CHAPTER 15

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CHAPTER 15

ACCIDENT ANALYSIS

15.0 GENERAL

This chapter addresses the representative initiating events listed on Table 15-1 of Regulatory Guide 1.70, Revision 3, "Standard Format and Content of Safety Analysis Reports," as they apply to a Westinghouse pressurized water reactor (PWR).

Item 6.2 of Table 15-1 in the Regulatory Guide 1.70 warrants comment, as follows:

Item 6.2 - No instrument lines from the reactor coolant system (RCS) boundary in the nuclear steam supply system (NSSS) PWR design penetrate the containment. (For the definition of the RCS boundary, refer to Section 5, ANSI-N18.2, "Nuclear Safety Criteria for the Design of Stationary PWR Plants," 1973).

15.0.1 Classification of Plant Conditions

Since 1970, the American Nuclear Society classification of plant conditions has been used to divide 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:

1. Condition I: Normal Operation and Operational Transients
2. Condition II: Faults of Moderate Frequency
3. Condition III: Infrequent Faults
4. 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 RTS 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 normal plant operation, refueling, and maintenance. As such, Condition I occurrences are accommodated with margin between any Beaver Valley Power Station - Unit 2 (BVPS-2) 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.

Typical Condition I events are as follows:

1. Steady state and shutdown operations

   a. Mode 1 - Power operation (>5 to 100 percent of rated thermal power).

   b. Mode 2 - Start-up (K_{eff} $\geq$0.99, $\leq$5 percent of rated thermal power).

   c. Mode 3 - Hot standby (K_{eff} <0.99, T_{avg} $\geq$350°F).

   d. Mode 4 - Hot shutdown (K_{eff} <0.99, 200°F <T_{avg}<350°F).

   e. Mode 5 - Cold shutdown (K_{eff} <0.99, T_{avg} $\leq$200°F).

   f. Mode 6 - Refueling (K_{eff} $\leq$0.95, T_{avg} $\leq$140°F).

2. Operation with permissible deviations

   Various deviations which may occur during continued operation as permitted by the BVPS-2 Technical Specifications must be considered in conjunction with other operational modes. These include:

   a. Operation with components or systems out of service.

   b. Radioactivity in the reactor coolant, including leakage from fuel with cladding defects.

      1) Fission products
      2) Corrosion products
      3) Tritium

   c. Operation with steam generator leaks up to the maximum allowed by the Technical Specifications.

   d. Testing as allowed by the Technical Specifications.

3. Operational transients

   a. Plant heatup and cooldown (up to 100°F/hr for the RCS; 200°F/hr for the pressurizer during cooldown; and 100°F/hr for the pressurizer during heatup).

   b. Step load changes (up to ±10 percent).

   c. Ramp load changes (up to 5 percent/minute).

   d. Load rejection (up to and including design 50 percent load rejection transient).

15.0.1.2 Condition II - Faults of Moderate Frequency

At worst, a Condition II fault results in a reactor trip with BVPS-2 being capable of returning to operation. By definition, these faults (or events) do not propagate to cause a more serious fault, that is, Condition III or IV events. In addition, Condition II events are not expected to result in fuel rod failures or RCS or secondary system overpressurization.

The following faults are included in this category:

1. Feedwater system malfunctions causing a reduction in feedwater temperature (Section 15.1.1).
2. Feedwater system malfunctions causing an increase in feedwater flow (Section 15.1.2).
3. Excessive increase in secondary steam flow (Section 15,1.3).
4. Inadvertent opening of a steam generator relief or safety valve causing a depressurization of the main steam system (Section 15.1.4).
5. Loss of external load (Section 15.2.2).
6. Turbine trip (Section 15.2.3).
7. Inadvertent closure of main steam isolation valves (Section 15.2.4).
8. Loss of condenser vacuum and other events resulting in turbine trip (Section 15.2.5).
9. Loss of nonemergency ac power to the station auxiliaries (Section 15.2.6).
10. Loss of normal feedwater flow (Section 15.2.7).

11. Partial loss of forced reactor coolant flow (Section 15.3.1).

12. Uncontrolled rod cluster control assembly (RCCA) bank withdrawal from a subcritical or low power start-up condition (Section 15.4.1).

13. Uncontrolled RCCA bank withdrawal at power (Section 15.4.2).

14. Control rod misalignment (dropped full length assembly or statically misaligned full length assembly) (Section 15.4.3).

15. Start-up of an inactive reactor coolant loop at an incorrect temperature (Section 15.4.4).

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

17. Inadvertent operation of emergency core cooling system during power operation (Section 15.5.1).

18. Chemical and volume control system malfunction that increases reactor coolant inventory (Section 15.5.2).

19. Inadvertent opening of a pressurizer relief valve (Section 15.6.1).

20. Failure of small lines carrying primary coolant outside containment (Section 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 BVPS-2. They will be accommodated with the failure of only a small fraction of the fuel rods although sufficient fuel damage might occur to preclude immediate resumption of the operation. The release of radioactivity will not be sufficient to interrupt or restrict public use of those areas beyond the exclusion area boundary. A Condition III fault will not, by itself, generate a Condition IV fault or result in a consequential loss of function of the RCS or containment barriers. The following faults are included in this category:

1. Minor steam system piping failures (Section 15.1.5).

2. Complete loss of forced reactor coolant flow (Section 15.3.2).

3. Control rod misalignment (single RCCA withdrawal at full power) (Section 15.4.3).

4. Inadvertent loading and operation of a fuel assembly in an improper position (Section 15.4.7).

5. Loss-of-reactor-coolant from small ruptured pipes or from cracks in large pipes, which actuate the ECCS (Section 15.6.5).

6. Waste gas system failure (Section 15.7.1).

7. Radioactive liquid waste system leak or failure (atmospheric release) (Section 15.7.2).

8. Liquid containing tank failure (Section 15.7.3).

15.0.1.4 Condition IV - Limiting Faults

Condition IV occurrences are faults which are not expected to occur, but are postulated because their consequences would include the potential for release of significant amounts of radioactive material. They are the most drastic events which must be designed against and represent limiting design cases. Plant design must be such as to preclude a fission product release to the environment resulting in an undue risk to public health and safety in excess of guideline values of 10 CFR 50.67. A single Condition IV fault must not cause a consequential loss of required functions of systems needed to mitigate the consequences of the fault including those of the ECCS and containment. The following faults have been classified in this category:

1. Steam system piping failure (Section 15.1.5).

2. Feedwater system pipe break (Section 15.2.8).

3. Reactor coolant pump shaft seizure (locked rotor) (Section 15.3.3).

4. Reactor coolant pump shaft break (Section 15.3.4).

5. Spectrum of RCCA ejection accidents (Section 15.4.8).

6. Steam generator tube failure (Section 15.6.3).

7. Loss-of-coolant accidents (LOCA) resulting from the spectrum of postulated piping breaks within the reactor coolant pressure boundary (RCPB) (Section 15.6.5).

8. Fuel handling accident (Section 15.7.4).

9. Spent fuel cask drop (Section 15.7.5).

15.0.1.5 Other Conditions

Concerns regarding the potential for water relief through pressurizer safety valves have led to analysis of events which could overfill the pressurizer. The concern arises from the possibility that water release through the safety valves could cause damage which could prevent a valve from reclosing. This could potentially result in the unacceptable progression of a Condition II event (e.g., inadvertent ECCS initiation which results in overfilling the pressurizer) degrading to a more severe Condition III event (e.g., a small break LOCA occurring due to failure of a safety valve to reseat), or to a similar degradation of a feedline break event resulting in a concurrent LOCA. Since assumptions which are made for the normal feedline break or inadvertent ECCS event are not necessarily conservative for the cases for valve operability, new cases are examined. The analysis is done only for the three loops operating case for Inadvertent Operation of Emergency Core Cooling System During Power Operation (Section 15.5.1) and for Feedwater System Pipe Break (Section 15.2.8).

15.0.2 Optimization of Control Systems

A control system automatically maintains prescribed conditions at BVPS-2 even under a conservative set of reactivity parameters with respect to both system stability and transient performance. For each mode of BVPS-2 operation, a group of optimum controller set points is determined. In areas where the resultant set points are different, compromises based on the optimum overall performance are made and verified. A consistent set of control system parameters is derived satisfying BVPS-2 operational requirements throughout the core life and for various levels of power operation.

The system set points are derived by an analysis of the following control systems: rod control, steam dump, steam generator level, pressurizer pressure, and pressurizer level.

15.0.3 Plant Characteristics and Initial Conditions Assumed in the Accident Analysis

Each of the three RCS loops are equipped with loop isolation valves. However, the station is not currently licensed to operate with less than all three RCS loops in service. Therefore, the UFSAR supports only three-loop operation.

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.

The guaranteed NSSS thermal power output is the power output including the thermal power generated by the RCPs.

Additionally, the engineered safety features (ESF) design rating is defined as a thermal power higher than the guaranteed value in order not to preclude realization of future potential power capability (including the thermal power generated by the RCPs).

Allowances for errors in the determination of the steady-state power level are made as described in Section 15.0.3.2. The values of pertinent BVPS-2 parameters utilized in the non-LOCA accident analyses are given in Table 15.0-2. The thermal power values used for each transient analyzed are given in Table 15.0-3.

15.0.3.2 Initial Conditions

For most events that 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 DNBR limit, as described in WCAP-11397 (Friedland and Ray, 1989). This methodology is known as the "Revised Thermal Design Procedure" (RTDP).

For accidents that 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 steady state errors were assumed in the analyses:

a.	Core Power	±0.6% allowance for calorimetric error
b.	Average Reactor Coolant System Temperature	±4°F allowance for controller deadband and measurement error
c.	Pressurizer Pressure	±45 psi allowance for steady state fluctuations and measurement error

Table 15.0-3 summarizes initial conditions and computer codes used in the accident analysis.

15.0.3.3 Core Power Distribution

The limiting conditions occurring during reactor transients are dependent on the core power distribution. The design of the core and the control system minimizes adverse power distribution through the placement of control rods and operating methods. In addition, the core power distribution is continuously monitored by the reactor protection system as described in Chapter 7. Audible alarms will be activated in the main control room whenever the power distribution exceeds the limits assumed as initial conditions for the transients presented in this chapter.

For transients which may be departure from nucleate boiling (DNB) limited both the radial and axial peaking factors are of importance. The core thermal limits illustrated on Figure 15.0-1 are based on a reference axial power shape. The radial peaking factor $F\Delta H$ increases with decreasing power and with increasing rod insertion. The increase in $F\Delta H$ is included in the core limits illustrated on Figure 15.0-1. All transients that may be DNB limited are assumed to begin with an $F\Delta H$ consistent with the initial power level as defined in the Technical Specifications.

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 an $F\Delta H$ consistent with the initial power level as defined in the Technical Specifications.

For overpower transients which are slow with respect to the fuel rod thermal time constant, fuel rod thermal evaluations are determined as discussed in Section 4.4. Examples of this are the uncontrolled boron dilution incident, which lasts many minutes, and the excessive load increase incident, which reaches equilibrium without causing a reactor trip. For overpower transients which are fast with respect to the fuel rod thermal time constant (for example, the uncontrolled RCCA bank withdrawal from subcritical and RCCA ejection incidents, which result in a large power rise over a few seconds), a detailed fuel heat transfer calculation is performed and discussed in the sections covering those specific accidents. 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 Analysis

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.

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 ruptures in the RCS, do not depend highly on reactivity feedback effects. The values used for each accident are given in Table 15.0-3. Reference is made in that table to Figure 15.0-2 which shows the upper and lower bound Doppler power coefficients as a function of power, used in the transient analysis. Table 15.0-3 shows the minimum moderator coefficients used in analyses. The justification for use of conservatively large versus small reactivity coefficient values is treated on an event-by-event basis. Conservative combinations of parameters are used for a given transient 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 RCCAs 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 percent of the rod cluster travel. For all accidents except the loss of flow events, the insertion time to dashpot entry is conservatively taken as 2.7 seconds. The RCCA position versus time assumed in accident analyses is shown on Figure 15.0-4.

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 an 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 RCCA negative reactivity insertion versus time is shown on Figure 15.0-6. The curve shown on this figure was obtained from Figures 15.0-4 and 15.0-5. A total negative reactivity insertion following a trip of 4% $\Delta\rho$ 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 Section 4.3.

The normalized RCCA 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 required use of three-dimensional or axial one-dimensional core models, the negative reactivity insertion resulting from the reactor trip is calculated directly to the reactor kinetics code and is not separable from the other reactivity feedback effects. In this case, the RCCA position versus time (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 four trip breakers, two per channel set, feeding power to the control rod drive mechanisms (CRDMs). The loss of power to the mechanism coils causes the mechanisms to release the RCCAs 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 set points assumed in accident analyses and the time delay assumed for each trip function are given in Table 15.0-4.

Reference is made in that table to overtemperature and overpower trip shown on Figure 15.0-1. This figure presents the allowable reactor coolant loop average temperature and ΔT for the flow and power distribution, 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 set point 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 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 the DNBR limit is not violated 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 set point); high pressure (fixed set point); low pressure (fixed set point); overpower and overtemperature ΔT (variable set points).

The limit value, which was used as the DNBR limit for all accidents, is conservative compared to the actual design DNBR value required to meet the DNB design basis as discussed in Section 4.4.

The difference between the limiting trip point, assumed for the analysis and the nominal trip point represents an allowance for instrumentation channel error and set point error. Nominal trip set points are specified in the BVPS-2 Technical Specifications, Chapter 16. During BVPS-2 start-up tests, it is 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 are verified in accordance with the BVPS-2 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 set point are presented in Table 15.0-5. 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 multiple 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.

15.0.8 Plant Systems and Components Available for Mitigation of Accident Effects

The Westinghouse NSSS is designed to afford proper protection against the possible effects of natural phenomena, postulated environmental conditions, and the dynamic effects of the postulated accident. In addition, the design incorporates features which minimize the probability and effects of fires and explosions. Chapter 17 discusses the quality assurance program which is implemented to ensure that BVPS-2 will be designed, constructed, and operated without undue risk to the health and safety of the general public. The incorporation of these features, coupled with the reliability of the design, ensures that the normally operating systems and components listed in Table 15.0-6 will be available for mitigation of the events discussed in Chapter 15. 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 events, the operation of the nonsafety-related rod control system, other than the reactor trip portion of the control rod drive system, is considered only if that action results in more severe consequences. 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 nonsafety-related control system operation to determine the worst case.

15.0.9 Fission Product Inventories

15.0.9.1 Activities in Core

The core fission product inventories and other nuclides utilized to support shielding adequacy and radwaste effluent assessments included with the original application were calculated as described in Section 11.1 using the computer program ACTIVITY 2. This program calculates the contributions from parent, daughter, and granddaughter nuclides by solving the differential equations given in Section 11.1. The resulting inventories which were based on a core power level of 2766 MWt and a one year fuel cycle length are presented in Table 12.2-3 and are considered historical.

Computer code ORIGEN-S is utilized to calculate the core radioactivity inventory used in performing design basis radiological dose consequence analyses. ORIGEN-S is distributed by the Radiation Safety Information Computational Center, Oak Ridge, TN. This code is readily available, and is a commonly used code for this purpose. The code input parameters used for this calculation are provided in Table 15.0-7. The inventory values are computed assuming an uprated core thermal power of 2918 MWt including allowance for calorimetric uncertainties and an eighteen month fuel cycle length. The isotopes of interest at the end of an equilibrium fuel cycle are listed in Table 15.0-7a.

15.0.9.2 Activities in the Fuel Pellet Clad Gap

The core gap activities are based on the guidance provided in Regulatory Guide 1.183. Table 3 in Regulatory Guide 1.183 specifies the fraction of fission product inventory assumed to be in the fuel rod gap to be used for the LOCA, the Control Rod Ejection Accident, and other Non-LOCA accidents, respectively. The footnote identifies that the applicability of Table 3 is limited to LWR fuel with peak burnups of 62 GWD/MTU "provided that the maximum linear heat generation rate does not exceed 6.3 kW/ft peak rod average power for burnups exceeding 54 GWD/MTU." The gap fractions utilized for events at BVPS which could result in fuel failure, are consistent with the requirements for RG 1.183 and are listed in Table 15.0-7b.

15.0.9.3 Primary and Secondary Side Coolant Activities

The design primary and secondary side coolant fission product inventories and other nuclides utilized to support shielding adequacy and radwaste effluent assessments included with the original application were calculated as described in Section 11.1 using the computer program ACTIVITY 2. The resulting primary and secondary coolant inventories which were based on a core power level of 2766 MWt and a one year fuel cycle length are presented in Tables 11.1-2 and 11.1-6, respectively, and are considered historical.

The equilibrium concentrations in the RCS and the secondary coolant system used in performing design basis radiological dose consequence analyses are calculated assuming full power operation with no VCT purge for the following cases: 1) one percent fuel defects, 2) plant Technical Specification iodine concentrations. The Technical Specifications for BVPS-2 restrict the concentration in the primary and secondary systems to 0.35 and 0.1 μCI/gm I-131 dose equivalent, respectively. The Technical Specification activities are used in the analysis of the main steam line break (MSLB), the failure of small lines carrying primary coolant outside containment, and the steam generator tube rupture. The waste gas system rupture analysis utilizes primary coolant concentrations with 1 percent fuel defects.

The parameters used to calculate the revised design primary and secondary side coolant radioactivities are presented in Table 15.0-8a. The revised core inventory values given in Table 15.0-7a are used as the basis for calculating the revised primary coolant and secondary side coolant and steam radioactivities. The computer codes and methodology used to determine these remain unchanged from those used to determine the former primary coolant and secondary side coolant and steam radioactivities.

The revised design primary and secondary side coolant radioactivities are presented in Table 15.0-8b. These concentrations reflect power operation at a core power of 2918 MWt and an eighteen month fuel cycle duration.

The primary coolant technical specification concentrations are presented in Table 15.0-8c. These concentrations reflect the change to the basis for primary coolant radioactivity from 1.0 μCi/gm I-131 dose equivalent (original license application) to a revised, lower value of 0.35 μCi/gm.

For the waste gas system rupture analysis, primary coolant concentrations with 1 percent fuel defects are assumed. These RCS concentrations are given in Table 15.0-8b. The calculation of releases due to a liquid-containing tank failure uses expected normal operation concentrations of 0.12 percent fuel defects. These concentrations are also presented in Table 11.1-2.

15.0.9.4 Iodine Spiking Concentrations

The analysis of an MSLB, steam generator tube rupture, and the failure of small lines carrying primary coolant outside containment include equilibrium coolant iodine concentrations augmented by iodine spiking. Both pre-accident and concurrent iodine spiking models are considered.

The pre-accident iodine spiking concentrations are determined by increasing the primary coolant iodine concentrations to the maximum value described in the Technical Specifications. For BVPS-2, the pre-accident iodine spike concentrations in the reactor coolant is 21 µCi/gm DE I-131 (transient Technical Specification limit for full power operation) or 60 times the reactor coolant iodine Technical Specification concentrations. The resulting primary coolant iodine concentrations are given in Table 15.0-9.

In accordance with Regulatory Guide 1.183, the concurrent iodine spike is modeled by increasing the iodine release rates from fuel rods into the primary coolant to an accident dependent value times the equilibrium iodine concentration release rates. The equilibrium iodine release rates are conservatively calculated based on the Technical Specification reactor coolant activities, along with the maximum design letdown rate, maximum Technical Specification allowed leakage, and an ion-exchanger iodine removal efficiency of 100%. Maximizing the reactor coolant cleanup results in maximizing the equilibrium iodine appearance rates. These parameters and the revised model are presented in Table 15.0-10a. Table 15.0-10 presents the maximum normal operational equilibrium iodine release rates and the accident dependent multipliers for concurrent iodine spiking.

15.0.10 Residual Decay Heat

15.0.10.1 Total Residual Heat

Residual heat in a subcritical core is calculated for the SBLOCA per the requirements of 10 CFR 50, Appendix K, 10 CFR 50.46, as described by Bordelon (et al 1974a, 1974b). These requirements include assuming infinite irradiation time before the core goes subcritical to determine fission product decay energy. For all other accidents, the same models are used except that fission product decay energy is based on core average exposure at the end of the equilibrium cycle.

15.0.11 Computer Codes Utilized

Summaries of some of the principal computer codes used in transient analyses are given as follows. 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 RCS pipe rupture (Section 15.6), are summarized in their respective accident analyses sections. The codes used in the analyses of each transient are listed in Table 15.0-3.

15.0.11.1 FACTRAN

FACTRAN calculates the transient temperature distribution in a cross section of a metal clad $U0_2$ fuel rod and the transient heat flux at the surface of the cladding, using as input the nuclear power and the time-dependent coolant parameters (pressure, flow, temperature, and density). The code uses a fuel model which exhibits the following features simultaneously:

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

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

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

FACTRAN is further discussed by Hargrove (1989).

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 generators (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 RPS is simulated to include reactor trips on neutron flux, overtemperature, overpower, 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 ECCS, 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 represent the minimum value of DNBR as calculated for typical or thimble cell.

LOFTRAN is further discussed by Burnett (1984).

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, or 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 2,000 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, and control rod motion. Various edits are provided, for example, channelwise 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 described by Risher and Barry (1975).

15.0.11.4 VIPRE

The VIPRE Code (Sung, et. al., 1999) is described in Section 4.4.

15.0.12 Radiological Consequences

The radiological consequences of each of the design basis accidents (DBA) were analyzed based on assumptions discussed in the respective sections. Specific parameters used in these analyses are tabulated in the corresponding sections. Table 6.4-1a lists key assumptions and input parameters associated with the BVPS control room design which was utilized in the dose consequence analyses. Section 15.6.5.4 discusses the control room design as related to dose consequences under a sub-section titled "Control Room Habitability".

Initial core and core gap activities, coolant Technical Specification equilibrium concentrations, pre-accident iodine spike primary coolant concentrations, and concurrent iodine spiking appearance rates are discussed in Section 15.0.9.

Accident atmospheric dispersion coefficients (X/Q) for the exclusion area boundary and low population zone were used to calculate the potential offsite doses. The 0.5 percent sector-dependent X/Q values, presented in Table 15.0-11, were determined as described in Section 2.3.4.

As part of the plant modifications associated with containment conversion and core power uprate, the control room X/Q values were re-calculated using the latest version of the "Atmospheric Relative Concentrations in Building Wakes" (ARCON96) methodology. The control room X/Q values applicable to release points associated with an accident at BVPS-1 or BVPS-2, are presented in Table 15.0-14 and 15.0-15, respectively. The Emergency Response Facility (ERF) X/Q values for the environmental release paths associated with the Loss-Of-Coolant Accident are also provided. The X/Q values for all of the release-receptor combinations utilized to develop the post-accident control room operator occupancy doses are summarized in Table 15.0-15. The X/Q values for all of the release-receptor combinations associated with BVPS-1 accidents addressed in Table 15.0-14 are taken into consideration when the dose consequences of the event are established based on an analysis that is bounding for both units. Occupancy factors are not included.

The atmospheric releases discussed in each accident section are used in conjunction with the appropriate X/Q values to calculate the potential offsite doses for the corresponding accidents and the potential control room doses. The methodology for determining the doses is discussed in Appendix 15A. The resulting EAB and LPZ doses are presented in Table 15.0-12 for all postulated accidents. The resulting doses to main control room personnel due to DBAs are presented in Table 15.0-13.

With the exception of the Waste Gas System Rupture which utilizes a licensing basis acceptance criteria of 500 mrem whole body, the potential offsite doses following all of the design basis accidents are within the limits of 10 CFR 50.67 while the potential doses for the main control room remain within the limits of GDC 19 of Appendix A to 10 CFR 50 (WGSR only) or 10 CFR 50.67.

15.0.13 References for Section 15.0

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Tables for Section 15.0

TABLE 15.0-1

NUCLEAR STEAM SUPPLY SYSTEM POWER RATINGS

Type	
Uprated reactor core thermal power output using the leading edge flow meter to reduce the allowance for calorimetric error (MWt)	2,900
Thermal power generated by the reactor coolant pumps (MWt), nominal	10
Guaranteed nuclear steam supply system thermal power output (MWt)	2,910

TABLE 15.0-2

BASES FOR VALUES OF PERTINENT PLANT PARAMETERS
UTILIZED IN ACCIDENT ANALYSES

	Low Tavg Program	High Tavg Program
Thermal output of nuclear steam supply system (MWt)	2,910	2,910
Reactor core thermal power output (MWt)	2,900	2,900
Core inlet temperature (°F)	528.5	543.1
Reactor coolant average temperature (°F)	566.2	580.0
Reactor coolant system pressure (psia)	2,250	2,250
Reactor coolant flow per loop (gpm)[5]	87,200	87,200
Total reactor coolant flow (10^6 lb/hr)	101.1	99.3
Total steam flow from NSSS (10^6 lb/hr)	12.05[1,3] 12.03[1,4] 13.00[2,3] 12.98[2,4]	12.08[1,3] 12.06[1,4] 13.04[2,3] 13.01[2,4]
Steam pressure at steam generator outlet (psia)	699[3] 641[4]	799[3] 735[4]
Maximum steam moisture content (percent)	0.25	0.25
Average core heat flux (10^3 BTU/hr-ft^2)	198.3	198.3

Footnotes:

(1) Feedwater temperature = 400°F
(2) Feedwater temperature = 455°F
(3) Minimum (0%) steam generator tube plugging
(4) Maximum (22%) steam generator tube plugging
(5) The value given is the Thermal Design Flow (TDF) rate used in the analysis for all non-DNB events. The analyses for DNB events assume the Minimum Measured Flow (MMF) of 266,800 gpm (total).

TABLE 15.0-3
SUMMARY OF INITIAL CONDITIONS
COMPUTER CODES USED AND
KINETIC PARAMETERS ASSUMED

	Faults	Computer Codes Utilized	Delayed Neutron Fraction	Moderator Coefficient	Doppler	DNB Correlation	Initial NSSS Thermal Power Output (MWt)	Reactor Vessel Coolant Flow (gpm)	Initial Vessel Average Temperature (°F)	Initial Pressurizer Pressure (psia)	Initial Pressurizer Water Volume ft^3	Feedwater Temperature (°F)
15.1	Increase in heat removal by the secondary system											
	Feedwater system malfunction causing an increase in feedwater flow	LOFTRAN	.0047	0.43 Δk/gm/cc (HZP cases assume the same reactivity parameters as the Steam System Piping Failure analysis - Section 15.1.5)	Minimum***	WRB-1 WRB-2M	2,910 0	266,800 261,600	584.5 547	2,242.5 2,250	834.3 364.3	400 32
	Excessive increase in secondary steam flow	LOFTRAN	NA*	NA*	NA*	WRB-1 WRB-2M	2,910	266,800	580.0	2,242.5	NA*	NA*
	Accidental depressurization of the main steam system	LOFTRAN	NA*	NA*	NA*	NA*	NA*	NA*	NA*	NA*	NA*	NA*
	Steam system piping failure (HZP Cases)	VIPRE, LOFTRAN	.0047	Function of moderator density (Section 15.1.5) (Figure 15.1-11)	Section 15.1.5	W-3	0 (Subcritical)	261,600	547	2,250	364.3	100
	Steam system piping failure (HFP Cases)	VIPRE, LOFTRAN	.0047	0.43 ΔK/gm/cc	Minimum***	W-3	2,910	266,800	581.0	2,242.5	746.8	400
15.2	Decrease in heat removal by the secondary system											
	Loss of external electrical load and/or turbine trip	LOFTRAN	.0075	+5 pcm/°F	Minimum***	WRB-1 WRB-2M	2,910 (DNB case) 2,927.5 (pressure case)	266,800 261,600	584.5 588.5	2,242.5 2,205	921.8 921.8	455 455

TABLE 15.0-3 (Cont)

	Faults	Computer Codes Utilized	Delayed Neutron Fraction	Moderator Coefficient	Doppler	DNB Correlation	Initial NSSS Thermal Power Output (MWt)	Reactor Vessel Coolant Flow (gpm)	Initial Vessel Average Temperature (°F)	Initial Pressurizer Pressure (psia)	Initial Pressurizer Water Volume ft^3	Feedwater Temperature (°F)
	Loss of non-emergency ac power to the station auxiliaries	LOFTRAN	.0075	0 pcm/°F‡	Maximum***	NA	2,927.5	261,600	588.5	2,205	921.8	400
	Loss of normal feedwater flow	LOFTRAN	.0075	0 pcm/°F‡	Maximum***	NA	2,927.5	261,600	588.5	2,205	921.8	400
	Feedwater system pipe break	LOFTRAN	.0075 .0047	0.43 Δk/gm/cc +5 pcm/°F	Minimum and Maximum***	NA	2,927.5	261,600	588.5	2,205	921.8	455
	Feedwater system pipe break (PSV operability cases)	LOFTRAN	.0047	0.43 Δk/gm/cc	Maximum***	NA	2,927.5	261,600	588.5	2,205	921.8	455
15.3	Decrease in reactor coolant system flow rate											
	Partial and complete loss of forced reactor coolant flow and locked rotor; rods-in-DNB	LOFTRAN, THINC, FACTRAN	.0075	0 pcm/°F‡	Maximum***	WRB-1 WRB-2M	2,910	266,800	584.5	2,242.5	834.3	455
	Reactor coolant pump shaft seizure (locked rotor), peak RCS pressure & clad temperature	LOFTRAN, FACTRAN	.0075	0 pcm/°F‡	Maximum***	WRB-1 WRB-2M	2,910 (DNB Case) 2,927.5 (Pressure Case)	266,800 261,600	584.5 588.5	2,242.5 2,295‡‡‡	834.3 921.8	455 455
15.4	Reactivity and power distribution anomalies											
	Uncontrolled rod cluster control assembly bank with-drawal from a sub-critical or low power start-up condition	TWINKLE, FACTRAN, VIPRE	.0075	+5 pcm/°F	Least negative Doppler defect - 962 pcm	W-3 WRB-1	0 (Subcritical)	162,192	547	2,205	NA*	NA*

TABLE 15.0-3 (Cont)

Faults	Computer Codes Utilized	Delayed Neutron Fraction	Moderator Coefficient	Doppler	DNB Correlation	Initial NSSS Thermal Power Output (MWt)	Reactor Vessel Coolant Flow (gpm)	Initial Vessel Average Temperature (°F)	Initial Pressurizer Pressure (psia)	Initial Pressurizer Water Volume ft^3	Feedwater Temperature (°F)
Uncontrolled rod cluster assembly bank withdrawal at power	LOFTRAN	.0075 .0047	0.43 Δk/gm/cc +5 pcm/°F (part power) 0 pcm/°F (full power)	Maximum and minimum***	WRB-1 WRB-2M	2,910, 1,746, 291	266,800	581.0, 567.8, 551.3	2,242.5	834.3, 646.3, 411.3	455, 392, 267
Control rod mis-alignment	VIPRE, LOFTRAN	NA*	NA*	NA*	WRB-1 WRB-2M	NA*	NA*	NA*	NA*	NA*	NA*
Chemical & volume control system mal-function that results in a decrease in boron concentration in the reactor coolant	NA*	NA*	NA*	NA*	NA*	NA*	NA*	NA*	NA*	NA*	NA*
Inadvertent loading and operating of a fuel assembly in an improper position	Section 4.3	NA	NA	NA	NA	2910	266,800	580.2	2,220	NA	NA
Spectrum of rod cluster control assembly ejection accidents	TWINKLE, FACTRAN	.0055 (BOL) .00474 (EOC)	Section 15.4.8 (BOC, EOC)**	Least negative Doppler defect 962 (BOC) 941 (EOC)	NA*	2,917.4‡‡ 0	261,600 162,192	588.5 547	NA*	NA*	NA*

TABLE 15.0-3 (Cont)

	Faults	Computer Codes Utilized	Delayed Neutron Fraction	Moderator Coefficient	Doppler	DNB Correlation	Initial NSSS Thermal Power Output (MWt)	Reactor Vessel Coolant Flow (gpm)	Initial Vessel Average Temperature (°F)	Initial Pressurizer Pressure (psia)	Initial Pressurizer Water Volume ft^3	Feedwater Temperature (°F)
15.5	Increase in coolant inventory											
	Inadvertent operation of ECCS during power operation	LOFTRAN	NA*	NA*	NA*	NA*	NA*	NA*	NA*	NA*	NA*	NA*
	Inadvertent operation of ECCS during power operation (PSV operability case)	LOFTRAN	.0047	0.43 Δk/gm/cc	Maximum***	NA*	2,927.5	261,600	556.7	2,205	709.3	455
15.6	Decrease in reactor coolant inventory											
	Inadvertent opening of a pressurizer relief valve	LOFTRAN	.0075	+5 pcm/°F	Minimum***	WRB-1 WRB-2M	2,910	266,800	584.5	2,242.5	834.3	455 400

NOTES:

* Cases were considered at 576.2°F + 7.5°F/-9.5°F and psia ±45 psi. The values given above yielded the most limiting results.
** BOC - Beginning of cycle
EOC - End of cycle
*** Reference Figure 15.0-2. Maximum refers to lower curve and minimum refers to upper curve.
‡ Analysis at full power with a 0 pcm/°F MTC bounds analysis at part power with a PMTC.
‡‡ 100.6% of nominal core power
‡‡‡ The FACTRAN portion of the analysis assumes an initial pressure of 2,205 psia which maximizes the peak clad temperature transient.

TABLE 15.0-4

TRIP POINTS AND TIME DELAYS TO TRIP
ASSUMED IN ACCIDENT ANALYSES**

Trip Function	Limiting Trip Point Assumed in Analysis	Time Delays (sec)
Power range high neutron flux, high setting	116 percent	0.5
Power range high neutron flux, low setting	35 percent	0.5
Overtemperature ΔT	Variable, (Figure 15.0-1)	2.0*
Overpower ΔT	Variable, (Figure 15.0-1)	2.0*
High pressurizer pressure	2,405 psig	2.0
Low pressurizer pressure	1,920 psig	2.0
Low reactor coolant flow (from loop flow detectors)	87 percent loop flow	1.0
Lo-lo steam generator level	0 percent of narrow range level span (FLB, LONF, and LOOP)	2.0

NOTE:

* Time delay given only includes channel electronics, trip logic and gripper release. Additional delays in the trip are a 6 second RTD response, a 2 second filter on the vessel Tavg signal and a 6 second filter on the vessel ΔT signal.

** The trip functions listed refer to direct reactor trips generated by the RPS. In addition, a feedwater isolation signal on a high-high steam generator level (assumed to be 100% narrow range level in the limiting case) results in an indirect reactor trip due to a turbine trip (assumed to occur in 2.5 seconds.)

The feedwater isolation is assumed to occur in 7.0 seconds (including valve closure time) which prevents the main steam lines from overstressing due to water fillup. This function is not credited in any accident analysis.

TABLE 15.0-5

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

Variable	Accuracy of Measurement of Variable (percent error)	Effect on Thermal Power Determination (percent 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 (percent of rated power)			±2(a)
Axial power distribution effects on total ion chamber current			
Estimated error (percent of rated power)		3	
Assumed error (percent of rated power)			±5(b)
Instrumentation channel drift and set point reproducibility			
Estimated error (percent of rated power)		1	

TABLE 15.0-5 (Cont)

Variable	Accuracy of Measurement of Variable (percent error)	Effect on Thermal Power Determination (percent error) (Estimated)	(Assumed)
Assumed error (percent of rated power)			±2(c)
Total assumed error in set point (a) + (b) ± (c)			±9

	Percent of Rated Power
Nominal set point	109
Maximum overpower trip assuming all individual errors are simultaneously in the most adverse direction	116

TABLE 15.0-6

PLANT SYSTEMS AND EQUIPMENT AVAILABLE FOR TRANSIENT AND ACCIDENT CONDITIONS

	Incident	Reactor Trip Functions	ESF Actuation Functions	Other Actuation Functions and Equipment	ESF Equipment
15.1	Increase in heat removed by the secondary system				
	Feedwater system malfunction causing an increase in feedwater flow	Power range high flux, turbine trip-reactor trip, manual, OTΔT, OPΔT	NA	Feedwater isolation valves, trip of turbine from high steam generator level*	NA
	Excessive increase secondary steam flow	Power range high flux, manual, OTΔT, OPΔT, low pressurizer pressure	NA	NA	NA
	Accidental depressurization of the main steam system	Low pressurizer pressure, manual, SIS, OTΔT, OPΔT, high neutron flux	Low pressurizer pressure, low compensated steam line pressure, Hi-1 containment pressure, manual	Feedwater isolation valves, steamline isolation valves (Hi-2 containment pressure, low steam line pressure, high negative steam pressure rate in any loop)	Auxiliary feed system; safety injection system
	Steam system piping failure	SIS, low pressurizer pressure, manual, OPΔT, OTΔT, high neutron flux	Low pressurizer pressure, low compensated steamline pressure, Hi-1 containment pressure, manual	Feedwater isolation valves, steamline isolation valves (Hi-2 containment pressure, low steam line pressure, high negative steam pressure rate in any loop)	Auxiliary feed system; safety injection system; Containment heat removal system (Hi-3 containment pressure)
15.2	Decrease in heat removal by the secondary system				
	Loss of external electrical load/turbine trip	High pressurizer pressure, OTΔT, high pressurizer water level, manual		Pressurizer safety valves, steam generator safety valves	Auxiliary feed system

TABLE 15.0-6 (Cont)

	Incident	Reactor Trip Functions	ESF Actuation Functions	Other Equipment	ESF Equipment
	Loss of Non-emergency ac power to the station auxiliaries	Steam generator lo-lo level, manual	Steam generator lo-lo level	Steam generator safety valves	Auxiliary feed system
	Loss of normal feedwater flow	Steam generator lo-lo level, manual	Steam generator lo-lo level	Steam generator safety valves	Auxiliary feed system
	Feedwater system pipe break	Steam generator lo-lo level, high pressurizer pressure, SIS, manual, OTΔT	Hi-1 containment pressure, steam generator lo-lo level, low compensated steam line pressure	Steam line isolation valves, (Hi-2 containment pressure) feedline isolation, pressurizer safety valves, steam generator safety valves	Auxiliary feed system; safety injection system; containment heat removal system (Hi-3 containment pressure)
15.3	Decrease in reactor coolant system flow rate				
	Partial and complete loss of forced reactor coolant flow	Low flow, manual	NA	Steam generator safety valves	NA
	Reactor coolant pump shaft seizure (locked rotor)	Low flow, manual	NA	Pressurizer safety valves, steam generator safety valves	NA

TABLE 15.0-6 (Cont)

	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 subcritical or low power start-up condition	Power range high flux (low set point), manual, Source range high neutron flux , Intermediate range high neutron flux, Power range high neutron flux (high setting), High nuclear flux rate	NA	NA	NA
	Uncontrolled rod cluster control assembly bank withdrawal at power	Power range high flux, Hi pressurizer pressure, manual, OTΔT, OPΔT, high pressurizer water level, positive neutron flux rate	NA	Pressurizer safety valves, steam generator safety valves	NA
	Control rod misalignment	Manual	NA	NA	NA
	Start-up of an inactive reactor coolant loop at an incorrect temperature	Modes 5 and 6 only	NA	NA	NA
	Chemical and volume control system malfunction that results in a decrease in boron concentration in the reactor coolant	Source range high flux, power range high flux, manual, OTΔT	NA	Low rod insertion limit annunciators manual actuation of VCT, outlet isolation valves	NA

TABLE 15.0-6 (Cont)

	Incident	Reactor Trip Functions	ESF Actuation Functions	Other Equipment	ESF Equipment
	Spectrum of rod cluster control assembly ejection accidents	Power range high flux, high positive flux rate, manual	NA	NA	NA
15.5	Increase in reactor coolant inventory				
	Inadvertent operation of ECCS during power operation	Lo pressurizer pressure, manual	NA	NA	NA
15.6	Decrease in reactor coolant inventory				
	Inadvertent opening of a pressurizer relief valve	Pressurizer low pressure, manual, OTΔT	Low pressurizer pressure	NA	Safety injection system
	Steam generator tube rupture	Pressurizer low pressure manual, OTΔT	Low pressurizer pressure	Service water system, steam generator safety valves, steam line isolation valves	Emergency core cooling auxiliary feedwater system, emergency power systems
	Loss of coolant accident from spectrum of postulated piping breaks within the system	Pressurizer low pressure manual, OTΔT	Low pressurizer pressure; Hi-1 containment pressure	Service water system, steam generator safety valves	Emergency core cooling; auxiliary feed-water system; containment heat removal system (Hi-3 containment pressure); emergency power system

NOTE:

* Performs no safety-related function.

TABLE 15.0-7

PARAMETERS AND ASSUMPTIONS USED FOR CALCULATING REACTOR CORE RADIONUCLIDE INVENTORY USING ORIGENS

A. Inputs for fuel, cladding and coolant material compositions

Reactor Core Thermal Power — 2918 MWt (1.006 times rated core power of 2900 MWt

Uranium mass per fuel assembly — 462.4 Kg/assembly

Uranium isotopic compositions*:

	U-234 (%)	U-235 (%)	U-236 (%)	U-238 (%)
Maximum Enrichment	0.0445	5.0	0.0230	94.933
Minimum Enrichment	0.0374	4.2	0.0193	95.743

* Core inventory calculations were performed using 4.2% and 5.0% average enrichments to represent bounding conditions. The maximum individual nuclide values are selected from these calculations for use in DBA analysis. See Table 15.0-7b.

