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DRAWINGS CITED IN THIS CHAPTER*

*The listed drawings are included as “General References” only; i.e., refer to the drawings to obtain additional detail or to obtain background information. These drawings are not part of the UFSAR. They are controlled by the Controlled Documents Program.

DRAWING*	SUBJECT
108D685-15	Auxiliary Feedwater Pumps Startup Diagram

CHAPTER 15.0 - ACCIDENT ANALYSES

15.0 ACCIDENT ANALYSES

This chapter addresses the representative initiating events listed on pages 15-10, 15-11, and 15-12 of Regulatory Guide 1.70, as they apply.

Certain items in the guide warrant comment, as follows:

Items 1.3 and 2.1 - There are no pressure regulators in the nuclear steam supply system (NSSS) pressurized water reactor (PWR) design whose malfunction or failure could cause a steam flow transient.

In compliance with the requirements of Regulatory Guide 1.70, a failure mode and effects analysis (FMEA) has been provided for each safety system needed to mitigate the consequences of the accidents analyzed in Chapter 15.0. A FMEA for the emergency core cooling system is provided in Section 6.3.2.5; for the residual heat removal system in Subsection 5.4.7.2.5; for the engineered safety features actuation system in Subsection 7.3.2; for the control rod drive system in Subsection 4.6.2; for the reactor trip system in Subsection 7.2.2.1; for the chemical and volume control system in Subsection 9.3.4.1.3.9; for the containment spray system in Subsection 6.5.2.2; and, for the auxiliary feedwater system in Subsection 10.4.9.3.

Once actuated, the equipment of a safety system operates in the same manner regardless of initiating accident. However, performance depends on the type of accident that is being mitigated.

15.0.1 Classification of Plant Conditions

Since 1970, the American Nuclear Society (ANS) classification of plant conditions has been used which divides plant conditions into four categories in accordance with anticipated frequency of occurrence and potential radiological consequences to the public. The four categories are as follows:

Condition I: Normal Operation and Operational Transients.
Condition II: Faults of Moderate Frequency.
Condition III: Infrequent Faults.
Condition IV: Limiting Faults.

The basic principle applied in relating design requirements to each of the conditions is that the most probable occurrences should yield the least radiological risk to the public and those extreme situations having the potential for the greatest risk to the public shall be those least likely to occur. Where applicable, reactor trip system and engineered safeguards functioning is assumed to the extent allowed by considerations such as the single failure criterion, in fulfilling this principle.

15.0.1.1 Condition I - Normal Operation and Operational Transients

Condition I occurrences are those which are expected frequently or regularly in the course of power operation, refueling, maintenance, or maneuvering of the plant. As such, Condition I occurrences are accommodated with margin between any plant parameter and the value of that parameter which would require either automatic or manual protective action. Inasmuch as Condition I occurrences occur frequently or regularly, they must be considered from the point of view of affecting the consequences of fault conditions (Conditions II, III and IV). In this regard, analysis of each fault condition described is generally based on a conservative set of initial conditions corresponding to adverse conditions which can occur during Condition I operation.

A typical list of Condition I events is listed below:

a. Steady-state and shutdown operations

   1. Power operation (>5 to 100% of rated thermal power),

   2. Startup (k_{eff} > 0.99 to ≤ 5% of rated thermal power),

   3. Hot standby (subcritical, residual heat removal system isolated),

   4. Hot shutdown (subcritical, residual heat removal system in operation),

   5. Cold shutdown (subcritical, residual heat removal system in operation), and

   6. Refueling

b. Operation with permissible deviations

   Various deviations which may occur during continued operation as permitted by the plant Technical Specifications must be considered in conjunction with other operational modes. These include:

   1. Operation with components or systems out of service,

   2. Radioactivity in the reactor coolant, due to leakage from fuel with cladding defects,

(a) Fission products

(b) Corrosion products

(c) Tritium

3. Operation with steam generator leaks up to the maximum allowed by the Technical Specification 3.4.13, and

4. Testing as allowed by Technical Specifications and the Technical Requirements Manual (TRM)

c. Operational transients

1. Plant heatup and cooldown (up to 100°F/hour for the reactor coolant system; 200°F/hour for the pressurizer during cooldown and 100°F/hour for the pressurizer during heatup),

2. Step load changes (up to ± 10%),

3. Ramp load changes (up to 5%/minute), and

4. Load rejection up to and including design full load rejection transient

15.0.1.2 Condition II - Faults of Moderate Frequency

These faults, at worst, result in the reactor trip with the plant being capable of returning to operation. By definition, these faults (or events) do not propagate to cause a more serious fault, i.e., Condition III or IV events. In addition, Condition II events are not expected to result in fuel rod failures or reactor coolant system or secondary system overpressurization.

For the purposes of this report, the following faults are included in this category:

a. Feedwater system malfunctions that result in a decrease in feedwater temperature (Subsection 15.1.1),

b. Feedwater system malfunctions that result in an increase in feedwater flow (Subsection 15.1.2),

c. Excessive increase in secondary steam flow (Subsection 15.1.3),

d. Inadvertent opening of a steam generator relief or safety valve (Subsection 15.1.4),

e. Loss of external electrical load (Subsection 15.2.2),

f. Turbine trip (Subsection 15.2.3),

g. Inadvertent closure of main steam isolation valves (Subsection 15.2.4),

h. Loss of condenser vacuum and other events resulting in turbine trip (Subsection 15.2.5),

i. Loss of nonemergency ac power to the station auxiliaries (Subsection 15.2.6),

j. Loss of normal feedwater flow (Subsection 15.2.7),

k. Partial loss of forced reactor coolant flow (Subsection 15.3.1),

l. Uncontrolled rod cluster control assembly bank withdrawal at a subcritical or low power startup condition (Subsection 15.4.1),

m. Uncontrolled rod cluster control assembly bank withdrawal at power (Subsection 15.4.2),

n. Rod cluster control assembly misalignment (dropped full length assembly, dropped full length assembly bank, or statically misaligned full length assembly) (Subsection 15.4.3),

o. Deleted

p. Chemical and volume control system malfunction that results in a decrease in the boron concentration in the reactor coolant (Subsection 15.4.6),

q. Inadvertent operation of the emergency core cooling system during power operation (Subsection 15.5.1),

r. Chemical and volume control system malfunction that increases reactor coolant inventory (Subsection 15.5.2),

s. Inadvertent opening of a pressurizer safety or relief valve (Section 15.6.1), and

t. Break in instrument line or other lines from reactor coolant pressure boundary that penetrate containment (Subsection 15.6.2).

15.0.1.3 Condition III - Infrequent Faults

By definition Condition III occurrences are faults which may occur very infrequently during the life of the plant. They will be accommodated with the failure of only a small fraction of the fuel rods although sufficient fuel damage might occur to preclude resumption of the operation for a considerable outage time. The release of radioactivity will not be sufficient to interrupt or restrict public use of those areas beyond the exclusion radius. A Condition III fault will not, by itself, generate a Condition IV fault or result in consequential loss of function of the reactor coolant system or containment barriers. For the purposes of this report the following faults are included in this category:

a. Steam system piping failure from zero power and full power (minor) (Subsections 15.1.5 and 15.1.6),

b. Complete loss of forced reactor coolant flow (Subsection 15.3.2),

c. Rod cluster control assembly misalignment (single rod cluster control assembly withdrawal at full power) (Subsection 15.4.3),

d. Inadvertent loading and operation of a fuel assembly in an improper position (Subsection 15.4.7),

e. Loss of coolant accidents resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary (small break) (Subsection 15.6.5),

f. Gaseous radwaste system leak or failure (Subsection 15.7.1),

g. Liquid radwaste system leak or failure (Subsection 15.7.2),

h. Postulated radioactive releases due to liquid tank failures (Subsection 15.7.3), and

i. Spent fuel cask drop accidents (Subsection 15.7.5).

15.0.1.4 Condition IV - Limiting Faults

Condition IV occurrences are faults which are not expected to take place, but are postulated because their consequences would include the potential of the release of significant amounts of radioactive material. They are the most drastic which must be designed against and represent limiting design cases. Condition IV faults are not to cause a fission product release to the environment resulting in an undue risk to public health and safety in excess of guidelines values of 10 CFR 100 for TID-14844 based dose analyses and 10 CFR 50.67 for AST based analyses. A single Condition IV fault is not to cause a consequential loss of required functions of systems needed to cope with the fault including those of the emergency core cooling system and the containment. For the purposes of this report, the following faults have been classified in this category:

a. Steam system piping failure from zero power and full power (major) (Subsections 15.1.5 and 15.1.6),

b. Feedwater system pipe break (Subsection 15.2.8),

c. Reactor coolant pump shaft seizure (locked rotor) (Subsection 15.3.3),

d. Reactor coolant pump shaft break (Subsection 15.3.4),

e. Spectrum of rod cluster control assembly ejection accidents (Subsection 15.4.8),

f. Steam generator tube failure (Subsection 15.6.3),

g. Loss of coolant accidents resulting from the spectrum of postulated piping breaks within the reactor coolant pressure boundary (large break) (Subsection 15.6.5), and

h. Design basis fuel handling accidents (Subsection 15.7.4).

15.0.1.5 Summary of Results

For all Condition II transients analyzed, the calculated minimum DNBR was greater than the limit value. For each of these transients, the peak RCS pressure was less than the safety limit of 110% of design pressure (2750 psia) and there was no failed fuel as a result of the transients. Since DNB does not occur for any Condition II transients, peak cladding temperature does not increase sufficiently above nominal values for these events.

For all of the applicable Condition III transients, the minimum DNBR was greater than the limit value and there was no failed fuel except for a single RCCA withdrawal at full power. For this transient, the upper bound of the number of fuel rods experiencing DNBR less than the limit value was 5% of the total rods in the core. All of the applicable Condition III transients experienced a peak RCS pressure less than 2750 psia. The only Condition III transient for which cladding temperature was calculated was the small LOCA and the peak value was less than 2200°F.

All the applicable Condition IV transients analyzed met the applicable condition IV acceptance criteria. For the locked rotor event, DNB was assumed to occur at the initiation of the transient and the peak cladding temperature was calculated to be less than the limit of 2375°F (associated with Optimized ZIRLO™ fuel cladding which bounds the 2700°F limit for ZIRLO® fuel cladding, Reference 21). For the LOCA the amount of failed fuel calculated was equal or less than 100% and for rod ejection it was equal or less than 10%. The locked rotor accident results in cladding failure in less than 2% of the fuel rods. There was no failed fuel predicted for the feed line break, the steam line break, or the steam generator tube rupture. All of the applicable Condition IV transients experienced a peak RCS pressure less than 2750 psia. The peak cladding temperature calculated for LOCA was less than 2200°. The average fuel pellet enthalpy at the hot spot for rod ejection was less than 200 cal/gm.

15.0.2 Optimization of Control Systems

A control system setpoint study is performed in order to simulate performance of the reactor control and protection systems. In this study, emphasis is placed on the development of a control system which will automatically maintain prescribed conditions in the plant even under a conservative set of reactivity parameters with respect to both system stability and transient performance. For each mode of plant operation, a group of optimum controller setpoints is determined. In areas where the resultant setpoints are different, compromises based on the optimum overall performance are made and verified. A consistent set of control system parameters is derived satisfying plant operational requirements throughout the core life and for various levels of power operation.

The study comprises an analysis of the following control systems: rod cluster control assembly, steam dump, steam generator, level, pressurizer pressure, and pressurizer level.

15.0.3 Plant Characteristics and Initial Conditions Assumed in the Accident Analyses

Each of the four RCS loops are equipped with loop isolation valves. However, the stations are not currently licensed for less than all loops in operation. Therefore, the UFSAR presents the licensing basis for four-loop operation only.

15.0.3.1 Design Plant Conditions

Table 15.0-1 lists the principal power rating values which are assumed in analyses performed in this report.

Allowances for errors in the determination of the steady-state power level are made as described in Subsection 15.0.3.2. The thermal power values used for each transient analyzed are given in Table 15.0-2.

The values of other pertinent plant parameters utilized in the accident analyses are given in Tables 15.0-3 (Unit 1) and 15.0-4 (Unit 2).

15.0.3.2 Initial Conditions

For most accidents which are DNB limited, nominal values of initial conditions are assumed. The allowances on power, temperature, and pressure are determined on a statistical basis and are included in the limit DNBR, as described in WCAP-11397 (Reference 10). This procedure is known as the "Revised Thermal Design Procedure," and is discussed more fully in Section 4.4.

For accidents which are not DNB limited, or in which the Revised Thermal Design Procedure is not employed, the initial conditions are obtained by adding the maximum steady state errors to rated values. The following conservative steady state errors were assumed in the analysis:

a. Core power — ± 0.345% allowance for calorimetric error

b.	Average reactor coolant system temperature	+9.1 -7.6°F allowance for controller deadband and measurement error
c.	Pressurizer pressure	± 43 pounds per square inch (psi) allowance for steady state fluctuations and measurement error

Table 15.0-2 summarizes initial conditions and computer codes used in the accident analysis, and shows which accidents employed a DNB analysis using the revised thermal design procedure.

15.0.3.3 <u>Power Distribution</u>

The transient response of the reactor system is dependent on the initial power distribution. The nuclear design of the reactor core minimizes adverse power distribution through the placement of control rods and operating instructions. Power distribution may be characterized by the radial factor ($F_{\Delta H}$) and the total peaking factor (F_Q). The peaking factor limits are given in the technical specifications.

For transients which may be DNB limited, the radial peaking factor is of importance. The radial peaking factor increases with decreasing power level due to rod insertion. This increase in $F_{\Delta H}$ is included in the core limits illustrated in Figure 15.0-1. All transients that may be DNB limited are assumed to begin with a $F_{\Delta H}$ consistent with the initial power level defined in the technical specifications.

The axial power shape used in the DNB calculation is discussed in Section 4.4.

The radial and axial power distributions described above are input to the VIPRE Code as described in Section 4.4.

For transients which may be overpower limited, the total peaking factor (F_Q) is of importance. All transients that may be overpower limited are assumed to begin with plant conditions including power distributions which are consistent with reactor operation as defined in the technical specifications.

For overpower transients which are slow with respect to the fuel rod thermal time constant, for example, the chemical volume control system malfunction that results in a decrease in the boron concentration in the reactor coolant incident which lasts many minutes, and the excessive increase in secondary steam flow incident which may reach equilibrium without causing a reactor trip, the fuel rod thermal evaluations are performed as discussed in Section 4.4. For overpower transients which are fast with respect to the fuel rod thermal time constant, for example, the uncontrolled rod cluster control assembly bank withdrawal from subcritical or low power startup and rod cluster control assembly ejection incidents which result in a large power rise over a few seconds, a detailed fuel heat transfer calculation must be performed. Although the fuel rod thermal time constant is a function of system conditions, fuel burnup and rod power, a typical value at beginning-of-life for high power rods is approximately 5 seconds.

15.0.4 Reactivity Coefficients Assumed in the Accident Analyses

The transient response of the reactor system is dependent on reactivity feedback effects, in particular the moderator temperature coefficient and the Doppler power coefficient. These reactivity coefficients and their values are discussed in detail in Chapter 4.0.

In the analysis of certain events, conservatism requires the use of large reactivity coefficient values whereas in the analysis of other events, conservatism requires the use of small reactivity coefficient values. Some analyses such as loss of reactor coolant from cracks or breaks in the reactor coolant system do not depend on reactivity feedback effects. The values are given in Table 15.0-2. Figure 15.0-2 shows the Technical Specifications limit for the moderator temperature coefficient as a function of power level. Figure 15.0-3 shows the least negative (upper bound), least negative at End-of-Life, and most negative (lower bound) bound Doppler power coefficients as a function of power, used in the transient analysis. The justification for use of conservatively large versus small reactivity coefficient values are treated on an event-by-event basis. In some cases conservative combinations of parameters are used to bound the effects of core life, although these combinations may not represent possible realistic situations.

15.0.5 Rod Cluster Control Assembly Insertion Characteristics

The negative reactivity insertion following a reactor trip is a function of the position versus time of the rod cluster control assemblies and the variation in rod worth as a function of rod position. With respect to accident analyses, the critical parameter is the time of insertion up to the dashpot entry or approximately 85% of the rod cluster travel.

The rod cluster control assembly position versus time assumed in accident analyses is shown in Figure 15.0-4. The rod cluster control assembly insertion time to dashpot entry is taken as 2.7 seconds. Drop time testing requirements are dependent on the type of cluster control assemblies actually used in the plant and are specified in the plant technical specifications.

Figure 15.0-5 shows the fraction of total negative reactivity insertion versus normalized rod position for a core where the axial distribution is skewed to the lower region of the core. An axial distribution which is skewed to the lower region of the core can arise from an unbalanced xenon distribution. This curve is used to compute the negative reactivity insertion versus time following a reactor trip which is input to all point kinetics core models used in transient analyses. The bottom skewed power distribution itself is not input into the point kinetics core model.

There is inherent conservatism in the use of Figure 15.0-5 in that it is based on a skewed flux distribution which would exist relatively infrequently. For cases other than those associated with unbalanced xenon distributions, significant negative reactivity would have been inserted due to the more favorable axial distribution existing prior to trip.

The normalized rod cluster control assembly negative reactivity insertion versus time is shown in Figure 15.0-6. The curve shown in this figure was obtained from Figures 15.0-4 and 15.0-5. A total negative reactivity insertion following a trip of 4% Δk is assumed in the transient analyses except where specifically noted otherwise. This assumption is conservative with respect to the calculated trip reactivity worth available as shown in Table 4.3-3. For Figures 15.0-4 and 15.0-6, the rod cluster control assembly drop time is normalized to 2.7 seconds, unless otherwise noted for a particular event.

The normalized rod cluster control assembly negative reactivity insertion versus time curve for an axial power distribution skewed to the bottom (Figure 15.0-6) is used in those transient analyses for which a point kinetics core model is used. Where special analyses require use of three dimensional or axial one dimensional core models, the negative reactivity insertion resulting from the reactor trip is calculated directly by the reactor kinetics code and is not separable from the other reactivity feedback effects. In this case, the rod cluster control assembly position versus time of Figure 15.0-4 is used as code input.

15.0.6 Trip Points and Time Delays to Trip Assumed in Accident Analyses

A reactor trip signal acts to open two trip breakers connected in series feeding power to the control rod drive mechanisms. The loss of power to the mechanism coils causes the mechanisms to release the rod cluster control assemblies which then fall by gravity into the core. There are various instrumentation delays associated with each trip function, including delays in signal actuation, in opening the trip breakers, and in the release of the rods by the mechanisms. The total delay to trip is defined as the time delay from the time that trip conditions are reached to the time the rods are free and begin to fall. Limiting trip setpoints assumed in accident analyses and the time delay assumed for each trip function are given in Table 15.0-5.

Reference is made in that table to overtemperature and overpower ΔT trips shown in Figure 15.0-1. This figure presents the allowable reactor coolant loop average temperature and ΔT for the design flow and power distribution, as described in Section 4.4, as a function of primary coolant pressure. The boundaries of operation defined by the overpower ΔT trip and the overtemperature ΔT trip are represented as "protection lines" on this diagram. The protection lines are drawn to include all adverse instrumentation and setpoint errors so that under nominal conditions trip would occur well within the area bounded by these lines. The utility of this diagram is in the fact that the limit imposed by any given DNBR can be represented as a line. The DNB lines represent the locus of conditions for which the DNBR equals the limit value (1.37 for the thimble cell and the typical cell). All points below and to the left of a DNB line for a given pressure have a DNBR greater than the limit value. The diagram shows that DNB is prevented for all cases if the area enclosed with the maximum protection lines is not traversed by the applicable DNBR line at any point.

The area of acceptable operation during transient conditions (power, pressure and temperature) is bounded by the combination of reactor trips: high neutron flux (fixed setpoint); high pressure (fixed setpoint); low pressure (fixed setpoint); overpower and overtemperature ΔT (variable setpoints).

A Tavg coastdown/power coastdown strategy may be used at end of cycle. The first part of the strategy consists of a Tavg coastdown from full power conditions to a specified reduced Tavg. In this portion of the strategy, extended full power capability is achieved by the increase in core reactivity that occurs with the increased moderator density due to the reduced Tavg. Following the Tavg coastdown, a power coastdown is then performed to extend plant operation at end of cycle. The Chapter 15 analyses support a constant full power Tavg window. An underlying operating strategy modeled in the safety analyses is that the OTΔT and OPΔT trip functions which are dependent on the full power Tavg are appropriately scaled and/or calibrated to the specified cycle-specific value.

To support end of cycle Tavg coastdown/power coastdown operating strategy, the events that are protected by the OTΔT and OPΔT reactor trips were reanalyzed or evaluated with a low Tavg without rescaling the OTΔT and OPΔT reactor trips. The analyses and evaluations concluded that these events continue to meet the acceptance criteria for the respective events.

The difference between the limiting trip point assumed for the analysis and the normal trip point represents an allowance for instrumentation channel error and setpoint error. Nominal trip setpoints are specified in the plant technical specifications. During plant startup tests, it will be demonstrated that actual instrument time delays are equal to or less than the assumed values. Additionally, protection system channels are calibrated and instrument response times determined periodically in accordance with the technical specifications.

15.0.7 Instrumentation Drift and Calorimetric Errors - Power Range Neutron Flux

The instrumentation drift and calorimetric errors used in establishing the power range high neutron flux setpoint are presented in Table 15.0-6. The calorimetric errors presented in Table 15.0-6 are applicable for the limiting condition in which the feedwater flow venturis are used in the determination of core thermal power. The calorimetric errors associated with the determination of core thermal power using the Leading Edge Flow Meters are bounded by the errors associated with the feedwater venturis.

The calorimetric error is the error assumed in the determination of core thermal power as obtained from secondary plant measurements. The total ion chamber current (sum of the top and bottom sections) is calibrated (set equal) to this measured power on a periodic basis.

The secondary power is obtained from measurement of feedwater flow, feedwater inlet temperature to the steam generators and steam pressure. High accuracy instrumentation is provided for these measurements with accuracy tolerances much tighter than those which would be required to control feedwater flow.

The nuclear instrumentation system is calibrated by comparing individual power level indications with the power obtained calorimetrically. The error assumed in the calorimetric determination of power level (for the purpose of establishing the maximum overpower trip setpoint) is ± 2% at the 95% confidence level, as given in Table 15.0-6.

An analysis has been prepared which took the effects of measurement errors and their confidence levels into account and provided the basis for the selection of protection system setpoints. The analysis was performed in accordance with NRC requirements.

15.0.8 Plant Systems and Components Available for Mitigation of Accident Effects

The NSSS is designed to afford proper protection against the possible effects of natural phenomena, postulated environmental conditions and dynamic effects of postulated accidents. In addition, the design incorporates features which minimize the probability and effects of fires and explosions. Chapter 17.0 discusses the quality assurance program which has been implemented to assure that the NSSS will satisfactorily perform its assigned safety functions. The incorporation of these features in the NSSS, coupled with the reliability of the design, ensures that the normally operating systems and components listed in Table 15.0-7 will be available for mitigation of the events discussed in Chapter 15.0. Table 15.0-7 identifies plant systems and equipment credited for transients and accident conditions. In determining which systems are necessary to mitigate the effects of these postulated events, the classification system of ANSI-N18.2-1973 is utilized. The design of "systems important to safety" (including protection systems) is consistent with IEEE Standard 379-1972 and Regulatory Guide 1.53 in the application of the single failure criterion.

In the analysis of the Chapter 15.0 events, control system action is considered only if that action results in more severe accident results. No credit is taken for control system operation if that operation mitigates the results of an accident. For some accidents, the analysis is performed both with and without control system operation to determine the worst case.

The response times for the air-operated and motor-operated valves in the auxiliary feedwater and main steam systems are verified during preoperational testing.

15.0.9 Fission Product Inventories

15.0.9.1 Activities in the Core

For the accidents evaluated using TID-14844, the calculation of the core iodine fission product inventory was modeled using the computer code ORIGEN2 (Reference 1) which is a versatile point-depletion and radioactive-decay code for use in simulating nuclear fuel cycles and calculating the nuclide compositions and characteristics of materials contained therein. This code takes into account the transmutation of all the isotopes in the fuel. The core fission product inventories were determined for end-of-cycle conditions, assuming an equilibrium fuel cycle. These inventories are given in Table 15.0-8. The isotopes included in Table 15.0-8 are the controlling isotopes from considerations of thyroid dose (iodines) and from external dose due to immersion (noble gases).

The isotopic yields used in the calculations are from the data of NEDO-12154-1, utilizing the isotopic yield data for thermal fissioning of U-235 as the sole fissioning source. The change in fission product inventory resulting from the fissioning of other fissionable atoms has been reviewed. The results of this review indicated that inclusion of all fission source data would result in a small (less than 10%) change in the isotopic inventories.

For the accidents evaluated using Alternative Source Terms, the calculation of the core iodine fission product inventory was also modeled using computer code ORIGEN2 (Reference 1), based on reactor operation at 3586.6 MWth and an equilibrium 542.9 Effective Full Power Days (EFPD) eighteen month cycle design. The maximum of the 100 EFPD and End of Cycle values for each isotope were selected to generate the bounding isotopic core inventory activity and composition results. The output from ORIGEN2 is adjusted to 3658.3 MWt such that the inputs to the accident dose analyses remain bounded for the MUR core licensed thermal power level including measurement uncertainties.

15.0.9.2 Activities in the Fuel Pellet Cladding Gap

The fuel clad gap activities are defined differently for the different accidents which model gap activity releases. These accidents include the locked pump rotor, rod ejection, and fuel handling accidents. The specific gap activity model used for each of these events is described in the associated accident analysis discussion.

The NRC staff has completed its review of the revised Westinghouse fuel rod internal pressure design criteria and has decided on an acceptable amended criterion:

> "The internal pressure of the lead fuel rod in the reactor will be limited to a value below that which could cause (1) the diametrical gap to increase due to outward cladding creep during steady-state operation, and (2) extensive DNB propagation to occur."

WCAP-8963, "Safety Analysis for the Revised Fuel Internal Design Basis," was found to be acceptable to support the conclusion that an insignificant number of additional DNB events would occur during transients and accidents as a result of operating with fuel rod pressure (1) greater than nominal system pressure, and (2) limited by the above criterion.

For all Condition III and IV overpower events, the number of rods that are assumed to fail is less than 10%. Therefore, the analyses for the Byron/Braidwood OFA amendment (Amendment 30) are bounded by the analysis presented in the WCAP. The results presented in the WCAP are based on the detailed probability analysis performed to determine the maximum extent of core damage that could lead to DNB propagation. It was shown that the propagation mechanism causes only a small incremental increase in the percentage of rods in DNB. In view of the conservative nature of the failure propagation scheme and the small percentage increase in the number of failed rods, the potential increase in site release is inconsequential.

Although this effect, resulting from the revised fuel rod internal pressure design criterion, is small, it was factored into the number of rods predicted to fail.

15.0.9.3 Activities in the Primary Coolant

The accident dose analyses consider the releases of radioactive iodines and noble gases. The noble gas activity in the primary coolant is based on operation with 1.0 percent fuel defects and with no credit for stripping of gases from the primary coolant for processing by the gaseous waste processing system. The primary coolant noble gas activities are provided in Subsection 11.1 and are also repeated here in Table 15.0-9.

While Subsection 11.1 identifies the iodine concentrations associated with operation with 1.0 percent fuel defects, the iodine activity in the primary coolant is controlled to an equilibrium concentration of no more than 1.0 µCi/g of dose equivalent (DE) I-131. The determination of DE I-131 is based on five iodine isotopes: I-131, I-132, I-133, I-134, and I-135. In the event of an accident involving depressurization of the primary coolant system or reactor trip, the rate at which iodine enters the primary system is assumed to increase. This is referred to as an accident-initiated or concurrent iodine spike.

There is also the potential for an accident to occur during the time in which an iodine spike has already occurred, initiated by changes in plant operation (e.g., power reduction). This is referred to as a pre-existing iodine spike. The maximum iodine concentration associated with a pre-existing iodine spike is 60 µCi/g of DE I-131.

Table 15.0-10 lists the equilibrium iodine concentrations associated with 1.0 µCi/g of DE I-131, the equilibrium iodine appearance rates into the primary coolant, the iodine spike appearance rates (500 times equilibrium), and the iodine concentrations associated with 60 µCi/g of DE I-131.

15.0.9.4 Activities in the Secondary Coolant

It is assumed that there is primary to secondary coolant leakage that transports activity to the secondary system. However, there is no significant amount of noble gas in the secondary coolant since non-condensable gases entering the secondary system are rapidly expelled to the environment. The iodine concentration in the secondary coolant is controlled to be less than 0.1 µCi/g of DE I-131 (values for individual isotopes are identified in Table 15.0-10).

15.0.10 Residual Decay Heat

15.0.10.1 Total Residual Heat

Residual heat in a subcritical core is calculated for the loss of coolant accident per the requirements of Appendix K of 10 CFR 50.46 (Reference 3) as described in (References 4 and 5). These requirements include assuming infinite irradiation time before the core goes subcritical to determine fission product decay energy. For all other accidents, unless otherwise noted in the text, the same models are used except that fission product decay energy is based on core average exposure at the end of the equilibrium cycle.

15.0.10.2 Distribution of Decay Heat Following Loss of Coolant Accident

During a loss-of-coolant accident, the core is rapidly shut down by void formation or rod cluster control assembly insertion, or both, and a large fraction of the heat generation to be considered comes from fission product decay gamma rays. This heat is not distributed in the same manner as steady state fission power. Local peaking effects which are important for the neutron dependent part of the heat generation do not apply to the gamma ray contribution. The steady state factor of 97.4% which represents the fraction of heat generated within the clad and pellet drops to 95% for the hot rod in a loss-of-coolant accident.

For example, in the transient resulting from the postulated double ended break of the largest reactor coolant system pipe, 1/2 second after the break, about 30% of the heat generated in the fuel rods is from gamma ray absorption. The power shape, reducing the energy deposited in the hot rod at the expense of adjacent colder rods. A conservative estimate of this effect is a reduction of 10% of the gamma ray contribution or 3% of the total. Since the water density is considerably reduced at this time, an average of 98% of the available heat is deposited in the fuel rods, the remaining 2% being absorbed by water, thimbles, sleeves and grids. The net effect is a factor of 0.95 rather than 0.974, to be applied to the heat production in the hot rod.

15.0.11 Computer Codes Utilized

Summaries of some of the principal computer codes used in transient analyses are given below. Other codes, in particular very specialized codes in which the modeling has been developed to simulate one given accident, such as those used in the analysis of the reactor coolant system pipe break (Section 15.6), are summarized in their respective accident analyses sections. The codes used in the analysis of each transient have been listed in Table 15.0-2.

15.0.11.1 FACTRAN

FACTRAN calculates the transient temperature distribution in a cross section of a metal clad UO_2 fuel rod and the transient heat flux at the surface of the clad using as input the nuclear power and time-dependent coolant parameters (pressure, flow, temperature, and density). The code uses a fuel model which exhibits the following features simultaneously.

a. A sufficiently large number of radial space increments to handle fast transients such as rod ejection accidents.

b. Material properties which are functions of temperature and a sophisticated fuel-to-clad gap heat transfer calculation.

c. The necessary calculations to handle post-DNB transients: film boiling heat transfer correlations, zirconium-water reaction and partial melting of the materials.

FACTRAN is further discussed in Reference 6.

15.0.11.2 LOFTRAN

The LOFTRAN program is used for studies of transient response of a PWR system to specified perturbations in process parameters. LOFTRAN simulates a multiloop system by a model containing reactor vessel, hot and cold leg piping, steam generator (tube and shell sides) and the pressurizer. The pressurizer heaters, spray, relief and safety valves are also considered in the program. Point model neutron kinetics, and reactivity effects of the moderator, fuel, boron and rods are included. The secondary side of the steam generator utilizes a homogeneous, saturated mixture for the thermal transients and a water level correlation for indication and control. The reactor protection system is simulated to include reactor trips on high neutron flux, overtemperature ΔT, overpower ΔT, high and low pressure, low flow, and high pressurizer level. Control systems are also simulated including rod control, steam dump, feedwater control and pressurizer pressure control. The emergency core cooling system, including the accumulators is also modeled.

LOFTRAN is a versatile program which is suited to both accident evaluation and control studies as well as parameter sizing.

LOFTRAN also has the capability of calculating the transient value of DNBR based on the input from the core limits illustrated on Figure 15.0-1. The core limits represents the minimum value of DNBR as calculated for typical or thimble cell.

LOFTRAN is further discussed in Reference 7.

15.0.11.3 TWINKLE

The TWINKLE program is a multi-dimensional spatial neutron kinetics code, which was patterned after steady state codes presently used for reactor core design. The code uses an implicit finite-difference method to solve the two-group transient neutron diffusion equations in one, two and three dimensions. The code uses six delayed neutron groups and contains a detailed multi-region fuel-clad-coolant heat transfer model for calculating pointwise Doppler and moderator feedback effects. The code handles up to 2000 spatial points, and performs its own steady state initialization. Aside from basic cross section data and thermal-hydraulic parameters, the code accepts as input basic driving functions such as inlet temperature, pressure, flow, boron concentration, control rod motion, and others. Various edits are provided e.g., channel-wise power, axial offset, enthalpy, volumetric surge, pointwise power, and fuel temperatures.

The TWINKLE Code is used to predict the kinetic behavior of a reactor for transients which cause a major perturbation in the spatial neutron flux distribution.

TWINKLE is further discussed in Reference 8.

15.0.11.4 Deleted

15.0.11.5 Deleted

15.0.11.6 RADTRAD

RADTRAD is used to determine accident doses at the appropriate dose points cited in Regulatory Guide 1.183; the Exclusion Area Boundary (EAB), Low Population Zone (LPZ), and Control Room (CR). RADTRAD is a simplified model of **RAD**ionuclide **T**ransport and **R**emoval **A**nd **D**ose Estimation developed for, and endorsed by, the NRC as an acceptable methodology for reanalysis of the radiological consequences of design basis accidents.

In accident analyses RADTRAD is used to estimate the releases using the Alternative Source Term assumptions. The RADTRAD code uses a combination of tables and/or numerical models of source term reduction phenomena to determine the time-dependent dose at user-specified locations for a given accident scenario. The code system also provides the inventory, decay chain, and dose conversion factor tables needed for the dose calculation. The technical basis for the RADTRAD code is documented in Reference 15.

15.0.11.7 ARCON96

ARCON96 was developed to calculate relative concentrations in plumes from nuclear power plants at control room air intakes in the vicinity of the release point.

ARCON96 implements a straight-line Gaussian dispersion model with dispersion coefficients that are modified to account for low wind meander and building wake effects. Hourly, normalized concentrations (χ/Q) are calculated from hourly meteorological data. The hourly values are averaged to form χ/Qs for periods ranging from 2 to 720 hours in duration. The calculated values for each period are used to form cumulative frequency distributions.

The technical basis for the ARCON96 code is documented in Reference 16.

15.0.11.8 PAVAN

PAVAN estimates down-wind ground-level air concentrations for potential accidental releases of radioactive material from nuclear facilities. Options can account for variation in the location of release points, additional plume dispersion due to building wakes, plume meander under low wind speed conditions, and adjustments to consider non-straight trajectories. It computes an effective plume height using the physical release height which can be reduced by inputted terrain features.

Using joint frequency distributions of wind direction and wind speed by atmospheric stability, the program provides relative air concentration (χ/Q) values as functions of direction for various time periods at the exclusion area boundary (EAB) and the outer boundary of the low population zone (LPZ). Calculations of χ/Q values can be made for assumed ground-level releases or elevated releases from free-standing stacks. The χ/Q calculations are based on the theory that material released to the atmosphere will be normally distributed (Gaussian) about the plume centerline. A straight-line trajectory is assumed between the point of release and all distances for which χ/Q values are calculated.

The technical basis for the PAVAN code is documented in Reference 17.

15.0.11.9 VIPRE

The VIPRE computer program performs thermal-hydraulic calculations. The code calculates coolant density, mass velocity, enthalpy, void fractions, static pressure and DNBR distributions along flow channels within a reactor core.

The VIPRE code is described in Reference 18.

15.0.11.10 ANC

ANC is an advanced nodal code capable of two-dimensional and three-dimensional neutronics calculations. ANC is the reference model for certain safety analysis calculations, power distributions, peaking factors, critical boron concentrations, control rod worths, reactivity coefficients, etc. In addition, three-dimensional ANC validates one-dimensional and two dimensional results and provides information about radial (x-y) peaking factors as a function of axial position. It can calculate discrete pin powers from nodal information as well.

The ANC code is described in Reference 19.

15.0.11.11 LOFTTR2

LOFTTR2 is a derivative of the LOFTRAN program discussed in Subsection 15.0.11.2 with a more realistic break flow model, a 2-region steam generator secondary side, and an improved capability to simulate operator actions during a steam generator tube rupture (SGTR) event.

LOFTTR2 is further discussed in Reference 20.

15.0.12 Radiological Consequences

This chapter also analyzes the effects of postulated accidents with respect to radiological consequences. The analysis considers a broad spectrum of events. The radiological consequences of each accident are shown to be within the guidelines of 10 CFR 100, or, for those accidents evaluated using Alternative Source Terms, the limits of 10 CFR 50.67 and the guidance of Regulatory Guide 1.183.

Chapter 15 also documents a set of radiological consequence analyses utilizing Alternative Source Terms (AST) methodology per Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" for the following accidents:

Loss of Coolant Accident (LOCA),
Fuel Handling Accident (FHA),
Control Rod Ejection Accident (CREA),
Locked Rotor Accident (LRA)
Main Steam Line Break Accident (MSLBA),
Steam Generator Tube Rupture Accident (SGTRA),and
Boron Recycle Holdup Tank Failure (Atmospheric Release)

The analyses use design-basis assumptions and parameters to demonstrate the adequacy of the plant design with regard to the guidelines of 10 CFR 100 and 10 CFR 50.67.

The parameters and assumptions used for each analysis, as well as the results, are presented in tabular form for each limiting accident. The sequence of events for each transient is listed in tables in the applicable sections of this chapter. In addition, the accident sequences are provided in Figure 15.0-7 through 15.0-25. Tables 15.0-11 and 15.0-12 present the accident doses for the Exclusion Area Boundary (EAB), the Low Population Zone (LPZ), and the control room for the Byron/Braidwood Stations respectively. This format of table identification is used consistently within the text of Chapter 15.0 to identify Braidwood Station.

Figures 15.0-24, 15.0-25a and 15.0-25b symbolically show event sequences of steam generator tube rupture and loss-of-coolant accidents. Included with the events illustrated are operator actions and systems necessary to mitigate the effects of the accidents. The figures were prepared using a protection sequence event diagram format to illustrate accident sequences. A description of operator actions required for system operation along with a reference to a listing of process instrumentation available to the operator following an accident is noted on the diagrams.

The sequence of events for a steam generator tube rupture is given in Tables 15.6-6a and 15.6-6b. This information is included in Table 15.6-1 for large and small break loss-of-coolant accidents.

Tables 15.0-13 and 15.0-14 list the atmospheric dilution factors (χ/Q) used in the TID-14844 analyses. The 5th percentile atmospheric dilution factors are used in the analyses.

The assumptions and methodology for the TID-14844 radiological dose analysis are discussed in Attachment 15A.

Table 15.0-17 lists the χ/Q factors used in the AST analyses, as derived in Section 2.3. The AST radiological dose analyses assumptions and methodology are discussed in the applicable accident's Radiological Consequences sections of this chapter.

The effects of Optimized and VANTAGE 5 fuel on the radiological consequence evaluations presented in Chapter 15.0 were assessed. The effects of extended fuel burnup were considered.

Analyses of core and fuel gap fission product inventories were performed for burnups of 33,000 MWd/Mtu and 60,000 MWd/Mtu (Reference 13), as well as 33,000 MWd/Mtu and 48,000 MWD/Mtu (Reference 14).

Analyses of core inventories show that increasing burnup from 33,000 MWd/Mtu to 60,000 MWd/Mtu produces negligible changes in the fission product inventories of short-lived noble gases and iodines. The effect of increased burnup on long-lived Kr-85 is essentially a linear increase in core inventory (Reference 14).

Since Kr-85 is not a significant contributor to the radiological impact of postulated accidents, the results presented in Chapter 15.0 change only slightly due to the use of extended burnup fuel and remain well within the NRC regulatory limits.

Analyses show that the fraction of an isotope that is released to the fuel rod gap can increase with fuel burnup increasing from 33,000 to 60,000 MWd/Mtu (Reference 13). This analysis was performed assuming the peak fuel rod in terms of burnup during normal operations and was applied to the fuel handling accidents in Section 15.7. Radiological consequences analyses, of the six Alternative Source Term (AST) Design Basis Accidents (DBAs) that result in control room and offsite exposure, were performed to support a full-scope implementation of AST as described in Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors". The results of the plant-specific AST analyses used the guidance in RG 1.183 and met the requirements of 10CFR50.67 "Accident source term". Technical Information Document (TID) 14844, "Calculation of Distance Factors for Power and Test Reactor Sites," will continue to be used as the radiation dose basis for equipment qualification. Support of full scope implementation of AST consisted of the following steps.

- Analysis of the atmospheric dispersion for the radiological propagation pathways
- Calculation of offsite Exclusion Area Boundary (EAB) and Low Population Zone (LPZ), Control Room (CR), and, for LOCA only, Technical Support Center (TSC) personnel Total Effective Dose Equivalent (TEDE) doses
- Identification of the AST based on plant-specific analysis of a bounding core fission product inventory
- Calculation of fission product deposition rates and transport and removal mechanisms
- Calculation of the release fractions for the DBAs that result in the most significant CR and offsite doses (i.e., LOCA, FHA, CREA, LRA, MSLB, and SGTR)

The analysis assumptions for the transport, reduction, and release of the radioactive material from the fuel and the reactor coolant are consistent with the guidance provided in applicable appendices of RG 1.183 for the analyzed DBAs.

AST calculations for the LOCA, FHA, CREA, LRA, MSLB, and SGTR were prepared for the simulation of the radionuclide release, transport, removal, and dose estimates associated with the postulated accidents. The RADTRAD computer code developed for and endorsed by the NRC for AST analyses was used in the calculations. The RADTRAD program is a radiological consequence analysis code used to estimate post-accident doses at plant offsite locations and in the control room.

Vendor fuel supply data were used in the calculation of the reactor core fission products for RADTRAD analysis. The inventories were determined based on the licensed core power level prior to the MUR power uprate of 3586.6 megawatts thermal (MWt) and further adjusted to 102% (3658.3 MWt) in support of the AST evaluations. This core power level bounds the MUR core licensed power level including measurement uncertainties.

Control room and offsite atmospheric dispersion factors (χ/Qs) developed for the releases from each plant at the containment wall, plant vent, Steam Generator (SG) power operated relief valves (PORVs)/safety valves, and through a main steam line break were utilized. Offsite χ/Qs were calculated with the PAVAN computer code, using the guidance of Regulatory Guide 1.145, and control room χ/Qs were calculated with the ARCON96 computer code. The PAVAN and ARCON96 codes calculate relative concentrations in plumes from nuclear power plants at offsite locations and control room air intakes, respectively (refer to section 2.3). All of these codes have previously been used by the NRC in their safety reviews.

The Byron Station and Braidwood Station meteorological databases for the five-year period 1994-1998 were applied in the ARCON96 modeling analysis. Wind measurements at Byron Station were taken at 30 ft and 250 ft; and the vertical temperature difference was measured between 250 ft and 30 ft. Braidwood Station wind measurements were taken at 34 ft and 203 ft; and the vertical temperature difference was measured between 199 ft and 30 ft. “Calm” wind speeds at both stations were assigned a value of 0.4 mph (i.e., one-half the threshold value) per UFSAR Section 2.3.4. The minimum wind speed (i.e. wind threshold) was set to the ARCON96 default value of 0.5 m/sec in accordance with RG 1.194, Table A-2. Other ARCON96 parameters are provided in section 2.3.

The atmospheric dispersion factors (χ/Qs) utilized are as found in Sections 2.3 and Table 15.0-17.

The key inputs/assumptions used in the AST analysis for each of the six major accidents and their results/conclusions are presented in the applicable UFSAR sections.

Hence the results of the radiological consequence evaluations presented in Chapter 15.0 remain acceptable as a result of the use of extended burnup fuel.

15.0.13 Limiting Single Failures

The most limiting single failure of safety-related equipment, where one exists, is identified in each analysis description, and the consequences of this failure are described therein. In some instances, because of redundancy in protection equipment, no single failure which could adversely affect the consequences of the transient has been identified. The failure assumed in each analysis is listed in Table 15.0-15.

All accident and safety analyses in the UFSAR that require safety valves actuation assume operation of three pressurizer safety valves to limit increases in RCS pressure. The overpressure protection analysis is also based on operation of three safety valves.

15.0.14 Operator Actions

For most of the events analyzed in Chapter 15.0, the plant will be in a safe and stable hot standby condition following the automatic actuation of reactor trip. This condition will in fact be similar to plant conditions following any normal, orderly shutdown of the reactor. At this point, the actions taken by the operator would be no different than normal operating procedures. The exact actions taken, and the time these actions would occur, will depend on what systems are available (e.g., steam dump system, main feedwater system) and the plans for further plant operation. As a minimum, to maintain the hot stabilized condition, decay heat must be removed via the steam generators. The main feedwater system and the steam dump or atmospheric relief system could be used for this purpose. Alternatively, the auxiliary feedwater system and the steam generator safety valves, both of which are safety grade systems, may be used. Although auxiliary feed may be started manually, it will be automatically actuated, if needed, by one of the signals shown in Drawing 108D685, Sheet 15, such as low steam generator level. If hot standby conditions are maintained for an extended period of time, operator action may be required to transfer the auxiliary feedwater system to the long-term source of auxiliary feedwater. The time when such action is required depends on the design of the auxiliary feed system, but will be sufficiently long to permit operator action. Also, if the hot standby condition is maintained for an extended period of time (greater than approximately 18 hours), operator action may be required to add boric acid via the CVCS to compensate for xenon decay and to maintain shutdown margin. The CVCS could also be used to control pressurizer level according to the operating procedures. Again, the actions taken by the operator would be no different than during a normal plant shutdown.

For several events involving breaks in the reactor coolant system or secondary system piping, additional requirements for operator action can be identified as described in the following.

Following the hypothetical steamline break incident, a safety injection signal (generated a few seconds after the break) will cause main feedwater isolation to occur. The only source of water available to the faulted steam generator is then the auxiliary feedwater system. Following steamline isolation, steam pressure in the steamline with the faulted steam generator will continue to fall rapidly, while the pressure stabilizes in the remaining three steamlines. The indication of the different steam pressures will be available to the operator within a few seconds of steamline isolation. This will provide the information necessary to identify the faulted steam generator so that auxiliary feedwater to it can be isolated. Manual controls are provided in the control room for start and stop of the auxiliary feedwater pumps and for the control valves associated with the auxiliary feedwater system. The means for detecting the faulted steam generator and isolating auxiliary feedwater to it requires only the use of safety grade equipment available following the break. The removal of decay heat in the long-term, (following the initial cooldown) using the remaining steam generators, requires only the auxiliary feedwater system as a water source and the secondary system safety valves to relieve steam.

For a feedwater line break, emergency operating procedures provide guidance to the operator during long-term cooling to prevent pressurization of the reactor coolant system. The operator is provided with the necessary controls to start, operate, and shutdown the units to assure safe and reliable operation under normal and accident conditions.

For the boron dilution transient, the operator is required to open RWST isolation valves LCA-112D and E, and close CVCS isolation valves LCV-112B and C. This action will allow 2300 ppm borated water to enter the RCS and terminate the RCS dilution.

Operator action (both short-term and long-term) required for the various modes of ECCS operation to mitigate the consequences of a loss-of-coolant accident (LOCA) or steamline break, as well as other accident conditions, are presented in the Emergency Operating Procedures. These procedures discuss the alarm/indications available to the operator to lead him to take the appropriate actions. The discussion provided below, constitutes an outline of the operator action required following a LOCA or steamline break.

The primary function of the safety injection system (SIS) is to provide emergency core cooling (ECC) in the event of a LOCA resulting from a break in the primary reactor coolant system (RCS) or to provide emergency boration in the event of a steamline break accident resulting from a break in the secondary steam system.

ECC following a LOCA is divided into three phases:

a. Short-Term Core Cooling/Cold Leg Injection Phase

   The cold leg injection phase is defined as that period during which borated water is delivered from the refueling water storage tank (RWST) and accumulators to the RCS cold legs. During this phase, no operator actions are required to ensure proper ECCS operation.

b. Long-Term Core Cooling/Cold Leg Recirculation

   The cold leg recirculation phase is that period during which borated water is recirculated from the containment sump to the RCS cold legs. Operator actions are required to establish the cold leg recirculation phase. These actions are detailed in Table 6.3-7 and are not required prior to 10 minutes following event initiation.

c. Long-Term Core Cooling/Hot Leg Recirculation Phase

   The hot leg recirculation phase is that period during which borated water is recirculated from the containment sump to both the RCS hot legs and RCS cold legs. Operator actions required to establish hot leg recirculation are detailed until approximately 6.0 hours following event initiation.

The emergency boration following a steam break accident would occur only during the injection phase. The function of the SIS during this phase would be to inject borated water into the RCS with sufficient shutdown reactivity to compensate for the change in RCS volume and counteract any reactivity increase caused by the resulting cooldown. The SIS would continue to inject borated water from the RWST until the RCS conditions have stabilized, the accident has been identified as a steam break, and the criteria for safety injection termination are satisfied. The operator should then take action to terminate ECCS operation.

For a secondary system break, such as a steamline or feedwater line break, the operator is instructed to complete specific actions in the Westinghouse Owners Group Emergency Response Guidelines (ERG). These actions instruct the operator to identify the faulted steam generator and terminate auxiliary feedwater to it. When acceptable plant conditions exist (i.e., RCS subcooling, RCS pressure, pressurizer level, secondary heat sink), the operator is instructed to terminate SI. Postaccident monitoring instrumentation which meets appropriate safety criteria is provided to monitor the course of various postulated accidents, including feedwater line break and steamline break (see Section 7.5).

In regard to overpressurizing the reactor coolant system, the small break LOCA is less limiting than the steamline break for the following reasons:

a. A small break LOCA is a relatively slow transient with the RCS liquid remaining at saturated condition for an extended period of time.

b. In all cases due to a break in the RCS, the system depressurizes.

c. Even for very small break sizes, a large enough amount of water leaks from the system so that level would not return to the pressurizer for a relatively long period of time.

d. For small and large LOCAs, the operator utilizes the emergency core cooling system for long-term cooling so that the operator does not initiate normal cooldown procedures.

The Byron/Braidwood abnormal and emergency operating procedures provide instructions to the operator on appropriate post-LOCA manual actions, as follows:

a. The operator is instructed to align the ECCS to cold leg recirculation when the RWST level reaches the auto switchover level setpoint and to hot leg recirculation at 6.0 hours.

b. The operator is instructed to control the steam generator and pressure levels utilizing the respective level indications.

c. If acceptable plant conditions exist (i.e., RCS subcooling, RCS pressure, pressurizer level, secondary heat sink), the operator is instructed to terminate SI.

d. If leakage past the PORVs is indicated by valve position or by conditions in the pressurizer relief tank, the operator is instructed to isolate the pressurizer PORVs.

e. If the RCS low pressure setpoint for the reactor coolant pumps is reached and SI flow is being delivered, the operator is instructed to trip the RCPs.

All of the previous indications are located inside the control room (see Section 7.5).

For a steam generator tube rupture (SGTR), the operator is instructed to complete specific actions per the plant-specific emergency procedures addressing the SGTR accident. The emergency procedures are based on the latest revision of the Westinghouse Owners Group generic emergency response guidelines and are, therefore, consistent with the latest guidelines regarding SGTR mitigation. The transient is identified as an SGTR in the E-0 procedure. Procedure E-3 directs the operator to mitigate the transient and terminate the break flow for the purpose of preventing SG overfill. The tube rupture results in a reactor trip and safety injection due to low pressurizer pressure, and the E-3 procedure instructs the operator to perform the following major actions:

1. The operator identifies the accident as a SGTR in the E-0 procedure based on steam generator level rising in an uncontrolled manner and isolates AFW flow to the ruptured steam generator. After all remaining post-trip checks have been performed, the operator enters the E-3 procedure based on abnormal secondary plant radiation levels to mitigate and recover from the accident. The operator then completes the isolation of the secondary side of the ruptured steam generator. Manual controls for the isolation of the ruptured steam generator are located on the main control board.

2. The next major step has the operator cool down the RCS below the saturation temperature at the ruptured steam generator pressure by opening the steam dumps or PORVs on the intact steam generators.

3. The steam generator PORV controls and RCS temperature indication are both available in the control room to allow the operator to cool down the RCS.

4. After verifying adequate RCS subcooling exists, the operator uses the pressurizer spray valve or PORV to depressurize the RCS below the ruptured steam generator pressure to terminate the break flow.

5. The final major step is for the operator to terminate the safety injection flow to prevent repressurization of the RCS, reestablishment of the break flow, and potential overfilling of the ruptured steam generator.

6. After completing these actions, the tube rupture break flow has been terminated and the operator is ready to transition the plant to cold shutdown conditions utilizing the appropriate recovery procedure.

See Subsection 15.6.3 for further information regarding the SGTR analysis performed.

15.0.15 References

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TABLE 15.0-1

NUCLEAR STEAM SUPPLY SYSTEM POWER RATINGS

1.	NSSS thermal power output (MWt) (nominal pump power)	3672.0
2.	Nominal thermal power generated by the reactor coolant pumps (MWt)	14
	Maximum thermal power generated by the reactor coolant pumps (MWt)	20
3.	Core thermal power (MWt)	3658.0/3648.0[(a)]

(a) The Byron/Braidwood MUR-PU increased the licensed reactor core power level from 3586.6 MWt to 3645.0 MWt. A Conservative nominal value of 3658.0 MWt was used in the transient analyses supporting the MUR-PU. A conservative nominal value of 3648.0 MWt was used in the DNB analyses supporting the MUR-PU.

TABLE 15.0-2

SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED
(REACTIVITY COEFFICIENTS ASSUMED)

FAULTS	COMPUTER CODES UTILIZED	MODERATOR TEMPERATURE ($\Delta K/^{\circ}F$)	MODERATOR DENSITY ($\Delta\rho$/gm/cc)	DOPPLER	DNB CORRELATION	REVISED THERMAL DESIGN PROCEDURE
15.1 Increase in Heat Removal by the Secondary System						
- Feedwater System Malfunction Causing a Decrease in Feedwater Temperature	LOFTRAN, VIPRE, ANC	NA	0.43	Minimum EOL*	WRB-2	Yes
- Feedwater System Malfunction Causing an Increase in Feedwater Flow	LOFTRAN, VIPRE, ANC	NA	0.43	Minimum EOL*	WRB-2	Yes
- Excessive Increase in Secondary Steam Flow	LOFTRAN	NA	0 and 0.54	Minimum* and Maximum	WRB-2	Yes

TABLE 15.0-2 (Cont'd)

SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED
(REACTIVITY COEFFICIENTS ASSUMED)

FAULTS	COMPUTER CODES UTILIZED	MODERATOR TEMPERATURE ($\Delta K/°F$)	MODERATOR DENSITY ($\Delta\rho/gm/cc$)	DOPPLER	DNB CORRELATION	REVISED THERMAL DESIGN PROCEDURE
- Steam System Piping Failure at Zero Power	VIPRE, ANC, LOFTRAN	NA	Function of Moderator Density, See Subsection 15.1.5 (Figure 15.1-11)	See Subsection 15.1.5	WLOP	No
- Steam System Piping Failure at Full Power	VIPRE, ANC, LOFTRAN	NA	0.43	Minimum EOL	WRB-2	Yes
15.2 Decrease in Heat Removal by the Secondary System						

TABLE 15.0-2 (Cont'd)

FAULTS	COMPUTER CODES UTILIZED	MODERATOR TEMPERATURE ($\Delta K/°F$)	MODERATOR DENSITY ($\Delta\rho$/gm/cc)	DOPPLER	DNB CORRELATION	REVISED THERMAL DESIGN PROCEDURE
- Loss of External Electrical Load and/or Turbine Trip	LOFTRAN	0	NA	Minimum*	WRB-2	Yes (DNB) NA (press.)
- Loss of Non-Emergency A-C Power to the Station Auxiliaries	LOFTRAN	0	NA	Maximum*	NA	NA
- Loss of Normal Feedwater Flow	LOFTRAN	0	NA	Maximum*	NA	NA
- Feedwater System Pipe Break	LOFTRAN	0	NA	Minimum*	NA	NA
15.3 Decrease in Reactor Coolant System Flow Rate						
- Partial and Complete - Loss of Forced Reactor Coolant Flow	LOFTRAN, VIPRE	Figure 15.0-2	NA	Maximum*	WRB-2	Yes
- Reactor Coolant Pump Shaft Seizure (Locked) Rotor	LOFTRAN, VIPRE	Figure 15.0-2	NA	Maximum*	WRB-2	Yes

TABLE 15.0-2 (Cont'd)

FAULTS	COMPUTER CODES UTILIZED	MODERATOR TEMPERATURE ($\Delta K/°F$)	MODERATOR DENSITY ($\Delta\rho$/gm/cc)	DOPPLER	DNB CORRELATION	REVISED THERMAL DESIGN PROCEDURE
15.4 Reactivity and Power Distribution Anomalies						
- Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Sub-critical or Low Power Startup Condition	TWINKLE, FACTRAN, VIPRE	Refer to Subsection 15.4.1.2	Refer to Subsection 15.4.1.2	Refer to Subsection 15.4.1.2	ABB-NV (first grid span) WRB-2 (remaining grid spans)	No
- Uncontrolled Rod Cluster Assembly Bank Withdrawal at Power	LOFTRAN	Figure 15.0-2	N/A	Maximum and Minimum*	WRB-2	Yes

TABLE 15.0-2 (Cont'd)

FAULTS	COMPUTER CODES UTILIZED	MODERATOR TEMPERATURE ($\Delta K/^{\circ}F$)	MODERATOR DENSITY ($\Delta\rho$/gm/cc)	DOPPLER	DNB CORRELATION	REVISED THERMAL DESIGN PROCEDURE
- Spectrum of Rod Cluster Control Assembly Ejection Accidents	TWINKLE, FACTRAN	Refer to Subsection 15.4.8 min., max. feedback	Refer to Subsection 15.4.8 min., max. feedback	Refer to Subsection 15.4.8.2.1	NA	NA
15.5 Increase in Coolant Inventory						
- Inadvertent Operation of ECCS During Power Operation	LOFTRAN	0	NA	Maximum and Minimum*	WRB-2	Yes(DNB) NA(PZR-Fill)
15.6 Decrease in Reactor Coolant Inventory						
- Inadvertent Opening of a Pressurizer Safety or Relief Valve	LOFTRAN	0	NA	Minimum*	WRB-2	Yes
- Steam Generator Tube Rupture Offsite Dose Case	LOFTTR2	0	NA	Maximum*	NA	NA
- Steam Generator Tube Rupture Margin To Overfill Case	LOFTTR2	0	NA	Minimum*	NA	NA

TABLE 15.0-2 (Cont'd)

FAULTS	INITIAL NSSS THERMAL POWER OUTPUT (MWt)	REACTOR VESSEL COOLANT FLOW (GPM)	VESSEL AVERAGE TEMPERATURE (°F)	PRESSURIZER PRESSURE (PSIA)	PRESSURIZER*** WATER VOLUME (ft^3)	FEEDWATER ** TEMPERATURE (°F)
15.1 Increase in Heat Removal by the Secondary System						
- Feedwater System Malfunction Causing a Decrease in Feedwater Temperature	3672.0	386,000	589.5	2250	1061.2	449.2
- Feedwater System Malfunction Causing an Increase in Feed-water Flow	3672.0	386,000	589.5	2250	1061.2	449.2
- Excessive Increase in Secondary Steam Flow	3672.0	386,000	589.5	2250	1061.2	449.2
- Steam System Piping Failure at Zero Power	0 (Subcritical)	368,000	557	2250	693.84	100.0
- Steam System Piping Failure at Full Power	3672.0	386,000	589.5	2250	979.6	449.2

TABLE 15.0-2 (Cont'd)

FAULTS	INITIAL NSSS THERMAL POWER OUTPUT (MWt)	REACTOR VESSEL COOLANT FLOW (GPM)	VESSEL AVERAGE TEMPERATURE (°F)	PRESSURIZER PRESSURE (PSIA)	PRESSURIZER*** WATER VOLUME (ft^3)	FEEDWATER ** TEMPERATURE (°F)
15.2 Decrease in Heat Removal by the Secondary System						
- Loss of External Electrical Load and/or Turbine Trip	3672.0 (DNB) 3672.6 (RCS press.)	386,000 (DNB) 368,000 (RCS press.)	589.5 (DNB) 597.1 (RCS press. high) 578.9 (RCS press. low)	2250 (DNB) 2207 (RCS press.)	1142.9 (DNB) 1149.9 (RCS press.)	449.2 (DNB) 446.6 (RCS press.)
	3672.0 (MSS press.)	368,000 (MSS press.)	597.1 (MSS press.)	2207 (MSS press.)	1142.9 (MSS press.)	449.2 (MSS press.)
- Loss of Non-Emergency A-C Power to the Station Auxiliaries	3672.6	368,000	597.1 (high) 567.4 (low)	2293 (high) 2207 (low)	1149.9	448.4

TABLE 15.0-2 (Cont'd)

FAULTS	INITIAL NSSS THERMAL POWER OUTPUT (MWt)	REACTOR VESSEL COOLANT FLOW (GPM)	VESSEL AVERAGE TEMPERATURE (°F)	PRESSURIZER PRESSURE (PSIA)	PRESSURIZER*** WATER VOLUME (ft^3)	FEEDWATER ** TEMPERATURE (°F)
- Loss of Normal Feedwater Flow	3672.6	368,000	597.1 (high) 567.4 (low)	2293 (high) 2207 (low)	1149.9	448.4
- Feedwater System Pipe Break	3672.6	368,000	597.1	2293 (high) 2207 (low)	1067	448.4
15.3 Decrease in Reactor Coolant System Flow Rate						
- Partial and Complete Loss of Forced Reactor Coolant Flow	3672.0	386,000	589.5	2250	1061.2	449.2
- Reactor Coolant Pump Shaft Seizure/Locked Rotor (Peak RCS Pressure and Peak Clad Temperature Core)	3672.6	368,000	597.1	2293	1149.9	446.6
Reactor Coolant Pump Shaft Seizure/Locked Rotor (Rods-in-DNB core)	3672.0	386,000	589.5	2250	1061.2	449.2
15.4 Reactivity and Power Distribution Anomalies						
- Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Sub-critical or Low Power Startup Condition	0	163,686	557.0	2207	NA	NA

TABLE 15.0-2 (Cont'd)

FAULTS	INITIAL NSSS THERMAL POWER OUTPUT (MWt)	REACTOR VESSEL COOLANT FLOW (GPM)	VESSEL AVERAGE TEMPERATURE (°F)	PRESSURIZER PRESSURE (PSIA)	PRESSURIZER*** WATER VOLUME (ft^3)	FEEDWATER ** TEMPERATURE (°F)
- Uncontrolled Rod Cluster Assembly Bank Withdrawal at Power (DNB)	3672.0/2203.2/367.2	386,000	589.5/577.1/ 561.6	2250	1061.2/832.6/ 546.9	433.0/379.5/ 264.0
- Uncontrolled Rod Cluster Assembly Bank Withdrawal at Power (Peak Pressure)	293.8	368,000	569.2	2207	541.8	272
- Spectrum of Rod Cluster Control Assembly Ejection Accidents	0 and 3672.6	163,686 and 368,000	557 and 597.1	2207	NA	NA

TABLE 15.0-2 (Cont'd)

FAULTS	INITIAL NSSS THERMAL POWER OUTPUT (MWt)	REACTOR VESSEL COOLANT FLOW (GPM)	VESSEL AVERAGE TEMPERATURE (°F)	PRESSURIZER PRESSURE (PSIA)	PRESSURIZER*** WATER VOLUME (ft^3)	FEEDWATER ** TEMPERATURE (°F)
15.5 Increase in Coolant Inventory						
- Inadvertent Operation of ECCS During Power Operation	3672.0	386,000(DNB) 358,800 (PZR-Fill)	589.5(DNB) 572.0 (PZR-Fill)	2250(DNB) 2207 (PZR-Fill)	1142.9	449.2
15.6 Decrease in Reactor Coolant Inventory						
- Inadvertent Opening of a Pressurizer Safety or Relief Valve	3672.0	386,000	589.5	2250	1061.2	435.0
- Steam Generator Tube Rupture Offsite Dose Case	3672	368,000	588.0	2207	1061.2 UNIT 1 1066.6 UNIT 2	433.0 UNIT 1 435.0 UNIT 2
- Steam Generator Tube Rupture Margin to Overfill Case	3672	368,000	580.0 UNIT 1 575.0 UNIT 2	2207	1061.2	433.0 UNIT 1 435.0 UNIT 2

*Reference Figure 15.0-3 Maximum refers to lower curve, minimum refers to upper curve, and minimum EOL refers to least negative EOL curve.

**Evaluations or analyses have been performed to support a nominal feedwater temperature window of 433°F to 449.2°F for Unit 1 and 435°F to 449.2°F for Unit 2.

***Evaluations have been performed to allow a decrease in nominal pressurizer water volume from 1066.6 ft^3 to 1061.2 ft^3. This reduction in volume is due to updated volume calculation, not a physical change. The evaluations also addressed initial volumes that included uncertainties based upon the reduced nominal volume.

NA - Not Applicable
BOC - Beginning of Cycle
EOC - End of Cycle

TABLE 15.0-3

NOMINAL VALUES OF PERTINENT PLANT PARAMETERS
UTILIZED IN ACCIDENT ANALYSES (BWI STEAM GENERATORS)

Thermal Output of NSSS (MWt)*	3672.0
Core Inlet Temperature (°F)	541.4 (Low) 555.1 (high)
Vessel Average Temperature (°F)	**575.0 (low) 588.0(high)
Reactor Coolant System Pressure (psia)	2250
Reactor Coolant Flow per Loop (gpm)	92,000 (TDF) 96,500 (MMF)
Total Reactor Coolant Flow (10^6 lb/hr)	140.0 (TDF, low T_{avg}) 137.4 (TDF, high T_{avg}) 146.8 (MMF, low T_{avg}) 144.1 (MMF, high T_{avg})
Steam Flow from NSSS (10^6 lb/hr)	15.98/16.36 (low T_{avg}, low/high T_{feed}, 0% SGTP) 15.98/16.35 (low T_{avg}, low/high T_{feed}, 5% SGTP) 16.06/16.43 (high T_{avg}, low/high T_{feed}, 0% SGTP) 16.05/16.43 (high T_{avg}, low/high T_{feed}, 5% SGTP)
Steam Pressure at Steam Generator Outlet (psia)	897 (low T_{avg}, 0% SGTP) 891 (low T_{avg}, 5% SGTP) 1008 (high T_{avg}, 0% SGTP) 1002 (high T_{avg}, 5% SGTP)
Assumed Feedwater Temperature at Steam Generator Inlet (°F)	433.0 (low) 449.2 (high)
Average Core Heat Flux (Btu/hr-ft^2)	210,815
Maximum Average Steam Generator Tube Plugging Level (%)	5
Maximum Steam Generator Tube Plugging Level in any Steam Generator (%)	5

*See Table 15.0-2

**Low Vessel Average Temperature is limited to 580.0 °F per SGTR analysis.

TABLE 15.0-4

NOMINAL VALUES OF PERTINENT PLANT PARAMETERS
UTILIZED IN ACCIDENT ANALYSES (D5 STEAM GENERATORS)

Thermal Output of NSSS (MWt)*	3672.0
Core Inlet Temperature (°F)	541.4 (low) 555.1 (high)
Vessel Average Temperature (°F)	575.0 (low) 588.0(high)***
Reactor Coolant System Pressure (psia)	2250
Reactor Coolant Flow per Loop (gpm)	92,000 (TDF) 96,500 (MMF)
Total Reactor Coolant Flow (10^6 lb/hr)	140.0 (TDF, low T_{avg}) 137.4 (TDF, high T_{avg}) 146.8 (MMF, low T_{avg}) 144.1 (MMF, high T_{avg})
Steam Flow from NSSS (10^6 lb/hr)	16.01/16.34 (low T_{avg}, low/high T_{feed}, 0% SGTP) 16.00/16.32 (low T_{avg}, low/high T_{feed}, 10% SGTP) 16.09/16.42 (high T_{avg}, low/high T_{feed}, 0% SGTP) 16.07/16.39 (high T_{avg}, low/high T_{feed}, 10% SGTP)
Steam Pressure at Steam Generator Outlet (psia)	837/829 (low T_{avg}, low/high T_{feed}, 0% SGTP) 809/802 (low T_{avg}, low/high T_{feed}, 10% SGTP) 953/945 (high T_{avg}, low/high T_{feed}, 0% SGTP) 923/914 (high T_{avg}, low/high T_{feed}, 10% SGTP)
Assumed Feedwater Temperature at Steam Generator Inlet (°F)	435.0 (low) 449.2 (high)
Average Core Heat Flux (Btu/hr-ft^2)	210,815
Maximum Average Steam Generator Tube Plugging Level (%)	10**
Maximum Steam Generator Tube Plugging Level in any Steam Generator (%)	10**

* See Table 15.0-2

** LBLOCA is currently limited to 5% to account for updated TCD assessment.

*** LBLOCA is currently limited to 583.5 °F to account for updated TCD assessment.

TABLE 15.0-5

TRIP POINTS AND TIME DELAYS TO TRIP ASSUMED IN ACCIDENT ANALYSES

TRIP FUNCTION	LIMITING TRIP POINT ASSUMED IN ANALYSIS	TIME DELAYS (SECONDS)
Power range high neutron flux, high setting	118%	0.5
Power range high neutron flux, low setting	35%	0.5
Overtemperature ΔT	Variable see Figure 15.0-1	8.0*
Overpower ΔT	Variable see Figure 15.0-1	8.0*
High pressurizer pressure	2456 psig	2.0
Low pressurizer pressure	1845 psig	2.0
Low reactor coolant flow (From loop flow detectors)	85.1% loop flow	1.0
Underfrequency trip	54.0 Hz	0.6
Undervoltage trip	Not Applicable	1.5
Turbine trip	Not applicable	2.0
Low-low steam generator level**	10% of narrow range level span(Unit 1 loss of normal feed) 0% of narrow range level span (Unit 1 feedline break) 28.6% of narrow range level span (Unit 2 loss of normal feed) 18.6% of narrow range level span (Unit 2 feedline break)	2.0

TABLE 15.0-5 (Cont'd)

TRIP POINTS AND TIME DELAYS TO TRIP
ASSUMED IN ACCIDENT ANALYSES

TRIP FUNCTION	LIMITING TRIP POINT ASSUMED IN ANALYSIS	TIME DELAYS (SECONDS)
High steam generator level trip of the feedwater pumps and closure of feedwater system valves, and turbine trip	100% of narrow range level span	2.5

* Total response time (including RTD time response, trip circuit delay time, electronic filtering and channel electronics delay) from the time the temperature difference in the coolant loops exceeds the trip setpoint until the rods are free to fall, including time for the trip breakers to open.
** See Subsections 15.2.6, 15.2.7, and 15.2.8 for details.

TABLE 15.0-6

DETERMINATION OF MAXIMUM OVERPOWER TRIP POINT - POWER RANGE NEUTRON FLUX CHANNEL - BASED ON NOMINAL SETPOINT CONSIDERING INHERENT INSTRUMENT ERRORS

VARIABLE	ACCURACY OF MEASUREMENT OF VARIABLE (% ERROR)	EFFECT ON THERMAL POWER DETERMINATION (% ERROR)	
		(ESTIMATED)	(ASSUMED)
Calorimetric Errors in the Measurement of Secondary System Thermal Power:			
Feedwater temperature	± 0.5		
Feedwater pressure (small correction on enthalpy)	± 0.5	± 0.3	
Steam pressure (small correction on enthalpy)	± 2		
Feedwater flow	± 1.25	1.25	
Assumed Calorimetric Error (% of rated power)			± 2 = a
Axial power distribution effects on total ion chamber current			
Estimated Error (% of rated power)		3	
Assumed Error (% of rated power)			± 5 = b

TABLE 15.0-6 (Cont'd)

VARIABLE	ACCURACY OF MEASUREMENT OF VARIABLE (% ERROR)	EFFECT ON THERMAL POWER DETERMINATION (% ERROR) (ESTIMATED)	(ASSUMED)
Instrumentation channel drift and setpoint reproducibility			
Estimated Error (% of rated power)		1	
Assumed Error (% of rated power)			$\pm$ 2 = c
Total assumed error in setpoint $\Sigma = a + b + c$			$\pm$ 9

	Percent of Rated Power
Nominal Setpoint	109
Maximum overpower trip point assuming all individual errors are simultaneously in the most adverse direction	118

TABLE 15.0-7

PLANT SYSTEMS AND EQUIPMENT CREDITED FOR TRANSIENTS AND ACCIDENT CONDITIONS

	INCIDENT	REACTOR TRIP FUNCTIONS	ESF ACTUATION FUNCTIONS	OTHER EQUIPMENT	ESF EQUIPMENT
15.1	INCREASE IN HEAT REMOVED BY THE SECONDARY SYSTEM				
	Feedwater system malfunction that results in a decrease in feedwater temperature	Power range high neutron flux, overpower ΔT, overtemperature ΔT	SI on low pressurizer pressure, feedwater isolation on SI	Feedwater isolation valves	NA
	Feedwater system malfunctions that result in an increase in feedwater flow	Power range high neutron flux, overpower ΔT, overtemperature ΔT	Feedwater isolation and turbine trip on hi-hi steam generator water level	Feedwater isolation valves, steam generator safety valves	NA
	Excessive increase in secondary steam flow	Power range high neutron flux, overtemperature ΔT, overpower ΔT, low pressurizer pressure	NA	Pressurizer safety valves, steam generator safety valves	NA
	Steam system piping failure	Power range high neutron flux, SI, low pressurizer pressure, overpower ΔT	SI on low steamline pressure or low pressurizer pressure, steamline isolation on low steamline pressure, feedwater isolation on SI	Feedwater isolation valves, steamline isolation valves	Auxiliary Feedwater System, Safety Injection System, Emergency Power System

TABLE 15.0-7 (Cont'd)

	INCIDENT	REACTOR TRIP FUNCTIONS	ESF ACTUATION FUNCTIONS	OTHER EQUIPMENT	ESF EQUIPMENT
15.2	DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM				
	Loss of external electrical load/turbine trip	High pressurizer pressure overtemperature ΔT, lo-lo steam generator water level	NA	Pressurizer safety valves, steam generator safety valves	NA
	Loss of non-emergency ac power to the station auxiliaries	Lo-lo steam generator water level	AF initiation on lo-lo steam generator water level	Steam generator safety valves, pressurizer safety valves	Auxiliary Feedwater System
	Loss of normal feedwater flow	Lo-lo steam generator water level	AF initiation on lo-lo steam generator water level	Steam generator safety valves, pressurizer safety valves	Auxiliary Feedwater System
	Feedwater system pipe break	Lo-lo steam generator water level, SI	SI on low steamline pressure or low pressurizer pressure, AF initiation on lo-lo steam generator water level, steamline isolation on SI, feedwater isolation on SI	Steamline isolation valves, feedwater isolation valves, pressurizer safety valves, steam generator safety valves	Auxiliary Feedwater System, Safety Injection System
15.3	DECREASE IN REACTOR COOLANT SYSTEM FLOW RATE				
	Partial and complete loss of forced reactor coolant flow	Low reactor coolant loop flow, reactor coolant pump undervoltage and underfrequency	NA	Steam generator safety valves	NA
	Reactor coolant pump shaft seizure (locked rotor)	Low reactor coolant loop flow	NA	Pressurizer safety valves, steam generator safety valves	NA

TABLE 15.0-7 (Cont'd)

	INCIDENT	REACTOR TRIP FUNCTIONS	ESF ACTUATION FUNCTIONS	OTHER EQUIPMENT	ESF EQUIPMENT
15.4	REACTIVITY AND POWER DISTRIBUTION ANOMALIES				
	Uncontrolled rod cluster control assembly bank withdrawal from a sub-critical or low power startup condition	Power range high neutron flux (low setpoint), source range high neutron flux	NA	NA	NA
	Uncontrolled rod cluster control assembly bank withdrawal at power	Power range high neutron flux, overtemperature ΔT, high pressurizer pressure	NA	Pressurizer safety valves, steam generator safety valves	NA
	Dropped rod cluster control assembly	Manual	NA	Pressurizer safety valves, steam generator safety valves	NA
	Single rod cluster control assembly withdrawal at full power	Overtemperature ΔT	NA	NA	NA
	Rod cluster control assembly misalignment	Manual	NA	NA	NA
	Chemical and volume control system mal-function that results in a decrease in boron concentration in the reactor coolant	Source range high neutron flux, power range high neutron flux (high and low setpoint), overtemperature ΔT	NA	"Boron Dilution Alert" alarms	NA
	Spectrum of rod cluster control assembly ejection accidents	Power range high neutron flux (high and low setpoint), high positive neutron flux rate	NA	Pressurizer safety valves	NA

TABLE 15.0-7 (Cont'd)

	INCIDENT	REACTOR TRIP FUNCTIONS	ESF ACTUATION FUNCTIONS	OTHER EQUIPMENT	ESF EQUIPMENT
15.5	INCREASE IN REACTOR COOLANT INVENTORY				
	Inadvertent operation of the ECCS during power operation	Low pressurizer pressure, SI	SI on low pressurizer pressure, AF initiation on SI	Pressurizer Power Operated Relief Valves and Safety Valves, Steam Generator Safety Valves	Safety Injection System, Auxiliary Feedwater System
15.6	DECREASE IN REACTOR COOLANT INVENTORY				
	Inadvertent opening of a pressurizer safety or relief valve	Low pressurizer pressure, overtemperature ΔT	NA	NA	NA
	Steam generator tube rupture	Low pressurizer pressure, overtemperature ΔT, SI	SI on low pressurizer pressure, AF initiation on SI or loss of offsite power	Service Water, Component Cooling Water, Steam Generator Safety Valves and Power Operated Relief Valves, Main Steam Isolation Valves, Pressurizer Power Operated Relief Valves, Main Steam-Line Area Radiation Monitors, Instrument Air System	Emergency Core Cooling System, Auxiliary Feedwater System, Emergency Power System, AF Accumulator Tanks

TABLE 15.0-7 (Cont'd)

INCIDENT	REACTOR TRIP FUNCTIONS	ESF ACTUATION FUNCTIONS	OTHER EQUIPMENT	ESF EQUIPMENT
Loss of coolant accidents resulting from the spectrum of postulated piping breaks within the reactor coolant pressure boundary	Low pressurizer pressure	SI on low pressurizer pressure or high containment pressure, AF initiation on SI	Service Water System, Component Cooling Water System, Steam Generator Safety and/or Relief Valves	Emergency Core Cooling System, Auxiliary Feedwater System, Containment Heat Removal System, Emergency Power System

TABLE 15.0-8

IODINE AND NOBLE GAS INVENTORY IN REACTOR CORE*
(For use with TID-14844 Based Analyses)

ISOTOPE	CORE ACTIVITY (CURIES)

Security - Related Information Withheld Under 10 CFR 2.390

* Based on end of equilibrium fuel cycle and power level of 3658.3 MWt. This core power level bounds the MUR core licensed power level including measurement uncertainties.

TABLE 15.0-9

PRIMARY COOLANT NOBLE GAS ACTIVITY BASED ON 1% FUEL DEFECTS
(For Use with TID-14844 Based Analyses and AST Analyses Using Regulatory Guide 1.183)

ISOTOPE	CONCENTRATION (μCi/gm)
Kr-85m	1.8
Kr-85	7.11
Kr-87	1.15
Kr-88	3.35
Xe-131m	3.31
Xe-133m	3.65
Xe-133	251
Xe-135m	0.488
Xe-135	7.72
Xe-138	0.663

TABLE 15.0-10

IODINE ACTIVITY IN THE PRIMARY AND SECONDARY COOLANT
(For use with TID-14844 Based Analysis and AST Analyses Using Regulatory Guide 1.183)

Primary Coolant

Isotope	Equilibrium Concentration (1.0 µCi/gm DE I-131) µCi/gm	Equilibrium Appearance Rate, Ci/min	Iodine Spike Appearance Rate,* Ci/min	Pre-Existing Iodine Spike (60 µCi/gm DE I-131) µCi/gm
I-131	0.742	0.416	208	44.5
I-132	0.979	1.754	877	58.7
I-133	1.350	0.923	462	81.0
I-134	0.243	0.926	463	14.6
I-135	0.842	0.826	413	50.5

Secondary Coolant

Isotope	Equilibrium Concentraion (0.1 µCi/gm DE I-131) µCi/gm
I-131	0.0742
I-132	0.0979
I-133	0.1350
I-134	0.0243
I-135	0.0842

*Based on spike factor of 500. The SGTR analysis uses a spike factor of 335 consistent with Regulatory Guide 1.183.

TABLE 15.0-11

POTENTIAL DOSES DUE TO ACCIDENTS

POTENTIAL DOSES DUE TO ACCIDENTS USING AST

AST Dose Results (rem TEDE)						
Accident	**Exclusion Area Boundary**		**Low Population Zone**		**Control Room**	
	Dose	Limit	Dose	Limit	Dose	Limit
Loss of Coolant Accident	15.03	25	4.99	25	4.94	5.0
Main Steam Line Break Accident	0.146 (1) 0.201 (2)	25 (1) 2.5 (2)	0.083 (1) 0.459 (2)	25 (1) 2.5 (2)	0.264 (1) 0.654 (2)	5.0
Control Rod Ejection Accident	5.358	6.3	2.278	6.3	4.538	5.0
Locked Rotor Accident	1.679	2.5	0.602	2.5	2.790	5.0
Steam Generator Tube Rupture Accident	3.7 (1) 2.1 (3)	25 (1) 2.5 (3)	0.69 (1) 0.41 (3)	25 (1) 2.5 (3)	2.0 (1) 0.56 (3)	5.0
Fuel Handling Accident	5.31	6.3	0.94	6.3	4.40	5.0
Recycle Holdup Tank Failure	0.852	6.3	0.152	6.3	0.459	5.0

Note (1): Pre-accident 60 µCi/gm DEI spike
Note (2): Accident initiated 500 times equilibrium iodine release rate spike
Note (3): Accident initiated 335 times equilibrium iodine release rate spike

TABLE 15.0-11 (Cont'd)
POTENTIAL DOSES DUE TO ACCIDENTS

POTENTIAL DOSES DUE TO ACCIDENTS USING TID-14844 (10 CFR 100):

		DOSE (2 HOURS) AT EXCLUSION AREA BOUNDARY (445 meters)		DOSE (COURSE OF ACCIDENT) AT LOW POPULATION ZONE (4828 meters)	
POSTULATED ACCIDENT	UFSAR SECTION	THYROID (REM)	WHOLE BODY (REM)	THYROID (REM)	WHOLE BODY (REM)
Process Gas System Rupture	15.7.1	0	5.4 (-1)	0	2.0 (-2)
Radioactive Liquid Waste System Failure	15.7.3				
Spent Resin Tank		4.5 (-1)	1.6 (-4)	1.4 (-2)	4.7 (-6)
10CFR100 limits		300	25	300	25

Note: 2.89 (+1) = 2.89 x 10^1

TABLE 15.0-12

POTENTIAL DOSES DUE TO ACCIDENTS

POTENTIAL DOSES DUE TO ACCIDENTS USING AST

AST Dose Results (rem TEDE)						
Accident	Exclusion Area Boundary		Low Population Zone		Control Room	
	Dose	Limit	Dose	Limit	Dose	Limit
Loss of Coolant Accident	15.03	25	4.99	25	4.94	5.0
Main Steam Line Break Accident	0.146 (1) 0.201 (2)	25 (1) 2.5 (2)	0.083 (1) 0.459 (2)	25 (1) 2.5 (2)	0.264 (1) 0.654 (2)	5.0
Control Rod Ejection Accident	5.358	6.3	2.278	6.3	4.538	5.0
Locked Rotor Accident	1.679	2.5	0.602	2.5	2.790	5.0
Steam Generator Tube Rupture Accident	3.7 (1) 2.1 (3)	25 (1) 2.5 (3)	0.69 (1) 0.41 (3)	25 (1) 2.5 (3)	2.0 (1) 0.56 (3)	5.0
Fuel Handling Accident	5.31	6.3	0.94	6.3	4.40	5.0
Recycle Holdup Tank Failure	0.852	6.3	0.152	6.3	0.459	5.0

Note (1): Pre-accident 60 µCi/gm DEI spike
Note (2): Accident initiated 500 times equilibrium iodine release rate spike
Note (3): Accident initiated 335 times equilibrium iodine release rate spike

TABLE 15.0-12 (Cont'd)

POTENTIAL DOSES DUE TO ACCIDENTS

Potential Doses Due To Accidents Using TID-14844 (10 CFR 100):

		DOSE (2 HOURS) AT EXCLUSION AREA BOUNDARY (485 meters)		DOSE (COURSE OF ACCIDENT) AT LOW POPULATION ZONE (1811 meters)	
POSTULATED ACCIDENT	UFSAR SECTION	THYROID (REM)	WHOLE BODY (REM)	THYROID (REM)	WHOLE BODY (REM)
Process Gas System Rupture	15.7.1	0	7.3 (-1)	0	7.0 (-2)
Radioactive Liquid Waste System Failure	15.7.3				
Spent Resin Tank		6.1 (-1)	2.1 (-4)	5.6 (-2)	2.0 (-5)
10CFR100 limits		300	25	300	25

Note: 3.90 (+1) = 3.90 x 10^{1}

TABLE 15.0-13

ACCIDENT ATMOSPHERIC DILUTION FACTORS (X/Q)* AT

EXCLUSION AREA BOUNDARY AND LOW POPULATION ZONE FOR THE BYRON STATION
(using TID-14844 Analyses)

TIME PERIOD (Hours)	EXCLUSION AREA BOUNDARY (445 Meters)		LOW POPULATION ZONE (4828 Meters)	
	5th PERCENTILE	50th PERCENTILE	5th PERCENTILE	50th PERCENTILE
0-2	5.7 (-4)	6.5 (-5)		
0-8			1.7 (-5)	1.5 (-6)
8-24			2.4 (-6)	3.1 (-7)
24-96			1.1 (-6)	1.5 (-7)
96-720			7.6 (-7)	1.6 (-7)

*X/Q values, expressed in sec/m^3, are based on hourly onsite meteorological data for the period of record, January 1974 - December 1976.

Note: 5.7 (-4) = 5.7×10^{-4}

TABLE 15.0-14

ACCIDENT ATMOSPHERIC DILUTION FACTORS (X/Q)* AT

EXCLUSION AREA BOUNDARY AND LOW POPULATION ZONE FOR THE BRAIDWOOD STATION
(using TID-14844 Analyses)

TIME PERIOD (Hours)	EXCLUSION AREA BOUNDARY (485 Meters)		LOW POPULATION ZONE (1811 Meters)	
	5th PERCENTILE	50th PERCENTILE	5th PERCENTILE	50th PERCENTILE
0-2	7.7 (-4)	7.1 (-5)		
0-8			7.1 (-5)	6.6 (-6)
8-24			1.4 (-5)	1.8 (-6)
24-96			7.1 (-6)	8.5 (-7)
96-720			4.1 (-6)	1.0 (-6)

*X/Q values, expressed in sec/m^3, are based on hourly onsite meteorological data for the period of record, January 1974 - December 1976.

Note: 7.7 (-4) = 7.7×10^{-4}

TABLE 15.0-15

SINGLE FAILURES ASSUMED IN ACCIDENT ANALYSES

EVENT DESCRIPTION	WORST FAILURE ASSUMED
NSSS:	
Feedwater temperature reduction	One Protection Train
Excessive feedwater flow	One Protection Train
Excessive steam flow	(1)
Inadvertent secondary depressurization	One Safety Injection Train
Steam system piping failure	One Safety Injection Train
Steam pressure regulatory malfunction	(2)
Loss of external load	One Protection Train
Turbine Trip	One Protection Train
Inadvertent closure of MSIV	One Protection Train
Loss of condenser vacuum	One Protection Train
Loss of ac power	One Aux, FW Train
Loss of normal feedwater	One Aux, FW Train
Feedwater system pipe break	One Aux, FW Train
Partial loss of forced reactor coolant flow	One Protection Train
Complete loss of forced reactor coolant flow	One Protection Train
RCP locked rotor	One Protection Train
RCP shaft break	One Protection Train
RCCA bank withdrawal from subcritical	One Protection Train
RCCA bank withdrawal at power	One Protection Train
Dropped RCCA, dropped RCCA bank	Nuclear Instrumentation System
Statically misaligned RCCA	(3)
Single RCCA withdrawal	One Protection Train
Flow controller malfunction	(2)
Uncontrolled boron dilution	Standby charging pump is operating
Improper fuel loading	(3)
RCCA ejection	One Protection Train
Inadvertent ECCS operation at power	One Protection Train
Increase in RCS inventory	One Protection Train
BWR transients	(2)
Inadvertent RCS depressurization	One Protection Train
Failure of small lines carrying primary coolant outside containment	(3)
BWR piping failures	(2)

TABLE 15.0-15 (Cont'd)

EVENT DESCRIPTION	WORST FAILURE ASSUMED
Steam generator tube rupture	
Offsite dose case	One SG PORV on ruptured steam generator (F.O.)
Margin to overfill case	One SG PORV on intact steam generator (F.C.) (6)
Spectrum of LOCA	
Small Break	One Electrical Train
Large Break	One SI Train (5)
For BOP:	
Break in instrument line or other lines from reactor coolant pressure boundary that penetrate containment (Subsection 15.6.2)	(3)
Radioactive gas waste system leak or failure (Subsection 15.7.1)	(1)
Radioactive Liquid Waste System Leak or Failure (Atmospheric Release)	
Spent Resin Storage Tank	(1)
Boron Recycle Holdup Tank	One Protection Train (5)
Postulated radioactive releases due to liquid tanks failures (Ground Release) (Subsection 15.7.3)	(1)
Postulated fuel handling accident inside Spent Fuel Storage Building (Subsection 15.7.4.2.1)	One Protection Train (4)
Postulated fuel handling accident inside containment (Subsection 15.7.4.2.2)	One Protection Train (4)
Spent fuel cask drop accidents (Subsection 15.7.5)	(1)

NOTES:

(1) No protective action required.
(2) Not applicable to Byron/Braidwood.
(3) No transient analysis involved.
(4) Only one of the two filtration trains was considered in determining offsite doses.
(5) For determining control room dose one train of the control room filtration system is assumed to fail to realign to the emergency mode of operation.
(6) Failures, active or passive, that result in only two SG PORVs available for RCS cooldown on intact steam generators are bounding.

B/B – UFSAR

TABLE 15.0-16

REACTOR CORE NUCLIDE INVENTORY USING AST

Security - Related Information Withheld Under 10 CFR 2.390

TABLE 15.0-17

BYRON-BRAIDWOOD MAXIMUM SITE-UNIT χ/Q SUMMARY

	Release Path		Recommended χ/Q (sec/m3)					
Model	Release Point	Receptor/ Intake	0-2 hr	2-8 hr (1)	8-24 hr	1-4 day	4-30 day	Notes
ARCON96	Containment Wall	CR Fresh Air	1.73E-03	1.24E-03	5.23E-04	3.55E-04	2.62E-04	Diffuse area source per RG 1.194 Sec. 3.2.4
ARCON96	Containment Wall	CR Turbine Building Emergency Air	1.01E-03	7.25E-04	3.07E-04	2.07E-04	1.46E-04	
ARCON96	Plant Vent	CR Fresh Air	2.22E-03	1.80E-03	7.20E-04	4.75E-04	3.81E-04	Building area perpendicular to wind utilized per RG 1.194 Table A-2
ARCON96	Plant Vent	CR Turbine Building Emergency Air	2.46E-03	1.92E-03	8.14E-04	5.52E-04	4.40E-04	
ARCON96	PORVs/Safety Valves	CR Fresh Air	1.77E-03	1.52E-03	6.98E-04	4.72E-04	3.50E-04	Includes factor of 5 reduction for vertical uncapped release per RG 1.194 Sec. 6
ARCON96	PORVs/Safety Valves	CR Turbine Building Emergency Air	8.14E-04 (2)	6.98E-04	3.12E-04	1.95E-04	1.67E-04	
ARCON96	MSLB	CR Fresh Air	3.205E-03	2.735E-03	1.1920E-03	8.205E-04	6.60E-04	"Taut String Length" to Fresh Air Intake and building area perpendicular to wind utilized per RG 1.194
ARCON96	MSLB	CR Turbine Building Emergency Air	1.70E-02	1.46E-02	6.68E-03	4.48E-03	3.31E-03	Building area perpendicular to wind utilized per RG 1.194
PAVAN	Outer Containment Wall	EAB	5.36E-04 (N)	2.65E-04 (SE)	1.89E-04 (SE)	9.04E-05 (SE)	3.16E-05 (SE)	In compliance with RG 1.145 (3)
PAVAN	Midpoint between two reactors	LPZ	9.32E-05 (ESE)	4.50E-05 (ESE)	3.12E-05 (ESE)	1.41E-05 (ESE)	4.54E-06 (ESE)	
PAVAN	Outer Containment Wall	EAB	6.18E-04 (SE)	3.08E-04 (SE)	2.17E-04 (SE)	1.02E-04 (SE)	3.44E-05 (SE)	In compliance with RG 1.23 Rev. 1(4)
PAVAN	Midpoint between two reactors	LPZ	1.10E-04 (W)	5.13E-05 (W)	3.51E-05 (W)	1.53E-05 (W)	4.68E-06 (W)	

(1) χ/Q values for the PAVAN results are for 0-8 hour time period.
(2) A slightly more conservative value of 8.16E-04 was used in the MSLB, LRA, and CREA DBA calculations.
(3) χ/Q values using the wind speed categories provided in RG 1.23 Rev. 0.
(4) Based on a commitment made to the NRC, finer wind speed categories from RG 1.23 Rev. 1 have been used in the latest revisions to the LOCA, MSLB, CREA, LRA, SGTR, and FHA offsite dose consequences analyses.

15.1 INCREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

A number of events have been postulated which could result in an increase in heat removal from the reactor coolant system by the secondary system. Analyses are presented for several such events which have been identified as limiting cases.

Discussions of the following reactor coolant system cooldown events are presented in this section:

a. Feedwater system malfunction causing a reduction in feedwater temperature.

b. Feedwater system malfunction causing an increase in feedwater flow.

c. Excessive increase in secondary steam flow.

d. Inadvertent opening of a steam generator relief or safety valve.

e. Steam system piping failure.

The above are considered to be ANS Condition II events, with the exception of a major steam system pipe break, which is considered to be an ANS Condition IV event. Subsection 15.0.1 contains a discussion of ANS classification and applicable acceptance criteria.

15.1.1 Feedwater System Malfunctions Causing a Reduction in Feedwater Temperature

15.1.1.1 Identification of Causes and Accident Description

Reductions in feedwater temperature will cause an increase in core power by decreasing reactor coolant temperature. Such transients are attenuated by the thermal capacity of the secondary plant and of the reactor coolant system (RCS). The overpower - overtemperature protection (neutron overpower, overtemperature and overpower ΔT trips) prevent any power increase which could lead to a DNBR less than the limit value.

A reduction in feedwater temperature may be caused by the accidental opening of a feedwater bypass valve which diverts flow around a portion of the feedwater heaters. In the event of an accidental opening of the bypass valve, there could be a sudden reduction in feedwater inlet temperature to the steam generators. At power, this increased subcooling will create a greater load demand on the RCS.

With the plant at no-load conditions, the addition of cold feedwater may cause a decrease in RCS temperature and thus a reactivity insertion due to the effects of the negative moderator temperature coefficient of reactivity. However, the rate of energy change is reduced as load and feedwater flow decrease, so the no-load transient is less severe than the full power case.

The net effect on the RCS due to a reduction in feedwater temperature would be similar to the effect of increasing secondary steam flow, i.e., the reactor will reach a new equilibrium condition at a power level corresponding to the new steam generator ΔT.

A decrease in normal feedwater temperature is classified as an ANS Condition II event, fault of moderate frequency. See Subsection 15.0.1 for a discussion of Condition II events.

The protection credited for mitigating the consequences of a decrease in feedwater temperature is the same as that for an excessive steam flow increase, as discussed in Section 15.0.8 and listed in Table 15.0-7.

15.1.1.2 Analysis of Effects and Consequences

Method of Analysis

The reduction in feedwater temperature due to a feedwater system malfunction transient is analyzed by using the detailed digital computer code, LOFTRAN (Reference 1). This code simulates a multiloop system, neutron kinetics, the pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables including temperature, pressure, and power level.

The system is analyzed to demonstrate plant behavior following a reduction in feedwater temperature due to the simultaneous opening of the low pressure feedwater heater string bypass valve and isolation of a low pressure feedwater heater string due to a high-two level in a first stage feedwater heater. The loss of a low pressure feedwater heater string causes a redistribution of flow to the other two parallel heater strings and the bypass line. The bypass line takes 40% of the total flow, which effectively increases the temperature differential since less flow passes through the heaters. This reduction in feedwater temperature results in cascading feedwater heater instability with the end result being a loss of all feedwater heating downstream of the fourth stage feedwater heaters. This bypass case results in the most severe decrease in feedwater temperature. For the purpose of this analysis, it is conservatively assumed that the temperature reduction that occurs is due to the loss of multiple trains of feedwater heaters along with the opening of the feedwater heater bypass valves.

This accident is analyzed with the revised thermal design procedure as described in WCAP-11397-P-A (Reference 8). The following assumptions are made:

a. Initial reactor power, pressure, and RCS temperatures are assumed to be at their normal values.

b. For the main feedwater malfunction transient, the limiting temperature reduction failure is the loss of an entire train of feedwater heaters. To conservatively bound this failure, a loss of multiple trains of feedwater heaters is modeled along with the opening of the feedwater heater bypass valves. It is assumed that 40% of the feedwater flows through the bypass line. A conservative feedwater temperature of 200°F is used after the initiating failure for this event. This event is only analyzed at hot full power (HFP) conditions.

c. The rate of feedwater flow to the steam generators is assumed to remain the same as it was immediately preceding the accident.

d. Changes in turbine performance due to changes in RCS performance and process steam extraction rates are neglected.

e. The temperature transient resulting from the bypass of an entire string of low pressure feedwater heaters is terminated by an overpower ΔT trip signal, which trips the reactor, and by a safety injection system low pressurizer pressure signal, which closes the feedwater control and isolation valves.

f. In order to determine the most limiting feedwater temperature reduction case for this accident, both of the Byron/Braidwood steam generator designs (BWI and D5) are analyzed for this event. The most limiting case of the two steam generator designs is documented and presented in the Results section that follows.

Plant characteristics and initial conditions are discussed in Subsection 15.0.3.

Normal reactor control systems and engineered safety systems are not required to function. The reactor protection system may function to trip the reactor due to an overpower condition. No single active failure will prevent operation of the reactor protection system.

Results

Following the initiation of this event as described above, feedwater temperature is conservatively modeled to instantaneously decrease from 449.2°F to 200°F. This reduction in feedwater temperature increases the thermal load on the primary system. The resultant temperature and power transient causes a reactor trip on an overpower ΔT signal. When the pressurizer pressure reaches the low pressure setpoint, the safety injection system is actuated, the feedwater control and isolation valves are closed, and feedwater isolation occurs.

Transient results for the most limiting feedwater temperature reduction case, as shown in Figures 15.1-1a and 15.1-1b, show the increase in nuclear power and loop ΔT associated with the increased thermal load on the reactor. Following reactor trip and feedwater isolation, the plant will approach a stabilized condition, which in this case, is hot standby. Normal plant operating procedures may then be followed. The operating procedures would call for operator action to control reactor coolant system boron concentration and pressurizer level using the chemical volume and control system (CVCS) and to maintain steam generator level through control of the main or auxiliary feedwater system.

Since the power level rises during the reduction in feedwater temperature event, the fuel temperatures will also rise until after reactor trip occurs. The core heat flux lags behind the neutron flux due to fuel rod thermal time constant and the peak linear rod power reached is limited to a value below that which would result in exceeding the fuel melting temperature. Hence, fuel melting is precluded for this event.

The transient results show that departure from nucleate boiling (DNB) does not occur at any time during the feedwater temperature reduction transient; thus, the ability of the primary coolant to remove heat from the fuel rod is not reduced. The fuel cladding temperature, therefore, does not rise significantly above its initial value during the transient.

The calculated sequence of events for the most limiting feedwater temperature reduction case for this accident is shown in Table 15.1-1.

15.1.1.3 <u>Radiological Consequences</u>

The radiological consequences will be less severe than the steamline break accident analyzed in Subsection 15.1.5.3.

15.1.1.4 <u>Conclusions</u>

The results show that the DNB ratios (DNBRs) encountered for the reduction in feedwater temperature at power are above the limit value at all times; therefore, no fuel or clad damage is predicted. The radiological consequences of this event are bounded by the steamline break accident analyzed in Subsection 15.1.5.3.

15.1.2 Feedwater System Malfunctions Causing an Increase in Feedwater Flow

15.1.2.1 Identification of Causes and Accident Description

Additions of excessive feedwater will cause an increase in core power by decreasing reactor coolant temperature. Such transients are attenuated by the thermal capacity of the secondary plant and of the RCS. The overpower - overtemperature protection (neutron overpower, overtemperature and overpower ΔT trips) prevent any power increase which could lead to a DNBR less than the limit value.

An example of excessive feedwater flow would be a full opening of one or more feedwater control valves due to a feedwater control system malfunction or an operator error. At power this excess flow causes a greater load demand on the RCS due to increased subcooling in the steam generators. With the plant at no-load conditions, the addition of an excess of feedwater may cause a decrease in RCS temperature and thus a reactivity insertion due to the effects of the negative moderator coefficient of reactivity. Excessive feedwater flow at no-load conditions results in a less severe transient than at full power. Therefore, only the full power case was analyzed.

Continuous addition of excessive feedwater is prevented by the steam generator high-high level trip, which closes the feedwater valves.

An increase in normal feedwater flow is classified as an ANS Condition II event, a fault of moderate frequency. (See Subsection 15.0.1 for a discussion of ANS Condition II events.)

Plant systems and equipment credited for mitigating the effects of the accident are discussed in Subsection 15.0.8 and listed in Table 15.0-7.

15.1.2.2 Analysis of Effects and Consequences

Method of Analysis

The excessive heat removal due to a feedwater system malfunction transient is analyzed by using the detailed digital computer code LOFTRAN (Reference 1). This code simulates a multiloop system, neutron kinetics, the pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level.

The system is analyzed to demonstrate plant behavior in the event that excessive feedwater addition, due to a control system malfunction or operator error which allows one or more feedwater control valves to open fully occurs. The limiting case analyzed is the accidental opening of four feedwater control valves with the reactor in automatic control at full power.

This accident is analyzed with the revised thermal design procedure as described in WCAP-11397-P-A. Plant characteristics and initial conditions are discussed in Subsection 15.0.3. The reactivity insertion rate following a feedwater system malfunction is calculated with the following assumptions:

a. Initial reactor power, pressure, and RCS temperatures are assumed to be at their nominal values. Uncertainties in initial conditions are included in the limit DNBR as described in WCAP-11397-P-A.

b. For the limiting excessive feedwater flow accident at full power, four feedwater control valves are assumed to malfunction resulting in a step increase to 129% of nominal feedwater flow to each steam generator. The temperature of the feedwater was reduced from 449.2°F to 360°F to all four steam generators.

c. No credit is taken for the heat capacity of the RCS and steam generator thick metal in attenuating the resulting plant cooldown.

d. The excessive feedwater flow results in rising steam generator level which is terminated by a steam generator high-high level trip signal which closes all feedwater control and isolation valves, trips the main feedwater pumps, and trips the turbine.

e. In order to determine the most limiting excessive feedwater flow case for this accident, both of the Byron/Braidwood steam generator designs (BWI and D5) are analyzed for this event. The most limiting case of the two steam generator designs is documented and presented in the Results section that follows.

Plant characteristics and initial conditions are further discussed in Subsection 15.0.3.

Normal reactor control systems and engineered safety systems are not required to function. The reactor protection system may function to trip the reactor due to an overpower condition. No single active failure will prevent operation of the reactor protection system.

Results

The accidental opening of four feedwater control valves at full power (maximum moderator reactivity feedback and minimum end-of-life Doppler-only power coefficients, automatic rod control) gives the largest reactivity feedback and results in the greatest power increase. Assuming the reactor to be in the manual rod control mode results in a slightly less severe transient. The rod control system is, however, not required to function for an excessive feedwater flow event.

The reactor trips on a overpower ΔT signal. When the steam generator water level in the faulted loop(s) reach the high-high level setpoint, all feedwater isolation valves and feedwater pump discharge valves are automatically closed and the main feedwater pumps are tripped. This prevents continuous addition of the feedwater.

Transient results for the most limiting excessive feedwater flow case, see Figures 15.1-2a and 15.1-2b, show the increase in nuclear power and ΔT associated with the increased thermal load on the reactor. Following reactor trip and feedwater isolation, the plant will approach a stabilized condition at hot standby. Normal plant operating procedures may then be followed. The operating procedures would call for operator action to control reactor coolant system boron concentration and pressurizer level using the CVCS and to maintain steam generator level through control of the main or auxiliary feedwater system. Any action required of the operator to maintain the plant in a stabilized condition will be in a time frame in excess of ten minutes following reactor trip.

Since the power level rises during the excessive feedwater flow event, the fuel temperatures will also rise until after the reactor trip occurs. The core heat flux lags behind the neutron flux due to fuel rod thermal time constant and the peak linear rod power reached is limited to a value below that which would result in exceeding the fuel melting temperature. Hence, fuel melting is precluded for this event.

The transient results show that DNB does not occur at any time during the excessive feedwater flow incident; thus, the ability of the primary coolant to remove heat from the fuel rod is not reduced. The fuel cladding temperature, therefore, does not rise significantly above its initial value during the transient.

The calculated sequence of events for the most limiting excessive feedwater flow case for this accident is shown in Table 15.1-1.

15.1.2.3 Radiological Consequences

The radiological consequences will be less severe than the steamline break accident analyzed in Subsection 15.1.5.3.

15.1.2.4 Conclusions

The results of the analysis show that the DNBRs encountered for an excessive feedwater addition at power are at all times above the limit value; hence, no fuel or clad damage is predicted. The radiological consequences of this event will be less than the steam line break accident analyzed in Subsection 15.1.5.3.

15.1.3 Excessive Increase In Secondary Steam Flow

15.1.3.1 Identification of Causes and Accident Description

An excessive increase in secondary system steam flow (excessive load increase incident) is defined as a rapid increase in steam flow that causes a power mismatch between the reactor core power and the steam generator load demand. The reactor control system is designed to accommodate a 10% step load increase of a 5% per minute ramp load increase in the range of 15% to 100% of full power. Any loading rate in excess of these values may cause a reactor trip actuated by the reactor protection system. Steam flow increases greater than 10% are analyzed in Subsections 15.1.4 and 15.1.5.

This accident could result from either an administrative violation such as excessive loading by the operator or an equipment malfunction in the steam dump control or turbine speed control.

During power operation, steam dump to the condenser is controlled by reactor coolant condition signals, i.e., high reactor coolant temperature indicates a need for steam dump. A single controller malfunction does not cause steam dump; an interlock is provided which blocks the opening of the valves unless a large turbine load decrease or a turbine trip has occurred.

Protection against an excessive load increase accident is provided by the following reactor protection system (RPS) signals:

a. Low pressurizer pressure,

b. Overtemperature ΔT, and

c. Power range high neutron flux.

An excessive load increase incident is considered to be an ANS Condition II event, a fault of moderate frequency. See Subsection 15.0.1 for a discussion of Condition II events.

15.1.3.2 Analysis of Effects and Consequences

Method of Analysis

This accident is analyzed using the LOFTRAN Code (Reference 1). The code simulates the neutron kinetics, reactor coolant system, (RCS) pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, steam generator safety valves, and feedwater system. The code computes pertinent plant variables including temperatures, pressures, and power level.

Four cases are analyzed to demonstrate the plant behavior following a 10% step load increase from rated load. These cases are as follows:

a. reactor control in manual with minimum reactivity feedback,

b. reactor control in manual with maximum reactivity feedback,

c. reactor control in automatic with minimum reactivity feedback, and

d. reactor control in automatic with maximum reactivity feedback.

For the minimum reactivity feedback cases, the core has a zero moderator temperature coefficient of reactivity and therefore, the least inherent transient capability. For the maximum reactivity feedback cases, the moderator temperature coefficient of reactivity has its highest absolute value. This results in the largest amount of reactivity feedback due to changes in coolant temperature.

A conservative limit on the turbine valve opening is assumed, and all cases are studied without credit taken for pressurizer heaters.

This accident is analyzed with the revised thermal design procedure as described in WCAP-11397-P-A. Plant characteristics and initial conditions are discussed in Subsection 15.0.3. Initial reactor power, pressure, and RCS temperatures are assumed to be at their nominal values. Uncertainties in initial conditions are included in the limit DNBR as described in WCAP-11397-P-A.

Normal reactor control systems and engineered safety systems are not required to function. The reactor protection system is assumed to be operable; however, reactor trip is not encountered for many cases due to the error allowances assumed in the setpoints. No single active failure will prevent the reactor protection system from performing its intended function.

The cases which assume automatic rod control are analyzed to ensure that the worst case is presented. The automatic rod control system is not required to mitigate the consequences of an accident.

Results

Figures 15.1-3 through 15.1-6 illustrate the transient with the reactor in manual rod control mode. As expected, for the minimum reactivity feedback case there is a slight power increase, and the average core temperature shows a large decrease. This results in a DNBR which increases above its initial value. For the maximum reactivity feedback manually controlled case, there is a much larger increase in reactor power due to the moderator feedback. A reduction in DNBR is experienced, but DNBR remains above the limit value. For these cases, the plant rapidly reaches a stabilized condition at the higher power level. Normal plant operating procedures would then be followed to reduce power.

Figures 15.1-7 through 15.1-10 illustrate the transient assuming the reactor is in the automatic rod control mode and no reactor trip signals occur. Both the minimum and maximum reactivity feedback cases show that core power increases, thereby reducing the rate of decrease in coolant average temperature and pressurizer pressure. For both of these cases, the minimum DNBR remains above the limit value.

The excessive load increase incident is an overpower transient for which the fuel temperatures will rise. Reactor trip may not occur for some of the cases analyzed, and the plant reaches a new equilibrium condition at a higher power level corresponding to the increase in steam flow.

Since the DNBR limit is not violated at any time during the excessive load increase transients, the ability of the primary coolant to remove heat from the fuel rod is not reduced. Thus, the fuel cladding temperature does not rise significantly above its initial value during the transient.

The calculated sequence of events for the excessive load increase incident is shown on Table 15.1-1.

As can be seen from Figures 15.1-4, 15.1-6, 15.1-8, and 15.1-10, for the excessive increase in secondary steam flow accident analysis, the conservative assumptions made in this analysis result in no reactor trip for all cases analyzed. The plant analyses show that a new steady-state condition is reached.

15.1.3.3 Radiological Consequences

There are no radiological consequences associated with this event and activity is contained within the fuel rods and reactor coolant system within design limits.

15.1.3.4 Conclusions

The analysis discussed above shows that for a 10% step load increase, the DNBR remains above the limit value, thereby precluding fuel or clad damage. The plant rapidly reaches a stabilized condition following the load increase.

15.1.4 Inadvertent Opening of a Steam Generator Relief or Safety Valve

The inadvertent opening of a steam generator relief or safety valve event (i.e., the credible steamline break) creates a depressurization of the secondary side with an effective opening size that is within the spectrum of break sizes analyzed by the hypothetical steamline break event. Therefore, the credible steamline break is bounded by the hypothetical steamline break discussed in Subsections 15.1.5 and 15.1.6.

Pages 15.1-11 through 15.1-13 have been deleted intentionally.

15.1.5 Steam System Piping Failure at Zero Power

15.1.5.1 Identification of Causes and Accident Description

The steam release arising from a break of a main steamline would result in an initial increase in steam flow which decreases during the accident as the steam pressure falls. The energy removal from the RCS causes a reduction of coolant temperature and pressure. In the presence of a negative moderator temperature coefficient, the cooldown results in an insertion of positive reactivity. If the most reactive rod cluster control assembly (RCCA) is assumed stuck in its fully withdrawn position after reactor trip, there is an increased possibility that the core will become critical and return to power. A return to power following a steamline break is a potential problem mainly because of the high power peaking factors which exist assuming the most reactive RCCA to be stuck in its fully withdrawn position. The core is ultimately shut down by the boric acid injection delivered by the safety injection system.

The analysis of a main steamline break is performed to demonstrate that the following criteria are satisfied:

a. Assuming a stuck RCCA with or without offsite power, and assuming a single failure in the engineered safety features, the core remains in place and intact. Radiation doses shall meet the applicable regulatory criteria.

b. Although DNB and possible clad perforation following a steam pipe break are not necessarily unacceptable, the following analysis, in fact, shows that DNBR never falls below the analysis limit for any break assuming the most reactive assembly stuck in its fully withdrawn position. The DNBR design basis is discussed in Section 4.4.

A major steamline break is classified as an ANS Condition IV event. See Subsection 15.0.1 for a discussion of Condition IV events.

Effects of minor secondary system pipe breaks are bounded by the analysis presented in this section. Minor secondary system pipe breaks are classified as Condition III events, as described in Subsection 15.0.1.3.

The major break of a steamline is the most limiting cooldown transient and is analyzed at zero power with no decay heat. Decay heat would retard the cooldown thereby reducing the return to power. A detailed analysis of this transient with the most limiting break size, a double ended break, is presented here.

The following functions provide the protection for a steamline break:

a. Safety injection system actuation from any of the following:

   1. Two-out-of-three low steamline pressure signals in any one loop

   2. Two-out-of-four low pressurizer pressure signals.

   3. Two-out-of-three high-1 containment pressure signals.

b. The overpower reactor trips (neutron flux and ΔT) and the reactor trip occurring in conjunction with receipt of the safety injection signal.

c. Redundant isolation of the main feedwater lines. Sustained high feedwater flow would cause additional cooldown. Therefore, in addition to the normal control action which will close the main feedwater valves a safety injection signal will rapidly close all feedwater control valves and backup feedwater isolation valves, trip the main feedwater pumps, and close the feedwater pump discharge valves.

d. Trip of the fast acting steamline stop valves (designed to close in less than 5 seconds) on:

   1. Two-out-of-three low steamline pressure signals in any one loop.

   2. Two-out-of-three high-2 containment pressure signals.

   3. Two-out-of-three high negative steamline pressure rate signals in any one loop (used only during cooldown and heatup operations).

The blocking of safety injection from the low pressurizer pressure and/or low steamline pressure signals is permitted following receipt of the P-11 permissive to allow the plant to be intentionally cooled down without the initiation of safety injection. To ensure that the hot zero power steamline break analysis is bounding when these automatic signals are blocked, the RCS must be borated to ensure subcriticality at 200°F. Prior to manually blocking these automatic signals, meeting the subcriticality boron concentration for 200°F and meeting the normal shutdown margin boron concentration at the current core conditions are both required.

For breaks downstream of the isolation valves, closure of all valves would completely terminate the blowdown. For any break, in any location, no more than one steam generator would experience an uncontrolled blowdown even if one of the isolation valves fails to close. A description of steamline isolation is included in Chapter 10.0.

Steam flow is measured by monitoring dynamic head in nozzles located in the throat of the steam generator. The effective throat area of the nozzles is 1.1 square feet for Unit 1 and 1.4 square feet for Unit 2, which is considerably less than the main steam pipe area; thus, the nozzles also serve to limit the maximum steam flow for a break at any location.

Table 15.1-2 lists the equipment required in the recovery from a high energy line break. Not all equipment is required for any one particular break, since the requirements will vary depending upon postulated break location and details of balance of plant design and pipe break criteria as discussed elsewhere in this application. Design criteria and methods of protection of safety-related equipment from the dynamic effects of postulated piping breaks are provided in Section 3.6.

15.1.5.2 Analysis of Effects and Consequences

Method of Analysis

The analysis of the steam pipe break has been performed to determine:

a. The core heat flux and RCS temperature and pressure resulting from the cooldown following the steamline break. The LOFTRAN code (Reference 1) has been used.

b. The thermal and hydraulic behavior of the core following a steamline break. A detailed thermal and hydraulic digital-computer code, VIPRE, has been used to determine if DNBR falls below the safety analysis limit for the core conditions computed in item a above.

The following conditions were assumed to exist at the time of a main steam break accident:

a. End-of-life shutdown margin at no-load, equilibrium xenon conditions, and the most reactive RCCA stuck in its fully withdrawn position. Operation of the control rod banks during core burnup is restricted in such a way that addition of positive reactivity in a steamline break accident will not lead to a more adverse condition than the case analyzed.

b. A negative moderator coefficient corresponding to the end-of-life rodded core with the most reactive RCCA in the fully withdrawn position. The variation of the coefficient with temperature and pressure has been included. The K_{eff} versus temperature at 1150 psi corresponding to the negative moderator temperature coefficient used is shown in Figure 15.1-11. The effect of power generation in the core on overall reactivity is shown in Figure 15.1-14.

The core properties associated with the sector nearest the affected steam generator and those associated with the remaining sector were conservatively combined to obtain average core properties for reactivity feedback calculations. Further, it was conservatively assumed that the core power distribution was uniform. These two conditions cause underprediction of the reactivity feedback in the high power region near the stuck rod. To verify the conservatism of this method, the reactivity as well as the power distribution was checked for the limiting statepoints for the cases analyzed.

This core analysis, performed with the ANC code, considered the Doppler reactivity from the high fuel temperature near the stuck RCCA, moderator feedback from the high water enthalpy near the stuck RCCA, power redistribution and non-uniform core inlet temperature effects. For cases in which steam generation occurs in the high flux regions of the core, the effect of void formation was also included. It was determined that the reactivity employed in the kinetics analysis was always larger than the reactivity calculated including the above local effects for the statepoints. These results verify conservatism; i.e., underprediction of negative reactivity feedback from power generation.

c. Minimum capability for injection of concentrated boric acid (2,300 ppm) solution corresponding to the most restrictive single failure in the high head safety injection (HHSI) system. The emergency core cooling system (ECCS), consists of three systems: (1) the passive accumulators, (2) the residual heat removal system (RHRS), and (3) the low head safety injection system (LHSIS), and the HHSI system. Only the HHSI system is modeled for the steamline break accident analysis.

The actual modeling of the HHSI system in LOFTRAN is described in Reference 1. The flow corresponds to that delivered by one charging pump delivering its full flow to the cold leg header. No credit has been taken for the low concentration borated water, which must be swept from the lines downstream of the refueling water storage tank prior to the delivery of concentrated boric acid to the reactor coolant loops.

For the cases where offsite power is assumed, the sequence of events in the HHSI system is the following. After the generation of the safety injection signal (appropriate delays for instrumentation, logic, and signal transport included), the appropriate valves begin to operate and the charging pump starts. In 17 seconds, the valves are assumed to be in their final position and the pump is assumed to be at full speed. This does not include sequential transfer of high head safety injection pump suction from the VCT to the RWST. The additional 10 seconds for valves CV112B and C to close after CV112D and E are open has been evaluated and is consistent with the accident analysis results. Transfer of the pump suction would be completed in 27 seconds. The volume containing the low concentration borated water is swept before the 2,300 ppm borated water reaches the core. This delay, described above, is inherently included in the modeling.

In cases where offsite power is not available, an additional 13-second delay is assumed to start the diesels and to load the necessary safety injection equipment onto them.

d. Design value of the steam generator heat transfer coefficient including allowance for fouling factor.

e. Since the steam generators are provided with integral flow restrictors with a 1.1 square foot throat area for Unit 1 and a 1.4 square foot throat area for Unit 2, any break with a break area greater than the area of the flow restrictor, regardless of location, would have the same effect on the NSSS as the break equal to the area of the flow restrictor. The following cases have been considered in determining the core power and RCS transients:

Case 1: Complete severance of a pipe, with the plant initially at no-load conditions, full reactor coolant flow with offsite power available.

Case 2: Case 1 with loss of offsite power coincident with the steamline break. Loss of offsite power results in reactor coolant pump coastdown, which is assumed to begin at 3 seconds.

f. Power peaking factors corresponding to one stuck RCCA and nonuniform core inlet coolant temperatures are determined at end of core life. The coldest core inlet temperatures are assumed to occur in the sector with the stuck rod. The power peaking factors account for the effect of the local void in the region of the stuck control assembly during the return to power phase following the steamline break. This void in conjunction with the large negative moderator coefficient partially offsets the effect of the stuck assembly. The power peaking factors depend upon the core power, temperature, pressure, and flow, and, thus, are different for each case studied.

The core parameters used for both with and without offsite power cases correspond to values determined from the respective transient analysis.

Both cases above assume initial hot shutdown conditions at time zero since this represents the most pessimistic initial condition. The hot shutdown initial conditions were considered for cases assuming full power operation at both the high (588.0°F) and low (575.0°F) HFP T_{avg} conditions. Should the reactor be just critical or operating at power at the time of a steamline break, the reactor will be tripped by the normal overpower protection system when power level reaches a trip point. Following a trip at power, the RCS contains more stored energy than at no-load, the average coolant temperature is higher than at no-load and there is appreciable energy stored in the fuel. Thus, the additional stored energy is removed via the cooldown caused by the steamline break before the no-load conditions of RCS temperature and shutdown margin assumed in the analyses are reached. After the additional stored energy has been removed, the cooldown and reactivity insertions proceed in the same manner as in the analysis which assumes no-load condition at time zero. A spectrum of steamline breaks at various power levels has been analyzed in Reference 4.

g. In computing the steam flow during a steamline break, the Moody Curve (Reference 3) for f(L/D) = 0 is used.

h. Perfect moisture separation in the steam generator is assumed.

These assumptions are discussed more fully in Reference 4.

Results

The calculated sequence of events for the limiting case (Unit 2, low T_{avg}, offsite power available) is shown in Table 15.1-1.

The results presented are a conservative indication of the events which would occur assuming a steamline break since it is postulated that all of the conditions described above occur simultaneously.

Core Power and Reactor Coolant System Transient

Figures 15.1-18 through 15.1-20 for Unit 2 show the RCS transient and core heat flux following a main steamline break (complete severance of a pipe) at initial no-load condition.

Offsite power is assumed available so that full reactor coolant flow exists. The transient shown assumes an uncontrolled steam release from only one steam generator. Should the core be critical at near zero power when the break occurs, the initiation of safety injection by low steamline pressure will trip the reactor. Steam release from more than one steam generator will be prevented by automatic trip of the fast acting isolation valves in the steamlines by low steamline pressure signals, high containment pressure signals, or high negative steamline pressure rate signals. Even with the failure of one valve, release is limited to no more than 10 seconds for the other steam generators while the one generator blows down. The steamline stop valves are designed to be fully closed in less than 5 seconds from receipt of a closure signal.

As shown in Figure 15.1-20 the core attains criticality with the RCCAs inserted (with the design shutdown assuming one stuck RCCA) before boron solution at 2,300 ppm enters the RCS. A peak core power lower than the nominal full power value is attained.

The calculation assumes the boric acid is mixed with, and diluted by, the water flowing in the RCS prior to entering the reactor core. The concentration after mixing depends upon the relative flow rates in the RCS and in the HHSI system. The variation of mass flow rate in the RCS due to water density changes is included in the calculation as is the variation of flow rate in the HHSI system due to changes in the RCS pressure. The HHSI system flow calculation includes the line losses in the system as well as the pump head curve.

The loss of offsite power case corresponds to the case discussed above with additional loss of offsite power at the time the safety injection signal is generated. The safety injection system delay time includes 13 seconds to start the diesel in addition to 17 seconds to start the safety injection pump and open the valves. An additional 10 seconds is required to close valves CV112B and C after CV112D and E are open to transfer the high head safety injection pump suction from the VCT to the RWST. This additional 10 second delay has been evaluated and is consistent with the accident analysis results. In 40 seconds, the diesel and pump are assumed to start and the valves are assumed to be in their final position with the pump suction transferred from the VCT to the RWST. Criticality is achieved later and the core power increase is slower than in the similar case with offsite power available. The ability of the emptying steam generator to extract heat from the RCS is reduced by the decreased flow in the RCS. The peak power remains well below the nominal full power value.

It should be noted that following a steamline break only one steam generator blows down completely. Thus, the remaining steam generators are still available for dissipation of decay heat after the initial transient is over. In the case of loss of offsite power this heat is removed to the atmosphere via the steamline safety valves.

Margin to Critical Heat Flux

DNB analyses were performed for both units, with and without offsite power and High and Low T_{avg} programs. The minimum DNBR is greater than the limit value in all cases and the limiting case is that for Unit 2 (D5 SGs), low HFP T_{avg} case, with offsite power available. The results of this case are presented herein.

15.1.5.3 Radiological Consequences of a Postulated Steamline Break Using AST

The key inputs and assumptions used in the Main Steam Line Break (MSLB) radiological consequence analysis are summarized below and provided in Table 15.1-3. Although this analysis is presented in the section describing a steam system piping failure at zero power, it is also applicable for the full power event (described in section 15.1.6).

The MSLB accident is postulated as a break of one of the large steam lines leading from a steam generator. This break results in the release of radioactive material from the Byron and Braidwood containment system. For the three intact Steam Generator (SG) loops, primary to secondary coolant leakage transfers activity into the Secondary Coolant. This makes it available for release into the environment via steaming through the SG Power-Operated Release Valves (PORV). For the coolant loop with the broken steam line (referred to as the faulted steam generator), primary to secondary coolant leakage is assumed to be released from the RCS directly into the environment without passing through any secondary coolant. This is due to assumed "dry-out" conditions in the faulted steam generator. Consistent with Regulatory Guide (RG) 1.183, two reactor transients that maximize the radioactivity available for release were modeled. In addition to these two transients, the release of the maximum allowed operational concentration of iodine activity in the secondary coolant system, 0.1 µCi/gm is analyzed. This Case simulates the initial blowdown of all fluid in the faulted SG (assuming a 2-minute duration), and the PORV release of secondary coolant activity of the intact SGs. The dose consequence of this simulation is added to each of the other modeled cases.

Case 1: Dose Due to Pre-accident Iodine Spike

The first case involves a 60 uCi/gm pre-accident Iodine spike. This 60 µCi/gm spike is consistent with the Technical Specification operational Reactor Coolant System (RCS) activity concentration limit for an assumed spike. In this scenario, it is assumed that all of the spike activity is homogeneously mixed in the primary coolant, prior to accident initiation.

Case 2: Dose Due to Accident Initiated Concurrent Iodine Spike

The second case involves an accident initiated iodine spike that occurs concurrently with the release of fluid from the primary and secondary coolant systems. Regulatory guidance specifies that this spike should result in a release rate from the operating limit defective fuel fraction that is 500 times the normal rate.

Case 3: Dose Due to Equilibrium Secondary Coolant System Iodine

The third case simulates the dose contribution that results from the initial blowdown of all fluid in the faulted SG (assuming a 2-minute duration), and the PORV release of secondary coolant through the intact SGs. These releases of specifically secondary coolant activity, existing prior to the MSLB accident, are analyzed, and the dose is added to each of the other modeled cases.

Fuel Damage and Core Source Term

The design basis assumes no fuel damage for the postulated main steam line break event. For this MSLB accident, the source terms are defined by the Technical Specification activity release rates from a maximum failed fuel fraction assumed during operation, which are characterized by the equilibrium 1.0 µCi/gm Dose Equivalent(DE) I-131 iodine activity concentration in the primary reactor coolant system. The noble gas inventory in the RCS is based on operation with a conservative worst-case 1% core fuel defects. Because no fuel damage is assumed for this accident, only iodine and noble gas isotopes are modeled to contribute to dose, as given in Table 15.0-16. To identify the worst-case MSLB accident, however, two different cases of iodine spiking are analyzed, per regulatory guidance.

Case 1: Pre-Accident Iodine Spike Source Terms

The first case is simply identified as a reactor pre-accident, transient induced, iodine spike, which raises the primary coolant iodine concentration to the maximum 60 µCi/gm DE I-131 value permitted by Technical Specifications at full power operations, prior to the initiation of the accident. Therefore this case is termed the pre-accident iodine spike case.

Case 2: Concurrent Iodine Spike Source Terms

The second case assumes that the postulated MSLB event causes a primary reactor system transient. This transient, in turn, is associated with an iodine spike which assumes that the iodine release rate from the fuel rods to the primary coolant increases to a value 500 times greater than the release rate corresponding to the 1.0 µCi/gm DE I-131 equilibrium iodine concentration as given in the Technical Specifications. This 500 times activity release rate spike is assumed to occur for a duration of 6 hours, as this period has been shown to conservatively deplete the available gap activity in the assumed operating damaged fuel fraction. Also, this assumption has historically been used as the design basis for this accident at Byron and Braidwood. In RADTRAD a Nuclide Inventory File (NIF) is designed to input the total isotopic iodine activity that is associated with 6 hours of activity release at the 500 times rate specified. Then, this NIF is used in conjunction with a modified Release Fraction and Timing (RFT) file, which defines the complete release of this activity over a 6-hour period.

Case 3: Equilibrium Secondary Coolant System Iodine Source Terms

The case 3 source term consists simply of the 0.1 µCi/gm DE I-131 equilibrium secondary coolant activity concentration limit in Technical Specifications.

Activity Removal Mechanisms in Containment

The design basis MSLB releases activity directly into the primary RCS, therefore no plateout, or other activity deposition, is credited.

Decay Credited:

Decay of radioactivity is credited in all compartments, prior to release. This is implemented in RADTRAD using the half-lives in the NIFs. The RADTRAD decay option is used.

Depletion from Leakage Credited:

For analyses of doses due to release from the RCS volume, the dose results from leakage. It is reasonable to credit the small amount of depletion from the available RCS activity inventory associated with this leakage. This is calculated inherently by the RADTRAD code.

Release Rates, Steaming Rates, and Partitioning Factors:

Activity that originates in the primary RCS is released to the secondary coolant by means of the primary-to-secondary coolant leak rate. This design basis leak rate value is 0.218 gpm, per intact SG, totaling 0.654 gpm, and 0.5 gpm for the faulted SG with the broken steam line. For input into RADTRAD these rates were converted from gallons per minute to cubic feet per minute, making them 0.02914 cfm, per intact SG, totaling 0.08743 cfm, and 0.06684 cfm for the faulted SG.

Primary to secondary coolant leakage through the faulted steam generator conservatively goes directly to the environment, without mixing with any secondary coolant. Therefore, under the assumed dry-out conditions, no partitioning of any nuclides is expected to occur in this release pathway.

For all post-accident releases through the PORVs of the intact SG loops, the mechanism for release to the environment is steaming of the secondary coolant. Because of this release dynamic, RG 1.183 allows for a reduction in the amount of activity released to the environment based on partitioning of nuclides between the liquid and gas states of water. For Iodine, the partitioning factor of 0.01 was taken directly from the suggested guidance of RG 1.183. Reviewing the specified AST release fractions, it is concluded that the only nuclides other than iodines to be released from the core source term are noble gas nuclides. Because of the volatility of noble gases, no partitioning is assumed for any such isotopes.

The methodology used to model steaming of activity through PORVs following the postulated MSLB event, assumes an average cumulative release rate through the SG valves that is reduced in steps. The partitioning factors are applied to these release rates, which were derived from the total time increment mass releases. Incremental steam mass releases are in pounds. Release rates were derived by dividing these totals by the time increment. This data was then converted using the assumption of cooled liquid conditions (i.e., 62.4 lbm/ft^3), as specified by the applicable guidance of RG 1.183. The steaming release and primary-to-secondary coolant leakage is postulated to end at 40 hours, when the RCS and secondary loop have equilibrated. The following table below shows the time steps, isotopic partitioning factors, and associated release rates:

MSLB Partitioning Factors And Associated Release Rates

Time Interval (hrs)	Total Steam Mass Release (lbm)	Iodine Partitioning Factor	Noble Gas Partitioning Factor	Steam Release Rate for Iodines (cfm)	Steam Release Rate for Noble Gases (cfm)
0 - 2.0	447,000	0.01	1	5.9696E-01	5.9696E+01
2.0 - 40	3,279,000	0.01	1	2.3047E-01	2.3047E+01

For the loop with the broken steam line, i.e., the faulted SG, it is postulated that the entire release of the secondary coolant of that loop will take 2 minutes. Therefore, for input into RADTRAD the faulted SG coolant volume of 2675 ft^3 is divided by 2 minutes to arrive at a design basis value of 1.3375E+03 cfm.

χ/Q Calculations (Meteorology)

Releases from the SG PORVs were considered elevated releases due to the high steaming rates and the associated χ/Qs were reduced by a factor of 5 per guidance in RG 1.194, as described in Section 15.4.8.3. The atmospheric dispersion factors are given in Table 15.0-17.

This event was reanalyzed using the finer wind speed categories of Regulatory Guide 1.23, Revision 1.

Assumptions and Inputs

The following inputs and assumptions were used in the MSLB analysis.

a. Core inventory is based on a DBA power level of 3658.3 MWt. This power level bounds the MUR power uprate Rated Thermal Power level including measurement uncertainties.

b. There is no fuel damage as a result of the postulated main steam line break accident.

c. In the case of a postulated Iodine activity release rate spike, the spike release is assumed to occur for a period of 6 hours, when the activity available for release from the fuel has been conservatively depleted.

d. The activity released from the fuel is assumed to be instantaneously mixed with the RCS.

e. All iodine released from the SGs is assumed to be of the elemental species. This is done for RADTRAD simulation considerations, and is consistent with the RG 1.183 specification of 97% elemental and 3% organic, because elemental and organic iodine are treated identically by the computer model.

f. The Control Room HVAC system is realigned to the emergency mode of operation 30 minutes after the initiation of this design basis accident.

g. The faulted steam generator is assumed to be in a "dry-out" condition, and does not inhibit activity release from the RCS through that coolant loop.

h. It is conservatively assumed that blowdown of the faulted steam generator 167,000 lbm fluid takes two minutes to complete.

Dose Results

Radiological doses resulting from a design basis MSLB for a control room operator and a person located at EAB or LPZ are to be less than the regulatory dose limits as given below.

Regulatory Dose Limits - MSLB

	Dose Type	Control Room (rem)	EAB and LPZ (rem)
Case 1	TEDE Dose	5[a]	25[b]
Case 2	TEDE Dose	5[a]	2.5[b]

Notes:
[a] 10 CFR 50.67
[b] 10 CFR 50.67 as modified by Regulatory Guide 1.183 (Table 6, Page 1.183-20)

The table below provides the results from the Case 1 and Case 2 simulations that were modeled using the RADTRAD 3.03 code. The total dose for the two cases includes the dose result from the Case 3 simulations. Therefore for convenience, the doses shown in the Table include the Case 3 dose.

Main Steam Line Break Accident
Radiological Analysis Results

Case 1: Pre-Accident 60 uCi/gm DE I-131 Spike Dose Assessment Results		
Control Room (REM TEDE)	EAB (REM TEDE)	LPZ (REM TEDE)
0.264	0.146	0.083
Case 2: Accident Initiated 500 times Equilibrium Iodine Release Rate Spike Dose Assessment Results		
Control Room (REM TEDE)	EAB (REM TEDE)	LPZ (REM TEDE)
0.654	0.201	0.459

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15.1.5.4 Conclusions

The analysis has shown that the criteria stated in Subsection 15.1.5.1 are satisfied for operation of all units at the uprated power conditions. Although DNB and possible cladding perforation following a steam pipe break are not necessarily unacceptable and not precluded by the criteria, the above analysis, in fact, shows that the DNB design basis is met as stated in Section 4.4.

The radiological consequences of this event are within the dose acceptance criteria of 10 CFR 50.67 and Regulatory Guide 1.183.

15.1.6 Steam System Piping Failure at Full Power

15.1.6.1 Identification of Causes and Accident Description

The steam system piping failure accident analysis described in Subsection 15.1.5 was performed assuming hot zero power conditions with the control rods fully inserted in the core with the exception for the most reactive rod. Such a condition could occur while the reactor is in hot shutdown at the minimum required shutdown margin or after the plant has been tripped automatically by the reactor protection system or manually by the operator. For an at-power steamline break, the analysis of Subsection 15.1.5 represents the limiting condition with respect to core protection for the period following reactor trip. Analysis of a steam system piping failure occurring from at-power initial conditions is performed to demonstrate that core protection is maintained prior to and immediately following reactor trip.

Depending on the size of the break, this event is classified as either a Condition III (infrequent fault) or Condition IV (limiting fault) event. The acceptance criteria for this event are defined in Subsection 15.0.1.

15.1.6.2 Analysis of Effects and Consequences

Method of Analysis

The analysis of the steamline break at-power was performed as follows:

a. The LOFTRAN code (Reference 1) was used to calculate the nuclear power, core heat flux, and reactor coolant system temperature and pressure transients resulting from the cooldown following the steamline break.

b. The core radial and axial peaking factors were determined using the thermal-hydraulic conditions from LOFTRAN as input to the nuclear core models. A detailed thermal-hydraulic code, VIPRE, was used to calculate the DNBR for the limiting time during the transient.

The analysis was performed with the revised thermal design procedure as described in WCAP-11397-P-A (Reference 8). Plant characteristics and initial conditions area discussed in Subsection 15.0.3.

Assumptions

a. Initial conditions - The initial core power, reactor coolant temperature, and reactor coolant system pressure were assumed to be at their nominal full-power values at uprated power conditions. Cases assuming full power operation at the high (588°F) HFP T_{avg} conditions with uniform initial loop flow are analyzed. In addition, cases assuming low (575 °F) HFP T_{avg} and asymmetric initial loop flow conditions are considered and determined to be non-limiting. The asymmetric flow cases assume a maximum 5% loop-to-loop asymmetric flow variation.

b. Break size – The limiting break size was calculated to be 0.95 ft^2 for Unit 1. The results for this case bound all other break sizes for both Unit 1 and Unit 2.

c. Break flow - In computing the steam flow during a steamline break, the Moody curve for f(L/D)=0 is used.

d. Reactivity coefficients - The analysis assumed maximum moderator reactivity feedback and least negative Doppler power feedback to maximize the power increase following the break.

e. Protection system - The protection system features that mitigate the effects of a steamline break are described in Subsection 15.1.5. This analysis only considers the initial phase of the transient from at-power conditions. Protection in this phase of the transient is provided by reactor trip, if necessary. Subsection 15.1.5 presents the analysis of the bounding transient following reactor trip, where other protection system features are actuated to mitigate the effects of the steamline break.

f. Control systems – The pressurizer sprays are modeled to minimize RCS pressure, which is conservative with respect to DNBR. Other control systems were not credited in mitigating the effects of the transient since doing so would not make the results more limiting.

Results

The sequence of events for the limiting case (Unit 1) is shown in Table 15.1-1. Although a break spectrum was analyzed, plots from only one break size (0.95 ft^2 break) are shown. Figures 15.1-27 through 15.1-29 show the transient responses for Unit 1.

Conclusions

The 0.95 ft^2 break with symmetric RCS flow for Unit 1 is the most limiting case for both kW/ft and DNB considerations. The Unit 1 results bound the results of Unit 2.

For radiological consequences of a postulated steamline break, see section 15.1.5.3.

15.1.7 References

1. Burnett, T. W. T., et al., "LOFTRAN Code Description," WCAP-7907-P-A (Proprietary), WCAP-7907-A (Non-Proprietary), April 1984.

2. Deleted.

3. Moody, F. S., "Transactions of the ASME," Journal of Heat Transfer, Figure 3, page 134, February 1965.

4. Scherder, W.J. (Editor), et. al. "Reactor Core Response to Excessive Secondary Steam Releases," WCAP-9226-P-A, Revision 1, (Proprietary), February 1998, and WCAP-9227, Revision 1, (Non-Proprietary), January 1978.

5. Deleted.

6. Deleted.

7. NUREG-0800, "Standard Review Plan," Subsection 15.1.5, Appendix A, "Radiological Consequences of Main Steamline Failure Outside Containment for a PWR," Revision 2, July 1981.

8. Freidland, A. J., et al., "Revised Thermal Design Procedure," WCAP-11397-P-A (Proprietary) and WCAP-11397-A (Nonproprietary), April 1989.

TABLE 15.1-1

TIME SEQUENCE OF EVENT FOR INCIDENTS WHICH CAUSE AN INCREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

ACCIDENT	EVENT	TIME (SEC.)
Limiting Excessive FW Flow Case	Four main feedwater valves fail open	0.0
Multi-Loop Excessive FW Flow to Model BWI steam generator - automatic rod control	Overpower ΔT reactor trip setpoint reached	36.0
	Rod motion occurs	44.0
	Minimum DNBR occurs	44.5
	High-High Steam Generator Water Level Setpoint Reached	69.5
	Feedwater isolation occurs	76.5
Limiting Feedwater Temperature Reduction Case	Feedwater heater bypass valves fail open and a loss of multiple trains of feedwater heaters occurs	0.0
Feedwater Temperature Reduction to Model D5 steam generator - manual rod control	Overpower ΔT reactor trip setpoint reached	5.9
	Rod motion occurs	13.9
	Minimum DNBR occurs	14.5
	Low Pressurizer Pressure SI setpoint reached	34.8
	Feedwater isolation occurs	41.8

TABLE 15.1-1 (Cont'd)

ACCIDENT	EVENT	TIME (SEC.)
Excessive Increase in Secondary Steam Flow		
1. Manual Reactor Control (Minimum moderator feedback)	10% step load increase	0.0
	Equilibrium conditions reached (approximate time only)	300
2. Manual Reactor Control (Maximum moderator feedback)	10% step load increase	0.0
	Equilibrium conditions reached (approximate time only)	150
3. Automatic Reactor Control (Minimum moderator feedback)	10% step load increase	0.0
	Equilibrium conditions reached (approximate time only)	300
4. Automatic Reactor Control (Maximum moderator feedback)	10% step load increase	0.0
	Equilibrium conditions reached (approximate time only)	200
Steam system piping failure at zero power		
1. Unit 2, low T_{avg} (Offsite power available)	Steamline breaks	0.0
	Criticality attained	26.8
	Pressurizer empties	~24.8
	Boron reaches core	~132.8

TABLE 15.1-1 (Cont'd)

ACCIDENT	EVENT	TIME (sec.)
Steam system piping failure at full power		
1. Unit 1 - 0.95 ft^2 break with uniform flow	Steamline breaks	0.0
	Overpower ΔT reactor trip setpoint reached	8.46
	Rods begin to drop	16.46
	Peak core heat flux occurs	17.10

TABLE 15.1-2

EQUIPMENT REQUIRED FOLLOWING A BREAK OF A MAIN STEAM LINE

SHORT TERM (REQUIRED FOR MITIGATION OF ACCIDENT)	HOT STANDBY	REQUIRED FOR COOLDOWN
Reactor trip and safeguards actuator channels including sensors, circuitry, and processing equipment (the protection circuits used to trip the reactor on undervoltage, underfrequency, and turbine trip may be excluded).	Auxiliary feedwater system including pumps, water supply, and system valves and piping (this system must be placed in service to supply water to operable steam generators no later than 10 minutes after the incident).	Steam generator power operated relief valves (can be manually operated locally).
HHSI system including the pumps, the refueling water storage tank, and the systems valves and piping.	Reactor containment ventilation cooling units. Capability for obtaining a reactor coolant system sample.	Residual heat removal system including pumps, heat exchanger, and system valves and piping necessary to cool and maintain the reactor coolant system in a cold shutdown condition.
Standby diesel generators and Class IE power distribution equipment.		
Essential service water and plant component cooling water system.		
Containment safeguards cooling equipment.		

TABLE 15.1-2 (Cont'd)

SHORT TERM (REQUIRED FOR MITIGATION OF ACCIDENT)	HOT STANDBY	REQUIRED FOR COOLDOWN
Auxiliary Feedwater System including pumps, water supplies, piping and valves.		
Main feedwater control valves (trip closed feature).		
Bypass feedwater control valves (trip closed feature).		
Primary and secondary safety valves.		
Circuits and/or equipment required to trip the main feedwater pumps.		
Main feedwater isolation valves (trip closed feature).		
Main steam line stop valves (trip closed feature).		
Main steam line stop valve bypass valves (trip closed feature).		
Steam generator blowdown isolation valves (automatic closure feature).		

TABLE 15.1-2 (Cont'd)

SHORT TERM (REQUIRED FOR MITIGATION OF ACCIDENT)	HOT STANDBY	REQUIRED FOR COOLDOWN
Batteries (Class 1E).		
Control Room air conditioning.		
Control Room equipment must not be damaged to an extent where any equipment will be spuriously actuated or any of the equipment contained elsewhere in this list cannot be operated.		
Emergency lighting.		
Post Accident Monitoring System[a].		
ESF and HHSI/charging pump cubicle unit coolers		

[a]See Section 7.5 for a discussion of the Postaccident Monitoring System.

TABLE 15.1-3

INPUT PARAMETERS FOR THE MSLB RADIOLOGICAL CONSEQUENCE ANALYSIS USING AST

Parameter	Unit	Value	Notes
Steam Released to Environment: Faulted SG (in 2 minutes) Intact SGs: 0 - 2 hours 2 - 40 hours Primary to Secondary Leakage: Faulted SG Intact SG (each)	 lbm lbm lbm gpm gpm	 167,000 447,000 3,279,000 0.5 0.218	The total leakage to be evenly divided for the four steam generators is 1 gpm. For events involving a faulted steam generator, 0.5 gpm leak rate shall be used for faulted generator and 0.218 gpm shall be used for each of the intact generators. Assumed to be based on water at cold conditions.
Duration of steam releases from intact SGs	hr	40	
Duration of activity release due to leakage of primary coolant to the faulted SG	hr	40	
Accident Iodine Spike		Factor of 500	In addition to pre-accident iodine spike case.
Primary and secondary coolant volumes: Primary Secondary: 3 Intact SGs Faulted SG	 gm gm gm	 2.063E8 1.000E8 7.575E7	

TABLE 15.1-3

INPUT PARAMETERS FOR THE MSLB RADIOLOGICAL CONSEQUENCE ANALYSIS USING AST
(continued)

Parameter	Unit	Value	Notes
Noble gas releases through the faulted SG due to primary to secondary leakage:			These values, based on operation with 1% fuel defects. The values are also consistent with a RCS "volume" of 2.063E8 gm.
KR-85m	Curies	3.713E2	
KR-85	Curies	1.467E3	
KR-87	Curies	2.372E2	
KR-88	Curies	6.911E2	
XE-131m	Curies	6.829E2	
XE-133m	Curies	7.530E2	
XE-133	Curies	5.178E4	
XE-135m	Curies	1.007E2	
XE-135	Curies	1.593E3	
XE-138	Curies	1.368E2	

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TABLE 15.1-3a

TABLE 15.1-4

Table 15.1-4 has been deleted intentionally.

TABLE 15.1-4a

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TABLE 15.1-4a

TABLE 15.1-4b

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15.2 DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

A number of transients and accidents have been postulated which could result in a reduction of the capacity of the secondary system to remove heat generated in the reactor coolant system (RCS). These events are discussed in this section. Detailed analyses are presented for several such events which have been identified as more limiting than the others.

Discussions of the following RCS coolant heatup events are presented in Section 15.2:

a. steam pressure regulator malfunction,

b. loss of external load,

c. turbine trip,

d. inadvertent closure of main steam isolation valves,

e. loss of condenser vacuum and other events resulting in turbine trip,

f. loss of nonemergency a-c power to the station auxiliaries,

g. loss of normal feedwater flow, and

h. feedwater system pipe break.

The above items are considered to be ANS Condition II events, with the exception of a feedwater system pipe break, which is considered to be an ANS Condition IV event. Subsection 15.0.1 contains a discussion of ANS classification and applicable acceptance criteria.

15.2.1 Steam Pressure Regulator Malfunction or Failure That Results in Decreasing Steam Flow

There are no pressure regulators whose failure or malfunction could cause a steam flow transient.

15.2.2 Loss of External Load

15.2.2.1 Identification of Causes and Accident Description

A major load loss on the plant can result from loss of external electrical load due to some electrical system disturbance. Offsite ac power remains available to operate plant components such as the reactor coolant pumps; as a result, the onsite emergency diesel generators are not required to function for this event. Following the loss of generator load, an immediate fast closure of the turbine control valves will occur. This will cause a sudden reduction in steam flow, resulting in an increase in pressure and temperature in the steam generator shell. As a result, the heat transfer rate in the steam generator is reduced, causing the reactor coolant temperature to rise, which in turn causes coolant expansion, pressurizer insurge, and RCS pressure rise.

For a loss of external electrical load without subsequent turbine trip, no direct reactor trip signal would be generated, as the plant would be expected to trip from the reactor protection system if a safety limit were approached. A continued steam load of approximately 5% would exist after total loss of external electrical load because of the steam demand of plant auxiliaries.

In the event that a safety limit is approached, protection would be provided by the high pressurizer pressure and overtemperature ΔT trips. Voltage and frequency relays associated with the reactor coolant pump provide no additional safety function for this event. Following a complete loss of load, the maximum turbine overspeed would be approximately 8% to 9%, resulting in an overfrequency of less than 6 Hz. This resulting overfrequency is not expected to damage the turbine protection trip sensors in any way. Testing of turbine overspeed protection equipment is required by Technical Requirements Manual (TRM) 3.3.g. Any degradation in their performance could be ascertained at that time. Any frequency increase to the reactor coolant pump motors will result in slightly increased flow rate and subsequent additional margin to safety limits. For postulated loss of load and subsequent turbine generator overspeed, any overfrequency condition is not seen by safety-related pump motors, reactor protection system equipment, or other safeguards loads. Safeguards loads are supplied from offsite power or, alternatively, from emergency diesels. Reactor protection system equipment is supplied from the inverters; the inverters are supplied from a d-c bus energized from batteries or by a rectified ac voltage from safeguards buses.

In the event the steam dump valves fail to open following a large loss of load, the steam generator safety valves may lift and the reactor may be tripped by the high pressurizer pressure signal, the high pressurizer water level signal, or the overtemperature ΔT signal. The steam generator shell side pressure and reactor coolant temperatures will increase rapidly. The pressurizer safety valves and steam generator safety valves are, however, sized to protect the reactor coolant system (RCS) and steam generator against overpressure for all load losses without assuming the operation of the steam dump system, pressurizer spray, pressurizer power-operated relief valves, or automatic rod cluster control assembly control.

A more complete discussion of overpressure protection can be found in Reference 1.

A loss of external load is classified as an ANS condition II event, fault of moderate frequency. See Subsection 15.0.1 for a discussion of condition II events.

A loss of external load event results in an NSSS transient that is less severe than the turbine trip event as analyzed in Subsection 15.2.3. Therefore, a detailed transient analysis is not presented for the loss of external load.

The primary-side transient is caused by a decrease in heat transfer capability from primary to secondary due to a rapid termination of steam flow to the turbine, accompanied by an automatic reduction of feedwater flow. (Should feed flow not be reduced, a larger heat sink would be available and the transient would be less severe.) Termination of steam flow to the turbine following a loss of external load occurs due to automatic fast closure of the turbine control valves. Following a turbine trip event, termination of steam flow occurs via turbine stop valve closure. The analysis presented in Subsection 15.2.3 assumes a valve closure time which is conservatively fast for both turbine stop valves and control valves. Therefore, the results of that analysis apply to both the loss of external load event and the turbine trip event.

The protection credited for mitigating the consequences of a loss of external load is the same as that for a turbine trip, as listed in Table 15.0-7.

15.2.2.2 Analysis of Effects and Consequences

Method of Analysis

Refer to Subsection 15.2.3.2 for the method used to analyze the limiting transient (turbine trip) in this grouping of events. The results of the turbine trip event analysis are bounding for those expected for the loss of external load, as discussed in Subsection 15.2.2.1.

Normal reactor control systems and engineered safety systems are not required to function. The auxiliary feedwater system may, however, be automatically actuated following a loss of main feedwater; this will further mitigate the effects of the transient.

The reactor protection system may be required to function following a complete loss of external load to terminate core heat input and prevent departure from nucleate boiling (DNB). Depending on the magnitude of the load loss, pressurizer safety valves and/or steam generator safety valves may be required to open to maintain system pressure below allowable limits. No single active failure will prevent operation of any system required to function.

15.2.2.3 Radiological Consequences

Loss of external load from full power would result in the operation of the steam dump system. This system keeps the main turbine generator operating to supply auxiliary electrical loads. Operation of the steam dump system results in bypassing steam to the condenser. If steam dumps are not available, steam generator safety and relief valves relieve to the atmosphere. The radiological consequences will be less severe than those for the steamline break accident analyzed in Subsection 15.1.5.3.

15.2.2.4 Conclusions

Based on results obtained for the turbine trip event (Subsection 15.2.3) and considerations described in Subsection 15.2.2.1, the applicable acceptance criteria for a loss of external load event are met.

15.2.3 Turbine Trip

15.2.3.1 Identification of Causes and Accident Description

For a turbine trip event, the turbine stop valves close rapidly (typically 0.1 sec.) on loss of trip fluid pressure actuated by one of a number of possible turbine trip signals. Turbine trip initiation signals include:

a. low condenser vacuum,

b. low bearing oil pressure,

c. turbine thrust bearing failure,

d. turbine overspeed,

e. manual trip,

f. Low emergency trip header pressure, and

g. Loss of both redundant controllers.

Upon initiation of stop valve closure, steam flow to the turbine stops abruptly. Sensors on the stop valves detect the turbine trip and initiate steam dump. The loss of steam flow results in an almost immediate rise in secondary system temperature and pressure with a resultant primary system transient as described in Subsection 15.2.2.1 for the loss of external load event. For a turbine trip, the reactor would be tripped directly (unless below approximately 30% (P-8) power) on a signal from the turbine stop valves.

The automatic steam dump system would normally accommodate the excess steam generation. Reactor coolant temperatures and pressure do not significantly increase if the steam dump system and pressurizer pressure control system are functioning properly. If the turbine condenser was not available, the excess steam generation would be dumped to the atmosphere and main feedwater flow would be lost. For this situation, feedwater flow would be maintained by the auxiliary feedwater system to ensure adequate residual and decay heat removal capability. Should the steam dump system fail to operate, the steam generator safety valves may lift to provide pressure control. See Subsection 15.2.2.1 for a further discussion of the transient.

A turbine trip is classified as an ANS condition II event, fault of moderate frequency. See Subsection 15.0.1 for a discussion of condition II events.

A turbine trip event is bounding for loss of external load, loss of condenser vacuum, inadvertent closure of the main steam isolation valves, and other turbine trip events. As such, this event has been analyzed in detail. Results and discussion of the analysis are presented in Subsection 15.2.3.2.

The plant systems and equipment credited for mitigating the consequences of a turbine trip are discussed in Subsection 15.0.8 and listed in Table 15.0-7.

15.2.3.2 Analysis of Effects and Consequences

Method of Analysis

In this analysis, the behavior of the unit is evaluated for a complete loss of steam load from full power primarily to show the adequacy of the pressure relieving devices and also to demonstrate core protection margins. The reactor is not tripped until conditions in the RCS result in a trip. No credit is taken for steam dump. Main feedwater flow is terminated at the time of turbine trip, with no credit taken for auxiliary feedwater to mitigate the consequences of the transient.

The turbine trip transients are analyzed by employing the detailed digital computer program LOFTRAN (Reference 3). The program simulates the neutron kinetics, RCS pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The program computes pertinent plant variables including temperatures, pressures, and power level.

This accident is analyzed for three specific cases, one for maximum RCS pressure, a second for maximum MSS pressure, and a third for minimum DNBR. For the pressure cases, the analysis is performed using the standard thermal design procedure. For the DNB case, the revised thermal design procedure is used as described in WCAP-11397-P-A (Reference 5). Plant characteristics and initial conditions are discussed in Subsection 15.0.3. This accident has also been analyzed separately for Unit 1 and for Unit 2 due to differences in steam generator designs.

Major assumptions used in all three cases are summarized below:

a. Moderator and Doppler Coefficients of Reactivity - minimum reactivity feedback is assumed conservatively in all cases. The analysis is performed conservatively at full power conditions assuming a moderator temperature coefficient of 0 pcm/°F and the least negative Doppler-only power and Doppler temperature coefficients. These conditions are bounding conservatively for all operating conditions anticipated throughout each cycle.

b. Reactor Control - from the standpoint of the adverse conditions of concern for this event, it is conservative to assume that the reactor is in manual control. If the reactor were in automatic control, the control rod banks would move prior to trip and reduce the severity of the transient.

c. Steam Release - no credit is taken for the operation of the steam dump system or steam generator power-operated relief valves. When the steam generator pressure rises to the safety valve setpoint, the steam release through the safety valve limits secondary steam pressure.

d. Feedwater Flow - main feedwater flow to the steam generators is assumed to be lost at the time of turbine trip. No credit is taken for auxiliary feedwater flow since a stabilized plant condition will be reached before auxiliary feedwater initiation is normally assumed to occur. The auxiliary feedwater flow would remove core decay heat following plant stabilization.

e. Reactor Trip - is actuated by the first reactor protection system trip setpoint reached. Trip signals are expected due to high pressurizer pressure and overtemperature ΔT.

f. Main Steam Safety Valves – The analysis assumes all main steam safety valves are operable. If one or more main steam safety valves become inoperable, Technical Specification 3.7.1 requires a reduction in power as well as a reduction in the power range neutron flux – high reactor trip setpoint. The required reductions are based on a heat balance equation as described in Reference 6. A sensitivity study was performed to demonstrate that the reductions are adequate.

Additional major assumptions for the RCS pressure case include the following:

a. Initial Operating Conditions - initial reactor power is assumed to be at the nominal value plus uncertainties. The nominal full power RCS temperature plus uncertainties, including the RCS average temperature bias, is modeled. Initial RCS pressure is assumed to be at its nominal value minus uncertainties. The RCS flow rate corresponds to thermal design flow. Maximum steam generator tube plugging is assumed.

b. Pressurizer Spray and Power-Operated Relief Valves - no credit is taken for the effect of pressurizer spray and power-operated relief valves in reducing or limiting the coolant pressure. Safety valves are operable.

Additional major assumptions for the MSS pressure case include the following:

a. Initial Operating Conditions – initial reactor power is assumed to be at the nominal value plus uncertainties. The nominal full power RCS temperature plus uncertainties, including the RCS average temperature bias, is modeled. Initial RCS pressure is assumed to be at its nominal value minus uncertainties. The RCS flow rate corresponds to thermal design flow. Minimum steam generator tube plugging is assumed.

b. Pressurizer Spray and Power-Operated Relief Valves – full credit is taken for the effect of pressurizer spray and power-operated relief valves in reducing or limiting the coolant pressure. Safety valves are also available.

Additional major assumptions for the DNB case include the following:

a. Initial Operating Conditions - initial reactor power and pressure are assumed to be at their nominal values. With the exception of the RCS average temperature bias, which is explicitly modeled in the analysis, uncertainties in initial conditions are included in the limit DNBR as described in WCAP-11397-P-A (Reference 5). The RCS flow rate corresponds to minimum measured flow. Maximum steam generator tube plugging is assumed.

b. Pressurizer Spray and Power-Operated Relief Valves - full credit is taken for the effect of pressurizer spray and power-operated relief valves in reducing or limiting the coolant pressure. Safety valves are also available.

Margin Between High Pressurizer Pressure Reactor Trip and Opening of Pressurizer Safety Valves

The Technical Specifications allow a ±2% tolerance around the nominal setting of 2460 psig for the pressurizer safety valves, with the lowest allowed lifting pressure being 2411 psig. The assumed high pressurizer pressure reactor trip setpoint in the peak pressure cases, including instrument uncertainty, is higher than the lowest setting for the pressurizer safety valves. As a result, the pressurizer safety valves could lift prior to reaching the high pressurizer pressure reactor trip setpoint and the reactor trip might be delayed.

The pressurizer safety valves are assumed to lift at the highest setpoint in the peak pressure cases to maximize RCS and secondary side pressure. A sensitivity study is performed assuming the lowest pressurizer safety valve lift setpoint to demonstrate that, with a delayed reactor trip, all applicable acceptance criteria are met.

Except as discussed above, normal reactor control system and engineered safety systems are not required to function.

The reactor protection system may be required to function following a turbine trip. Pressurizer safety valves and/or steam generator safety valves may be required to open to maintain system pressures below allowable limits. No single active failure will prevent operation of any system required to function.

Results

The transient responses for a turbine trip from full power operation are shown for three cases for each unit. The calculated sequence of events for the accident is shown in Table 15.2-1.

RCS Pressure Case

The transient responses for the total loss of steam load from full power for the RCS overpressure case are shown in Figures 15.2-1 and 15.2-2 for Unit 1 and Figures 15.2-5 and 15.2-6 for Unit 2, respectively. No credit is taken for the pressurizer spray, pressurizer power-operated relief valves, or for the steam dump. The reactor is tripped by the high pressurizer pressure trip channel. The pressurizer safety valves are actuated and the primary system pressure remains below the 110% design value.

MSS Pressure Case

The transient responses for the total loss of steam load from full power for the MSS overpressure case are shown in Figures 15.2-1a and 15.2-2a for Unit 1 and Figures 15.2-5a and 15.2-6a for Unit 2, respectively. Full credit is taken for the pressurizer spray and pressurizer power-operated relief valves. No credit is taken for the steam dump. The reactor is tripped by the overtemperature ΔT trip channel. The steam generator safety valves maintain the secondary side steam pressure below 110% of the steam generator shell design pressure.

DNB Case

The transient responses for the total loss of steam load from full power for the DNB case are shown in Figures 15.2-3 and 15.2-4 for Unit 1 and Figures 15.2-7 and 15.2-8 for Unit 2, respectively. Full credit is taken for the pressurizer spray and pressurizer power-operated relief valves. No credit is taken for the steam dump. The reactor is tripped by the overtemperature ΔT trip channel. The minimum DNBR remains well above the limit value.

In the event that feedwater flow is not terminated at the time of turbine trip for these cases, flow would continue under automatic control with the reactor at a reduced power. The operator would take action to terminate the transient and bring the plant to a stabilized condition. If no action were taken by the operator, the reduced power operation would continue until the condenser hotwell is emptied. A low-low steam generator water level reactor trip would be generated along with auxiliary feedwater initiation signals. Auxiliary feedwater would then be used to remove decay heat with the results less severe than those presented in Subsection 15.2.7.

For Unit 2, Reference 1 presents additional results of the analysis for a complete loss of heat sink including loss of main feedwater. This analysis shows the overpressure protection that is afforded by the pressurizer and steam generator safety valves.

15.2.3.3 Radiological Consequences

The turbine trip transient and steam released for this event are similar to the loss of load transient described in Subsection 15.2.2.3.

There are only minimal radiological consequences associated with this event, therefore, this event is not limiting. The radiological consequences resulting from atmosphere steam dump are less severe than the steamline break event analyzed in Subsection 15.1.5.3.

15.2.3.4 Conclusions

Results of the analyses show that the plant design is such that a turbine trip presents no hazard to the integrity of the RCS or the main steam system. Pressure relieving devices incorporated in the two systems are adequate to limit the maximum pressures to within the design limits.

The integrity of the core is maintained by operation of the reactor protection system, i.e., the DNBR will be maintained above the limit value. The applicable acceptance criteria as listed in Subsection 15.0.1 have been met. The above analysis demonstrates the ability of the NSSS to safely withstand a full load rejection. The radiological consequences in this event will be less than the steam break event analyzed in Subsection 15.1.5.3.

15.2.4 Inadvertent Closure of Main Steam Isolation Valves

Inadvertent closure of the main steam isolation valves would result in a turbine trip. Turbine trips are discussed in Subsection 15.2.3.

15.2.5 Loss of Condenser Vacuum and Other Events Causing a Turbine Trip

Loss of condenser vacuum is one of the events that can cause a turbine trip. Turbine trip initiating events are described in Subsection 15.2.3. A loss of condenser vacuum would preclude the use of steam dump to the condenser; however, since steam dump is assumed not to be available in the turbine trip analysis, no additional adverse effects would result if the turbine trip were caused by loss of condenser vacuum. Therefore, the analysis results and conclusion contained in Subsection 15.2.3 apply to loss of condenser vacuum. In addition, analyses for the other possible causes of a turbine trip, as listed in Subsection 15.2.3.1 are covered by Subsection 15.2.3. Possible over-frequency effects due to a turbine overspeed condition are discussed in Subsection 15.2.2.1 and are not a concern for this type of event.

Pages 15.2-10 through 15.2-13 have been deleted intentionally.

15.2.6 Loss of Nonemergency AC Power to the Plant Auxiliaries

15.2.6.1 Identification of Causes and Accident Description

A complete loss of nonemergency ac power may result in the loss of all power to the plant auxiliaries, i.e., the reactor coolant pumps, condensate pumps, etc. The loss of power may be caused by a complete loss of the offsite grid accompanied by a turbine generator trip at the station, or by a loss of the onsite ac distribution system.

This transient is more severe than the turbine trip event analyzed in Subsection 15.2.3 because for this case the decrease in heat removal by the secondary system is accompanied by a flow coastdown which further reduces the capacity of the primary coolant to remove heat from the core. The reactor will trip: (1) upon reaching one of the trip setpoints in the primary and secondary systems as a result of the flow coastdown and decrease in secondary heat removal; or (2) due to loss of power to the control rod drive mechanisms as a result of the loss of power to the plant.

Following a loss of ac power with turbine and reactor trips, the sequence described below will occur:

a. The standby diesel generators, started on loss of voltage on the plant emergency buses, begin to supply plant vital loads.

b. Plant vital instruments are supplied from emergency d-c power sources.

c. As the steam system pressure rises following the trip, the steam generator power-operated relief valves may be automatically opened to the atmosphere. The condenser is assumed not to be available for steam dump. If the steam flow rate through the power relief valves is not available, the steam generator safety valves may lift to dissipate the sensible heat of the fuel and coolant plus the residual decay heat produced in the reactor.

d. As the no load temperature is approached, the steam generator power-operated relief valves (or safety valves, if the power operated relief valves are not available) are used to dissipate the residual decay heat and to maintain the plant at the hot standby condition.

The auxiliary feedwater system is started automatically as follows:

One motor driven and one diesel-driven auxiliary feedwater pump are started on any of the following:

a. Low-low level in any steam generator,

b. Any safety injection signal,

c. Loss of offsite power, and

d. Manual actuation.

The motor-driven auxiliary feedwater pump is supplied power by the diesel generators. The diesel-driven auxiliary feedwater pump is driven by its own diesel engine. Both type pumps are designed to supply rated flow within approximately one minute of the initiating signal even if a loss of all nonemergency a-c power occurs simultaneously with loss of normal feedwater.

The motor-driven auxiliary feedwater pump is supplied power by the diesel generators. The diesel-driven auxiliary feedwater pump is driven by its own diesel engine. Both type pumps are designed to supply rated flow within 63 seconds of a low-low steam generator signal if a loss of all nonemergency a-c power occurs with a loss of normal feedwater. The motor driven auxiliary feedwater pump delay and startup sequence times after the low-low steam generator level is reached and assuming a loss of offsite power occurs when the reactor trip occurs are given below. The starting sequence is within the 63 seconds of the low-low steam generator signal assumed for establishing rated auxiliary feedwater flow with a loss of offsite power.

Start of rod insertion and loss of offsite power occurs	2.0	seconds
CV-7 Undervoltage Relay Allowable Response Time	5.9	
Diesel Starting	10.0	
CV-7 Overvoltage Relay Operation	2.0	
Auxiliary Feedwater Pump Sequence Delay	35.0	
Auxiliary Feedwater Pump Maximum Acceleration Time	6.1	
Miscellaneous Auxiliary Relay/Breaker Operation	1.0	
Total	62.0	seconds

The pumps take suction from the condensate storage tank for delivery to the steam generators. If the condensate storage tank is not available, then the source for the AFW is switched to the Essential Service Water system causing an additional delay of 12 seconds in the initiation of AFW so that sufficient water is provided to the AFW system to operate the pump.

Upon the loss of power to the reactor coolant pumps, coolant flow necessary for core cooling and the removal of residual heat is maintained by natural circulation in the reactor coolant loops.

A loss of nonemergency ac power to the station auxiliaries is classified as an ANS Condition II event, fault of moderate frequency. See Subsection 15.0.1 for a discussion of Condition II events.

The first few seconds after the loss of power to the reactor coolant pumps will closely resemble the simulation of the complete loss of flow incident (see Subsection 15.3.2, where it is shown that the DNBR is maintained above the limit value). Therefore, the DNB aspects of this event were not reevaluated for this analysis.

Following the reactor coolant pump coastdown caused by the loss of ac power, the natural circulation capability of the RCS will remove residual and decay heat from the core, aided by auxiliary feedwater in the secondary system. An analysis is presented here to show that the natural circulation flow in the RCS following a loss of ac power event is sufficient to remove residual heat from the core.

Like the analysis of the Loss of Normal Feedwater Flow event described in UFSAR Section 15.2.7, the primary purpose of the analysis for the Loss of Non-emergency AC Power to the Plant Auxiliaries described herein is to demonstrate the long-term heat removal capacity of the auxiliary feedwater system. Following a loss of offsite power signal and resultant loss of instrument air, the letdown isolation valves close, the normal charging path is isolated, and the charging pumps are initiated. Unlike the letdown and normal charging flow paths, the reactor coolant pump seal injection flow path is not isolated following a loss of instrument air; thus, coolant would be added to the RCS via seal injection flow. These actions would provide a net addition of relatively cold water to the reactor coolant system inventory. Since this addition of colder water would provide a benefit relative to the post-trip primary-side temperature increase, it is conservative not to credit these actions in demonstrating the heat removal capacity of the auxiliary feedwater system. However, the potential effect of the reactor coolant pump seal injection flow on the reactor coolant system inventory following a loss of offsite power are considered as part of the overall evaluation of this event.

The plant systems and equipment credited for mitigating the consequences of a loss of ac power event are discussed in Subsection 15.0.8 and listed in Table 15.0-7.

15.2.6.2 Analysis of Effects and Consequences

An analysis was performed for both Unit 1 and Unit 2. However, the Unit 1 analysis is more limiting and, therefore, is presented here.

Method of Analysis

A detailed analysis using the LOFTRAN Code (Reference 3) is performed to obtain the plant transient following a station loss of nonemergency a-c power. The simulation describes the plant thermal kinetics, reactor coolant system (RCS) including the natural circulation, pressurizer, steam generators and feedwater system. The digital program computes pertinent variables including the steam generator mass, pressurizer water level, and reactor coolant average temperature.

Assumptions made in the analysis are:

a. The initial power level bounds 100% of the nominal NSSS power level of 3672 MWt (including power uncertainty), which includes the nominal reactor coolant pump heat.

b. The initiating event is a loss of all non-emergency ac power that results in the loss of power supply for the condensate pumps. The loss of the condensate pumps results in a loss of normal feedwater.

c. The RCPs are conservatively assumed to operate until the time of reactor trip providing a constant reactor coolant volumetric flow equal to the Thermal Design Flow value. This is to maximize the amount of stored energy in the RCS. The loss of power to the RCPs is not assumed to occur until after the start of rod motion following the reactor trip on a low-low steam generator water level condition.

d. No credit is taken for the immediate insertion of the control rods because of the loss of ac power to the station auxiliaries.

e. Cases are analyzed assuming initial HFP reactor vessel average coolant temperatures at the upper and lower ends of the reactor vessel average temperature window. The vessel average temperature assumed at the upper end of the temperature window is 588°F plus an uncertainty of 9.1°F, which includes a bias of 1.5°F. The average temperature assumed at the lower end of the average temperature window is 575°F minus an uncertainty of 7.6°F.

   The limiting Unit 1 case assumes a vessel average temperature of 575°F minus an uncertainty of 7.6°F.

f. Initial pressurizer pressure is assumed to be 2250 psia with an initial pressurizer pressure uncertainty of ±43 psi. Cases are considered with the pressure uncertainty applied in both the positive and negative direction to conservatively encompass the potential operating conditions.

   The limiting Unit 1 case assumes an initial pressurizer pressure of 2250 psia plus the uncertainty of 43 psi.

g. Reactor trip occurs on steam generator low-low water level at 10% of narrow range span for the limiting case (Unit 1).

h. The worst single failure, which is modeled in the analysis, is the loss of the diesel-driven AFW pump. This results in the availability of one motor-driven AFW pump supplying a minimum total AFW flow of 560 gpm distributed equally to each of the four steam generators.

i. Cases are analyzed assuming different sources for AFW resulting in different delay times to the initiation of AFW. When the AFW source is the condensate storage tank, AFW flow is assumed to be initiated 63 seconds following a low-low steam generator water level signal to allow time for sensor response, signal processing, startup of the emergency diesel generators and the AFW pump. AFW flow is assumed to be initiated 75 seconds following a low-low steam generator water level signal when the condensate storage tank is not available and the AFW source is switched to the Essential Service Water system. The additional 12 seconds allows time for the Essential Service Water system to provide sufficient water to operate the pump. The limiting case (Unit 1) assumes a delay of 75 seconds in AFW initiation because the Essential Service Water system is the AFW source.

j. A maximum steam generator blowdown valve leak rate of 10 gpm per steam generator is accounted for in the analysis.

k. The pressurizer sprays and PORVs are assumed to be operable to maximize the pressurizer water volume. These control systems are not credited for event mitigation since the pressurizer safety valves alone would prevent the RCS pressure from exceeding the RCS design pressure limit during this transient. An evaluation was performed modeling the PORVs as inoperable. The evaluation proved that it is conservative to model the PORVs as inoperable and acceptable results were obtained.

l. Secondary system steam relief is achieved through the steam generator safety valves that are modeled assuming a +4% lift point tolerance.

m. A conservative core residual heat generation based upon long term operation at the initial power level preceding the trip is assumed in the analysis. This core residual heat generation model is based on the 1979 version of ANS 5.1 (Reference 4). ANSI/ANS-5.1-1979 is a conservative representation of the decay energy release rates.

n. A steam generator tube plugging level of 0% is modeled for the limiting case (Unit 1).

o. Cases are analyzed assuming a maximum AFW enthalpy of 91.12 Btu/lbm when the AFW source is condensate storage tank and assuming a maximum AFW enthalpy of 78.2 Btu/lbm for Byron and 80.2 Btu/lbm for Braidwood (AF temperature of 104°F) when the AFW source is the Essential Service Water system. An AFW line purge volume of 160 ft^3 is also modeled. The limiting case (Unit 1) modeled an AFW enthalpy of 78.2 Btu/lbm for Byron and 80.2 Btu/lbm for Braidwood (AF temperature of 104°F).

p. A heat transfer coefficient in the steam generators associated with RCS natural circulation is assumed following the RCP coastdown.

The assumptions used in the analysis are essentially identical to the loss of normal feedwater flow incident (Subsection 15.2.7), except that power is assumed to be lost to the reactor coolant pumps at the time of reactor trip and the pressurizer heaters are not operable during a loss of offsite power event.

Plant characteristics and initial conditions are further discussed in Subsection 15.0.3.

Results

The limiting results were calculated for Unit 1 modeling the Essential Service Water system as the AFW source. The transient response of the RCS following a loss of ac power is shown in Figures 15.2-9a and 15.2-10a. The calculated sequence of events for this event is listed in Table 15.2-1.

The first few seconds after the loss of power to the reactor coolant pumps will closely resemble a simulation of the complete loss of flow incident (see Subsection 15.3.2) i.e., core damage due to rapidly increasing core temperatures is prevented by promptly tripping the reactor. After the reactor trip, stored and residual decay heat must be removed to prevent damage to either the RCS or the core.

For the purpose of demonstrating adequate heat removal capacity of the emergency auxiliary feedwater system, operation of the charging pumps, initiated on a loss of offsite power signal, are not assumed to function for this event since their operation is a benefit with respect to long term core decay heat removal. A loss of offsite power will lead to increased reactor coolant system inventory because the following events occur: 1) the charging pumps actuate and mass is added to the reactor coolant system via reactor coolant seal injection flow, and 2) the letdown isolation valves close due to a loss of instrument air. This scenario is examined separately. From the evaluation of this scenario, it was determined that appropriate operator actions can be taken within 1 hour of event initiation to terminate water relief through the pressurizer safety valves and preclude valve damage. These operator actions are included in the plant emergency response guidelines following a loss of offsite power event.

15.2.6.3 Radiological Consequence

A loss of nonessential ac power to plant auxiliaries would result in a turbine and reactor trip and loss of condenser vacuum. Heat removal from the secondary system would occur through the steam generator power-operated relief valves or safety valves. The radiological consequences will be less severe than the steamline break accident analyzed in Subsection 15.1.5.3.

15.2.6.4 Conclusions

Analysis of the natural circulation capability of the reactor coolant system has demonstrated that sufficient heat removal capability exists following RCP coastdown to prevent fuel or clad damage. The radiological consequences of this event would be less severe than the steamline break event analyzed in Subsection 15.1.5.3.

Pages 15.2-14 through 15.2-16 have been deleted intentionally.

15.2.7 Loss of Normal Feedwater Flow

15.2.7.1 Identification of Causes and Accident Description

A loss of normal feedwater (from pump failures, valve malfunctions, or loss of offsite ac power) results in a reduction in capability of the secondary system to remove the heat generated in the reactor core. If an alternative supply of feedwater were not supplied to the plant, core residual heat following reactor trip would heat the primary system water to the point where water relief from the pressurizer would occur, resulting in a substantial loss of water from the reactor coolant system (RCS). Since the plant is tripped well before the steam generator heat transfer capability is reduced, the primary system variables never approach a DNB condition.

The following occur upon loss of normal feedwater (assuming main feedwater pump failures or valve malfunctions):

a. As the steam system pressure rises following the trip, the steam generator power-operated relief valves are automatically opened to the atmosphere.

   Steam dump to the condenser is assumed not to be available. If the steam flow rate through the power-operated relief valves is not available, the steam generator safety valves may lift to dissipate the sensible heat of the fuel and coolant plus the residual decay heat produced in the reactor.

b. As the no load temperature is approached, the steam generator power-operated relief valves (or safety valves, if the power-operated relief valves are not available) are used to dissipate the residual decay heat and to maintain the plant at the hot standby condition.

A loss of normal feedwater is classified as an ANS Condition II event, fault of moderate frequency. See Subsection 15.0.1 for a discussion of Condition II events.

The auxiliary feedwater system is started automatically as discussed in Subsection 15.2.6.1. Both the motor-driven auxiliary feedwater pump and the independently powered diesel-driven auxiliary feedwater pump take suction directly from the condensate storage tank for delivery to the steam generators. If the condensate storage tank is not available, then the source for the AFW is switched to the Essential Service Water system causing an additional delay of 12 seconds in the initiation of AFW so that sufficient water is provided to the AFW system to operate the pump.

An analysis of the system transient is presented below to show that following a loss of normal feedwater, the auxiliary feedwater system is capable of removing the stored and residual heat, thus preventing either overpressurization of the RCS or loss of water from the reactor core, and returning the plant to a safe condition.

15.2.7.2 Analysis of Effects and Consequences

An analysis was performed for both Unit 1 and Unit 2. Because the Unit 2 analysis is more limiting, it is presented here.

Method of Analysis

A detailed analysis using the LOFTRAN Code (Reference 3) is performed in order to obtain the plant transient following a loss of normal feedwater. The simulation describes the plant thermal kinetics, RCS, pressurizer, steam generators and feedwater system. The digital program computes pertinent variables including the steam generator mass, pressurizer water level, and reactor coolant average temperature.

Assumptions made in the analysis are:

a. The initial power level bounds 100% of the nominal NSSS power level of 3672 MWt (including power uncertainty), which includes applicable uncertainties and a maximum reactor coolant pump heat of 16.65 MWt.

b. The RCPs are assumed to operate continuously throughout the transient providing a constant reactor coolant volumetric flow equal to the Thermal Design Flow.

c. Cases are analyzed assuming initial HFP reactor vessel average coolant temperatures at the upper and lower ends of the reactor vessel average temperature window. The vessel average temperature assumed at the upper end of the temperature window is 588°F plus an uncertainty of 9.1°F, which includes a bias of 1.5°F. The average temperature assumed at the lower end of the average temperature window is 575°F minus an uncertainty of 7.6°F.

   The limiting Unit 2 case assumes a vessel average temperature of 588°F plus an uncertainty of 9.1°F.

d. Initial pressurizer pressure is assumed to be 2250 psia with an initial pressurizer pressure uncertainty of ±43 psi. Cases are considered with the pressure uncertainty applies in both the positive and negative direction to conservatively encompass the potential operating conditions.

The limiting Unit 2 case assumes an initial pressurizer pressure of 2250 psia plus the uncertainty of 43 psi.

e. Reactor trip occurs on steam generator low-low water level at 28.6% of narrow range span for the limiting case (Unit 2).

f. The worst single failure, which is modeled in the analysis, is the loss of the diesel-driven AFW pump. This results in the availability of one motor-driven AFW pump supplying a minimum total AFW flow of 560 gpm distributed equally among each of the four steam generators.

g. Cases are analyzed assuming different sources for AFW resulting in different delay times to the initiation of AFW. AFW flow is assumed to be initiated 55 seconds following a low-low steam generator water level signal when the AFW sources is the condensate storage tank to allow time for sensor response, signal processing, and startup of the AFW pump. AFW flow is assumed to be initiated 67 seconds following a low-low steam generator water level signal when the condensate storage tank is not available and the AFW source is switched to the Essential Service Water system to provide sufficient water to operate the pump. The limiting case (Unit 2) assumes a delay of 55 seconds in AFW initiation because the condensate storage tank is the AFW source.

h. A maximum steam generator blowdown valve leak rate of 3 gpm per steam generator is accounted for in the analysis.

i. In order to conservatively maximize the pressurizer water volume, the pressurizer sprays are assumed to be operable and the pressurizer PORVs are assumed to be inoperable. For event mitigation, the pressurizer safety valves alone would prevent the RCS pressure from exceeding the RCS design pressure limit during this transient.

j. Secondary system steam relief is achieved through the steam generator safety valves that are modeled assuming at +3% lift point tolerance.

k. A conservative core residual heat generation based upon long term operation at the initial power level preceding the trip is assumed in the analysis. This core residual heat generation model is based on the 1979 version of ANS 5.1 (Reference 4). ANSI/ANS-5.1-1979 is a conservative representation of the decay energy release rates.

l. A steam generator tube plugging level of 0% is modeled for the limiting case (Unit 2).

m. Cases are analyzed assuming a maximum AFW enthalpy of 91.12 Btu/lbm when the AFW source is the condensate storage tank and assuming a maximum AFW enthalpy of 78.2 Btu/lbm for Byron and 80.2 Btu/lbm for Braidwood when the AFW source is the Essential Service Water system. An AFW line purge volume of 160 ft^3 is also modeled. The limiting case (Unit 2) modeled an AFW enthalpy of 91.12 Btu/lbm.

The loss of normal feedwater analysis is performed to demonstrate the adequacy of the reactor protection and engineered safeguards systems (e.g., the auxiliary feedwater system) in removing long term decay heat and preventing excessive heatup of the RCS with possible resultant RCS overpressurization or loss of RCS water.

As such, the assumptions used in this analysis are designed to minimize the energy removal capability of the system and to maximize the possibility of water relief from the coolant system by maximizing the coolant system expansion, as noted in the assumptions listed above.

For the loss of normal feedwater transient, the reactor coolant volumetric flow remains at its normal value and the reactor trips via the low-low steam generator level trip. The reactor coolant pumps may be manually tripped at some later time to reduce heat addition to the RCS.

Plant characteristics and initial conditions are further discussed in Subsection 15.0.3.

Plant systems and equipment credited for mitigating the effects of a loss of normal feedwater accident are discussed in Subsection 15.0.8 and listed in Table 15.0-7. Normal reactor control systems are not required to function. The reactor protection system is required to function following a loss of normal feedwater as analyzed here. The auxiliary feedwater system is required to deliver a minimum auxiliary feedwater flowrate. No single active failure will prevent operation of any system required to function.

Results

The limiting results were calculated for Unit 2, modeling the condensate storage tank as the AFW source. Figures 15.2-11 and 15.2-12 show the significant plant parameters following a loss of normal feedwater.

Following the reactor and turbine trip from full load, the water level in the steam generators will fall due to the reduction of steam generator void fraction and because steam flow through the safety valves continues to dissipate the stored and generated heat. Approximately one minute following the initiation of the low-low level trip, at least one auxiliary feedwater pump is automatically started, reducing the rate of water level decrease.

The capacity of the auxiliary feedwater pumps combined with the large steam generator inventory is such that the water level in the steam generators being fed does not recede below the lowest level at which sufficient heat transfer area is available to dissipate core residual heat without water relief from the RCS safety valves. Figure 15.2-11 shows that at no time is there water relief from the pressurizer.

The calculated sequence of events for this accident is listed in Table 15.2-1. As shown in Figures 15.2-11 and 15.2-12, the plant approaches a stabilized condition following reactor trip and auxiliary feedwater initiation. Plant procedures may be followed to further cool down the plant.

15.2.7.3 Radiological Consequences

The steam release and resulting radiological consequence from this transient would be the same as that for the loss of offsite ac power, and similarly, radiological consequences resulting from this transient are less severe than the steamline break accident analyzed in Subsection 15.1.5.3.

15.2.7.4 Conclusions

Results of the analysis show that a loss of normal feedwater does not adversely affect the core, the RCS, or the steam system and the auxiliary feedwater capacity is such that reactor coolant water is not relieved from the pressurizer relief or safety valves. The radiological consequences of this event would be less severe than the steamline break accident analyzed in Subsection 15.1.5.3.

Pages 15.2-16f through 15.2-24 have been deleted intentionally.

15.2.8 Feedwater System Pipe Break

It should be noted that the results and plots presented in this section are for Unit 2 and are not intended to be representative of Unit 1. The cooldown that occurs during a Unit 1 feedline break event is much more severe than that for Unit 2. This is because of the difference between the main feedwater system design in the Unit 1 (feedring) and Unit 2 (preheat) steam generators. The cooldown portion of the event is bounded by the steamline break event, as analyzed in Subsections 15.1.5 and 15.1.6. The results of the heatup portion of the feedline break show that the Unit 2 analysis is more limiting than the Unit 1 analysis with respect to hot leg saturation margin. Therefore, the Unit 1 results are not presented here.

15.2.8.1 Identification of Causes and Accident Description

A major feedwater line break is defined as a break in a feedwater line large enough to prevent the addition of sufficient feedwater to the steam generators to maintain shell side fluid inventory in the steam generators. If the break is postulated in a feedline between the check valve and the steam generator, fluid from the steam generator may also be discharged through the break. (A break upstream of the feedline check valve would affect the nuclear steam supply system NSSS only as a loss of feedwater. This case is covered by the evaluation in Subsection 15.2.7.)

Depending upon the size of the break and the plant operating conditions at the time of the break, the break could cause either a RCS cooldown (by excessive energy discharge through the break) or a RCS heatup. Potential RCS cooldown resulting from a secondary pipe break is evaluated in Subsection 15.1.5. Therefore, only the RCS heatup effects are evaluated for a feedwater line break.

A feedwater line break reduces the ability of the steam generators to remove heat generated by the core from the RCS for the following reasons:

a. Feedwater flow to the steam generators is reduced. Since feedwater is subcooled, its loss may cause reactor coolant temperatures to increase prior to reactor trip.

b. Fluid in the steam generator may be discharged through the break, and would then not be available for decay heat removal after trip.

c. The break may be large enough to prevent the addition of any main feedwater to the steam generators.

