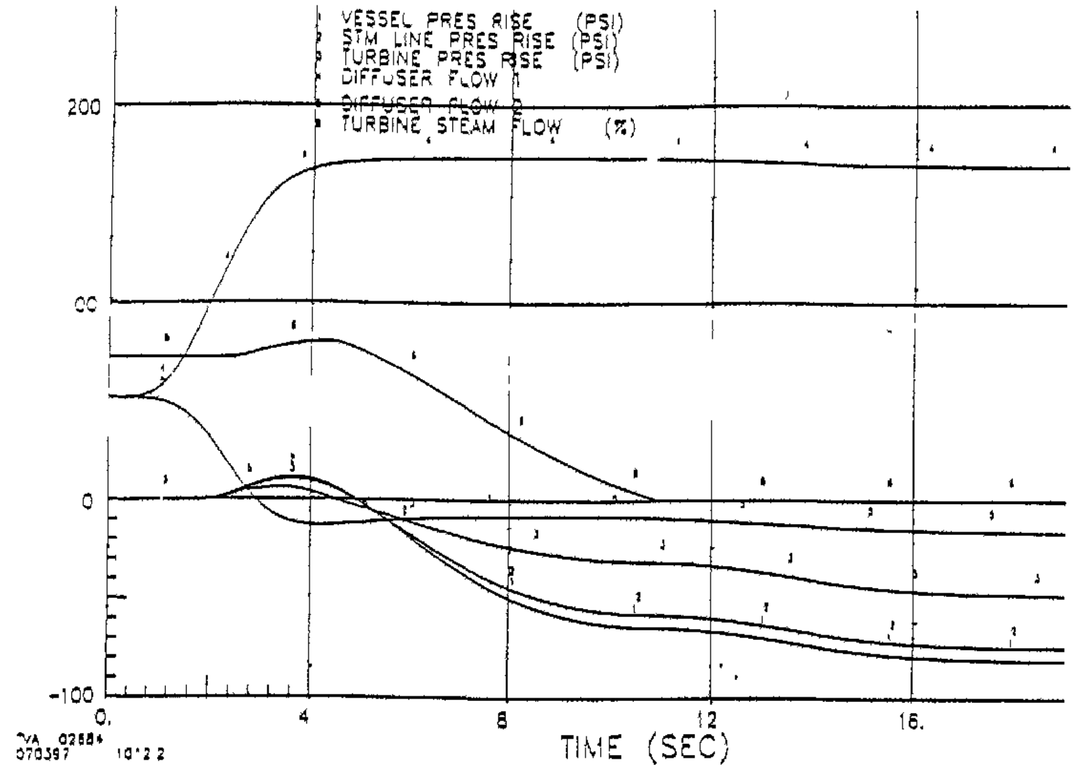


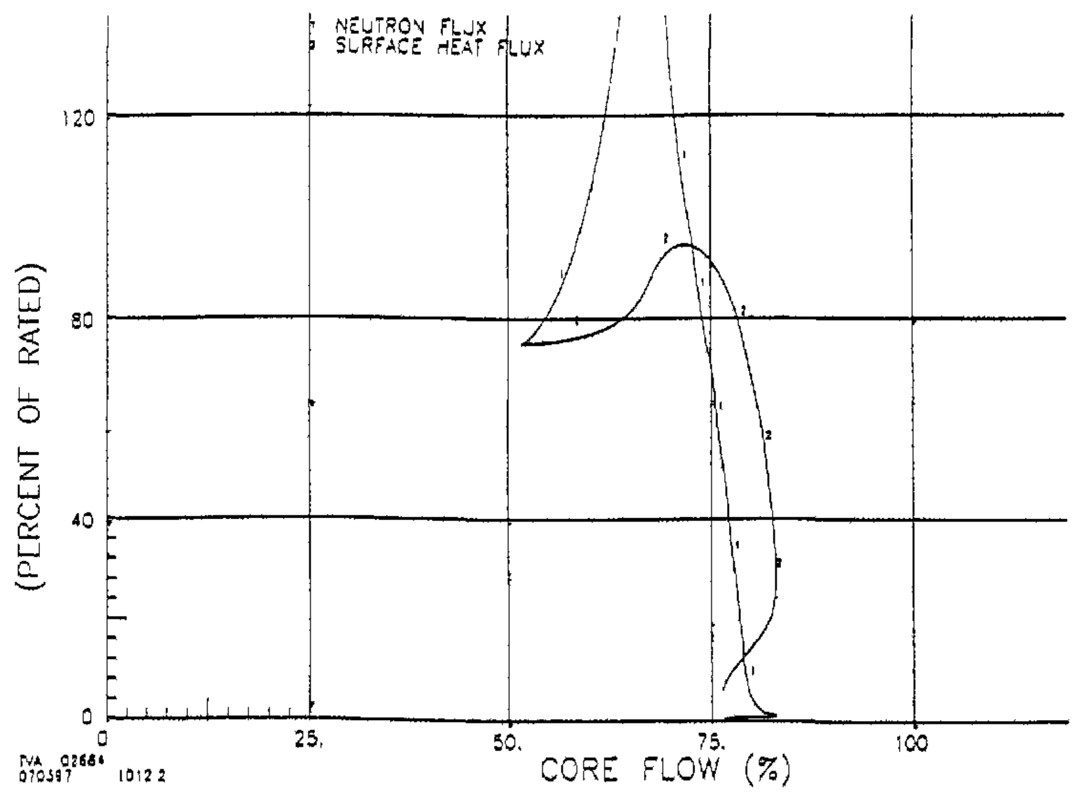
AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

RECIRCULATION FLOW CONTROL
FAILURE-INCREASING FLOW
75P/52F

FIGURE 14.5-24a



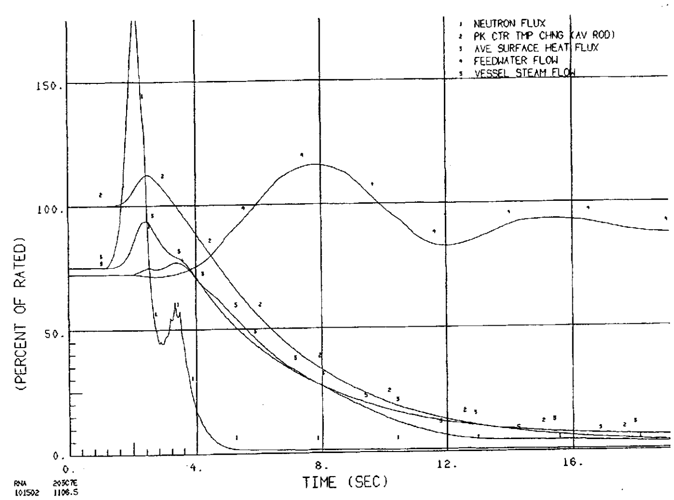


AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

RECIRCULATION FLOW CONTROL
FAILURE-INCREASING FLOW
75P/52F

FIGURE 14.5-24b

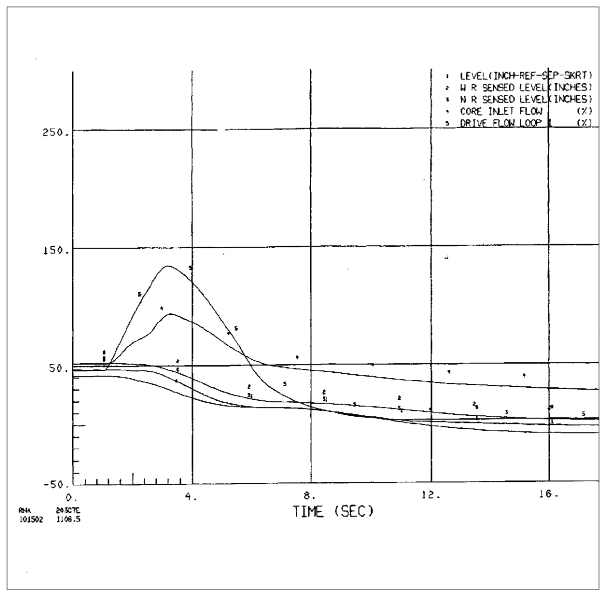


ADMENDMENT 20

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

RECIRCULATION FLOW CONTROL
FAILURE-INCREASING FLOW 75P/52F
VFD SPEED CONTROL
MAXIMUM PUMP TORQUE
SET BY BREAKDOWN TORQUE

FIGURE 14.5-24c

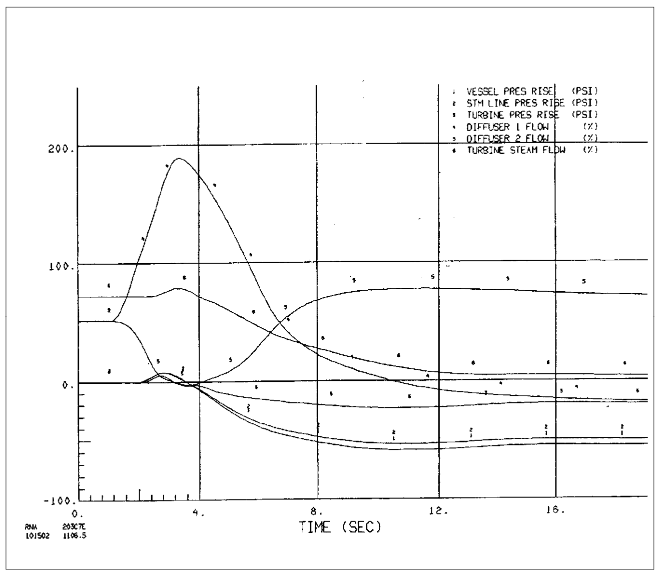


ADMENDMENT 20

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

RECIRCULATION FLOW CONTROL
FAILURE-INCREASING FLOW 75P/52F
VFD SPEED CONTROL
MAXIMUM PUMP TORQUE
SET BY BREAKDOWN TORQUE

FIGURE 14.5-24d

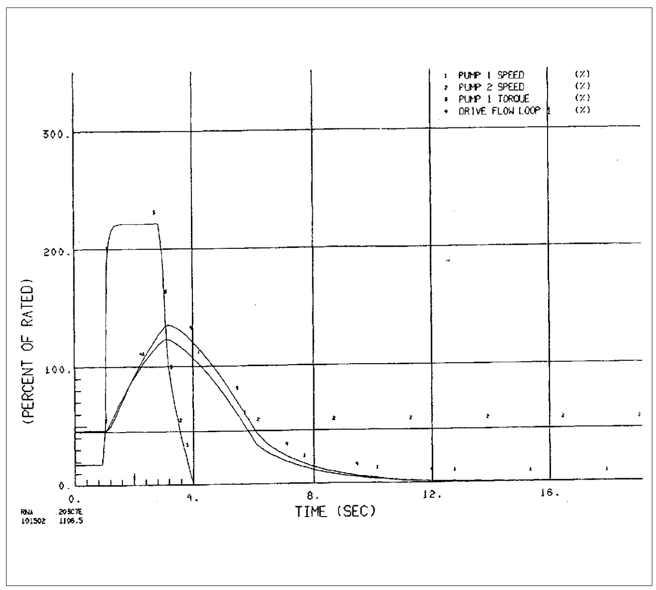


ADMENDMENT 20

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

RECIRCULATION FLOW CONTROL
FAILURE-INCREASING FLOW 75P/52F
VFD SPEED CONTROL
MAXIMUM PUMP TORQUE
SET BY BREAKDOWN TORQUE

FIGURE 14.5-24e

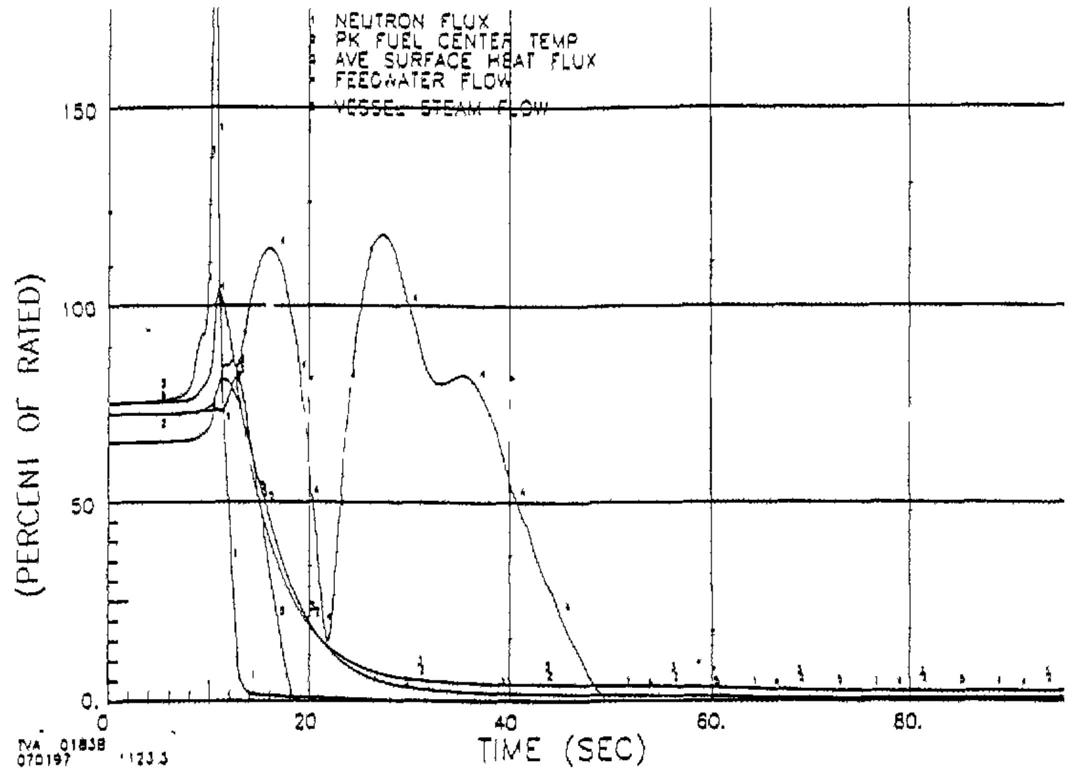


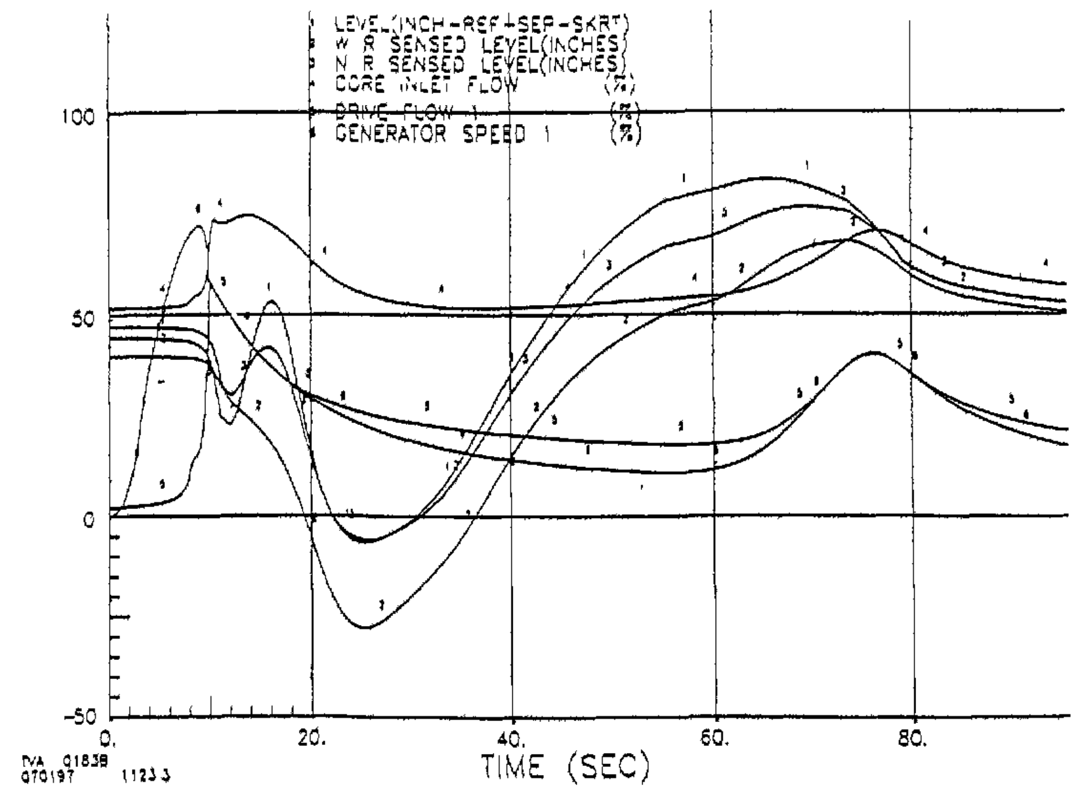
ADMENDMENT 20

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

RECIRCULATION FLOW CONTROL
FAILURE-INCREASING FLOW 75P/52F
VFD SPEED CONTROL
MAXIMUM PUMP TORQUE
SET BY BREAKDOWN TORQUE

FIGURE 14.5-24f



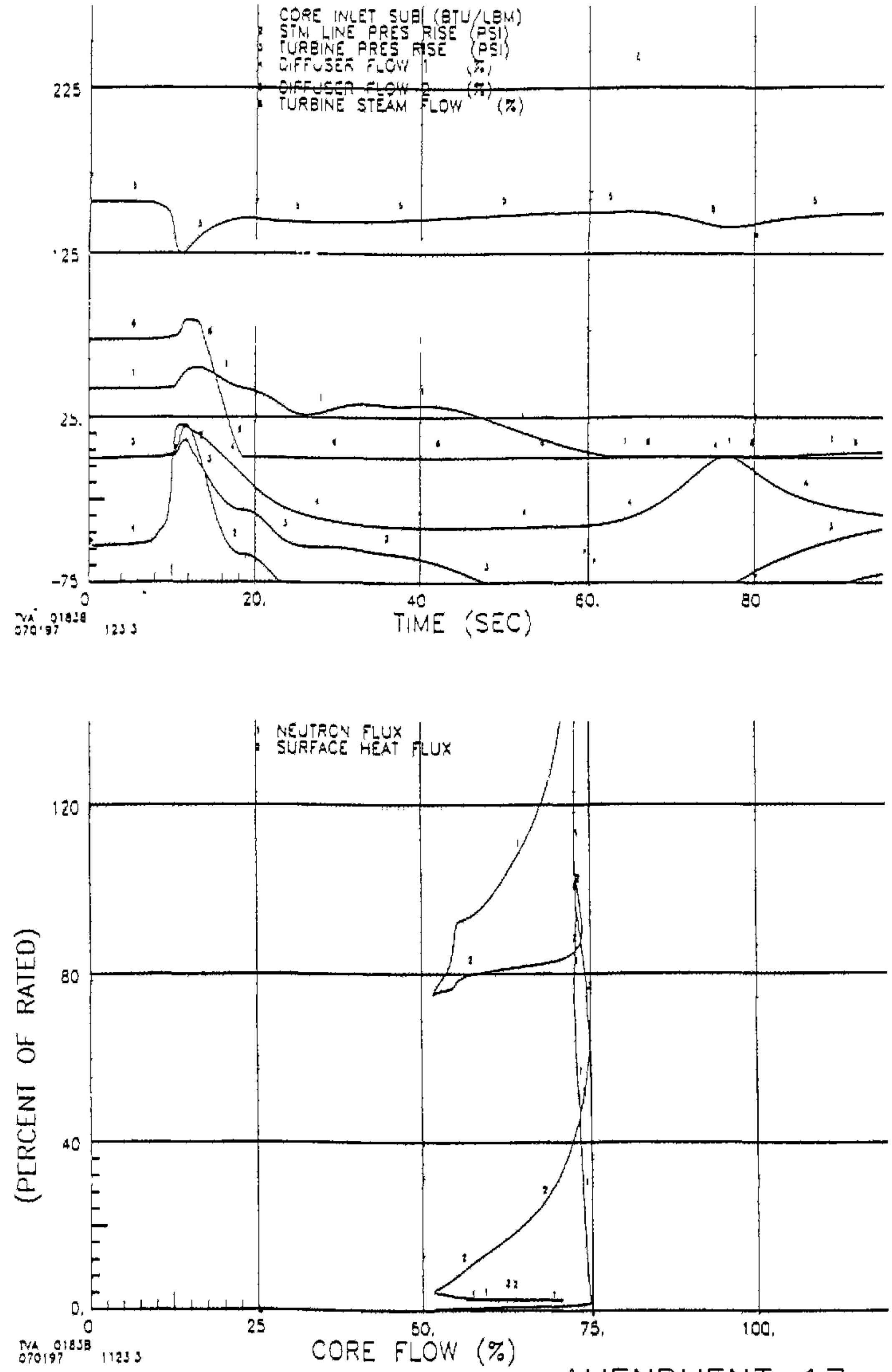


AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

IDLE RECIRCULATION LOOP
STARTUP 75P/52F
COUPLER POSITION 19%

FIGURE 14.5-25a

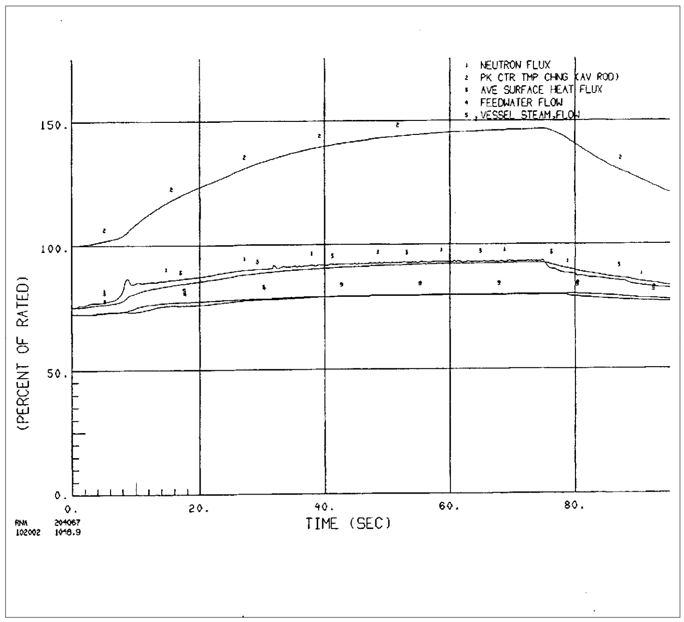


AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

IDLE RECIRCULATION LOOP
STARTUP 75P/52F
COUPLER POSITION 19%

FIGURE 14.5-25b

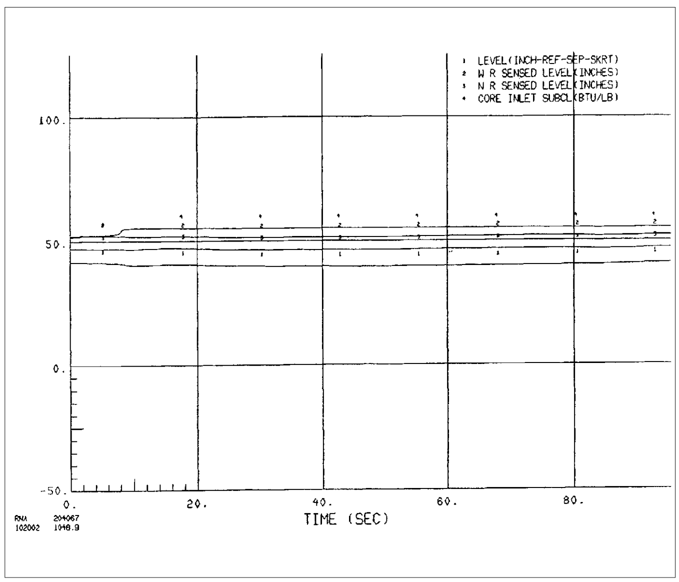


ADMENDMENT 20

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

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

FIGURE 14.5-25c

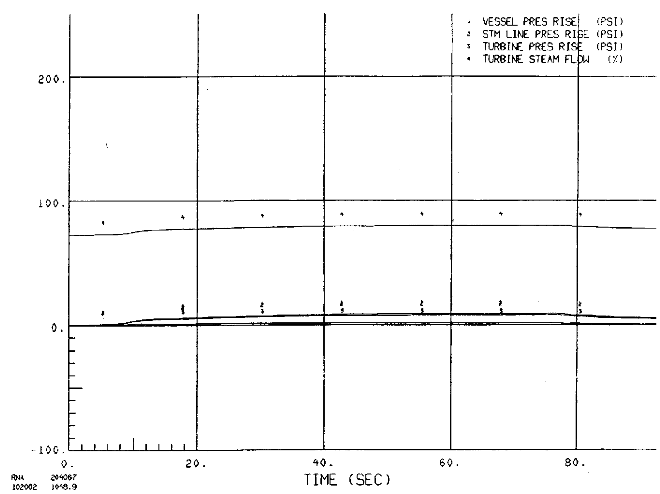


ADMENDMENT 20

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

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

FIGURE 14.5-25d

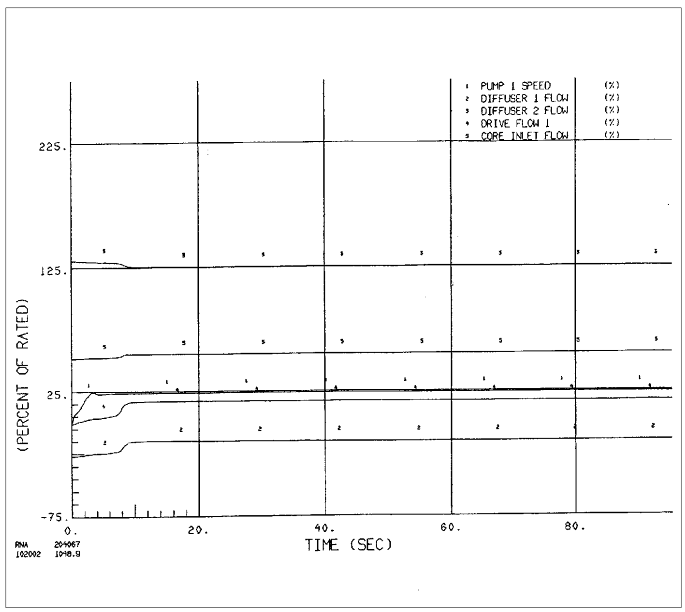


ADMENDMENT 20

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

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

FIGURE 14.5-25e

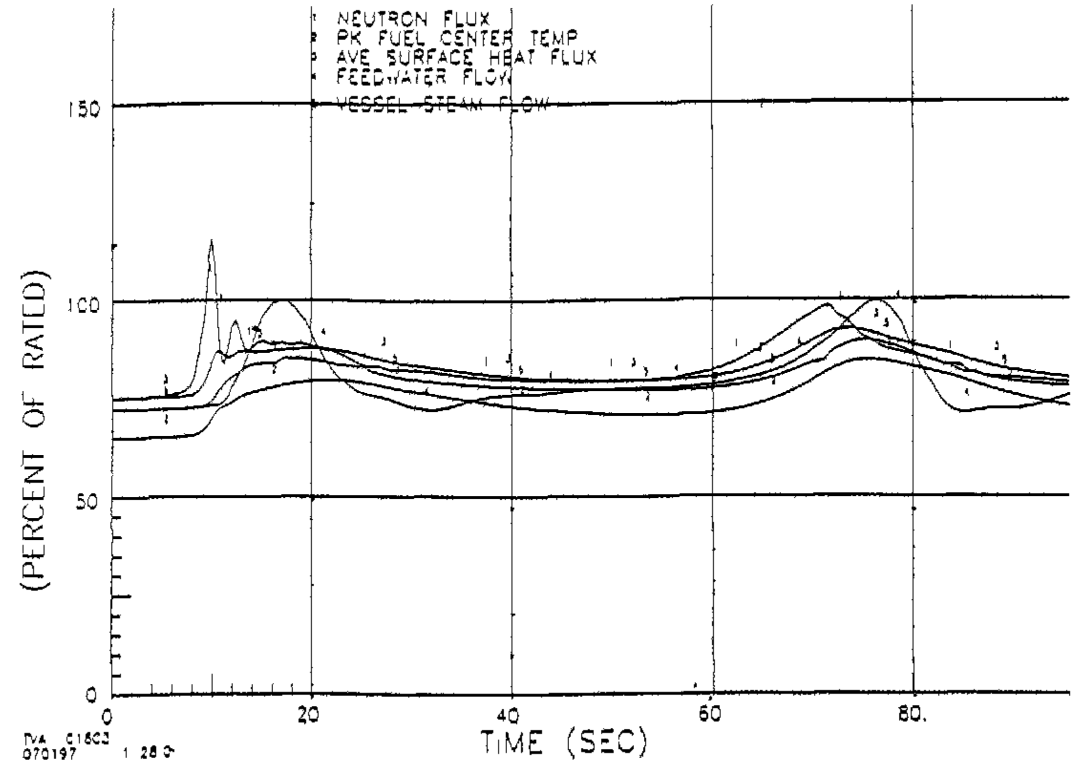


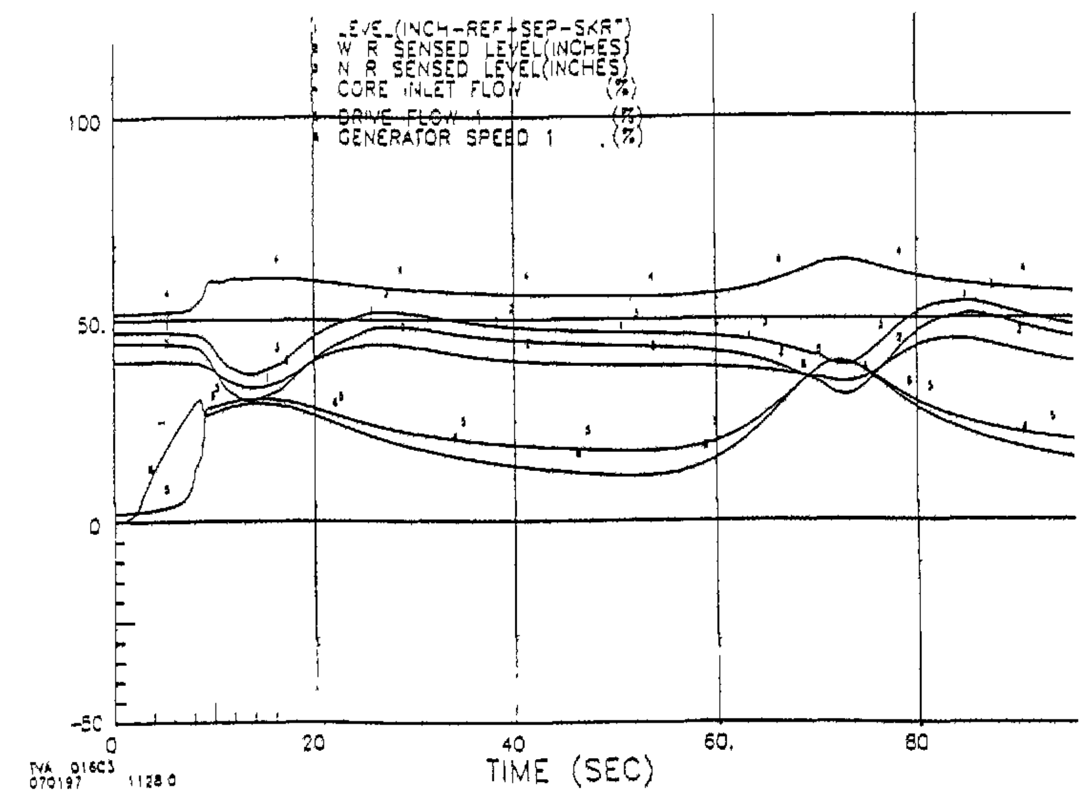
ADMENDMENT 20

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

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

FIGURE 14.5-25f



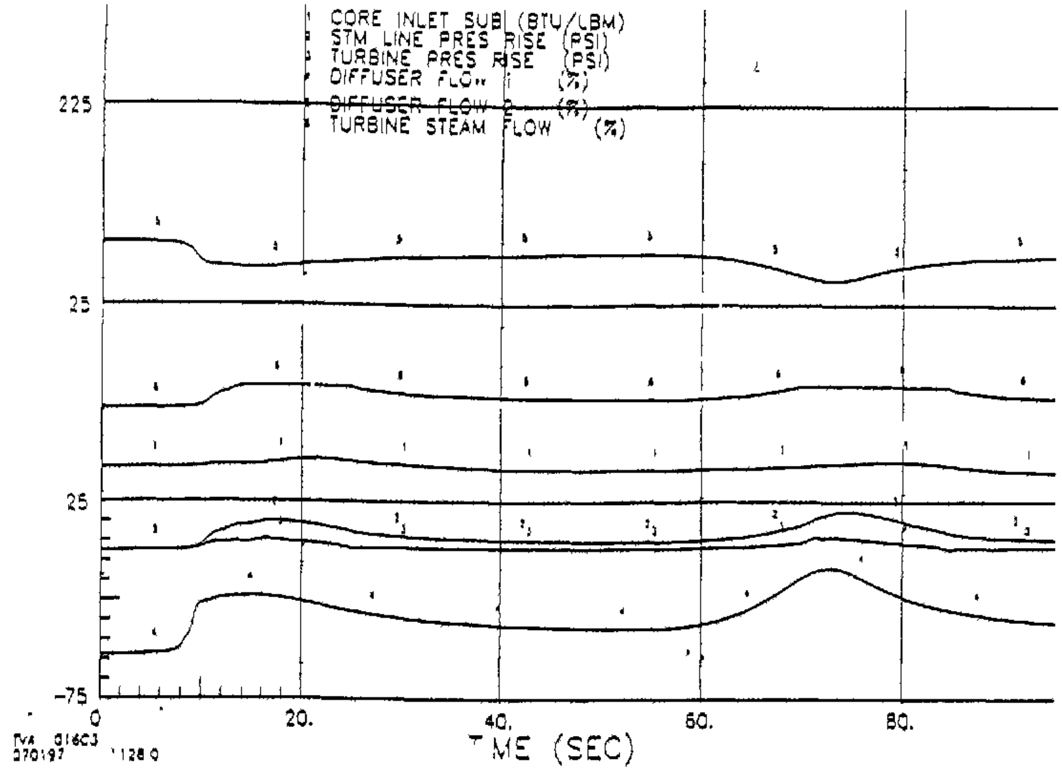


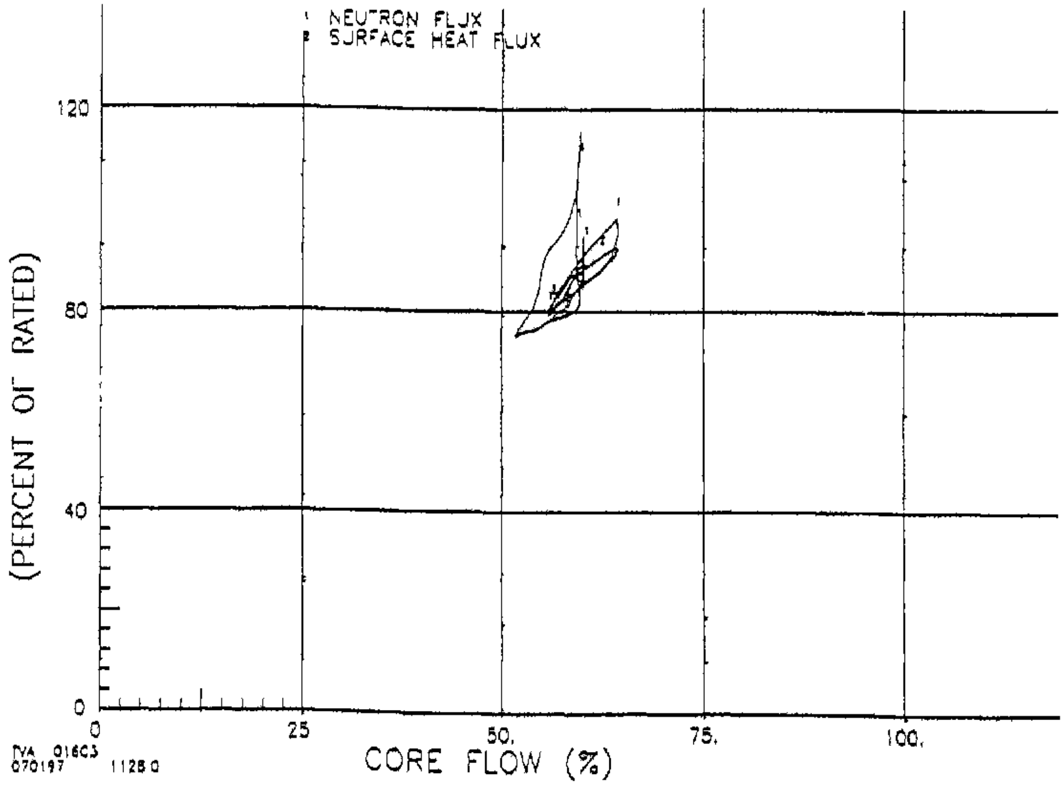
AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

IDLE RECIRCULATION LOOP
STARTUP 75P/52F
COUPLER POSITION 11%

FIGURE 14.5-26a



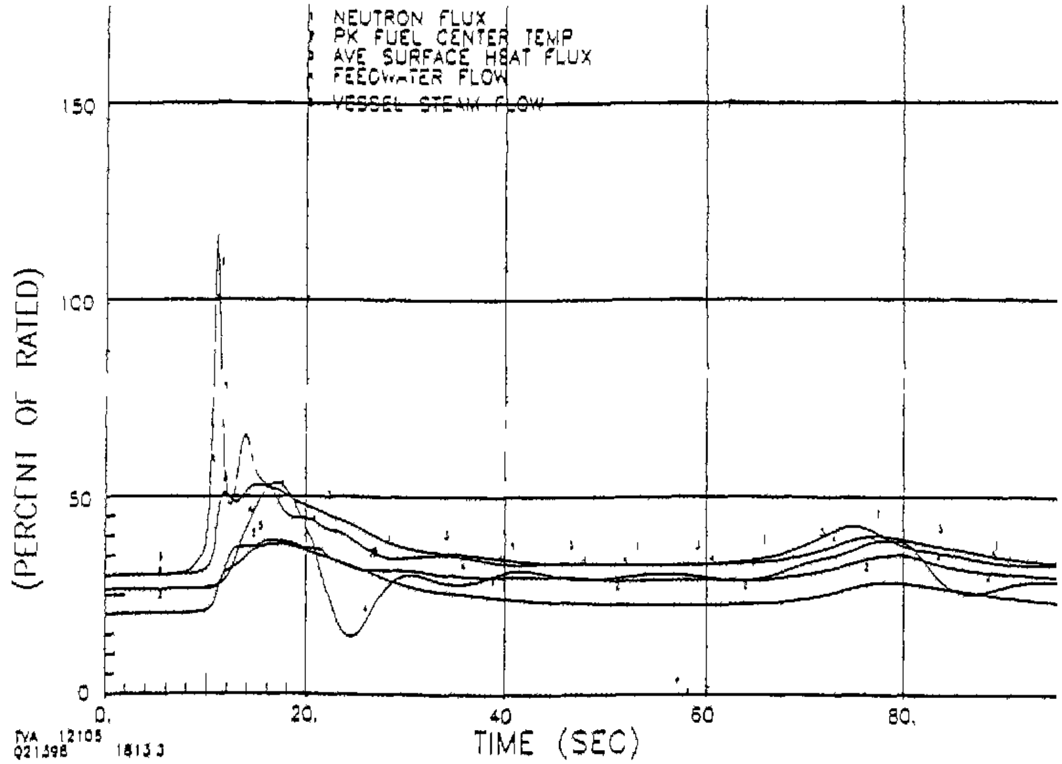


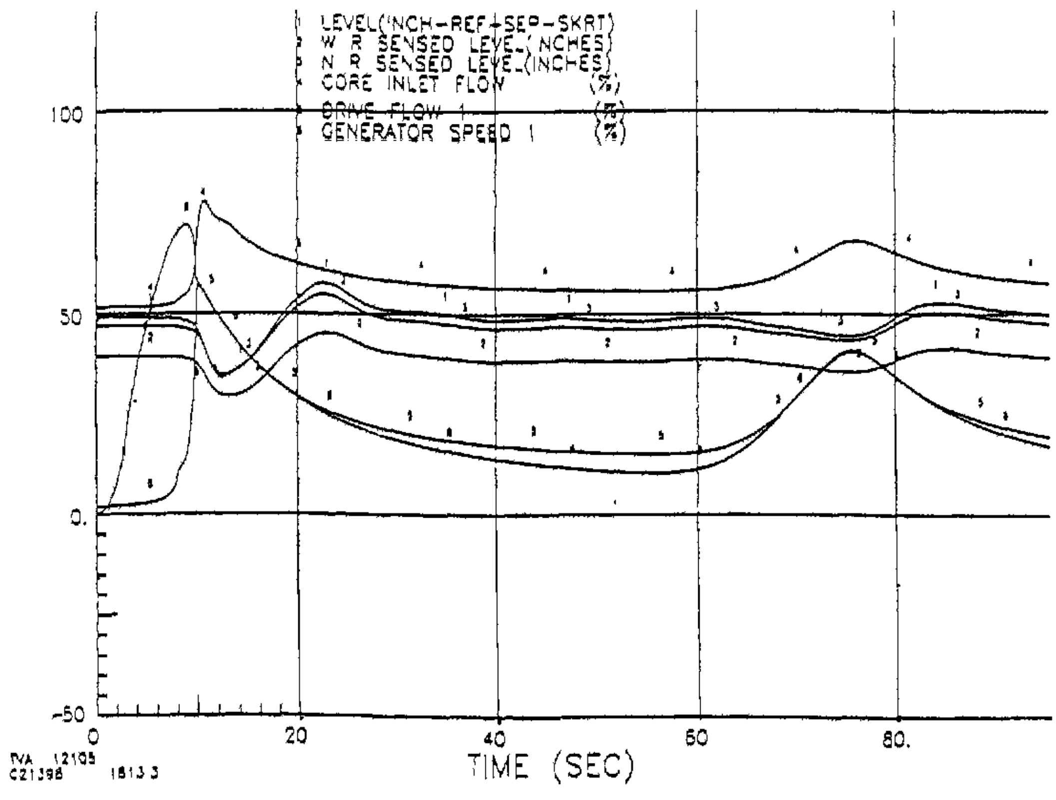
AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

IDLE RECIRCULATION LOOP
STARTUP 75P/52F
COUPLER POSITION 11%

FIGURE 14.5-26b



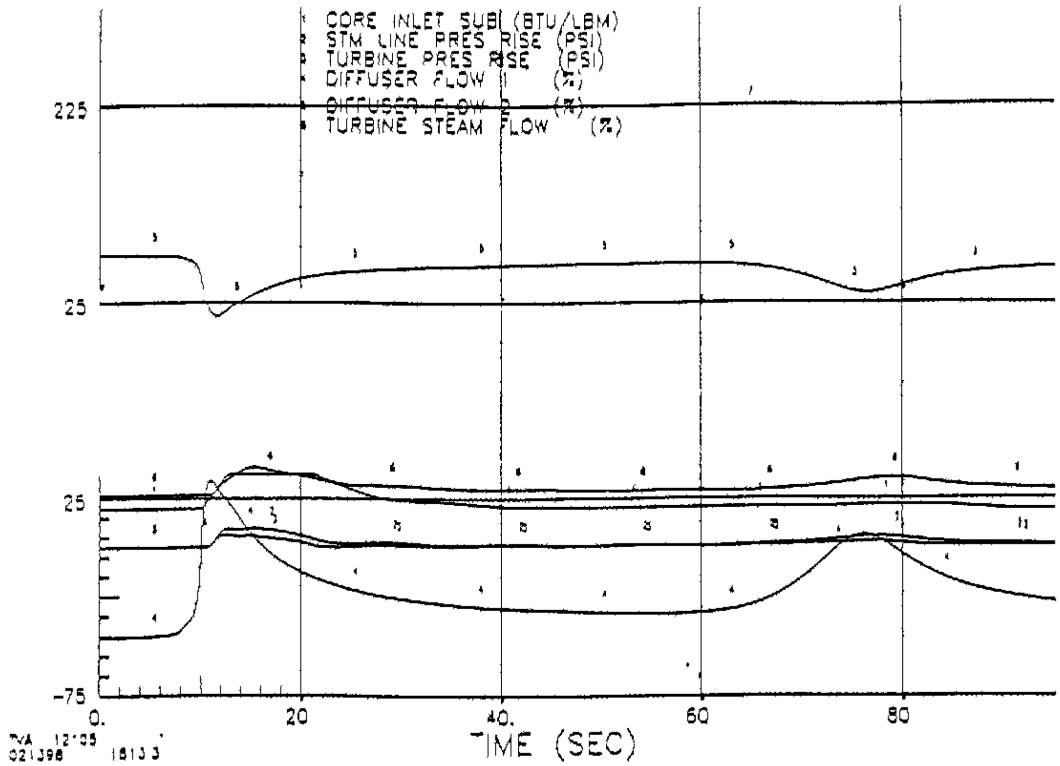


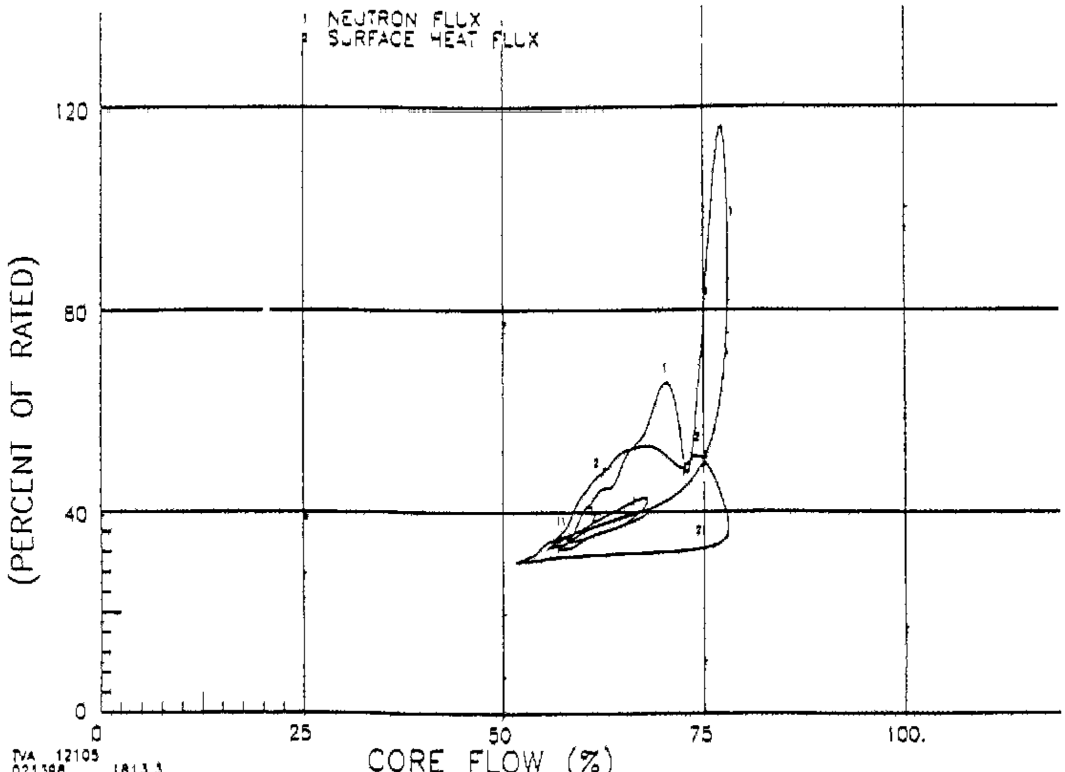
AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