Cladding material and density — ZIRLO - 6.425 gm/cm^3

Average coolant density in active core region — 45.46 lb/ft^3

Average boron concentration in reactor coolant during a fuel cycle

GWD/MTU	ppm
0.0	1879
0.150	1470
1.000	1491
2.000	1530
3.000	1544
4.000	1530
6.000	1435
8.000	1281
10.00	1105
12.00	913
14.00	711
16.00	505
18.00	301
EOC	10

Average temperature in fuel — 1100°F - 1260°F

Average temperature in cladding — 641°F

Average temperature in coolant — 566.2°F - 580°F

Table 15.0-7 (Cont)

PARAMETERS AND ASSUMPTIONS USED FOR CALCULATING
REACTOR CORE RADIONUCLIDE INVENTORY USING ORIGENS

B. Inputs for fuel cell geometry

Fuel cell type	Square cell (17x17)
Cell dimension	0.496 inches
Clad outer diameter	0.374 inches
Clad inner diameter	0.329 inches

C. Inputs for fuel assembly geometry

Number of fuel rods per assembly	264
Number of instrument tubes per assembly	1
Number of control rod guide tubes	24
Active fuel length	144 inches
Outside radius of control rod guide tube	0.237 inches
Inner radius of control rod guide tube	0.221 inches

D. Inputs for Fuel Irradiation (Equilibrium Fuel Cycle)

Number of fuel assemblies in core	157
Average power level per assembly	18.586 MW/Assembly
Irradiation period per fuel cycle	518 days
Refueling shutdown period between irradiation	30 days

E. Miscellaneous inputs

Data for computing light element weight per assembly:

Compositions:
Fe - 0.09 - 0.13%
Nb - 0.8 - 1.2%
Sn - 0.8 - 1.2%
Zr - 98.0%

Total weight of ZIRLO in core = 41,400 lbs

TABLE 15.0-7a

EQUILIBRIUM CORE INVENTORY BASED ON UPRATE CORE POWER OF 2918 MWT

ISOTOPE	PARENT RELATIONSHIP	PARENT ISOTOPE	ACTIVITY (CURIES)	ISOTOPE	PARENT RELATIONSHIP	PARENT ISOTOPE	ACTIVITY (CURIES)
AG-111			5.05E+06	PU-239			2.86E+04
	PARENT:	AG-111M	5.06E+06		PARENT:	NP-239	1.66E+09
	GRAND PARENT:	PD-111	5.04E+06		GRAND PARENT:	U-239	1.66E+09
AG-112			2.28E+06	PU-240			3.87E+04
	PARENT:	PD-112	2.27E+06		PARENT:	NP-240	4.32E+06
AM-241			1.17E+04	PU-241			1.13E+07
	PARENT:	PU-241	1.13E+07	PU-242			2.01E+02
BA-137M			9.35E+06		PARENT:	AM-242	7.04E+06
	PARENT:	CS-137	9.81E+06	RB-86			1.69E+05
	GRAND PARENT:	XE-137	1.46E+08	RB-88			5.57E+07
BA-139			1.41E+08		PARENT:	KR-88	5.43E+07
	PARENT:	CS-139	1.37E+08		GRAND PARENT:	BR-88	2.99E+07
	GRAND PARENT:	XE-139	1.01E+08	RB-89			7.26E+07
BA-140			1.42E+08		PARENT:	KR-89	6.75E+07
	PARENT:	CS-140	1.23E+08		GRAND PARENT:	BR-89	2.08E+07
	GRAND PARENT:	XE-140	7.06E+07	RB-90			6.69E+07
BA-142			1.21E+08		PARENT:	KR-90	7.24E+07
	PARENT:	CS-142	5.48E+07		GRAND PARENT:	BR-90	1.13E+07
	GRAND PARENT:	XE-142	1.07E+07		2ND PARENT:	RB-90M	2.11E+07
BR-82			3.02E+05	RB-90M			2.11E+07
	PARENT:	BR-82M	2.62E+05		PARENT:	KR-90	7.24E+07
BR-83			9.37E+06		GRAND PARENT:	BR-90	1.13E+07
	PARENT:	SE-83M	4.69E+06	RH-103M			1.26E+08
	2ND PARENT:	SE-83	4.42E+06		PARENT:	RU-103	1.26E+08
BR-85			1.95E+07	RH-105			8.16E+07

Table 15.0-7a (Cont)

EQUILIBRIUM CORE INVENTORY BASED ON UPRATE CORE POWER OF 2918 MWT

ISOTOPE	PARENT RELATIONSHIP	PARENT ISOTOPE	ACTIVITY (CURIES)	ISOTOPE	PARENT RELATIONSHIP	PARENT ISOTOPE	ACTIVITY (CURIES)
CE-141			1.30E+08		PARENT:	RH-105M	2.53E+07
	PARENT:	LA-141	1.29E+08		GRAND PARENT:	RU-105	8.90E+07
	GRAND PARENT:	BA-141	1.28E+08		2ND PARENT:	RU-105	8.90E+07
CE-143			1.21E+08	RH-105M			2.53E+07
	PARENT:	LA-143	1.20E+08		PARENT:	RU-105	8.90E+07
CE-144			9.82E+07		GRAND PARENT:	TC-105	8.76E+07
CM-242			4.22E+06	RH-106			5.13E+07
	PARENT:	AM-242	7.04E+06		PARENT:	RU-106	4.63E+07
CM-244			5.97E+05	RU-103			1.26E+08
	PARENT:	AM-244	1.89E+07		GRAND PARENT:	MO-103	1.24E+08
CS-134			1.57E+07	RU-106			4.63E+07
	PARENT:	CS-134M	3.69E+06		2ND PARENT:	SN-125M	1.20E+06
CS-134M			3.69E+06	SB-127			6.92E+06
CS-135M			4.39E+06		PARENT:	SN-127	2.78E+06
CS-136			4.97E+06		2ND PARENT:	SN-127M	3.76E+06
CS-137			9.81E+06	SB-129			2.52E+07
	PARENT:	XE-137	1.46E+08		PARENT:	SN-129	9.90E+06
	GRAND PARENT:	I-137	7.47E+07		2ND PARENT:	SN-129M	9.29E+06
CS-138			1.48E+08	SB-130			8.37E+06
	PARENT:	XE-138	1.36E+08	SB-130M			3.47E+07
	GRAND PARENT:	I-138	3.80E+07		PARENT:	SN-130	2.61E+07
CS-139			1.37E+08	SB-131			6.09E+07
	PARENT:	XE-139	1.01E+08		PARENT:	SN-131	2.24E+07
	GRAND PARENT:	I-139	1.83E+07	SB-132			3.67E+07
CS-140			1.23E+08		PARENT:	SN-132	1.81E+07
	PARENT:	XE-140	7.06E+07	SB-133			5.08E+07

Table 15.0-7a (Cont)

EQUILIBRIUM CORE INVENTORY BASED ON UPRATE CORE POWER OF 2918 MWT

ISOTOPE	PARENT RELATIONSHIP	PARENT ISOTOPE	ACTIVITY (CURIES)	ISOTOPE	PARENT RELATIONSHIP	PARENT ISOTOPE	ACTIVITY (CURIES)
	GRAND PARENT:	I-140	4.81E+06	SE-83			4.42E+06
	PARENT:	SM-155	3.11E+06	SM-153			4.02E+07
EU-156			2.29E+07		PARENT:	PM-153	7.37E+06
	PARENT:	SM-156	1.93E+06	SN-127			2.78E+06
EU-157			2.41E+06	SR-89			7.61E+07
H-3			4.36E+04		PARENT:	RB-89	7.26E+07
I-129			2.86E+00		GRAND PARENT:	KR-89	6.75E+07
	PARENT:	TE-129	2.40E+07	SR-90			7.21E+06
	GRAND PARENT:	TE-129M	4.87E+06		PARENT:	RB-90	6.69E+07
	2ND PARENT:	TE-129M	4.87E+06		GRAND PARENT:	KR-90	7.24E+07
I-130			2.07E+06		2ND PARENT:	RB-90M	2.11E+07
	PARENT:	I-130M	1.10E+06	SR-91			9.50E+07
I-131			7.78E+07		PARENT:	RB-91	8.85E+07
	PARENT:	TE-131	6.54E+07		GRAND PARENT:	KR-91	4.98E+07
	GRAND PARENT:	TE-131M	1.57E+07	SR-92			1.01E+08
	2ND PARENT:	TE-131M	1.57E+07		PARENT:	RB-92	7.83E+07
I-132			1.14E+08		GRAND PARENT:	KR-92	2.66E+07
	PARENT:	TE-132	1.12E+08	SR-93			1.14E+08
	GRAND PARENT:	SB-132	3.67E+07		GRAND PARENT:	KR-93	9.04E+06
I-133			1.60E+08	SR-94			1.14E+08
	PARENT:	TE-133	8.66E+07		GRAND PARENT:	KR-94	4.18E+06
	GRAND PARENT:	SB-133	5.08E+07	TC-99M			1.29E+08
	2ND PARENT:	TE-133M	7.12E+07		PARENT:	MO-99	1.45E+08
I-134			1.77E+08		GRAND PARENT:	NB-99	8.50E+07
	PARENT:	TE-134	1.41E+08	TC-101			1.33E+08
	2ND PARENT:	I-134M	1.59E+07		PARENT:	MO-101	1.33E+08

Table 15.0-7a (Cont)

EQUILIBRIUM CORE INVENTORY BASED ON UPRATE CORE POWER OF 2918 MWT

ISOTOPE	PARENT RELATIONSHIP	PARENT ISOTOPE	ACTIVITY (CURIES)	ISOTOPE	PARENT RELATIONSHIP	PARENT ISOTOPE	ACTIVITY (CURIES)
I-135			1.52E+08	TC-104			1.05E+08
I-136			6.99E+07		PARENT:	MO-104	9.99E+07
KR-83M			9.46E+06	TC-105			8.76E+07
	PARENT:	BR-83	9.37E+06		PARENT:	MO-105	7.38E+07
	GRAND PARENT:	SE-83M	4.69E+06	TE-127			6.81E+06
KR-85			8.27E+05		PARENT:	TE-127M	1.13E+06
	PARENT:	KR-85M	1.95E+07		GRAND PARENT:	SB-127	6.92E+06
	GRAND PARENT:	BR-85	1.95E+07		2ND PARENT:	SB-127	6.92E+06
	2ND PARENT:	BR-85	1.95E+07	TE-127M			1.13E+06
KR-85M			1.95E+07		PARENT:	SB-127	6.92E+06
	PARENT:	BR-85	1.95E+07		GRAND PARENT:	SN-127	2.78E+06
KR-87			3.91E+07	TE-129			2.40E+07
	PARENT:	BR-87	3.09E+07		PARENT:	TE-129M	4.87E+06
KR-88			5.43E+07		GRAND PARENT:	SB-129	2.52E+07
	PARENT:	BR-88	2.99E+07		2ND PARENT:	SB-129	2.52E+07
KR-89			6.75E+07	TE-129M			4.87E+06
	PARENT:	BR-89	2.08E+07		PARENT:	SB-129	2.52E+07
KR-90			7.24E+07		GRAND PARENT:	SN-129	9.90E+06
	PARENT:	BR-90	1.13E+07	TE-131			6.54E+07
LA-140			1.46E+08		PARENT:	SB-131	6.09E+07
	PARENT:	BA-140	1.42E+08		GRAND PARENT:	SN-131	2.24E+07
	GRAND PARENT:	CS-140	1.23E+08		2ND PARENT:	TE-131M	1.57E+07
LA-141			1.29E+08	TE-131M			1.57E+07
	PARENT:	BA-141	1.28E+08		PARENT:	SB-131	6.09E+07
LA-142			1.26E+08		GRAND PARENT:	SN-131	2.24E+07
	PARENT:	BA-142	1.21E+08	TE-132			1.12E+08

Table 15.0-7a (Cont)

EQUILIBRIUM CORE INVENTORY BASED ON UPRATE CORE POWER OF 2918 MWT

ISOTOPE	PARENT RELATIONSHIP	PARENT ISOTOPE	ACTIVITY (CURIES)	ISOTOPE	PARENT RELATIONSHIP	PARENT ISOTOPE	ACTIVITY (CURIES)
	GRAND PARENT:	CS-142	5.48E+07		PARENT:	SB-132	3.67E+07
LA-143			1.20E+08		GRAND PARENT:	SN-132	1.81E+07
MO-99			1.45E+08	TE-133			8.66E+07
	PARENT:	NB-99M	5.82E+07		PARENT:	TE-133M	7.12E+07
	2ND PARENT:	NB-99	8.50E+07		GRAND PARENT:	SB-133	5.08E+07
MO-101			1.33E+08		2ND PARENT:	SB-133	5.08E+07
NB-95			1.34E+08	TE-133M			7.12E+07
	PARENT:	ZR-95	1.33E+08		PARENT:	SB-133	5.08E+07
	GRAND PARENT:	Y-95	1.28E+08	TE-134			1.41E+08
	2ND PARENT:	NB-95M	1.52E+06	XE-131M			1.08E+06
NB-95M			1.52E+06		PARENT:	I-131	7.78E+07
	PARENT:	ZR-95	1.33E+08		GRAND PARENT:	TE-131M	1.57E+07
	GRAND PARENT:	Y-95	1.28E+08	XE-133			1.60E+08
NB-97			1.27E+08		PARENT:	I-133	1.60E+08
	PARENT:	NB-97M	1.19E+08		GRAND PARENT:	TE-133M	7.12E+07
	GRAND PARENT:	ZR-97	1.26E+08		2ND PARENT:	XE-133M	5.05E+06
	2ND PARENT:	ZR-97	1.26E+08	XE-133M			5.05E+06
NB-97M			1.19E+08		PARENT:	I-133	1.60E+08
	PARENT:	ZR-97	1.26E+08		GRAND PARENT:	TE-133M	7.12E+07
ND-147			5.22E+07	XE-135			4.84E+07
	PARENT:	PR-147	5.18E+07		PARENT:	I-135	1.52E+08
	GRAND PARENT:	CE-147	4.92E+07		2ND PARENT:	XE-135M	3.36E+07
NP-239			1.66E+09	XE-135M			3.36E+07
	GRAND PARENT:	PU-243	4.23E+07		PARENT:	I-135	1.52E+08
	2ND PARENT:	U-239	1.66E+09	XE-137			1.46E+08
PD-109			3.26E+07		PARENT:	I-137	7.47E+07

Table 15.0-7a (Cont)

EQUILIBRIUM CORE INVENTORY BASED ON UPRATE CORE POWER OF 2918 MWT

ISOTOPE	PARENT RELATIONSHIP	PARENT ISOTOPE	ACTIVITY (CURIES)	ISOTOPE	PARENT RELATIONSHIP	PARENT ISOTOPE	ACTIVITY (CURIES)
PM-147			1.38E+07	XE-138			1.36E+08
	PARENT:	ND-147	5.22E+07		PARENT:	I-138	3.80E+07
	GRAND PARENT:	PR-147	5.18E+07	Y-90			7.49E+06
PM-148			1.41E+07		PARENT:	SR-90	7.21E+06
	PARENT:	PM-148M	2.37E+06		GRAND PARENT:	RB-90	6.69E+07
PM-148M			2.37E+06	Y-91			9.87E+07
PM-149			4.82E+07		PARENT:	SR-91	9.50E+07
	PARENT:	ND-149	3.02E+07		GRAND PARENT:	RB-91	8.85E+07
	GRAND PARENT:	PR-149	2.80E+07		2ND PARENT:	Y-91M	5.51E+07
PM-151			1.60E+07	Y-91M			5.51E+07
	PARENT:	ND-151	1.58E+07		PARENT:	SR-91	9.50E+07
PR-142			5.57E+06		GRAND PARENT:	RB-91	8.85E+07
PR-143			1.18E+08	Y-92			1.02E+08
	PARENT:	CE-143	1.21E+08		PARENT:	SR-92	1.01E+08
	GRAND PARENT:	LA-143	1.20E+08		GRAND PARENT:	RB-92	7.83E+07
PR-144			9.89E+07	Y-93			7.73E+07
	PARENT:	CE-144	9.82E+07		PARENT:	SR-93	1.14E+08
	2ND PARENT:	PR-144M	1.38E+06	Y-94			1.23E+08
PU-238			3.40E+05		PARENT:	SR-94	1.14E+08
	2ND PARENT:	NP-238	3.98E+07	Y-95			1.28E+08
				ZR-95			1.33E+08
					PARENT:	Y-95	1.28E+08
				ZR-97			1.26E+08

TABLE 15.0-7b

FRACTION OF CORE INVENTORY IN THE FUEL GAP
DESIGN BASIS ACCIDENT ANALYSES

Nuclide Group	Regulatory Guide 1.183 Gap Fraction for Non-LOCA Events
I-131	0.08
Kr-85	0.10
Other Noble Gases	0.05
Other Halogens	0.05
Alkali Metals	0.12

NOTES:

- In accordance with Regulatory Guide 1.183 the gap fraction associated with the Large Break LOCA is as follows:

  Noble gases: 5%
  Halogens: 5%
  Alkali Metals: 5%

- In accordance with Regulatory Guide 1.183 the gap fraction associated with the Control Rod Ejection accident is as follows:

  Noble gases: 10%
  Halogens: 10%

TABLE 15.0-8a

PARAMETERS AND ASSUMPTIONS USED FOR CALCULATING PRIMARY SYSTEM COOLANT AND SECONDARY SYSTEM COOLANT AND STEAM RADIOACTIVITY CONCENTRATIONS

1% failed fuel primary coolant activities given in Table 15.0-8b

Reactor Core Thermal Power Level	2918 MWt
Total primary to secondary leak rate (all SGs)	3750 lbm/day (450 gpd)
Weight of water in all SG	2.46E+05 lbm
Total steam generator blowdown flow	2.24E+04 lbm/hour
SG blowdown flash tank partition factor	Halogens - 0.05 Others - 0.05
Destination of steam generator blowdown vapor	Main condenser
Decontamination factors for the SG blowdown demineralizer	Cs, Rb - 10 Others - 100
Ratio of concentration in steam to that in water in the steam generator	Halogens - 0.01 Others - 0.005
Reactor operation time	21,600 hours
Bleed flow to model depletion by steam generator blowdown demineralizer	Cs/Rb - 38.25 gpm Other - 42.08 gpm
Steam flow rate	1.21E+07 lbm/hr
Expected design corrosion product concentrations in RCS (three times these values are used in the analysis)	Na-24 - 4.7E-2 µCi/g Cr-51 - 3.1E-3 µCi/g Mn-54 - 1.6E-3 µCi/g Fe-55 - 1.2E-3 µCi/g Fe-59 - 3.0E-4 µCi/g Co-58 - 4.6E-3 µCi/g Co-60 - 5.3E-4 µCi/g Zn-65 - 5.1E-4 µCi/g Np-239 - 2.2E-3 µCi/g

TABLE 15.0-8b

DESIGN PRIMARY AND SECONDARY COOLANT CONCENTRATIONS - 2918 MWt
(1% FUEL DEFECTS)

Nuclide	Reactor Coolant (μCi/gm)	Secondary Liquid (μCi/gm)	Secondary Steam (μCi/gm)
KR 83M	4.10E-01		6.29E-06
KR 85M	1.42E+00		1.84E-05
KR 85	1.25E+02		1.62E-03
KR 87	9.48E-01		1.23E-05
KR 88	2.65E+00		3.44E-05
KR 89	7.63E-02		9.90E-07
XE131M	5.10E+00		6.62E-05
XE133M	4.20E+00		5.47E-05
XE133	3.11E+02		4.04E-03
XE135M	9.56E-01		8.25E-05
XE135	9.64E+00		1.36E-04
XE137	1.98E-01		2.57E-06
XE138	6.70E-01		8.69E-06
BR83	7.55E-02	1.30E-04	1.30E-06
BR84	3.73E-02	1.74E-05	1.74E-07
BR85	3.94E-03	1.76E-07	1.76E-09
BR87	2.04E-03	2.97E-08	2.97E-10
I129	1.11E-07	8.25E-10	8.25E-12
I130	4.62E-02	2.09E-04	2.09E-06
I131	2.89E+00	2.06E-02	2.06E-04
I132	1.13E+00	3.46E-03	3.46E-05
I133	4.32E+00	2.33E-02	2.33E-04
I134	6.32E-01	4.84E-04	4.84E-06
I135	2.48E+00	8.39E-03	8.39E-05
I136	6.84E-03	1.48E-07	1.48E-09
SE81	6.09E-07	1.70E-10	8.48E-13
SE83	8.34E-07	2.81E-10	1.40E-12
SE84	4.87E-07	2.50E-11	1.25E-13
RB86	5.45E-02	4.37E-04	2.18E-06
RB88	2.75E+00	7.41E-04	3.71E-06
RB89	1.57E-01	3.63E-05	1.82E-07
RB90	1.22E-02	4.85E-07	2.42E-09
RB91	6.04E-03	9.16E-08	4.58E-10
RB92	4.28E-04	4.99E-10	2.50E-12
SR89	3.49E-03	2.60E-05	1.30E-07
SR90	2.16E-04	1.60E-06	8.02E-09
SR91	1.45E-03	5.87E-06	2.94E-08
SR92	1.03E-03	1.95E-06	9.76E-09
SR93	4.60E-05	5.32E-09	2.66E-11
SR94	7.82E-06	1.54E-10	7.69E-13

TABLE 15.0-8b

DESIGN PRIMARY AND SECONDARY COOLANT CONCENTRATIONS - 2918 MWt
(1% FUEL DEFECTS)

Nuclide	Reactor Coolant (μCi/gm)	Secondary Liquid (μCi/gm)	Secondary Steam (μCi/gm)
Y90	5.94E-05	5.69E-07	2.85E-09
Y91M	7.91E-04	3.66E-06	1.83E-08
Y91	4.78E-04	3.57E-06	1.78E-08
Y92	8.84E-04	3.38E-06	1.69E-08
Y93	3.13E-04	1.31E-06	6.53E-09
Y94	2.77E-05	8.11E-09	4.05E-11
Y95	1.15E-05	1.85E-09	9.25E-12
ZR95	6.32E-04	4.67E-06	2.34E-08
ZR97	3.91E-04	1.98E-06	9.89E-09
NB95M	7.26E-06	5.22E-08	2.61E-10
NB95	6.41E-04	4.76E-06	2.38E-08
NB97M	3.70E-04	1.87E-06	9.37E-09
NB97	4.18E-04	2.13E-06	1.06E-08
MO99	7.62E-01	5.05E-03	2.53E-05
MO101	2.07E-02	4.60E-06	2.30E-08
MO102	1.57E-02	2.67E-06	1.33E-08
MO105	7.15E-04	1.01E-08	5.04E-11
TC99M	4.09E-01	3.83E-03	1.91E-05
TC101	2.02E-02	8.82E-06	4.41E-08
TC102	1.57E-02	2.69E-06	1.34E-08
TC105	7.63E-04	1.04E-07	5.21E-10
RU103	5.97E-04	4.40E-06	2.20E-08
RU105	1.39E-04	4.38E-07	2.19E-09
RU106	2.21E-04	1.64E-06	8.20E-09
RU107	1.92E-06	1.25E-10	6.26E-13
RH103M	5.97E-04	4.01E-06	2.01E-08
RH105M	3.97E-05	1.21E-07	6.06E-10
RH105	3.69E-04	2.31E-06	1.15E-08
RH106	2.45E-04	1.64E-06	8.20E-09
RH107	1.15E-05	3.86E-09	1.93E-11
SN127	2.49E-06	3.88E-09	1.94E-11
SN128	5.11E-06	4.19E-09	2.10E-11
SN130	8.41E-07	4.85E-11	2.43E-13
SB127	3.01E-05	2.06E-07	1.03E-09
SB128	2.80E-06	1.11E-08	5.54E-11
SB129	3.87E-05	1.02E-07	5.08E-10
SB130	2.72E-06	1.57E-09	7.85E-12
SB131	1.17E-05	4.02E-09	2.01E-11
SB132	8.92E-07	3.88E-11	1.94E-13
SB133	1.06E-06	3.97E-11	1.98E-13

TABLE 15.0-8b

DESIGN PRIMARY AND SECONDARY COOLANT CONCENTRATIONS - 2918 MWt
(1% FUEL DEFECTS)

Nuclide	Reactor Coolant (μCi/gm)	Secondary Liquid (μCi/gm)	Secondary Steam (μCi/gm)
TE125M	4.42E-04	3.27E-06	1.63E-08
TE127M	3.36E-03	2.49E-05	1.24E-07
TE127	1.17E-02	5.86E-05	2.93E-07
TE129M	1.44E-02	1.06E-04	5.30E-07
TE129	1.43E-02	7.37E-05	3.69E-07
TE131M	3.60E-02	2.12E-04	1.06E-06
TE131	1.30E-02	4.91E-05	2.45E-07
TE132	3.00E-01	2.02E-03	1.01E-05
TE133M	1.95E-02	1.51E-05	7.56E-08
TE133	8.76E-03	3.58E-06	1.79E-08
TE134	2.99E-02	1.80E-05	9.00E-08
CS134M	4.41E-02	9.01E-05	4.50E-07
CS134	6.05E+00	4.94E-02	2.47E-04
CS136	1.50E+00	1.19E-02	5.97E-05
CS137	3.79E+00	3.10E-02	1.55E-04
CS138	1.03E+00	4.89E-04	2.44E-06
CS139	9.11E-02	1.32E-05	6.58E-08
CS140	9.25E-03	1.53E-07	7.66E-10
CS142	1.10E-04	4.88E-11	2.44E-13
BA137M	3.59E+00	2.93E-02	1.47E-04
BA139	7.97E-02	9.89E-05	4.94E-07
BA140	4.10E-03	2.97E-05	1.49E-07
BA141	1.24E-04	3.42E-08	1.71E-10
BA142	1.71E-04	2.81E-08	1.40E-10
LA140	1.41E-03	1.36E-05	6.82E-08
LA141	2.72E-04	6.94E-07	3.47E-09
LA142	2.52E-04	3.29E-07	1.64E-09
LA143	1.43E-05	3.05E-09	1.52E-11
CE141	6.18E-04	4.55E-06	2.28E-08
CE143	4.56E-04	2.73E-06	1.37E-08
CE144	4.69E-04	3.48E-06	1.74E-08
CE145	2.14E-06	9.99E-11	4.99E-13
CE146	7.89E-06	1.70E-09	8.51E-12
PR143	5.75E-04	4.24E-06	2.12E-08
PR144	4.72E-04	3.48E-06	1.74E-08
PR145	1.57E-04	5.02E-07	2.51E-09
PR146	2.09E-05	9.16E-09	4.58E-11
ND147	2.41E-04	1.74E-06	8.69E-09
ND149	2.31E-05	3.09E-08	1.54E-10
ND151	1.67E-06	3.16E-10	1.58E-12

TABLE 15.0-8b

DESIGN PRIMARY AND SECONDARY COOLANT CONCENTRATIONS - 2918 MWt
(1% FUEL DEFECTS)

Nuclide	Reactor Coolant (μCi/gm)	Secondary Liquid (μCi/gm)	Secondary Steam (μCi/gm)
PM147	1.01E-04	7.51E-07	3.75E-09
PM149	2.02E-04	1.31E-06	6.55E-09
PM151	5.83E-05	3.39E-07	1.69E-09
SM151	4.93E-07	3.67E-09	1.83E-11
SM153	1.61E-04	1.02E-06	5.12E-09
NA 24	1.41E-01	6.87E-04	3.43E-06
CR 51	9.30E-03	6.83E-05	3.41E-07
MN 54	4.80E-03	3.56E-05	1.78E-07
FE 55	3.60E-03	2.67E-05	1.34E-07
FE 59	9.00E-04	6.64E-06	3.32E-08
CO 58	1.38E-02	1.02E-04	5.10E-07
CO 60	1.59E-03	1.18E-05	5.91E-08
ZN 65	1.53E-03	1.14E-05	5.68E-08
NP 239	6.60E-03	4.30E-05	2.15E-07
H3	3.50E+00	3.50E-03	3.50E-03

NOTE:

All noble gas activity is assumed to leak directly from the primary coolant into the steam phase.

TABLE 15.0-8c

BVPS-2 PRIMARY AND SECONDARY COOLANT TECHNICAL SPECIFICATION IODINE AND NOBLE GAS CONCENTRATIONS BASED ON THE UPRATE CORE POWER OF 2918 MWT

Nuclide	Primary Coolant (µCi/gm)	Secondary Coolant (µCi/gm)
I-131	2.74E-01	8.33E-02
I-132	1.08E-01	1.40E-02
I-133	4.10E-01	9.39E-02
I-134	6.00E-02	1.95E-03
I-135	2.36E-01	3.39E-02
Kr-83m	3.89E-02	--
Kr-85m	1.35E-01	--
Kr-85	1.18E+01	--
Kr-87	9.00E-02	--
Kr-88	2.52E-01	--
Xe-131m	4.84E-01	--
Xe-133m	3.99E-01	--
Xe-133	2.95E+01	--
Xe-135m	9.09E-02	--
Xe-135	9.16E-01	--

NOTES:

1. Technical Specification primary coolant concentrations correspond to 0.35 µCi/gm I-131 dose equivalent.
2. Technical Specification secondary liquid concentrations correspond to 0.1 µCi/gm I-131 dose equivalent.

TABLE 15.0-9

PRE-ACCIDENT IODINE SPIKE
PRIMARY COOLANT IODINE CONCENTRATIONS

Nuclide	Primary Coolant Concentration (μCi/gm)
I-131	1.64E+01
I-132	6.46E+00
I-133	2.46E+01
I-134	3.60E+00
I-135	1.41E+01

TABLE 15.0-10

IODINE RELEASE RATES
INTO REACTOR COOLANT DUE TO
A CONCURRENT IODINE SPIKE

Nuclide	Equilibrium Iodine Appearance Rates (μCI/sec)
I-131	2.53E+03
I-132	2.66E+03
I-133	4.42E+03
I-134	3.00E+03
I-135	3.41E+03

NOTES:

1. Per Regulatory Guide 1.183, the equilibrium iodine activity release rate at the Technical Specification concentration is assumed to be increased by a factor of 335 (for the Steam Generator Tube Rupture) and a factor of 500 (for the Main Steam Line Break).

2. Per current licensing basis, the equilibrium iodine activity release rate at the Technical Specification concentration is assumed to be increased by a factor of 500 for the small line break outside containment.

TABLE 15.0-10a

PARAMETERS AND ASSUMPTIONS AND MODEL USED
FOR CALCULATING IODINE RELEASE RATES INTO REACTOR COOLANT
DUE TO A CONCURRENT IODINE SPIKE

Thyroid dose conversion factors	Nuclide	mrem/μCi
	I-131	1.08E+03
	I-132	6.44E+00
	I-133	1.80E+02
	I-134	1.07E+00
	I-135	3.13E+01
Nuclide decay constants (λ_r)	Nuclide	$second^{-1}$
	I-131	9.9783E-07
	I-132	8.3713E-05
	I-133	9.2568E-06
	I-134	2.1963E-04
	I-135	2.9129E-05
Reactor coolant system leakage (L)	Technical Specification maximum allowable values: Identified 10 gpm; Unidentified 1 gpm	
Reactor coolant system mass (M)	4.11E+05 lbm	
Letdown purification removal Efficiency (E)	1	
Letdown purification flow rate (F)	135 gpm	
Technical Specification equilibrium concentrations (EQ)	Table 15.0-8c	
Formula for iodine loss constant	$\lambda_{total} = (F*E/M) + (L/M) + \lambda_r$	
Concurrent iodine spike release rate (RR)	$RR = EQ * M * \lambda_{total}$	

NOTES:

Formulas for iodine release rates from EPRI Report, "Review of Iodine Spike Data from PWR Power Plants in Relation to SGTR with MSLB, TR-103680)"

TABLE 15.0-11

ACCIDENT SITE BOUNDARY ATMOSPHERIC DISPERSION FACTORS

Exclusion Area Boundary (547 m, NW) Averaging Period				
Release Point	0-2 hr	-	-	-
BVPS-1 Release Points	1.04E-3	-	-	-
BVPS-2 Release Points	1.25E-3	-	-	-

Low Population Zone (5,794 m, NW) Averaging Period				
Release Point	0-8 hr	8-24 hr	1-4 day	4-30 day
BVPS-1 and BVPS-2 Release Points	6.04E-5	4.33E-5	2.10E-5	7.44E-6

<u>NOTE</u>:

All values are in X/Q (x 10^{-3} sec/m^3)

TABLE 15.0-12

POTENTIAL DOSES AT THE EXCLUSION AREA BOUNDARY AND LOW POPULATION ZONE
DUE TO POSTULATED ACCIDENTS
(TEDE)

ACCIDENT	EAB Dose (rem)[1,3]	LPZ Dose (rem)[2]	Regulatory Limit (rem)
Loss of Coolant Accident (LOCA)	16.5	3.0	25
Control Rod Ejection Accident (CREA)[4]	3.1	1.5	6.3
Main Steam Line Break (MSLB)[5][7]	0.4	0.1	25(PIS)
	2.5	0.7	2.5(CIS)
Steam Generator Tube Rupture (SGTR)[7]	1.3	0.07	25(PIS)
	0.68	0.05	2.5(CIS)
Locked Rotor Accident (LRA)	2	0.33	2.5
Loss of AC Power (LACP)	(Note 6)	(Note 6)	2.5
Fuel Handling Accident (FHA)	2.43	0.12	6.3
Small Line Break Outside Containment (SLBOC)	0.23	0.012	2.5

NOTES:

(1) EAB Doses are based on the worst 2-hour period following the onset of the event.
(2) LPZ Doses are based on the duration of the release.
(3) Except as noted, the maximum 2 hr dose for the EAB is based on the 0 to 2 hr period:
- LOCA: 0.5 to 2.5 hr
- MSLB (CIS): 4 to 6 hr
- LRA: 6 to 8 hr

(4) Dose values are based on the containment release scenario. The dose consequences based on the secondary side release scenario are 1 Rem (EAB) and 0.1 Rem (LPZ).
(5) Doses are based on the maximum allowable Accident Induced Leakage (2.1 gpm) into the affected SG.
(6) Dose from a postulated Loss of AC Power is bounded by the Locked Rotor Accident.
(7) PIS: Pre-accident iodine spike; CIS: Concurrent iodine spike.
(8) For WGSR, the reported dose is shown in Table 11.3.4-3.

TABLE 15.0-13

CONTROL ROOM DOSES (TEDE)
FROM DESIGN BASIS ACCIDENTS

	Control Room Operator	
Accident	Dose (rem)	Reg. Limit (rem)
Loss of Coolant Accident[1] (LOCA)	2.5 (0.61)	5
Control Rod Ejection Accident[2] (CREA)	1.3	5
Main Steam Line Break (MSLB)[3][5]	0.6	5
Steam Generator Tube Rupture (SGTR)[5]	0.32	5
Fuel Handling Accident (FHA)[6]	1.4	5
Locked Rotor Accident[6] (LRA)	2.2	5
Loss of AC Power[6] (LACP)	(Note 4)	5
Small Line Break Outside Containment[6] (SLBOC)	0.7	5

NOTES:

(1) Portion shown in parentheses for the LOCA represents that portion of the total dose that is the contribution of direct shine from contained sources/external cloud.

(2) Dose values are based on the containment release scenario. The dose consequences based on the secondary side release scenario is 0.06 Rem.

(3) Dose is based on the maximum allowable Accident Induced Leakage (2.1 gpm) into the affected SG.

(4) Dose from a postulated Loss of AC Power is bounded by the Locked Rotor Accident.

(5) The CR is purged for 30 minutes at 16,200 cfm following termination of the environmental releases and by:

- MSLB: Purge within 24 hrs
- SGTR: Purge within 8 hrs

(6) The following accidents do not take credit for CREVS operations: SGTR, LRA, LACP, SLBOC, and FHA.

(7) For WGSR, the reported dose is shown in Table 11.3.4-3.

TABLE 15.0-14

BVPS-1 ON-SITE ATMOSPHERIC DISPERSION FACTORS (SEC/M^3)

Release	Receptor	0-2 hr	2-8 hr	8-24 hr	1-4 d	4-30 d
U 1 Containment Edge	BVPS-1 CR Intake	7.48E-04	5.77E-04	2.53E-04	2.00E-04	1.78E-04
U 1 Containment Top	BVPS-1 CR Intake	8.16E-04	5.78E-04	2.27E-04	1.71E-04	1.47E-04
U 1 Ventilation Vent	BVPS-1 CR Intake	4.75E-03	3.66E-03	1.43E-03	1.02E-03	8.84E-04
U 1 RWST Vent	BVPS-1 CR Intake	7.34E-04	6.17E-04	2.54E-04	1.96E-04	1.57E-04
U 1 MS Relief Valves	BVPS-1 CR Intake	1.24E-03	9.94E-04	4.08E-04	3.03E-04	2.51E-04
U 1 MSL (break)/AEJ	BVPS-1 CR Intake	1.05E-02	7.72E-03	3.01E-03	2.14E-03	2.00E-03
U 1 Gaseous Waste Storage Vault	BVPS-1 CR Intake	1.40E-03	8.78E-04	3.16E-04	2.93E-04	2.62E-04
U 1 Containment Equipment Hatch	BVPS-1 CR Intake	6.25E-04	4.23E-04	1.76E-04	1.27E-04	1.11E-04
U 1 Cooling Tower	BVPS-1 CR Intake	1.19E-04	8.79E-05	3.41E-05	2.76E-05	2.09E-05
U 1 Containment Edge	BVPS-2 CR Intake	4.88E-04	4.07E-04	1.79E-04	1.41E-04	1.22E-04
U 1 Containment Top	BVPS-2 CR Intake	5.93E-04	4.63E-04	1.84E-04	1.34E-04	1.16E-04
U 1 Ventilation Vent	BVPS-2 CR Intake	2.00E-03	1.62E-03	6.76E-04	5.05E-04	4.06E-04
U 1 RWST Vent	BVPS-2 CR Intake	4.76E-04	4.10E-04	1.70E-04	1.33E-04	1.07E-04
U 1 MS Relief Valves	BVPS-2 CR Intake	7.46E-04	6.31E-04	2.62E-04	1.98E-04	1.62E-04
U 1 MSL (break)/AEJ	BVPS-2 CR Intake	4.24E-03	3.87E-03	1.69E-03	1.18E-03	1.06E-03
U 1 Gaseous Waste Storage Vault	BVPS-2 CR Intake	1.42E-03	8.19E-04	3.38E-04	2.78E-04	2.49E-04
U 1 Containment Equipment Hatch	BVPS-2 CR Intake	4.48E-04	3.33E-04	1.36E-04	1.02E-04	8.70E-05
U 1 Cooling Tower	BVPS-2 CR Intake	1.33E-04	9.49E-05	3.61E-05	2.87E-05	2.25E-05
U 1 Containment Edge	BVPS-2 Aux. Bldg. NW Corner	3.34E-04	2.85E-04	1.23E-04	9.62E-05	8.37E-05
U 1 Containment Top	BVPS-2 Aux. Bldg. NW Corner	4.37E-04	3.41E-04	1.39E-04	1.02E-04	8.79E-05
U 1 RWST Vent	BVPS-2 Aux. Bldg. NW Corner	3.23E-04	2.83E-04	1.18E-04	9.32E-05	7.52E-05
U 1 Cooling Tower	BVPS-2 Aux. Bldg. NW Corner	1.57E-04	1.12E-04	4.13E-05	3.35E-05	2.60E-05
U 1 Containment Edge	BVPS-1 Service Bldg.	1.90E-03	1.57E-03	4.54E-04	5.08E-04	4.55E-04

TABLE 15.0-14 (Contd)

BVPS-1 ON-SITE ATMOSPHERIC DISPERSION FACTORS (SEC/M^3)

Release	Receptor	0-2 hr	2-8 hr	8-24 hr	1-4 d	4-30 d
U 1 Containment Top	BVPS-1 Service Bldg.	1.64E-03	8.59E-04	3.35E-04	2.71E-04	2.29E-04
U 1 RWST Vent	BVPS-1 Service Bldg.	2.37E-03	1.88E-03	7.58E-04	5.71E-04	4.48E-04
U 1 Cooling Tower	BVPS-1 Service Bldg.	1.09E-04	8.10E-05	3.28E-05	2.65E-05	1.92E-05
U 1 Containment Edge	ERF Intake	4.53E-05	2.97E-05	1.41E-05	1.23E-05	1.09E-05
U 1 Containment Top	ERF Intake	4.57E-05	3.74E-05	1.50E-05	1.44E-05	1.23E-05
U 1 RWST Vent	ERF Intake	4.53E-05	2.87E-05	1.39E-05	1.21E-05	1.05E-05
U 1 Cooling Tower	ERF Intake	5.75E-05	4.97E-05	2.31E-05	1.80E-05	1.66E-05
U 1 Containment Edge	ERF Edge Closest to Cont.	4.70E-05	3.16E-05	1.54E-05	1.32E-05	1.14E-05
U 1 Containment Top	ERF Edge Closest to Cont.	5.00E-05	3.94E-05	1.62E-05	1.52E-05	1.30E-05
U 1 RWST Vent	ERF Edge Closest to Cont.	4.54E-05	3.14E-05	1.50E-05	1.29E-05	1.13E-05
U 1 Cooling Tower	ERF Edge Closest to Cont.	7.67E-05	6.28E-05	3.10E-05	2.36E-05	2.17E-05

Note: The Control Room In-leakage X/Q values can be represented by the Control Room air intake X/Q values. The higher values from among the Unit 1 and Unit 2 Control Room Intake X/Qs are conservatively used for this purpose.

TABLE 15.0-15

BVPS-2 ON-SITE ATMOSPHERIC DISPERSION FACTORS (SEC/M^3)

Release	Receptor	0-2 hr	2-8 hr	8-24 hr	1-4 d	4-30 d
U 2 Contain. Edge	BVPS-1 CR Intake	3.19E-04	2.38E-04	1.06E-04	8.08E-05	6.19E-05
U 2 Containment Top	BVPS-1 CR Intake	3.83E-04	3.10E-04	1.34E-04	9.83E-05	6.65E-05
U 2 Ventilation Vent	BVPS-1 CR Intake	5.32E-04	3.89E-04	1.75E-04	1.30E-04	9.02E-05
U 2 RWST Vent	BVPS-1 CR Intake	1.70E-04	1.30E-04	5.56E-05	4.40E-05	3.31E-05
U 2 MS Relief Valves	BVPS-1 CR Intake	3.33E-04	2.38E-04	1.09E-04	7.88E-05	5.66E-05
U 2 MSL (break)/AEJ	BVPS-1 CR Intake	6.21E-04	4.87E-04	2.30E-04	1.65E-04	1.10E-04
U 2 Gaseous Waste Storage Vault	BVPS-1 CR Intake	7.71E-04	4.90E-04	2.26E-04	1.76E-04	1.31E-04
U 2 Containment Equipment Hatch	BVPS-1 CR Intake	2.47E-04	1.69E-04	7.94E-05	6.05E-05	4.56E-05
U 2 Contain. Edge	BVPS-2 CR Intake	4.82E-04	3.59E-04	1.55E-04	1.21E-04	9.18E-05
U 2 Containment Top	BVPS-2 CR Intake	5.56E-04	4.45E-04	1.91E-04	1.39E-04	9.35E-05
U 2 Ventilation Vent	BVPS-2 CR Intake	9.39E-04	6.69E-04	3.08E-04	2.23E-04	1.54E-04
U 2 RWST Vent	BVPS-2 CR Intake	2.18E-04	1.58E-04	7.31E-05	5.53E-05	4.12E-05
U 2 MS Relief Valves	BVPS-2 CR Intake	5.01E-04	3.58E-04	1.61E-04	1.19E-04	8.32E-05
U 2 MSL (break)/AEJ	BVPS-2 CR Intake	1.03E-03	7.84E-04	3.57E-04	2.64E-04	1.86E-04
U 2 Gaseous Waste Storage Vault	BVPS-2 CR Intake	1.55E-03	9.04E-04	4.08E-04	3.30E-04	2.45E-04
U 2 Containment Equipment Hatch	BVPS-2 CR Intake	3.45E-04	2.23E-04	1.06E-04	8.29E-05	6.14E-05
U 2 Contain. Edge	BVPS-2 Aux. Bldg. NW Corner	9.12E-04	7.13E-04	3.05E-04	2.35E-04	1.79E-04
U 2 Containment Top	BVPS-2 Aux. Bldg. NW Corner	1.14E-03	8.87E-04	3.83E-04	2.74E-04	1.83E-04
U 2 RWST Vent	BVPS-2 Aux. Bldg. NW Corner	3.19E-04	2.25E-04	1.06E-04	7.95E-05	5.84E-05
U 2 Contain. Edge	BVPS-1 Service Bldg.	1.96E-04	1.54E-04	6.37E-05	5.05E-05	3.89E-05
U 2 Containment Top	BVPS-1 Service Bldg.	2.46E-04	2.07E-04	8.84E-05	6.56E-05	4.49E-05
U 2 RWST Vent	BVPS-1 Service Bldg.	1.24E-04	9.81E-05	4.10E-05	3.24E-05	2.51E-05
U 2 Contain. Edge	ERF Intake	6.02E-05	4.67E-05	2.22E-05	1.78E-05	1.59E-05
U 2 Containment Top	ERF Intake	6.16E-05	5.36E-05	2.42E-05	2.08E-05	1.81E-05

TABLE 15.0-15 (Contd)

BVPS-2 ON-SITE ATMOSPHERIC DISPERSION FACTORS (SEC/M^3)

Release	Receptor	0-2 hr	2-8 hr	8-24 hr	1-4 d	4-30 d
U 2 RWST Vent	ERF Intake	7.28E-05	6.58E-05	3.01E-05	2.31E-05	2.08E-05
U 2 Contain. Edge	ERF Edge Closest to Containment	6.72E-05	5.69E-05	2.65E-05	2.13E-05	1.89E-05
U 2 Containment Top	ERF Edge Closest to Containment	7.22E-05	6.43E-05	2.96E-05	2.48E-05	2.15E-05
U 2 RWST Vent	ERF Edge Closest to Containment	9.42E-05	8.37E-05	3.81E-05	2.97E-05	2.58E-05

Note: The Control Room In-leakage X/Q values can be represented by the Control Room air intake X/Q values. The higher values from among the Unit 1 and Unit 2 Control Room Intake X/Qs are conservatively used for this purpose.