An auxiliary feedwater system is provided to assure that adequate feedwater will be available such that:

a. No substantial overpressurization of the RCS shall occur.

b. Sufficient liquid in the RCS shall be maintained in order to provide adequate decay heat removal.

Refer to Subsection 10.4.9 for a description of the auxiliary feedwater system interfaces.

Although overpressurization of the RCS is stated as a concern, the main feedwater line break event is bounded by the loss of load and turbine trip event (see subsections 15.2.2 and 15.2.3), in which assumptions are made to conservatively calculate the RCS pressure transient.

A major feedwater line break is classified as an ANS Condition IV event. See Subsection 15.0.1 for a discussion of Condition IV events.

The severity of the feedwater line break transient depends on a number of system parameters including break size, initial reactor power, and credit taken for the functioning of various control and safety systems. A number of cases of feedwater line breaks have been analyzed. Based on these analyses, it has been shown that the most limiting feedwater line breaks are the double ended break of the largest feedwater line, occurring at full power with and without loss of offsite power. These cases are discussed below.

The following provides the necessary protection for a main feedwater break:

a. A reactor trip on any of the following conditions:

   1. High pressurizer pressure.

   2. Overtemperature ΔT.

   3. Low-low steam generator water level in any steam generator.

   4. Safety injection signals from any of the following:

      a) 2/3 low steam line pressure in any loop.

      b) 2/3 high containment pressure.

   (Refer to Chapter 7.0 for a description of the actuation system.)

b. An auxiliary feedwater system to provide an assured source of feedwater to the steam generators for decay heat removal. The auxiliary feedwater system can draw water from either the condensate storage tank or the Essential Service Water system. If the condensate storage tank is not available, then the auxiliary feedwater source is switched to the Essential Service Water System causing an additional delay of 12 seconds in the initiation of auxiliary feedwater because of the time required to provide sufficient water to operate the AFW pump. (Refer to Subsection 10.4.9 for a description of the auxiliary feedwater system.)

Receipt of a low-low steam generator water level signal in at least one steam generator starts the motor-driven and diesel-driven auxiliary feedwater pumps, which then deliver auxiliary feedwater flow to the steam generators. Similarly, receipt of a low steamline pressure signal in at least one steam line initiates a steamline isolation signal which closes the main steamline isolation valves in all steam lines. This signal also gives a safety injection signal which initiates flow of borated water into the RCS. The amount of safety injection flow is a function of RCS pressure.

Emergency operating procedures call for the following actions to be taken by the reactor operator following a secondary system line break:

a. Isolate feedwater flow spilling out the break of the faulted steam generator.

b. Stop high head safety injection charging pumps if (1) wide range reactor coolant pressure is stable or increasing, (2) pressurizer water level is greater than 12% of span (normal containment) and 28% of span (adverse containment) for Byron (Unit 1 and Unit 2) and 14% of span (normal containment) and 28% of span (adverse containment) for Braidwood (Unit 1 and Unit 2), (3) total feed flow to an intact steam generator is > 500 gpm or the narrow range level in at least one intact steam generator is > 10% of narrow range span (normal containment) and 31% of span (adverse containment) for Unit 1 (Byron and Braidwood) and 14% of narrow range span (normal containment) and 34% of span (adverse containment) for Unit 2 (Byron and Braidwood), and (4) RCS subcooling is acceptable.

Isolating feedwater flow through the break allows additional auxiliary feedwater flow to be diverted to the intact steam generators.

Subsequent to recovery of level in the intact steam generators, the high head safety injection pumps will be turned off and plant operating procedures will be followed in cooling the plant to hot shutdown conditions.

15.2.8.2 Analysis of Effects and Consequences

Method of Analysis

A detailed analysis using the LOFTRAN Code (Reference 3) is performed in order to determine the plant transient following a feedwater line break.

The code describes the plant neutron kinetics, RCS including natural circulation, pressurizer, steam generators and feedwater system, and computes pertinent variables including the pressurizer pressure, pressurizer water level, and reactor coolant average temperature.

The limiting cases analyzed (with and without offsite power) assume a double ended break of the largest feedwater pipe at full power. Major assumptions made in the analyses are as follows:

a. The initial power level bounds 100% of the nominal NSSS power level of 3672 MWt (including power uncertainty).

b. Initial reactor coolant average temperature is 9.1°F above the nominal value which includes a 1.5°F bias. The initial pressurizer pressure is 43 psi below the nominal value.

c. No credit is taken for the pressurizer spray. To determine that sufficient decay heat removal capability is maintained, this analysis does assume operation of the pressurizer power-operated relief valve in order to minimize RCS pressure (Tsat).

d. Initial pressurizer level is at the nominal programmed value for the cases that assume the condensate storage tank for the auxiliary feedwater source. An evaluation of the effect of the pressurizer level uncertainty concluded that this analysis remains valid. For the cases that assume the Essential Service Water system for the auxiliary feedwater source, the initial pressurizer level is at the nominal programmed value plus 5% span.

e. Initial steam generator water level is at the nominal value plus 5% NRS in the faulted steam generator and at the nominal value minus 5% NRS in the intact steam generators.

f. Main feedwater flow to all steam generators is assumed to be lost at the time the break occurs (all main feedwater spills out through the break).

g. A double-ended break area of 0.223 ft2 is assumed based on the maximum flow through the steam generator flow orifices.

h. A conservative feedline break discharge quality is assumed prior to the time the reactor trip occurs, thereby maximizing the time the trip setpoint is reached. After the trip occurs, a saturated liquid discharge is assumed until all the water inventory is discharged from the faulted steam generator. This minimizes the heat removal capability of the faulted steam generator.

i. Reactor trip is assumed to be initiated when the steam generator water level reaches the low-low steam generator water level setpoint of 18.6% of the narrow range span in the faulted steam generator, to allow for error.

j. The auxiliary feedwater system is actuated by the low-low steam generator water level signal. The auxiliary feedwater system is assumed to supply a total of 453 gpm for all steam generator pressures. This is consistent with the auxiliary feedwater system control valves functioning as designed (to control flow to 160 gpm less measurement uncertainties).

   The air supply to the control valves has been modified to provide a 30 minute safety related supply of air. After the 30 minute supply of air is exhausted, the control valves will fail open. Since the supply of air is limited, the case in which the valves are assumed to conservatively fail (open) at event initiation has also been considered. Flow may be diverted out the break, but more flow may be provided to the intact steam generators at lower pressure. This case has been found to be less limiting. For this case, credit is taken for operators to isolate the faulted steam generator 20 minutes after the reactor trip.

k. Separate cases are analyzed assuming different sources for auxiliary feedwater resulting in different delay times for the initiation of auxiliary feedwater flow.

For the cases where the condensate storage tank is the auxiliary feedwater source and offsite power is available, a 55 second delay is assumed following the low-low steam generator water level setpoint being reached to allow time for sensor response, signal processing, and startup of the pump. For the cases where offsite power is lost, the delay time assumed is 63 seconds to account for additional delays for startup of the emergency diesel generators. In all of the cases analyzed, there is an additional 181 seconds before the feedwater lines are purged and the relatively cold (120°F) auxiliary feedwater enters the intact steam generators.

Additional cases assume the condensate storage tank is not available such that a switch is made to the Essential Service Water system as the source for the auxiliary feedwater system, which causes an additional 12-second delay to initiate the auxiliary feedwater flow. For the case with offsite power available, a 67 second delay is assumed following the low-low steam generator water level setpoint being reached to allow time for sensor response, signal processing, provision of sufficient water from the Essential Service Water system to operate the pump, and startup of the pump. For the case where offsite power is lost, the delay time assumed is 75 seconds to account for additional delays for startup of the emergency diesel generators. In all of the cases analyzed, there is an additional 181 seconds before the feedwater lines are purged and the relatively cold (102°F for Byron and 104°F for Braidwood) auxiliary feedwater enters the intact steam generators.

The limiting results were calculated for Unit 2 modeling the condensate storage tank as the AFW source.

l. Credit is taken for heat energy deposited in some portions of the RCS metal during the RCS heatup.

m. No credit is taken for charging or letdown.

n. Steam generator heat transfer area is assumed to decrease as the shell side liquid inventory decreases.

o. Conservative core residual heat generation is assumed based upon long-term operation at the initial power level preceding the trip (Reference 4).

p. No credit is taken for the following potential protection logic signals to mitigate the consequences of the accident:

1. high pressurizer pressure,

2. overtemperature ΔT,

3. high pressurizer level, and

4. high containment pressure.

q. To account for potential variations associated with steam generator tube plugging, the analysis considers conservatively a maximum loop-to-loop flow variation of 7% and a maximum tube plugging level of 10% in any steam generator.

r. Pressurizer heaters are modeled in the cases with offsite power available since their operation slightly reduces the margin to the acceptance criterion.

s. A maximum steam generator blowdown valve leak rate of 3 gpm per steam generator was accounted for in the analysis.

Plant characteristics and initial conditions are further discussed in Subsection 15.0.3.

Only reactor control systems that reduce the margin to the acceptance criteria are assumed to function including the power-operated relief valves as previously discussed and, for cases with offsite power available, pressurizer heaters. The reactor protection system is required to function following a feedwater line break as analyzed here. No single active failure will prevent operation of this system.

The engineered safety systems that are assumed to function are the safety injection system, auxiliary feedwater system, and the main steamline isolation system. One train of safety injection has been assumed to be available. For the auxiliary feedwater system, the worst-case configuration has been assumed, i.e., three intact steam generators receive auxiliary feedwater following the break. The diesel-driven auxiliary feedwater pump has been assumed to fail also. The motor-driven auxiliary feedwater pump is assumed to deliver flow to each of the three intact steam generators with the auxiliary feedwater control valves operating properly. This configuration normally controls the flow to each steam generator at 160 gpm; however, after measurement uncertainties are considered, the minimum flow of 151 gpm is assumed to be provided to each of the three intact steam generators. Receipt of a low steamline pressure signal in at least one steam line initiates a steamline isolation signal that closes the main steamline isolation valves in all of the steam lines.

Following the trip of the reactor coolant pumps for the feedline break without offsite power, there will be a flow coastdown until flow in the loops reaches the natural circulation value. The natural circulation capability of the RCS has been shown in Subsection 15.2.6, for the loss of ac power transient, to be sufficient to remove core decay heat following reactor trip.

Pump coastdown characteristics are demonstrated in Subsections 15.3.1 and 15.3.2 for single and multiple reactor coolant pump trips, respectively.

A detailed description and analysis of the safety injection system is provided in Section 6.3. The auxiliary feedwater system is described in Subsection 10.4.9. The main steamline isolation system is described in Subsection 10.3.2.

Operator Action

The only operator action for which credit is taken in this transient is ensuring an adequate supply of auxiliary feedwater to maintain the secondary heat sink. This includes isolation of the faulted steam generator 20 minutes after reactor trip. This is not modeled in the limiting case, because the control valves are assumed to be operating correctly. The only other possible operator action before 20 minutes is the termination of safety injection (SI) according to the plant emergency operating procedures. However, the manual termination of SI is not required for this event since continued operation of SI would not degrade the results of this analysis.

Results

The limiting results were calculated for Unit 2 modeling the condensate storage tank AFW source. Calculated plant parameters following a major feedwater line break analyzed to maximize the potential for losing RCS inventory are shown in Figures 15.2-13 through 15.2-20. Results for the case with offsite power available are presented in Figures 15.2-13 through 15.2-16. Results for the case where offsite power is lost are presented in Figures 15.2-17 through 15.2-20. The calculated sequence of events for both cases analyzed is listed in Table 15.2-1.

Figures 15.2-13 and 15.2-17 show that following reactor trip, the plant remains subcritical. RCS pressure will be maintained at the power-operated relief valve setpoint until safety injection flow is terminated by the operator, as mentioned in Subsection 15.2.8.2. However, the reactor core remains covered with water throughout the transient, as water relief due to thermal expansion is limited by the heat removal capability of the auxiliary feedwater system and makeup is provided by the safety injection system.

The major difference between the two cases can be seen in the plots of hot and cold leg temperatures, Figure 15.2-15 (with offsite power available) and Figure 15.2-19 (without offsite power). The case with offsite power results in a more severe rise in temperature. The pressurizer fills for the case with power due to the increased coolant expansion resulting from the reactor coolant pump heat addition; hence, water is relieved for the case with power. As previously stated, however, the core remains covered with water for both cases.

15.2.8.3 Radiological Consequences

The feedwater line break with the most significant consequences would be one that occurred inside the containment between a steam generator and the feedwater check valve. In this case, the contents of the steam generator would be released to the containment. The radioactivity released would be less than that for the steamline break, as analyzed in Subsection 15.1.5.3. Furthermore, automatic isolation of the containment would further reduce any radiological consequences from this postulated accident.

15.2.8.4 Conclusions

Results of the analyses show that for the postulated feedwater line break, the assumed auxiliary feedwater system capacity is adequate to remove decay heat, to prevent overpressurizing the RCS, and to prevent uncovering the reactor core. Radiological doses from the postulated feedwater line break would be less than those previously presented for the postulated steam line break.

15.2.9 References

1. Mangan, M. A., "Overpressure Protection for Westinghouse Pressurized Water Reactors," WCAP-7769, Rev. 1, June, 1972.

2. Deleted.

3. Burnett, T. W. T., et al., "LOFTRAN Code Description," WCAP-7907-P-A (Proprietary), WCAP 7907-A (Non-Proprietary), April 1984.

4. "American National Standard for Decay Heat Power in Light Water Reactors," ANSI/ANS-5.1-1979, August 1979.

5. Freidland, A. J., et al., "Revised Thermal Design Procedure," WCAP-11397-P-A (Proprietary), and WCAP-11397-A (Non-Proprietary), April 1989.

6. NRC Information Notice 94-60, “Potential Overpressurization of the Main Steam System,” August 22, 1994.

TABLE 15.2-1

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH CAUSE A DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

ACCIDENT	EVENT	TIME (sec)
Turbine Trip		
1. Unit 1 RCS Pressure Case	Turbine trip, loss of main feedwater flow	0.0
	High pressurizer pressure reactor trip setpoint reached	5.4
	Rods begin to drop	7.4
	Peak pressurizer pressure	7.6
	Initiation of steam release from steam generator safety valves	9.8
2. Unit 1 MSS Pressure Case	Turbine trip, loss of main feedwater flow	0.0
	Initiation of steam release from steam generator safety valves	3.4
	Overtemperature ΔT reactor trip setpoint reached	3.7
	Rods begin to drop	11.7
	Maximum SG pressure occurs	15.6
3. Unit 1 DNB Case	Turbine trip, loss of main feedwater flow	0.0
	Overtemperature ΔT reactor trip setpoint reached	5.4
	Initiation of steam release from steam generator safety valves	6.0

TABLE 15.2-1(Cont'd)

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH CAUSE A DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

ACCIDENT	EVENT	TIME (sec)
	Rods begin to drop	13.4
	Minimum DNBR occurs	14.1
4. Unit 2 RCS Pressure Case	Turbine trip, loss of main feedwater flow	0.0
	High pressurizer pressure reactor trip setpoint reached	4.5
	Initiation of steam release from steam generator safety valves	5.9
	Peak pressurizer pressure occurs	6.3
	Rods begin to drop	6.5
5. Unit 2 MSS Pressure Case	Turbine trip, loss of main feedwater flow	0.0
	Overtemperature ΔT reactor trip setpoint reached	1.8
	Initiation of steam release from steam generator safety valves	4.8
	Rods begin to drop	9.8
	Maximum SG pressure occurs	15.1

TABLE 15.2-1 (Cont'd)

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH CAUSE A

DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

ACCIDENT	EVENT	TIME (sec)
6. Unit 2 DNB Case	Turbine trip, loss of main feedwater flow	0.0
	Overtemperature ΔT reactor trip setpoint reached	3.5
	Initiation of steam release from steam generator safety valves	7.9
	Rods begin to drop	11.5
	Minimum DNBR occurs	12.6

TABLE 15.2-1 (Cont'd)

ACCIDENT	EVENT	TIME (sec)
Loss of Nonemergency a-c Power	Main feedwater flow stops	10.1
	Low-low steam generator water level trip	67.4
	Rods begin to drop	69.4
	Reactor coolant pumps begin to coastdown	71.4
	Opening of SG safety valves	~75.6
	Four steam generators begin to receive auxiliary feedwater from one motor driven auxiliary feedwater pump	142.4
	Core decay heat decreases to auxiliary feedwater heat removal capacity	~328
	Cold auxiliary feedwater is delivered to the steam generators	696.0
	Peak water level in pressurizer occurs	800.0

TABLE 15.2-1 (Cont'd)

ACCIDENT	EVENT	TIME (sec)
Loss of Nonemergency a-c Power	Main feedwater flow stops	10.1
	Low-low steam generator water level trip	67.4
	Rods begin to drop	69.4
	Reactor coolant pumps begin to coastdown	71.4
	Opening of SG safety valves	~75.6
	Four steam generators begin to receive auxiliary feedwater from one motor driven auxiliary feedwater pump	142.4
	Core decay heat decreases to auxiliary feedwater heat removal capacity	~328
	Cold auxiliary feedwater is delivered to the steam generators	696.0
	Peak water level in pressurizer occurs	790.0

TABLE 15.2-1 (Cont'd)

ACCIDENT	EVENT	TIME (sec)
Loss of Normal Feed-water Flow	Main feedwater flow stops	10
	Low-low steam generator water level trip	51.3
	Rods begin to drop	53.3
	Four steam generators begin to receive auxiliary feedwater from one motor-driven auxiliary feedwater pump	106.3
	Cold auxiliary feedwater is delivered to the steam generators	302
	Peak water level in pressurizer occurs	1774
	Core decay heat plus pump heat decreases to auxiliary feedwater heat removal capacity	~1950

TABLE 15.2-1 (Cont'd)

ACCIDENT	EVENT	TIME (sec)
Feedwater System Pipe Break		
1. With offsite power available	Main feedline break occurs	10
	Low-low steam generator water level reactor trip setpoint reached in faulted steam generator	32
	Rods begin to drop	34
	One motor-driven auxiliary feedwater pump starts and supplies three intact steam generators	87
	Low steam line pressure setpoint reached in faulted steam generator	204
	All main steam line isolation valves close	212
	Cold auxiliary feedwater is delivered to intact steam generators	268
	Steam generator safety valve setpoint reached in intact steam generators	510
	Pressurizer water relief begins	1234
	Core decay heat and pump heat decreases to auxiliary feedwater heat removal capacity	$\sim$4900

TABLE 15.2-1 (Cont'd)

ACCIDENT	EVENT	TIME (sec)
2. Without offsite power	Main feedline break occurs	10
	Low-low steam generator level reactor trip setpoint reached in faulted steam generator	32
	Rods begin to drop	34
	Power lost to the reactor coolant pumps.	36
	One auxiliary feedwater pump starts and supplies three intact steam generators	95
	Low steam line pressure setpoint reached in faulted steam generator	256
	All main steam line isolation valves close	264
	Cold auxiliary feedwater is delivered to intact steam generators	276
	Steam generator safety valve setpoint reached in intact steam generators	795
	Core decay heat decreases to auxiliary feedwater heat removal capacity	$\underset{\sim}{}$1800

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TABLE 15.2-4

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15.3 DECREASE IN REACTOR COOLANT SYSTEM FLOW RATE

A number of faults are postulated which could result in a decrease in reactor coolant system flow rate. These events are discussed in this section. Detailed analyses are presented for the most limiting of these events.

Discussions of the following flow decrease events are presented in Section 15.3:

a. partial loss of forced reactor coolant flow,

b. complete loss of forced reactor coolant flow,

c. reactor coolant pump shaft seizure (locked rotor), and

d. reactor coolant pump shaft break.

Item a. above is considered to be an ANS Condition II event, item b. an ANS Condition III event, and items c. and d. ANS Condition IV events. Subsection 15.0.1 contains a discussion of ANS classifications.

15.3.1 Partial Loss of Forced Reactor Coolant Flow

15.3.1.1 Identification of Causes and Accident Description

A partial loss of coolant flow accident can result from a mechanical or electrical failure in a reactor coolant pump, or from a fault in the power supply to the pump or pumps supplied by a reactor coolant pump bus. If the reactor is at power at the time of the accident, the immediate effect of loss of coolant flow is a rapid increase in the coolant temperature. This increase could result in DNB with subsequent fuel damage if the reactor is not tripped promptly.

Normal power for two of the reactor coolant pumps is supplied through individual buses connected to the generator, whereas the other two reactor coolant pumps are supplied from offsite power. When a generator trip occurs, the buses which are normally fed from the generator are automatically transferred to an offsite power supply. The pumps will continue to supply coolant flow to the core. Following any turbine trip where there are no electrical faults or thrust bearing failure, which require tripping the generator from the network, the generator remains connected to the network for approximately 30 seconds. The two reactor coolant pumps normally fed from the generator remain connected to the generator thus ensuring full flow for approximately 30 seconds after the reactor trip before any transfer is made.

This event is classified as an ANS Condition II incident (an incident of moderate frequency) as defined in Subsection 15.0.1.

The necessary protection against a partial loss of coolant flow accident is provided by the low primary coolant flow reactor trip signal which is actuated in any reactor coolant loop by two out of three low flow signals. Above Permissive 8 (refer to Table 7.2-2 for a discussion of permissives), low flow in any loop will actuate a reactor trip. Between approximately 10% power (Permissive 7) and the power level corresponding to Permissive 8, low flow in any two loops will actuate a reactor trip. A reactor trip signal from the pump breaker position is provided as a backup to the low flow signal. When operating above Permissive 7, a breaker open signal from any two pumps will actuate a reactor trip. Reactor trip on reactor coolant pump breakers open signal is blocked below Permissive 7.

15.3.1.2 Analysis of Effects and Consequences

Method of Analysis

One case has been analyzed:

Loss of two pumps with four loops in operation.

This transient is analyzed by two digital computer codes. First, the LOFTRAN Code (Reference 1) is used to calculate the loop and core flow during the transient, the time of reactor trip based on the calculated flows, the nuclear power transient, and the primary system pressure and temperature transients. The VIPRE code (see Section 4.4) is then used to calculate the hot-channel heat flux transient and DNBR, based on the nuclear power and RCS temperature (enthalpy), pressure, and flow from LOFTRAN. The DNBR transients presented represent the minimum of the typical or thimble cell.

This accident is analyzed with the revised thermal design procedure as described in WCAP-11397-P-A (Reference 3). Plant characteristics and initial conditions are discussed in Subsection 15.0.3.

Initial Conditions

Initial reactor power (consistent with uprated power conditions) and pressure are assumed to be at their nominal values. The initial temperature is assumed to be at the nominal value for the high T_{avg} program plus a 1.5°F bias. Uncertainties in initial conditions are included in the limit DNBR as described in WCAP-11397-P-A.

Reactivity Coefficients

A conservatively large absolute value of the Doppler-only power coefficient is used (see Figure 15.0-3). This is equivalent to a total integrated Doppler reactivity from 0% to 100% power of 0.016 δk.

A moderator temperature coefficient (MTC) of 0 pcm/°F is assumed in the analysis. The use of this MTC is consistent with the analysis initial conditions assumptions and corresponds to the applicable MTC limit at hot full power (HFP) initial conditions. The HFP analysis results using a 0 pcm/°F MTC bound those for part-power initial conditions with a positive MTC at the licensed allowable MTC limit.

Flow Coastdown

The flow coastdown analysis is based on a momentum balance around each reactor coolant loop and across the reactor core. This momentum balance is combined with the continuity equation, a pump momentum balance and the pump characteristics and is based on high estimates of system pressure losses.

Plant systems and equipment credited for mitigating the effects of the accident are discussed in Subsection 15.0.8 and listed in Table 15.0-7. No single active failure in any of these systems or equipment will adversely affect the consequences of the accident.

Results

Figures 15.3-1 and 15.3-2 show the transient response for the loss of two reactor coolant pumps with four loops in operation. The reactor is assumed to be tripped on a low flow signal. Figure 15.3-2 shows the DNBR to be greater than the limit value.

For the case analyzed, since DNB does not occur, the ability of the primary coolant to remove heat from the fuel rod is not greatly reduced. Thus, the average fuel and clad temperatures do not increase significantly above their respective initial values.

The calculated sequence of events for the case analyzed is shown on Table 15.3-1. The affected reactor coolant pumps will continue to coast down, and the core flow will reach a new equilibrium value corresponding to the number of pumps still in operation. With the reactor tripped, a stable plant condition will eventually be attained. Normal plant shutdown may then proceed.

15.3.1.3 Radiological Consequences

A partial loss of reactor coolant flow from full load would result in a reactor and turbine trip. Assuming that the condenser is not available, atmospheric steam dump may be required.

There are only minimal radiological consequences associated with this event. Therefore, this event is not limiting.

The radiological consequences resulting from atmospheric steam dump would be less severe than the steamline break event analyzed in Subsection 15.1.5.3.

15.3.1.4 Conclusions

The analysis shows that the DNBR will not decrease below the limit value at any time during the transient. Thus, no fuel or clad damage is predicted, and all applicable acceptance criteria are met.

The radiological consequences of this event would be less than the steamline break event analyzed in Subsection 15.1.5.3.

15.3.2 Complete Loss of Forced Reactor Coolant Flow

15.3.2.1 Identification of Causes and Accident Description

A complete loss of forced reactor coolant flow may result from a simultaneous loss of electrical supplies to all reactor coolant pumps. If the reactor is at power at the time of the accident, the immediate effect of loss of coolant flow is a rapid increase in the coolant temperature. This increase could result in DNB with subsequent fuel damage if the reactor were not tripped promptly.

Normal power for two of the reactor coolant pumps is supplied through buses from a transformer connected to the generator, whereas the other two reactor coolant pumps are supplied from offsite power. When a generator trip occurs, the buses which are normally fed from the generator are automatically transferred to an offsite power supply. The pumps will continue to supply coolant flow to the core. Following any turbine trip where there are no electrical faults or thrust bearing failure which require tripping the generator from the network, the generator remains connected to the network for approximately 30 seconds. The two reactor coolant pumps normally fed from the generator remain connected to the generator thus ensuring full flow for 30 seconds after the reactor trip before any transfer is made.

This event is classified as an ANS Condition III incident (an infrequent incident) as defined in Subsection 15.0.1.

The following signals provide the necessary protection against a complete loss of flow accident:

a. Reactor coolant pump power supply undervoltage or underfrequency.

b. Low reactor coolant loop flow.

The reactor trip on reactor coolant pump bus undervoltage is provided to protect against conditions which can cause a loss of voltage to all reactor coolant pumps, i.e., loss of nonemergency a-c power. This function is blocked below approximately 10% power (Permissive 7).

The reactor trip on reactor coolant pump underfrequency is provided to trip the reactor for an underfrequency condition, resulting from frequency disturbances on the power grid. Reference 4 provides analyses of grid frequency disturbances and the resulting nuclear steam supply system (NSSS) protection requirements which are generally applicable.

The reactor trip on low primary coolant loop flow is provided to protect against loss of flow conditions which affect only one reactor coolant loop. This function is generated by two out of three low flow signals per reactor coolant loop. Above Permissive 8, low flow in any loop will actuate a reactor trip. Between approximately 10% power (Permissive 7) and the power level corresponding to Permissive 8, low flow in any two loops will actuate a reactor trip.

15.3.2.2 Analysis of Effects and Consequences

The following cases for complete loss of forced reactor coolant flow are analyzed:

1. Complete loss of all four reactor coolant pumps (RCPs) with four loops in operation; and

2. Frequency decay event resulting in a complete loss of forced reactor coolant flow.

Case 1 of the complete loss of flow event assumes that the RCPs begin to coastdown upon reaching a undervoltage trip setpoint (modeled to occur at t=0 seconds in this analysis). Rod motion following the undervoltage trip is modeled to occur at t=1.5 seconds, reflecting an undervoltage trip time delay of 1.5 seconds. For the underfrequency complete loss of flow event (Case 2), a frequency decay of 5 Hz/sec is assumed to occur at t=0 seconds, decreasing RCS flow to all loops. At t=1.2 seconds, the underfrequency trip setpoint of 54.0 Hz is reached. Rod motion occurs at t=1.8 seconds, following a 0.6 second underfrequency trip time delay.

These transients are analyzed by two digital computer codes. First, the LOFTRAN code (Reference 1) is used to calculate the loop and core flow transients, the nuclear power transient, and the primary system pressure and temperature transients. The flow coastdown analysis performed by LOFTRAN is based on a momentum balance around each reactor coolant loop and across the reactor core.

This momentum balance is combined with the continuity equation, a pump momentum balance, and the as-built pump characteristics and is based on conservative system pressure loss estimates.

The VIPRE code (see Section 4.4) is then used to calculate the hot-channel heat flux transient and DNBR, based on the nuclear power and RCS temperature (enthalpy), pressure, and flow from LOFTRAN. The DNBR results are based on the minimum of the typical and thimble cells.

Results

Figures 15.3-3 and 15.3-4 show the transient response for the limiting case; the frequency decay complete loss of flow event (Case 2). The reactor is assumed to be tripped on an underfrequency signal. Figure 15.3-4 shows the DNBR to be greater than the limit value.

Since DNB does not occur for the cases analyzed, the ability of the primary coolant to remove heat from the fuel rod is not greatly reduced. Thus, the average fuel and clad temperatures do not increase significantly above their respective initial values.

The calculated sequence of events for the case analyzed is shown on Table 15.3-1. The speed of the reactor coolant pumps will continue to decrease from the 5 hz/sec frequency decay until a pump trip occurs on an underfrequency condition. Following pump trip, the reactor coolant pumps will continue to coast down, and natural circulation flow will eventually be established, as demonstrated in Subsection 15.2.6. With the reactor tripped, a stable plant condition would be attained. Normal plant shutdown may then proceed.

15.3.2.3 Radiological Consequences

A complete loss of reactor coolant flow from full load results in a reactor and turbine trip. Assuming, in addition, that the condenser is not available, atmospheric steam dump would be required. The quantity of steam released would be the same as for a loss of offsite power.

There are only minimal radiological consequences associated with this event. Therefore, this event is not limiting.

The radiological consequences resulting from atmospheric steam dump would be less severe than the steamline break analyzed in Subsection 15.1.5.

15.3.2.4 Conclusions

The analysis performed has demonstrated that for the complete loss of forced reactor coolant flow, the DNBR does not decrease below the limit value at any time during the transient. Thus, no fuel or clad damage is predicted, and all applicable acceptance criteria are met.

15.3.3 Reactor Coolant Pump Shaft Seizure (Locked Rotor)

15.3.3.1 Identification of Causes and Accident Description

A transient analysis is performed for the instantaneous seizure of an RCP rotor (locked rotor). Flow through the affected reactor coolant loop is rapidly reduced, leading to a reactor trip on a low flow signal. Following the trip, heat stored in the fuel rods continues to pass into the core coolant, causing the coolant to expand. At the same time, heat transfer to the shell side of the steam generator is reduced, first because the reduced flow results in a decreased tube side film coefficient, and then because the reactor coolant in the tubes cools down while the shell side temperature increases (turbine steam flow is reduced to zero upon plant trip). The rapid expansion of the coolant in the reactor core, combined with the reduced heat transfer in the steam generator causes an insurge into the pressurizer and a pressure increase throughout the RCS. The insurge into the pressurizer causes a pressure increase which in turn actuates the automatic spray system, opens the power-operated relief valves (PORVs), and opens the pressurizer safety valves. The sequence of events initiated by the insurge depends on the rate of insurge and pressure increase. The PORVs are designed for reliable operation and would be expected to function properly during the accident. However, for conservatism, their pressure-reducing effect as well as the pressure-reducing effect of the spray is not included in this analysis.

This event is classified as an ANS Condition IV incident (a limiting fault) as defined in Subsection 15.0.1.

The consequences of a locked rotor (i.e., an instantaneous seizure of a pump shaft) are very similar to those of a pump shaft break. The initial rate of the reduction in coolant flow is slightly greater for the locked rotor event. However, with a broken shaft, the impeller could conceivably be free to spin in the reverse direction. The effect of reverse spinning is to decrease the steady-state core flow when compared to the locked rotor scenario. Only one analysis, which permits reverse spinning but no forward flow, has been performed and represents the most limiting condition for the locked rotor and pump shaft break accidents.

15.3.3.2 Analysis of Effects and Consequences

Method of Analysis

Two digital computer codes are used to analyze this transient. The LOFTRAN Code (Reference 1) is used to calculate the resulting loop and core flow transients following the pump seizure, the time of reactor trip based on the loop flow transients, the nuclear power following reactor trip, and the peak RCS pressure. The thermal behavior of the fuel located at the core hot spot is investigated using the FACTRAN Code (Reference 2), which uses the core flow and the nuclear power values calculated by LOFTRAN. The FACTRAN Code includes a film boiling heat transfer coefficient.

One case is analyzed: one locked rotor/shaft break with four loops in operation, concurrent with a loss of offsite power at the time of trip.

Initial Operating Conditions

At the beginning of the postulated locked rotor accident, the plant is assumed to be operating at nominal reactor power consistent with uprated power conditions and pressure. Initial temperature is assumed to be nominal plus a 1.5°F bias. The revised thermal design procedure as described in WCAP-11397-P-A (Reference 3) is used. Plant characteristics and initial conditions are discussed in Subsection 15.0.3.

For the peak pressure and peak clad temperature evaluations, one analysis is performed and the initial pressure is conservatively estimated as 43 psi above the nominal pressure of 2250 psia to allow for errors in the pressurizer pressure measurement and control channels. This is done to obtain the highest possible rise in the coolant pressure during the transient. To obtain the maximum pressure in the primary side, conservatively high loop pressure drops are added to the calculated pressurizer pressure. The pressure response shown in Figure 15.3-6 is at the point in the RCS having the maximum pressure (i.e., the outlet of the faulted loop's RCP). The remainder of the plant is assumed to be operating under the most adverse steady-state operating condition, e.g., 100% of the NSSS MUR thermal power including applicable uncertainties and the maximum steady-state coolant average temperature, including uncertainties and a 1.5°F bias.

For a conservative analysis of fuel rod behavior, the hot spot evaluation assumes that DNB occurs at the initiation of the transient and continues throughout the event. This assumption reduces heat transfer to the coolant and results in conservatively high hot spot temperatures.

The reactor coolant flow coastdown analysis is based on a momentum balance around each reactor coolant loop and across the reactor core. This momentum balance is combined with continuity equation, a pump momentum balance, and the as-built pump characteristics and is based on high estimates of system pressure losses. No single active failure in any of these systems or equipment will adversely affect the consequences of the accident.

A conservatively large absolute value of the Doppler-only power coefficient is used (see Figure 15.0-3). The total integrated Doppler reactivity from 0 to 100% power is assumed to be 0.016 δK. A moderator temperature coefficient (MTC) of 0 pcm/°F is assumed in the analysis. The use of this MTC is consistent with the analysis initial conditions assumptions and corresponds to the applicable MTC limit at hot full power (HFP) initial conditions. The HFP analysis results using a 0 pcm/°F MTC bound those for part-power initial conditions with a positive MTC at the licensed allowable MTC limit. For this analysis, the curve of trip reactivity versus time (Figure 15.0-6) was used with a 4% δK trip reactivity, which includes the most reactive rod cluster control assembly (RCCA) stuck out of the core.

Evaluation of the Pressure Transient

A detailed model was used to determine the peak pressure in the RCS under postulated accident conditions and to obtain the neutron flux response as a function of time, which is used later in the analysis.

After pump seizure, neutron flux is rapidly reduced because of the control rod insertion upon plant trip. In this analysis, rod motion is assumed to begin one second after the flow in the affected loop reached 85.1% of nominal flow.

No credit was taken for the pressure-reducing effect of the pressurizer relief valves, pressurizer spray, steam dump or controlled feedwater flow after plant trip. Although these operations are expected to occur and would result in a lower peak pressure, an additional degree of conservatism is provided by ignoring their effect.

Upon actuation of the pressurizer safety valves at an opening pressure of 2549.9 psia (including 2% allowance for drift and 1% for pressure shift), purge of the water in the safety valve loop seal occurs and full valve relief capacity is achieved within 1 second. The pressurizer safety valves capacity for steam relief is described in Chapter 5.

Evaluation of DNB in the Core During the Accident

For this accident, DNB is assumed to occur in the core. Therefore, an evaluation of the consequences with respect to fuel rod thermal transients is performed. Results obtained from analysis of this hot spot condition represent the upper limit with respect to clad temperature and zirconium water reaction. In the evaluation, the rod power at the hot spot is assumed to be 2.6 times the average rod power (i.e., F_Q=2.6) at the initial core power level.

Calculation of the extent of DNB in the core during the locked rotor event is performed with the VIPRE computer code (See Section 4.4). The VIPRE code calculates the hot-channel heat flux transient and DNBR, based on the nuclear power and RCS temperature (enthalpy), pressure, and flow from LOFTRAN.

Film Boiling Coefficient

The film boiling coefficient is calculated in the FACTRAN Code using the Bishop-Sandberg-Tong film boiling correlation (Reference 5). The fluid properties are evaluated at film temperature (average between wall and bulk temperatures). The program calculates the film coefficient at every time step based upon the actual heat transfer conditions at the time. The neutron flux, system pressure, bulk density, and mass flow rate as a function of time are used as program input.

For this analysis, the initial values of the pressure and bulk density are used throughout the transient since they are the most conservative with respect to the clad temperature response. As indicated earlier, DNB was assumed to start at the beginning of the accident.

Fuel-Clad Gap Coefficient

The magnitude and time dependence of the heat transfer coefficient between fuel and cladding (gap coefficient) have a pronounced influence on the thermal results. The larger the value of the gap coefficient, the more heat is transferred between pellet and clad. For the first part of the transient, a high gap coefficient produces higher clad temperatures since the heat stored and generated in the fuel pellet tries to redistribute itself in the cooler clad. This effect of the gap coefficient is reversed when the clad temperature exceeds the pellet temperature in cases where a zirconium-steam reaction is present. The gap coefficient was taken to be the conservatively large value of 10,000 Btu/hr-ft^2-°F which is greater than the highest value calculated during core life.

Zirconium-Steam Reaction

The zirconium-steam reaction can become significant above 1800°F (clad temperature). In order to take this phenomenon into account, the following correlation, which defines the rate of the zirconium-steam reaction, was introduced into the model (see Reference 4):

$$\frac{d(w^2)}{dt} = (33.3 \times 10^6)\, e^{-(45500/1.986T)}$$

where w = amount reacted, mg/cm^2
t = time, seconds
T = temperature, K

Results

The transient results without offsite power available are shown in Figures 15.3-5 through 15.3-6. The peak RCS pressure reached during the transient is less than that which would cause stresses to exceed the faulted condition stress limits. Also, the peak clad surface temperature is considerably less than the limit of 2375°F (associated with Optimized ZIRLO™ fuel cladding which bounds the 2700°F limit for ZIRLO® fuel cladding, Reference 7). It should be noted that the clad temperature was conservatively calculated assuming the DNB occurs at the initiation of the transient. The results of these calculations (peak pressure, peak clad temperature, and zirconium-steam reaction) are also summarized in Table 15.3-2.

The calculated sequence of events is shown in Table 15.3-1. Figure 15.3-6 shows that the core flow rapidly coasts down to a new equilibrium value. With the reactor tripped, a stable plant condition will eventually be attained. Normal plant shutdown may then proceed.

15.3.3.3 Conclusion

The results of the transient analysis show that 0.10% of the fuel rods will experience DNBR values below the safety analysis limit value. This is considerably less than the transient limit of 2.0%.

Since the peak RCS pressure reached during the transient is less than that which would cause stresses to exceed the faulted condition stress limits, the integrity of the primary coolant system is not endangered. Also, since the peak clad average temperature calculated for the hot spot during the worst transient remains considerably less than the limit of 2375°F (associated with Optimized ZIRLO™ fuel cladding which bounds the 2700°F limit for ZIRLO® fuel cladding, Reference 7) the core will remain in place and intact with no loss of core cooling capability.

15.3.3.4 Radiological Consequences

The evaluation of the radiological consequences of a postulated seizure of a reactor coolant pump rotor (Locked Rotor Accident-LRA) assumes that the reactor has been operating with a small percent of defective fuel and leaking steam generator tubes for sufficient time to establish equilibrium concentrations of radionuclides in the reactor coolant and in the secondary coolant. A concurrent PORV failure is also assumed.

It is assumed conservatively that, as a result of the locked rotor accident, 2% of the fuel rods in the core undergo sufficient clad damage to result in the release of their gap activity.

As a result of the accident, radionuclides carried by the primary coolant to the steam generators, via the leaking tubes, are released to the environment via the steam line safety or power operated relief valves.

The key inputs and assumptions used in the AST Locked Rotor Accident (LRA) radiological consequence analysis using AST methodology are summarized below and provided in Table 15.3-4.

This LRA analysis postulates the instantaneous seizure of a Reactor Coolant Pump (RCP) rotor, where the reactor is tripped on the subsequent low flow signal. Following the trip, heat stored in fuel rods continues to pass into the reactor coolant, causing the coolant to expand. At the same time, heat transfer to the shell side of the Steam Generator (SG) is reduced, first because the reduced flow results in a decreased tube side film coefficient, and then because the primary reactor coolant in the tubes cools down while the shell side temperature increases (turbine steam flow is reduced to zero upon plant trip). The rapid expansion of the coolant in the reactor core, combined with the reduced heat transfer in the SG, causes an insurge of coolant into the pressurizer and a pressure increase throughout the Reactor Coolant System (RCS). This insurge into the pressurizer causes a pressure increase, which in turn actuates the automatic spray system, opens the power-operated relief valves (PORVs), and also opens the pressurizer safety valves.

The SG PORVs are designed for reliable operation and would be expected to function properly during the accident. However, for conservatism, their pressure reducing effect is ignored, and an SG PORV failure, in the open position at the onset of accident releases, is assumed. In addition, the pressure reducing effect of the spray is also ignored for this analysis.

This evaluation of the radiological consequences of an LRA assumes that the reactor has been operating with a small percent of defective fuel and leaking SG tubes. The reactor is assumed to have been operating in this condition for sufficient time to establish equilibrium concentrations of radionuclides in the reactor coolant and secondary coolant. Additionally, prior to accident initiation, the reactor is postulated to experience an Iodine spike, thereby increasing the RCS iodine activity above that of equilibrium levels.

As a result of this accident, radionuclides carried by the primary coolant to the Steam Generators, via the leaking tubes, are released to the environment via the steam line safety valves or PORVs.

This LRA dose assessment is analyzed using two modeled simulations. The first simulation, Case 1, is modeled to calculate the doses due to the activity that was instantaneously released into the primary RCS from the postulated damaged fuel fraction, and the activity resulting from a pre-accident 60 µCi/gm Dose Equivalent (DE) I-131 spike. Leak and steaming rates are used to model the transport of activity from the RCS to the environment, through the PORVs of intact SGs, and through the failed PORV of the faulted SG, which is the postulated single-failure for this analysis.

The second simulation, Case 2, is modeled to calculate the doses due to 0.1 µCi/gm DE I-131 equilibrium activity existing in the secondary coolant prior to accident initiation. This iodine activity is released using the same partitioned steaming rates that are associated with Case 1 SG PORV release to the environment. For the intact SGs this iodine activity is released the same partitioned steaming rates that are associated with Case 1. However, for the SG with the failed PORV, referred to as the faulted SG, it is postulated that the activity initially contained in the faulted SG is released to the environment for 20 minutes. The failed SG PORV is isolated by locally closing the associated isolation valve. The operator can identify the failed SG PORV when the faulted SG pressure drops below SG PORV reset value. There is also a positive valve position indication of the open SG PORV on the control board. As a result of the failed SG PORV, the operator would enter the emergency procedure for "Faulted Steam Generator Isolation". The dispatch, travel time, and local isolation of this valve by an operator was conservatively assumed to require 20 minutes, from the time the SG PORV fails open.

Fuel Damage and Core Source Term

The LRA core source terms are those associated with a DBA power level of 3658.3 MWt. This power level bounds the MUR power uprate Rated Thermal Power level including measurement uncertainties.

The instantaneous seizure of the RCP rotor associated with the LRA results in the damage of 2% of the core. The design basis of this accident assumes that no fuel melt is postulated to occur. Therefore for Case 1, the source term available for release is associated with this fraction of damaged fuel and the fraction of core activity existing in the gap, plus the iodine in the RCS due to a design basis pre-accident 60 µCi /gm DE I-131 spike, and the noble gas activity associated with 1% fuel defects.

The additional source activity modeled in the second case consists simply of the 0.1 µCi/gm DE I-131 equilibrium secondary coolant activity concentration limit from the Technical Specifications. The total activity available for release from both the intact SGs and the SG with a failed SG PORV are input to the RADTRAD NIF for Case 2.

Activity Release Fractions

Release fractions and transport fractions are per Regulatory Guide 1.183, Appendix G and Table 3. To comply with this regulatory guidance, 5% of the core inventory of iodine and noble gas is assumed to be in the fuel-clad gap, excluding I-131 and Kr-85, where 8% and 10% are assumed, respectively. Additionally, Table 3 of RG 1.183 shows that 12% of the core cesium and rubidium should be assumed to be in the fuel-clad gap. However, to accommodate the consideration of extended fuel burnup, in excess of the RG 1.183 assumptions, all RG 1.183 Table 3 "Other Noble Gases", "Other Halogens", and "Alkali Metals" isotopic release fractions are doubled. Although analyses have shown that isotopic activity fractions in the fuel-clad gap may in fact decrease when "burning" the fuel longer than the 62 GWd/MTU specified in RG 1.183, this 100% increase in the gap fractions is used as an accepted and conservative means of bounding all extended burnup phenomena. All of the gap activity in the damaged fuel is released in its entirety, instantaneously into the RCS and mixed homogeneously therein.

These activity release fractions are input to the RADTRAD code through the use of the Release Fractions and Timing (RFT) file.

Airborne Activity Removal Mechanisms in Containment

As discussed below only decay and leakage are credited.

Decay Credited:

Decay of radioactivity is credited in all compartments, prior to release. This is implemented in RADTRAD using the half-lives in the Nuclide Inventory File (NIF). The RADTRAD decay plus daughter option is used. In reality, daughter products such as xenon from iodines or iodines from tellurium are unlikely to readily escape from the matrix in which the parent iodine or tellurium is contained. Nevertheless, the RADTRAD feature to include daughter effects is selected for conservatism.

Depletion due to Leakage Credited:

For analyses of doses due to release from the RCS volume, the dose results from leakage, and it is reasonable to credit the small amount of depletion from the available RCS activity inventory associated with this leakage. This is calculated inherently by the RADTRAD code.

Release Rates, Steaming Rates, and Partitioning Factors:

Activity that originates in the primary RCS is released to the secondary coolant by means of the primary-to-secondary coolant leak rate. This design basis leak rate value is 0.218 gpm, per intact SG, totaling 0.654 gpm, and 0.5 gpm for the SG with the failed SG PORV. For input into RADTRAD, these rates were converted from gallons per minute to cubic feet per minute, making them 0.02914 cfm, per intact SG, totaling 0.08743 cfm, and 0.06684 cfm for the SG with the failed SG PORV.

Releases to the environment are associated with the secondary coolant steaming from the Steam Generators with intact SG PORVs and releases directly out of an SG with a failed SG PORV. Because of the release dynamic of the activity from the intact SG PORVs, RG 1.183 allows for a reduction in the amount of activity released to the environment based on partitioning of nuclides between the liquid and gas states of water for this release path. For Iodine, the partitioning factor of 0.01 was taken directly from the suggested guidance. However, there is no explicit guidance with regard to other particulate nuclides. Reviewing the specified AST release fractions, it is concluded that the only nuclides other than iodines to be released from the core source term are cesiums, rubidium, and noble gases. For cesiums and rubidium, a bounding partitioning factor of 0.0055 is used, as shown in the applicable ANSI Standard, ANS/ANSI-18.1-1999. This value bounds the actual 0.00529 factor that is shown for Cs-134 in the ANSI Standard, which shows the largest partitioning factor of these such isotopes. Because of the volatility of noble gases, no partitioning is assumed for any such isotopes.

The methodology used to model steaming of activity through intact SG PORVs following the postulated LRA event is applicable to both the Case 1 and Case 2 simulations. This methodology assumes an average cumulative release rate through the SG valves that is reduced in steps. The partitioning factors are applied to these release rates, which were derived from the total time increment mass releases. Incremental steam mass releases are in pounds. Release rates were derived by dividing these totals by the time increment. Then, this data was converted using the assumption of cooled liquid conditions (i.e., 62.4 lbm/ft^3), as specified by the guidance of RG 1.183. The steaming release and primary-to-secondary coolant leakage is postulated to end at 40 hours, when the RCS and secondary loop have equilibrated. The table below shows the time steps, isotopic partitioning factors, and associated release rates:

LRA Partitioning Factors And Associated Release Rates

Time Interval (hrs)	Total Steam Mass Release (lbm)	Iodine Partitioning Factor	Cesium Partitioning Factor	Noble Gas Partitioning Factor	Steam Release Rate for Iodines (cfm)	Steam Release Rate for Cesiums (cfm)	Steam Release Rate for Noble Gases (cfm)
0 - 2.0	719,000	0.01	0.0055	1.0	9.5977E-01	5.2788E-01	9.5977E+01
2.0 - 8.0	1,109,000	0.01	0.0055	1.0	4.9346E-01	2.7140E-01	4.9346E+01
8.0 - 40	2,664,000	0.01	0.0055	1.0	2.2226E-01	1.2224E-01	2.2226E+01
40 - 64	0	0.01	0.0055	1.0	0	0	0

The release rate through the failed SG PORV is conservatively assumed to be un-partitioned, and therefore no isotopic partitioning factors are applied. The rate at which activity is released from this pathway is therefore equal to the primary-to-secondary coolant leak rate discussed above.

In Case 2, for the intact SGs, the iodine activity is released using the same partitioned steaming rates that are associated with Case 1 SG PORV release to the environment. For the faulted SG the release rate is based on a 20-minute failed SG PORV release. The total mass released is 167,000 lbm. Converted to a volume, and divided by the 20-minute release time, this becomes the 133.75 cfm volumetric flow rate.

χ/Q Calculations (Meteorology)

Releases from the SG PORVs are considered elevated releases due to the high steaming rates, and the associated χ/Q's have been reduced by a factor of 5 per guidance in RG 1.194. The atmospheric dispersion factors are given in Table 15.0-17.

Assumptions and Inputs

The following inputs and assumptions were used in the LRA analysis.

a. Core inventory is based on a DBA power level of 3658.3 MWt. This power level bounds the MUR power uprate Rated Thermal Power level including measurement uncertainties.

b. Two percent (2%) of the fuel is damaged during the initiation of this accident, and is assumed to have failed.

c. No fuel melts following the postulated LRA.

d. Five percent (5%) of the core inventory of noble gases and iodines are released from the fuel gap, excluding I-131 and Kr-85, where 8% and 10% are respectively released. Release fractions of other nuclide groups contained in the fuel gap are detailed in Table 3 of Regulatory Guide 1.183, and to account for gap fraction uncertainty due to expected extended fuel burnup, these fractions from the referenced table are doubled.

e. All iodine released from the SGs is assumed to be of the elemental species. This is done for RADTRAD simulation considerations, and is consistent with the RG 1.183 specification of 97% elemental and 3% organic, because elemental and organic iodine are treated identically by the computer model.

f. The Control Room HVAC system is realigned to the emergency mode of operation 30 minutes after the initiation of this design basis accident.

g. The activity released from the fuel from either the gap or from fuel pellets is assumed to be instantaneously mixed with the reactor coolant within the pressure vessel per RG 1.183.

h. SG PORV releases end at 40 hours, when the RCS has seen a large enough reduction in residual heat to no longer require steaming via the PORVs for temperature reduction.

i. A failure of one SG PORV, in the open position, that takes 20 minutes to isolate, is assumed to conservatively maximize activity release.

Dose Results

Radiological doses resulting from a design basis LRA for a control room operator and a person located at EAB or LPZ are to be less than the regulatory dose limits as given below:

Regulatory Dose Limits – LRA

Dose Type	Control Room (rem)	EAB and LPZ (rem)
TEDE Dose	5[a]	2.5[b]

Notes:
[a] 10 CFR 50.67
[b] 10 CFR 50.67 as modified by Regulatory Guide 1.183 (Table 6, Page 1.183-20)

The table below provides the results from the simulations modeled using the RADTRAD 3.03 code, as well as the summed result:

Locked Rotor Accident
Radiological Analysis Results
(Maximum for Both Byron or Braidwood)

Case 1: Doses from Iodine Spike and Fuel Damage RCS Activity Dose Assessment Results		
Control Room (rem TEDE)	EAB (rem TEDE)	LPZ (rem TEDE)
2.729	1.639	0.594
Case 2: Doses from Equilibrium Secondary Coolant Activity Dose Assessment Results		
Control Room (rem TEDE)	EAB (rem TEDE)	LPZ (rem TEDE)
0.061	0.040	0.008
Total Dose from Design Basis Locked Rotor Accident (LRA)		
Control Room (rem TEDE)	EAB (rem TEDE)	LPZ (rem TEDE)
2.790	1.679	0.602

These doses are below the Regulatory Dose Limits, so it is verified that the LRA is sufficiently mitigated at both Byron and Braidwood Stations.

15.3.3.4.1 Source Term

The concentration of nuclides in the primary and secondary system prior to and following the accident are determined as follows:

a. The iodine concentrations in the reactor coolant will be based upon a preaccident iodine spike and 2% failed fuel.

   1. Preaccident Spike - A reactor transient has occurred prior to the LRA and has raised the primary coolant iodine concentration to 60 μCi/gm of Dose Equivalent (DE) I-131 (Table 15.0-10).

   2. Failed Fuel - 2% of the fuel rods in the core suffer clad damage due to the LRA and release all their iodine gap activity to the primary coolant.

b. The noble gas concentrations in the primary coolant are based on 1 percent defective fuel existing prior to the LRA, plus 5% of the inventory of noble gases and iodines released from the fuel gap of the damaged fuel, excluding I-131 and Kr-85, where 8% and 10% are respectively released. Release fractions of other nuclide groups contained in the fuel gap are detailed in Regulatory Guide 1.183 Table 3. To account for gap fraction uncertainty due to expected extended fuel burnup, fractions from Table 3 are doubled.

c. The secondary coolant activity is based on the DE of 0.1 μCi/gm of I-131 (Table 15.0-10).

15.3.3.5 Radiological Conclusions

The resulting radiation doses in the control room, at the exclusion area boundary, and at the low-population zone outer boundary are presented in Tables 15.0-11 and 15.0-12. The doses from this accident are within the NRC's dose acceptance criterion of 10 CFR Part 50.67 guidelines.

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15.3.4 Reactor Coolant Pump Shaft Break

Refer to Section 15.3.3

Pages 15.3-14 through 15.3-16 have been deleted intentionally.

15.3.5 Locked Rotor with Loss of Offsite Power

Subsection 15.3.3 assumes a locked-rotor/shaft break concurrent with a loss of offsite power at time of trip.

15.3.6 References

1. Burnett, T. W. T., et al., "LOFTRAN Code Description", WCAP-7907-P-A (proprietary), WCAP 7907-A (non-proprietary), April 1984.

2. Hargrove, H. G., "FACTRAN - A Fortran - IV Code for Thermal Transients in a $U0_2$ Fuel Rod", WCAP-7908-A, December 1989.

3. Friedland, A.J. and Ray, S., "Revised Thermal Design Procedure," WCAP-11397-P-A (proprietary) and WCAP-11397-A (non-proprietary), April 1989.

4. Baldwin, M. S., Merrian, M. M., Schenkel, H. S. and Van De Walle, D. J., "An Evaluation of Loss of Flow Accidents Caused by Power System Frequency Transients in Westinghouse PWRs," WCAP-8424, Revision 1, June 1975.

5. Bishop, A. A., Sandberg, R. O. and Tong, L. S., "Forced Convection Heat Transfer at High Pressure After the Critical Heat Flux," ASME 65-HT-31, August 1965.

6. NUREG-0800, Standard Review Plan Section 15.3.3-15.3.4, "Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break," Revision 2, July 1981.

7. Shah, H. H., "Optimized ZIRLO™," WCAP-12610-P-A & CENPD-404-P-A Addendum 1-A, July 2006.

TABLE 15.3-1

TIME SEQUENCE OF EVENTS FOR INCIDENTS

WHICH RESULT IN A DECREASE IN REACTOR COOLANT

SYSTEM FLOW

ACCIDENT	EVENT	TIME (sec.)
Partial Loss of Forced Reactor Coolant Flow		
Four loops operating two pumps coasting down	Coastdown begins	0.0
	Low flow reactor trip	1.7
	Rods begin to drop	2.7
	Minimum DNBR occurs	3.8
Complete Loss of Forced Reactor Coolant Flow (Underfrequency)		
Frequency decay to all four RCPs begins		0.0
Underfrequency trip setpoint is reached		1.2
Rods begin to drop		1.8
Minimum DNBR occurs		3.9

TABLE 15.3-1 (Cont'd)

ACCIDENT	EVENT	TIME (sec.)
		Four Loop Operation
Reactor Coolant Pump Locked Rotor/Shaft Break		
	Rotor on one pump locks	0.0
	Low flow trip point reached	0.04
	Rods begin to drop	1.04
	Maximum RCS pressure occurs	3.60
	Maximum clad temperature occurs	3.7

TABLE 15.3-2

SUMMARY OF RESULTS FOR LOCKED ROTOR/SHAFT BREAK TRANSIENT

	FOUR LOOPS OPERATING INITIALLY
Maximum Reactor Coolant System Pressure (psia)	2736
Maximum Cladding Temperature (°F) Core Hot Spot	1954
Zr-H_2O reaction at core hot spot (% by weight)	0.54%

TABLE 15.3-3

THIS TABLE HAS BEEN INTENTIONALLY DELETED.

TABLE 15.3-3a

TABLE 15.3-4

INPUT PARAMETERS FOR THE LOCKED ROTOR ACCIDENT RADIOLOGICAL

CONSEQUENCE ANALYSIS USING AST

Parameter	Unit	Value	Notes
Fraction of Core Experiencing Cladding Damage with Failed-Open PORV	%	2.0	The value of 2% is used, the reload limit for safety analysis, bounding the 0.1% value in the previous design basis calculation.
Gap Fractions: I-131 Kr-85 Other Noble Gases Other Halogens Alkali Metals	 - - - - -	(2 times the following): 0.08 0.10 0.05 0.05 0.12	As a significant number of fuel assemblies not qualifying for AST due to their containing fuel rods with maximum linear heat generation rates exceeding 6.3 kilowatts per foot peak rod average power for burnups exceeding 54 GWD/MTU, the fuel will be treated as having gap fractions a factor of 2 greater than the Regulatory Guide 1.183 values.
SG Iodine Partition Factor SG Aerosol Carryover	- -- -	0.01 0.001	Credited only for non-faulted SG, with PORV failure assumed in faulted SG
Steam Released to Environment: 0 - 2 hrs 2 - 8 hrs 8 - 40 hrs 40 - 64 hrs	 lbm lbm lbm lbm	 719,000 1,109,000 2,664,000 0	Per RG 1.183, release duration is until cold shutdown is established and releases from the steam generators have been terminated.
RHR Cut-in Time	hours	40	Time for termination of release due to PORV steaming.
Chemical form of radioiodine released from the fuel: Cesium iodide Elemental iodide Organic iodide	 % % %	 95 4.85 0.15	
Iodine Releases from steam generator to environment: Elemental Organic	 % %	 97 3	

15.4 REACTIVITY AND POWER DISTRIBUTION ANOMALIES

A number of faults have been postulated which could result in reactivity and power distribution anomalies. Reactivity changes could be caused by control rod motion or ejection, boron concentration changes, or addition of cold water to the reactor coolant system (RCS). Power distribution changes could be caused by control rod motion, misalignment, or ejection, or by static means such as fuel assembly mislocation. These events are discussed in this section. Detailed analyses are presented for the most limiting of these events.

Discussion of the following incidents is presented in Section 15.4:

a. Uncontrolled rod cluster control assembly bank withdrawal from a subcritical or low power startup condition,

b. Uncontrolled rod cluster control assembly bank withdrawal at power,

c. Rod cluster control assembly misalignment,

d. Startup of an inactive reactor coolant pump at an incorrect temperature (detailed analysis is deleted),

e. Chemical and volume control system malfunction that results in a decrease in boron concentration in the reactor coolant,

f. Inadvertent loading and operation of a fuel assembly in an improper position, and

g. Spectrum of rod cluster control assembly ejection accidents.

Items a, b, c, d and e are considered to be ANS Condition II events, Item f and ANS Condition III event, and Item g of ANS Condition IV event. Item c entails both Condition II and III events. Subsection 15.0.1 contains a discussion of ANS classifications.

15.4.1 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal From a Subcritical or Low Power Startup Condition

15.4.1.1 Identification of Causes and Accident Description

A rod cluster control assembly (RCCA) withdrawal accident is defined as an uncontrolled addition of reactivity to the reactor core caused by withdrawal of RCCAs resulting in a power excursion. Such a transient could be caused by operator action or by a malfunction of the reactor control or rod control systems.

This could occur with the reactor either subcritical, hot zero power or at power. The "at power" case is discussed in Subsection 15.4.2.

Although the reactor is normally brought to power from a subcritical condition by means of RCCA withdrawal, initial startup procedures with a clean core call for boron dilution. The maximum rate of reactivity increase in the case of boron dilution is less than that assumed in this analysis (see Subsection 15.4.6).

The RCCA drive mechanisms are wired into preselected bank configurations which are not altered during reactor life. These circuits prevent the RCCAs from being automatically withdrawn in other than their respective banks. Power supplied to the banks is controlled such that no more than two banks can be withdrawn at the same time and in their proper withdrawal sequence. The RCCA drive mechanisms are of the magnetic latch type and coil actuation is sequenced to provide variable speed travel. The maximum reactivity insertion rate analyzed in the detailed plant analysis is that occurring with the simultaneous withdrawal of the combination of two sequential control banks having the maximum combined worth at maximum speed.

This event is classified as an ANS Condition II incident (an incident of moderate frequency) as defined in Subsection 15.0.1.

The neutron flux response to a continuous reactivity insertion is characterized by a very fast rise terminated by the reactivity feedback effect of the negative Doppler coefficient. This self limitation of the power excursion is of primary importance since it limits the power to a tolerable level during the delay time for protective action. Should a continuous RCCA withdrawal accident occur, the transient will be terminated by the following automatic features of the reactor protection system:

a. <u>Source Range High Neutron Flux Reactor Trip</u> - actuated when either of two independent source range channels indicates a neutron flux level above a preselected manually adjustable setpoint. This trip function may be manually bypassed only after an intermediate range flux channel indicates a flux level above a specified level. It is automatically reinstated when both intermediate range channels indicate a flux level below a specified level.

b. <u>Intermediate Range High Neutron Flux Reactor Trip</u> - actuated when either of two independent intermediate range channels indicates a flux level above a preselected manually adjustable setpoint. This trip function may be manually bypassed only after two of the four power range channels are reading above approximately 10% of full power and is automatically reinstated when three of the four channels indicate a power level below this value.

c. Power Range High Neutron Flux Reactor Trip (Low Setting) - actuated when two out of the four power range channels indicate a power level above approximately 25% of full power. This trip function may be manually bypassed when two of the four power range channels indicate a power level above approximately 10% of full power and is automatically reinstated only after three of the four channels indicate a power level below this value.

d. Power Range High-Neutron Flux Reactor Trip (High Setting) - actuates when two out of the four power range channels indicate a power level above a preset setpoint. This trip function is always active.

e. High Nuclear Flux Rate Reactor Trip - actuated when the positive rate of change of neutron flux on two out of four nuclear power range channels indicate a rate above the preset setpoint. This trip function is always active.

In addition, control rod stops on high intermediate range flux level (one of two) and high power range flux level (one out of four) serve to discontinue rod withdrawal and prevent the need to actuate the intermediate range flux level trip and the power range flux level trip, respectively.

15.4.1.2 Analysis of Effects and Consequences

Method of Analysis

The analysis of the uncontrolled RCCA bank withdrawal from subcritical accident is performed in three stages: first, an average core nuclear power transient calculation, then an average core heat transfer calculation, and finally the DNBR calculation. The average core nuclear calculation is performed using spatial neutron kinetics methods TWINKLE (Reference 1) to determine the average power generation with time including the various total core feedback effects, i.e., Doppler reactivity and moderator reactivity. The average heat flux and temperature transients are determined by performing a fuel rod transient heat transfer calculation in FACTRAN (Reference 2). The average heat flux is next used in VIPRE (described in Section 4.4) for transient DNBR calculation.

Plant characteristics and initial conditions are discussed in Subsection 15.0.3. In order to give conservative results for a startup accident, the following assumptions are made:

a. Since the magnitude of the power peak reached during the initial part of the transient for any given rate of reactivity insertion is strongly dependent on the Doppler coefficient, conservatively low values as a function of power are used. The Doppler reactivity defect is determined as a function of power level using a one-dimensional steady-state computer code with a Doppler weighting factor of 1.0. The Doppler coefficient used does not directly correlate with Figure 15.0-3 because the TWINKLE code, on which the neutronic analysis is based, is a diffusion-theory code, rather than a point-kinetics approximation. The Doppler defect used as an initial condition is 965 pcm.

b. Contribution of the moderator reactivity coefficient is negligible during the initial part of the transient because the heat transfer time between the fuel and the moderator is much longer than the neutron flux response time. However, after the initial neutron flux peak, the succeeding rate of power increase is affected by the moderator reactivity coefficient. A highly conservative value is used in the analysis to yield the maximum peak heat flux.

c. The reactor is assumed to be just critical at hot zero power (no load) T_{avg}. This assumption is more conservative than that of a lower initial system temperature. The higher initial system temperature yields a large fuel-water heat transfer coefficient, larger specific heats, and a less negative (smaller absolute magnitude) Doppler coefficient, all of which tend to reduce the Doppler feedback effect thereby increasing the neutron flux peak. The initial effective multiplication factor is assumed to be 1.0 since this results in the worst nuclear power transient.

d. Reactor trip is assumed to be initiated by power range high neutron flux (low setting). The most adverse combination of instrument and setpoint errors, as well as delays for trip signal actuation and rod cluster control assembly release, is taken into account. A 10% increase is assumed for the power range flux trip setpoint raising it from the nominal value of 25% to 35%. Since the rise in the neutron flux is so rapid, the effect of errors in the trip setpoint on the actual time at which the rods are released is negligible. In addition, the reactor trip insertion characteristic is based on the assumption that the highest worth rod cluster control assembly is stuck in its fully withdrawn position. See Subsection 15.0.5 for rod cluster control assembly insertion characteristics.

e. The maximum positive reactivity insertion rate assumed is greater than that for the simultaneous withdrawal of the combination of the two sequential control banks having the greatest combined worth at maximum speed (45 inches/minute). Control rod drive mechanism design is discussed in Section 4.6.

f. The most limiting axial and radial power shapes associated with having the two highest combined worth banks in their high worth position are assumed in the DNB analysis.

g. The initial power level was assumed to be below the power level expected for any shutdown condition (10^{-9} of nominal power). This combination of highest reactivity insertion rate and lowest initial power produces the highest peak heat flux.

h. Two reactor coolant pumps are assumed to be in operation. This is conservative with respect to DNB.

Plant systems and equipment credited for mitigating the effects of the accident are discussed in Subsection 15.0.8 and listed in Table 15.0-7. No single active failure in any of these systems or equipment will adversely affect the consequences of the accident.

Results

Figures 15.4-1 through 15.4-3 show the transient behavior for the uncontrolled RCCA bank withdrawal incident, with the accident terminated by reactor trip at 35% of nominal power. The reactivity insertion rate used is greater than that calculated for the two highest worth sequential control banks, both assumed to be in their highest incremental worth region. Figure 15.4-1 shows the neutron flux transient.

The energy release and the fuel temperature increases are relatively small. The thermal flux response, of interest for DNB considerations, is shown on Figure 15.4-2. The beneficial effect of the inherent thermal lag in the fuel is evidenced by a peak heat flux much less than the full power nominal value. There is a large margin to DNB during the transient since the rod surface heat flux remains below the design value, and there is a high degree of subcooling at all times in the core. Figure 15.4-3 shows the response of the hot spot fuel average temperature and the hot spot clad temperature. The average fuel temperature increases to a value lower than the nominal full power value. The minimum DNBR at all times remains above the limit value.

The calculated sequence of events for this accident is shown on Table 15.4-1. With the reactor tripped, the plant returns to a stable condition. The plant may subsequently be cooled down further by following normal plant shutdown procedures.

The operating procedures would call for operator action to control RCS boron concentration and pressurizer level using the CVCS, and to maintain steam generator level through control of the main or auxiliary feedwater system. Any action required of the operator to maintain the plant in a stabilized condition will be in a time frame in excess of 10 minutes following reactor trip.

15.4.1.3 Radiological Consequences

There are no radiological consequences associated with an uncontrolled rod cluster control assembly bank withdrawal from a subcritical or low power start-up condition event since radioactivity is contained within the fuel rods and reactor coolant system within design limits.

15.4.1.4 Conclusions

In the event of a RCCA withdrawal accident from the subcritical condition, the core and the reactor coolant system are not adversely affected, since the combination of thermal power and the coolant temperature result in a DNBR greater than the limit value. Thus, no fuel or clad damage is predicted as a result of DNB.

15.4.2 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power

15.4.2.1 Identification of Causes and Accident Description

Uncontrolled rod cluster control assembly (RCCA) bank withdrawal at power results in an increase in the core heat flux. Since the heat extraction from the steam generator lags behind the core power generation until the steam generator pressure reaches the relief or safety valve setpoint, there is a net increase in the reactor coolant temperature. Unless terminated by manual or automatic action, the power mismatch and resultant coolant temperature rise could eventually result in DNB. Therefore, in order to avert damage to the fuel clad, the reactor protection system is designed to terminate any such transient before the DNBR falls below the limit value.

This event is classified as an ANS Condition II incident (an incident of moderate frequency) as defined in Subsection 15.0.1.

The automatic features of the reactor protection system which prevent core damage following the postulated accident include the following:

a. Power range neutron flux instrumentation actuates a reactor trip if two-of-four channels exceed an overpower setpoint.

b. Reactor trip is actuated if any two-out-of-four channels exceed a positive neutron flux rate setpoint.

c. Reactor trip is actuated if any two-out-of-four ΔT channels exceed an overtemperature ΔT setpoint. This setpoint is automatically varied with axial power imbalance, coolant temperature and pressure to protect against DNB.

d. Reactor trip is actuated if any two-out-of-four ΔT channels exceed an overpower ΔT setpoint. This setpont is automatically varied with coolant average temperature so that the allowable heat generation rate (kw/ft) is not exceeded.

e. A high pressurizer pressure reactor trip actuated from any two-out-of four pressure channels which is set at a fixed point.

f. A high pressurizer water level reactor trip actuated from any two-out-of-three level channels when the reactor power is above approximately 10% (Permissive-7).

In addition to the above listed reactor trips, there are the following RCCA withdrawal blocks:

a. high neutron flux (one-out-of-four power range),

b. Overpower ΔT (two-out-of-four), and

c. Overtemperature ΔT (two-out-of-four).

The manner in which the combination of overpower and overtemperature ΔT trips provide protection over the full range of reactor coolant system conditions is described in Chapter 7.0. Figure 15.0-1 presents allowable reactor coolant loop average temperature and ΔT for the design power distribution and flow as a function of primary coolant pressure. The boundaries of operation defined by the overpower ΔT trip and the over temperature ΔT trip are represented as "protection lines" on this diagram. The protection lines are drawn to include all adverse instrumentation and setpoint errors so that under nominal conditions trip would occur well within the area bounded by these lines. The utility of this diagram is in the fact that the limit imposed by a given DNBR can be represented as a line. The DNB lines represent the locus of conditions for which the DNBR equals the limit value. All points below and to the left of a DNB line for a given pressure have a DNBR greater than the limit value. The diagram shows that DNB is prevented for all cases if the area enclosed with the maximum protection lines is not traversed by the applicable DNBR line at any point.

The area of permissible operation (power, pressure and temperature) is bounded by the combination of reactor trips: high neutron flux (fixed setpoint); high pressure (fixed setpoint); low pressure (fixed setpoint); overpower and overtemperature ΔT (variable setpoints).

15.4.2.2 Analysis of Effects and Consequences

Method of Analysis

The transient is analyzed by the LOFTRAN Code (Reference 3). This code simulates the neutron kinetics, reactor coolant system, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables including temperature, pressures, and power level. The core limits as illustrated in Figure 15.0-1 are used as input to LOFTRAN to determine the minimum DNBR during the transient.

Bryon/Braidwood Unit 1 have the BWI steam generators while Byron/Braidwood Unit 2 have Westinghouse D5 steam generators. For this reason, the limiting cases are analyzed separately using input for each steam generator design to insure limiting results are calculated.

This accident is analyzed with the Revised Thermal Design Procedure as described in WCAP-11397-P-A for minimum DNBR. Plant characteristics and initial conditions are discussed in Subsection 15.0.3.

a. Initial reactor power, pressure, and RCS temperatures are assumed to be at their nominal values. With the exception of the RCS average temperature bias, which is explicitly modeled in the analysis, uncertainties in initial conditions are included in the limit DNBR as described in WCAP-11397-P-A.

b. Reactivity Coefficients - Two cases are analyzed:

    1. Minimum Reactivity Feedback. The most positive moderator coefficient of reactivity allowed by the Technical Specifications is assumed corresponding to the beginning of core life. A variable Doppler power coefficient with core power is used in the analysis. A conservatively small (in absolute magnitude) value is assumed.

    2. Maximum Reactivity Feedback. A conservatively large positive moderator density coefficient and a large (in absolute magnitude) negative Doppler power coefficient are assumed.

c. The reactor trip on high neutron flux is assumed to be actuated at a conservative value of 118% of nominal full power. The ΔT trips include all adverse instrumentation and setpoint errors; the delays for trip actuation are assumed to be the maximum values.

d. The RCCA trip insertion characteristic is based on the assumption that the highest worth assembly is stuck in its fully withdrawn position.

e. The maximum positive reactivity insertion rate is greater than that for the simultaneous withdrawal of the combinations of the two control banks have the maximum combined worth at maximum speed.

f. The analysis assumes all main steam safety valves are operable. If one or more main steam safety valves become inoperable, Technical Specification 3.7.1 requires a reduction in power as well as a reduction in the power range neutron flux - high reactor trip setpoint. The required reductions are based on a heat balance equation as described in Reference 12. A sensitivity study was performed to demonstrate that the reductions are adequate.

The effect of RCCA movement on the axial core power distribution is accounted for by causing a decrease in overtemperature ΔT trip setpoint proportional to a decrease in margin to DNB.

The peak RCS pressure case for this event is analyzed using the standard thermal design procedure. Initial reactor power, pressure, and RCS temperatures are assumed to be at their nominal values with uncertainties applied in the conservative direction. No credit is taken for the effect of pressurizer spray and power operated relief valves in reducing or limiting the coolant pressure. Pressurizer safety valves are operable. The Technical Specifications allow a ±2% tolerance around the nominal setting of 2460 psig for the pressurizer safety valves, with the lowest allowed lifting pressure being 2411 psig. The assumed high pressurizer pressure reactor trip setpoint in the peak pressure cases, including instrument uncertainty, is higher than the lowest setting for the pressurizer safety valves. As a result, the pressurizer safety valves could lift prior to reaching the high pressurizer pressure reactor trip setpoint and the reactor trip might be delayed.

The pressurizer safety valves are assumed to lift at the highest setpoint in the peak pressure cases to maximize RCS and secondary side pressure. A sensitivity study is performed assuming the lowest pressurizer safety valve lift setpoint to demonstrate that, with a delayed reactor trip, all applicable acceptance criteria are met.

Plant systems and equipment credited for mitigating the effects of the accident are discussed in Subsection 15.0.8 and listed in Table 15.0-7. No single active failure in any of these systems or equipment will adversely offset the consequences of the accident.

Results

Figures 15.4-4 and 15.4-5 show the transient response for a rapid RCCA withdrawal incident starting from full power. Reactor trip on high neutron flux occurs shortly after the start of the accident. Since this is rapid with respect to the thermal time constants of the plant, small changes in T_{avg} and pressure result and margin to the DNBR limit is maintained.

The transient response for a slow RCCA withdrawal from full power is shown in Figures 15.4-6 and 15.4-7. Reactor trip on overtemperature ΔT occurs after a longer period and the rise in temperature and pressure is consequently larger than for rapid RCCA withdrawal. Again, the minimum DNBR is greater than the limit value.

Figure 15.4-8 shows the minimum DNBR as a function of reactivity insertion rate from initial full power operation for minimum and maximum reactivity feedback. It can be seen that two reactor trip channels provide protection over the whole range of reactivity insertion rates. These are the high neutron flux and overtemperature ΔT channels. The minimum DNBR is never less than the limit value.

Figures 15.4-9 and 15.4-10 show the minimum DNBR as a function of reactivity insertion rate for RCCA withdrawal incidents starting at 60% and 10% power, respectively. In neither case does the DNBR fall below the limit value.

The shape of the curves of minimum DNB ratio versus reactivity insertion rate in the reference figures is due both to reactor core and coolant system transient response and to protection system action in initiating a reactor trip.

Referring to Figure 15.4-10, for example, it is noted that:

a. For high reactivity insertion rates (i.e., between ~ 2.0×10^{-4} δK/sec and 11.0×10^{-4} δK/sec) reactor trip is initiated by the high neutron flux trip for the minimum reactivity feedback cases. For the higher insertion rates in this range, the neutron flux level in the core rises rapidly while core heat flux and coolant system fluid lag behind due to the thermal capacity of the fuel and coolant system fluid. Thus, the reactor is tripped prior to any significant increase in heat flux or coolant temperature resulting in higher minimum DNBRs for these reactivity insertion rates and more margin to the applicable safety analysis limit.

As reactivity insertion rate decreases, core heat flux and coolant temperatures can remain more nearly in equilibrium with the neutron flux; minimum DNB ratio during the transient thus decreases with decreasing insertion rate.

b. The overtemperature ΔT reactor trip circuit initiates a reactor trip when measured coolant loop ΔT exceeds a setpoint based on measured reactor coolant system average temperature and pressure. This trip circuit is described in detail in Chapter 7.0; however, it is important in this context to note that the average temperature contribution to the circuit is lead-lag compensated in order to account for the effect of the thermal capacity of the reactor coolant system in response to power increases.

c. With further decrease in reactivity insertion rate, the overtemperature ΔT and high neutron flux trips become equally effective in terminating the transient (e.g., at ~ 2.0×10^{-4} δK/sec reactivity insertion rate).

For reactivity insertion rates between ~ 2.0×10^{-4} δK/sec and ~ 5.0×10^{-5} δK/sec the effectiveness of the overtemperature ΔT trip increases (in terms of increased minimum DNB ratio) due to the fact that with lower insertion rates the power increase rate is slower, the rate of rise of average coolant temperature is slower and the system lags and delays become less significant.

For reactivity insertion rates less than ~ 5.0×10^{-5} δK/sec, the rise in the reactor coolant temperature is sufficiently high so that the setpoint for the first bank of steam generator safety valves is reached before the reactor trip setpoint. This delays reactor trip due to dynamic compensation associated with the overtemperature ΔT trip resulting in lower minimum DNB ratios.

Figures 15.4-8, 15.4-9 and 15.4-10 illustrate minimum DNBR calculated for minimum and maximum reactivity feedback for the BWI steam generators which bound the D5 steam generators.

Since the RCCA withdrawal at power incident is an overpower transient, the fuel temperatures rise during the transient until after reactor trip occurs. For high reactivity insertion rates, the overpower transient is fast with respect to the fuel rod thermal time constant, and the core heat flux lags behind the neutron flux response. Due to this lag, the peak core heat flux does not exceed 118% of its nominal value (i.e., the high neutron flux trip setpoint assumed in the analysis). Taking into account the effect of the RCCA withdrawal on the axial core power distribution, the peak fuel temperature will still remain below the fuel melting temperature.

For slow reactivity insertion rates, the core heat flux remains more nearly in equilibrium with the neutron flux. The overpower transient is terminated by the overtemperature ΔT reactor trip before a DNB condition is reached. The peak heat flux again is maintained below 118% of its nominal value. Taking into account the effect of the RCCA withdrawal on the axial core power distribution, the peak clad centerline temperature will remain below the fuel melting temperature.

Since DNB does not occur at any time during the RCCA withdrawal at power transient, the ability of the primary coolant to remove heat from the fuel rod is not reduced. Thus, the fuel cladding temperature does not rise significantly above its initial value during the transient.

The calculated sequence of events for this accident is shown on Table 15.4-1. With the reactor tripped, the plant eventually returns to a stable condition. The plant may subsequently be cooled down further by following normal plant shutdown procedures.

The peak RCS pressure for this event is analyzed using the same range of reactivity insertion rates as the DNB cases discussed above. A limit is placed on the maximum reactivity insertion rate during an RWAP event to demonstrate compliance with the TS RCS pressure SL. Subsequently, the plant-specific core design must ensure this maximum reactivity insertion rate will not be exceeded as part of the reload evaluation process. This ensures the RWAP RCS overpressure analysis remains valid on a cycle specific basis.

15.4.2.3 Radiological Consequences

There are only minimal radiological consequences associated with an uncontrolled rod cluster control assembly bank withdrawal at power event. The reactor trip causes a turbine trip, and heat is removed from the secondary system through the steam generator power relief valves or safety valves. The radiological consequences associated with atmospheric steam release from this event are less severe than the steamline break accident analyzed in Subsection 15.1.5.

15.4.2.4 Conclusions

The high neutron flux and overtemperature ΔT trip channels provide adequate protection over the entire range of possible reactivity insertion rates, i.e., the minimum value of DNBR is always larger than the limit value. The radiological consequences would be less severe than the steamline break accident analyzed in Subsection 15.1.5.

15.4.3 Rod Cluster Control Assembly Misoperation (System Malfunction or Operator Error)

15.4.3.1 Identification of Causes and Accident Description

Rod cluster control assembly (RCCA) misoperation accidents include:

a. One or more dropped RCCAs within the same group,

b. A dropped RCCA bank,

c. Statically misaligned RCCA,

d. Withdrawal of a single RCCA.

Each RCCA has a position indicator channel which displays position of the assembly. The displays of assembly positions are grouped for the operator's convenience. Fully inserted assemblies are further indicated by a rod at bottom signal, which actuates a local alarm and a control room annunciator. Group demand position is also indicated.

Full length RCCAs are always moved in preselected banks, and the banks are always moved in the same preselected sequence. Each bank of RCCAs is divided into two groups. The rods comprising a group operate in parallel through multiplexing thyristors. The two groups in a bank move sequentially such that the first group is always within one step of the second group in the bank. A definite schedule of actuation (or deactuation of the stationary gripper, movable gripper, and lift coils of a mechanism) is required to withdraw the RCCA that is attached to the mechanism. Since the stationary gripper, movable gripper, and lift coils associated with the four RCCAs of a rod group are driven in parallel, any single failure which would cause rod withdrawal would affect a minimum of one group. Mechanical failures are in the direction of insertion, or immobility.

The dropped RCCAs, dropped RCCA bank, and statically misaligned RCCA events are classified as ANS Condition II incidents (incidents of moderate frequency) as defined in Subsection 15.0.1. The single RCCA withdrawal incident is classified as an ANS Condition III event, as discussed below.

No single electrical or mechanical failure in the rod control system could cause the accidental withdrawal of a single rod cluster control assembly (RCCA) from the inserted bank at full power operation. The operator could deliberately withdraw a single RCCA in a control bank since this feature is necessary in order to retrieve an assembly should one be accidentally dropped. The event analyzed must result from multiple wiring failures (probability for single random failure is on the order of 10^{-4}/year refer to Subsection 7.7.2.2) or multiple deliberate operator actions and subsequent and repeated operator disregard of event indication. The probability of such a combination of conditions is considered so low that the limiting consequences may include slight fuel damage.

Thus, consistent with the philosophy and format of ANSI N18.2, the event is classified as a Condition III event. By definition "Condition III occurrences include incidents, any one of which may occur during the lifetime of a particular plant," and "shall not cause more than a small fraction of fuel elements in the reactor to be damaged..."

This selection of criterion is not in violation of GDC 25 which states, "The protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods." (Emphases have been added.) It has been shown that single failures resulting in RCCA bank withdrawals do not violate specified fuel design limits. Moreover, no single malfunction can result in the withdrawal of a single RCCA. Thus, it is concluded that the criterion established for the single rod withdrawal at power is appropriate and in accordance with GDC 25.

A dropped RCCA or RCCA bank is detected by:

a. Sudden drop in the core power level as seen by the nuclear instrumentation system;

b. Asymmetric power distribution as seen on out-of-core neutron detectors or core exit thermocouples;

c. Rod at bottom signal;

d. Rod deviation alarm; and

e. Rod position indication.

Misaligned assemblies are detected by:

a. Asymmetric power distribution as seen on out-of-core neutron detectors or core exit thermocouples;

b. Rod deviation alarm; and

c. Rod position indicators.

The resolution of the rod position indicator channel is ± 6 steps at full accuracy and ± 12 steps at half accuracy. Deviation of any assembly from its group by twice this distance (24 steps) will not cause power distributions worse than the design limits. The deviation alarm alerts the operator to rod deviation with respect to the group position in excess of 12 steps. If the rod deviation alarm is not operable, procedures require the operator to take appropriate actions.

If one or more rod position indicator channels should be out of service, detailed operating instructions shall be followed to assure the alignment of the non-indicated RCCAs. The operator is also required to take action as required by the Technical Specifications.

In the extremely unlikely event of simultaneous electrical failures which could result in single RCCA withdrawal, rod deviation and rod control urgent failure would both be displayed on the plant annunciator, and the rod position indicators would indicate the relative positions of the RCCAs in the bank. The urgent failure alarm also inhibits automatic rod motion in the group in which it occurs. Withdrawal of a single RCCA by operator action, whether deliberate or by a combination of errors, would result in activation of the same alarm and the same visual indications. Withdrawal of a single RCCA results in both positive reactivity insertion tending to increase core power, and an increase in local power density in the core area associated with the RCCA. Automatic protection for this event is provided by the overtemperature ΔT reactor trip, although due to the increase in local power density it is not possible in all cases to provide assurance that the core safety limits will not be violated.

Plant systems and equipment credited for mitigating the effects of the various control rod misoperations are discussed in Subsection 15.0.8 and listed in Table 15.0-7. No single active failure in any of these systems or equipment will adversely affect the consequences of the accident.

15.4.3.2 <u>Analysis of Effects and Consequences</u>

a. Dropped RCCAs, dropped RCCA bank, and statically misaligned RCCA.

<u>Method of Analysis</u>

1. One or more dropped RCCAs from the same group.

For evaluation of the dropped RCCA event, the transient system response is calculated using the LOFTRAN code. The code simulates the neutron kinetics, reactor coolant system, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level. Statepoints are calculated and nuclear models are used to obtain a hot channel factor consistent with the primary system conditions and reactor power. By incorporating the primary conditions from the transient and the hot channel factor from the nuclear analysis, the

DNB design basis is shown to be met using the VIPRE code. The transient response, nuclear peaking factor analysis, and DNB design basis confirmation are performed in accordance with the methodology described in Reference 9.

2. Dropped RCCA Bank

   A LOFTRAN calculation is not necessary for the dropped RCCA event. Westinghouse WCAP 11394-P-A concludes that sufficient DNB margin exists, subject to plant/cycle specific analysis.

3. Statically Misaligned RCCA.

   Steady state power distributions are analyzed using the computer codes as described in Table 4.1-2. The peaking factors are then compared to the peaking factor limit determined by the VIPRE code at the safety analysis DNBR limit.

Results

1. One or more dropped RCCAs.

   Single or multiple dropped RCCAs within the same group result in a negative reactivity insertion. When detected, the reactor is manually tripped following the drop of the RCCAs. The core is not adversely affected during this period, since power is decreasing rapidly. Following manual reactor trip, normal shutdown procedures are followed. RCCA may be manually retrieved by following approved operating procedures.

   For those dropped RCCAs which do not result in a manual reactor trip, power may be reestablished either by reactivity feedback or control bank withdrawal. Following a dropped rod event in manual rod control, the plant will establish a new equilibrium condition. The equilibrium process without control system interaction is monotonic, thus removing power overshoot as a concern and establishing the automatic rod control mode of operation as the limiting case.

   For a dropped RCCA event in the automatic rod control mode, the rod control system detects the drop in power and initiates control bank withdrawal. Power overshoot may occur due to this action by the automatic rod controller after which the control system will insert the control bank to restore nominal power. Figure 15.4-12a shows a typical transient response to a dropped RCCA (or RCCAs) in automatic control. Uncertainties in the initial condition are included in the DNB evaluation as described in Reference 11. In all cases, the minimum DNBR remains above the limit value.

2. Dropped RCCA Bank.

   A dropped RCCA bank typically results in a reactivity insertion greater than 500 pcm. The reactor is manually tripped following the drop of a RCCA bank. The core is not adversely affected during this period, since power is decreasing rapidly. Following reactor trip, normal shutdown procedures are followed to further cool down the plant. Any action required of the operator to maintain the plant in a stabilized condition will be in a time frame in excess of ten minutes following the incident.

3. Statically Misaligned RCCA

   The most severe misalignment situations with respect to DNBR at significant power levels arise from cases in which one RCAA is fully inserted, or where bank D is fully inserted with one RCCA fully withdrawn. Multiple independent alarms, including a bank insertion limit alarm, alert the operator well before the postulated conditions are approached. The bank can be inserted to its insertion limit with any one assembly fully withdrawn without the DNBR falling below the limit value.

   The insertion limits in the Technical Specifications may vary from time to time depending on a number of limiting criteria. It is preferable, therefore, to analyze the misaligned RCCA case at full power for a position of the control bank as deeply inserted as the criteria on minimum DNBR and power peaking factor will allow. The full power insertion limits on control bank D must then be chosen to be above that position and will usually be dictated by other criteria. Detailed results will vary from cycle to cycle depending on fuel arrangements.

   For the RCCA misalignment in which control bank D is inserted to its full power insertion limit and one RCCA is fully withdrawn, the DNBR does not fall below the limit value. This case was analyzed assuming the initial reactor power, pressure, and RCS temperatures were at their nominal values, but with the increased radial peaking factor associated with the misaligned RCCA. Uncertainties in initial conditions were included as described in WCAP-11397-P-A (Reference 11).

DNB calculations have not been performed specifically for assemblies missing from other banks; however, power shape calculations have been done as required for the RCCA ejection analysis. Inspection of the power shapes shows that the DNB and peak kW/ft situation is less severe than the bank D case discussed above assuming insertion limits on the other banks equivalent to a bank D full-in insertion limit.

For the RCCA misalignments with one RCCA fully inserted, the DNBR does not fall below the limit value. This case was analyzed assuming the initial reactor power, pressure, and RCS temperatures are at their nominal values, but with the increased radial peaking factor associated with the misaligned RCCA. Uncertainties in the initial conditions are included as described in WCAP-11397-P-A.

DNB does not occur for the RCCA misalignment incident and thus the ability of the primary coolant to remove heat from the fuel rod is not reduced. The peak fuel temperature corresponds to a linear heat generation rate based on the radial peaking factor penalty associated with the misaligned RCCA and the design axial power distribution. The resulting linear heat generation is well below that which would cause fuel melting.

Following the identification of a RCCA group misalignment condition by the operator, the operator is required to take action as required by the plant Technical Specifications and operating instructions.

b. Single RCCA Withdrawal

Method of Analysis

Power distributions within the core are calculated using the computer codes as described in Table 4.1-2. The peaking factors are then used by VIPRE to calculate the minimum DNBR for the event. The case of the worst rod withdrawn from control bank D inserted at the insertion limit, with the reactor initially at full power, was analyzed. This incident is assumed to occur at beginning-of-life since this results in the minimum value of moderator temperature coefficient. This assumption maximizes the power rise and minimizes the tendency of increased moderator temperature to flatten the power distribution.

Results

For the single rod withdrawal event, two cases have been considered as follows:

1. If the reactor is in the manual control mode, continuous withdrawal of a single RCCA results in both an increase in core power and coolant temperature, and an increase in the local hot channel factor in the area of the withdrawing RCCA. In terms of the overall system response, this case is similar to those presented in Subsection 15.4.2; however, the increased local power peaking in the area of the withdrawn RCCA results in lower minimum DNBRs than for the withdrawn bank cases. Depending on initial bank insertion and location of the withdrawn RCCA, automatic reactor trip may not occur sufficiently fast to prevent the minimum core DNBR from falling below the limit value. Evaluation of this case at the power and coolant conditions at which the overtemperature ΔT trip would be expected to trip the plant shows that an upper limit for the number of rods with a DNBR less than the limit value is 5%.

2. If the reactor is in the automatic control mode, the multiple failures that result in the withdrawal of a single RCCA will result in the immobility of the other RCCAs in the controlling bank. The transient will then proceed in the same manner as Case 1 described above.

For such cases as above, a reactor trip will ultimately ensue, although not sufficiently fast in all cases to prevent a minimum DNBR in the core of less than the limit value. Following reactor trip, normal shutdown procedures are followed.

15.4.3.3 Radiological Consequences

The most limiting rod cluster control assembly misoperation, accidental withdrawal of a single RCCA, is predicted to result in limited fuel damage (< 5% of the total). The subsequent reactor and turbine trip would result in atmospheric steam release, assuming the condenser was not available for use. The radiological consequences from this event would be no greater than the locked rotor event, analyzed in Subsection 15.3.3.

15.4.3.4 Conclusions

For cases of dropped RCCAs or dropped banks, for which the reactor is manually tripped, there is no reduction in the margin to core thermal limits, and consequently the DNB design basis is met. It is shown for all cases which do not result in reactor trip that the DNBR remains greater than the limit value and, therefore, the DNB design is met.

For all cases of any RCCA fully inserted, or control bank D inserted to its rod insertion limits with any single RCCA in that bank fully withdrawn (static misalignment), the DNBR remains greater than the limit value.

For the case of the accidental withdrawal of a single RCCA, with the reactor in the automatic or manual control mode and initially operating at full power with control bank D at the insertion limit, an upper bound of the number of fuel rods experiencing DNB is 5 percent of the total fuel rods in the core.

15.4.4 Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature

The Technical Specifications require that all four reactor coolant pumps be operating for reactor power operation; therefore, operation with an inactive loop is precluded. This event was originally included in the FSAR licensing basis when operation with a loop out of service was considered. Based on the Technical Specifications, which prohibit at-power operation with an inactive loop, and changes to the Technical Specifications that deleted all references to three-loop operation, this event has been deleted from the UFSAR.

Pages 15.4-20 and 15.4-21 have been deleted intentionally.

15.4.5 A Malfunction or Failure of the Flow Controller in a BWR Loop That Results in an Increased Reactor Coolant Flow Rate

(Not applicable in PWRs)

15.4.6 Chemical and Volume Control System Malfunction That Results in a Decrease in Boron Concentration in the Reactor Coolant

15.4.6.1 Identification of Causes and Accident Description

The principal means of causing an inadvertent boron dilution are the opening of the primary water makeup control valve and failure of the blend system, either by controller or mechanical failure. The CVCS and RMCS are designed to limit, even under various postulated failure modes, the potential rate of dilution to values which, with indication by alarms and instrumentation, will allow sufficient time for operator response, depending on the mode of operation, to terminate the dilution. An inadvertent dilution from the RMCS may be terminated by closing the primary water makeup control valve, CV-111A. All expected sources of dilution may be terminated by closing isolation valves in the CVCS, LCV-112B and C. The lost shutdown margin (SDM) may be regained by the opening of isolation valves to the RWST, LCV-112D and E, thus allowing the addition of 2300 ppm borated water to the RCS.

It is assumed that the addition rate of unborated water to the RCS is limited to 205 gpm by the capacity of the primary water makeup pumps and two charging pumps for Modes 1 and 2. The addition rate of unborated water to the RCS is assumed to be limited to 168 gpm by the high charging flow alarm in Modes 3, 4, and 5. Flows higher than 168 gpm are not considered in these modes of operation since the VCT will be filled faster and the "boron dilution alert" alarms will be initiated earlier. Unborated water sources are isolated from the RCS if conditions prescribed by Technical Specifications 3.3.9 cannot be met.

Generally, to dilute, the operator must perform two distinct actions:

a. Switch control of the RMCS from the automatic makeup mode to the dilute mode and

b. Take RMCS control switch to start.

Failure to carry out either of the above actions prevents initiation of dilution. Also during normal operation the operator may add borated water to the RCS by blending boric acid from the boric acid storage tanks with primary grade water. This requires the operator to determine the concentration of the addition and setting the blended flow rate and the boric acid flow rate. The makeup controller will then limit the sum of the boric acid flow rate and primary grade water flow rate to the blended flow rate, i.e., the controller determines the primary grade water flow rate, after the RMCS control switch is taken to start.

The status of the RCS makeup is continuously available to the operator by:

a. Indication of the boric acid and blended flow rates,

b. CVCS, AB, and PW pump status lights,

c. Deviation alarms if the boric acid or blended flow rates deviate by more than 10 percent from the preset values,

d. Source range neutron flux - when reactor is subcritical;

    1. high flux at shutdown alarm,

    2. indicated source range neutron flux count rates,

    3. audible source range neutron flux count rate, and

    4. source range neutron flux - doubling alarm.

e. With the reactor critical

1. Axial flux difference alarm (reactor power ≥ 50 percent RTP),

2. Control rod insertion limit low and low-low alarms,

3. Overtemperature ΔT alarm (at power),

4. Overtemperature ΔT turbine runback (at power), and

5. Overtemperature ΔT reactor trip.

f. Power Range Neutron Flux - High, both high and low setpoint reactor trips.

g. "Boron Dilution Alert" alarm(s) when reactor is subcritcal

1. VCT high level,

2. Divert valve CV112A is not in VCT position, and

3. Source Range neutron flux-doubling.

This event is classified as an ANS Condition II incident (a fault of moderate frequency) as defined in Subsection 15.0.1.

15.4.6.2 Analysis of Effects and Consequences

To cover all phases of plant operation, boron dilution during refueling, cold shutdown, hot shutdown, hot standby, startup, and power modes of operation are considered in this analysis. Conservative values for necessary parameters were used, i.e., high RCS critical boron concentrations, high boron worths, minimum shutdown margins, and lower than actual RCS volumes. These assumptions result in conservative determinations of the time available for operator or system response after detection of a dilution transient in progress.

Conservative analysis methods are used to analyze a CVCS malfunction that results in a decrease in boron concentration in the reactor coolant. Minimum reactor coolant volumes and maximum dilution flow rates are conservatively assumed for each case analyzed. The result is a logarithmic decrease in coolant boron concentration according to the equation:

$$\frac{d\,C_B}{dt} = -\frac{Q_{in}}{V}\,C_B$$

where

C_B = Boron concentration in the RCS

Q_{in} = Maximum dilution flow rate

V = Active volume in RCS.

This equation can be solved for the time at which the core would become critical or all shutdown margin would be lost. The rate of reactivity insertion due to the dilution can be calculated from the dilution rate and the differential boron worth. The results of this analysis are conservative for all cases analyzed.

A comprehensive review of the primary system has been completed. This review showed that a single failure in the CVCS system in the cold shutdown condition would not result in a boron dilution of the reactor coolant system and that the CVCS malfunction represents the most limiting potential source of dilution. Based on this review, it is clear that the analysis results presented in the UFSAR bound all potential sources of inadvertent dilution under all modes of operation.

Dilution During Refueling

An uncontrolled boron dilution transient cannot occur during this mode of operation. Inadvertent dilution is prevented by administrative controls which isolate the RCS from the potential source of unborated water. CVCS valves, specified in Technical Specification 3.9.2 Bases will be verified closed and secured in position by mechanical stops or by removal of air or electrical power. These valves block all automatic flow paths that could allow unborated makeup water to reach the RCS. Any makeup which is required during refueling will be borated water supplied from the RWST. Administrative controls will strictly control and limit the volume of unborated water which can be manually added to the refueling pool, such as for decontamination activities, in order to prevent diluting the refueling cavity or canal below the required refueling boron concentrations.

Dilution During Cold Shutdown

During this mode, the reactor is shutdown and meets the minimum shutdown margin described by the Core Operating Limits Report (COLR) per Technical Specification 3.1.1. The following conditions are assumed for inadvertent boron dilution while in this operating mode:

a) The minimum ratio of initial to critical boron concentration required to ensure the core does not reach criticality in this mode is 1.065 for Byron Unit 1 and Braidwood Unit 1 and 1.075 for Byron Unit 2 and Braidwood Unit 2. The initial boron concentrations used for these ratios must meet the shutdown margin requirements as specified in the COLR.

b) Dilution flow is limited to a maximum dilution rate of 168 gpm, which is equivalent to the high charging flow alarm plus uncertainties. Any additional dilution flow would increase the VCT level, without an immediate decrease in RCS boron concentration.

c) The minimum RCS water volume of 10583 ft^3 for Byron Unit 1 and Braidwood Unit 1 and 9260 ft^3 for Byron Unit 2 and Braidwood Unit 2 is used. This is a conservative estimate of the active volume of the RCS with one RCP running and all four loop stop valves open.

Dilution During Hot Shutdown

During this mode, the reactor is shutdown and meets the minimum shutdown margin described by the Core Operating Limits Report (COLR) per Technical Specification 3.1.1. The following conditions are assumed for inadvertent boron dilution while in this operating mode:

a) The minimum ratio of initial to critical boron concentration required to ensure the core does not reach criticality in this mode is 1.071 for Byron Unit 1 and Braidwood Unit 1 and 1.082 for Byron Unit 2 and Braidwood Unit 2. The initial boron concentrations used for these ratios must meet the shutdown margin requirements as specified in the COLR.

b) Dilution flow is limited to a maximum dilution rate of 168 gpm, which is equivalent to the high charging flow alarm plus uncertainties. Any additional dilution flow would increase the VCT level, without an immediate decrease in RCS boron concentration.

c) The minimum RCS water volume of 10583 ft^3 for Byron Unit 1 and Braidwood Unit 1 and 9260 ft^3 for Byron Unit 2 and Braidwood Unit 2 is used. This is a conservative estimate of the active volume of the RCS with one RCP running and all four-loop stop valves open.

Dilution During Hot Standby

During the mode, the reactor is shutdown and meets the minimum shutdown margin described by the Core Operating Limits Report (COLR) per Technical Specification 3.1.1. The following conditions are assumed for inadvertent boron dilution while in this operating mode:

a) The minimum ratio of initial to critical boron concentration required to ensure the core does not reach criticality in this mode is 1.088 for Byron Unit 1 and Braidwood Unit 1 and 1.101 for Byron Unit 2 and Braidwood Unit 2. The initial boron concentrations used for these ratios must meet the shutdown margin requirements as specified in the COLR.

b) Dilution flow is limited to a maximum dilution rate of 168 gpm, which is equivalent to the high charging flow alarm plus uncertainties. Any additional dilution flow would increase the VCT level, without an immediate decrease in RCS boron concentration.

c) The minimum RCS water volume of 10583 ft^3 for Byron Unit 1 and Braidwood Unit 1 and 9260 ft^3 for Byron Unit 2 and Braidwood Unit 2 is used. This is a conservative estimate of the active volume of the RCS with one RCP running and all four-loop stop valves open.

Dilution During Startup

Startup is a transitory mode of operation. In this mode the plant is being taken from one long-term mode of operation, hot standby, to another, power operation. The plant is maintained in the startup mode only for the purpose of startup testing at the beginning of each cycle.

During this mode of operation the plant is in manual control, i.e., T_{avg}/rod control is in manual. All normal actions required to change power level, either up or down, require operator initiation. The Technical Specifications and TRM require a SDM as specified in the Core Operating Limits Report and four reactor coolant pumps operating. Other conditions assumed are:

a. Dilution flow is limited by the reactor makeup water system. The maximum anticipated flow rate of unborated primary water to the RCS is 205 gpm.

b. A minimum RCS water volume of 9941.9 ft^3 for Byron Units 1 and 2 and Braidwood Units 1 and 2. This is a conservative estimate of the active RCS volume, minus the pressurizer volume.

c. The C_B for criticality (ARI-1) is very conservatively assumed with a very conservative, constant boron worth.

Dilution During Full Power Operation

The plant may be operated at power two ways, automatic T_{avg}/rod control and under operator control. The TRM requires an available trip reactivity as specified in the Core Operating Limiting Report. Technical Specifications require operation with four reactor coolant pumps. With the plant at power and the RCS at pressure, the dilution rate is limited by the capacity of the reactor makeup water system with two centrifugal charging pumps in operation (analysis is performed assuming two charging pumps are in operation even though normal operation is with one pump). Conditions assumed for this mode are:

a. For manual and automatic reactor control at full power conditions, the dilution flow is limited by the reactor makeup water system (205 gpm).

b. A minimum RCS water volume of 9941.9 ft^3 for Byron Units 1 and 2 and Braidwood Units 1 and 2. This is a conservative estimate of the active RCS volume, minus the pressurizer volume.

c. The C_B for criticality (ARI-1) is very conservatively assumed with a very conservative, constant boron worth.

15.4.6.3 Results and Conclusions

Dilution During Refueling

Dilution during this mode has been precluded through administrative control of valves in the possible dilution flow paths, see Subsection 15.4.6.2.

Dilution During Cold Shutdown

In this mode of operation, unborated water sources will be isolated from the RCS if the conditions prescribed by Technical Specifications 3.3.9 cannot be met. In the event of an inadvertent boron dilution transient while in this mode of operation, the "boron dilution alert" alarms will sound upon detection of VCT high level, and the operator will administratively align CVCS valves to terminate dilution and start boration (note that this alarm could also be initiated by the flux doubling signal or CV112A valve not in VCT position, but these signals are not credited in the analysis). Valves LCV-112D and E (isolation valves to the RWST) are opened to supply 2300 ppm borated water to the suction of the charging pumps and valves LCV-112B and C (isolation valves in the CVCS) are closed to terminate the dilution. These actions are carried out to minimize the approach to criticality and regain lost shutdown margin. The operator has at least 15 minutes to complete these actions from the time of the alarms until shutdown margin is lost.

Dilution During Hot Shutdown and Hot Standby

In this mode of operation, unborated water sources will be isolated from the RCS if the conditions prescribed by Technical Specifications 3.3.9 cannot be met. In the event of an inadvertent boron dilution transient while in this mode of operation, the “boron dilution alert” alarms will sound upon detection of VCT high level, and the operator will administratively align CVCS valves to terminate dilution and start boration (note that this alarm could also be initiated by the flux doubling signal or CV112A valve not in VCT position, but these signals are not credited in the analysis). Valves LCV-112D and E (isolation valves to the RWST) are opened to supply 2300 ppm borated water to the suction of the charging pumps and valves LCV-112B and C (isolation valves in the CVCS) are closed to terminate the dilution. These actions are carried out to minimize the approach to criticality and regain lost shutdown margin. The operator has at least 15 minutes to complete these actions from the time of the alarm(s) until shutdown margin is lost.

Dilution During Startup

This mode of operation is a transitory mode to go to power and is the operational mode in which the operator intentionally dilutes and withdraws control rods to take the plant critical. During this mode the plant is in manual control with the operator required to maintain a very high awareness of the plant status. For a normal approach to criticality the operator must manually initiate a limited dilution and subsequently manually withdraw the control rods, a process that takes several hours. The plant Technical Specifications require that the operator determine the estimated critical position of the control rods prior to approaching criticality thus assuring that the reactor does not go critical with the control rods below the insertion limits. Once critical, the power escalation must be sufficiently slow to allow the operator to manually block the source range reactor trip (nominally at 10^5 cps) after receiving P-6 from the intermediate range. Too fast a power escalation (due to an unknown dilution) would result in reaching P-6 unexpectedly leaving insufficient time to manually block the source range reactor trip. Failure to perform this manual action results in a reactor trip and immediate shutdown of the reactor.

However, in the event of an unplanned approach to criticality or dilution during power escalation while in the startup mode, the plant status is such that minimal impact will result. The plant will slowly escalate in power to a reactor trip on the power range neutron flux - high, low setpoint (nominally 25 percent RTP). After reactor trip, there are more than 15 minutes for operator action prior to return to criticality. The required operator action is to initiate and continue boration until adequate shutdown margin is restored and to terminate the dilution.

Dilution During Full Power Operation

With the reactor in manual control and no operator action taken to terminate the transient, the power and temperature rise will cause the reactor to reach the overtemperature ΔT or Power Range High Neutron Flux trip setpoint resulting in a reactor trip. After reactor trip, there are more than 15 minutes for operator action prior to return to criticality. The required operator action is to initiate and continue boration until adequate shutdown margin is restored and to terminate the dilution. The boron dilution transient in this case is essentially the equivalent to an uncontrolled rod withdrawal at power. The maximum reactivity insertion rate for a boron dilution transient is conservatively estimated to be 2.0 pcm/sec and is within the range of insertion rates analyzed for uncontrolled rod withdrawal at power. It should be noted that prior to reaching the overtemperature ΔT reactor trip, the operator will have received an alarm on overtemperature ΔT turbine runback.

Thus with the reactor in automatic rod control, a boron dilution will result in a power and temperature increase such that the rod controller will attempt to compensate by slow insertion of the control rods. This action by the controller will result in at least one of three alarms to the operator:

a. rod insertion limit - low level alarm,

b. rod insertion limit - low-low level alarm if insertion continued after (1), and

c. axial flux difference alarm (ΔI outside of the target band).

Given the many alarms, indications, and the inherent slow process of dilution at power, the operator has sufficient time for action. For example, the operator has more than 15 minutes from the rod insertion limit low-low alarm until 1.3 percent ΔK/K is inserted at beginning-of-life. The time would be significantly longer at end-of-life, due to the low initial boron concentration, when shutdown margin is a concern.

The above results demonstrate that in all modes of operation an inadvertent boron dilution is precluded, or sufficient time is available for operator action to terminate the transient. Following termination of the dilution flow and initiation of boration the reactor is in a stable condition with the operator regaining the required shutdown margin.

15.4.7 Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position

15.4.7.1 Acceptance Criteria

This event is identified as a Condition III event (Infrequent Fault) as defined in Subsection 15.0.1. The specific acceptance criteria for this event are as follows:

a. To meet the requirements of General Design Criteria 13, plant operating procedures should include a provision requiring that reactor instrumentation be used to search for potential fuel loading errors after fueling operations.

b. In the event the error is not detectable by the instrumentation system and fuel rod failure limits could be exceeded during normal operations, the offsite consequences should be a small fraction of the 10 CFR Part 100 guidelines.

15.4.7.2 Identification of Causes and Accident Description

The Inadvertent Loading Event comprises core misloading scenarios such as the loading of one or more fuel assemblies into improper positions, loading a fuel rod during manufacture with one or more pellets of the wrong enrichment, or the loading of a full fuel assembly during manufacture with pellets of the wrong enrichment. In addition to these scenarios, misloading events involving burnable absorbers are theoretically possible, scenarios such as the placement of a cluster of 20 burnable absorbers into a core location slated to have 24 burnable absorbers. All of these misloading scenarios potentially result in a core reactivity distribution that differs from the intended core reactivity distribution. As a result, the core power distribution and peaking factors may differ from predictions. Specifically, misloading errors can lead to increased local power peaking at the location of the misloading if the misloading results in a local reactivity increase relative to the intended pattern. If the misloading results in a local reactivity decrease, power peaking increases away from the location of the misloading are possible due to unintended power tilts. These kinds of increases, however, are generally distributed over a large core volume and are small relative to those where the local reactivity is increased.

Fuel misloads are prevented by the manufacturing controls employed to build the fuel and the core loading controls used to assemble the core. The manufacturing controls include checks on fuel rod weight to confirm fuel enrichments, pellet stack lengths, pellet types, and the absence of pellet gaps during fuel manufacturing, and bar coding of each fuel rod to confirm its proper placement in the fuel assembly.

To reduce the probability of core loading errors during fuel loading, each fuel assembly and core component is marked with an identification number and loaded in accordance with a core loading diagram. During core loading, the identification numbers are checked before each assembly is moved into the core. Identification numbers read during fuel movement are subsequently recorded on the loading diagram as a further check on proper placement after loading is completed. These procedures make the likelihood of core misloadings very small.

The severity and detectability of fuel misloads are influenced by several factors: the local reactivity perturbation relative to the intended core loading pattern, the core position of the misload, the local environment of the misloaded fuel assembly, and the number of operable incore detector locations and their proximity to the misload location. Should misloadings occur, the incore system of movable flux detectors, which is used to verify power distributions during startup and throughout the operating cycle, is capable of revealing enrichment errors or misloadings which would cause the kind of substantial power distribution perturbation that would be necessary to induce large numbers of fuel rod failures. In addition, thermocouples and excore detectors can provide additional indications of power distribution anomalies. The instrumentation, along with the startup testing performed each cycle, make the detection of severe misloadings highly likely.

15.4.7.3 Evaluation

The incore moveable detector system is used to search for potential fuel misloads at the start of each operating cycle. Following fuel loading and low power physics testing, an initial core power distribution measurement is made. The core power level of this initial flux map is typically between ~30% and ~50% of rated thermal power. This initial power distribution measurement is used to confirm that the measured power distribution is consistent with the predicted power distribution. Observed flux map deviations in excess of the flux map review criteria (See Table 15.4-6) would prompt an investigation of a possible core anomaly. This satisfies the first acceptance criterion given above.

In Reference 13, a larger number of misloads were evaluated for representative core designs employing current fuel types and fuel failures. The simulated misloads, involving one or two fuel assemblies, covered a wide range of local reactivity perturbations and core positions. The resulting hot full power (HFP) $F_{\Delta H}$ peaking factors ranged from benign to severe. Severe misloads with peaking factors that exceed the $F_{\Delta H}$ limit for DNB at normal operation conditions have the potential for fuel failure if they remain undetected. The simulated misloads were assessed with respect to severity and detectability.

The detectability assessments of Reference 13 demonstrated that the incore detector system is very robust with respect to detection of misloads severe enough to fail fuel during normal operation. By examining a large number of moveable detector thimble patterns, the detectability assessments considered the effect of inoperable moveable detector thimbles on misload detectability. Even when the minimum number of operable detector locations allowed per the plant licensing bases was assumed, the incore detector system was capable of reliably detecting misloads severe enough to fail fuel during normal operation.

Fuel misloads involving a single fuel rod or fuel pellet were not evaluated as part of Reference 13. Such misloads, in general, will not be detectable using the incore detector system due to the very small power distribution perturbation. In terms of increased peaking factors and reduced DNBR values, however, the consequences of such misloads will be very small and limited to the affected fuel rod and the immediately adjacent fuel rods.

Detection of fuel misloads is, in part, a function of the number of available incore detector locations. Reference 13 demonstrated that the flux map review criteria of Table 15.4-6 are effective in detecting fuel misloads that could lead to fuel failures during normal operation. To enhance the probability that significant misloads will be detected, tighter review criteria are employed when the number of available detector locations is reduced. These review criteria will be used for startup and subsequent at-power flux maps.

The detectability assessments of Reference 13 confirm that the moveable detector system can reliably detect fuel misloads that could fail fuel during normal operation when the Table 15.4-6 review criteria are employed. Specifically, Reference 13 demonstrated that only a small fraction of 1% of misloads severe enough to fail fuel during normal operation would be undetected at startup using these limited review criteria. Furthermore, it was judged that even these “undetected” misloads would very likely be detected if other attributes of the startup power distribution measurement (e.g., tilts and reaction rate error contours) were considered along with the results of low power physics testing. Given that detection of >99% of misloads severe enough to fail fuel is expected using these review criteria, a radiological consequences analysis is deemed unnecessary. Failures in fresh fuel during startup would have negligible radiological consequences since there is only a small fission product inventory. Following startup, any fuel rod failures would occur gradually and would be detected by coolant activity monitoring. Since the number of potential fuel rod failures due to a core misload would be extremely small and such failures would occur gradually, any coolant activity releases would initially be well within the cleanup capacity of the plant. Any trend in increased coolant activity would warrant further investigation and evaluation. Therefore, the second acceptance criterion for this event would be satisfied since failures would be gradual, detectable, and the operations would be maintained within Technical Specification coolant activity guidance.

15.4.7.4 Conclusions

Fuel misloads are prevented by manufacturing controls and core loading controls. In the unlikely event that a fuel misload should occur, the incore moveable detector system is capable of reliably detecting misloads that could fail fuel at normal operation conditions. Exceeding the review criteria herein would initiate an investigation to identify potential core anomalies. Any failures associated with an undetected fuel misload would be gradual, detectable, and the operations would be maintained within Technical Specification coolant activity guidelines.

15.4.8 Spectrum of Rod Cluster Control Assembly Ejection Accidents

15.4.8.1 Identification of Causes and Accident Description

This accident is defined as the mechanical failure of a control rod mechanism pressure housing resulting in the ejection of a rod cluster control assembly (RCCA) and drive shaft. The consequence of this mechanical failure is a rapid positive reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage.

15.4.8.1.1 Design Precautions and Protection

Certain features are intended to preclude the possibility of rod ejection accident, or to limit the consequences if the accident were to occur. These include a sound, conservative mechanical design of the rod housings, together with a thorough quality control (testing) program during assembly, and a nuclear design which lessens the potential ejection worth of RCCAs, and minimizes the number of assemblies inserted at high power levels.

Mechanical Design

The mechanical design is discussed in Section 4.6. Mechanical design and quality control procedures intended to preclude the possibility of a RCCA drive mechanism housing failure are listed below:

a. Each full length control rod drive mechanism housing is completely assembled and shop tested at 4100 psi.

b. The mechanism housings are individually hydrotested after they are attached to the head adapters in the reactor vessel head, and checked during the hydrostatic test of the completed reactor coolant system.

c. Stress levels in the mechanism are not affected by anticipated system transients at power, or by the thermal movement of the coolant loops. Moments induced by the design-basis earthquake can be accepted within the allowable primary working stress range specified by the ASME Code, Section III, for Class 1 components.

d. The latch mechanism housings are each a single length of forged or cast Type-304 stainless steel. The rod travel housings are each a single length of forged Type-304 stainless steel. This material exhibits excellent notch toughness at all temperatures that will be encountered.

A significant margin of strength in the elastic range together with the large energy absorption capability in the plastic range gives additional assurance that gross failure of the housing will not occur. The joints between the latch mechanism housing and head adapter, and between the latch mechanism housing and rod travel housing, are threaded joints reinforced by canopy type welds. Administrative regulations require periodic inspections of these (and other) welds.

Nuclear Design

Even if a rupture of a RCCA drive mechanism housing is postulated, the operation of a plant utilizing chemical shim is such that the severity of an ejected RCCA is inherently limited. In general, the reactor is operated with the RCCAs inserted only far enough to permit load follow. Reactivity changes caused by core depletion and xenon transients are compensated by boron changes. Further, the location and grouping of control RCCA banks are selected during the nuclear design to lessen the severity of a RCCA ejection accident. Therefore, should a RCCA be ejected from its normal position during full power operation, only a minor reactivity excursion, at worst, could be expected to occur.

However, it may be occasionally desirable to operate with larger than normal insertions. For this reason, a rod insertion limit is defined as a function of power level. Operation with the RCCAs above this limit guarantees adequate shutdown capability and acceptable power distribution. The position of all RCCAs is continuously indicated in the control room. An alarm will occur if a bank of RCCAs approaches its insertion limit or if one RCCA deviates from its bank. Operating instructions may require boration, as necessary, at the low level alarm. The low-low alarm alerts the operator to stop diluting if in progress, and verify Shutdown Margin is within the limits specified in the COLR or initiate boration to restore Shutdown Margin to within limit.

Reactor Protection

The reactor protection in the event of a rod ejection accident has been described in Reference 5. The protection for this accident is provided by high neutron flux trip (high and low setting) and high positive rate neutron flux trip. These protection functions are described in detail in Section 7.2.

Effects on Adjacent Housings

Disregarding the remote possibility of the occurrence of a RCCA mechanism housing failure, investigations have shown that failure of a housing due to either longitudinal or circumferential cracking would not cause damage to adjacent housings. However, even if damage is postulated, it would not be expected to lead to a more severe transient since RCCAs are inserted in the core in symmetric patterns, and control rods immediately adjacent to worst ejected rods are not in the core when the reactor is critical. Damage to an adjacent housing could, at worst, cause that RCCA not to fall on receiving a trip signal; however, this is already taken into account in the analysis by assuming a stuck rod adjacent to the ejected rod.

15.4.8.1.2 Limiting Criteria

This event is classified as an ANS Condition IV incident. See Subsection 15.0.1 for a discussion of ANS classifications. Due to the extremely low probability of a RCCA ejection accident, some fuel damage could be considered an acceptable consequence.

Comprehensive studies of the threshold of fuel failure and of the threshold or significant conversion of the fuel thermal energy to mechanical energy, have been carried out as part of the SPERT project by the Idaho Nuclear Corporation (Reference 6). Extensive tests of UO_2 zirconium clad fuel rods representative of those in pressurized water reactor type cores have demonstrated failure thresholds in the range of 240 to 257 cal/gm. However, other rods of a slightly different design have exhibited failures as low as 225 cal/gm. These results differ significantly from the TREAT (Reference 7) results, which indicated a failure threshold of 280 cal/gm. Limited results have indicated that this threshold decreases by about 10% with fuel burnup. The clad failure mechanism appears to be melting for zero burnup rods and brittle fracture for irradiated rods. Also important is the conversion ratio of thermal to mechanical energy. This ratio becomes marginally detectable above 300 cal/gm for unirradiated rods and 200 cal/gm for irradiated rods; catastrophic failure, (large fuel dispersal, large pressure rise) event for irradiated rods, did not occur below 300 cal/gm.

In view of the above experimental results, criteria are applied to ensure that there is little or no possibility of fuel dispersal in the coolant, gross lattice distortion, or severe shock waves. These criteria are:

a. Average fuel pellet enthalpy at the hot spot below 225 cal/gm for unirradiated fuel and 200 cal/gm for irradiated fuel.

b. Peak reactor coolant pressure less than that which could cause stresses to exceed the faulted condition stress limits.

c. Fuel melting will be limited to less than ten percent of the fuel volume at the hot spot even if the average fuel pellet enthalpy is below the limits of criterion 1 above.

It should be noted that the FSAR included an additional criterion that the average clad temperature at the hot spot must remain below 2,700°F. The elimination of this criterion as a basis for evaluating the RCCA ejection accident results is consistent with the revised Westinghouse acceptance criteria for this event (Reference 10).

15.4.8.2 Analysis of Effects and Consequences

Method of Analysis

The calculation of the RCCA ejection transient is performed in two stages, first an average core channel calculation and then a hot region calculation. The average core calculation is performed using spatial neutron kinetics methods to determine the average power generation with time including the various total core feedback effects, i.e., Doppler reactivity and moderator reactivity. Enthalpy and temperature transients in the hot spot are then determined by multiplying the average core energy generation by the hot channel factor and performing a fuel rod transient heat transfer calculation. The power distribution calculated without feedback is pessimistically assumed to persist throughout the transient.

A detailed discussion of the method of analysis can be found in Reference 5.

Average Core Analysis

The spatial kinetics computer code, TWINKLE (Reference 1), is used for the average core transient analysis. This code solves the two group neutron diffusion theory kinetic equation in one, two or three spatial dimensions (rectangular coordinates) for six delayed neutron groups and, up to 2000 spatial points. The computer code includes a detailed multiregion, transient fuel-clad-coolant heat transfer model for calculation of pointwise Doppler and moderator feedback effects. In this analysis, the code is used as a one dimensional axial kinetics code since it allows a more realistic representation of the spatial effects of axial moderator feedback and RCCA movement. However, since the radial dimension is missing, it is still necessary to employ very conservative methods (described in the following) of calculating the ejected rod worth and hot channel factor. Further description of TWINKLE appears in Subsection 15.0.11.

Hot Spot Analysis

In the hot spot analysis, the initial heat flux is equal to the nominal times the design hot channel factor. During the transient, the heat flux hot channel factor is linearly increased to the transient value in 0.1 second, the time for full ejection of the rod. Therefore, the assumption is made that the hot spots before and after ejection are coincident. This is very conservative since the peak after ejection will occur in or adjacent to the assembly with the ejected rod, and prior to ejection the power in this region will necessarily be depressed.

The hot spot analysis is performed using the detailed fuel-and cladding transient heat transfer computer code, FACTRAN (Reference 2). This computer code calculates the transient temperature distribution in a cross section of a metal clad UO_2 fuel rod, and the heat flux at the surface of the rod, using as input the nuclear power versus time and the local coolant conditions. The zirconium-water reaction is explicitly represented, and all material properties are represented as functions of temperature. A conservative pellet radial power distribution is used within the fuel rod.

FACTRAN uses the Dittus-Boelter or Jens-Lottes correlation to determine the film heat transfer before DNB, and the Bishop-Sandburg-Tong correlation (Reference 8) to determine the film boiling coefficient after DNB. The BST correlation is conservatively used assuming zero bulk fluid quality. The DNB ratio is not calculated, instead the code is forced into DNB by specifying a conservative DNB heat flux. The gap heat transfer coefficient can be calculated by the code; however, it is adjusted in order to force the full power steady-state temperature distribution to agree with the fuel heat transfer design codes. Further description of FACTRAN appears in Subsection 15.0.11.

<u>System Overpressure Analysis</u>

Because safety limits for fuel damage specified earlier are not exceeded, there is little likelihood of fuel dispersal into the coolant. The pressure surge may therefore be calculated on the basis of conventional heat transfer from the fuel and prompt heat generation in the coolant.

The pressure surge is calculated by first performing the fuel heat transfer calculation to determine the average and hot spot heat flux versus time. Using this heat flux data, a THINC (Section 4.4) calculation is conducted to determine the volume surge. Finally, the volume surge is simulated in a plant transient computer code. This code calculates the pressure transient taking into account fluid transport in the reactor coolant system and heat transfer to the steam generators. No credit is taken for the possible pressure reduction caused by the assumed failure of the control rod pressure housing.

15.4.8.2.1 <u>Calculation of Basic Parameters</u>

Input parameters for the analysis are conservatively selected on the basis of values calculated for this type of core. The more important parameters are discussed below. Table 15.4-3 presents the parameters used in this analysis.

<u>Ejected Rod Worths and Hot Channel Factors</u>

The values for ejected rod worths and hot channel factors are calculated using either three dimensional static methods or by a synthesis method employing one dimensional and two dimensional calculations. Standard nuclear design codes are used in the analysis. No credit is taken for the flux flattening effects of reactivity feedback. The calculation is performed for the maximum allowed bank insertion at a given power level, as determined by the rod insertion limits. Adverse xenon distributions are considered in the calculation.

Appropriate margins are added to the ejected rod worth and hot channel factors to account for any calculational uncertainties, including an allowance for nuclear power peaking due to densification.

Power distributions before and after ejection for a "worst case" can be found in Reference 5. During plant startup physics testing, ejected rod worths and power distributions are measured in the zero and full power rodded configurations and compared to values used in the analysis. It has been found that the ejected rod worth and power peaking factors are consistently overpredicted in the analysis.

Reactivity Feedback Weighting Factors

The largest temperature rises, and hence the largest reactivity feedbacks occur in channels where the power is higher than average. Since the weight of a region is dependent on flux, these regions have high weights. This means that the reactivity feedback is larger than that indicated by a simple channel analysis. Physics calculations have been carried out for temperature changes with a flat temperature distribution, and with a large number of axial and radial temperature distributions. Reactivity changes were compared and effective weighting factors determined. These weighting factors take the form of multipliers which when applied to single channel feedbacks correct them to effective whole core feedbacks for the appropriate flux shape. In this analysis, since a one dimensional (axial) spatial kinetics method is employed, axial weighting is not necessary if the initial condition is made to match the ejected rod configuration. In addition, no weighting is applied to the moderator feedback. A conservative radial weighting factor is applied to the transient fuel temperature to obtain an effective fuel temperature as a function of time accounting for the missing spatial dimension. These weighting factors have also been shown to be conservative compared to three dimensional analysis (Reference 5).

Moderator and Doppler Coefficient

The critical boron concentrations at the beginning of life and end of life are adjusted in the nuclear code in order to obtain moderator density coefficient curves which are conservative compared to actual design conditions for the plant. As discussed above, no weighting factor is applied to these results.

The Doppler reactivity defect is determined as function of power level using a one dimensional steady-state computer code with a Doppler weighting factor of 1.0. The Doppler defect used is ≤0.965% $\Delta\rho$ for the BOL cases and ≤0.893% $\Delta\rho$ for the EOL cases. The Doppler weighting factor will increase under accident conditions, as discussed above.

Delayed Neutron Fraction, β

Calculations of the effective delayed neutron fraction (β_{eff}) typically yield values no less than 0.70% at beginning of life and 0.50% at end of life for the first cycle. The accident is sensitive to β if the ejected rod worth is equal to or greater than β as in zero power transients. In order to allow for future cycles, pessimistic estimates of β of 0.55% at beginning of cycle and 0.44% at end of cycle were used in the analysis.

Trip Reactivity Insertion

The trip reactivity insertion assumed is given in Table 15.4-3 and includes the effect of one stuck RCCA. These values are reduced by the ejected rod reactivity. The shutdown reactivity was simulated by dropping a rod of the required worth into the core. The start of rod motion occurred 0.5 seconds after the high neutron flux trip point was reached. This delay is assumed to consist of 0.2 seconds for the instrument channel to produce a signal, 0.15 seconds for the trip breaker to open and 0.15 seconds for the coil to release the rods. A curve of trip rod insertion versus time was used which assumed that insertion to the dashpot does not occur until 2.7 seconds after the start of fall. The choice of such a conservative insertion rate means that there is over one second after the trip point is reached before significant shutdown reactivity is inserted into the core. This is a particularly important conservatism for hot full power accidents.

The minimum design shutdown available for this plant at HZP may be reached only at end of life in the equilibrium cycle. This value includes an allowance for the worst stuck rod, adverse xenon distribution conservative Doppler and moderator defects, and an allowance for calculational uncertainties. Physics calculations for this plant have shown that the effect of two stuck RCCAs (one of which is the worst ejected rod) is to reduce the shutdown by about an additional one percent Δk. Therefore, following a reactor trip resulting from an RCCA ejection accident, the reactor will be subcritical when the core returns to HZP.

Depressurization calculations have been performed for a typical four-loop plant assuming the maximum possible size break (2.75 inch diameter) located in the reactor pressure vessel head. The results show a rapid pressure drop and a decrease in system water mass due to the break. The safety injection system is actuated by the low pressurizer pressure trip within one minute after the break. The reactor coolant pressure continues to drop and reaches saturation (1100 to 1300 psi depending on the system temperature) in about two to three minutes. Due to the large thermal inertia of primary and secondary system, there has been no significant decrease in the reactor coolant system temperature below no-load by this time, and the depressurization itself has caused an increase in shutdown margin by about 0.2% Δk due to the pressure coefficient. The cooldown transient could not absorb the available shutdown margin until more than ten minutes after the break. The addition of high borated (2300 ppm) safety injection flow starting one minute after the break is much more than sufficient to ensure that the core remains subcritical during the cooldown.

Reactor Protection

As discussed in Subsection 15.4.8.1.1, reactor protection for a rod ejection is provided by high neutron flux trip (high and low setting) and high positive rate neutron flux trip. These protection functions are part of the reactor trip system. No single failure of the reactor trip system will negate the protection functions required for the rod ejection accident, or adversely affect the consequences of the accident.

15.4.8.2.2 Results

Cases are presented for both beginning and end of life at zero and full power.

1. Beginning of Cycle, Full Power

   Control bank D was assumed to be inserted to its insertion limit. The worst ejected rod worth and hot channel factor were conservatively calculated to be 0.2% Δk and 6.10, respectively. The peak hot spot clad average temperature was 2434°F. The peak hot spot fuel center temperature reached melting, conservatively assumed at 4900°F. However, melting was restricted to less than 10% of the pellet.

2. Beginning-of-Cycle, Zero Power

   For this condition, control bank D was assumed to be fully inserted and banks B and C were at their insertion limits. The worst ejected rod is located in control bank D and has a worth of 0.765% Δk and a hot channel factor of 11.5. The peak hot spot clad average temperature reached 2348°F, the fuel center temperature was 3616°F.

3. End of Cycle, Full Power

   Control bank D was assumed to be inserted to its insertion limit. The ejected rod worth and hot channel factors were conservatively calculated to be 0.25% Δk and 6.40, respectively. This resulted in a peak clad average temperature of 2235°F. The peak hot spot fuel temperature reached melting conservatively assumed at 4800°F. However, melting was restricted to less than 10% of the pellet.

4. End of Cycle, Zero Power

   The ejected rod worth and hot channel factor for this case were obtained assuming control bank D to be fully inserted and banks C and B at its insertion limit. The results were 0.8% Δk and 23.0, respectively. The peak clad average and fuel center temperatures were 2337 and 3479°F. The Doppler weighting factor for this case is significantly higher than for the other cases due to the very large transient hot channel factor.

A summary of the cases presented above is given in Table 15.4-3. The nuclear power and hot spot fuel and clad temperature transients for the worst cases are presented in Figures 15.4-18 through 15.4-21. (Beginning of life full power and end of life full power.)

The calculated sequence of events for the worst case rod ejection accidents, as shown in Figures 15.4-18 through 15.4-21, is presented in Table 15.4-1. For all cases, reactor trip occurs very early in the transient, after which the nuclear power excursion is terminated. As discussed previously in Subsection 15.4.8.2.2, the reactor will remain subcritical following reactor trip.

The ejection of an RCCA constitutes a break in the reactor coolant system, located in the reactor pressure vessel head. The effects and consequences of loss of coolant accidents are discussed in Subsection 15.6.5. Following the RCCA ejection, the operator would follow the same emergency instructions as for any other loss of coolant accident to recover from the event.

Fission Product Release

It is assumed that fission products are released from the gaps of all rods entering DNB. In all cases considered, less than 10% of the rods entered DNB based on a detailed three dimensional THINC analysis (Reference 5).

Pressure Surge

A detailed calculation of the pressure surge for an ejection worth of one dollar at beginning of life, hot full power, indicates that the peak pressure does not exceed that which would cause stress to exceed the faulted condition stress limits (Reference 5). Since the severity of the present analysis does not exceed the "worst case" analysis, the accident for this plant will not result in an excessive pressure rise or further damage to the reactor coolant system.

Lattice Deformations

A large temperature gradient will exist in the region of the hot spot. Since the fuel rods are free to move in the vertical direction, differential expansion between separate rods cannot produce distortion. However, the temperature gradients across individual rods may produce a differential expansion tending to bow the midpoint of the rods toward the hotter side of the rod. Calculations have indicated that this bowing would result in a negative reactivity effect at the hot spot since Westinghouse cores are under-moderated, and bowing will tend to increase the under-moderation at the hot spot. Since the 17 x 17 fuel design is also under-moderated, the same effect would be observed. In practice, no significant bowing is anticipated, since the structural rigidity of the core is more than sufficient to withstand the forces produced. Boiling in the hot spot region would produce a net flow away from that region. However, the heat from the fuel is released to the water relatively slowly, and it is considered inconceivable that cross flow will be sufficient to produce significant lattice forces. Even if massive and rapid boiling, sufficient to distort the lattice, is hypothetically postulated, the large void fraction in the hot spot region would produce a reduction in the total core moderator to fuel ratio, and a large reduction in this ratio at the hot spot. The net effect would therefore be a negative feedback. It can be concluded that no conceivable mechanism exists for a net positive feedback resulting from lattice deformation. In fact, a small negative feedback may result. The effect is conservatively ignored in the analysis.

15.4.8.3 Radiological Consequences of a Postulated Rod Ejection Accident (AST)

The key inputs and assumptions used in the AST Control Rod Ejection Accident (CREA) radiological consequence analysis are summarized below and provided in Table 15.4-4.

Two cases are considered when analyzing the radioactive release due to a CREA.

Case 1: Containment Leakage

For Case 1, the ejected control rod is assumed to breach the reactor pressure vessel (RPV), effectively causing the equivalent of a small break loss of coolant accident. In this case, all activity from damaged fuel that has been mixed with the primary coolant of the reactor coolant system (RCS) leaks directly to the containment volume. This flashed release is assumed to instantaneously and homogeneously mix with the containment atmosphere and subsequently be available for release to the environment via a containment leak rate limit, or L_a, conservatively assumed to be 0.2% per day for this accident analysis. In accordance with RG 1.183 guidance, the leak rate is reduced by 50% after 24 hours, based on the containment pressure decreasing over time.

Case 2: Steam Generator PORV Release

For Case 2, no breach of the RPV is assumed following the rod ejection. In this case, reactor coolant system (RCS) integrity is maintained and all activity from damaged fuel that has been mixed with the RCS leaks to the secondary side through the steam generator (SG) tubes at a conservative rate of 1.0 gpm total leakage. From here, activity is available for release to the environment by steaming of the SG power operated relief valves (PORVs). An average rate of release is assumed. In addition to the activity released from the primary to secondary coolant, iodine activity in the secondary coolant at the TS limit (i.e., 0.1 μCi/gm Dose Equivalent (DE) I-131) is also assumed to be released.

Fuel Damage and Core Source Term

The CREA core source terms are those associated with a DBA power level of 3658.3 MWt. This power level bounds the MUR power uprate Rated Thermal Power level including measurement uncertainties.

For the radiological dose analysis, the sudden rod ejection and localized temperature spike associated with the CREA is assumed to damage of 10% of the core fuel rods. Only 2.5 % of the damaged core fuel rods release melted fuel activity (i.e., 0.25% of the total core melts). Therefore for both cases, the source term available for release is associated with this fraction of melted fuel and the fraction of core activity existing in the gap.

The damaged fuel is assumed to have operated at a radial peaking factor of 1.7.

Activity Release Fractions

Release fractions and transport fractions conform to RG 1.183, Appendix H and Table 3. To conform with this regulatory guidance, 10% of the core inventory of iodine and noble gas is assumed to be in the fuel-clad gap. Additionally, Table 3 of Regulatory Guide 1.183 shows that 12% of the core cesium and rubidium should be assumed to be in the fuel-clad gap and should be released in its entirety from the damaged 10% of the total core. However, to account for fuel burnup in excess of the referenced assumptions, the cesium and rubidium release fraction is doubled. Although analyses have shown that isotopic activity fractions in the fuel-clad gap may in fact decrease when "burning" the fuel longer than the 54 GWd/MTU specified in Regulatory Guide 1.183, this 100% increase in the gap fractions is used as an accepted and conservative means of bounding all extended burnup phenomena. With regard to the fraction released from melted fuel, it is assumed that 90% of the core inventory of iodine and noble gas, and 76% of the core cesium and rubidium remain available for release due to melting (i.e., these are the remaining fractions of activity that are not in the fuel-clad gap). Again, in conformance with RG 1.183, it would be assumed that 100% of the noble gases, 25% of the iodines, and 50% of the cesium and rubidium (i.e., considered particulate/aerosol nuclides) released from the melted fuel would be available for release from containment. However, for this analysis the assumption of 25% of the iodines being available for release was increased to 50%. This was done to prevent a "double counting" of the iodine removal due to plate-out in containment, because this analysis credits Powers' Natural Deposition model for plateout, as opposed to the historically assumed 50% plateout.

These activity release fractions are input to the RADTRAD code through the use of the Release Fractions and Timing (RFT) file.

Airborne Activity Removal Mechanisms in Containment

As discussed below only natural deposition, decay, and leakage are credited.

Natural Deposition:

The RADTRAD computer program, including the Powers' Natural Deposition algorithm based on NUREG/CR-6189, is used for modeling aerosol deposition in containment. No natural deposition is assumed for elemental or organic iodine. The RADTRAD lower bound level (i.e., 10 percent) of deposition credit is used.

Decay Credited:

Decay of radioactivity is credited in all compartments, prior to release. This is implemented in RADTRAD using the half-lives in the Nuclide Inventory File (NIF). The RADTRAD decay plus daughter option is used. In reality, daughter products such as xenon from iodines or iodines from tellurium are unlikely to readily escape from the matrix in which the parent iodine or tellurium is contained. Nevertheless, the RADTRAD feature to include daughter effects is selected for conservatism.

Steaming Release Rates and Partition Factors:

Activity that originates in the RCS is released to the secondary coolant by means of the primary-to-secondary coolant leak rate. This assumed leak rate value is a total of 1.0 gpm. For input into RADTRAD, this rate is converted from gallons per minute to cubic feet per minute, making it equal to 0.1337 cfm.

For Case 2, the release to the environment is associated with the secondary coolant steaming from the SGs. Because of this release dynamic, RG 1.183 allows for a reduction in the amount of activity released to the environment based on partitioning of nuclides between the liquid and gas states of water. For iodine, the partition factor of 0.01 was taken directly from the suggested guidance. However, there is no explicit guidance with regard to other particulate nuclides. Reviewing the specified AST release fractions, it is concluded that the only nuclides other than iodines to be released from the core source term are cesiums, rubidium, and noble gases. For cesiums and rubidium, a partition factor of 0.0055 is used which bounds the value of 0.00529 shown in ANSI Standard, ANS/ANSI 18.1 - 1999, for Cs-134 which has the largest partition factor of these isotopes. Because of their volatility, 100% of the noble gases are assumed to be released.

The methodology used to model steaming of activity through PORVs following the postulated CREA event of Case 2 assumes an average cumulative release rate through the SG PORVs that, for simplicity and conservatism, is reduced in steps. The partition factors discussed above are applied to these release rates, which were derived from the total time increment mass releases. The following table shows the time steps, isotopic partition factors, and associated release rates; conversion of this data from mass to volumetric flow rates was performed based on cooled liquid conditions (i.e., 62.4 lbm/ft^3), as specified by RG 1.183.

CREA Partition Factors and Release Rates

Time Interval (hrs)	Total Steam Mass Release (lbm)	Iodine Partition Factor	Cesium Partition Factor	Noble Gas Partition Factor	Steam Release Rate for Iodines (cfm)	Steam Release Rate for Cesiums (cfm)	Steam Release Rate for Noble Gases (cfm)
0 - 0.0556	600,000	0.01	0.0055	1	2.8833E+01	1.5858E+01	2.8833E+03
0.0556 - 1.1111	1,900,000	0.01	0.0055	1	4.8055E+00	2.6430E+00	4.8055E+02
1.1111 - 720	0	0.01	0.0055	1	0	0	0

χ/Q Calculations (Meteorology)

Releases from the SG PORVs are considered elevated releases due to the high steaming rates, and the associated χ/Qs have been reduced by a factor of five per guidance in RG 1.194 for energetic releases from steam relief valves, if (1) the release point is uncapped and vertically oriented, and (2) the time-dependent vertical velocity exceeds the 95th-percentile wind speed, at the release point height. Byron Station and Braidwood Station meet these criteria.

Assumptions and Inputs

The following inputs and assumptions were used in the CREA analysis.

a. Core inventory is based on a DBA power level of 3658.3 MWt. This power level bounds the MUR power uprate Rated Thermal Power level including measurement uncertainties.

b. 10% of the fuel is damaged during the initiation of this accident, and is assumed to have failed.

c. The damaged fuel is assumed to have operated at a radial peaking factor of 1.7.

d. 10% of the core inventory of noble gases and iodines are released from the fuel gap (RG 1.183, Appendix H). Release fractions of other nuclide groups contained in the fuel gap are detailed in Table 3 of RG 1.183, and to account for gap fraction uncertainty. Due to fuel burnup, fractions from the referenced table are doubled.

e. 2.5% of the damaged fuel rods will experience melting during the CREA.

f. 100% of noble gases and 50% of the iodines contained in the melted fuel fraction are assumed to be released to the reactor coolant in accordance with Appendix H of RG 1.183. Fractions of other nuclides released from the melted fuel are used from Table 2 of RG 1.183. Though these are described as LOCA values for fuel melt release, they are conservatively used for the other nuclide groups.

g. The activity released from the fuel from either the gap or from fuel pellets is assumed to be instantaneously mixed with the reactor coolant within the pressure vessel.

h. All iodine released from the SGs is assumed to be of the elemental species. This is done for RADTRAD simulation considerations, and is consistent with the RG 1.183 specification of 97% elemental and 3% organic, because elemental and organic iodine are identically treated by the computer model.

i. The CR ventilation system is assumed to realign to the emergency mode of operation 30 minutes after the initiation of this design basis accident.

j. For the containment leakage case, all leakage is assumed to be at an L_a of 0.2% per day for the first 24 hours and 0.1% per day thereafter.

Dose Results

Radiological doses resulting from a design basis CREA for a CR operator and a person located at the EAB or LPZ are to be less than the regulatory dose limits as given in the Table below.

Regulatory Dose Limits - CREA

Dose Type	Control Room (rem)	EAB and LPZ (rem)
TEDE Dose	5[a]	6.3[b]

Notes:
[a] 10 CFR 50.67
[b] 10 CFR 50.67 as modified by Regulatory Guide 1.183 (Table 6, Page 1.183-20)

The table below provides the results from the simulations modeled using the RADTRAD code.

Control Rod Ejection Accident
Radiological Analysis Results
(Maximum of Either Byron or Braidwood)

Case 1: Containment Leakage CREA Dose Assessment Results		
Control Room (rem TEDE)	EAB (rem TEDE)	LPZ (rem TEDE)
4.538	5.358	2.278
Case 2: Steam Generator PORV Release CREA Dose Assessment Results		
Control Room (rem TEDE)	EAB (rem TEDE)	LPZ (rem TEDE)
0.424	1.707	0.304

For the cases analyzed in this calculation, it is shown that a Case 1 CREA that breaches the RPV, and causes a containment release, would be the bounding CREA scenario. All doses are below the Regulatory Dose Limits, so it is verified that this design basis Control Rod Ejection Accident is sufficiently mitigated at both Byron and Braidwood Stations.

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15.4.8.4 Conclusions

Conservative analyses indicate that the described fuel and cladding limits are not exceeded. It is concluded that there is no danger of sudden fuel dispersal into the coolant. Since the peak pressure does not exceed that which would cause stresses to exceed the faulted condition stress limits, it is concluded that there is no danger of further consequential damage to the reactor coolant system. The analyses have demonstrated that the number of fuel rods entering DNB is limited to less than 10% of the fuel rods in the core. The radiological consequences of this event, based on 10% of the fuel rods being damaged, are well within the dose limits of 10 CFR Part 50.67.

15.4.9 References

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4. NUREG-0800, Standard Review Plan Section 15.4.8, Appendix A, “Radiological Consequences of a Control Rod Ejection Accident (PWR),” Revision 1, July 1981.

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6. Taxelius, T. G. (Ed), "Annual Report - SPERT Project, October 1968, September 1969," Idaho Nuclear Corporation IN-1370, June 1970.

7. Liimataninen, R. C. and Testa, F. J., "Studies in TREAT of Zircaloy-2-Clad, UO2-Core Simulated Fuel Elements" ANL-7225, January - June 1966, p. 177, November 1966.

8. Bishop, A. A., Sandberg, R. O., and Tong, L. S., "Forced Convection Heat Transfer at High Pressure After the Critical Heat Flux," ASME 65-HT-31, August 1965.

9. Hassler, R. L., et al, "Methodology for the Analysis of the Dropped Rod Event," WCAP-11394-P-A (Proprietary), Revision 1, January 1990, and WCAP-11395 (Non-Proprietary), April 1987.

10. Letter from W. J. Johnson of Westinghouse to R. C. Jones of the NRC, "Use of 2700°F PCT Acceptance Limit in Non-LOCA Accidents," NS-NRC-89-3466, October 1989.

11. Friedland, A.J., and Ray, S., "Revised Thermal Design Procedure," WCAP-11397-P-A (Proprietary), WCAP-11397-A (Non-Proprietary), April 1989.

12. NRC Information Notice 94-60, "Potential Overpressurization of the Main Steam System," August 22, 1994.

13. Ankney, R. D. and Grover, J. L., "Analysis Update for the Inadvertent Loading Event," WCAP-16676-NP, March 2009.

TABLE 15.4-1

TIME SEQUENCE OF EVENTS FOR INCIDENT WHICH CAUSE REACTIVITY AND POWER DISTRIBUTION ANOMALIES

ACCIDENT	EVENT	TIME (sec.)
Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical or Low Power Startup Condition	Initiation of uncontrolled rod withdrawal from 10^{-9} of nominal power	0.0
	1 Power range high neutron flux low setpoint reached	11.1
	2 Peak nuclear power occurs	11.3
	3 Rod begin to fall into core	11.6
	4 Peak heat flux occurs	13.6
	5 Minimum DNBR occurs	13.6
	6 Peak average clad temperature occurs	13.9
	7 Peak average fuel temperature occurs	14.0
Uncontrolled RCCA bank withdrawal at power		
1. Case A	Initiation of uncontrolled RCCA withdrawal at a high reactivity insertion rate (80 pcm/sec)	0
	Power range high neutron flux high trip point reached	1.6
	Rods begin to fall into core	2.1
	Minimum DNBR occurs	3.2

TABLE 15.4-1 (Cont'd)

2. Case B	Initiation of uncontrolled RCCA withdrawal at a small reactivity insertion rate (3.0 pcm/sec)	0
	Overtemperature ΔT trip signal initiated	35.7
	Rods begin to fall into core	43.7
	Minimum DNBR occurs	44.3
Rod Cluster Control Assembly Ejection		
1. Beginning-of-Life, Full Power	Initiation of rod ejection	0.0
	Power range high neutron flux setpoint reached	0.05
	Peak nuclear power occurs	0.13
	Rods begin to fall into core	0.55
	Peak fuel average temperature occurs	2.28
	Peak clad temperature occurs	2.38
	Peak heat flux occurs	2.39

TABLE 15.4-1 (Cont'd)

2. End-of-Life, Full Power	Initiation of rod ejection	0.0
	Power range high neutron flux setpoint reached	0.04
	Peak nuclear power occurs	0.13
	Rods begin to fall into core	0.54
	Peak fuel average temperature occurs	2.31
	Peak clad temperature occurs	2.40
	Peak heat flux occurs	2.41

Table 15.4-2 has been deleted.

TABLE 15.4-3

PARAMETERS USED IN THE ANALYSIS OF THE ROD CLUSTER CONTROL

ASSEMBLY EJECTION ACCIDENT

TIME IN LIFE	BOL-HFP BEGINNING	BOL-HZP BEGINNING	EOL-HFP END	EOL-HZP END
Power Level, %	100 (including uncertainties)	0	100 (including uncertainties)	0
Ejected rod worth, Δ%WK	0.2	0.765	0.25	0.8
Delayed neutron fraction, %	0.55	0.55	0.44	0.44
Feedback reactivity weighting	1.30	2.07	1.30	3.55
Trip reactivity, Δ%WK	4.0	2.0	4.0	2.0
F_q before rod ejection	2.60		2.60	
F_q after rod ejection	6.10	11.5	6.40	23.0
Number of operational pumps	4	2	4	2

TABLE 15.4-3 (Cont'd)

TIME IN LIFE	BOL-HFP BEGINNING	BOL-HZP BEGINNING	EOL-HFP END	EOL-HZP END
Max. fuel pellet average temperature, °F	4128	3123	4044	3056
Max. fuel center temperature, °F	>4900	3616	>4800	3479
Max. clad average temperature, °F	2434	2348	2369	2337
Max. fuel store energy, cal/gm	181	130	177	127
% Fuel Melt	<10%	0	<10%	0

TABLE 15.4-4

INPUT PARAMETERS FOR THE CREA RADIOLOGICAL CONSEQUENCES ANALYSIS USING AST

Parameter	Unit	Value	Notes
Portion of Core Fuel Rods Experiencing Cladding Damage	%	10	Bounds value predicted by safety analysis. The acceptance criteria for fuel melting is a maximum of 10%.
Melted fuel	% of core	0.25	Based on 50% of the rods that violate the DNB limit having melting in the inner 10% over 50% of the axial length.
Gap Activity Alkali Metals	 %	 0.24	As a significant number of fuel assemblies not qualifying for AST due to their containing fuel rods with maximum linear heat generation rates exceeding 6.3 kilowatts per foot peak rod average power for burnups exceeding 54 GWD/MTU, the fuel will be treated as having gap fractions a factor of 2 greater than the RG 1.183 values.
Fraction of activity released to containment: From gap inventory Iodine Noble Gases From melted fuel Iodine Noble gas Iodine plateout onto containment surfaces	 - - - - -	 1.0 1.0 0.5 1.0 0.5	
Iodine Chemical Species in Containment: Aerosol Elemental Organic Iodine Chemical Species in Release from SG to Environment: Elemental Organic	 % % % % %	 95 4.85 0.15 97 3	.

TABLE 15.4-4 (Cont'd)

INPUT PARAMETERS FOR THE CREA RADIOLOGICAL CONSEQUENCES ANALYSIS USING AST

Parameter	Unit	Value	Notes
Iodine Removal Coefficients in Containment	N/A	See Notes	Typically, no credit is taken for continuing iodine removal in the containment for the rod ejection accident, however under provisions allowed by the AST governing RG 1.183, Power's model for particulate deposition removal may be credited, if 50% plateout is not credited.
Containment Leak Rate: 0-24 Hours > 24 Hours	weight %/day	0.20 0.10	The design basis containment leak rate at 24 hours is consistent with guidance of RG 1.183.
Fraction of activity released to primary coolant (for primary to secondary leakage pathway): From gap inventory Iodine Noble Gases From melted fuel Iodine Noble gas	 - - - -	 1.0 1.0 0.5 1.0	
Iodine Chemical Species in Primary Coolant: Elemental Organic Particulate	 % % %	 100 0 0	
Steam Released to Environment: 0 - 200 sec 200 - 4000 sec	 lb/sec lb/sec	 3000 500	Bounds release predicted by small break LOCA analysis.
Chemical form of radioiodine released to the containment atmosphere: Cesium iodine Elemental iodine Organic iodide	 % % %	 95 4.85 0.15	

TABLE 15.4-4 (Cont'd)

ASSUMPTIONS USED FOR THE ROD CLUSTER CONTROL ASSEMBLY EJECTION ACCIDENT

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TABLE 15.4-5

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TABLE 15.4-6

Flux Map Review Criteria

Number of Available Detector Locations	**Measured vs. Predicted Detector Reaction Rate Comparison***	**Symmetric Thimble Reaction Rate Comparison****
55 to 58	10%	7%
49 to 54	8%	5%
44 to 48	6%	5%

*The review criterion is the table value (%) or an absolute normalized reaction rate difference equal to the table value divided by 100% (e.g., 10% / 100% = 0.1), whichever is greater.

**Applicable to symmetric thimbles with normalized reaction rates above 0.7. The review criterion is relative to the expected reaction rate difference

15.5 INCREASE IN REACTOR COOLANT INVENTORY

Discussion and analysis of the following events is presented in this section:

a. inadvertent operation of emergency core cooling system during power operation,

b. chemical and volume control system malfunction that increases reactor coolant inventory, and

c. a number of BWR transients (not applicable to the Byron/Braidwood Stations).

These events, considered to be ANS Condition II, cause an increase in reactor coolant inventory. Subsection 15.0.1 contains a discussion of ANS classifications.

15.5.1 Inadvertent Operation of Emergency Core Cooling System During Power Operation

15.5.1.1 Identification of Causes and Accident Description

Inadvertent operation of the emergency core cooling system (ECCS) at power could be caused by operator error, test sequence error, or a false electrical actuation signal. A spurious signal initiated after the logic circuitry in one solid-state protection system train for any of the following engineered safety feature (ESF) functions could cause this incident by actuating the ESF equipment associated with the affected train.

a. High containment pressure,

b. Low pressurizer pressure, or

c. Low steamline pressure.

Following the actuation signal, the suction of the coolant charging pumps diverts from the volume control tank to the refueling water storage tank. Simultaneously, the valves isolating the charging pumps from the injection header automatically open and the normal charging line isolation valves close. The charging pumps force the borated water from the refueling water storage tank (RWST) through the pump discharge header, the injection line, and into the cold leg of each loop. The passive accumulator tank safety injection and low head system are available. However, they do not provide flow when the reactor coolant system (RCS) is at normal pressure.

A safety injection (SI) signal normally results in a direct reactor trip and a turbine trip. However, any single fault that actuates the ECCS will not necessarily produce a reactor trip.

If an SI signal generates a reactor trip, the operator should determine if the signal is spurious. If the SI signal is determined to be spurious, the operator should terminate SI and maintain the plant in the hot-standby condition as determined by appropriate recovery procedures. If repair of the ESF actuation system instrumentation is necessary, future plant operation will be in accordance with the Technical Specifications. If the SI results in discharge of coolant through the pressurizer safety valves, the operators will bring the plant to cold shutdown in order to inspect the valves.

If the reactor protection system does not produce an immediate trip as a result of the spurious SI signal, the reactor experiences a negative reactivity excursion due to the injected boron, which causes a decrease in reactor power. The power mismatch causes a drop in T_{avg} and consequent coolant shrinkage. The pressurizer pressure and water level decrease. Load decreases due to the effect of reduced steam pressure on load after the turbine throttle valve is fully open. If automatic rod control is used, these effects will lessen until the rods have moved out of the core. The transient is eventually terminated by the reactor protection system low pressurizer pressure trip or by manual trip.

The time to trip is affected by initial operating conditions. These initial conditions include the core burnup history which affects initial boron concentration, rate of change of boron concentration, and Doppler and moderator coefficients.

15.5.1.2 Analysis of Effects and Consequences

Method of Analysis

Inadvertent operation of the ECCS is analyzed using the LOFTRAN computer code (Reference 1). The code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, the feedwater system, the steam generator, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level.

Inadvertent operation of the ECCS at power is classified as a Condition II event, a fault of moderate frequency. The criteria established for Condition II events include the following.

a. Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design values,

b. Fuel cladding integrity shall be maintained by ensuring that the minimum departure from nucleate boiling ratio (DNBR) remains above the DNBR limit, derived at a 95% confidence level and 95% probability, and

c. An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

The inadvertent ECCS actuation at power event is analyzed to determine the maximum RCS pressure encountered throughout the accident. The most limiting case with respect to RCS pressure is an SI at Hot Full Power coincident with a reactor trip. Because of the pressure reduction from the reactor trip, the SI flow is maximized. The SI flow refills the pressurizer until the pressurizer is water solid, and the SI flow results in liquid discharge through the pressurizer safety valves.

The performance of the pressurizer safety valve system and the loads on pressurizer safety valves, associated piping, and supports as a result of liquid discharge through the pressurizer safety valves, was determined to be acceptable (References 4 and 5).

The Inadvertent Operation of the ECCS During Power Operation event does not progress into a stuck open Pressurizer Safety Valve LOCA event. All three valves may lift in response to the event, but they will reclose. The resulting leakage from up to three pressurizer safety valves that are seated is bounded by flow through one fully open valve. The consequences of the event are bounded by the analysis described in UFSAR Section 15.6.1, “Inadvertent opening of Pressurizer Safety or Relief Valve” (References 6 and 7). This event is also classified as an event of moderate frequency.

American Nuclear Society standard 51.1/N18.2-1973 (Reference 2) describes example 15 of a condition II event as a "minor reactor coolant system leak which would not prevent orderly reactor shutdown and cooldown assuming makeup is provided by normal makeup systems only." In Reference 2, normal makeup systems are defined as those systems normally used to maintain reactor coolant inventory under respective conditions of startup, hot standby, power operation, or cooldown, using onsite power. Since the cause of the water relief is the ECCS flow, the magnitude of the leak will be less than or equivalent to that of the ECCS (i.e., operation of the ECCS maintains RCS inventory during the postulated event and establishes the magnitude of the subject leak). Therefore, the above example of a Condition II event is met.

The inadvertent ECCS actuation at power event is also analyzed to determine the minimum DNBR value.

The most limiting case is a minimum reactivity feedback condition with the plant assumed to be in manual rod control. Because of the power and temperature reduction during the transient, operating conditions do not approach the core limits.

The minimum DNBR was obtained at time zero for both units. The Unit 1 specific results are presented here. However, they are representative of the results for both Unit 1 and Unit 2.

The analysis assumptions for the DNBR evaluation are as follows:

a. Initial Operating Conditions

   The event is analyzed with the revised thermal design procedure as described in WCAP-11397-P-A (Reference 3). Initial reactor power, RCS pressure and temperature are assumed to be at the nominal full power values. With the exception of the RCS average temperature bias, which is explicitly modeled in the analysis, uncertainties in initial conditions are included in the limit DNBR as described in Reference 3.

b. Moderator and Doppler Coefficients of Reactivity

   The analysis assumes a zero moderator temperature coefficient and a low absolute value Doppler power coefficient at beginning of life.

c. Reactor Control

   The reactor is assumed to be in manual rod control.

d. Pressurizer Pressure Control

Pressurizer heaters are assumed to be inoperable. This assumption yields a higher rate of pressure decrease which is conservative. Pressurizer spray and PORVs are assumed available in order to minimize RCS pressure.

e. Boron Injection

At the initiation of the event, two charging pumps inject borated water into the cold leg of each loop. The analysis assumes zero injection line purge volume for calculational simplicity; thus, the boration transient begins immediately in the analysis. The positive displacement charging pump is assumed to be inoperable at event initiation.

f. Turbine Load

The turbine load remains constant until the governor drives the throttle valve wide open. After the throttle valve is full open, turbine load decreases as steam pressure drops.

g. Reactor Trip

Reactor trip is initiated by a low pressurizer pressure signal at 1860 psia.

h. Decay Heat

The decay heat has no impact on the DNB case (i.e., minimum DNBR occurs prior to reactor trip). A conservative core residual heat generation based upon long-term operation at the initial power level is assumed.

i. Operator Action Time

Operator action is not required to mitigate the consequences of this event. Operator action is assumed to occur after the event to stabilize the plant in accordance with approved procedures to bring the plant to the applicable condition.

j. Pressurizer Safety Valves

The safety valves setpoints do not impact the minimum DNBR since the PORVs are assumed available to maintain low RCS pressure; this assumption is conservative with respect to DNBR.

k. Auxiliary Feedwater

Auxiliary feedwater was not credited.

l. Main Steam Safety Valves

The main steam safety valves are assumed conservatively to open at +5% above their nominal set pressure for the DNB case. No credit for steam dump is assumed in this analysis.

Plant systems and equipment credited for mitigating the effects of the accident are discussed in Subsection 15.0.8 and listed in Table 15.0-7. No single active failure in any of these systems or equipment will adversely affect the consequences of the accident.

Results

The transient response is shown in Figures 15.5-1 through 15.5-3. Table 15.5-1 shows the calculated sequence of events.

Nuclear power starts decreasing immediately due to boron injection, but steam flow does not decrease until later in the transient when the turbine throttle valve is wide open. The mismatch between load and nuclear power causes T_{avg}, pressurizer water level, and pressurizer pressure to drop. The reactor trips and control rods start moving into the core when the pressurizer pressure reaches the pressurizer low pressure trip setpoint. The DNBR increases throughout the transient.

15.5.1.3 Radiological Consequences

There are only minimal radiological consequences associated with inadvertent ECCS operation. The reactor trip causes a turbine trip and heat is removed from the secondary system through the steam generator power relief valves or safety valves. Since no fuel damage is postulated to occur for this transient, the radiological consequences associated with an atmospheric steam release from this event would be less severe than the steamline break event analyzed in Subsection 15.1.5.3.

Water relief from the pressurizer PORVs and safeties may result in overpressurization of the pressurizer relief tank (PRT), breaching the rupture disk and spilling contaminated fluid into containment. The radiological releases (offsite doses) resulting from breaking the PRT rupture disk are limited by isolation of the containment.

15.5.1.4 Conclusions

Results of the analysis show that spurious ECCS operation at measurement uncertainty recapture conditions without immediate reactor trip does not present any hazard to the integrity of the core or the RCS with respect to DNBR. The minimum DNBR is never less than the initial value. If the reactor does not trip immediately, the low pressurizer pressure reactor trip will provide protection. This trips the turbine and prevents excess cooldown, which expedites recovery from the incident.

With respect to pressurizer filling, RCS pressure will stabilize well below the RCS pressure safety limit of 2735 psig. The performance of the pressurizer safety valve system and the loads on pressurizer safety valves, associated piping, and supports will be within acceptable limits.

15.5.2 Chemical and Volume Control System Malfunction That Increases Reactor Coolant Inventory

An increase in reactor coolant inventory which results from the addition of cold, unborated water to the reactor coolant system is analyzed in Subsection 15.4.6, chemical and volume control system malfunction that results in a decrease in boron concentration in the reactor coolant. An increase in reactor coolant inventory which results from the injection of highly borated water into the reactor coolant system is analyzed in Subsection 15.5.1, inadvertent operation emergency core cooling system during power operation.

15.5.3 A Number of BWR Transients

(Not applicable to the Byron/Braidwood Stations.)

15.5.4 References

1. Burnett, T. W. T., et al., "LOFTRAN Code Description," WCAP-7907-P-A (Proprietary), WCAP-7907-A (Non-Proprietary), April, 1984.

2. ANS-51.1/Nl8.2-1973, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants."

3. Freidland, A. J. and Ray, S., "Revised Thermal Design Procedure," WCAP-11397-P-A (Proprietary), WCAP-11397-A (Non-Proprietary) April 1989.

4. “Technical Evaluation Report, TMI Action Plan - NUREG-0737 (II.D.1) Braidwood Unit 1 & 2. Docket No. 50-456, 50-457,” G. K. Miller et. al, Idaho National Engineering Laboratory, January 1988.

5. NRC Letter from Mr. Leonard N. Olshan to ComEd Henry E. Bliss, dated August 18, 1988, Subject: NUREG-0737, Item II.D.1, Performance Testing on Relief and Safety Valves for Byron Station, Units 1 and 2 (TAC Nos. 56200 and 63240) transmitting Technical Evaluation Report (TER) providing the results of the NRC’s review on Byron Units 1 and 2 response to NUREG-0737, Item II.D.1.

6. NRC Letter from Mr. George F. Dick (NRR) to Oliver D. Kingsley (Exelon), dated May 4, 2001. Subject: Issuance of Amendments; Increase in Reactor Power, Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2” (TAC Nos. MA9428, MA9429, MA9426 and MA9427).

7. NRC Letter from Mr. George F. Dick (NRR) to Mr. Christopher M. Crane (Exelon), dated August 26, 2004. Subject: Byron Station, Units 1 and 2, and Braidwood Station Units 1 and 2 – Issuance of Amendments, RE: Pressurizer Safety Valve Setpoints, (TAC Nos. MB9762, MB9763, MB9760 and MB9761).

TABLE 15.5-1

TIME SEQUENCE OF EVENTS FOR INCREASE IN REACTOR COOLANT INVENTORY EVENTS

ACCIDENT	EVENT	TIME (sec.)
Inadvertent Actuation of ECCS During Power Operation		
	Spurious SI signal generated; two charging pumps begin injecting borated water	0
	Turbine throttle valve wide open, load begins to drop with steam pressure	51.5
	Low pressurizer pressure reactor trip setpoint reached	74.1
	Control Rod Motion Begins	76.1
	Minimum DNBR occurs	(*)

(*) - DNBR does not decrease below its initial value

15.6 DECREASE IN REACTOR COOLANT INVENTORY

Events which result in a decrease in reactor coolant inventory are discussed in this section, as follows:

a. inadvertent opening of a pressurizer safety or relief valve,

b. failure of small lines carrying primary coolant outside containment,

c. steam generator tube rupture,

d. BWR piping failure outside containment (not applicable), and

e. loss-of-coolant accident resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary.

15.6.1 Inadvertent Opening of a Pressurizer Safety or Relief Valve

15.6.1.1 Identification of Causes and Accident Description

An accidental depressurization of the reactor coolant system could occur as a result of an inadvertent opening of a pressurizer relief or safety valve. Since a safety valve is sized to relieve approximately twice the steam flow-rate of a relief valve, and will therefore allow a much more rapid depressurization upon opening, the most severe core conditions resulting from an accidental depressurization of the reactor coolant system are associated with an inadvertent opening of a pressurizer safety valve. Initially the event results in a rapidly decreasing reactor coolant system pressure that could reach the hot leg saturation pressure if a reactor trip did not occur. The pressure continues to decrease throughout the transient. The effect of the pressure decrease is to decrease power via the moderator density feedback, but the reactor control system (if in the automatic mode) functions to maintain the power and average coolant temperature until reactor trip occurs. Pressurizer level increases initially due to expansion caused by depressurization and then decreases following reactor trip.

The reactor may be tripped by the following reactor protection system signals:

a. overtemperature ΔT and

b. pressurizer low pressure.

An inadvertent opening of a pressurizer safety valve is classified as an ANS Condition II event, a fault of moderate frequency. See Subsection 15.0.1 for a discussion of Condition II events.

15.6.1.2 Analysis of Effects and Consequences

Method of Analysis

The accidental depressurization transient is analyzed by employing the detailed digital computer code LOFTRAN (Reference 1). The code simulates the neutron kinetics, reactor coolant system, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level.

This accident is analyzed with the revised thermal design procedure as described in WCAP-11397-P-A. Plant characteristics and initial conditions are discussed in Subsection 15.0.3.

In order to give conservative results in calculating the DNBR during the transient, the following assumptions are made:

a. Initial reactor power, pressure, and RCS temperatures (consistent with the uprated power conditions) are assumed to be at their nominal values. With the exception of the RCS average temperature bias, which is explicitly modeled in the analysis, uncertainties in initial conditions are included in the limit DNBR as described in WCAP-11397-P-A.

b. A zero moderator temperature coefficient is assumed. The spatial effect of void due to local or subcooled boiling is not considered in the analysis with respect to reactivity feedback or core power shape.

c. A least negative Doppler-only power coefficient is assumed (see Figure 15.0-3) such that the resultant amount of negative feedback is conservatively low in order to maximize any power increase due to moderator reactivity feedback.

D. Cases assuming both D5 and BWI steam generator models at maximum steam generator tube plugging levels with both minimum and maximum feedwater temperatures were analyzed.

Plant systems and equipment credited for mitigating the effects of a reactor coolant system depressurization caused by an inadvertent safety valve opening are discussed in Subsection 15.0.8 and listed in Table 15.0-7.

Normal reactor control systems are not required to function. The reactor protection system functions to trip the reactor on the appropriate signal. No single active failure will prevent the reactor protection system from functioning properly.

Results

The most limiting case (D5 steam generators at the maximum steam generator tube plugging level and minimum feedwater temperature) for an inadvertent opening of a pressurizer safety or relief valve is shown on Figures 15.6-1 and 15.6-2. Figure 15.6-1 illustrates the nuclear power transient following the depressurization. Nuclear power is maintained at the initial value until reactor trip occurs on low pressurizer pressure. The pressure decay transient and average temperature transient following the accident are given in Figure 15.6-2. Pressure drops more rapidly while core heat generation is reduced via the trip, and would then slow once saturation temperature is reached in the hot leg. The DNBR decreases initially, but increases rapidly following the trip, as shown in Figure 15.6-1. The DNBR remains above the limit value throughout the transient.

The calculated sequence of events for the inadvertent opening of a pressurizer safety valve incident is shown on Table 15.6-1.

15.6.1.3 Radiological Consequences

An inadvertent opening of a pressurizer safety or relief valve releases primary coolant to the pressurizer relief tank; however, even assuming a direct release to the containment atmosphere, the radiological consequences of this event would be substantially less than that of a LOCA (Subsection 15.6.5) because less primary coolant is released and the activity is lower as fuel damage is not predicted as a result of this event.

15.6.1.4 Conclusions

The results of the analysis show that the pressurizer low pressure and the overtemperature ΔT reactor protection system signals provide adequate protection against the RCS depressurization event. No fuel or clad damage is predicted for this accident. The radiological consequences of this event would be substantially less than that of the LOCA analyzed in Subsection 15.6.5.

15.6.2 Failure of Small Lines Carrying Primary Coolant Outside Containment

15.6.2.1 Identification of Causes and Accident Description

An assumed CVCS line break outside containment releases primary coolant at a rate of 140 gpm. Primary coolant activities are given in Tables 15.0-9 and 15.0-10. Iodine spiking is taken into consideration.

An accident which results from a break in small sample lines connected to the primary coolant system and penetrating the containment will cause expulsion of the coolant at a rate which can be accommodated by a charging pump. The charging pump would maintain an operational water level in the pressurizer, permitting the operator to conduct an orderly shutdown. The release contains the radionuclide concentration of the primary coolant.

The maximum break size for which the normal makeup system can maintain the pressurizer level is obtained by comparing the calculated flow from the reactor coolant system (RCS) through the postulated break against the charging pump makeup flow at normal RCS pressure, i.e., 2250 psia. A makeup flow rate from one centrifugal charging pump is adequate to sustain pressurizer level and a pressure of 2250 psia for a break through a 0.375 inch diameter hole. This break results in a loss of approximately 17.5 lb/sec, and, due to the use of a 0.245 inch restriction, is the maximum flow available for all reactor coolant sample line breaks outside of the containment. In addition, all such lines meet the requirements of General Design Criterion 55 of Appendix A 10 CFR 50. There are no instrument lines which pass through the containment and connect directly to the RCS. A failure of a small line carrying primary coolant outside containment is classified as an ANS Condition II event, a fault of moderate frequency. See Subsection 15.0.2 for a discussion of Condition II events.

15.6.2.2 Analysis of Effects and Consequences

A break assumed to occur immediately downstream of valve CV8152 would result in low level in the volume control tank if other remedial action were not taken. Fifteen minutes is allowed for the operator to determine what has happened and to close the valve. Spring closure of the valve is assumed to take 5 seconds.

The break of a small sample line does not result in a leakage rate greater than the capacity of a charging pump and pressurizer level does not decrease, normal shutdown procedures can be employed. There are no significant consequences to the reactor or its essential auxiliary systems.

15.6.2.3 Radiological Consequences

A partition factor of 0.1 and a DF of 10 in the building filtration system were applied to the release of radioiodine assuming a CVCS line break outside containment.

Conservative (fifth percentile) values of χ/Q at the exclusion area boundary (Tables 15.0-13 and 15.0-14) were used in the analysis.

The following radiological doses are calculated as a consequence of this accident:

	TWO HOUR DOSE AT EXCLUSION AREA BOUNDARY (rem)	
	WHOLE BODY	THYROID
Byron	0.03	1.0
Braidwood	0.04	1.4

These doses meet the dose acceptance criterion of being a small fraction of the 10 CFR Part 100 dose guidelines where "small fraction" is defined as being 10% (Reference 41).

There are no unusual features associated with the plant design which would prevent the limitation of the radiological consequences of a failure of a small sample line carrying primary coolant outside containment to an acceptable level by utilizing the appropriate limits or reactor coolant activity concentrations or isolation valve closing time and leak rates.

15.6.3 STEAM GENERATOR TUBE RUPTURE

The SGTR analysis was performed to evaluate the two major SGTR potential consequences of concern, specifically:

- margin-to-overfill (MTO) case - the potential for overfilling the ruptured steam generator before the AFW can be isolated and the break flow terminated.

- offsite dose case - the potential to release primary system activity through the secondary side in excess of 10 CFR 50.67 limits.

The background information and results of the SGTR analysis are presented in the following sections.

15.6.3.1 Identification of Causes and Accident Description

The accident examined is the complete severance of a single steam generator tube. The accident is assumed to take place at steady-state reactor power with the reactor coolant system activity at the allowable technical specification operating limit. The accident leads to an increase in secondary system activity due to the leakage of radioactive primary system water into the ruptured steam generator. In the event of a coincident loss of offsite power and subsequent failure of the steam dump system, there is a discharge of activity to the atmosphere through the steam generator safety and/or power-operated relief valves (PORVs).

Since the steam generator tube material is Inconel, a highly ductile material, the assumption of a complete tube severance is conservative. Steam generator operating experience has demonstrated the most probable mode of failure would be one or more minor tube leaks of varying sizes of undetermined origin. An aggressive nondestructive examination policy is implemented during refueling outages to identify failing steam generator tubes and repair those with indications exceeding conservatively established limits. In addition, the steam and power conversion system is continually monitored for activity indicating tube leakage and an accumulation of minor leaks exceeding the technical specification limits is not permitted during operation.

The response of the operator and his ability to implement recovery actions is critical in mitigating the consequence of a SGTR event. The operator actions assumed in these two analyses are based upon the plant-specific emergency procedures addressing the SGTR accident. The emergency procedures are based upon the latest revision of the Westinghouse Owners Group generic emergency response guidelines and are, therefore, consistent with the latest guidelines regarding SGTR mitigation.

The operator action times used in this analysis are based upon the Westinghouse WCAP-10698-P-A SGTR analysis methodology and operator response times observed during initial and requalification operator license simulator training. The operator action times used in the two analyses are shown in Table 15.6-5a.

The MTO analysis assumption time intervals in Table 15.6-5a are validated on the simulator. The operator action times for simulator verification do not include the calculated plant thermal hydraulic response.

A simple validation of the operator action times modeled in the MTO analyses requires that the operators perform each of the actions within the time listed in Table 15.6-5a. However, evaluations can be performed to show that acceptable results continue to be obtained if some of these action times are traded off. For example, performing the earlier actions (e.g. isolating AFW) in a shorter time than analyzed allows for the possibility that the later times (e.g. starting the depressurization) may be increased over the values used in the analyses and still demonstrate acceptable results.

The SGTR offsite dose case does not require operator simulator verification, since the scenario is mitigated by isolating the stuck open SG PORV within 30 minutes and the remaining required operator actions are the same as the MTO case. All of the individual, assumed analysis operator action time intervals for the UFSAR design basis SGTR cases are discussed in detail for reference purposes.

There are six major recovery phases during the SGTR event. The first five phases require timely completion to terminate the primary to secondary tube leakage and ensure overfill does not occur or offsite and Control Room dose limits are not exceeded. These six recovery phases are given below and are discussed in detail later for both the margin-to-overfill and offsite dose case analyses.

1) Identify The Ruptured Steam Generator

The first critical step is for the operator to identify a SGTR exists and determine which steam generator is ruptured. Since the SGTR generates a reactor trip and safety injection, the operator enters the E-0 procedure to verify proper automatic system responses, assess the plant conditions, and determine the appropriate recovery procedure. While in the E-0 procedure, the SGTR identification can be made by several possible indications. High secondary activity on the steam jet air ejector radiation monitor or other secondary monitors would positively identify a SGTR. These monitors would lose direct sampling capability at the time of the loss of offsite power (LOOP), but the recorder readouts and residual meter indication would show above normal secondary activity. For scenarios where the reactor trip occurs some time after the tube rupture, there is a definite possibility the operator will identify the SGTR prior to trip based upon a reduction in feedwater flow to the ruptured steam generator or steam generator level deviation. The main identification factor for analysis purposes is the increase in the ruptured steam generator level and recovery rate compared to the intact steam generators following the reactor trip. This is due to the addition of break flow and auxiliary feedwater (AFW) flow. The identification of the SGTR directs the operator to enter the E-3 SGTR procedure where the remaining recovery actions are performed.

2) Isolation of the Ruptured Steam Generator

The next major step is to isolate the ruptured steam generator. The operator first isolates AFW flow. Isolation is completed by closing the MSIV and MSIV bypass valves after ensuring the PORV and the blowdown valves are closed.

The isolation of the ruptured steam generator provides three functions: 1) it minimizes feedwater accumulation in the steam generator and decreases the potential for overfill, 2) it isolates the ruptured steam generator from the intact steam generators, preventing blowdown and release from the ruptured steam generator when the intact steam generator PORVs are used to cool the RCS, 3) it maximizes the ruptured steam generator pressure, thereby reducing the amount of RCS depressurization needed to terminate the primary to secondary leakage.

3) Cooldown of the Reactor Coolant System

For this step, the RCS must be cooled down below the saturation temperature of the ruptured steam generator. Cooldown would normally be performed by using the steam dumps. However, the LOOP makes the condenser unavailable, and the operator initiates the RCS cooldown by opening the intact steam generator PORVs. The cooldown ensures the RCS remains adequately subcooled when the primary system is depressurized below the ruptured SG pressure.

4) Depressurization of the Reactor Coolant System

As stated previously, the RCS must then be depressurized to equalize pressure with the ruptured steam generator and terminate the primary to secondary leakage. The operator would normally use the pressurizer spray system to depressurize the RCS; however, in the advent of a LOOP, the sprays are not available since the RCPs are tripped. Therefore, the operator opens the pressurizer PORV to decrease the RCS pressure below the ruptured steam generator. The RCS depressurization increases the rate at which SI flow can restore the RCS mass inventory and terminates the primary to secondary leakage.

5) Termination of the Safety Injection Flow

Unless terminated, the SI flow continues repressurizing the RCS until the break flow reinitiates. Therefore, the E-3 procedure specifies that the operator must verify RCS pressure is stable or increasing and then terminate all SI flow except for one centrifugal charging pump. The operator then establishes normal charging flow and adjusts it as necessary to maintain adequate pressurizer level.

6) Place the Reactor in the Cold Shutdown Condition

At this point, the primary to secondary leakage has been terminated and the RCS is stable and ready for transition to the cold shutdown condition. This requires depressurization and cooldown of the ruptured steam generator and cooldown of the RCS to less than 200°F. The three different methods available per the E-3 series procedures to perform the transition to cold shutdown are:

- post-SGTR cooldown using backfill - this method drains the ruptured steam generator through the ruptured tube into the RCS by incrementally decreasing RCS pressure below the steam generator pressure.

- post-SGTR cooldown using blowdown - this method depressurizes the ruptured steam generator by draining it through the blowdown lines.

- post-SGTR cooldown using steam dump - this method depressurizes the ruptured steam generator by dumping steam via the steam dumps to the condenser or via the SG PORV.

The transition to the cold shutdown method used is determined by the TSC personnel based upon available plant equipment and system conditions.

15.6.3.2 Analysis of Effects and Consequences

The analyses for the offsite and Control Room dose and margin-to-overfill cases were performed utilizing the most limiting plant parameters as identified in Table 15.6-5 and the limiting single failures as identified in Table 15.0-15.

The following analysis assumptions were made for both the offsite and Control Room dose and margin-to-overfill transient cases.

- Conservative operator action times for the two cases were used as shown in Table 15.6-5a.
- Loss of offsite power (LOOP) is assumed to occur concurrent with the reactor trip.
- No credit is taken for the chemical and volume control system charging or letdown flow.
- Prior to the reactor trip, the SG level control system maintains the faulted steam generator water level essentially constant, compensating for the additional break flow by reducing the feed flow.
- The operator throttles the AFW flow to the intact SGs as necessary to maintain adequate narrow range level.
- During the RCS depressurization, the operator ensures the pressurizer level stays within the limits specified in the E-3 procedure.

Offsite and Control Room Dose Case

The major sequence of operator actions for the offsite and Control Room dose case is the same as the above summary, but there is one extra operator manipulation required during the isolation of the ruptured SG. The limiting single failure involves the PORV on the ruptured SG failing in the full open position. This causes the SG to be both ruptured and faulted (ruptured/faulted).

The time sequence of events for the offsite and Control Room dose case is shown in Table 15.6-6a and the key transient parameter responses are shown in Figures 15.6-3a, 15.6-3b, and 15.6-3I for Unit 1 and Figures 15.6-3c, 15.6-3d, and 15.6-3J for Unit 2.

Automatic Actions

After the tube rupture occurs, reactor coolant immediately begins flowing from the primary system into the secondary side of the ruptured steam generator causing the RCS pressure to decrease until a reactor trip occurs on overtemperature Delta-T. The reactor trip signal closes the turbine stop valves isolating steam flow to the turbine. The normal feedwater flow to the SGs is also isolated due to the reactor trip.

The LOOP occurs coincident with the reactor trip causing the RCPs to trip and the main condenser to become unavailable when the circulating water pumps are lost.

After the reactor trips, the core power quickly decreases to decay heat levels. The steam dump system cannot be used to dissipate the core decay heat due to the unavailable condenser. Therefore, the secondary pressure increases in the SGs until the PORVs open.

The RCS pressure continues decreasing, and a low pressurizer pressure safety injection (SI) signal is generated. The RCS break flow steadily decreases from time zero as the RCS pressure drops, and it reduces substantially following the reactor trip as the pressure increases in the ruptured SG secondary side, thereby reducing the primary to secondary pressure differential. The RCS pressure then begins increasing again, as the ECCS injection flow exceeds the RCS break flow restoring RCS mass inventory.

The RCS temperature drops after the reactor trip. With the RCPs tripped, the reactor coolant system evolves into the natural circulation cooling mode.

Major Operator Actions

The operator isolates AFW flow to the ruptured SG and then closes the ruptured SG MSIV. The ruptured SG PORV is assumed to fail open at the time the MSIV is closed.

The ruptured SG failed PORV is isolated when the block valve is manually closed thirty minutes after the PORV fails open.

After the isolation of the ruptured SG, the operator initiates the RCS cooldown by opening the intact SG PORVs. The operator continues the cooldown until the RCS temperature is subcooled below the saturation temperature at the ruptured SG pressure.

The operator initiates the RCS depressurization by opening the pressurizer PORV. The operator maintains the pressurizer PORV open until the RCS pressure decreases below the pressure in the ruptured SG. The pressurizer PORV is reclosed and the depressurization terminated.

After the RCS depressurization is completed the operator terminates ECCS flow.

Margin-To-Overfill Case

The sequence of events for the margin-to-overfill case is shown in Table 15.6-6b, and the key parameter transient responses are shown in Figures 15.6-3e and 15.6-3f for Unit 1 and 15.6-3g and 15.6-3h for Unit 2.

For the tube rupture event, the overtemperature ΔT reactor trip setpoint is reached earlier than the low pressurizer pressure reactor trip setpoint. Since the time from tube rupture to auxiliary feedwater isolation is an assumed operator action time, an early reactor trip results in auxiliary feedwater going to the ruptured steam generator for a longer period of time which leads to a reduction in margin-to-overfill.

Automatic Actions

The initial sequence for the MTO case is similar to the OD case as the tube rupture occurs at time, t = 0, and the primary to secondary break flow initiates. The reactor trips on overtemperature ΔT. The reactor trip signal closes the turbine stop valves isolating steam flow to the turbine.

The loss of offsite power occurs coincident with the reactor trip, resulting in the RCPs tripping and the condenser becoming unavailable due to the loss of the circulating water pumps. The loop also results in a trip of the main feedwater pumps.

The auxiliary feedwater pumps automatically start on loss-of-offsite power, and the auxiliary feedwater flow immediately begins entering the steam generators. The RCS pressure continues decreasing and initiates a safety injection signal on low pressurizer pressure.

After the trip, the core power quickly decreases to decay heat levels. However, since the steam dump system cannot be used to dissipate the core decay heat due to the unavailable condenser, the secondary pressure increases until the ruptured SG PORV setpoint is reached. The valve relieves pressure produced by the decay heat and the rising liquid level (see Figures 15.6-3f and 15.6-3h).

The liquid volume in the steam generator with the tube rupture begins increasing immediately after the reactor trip and continues to increase up to the termination of the transient.

Major Operator Actions

The operator is able to identify and isolate AFW flow to the ruptured SG within 9 minutes after tube rupture occurs. The ruptured SG water level continues increasing after isolation due to break flow, but at a reduced rate.

The operator then initiates the RCS cooldown by opening the intact SG PORVs. Since one of the intact SG PORVs fails to open, there is a resulting reduced cooldown rate and a longer time to reach the E-3 procedure RCS subcooling requirement. The operator terminates the RCS cooldown after meeting the subcooling requirement.

The operator initiates the RCS depressurization by opening the pressurizer PORV.