IDLE RECIRCULATION LOOP
STARTUP 30P/52F
COUPLER POSITION 19%

FIGURE 14.5-27a



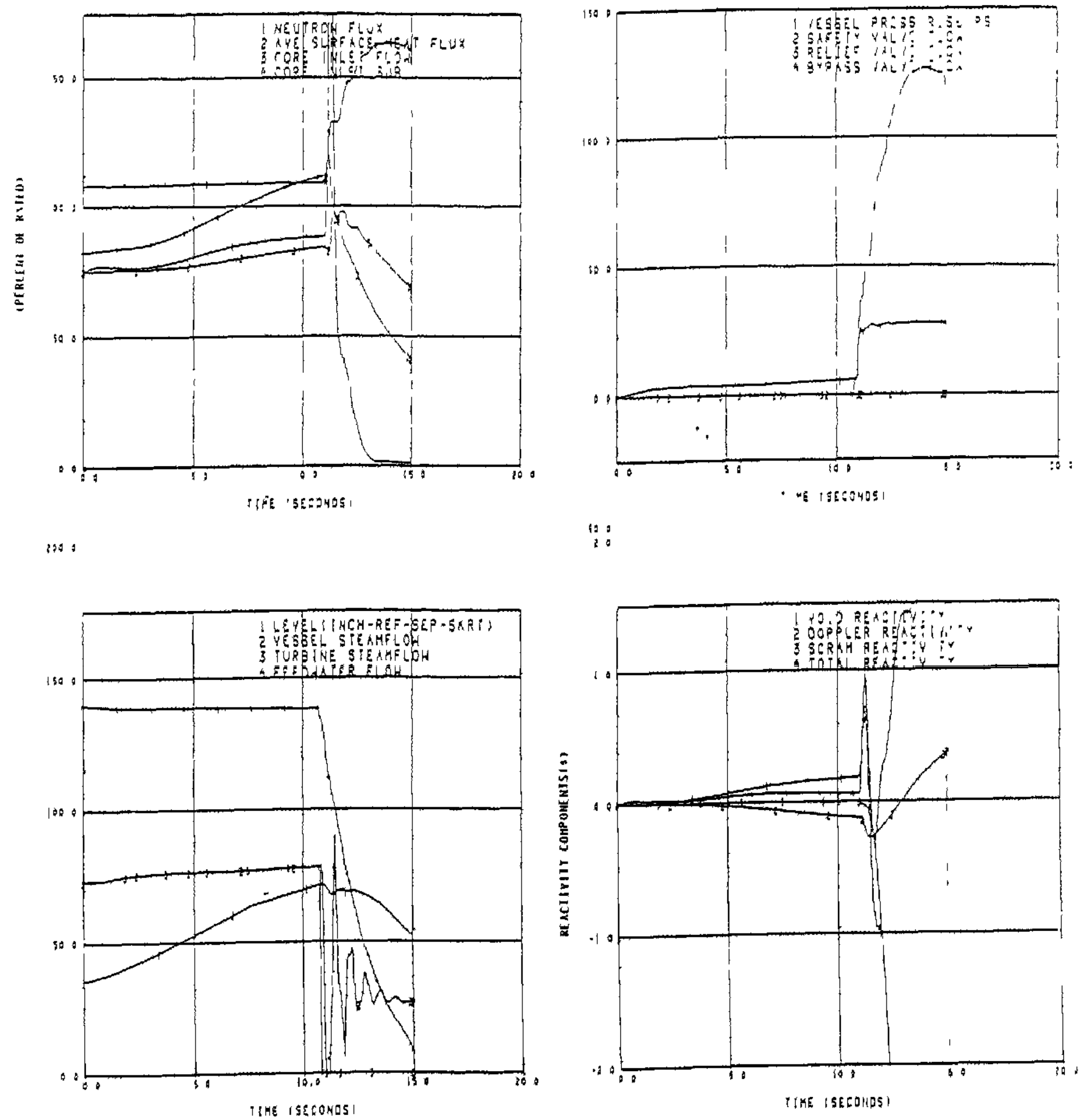


AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

IDLE RECIRCULATION LOOP
STARTUP 30P/52F
COUPLER POSITION 19%

FIGURE 14.5-27b

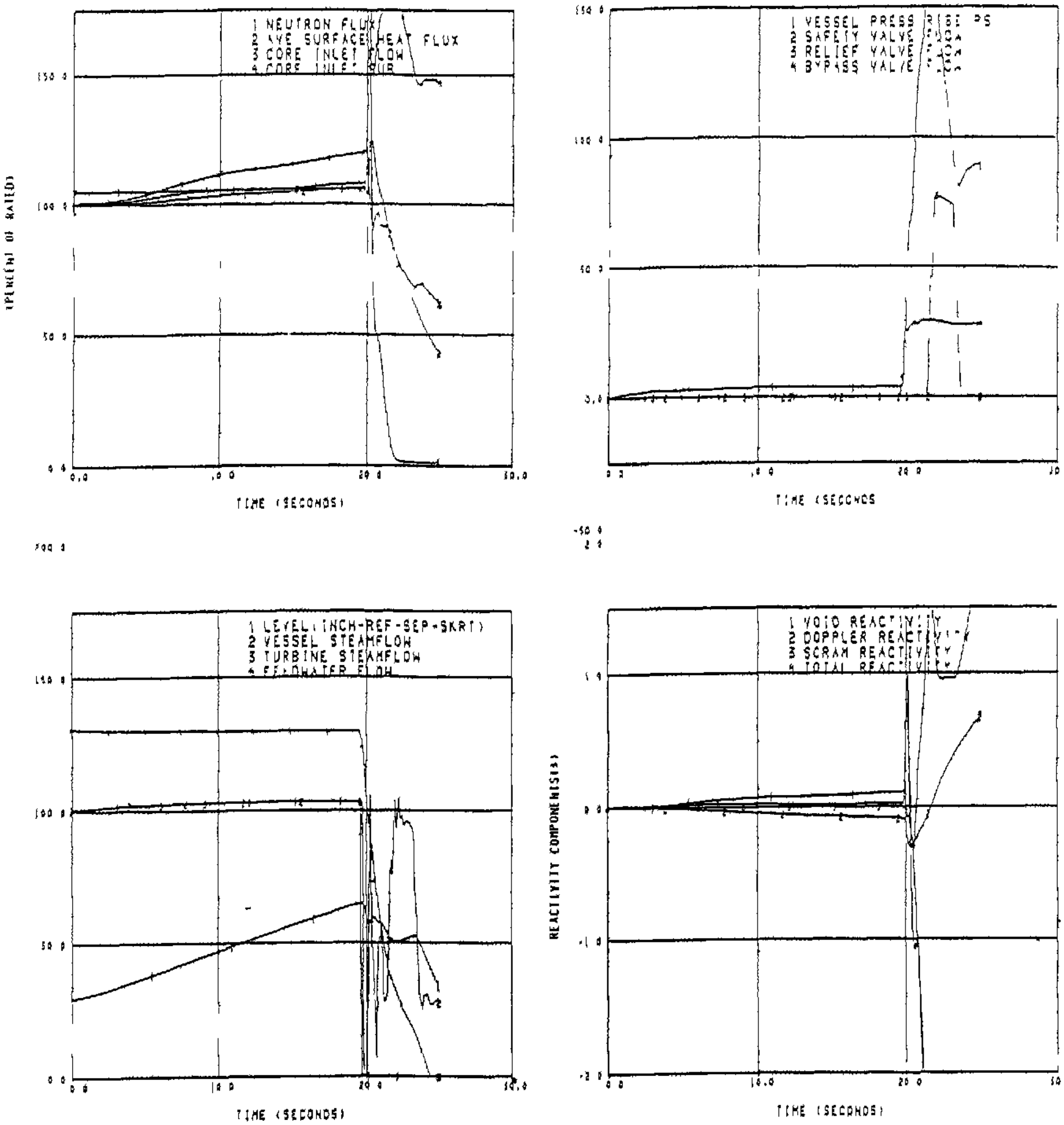


AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

FEEDWATER CONTROLLER
FAILURE-MAXIMUM DEMAND 75P/108F

FIGURE 14.5-28

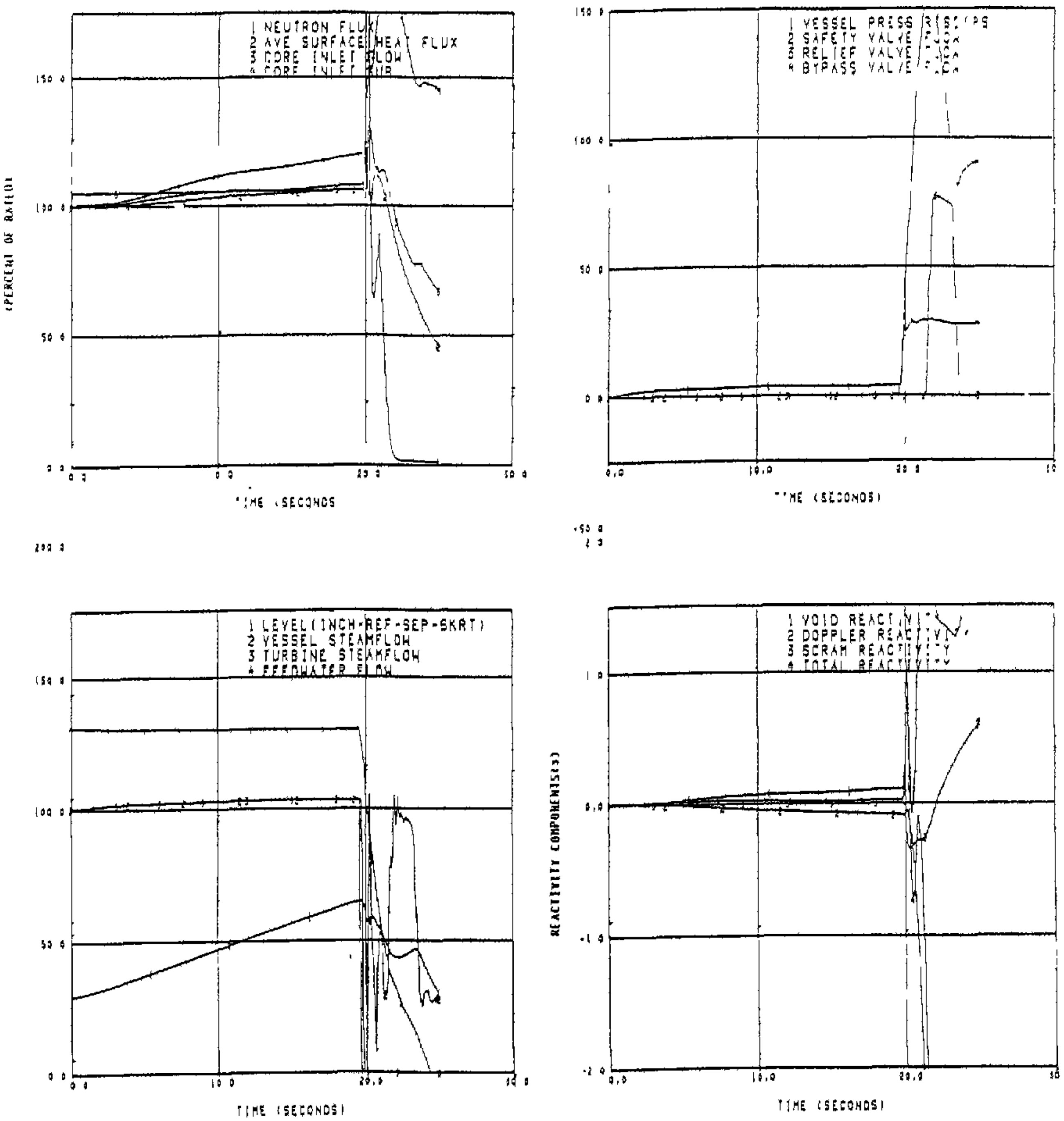


AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

FEEDWATER CONTROLLER
FAILURE-MAXIMUM DEMAND 100P/105F

FIGURE 14.5-29

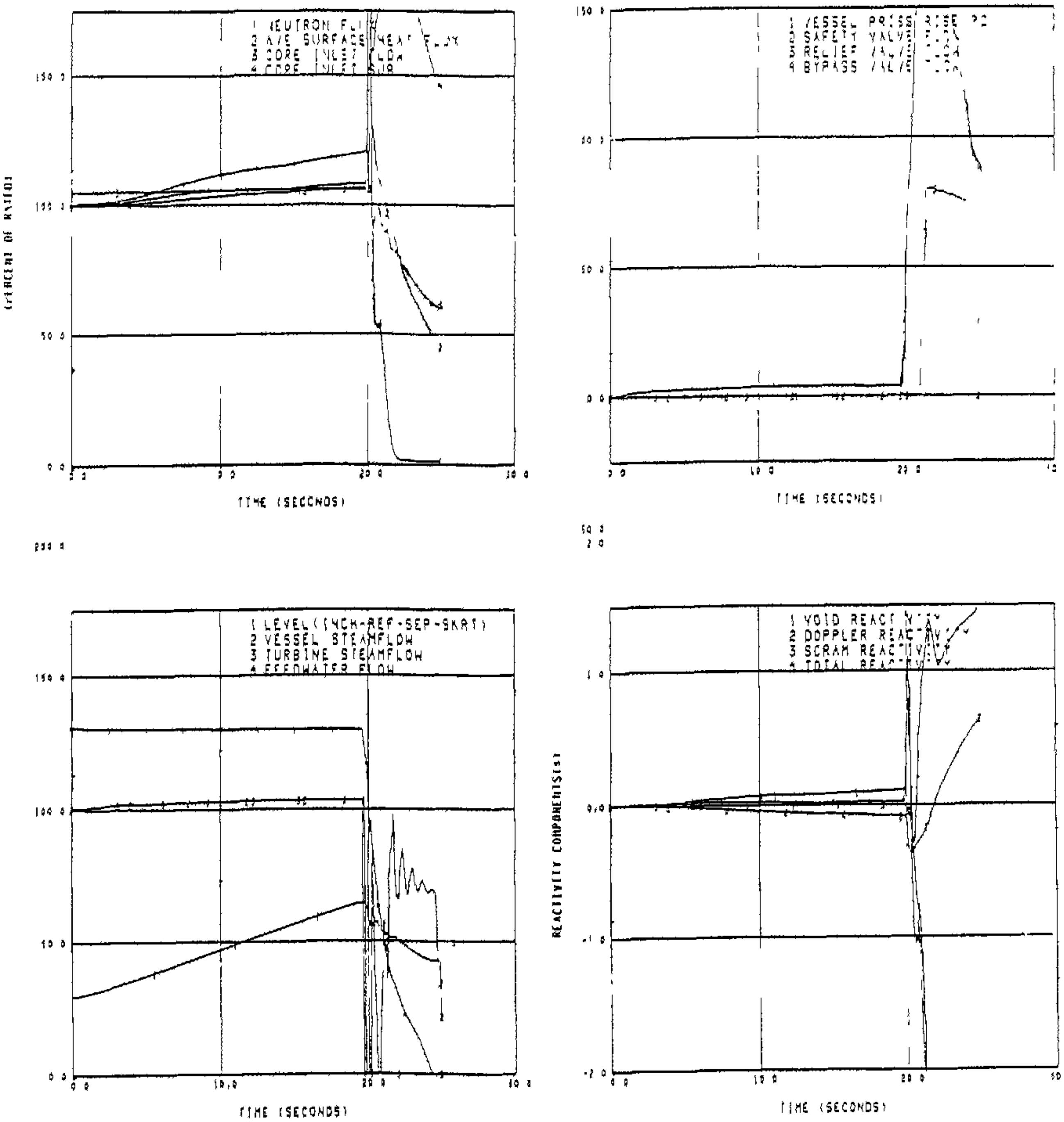


AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

FEEDWATER CONTROLLER
FAILURE-MAXIMUM DEMAND/EOC-RPT-OOS
100P/105F

FIGURE 14.5-30



AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

FEEDWATER CONTROLLER
FAILURE-MAXIMUM DEMAND/TBP-OOS
100P/105F

FIGURE 14.5-31

Table 14.5-1
TRANSIENT ANALYSES POWER/FLOW STATE POINTS
(100P = 3458 MWt, 100F = 102.5E6 lbm/hr)

TRANSIENT	30P/52F	75P/52F	75P/108F	100P/81F	102P/81F	100P/100F	100P/105F	102P/105F	102P/100F
Load Rejection No Bypass							X		
LRNBP/EOC-RPT-OOS							X		
Loss of Condenser Vacuum								X	
Turbine Stop Valve Closure/TT								X	
TSVC/TT-NBP, HP							X		
TSVC/TT-NBP, LP	X								
Closure of All MSIVs								X	
Closure of One MSIV								X	
Loss of Feedwater Heater-REDY					X				
Loss of Feedwater Heater-PANACEA				X					
Inadvertent Pump Start					X				
Pressure Regulator Failure Open									X
Inadvertent Opening of a Relief Valve									X
Loss of Feedwater Flow									X
Loss of Auxiliary Power Transformers									X
Loss of Auxiliary Power - Grid Connections								X	
Recirculation Flow Control Failure-Decreasing Flow									X
One Recirculation Pump Trip									X
Two Recirculation Pumps Trip						X (VFD)			X (MG Set)
One Recirculation Pump Seizure									X
Recirculation Flow Control Failure-Increasing Flow		X							
Startup of Idle Recirculation Loop	X	X							
FW Control Failure-Maximum Demand			X				X		
FWCF - EOC-RPT-OOS							X		
FWCF - TBP-OOS							X		

Table 14.5-2
TRANSIENT ANALYSES INITIAL CONDITIONS
(Power Uprate)

Parameter	GE UFSAR Analysis (at percent power)	Framatome ANP Reload Analysis
Thermal Power, MWt	3458 (100%) / 3527 (102%)	3458 (100%)
Core Flow, Mlb/hr	102.5	102.5
Core Flow Range (% of current rated)	81-105	81-105
Vessel Steam Flow and FW flow, Mlb/hr	14.24 (100%) / 14.57 (102%)	14.15
Analysis Dome Pressure, psia	1055 (100%) / 1070 (102%)	1050
Analysis Turbine Pressure, psia	995 (100%) / 1010 (102%)	985
Feedwater Temperature, °F	382 (100%) / 384 (102%)	382
Turbine Bypass Capacity	25.2% of rated vessel steam flow	
Number of MSRVs	13	13
MSRV type	Target Rock	Target Rock
Opening response of relief functions	0.15 s	0.15 s
Opening delay of relief functions	0.4 s	0.45s
MSRV Capacity ,% rated steam flow (Based on 1090 psig setpoint)	73.8% [i] (12 valves)	73.8% [i] (12 valves)
MSRV Setpoint, (number of valves @ psig) (+3% setpoint tolerance included)	4 @ 1174[ii] 4 @ 1185 5 @ 1195	4 @ 1169[ii] 4 @ 1179 4 @ 1190
MCPR Safety Limit	1.10	Cycle Specific
Recirculation Flow Control	VFD Flow Control	VFD Flow Control
Core Average Gap Conductance (Btu/s-sq. ft -Deg F)	0.3972	Case Dependent
High Neutron Flux Scram Setpoint	125.4% of rated power	125.4% of rated power
High Pressure Scram Setpoint, psig	1106	1101
High Pressure ATWS-RPT setpoint, psig	1153	1177
Reactor L8 Water Level, in avz[iii]	588	588
Reactor L3 Water Level, in avz[iii]	518	518
Reactor L2 Water Level, in avz[iii]	448	448
Reactor L1 Water Level, in avz[iii]	372.5	372.5

i Referenced to rated vessel steam flow at 3458 MWt. The absolute MSRV capacity at 1090 psig does not change with power uprate.
ii Considered only 3 out of 4 due to 1 MSRV-OOS
iii Above vessel zero

14.6 ANALYSIS OF DESIGN BASIS ACCIDENTS - UPRATED

This section contains general descriptions of the evaluation of design basis accidents for BFN Units 2 and 3 at uprated conditions. The similar results at pre-uprated conditions can be found in Section 14.11.

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. For GE analyzed reload cycles in which the BPWS is utilized, a cycle specific CRDA analysis is not required. For GE analyzed reload cycles, the cycle specific CRDA results or a commitment to employ BPWS are contained in the Reload Licensing Report. For FANP analyzed reload cores, the cycle specific CRDA results are provided in the Reload Licensing Analysis Report.

The BPWS is an improvement over previous group notch sequences with regards to reducing maximum incremental control rod worths. It virtually eliminates the CRDA as an accident of any concern not because it eliminates the possibility of a rod drop occurring, but because the BPWS maintains incremental rod worths to such low values.[2, 3]

The 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.

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1 NEDE 24011-P-A, GESTAR II

2 NEDO 10527 including Supplements 1 and 2, Rod Drop Accident Analysis for Large BWRs, March 1972

3 NEDO 21231, Bank Position Withdrawal Sequence, January 1977

14.6.2.1 Excursion Analysis Assumptions for GE Analyzed Reload Cores

The following assumptions are used in the analysis of the nuclear excursion for each case:

a. The velocity at which the control rod falls out of the core is assumed to be 5 ft/sec. The control rod velocity limiter[4] an engineered safeguard, limits the rod drop velocity to less than this value.

b. No credit is taken for the IRM or 15% APRM scram signals. Control rod scram motion is assumed to start at about 200 milliseconds after the neutron flux has attained 120 percent of rated flux. This assumption allows the power transient to be terminated initially by the Doppler reactivity effect of the fuel.

c. No credit is taken for the negative reactivity effect resulting from the increased temperature of, or void formation in the moderator because the time constant for heat transfer between the fuel and the moderator is long compared with the time required for control rod motion.

d. No credit is taken for the prompt negative reactivity effect of heating in the moderator due to gamma heating and neutron thermalization.

e. Scram times for the control rods is conservatively assumed to be equal to or greater than the Technical Specification limits. The scram rates which were used in this analysis are tabulated below.

Percent of Rod Insertion	Time (second)
5%	0.475
20%	1.10
50%	2.0
90%	5.0

f. The rod drop accident was evaluated at the time in the fuel cycle at which the consequences are worst.

4 "Control Rod Velocity Limiter," General Electric Company, Atomic Power Equipment Department, March 1967 (APED-5446).

14.6.2.2 CRDA Analysis and Results for FANP Licensed Reload Cores

The FANP analytical methods, assumptions, and conditions for evaluating the excursion aspects of the control rod drop accident have been reviewed and approved by the NRC. FANP has performed and submitted a generic analysis that correlates deposited enthalpy from a postulated CRDA to steady state parameters calculated on a cycle specific basis. Analyses are performed assuming BPWS rules or equivalent are in force to limit dropped rod worths to reasonable values. The FANP cycle specific application of the generic CRDA methodology shows that peak deposited enthalpies do not exceed 380 cal/gm. For FANP 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 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 FANP analysis, rods with deposited enthalpies exceeding 170 cal/gm are assumed to fail. If the number of rods exceeding the failure threshold is shown to be below 850, 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/gm 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/gm threshold is protected and that the radiological evaluation accounting for 850 failed fuel rods remains applicable for FANP fuel.

14.6.2.3 Fuel Damage

Fuel rod damage estimates were initially based upon the UO_2 vapor pressure data of Ackerman[5] 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/gm. Two especially applicable sets of data come from the PULSTAR[6] and ANL-TREAT[7] 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/gm with a minimum period of 2.83 milliseconds. The coolant flow was by natural 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/gm)	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/gm is the threshold for eventual fuel cladding perforation. Fuel melting is estimated to occur in the 220 to 280 cal/gm range, and a minimum of 425 cal/gm is required to cause immediate rupture of the fuel rods due to UO_2 vapor pressures.

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5 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.

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

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

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/gm. 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/gm which is less than both the 170 cal/gm and 280 cal/gm 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 37 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[8].

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[8]:

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[8].

8 Regulatory Guide 1.183 and NUREG-0800, Section 15.4.9.

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 temperature for GE fuel, the limiting break is a 4.2 sq ft break in a recirculation suction line as documented in NEDC-32484(P)(A) and for FANP fuel, the limiting break is a 0.5 sq ft split in a recirculation discharge line as documented in EMF-2950(P)[B].

Information on GE LOCA models currently in use is given in NEDO-20566[9] and NEDC-32484P[10]. LOCA models used for FANP reload fuel analyses are described in EMF-2361(P)(A)[A]. 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 NEDC-32484P and NEDO-10320[11].

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.

A EMF-2361(P)(A) Revision 0, EXEM BWR-2000 ECCS Evaluation Model, Framatome ANP, May 2001.

B EMF-2950(P) Revision 1, Browns Ferry Units 1, 2, and 3 Extended Power Uprate LOCA Break Spectrum Analysis, April 2004

9 General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10CFR50 Appendix K. NEDO-20566.

10 General Electric SAFER/GESTR-LOCA, Loss of Coolant Analysis, Browns Ferry Units 1, 2, and 3, NEDC-32484P, Rev. 6.

11 The General Electric Pressure Suppression Containment Analytical Model, NEDO-10320.

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.

14.6.3.3 Primary Containment Response

BFN Units 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 reactor pressure vessel (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 SRVs, 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 Residual Heat Removal (RHR) System to remove decay heat which is transferred from the RPV to the suppression pool.

The Primary Containment System requirements are:

Design Pressure	56 psig
Design Temperature	281°F

Minimum containment overpressure following a LOCA and its affect on NPSH for Core Spray and RHR pumps is discussed in Chapter 6.5.5.

14.6.3.3.1 Initial Conditions and Assumptions

The following assumptions and initial conditions were used in calculating the effects of a loss of coolant accident on the primary containment. (These assumptions are in addition to those specified for the loss of coolant accident described in paragraph 14.6.3.1.)

a. The reactor is assumed to be initially operating at the conditions specified in Table 14.6-3. Tables 14.6-4 and 14.6-5 provide additional conditions that apply for the short term containment response and long term containment response, respectively.

b. The reactor is assumed to go subcritical at the time of accident initiation due to void formation in the core region. Scram also occurs in less than one second from receipt of the high drywell pressure and low water level signals, but the difference in shutdown time between zero and one second is negligible.

c. The sensible heat released in cooling the fuel to the normal primary system operating saturation temperature and the core decay heat were included in the reactor vessel depressurization calculation. Initial high vessel pressure increases the calculated flow rates out of the break; this is conservative for containment analysis purposes.

d. The main steam isolation valves were assumed to start closing at 0.5 seconds after the accident, and the valves were assumed to be fully closed in the shortest possible time of three seconds following closure initiation. Actually, the closures of the main steam isolation valves are expected to be the result of low water level, so these valves may not receive a signal to close for over four seconds; and the closing time could be as high as 10 seconds. By assuming rapid closure of these valves, the reactor vessel is maintained at a high pressure which maximizes the discharge of high energy steam and water into the primary containment.

e. For the short term containment response analysis, the feedwater flow is assumed to coast down to zero at four seconds into the event. This conservatism is used because the relatively cold feedwater flow, if considered to continue, tends to depressurize the reactor vessel, thereby, reducing the discharge of steam and water into the primary containment.

f. For the long term containment response analysis, the reactor feedwater flow into the reactor continues until all the high energy feedwater (water that would contribute to heating the pool) is injected into the vessel.

g. The pressure response of the containment is calculated assuming:

    1. Thermodynamic equilibrium in the drywell and pressure suppression chamber. Because complete mixing is nearly achieved, the error

introduced by assuming complete mixing is negligible and in the conservative direction.

2. The constituents of the fluid flowing in the drywell to pressure suppression chamber vents are based on a homogeneous mixture of the fluid in the drywell. The consequences of this assumption result in complete liquid carryover into the drywell vents. Actually, some of the liquid will remain behind in a pool on the drywell floor so that the calculated drywell pressure is conservatively high.

3. The flow in the drywell pressure suppression pool vents is compressible except for the liquid phase.

4. No heat loss from the gases inside the primary containment is assumed.

h. The limiting core/containment cooling configuration assumed is the availability of one reactor core spray loop and one RHR loop consisting of two RHR pumps and associated heat exchangers and two associated RHR service water pumps.

i. For the long term containment response analysis, LPCI and core spray are used to cool the core for the first 600 seconds. After 600 seconds, it is assumed that containment cooling is manually initiated using containment spray.

14.6.3.3.2 Containment Response

The containment performance for the DBA-LOCA response is typically divided into two phases: the short-term initial blowdown period (approximately 30 seconds following a LOCA) and the long-term period which includes the time period after the containment cooling system starts. The short-term containment response determines the peak drywell pressure and the peak drywell LOCA temperature. The long-term containment response determines the peak wetwell (suppression pool) temperature and pressure.

The following subsections provide a description of the dynamics of the containment response during a LOCA along with the calculational methods and results of the short term and long term containment response at power uprated conditions.

14.6.3.3.2.1 LOCA Dynamics

Following the initiation of the LOCA, the primary coolant from the reactor vessel is discharged into the drywell. Most of the noncondensible gases are forced into the suppression chamber during the vessel depressurization phase. However, the noncondensibles soon redistribute between the drywell and the suppression chamber via the vacuum breaker system as the drywell pressure decreases due to steam condensation. The Core Spray System removes decay heat and stored heat from the core, thereby controlling core heatup. The core spray water transports the core heat out of the reactor vessel through the broken recirculation line in the form of hot water. This hot water flows into the pressure suppression chamber via the drywell-to-pressure suppression chamber vent pipes. Steam flow is negligible. The energy transported to the pressure suppression chamber water is then removed from the primary containment system by the RHRS heat exchangers.

Prior to activation of the RHRS containment cooling mode (arbitrarily assumed at 600 seconds after the accident), the RHRS pumps (LPCI mode) have been adding liquid to the reactor vessel. After the reactor vessel is flooded to the height of the jet pump nozzles, the excess flow discharges through the recirculation line break into the drywell. This flow offers considerable cooling to the drywell and causes a depressurization of the containment as the steam in the drywell is condensed. At 600 seconds, the RHRS pumps are assumed to be switched from the LPCI mode to the containment cooling mode. The containment spray would normally not be activated at all, and the changeover to the containment cooling mode need not be made for several hours. There is considerable time available to place the containment cooling system in operation because about eight hours will pass before the maximum allowable pressure is reached with no containment cooling.

14.6.3.3.2.2 Short-Term Response

The short-term containment pressure and temperature response was re-analyzed at power uprate conditions in accordance with Regulatory Guide 1.49[12] and NEDO-31897[13], using the GE proprietary computer code M3CPT05V. The modeling used in M3CPT is described in NEDO -10320[14], NEDO-20533[15], and NEDE-20566-P-A[16]. The short-term containment response is controlled by the reactor blowdown during the LOCA. The reactor blowdown rate is dependent on the reactor initial thermal hydraulics conditions, such as vessel dome pressure and the mass and energy of the fluid inventory in the RPV. However, the reactor blowdown is relatively insensitive to the initial reactor power.

12 Regulatory Guide 1.49

13 GE Nuclear Energy, "Generic Guidelines for GE Boiling Water Reactor Power Uprate," Licensing Topical Report NEDO-31897, Class I (Non-proprietary), February 1992; and NEDC-31897P-A, Class III (Proprietary), May 1992

14 NEDO-10320, "The GE Pressure Suppression Containment Analytical Model," April 1971

15 NEDO-20533, "The General Electric Mark III Pressure Suppression Containment System Analytical Model," June 1974

16 NEDO-21052, "Maximum Discharge of Liquid-Vapor Mixtures from Vessels," September 1975

The M3CPT analyses were performed using blowdown flow rates based on the GE code LAMB08 blowdown model[17]. In using the LAMB blowdown model, the blowdown flow rates were calculated first. The LAMB flow rates were then used as input to M3CPT.
The following four reactor operating points on the power/flow map were selected for evaluation to envelope the full range of reactor operating conditions:

Case 1 -	102% of uprated power, 100% core flow with normal feedwater temperature.
Case 2 -	102% of uprated power, 100% core flow with feedwater temperature reduction.
Case 3 -	102% of uprated power, 81% core flow with feedwater temperature reduction [MELLLA point].
Case 4 -	63% of uprated power, 38% core flow with feedwater temperature reduction [natural circulation line-MELLLA rod line intersection].

The containment response for the Increased Core Flow (ICF) state point was not analyzed since it is bounded by the containment response for the above power/flow state points.

Table 14.6-6 presents the results of all the power/flow state points analyzed. The results demonstrate that the maximum drywell pressure and maximum differential pressure between the drywell and wetwell during operation at uprated power remain within the containment design limits.

The peak drywell pressures for all points analyzed are well below the design limit. The highest peak short-term drywell pressure and temperature for power uprate conditions (50.6 psig, 297°F) occur at the MELLLA point (Case 3). Although the calculated peak drywell atmosphere temperature is higher than the drywell shell design value of 281°F, the shell temperature will not exceed 281°F. This is because drywell atmosphere temperature exceeds 281°F for a short duration following the blowdown, and it would take a longer time for the drywell shell to heat up to 281°F. Thus, the drywell shell is expected to remain below the design temperature of 281°F. Additionally, the safety components in the drywell that must function following a LOCA have been successfully tested in a steam atmosphere at higher temperatures than the containment design temperature of 281°F (FSAR Section 12.2.2.7.3).

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17 NEDE-20566-P-A, "General Electric Model for LOCA Analysis in Accordance with 10CFR50 Appendix K," September 1986

Plots showing the limiting DBA-LOCA short-term temperature and pressure response in the drywell and wetwell at power uprate conditions are given in Figures 14.6-1 and 14.6-2, respectively.

14.6.3.3.2.3 Long-Term Response

As the operating power level is increased due to power uprate, the decay power increases and the long-term pressure suppression pool temperature will potentially increase. The most limiting DBA-LOCA case with respect to peak pressure suppression pool temperature, a double-ended recirculation suction line break, was analyzed at power uprate conditions using the SHEX-04V code[18]. In the long-term response evaluation at power uprate conditions, the ANSI/ANS 5.1 - 1979 decay heat model plus 2σ uncertainty was used.

The results of the analysis shows the peak pressure suppression pool temperature is less than 177°F for 105% power uprate. Figures 14.6-3 and 14.6-4 show the long term wetwell and drywell temperature response on Units 2 and 3. Figure 14.6-5 provides the long term pressure response of the drywell and wetwell on Units 2 and 3. The same case was re-analyzed at the pre-uprate power conditions to assess the impact of power uprate on peak pool temperature on a common analysis basis. The comparison indicates that power uprate increases the peak suppression pool temperature by 2°F. For Unit 1 the peak suppression pool temperature is 187.3°F, which is based on a 120% power uprate analysis.

For Units 2 and 3, the unlikely occurrence that the RHR service water temperature exceeds the design value of 92°F, an allowable derated operating power map has been developed to enable the operator to determine the maximum allowed operating power limit for a range of service water temperatures. This power map is included in the Technical Specification for the ultimate heat sink. The limit assumes the plant power level has been within the limit for a long enough period of time such that it can be considered a steady state condition. This assumption is required because during a power reduction the total decay heat lags the instantaneous power level. Based on historical operating plant data, 95°F is chosen as the upper bound for the RHR service water temperature range. A long-term containment sensitivity study was performed to identify the maximum acceptable core thermal power as a function of RHR service water temperature in order to maintain the peak pressure suppression pool temperature at 177°F and, thus, satisfy the temperature

18 Letter to Patrick W. Marriot (GE) from William T. Russel (NRC) forwarding the Staff Position Power on General Electric Boiling Water Reactor Power Uprate Program (TAC No. M79384), September 30, 1991

limit per the Torus Integrity Long-Term Program Plant Unique Analysis Report[19]. The Unit 1 analysis is based on a RHR service water temperature of 95°F and the Unit 1 Technical Specifications do not include a derated power operating map. On Unit 1, the peak suppression pool temperature is 187.3°F.

14.6.3.3.3 Metal Water Reaction Effects on the Primary Containment

If Zircaloy in the reactor core is heated to temperatures above about 2000°F in the presence of steam, a chemical reaction occurs in which zirconium oxide and hydrogen are formed. This is accompanied with an energy release of about 2800 Btu per pound of zirconium reacted. The energy produced is accommodated in the pressure suppression chamber pool. The hydrogen formed, however, will result in an increased long term drywell pressure due simply to the added volume of gas to the fixed containment volume. Although very small quantities of hydrogen are produced during the accident, the containment has the inherent ability to accommodate a much larger amount as discussed below. The containment pressure response curves presented in Section 14.6.3.3.2 do not reflect the negligible long term pressure increase due to this phenomena.

The basic approach to evaluating the capability of a containment system with a given containment spray design is to assume that the energy and gas are liberated from the reactor vessel over some time period. The rate of energy release over the entire duration of the release is arbitrarily taken as uniform, since the capability curve serves as a capability index only, and is not based on any given set of accident conditions as an accident performance evaluation might be.

It is conservatively assumed that the pressure suppression pool is the only body in the system which is capable of storing energy. The considerable amount of energy storage which would take place in the various structures of the containment is neglected. Hence, as energy is released from the core region, it is absorbed by the pressure suppression pool. Energy is removed from the pool by heat exchangers which reject heat to the service water. Because the energy release is taken as uniform and the service-water temperature and exchanger flow rate are constant, the temperature response of the pool can be determined. It is assumed that the pressure suppression chamber gases are at the pressure suppression chamber water temperature.

The extent of the metal-water reaction is less than 0.1 percent of all the zirconium in the core. As an index of the containment's ability to tolerate postulated metal-water reactions, the concept of "Containment Capability" is used. Since this capability

19 Report CEB-83-34 R2, "Browns Ferry Nuclear Plant Torus Integrity Long-Term Program Plant Unique Analysis Report (PUAR)"

depends on the time domain, the duration over which the metal-water reaction is postulated to occur is one of the parameters used.

Containment capability is defined as the maximum percent of fuel channels and fuel cladding material which can enter into a metal-water reaction during a specified duration without exceeding the maximum allowable pressure of the containment. To evaluate the containment capability, various percentages of metal-water reaction are assumed to take place over certain time period. This analysis presents a method of measuring system capability without requiring prediction of the detailed events in a particular accident condition.

Since the percent metal-water reaction capability varies with the duration of the uniform energy and gas release, the percent metal-water reaction capability is plotted against the duration of release. This constitutes the containment capability curves as shown in Figure 14.6-6. All points below the curves represent a given metal-water reaction and a given duration which will result in a containment peak pressure which is below the maximum allowable pressure. The calculations are made at the end of the energy release duration because the number of moles of gases in the system in then at a maximum, and the pressure suppression pool temperature is higher at this time than at any other time during the energy release.

It should be noted that the curves are actually derived from separate calculations of two conditions: the "steaming" and the "nonsteaming" situation. The minimum amount of metal-water reaction which the containment can tolerate for a given duration is given by the condition where all of the noncondensible gases are stored in the pressure suppression chamber. This condition assumes that "steaming" from the drywell to the pressure suppression chamber results in washing all of the noncondensible gases into the pressure suppression chamber. This is shown as the flat portion of the containment capability characteristic curve. Activation of containment sprays condense the drywell steam so that no steaming occurs, thus allowing noncondensibles to also be stored in the drywell. This is denoted by the rising (spray) curve. The intersection between the no spray curve and the spray curve represents the duration and metal water reaction energy release which just raises all the spray water to the saturation temperature at the maximum allowable containment pressures.

For durations to the left of the intersection, some steam is generated and all the gases are stored in the pressure suppression chamber. For durations to the right of the intersection, the spray flow is subcooled as it exits from drywell by increasing amounts as the duration is increased.

The energy release rate to the containment is calculated as follows:

$$q_{IN} = \frac{Q_O + Q_{MW} + Q_S}{T_D}$$

where:

q_{IN} = Arbitrary energy release rate to the containment Btu per second,

Q_O = Integral of decay power over selected duration of energy gas release, Btu,

Q_{MW} = Total chemical energy released exothermically from selected metal-water reaction, Btu,

Q_S = Initial internal sensible energy of core fuel and cladding, Btu, and

T_D = Selected duration of energy and gas release, seconds.

The total chemical energy released from the metal-water reaction is proportional to the percent metal-water reaction. The initial internal sensible energy of the core is taken as the difference between the energy in the core after the blowdown and the energy in the core at a datum temperature of 250°F.

The temperature of the drywell gas is found by considering an energy balance on the spray flows through the drywell.

Based upon the drywell gas temperature, pressure suppression chamber gas temperature, and the total number of moles in the system, as calculated above, the containment pressure is determined. The containment capability curves in Figure 14.6-6 present the results of the parametric investigation.

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 37 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 to the top of the stack via leakage of the hardened wetwell vent 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. CAD system flow rate is 139 cfm for 24 hours at 10 days, 20 days, and 29 days.

e. The hardened wetwell vent isolation valves leak a total of 10 scfh to the top of the offgas stack. This leakage is assumed to begin at 8 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 Atmospheric Dilution (CAD) system 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 1.31E5ft^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 three 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. The hardened wetwell vent isolation valves leak a total of 10 scfh to the top of the offgas stack with a delay of 8 hours for the leakage to reach the stack. The hardened wetwell vent isolation valve leakage enters the stack above the divider deck and exits the top of the stack.

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.

<u>Main Steam Isolation Valve Leakage Releases</u>

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 vents 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 has been performed as a comparison of the fuel damage to the 7x7 fuel design. Fuel types evaluated include 8x8, 8x8R, 9x9, GE-14 (10x10) and Framatome Atrium 10 (A-10). The activity release for each of these 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 FANP ATRIUM-10 fuel, including ATRIUM-10 fuel containing blended, low-enriched uranium (BLEU). The FANP ATRIUM-10 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 ATRIUM-10 assembly will produce similar number of rod failures as in the GE14 design. The EOC exposure of the initial reload of ATRIUM-10 fuel at the end of the first cycle or irradiation is bounded by the EOC exposure assumed for the source terms of the GE fuel used in the current licensing basis analysis. Furthermore, the mass of the GE fuel rods assumed to fail in the current licensing basis analysis bounds that of the ATRIUM-10 fuel. For these reasons, fuel cycles containing initial reloads of new ATRIUM-10 fuel coresident with previously exposed GE fuel are bounded by the current licensing basis refueling accident analysis.

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:

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

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

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:

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

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

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 37 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 Commissions 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 10.5 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 10-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 feedpumps 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.[20] 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.

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

20 Moody, F. J.: "Two Phase Vessel Blowdown From Pipes", Journal of Heat Transfer, ASME Vol, 88, August 1966, page 285.

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 core flow and MCHFR during the first 10.5 seconds of the accident are shown in Figures 14.6-9 and 14.6-10. Since 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 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 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 <u>Fission Product Release From Break</u>

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

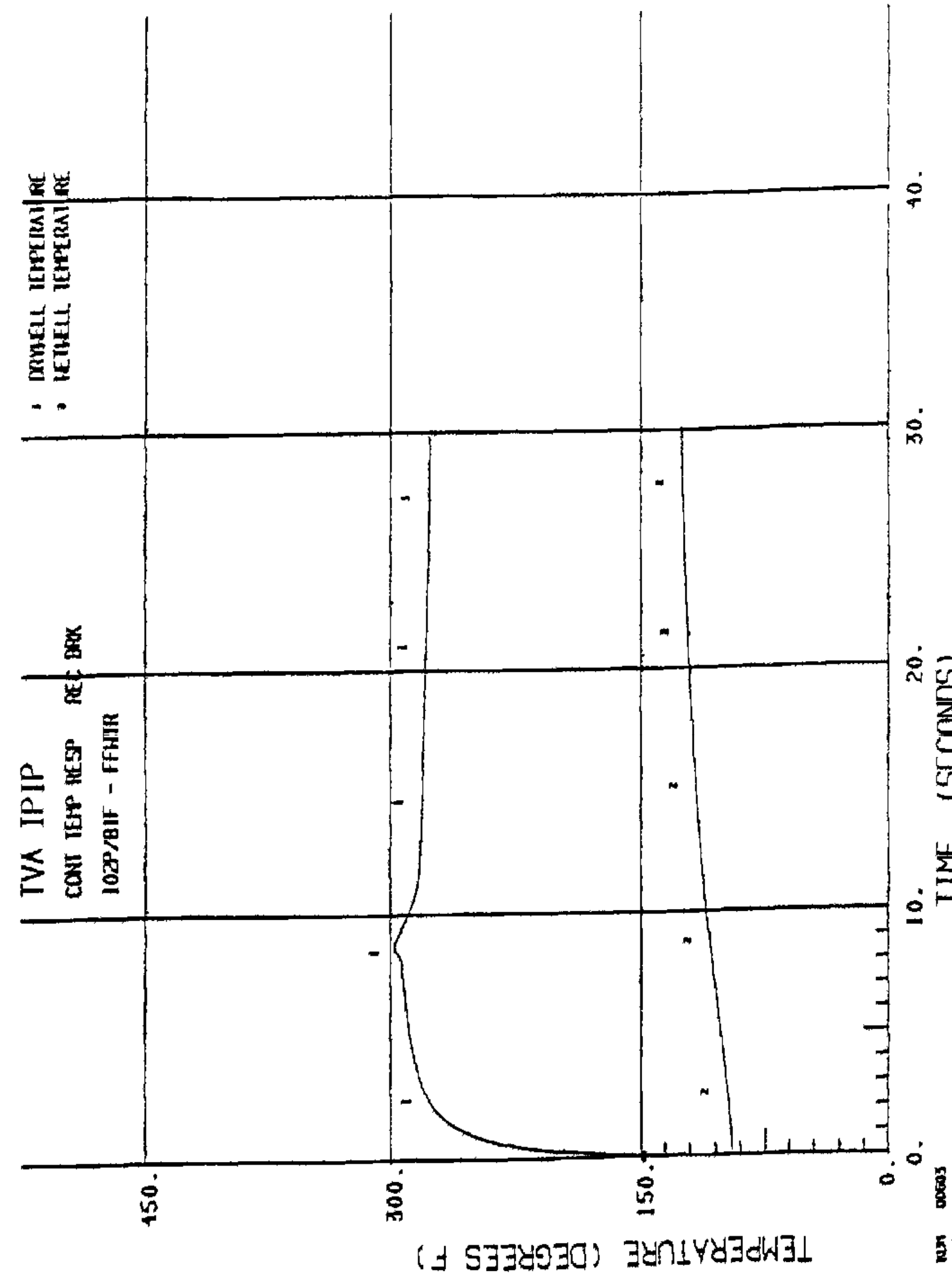
Noble gases	1.342×10^3 Ci
Iodine 131	5.254×10^1 Ci
Iodine 132	4.737×10^2 Ci
Iodine 133	3.533×10^2 Ci
Iodine 134	8.549×10^2 Ci
Iodine 135	5.031×10^2 Ci

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 <u>Radiological Effects</u>

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.