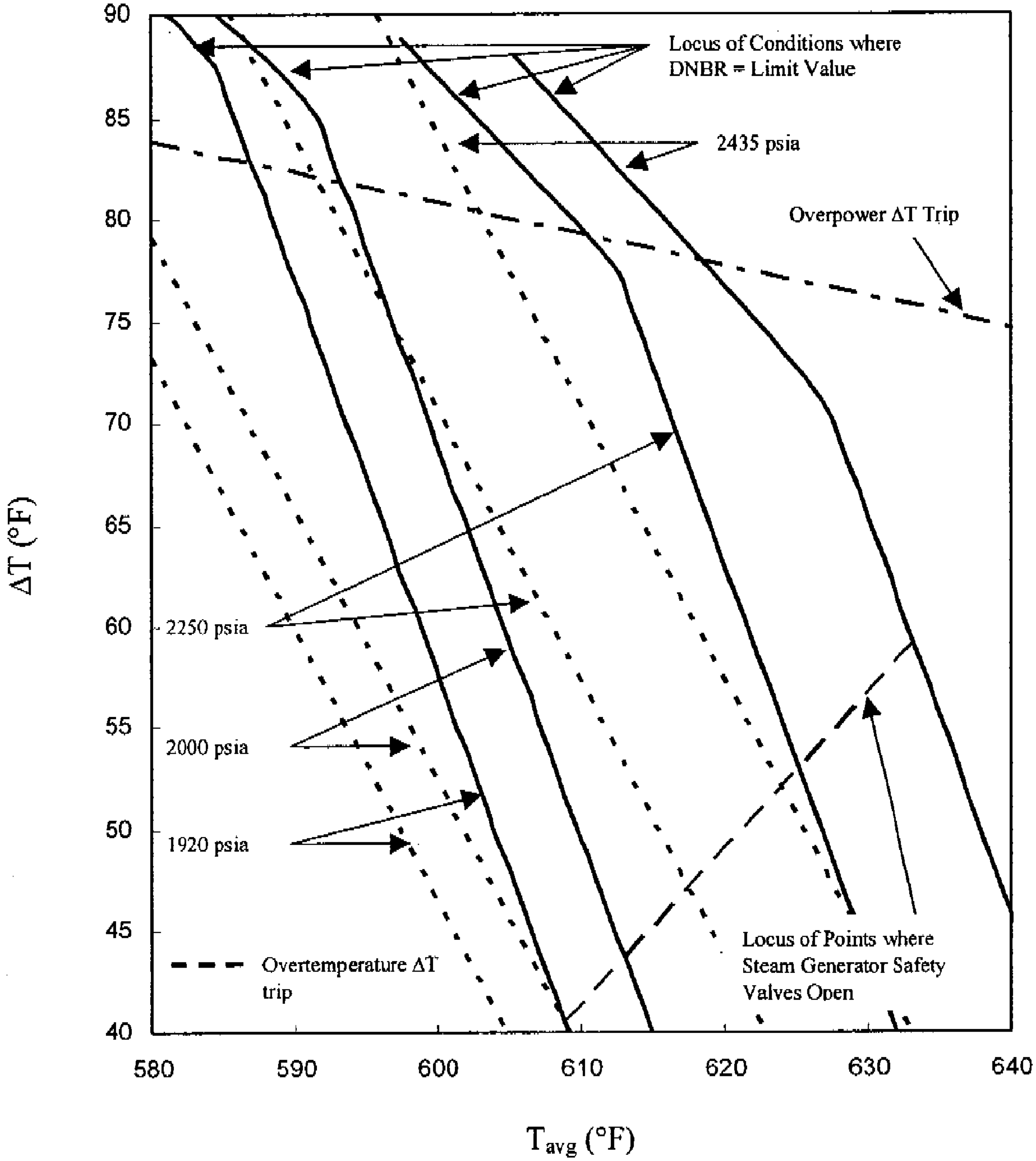
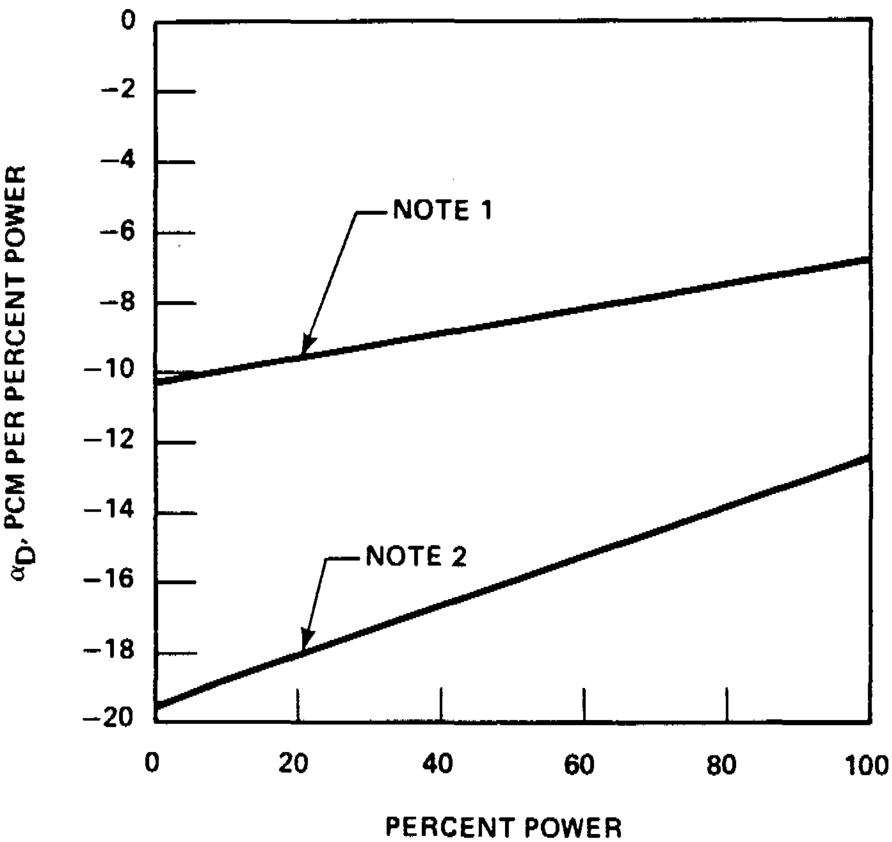


Figure 15.0-1

Illustration of Overtemperature and Overpower ΔT Protection

Beaver Valley Power Station Unit No. 2 Updated Final Safety Analysis Report



NOTE 1 – UPPER CURVE, LEAST NEGATIVE DOPPLER ONLY POWER DEFECT = –0.84% $\Delta\rho$ (0 TO 100% POWER)

NOTE 2 – LOWER CURVE, MOST NEGATIVE DOPPLER ONLY POWER DEFECT = –1.6% $\Delta\rho$ (0 TO 100% POWER)

FIGURE 15.0-2
DOPPLER POWER COEFFICIENT
USED IN ACCIDENT ANALYSIS
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

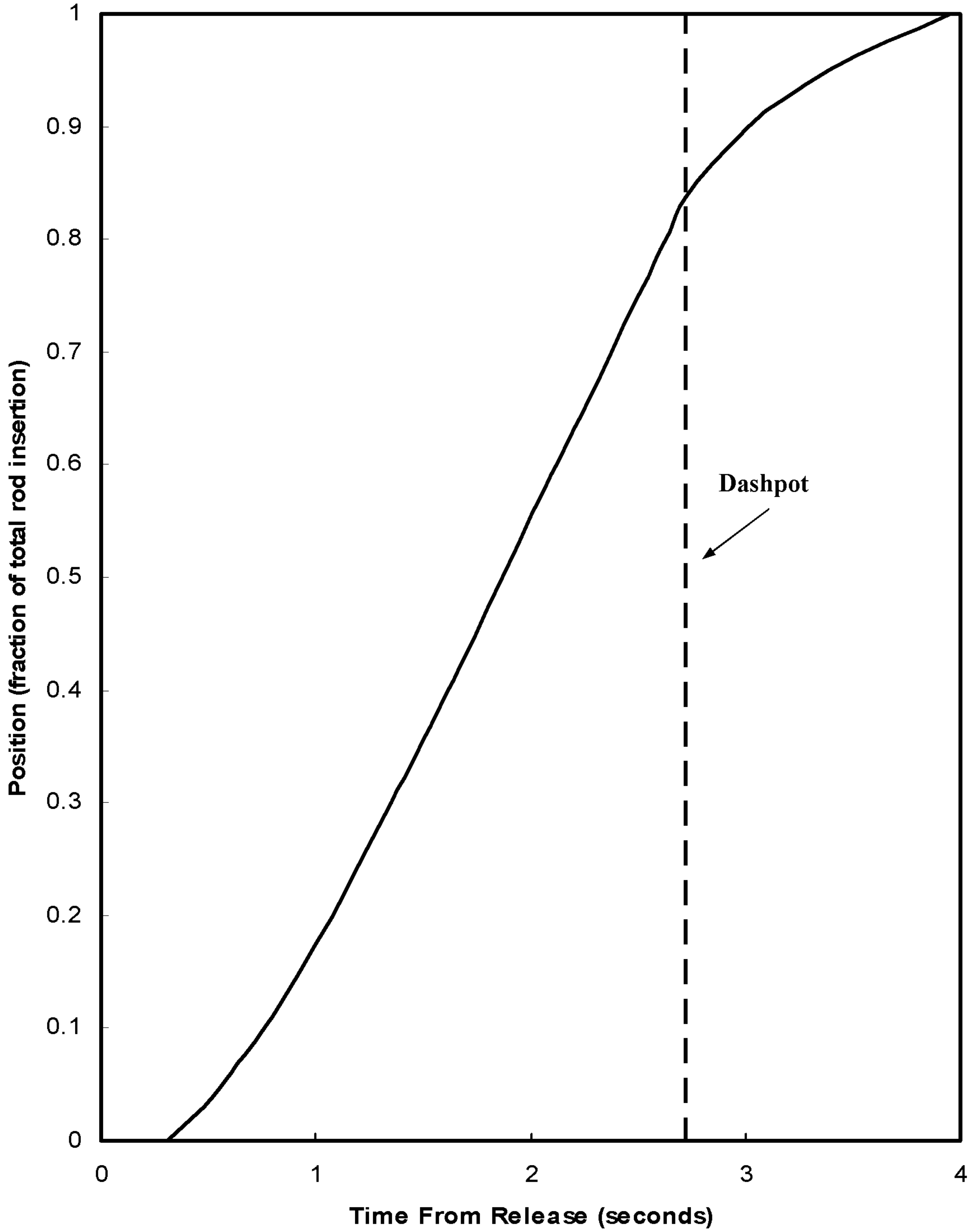


Figure 15.0-4

RCCA Position vs. Time

Beaver Valley Power Station Unit No. 2
Updated Final Safety Analysis Report

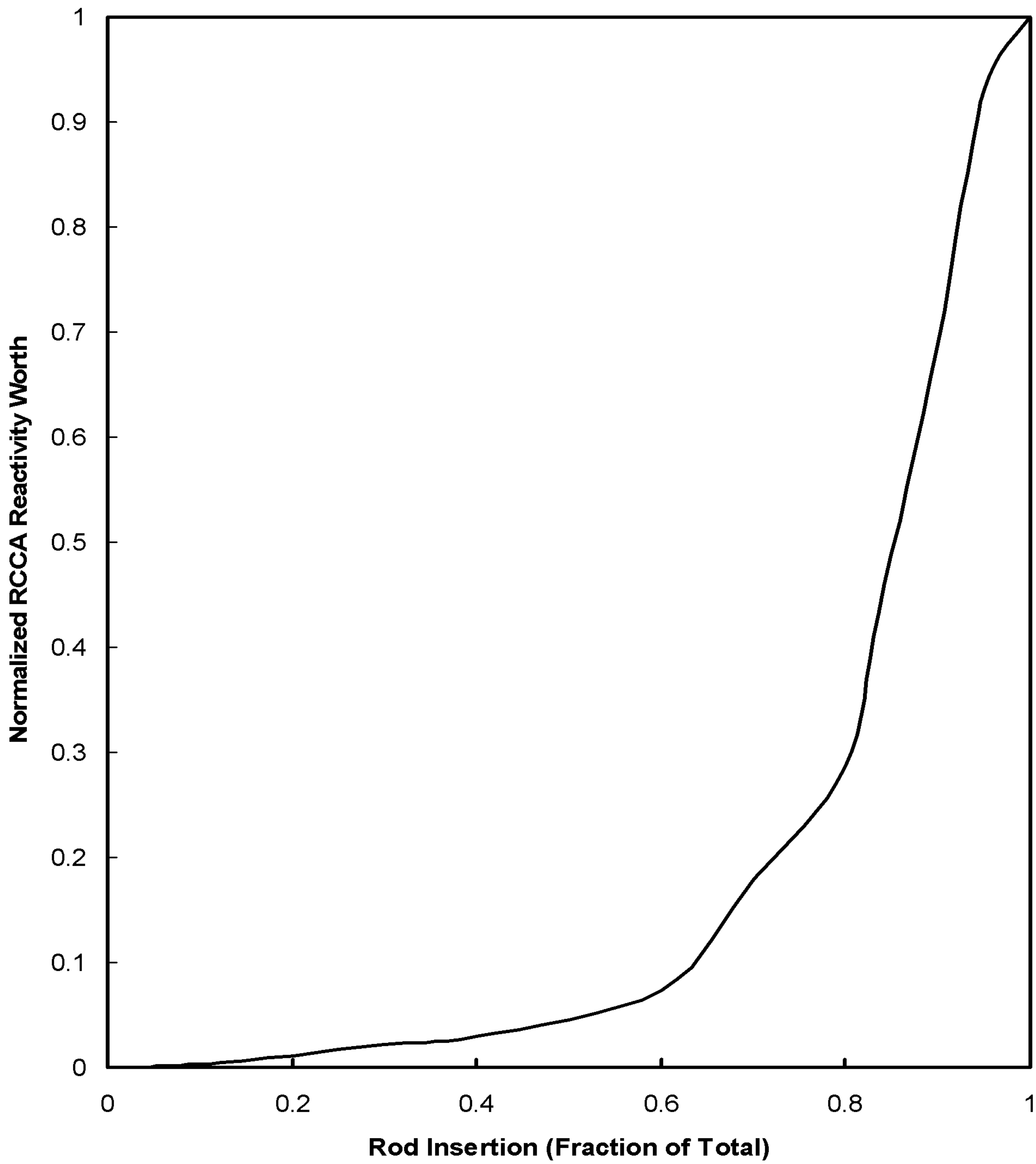


Figure 15.0-5

Normalized RCCA Reactivity Worth vs. Rod Position

Beaver Valley Power Station Unit No. 2
Updated Final Safety Analysis Report

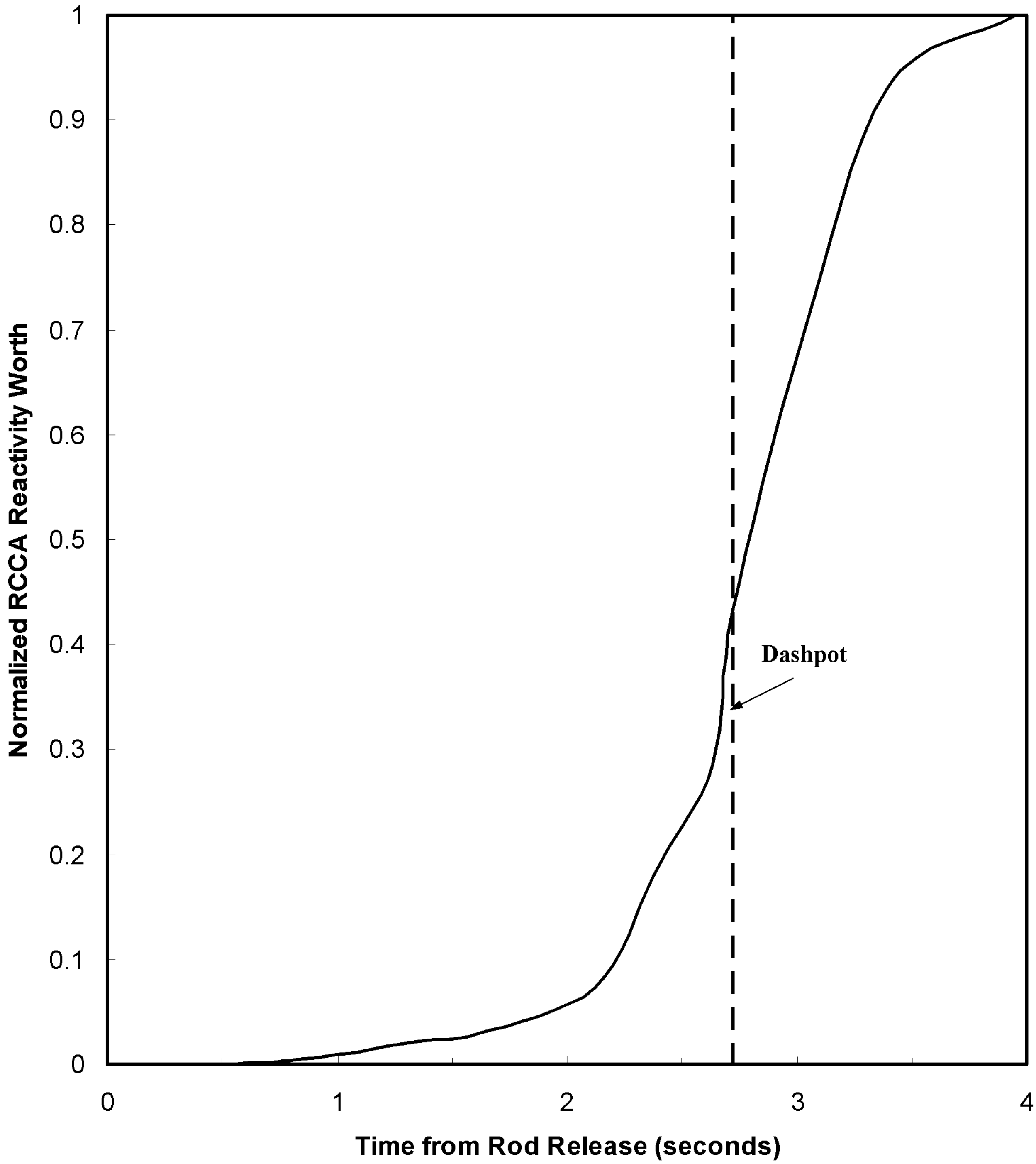


Figure 15.0-6

Normalized RCCA Bank Reactivity Worth vs. Time After Trip

Beaver Valley Power Station Unit No. 2
Updated Final Safety Analysis Report

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 (RCS) by the secondary system. Analyses are presented for several such events which have been identified as limiting cases.

Discussions of the following RCS cooldown events are presented:

1. Feedwater system malfunctions causing a reduction in feedwater temperature (Section 15.1.1).

2. Feedwater system malfunction causing an increase in feedwater flow (Section 15.1.2).

3. Excessive increase in secondary steam flow (Section 15.1.3).

4. Inadvertent opening of a steam generator relief or safety valve causing a depressurization of the main steam system (MSS) (Section 15.1.4).

5. Spectrum of steam system piping failures inside and outside containment (Section 15.1.5).

These are considered to be American Nuclear Society (ANS) Condition II events, with the exception of a major steam system pipe break, which is considered to be an ANS Condition IV event (Section 15.0.1).

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 result in an increase in core power by initially decreasing reactor coolant temperature. Such transients are attenuated by the thermal capacity of the secondary plant and of the RCS. The high neutron flux trip, overtemperature ΔT trip, and overpower ΔT trip are available to prevent any power increase which could lead to a departure from nucleate boiling ratio (DNBR) less than the limit value.

A reduction in feedwater temperature may be caused by the accidental opening of a feedwater heater bypass valve which diverts flow around a portion of the feedwater heaters. In the event of an accidental opening of the bypass valve, there is 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 transient is less severe than the full power case.

The net effect on the RCS due to a reduction in feedwater temperature is similar to the effect of increasing feedwater flow; that is, the reactor power will increase slowly until the transient is terminated via feedwater isolation and reactor trip.

A decrease in normal feedwater temperature is classified as an ANS Condition II event, fault of moderate frequency (Section 15.0.1).

The protection available to mitigate 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 15.0-6.

15.1.1.2 Analysis of Effects and Consequences

The feedwater system malfunction causing a reduction in feedwater temperature event is analyzed by using the detailed digital computer code LOFTRAN (Burnett, et al. 1984).

The following assumptions are made:

1. The initial pressure, temperature and power are assumed to be at their nominal full-power values consistent with Revised Thermal Design Procedure (Friedland and Ray 1989).

2. The initial feedwater temperature is 455°F and the final feedwater temperature is conservatively assumed to be 300°F.

3. End-of-life reactivity coefficients are assumed (i.e., a conservatively large negative moderator temperature coefficient).

4. Reactor trip via Over Power Delta T (OPDT) is assumed.

5. Feedwater isolation occurs via the Low Pressurizer Pressure Safety Injection function.

6. Simultaneous actuation of a low pressure heater bypass and isolation of one string of low pressure feedwater heaters.

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

<u>Results</u>

Opening of a low pressure heater bypass valve causes a reduction in feedwater temperature which increases the thermal load on the primary system. The calculated reduction in feedwater temperature is less than 155°F. The increased thermal load, due to opening of the low pressure heater bypass valve, thus would result in a transient very similar to that presented in Section 15.1.2 for an increase in feedwater flow incident. Therefore, the results of this analysis are not presented.

15.1.1.3 Radiological Consequences

There are no radiological consequences associated with a decrease in feedwater temperature event. The activity is contained within the fuel rods and the RCS radionuclide concentrations remain within Technical Specification limits.

15.1.1.4 Conclusions

The decrease in feedwater temperature transient has been evaluated and the applicable acceptance criteria for the decrease in feedwater temperature event have been met. There are no radiological consequences for this event.

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 high neutron flux trip (primary), overtemperature ΔT trip (backup), and overpower ΔT trip (backup) 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 a feedwater control valve 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 generator. 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.

Continuous addition of excessive feedwater will normally be prevented by the steam generator hi-hi level trip, which closes all feedwater control and isolation valves, trips the main feedwater pumps, and trips the main turbine. This trip function, however, is not required to function.

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

Plant systems and equipment which are available to mitigate the effects of the accident are discussed in Section 15.0.8 and listed in Table 15.0-6.

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.

A control system malfunction or operator error is assumed to cause a feedwater control valve to open fully. Two cases are analyzed as follows:

1. Accidental opening of one feedwater control valve with the reactor just critical at zero load conditions and in manual rod control.

2. Accidental opening of one feedwater control valve with the reactor in automatic control at full power.

Both of these cases are analyzed for operation with three loops in service.

The reactivity insertion rate following a feedwater system malfunction is calculated with the following assumptions:

1. For the feedwater control valve accident at full power, one feedwater control valve is assumed to malfunction resulting in a step increase to 156 percent of nominal feedwater flow to one steam generator.

2. For the feedwater control valve accident at zero load conditions, a feedwater control valve malfunction occurs which results in an increase in flow to one steam generator from zero to 175 percent of the nominal full load value.

3. For the zero load condition, feedwater temperature is at a conservatively low value of 32°F.

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

5. The feedwater flow resulting from a fully open control valve is terminated by a steam generator hi-hi level trip signal which closes all feedwater control and isolation valves, trips the main feedwater pumps, and trips the turbine.

Initial operating conditions are assumed at values consistent with steady-state operation.

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

Normal reactor control systems and engineered safety feature (ESF) systems are not required to function. The reactor protection system (RPS) will function to trip the reactor due to high neutron flux or, when above P-9, high steam generator water level conditions. As described in UFSAR Sections 7.2 and 7.3, a high steam generator water level will cause a turbine trip. Above the P-9 setpoint, a turbine trip will cause a reactor trip. No single active failure will prevent operation of the RPS. A discussion of anticipated transients without SCRAM (ATWS) considerations is provided in Section 15.8.

Results

The calculated sequence of events for this accident is shown in Table 15.1-1.

An accidental full opening of one feedwater control valve with the reactor at zero power conditions with manual rod control is less limiting than an accidental full opening of one feedwater control valve at hot full power. Therefore, the results of the analysis are not presented here. It should be noted that if the incident occurs with the unit just critical at no-load conditions, the reactor may be tripped by the power range high neutron flux trip (low setting) set at approximately 25 percent of nominal full power.

The full power case (with rod control) gives the largest reactivity feedback and results in the greatest power increase. Assuming the reactor to be in the manual control mode results in a slightly less severe transient. The rod control system is not required to function for an excessive feedwater flow event.

When the steam generator water level in the faulted loop reaches the hi-hi level set point, all feedwater control and isolation valves and the main feedwater pumps are tripped. This prevents continuous addition of feedwater. In addition, a reactor trip and turbine trip are initiated. This function prevents moisture carryover to the turbine but is not required to prevent DNB for this event as described below.

Transient results (Figures 15.1-1 and 15.1-2) show the core heat flux, pressurizer pressure, T_{avg}, and DNBR, as well as the increase in nuclear power and loop ΔT associated with the increased thermal load on the reactor. The DNBR decreases due to the increased power needed to heat the additional feedwater being added, but the DNBR does not drop below the limit value. The DNBR settles out to a new, lower, steady-state but acceptable value before the assumed high steam generator water level function occurs. However, the conclusions stated in Section 15.1.2.4 would not change if the high steam generator water level function is assumed not to actuate and the transient is terminated by the operator.

Following the reactor trip and feedwater isolation, Beaver Valley Power Station - Unit 2 (BVPS-2) 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 RCS boron concentration and pressurizer level using the chemical and volume control system and to maintain steam generator level through control of the main or auxiliary feedwater system (AFWS). Any action required of the operator to terminate the event or to maintain BVPS-2 in a stabilized condition following reactor trip will be in a time frame in excess of ten minutes.

Since the power level rises during the excessive feedwater flow incident, the fuel temperatures will also rise until after reactor trip occurs. The core heat flux lags behind the neutron flux response due to the fuel rod thermal time constant, hence the peak value does not exceed 116 percent of its normal value (that is, the assumed high neutron flux trip point). The peak fuel temperature will thus remain well below the fuel melting temperature.

The transient results show that the departure from nucleate boiling ratio (DNBR) limit is not violated 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.

15.1.2.3 Radiological Consequences

There are only minimal radiological consequences from feedwater system malfunctions causing an increase in feedwater flow. The reactor trip causes a turbine trip and heat is removed from the secondary system through the steam generator power relief or safety valves. Since no fuel damage is postulated to occur from this transient, the radiological consequences are less severe than those of the loss of non-emergency ac power to the station auxiliaries analyzed in Section 15.2.6.

15.1.2.4 Conclusions

The results of the analysis show that the DNBRs encountered for an excessive feedwater addition at power or at no-load are at all times above the limit value; hence, the DNB design basis, as described in Section 4.4, is met.

The radiological consequences of this event are not limiting.

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 percent step load increase, or a 5 percent per minute ramp load increase in the range of 15 to 100 percent of full power. Any loading rate in excess of these values may cause a reactor trip actuated by the RPS. Steam flow increases greater than 10 percent are analyzed in Section 15.1.4 and 15.1.5. Any steam flow increase caused by a malfunction or failure of any steam pressure regulator is conservatively bounded by steam increase events in Sections 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, that is, 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.

Available protection for an excessive load increase accident is provided by the following RPS signals:

1. Overtemperature ΔT.

2. Overpower ΔT.

3. Power range high neutron flux.

4. Low pressurizer pressure.

An excessive load increase incident is considered to be an ANS Condition II event, fault of moderate frequency (Section 15.0.1).

15.1.3.2 Analysis of Effects and Consequences

Method of Analysis

Historically, four cases are analyzed to demonstrate the plant behavior following a 10 percent step load increase from the rated load. These cases are as follows:

1. Reactor control in manual with minimum moderator reactivity feedback.

2. Reactor control in manual with maximum moderator reactivity feedback.

3. Reactor control in automatic with minimum moderator reactivity feedback.

4. Reactor control in automatic with maximum moderator reactivity feedback.

For the minimum moderator feedback cases, the core has the least negative moderator temperature coefficient of reactivity and, therefore, the least inherent transient capability. For the maximum moderator 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 being taken for pressurizer heaters.

The Revised Thermal Design Procedure (Friedland and Ray, 1989) is used. Therefore, uncertainties on initial conditions are included in the DNBR limit and initial operating conditions are assumed to be at values consistent with nominal steady-state N loop operation. A pressurizer pressure bias of 7.5 psia is included in the transient pressurizer pressure initial condition assumption.

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

Normal reactor control systems and ESF systems are not required to function. The RPS is assumed to be operable; however, reactor trip is not encountered in any cases due to the error allowances assumed in the set points. No single active failure will prevent the RPS from performing its intended function.

Given the non-limiting nature of this event with respect to the DNBR safety analysis criterion, an explicit analysis was not performed as part of the 9.4% Uprating Program. Instead, an evaluation of this event was performed. The evaluation model consists of the generation of statepoints based on generic conservative data. These statepoints are then compared to the core thermal limits to ensure that the DNBR limit is not violated. A total of three cases are included in this evaluation. These are:

- Reactor in manual rod control with BOL (minimum moderator) reactivity feedback

- Reactor in manual rod control with EOL (maximum moderator) reactivity feedback

- Reactor in automatic rod control

The case which assumes automatic rod control is evaluated to ensure that the worst case is presented. However, the automatic function is not required.

15.1.3.3 Radiological Consequences

There will be no radiological consequences associated with this event. The activity is contained within the fuel rods and the RCS radionuclide concentrations remain within Technical Specification limits.

15.1.3.4 Results and Conclusions

The evaluation confirms that for a 10 percent step load increase, the DNBR remains above the limit value; thus the DNB design basis, as described in Section 4.4, is met. The plant reaches a stabilized condition rapidly following the load increase.

15.1.4 Inadvertent Opening of a Steam Generator Relief or Safety Valve Causing a Depressurization of the Main Steam System

15.1.4.1 Identification of Causes and Accident Description

The results from an inadvertent opening of a main steam generator relief or safety valve causing a depressurization of the main steam system are always bounded by the results from a steam system piping failure presented in Section 15.1.5. The description of the inadvertent opening of a main steam generator relief or safety valve causing a depressurization of the main steam system is presented in this section for historical purposes only.

The most severe core conditions resulting from an accidental depressurization of the MSS result from an inadvertent opening, with failure to close of the largest of any single steam dump, relief, or safety valve. The analyses performed assuming a rupture of a main steam line are given in Section 15.1.5.

The steam release as a consequence of this accident results 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 in coolant temperature and pressure. In the presence of a negative moderator temperature coefficient, the cooldown results in an insertion of positive reactivity.

The analysis is performed to demonstrate that the following criterion is satisfied: assuming a stuck rod cluster control assembly (RCCA), with offsite power available, and assuming a single failure in the ESF, there will be no consequential damage to the core or RCS after reactor trip for a steam release equivalent to the spurious opening, with failure to close, of the largest of any single steam dump, relief, or safety valve.

Accidental depressurization of the secondary system is classified as an ANS Condition II event (Section 15.0.1).

The following systems provide the necessary protection against an accidental depressurization of the MSS:

1. Safety injection actuation from any of the following:

    a. Low steam line pressure,

    b. Low pressurizer pressure, or

    c. High-1 containment pressure.

2. A reactor trip from:

    a. Overtemperature ΔT, overpower ΔT, or high neutron flux,

    b. Low pressurizer pressure, or

    c. Safety injection signal.

3. 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 control valves following reactor trip, a safety injection signal will rapidly close all feedwater control and bypass valves and backup feedwater isolation valves, trip the main feedwater pumps, and trip the turbine.

4. Trip of the fast-acting steam line stop valves (designed to close in less than 5 seconds) on any one of the following:

    a. Low steam line pressure,

    b. High negative steam pressure rate in any loop, or

    c. High-2 containment pressure.

Systems and equipment which are available to mitigate the effects of the accident are also discussed in Section 15.0.8 and listed in Table 15.0-6.

15.1.4.2 Analysis of Effects and Consequences

Method of Analysis

The following analyses of a secondary system steam release are performed for this section:

1. A full plant digital computer simulation using the LOFTRAN Code to determine RCS temperature and pressure during cooldown, and the effect of safety injection.

2. Analyses to determine that there is no damage to the core or RCS.

The following conditions are assumed to exist at the time of a secondary steam system release:

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

2. 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 is included.

3. Minimum capability for injection of boric acid solution corresponding to the most restrictive single failure in the safety injection system (SIS). This corresponds to the flow delivered by one safety injection pump delivering its full contents to the cold leg header. Unborated water must be swept from the safety injection lines downstream of the refueling water storage tank prior to the delivery of boric acid to the reactor coolant loops. This effect has been allowed for in the analysis.

4. The case studied is an initial steam flow from one steam generator with offsite power available. This is the maximum capacity of any single steam dump, relief, or safety valve. Initial hot shutdown conditions at time zero are assumed since this represents the most conservative initial condition. Cases evaluated in Section 15.1.3, Excessive Increase in Secondary Steam Flow, bound a failure of a steam generator dump, safety, or relief valve from full power.

5. In computing the steam flow, the Moody (1965) Curve for FL/D = 0 is used.

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

7. Cases are shown for three loops in operation.

Results

The calculated time sequence of events for this accident is listed in Table 15.1-1.

The results presented are a conservative indication of the events which would occur assuming a secondary system steam release since it is postulated that all of the conditions described in this section occur simultaneously.

Figures 15.1-13 and 15.1-14 show the transient results for a steam flow of 225 lb/sec at 1,000 psia from one steam generator.

The assumed steam release is typical of the capacity of any single steam dump, relief, or safety valve.

Safety injection (SI) is initiated automatically by low pressurizer pressure. Operation of one charging/high head SI pump is assumed. Borated water enters the RCS providing sufficient negative reactivity to prevent core damage. The cooldown for the case shown on Figures 15.1-13 and 15.1-14 is more rapid than the case of steam release from all steam generators through one steam dump, relief, or safety valve. The calculated transient is quite conservative with respect to cooldown, since no credit is taken for the energy stored in the system metal other than that of the fuel elements or the energy stored in the other steam generators. Since the transient occurs over a period of about 5 minutes, the neglected stored energy will have a significant effect in slowing the cooldown.

Following blowdown of the faulted steam generator, BVPS-2 can be brought to a stabilized hot standby condition through control of auxiliary feedwater flow and SI flow as described by plant operating procedures. The operating procedures would call for operator action to limit RCS pressure and pressurizer level by terminating SI flow and to control steam generator level and RCS coolant temperature using the AFWS. Any action required of the operator to maintain BVPS-2 in a stabilized condition will be in a time frame in excess of 10 minutes following SI actuation.

15.1.4.3 Radiological Consequences

The inadvertent opening of a single steam dump, relief, or safety valve can result in steam release from the secondary system. Normally, no activity release is expected. However, if steam generator tube leakage exists coincident with fuel defects, some activity will be released.

The dose, being a function of steam release, will be less than that calculated for the loss of nonemergency ac power to the station auxiliaries accident (Section 15.2.6).

15.1.4.4 Conclusions

The analysis shows that the criteria stated earlier in this section are satisfied. For an accidental depressurization of the MSS, the minimum DNBR remains well above the limiting value and no system design limits are exceeded. The radiological consequences of this accident are not limiting.

15.1.5 Spectrum of Steam System Piping Failure Inside and Outside Containment

15.1.5.1 Identification of Causes and Accident Description

The steam release arising from a rupture of a main steam line would result in an initial increase in steam flow which decreases during the accident as the steam pressure decreases. 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 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. The core is ultimately shut down by the boric acid delivered by the SIS and accumulators.

The analysis of a main steam line rupture is performed to demonstrate that the following criteria are satisfied:

1. Assuming the most limiting single failure of a stuck RCCA, with or without offsite power, and assuming the most limiting single failure in the ESF, the core remains in place and intact. Radiation doses do not exceed the guidelines of 10 CFR 67.

2. Although DNB and possible cladding perforation following a steam pipe rupture are not necessarily unacceptable, the following analysis, in fact, shows that the DNB design basis is met as stated in Section 4.4 for any rupture, assuming the most reactive RCCA assembly stuck in its fully withdrawn position.

A major steam line rupture, or main steam line break (MSLB) is classified as an ANS Condition IV event (Section 15.0.1).

The rupture of a major steam line is the most limiting cooldown transient and, thus, is analyzed at zero power with no decay heat. Decay heat would retard the cooldown thereby reducing the magnitude of the return to power. A detailed analysis of this transient with the most limiting break size, a double-ended rupture, is presented as follows.

The following functions provide the necessary protection for a steam line rupture:

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

    a. Low steam line pressure,

    b. Low pressurizer pressure, or

    c. High-1 containment pressure.

2. A reactor trip from:

    a. Overtemperature ΔT, Overpower ΔT or high neutron flux,

    b. Low pressurizer pressure, or

    c. Safety injection signal.

3. 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 control valves following reactor trip, a safety injection signal will rapidly close all feedwater control and bypass valves and backup feedwater isolation valves, trip the main feedwater pumps, and trip the turbine.

4. Trip of the fast-acting steam line stop valves (designed to close in less than 5 seconds) on:

    a. Low steam line pressure,

    b. High negative steam pressure rate in any loop, or

    c. High-2 containment pressure.

Fast-acting isolation valves are provided in each steam line that will fully close within 6 seconds of actuation following a steam line isolation signal from the integrated protection system. An additional delay of 2.0 seconds is included for sensor and protection system delays. Therefore, the steam line stop valves are credited to be closed within 8.0 seconds. 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 steam line isolation is included in Chapter 10.

Table 15.1-2 lists the equipment required in the recovery from a high energy steam line rupture. Not all equipment is required for any one particular break, since it will vary depending upon postulated break location and details criteria. Design criteria and methods of protection of safety-related equipment from the dynamic effects of postulated piping ruptures are provided in Section 3.6.

15.1.5.2 Analysis of Effects and Consequences

Method of Analysis

The analysis of the main steam line break (MSLB) has been performed to determine the following:

1. The core heat flux and RCS temperature and pressure resulting from the cooldown following the main steam line break (MSLB). The LOFTRAN Code (Burnett 1984) has been used.

2. The thermal and hydraulic behavior of the core following an MSLB. A detailed thermal and hydraulic digital-computer code, VIPRE (Sung 1999), has been used to determine if DNB occurs for the core conditions computed in item 1.

The analysis has been performed with three reactor coolant loops in operation.

The following conditions were assumed to exist at the time of an MSLB accident:

1. 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 an MSLB accident will not lead to a more adverse condition than the case analyzed.

2. 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 average coolant temperature at 1,000 psi corresponding to the negative moderator temperature coefficient used is shown on Figure 15.1-11. (The effect of power generation in the core on overall reactivity is shown on Figure 15.1-15.)

   The core properties associated with the sector nearest the affected steam generator and those associated with the remaining sectors 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 conditions for the cases analyzed. This core analysis 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 nonuniform 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 local effects for the conditions. These results verify conservatism: underprediction of negative reactivity feedback from power generation.

3. Minimum capability for injection of concentrated boric acid solution (2,400 ppm boron) corresponding to the most restrictive single failure in the SI portion of the emergency core cooling system (ECCS). Minimum boron concentration of injection fluid of 2,300 ppm is assumed for the accumulators. The ECCS consists of three systems:

   a. The passive accumulators,

   b. The low head SIS, and

   c. The high head SIS.

   The high head system and passive accumulators are modeled for the MSLB accident analysis.

The modeling of the SIS in LOFTRAN is described by Burnett (1984). The flow corresponds to that delivered by one SI pump delivering its full flow to the cold leg header. The water, which must be swept from the lines downstream of the RWST prior to the delivery of concentrated boric acid to the reactor coolant loops, is assumed to have no boron concentration.

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 SIS. 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 SIS due to changes in the RCS pressure. The SIS flow calculation includes the line losses in the system as well as the SI pump head curve.

The boric acid solution from the SIS is assumed to be uniformly delivered to the active reactor coolant loops. The boron in the loops is then delivered to the inlet plenum where the coolant (and boron) from each loop is mixed and delivered to the core. The stuck RCCA is conservatively assumed to be located in the core sector near the broken steam generator. Because the cold leg pressure is lowest in the broken loop due to larger loop flow and a larger pressure drop, more boron would actually be delivered to the core sector where the power is being generated, enhancing the effect of the boric acid on the transient. No credit was taken for this in the analysis. Furthermore, sensitivity studies have demonstrated that the transient is insensitive to boron worth or distribution.

For the cases where offsite power is assumed, the sequence of events in the SIS is the following. After the generation of the SI signal (appropriate delays for instrumentation, logic, and signal transport included), the appropriate valves begin to operate and the SI pump starts. Injection of borated water is assumed not to occur until the VCT charging pump suction valves are closed following opening of the RWST charging pump suction valves. In 27 seconds, total, the valves are assumed to be in their final position and the pump is assumed to be at full speed. The volume containing the unborated water is swept into the core before the 2,400 ppm borated water from the RWST reaches the core. This delay, described herewith, is inherently included in the modeling.

In cases where offsite power is not available, a 10-second delay to start the emergency diesel generators in addition to the time necessary to start the SI equipment (mentioned previously) is included.

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

5. Since the steam generators are provided with integral flow restrictors with a 1.069 ft^2 throat area, any rupture with a break area greater than 1.069 ft^2, regardless of location, would have the same effect on the NSSS as the 1.069 ft^2 break. The following cases have been considered in determining the core power and RCS transients:

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

   b. Case a with loss of offsite power (LOOP) simultaneous with the MSLB and initiation of the SI signal. Loss of offsite power results in reactor coolant pump coastdown.

6. 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 MSLB. This void, in conjunction with the large negative moderator coefficient, partially offsets the effect of the stuck RCCA. The power peaking factors depend upon the core power, temperature, pressure, and flow, and are different for each case studied.

   The core parameters used for each of the two cases correspond to values determined from the respective transient analysis.

   Both cases listed previously assume initial hot shutdown conditions at time zero since this represents the most pessimistic initial condition. Should the reactor be just critical or operating at power at the time of an MSLB, 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 cooldown caused by the MSLB 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.

7. In computing the steam flow during an MSLB, the Moody (1965) Curve for FL/D = 0 is used.

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

9. Feedwater addition aggravates cooldown accidents like the steam line rupture. Therefore, feedwater flow is assumed. All the main and auxiliary feedwater pumps are assumed to be operating at full capacity when the rupture occurs, even though the plant is assumed to be in a hot standby condition. Full main and auxiliary feedwater flow is maintained for 5 seconds following the receipt of a feedwater isolation signal from the integrated protection system following SI actuation. An additional 2.0 second delay is added for sensor and protection system delays. During the first 5 seconds following the start of the transient, a feedwater isolation signal is generated to close both the feedwater control and feedwater isolation valves. The feedwater isolation valves have a 5 second closure time. The feedwater control valve's closure time plus the ESF actuation time will be ≤ 7 seconds. All the auxiliary feedwater is assumed to be pumped into the depressurizing steam generator.

10. The effect of heat transferred from thick metal in the pressurizer and reactor vessel upper head is not included in the cases analyzed. Studies previously performed have shown that the heat transferred to the coolant from these latent sources is a net benefit in DNB and RCS energy when the effect of the extra heat on reactivity and peak power is considered.

Results

The calculated sequence of events for complete severance of a pipe at no-load conditions with offsite power available is shown in Table 15.1-1.

The results presented are a conservative indication of the events which would occur assuming an MSLB since it is postulated that all of the preceding conditions occur simultaneously.

Core Power and Reactor Coolant System Transient

Figures 15.1-16, 15.1-17, 15.1-18, 15.1-19, and 15.1-20 show the RCS transient and core heat flux following a main steamline rupture (complete severance of a pipe) at initial no-load conditions (Case a). 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 rupture occurs, the initiation of SI by low steam line 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 steam lines by high containment pressure signals or by low steam line pressure 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 steam line stop valves are designed to be fully closed in less than 5 seconds from receipt of a closure signal. The steam line stop valves are credited to be closed within 8 seconds to account for valve degradation and sensor delay.

As shown on Figure 15.1-19, the core attains criticality with the RCCAs inserted (with the design shutdown assuming one stuck RCCA) before boron solution at 2,400 ppm (in the RWST) and 2,300 ppm (in the accumulators) enters the RCS.

Based on the results of the generic study in WCAP-9227 (Hollingsworth 1998), the case with LOOP (Case b) is less severe than Case a described previously. The case without offsite power affects the transient in basically two ways:

1. The delay to start SI is increased by the time required to start the diesel, that is, 10 seconds.

2. The loss of the reactor coolant pump reduces the flow in the RCS.

The additional 10 seconds do not significantly change the conclusions. The loss of forced circulation, however, makes the transient less severe than the case with forced circulation. This is due to:

1. The core and steam generator are more decoupled, hence the time to go critical is much longer;

2. Without forced coolant, the rate of heat transfer is reduced in the steam generator, thereby reducing the core cooldown rate; and

3. Local density feedback effects minimize the power level following the assumed MSLB.

All of these effects result in the case with forced coolant flow being much more severe than the case without offsite power for the potential of clad damage.

It has been noted that following an MSLB, 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 LOOP, this heat is removed to the atmosphere via the steam line safety and atmospheric dump valves.

Following blowdown of the faulted steam generator, BVPS-2 can be brought to a stabilized hot standby condition through control of the auxiliary feedwater flow and SI flow as described by plant operating procedures. The operating procedures would call for operator action to limit RCS pressure and pressurizer level by terminating SI flow and to control steam generator level and RCS coolant temperature using the AFWS.

The ability of the intact steam generators to remove residual energy from the RCS in the long term is demonstrated by the major rupture of a main feedwater line (Section 15.2.8). The MSLB is less limiting with respect to cooldown without offsite power because temperatures are much lower, all of the auxiliary feedwater can be delivered to the steam generators, and the steam blowdown leaves a higher water inventory than the feedline blowdown. The feedline rupture demonstrates that the intact steam generators and auxiliary feedwater provide sufficient heat sink to remove long term heat following the transient.

Margin to Critical Heat Flux

A DNB analysis was performed for the limiting case with offsite power available. It was found that the DNB design basis, as stated in Section 4.4, was met for both cases.

15.1.5.3 Radiological Consequences

The MSLB is postulated to occur in a main steam line outside the containment. Steam and feedwater isolation valves shut automatically, and the AFWS starts, supplying feedwater to each of the steam generators until the feedwater system is manually isolated from the affected steam generator (that is, the one associated with the broken main steam line).

The conservative analysis assumes that BVPS-2 is operating with Technical Specification iodine concentrations in the primary coolant and secondary coolant systems and with a primary to secondary Technical Specification leakage of 450 gpd. The equilibrium technical specification primary and secondary coolant systems noble gas and iodine concentrations are presented in Table 15.0-8c.

Since there is no fuel damage associated with this event, the radiological consequences are determined assuming each of the following occurrences which results in an increased inventory of iodine in the primary coolant:

1. Pre-accident iodine spike, and
2. Accident-initiated concurrent iodine spike.

The pre-accident iodine spike is the result of a primary plant transient which will increase the primary system iodine concentrations to the levels discussed in Section 15.0.9.4 and shown in Table 15.0-9. The accident-initiated or concurrent iodine spike is modeled by assuming that the iodine release rates from the fuel rods into the primary coolant are 500 times the Technical Specification equilibrium release rates. The iodine release rates for the concurrent iodine spiking conditions are calculated for the limiting MSLB as detailed in Section 15.0.9.4 and Tables 15.0-10 and 15.0-10a.

In 1999, the radiological consequences of a MSLB outside of containment was re-analyzed in support of the Alternate Plugging Criteria (APC) for steam generators (ref. T/S amendment 115). The MSLB is of interest due to the rapid depressurization of the secondary side and the high differential pressure across the steam generator tubes that can occur. The APC allows steam generator tubes having outside diameter stress corrosion cracking (ODSCC) to remain in service with higher NDE indications than would be allowed under prior repair criteria, subject to conditions established in technical specifications. One such requirement is to project, on the basis of the NDE indication (voltage), the potential leakage (95 percentile/95% confidence) should a MSLB occur, and, on the basis of this projected leakage, the resulting offsite and control room doses.

Consequently, in support of APC, and in lieu of calculating the radiological consequence of this event for each operating cycle, an analysis was performed to establish a maximum allowable accident-induced tube leakage, against which the leakage projections could be compared. For this purpose, the dose was maximized to the most limiting acceptance criteria in regulatory guidance to establish a maximum acceptable accident initiated leak rate in addition to the traditionally assumed technical specification primary-to-secondary leak rate.

In 2006, as part of core power uprate, the methodology used to analyze the dose consequences following a MSLB was updated to be consistent with the alternative source term guidance provided in RG 1.183.