After termination of the RCS depressurization, the operator halts the ECCS injection flow and the break flow begins steadily decreasing the RCS pressure. Detailed operator actions involved in termination of ECCS flow and subsequent actions to reduce RCS pressure and terminate break flow are not modeled in the analysis. This leads to a conservatively long period of continued break flow after ECCS termination is modeled before the RCS and ruptured SG pressures equalize.

The SGTR Margin to Overfill transient is terminated when the RCS pressure is equalized with the ruptured SG.

Results

In the SGTR offsite and Control Room dose case there is no concern for steam generator overfill since the failed open PORV reduces the secondary mass and results in a much lower SG level at termination. At the time the SGTR margin-to-overfill case is terminated, the ruptured SG still contains more than 50 cubic feet and more than 250 cubic feet of gas volume margin for Unit 1 and Unit 2, respectively. This demonstrates there are sufficient indications and controls which provide the operator adequate time to terminate the break flow before steam generator overfill occurs.

15.6.3.3 Radiological Consequences

It was determined that the doses resulting from a SGTR at Unit 2 are higher than those from a SGTR at Unit 1. The Unit 2 analysis is presented.

The key inputs and assumptions used in the AST Steam Generator Tube Rupture (SGTR) radiological consequence analysis are summarized below and provided in Table 15.6-6.

The SGTR accident is postulated as a complete severance of a single Steam Generator (SG) tube. This is a conservative assumption because tube material is Inconel, a highly ductile metal alloy, and the most probable mode of failure would be one or more minor tube leaks of varying sizes and undetermined origin.

The tube rupture results in the release of radioactive material from the Containment system. For the three intact SGs, primary to secondary coolant leakage continues to transfer activity into the Secondary Coolant side. This makes it available for release into the environment via steaming through the SG Power Operated Release Valves (PORV). For the SG with the ruptured tube, referred to as the ruptured SG, coolant release will take two forms:

- Break Flow - un-flashed release of RCS coolant directly into the secondary loop, and made available for steaming release to the environment through the SG PORV.

- Flashed Break Flow - RCS coolant that flashes directly to steam when released from the ruptured tube, and is sent through the SG PORV to the environment.

SGTR accident mitigation can be described in recovery phases. The major phases that this analysis uses to model the dose consequences from this event are shown in Table 15.6-6, along with the approximate time increments that are associated with the sequence of events:

Consistent with RG 1.183, two reactor transients that maximize the radioactivity available for release were modeled.

Case 1: Dose Due to Pre-accident Iodine Spike

The first case involves a 60 µCi/gm pre-accident Iodine spike. This 60-µCi/gm spike is consistent with the Technical Specification operational Reactor Coolant System (RCS) activity concentration limit for an assumed spike. In this scenario, it is assumed that all of the spike activity is homogeneously mixed in the primary coolant, prior to accident initiation.

Case 2: Dose Due to Accident Initiated Concurrent Iodine Spike

The second case involves an accident initiated iodine spike that occurs concurrently with the release of fluid from the primary and secondary coolant systems. Regulatory guidance specifies that this spike should result in a release rate from defective fuel that is 335 times the normal rate, and lasts for an 8-hour duration.

Fuel Damage and Core Source Term

The design basis assumes no fuel damage for the postulated steam generator tube rupture event. For this SGTR accident, the source terms are defined by the Technical Specification Dose Equivalent (DE) I-131 iodine and DE Xe-133 noble gas activity in the primary reactor coolant system. Because no fuel damage is assumed for this accident, only iodine and noble gas isotopes are modeled to contribute to dose. To identify the worst-case SGTR accident, however, the two different cases of iodine spiking described above are analyzed, per regulatory guidance.

In addition, for both cases, a 0.1 μCi/gm DE I-131 equilibrium secondary coolant activity concentration limit from Technical Specifications, is included.

Activity Removal Mechanisms in Containment

The design basis SGTR releases activity directly into the primary RCS, therefore no plateout, or other activity deposition, is credited.

Decay Credited:

Decay of radioactivity is credited, prior to release to the environment.

Depletion from Leakage Credited:

For analyses of doses due to release from the RCS volume, the dose results from leakage. It is reasonable to credit the small amount of depletion from the available RCS activity inventory associated with this leakage. This is calculated inherently by the RADTRAD code.

Release Rates and Partitioning Factors:

As discussed above, a number of modes of release are indicative of this particular accident scenario. Therefore, the varying releases associated with the timing and sequence of events of this accident was derived.

Activity that originates in the primary RCS is released to the secondary coolant by means of the primary-to-secondary coolant leak rate. This design basis leak rate value is 0.218 gpm, per intact SG, totaling 0.654 gpm. This volumetric flow rate is converted to mass flow using the assumption of cooled liquid conditions (i.e., 62.4 lbm/ft^3), as specified by RG 1.183.

The methodology used to model steaming of activity through SG PORVs following the postulated SGTR event assumes an average cumulative release rate through the SG valves. The partitioning factors are applied to these release rates. For the time increments used in this accident scenario, release rates were derived by taking the averages of these rates over each specified time increment.

The ruptured/faulted steam generator sees two simultaneous release mechanisms. Primary to secondary coolant leakage through the ruptured tube that flashes conservatively goes directly to the environment, without mixing with any secondary coolant. Therefore, with this release mechanism, no partitioning of iodine is expected to occur in this release. However, leakage that does mix with the volume of coolant in the ruptured SG is released by flashing to the environment, and the applicable partition factor is applied, as discussed below.

For all post-accident releases through the SG PORVs, the mechanism for release to the environment is steaming of the coolant in the secondary system. Because of this release dynamic, RG 1.183 allows for a reduction in the amount of activity released to the environment based on partitioning of nuclides between the liquid and gas states of water. For Iodine, the partitioning factor of 0.01 was taken directly from the suggested guidance of RG 1.183. Reviewing the specified AST release fractions, it is concluded that the only nuclides to be released from the core source term, other than iodines, are noble gas nuclides, and because of the volatility of noble gases, no partitioning is assumed for any such isotopes.

In addition to the steam released through the SG PORVs, the steam release through the Condenser until the time of Reactor trip and loss of offsite power is also accounted for. For steam flow that is released through the Condenser an additional 0.01 factor is applied to model partitioning in this pathway.

χ/Q Calculations (Meteorology)

Releases from the SG PORVs are considered elevated releases due to the high steaming rates, and the associated χ/Qs have been reduced by a factor of 5, per guidance in RG 1.194, as described in Section 15.4.8.3.

This event was reanalyzed using the finer wind speed categories of Regulatory Guide 1.23, Revision 1.

Assumptions and Inputs

The following inputs and assumptions were used in the SGTR analysis.

a. There is no fuel damage as a result of the postulated steam generator tube rupture accident.

b. In the case of a postulated iodine activity release rate spike, the spike release is assumed to occur for a period of 8 hours.

c. The activity released from the fuel is assumed to be instantaneously mixed with the RCS.

d. All iodine released from the SGs is modeled consistent with the RG 1.183 specification of 97% elemental and 3% organic.

e. The Control Room HVAC system is realigned to the emergency mode of operation 30 minutes after the initiation of this design basis accident.

f. The break flow modeled in the dose analysis is shown in Figure 15.6-3C.

g. The break flashing fraction modeled in the dose analysis is shown in Figure 15.6-3J.

h. The PORV steam releases until break flow termination modeled in the dose analysis are shown in Figure 15.6-3D.

i. In addition to PORV steam releases, the full power steam release through the Condenser until the time of reactor trip and loss of offsite power is accounted for.

Dose Results

Radiological doses resulting from a design basis SGTR for a control room operator and a person located at EAB or LPZ are to be less than the regulatory dose limits as given below.

Regulatory Dose Limits - SGTR

	Dose Type	Control Room (rem)	EAB and LPZ (rem)
Case 1	TEDE Dose	5[a]	25[b]
Case 2	TEDE Dose	5[a]	2.5[b]

Notes:
[a] 10 CFR 50.67
[b] 10 CFR 50.67 as modified by Regulatory Guide 1.183 (Table 6, Page 1.183-20)

The table below provides the results.

Steam Generator Tube Rupture Accident
Radiological Analysis Results

Case 1: Pre-Accident 60 µCi/gm DE I-131 Spike Dose Assessment Results		
Control Room (rem TEDE)	EAB (rem TEDE)	LPZ (rem TEDE)
2.0	3.7	0.69
Case 2: Accident Initiated 335 times Equilibrium Iodine Release Rate Spike Dose Assessment Results		
Control Room (rem TEDE)	EAB (rem TEDE)	LPZ (rem TEDE)
0.56	2.1	0.41

These doses are below the Regulatory Dose Limits, so it is verified that this design basis Steam Generator Tube Rupture accident is sufficiently mitigated at both Byron and Braidwood Stations.

15.6.3.4 Conclusions

A steam generator tube rupture will cause no subsequent damage to the reactor coolant system or the reactor core. An orderly recovery from the accident can be completed before steam generator overfill occurs even assuming simultaneous loss of offsite power. The radiological consequences are within the 10 CFR 50.67 and RG 1.183 guidelines.

15.6.4 Spectrum of BWR Steam System Piping Failures Outside of Containment

This section is not applicable to the Byron/Braidwood Stations.

15.6.5 Loss of Coolant Accidents Resulting From a Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary

A comprehensive safety analysis of postulated pipe breaks within the reactor coolant system (RCS) boundary has been performed. This analysis has included cases of the loss-of-coolant accident (LOCA) resulting from a broad spectrum of small and large pipe breaks up to and including the double-ended break of the largest RCS pipe.

The objective of the analysis was to determine the conditions of the RCS, core, and containment in the event of a postulated LOCA and to demonstrate that the emergency core cooling system (ECCS) has the capability to mitigate each LOCA.

For the analysis reported here, a major pipe break (large break) is defined as a break with a total cross-sectional area equal to or greater than 1.0 ft^2. This event is considered an ANS Condition IV event, a limiting fault that is not expected to occur during the lifetime of the plant, but is postulated as a conservative design basis.

A minor pipe break (small break) is defined as a break of the reactor coolant pressure boundary with a total cross sectional area less than 1.0 ft^2. This event is considered an ANS Condition III event, an infrequent event fault which may occur during the life of the plant.

15.6.5.1 General

15.6.5.1.1 Performance Criteria for Emergency Core Cooling System

The reactor is designed to withstand thermal effects caused by a loss-of-coolant accident including the double-ended severance of the largest reactor coolant system pipe. The reactor core and internals together with the emergency core cooling system are designed so that the reactor can be safely shut down and the essential heat transfer geometry of the core is preserved following the accident. The emergency core cooling system, even when operating during the injection mode with the most severe single active failure, is designed to meet the requirements of 10 CFR 50.46 (Reference 2). The requirements are:

a. The calculated maximum fuel element cladding temperature shall not exceed 2200°F.

b. The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.

c. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

d. Calculated changes in core geometry shall be such that the core remains capable of being cooled.

e. After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptable low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

For the Best Estimate LB LOCA analysis, it is noted that criteria a through c above are satisfied by ensuring that there is a high level of probability that when uncertainties in the analysis method and inputs are accounted for, the criteria are not exceeded.

15.6.5.1.2 Accident Description

A LOCA would result from a break of the RCS piping or of any line connected to that system up to the first closed valve. The charging pumps have the capability to make up for leakage resulting from ruptures of a small cross section, thus permitting an orderly shutdown. The coolant released would remain in the containment.

For a postulated break, reactor trip is initiated when the pressurizer low pressure setpoint is reached while the safety injection system (SI) signal is actuated by pressurizer low pressure or containment high pressure. The consequences of the accident are limited in two ways:

a. Reactor trip and borated water injection supplement void formation in causing rapid reduction of the nuclear power to a residual level corresponding to the delayed fission product decay; and

b. Injection of borated water ensures sufficient flooding of the core to prevent excessive temperatures.

Before the reactor trip occurs, the reactor is in a equilibrium condition, i.e., the heat generated in the core is being removed via the secondary system. After reactor trip and turbine trip, core heat and heat from hot internals and the vessel is transferred to the RCS fluid and then to the secondary system. The secondary system pressure increases and steam dump may occur.

Makeup to the secondary side is provided automatically by the auxiliary feedwater pumps. The SI signal stops normal feedwater flow by closing the main feedwater line isolation valves and initiates auxiliary feedwater flow by starting the auxiliary feedwater pumps. If offsite power is available, the steam is dumped to the condenser; if not, the steam is dumped to the atmosphere. The secondary flow aids in the reduction of RCS pressure. When the RCS pressure falls below 600 psia, the accumulators begin to inject borated water.

Reactor coolant pump (RCP) operation is assumed as follows. For small break LOCA analysis, the RCPs are assumed to trip, with attendant time delay, at the time of loss of offsite power, which is assumed to occur simultaneously with low pressurizer pressure reactor trip. Large break LOCA analyses cases for both with and without loss of offsite power are performed. Large break LOCA cases with loss of offsite power assume RCP trip at the transient initiation, with no time delay, and the effects of RCP coastdown are included in the blowdown analysis. Large break LOCA cases without loss of offsite power assume powered RCP operation throughout the transient.

15.6.5.2 Thermal Analysis

The analysis specified by 10 CFR 50.46 is presented in this subsection. The time sequence of events for the LOCA analysis is provided in Tables 15.6-1a, 15.6-1b, 15.6-1c, 15.6-1d and 15.6-1e. The results of the loss-of-coolant accident analysis are shown in Tables 15.6-3, 15.6-3a, 15.6-4, 15.6-4a, 15.6-4b and 15.6-4c and show compliance with acceptance criteria.

The SBLOCA analysis is based on reactor conditions shown in Table 15.6-2. The detailed description of the SBLOCA analysis methodology is given in References 11, 13, and 34. These documents describe the major phenomena modeled, the interfaces among the computer codes, and the features of the codes which ensure compliance with the acceptance criteria. The NOTRUMP (References 11 and 13) and LOCTA-IV (Reference 8) codes are used to assess the core heat transfer geometry and to determine if the core remains capable of being cooled throughout the SBLOCA. The input assumptions in the Best Estimate large break analysis are summarized in Table 15.6-2a and 15.6-3b, as discussed in further detail in Section 15.6.5.2.1.3. The codes, interfaces, and modeling consideration associated with the Best Estimate large break analysis are described in detail in Section 15.6.5.2.1.4. The conditions of Table 15.6-2 reflect four-loop operation. Containment parameters used in the large break analysis are given in Section 6.2.1.5.

The method of analysis to determine peak clad temperature is divided into two types of analysis: 1) large break LOCA, and 2) small break LOCA. The methods of analysis for large and small break LOCA are described below and results are given.

15.6.5.2.1 Large Break Analysis

15.6.5.2.1.1 General

When the Final Acceptance Criteria (FAC) governing the loss-of-coolant accident (LOCA) for Light Water Reactors was issued in Appendix K of 10 CFR 50.46, both the Nuclear Regulatory Commission (NRC) and the industry recognized that the stipulations of Appendix K were highly conservative. That is, using the then accepted analysis methods, the performance of the Emergency Core Cooling System (ECCS) would be conservatively underestimated, resulting in predicted Peak Clad Temperatures (PCTs) much higher than expected. At that time, however, the degree of conservatism in the analysis could not be quantified. As a result, the NRC began a large-scale confirmatory research program with the following objectives:

1) Identify, through separate effects and integral effects experiments, the degree of conservatism in those models permitted in the Appendix K rule. In this fashion, those areas in which a purposely prescriptive approach was used in the Appendix K rule could be quantified with additional data so that a less prescriptive future approach might be allowed.

2) Develop improved thermal-hydraulic computer codes and models so that more accurate and realistic accident analysis calculations could be performed. The purpose of this research was to develop an accurate predictive capability so that the uncertainties in the ECCS performance and the degree of conservatism with respect to the Appendix K limits could be quantified.

Since that time, the NRC and the nuclear industry have sponsored reactor safety research programs directed at meeting the above two objectives. The overall results have quantified the conservatism in the Appendix K rule for LOCA analyses and confirmed that some relaxation of the rule can be made without a loss in safety to the public. It was also found that some plants were being restricted in operating flexibility by the overly conservative Appendix K requirements. In recognition of the Appendix K conservatism that was being quantified by the research programs, the NRC adopted an interim approach for evaluation methods. This interim approach is described in SECY-83-472 (Reference 5). The SECY-83-472 approach retained those features of Appendix K that were legal requirements, but permitted applicants to use best-estimate thermal-hydraulic models in their ECCS evaluation model. Thus, SECY-83-472 represented an important step in basing licensing decisions on realistic calculations, as opposed to those calculations prescribed by Appendix K.

In 1998, the NRC Staff amended the requirements of 10 CFR 50.46 and Appendix K, "ECCS Evaluation Models" (Reference 2), to permit the use of a realistic evaluation model to analyze the performance of the ECCS during a hypothetical LOCA. This decision was based on an improved understanding of LOCA thermal-hydraulic phenomena gained by extensive research programs. Under the amended rules, best-estimate thermal-hydraulic models may be used in place of models with Appendix K features. The rule change also requires, as part of the LOCA analysis, an assessment of the uncertainty of the best-estimate calculations. It further requires that this analysis uncertainty be included when comparing the results of the calculations to the prescribed acceptance criteria of 10 CFR 50.46. Further guidance for the use of best-estimate codes is provided in Regulatory Guide 1.157 (Reference 37).

To demonstrate use of the revised ECCS rule, the NRC and its consultants developed a method called the Code Scaling, Applicability, and Uncertainty (CSAU) evaluation methodology (Reference 38). This method outlined an approach for defining and qualifying a best-estimate thermal-hydraulic code and quantifying the uncertainties in a LOCA analysis.

A LOCA evaluation methodology for three- and four-loop Pressurized Water Reactor (PWR) plants based on the revised 10 CFR 50.46 rules was developed by Westinghouse with the support of EPRI and Consolidated Edison and has been approved by the NRC (Reference 36).

More recently, Westinghouse developed an alternative uncertainty methodology called ASTRUM, which stands for Automated Statistical TReatment of Uncertainty Method (Reference 4). This method is still based on the CQD methodology and follows the steps in the CSAU methodology (Reference 38). However, the uncertainty analysis (Element 3 in the CSAU) is replaced by a technique based on order statistics. The ASTRUM methodology replaces the response surface technique with a statistical sampling method where the uncertainty parameters are simultaneously sampled for each case. The ASTRUM methodology has received NRC approval for referencing in licensing calculations in Reference 4.

The three 10 CFR 50.46 criteria (peak clad temperature, maximum local oxidation, and core-wide oxidation) are satisfied by running a sufficient number of WCOBRA/TRAC calculations (sample size). In particular, the statistical theory predicts that 124 calculations are required to simultaneously bound the 95th percentile values of three parameters with a 95-percent confidence level.

The Byron/Braidwood analysis is in accordance with the applicability limits and usage conditions defined in Section 13-3 of Reference 4, as applicable to the ASTRUM methodology. Section 13-3 of Reference 4 was found to acceptably disposition each of the identified conditions and limitations related to WCOBRA/TRAC and the CQD uncertainty approach per Section 4.0 of the ASTRUM Final Safety Evaluation Report.

Two analytical models for the Byron/Braidwood Stations were developed for the best estimate large break LOCA analyses. One analysis (Unit 1) utilized the BWI Replacement Steam Generator geometry and the other (Unit 2) utilized the Westinghouse D-5 Steam Generator Geometry. The remaining analysis parameters were identical. Very slight differences were observed in the vessel design and these differences were blended together in a generally conservative fashion to create one common vessel model as characterized in detail in Section 3 of Reference 40.

Input parameters used for the Byron/Braidwood minimum containment pressure analysis are presented in Section 6.2.1.5. Mass and energy releases from the WCOBRA/TRAC Reference transient were utilized to confirm the containment back pressure utilized in the Reference Transient as described in detail in Section 6.2.1.5.

15.6.5.2.1.2 Design Basis Accident

The Byron/Braidwood PCT-limiting transients are double-ended cold leg guillotine breaks which analyze conditions that fall within those listed in Table 15.6-3b. Traditionally, cold leg breaks have been limiting for large break LOCA. This location is the one where flow stagnation in the core appears most likely to occur. Scoping studies with WCOBRA/TRAC have confirmed that the cold leg remains the limiting break location (Reference 36). The design basis accident is described in more detail in Section 15.6.5.2.3.1.

15.6.5.2.1.3 Analysis Assumptions

The expected PCT and its uncertainty developed are valid for a range of plant operating conditions. The range of variation of the operating parameters has been accounted for in the uncertainty evaluation. Table 15.6-3b summarizes the operating ranges for Byron/Braidwood as defined for the proposed operating conditions, which are supported by the Best-Estimate LBLOCA analysis. Section 6.2.1.5 summarizes the LBLOCA containment data used for calculating containment pressure. If operation is maintained within these ranges, the LBLOCA results using WCOBRA/TRAC are considered to be valid. Note that some of these parameters vary over their range during normal operation (accumulator temperature) and other ranges are fixed for a given operational condition (T_{avg}) (see Table 15.6-2a).

15.6.5.2.1.4 Method of Analysis. Large Break

The methods used in the application of WCOBRA/TRAC to the large break LOCA with ASTRUM are described in Reference 36 and Reference 4. A detailed assessment of the computer code WCOBRA/TRAC was made through comparisons to experimental data. These assessments were used to develop quantitative estimates of the code's ability to predict key physical phenomena in a PWR large break LOCA. Modeling of a PWR introduces additional uncertainties which are identified and quantified in the plant-specific analysis. WCOBRA/TRAC MOD7A was used for the execution of ASTRUM for Byron/Braidwood.

WCOBRA/TRAC combines two-fluid, three-field, multi-dimensional fluid equations used in the vessel with one-dimensional drift-flux equations used in the loops to allow a complete and detailed simulation of a PWR. This best-estimate computer code contains the following features:

1) Ability to model transient three-dimensional flows in different geometries inside the vessel.

2) Ability to model thermal and mechanical non-equilibrium between phases

3) Ability to mechanistically represent interfacial heat, mass,

and momentum transfer in different flow regimes.

4) Ability to represent important reactor components such as fuel rods, steam generators, reactor coolant pumps, etc.

A typical calculation using WCOBRA/TRAC begins with the establishment of a steady-state, initial condition with all loops intact. Following the establishment of an acceptable steady-state condition, the transient calculation is initiated by introducing a break into one of the loops. The evolution of the transient through blowdown, refill, and reflood proceeds continuously, using the same computer code (WCOBRA/TRAC) and the same modeling assumptions. Containment pressure is modeled with the BREAK component using a time dependent pressure table. Containment pressure is calculated using the COCO code (Reference 7) and mass and energy releases from the WCOBRA/TRAC calculation.

The final step of the best-estimate methodology, in which all uncertainties of the LOCA parameters are accounted for to estimate a PCT, Local Maximum Oxidation (LMO), and CoreWide Oxidation (CWO) at 95-percent probability, is described in the following sections.

1) Plant Model Development:

   In this step, a WCOBRA/TRAC model of the plant is developed. A high level of nodding detail is used in order to provide an accurate simulation of the transient. However, specific guidelines are followed to ensure that the model is consistent with models used in the code validation. This results in a high level of consistency among plant models, except for specific areas dictated by hardware differences, such as in the upper plenum of the reactor vessel or the ECCS injection configuration.

2) Determination of Plant Operating Conditions:

   In this step, the expected or desired operating range of the plant to which the analysis applies is established. The parameters considered are based on a "key LOCA parameters" list that was developed as part of the methodology. A set of these parameters, at mostly nominal values, is chosen for input as initial conditions to the plant model. A transient is run utilizing these parameters and is known as the "initial transient".

   The list of these parameters is given in Table 15.6-2a. Next, several confirmatory runs are made, which vary a subset of the key LOCA parameters over their expected operating range in one-at-a-time sensitivities. Because certain parameters are not included in the uncertainty analysis, these parameters are set at their bounding condition. This analysis is commonly referred to as the confirmatory analysis. The most limiting input conditions, based on these confirmatory runs, are then combined into the model that will represent the limiting state for the plant, which is the starting point for the assessment of uncertainties. The results of the confirmatory analyses are given in Table 15.6-3.

3) Assessment of Uncertainty:The ASTRUM methodology is based on order statistics. The technical basis of the order statistics is described in Section 11 of Reference 4. The determination of the PCT uncertainty, LMO uncertainty, and CWO uncertainty relies on a statistical sampling technique. According to the statistical theory, 124 WCOBRA/TRAC calculations are necessary to assess against the three 10 CFR 50.46 criteria (PCT, LMO, CWO).

   The uncertainty contributors are sampled randomly from their respective distributions for each of the WCOBRA/TRAC calculations. The list of uncertainty parameters, which are randomly sampled for each time in the cycle, break type (split or double-ended guillotine), and break size for the split break are also sampled as uncertainty contributors within the ASTRUM methodology.

   Results from the 124 calculations are tallied by ranking the PCT from highest to lowest. A similar procedure is repeated for LMO and CWO. The highest rank of PCT, LMO, and CWO will bound 95 percent of their respective populations with 95-percent confidence level.

4) Plant Operating Range:

   The plant operating range over which the uncertainty evaluation applies is defined. Depending on the results obtained in the above uncertainty evaluation, this range may be the desired range or may be narrower for some parameters to gain additional margin.

   Note that the LBLOCA analysis was performed with ZIRLO® cladding. However, Reference 6 concluded that the LOCA ZIRLO cladding models are acceptable for application to Optimized ZIRLO™ cladding in the Large Break analysis, and that no additional calculations are necessary for evaluating the use of Optimized ZIRLO cladding provided that plant specific ZIRLO calculations were previously performed.

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15.6.5.2.2 Small Break LOCA Analysis

15.6.5.2.2.1 Description of Analysis and Assumptions

The small break LOCA analysis was performed using the Westinghouse ECCS small break evaluation model (Reference 12) which utilizes the NOTRUMP (References 11 and 13) and LOCTA-IV (Reference 8) computer codes. Figure 15.6-5 shows the interaction of the computer codes used to evaluate the small break cases. Two complete break spectrums were analyzed, one for Unit 1 (1.5, 2, 3, and 4-inch diameter) and one for Unit 2 (1.5, 2, 3, and 4-inch diameter). These spectrums were selected in order to ensure that the limiting PCT is bounded for the small break LOCA event.

In addition to the description of the NOTRUMP code in References 11 and 13, safety injection was explicitly modeled in the broken loop. Figure 15.6-13b presents the broken loop safety injection flow from one safety injection line for Unit 1 and 2. The charging pump line discharges to the intact loop at RCS pressure, consistent with the implemented COSI methodology (Reference 34).

The core power level utilized in the small break LOCA analysis was 3659 MWt. Since accumulator water volume is not a significant parameter in small beak LOCA analyses, the large break assumption was employed. The core power and peaking factors used in the analyses are also given in Table 15.6-2. Additionally the main steam safety valves were set 5% above the Technical Specifications setpoint value and required an additional 3% accumulation before being fully open.

Figure 15.6-13c presents the hot rod power shape utilized to perform the small break analysis. This actual power shape was chosen because it provides that distribution of power versus core height which will maximize peak clad temperature given an upper limit of +13% for the axial flux difference.

Reference 12 covered a range of break locations for the small break LOCA. This study determined that the bottom of the cold leg was the most limiting location. Therefore, only the spectrum of cold leg breaks covering the range of 1.5-, 2-, 3-, and 4-inch diameter breaks for Unit 1 and Unit 2 were analyzed in order to determine the most limiting break size. Once the limiting break size was determined, the range of RCS operating temperatures was analyzed. The operating range of temperatures is the nominal T_{avg} of 588.0°F plus uncertainties to the reduced T_{avg} of 575.0°F minus uncertainties. These breaks were analyzed following the method presented in Subsection 15.6.5.2.2.2.

The limiting single failure in the small break LOCA analysis is the loss of one diesel, which results in the loss of one train of SI pumps under loss of offsite power.

The limiting conditions of burnup for small break LOCA need to be specifically determined due to the effects of burst upon the small break LOCA peak clad temperature. For peak clad temperatures (PCTs) less than 1700°F, a beginning-of-life (BOL) burnup is assumed. No burnup spectrum study is performed since the burst PCT at higher burnups will not exceed BOL PCT. If the PCT exceeds 1700°F, a burnup study is performed to obtain the limiting time in life for rod burst. For Unit 1 and Unit 2, a burnup study was not performed since the PCT was less than 1700°F (Unit 1 - 1624°F, Unit 2 - 1627°F).

Figures 15.6-13a and 15.6-13b provide the intact and broken loop safety injection (total ECCS) flows assumed in the small break LOCA analysis. Consistent with the implemented COSI model (Reference 34), the intermediate safety injection and charging lines for Unit 1 and Unit 2 discharge to the broken and intact loop at RCS pressure. The safety injection flows also incorporate a 5% flow reduction for future pump degradation.

The small break LOCA analysis was performed with the assumptions appearing on Table 15.6-2. The analysis assumed operation at a vessel thermal design flow rate of 368,000 gpm, 5% steam generator tube plugging for Unit 1 and 10% steam generator tube plugging for Unit 2, and a 40-second delay in delivery of pumped ECCS assuming loss of offsite power. Finally, to address the flexibility of a range of RCS operating temperatures, a sensitivity was performed to compare reduced RCS operating temperature to nominal RCS operating temperature. Table 15.6-2 identifies the nominal RCS operating temperature (upper range), and the reduced RCS operating temperature (lower range) supported by the small break LOCA analysis.

Note that the SBLOCA analysis was performed with ZIRLO cladding. However, Reference 6 concluded that the LOCA ZIRLO cladding models are acceptable for application to Optimized ZIRLO™ cladding in the Small Break analysis, and that no additional calculations are necessary for evaluating the use of Optimized ZIRLO cladding provided that plant specific ZIRLO calculations were previously performed.

15.6.5.2.2.2 Method of Analysis, Small Break

The NOTRUMP and LOCTA-IV computer codes are used to perform the analysis of loss-of-coolant accidents due to small breaks in the reactor coolant system (RCS). The NOTRUMP computer code, approved for this use by the Nuclear Regulatory Commission, is used to calculate the transient depressurization of the RCS as well as to describe the mass and enthalpy of the flow through the reactor core and break. This code is a state-of-the-art one-dimensional general network code incorporating a number of advanced features. Among these new features are the utilization of a nonequilibrium thermal calculation in all fluid volumes, flow regime-dependent drift flux calculations with counter-current flooding limitations, mixture level tracking logic in multiple-stack fluid nodes, and regime-dependent heat transfer correlations. The NOTRUMP small break LOCA emergency core cooling system evaluation model was developed to determine the RCS response to design basis small break LOCAs and to address the NRC concerns expressed in NUREG-0611, "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse-Designed Operating Plants."

In NOTRUMP, the RCS is subdivided into fluid-filled control volumes (fluid nodes) and metal nodes interconnected by flowpaths and heat transfer links. The transient behavior of the system is determined from the governing conservation equations of mass, energy, and momentum applied to these nodes. The broken loop is modeled explicitly, and the intact loops are lumped into a second loop. A detailed description of the NOTRUMP code is provided in References 11 and 13.

In the NOTRUMP model, the reactor core is represented as a vertical stack of heated control volumes with an associated bubble rise model to permit a transient mixture height calculation. The multi-node capability of the program enables the explicit and detailed spatial representation of various system components. In particular, it enables a proper calculation of the behavior of the loop seal during a loss-of-coolant accident.

The COSI condensation model was used in the NOTRUMP portion of the Unit 1 and Unit 2 small break LOCA analyses. As explained in Reference 34, the COSI model is an improved condensation model based on data that modeled the configuration of the SI piping to the RCS cold leg. With this model, improved condensation of steam in the intact loops results in lower RCS pressure and larger SI flow rates. Further, increased condensation of steam by the SI water in the intact cold legs results in additional warming of the SI water prior to reaching the core. Warmer water entering the core results in increased steaming in the core, increased mixture void fraction, and increased mixture level. Additionally, improvements in condensation in the broken loop by the broken loop SI can further decrease RCS pressure and may partially or completely offset any "plugging" effect on the break.

Clad thermal analyses are performed with the LOCTA-IV computer code which uses as input the RCS pressure, fuel rod power history, steam flow past the uncovered part of the core, and mixture height history from the NOTRUMP hydraulic calculations as input. For all computations, the NOTRUMP and LOCTA-IV calculations were terminated slightly after the time the core mixture level returned to the top of the core following core uncovery.

A schematic representation of the computer code interfaces is given in Figure 15.6-5

15.6.5.2.3 Results

15.6.5.2.3.1 Results of Large Break LOCA Analysis

The large break LOCA transient can be divided into convenient time periods in which specific phenomena occur, such as various hot assembly heatup and cooldown transients. For a typical large break, the blowdown period can be divided into the Critical Heat Flux (CHF) phase, the upward core flow phase, and the downward core flow phase. These are followed by the refill, reflood, and long-term cooling periods. Specific important transient phenomena and heat transfer regimes are discussed below, with the transient results shown in Figures 15.6-8a through 15.6-8m (Unit 1) and 15.6-9a through 15.6-9m (Unit 2). (The PCT-limiting case was chosen to show a conservative representation of the response to a large break LOCA.)

1) Critical Heat Flux (CHF) Phase:

   Immediately following the cold leg rupture, the break discharge rate is subcooled and high (Figures 15.6-8b, -8c and Figure 15.6-9b, -9c). The regions of the RCS with the highest initial temperatures (core, upper plenum, upper head, and hot legs) begin to flash to steam, the core flow reverses and the fuel rods begin to go through departure from nucleate boiling (DNB). The fuel cladding rapidly heats up (Figures 15.6-8a and 9a) while the core power shuts down due to voiding in the core. This phase is terminated when the water in the lower plenum and downcomer begins to flash (Figures 15.6-8g, 8k and Figures 15.6-9g, 9k respectively). The mixture swells and intact loop pumps, still rotating in single-phase liquid, push this two-phase mixture into the core.

2) Upward Core Flow Phase:

   Heat transfer is improved as the two-phase mixture is pushed into the core. This phase may be enhanced if the pumps are not degraded, or if the break discharge rate is low due to saturated fluid conditions at the break. If pump degradation is high or the break flow is large, the cooling effect due to upward flow may not be significant. Figures 15.6-8d and 15.6-9d show the void fraction for one intact loop pump and the broken loop pump. The figure shows that the intact loop remains in single-phase liquid flow for several seconds, resulting in enhanced upward core flow cooling. This phase ends as the lower plenum mass is depleted, the loop flow becomes two-phase, and the pump head degrades.

3) Downward Core Flow Phase:

The loop flow is pushed into the vessel by the intact loop pumps and decreases as the pump flow becomes two-phase. The break flow begins to dominate and pulls flow down through the core, up the downcomer to the broken loop cold leg, and out the break. While liquid and entrained liquid flow provide core cooling, the top of core vapor flow (Figures 15.6-8e, 15.6-8f, 15.6-ge and 15.6-9f) best illustrates this phase of core cooling. Once the system has depressurized to the accumulator pressure, the accumulators begin to inject cold borated water into the intact cold legs (Figures 15.6-8h and 15.6-9h). During this period, due to steam upflow in the downcomer, a portion of the injected ECCS water is calculated to be bypassed around the downcomer and out the break. As the system pressure continues to fall, the break flow, and consequently the downward core flow, are reduced. The core begins to heat up as the system pressure approaches the containment pressure and the vessel begins to fill with ECCS water (Figures 15.6-81 and 15.6-91).

4) Refill Period:

As the refill period begins, the core begins a period of heatup and the vessel begins to fill with ECCS water (Figures 15.6-81 and 15.6-91). This period is characterized by a rapid increase in cladding temperatures at all elevations due to the lack of liquid and steam flow in the core region. This period continues until the lower plenum is filled and the bottom of the core begins to reflood and entrainment begins.

5) Reflood Period:

During the early reflood phase, the accumulators begin to empty and nitrogen enters the system. This forces water into the core, which then boils, causing system repressurization, and the lower core region begins to quench (Figures 15.6-8m and 15.6-9m). During this time, core cooling may increase due to vapor generation and liquid entrainment. During the reflood period, the core flow is oscillatory as cold water periodically rewets and quenches the hot fuel cladding, which generates steam and causes system re-pressurization. The steam and entrained water must pass through the vessel upper plenum, the hot legs, the steam generators, and the reactor coolant pumps before it is vented out of the break. This flow path resistance is overcome by the downcomer water elevation head, which provides the gravity driven reflood force. From the later stage of blowdown to the beginning of reflood, the accumulators rapidly discharge borated cooling water into the RCS, filling the lower plenum and contributing to the filling of the downcomer. The pumped ECCS water (Figures 15.6-8i and 15.6-9i) aids in the filling of the downcomer and subsequently supplies water to maintain a full down comer and complete the reflood period. As the quench front progresses up the core (Figures 15.6-8j and 15.6-9j), the PCT location moves higher into the top core region (Figures 15.6-8m and 15.6-9m). As the vessel continues to fill, the PCT location is cooled the early reflood period is terminated.

A second cladding heatup transient may occur due to boiling in the downcomer. The mixing of ECCS water with hot water and steam from the core, in addition to the continued heat transfer from the hot vessel and vessel metal, reduces the subcooling of ECCS water in the lower plenum and downcomer. The saturation temperature is dictated by the containment pressure. If the liquid temperature in the downcomer reaches saturation, subsequent heat transfer from the vessel and other structures will cause boiling and level swell in the downcomer. The downcomer liquid will spill out of the broken cold leg and reduce the driving head, which can reduce the reflood rate, causing a late reflood heatup at the upper core elevations. Figures 15.6-8k and 15.6-9k shows only a slight reduction in downcomer level and indicates that a late reflood heatup does not occur.

Additionally, it must be demonstrated that there is a high level of probability that the limits set forth in 10 CFR 50.46 are met. The demonstration that these limits are met is as follows:

(b)(1) The limiting PCT corresponds to a bounding estimate of the 95th percentile PCT at the 95-percent confidence level. Since the resulting PCT for the limiting cases are 1913 OF for Unit 1 and 2041 OF for Unit 2, the analyses confirm that 10 CFR 50.46 acceptance criterion (b)(1), i.e., "Peak Clad Temperature less than 2200°F, is demonstrated. The results are shown in Table 15.6-3a.

(b)(2) The limiting cladding oxidation corresponds to a bounding estimate of the 95th percentile LMO at the 95-percent confidence level. Since the resulting LMO for the limiting cases are 5.51% for Unit 1 and 8.27% for Unit 2, the analyses confirm that 10 CFR 50.46 acceptance criterion (b)(2), i.e., "Local Maximum Oxidation of the cladding less than 17 percent", is demonstrated. The results are shown in Table 15.6-3a.

(b)(3) The limiting core-wide oxidation corresponds to a bounding estimate of the 95^{th} percentile CWO at the 95-percent confidence level. The limiting Hot Assembly Rod (HAR) total maximum oxidation is 0.25 percent for Unit 1 and 0.33 percent for Unit 2. A detailed CWO calculation takes advantage of the core power census that includes many lower power assemblies. Because there is significant margin to the regulatory limit, the CWO value can be conservatively chosen as that calculated for the limiting HAR. A detailed CWO calculation is therefore not needed because the outcome will always be less than 0.25 percent for Unit 1 and 0.33 for Unit 2. Since the resulting CWO is 0.25 percent for Unit 1 and 0.33 percent for Unit 2, the analyses confirm that 10 CFR 50.46 acceptance criterion (b)(3), i.e., "Core-Wide Oxidation less than 1 percent", is demonstrated. The results are shown in Table 15.6-3a.

(b)(4) 10 CFR 50.46 acceptance criterion (b)(4) requires that the calculated changes in core geometry are such that the core remains amenable to cooling. This criterion has historically been satisfied by adherence to criteria (b)(1) and (b)(2), and by assuring that fuel deformation due to combined LOCA and seismic loads is specifically addressed. It has been demonstrated that the PCT and maximum cladding oxidation limits remain in effect for Best-Estimate LOCA applications. The approved methodology (Reference 36) specifies that effects of LOCA and seismic loads on core geometry do not need to be considered unless grid crushing extends beyond the 44 assemblies in the low-power channel. This situation has not been calculated to occur for Byron/Braidwood. Therefore, acceptance criterion (b)(4) is satisfied.

(b)(5) 10 CFR 50.46 acceptance criterion (b)(5) requires that long-term core cooling be provided following the successful initial operation of the ECCS. Long-term cooling is dependent on the demonstration of continued delivery of cooling water to the core. The actions, automatic or manual, that are currently in place at these plants to maintain long-term cooling remain unchanged with the application of the ASTRUM methodology (Reference 4).

Based on the ASTRUM Analysis results (Table 15.6-3a), it is concluded that Byron/Braidwood continue to maintain a margin of safety to the limits prescribed by 10 CFR 50.46.

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15.6.5.2.3.2 Results of Small Break LOCA Analysis

15.6.5.2.3.2.1 Unit 1 Analysis

The time sequence of events and results of interest for the 1.5, 2, 3 and 4-inch diameter breaks at the reduced T_{avg} conditions are shown in Tables 15.6-1c and 15.6-4a, respectively.

The analysis included both Zircaloy-4 and ZIRLO® fuel assemblies in accordance with the licensing guidelines of Reference 33. Zircaloy-4 results for the limiting case have been provided in Tale 15.6-4d.

Figures 15.6-14a through 15.6-17n show the pertinent small break LOCA parameters during the transient.

Figures 15.6-14a - 15.6-17a (reduced) 15.6-14h – 15.6-17h (nominal)	Peak Clad Temperature
Figures 15.6-14b - 15.6-17b (reduced) 15.6-14i – 15.6-17i (nominal)	Local fluid temperatures at the peak clad temperature elevation
Figures 15.6-14c - 15.6-17c (reduced) 15.6-14j – 15.6-17j (nominal)	Local heat transfer coefficient at the peak clad temperature elevation
Figures 15.6-14d - 15.6-17d (reduced) 15.6-14k – 15.6-17k (nominal)	RCS pressure
Figures 15.6-14e - 15.6-17e (reduced) 15.6-14l – 15.6-17l (nominal)	Core mixture height
Figures 15.6-14f - 15.6-17f (reduced) 15.6-14m – 15.6-17m (nominal)	Core steam flow
Figures 15.6-14g - 15.6-17g (reduced) 15.6-14n – 15.6-17n (nominal)	Break flow

The PCT for the limiting break, 2 inches in diameter at reduced T_{avg} was 1624°F, occurring at an assembly average burnup of 0 MWD/MTU with either Zircaloy-4 or ZIRLO fuel assembly materials. The maximum local zirconium oxidation of 1.47%, which is less than the 17% criterion, and the core wide oxidation was less than 0.22% (Refer to Reference 35), which meets the 1% criterion. These results indicate that a coolable geometry was maintained for the small break LOCAs and that, therefore, long-term cooling is assured by continued operation of the ECCS. Sometimes the small break LOCA PCT occurs at a later time in life due to rod burst and an associated zirconium-water driven spike in PCT. However, since the PCT was less than 1700°F, burnup studies were not performed on the limiting 2-inch, reduced T_{avg} case. Instead, a BOL burnup of 0 MWD/MTU was assumed.

15.6.5.2.3.2.2 Unit 2 Analysis

The time sequence of events and results of interest for the 1.5-, 2-, 3-, and 4-inch diameter breaks at the nominal T_{avg} conditions are shown in Tables 15.6-1d and 15.6-4b, respectively.

The analysis included both Zircaloy-4 and ZIRLO fuel assemblies in accordance with the licensing guidelines of Reference 33. Zircaloy-4 fuel assemblies were found to be more limiting than the ZIRLO fuel assemblies for the 3-inch case at a burnup of 0 Mwd/Mtu. As shown in Table 15.6-4d, a burnup of 6000 Mwd/Mtu was assumed for Zircaloy-4 fuel to make ZIRLO clad fuel limiting.

Figures 15.6-18a through 15.6-21n show the pertinent small break LOCA parameters during the transient.

Figures 15.6-18a - 15.6-21a (reduced) 15.6-18h – 15.6-21h (nominal)	Peak clad temperature
Figures 15.6-18b - 15.6-21b (reduced) 15.6-18i – 15.6-21I (nominal)	Local fluid temperature at the peak clad temperature elevation
Figures 15.6-18c - 15.6-21c (reduced) 15.6-18j – 15.6-21j (nominal)	Local heat transfer coefficient at the peak clad temperature elevation
Figures 15.6-18d - 15.6-21d (reduced) 15.6-18k – 15.6-21k (nominal)	RCS pressure
Figures 15.6-18e - 15.6-21e (reduced) 15.6-18l – 15.6-21l (nominal)	Core mixture height
Figures 15.6-18f - 15.6-21f (reduced) 15.6-18m – 15.6-21m (nominal)	Core steam flow
Figures 15.6-18g - 15.6-21g (reduced) 15.6-18n – 15.6-21n (nominal)	Break flow

The PCT for the limiting break, 2-inch diameter at reduced T_{avg}, and ZIRLO fuel assembly materials was 1627°F occurring at an assembly average burnup of 0 MWd/MTU. The maximum local zirconium oxidation was 1.59% and the core wide oxidation was less than 0.24% (See Reference 35) which meets the 1% criteria. These results indicate that a coolable geometry was maintained for small break LOCAs and therefore, long-term core cooling is assured by continued operation of the ECCS. Sometimes the small break LOCA PCT occurs at a later time in life due to rod burst and an associated Zirconium-water driven spike in PCT.

15.6.5.2.3.2.3 Safety Injection Evaluation

Subsequent to the completion of the break spectrum study, a non-conservative discrepancy was discovered in the safety injection flows used in the analysis. Two data points were not included in the original cases and thus the flow modeled were non-conservatively higher in the analysis. These points more appropriately modeled the shutoff pressure of the high pressure safety injection pumps. The effect of this is greater on the 2-inch breaks and negligible for the 3-inch breaks and larger, greatly due to the increased dependency on safety injection for the smaller break sizes. The result for Unit 1 was a more limiting Peak Cladding Temperature (PCT) and the result for Unit 2 was a shift in break size and conditions to the 2-inch Low Tavg break. The prior Unit 2 limiting case was the 3-inch High Tavg case. Tables 15.6-1c, 15.6-1e, 15.6-4a, and 15.6-4c have been updated to reflect the results. Also, Figures 15.6-15a through 15.6-15g and Figures 15.6-18h through 15.6-18n have been updated to reflect the new 2-inch Low Tavg limiting case for Unit 1 and Unit 2, respectively. Note that for the Zirc-4 cladding evaluation, the results for Unit 2 are based on the High Tavg 3-inch prior limiting case. Although the new Unit 2 Low Tavg 2-inch limiting case has not been performed with Zirc-4 cladding, the prior results are conservatively being applied to the new limiting case because the effects are expected to be similar. The impact of the SI discrepancy has also been evaluated on the other break sizes and resulted in a negligible impact on those. Thus, the original results demonstrated herein remain applicable to those break sizes.

15.6.5.2.3.3 Post Analysis of Record Evaluations

In addition to the analyses presented in Subsections 15.6.5.2.1 and 15.6.5.2.2, evaluations and reanalyses may be performed as needed to address emergent issues or to support plant changes. The issues or changes are evaluated, and the impact on the PCT is determined. The resultant increase or decrease in PCT is added to the analysis of record.

The peak clad temperatures, including all penalties and benefits are presented in Table 15.6-15.

15.6.5.2.3.3.1 Large Break LOCA

As documented in the preceding section two complete large break LOCA analyses were performed for both the Byron and Braidwood Units with a F_Q^T of 2.6 and $F_{\Delta H}^N$ of 1.70 as reported in Table 15.6.3b. The 95 percentile PCT results are reported in Table 15.6-3a for these analyses.

In addition, evaluations/reanalysis are also performed from time to time to address the various issues or to support plant changes as they arise. On a cycle specific basis, if axial power shapes are beyond those reflected in Figure 15.6-7, the violating axial shapes are evaluated. The plant changes or the issues are evaluated and the impact on the PCT determined. The resultant increase or decrease in PCT is added to the analysis of record PCT. These issues and their evaluations are reported to the NRC via the normal 10CFR50.46 reporting requirement. The latest 10 CFR 50.46 report is publicly available on the NRC website in the Agencywide Documents Access and Management System (ADAMS). Between UFSAR updates the latest PCT is tracked by the cognizant organization.

The current PCT (including PCT penalties/benefits associated with all the evaluations) for the four Byron and Braidwood Units are maintained in Table 15.6-15.

15.6.5.2.3.3.2 Small Break LOCA - Unit 1

As documented in the preceding section a complete break spectrum small break LOCA analysis was performed for Byron and Braidwood Unit 1. The results of this analysis are reported in Tables 15.6-4 and 15.6.4a. The limiting break was determined to be the 2-inch diameter break at reduced T_{avg} conditions with a PCT of 1624°F, the analysis of record. In addition, evaluations/reanalysis are performed from time to time to address the various issues or to support plant changes as they arise. The plant changes or the issues are evaluated and the impact on the PCT determined. The resultant increase or decrease in PCT is added to the analysis of record PCT (1624°F). These issues and their evaluations are reported to the NRC via the normal 10CFR50.46 reporting requirement. The latest 10 CFR 50.46 report is publicly available on the NRC website in the Agencywide Documents Access and Management System (ADAMS).

An evaluation of reduced recirculation flows was performed to support GSI-191 modifications using the flows provided in Figures 15.6-13d for intact loop and 15.6-13e for broken loop. The current PCT (including PCT penalties/benefits associated with all the evaluations) for the two Byron and Braidwood Unit 1s are maintained in Table 15.6-15.

15.6.5.2.3.3.3 Small Break LOCA - Unit 2

As documented in the preceding section a complete break spectrum small break LOCA analysis was performed for Byron and Braidwood Unit 2. The results of this analysis are reported in Tables 15.6-4b and 15.6.4c. The limiting break was determined to be the 2-inch diameter break at reduced T_{avg} conditions with a PCT of 1627°F, the analysis of record.

In addition, evaluations/reanalysis are also performed from time to time to address the various issues or to support plant changes as they arise. The plant changes or the issues are evaluated and the impact on the PCT determined. The resultant increase or decrease in PCT is added to the analysis of record PCT (1627°F) These issues and their evaluations are reported to the NRC via the normal 10CFR50.46 reporting requirement. The latest 10 CFR 50.46 report is publicly available on the NRC website in the Agencywide Documents Access and Management System (ADAMS).

An evaluation of reduced recirculation flows was performed to support GSI-191 modifications using the flows provided in Figures 15.6-13d for intact loop and 15.6-13e for broken loop. The current PCT (including PCT penalties/benefits associated with all the evaluations) for the two Byron and Braidwood Unit 2s are maintained in Table 15.6-15.

15.6.5.2.4 Post-LOCA Long-Term Core Cooling/Subcriticality

10CFR50.46, "Acceptance Criteria for Emergency Core Cooling

Systems for Light Water Nuclear Power Reactors" paragraph (b) item (5) sets forth the requirements for post-LOCA long-term core cooling. To satisfy the requirements, the core is maintained in a shutdown state solely by the soluble boron contained in the ECCS water after a LOCA because credit for the shutdown provided by the control rods was not taken for breaks $\geq$ 1.0 ft^2. Since safety injection flow is drawn from the sump following switchover from the RWST, the containment sump post-LOCA boron concentration must be higher than the boron concentration required to ensure subcritical conditions.

To determine if the requirements for post-LOCA long-term core cooling subcriticality are met, a calculation is performed for each reload to determine the boron concentration required to keep the core subcritical (K_{eff}<1.0) and the mixed mean boron concentration (MMBC) of the post-LOCA sump water. This calculation, documented in the cycle-specific Reload Safety Analysis Checklist (RSAC), confirms that the post-LOCA sump MMBC exceeds the core critical boron concentration, thereby ensuring the reload core remains subcritical post-LOCA. Note: The post-LOCA long-term core cooling critical boron concentration is determined at the most reactive time in life, assuming an all rods out (ARO) no Xenon condition and a post-LOCA fluid temperature range of 68-212°F. All sources of water that may eventually reside in the containment sump at cold leg recirculation switchover time and their respective pre-accident boron concentrations are considered.

Westinghouse has identified a potential safety issue concerning core recriticality following a large break cold leg break LOCA (Reference 43). The potential safety issue is that during hot leg switchover the core will be flushed with a diluted sump solution, which may cause the core to return to criticality. The sump solution would become diluted as boron accumulates in the core during the cold leg recirculation phase due to core boiling. The accumulation of boron in the core prevents the boron from being displaced to the sump which leads to a diluted sump solution.

However, a generic assessment (Reference 43) concludes that for any given plant and fuel cycle, the boron worth of the inserted control rods plus the equivalent boron worth due to the presence of Xenon at the time of hot leg switchover, would offset any reasonable calculation of sump dilution. As documented in WCAP-15704 (Reference 44), it was demonstrated that control rods will insert following a licensing basis cold leg LOCA for 3-loop and 4-loop Westinghouse plant designs. Control rod insertion will result in negative reactivity benefits on the order of 400 ppm or more (boron equivalent rod worth). Thus, for Byron and Braidwood, the negative reactivity credit associated with control rods can be applied when evaluating recriticality at the time of switchover to hot leg ECCS recirculation. Note that for non-Westinghouse fuel further assessment is required before credit for control rod insertion can be taken to address the post-LOCA subcriticality issue. See section 7.4 of Reference 44 for details.

Consequently, post-LOCA subcriticality will continue to be confirmed using sump boron calculations that do not consider sump dilution, but also use the conservative assumptions of ARO and no Xenon at the most reactive time in life, with an assumed post-LOCA core/fluid temperature in the range of 68-212°F.

15.6.5.2.5 Conclusions - Thermal Analysis

For breaks up to and including the double-ended severance of a reactor coolant pipe, the emergency core cooling system will meet the acceptance criteria as presented in Subsection 15.6.5.1.2 for a total core peaking factor, F_Q^T , of 2.60, and an $F_{\Delta H}^N$ of 1.70.

15.6.5.3 Radiological Consequences of a Postulated Loss-of-Coolant Accident

The results of analyses presented in this section demonstrate that the amounts of radioactivity released to the environment in the event of a loss-of-coolant accident do not result in doses which exceed the guideline values specified in 10 CFR 50.67.

The key inputs and assumptions used in the AST LOCA radiological consequence analysis are summarized below and provided in Table 15.6-7.

EAB and LPZ Dose Model

The EAB and LPZ χ/Qs have been determined, and the worst-case values are used for either of the Byron and Braidwood Stations as located at 445m and 1810m, respectively, from the postulated release locations.

Control Room Dose Model

The following Control Room (CR) dose model, with assumed CR HVAC system operation, is used to calculate CR personnel dose for AST re-analyzed accident scenarios.

CR HVAC System Operation:

Actuation of the CR filtration system places the system in the emergency mode of operation. Actuation of the system to the emergency mode of operation starts the makeup fan, opens the turbine building intake damper, isolates the normal intake from outside dampers, isolates the purge dampers (if open), opens the recirculation charcoal adsorber dampers, and closes the recirculation charcoal adsorber bypass dampers. The operating supply and return fans continue to operate. Outside air from the turbine building is filtered and added to the air being recirculated through the CR.

Following CR emergency mode of operation isolation, the credited CR filtration is 99% for the HEPA and 95% for the charcoal filters. The maximum intake rate is used because sensitivity analyses have shown that it is conservative to maximize the potentially contaminated emergency mode of operation make-up airflow into the CR.

The CR HVAC system emergency mode of operation is not credited during the first 30 minutes of the accident, and therefore, unfiltered make-up air is assumed. This assumption is intended to allow for realignment of the CR HVAC system to the emergency mode of operation. During this 30-minute period, it is conservative not to consider the potential filtered intake entering the CR (i.e., the flow from the unused CR make-up train). This assumption is used because filtered flow would act to "clean" the unfiltered air being brought in by the other unfiltered flows (i.e., normal intake and unfiltered inleakage) during these 30 minutes. Sensitivity analyses using RADTRAD confirmed that such additional filtered intake during the first 30 minutes of an accident (i.e., with the CR filtration system in the normal mode of operation) would lower the CR dose consequences.

The CRs are modeled using a conservatively reduced recirculation train flow rate of 39,150 cfm. This recirculation train filtration consists of a 90% elemental and organic iodine filter (i.e., charcoal filter). No reduction in the efficiency of this filter is proposed.

For additional conservatism, an unfiltered inleakage rate allowance of 500 cfm is modeled for the accident duration.

The following table summarizes the CR dose model inputs and assumptions.

Time	Intake (cfm)	Intake E (%)	Recirc. (cfm)	Recirc. E (%)	Unfiltered Inleakage (cfm)	Flow to UCSR (cfm)	CR Exfiltration (cfm)
0-30 min	6424	0	39,150	0	500	Byron: 1319 Braidwood: 2430	6924
30 min to 30 days	8575	99 HEPA 95 Charcoal	39,150	90 Charcoal	500	Byron: 1319 Braidwood: 2430	9075

Where:
E = Efficiency
UCSR = Upper Cable Spreading Room

Assumptions and Inputs

Input parameters used for the LOCA analysis are given in Table 15.6-7.

The following primary assumptions from previous LOCA analyses continue to apply:

a. Two release pathways are considered: containment leakage and ECCS recirculation leakage and
b. A single electrical train failure is assumed to remove the following equipment from service: one containment spray train, one ECCS train, and two of the four reactor containment fan coolers (RCFCs).

Primary Containment and ECCS Leakage

Primary Containment Leakage:

Primary containment leakage is the main contributor to LOCA doses. The impacts of the design basis LOCA are mitigated by: (1) a controlling design basis leak rate; and (2) operation of the containment spray system. While most penetrations are into the auxiliary building that has an ESF filtered exhaust, this filtration system is not credited for primary containment leakage.

Primary containment leakage is modeled as a diffuse area source in conformance with RG 1.194.

The assumed containment leak rate is 0.2% per day. This leak rate is assumed to be reduced to one-half the initial value after 24 hours due to expected reductions in containment pressure.

Containment spray removal coefficients continue to be based on Standard Review Plan 6.5.2, "Containment Spray as a Fission Product Cleanup System," Draft Revision 3, April 1996, with "particulate" removal coefficients applied to "aerosols." Spray timing reflects AST-caused differences in time to reach decontamination factor (DF) credit limits.

Dispersion factors, developed in conformance with the guidance in RG 1.145 and RG 1.194 are used. Dispersion factors for primary containment leakage for the CR are based on a diffuse area source.

ECCS Leakage:

ECCS leakage is a minor contributor to LOCA doses. The ECCS leakage rate assumed in the AST LOCA analysis is 276,000 cc/hr. The ECCS leakage flashing fractions assumption is 10% for the duration of the accident.

Fuel Damage and Core Source Term

The LOCA core source terms are those associated with a DBA power level of 3658.3 MWt. This power level bounds the MUR power uprate Rated Thermal Power level including measurement uncertainties.

The AST values used in this analysis were derived using guidance outlined in RG 1.183. A list of 60 core isotopic nuclides and their curie per megawatt activities was extracted from the RADTRAD "NIF" files. The release fractions associated with all of these nuclide groups, as detailed in RG 1.183, were applied to their given groups, and input into the RADTRAD "RTF" files. RADTRAD uses these files combined with the power of 3658.3 MWth to develop the source terms for the DBA LOCA.

As discussed below, containment spray, natural deposition, decay, and leakage are credited as airborne activity removal mechanisms.

Removal by Containment Spray

Iodine removal by containment spray is in accordance with Standard Review Plan Chapter 6.5.2. The major impacts of application of AST are that no initial plateout fraction is assumed and the duration of credited spray is modified.

Sprayed and Unsprayed Volumes, and Air Exchange Rate:

The containment volume is 2.85E6 cubic feet with 82.5% of the containment volume sprayed; i.e., the sprayed volume is 2.35125E6 cubic feet and the unsprayed volume is 4.9875E5 cubic feet. These values are rounded to 2.35E6 and 5.0E5 cubic feet, respectively, in the analysis. Initial activity distribution is accordingly, 82.5% in the sprayed region, and 17.5% in the unsprayed region.

Transfer between these two volumes is assumed to be limited to that provided by the Reactor Containment Fan Coolers (RCFCs). Even without the RCFCs, there would be significant mixing induced by the containment sprays and by the combination of steaming and heat transfer. Two of four RCFCs are credited in the analysis. The assumed flow rate per RCFC is 65,000 cfm for a total of 130,000 cfm.

Spray Removal Coefficients for Aerosols (SRP 6.5.2 particulate removal):

From SRP 6.5.2, the first order removal coefficient for particulate (or, effectively, aerosols) may be estimated by:

$$\lambda_p = \frac{3\ (h)\ (F)\ (E)}{2\ (V)\ (D)}$$

where:

λ_p	=	spray removal constant, hr^{-1}
h	=	drop fall height = 141 ft
F	=	volume flow rate of sprays, ft^3/hr
	=	2950 gpm (applicable to both injection and recirculation phases)
	=	(2950 gpm)(60 min/hr) / (7.4805 gal/ft^3)
	=	23,661.5 ft^3/hr
V	=	sprayed volume, ft^3
	=	(0.825)(2.85E6 ft^3)
	=	2.35125E6 ft^3
E/D	=	ratio of a dimensionless collection efficiency "E" to the average spray drop diameter "D"
	=	10 m^{-1} for $M_o/M_t \leq 50$
	=	1 m^{-1} for $M_o/M_t \overline{<} 50$
		where M_o/M_t is the ratio of the initial aerosol mass to the aerosol mass at time t (note that this ratio also defines the DF achieved)

$$\lambda_{P1} = \frac{3\ (141\ ft)\ (23{,}661.5\ ft^3/hr)\ (10\ m^{-1})\ (0.3048\ m/ft)}{2\ (2.35E6\ ft^3)} = 6.491\ hr^{-1}$$

$$\lambda_{P2} = 0.1 \times 6.491\ hr^{-1} = 0.65\ hr^{-1}$$

(The calculated λ_{P1} and λ_{P2} values above are conservatively reduced to 6.0 hr^{-1} and 0.6 hr^{-1}, respectively).

It is assumed that after the end of the core activity release process the aerosols would continue to be removed at a λ of 6.0 hr^{-1} until an overall DF of 50 is achieved.

Spray Removal Coefficients for Elemental Iodine (same as SRP 6.5.2 based elemental iodine removal):

The current SRP 6.5.2 based assessment of elemental iodine removal coefficients during containment spray are used. The design basis derivation of the spray removal coefficient is 20 hr^{-1}, per SRP 6.5.2. Elemental iodine removal is limited to a DF of 100.

Spray Timing:

The injection spray is initiated at 90 seconds and continues for 20.9 minutes from the time of initiation. There may be a delay of as much as ten minutes between the termination of containment spray during the ECCS injection phase and the initiation of containment spray during the ECCS recirculation phase.

For aerosol removal, the DF of 50 is reached at 2.21 hours. From that point until eight hours, the removal coefficient of 0.6 hr^{-1} is used. For elemental iodine removal, the DF of 100 is reached at 1.926 hours. After that time, no elemental iodine removal is credited.

Natural Deposition:

The RADTRAD computer program, including the Powers Natural Deposition algorithm based on NUREG/CR-6189, is used for modeling aerosol deposition in containment. No natural deposition is assumed for elemental or organic iodine. The lower bound (i.e., -10%) level of deposition credit is used.

Decay Credited:

Decay of radioactivity is credited in all compartments prior to release. This is implemented in RADTRAD using the half-lives in the "NIF" files. The RADTRAD decay plus daughter option is used. In reality, daughter products such as xenon from iodines or iodines from tellurium are unlikely to readily escape from the medium in which the parent iodine or tellurium is contained. The RADTRAD feature to include daughter effects is selected for conservatism.

Depletion from Leakage Credited:

For analyses of doses due to leakage from containment, it is reasonable to credit the small amount of depletion from the containment inventory associated with this leakage. This is done inherently by RADTRAD.

Mechanisms for Environmental Releases:

The applicable release paths are leakage of airborne activity from primary containment, and leakage from ECCS system carrying reactor coolant outside containment into the Auxiliary Building. These release paths are discussed below.

Containment source dose contributions are not significantly impacted by AST. Airborne containment activity is comparable or slightly smaller with AST due to time dependent release. Therefore, no change in the containment component is assumed. Auxiliary Building contained sources such as ECCS piping are not expected to see an increase in dose. However, airborne activity may be increased since the containment leak rate allowable has been doubled. Since dose contributors are not itemized, the values in UFSAR Tables E.20-1and E.20-2 were doubled for conservatism.

Control Room Direct Gamma Dose

The contributors to Control Room doses due to gamma shine are Containment Building airborne activity; the post-LOCA plume surrounding the Control Room; and radioactivity accumulated on the Control Room Filter.

The pre-AST containment activity dose of 0.023 rem whole body is slightly conservative for AST conditions, and therefore is treated as a 0.023 rem TEDE dose contribution.

The pre-AST external plume shine dose of 0.003 rem whole-body to control room personnel is multiplied by a factor of 5 to yield 0.015 rem TEDE. Generally these doses are noble gas dominated. This conservative multiplier accounts for the factor of 2 increase in primary containment leak rate assumptions, and other increase contributors such as ECCS iodine.

Fission product filter loading was reanalyzed using AST assumptions, and compared with those that would be determined using RG 1.3 assumptions. The pre-AST source terms were found to be bounding. Therefore, the pre-AST 0.013 rem whole body dose will continue to be used as an 0.013 rem TEDE dose. The total Direct Gamma Shine Dose is 0.05 rem TEDE.

χ/Q Calculations (Meteorology)

Control Room

The CR χ/Q values input to RADTRAD were taken from the ARCON96 results of the design bases analyses. The limiting χ/Q values used are conservatively the worst-case combination of values from all four units at Byron Station and Braidwood Station. The atmospheric dispersion factors are given in Table 15.0-17.

Releases from primary containment to the environment and subsequently to the CR, utilize the χ/Q value associated with a diffuse area source in conformance with Regulatory Guide 1.194 (Diffuse Area Source Guidance).

Activity released during the initial 30 minutes of the accident is introduced to the CR via the normal CR fresh air intake. After this period, when the CR is assumed to be in the emergency mode of operation, the emergency air intake located in the turbine building is used.

The CR, EAB, and LPZ χ/Q two-hour values are selected such that they coincide with the release period that caused the highest doses.

Dose Results

Radiological doses resulting from a design basis LOCA to a control room operator and a person located at EAB or LPZ are to be less than the regulatory dose limits given below.

Regulatory Dose Limits - LOCA

Dose Type	Control Room (rem)	EAB and LPZ (rem)
TEDE Dose	5[a]	25[b]

Notes:
[a] 10 CFR 50.67
[b] 10 CFR 50.67 as modified by Regulatory Guide 1.183 (Table 6, Page 1.183-20)

Loss of Coolant Accident
Radiological Analysis Results

Source	Control Room (rem TEDE)	EAB (rem TEDE)	LPZ (rem TEDE)
Primary Containment Leakage	3.35	13.50 Maximum 2 hrs	3.21
ECCS Leakage	1.54	1.53 Maximum 2 hrs	1.78
Direct Dose from Containment, External Plume, & CR Filters	0.05	None	None
Total Dose	4.94	15.03	4.99

The total doses are below the Regulatory Dose Limits, so it is verified that this design basis LOCA is sufficiently mitigated at both Byron Station and Braidwood Station.

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15.6.6 BWR Transient

This is not applicable to Byron/Braidwood.

15.6.7 References

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TABLE 15.6-1

TIME SEQUENCE OF EVENTS FOR INADVERTENT OPENING

OF A PRESSURIZER SAFETY VALVE

EVENT	TIME (SEC)
Safety valve opens fully	0.0
Low Pressurizer Pressure reactor trip setpoint reached	29.59
Rods begin to drop	31.59
Minimum DNBR occurs	32.20

TABLE 15.6-1a

TIME SEQUENCE OF EVENTS FOR LARGE BREAK LOCA ANALYSIS
LIMITING PCT CASE

EVENT	Unit 1	Unit 2
Start	0.0	0.0
Reactor trip signal	N/A	N/A
Safety injection signal	6	6
Accumulator injection begins	14	12
End-of-bypass	N/A	N/A
End-of-blowdown	26	24
Pump injection begins	46	46

Note: All times are in seconds.

TABLE 15.6-1a (Cont'd)

TIME SEQUENCE OF EVENTS FOR LARGE BREAK LOCA ANALYSIS
LIMITING PCT CASE

EVENT	Unit 1	Unit 2
Bottom of core recovery	36	33
Hot rod burst	36	32
Peak clad temperature Occurs	102	96
Accumulators empty	46	56

Note: All times are in seconds.

TABLE 15.6-1b

TIME SEQUENCE OF EVENTS FOR SMALL BREAK LOCA ANALYSIS
UNIT 1 NOMINAL TAVG

Event Time (sec)	1.5 Inch	2 Inch	3 Inch	4 Inch
Break Initiation	0	0	0	0
Reactor Trip Signal	147.1	82.3	54.4	24.7
Safety Injection Signal	159.3	93.9	66.5	35.6
Safety Injection Delivered	199.3	133.9	106.5	75.6
Top of Core Uncovered	15020	2112	863	637
Accumulator Injection Begins	N/A	N/A	2002	920
Peak Clad Temperature Occurs	22680.6	3434.5	1834.7	1019.1
Top of Core Recovered	>TMAX	>TMAX	2960	2200

Note: All times are in seconds.

TABLE 15.6-1c

TIME SEQUENCE OF EVENTS FOR SMALL BREAK LOCA ANALYSIS
UNIT 1 REDUCED TAVG

Event Time (sec)	1.5 Inch	2 Inch	3 Inch	4 Inch
Break Initiation	0	0	0	0
Reactor Trip Signal	79.0	41.8	17.7	10.2
Safety Injection Signal	123.3	65.2	27.1	14.1
Safety Injection Delivered	163.3	105.2	67.1	54.1
Top of Core Uncovered	16040	2268	1032	731.1
Accumulator Injection Begins	N/A	N/A	2119	991
Peak Clad Temperature Occurs	22894.8	3455.5	2013.5	1099.7
Top of Core Recovered	>TMAX	>TMAX	2955	2150

Note: All times are in seconds.

TABLE 15.6-1d

TIME SEQUENCE OF EVENTS FOR SMALL BREAK LOCA ANALYSIS
UNIT 2 NOMINAL TAVG

Event Time (sec)	1.5 Inch	2 Inch	3 Inch	4 Inch
Break Initiation	0	0	0	0
Reactor Trip Signal	142.1	80.2	59.0	24.6
Safety Injection Signal	154.6	91.8	71.6	36.4
Safety Injection Delivered	194.6	131.8	111.6	76.4
Top of Core Uncovered	9810	1809.3	771	510
Accumulator Injection Begins	N/A	N/A	1732	990.8
Peak Clad Temperature Occurs	16234.8	2804.5	1618.5	889.0
Top of Core Recovered	37098	4740	2754	2378

Note: All times are in seconds.

TABLE 15.6-1e

TIME SEQUENCE OF EVENTS FOR SMALL BREAK LOCA ANALYSIS
UNIT 2 REDUCED TAVG

Event Time (sec)	1.5 Inch	2 Inch	3 Inch	4 Inch
Break Initiation	0	0	0	0
Reactor Trip Signal	77.7	41.3	17.6	10.2
Safety Injection Signal	117.8	59.6	27.2	14.3
Safety Injection Delivered	157.8	67.2	96.6	54.3
Top of Core Uncovered	10750	1930	717.9	614.5
Accumulator Injection Begins	N/A	N/A	1928.1	882.6
Peak Clad Temperature Occurs	17491.8	3071.7	1805.6	992.3
Top of Core Recovered	36950	5543	2826	2103

Note: All times are in seconds.

TABLE 15.6-2

INPUT PARAMETERS USED IN THE SBLOCA ANALYSIS[c]

Analyzed Core Power[a], (MWt)	3659
Total peaking factor, F_Q	2.60
Axial peaking factor, F_Z ($F_Q/F_{\Delta H}$)	1.53
Power shape	See Figure 15.6-13c
Accumulator water volume, nominal (ft^3/accumulator)	950
Accumulator tank volume, nominal (ft^3/accumulator)	1350
Accumulator gas pressure, minimum (psia)	600
Accumulator water temperature (°F)	130
Safety injection pumped flow	See Figures 15.6-13a and 15.6-13b
Pressurizer pressure reactor trip signal (psia)	1857
Pressurizer pressure SI signal (psia)	1715

TABLE 15.6-2 (Cont'd)

INPUT PARAMETERS USED IN THE SBLOCA ANALYSIS[c]

Initial vessel flowrate (lb/sec)	
Reduced Tavg	39,416 (368,000 gpm)
Nominal Tavg	37,595 (368,000 gpm)
Vessel Average Temperature (°F)[b]	
Reduced Tavg	565
Nominal Tavg	598.0
Average reactor coolant pressure (psia)	2300
Steam generator pressure (psia)	
SBLOCA (Nominal T_{avg})	1108 (Unit 1)
	1012 (Unit 2)
SBLOCA (Reduced T_{avg})	824 (Unit 1)
	727 (Unit 2)
Steam generator tube plugging level (%)	
SBLOCA	5 (Unit 1)
	10 (Unit 2)

(a) Two percent is included in this power to account for calorimetric error. Therefore, the analyzed core power is bounding for the MUR power uprate up to a total core power(licensed core power plus calorimetric uncertainties) of 3659 MW_t.

(b) The vessel average temperature used in the LOCA analyses is based on the Tavg window of 575.0 °F through 588.0 °F ± 10 °F in uncertainties.

(c) The analysis is applicable for VANTAGE 5/VANTAGE+ fuel by accounting for the following features: IFMs, 100 and 200 psig IFBA, 275 psig Non-IFBA, Zircaloy-4 and ZIRLO cladding, and solid/mid-enriched annular blankets.

TABLE 15.6-2a

KEY LBLOCA PARAMETERS AND INITIAL TRANSIENT ASSUMPTIONS

Parameter	Initial Transient	Uncertainty or Bias
1.0 Plant Physical Description		
a. Dimensions	Nominal	-
b. Flow resistance	Nominal	Sample**
c. Pressurizer location	On Intact loop	Bounded
d. Hot assembly location	Under limiting location	Bounded
e. Hot assembly type	17x17 Vantage+ w/ZIRLO™ clad	-
f. SG tube plugging level	High (U1 5%, U2 10% ****)	Bounded*
2.0 Plant Initial Operating Conditions		
2.1 Reactor Power		
a. Core average linear heat rate (AFLUX)	Nominal - Based on 3658.33 MWt (102% of 3586.6 MWt)	Sample**
b. Hot Rod Peak linear heat rate (PLHR)	Derived from desired Tech Spec (TS) limit FQ = 2.60 and maximum baseload FQ	Sample**
c. Hot rod average linear heat rate (HRFLUX)	Derived from TS $F_{\Delta H}$ = 1.70	Sample**
d. Hot assembly average heat rate (HAFLUX)	HRFLUX/1.04	Sample**
e. Hot assembly peak heat rate (HAPHR)	PLHR/1.04	Sample**
f. Axial power distribution (PBOT, PMID)	Figure 15.6-7	Sample**
g. Low Power region relative power (PLOW)	High (0.62)	Bounded*
h. Hot assembly burnup	2000 MWD/MTU	Sample**
i. Prior operating history	Equilibrium decay heat	Bounded
j. Moderator Temperature Coefficient (MTC)	Tech Spec Maximum (0)	Bounded

* Confirmatory Sequence Parameter

TABLE 15.6-2a (Cont)

KEY LBLOCA PARAMETERS AND INITIAL TRANSIENT ASSUMPTIONS

Parameter	Initial Transient	Uncertainty or Bias
2.0 Plant Initial Operating Conditions (cont.)		
2.2 Fluid Conditions		
a. T_{AVG}	High Nominal (588.0°F***)	Bounded*, Sample**
b. Pressurizer pressure	Nominal (2250.0 psia)	Sample**
c. Loop flow	92,000 gpm	Bounded
d. T_{UH}	T_{COLD}	0
e. Pressurizer level	Nominal (60% span)	0
f. Accumulator temperature	Nominal (95°F)	Sample**
g. Accumulator pressure	Nominal 639.5 psia)	Sample**
h. Accumulator liquid volume	Nominal (950 ft^3)	Sample**
i. Accumulator line resistance	Nominal	Sample**
j. Accumulator boron	Minimum (2150 ppm)	Bounded
3.0 Accident Boundary Conditions		
a. Break location	Cold leg	Bounded
b. Break type	Guillotine (DEGCL)	Sample**
c. Break Size	Nominal (cold leg area)	Sample**
d. Offsite Power	Unavailable (RCS pumps de-energized at break initiation)	Bounded*

TABLE 15.6-2a (Cont)

KEY LBLOCA PARAMETERS AND INITIAL TRANSIENT ASSUMPTIONS

Parameter	Initial Transient	Uncertainty or Bias
3.0 Accident Boundary Conditions (Cont.)		
e. Safety injection flow	Minimum	Bounded
f. Safety injection temperature	Nominal (76°F)	Sample**
g. Safety injection delay	Max delay (40.0 sec Loop Initial Transient)	Bounded
h. Containment pressure	Bounded – See Section 6.2.1.5	Bounded
i. Single failure	ECCS: Loss of 1 SI train. Containment pressure: all trains operating	Bounded
j. Control rod drop time	No control rods	Bounded
4.0 Model Parameters		
a. Critical flow	Nominal (C_D = 1.0)	Sample**
b. Resistance uncertainties in broken loop	Nominal (as coded)	Sample**
c. Initial stored energy/fuel rod behavior	Nominal (as coded)	Sample**
d. Core heat transfer	Nominal (as coded)	Sample**
e. Delivery and bypassing of ECC	Nominal (as coded)	Conservative
f. Steam binding/entrainment	Nominal (as coded)	Conservative
g. Non-condensable bases/accumulator nitrogen	Nominal (as coded)	Conservative
h. Condensation	Nominal (as coded)	Sample**

Notes:
- * Bounded by Confirmatory Analysis for Each Plant
- ** Sampling distribution defined in Table 15.6-3b
- *** LBLOCA is currently limited to 583.5°F to account for updated TCD assessment.
- **** LBLOCA is currently limited to 5% to account for updated TCD assessment.

TABLE 15.6-3

LBLOCA CONFIRMATORY CASES PCT RESULTS SUMMARY

Results from WCOBRA/TRAC Confirmatory Studies for Byron/Braidwood Unit 1	
Transient Description	**PCT (°F)**
Initial Transient (High T_{avg}, High SGTP, High PLOW, LOOP)	1660
SGTP Confirmatory Transient (High T_{avg}, Low SGTP, High PLOW, LOOP)	1616
PLOW Confirmatory Transient (High T_{avg}, High SGTP, Low PLOW, LOOP)	1601
No-LOOP Confirmatory Transient (High T_{avg}, High SGTP, High PLOW, no-LOOP)	1412
T_{avg} Confirmatory Transient (Low T_{avg}, High SGTP, High PLOW, LOOP)	1451
Reference Transient (High T_{avg}, High SGTP, High PLOW, LOOP)	**1660**

Results from WCOBRA/TRAC Confirmatory Studies for Byron/Braidwood Unit 2	
Transient Description	**PCT (°F)**
Initial Transient (High T_{avg}, High SGTP, High PLOW, LOOP)	1578
SGTP Confirmatory Transient (High T_{avg}, Low SGTP, High PLOW, LOOP)	1543
PLOW Confirmatory Transient (High T_{avg}, High SGTP, Low PLOW, LOOP)	1609
No-LOOP Confirmatory Transient (High T_{avg}, High SGTP, High PLOW, no-LOOP)	1485
T_{avg} Confirmatory Transient (Low T_{avg}, High SGTP, High PLOW, LOOP)	1509
Reference Transient (High T_{avg}, High SGTP, Low PLOW, LOOP)	**1609**

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TABLE 15.6-3a

OVERALL LARGE BREAK LOCA RESULTS FOR THE BYRON/BRAIDWOOD NUCLEAR STATIONS

Result for Byron/Braidwood Unit 1		
	IFBA	**NON-IFBA**
95/95 PCT (^{o}F)	1912	1913
95/95 LMO (%)	5.51	5.51
95/95 CWO (%)	N/A	0.25

Result for Byron/Braidwood Unit 2		
	IFBA	**NON-IFBA**
95/95 PCT (^{o}F)	2036	2041
95/95 LMO (%)	7.02	8.27.51
95/95 CWO (%)	N/A	0.33

PCT = Peak Cladding Temperature
LMO = Local Maximum Oxidation
CWO = Core Wide Oxidation

TABLE 15.6-3b

PLANT OPERATING RANGE ALLOWED BY THE
BEST ESTIMATE LARGE BREAK LOCA ANALYSIS
(BYRON/BRAIDWOOD NUCLEAR STATIONS)

Parameter		As-Analyzed Value or Range
1.0	**Plant Physical Description**	
	a) Dimensions	Nominal
	b) Pressurizer location	On an intact loop(3)
	c) Hot assembly location	Anywhere in core(1)
	d) Hot assembly type(2)	17X17 Vantage+ fuel with, ZIRLO™ cladding, non-IFBA or IFBA(4)
	e) Steam generator tube plugging level	≤ 5% (U1) or ≤ 5%(5) (U2)
	f) Fuel assembly type(2)	17X17 Vantage+ fuel with, ZIRLO™ cladding, non-IFBA or IFBA(4)
2.0	**Plant Initial Operating Conditions**	
	2.1 Reactor Power	
	a) Core power	3658.3 MWt (including calorimetric uncertainties)
	b) Peak heat flux hot channel factor (F_Q)(2)	≤ 2.6
	c) Peak hot rod enthalpy rise hot channel factor ($F_{\Delta H}$)(2)	≤ 1.70
	d) Hot assembly radial peaking factor (P_{HA})(2)	≤1.70/1.04
	e) Hot assembly heat flux hot channel factor (F_{QHA})	≤ 2.6/1.04
	f) Axial power dist (P_{BOT}, P_{MID})(2)	Figure 15.6-7
	g) Low power region relative power (P_{LOW})(2)	$0.35 \leq P_{LOW} \leq 0.62$
	h) Hot assembly burnup	≤ 75,000 MWD/MTU, lead rod(1)
	i) MTC	≤ 0 at hot full power (HFP)
	j) Typical cycle length	18 months
	k) Minimum core average burnup(2)	≥ 10,000 MWD/MTU
	l) Maximum steady state depletion, F_Q(2)	2.1

TABLE 15.6-3b (Cont)

PLANT OPERATING RANGE ALLOWED BY THE BEST ESTIMATE LARGE BREAK LOCA ANALYSIS (BYRON/BRAIDWOOD NUCLEAR STATIONS)

Parameter		**As-Analyzed Value or Range**
	2.2 Fluid Conditions	
	a) T_{AVG}	575.0 -10°F ≤ T_{AVG} ≤ 588.0 (U1) or 583.5 (U2)(6) +10°F
	b) Pressurizer pressure	2250 – 43 psia ≤ P_{RCS} ≤ 2250 + 43 psia
	c) Loop flow	TDF ≥ 92,000 gpm/loop
	d) Upper head design	T_{COLD}
	e) Pressurizer level	60.0% of span
	f) Accumulator temperature	60°F ≤ T_{ACC} ≤ 130°F
	g) Accumulator pressure	587 psia ≤ P_{ACC} ≤ 692 psia
	h) Accumulator liquid volume	920 ft^3 ≤ V_{ACC} ≤ 980 ft^3
	i) Accumulator fL/D	Current Line Configuration
	j) Minimum accumulator boron	2150 ppm
3.0	**Accident Boundary Conditions**	
	a) Minimum safety injection flow	Table 15.6-3d
	b) Safety injection temperature	32°F ≤ SI Temp ≤ 120°F
	c) Safety injection delay	27 seconds (with offsite power) 40 seconds (with LOOP)
	d) Containment modeling	See Section 6.2.1.5
	e) Minimum containment pressure	14.2 psia
	f) Containment spray initiation delay	0 seconds
	g) Recirculation spray initiation delay	Not Applicable
	h) Single failure	Loss of one SI train

Notes:

1. 44 peripheral locations will not physically be lead power assembly.
2. In the Westinghouse Reload Safety Analysis Checklist (RSAC) process, this parameter is identified as a key safety analysis parameter that could be impacted by a fuel reload.
3. Analyzing the pressurizer as being located on an intact loop is limiting per Westinghouse methodology.
4. Analysis models thimble plugs removed, which is judged to bound plugs installed. Any combination of thimble plugs installed/removed is supported.
5. LBLOCA is currently limited to 5% to account for updated TCD assessment.
6. LBLOCA is currently limited to 583.5°F for Unit 2 to account for updated TCD assessment.

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TABLE 15.6-3c

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TABLE 15.6-3d

TOTAL MINIMUM INJECTED SAFETY INJECTION FLOW USED IN BEST-ESTIMATE LARGE-BREAK LOCA ANALYSIS FOR BYRON/BRAIDWOOD

RCS Pressure (psia)	High Head Safety Injection Flow (ft^3/sec)	Low Head and Intermediate Head Total Safety Injection - LHSI Miniflow Closed* (ft^3/sec)	Low Head and Intermediate Head Total Safety Injection – LHSI Miniflow Open* (ft^3/sec)
14.7	0.671	6.620	5.15
34.7	0.666	6.091	4.621
54.7	0.661	5.348	3.878
74.7	0.657	4.169	2.699
94.7	0.652	2.682	1.212
106.9	0.648	0.867	0.867
120.0	0.644	0.860	0.860
214.7	0.617	0.812	0.812
314.7	0.592	0.760	0.760
414.7	0.567	0.707	0.707
≥ 414.8	0.000	0.000	0.000

Note:

*LHSI Miniflow valve is conservatively modeled to close 15 seconds following the SI delay signal.

TABLE 15.6-3e

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TABLE 15.6-3f

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TABLE 15.6-4

SMALL BREAK LOCA RESULTS FOR ZIRLO FUEL CLADDING - NOTRUMP EM-

UNIT 1 NOMINAL TAVG

Results	1.5 Inch	2 Inch	3 Inch	4 Inch
Peak Clad Temperature (°F)	765	1570	1514	1428
Peak Clad Temperature Times (s)	22680.6	3434.5	1834.7	1019.1
Peak Clad Temperature Elevation (ft)	11.25	11.75	11.5	11.25
Max. Local ZrO_2 (%)	0.01	1.17	0.89	0.29
Max. Local ZrO_2 Elevation (ft)	11.25	11.75	11.50	11.25
Core-Wide Avg. ZrO_2 (%)	0.00	0.17	0.14	0.05
Burst Time(s)	N/A	N/A	N/A	N/A
Burst Elevation (ft)	N/A	N/A	N/A	N/A

TABLE 15.6-4a

SMALL BREAK LOCA RESULTS FOR ZIRLO FUEL CLADDING - NOTRUMP EM-

UNIT 1 REDUCED TAVG

Results	1.5 Inch	2 Inch	3 Inch	4 Inch
Peak Clad Temperature ($^{\circ}F$)	731	1624	1457	1292
Peak Clad Temperature Times (s)	22894.8	3455.5	2013.5	1099.7
Peak Clad Temperature Elevation (ft)	11.50	11.50	11.5	11.25
Max. Local ZrO_2 (%)	0.00	1.47	0.62	0.11
Max. Local ZrO_2 Elevation (ft)	11.50	11.50	11.50	11.25
Core-Wide Avg. ZrO_2 (%)	0.00	0.22	0.10	0.02
Burst Time(s)	N/A	N/A	N/A	N/A
Burst Elevation (ft)	N/A	N/A	N/A	N/A

TABLE 15.6-4b

SMALL BREAK LOCA RESULTS FOR ZIRLO FUEL CLADDING - NORTUMP EM-

UNIT 2 NOMINAL TAVG

Results	1.5 Inch	2 Inch	3 Inch	4 Inch
Peak Clad Temperature (^{o}F)	912	1086	1614	1537
Peak Clad Temperature Time (s)	16234.8	2804.5	1618.5	889.0
Peak Clad Temperature Elevation (ft)	11.25	11.25	11.50	11.25
Max. Local ZrO_2 (%)	0.03	0.06	1.48	0.65
Max. Local ZrO_2 Elevation (ft)	11.25	11.25	11.50	11.25
Core-Wide Avg. ZrO_2 (%)	0.00	0.01	0.23	0.11
Burst Time(s)	N/A	N/A	N/A	N/A
Burst Elevation (ft)	N/A	N/A	N/A	N/A

TABLE 15.6-4c

SMALL BREAK LOCA RESULTS FOR ZIRLO FUEL CLADDING – NORTUMP EM-

UNIT 2 REDUCED TAVG

Results	1.5 Inch	2 Inch	3 Inch	4 Inch
Peak Clad Temperature (^{o}F)	874	1627	1452	1313
Peak Clad Temperature Time (s)	17491.8	3071.7	1805.6	992.3
Peak Clad Temperature Elevation (ft)	11.00	11.50	11.50	11.25
Max. Local ZrO_2 (%)	0.03	1.59	0.61	0.14
Max. Local ZrO_2 Elevation (ft)	11.00	11.50	11.50	11.25
Core-Wide Avg. ZrO_2 (%)	0.00	0.24	0.09	0.02
Burst Time(s)	N/A	N/A	N/A	N/A
Burst Elevation (ft)	N/A	N/A	N/A	N/A

TABLE 15.6-4d

SMALL BREAK LOCA RESULTS FOR ZIRCALOY-4 FUEL CLADDING -

NORTRUMP EM – UNIT 2 REDUCED TAVG

Results	CAE/CCE REDUCED Tavg 2 Inch	CBE/CDE NOMNIAL Tavg 3 Inch BOL	CBE/CDE NOMINAL Tavg 3 Inch BU=6K MWD/MTU
Peak Clad Temperature (°F)	1601	1615	1601
Peak Clad Temperature Time (s)	3494.5	1618.5	1624.7
Peak Clad Temperature Elevation (ft)	11.75	11.50	11.75
Max. Local ZrO_2 (%)	1.30	1.5	1.48
Max. Local ZrO_2 Elevation (ft)	11.75	11.5	11.50
Core-Wide Avg. ZrO_2 (%)	0.19	0.23	0.45
Burst Time(s)	N/A	N/A	N/A
Burst Elevation (ft)	N/A	N/A	N/A

TABLE 15.6-5

SGTR INITIAL CONDITIONS

PARAMETER	VALUE (Typical) Offsite Dose Case	VALUE MTO Case
NSSS Power	3672 Mw	3672 Mw
RCS Flow	368,000 gpm Standard Thermal Design	368,000 gpm Standard Thermal Design
RCS Pressure	2192 psig	2192 psig
RCS T_{avg}	588.0°F	580.0°F (Unit 1) 575.0°F (Unit 2)
RCS Initial Water Mass	5.48E5 lbm (Unit 1) 4.88E5 lbm (Unit 2)	Not Calculated
Percent Tube Plugging (per steam generator)	0%	5% (Unit 1) 10% (Unit 2)
S/G Initial Mass	1.37E5 lbm (Unit 1) 1.26E5 lbm (Unit 2)	1.36E5 lbm (Unit 1) 1.22E5 lbm (Unit 2)
Break Location	First Row Cold Leg at Tubesheet (Shortest Tube)	First Row Cold Leg at Tubesheet (Shortest Tube)

TABLE 15.6-5 (Cont'd)

PARAMETER	VALUE (Typical) Offsite Dose Case	VALUE MTO Case
Decay Heat	1.2 X ANS 5.1 1971	ANS 5.1 1979-2sigma
AFW Flow	302 gpm per S/G	180 gpm per SG for 40 seconds, 360 gpm/SG after 40 seconds (Unit 1). 263 gpm per SG for 40 seconds, 450.22 gpm/SG after 40 seconds (Unit 2).
AFW Temperature	120°F	32°F

TABLE 15.6-5A

SGTR OPERATOR ACTIONS

OPERATOR ACTION	VALUE Offsite Dose Case	VALUE MTO Case
Isolate AFW Flow to the Ruptured Steam Generator	5 Minutes* After Tube Rupture Occurs Provided Level is Adequate	9 Minutes After Tube Rupture Occurs
Isolate MSIV for the Ruptured Steam Generator	18 Minutes After Tube Rupture Occurs	18 Minutes After Tube Rupture Occurs
Isolate the Failed Open PORV one the Ruptured Steam Generator	30 Minutes After PORV Fails Open	----------
Initiate RCS Cooldown	3 Minutes After Failed PORV is Isolated	3 Minutes After MSIV Isolation
Initiate RCS Depressurization	4 Minutes After Cooldown is Completed	4 minutes After Cooldown is Completed
Terminate ECCS Flow	3 Minutes After Depressurization is Completed	3 Minutes After Depressurization is Completed

*The assumption of a minimum of 5 minutes from event initiation until AFW isolation used in the input to dose analyses is not a critical operator action time and does not impose a requirement on the operators.

TABLE 15.6-6

INPUT PARAMETERS FOR THE SGTR RADIOLOGICAL CONSEQUENCE ANALYSIS USING AST

Parameter	Unit	Value
Accident mitigation recovery phases associated with the sequence of events:		
Start of event until Reactor trip	seconds	0 – 160
Reactor Trip until ruptured SG PORV fails open	seconds	160 – 1082
Ruptured SG PORV failure until 15 minutes later	seconds	1082 – 1982
15 minutes after Ruptured SG PORV failure until failed ruptured SG PORV is isolated	seconds	1982 – 2882
Ruptured SG PORV Isolation until RCS cooldown is initiated	seconds	2882 – 3062
RCS cooldown initiation until Break flow flashing ceases	seconds	3062 – 3390
End of Break flow flashing until RCS depressurization initiation	seconds	3390 – 5164
RCS depressurization initiation until all Break Flow ceases	seconds	5164 - 6234
Iodine Spike Release Period	hours	8
Full power steam release through the Condenser until the time of reactor trip and loss of offsite power		
Ruptured SG	lbm	1.830E5
Intact SGs	lbm	5.445E5
Primary and secondary coolant masses:		
Ruptured SG	lbm	6.89E4
Intact SGs	lbm	2.06E5
RCS	lbm	4.88E5

TABLE 15.6-6a

SEQUENCE OF EVENTS FOR SGTR (TYPICAL OFFSITE DOSE CASE)

SYSTEM RESPONSE/OPERATOR ACTION	Unit 1 Time (sec)	Unit 2 Time (sec)
SG Tube Rupture Occurs	0	0
Reactor Trip	217	160
AFW Actuated	280	223
AFW Flow to the Ruptured SG Isolated	300	300
SI Actuated	387	305
Ruptured SG MSIV Closed	1080	1080
Ruptured SG PORV Fails Open	1082	1082
Ruptured SG PORV Isolated	2882	2882
RCS Cooldown Initiated	3062	3062
Break Flow Flashing Terminated	3304	3390
RCS Cooldown Terminated	4284	4922
RCS Depressurization Initiated	4526	5164
RCS Depressurization Terminated	4618	5250
SI Flow Terminated	4798	5430
Break Flow Terminated	5478	6234

TABLE 15.6-6b

SEQUENCE OF EVENTS FOR SGTR (MARGIN TO OVERFILL CASE)

SYSTEM RESPONSE/OPERATOR ACTION	Unit 1 Time (sec)	Unit 2 Time (sec)
SG Tube Rupture Occurs	0	0
Reactor Trip	200	139
AFW Actuated	201	140
SI Actuated	474	317
AFW Flow to the Ruptured SG Isolated	540	540
Ruptured SG MSIV Closed	1080	1080
RCS Cooldown Initiated	1260	1260
RCS Cooldown Terminated	1838	1958
RCS Depressurization Initiated	2080	2200
RCS Depressurization Terminated	2178	2302
SI Flow Terminated	2359	2482
Break Flow Terminated	3360	3258

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TABLE 15.6-7

INPUT PARAMETERS FOR THE LOCA RADIOLOGICAL CONSEQUENCES ANALYSIS USING AST

Parameter	Unit	Value	Notes
Core Release Fraction: Iodine Noble Gases	 - -	Per Table 2 in RG 1.183	
Iodine Plate-out Fraction on Containment Surfaces	-	Time-dependent	Powers natural deposition model in RADTRAD
Initial Iodine Species in Containment Aerosol Elemental Organic	 % % %	 95 4.85 0.15	
Iodine released from ECCS to the environment: Elemental Organic	 % %	 97 3	
Containment Leak Rate: 0-24 Hours > 24 Hours	 weight %/day	 0.20 0.10	The reduction in the design basis containment leak rate by 50% at 24 hours is consistent with guidance of RG 1.183.
Containment Spray (CS) Flow Actuation Time Including Delay	seconds	90	Biased high for conservatism based on 88.1 or 53.1+X* where X* denotes the time at which the containment pressure setpoint is reached (High-3 for CS).
Spray Injection Flow Duration	minutes	20.9	This value is based on containment spray switchover at lo-3 RWST level, maximum SI, and one containment spray pump running (because the analysis assumes the failure of one containment spray pump). The duration is from spray initiation.
Containment Spray Parameters: Containment Spray Flow Injection Recirculation	 gpm gpm	 2950 2950	The minimum amount of delivered flow based on the minimum delivered flow for 1 spray pump is 3113 gpm. For conservatism, a value of approximately 95 % is used.
Fraction of Containment Sprayed	-	0.825	The value is based on the most conservative approach of minimum net sprayed containment volume and maximum net containment volume. Since the same flow rate is used during injection and recirculation phases, this value is also valid for both phases.
Containment Volume Sprayed Unsprayed	ft^3 ft^3 ft^3	2.85E6 2.35E6 5.0E5	
Containment Spray Fall Height	ft	141	This value bounds the minimum height between the spray ring header (elevation 567 feet), and the operating floor (elevation 426 feet).

TABLE 15.6-7 (Cont'd)

INPUT PARAMETERS FOR THE LOCA RADIOLOGICAL CONSEQUENCES ANALYSIS USING AST

Parameter	Unit	Value	Notes
Containment Spray Temperature Nominal (norm. op.) Containment Temperature Maximum during LOCA	°F °F	120 260	Conservatively greater than maximum RWST temperature in Technical Specification 3.5.4.
Number of Deck Fans Operating	#	2 of 4	
Deck Fan Flow Rate, per fan	cfm	65,000	Based on minimum heat removal and service water temperature of 100 ºF.
Time from Spray Cessation Before Spray Recirculation Starts	min	10	
ECCS Recirculation Leakage	cc/hr	276,000	
Recirculation Loop Water Volume at 11.6 minutes	ft^3	38,979	This value is considered to be the minimum containment sump volume at ECCS switchover. The proposed value assumes the RCS volume for unit 2, which bounds unit 1 and takes into account maximum steam generator tube plugging level. At CS switchover, an additional 19,527 ft^3 should be added, for a total of 58,506 ft^3.
Fraction of core iodine in sump solution	-	0.4	
Iodine partition coefficient for recirculation leakage	-	0.1	
Sump Water Temperature	ºF	260	
Auxiliary Building Release Path Iodine Removal Efficiency: Elemental iodine Organic iodine Particulates	 % % %	 90 90 99	Flashed fluid Iodine is assumed to contain no aerosol.
Passive failure release from the ECCS recirculation path	-	N/A	Since credit is taken for safety grade filtration system in the auxiliary building for ECCS leakage, the passive failure at 24 hours is not applicable.
Offsite Dose Acceptance Criteria Analysis Release Duration	days	30	

TABLE 15.6-8

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TABLE 15.6-9

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TABLE 15.6-9 (Cont'd)

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TABLE 15.6-10

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TABLE 15.6-11

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TABLE 15.6-12

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TABLE 15.6-13

RECIRCULATION LOOP LEAKAGE EXTERNAL TO CONTAINMENT

ITEMS	TYPE OF LEAKAGE CONTROL AND UNIT LEAKAGE RATE USED IN THE ORIGINAL ANALYSIS	LEAKAGE TO ATMOSPHERE VIA LOCAL LEAKAGE (cc/hr)	LEAKAGE TO ATMOSPHERE VIA VENTED DRAIN TANK (cc/hr)
1. Residual heat removal (low head safety injection)	Mechanical seal 10 cc/hr/seal* (Note 1) (while pump is in operation)	20	0
2. Safety injection pumps	Same as residual heat removal pump	40	0
3. Charging Pumps	Same as residual heat removal pump	40	0
4. Flanges:			
a. Pumps	Gasket - adjusted to zero leakage following any test 10 drops/min/gauge used (30 cc/hr). Due to leaktight flanges on pumps, no leakage is assumed to atmosphere	0	0
b. Valves bonnet to body (larger than 2 inches)		2400	0
c. Control valves		480	0
d. Heat exchangers		240	0
5. Valves - stem leakoffs	Backed seated double packing with leakoff- 1 cc/hr/in. of steam diameter used	0	50
6. Miscellaneous small valves	Flanged boyd packed stems - 1 drop/min used (3 cc/hr)	600	0
7. Miscellaneous large valves (Larger than 2 inches)	Double packing 1 cc/hr/in. stem diameter used	40	0
		3860	50
		Total - 3910 cc/hr (Note 2)	

Note 1 *During equipment shutdown, mechanical seals may develop minor leakage. However, this leakage typically stops when the pump is returned to operation.

Note 2 The maximum permitted recirculation loop leakage is 138,000 cc/hr. The design basis accident analysis assumes a value two times the maximum permitted recirculation loop leakage (i.e., 276,000 cc/hr) in accordance with the Standard Review Plan.

TABLE 15.6-14

TABLE 15.6-15

PEAK CLAD TEMPERATURE INCLUDING ALL PENALTIES AND BENEFITS

ANALYSIS	PEAK CLAD TEMPERATURE (°F)
Large Break LOCA	
Byron Unit 1	2025
Byron Unit 2	2047
Braidwood Unit 1	2025
Braidwood Unit 2	2047
Small Break LOCA	
Byron Unit 1	1749
Byron Unit 2	1755
Braidwood Unit 1	1749
Braidwood Unit 2	1755

15.7 RADIOACTIVE RELEASE FROM A SUBSYSTEM OR COMPONENT

Events which may result in a radioactive release from a subsystem or component are as follows:

a. waste gas system leak or failure (Subsection 15.7.1),

b. liquid waste system leak or failure - release to atmosphere (Subsection 15.7.2),

c. liquid tank failure - ground release (Subsection 15.7.3),

d. fuel handling accidents (Subsection 15.7.4),

e. spent fuel cask drop accidents (Subsection 15.7.5).

15.7.1 Gas Waste System Leak or Failure

15.7.1.1 Identification of Causes and Accident Description

The gaseous waste processing system, as discussed in Chapter 11.0, is designed to remove fission product gases from the reactor coolant. The system consists of a closed loop with waste gas compressors, hydrogen recombiners, waste gas decay tanks for service at power, and other waste gas decay tanks for service at shutdown and startup.

The accident is defined as the postulated uncontrolled release of the xenon and krypton fission product gases stored in a waste decay tank as a consequence of a failure of a single gas decay tank or associated piping.

15.7.1.2 Analysis of Effects and Consequences

The maximum noble gas activities in the gas decay tank assumed to fail, the assumptions on which the activities are based, and the offsite dose consequences of releasing the activity are discussed in Subsection 15.7.1.3.

15.7.1.3 Radiological Consequences of a Postulated Waste Gas Decay Tank Rupture

The analysis of the postulated waste gas decay tank rupture is performed based on Regulatory Guide 1.24 (May 1972). The parameters used for this analysis are listed in Table 15.7-1.

The assumptions for the Regulatory Guide 1.24 analysis are:

a. The reactor has been operating at full power with 1% defective fuel and a shutdown to cold condition has been conducted near the end of an equilibrium core cycle. As soon as possible after shutdown, all noble gases have been removed from the reactor coolant system and transferred to the gas decay tank that is assumed to fail.

The inventory of noble gases in the primary coolant conservatively assumes that the unit has been operated for a full cycle without the removal of fission gases from the primary coolant, thus maximizing the coolant noble gas inventory. This coolant noble gas concentration is provided in Table 15.0-9.

b. Radiological decay is taken into account in the computation only for the minimum time period required to transfer the gases from the reactor coolant system to the decay tank. The noble gas inventories of the tank are given in Table 15.7-2.

c. The tank rupture is assumed to occur immediately upon completion of the waste gas transfer, releasing the entire contents of the tank at ground level to the outside atmosphere. The assumption of the release of the noble gas inventory from only a single tank is based on the fact that all gas decay tanks will be isolated from each other whenever they are in use.

d. The 5th percentile atmospheric, diffusion factors given in Tables 15.0-13 and 15.0-14 are used to evaluate the doses from the released activity. Doses are based on the dose models presented in Attachment 15A.

The doses at the exclusion area boundary and the LPZ resulting from this accident are listed in Tables 15.0-11 and 15.0-12. These doses are well within 10 CFR 100 limits.

15.7.2 Radioactive Liquid Waste System Leak or Failure (Atmospheric Release)

15.7.2.1 Identification of Causes and Frequency Classification

An unspecified event causes the complete release of the worst case radionuclide inventory in the tanks containing the largest quantities of significant radionuclides in the liquid radwaste system. These are the spent resin storage tank and the boron recycle holdup tanks in the auxiliary building. The airborne radioactivity release during the accident passes directly to the environment via the station vent stack.

Postulated events that could cause release of the radioactive inventory of the spent resin storage tank and the boron recycle holdup tanks are cracks in the tanks and operator error. The possibility of small cracks and consequent low-level release rates receives primary consideration in system and component design. The spent resin storage tank is designed to operate at a pressure of 125 psi and 150°F maximum temperature. The boron recycle holdup tank is designed for a pressure of 15 psig and a temperature of 200°F. The possibility of failure of either tank is considered small. A radioactive liquid release from either of these tanks caused by operator error is also considered a remote possibility. Operating techniques and administrative procedures emphasize detailed system and equipment operating instructions. Should a release of radioactive liquid occur, floor drain sump pumps in the floor of the auxiliary building will receive a high water level alarm, activate automatically, and remove the spilled liquid. This accident is expected to occur with the frequency of a limiting fault.

The probability of a complete rupture of the radwaste system or a complete malfunction accident is considered to be much lower than the rupture of a single tank, as described above. This accident is not analyzed.

Although not built to the standards of Seismic Category I equipment, the liquid radwaste system tanks are constructed in accordance with sound engineering principles and the ASME Code. Therefore, simultaneous failure of all the tanks is not considered credible.

15.7.2.2 Sequence of Events and Systems Operation

The sequence of events expected to occur are as follows;

Sequence of Events	Elapsed Time
1. Event begins--failure occurs	0
2. Area radiation alarms alert plant personnel	1 minute
3. Operator actions begin	5 minutes

The rupture of the spent resin storage tank or the boron recycle holdup tank would leave little recourse to the operator. No method of recontaining the gaseous phase discharge is available, however, isolation of the auxiliary building would minimize the results. High radiation alarms both in the ventilation exhaust and in the auxiliary building would alert the operator to the failure. No credit is taken for operator action to isolate the auxiliary building ventilation system in evaluating these events.

15.7.2.3 Modeling of Accident Sequence

15.7.2.3.1 Mathematical Models

a. Spent Resin Storage Tank

The release from the station vent stack is considered a ground level release with accident atmospheric dilution factors calculated by the methods discussed in Reference 1. Doses are calculated at the exclusion area boundary and the LPZ boundary using the fifth percentile χ/Q. Values are given in Tables 15.0-13 and 15.0-14.

The atmospheric dilution factor is assumed constant throughout the duration of the accident. Doses are calculated using the models of Attachment 15A.

b. Boron Recycle Holdup Tank

The doses due to a Boron Recycle Tank failure were calculated using the Alternate Source Term methodology using Regulatory Guide 1.183 with dose acceptance criteria per 10 CFR 50.67. The RADTRAD computer program was used to model the associated radiological releases.

15.7.2.3.2 Input Parameters and Initial Conditions

The tank failures are evaluated in accordance with the following sets of assumptions and conditions:

a. Spent Resin Storage Tank

1. One hundred percent of the liquid volume of a spent resin storage tank is released into the spent resin storage tank cubicle.

2. The radionuclide inventory is based on a primary coolant concentration of 1.0 μCi/g of Dose Equivalent I-131 (see Table 15.0-10) and letdown demineralizer resins. The tank is assumed to be filled to 80% of capacity with resin.

3. The partition factor for radioiodine from water to air is 0.1 and 0.01 for resin to water.

4. The airborne radioactivity release into the spent resin storage tank cubicle passes to the environment via the station vent stack.

5. The accident (χ/Q) values are for a ground level release as described in Subsection 15.7.2.3.1 and presented in Tables 15.0-13 and 15.0-14.

b. Boron Recycle Holdup Tank

A rupture of the Boron Recycle Holdup Tank (HUT) could potentially release activity from the following:

- Activity in the water stored in the ruptured HUT.
- Activity in the water stored in the un-ruptured HUT that could drain via the cross-tie piping that interconnects the two tanks.
- Activity in gas spaces from both HUTs. The gas spaces in both HUTs are connected to a common header.
- Activity from one waste gas decay tank. As described in UFSAR Section 11.3.2.5, one waste gas decay tank is normally aligned to provide cover gas for the HUTs.
- Activity from the discharge of RCS fluid from a Residual Heat Removal (RHR) suction relief valve.

The following assumptions and inputs are used in the dose analysis:

1. The iodine activity in the primary coolant is the equilibrium concentration based on 1.0 μCi/g dose equivalent I-131 as indicated in UFSAR Table 15.0-10.
2. The primary coolant noble gas activity is based on 1% failed fuel. Isotope activity concentrations are indicated in UFSAR Table 15.0-9.
3. The HUT rupture event is evaluated for two cases, minimum and maximum initial water level.
4. The letdown demineralizer is assumed to have a decontamination factor of 10, so that the initial water collected in the HUT has one tenth of the RCS iodine concentration.
5. The initial noble gas activity in the combined air space of the two HUTs at minimum initial water volume is calculated by assuming that the entire inventory of noble gas in the RCS is transferred to the HUTs. This noble gas concentration is also assumed for the maximum initial water volume case.
6. In addition to the water initially in the HUTs, the calculated dose assumes the ruptured HUT also releases RCS water from a RH suction relief valve discharge for one hour. The RCS water from a RH suction relief valve discharges directly into one of the HUTs. Therefore, no decontamination factor is credited.

7. The noble gas in one waste gas decay tank is postulated to be released to the environment along with the noble gas released from the HUT air space. The noble gas activity is based on a maximum 50,000 curies of dose equivalent Xe-133 in the waste gas decay tank.

8. Following the rupture all of the water in both HUTs, the RH suction relief valve water, all of the noble gas in both HUTs, and the gas in one waste gas decay tank is released to the HUT cubicle.

9. An iodine flashing factor (FF) of 10% is assumed for the liquid released from the HUTs.

10. All of the activity is released to the environment within a 5 minute time period following the HUT rupture.

11. The accident χ/Q values (using finer wind speed categories from Regulatory Guide 1.23 Revision 1) as described in Subsection 2.6.3.3 and presented in Table 15.0-17.

12. The Main Control Room ventilation system is assumed to be manually realigned to the emergency mode of operation 30 minutes after the initiation of the accident.

15.7.2.3.3 Results

The conservative assessment of the tank failures leads to consideration of iodine partitioning from a spill which could be drained in a rapid manner, thereby minimizing the release of iodine to the air.

The auxiliary building exhaust is filtered through HEPA filters for which no reduction in released iodine has been assumed.

Because of the design features incorporated in the Byron/Braidwood Stations, i.e., as most of the tanks are each located in vented and drained cells, the failure of a tank will not result in significant increases in in-plant doses.

a. Spent Resin Storage Tank

   Table 15.7-3 lists the contained radionuclide activity for the spent resin storage tank. Tables 15.0-11 and 15.0-12 list the doses at the exclusion area boundary and the LPZ boundary for the release of the spent resin storage tank. These doses are well within 10 CFR 100 limits.

b. Boron Recycle Holdup Tank

   The minimum initial water level case is bounding for the offsite doses. The maximum initial water level case is bounding for the control room dose. The results and regulatory dose limits are provided in UFSAR Tables 15.0-11 and 15.0-12.

15.7.3 Postulated Radioactive Release Due to Liquid Tank Failure (Ground Release)

15.7.3.1 Identification of Causes of Frequency Classification

This accident is defined as an unexpected and uncontrolled postulated rupture of the boron recycle holdup tank. This tank is located in the Seismic Category I auxiliary building at elevation 346.0 feet. Since grade elevation is 400.0 feet, the only way any effluents can be released accidentally is through postulated cracks in the auxiliary building which would allow the contents of the tank to enter groundwater. This accident is expected to occur with the frequency of a limiting fault (see Subsection 15.7.2.1).

15.7.3.2 Sequence of Events and Systems Operation

See Subsection 2.4.12.

15.7.3.3 Modeling of Accident Sequence

15.7.3.3.1 Mathematical Model

Subsection 2.4.12 gives the dispersion, dilution, and travel times of accidental releases of liquid effluents in surface water.

15.7.3.3.2 Input Parameters and Initial Conditions

The tank failure is evaluated in accordance with the following sets of assumptions and conditions:

a. One hundred percent of the liquid volume of the boron recycle holdup tank is released into the boron recycle holdup tank cubicle.

b. The liquid enters the groundwater environment through postulated cracks in the auxiliary building.

15.7.3.4 Radiological Consequences

UFSAR Section 2.4.12 provides an evaluation of an accidental release of effluents through postulated cracks in the auxiliary building.

The concentrations of any postulated accidental release of radioactive effluents from the boron recycle holdup tank would not exceed 10 CFR 20 limits at the nearest surface water intake.

15.7.4 Fuel Handling Accidents

15.7.4.1 Accident Description

The accident is defined as dropping of a spent fuel assembly onto the spent fuel pool floor or onto the core resulting in the postulated rupture of the cladding of all the fuel rods in the assembly despite many administrative controls and physical limitations imposed on fuel handling operations. All refueling operations are conducted in accordance with prescribed procedures under direct surveillance of a supervisor.

15.7.4.2 Analysis of Effects and Consequences

The reactor is assumed to have been operating at 3658.3 MWt. The assembly inventory assumes that the reactor has been operating at this power level up to the end of the cycle. It is assumed that 48 hours has passed from shutdown to the beginning of fuel handling operations. The resulting source term in an average fuel assembly is given in Table 15.7-6.

For the fuel handling accident it is conservatively assumed that the damaged fuel assembly has been operating at 1.7 times average assembly power (this is the lead rod peaking factor) and the activity in the damaged assembly is 1.7 times the values in Table 15.7-6. Use of this peaking factor is conservative since it would not occur in more than a small fraction of the rods in the assembly. The gap model discussed in Regulatory Guide 1.25 (May 1972), as modified by the recommendations of NUREG/CR-5009 (Reference 1), is used to determine the activity that is available for release from the damaged fuel. The gap fractions are listed in Table 15.7-7.

15.7.4.2.1 Fuel Handling Accident Inside Spent Fuel Storage Building

The accident is defined as the dropping of a spent fuel assembly in the spent fuel pool resulting in the rupture of the cladding of fuel rods.

15.7.4.2.1.1 Identification of Causes and Frequency Classification

The cause of the event can be identified as any mechanical failure or operating error which results in the dropping of a fuel assembly into the refueling pool during its transfer from one position in the pool to another.

The frequency classification, as defined in Regulatory Guide 1.70, can be categorized as one of the limiting faults. This means that it is an occurrence that is not expected to occur but is postulated because its consequences would include the potential for the release of significant amounts of radioactive material.

15.7.4.2.1.2 Sequence of Events and Systems Operation

The bounding fuel handling accident analysis assumes that the fuel handling building and containment are not isolated at the time of the event and remain open to the environment during the event. Control room HVAC system filtration is delayed for the first 30 minutes of the event for conservatism.

As indicated in Regulatory Guide 1.183, if the fuel handling building and containment are not isolated at the time of the event, the release of all activity to the environment is assumed to occur over a 2-hour time period. To assure this, building exhaust rates are set artificially high in the analysis.

The bounding exhaust point is the plant vent, with specific atmospheric dispersion factors (χ/Qs) for the control room or offsite receptor locations. Alternative release points through major openings such as the fuel handling building inner or outer rail bay roll-up doors or the personnel/equipment hatch from the containment would have lower χ/Qs and would therefore result in lower calculated doses.

15.7.4.2.1.3 System Operation

During handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical core within the previous 48 hours), the fuel handling building ventilation system is required to be operable. The normal supply system is designed to provide 21,000 cfm of outside air to the fuel handling building general area. The exhaust inlets are located at the pool edge. The shortest distance between the exhaust inlets and the inboard isolation valve is 222 feet.

During movement of irradiated fuel in the fuel handling building, the following compensatory measures are required per the Alternative Source Term license amendment:

a. At least one train of the fuel handling building ventilation system will be available.

b. At least one fuel handling building trackway door will be maintained closed.

c. Within one hour of a fuel handling accident, airflow in the proper direction will be established in the fuel handling building.

d. Containment and fuel handling building effluents will be monitored consistent with the Technical Specification Radioactive Effluent Controls Program.

15.7.4.2.1.4 Radiation Monitoring System

Redundant GM type gamma detectors are mounted on the walls near the edge of the pool to provide reliable and rapid detection of radioactivity released from the pool surface.

If predetermined levels are exceeded, the monitors alarm locally and in the main control room and initiate control action to route the released activity through the emergency exhaust system.

The monitors have an operating range which extends from 0.1 to 10^4 mR/hr. The lower range level is chosen to assure that normal operating levels are on scale (provides indication that the instrument is operational). Operating levels below 0.1 or greater than 50 mR/hr are unlikely. Initial setpoints are listed in Table 12.3-3.

The worst case fuel handling accident has the potential of exceeding the 10 R/hr maximum range of the fuel handling accident monitors, but this environment will not prevent the monitor from completing its design function. General Atomics (GA) has tested this monitor to 500 R/hr, and based on this test, they have determined that this monitor will perform its function up to 1000 R/hr.

The monitor has been selected to assure initiation of control action within 6 seconds or less. Commercially available area monitors are suitable for this application.

Two separate and independent (nuclear safety-related) monitors are provided for the spent fuel pool. Two nuclear safety-related recorders are provided in the control room for the spent fuel pool.

15.7.4.2.2 Fuel Handling Accident Inside Containment

This accident is defined as the dropping of a spent fuel assembly onto the core during refueling which results in the rupture of the cladding of fuel rods.

15.7.4.2.2.1 Identification of Causes and Frequency Classification

The cause of the event can be identified as any event which would result in the dropping of a fuel assembly onto the reactor core during refueling.

The frequency classification, as defined in Regulatory Guide 1.70, can be categorized as one of limiting faults. This means that it is an occurrence that is not expected to occur but is postulated because its consequences would include the potential for the release of significant amounts of radioactive material.

15.7.4.2.2.2 Sequence of Events and Systems Operation

The bounding fuel handling accident analysis assumes that the fuel handling building and containment are not isolated at the time of the event and remain open to the environment during the event. Control room HVAC system filtration is delayed for the first 30 minutes of the event for conservatism.

As indicated in Regulatory Guide 1.183, if the fuel handling building and containment are not isolated at the time of the event, the release of all activity to the environment is assumed to occur over a 2-hour time period. To assure this, building exhaust rates are set artificially high in the analysis.

The bounding exhaust point is the plant vent, with specific atmospheric dispersion factors (χ/Qs) for the control room or offsite receptor locations. Alternative release points through major openings such as the fuel handling building inner or outer rail bay roll-up doors or the personnel/equipment hatch from the containment would have lower χ/Qs and would therefore result in lower calculated doses.

15.7.4.2.2.3 System Operation

During handling of recently irradiated fuel assemblies in containment (i.e., fuel that has occupied part of a critical reactor core within the previous 48 hours), the Technical Specification requirements for containment closure ensure that release of fission product radioactivity within containment will be restricted from escaping to the environment.

During movement of irradiated fuel in the containment, the following compensatory measures are required per the Alternative Source Term license amendment:

a. At least one train of the fuel handling building ventilation system will be available.

b. At least one fuel handling building trackway door will be maintained closed.

c. Within one hour of a fuel handling accident, with the equipment hatch off or with the equipment hatch installed and both personnel air lock doors open, airflow in the proper direction will be established in the containment/fuel handling building and all other penetrations in the containment to the outside air will be closed.

d. Within one hour of a fuel handling accident, with the equipment hatch installed and at least one of the personnel air lock doors closed, all other penetrations in the containment to the outside air will be closed.

e. Containment and fuel handling building effluents will be monitored consistent with the Technical Specification Radioactive Effluent Controls Program.

15.7.4.2.2.4 Radiation Monitoring System

Redundant GM type gamma detectors are mounted on the shield walls near the edge of the refueling cavity to provide reliable and rapid detection of radioactivity released from the pool surface. If predetermined levels are exceeded, the monitors alarm in the main control room and initiate control action to close the purge system isolation valves.

The monitors have an operating range which extends from 0.1 to 10^4 mR/hr. The lower range level is chosen to assure that normal operating levels are on scale (indicates that the instrument is operational). Operating levels below 0.1 or greater than 50 mR/hr are unlikely. Initial setpoints are listed in Table 12.3-3.

The worst case fuel handling accident has the potential of exceeding the 10 R/hr maximum range of the fuel handling accident monitors, but this environment will not prevent the monitor from completing its design function. General Atomics (GA) has tested this monitor to 500 R/hr, and based on this test, they have determined that this monitor will perform its function up to 1000 R/hr.

The monitor has been selected to assure initiation of control action within 6 seconds or less. Commercially available area monitors are suitable for this application.

Two separate and independent (nuclear safety-related) monitors are provided for each reactor unit. Unit 1 is provided with two nuclear safety-related recorders for the fuel building fuel handling accident and the containment building fuel handling accident. Unit 2 is provided with two nuclear safety-related recorders for the containment building fuel handling accident. All recorders are located in the control room for each unit.

15.7.4.3 Radiological Consequences of a Postulated Fuel Handling Accident (AST)

The key inputs and assumptions used in the AST Fuel Handling Accident (FHA) radiological consequence analysis are summarized below and provided in Tables 15.7-6 and Table 15.7-7.

This event is defined as the drop of a spent fuel assembly onto the spent fuel pool floor or the core, resulting in the postulated rupture of the cladding of all the fuel rods in one assembly. Consistent with RG 1.183, two potential accident locations were considered; the FHB and the containment. The dropped fuel assembly is assumed to have been subcritical for ≥48 hrs. The RADTRAD computer program was used to model the associated radiological releases.

The assumed fuel assembly source term is based on the reactor core source terms previously described in the LOCA section. The fraction of the core fuel damaged is based on a postulated rupture of all the fuel rods in one assembly. With 193 fuel assemblies in the core operating at full core power of 3658.3 MWth, the damaged fuel assembly is assumed to have been operating with a 1.7 peaking factor; therefore, the damaged fuel assembly power = 3658.3 MWth x 1.7/193 = 32.22 MWth.

The analysis assumes 23 feet of water above damaged fuel. This value corresponds to the minimum depth of water coverage over the top of irradiated fuel assemblies seated in the spent fuel pool racks as required by TS 3.7.14, "Spent Fuel Pool Water Level", and is also assumed for an assembly drop in the core. TS 3.9.7, "Refueling Cavity Water Level," requires maintaining at least 23 feet of water above the top of the reactor vessel flange during movement of irradiated fuel assemblies within containment. Since the fuel assemblies are seated below the reactor vessel flange, in actuality, more than 23 feet of water covers the assemblies in the core; therefore, this assumption is conservative for an assembly drop in the core. This assumption is consistent with RG 1.183. As prescribed in Appendix C of RG 1.183, an overall DF of 200 is used as the overall effective iodine DF for this 23-foot water depth, with a DF of 1 for noble gases. Particulate radionuclides are assumed to be retained in the pool water (i.e., a DF of infinity).

RG 1.183, Table 3 allows application of the following gap activity fractions for non-LOCA events. These gap activities apply to fuel whose burnup and power are bounded by those specified in RG 1.183, footnote 11.

- 5% of the noble gases (excluding Kr-85)
- 10% of the Kr-85
- 5% of the iodine inventory (excluding I-131)
- 8% of the I-131
- 12% of the alkali metal inventory

Both Byron and Braidwood Stations may utilize some fuel assemblies with linear heat generation rates in excess of RG 1.183 footnote 11 limits. To account for high burnup assemblies, the above gap activity fractions were doubled.

Release modeling uses the refueling floor air space (i.e., in the FHB), with the initial air change rate based on the 525,460 cubic feet (86′ x 130′ x 47′ high) volume "exposed to the monitor" as developed for the post-accident radiation monitor response time calculation, divided into the spent fuel storage pool total ventilation exhaust rate of 12,400 cfm. The initial air change rate is therefore 0.0236 per minute, assumed to last for the entire period until initiation of the CR emergency mode of operation. Consistent with RG 1.183, the release from the FHB to the environment is assumed over a two-hour time period. To assure this, the refueling floor exhaust rate is set artificially high at five times this value or 0.118 air changes per minute during the CR emergency mode of operation. The postulated exhaust point is the plant vent, with specific dispersion characteristics to the CR or offsite receiving locations which are defined by unique dispersion factors, or χ/Qs. The alternative release point through a major opening such as a FHB inner or outer trackway roll-up door would have lower χ/Qs and therefore lower calculated dose results.

For the potential accident location in the containment, the corresponding air change rate based on a containment volume of 2,850,000 cubic feet and containment purge ventilation exhaust rate of 40,000 cfm is 0.0140 air changes per minute, considerably lower than the FHB exhaust rates developed above. The purge exhaust point would again be the plant vent, with the same assumed χ/Qs for the CR or offsite receiving locations as for the potential accident location in the FHB. The FHB potential accident location would therefore be bounding. For the alternative release points through the major opening of the personnel/equipment hatch from the containment to the outside, lower χ/Qs apply which would result in lower calculated doses. For an alternative release point through the major opening of the personnel/equipment hatch from the containment to the auxiliary building, the release would be exhausted through the auxiliary building ventilation system to the plant vent, with the same CR χ/Qs as for the FHB potential accident location which would again be bounding.

Inputs and Assumptions

The following inputs and assumptions and bounding analyzed conditions are utilized in the analysis.

a. Core inventory is based on a DBA power level of 3658.3 MWt. This power level bounds the MUR power uprate Rated Thermal Power level including measurement uncertainties.

b. Spent fuel source terms are based on reactor core source terms as previously discussed, with a conservative factor of 2.0 multiplier to account for the gap fractions of fuel exceeding 54 GWD/MTU burnup with a maximum linear heat generation rate exceeding the 6.3 kW/ft peak rod average power limit to address RG 1.183 footnote 11.

c. The damaged fuel is assumed to have operated at a radial peaking factor of 1.7. The damage is assumed to be the rupture of the cladding of all the fuel rods in the one assembly.

d. Movement of fuel will not occur less than 48 hours after the associated reactor shutdown.

e. A water depth above the damaged fuel of 23 feet is the limiting case, corresponding to the minimum depth of water coverage over the top of irradiated fuel assemblies seated in the spent fuel pool racks within the spent fuel pool.

f. No filtration of the airborne radioactivity released from the pool or automatic isolation of the accident location is assumed. Essentially all of the activity released to the environment is assumed to reach the containment refueling floor or spent fuel pool refueling floor airspace within two hours after the accident.

g. Delayed realignment of the CR HVAC system into the emergency mode of operation with filtered make-up (i.e., with an assumed charcoal filtration efficiency of 95%) at 30 minutes after the accident.

h. An amount of unfiltered inleakage into the CR of 500 cfm, added continuously throughout the accident duration.

Dose Results

Radiological doses resulting from a design basis FHA for a CR operator and a person located at the EAB or LPZ are to be less than the regulatory dose limits as given as indicated below.

Regulatory Dose Limits - FHA

Dose Type	Control Room (rem)	EAB and LPZ (rem)
TEDE Dose	5[a]	6.3[b]

Notes:
[a] 10 CFR 50.67
[b] 10 CFR 50.67 as modified by Regulatory Guide 1.183 (Table 6, Page 1.183-20)

The table below provides the results from the limiting case of the unfiltered FHB FHA simulation modeled using the RADTRAD code and the AST assumptions and design input parameters described above.

Fuel Handling Accident
Radiological Analysis Results
(Maximum of Byron and Braidwood)

Fuel Handling Accident Dose Assessment Results		
Control Room (rem TEDE)	EAB (rem TEDE)	LPZ (rem TEDE)
4.40	5.31	0.94

These doses were below the Regulatory Dose Limit, so it is verified that this design basis FHA is sufficiently mitigated at both Byron Station and Braidwood Station.

An additional FHA analysis was performed for recently irradiated fuel with containment closure established or with the FHB ventilation system operable. The results of this analysis also met the limits of 10 CFR 50.67 assuming a minimum decay time of seven hours. The seven-hour minimum decay time is inconsequential, as it is physically impossible to remove the reactor head and move fuel within the first seven hours after the reactor is subcritical.

15.7.5 Spent Fuel Cask Drop Accident

15.7.5.1 Identification of Causes and Frequency Classification

A spent fuel cask will follow the path outlined in Figure 15.7-1. The fuel handling building overhead crane meets the single-failure proof criteria of ASME NOG-1-2004, NUREG-0554 and NUREG-0612, Appendix C, therefore, a drop of the main hook/lower load block is not credible. The main hoist is classified as a Type I main hoist per ASME NOG-1-2004 – single – failure proof for loads up to 125 tons. The auxiliary hoist is single-failure proof up to 15 tons per NUREG-0554.

The upgraded overhead crane is designed to ensure that a single failure of any one of the main components of the crane system will not result in the loss of capability of the system to safely retain a load.

A failure in the lifting gear (sling) would result in an unsafe condition but would not affect the ability to raise or lower the cask.

The following items are assumed to be incapable of failure:

a. the cable drum;

b. any structural component of the crane (bridge beams, rails, etc.);

c. any support for the crane rails which is tied into the building steel; and

d. electrical crane interlocks which prevent movement of the cask over the spent fuel pool.

Therefore, the only single failure in the crane which could cause a fuel cask drop is a failure in the main hook, which is not credible since a significant safety factor (10 to 1) is considered in the hook design.

15.7.5.2 Evaluation and Analysis

Figure 15.7-1 shows the general arrangement of the fuel handling building and the recommended route for the cask to move the spent fuel.

Guard walls are provided around the cask loading area in the spent fuel storage pool as shown on the plan (Figure 15.7-1) and in Section A-A (Figure 15.7-2). These walls which surround the cask loading area, rise the full height of the pool and are structurally designed to withstand the impact force due to a falling cask. If the cask is positioned over the cask loading area and tips and falls, it will land on the guard walls. Since the center of gravity of the cask is within the loading area, as noted in Figure 15.7-2, the cask cannot tip over into the spent fuel storage pool.

A cask drop incident is not credible since the FHB crane is single-failure-proof in accordance with ASME NOG-1-2004, NUREG-0554, and NUREG-0612, Appendix C and the lifting devices and cask attachment points meet the requirements of NUREG-0612.

In addition, the fuel handling building crane will be restricted from operating over the spent fuel pool storage pool and the cask fill area by providing an electrical interlock for the crane bridge as indicated in Figure 15.7-1 (Interlock #1). The fuel handling building crane will also be restricted from leaving the cask fill area and operating over the spent fuel pool by a second interlock for the crane bridge as indicated in Figure 15.7-1 (Interlock #2). The second electrical interlock also prevents the trolley from traveling over the spent fuel pool.

Any other heavy loads (loads exceeding 2000 pounds) will be carried in the spent fuel pool area in accordance with NUREG-0612.

During new fuel loading, the 15-ton auxiliary hook is used to remove the new fuel from the transport vehicle to the new fuel storage racks or new fuel elevator. It is required to have full freedom of travel horizontally to perform this task, so there are no interlocks or stops to prevent hook movement during this period. The auxiliary hook can travel up to 5 feet 6 inches over the spent fuel pool where it is restrained from further motion by electrical interlocks. The 125-ton main hook (load block) is considered a heavy load, but because the upgraded load block is supported by four redundant wire ropes and the maximum load on each wire rope with the maximum critical load attached does not exceed 10% of the manufacturer’s published breaking strength, which is much greater than the weight of the main hoist load block, travel over the spent fuel pool is permitted without a load attached. When the crane does travel over the spent fuel pool, the interlocks must be bypassed and the hoist selector switch must be placed in the “Aux Hoist” position per procedure to prevent inadvertent lowering of the main hoist. These interlocks may be bypassed with loads of less than 2000 pounds on the crane. The interlocks will be controlled administratively using the guidelines listed in NUREG-0612, Table 2.1-1.

The electrical interlocks (limit switches) on the bridge and trolley prevent the main hook of the crane from travelling over the spent fuel storage area when handling a spent fuel cask. These interlocks may be bypassed when the crane is used for new fuel handling operations over the spent fuel pool.

New fuel operations and cask handling will not be performed simultaneously, thus minimizing the possibility of improper movement of the cask.

The main hoist/load block is considered a heavy load, but because the reeving of the upgraded load block is supported by four redundant wire ropes, and the maximum load on each wire rope with the maximum critical load attached does not exceed 10% of the manufacturer's published breaking strength, which is much greater than the weight of the main hoist load block, it is very unlikely that the main hook and lower load block could be dropped on the spent fuel. Therefore, it can travel over the spent fuel pool. Even if such an event were to occur, the resulting damage to the fuel would not result in a release which exceeds the limits of 10 CFR 100. This can be seen by extrapolation of the results of a postulated single fuel element drop in Subsection 15.7.4. This shows that a large number of elements must be damaged to exceed the 10 CFR 100 limits. The lower load block is not large enough to cause this damage.

All potential accidents involving lifting and transporting of loads heavier than a fuel element are addressed in a report submitted in response to NUREG-0612. The fuel handling building crane and loads are included in this report.

The consequences of the drop of loads lighter than a fuel element will be less than the drop of a single fuel element as reported in Chapter 15.0. The design of the tools and the fuel building cranes prevents the tools from dropping onto the fuel from a great height. The heaviest of these loads is the RCC change tool which weighs less than 1100 pounds. This tool is over 30 feet long. Because of the height of the fuel building crane, the RCC change tool can only be carried a few feet above the fuel. With the short vertical drop distance and the low weight per foot of length involved, there is no real probability of damage to the fuel. The burnable poison assembly handling tool, the thimble plug handling tool, and the spent fuel assembly handling tool all have weights under 30 pounds per foot and are not carried high above the fuel. All other tools have gross weights under 100 pounds. The single fuel assembly drop accident in Subsection 15.7.4 is the maximum credible accident involving dropped loads and spent fuel damage.

An unrestrained stack-up configuration is used during the loading or unloading process for transferring the Multi-Purpose Canister (MPC) (loaded with spent fuel assemblies) between the HI-TRAC transfer cask (HI-TRAC) and the HI-STORM 100 Version B overpack (HI-STORM) in the Fuel Handling Building (FHB). In the unrestrained stack-up configuration, the HI-TRAC transfer cask is located on top of the HI-STORM overpack and is bolted to the Mating Device (MD) which is bolted to the top of the HI-STORM (without HI-STORM lid in place). The unrestrained stack-up configuration is shown in Figure 9.1-23.

This stack-up configuration has been demonstrated via analysis (Section 9.1.2.3.11) to maintain its stability (rocking and sliding) during a seismic event without tip-over. In the unlikely event of a tip-over of the casks in the FHB track-way, there is more than 15 feet of separation between the top of the HI-TRAC and the closest edge of the Spent Fuel Pool. Therefore, due to the spatial separation, a tip-over of the HI-TRAC/ HI-STORM stack-up will not result in a compromise to Spent Fuel Pool integrity.

15.7.5.3 Barrier Performance

The spent fuel pool walls are designed to withstand increased water pressure caused by a vertical drop of the cask.

If a cask drops on the exterior pool wall, and tips, it will land on the guard walls which are designed to withstand the impact force due to a falling cask. The cask will not fall outside the cask storage well and thus will not affect the fuel in the fuel storage pool.

15.7.5.4 Modeling of Accident Analysis

15.7.5.4.1 Mathematical Model

A free drop of a fuel cask from a height of 30 feet or more onto an unyielding surface (see 10CFR 71) is not a credible event since a single failure proof crane and lifting device will be employed during cask handling activities. Therefore, there will be no resulting damage to the cask that will cause a release of radioactive materials to the public.

15.7.5.4.2 Input Parameters and Initial Conditions

There are no input parameters and initial conditions of variables relevant to the evaluation of barrier performance that were not presented in Subsection 15.7.5.3.

15.7.5.4.3 Results

Inasmuch as the cask cannot be free dropped onto an unyielding surface from a height of 30 feet or more, the cask will not be damaged and there will be no release of radioactive materials to the public.

Moreover, if damage to the cask compartment liner resulted from a drop of the cask from maximum hook height, the fuel pool would not become dewatered as a result of such damage. This is due to the watertight gate that separates the cask compartment from the main portion of the fuel pool.

Since the cask will always be brought to the pool so as to enter the storage well without passing over the pool, there is no possibility of cask drop into the fuel storage pool.

The fuel pool slab rests directly on rock (Byron) or on stiff soil (Braidwood). This slab was designed for a vertical drop of the cask in the storage well.

15.7.5.5 Radiological Consequences

A spent fuel cask drop accident is not a credible event when a single failure proof crane and lifting device are employed to move the cask and therefore, is less severe than the fuel handling accident analyzed in Subsection 15.7.4.

For Braidwood Station a shipment specific radiological analysis was performed for the dropping of an NAC International Shipping Cask with five fuel rods. Impact limiting devices are not employed during the movement of the NAC International Shipping Cask in the fuel handling building. A drop of the cask would result in damage and subsequent release of radioactive materials to the public. Results of the radiological analysis show that the dropping of the loaded NAC International Shipping Cask is not limiting and is much less severe than the fuel handling accident analyzed in Subsection 15.7.4. This analyses is applicable only to Braidwood Station and only for this specific shipment.

15.7.6 References

1. NUREG/CR-5009, "Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors," February 1988.

2. WCAP-7828, "Radiological Consequences of a Fuel Handling Accident," December 1971.

3. NUREG-0800, Section 15.7.4, "Radiological Consequences of Fuel Handling Accidents," Revision 1, July 1981.

TABLE 15.7-1

PARAMETERS USED IN WASTE GAS DECAY TANK RUPTURE ANALYSES
(using TID 14844)

	REGULATORY GUIDE 1.24 ANALYSIS
Fuel defects	1%
Primary Coolant Activity	Table 15.0-9
Time of accident	Immediately after shutdown at end of equilibrium core cycle
Meteorology	See Tables 15.0-13 and 15.0-14
Nuclide data	Table 15A-1

TABLE 15.7-2

WASTE GAS DECAY TANK INVENTORY (ONE UNIT)
(using TID 14844)

Security - Related Information Withheld Under 10 CFR 2.390

TABLE 15.7-3

SPENT RESIN STORAGE TANK IVENTORY
(using TID 14844)

Security - Related Information Withheld Under 10 CFR 2.390

TABLE 15.7-4

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TABLE 15.7-5

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TABLE 15.7-6

BOUNDING ISOTOPIC CORE INVENTORY

Security - Related Information Withheld Under 10 CFR 2.390

TABLE 15.7-7

INPUT PARAMETERS FOR THE FHA RADIOLOGICAL CONSEQUENCE ANALYSIS USING AST

Parameter	Unit	Value	Notes
Gap Activity: I-131 Kr-85 Other Noble Gases Other Halogens Alkali Metals	 % % % % %	2 times the following: 0.08 0.10 0.05 0.05 0.12	As a significant number of fuel assemblies not qualifying for AST due to their containing fuel rods with maximum linear heat generation rates exceeding 6.3 kilowatts per foot peak rod average power for burnups exceeding 54 GWD/MTU, the fuel will be treated as having gap fractions a factor of 2 greater than the RG 1.183 values.
Number of Assemblies Damaged	#	1	264 rods are assumed to be damaged. This is one full assembly.
Pool Scrubbing Factor: Overall Elemental Organic (and Noble Gases)	 - - -	 200 500 1	Per RG 1.183, based on 23 feet of water coverage.
Release Path Filter Efficiency	%	0	Filtration is not credited for AST analysis.
Duration of release	hr	2	
Chemical form of radioiodine released from the fuel to the spent fuel pool: Cesium iodide Elemental iodine Organic iodine	 % % %	 95 4.85 0.15	
Chemical form of radioiodine released from the pool to the building: Elemental iodine Organic iodine	 % %	 57 43	
Depth of Water above the Top of Reactor Vessel Flange and Fuel Assemblies in Spent Fuel Pool	ft	23	

TABLE 15.7-7 (Cont'd)

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TABLE 15.7-8

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15.8 ANTICIPATED TRANSIENTS WITHOUT SCRAM (ATWS)

An anticipated transient without scram (ATWS) is an anticipated operational occurrence (such as a loss of feedwater, loss of condenser vacuum, or loss of offsite power) that is accompanied by a failure of the reactor trip system to shut down the reactor. A series of generic studies (References 1 and 2) on ATWS showed that acceptable consequences would result, provided that the turbine trips and that auxiliary feedwater flow is initiated in a timely manner.

The effects of an ATWS are not considered as part of the design basis for transients analyzed in Chapter 15. However, 10 CFR 50.62, "Requirements for Reduction of Risk from Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants," the ATWS Rule, requires that each pressurized water reactor have equipment that is diverse from the reactor trip system to automatically initiate the auxiliary feedwater system and initiate a turbine trip under the conditions indicative of an ATWS.

In addition, compliance with the ATWS rule is demonstrated on a cycle-by-cycle basis by focusing on two aspects of WCAP-11992, "ATWS Rule Administration," December 1988 methodology (Reference 3). These aspects are the unfavorable exposure time (UET) and critical trajectory methodologies. The critical trajectories are calculated loci of plant conditions (e.g., power vs. inlet temperature) that provide a peak pressure in the transient analysis of the limiting ATWS event, which is then compared to the specified limit (3200 psig). The UET is the time during the cycle when reactivity feedback is insufficient to maintain pressure under 3200 psig for a given reactor state.

In the application of the UET methodology, the ATWS transient point kinetics information is transferred into steady-state conditions for comparison with cycle-specific core condition evaluation calculations, and the critical trajectories are determined. During peak ATWS pressure conditions, heatup is relatively slow so that steady-state analysis is acceptable. The methodology uses the "base case" conditions from the ATWS submittal, with 100-percent power-operated relief valve capacity, 100-percent auxiliary feedwater, and no control rod insertion. The cycle-specific calculations are done with appropriate ATWS initial conditions of full power, rods out, equilibrium xenon, and 3200 psig pressure. These calculations are compared to the critical trajectory from the transient analysis. This comparison provides cycle-specific design conditions that would result in transient conditions exceeding 3200 psig. These calculations show any core design conditions that would result in exceeding the 3200 psig pressure limit. Calculations as a function of the time in cycle and, thus, as a function of moderator temperature coefficient (MTC) show the time during the cycle that the core design critical trajectory is greater than the transient trajectory. From this, the UET is determined. The analysis must show that the UET, given the cycle design, will be less than five percent, or equivalently, that ATWS pressure limit will be met for at least 95 percent of the cycle. If the limit is not met, the core design would be changed until the 95-percent level is achieved.

This 95-percent probability level for the UET is equivalent to the probability level in the reference analyses for the ATWS rule basis. All parameters should be best estimate values with the exception of the MTC initial condition. That was to be at a level not to be exceeded (i.e., not less negative) at full power conditions for at least 95 percent of the cycle.

15.8.1 REFERENCES

1. Burnett, T. W. T., et al., "Westinghouse Anticipated Transients Without Trip Analysis," WCAP-8330, August 1974.

2. Anderson, T. M., (Westinghouse) letter to S. H. Hanauer (USNRC), "ATWS Submittal," NS-TMA-2182, December 1979.

3. Sloane, B. D., et al, “Joint Westinghouse Owners Group / Westinghouse Program: ATWS Rule Administration Process,” WCAP 11992, December 1988.

ATTACHMENT 15A

DOSE MODELS USED TO EVALUATE THE ENVIRONMENTAL CONSEQUENCES OF ACCIDENTS

TABLE OF CONTENTS

LIST OF TABLES

ATTACHMENT 15A

DOSE MODELS USED TO EVALUATE THE RADIOLOGICAL CONSEQUENCES OF ACCIDENTS

15A.1 INTRODUCTION

This Attachment identifies the models used to calculate the radiological doses that would result from releases of radioactivity due to various postulated accidents.

The major design basis accidents were analyzed using Alternative Source Term methodology using Regulatory Guide 1.183 with dose acceptance criteria per 10 CFR 50.67. These accidents include:

- Loss of Coolant Accident (LOCA)
- Main Steam Line Break (MSLB)
- Control Rod Ejection Accident (CREA)
- Locked Rotor Accident (LRA)
- Steam Generator Tube Rupture (SGTR)
- Fuel Handling Accident (FHA)

Conformance with Regulatory Guides 1.183 and 1.194 are documented in Appendix A.

Other accidents and events not listed above (including doses related to equipment qualification) were analyzed using the TID-14844 methodology per Regulatory Guide 1.4 with 10 CFR 100 dose acceptance criteria.

15A.2 ASSUMPTIONS (TID-14844)

The following assumptions are basic to all dose models:

a. Direct radiation from the source point is negligible compared to gamma radiation due to immersion in the semi-infinite radioactive cloud.

b. All radioactivity releases are treated as ground level releases regardless of the point of discharge.

c. The dose receptor is a standard man as defined by the International Commission on Radiological Protection (ICRP) (Reference 1).

d. Radioactive decay from the point of release to the dose receptor is neglected.

e. The offsite atmospheric dispersion factors used are the 5^{th} percentile values from Table 15.0-13 and 15.0-14.

15A.3 OFFSITE DOSE MODELS (TID-14844)

Whole body (acute) doses and thyroid doses from inhaled radioactive iodine are calculated for the 0 - 2 hour interval at the exclusion area boundary and for the duration of the accident at the low population zone (LPZ) outer boundary.

15A.3.1 WHOLE BODY DOSE (TID-14844)

The whole body dose delivered to a dose receptor is obtained by considering the dose receptor to be immersed in a radioactive cloud which is infinite in all directions above the ground plane, i.e., an "infinite semispherical cloud." The concentration of radioactive material within this cloud is taken to be uniform and equal to the maximum centerline ground level concentration that would exist in the cloud at the appropriate distance from the point of release.

The whole body dose is a result of external gamma radiation.

Two different models are used for calculating the whole body dose. For the noble gases, the model taken from Regulatory Guide 1.109 (Reference 2) is used. The equation is:

$$D_{\gamma} = (X / Q) \sum_{i} (A_i)\,(DCF_{\gamma-i})$$

Where: D_{γ} = Whole body dose, rem

X/Q = the 5th percentile atmospheric dispersion factor for a given time period and distance (see Tables 15.0-13 and 15.0-14), sec / m^3

A_i = the activity of nuclide i released during a given time period, Ci

$DCF_{\gamma-i}$ = gamma dose conversion factor for nuclide i (see Table 15A-1), rem • m^3 / Ci • sec

Regulatory Guide 1.109 does not include dose conversion factors submersion in a cloud of radioactive iodine. The whole body dose contribution from the iodine in the cloud of activity is calculated using the following equation from Reference 3.

$$D_{\gamma} = (0.25)(X/Q) \sum_{i} (A_i)(E_{\gamma-i})$$

Where: $E_{\gamma-i}$ = average gamma disintegration energy for nuclide i (see Table 15A-1), MeV per disintegration

15A.3.2 THYROID INHALATION DOSE (TID-14844)

The thyroid dose for a given time period is obtained from the following expression:

$$D_{thy} = B(X/Q) \sum_{i} (A_i)(DCF_{thy-i})$$

Where: D_{thy} = thyroid dose, rem

B = breathing rate, m^3/sec (from Reference 3), the breathing rate offsite varies with time as follows:

0 - 8 hours	3.47E-4 m^3/sec
8 - 24 hours	1.75E-4 m^3/sec
>24 hours	2.32E-4 m^3/sec

X/Q	=	the 5th percentile atmospheric dispersion factor for a given time period and distance (see Tables 15.0-13 and 15.0-14), sec / m^3
A_i	=	the activity of iodine isotope i released during a given time period, Ci
DCF_{thy-i}	=	thyroid dose conversion factor for iodine isotope i (see Table 15A-1), rem / Ci inhaled

15A.4 CONTROL ROOM DOSE MODELS (TID-14844)

The operators in the control room will accumulate doses due to the activity entering the control room as the result of an accident. The integrated activity in the control room during a given time interval is found by multiplying the release by the appropriate X/Q (see Table 6.4-1a) to give the concentration at the control room intake. This activity is brought into the control room through the filtered intake pathway and by unfiltered inleakage. The activity in the control room atmosphere is reduced by recirculation through filters, by exhausting a portion of the air to the environment (balancing the air inflow and inleakage), and by radioactive decay. The flows and filter efficiencies are provided in Table 6.4-1a.

Using the integrated activity in the control room, the whole body (acute) doses, beta-skin doses, and thyroid doses are calculated for the operators in the control room from the models described in the following subsections.

15A.4.1 WHOLE BODY DOSE (TID-14844)

As with the determination of offsite doses, there are two different models used for calculating the whole body dose. For the noble gases, the model is based on Regulatory Guide 1.109 (Reference 2). The equation is:

$$D\gamma = (GF)(OF)\sum_i (A_i)(DCF_{\gamma-i})$$

Where:	D_γ	=	whole body dose, rem
	GF	=	geometry correction factor used to convert the semi-infinite cloud to a finite cloud (from Reference 4, GF = 1173 / $V^{0.338}$ where "V" is the control room volume in cubic feet)

OF	=	the occupancy factor (i.e., the fraction of the time period that the operator is assumed to be present in the control room), from Reference 4, the values are: 0 - 24 hours OF = 1.0 24 - 96 hours OF = 0.6 96 - 720 hours OF = 0.4
A_i	=	the average concentration of nuclide i in the control room atmosphere during a given time period, Ci/m^3
$DCF_{\gamma-i}$	=	whole body dose conversion factor for nuclide i (see Table 15A-1), rem • m^3 / Ci • sec

Regulatory Guide 1.109 does not include dose conversion factors for sumbersion in a cloud of radioactive iodine. The whole body dose contribution from the iodine in the cloud of activity is calculated using the following equation:

$$D_\gamma = (GF)(0.25)(OF)\sum_i (A_i)(E_{\gamma-i})$$

Where: $E_{\gamma-i}$ = average gamma disintegration energy for nuclide i (Table 15A-1), MeV per disintegration

15A.4.2 BETA-SKIN DOSE (TID-14844)

As with the whole body dose, two different models are used for calculating the beta-skin dose. For the noble gases, the model is based on Regulatory Guide 1.109 (Reference 2). The equation is:

$$D_\beta = OF \sum_i (C_i)(DCF_{\beta-i})$$

Where:	D_β	=	Beta-skin dose, rem
	$DCF_{\beta-i}$	=	Beta-skin dose conversion factor for nuclide i (see Table 15A-1), rem • m^3 / Ci • rem

Regulatory Guide 1.109 does not include dose conversion factors for submersion in a cloud of radioactive iodine. The beta-skin dose contribution form the iodine in the cloud of activity is calculated using the following equation:

$$D_\beta = (0.23)(OF) \sum_i (A_i)(E_{\beta-1})$$

Where: $E_{\beta-i}$ = average beta disintegration energy for nuclide i (Table 15A-1), MeV per disintegration

15A.4.3 THYROID DOSE (TID-14844)

The equation for the thyroid dose is:

$$D_{thy} = (OF)(B) \sum_i (A_i)(DCF_{thy-i})$$

Where: D_{thy} = thyroid dose, rem

B = breathing rate, m^3/sec (from Reference 4, the breathing rate in the control room is 3.47E-4 m^3/sec and does not vary with time)

DCF_{thy-i} = thyroid dose conversion factor for nuclide i (see Table 15A-1), rem / Ci inhaled

15A.5 REFERENCES

1. "Report of ICRP Committee II on Permissive Dose for Internal Radiation (1959)," Health Physics Vol. 3, pp. 30, 146-153, 1970.

2. Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977.

3. Regulatory Guide 1.4, Revision 2, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactor," USAEC, June 1974.

4. Murphy, K. G. and K. M. Campe, "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19," presented at the 13th AEC Air Cleaning Conference, November 1974.

5. ICRP Publication 38, "Radionuclide Transformations - Energy and Intensity of Emissions," 1983.

6. ICRP Publication 30, "Limits for Intakes of Radionuclides by Workers," 1979.

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

8. NUREG/CR-6604, "RADTRAD: A Simplified Model for RADionuclide Transport and Removal And Dose Estimation," April 1998; Supplement 1, June 1999; and Supplement 2, October 2002.

9. U.S. Federal Guidance Report No.11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," 1988.

10. U.S. Federal Guidance Report No.12, "External Exposure to Radionuclides in Air, Water, and Soil," 1993.

11. Regulatory Guide 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," June 2003.

12. Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Revision 1, November 1982.

TABLE 15A-1

PHYSICAL DATA FOR ISOTOPES

Isotope	Decay Constant*, hr^{-1}	Thyroid Dose Conversion Factors**, rem/Ci	Average Gamma Disintegration Energy*, Mev/Disintegration	Average Beta Disintegration Energy*, Mev/Disintegration
I-131	3.59E-3	1.07E6	3.81E-1	3.81E-1
I-132	3.01E-1	6.29E3	2.28E0	2.28E0
I-133	3.33E-2	1.81E5	6.07E-1	6.07E-1
I-134	7.91E-1	1.07E3	2.62E0	2.62E0
I-135	1.05E-1	3.14E4	1.58E0	1.58E0

Isotope	Decay Constant*, hr-1	Whole-Body Dose Conversion Factor ***, rem-m^3/Ci-sec	Beta-Skin Dose Conversion Factor***, rem-m^3/Ci-sec
Kr-85m	1.55E-1	3.71E-2	4.63E-2
Kr-85	7.38E-6	5.10E-4	4.25E-2
Kr-87	5.45E-1	1.88E-1	3.085E-1
Kr-88	2.44E-1	4.66E-1	7.51E-2
Xe-131m	2.43E-3	2.90E-3	1.50E-2
Xe-133m	1.32E-2	7.96E-3	3.15E-2
Xe-133	5.51E-3	9.32E-3	9.70E-3
Xe-135m	2.72E0	9.89E-2	2.25E-2
Xe-135	7.63E-2	5.74E-2	5.90E-2
Xe-138	2.93E0	2.80E-1	1.31E-1

* Reference 5

** Reference 6

*** Reference 2

Attachment 15B has been deleted intentionally.

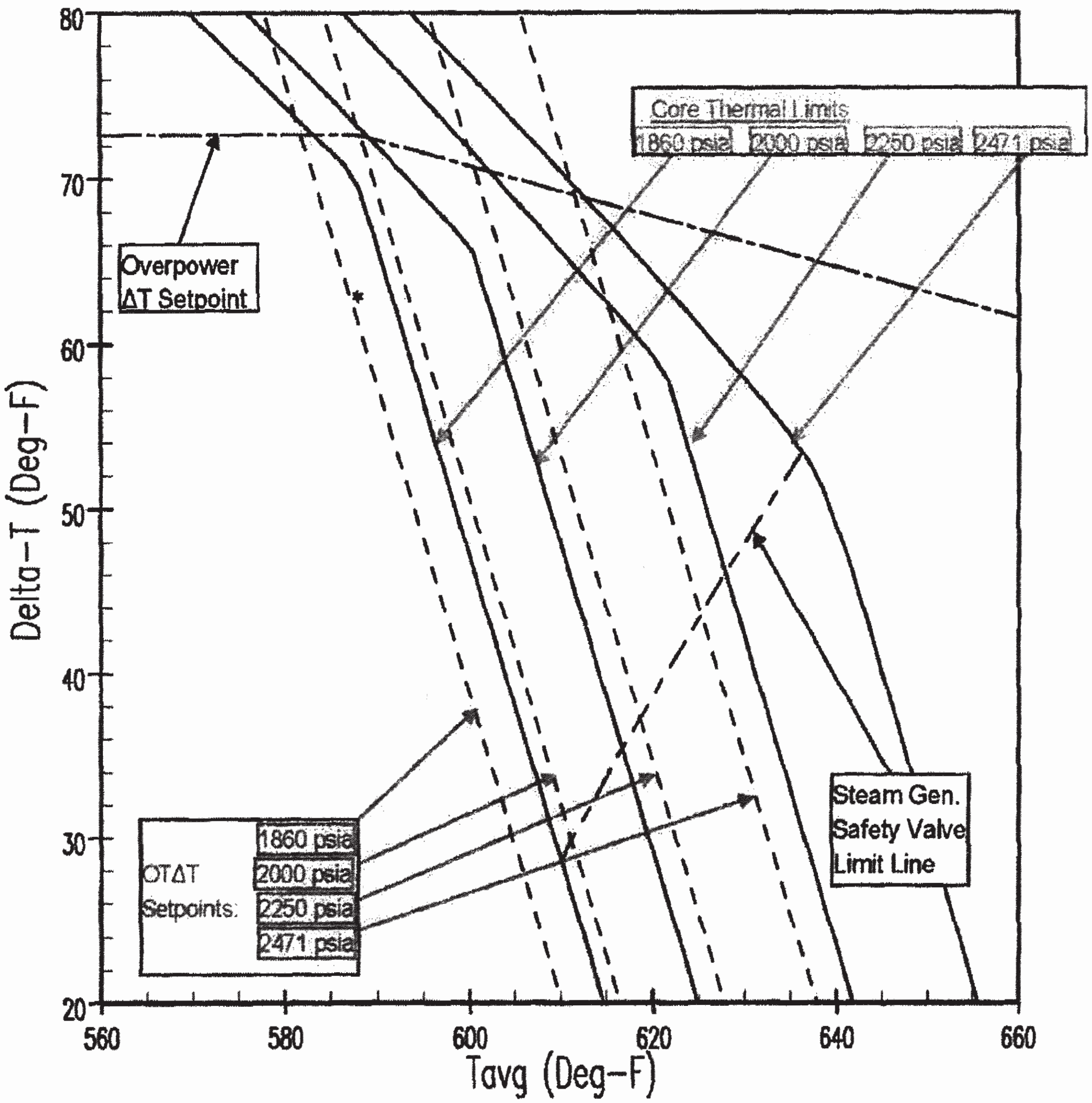

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FIGURE 15.0-1

ILLUSTRATION OF OVERTEMPERATURE AND OVERPOWER ΔT PROTECTION

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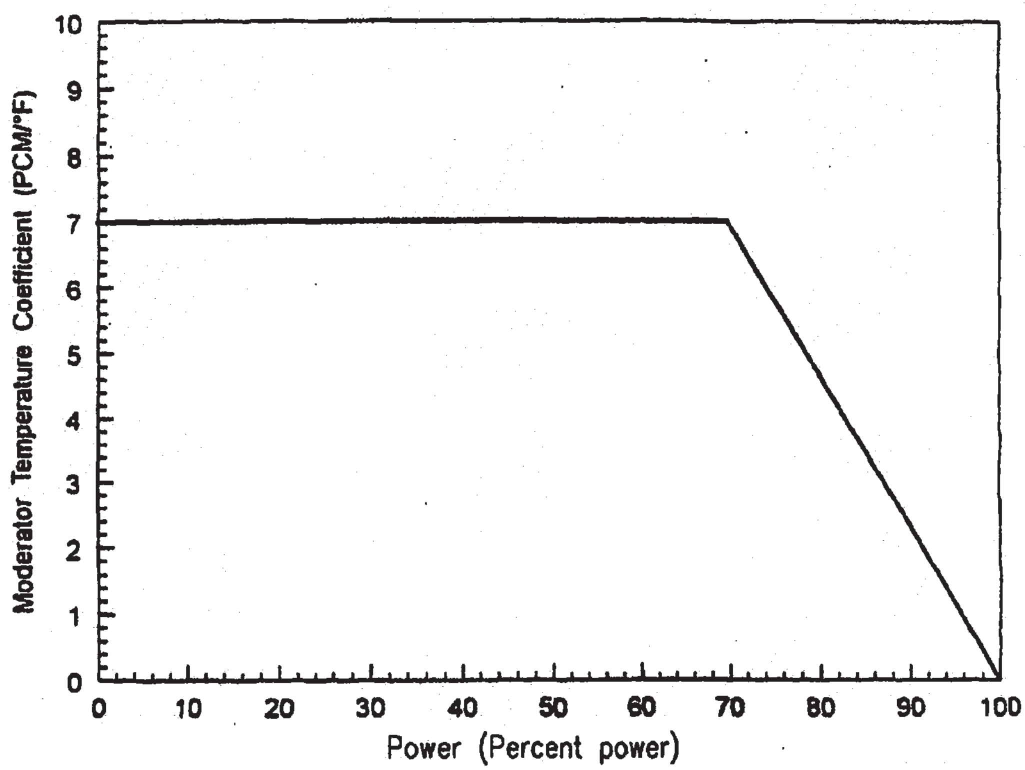

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FIGURE 15.0-2 MODERATOR TEMPERATURE COEFFICIENT USED IN ANALYSIS

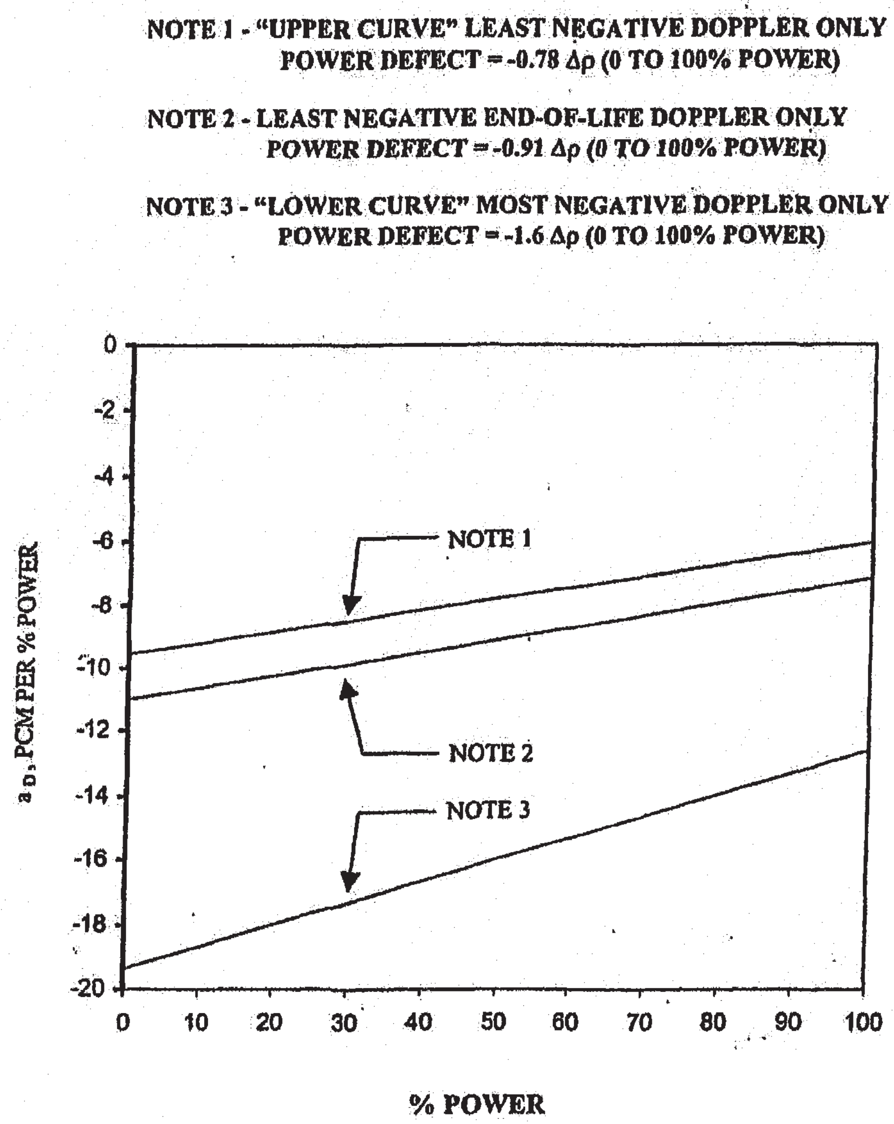

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FIGURE 15.0-3

DOPPLER POWER COEFFICIENT
USED IN ACCIDENT ANALYSIS

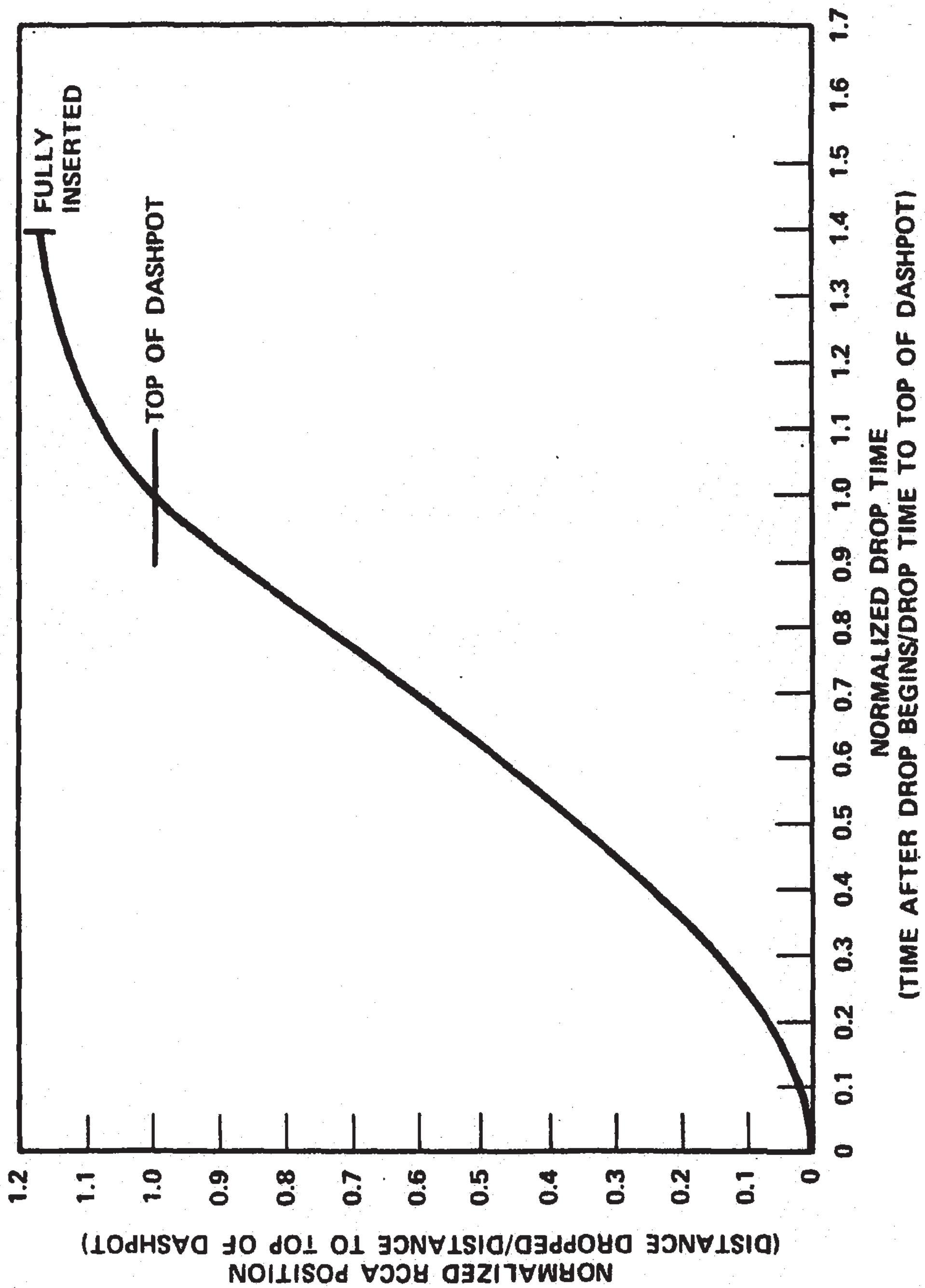

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FIGURE 15.0-4

RCCA POSITION VERSUS TIME
TO DASHPOT

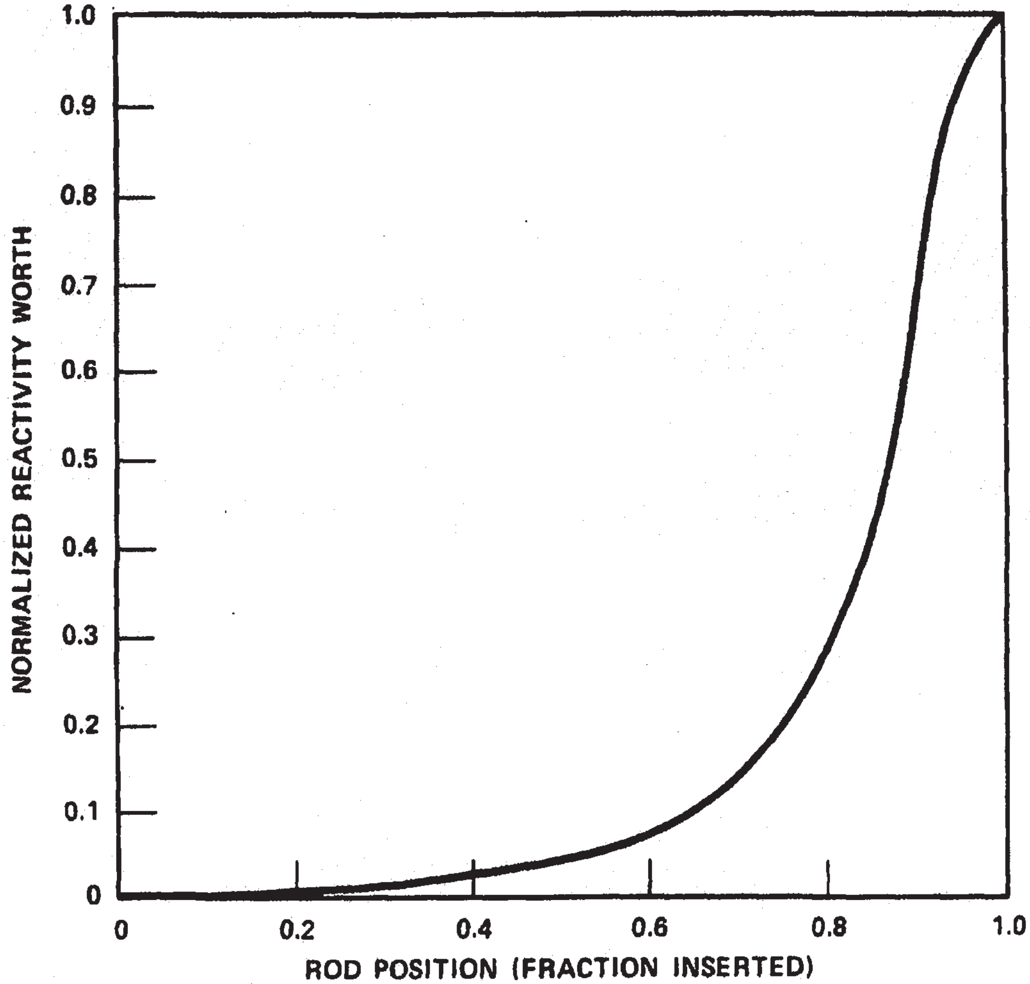

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FIGURE 15.0-5

NORMALIZED ROD WORTH
VERSUS PERCENT INSERTED

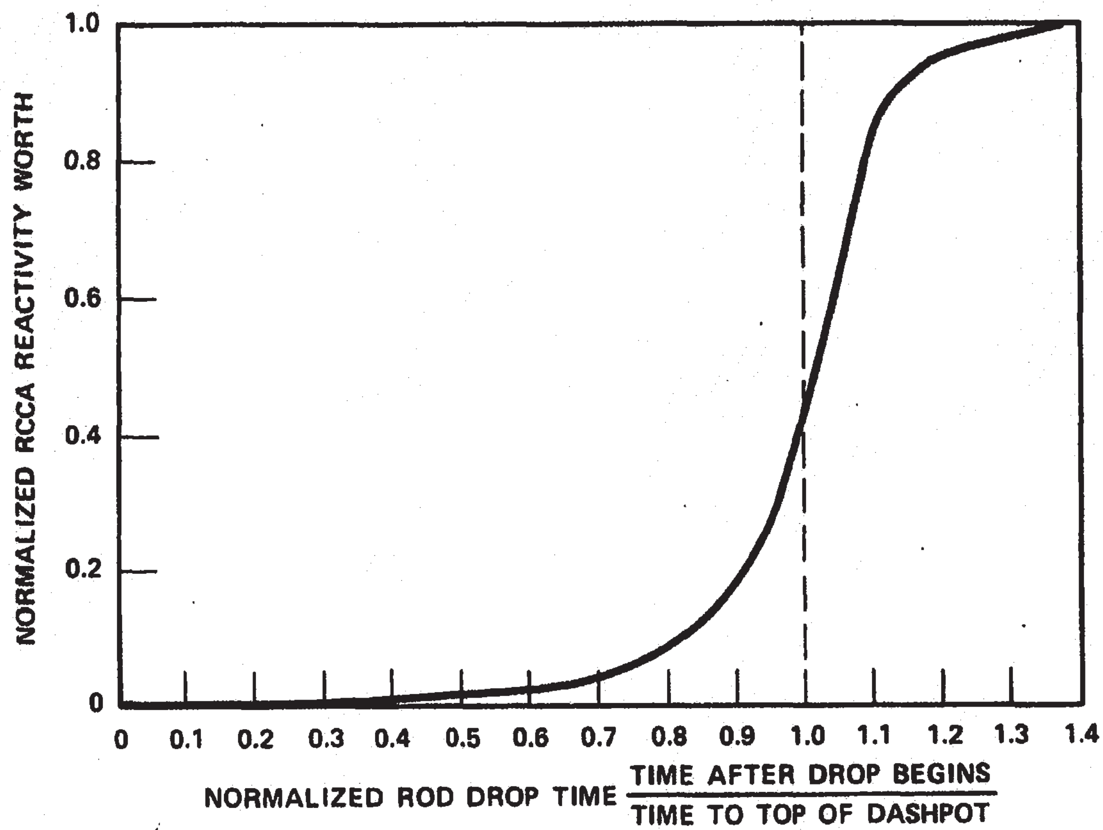

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FIGURE 15.0-6

NORMALIZED RCCA BANK REACTIVITY WORTH
VS. NORMALIZED DROP TIME

ABBREVIATIONS USED:

AFWS	- AUXILIARY FEEDWATER SYSTEM	ECCS	- EMERGENCY CORE COOLING SYSTEM
CVCS	- CHEMICAL AND VOLUME CONTROL SYSTEM	HL	- - HOT LEG
ESFAS	- ENGINEERED SAFETY FEATURES ACTUATION SYSTEM	CL	- COLD LEG
FW	- FEEDWATER	CCWS	- COMPONENT COOLING WATER SYSTEM
RTS	- REACTOR TRIP SYSTEM	RCS	- REACTOR COOLANT SYSTEM
SIS	- SAFETY INJECTION SYSTEM	SWS	- SERVICE WATER SYSTEM
SI	- SAFETY INJECTION	HPI	- HIGH PRESSURE INJECTION
RT	- REACTOR TRIP	LPI	- LOW PRESSURE INJECTION
CS	- CONTAINMENT SPRAY	CI	- CONTAINMENT ISOLATION
		SG	- STEAM GENERATOR

NOTES:

1. FOR TRIP INITIATION AND SAFETY SYSTEM ACTUATION. MULTIPLE SIGNALS ARE SHOWN BUT ONLY A SINGLE SIGNAL IS REQUIRED. THE OTHER SIGNALS ARE BACKUPS.
2. NO TIMING SEQUENCE IS IMPLIED BY POSITION OF VARIOUS BRANCHES. REFER TO EVENT TIMING SEQUENCES PRESENTED IN TABULAR FORM IN PERTINENT ACCIDENT ANALYSIS SECTION OF CHAPTER 15.0 OF THE FSAR.

DIAGRAM SYMBOLS:


- EVENT TITLE


- BRANCH POINT FOR DIFFERENT PLANT CONDITIONS


- SAFETY SYSTEM


- SAFETY ACTION

- SYSTEM REQUIRED TO MEET SINGLE-FAILURE CRITERIA

- MANUAL ACTION REQUIRED DURING SYSTEM OPERATION

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FIGURE 15.0-7

SYMBOLS USED IN ACCIDENT SEQUENCES

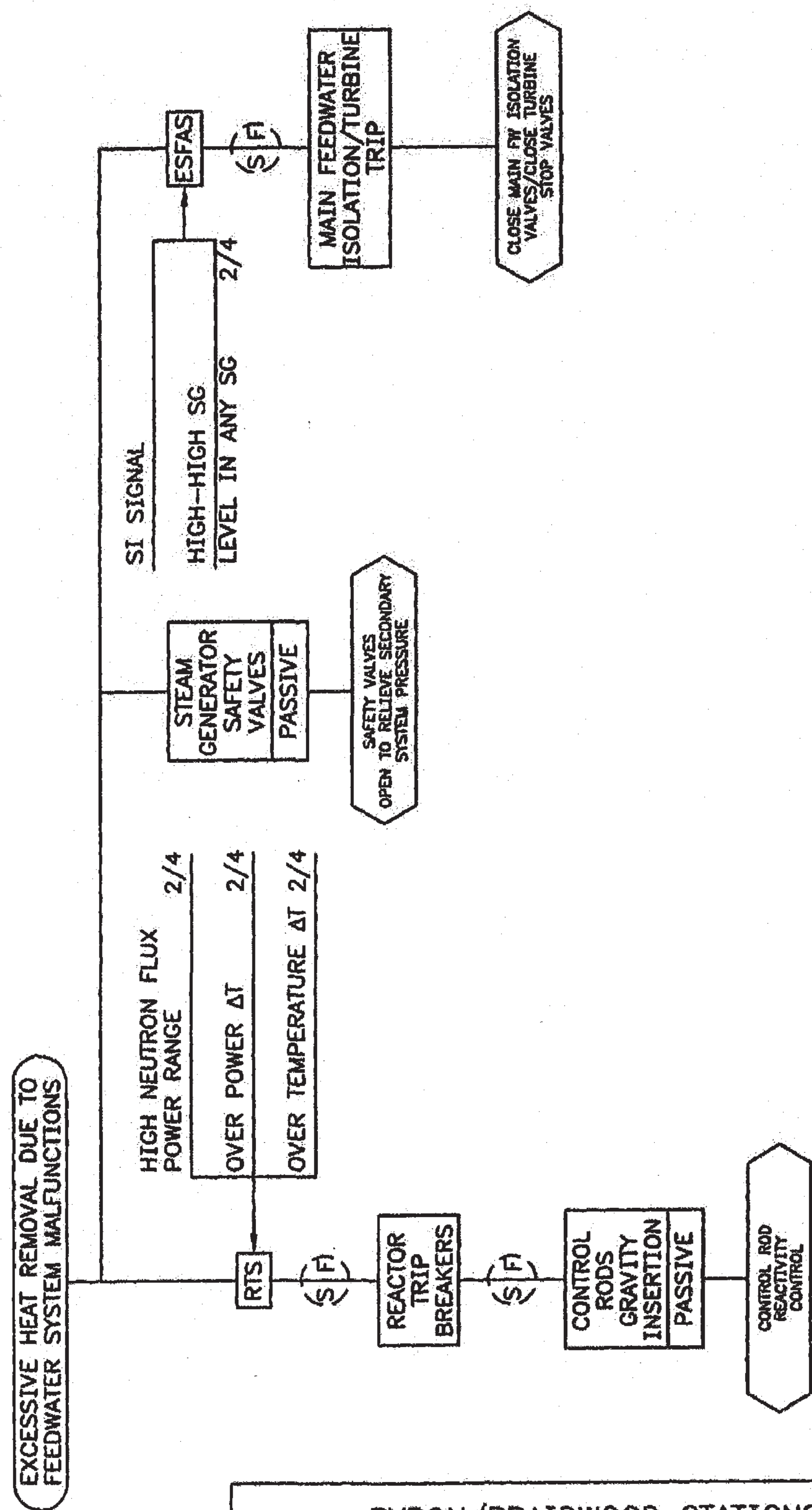

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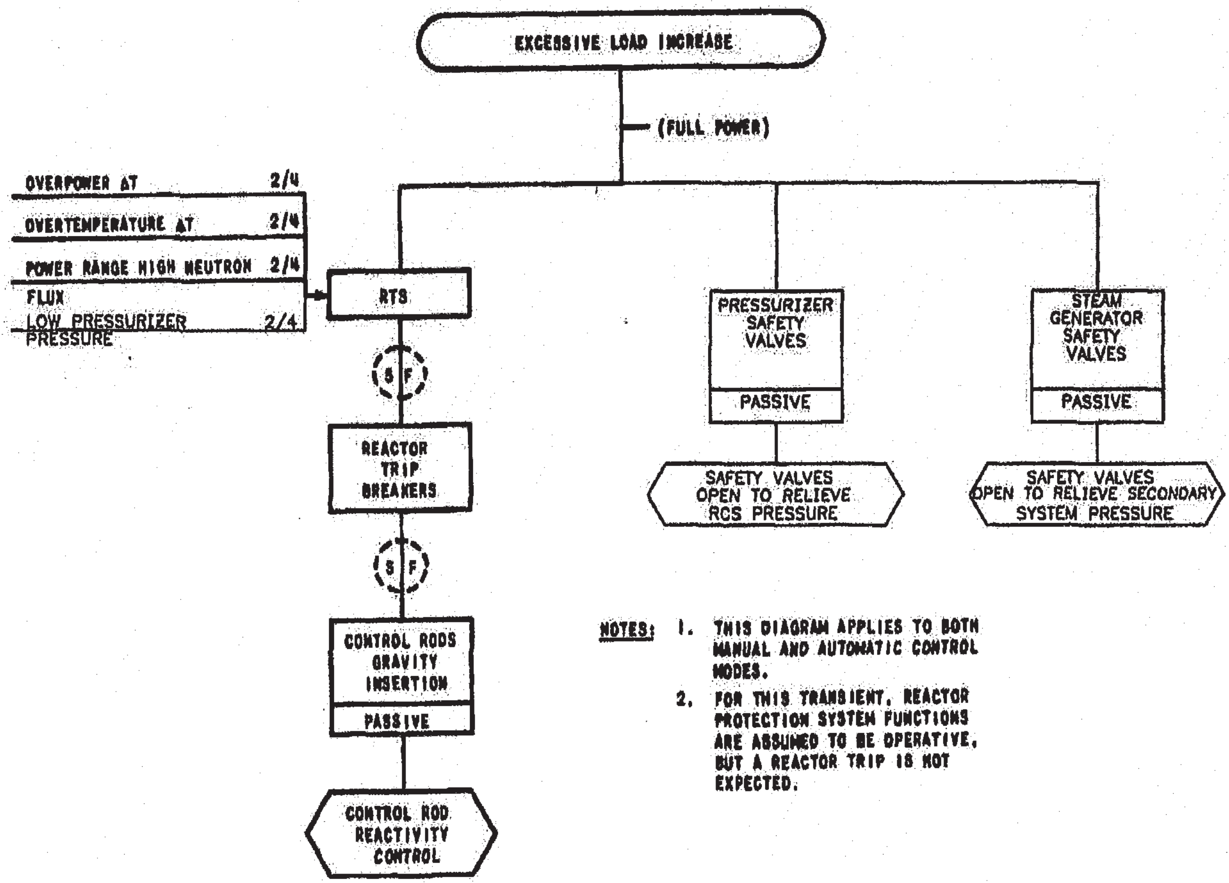
FIGURE 15.0-8

EXCESSIVE HEAT REMOVAL DUE TO FEEDWATER SYSTEM MALFUNCTIONS



NOTES:
1. THIS DIAGRAM APPLIES TO BOTH MANUAL AND AUTOMATIC CONTROL MODES.
2. FOR THIS TRANSIENT, REACTOR PROTECTION SYSTEM FUNCTIONS ARE ASSUMED TO BE OPERATIVE, BUT A REACTOR TRIP IS NOT EXPECTED.