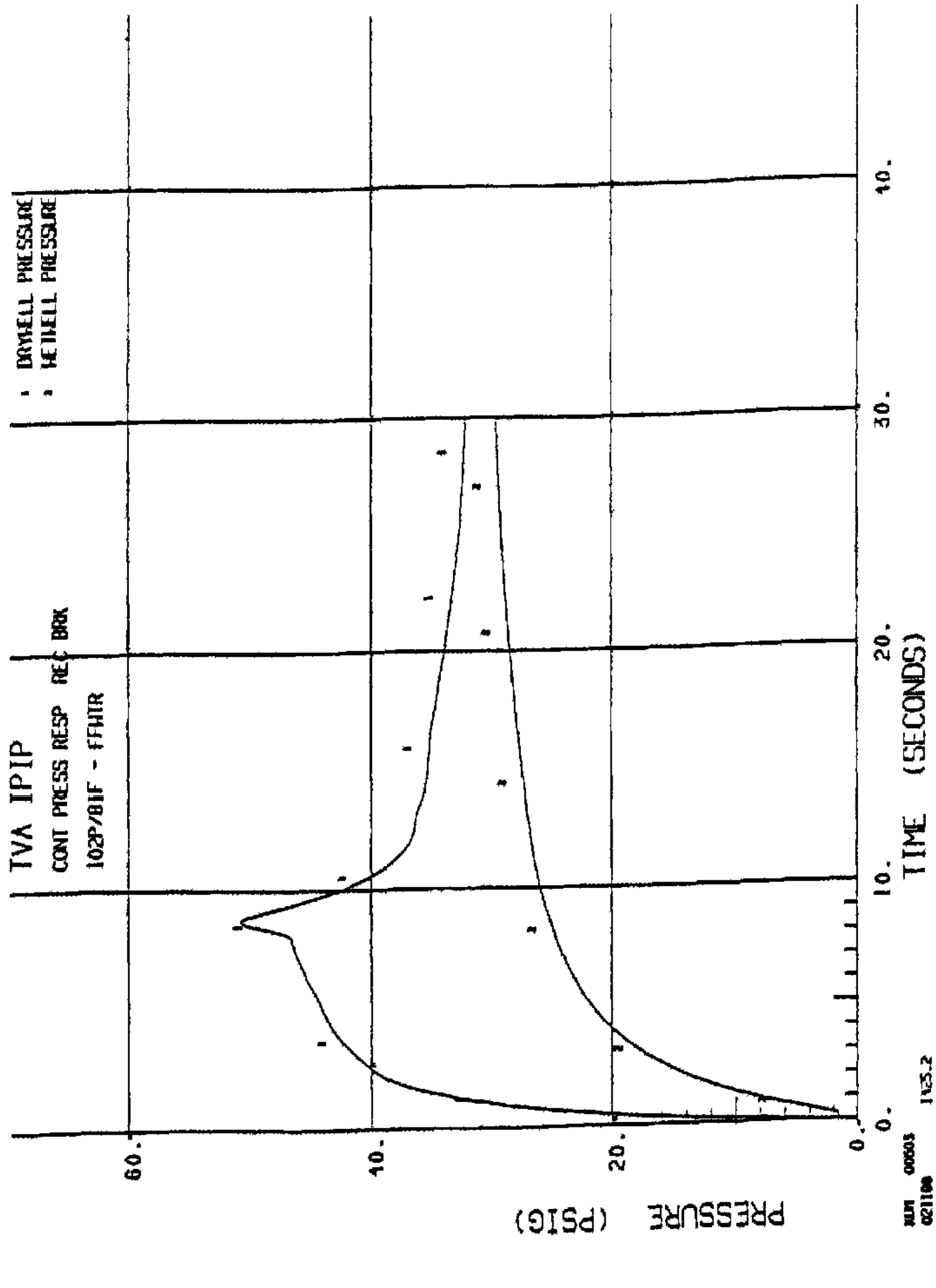


AMENDMENT 17

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

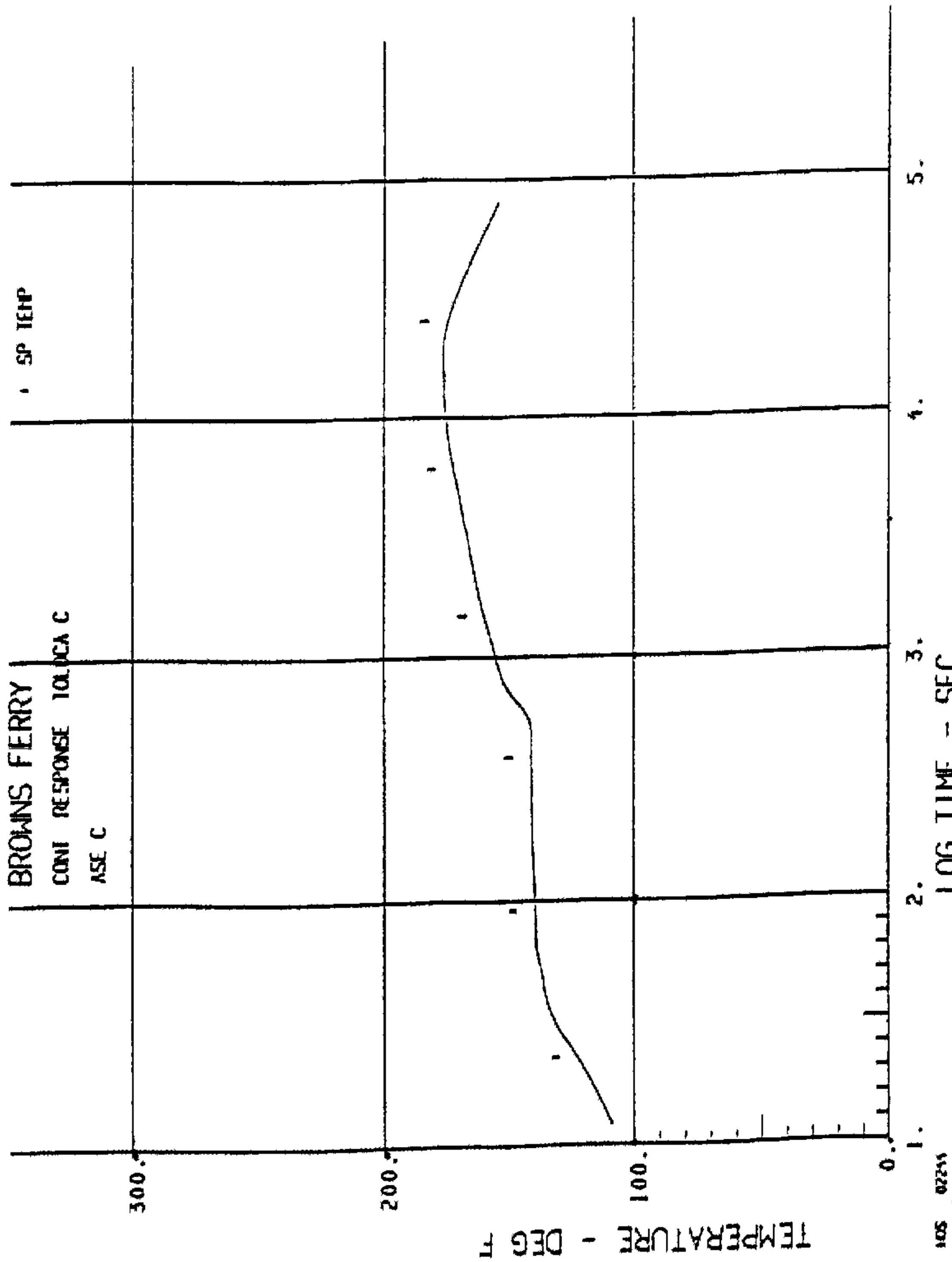


AMENDMENT 17

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

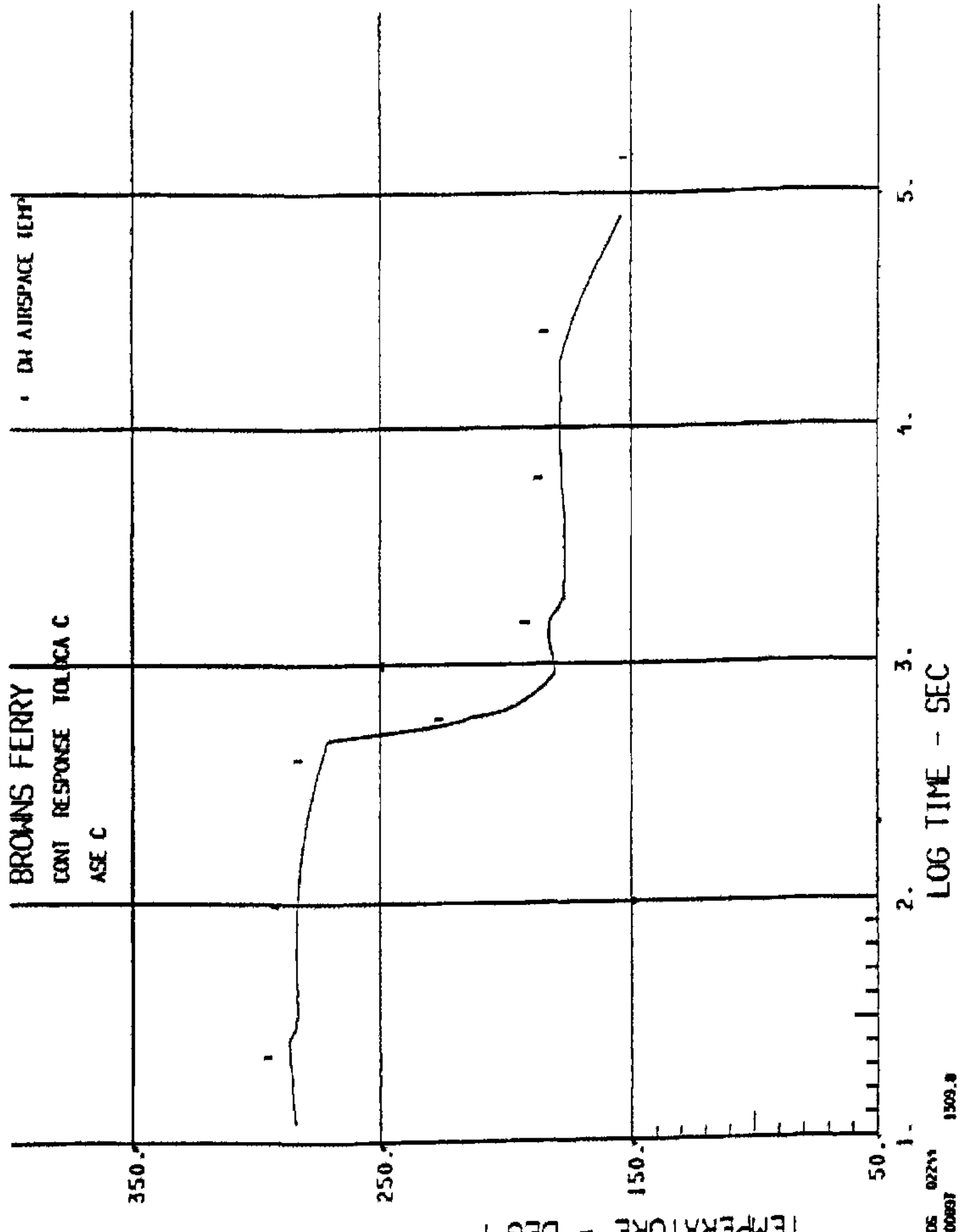


AMENDMENT 17

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

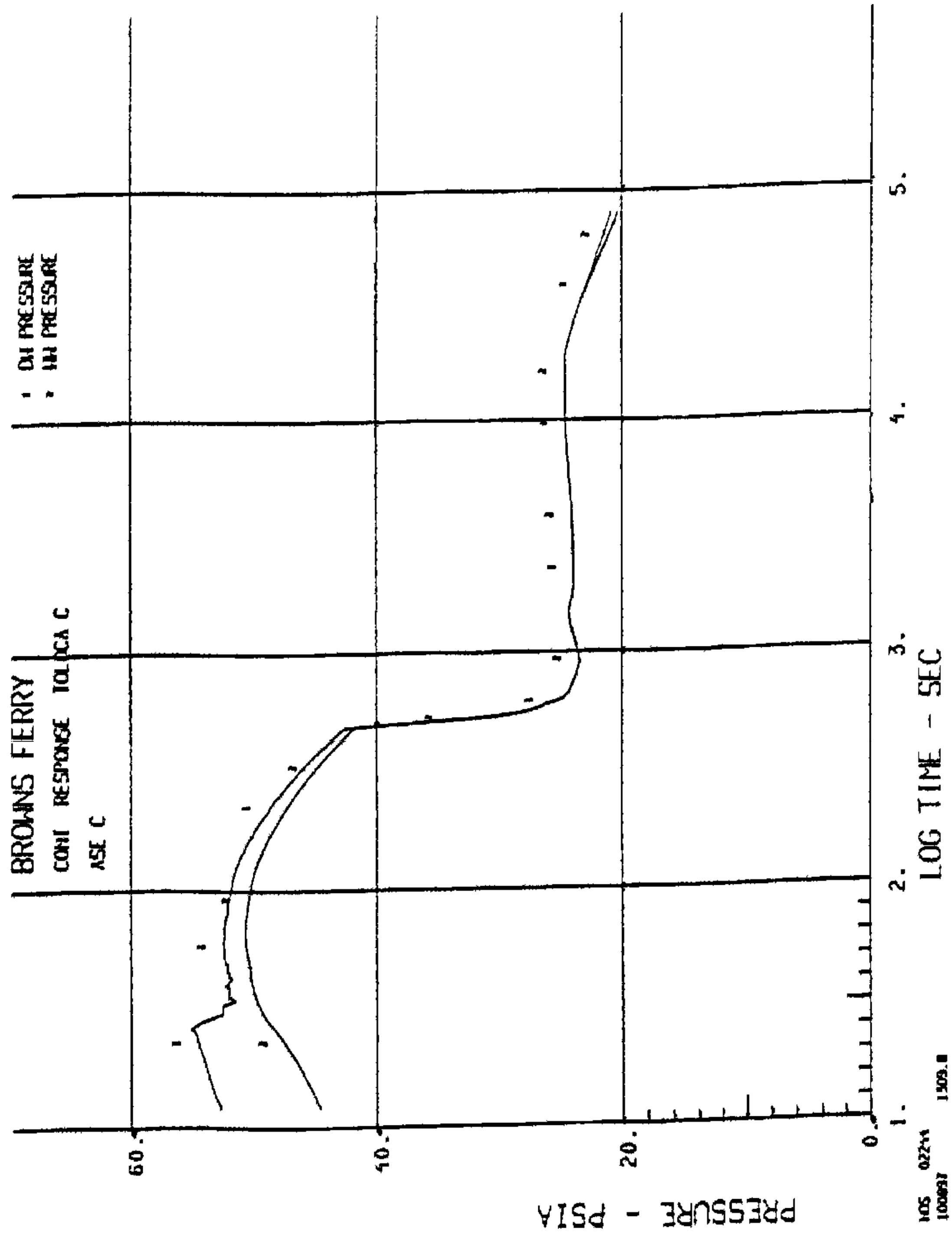


AMENDMENT 17

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

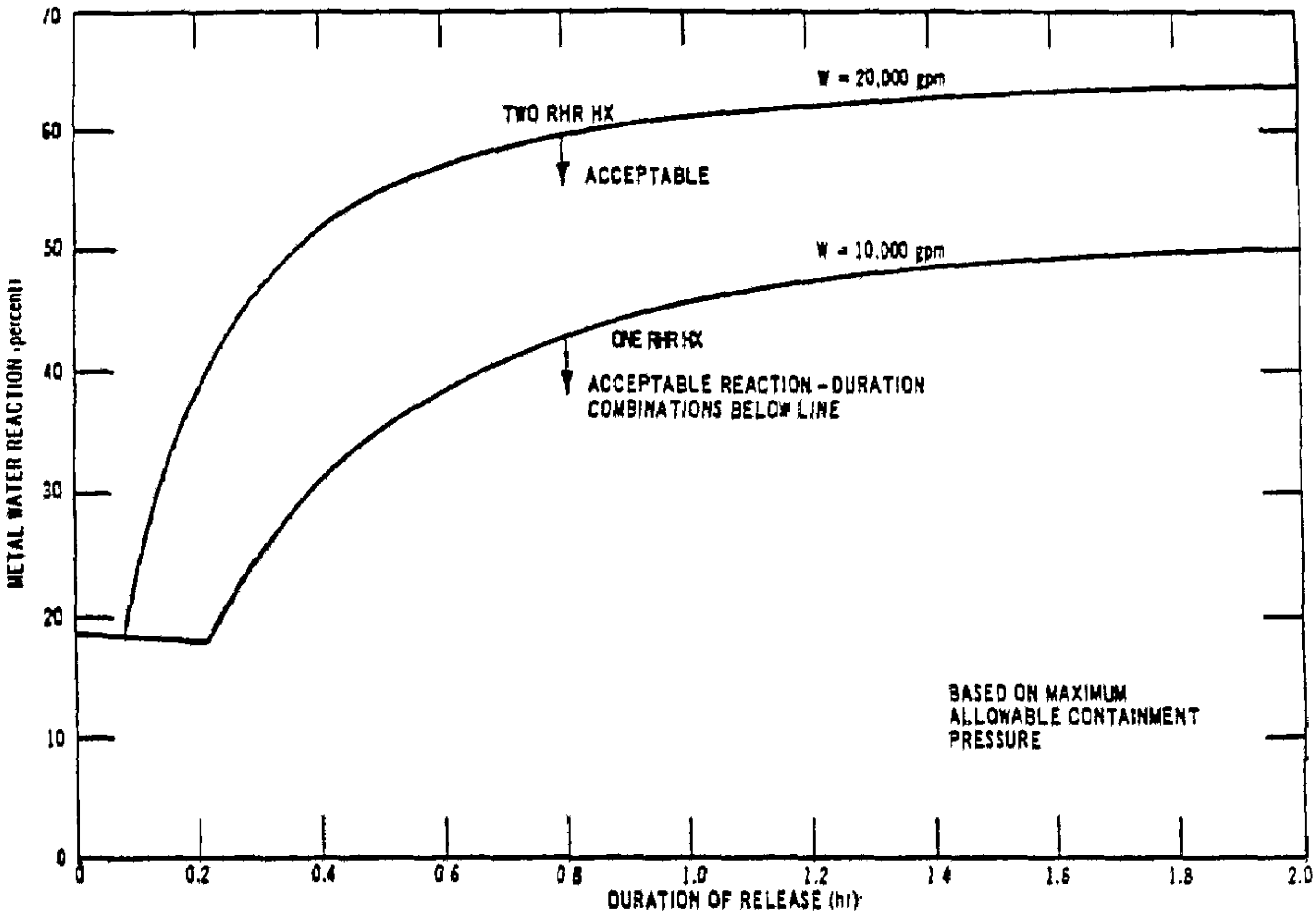


AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

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

FIGURE 14.6-5

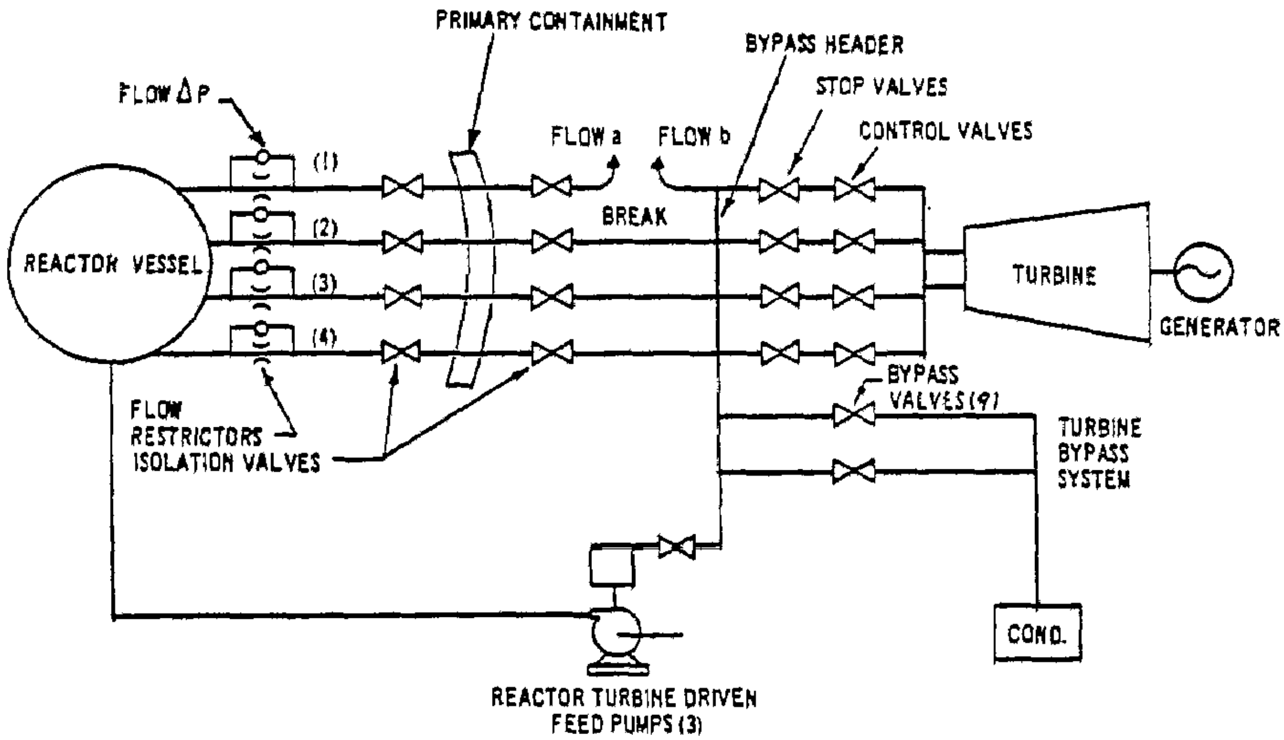


AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Loss-of-Coolant Accident,
Primary Containment Capability
for Metal-Water Reaction

FIGURE 14.6-6

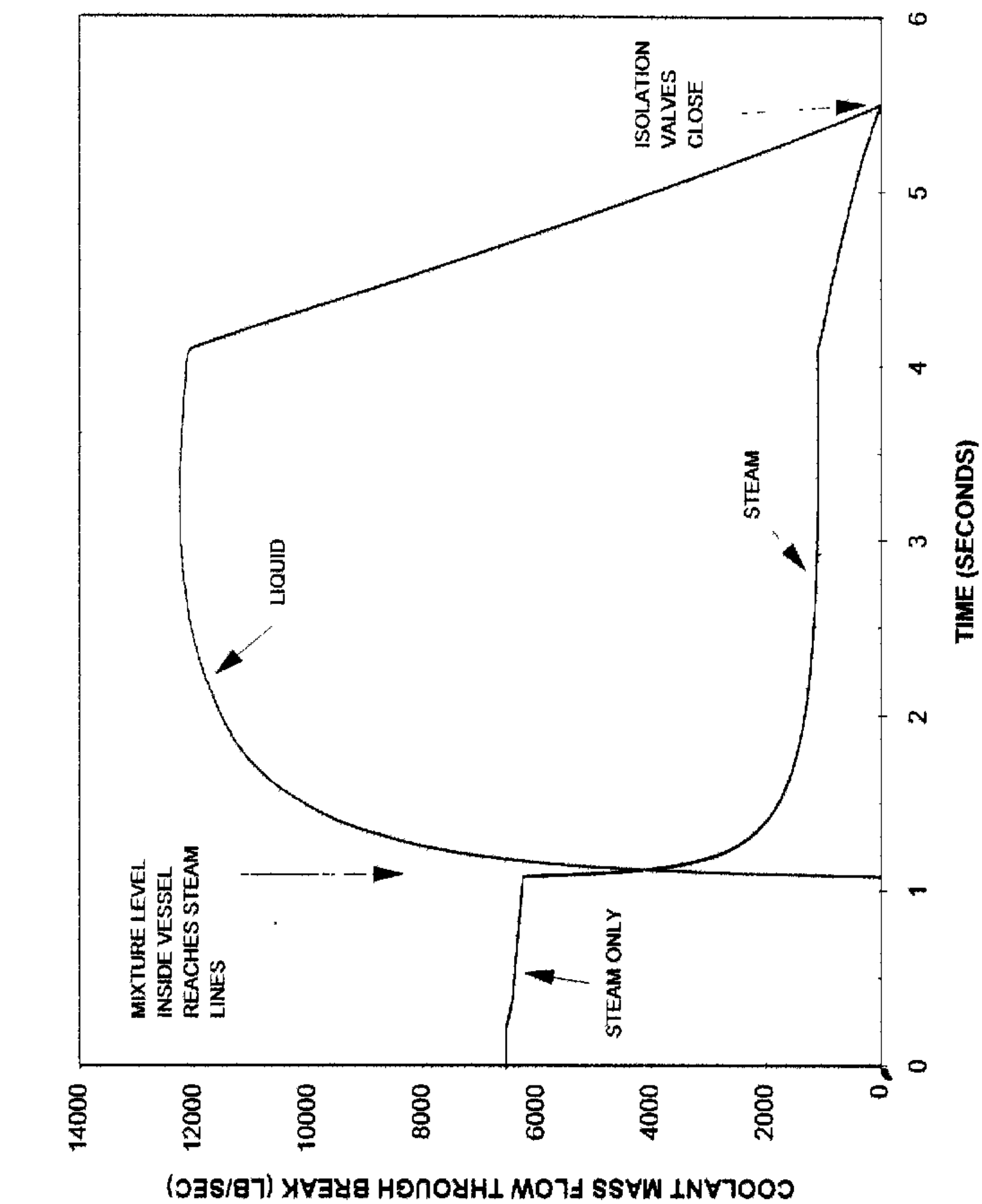


AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Main Steamline Break Accident
Break Location
FIGURE 14.6-15

FIGURE 14.6-7

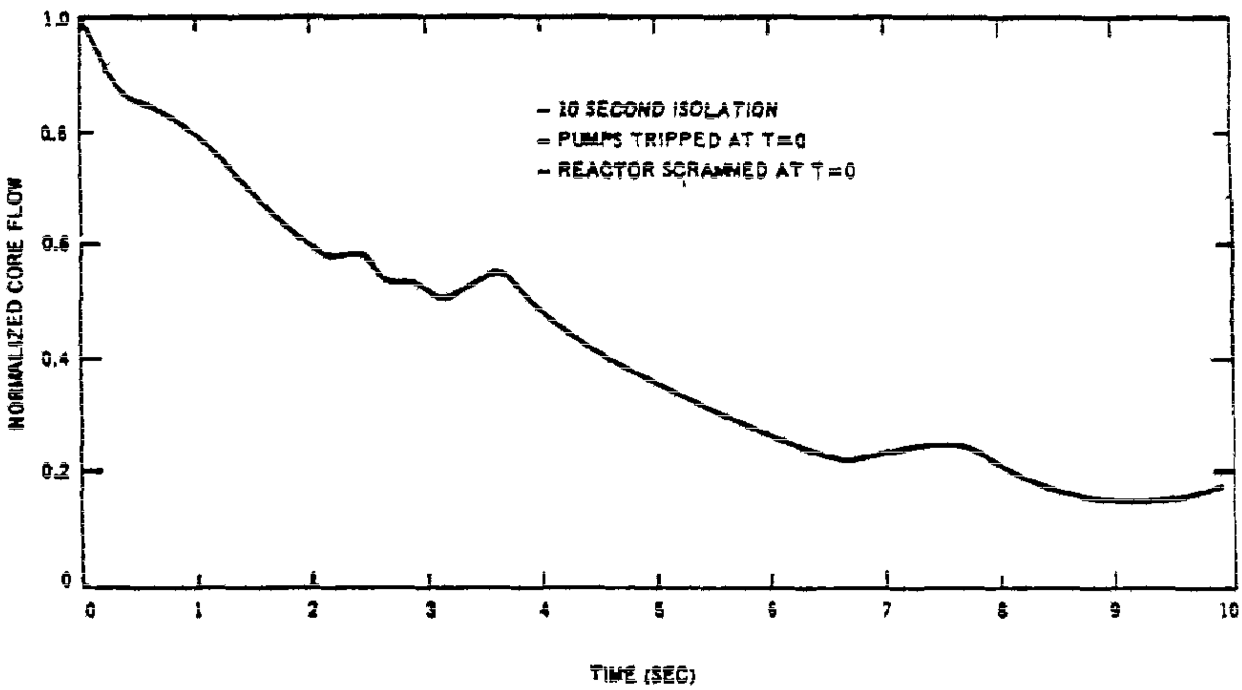


AMENDMENT 17

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

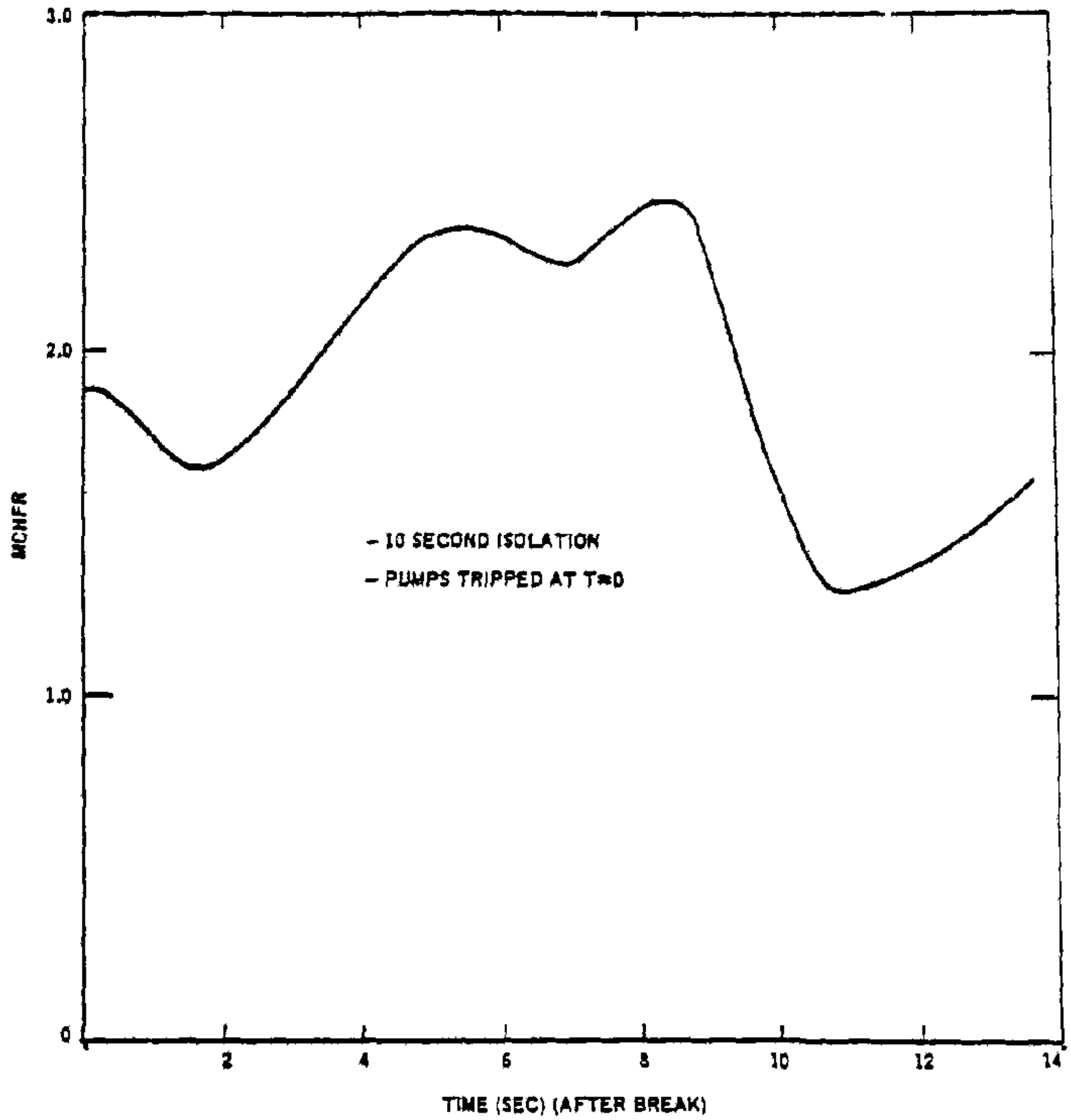


AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Main Steamline Break Accident
Normalized Core Inlet Flow

FIGURE 14.6-9

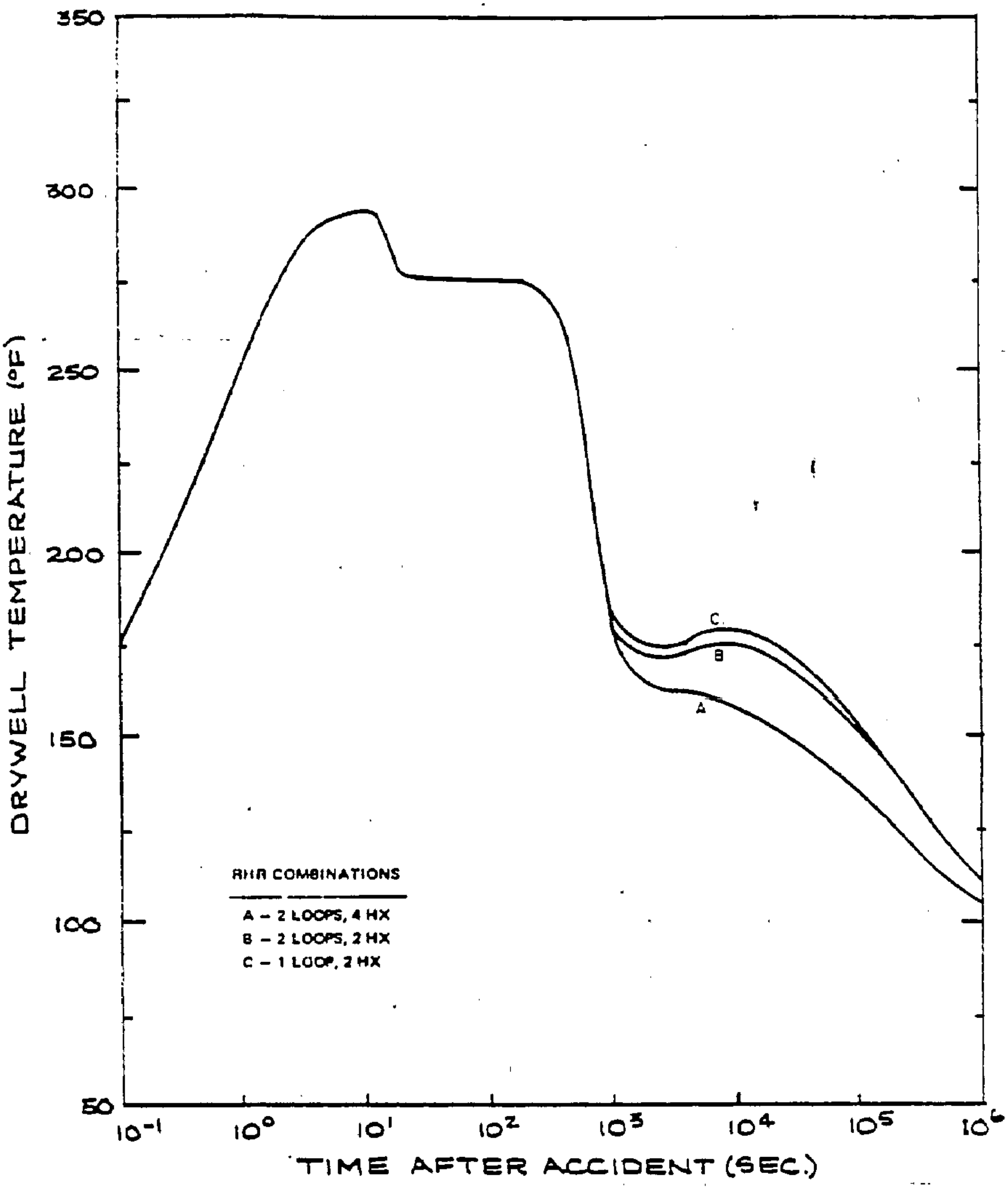


AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Main Steamline Break Accident
Minimum Critical Heat Flux Ratio

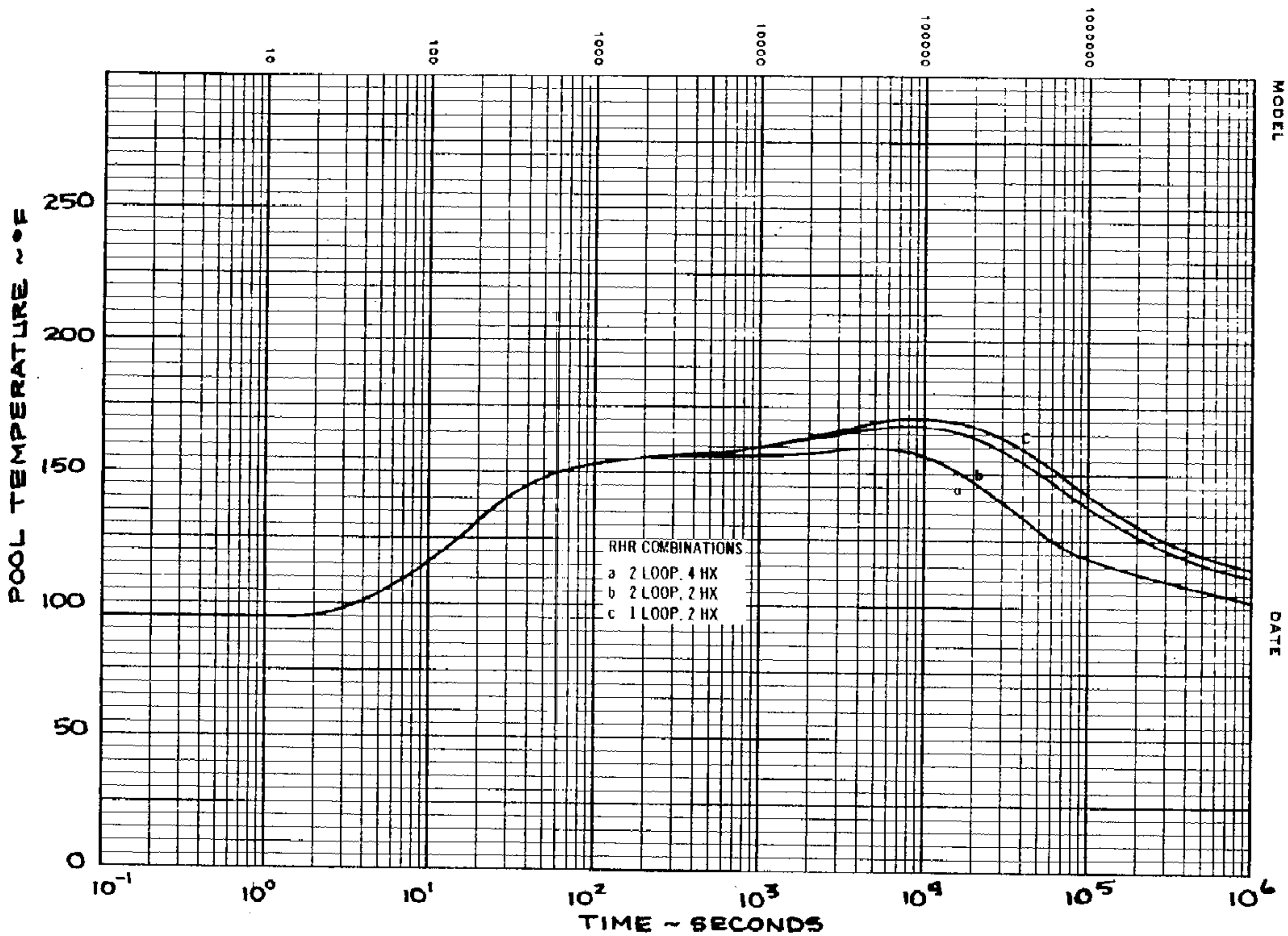
FIGURE 14.6-10



AMENDMENT 16

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Loss-of-Coolant Accident
Drywell Temperature Response
FIGURE 14.6-11



AMENDMENT 16

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

Loss-of-Coolant Accident,
Suppression Pool Temperature Response
FIGURE 14.6-12

BFN-16

Figure 14.6-13

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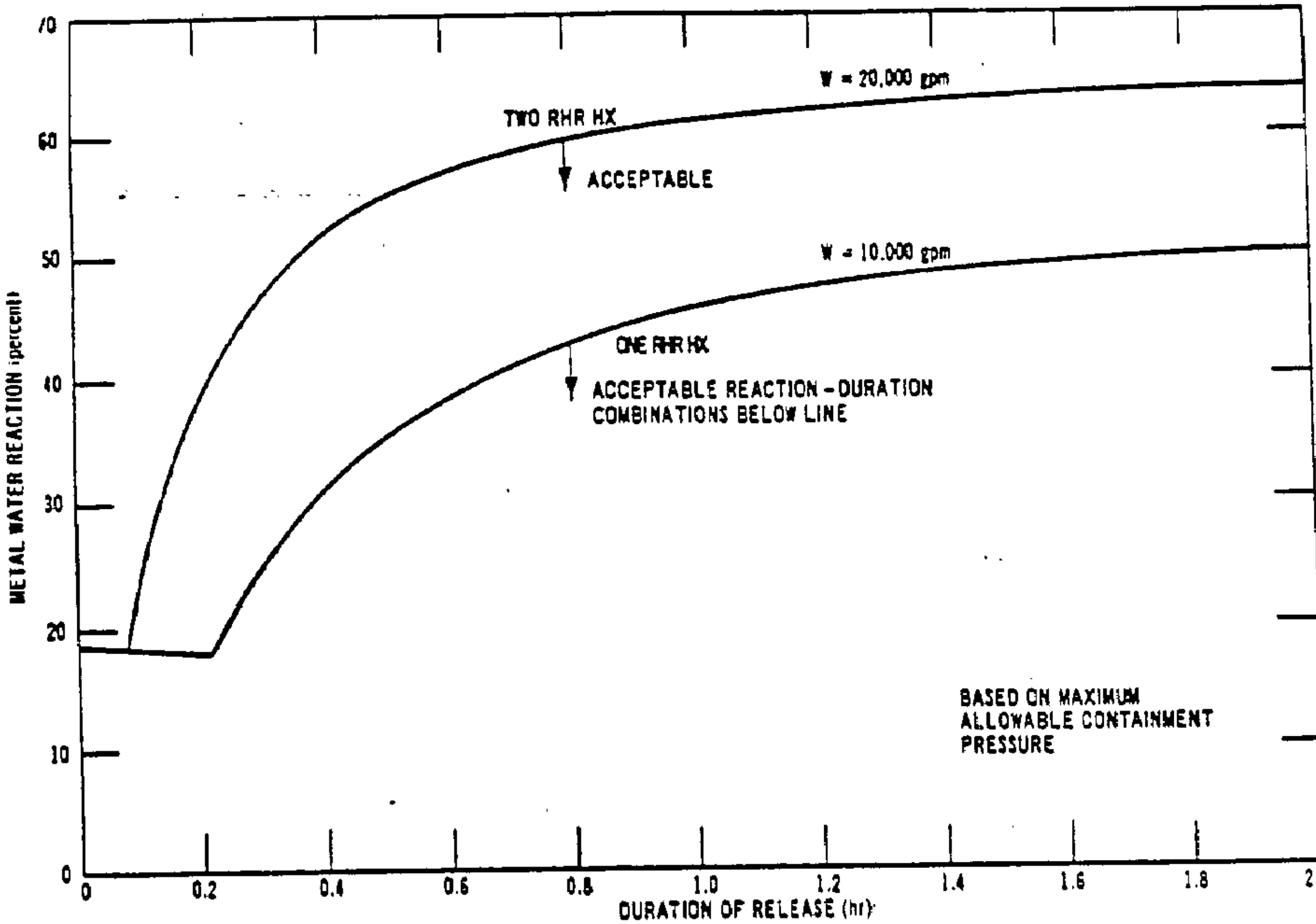
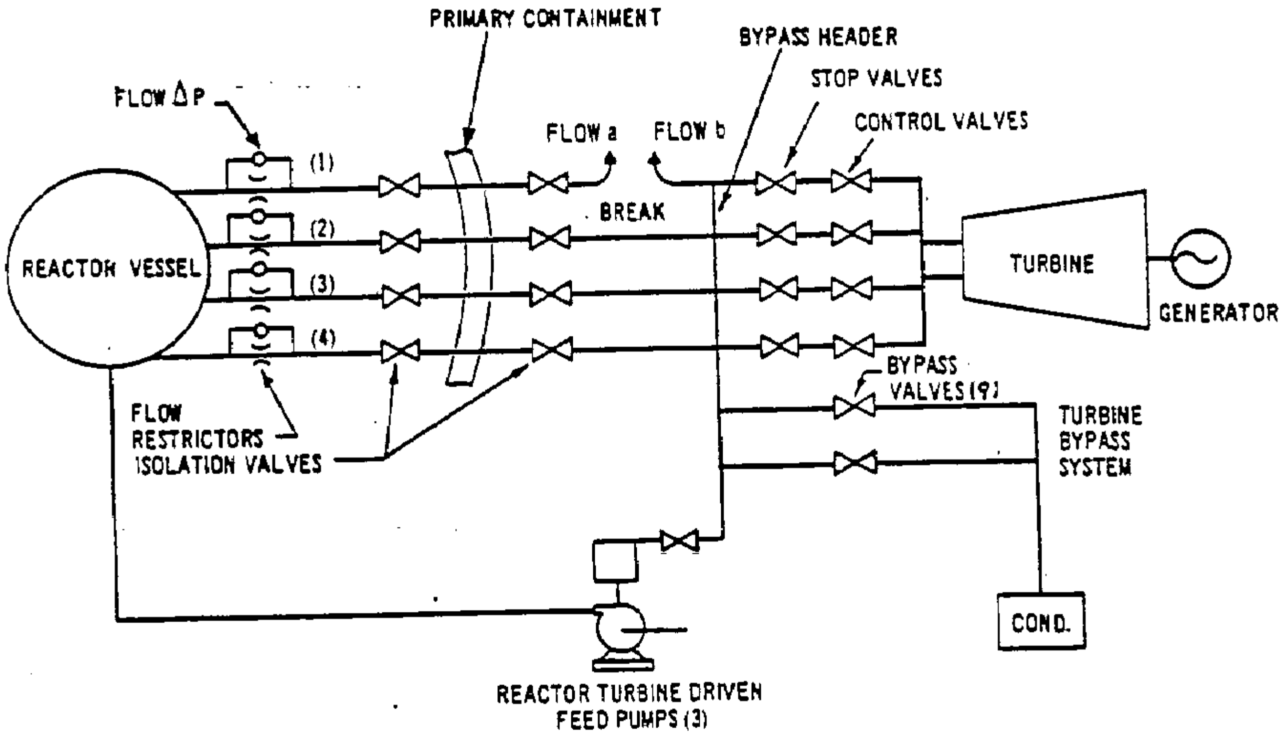


AMENDMENT 16

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

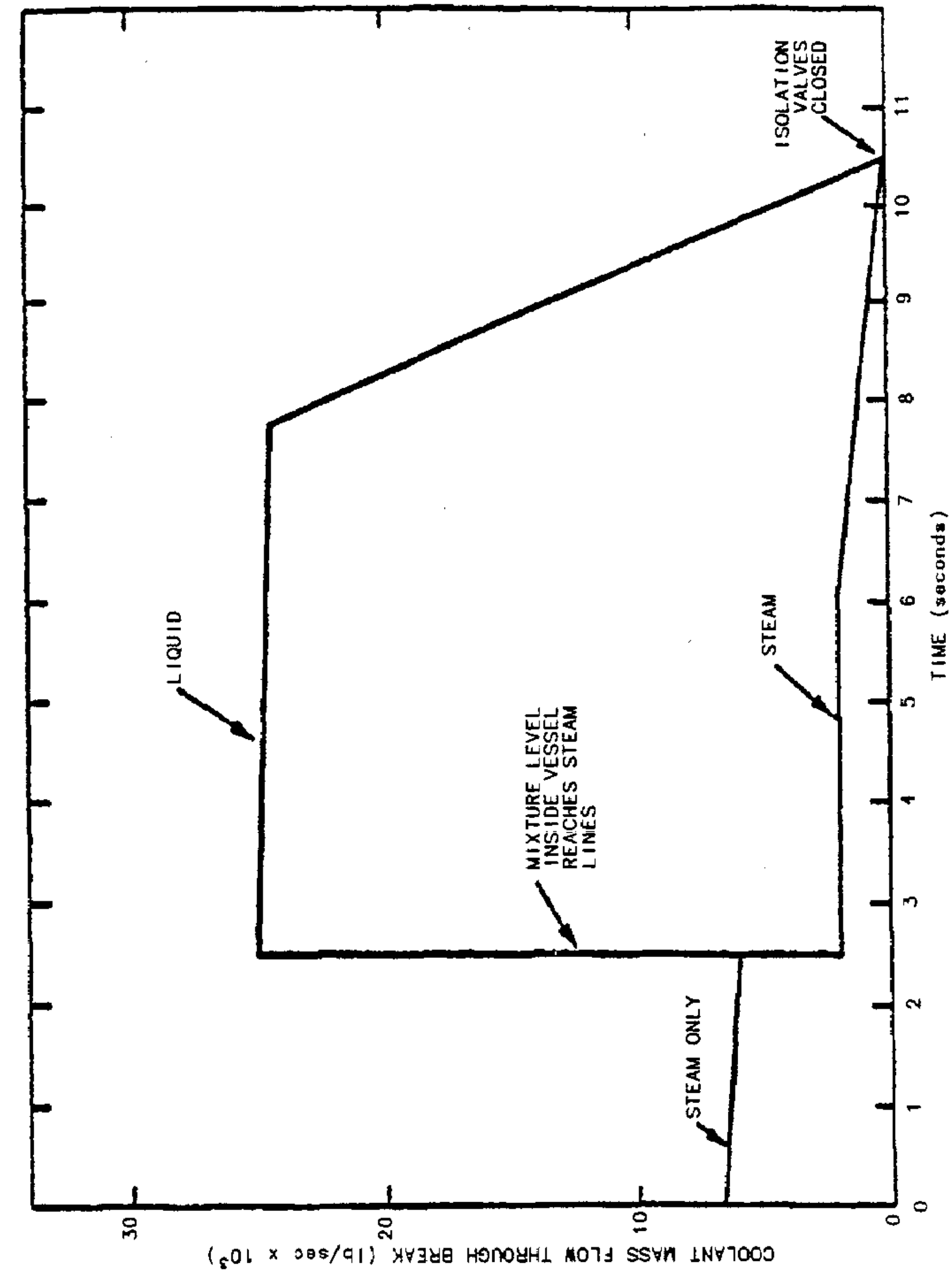
Loss-of-Coolant Accident,
Primary Containment Capability
for Metal-Water Reaction
FIGURE 14.6-14



AMENDMENT 16

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Main Steamline Break Accident
Break Location
FIGURE 14.6-15

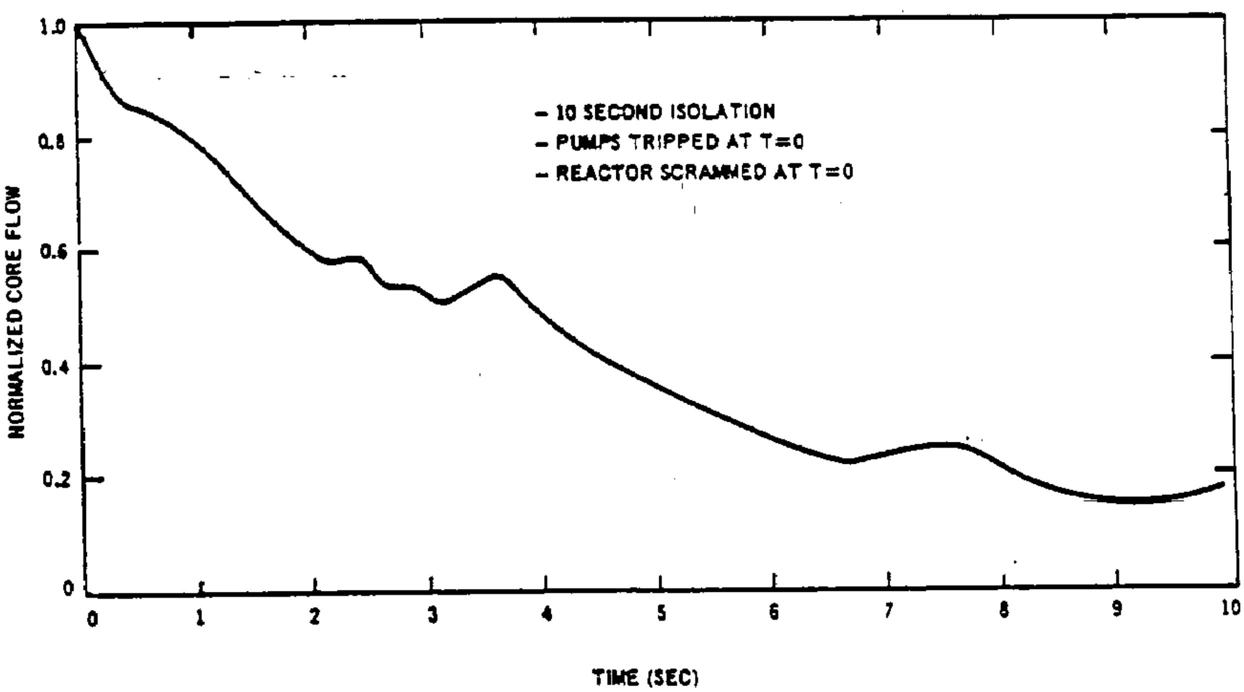


AMENDMENT 16

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

MAIN STEAMLINE BREAK ACCIDENT
MASS OF COOLANT LOST THROUGH BREAK

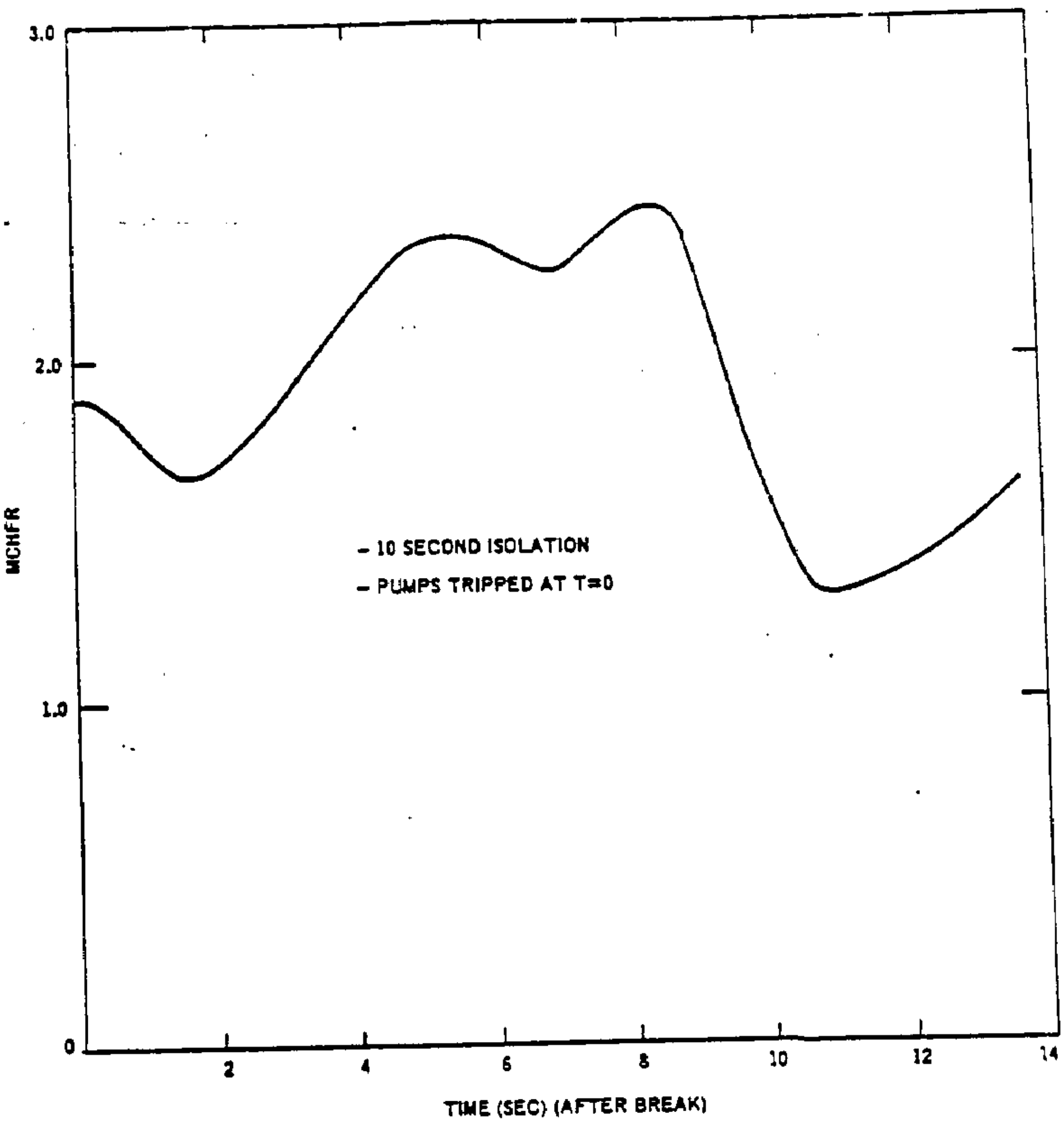
FIGURE 14.6-16



AMENDMENT 16

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Main Steamline Break Accident
Normalized Core Inlet Flow
FIGURE 14.6-17



AMENDMENT 16

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Main Steamline Break Accident
Minimum Critical Heat Flux Ratio
FIGURE 14.6-18

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	6.76E+00	2.31E-02	6.24E+02
0.5	2.22E+04	8.18E+00	2.56E-02	7.56E+02
0.8	1.85E+04	1.20E+01	1.44E-02	1.11E+03
1.1	1.55E+04	1.51E+01	8.16E-03	1.41E+03
1.4	1.29E+04	1.76+01	4.63E-03	1.66E+03
1.7	1.08E+04	1.97E+01	2.63E-03	1.87E+03
2	9.05E+03	2.14E+01	1.51E-03	2.05E+03
2.3	7.56E+03	2.27E+01	8.55E-04	2.19E+03
2.6	6.32E+03	2.38E+01	4.87E-04	2.31E+03
2.9	5.29E+03	2.46E+01	2.79E-04	2.41E+03
3.2	4.42E+03	2.53E+01	1.61E-04	2.50E+03
3.5	3.69E+03	2.57E+01	9.44E-05	2.57E+03
3.8	3.09E+03	2.61E+01	5.62E-05	2.63E+03
4.1	2.58E+03	2.63E+01	3.43E-05	2.68E+03
4.4	2.16E+03	2.65E+01	2.15E-05	2.72E+03
4.7	1.80E+03	2.66E+01	1.39E-05	2.76E+03
5	1.51E+03	2.66E+01	9.39E-06	2.79E+03
5.3	1.26E+03	2.65E+01	6.58E-06	2.81E+03
5.6	1.05E+03	2.64E+01	4.79E-06	2.83E+03
5.9	8.81E+02	2.63E+01	3.60E-06	2.85E+03
6.2	7.37E+02	2.62E+01	2.79E-06	2.86E+03
6.5	6.16E+02	2.60E+01	2.20E-06	2.87E+03
6.8	5.15E+02	2.58E+01	1.77E-06	2.88E+03
7.1	4.30E+02	2.56E+01	1.44E-06	2.89E+03
7.4	3.60E+02	2.54E+01	1.18E-06	2.90E+03
7.7	3.01E+02	2.51E+01	9.75E-07	2.91E+03
8	2.51E+02	2.49E+01	8.09E-07	2.91E+03
8.3	2.10E+02	2.47E+01	5.77E-07	2.91E+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	2.45E+01	4.27E-07	2.92E+03
8.9	1.47E+02	2.42E+01	3.26E-07	2.92E+03
9.2	1.23E+02	2.40E+01	2.55E-07	2.92E+03
9.5	1.03E+02	2.37E+01	2.03E-07	2.92E+03
9.8	8.58E+01	2.34E+01	1.64E-07	2.93E+03
10.1	7.17E+01	2.32E+01	1.34E-07	2.93E+03
10.4	6.00E+01	2.29E+01	1.10E-07	2.93E+03
24	1.78E-02	1.37E+01	0	2.94E+03
96	0	8.66E-01	0	2.94E+03
720	0	0	0	2.94E+03

TABLE 14.6-2

(Deleted by Amendment 19)

TABLE 14.6-3

SUMMARY OF POWER UPRATE INPUT PARAMETERS USED FOR ALL CONTAINMENT ANALYSES

Parameter	Unit	Analysis Value for Power Uprate
Core Thermal Power 102% of uprated power (3458 MWt)	MWt	3527
Initial Reactor Core Flow (100% rated)	Mlbm/hr	102.5
Vessel dome pressure At 102% of uprated power (3458 MWt)	psia	1053
Initial drywell pressure	psia	17.0/15.1(1)
Initial drywell temperature (Maximum value used to maximize the drywell temp. response)	°F	150
Initial drywell relative humidity (Minimum)	%	20
Initial wetwell pressure	psia	15.9/15.1(1)
Initial wetwell airspace temperature (Maximum)	°F	95
Initial wetwell airspace relative humidity (Maximum)	%	100
Initial pressure suppression pool temperature (Maximum value used to maximize the suppression pool temp. response)	°F	95
Downcomer submergence at high water level	ft.	3.83
Initial pressure suppression pool volume:		
Maximum (at high water level corresponding to 3'-10" downcomer submergence with a drywell to torus differential pressure of 1.1 psid)	ft^3	131,400
Minimum (at low water level corresponding to 2'-11" downcomer submergence with zero drywell to torus differential pressure)	ft^3	121,500
Total pressure suppression chamber volume (including pool)	ft^3	250,800
Drywell free volume (including vent system)	ft^3	171,000/159,000(1)
Torus-to-drywell vacuum breaker full open Δp	psid	0.5
Number of downcomers		96
Downcomer I.D.	ft.	1.956
Vent flow loss coefficient	-	5.32

(1): The value that was most limiting for a specific analysis was used.