Table 15.1-3 lists the key parameters utilized to develop the radiological consequences following a MSLB.

Computer program PERC2 is used to calculate the control room and site boundary doses due to airborne radioactivity releases following a MSLB.

Faulted Steam Generator

The radiological model used for the assessment conservatively assumes immediate dry-out of the faulted SG following a MSLB resulting in the instantaneous release of all of the liquid contents of the steam generator, which are assumed at maximum Technical Specification concentrations. The secondary steam initially contained in the faulted steam generator is also released; however, this contribution is not included in this analysis since the associated radioactivity is insignificant compared to the other contributions.

The maximum technical specification primary to secondary leakage into the faulted steam generator is 150 gpd at STP. The elevated iodine activity in the primary coolant due to a postulated pre-accident or concurrent iodine spike, as well as the noble gas, leak into the faulted steam generator and are released to the environment from the break point without hold-up or decontamination. The releases from the faulted SG due to primary to secondary leakage continues until T=21 hrs (i.e., estimated time for the RHR System to bring the primary coolant temperature down to 212°F).

In support of APC and in accordance with the guidance provided in Nuclear Regulatory Commission (NRC) Generic Letter 95-05, an accident-induced primary-to-secondary leakage is also postulated to occur (via pre-existing tube defects) as a result of the rapid depressurization of the secondary side due to the MSLB and the consequent high differential pressure across the faulted steam generator. The MSLB dose analysis is performed to establish a maximum allowable accident-induced leakage, against which the cycle leakage projections can be compared. The accident induced leakage rate is the maximum primary-to-secondary SG tube leakage that could occur with the offsite or control room operator doses remaining within applicable limits. For analysis purposes, this tube leakage is conservatively assigned to the faulted SG. Consequently, the primary-to-secondary leakage in the faulted steam generator reflects 150 gpd at STP, plus the maximum allowable accident induced tube leakage that results in dose consequences that are just within the most limiting of the regulatory limits associated with the EAB/LPZ and the control room.

Intact Steam Generator

Based on an assumption of a simultaneous Loss of Offsite Power, the condenser is assumed to be unavailable, and environmental steam releases via the MSSVs/ADVs from the intact steam generators are used to cool down the reactor until the Residual Heat Removal (RHR) system starts shutdown cooling. The elevated iodine activity in the primary coolant due to a postulated pre-accident or concurrent iodine spike, as well as the noble gas, leak into the intact steam generators, and are released to the environment from the MSSVs/ADVs.

The iodine activity in the intact SG liquid is released to the environment in proportion to the steaming rate and the partition factor. The steam releases from the intact SGs continue until shutdown cooling is initiated via operation of the RHR system at T=8 hrs, resulting in the termination of environmental releases via this pathway.

15.1.5.4 Conclusions

The analysis has shown that the criteria stated in Section 15.1.5.1 are satisfied with the exclusion of the radiological criteria. Although DNB and possible cladding perforation following a steam pipe rupture are not necessarily unacceptable and not precluded by the criteria, the analysis, in fact, shows that the DNB design bases is met as stated in Section 4.4. The radiological consequences remain within the dose limits provided in 10 CFR 50.67 as supplemented by SRP 15.0.1 and Regulatory Guide 1.183.

15.1.5.5 Steam System Piping Failure at Full Power

15.1.5.5.1 Identification of Causes and Accident Description

A rupture in the main steam system piping from an at-power condition creates an increased steam load, which extracts an increased amount of heat from the RCS via the steam generators. This results in a reduction in RCS temperature and pressure. In the presence of a strong negative moderator temperature coefficient, typical of end-of-cycle life conditions, the colder core inlet coolant temperature causes the core power to increase from its initial level due to the positive reactivity insertion. The power approaches a level equal to the total steam flow. Depending on the break size, a reactor trip may occur due to overpower conditions or as a result of a steam line break protection function actuation.

The steam system piping failure accident analysis described in subsection 15.1.5.1 is performed assuming a hot zero power initial condition with the control rods inserted in the core, except for the most reactive rod in the fully withdrawn position, out of the core. That condition could occur while the reactor is at hot shutdown at the minimum required shutdown margin or after the plant has been tripped manually or by the reactor protection system following a steam line break from an at-power condition. For an at-power break, the analysis in this section represents the limiting condition with respect to core protection for the time period following reactor trip. The purpose of this section is to describe the analysis of a steam system piping failure occurring from an at-power initial condition, 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 an ANS Condition III (infrequent fault) or Condition IV (limiting fault), as defined in subsection 15.0.1.

15.1.5.5.2 Analysis of Effects and Consequences

Method of Analysis

The analysis of the steam line rupture is performed in the following stages:

1. The LOFTRAN code (Burnette 1984) is used to calculate the nuclear power, core heat flux, and RCS temperature and pressure transients resulting from the cooldown following the steam line break.

2. The core radial and axial peaking factors are determined using the thermal-hydraulic conditions from the transient analysis as input to the nuclear core models. The VIPRE code (Sung 1999) is then used to calculate the DNBR for the limiting time during the transient.

This accident is analyzed with the Revised Thermal Design Procedure as described in WCAP-11398-A (Friedland 1989). Plant characteristics and initial conditions are discussed in subsection 15.1.5.2.

The following assumptions are made in the transient analysis:

1. Initial Conditions - The initial core power, reactor coolant temperature, and RCS pressure are assumed to be at their nominal full-power values. The full power condition is more limiting than part-power in terms of DNBR. The RCS Minimum Measured Flow is used. Uncertainties in initial conditions are included in the DNBR limit as described in WCAP-11398-A (Friedland 1989).

2. Break size - A spectrum of break sizes is analyzed. Small breaks do not result in a reactor trip. Intermediate size breaks result in a reactor trip on Overpower ΔT. Larger break sizes result in a reactor trip as a consequence of the lead-lag compensated Low Steam Pressure Safety Injection actuation.

3. Break flow - In computing the steam flow during a steam line break, the Moody curve[2] for fL/D = 0 is used.

4. Reactivity Coefficients - The analysis assumes maximum moderator reactivity feedback and minimum Doppler power feedback to maximize the power increase following the break.

5. Protection System - The protection system features that mitigate the effects of a steam line break are described in subsection 15.1.5.1. This analysis only considers the initial phase of the transient initiated from an at-power condition. Protection in this phase of the transient is provided by a reactor trip, if necessary.

   Subsection 15.1.5.1 presents the analysis of the bounding transient following reactor trip, where other protection system features are actuated to mitigate the effects of the steam line break.

6. Control Systems - The results of the analysis would not be more severe as a result of control system actuation; therefore, their effects have been ignored in the analysis. Control systems are not credited in mitigating the effects of the transient.

15.1.5.5.3 Results

A spectrum of steam line break sizes was analyzed from 0.1 ft^2 to 1.4 ft^2. The results show that for small break sizes up to 0.3 ft^2 a reactor trip is not generated. In this case, the event is similar to an excessive load increase event as described in subsection 15.1.3. The core reaches a new equilibrium condition at a higher power equivalent to the increased steam release. For break sizes of 0.4 ft^2 to 0.8 ft^2 the power increase results in a reactor trip on overpower ΔT. For break sizes larger than 0.8 ft^2 a reactor trip is generated within a few seconds of the break on the lead-lag compensated Low Steam Pressure Safety Injection actuation signal.

The limiting case for demonstrating DNB protection is the 0.8 ft^2 break, the largest break size that results in a trip on overpower ΔT. The time sequence of events for this case is shown on Table 15.1-1. Figures 15.1-21, 15.1-22, 15.1-23, and 15.1-24 show the transient response.

15.1.5.5.4 Conclusions

A detailed DNB analysis is performed as part of each cycle-specific reload safety evaluation, using radial and axial core peaking factors which are dependent on the cycle-specific loading pattern. The analysis concludes that the DNB design basis is met for the limiting case. Although DNB and possible clad perforation following a steam pipe rupture are not necessarily unacceptable and not precluded by the criteria, the above analysis, in fact, shows that the minimum DNBR remains above the limit value for any rupture occurring from an at-power condition prior to and immediately following a reactor trip.

15.1.6 References for Section 15.1

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

BVPS, Safety Analysis of the EAB, LPZ and Control Room Doses from a Main Steam Line Break Outside of CNMT at Unit 2 with Increased Primary-to-secondary Leakage (SG APC), ERS-SFL-96-010.

Technical Evaluation Report No. 12075, BVPS-2 Design Basis for Safety Limit Associated with Overpower ΔT Trip Setpoint.

Hollingsworth, S. D. and Wood, D. C., "Reactor Core Response to Excessive Secondary Steam Releases," WCAP-9227-A, Revision 1 (Non-Proprietary), WCAP-9226-P-A (Proprietary), February 1998.

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

USNRC Voltage Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking, Generic Letter 95-05.

Friedland, A. J. and Ray, S., “Revised Thermal Design Procedure,” WCAP-11397-P-A (Proprietary) and WCAP-11398-A (Non-Proprietary), April 1989.

Sung, Y. X., et. al., “VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis,” WCAP-14565-P-A (Proprietary) and WCAP-15306-NP-A (non-Proprietary), October 1999.

Tables for Section 15.1

TABLE 15.1-1

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

Accident	Event	Time (sec)
Excessive Feedwater Flow	One main feedwater control valve fails fully open	0.0
	Minimum DNBR occurs	72.5
	Hi-hi steam generator water level signal generated	114.9
	Turbine trip occurs due to hi-hi steam generator level	117.4
	Reactor trip occurs	119.4
	Feedwater isolation valves begin to close	121.9
Accidental depressurization of the main steam system	Inadvertent opening of one main steam safety or relief valve	0.0
	Pressurizer empties	190.0
	Boron reaches core	243.0
Steam System Piping Failure		
Reactor at hot zero power with offsite power available. All control rods inserted except most reactive RCCA, Shutdown Margin = 1.77% $\Delta k/k$.	Double-ended guillotine break occurs	0.0
	Low Steam Pressure SIS Actuation Setpoint Reached	1.0
	Main Feedwater flow is isolated	8.0
	MSIVs close	9.0
	Head-head SI pump reaches rated speed	28.0
	Reactor becomes critical	30.8
	Power reaches maximum level	203.2
	Reactor goes subcritical	269.6

TABLE 15.1-1 (Cont)

Accident	Event	Time (sec)
Major Secondary System Pipe Rupture at Hot Full Power Conditions	Steam Line Ruptures	0
	Overpower ΔT Reactor Trip Setpoint Reached	26.5
	Rods Begin to Drop	28.5
	Minimum DNBR Occurs	29.1
	Peak Core Heat Flux Occurs	29.1

TABLE 15.1-2

EQUIPMENT REQUIRED FOLLOWING A MAIN STEAM LINE BREAK

Short Term (Required for Mitigation of Accident)	Hot Standby	Required for Cooldown
Reactor trip and safeguards actuation channels including sensors, circuitry, and processing equipment (the protection circuits use to trip the reactor on overtemperature ΔT and overpower ΔT may be excluded).	Auxiliary feedwater system including pumps, water supply, and system valves and piping	Steam Line Atmospheric Dump Valves Controls for defeating automatic safety injection actuation during a cooldown and depressurization.
Safety Injection System including the pumps, the refueling water storage tank, and the systems valves and piping.	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.
Emergency diesel generators and Class 1E Power distribution equipment	Main steam safety valves	Service water and component cooling for residual heat removal.
Service water to charging pump coolers and recirculation spray coolers.		Redundant accumulator vents
Containment spray system equipment		High head safety injection throttling.
Pressurizer and main steam safety valves.		Pressurizer power-operated relief valves
Circuits and/or equipment required to trip the main feedwater pumps.		Reactor vessel head letdown
Main feedwater isolation valves (trip closed feature). Feedwater control and bypass valves (trip closed feature).		

TABLE 15.1-2 (Cont)

Short Term (Required for Mitigation of Accident)	Hot Standby	Required for Cooldown
Main steam line isolation and bypass valves (trip closed feature).		
Steam generator blowdown isolation valves (automatic closure feature).		
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.		

TABLE 15.1-3

ASSUMPTIONS USED FOR THE MAIN STEAM LINE BREAK ACCIDENT

Core Power Level	2918 MWt
Reactor Coolant Mass (min)	341,332 lbm
Leakrate into Faulted Steam Generator	150 gpd @ STP
Amount of Accident Induced Leakage (AIL) into	2.1 gpm @ STP
Faulted SG.	21 hrs
Maximum time to cool RCS to 212F	300 gpd total from 2 SGs @ STP
Leakrate to Intact Steam Generators	0%
Failed/Melted Fuel Percentage	Table 15.0-8c (0.35 µCi/gm DE-I131)
RCS Tech Spec Iodine & NG Concentration	Table 15.0-10 (0.35 µCi/gm DE-I131)
RCS Equilibrium Iodine Appearance Rates	Table 15.0-9 (21 µCi/gm DE-I131)
Pre-Accident Iodine Spike Activity	500 times equilibrium appearance
Accident Initiated Spike Appearance Rate	rates
Duration of Accident Initiated Spike	4 hours
Secondary System Release Parameters:	
Iodine Species released to Environment	97% elemental; 3% organic
Tech Spec Activity in SG liquid	Table 15.0-8c (0.1 µCi/gm DE-I131)
Iodine Partition Coefficient in Intact SG	100 (all tubes submerged)
Fraction of Noble Gas Released from Intact SG	1.0 (Released without holdup)
Fraction of Iodine Released form Faulted SG	1.0 (Released without holdup)
Fraction of Noble Gas Released from faulted SG	1.0 (Released without holdup)
Minimum Post-Accident Intact SG Liquid Mass	105,076 lbm per SG
Maximum Initial Liquid in each SG	105,076 lbm
Steam Releases from Intact SG	0-2 hr (350,000 lbm)
	2-8 hr (730,000 lbm)
Dryout of Faulted SG	Instantaneous
Termination of release from Faulted SG	21 hours
Termination of release from Intact SG	8 hours
Release Point: Faulted SG	Break Point
Release Point: Intact SG	MSSV/ADVs
CR emergency Ventilation: Initiation	T= 30 minutes
Signal/Timing	Manual
CR pressurized and in Emergency Mode	24 hours after DBA
Control Room Purge (Time/Rate)	@16,200 cfm (min) for 30 min

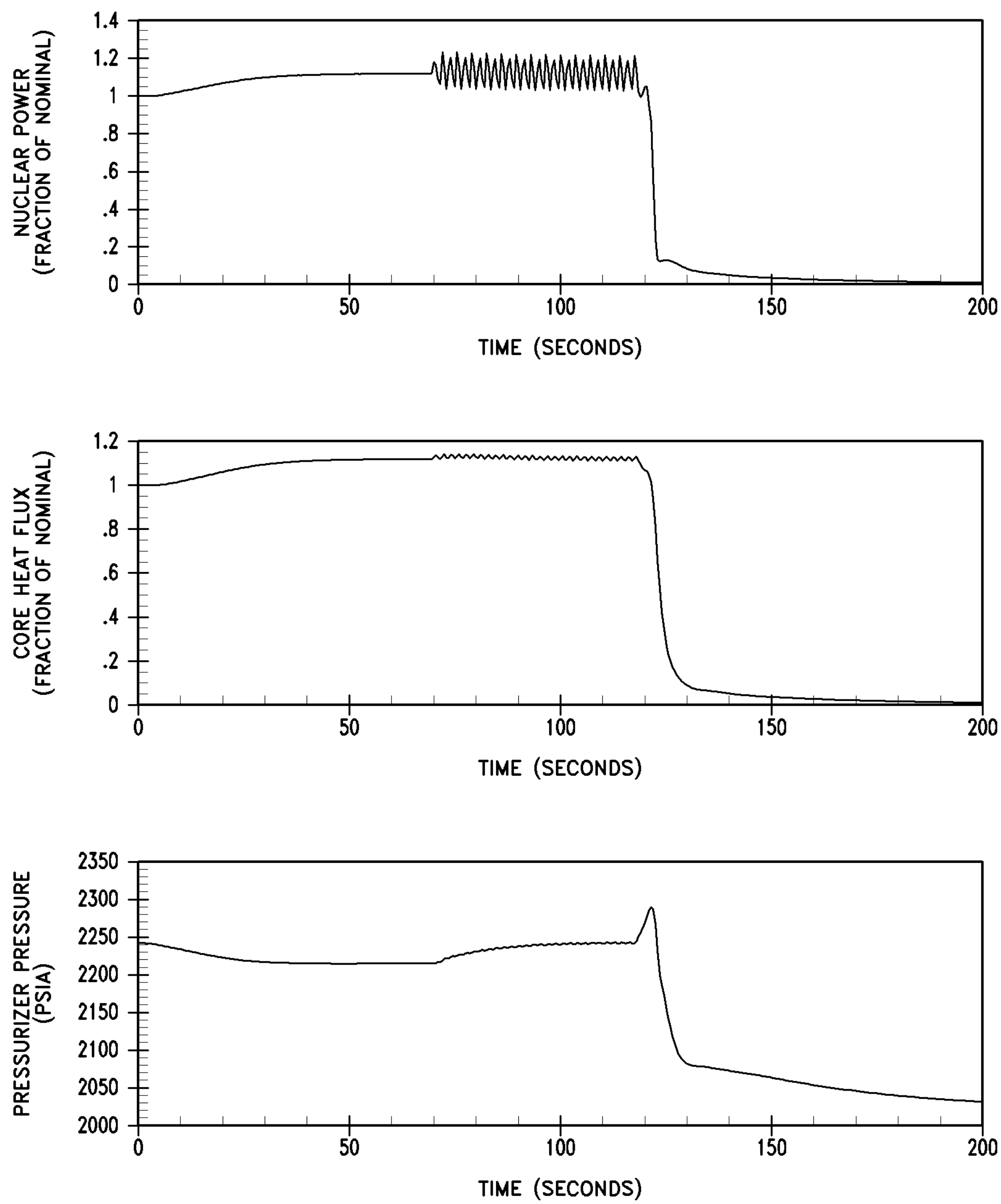


Figure 15.1-1

Nuclear Power, Core Heat Flux, and Pressurizer Pressure Transients for Feedwater Control Valve Malfunction at Full Power

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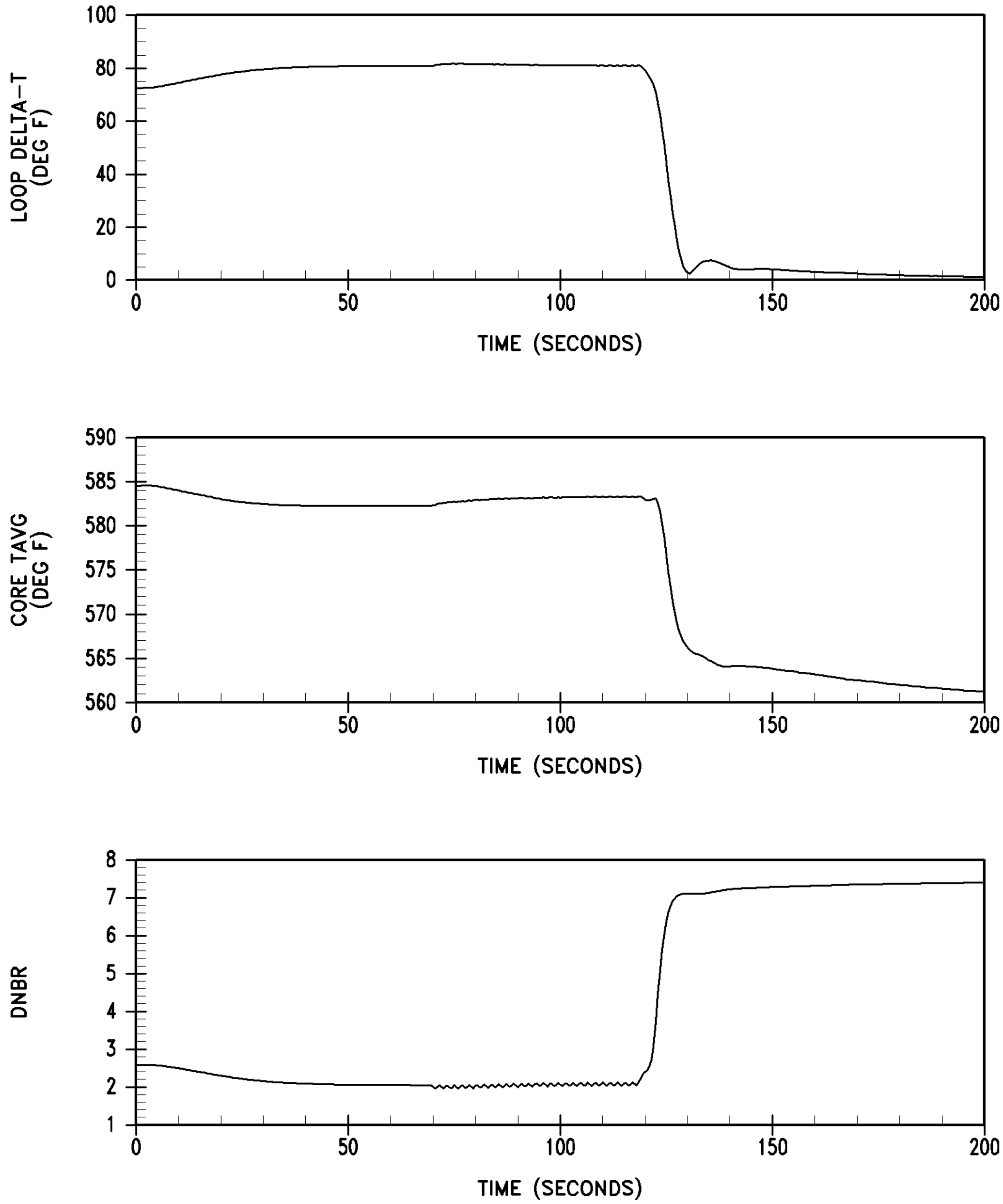

Figure 15.1-2

Loop Delta-T, Core Average Temperature, and DNBR Transients for Feedwater Control Valve Malfunction at Full Power

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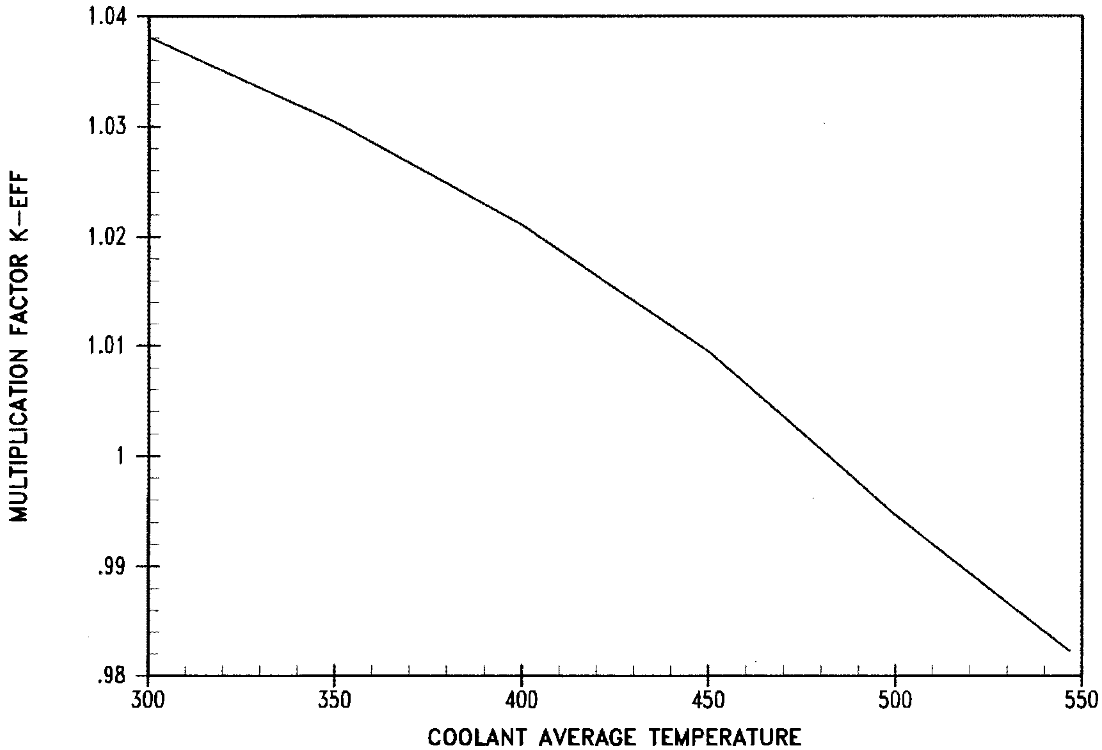

Figure 15.1-11

Keff versus Coolant Average Temperature

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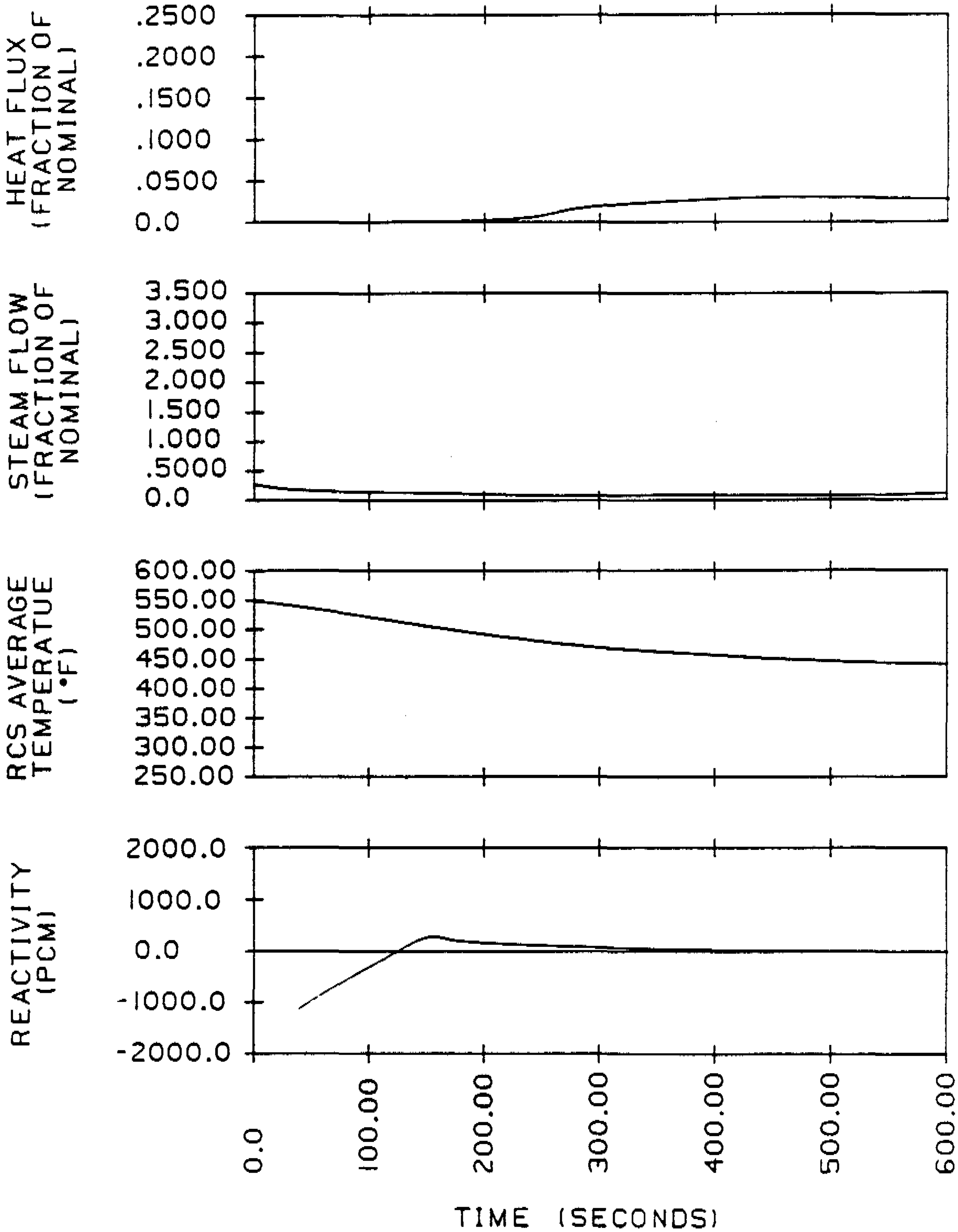

FIGURE 15.1-13
FAILURE OF A STEAM GENERATOR SAFETY OR DUMP VALVE-HEAT FLUX VS. TIME, STEAM FLOW VS. TIME, AVERAGE TEMPERATURE VS. TIME, REACTIVITY VS. TIME
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

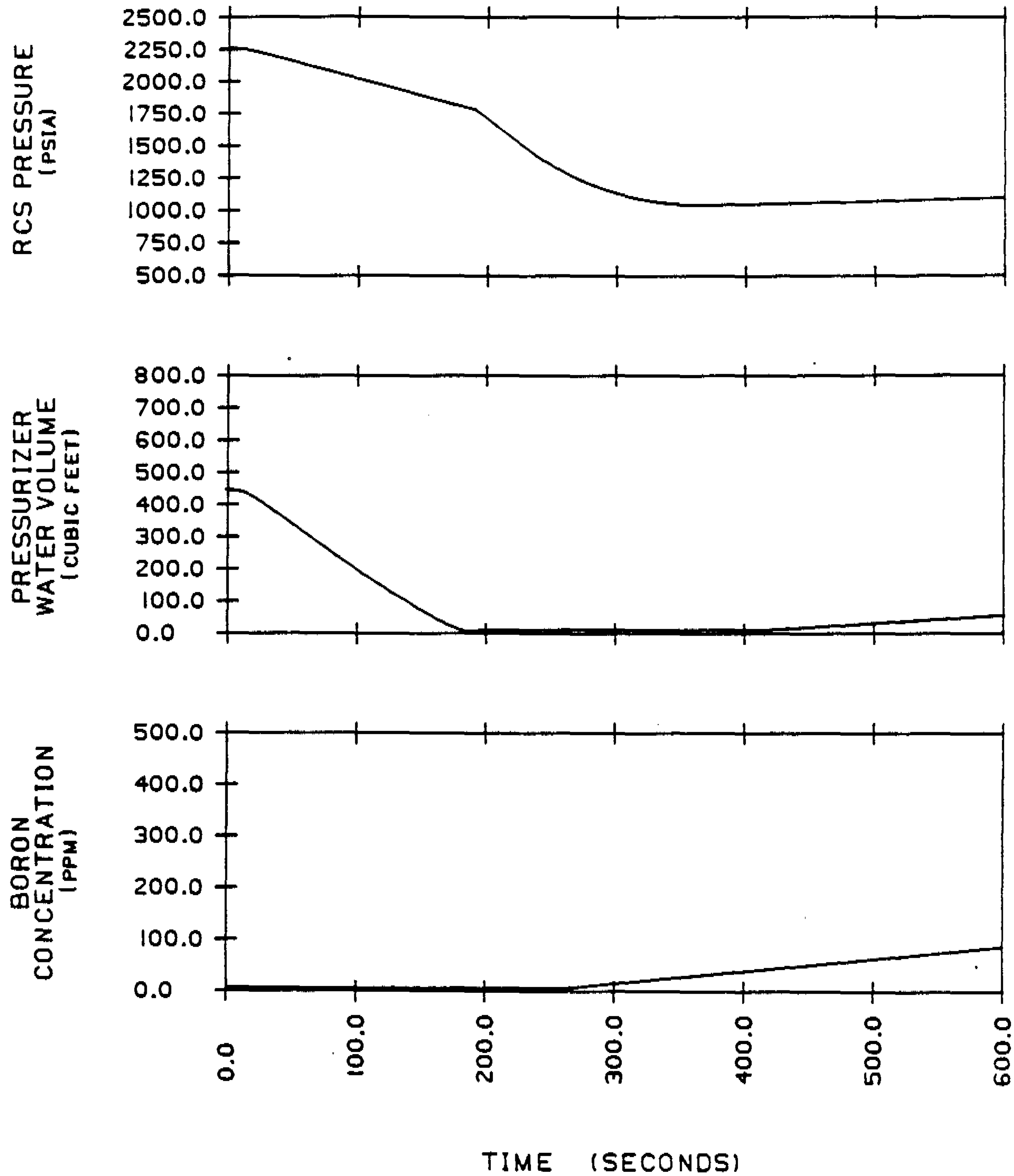

FIGURE 15.1-14
FAILURE OF A STEAM GENERATOR SAFETY OR DUMP VALVE - RCS PRESSURE VS. TIME, PRESSURIZER WATER VOLUME VS. TIME, BORON CONCENTRATION VS. TIME
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

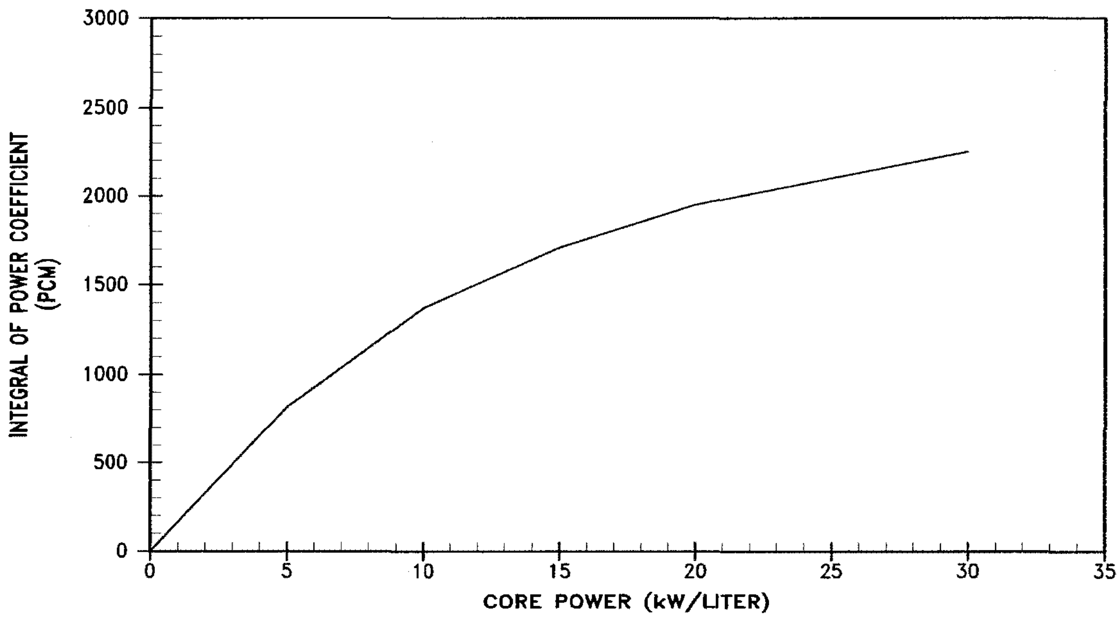

Figure 15.1-15

Doppler Power Coefficient

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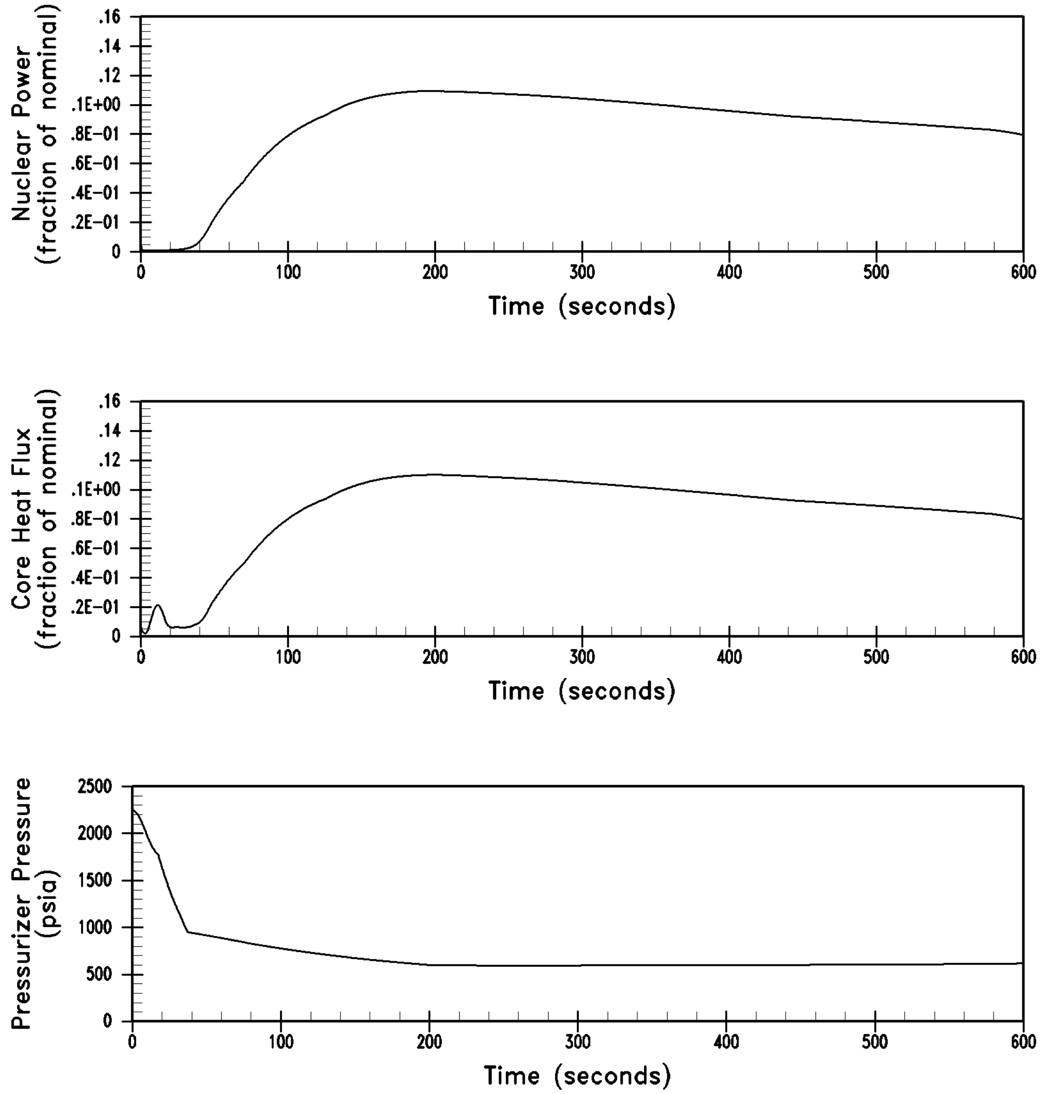

Figure 15.1-16

1.069 ft^2 Steamline Rupture at Hot Zero Power with Offsite Power Available – Nuclear Power, Core Heat Flux and Pressurizer Pressure Transients

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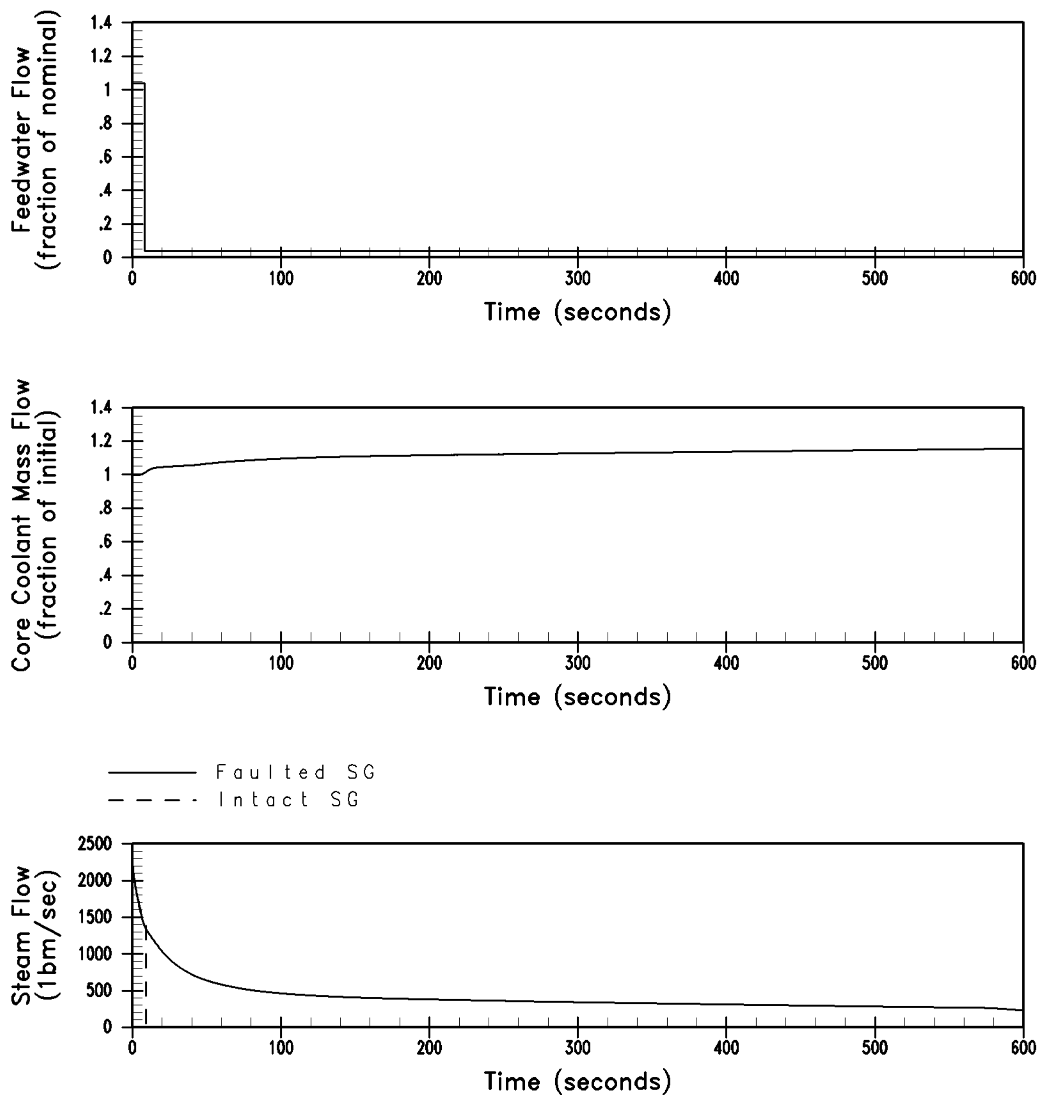

Figure 15.1-17

1.069 ft^2 Steamline Rupture at Hot Zero Power with Offsite Power Available – Feed Flow, Core Flow and Steam Flow Transients

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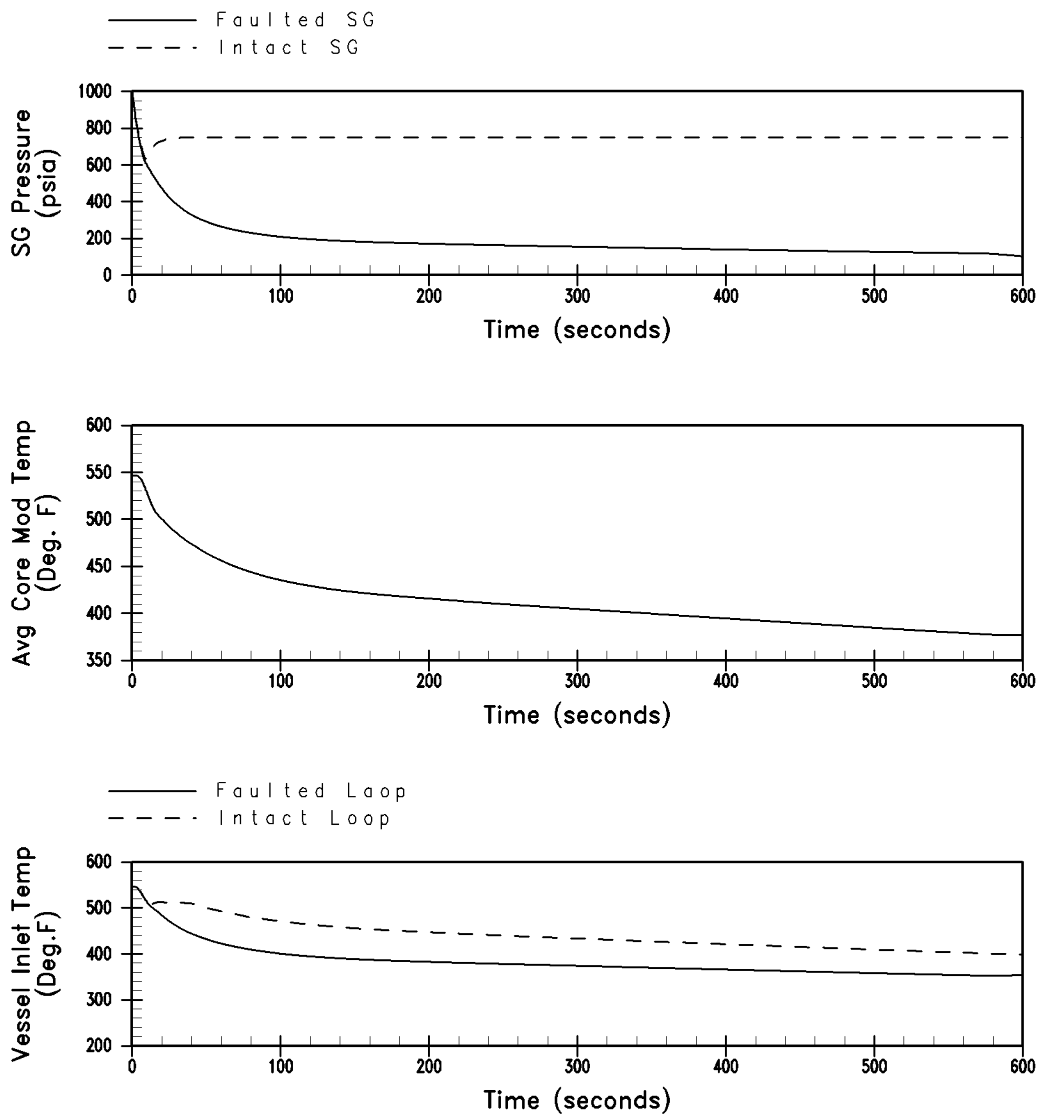