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FIGURE 15.0-9

EXCESSIVE LOAD INCREASE

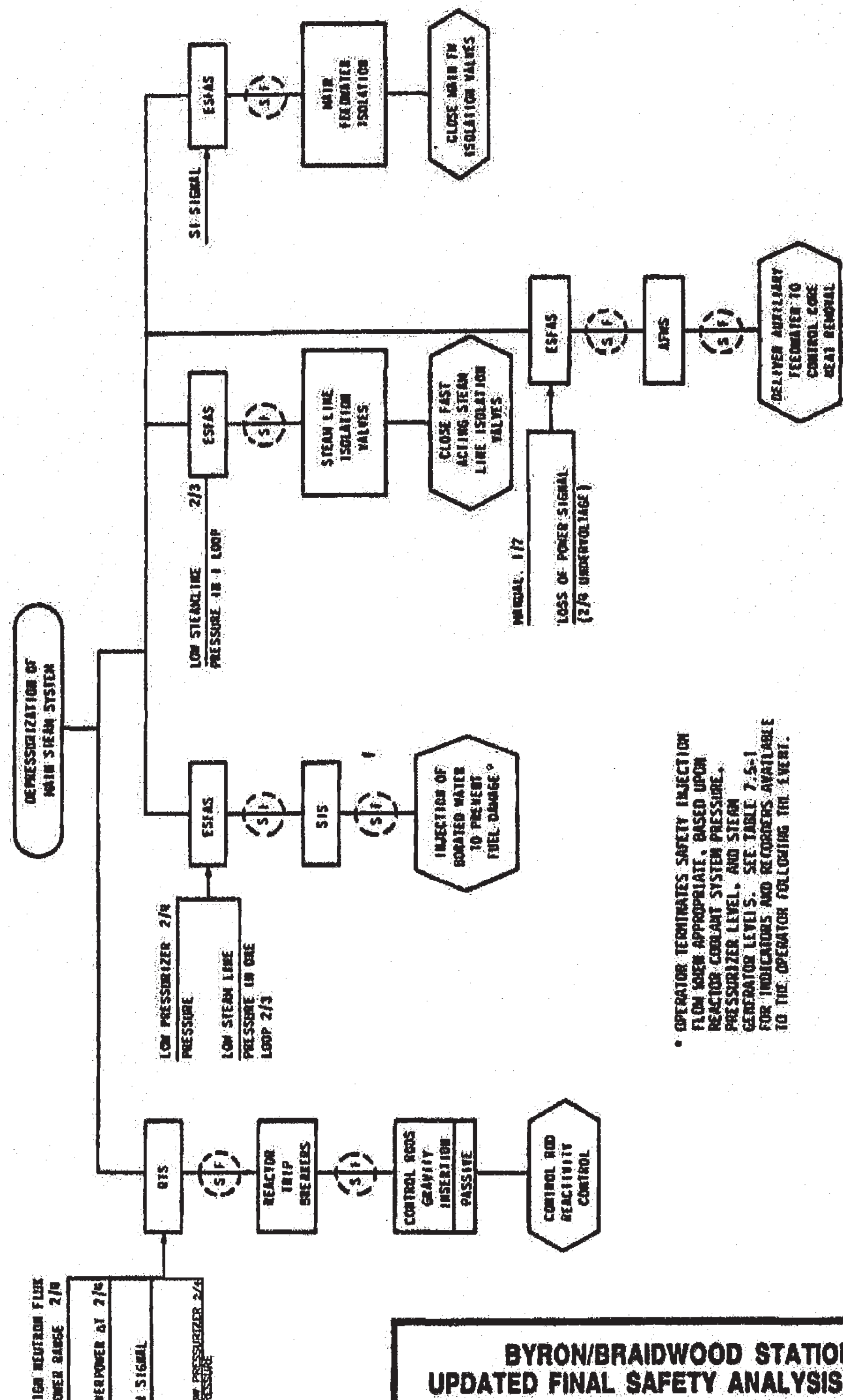

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FIGURE 15.0-10

DEPRESSURIZATION OF MAIN STEAM SYSTEM

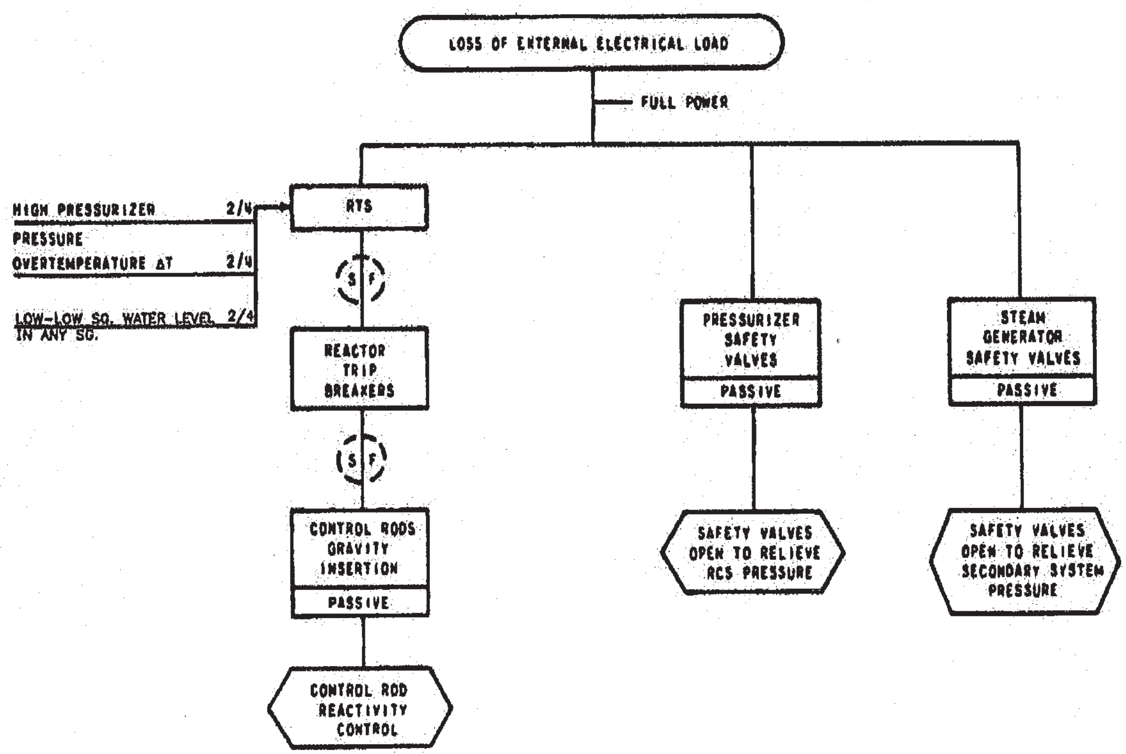

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FIGURE 15.0-11

LOSS OF EXTERNAL LOAD

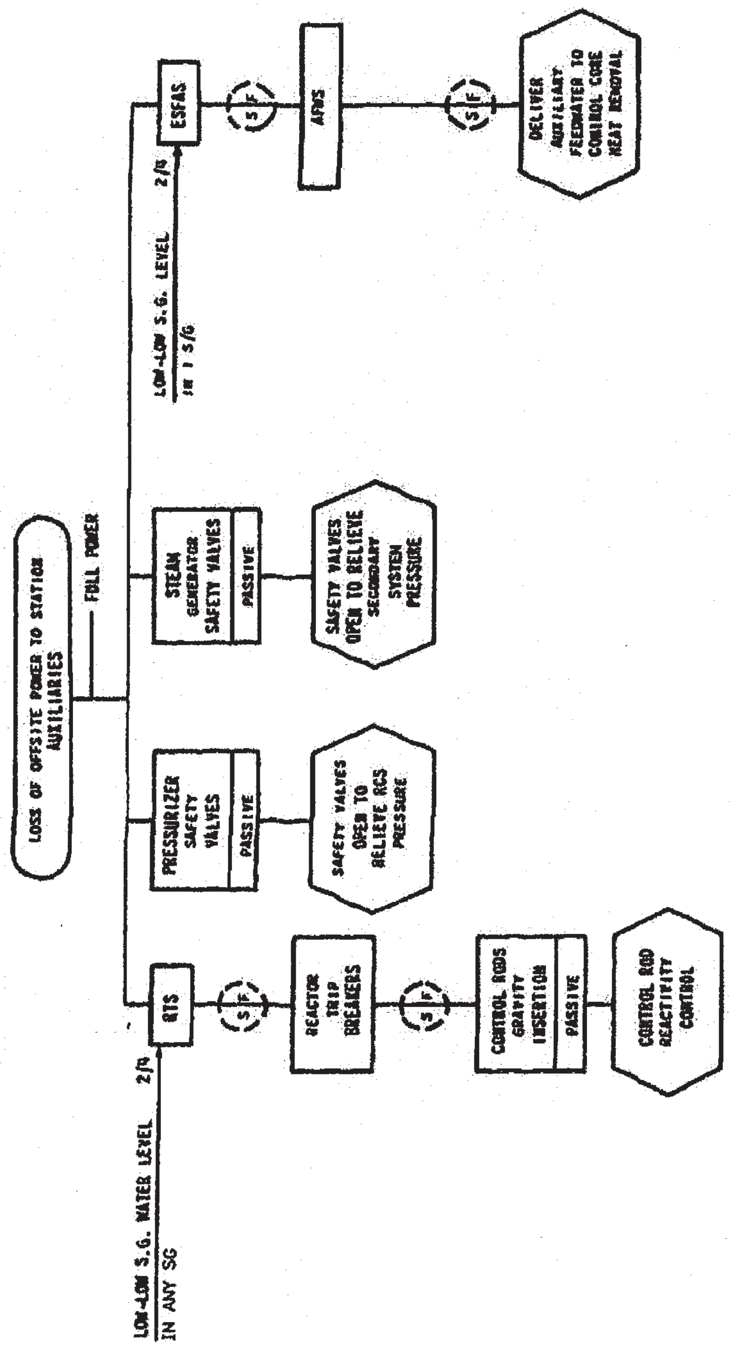

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FIGURE 15.0-12

LOSS OF OFFSITE POWER TO
STATION AUXILIARIES

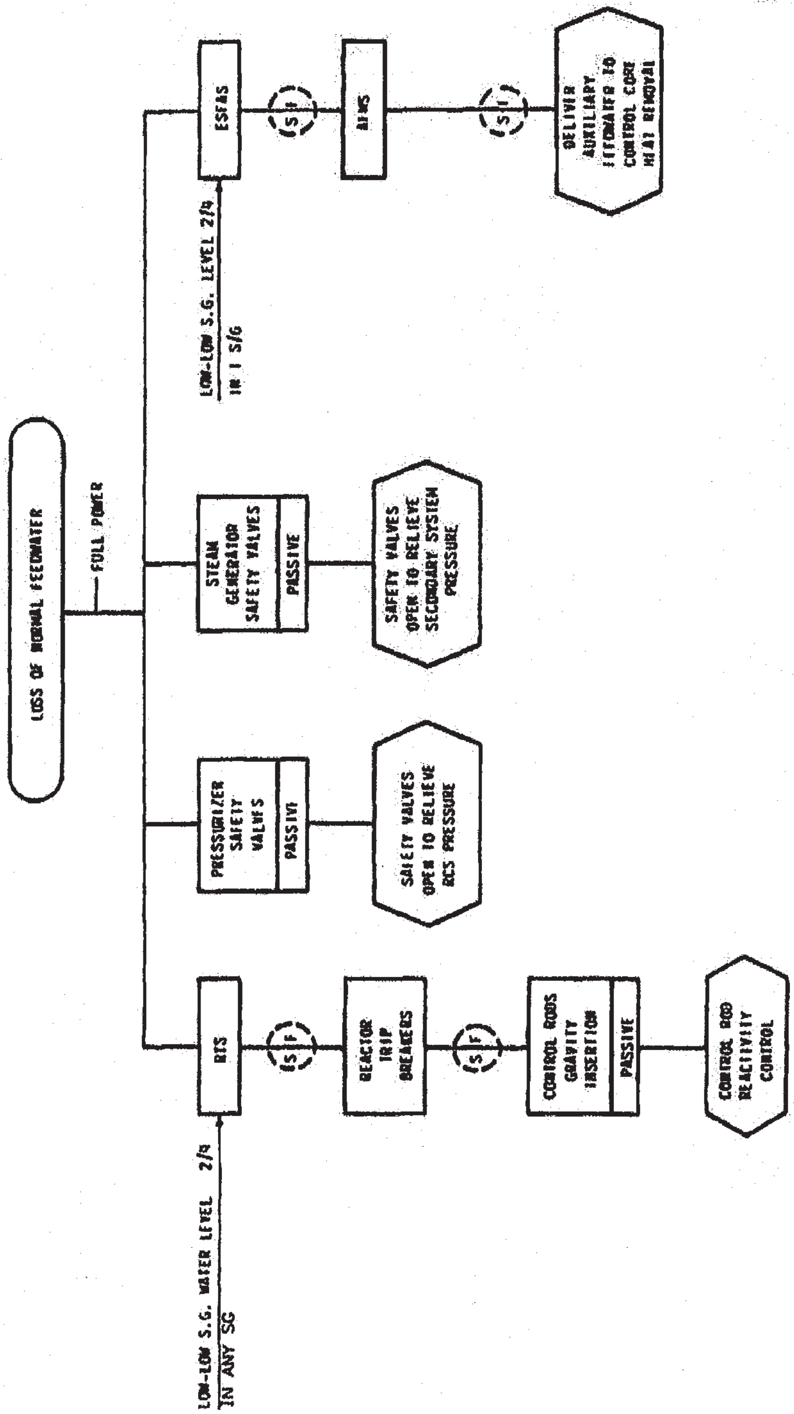

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FIGURE 15.0-13

LOSS OF NORMAL FEEDWATER

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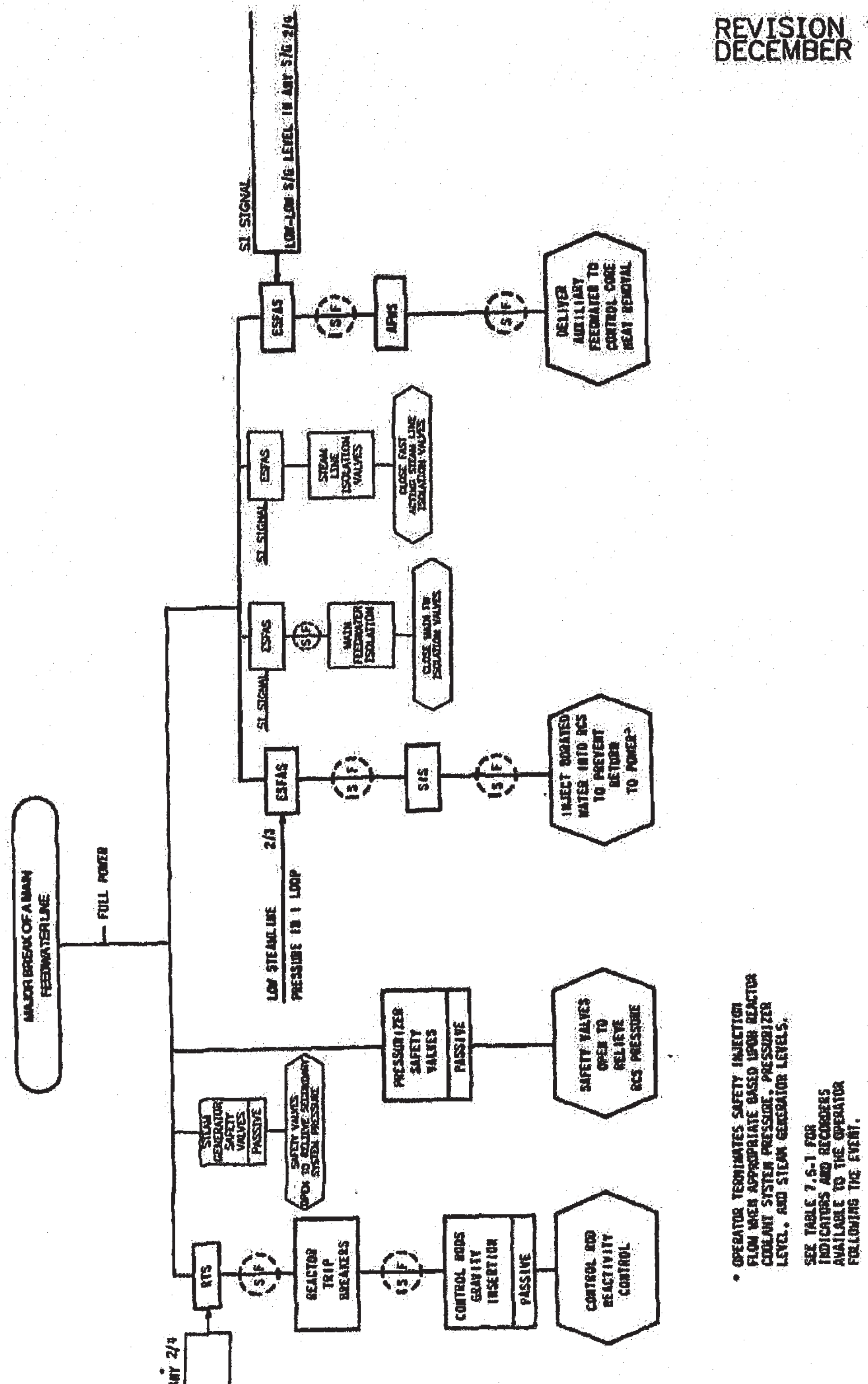

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FIGURE 15.0-14

MAJOR BREAK OF A
MAIN FEEDWATER LINE

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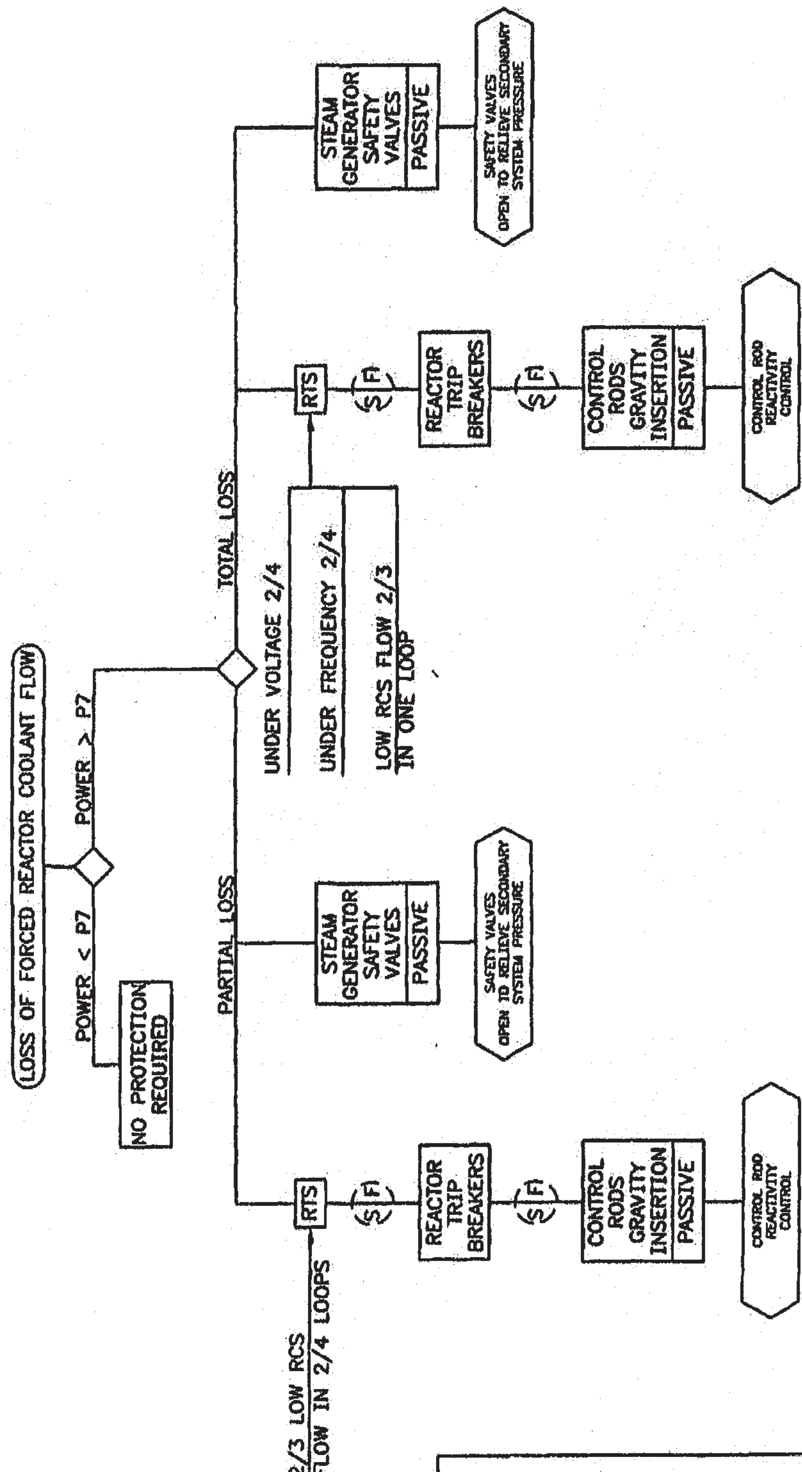

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FIGURE 15.0-15a
PARTIAL AND COMPLETE LOSS
OF FORCED REACTOR
COOLANT FLOW

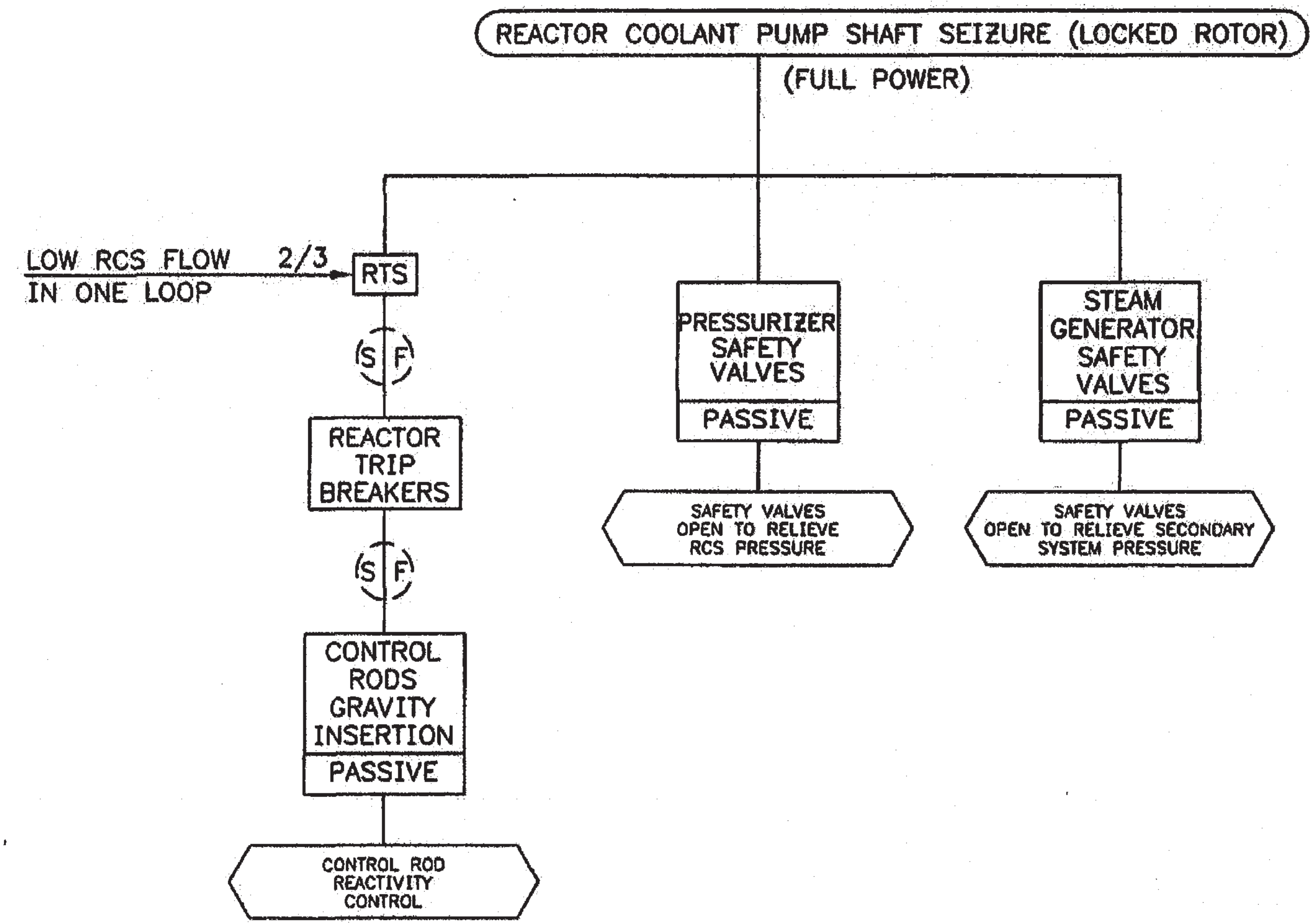

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FIGURE 15.0-15b

REACTOR COOLANT PUMP SHAFT SEIZURE (LOCKED ROTOR)

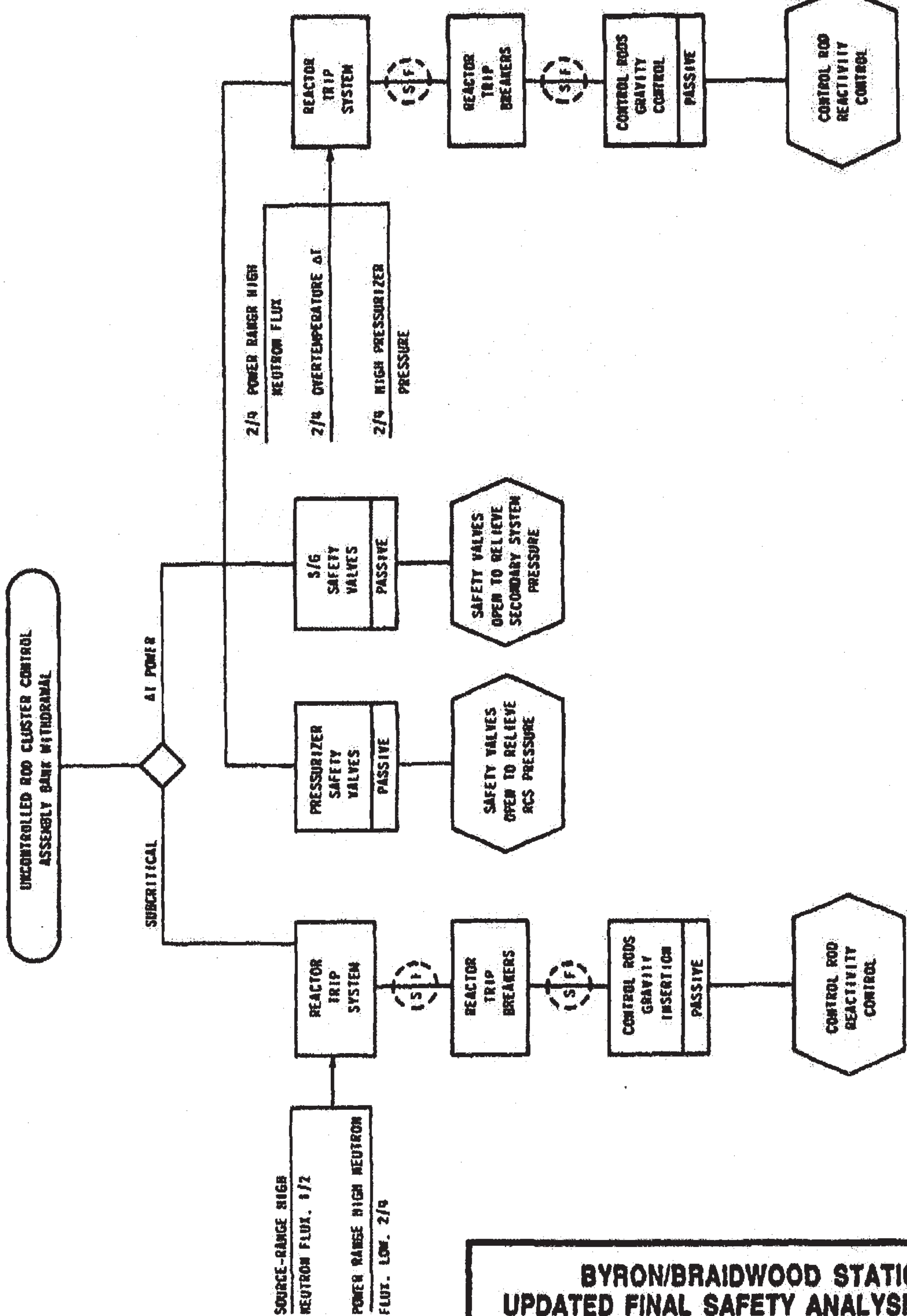

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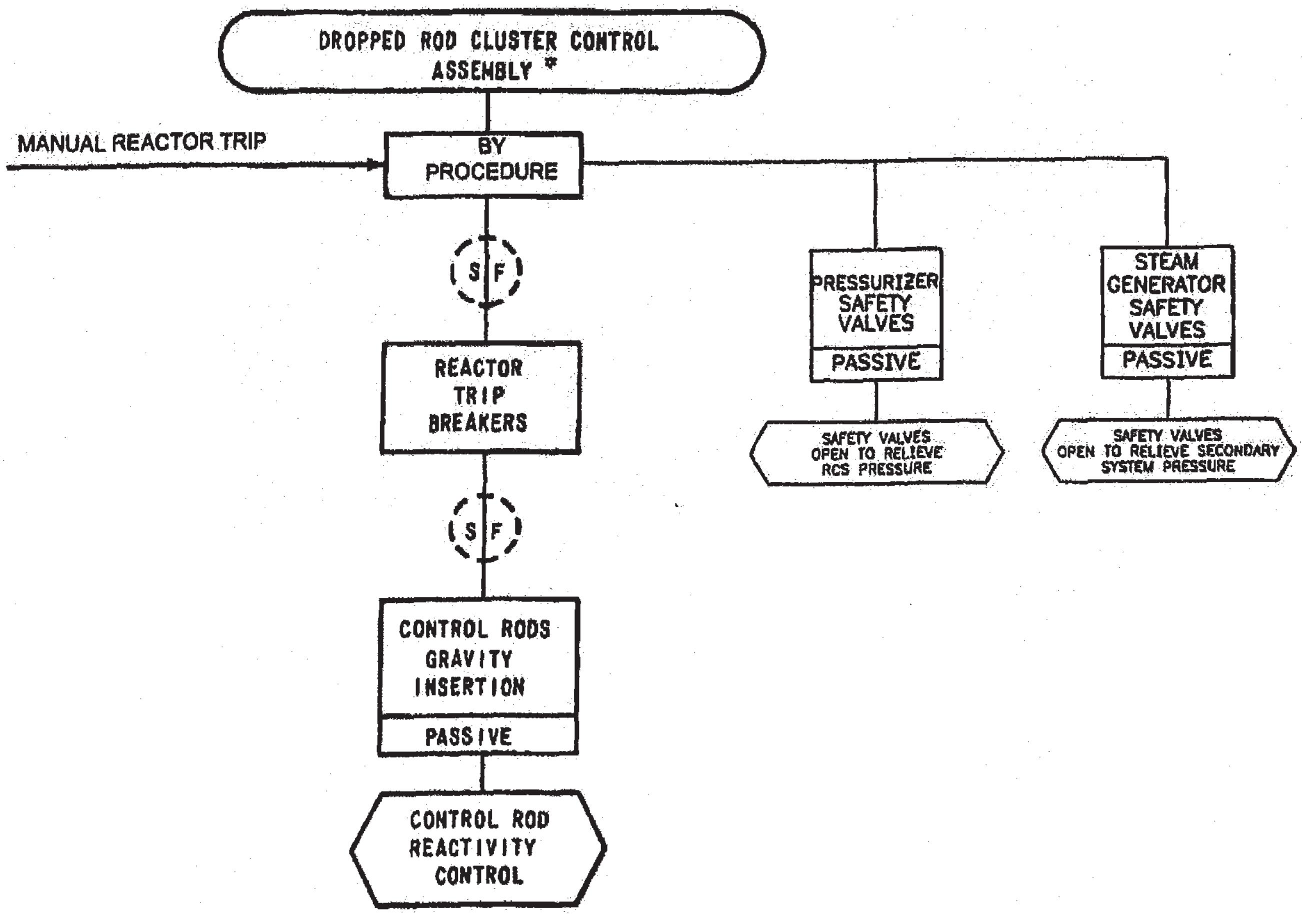
FIGURE 15.0-16

UNCONTROLLED ROD CLUSTER CONTROL ASSEMBLY BANK WITHDRAWAL



* TRIP SEQUENCE OCCURS ONLY IF REQUIRED BY PROCEDURE, OR IF AN ENTIRE RCCA BANK DROPS, FOR OTHER ROD WORTH, OR FOR MISALIGNMENT, NO AUTOMATIC PROTECTION REQUIRED.

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FIGURE 15.0-17

DROPPED ROD CLUSTER CONTROL ASSEMBLY

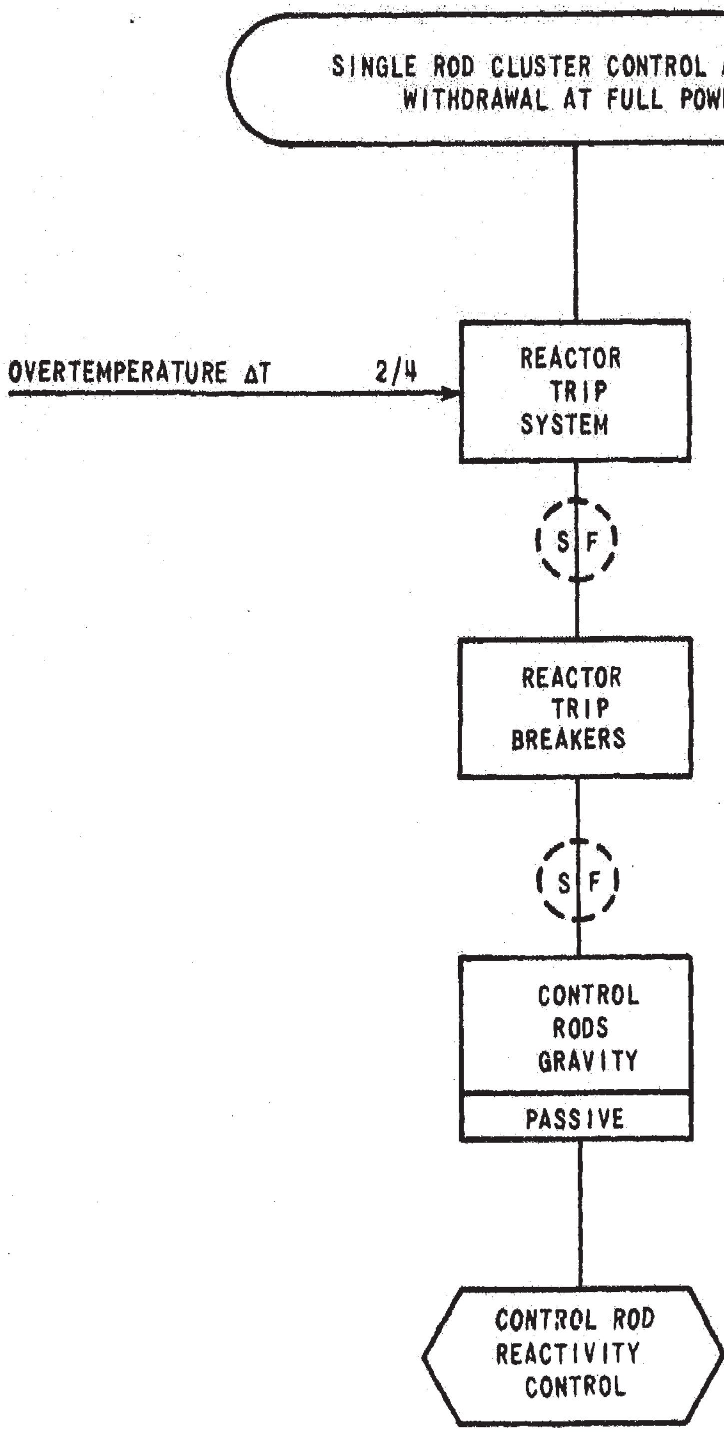

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FIGURE 15.0-18

SINGLE ROD CLUSTER CONTROL ASSEMBLY
WITHDRAWAL AT FULL POWER

Figure 15.0-19 has been deleted intentionally.

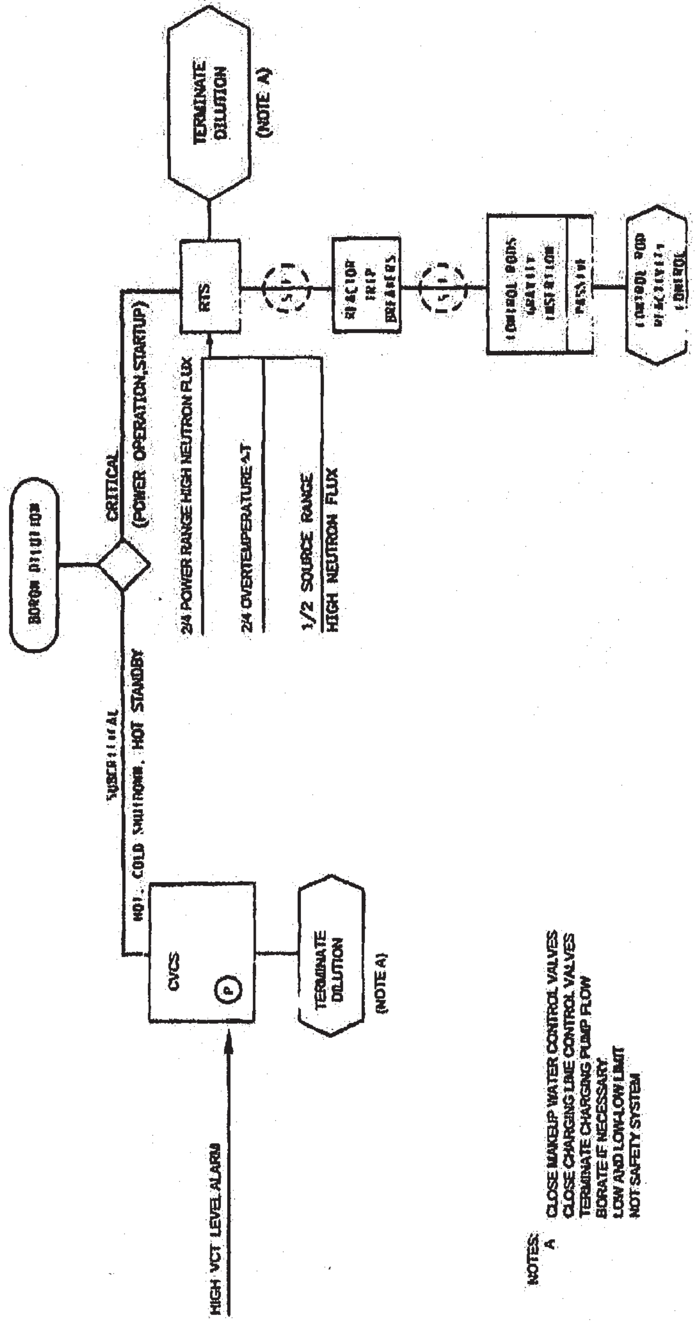

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FIGURE 15.0-20

BORON DILUTION

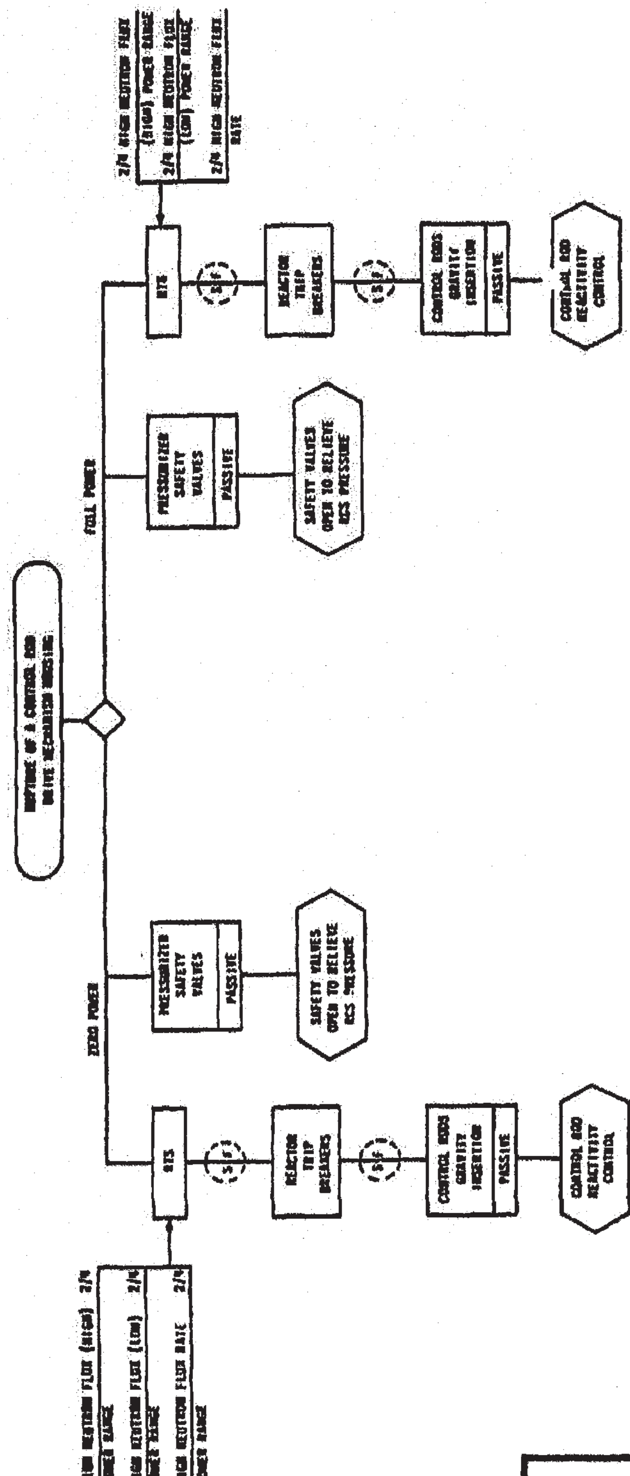

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FIGURE 15.0-21

RUPTURE OF CONTROL ROD DRIVE MECHANISM HOUSING

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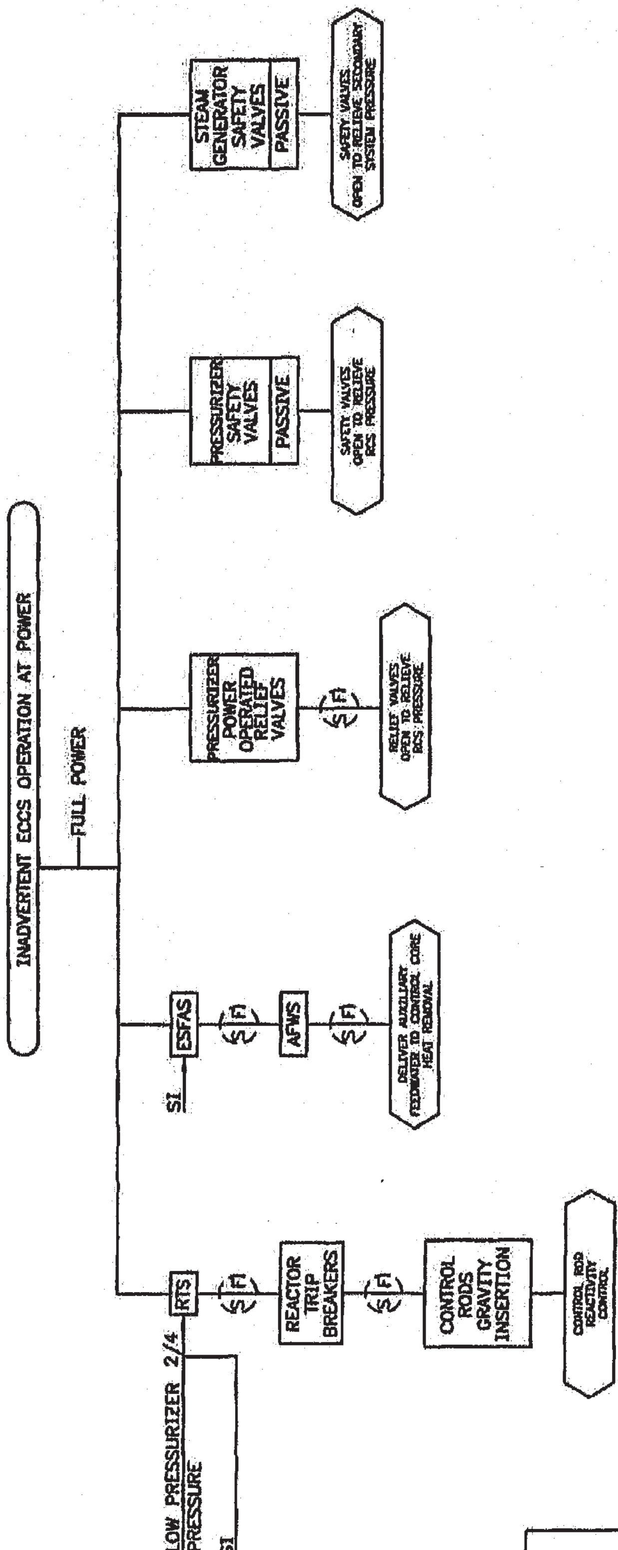

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FIGURE 15.0-22

INADVERTENT ECCS OPERATION AT POWER

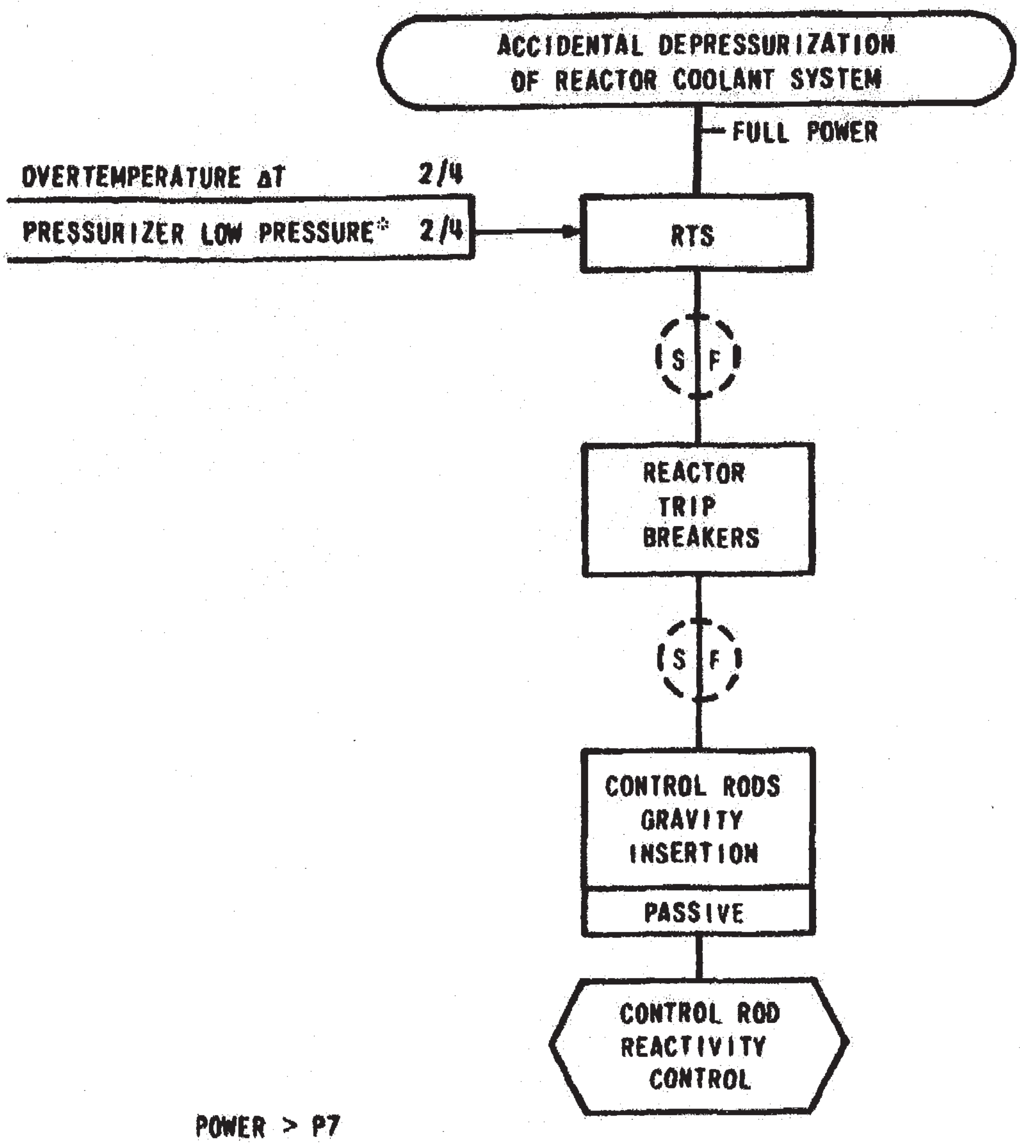

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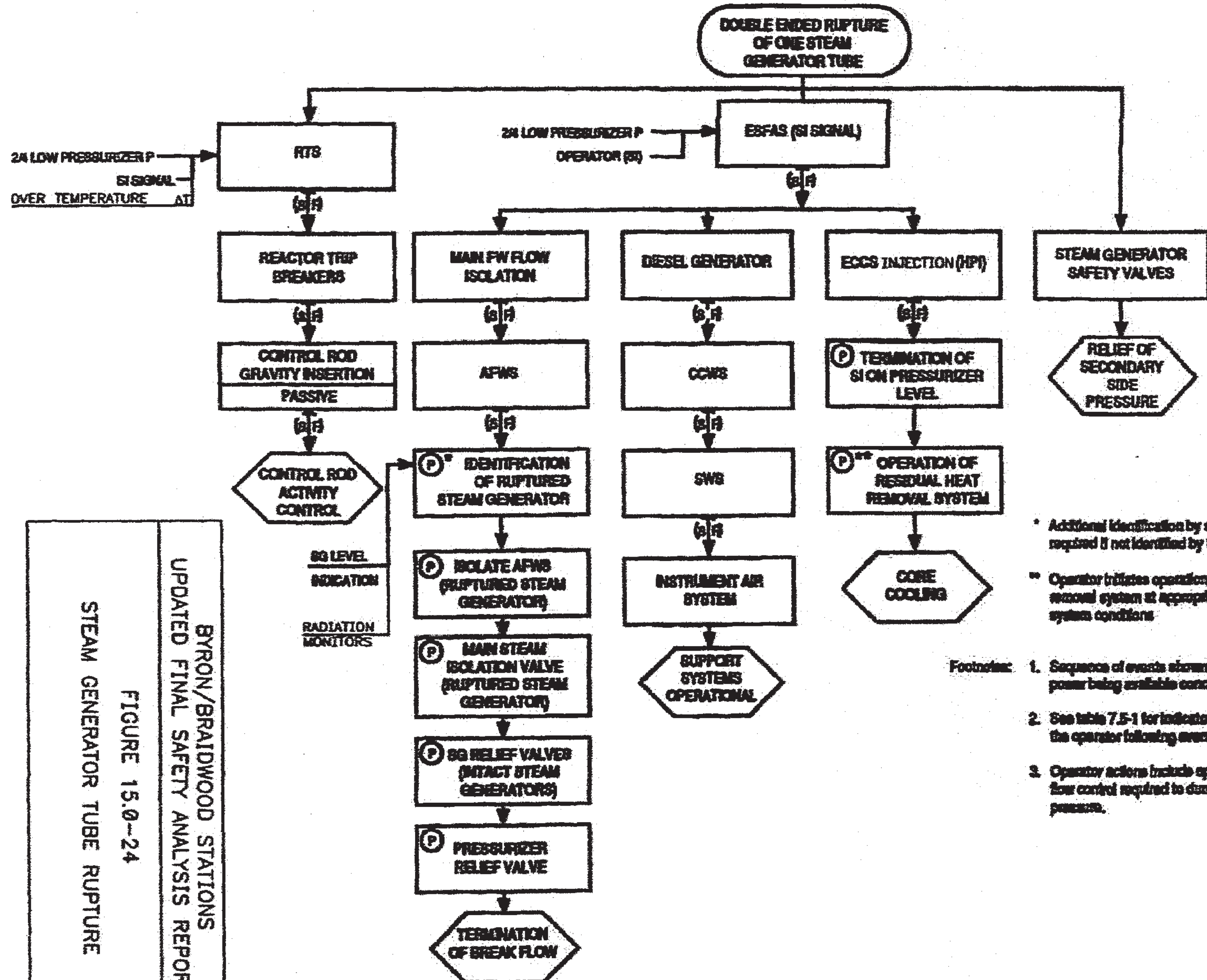
FIGURE 15.0-23

ACCIDENTAL DEPRESSURIZATION OF
REACTOR COOLANT SYSTEM



* Additional identification by sample SG shell side fluid required if not identified by SG level indication

** Operator initiates operation of residual heat removal system at appropriate reactor coolant system conditions

Footnotes:
1. Sequence of events shown is for a condition of no offsite power being available coincident with SG tube rupture.
2. See table 7.5-1 for indications and recorders available to the operator following event.
3. Operator actions include operation of relief valves and flow control required to dump steam and reduce RCS pressure.

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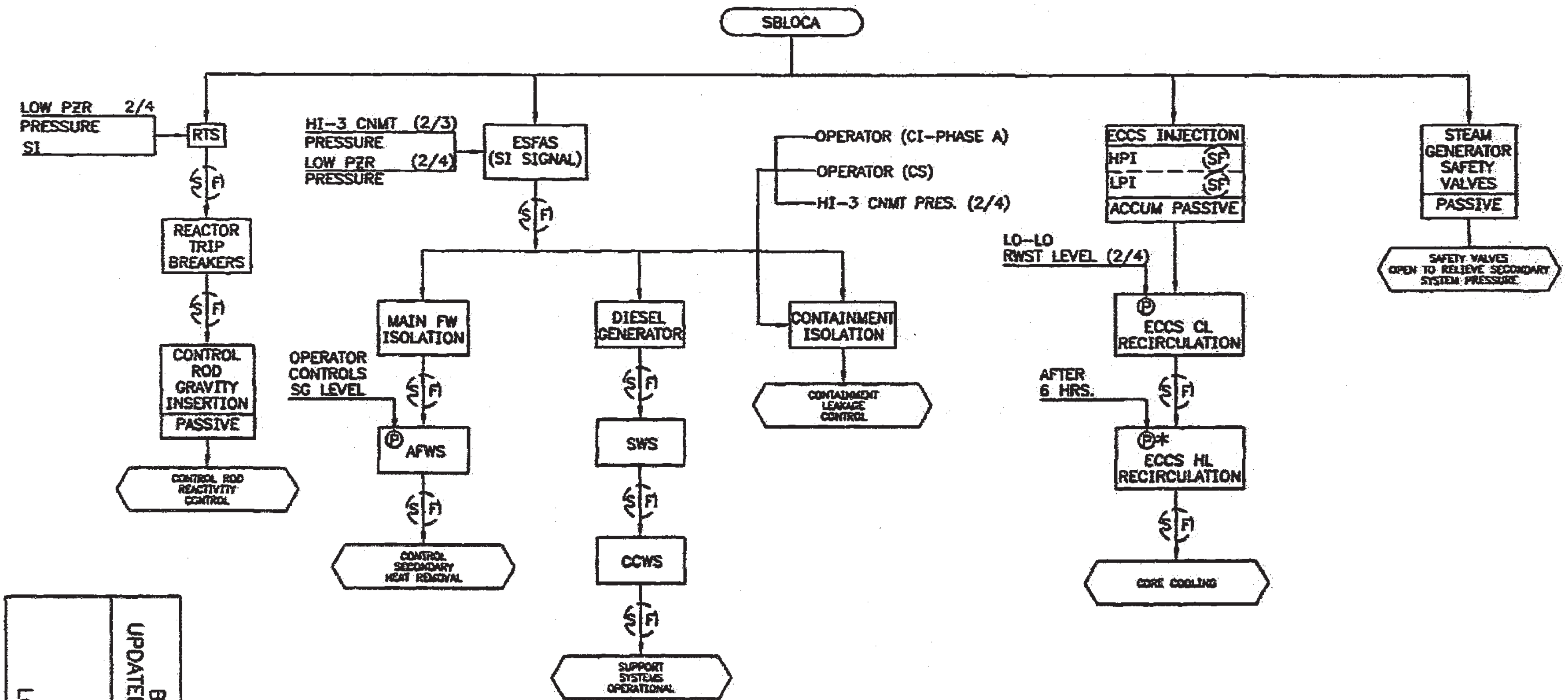
FIGURE 15.0-24

STEAM GENERATOR TUBE RUPTURE



* OPERATOR ACTION INCLUDES VALVE ALIGNMENT AND PUMP CONTROL REQUIRED FOR RECIRCULATION.

FOOTNOTE: SEE TABLE 7.5-1 FOR INDICATORS AND RECORDERS AVAILABLE TO THE OPERATOR FOLLOWING EVENT.

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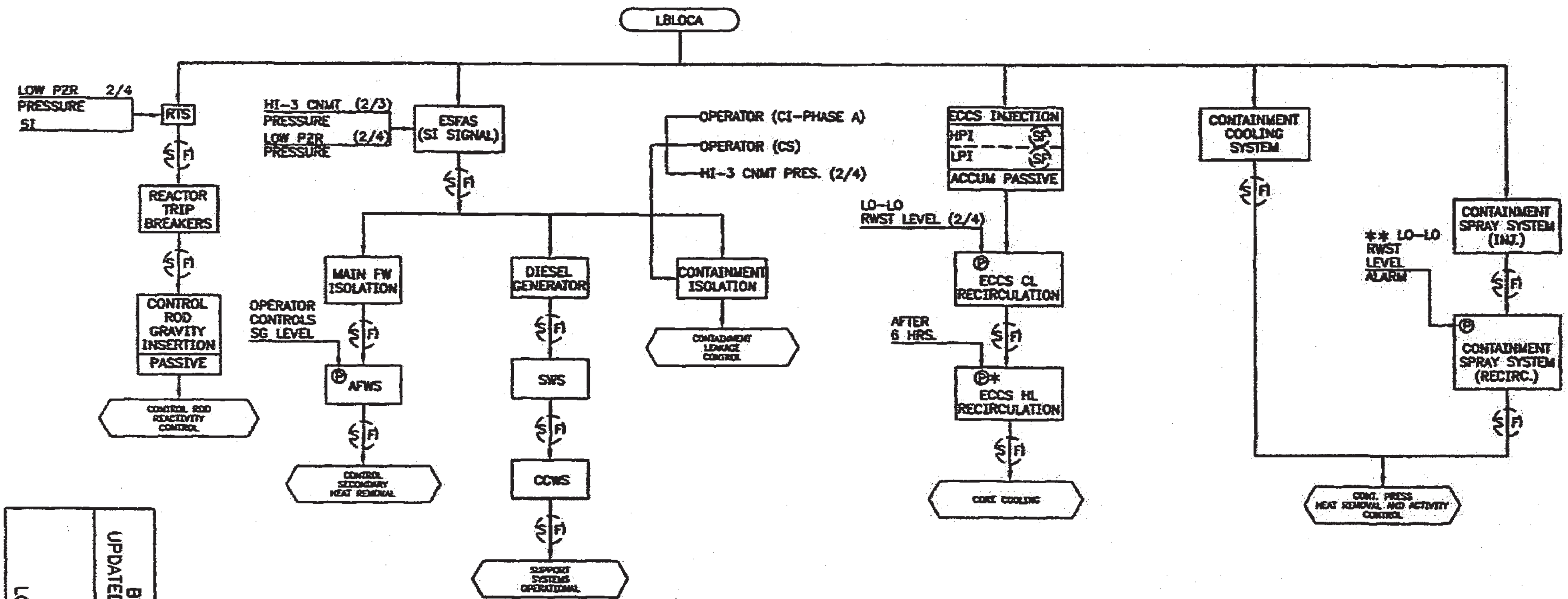
FIGURE 15.0-25a

SMALL BREAK
LOSS-OF-COOLANT ACCIDENT



* OPERATOR ACTION INCLUDES VALVE ALIGNMENT AND PUMP CONTROL REQUIRED FOR RECIRCULATION.

** OPERATOR ALIGNS SPRAY PUMP SUCTION FOR RECIRCULATION TO CONTAINMENT RECIRCULATION SUMPS ON LO-LO RWST LEVEL ALARM.

FOOTNOTE: SEE TABLE 7.5-1 FOR INDICATORS AND RECORDERS AVAILABLE TO THE OPERATOR FOLLOWING EVENT.

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FIGURE 15.0-25b

LARGE BREAK
LOSS-OF-COOLANT ACCIDENT

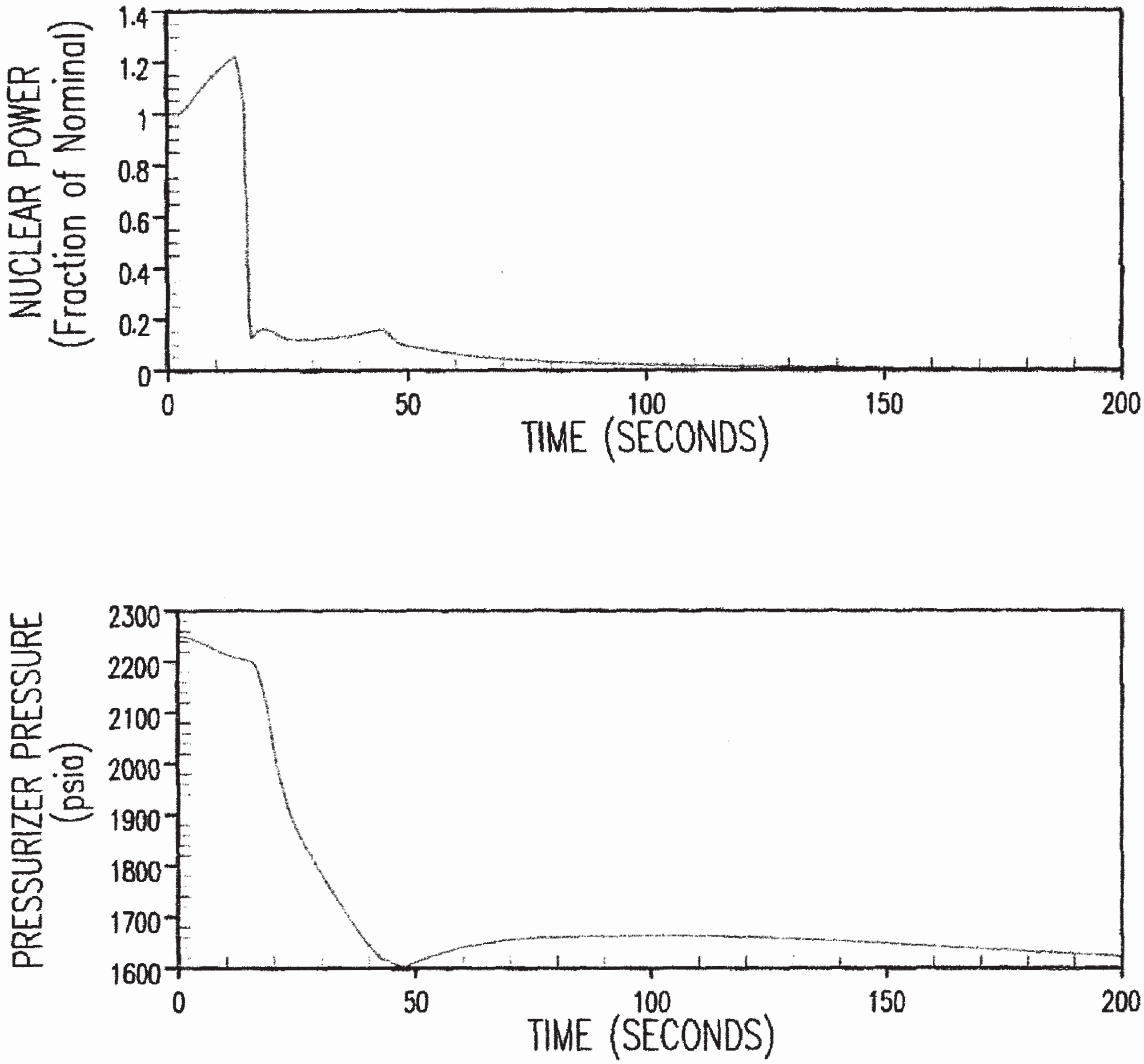

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FIGURE 15.1-1a

NUCLEAR POWER TRANSIENT
PRESSURIZER PRESSURE TRANSIENT FOR
FEEDWATER TEMPERATURE REDUCTION

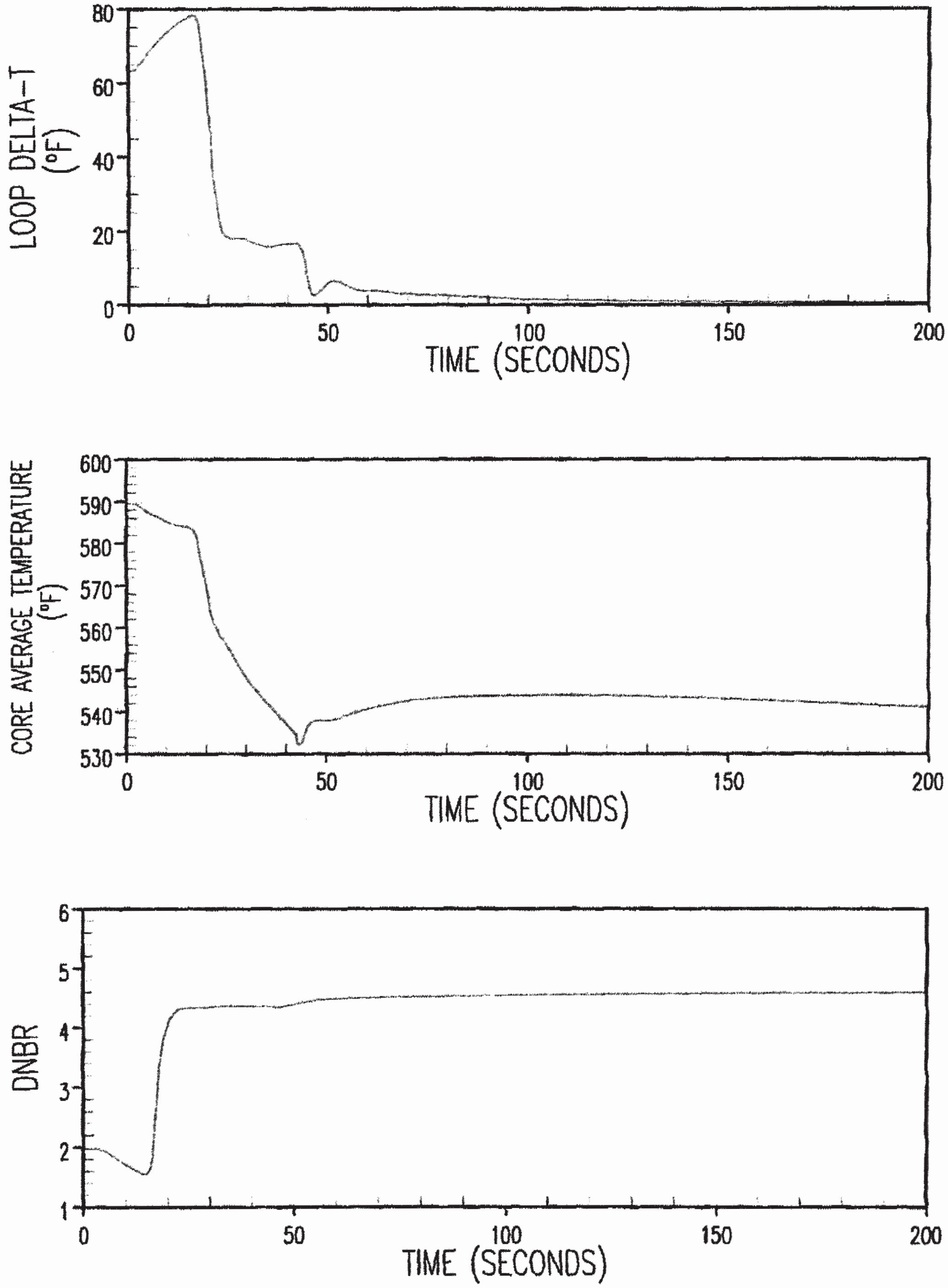

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FIGURE 15.1-1b
REACTOR COOLANT LOOP ΔT TRANSIENT
CORE AVERAGE TEMPERATURE TRANSIENT
AND DNBR TRANSIENT FOR
FEEDWATER TEMPERATURE REDUCTION

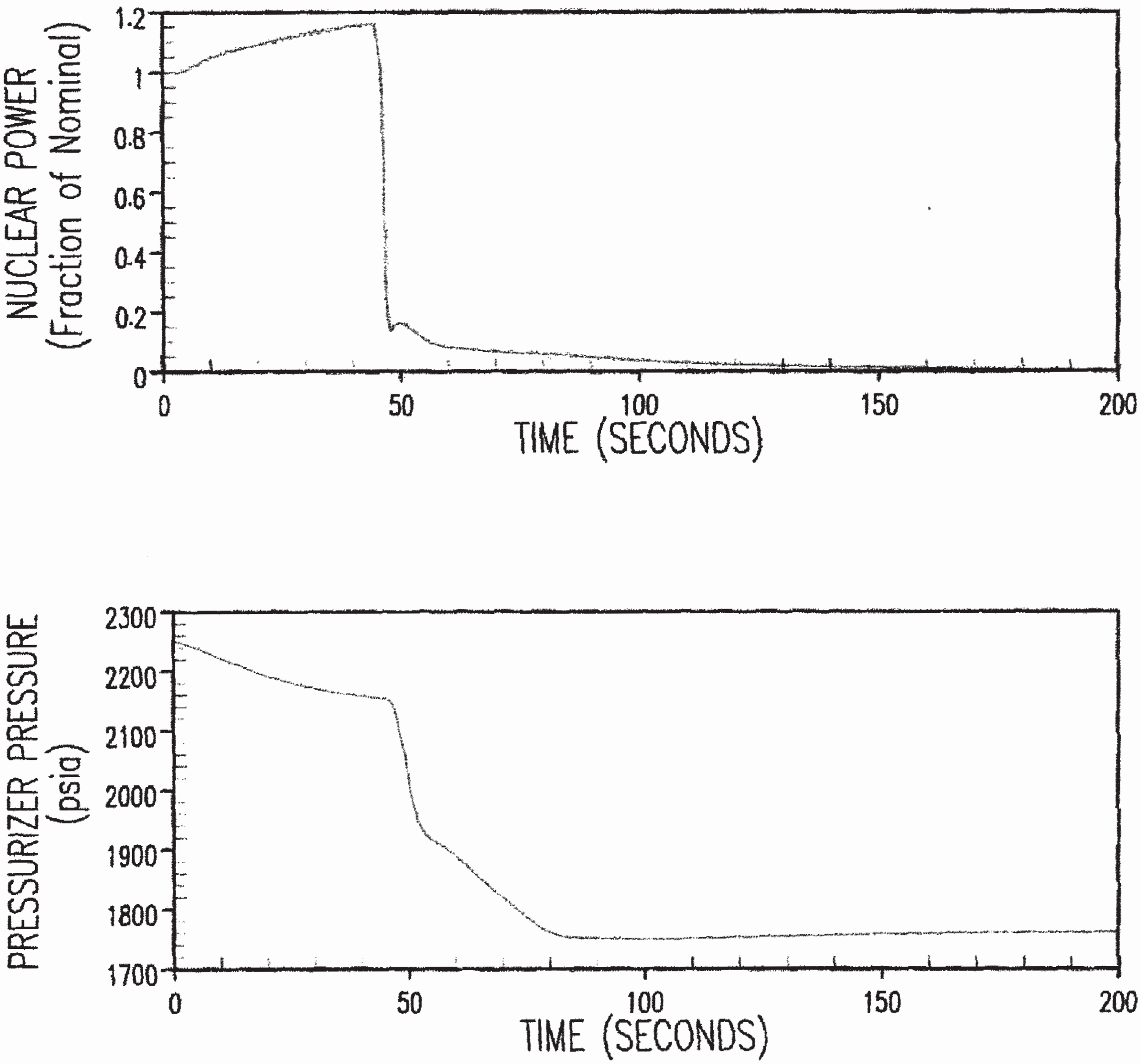

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FIGURE 15.1-2a

NUCLEAR POWER TRANSIENT
PRESSIRIZER PRESURE TRANSIENT FOR
FEEDWATER CONTROL VALVE MALFUNCTION

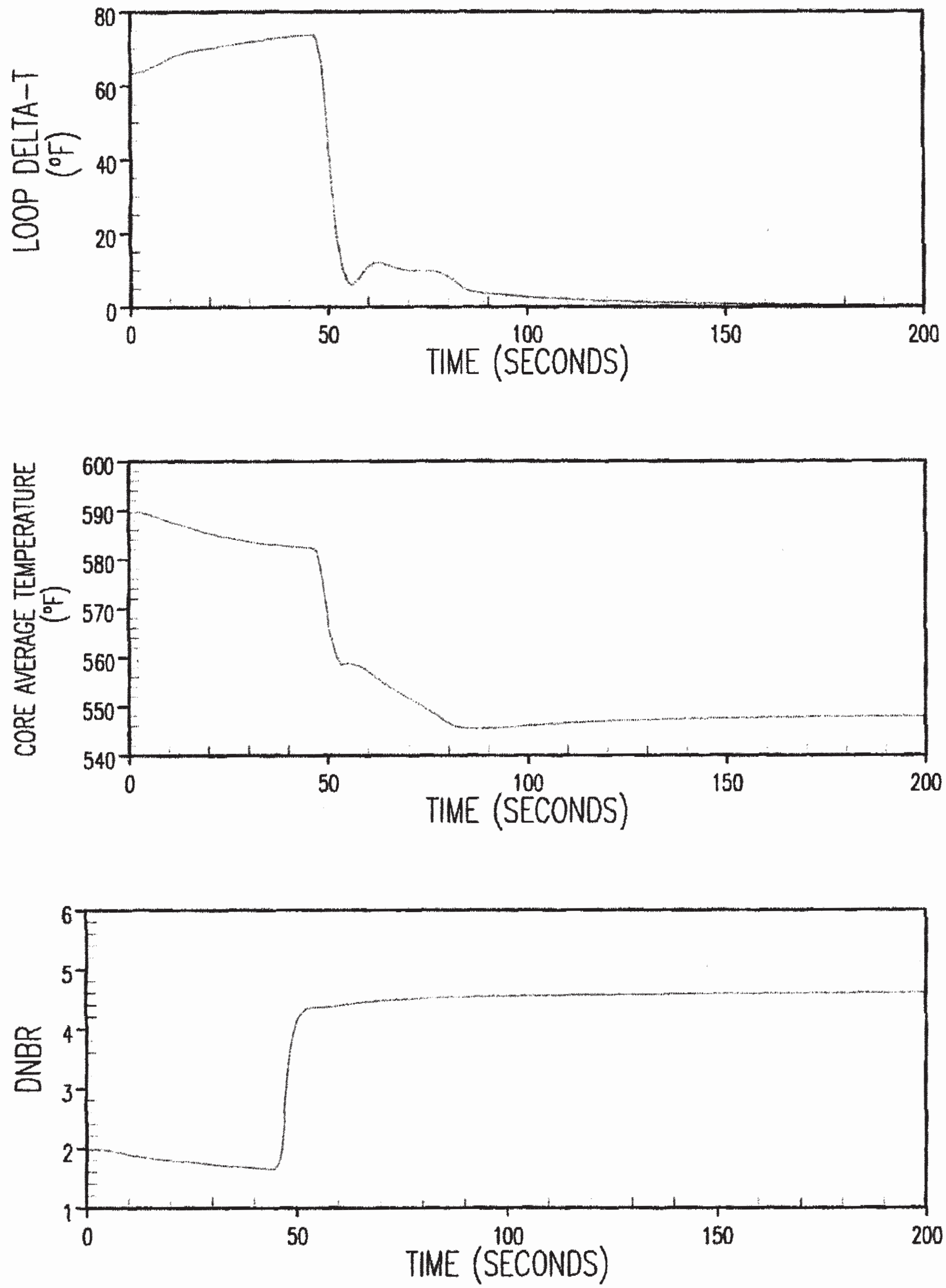

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FIGURE 15.1-2b
REACTOR COOLANT LOOP DELTA T TRANSIENT
CORE AVERAGE TEMPERATURE TRANSIENT
AND DNBR TRANSIENT FOR
FEEDWATER CONTROL VALVE MALFUNCTION

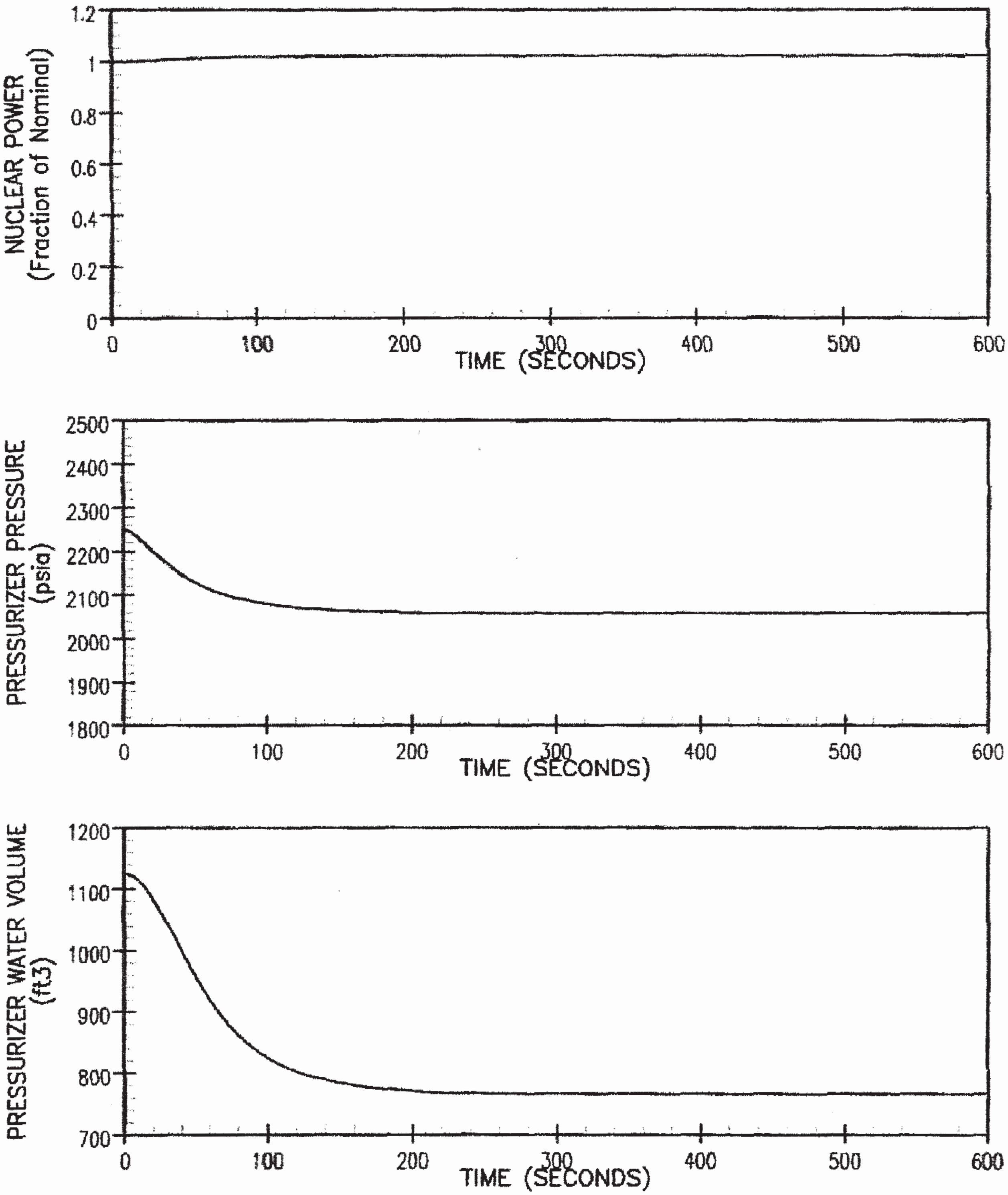

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FIGURE 15.1-3

TEN PERCENT STEP LOAD INCREASE
MINIMUM REACTIVITY FEEDBACK
MANUAL REACTOR CONTROL

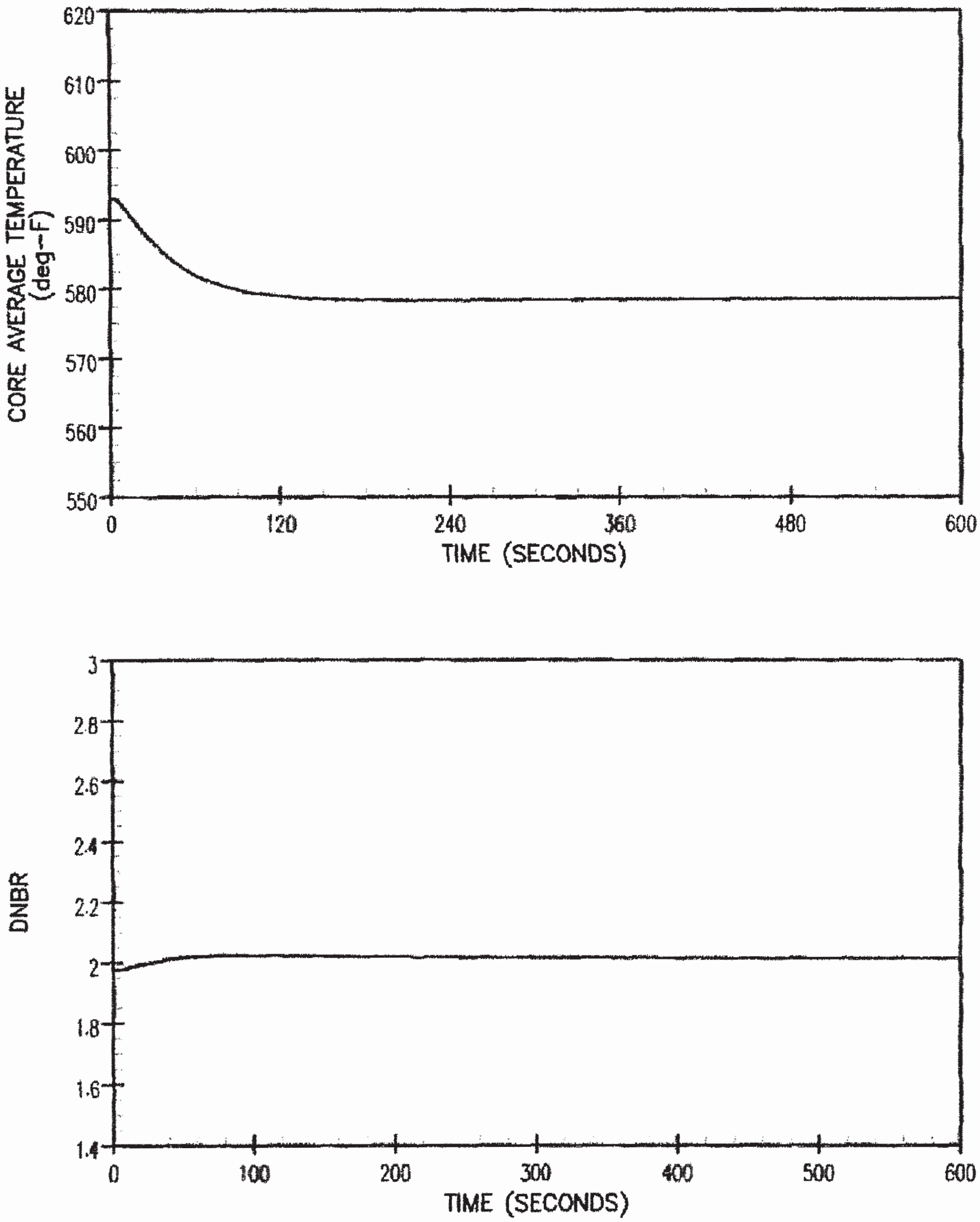

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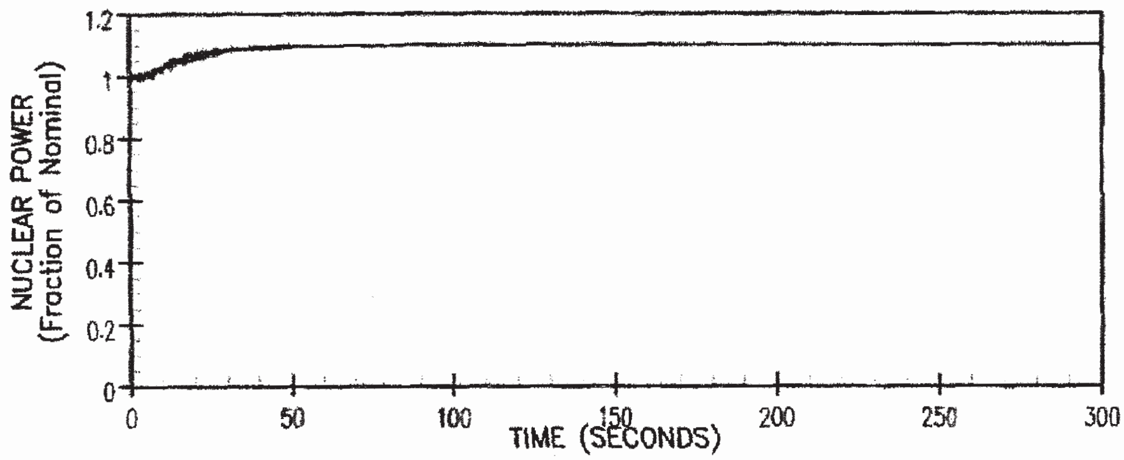
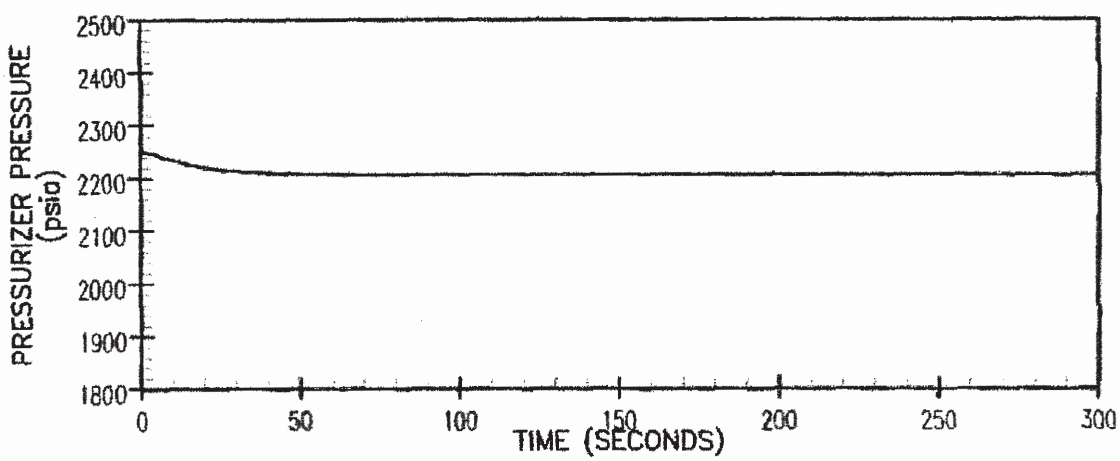
FIGURE 15.1-4

TEN PERCENT STEP LOAD INCREASE
MINIMUM REACTIVITY FEEDBACK
MANUAL REACTOR CONTROL





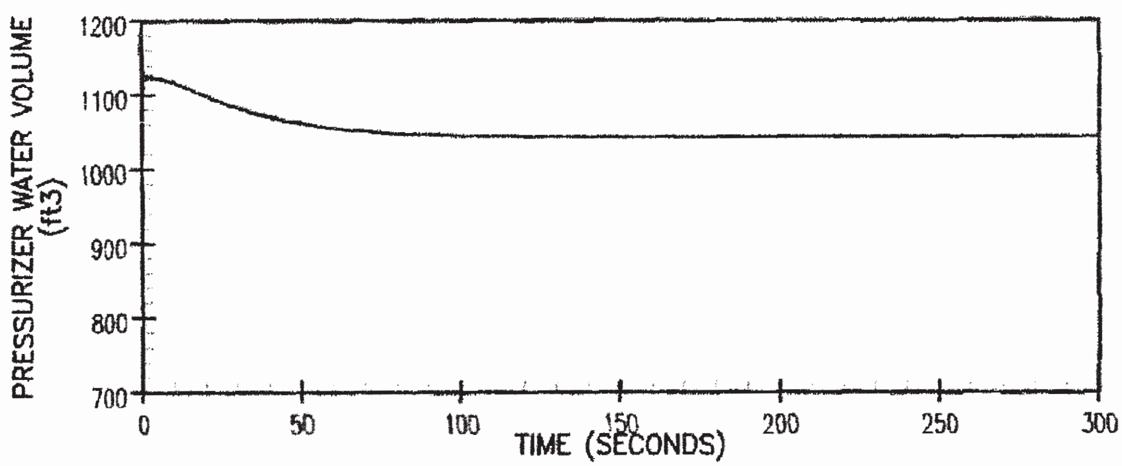


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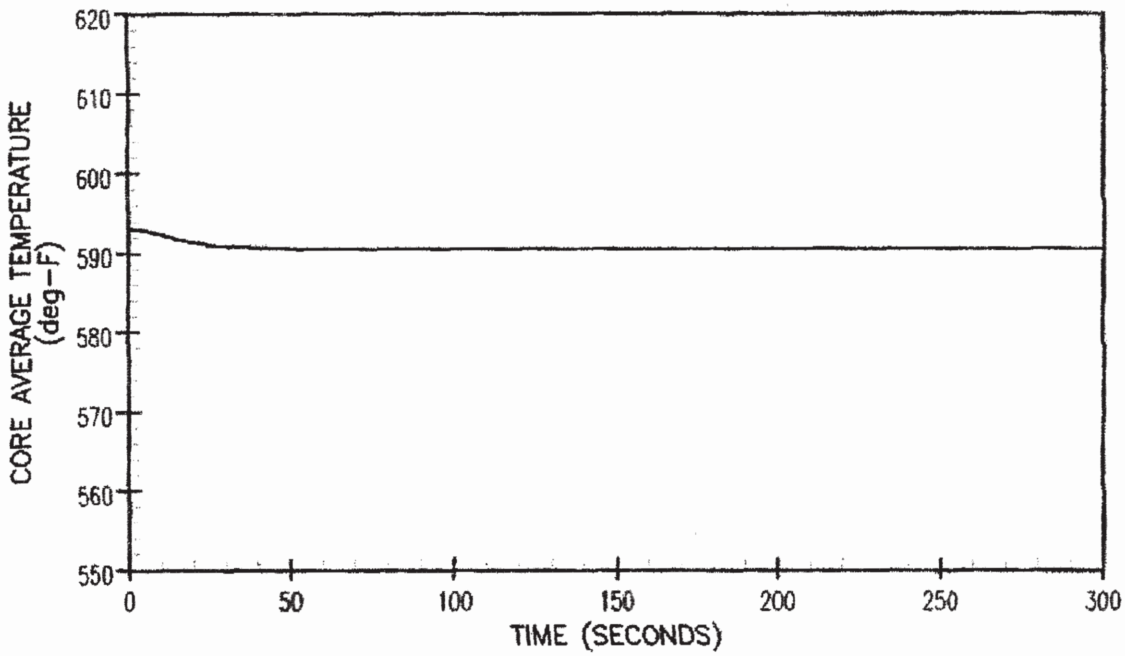
FIGURE 15.1-5

TEN PERCENT STEP LOAD INCREASE
MAXIMUM REACTIVITY FEEDBACK
MANUAL REACTOR CONTROL



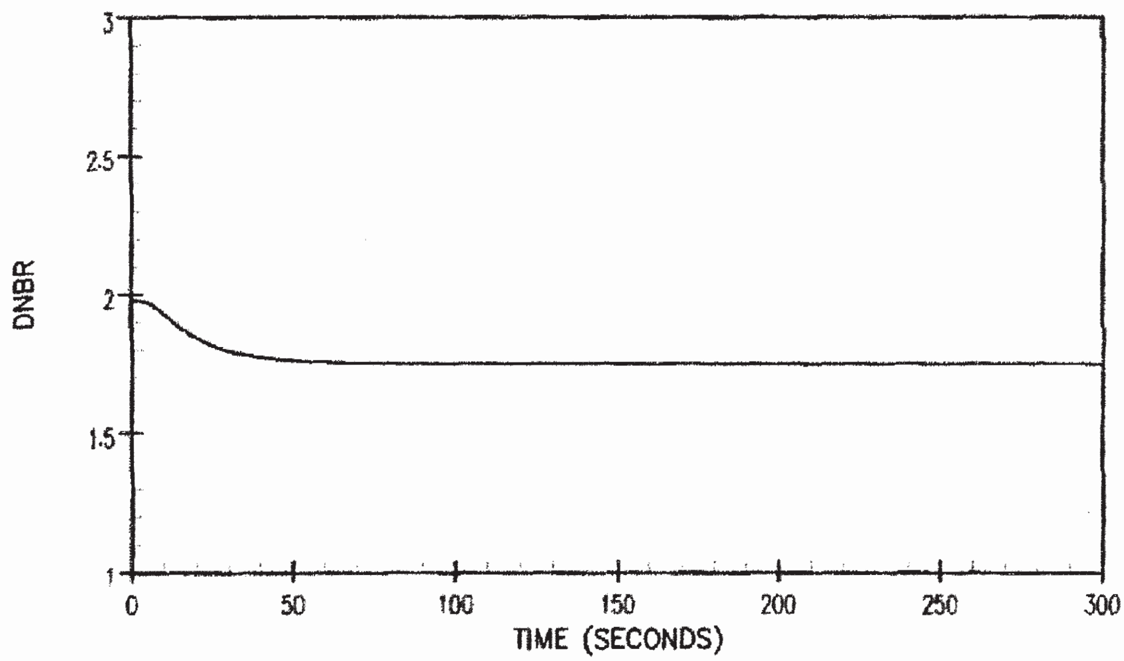


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FIGURE 15.1-6

TEN PERCENT STEP LOAD INCREASE
MAXIMUM REACTIVITY FEEDBACK
MANUAL REACTOR CONTROL

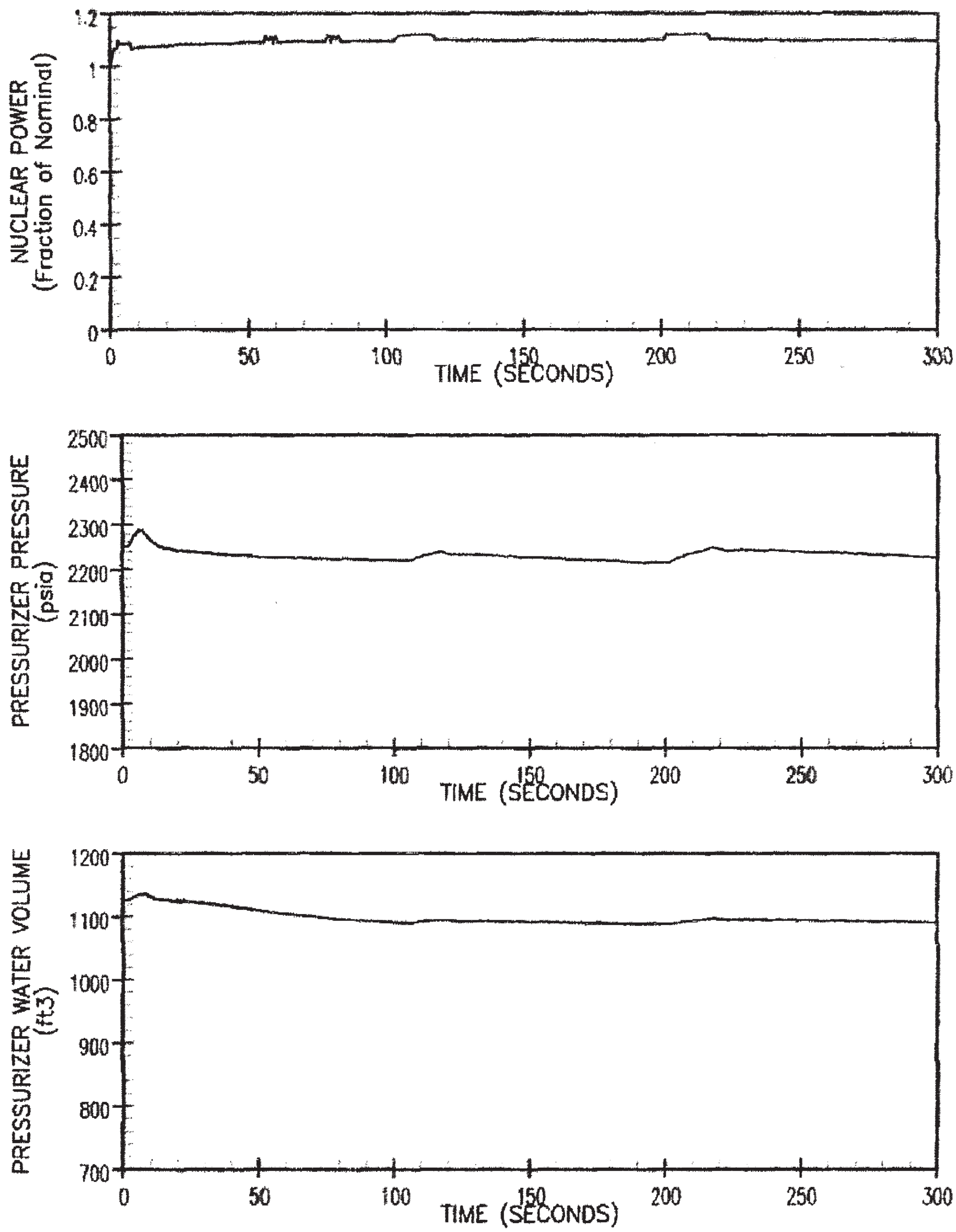

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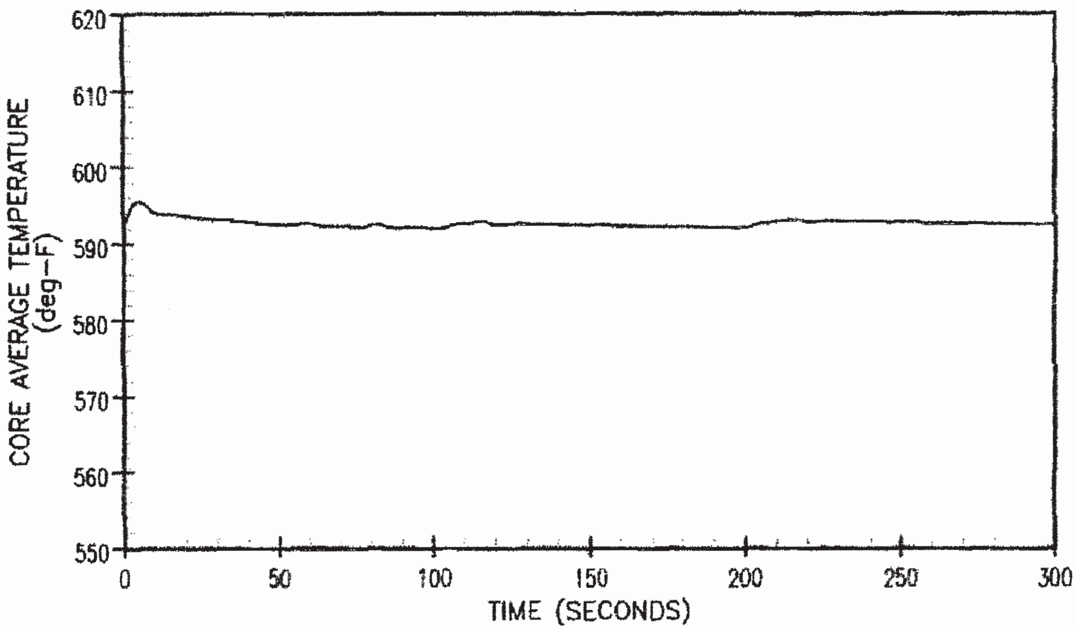
FIGURE 15.1-7

TEN PERCENT STEP LOAD INCREASE
MINIMUM REACTIVITY FEEDBACK
AUTOMATIC REACTOR CONTROL



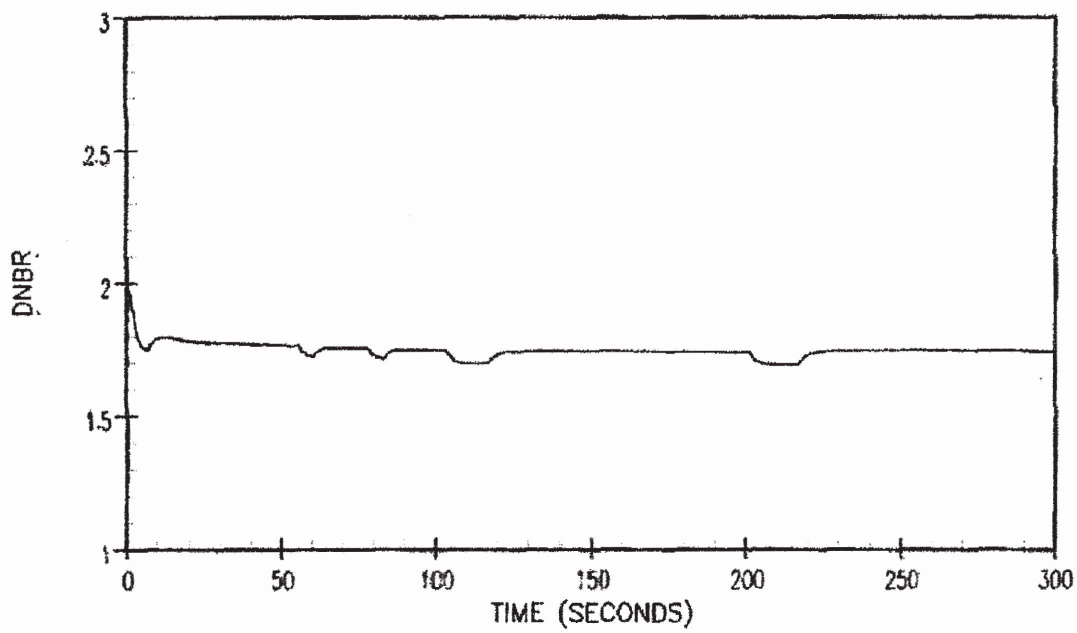


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FIGURE 15.1-8

TEN PERCENT STEP LOAD INCREASE
MINIMUM REACTIVITY FEEDBACK
AUTOMATIC REACTOR CONTROL

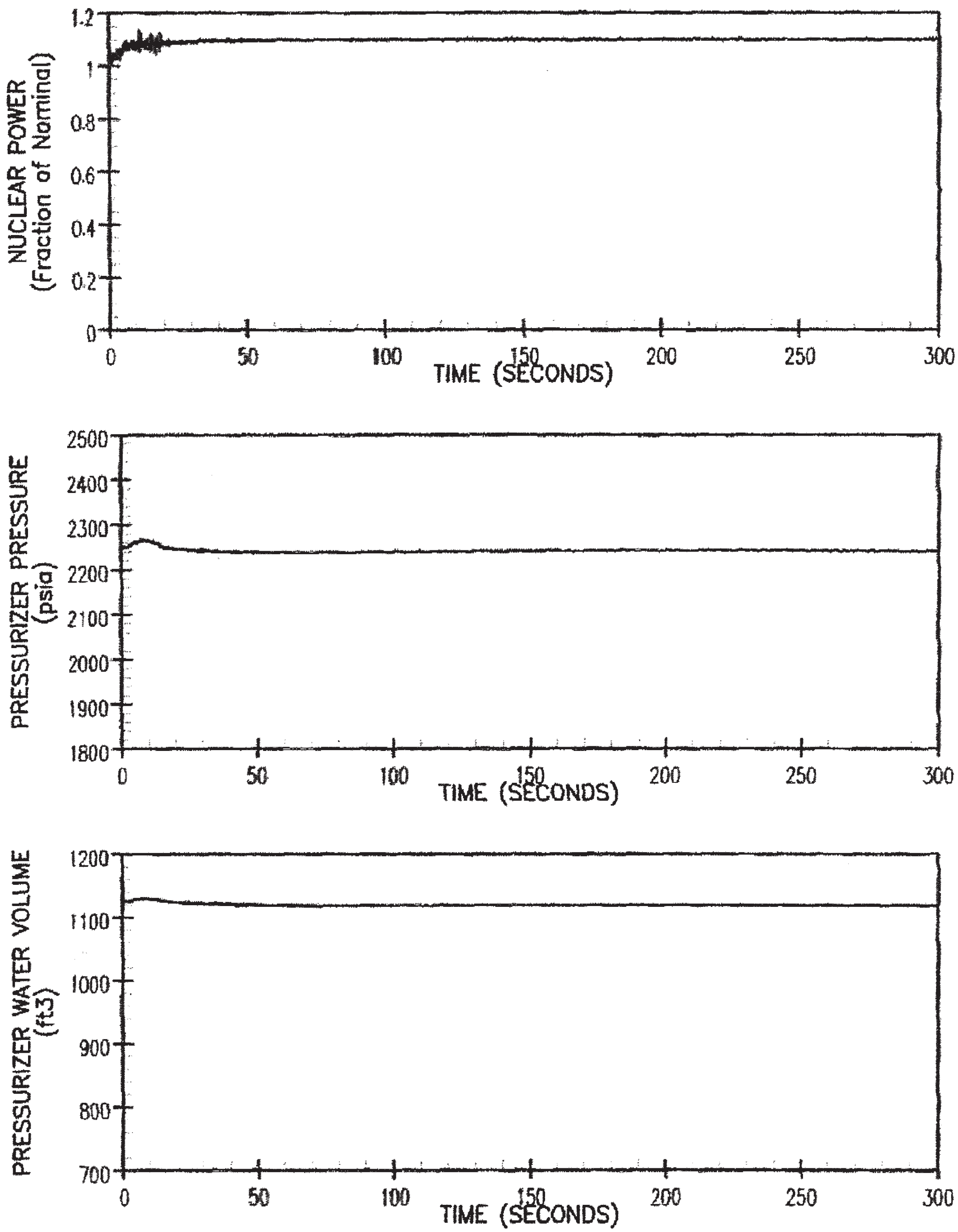

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FIGURE 15.1-9

TEN PERCENT STEP LOAD INCREASE
MAXIMUM REACTIVITY FEEDBACK
AUTOMATIC REACTOR CONTROL

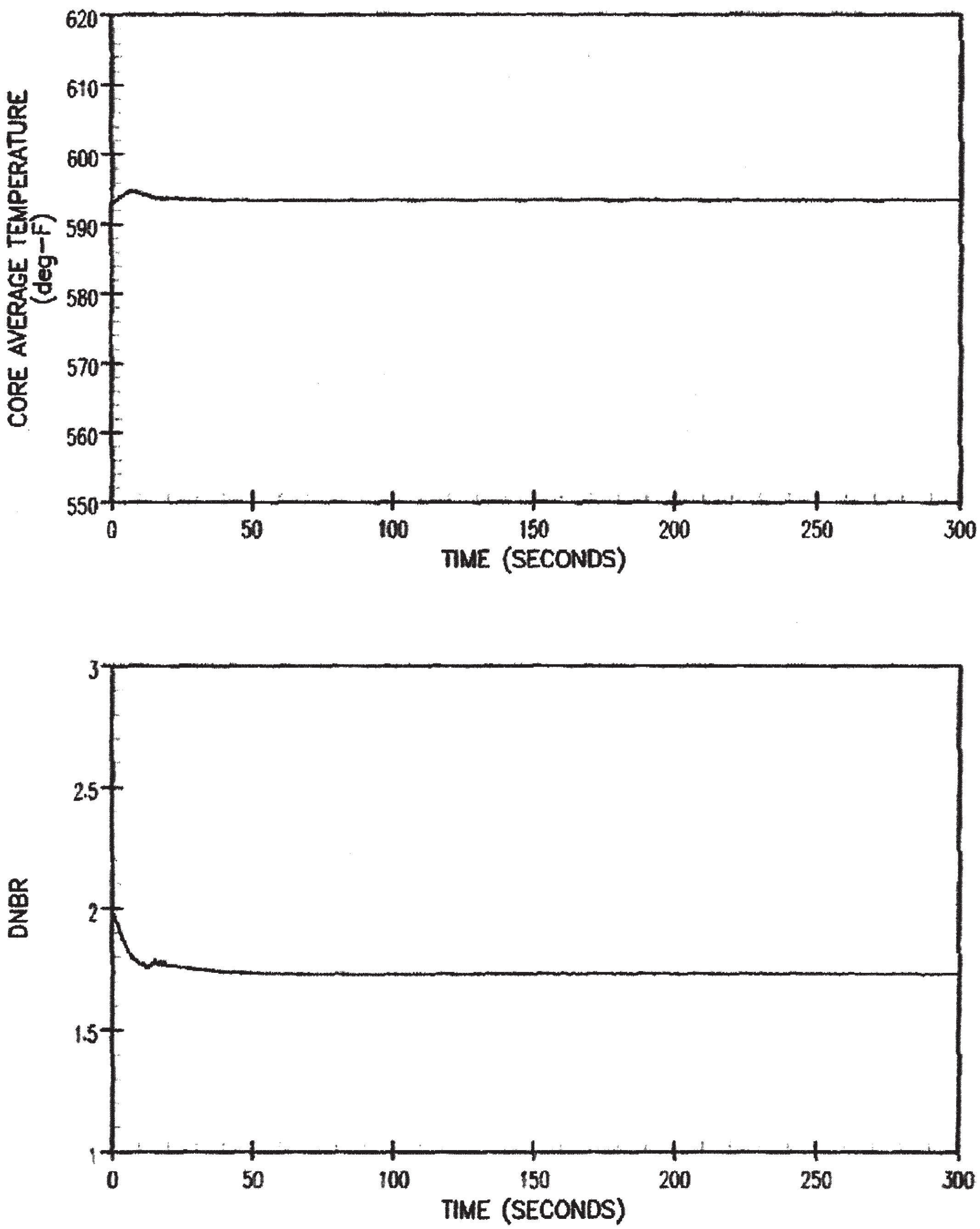

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FIGURE 15.1-10

TEN PERCENT STEP LOAD INCREASE
MAXIMUM REACTIVITY FEEDBACK
AUTOMATIC REACTOR CONTROL

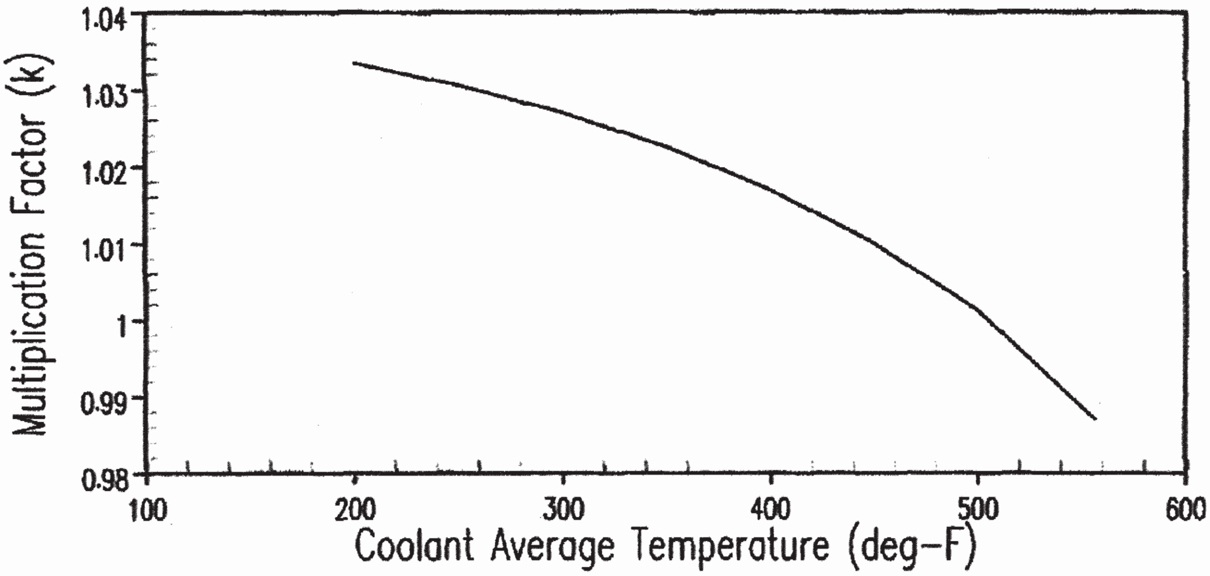

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FIGURE 15.1-11

KEFF VS COOLANT AVERAGE TEMPERATURE

Figure 15.1-11a has been deleted intentionally.

Figures 15.1-12 and 15.1-13 have been deleted intentionally.

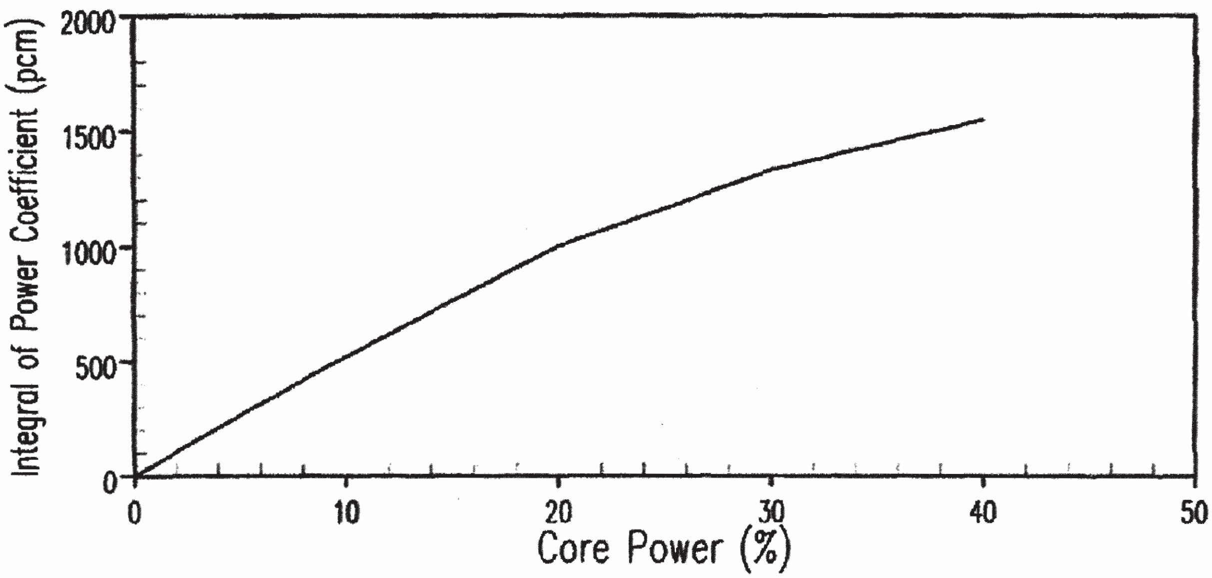

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FIGURE 15.1-14

DOPPLER POWER FEEDBACK

Figures 15.1-14a through 15.1-17 have been deleted intentionally.

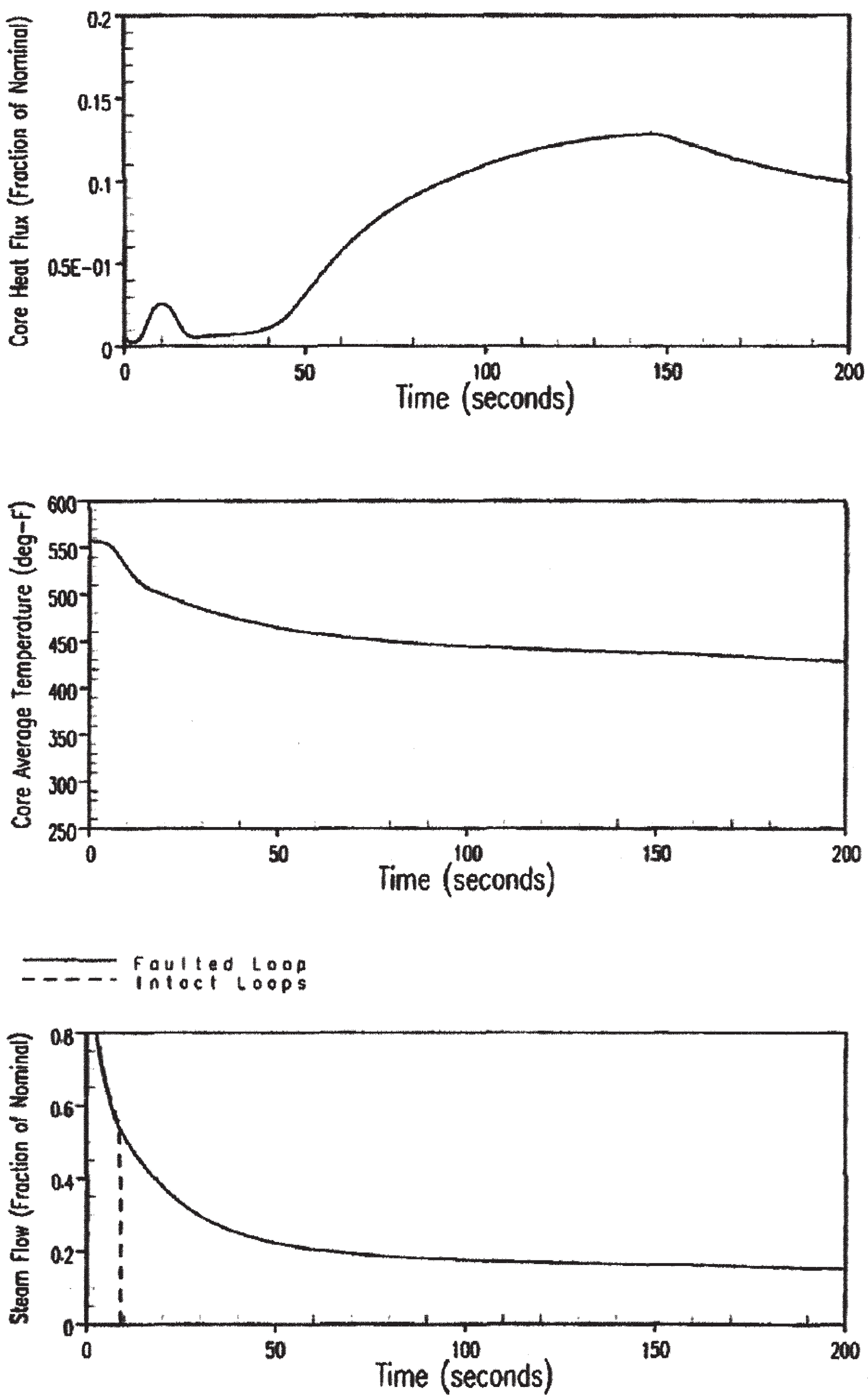

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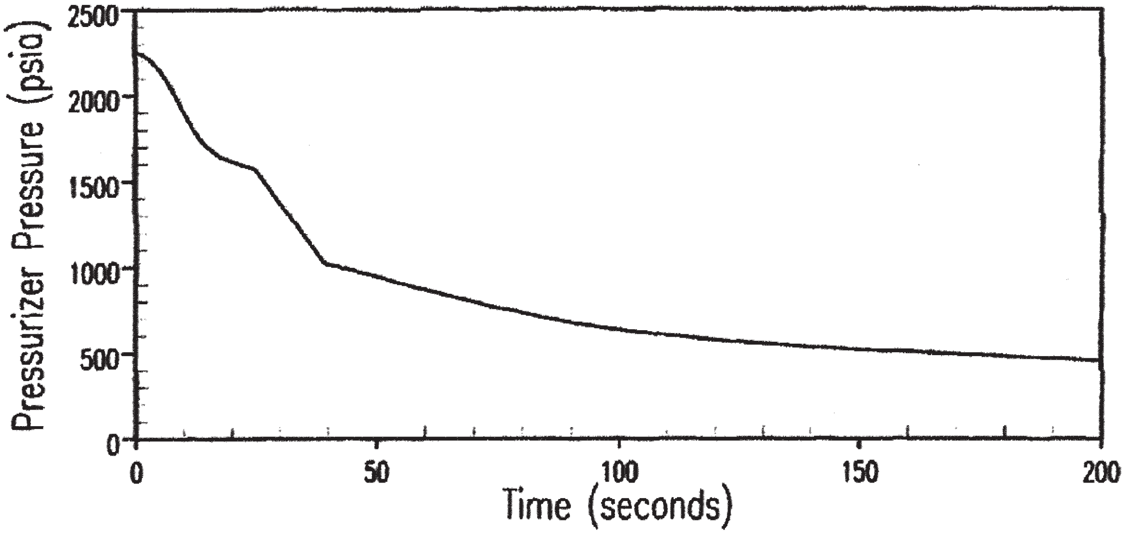
FIGURE 15.1-18

1.4 FT^2 STEAMLINE BREAK
OFFSITE POWER AVAILABLE
UNIT 2



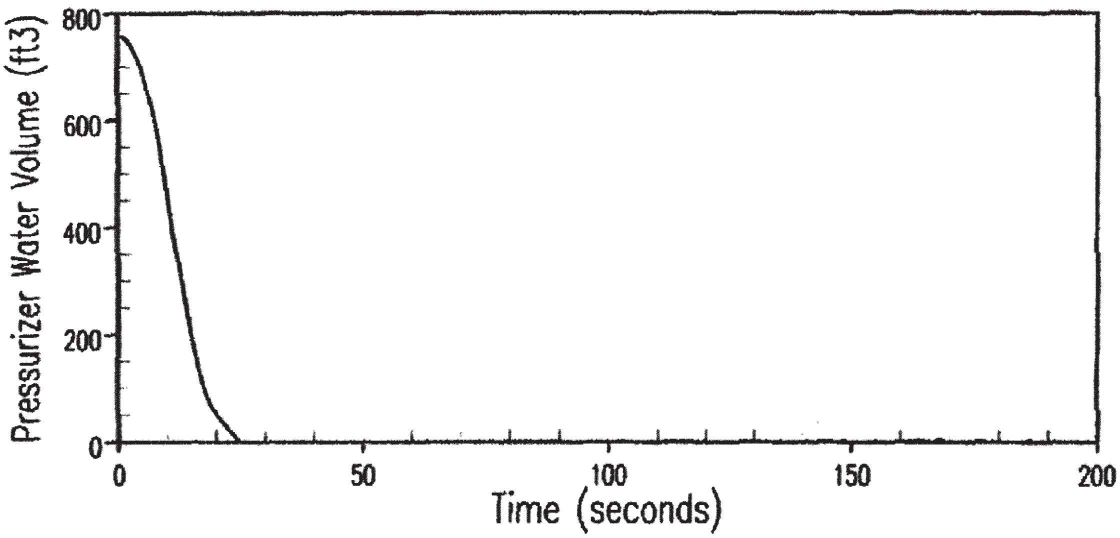


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FIGURE 15.1-19

1.4 FT^2 STEAMLINE BREAK
OFFSITE POWER AVAILABLE
UNIT 2

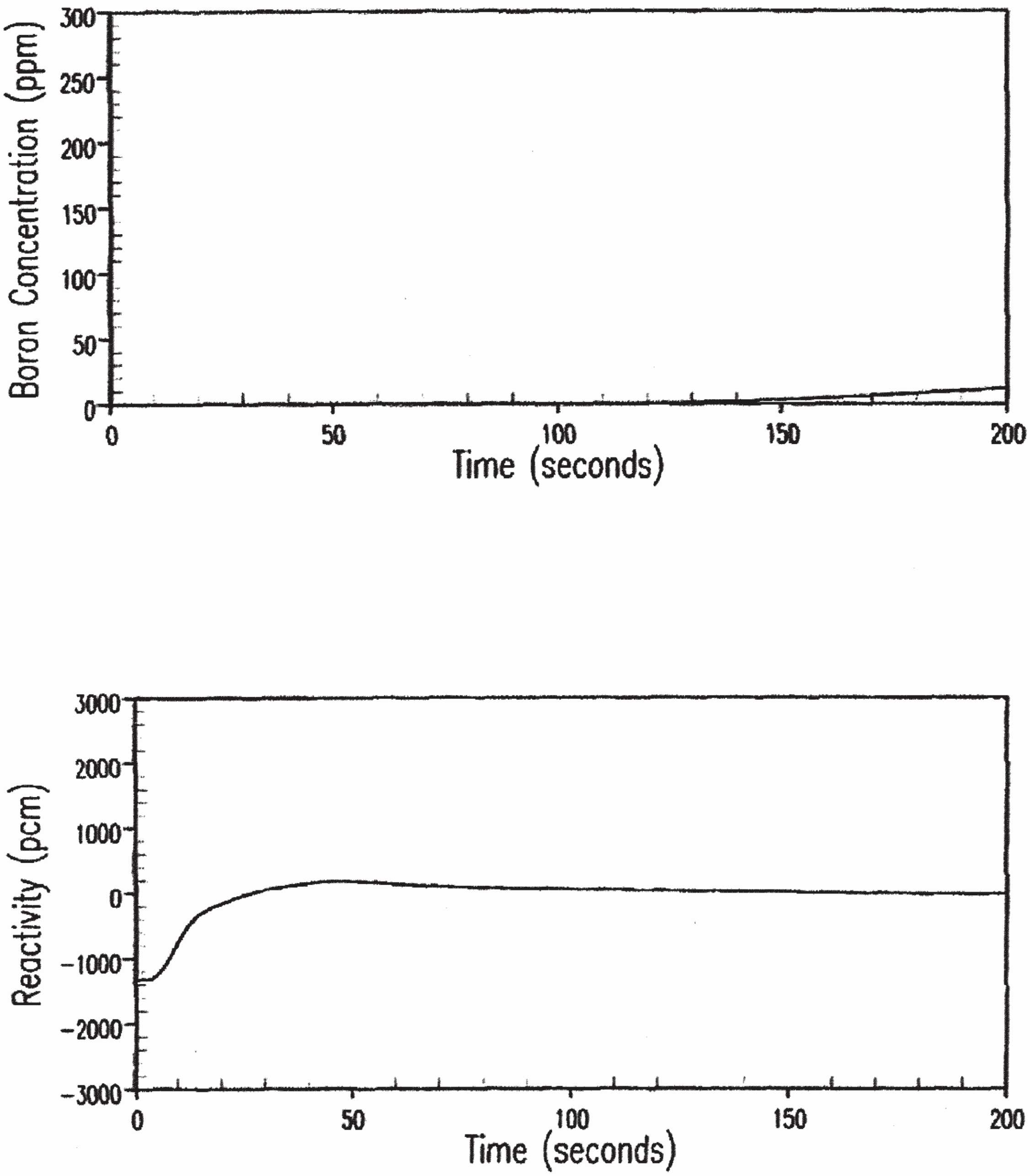

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FIGURE 15.1-20

1.4 FT^2 STEAMLINE BREAK
OFFSITE POWER AVAILABLE
UNIT 2

Figures 15.1-21 through 15.1-26 have been deleted intentionally.

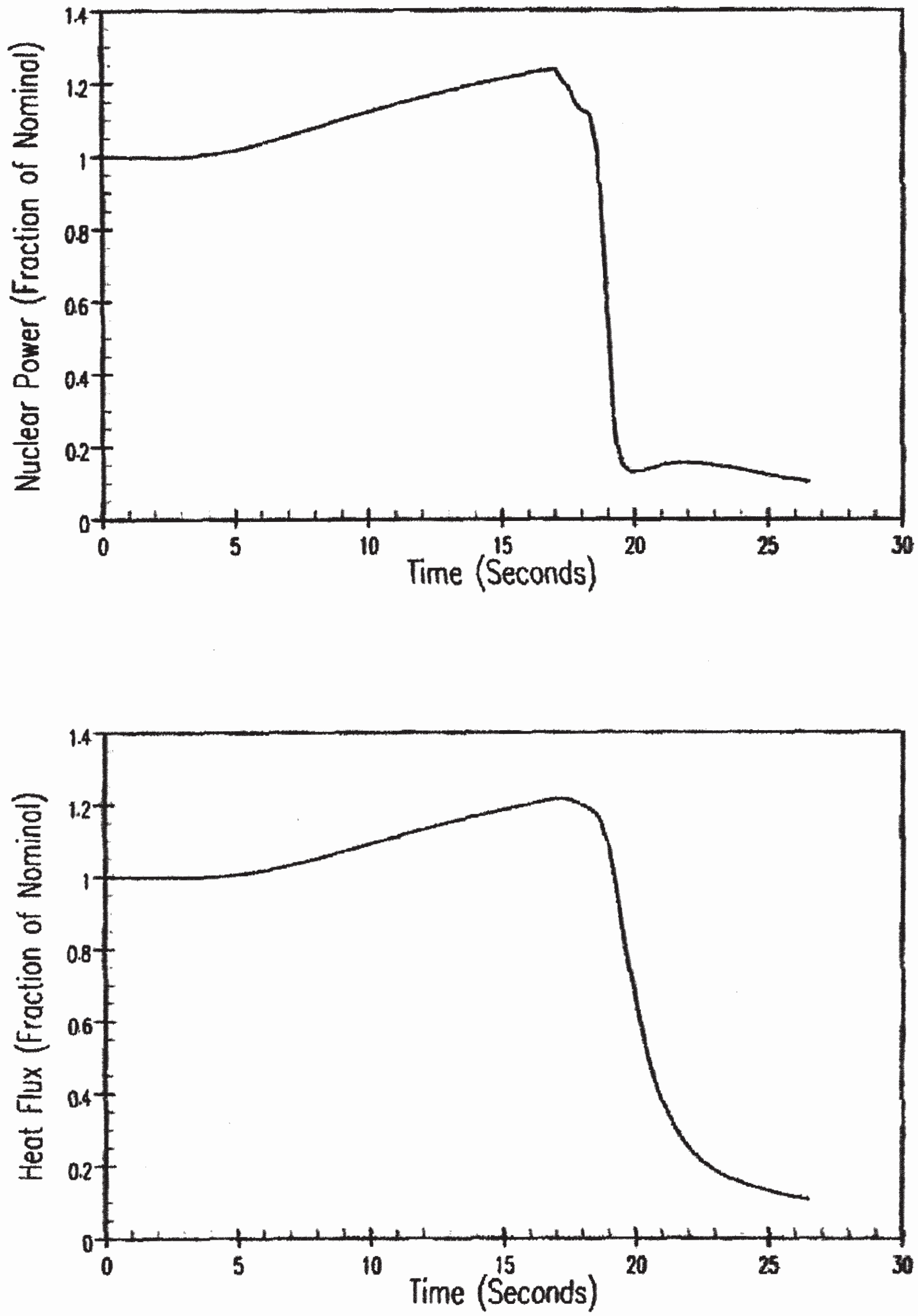

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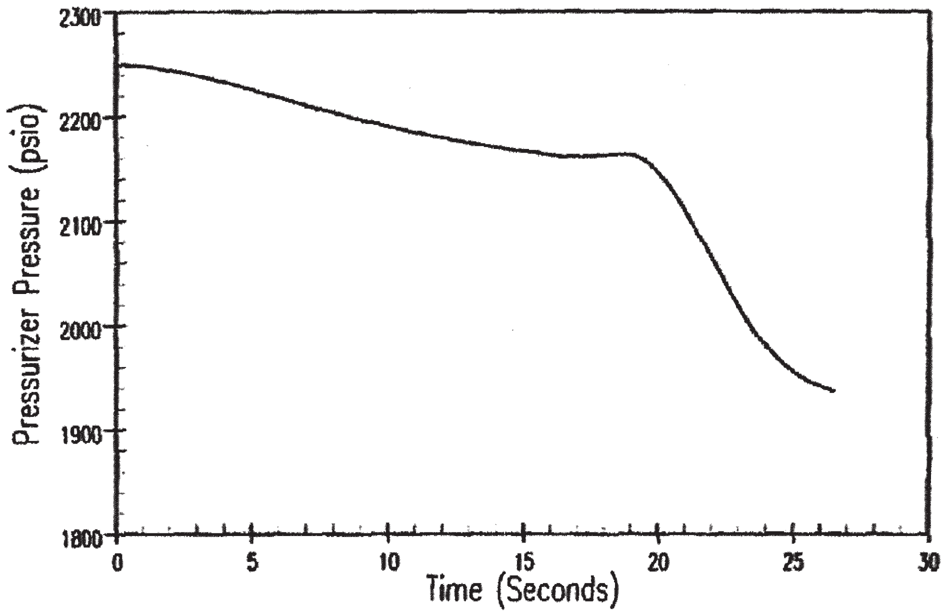
FIGURE 15.1-27

STEAM SYSTEM PIPING FAILURE
AT FULL POWER – 0.95 FT^2 BREAK
UNIT 1



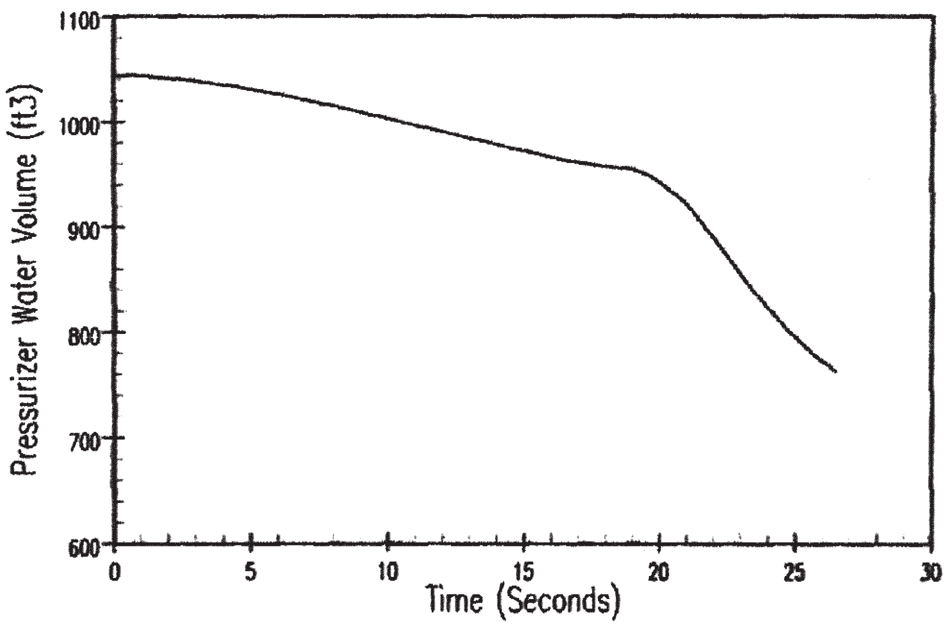


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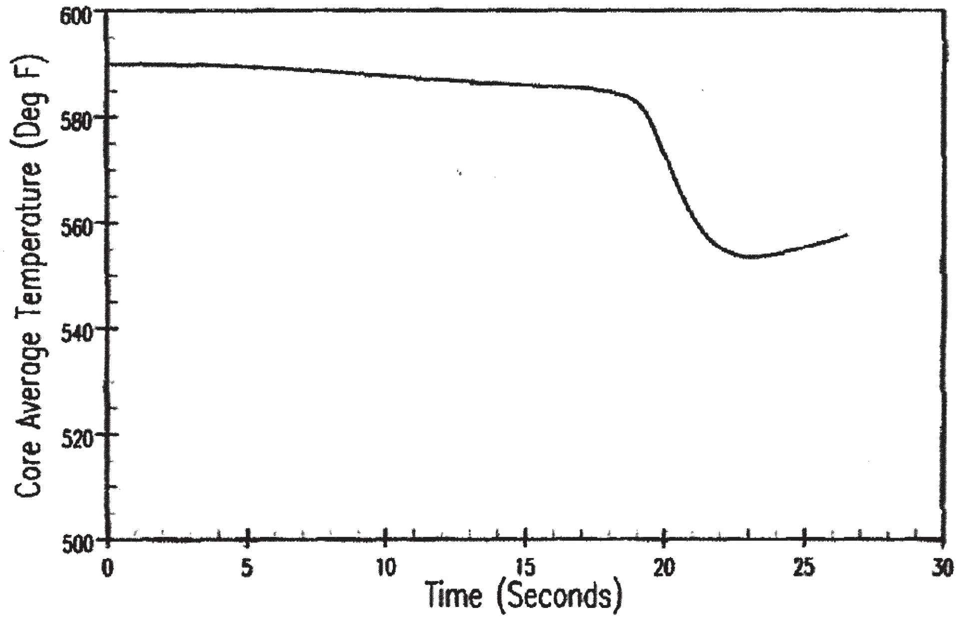
FIGURE 15.1-28

STEAM SYSTEM PIPING FAILURE
AT FULL POWER – 0.95 FT^2 BREAK
UNIT 1



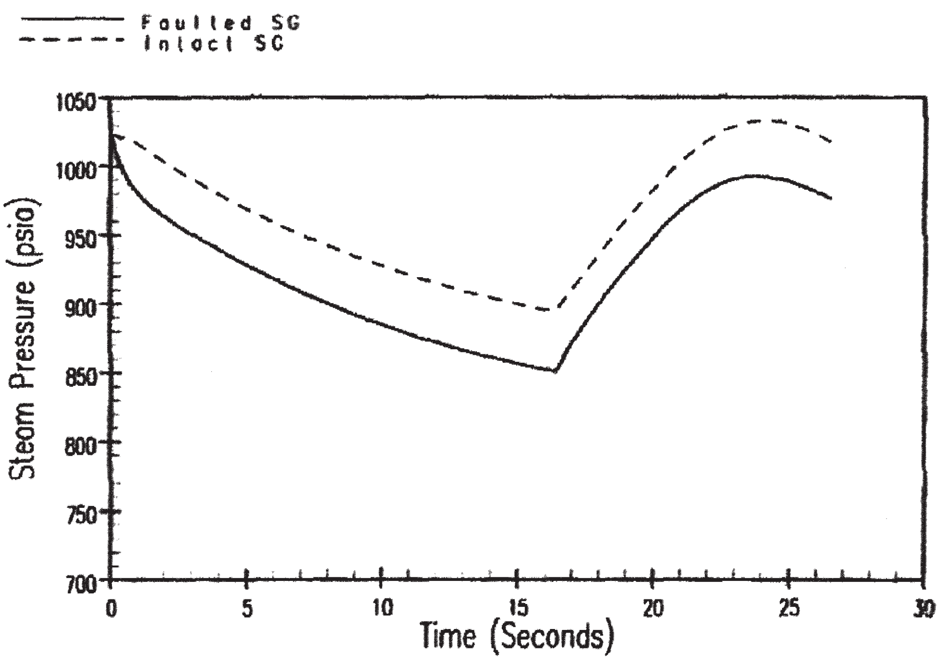


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FIGURE 15.1-29 STEAM SYSTEM PIPING FAILURE AT FULL POWER – 0.95 FT^2 BREAK UNIT 1

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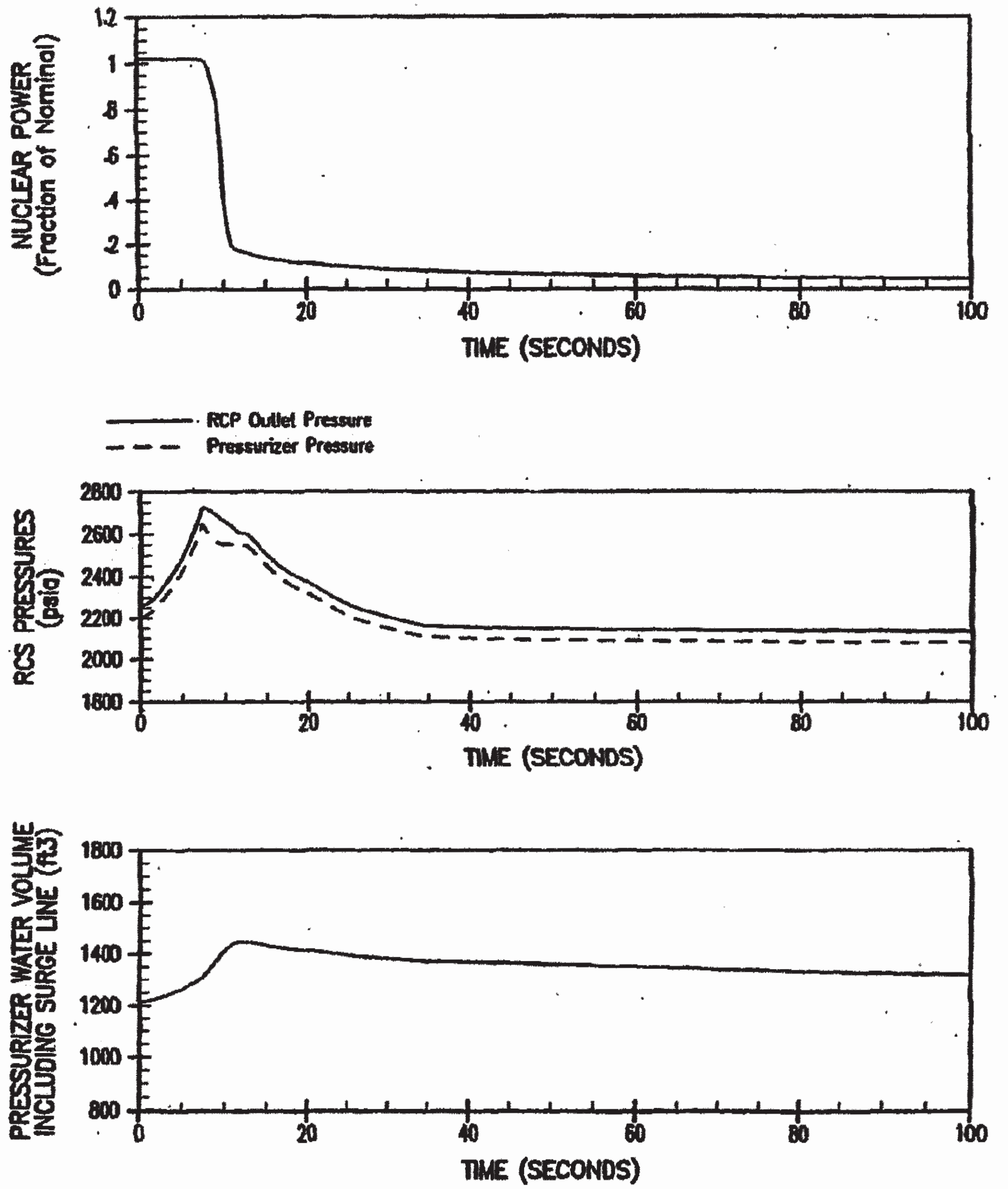

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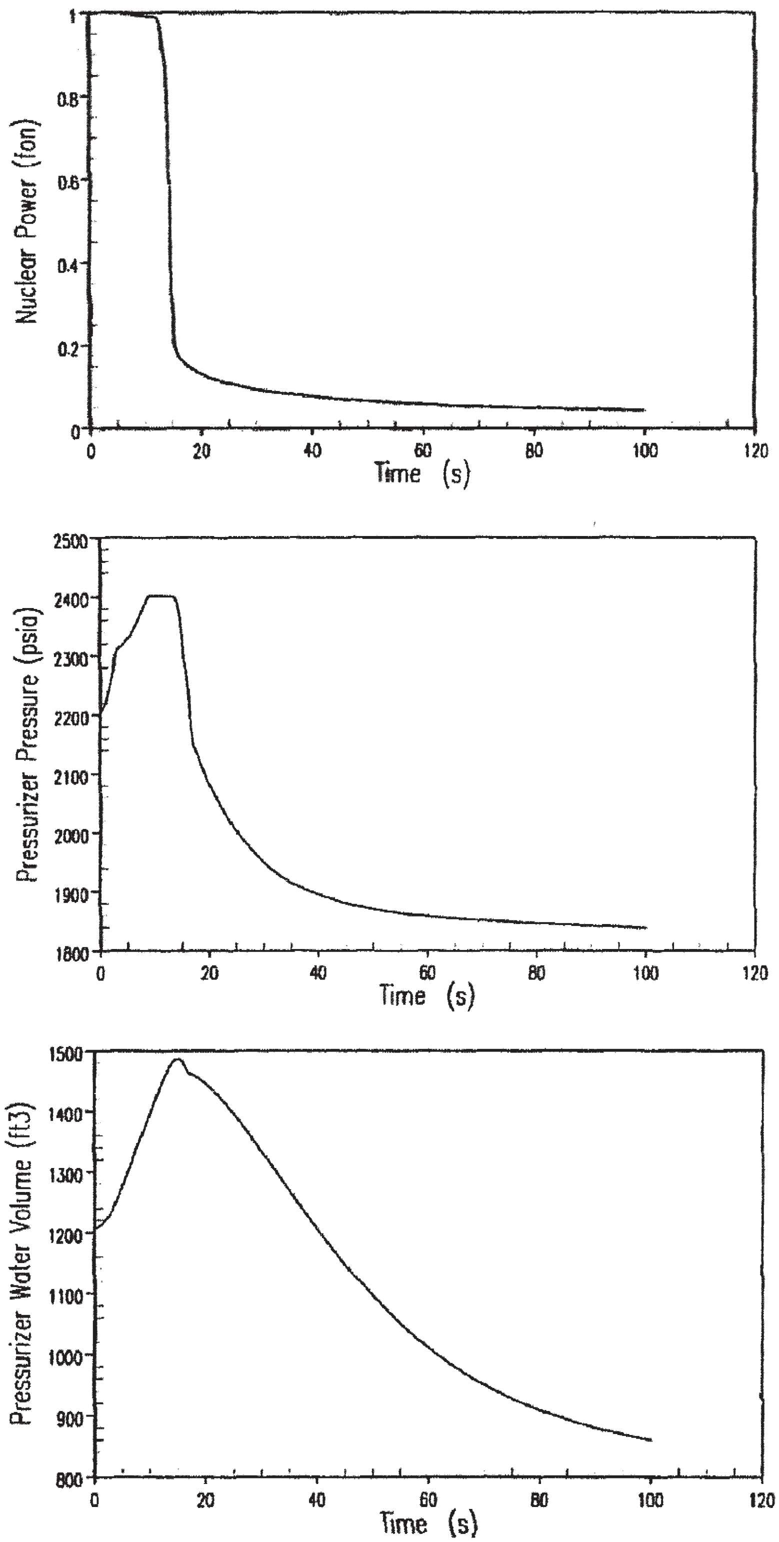
FIGURE 15.2-1 LOSS OF LOAD/TURBINE TRIP UNIT 1 – RCS PRESSURE CASE



BYRON/BRAIDWOOD STATIONS UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE 15.2-1a LOSS OF LOAD/TURBINE TRIP UNIT 1 - MSS PRESSURE CASE

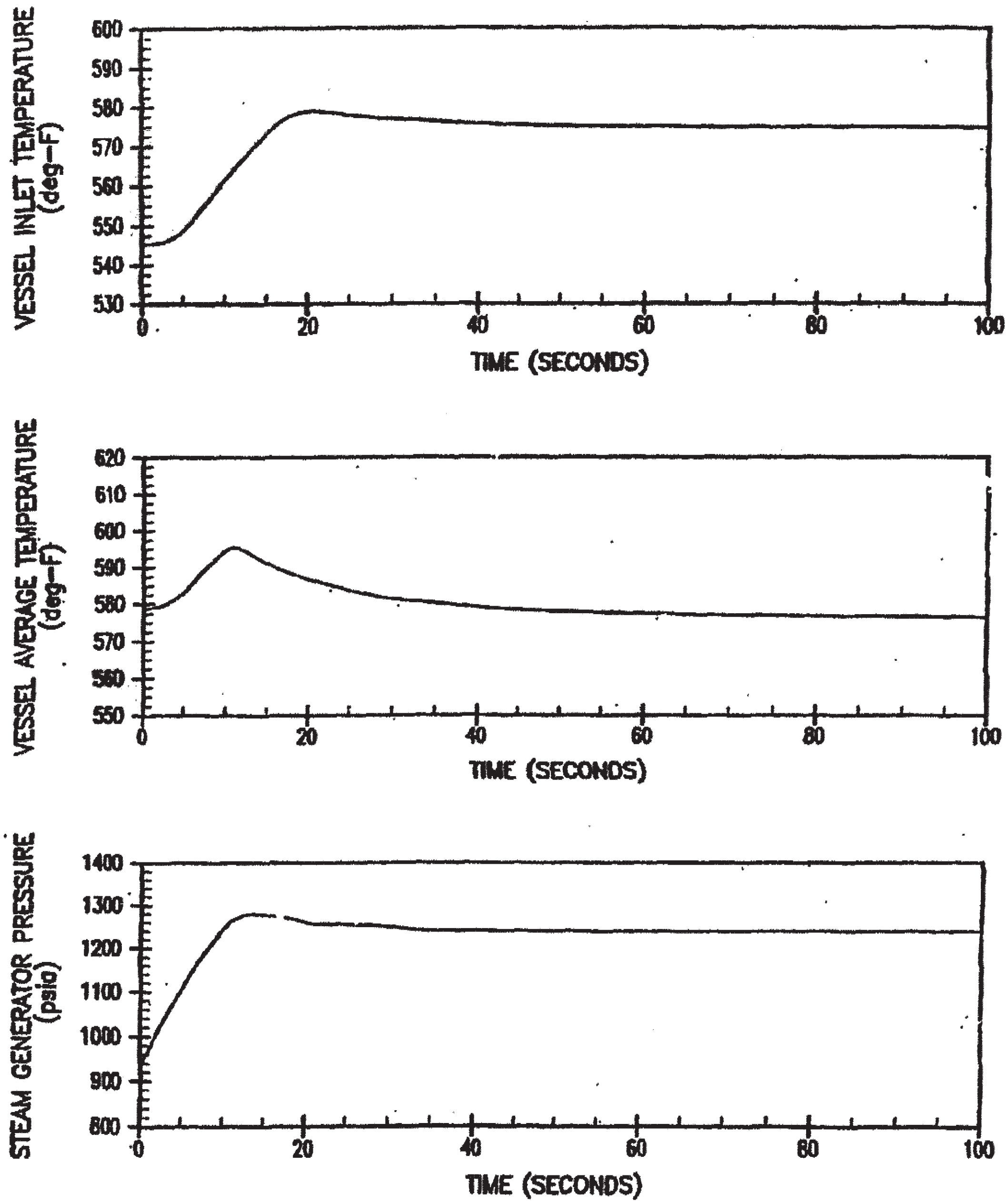


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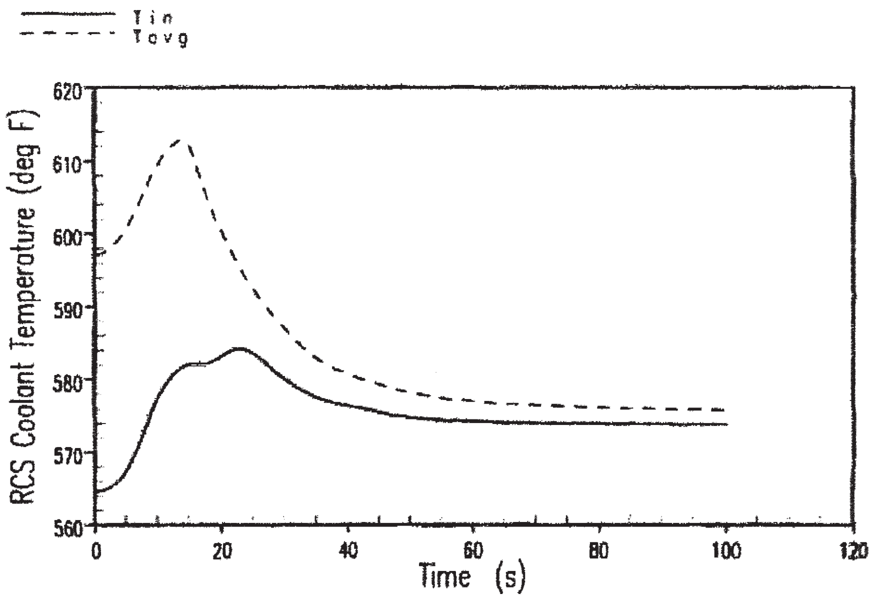
FIGURE 15.2-2

LOSS OF LOAD/TURBINE TRIP
UNIT 1 - RCS PRESSURE CASE



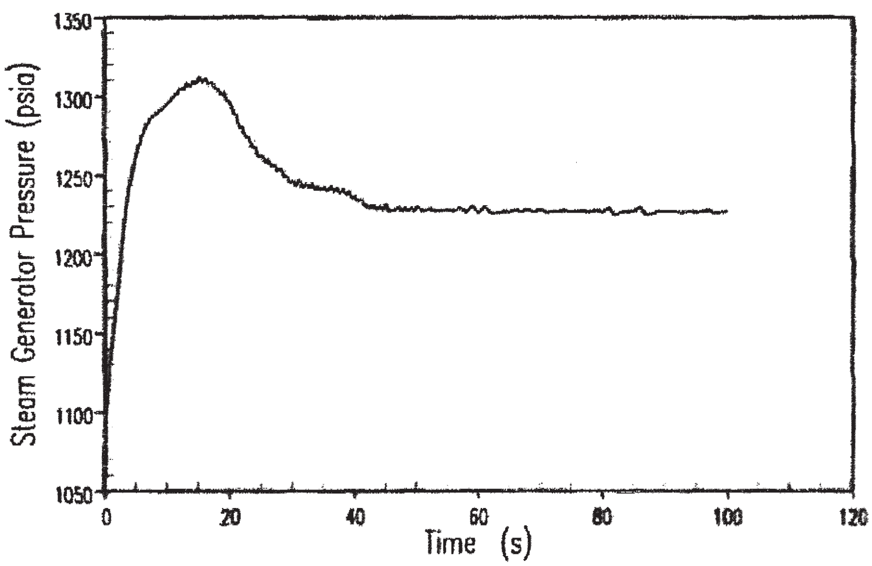


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FIGURE 15.2-2a LOSS OF LOAD/TURBINE TRIP UNIT 1 - MSS PRESSURE CASE

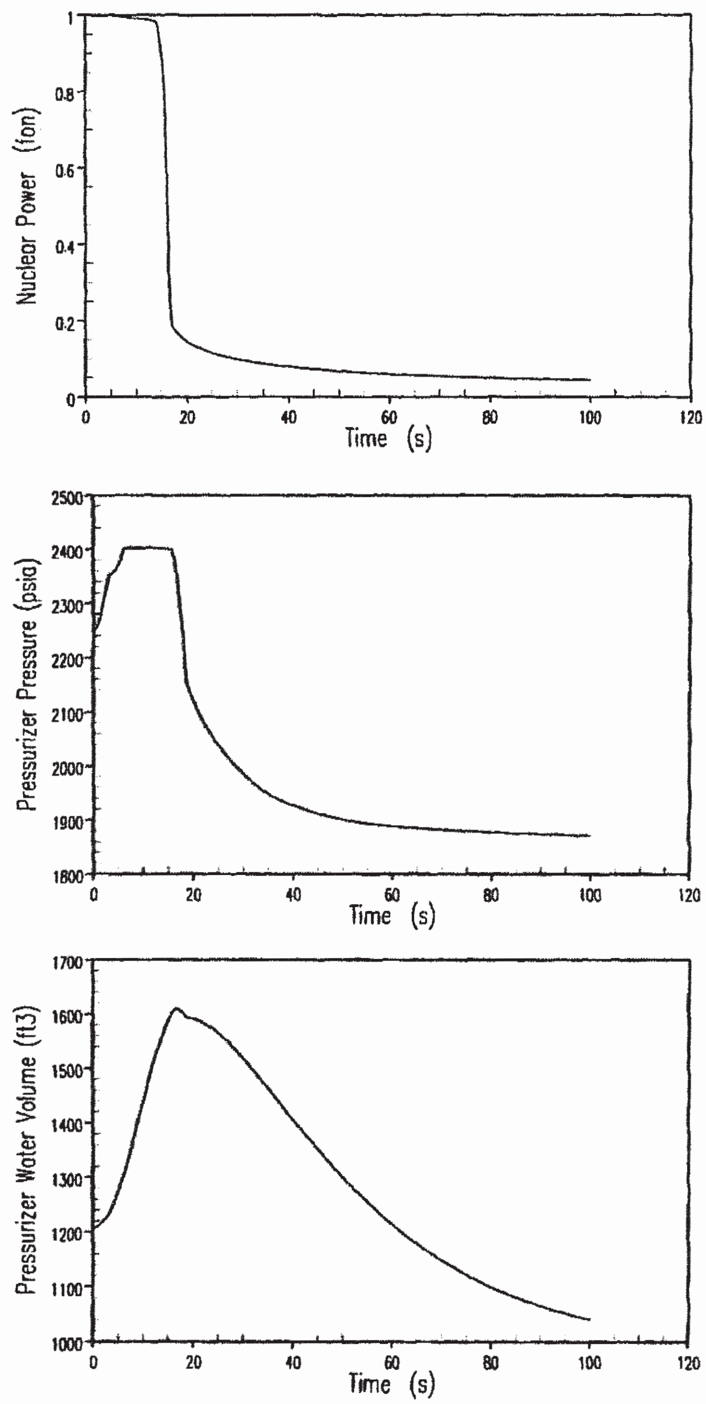

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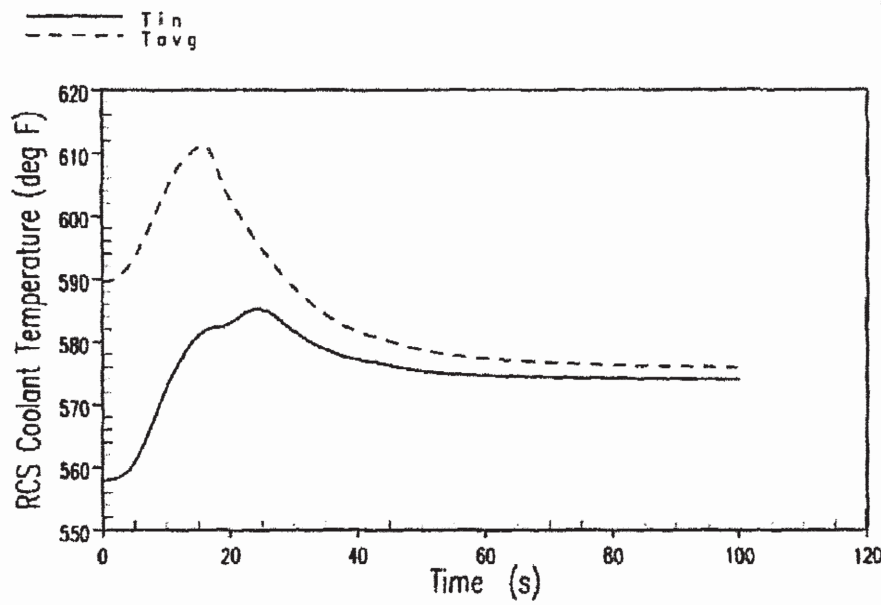
FIGURE 15.2-3

LOSS OF LOAD/TURBINE TRIP
UNIT 1 – DNB CASE



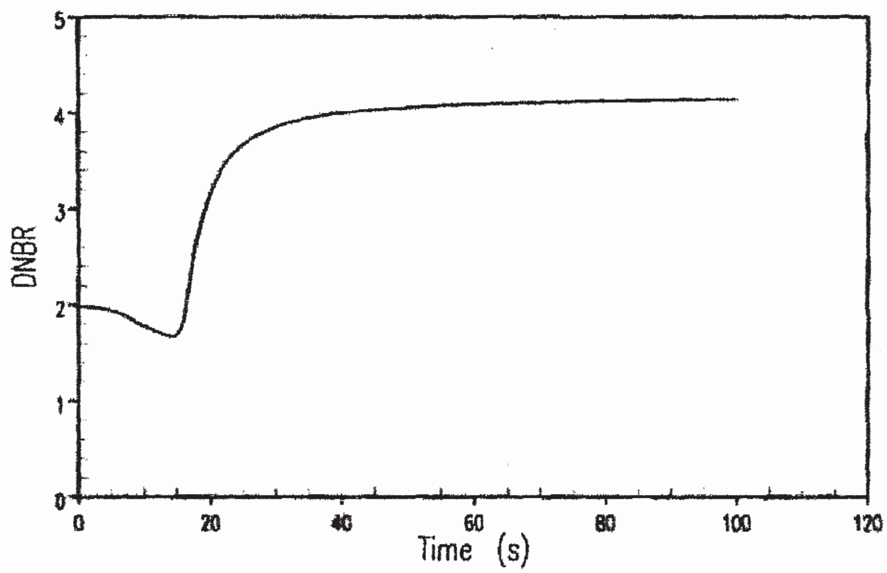


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FIGURE 15.2-4 LOSS OF LOAD/TURBINE TRIP UNIT 1 – DNB CASE

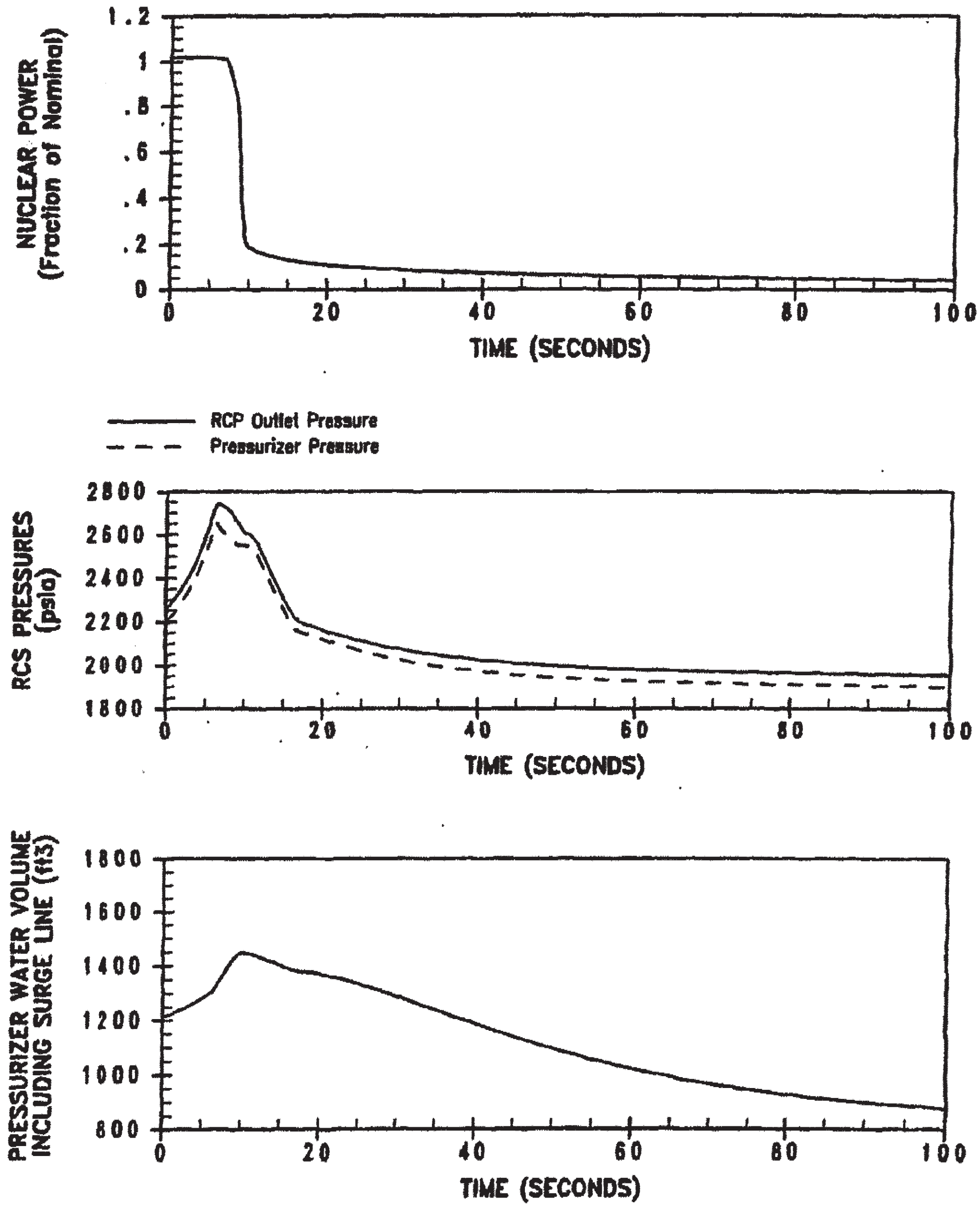

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FIGURE 15.2-5

LOSS OF LOAD/TURBINE TRIP
RCS PRESSURE CASE
UNIT 2

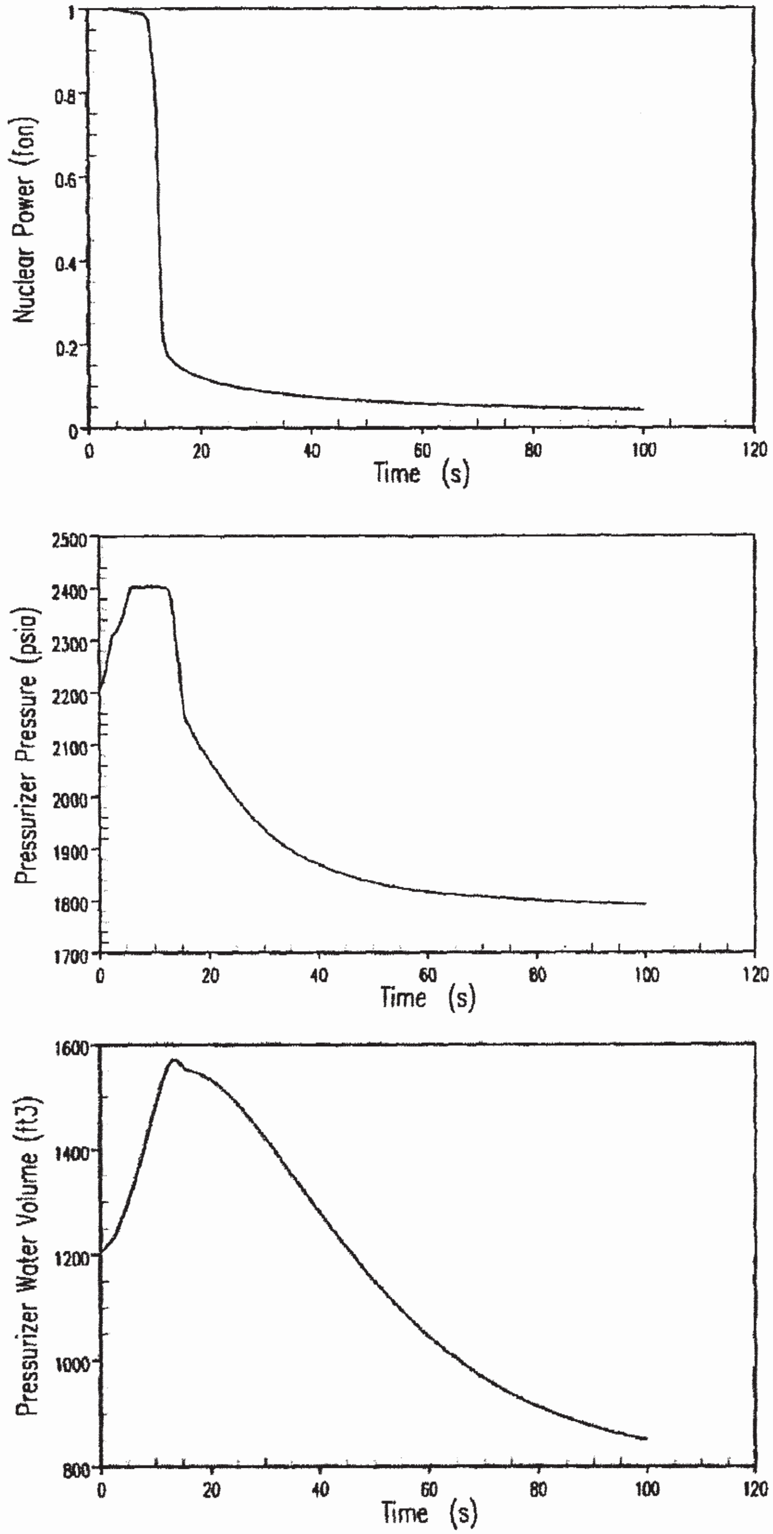

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FIGURE 15.2-5a

LOSS OF LOAD/TURBINE TRIP
UNIT 2 - MSS PRESSURE CASE

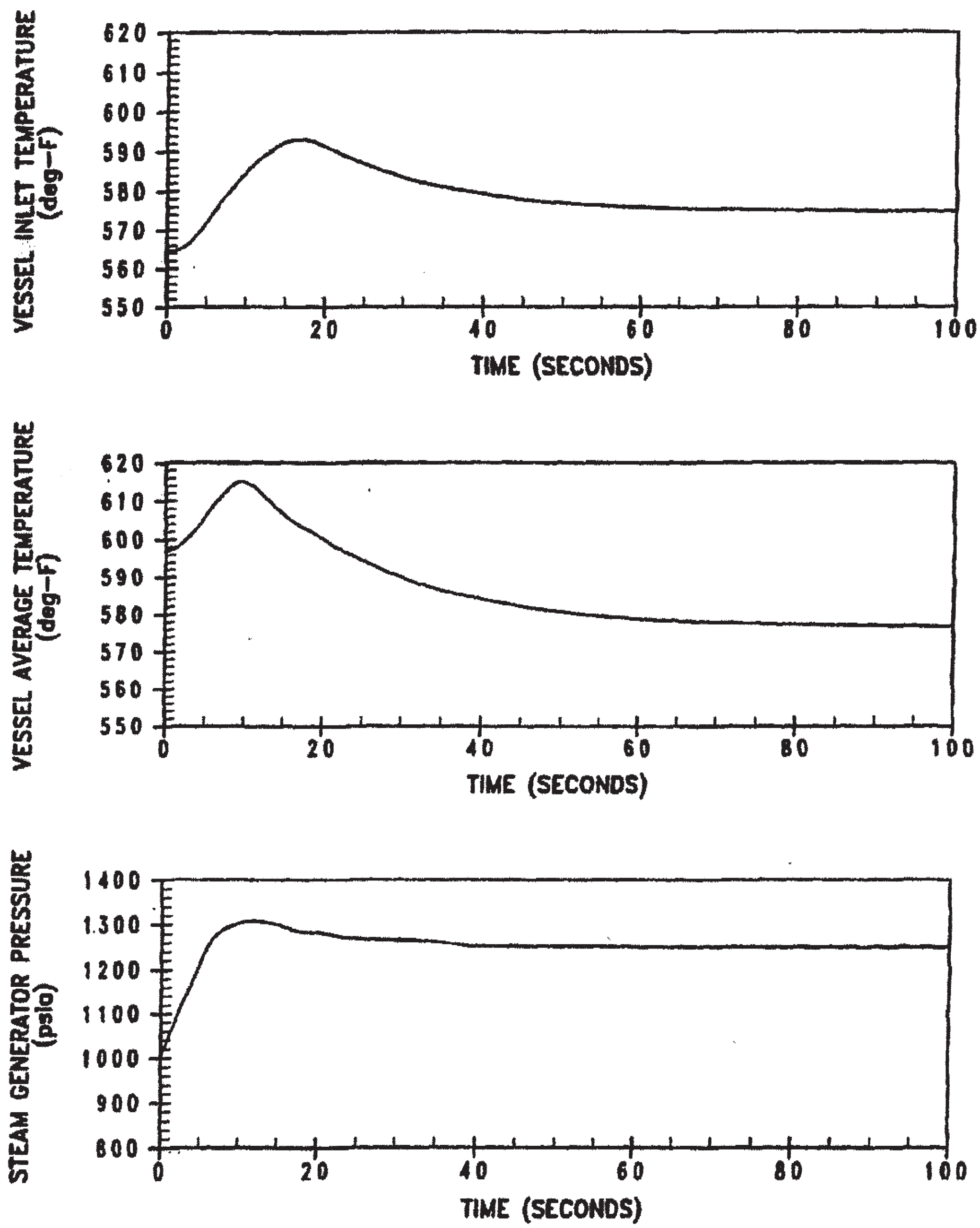

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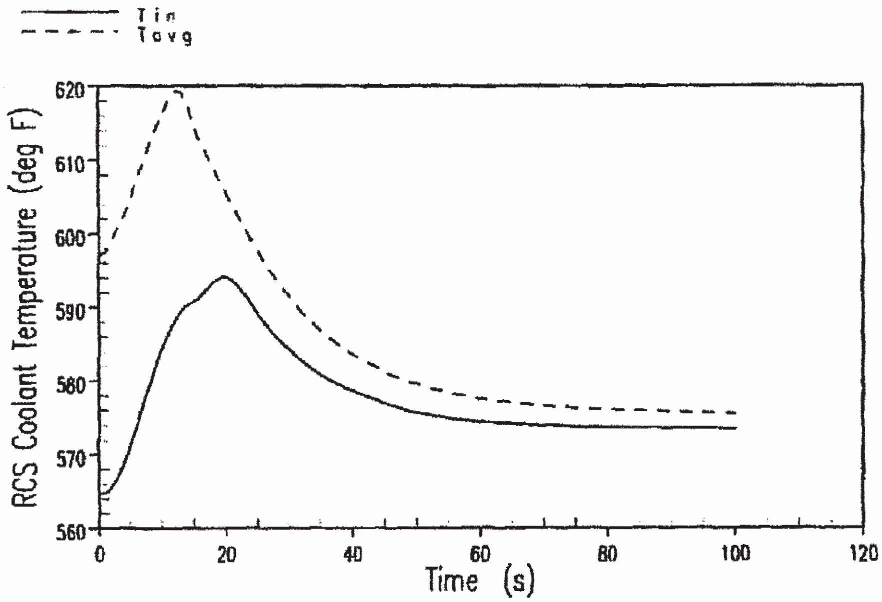
FIGURE 15.2-6

LOSS OF LOAD/TURBINE TRIP
RCS PRESSURE CASE
UNIT 2



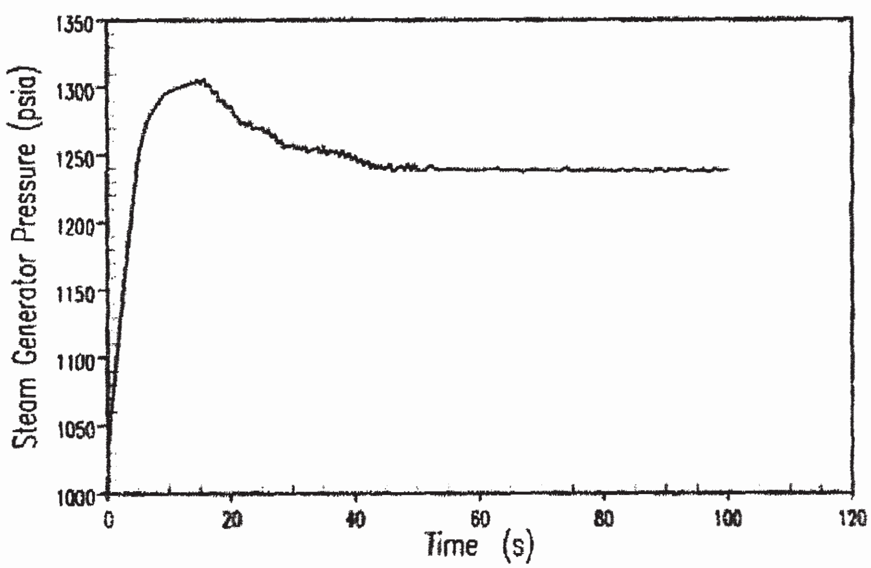


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FIGURE 15.2-6a

LOSS OF LOAD/TURBINE TRIP
UNIT 2 - MSS PRESSURE CASE

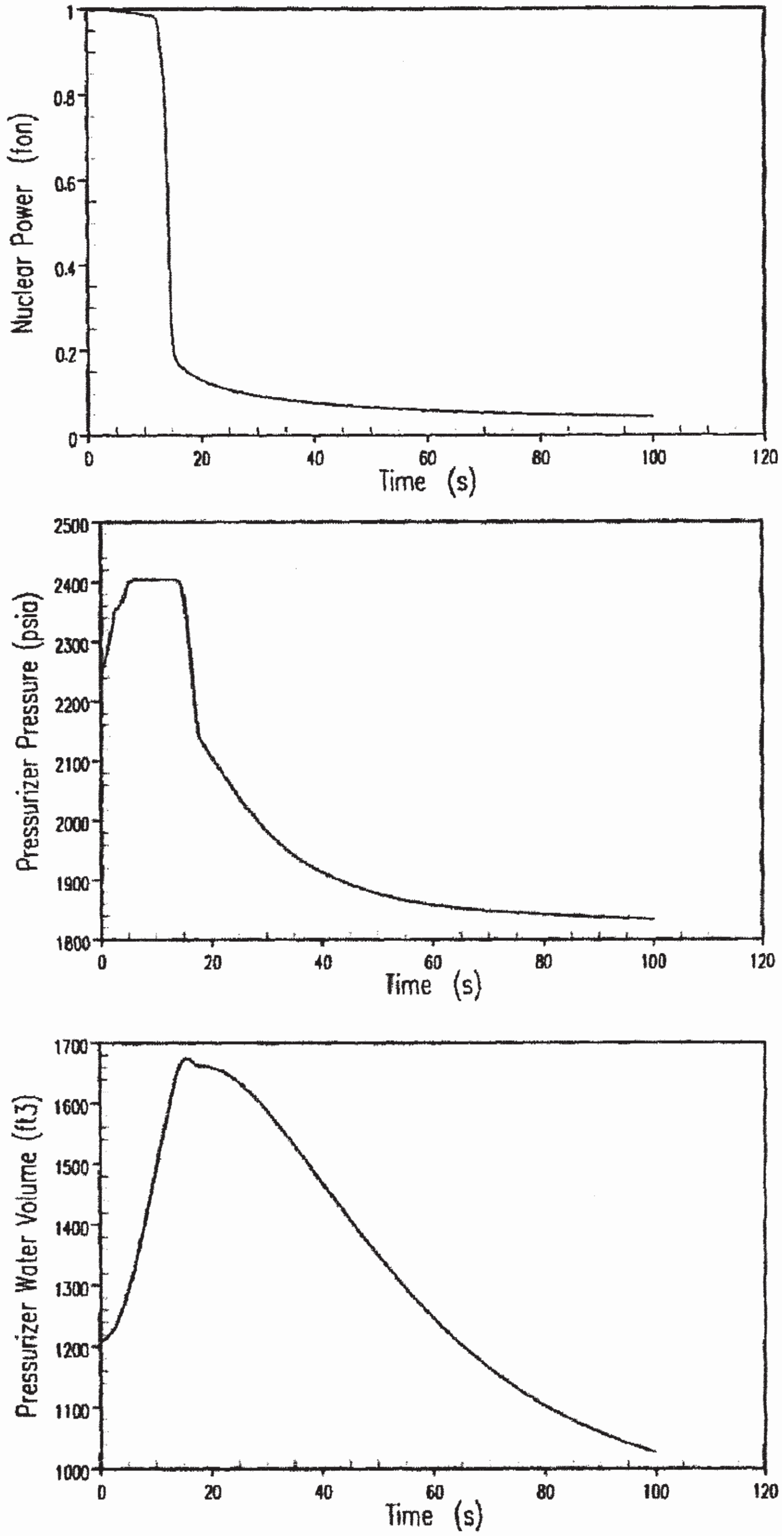

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FIGURE 15.2-7

LOSS OF LOAD/TURBINE TRIP
UNIT 2 - DNB CASE

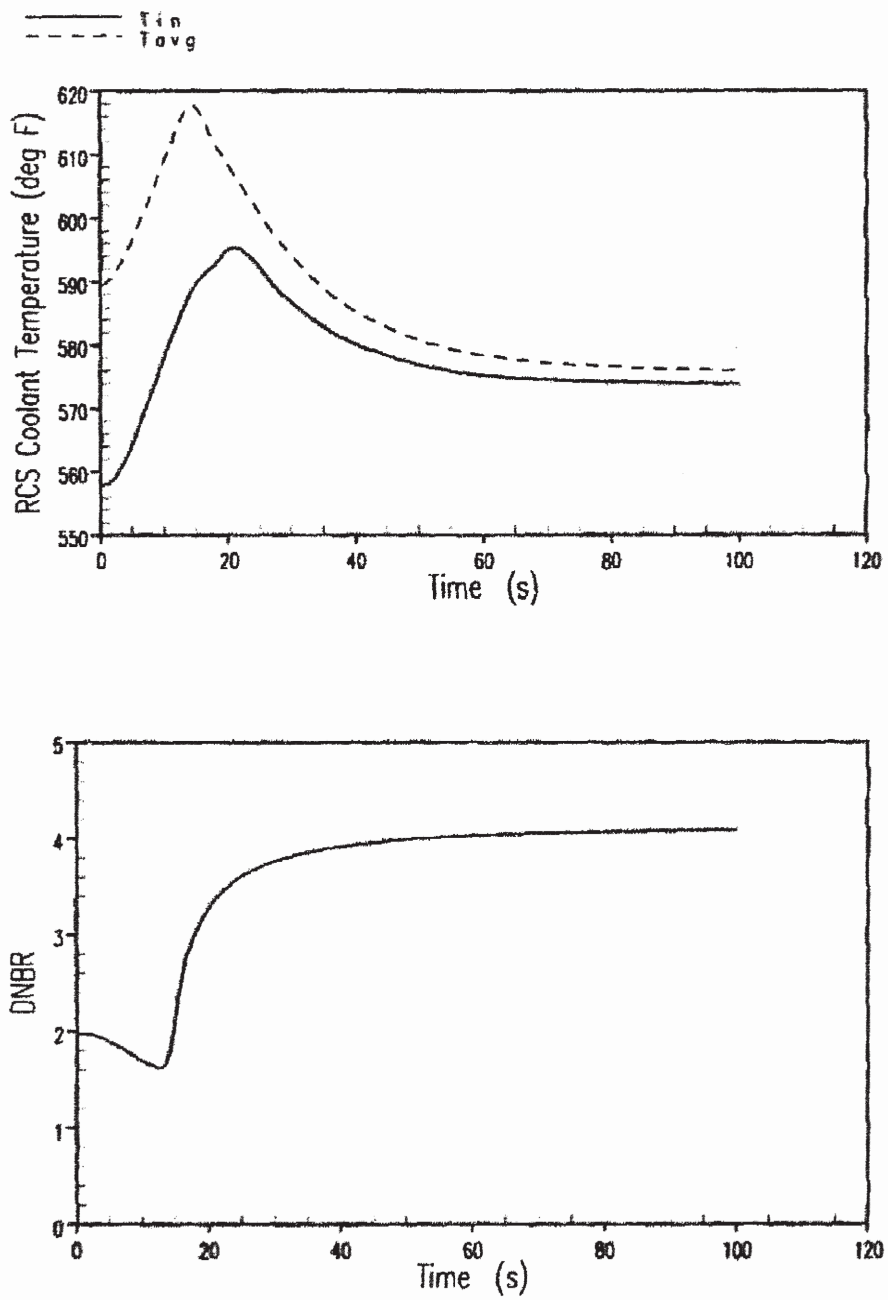

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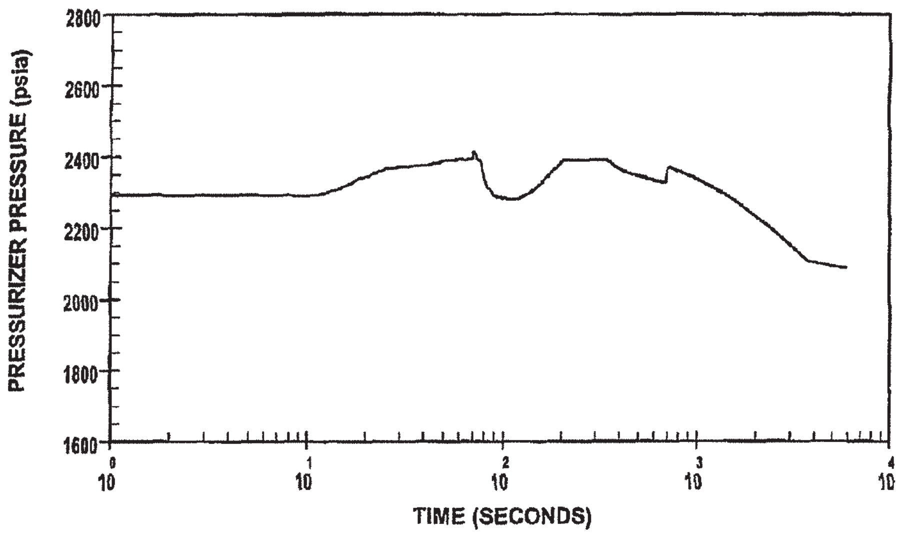
FIGURE 15.2-8

LOSS OF LOAD/TURBINE TRIP
UNIT 2 – DNB CASE

Figure 15.2-9 has been deleted intentionally.



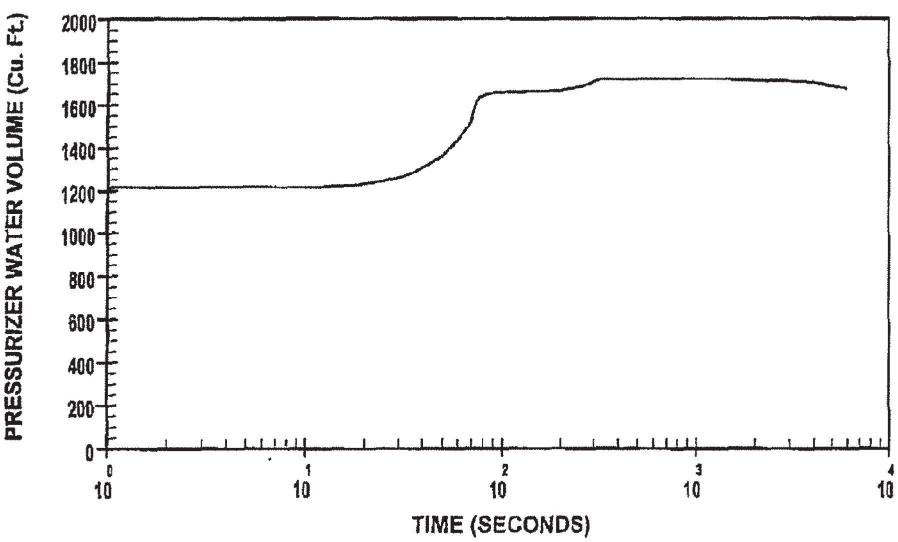


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FIGURE 15.2–9a

PRESSURIZER PRESSURE AND WATER VOLUME TRANSIENTS FOR LOSS OF OFFSITE POWER

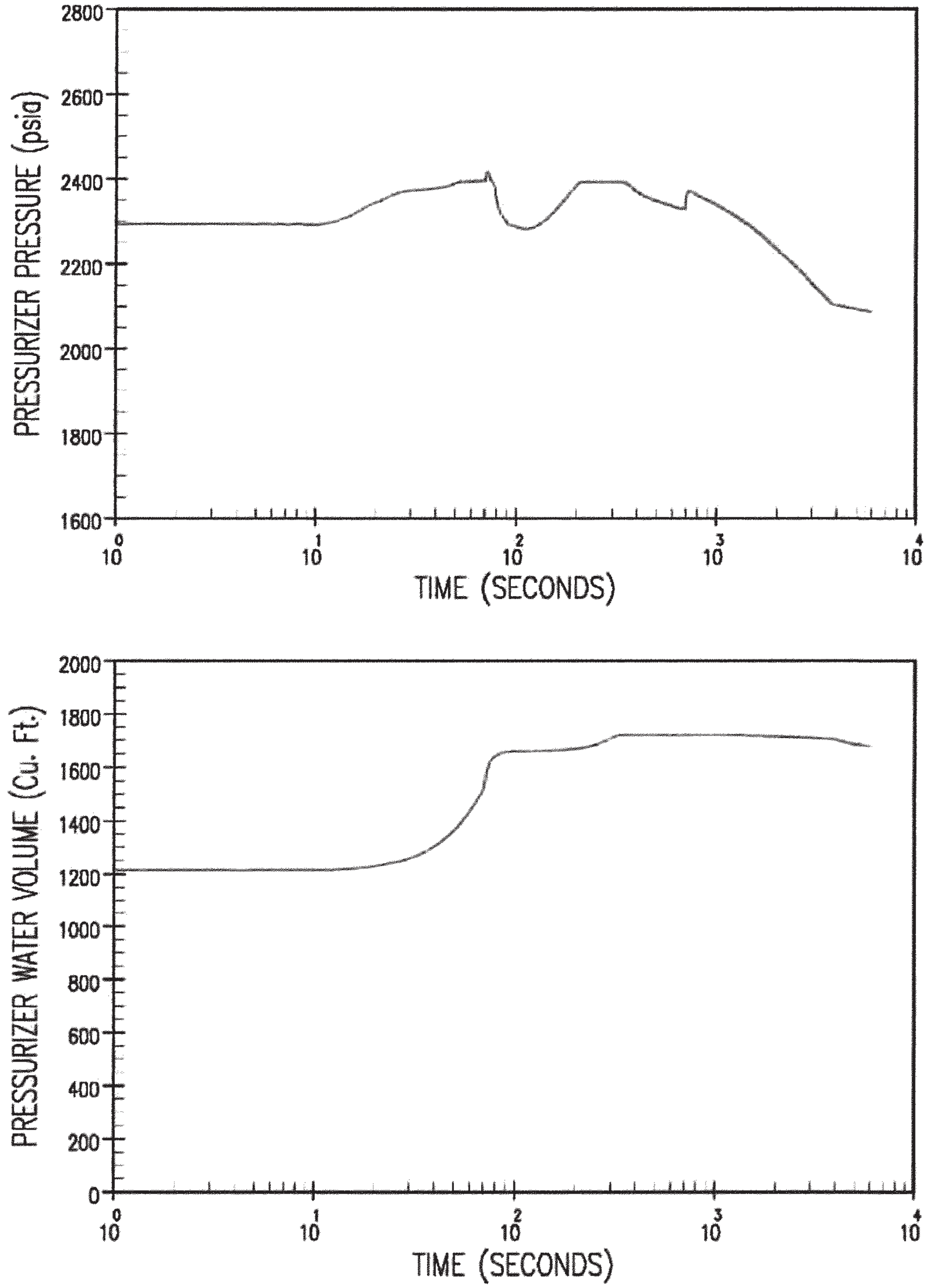

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FIGURE 15.2-9a

PRESSURIZER PRESSURE AND WATER VOLUME TRANSIENTS FOR LOSS OF OFFSITE POWER

Figure 15.2-10 has been deleted intentionally.