Table 14.6-4

SUMMARY OF POWER UPRATE INPUT PARAMETERS USED FOR DBA-LOCA SHORT TERM CONTAINMENT RESPONSE

Parameter	Units	Case 1(1)	Case 2	Case 3(2)	Case 4
Initial Reactor Thermal Power	MWt	3527	3527	3527	2179
Initial Reactor Thermal Power	% of uprated	102	102	102	63
Initial Reactor Core Flow	Mlb/hr	102.5	102.5	83.0	39.0
Initial Reactor Core Flow	% of uprated	100	100	81	38
Feedwater Temperature	°F	384	328	328	328
Initial suppression pool volume	ft^3	131,400	131,400	131,400	131,400
Drywell Free Volume (including vent system)	ft^3	159000	159000	171000	159000
Initial Drywell Pressure	psia	15.1	17.0	17.0	17.0
Drywell-Wetwell Pressure difference	psid	0.0	1.1	1.1	1.1

(1): This is the most limiting case for hydrodynamic loads.
(2): This is the most limiting case for DBA-LOCA drywell pressure and temperature response.

Table 14.6-5

SUMMARY OF POWER UPRATE INPUT PARAMETERS USED FOR DBA-LOCA LONG-TERM CONTAINMENT RESPONSE

Parameter	Unit	Analysis Value for Power Uprate
Core Thermal Power 102% of uprated power (3458 MWt)	MWt	3527
Core Flow (100% of rated)	Mlbm/hr	102.5
Initial suppression pool volume	ft^3	121,500
Decay Heat Model	-	ANS/ANSI 5.1 + 2σ uncertainty
RHRS Heat Exchanger Heat Removal Rate (BTU per hr) calculated at the following containment spray mode conditions:		
- Shell side (RHR) flow (Minimum)	gpm	6500
- Shell side inlet temperature (Maximum)	°F	177
- Tube side (RHRSW) flow (Minimum)	gpm	4000
- Tube side inlet temperature	°F	92[(1)]
- Heat exchanger k-factor	Btu/sec-°F	223

(1) Operation with inlet temperature above 92°F is governed by Technical Specification limits for the Ultimate Heat Sink.

Table 14.6-6

DBA-LOCA SHORT TERM PRESSURE AND TEMPERATURE RESPONSE

Parameter	Units	Case 1[1]	Case 2	Case 3[2]	Case 4
Peak Drywell Pressure	psig	48.6	47.1	50.6	44.2
Peak Drywell Gas Temperature	°F	295	294	297	290

(1): This is the most limiting case for hydrodynamic loads.
(2): This is the most limiting case for DBA-LOCA drywell pressure and temperature response.

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 Only)	U1 Intake	Unit 3 Intake		
0-2 hrs	4.60E-4	**		
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

Time Period	Control Room U1 Intake (sec/m^3)	Control Room Unit 3 Intake (sec/m^3)	Site Boundary (sec/m^3)	LPZ Boundary (sec/m^3)
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)

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

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:

$$V = M_f v_f + M_g v_g = \text{constant}, \tag{14.1}$$

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

$$M = M_f + M_g \tag{14.2}$$

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
$\dot{q}$ = 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:

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

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

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:

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

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

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).

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

where:

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

The mass flow rate is:

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

14.8.2.3 Numerical Solution

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:

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

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,

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

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

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

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

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

$$P_D = P_S$$

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

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

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

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

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

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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:

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

where:

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

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

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}	$3.3 \times 10\text{-}^{-1}$	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.

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

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:

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

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 K^{th} 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

$$Q\ dep. = \chi_i B_r \qquad (14.42)$$

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

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

Where

D_i = Total inhalation dose (rad)

λ_i = Effective decay constant of the i^{th} 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:

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

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

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.

FUEL ROD DETAIL

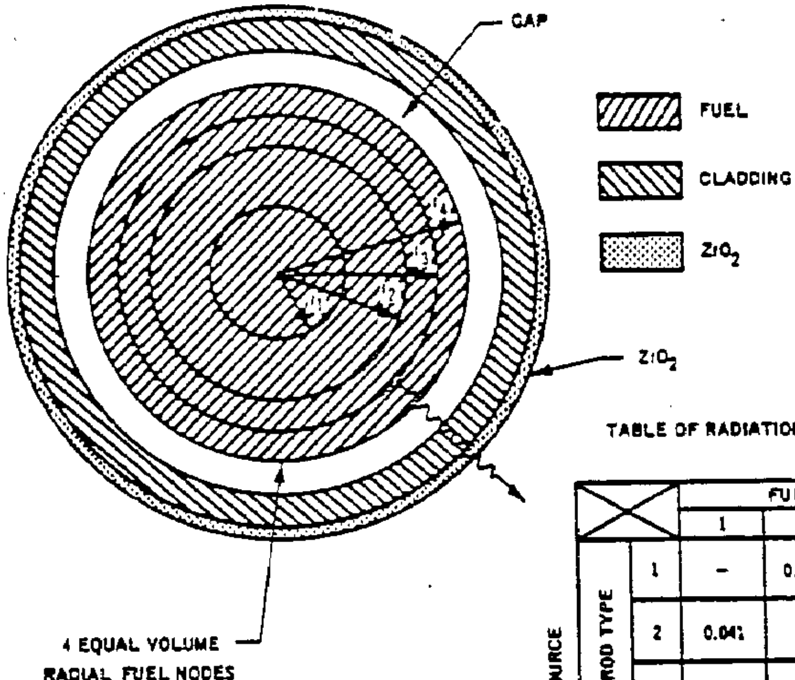
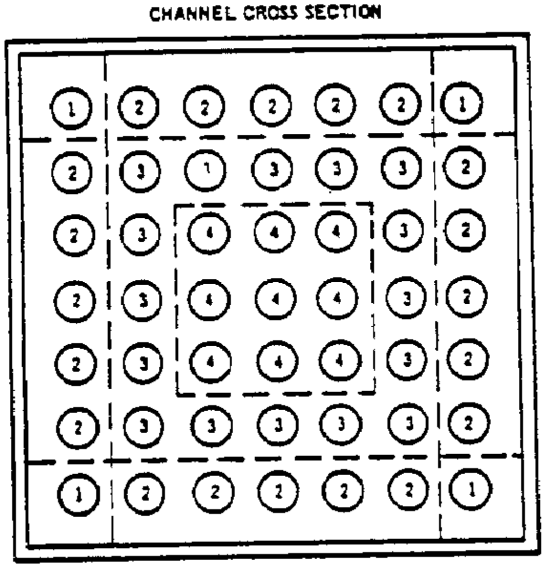


TABLE OF RADIATION HEAT TRANSFER COEFFICIENTS

RECEPTOR

SOURCE		FUEL ROD TYPE 1	FUEL ROD TYPE 2	FUEL ROD TYPE 3	FUEL ROD TYPE 4	CHANNEL SIDE WALL
FUEL ROD TYPE	1	–	0.205	0.0849	0.007	0.2914
FUEL ROD TYPE	2	0.041	–	0.177	0.0287	0.1744
FUEL ROD TYPE	3	0.0212	0.221	–	0.150	0.0235
FUEL ROD TYPE	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

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*	$\sigma \varnothing$ $\underline{u}$		
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 X 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 X 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 X 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

($Curie\text{-}sec/m^3$/curie released)

			Meteorological Conditions				
Distance (meters)	Activity of Interest	VS-1	MS-1	N-1	N-5	U-1	U-5
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

($Curie\text{-}sec/m^3$/curie released)

			Meteorological Conditions				
Distance (meters)	Activity of Interest	VS-1	MS-1	N-1	N-5	U-1	U-5
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}

14.9 DOSE SENSITIVITY EVALUATION USING ASSUMPTIONS OF THE AEC/DRL (INCORPORATED WITH TID 14844)

The dose sensitivity analysis utilizes the original evaluations of the radiological consequences of the four design basis accidents. The original sensitivity analysis which provides the effects of variation of design parameters on accident doses remains useful and is retained as background material.

14.9.1 Loss-of-Coolant Accident (183 meter release height)

1. The reactor has operated for an extended period at 3440 MWt.

2. 100 percent of the noble gases in the reactor and 25 percent of the iodine instantaneously become available for leakage from the primary containment as an aerosol based on TID 14844.

3. The primary containment volume leaks at a rate of 0.635 percent per day for 30 days.

4. The escaping aerosol immediately flows through the standby gas treatment system and the stack without mixing in the secondary containment building.

5. 90 percent of the iodine entering the standby gas treatment is retained by charcoal filters.

6. Meteorology - For the exclusion area calculations, the concentrations are those at the plume centerline with Pasquill C conditions. For the low population zone calculations, the concentrations are those for Pasquill C, 1 m/sec wind speed for the first day. For the remaining 29 days the conditions are 50 percent Pasquill C, 3 m/sec wind speed and 50 percent Pasquill F, 2 m/sec wind speed. During the first eight hours, the concentrations are at the plume centerline. During the 8-24 hour period, the plume stays within a 22.5° sector. For the 1-4 day period, the plume stays within a 22.5° sector 1/2 of the three days. The plume stays within the 22.5° sector 1/3 of the remaining 26 days.

7. There is a ground reflection factor of 2 for the plume and there is no ground deposition or rain wash out of the plume.

8. The breathing rate is 347 cc/sec for the first 8 hours and 232 cc/sec thereafter.

14.9.2 Refueling Accident (183 meter release height)

1. Assumptions "1", "4", "5", "7", and "8" of the loss-of-coolant accident.

2. Each damaged fuel rod contains 50 percent more activity than the average fuel rod in the core.

3. 20 percent of the noble gases and 10 percent of the iodine contained within the damaged rods are released within two hours.

4. 90 percent of the iodine released from the rods is retained by the refueling pool water.

5. One bundle is assumed to be damaged (49 rods). Appendix N and NEDE-24011-P-A contain or reference current fuel design information.

6. Meteorology - For the duration of the accident the concentrations are those at the plume centerline during Pasquill C, 1 m/sec wind speed.

14.9.3 Steam Line Break Accident (ground level release)

1. Assumptions "1", "7", and "8", for the loss-of-coolant accident.

2. The concentration of radionuclides in the reactor water are those associated with the maximum stack gas release limit which may be proposed as an operating limit.

3. The total mass of steam and water released from the steamline contain concentrations of radionuclides identical with those in the reactor water.

4. All of the radionuclides contained in the steam and water mass released from the steamline are released to the atmosphere at ground level.

5. It is assumed that there is no thermal rise of the steam cloud.

6. Meteorology - For the duration of the accident, the concentrations are those for Pasquill F, 1 m/sec wind speed. Building wake effects are accounted for with the plume shape factors equal to 0.5 and the building cross- sectional area equal to 2660 square meters.

14.9.4 Control Rod Drop Accident (ground level release)

1. Assumptions "1", "6", "7", and "8" for the loss-of-coolant accident.

2. The damaged fuel rods are from the highest burnup (activity) regions of core.

3. 100 percent of the noble gases and 50 percent of the iodine are released from the damaged rods.

4. 90 percent of the iodine released from the damaged fuel rods is retained by the reactor water.

5. The radionuclides released from the reactor water travel to the condenser where 50 percent of the iodine plateout.

6. The leak rate from the condenser is 0.5 percent per day. (Only if mechanical vacuum pump is not operating.)

7. The accident duration is 24 hours.

8. Meteorology - For the duration of the accident, the concentrations are those for Pasquill F, 1 m/sec wind speed. For the exclusion area calculations, the concentrations are those at the plume centerline. For the low population zone calculations, the concentrations are those at the plume centerline for the first 8 hours. For the remaining 16 hours, the plume stays within a 22.5° sector.

14.9.5 Radiological Consequences

Radiological consequences of the design basis accidents have been evaluated in Chapter 14 using the General Electric method of analysis as described in a topical report APED 5756 "Analytical Methods for Evaluating the Radiological Aspects of the General Electric Boiling Water Reactor" (March 1969). Table 14.9-1 contains the doses calculated using the assumptions listed in paragraphs 14.9.1 through 14.9.4 above.

To demonstrate the effects of the various factors involved in making radiological dose calculations, Table 14.9-2 lists many assumptions and their effect of the resulting dose. Note that many of these factors are nonlinear in nature and, therefore, cannot be interpolated or extrapolated without performing sophisticated calculations.

14.9.6 Discussion of Assumptions

The loss-of-coolant accident has generally been interpreted as a complete core melt (10 CFR 100.11(a) Note 1) without consideration of the geometry aspects of molten fuel and its resultant consequences. Only the fission product release has been considered. Such a situation would only be evaluated in light of little or no core cooling protection. It states in 10 CFR 100.10 that ". . . the Commission will take the following factors into consideration in determining the acceptability of a site for a power or testing reactor:

a. Characteristics of reactor design and proposed operation including:

   (1) Intended use of the reactor including the proposed maximum power level and the nature and inventory of contained radioactive materials,

   (2) The extent to which generally accepted engineering standards are applied to the design of the reactor,

   (3) The extent to which the reactor incorporates unique or unusual features having a significant bearing on the probability or consequences of accidental release of radioactive materials, and

   (4) The safety features that are to be engineered into the facility and those barriers that must be breached as a result of an accident before a release of radioactive materials to the environment can occur.

The plant is designed to keep the thermal response of the core below a clad temperature of 2200°F. Because of this the use of TID 14844 assumptions of core melt fission product release do not apply for the plant.

The above referenced topical report (APED 5756) summarizes the technical basis for all assumptions and models used on current generation GE Boiling Water Reactors. The use of this topical report in evaluating the radiological aspects of the plant is consistent with good engineering and actual design.

TABLE 14.9-1

DESIGN BASIS ACCIDENT RADIOLOGICAL DOSES
(REM)

	WHOLE BODY		THYROID	
Accident	2 hour (1400 m)	30 day (3218 m)	2 hour (1400 m)	30 day (3218 m)
Loss of Coolant	2.2×10^{-1}	6.1×10^{-1}	2.9×10^{0}	3.6×10^{1}
Refueling	4.9×10^{-2}	1.9×10^{-2}	3.1×10^{1}	4.2×10^{-1}
Control Rod Drop	1.2×10^{-2}	4.3×10^{-2}	6.1×10^{0}	7.0×10^{0}
Steam Line Break	1.7×10^{-2}	8.0×10^{-3}	3.0×10^{1}	1.0×10^{1}

Table 14.9-2
(Sheet 1)

SENSITIVITY OF DOSES TO VARIATION OF ASSUMPTIONS
LOSS-OF-COOLANT ACCIDENT

Assumptions	Design "Base" Case	Assumed AEC Case	Factor Affecting Thyroid Dose	Factor Affecting Whole Body Dose
Fission products released to drywell	1.8 percent noble gases[1] 0.32 percent iodines from 25 percent of the fuel rods which are assumed to be perforated. 1 percent of total iodine in organic form. Negligible solids.	100 percent noble gases 50 percent iodines and 1 percent solids in total core inventory. 5 percent of total iodines in organic form.	625	220
Iodine retained in water	Based on partition factor of 100 between the volumes of air and water in pressure suppression chamber and drywell.	None	12.5[3]	1
Elemental iodine plateout in drywell	50 percent	50 percent	1	1
Leakage rate from primary containment	Function of drywell pressure; peaks close to 0.5 percent volume per day	0.635 percent volume per day, constant throughout accident	1.3 (2-hr)[2] ~1 (30-day)	1.3 (2-hr) ~(30-day)
Uniform mixing in Reactor Building	Yes	No	22 (2-hr) 1.2 (30-day)	28 (2-hr) 1.1 (30-day)
Iodine filter efficiency	99 percent (95 percent for solids)	90 percent	10	1
Effectiveness of stack	Yes	Yes	1	1

NOTE:

[1] 1 percent of iodines released in organic form, which is not reduced by fallout or drywell and reactor building. Elemental iodines are carried into pressure suppression pool during blowdown, and a fraction retained according to the assumed equilibrium partition factor of 100. Iodines become airborne in the pressure suppression chamber and drywell before leaking out to the secondary containment.

[2] 2-hr dose is evaluated at site boundary of 1400 meters; 30-day dose is evaluated at low-population zone of 3218 meters.

[3] Takes into account the organic iodine fraction.

Table 14.9-2

(Sheet 2)

SENSITIVITY OF DOSES TO VARIATION OF ASSUMPTIONS

REFUELING ACCIDENT

Assumptions	Design "Base" Case	Assumed AEC Case	Factor Affecting Thyroid Dose	Factor Affecting Whole Body Dose
Fission product release to reactor water[1]	1.8 percent noble gases, 0.32 percent iodines from 111 perforated fuel rods, solids negligible	20 percent noble gases, 10 percent iodines from 49 perforated fuel rods[2]	13.8	4.9
Iodines retained in water	Equilibrium partition[4] factor of 100 for iodines and water	90 percent[2]	0.4	1
Plateout of iodines in Reactor Building	None	50 percent	0.5	1
Uniform mixing in refueling chamber	Yes	No	14 (2-hr)[3] ~1.3 (30-day)	18 (2-hr) ~1.1 (30-day)
	Fission Products exponentially released from water to Reactor Building till exhausted	Fission products exponentially released from water in 2 hours		
Iodine filter efficiency	99 percent	90 percent (95 percent for solids)	10	1
Effectiveness of stack	Yes	Yes	1	1

NOTE:

[1]Accident occurs 24 hours after shutdown.

[2]Assumptions in Hatch (Docket No. 50.321) evaluation.

[3]2-hr dose is evaluated at site boundary of about 1400 meters. 30-day dose is evaluated at low-population zone of 3128 meters.

[4]Amount of retention depends on the ratio of air space to water space. In this case, the equivalent value of 75 percent is obtained.

Table 14.9-2

(Sheet 3)

SENSITIVITY OF DOSES TO VARIATION OF ASSUMPTIONS
CONTROL ROD DROP ACCIDENT

Assumptions	Design "Base" Case	Assumed AEC Case	Factor Affecting Thyroid Dose	Factor Affecting Whole Body Dose
Fission products released to water	1.8 percent noble gases, 0.32 percent iodines from 330 perforated fuel rods, solids negligible	100 percent noble gases, 50 percent iodines from 330 perforated fuel rods	156	55
Noble gases carry-over to condenser hotwell	Uniformly mixed with steam, carried over at 5.0 percent steam flow rate, isolation valve closure at 10.5 sec.	100 percent	1	10
Iodine carryover to condenser hotwell	Retention in water,[1] uniform mixing in steam dome, carryover at 5.0 percent steam flow, and isolation at 10.5 seconds	10 percent	2700	1.0
Iodine plateout in condenser hotwell	None	50 percent	0.5	1
Release mechanism	1800 cfm from vapor space of condenser and turbine, stack release	Leak rate of 0.5 percent per day from condenser to environs	5.5×10^{-4} (2-hrs) 1.13×10^{-2} (30-days)	5.5×10^{-4} (2-hrs) 1.13×10^{-2} (30-days)

NOTE:

[1]Amount of retention in condenser hotwell water depends on relative ratio of steam space to water space. The "base" case uses an equilibrium partition factor of 100 and a steam-water space ratio of about 12.

Table 14.9-2

(Sheet 4)

SENSITIVITY OF DOSES TO VARIATION OF ASSUMPTIONS
STEAM LINE BREAK ACCIDENT

Assumptions	Design "Base" Case	Assumed AEC Case	Factor Affecting Thyroid Dose	Factor Affecting Whole Body Dose
Steam and Water Mass Lost in blowdown (10.5 sec. closure)	185,000 lb (25,000 lb steam 160,000 lb water)	185,000 lb	1	1
Total fission gases released	146 curies iodines and 5.7 curies noble gases[1]	Proportional to operating limit, 10.5 times the base case value	10.5	10.5[2]
Concentration in water and steam	Equilibrium separation	Equilibrium separation	1	1
Steam cloud rise	No	No	1	1

NOTE:

[1]Based on fission product concentrations in coolant such that the offgas release rate at stack reaches the maximum expected value of 10,000 FCi/sec.

[2]In the steamline break accident, the noble gases contribution to the whole body dose is insignificant.

14.10 ANALYSES OF ABNORMAL OPERATIONAL TRANSIENTS PRE-UPRATED

This section contains general descriptions of abnormal operational transients analyzed for the initial operating cycle of Browns Ferry Units l, 2, and 3. The bounding transients are reanalyzed 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 newer fuel reload specific analysis have been performed will be noted; however, the original analysis descriptions will be retained in this chapter.

This section does not reflect the effects from power uprate. For power uprated conditions, the results of the re-analyses at 3458 MWt using the latest transients methodologies for the non-limiting events not included in the cycle-specific reload analyses are provided in Section 14.5.

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

14.10.1.1 Generator Trip (Turbine Control Valve (TCV) Fast Closure)

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

a. Turbine-generator acceleration protection devices trip to initiate turbine control valve fast (about 0.20 second) closure,

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 30 percent rated),

c. The turbine bypass valves are opened simultaneously with turbine control valve closure,

d. Reactor vessel pressure rises to the main steam relief valve setpoints, causing them to open for a short period. The steam passed by the main steam relief valves is discharged into the pressure suppression pool, and

e. The turbine bypass system controls nuclear system pressure after the main steam relief valves close.

Below about 25 percent of rated power, the bypass system will transfer steam around the turbine and thereby avoid reactor scram. Between about 25 percent to 30 percent power, high-pressure scram will result unless operator action can reduce power to within the bypass capacity.

14.10.1.1.1 Generator Trip (TCV Fast Closure) With Bypass Valve Failure

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 14.10.1.1 except the turbine bypass valves are assumed to remain closed.

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 Reload Licensing Report. A typical generator load rejection bypass is shown in Figure 14.10-20. Assuming the initial reactor power level is 105 percent nuclear boiler rated steam flow, the neutron flux peaks at 281 percent nuclear boiler rated, the average heat flux peaks at 111 percent of its initial value and MCPR remains greater than the safety limit MCPR. The peak pressure at the bottom of the vessel is approximately 1245 psig which is well below the established transient limit of 1375 psig. (Reference - "Basis For Installation of Recirculation Pump Trip System", NEDO-24119A, April 1978).

14.10.1.2 Loss of Condenser Vacuum

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 and closure of the turbine stop valves and bypass valves with a turbine trip scram. It is assumed that the plant is initially operating at design power (105 percent rated).

Figure 14.10-1 shows a typical transient with relief capacity equal to 61 percent of rated steam flow. Peak neutron flux reaches 198 percent of the rated power; however, the fuel surface heat flux does not exceed 107 percent of its initial value and peak fuel center temperature increases less than 150°F. No damage to the fuel results from the transient. The main steam relief valves open fully to limit the pressure rise, then sequentially reclose as the stored energy is dissipated. The peak nuclear system pressure at the bottom of the vessel is also well below the nuclear process barrier transient pressure limit of 1375 psig.

14.10.1.3 Turbine Trip

This case represents the events that would follow an assumed trip of the turbine stop valve producing the fastest possible steam flow shutoff and severe nuclear system pressure increase. It is assumed that the plant is initially operating at design power.

The sequence of events for a turbine trip is very similar to that for a generator load rejection. However, the valve closure is faster, occurring over a period of about 0.1 second. Position switches on the stop valves provide the means of sensing the trip and initiating immediate reactor scram, recirculation pump trip, and bypass valve opening. Figure 14.10-2 shows a typical transient expected from design power conditions with 61 percent relief capacity. The main steam relief valves open for a short time to help relieve the pressure transient, and then the bypass valves control the reactor pressure for post-trip conditions. The fuel thermal transient is mild. Peak pressure in the bottom of the vessel and at the steam lines is below the ASME Code limits for the nuclear process barrier. Turbine trips from lower initial power levels decrease in severity to the point where scram may even be avoided within the bypass capacity if auxiliary power is available from an external source.

14.10.1.4 Bypass Valves Failure Following Turbine Trip, High Power

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

14.10.1.5 Bypass Valves Failure Following Turbine Trip, Low Power

This abnormal operational transient is of interest because turbine stop valve closure and turbine control valve fast closure scrams are automatically bypassed when the reactor power level is low. Turbine first-stage pressure is used to initiate this bypass at 154 psig. The highest power level for which these scrams remain bypassed is about 30 percent of rated power. Figure 14.10-3 graphically shows the transient starting with the recirculation pumps at about 20 percent speed producing 40 percent core flow at 31 percent rated power. Reactor scram is initiated at about 3.0 seconds by high vessel pressure. No bypass flow is assumed; however, the main steam relief valves partially open to relieve the pressure transient. The peak pressure at the main steam relief valves is well below the ASME Code limits. Since pressure remains below 1375 psig at the bottom of the vessel, no damage occurs to the nuclear process system barrier. No fuel damage occurs since peak heat flux is significantly lower than rated conditions.

14.10.1.6 Main Steam Isolation Valve 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 reactor pressure is greater than 1,055 psig or 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 Operating States C or E) will produce no significant transient. Closures during plant heatup (Operating State D) will be less severe than the maximum power cases (maximum stored and decay heat) which follow.

14.10.1.6.1 Closure of All Main Steam Isolation Valves

Figure 14.10-4 shows typical changes in important nuclear system variables for the simultaneous isolation of all main steam lines while the reactor is operating at design power. Reactor scram is initiated by the isolation valve position switches before the valves have traveled more than 10 percent from the open position. A three-second nonlinear valve closure was simulated, which is the fastest closure attainable. Scram is initiated very early into the event; therefore, no fuel center temperature, or fuel surface heat flux peaks occur. A small neutron flux peak occurs near 2 seconds. The nuclear system main steam relief valves begin to open when pressure reaches the lowest setpoint at about 2.5 seconds after the start of the isolation. They close sequentially as the stored heat is dissipated and will continue to intermittently discharge the decay heat. The peak pressure in the main steam line near the spring setpoint main steam relief valves is well below their setpoint. Peak pressure at the bottom of the vessel is also below the pressure limits of the nuclear system process barrier.

14.10.1.6.2 Closure of One Main Steam Isolation Valve

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 about 80-90 percent of design conditions in order to avoid high flux or pressure scram. Figure 14.10-5 graphically shows typical changes of important nuclear system variables during the simulated three-second closure of one main steam isolation valve from design power conditions. The steam flow disturbance raises vessel pressure and reactor power causing a high neutron flux scram. Peak pressures remain below the setting of the lowest main steam relief valves and peak fuel parameters are well below the point at which damage might occur.

14.10.1.7 Pressure Regulator Failure

Pressure regulator malfunctions that result in the turbine steam flow shutoff and a nuclear system pressure increase are similar to but of milder consequence than the generator trip described previously.

14.10.2 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 (RHRS) malfunction-decreasing temperature, and
c. Inadvertent pump start.

14.10.2.1 Loss of a Feedwater Heater

A feedwater heater can be lost if the steam extraction line to the heater is shut, the heat supply to the heater is removed, producing a gradual cooling of the feedwater. The reactor vessel receives cooler feedwater which produces an increase in core inlet subcooling. Due to the negative void reactivity coefficient, an increase in core power results. An assumed loss of feedwater heating event is analyzed for each reload cycle using the methodology in NEDE-24011-P-A and the results are contained in the Reload Licensing Report.

Figure 14.10-6 shows a typical response of the plant to the loss of 100°F of the feedwater heating capability of the plant. This represents the maximum expected single heater (or group of heaters) which can be tripped or bypassed by a single event. The reactor is assumed to be at design power conditions on automatic recirculation flow control when the heater is lost. For this analyzed case, the feedwater flow delay time of approximately 25 seconds between the heaters and the feedwater sparger is neglected. The plant would continue at steady-state conditions during this delay period. The recirculation flow control system responds to the power increase by reducing core flow so that steam flow from the reactor vessel to the turbine remains essentially constant throughout the transient. Neutron flux increases above the initial value to produce turbine design steam flow with the higher inlet subcooling. Normally the reactor would be on manual flow control, and this neutron flux increase would have reached within 1 percent of the scram setting. In the case with automatic control, reactor power settles out slightly below the scram setting, but with core flow reduced to about 90 percent. 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 below the limiting values at which fuel damage could occur.

This transient is less severe from lower power levels for two main reasons: (1) lower initial power levels will have initial fuel parameter values less limiting than the values assumed here, and (2) the magnitude of the power rise decreases with the initial power condition. Therefore, transients from other reactor operating states or lower power levels within operating state F will be less severe.

14.10.2.2 Shutdown Cooling (RHRS)

Malfunction-Decreasing Temperature

A shutdown cooling malfunction leading to a moderator temperature decrease could result from misoperation of the cooling water controls for the RHRS heat exchangers. The resulting temperature decrease causes a slow insertion of positive reactivity into the core. If the reactor were critical or near critical (operating states B or D), 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.

14.10.2.3 Inadvertent Pump Start

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 RCIC System or the HPCI System. In all of these cases, the normal feedwater flow would be correspondingly reduced by the water level controls. The net result is simply a replacement of a portion of the 370°F feedwater flow (at design power operation) by approximately 100°F flow.

The severity of the resulting transient is highest for the largest of these abnormal events: the inadvertent startup of the large, 5000 gpm HPCI System.

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 given above. As in that case, the most threatening transient would occur where minimum initial fuel thermal margins exist (maximum power within reactor operating state F). The HPCI startup transient is clearly less severe than the loss of feedwater heater case because its effect on mixed feedwater temperature will produce a change of only 46°F compared to the 100°F change previously analyzed. For this reason, no fuel clad barrier damage will result for the malfunction or inadvertent startup of any of these auxiliary cold water supply systems.

14.10.3 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, and
d. Fuel assembly insertion error during refueling.

14.10.3.1 Continuous Rod Withdrawal During Power Range Operation

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 (approximately 0.01 Δk) 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 for each reload cycle using the methodology in NEDE-24011-P-A and the results presented in the Reload Licensing Report.

Figure 14.10-7 shows typical results of an analysis of the continuous withdrawal, at design power, of the rod with the maximum possible worth. For this analysis, the central rod was left fully inserted in the core and all other rods withdrawn such that the worth of the central rod was maximized. This rod configuration could only be achieved by deliberate operator action or by numerous operator errors during rod pattern manipulation prior to the selection and complete withdrawal of the rod. Abnormal indications and APRM alarms would warn the operator as he approaches this abnormal rod pattern. The rod block monitors (RBM) stop the rod withdrawal. The increase in nuclear system pressure is less than 50 psi. Thus, neither nuclear system process barrier damage nor fuel damage occur.

14.10.3.2 Continuous Rod Withdrawal During Reactor Startup

Control rod withdrawal errors are considered when the reactor is at power levels below the power range. 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 APRMs at 15 percent and at 120 percent of rated power.

The analysis was performed for a 2.5 percent Δk control rod withdrawal at the rod drive speed of 3 in./sec starting from an average moderator temperature of 82°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 approximately 1 percent plastic strain, which corresponds to approximately 3.3 mils radial expansion.

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

14.10.3.3 Control Rod Removal Error During Refueling

The nuclear characteristics of the core assure 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.

14.10.3.4 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.

14.10.4 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 are identified as causing the most severe transients in this category:

a. Pressure regulator failure,
b. Inadvertent opening of a main steam relief valve,
c. Loss of feedwater flow, and
d. Loss of auxiliary power.

14.10.4.1 Pressure Regulator Failure

If either the controlling pressure regulator or the backup pressure regulator fails in an open direction, the turbine admission valves can be fully opened, and the turbine bypass valves can be partially or fully opened. This action initially results in decreasing 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 regulator malfunction is limited by the turbine controls to about 125 percent of design flow.

Figure 14.10-8 graphically shows a typical transient, starting at design power, resulting from a pressure regulator malfunction in which a steam flow demand capable of fully opening the turbine control and bypass valves is assumed as a most severe case. The depressurization results in the formation of voids in the reactor coolant causing a rapid rise in reactor vessel water level up to the high level trips (level 8). The reactor scrams after about 2 seconds due to the trip of the main turbine. A typical turbine trip response occurs, but it is milder than the limiting cases since power had begun to drop due to the depressurization. The peak neutron flux and fuel surface heat flux do not exceed the initial power. There is no reduction in fuel thermal margins. The bypass system is also already open (due to the failed regulator), therefore the pressure increase is mild, opening only part of the main steam relief valves. They quickly reclose and the depressurization trend is reestablished. The main steam isolation valves automatically close when pressure at the turbine decreases below 840 psia (and the reactor mode switch is in RUN) near 50 seconds. (See Subsection 7.3, "Primary Containment and Reactor Vessel Isolation Control System"). The reactor vessel isolation limits the duration and severity of the final depressurization so that no significant thermal stresses are imposed on the nuclear system process barrier. After the rapid portion of the transient is complete and isolation is effective, the nuclear system main steam relief valves may again operate intermittently to relieve the pressure rise resulting from decay heat generation. Because the initial portion of the transient results in depressurization of the nuclear system and power reduction, only a portion of the main steam relief valves need to operate to relieve the pressure increase due to the nuclear system process barrier.

14.10.4.2 Inadvertent Opening of a Main Steam Relief Valve

The opening of a main steam relief valve 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. Figure 14.10-9 shows a typical transient resulting from the opening of a main steam relief valve with the capacity to pass 6.5 percent of rated nuclear system steam flow. An initial power level corresponding to design power conditions is assumed.

The pressure regulator senses the nuclear system pressure decrease and closes the turbine control valves far enough to maintain constant reactor vessel pressure. Reactor power settles out at nearly the initial power level. Automatic recirculation flow control (assumed to be active) increases recirculation flow to the maximum. Because the recirculation flow cannot satisfy the additional load demand, the pressure regulator setpoint is automatically reduced to its lower limit, and nuclear system pressure decreases. No fuel damage results from the transient. 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 main steam relief valves to remove decay heat.

14.10.4.3 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.

Feedwater control system failures or feedwater pump trips can lead to partial or complete loss of feedwater flow. Figure 14.10-10 graphically shows a typical transient resulting from the trip of all feed pumps from design power. Feedwater system inertia results in a 5 second feedwater flow decrease before flow is completely stopped. The decrease in feedwater flow produces a slight pressure drop and a decrease in core inlet subcooling which both increase core void fraction, and reduce reactor power initially and helps moderate the decrease in actual reactor vessel water level for the first few seconds of the transient. However, sensed reactor vessel water level decreases quickly, causing a reactor low water level scram at about 6 seconds. The maximum rate of actual level decrease is about 7 inches/second. Startup of the RCICS, HPCI, isolation of the main steam lines and recirculation pump trip occurs near 16 seconds when wide range level reaches about 50 inches below the separator skirt (-51.5 inches in Figure 14.10-10). The ability of the RCICS alone to maintain adequate core coverage is described under "Loss of Auxiliary Power" below.

Pressure in the reactor vessel decreases gradually with the power reduction so that no threat is posed for the nuclear system process barrier. After the main steam isolation valves close, decay heat slowly raises nuclear system pressure to the lowest main steam relief valve setting, but no excessive overpressure occurs.

This transient is most severe from high power (operating state F) conditions, since the rate of level decrease is greatest and the amount of stored and decay heat to be dissipated is highest.

After lowering the Main Steam Isolation Valve (MSIV) reactor water level set point (Reference NEDE-30012 December 1982), the transient for loss of feedwater flow was re-evaluated. This evaluation was performed by General Electric using the Appendix K evaluation models with the following conservative assumptions:

a. Conservative decay heat values (1973 ANS + 20%) are used to maximize heat addition to the vessel, main steam relief valve challenges, and inventory loss.

b. The initial reactor power is assumed at 102% of licensed power which also maximizes the above parameters.

c. The initial water level in the reactor vessel is assumed to be at the scram level (Level 3) and the reactor is scrammed at time zero. This is consistent with Appendix K LOCA analysis.

d. The feedwater pumps are assumed to coast down in 1 second. This is also consistent with the Appendix K LOCA analysis.

e. Only RCIC will initiate at Level 2. Since the HPCI injection rate is about 10 times that of RCIC, this assumption provides the most severe challenge to the reactor core cooling.

The major change in the transient is that the main steam lines are not isolated with the startup of HPCI and RCIC when the reactor water level reaches the reactor water level 2 setpoint. As shown in Figure 14.10-10a, RCIC alone is still capable of maintaining adequate core coverage with the MSIV's open. RCIC also maintains reactor water level above the MSIV water level isolation setpoint; therefore, the MSIV's remain open and the main condenser remains as a heat sink. As shown in Figure 14.10-10b, reactor pressure is maintained at approximately 950 psig by the turbine bypass valves. Pressure suppression pool heatup which could occur as a result of main steam relief valve actuation is totally eliminated from this event with the new MSIV reactor water level setpoint.

14.10.4.4 Loss of Auxiliary Power

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. 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 time = 0. Normal coastdown times were used for the recirculation and feedwater pumps.

b. At time = 5 seconds, the reactor protection system MG sets are assumed to coast down to the point that RPS instrumentation power is lost. This initiates closure of the MSIV's which also produces a scram signal after the valves have moved 10 percent of their total movement.

c. The condenser vacuum was assumed to continue dropping and reaches the turbine trip setting by 6 seconds. The turbine bypass valves open for a short period (about 2 seconds), then they close due to the loss of pressure in the main steam lines downstream of the MSIV's once the MSIV's complete their closure at time = 8 seconds.

Figure 14.10-11 graphically shows for loss of auxiliary power the simulated transients from design power. The initial portion of the transient is very similar to the loss of all feedwater described above except for the recirculation pump trip. Initiation of scram, isolation valve closure, and turbine trip all occur between 5 to 6 seconds and the transient changes to that of an isolation. The main steam relief valves open for a short time then sequentially reclose as the remainder of the stored heat is dissipated. Peak pressures reached only 100 psi above nominal operating pressures; therefore, no safety valve lifting was initiated nor were the vessel pressure limits approached. Note how the lowest main steam relief valve group in the model was reopened and reclosed as the generated heat drops down into the decay heat characteristic. This pressure and relief cycle will be continued with slower frequency and shorter relief discharges as the decay heat drops off up to the time the RHRS, in the shutdown cooling mode, can dissipate the heat. Sensed level dropped to the RCIC, HPCI and isolation initiation setpoint (about -51.5 inches in Figure 14.10-11) 25 seconds after the loss of auxiliary power.

A different transient results if complete connection with the external grids is lost at time = 0. The same sequence as above would be followed except that the reactor would also experience a generator load rejection and its associated trip scram at the beginning of the transient. Figure 14.10-12 shows this simulated loss of auxiliary power event from design power. No increase in neutron flux occurs due to the trip scram and the recirculation pump trips. No increase in fuel surface heat flux occurs, and the thermal behavior is again much like a simple recirculation pump trip. Peak pressures are virtually identical to the previous case; however, they occur sooner during the transient. Wide range (WR) sensed level dropped to the RCIC, HPCI and isolation initiation point by 30 seconds.

No fuel damage occurs in either case, since the only critical fuel transient is almost exactly the same as that experienced during the trip of both recirculation motor generator (MG) set drive motors. By about 20 seconds after the simulated losses of power, both transients look essentially identical. Pressure is cycling about the lowest main steam relief valve setpoints and water level is dropping gradually, waiting for RCIC (or HPCI) operation to restore water level control. Figure 14.10-13 shows the calculated long-term water level transient conservatively considering RCIC operation only beginning at 90 seconds and reaching full RCIC flow (600 gpm) at 120 seconds. The minimum calculated water level is 100 inches above the top of the active fuel, providing ample margin.

14.10.5 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 result in the most significant transients in this category:

a. Recirculation Flow Control Failure-Decreasing Flow,
b. Trip of One Recirculation Pump,
c. Trip of Two Recirculation Pump MG Set Drive Motors, and
d. Recirculation Pump Seizure.

14.10.5.1 Recirculation Flow Control Failure-Decreasing Flow

Several varieties of recirculation flow control malfunctions can cause a decrease in core coolant flow. The master controller could malfunction in such a way that a zero speed signal is generated for both recirculation pumps. The recirculation flow control system is provided with a speed demand limiter which is set so that this situation cannot be more severe than the simultaneous tripping of both recirculation pump MG set drive motors. A simultaneous trip of both recirculation pump MG set drive motors is evaluated in paragraph 14.10.5.3.

The remaining recirculation flow controller malfunction is one in which the speed controller for one recirculation pump set fails in such a way that the speed controller output signal changes in the direction of zero speed. This transient is similar but less severe than the trip of one recirculation pump. A trip of one recirculation pump is evaluated in paragraph 14.10.5.2.

14.10.5.2 Trip of One Recirculation Pump

Normal trip of one recirculation pump is accomplished through the drive motor breaker. However, a worse coastdown transient occurs if the generator field excitation breaker is opened, separating the pump and its motor from the inertia of the MG set. This condition was assumed for this calculation. Figure 14.10-14 shows a typical transient from design power conditions. Diffuser flows on the tripped side reverse at about 3 seconds; however, M-ratio in the active jet pumps increases greatly, producing about 150 percent of normal diffuser flow. No fuel damage results from this transient.

14.10.5.3 Trip of Two Recirculation Pump MG Set Drive Motors

This two-loop trip provides the evaluation of the fuel thermal margins maintained by the rotating inertia of the recirculation drive equipment. No single operator act or plant protection action can produce simultaneous trip of the generator field breakers in both loops. Plant protection action can, however, simultaneously trip the power supplying the MG set drive motors. Also, the recirculation pump trip (RPT) system can trip both pumps.

Figure 14.10-15 graphically shows the transient resulting from the trip of both MG set drive motors with the minimum design rotating inertia from design power. Fuel thermal margin reached its worst condition near 2.0 seconds; however, no damage to the fuel barrier occurs. No scram is initiated directly by the simultaneous MG set motor trip and the power will settle out at part-load, natural circulation conditions. An inadvertent RPT has also been analyzed and shown to have similar results.

14.10.5.4 Recirculation Pump Seizure*

This case represents the instantaneous stoppage of the pump motor shaft of one recirculation pump. It produced the most rapid decrease of core flow. The reactor is assumed to be operating at design power. Figure 14.10-16 shows a typical transient. The fast decrease in recirculation flow in the seized loop is due to the large hydraulic resistance introduced by the stopped rotor. Core coolant flow reaches its minimum value at about 1.5 seconds. Nucleate boiling is maintained throughout the transient and no damage occurs to the fuel barrier. No scram occurs. The initial pressure regulator maintains pressure control as the reactor settles out at the final, lower power condition. Because nuclear system pressure decreases throughout the transient, the nuclear system process barrier is not threatened by overpressure.

*This event has been reclassified as an accident (see NEDE-24011-P-A-US)

14.10.6 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 result in the most significant transients in this category:

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

14.10.6.1 Recirculation Flow Controller Failure - Increasing Flow

Several possibilities exist for an unplanned increase in core coolant flow resulting from a recirculation flow control system malfunction. Failure of the master controller can result in a speed increase for both recirculation pumps. On Unit 1, the maximum output signal of the master controller is provided with rate limits which are adjusted in such a way that a master controller failure is less severe than a failure of one of the MG set speed controllers. The most severe case of increasing coolant flow results when the MG set fluid coupler for one recirculation pump attempts to achieve full speed at maximum acceleration. The maximum acceleration for this failure is 25 percent of full speed per second. The most severe transient results when reactor power is initially at 68 percent of rated, which is at the lower end of the automatic flow control range. These conditions correspond to the lowest power and flow conditions on the automatic flow control characteristic curve for the reactor.

Figure 14.10-17 shows typical results of the transient. The changes in nuclear system pressure are not significant with regard to overpressure. The pressure decreases over most of the transient. The rapid increase in core coolant flow causes an increase in neutron flux, which initiates a reactor scram. The transient fuel surface heat flux reaches 83 percent of rated heat flux, but it barely exceeds the steady state power-flow control curve. No fuel damage occurs.