Figure 15.1-18

1.069 ft^2 Steamline Rupture at Hot Zero Power with Offsite Power Available – SG Pressure, Average Core Moderator Temperature and Vessel Inlet Temperature Transients

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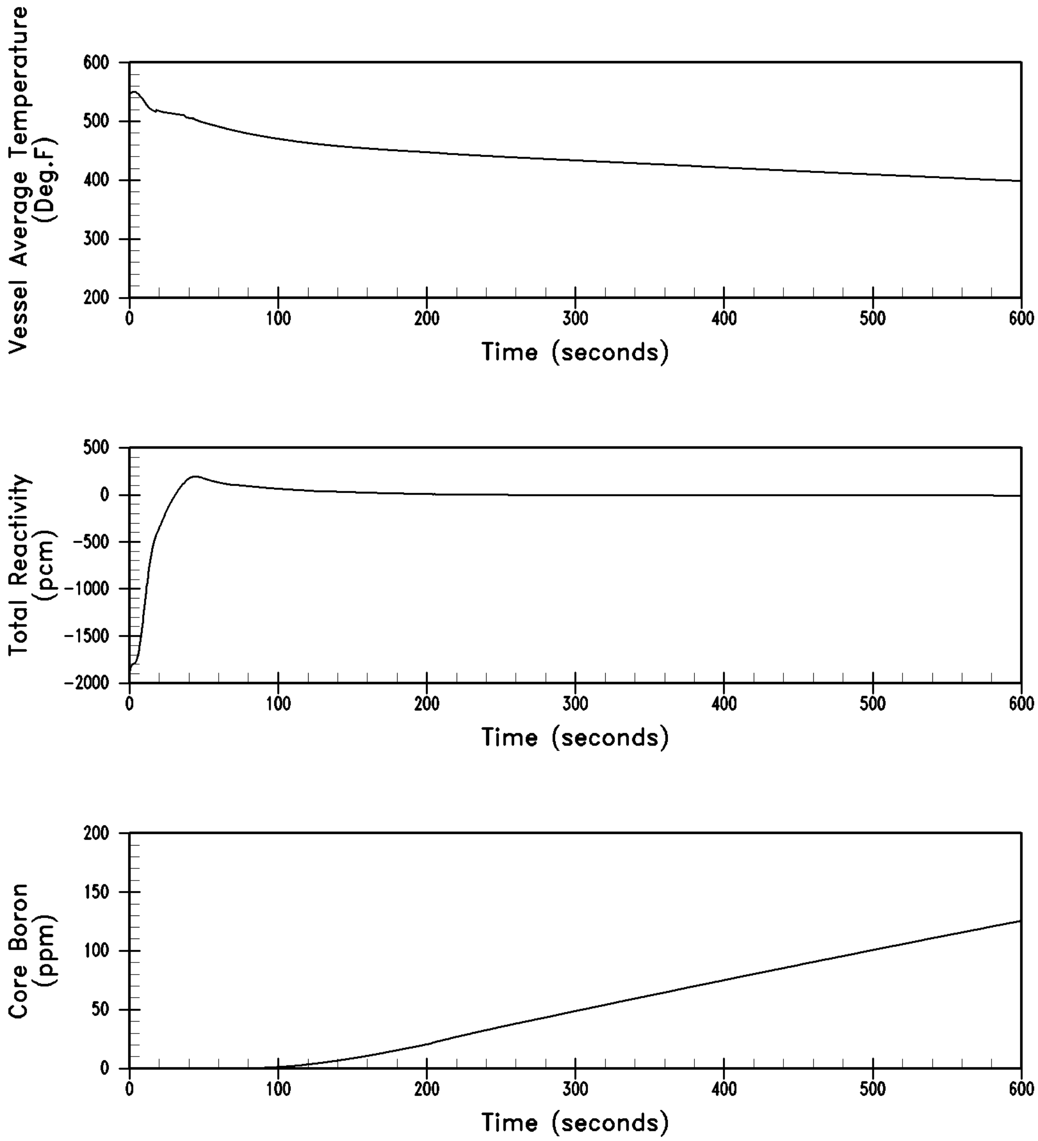

Figure 15.1-19

1.069 ft^2 Steamline Rupture at Hot Zero Power with Offsite Power Available – Vessel Average Temperature, Reactivity and Core Boron Transients

Beaver Valley Power Station Unit No. 2

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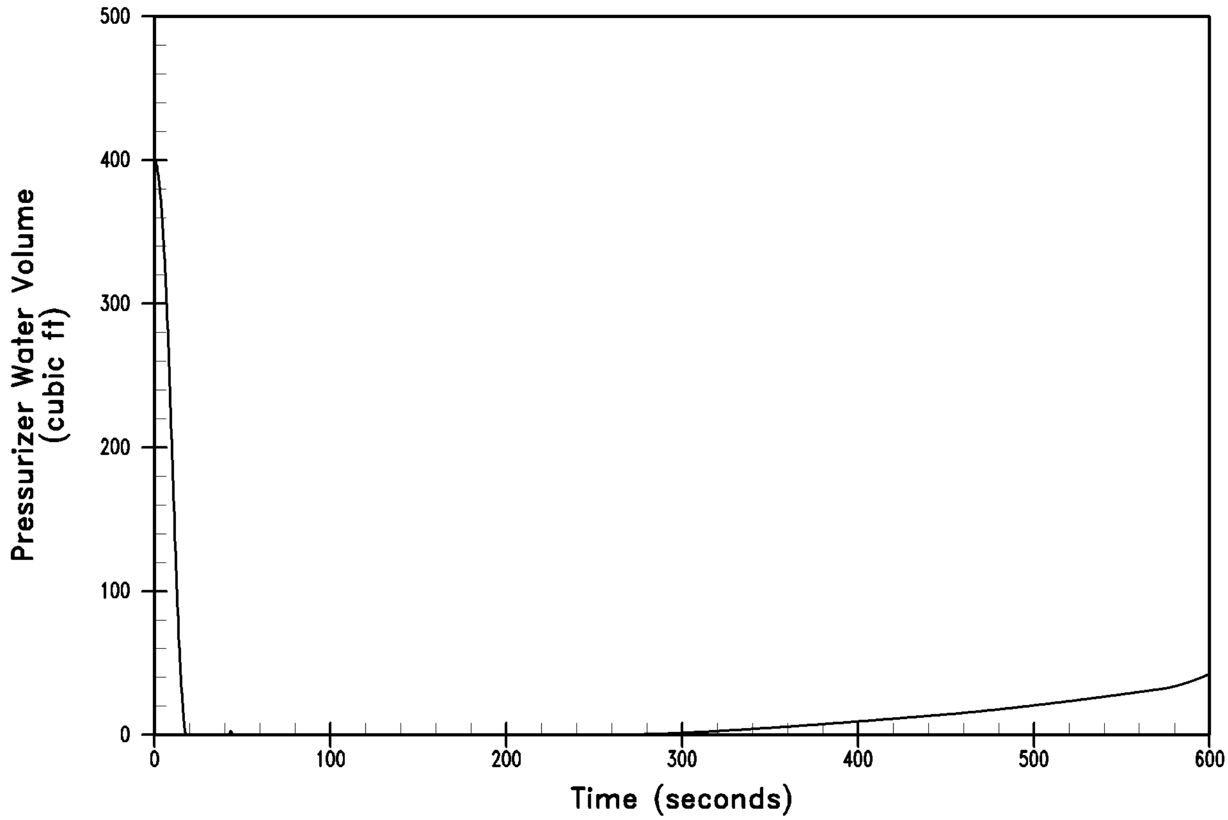

Figure 15.1-20

1.069 ft^2 Steamline Rupture at Hot Zero Power with Offsite Power Available – Pressurizer Water Volume Transient

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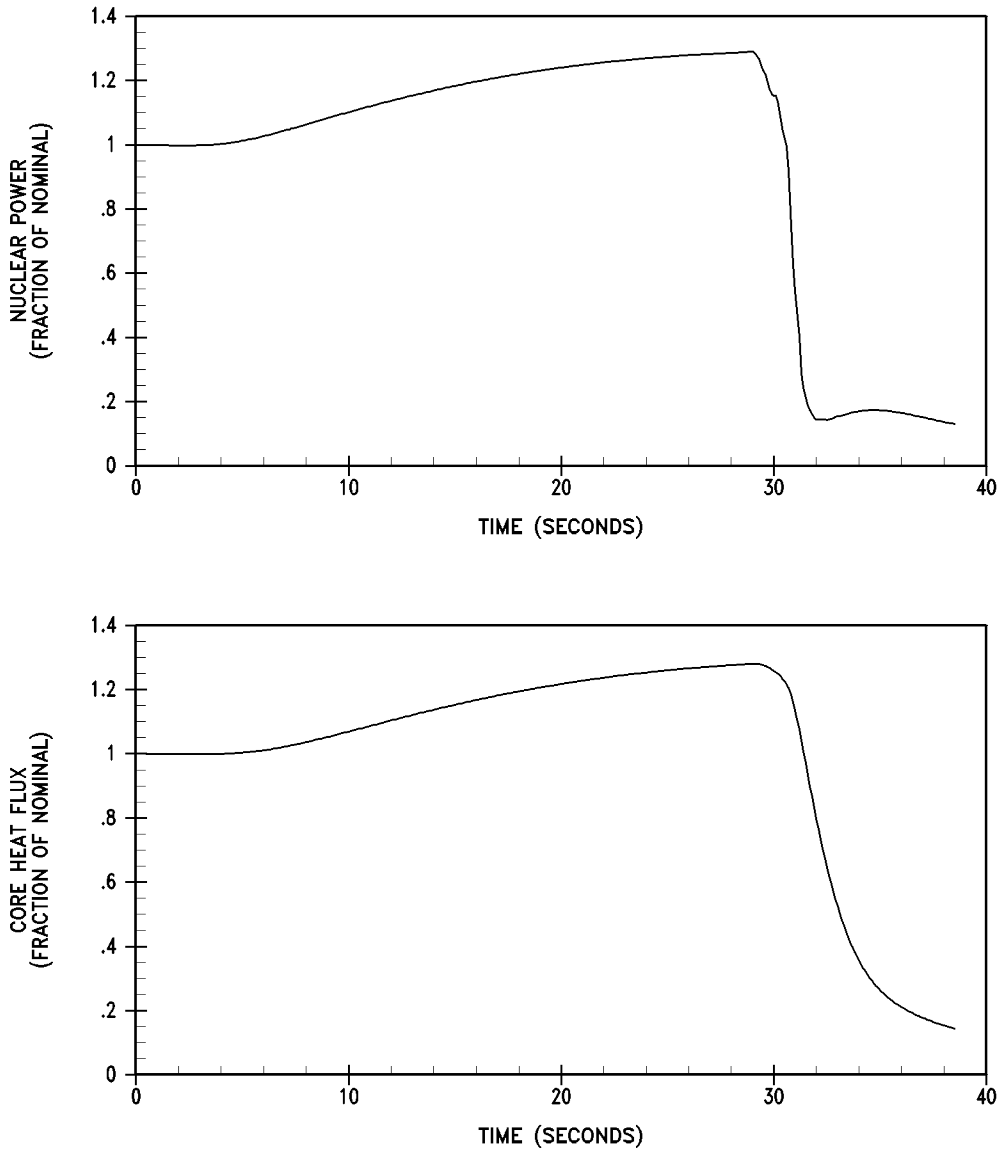

Figure 15.1-21

Nuclear Power and Core Average Heat Flux Transients for Steam System Piping Failure At Power – 0.8 ft^2 Break

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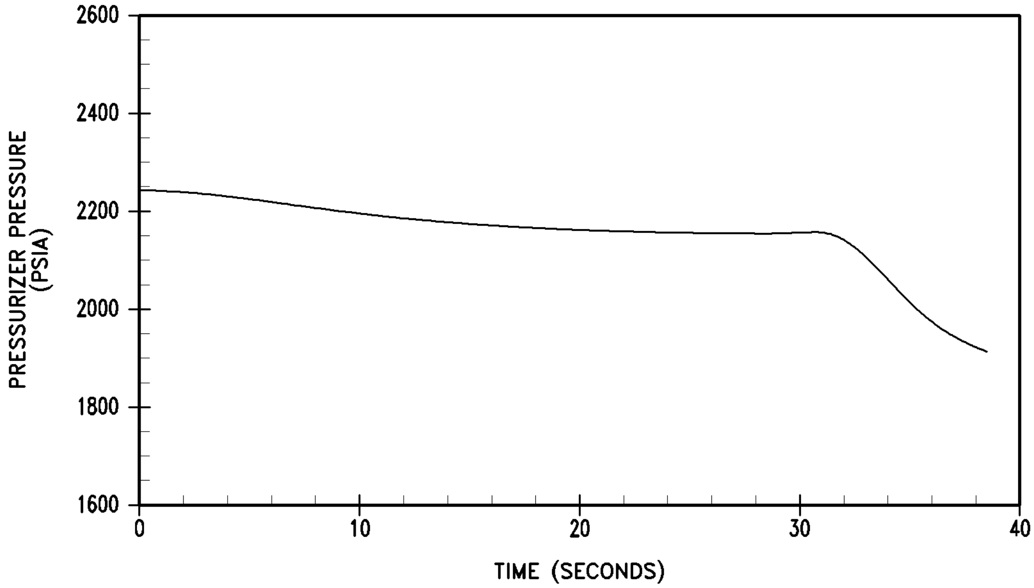

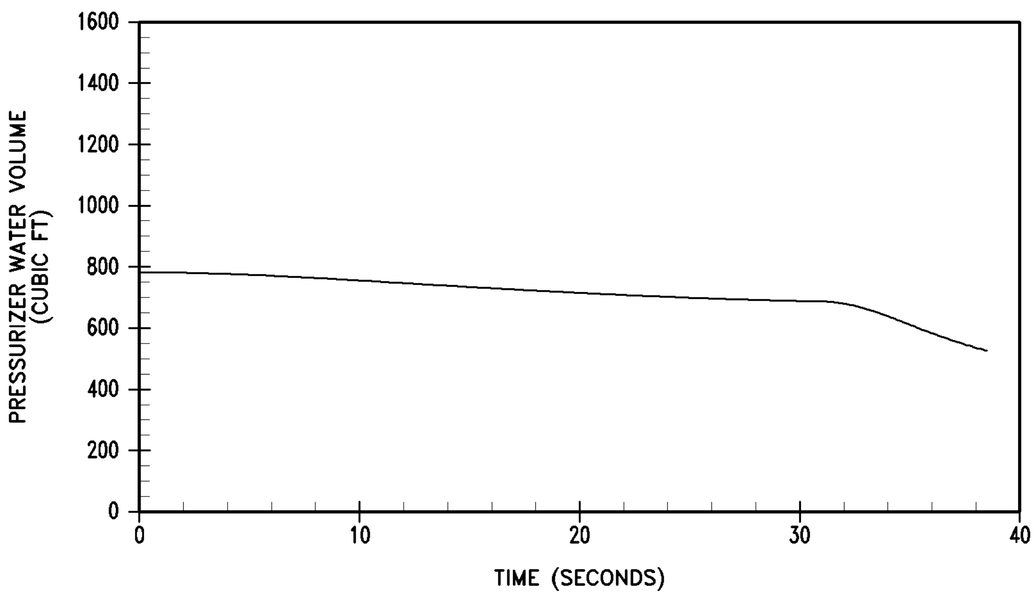

Figure 15.1-22

Pressuizer Pressure and Pressurizer Water Volume Transients for Steam System Piping Failure At Power – 0.8 ft^2 Break

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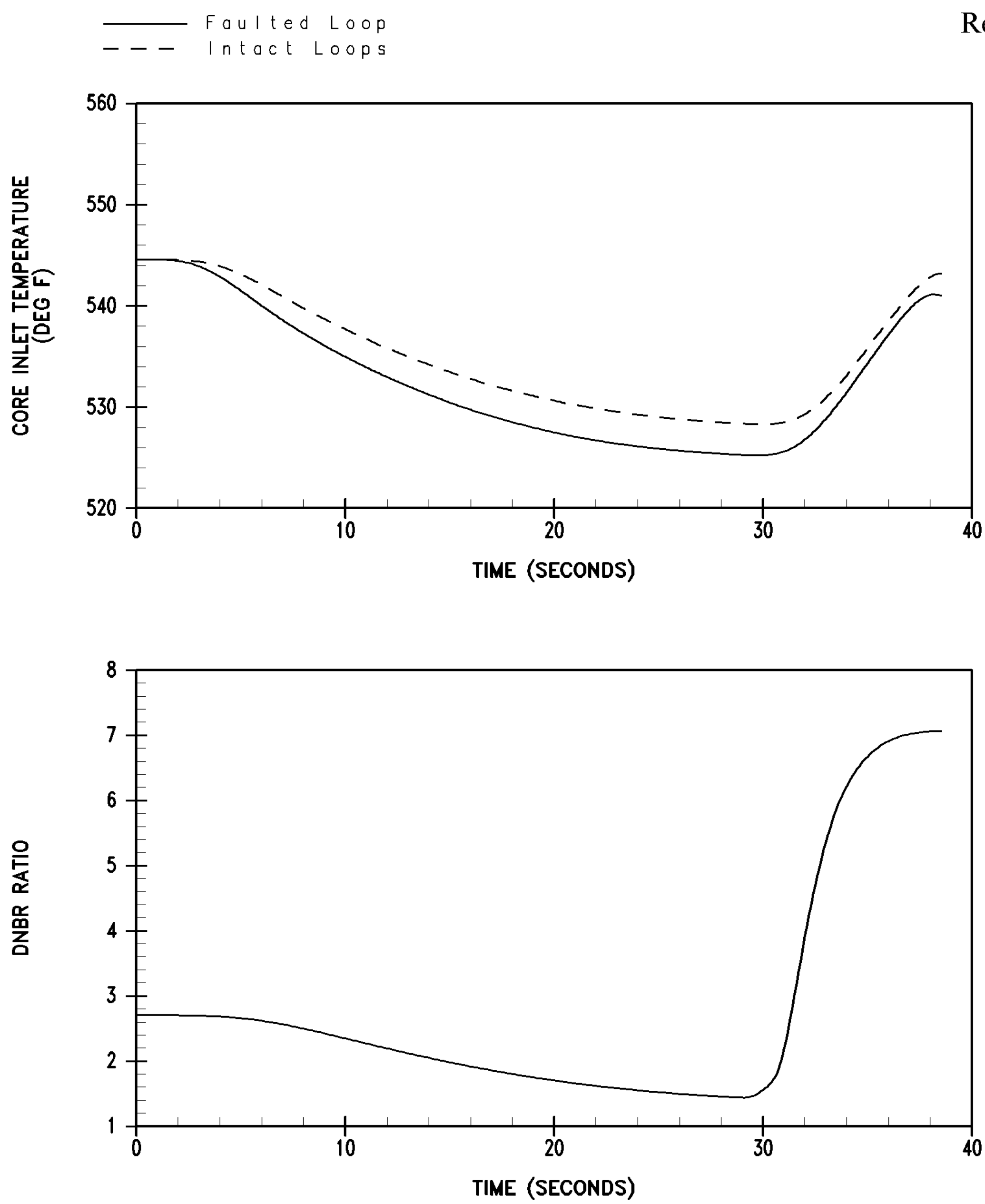


Figure 15.1-23

Core Inlet Temperature Transient and DNB Ratio versus Time for Steam System Piping Failure At Power – 0.8 ft^2 Break

Beaver Valley Power Station Unit No. 2 Updated Final Safety Analysis Report

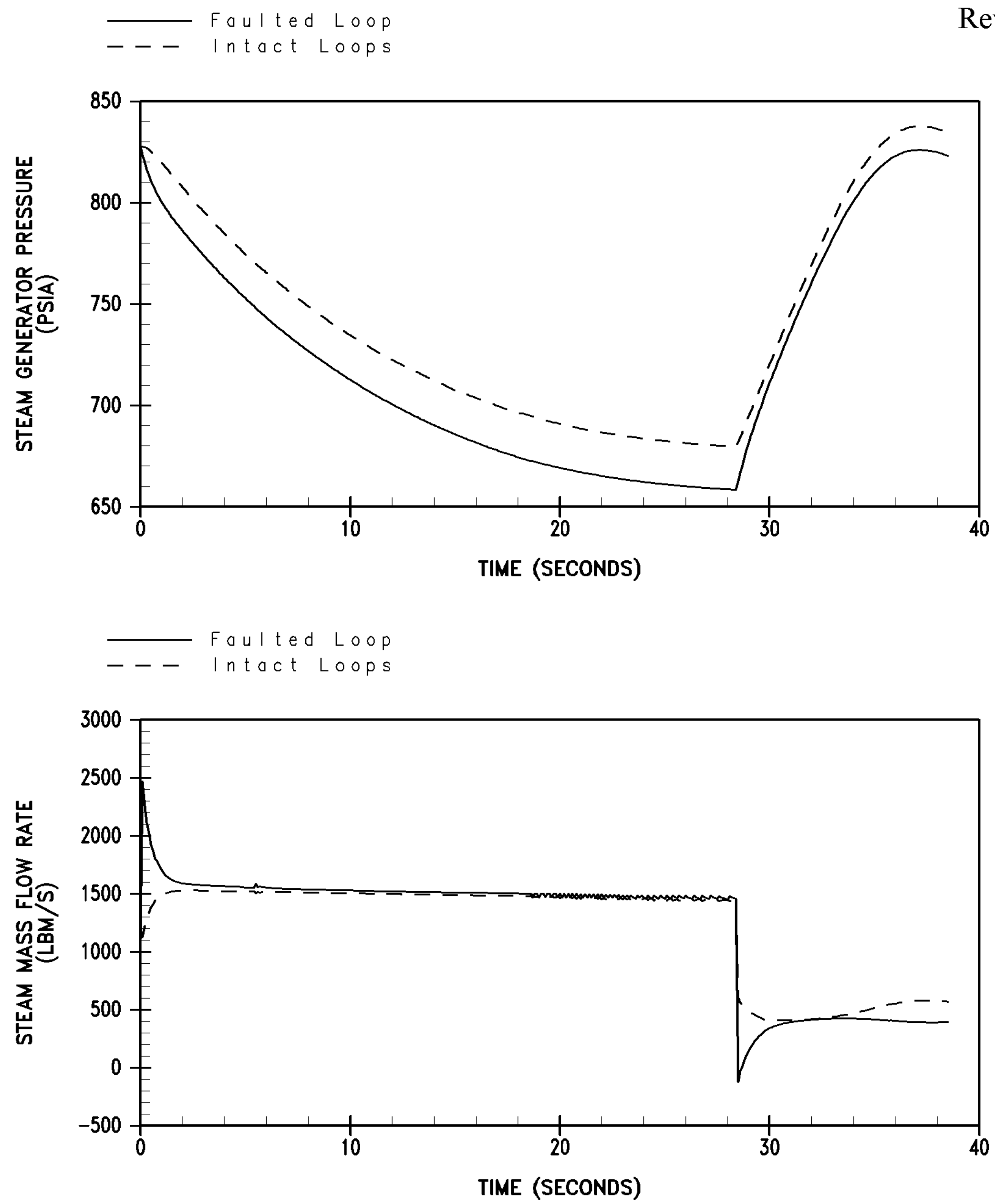


Figure 15.1-24

Steam Generator Pressure and Steam Mass Flow Transients for Steam System Piping Failure At Power – 0.8 ft^2 Break

Beaver Valley Power Station Unit No. 2 Updated Final Safety Analysis Report

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). Detailed analyses are presented in this section 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 the following sections:

1. Steam pressure regulator malfunction or failure that results in decreasing steam flow (Section 15.2.1).

2. Loss of external electrical load (Section 15.2.2).

3. Turbine trip (Section 15.2.3).

4. Inadvertent closure of main steam isolation valves (MSIVs) (Section 15.2.4).

5. Loss of condenser vacuum and other events resulting in turbine trip (Section 15.2.5).

6. Loss of nonemergency ac power to the Beaver Valley Power Station - Unit 2 (BVPS-2) auxiliaries (loss of offsite power) (LOOP) (Section 15.2.6).

7. Loss of normal feedwater flow (Section 15.2.7).

8. Feedwater system pipe break (Section 15.2.8).

These items are considered to be American Nuclear Society (ANS) Condition II events, with the exception of a feedwater system pipe break, which is considered to be an ANS Condition IV event.

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

Any steam flow decrease caused by a malfunction or failure of any steam pressure regulator is conservatively bounded by the turbine trip event analyzed in Section 15.2.3.

15.2.2 Loss of External Electrical Load

15.2.2.1 Identification of Causes and Accident Description

A loss of external electrical load may occur due to some electrical system disturbance. Offsite ac power remains available to operate BVPS-2 components such as the reactor coolant pumps (RCPs); as a result, the onsite emergency generators are not required to function for this event. Following the loss of the turbine 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, below the P-4 setpoint a reactor trip signal may be generated. The actual response of the plant may or may not include a reactor trip depending on the configuration. Beaver Valley Power Station - Unit 2 would be expected to trip from the reactor protection system (RPS) if a safety limit were approached. If sufficient steam dump capacity exists, however, it is not expected that the reactor would trip due to such a load rejection. A continued steam load of approximately 5 percent would exist after total loss of external electrical load because of the steam demand of BVPS-2 auxiliaries.

In the event that a safety limit is approached, protection would be provided by the high pressurizer pressure and overtemperature ΔT trip. Following a complete loss of load, the maximum turbine overspeed would be approximately 8 to 9 percent, resulting in an overfrequency of less than 6 Hz. Any increased frequency to the RCP motors will result in a corresponding increase in flow rate and subsequent additional margin to safety limits. For postulated loss of load and subsequent turbine generator overspeed, any overfrequency condition will not affect other safety-related pump motors, RPS equipment, or other safeguard loads. Safeguard loads are supplied from offsite power or, alternatively, from emergency diesel generators. The RPS equipment is supplied from 118 V ac vital instrument power supply system, which in turn is supplied from the inverters. The inverters are supplied from a Class 1E 125 V dc bus energized from batteries or by a rectified Class 1E ac voltage from safeguard buses.

In the event there is insufficient steam dump capacity (either condenser or atmospheric) 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 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 (PORVs), automatic rod control, or direct trip on turbine trip.

The pressurizer safety valves and steam generator safety valves are able to relieve sufficient steam to maintain the RCS pressure within 110 percent of the RCS design pressure.

Mangan (1972) presents a complete discussion of overpressure protection.

A loss of external load is classified as an ANS Condition II event, a fault of moderate frequency (Section 15.0.1).

A loss of external load event results in a nuclear steam supply system (NSSS) transient that is less severe than a turbine trip event (Section 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 the 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. Therefore, the transient in reactor coolant pressure, temperature and water volume will be less severe for the loss of external load than for the turbine trip due to a slightly slower loss of heat transfer capability.

The protection available to mitigate the consequences of a loss of external load is the same as that for a turbine trip, as listed in Table 15.0-6.

15.2.2.2 Analysis of Effects and Consequences

Method of Analysis

Section 15.2.3.2 discusses the method used to analyze the limiting transient (turbine trip) in this grouping of events. The results of the turbine trip event analysis are more severe than those expected for the loss of external load as discussed in Section 15.2.2.1.

Normal reactor control systems and engineered safety feature (ESF) systems are not required to function during a loss of external load. The auxiliary feedwater system (AFWS) may be automatically actuated following a trip of the main feedwater pumps, however, no credit is taken for the AFWS in the loss of external load transient analysis.

The RPS may be required to function following a complete loss of external load to terminate core heat input and prevent a violation of the departure from nucleate boiling ratio (DNBR) limit. 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. A discussion of anticipated transients without scram (ATWS) considerations is found in Section 15.8.

15.2.2.3 Radiological Consequences

Loss of external load from full power would result in the operation of the turbine bypass system (TBS). This system keeps the main turbine generator operating to supply auxiliary electrical loads. Operation of the TBS results in bypassing steam to the condenser. If TBS is not available, steam generator safety and relief valves relieve to the atmosphere. Since no fuel damage is postulated for this transient, the radiological releases are less severe than those for the loss of nonemergency ac power to the station auxiliaries (Section 15.2.6).

15.2.2.4 Conclusions

Based on results obtained for the turbine trip event (Section 15.2.3) and considerations described in Section 15.2.2.1, the applicable acceptance criteria for a loss of external load event are met. The radiological consequences of this event are not limiting.

15.2.3 Turbine Trip

15.2.3.1 Identification of Causes and Accident Description

For a turbine trip event, the reactor would be tripped directly (unless below approximately 49% reactor power) by a signal derived from the turbine auto-stop oil pressure and turbine stop valves. The turbine stop valves close rapidly on loss of trip-fluid pressure actuated by one of a number of turbine trip signals. Turbine trip initiation signals include:

1. Low condenser vacuum,
2. Low bearing oil pressure,
3. Turbine thrust bearing failure,
4. Turbine overspeed,
5. Electro hydraulic dc power failure,
6. Manual trip,
7. Generator trip (remote), and
8. AMSAC

Upon initiation of stop valve closure, steam flow to the turbine stops abruptly. Sensors on the stop valves detect the turbine trip and if above approximately 49% reactor power, trip the reactor. The loss of steam flow results in a rapid rise in secondary system temperature and pressure with a resultant primary system transient as described in Section 15.2.2.1 for the loss of external load event. The turbine trip event is analyzed because it results in the most rapid reduction in steam flow.

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 is not available, the excess steam generation would be dumped to the atmosphere. Feedwater flow would be maintained by the AFWS 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. Section 15.2.2.1 provides further discussion of the transient.

A turbine trip is classified as an ANS Condition II event, a fault of moderate frequency (Section 15.0.1).

The BVPS-2 systems and equipment available to mitigate the consequences of a turbine trip are discussed in Section 15.0.8, and listed in Table 15.0-6.

15.2.3.2 Analysis of Effects and Consequences

<u>Method of Analysis</u>

In this analysis, the behavior of BVPS-2 is evaluated for a complete loss of steam load from full power without direct reactor trip primarily to show the adequacy of the pressure relieving devices and also to demonstrate core protection margins; that is, the turbine is assumed to trip without actuating all the sensors for reactor trip on the turbine stop valves and auto stop oil pressure. The assumption delays reactor trip until conditions in the RCS result in a trip due to other signals. Thus, the analysis assumes a worst case transient. In addition, 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 (Burnett, et al. 1984).

Initial operating conditions are assumed at values consistent with steady-state operation. Plant characteristics and initial conditions are discussed in Section 15.0.3.

Major assumptions are summarized as follows:

1. Initial operating conditions

   Two cases have been considered -- one to demonstrate the adequacy of the pressure relieving devices and the other to demonstrate that the Departure from Nucleate Boiling Ratio (DNBR) limit is not violated. In the pressure case, the initial reactor power and RCS temperatures are assumed to be at their maximum values consistent with steady-state full power operation including allowances for calibration and instrument errors. The initial RCS pressure is assumed to be at a minimum value consistent with steady-state full power operation including allowances for calibration and instrument errors. The RCS flow rate assumed is the Thermal Design Flow (TDF).

   For the DNB case, the initial reactor power and RCS temperatures are assumed to be at their nominal full power values. The initial RCS pressure is assumed to be at its nominal full power value, minus a pressure bias. The RCS flow rate assumed is the Minimum Measured Flow (MMF). This is consistent with the Revised Thermal Design Procedure (RTDP) which is discussed in WCAP-11397-A (Friedland, April 1989).

2. Moderator and Doppler coefficients of reactivity

   The turbine trip is analyzed with a least negative Doppler power coefficient (Figure 15.0-2). The moderator temperature coefficient assumed is +5 pcm/°F, which conservatively bounds the maximum allowable full power moderator temperature coefficient of 0 pcm/°F.

3. Reactor control

   From the standpoint of the maximum pressures attained, 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.

4. Steam release

   No credit is taken for the operation of the condenser steam dump system or steam line ADVs. The steam generator pressure rises to the safety valve set points where steam release through safety valves limits secondary steam pressure at the set point value.

5. Pressurizer spray and PORVs

The following two cases are analyzed:

DNBR Case (a):

Full credit is taken for the effect of pressurizer spray and PORVs in reducing or limiting the coolant pressure. Safety valves are also available.

Pressure Case (b):

No credit is taken for the effect of pressurizer spray and PORVs in reducing or limiting the coolant pressure. Safety valves are operable.

6. 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 BVPS-2 condition will be reached before auxiliary feedwater initiation is normally assumed to occur.

The auxiliary feedwater flow would remove core decay heat following BVPS-2 stabilization.

7. Reactor trip

Reactor trip is actuated by the first RPS trip set point reached with no credit taken for the direct reactor trip on the turbine trip. Trip signals are expected due to high pressurizer pressure, overtemperature ΔT, or high pressurizer water level. Only the overtemperature ΔT and high pressurizer pressure trips are credited in the analysis.

Except as discussed previously, the normal reactor control system and engineered safety systems are not required to function. In the DNBR case (Case a.) the pressurizer sprays and PORVs are assumed to operate, while in the pressure case (Case b.) they are not since this yields more limiting RCS pressure results.

The RPS may be required to function following a turbine trip. Opening of pressurizer safety valves and/or steam generator safety valves may be required to maintain system pressures below allowable limits. No single active failure will prevent operation of any system required to function. A discussion of ATWS considerations is presented in Section 15.8.

Results

The transient responses for a turbine trip from full power operation are presented for two cases: one case with pressurizer pressure control and one case without pressure control. Both cases assume minimum reactivity feedback.

In the case with pressurizer pressure control, pressurizer sprays and pressurizer PORVs are modeled. This case is analyzed to demonstrate that the DNBR limit is met and, for this case, minimizing RCS pressure is conservative. The transient responses for this case are shown in Figures 15.2-1, 15.2-2, 15.2-3 and 15.2-4. No credit is taken for the steam dump system. The reactor is tripped by the High Pressurizer Pressure trip signal. The minimum departure from nucleate boiling ratio (DNBR) remains well above the applicable limit value. Pressurizer PORVs and safety valves prevent overpressurization of the primary system. The steam generator safety valves also actuate to maintain the secondary system pressure below 110 percent of the design value.

In the event that feedwater flow is not terminated at the time of turbine trip for this case, 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 was taken by the operator, the reduced power operation would continue until the condenser hotwell was 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. The results would be less severe than those presented in Section 15.2.7, Loss of Normal Feedwater Flow.

The turbine trip accident is also analyzed assuming BVPS-2 to be initially operating at 100.6 percent of full power with no credit taken for the pressurizer spray, pressurizer PORVs or steam dump. In this case, the reactor is tripped on the high pressurizer pressure signal. The transient responses for this case are shown in Figures 15.2-5 through 15.2-8. The nuclear power and core heat flux remain essentially constant until the reactor is tripped. In this case, the pressurizer safety valves are actuated and maintain the primary system pressures below 110 percent of the design value. The steam generator safety valves also actuate to maintain the secondary system pressure below 110 percent of the design value.

Following reactor trip, BVPS-2 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 RCS boron concentration and pressurizer level using the chemical and volume control system (CVCS), and to maintain steam generator level through control of the main feedwater system or AFWS. Any action required of the operator to maintain BVPS-2 in a stabilized condition will be in a time frame in excess of ten minutes following reactor trip.

Mangan (1972) presents additional results of analysis for 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 Section 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 atmospheric steam dump are less severe than those of the loss of nonemergency ac power to the station auxiliaries (Section 15.2.6).

15.2.3.4 Conclusions

Results of the analyses, including those in Mangan (1972), show that the BVPS-2 design is such that a turbine trip, without a direct or immediate reactor trip, presents no hazard to the integrity of the RCS or the main steam system (MSS). Pressure relieving devices incorporated in the two systems are adequate to limit the maximum pressures to within the design limits.

The DNBR remains above the limit value for all cases analyzed; thus, the DNB design basis as described in Section 4.4 is met.

The preceding analysis demonstrates the ability of the NSSS to safely withstand a full load rejection. The radiological consequences of this event are not limiting.

15.2.4 Inadvertent Closure of Main Steam Isolation Valves

The inadvertent closure of the MSIVs would result in a turbine trip and other consequences as discussed in Section 15.2.5.

15.2.5 Loss of Condenser Vacuum and Other Events Resulting in Turbine Trip

Loss of condenser vacuum is one of the events that can cause a turbine trip. Turbine trip initiating events are described in Section 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 conclusions contained in Section 15.2.3 apply to loss of condenser vacuum. In addition, analyses for the other possible causes of a turbine trip, as listed in Section 15.2.3.1, are covered by Section 15.2.3. Possible overfrequency effects due to a turbine overspeed condition are discussed in Section 15.2.2.1 and are not a concern for this type of event.

15.2.6 Loss of Non-emergency AC Power to the Plant Auxiliaries (Loss of Offsite Power)

15.2.6.1 Identification of Causes and Accident Description

A complete loss of non-emergency AC power shall result in the loss of all power to the non-essential station auxiliaries, that is, the RCPs, 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 BVPS-2.

This transient is more severe than the turbine trip event analyzed in Section 15.2.3, because for this case the decrease in heat removal by the secondary is accompanied by a flow coastdown which further reduces the capacity of the primary coolant to remove heat from the core.

Following a loss of AC power with turbine and reactor trips, the following sequence will occur:

1. Plant vital instruments are supplied from emergency DC power sources.

2. As the steam system pressure rises following the trip, the steam line ADVs may be automatically opened to the atmosphere. Steam dump to the condenser is assumed not to be available. If the steam flow rate through ADVs is not available, the steam generator self-actuated safety valves may lift to dissipate the sensible heat of the fuel and coolant plus the residual decay produced in the reactor.

3. As the no-load temperature is approached, the steam line ADVs (or the safety valves, if the ADVs are not available) are used to dissipate the residual decay heat and to maintain BVPS-2 at the hot shutdown condition.

4. The emergency diesel generators, started on loss of voltage to the BVPS-2 emergency buses, begin to supply BVPS-2 Class lE loads.

The AFWS is started automatically as described in Section 7.3.2.

The motor driven auxiliary feedwater pumps are supplied by power from the Class 1E buses. The turbine-driven auxiliary feedwater pump is driven by steam from the secondary system and exhausts to the atmosphere. Both types of pumps are designed to start and supply rated flow within one minute of the initiating signal. The auxiliary pumps take suction from the primary plant demineralized water storage tank (PPDWST) for delivery to the steam generators.

Upon the loss of power to the RCPs, 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 non-emergency AC power to the station auxiliaries is classified as an ANS Condition II event, a fault of moderate frequency. Section 15.0.1 discusses Condition II events.

A loss of non-emergency AC power event is more limiting than the turbine-trip-initiated decrease in secondary heat removal without loss of AC power, which was analyzed in Section 15.2.3. However, a loss of AC power to the BVPS-2 auxiliaries as postulated previously also results in a loss of normal feedwater since the condensate pumps lose their power supply.

Following the RCP coastdown caused by the loss of AC power, the natural circulation capability of the RCS will remove residual 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.

The BVPS-2 systems and equipment available to mitigate the consequences of a loss of ac power event are discussed in Section 15.0.8, and listed in Table 15.0-6.

15.2.6.2 Analysis of Effects and Consequences

Method of Analysis

A detailed analysis using the LOFTRAN Code (Burnett et al 1984) is performed to obtain the BVPS-2 transient following a loss of AC power event.

The assumptions used in the analysis are as follows:

1. The plant is initially operating at 100.6 percent of the nominal NSSS power (100.6% of 2,910 MWt).

2. A conservative core residual heat generation based upon long term operation at the initial power level preceding the trip.

3. A heat transfer coefficient in the steam generator associated with RCS natural circulation, following the RCP coastdown.

4. Reactor trip occurs on steam generator low-low level (0% Narrow Range Span). No credit is taken for immediate release of the control rod drive mechanisms (CRDM) caused by a LOOP.

5. Auxiliary feedwater is delivered by two motor driven auxiliary feed pumps to three steam generators.

6. Secondary system steam relief is achieved through the steam generator safety valves.

7. The pressurizer PORVs and pressurizers heaters are assumed to function.

8. The initial reactor coolant average temperature is 8.5°F higher than the nominal value.

9. The initial pressurizer pressure is 45 psi lower than the nominal value.

The assumptions used in the analysis are similar to the loss of normal feedwater flow incident (Section 15.2.7) except that power is assumed to be lost to the RCPs at the time of reactor trip.

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

Results

The transient response of the RCS following a loss of ac power is shown on Figures 15.2-9, 15.2-10, 15.2-11, 15.2-12 and 15.2-13 for three loops initially in operation.

The first few seconds after the loss of power to the RCPs will closely resemble the complete loss of flow incident (Section 15.3.2), that is, 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. The LOFTRAN results show that the natural circulation flow available is sufficient to provide adequate core decay heat removal following reactor trip and RCP coastdown.

The calculated sequence of events for this accident is listed in Table 15.2-1.

15.2.6.3 Radiological Consequences

A loss of nonemergency ac power (LACP) to BVPS-2 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 relief valves or safety valves.

No fuel damage is postulated to occur from this transient. Table 15.2-2 lists the key parameters associated with the LACP. Table 15.3-3 lists the key assumptions and parameters utilized to develop the radiological consequences following a Locked Rotor Accident. The transport models associated with the two events are similar with the exception that the locked rotor event results in fuel damage and associated release of gap activity, whereas the LACP has no fuel damage, and the maximum release is associated with Technical Specification concentrations. Since the reactor coolant Technical Specification activity is significantly smaller than the gap activity associated with failed fuel, it is concluded that the dose consequences of the locked rotor bound that of the LACP. As shown in Tables 15.0-12 and 15.0-13, the dose consequences of the loss of non-emergency ac power are within a small fraction of the regulatory dose limits provided in 10 CFR 50.67.

15.2.6.4 Conclusions

Analysis of the natural circulation capability of the RCS has demonstrated that sufficient heat removal capability exists following RCP coastdown to prevent fuel or clad damage. The radiological consequences of this event are within a small fraction of the regulatory dose limits provided in 10 CFR 50.67.

15.2.7 Loss of Normal Feedwater Flow

15.2.7.1 Identification of Causes and Accident Description

A loss of normal feedwater flow (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 BVPS-2, 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 RCS. Since BVPS-2 is tripped well before the steam generator heat transfer capability is reduced, the primary system variables never approach a DNB condition.

In addition to the primary means of auxiliary feedwater initiation, a backup system has been implemented to meet the requirements of 10CFR 50.62(c)(1), Diverse and Independent Auxiliary Feedwater Initiation and Turbine Trip for PWR's (AMSAC). This system provides an automatic start signal for the motor driven and steam driven auxiliary feedwater pumps and a turbine trip signal after a predetermined time delay.

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

1. As the steam system pressure rises following the trip, the steam line ADVs are automatically opened to the atmosphere. Steam dump to the condenser is assumed not to be available. If the steam flow through the PORVs 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.

2. As the no-load temperature is approached, the steam line ADVs (or the safety valves, if the ADVs are not available) are used to dissipate the residual decay heat and to maintain BVPS-2 at the hot shutdown condition.

A loss of normal feedwater is classified as an ANS Condition II event, a fault of moderate frequency.

Reactor trip on low-low water level in any steam generator provides protection for a loss of normal feedwater.

The AFWS is started automatically as discussed in Section 15.2.6.1. The motor-driven auxiliary feedwater pumps are supplied by power from the Class 1E buses. The turbine-driven auxiliary feedwater pump is driven by steam from the secondary system and exhausts to the atmosphere. The pumps take suction directly from the primary plant DWST for delivery to the steam generators.

An analysis of the system transient is presented as follows to show that following a loss of normal feedwater, the AFWS 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 BVPS-2 to a safe condition.

15.2.7.2 Analysis of Effects and Consequences

Method of Analysis

A detailed analysis using the LOFTRAN Code (Burnett et al 1984) is performed in order to obtain BVPS-2 transient following a loss of normal feedwater.

Assumptions made in the analysis are:

1. The plant is initially operating at 100.6 percent of the nominal NSSS power (100.6% of 2,910 MWt).

2. A conservative core residual heat generation based upon long term operation at the initial power level preceding the trip.

3. Reactor trip occurs on steam generator low-low level (0% NRS).

4. Auxiliary feedwater is delivered by two motor-driven auxiliary feed pumps to three steam generators.

5. Secondary system steam relief is achieved through the steam generator safety valves.

6. The pressurizer relief valves and pressurizer heaters are assumed to function.

7. The initial reactor coolant average temperature is 8.5°F higher than the nominal value.

8. The initial pressurizer pressure is 45 psi lower than the nominal value.

The loss of normal feedwater analysis is performed to demonstrate the adequacy of the RPS and engineered safeguards systems (for example, AFWS) 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 previously.

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 assumptions used in the analysis are similar to the loss of ac power incident, (Section 15.2.6), except that the RCPs are assumed to continue to operate.