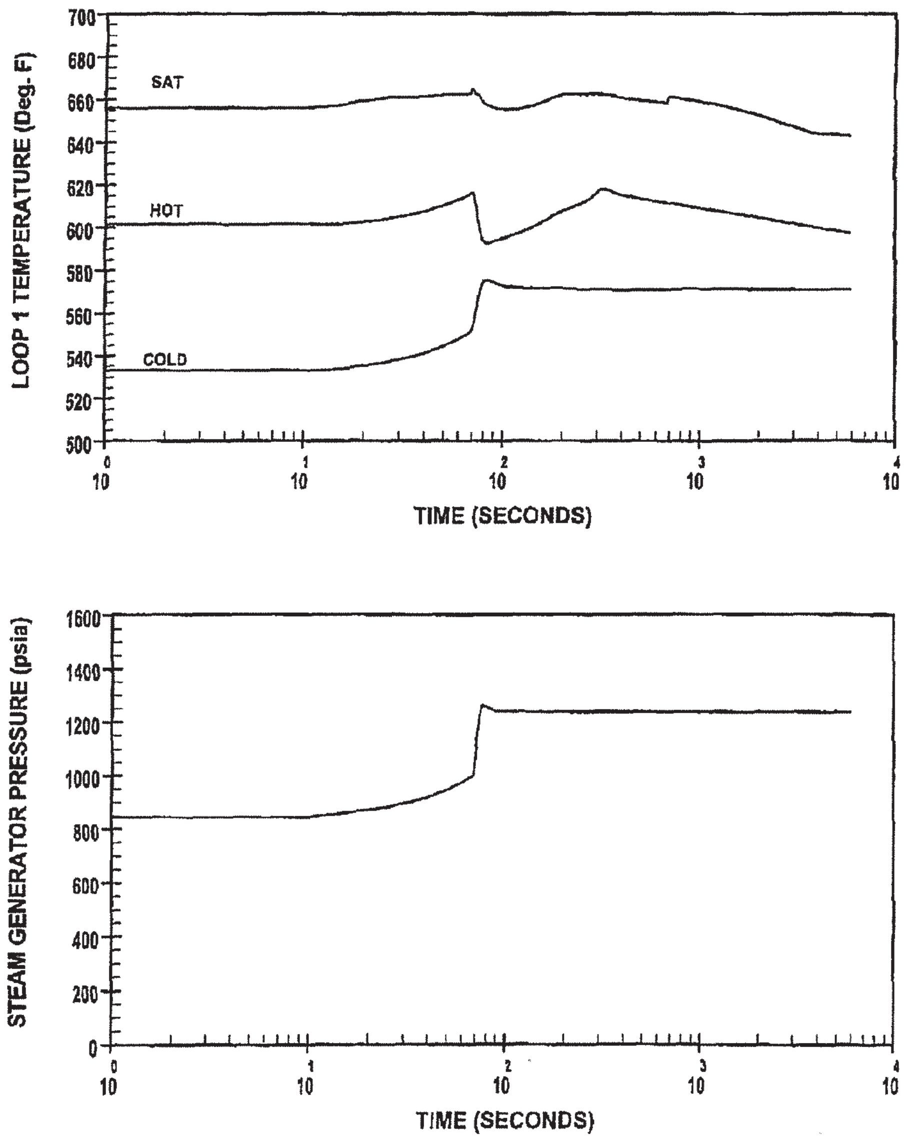

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FIGURE 15.2-10a

LOOP TEMPERATURES AND STEAM GENERATOR PRESSURE FOR LOSS OF OFFSITE POWER

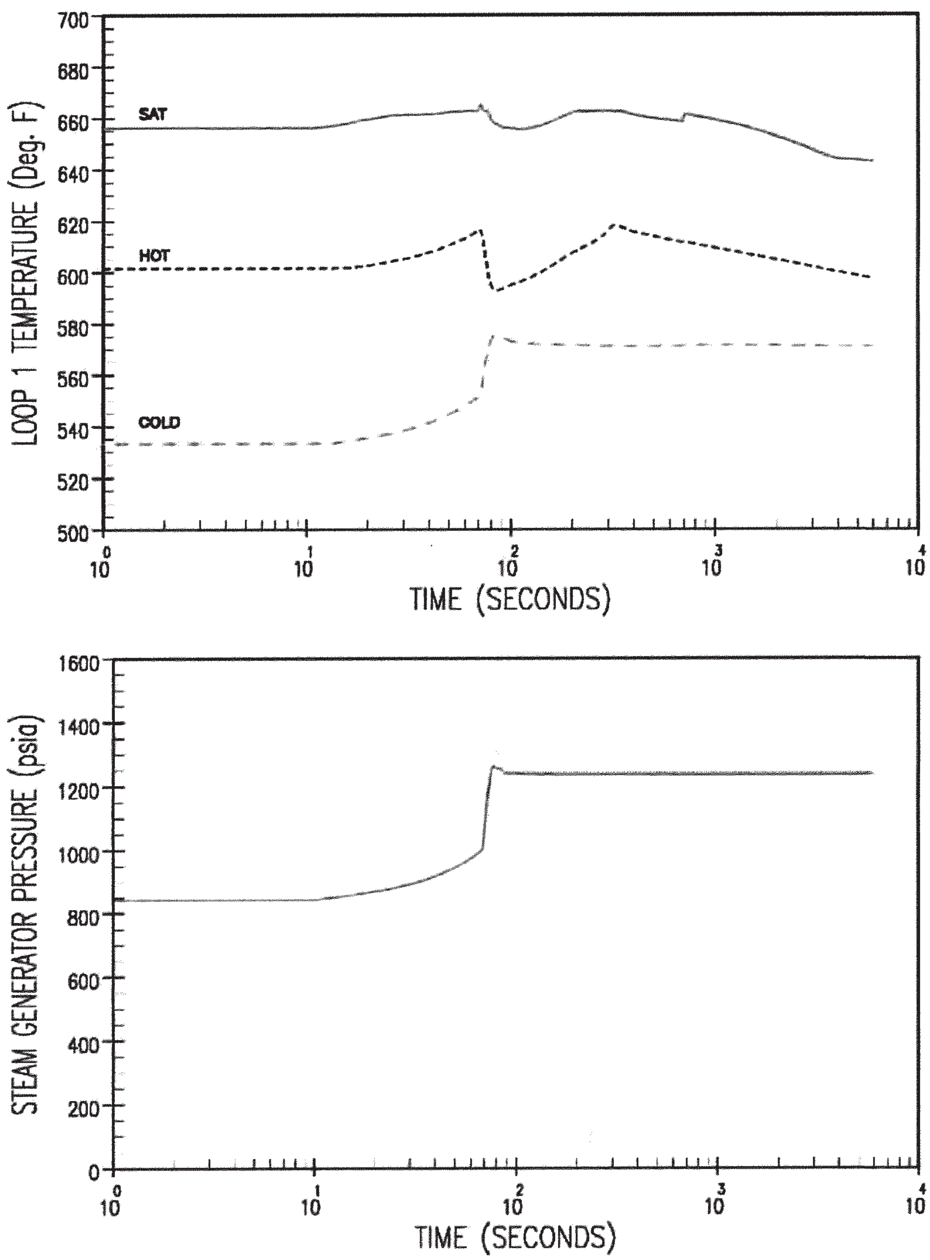

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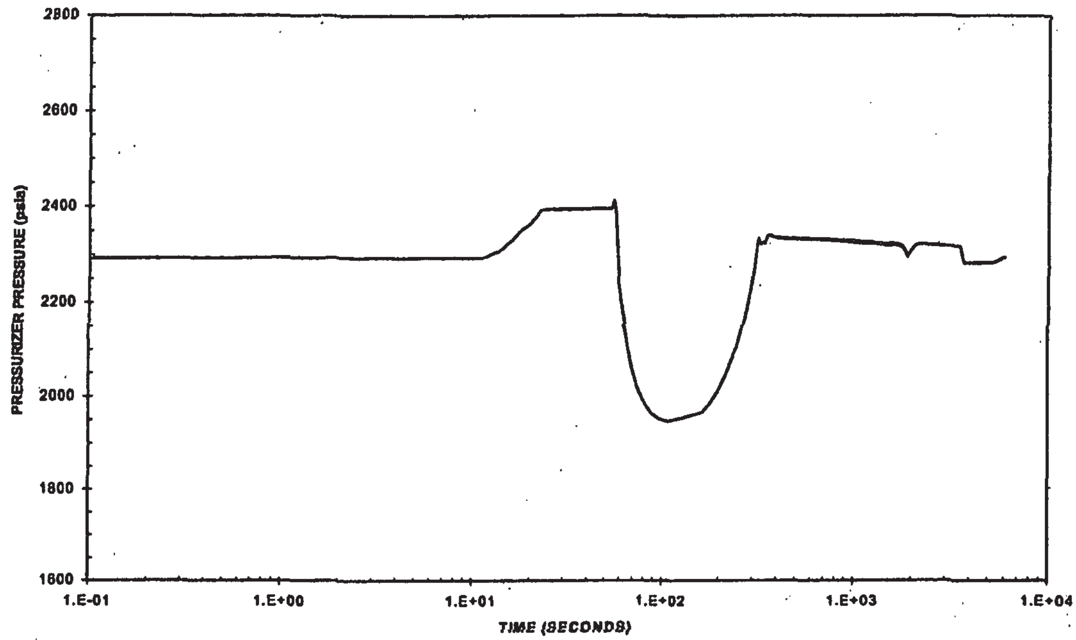
FIGURE 15.2-10a

LOOP TEMPERATURES AND STEAM GENERATOR PRESSURE FOR LOSS OF OFFSITE POWER



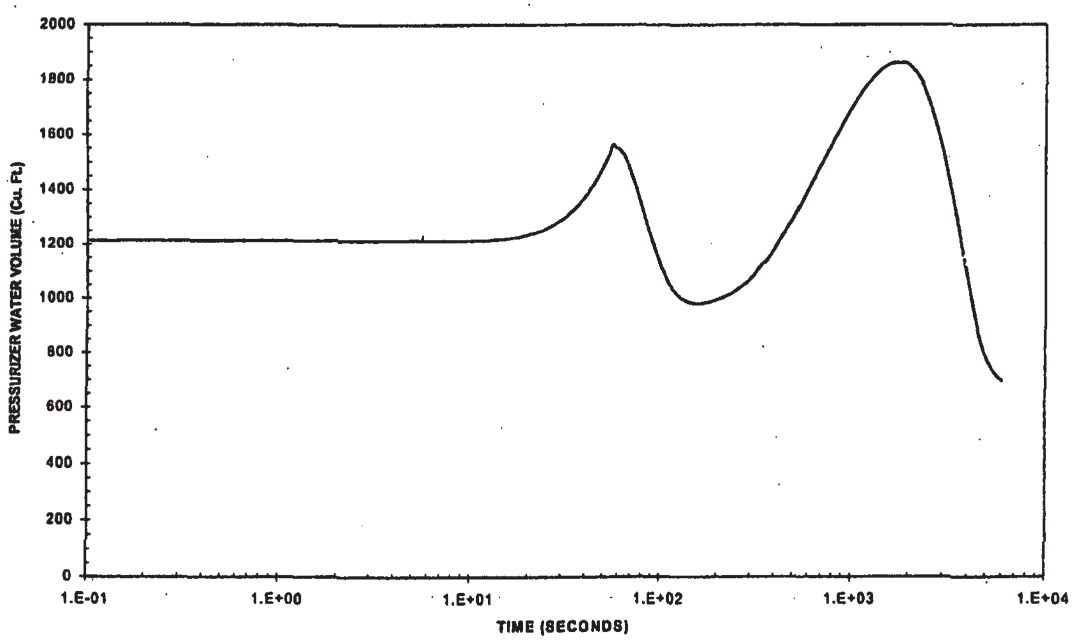


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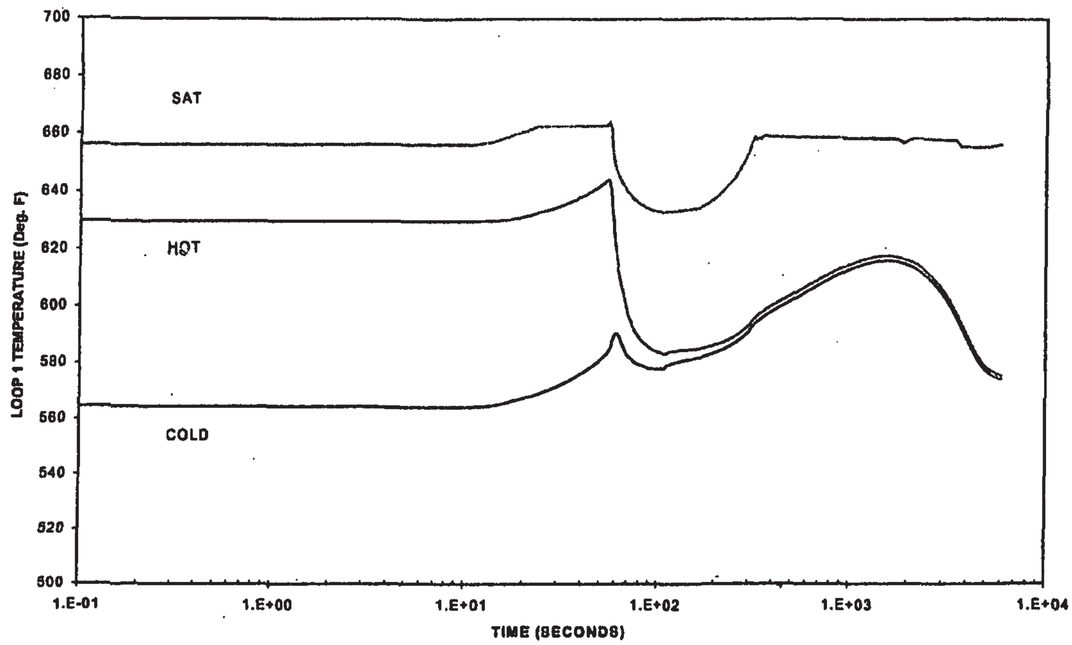
FIGURE 15.2-11

PRESSURIZER PRESSURE AND WATER VOLUME TRANSIENTS FOR LOSS OF NORMAL FEEDWATER



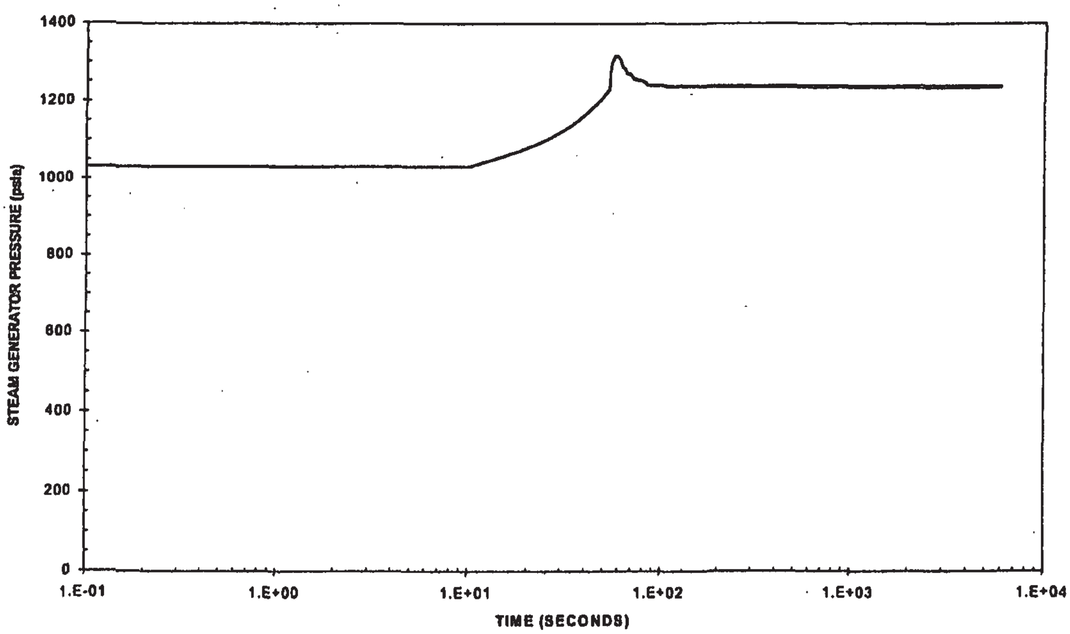


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FIGURE 15.2-12 LOOP TEMPERATURES AND STEAM GENERATOR PRESSURE FOR LOSS OF NORMAL FEEDWATER

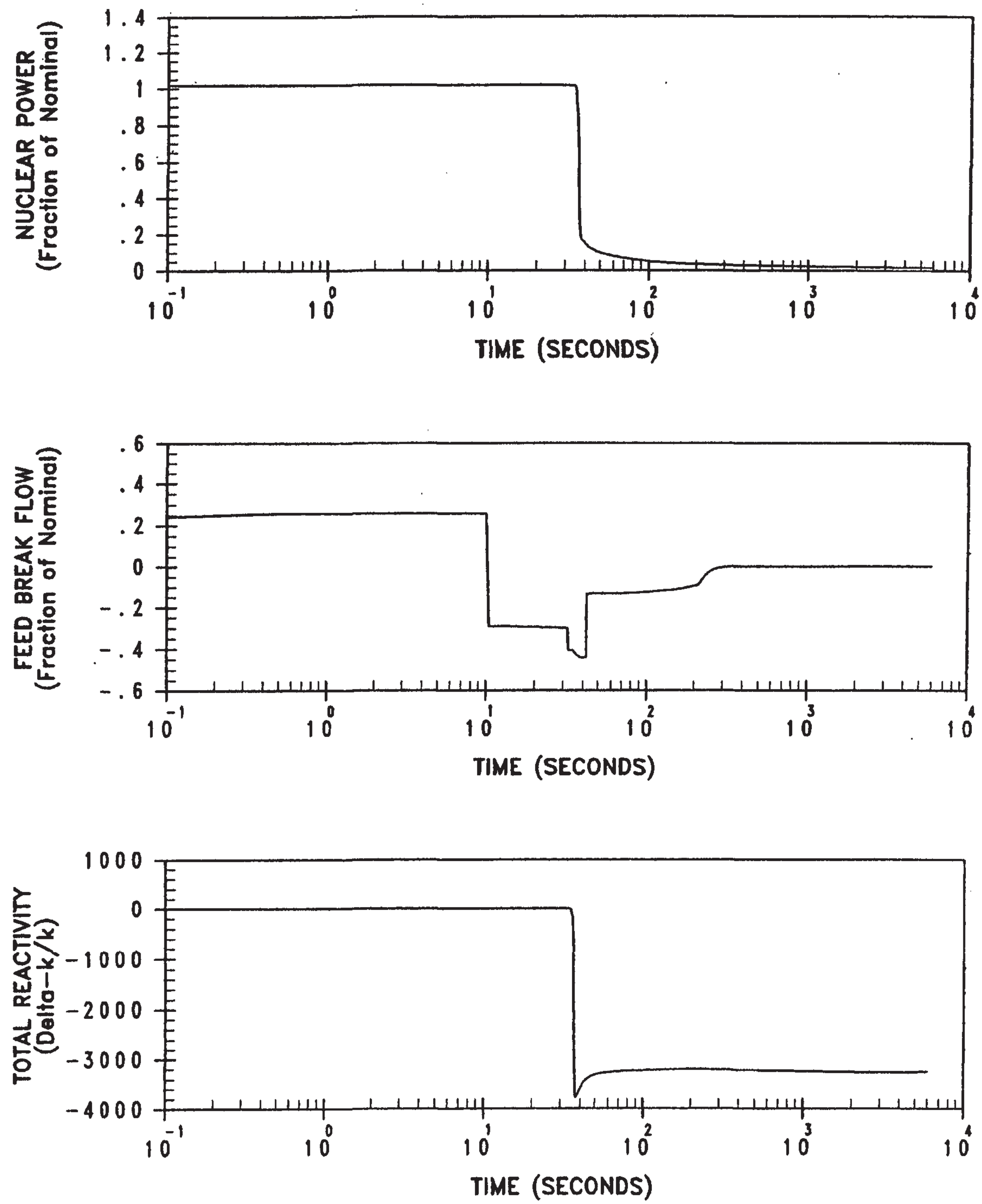

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FIGURE 15.2-13

NUCLEAR POWER, TOTAL CORE REACTIVITY AND FEEDLINE BREAK FLOW TRANSIENTS FOR MAIN FEEDLINE BREAK WITH OFFSITE POWER AVAILABLE

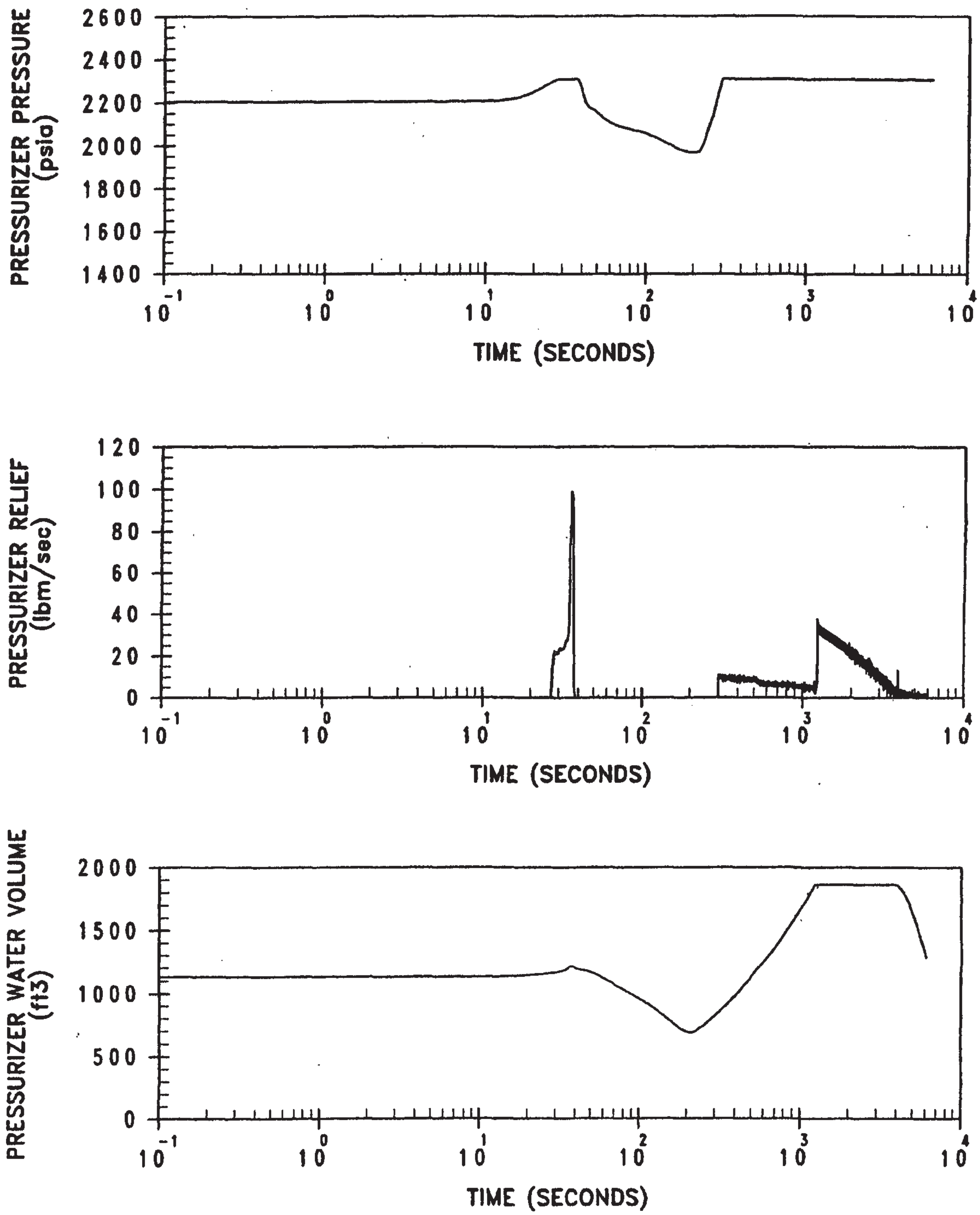

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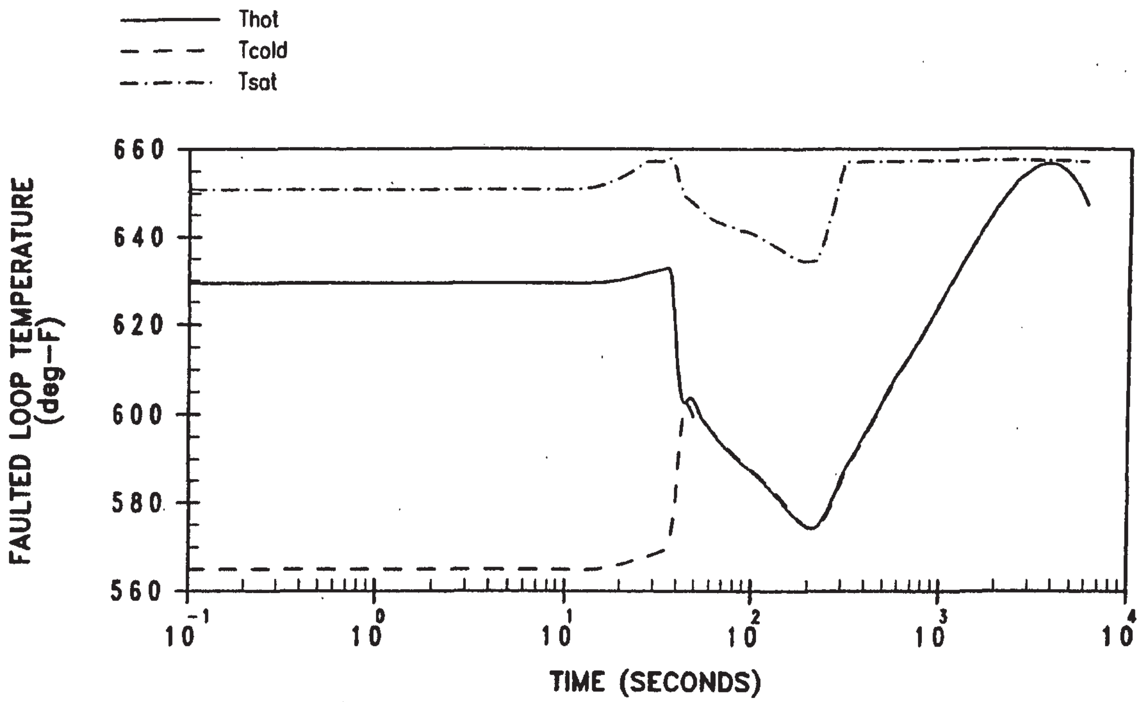
FIGURE 15.2-14

PRESSURIZER PRESSURE, PRESSURIZER RELIEF AND PRESSURIZER WATER VOLUME TRANSIENTS FOR MAIN FEEDLINE BREAK WITH OFFSITE POWER AVAILABLE



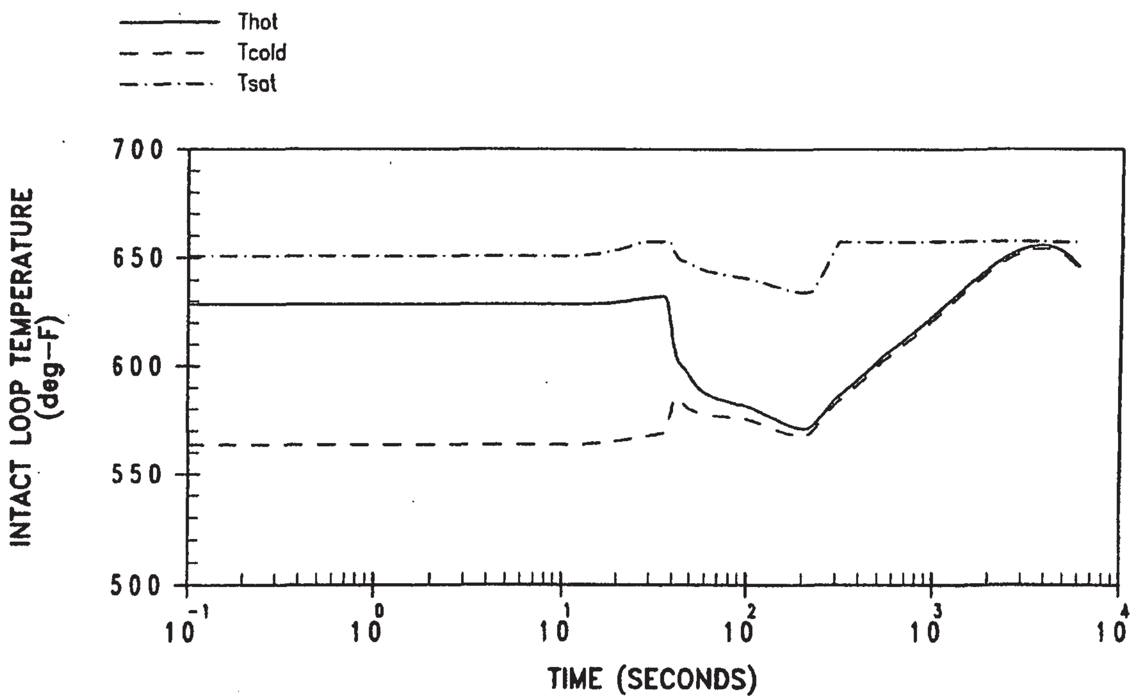


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FIGURE 15.2-15

FAULTED AND INTACT LOOP
COOLANT TEMPERATURE TRANSIENTS
FOR MAIN FEEDLINE BREAK WITH
OFFSITE POWER AVAILABLE

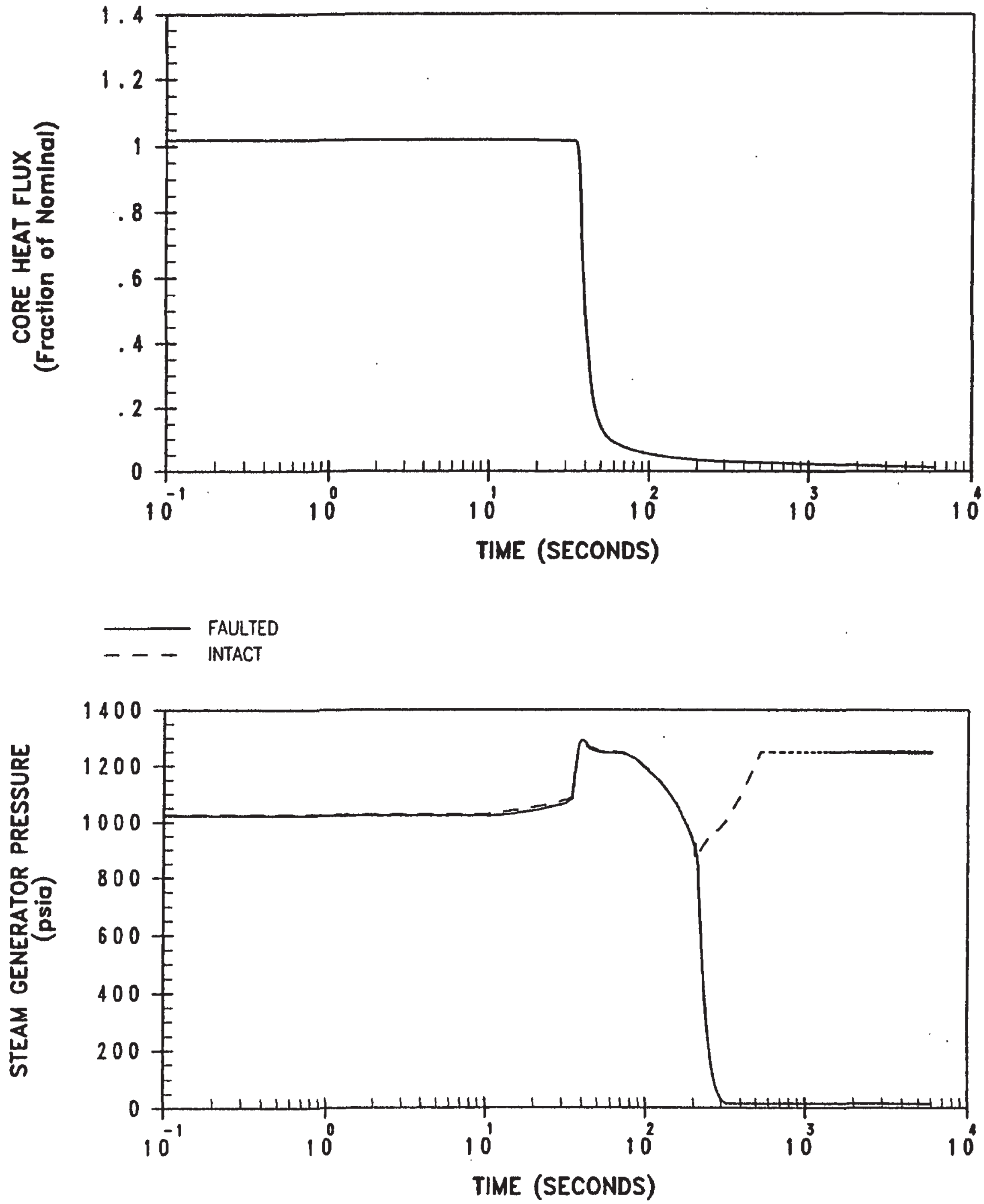

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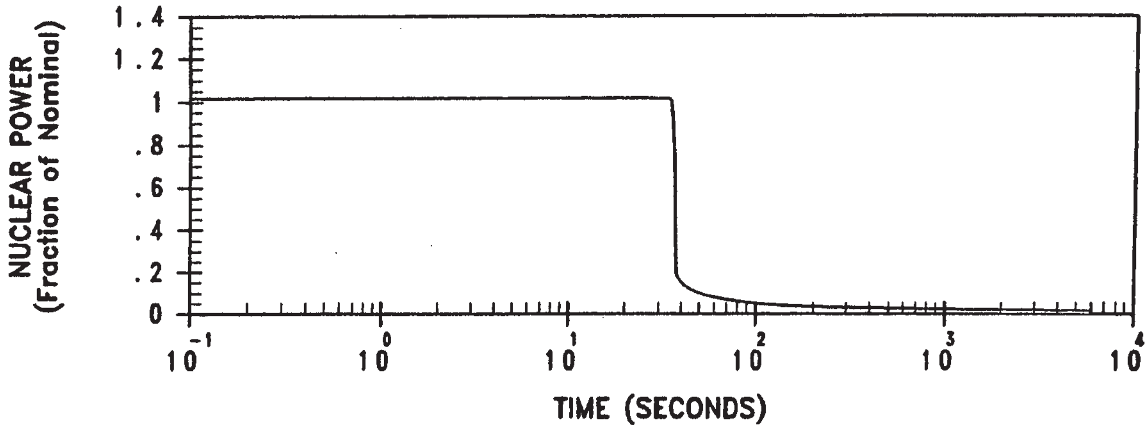
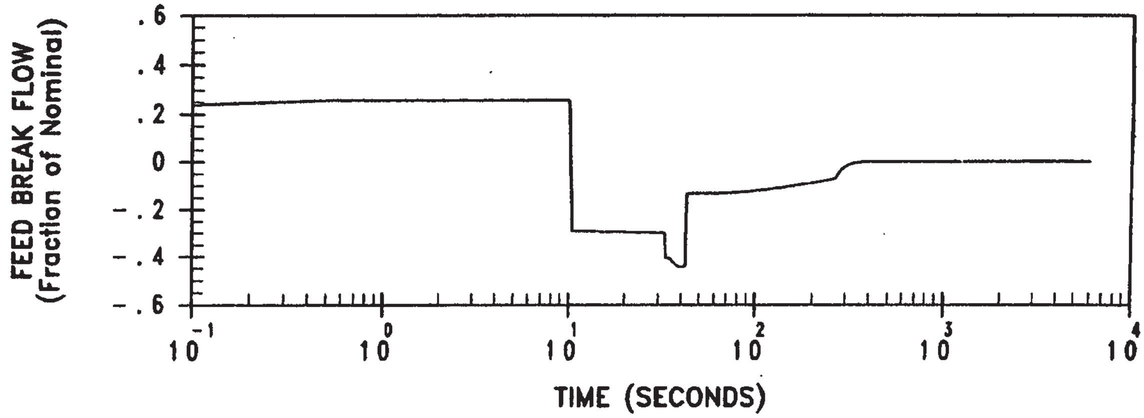
FIGURE 15.2-16

CORE HEAT FLUX AND STEAM
GENERATOR PRESSURE TRANSIENTS
FOR MAIN FEEDLINE BREAK WITH
OFFSITE POWER AVAILABLE





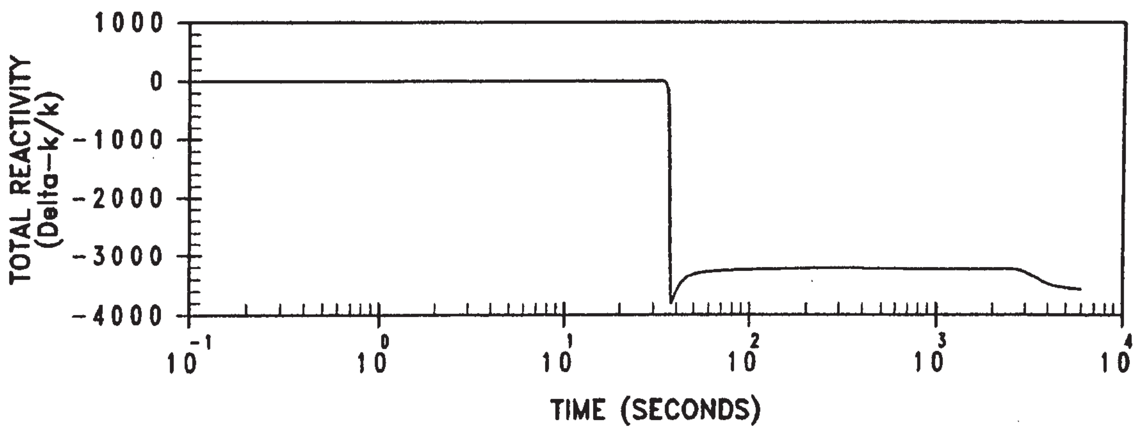


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FIGURE 15.2-17

NUCLEAR POWER, TOTAL CORE REACTIVITY AND FEEDLINE BREAK FLOW TRANSIENTS FOR MAIN FEEDLINE BREAK WITHOUT OFFSITE POWER AVAILABLE

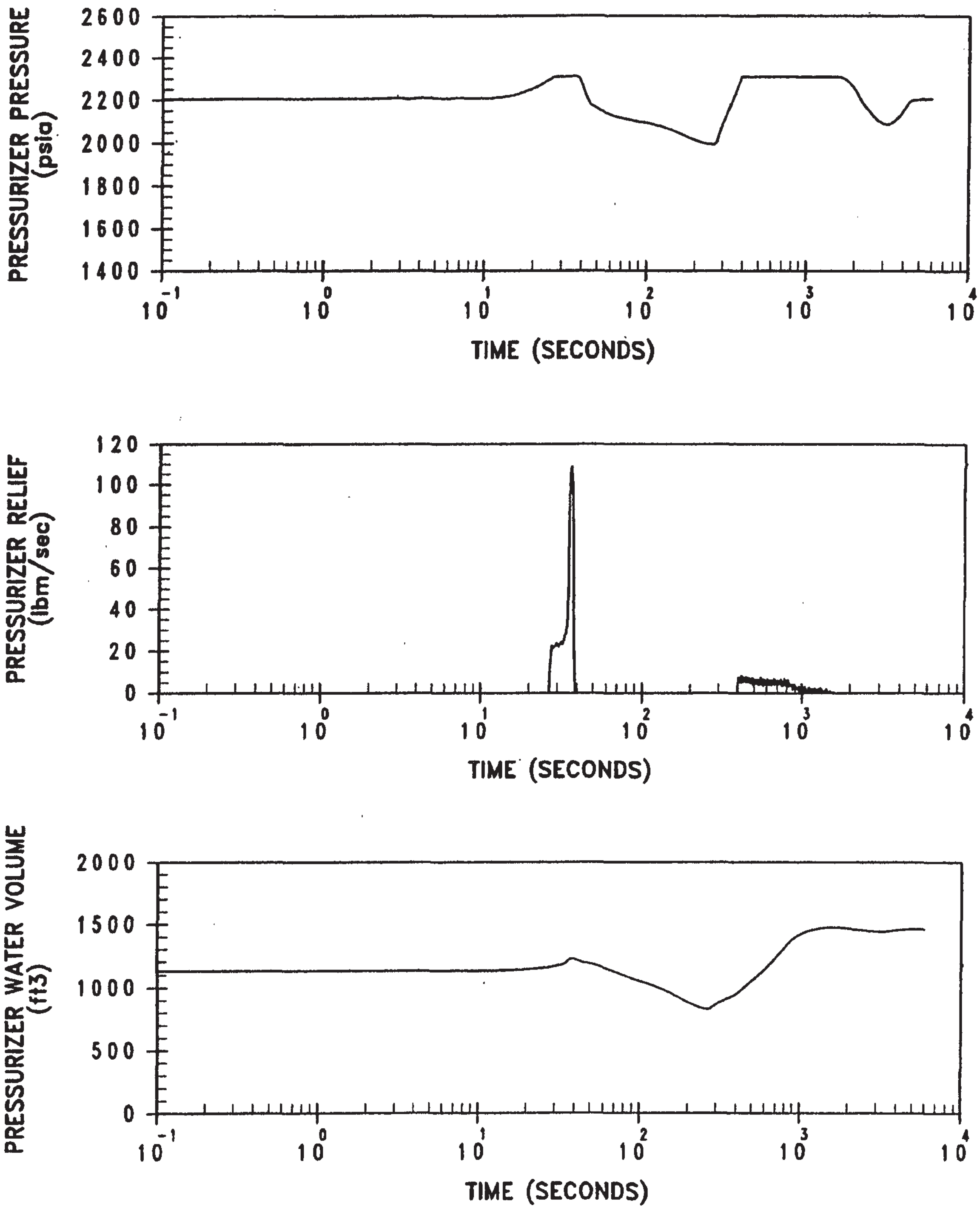

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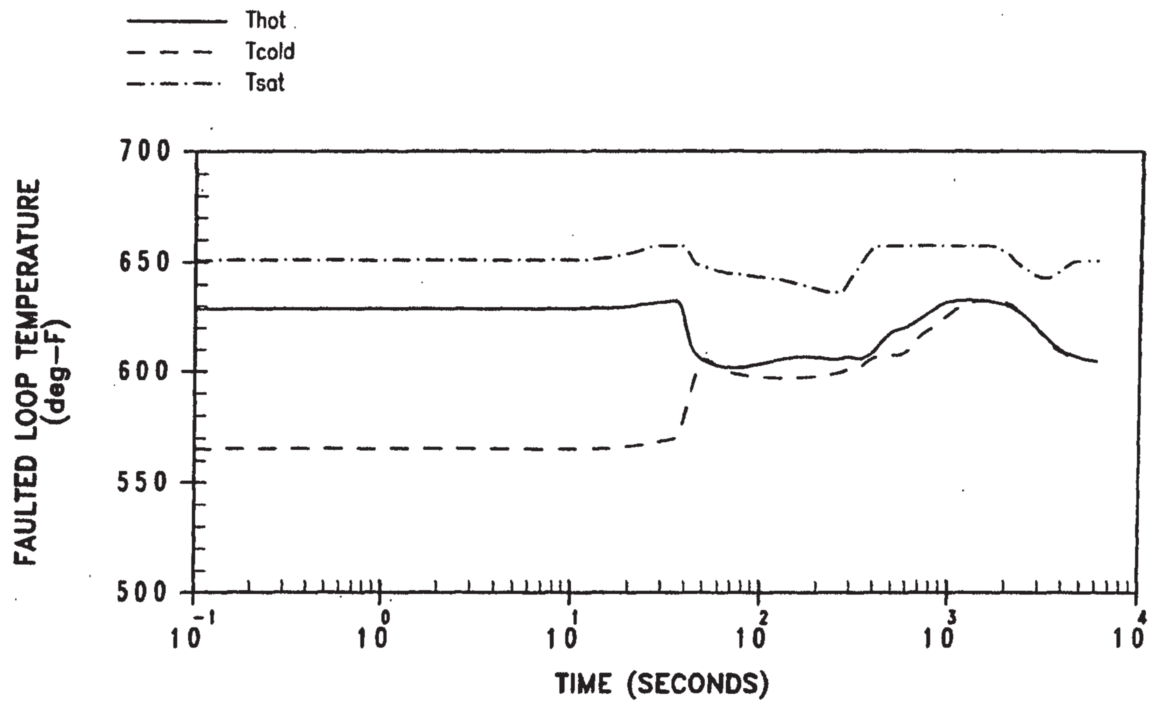
FIGURE 15.2-18

PRESSURIZER PRESSURE, PRESSURIZER RELIEF AND PRESSURIZER WATER VOLUME TRANSIENTS FOR MAIN FEEDLINE BREAK WITHOUT OFFSITE POWER AVAILABLE



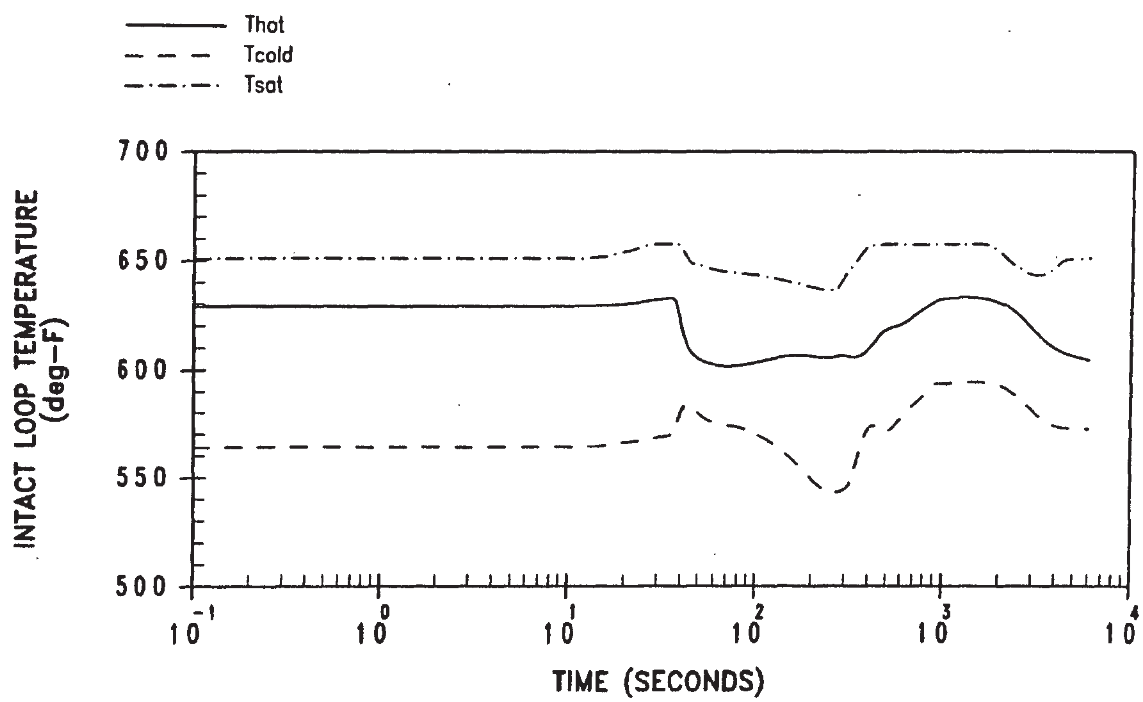


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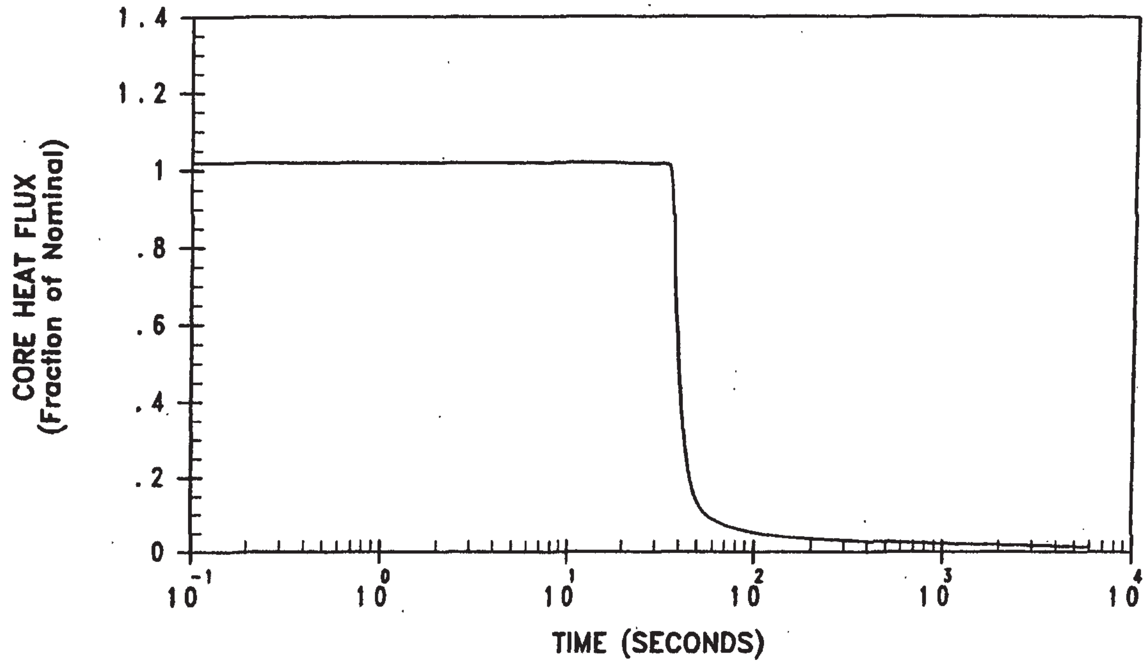
FIGURE 15.2-19

FAULTED AND INTACT LOOP
COOLANT TEMPERATURE TRANSIENTS
FOR MAIN FEEDLINE BREAK WITHOUT
OFFSITE POWER AVAILABLE



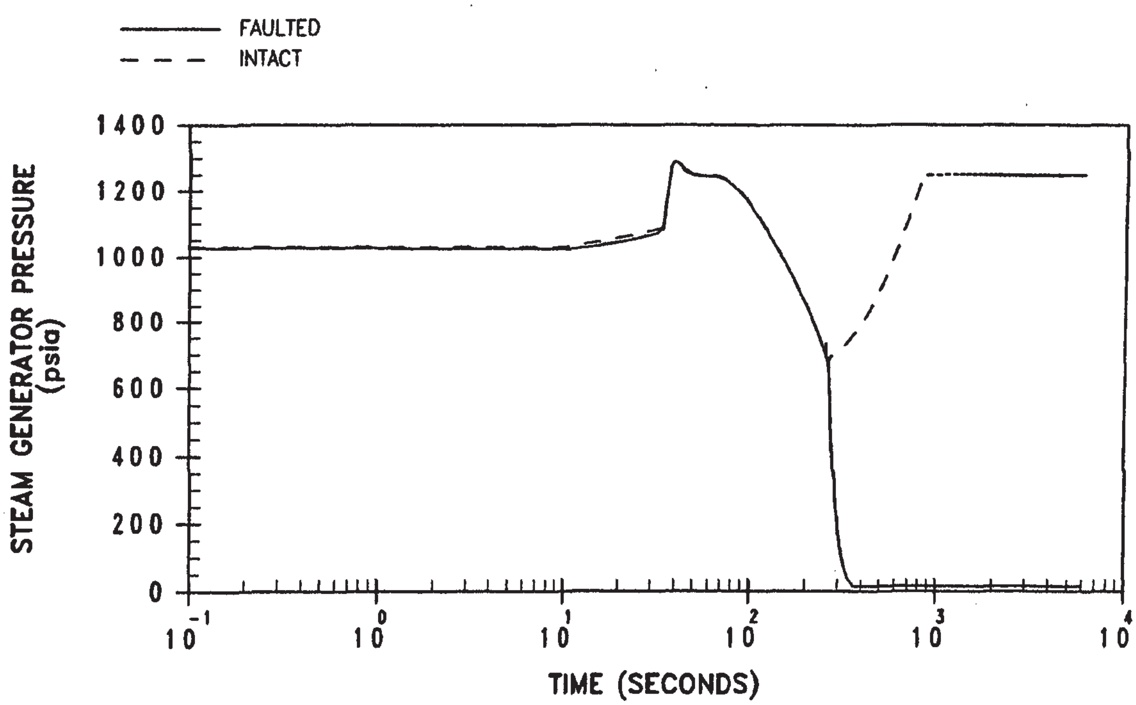


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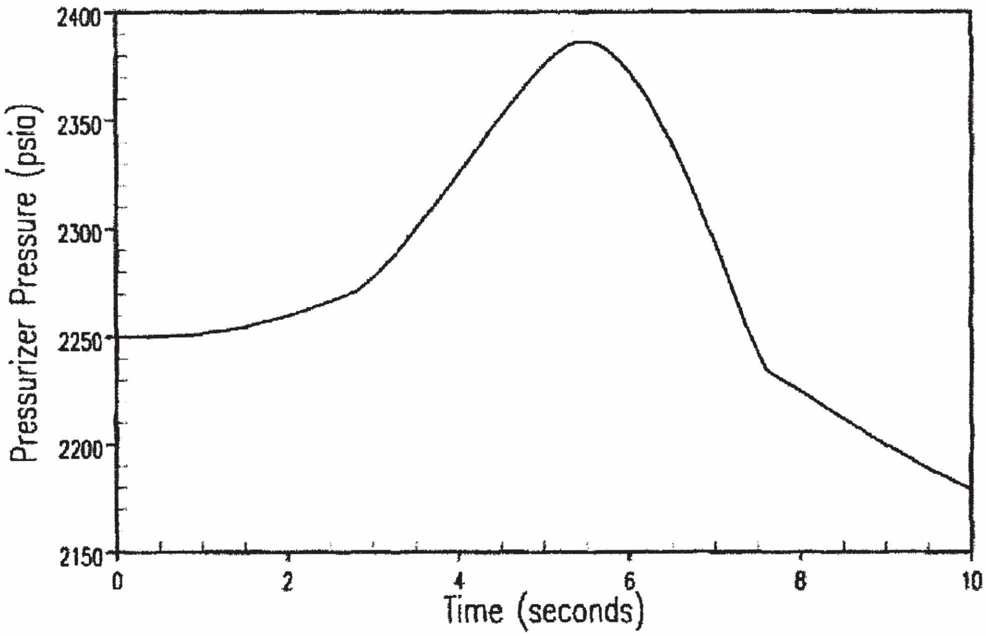
FIGURE 15.2-20

CORE HEAT FLUX AND STEAM
GENERATOR PRESSURE TRANSIENTS
FOR MAIN FEEDLINE BREAK WITHOUT
OFFSITE POWER AVAILABLE



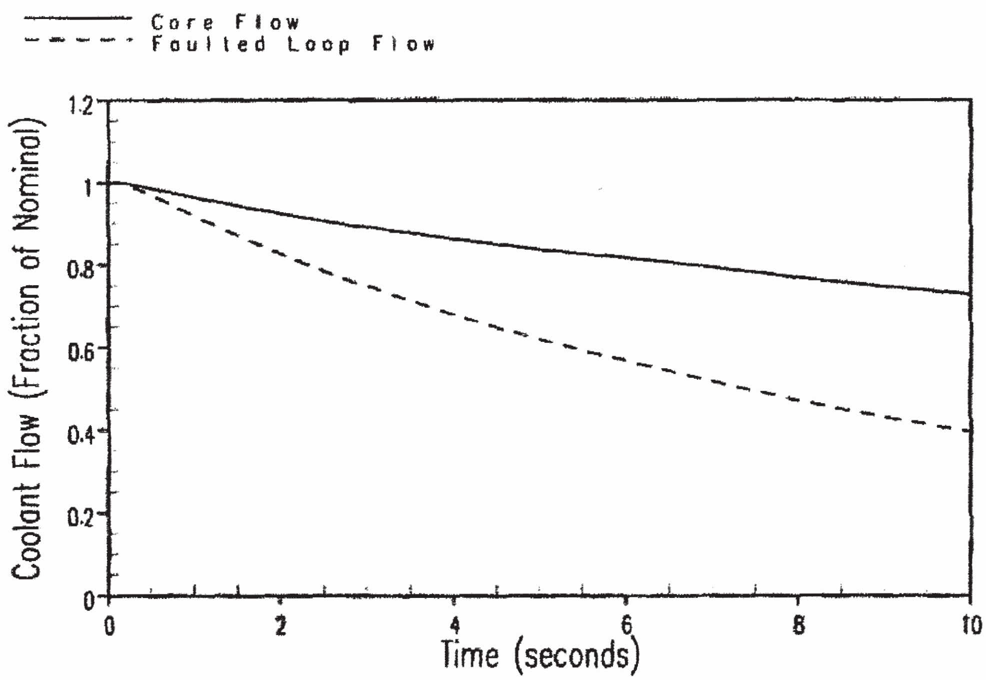


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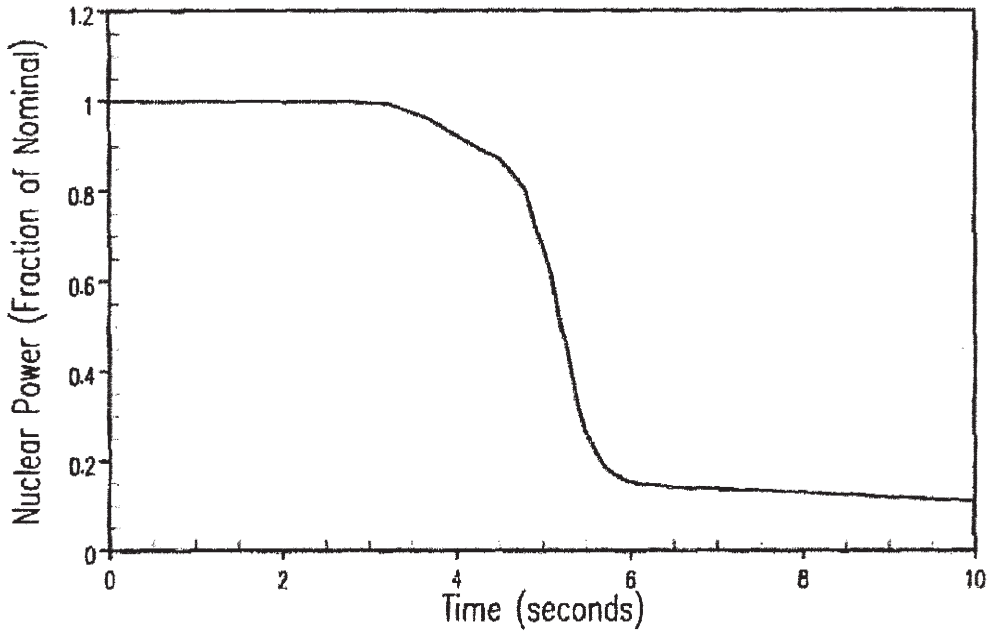
FIGURE 15.3-1

PARTIAL LOSS
OF FORCED
REACTOR COOLANT FLOW



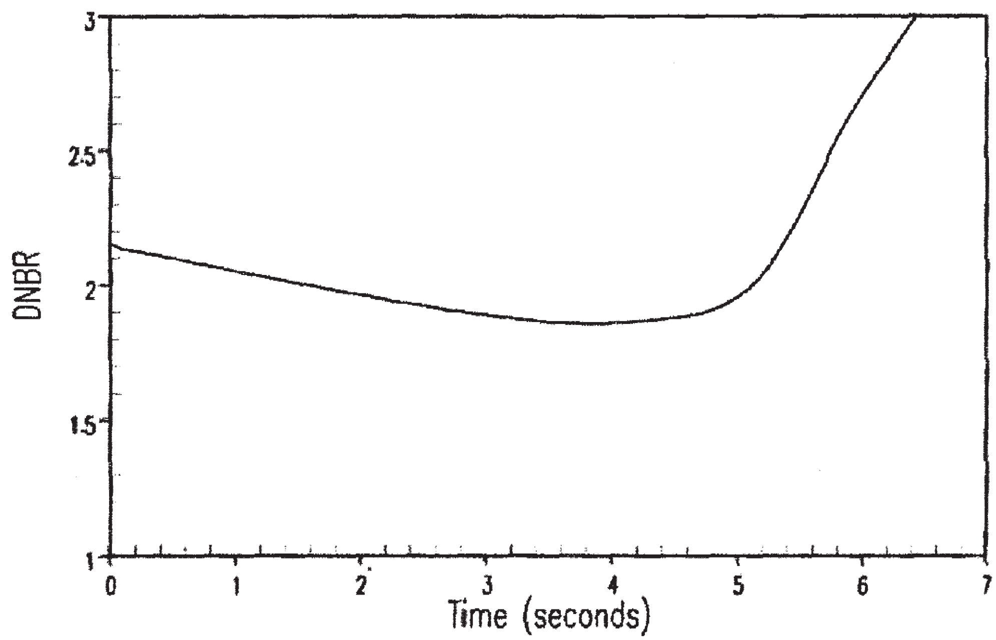


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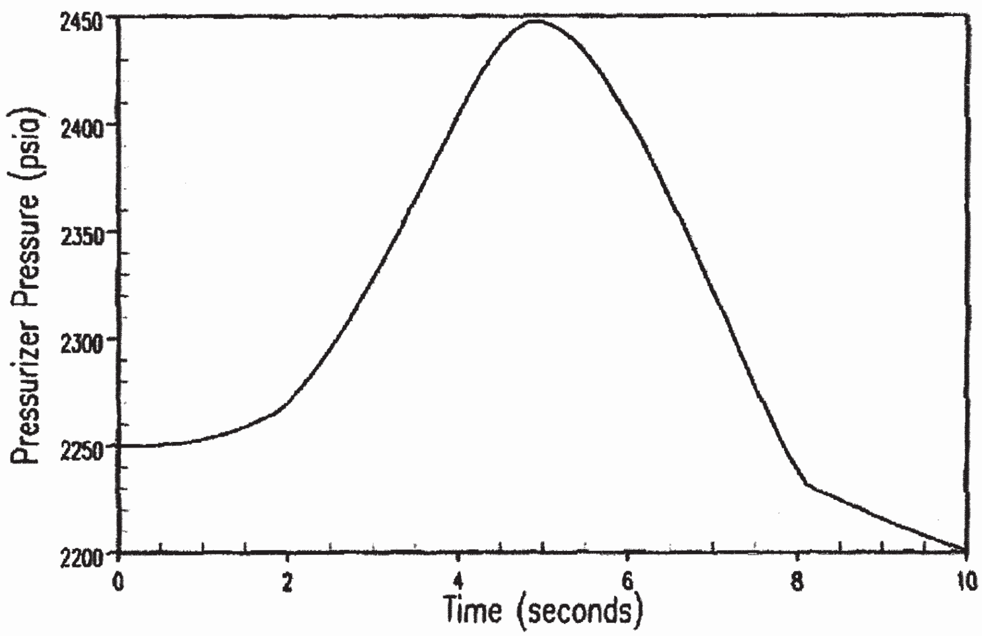
FIGURE 15.3-2

PARTIAL LOSS
OF FORCED
REACTOR COOLANT FLOW



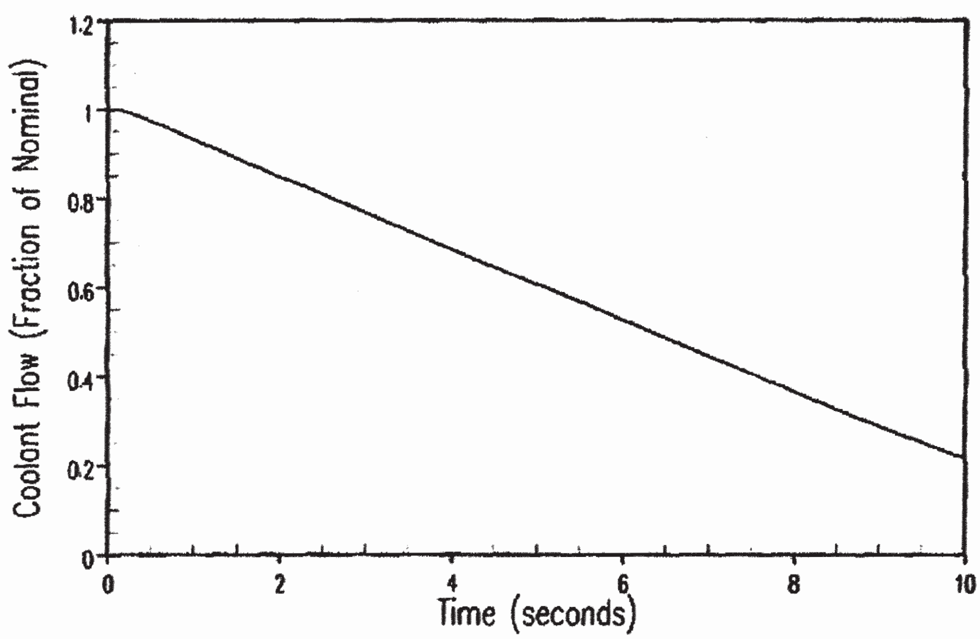


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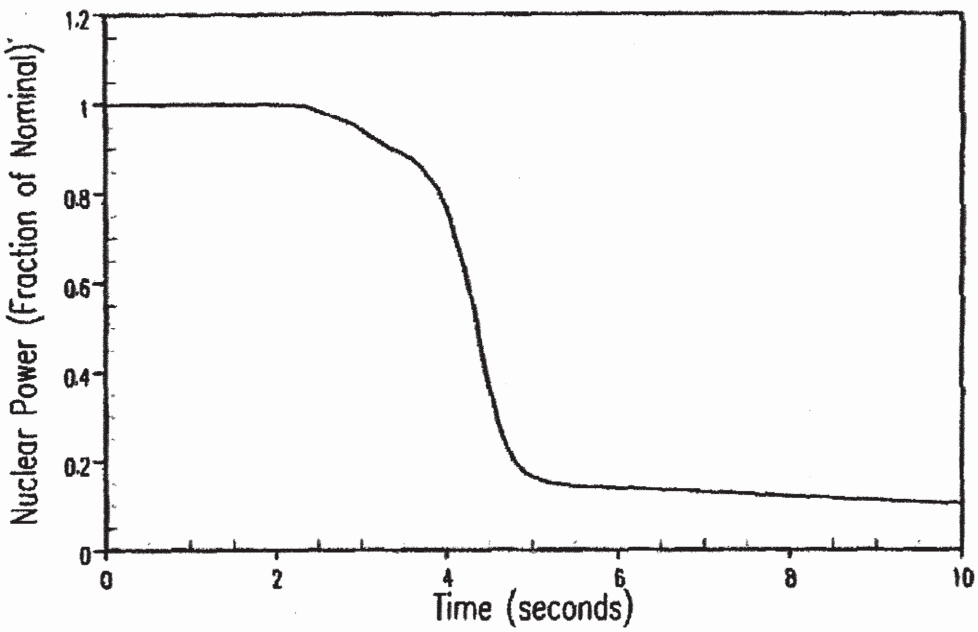
FIGURE 15.3-3

COMPLETE LOSS
OF FORCED
REACTOR COOLANT FLOW



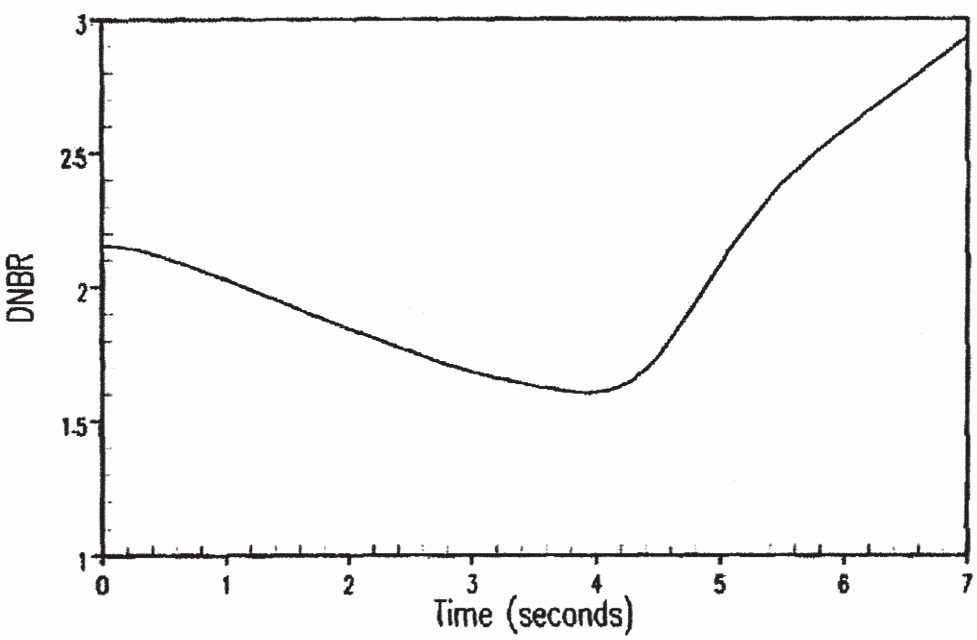


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FIGURE 15.3-4

COMPLETE LOSS
OF FORCED
REACTOR COOLANT FLOW

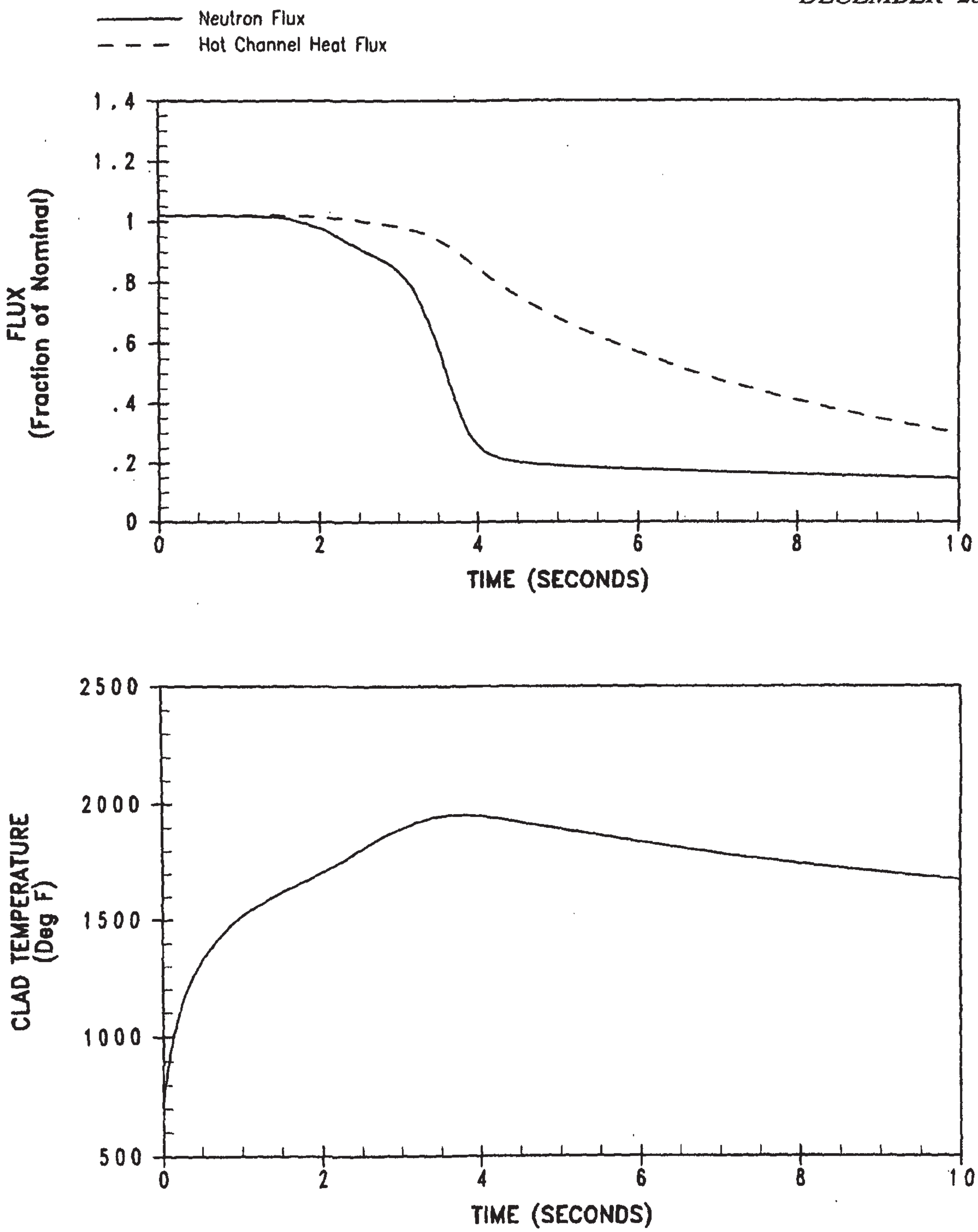

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FIGURE 15.3-5

SINGLE REACTOR
COOLANT PUMP
LOCKED ROTOR

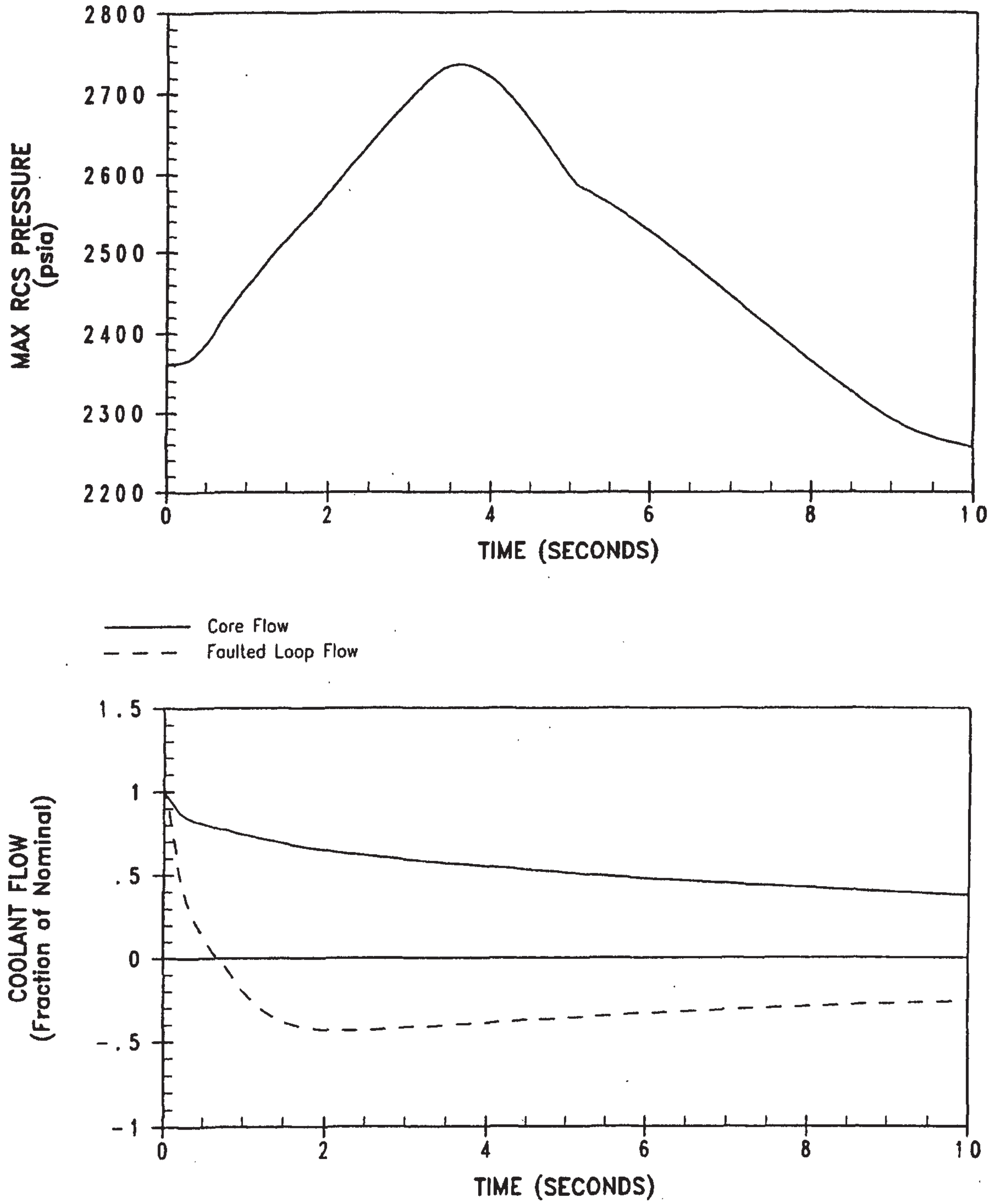

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FIGURE 15.3-6

SINGLE REACTOR
COOLANT PUMP
LOCKED ROTOR

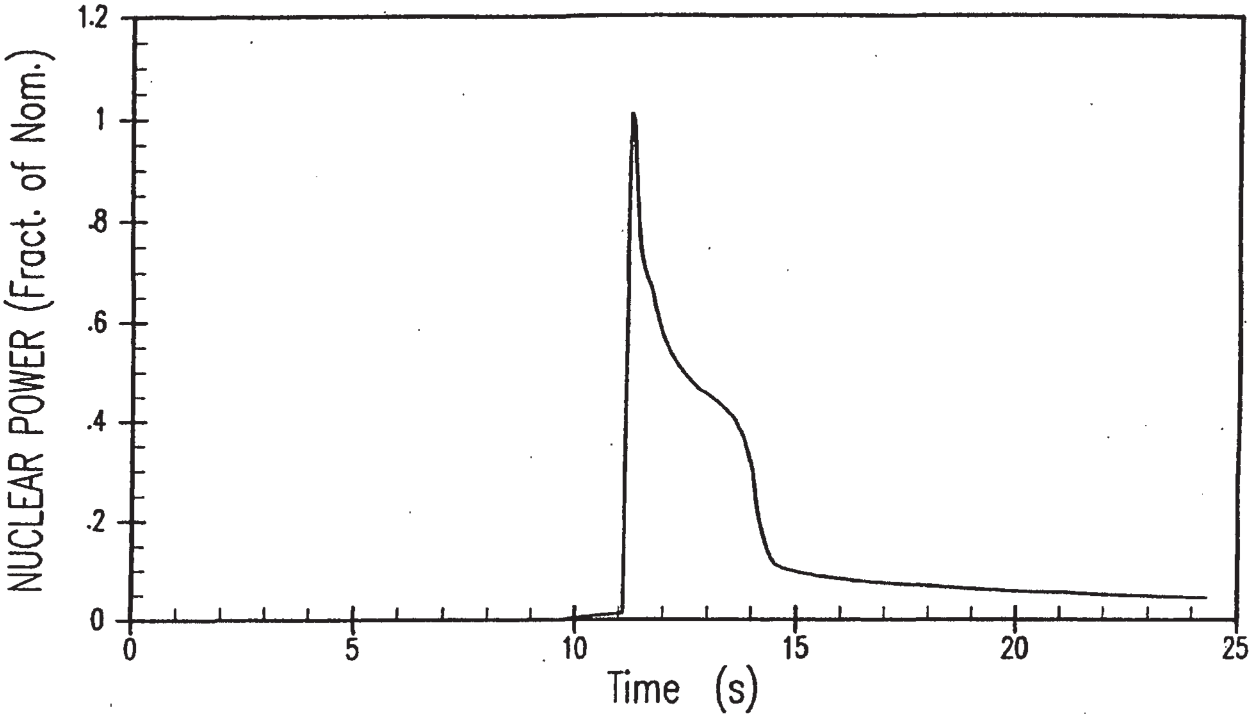

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FIGURE 15.4-1

NEUTRON FLUX TRANSIENT FOR
UNCONTROLLED ROD WITHDRAWAL
FROM A SUBCRITICAL CONDITION

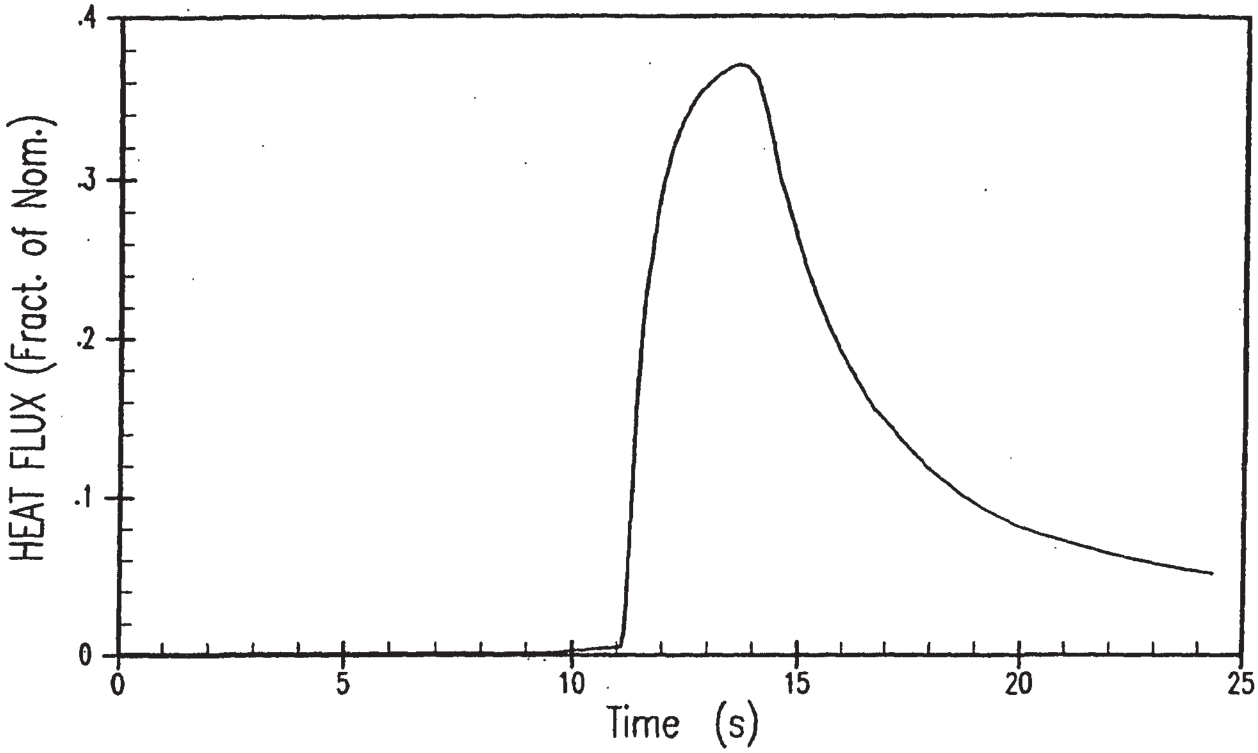

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FIGURE 15.4-2

THERMAL FLUX TRANSIENT FOR
UNCONTROLLED ROD WITHDRAWAL
FROM A SUBCRITICAL CONDITION

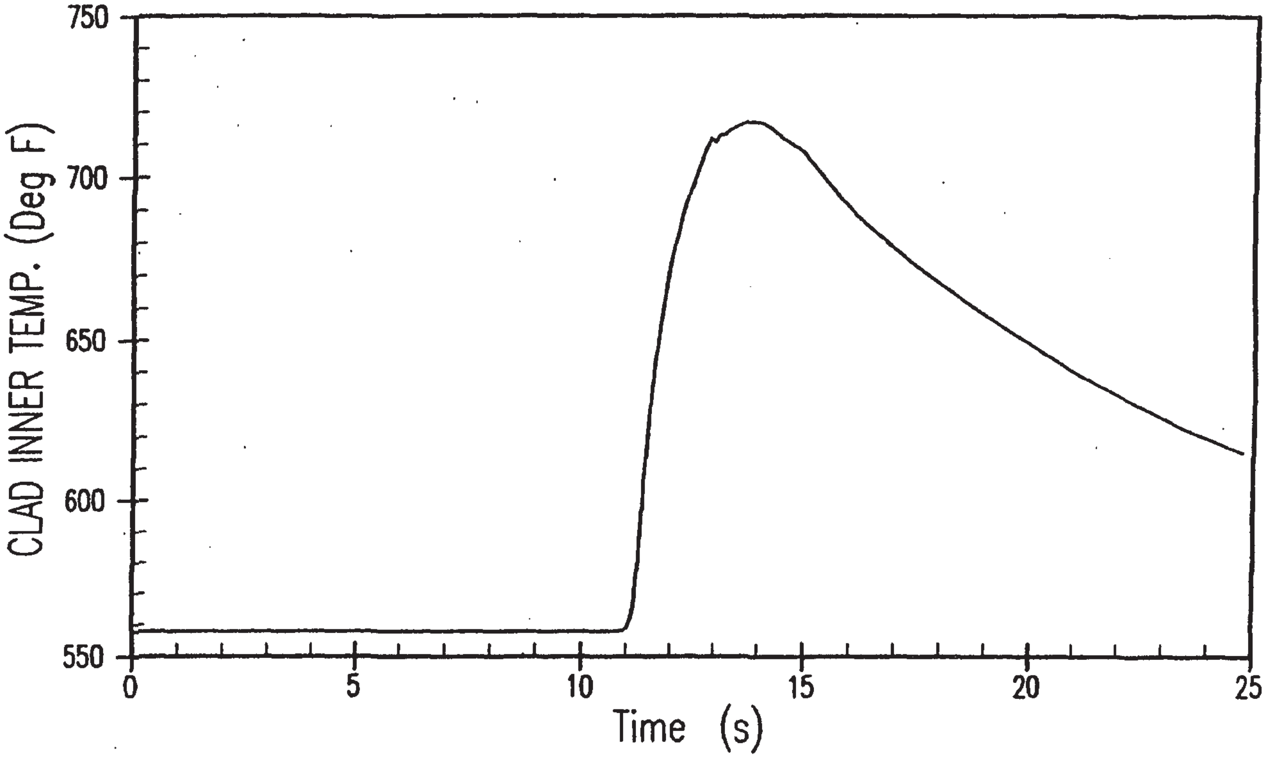

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FIGURE 15.4-3 FUEL AND CLAD TEMPERATURE TRANSIENTS FORUNCONTROLLED ROD WITHDRAWAL FROM A SUBCRITICAL CONDITION (SHEET 1 OF 2)

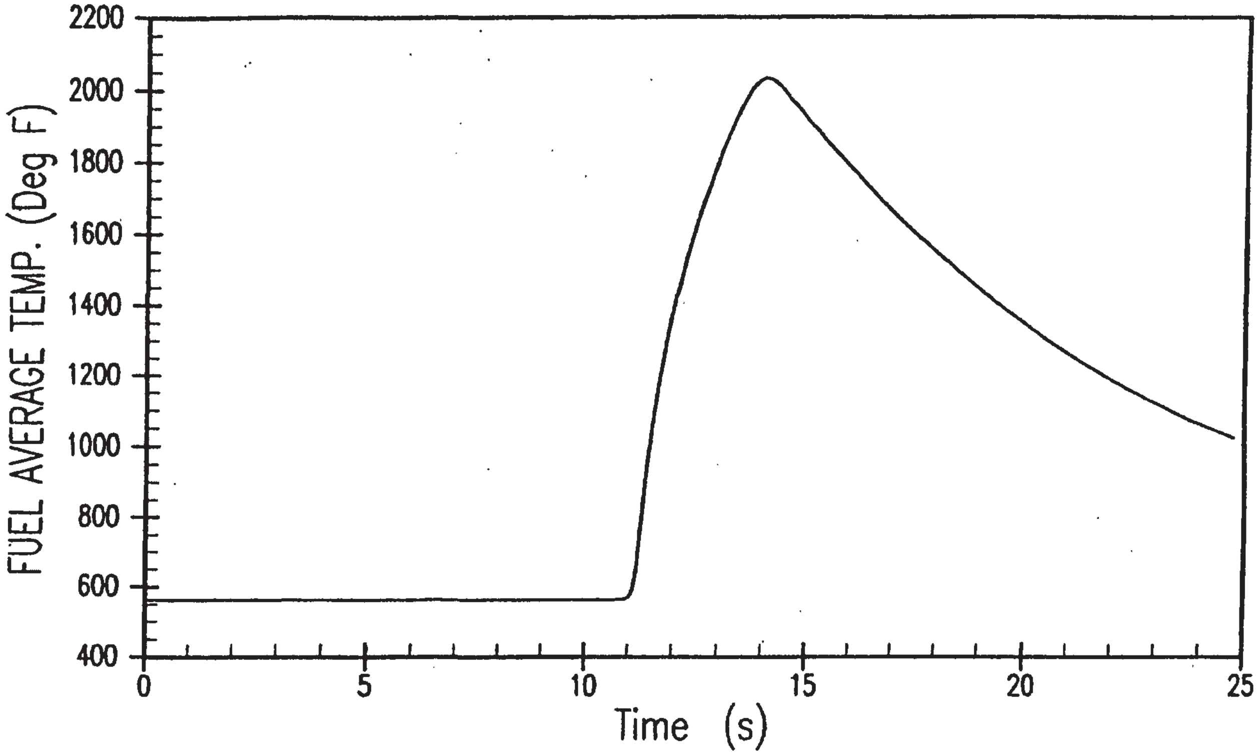

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FIGURE 15.4-3 FUEL AND CLAD TEMPERATURE TRANSIENTS FORUNCONTROLLED ROD WITHDRAWAL FROM A SUBCRITICAL CONDITION (SHEET 2 OF 2)

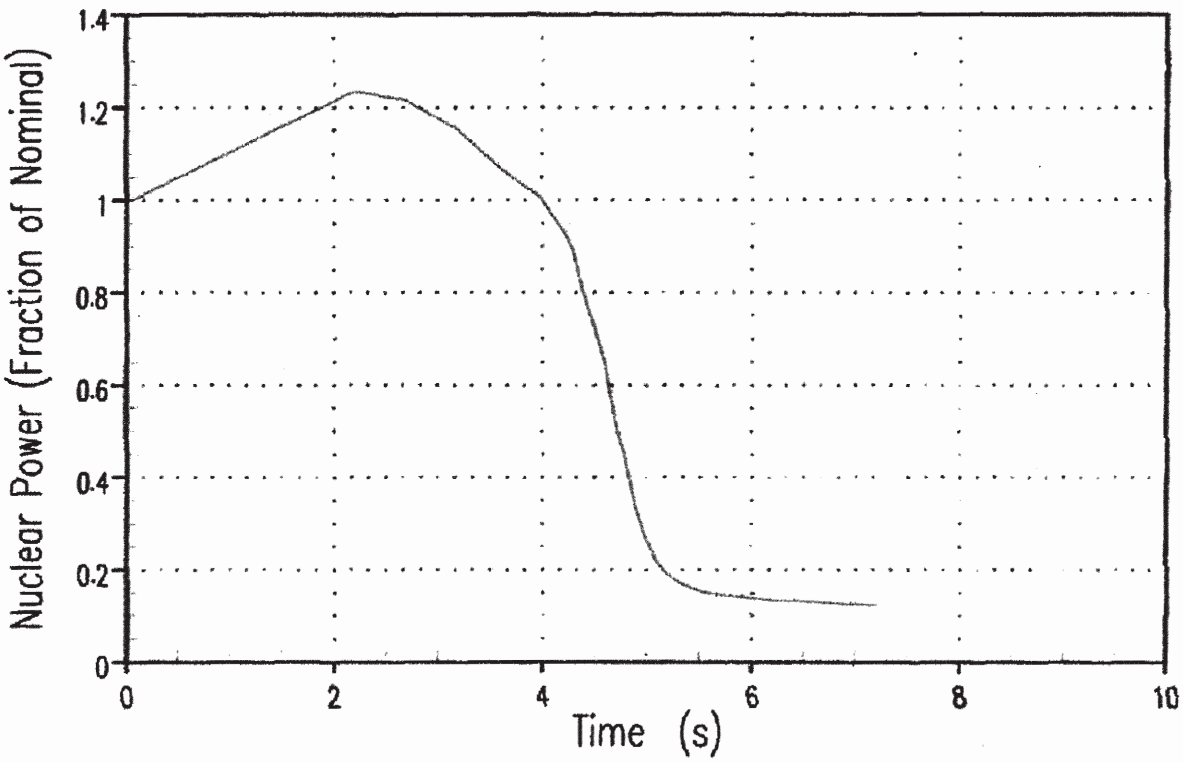

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FIGURE 15.4-4
UNCONTROLLED RCCA BANK WITHDRAWAL FROM FULL POWER WITH MINIMUM REACTIVITY FEEDBACK
(80 PCM/SEC WITHDRAWAL RATE)
(SHEET 1 OF 3)

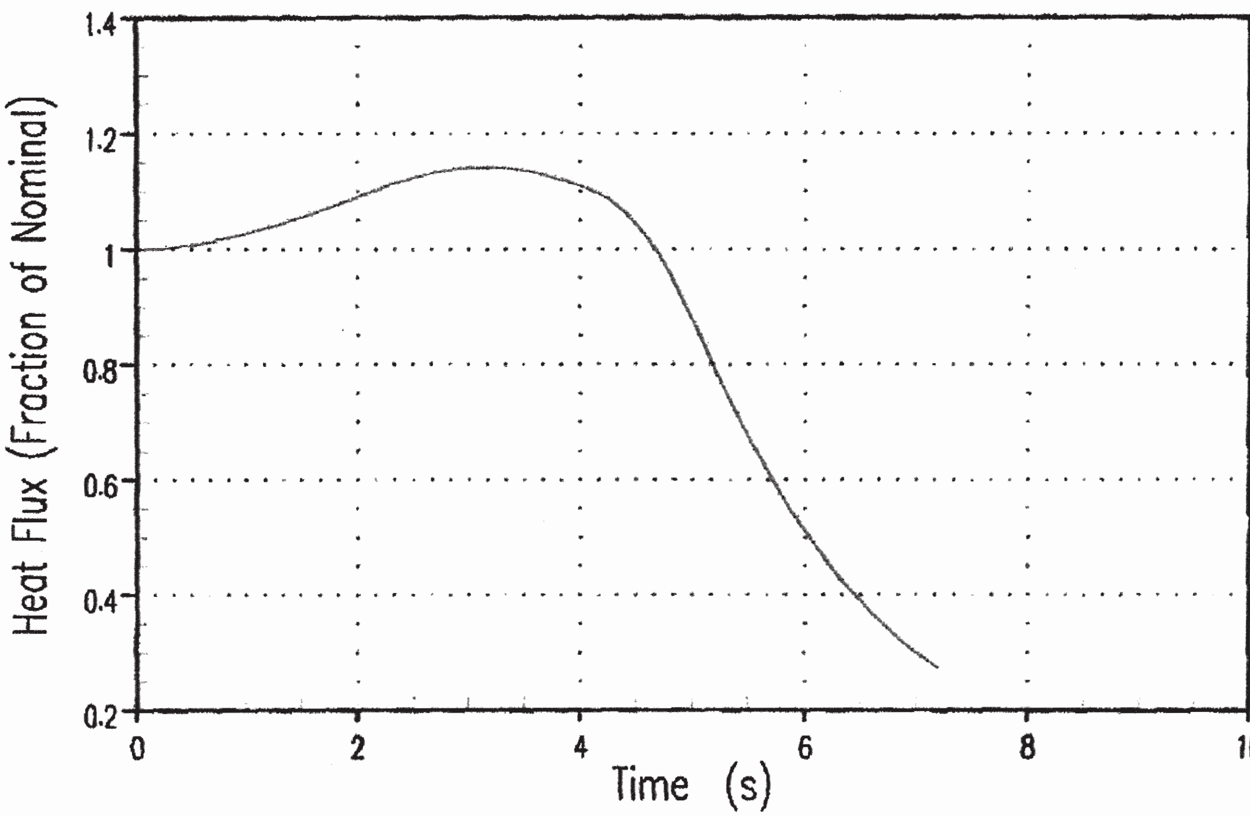

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FIGURE 15.4-4
UNCONTROLLED RCCA BANK WITHDRAWAL FROM FULL POWER WITH MINIMUM REACTIVITY FEEDBACK (80 PCM/SEC WITHDRAWAL RATE)
(SHEET 2 OF 3)

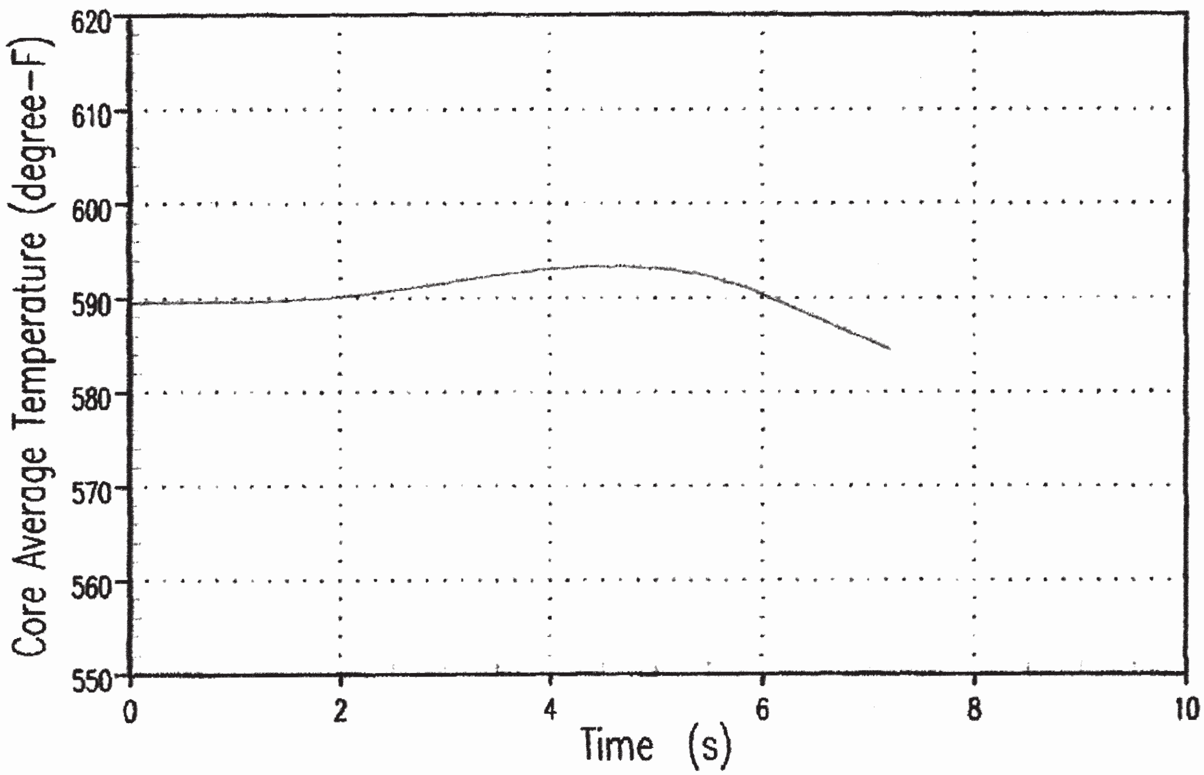

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FIGURE 15.4-4
UNCONTROLLED RCCA BANK WITHDRAWAL FROM FULL POWER WITH MINIMUM REACTIVITY FEEDBACK (80 PCM/SEC WITHDRAWAL RATE)
(SHEET 3 OF 3)

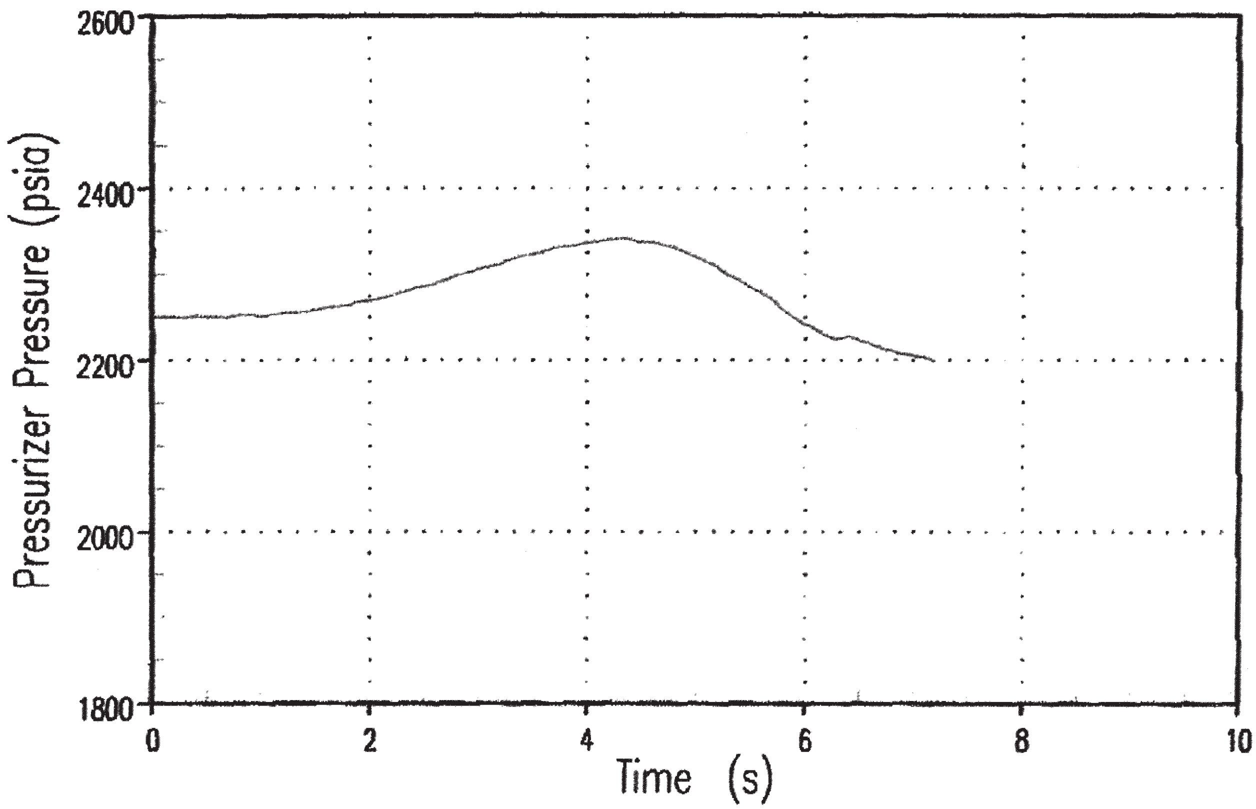

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FIGURE 15.4-5
UNCONTROLLED RCCA BANK WITHDRAWAL FROM FULL POWER WITH MINIMUM REACTIVITY FEEDBACK (80 PCM/SEC WITHDRAWAL RATE)
(SHEET 1 OF 3)

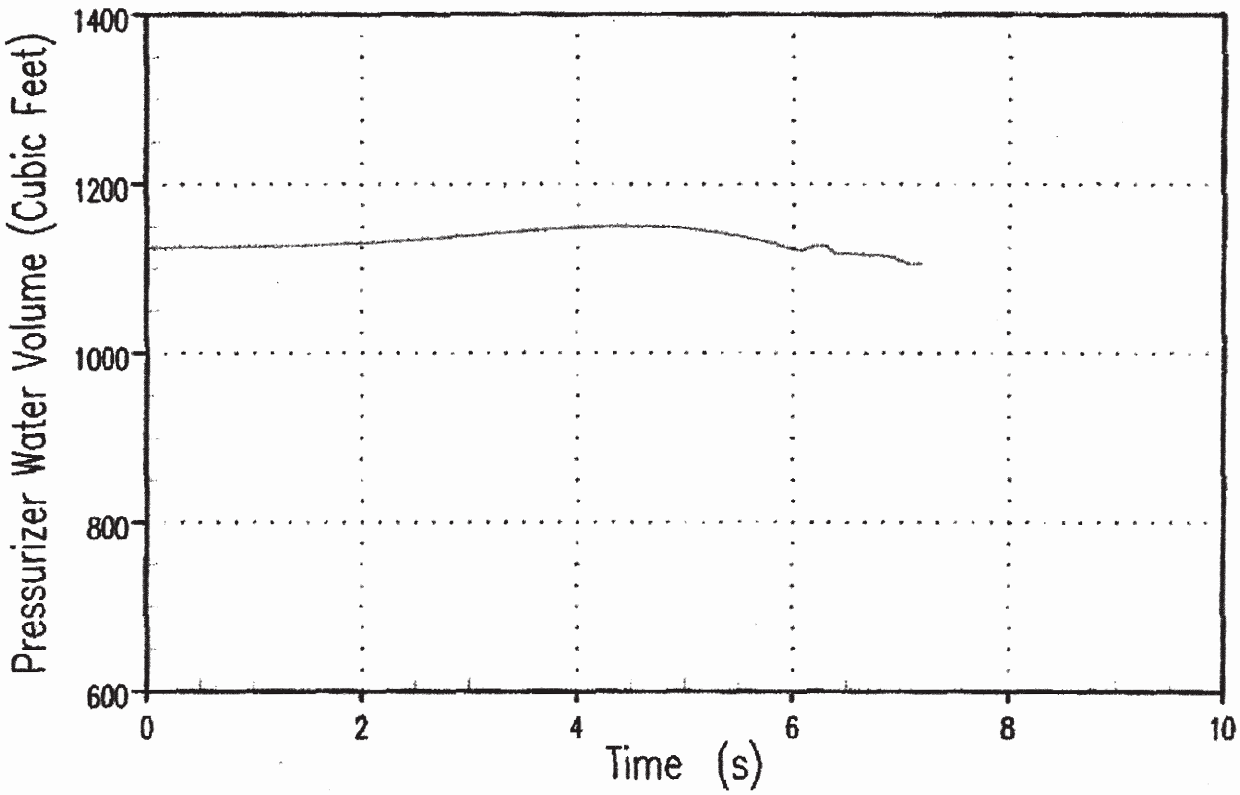

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FIGURE 15.4-5
UNCONTROLLED RCCA BANK WITHDRAWAL FROM FULL POWER WITH MINIMUM REACTIVITY FEEDBACK
(80 PCM/SEC WITHDRAWAL RATE)
(SHEET 2 OF 3)

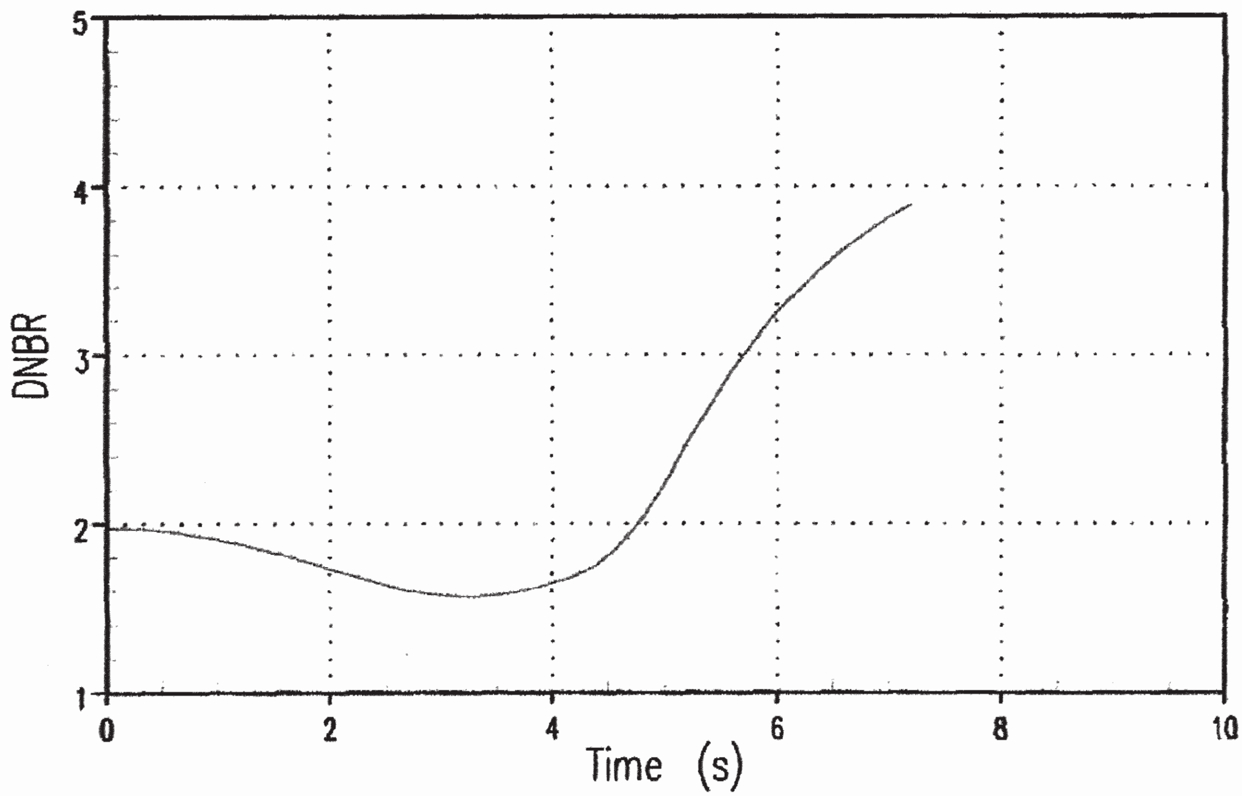

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FIGURE 15.4-5
UNCONTROLLED RCCA BANK WITHDRAWAL FROM FULL POWER WITH MINIMUM REACTIVITY FEEDBACK
(80 PCM/SEC WITHDRAWAL RATE)
(SHEET 3 OF 3)

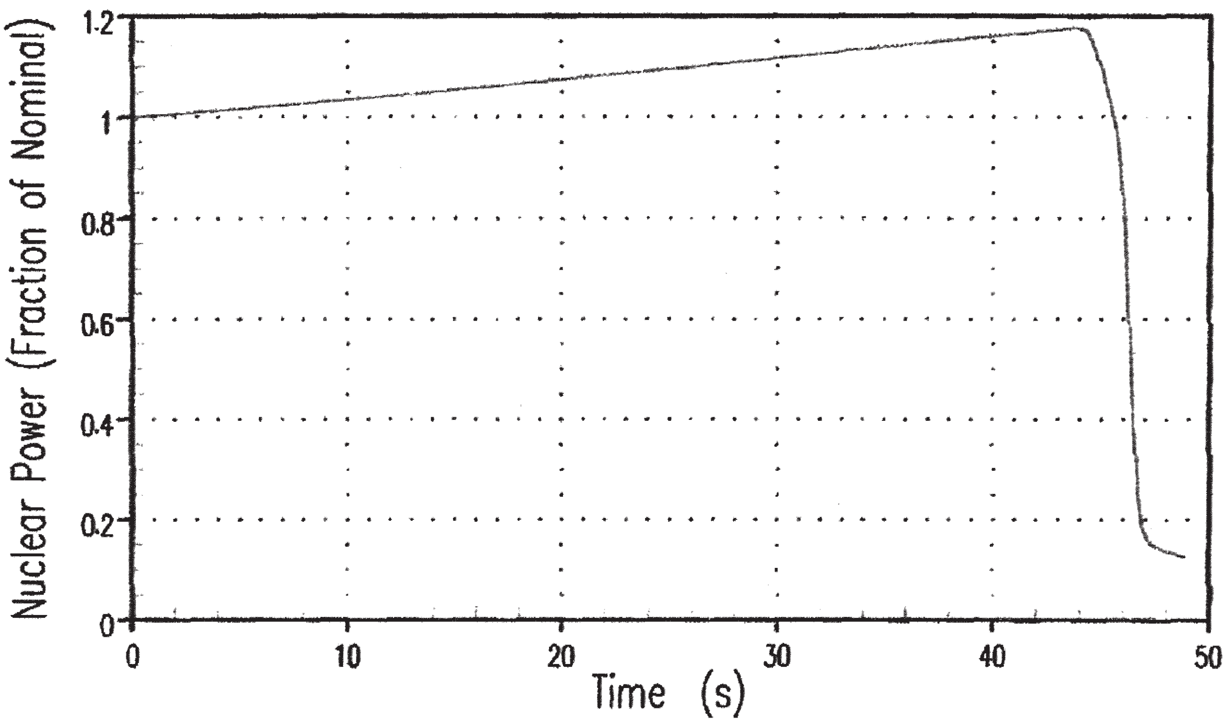

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FIGURE 15.4-6
UNCONTROLLED RCCA BANK WITHDRAWAL FROM FULL POWER WITH MINIMUM REACTIVITY FEEDBACK
(3.0 PCM/SEC WITHDRAWAL RATE)
(SHEET 1 OF 3)

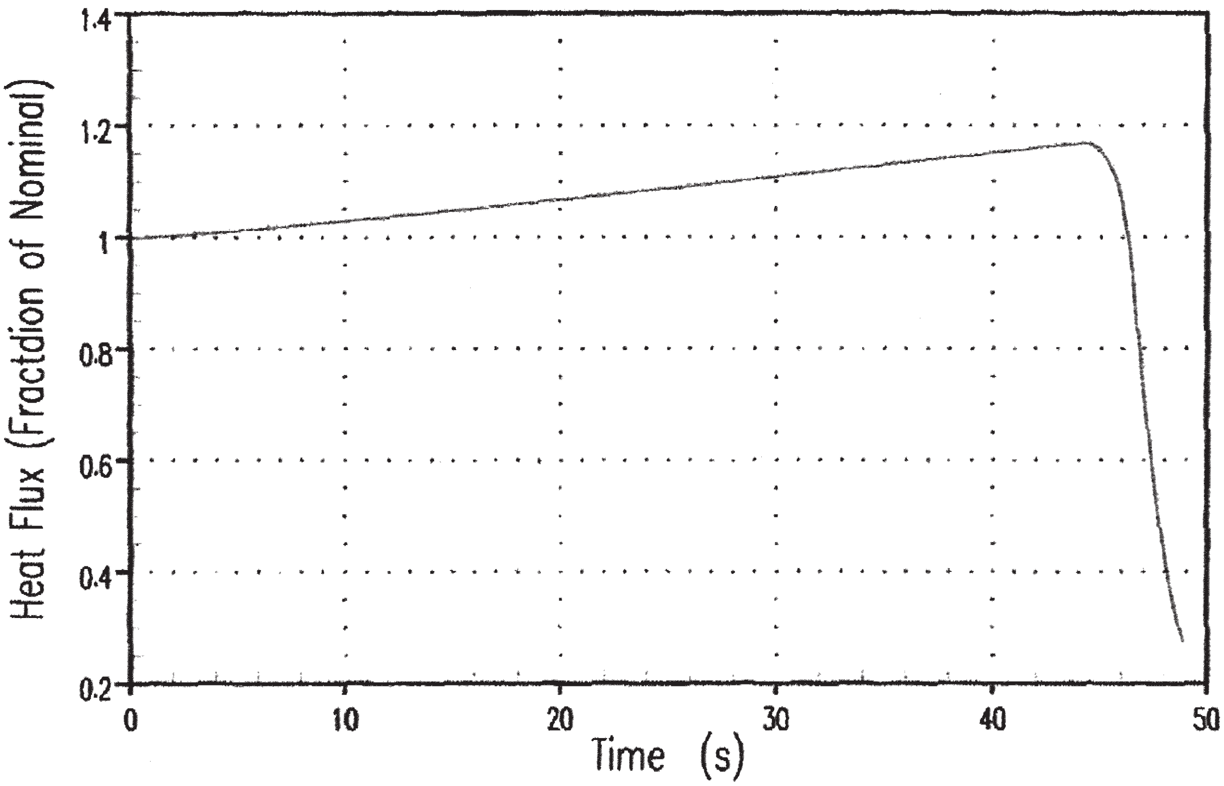

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FIGURE 15.4-6
UNCONTROLLED RCCA BANK WITHDRAWAL FROM FULL POWER WITH MINIMUM REACTIVITY FEEDBACK
(3.0 PCM/SEC WITHDRAWAL RATE)
(SHEET 2 OF 3)

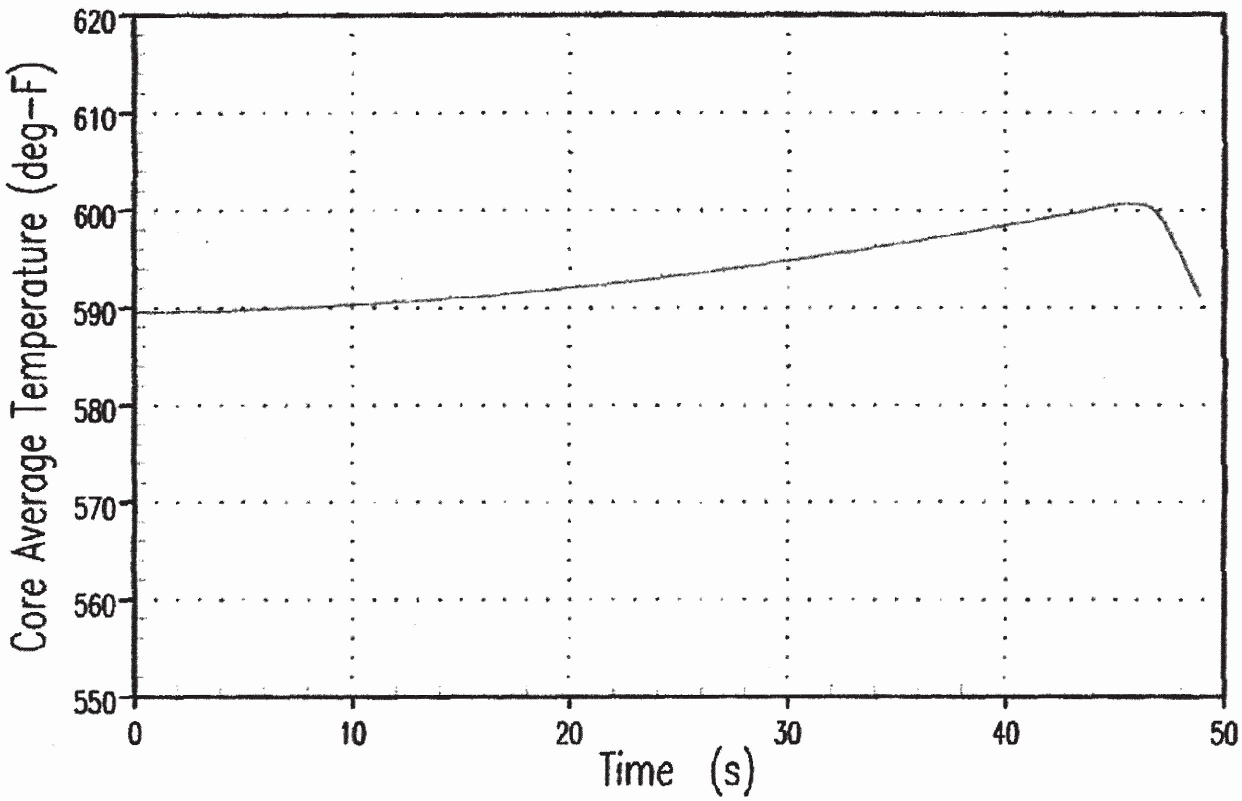

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FIGURE 15.4-6
UNCONTROLLED RCCA BANK WITHDRAWAL FROM FULL POWER WITH MINIMUM REACTIVITY FEEDBACK
(3.0 PCM/SEC WITHDRAWAL RATE)
(SHEET 3 OF 3)

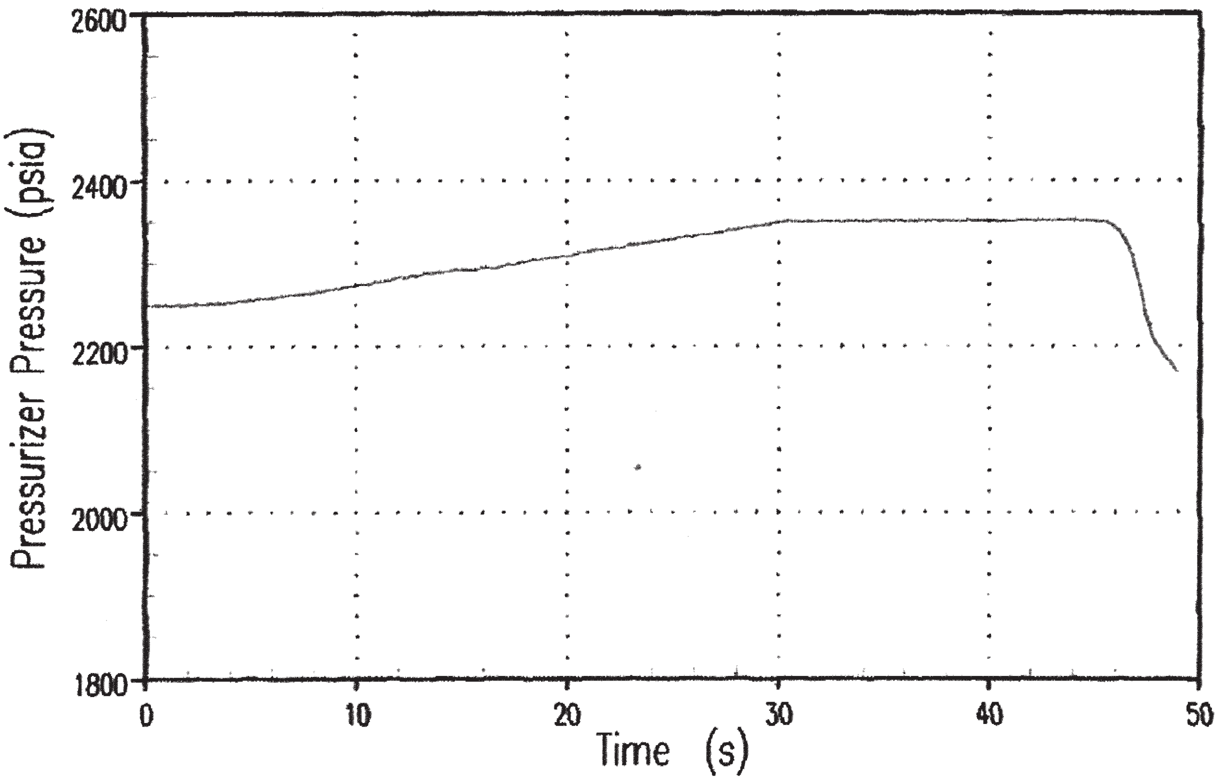

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FIGURE 15.4-7
UNCONTROLLED RCCA BANK WITHDRAWAL FROM FULL POWER WITH MINIMUM REACTIVITY FEEDBACK
(3.0 PCM/SEC WITHDRAWAL RATE)
(SHEET 1 OF 3)

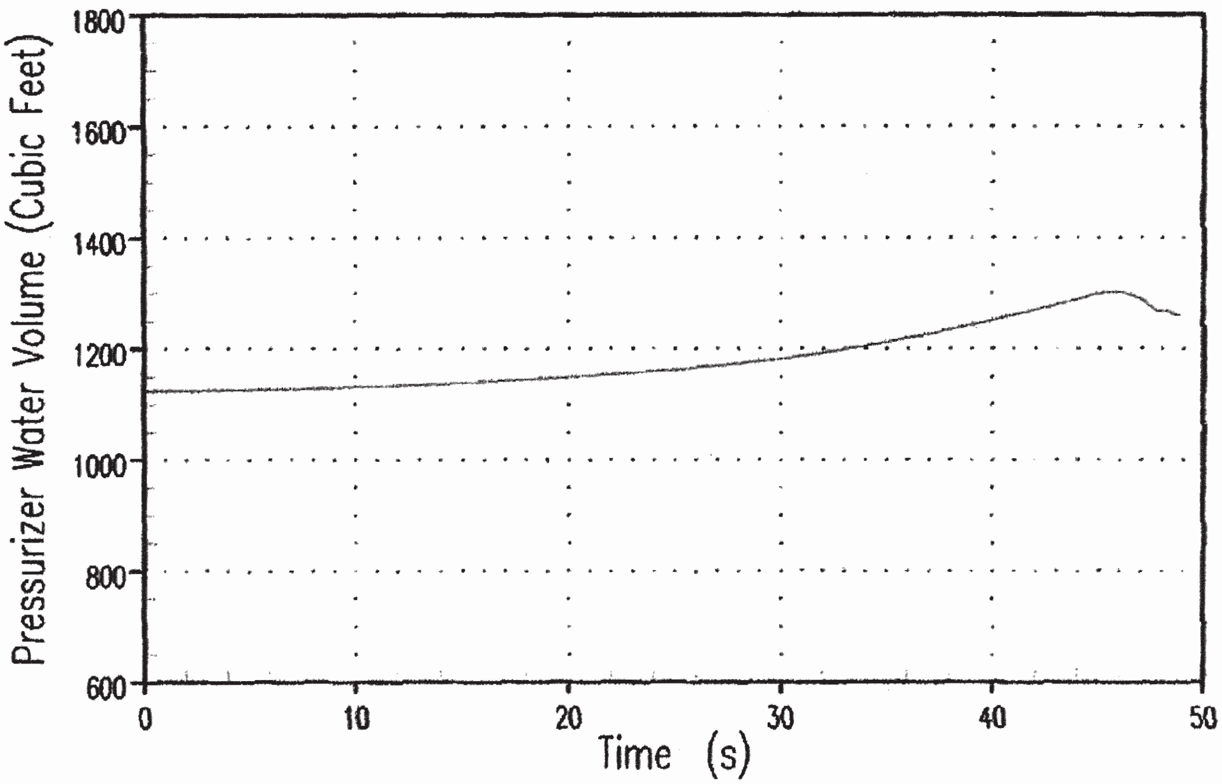

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FIGURE 15.4-7
UNCONTROLLED RCCA BANK WITHDRAWAL FROM FULL POWER WITH MINIMUM REACTIVITY FEEDBACK (3.0 PCM/SEC WITHDRAWAL RATE)
(SHEET 2 OF 3)

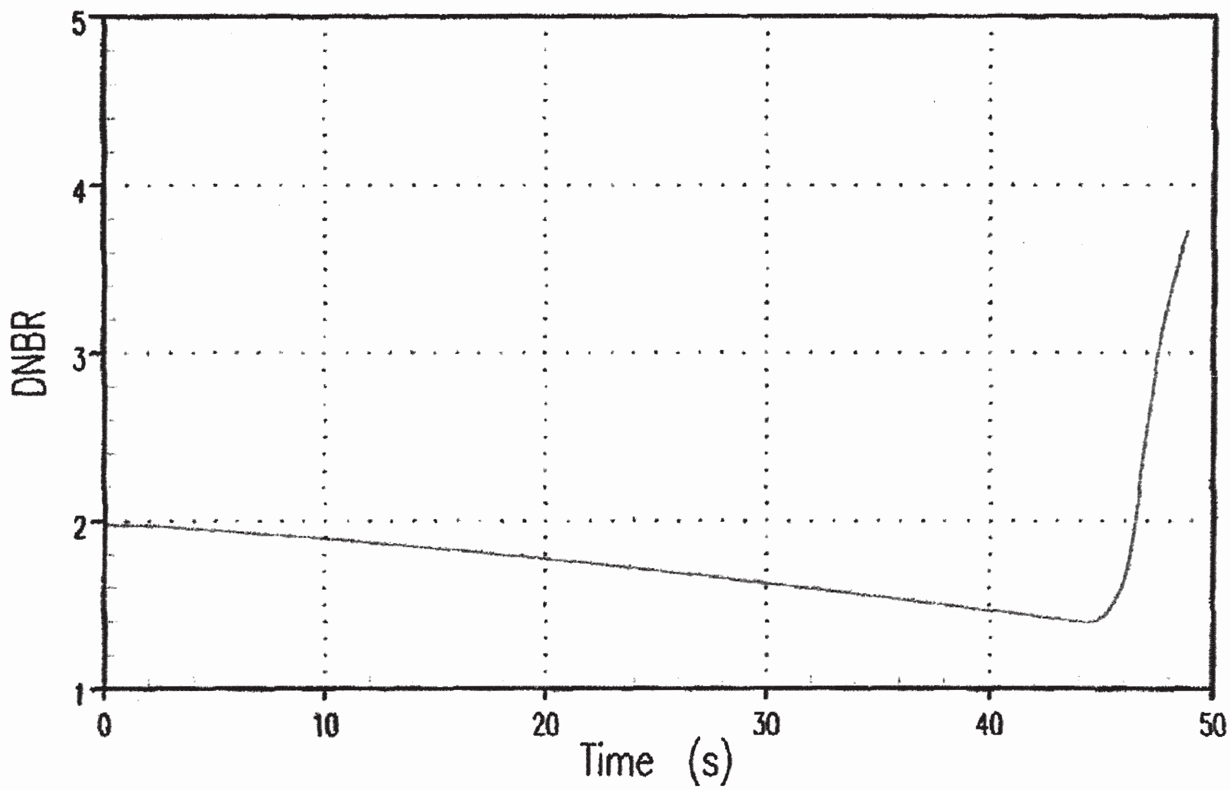

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FIGURE 15.4-7
UNCONTROLLED RCCA BANK WITHDRAWAL FROM FULL POWER WITH MINIMUM REACTIVITY FEEDBACK (3.0 PCM/SEC WITHDRAWAL RATE)
(SHEET 3 OF 3)

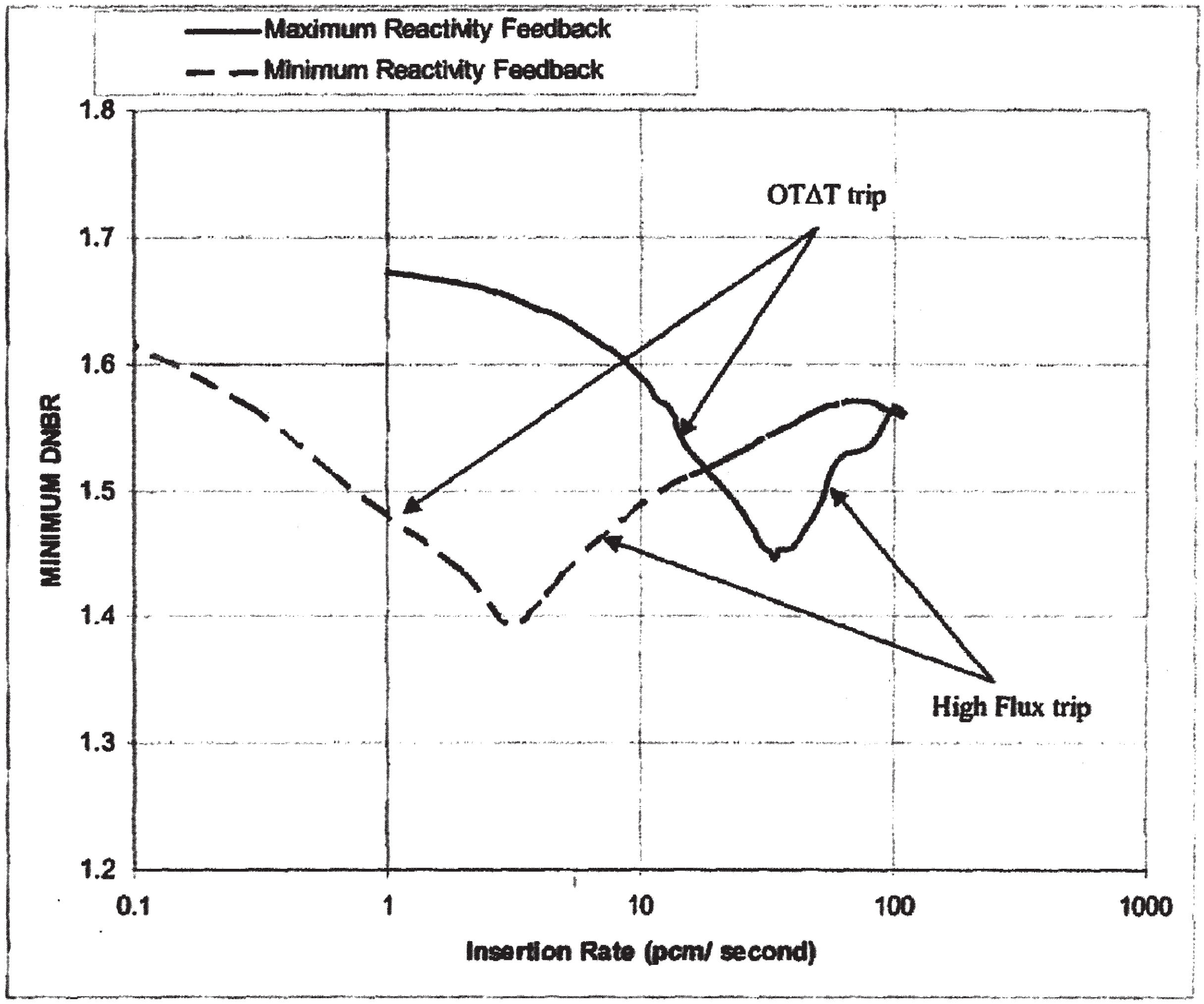

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FIGURE 15.4-8

UNCONTROLLED RCCA BANK WITHDRAWL
FROM 100% POWER
MINIMUM DNBR VS REACTIVITY INSERTION RATE

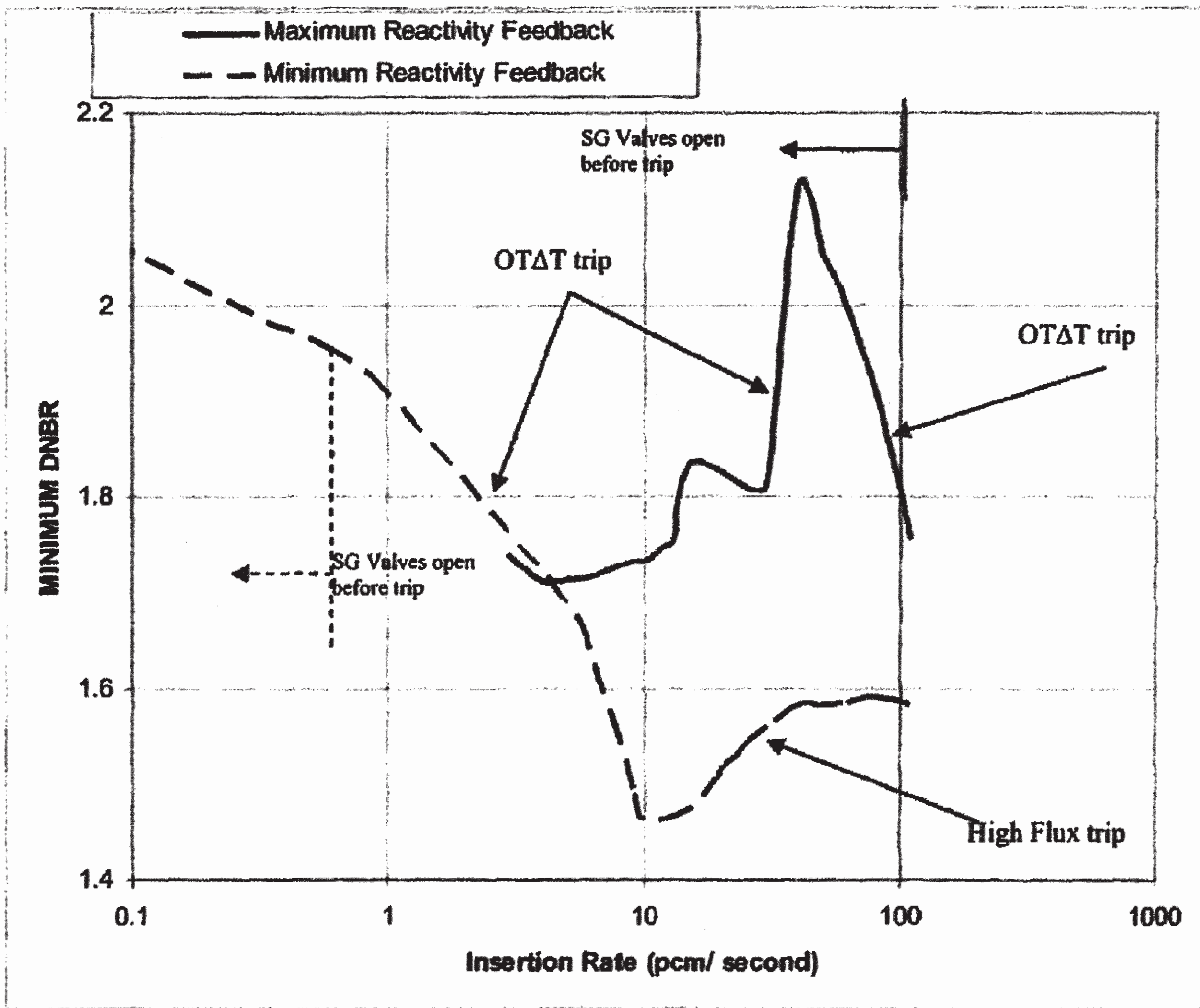

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FIGURE 15.4-9 UNCONTROLLED RCCA BANK WITHDRAWAL FROM 60% POWER MINIMUM DNBR VS REACTIVITY INSERTION RATE

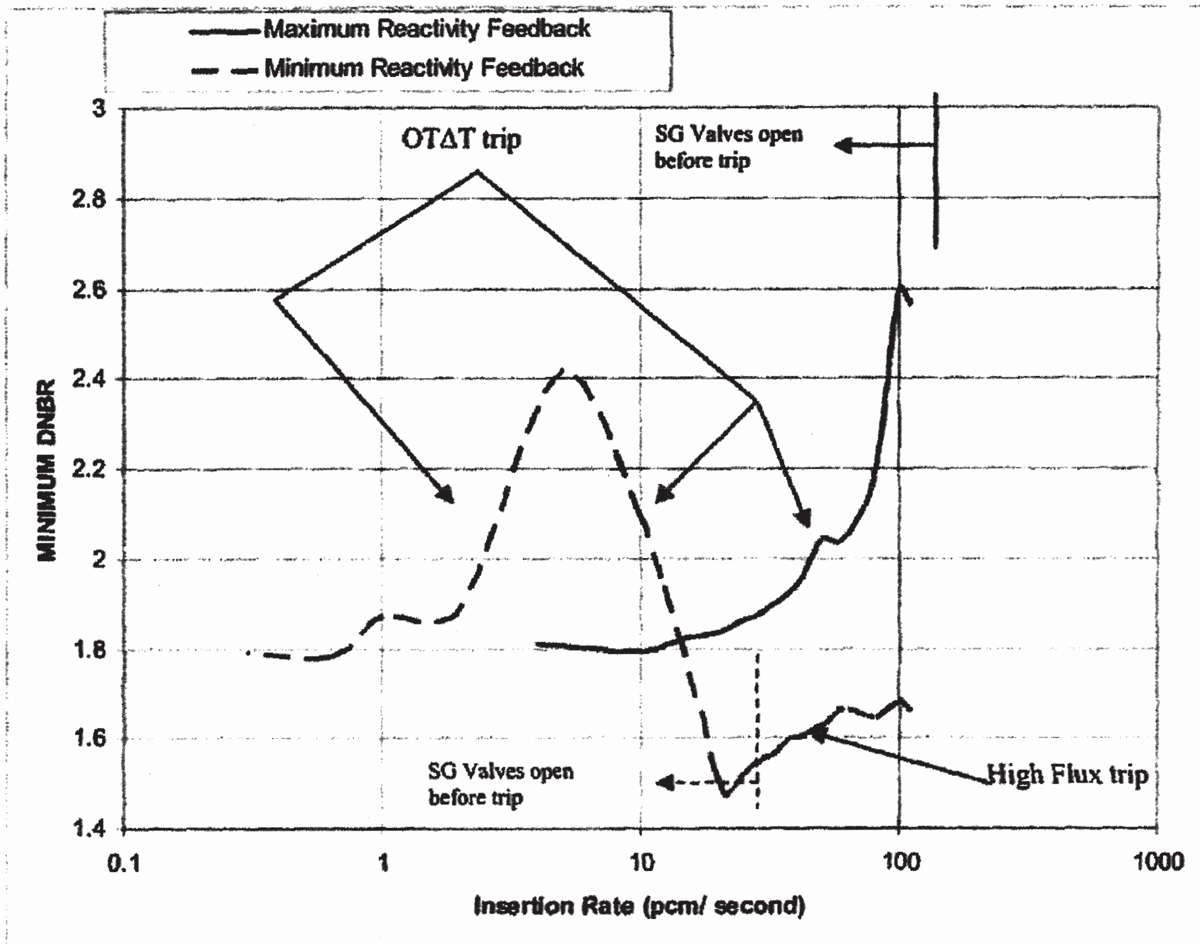

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FIGURE 15.4-10 UNCONTROLLED RCCA BANK WITHDRAWAL FROM 100% POWER MINIMUM DNBR VS REACTIVITY INSERTION RATE

Figures 15.4-11 and 15.4-12 have been deleted intentionally.

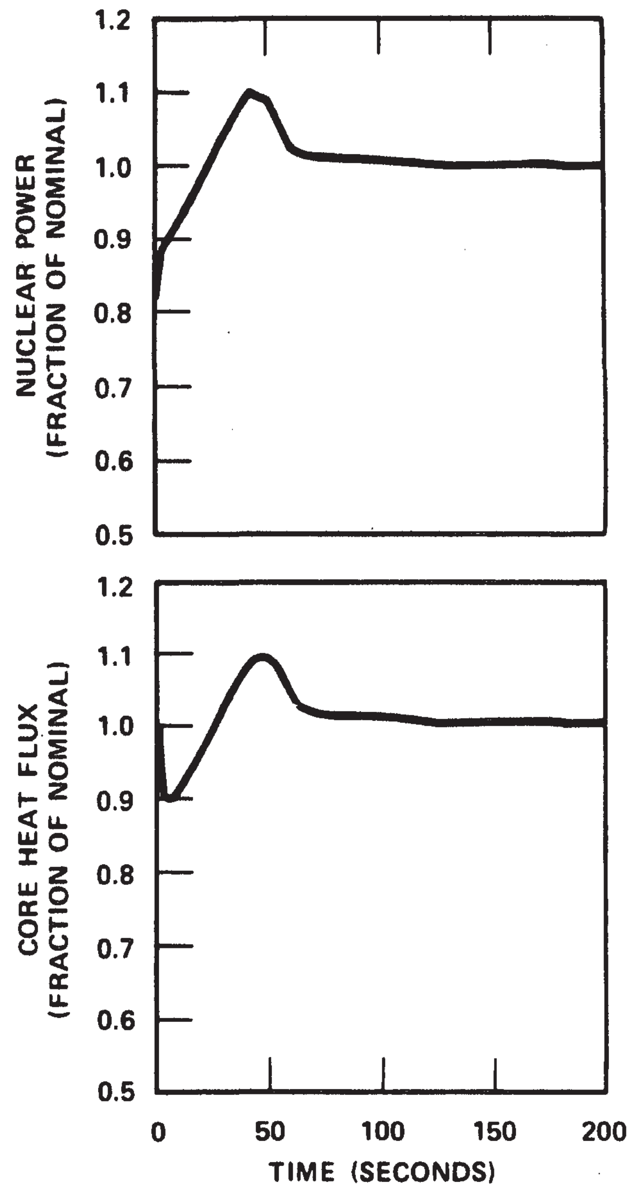

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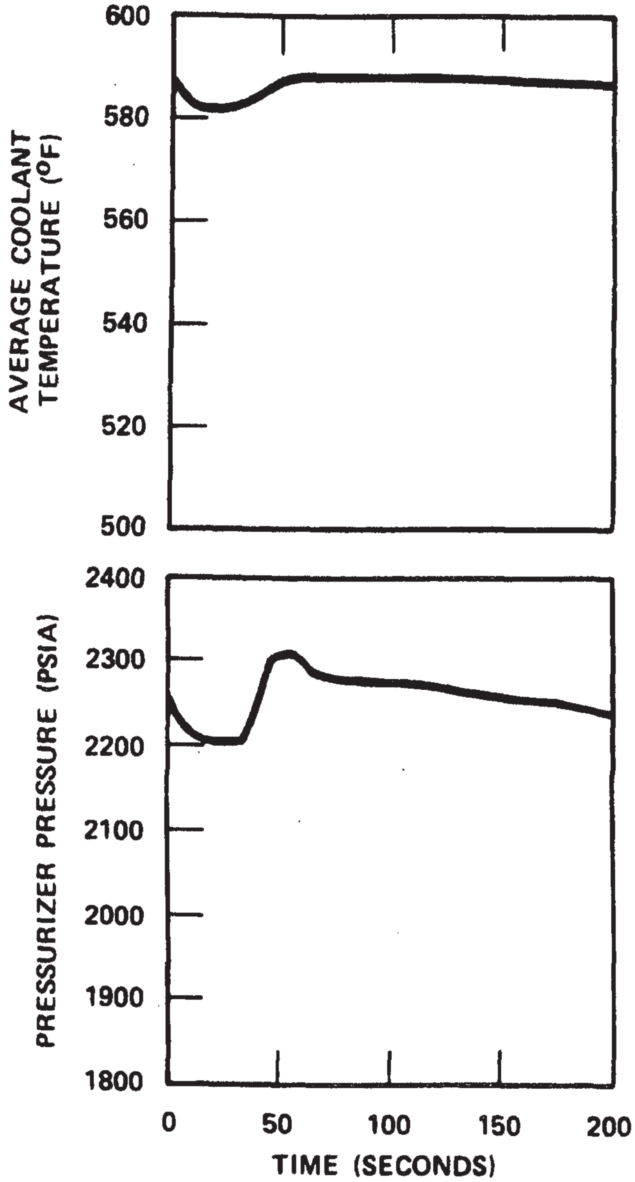
FIGURE 15.4-12a

TYPICAL TRANSIENT RESPONSE TO A DROPPED RCCA (OR RCCAS) IN AUTOMATIC CONTROL
(SHEET 1 OF 2)



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UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 15.4-12a

TYPICAL TRANSIENT RESPONSE TO A DROPPED RCCA (OR RCCAS) IN AUTOMATIC CONTROL (SHEET 2 OF 2)

Figures 15.4-13 thru 15.4-17 have been deleted intentionally.

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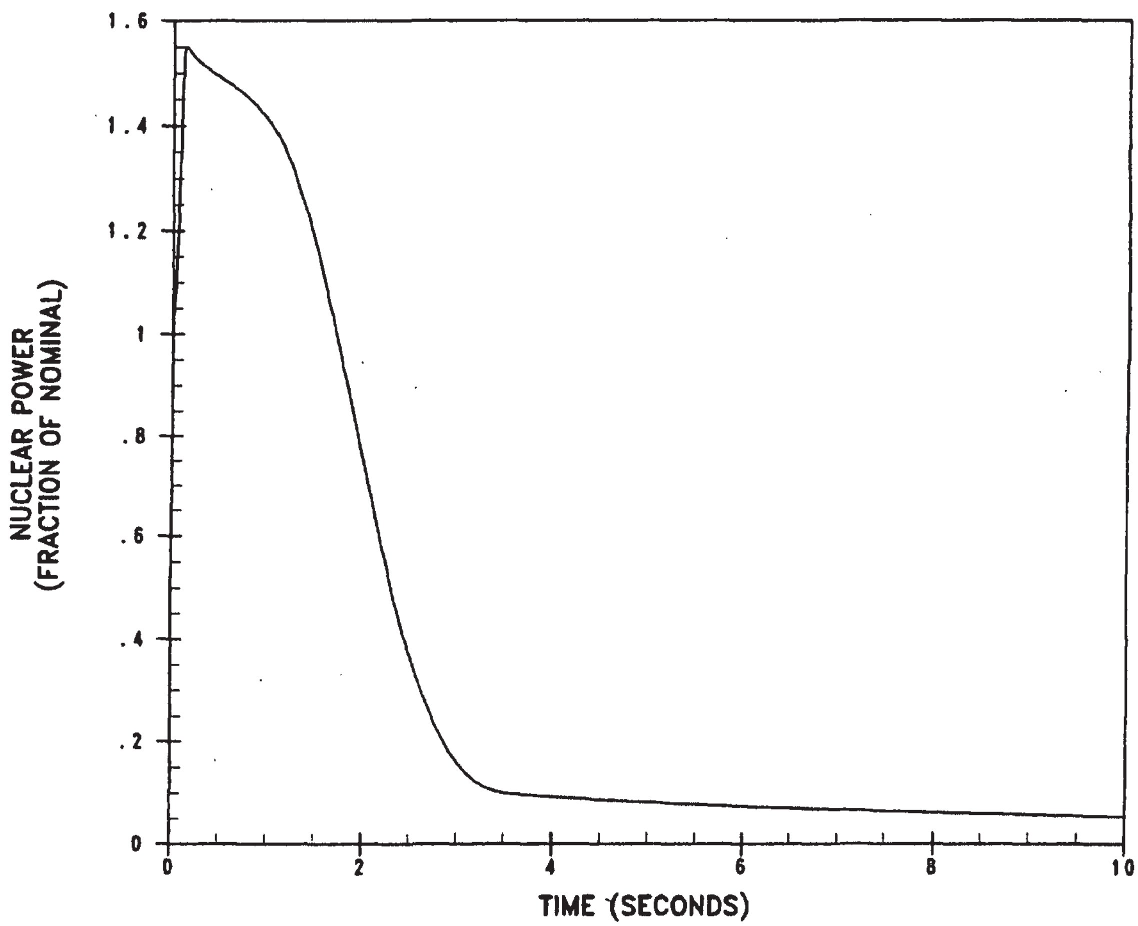

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FIGURE 15.4-18

NUCLEAR POWER TRANSIENT,
BOL HFP ROD EJECTION ACCIDENT

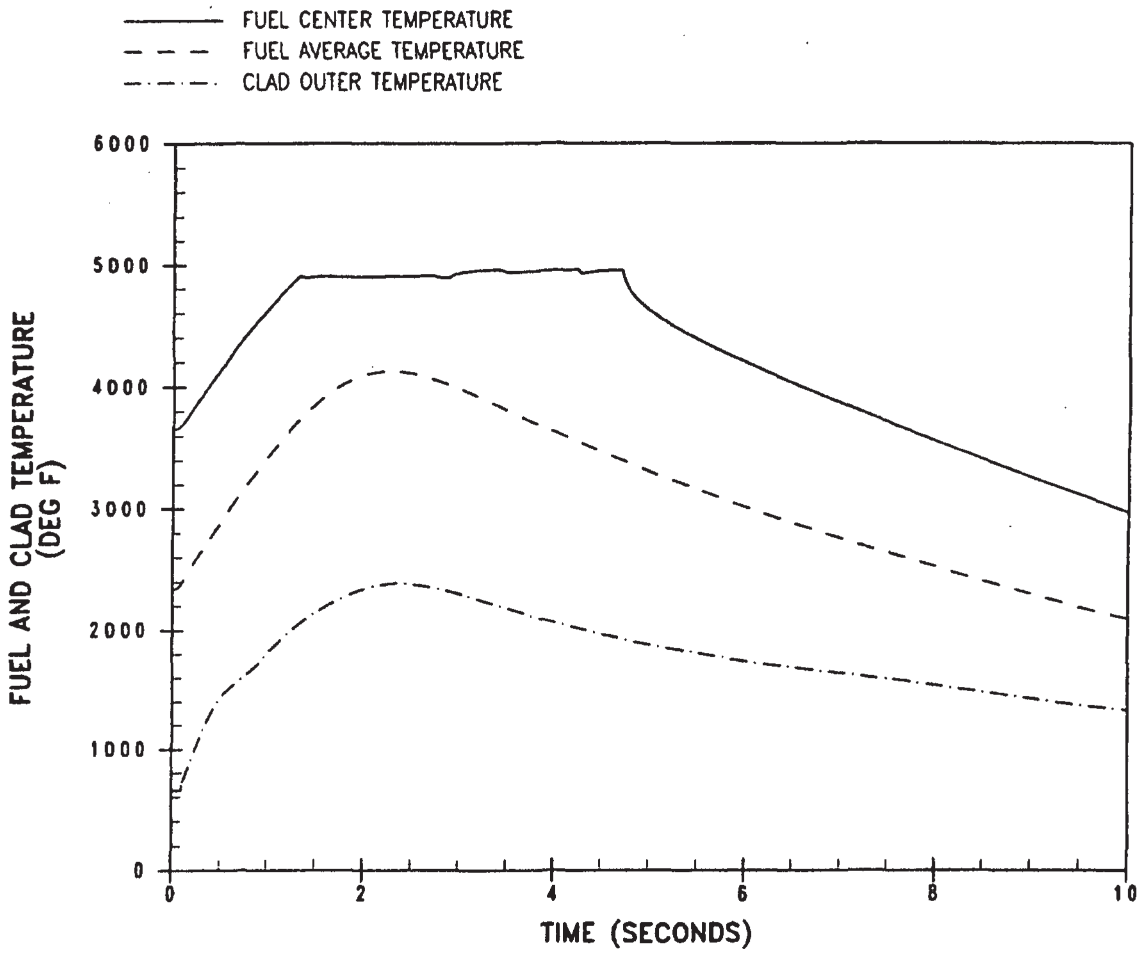

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FIGURE 15.4-19

HOT SPOT FUEL AND CLAD
TEMPERATURES VERSUS TIME
BOL HFP ROD EJECTION ACCIDENT

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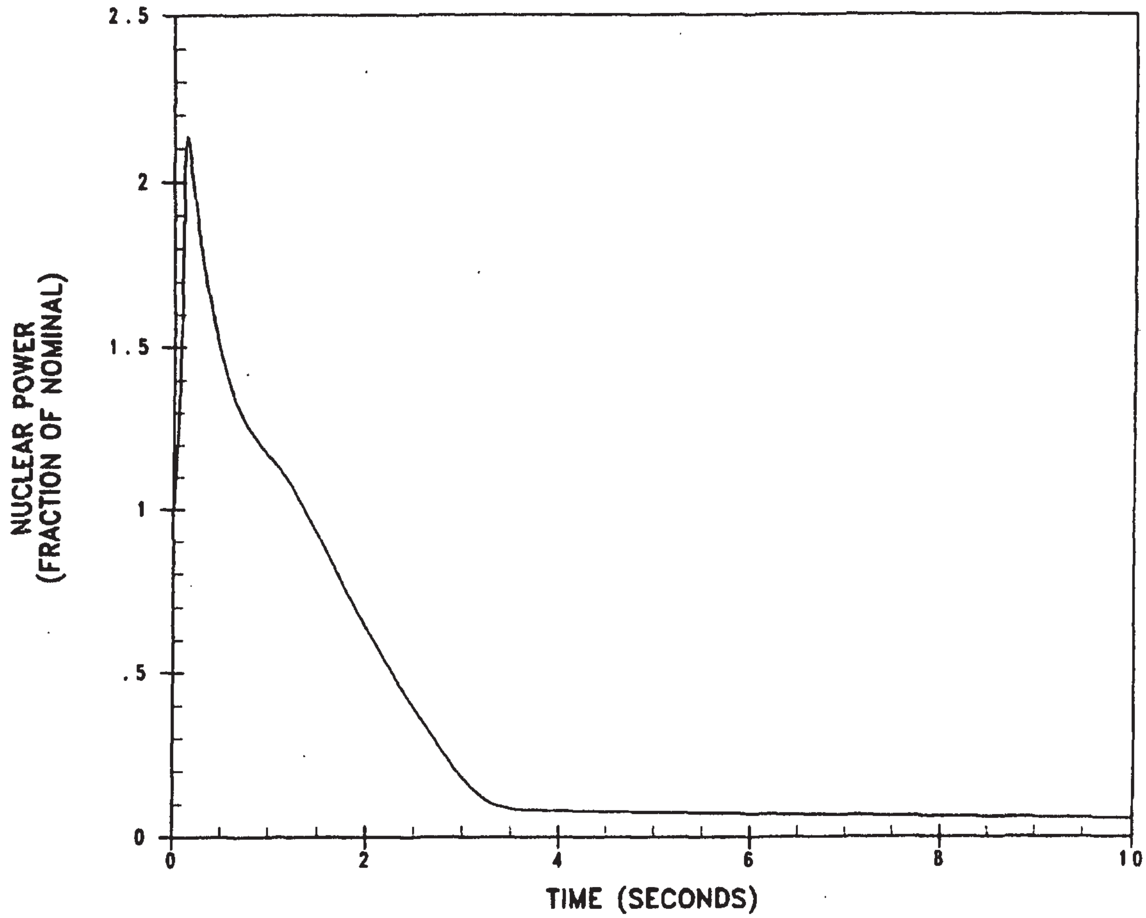

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FIGURE 15.4-20

NUCLEAR POWER TRANSIENT,
EOL HFP ROD EJECTION ACCIDENT

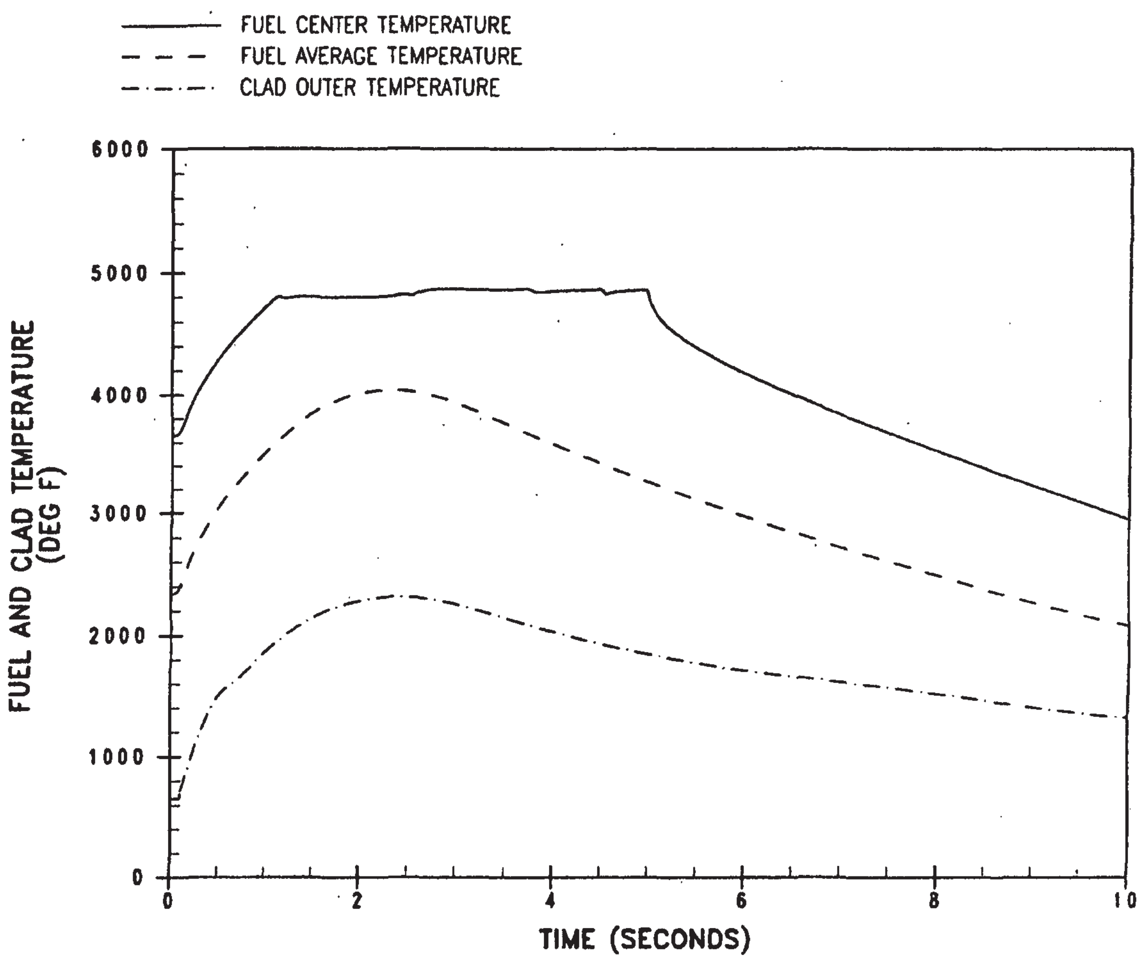

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FIGURE 15.4-21

HOT SPOT FUEL AND CLAD
TEMPERATURES VERSUS TIME
EOL HFP ROD EJECTION ACCIDENT

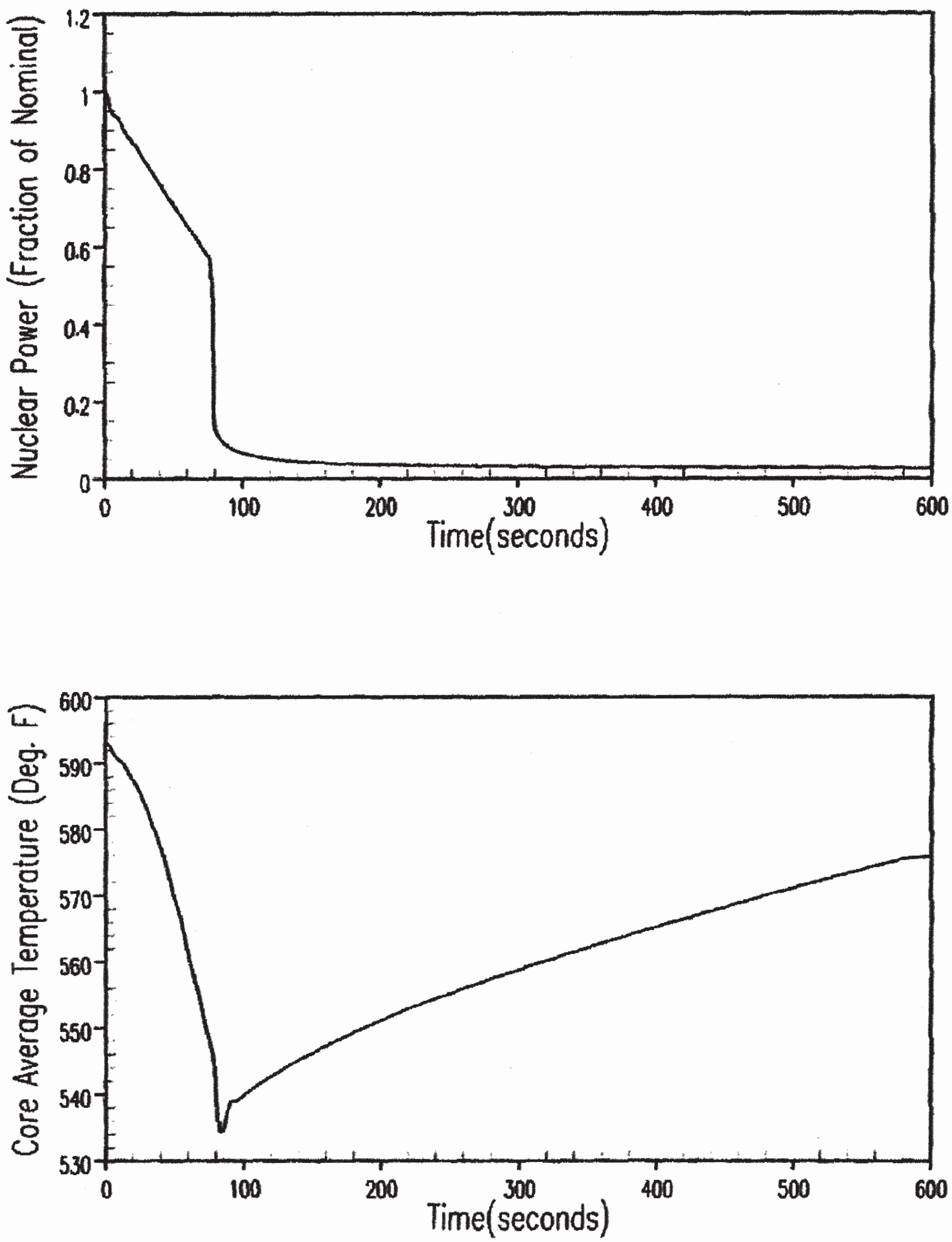

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FIGURE 15.5-1 NUCLEAR POWER TRANSIENT CORE AVERAGE WATER TEMPERATURE TRANSIENT FOR INADVERTENT ACTUATION OF ECCS DURING POWER OPERATION

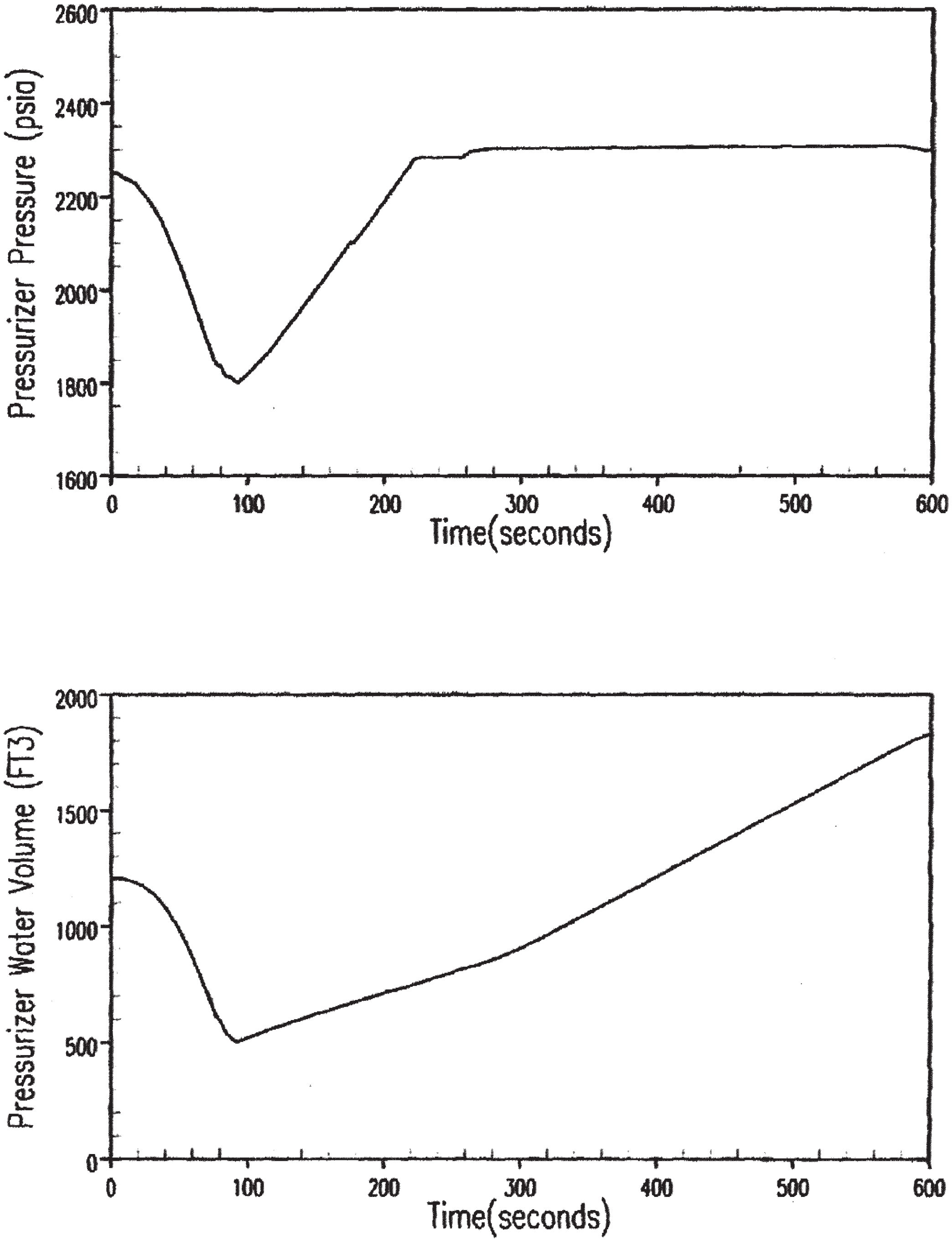

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FIGURE 15.5-2
PRESSURIZER PRESSURE TRANSIENT
PRESSURIZER WATER VOLUME TRANSIENT
FOR INADVERTENT ACTUATION OF ECCS
DURING POWER OPERATION

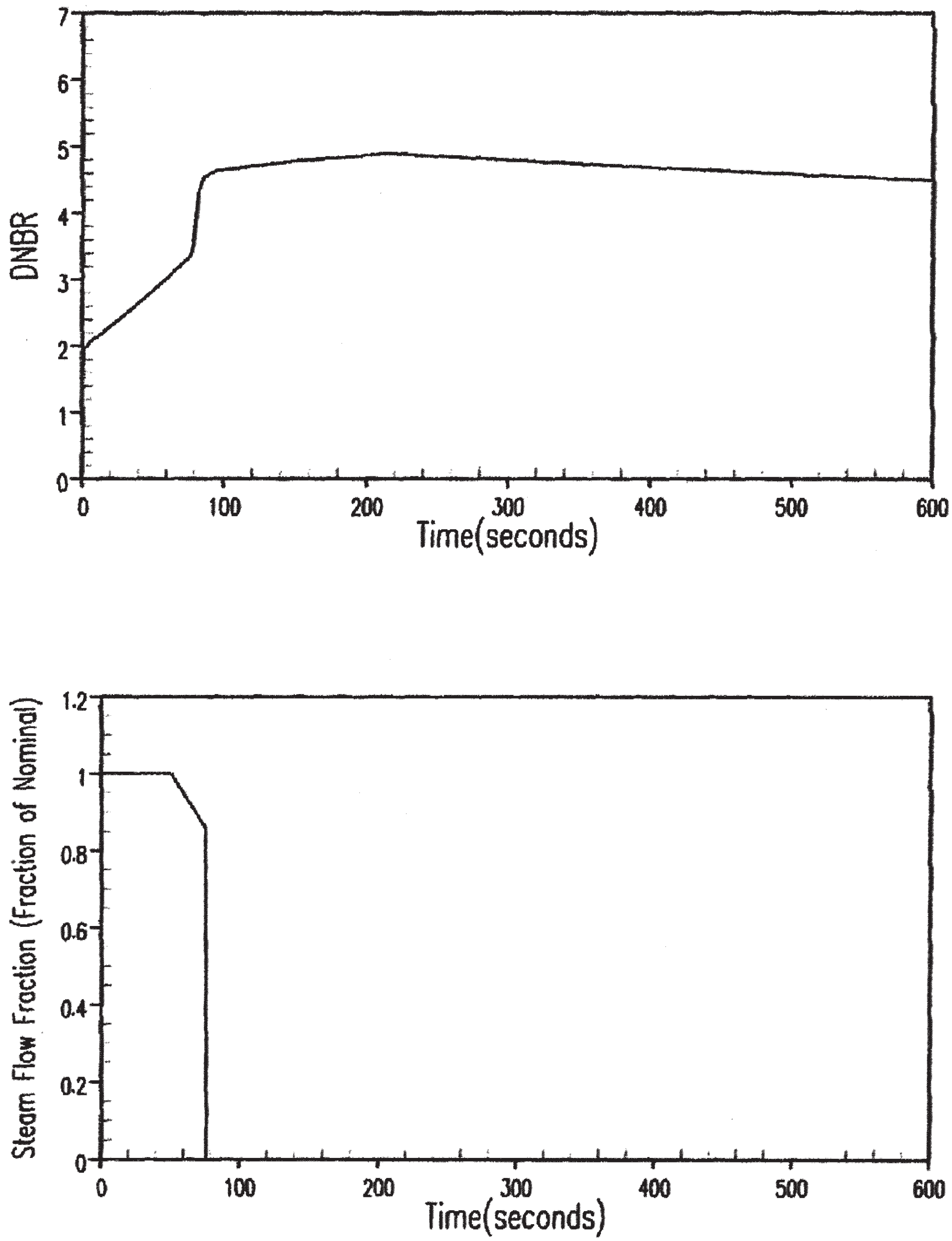

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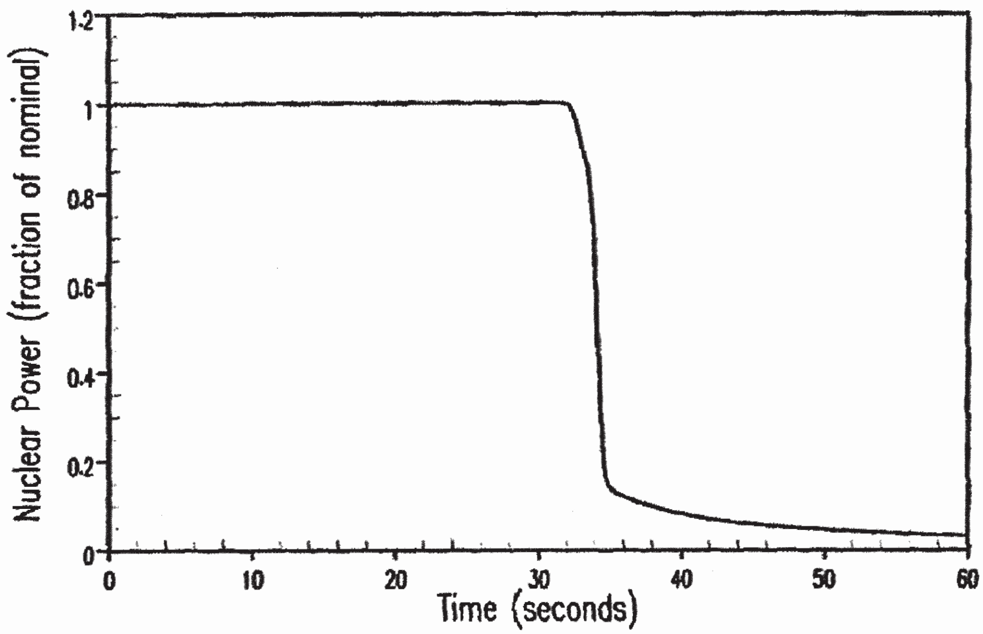
FIGURE 15.5-3

DNBR AND STEAM FLOW TRANSIENTS FOR INADVERTENT ACTUATION OF ECCS DURING POWER OPERATION



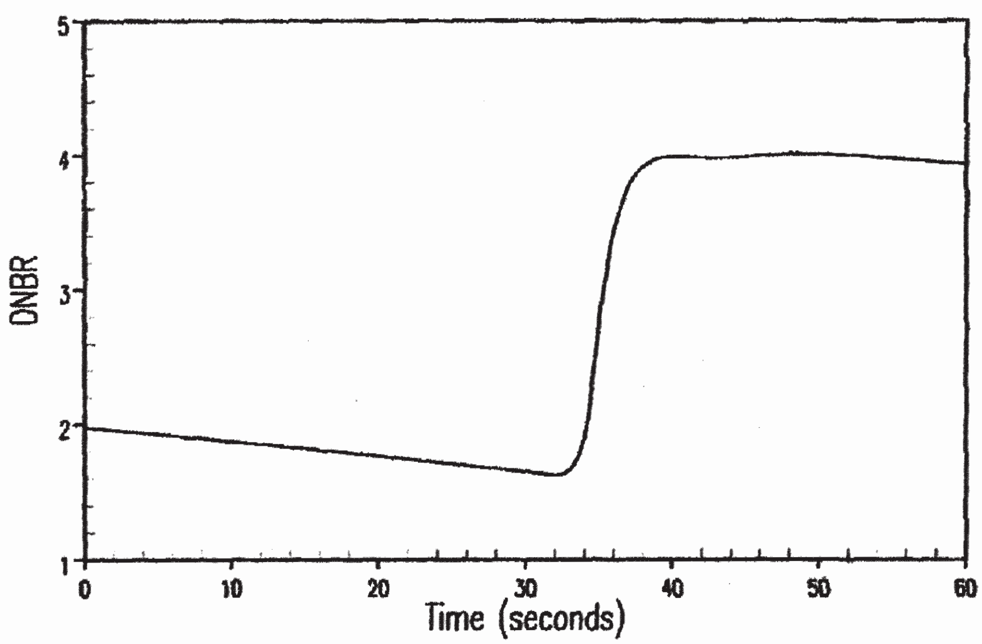


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FIGURE 15.6-1

INADVERTENT OPENING OF A
PRESSURIZER SAFETY VALVE

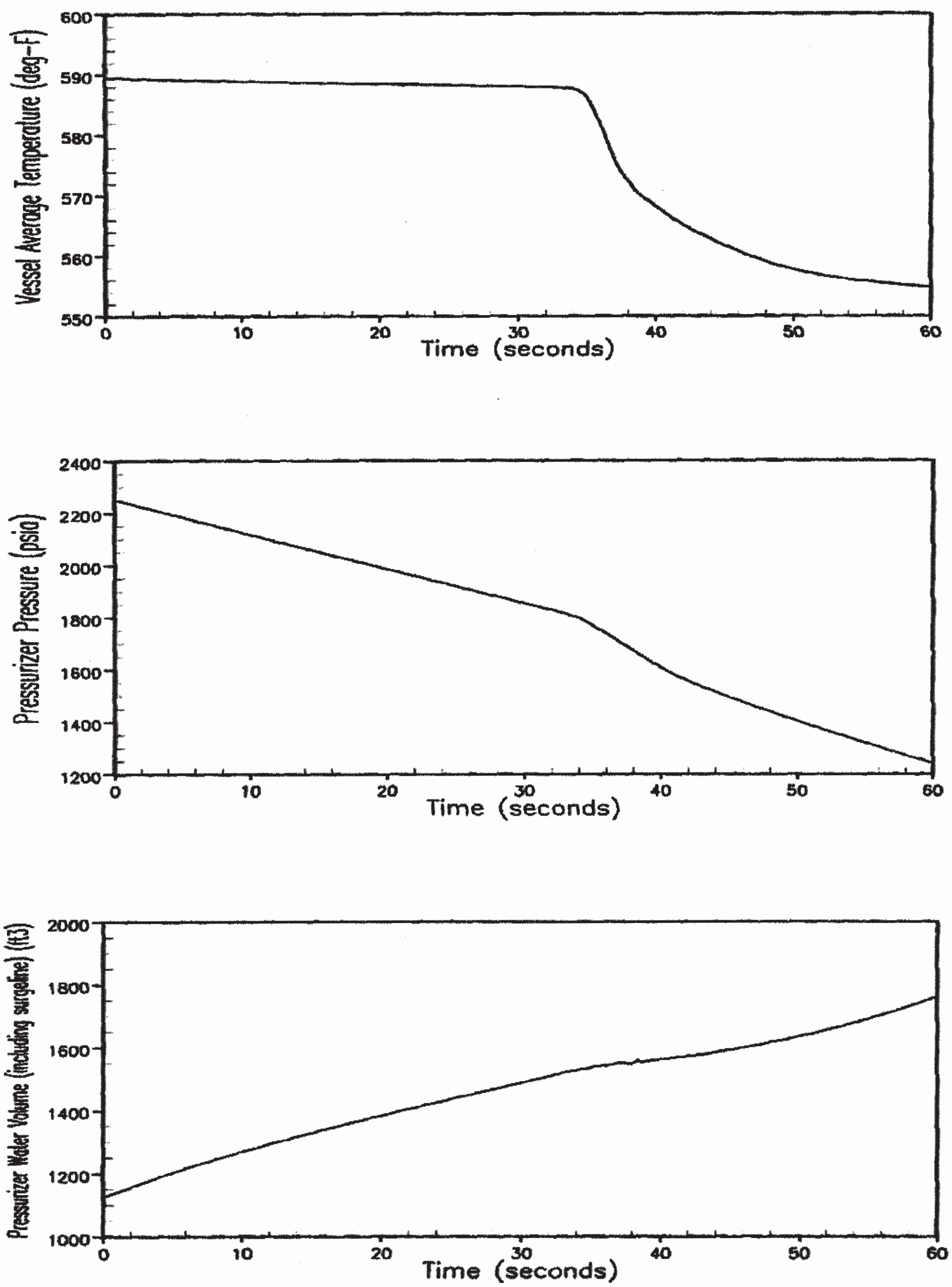

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FIGURE 15.6-2

INADVERTENT OPENING OF A
PRESSURIZER SAFETY VALVE

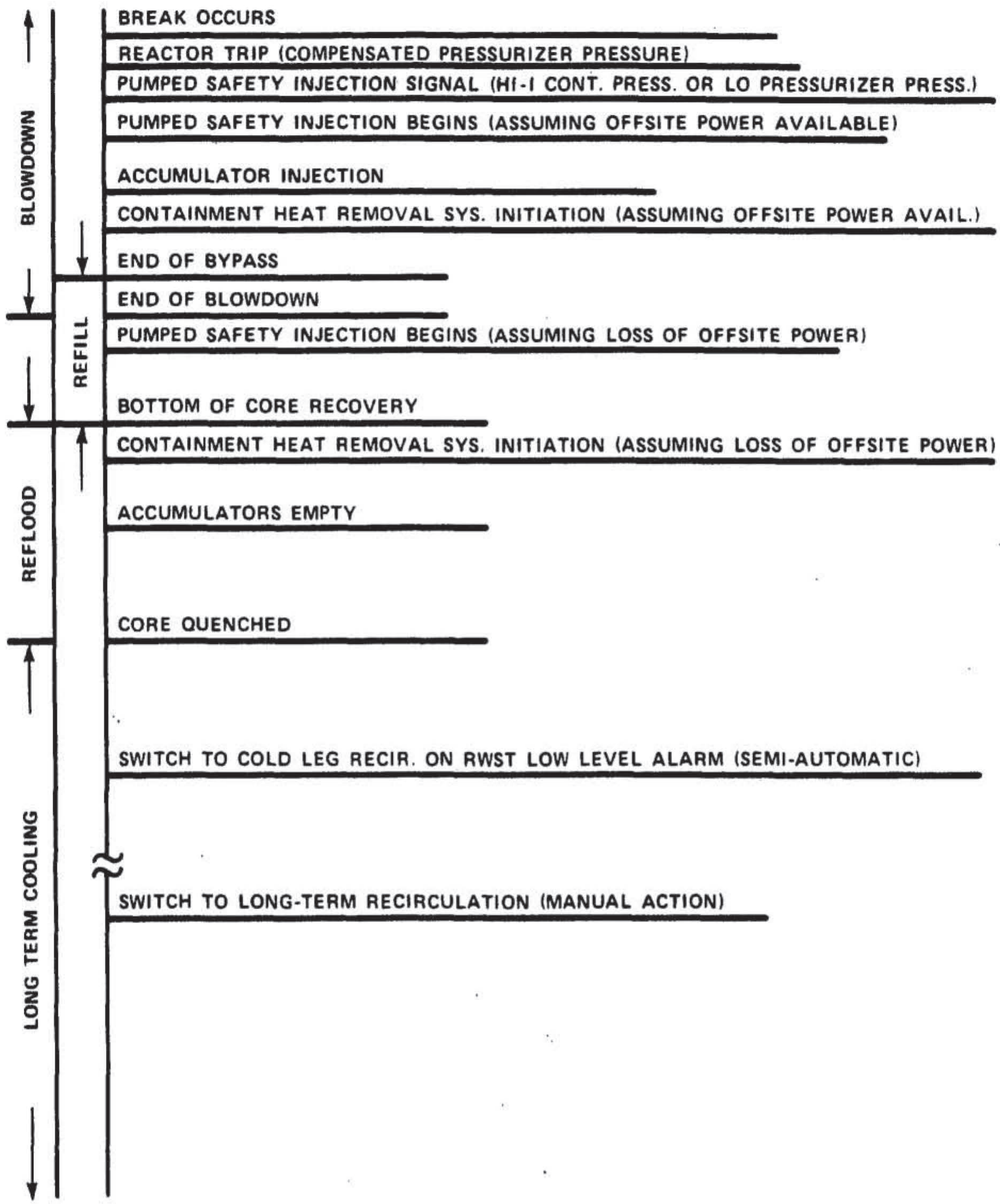

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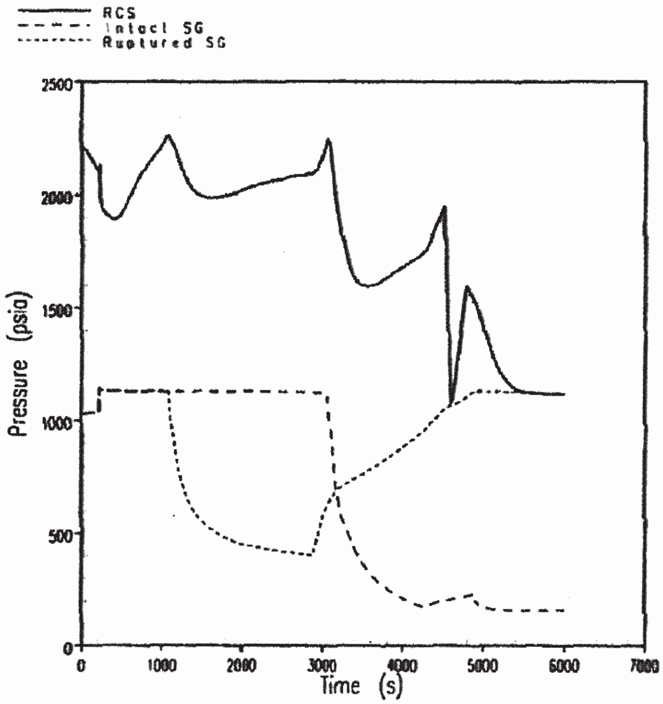
FIGURE 15.6-3

SEQUENCE OF EVENTS FOR
LARGE BREAK LOCA ANALYSIS



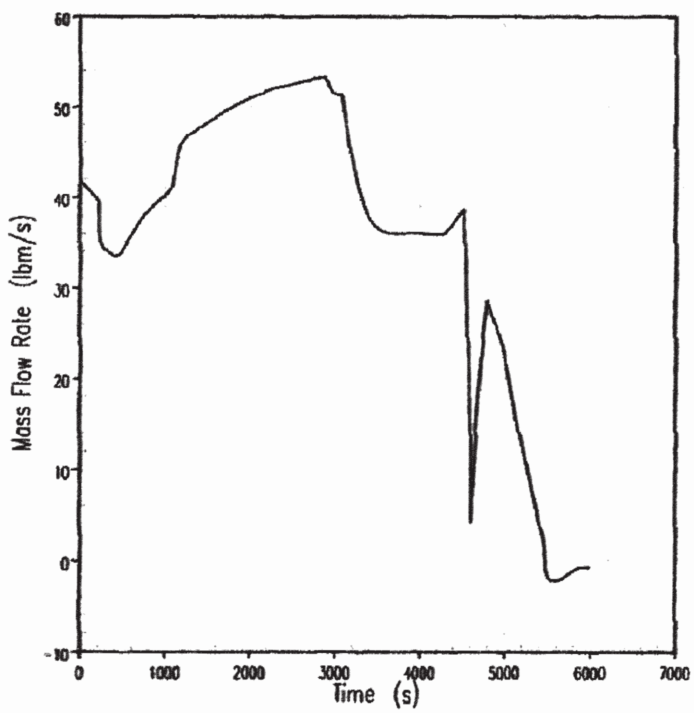


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FIGURE 15.6-3A

PRESSURIZER AND STEAM GENERATOR
PRESSURE AND RUPTURED TUBE FLOW RATE
SGTR RESULTS (OFFSITE DOSE CASE)-UNIT 1

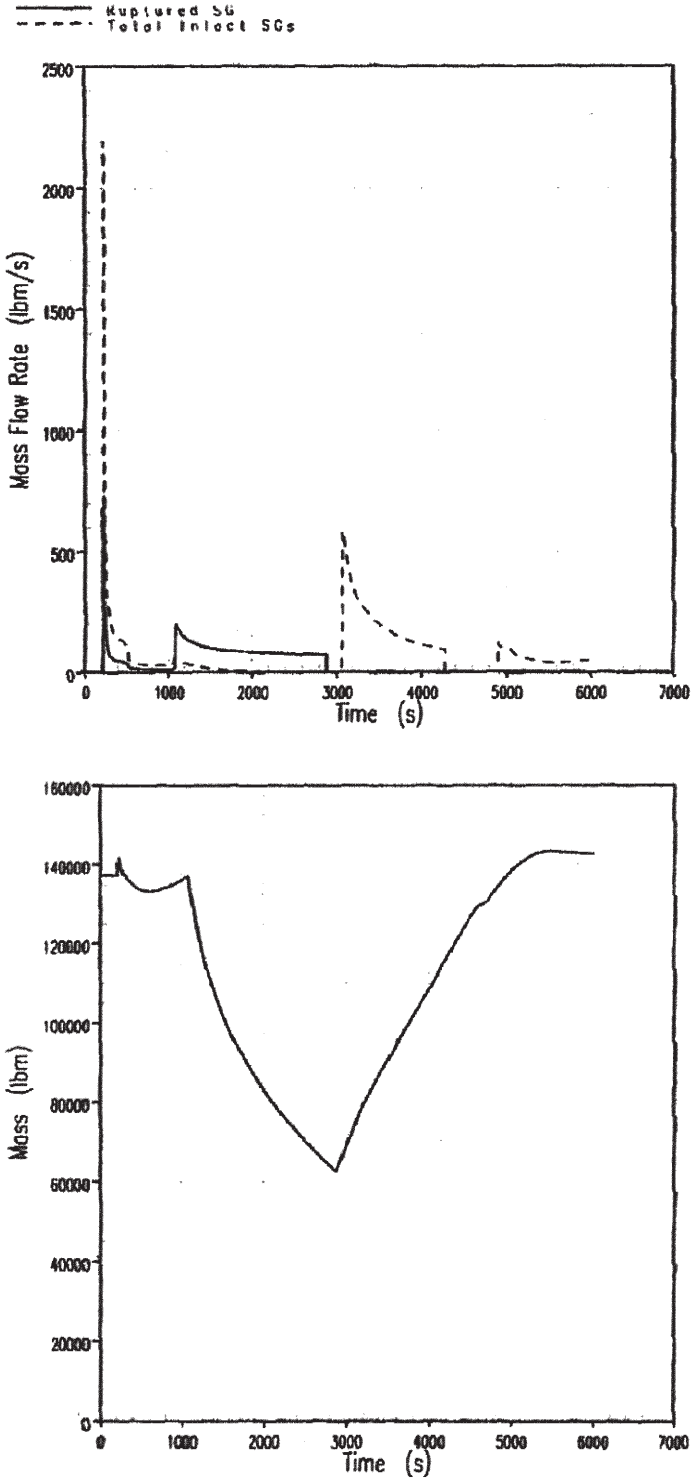

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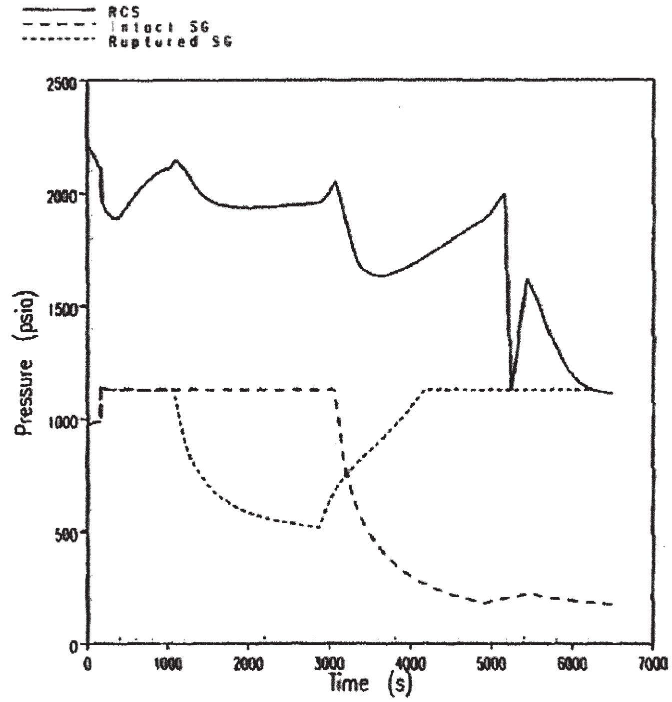
FIGURE 15.6-3B

STEAM GENERATOR RELEASE RATES AND
MASS RETAINED
SGTR RESULTS (OFFSITE DOSE CASE)-UNIT 1



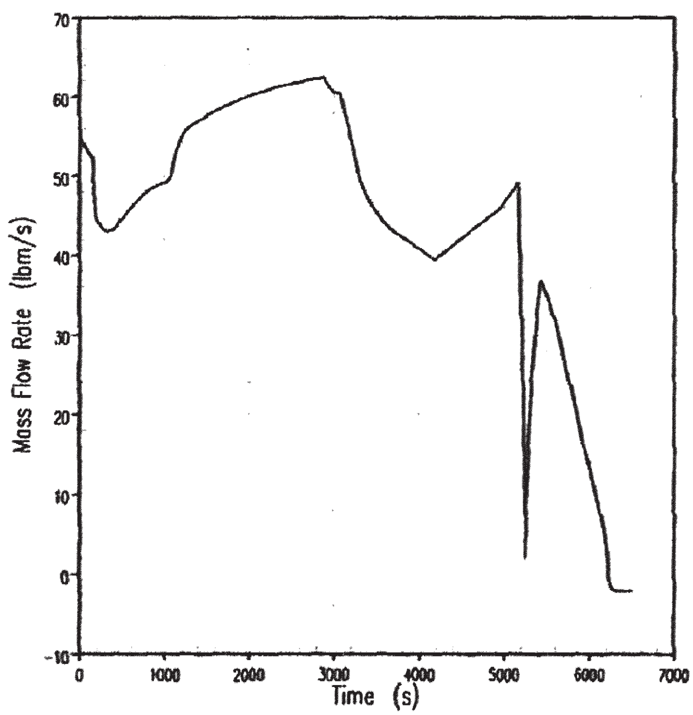


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FIGURE 15.6-3C

PRESSURIZER AND STEAM GENERATOR
PRESSURE AND RUPTURED TUBE FLOW RATE
SGTR RESULTS (OFFSITE DOSE CASE)-UNIT 2

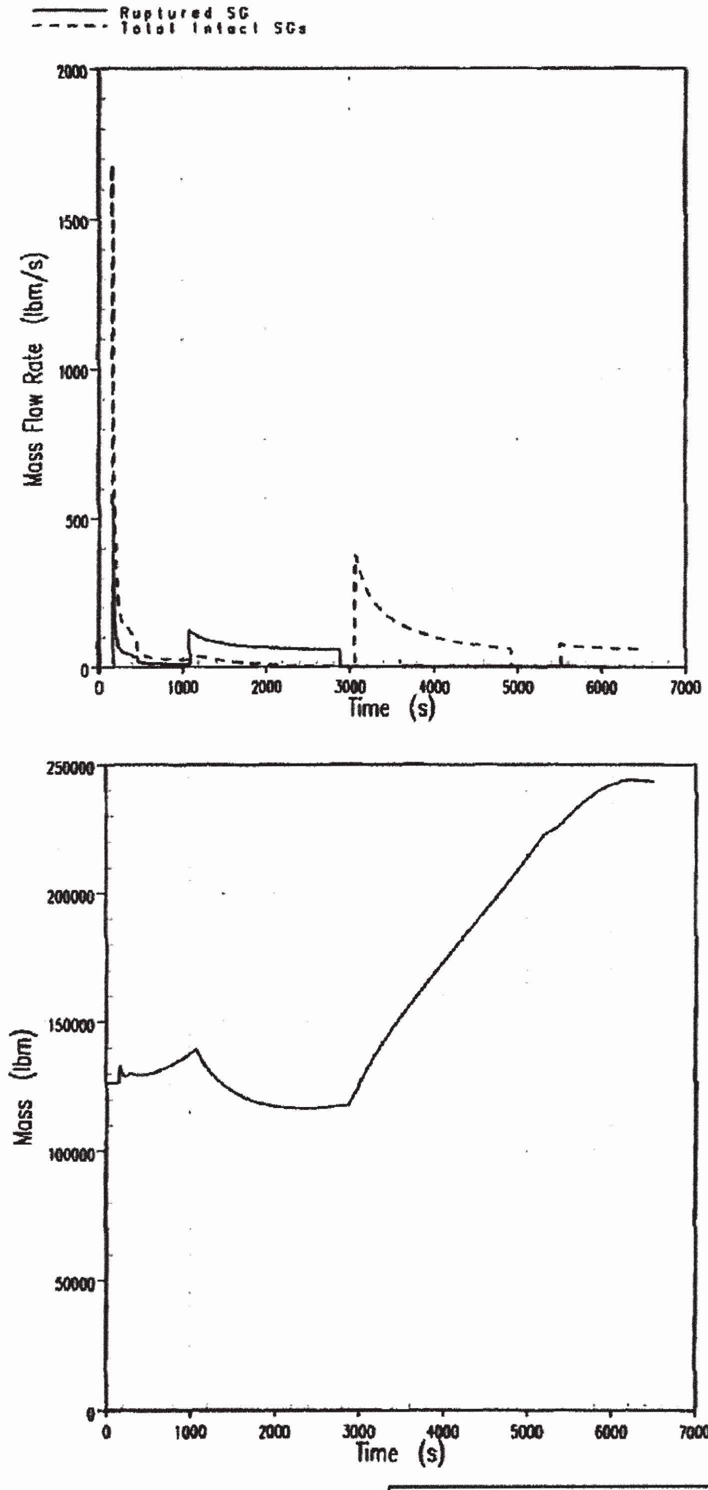

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FIGURE 15.6-3D

STEAM GENERATOR RELEASE RATES AND MASS RETAINED
SGTR RESULTS (OFFSITE DOSE CASE)-UNIT 2

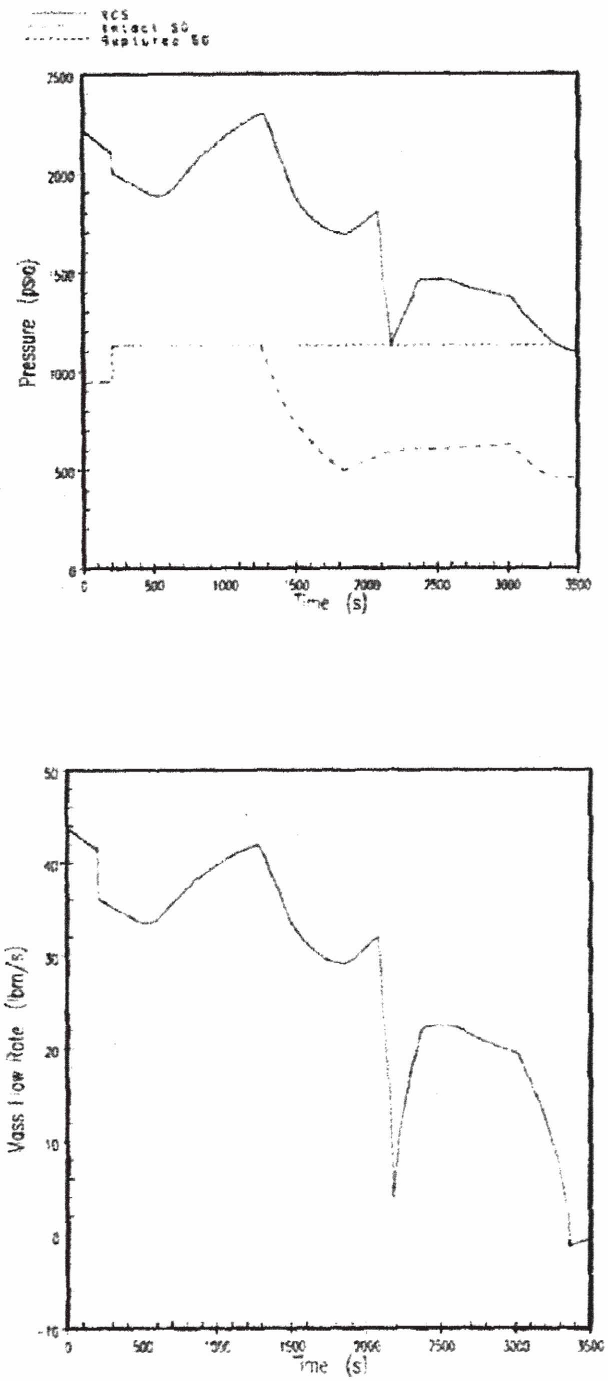

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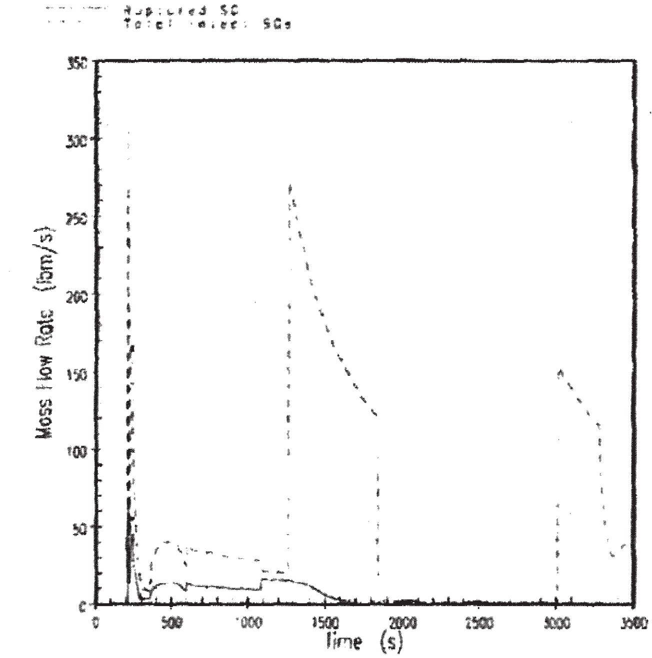
FIGURE 15.6-3E

PRESSURIZER AND STEAM GENERATOR PRESSURE AND RUPTURED TUBE FLOW RATE
SGTR RESULTS (MARGIN TO OVERFILL CASE)-UNIT 1



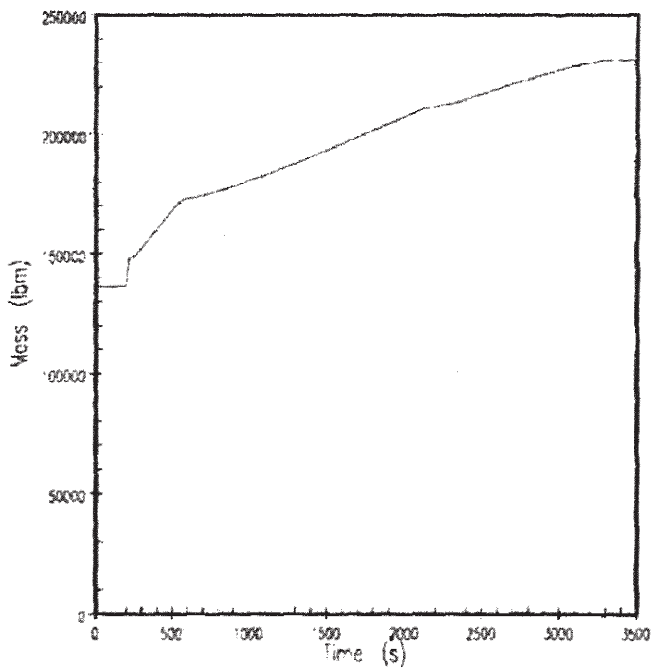


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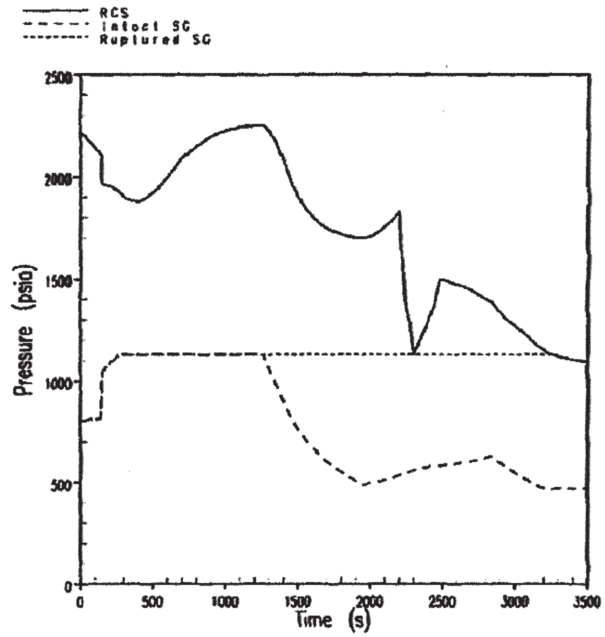
FIGURE 15.6-3F

STEAM GENERATOR RELEASE RATES
AND MASS RETAINED
SGTR RESULTS (MARGIN TO OVERFILL CASE)-UNIT 1



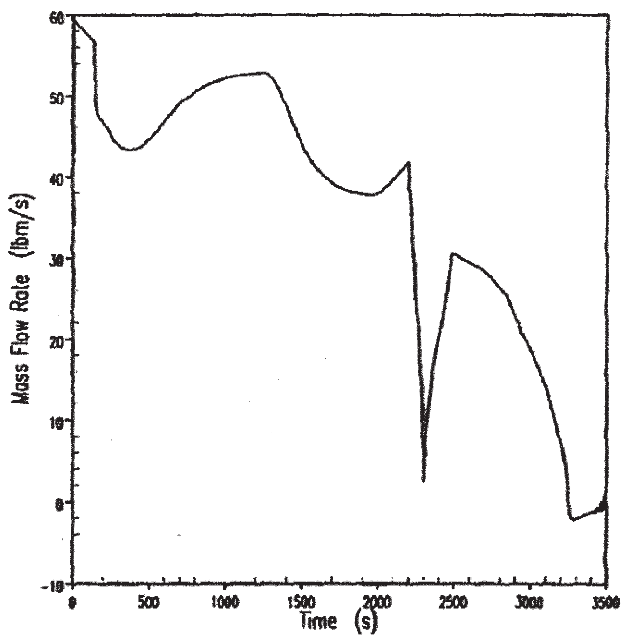


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FIGURE 15.6-3G

PRESSURIZER AND STEAM GENERATOR PRESSURE AND RUPTURED TUBE FLOW RATE
SGTR RESULTS (MARGIN TO OVERFILL CASE)-UNIT 2

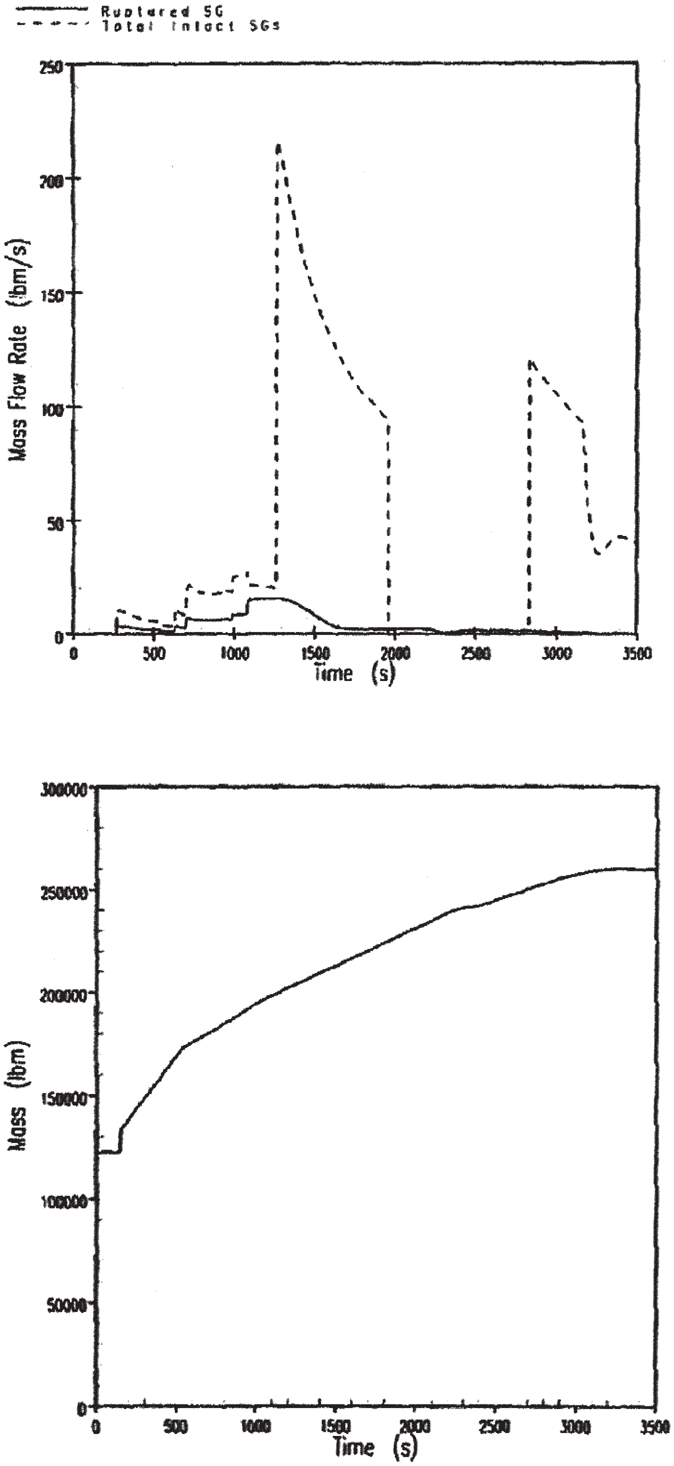

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FIGURE 15.6-3H

STEAM GENERATOR RELEASE RATES
AND MASS RETAINED
SGTR RESULTS (MARGIN TO OVERFILL CASE)-UNIT 2

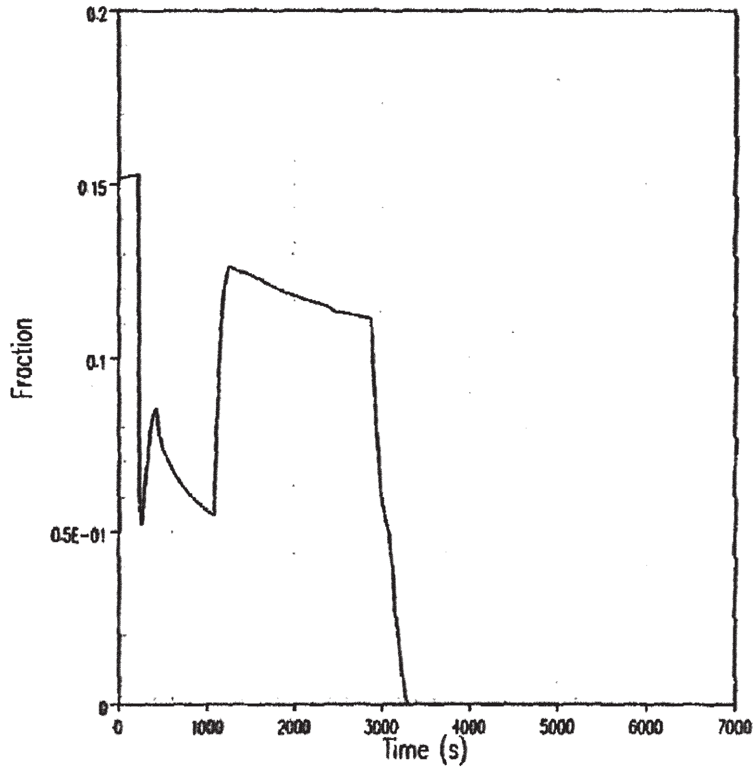

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FIGURE 15.6-3I

BREAK FLOW
FLASHING FRACTION
SGTR RESULTS (OFFSITE DOSE CASE) - UNIT 1

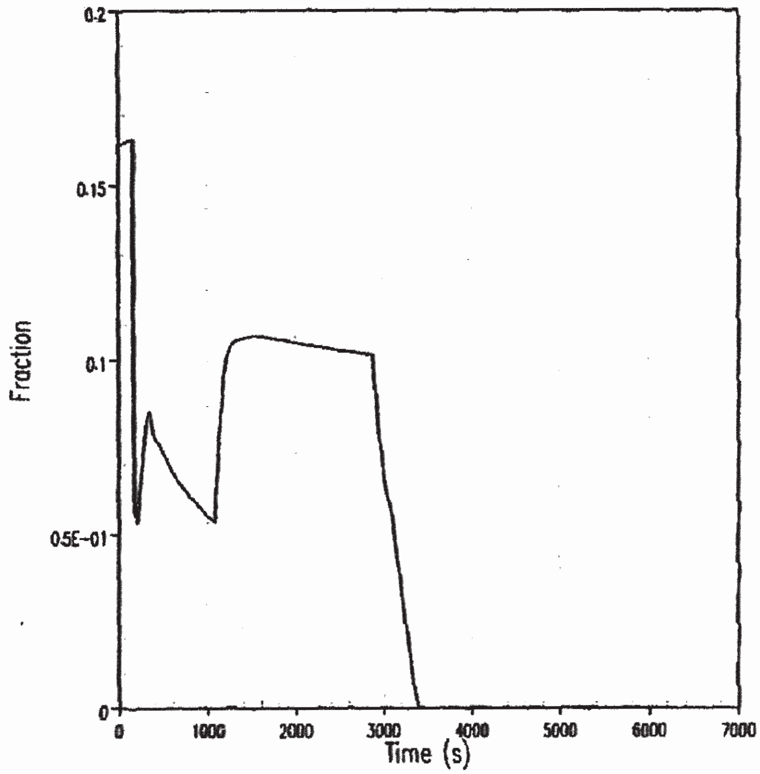

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FIGURE 15.6-3J

BREAK FLOW
FLASHING FRACTION
SGTR RESULTS (OFFSITE DOSE CASE) - UNIT 2

Figure 15.6-4 has been deleted intentionally.