14.10.6.2 Startup of Idle Recirculation Pump

The transient response to the starting of an idle recirculation loop without warming the drive loop water is shown in Figure 14.10-18. The assumed initial conditions are as follows:

a. One recirculation loop is idle and filled with cold water (100°F). (Normal procedure requires warming this loop.),

b. The active recirculation pump is operating at a speed that produces about 125 percent of normal rated jet pump diffuser flow in the active jet pumps,

c. The core is receiving 48 percent of its normal rated flow; the remainder of the coolant flows in the reverse direction up through the inactive jet pumps,

d. Reactor power is 68 percent of design power. This is the highest initial power for which a high neutron flux scram is not initiated. (Normal procedures require startup of an idle loop at a much lower power.) If transient is initiated from higher power scram will occur and the results will be less severe,

e. The idle recirculation pump suction and bypass valves are open: the pump discharge valve is closed, and

f. The idle pump fluid coupler is at a setting which approximates 50 percent of generator speed demand.

The loop startup transient sequence is:

a. The recirculation pump MG set breaker is closed at t = 0,

b. The motor reaches near synchronous speed quickly, while the generator approaches full speed in approximately 5 seconds,

c. Next, the generator field breaker is closed, loading the generator and applying starting torque to the pump motor and generator speed decreases as the generator tries to start the stopped rotor of the pump. Pump breakaway is modeled to occur at 8 seconds. Speed demand is sequentially programmed back to 20 percent of rated speed, and

d. The pump discharge valve is started open as soon as its interlock with the drive motor breaker is cleared. (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 reactor vessel.) A nonlinear 30-second valve opening characteristic is used.

Shortly after the pump begins to move, a surge in flow from the started up jet pump diffusers gives the core inlet flow a sharp rise. A short-duration neutron flux peak of almost 105 percent (no scram occurs) is produced; however, surface heat flux follows the slower response of the fuel. No damage occurs to the fuel barrier and no significant changes in nuclear system pressure result from the transient.

Throughout the transient, diffuser flow in the startup loop jet pumps is either reversed or less than about 10 percent of rated. For this reason, the cold loop water does not significantly affect the transient.

14.10.7 Event Resulting in Excess of Coolant Inventory

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 Feedwater Control System Failure - Maximum Demand is the limiting event of the excess coolant inventory type. The analysis results for the feedwater controller failure to maximum demand for the current cycle are presented in the Reload Licensing Report. The methodology and analysis assumptions for the current reload cycle analysis described in NEDE-24011-P-A differ from the older analysis described below.

The typical response of the plant to a failure of the feedwater controller which demanded maximum flow is shown in Figure 14.10-19. The transient was initiated from the low end of the analytical automatic flow control range (68 percent rated power) producing a more severe steam/feed flow mismatch and level transient than would be produced at higher power. The feedwater pumps were assumed to accelerate to their maximum capability.

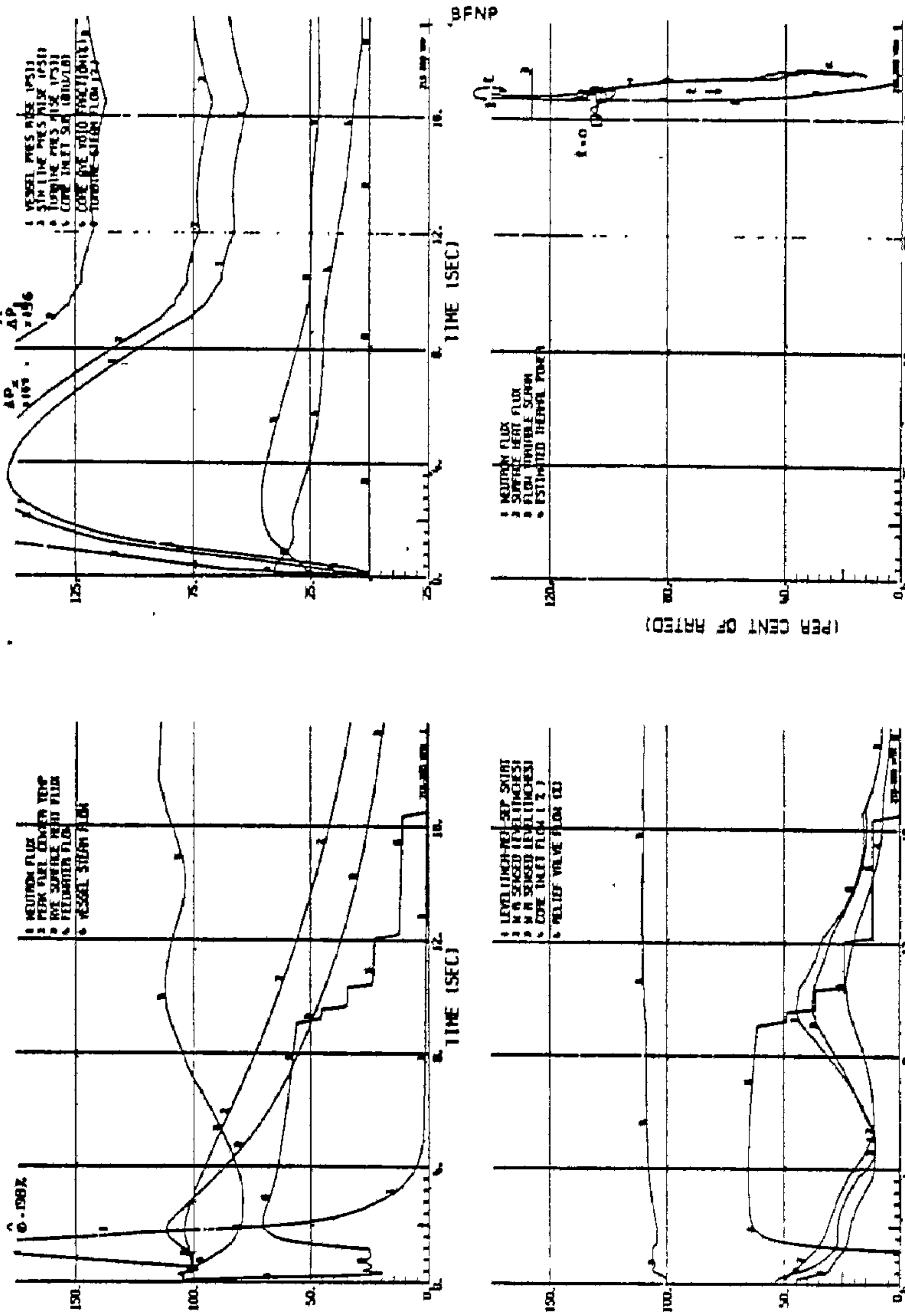
Sensed and actual water level increase during the initial part of the transient at about 4.0 inches/sec. The high water level main turbine trip and feedwater turbine trip was initiated at 5 seconds when sensed level had increased about 19-21 inches preventing excessive carryover from damaging the turbines. Scram occurs simultaneously with the turbine trip, limiting the neutron flux peak and fuel thermal transient so that no fuel damage occurs.

The turbine bypass system opens to limit the pressure rise. The lower set main steam relief valves open only momentarily and no excessive overpressure of the nuclear system process barrier occurs. The bypass valves close at about 24 seconds, bringing the pressure in the vessel under control during reactor shutdown.

Although lower initial power conditions would result in more rapid increases in level, high power cases represent the maximum threat to fuel clad and nuclear system process barriers. Obviously, no power transient will occur if the reactor is shut down (operating States C and E).

14.10.8 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 shut down the reactor from outside the control room. (For additional information see Section 7.18 - Backup-Control System)

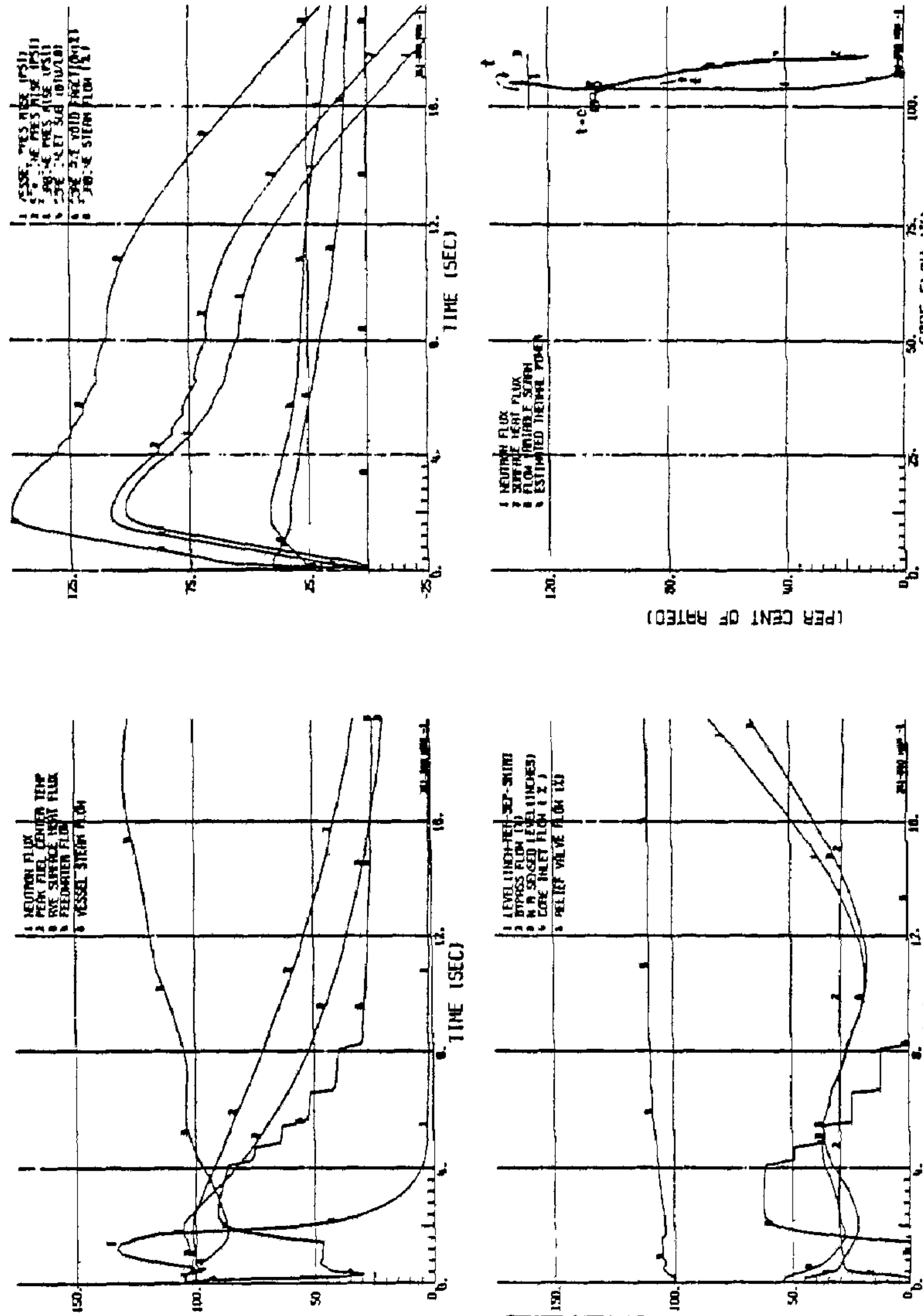


BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

Transient Results, Turbine Trip from High Power without Bypass and Loss of Condenser Vacuum

FIGURE 14.10-1

AMENDMENT 17



AMENDMENT 17

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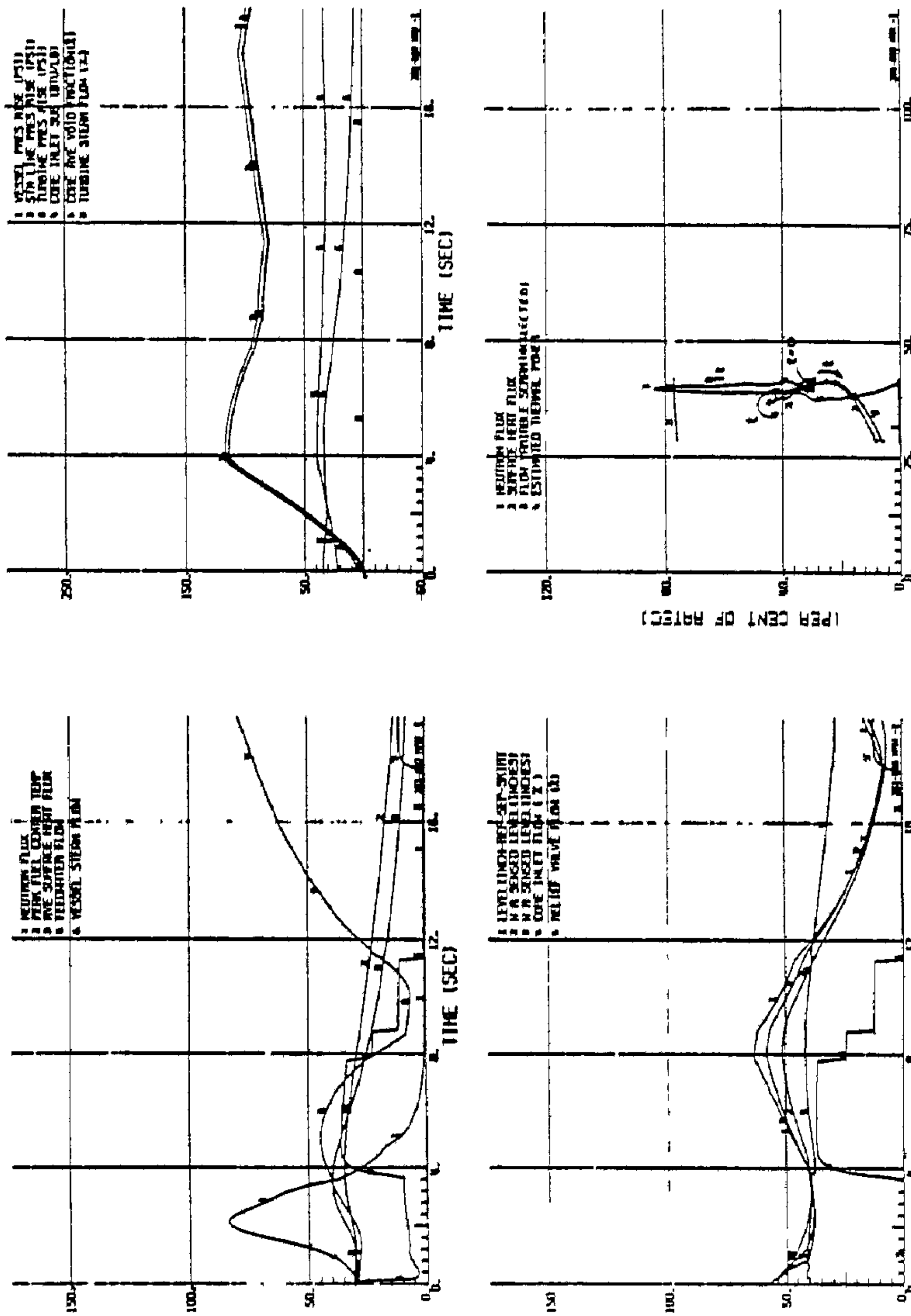
Transient Results, Turbine Trip

FIGURE 14 10-2



AMENDMENT 17

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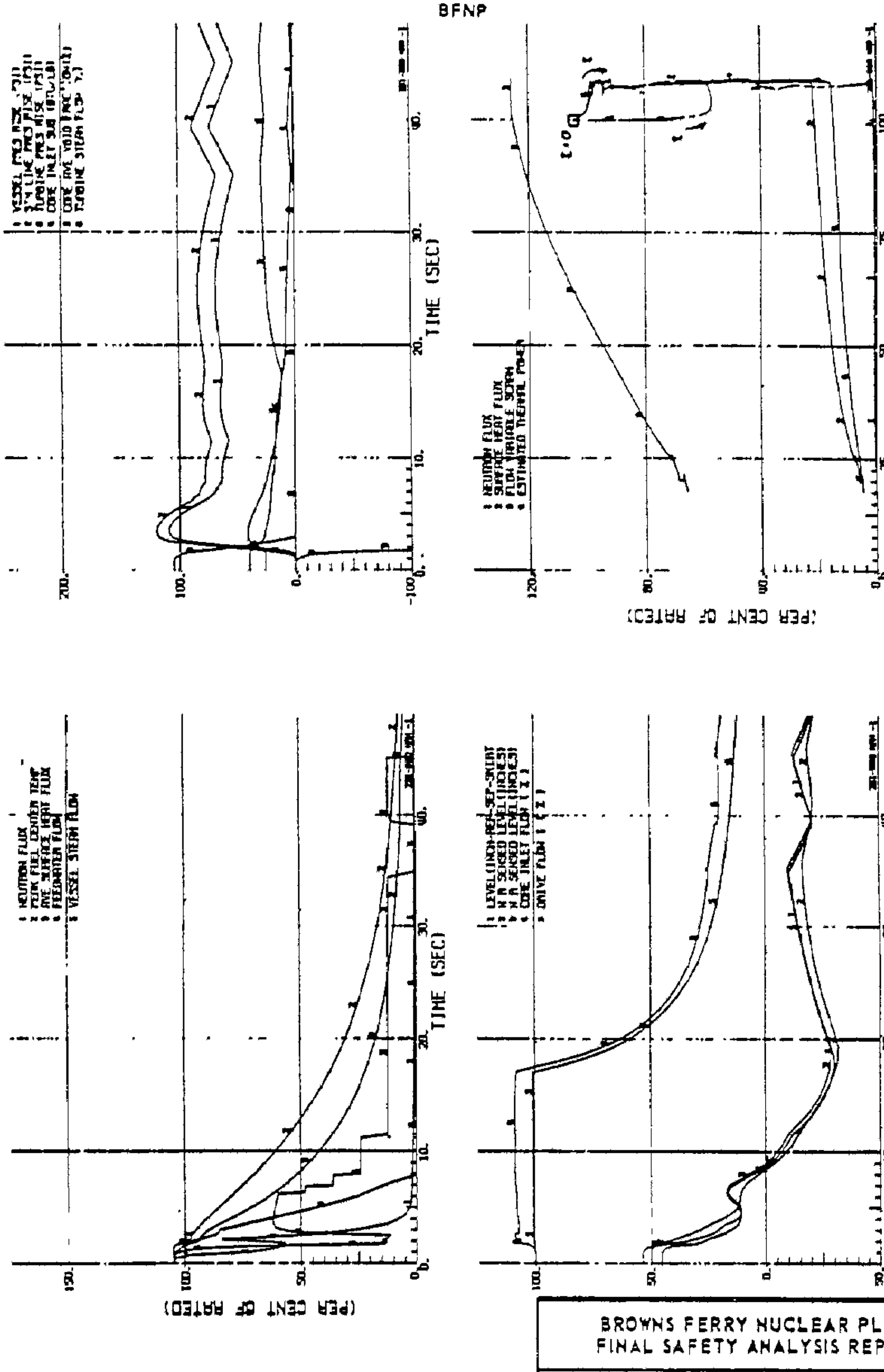
Transient Results, Turbine Trip from Low Power without Bypass

FIGURE 14 10-3



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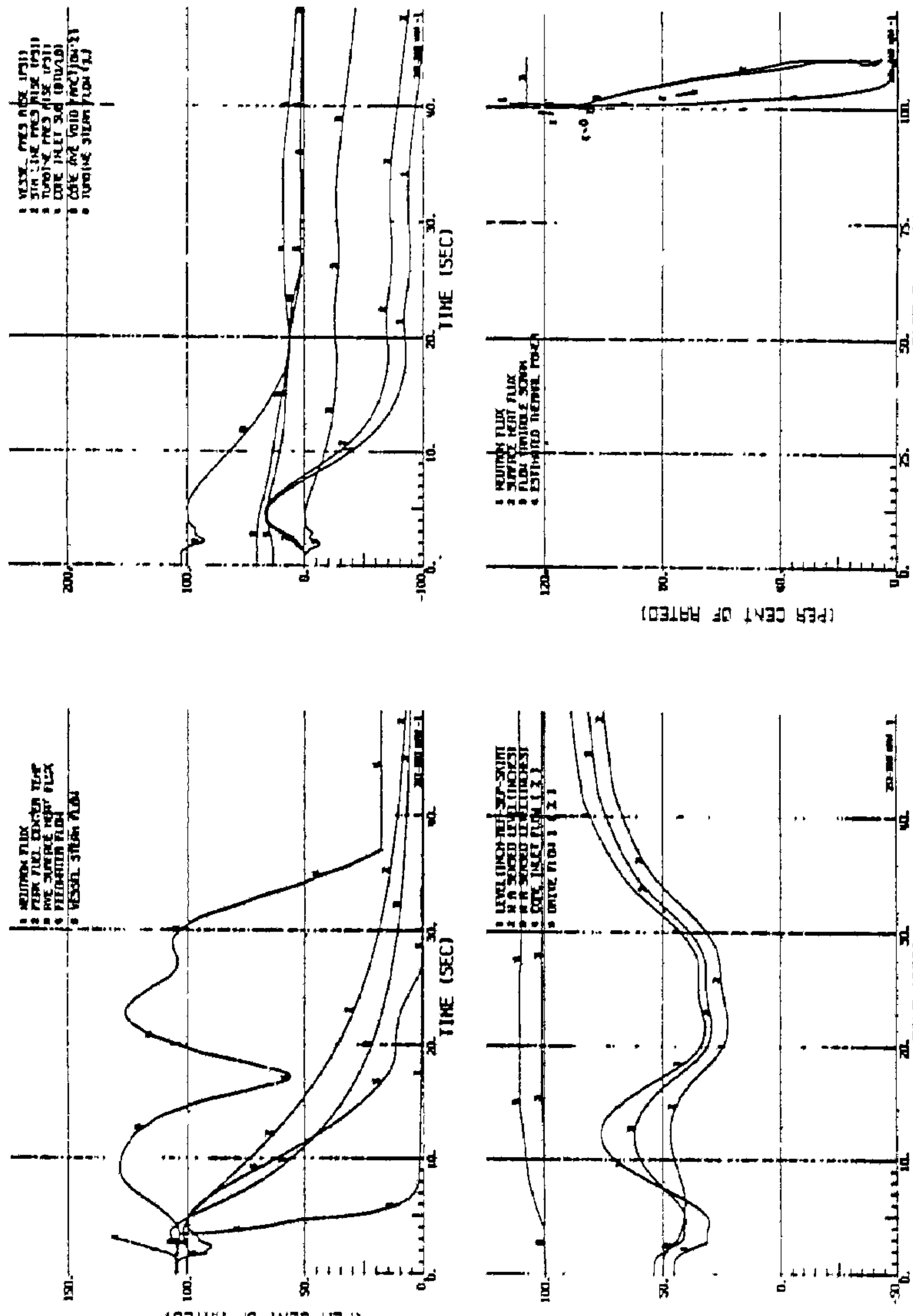
Transient Results, Closure of All Main Steam Isolation Valves

FIGURE 14.10-4

AMENDMENT 17



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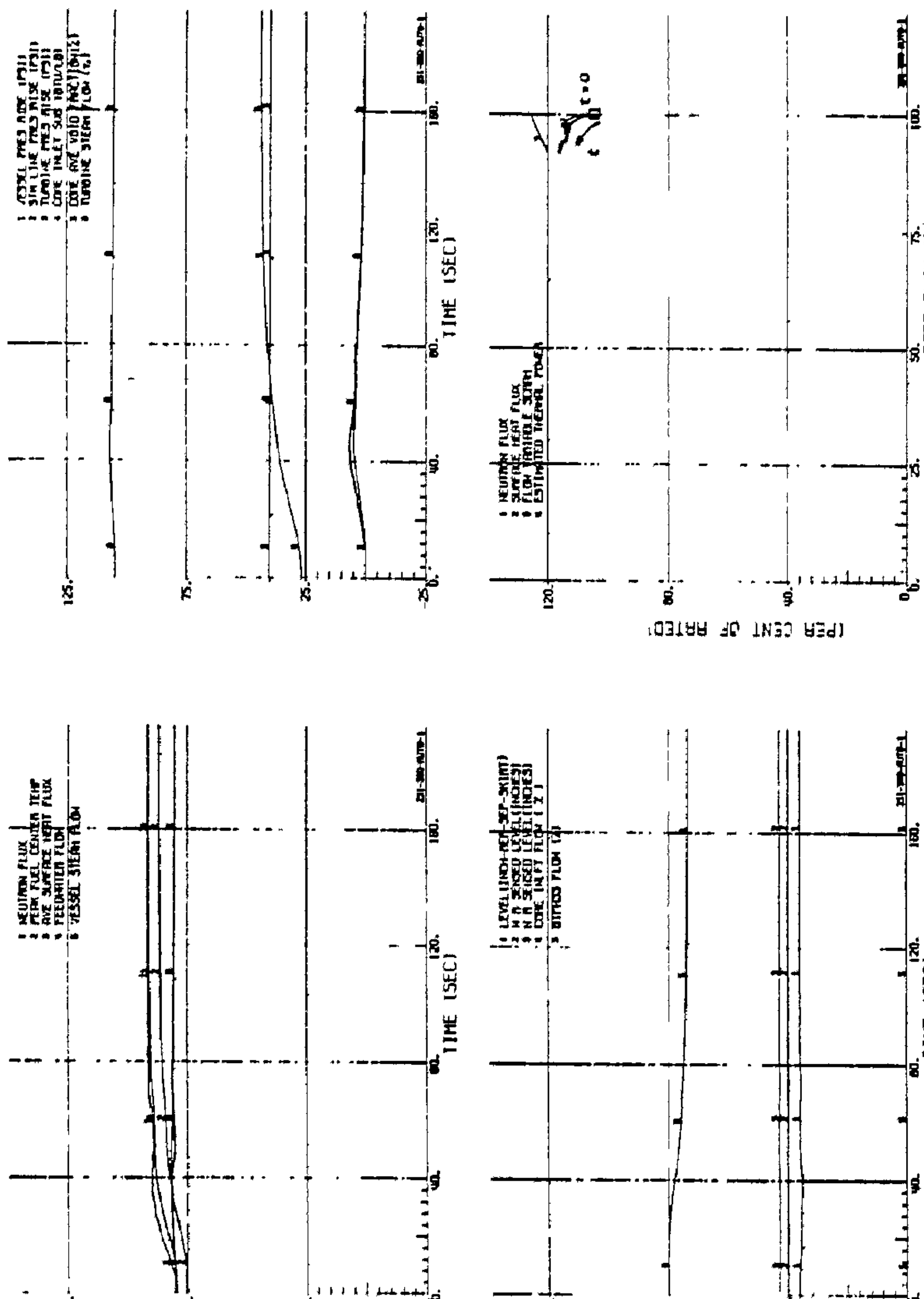
Transient Results, Closure of One Main Steam Isolation Valve

FIGURE 14 10-5



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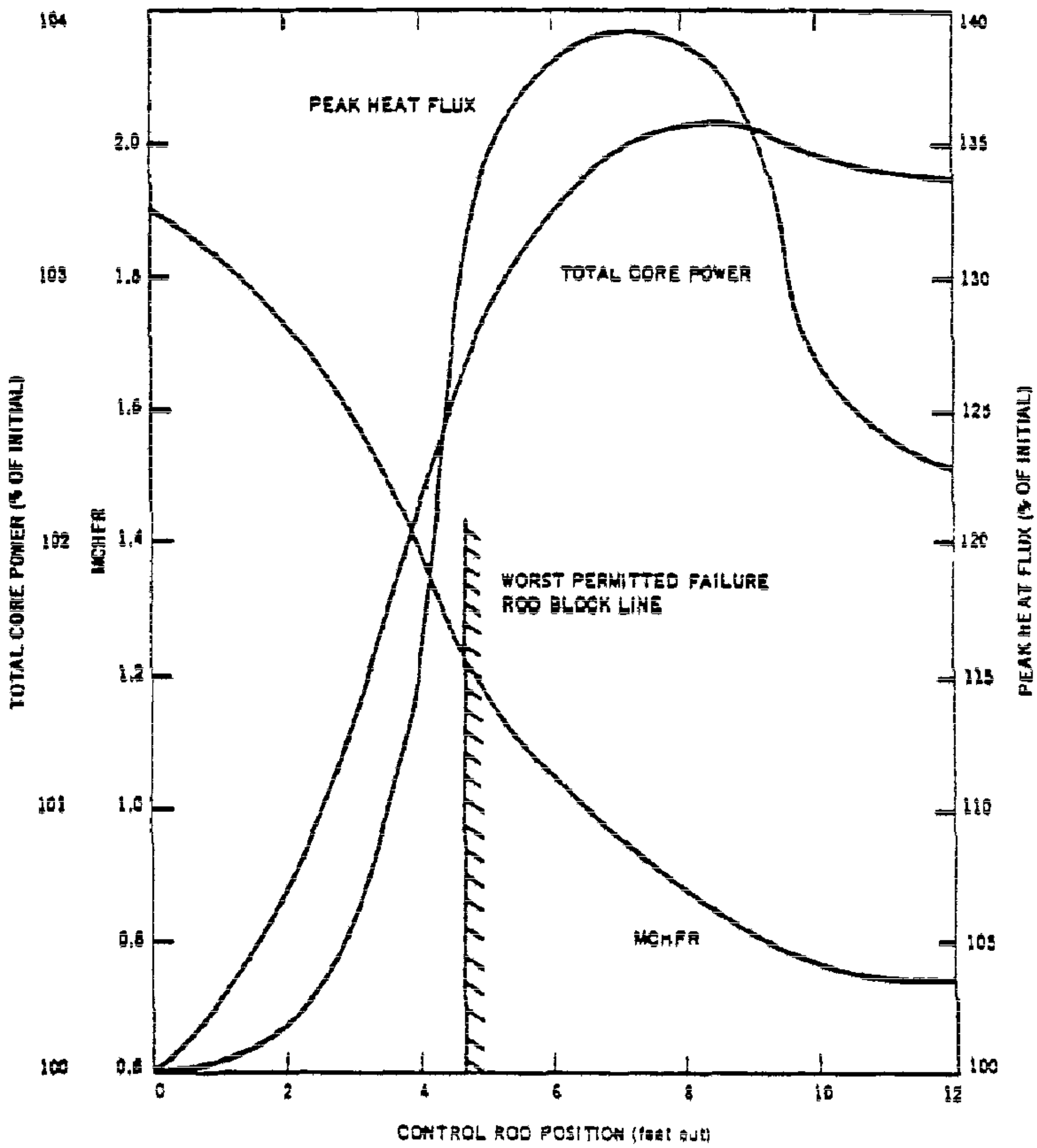
Transients Results,
Loss of Feedwater Heater

FIGURE 14.10-6



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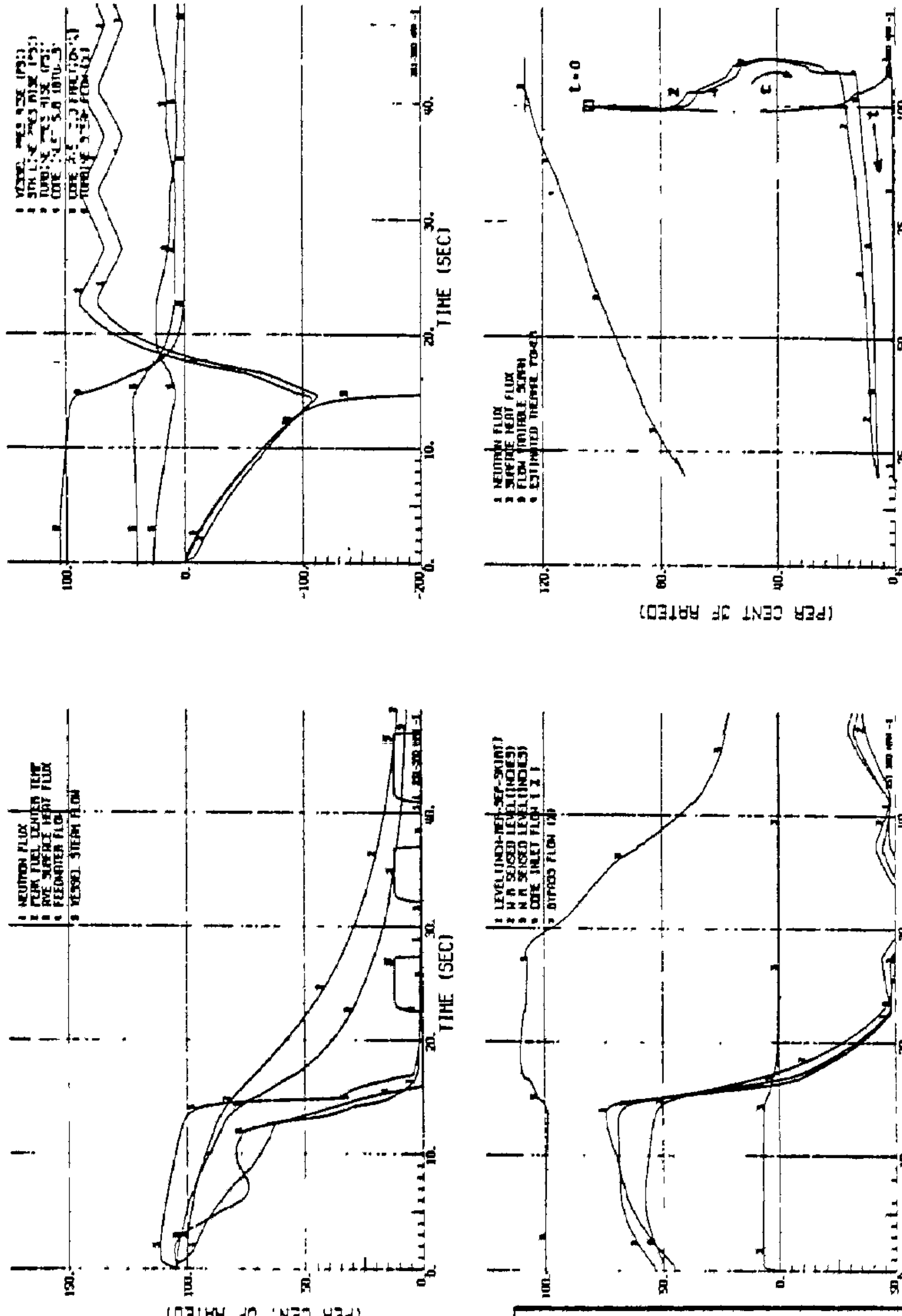
FINAL SAFETY ANALYSIS REPORT

Transient Results,
Continuous Rod Withdrawal
During Power Range Operation
FIGURE 14.10-7



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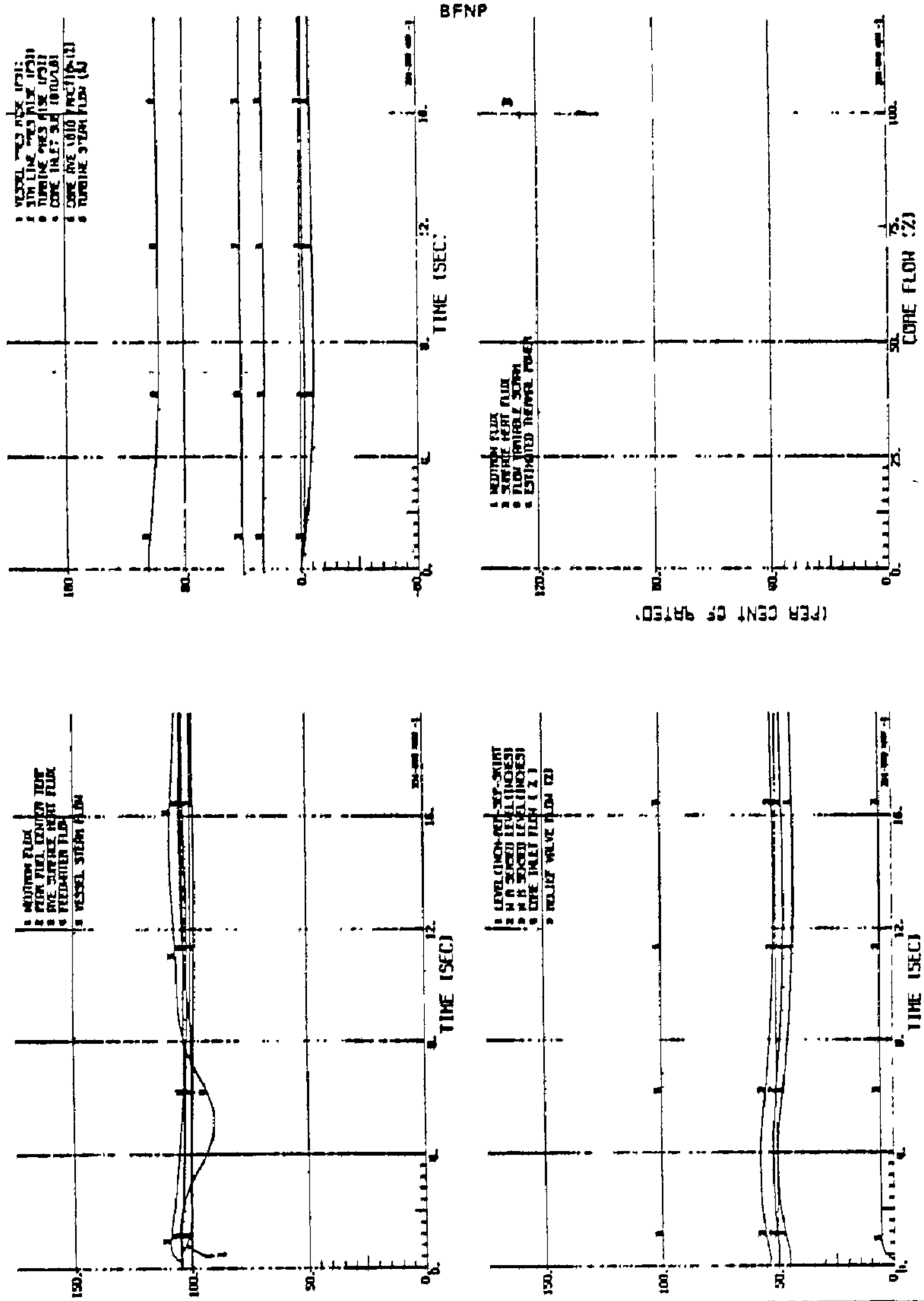
BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

Transient Results,
Pressure Regulator Failure

FIGURE 14.10-8

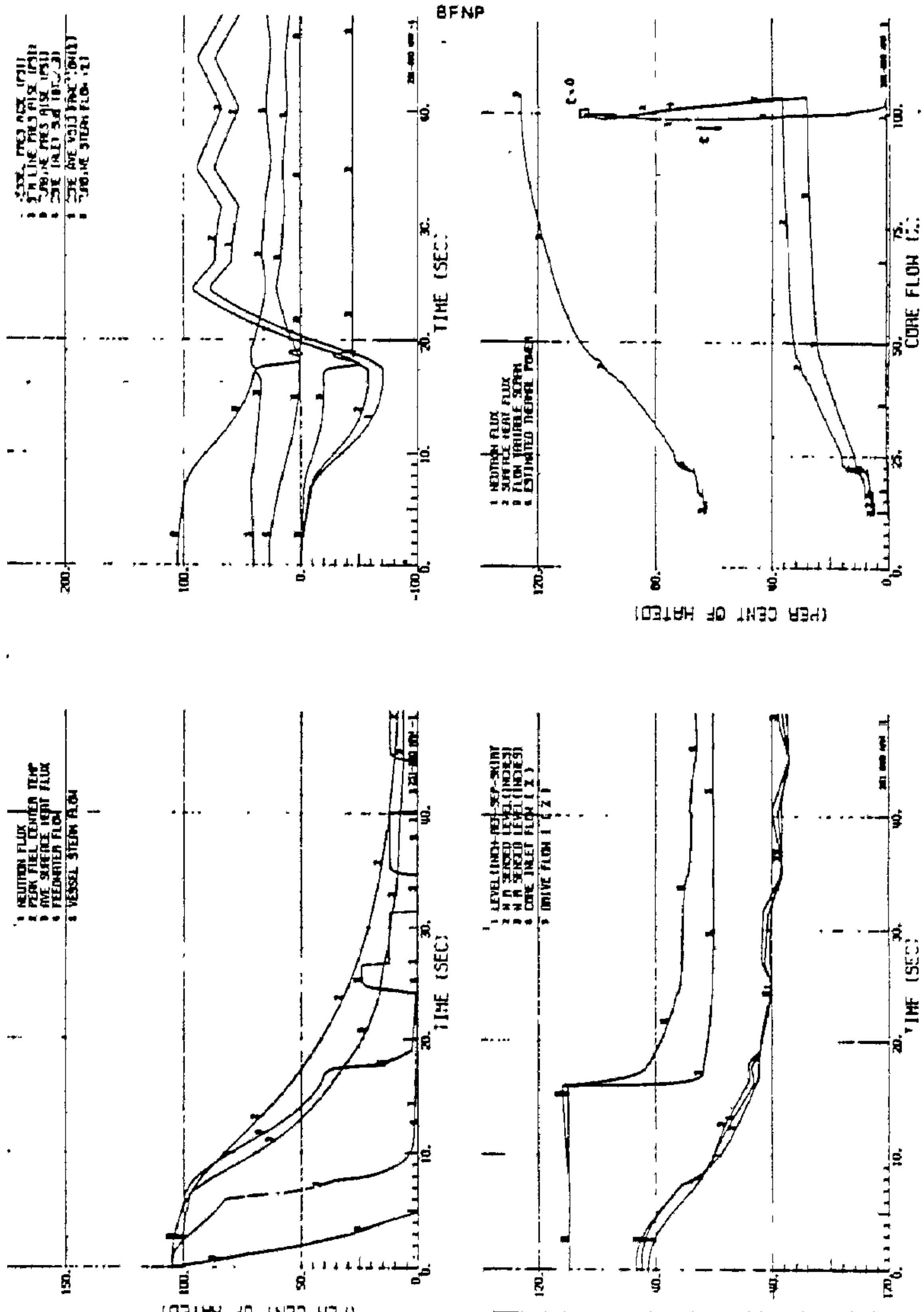


AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

Transient Results,
Inadvertent Opening of a Relief Valve
or Safety Valve

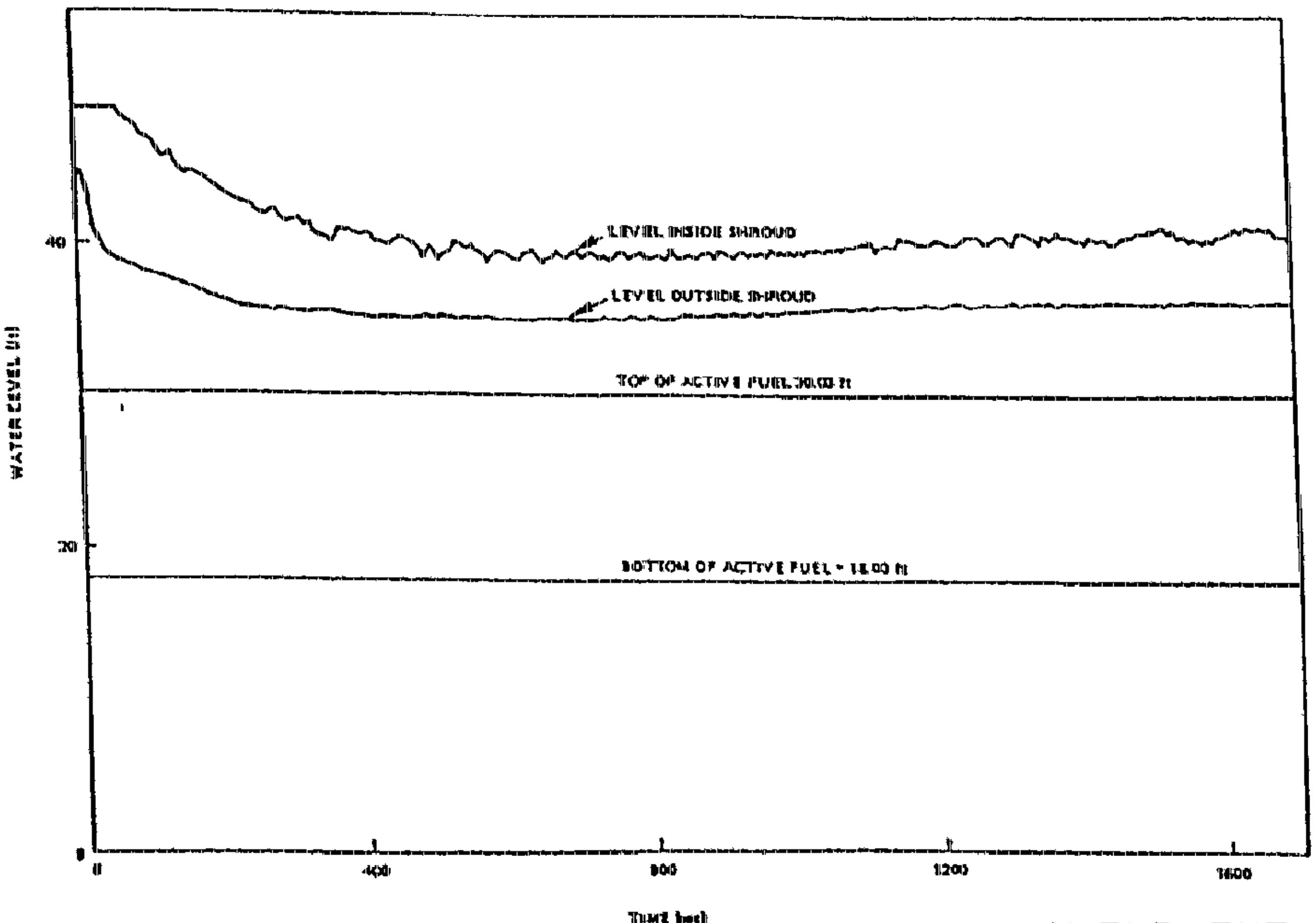
FIGURE 14.10-9



AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

Transient Results,
Loss of Feedwater Flow
FIGURE 14 10-10



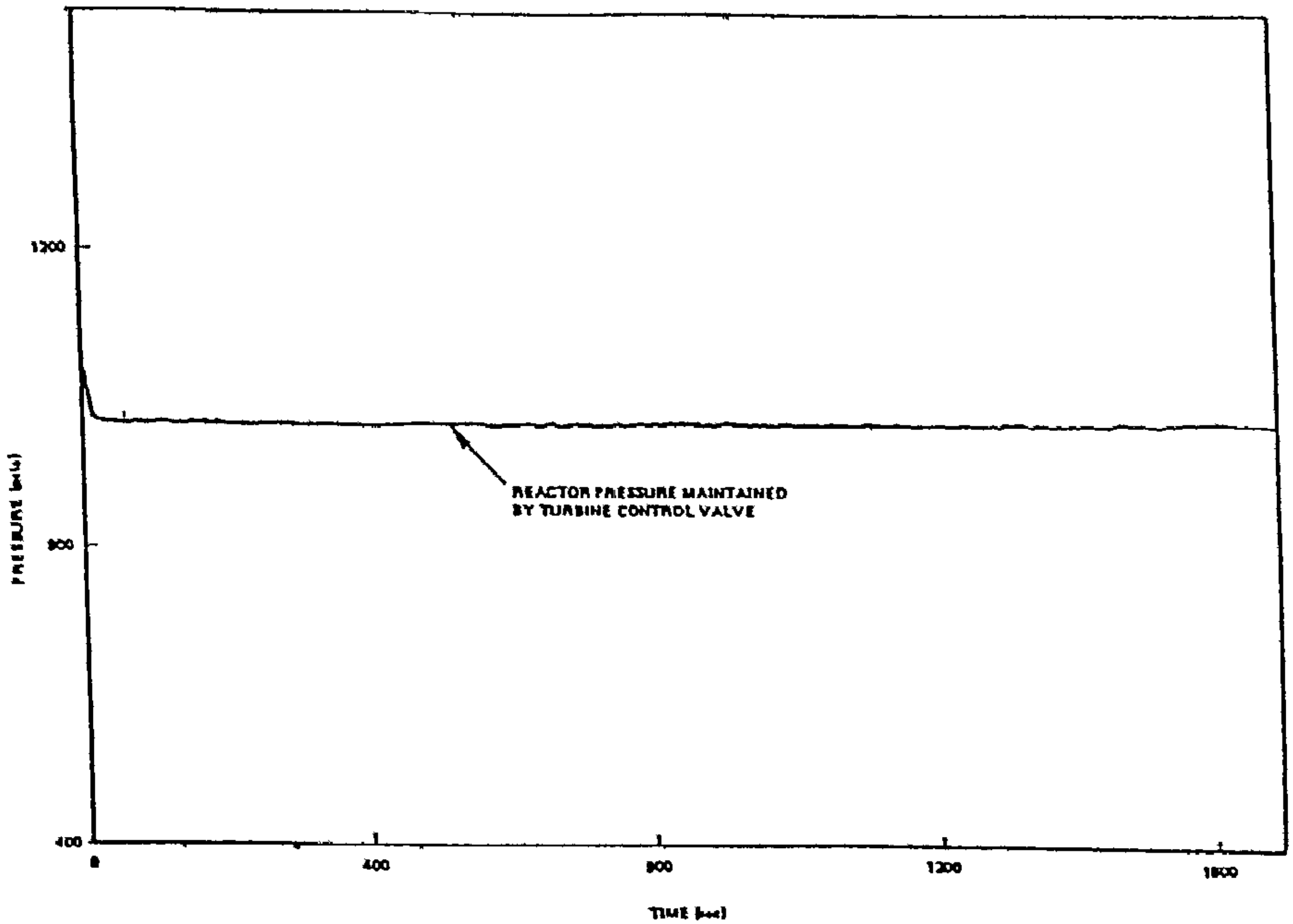
AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT