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

Beaver Valley Power Station - Unit 2 systems and equipment which are available to mitigate effects of a loss of normal feedwater accident are discussed in Section 15.0.8 and listed in Table 15.0-6. Normal reactor control systems are not required to function. The RPS is required to function following a loss of normal feedwater as analyzed here. The AFWS is required to deliver a minimum auxiliary feedwater flow rate. No single active failure will prevent operation of any system required to function. A discussion of ATWS considerations is presented in Section 15.8.

Results

Figures 15.2-14, 15.2-15, 15.2-16, 15.2-17 and 15.2-18 show the significant BVPS-2 parameters following a loss of normal feedwater with three loops initially in operation.

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. One minute following the initiation of the low-low level trip, the auxiliary feedwater pumps are automatically started, reducing the rate of water level decrease.

The calculated sequence of events for this accident is listed in Table 15.2-1.

As shown on Figures 15.2-14, 15.2-15 and 15.2-16, BVPS-2 will slowly approach a stabilized condition at hot standby with auxiliary feedwater removing decay heat. The plant may be maintained at hot standby or further cooled through manual control of the auxiliary feed flow. The operating procedures would also 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 AFWS. Any action required of the operator to maintain BVPS-2 in a stabilized condition will be in a time frame in excess of ten minutes following reactor trip.

15.2.7.3 Radiological Consequences

The steam release and resulting radiological consequences from this transient would be the same as those for the loss of nonemergency ac power.

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 since the auxiliary feedwater capacity is such that the RCS does not overpressurize. The radiological consequences of this event are within the limits described by 10 CFR 20.

15.2.8 Feedwater System Pipe Break

15.2.8.1 Identification of Causes and Accident Description

A major feedwater line rupture 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 NSSS only as a loss of feedwater. This case is covered by the evaluation in Section 15.2.7.

Depending upon the size of the break and BVPS-2 operating conditions at the time of the break, the break could cause either an RCS cooldown (by excessive energy discharge through the break) or an RCS heatup. Potential RCS cooldown resulting from a secondary pipe rupture is evaluated in Section 15.1.5. Therefore, only the RCS heatup effects are evaluated for a feedwater line rupture. (As indicated in Section 15.0.1.5, there is the potential for overfilling the pressurizer and subsequent water relief through the pressurizer relief and/or safety valves during a feedwater line rupture. If it is assumed that the pressurizer relief valves are not available, then the water relief would be through the pressurizer safety valves that are not designed to pass water. As such, some additional cases are considered to address pressurizer safety valve (PSV) operability concerns. These additional cases are discussed in Section 15.2.8.2.)

A feedwater line rupture reduces the ability to remove heat generated by the core from the RCS for the following reasons:

1. 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.

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

3. The break may be large enough to prevent the addition of any main feedwater after trip.

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

1. No substantial overpressurization of the RCS shall occur, and

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

A major feedwater line rupture is classified as an ANS Condition IV event. Section 15.0.1 discusses Condition IV events.

The severity of the feedwater line rupture 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 rupture is a double-ended rupture of the largest feedwater line. Analyses have been performed at full power with and without LOOP.

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

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

    a. High pressurizer pressure,

    b. Low-low steam generator water level in any steam generator, or

    c. Safety injection signals from any of the following:

        1) 2/3 low steam line pressure in any one loop,

        2) 2/3 high containment pressure (Hi-1).

        Chapter 7 describes the actuation system.

2. An AFWS to provide an assured source of feedwater to the steam generators for decay heat removal in case the main feedwater system is tripped.

15.2.8.2 Analysis of Effects and Consequences

Method of Analysis

A detailed analysis using the LOFTRAN Code (Burnett, et al. 1984) is performed in order to determine the BVPS-2 transient following a feedwater line rupture.

The cases analyzed assume a double-ended rupture of the largest feedwater pipe at full power. Major assumptions made in the analyses are as follows:

1. The plant is initially operating at 100.6 percent of nominal rated power (100.6% of 2,910 MWt).

2. Initial reactor coolant average temperature is 8.5°F above the nominal value to account for uncertainties and loop-to-loop temperature asymmetry, and the initial pressurizer pressure is 45 psi below its nominal value.

3. Initial pressurizer level is at the nominal programmed value plus 7 percent (error); initial steam generator water level is at the nominal value plus 7 percent in the faulted steam generator and at the nominal value minus 10.3 percent in the intact steam generators.

4. No credit is taken for the high pressurizer pressure reactor trip.

5. 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).

6. The full double-ended break area is assumed.

7. A conservative feedwater line break discharge quality is assumed. This minimizes the heat removal capability of the affected steam generator.

8. Reactor trip is assumed to be actuated when the low-low level trip set point of 0% narrow range span in the ruptured steam generator is reached.

9. The AFWS is actuated by the low-low steam generator water level signal. The AFWS is assumed to supply a total of 250 gpm split equally to the two intact steam generators prior to isolation of the faulted steam generator and 400 gpm split equally to the two intact steam generators after isolation of the faulted steam generator. This allows for spillage through the main feedwater line break prior to isolation. A 60-second delay was assumed following the low-low level signal to allow time for start-up of the emergency diesel generators and the auxiliary feedwater pumps. An additional time was assumed before the feedwater lines were purged and the relatively cold (120°F) auxiliary feedwater entered the unaffected steam generators.

10. An operator action time of 15 minutes to isolate the faulted steam generator following the time of reactor trip is assumed.

11. No credit is taken for heat energy deposited in RCS metal during the RCS heatup.

12. No credit is taken for charging or letdown.

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

14. Conservative core residual heat generation is assumed based upon long-term operation at the initial power level preceding the trip.

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

    a. High pressurizer pressure,

    b. High pressurizer level, and

    c. High containment pressure.

16. Asymmetric loop-to-loop RCS flow was considered in this analysis. It was found that the symmetric RCS flow conditions produced more conservative results than the asymmetric RCS flow conditions and are therefore presented herein.

Receipt of a low-low steam generator water level signal in at least two steam generators starts the motor driven auxiliary feedwater pumps, which then deliver auxiliary feedwater flow to the steam generators. The turbine driven auxiliary feedwater pump is started if the low-low steam generator water level signal is reached in at least one steam generator. Similarly, receipt of a low steam line pressure signal in at least one steam line initiates a steam line isolation signal which closes the MSIVs in all steam lines.

Emergency operating procedures following a main feed line rupture require the following actions to be taken.

1. Isolate the feedwater flow spilling out from the break in the ruptured feedwater line and align the system so that the water level in the intact steam generators recovers. The AFWS flow is limited by cavitating venturis in the headers to each steam generator so that a maximum of 310 gpm to each steam generator is maintained.

2. Stop high head safety injection and initiate charging flow.

Isolating feedwater flow through the break allows additional auxiliary feedwater flow to be diverted to the intact steam generators; up to 310 gpm per steam generator may be allowed by the cavitating venturis in the AFWS.

Subsequent to recovery of water level in the intact steam generators, BVPS-2 operating procedures will be followed in cooling the plant to hot shutdown conditions.

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

The RPS is required to function following a feedwater line rupture as analyzed here. No single active failure will prevent operation of this system.

The engineered safety systems assumed to function are the AFWS and the safety injection system (SIS). The turbine driven auxiliary feedwater pump has been assumed to fail; the motor-driven pumps deliver 250 gpm split equally to the two intact steam generators prior to isolation of the faulted steam generator and 400 gpm split equally to the two intact steam generators after isolation of the fault steam generator.

Following the trip of the RCPs, 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 Section 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 Sections 15.3.1 and 15.3.2 for single and multiple RCP trips, respectively.

A detailed description and analysis of the SIS is provided in Section 6.3. The AFWS is described in Section 10.4.9.

Results

Calculated BVPS-2 parameters following a major feedwater line rupture are shown on Figures 15.2-19, 15.2-20, 15.2-21, 15.2-22, 15.2-23, 15.2-24, 15.2-25, 15.2-26, 15.2-27, 15.2-28, 15.2-29, and 15.2-30 for three loops initially in operation. Results for the case with offsite power available and a break size of 0.717 ft^2, are presented on Figures 15.2-19, 15.2-20, 15.2-21, 15.2-22, 15.2-23 and 15.2-24. Results for the case where offsite power is lost and a break size of 1.36 ft^2 are presented on Figures 15.2-25, 15.2-26, 15.2-27, 15.2-28, 15.2-29 and 15.2-30. For the case with offsite power available it was found that the 0.717 ft^2 break size was more limiting than the 1.36 ft^2 break size. However, for the case where offsite power is lost, the 1.36 ft^2 break size is found to be more limiting than the 0.717 ft^2 break size. The calculated sequence of events for both cases analyzed are listed in Table 15.2-1.

The system response following the feedwater line rupture is similar for both cases analyzed. Results presented on Figures 15.2-19, 15.2-20, 15.2-21, 15.2-22, 15.2-23 and 15.2-24 (with offsite power available) and Figures 15.2-25, 15.2-26, 15.2-27, 15.2-28, 15.2-29 and 15.2-30 (without offsite power) show that pressures in the RCS and MSS remain below 110 percent of the respective design pressures. Pressurizer pressure increases until reactor trip occurs on low steam generator water level. Pressure then decreases, due to the loss of heat input, until the time at which the mass inventory in the intact steam generators is not sufficient to remove the core decay heat, and until steam line isolation and safety injection actuation occur.

Addition of the safety injection flow aids in cooling down the primary system and helps to ensure that sufficient fluid exists to keep the core covered with water.

The reactor core remains covered with water throughout the transient, water relieved due to thermal expansion is limited by the heat removal capability of the AFWS, and makeup is provided by the high head safety injection pumps. The pressurizer does not fill in either case.

The major difference between the two cases analyzed can be seen in the plots of hot- and cold-leg temperatures, Figures 15.2-21 and 15.2-22 (with offsite power available) and Figures 15.2-27 and 15.2-28 (without offsite power). It is apparent that for the initial transient (300 seconds), the case without offsite power results in higher temperatures in the hot-leg. For longer times, however, the case with offsite power results in a more severe rise in temperature due to the addition of pump heat.

Figure 15.2-20 (with offsite power available) and Figure 15.2-26 (without offsite power available) show that the pressurizer does not become water solid; thus, there is no water relief through the pressurizer relief valves.

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. Since no fuel failures are postulated, the radioactivity released is less than that for the main steam line break (MSLB), Section 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 rupture, AFWS 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 rupture would be less than those previously presented for the postulated MSLB.

15.2.9 References for Section 15.2

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

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

Friedland, A. J., and Ray, S., "Revised Thermal Design Procedure," WCAP-11397-P-A (Proprietary) and WCAP-11398-A (Non-proprietary), April 1989.

Tables for Section 15.2

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. With pressurizer control	Turbine trip, loss of main feedwater flow	0.0
	Initiation of steam release from steam generator safety valves	9.4
	High pressurizer pressure reactor trip point reached	11.2
	Rods begin to drop	13.2
	Peak RCS pressure occurs	13.6
	Minimum DNBR occurs	14.6
2. Without pressurizer control	Turbine trip, loss of main feed flow	0.0
	High pressurizer pressure reactor trip point reached	5.4
	Rods begin to drop	7.4
	Initiation of steam release from steam generator safety valves	8.2
	Peak RCS pressure occurs	8.4

TABLE 15.2-1 (Cont)

Accident	Event	Time (sec)
Loss of Non-Emergency ac Power	Main feedwater flow stops	10.0
	Low-low steam generator water level trip	44.7
	Rods begin to drop	46.7
	Reactor coolant pumps begin to coast down	48.7
	Peak water level in pressurizer occurs	50.0
	Three steam generators begin to receive auxiliary feedwater from two auxiliary feedwater pumps	104.7
	Core decay heat decreases to auxiliary feedwater heat removal capacity	~1,200
Loss of Normal Feed-water Flow	Main feedwater flow stops	10.0
	Low-low steam generator water level trip	44.7
	Rods begin to drop	46.7
	Peak water level in pressurizer occurs	50.0
	Three steam generators begin to receive auxiliary feed from two auxiliary feedwater pumps	104.7
	Core decay heat decreases to auxiliary feedwater heat removal capacity	~2,800

TABLE 15.2-1 (Cont)

Accident	Event	Time (sec)
Feedwater System Pipe Break		
1. With offsite power available ($0.717\ ft^2$)	Main feed line rupture occurs	10
	Low-low steam generator level reactor trip setpoint reached in ruptured steam generator	15.8
	Rods begin to drop	17.8
	Auxiliary feedwater is started	75.8
	Low steam line pressure setpoint reached in ruptured steam generator	185.6
	All main steam line isolation valves close	193.6
	Feedwater lines are purged and auxiliary feedwater is delivered to intact steam generators	620.0
	Steam generator safety valve setpoint reached in intact steam generators	649.3
	Core decay heat plus pump heat decreases to auxiliary feedwater heat removal capacity	~2,736

TABLE 15.2-1 (Cont)

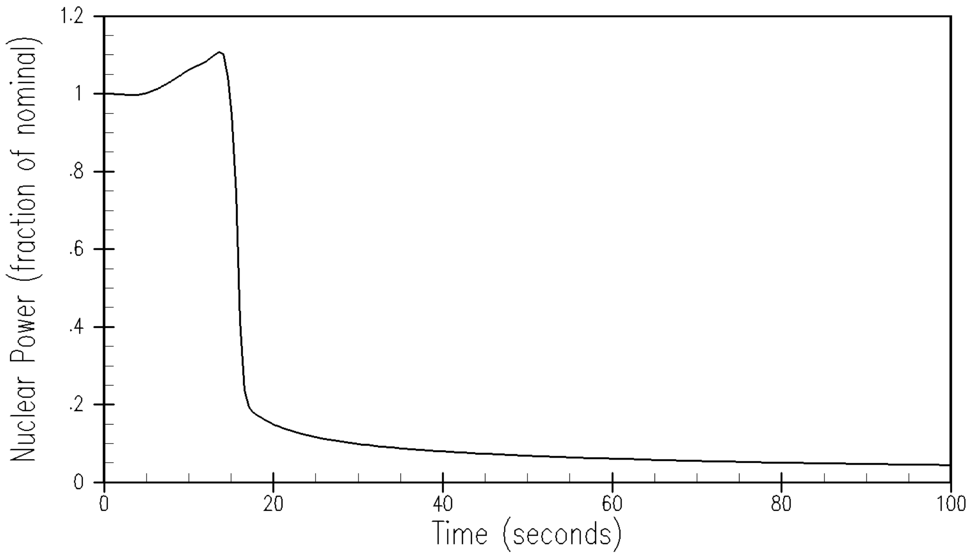
Accident	Event	Time (sec)
2. Without offsite power (1.36 ft^2)	Main feed line rupture occurs	10
	Low-low steam generator level reactor trip setpoint reached in ruptured steam generator	13.4
	Rods begin to drop, power lost to the reactor coolant pumps	15.4
	Low steam line pressure setpoint reached in ruptured steam generator	15.5
	All main steam line isolation valves close	23.5
	Steam generator safety valve setpoint reached in intact steam generators	48.4
	Auxiliary feedwater is started	73.4
	Feedwater lines are purged and "cold" auxiliary feedwater is delivered to intact steam generators	616.0
	Core decay heat decreases to auxiliary feedwater heat removal capacity	~718

TABLE 15.2-2

PARAMETERS USED FOR THE LOSS OF NONEMERGENCY AC POWER TO THE STATION AUXILIARIES ACCIDENT

Core Power Level	2918 MWt
Minimum Reactor Coolant Mass	340,711 lbm
Primary to Secondary SG tube leakage	450 gpd @ STP
Melted Fuel Percentage	0%
Failed Fuel Percentage	0%
RCS Tech Spec Iodine & NG Concentration	Table 15.0-8c (0.35 µCi/gm DE-I131)
Secondary Side Parameters:	
Minimum Post-Accident SG Liquid Mass	101,799 lbm per SG
Iodine Species released to Environment	97% elemental; 3% organic
Tech Spec Activity in SG liquid	Table 15.0-8c (0.1 µCi/gm DE-I131)
Iodine Partition Coefficient in SGs	100 (all tubes submerged)
Fraction of Noble Gas Released from SGs	1.0 (Released without holdup)
Steam Releases from SGs	0-2 hr (348,000 lbm) 2-8 hr (773,000 lbm)
Termination of releases from SGs	8 hours
Environmental Release Point	MSSVs/ADVs
CR emergency Ventilation: Initiation Signal/Timing	CR is maintained under Normal Operation ventilation

Note:
(1) Bounding parameter values are used to encompass an event at either unit.



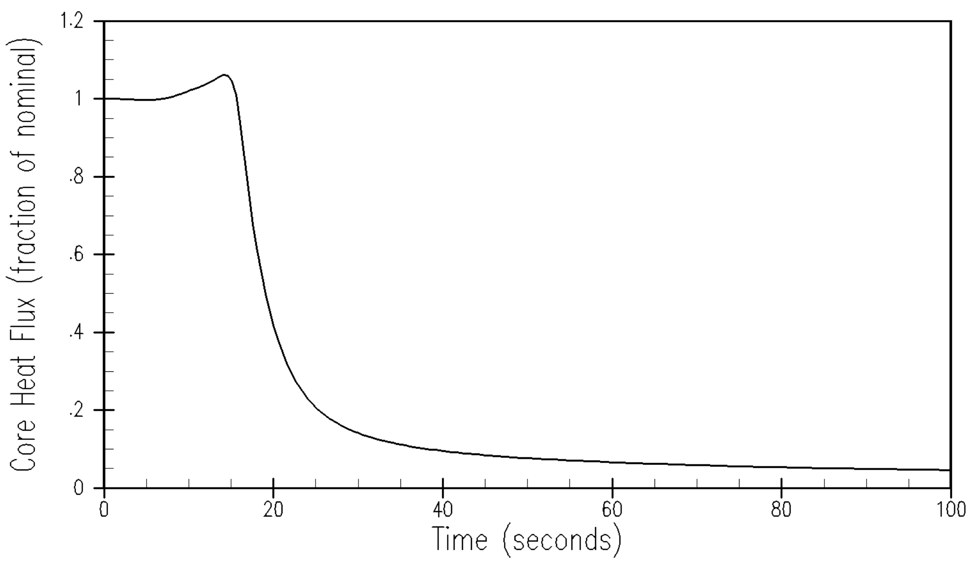


Figure 15.2-1

Turbine Trip Accident With Pressurizer Spray and Power-Operated Relief Valves

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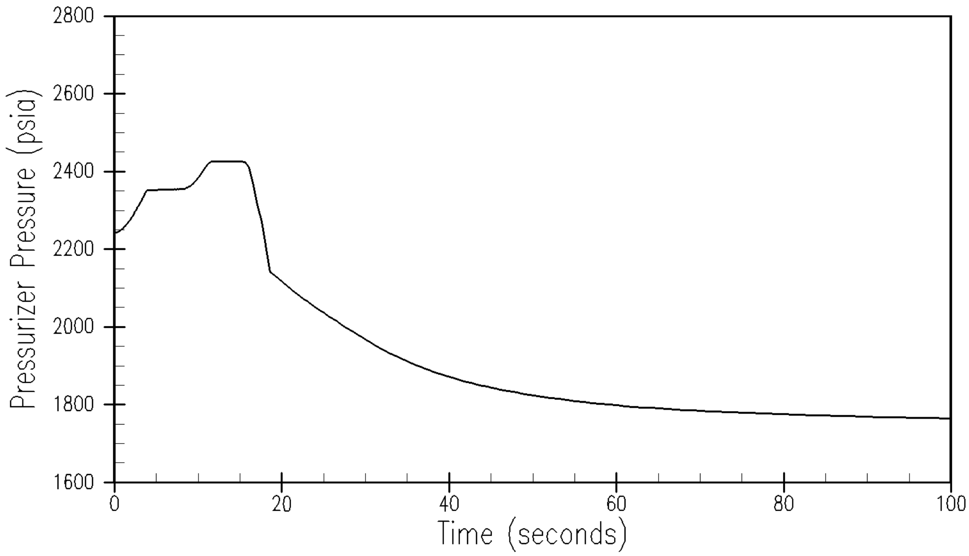

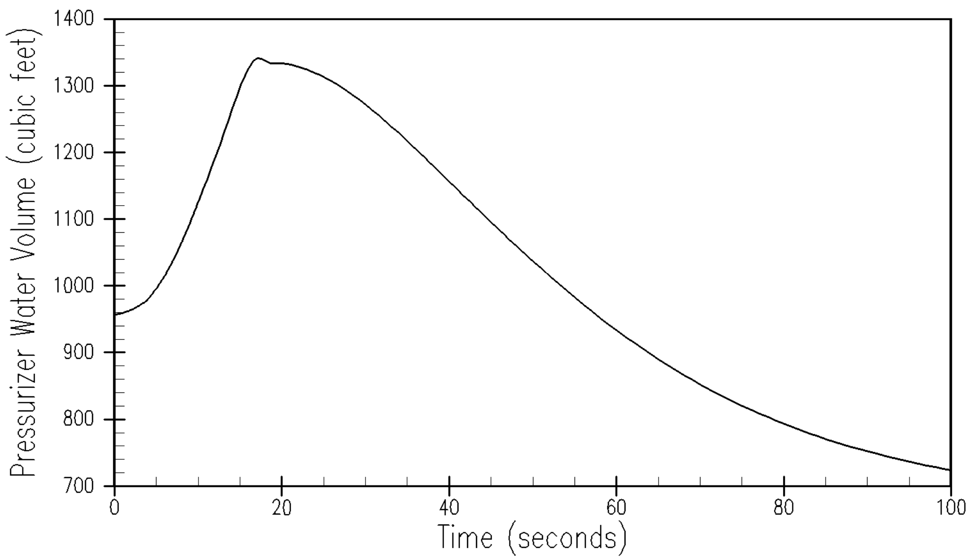

Figure 15.2-2

Turbine Trip Accident With Pressurizer Spray and Power-Operated Relief Valves

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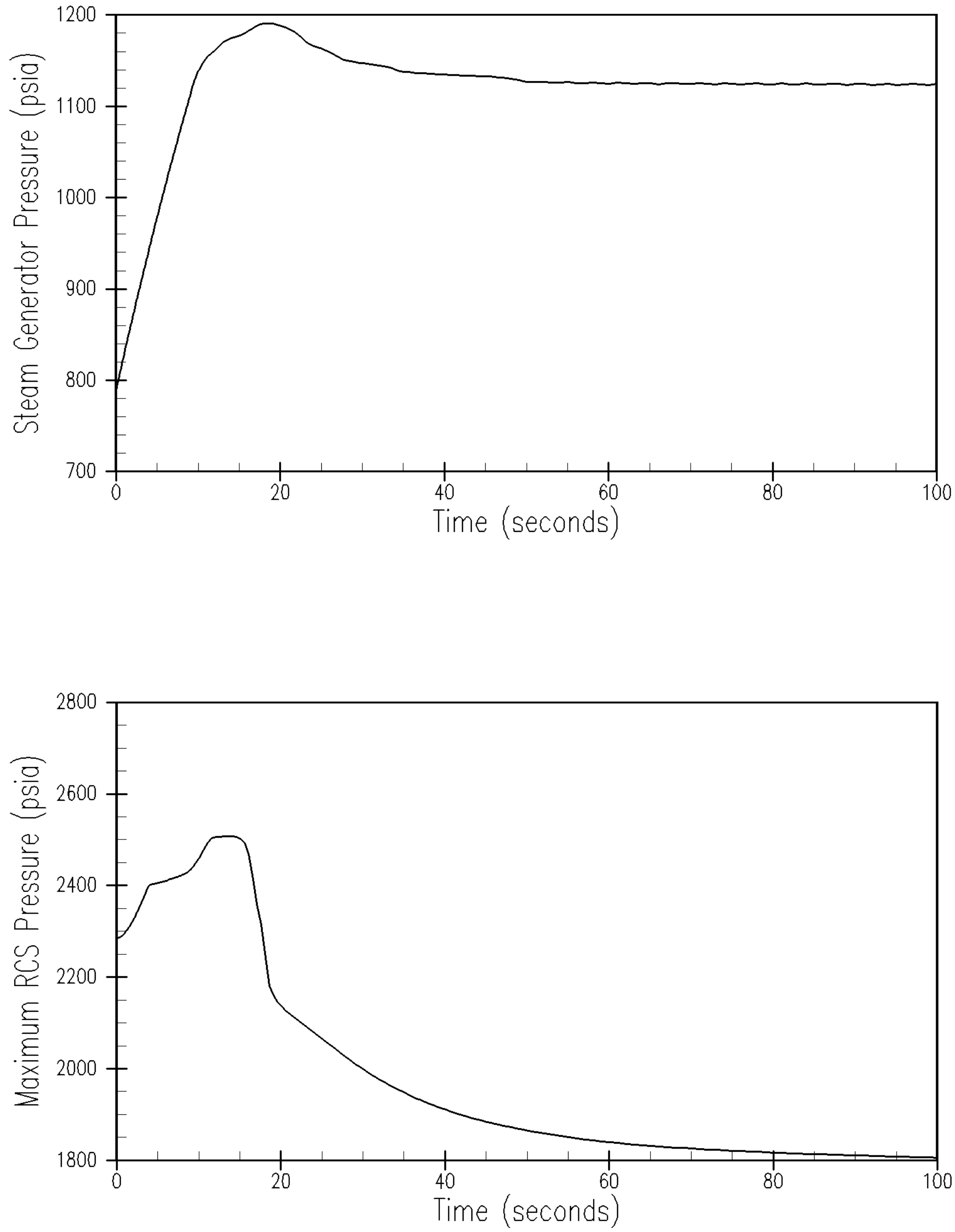

Figure 15.2-3

Turbine Trip Accident With Pressurizer Spray and Power-Operated Relief Valves

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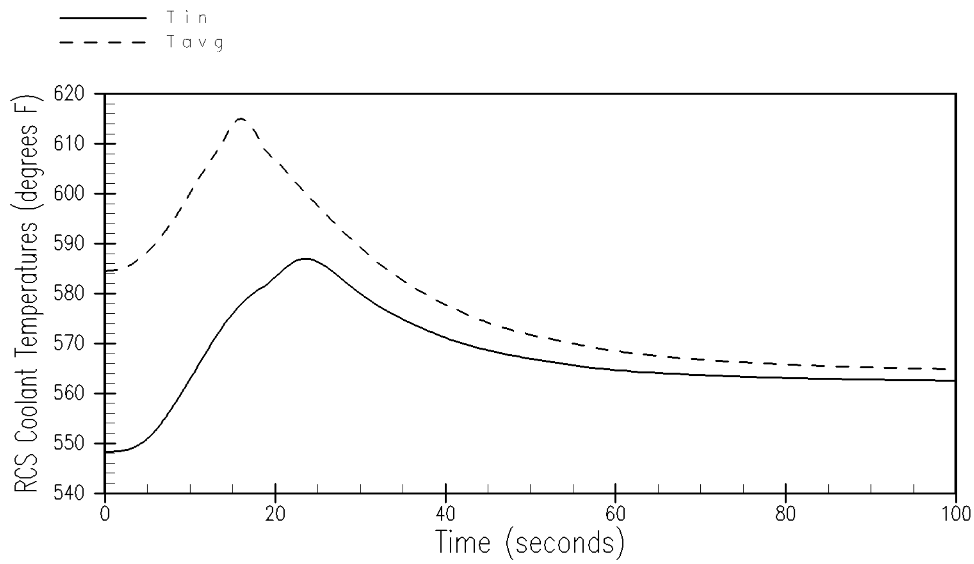

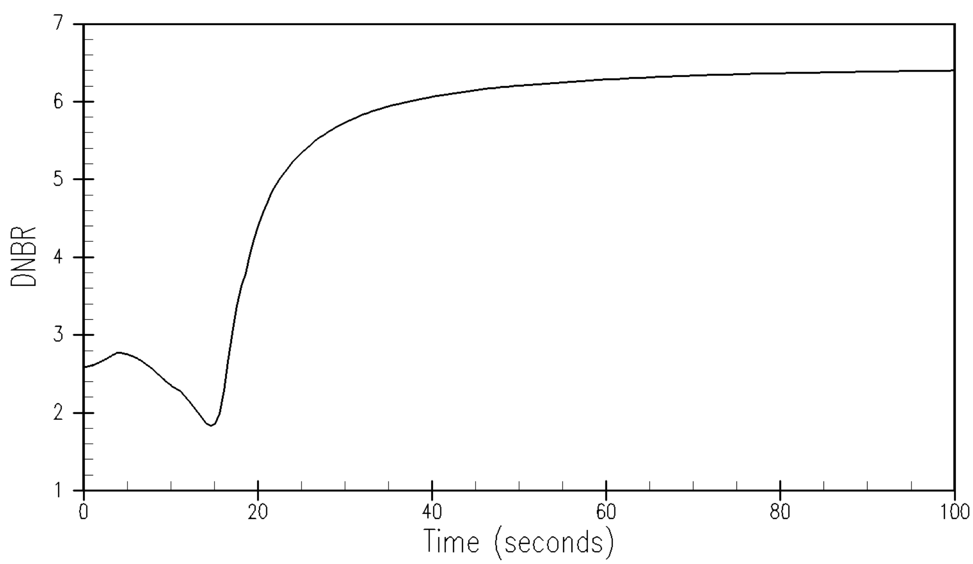

Figure 15.2-4

Turbine Trip Accident With Pressurizer Spray and Power-Operated Relief Valves

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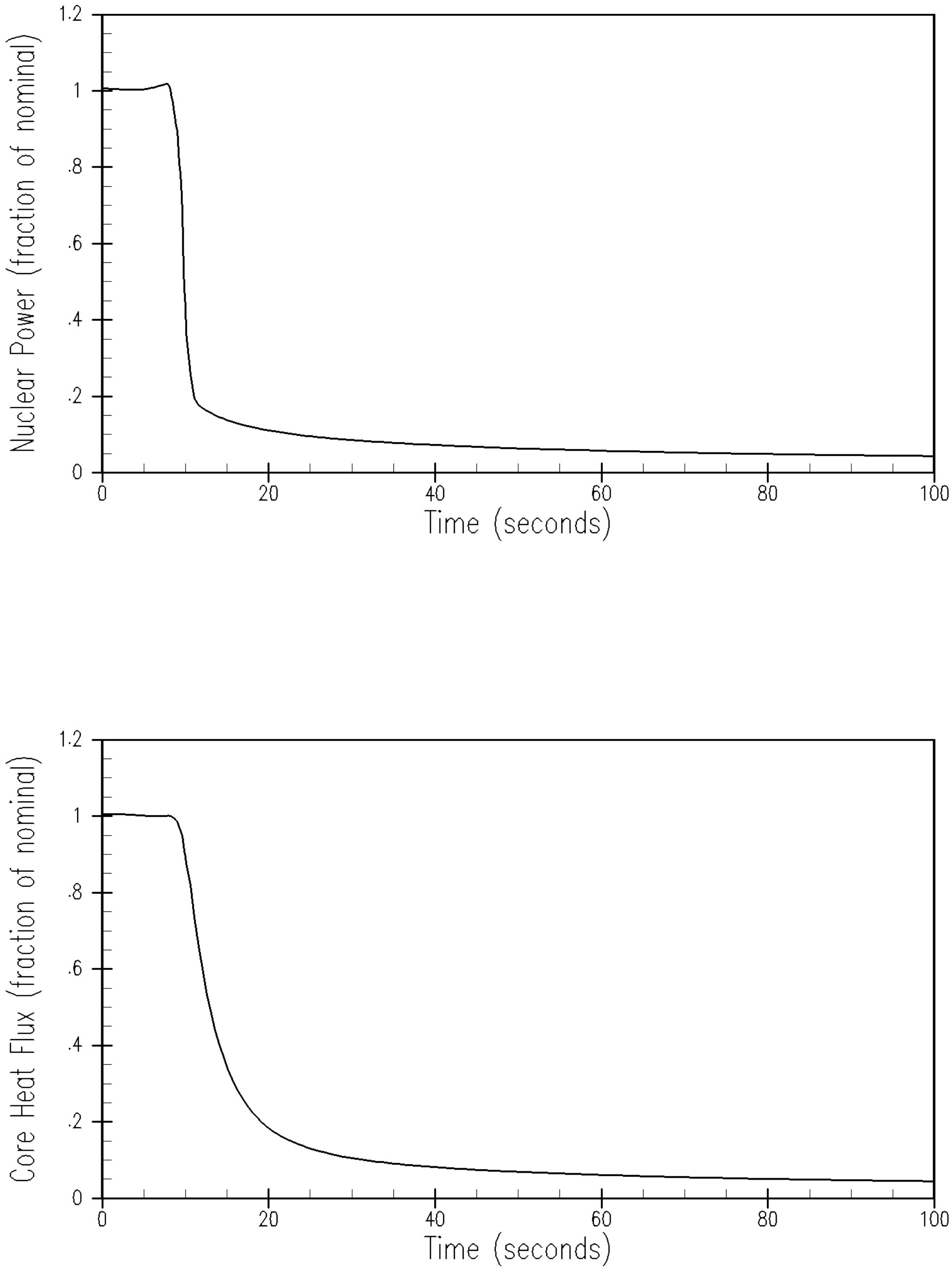

Figure 15.2-5

Turbine Trip Accident Without Pressurizer Spray and Power-Operated Relief Valves

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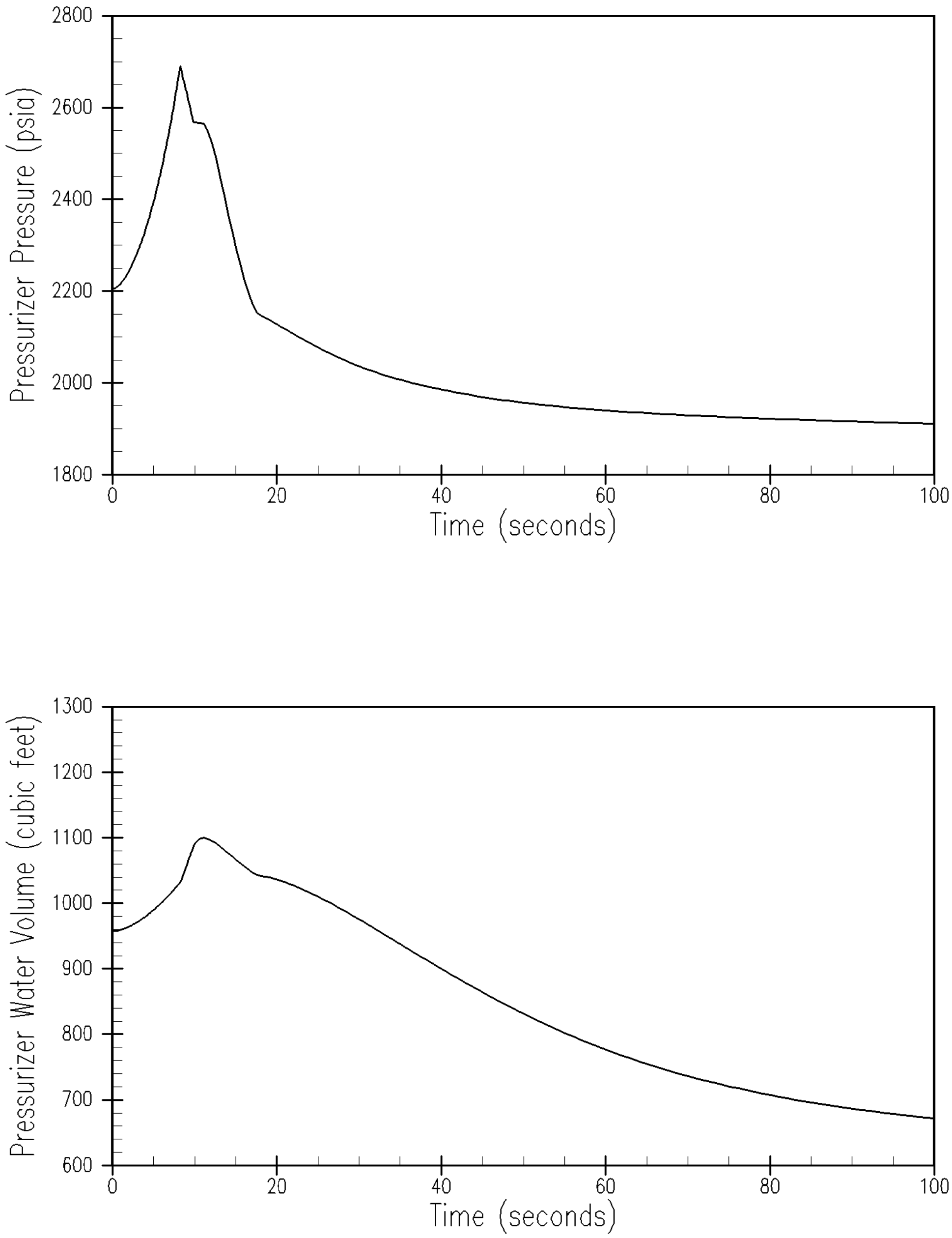

Figure 15.2-6

Turbine Trip Accident Without Pressurizer Spray and Power-Operated Relief Valves

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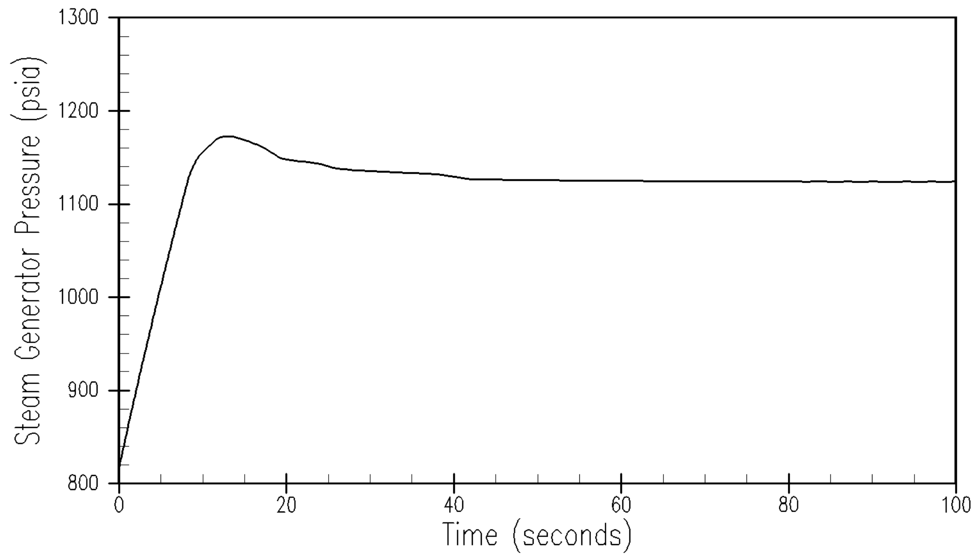

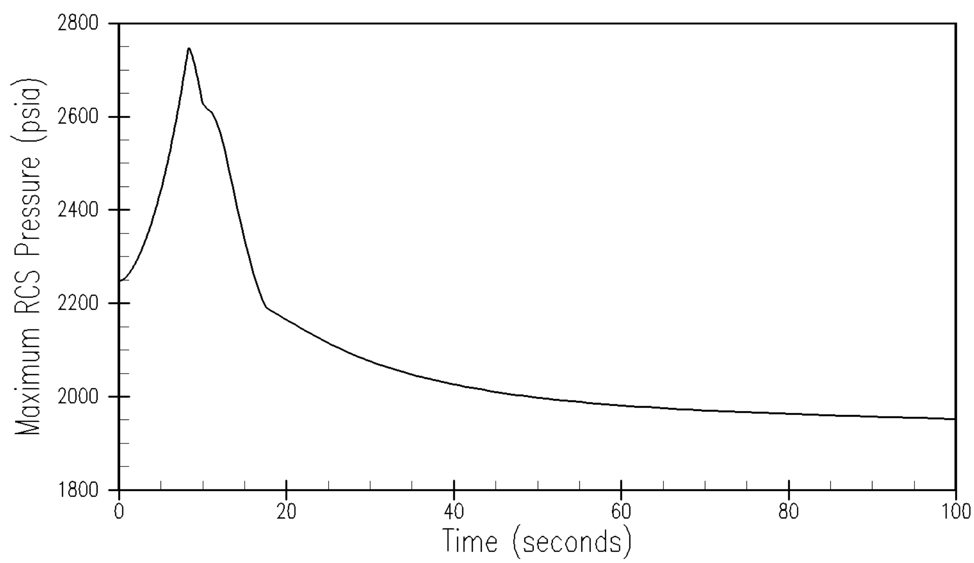

Figure 15.2-7

Turbine Trip Accident Without Pressurizer Spray and Power-Operated Relief Valves

Beaver Valley Power Station Unit No. 2
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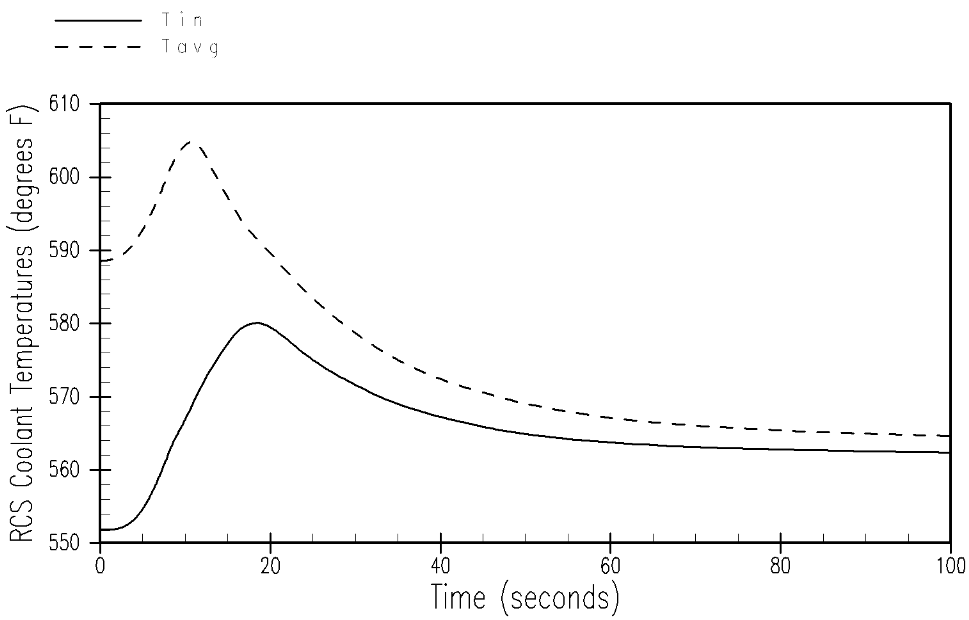

Figure 15.2-8

Turbine Trip Accident Without Pressurizer Spray and Power-Operated Relief Valves

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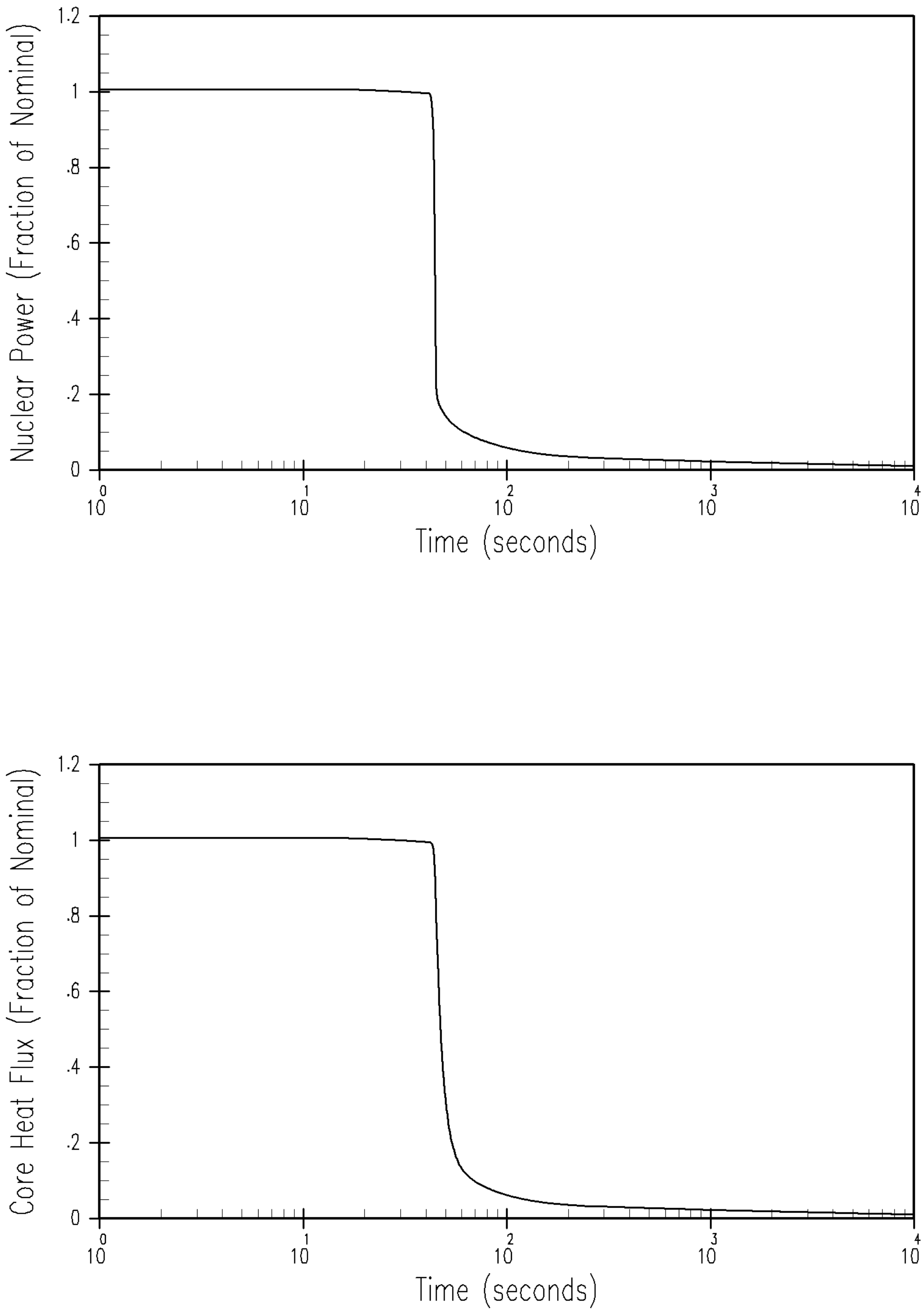