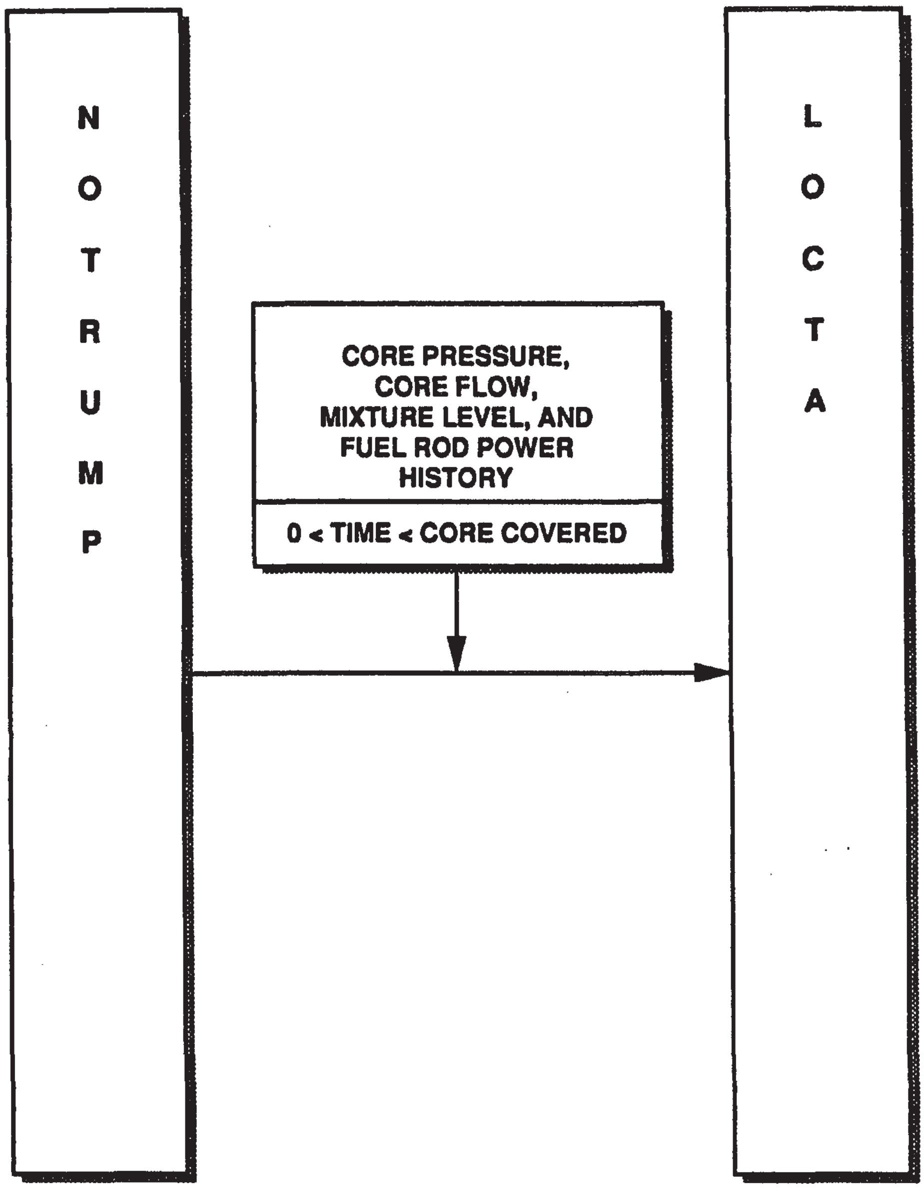

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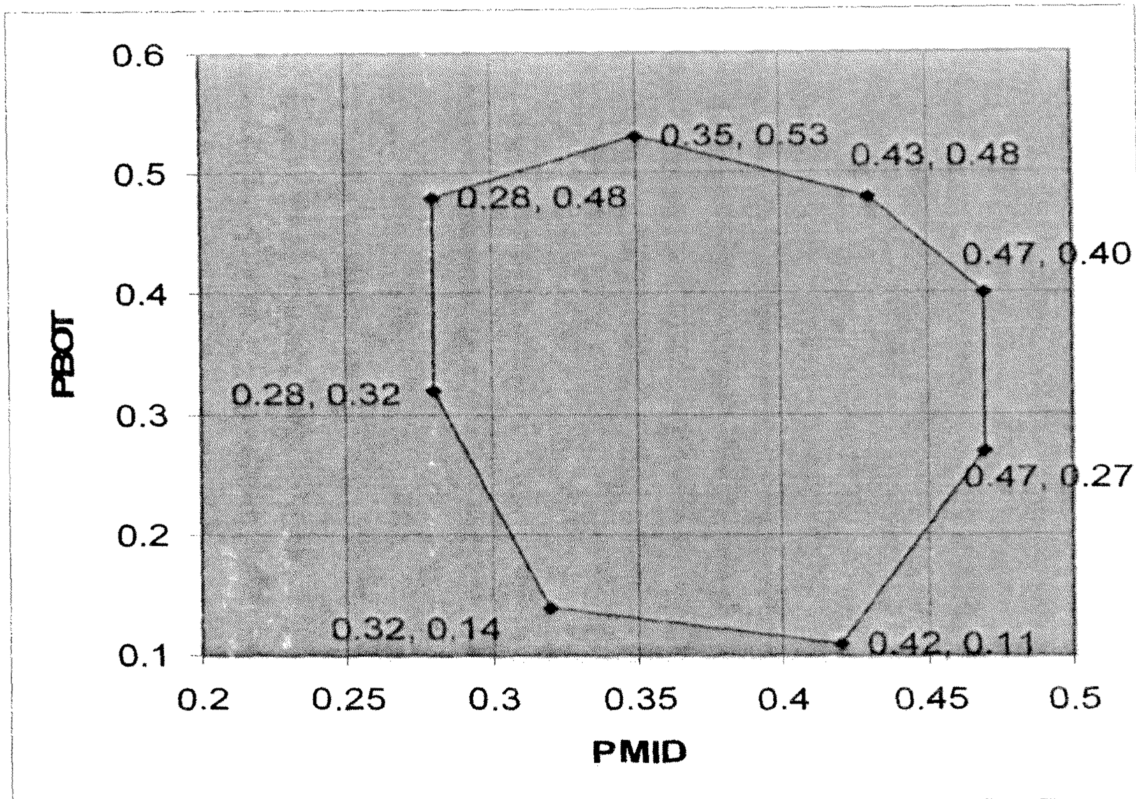
FIGURE 15.6-5

CODE INTERFACE DESCRIPTION
FOR SMALL BREAK MODEL

Figure 15.6-6 has been deleted intentionally.

Figure 15.6-6a has been deleted intentionally.



PBOT = integrated power fraction in the bottom third of the core

PMID = integrated power fraction in the middle third of the core

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FIGURE 15.6-7 Pbot/Pmid OPERATING LIMITS

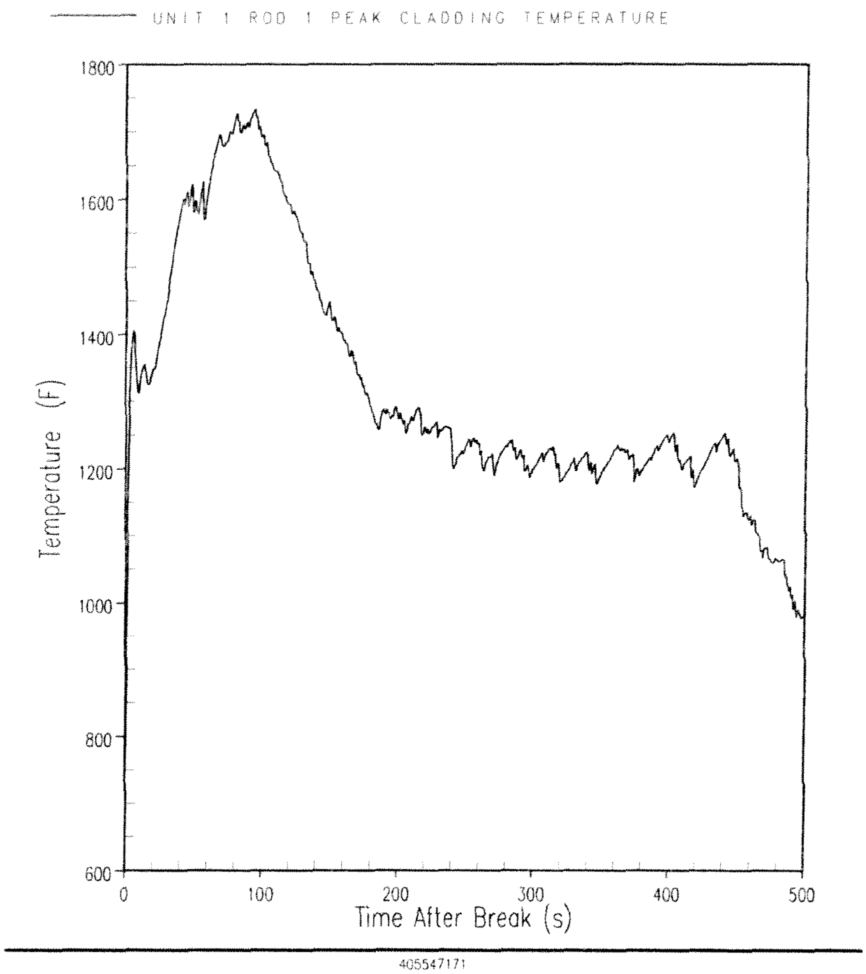

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FIGURE 15.6-8a

HOT ROD PEAK CLADDING TEMPERATURE

FOR ECCS LB LOCA PCT LIMITING CASE-UNIT 1

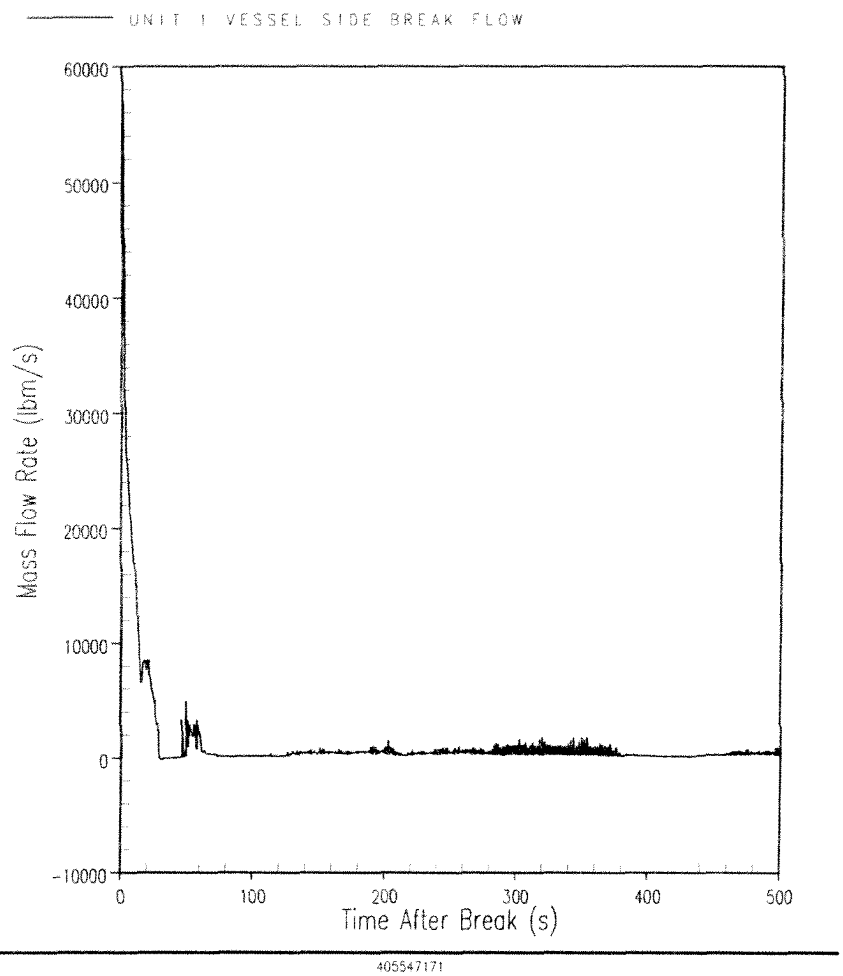

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FIGURE 15.6-8b

BREAK FLOW ON VESSEL SIDE OF BROKEN COLD LEG FOR ECCS LB LOCA PCT LIMITING CASE-UNIT 1

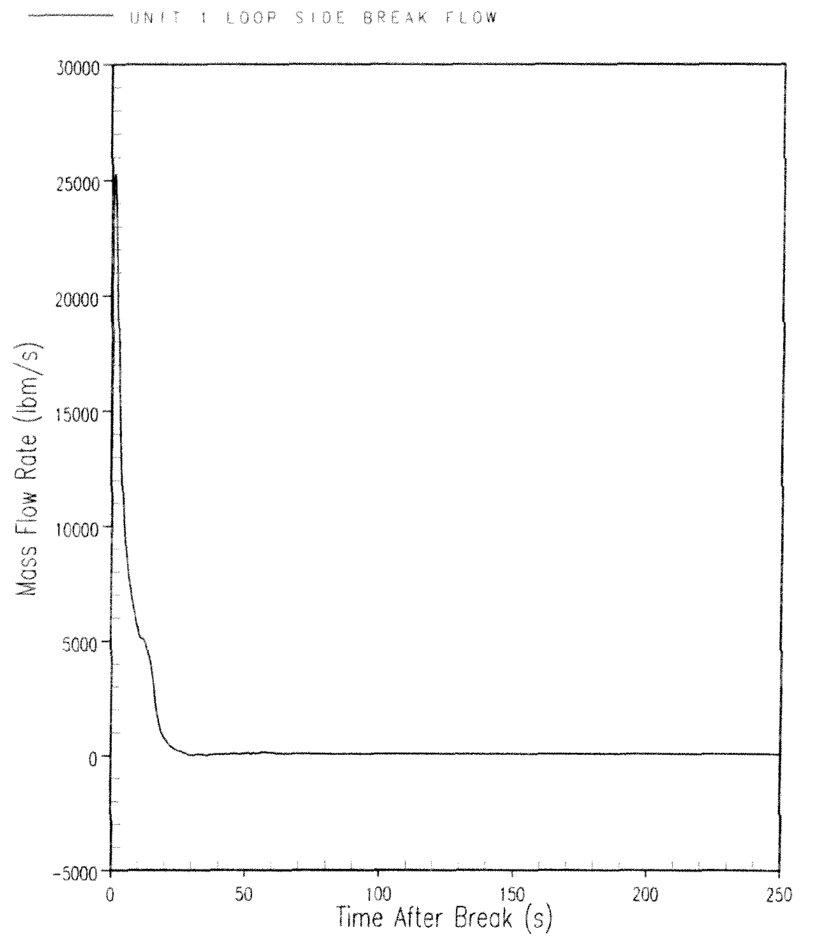

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FIGURE 15.6-8c

BREAK FLOW ON LOOP SIDE OF
BROKEN COLD LEG
FOR ECCS LB LOCA PCT LIMITING CASE-UNIT 1

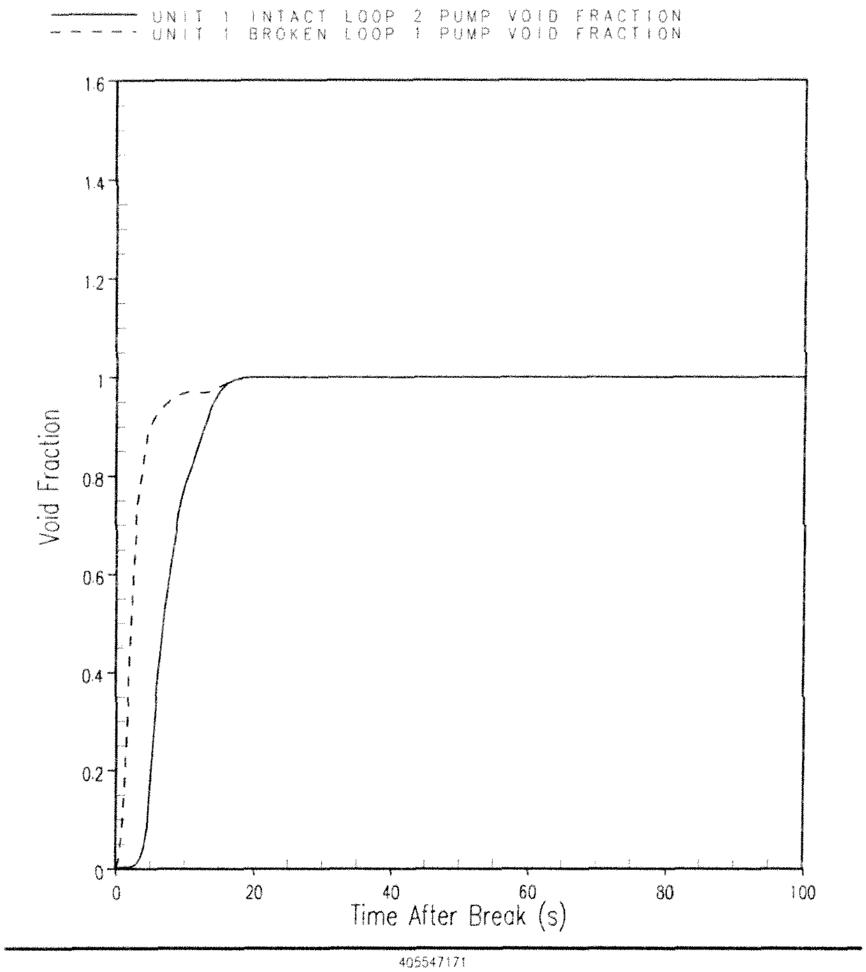

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FIGURE 15.6-8d

VOID FRACTION AT THE INTACT AND BROKEN LOOP PUMP INLET FOR ECCS LB LOCA PCT LIMITING CASE-UNIT 1

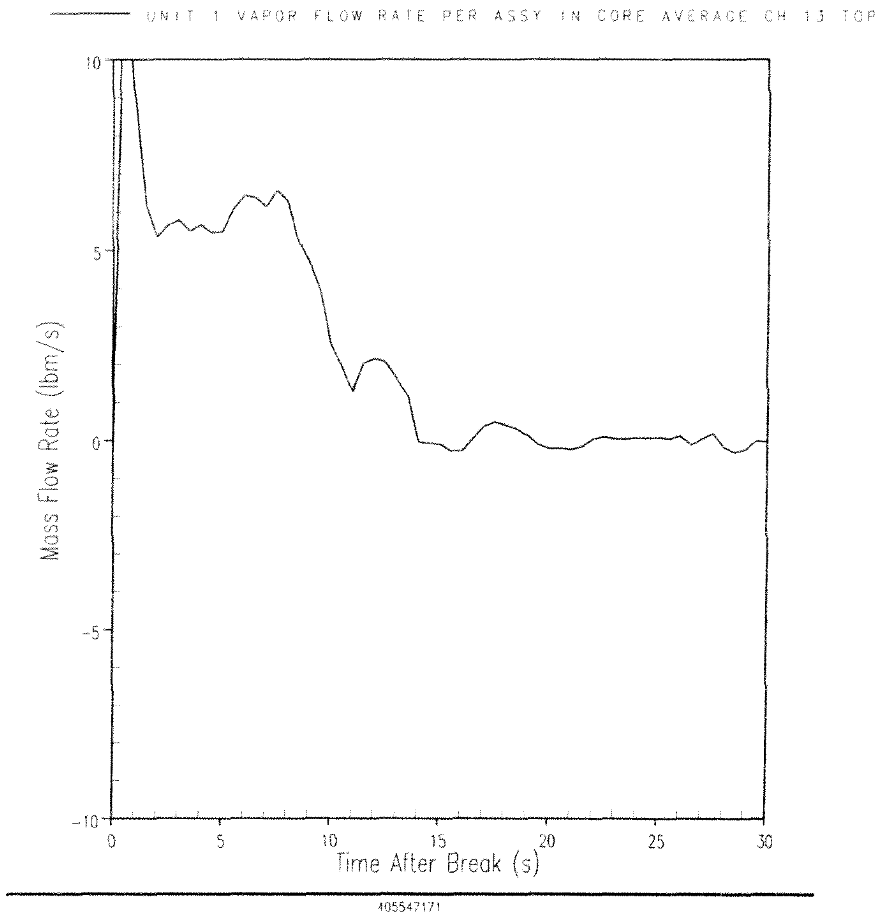

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FIGURE 15.6-8e

VAPOR FLOW RATE AT TOP OF CORE AVERAGE CHANNEL DURING BLOWDOWN FOR ECCS LB LOCA PCT LIMITING CASE-UNIT 1

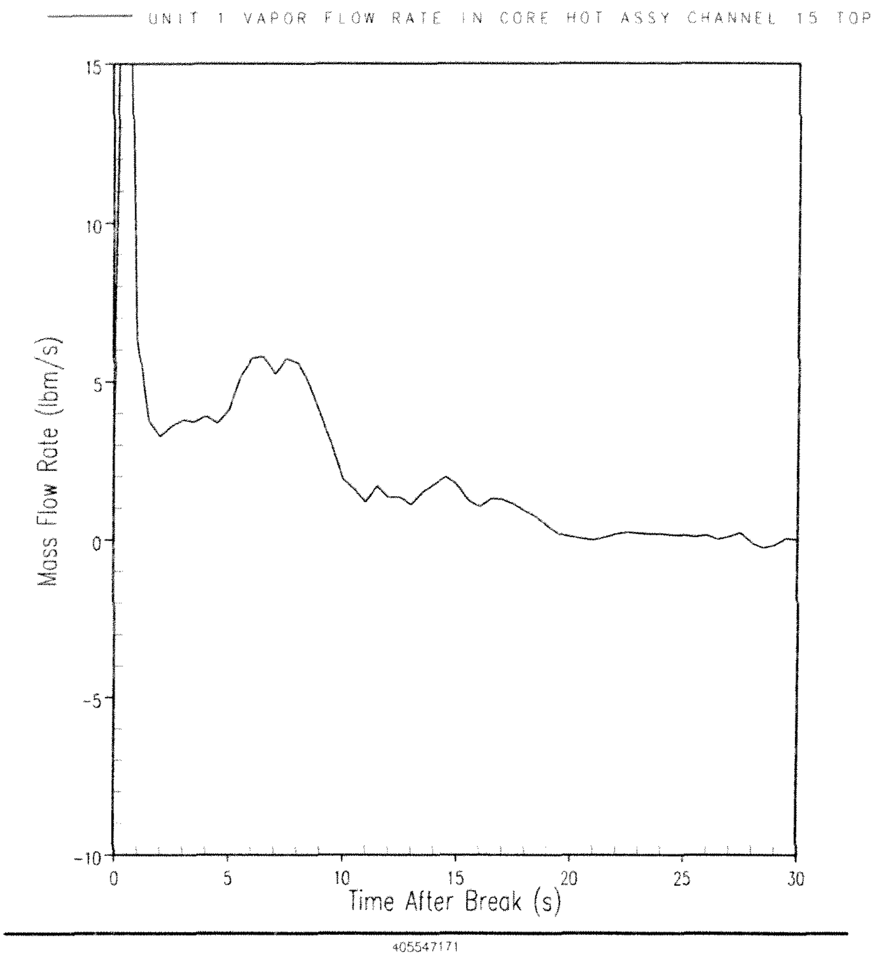

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FIGURE 15.6-8f

VAPOR FLOW RATE AT TOP OF HOT ASSEMBLY CHANNEL DURING BLOWDOWN FOR ECCS LB LOCA PCT LIMITING CASE-UNIT 1

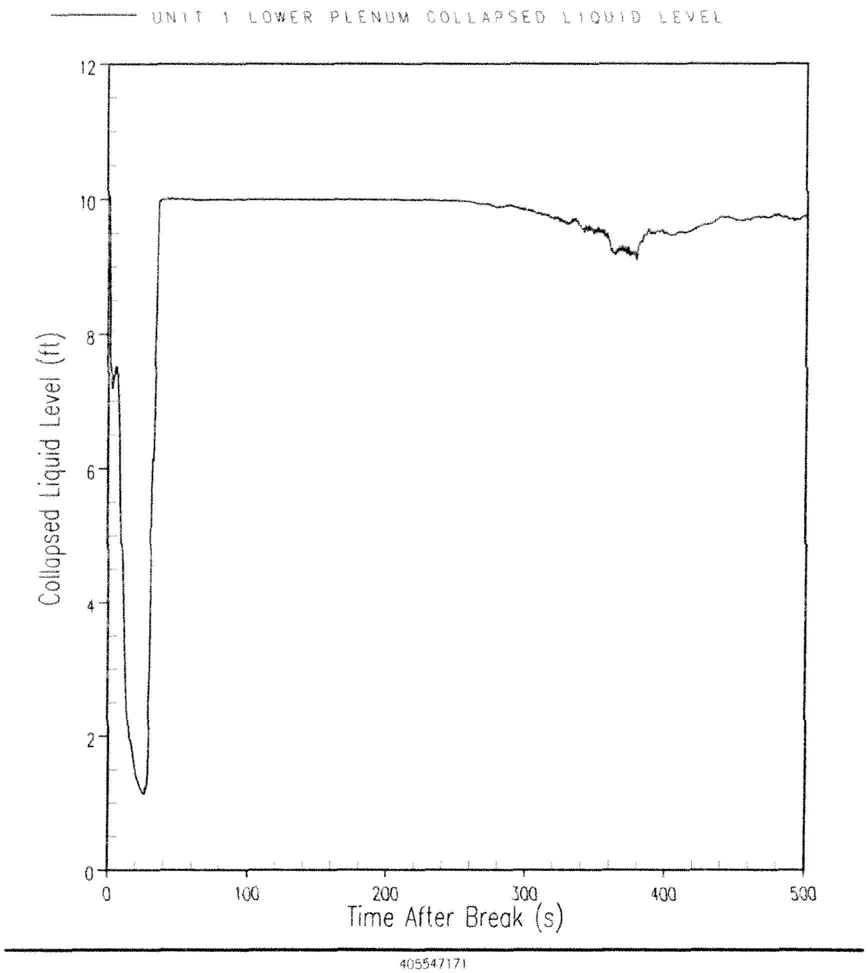

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FIGURE 15.6-8g

COLLAPSED LIQUID LEVEL IN LOWER PLENUM

FOR ECCS LB LOCA PCT LIMITING CASE-UNIT 1

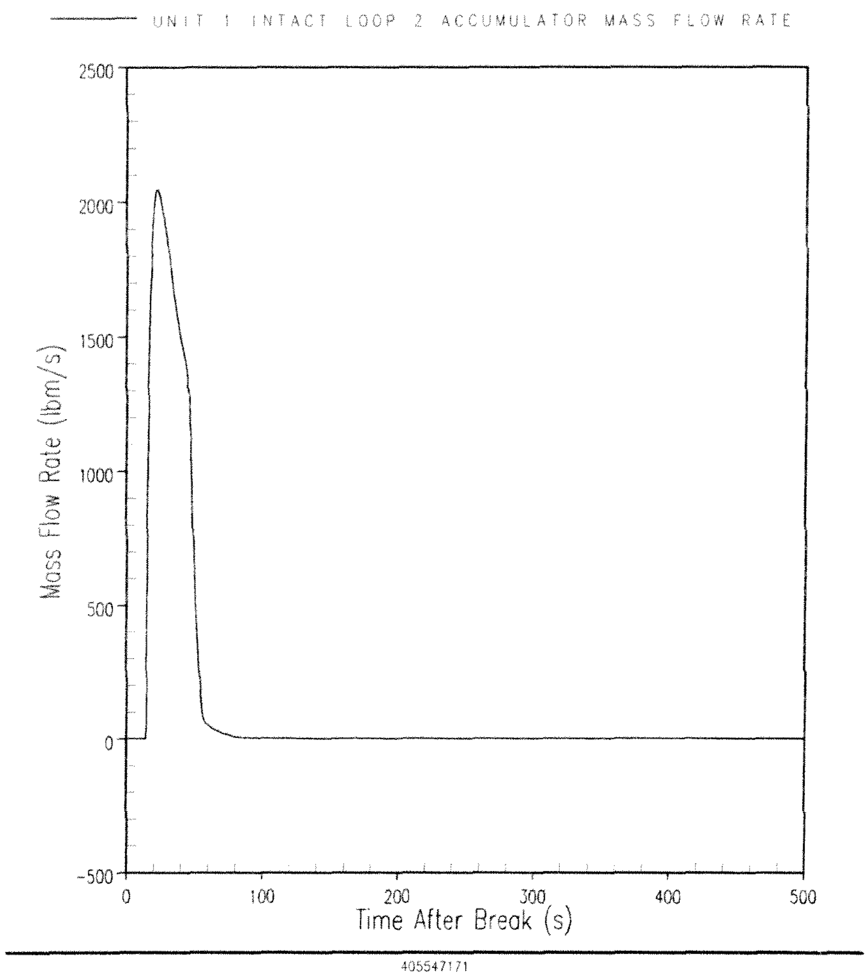

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FIGURE 15.6-8h

ACCUMULATOR MASS FLOW RATE
VAPOR FLOW RATE
FOR ECCS LB LOCA PCT LIMITING CASE-UNIT 1

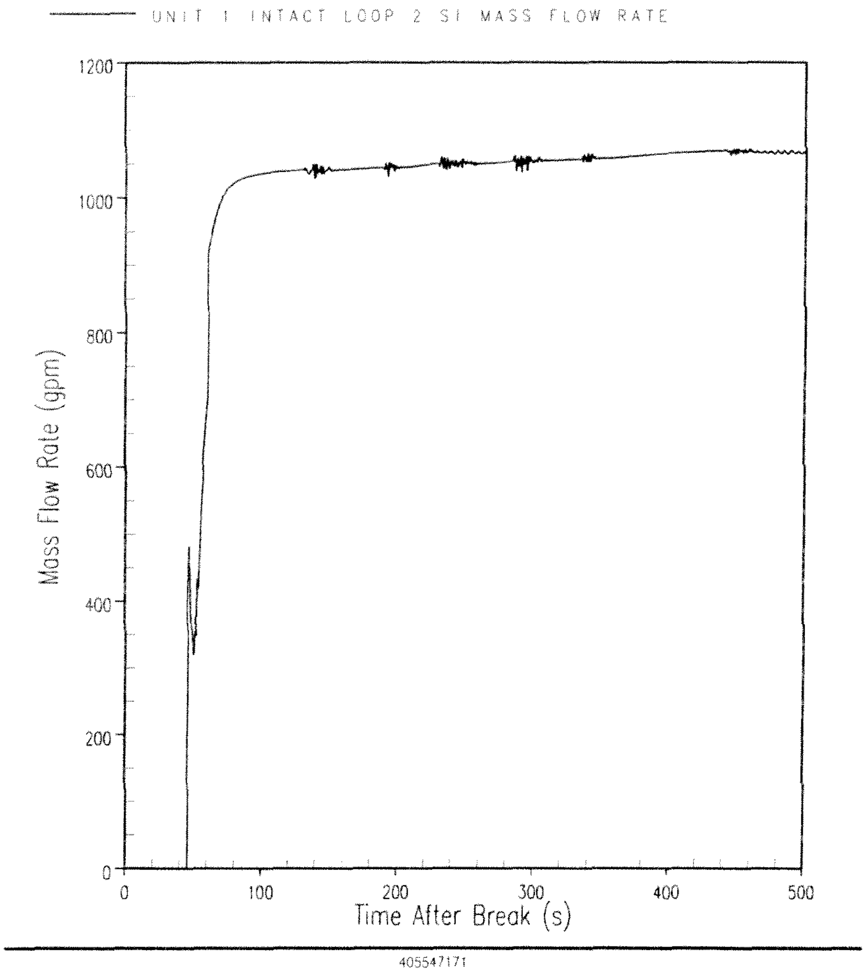

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FIGURE 15.6-8i

TOTAL SI MASS FLOW RATE FOR ONE INTACT LOOP FOR ECCS LB LOCA PCT LIMITING CASE-UNIT 1

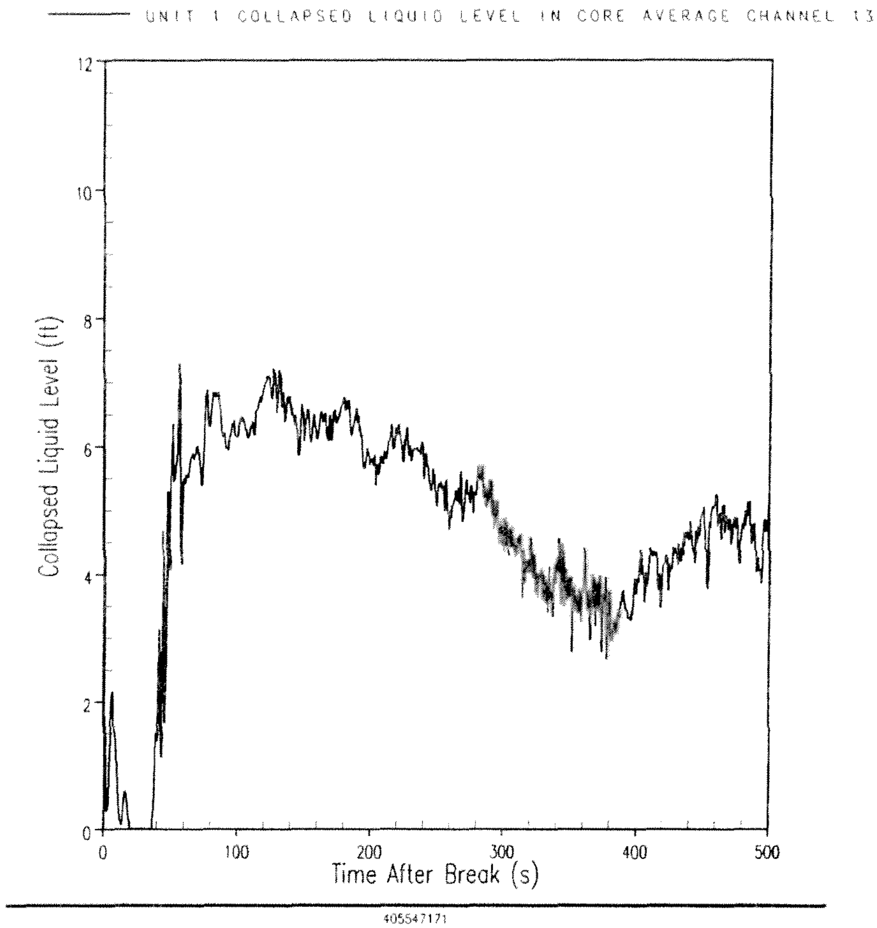

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FIGURE 15.6-8j

COLLAPSED LIQUID LEVEL IN CORE
AVERAGE CHANNEL
FOR ECCS LB LOCA PCT LIMITING CASE-UNIT 1

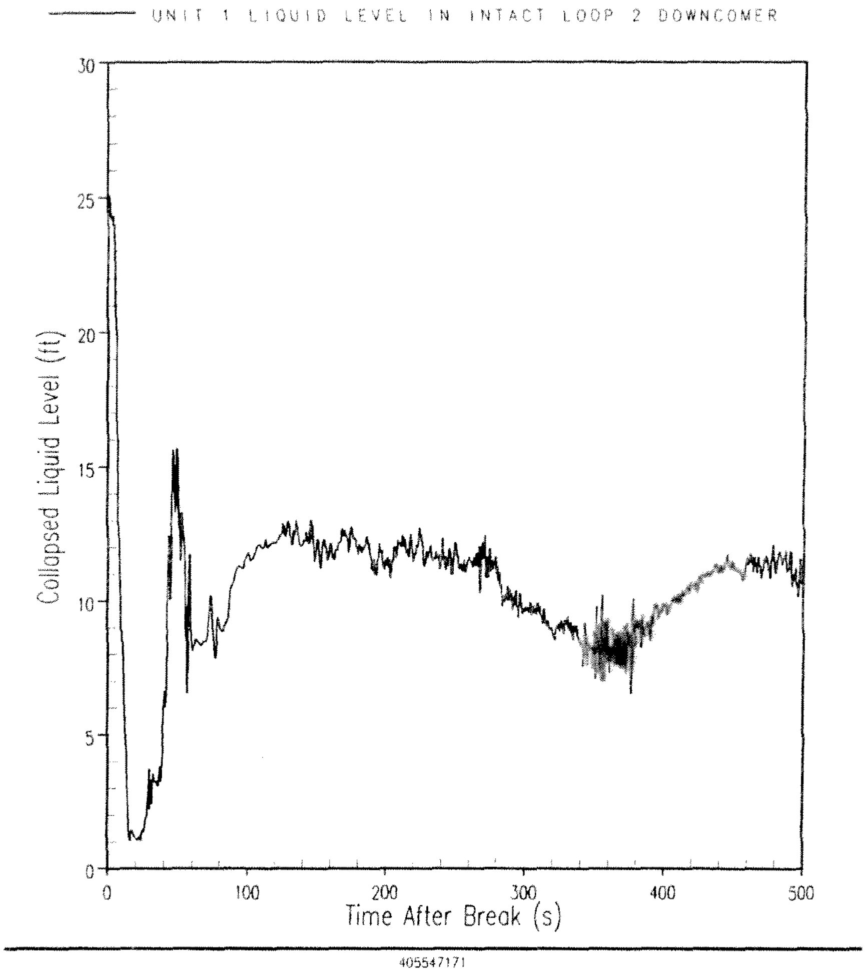

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FIGURE 15.6-8k

COLLAPSED LIQUID LEVEL IN DOWNCOMER CHANNEL FOR ECCS LB LOCA PCT LIMITING CASE-UNIT 1

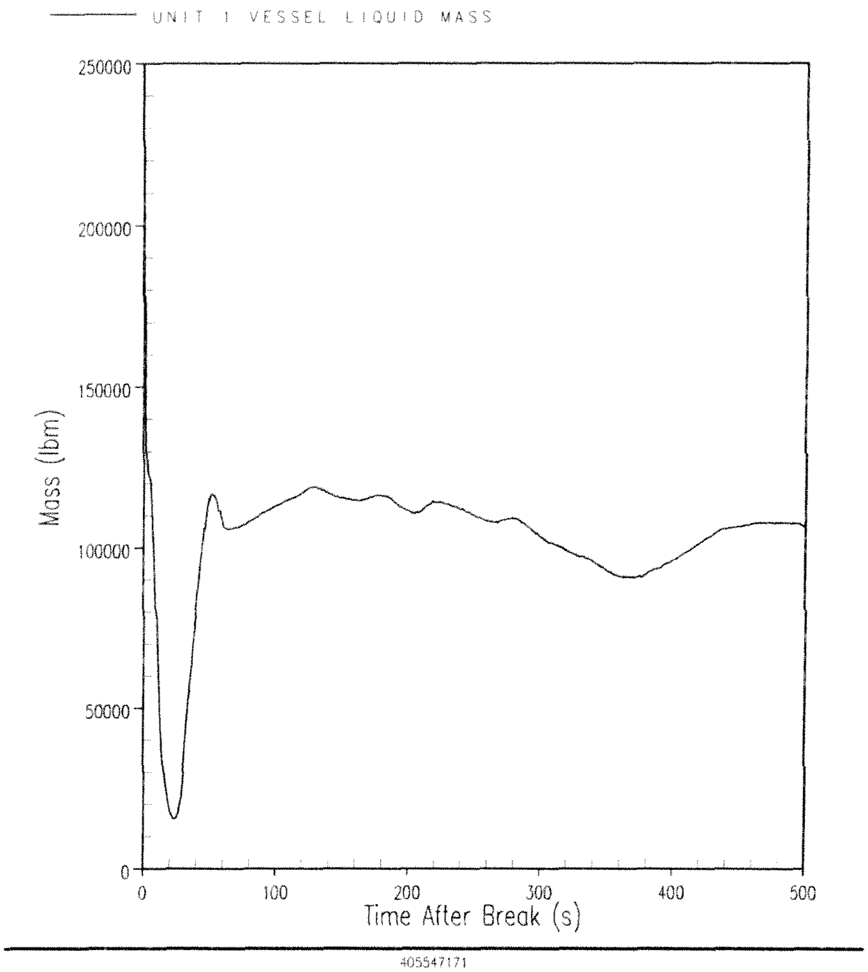

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FIGURE 15.6-81

VESSEL FLUID MASS

FOR ECCS LB LOCA PCT LIMITING CASE-UNIT 1

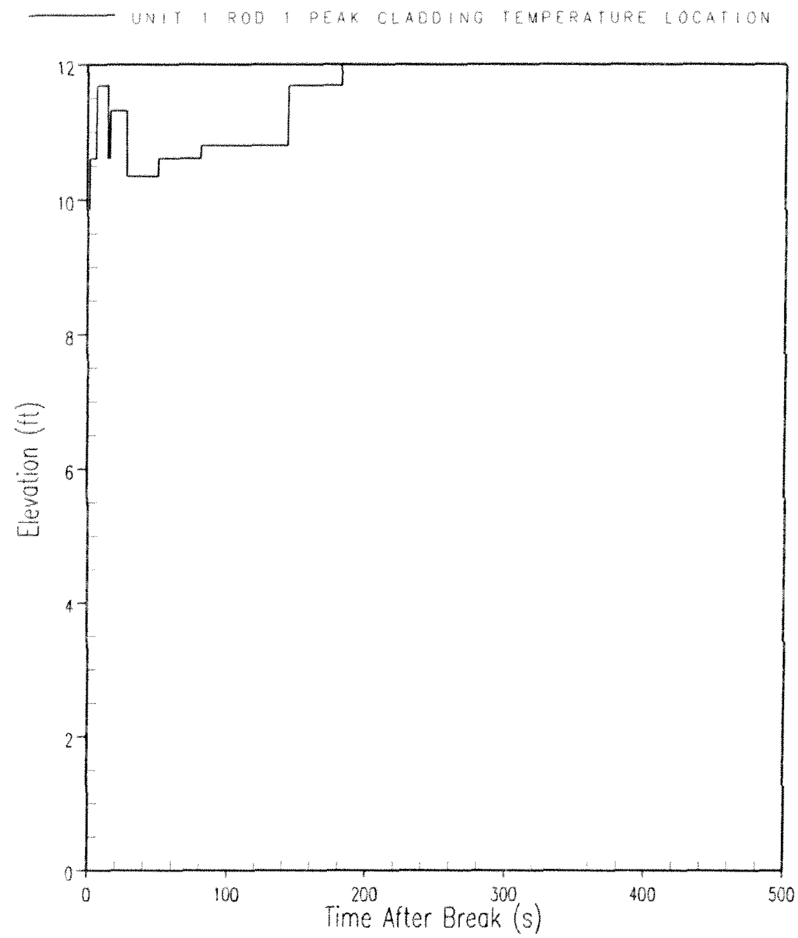

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FIGURE 15.6-8m

HOT ROD PEAK CLADDING TEMPERATURE LOCATION
FOR ECCS LB LOCA PCT LIMITING CASE-UNIT 1

Figures 15.6-8n through 15.6-8p have been deleted intentionally.

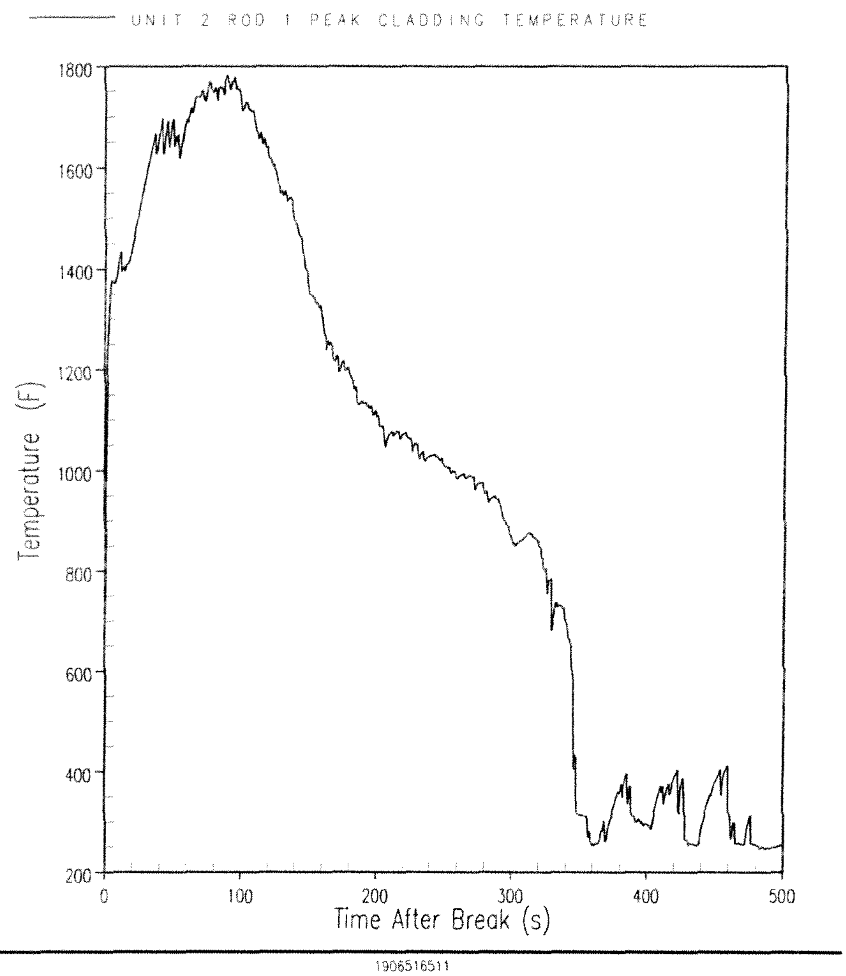

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FIGURE 15.6-9a

HOT ROD PEAK CLADDING TEMPERATURE

FOR ECCS LB LOCA PCT LIMITING CASE-UNIT 2

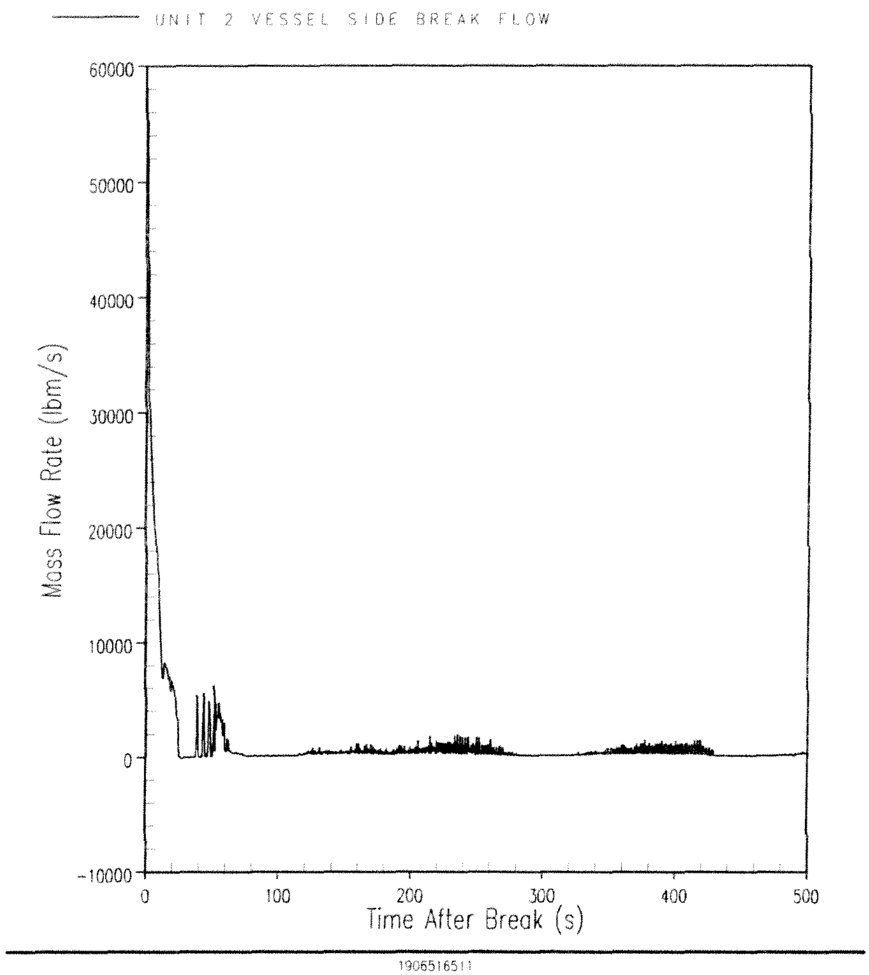

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FIGURE 15.6-9b

BREAK FLOW ON VESSEL SIDE OF BROKEN COLD LEG FOR ECCS LB LOCA PCT LIMITING CASE-UNIT 2

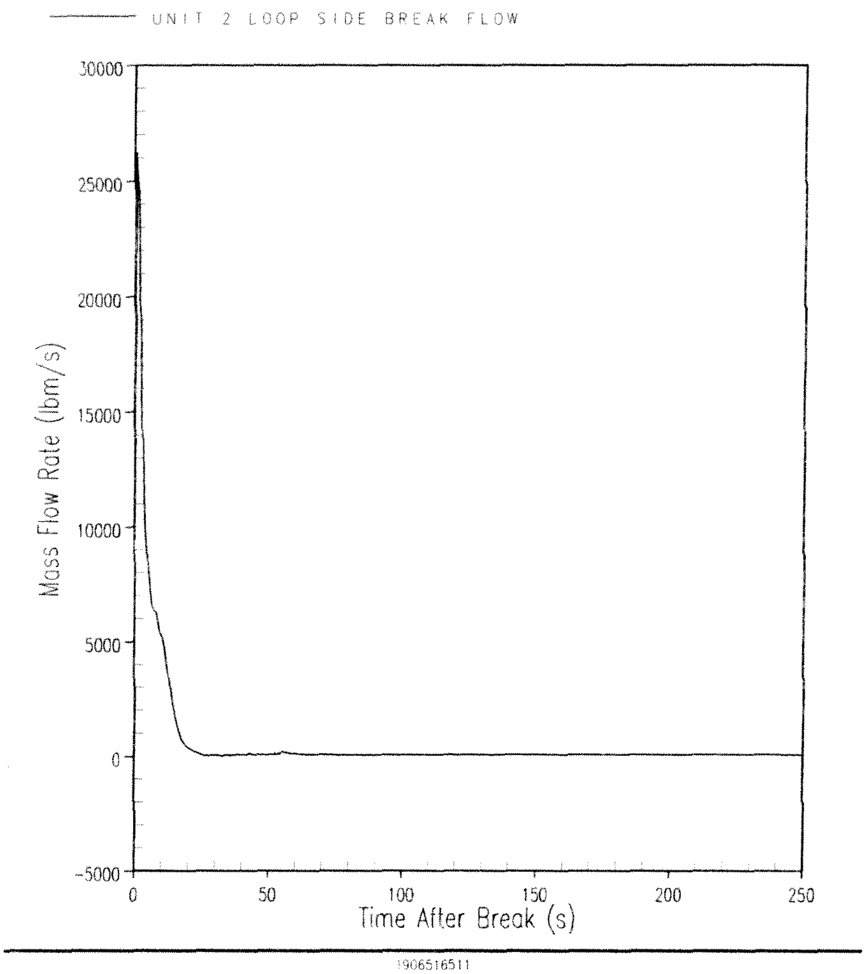

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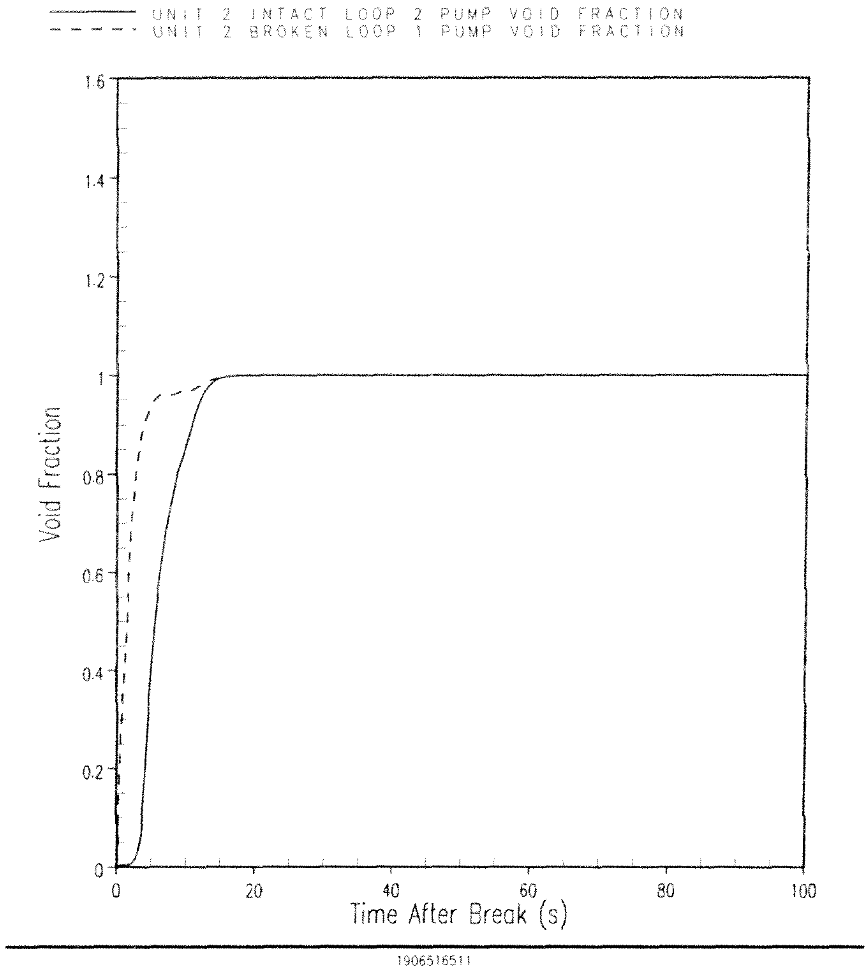
FIGURE 15.6-9c

BREAK FLOW ON LOOP SIDE OF
BROKEN COLD LEG
FOR ECCS LB LOCA PCT LIMITING CASE-UNIT 2



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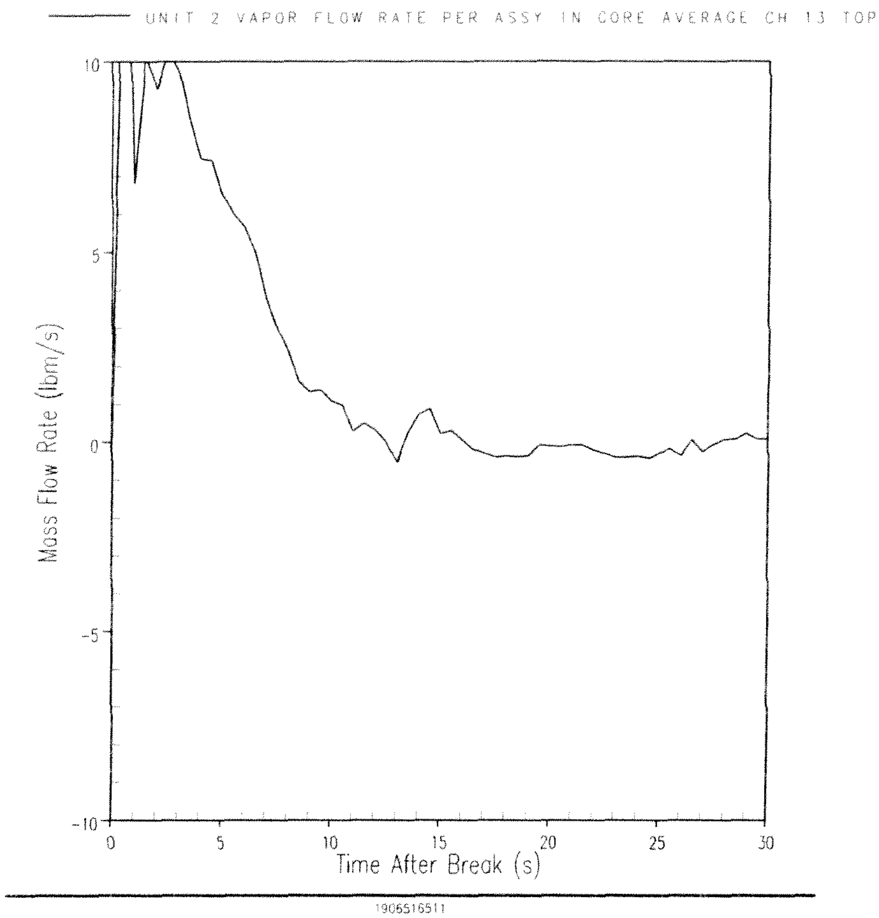
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 15.6-9d

VOID FRACTION AT THE INTACT AND BROKEN LOOP PUMP INLET FOR ECCS LB LOCA PCT LIMITING CASE-UNIT 2



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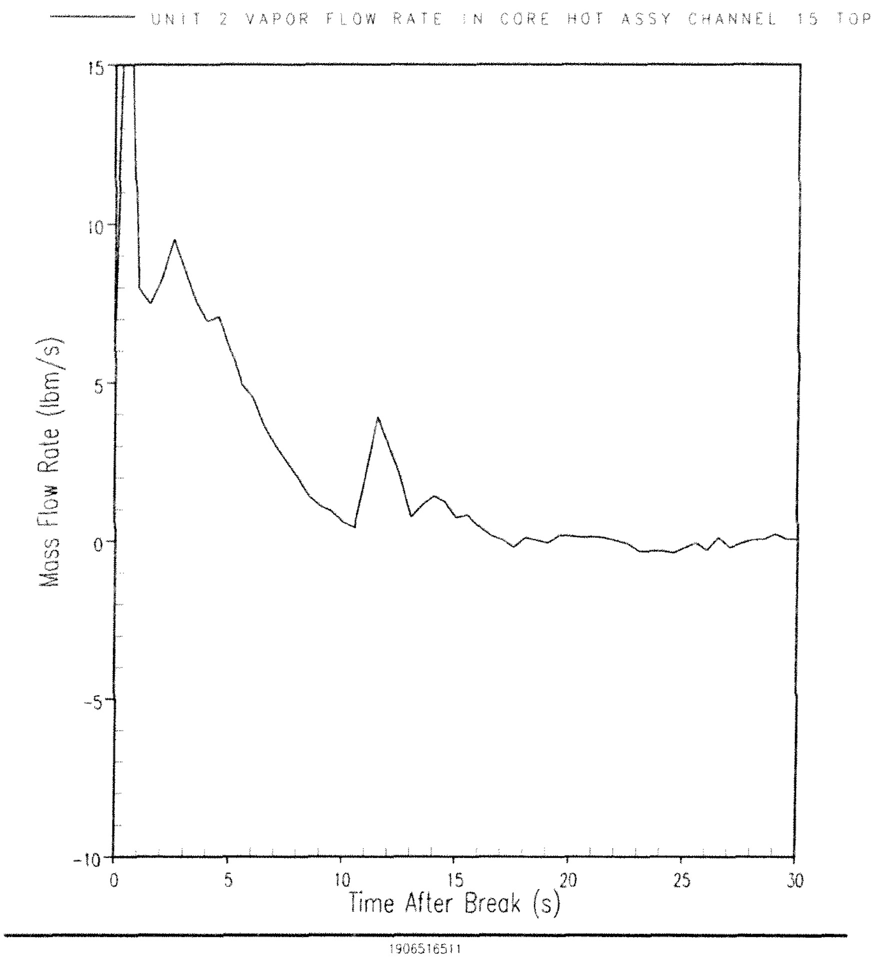
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 15.6-9e

VAPOR FLOW RATE AT TOP OF CORE AVERAGE CHANNEL DURING BLOWDOWN FOR ECCS LB LOCA PCT LIMITING CASE-UNIT 2



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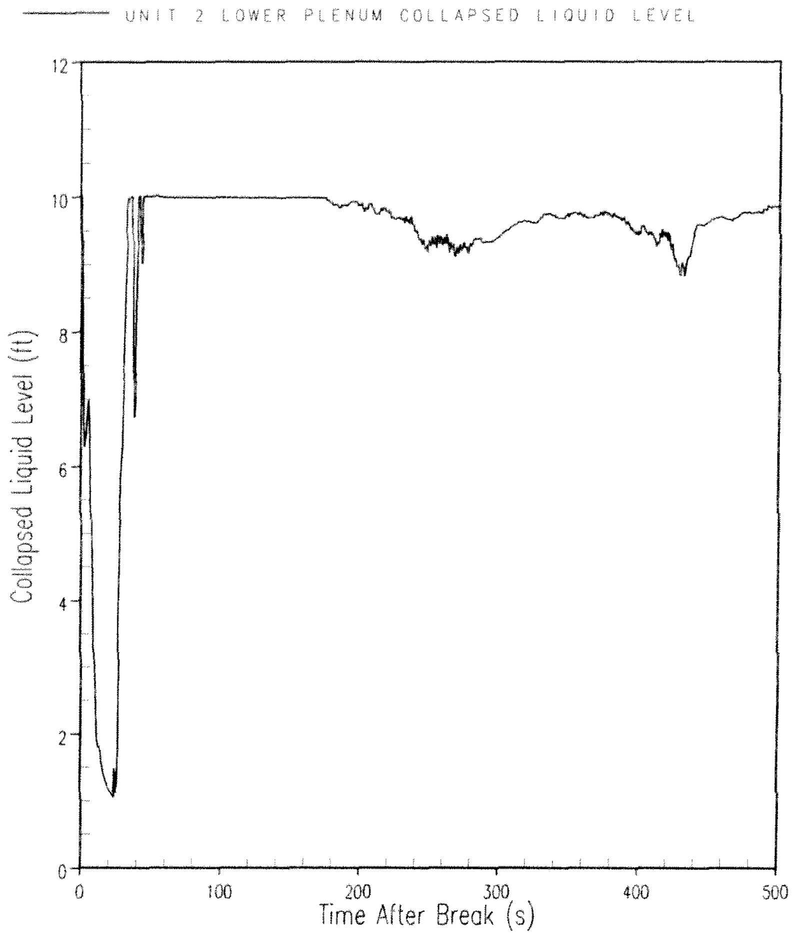
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 15.6-9f

VAPOR FLOW RATE AT TOP OF HOT ASSEMBLY CHANNEL DURING BLOWDOWN FOR ECCS LB LOCA PCT LIMITING CASE-UNIT 2



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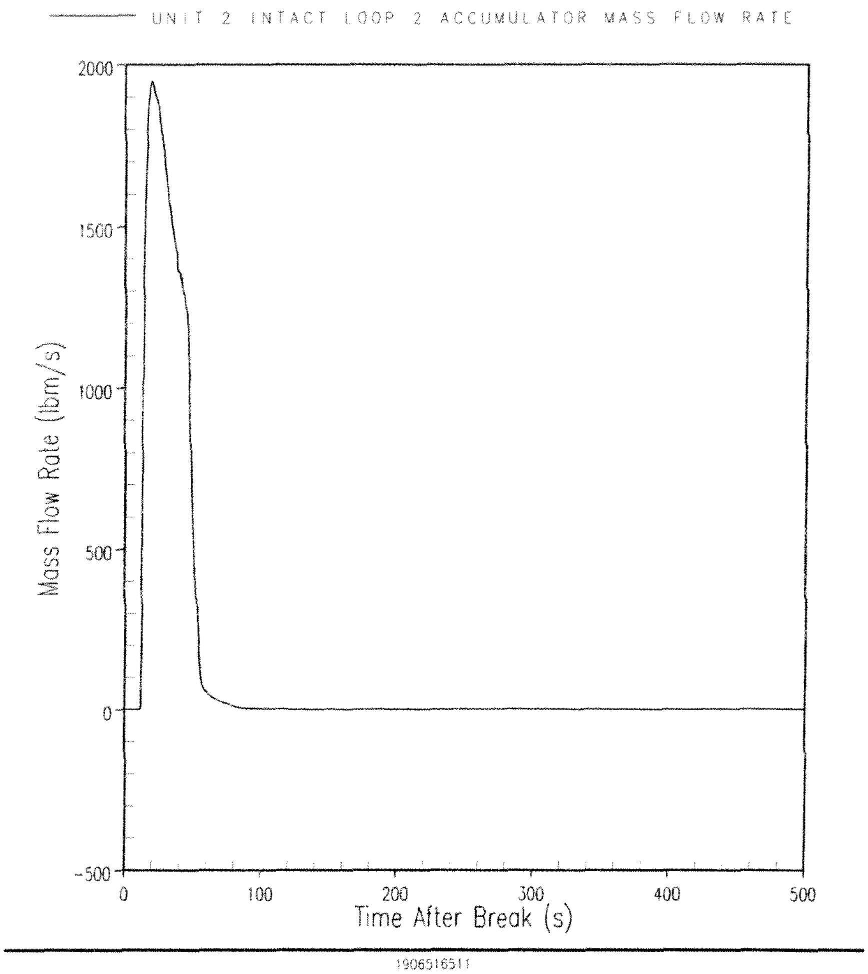
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 15.6-9g

COLLAPSED LIQUID LEVEL IN LOWER PLENUM

FOR ECCS LB LOCA PCT LIMITING CASE-UNIT 2



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UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 15.6-9h

ACCUMULATOR MASS FLOW RATE
VAPOR FLOW RATE
FOR ECCS LB LOCA PCT LIMITING CASE-UNIT 2

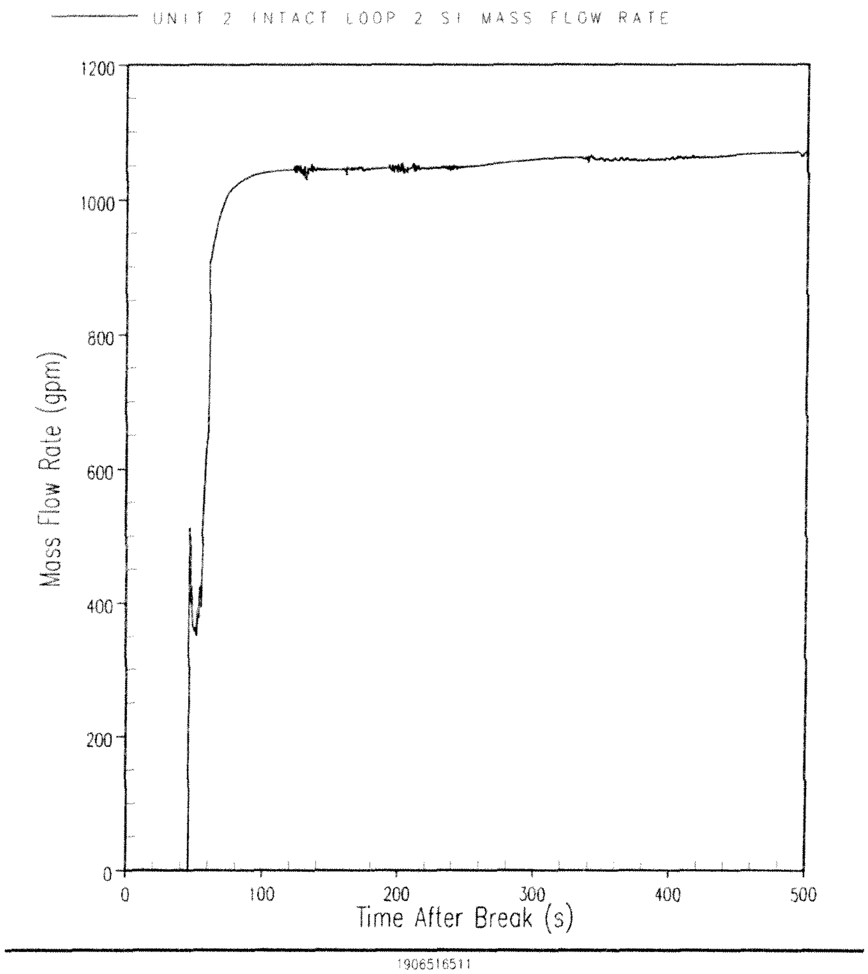

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FIGURE 15.6-9i

TOTAL SI MASS FLOW RATE FOR ONE INTACT LOOP FOR ECCS LB LOCA PCT LIMITING CASE-UNIT 2

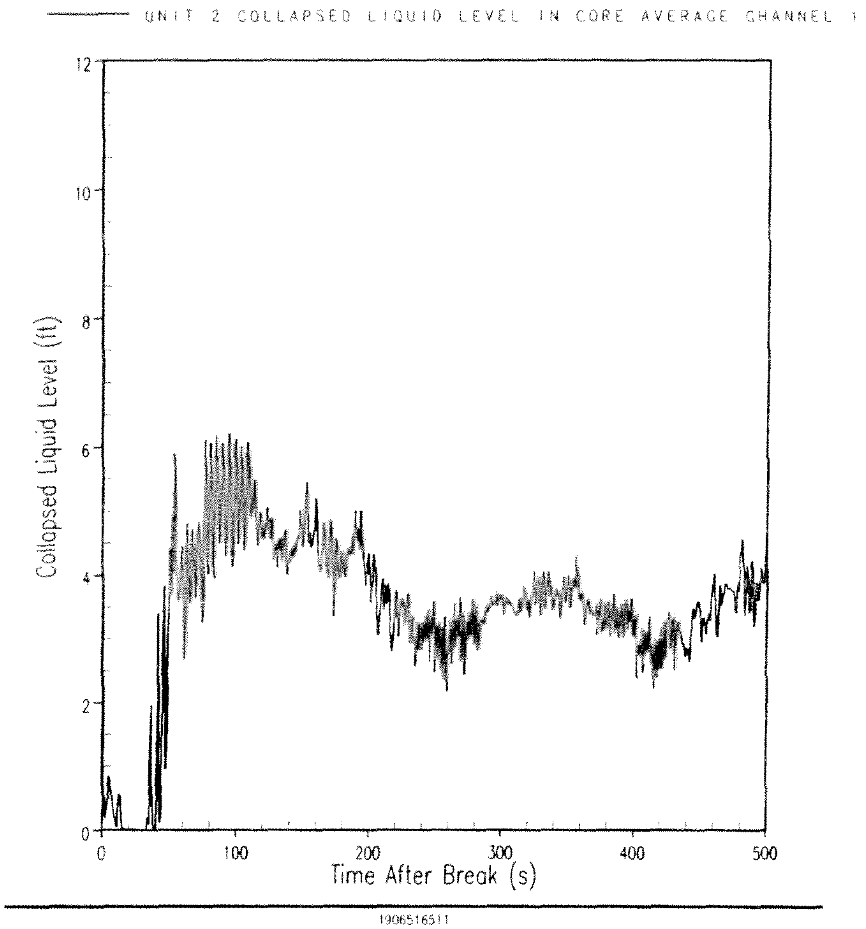

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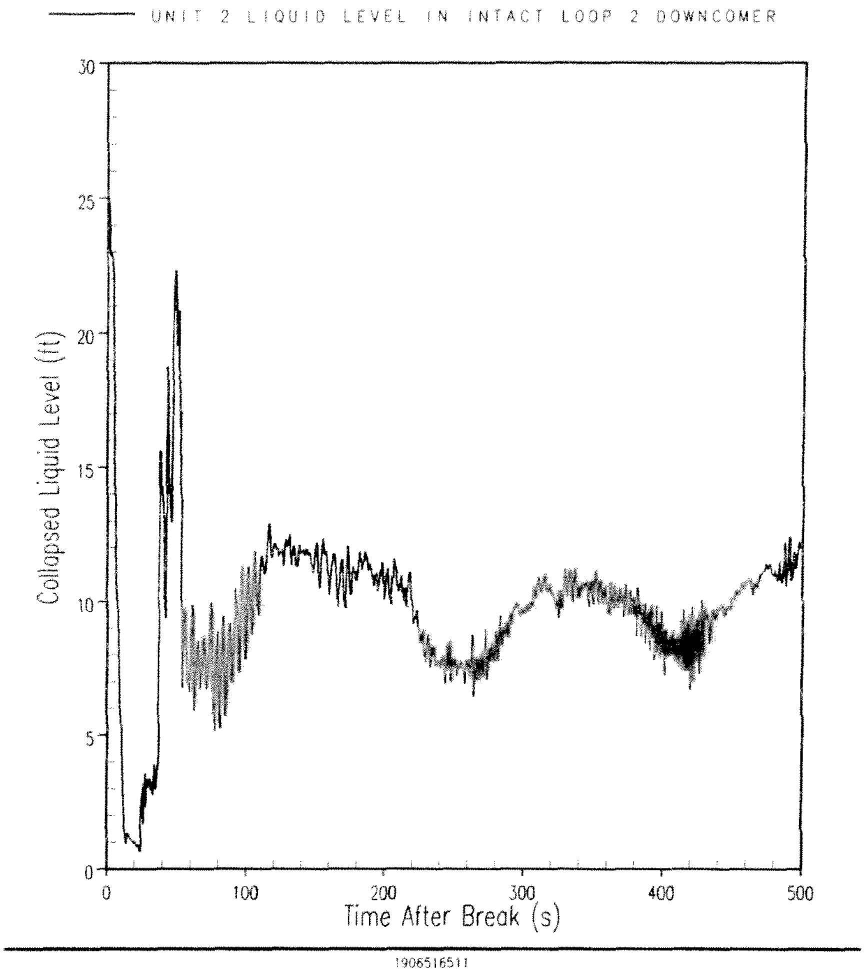
FIGURE 15.6-9j

COLLAPSED LIQUID LEVEL IN
CORE AVERAGE CHANNEL
FOR ECCS LB LOCA PCT LIMITING CASE-UNIT 2



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UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 15.6-9k

COLLAPSED LIQUID LEVEL IN DOWNCOMER CHANNEL FOR ECCS LB LOCA PCT LIMITING CASE-UNIT 2

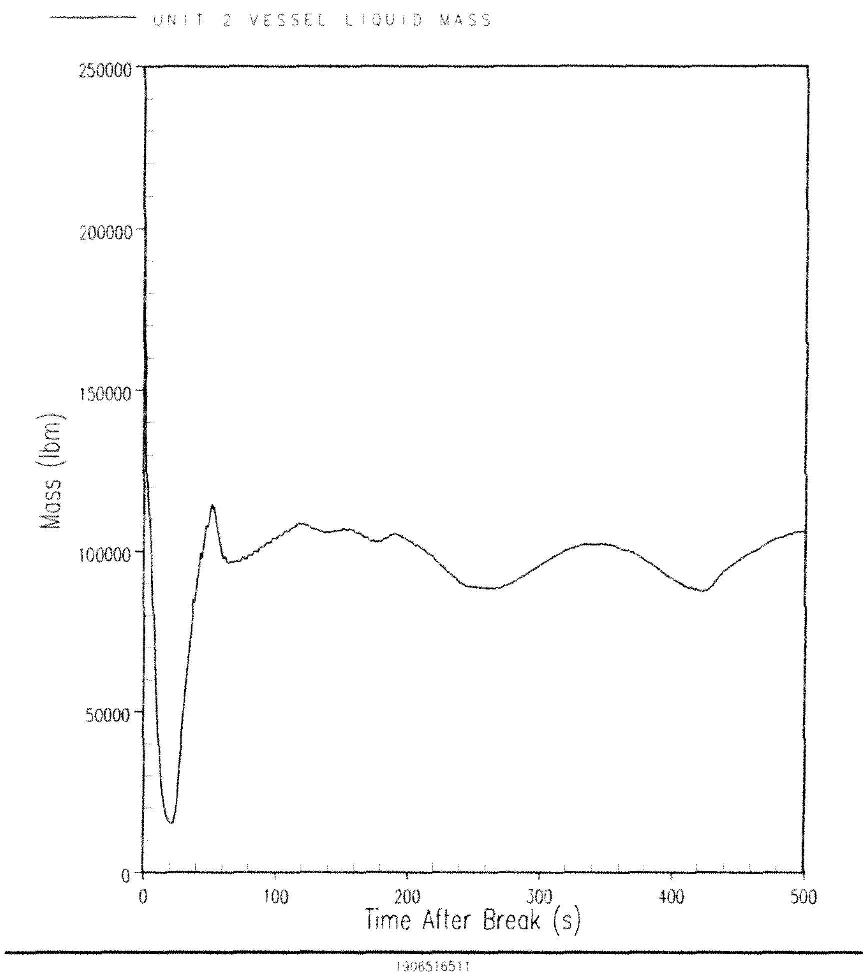

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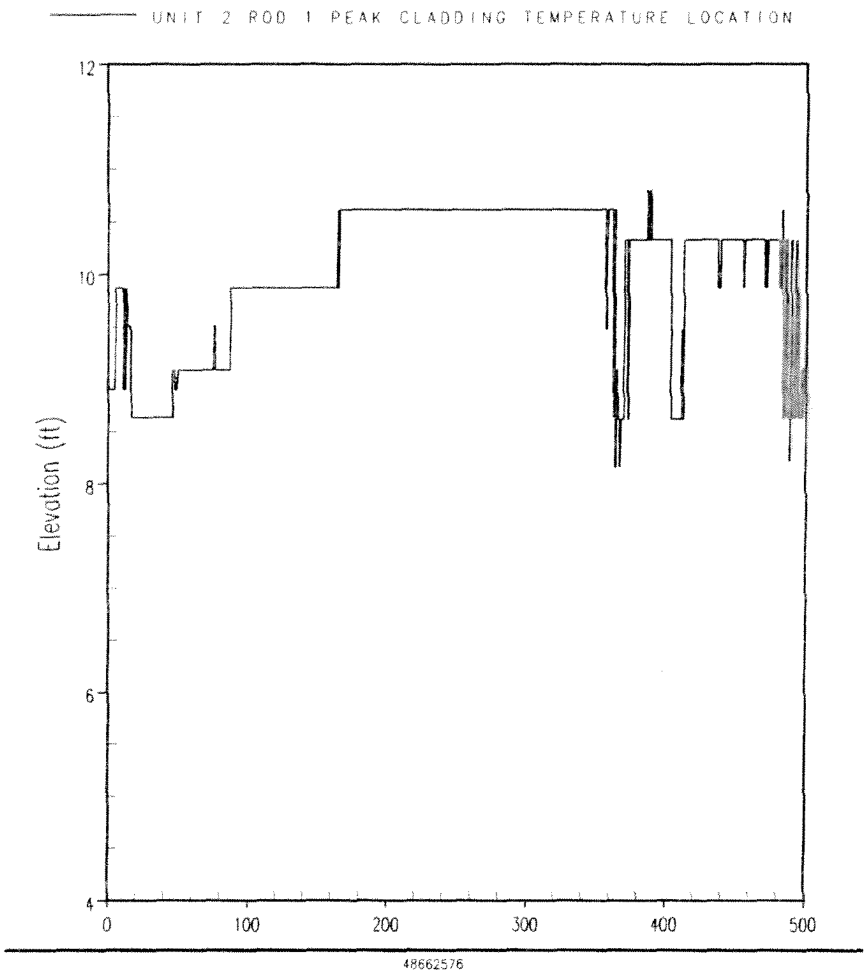
FIGURE 15.6-91

VESSEL FLUID MASS

FOR ECCS LB LOCA PCT LIMITING CASE-UNIT 2



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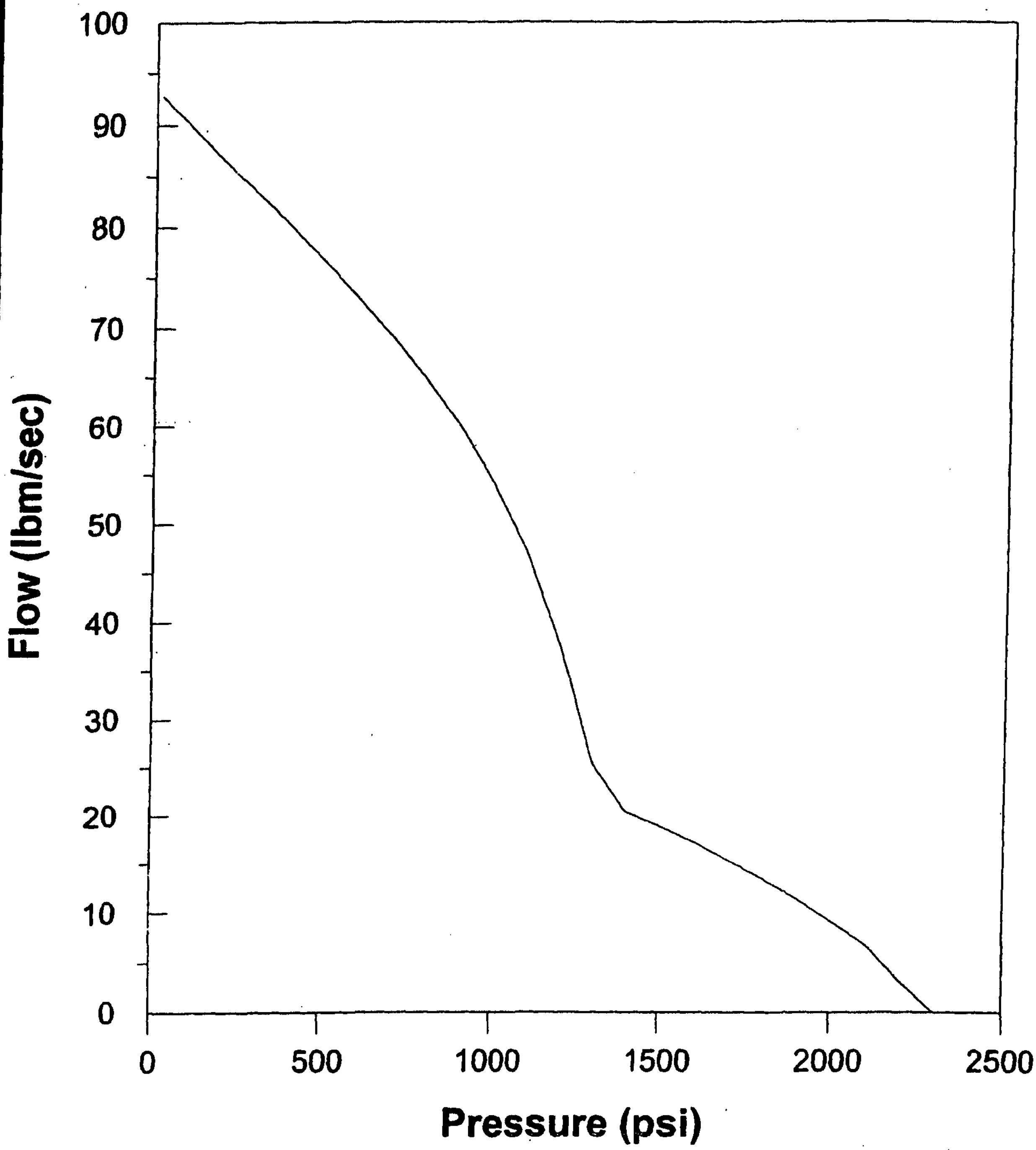
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 15.6-9m

HOT ROD PEAK CLADDING
TEMPERATURE LOCATION
FOR ECCS LB LOCA PCT LIMITING CASE-UNIT 2

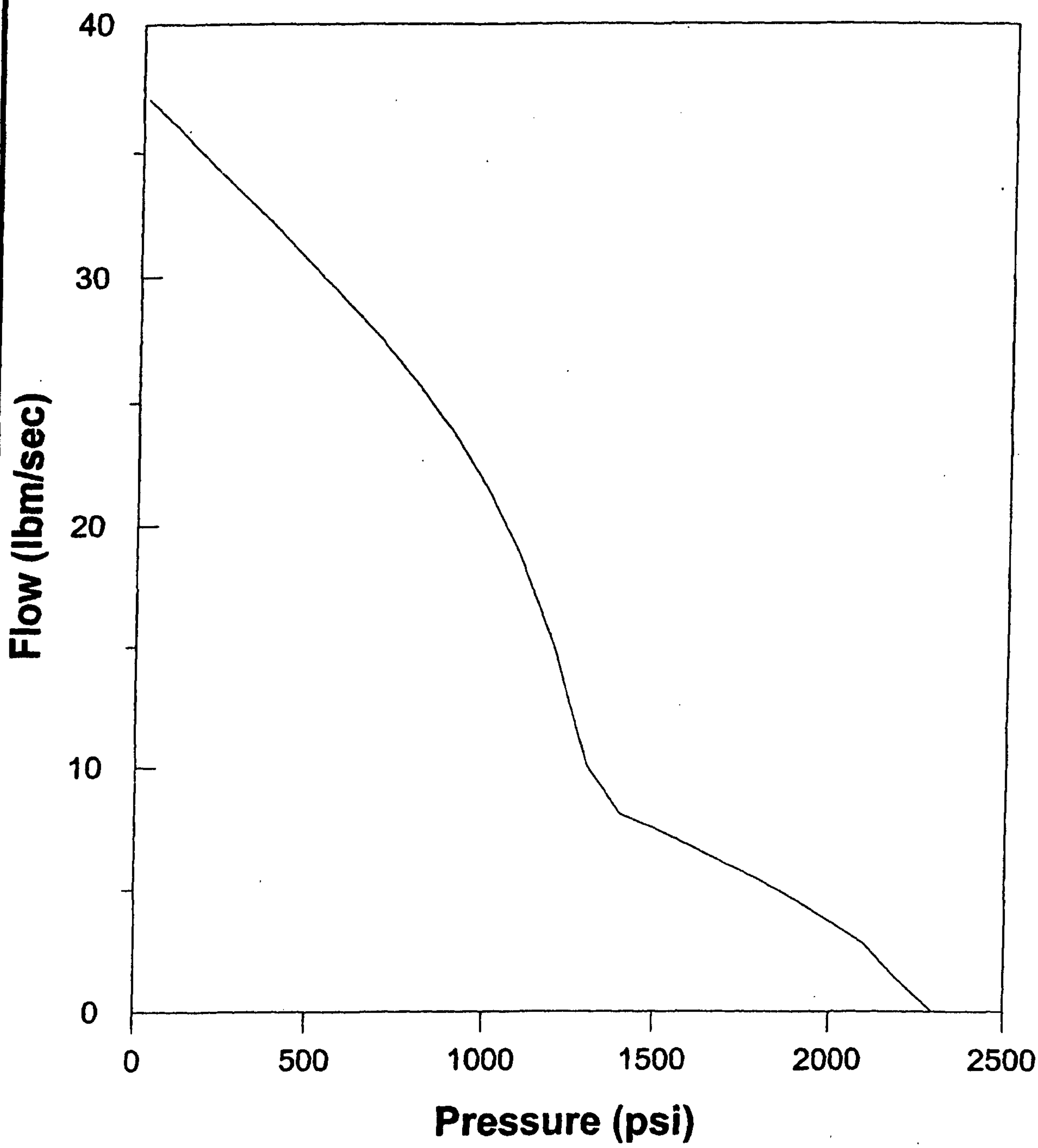
Figures 15.6-9n through 15.6-12p have been deleted intentionally.



**Byron/Braidwood Stations
Updated Final Safety Analysis
Report**

**Intact Loop Safety Injection Flow Rates
Used in Small Break**

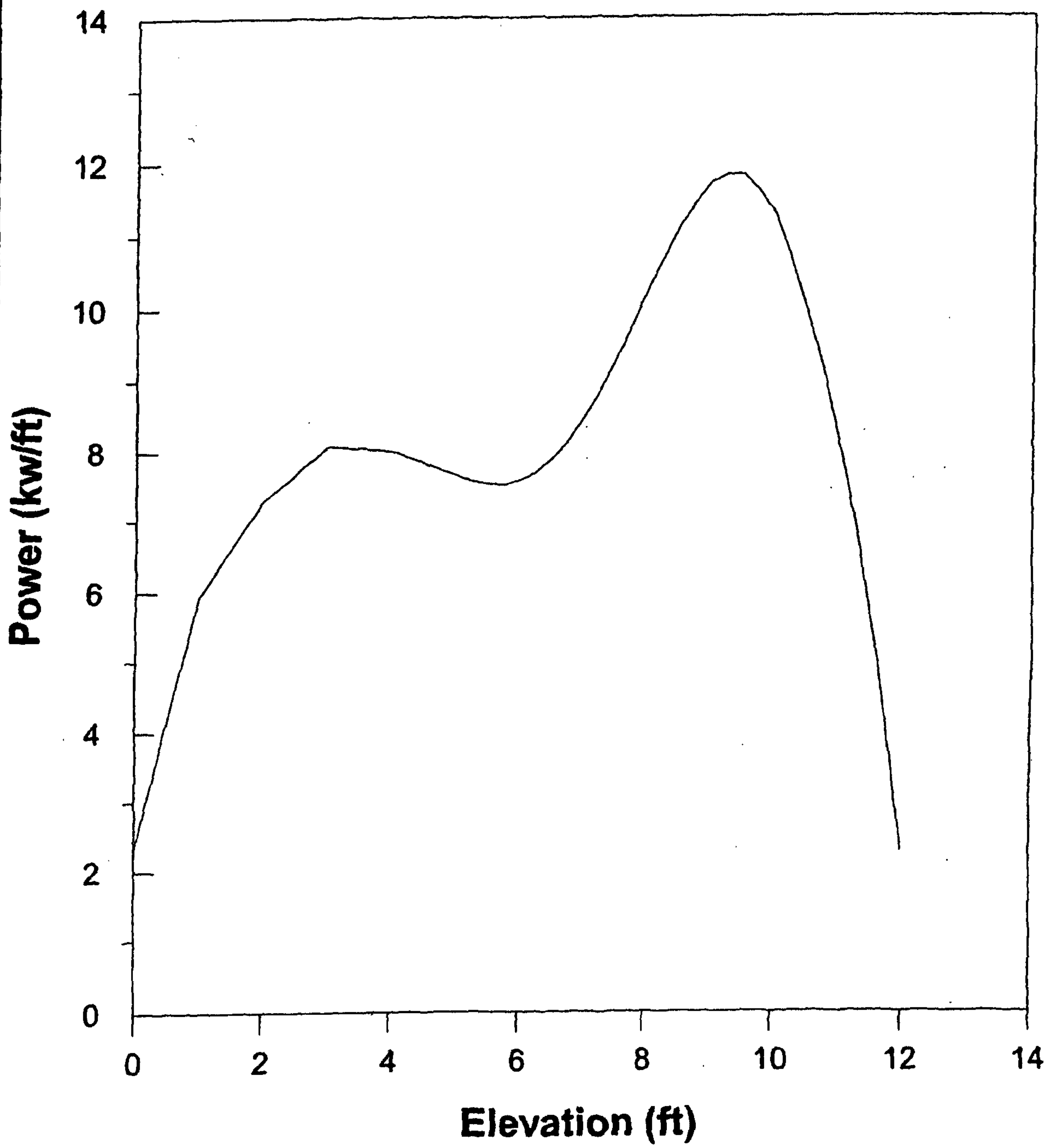
Figure 15.6-13a



Byron/Braidwood Stations
Updated Final Safety Analysis Report

Broken Loop Safety Injection Flow Rates Used in Small Break

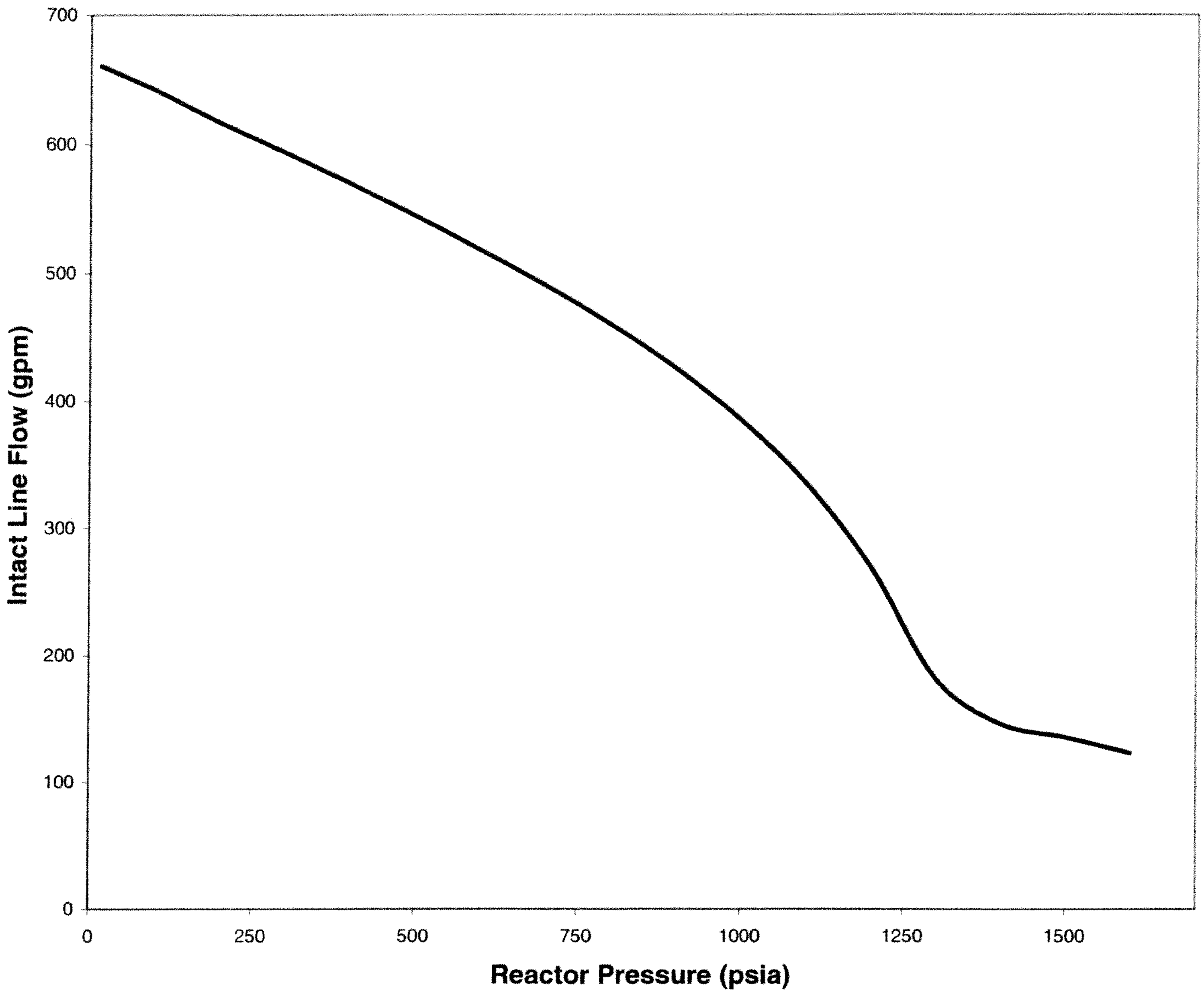
Figure 15.6-13b



Byron/Braidwood Stations
Updated Final Safety Analysis Report

Core Power Shape
Used in Small Break

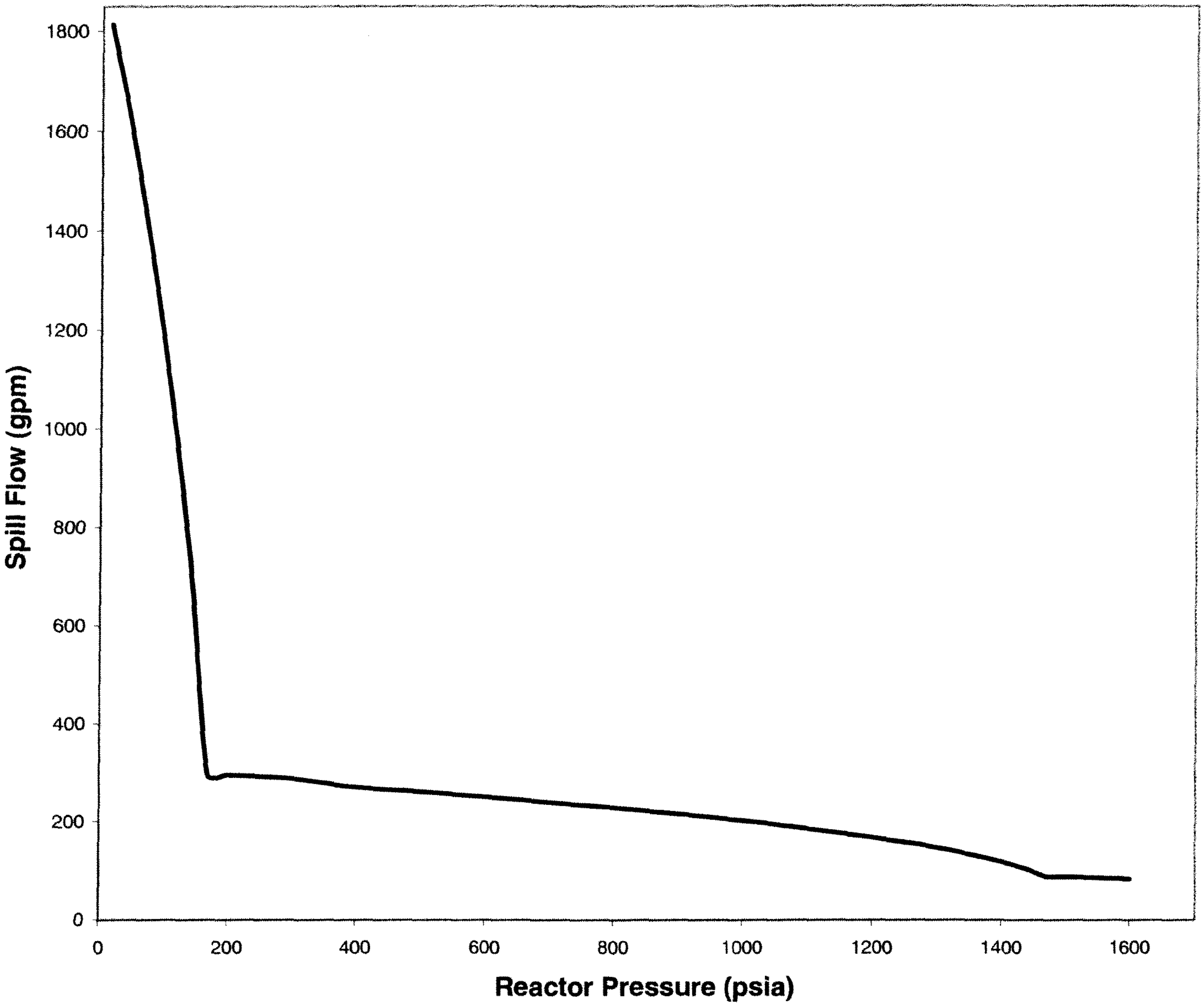
Figure 15.6-13c



BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 15.6-13d

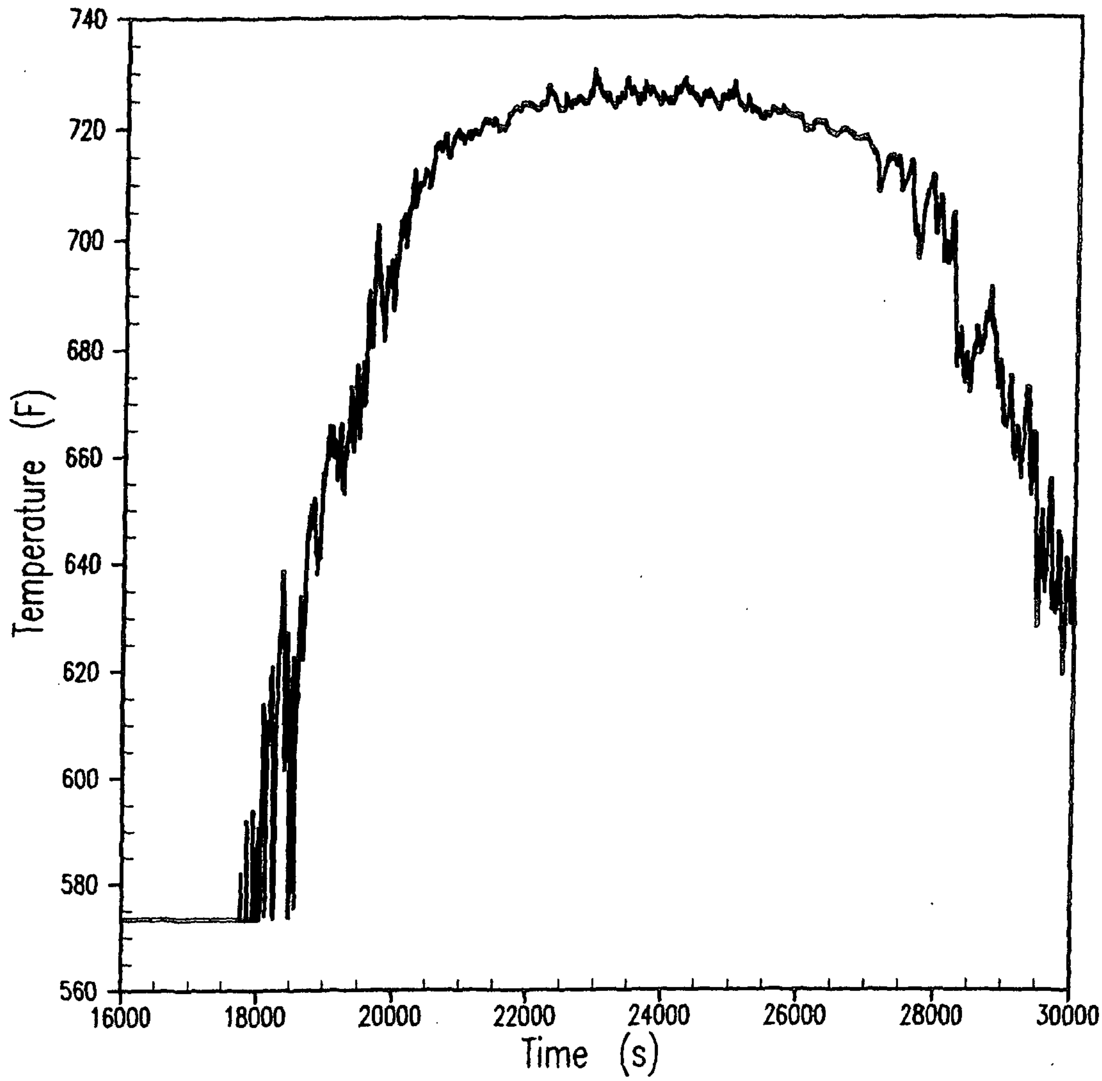
RECIRCULATION PHASE INTACT LINE ECCS
FLOW RATES USED IN SMALL BREAK LOCA



BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 15.6-13e

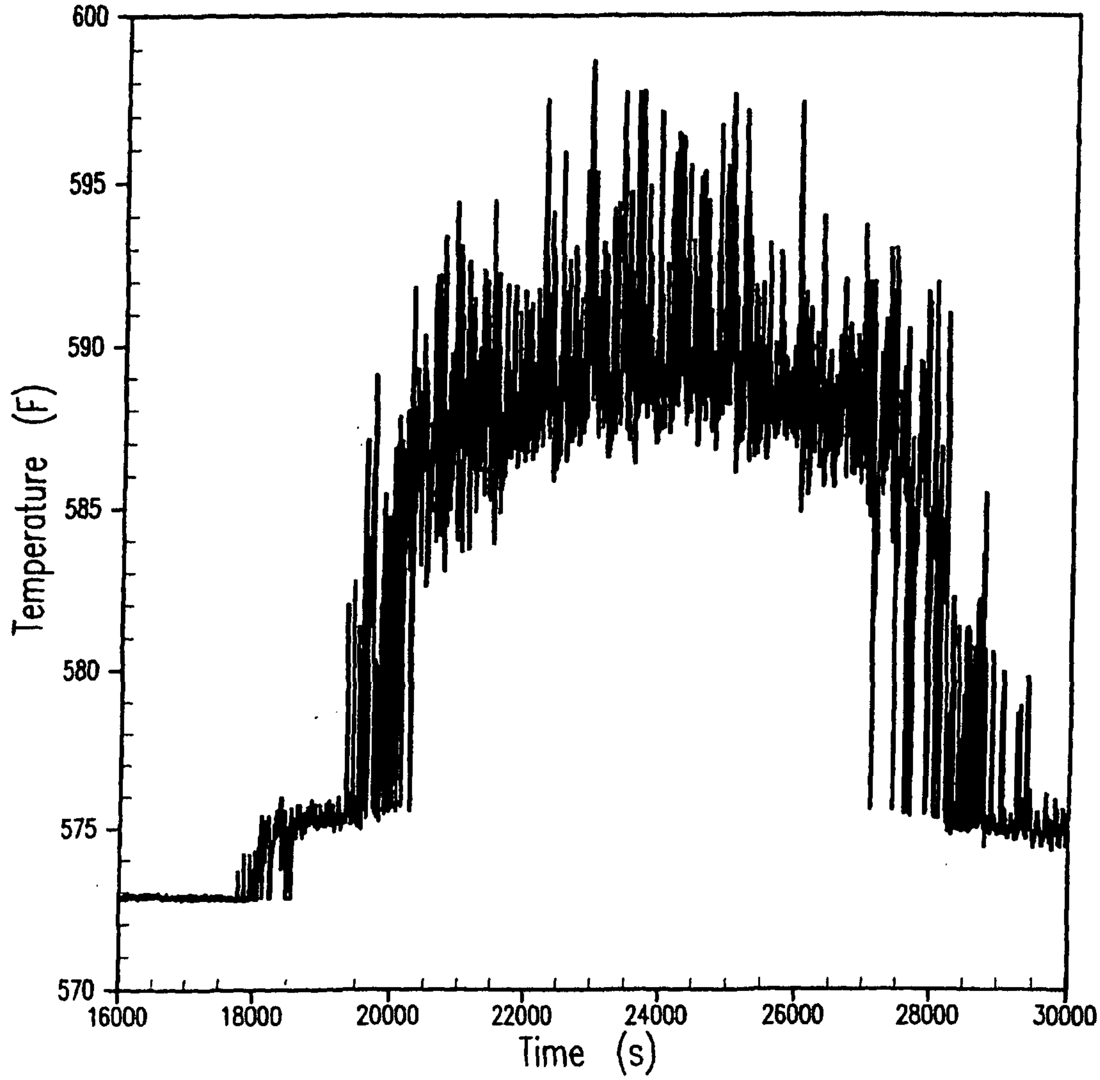
RECIRCULATION PHASE SPILL LINE ECCS FLOW RATES USED IN SMALL BREAK LOCA



Byron/Braidwood Stations
Updated Final Safety Analysis Report

Peak Clad Temperature - Unit 1
(1.5 Inch, Reduced T_{AVG})

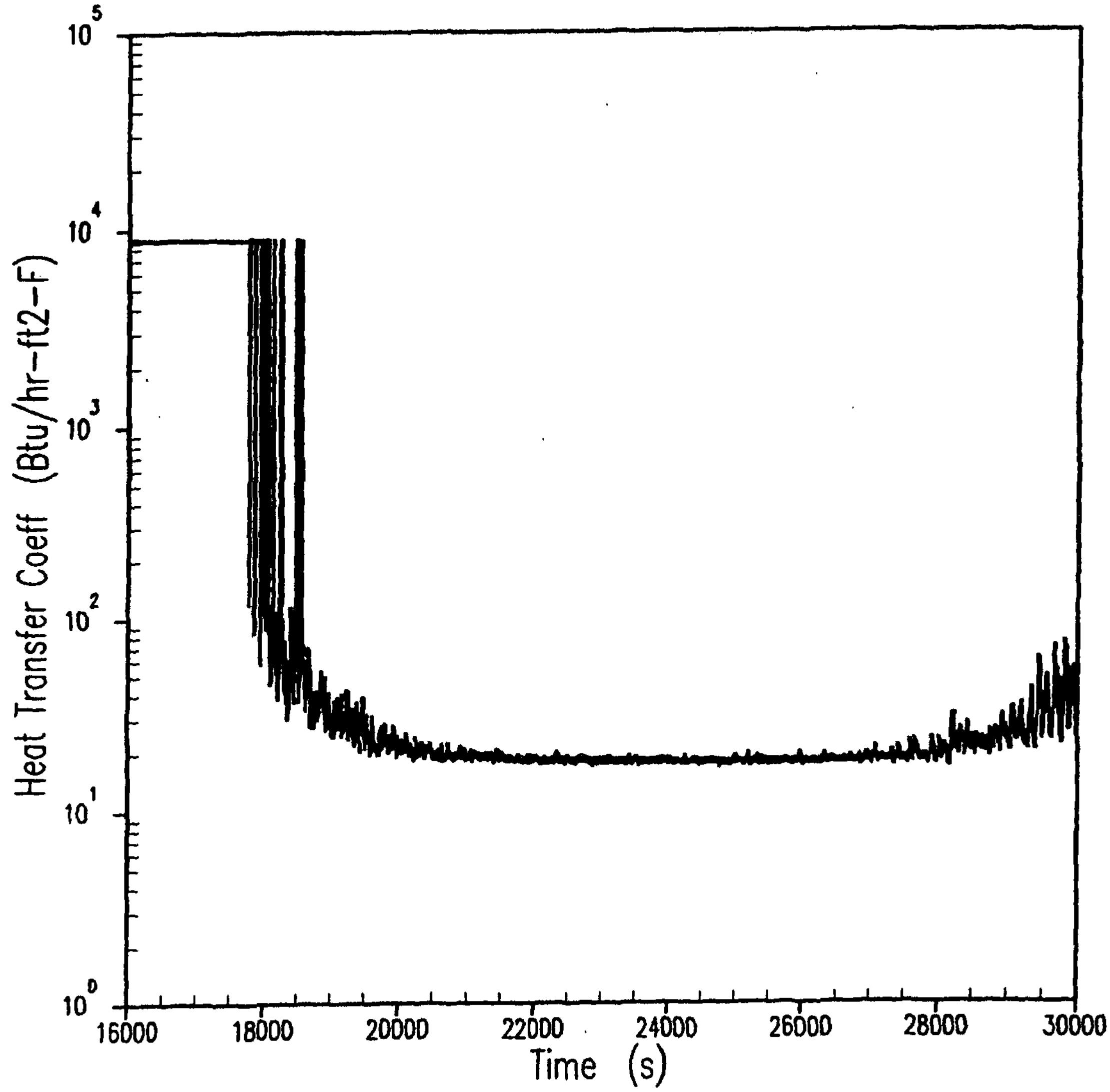
Figure 15.6-14a



Byron/Braidwood Stations
Updated Final Safety Analysis Report

Fluid Temperature - Unit 1
(1.5 Inch, Reduced T_{AVG})

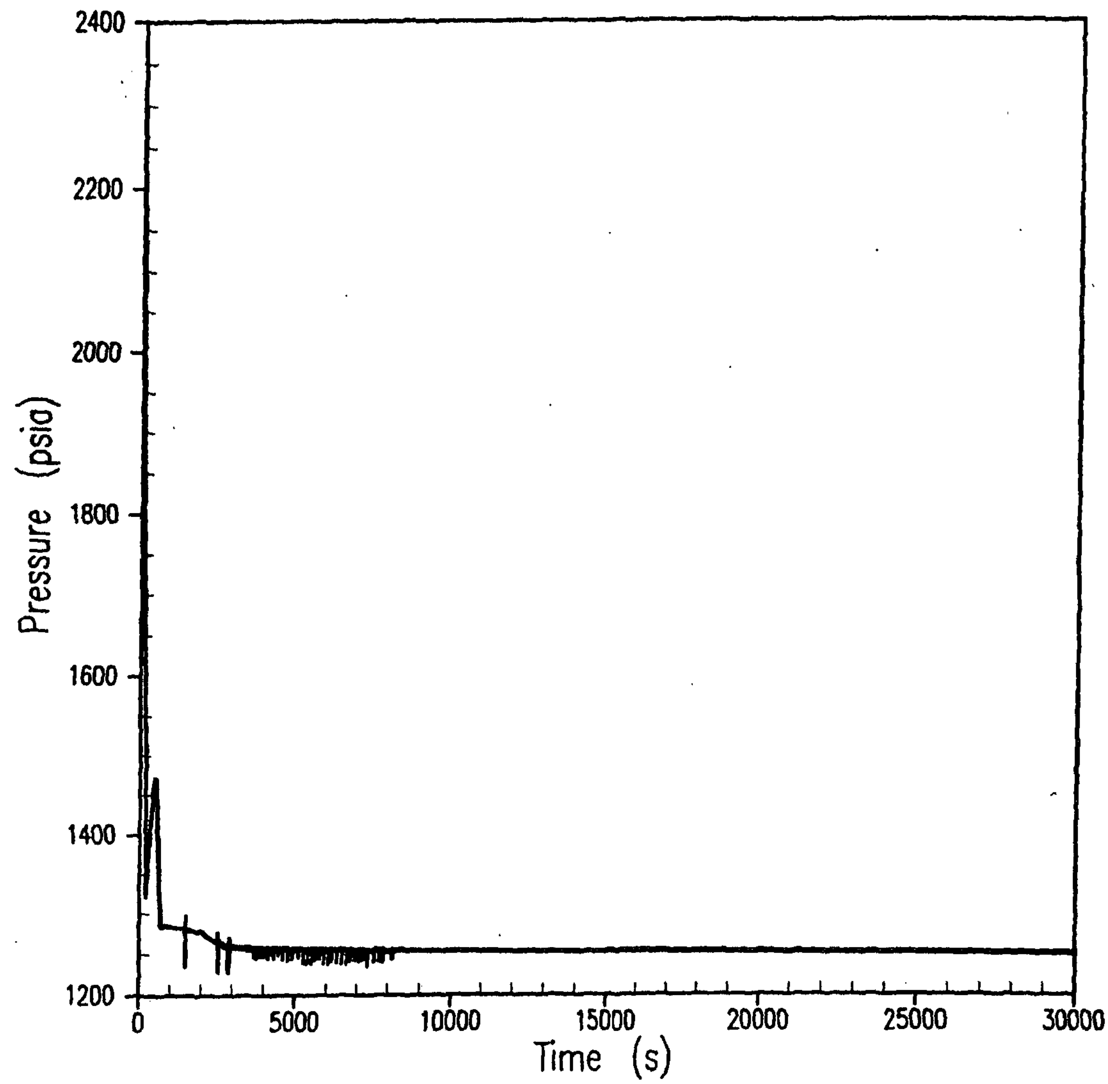
Figure 15.6-14b



Byron/Braidwood Stations Updated Final Safety Analysis Report

Heat Transfer Coefficient - Unit 1 (1.5 Inch, Reduced T_{AVG})

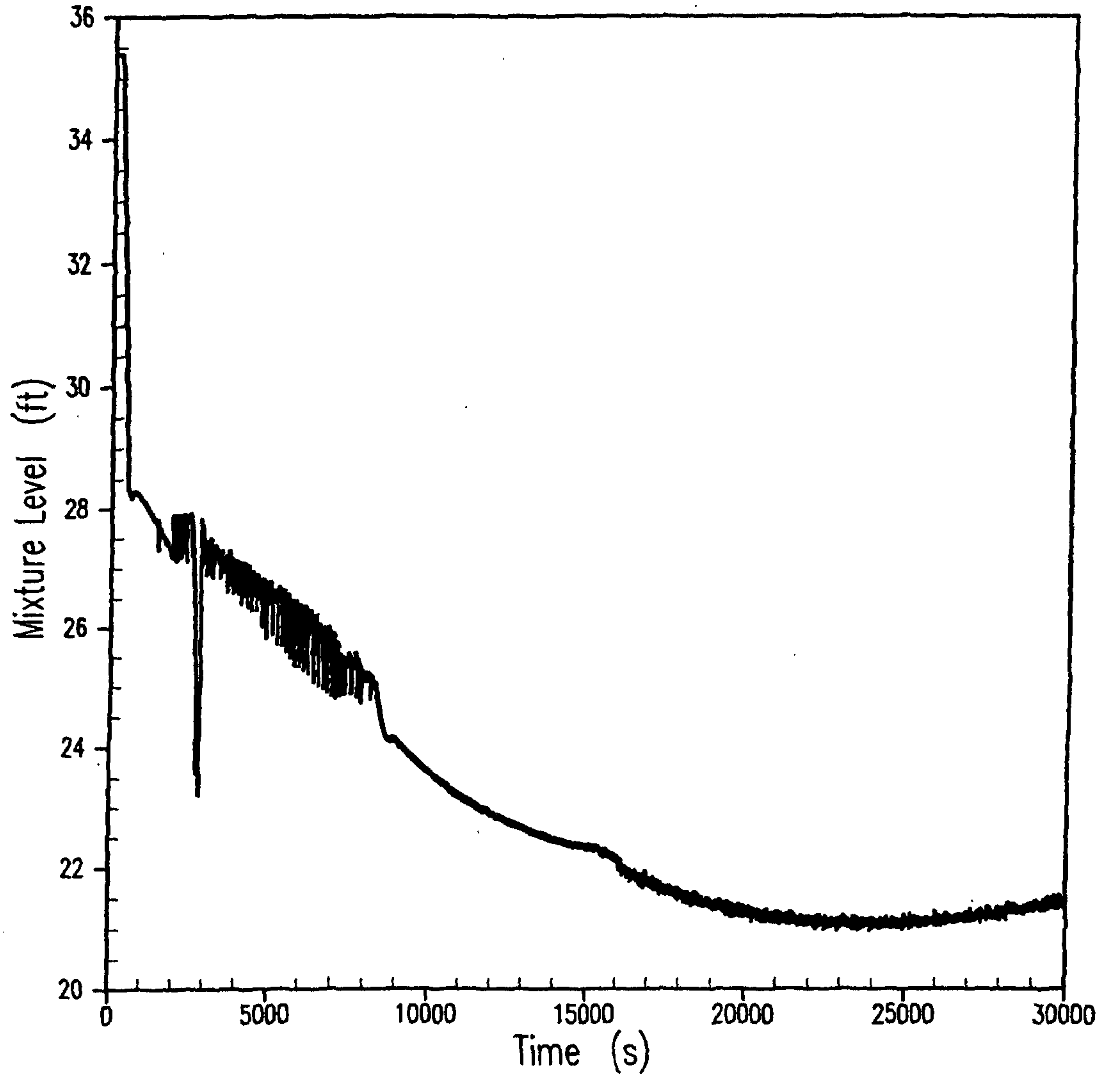
Figure 15.6-14c



Byron/Braidwood Stations Updated Final Safety Analysis Report

RCS Pressure - Unit 1
(1.5 Inch, Reduced T_{AVG})

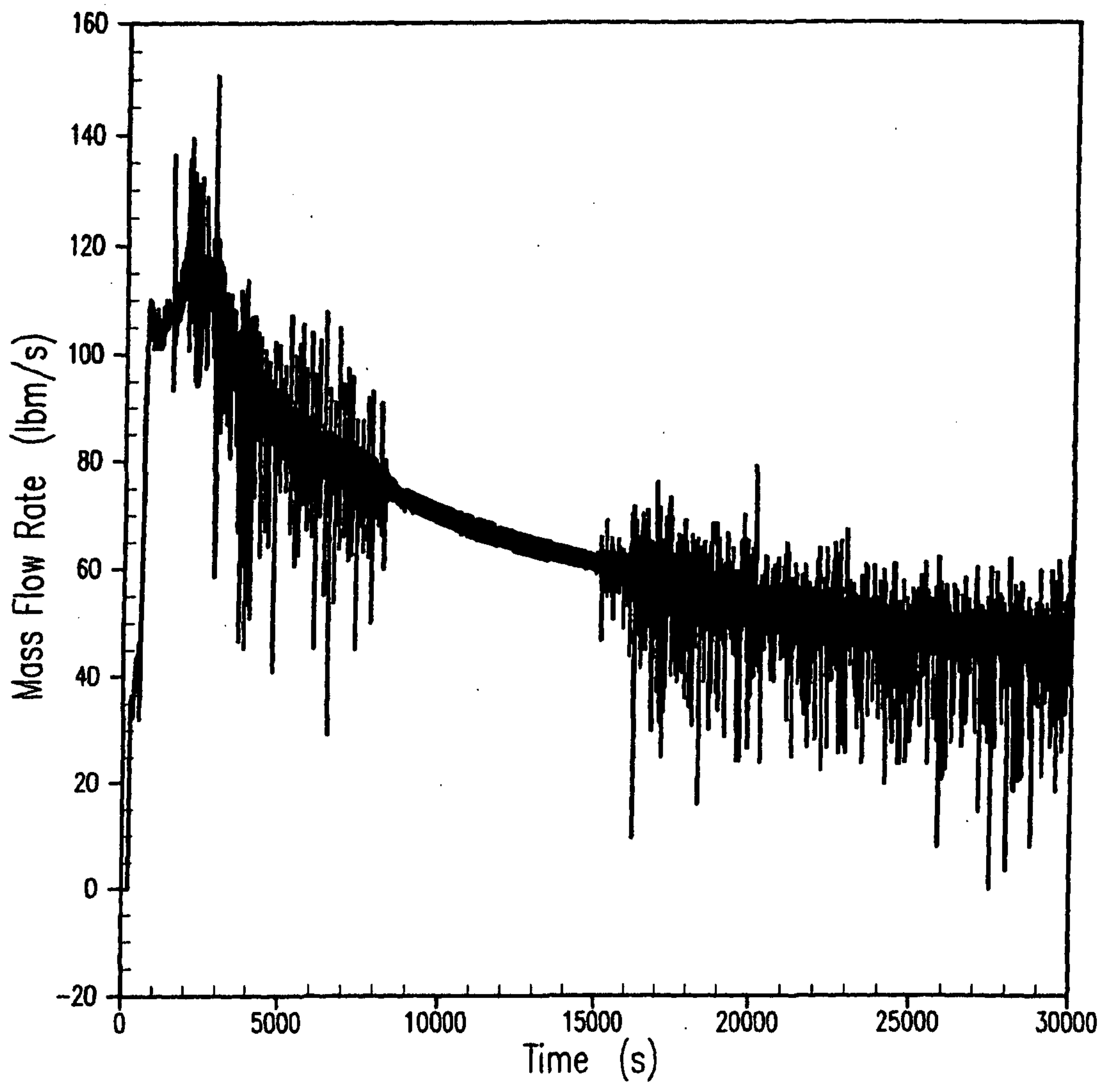
Figure 15.6-14d



Byron/Braidwood Stations
Updated Final Safety Analysis Report

Core Mixture Level - Unit 1
(1.5 Inch, Reduced T_{AVG})

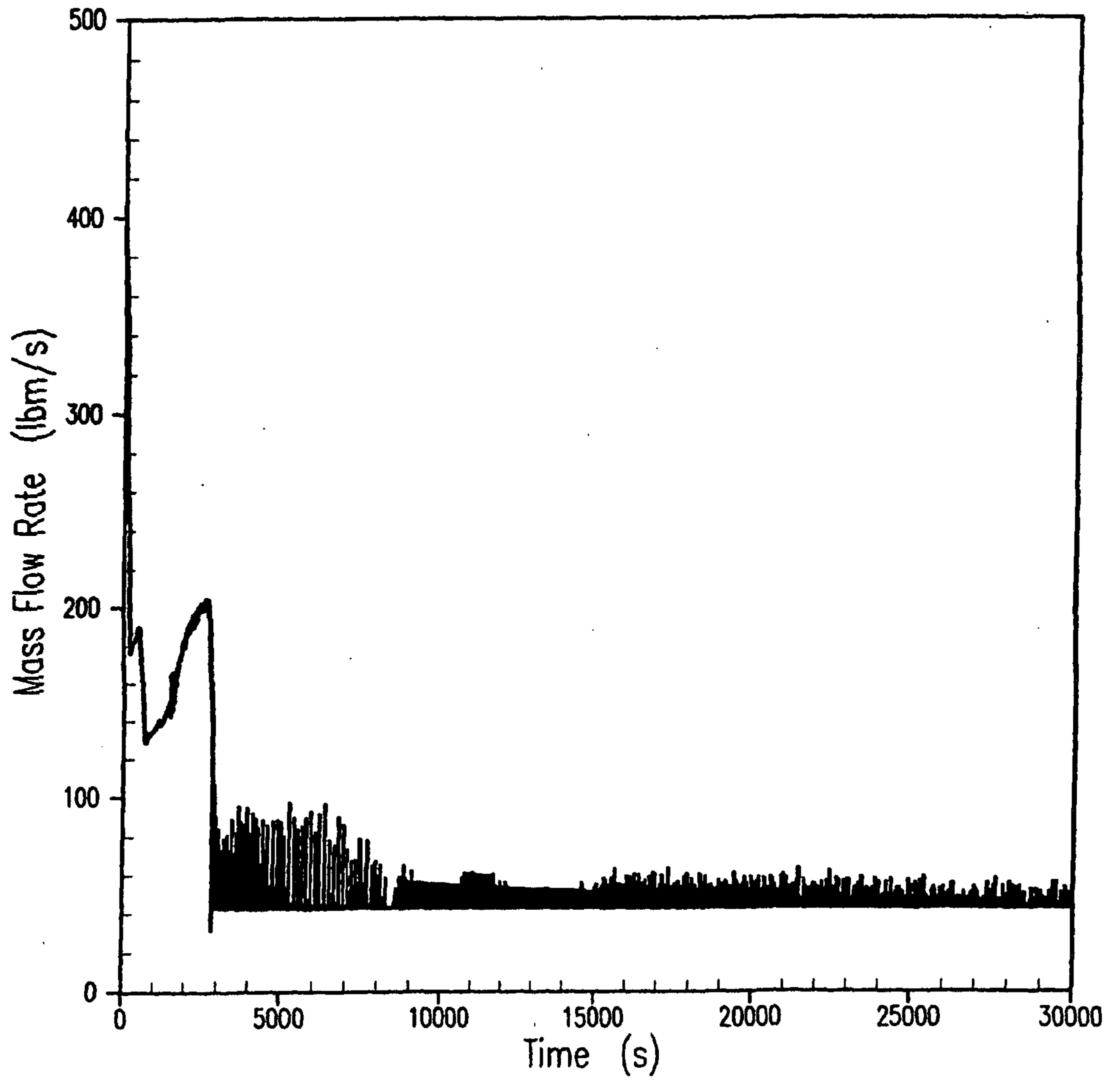
Figure 15.6-14e



Byron/Braidwood Stations Updated Final Safety Analysis Report

Core Steam Flow Rate - Unit 1 (1.5 Inch, Reduced T_{AVG})

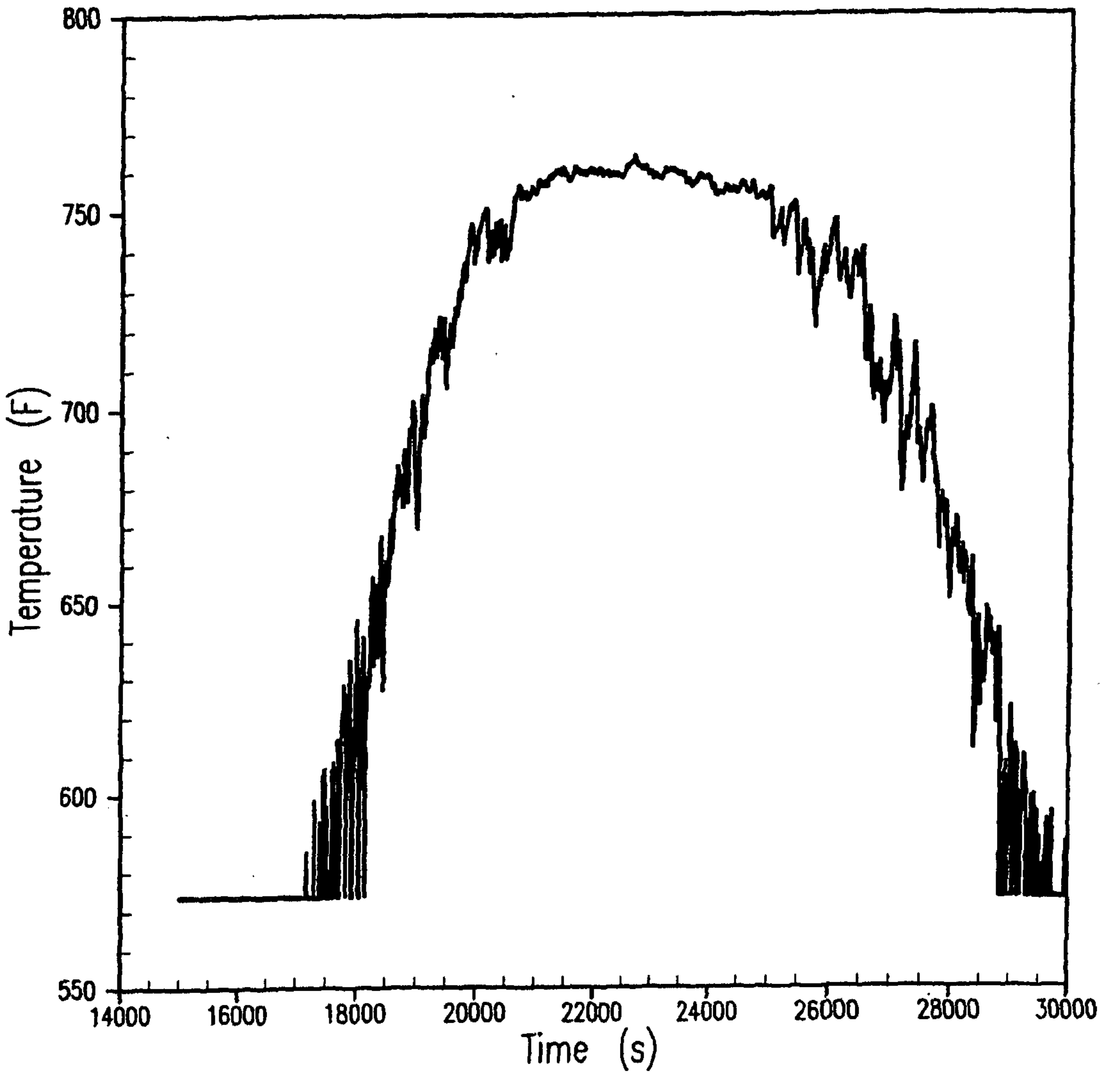
Figure 15.6-14f



Byron/Braidwood Stations Updated Final Safety Analysis Report

Break Flow - Unit 1 (1.5 Inch, Reduced T_{AVG})

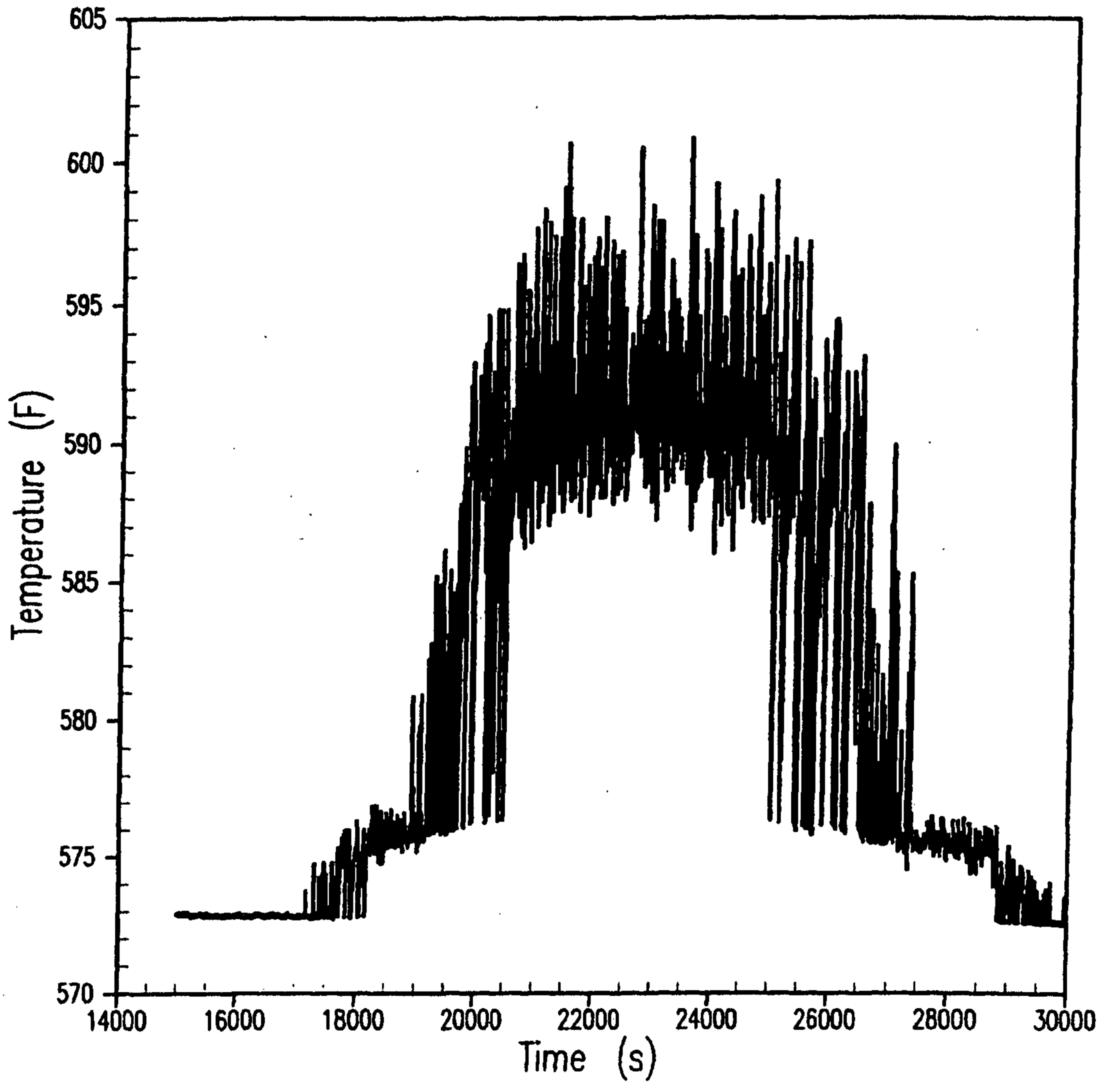
Figure 15.6-14g



Byron/Braidwood Stations Updated Final Safety Analysis Report

Peak Clad Temperature - Unit 1 (1.5 Inch, Nominal T_{AVG})

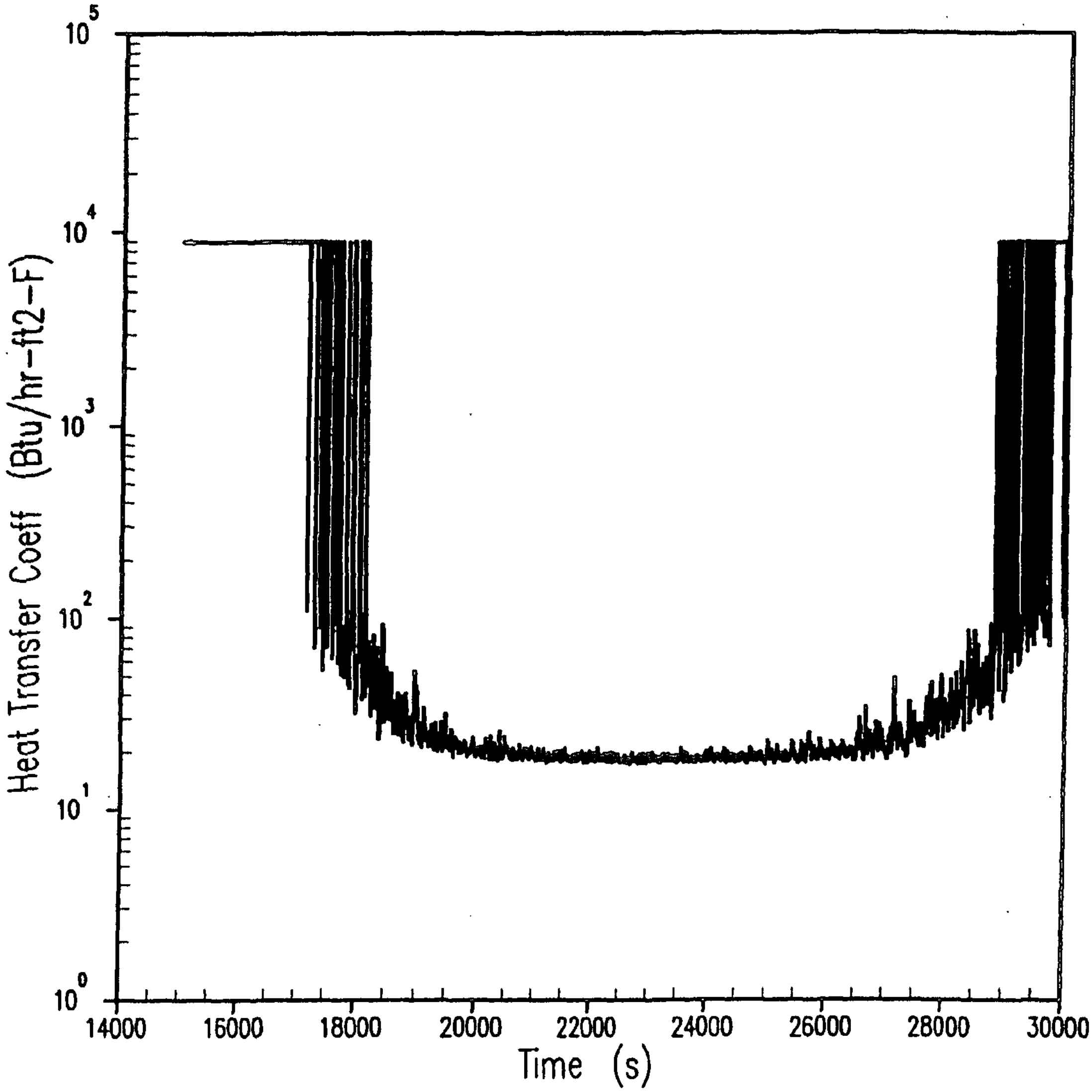
Figure 15.6-14h



Byron/Braidwood Stations
Updated Final Safety Analysis Report

Fluid Temperature - Unit 1
(1.5 Inch, Nominal T_{AVG})

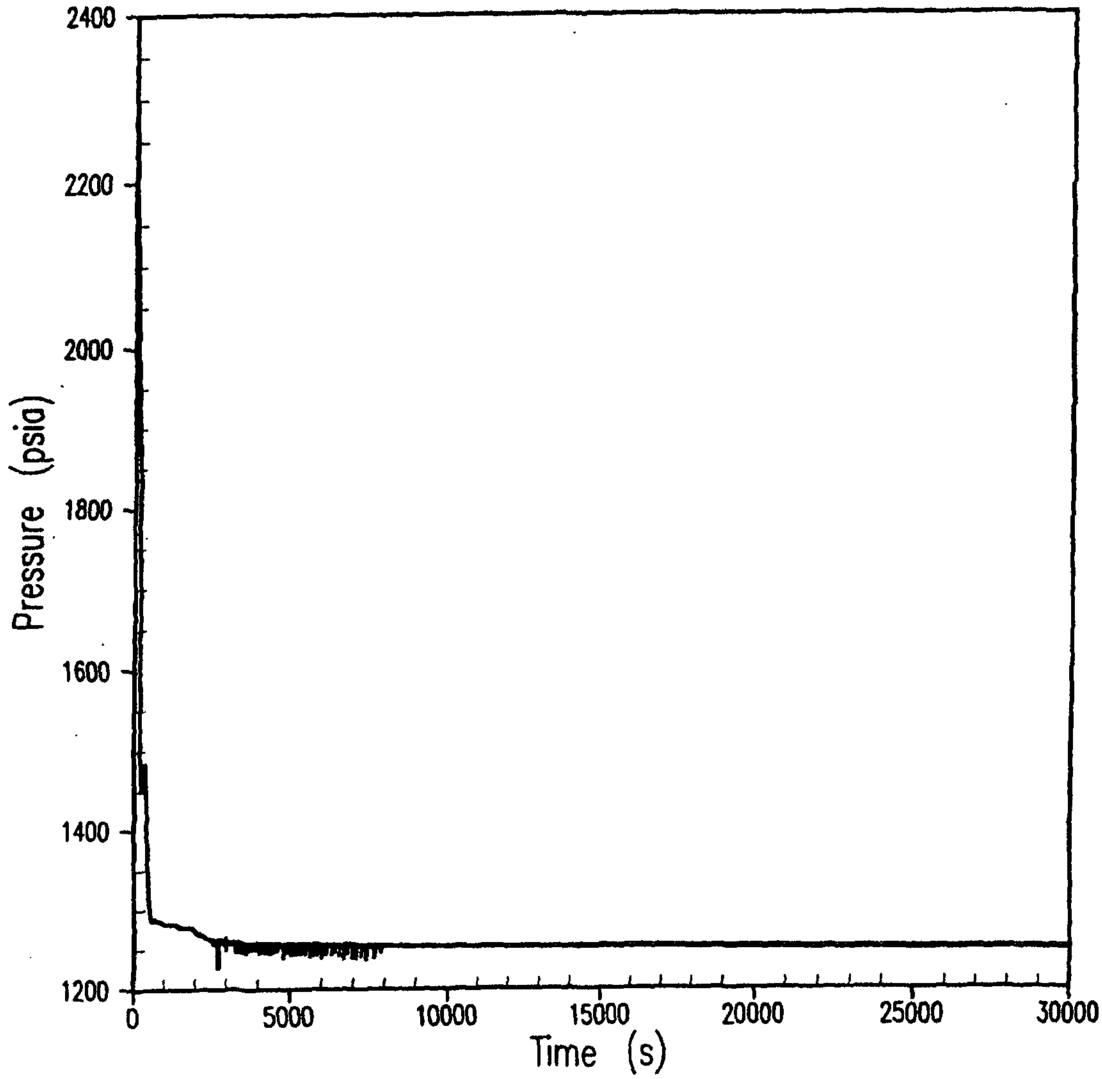
Figure 15.6-14i



Byron/Braidwood Stations Updated Final Safety Analysis Report

Heat Transfer Coefficient - Unit 1 (1.5 Inch, Nominal T_{AVG})

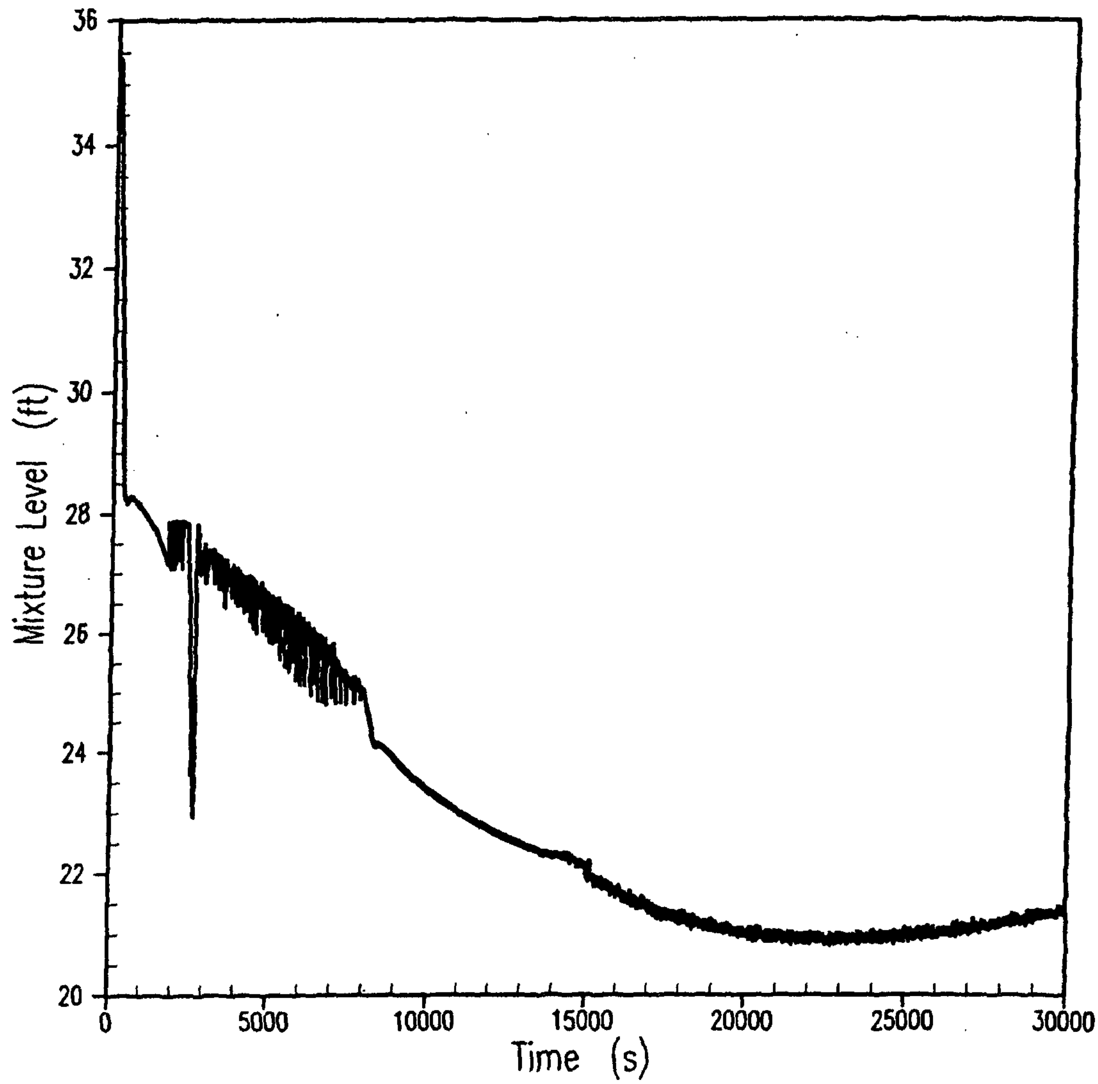
Figure 15.6-14j



Byron/Braidwood Stations Updated Final Safety Analysis Report

RCS Pressure - Unit 1
(1.5 Inch, Nominal T_{AVG})

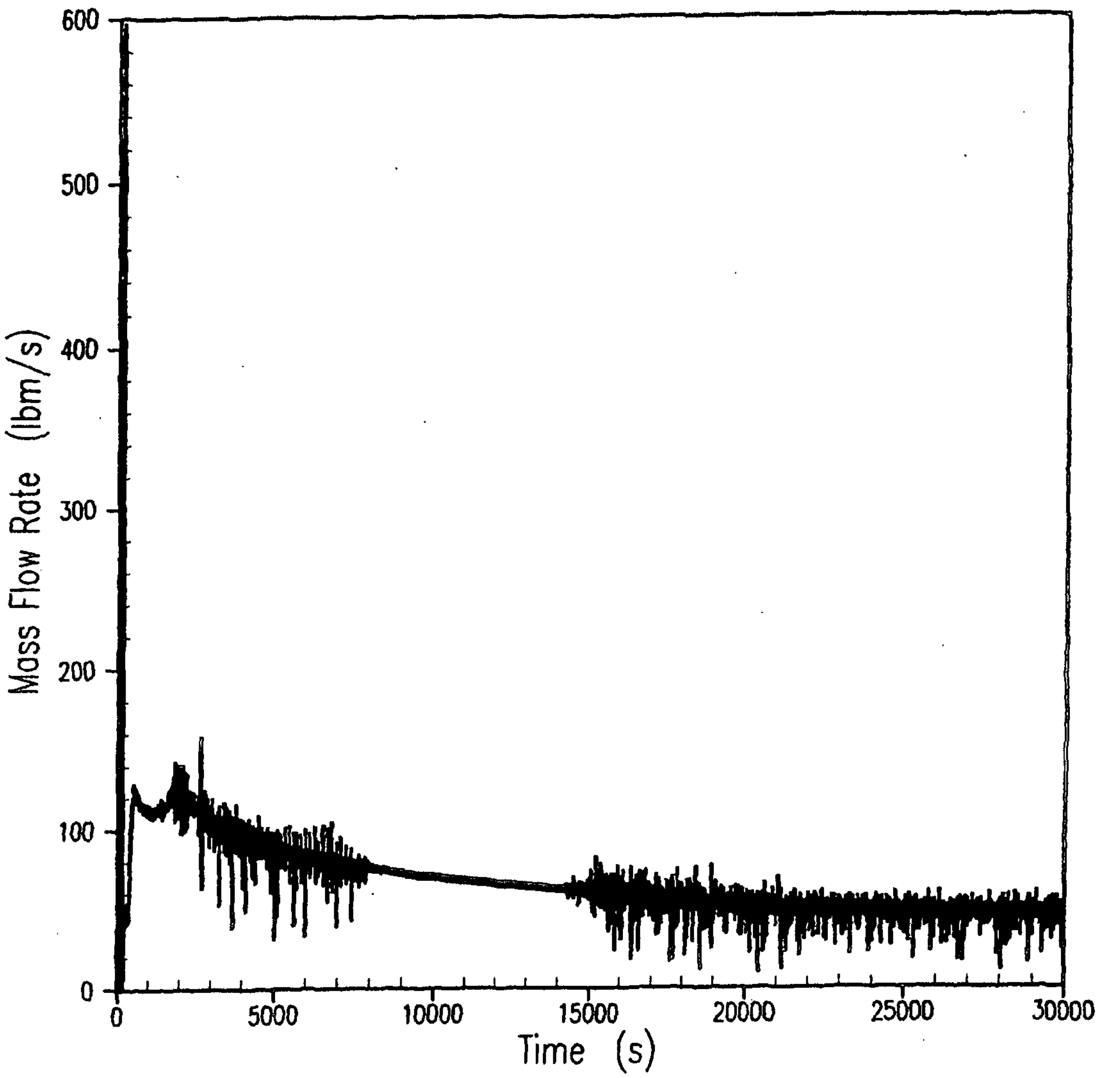
Figure 15.6-14k



Byron/Braidwood Stations Updated Final Safety Analysis Report

Core Mixture Level - Unit 1 (1.5 Inch, Nominal T_{AVG})

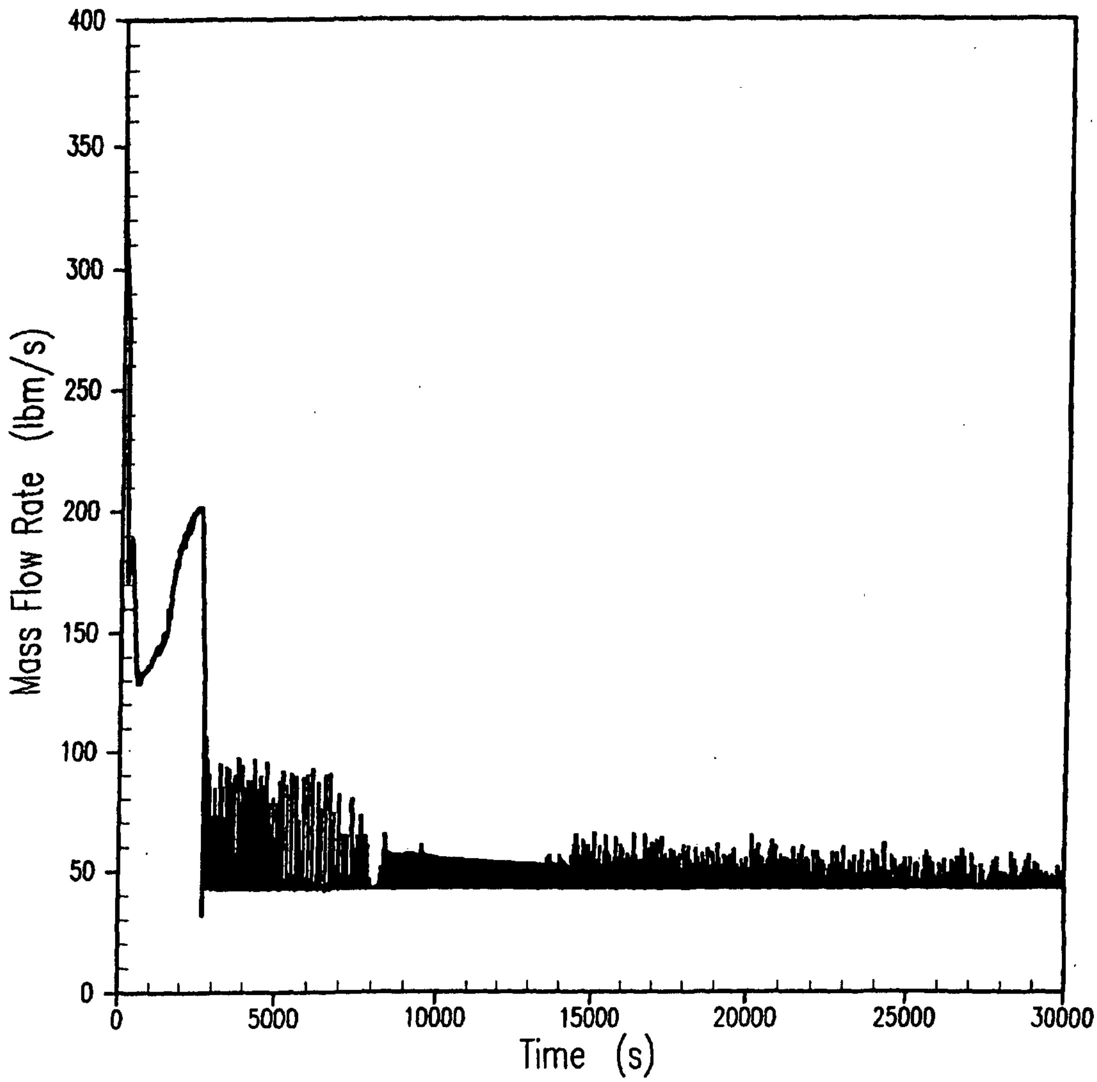
Figure 15.6-14I



Byron/Braidwood Stations Updated Final Safety Analysis Report

Core Steam Flow Rate - Unit 1 (1.5 Inch, Nominal T_{AVG})

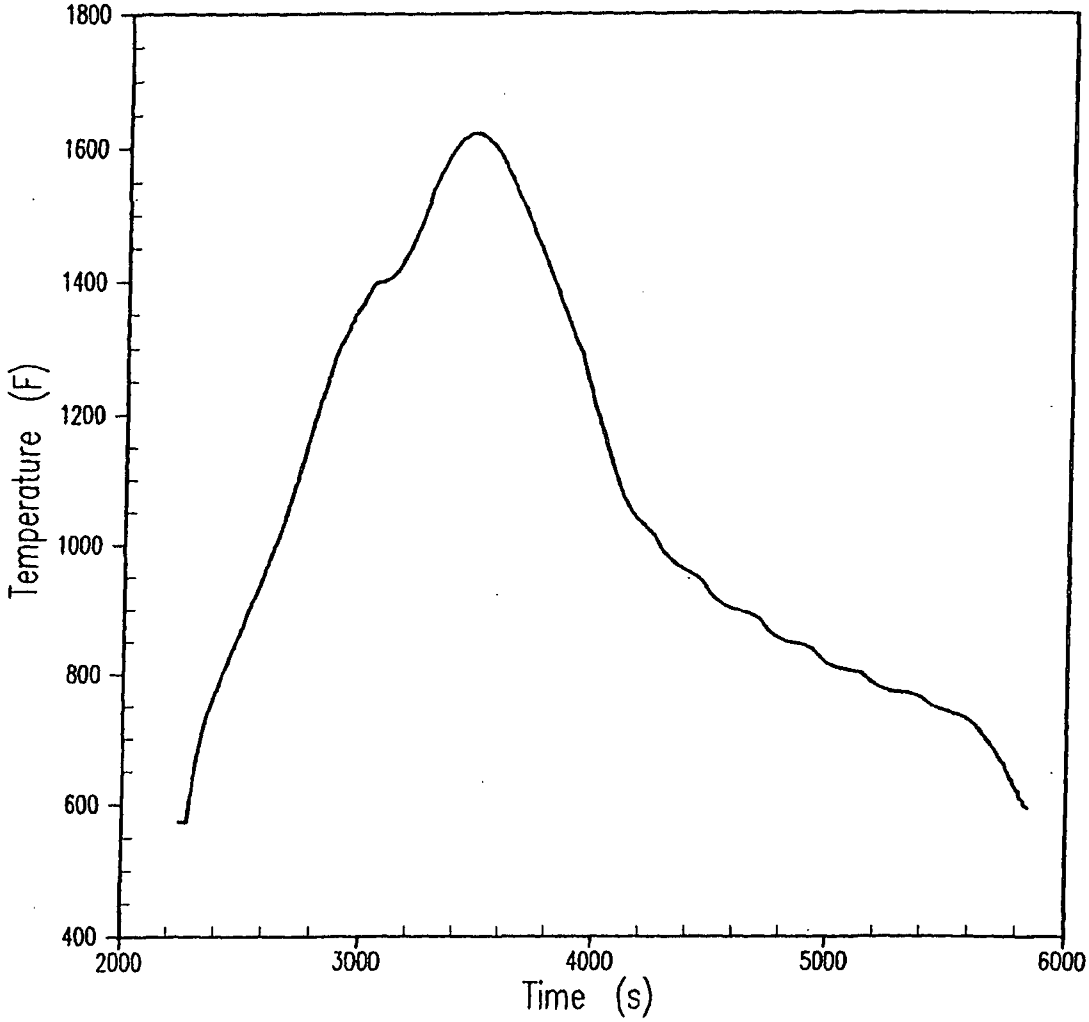
Figure 15.6-14m



**Byron/Braidwood Stations
Updated Final Safety Analysis
Report**

**Break Flow - Unit 1
(1.5 Inch, Nominal T_{AVG})**

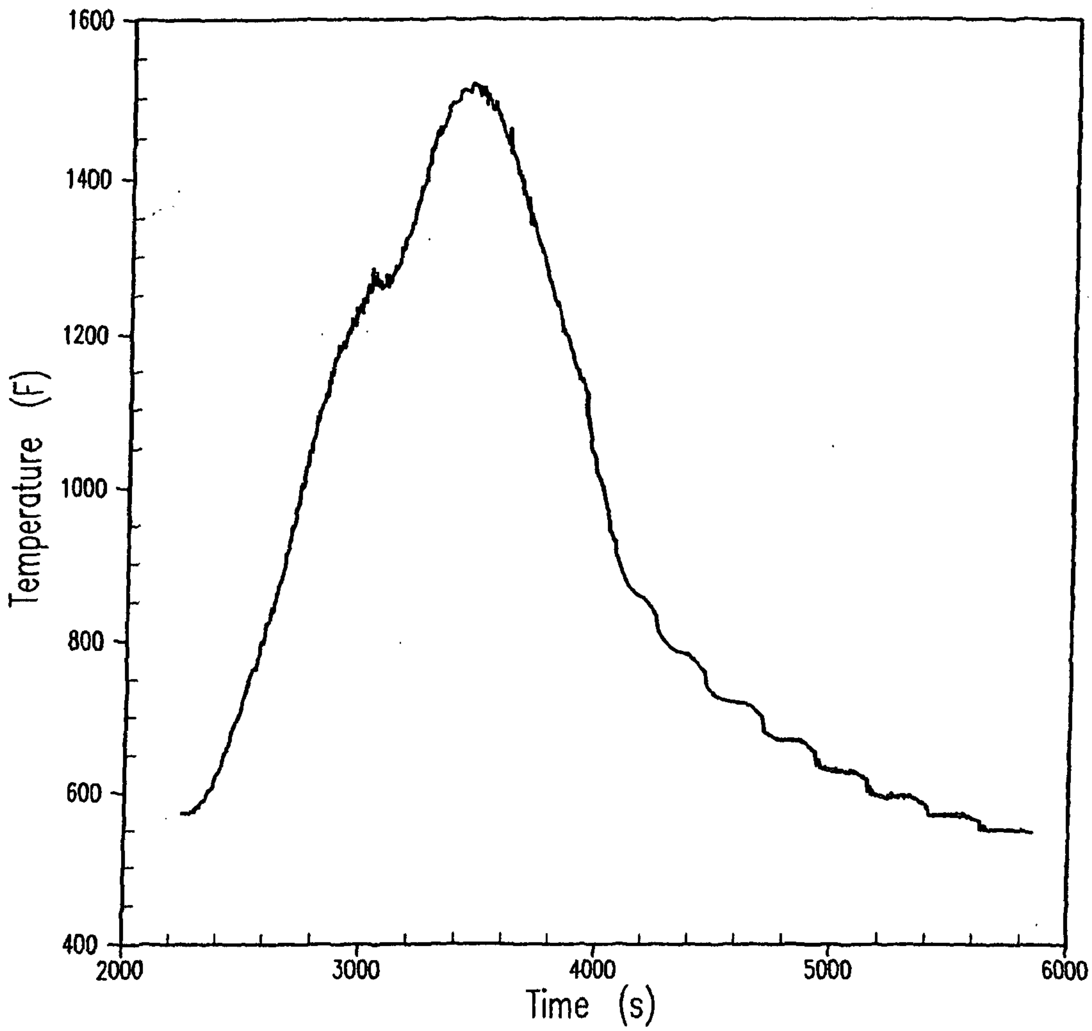
Figure 15.6-14n



Byron/Braidwood Stations Updated Final Safety Analysis Report

Peak Clad Temperature - Unit 1 (2 Inch, Reduced T_{AVG})