FINAL SAFETY ANALYSIS REPORT

Water Level Response with RCIC for Loss of Feedwater Flow Event

FIGURE 14.10-10a



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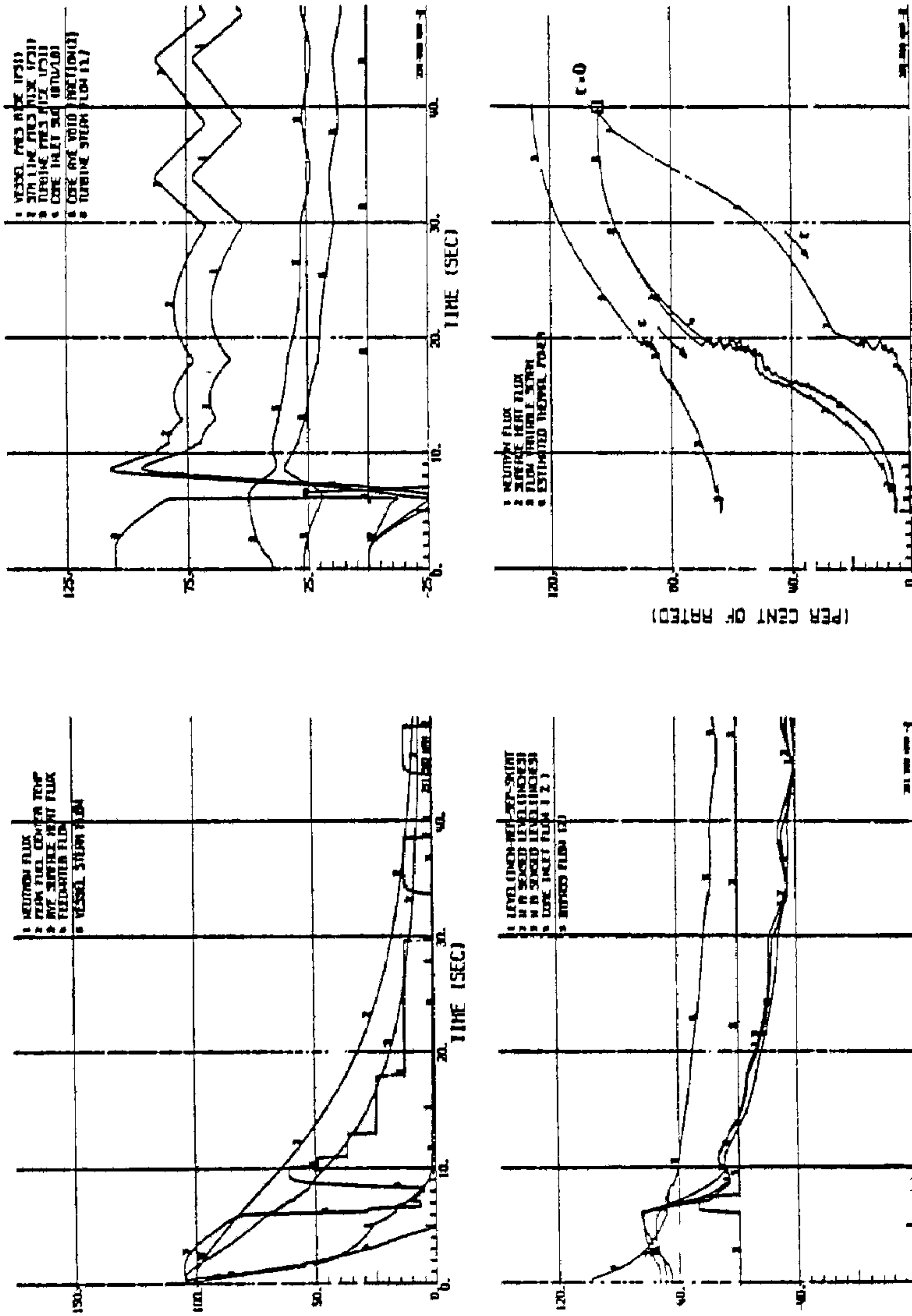
BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

Pressure Response with RCIC for Loss of Feedwater Flow Event

FIGURE 14.10-10b

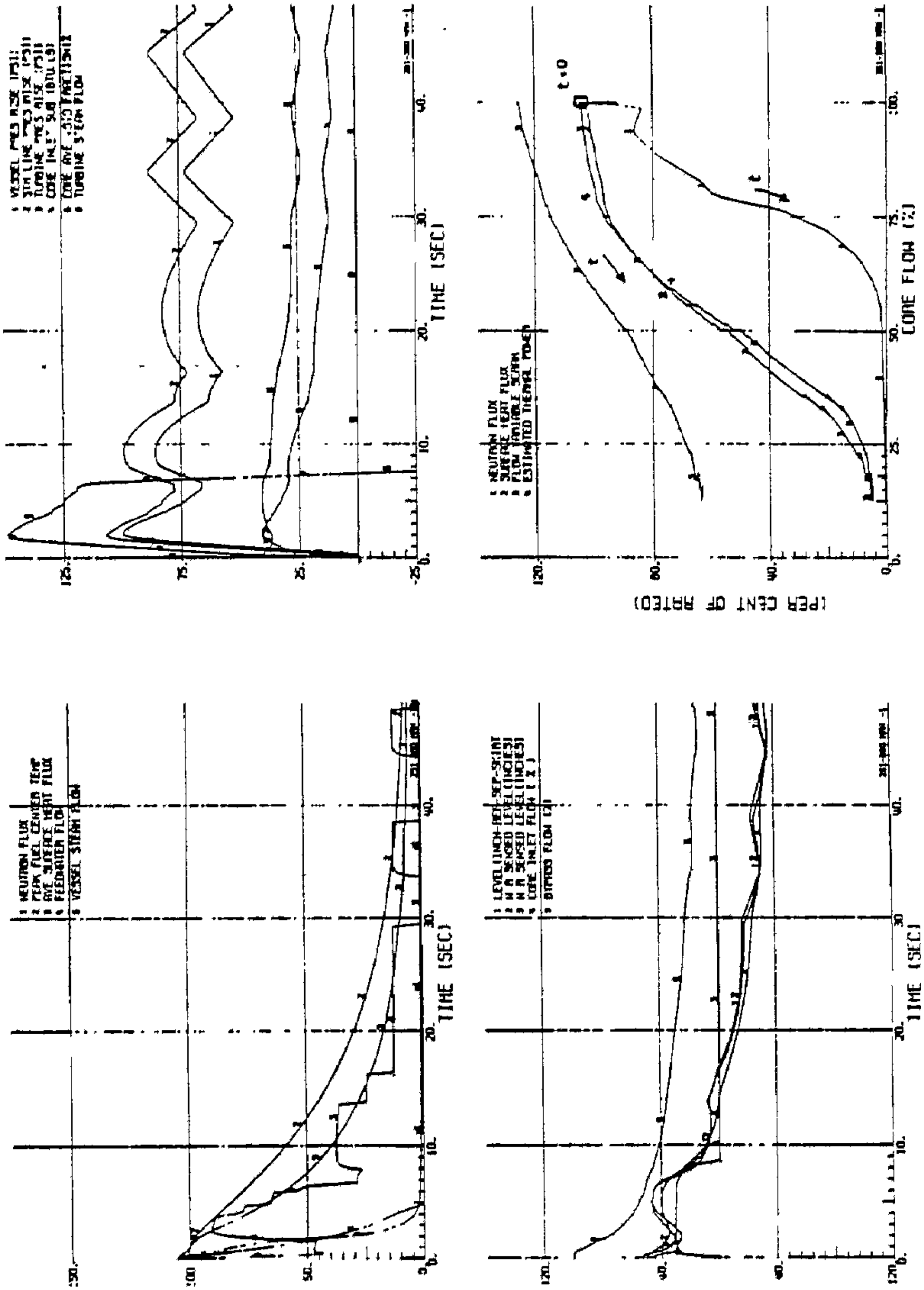


AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

Transient Results,
Loss of Auxiliary Power

FIGURE 14.10-11

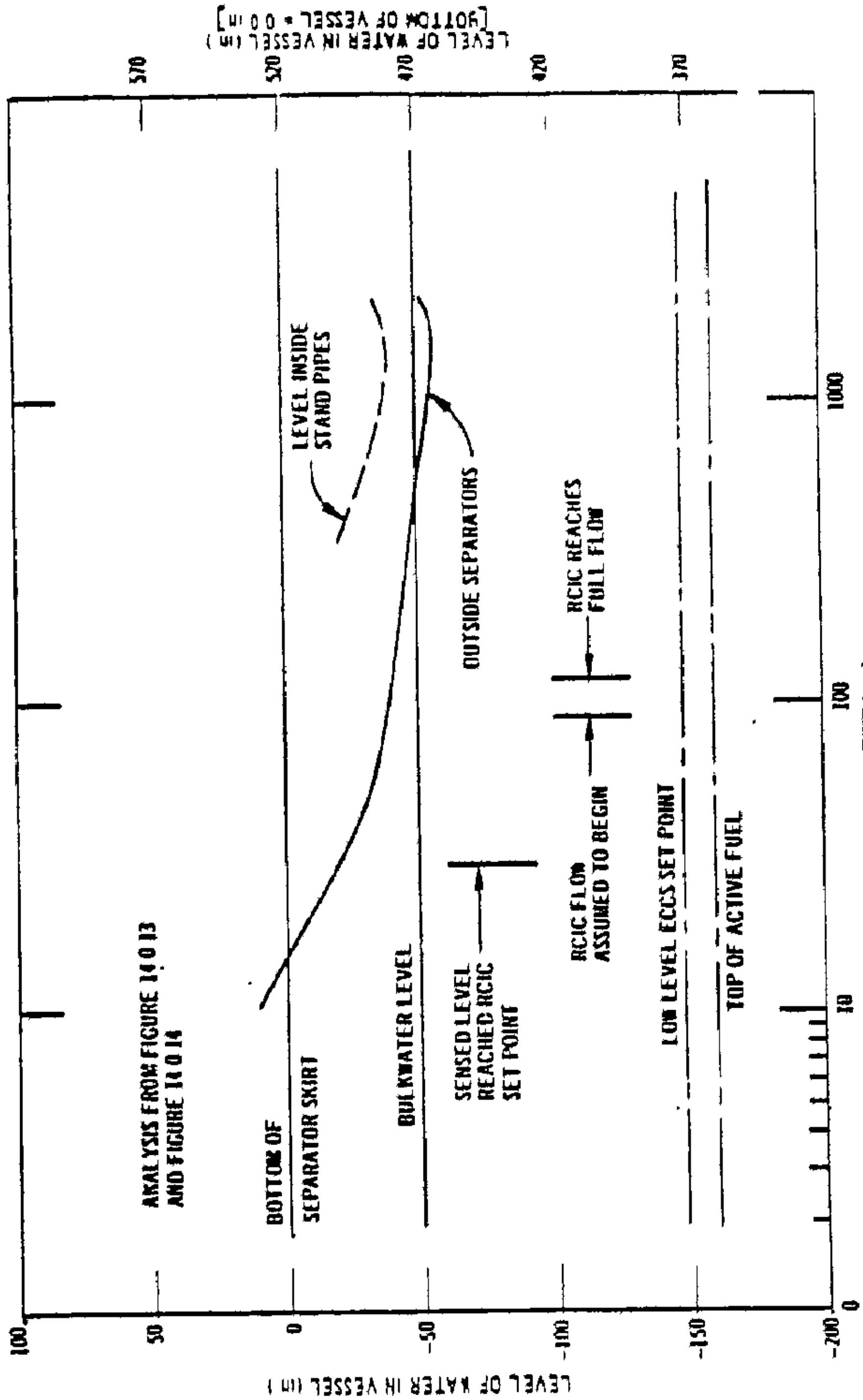


AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

Transient Results,
Loss of Auxiliary Power—All
Grid Connections

FIGURE 14 10-12

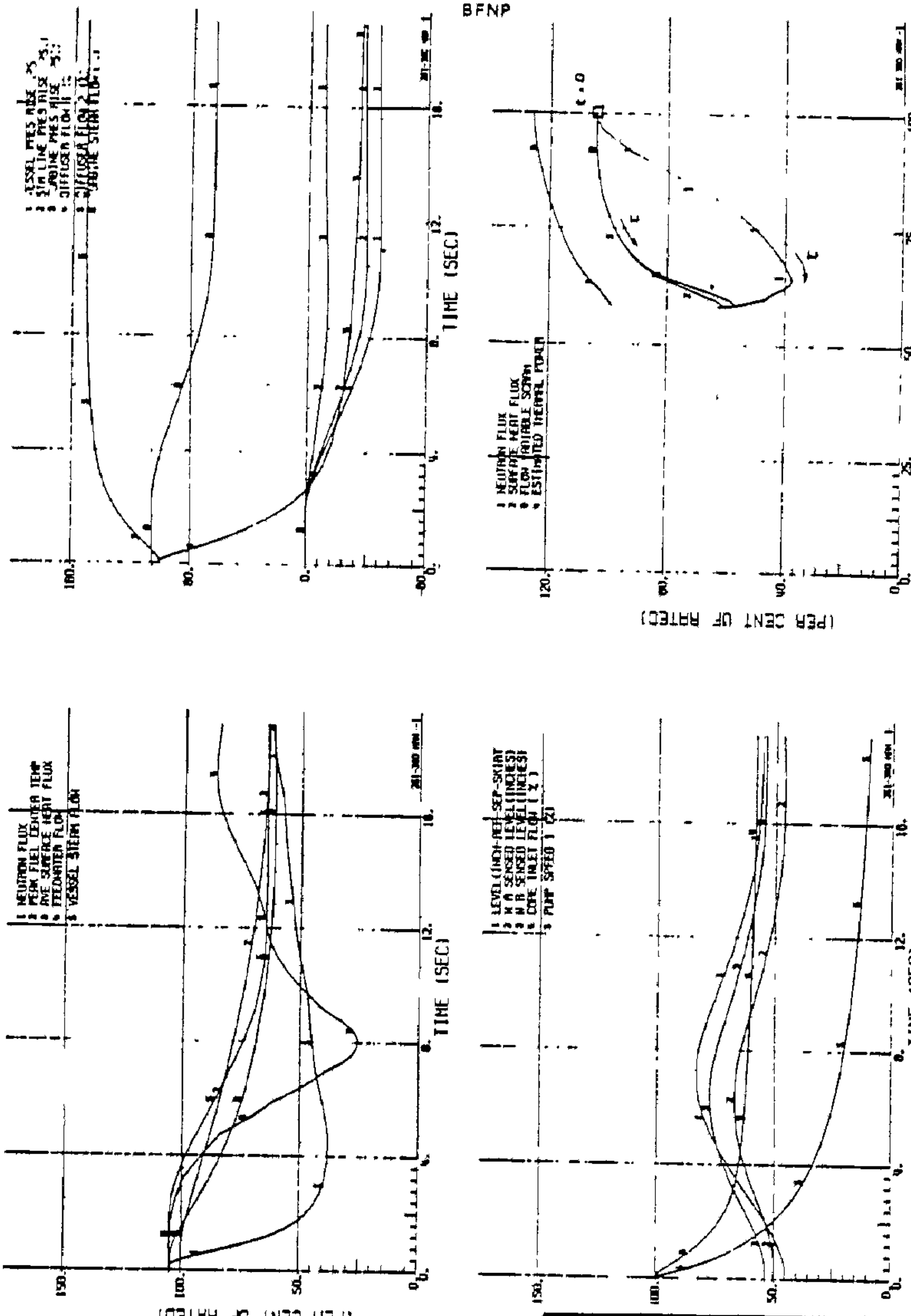


AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

Water Level vs. Time
Following Loss of Auxiliary Power
(RCIC Only)

FIGURE 14.10-13

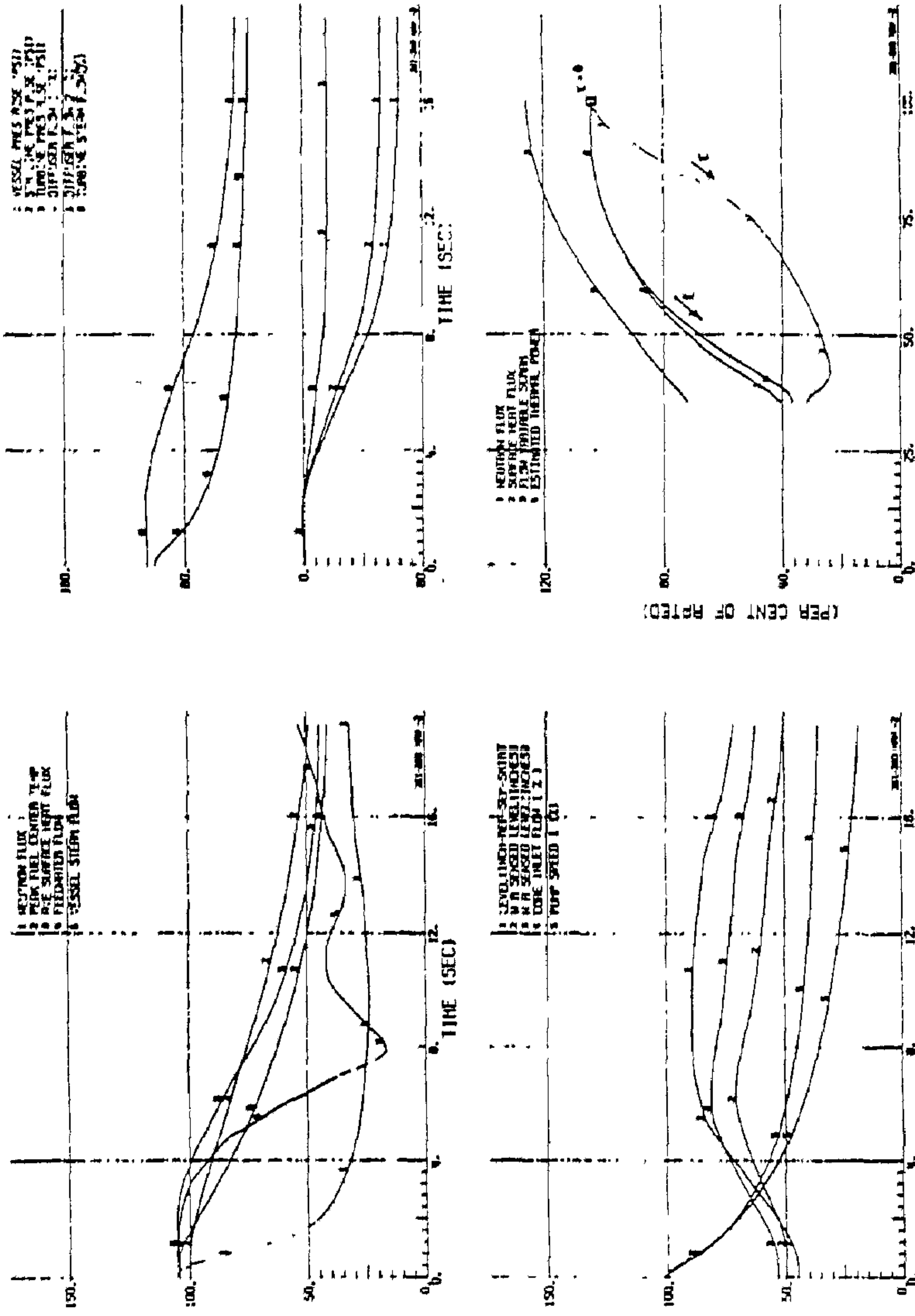


AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

Transient Results,
Trip of One Recirculation System
Generator Field Breaker

FIGURE 14 10-14

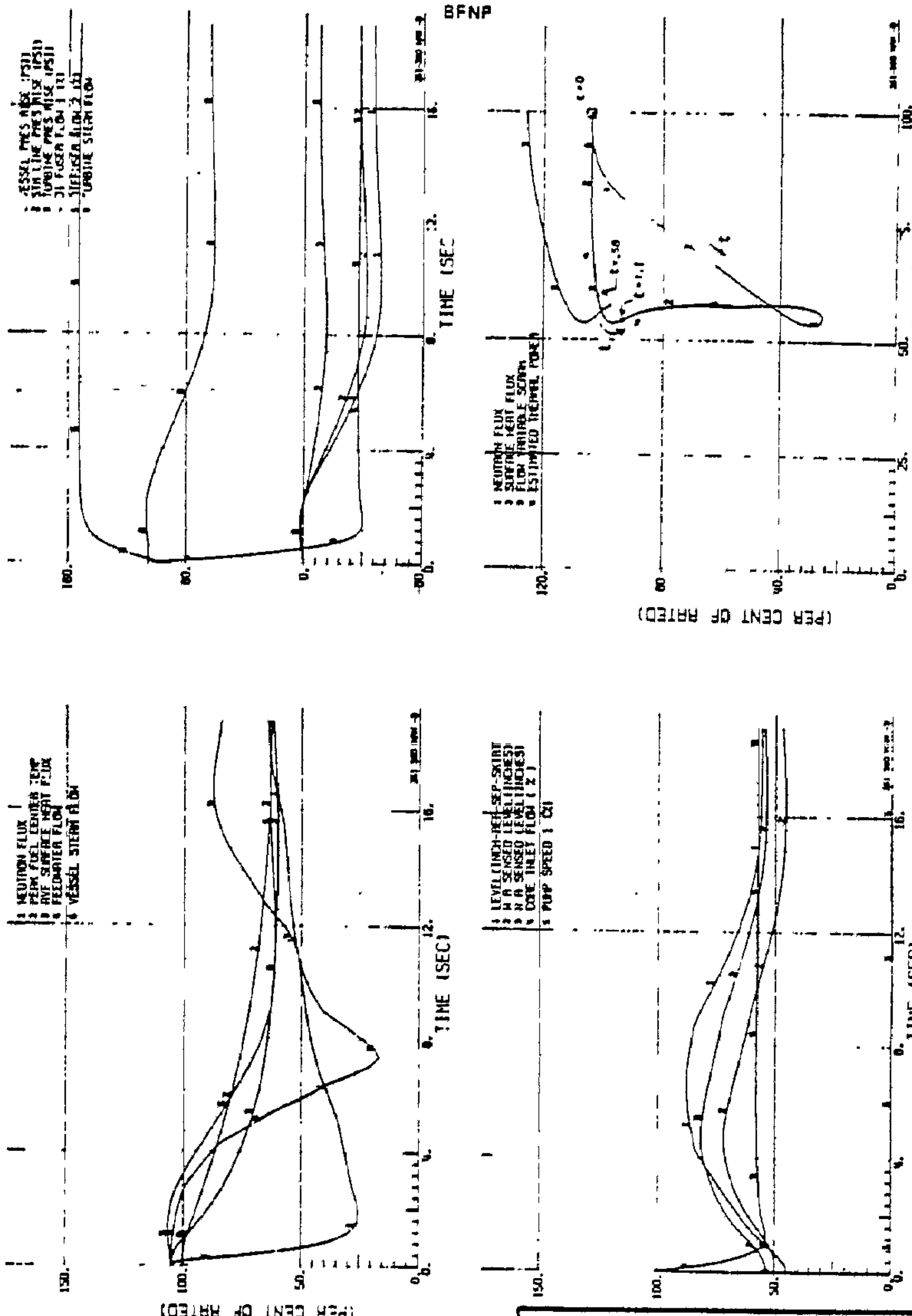


AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

Transient Results,
Trip of Two Recirculation Pump
M-G Set Drive Motors

FIGURE 14 10-15

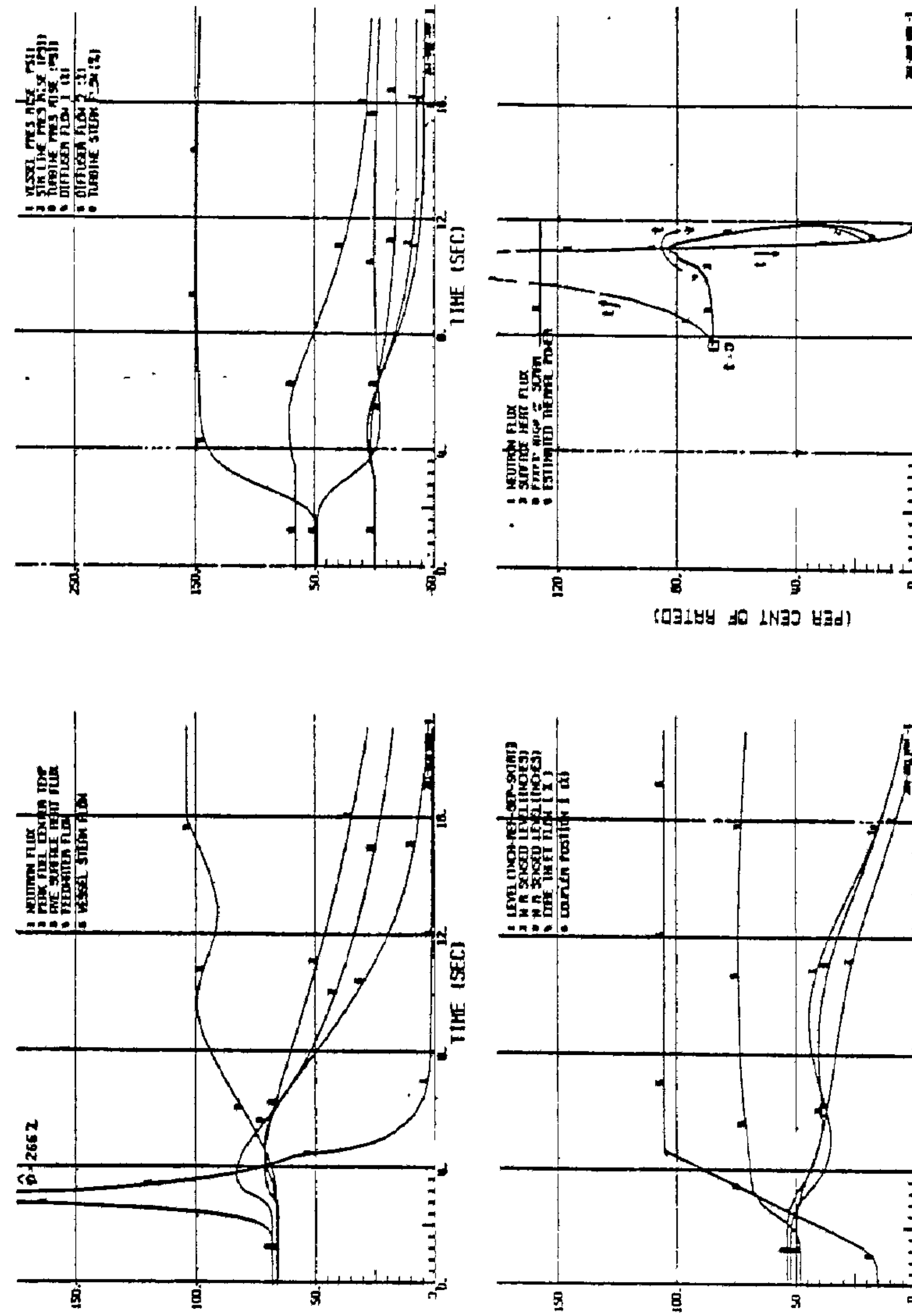


AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

Transient Results,
Recirculation Pump Seizure

FIGURE 14 10-16

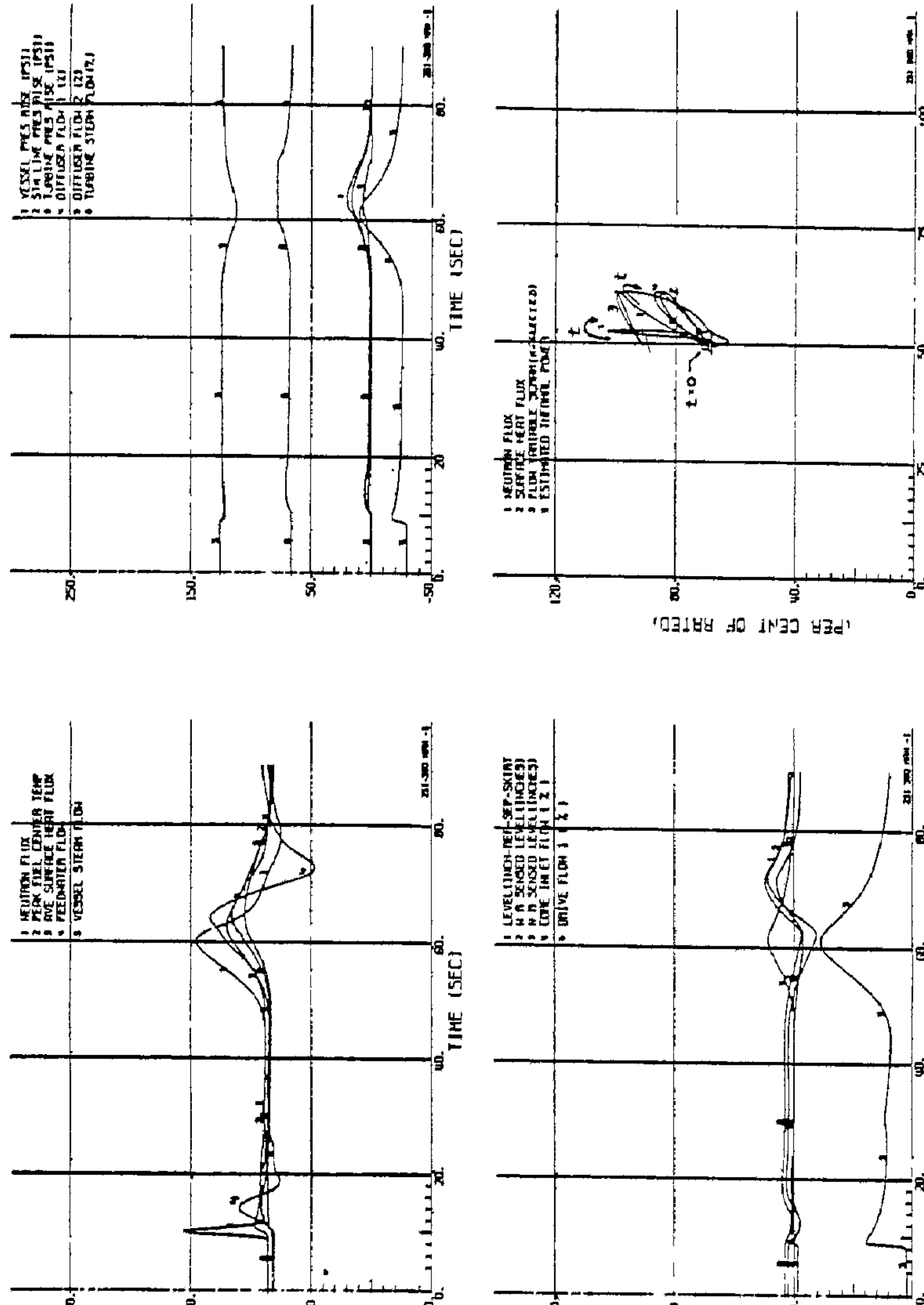


AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

Transient Results,
Recirculation Flow Control Failure
Increasing Flow

FIGURE 14.10-17

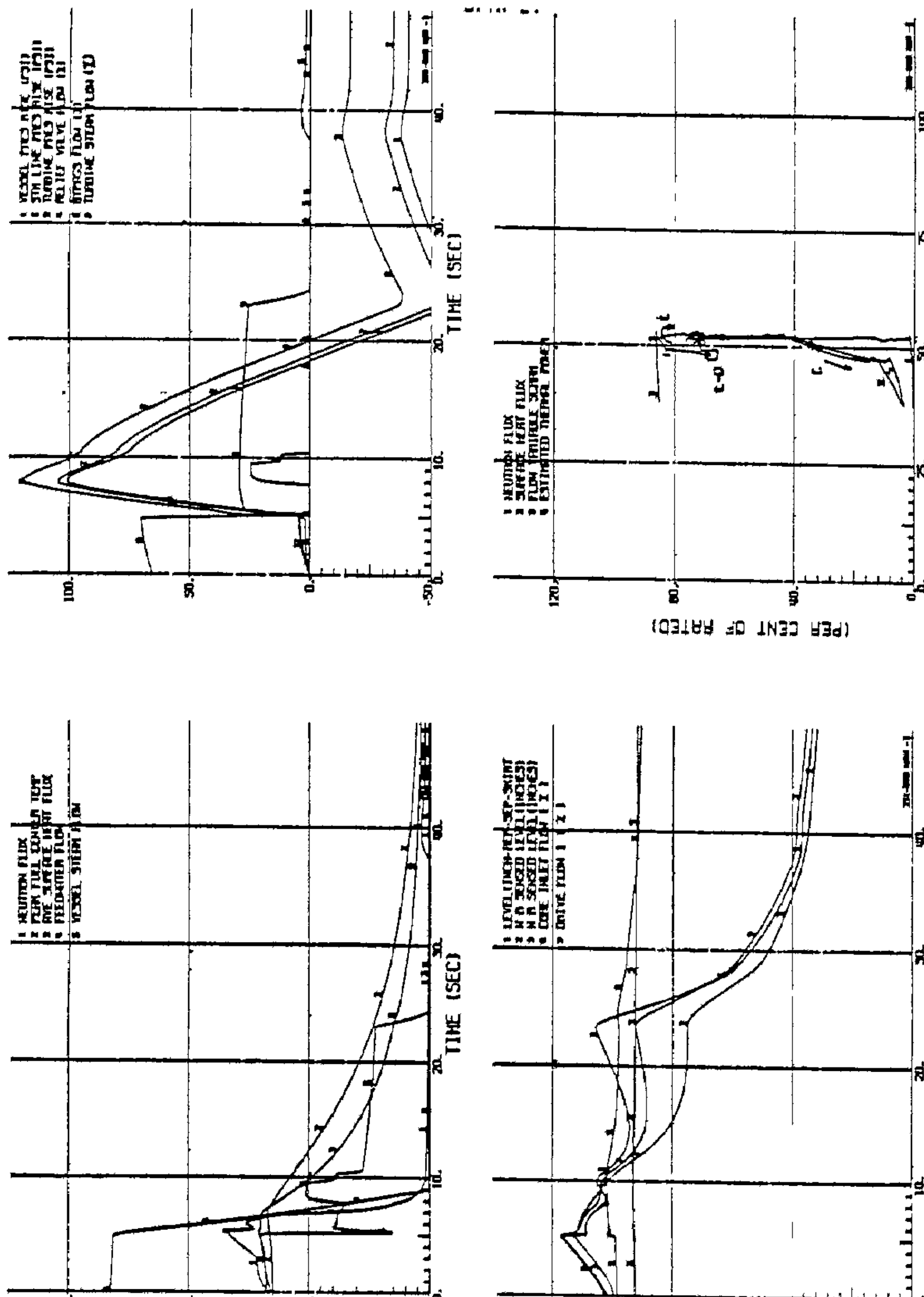


AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

Transient Results,
Startup of Idle Recirculation Pump

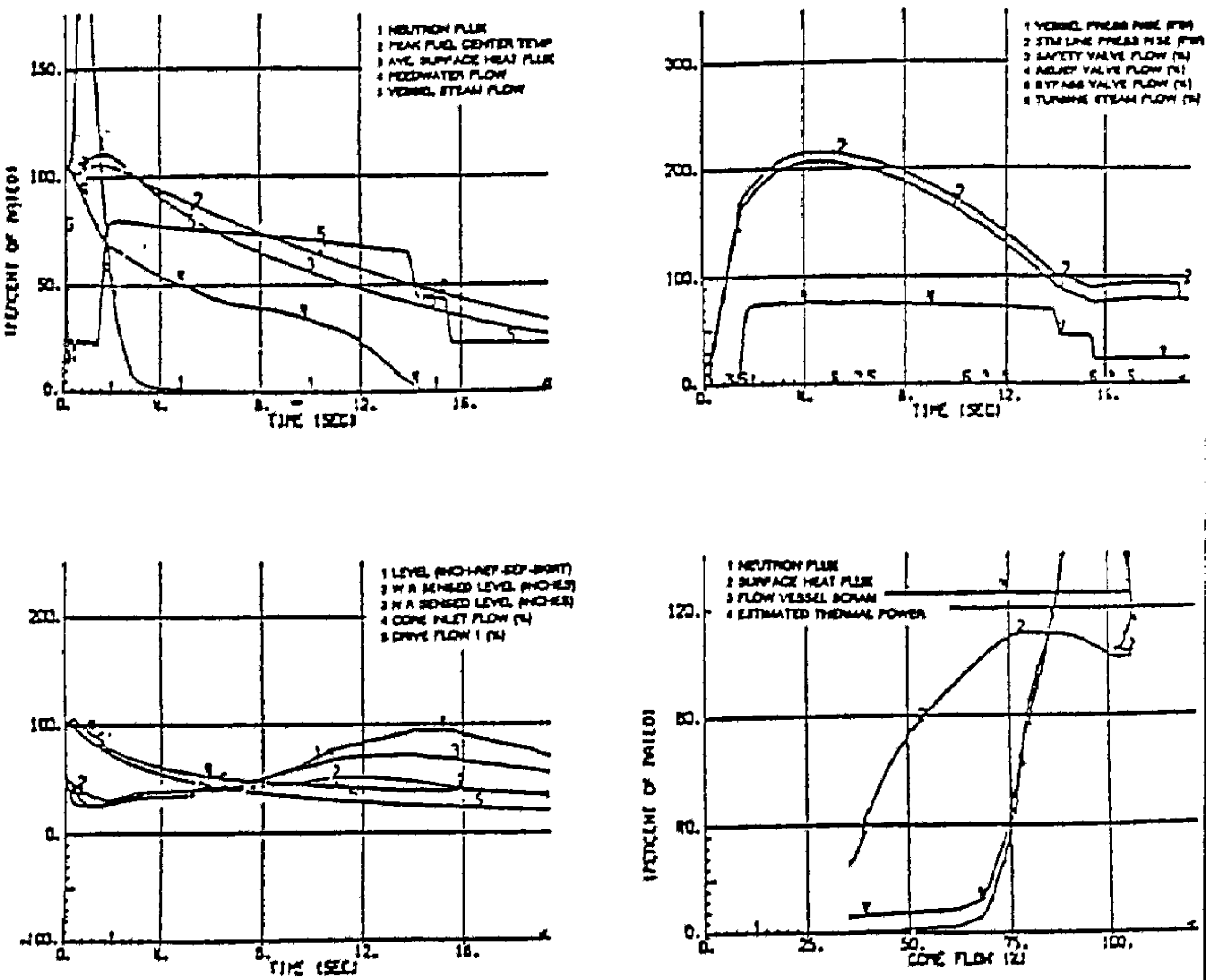
FIGURE 14 10-18



AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

Transient Results,
Feedwater Controller Failure
Maximum Demand
FIGURE 14.10-19



AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT

FINAL SAFETY ANALYSIS REPORT

Browns Ferry Nuclear Plant Generator Load Rejection Without Bypass, EOC2, RPT

FIGURE 14.10-20

14.11 ANALYSIS OF DESIGN BASIS ACCIDENTS - PRE-UPRATED

This section does not reflect the effects from power uprate. For power uprated conditions, the results of the evaluation at 3458 MWt are provided in Section 14.6.

14.11.1 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 into the fuel pool 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.

14.11.2 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 accident is analyzed over the full spectrum of power conditions. Nuclear excursion results are presented for three points in this range: the cold (68°F) critical condition for moderator and fuel, a hot (547°F) critical condition, and the 10 percent of rated power condition. The results of the rod drop accident initiated from higher than 10 percent power are less severe than the 10 percent power case because of the faster doppler response. Only the radiological results of the most severe case are presented.

Subsections 14.11.2.1 through 14.11.2.7 discuss the CRDA and the analysis performed for the initial core loading. It is retained in the FSAR because the information presented is useful to understanding the Control Rod Drop accident and the related licensing bases for the initial core and cycle. A complete and detailed discussion of the CRDA including accident description, causes, sequence of events, consequences of the accident, and the accident analysis (analysis methods, assumptions, conditions, results and consequences) using refined analytical methods is given in the licensing topical report, "GESTAR II," NEDE - 24011-P-A, May 1986 and revisions thereto. Subsection 14.11.2.8 discusses the current CRDA performed for BFN. This analysis documents the safety design basis for eliminating the Main Steam Line Radiation Monitors as required components to mitigate a CRDA.

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. For reload cycles in which the BPWS is utilized a cycle specific CRDA analysis is not required. The cycle specific CRDA results or a commitment to employ BPWS are contained in the Reload Licensing Report.

14.11.2.1 Initial Conditions

The following initial conditions were assumed for the three cases presented in the initial CRDA analysis:

Case A (cold):	Reactor critical Moderator and fuel at 68°F Power level 10^{-8} x design Rod worth (for dropped rod) 0.025 Δk.
Case B (hot):	Reactor critical Moderator and fuel at 547°F Power level 10-6 x design Rod worth (for dropped rod) 0.025 Δk.
Case C (power)	Reactor critical Moderator and fuel at 547°F Power level 10-1 x design Rod worth (for dropped rod) 0.038 Δk.

In considering the possibilities of a control rod drop accident, only the rod worths of the lower curve of Figure 14.11-1 are pertinent at less than ten percent power. These are the rods which are normally allowed to be moved by operating procedures and the rod worth minimizer. The non scheduled rods, those described by the central envelope, do not have a withdrawal permissive during the time their worths are greater than the lower curve, so they are held full in by the control rod drive and cannot drop from the core. If a nonscheduled rod were selected, the rod worth minimizer blocks rod movement. Therefore, the worth of the strongest rod which could be stuck is limited to about 0.01 Δk, and the 0.025 Δk worth assumed for cases A and B is considerably above the rod worth values available for stuck rods under the assumed reactor conditions. In the greater than ten percent power range, the maximum rod worth is determined by the FLARE[1] and WANDA[2] computer codes and is shown in Figure 14.11-2. Thus, in case C the rod worth is assumed to be 0.038 Δk.

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[1] Delp, D. L., et al.: "FLARE-A Three Dimensional Boiling Water Reactor Simulator," GEAP-4598, July 1964.

[2] Marlowe, O. C., and Suggs, M. C.: "WANDA-5-A One Dimensional Neutron Diffusion Equation Program for the Philco 2000 Computer," WAPD-TM-241, November, 1960.

14.11.2.2. Excursion Analysis Assumptions

The following assumptions are used in the analysis of the nuclear excursion for each case:

a. The velocity at which the control rod falls out of the core is assumed to be 5 ft/sec. The control rod velocity limiter[3] an engineered safeguard, limits the rod drop velocity to less than this value.

b. Control rod scram motion is assumed to start at about 200 milliseconds after the neutron flux has attained 120 percent of rated flux. This assumption allows the power transient to be terminated initially by the Doppler reactivity effect of the fuel. This assumption is particularly conservative for cases A and B because a high neutron flux scram would be initiated earlier by the intermediate range neutron monitoring channels (IRM).

c. No credit is taken for the negative reactivity effect resulting from the increased temperature of, or void formation in the moderator because the time constant for heat transfer between the fuel and the moderator is long compared with the time required for control rod motion.

d. No credit is taken for the prompt negative reactivity effect of heating in the moderator due to gamma heating and neutron thermalization.

14.11.2.3 Fuel Damage

Fuel rod damage estimates (initial core) were based upon the UO_2 vapor pressure data of Ackerman[4] 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/gm. Two especially applicable sets of data come from the PULSTAR[5] and ANL-TREAT[6] 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/gm with a minimum period of 2.83 msec. The coolant flow was by natural convection. Film boiling occurred and there were local clad bulges; however, fuel pin integrity was maintained and there were no abnormal pressure rises.

---

[3] "Control Rod Velocity Limiter," General Electric Company, Atomic Power Equipment Department, March 1967 (APED-5446).

[4] 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.

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

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

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

	Test 1	Test 2
Input Energy (cal/gm)	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/gm is the threshold for eventual fuel cladding perforation. Fuel melting is estimated to occur in the 220 to 280 cal/gm range and a minimum of 425 cal/gm is required to cause immediate rupture of the fuel rods due to UO_2 vapor pressures.

A parametric analysis was made of the rod drop accident for various starting conditions and rod worths. The results are shown in Figures 14.11-3 and 14.11-4, and the reduction in final peak fuel enthalpy with increasing initial power level is clearly shown. The cold critical case (case A) is shown as point A on Figure 14.11-3, and the hot standby critical case (case B) is shown as point B on Figure 14.11-4. Figure 14.11-5 is a conservative description of the consequences when the core is at rated temperature and the coolant is boiling. Here the ten percent of power case (case C) is represented by point C. In these cases the maximum initial enthalpy generally is not in the fuel which experiences the greatest enthalpy addition during the excursion. If a rod were dropped from a high initial enthalpy region, the results would not be as great as with one dropped from a lower enthalpy region. However, for conservatism, it is assumed that the peak enthalpy increment is added to the maximum fuel enthalpy that existed in the vicinity of the excursion center prior to the accident.

In the hot standby critical case, the power transient is calculated to have a total energy generation of 4000 MW-seconds (approximately 1.2 full power seconds). The excursion energy is calculated to be distributed in the fuel such that about 330 fuel rods have enthalpies greater than 170 cal/gm. The maximum UO_2 enthalpy is calculated to be 220 cal/gm. Essentially no fuel will melt because fuel melting occurs in the range from 220 to 280 cal/gm.

The power transient in the ten percent of power rod drop accident is less severe than the one at hot standby. The peak enthalpy is about 200 cal/gm and only about 50 fuel rods have enthalpies exceeding 170 cal/gm.

The power transient in the cold condition rod drop accident is calculated to be distributed in the fuel such that about 200 fuel rods have enthalpies greater than 170 cal/gm. The maximum UO_2 enthalpy is calculated to be 250 cal/gm. Approximately 50 pounds of UO_2 have enthalpies in excess of 220 cal/gm. Because fewer fuel rods are perforated and because the shutdown cooling system would be operating, allowing no radioactivity release to the main condenser, the radiological results of the cold rod drop accident are insignificant when compared to the hot standby critical case.

All of these peak enthalpies are far below 425 cal/gm which is estimated to be the threshold for immediate rupture of fuel rods due to UO_2 vapor pressure. Furthermore, the above peak enthalpies are well below the design limit of 280 cal/gm. Thus, there are no damaging pressure pulses as a result of the rod drop accident; and the only damage expected would be the failed fuel rods.

14.11.2.4 Fission Product Release From Fuel

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

a. Three hundred thirty fuel rods fail. This is the largest number of failed fuel rods resulting from the analysis of the rod drop accident over the full spectrum of power conditions.

b. The reactor has been operating at design power until 30 minutes before accident initiation. When translated into actual plant operations, this assumption means that the reactor was shut down from design power, taken critical, and brought to the initial temperature conditions within 30 minutes of the departure from design power. The 30-minute time represents a conservative estimate of the shortest period in which the required plant changes could be accomplished and defines the decay time to be applied to the fission product inventory for the calculation.

c. The reactor has been operating at design power for 1,000 days prior to the accident. This assumption results in equilibrium concentration of fission products in the fuel. Longer operating histories do not increase the concentration of longer lived fission products significantly.

d. An average of 1.8 percent of the noble gas activity and 0.32 percent of the halogen activity in a perforated fuel rod are assumed released. These release percentages are consistent with actual measurements made on defective fuel experiments. The basis for these values is presented in APED-5756.[7]

e. The following fission product concentrations in all fuel rod plenums are applicable for the core at the time the accident occurs:

Noble gases	4.5×10^8 Ci
Halogens	8.3×10^8 Ci

These concentrations are the result of a nuclear analysis of the fuel assuming operation at design power for 1,000 days followed by a 30-minute decay period.

f. None of the solid fission products is released from the fuel. Because the fraction of solid fission product activity available for release from the fuel is negligible, this assumption is reasonable.

g. The fission products produced during the nuclear excursion are neglected. The excursion is of such short duration that the fission products generated are negligible in comparison with the concentration of fission products already assumed present in the fuel.

Using the above assumptions, the following amounts of fission product activity are released from the failed fuel rods to the reactor coolant:

Noble gases (Xe,Kr)	7.1×10^4 Ci
Halogens (Br,I)	2.4×10^4 Ci

14.11.2.5 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. The recirculation flow rate is 25 percent of rated, and the steam flow to the condenser is five percent of rated. The 25 percent recirculation flow and five percent steam flow are the maximum flow rates expected when the reactor is being taken to power and the main condenser is still being evacuated by the

[7] Horton, N. R., Williams, W. A., And Holtzclaw, J. W. "Analytical Methods for Evaluating the Radiological Aspects of the General Electric Boiling Water Reactor," General Electric Company, Atomic Power Equipment Department, March 1969, (APED-5756).

mechanical vacuum pump. The recirculation flow rate is used in determining the volume of coolant in which the activity released from the fuel is deposited. The five percent steam flow rate is greater than that which would be in effect at the reactor power level assumed in the initial conditions for the accident. This assumption is conservative because it results in the transport of more fission products through the steam lines than would actually be expected. Because of the relatively long fuel-to-coolant heat transfer time constant, steam flow is not significantly affected by the increased core heat generation within the time required for the main steam isolation valves to achieve full closure.

b. The main steam isolation valves are assumed to receive an automatic closure signal 0.5 seconds after the radiation monitors are tripped and to be fully closed at 10 seconds from the receipt of the closure signal. The automatic closure signal originates from the main steam line radiation monitors. The 10-second closure time of the main steam isolation valves is the maximum closing time permitted by valve setting. The total time required to isolate the main steam lines (10.5 seconds), combined with the assumptions in "a", allows calculation of the total amount of fission product activity transported to the condenser before the steam lines are isolated.

c. All of the noble gas activity is assumed released to the steam space of the reactor vessel. None is retained in the liquid reactor coolant.

d. The ratio of the halogen concentration in steam to that of water is assumed to be 3×10^{-5} by volume. Measurements, taken under applicable chemical and physical conditions at Dresden Nuclear Power Station Unit No. 1, indicate that the steam-to-water halogen concentration ratio is in the range of 1×10^{-5} to 3×10^{-5}.

e. Water carryover in the main steam lines is assumed to be 0.1 percent of the total mass of steam transferred to the condenser. Measurements of the steam separation effected by the same types of separators used in this reactor vessel show that water carryover is less than 0.1 percent even at rated steam flow. The carryover fraction permits computation of the halogen activity carried to the main condenser in the water entrained in the steam.

f. None of the fission products released from the fuel is assumed to plate out.