Figure 15.2-9

Nuclear Power and Core Heat Flux Transients for Loss of AC Power

Beaver Valley Power Station Unit No. 2
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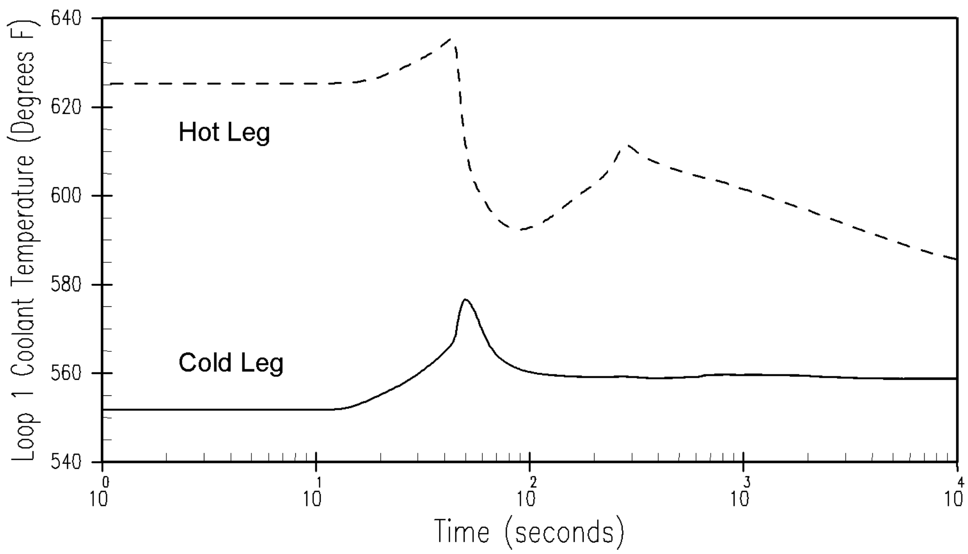

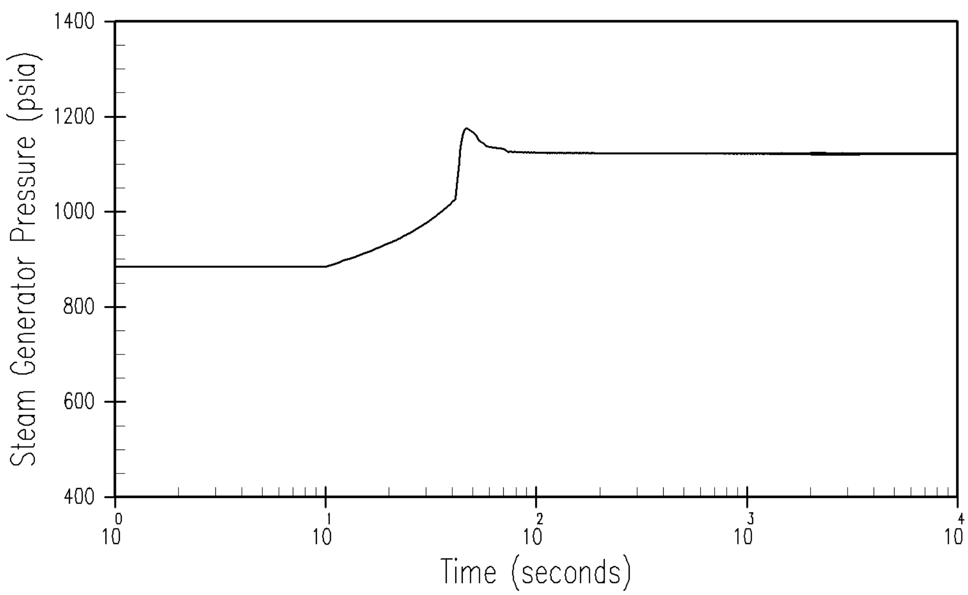

Figure 15.2-10

Reactor Coolant Temperature and Steam Generator Pressure Transients for Loss of AC Power

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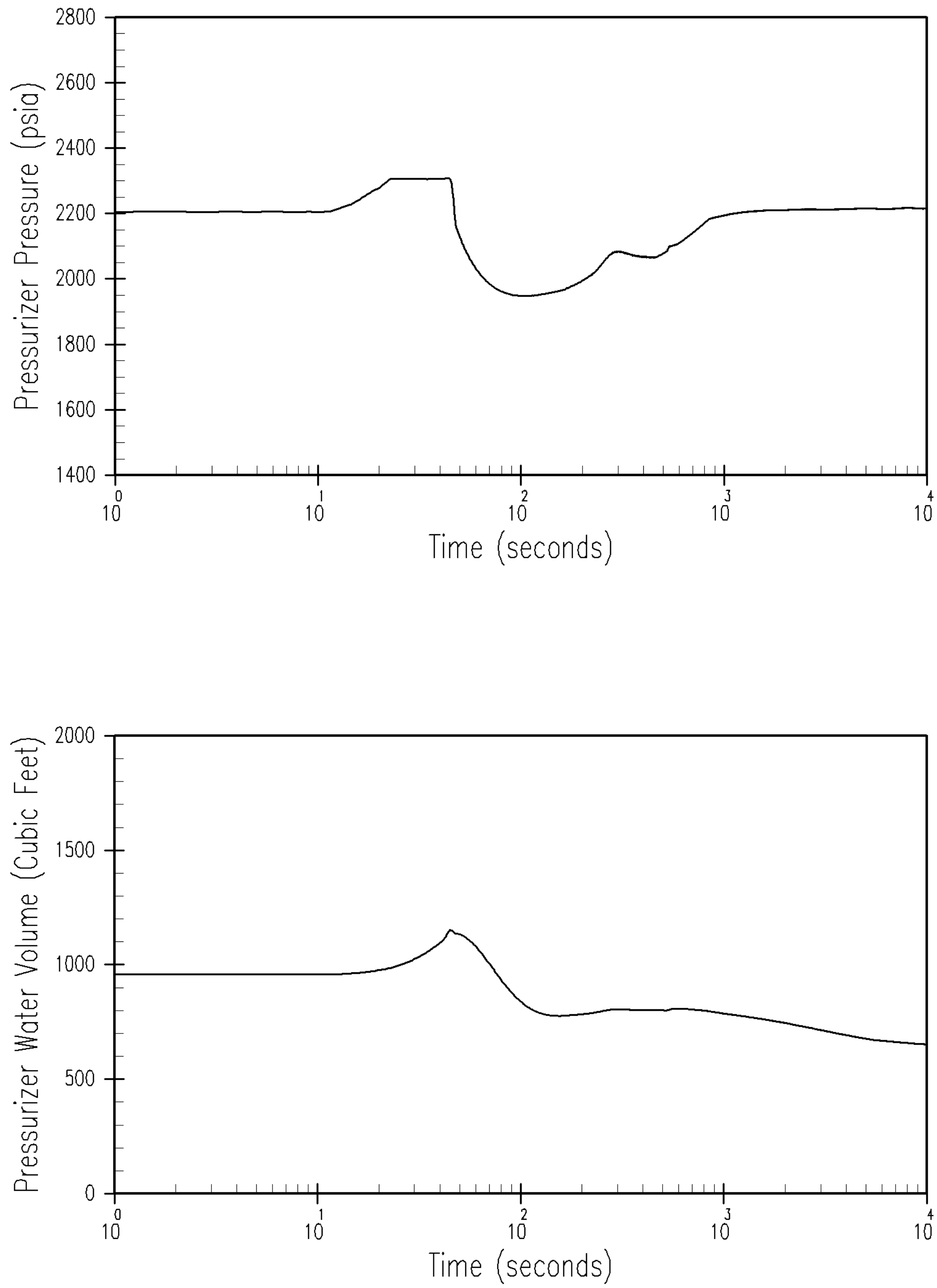

Figure 15.2-11

Pressurizer Pressure and Water Volume Transients for Loss of AC Power

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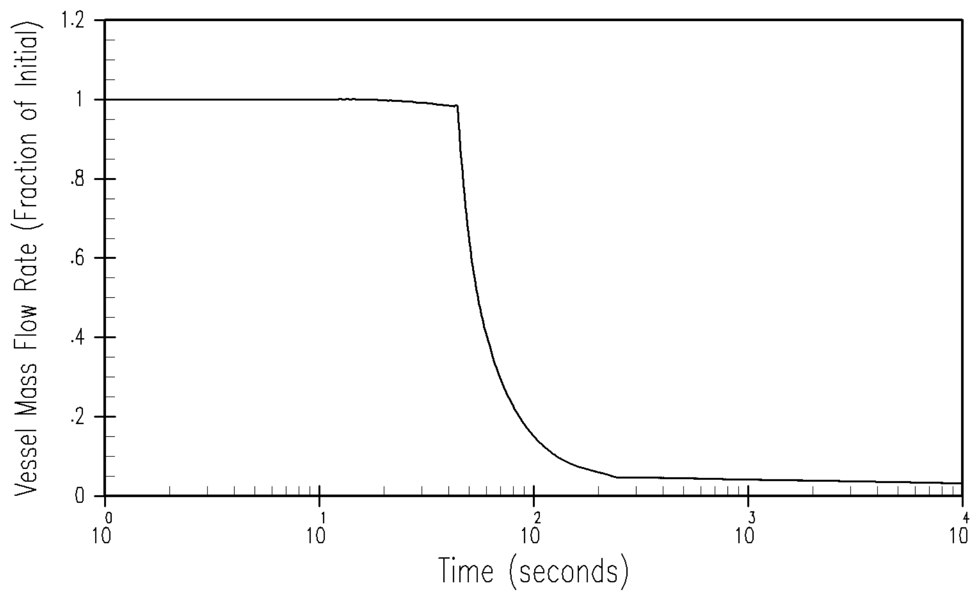

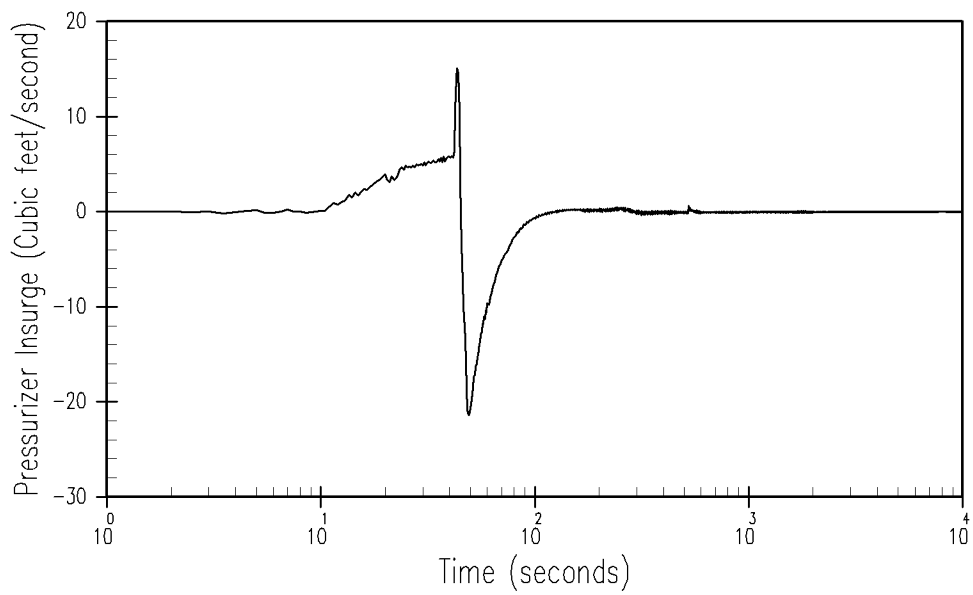

Figure 15.2-12

Vessel Mass Flow Rate and Pressurizer Insurge Transient for Loss of AC Power

Beaver Valley Power Station Unit No. 2 Updated Final Safety Analysis Report

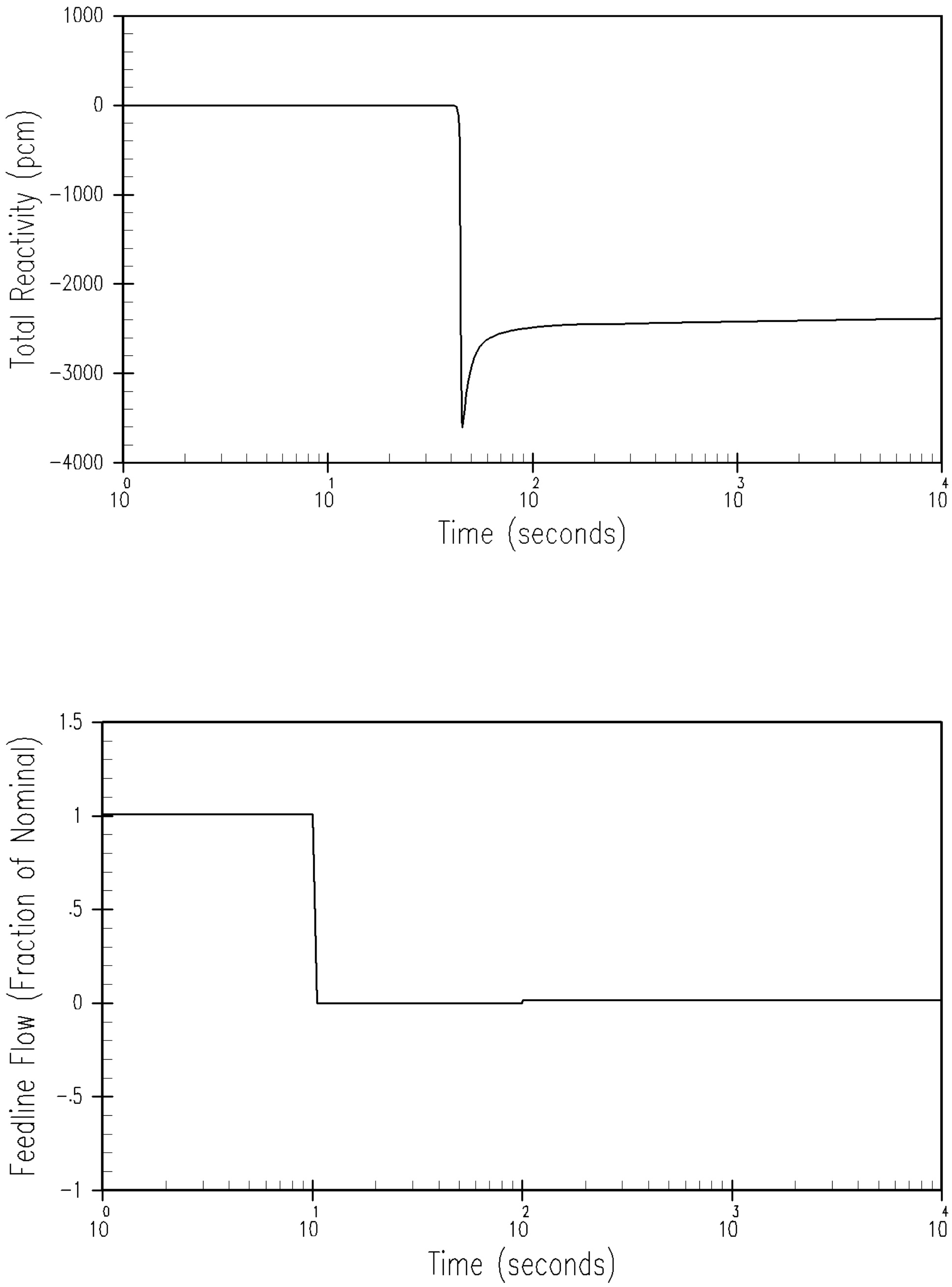

Figure 15.2-13

Core Reactivity Transient and Feedline Flow Transient for Loss of AC Power

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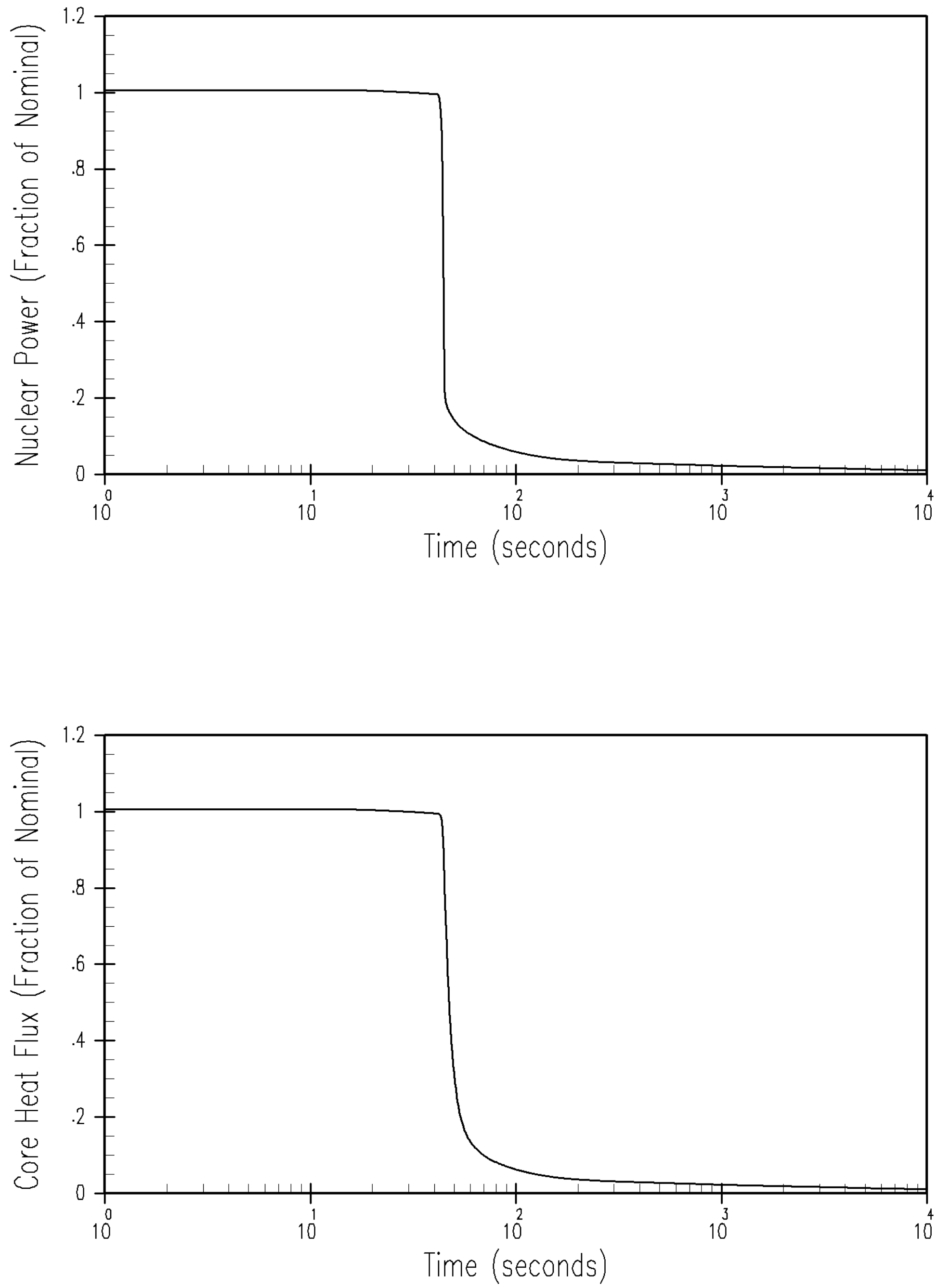

Figure 15.2-14

Nuclear Power and Core Heat Flux Transients for Loss of Feedwater

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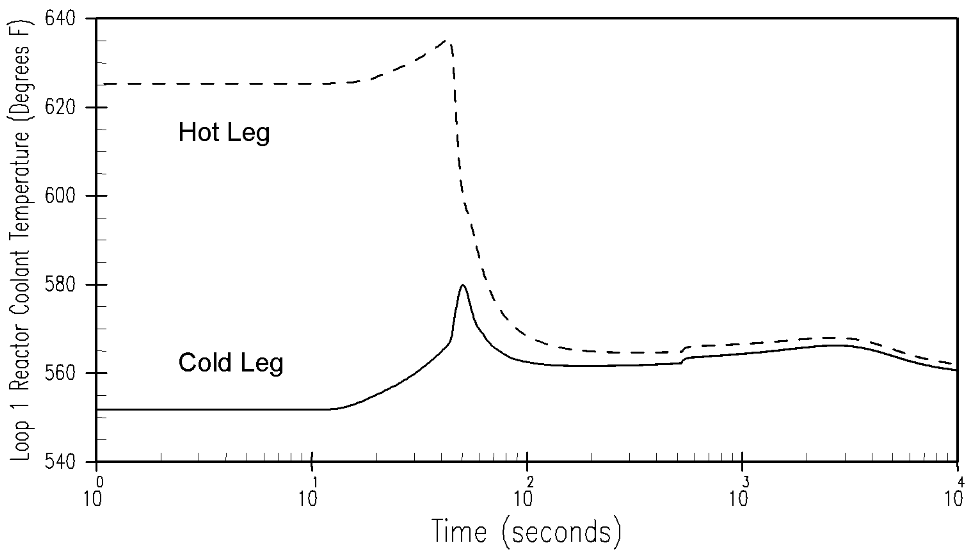

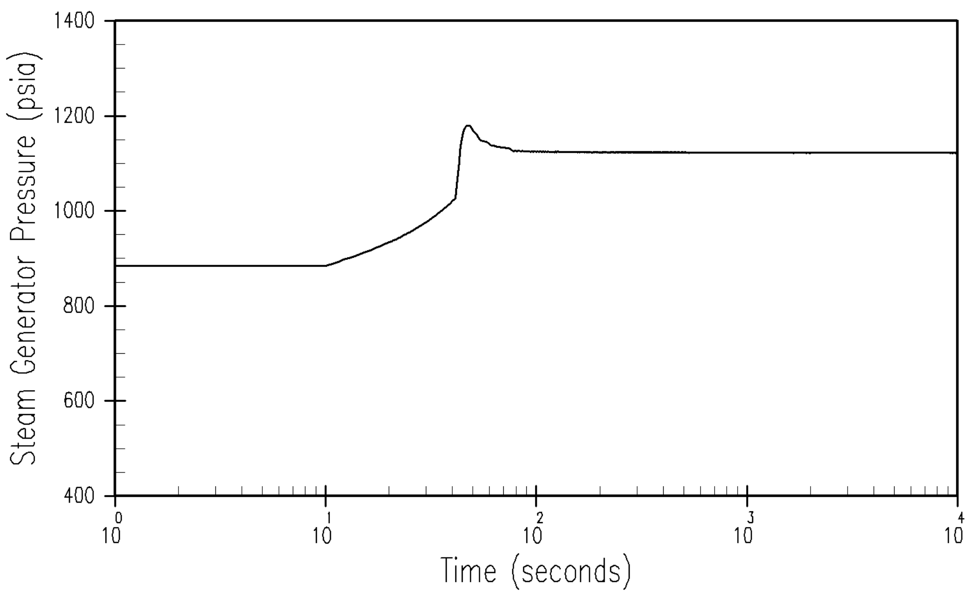

Figure 15.2-15

Reactor Coolant Temperature and Steam Generator Pressure Transients for Loss of Feedwater

Beaver Valley Power Station Unit No. 2
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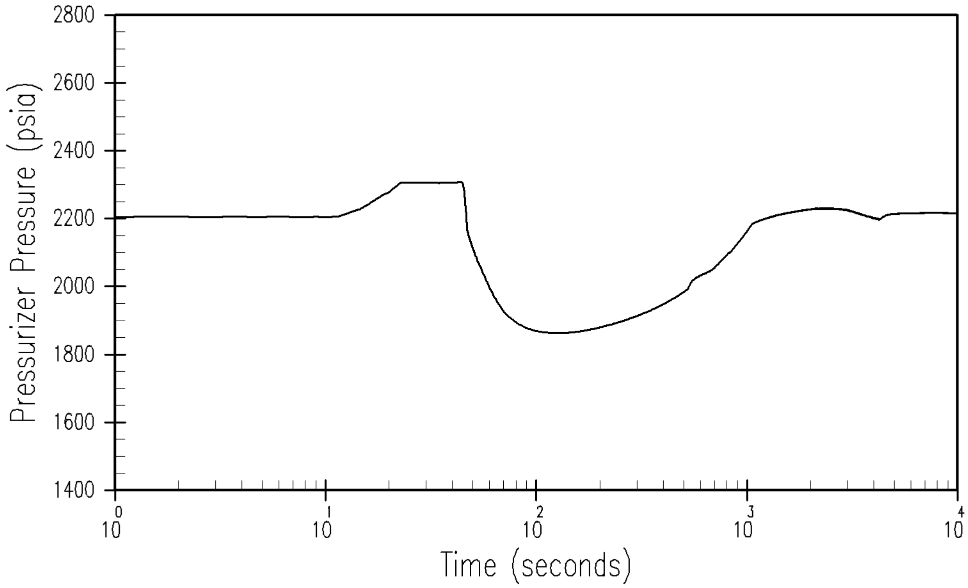

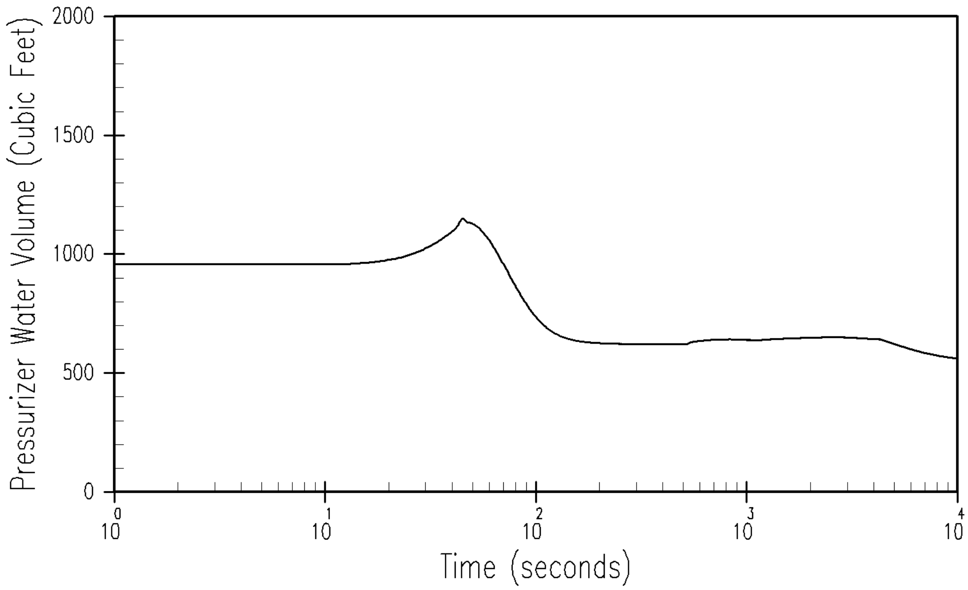

Figure 15.2-16

Pressurizer Pressure and Water Volume Transients for Loss of Feedwater

Beaver Valley Power Station Unit No. 2
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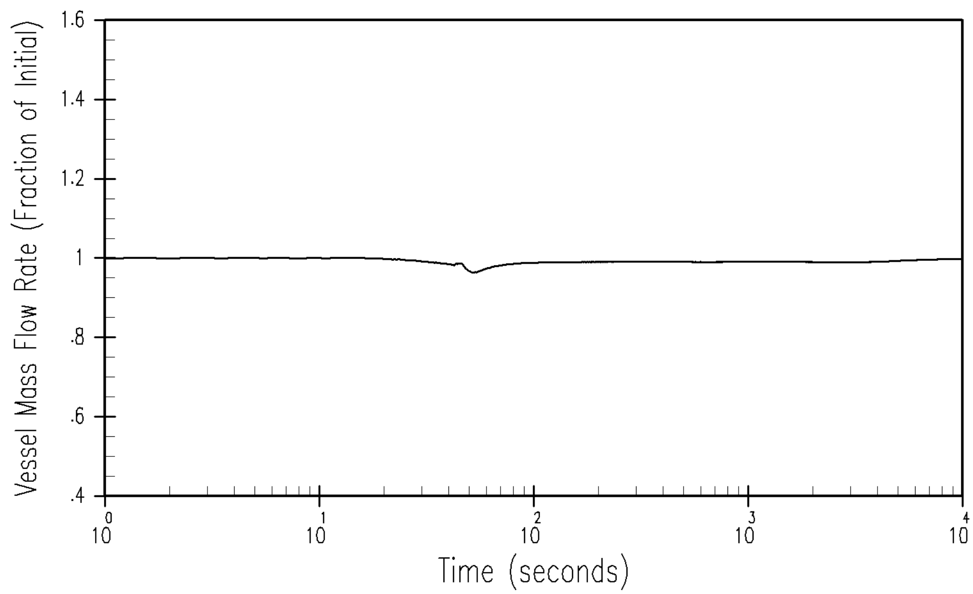

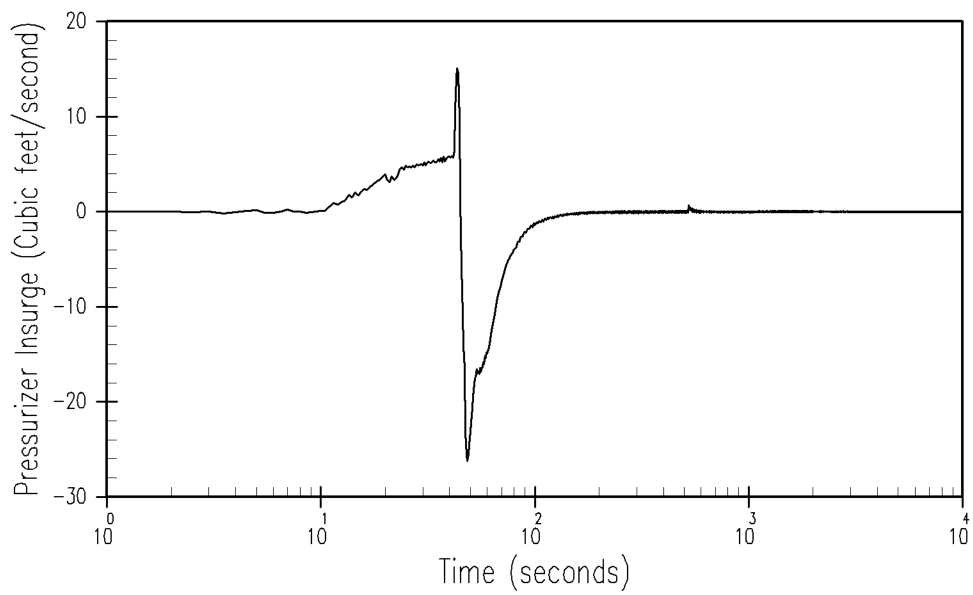

Figure 15.2-17

Vessel Mass Flow Rate and Pressurizer Insurge Transients for Loss of Feedwater

Beaver Valley Power Station Unit No. 2
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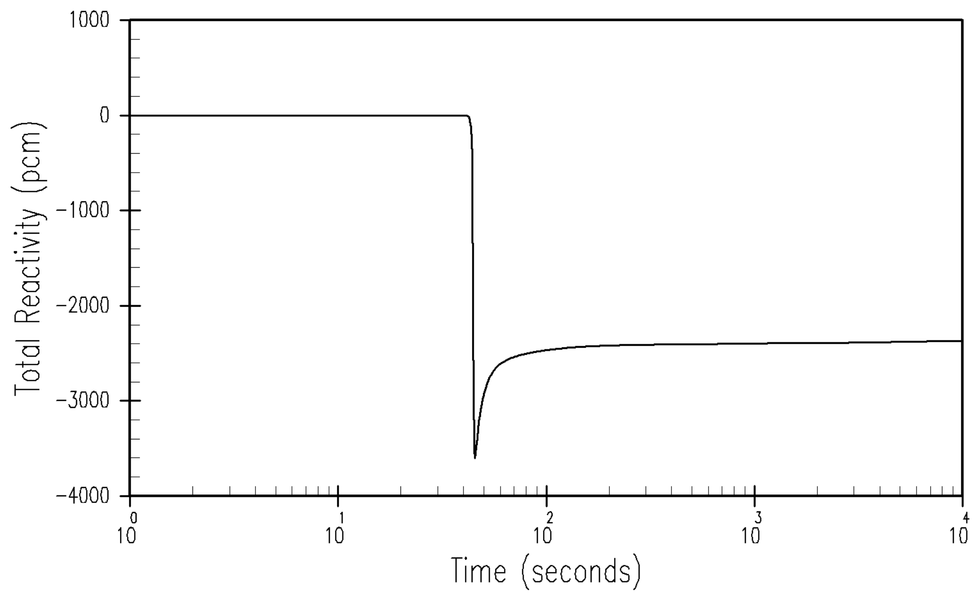

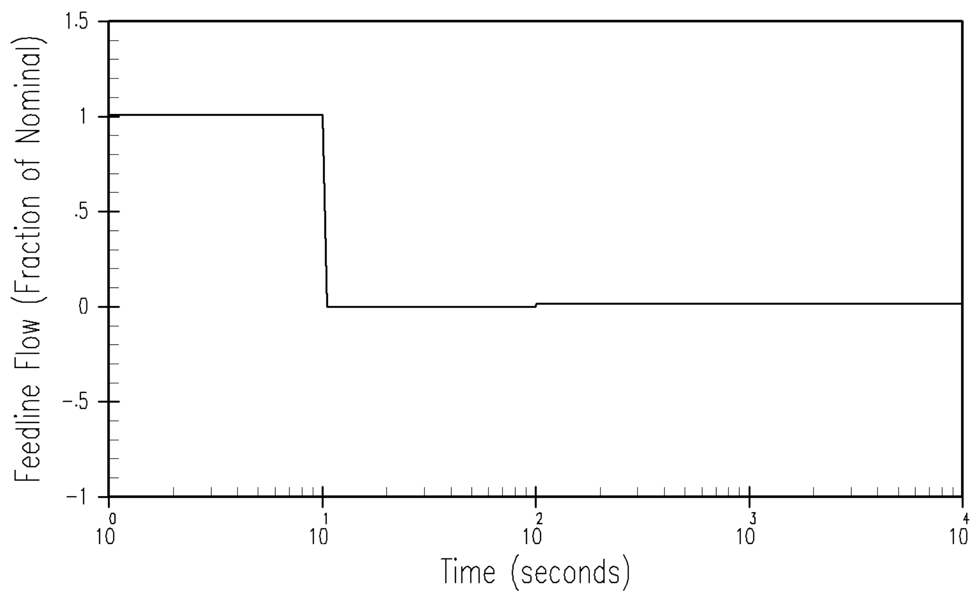

Figure 15.2-18

Core Reactivity Transient and Feedline Flow Transient for Loss of Feedwater

Beaver Valley Power Station Unit No. 2
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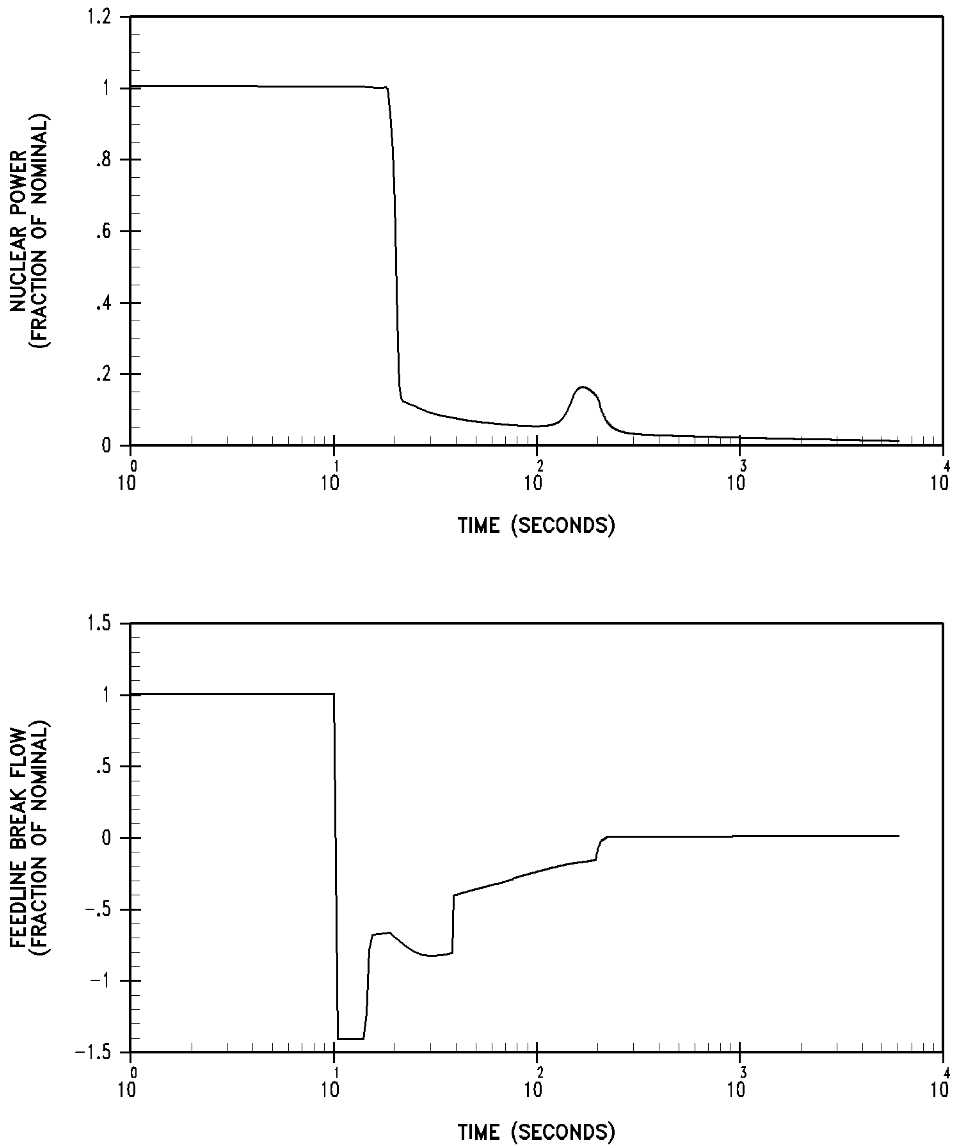

Figure 15.2-19

Nuclear Power and Feedline Break Flow Transients for Main Feedline Rupture with Offsite Power Available (0.717 ft^2)

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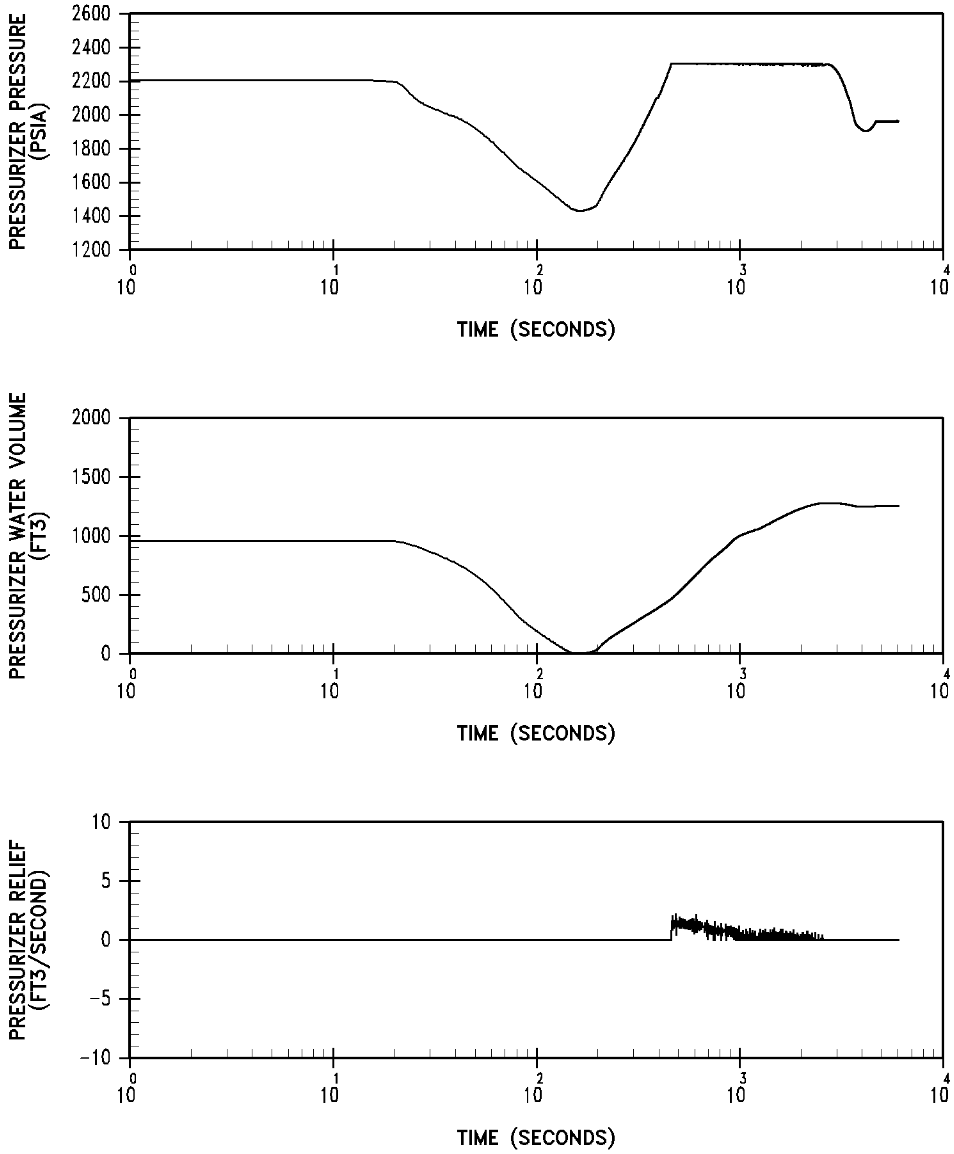

Figure 15.2-20

Pressurizer Pressure, Pressurizer Water Volume and Pressurizer Relief Transients for Main Feedline Rupture with Offsite Power Available (0.717 ft^2)

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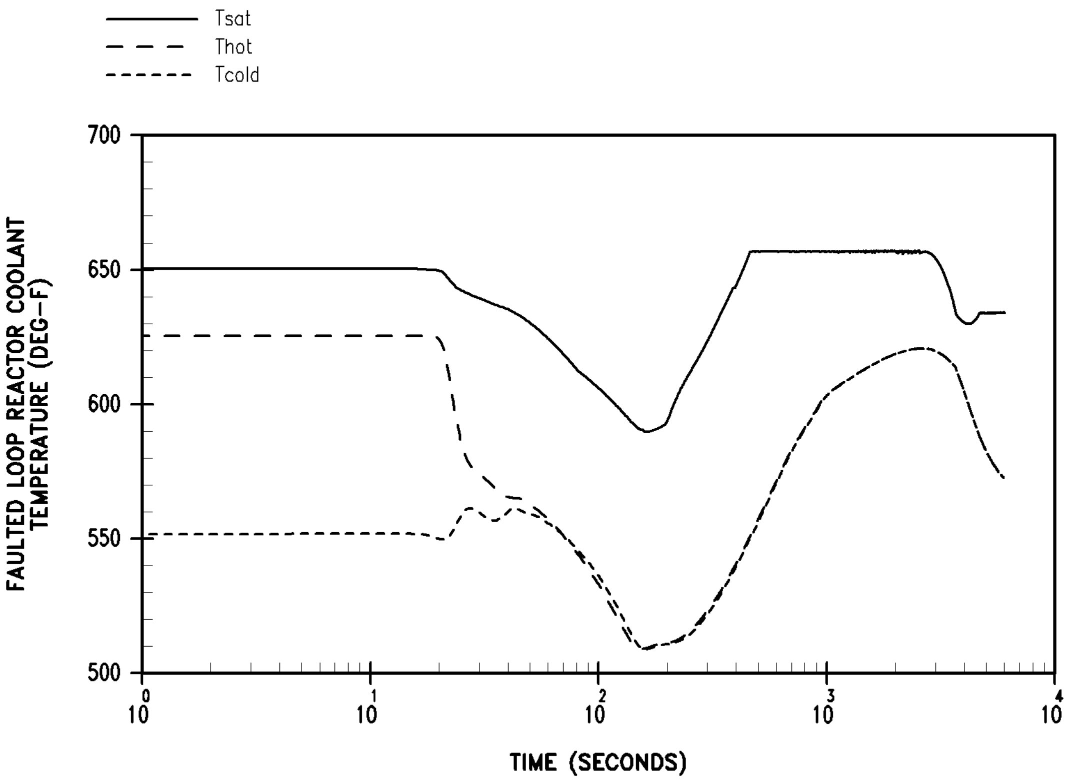