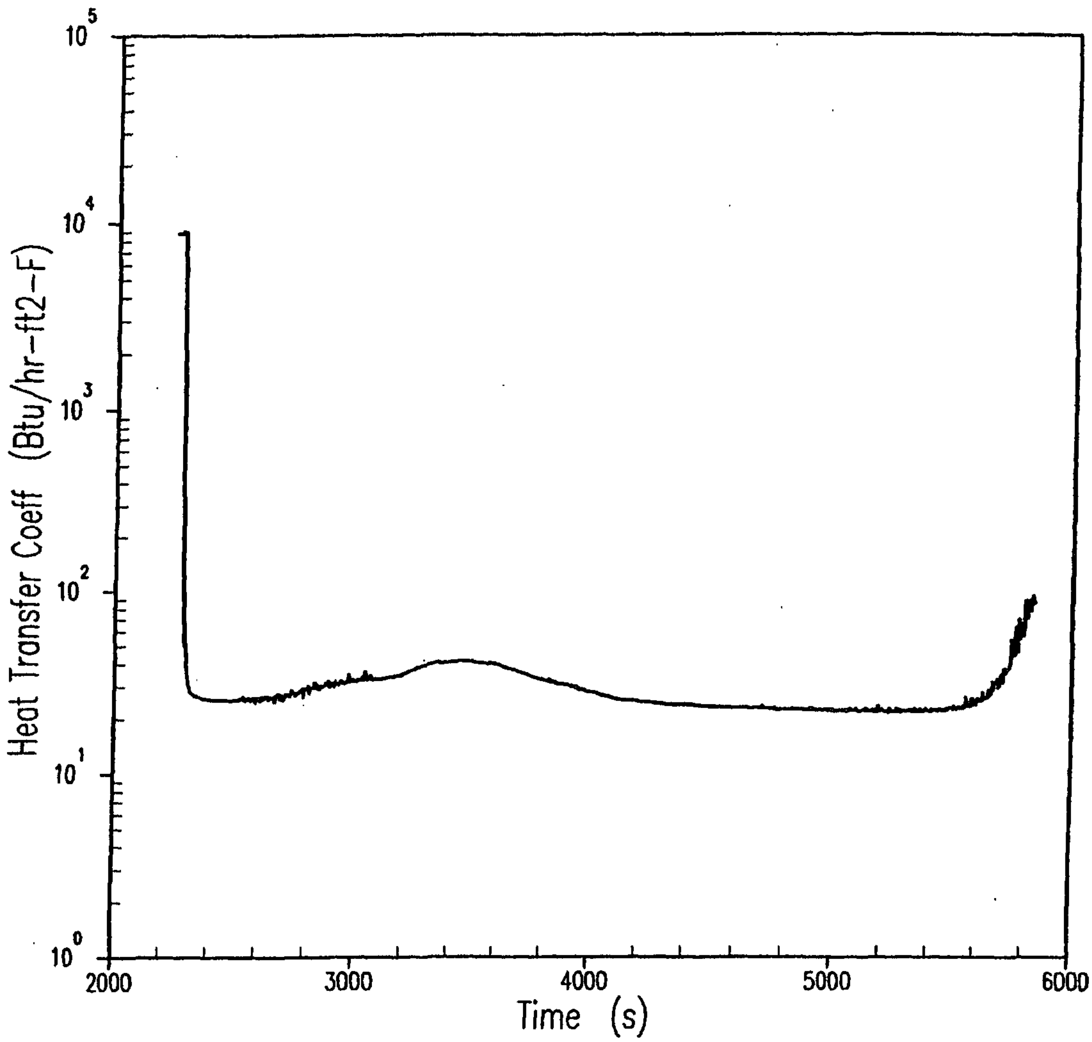
Figure 15.6-15a



Byron/Braidwood Stations
Updated Final Safety Analysis Report

Fluid Temperature - Unit 1
(2 Inch, Reduced T_{AVG})

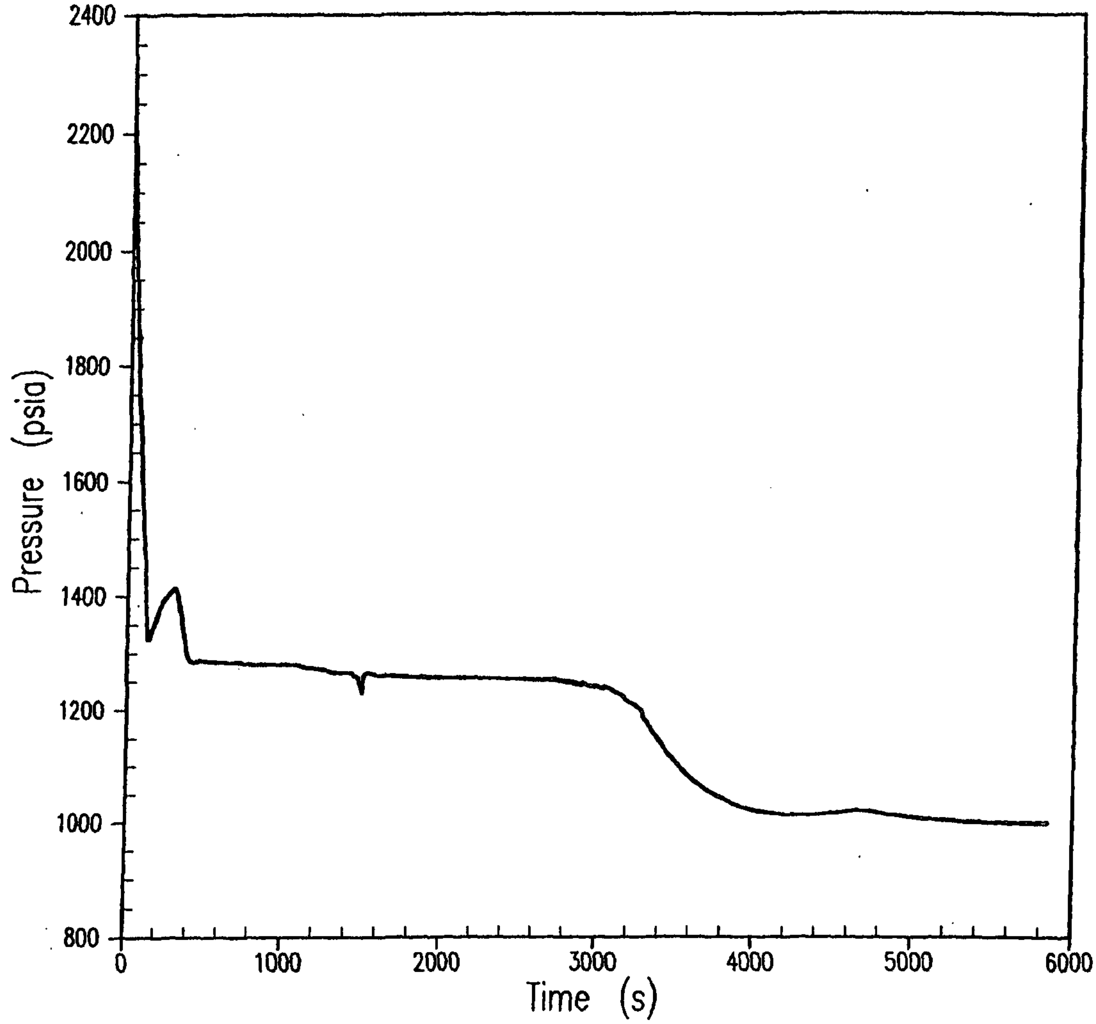
Figure 15.6-15b



**Byron/Braidwood Stations
Updated Final Safety Analysis
Report**

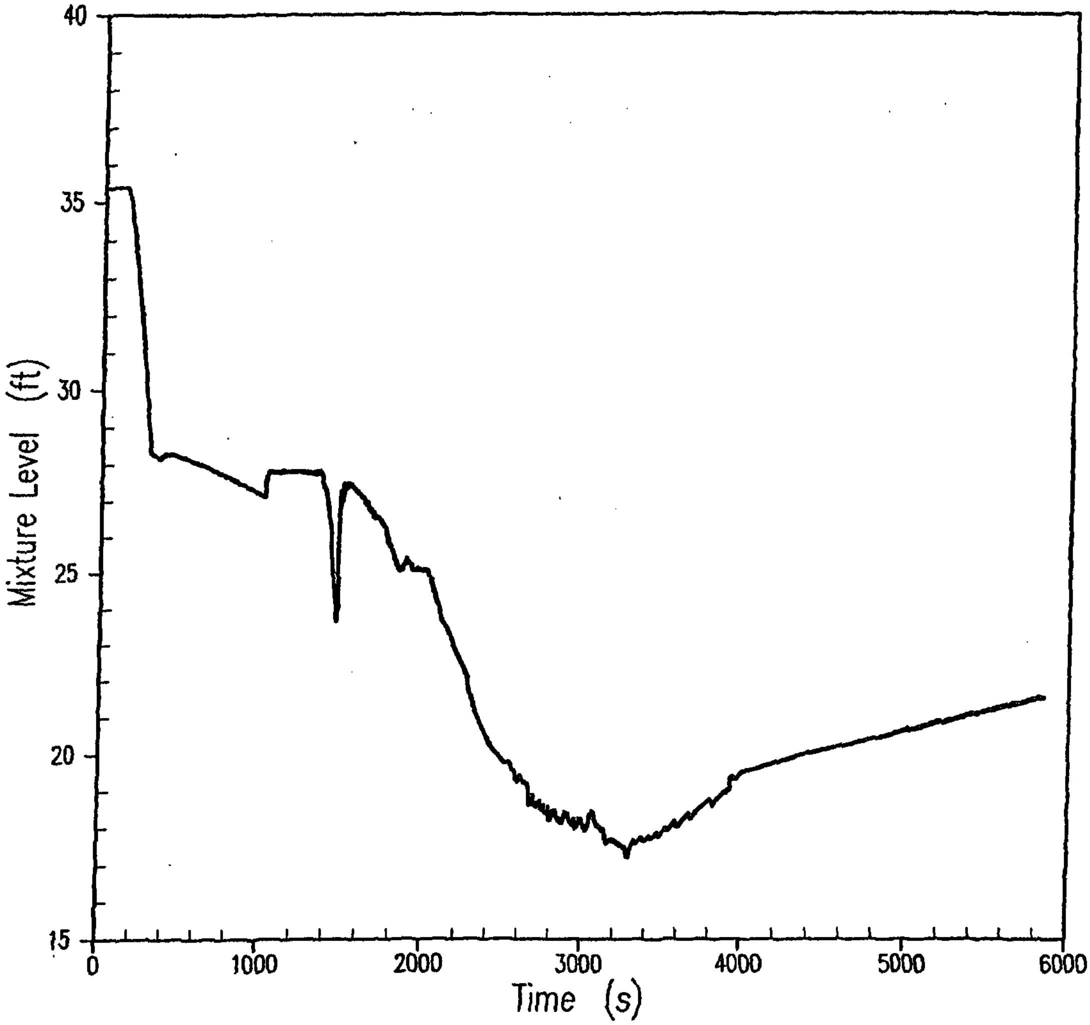
**Heat Transfer Coefficient - Unit 1
(2 Inch, Reduced T_{AVG})**

Figure 15.6-15c



RCS Pressure - Unit 1
(2 Inch, Reduced T_{AVG})

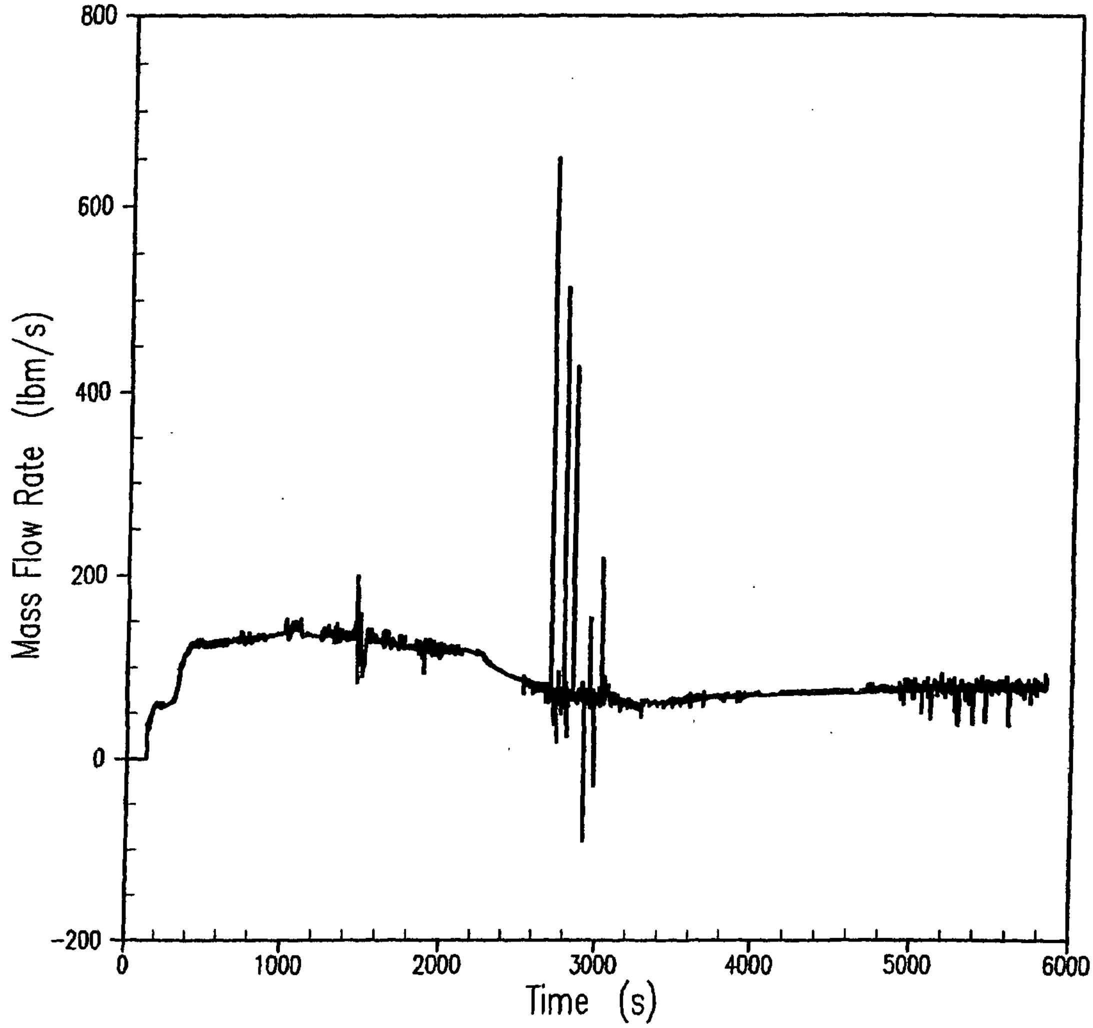
Figure 15.6-15d



Byron/Braidwood Stations Updated Final Safety Analysis Report

Core Mixture Level - Unit 1
(2 Inch, Reduced T_{AVG})

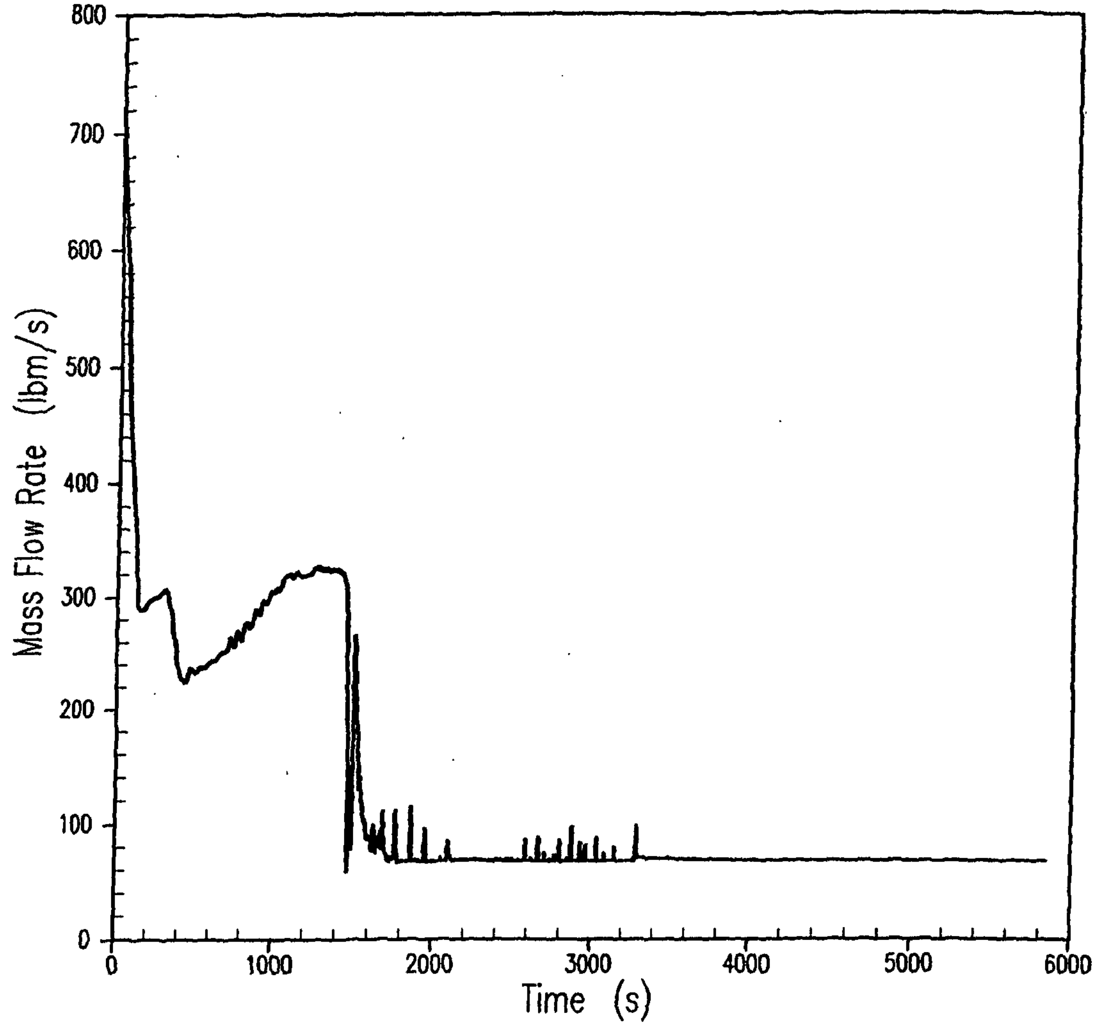
Figure 15.6-15e



Byron/Braidwood Stations Updated Final Safety Analysis Report

Core Steam Flow Rate - Unit 1
(2 Inch, Reduced T_{AVG})

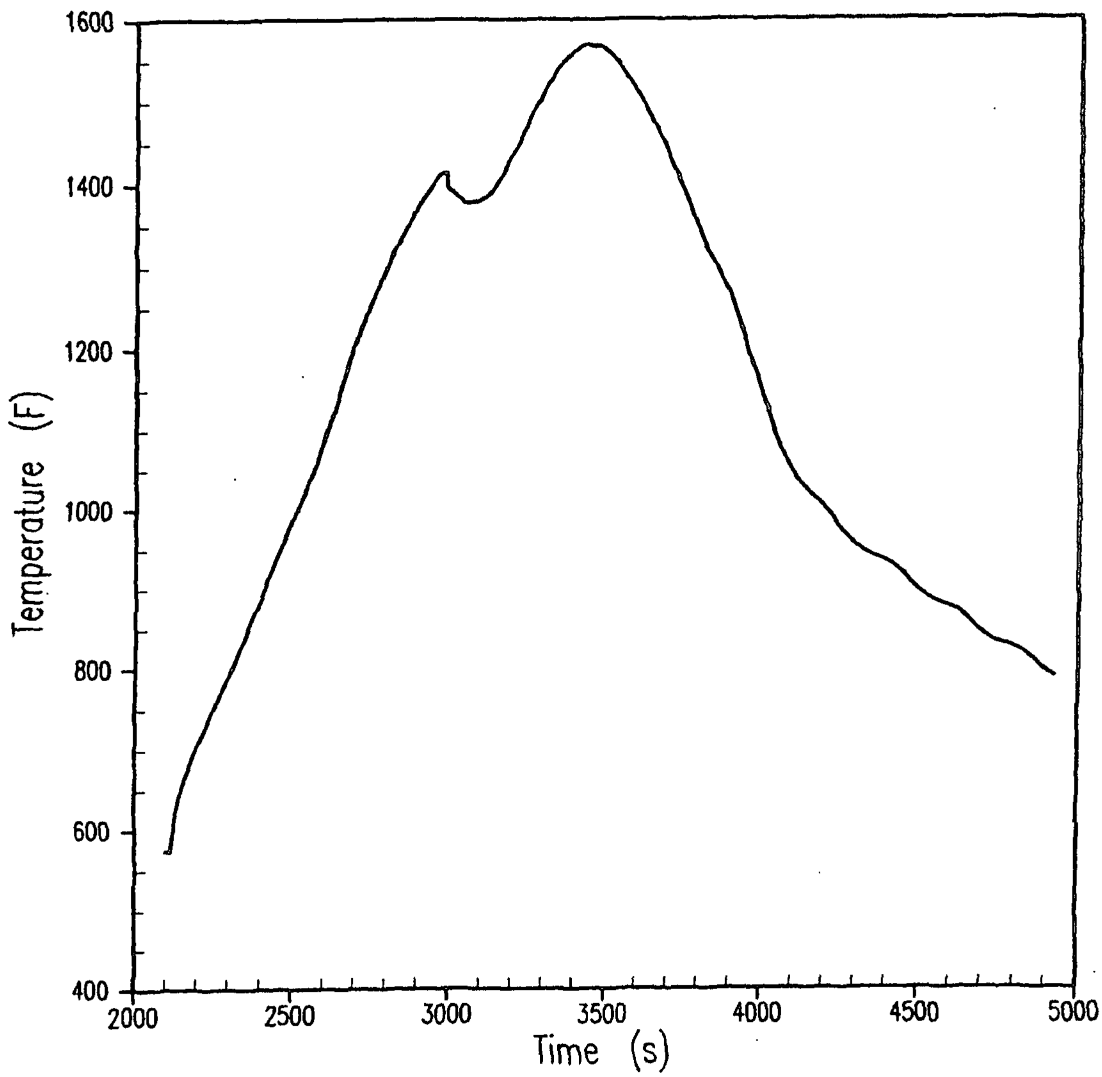
Figure 15.6-15f



Byron/Braidwood Stations Updated Final Safety Analysis Report

Break Flow - Unit 1
(2 Inch, Reduced T_{AVG})

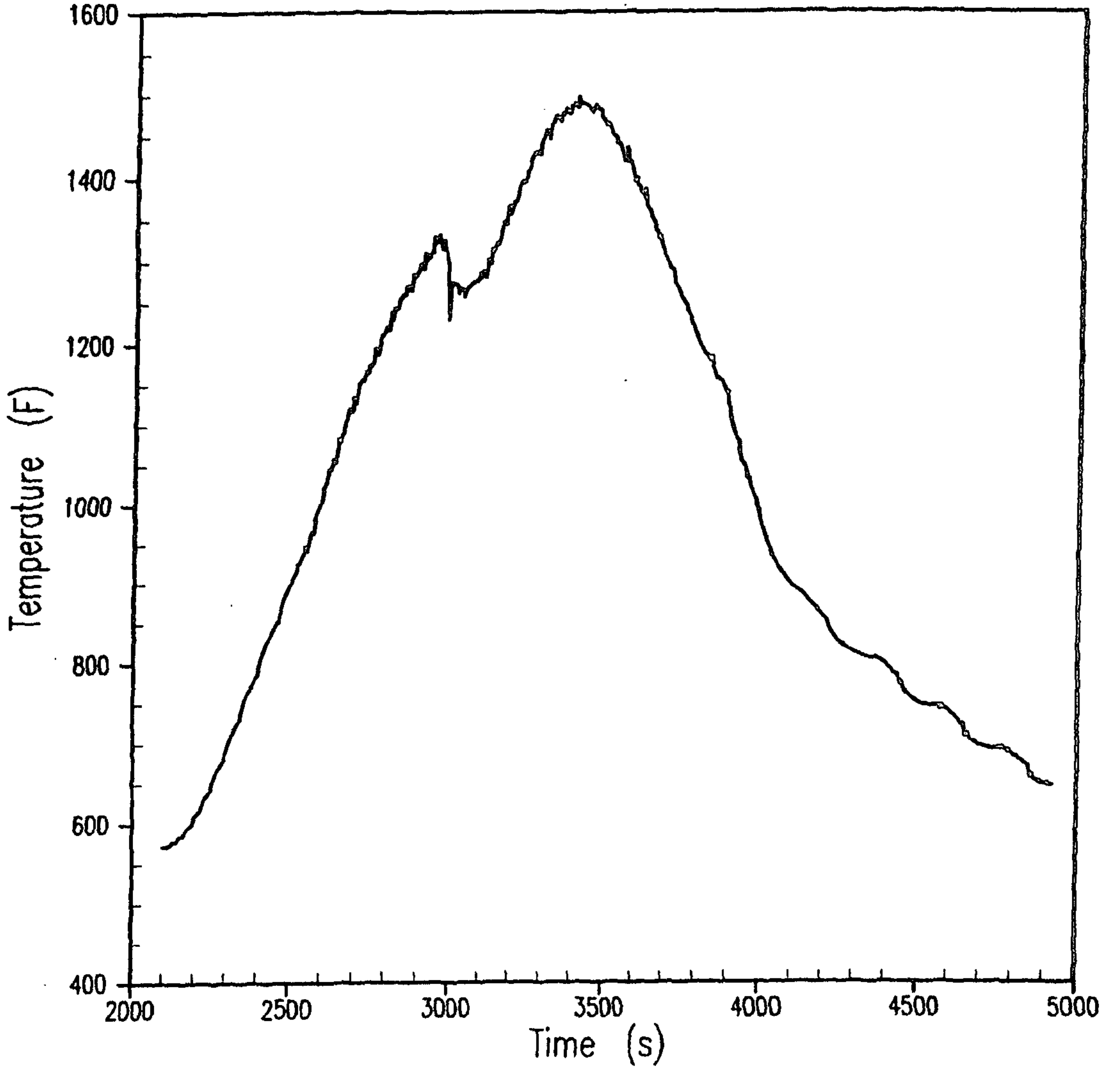
Figure 15.6-15g



**Byron/Braidwood Stations
Updated Final Safety Analysis
Report**

**Peak Clad Temperature - Unit 1
(2 Inch, Nominal T_{AVG})**

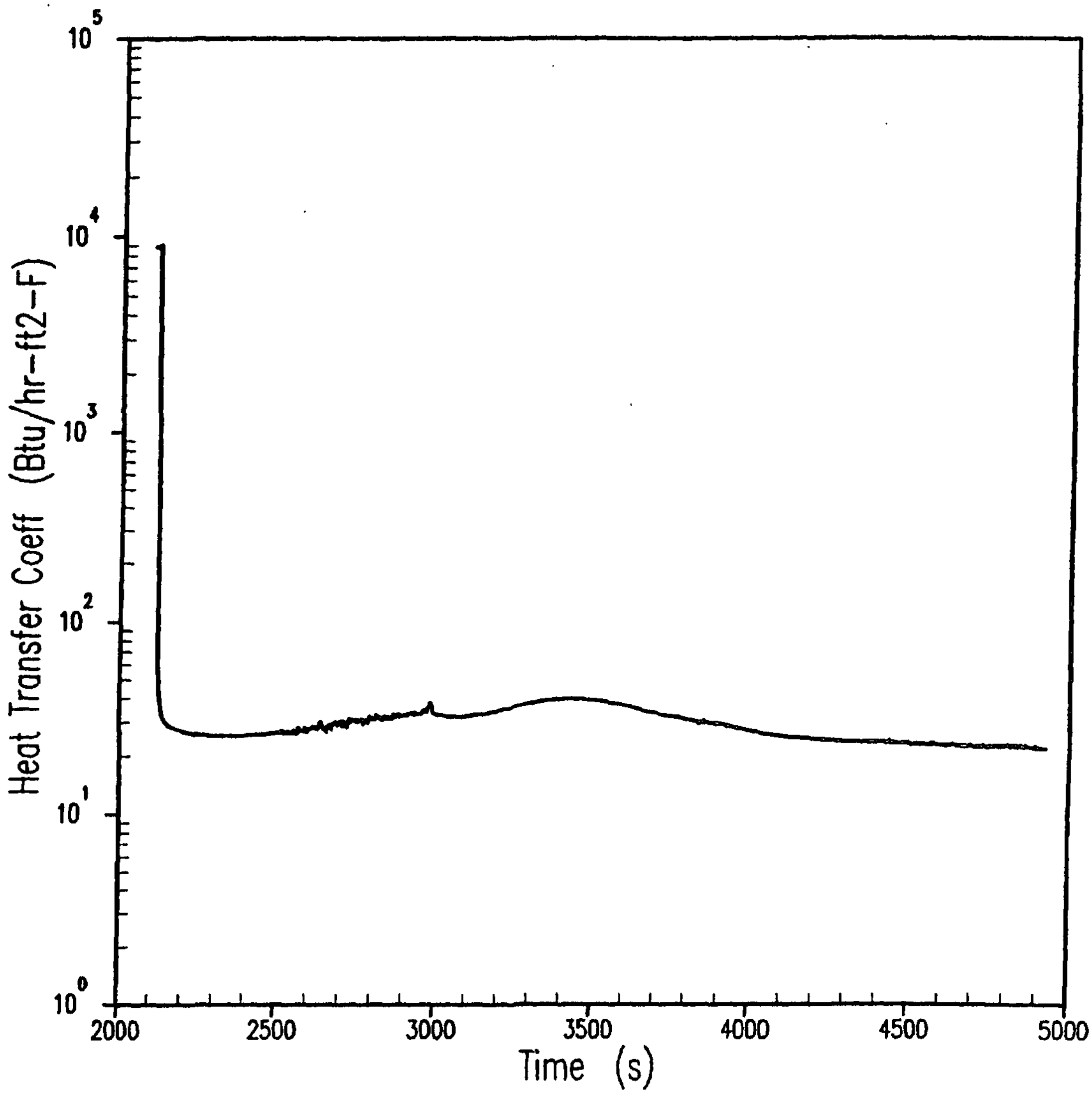
Figure 15.6-15h



**Byron/Braidwood Stations
Updated Final Safety Analysis Report**

**Fluid Temperature - Unit 1
(2 Inch, Nominal T_{AVG})**

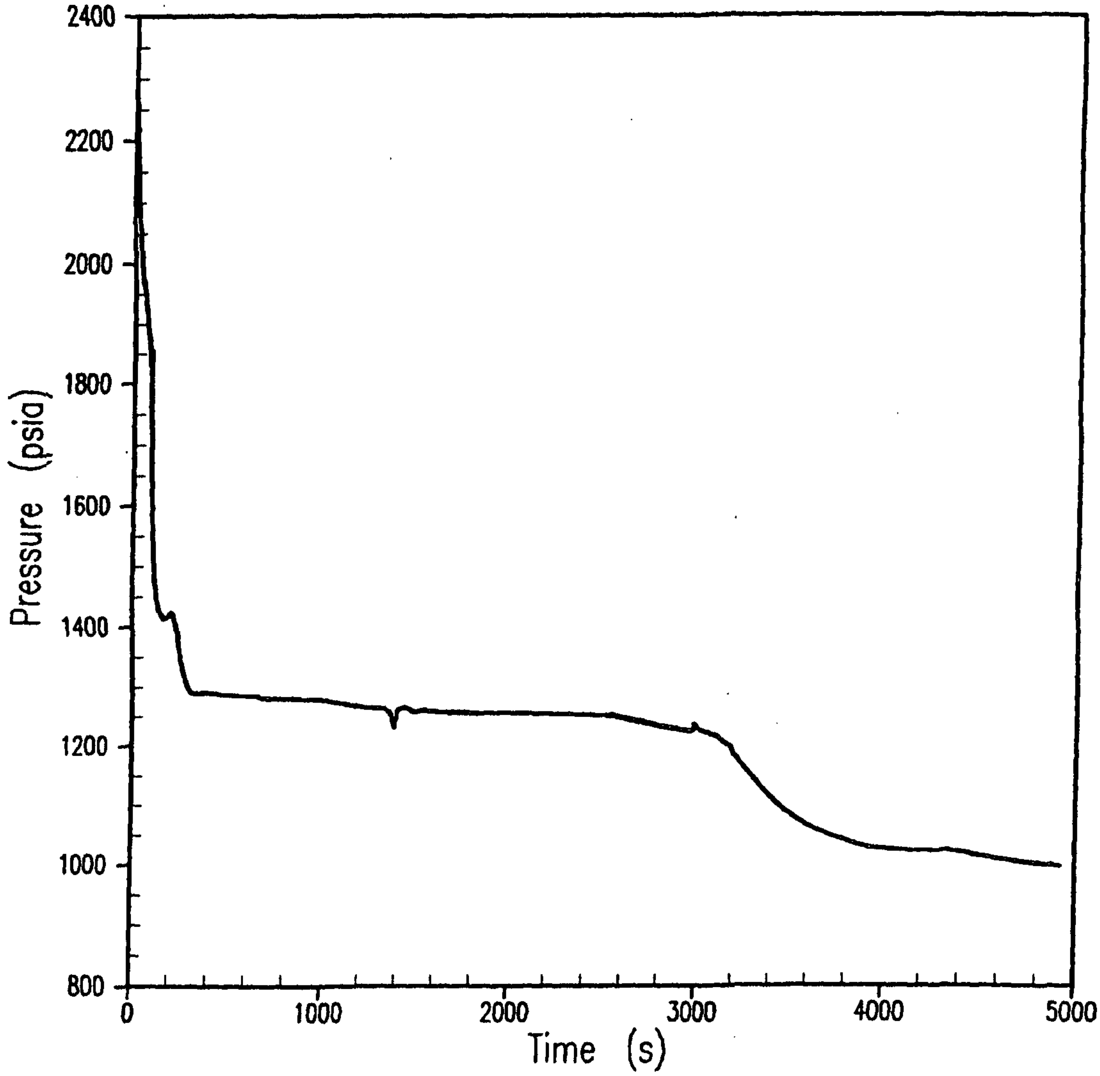
Figure 15.6-15i



**Byron/Braidwood Stations
Updated Final Safety Analysis
Report**

**Heat Transfer Coefficient - Unit 1
(2 Inch, Nominal T_{AVG})**

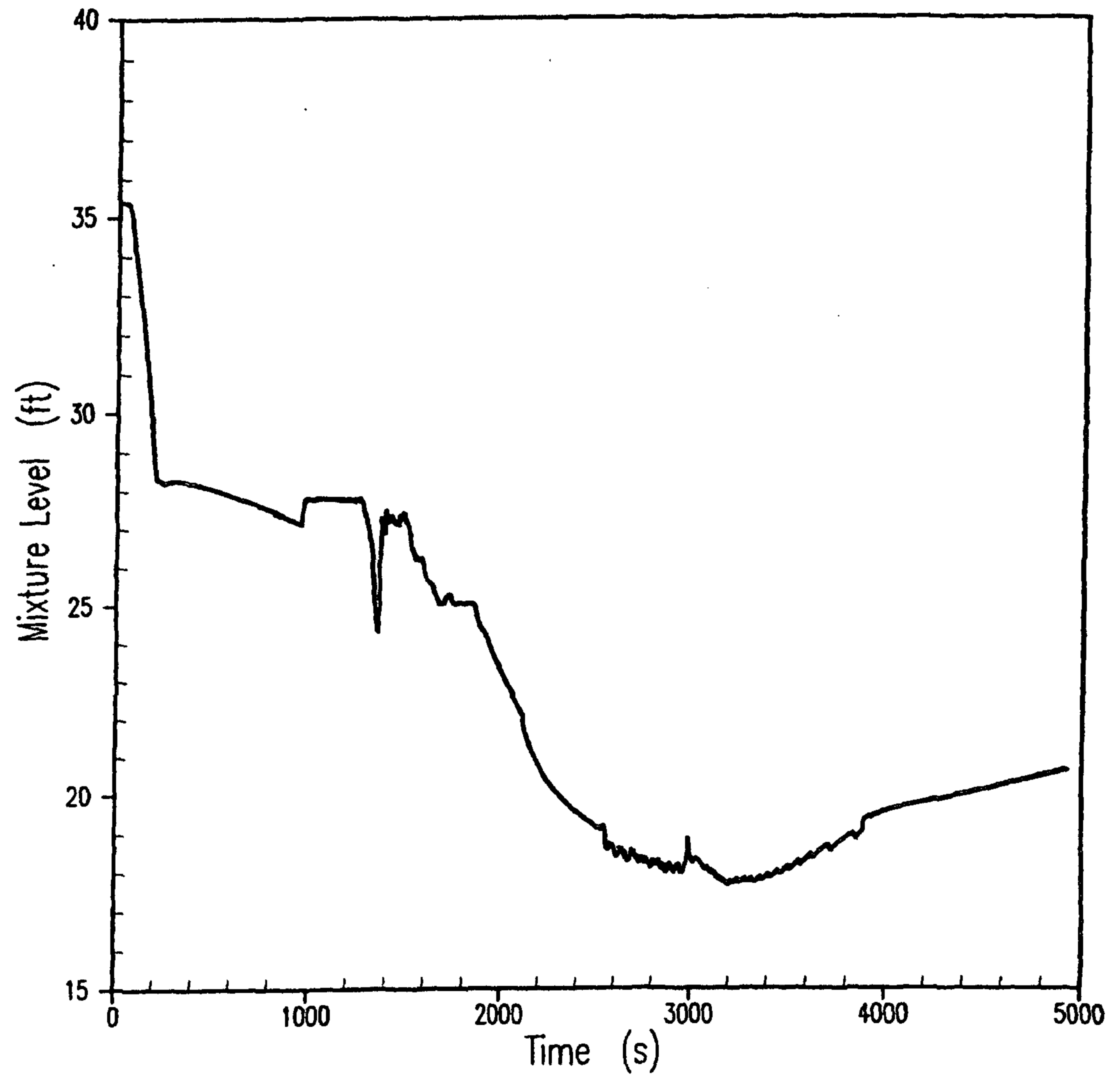
Figure 15.6-15j



**Byron/Braidwood Stations
Updated Final Safety Analysis
Report**

**RCS Pressure - Unit 1
(2 Inch, Nominal T_{AVG})**

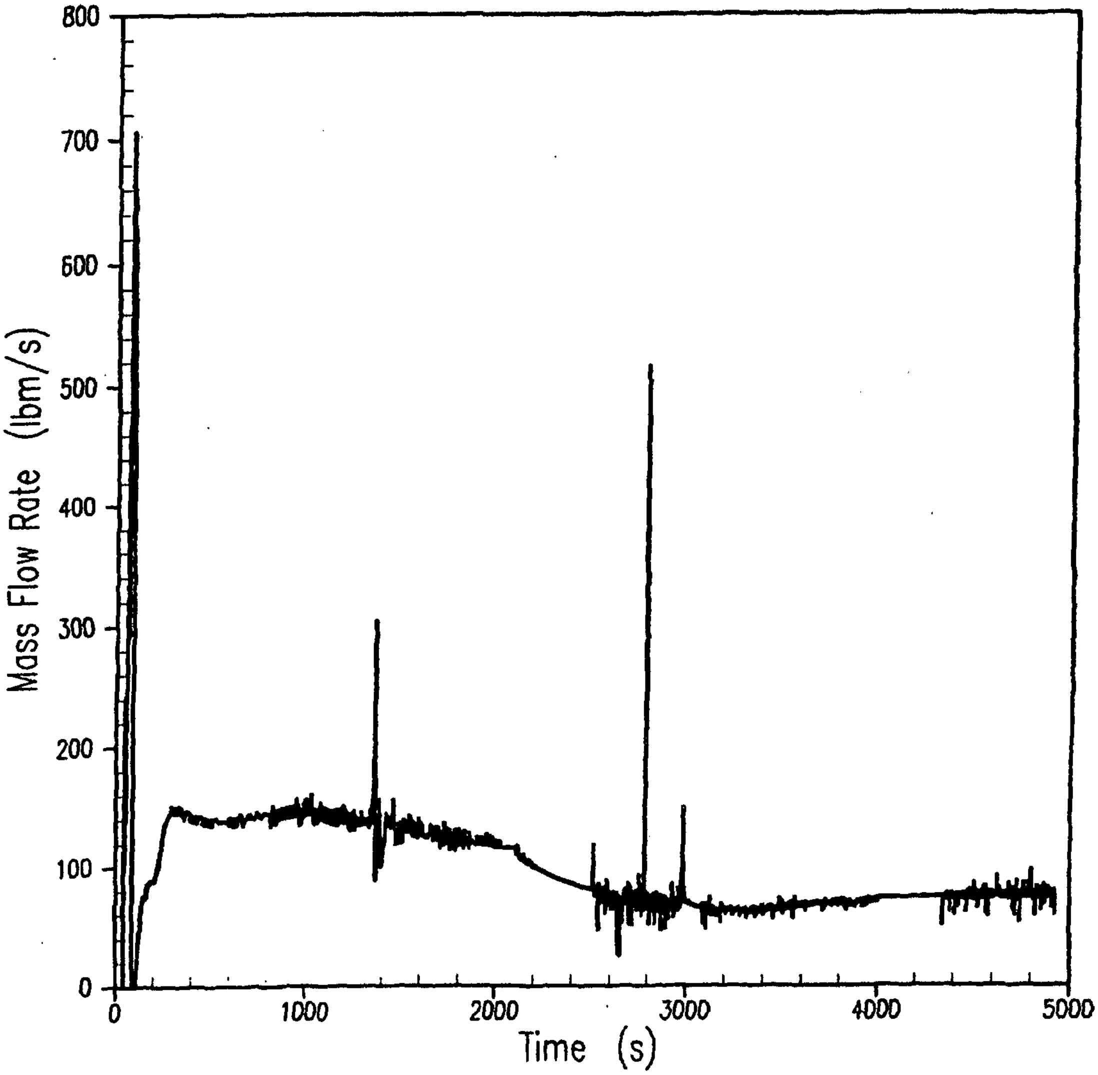
Figure 15.6-15k



Byron/Braidwood Stations Updated Final Safety Analysis Report

Core Mixture Level - Unit 1
(2 Inch, Nominal T_{AVG})

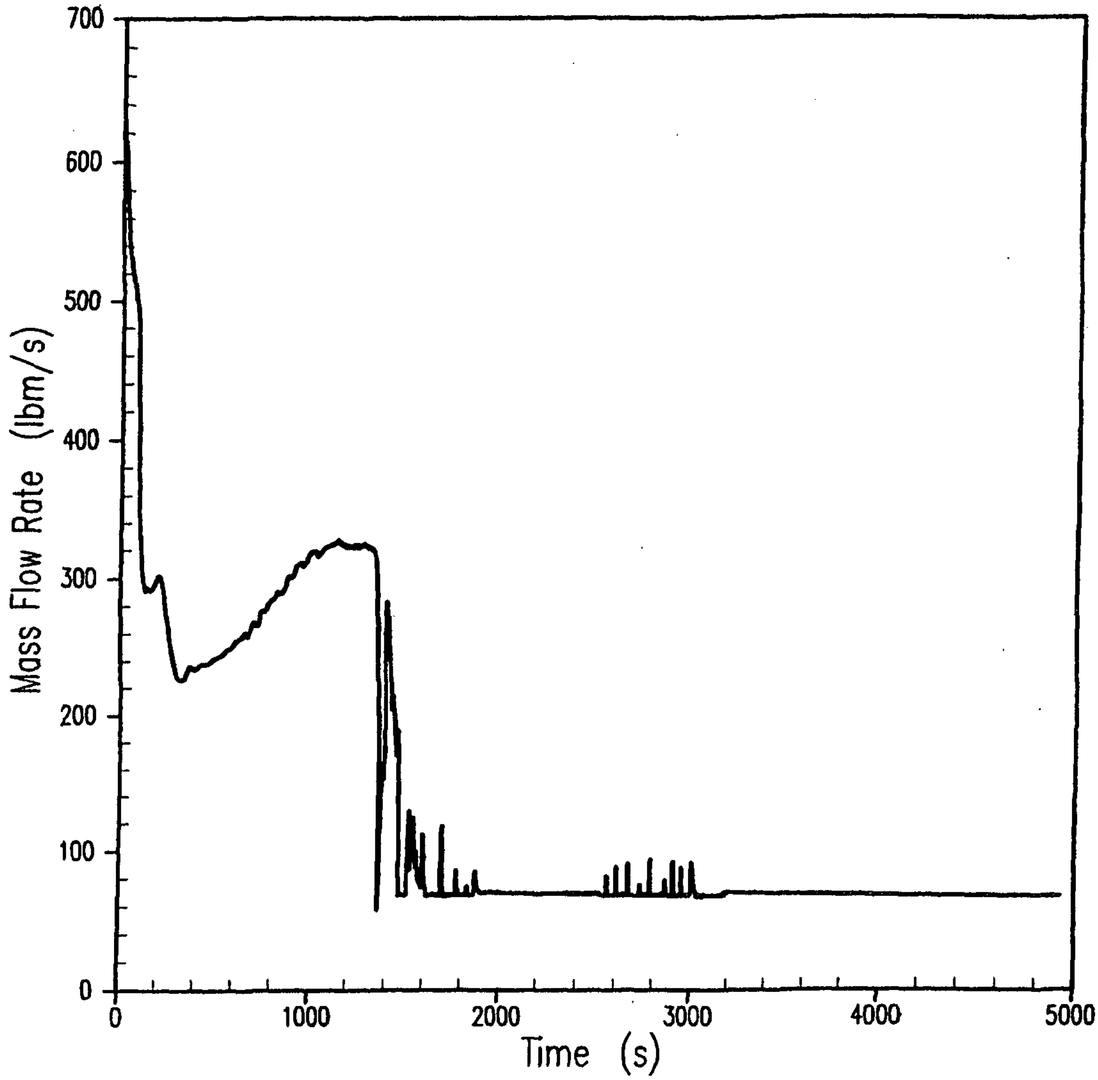
Figure 15.6-15l



Byron/Braidwood Stations Updated Final Safety Analysis Report

Core Steam Flow Rate - Unit 1 (2 Inch, Nominal T_{AVG})

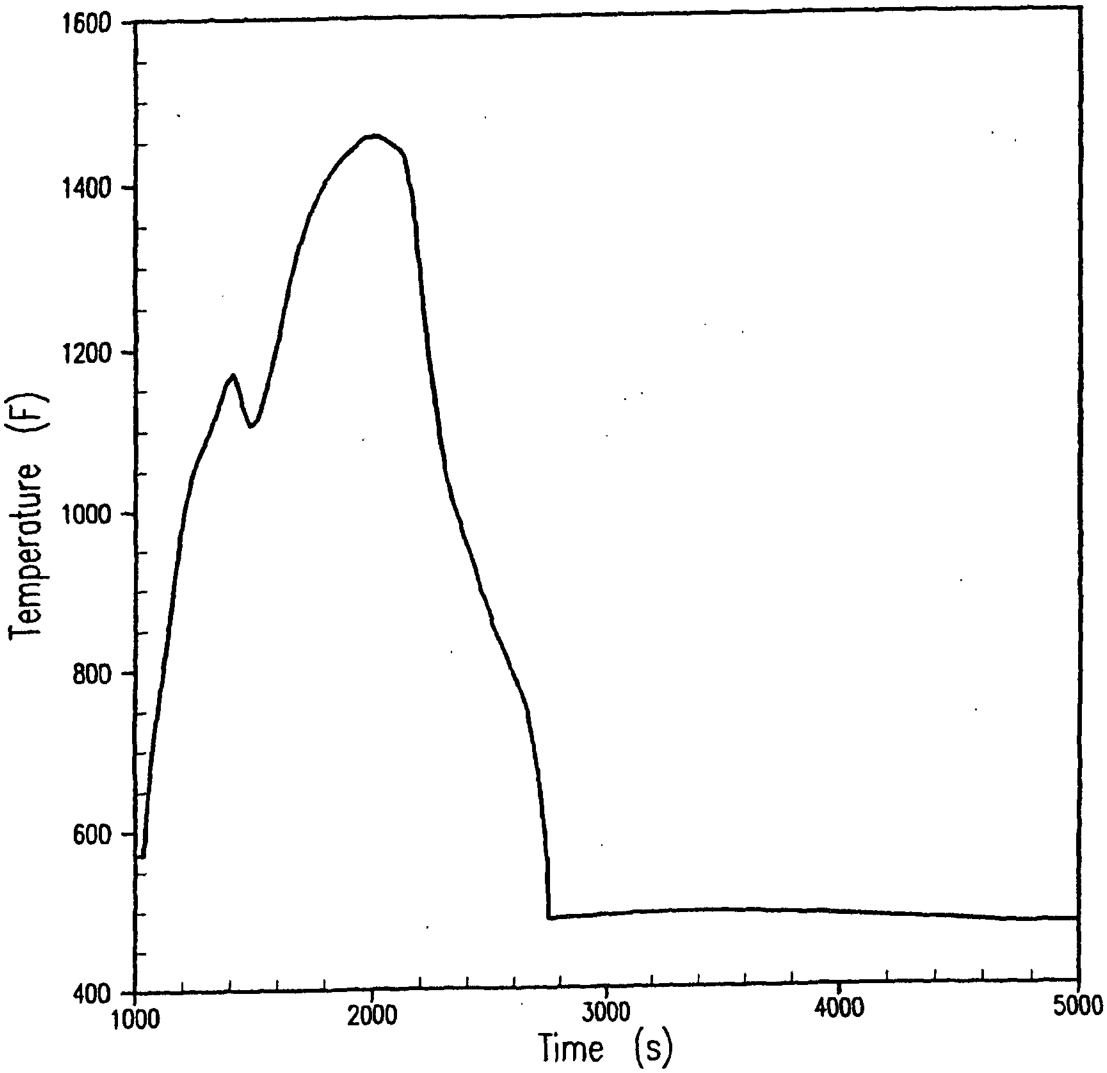
Figure 15.6-15m



**Byron/Braidwood Stations
Updated Final Safety Analysis Report**

**Break Flow - Unit 1
(2 Inch, Nominal T_{AVG})**

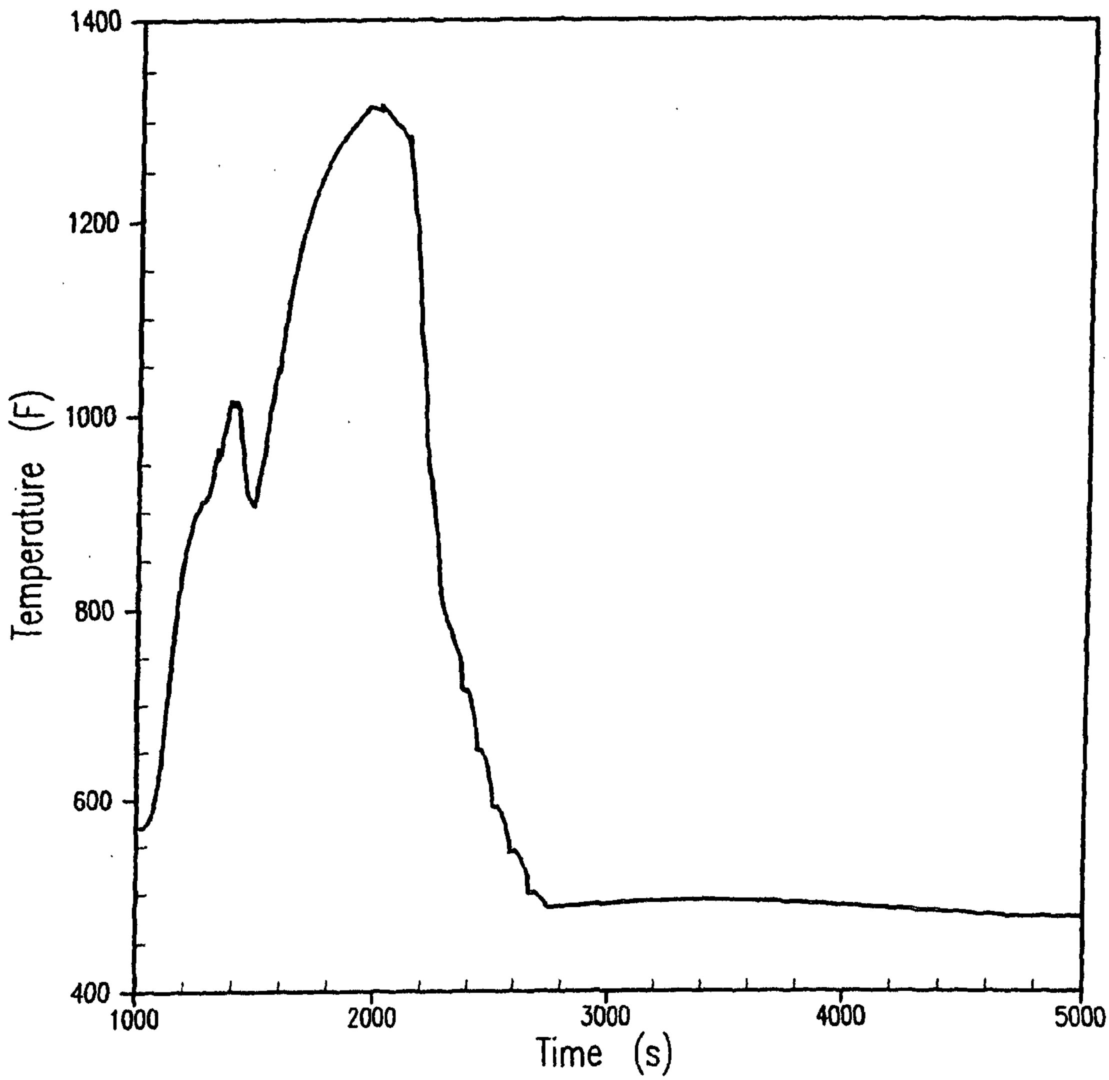
Figure 15.6-15n



Byron/Braidwood Stations Updated Final Safety Analysis Report

Peak Clad Temperature - Unit 1 (3 Inch, Reduced T_{AVG})

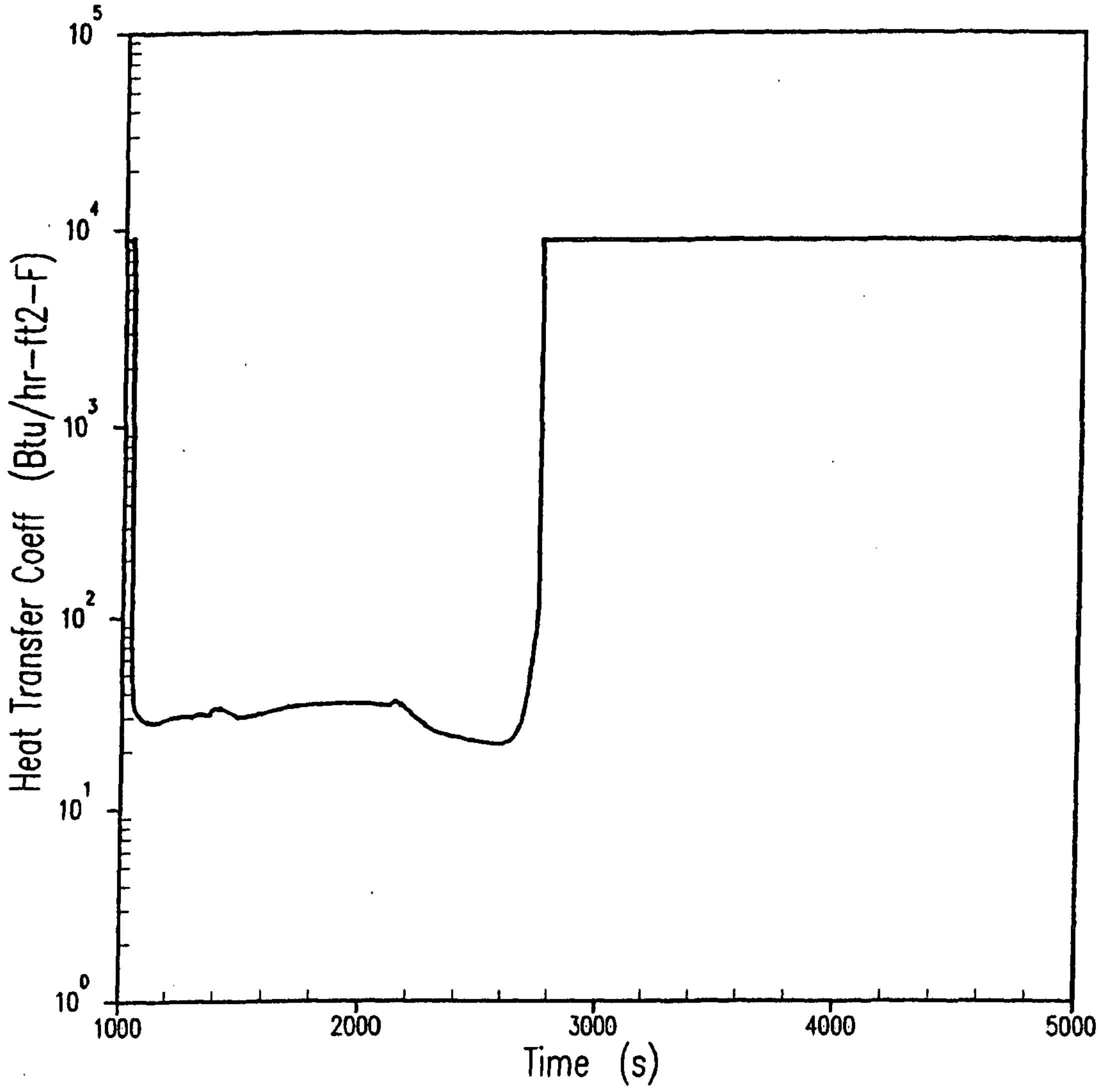
Figure 15.6-16a



Byron/Braidwood Stations Updated Final Safety Analysis Report

Fluid Temperature - Unit 1 (3 Inch, Reduced T_{AVG})

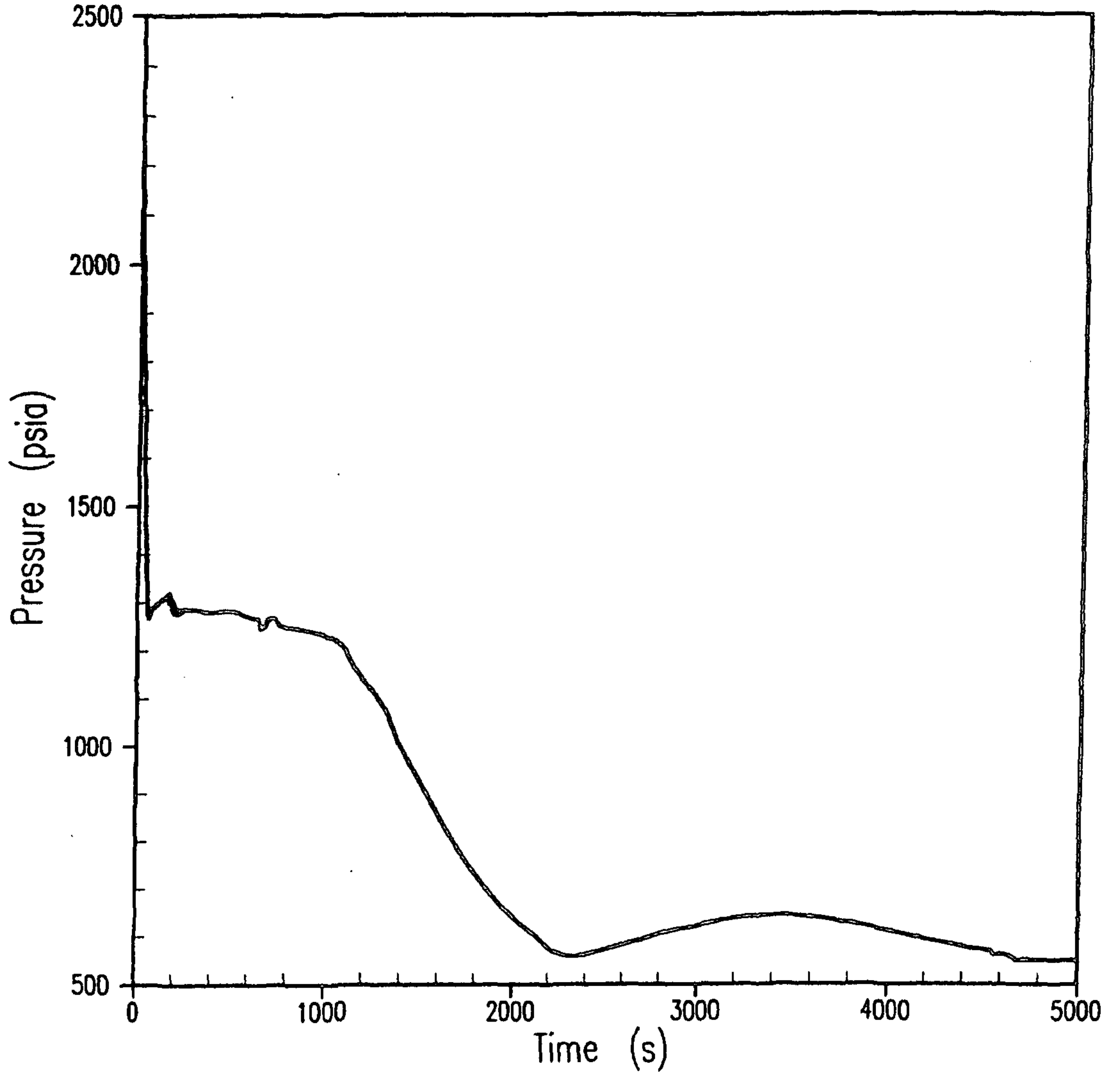
Figure 15.6-16b



Byron/Braidwood Stations Updated Final Safety Analysis Report

Heat Transfer Coefficient - Unit 1 (3 Inch, Reduced T_{AVG})

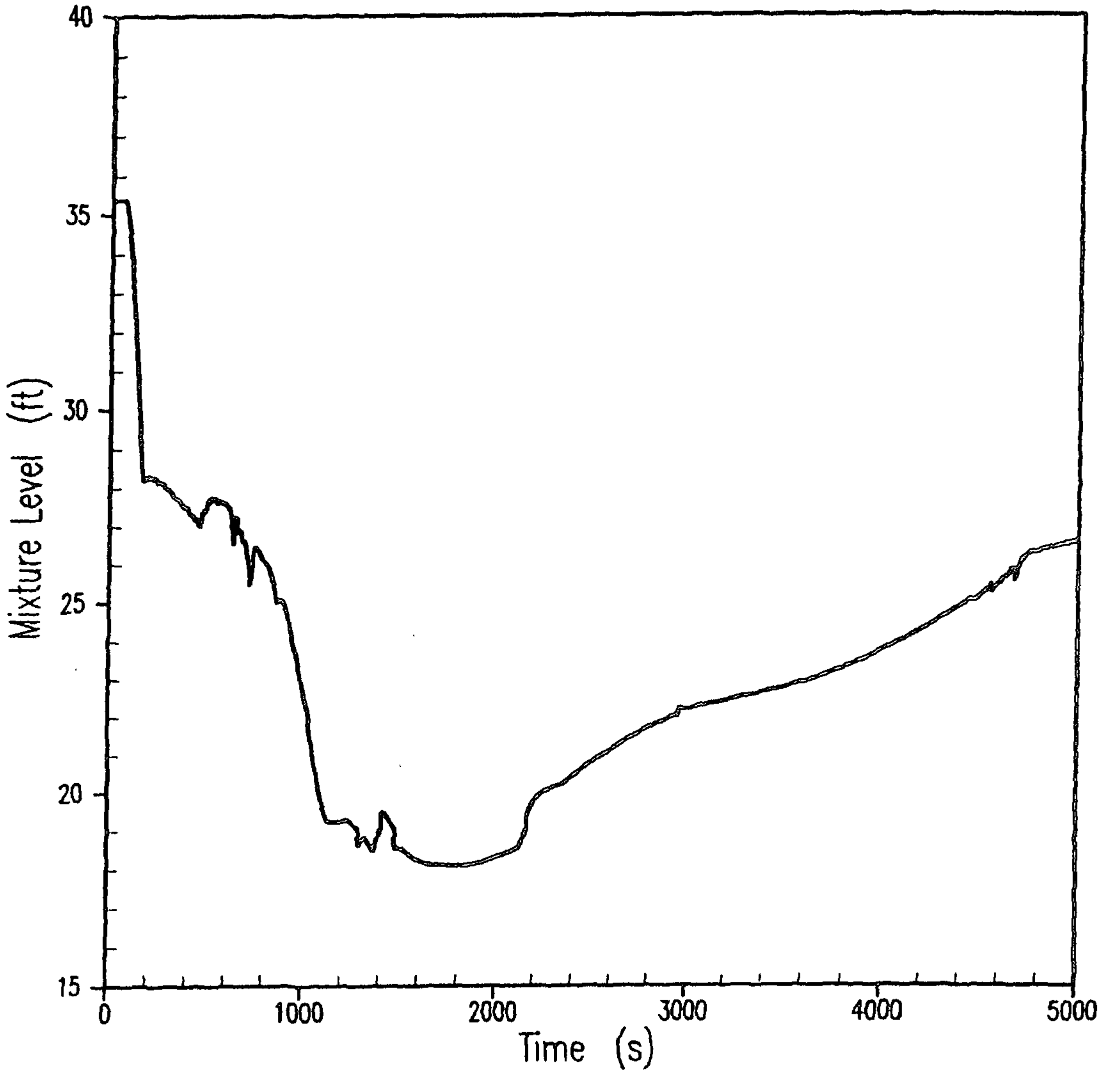
Figure 15.6-16c



Byron/Braidwood Stations Updated Final Safety Analysis Report

RCS Pressure - Unit 1
(3 Inch, Reduced T_{AVG})

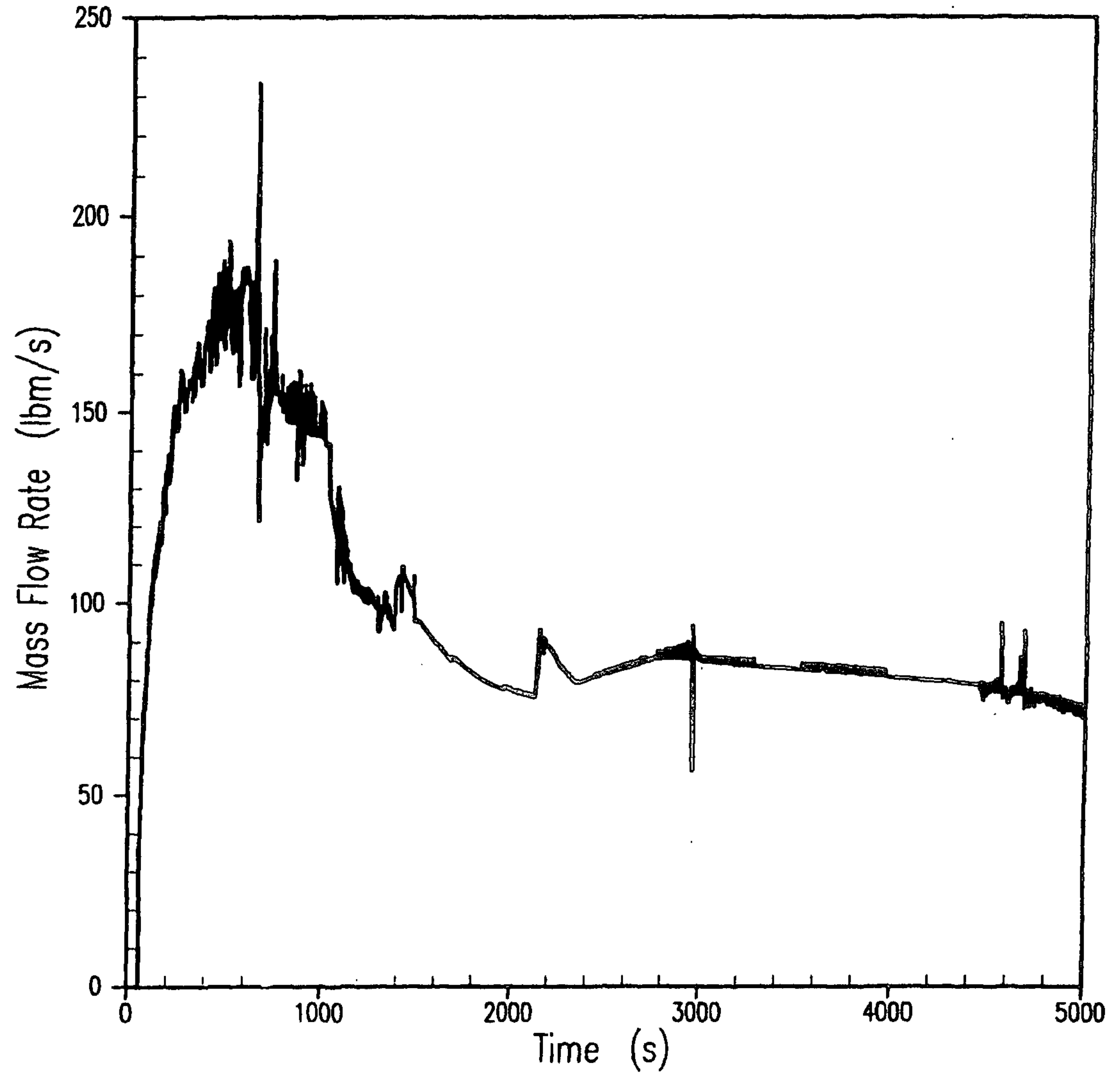
Figure 15.6-16d



Byron/Braidwood Stations
Updated Final Safety Analysis Report

Core Mixture Level - Unit 1
(3 Inch, Reduced T_{AVG})

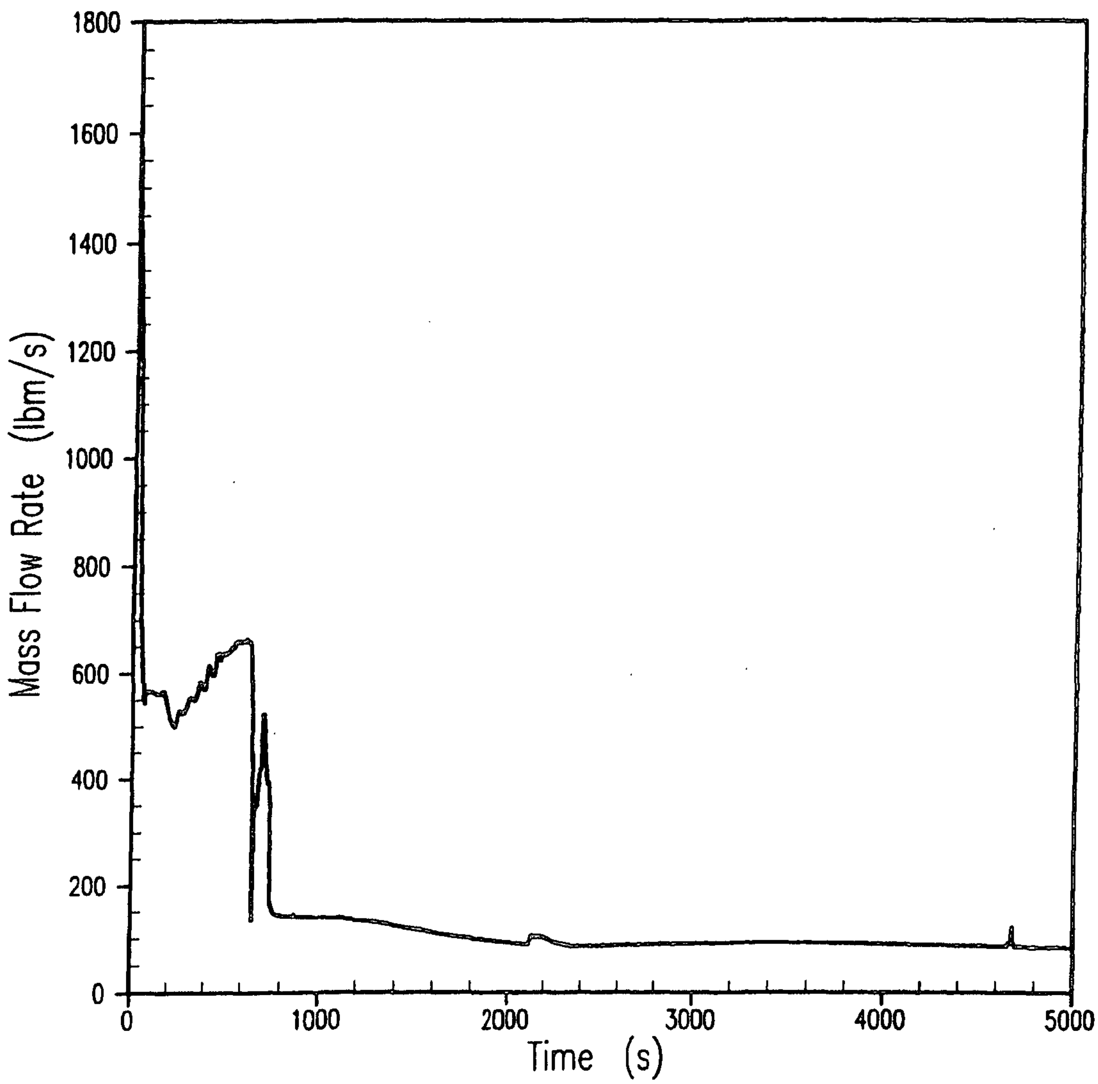
Figure 15.6-16e



Byron/Braidwood Stations Updated Final Safety Analysis Report

Core Steam Flow Rate - Unit 1
(3 Inch, Reduced T_{AVG})

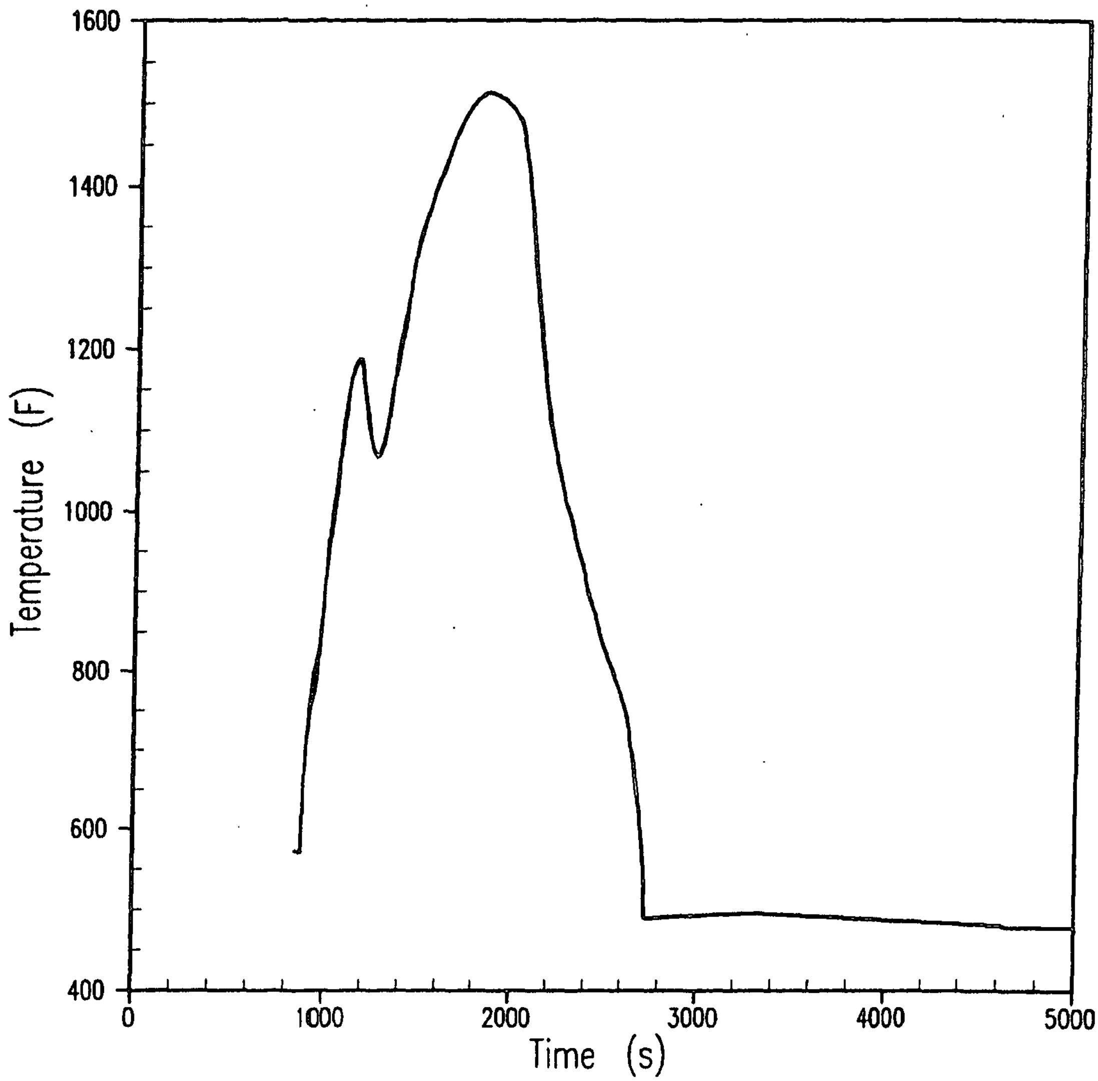
Figure 15.6-16f



Byron/Braidwood Stations Updated Final Safety Analysis Report

Break Flow - Unit 1 (3 Inch, Reduced T_{AVG})

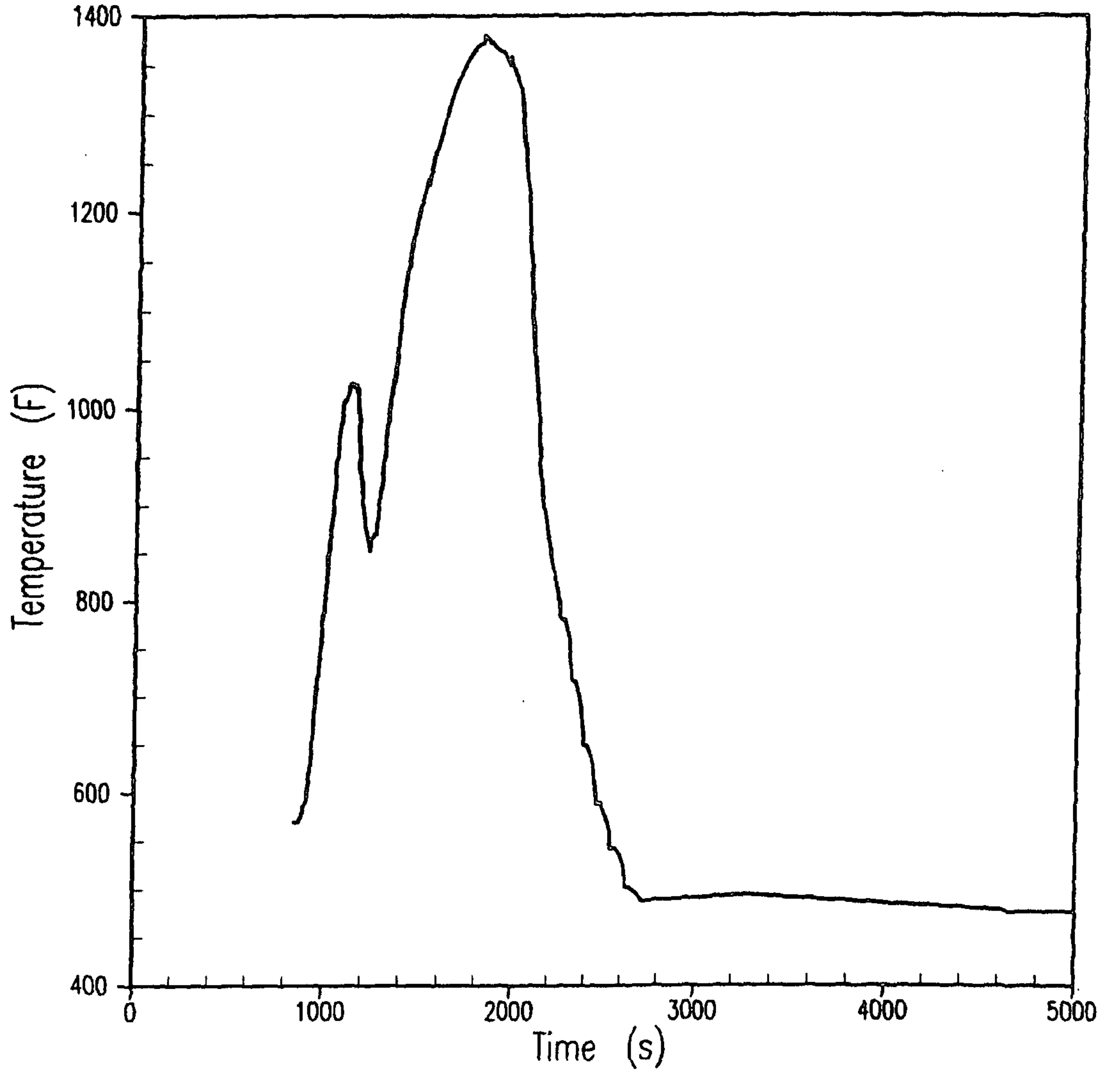
Figure 15.6-16g



Byron/Braidwood Stations
Updated Final Safety Analysis Report

Peak Clad Temperature - Unit 1
(3 Inch, Nominal T_{AVG})

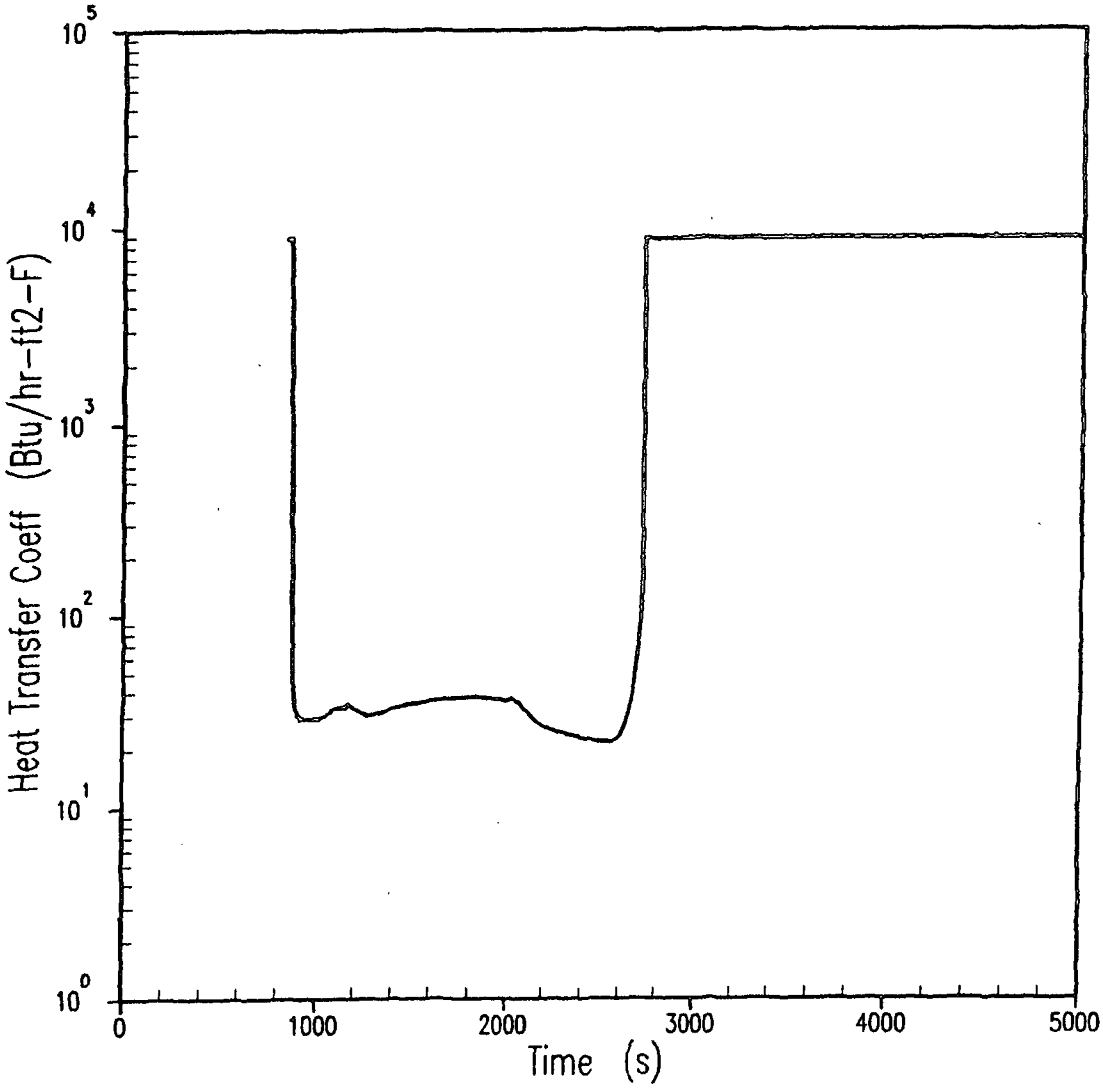
Figure 15.6-16h



Byron/Braidwood Stations Updated Final Safety Analysis Report

Fluid Temperature - Unit 1
(3 Inch, Nominal T_{AVG})

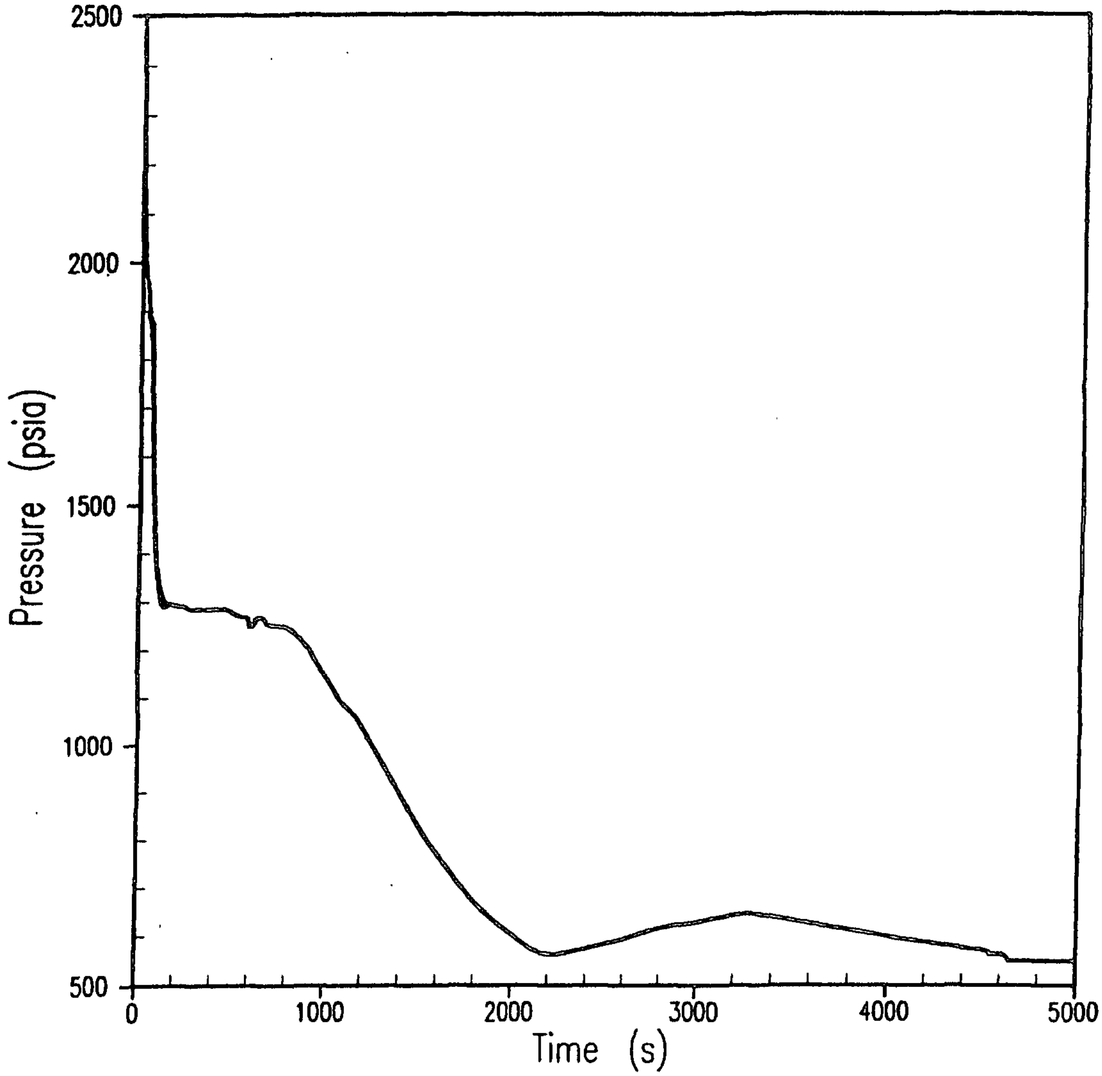
Figure 15.6-16i



Byron/Braidwood Stations Updated Final Safety Analysis Report

Heat Transfer Coefficient - Unit 1 (3 Inch, Nominal T_{AVG})

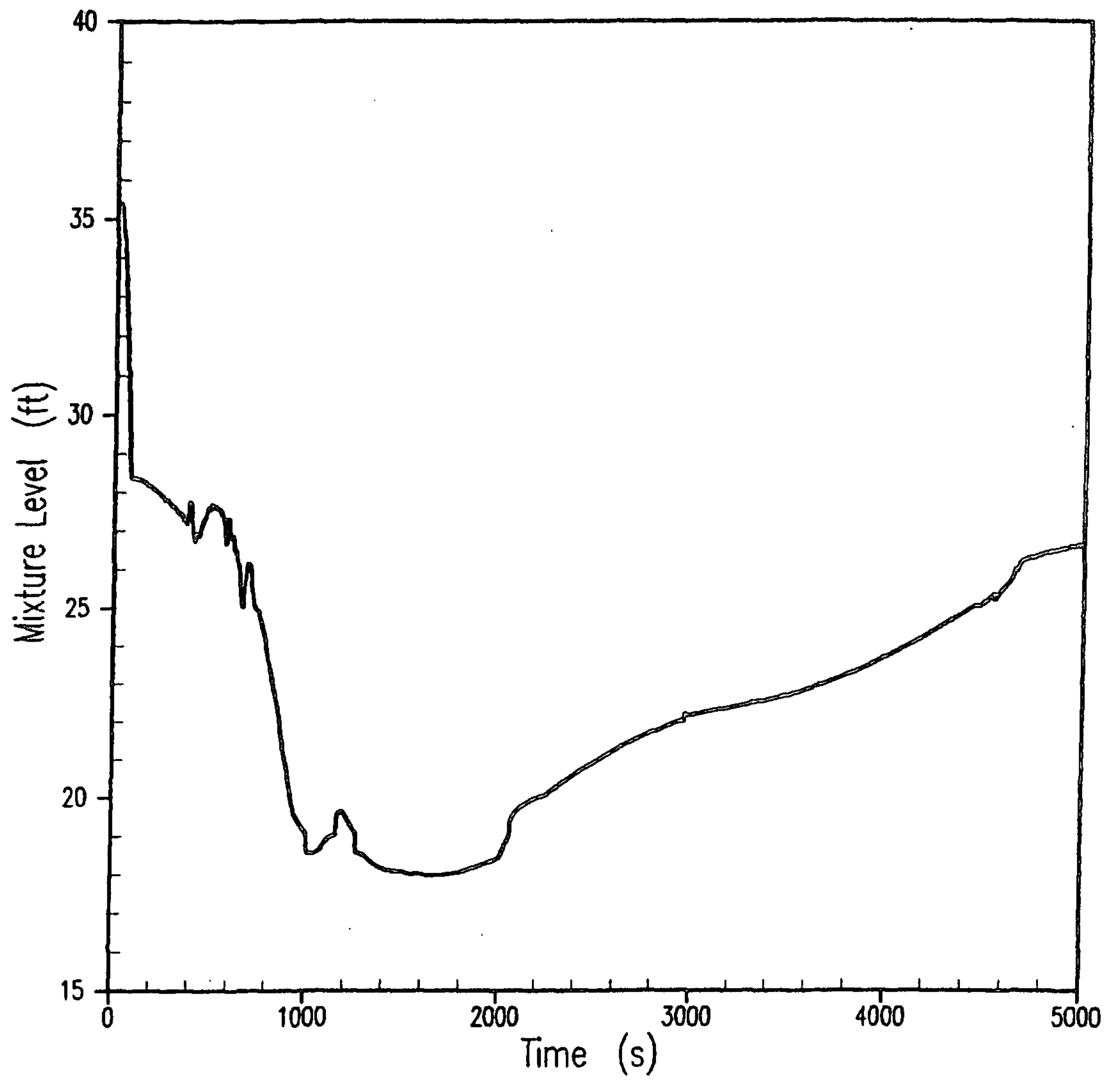
Figure 15.6-16j



Byron/Braidwood Stations Updated Final Safety Analysis Report

RCS Pressure - Unit 1
(3 Inch, Nominal T_{AVG})

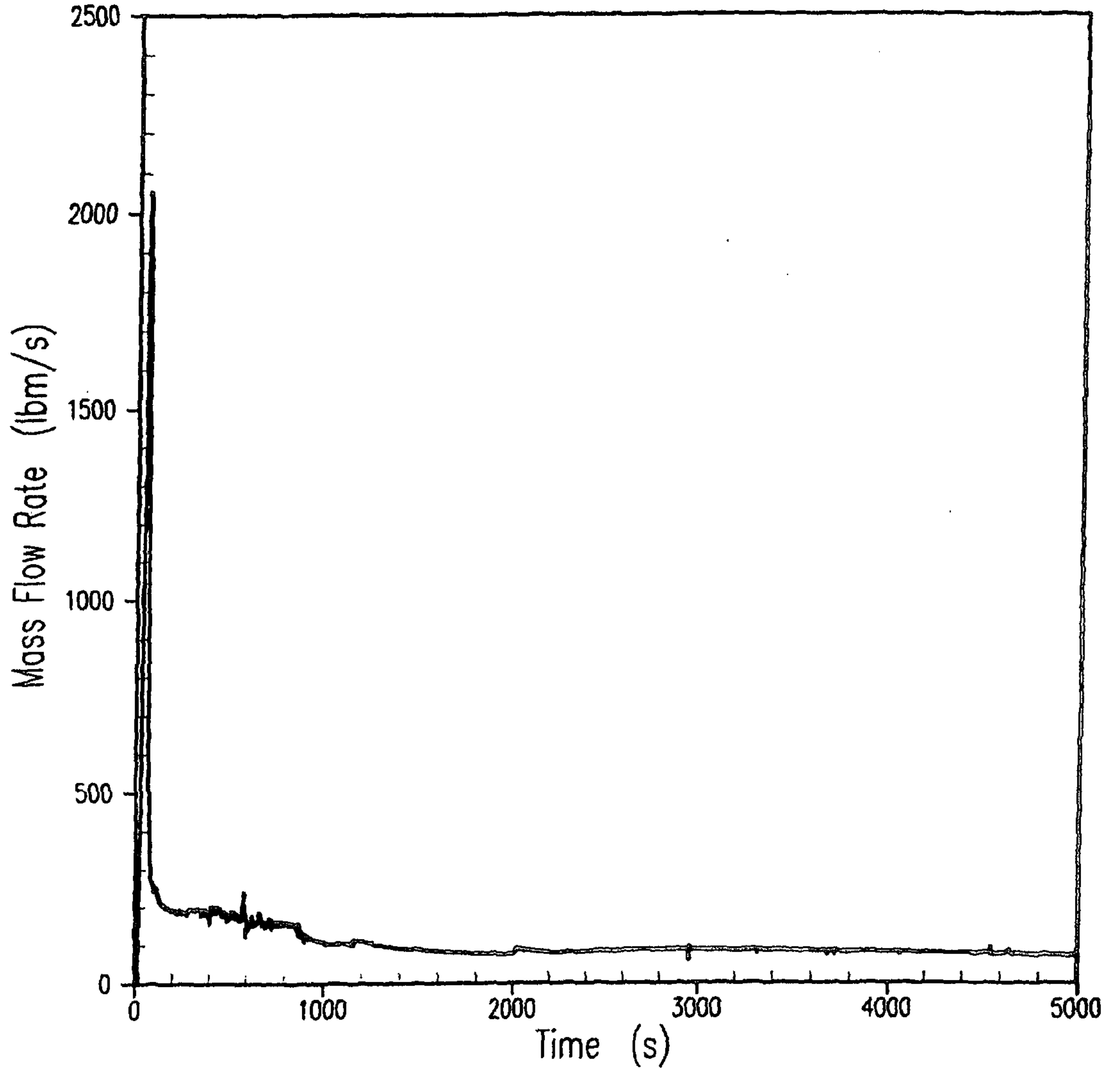
Figure 15.6-16k



**Byron/Braidwood Stations
Updated Final Safety Analysis
Report**

**Core Mixture Level - Unit 1
(3 Inch, Nominal T_{AVG})**

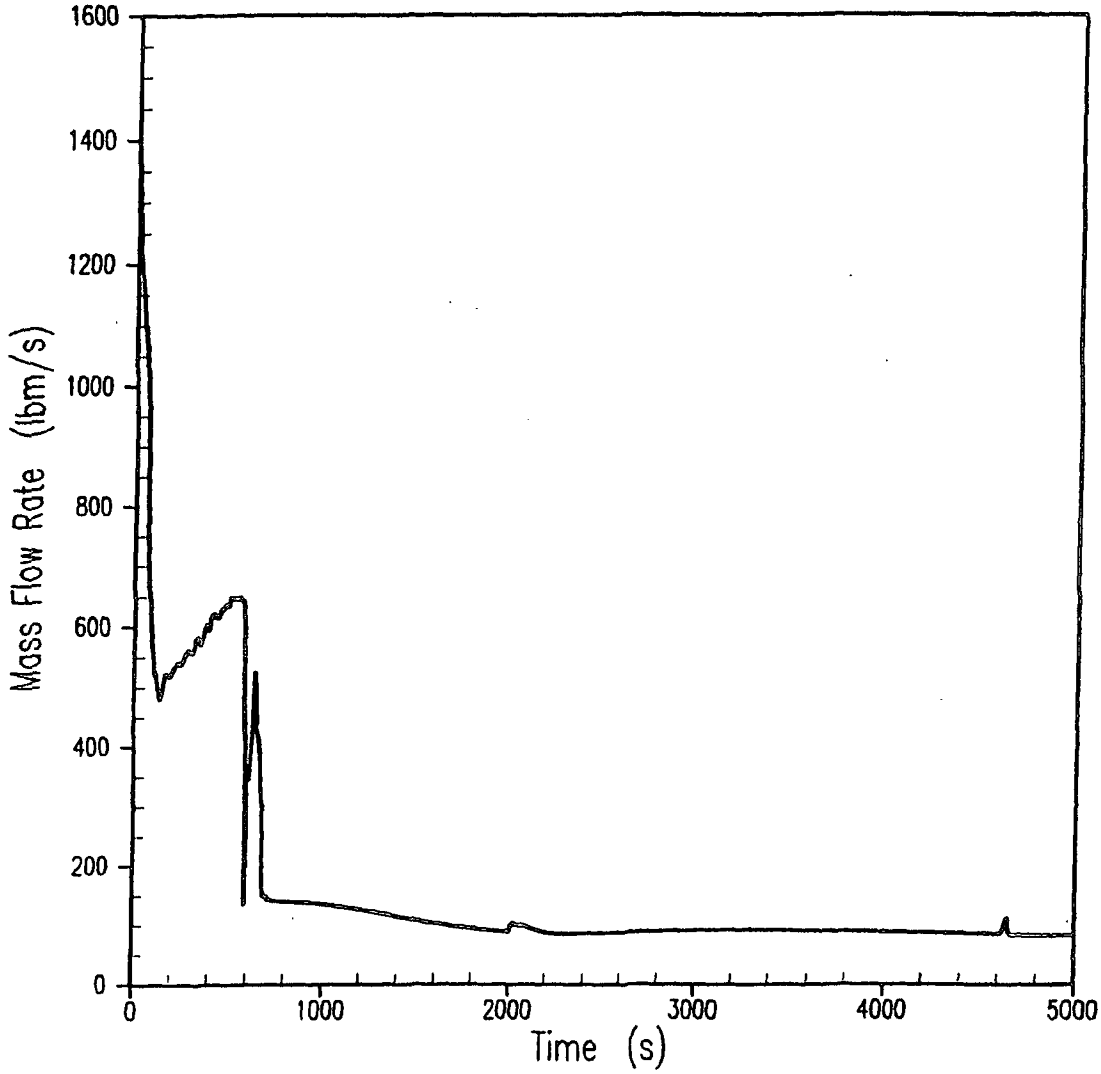
Figure 15.6-16I



**Byron/Braidwood Stations
Updated Final Safety Analysis
Report**

Core Steam Flow Rate - Unit 1
(3 Inch, Nominal T_{AVG})

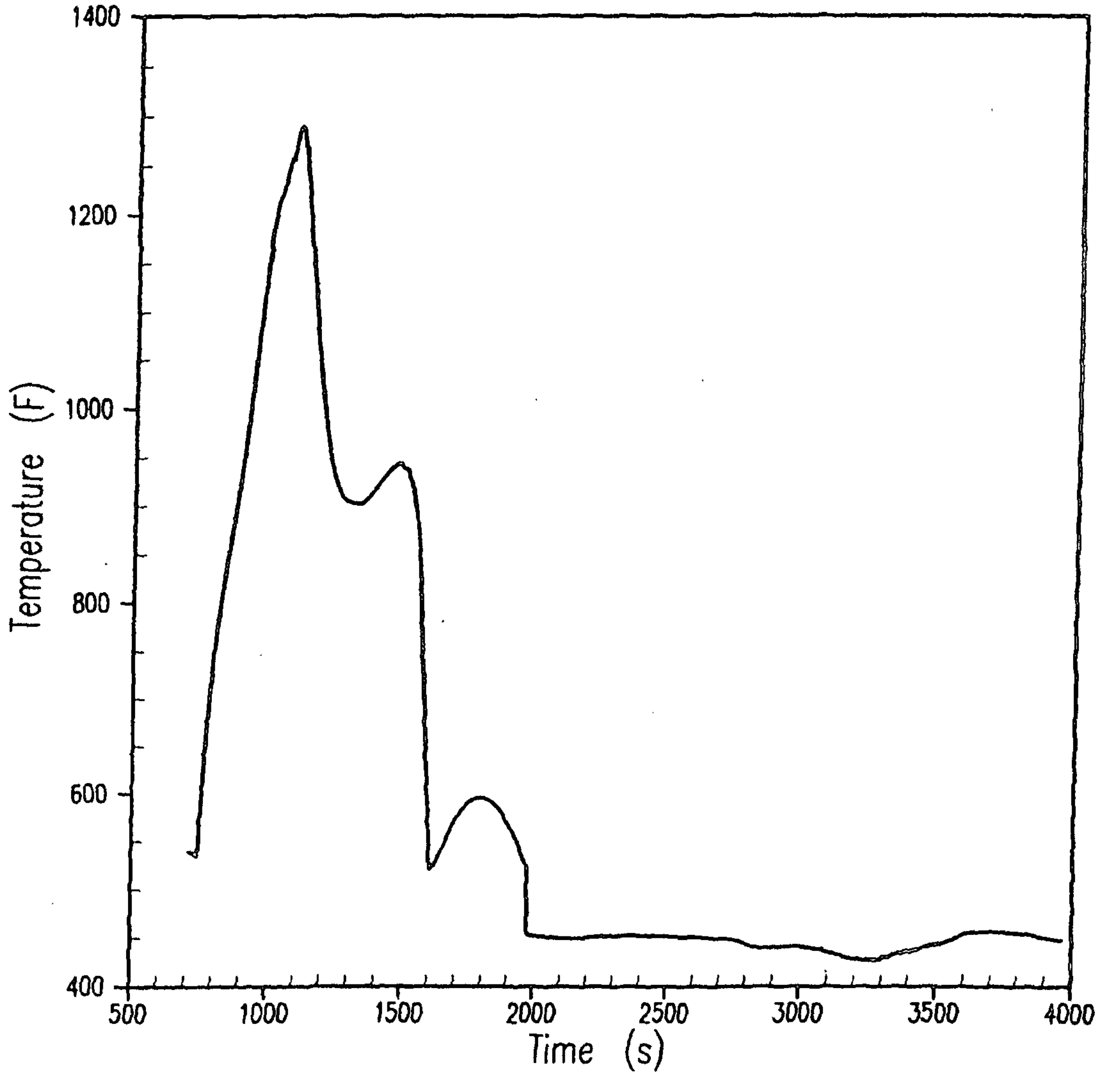
Figure 15.6-16m



**Byron/Braidwood Stations
Updated Final Safety Analysis
Report**

**Break Flow - Unit 1
(3 Inch, Nominal T_{AVG})**

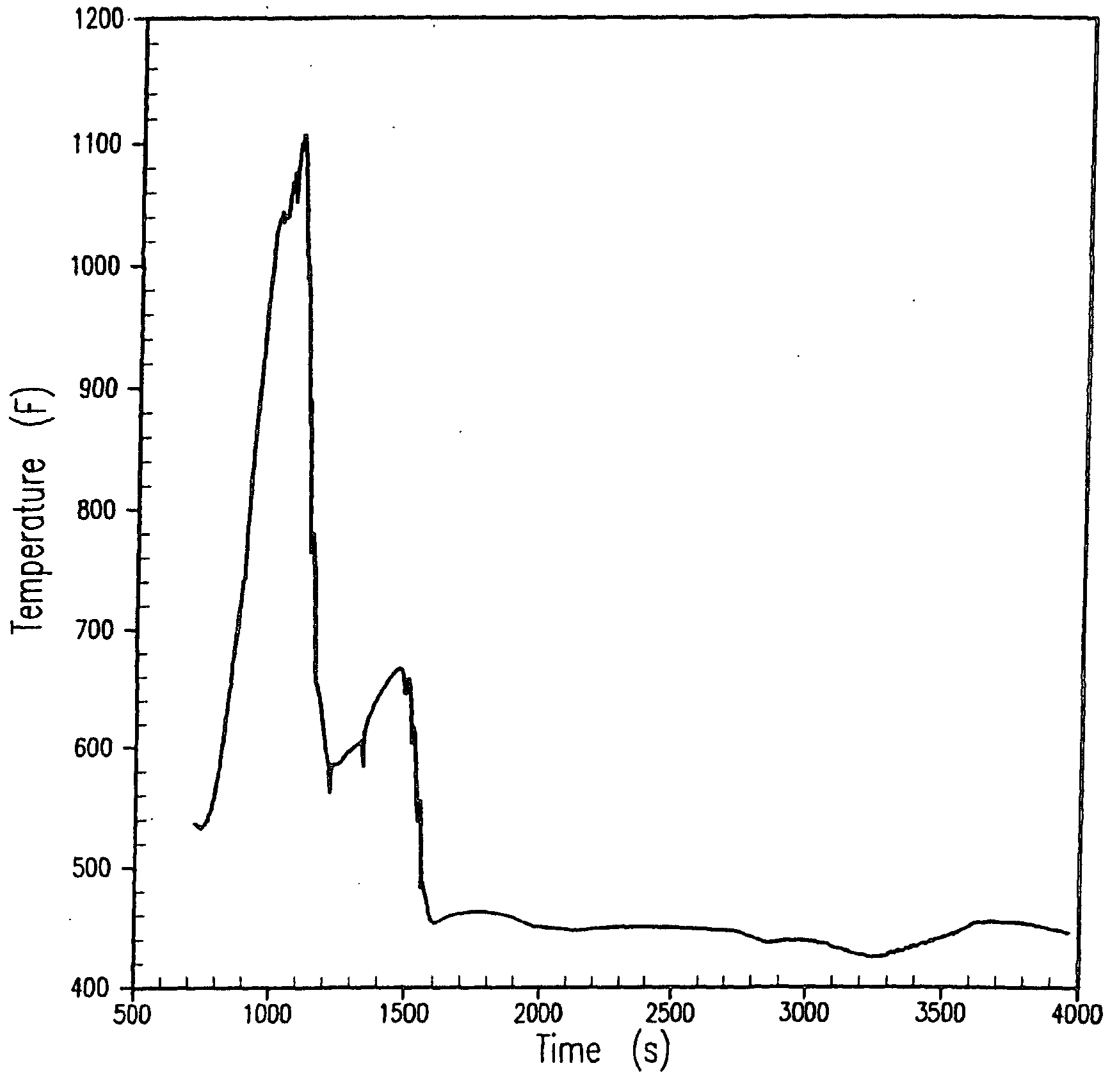
Figure 15.6-16n



Byron/Braidwood Stations Updated Final Safety Analysis Report

Peak Clad Temperature - Unit 1
(4 Inch, Reduced T_{AVG})

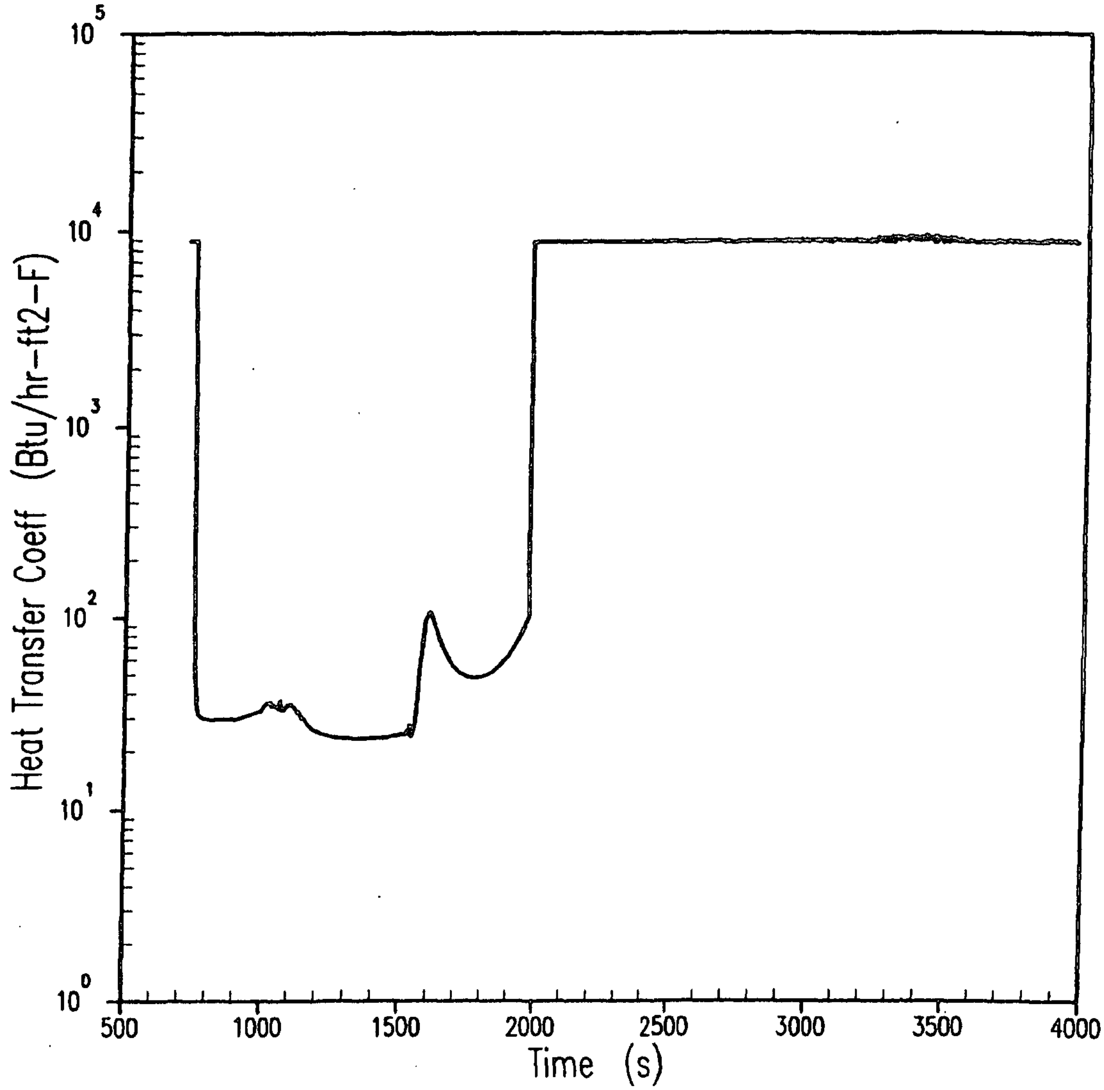
Figure 15.6-17a



**Byron/Braidwood Stations
Updated Final Safety Analysis Report**

**Fluid Temperature - Unit 1
(4 Inch, Reduced T_{AVG})**

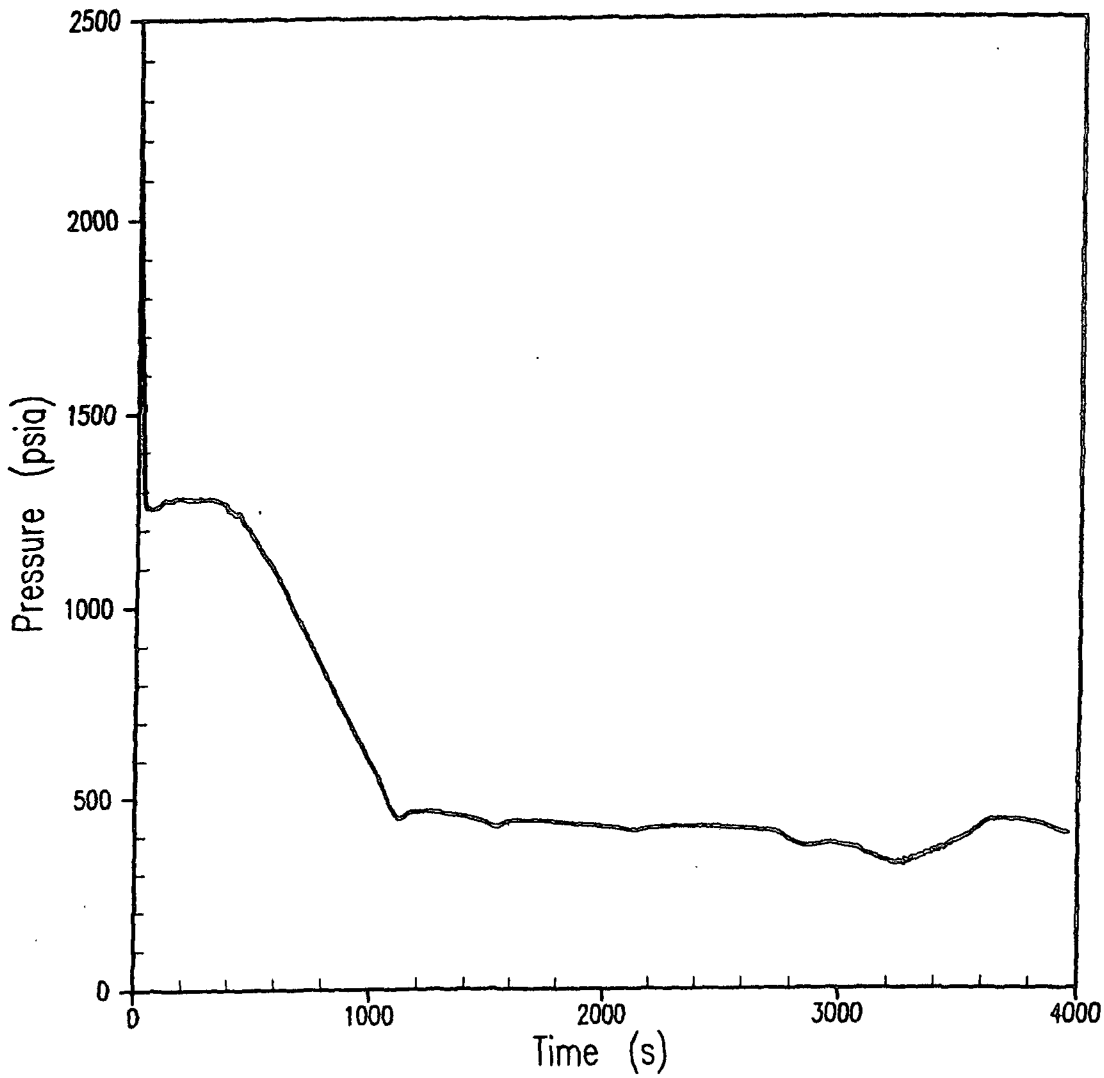
Figure 15.6-17b



Byron/Braidwood Stations Updated Final Safety Analysis Report

Heat Transfer Coefficient - Unit 1 (4 Inch, Reduced T_{AVG})

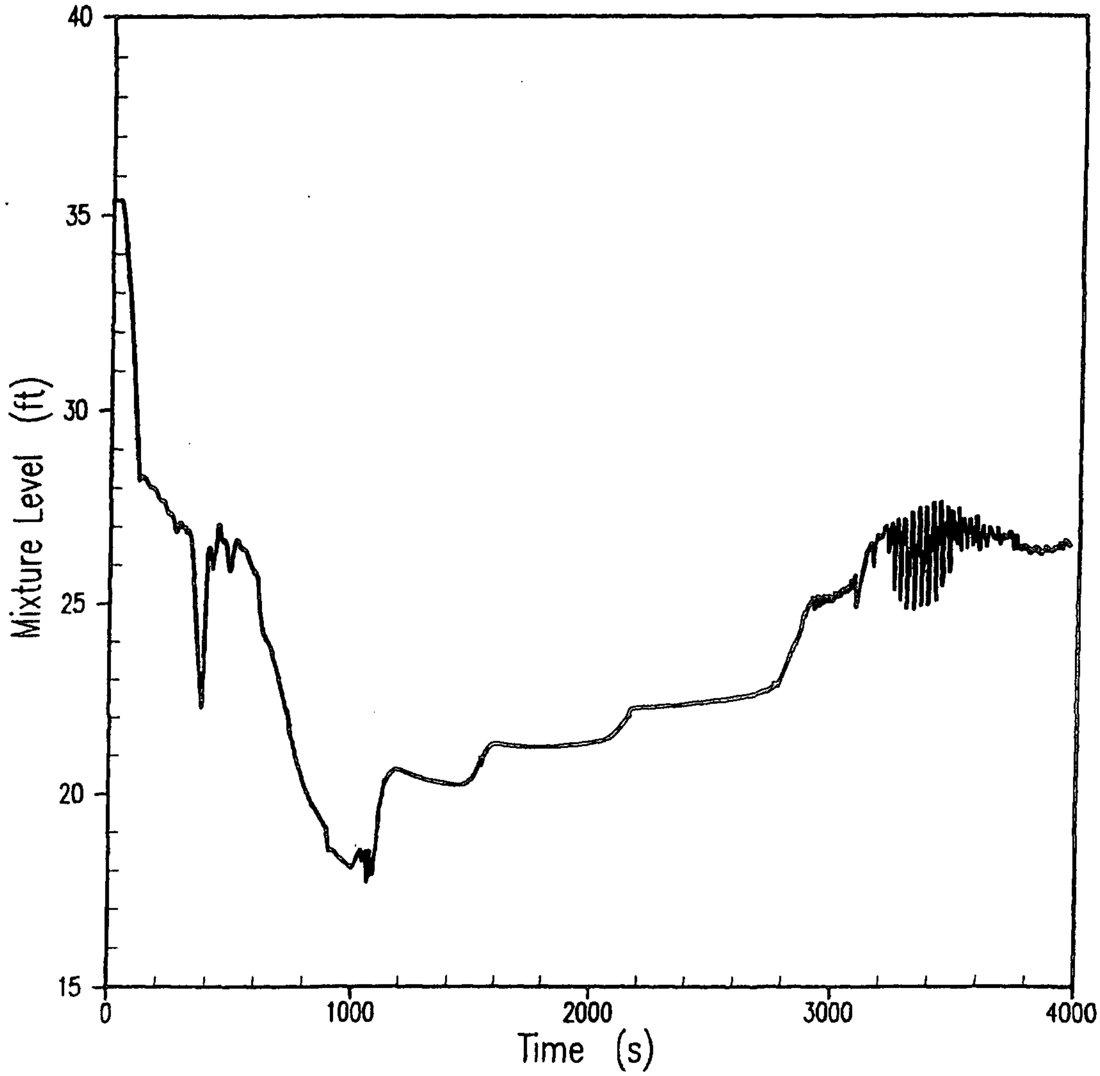
Figure 15.6-17c



Byron/Braidwood Stations Updated Final Safety Analysis Report

RCS Pressure - Unit 1
(4 Inch, Reduced T_{AVG})

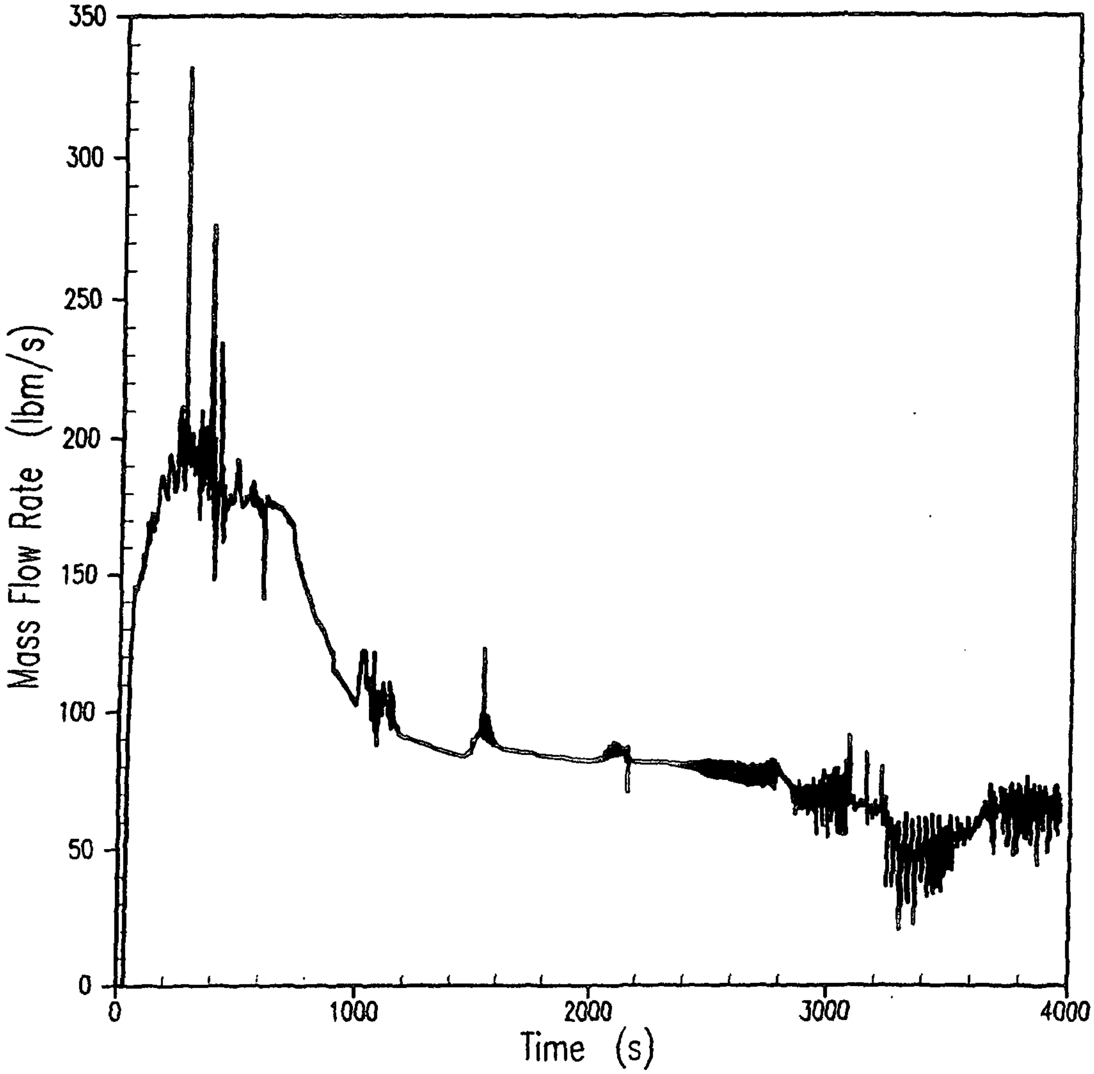
Figure 15.6-17d



Byron/Braidwood Stations Updated Final Safety Analysis Report

Core Mixture Level - Unit 1
(4 Inch, Reduced T_{AVG})

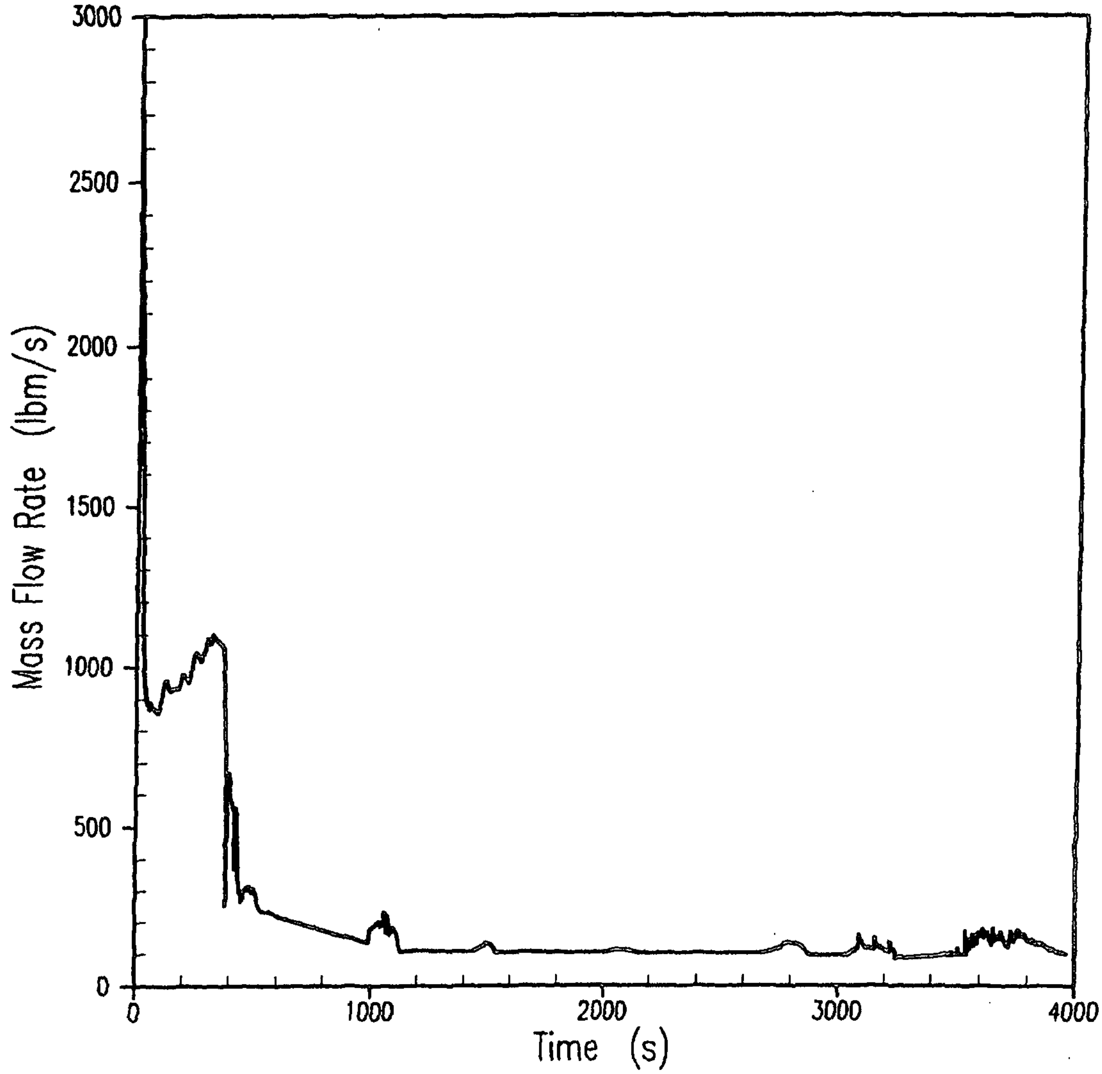
Figure 15.6-17e



Byron/Braidwood Stations Updated Final Safety Analysis Report

Core Steam Flow Rate - Unit 1
(4 Inch, Reduced T_{AVG})

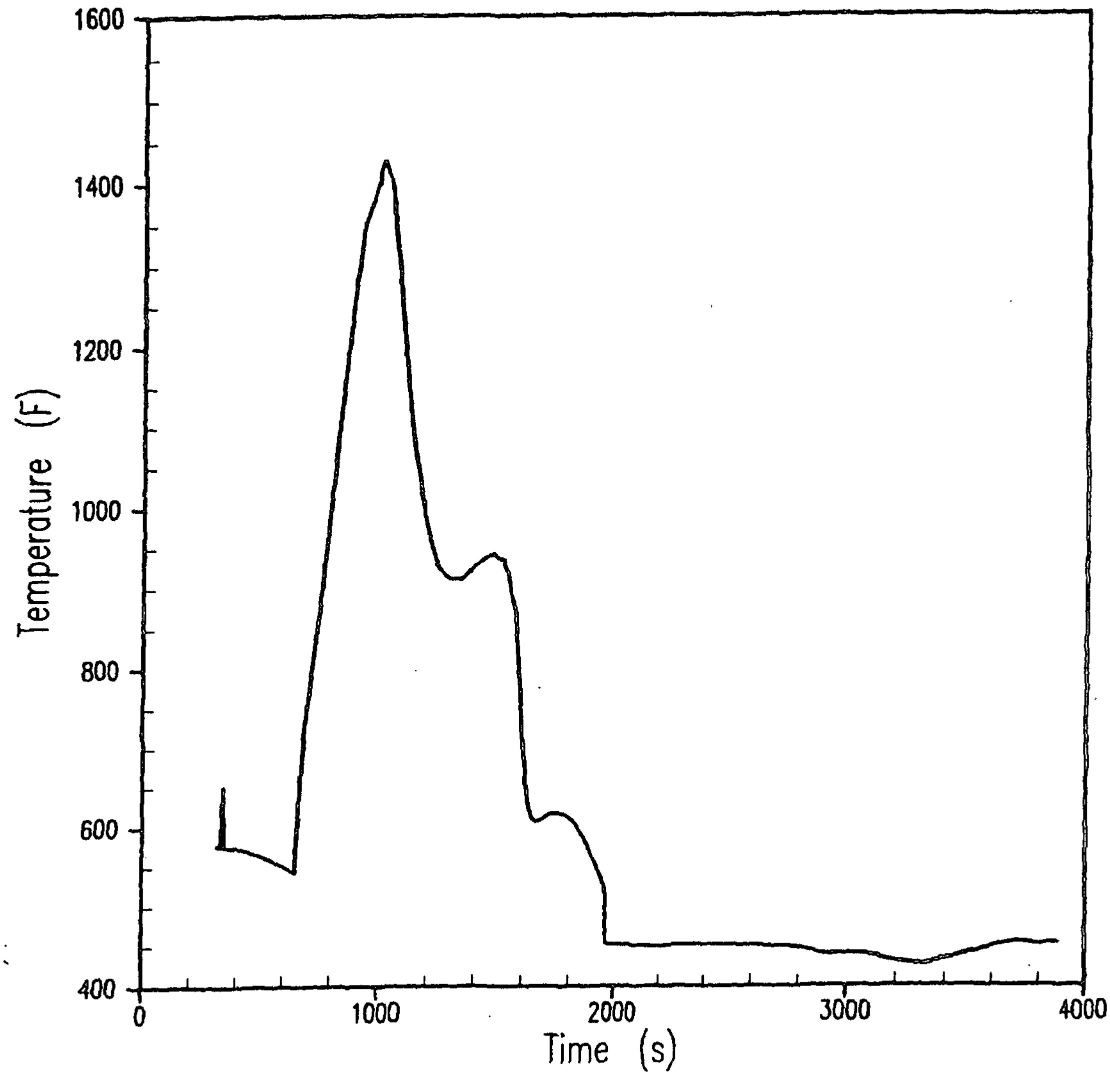
Figure 15.6-17f



Byron/Braidwood Stations Updated Final Safety Analysis Report

Break Flow - Unit 1
(4 Inch, Reduced T_{AVG})

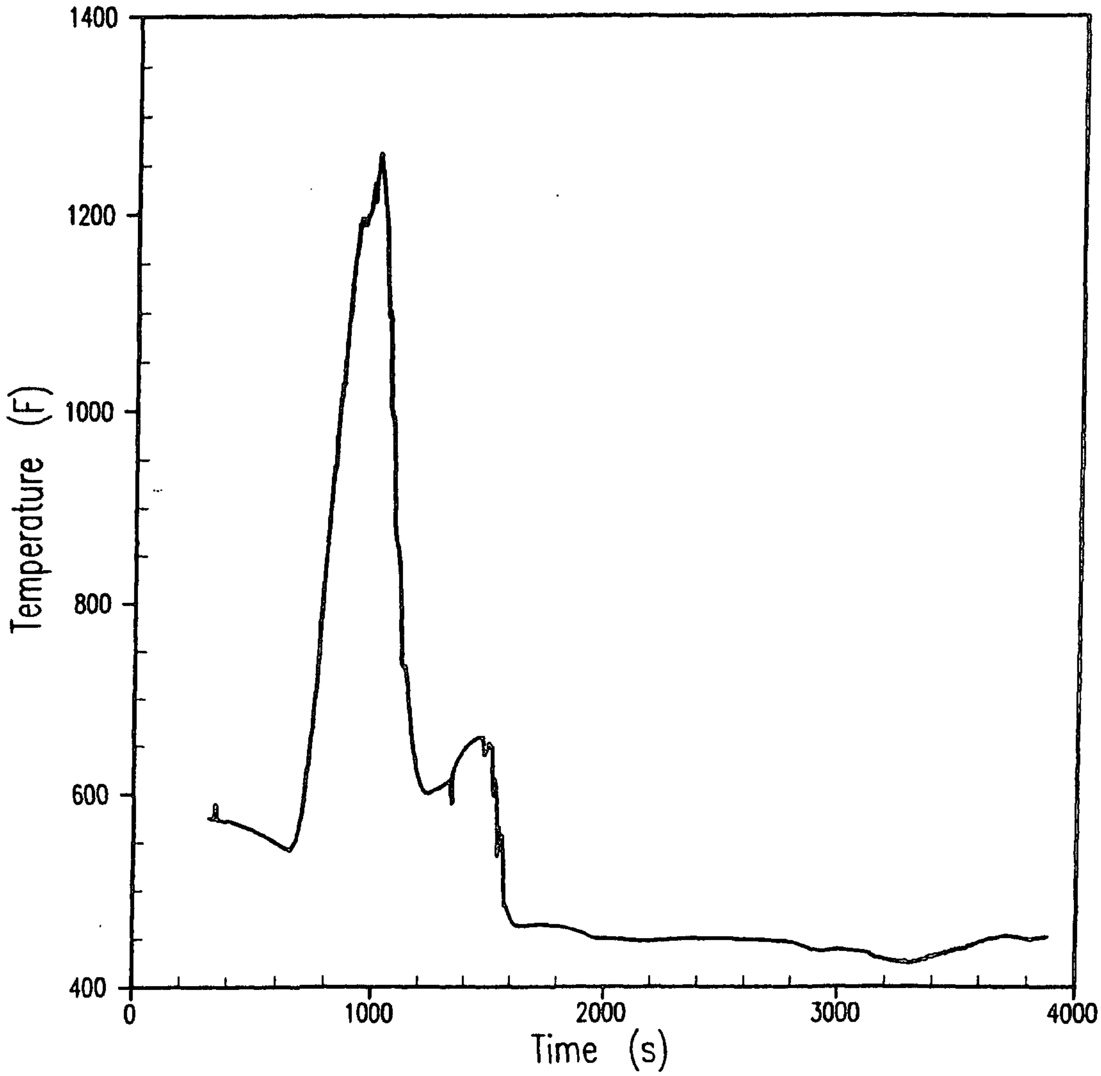
Figure 15.6-17g



Byron/Braidwood Stations Updated Final Safety Analysis Report

Peak Clad Temperature - Unit 1 (4 Inch, Nominal T_{AVG})

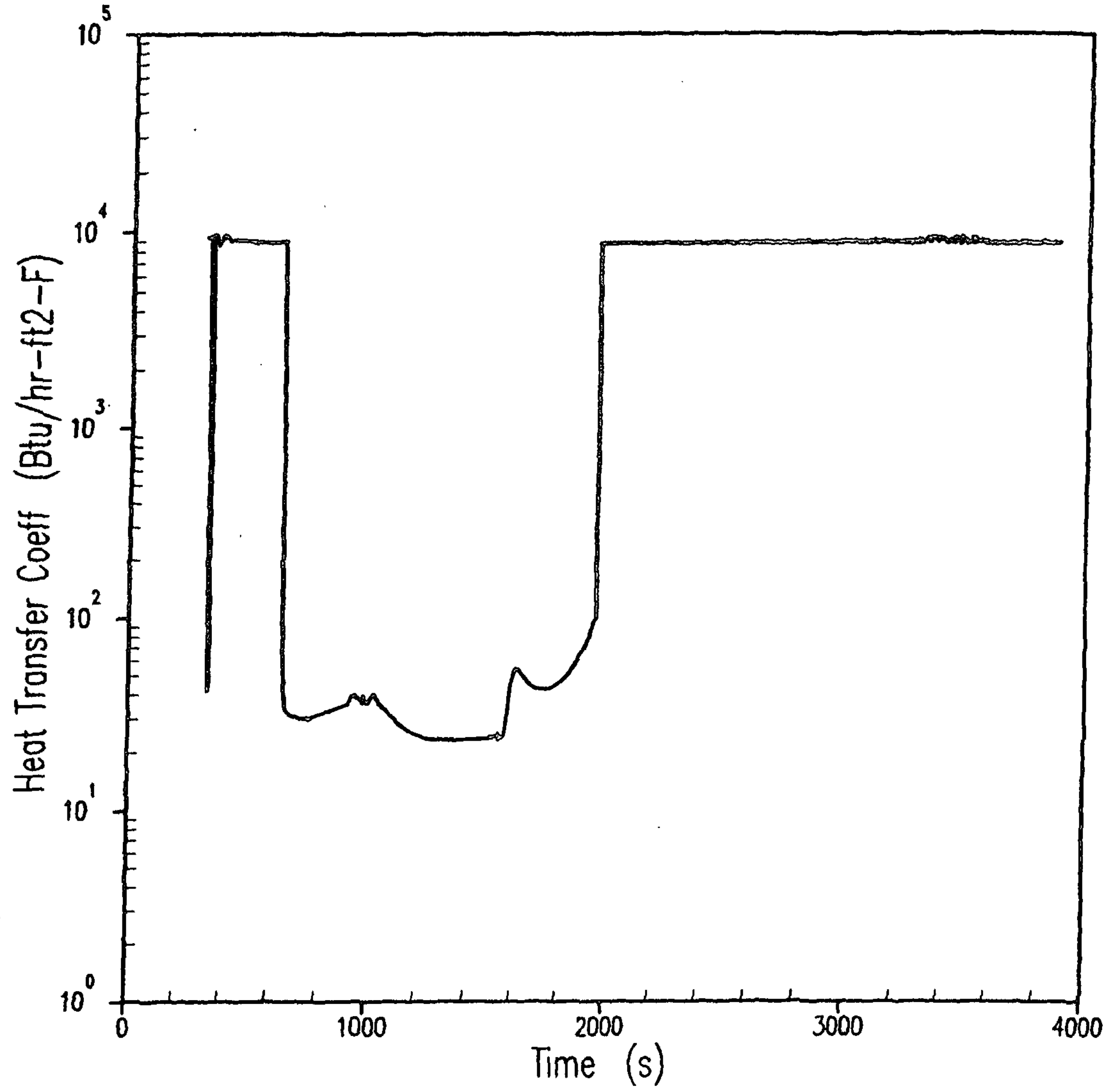
Figure 15.6-17h



Byron/Braidwood Stations Updated Final Safety Analysis Report

Fluid Temperature - Unit 1 (4 Inch, Nominal T_{AVG})

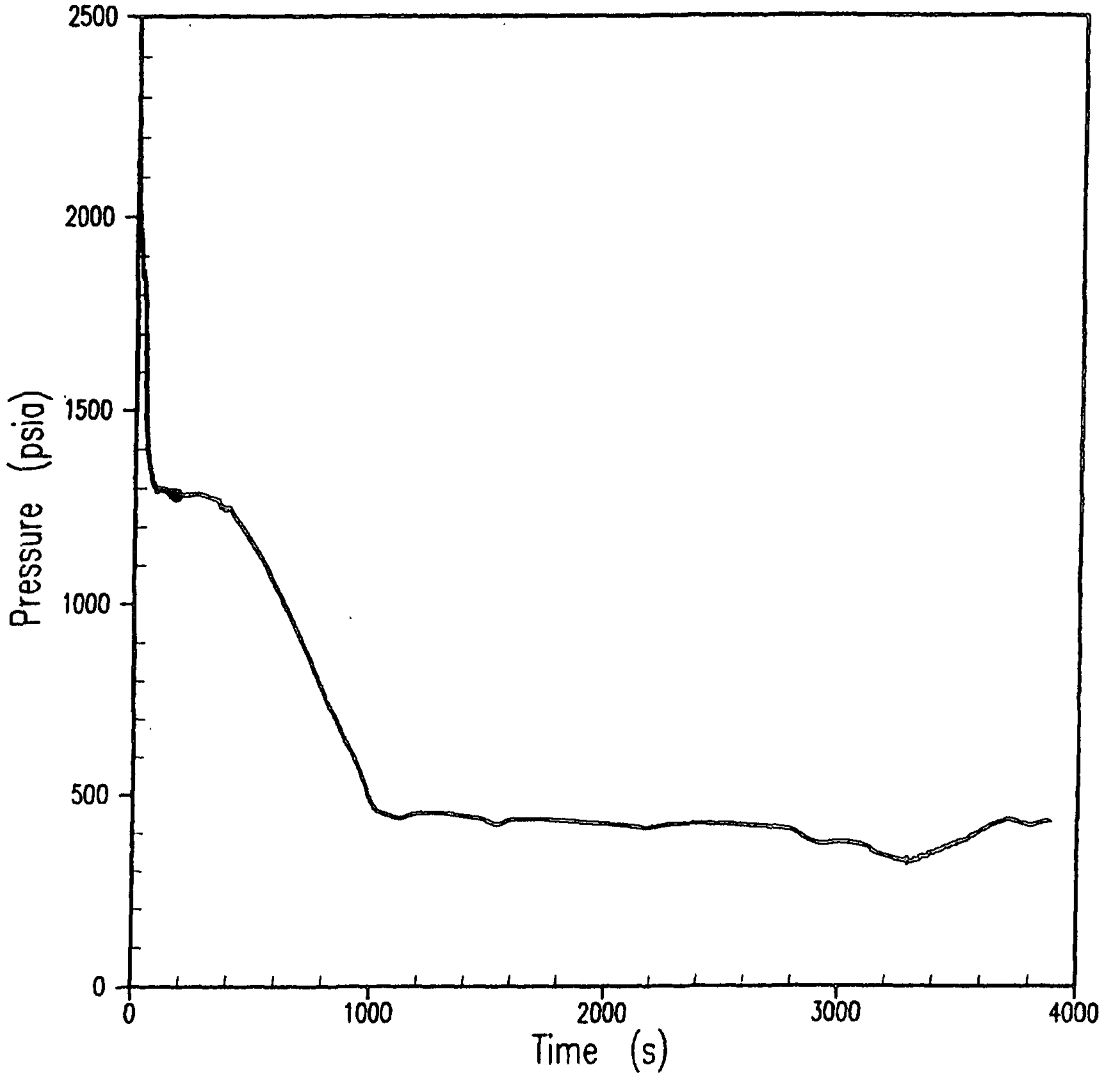
Figure 15.6-171



Byron/Braidwood Stations Updated Final Safety Analysis Report

Heat Transfer Coefficient - Unit 1 (4 Inch, Nominal T_{AVG})

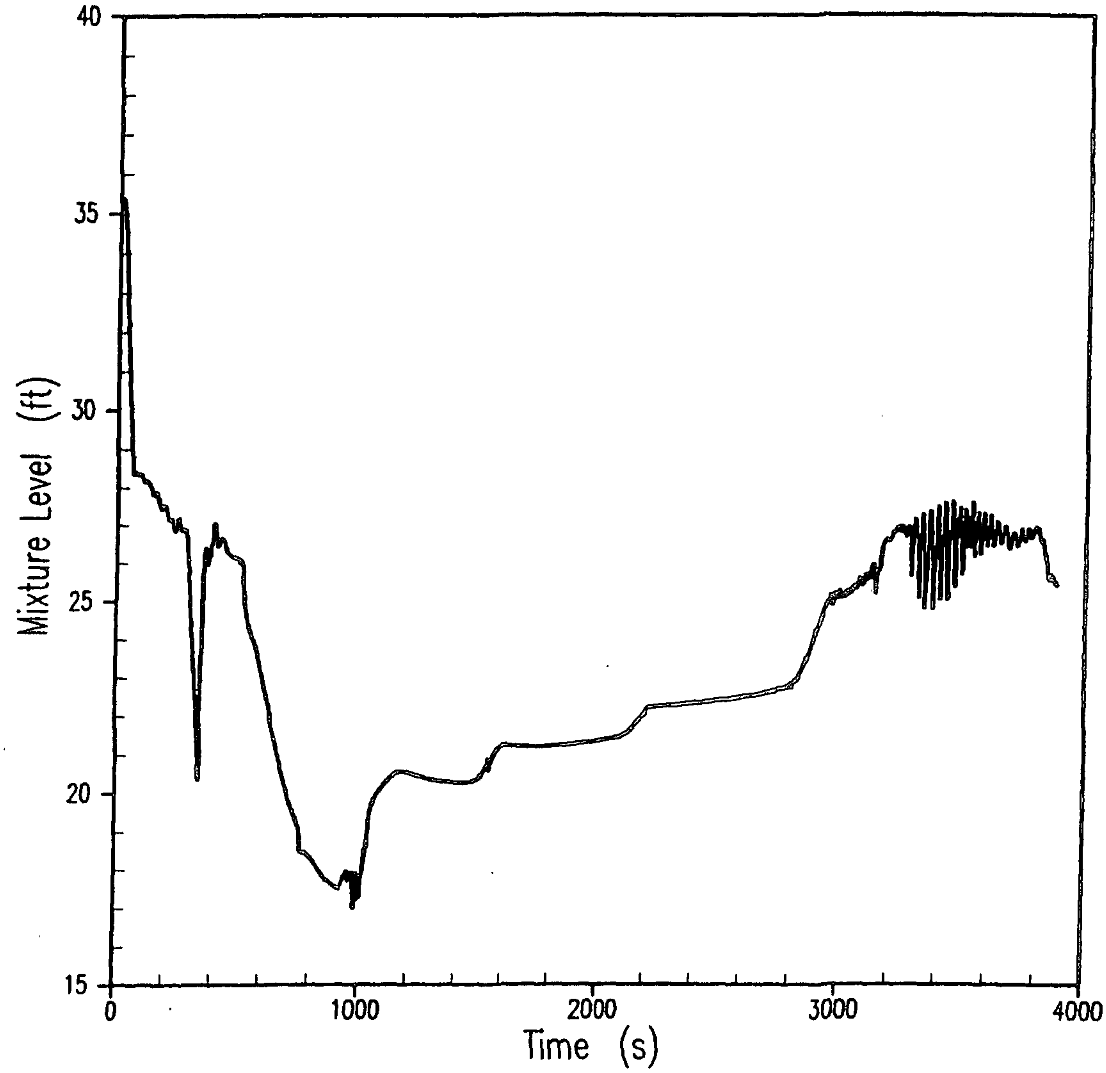
Figure 15.6-17j



Byron/Braidwood Stations Updated Final Safety Analysis Report

RCS Pressure - Unit 1
(4 Inch, Nominal T_{AVG})

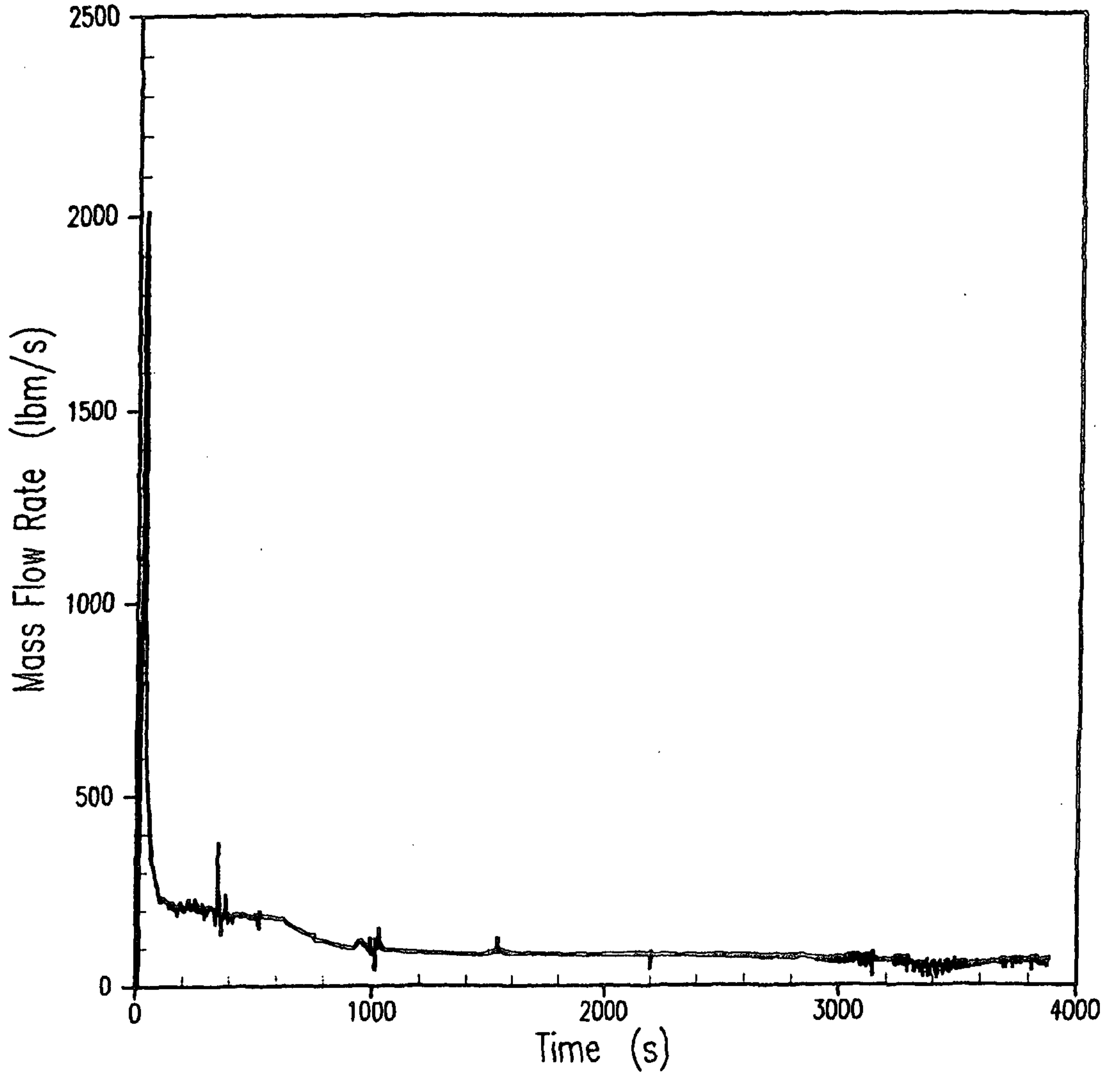
Figure 15.6-17k



**Byron/Braidwood Stations
Updated Final Safety Analysis
Report**

**Core Mixture Level - Unit 1
(4 Inch, Nominal T_{AVG})**

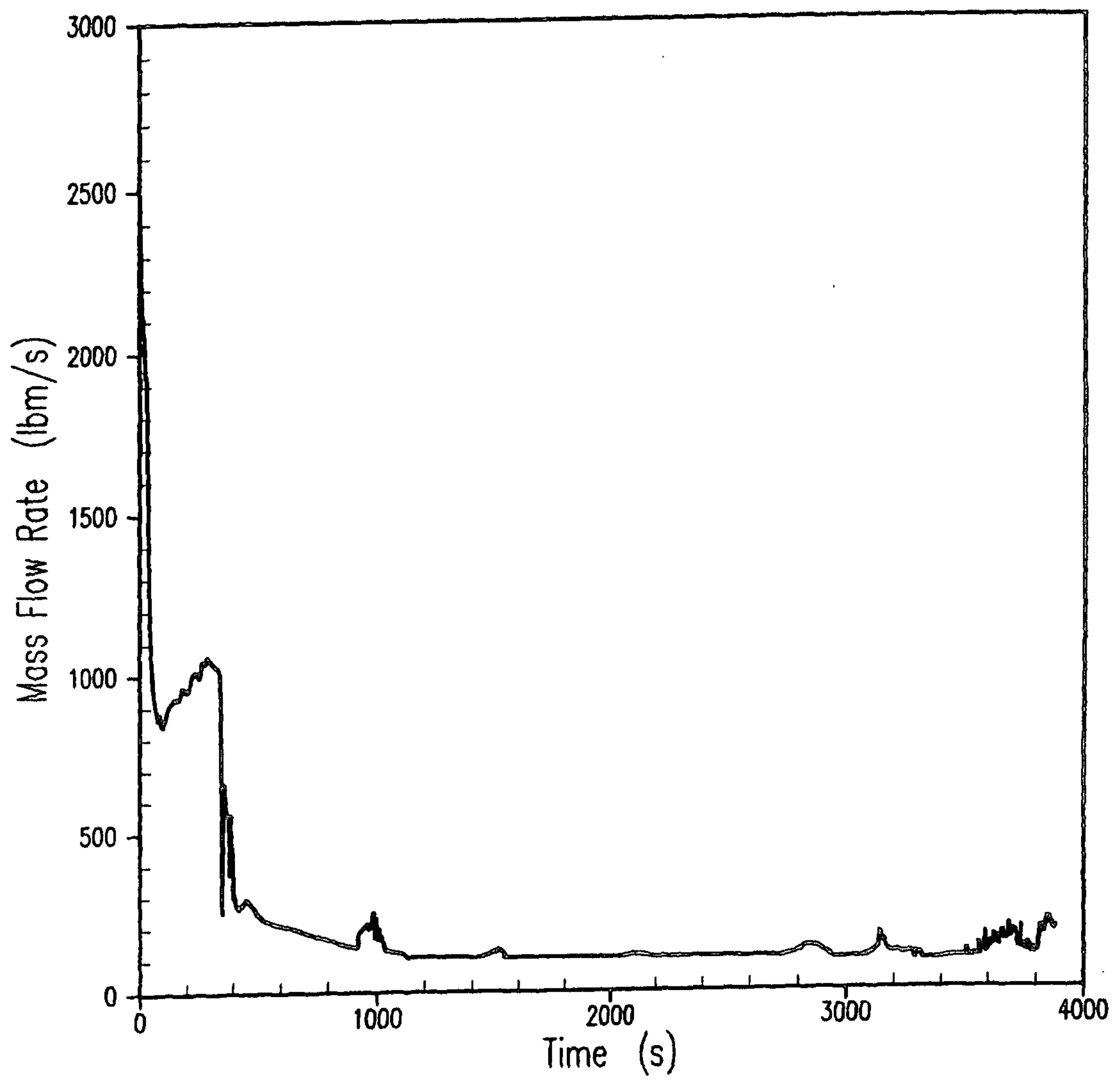
Figure 15.6-171



Byron/Braidwood Stations
Updated Final Safety Analysis Report

Core Steam Flow Rate - Unit 1
(4 Inch, Nominal T_{AVG})

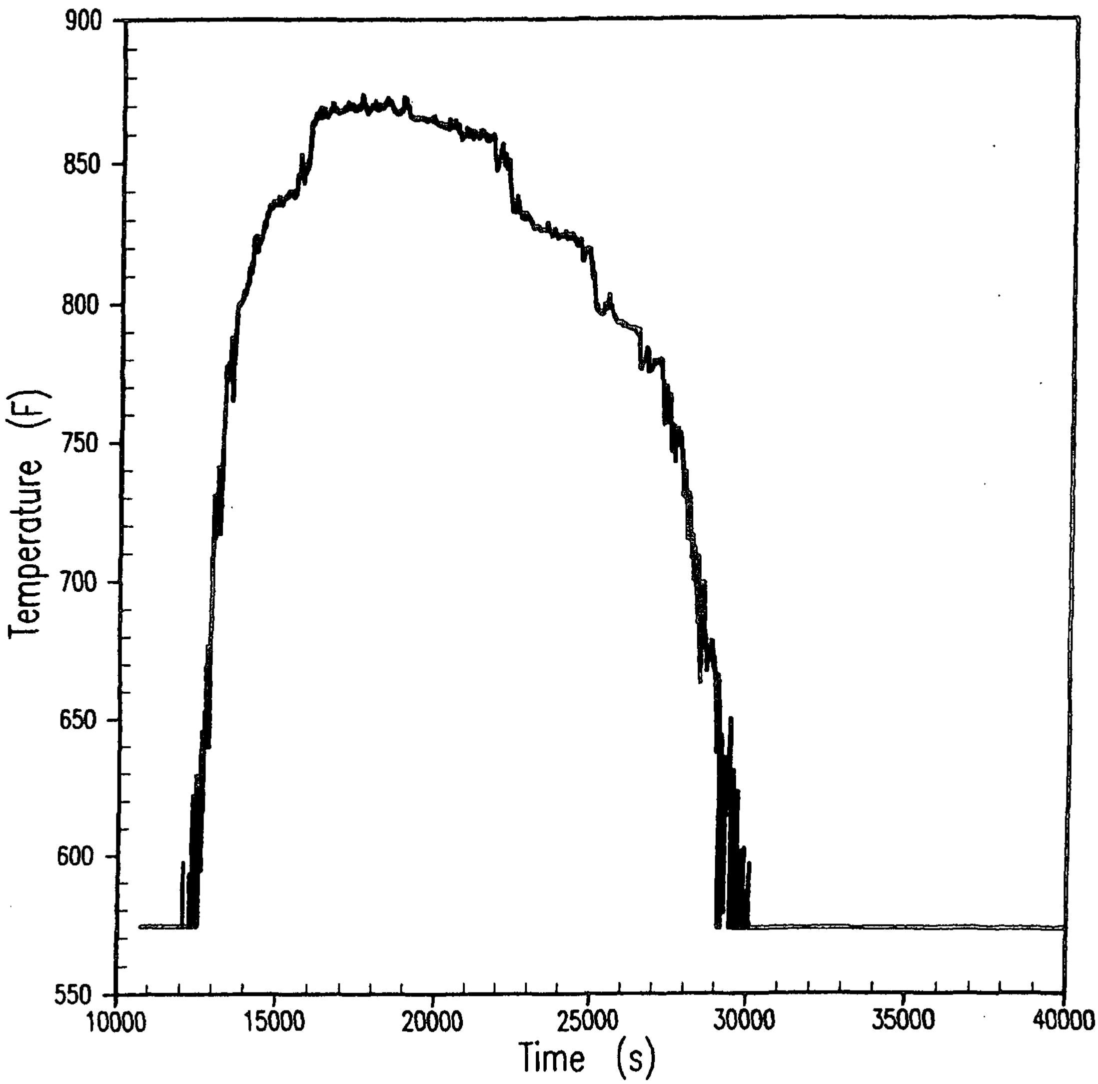
Figure 15.6-17m



Byron/Braidwood Stations Updated Final Safety Analysis Report

Break Flow - Unit 1 (4 Inch, Nominal T_{AVG})

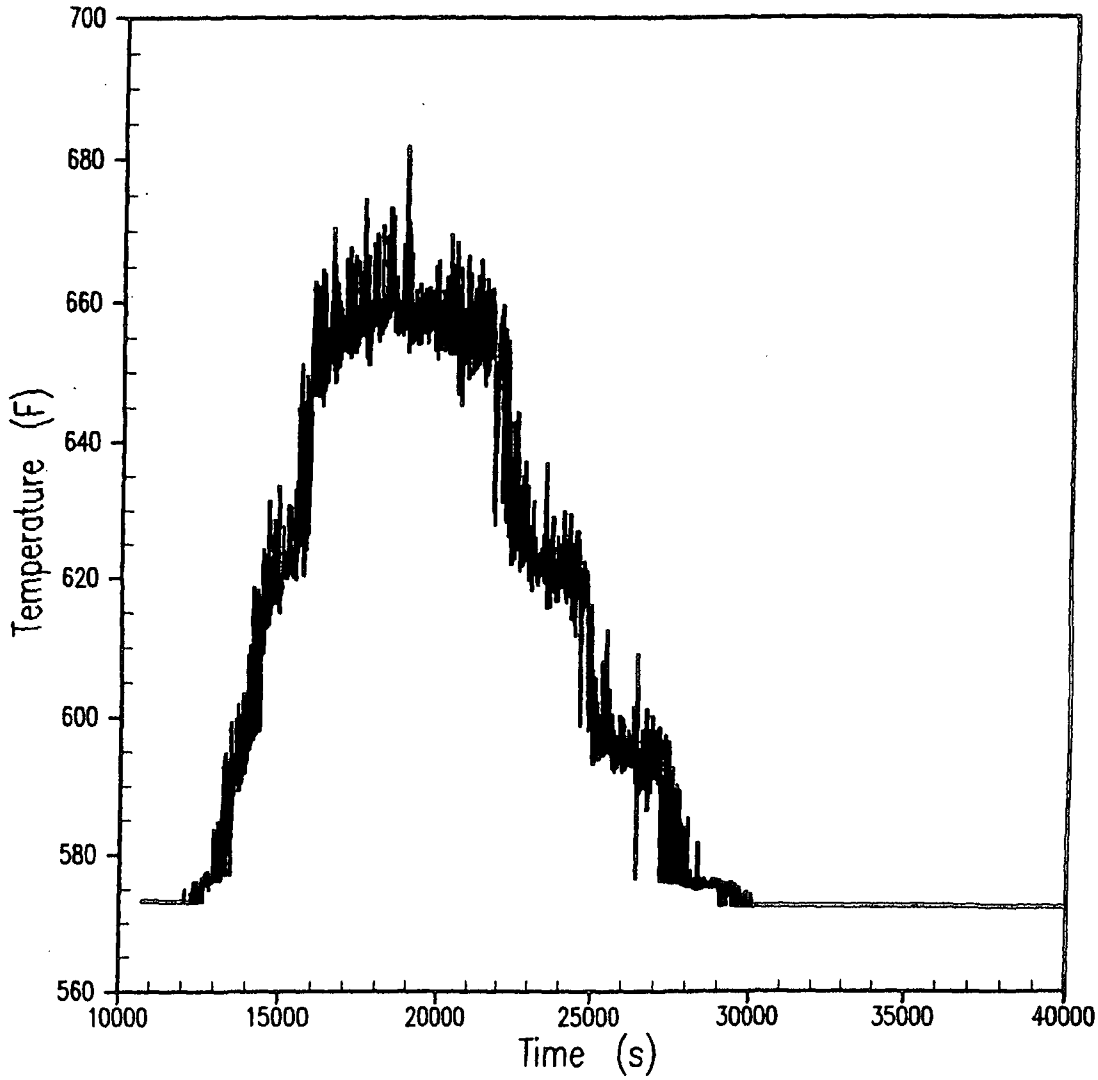
Figure 15.6-17n



Byron/Braidwood Stations Updated Final Safety Analysis Report

Peak Clad Temperature - Unit 2 (1.5 Inch, Reduced T_{AVG})

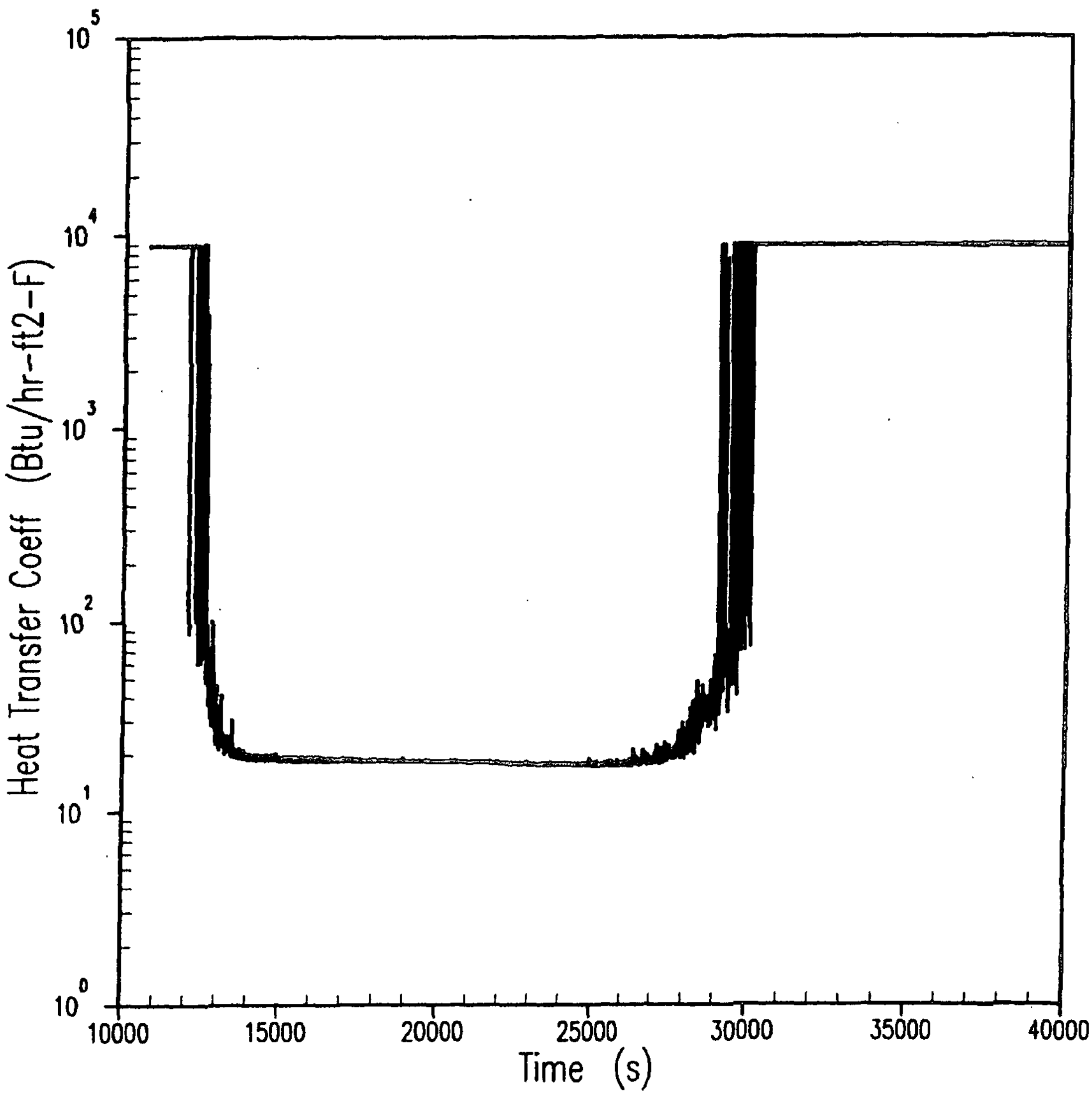
Figure 15.6-18a



Byron/Braidwood Stations Updated Final Safety Analysis Report

Fluid Temperature - Unit 2 (1.5 Inch, Reduced T_{AVG})

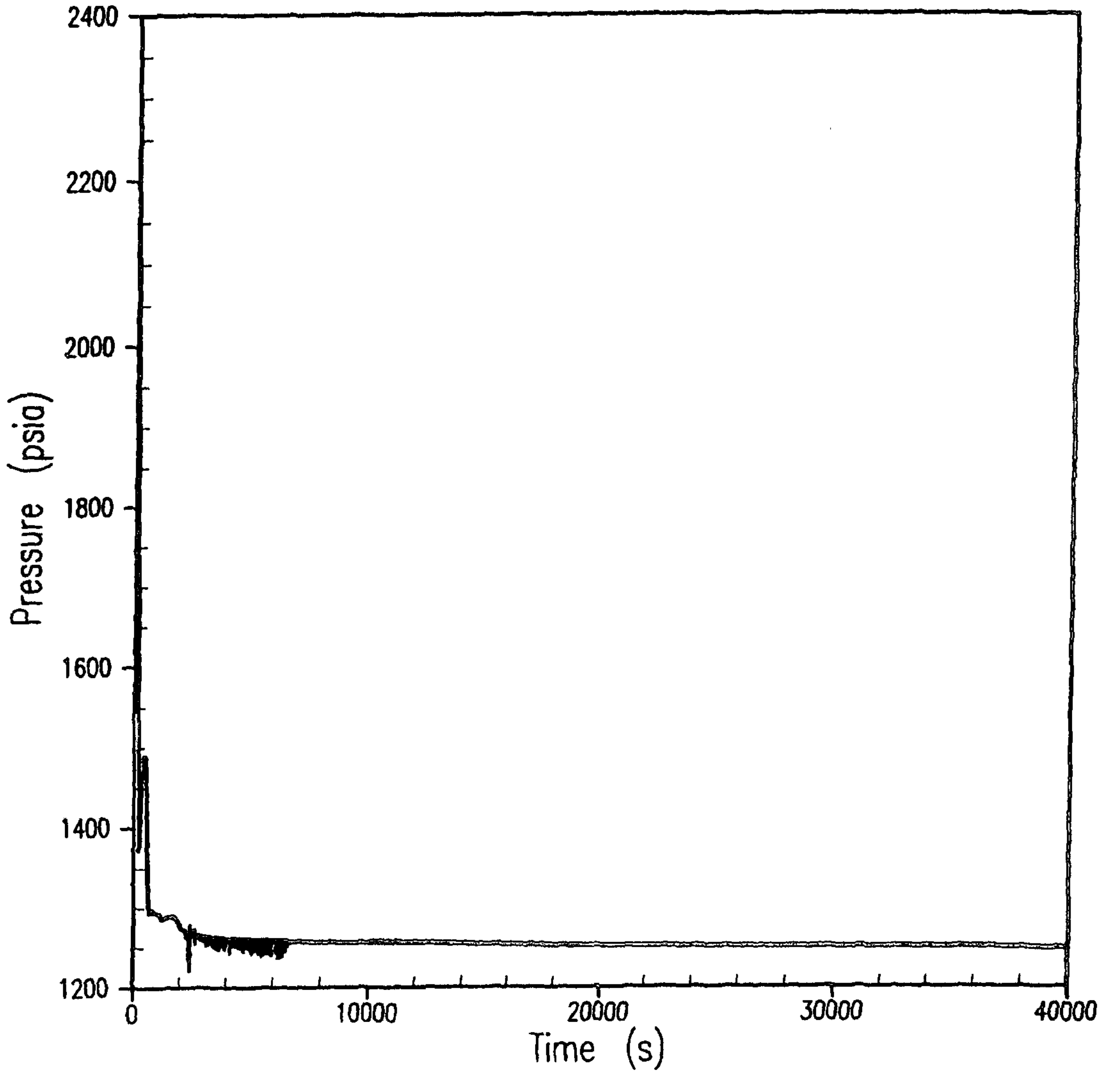
Figure 15.6-18b



**Byron/Braidwood Stations
Updated Final Safety Analysis
Report**

**Heat Transfer Coefficient - Unit 2
(1.5 Inch, Reduced T_{AVG})**

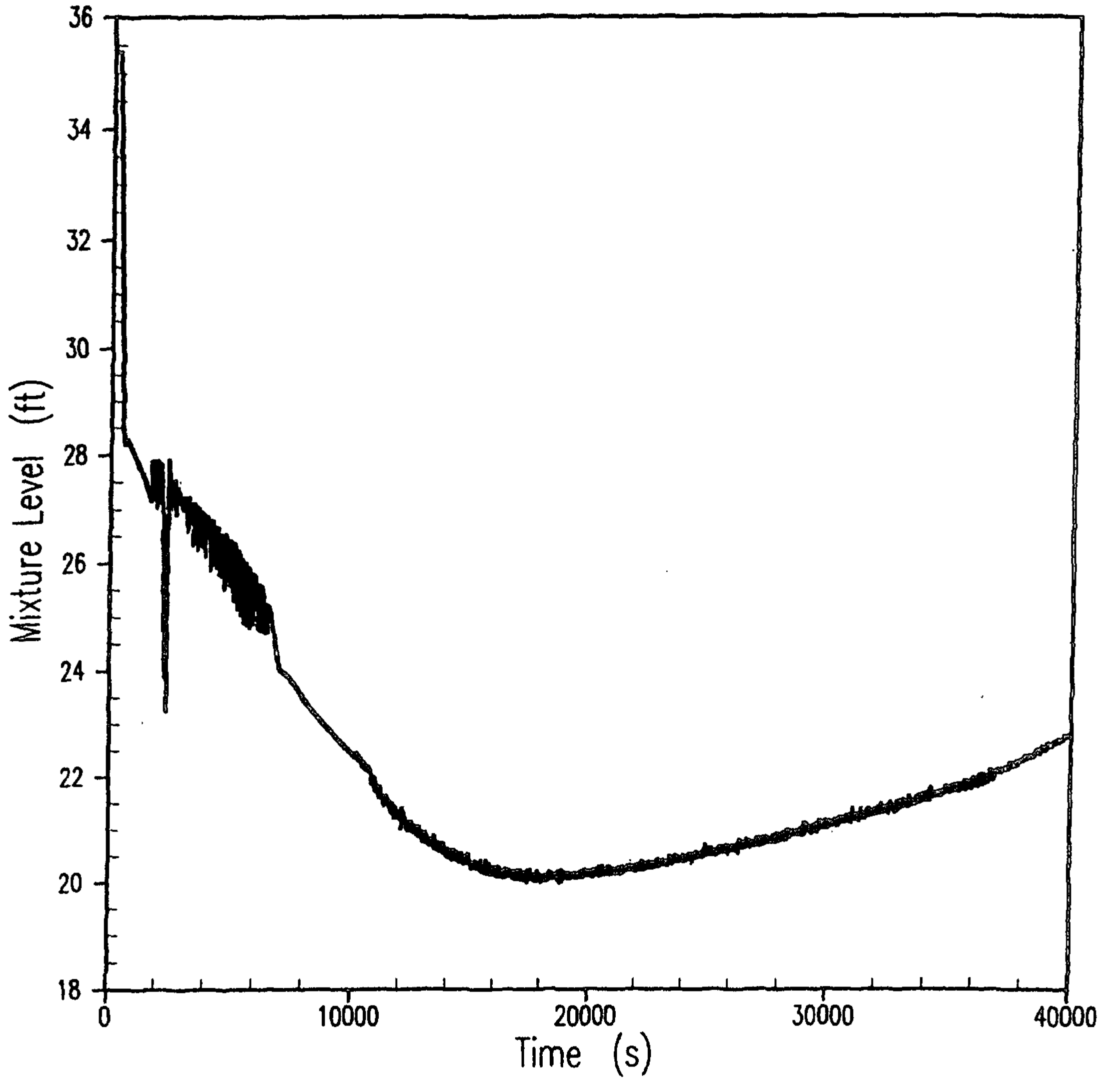
Figure 15.6-18c



Byron/Braidwood Stations
Updated Final Safety Analysis Report

RCS Pressure - Unit 2
(1.5 Inch, Reduced T_{AVG})

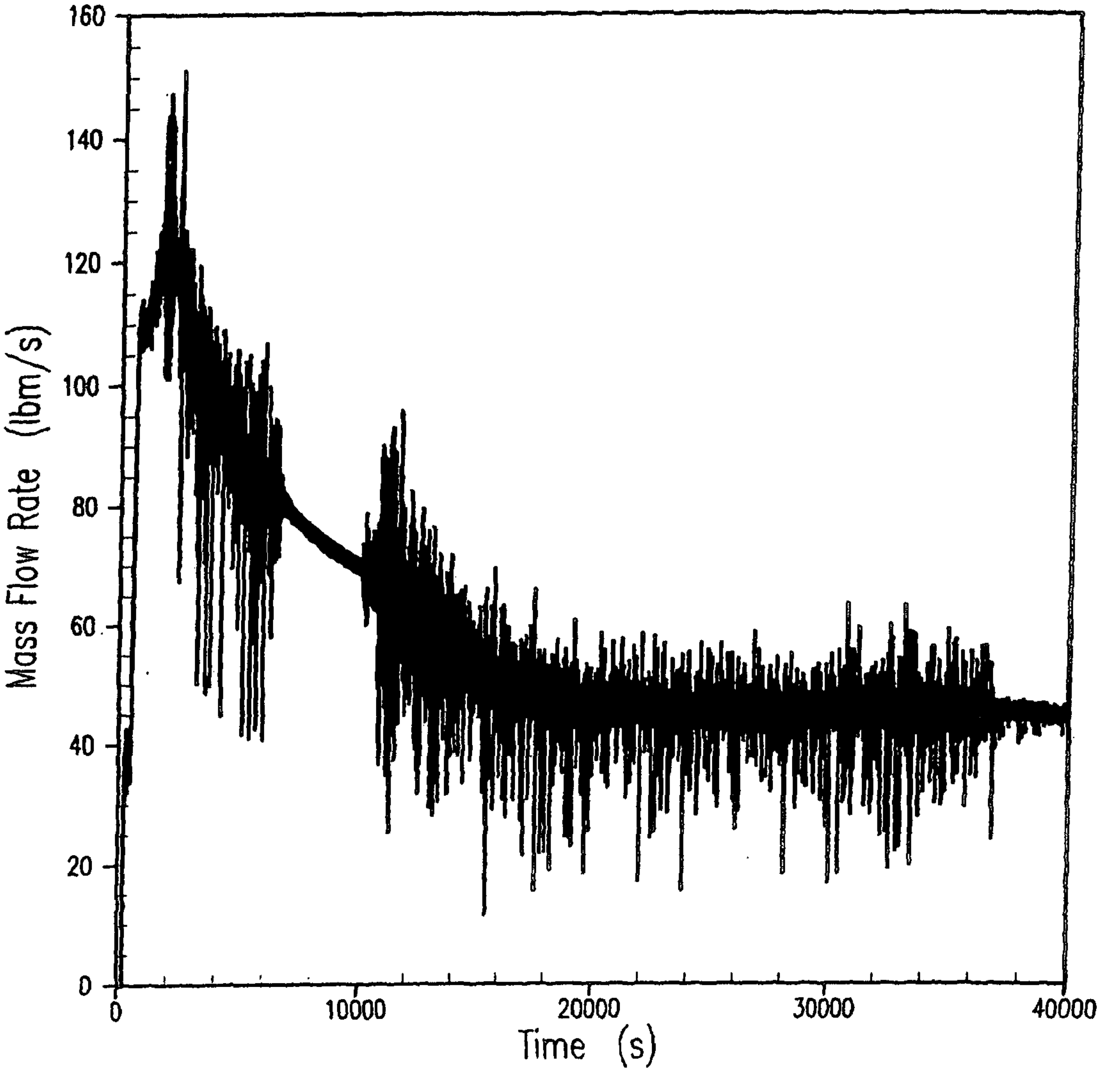
Figure 15.6-18d



Byron/Braidwood Stations Updated Final Safety Analysis Report

Core Mixture Level - Unit 2 (1.5 Inch, Reduced T_{AVG})

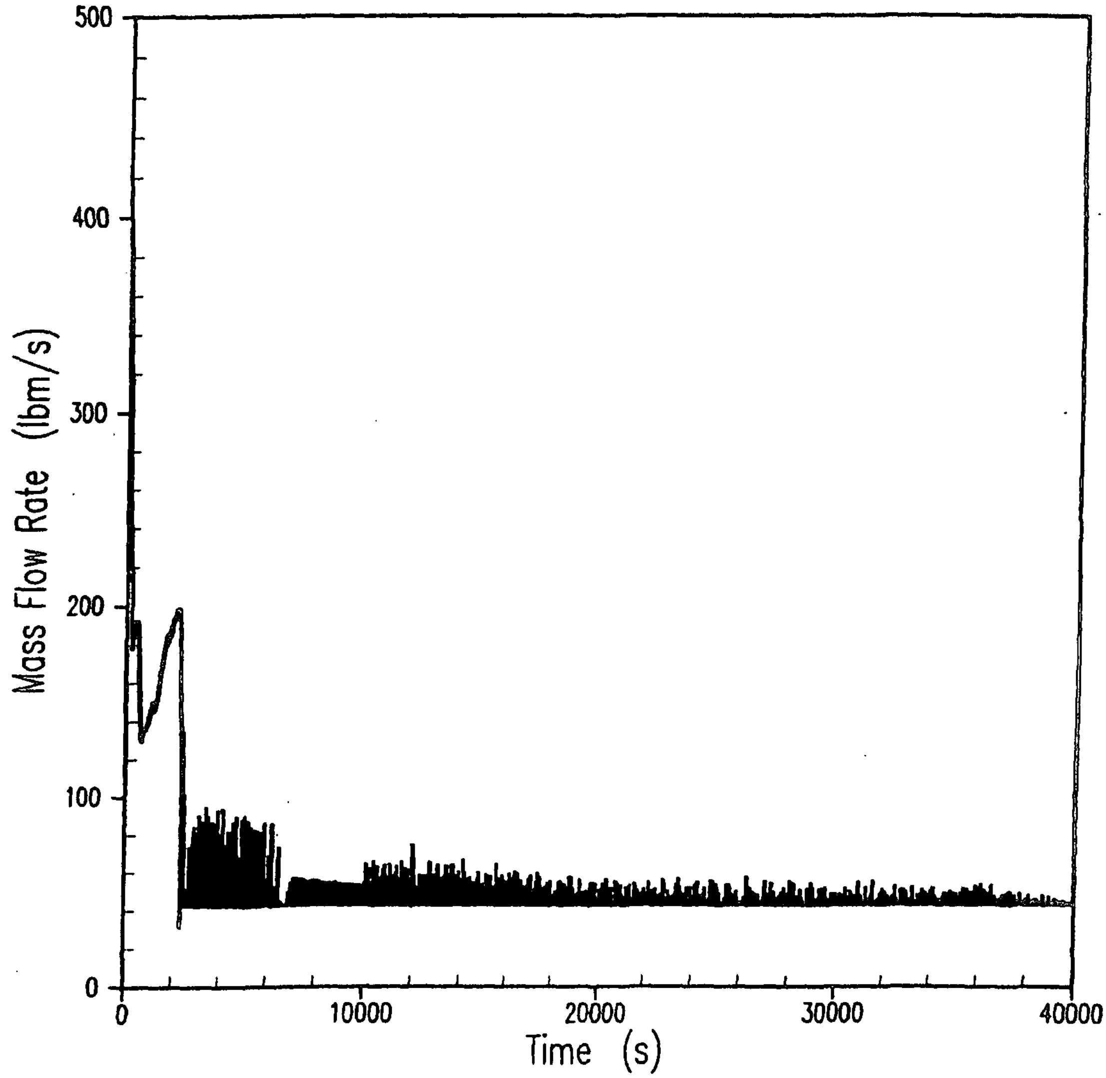
Figure 15.6-18e



**Byron/Braidwood Stations
Updated Final Safety Analysis
Report**

**Core Steam Flow Rate - Unit 2
(1.5 Inch, Reduced T_{AVG})**

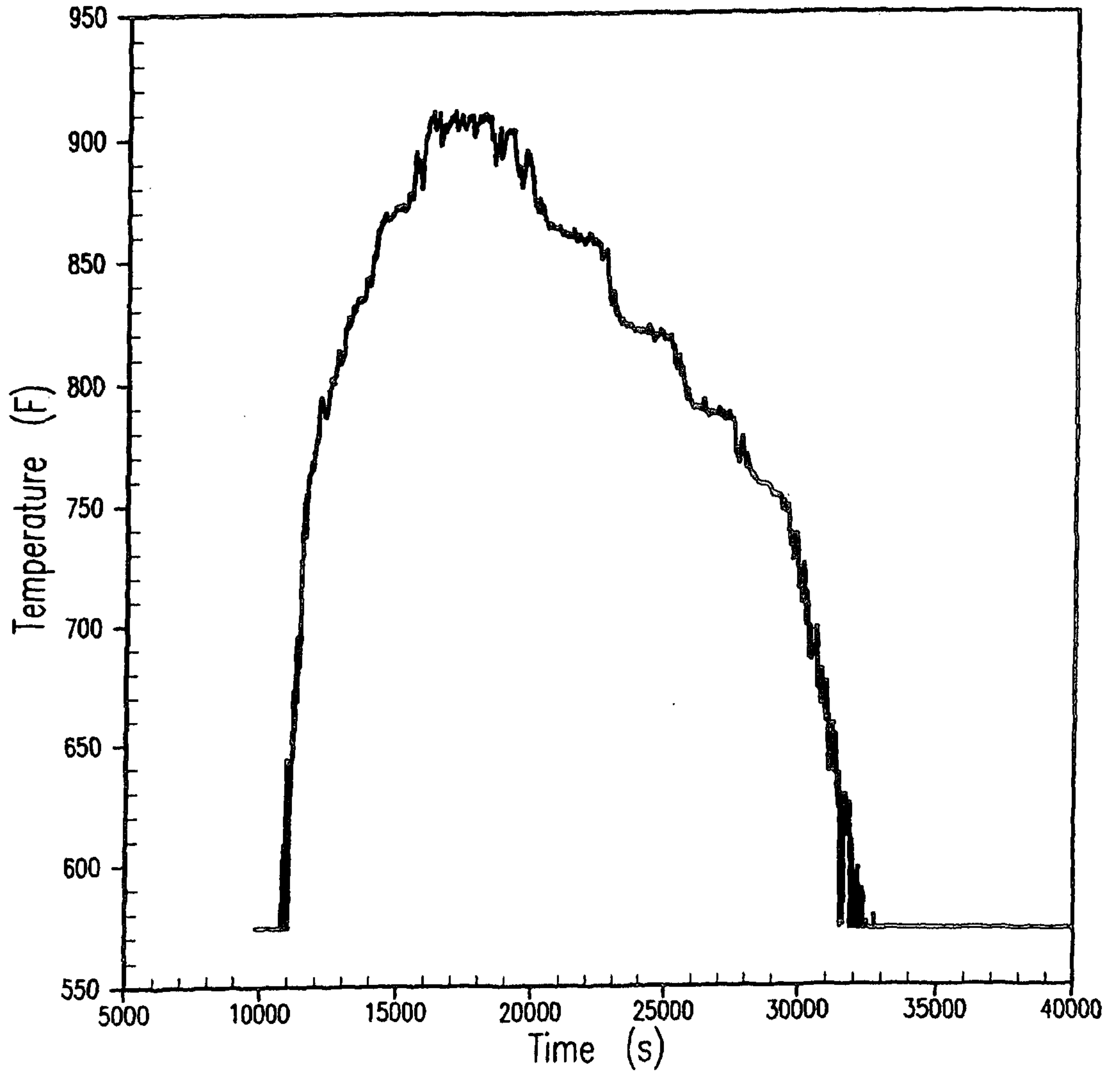
Figure 15.6-18f



Byron/Braidwood Stations Updated Final Safety Analysis Report

Break Flow - Unit 2 (1.5 Inch, Reduced T_{AVG})