The main steam line radiation monitors initiate closure of all main steam isolation valves when a preestablished radiation level is exceeded. This action prevents further transport of the fission products to the condenser. Using the listed assumptions, the following amounts of fission product activities are carried to the condenser:

Noble gases	8.0×10^{3} Ci
Iodine 131	4.2×10^{-1} Ci
Iodine 132	6.4×10^{-2} Ci
Iodine 133	2.2×10^{-1} Ci
Iodine 134	5.5×10^{-2} Ci
Iodine 135	1.4×10^{-1} Ci

14.11.2.6 Fission Product Release to Environs

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

a. The accident is assumed to occur while condenser vacuum is being maintained with the mechanical vacuum pump. 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.

b. All of the noble gas activity transferred to the condenser during the assumed 10.5 second isolation valve closure time is assumed to be airborne in the condenser. The halogen activity transferred to the condenser experiences the removal effects of the condensate and forms an equilibrium condition between the condensate and the vapor volume.

c. The rate at which the condenser activity is discharged to the environment is dependent upon the free volume of the turbine and condenser, the volume of liquid in the condenser, and the discharge rate of the mechanical vacuum pump. The numerical values appropriate to these parameters are 208,000 ft^3 turbine plus condenser free volume, 12,500 ft^3 condenser liquid, and 1,800 cfm mechanical vacuum pump discharge rate.

d. If the mechanical vacuum pump is isolated, the activity released will be contained within the condenser. Due to the condenser air being at lower pressure than its surroundings, the only leakage, if any, would be inward. Therefore, no activity would be transported to the environs.

Based upon these conditions, the fission product release rate to the environment is shown in Table 14.11-l.

14.11.2.7 Radiological Effects

The radiological exposure resulting from the activity discharged to the environment (initial core) was determined for six meterological conditions. These conditions range from very stable to unstable and consider wind speeds of 1 meter/sec and 5 meters/sec. Table 14.11-2 shows that the maximum offsite exposure occurs at the site boundary, which is approximately 0.9 miles from the release point. The maximum radiological exposures at the site boundary are 1 x 10^{-3} rem thyroid and 1.2 x 10^{-2} rem whole body, which are well below the respective thyroid and whole body gamma reference values of 300 rem and 25 rem, respectively, set forth in 10 CFR 100. Due to the large flow rate of the mechanical vacuum pump, essentially all of the activity is exhausted to the environment in the 24-hour release period investigated. Therefore, a 30-day dose would be essentially equivalent to the dose obtained for the 24-hour period. NEDE-24011-P-A-9-US describes how the radiological effects of a CRDA have been affected by the change from 7 x 7 fuel to 8 x 8 fuel. The radiological effects are still orders of magnitude below those set forth in l0 CFR l00.

14.11.2.8 Elimination of the Main Steam Line High Radiation Signal

Upon detection of high radiation in the main steam lines, the main steam line radiation monitors (MSLRMs) originally performed the following actions: 1) scram the reactor, 2) close the main steam isolation valves, 3) close the Main Steam Line (MSL) drain isolation valves, 4) isolate the reactor coolant sample lines and 5) isolate and trip the condenser mechanical vacuum pump (MVP).

General Electric (GE) Licensing Topical Report, NEDO-31400A, October 1992, presents a generic bounding safety analysis which supports the removal of the automatic MSIV closure, MSL drain line isolation valve closure and the automatic reactor shutdown functions of the Main Steam Line Radiation Monitor (MSLRM). BFN has performed additional analyses as described below to justify eliminating the remaining trip/isolation functions of the MSLRMs. Eliminating the MSLRM automatic functions will reduce the potential for unnecessary reactor shutdowns and increase the plant operational flexibility. Following the elimination of these functions, the calculated radiological release consequences of the CRDA will not exceed the acceptable dose limits as specified in 10 CFR 100 and Standard Review Plan (SRP) 15.4.9, "Radiological Consequences of Control Rod Drop Accident (BWR)."

14.11.2.8.1 Summary of Revised Rod Drop Accident Analysis

The NEDO-31400A analyzes the consequences of the CRDA by assuming the CRDA source term carried away by the reactor steam is instantaneously deposited in the turbine and condenser and either leaks out of the condenser into the turbine building and then enters the environment or is released to the environment through the steam jet air ejector (SJAE) off-gas system via the plant stack. The bounding analysis is performed without the MSLRM automatic reactor shutdown and MSIV closure functions. The CRDA source term and radiological transport from the condenser are consistent with the standard approach outlined in the SRP 15.4.9. The NEDO-31400A generic results are presented in graphs such that site specific atmospheric dispersion coefficient (X/Q) values and offgas holdup times can be applied to determine the resulting CRDA doses.

Two additional analyses have been performed to demonstrate that the MVP trip/isolation function and the reactor coolant sample line isolation are not required in order to limit the consequences of an CRDA within acceptable values. Without the MVP trip/isolation, the CRDA source term in the condenser was assumed to be exhausted from the condenser at the MVP flow rate and released to the atmosphere via the plant stack. This release path goes directly to the stack with no holdup or filtering. Without the reactor coolant sample line isolation, the analysis assumes reactor coolant containing CRDA iodine source term is released into the secondary containment and is released to the environment via the standby gas treatment system (SGTS) to the plant stack. Fission products exiting the secondary containment prior to SGTS initiation are considered.

The results of these analyses are considered acceptable if the resulting doses are well within the 10 CFR 100 limits for offsite doses. SRP 15.4.9 defines "well within" as being below 25 percent of the 10 CFR 100 limits.

14.11.2.8.2 Application of NEDO-31400A to BFN

NEDO-31400A explicitly discusses the elimination of the automatic reactor shutdown and MSIV closure functions of the MSLRM. Since the MSL drain discharges to the condenser (as do the main steam lines), the NEDO analysis is also applicable to and bounds the elimination of the MSL drain isolation function. In order to apply the generic NEDO-31400A analysis to BFN, it must be demonstrated that the assumptions made and analysis performed bound those of BFN for a CRDA. The following is a comparison of the key input parameters used in the BFN CRDA analysis to demonstrate NEDO-31400A applicability.

Parameters	BFN	NEDO-31400A
Power	0.109 MW/Rod* (105%)	0.12 MW/Rod (105%)
Failed Fuel Rods	850 (NEDO 31400A and GESTAR II, NEDE-24011-P-A)	850
Operation	1000 days (FSAR Section 14.11.2.4(e))	Long Term
Releases from Fuel (non melted) (melted)	10% Noble 10% Iodine 100% Noble 50% Iodine	10% Noble 10% Iodine 100% Noble 50% Iodine
X/Q Ground (EAB)	$1.22x10^{-4}$ sec/m^3 (FSAR Table 14.11-8)	$2.5x10^{-3}$ sec/m^3
X/Q Fumigation (EAB)	$2.4x10^{-5}$ sec/m^3	N/A
X/Q Elevated (EAB)	9.70×10^{-7} sec/m^3 (FSAR Table 14.11-8)	$3.0x10^{-4}$ sec/m^3
Holdup (Delay Time)	7.3 days for Xenon, 9.7 hours for Kr (FSAR Section 9.5.4)	Graphs provided for various holdup times

*Calculated as: (3293 x 1.05 x 1.5) MW/ (764 x (64-2)) rods
= 0.109 MW/rod

Utilizing the BFN site specific parameters and the graphs provided in the NEDO-31400A analysis, the resulting BFN Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) doses for the condenser 1% per day leakage and off gas system release paths following a CRDA are well below the 10 CFR 100 limits and SRP 15.4.9 guidelines.

14.11.2.8.3 Elimination of Additional Main Steam Line Radiation Monitor Functions

The BFN MSLRM also isolates and trips the condenser mechanical vacuum pump (MVP) and isolates the reactor coolant sample lines. The NEDO-31400A assumes the MVP trips and is isolated such that the CRDA source term either leaks out of the condenser at 1% per day or is processed through the off-gas system with holdup volumes and charcoal filters. However, the MVP trip and isolation on high MSL radiation is not a safety-related function. This release path was not analyzed with the CRDA condenser source term in the NEDO or in the original BFN CRDA analysis. Therefore, the offsite dose impact with the CRDA source term being exhausted from the condenser via the MVP has been analyzed. The CRDA source term in the condenser as described in the NEDO was exhausted to the environment via the plant stack at the MVP flow rate of 1850 cfm. The flow in the stack was split between the base and top of the stack and the atmospheric dispersion coefficients were used as discussed in Section 14.11.3.6.f, g, and h. The resulting EAB and LPZ doses from the MVP release path are well below the 10 CFR 100 limits and the SRP 15.4.9 guidelines.

The release due to the elimination of the reactor water sample line isolation is another potential path that was not analyzed by the generic NEDO-31400A. This 3/4" sample line is connected to the discharge of a reactor recirculation pump but is normally isolated by its primary containment isolation valves. The line is used as an alternate means of obtaining samples for continuous conductivity monitoring of the reactor coolant. Thus, this line is normally closed unless the normal sample path from the RWCU demineralizers is out of service. The recirculation sample station is protected from overpressurization by pressure control valves, sample coolers, and relief valves (relief valves are in Unit 3 only). However, these overpressurization protection devices are not safety related. The samples are discharged directly to Radwaste.

If this alternate sample path is in operation at the time of a CRDA and the nonsafety-related overpressurization protection devices failed, the result could potentially overpressurize the instruments and produce a continuous blowdown of reactor coolant into the reactor building. This scenario is essentially a small break LOCA outside containment which is releasing CRDA source term directly to the secondary containment. Since the break is from a subcooled section of reactor coolant piping which is well below the reactor vessel normal water line, the analysis assumes that only CRDA source term iodines (i.e., no noble gases) exit through the break. This analysis demonstrates that this unlikely blowdown release to the reactor building would initiate isolation of secondary containment and start the standby gas treatment system. Fission products exiting the secondary containment prior to SGTS initiation have been considered and are treated as a ground level release. The fission products removed via SGTS were exhausted to the environment via the plant stack as per the assumptions in Section 14.11.3.6.b, c, e, f, g, and h. The resulting EAB and LPZ doses from the reactor coolant sample line release path are well below the 10 CFR 100 limits and the SRP 15.4.9 guidelines.

14.11.2.8.4 Radiological Effects of Eliminating the MSLRM Functions

The BFN analysis for the CRDA without MSLRM automatic reactor shutdown and isolation functions now consists of four potential release paths; condenser leakage at 1% per day into the turbine building or through SJAE and off-gas system as analyzed by the NEDO-31400A, and the MVP discharge and recirculation sample line discharge as analyzed in accordance with SRP 15.4.9. The “worst-case” radiological exposure resulting from the activity discharged from a CRDA and a SRP 15.4.9 source term would be from the MVP and recirculation sample line release paths combined. The combination of these paths maximizes the CRDA source term released and could occur simultaneously. The resulting combined EAB and LPZ doses from the MVP and reactor coolant sample line are well below the SRP 15.4.9 reference values of 75 REM thyroid and 6 REM whole body.

Based on the analyses above, the MVP, recirculation sample line, and MSL drain release paths have been analyzed and their isolation on a MSL high radiation signal is not required to mitigate the consequences of a CRDA. Units 2 and 3 have physically disconnected the MSLRM functions for automatic reactor shutdown, MSIV closure, MSL drain isolation, and recirculation sample line isolation but has retained the MSLRM function for MVP trip and isolation as an additional nonsafety- related preventative means of reducing the consequences of a CRDA. Unit 1 has not physically disconnected these functions.

14.11.3 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.

Subsection 14.11.3 presents information on the analytical models used to analyze the LOCA for the initial operating cycle including the results of the analyses. This description is applicable only to the initial operating cycle but is generally applicable to analytical LOCA work performed for subsequent cycles. Additional information on

LOCA models currently in use is given in NEDO-20566[8] and NEDC-32484P.[9] Detailed 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 NEDO-10320[10] and NEDC-32484P.[11]

14.11.3.1 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 Core Standby Cooling Systems to become operational.

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.

14.11.3.2 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, the heatup of the fuel, and the perforation of fuel rods. Analysis shows that a maximum of 9.0 percent of the fuel rods reach the pressure and temperature conditions necessary for perforation.

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[8] General Electric Company Analytical Model for Loss-of Coolant Analysis in Accordance with 10CFR50 Appendix K, NEDO-20566.

[9] General Electric SAFER/GESTR-LOCA, Loss of Coolant Analysis, Browns Ferry Units 1, 2, and 3, NEDC-32484P.

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

[11] General Electric SAFER/GESTR-LOCA, Loss of Coolant Analysis, Browns Ferry Units 1, 2, and 3, NEDC-32484P.

14.11.3.3 Primary Containment Response

14.11.3.3.1 Initial Conditions and Assumptions

The following assumptions and initial conditions were used in calculating the effects of a loss of coolant accident on the primary containment. (These assumptions are in addition to those specified for the loss of coolant accident described in paragraph 14.11.3.1.)

a. The reactor is assumed to be operating at the maximum possible steady-state power level and pressure at the time the accident occurs. This maximizes the reactor pressure during the blowdown which in turn maximizes the blowdown flow rate.

b. The break area through which the reactor coolant can escape to the drywell is maximized by assuming the reactor is operating on one recirculation loop with the equalizer valves open. In this configuration, mass escapes from the reactor vessel via the broken loop as well as from jet pump backflow from the unbroken loop through the equalizer valves to the broken loop. This results in the most severe primary containment pressure transient. For the equalizer line to be open, an interlock requires the reactor to be operating on only one recirculation pump with the idle pump's discharge valve closed. The maximum power level under this condition is approximately 80 percent. It is recognized that this assumption is inconsistent with the assumption regarding initial reactor power but is used to maximize the break area. It is also recognized that this assumption is conservative for Unit 3 since the recirculation ring header has been split into two independent halves and the equalizer valves removed. Removal of the equalizer valves prevents the cross flow from the unbroken loop and thus reduces the break effluent.

c. The reactor is assumed to go subcritical at the time of accident initiation due to void formation in the core region. Scram also occurs in less than one second from receipt of the high drywell pressure and low water level signals, but the difference in shutdown time between zero and one second is negligible.

d. The sensible heat released in cooling the fuel to 545°F (the normal primary system operating temperature) and the core decay heat were included in the reactor vessel depressurization calculation. The rate of energy release was calculated using a conservatively high heat transfer coefficient throughout the depressurization. Because of this assumed high energy release rate the vessel is maintained at near rated pressure about ten seconds. The high vessel pressure increases the calculated flow rates out of the break; this is conservative for containment analysis purposes. With the vessel fluid temperature remaining near 545°F, however, the release of sensible energy stored below 545°F is negligible during the first ten seconds. The later release of this sensible energy does not affect the peak drywell pressure. The small effect of this energy on the end-of-transient pressure suppression pool temperature is included in the calculations.

e. The main steam isolation valves were assumed to start closing at 0.5 seconds after the accident, and the valves were assumed to be fully closed in the shortest possible time of three seconds following closure initiation. Actually, the closures of the main steam isolation valves are expected to be the result of low water level, so these valves may not receive a signal to close for over four seconds, and the closing time could be as high as 10 seconds. By assuming rapid closure of these valves, the reactor vessel is maintained at a high pressure which maximizes the discharge of high energy steam and water into the primary containment.

f. The feedwater flow was assumed to stop instantaneously at time zero. This conservatism is used because the relatively cold feedwater flow, if considered to continue, tends to depressurize the reactor vessel, thereby reducing the discharge of steam and water into the primary containment.

g. The vessel depressurization flow rates were calculated using Moody's critical flow model[12] assuming "liquid only" outflow because this maximizes the energy release to the containment. "Liquid only" outflow means that all vapor formed in the vessel due to bulk flashing rises to the surface rather than being entrained in the exiting flow.

   Some entrainment of the vapor would occur and would significantly reduce the reactor vessel discharge flow rates. Moody's critical flow model, which assumes annular, isentropic flow, thermodynamic phase equilibrium, and maximized slip ratio, accurately predicts vessel outflows through small diameter orifices. However, actual flow rates through larger flow areas are less than the model indicates due to the effects of a near homogeneous two-phase flow pattern and phase nonequilibrium. These effects are in addition to the reduction due to vapor entrainment discussed above.

h. The pressure response of the containment is calculated assuming:

   1. Thermodynamic equilibrium in the drywell and pressure suppression chamber. Because complete mixing is nearly achieved, the error introduced by assuming complete mixing is negligible and in the conservative direction.

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[12] Moody, F. J. "Maximum Flow of a Rate Single Component Two-Phase Mixture," Journal of Heat Transfer ASME Series C, Vol 83, p. 134.

2. The constituents of the fluid flowing in the drywell to pressure suppression chamber vents are based on a homogeneous mixture of the fluid in the drywell. The consequences of this assumption result in complete liquid carryover into the drywell vents. Actually, some of the liquid will remain behind in a pool on the drywell floor so that the calculated drywell pressure is conservatively high.

3. The flow in the drywell pressure suppression pool vents is compressible except for the liquid phase.

4. No heat loss from the gases inside the primary containment is assumed.

i. The initial conditions within the containment assumed for the analysis were:

Drywell	
Pressure, psig	0.75
Temperature, °F	135
Humidity, percent	20
Pressure Suppression Chamber	
Pressure, psig	0.75
Water Temperature, °F	95
Humidity, percent	100

14.11.3.3.2 Containment Response

The calculated pressure and temperature responses of the containment are shown in Figures 14.11-10 and 14.11-11. Figure 14.11-10 shows that the calculated drywell peak pressure is 49.6 psig, which is well below the maximum allowable pressure of 62 psig. After the discharge of the primary coolant from the reactor vessel into the drywell, the temperature of the pressure suppression chamber water approaches 170°F (Figure 14.11-12), and the primary containment pressure stabilizes at about 27 psig, as shown on Figure 14.11-10. Most of the noncondensible gases are forced into the pressure suppression chamber during the vessel depressurization phase. However, the noncondensibles soon redistribute between the drywell and the pressure suppression chamber via the vacuum breaker system as the drywell pressure decreases due to steam condensation. The Core Spray System removes decay heat and stored heat from the core, thereby controlling core heatup and limiting metal-water reaction to less than 0.1 percent. The core spray water transports the core heat out of the reactor vessel through the broken recirculation line in the form of hot water. This hot water flows into the pressure suppression chamber via the drywell-to-pressure suppression chamber vent pipes. Steam flow is negligible. The energy transported to the pressure suppression chamber water is then removed from the primary containment system by the RHRS heat exchangers.

Prior to activation of the RHRS containment cooling mode (arbitrarily assumed at 600 seconds after the accident), the RHRS pumps (LPCI mode) have been adding liquid to the reactor vessel. After the reactor vessel is flooded to the height of the jet pump nozzles, the excess flow discharges through the recirculation line break into the drywell. This flow offers considerable cooling to the drywell and causes a depressurization of the containment as the steam in the drywell is condensed. At 600 seconds, the RHRS pumps are assumed to be switched from the LPCI mode to the containment cooling mode. The containment spray would normally not be activated at all and the changeover to the containment cooling mode need not be made for several hours. There is considerable time available to place the containment cooling system in operation because about eight hours will pass before the maximum allowable pressure is reached with no containment cooling.

To assess the primary containment long term response after the accident, an analysis was made of the effects of various containment spray and containment cooling combinations. For all cases, one of the core spray loops is assumed to be in operation. The long term pressure and temperature response of the primary containment was analyzed for the following RHRS containment cooling mode conditions:

Case A — Operation of both RHRS cooling loops - four RHRS pumps, four service water pumps, and four RHRS heat exchangers - with containment spray.

Case B — Operation of two RHRS cooling loops with one RHRS pump, one service water pump, and one RHRS heat exchanger on each loop - with containment spray.

Case C — Operation of one RHRS cooling loop with two RHRS pumps, two service water pumps, and two RHRS heat exchangers - with containment spray.

The initial pressure response of the containment (the first 30 seconds after break) is the same for each of the above conditions. During the long term containment response (after depressurization of the reactor vessel is complete), the pressure suppression pool is assumed to be the only heat sink in the containment system. The effects of decay energy, stored energy, and energy from the metal-water reaction on the pressure suppression pool temperature are considered.

Case A

This case assumes that both RHRS loops are operating in the containment cooling mode. This includes four RHRS heat exchangers, four RHRS pumps, and four RHR service water pumps. The RHRS pumps draw suction from the pressure suppression pool and pump water through the RHRS heat exchangers and into the drywell as containment spray. This forms a closed cooling loop with the pressure suppression pool. This pressure suppression pool cooling condition is arbitrarily assumed to start at 600 seconds after the accident. Prior to this time the RHRS pumps are used to flood the core (LPCI mode).

The containment pressure response to this set of conditions is shown as curve "a" in Figure 14.11-10. The corresponding drywell and pressure suppression pool temperature responses are shown as curves "a" in Figures 14.11-11 and 14.11-12. After the initial rapid temperature rise in the containment. When the energy removal rate of the RHRS exceeds the energy addition rate from the decay heat, the containment pressure and temperature decrease to their preaccident values. Table 14.11-3 summarizes the cooling equipment operation, the peak containment pressure following the initial blowdown peak, and the peak pressure suppression pool temperature.

Case B

This case assumes that both RHRS loops are operating in the containment cooling mode. However, only one RHR heat exchanger, one RHR pump, and one RHR service water pump on each loop are assumed to be in operation. As in the previous case, the RHRS containment cooling mode is assumed to be activated at 600 seconds after the accident. The containment pressure response to this set of conditions is shown as curve "b" in Figure 14.11-10. The corresponding drywell and pressure suppression pool temperature responses are shown as curves "b" in Figures 14.11-11 and 14.11-12. A summary of this case is shown in Table 14.11-3.

Case C

This case assumes that one RHRS loop is operating in the containment cooling mode. This includes two RHRS heat exchangers, two RHRS pumps, and two RHR service water pumps.

This case represents the most degraded condition of heat removal while in the containment cooling mode. It is assumed that this condition is established at 600 seconds after the accident.

The containment response to this set of conditions is shown as curve "c" in Figure 14.11-10. The corresponding drywell and pressure suppression pool temperatures are shown as curves "c" in Figures 14.11-11 and 14.11-12. A summary of this case is shown in Table 14.11-3. Case C pressure suppression pool responses have been reanalyzed by NEDC-32484P, Revision 2, and GE-NE-B13-01755-2, Revision 2.

14.11.3.3.3 Metal Water Reaction Effects on the Primary Containment

If Zircaloy in the reactor core is heated to temperatures above about 2000°F in the presence of steam, a chemical reaction occurs in which zirconium oxide and hydrogen are formed. This is accompanied with an energy release of about 2800 Btu per pound of zirconium reacted. The energy produced is accommodated in the pressure suppression chamber pool. The hydrogen formed, however, will result in an increased drywell pressure due simply to the added volume of gas to the fixed containment volume. Although very small quantities of hydrogen are produced during the accident, the containment has the inherent ability to accommodate a much larger amount as discussed below.

The basic approach to evaluating the capability of a containment system with a given containment spray design is to assume that the energy and gas are liberated from the reactor vessel over some time period. The rate of energy release over the entire duration of the release is arbitrarily taken as uniform, since the capability curve serves as a capability index only, and is not based on any given set of accident conditions as an accident performance evaluation might be.

It is conservatively assumed that the pressure suppression pool is the only body in the system which is capable of storing energy. The considerable amount of energy storage which would take place in the various structures of the containment is neglected. Hence, as energy is released from the core region, it is absorbed by the pressure suppression pool. Energy is removed from the pool by heat exchangers which reject heat to the service water. Because the energy release is taken as uniform and the service-water temperature and exchanger flow rate are constant, the temperature response of the pool can be determined. It is assumed that the pressure suppression chamber gases are at the pressure suppression chamber water temperature.

The metal-water reaction during core heatup is calculated by the core heat-up mode described in Subsection 14.8. The extent of the metal-water reaction thus calculated is less than 0.1 percent of all the zirconium in the core. As an index of the containment's ability to tolerate postulated metal-water reactions, the concept of "Containment Capability" is used. Since this capability depends on the time domain, the duration over which the metal-water reaction is postulated to occur is one of the parameters used.

Containment capability is defined as the maximum percent of fuel channels and fuel cladding material which can enter into a metal-water reaction during a specified duration without exceeding the maximum allowable pressure of the containment. To evaluate the containment capability, various percentages of metal-water reaction are assumed to take place over certain time period. This analysis presents a method of measuring system capability without requiring prediction of the detailed events in a particular accident condition.

Since the percent metal-water reaction capability varies with the duration of the uniform energy and gas release, the percent metal-water reaction capability is plotted against the duration of release. This constitutes the containment capability curves as shown in Figure 14.11-14. All points below the curves represent a given metal-water reaction and a given duration which will result in a containment peak pressure which is below the maximum allowable pressure. The calculations are made at the end of the energy release duration because the number of moles of gases in the system in then at a maximum, and the pressure suppression pool temperature is higher at this time than at any other time during the energy release.

It should be noted that the curves are actually derived from separate calculations of two conditions: the "steaming" and the "nonsteaming" situation. The minimum amount of metal-water reaction which the containment can tolerate for a given duration is given by the condition where all of the noncondensible gases are stored in the pressure suppression chamber. This condition assumes that "steaming" from the drywell to the pressure suppression chamber results in washing all of the noncondensible gases into the pressure suppression chamber. This is shown as the flat portion of the containment capability characteristic curve. Activation of containment sprays condense the drywell steam so that no steaming occurs, thus allowing noncondensibles to also be stored in the drywell. This is denoted by the rising (spray) curve. The intersection between the no spray curve and the spray curve represents the duration and metal water reaction energy release which just raises all the spray water to the saturation temperature at the maximum allowable containment pressures.

For durations to the left of the intersection, some steam is generated and all the gases are stored in the pressure suppression chamber. For durations to the right of the intersection, the spray flow is subcooled as it exits from drywell by increasing amounts as the duration is increased.

The energy release rate to the containment is calculated as follows:

$$q_{IN} = \frac{Q_O + Q_{MW} + Q_S}{T_D}$$

where:

q_{IN} = Arbitrary energy release rate to the containment Btu per sec,

Q_O = Integral of decay power over selected duration of energy gas release, Btu,

Q_{MW} = Total chemical energy released exothermically from selected metal-water reaction, Btu,

Q_S = Initial internal sensible energy of core fuel and cladding, Btu. and

T_D = Selected duration of energy and gas release, seconds.

The total chemical energy released from the metal-water reaction is proportional to the percent metal-water reaction. The initial internal sensible energy of the core is taken as the difference between the energy in the core after the blowdown and the energy in the core at a datum temperature of 250°F.

The temperature of the drywell gas is found by considering an energy balance on the spray flows through the drywell as described in Subsection 14.8.

Based upon the drywell gas temperature, pressure suppression chamber gas temperature and the total number of moles in the system, as calculated above, the containment pressure is determined. The containment capability curves in Figure 14.11-14 present the results of the parametric investigation.

14.11.3.4 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 TID 14844 methodology. These source terms are generally comparable to those based on the methodology utilized by the ORIGEN Code.

b. The reactor has been operating at design power (3458 MWt) for 1,000 days prior to the accident. This is appropriate for irradiation times up to 1400 days as noted by calculations performed utilizing the ORIGEN Code.

c. One hundred percent of the equilibrium radioactive noble gas inventory developed as a result of such operation is released.

d. Twenty-five percent of the equilibrium radioactive iodine inventory developed as a result of such operation is released. Of this 25 percent, 91 percent is assumed to be elemental iodine, 5 percent in particulate form, and 4 percent in the form of organic iodides. Table 14.11-4 gives the inventory of each isotope in the primary containment available for leakage.

14.11.3.5 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. The following assumptions were used in calculating the amounts of fission products released from the primary containment:

a. The primary containment free volume is 283,000 ft^3.

b. The primary to secondary containment leak rate was taken as two percent volume per day (235 cfh).

14.11.3.6 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 leakage from primary containment to secondary containment mixes instantaneously and uniformly within the secondary containment.

b. The effective volume of the secondary containment is 50 percent of the total free volume of a single reactor zone plus 50 percent of the refueling zone. The resulting effective secondary containment volume is 1,931,502 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.

During this short time period, a very small amount of secondary containment atmosphere (~35 ft^3) will be released directly to the environment unfiltered from the Reactor Building. 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 22,000 cfm), all releases to the environment from secondary containment are through the SGTS filters via the plant stack.

d. The Containment Atmospheric Dilution (CAD) system 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. Filter efficiency for the SGTS was taken as 90 percent for organic and inorganic (elemental) iodine.

f. Release to the environment from the plant stack is composed of two flow paths. A continuous ground level release of 10 cfm occurs at the base of the stack. This flow results from leakage through the backdraft dampers in the base of the stack. Subsection 5.3.3, "Secondary Containment System Description" describes the backdraft dampers. The 10 cfm leakage mixes uniformly within the rooms at the base of the stack (34,560 ft^3). The remaining SGTS flow exits the stack at a height of 183 meters above ground elevation.

g. Fumigation conditions exist for the first 30 minutes post accident.

h. 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 14.11-8.

14.11.3.7 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 2-hour whole body gamma dose, beta dose and the thyroid inhalation dose at the site boundary (1,465 meters), and the corresponding 30-day doses at the low population zone (LPZ) boundary 3,200 meters).

The offsite doses are calculated using a combination of the STP and FENCEDOSE computer programs. The STP program models the fission product transport from the primary containment to release to the environment. The model accounts for fission product decay, flow rates, filter absorption, dilution, release rates and release points. The FENCEDOSE computer program models the atmospheric dispersion to the offsite receptor points by use of appropriate X/Qs and calculates the gamma, beta, and thyroid doses.

The largest calculated total offsite dose is well within the 10 CFR 100 guideline values.

Control Room

The control room doses are calculated using a combination of the STP and COROD computer programs. The STP program models the fission product transport from the primary containment to release to the environment. The model accounts for fission product decay, flow rates, filter absorption, dilution, release rates, and release points. The COROD computer program accounts for the atmospheric dispersion to the control room intakes by use of appropriate X/Qs and models the control bay habitability zone filtered pressurization flow, unfiltered inleakage, occupancy times, breathing rates and calculates the gamma, beta, and thyroid doses. 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 COROD computer code calculates the gamma dose by a typical point-kernel methodology accounting for the control room geometry. The thyroid dose was reduced by ratioing to the ICRP-30 conversion factors. This resulted in a reduction factor of 1.7 for the dose for the 0 to 30 minute time frame and a factor of 1.35 for times after 30 minutes.

The direct gamma dose contribution from the piping inside secondary containment, the secondary containment atmosphere and the cloud dose 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 pressure suppression chamber water in the event of a LOCA.

All of these exposure mechanisms (filtered pressurization flow, unfiltered inleakage, cloud dose and direct dose) are combined to produce a total control room dose for the duration of the accident. It was determined that the differences between the case with two SGTS fans in operation with a small amount of unfiltered secondary containment release and the case with three SGTS fans in operation with all releases being filtered and via the plant stack are negligible. The 30 day integrated post-accident doses in the control room are within the limits of 5 REM whole body gamma dose, 30 REM beta and 30 REM to the thyroid as specified in 10 CFR 50, Appendix A General Design Criteria 19.

The Committed Effective Dose Equivalent (CEDE) for the thyroid plus the whole body gamma Deep Dose Equivalent (DDE) is below the 5 REM Total Effective Dose Equivalent (TEDE) limit.

14.11.4 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 Subsection 14.11.4.1 applies to the dropping of 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. For the 8x8 fuel design, the activity released due to a fuel handling accident will be less than 84% of the activity released by the original 7x7 fuel design. For the 9x9 fuel design the activity will be less than 83.5% of the activity released by the original 7x7 fuel design. The historical and current calculated doses are much less than the regulatory guidelines.

14.11.4.1 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).

14.11.4.2 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 18,150 ft-lb and 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 where

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

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 18,150 or 14,500 ft-lbs of energy. 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 62 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 54 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 54 x 4 or 54,000 ft-lbs 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:

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

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

Dropped assembly	-	62	rods (bending)
Struck assemblies	-	32	tie rods (bending)
Struck assemblies	-	12	rods (compression)
		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:

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

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 (GE 8x8 fuel design) resulting from the accident is as follows:

First impact	-	106	rods
Second impact	-	19	rods
Third impact	-	0	rods
		125	failed rods (total)

14.11.4.3 Fission Product Release From Fuel

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

a. The reactor fuel has an average irradiation time of 1000 days at design power up to 24 hours prior to the accident. This assumption results in an equilibrium fission product concentration at the time the reactor is shut down. Longer operating histories do not significantly increase the concentration of the fission products of concern. 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 from

$$q(Ci) = \frac{0.865\ x\ 10^6\ x\ f}{n} \times\ P_o \gamma_i \left(1 - e^{-\lambda_i T_o} \right) \times \left(e^{\lambda_i t_D} \right)$$

where

f	=	peaking factor, taken as 1.5
n	=	number of fuel bundles in core (n = 764)
P_o	=	thermal power level (P_o = 3458 MWt)
γ_i	=	fission yield for isotope i
λ_i	=	decay constant of isotope i
T_o	=	residence time in core ($T_o = 8.64 \times 10^7$ sec)
t_D	=	decay time between shutdown and removal of the vessel head (24 hrs)

c. Due to the negligible particulate activity available for release in the fuel plenums or from the unmelted fuel, none of the solid fission products is assumed to be released from the fuel.

d. One hundred twenty-five fuel rods are assumed to fail. This was the conclusion of the analysis of mechanical damage to the fuel based on the GE 8x8 fuel design.

14.11.4.4 Fission Product Release to Secondary Containment

The following assumptions were used to calculate the fission product release to the secondary containment:

a. Fraction of Fuel Rod Inventory Released

Noble Gases (Except Kr 85)	10 percent
Kr 85	30 percent
Iodines	10 percent

b. Iodine Decontamination Factor in Reactor Cavity Pool Water — 100

14.11.4.5 Fission Product Release to Environs

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

a. High radiation levels in the reactor building will isolate the normal ventilation system and actuate the Standby Gas Treatment System. The isolation dampers were assumed to close in 15 seconds.

b. The relative humidity in the secondary containment is 70 percent. Since the refueling accident does not result in the release of any liquid or vapor to the secondary containment, the normal environmental condition existing prior to the accident will also exist after the accident, except for the addition of the released fission products. The relative humidity in the secondary containment will therefore be considerably below any levels which may be detrimental to the filter media in the Standby Gas Treatment System. However, as mentioned previously, the charcoal beds and absolute filter media, as well as the air flowing through the filter system, are heated 5°F above the mixture entering the system, reducing the relative humidity to 70 percent or less.

c.	Standby Gas Treatment System Filter Efficiency	0.90
d.	Height of the Main Stack	183 meters
e.	Distance to Exclusion Area Boundary	1,300 meters
f.	Distance to Low Population Zone	3,200 meters
g.	Mixing Air Volume	4,900 FT^3
h.	Ventilation Air Flow Prior to Damper Isolation	22,000 CFM

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 125 fuel pins (assuming GE 8x8 fuel design). Ten percent of the halogen isotopes inventory plus 10 percent of all noble gases inventory (except Kr 85 which is 30 percent of this inventory) will be released from the fractured fuel rods. An overall effective decontamination factor of 100 is applicable for iodine released at depth under water. The radioactive releases to the air space above the pool are released through the refueling zone ventilation and the Standby Gas Treatment Systems. The assumptions used to evaluate the fuel handling design basis accident event are defined in Nuclear Regulatory Commissions Regulatory Guide 1.25. Further guidance is contained in the standard review plans in NUREG-800, Section 15.7.4.

In order to evaluate the effect of refueling zone ventilation damper closure time, the analysis includes doses from air bypassing the Standby Gas Treatment System.

The bypass is occurring through the Refueling Zone Ventilation System. For this evaluation, it is assumed that the portion of the ventilation system dedicated to the reactor vessel pool and the spent fuel storage pool provides the bypass flow. The gases released from the damaged fuel bundles are assumed to be confined to an air volume bounded by the perimeter of the pool and mixed to a height of no more than 4 feet above the pool. The activity released to the environment before the dampers close is taken from the air volume over the pool expelled through the ventilation system. 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. However, the ventilation flow from the storage pool area is twice the size of the flow from the reactor vessel area. The difference in flows transports more activity to the environment in a given time period. Therefore, for conservatism the number of rods damaged and resulting activity released is based on a fuel handling accident in the reactor cavity, and the mixing volume and ventilation is based on a release over the spent fuel pool.

The bypass flow not only bypasses the SGTS filters, it is also released from a roof vent rather than the main stack. The atmospheric dispersion, X/Q, of releases from the top of the stack is significantly smaller than the atmospheric dispersion factors for the roof vent releases. The result of this change is to make the dose contribution from the roof vent releases more important than if all releases were through the stack. Almost all the dose is from the roof vent release.

The fuel handling accident was evaluated using the STP, FENCEDOSE, and COROD computer programs described in Section 14.11.3.7. The calculations simulate an initial time period without filtration of the releases. Following the initial time period, the releases are filtered. Computations were prepared with an atmospheric dispersion, X/Q, for elevated releases and with X/Q data for ground level releases appropriate for the EAB and LPZ boundaries. The final dose evaluations become the dose contributions from the initial ground level release plus the contribution from the release of the balance of the activity through the stack (base and top).

14.11.4.6 Radiological Effects

The radiological exposures following the refueling accident have been evaluated at the site boundary and at the LPZ boundary. The calculated dose assumes that the bypass activity is exhausted through a roof vent and, after the dampers close, the activity is processed through the SGTS and the plant stack.

Boundary dose resulting from design basis accident events has been judged by comparing the dose to the dose in 10 CFR 100, Reactor Site Criteria. This regulation uses radiation doses of 300 rem to the thyroid and 25 rem whole body as guides for doses to the public under accident conditions. Fuel handling accidents in the past have been judged as having acceptable consequences if the dose is a small part of 10 CFR 100. In the standard review plan, NUREG-800, a small part has been defined as 25 percent. The calculated doses are much less than the guidelines.

14.11.5 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 14.11-15 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 offsite persons.

14.11.5.1 Nuclear System Transient Effects

14.11.5.1.1 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 design power.

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 10.5 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 10-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. In calculating the rate of water level rise inside the vessel, it is assumed that the steam bubbles formed during depressurization rise at an average velocity of about 1 foot per second relative to the liquid. This assumption is predicted by analysis[13] and confirmed experimentally.[14]

i. After the level of the mixture inside the reactor vessel rises to the top of the steam dryers, the quality of the two-phase mixture discharged through the break is assumed constant at its minimum value. This assumption maximizes

[13] Moody, F. J.: "Liquid/Vapor Action In a Vessel During Blowdown" APED-5177, June 1966, Wilson, J.F., et al: "The Velocity of Rising Steam In A Bubbling Two-Phase Mixture," ANS Transation, Vol 5, No. 1, Page 151 (1962).

[14] Ianna, P.W., et al: "Design and Operating Experience Of The ESADA Vallecitos Experimental Superheat Reactor (Eversr)";

the total mass of coolant discharged through the break because most of the mixture flow will actually be at a higher quality.

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

k. 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 according to momentum computations for the recirculation system.

l. Recirculation system drive pump head is assumed to be zero when the coolant at the pump suction reaches 1 percent quality. This assumption accounts for the effects of cavitation on recirculation drive pump capacity as the pumps coast down.

14.11.5.1.2 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.

14.11.5.1.3 Coolant Loss and Reactor Vessel Water Level

The steam flow rate through the downstream side of the break increases from the initial value of 1000 lb/sec in the line to 2000 lb/sec (about 200 percent of rated flow for one steam line) with critical flow initially occurring at the flow restrictor. The steam flow rate was calculated using an ideal nozzle model. Tests conducted on a scale model over a variety of pressure, temperature, and moisture conditions have been used to substantiate the flow models capability to predict the steam flow behavior in the presence of a flow restrictor.

The steam flow rate through the downstream side of the break consists of equal flow components from each of the unbroken lines. The pipe resistance and local restrictions in the unbroken lines result in critical flow initially occurring at the downstream side break location. The steam flow rate in each of the unbroken lines increases from an initial value of 1000 lb/sec to 1530 lb/sec.

The total steam flow rate leaving the vessel is approximately 6600 lb/sec, which is in excess of the steam generation rate of 4000 lb/sec. The steam flow-steam generation mismatch causes an initial depressurization of the reactor vessel at a rate of 35 psi/sec. 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 2 to 3 seconds after the break, as shown in Figure 14.11-16. 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.[15]

The vessel depressurization is calculated using a digital computer code in which the reactor vessel is modeled as five major nodes. The model includes the flow resistance between nodes, as well as heat addition from the core.

As shown in Figure 14.11-16, 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 isolation valve closure:

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

Steam	25,000 pounds (19,874 pounds for dose evaluation)
Liquid	160,000 pounds (43,740 pounds for dose evaluation)

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.

After the main steam isolation valves close, depressurization stops and natural convection is established through the reactor core. 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 core flow and MCHFR during the first 10.5 seconds of the accident are shown in Figures 14.11-17 and 14.11-l8. Since 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.

14.11.5.2 Radioactive Material Release

14.11.5.2.1 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, 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/cc which is 10 times the equilibrium value for continued full power operation allowed by Technical Specifications. Since this value is 10 times the equilibrium value for continued full power operation allowed by Technical Specifications and several orders of magnitude higher than normal reactor coolant concentrations, considerable conservatism is included in the analysis.

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.

14.11.5.2.2 <u>Fission Product Release From Break</u>

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

Noble gases	1.5×10^1 Ci
Iodine 131	1.3×10^2 Ci
Iodine 132	1.1×10^3 Ci
Iodine 133	8.6×10^2 Ci
Iodine 134	1.8×10^3 Ci
Iodine 135	1.2×10^3 Ci

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

14.11.5.2.3 <u>Steam Cloud Movement</u>

The following initial conditions and assumptions are used in calculating the movement of the steam cloud:

a. Additional flashing to steam of the liquid exiting from the steam line break will occur due to its superheated condition in the atmosphere.

b. The pressure buildup inside the turbine building will cause the blowout panels to function, resulting in release of the steam cloud in a matter of seconds.

c. Steam cloud rise as predicted by the following equation could vary between 100 and 600 meters depending upon the assumptions made regarding wind speed.[16]

$$h = \frac{11Q^{1/3}}{u}$$

where:

h = Height of cloud rise (ft)
u = Wind speed (ft/sec)
Q = Heat output of cloud (cal/sec)

While the effect of cloud rise is a physical reality, this effect has been neglected for this accident and the assumption is made that the steam cloud does not attain an elevation greater than the height of the turbine building.

The following assumptions and initial conditions are used in calculating the radiological effects of the steam line break accident:

a. The steam cloud movement parameters of paragraph 14.11.5.2.3, and

b. All of the activity released from the reactor vessel to the turbine building is conservatively assumed to escape to the environment.

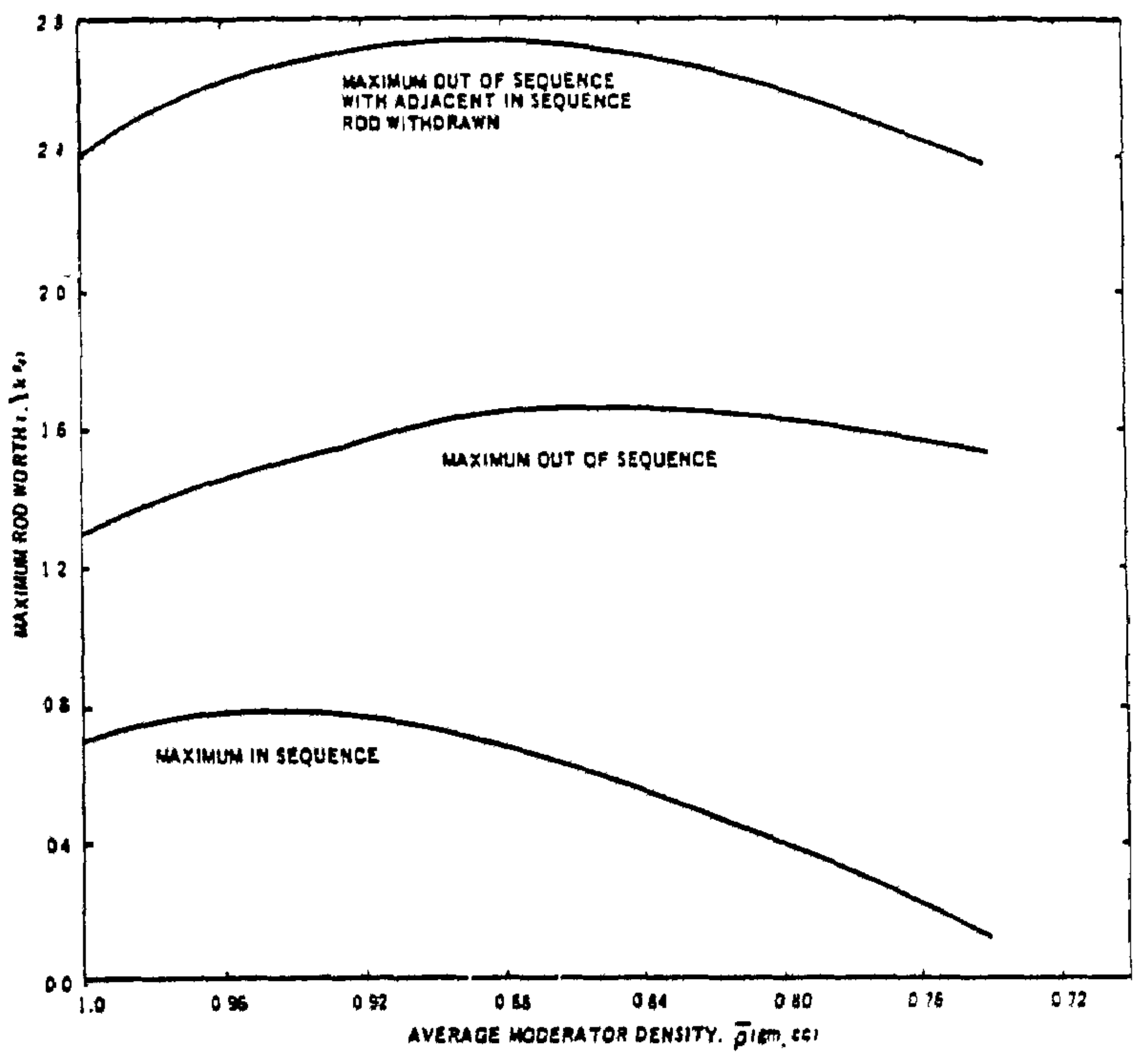
14.11.5.3 Radiological Effects

The resulting radiological exposures are shown in Table 14.11-11. These values are well within the guideline doses set forth in 10 CFR 100.

Since all of the activity is released to the environment in the form of a puff, the doses indicated are maximum values regardless of what dose period is being evaluated.

It is concluded that no danger to the health and safety of the public results as a consequence of this accident.

[16] Singer, I. A., Frizzola, J. A., Smith M . E., "The Predition of the Rise Of A Hot Cloud From Field Experiments, "Journal of the Air Pollution Control Association, November, 1964.

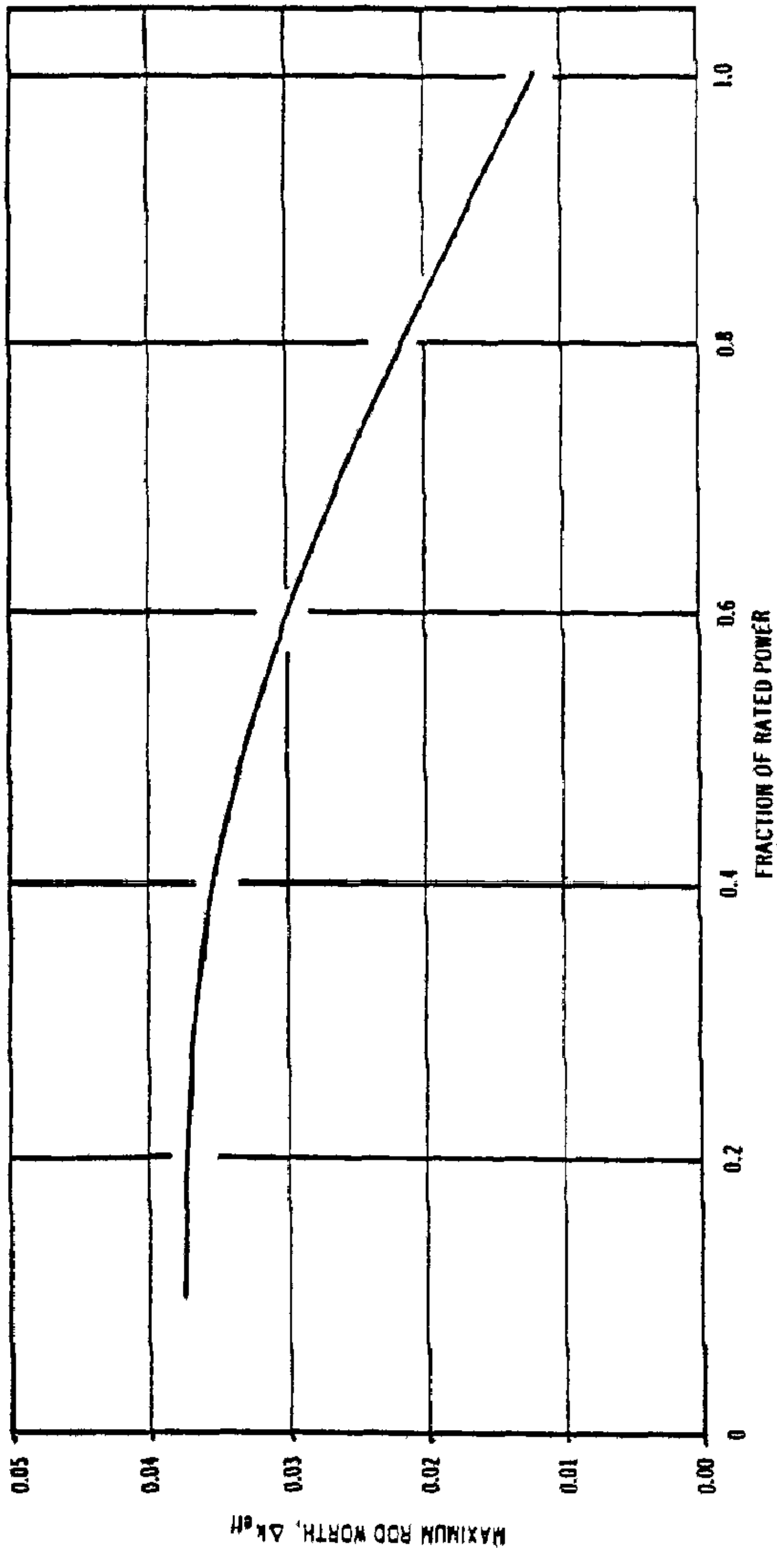


AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

Maximum Rod Worth
Versus Moderator Density

FIGURE 14.11-1

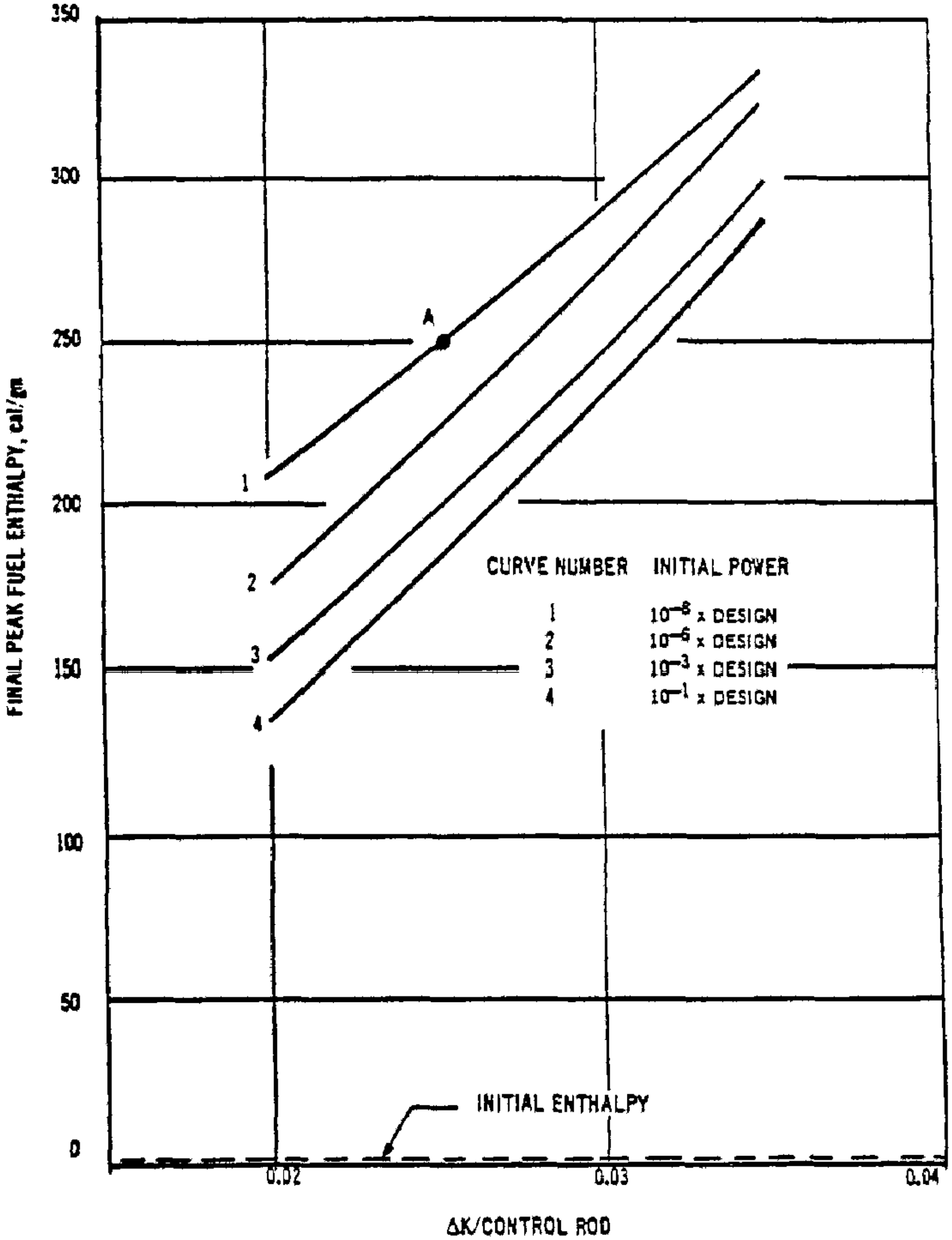


AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

Maximum Rod Worth
Versus Power Level

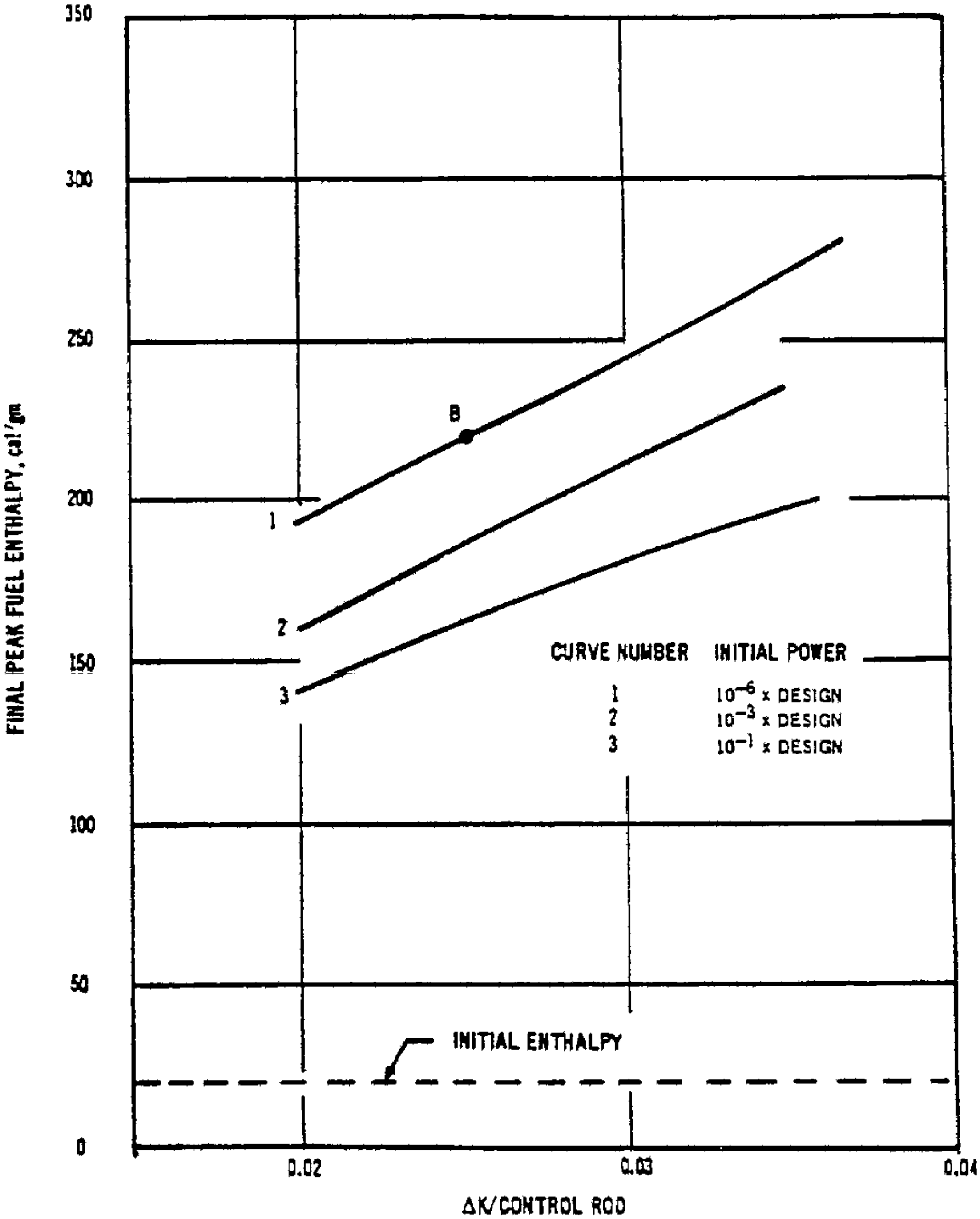
FIGURE 14.11-2



AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

Rod Drop Accident (Cold, Critical)
Peak Fuel Enthalpy
FIGURE 14 11-3

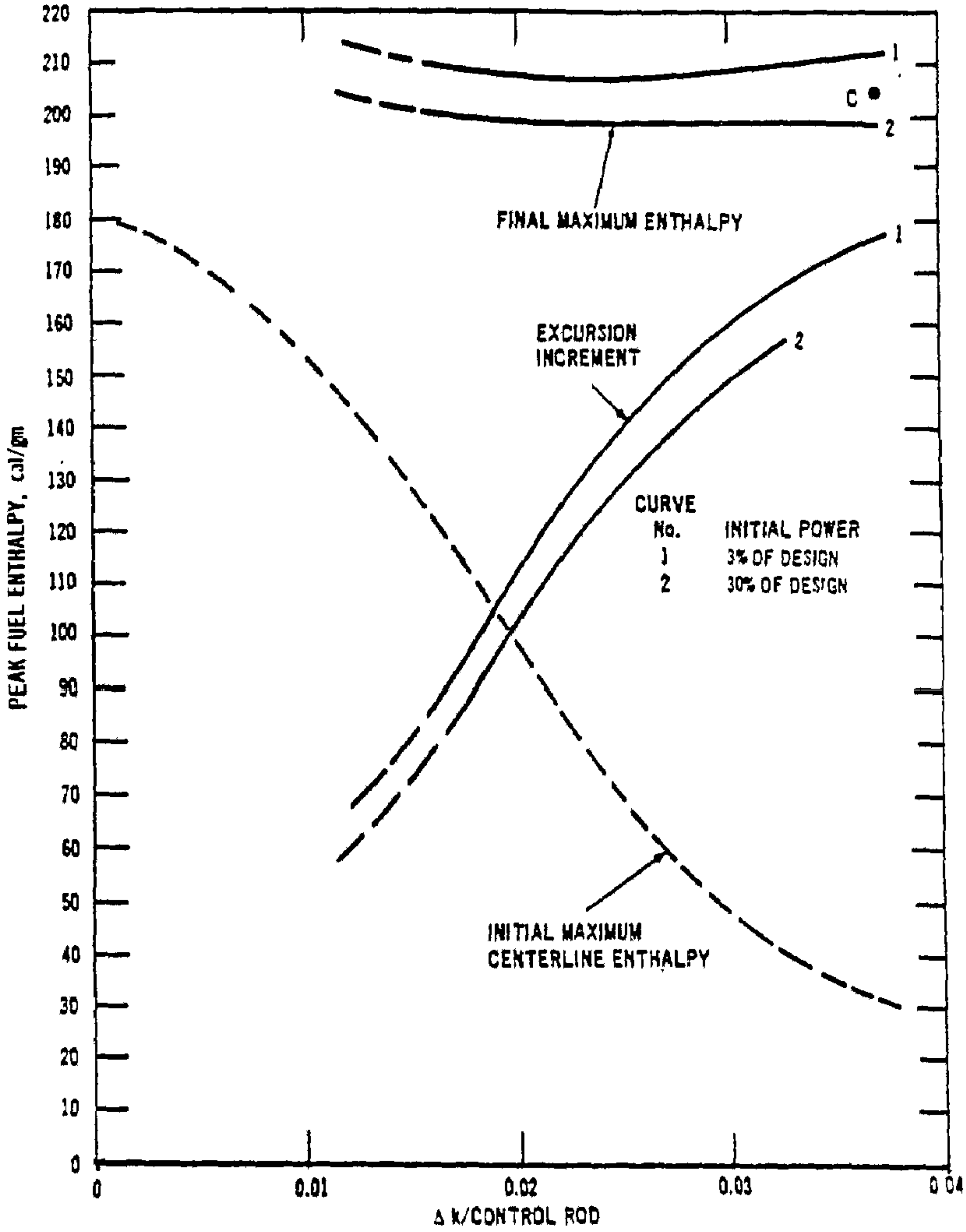


AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

Rod Drop Accident (Hot, Critical)
Peak Fuel Enthalpy

FIGURE 14.11-4



AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

Rod Drop Accident (Power Range)
Peak Fuel Enthalpy

FIGURE 14.11-5

BFN-17

Figure 14.11-6 through 9

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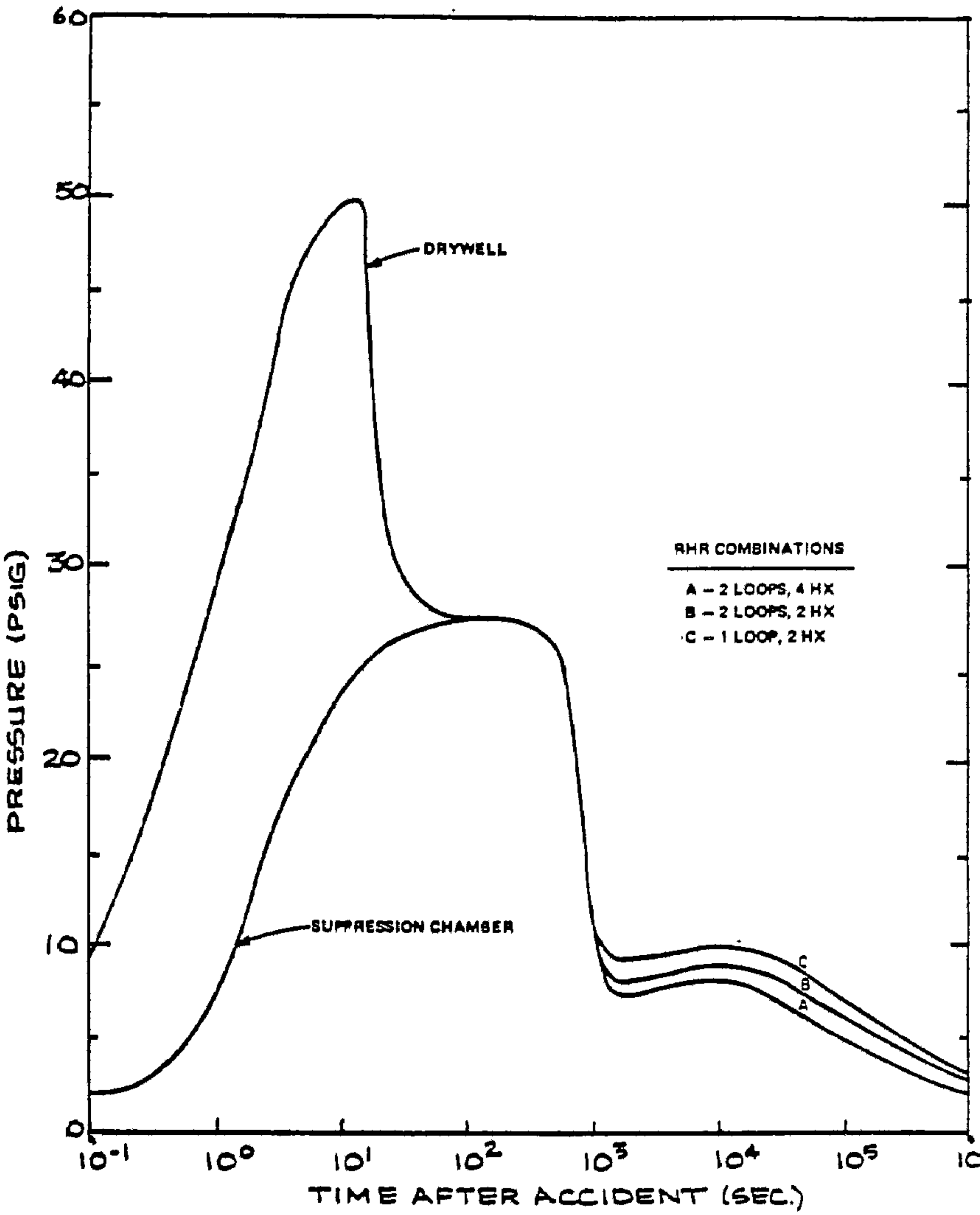
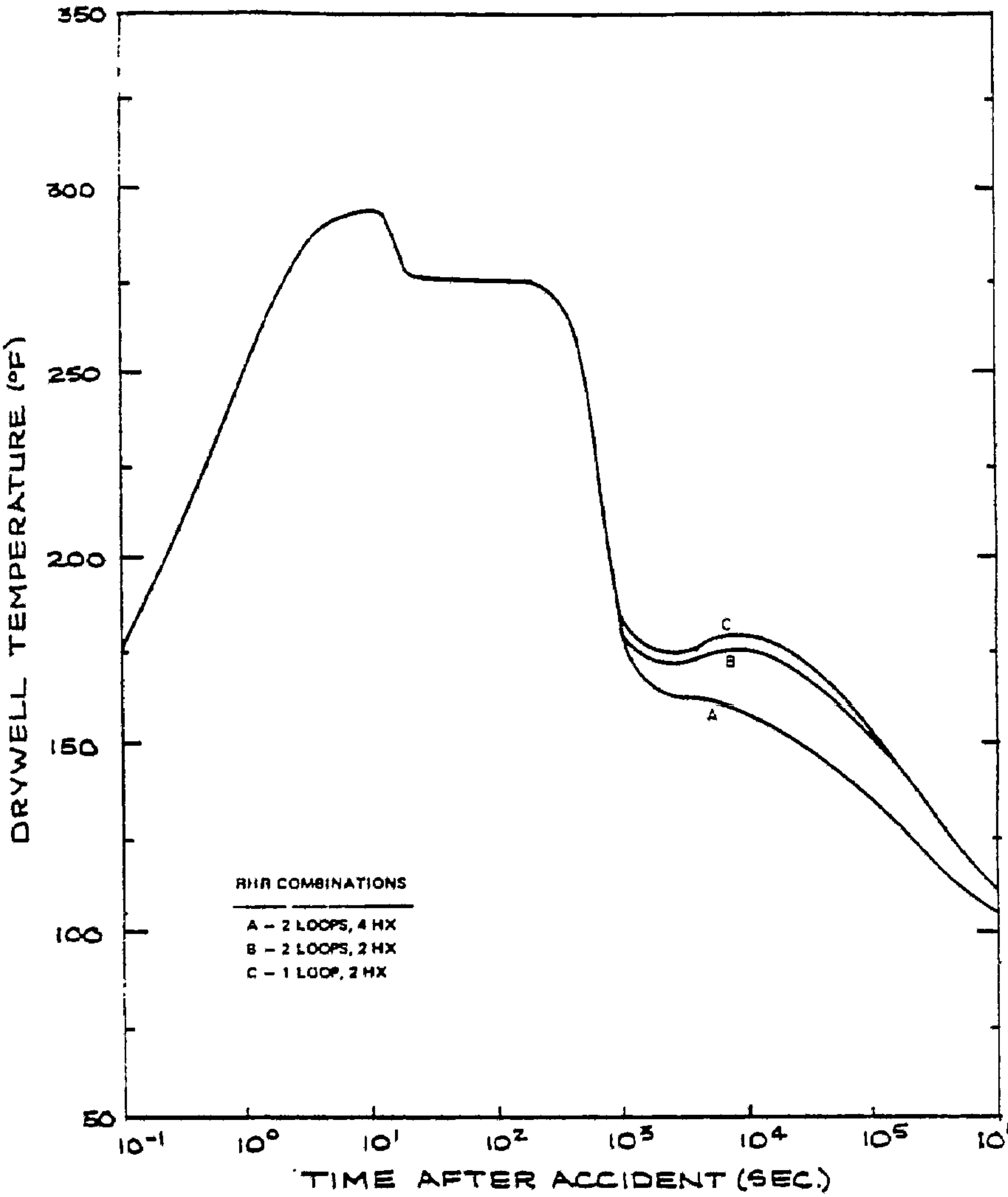


AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Loss-of-Coolant Accident
Primary Containment Pressure
Response

FIGURE 14.11-10

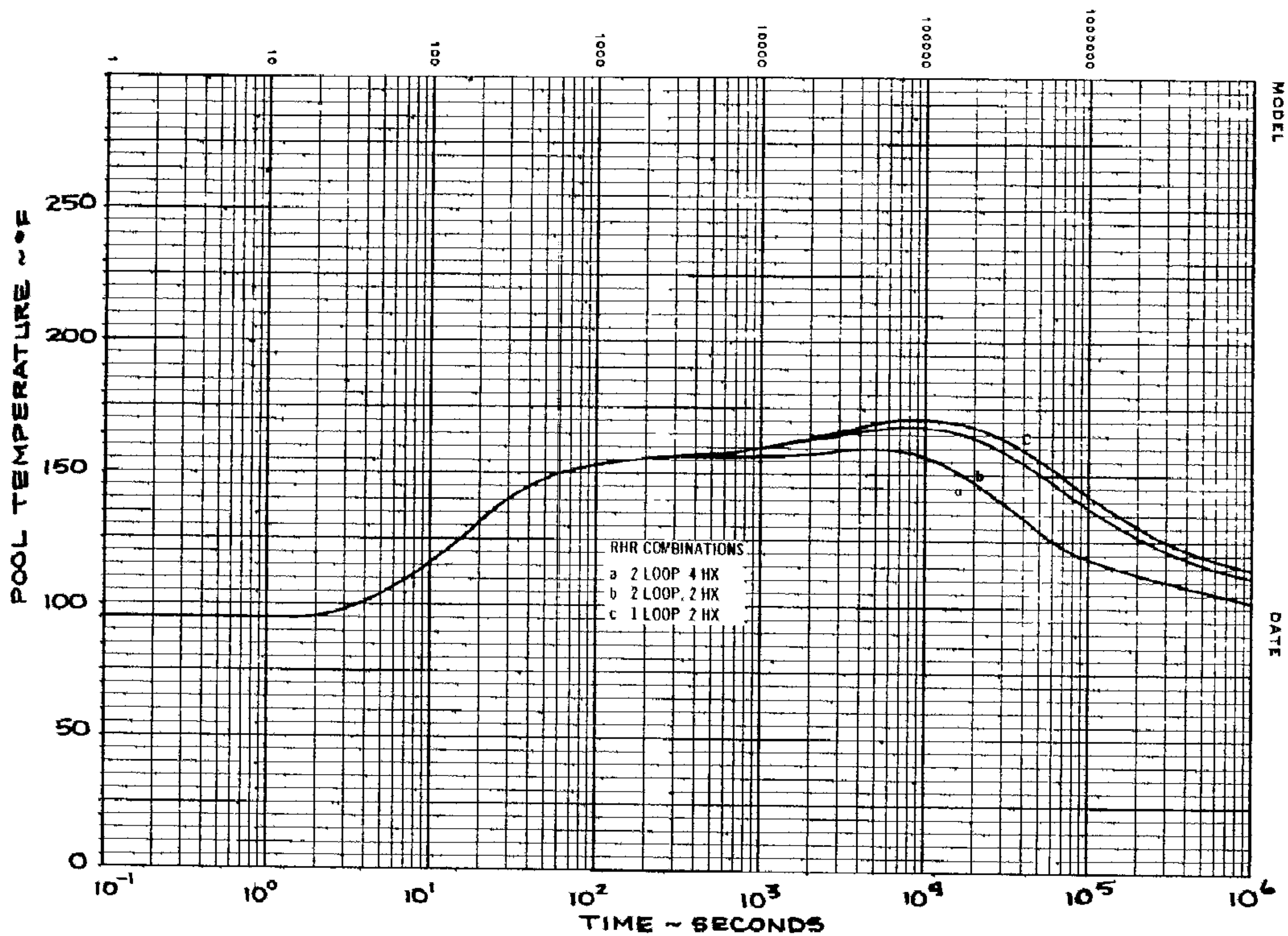


BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Loss-of-Coolant Accident
DrywEll Temperature Response

FIGURE 14.11-11

AMENDMENT 17



AMENDMENT 18

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

Loss-of-Coolant Accident,
Pressure Suppression
Pool Temperature Response
FIGURE 14.11-12

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Figure 14.11-13

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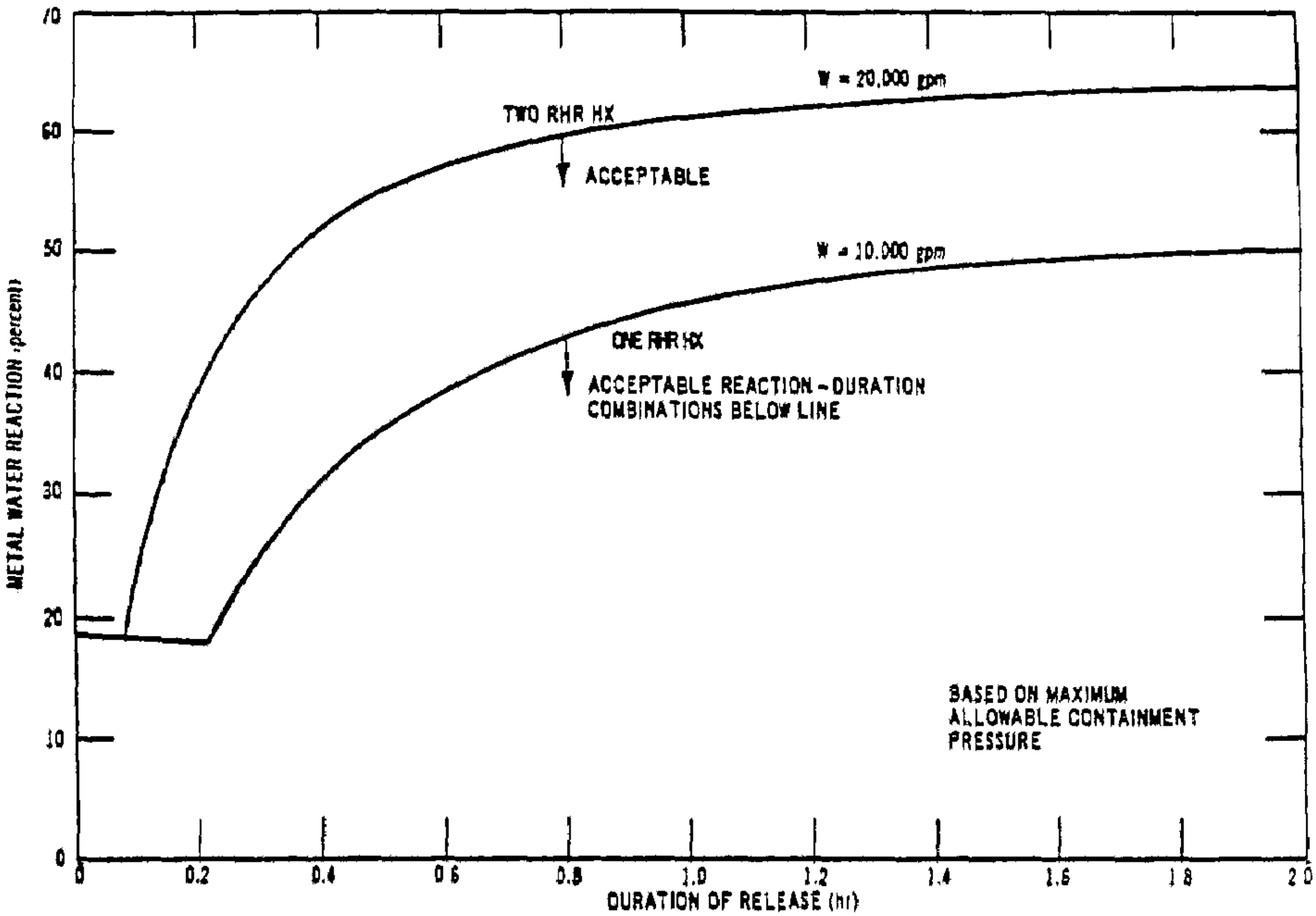
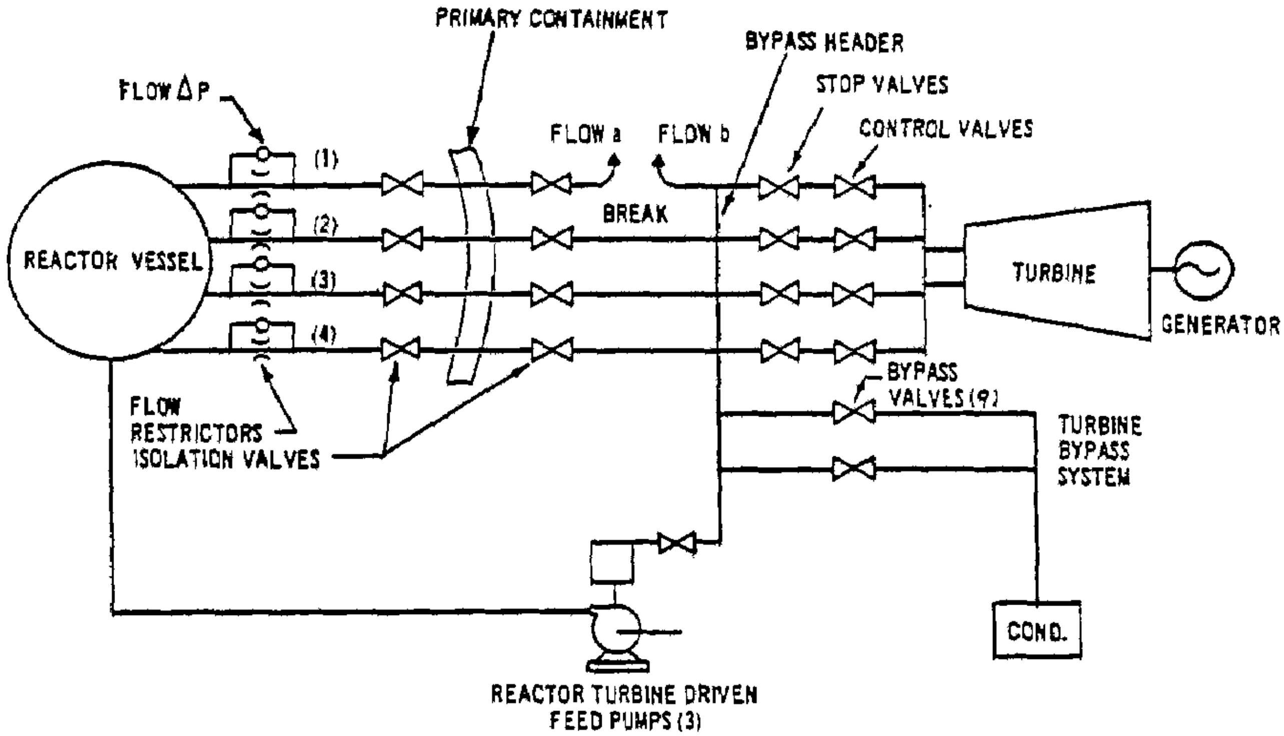


AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Loss-of-Coolant Accident,
Primary Containment Capability
for Metal-Water Reaction

FIGURE 14.11-14

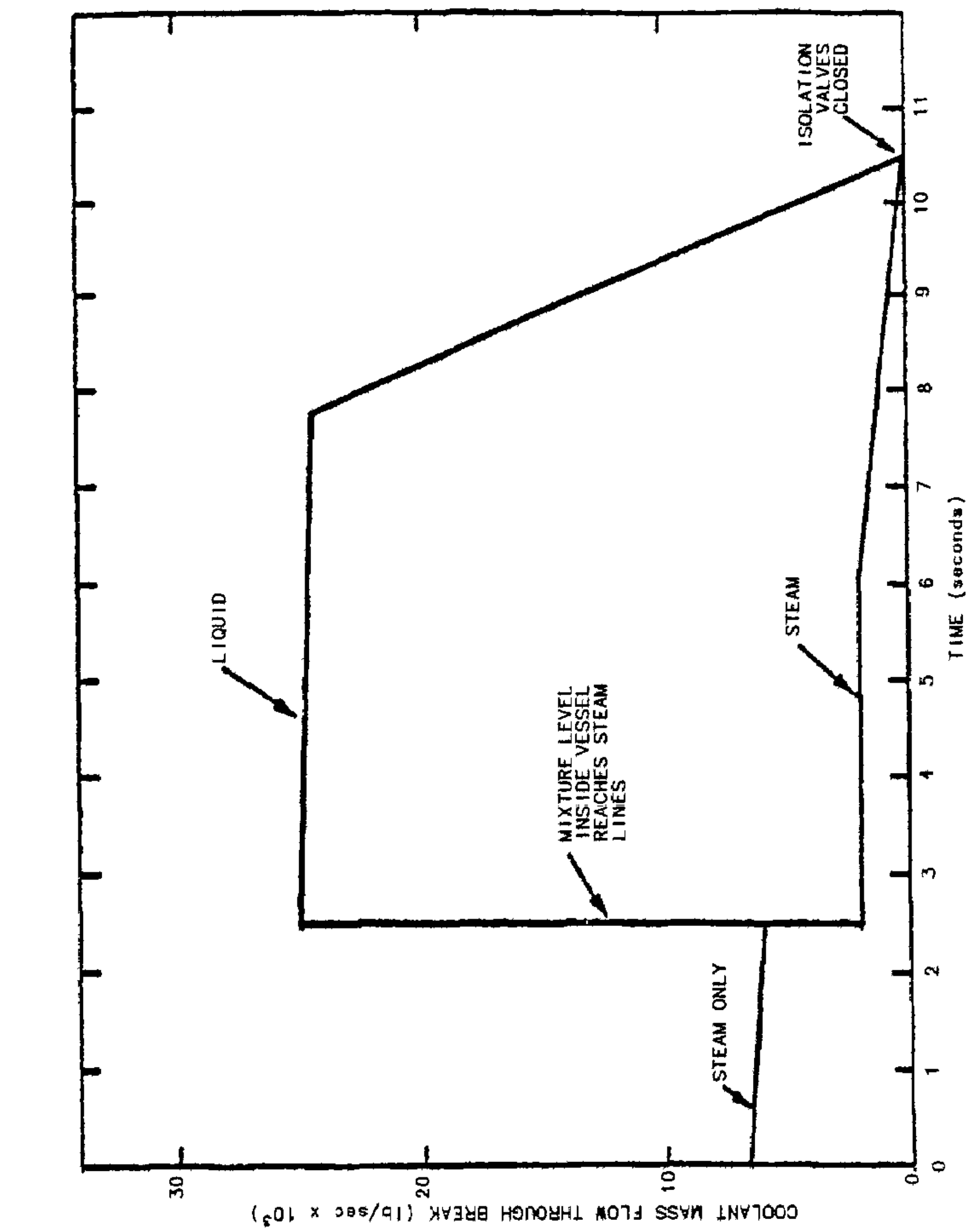


AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Main Steamline Break Accident
Break Location

FIGURE 14.11-15

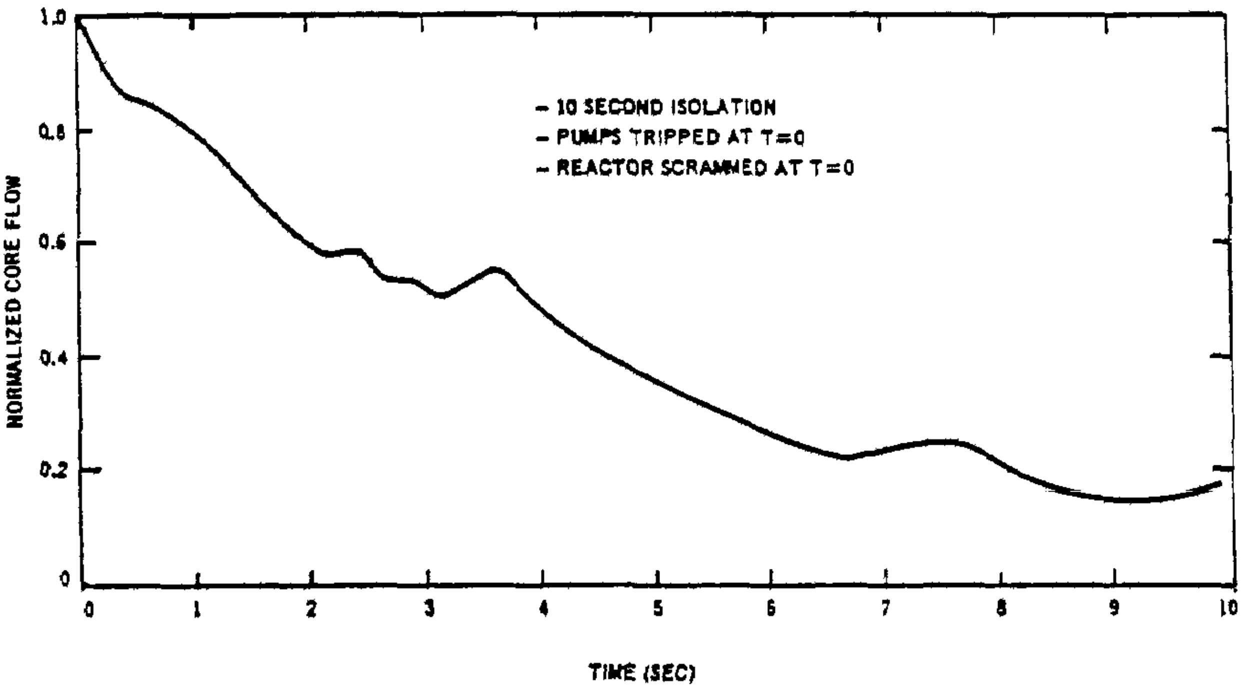


AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

MAIN STEAMLINE BREAK ACCIDENT
MASS OF COOLANT LOST THROUGH BREAK

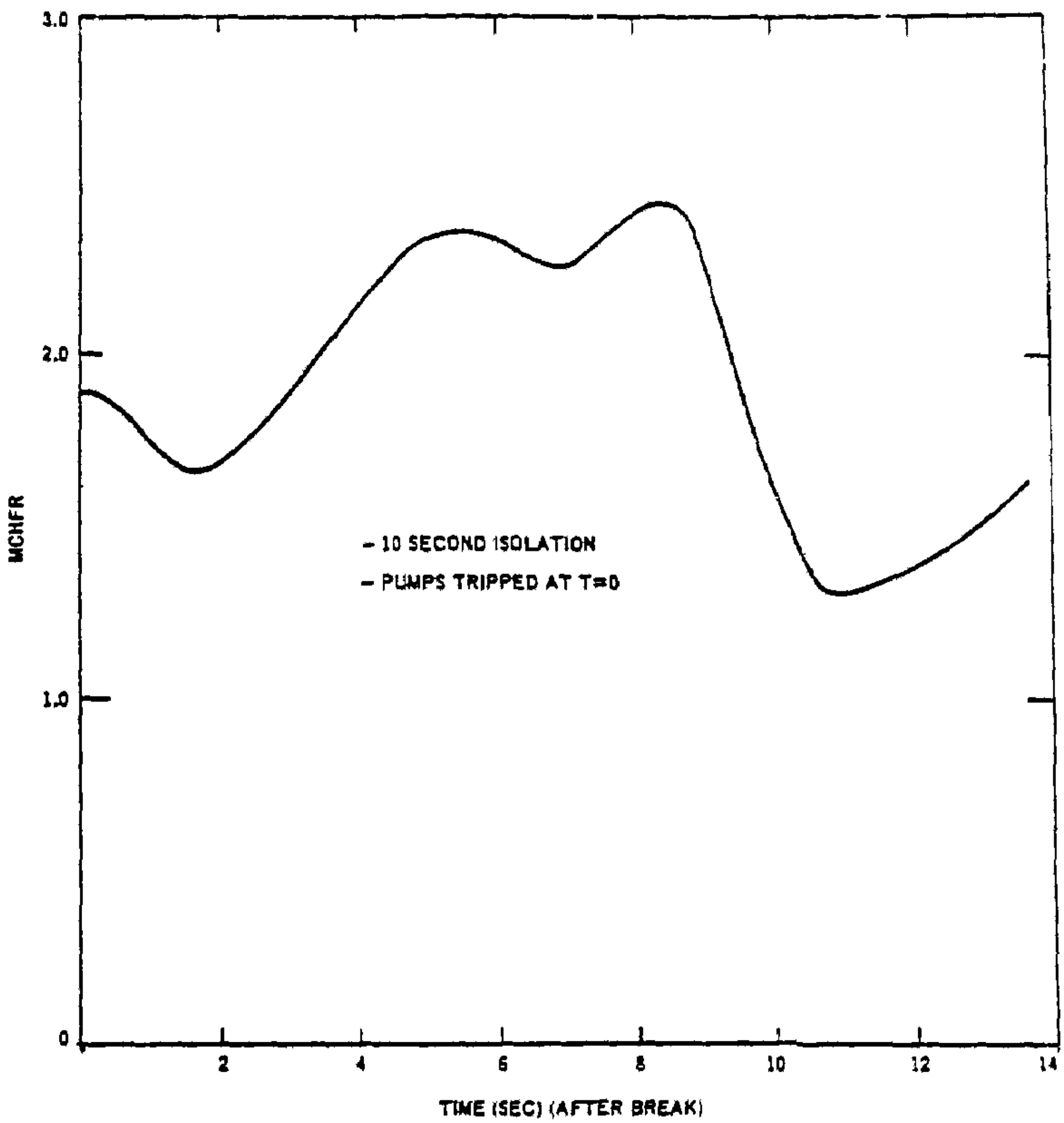
FIGURE 14.11-16



AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Main Steamline Break Accident
Normalized Core Inlet Flow
FIGURE 14.6-17
FIGURE 14 11-17



AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Main Steamline Break Accident
Minimum Critical Heat Flux Ratio

FIGURE 14.11-18

Table 14.11-1

CONTROL ROD DROP ACCIDENT

FISSION PRODUCT RELEASE RATE TO ENVIRONS

(INITIAL CORE)

	Fission Product Activity Being Released to Environment	
Time After Accident	Noble Gases (curies/sec)	Iodines (curies/sec)
1 min	1.7 X 10°	$1.8 X 10^{-5}$
30 min	1.1 X 10°	$1.7 X 10^{-5}$
1 hr	$7.4 X 10^{-1}$	$1.6 X 10^{-5}$
2 hr	$3.4 X 10^{-1}$	$1.4 X 10^{-5}$
12 hr	$1.4 X 10^{-4}$	$5.0 X 10^{-6}$
1 day	$1.5 X 10^{-7}$	$1.7 X 10^{-6}$
2 days	$1.6 X 10^{-16}$	$2.4 X 10^{-7}$
5 days	$7.9 x 10^{-39}$	$8.4 X 10^{-10}$

TABLE 14.11-2
CONTROL ROD DROP ACCIDENT RADIOLOGICAL EFFECTS
(INITIAL CORE)

Meteorology Distance (meters)	VS-1	MS-1	N-1	N-5	U-1	U-5
		0.2 Hour Cloud Gamma Doses (rem)				
1,400*	4.4×10^{-3}	4.4×10^{-3}	5.6×10^{-3}	7.8×10^{-4}	9.5×10^{-3}	1.3×10^{-3}
3,000	2.7×10^{-3}	2.9×10^{-3}	5.1×10^{-3}	7.3×10^{-4}	4.6×10^{-3}	7.6×10^{-3}
8,000	1.3×10^{-3}	1.4×10^{-3}	1.8×10^{-3}	4.4×10^{-4}	9.5×10^{-4}	2.2×10^{-4}
16,000	6.4×10^{-4}	8.0×10^{-4}	5.8×10^{-4}	2.0×10^{-4}	2.5×10^{-4}	8.2×10^{-5}
		0-2 Hour Thyroid Inhalation Doses (rem)				
1,400*	a	a	1.3×10^{-5}	1.4×10^{-7}	2.4×10^{-4}	2.7×10^{-5}
3,000	a	2.5×10^{-10}	1.1×10^{-4}	9.1×10^{-6}	1.2×10^{-5}	1.7×10^{-5}
8,000	a	8.7×10^{-6}	5.8×10^{-5}	1.0×10^{-5}	2.7×10^{-5}	4.9×10^{-6}
16,000	a	8.4×10^{-6}	2.4×10^{-5}	4.7×10^{-6}	9.6×10^{-6}	1.8×10^{-6}
		0-24 Hour Cloud Gamma Doses (rem)				
1,400	5.4×10^{-3}	5.4×10^{-3}	7.1×10^{-3}	9.8×10^{-4}	1.2×10^{-2}	1.7×10^{-3}
3,000	3.5×10^{-3}	3.6×10^{-3}	6.4×10^{-3}	9.1×10^{-4}	5.6×10^{-3}	9.6×10^{-4}
8,000	1.6×10^{-3}	1.8×10^{-3}	2.4×10^{-3}	5.4×10^{-4}	1.2×10^{-3}	2.7×10^{-4}
16,000	8.0×10^{-4}	1.0×10^{-3}	7.5×10^{-4}	2.4×10^{-4}	3.1×10^{-4}	9.8×10^{-5}
		0-24 Hour Thyroid Inhalation Doses (rem)				
1,400*	a	a	5.5×10^{-5}	6.0×10^{-7}	1.0×10^{-3}	1.1×10^{-4}
3,000	a	1.1×10^{-9}	4.6×10^{-4}	3.8×10^{-5}	4.9×10^{-4}	7.3×10^{-5}
8,000	a	3.8×10^{-6}	2.7×10^{-4}	4.4×10^{-5}	1.2×10^{-4}	2.2×10^{-5}
16,000	a	3.5×10^{-5}	1.0×10^{-4}	2.0×10^{-5}	4.2×10^{-5}	7.8×10^{-6}

Symbol Definitions:

VS-1 - Very stable 1m/sec winds
MS-1 - Moderately stable 1m/sec winds
N-1 - Neutral 1m/sec winds
N-5 - Neutral 5m/sec winds
U-1 - Unstable 1m/sec winds
U-5 - Unstable 5m/sec winds

* - Nearest site boundary
a - Dose value 10^{-10}

Table 14.11-3

LOSS-OF-COOLANT ACCIDENT

PRIMARY CONTAINMENT RESPONSE SUMMARY

Case	RHR Loops	RHR HX	RHR Pumps	Service Water Pumps	Pressure Suppression Chamber Cooling	Core Spray (gpm)	Peak Pool Temperature (°F)	Secondary Peak Pressure (psig)
A	2	4	4	4	30,000	6250	158	8.0
B	2	2	2	2	20,000	6250	169	9.8
C	1	2	2	2	16,000	6250	173	10.7
C*	1	2	2	2	11,700	5600	172	9.0

*Reanalysis based on NEDC-32484P, Revision 2.

Table 14.11-4

INVENTORY IN PRIMARY CONTAINMENT AVAILABLE FOR LEAKAGE

Isotope	Activity (Ci)	λ (hr^{-1})	Isotope	Activity (Ci)	λ (hr^{-1})
^{131}I	7.7173E7	3.59E-3	^{85}Kr	1.3977E6	7.34E-6
^{132}I	1.1455E8	3.01E-1	^{87}Kr	7.6039E7	5.47E-1
^{133}I	1.8416E8	3.30E-2	^{88}Kr	1.0720E8	2.48E-1
^{134}I	2.0719E8	7.88E-1	^{89}Kr	1.4002E8	1.31E1
^{135}I	1.7438E8	1.03E-1	$^{131}Xe^m$	1.1873E6	2.41E-3
$^{131}I^*$	3.3922E6	3.59E-3	$^{133}Xe^m$	5.7251E6	1.28E-2
$^{132}I^*$	5.0351E6	3.01E-1	^{133}Xe	2.0252E8	5.48E-3
$^{133}I^*$	8.0945E6	3.30E-2	$^{135}Xe^m$	3.1599E7	2.65E0
$^{134}I^*$	9.1072E6	7.88E-1	^{135}Xe	1.9842E8	7.57E-2
$^{135}I^*$	7.6652E6	1.03E-1	^{137}Xe	1.8343E8	1.09E1
$^{85}Kr^m$	3.8936E7	1.58E-1	^{138}Xe	1.8795E8	2.93E0

Note: * denotes organic form, m denotes metastable state

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Table 14.11-5

(Deleted by Amendment 17)

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Table 14.11-6

(Deleted by Amendment 17)

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Table 14.11-7

(Deleted by Amendment 17)

Table 14.11-8

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	Note 1 U1 Intake	Note 2	Note 1 Unit 3 Intake		
0-0.5 hrs*	3.40E-5	3.31E-5	3.02E-5	2.40E-5	1.30E-5
0.5-2 hrs	5.90E-15	5.90E-15	9.64E-7	9.70E-7	8.00E-7
2-8 hrs	4.29E-15	3.80E-15	1.89E-7		8.00E-7
8-24 hrs	3.65E-15	3.02E-15	8.37E-8		4.00E-7
1-4 days	2.58E-15	1.90E-15	1.43E-8		2.00E-7
4-30 days	1.57E-15	9.60E-16	1.13E-9		6.50E-8
*Fumigation					
Base of Stack Releases	Note 1	Note 2	Note 1		
0-2 hrs	3.70E-3	8.89E-4	1.20E-3	1.22E-4	5.65E-5
2-8 hrs	2.38E-3	7.30E-4	7.91E-4		5.65E-5
8-24 hrs	1.91E-3	6.60E-4	6.42E-4		2.24E-5
1-4 days	1.19E-3	5.40E-4	4.09E-4		7.94E-6
4-30 days	5.97E-4	4.00E-4	2.14E-4		1.71E-6
Refuel Floor Damper Bypass (FHA Only)					
0-15 secs		1.46E-4		1.22E-4	5.65E-5
Turb. Bldg. Release (MSLB only)		6.56E-4		2.70E-4	1.32E-4

Note 1: The control room X/Q values used in these columns are for the LOCA analysis only.
Note 2: The control room X/Q values used in this column is for all other radiological evaluations.

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Table 14.11-9

(Deleted by Amendment 17)

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Table 14.11-10

(Deleted by Amendment 17)

Table 14.11-11

STEAM LINE BREAK ACCIDENT

RADIOLOGICAL EFFECTS

Distance (m)	Gamma Dose (rem)	Thyroid Dose (rem)
1465 (EAB)	0.657	32.05
3200 (LPZ)	0.321	15.67