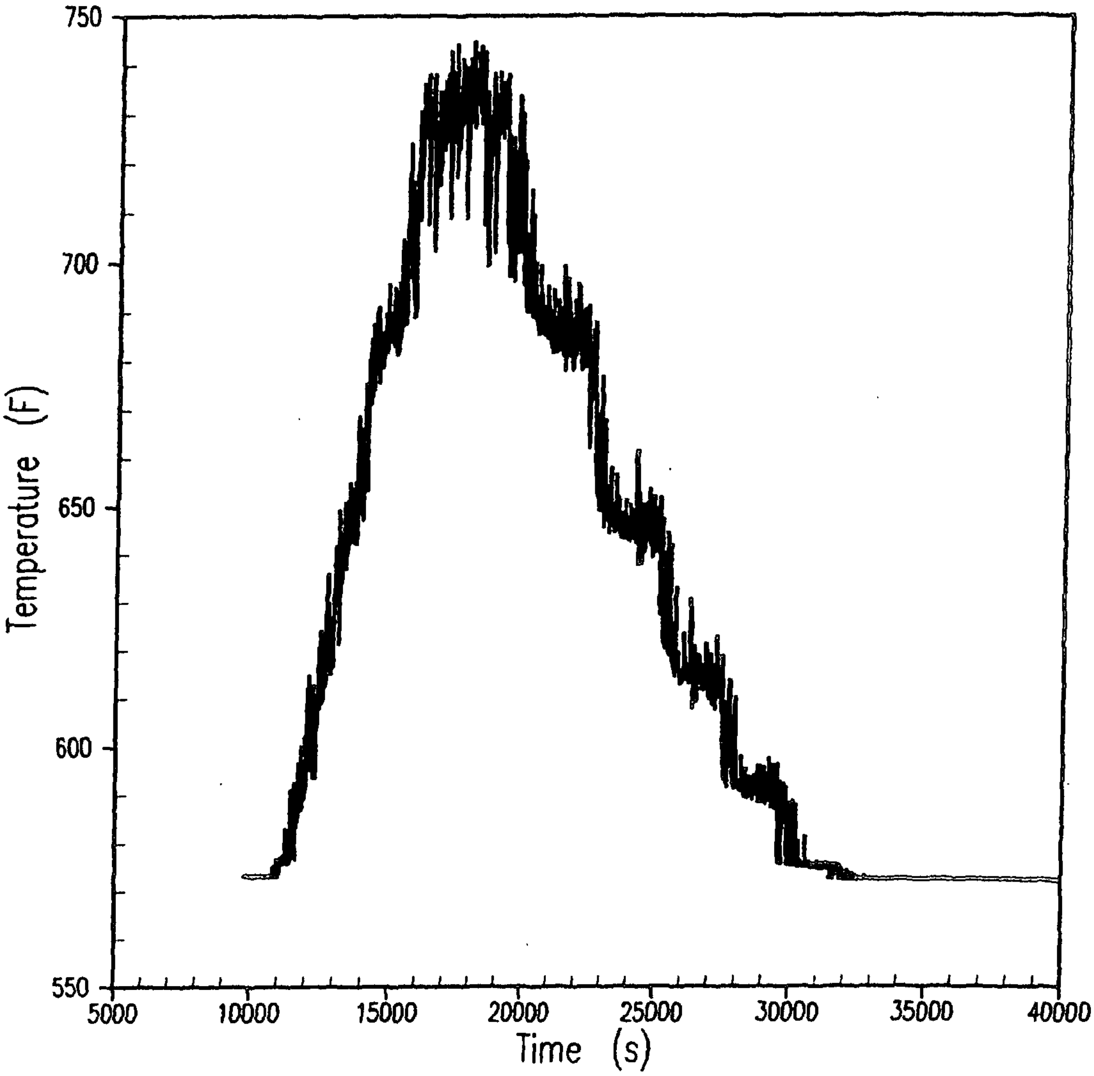
Figure 15.6-18g



Byron/Braidwood Stations Updated Final Safety Analysis Report

Peak Clad Temperature - Unit 2
(1.5 Inch, Nominal T_{AVG})

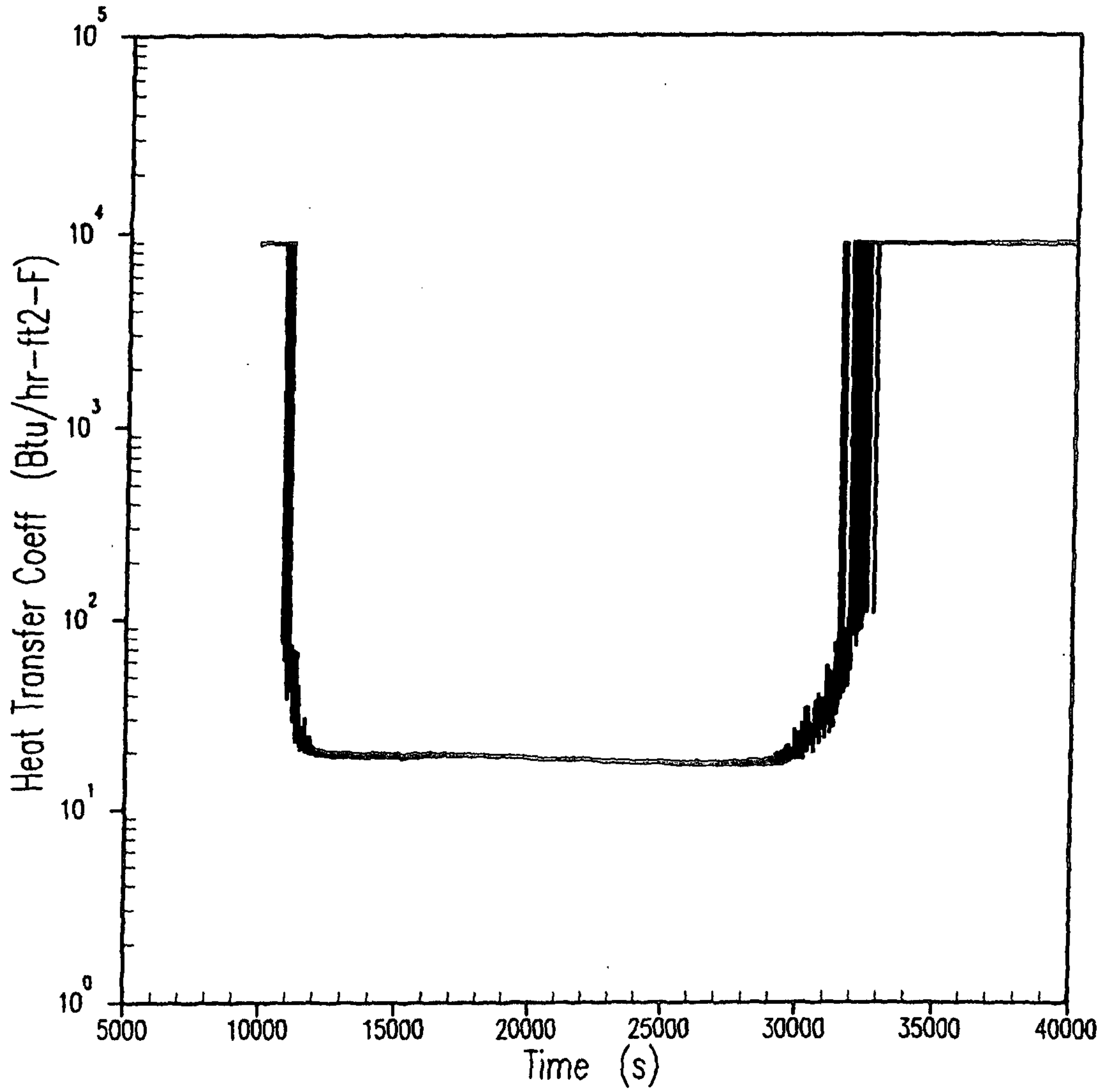
Figure 15.6-18h



Byron/Braidwood Stations
Updated Final Safety Analysis Report

Fluid Temperature - Unit 2
(1.5 Inch, Nominal T_{AVG})

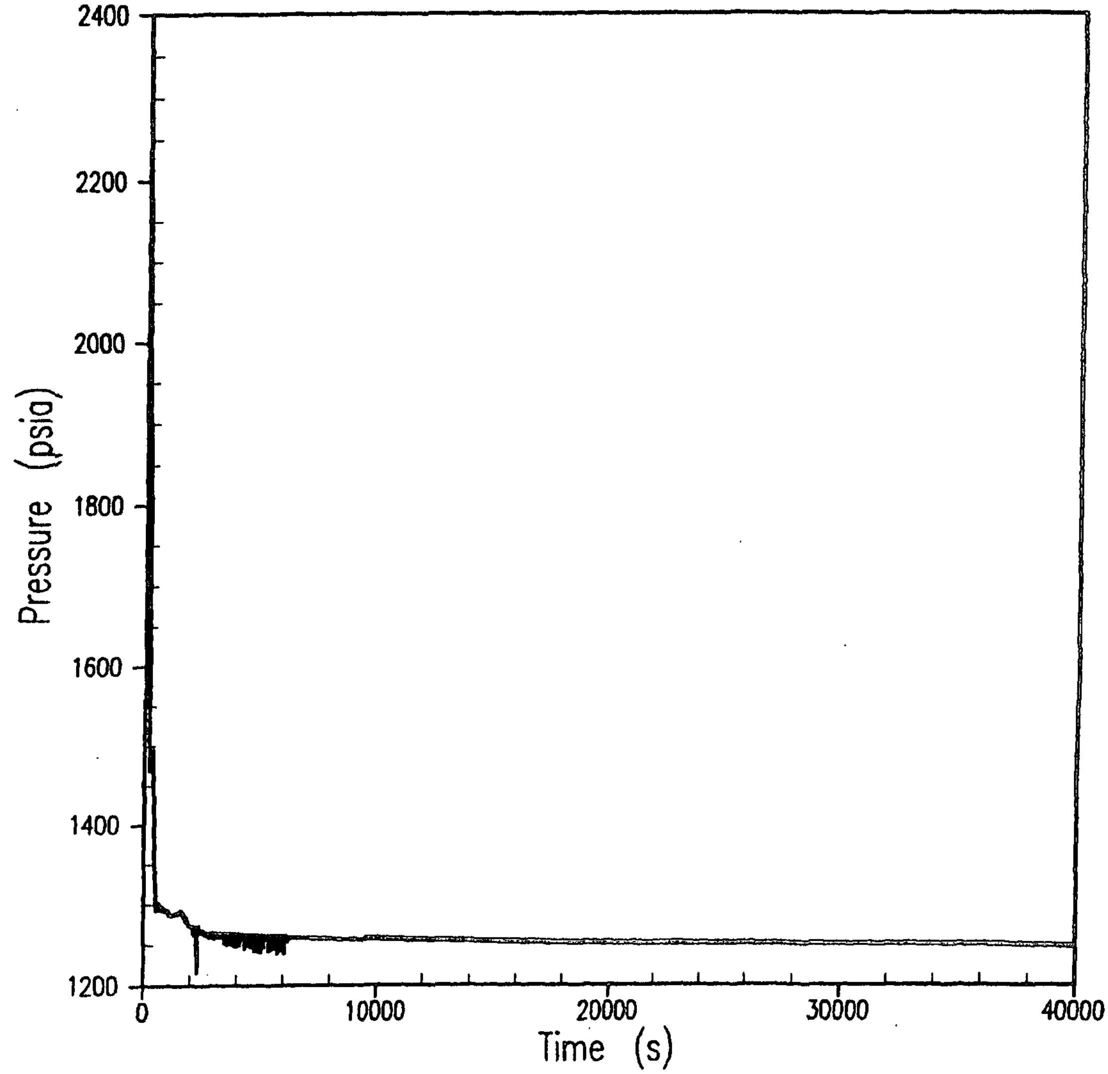
Figure 15.6-18I



Byron/Braidwood Stations Updated Final Safety Analysis Report

Heat Transfer Coefficient - Unit 2 (1.5 Inch, Nominal T_{AVG})

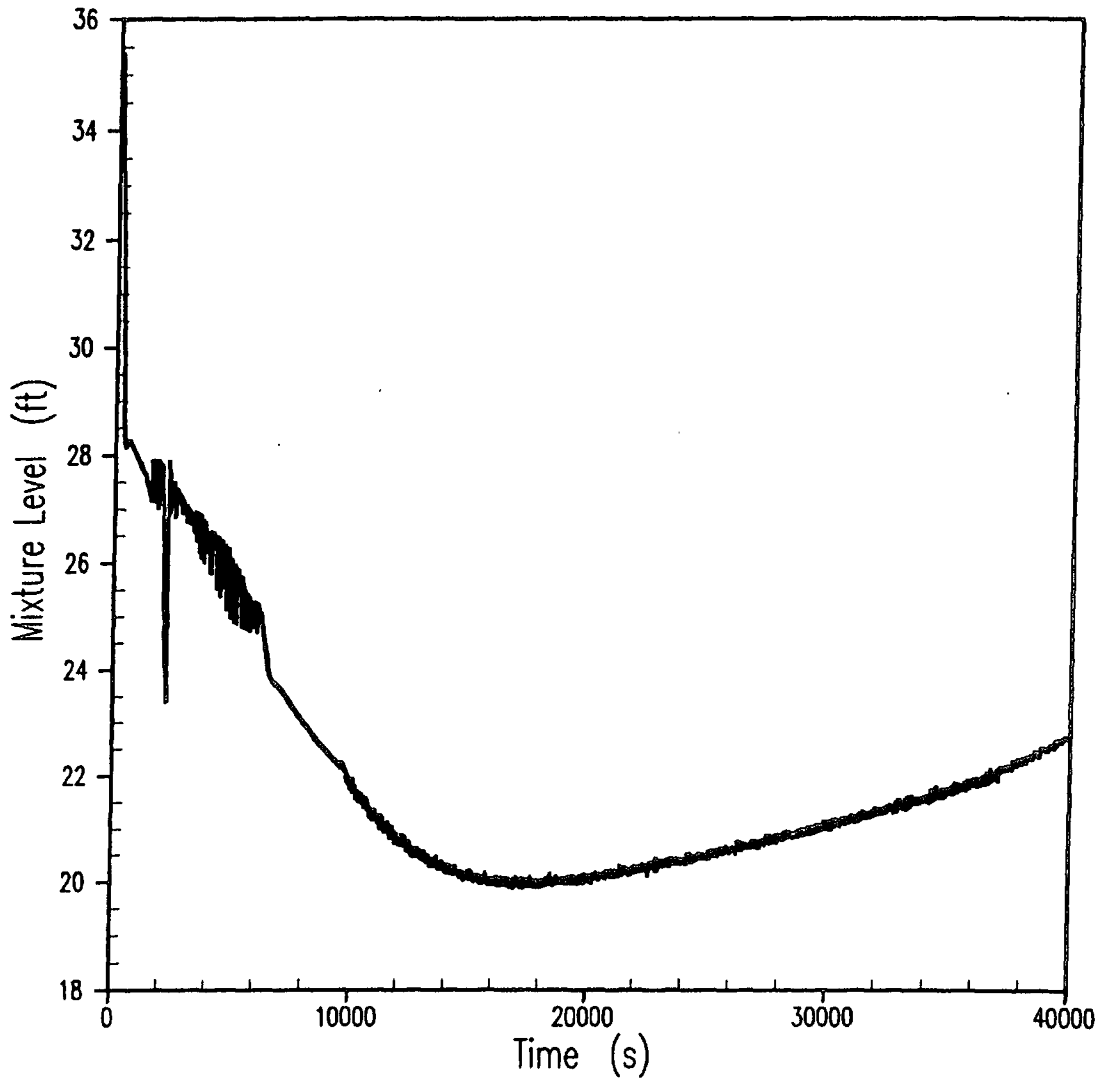
Figure 15.6-18j



Byron/Braidwood Stations Updated Final Safety Analysis Report

RCS Pressure - Unit 2
(1.5 Inch, Nominal T_{AVG})

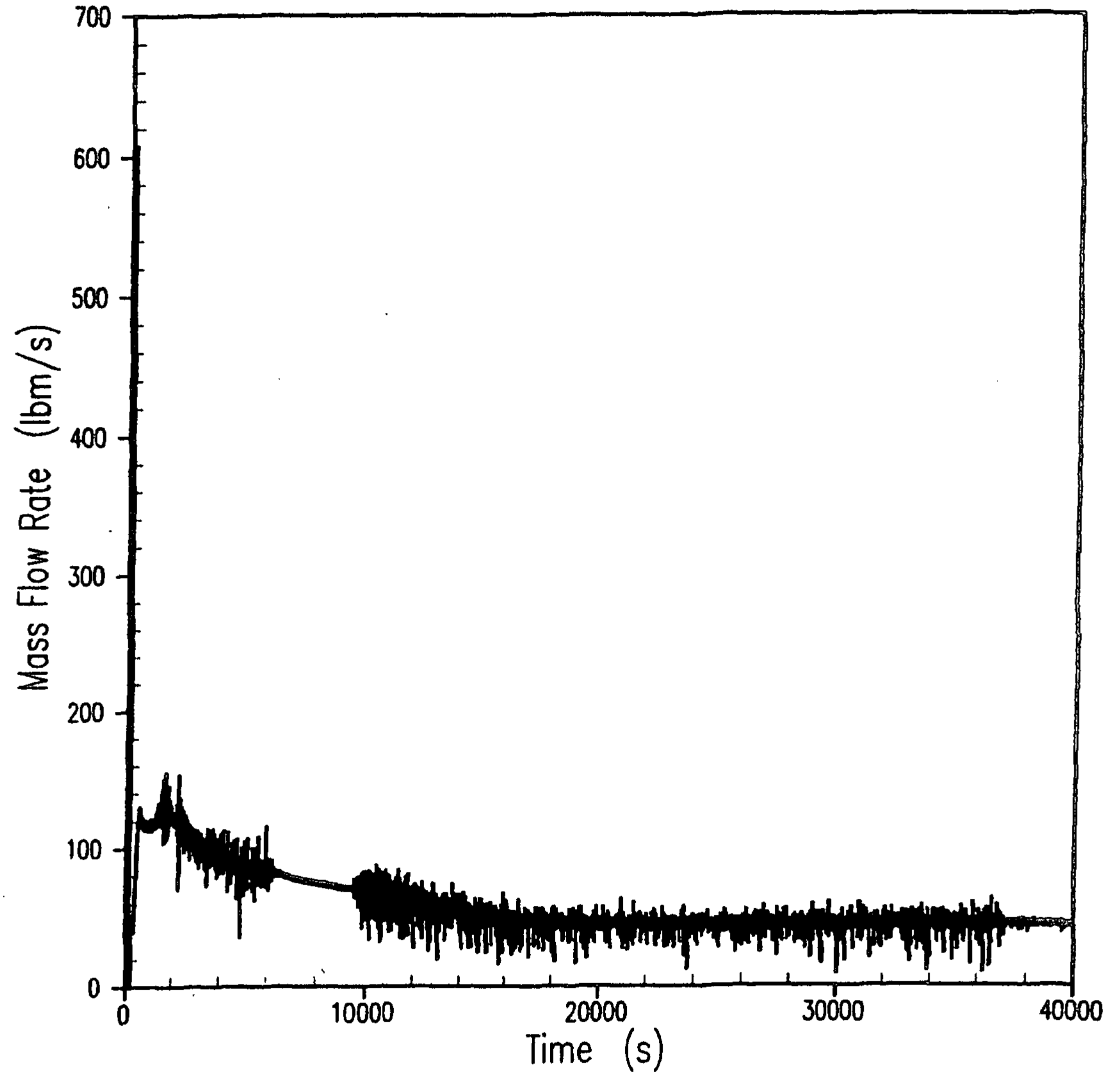
Figure 15.6-18k



Byron/Braidwood Stations Updated Final Safety Analysis Report

Core Mixture Level - Unit 2 (1.5 Inch, Nominal T_{AVG})

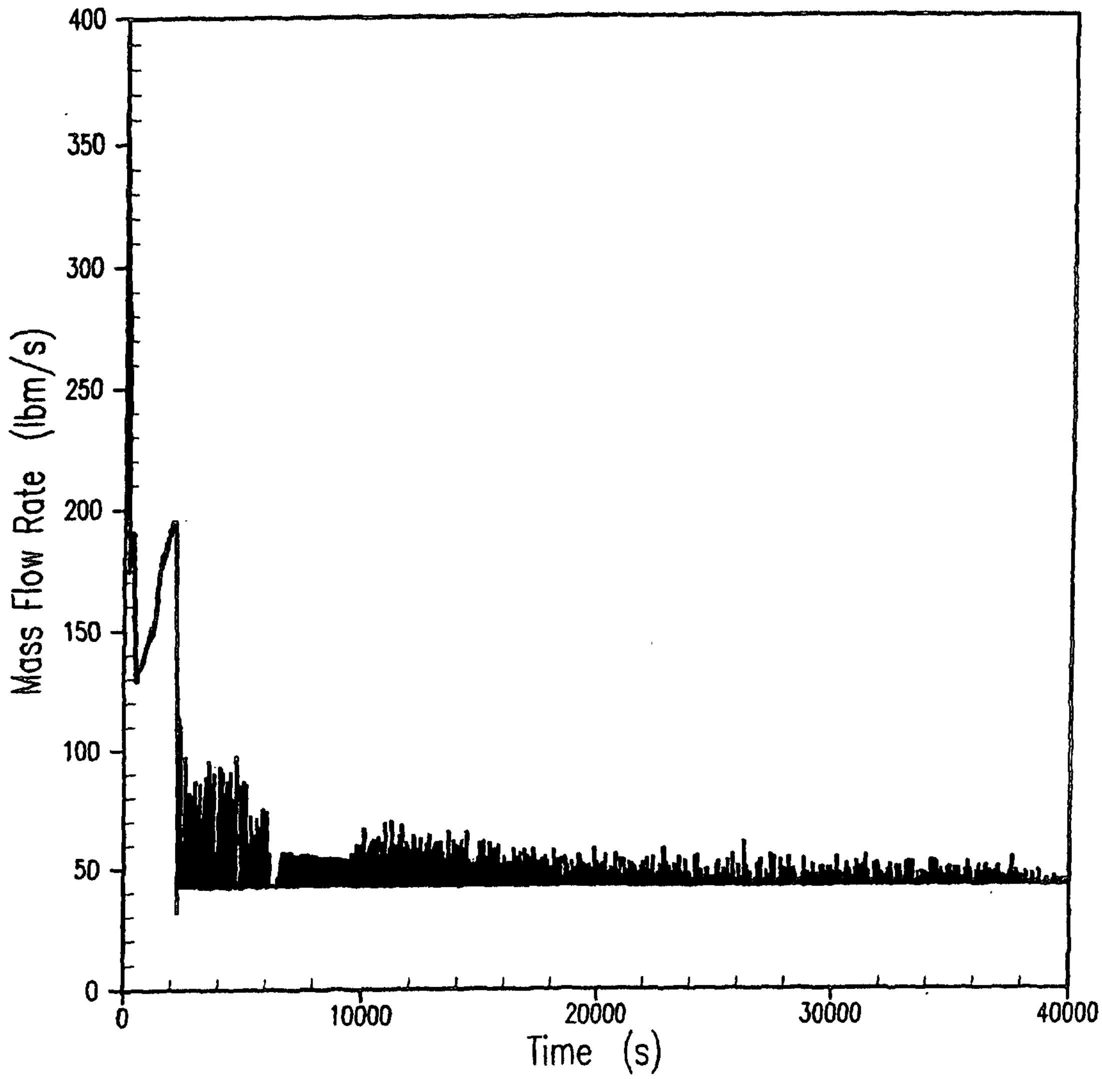
Figure 15.6-18I



Byron/Braidwood Stations Updated Final Safety Analysis Report

Core Steam Flow Rate - Unit 2 (1.5 Inch, Nominal T_{AVG})

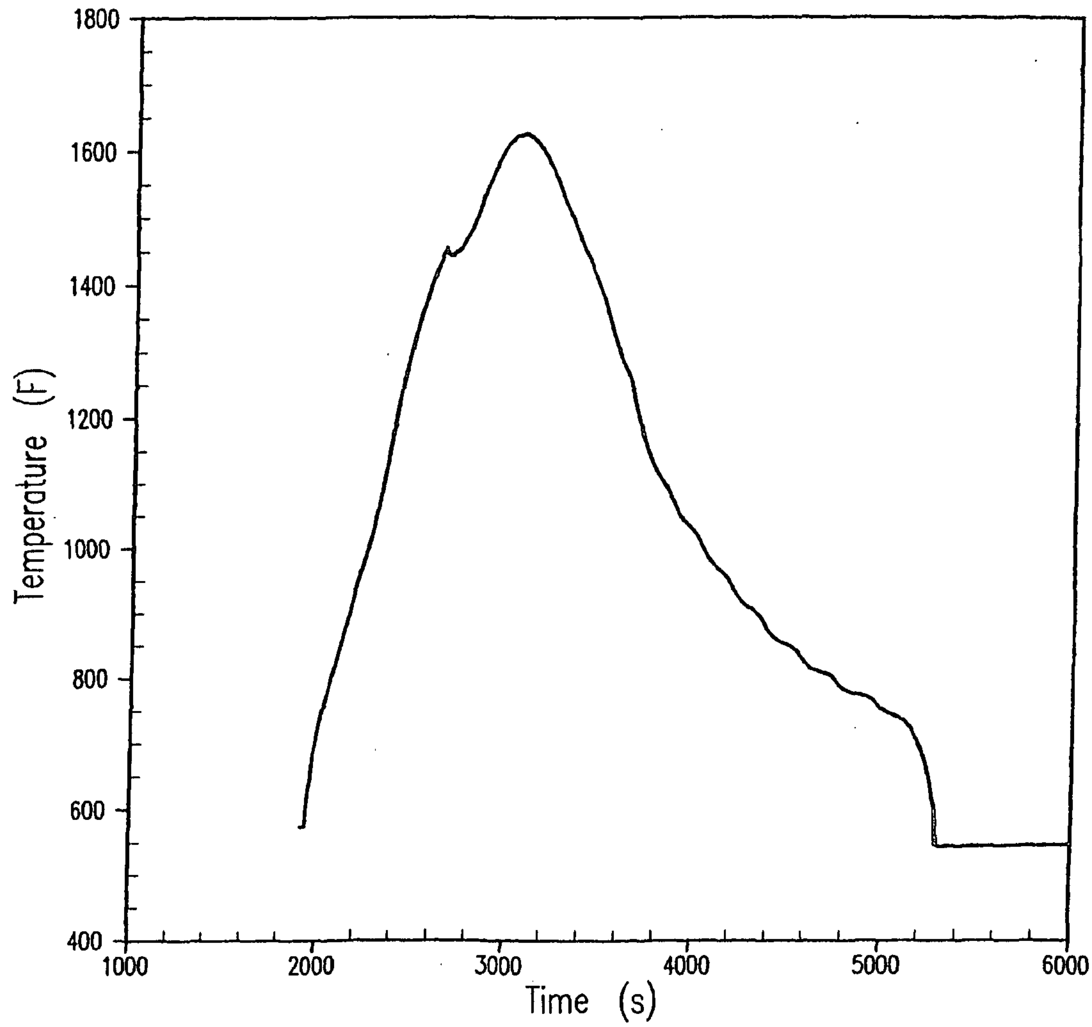
Figure 15.6-18m



Byron/Braidwood Stations
Updated Final Safety Analysis Report

Break Flow - Unit 2
(1.5 Inch, Nominal T_{AVG})

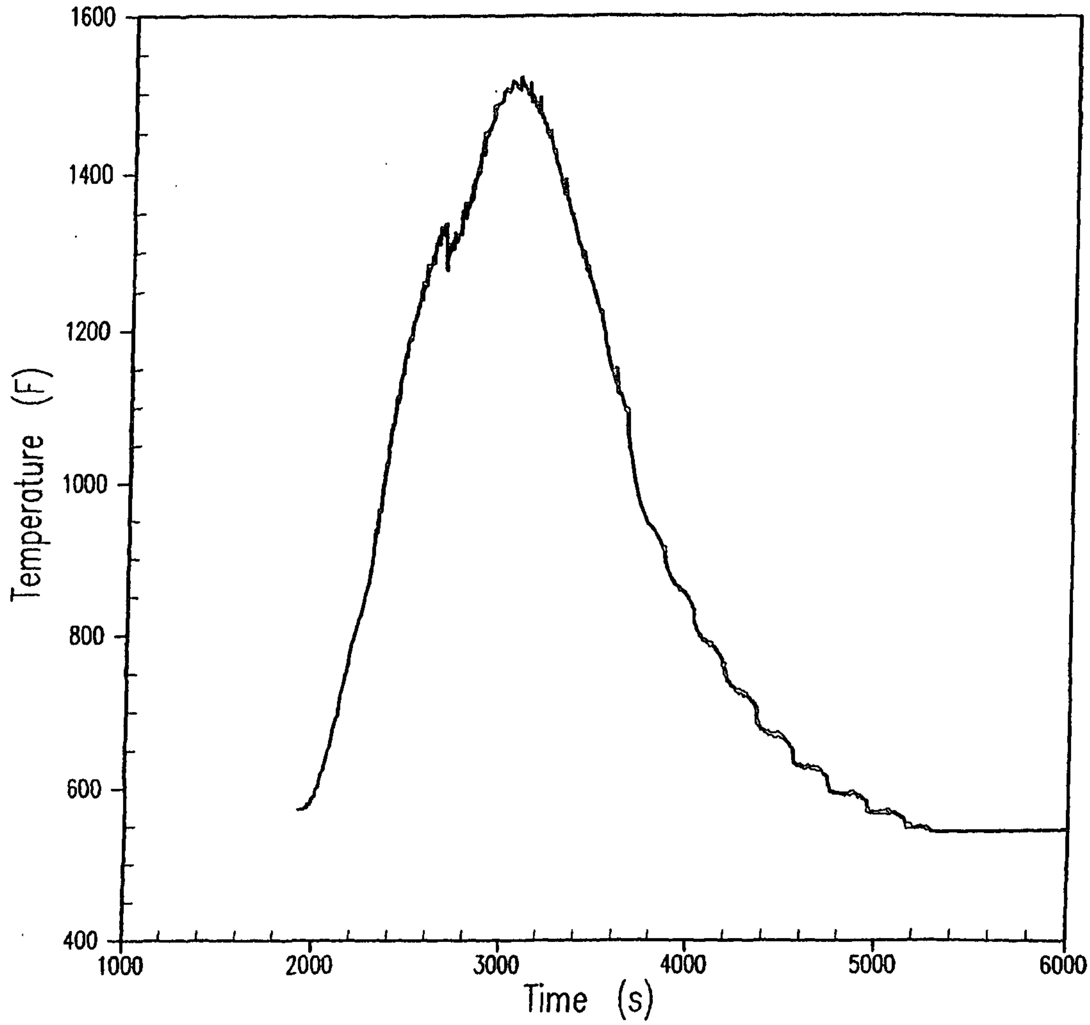
Figure 15.6-18n



**Byron/Braidwood Stations
Updated Final Safety Analysis
Report**

**Peak Clad Temperature - Unit 2
(2 Inch, Reduced T_{AVG})**

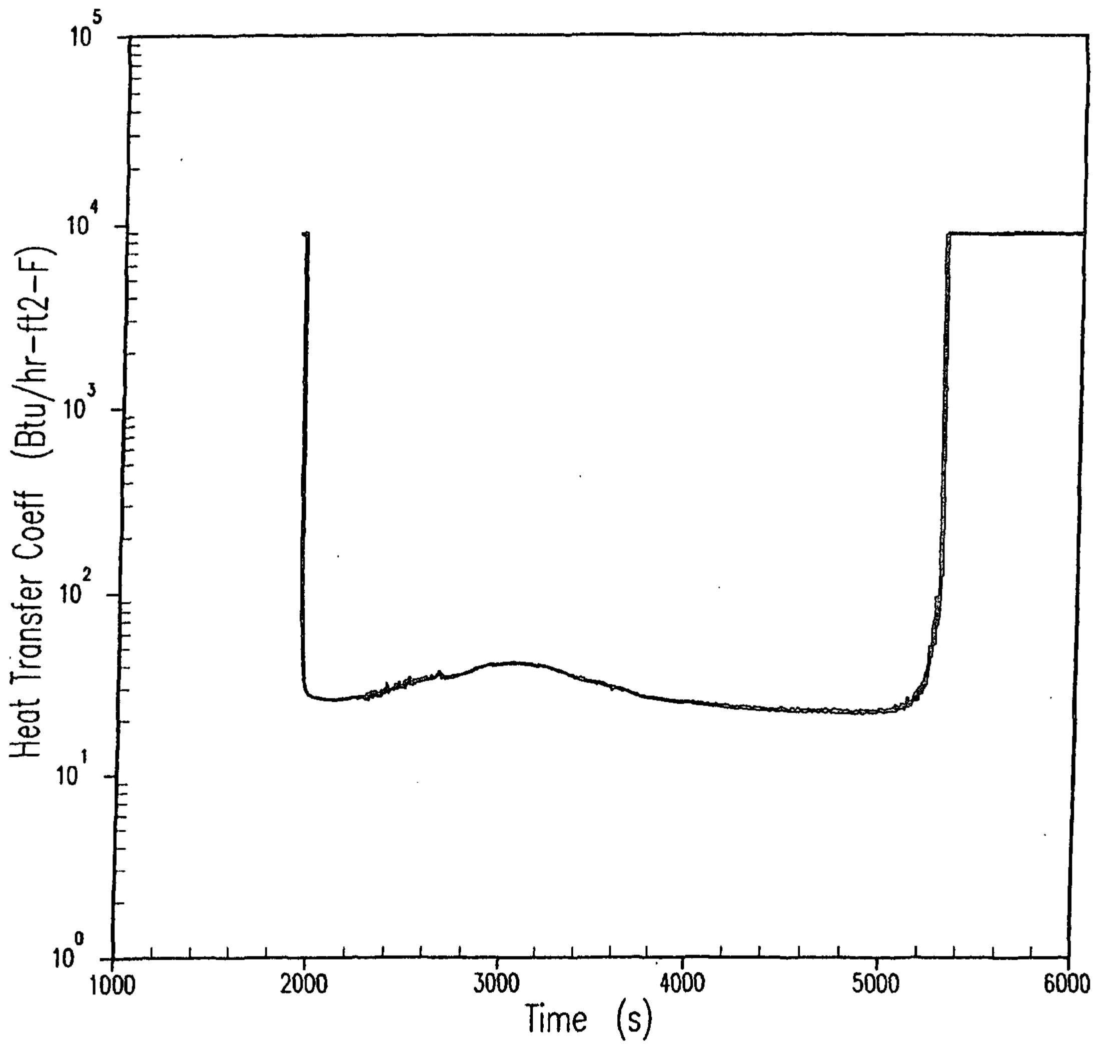
Figure 15.6-19a



Byron/Braidwood Stations Updated Final Safety Analysis Report

Fluid Temperature - Unit 2 (2 Inch, Reduced T_{AVG})

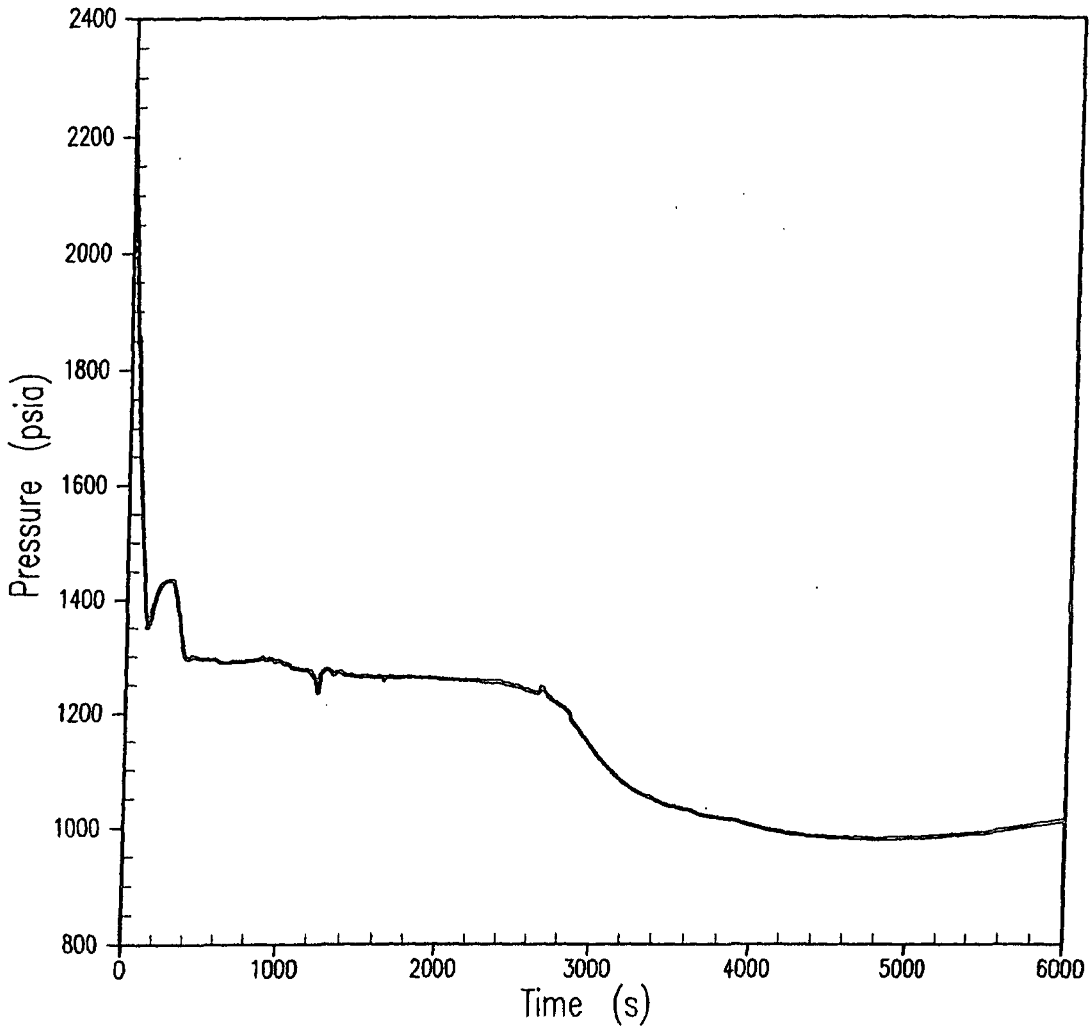
Figure 15.6-19b



Byron/Braidwood Stations
Updated Final Safety Analysis Report

Heat Transfer Coefficient - Unit 2
(2 Inch, Reduced T_{AVG})

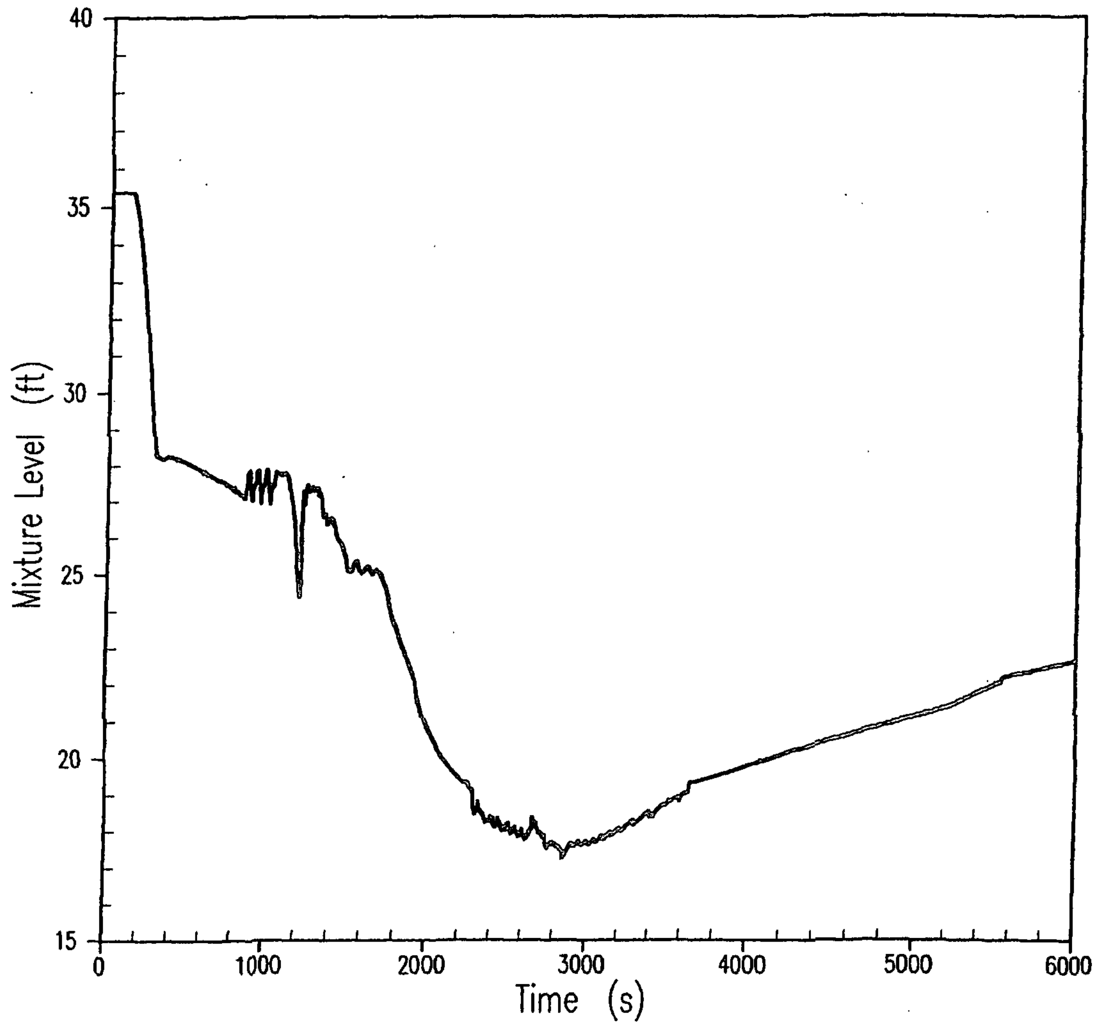
Figure 15.6-19c



Byron/Braidwood Stations Updated Final Safety Analysis Report

RCS Pressure - Unit 2
(2 Inch, Reduced T_{AVG})

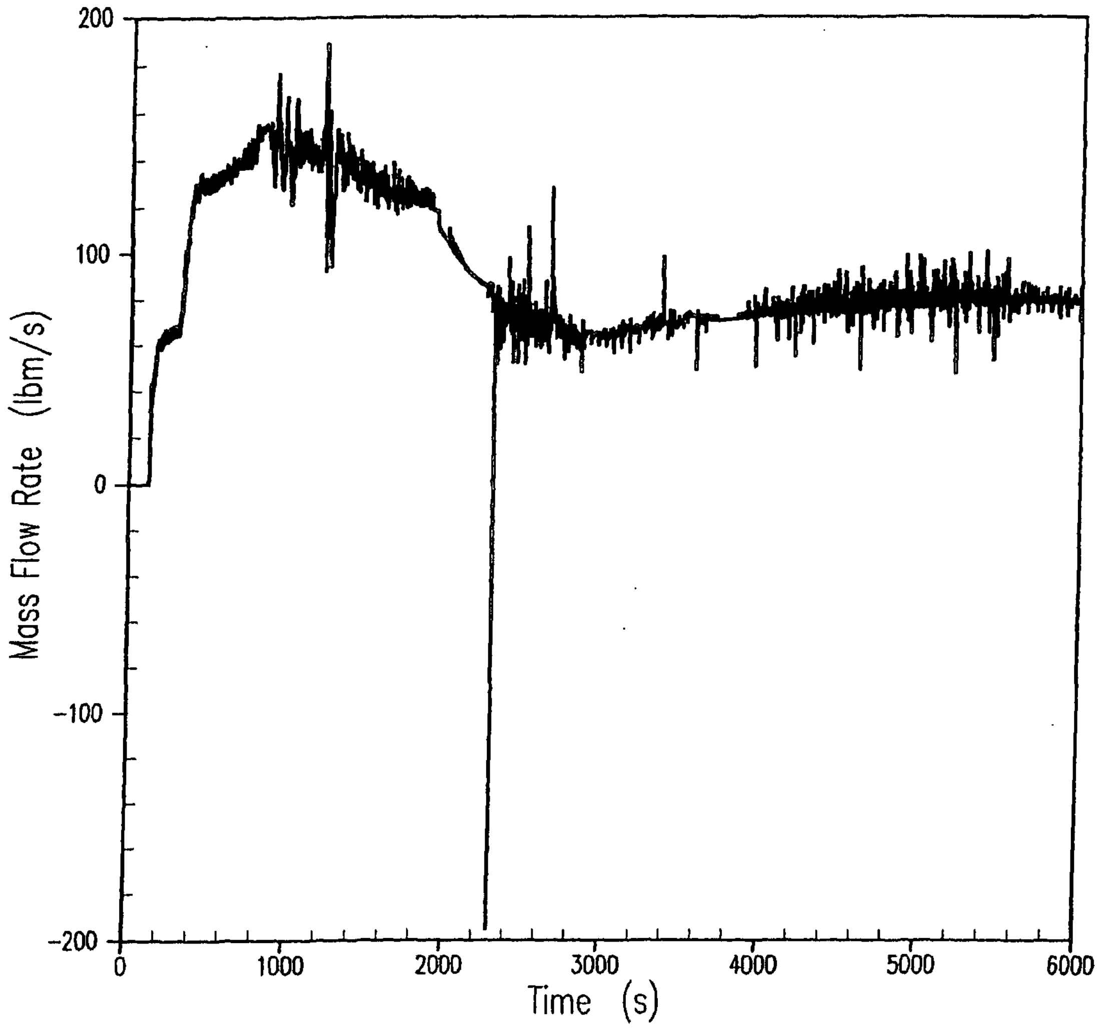
Figure 15.6-19d



Byron/Braidwood Stations Updated Final Safety Analysis Report

Core Mixture Level - Unit 2
(2 Inch, Reduced T_{AVG})

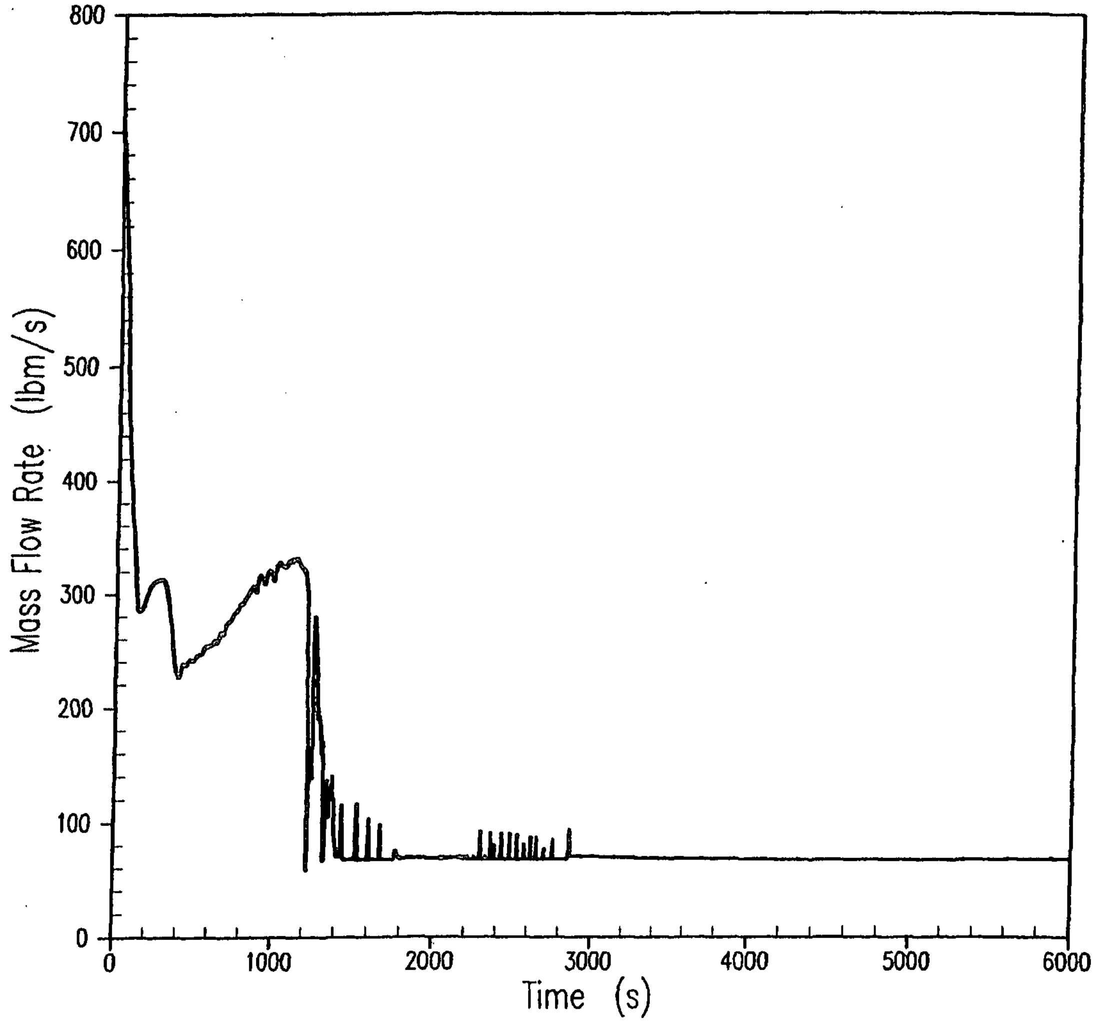
Figure 15.6-19e



Byron/Braidwood Stations Updated Final Safety Analysis Report

Core Steam Flow Rate - Unit 2
(2 Inch, Reduced T_{AVG})

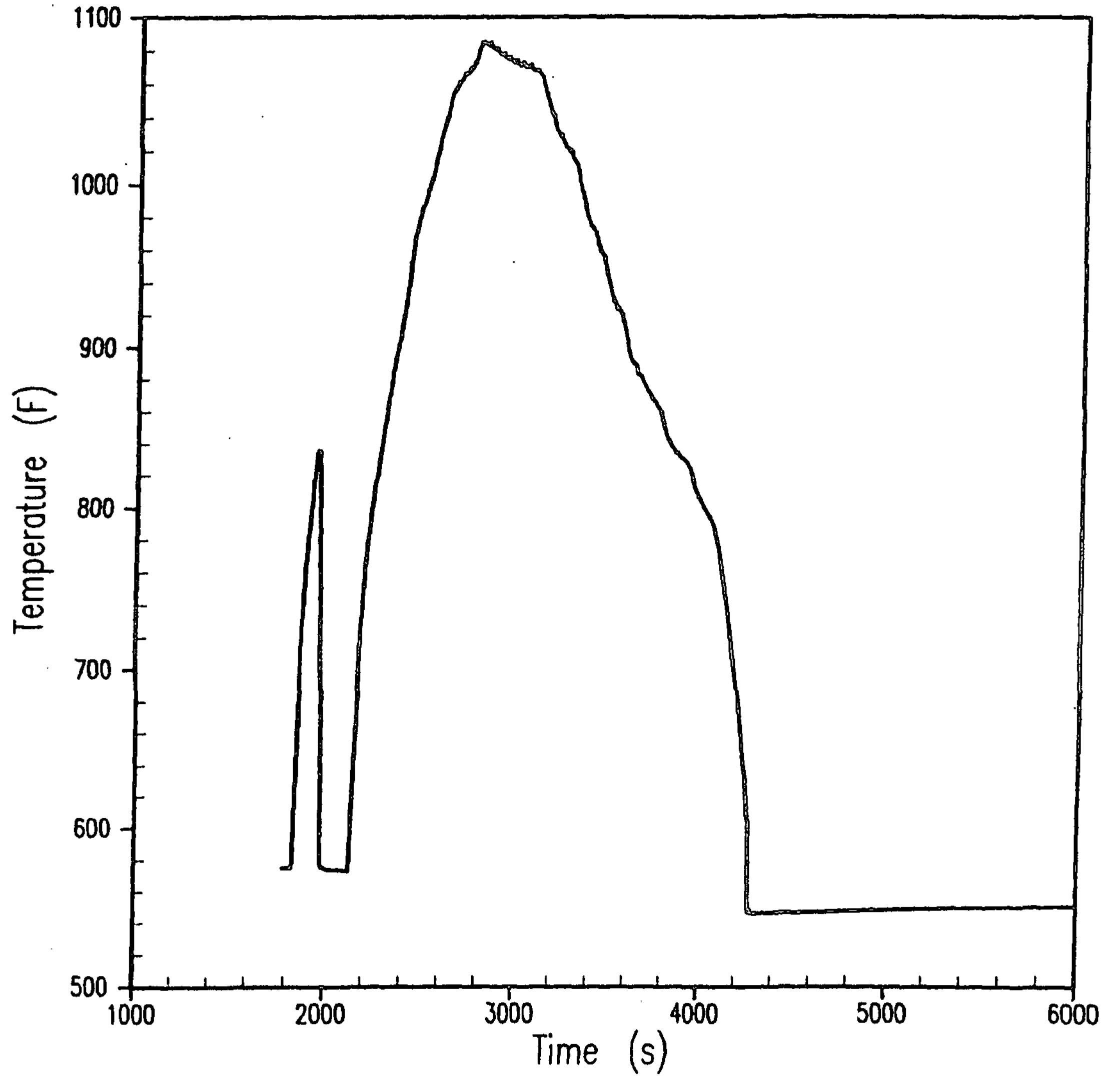
Figure 15.6-19f



Byron/Braidwood Stations Updated Final Safety Analysis Report

Break Flow - Unit 2 (2 Inch, Reduced T_{AVG})

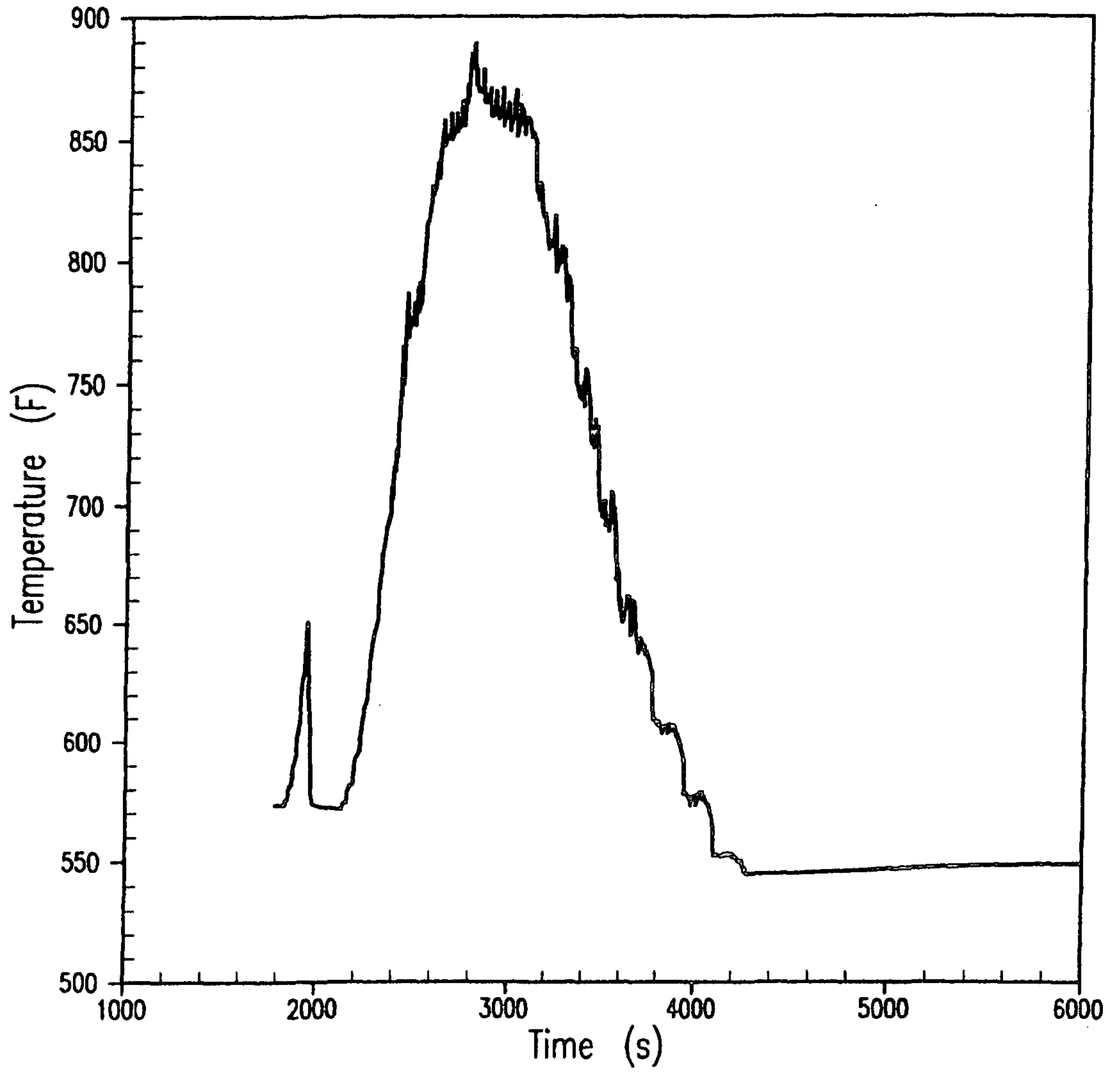
Figure 15.6-19g



Byron/Braidwood Stations Updated Final Safety Analysis Report

Peak Clad Temperature - Unit 2 (2 Inch, Nominal T_{AVG})

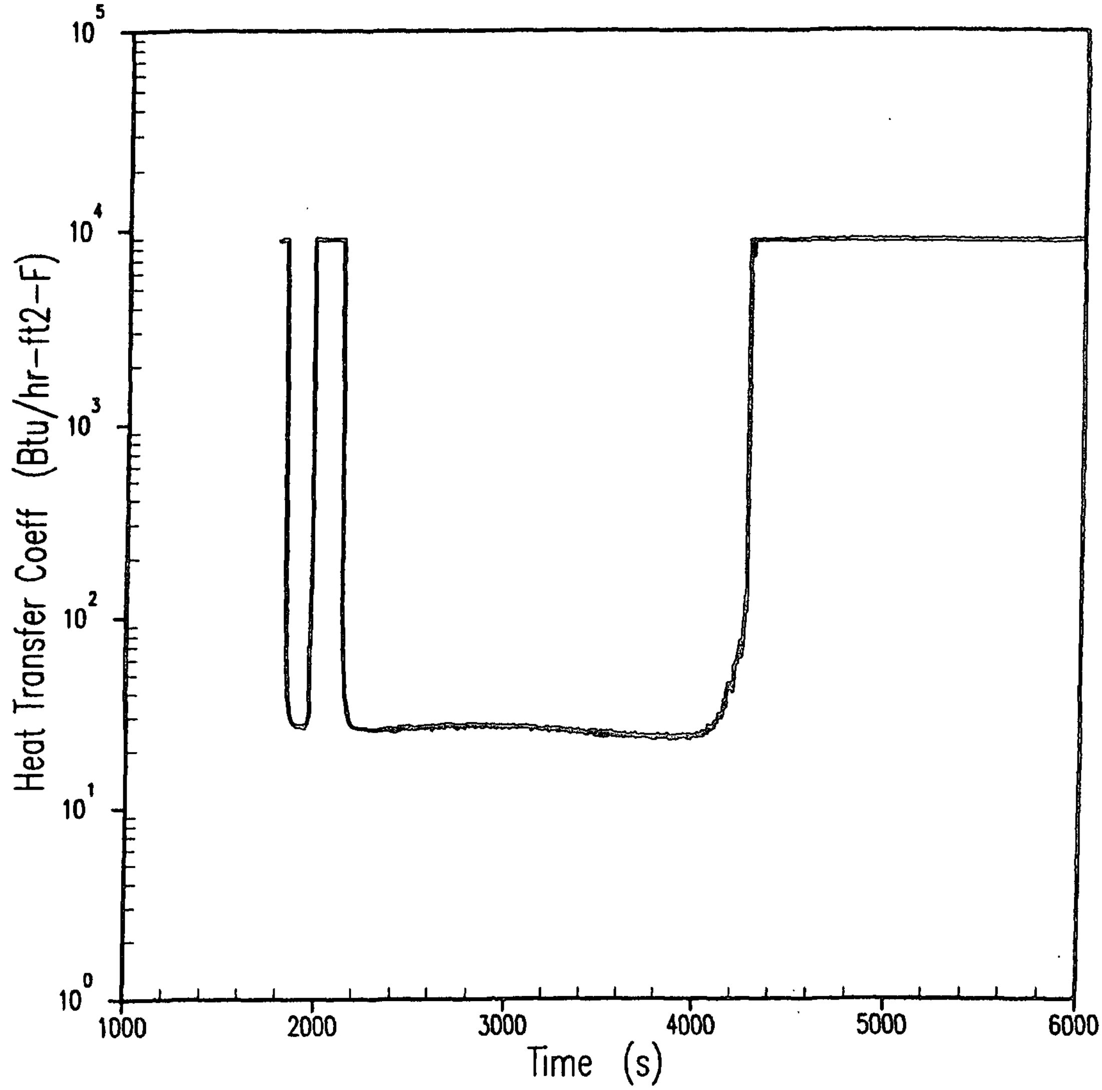
Figure 15.6-19h



Byron/Braidwood Stations Updated Final Safety Analysis Report

Fluid Temperature - Unit 2 (2 Inch, Nominal T_{AVG})

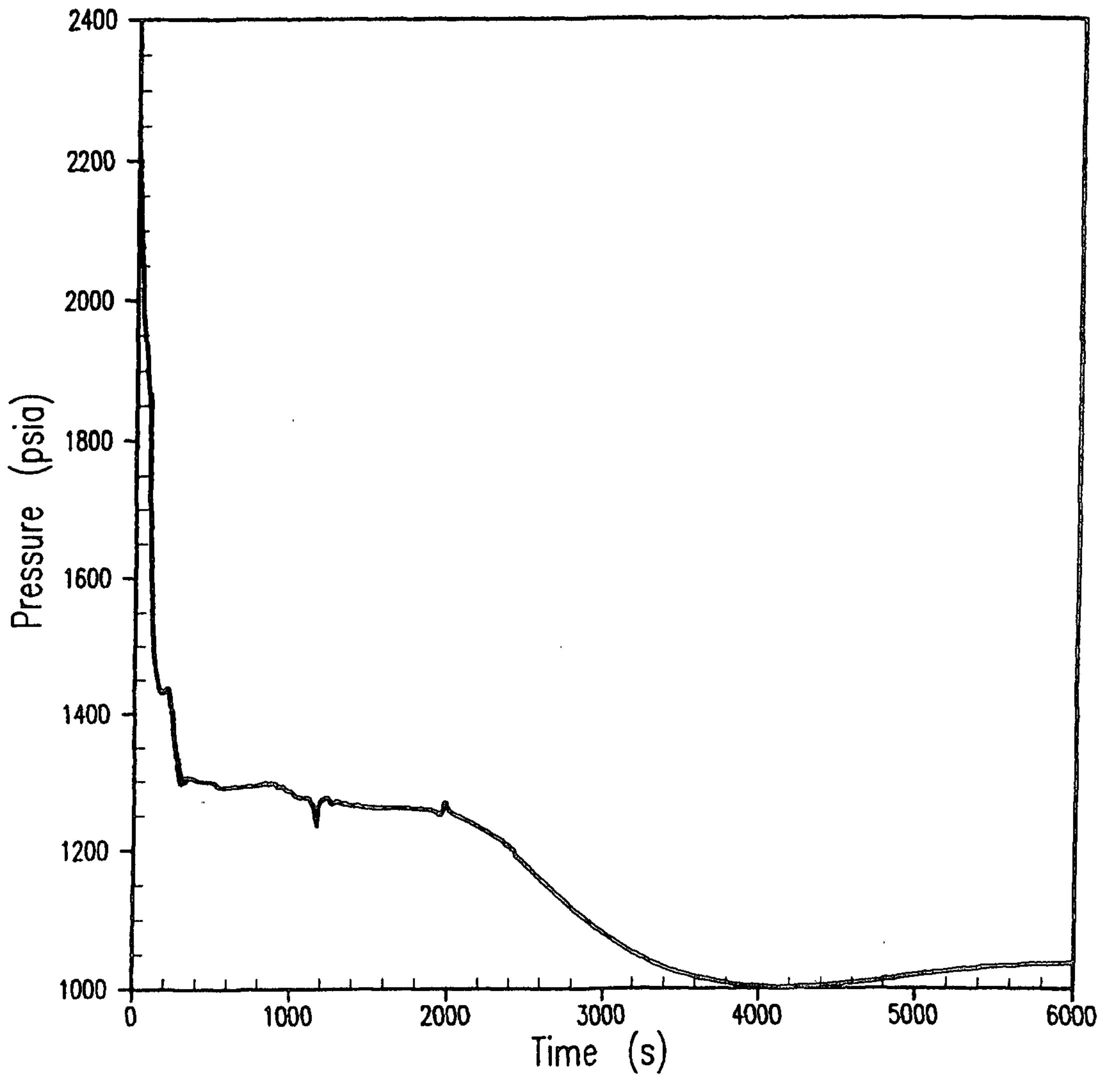
Figure 15.6-19I



Byron/Braidwood Stations
Updated Final Safety Analysis Report

Heat Transfer Coefficient - Unit 2
(2 Inch, Nominal T_{AVG})

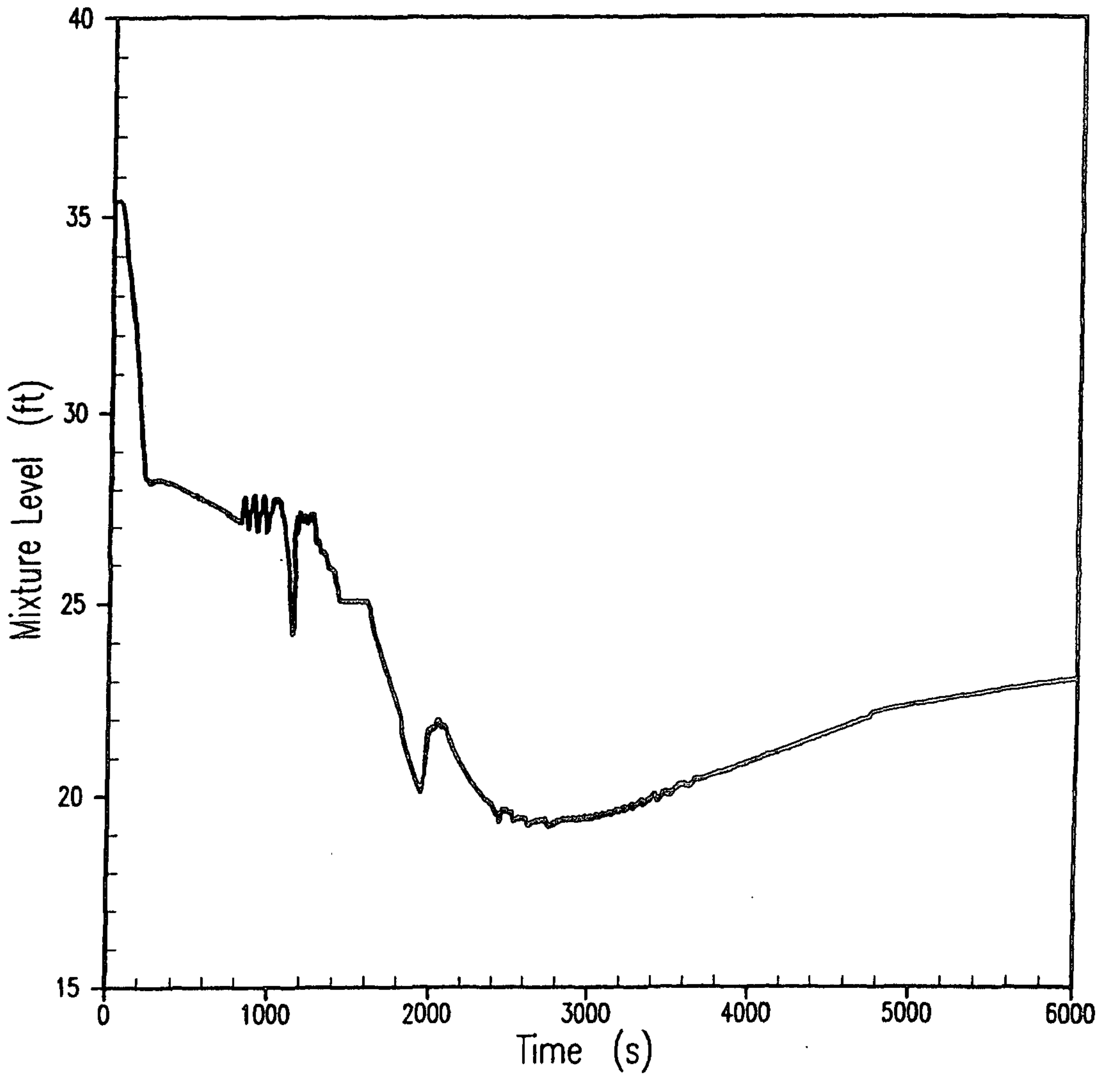
Figure 15.6-19j



Byron/Braidwood Stations
Updated Final Safety Analysis Report

RCS Pressure - Unit 2
(2 Inch, Nominal T_{AVG})

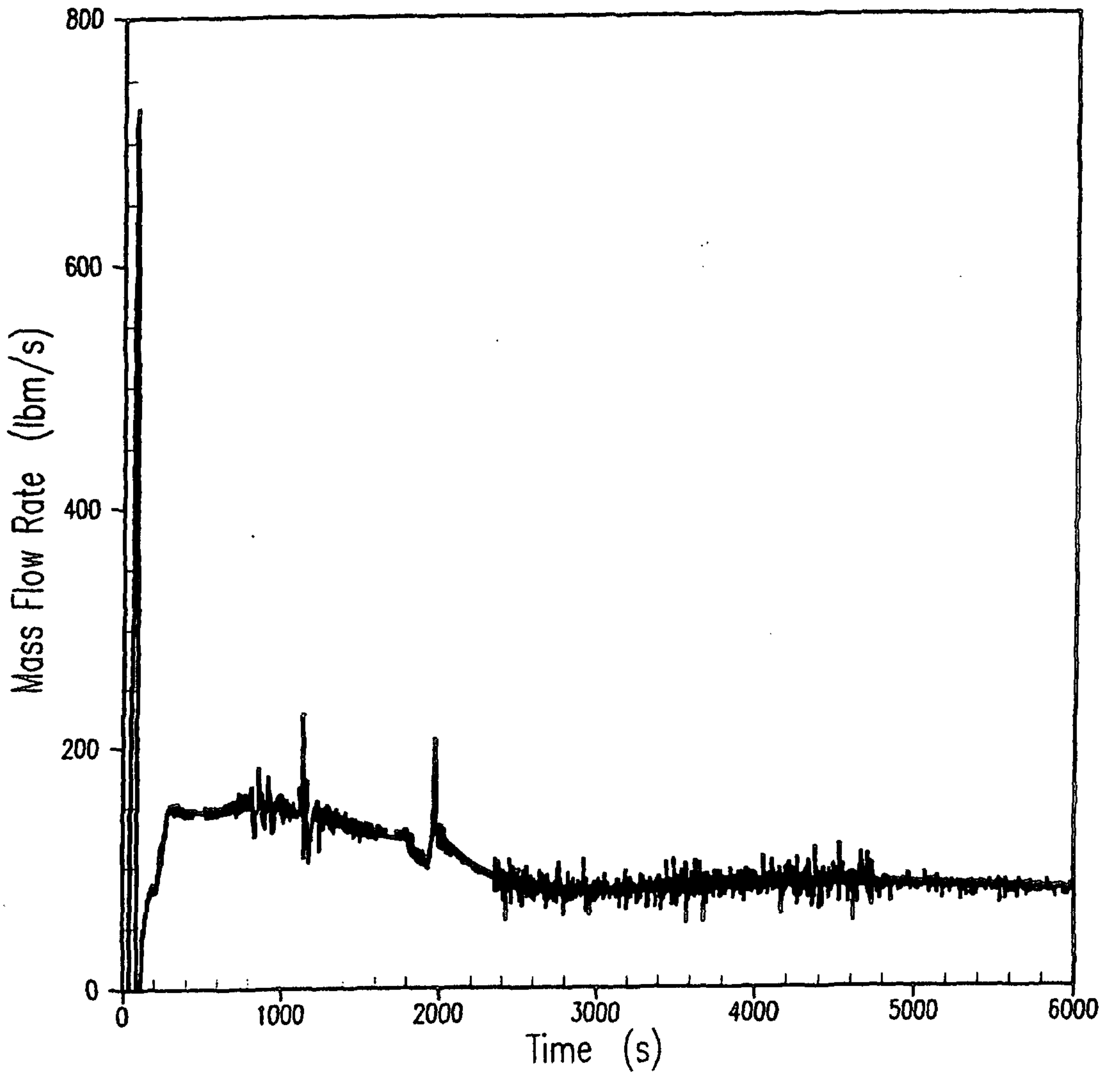
Figure 15.6-19k



Byron/Braidwood Stations Updated Final Safety Analysis Report

Core Mixture Level - Unit 2 (2 Inch, Nominal T_{AVG})

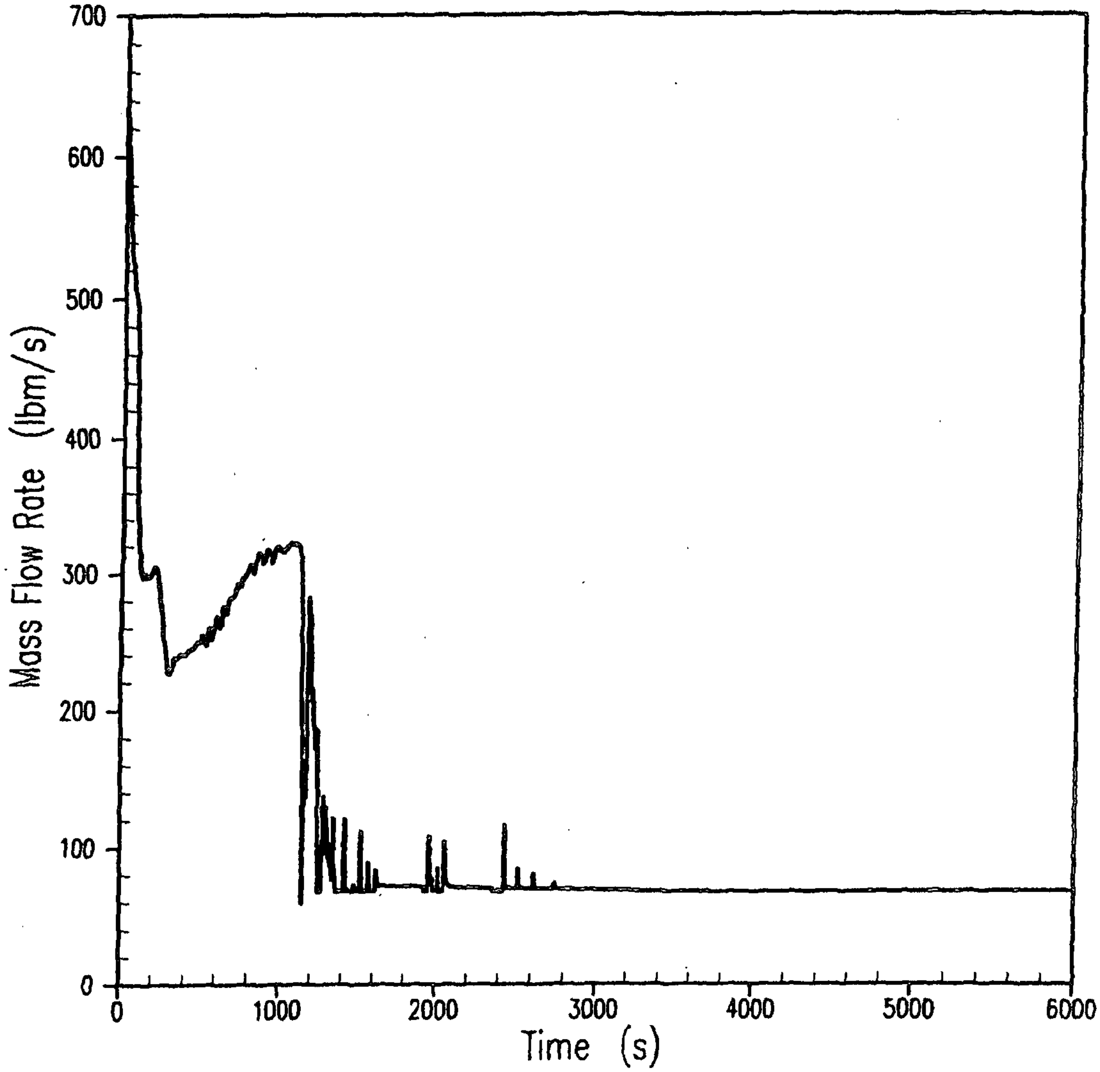
Figure 15.6-191



Byron/Braidwood Stations Updated Final Safety Analysis Report

Core Steam Flow Rate - Unit 2
(2 Inch, Nominal T_{AVG})

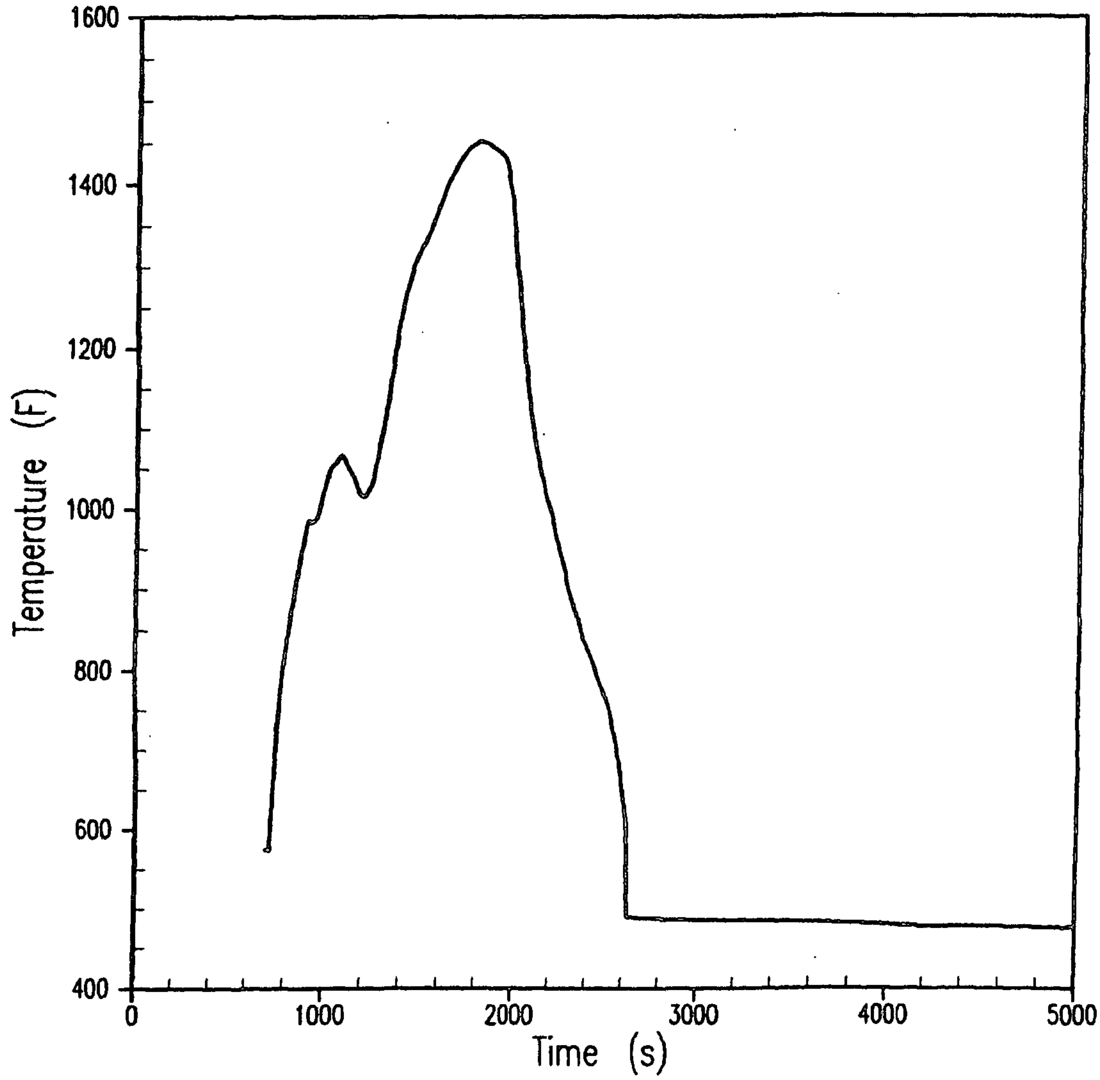
Figure 15.6-19m



Byron/Braidwood Stations Updated Final Safety Analysis Report

Break Flow - Unit 2 (2 Inch, Nominal T_{AVG})

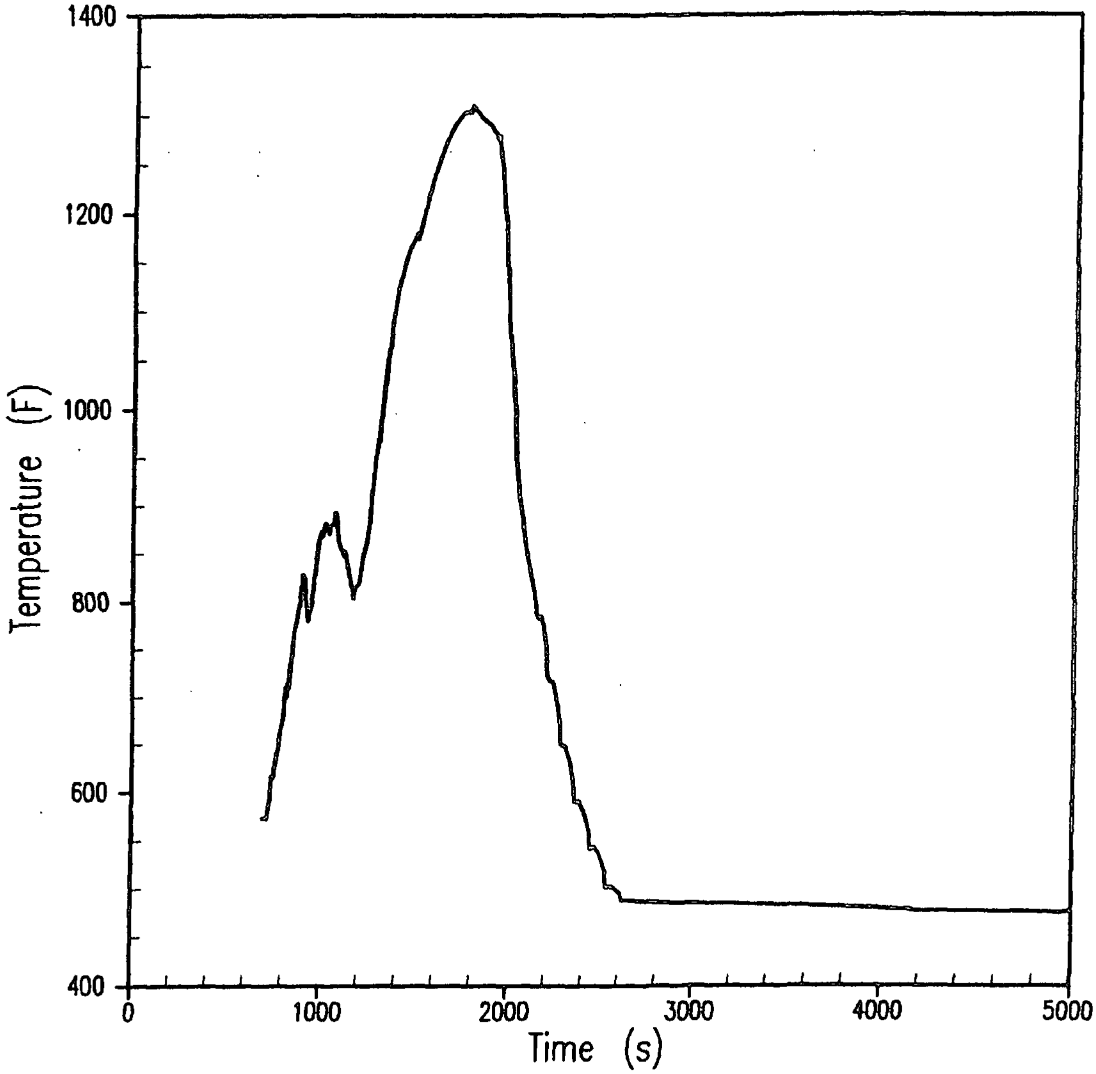
Figure 15.6-19n



Byron/Braidwood Stations Updated Final Safety Analysis Report

Peak Clad Temperature - Unit 2 (3 Inch, Reduced T_{AVG})

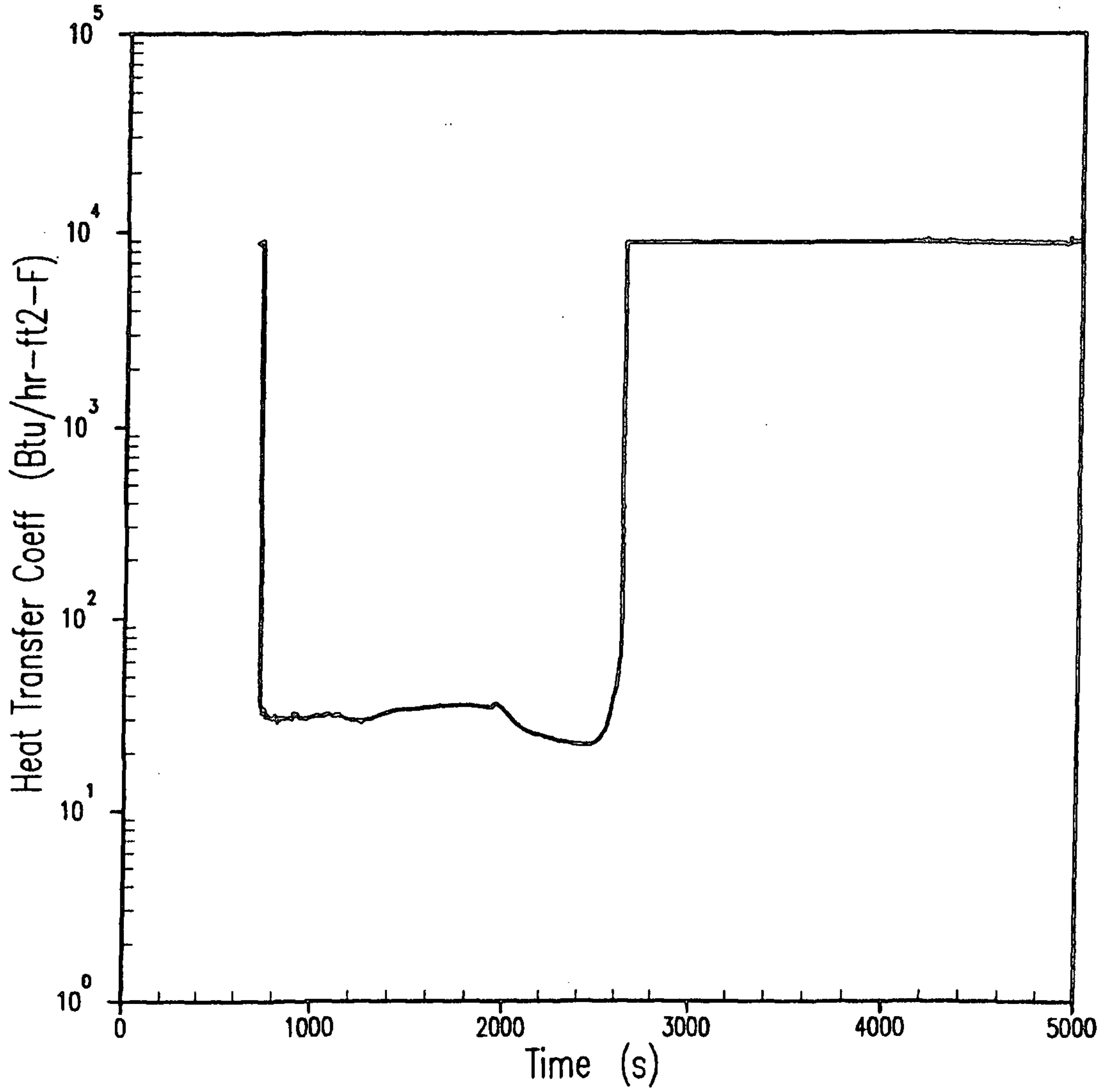
Figure 15.6-20a



Byron/Braidwood Stations Updated Final Safety Analysis Report

Fluid Temperature - Unit 2
(3 Inch, Reduced T_{AVG})

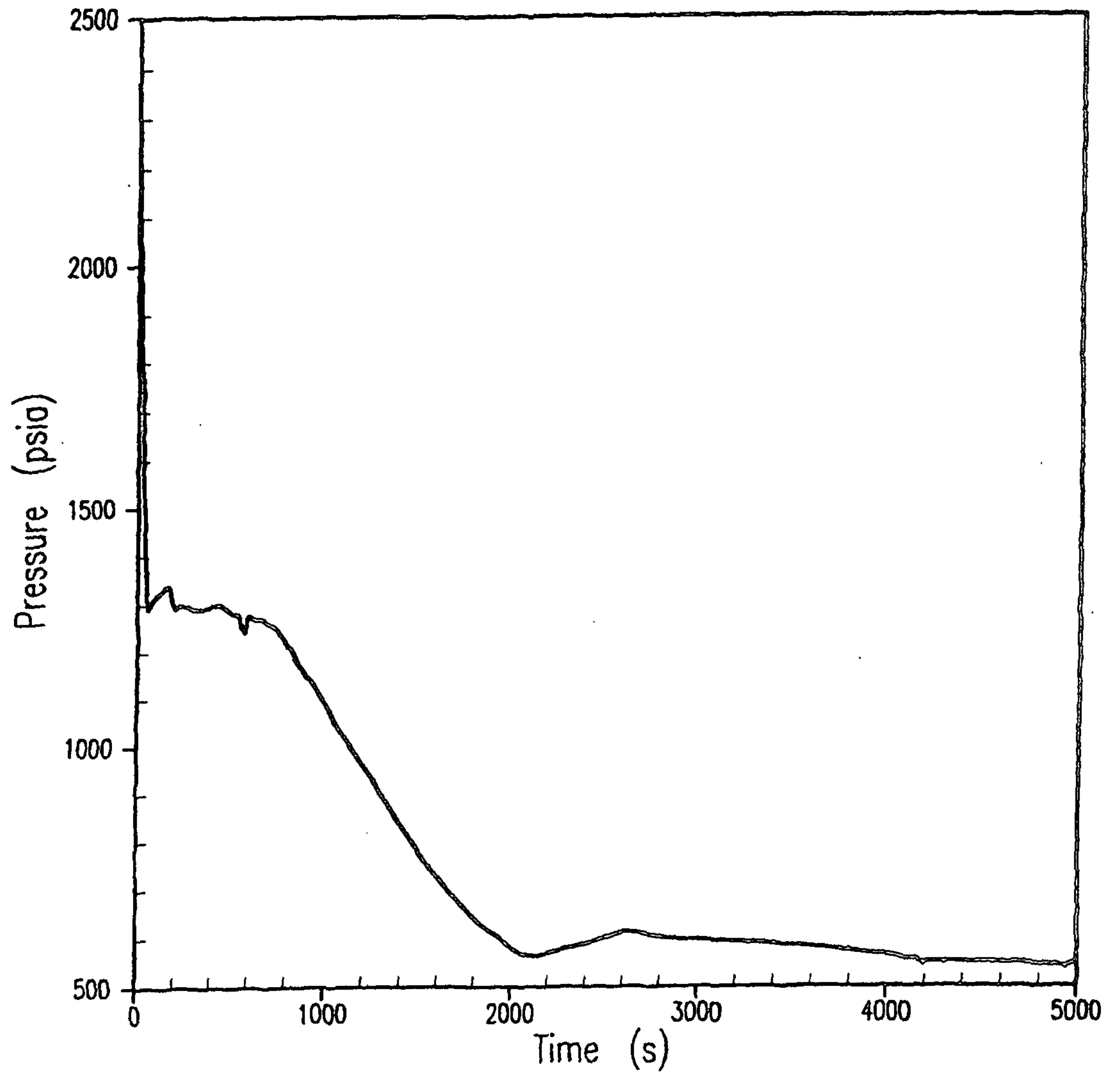
Figure 15.6-20b



Byron/Braidwood Stations Updated Final Safety Analysis Report

Heat Transfer Coefficient - Unit 2 (3 Inch, Reduced T_{AVG})

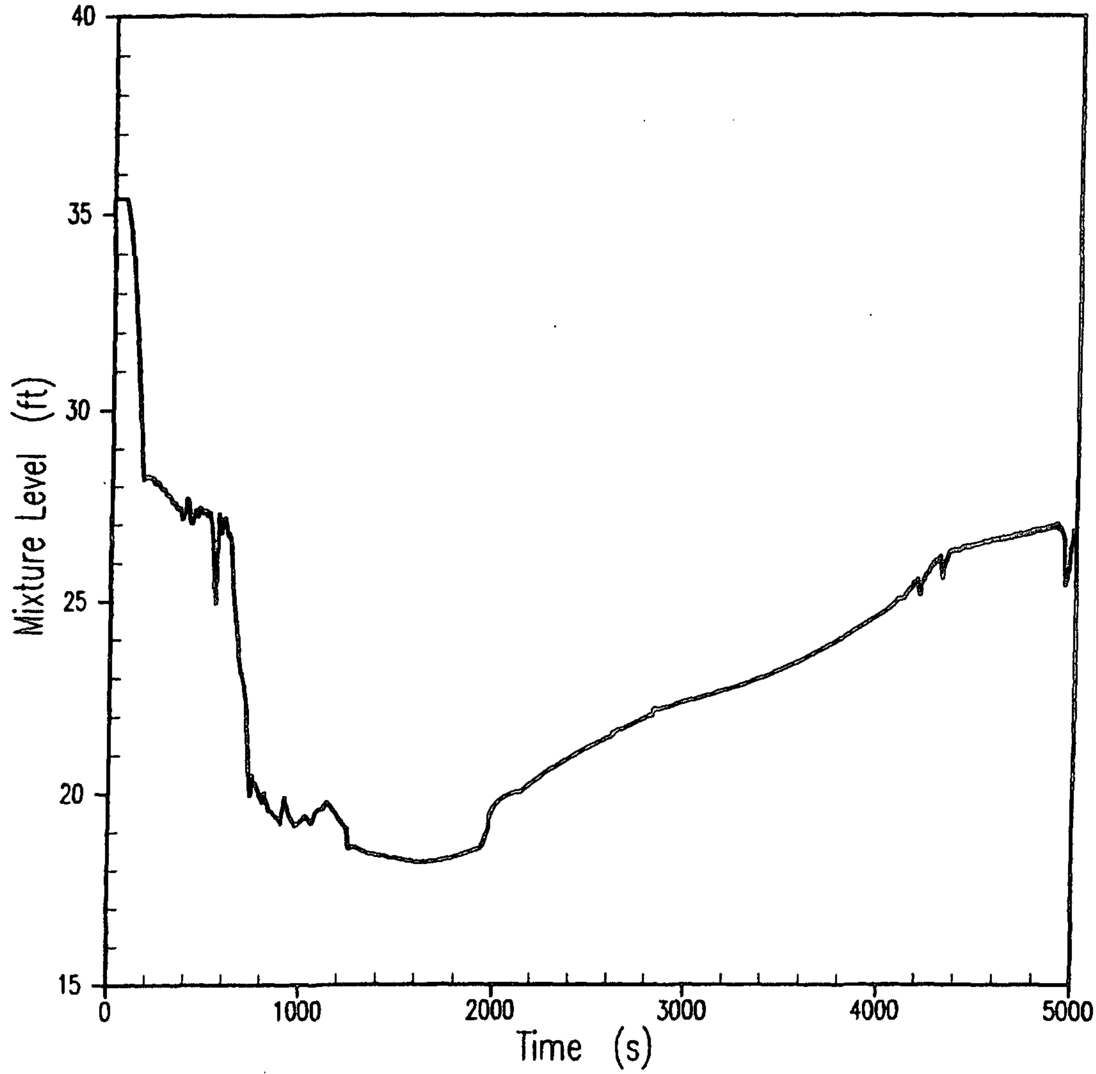
Figure 15.6-20c



Byron/Braidwood Stations Updated Final Safety Analysis Report

RCS Pressure - Unit 2
(3 Inch, Reduced T_{AVG})

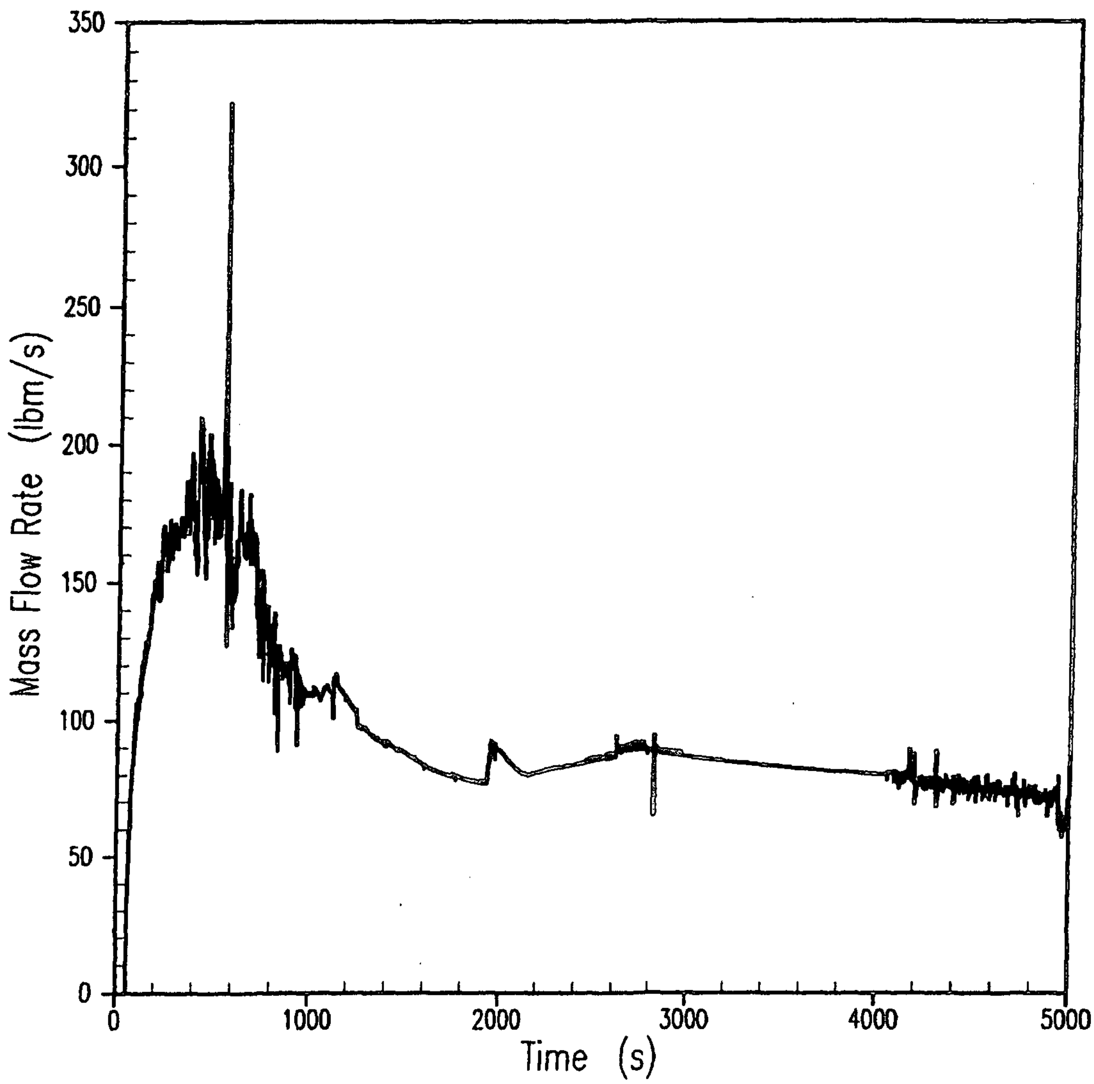
Figure 15.6-20d



Byron/Braidwood Stations Updated Final Safety Analysis Report

Core Mixture Level - Unit 2 (3 Inch, Reduced T_{AVG})

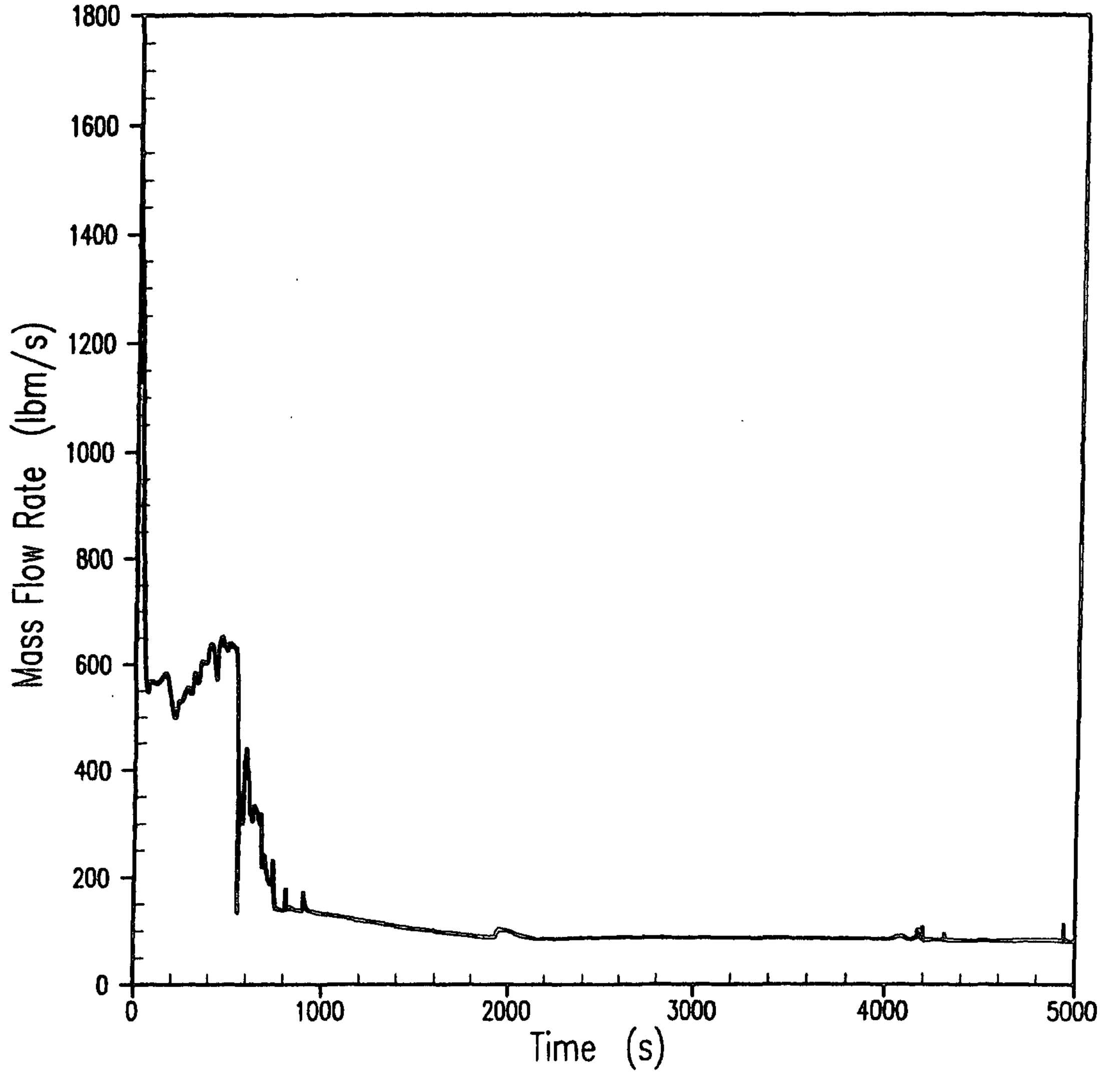
Figure 15.6-20e



**Byron/Braidwood Stations
Updated Final Safety Analysis
Report**

**Core Steam Flow Rate - Unit 2
(3 Inch, Reduced T_{AVG})**

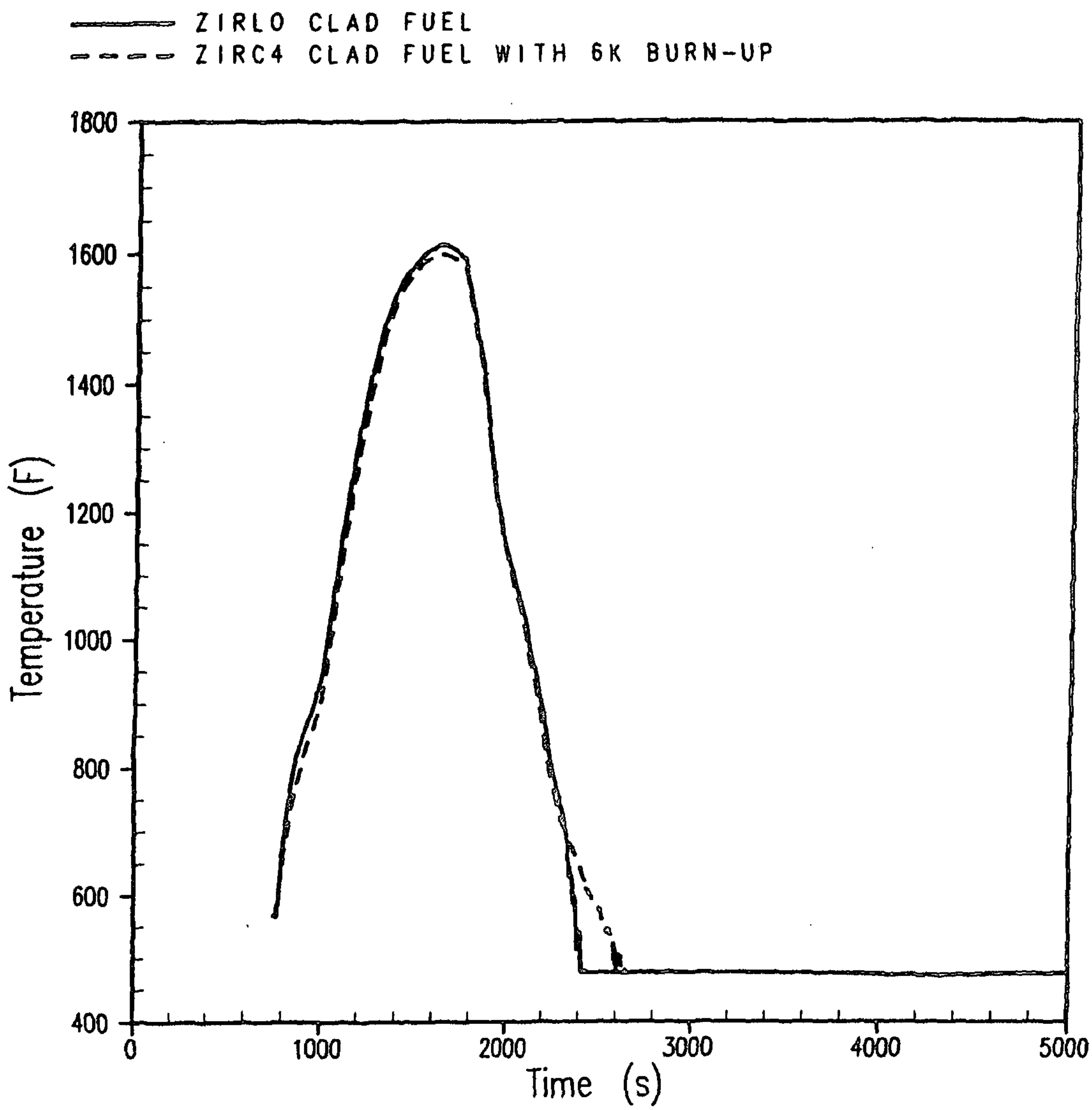
Figure 15.6-20f



Byron/Braidwood Stations Updated Final Safety Analysis Report

Break Flow - Unit 2
(3 Inch, Reduced T_{AVG})

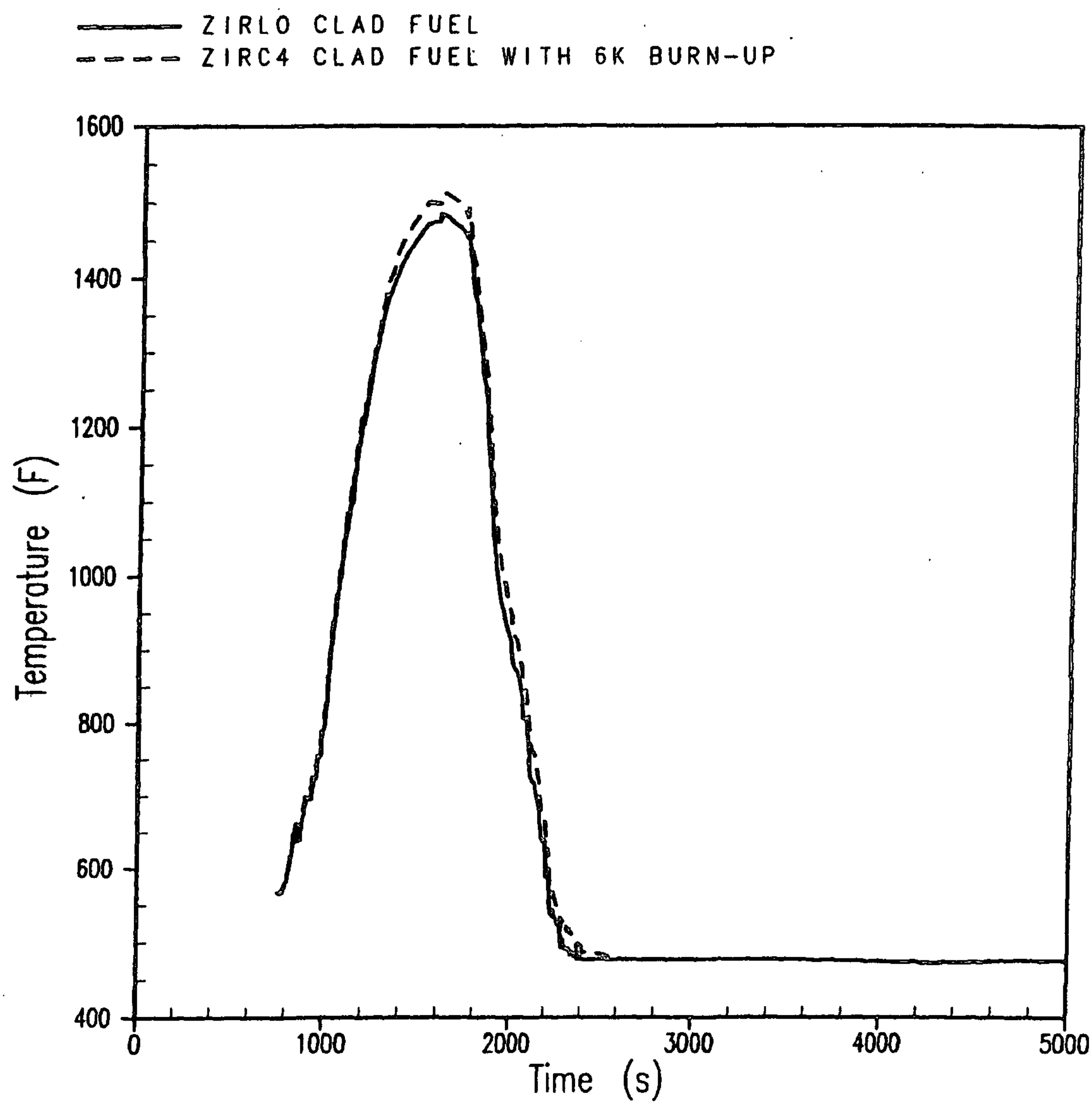
Figure 15.6-20g



Byron/Braidwood Stations Updated Final Safety Analysis Report

Peak Clad Temperature - Unit 2 (3 Inch, Nominal T_{AVG})

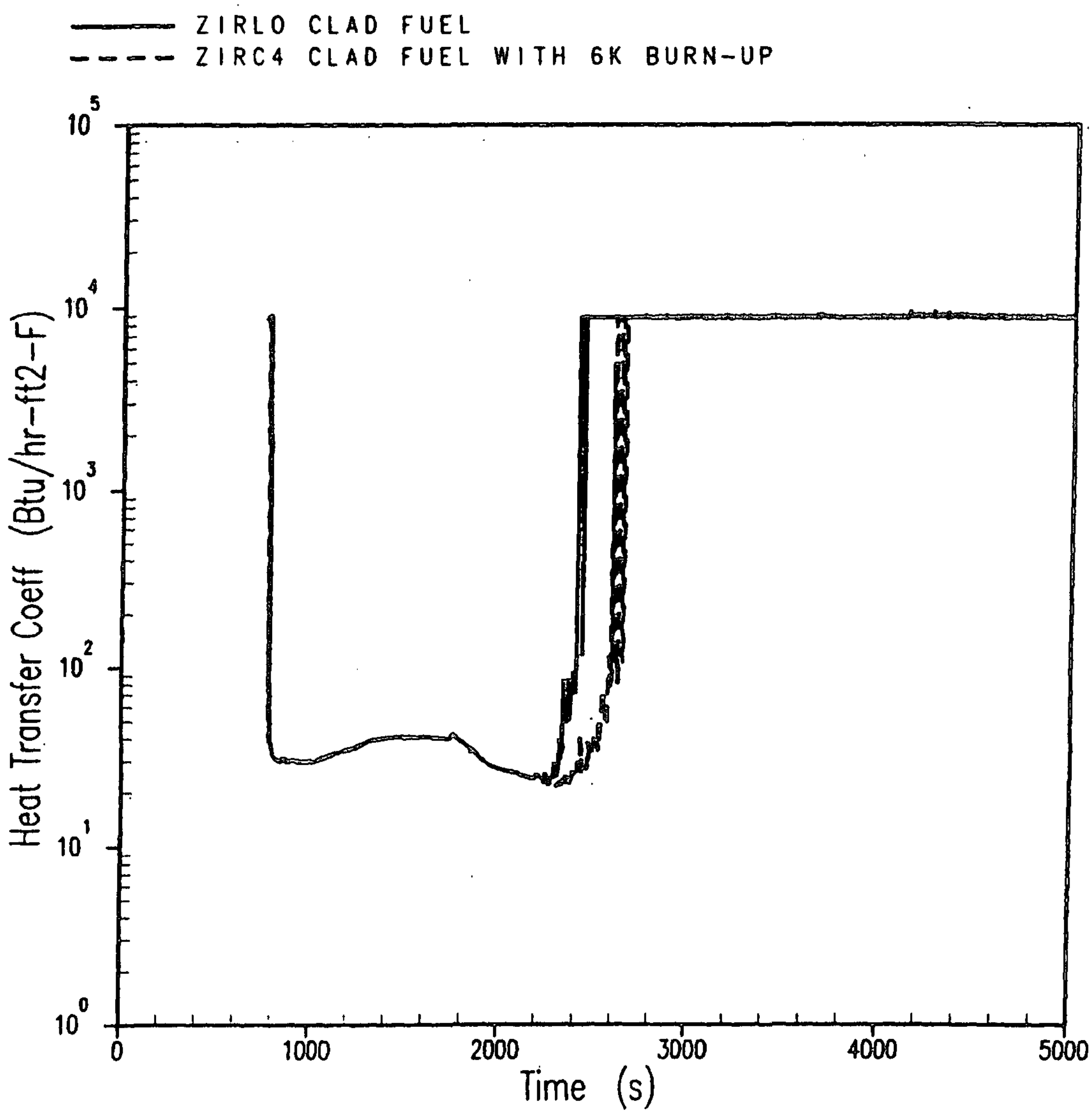
Figure 15.6-20h



Byron/Braidwood Stations Updated Final Safety Analysis Report

Fluid Temperature - Unit 2 (3 Inch, Nominal T_{AVG})

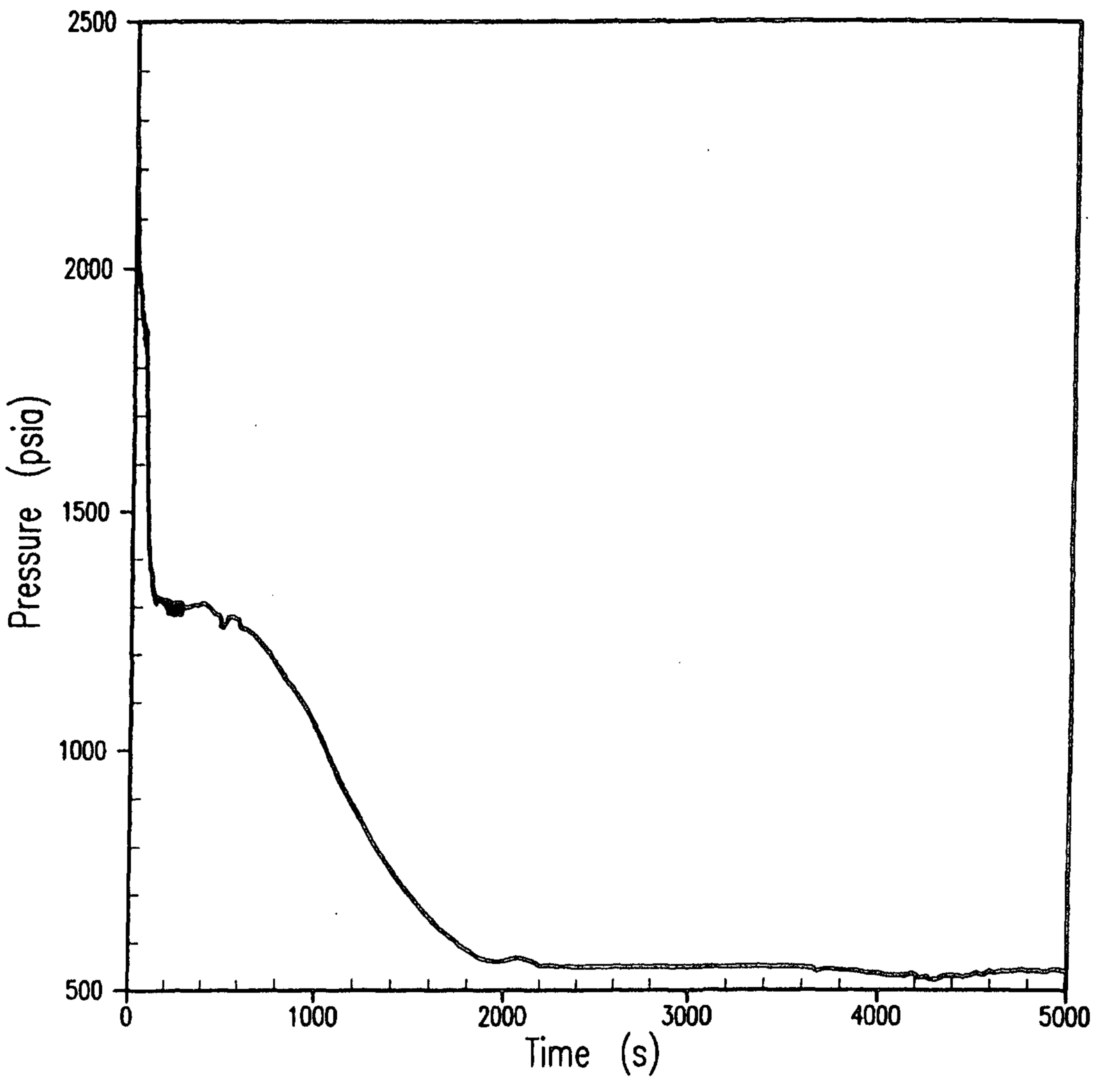
Figure 15.6-201



Byron/Braidwood Stations Updated Final Safety Analysis Report

Heat Transfer Coefficient - Unit 2 (3 Inch, Nominal T_{AVG})

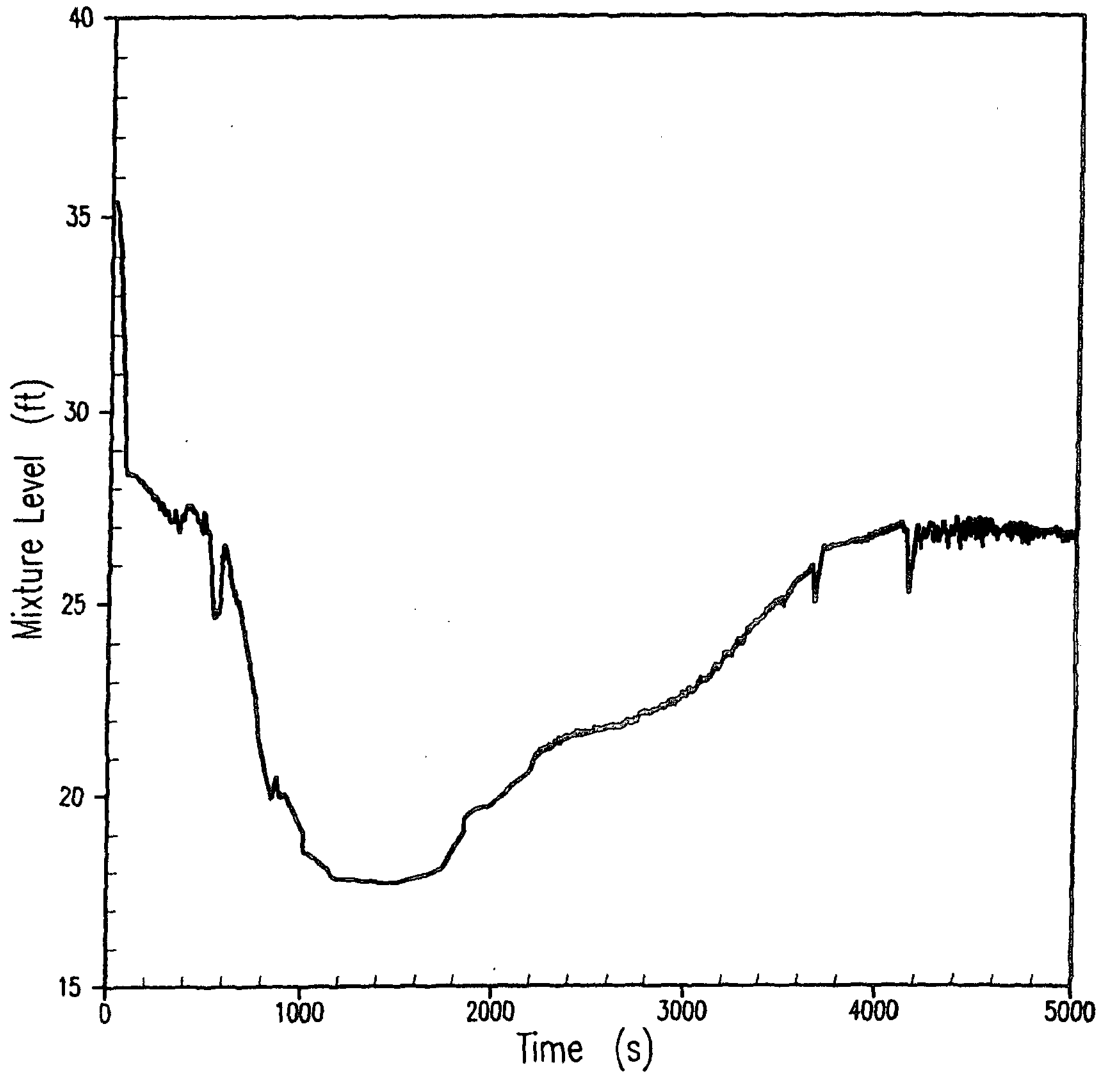
Figure 15.6-20j



Byron/Braidwood Stations Updated Final Safety Analysis Report

RCS Pressure - Unit 2 (3 Inch, Nominal T_{AVG})

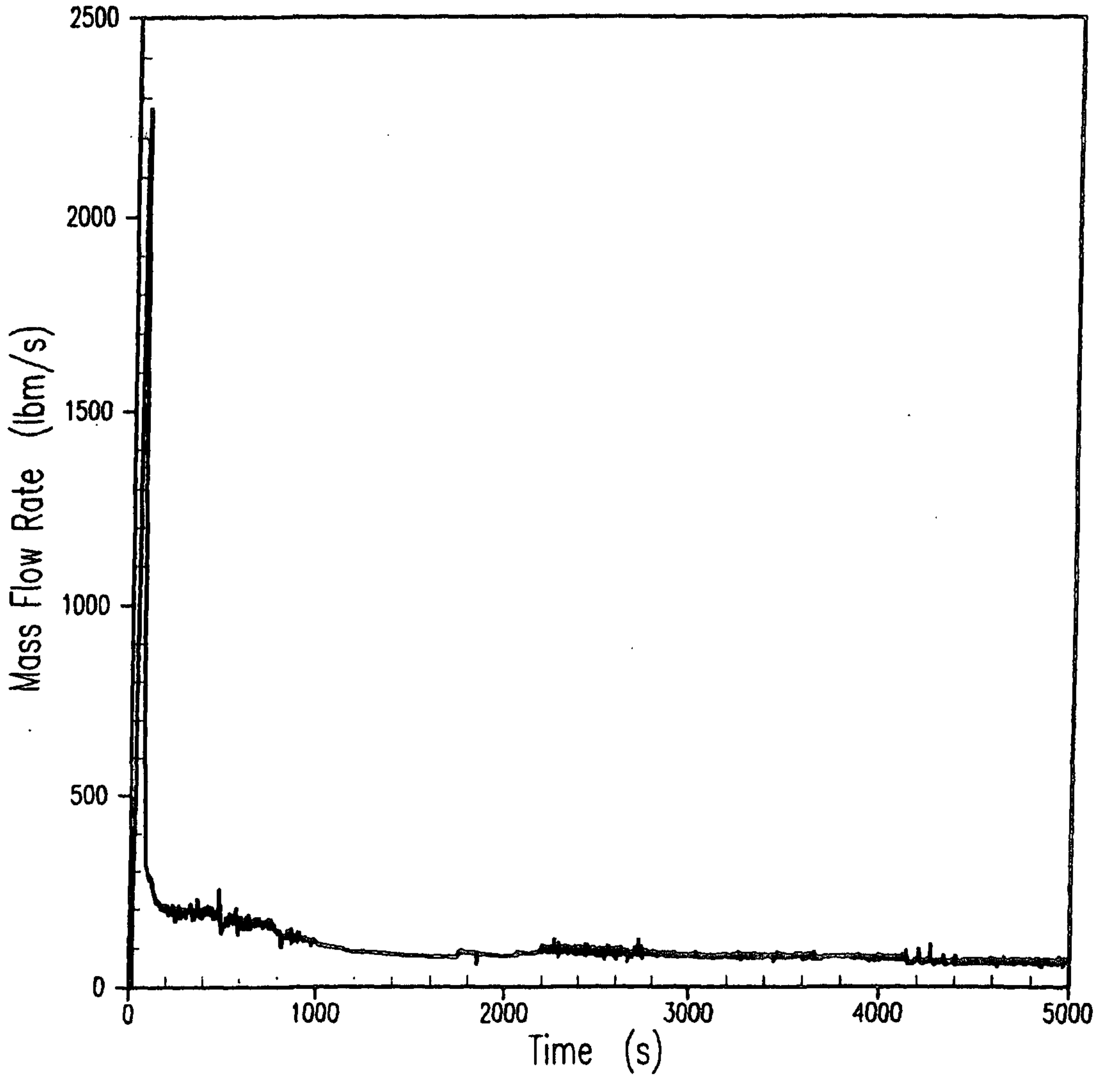
Figure 15.6-20k



Byron/Braidwood Stations Updated Final Safety Analysis Report

Core Mixture Level - Unit 2
(3 Inch, Nominal T_{AVG})

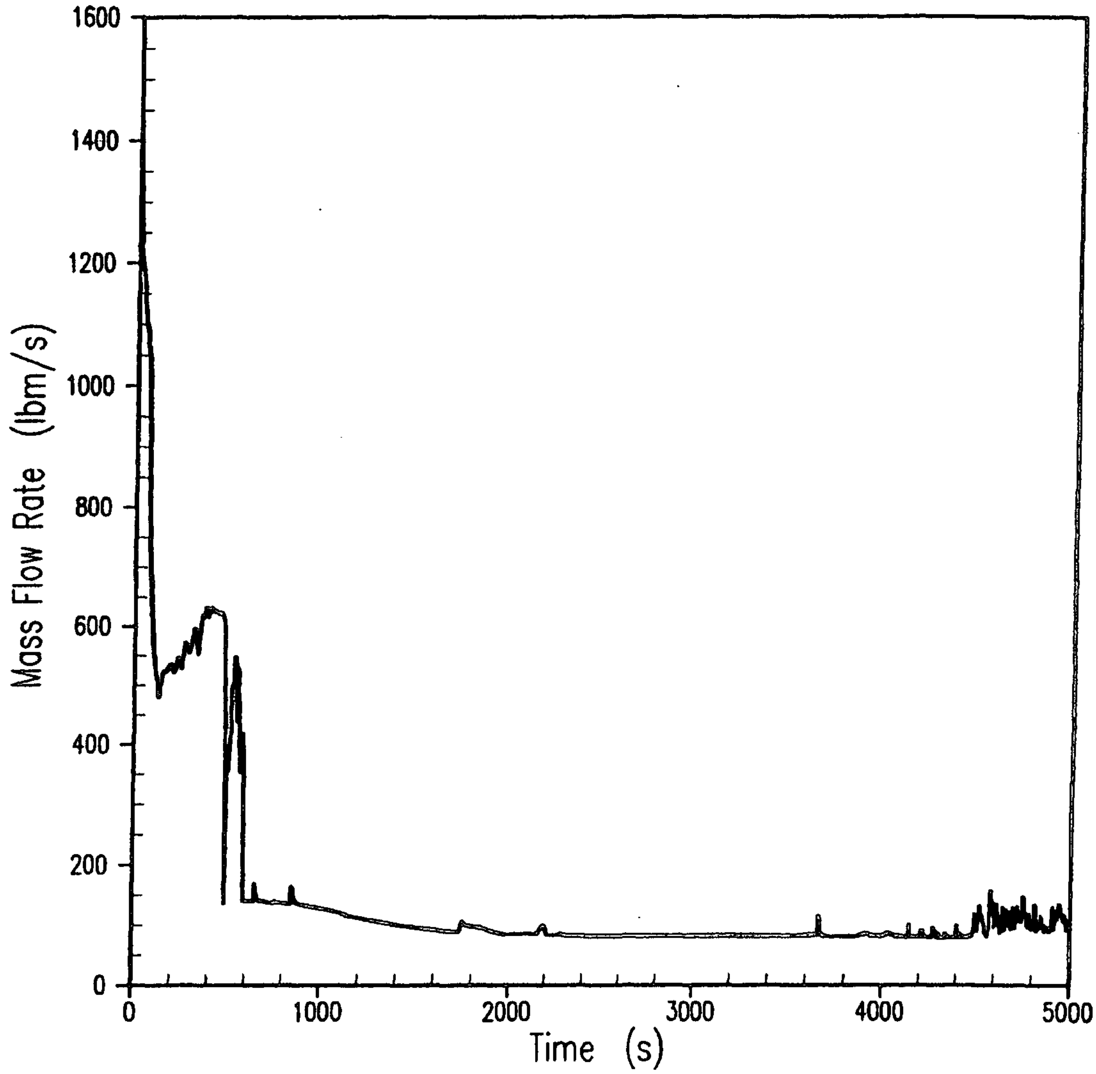
Figure 15.6-201



Byron/Braidwood Stations Updated Final Safety Analysis Report

Core Steam Flow Rate - Unit 2 (3 Inch, Nominal T_{AVG})

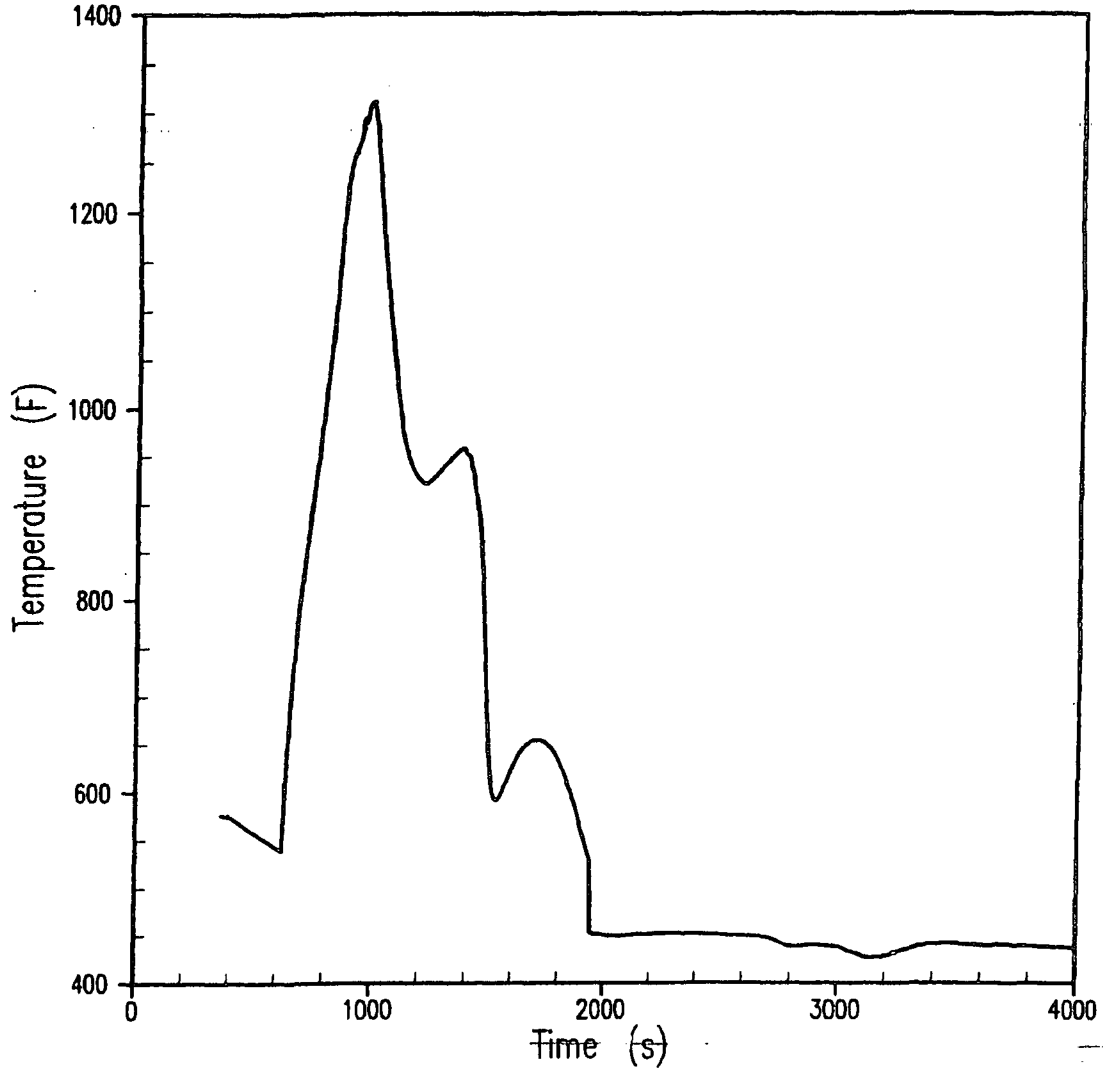
Figure 15.6-20m



Byron/Braidwood Stations Updated Final Safety Analysis Report

Break Flow - Unit 2
(3 Inch, Nominal T_{AVG})

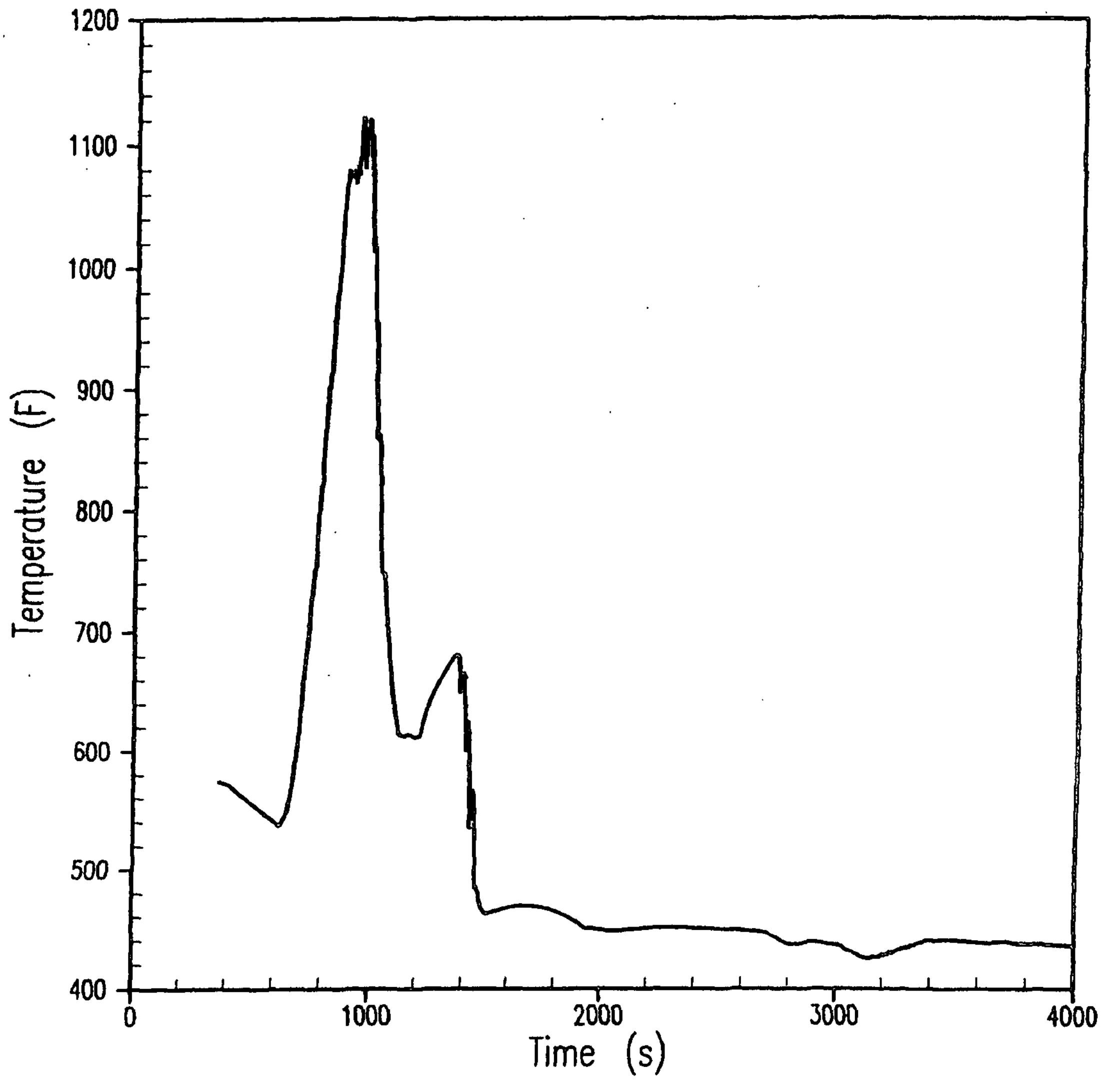
Figure 15.6-20n



**Byron/Braidwood Stations
Updated Final Safety Analysis
Report**

**Peak Clad Temperature - Unit 2
(4 Inch, Reduced T_{AVG})**

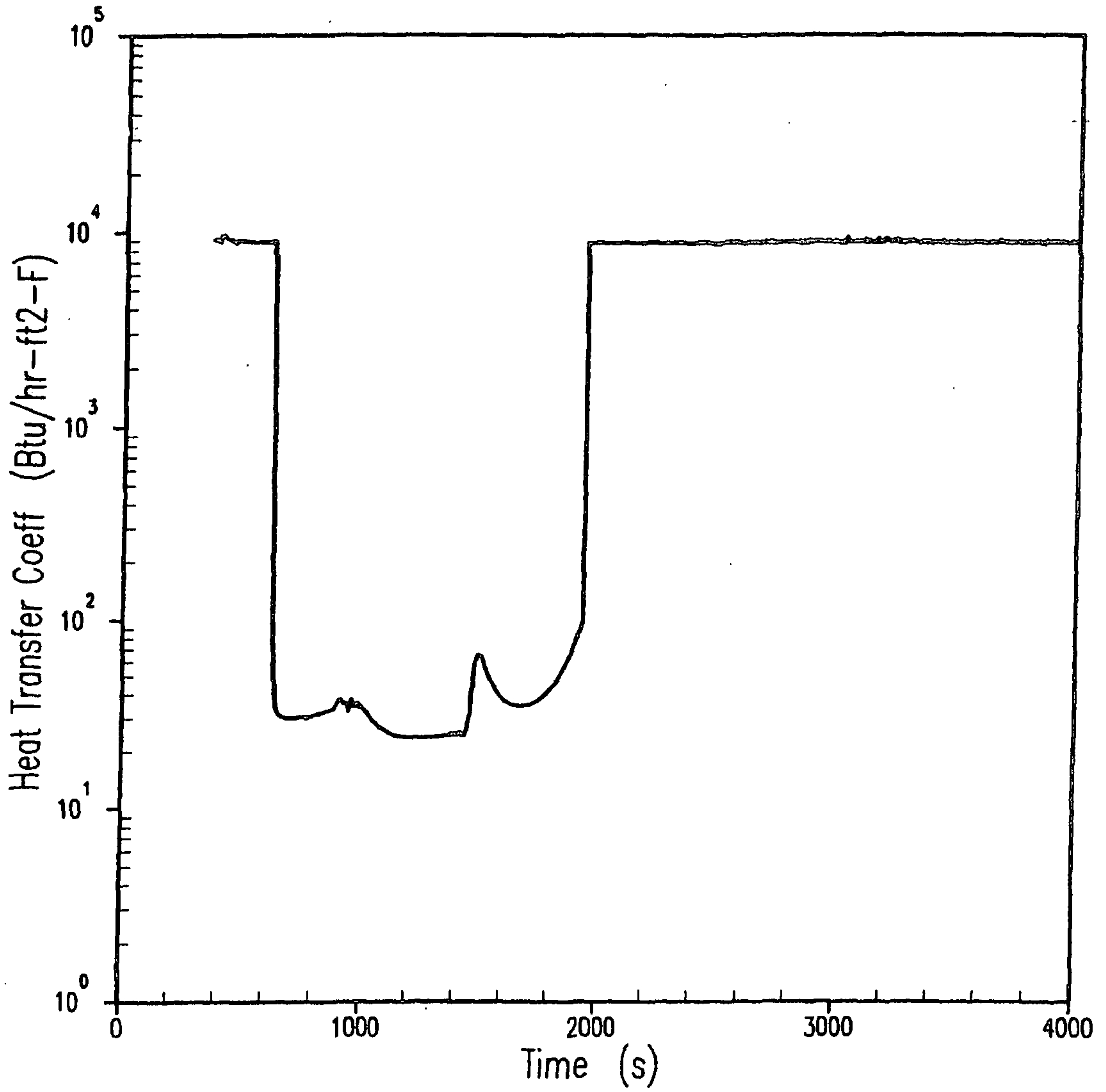
Figure 15.6-21a



Byron/Braidwood Stations Updated Final Safety Analysis Report

Fluid Temperature - Unit 2
(4 Inch, Reduced T_{AVG})

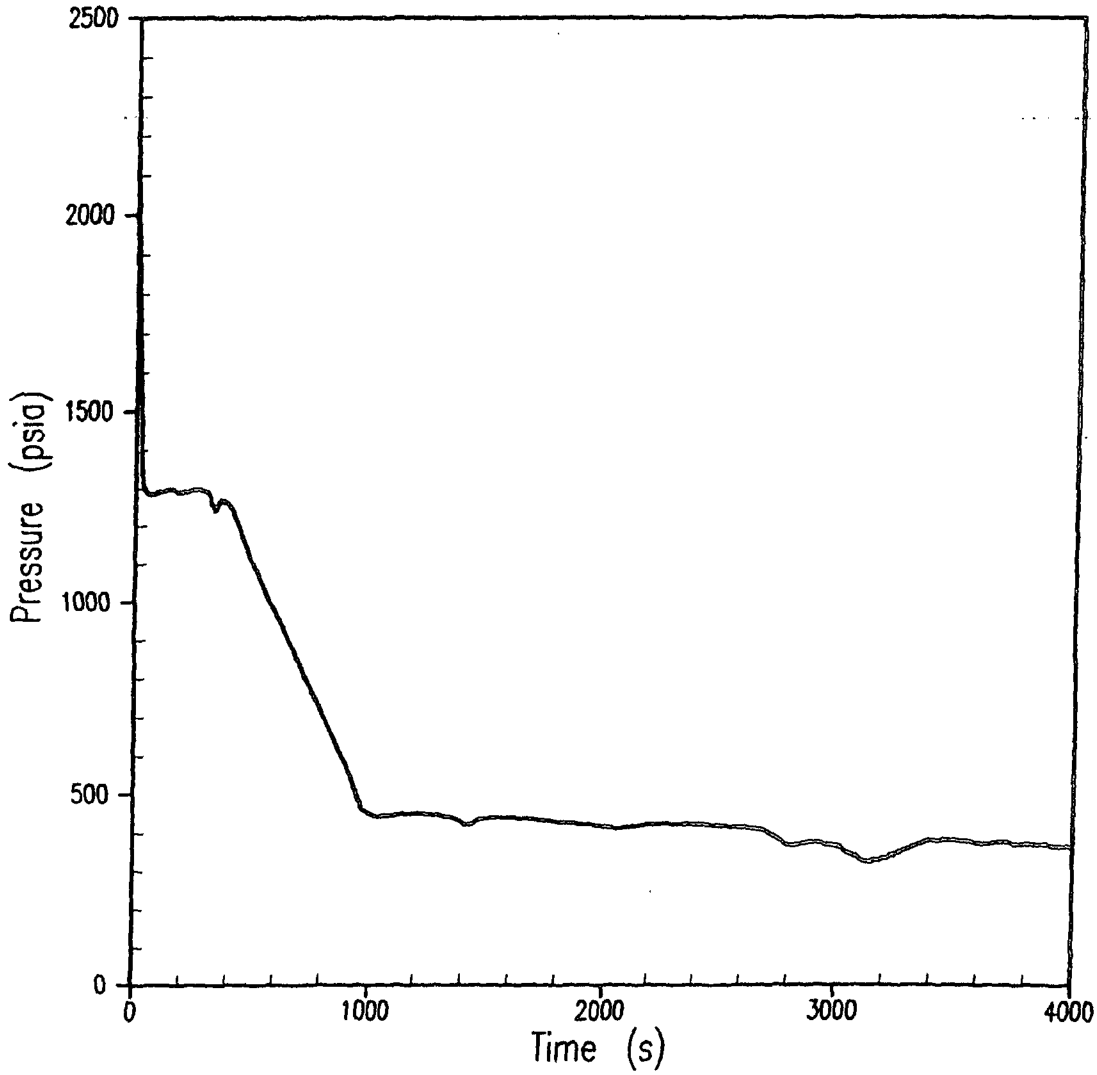
Figure 15.6-21b



Byron/Braidwood Stations Updated Final Safety Analysis Report

Heat Transfer Coefficient - Unit 2 (4 Inch, Reduced T_{AVG})

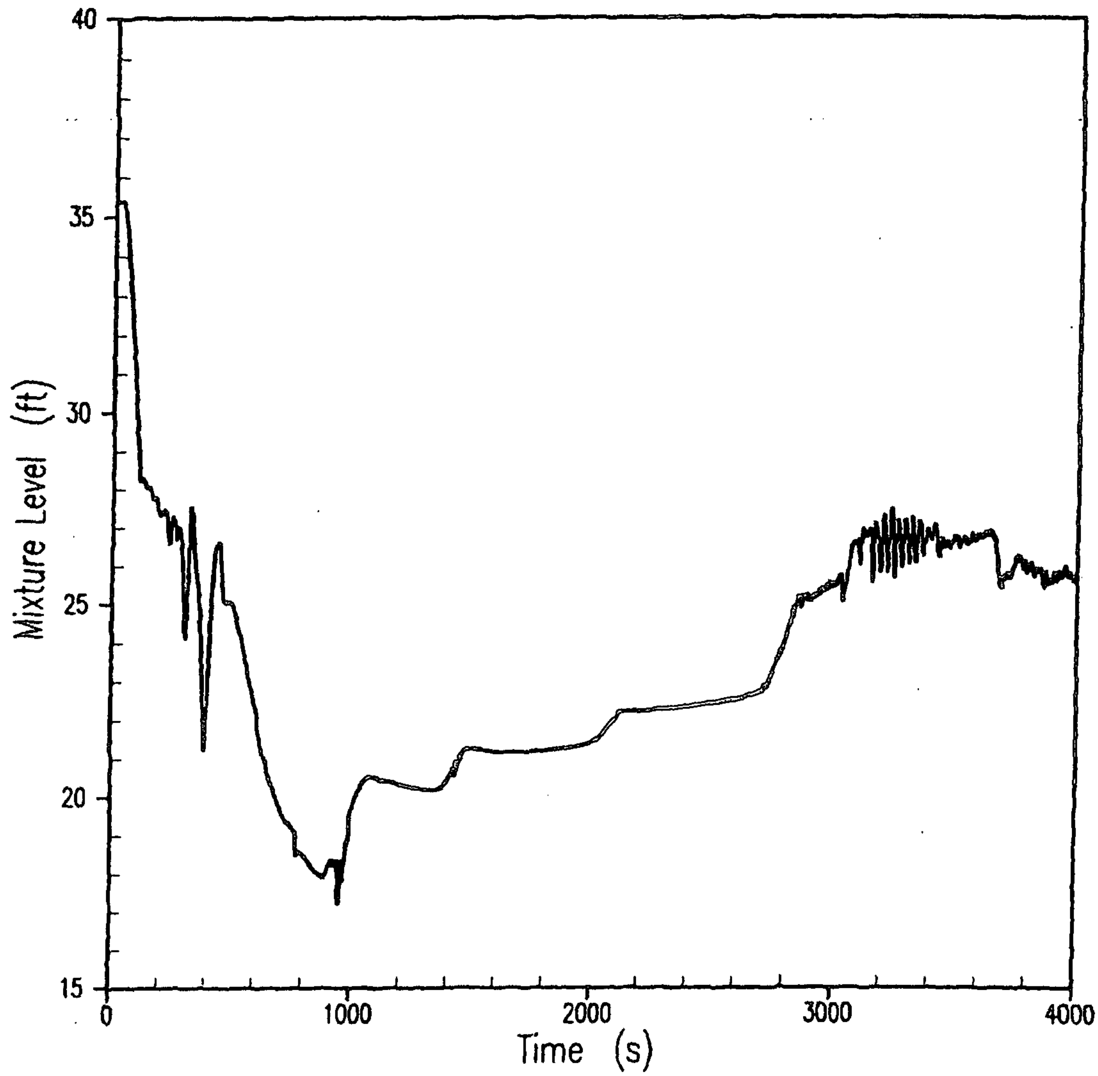
Figure 15.6-21c



Byron/Braidwood Stations Updated Final Safety Analysis Report

RCS Pressure - Unit 2
(4 Inch, Reduced T_{AVG})

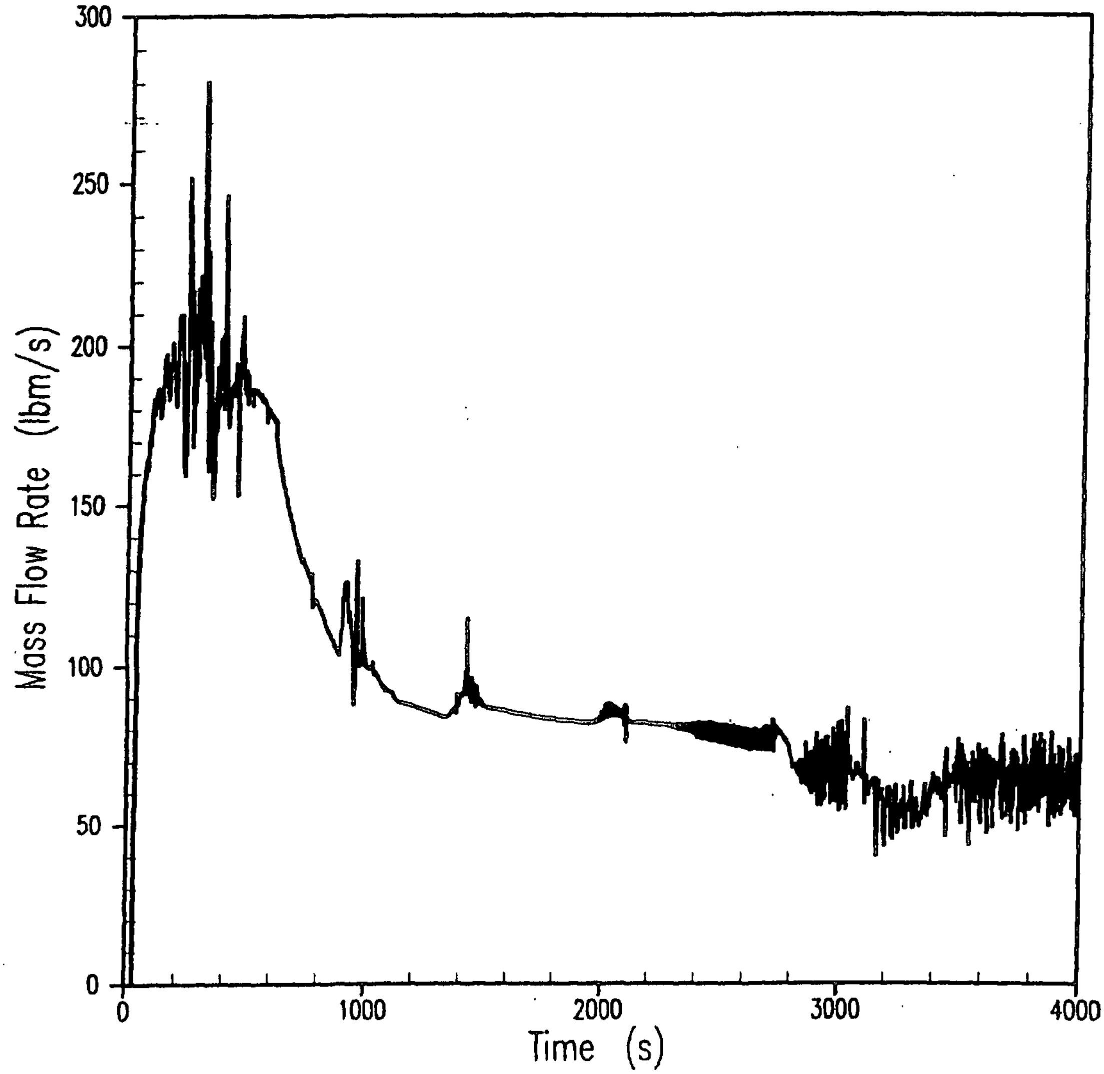
Figure 15.6-21d



Byron/Braidwood Stations Updated Final Safety Analysis Report

Core Mixture Level - Unit 2
(4 Inch, Reduced T_{AVG})

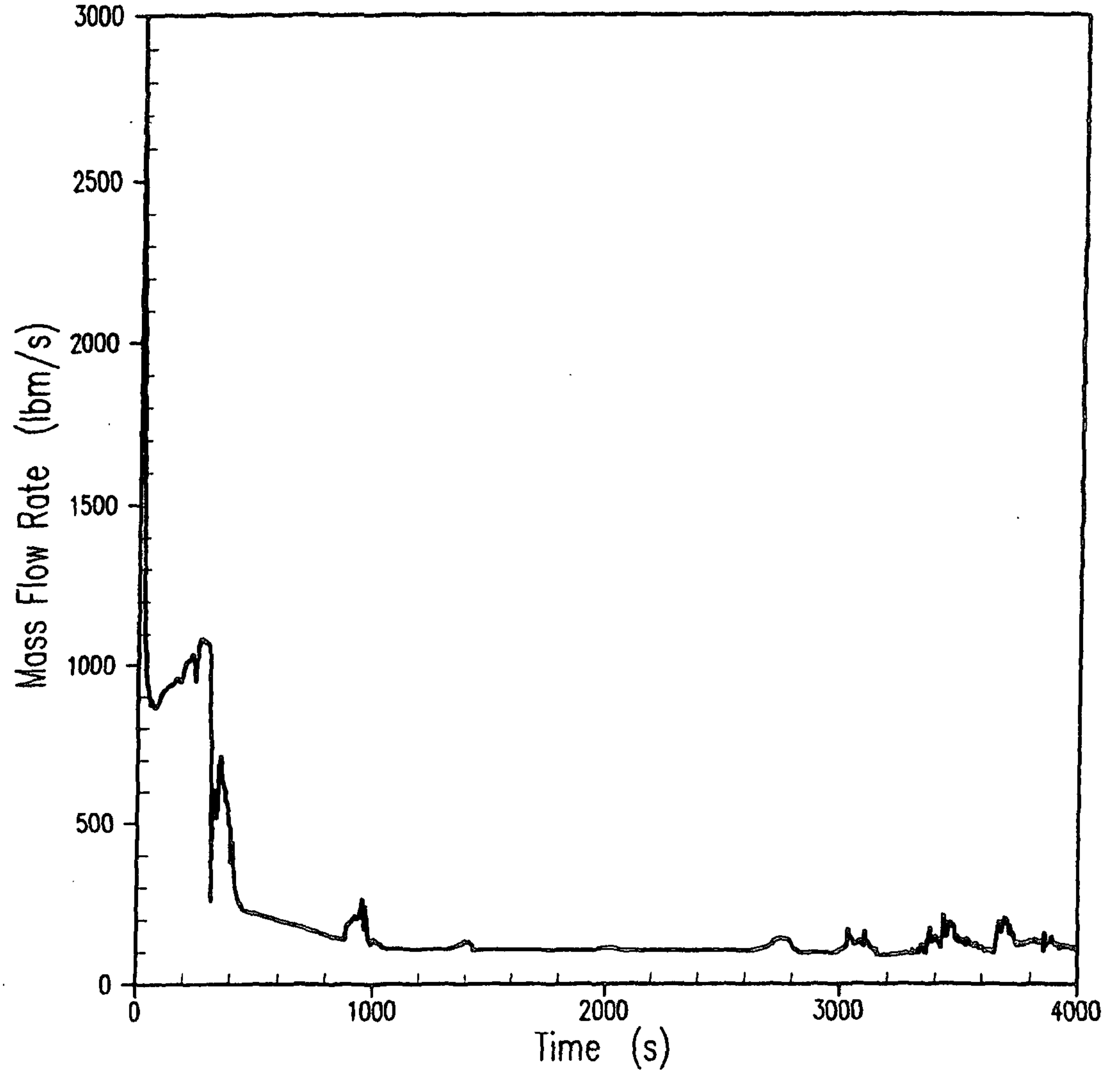
Figure 15.6-21e



Byron/Braidwood Stations Updated Final Safety Analysis Report

Core Steam Flow Rate - Unit 2
(4 Inch, Reduced T_{AVG})

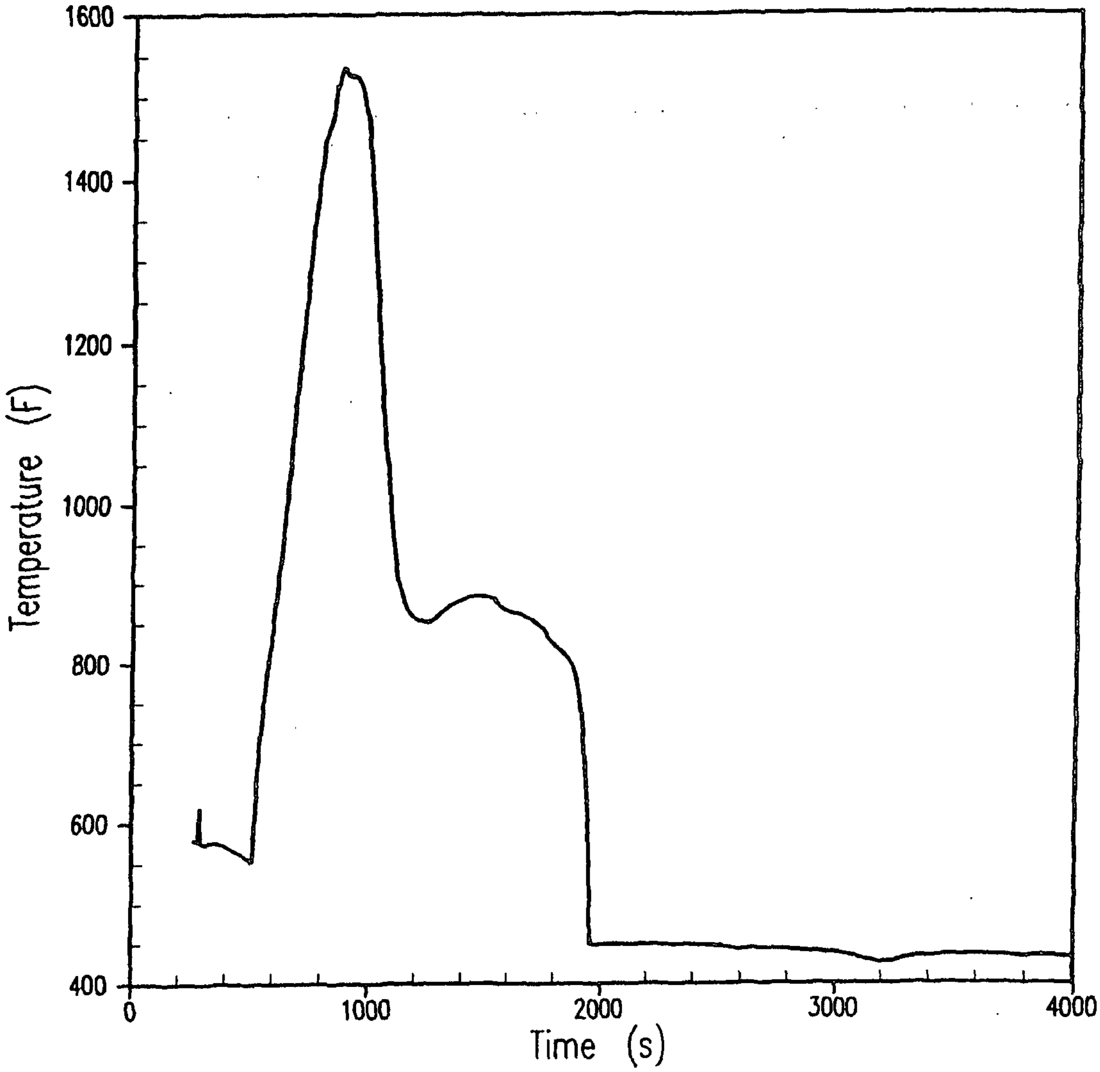
Figure 15.6-21f



Byron/Braidwood Stations Updated Final Safety Analysis Report

Break Flow - Unit 2 (4 Inch, Reduced T_{AVG})

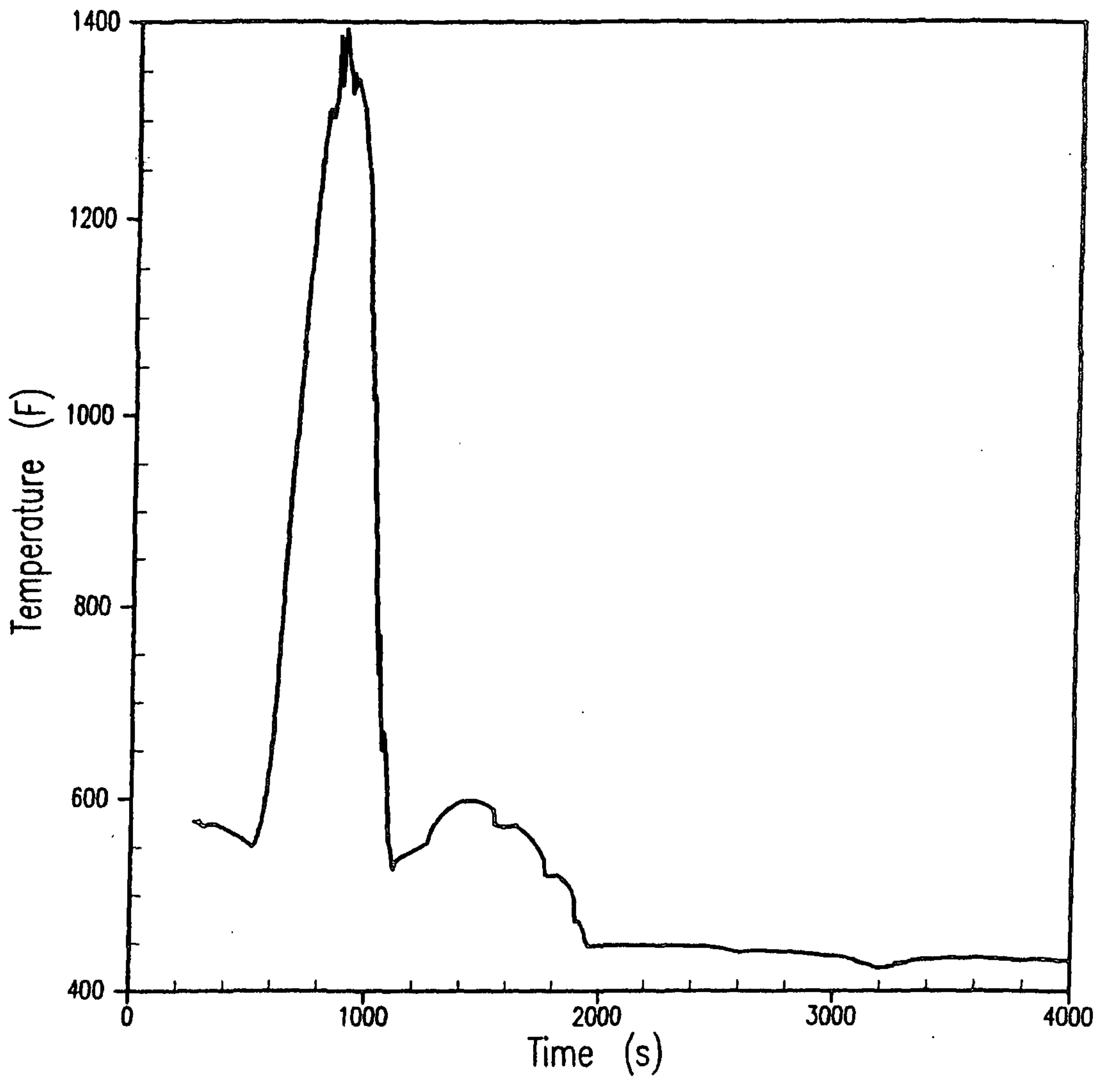
Figure 15.6-21g



Byron/Braidwood Stations Updated Final Safety Analysis Report

Peak Clad Temperature - Unit 2 (4 Inch, Nominal T_{AVG})

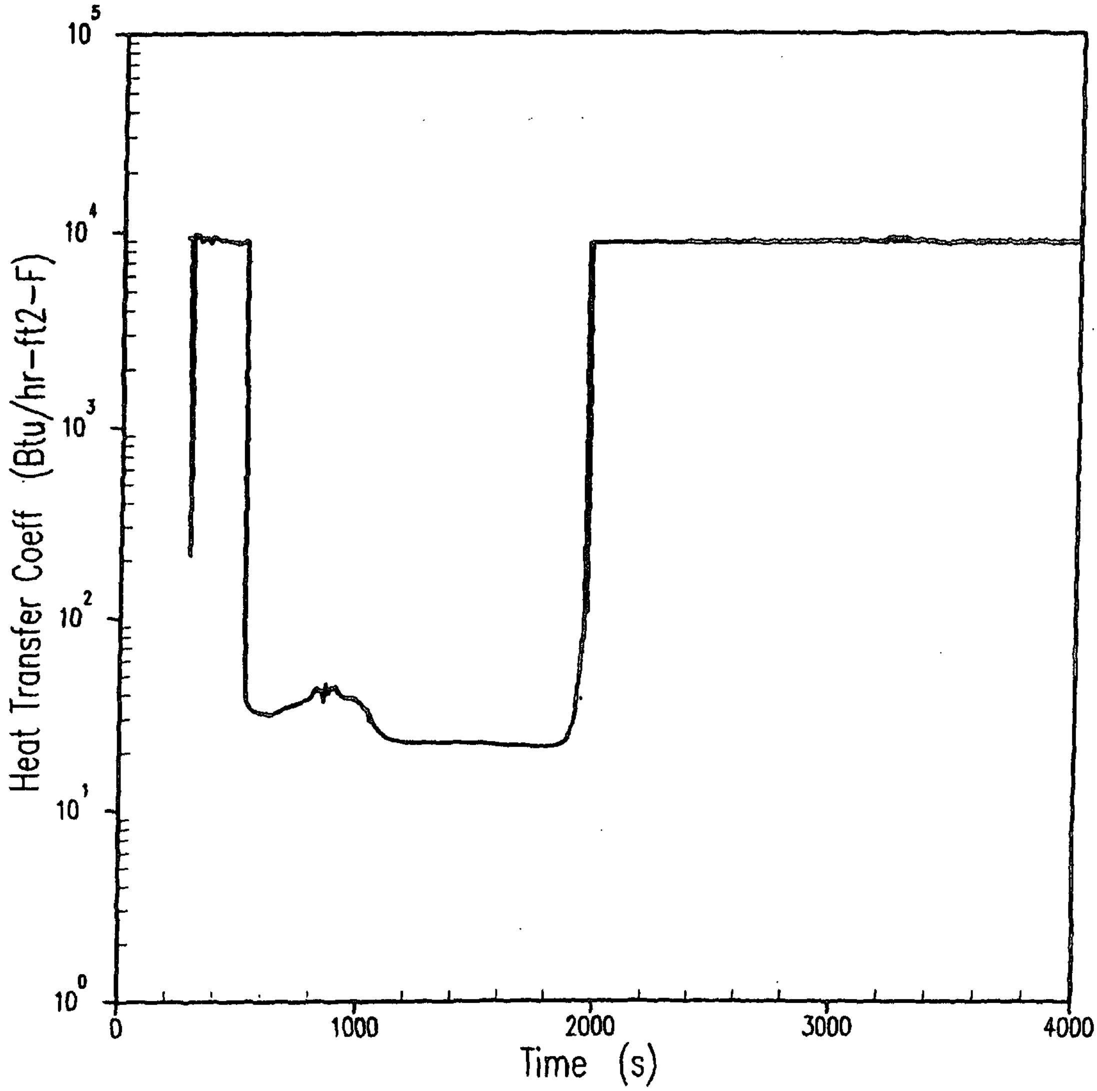
Figure 15.6-21h



Byron/Braidwood Stations Updated Final Safety Analysis Report

Fluid Temperature - Unit 2
(4 Inch, Nominal T_{AVG})

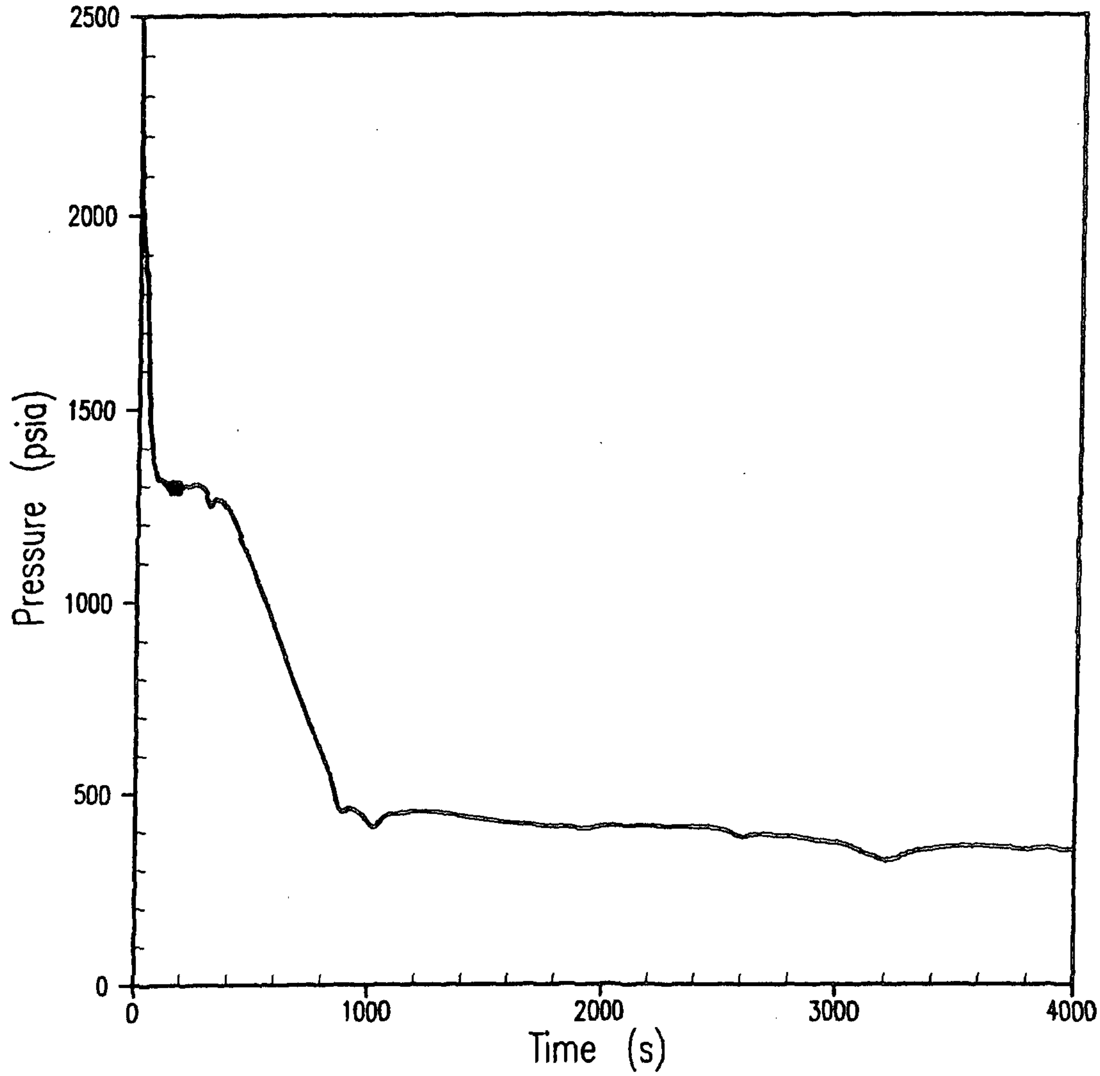
Figure 15.6-21l



Byron/Braidwood Stations Updated Final Safety Analysis Report

Heat Transfer Coefficient - Unit 2 (4 Inch, Nominal T_{AVG})

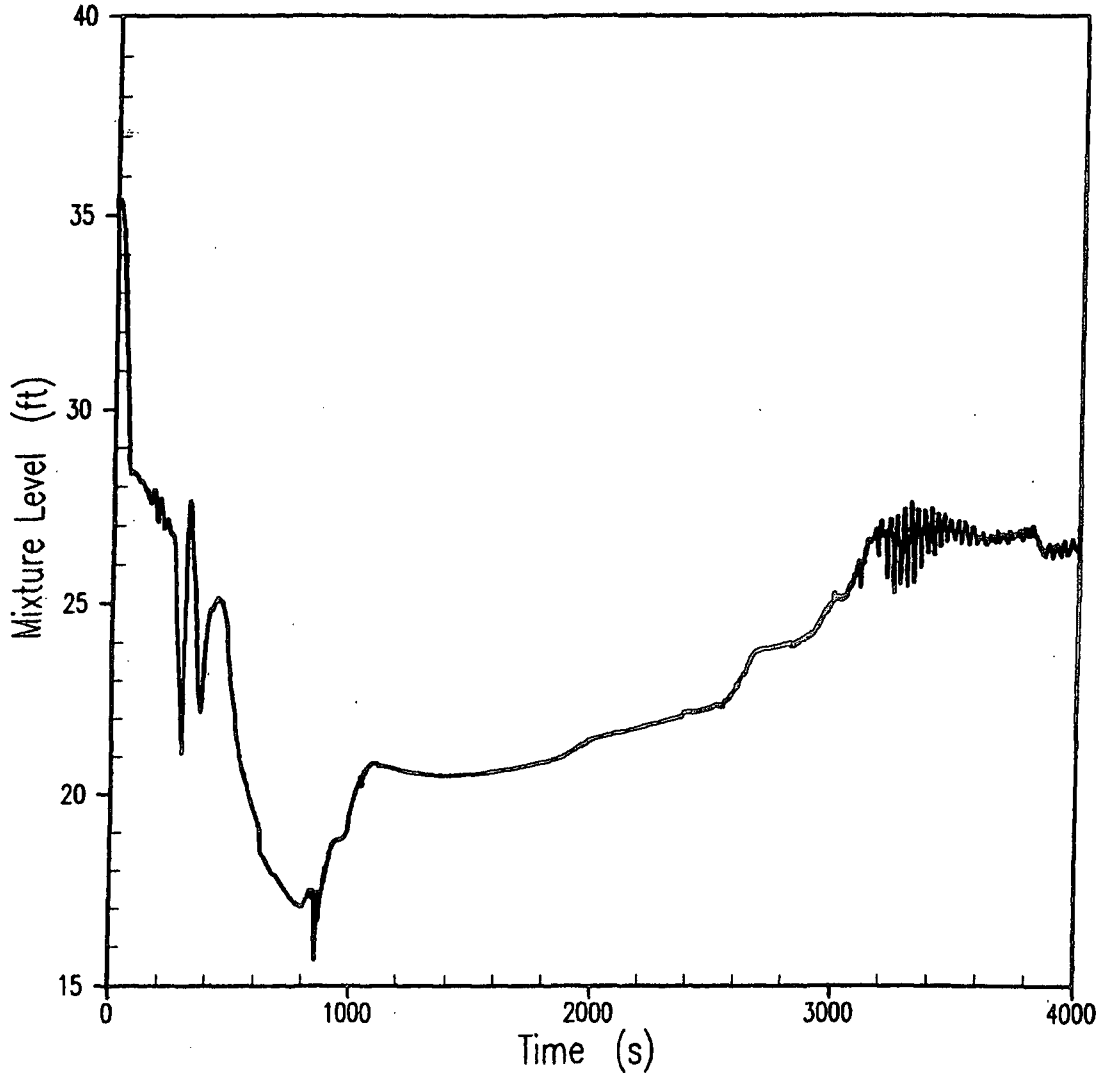
Figure 15.6-21)



Byron/Braidwood Stations Updated Final Safety Analysis Report

RCS Pressure - Unit 2 (4 Inch, Nominal T_{AVG})

Figure 15.6-21k



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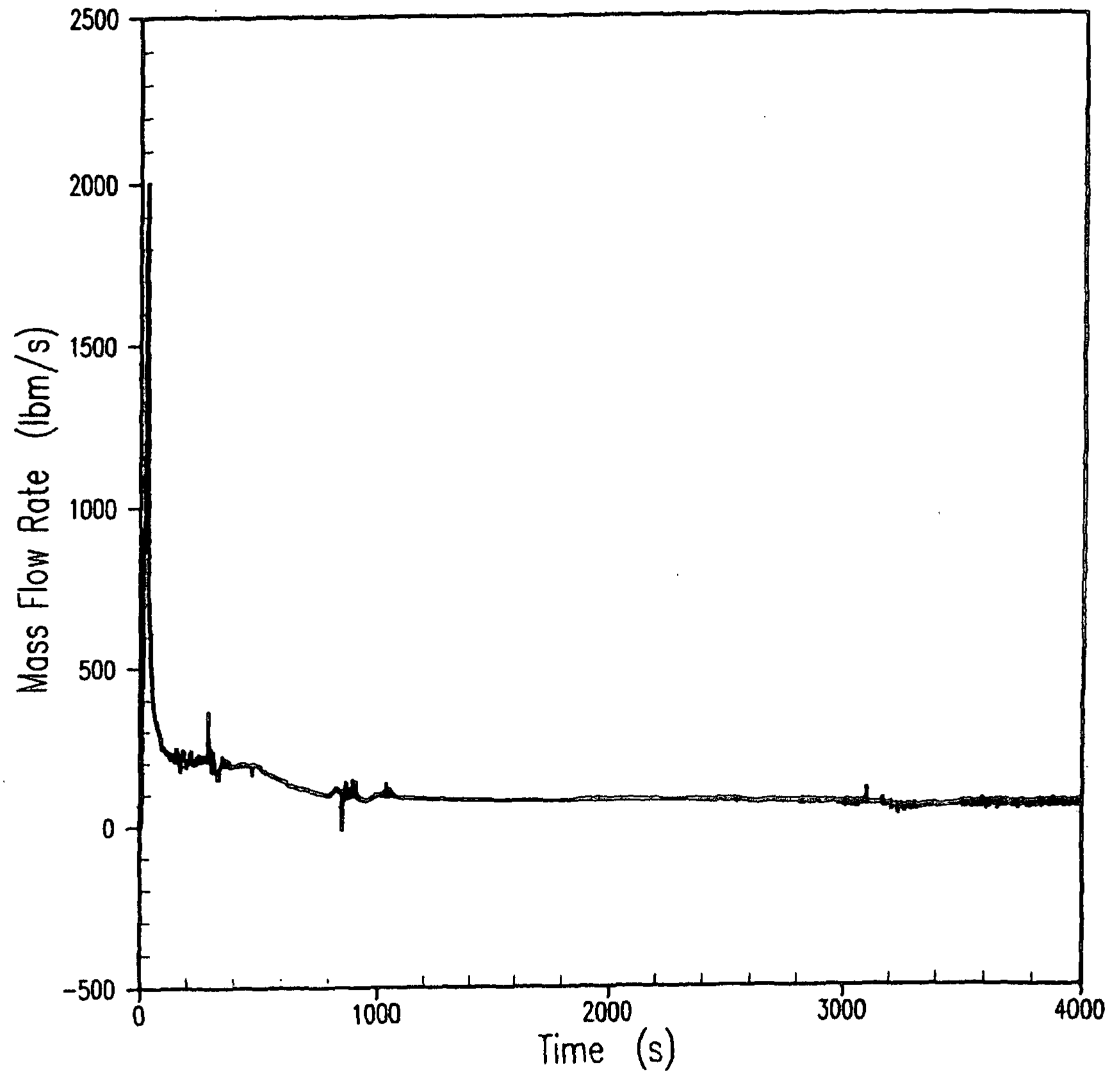
Core Mixture Level - Unit 2
(4 Inch, Nominal T_{AVG})

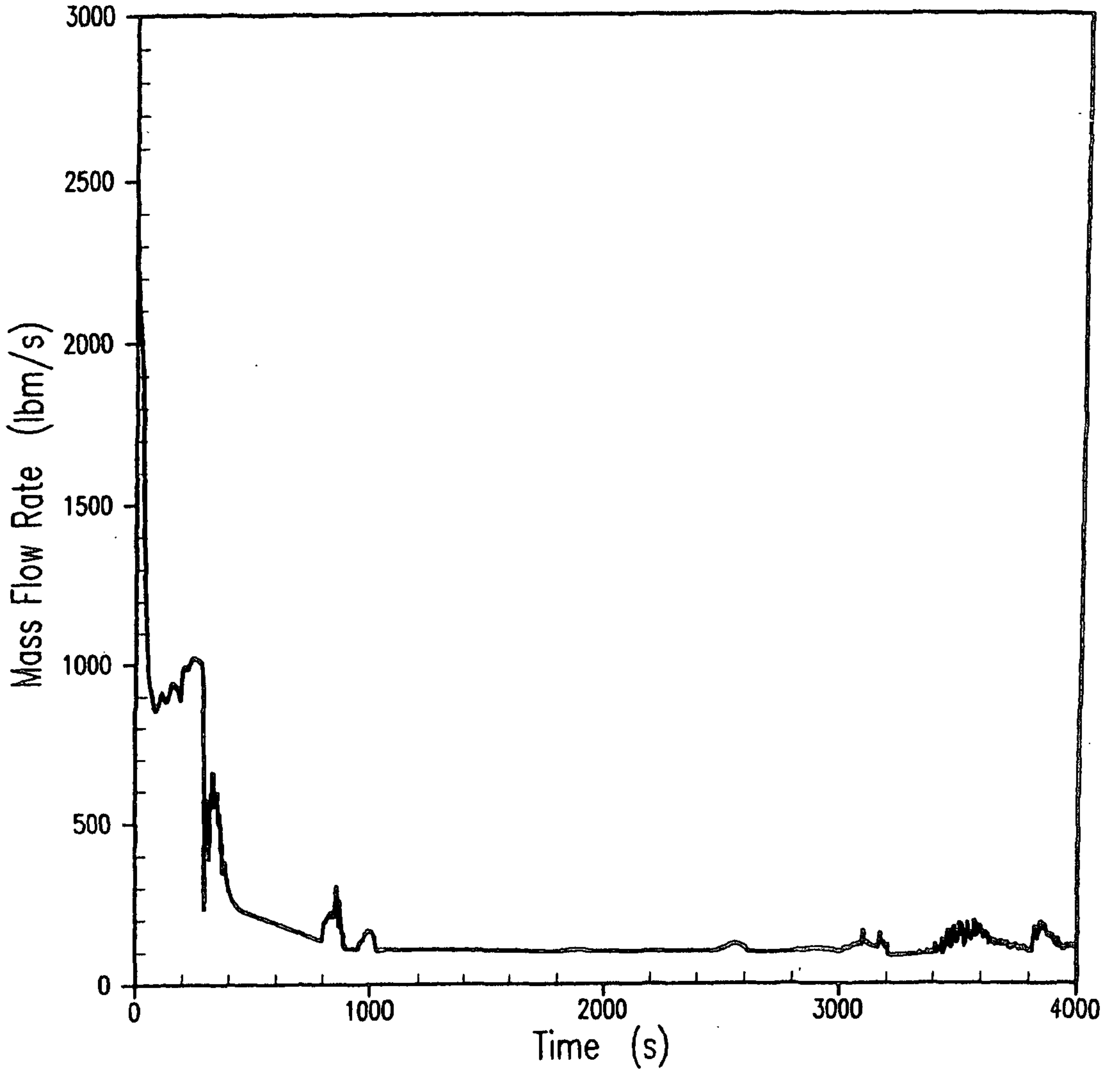
Figure 15.6-211



Byron/Braidwood Stations Updated Final Safety Analysis Report

Core Steam Flow Rate - Unit 2
(4 Inch, Nominal T_{AVG})

Figure 15.6-21m



Byron/Braidwood Stations Updated Final Safety Analysis Report

Break Flow - Unit 2
(4 Inch, Nominal T_{AVG})

Figure 15.6-21n

Security - Related Information Figure Withheld Under 10 CFR 2.390

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FIGURE 15.7-1

SPENT FUEL STORAGE POOL PLAN

DECEMBER 2010

Security - Related Information Figure Withheld Under 10 CFR 2.390

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FIGURE 15.7-2

SPENT FUEL STORAGE POOL SECTION