Figure 15.2-21

Reactor Coolant Temperature Transients for the Faulted Loop for Main Feedline Rupture with Offsite Power Available (0.717 ft^2)

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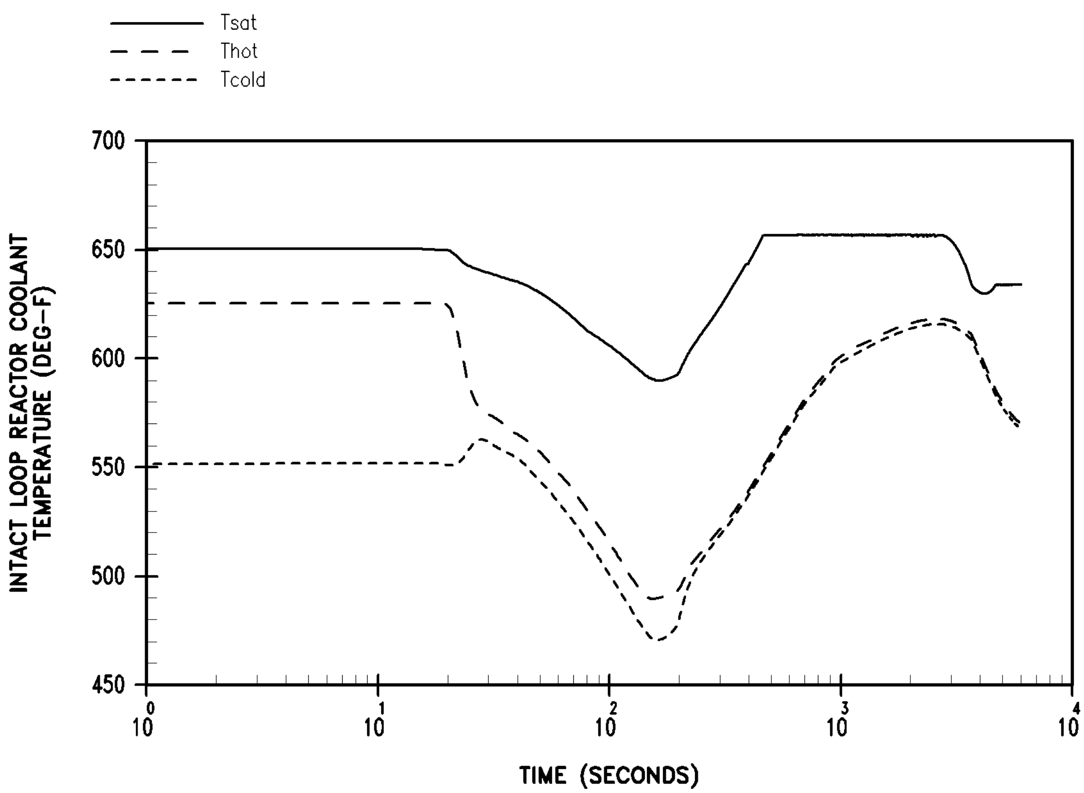

Figure 15.2-22

Reactor Coolant Temperature Transients for an Intact Loop for Main Feedline Rupture with Offsite Power Available (0.717 ft^2)

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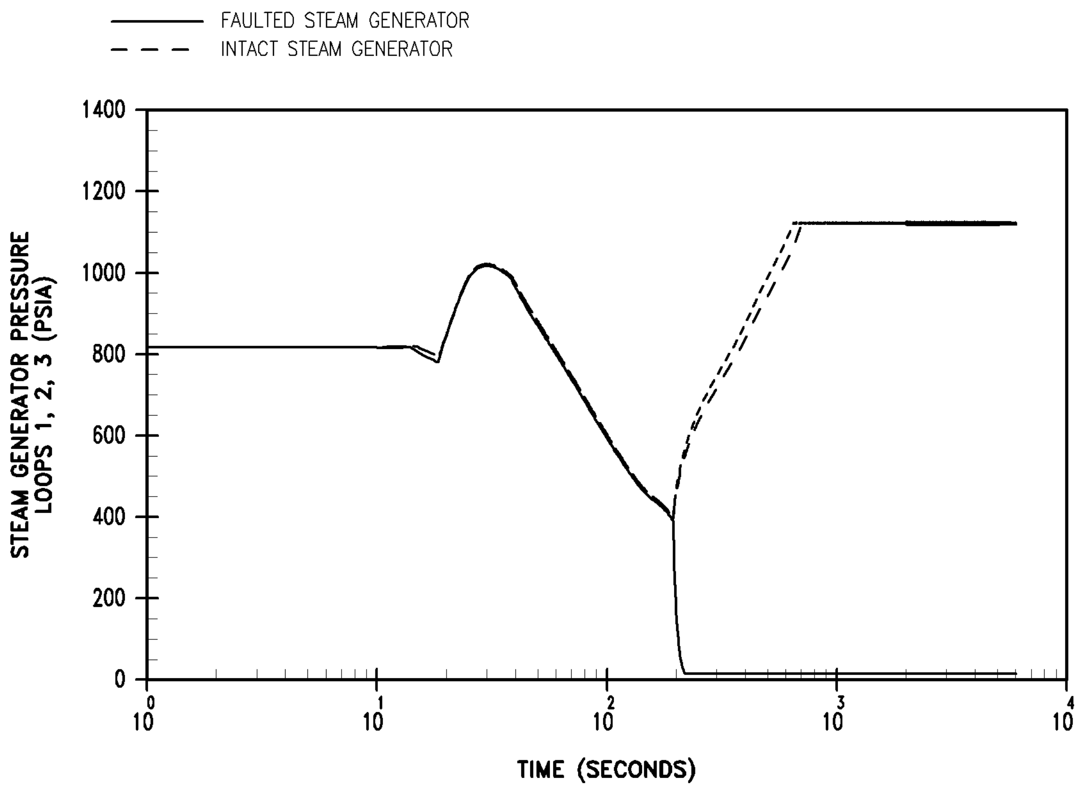


Figure 15.2-23

Steam Generator Pressure Transients for Main Feedline Rupture with Offsite Power Available (0.717 ft^2)

Beaver Valley Power Station Unit No. 2 Updated Final Safety Analysis Report

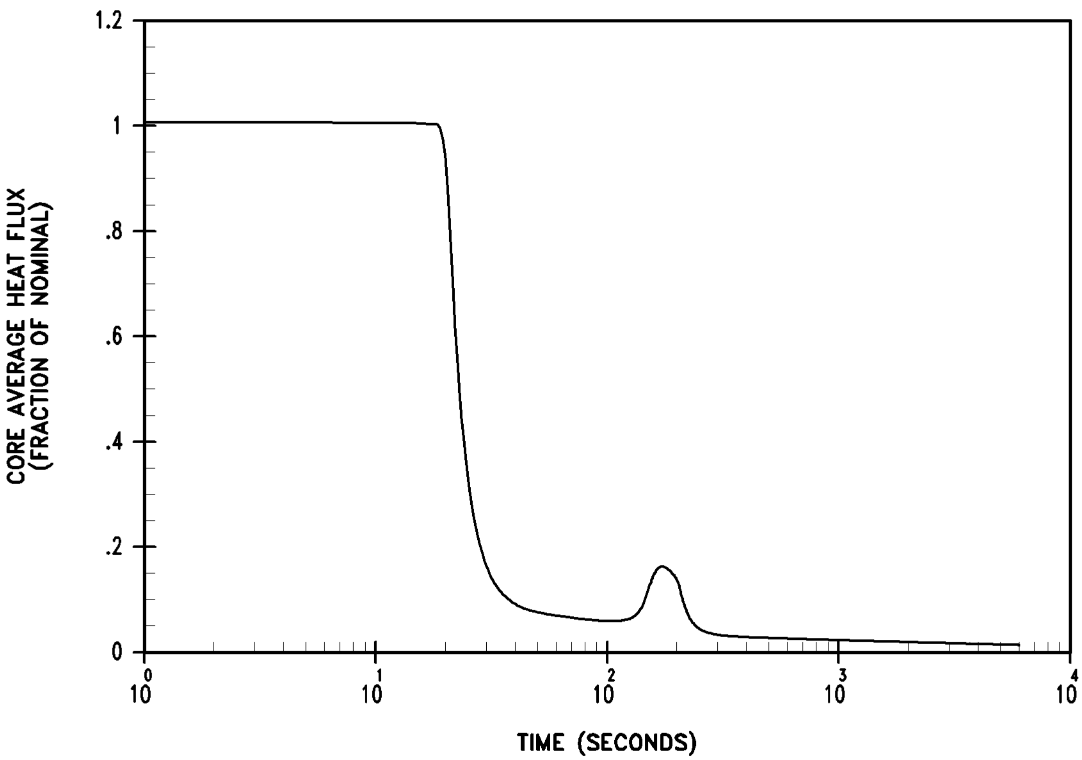
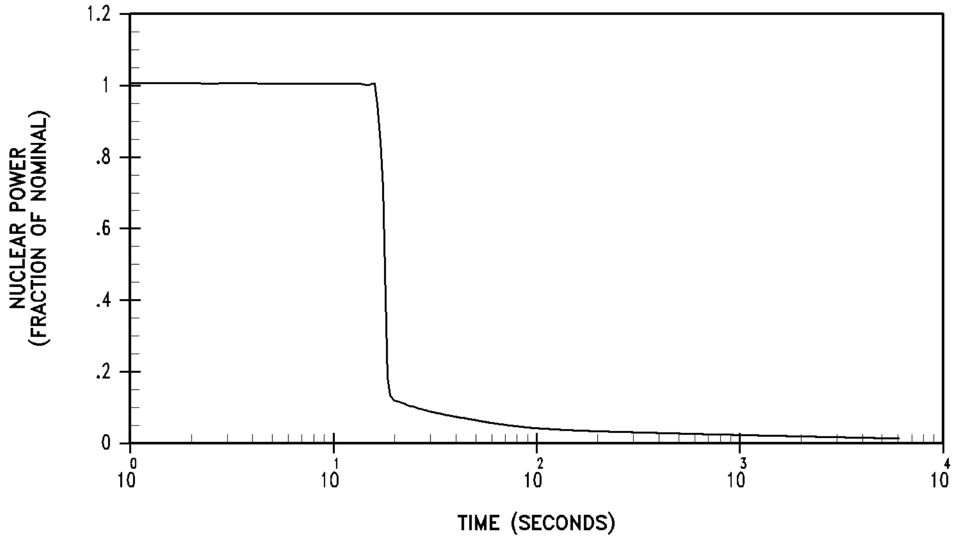


Figure 15.2-24

Core Average Heat Flux Transient for Main Feedline Rupture with Offsite Power Available (0.717 ft^2)

Beaver Valley Power Station Unit No. 2 Updated Final Safety Analysis Report



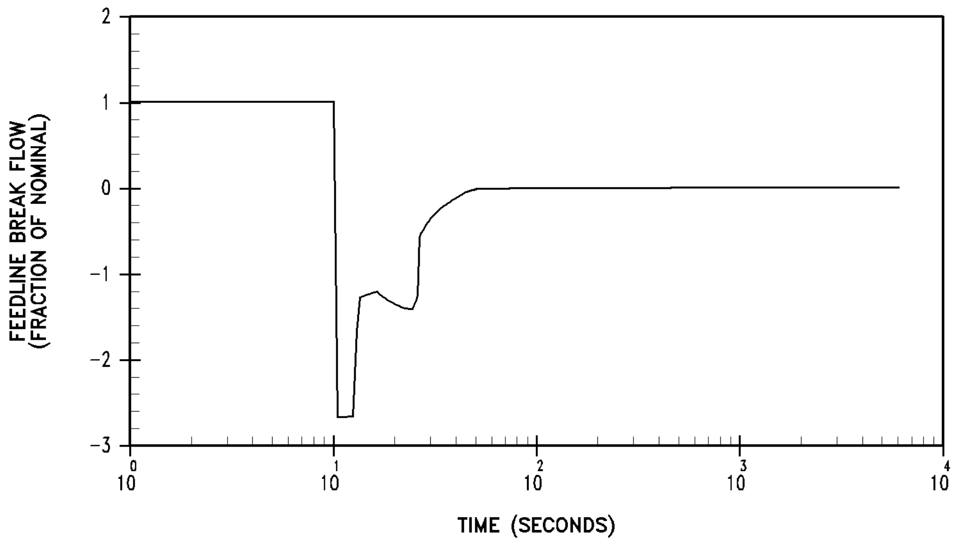


Figure 15.2-25

Nuclear Power and Feedline Break Flow Transients for Main Feedline Rupture without Offsite Power Available (1.36 ft^2)

Beaver Valley Power Station Unit No. 2
Updated Final Safety Analysis Report

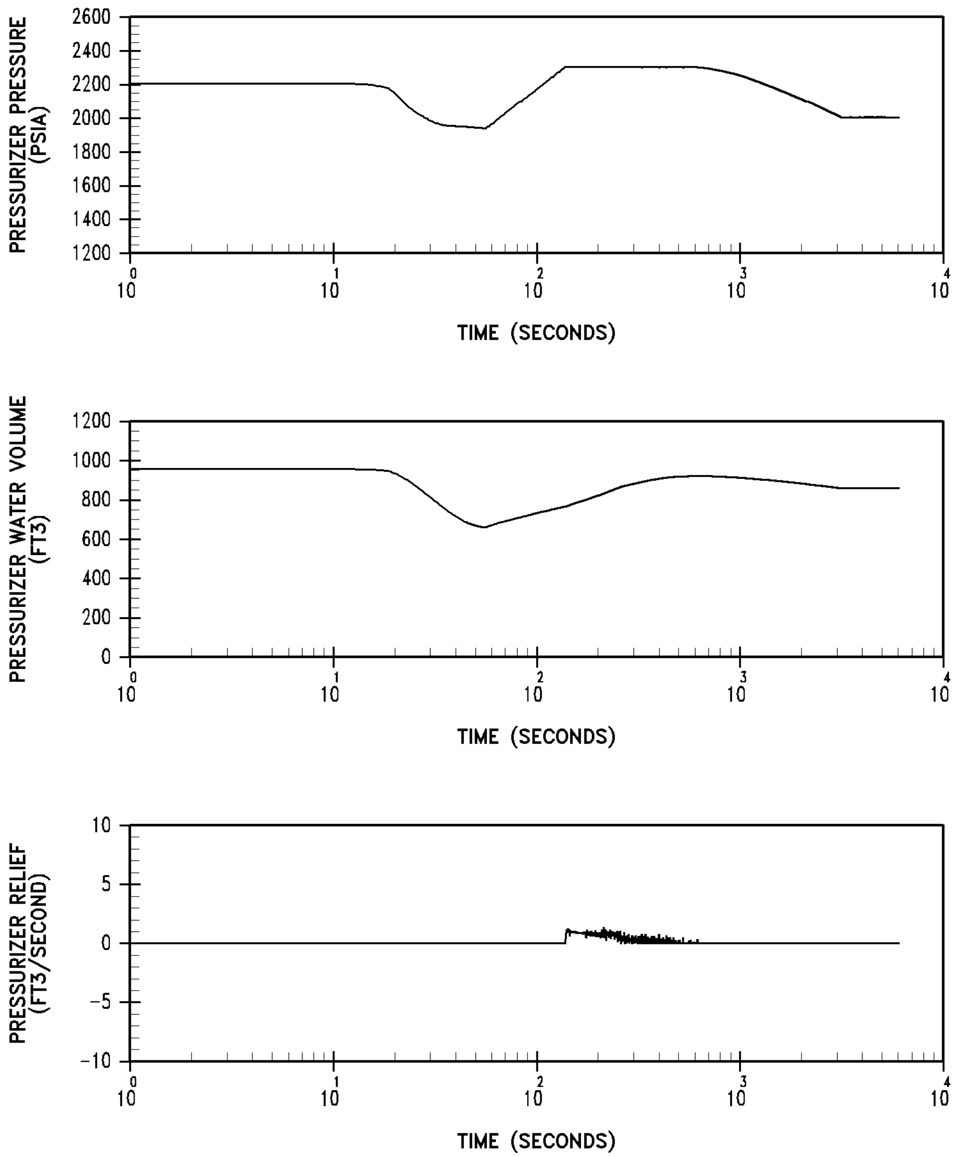


Figure 15.2-26

Pressurizer Pressure, Pressurizer Water Volume and Pressurizer Relief Transients for Main Feedline Rupture without Offsite Power Available (1.36 ft^2)

Beaver Valley Power Station Unit No. 2
Updated Final Safety Analysis Report

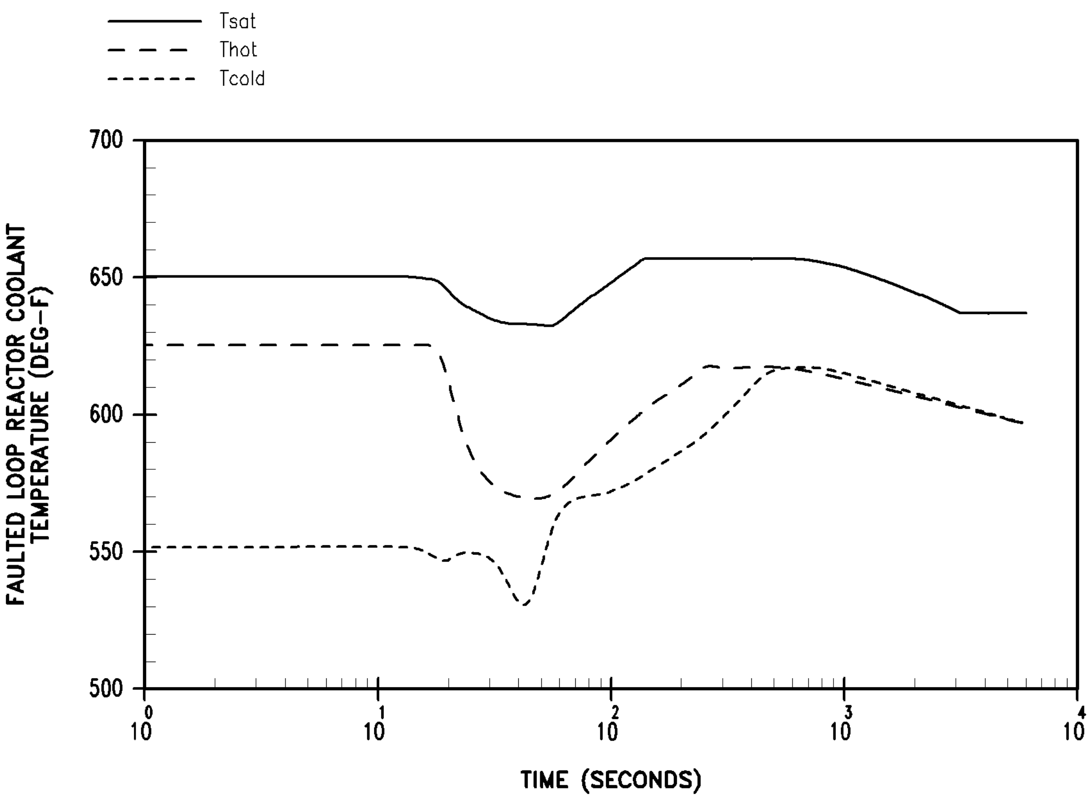


Figure 15.2-27

Reactor Coolant Temperature Transients for the Faulted Loop for Main Feedline Rupture without Offsite Power Available (1.36 ft^2)

Beaver Valley Power Station Unit No. 2
Updated Final Safety Analysis Report

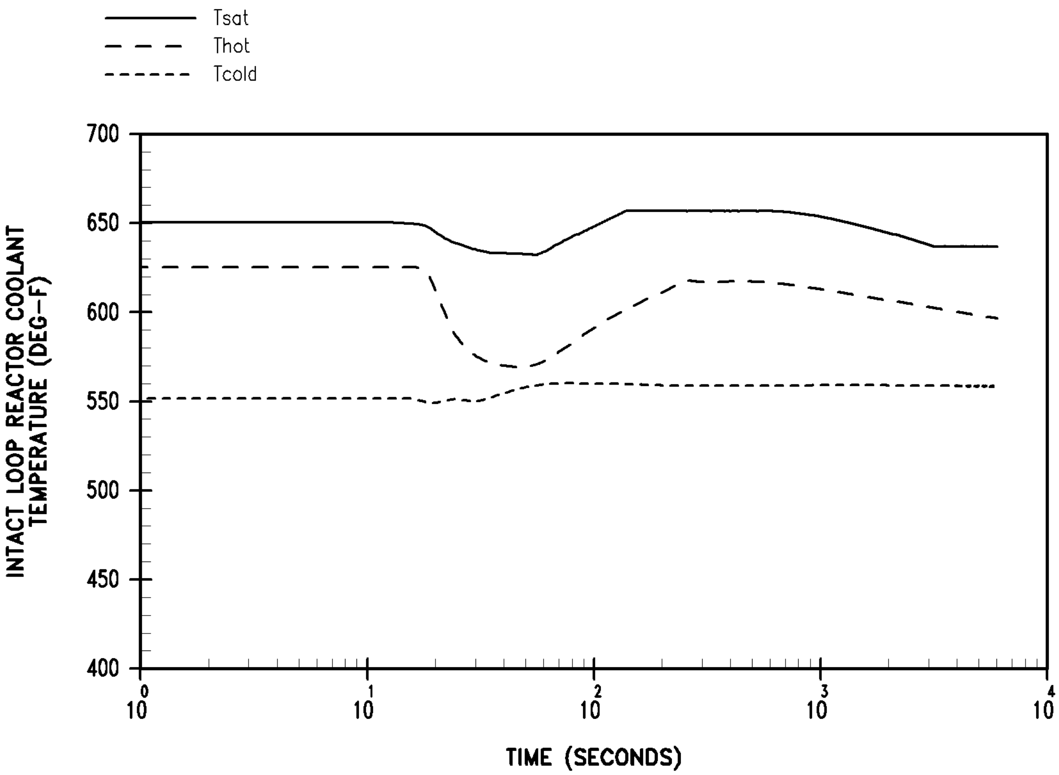


Figure 15.2-28

Reactor Coolant Temperature Transients for an Intact Loop for Main Feedline Rupture without Offsite Power Available (1.36 ft^2)

Beaver Valley Power Station Unit No. 2
Updated Final Safety Analysis Report

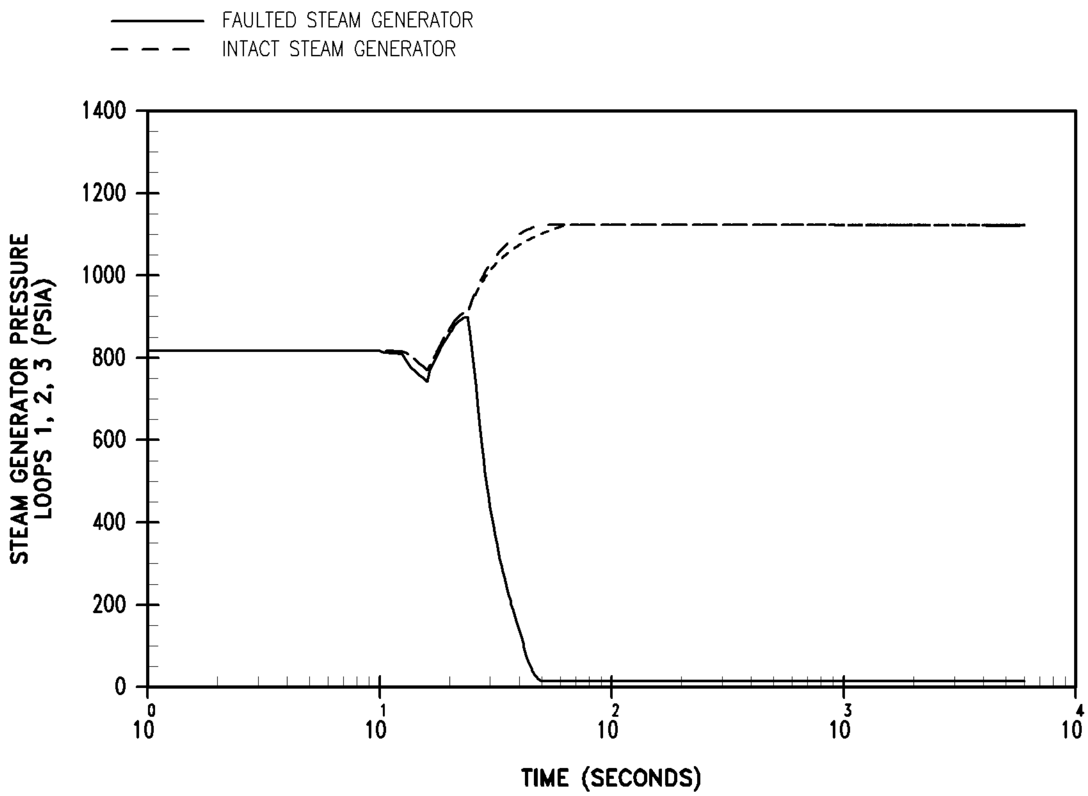


Figure 15.2-29

Steam Generator Pressure Transients for Main Feedline Rupture without Offsite Power Available (1.36 ft^2)

Beaver Valley Power Station Unit No. 2 Updated Final Safety Analysis Report

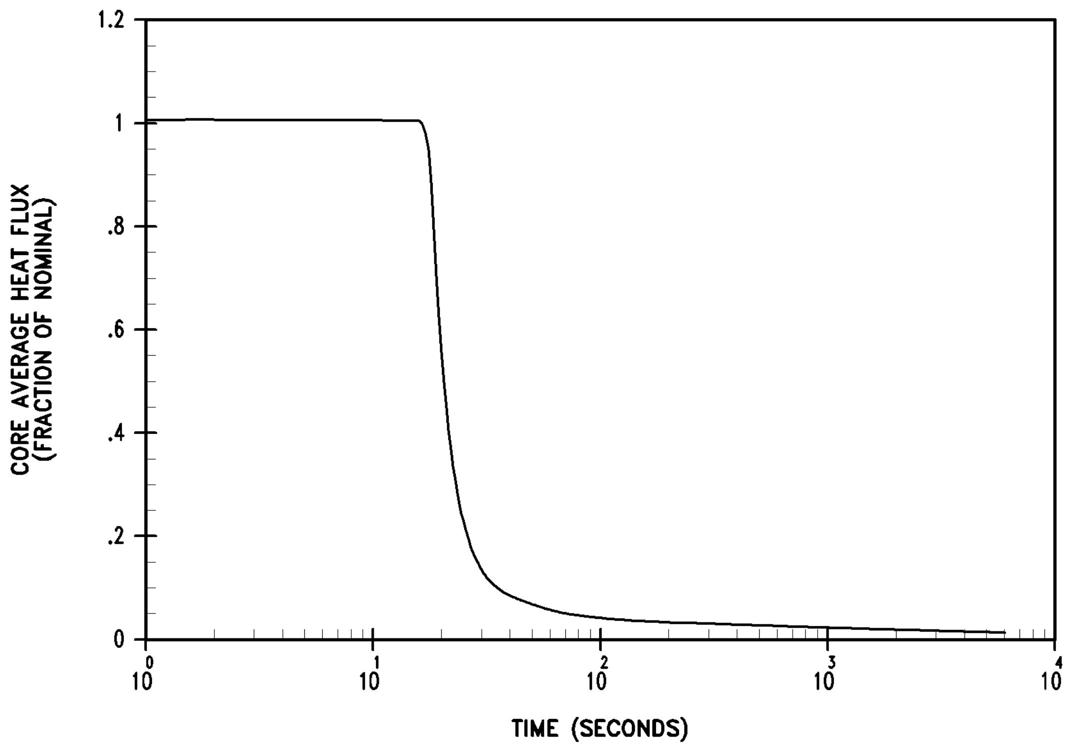


Figure 15.2-30

Core Average Heat Flux Transient for Main Feedline Rupture without Offsite Power Available (1.36 ft^2)

Beaver Valley Power Station Unit No. 2 Updated Final Safety Analysis Report

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 (RCS) flow. These events are discussed in this section. Detailed analyses are presented for the most limiting of these events.

Discussions of the following events which result in a flow decrease are presented:

1. Partial loss of forced reactor coolant flow.

2. Complete loss of forced reactor coolant flow.

3. Reactor coolant pump (RCP) shaft seizure (locked rotor).

4. Reactor coolant pump shaft break.

Item 1 is considered to be an American Nuclear Society (ANS) Condition II event, item 2 an ANS Condition III event, and items 3 and 4, ANS Condition IV events (Section 15.0.1).

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 (LOCF) accident can result from a mechanical or electrical failure in a RCP, or from a fault in the power supply to the pump supplied by a RCP bus. If the reactor is at power at the time of the accident, the immediate effect of LOCF is a rapid increase in the coolant temperature. This transient is analyzed to ensure that a reactor trip maintains an adequate safety margin to ensure that there is no departure from nucleate boiling (DNB) or subsequent fuel damage.

Normal power for the RCPs is supplied through individual buses connected to the main generator. When a generator trip occurs, the buses are automatically transferred to an offsite power supply. The RCPs will continue to supply coolant flow to the core. Following any turbine trip where there are no electrical faults or thrust bearing failures, which require tripping the generator from the network, the generator remains connected to the network for approximately 30 seconds, 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 Section 15.0.1. The necessary protection against a partial LOCF accident is provided by the low primary coolant flow reactor trip which is actuated by two out of three low flow signals in any reactor coolant loop. Above Permissive 8, low flow in any loop will actuate a reactor trip. Between approximately 10 percent power (Permissive 7) and the power level corresponding to Permissive 8, low flow in any two loops will actuate a reactor trip. Above Permissive 7, two or more RCP circuit breakers opening will actuate a reactor trip which serves as a backup to the low flow trip.

15.3.1.2 Analysis of Effects and Consequences

Method of Analysis

The loss of one reactor coolant pump with three loops in operation has been analyzed. The analysis is performed to bound operation with steam generator tube plugging levels up to 22% (maximum loop-to-loop plugging difference of 10%) with a maximum loop-to-loop flow asymmetry of 5%.

This transient is analyzed by three digital computer codes: 1) the LOFTRAN (Burnett et al 1984) Code 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; 2) the FACTRAN (Hargrove 1989) Code is then used to calculate the heat flux transient based on the nuclear power and flow from LOFTRAN; and 3) the VIPRE Code (Section 4.4 Sung, et al., 1999) is used to calculate the departure from nucleate boiling ratio (DNBR) during the transient based on the heat flux determined by FACTRAN 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-A (Friedland and Ray 1989).

Initial Conditions

Initial core power is assumed to be at its nominal value consistent with steady-state full-power operation. RCS pressure is at its nominal value minus a 7.5 psi bias and the RCS vessel average temperature is at its nominal value plus a 4.5°F bias. Uncertainties in initial conditions are included in the DNBR limit as described in WCAP-11397-P-A.

Reactivity Coefficients

The most negative Doppler-only power coefficient is used (Figure 15.0-2). This is equivalent to a total integrated Doppler reactivity from 0 to 100 percent power of 0.016 $\Delta\rho$.

A zero moderator temperature coefficient was assumed in the analysis.

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 which are necessary to mitigate the effects of the accident are discussed in Section 15.0.8 and listed in Table 15.0-6. No single active failure in any of these systems or equipment will adversely affect the consequences of the accident.

Results

Figures 15.3-1, 15.3-2, 15.3-3 and 15.3-4 show the transient response for the loss-of-reactor coolant pump with three loops initially in operation. Figure 15.3-4 shows the DNBR versus time.

The results of the partial loss of flow transient confirm that the minimum DNBR acceptance criterion is met.

Since DNB is not expected to occur, the ability of the reactor coolant to remove heat from the fuel is not significantly 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 in Table 15.3-1. The affected RCP will continue to coast down, and the core flow will reach a new equilibrium value associated with the two pumps still in operation. Following reactor trip, the plant will come to a stabilized condition at hot standby with one or more RCPs in operation. Normal operating procedures may then be followed. The operating procedures would call for operator action to control RCS boron concentration and pressurizer level using the chemical and volume control system (CVCS), and to maintain steam generator level through control of the main or auxiliary feedwater system (AFWS). 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.

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, in addition, that the condenser is not available, atmospheric steam dump may be required.

There are only minimal radiological consequences associated with this event. Fuel damage as a result of this transient is not postulated. The radiological consequences resulting from atmospheric steam dump are less severe than those of the loss of non-emergency ac power to station auxiliaries described in Section 15.2.6.

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 power supply or a reduction in power supply frequency to all RCPs. If the reactor is at power at the time of the accident, the immediate effect of LOCF is a rapid increase in the coolant temperature. This transient is analyzed to ensure that a reactor trip maintains an adequate safety margin to ensure that there is no DNB or subsequent fuel damage.

Normal power for the RCPs is supplied through buses from a transformer connected to the main generator. When a generator trip occurs, the buses are automatically transferred to an offsite power supply. The RCPs will continue to supply coolant flow to the core. Following any turbine trip where there are no electrical faults or thrust bearing failures which require tripping the generator from the network, the generator remains connected to the network for approximately 30 seconds, 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 Section 15.0.1.

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

1. Reactor coolant pump power supply undervoltage or underfrequency. (anticipatory)

2. Low reactor coolant loop flow. (credited in the analysis)

3. Overpower ΔT reactor trip function. (diverse trip)

The reactor trip on RCP undervoltage is provided to protect against conditions which can cause a loss of voltage to all RCPS, that is, loss of offsite power. However, due to the possibility of a common mode failure in the cabinets containing the circuitry, the Undervoltage and Underfrequency reactor trip functions may not be available. The Undervoltage and Underfrequency trip functions are blocked below approximately 10% power (Permissive P-7). If the Undervoltage and Underfrequency reactor trips were not available, the low reactor coolant flow trip is available to provide the reactor trip function. Reactor protection system diversity is provided by the overpower ΔT reactor trip function.

The reactor trip on RCP underfrequency is provided to trip the reactor for an underfrequency condition resulting from frequency disturbances on the power grid and provides an anticipatory trip to the reactor coolant loop loss flow trip. If the maximum grid frequency decay rate is less than approximately 5 Hz/sec, this trip function will enhance protection of the core from underfrequency events without requiring tripping of the RCP breakers. Chapter 7 discusses the interface requirements concerning tripping of the RCP breakers for underfrequency events. Baldwin (et al 1975) provides analyses of grid frequency disturbances and the resulting nuclear steam supply system protection requirements which are generally applicable.

The reactor trip on low reactor 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 flow signals per reactor coolant loop. Above Permissive 8, low flow in any loop will actuate a reactor trip. Between approximately 10 percent power (Permissive 7) and the power level corresponding to Permissive 8, low flow in any two loops will actuate a reactor trip. If the maximum grid frequency decay rate is low enough, this trip function will protect the core from underfrequency events. This effect is fully described by Baldwin (et al 1975).

Normal power for the reactor coolant pumps is supplied through busses from a transformer connected to the generator. Each pump is on a separate bus. When the generator trip occurs, the busses are automatically transferred to a transformer supplied from external power lines, and the pumps will continue to supply coolant flow to the core. Following any turbine trip, where there are no electrical faults which require tripping the generator from the network, the generator remains connected to the network for approximately 30 seconds. The reactor coolant pumps remain connected to the generator thus ensuring full flow for 30 seconds after the reactor trip before any transfer is made.

15.3.2.2 Analysis of Effects and Consequences

Method of Analysis

These transients are analyzed by three digital computer codes. First, the LOFTRAN Code 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 FACTRAN (Hargrove 1989) Code is then used to calculate the heat flux transient based on the nuclear power flow from LOFTRAN. Finally, the VIPRE Code is used to calculate the DNBR during the transient based on the heat flux determined by FACTRAN and flow from LOFTRAN. The DNBR transients presented represent the minimum of the typical or thimble cell.

The complete loss of flow event results in a loss of forced reactor coolant flow to all loops. Hence, the modeling of initial asymmetric loop-to-loop flow variations is not necessary in the analysis for this event.

Initial core power is assumed to be at its nominal value consistent with steady-state, full-power operation. RCS pressure is at its nominal value minus a 7.5 psi bias and the RCS vessel average temperature is at its nominal value plus a 4.5°F bias. Uncertainties in initial conditions are included in the limit DNBR as described in WCAP-11397-P-A.

Results

Figures 15.3-9, 15.3-10, 15.3-11 and 15.3-12 show the transient response for the more limiting case; the frequency decay complete loss of flow event with three loops in operation.

Since a violation of the DNBR limit is not expected to occur, the ability of the primary coolant to remove heat from the fuel rod is not greatly reduced. Thus, the average fuel and cladding temperatures do not increase significantly above their respective initial values. The calculated sequence of events for the case analyzed is shown in Table 15.3-1. A frequency decay of 5 hz/sec is assumed to occur, eventually tripping the reactor on a low Reactor Coolant loop flow signal. Eventually natural circulation flow will be established. With the reactor tripped, a stable plant condition will be attained. Normal plant shutdown may then proceed. 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 feedwater system or AFWS.

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 less than that of the main steam line break described in Section 15.1.5.

There are only minimal radiological consequences associated with this event. Since fuel damage is not postulated, the radiological consequences resulting from this event are less severe than those of the MSLB analyzed in Section 15.1.5.3.

15.3.2.4 Conclusions

The analysis performed has demonstrated that for the complete loss of forced reactor coolant flow, the minimum DNBR acceptance criterion is met. Thus, the DNB design-basis as described in Section 4.4 is met. The radiological consequences are not limiting.

15.3.3 Reactor Coolant Pump Shaft Seizure (Locked Rotor)

15.3.3.1 Identification of Causes and Accident Description

The accident postulated is an instantaneous seizure of a RCP rotor (Section 5.4). Flow through the affected reactor coolant loop is rapidly reduced, leading to an initiation of a reactor trip on a low flow signal.

Following initiation of the reactor trip, heat stored in the fuel rods continues to be transferred to the coolant causing the coolant to expand. At the same time, heat transfer to the shell side of the steam generators 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 reduced heat transfer in the steam generators causes an insurge into the pressurizer and a pressure increase throughout the RCS. The insurge into the pressurizer compresses the steam volume, actuates the automatic spray system, opens the power-operated relief valves (PORVs), and opens the pressurizer safety valves in that sequence. 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 the analysis.

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

15.3.3.2 Analysis of Effects and Consequences

Method of Analysis

Two locked rotor cases are analyzed; they are:

1. Peak RCS pressure resulting from a locked rotor in one of three loops

2. Number of rods-in-DNB resulting from a locked rotor in one of three loops

The first case is aimed at maximizing the RCS pressure transient. This is done using the Standard Thermal Design Procedure. Thermal design flow is assumed. Initial core power, reactor coolant temperature and pressure are assumed allowed to be at their maximum values consistent with full-power conditions including allowances for calibration and instrument errors (i.e., initial power includes a 0.6% power calorimetric uncertainty, initial pressure includes a +45 psi uncertainty and initial RCS vessel average temperature includes a +8.5°F uncertainty). This assumption results in a conservative calculation of the coolant insurge into the pressurizer, which in turn results in a maximum calculated peak RCS pressure. The pressure responses shown on Figure 15.3-18 are the responses at the point in the RCS having the maximum pressure.

The peak pressure case is analyzed using two digital computer codes. The LOFTRAN Code 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, which used the core flow and nuclear power calculated by LOFTRAN. The FACTRAN Code includes a film boiling heat transfer coefficient.

The second case is an evaluation of DNB in the core during the transient. This case is analyzed using the Revised Thermal Design Procedure (Friedland and Ray 1989). Initial core power is assumed to be at its nominal value consistent with steady-state, full-power operation. RCS pressure is at its nominal value minus a 7.5 psi bias, and RCS vessel average temperature is at its nominal value plus a 4.5°F bias. Uncertainties in initial conditions are included in the limit DNBR as described in WCAP-11397-P-A.

The rods-in-DNB case is analyzed using two digital computer codes, LOFTRAN and VIPRE. The LOFTRAN Code is used in the same manner as in the peak pressure case discussed above. The VIPRE Code is used to calculate the core heat flux and DNBR during the transient based on the nuclear power and core flow from LOFTRAN.

The analyses for both cases assume a zero moderator temperature coefficient (MTC) and a conservatively large (absolute value) of the doppler-only power coefficient. The negative reactivity from control rod insertion/scram for both cases is based on 4.0% $\Delta k/k$ trip reactivity from hot full power.

In both cases the analysis is performed to bound operation with steam generator tube plugging levels up to 22% (maximum loop-to-loop plugging difference of 10%) with a maximum loop-to-loop flow asymmetry of 5%.

<u>Evaluation of the Pressure Transient</u>

After pump seizure, the neutron flux is rapidly reduced by control rod insertion. Rod motion is assumed to begin one second after the flow in the affected loop reaches 87 percent of nominal flow. The time delay of 1.0 second used in connection with the low flow reactor trip is a very conservative allowance for the total time delay between the time the flow reaches 87 percent of the nominal and the time the rods begin moving into the core. This total includes individual delays associated with the following: Flow sensors/transmitters, solid state protection system input relays, solid state protection system, voltage drop on reactor trip breaker undervoltage, and control rod gripper release. No credit is taken for the pressure reducing effect of the pressurizer power-operated 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. The pressurizer safety valves are full open at 2,605 psia (including 3% uncertainty and 1% set point shift) and their capacity for steam relief is as described in Section 5.4.

Evaluation of DNB in the Core During the Accident

For this accident, DNB is assumed to occur in the core, and 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 cladding temperature and zirconium water reaction.

In the evaluation, the rod power at the hot spot is assumed to be 2.52 times the average rod power level (that is, F_Q at the initial core power = 2.52). The number of rods in DNB was conservatively calculated to be less than 20% of the total rods in the core.

Film Boiling Coefficient

The film boiling coefficient is calculated in the FACTRAN Code using the Bishop-Sandberg-Tong film boiling correlation. 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 the bulk density are used throughout the transient since they are the most conservative with respect to cladding temperature response. For conservatism, 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 clad (gap coefficient) has a pronounced influence on the thermal results. The larger the value of the gap coefficient, the more heat is transferred between pellet and cladding. Based on investigations on the effect of the gap coefficient upon maximum cladding temperature during the transient, the gap coefficient was assumed to increase from a steady-state value consistent with the initial fuel temperature to 10^4 Btu/hr-ft^2-°F at the initiation of the transient. Thus, the large amount of energy stored in the fuel because of the small initial value is released to the cladding at the initiation of the transient.

Zirconium Steam Reaction

The zirconium-steam reaction can become significant above 1,800°F (cladding temperature). The Baker-Just parabolic rate equation shown as follows is used to define the rate of the zirconium-steam reaction.

$$\frac{d(w^2)}{dt} = 33.3 \ x \ 10^6 \exp\left[\frac{(-45{,}500)}{1.986T}\right]$$

where:

w = amount reacted (mg/cm^2)

t = time (sec)

T = temperature (°K)

The reaction heat is 1,510 cal/gm.

The effect of zirconium-steam reaction is included in the calculation of the "hot spot" cladding temperature transient.

Plant systems and equipment which are available to mitigate the effects of the accident are discussed in Section 15.0.8 and listed in Table 15.0-6. No single active failure in any of these systems or equipment will adversely affect the consequences of the accident.

Results

The transient results for the peak pressure case are shown on Figures 15.3-17, 15.3-18, 15.3-19 and 15.3-20. The results of these calculations are summarized in Table 15.3-2. 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 cladding surface temperature is considerably less than 2,700°F. It should be noted that the cladding temperature was conservatively calculated assuming that DNB occurs at the initiation of the transient.

The calculated sequence of events for the peak pressure case analyzed is shown in Table 15.3-1. With the reactor tripped, a stable plant condition will eventually be attained. Normal plant shutdown may then proceed.

Following reactor trip, Beaver Valley Power Station - Unit 2 (BVPS-2) will approach a stabilized condition at hot standby; normal plant operating procedures may then be followed to maintain a hot condition or to cool the plant to cold shutdown. 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 feedwater system or AFWS. Any action required of the operator to maintain BVPS-2 in a stabilized condition will be in a time frame in excess of ten minutes following reactor trip.

15.3.3.3 Radiological Consequences

Computer program PERC2 is used to calculate the control room and site boundary doses due to airborne radioactivity releases following a locked rotor accident at BVPS at power uprate conditions. Bounding parameter values are used to encompass an event at either unit.

The dose assessment follows the guidance provided in Regulatory Guide 1.183. Table 15.3-3 lists the key assumptions and parameters utilized to develop the radiological consequences following a BVPS locked rotor accident.

A locked rotor accident is assumed to result in 20% failed fuel and a release of the associated gap activity. The gap activity (consisting of noble gases, halogens and alkali metals) are instantaneously and homogeneously mixed in the reactor coolant system and transmitted to the secondary side via primary to secondary steam generator tube leakage assumed to be at the value of 450 gpd.

A radial peaking factor of 1.75 is applied to the activity release. The chemical form of the iodines in the gap are assumed to be 95% CsI, 4.85% elemental and 0.15% organic. At BVPS, the SG tubes remain submerged for the duration of the event; therefore, the gap iodines are assumed to have a partition coefficient of 100 in the steam generators. The iodine releases from the steam generators are assumed to be 97% elemental and 3% organic. The gap noble gases are assumed to be released freely to the environment without retention in the steam generators, whereas the particulates are carried over in accordance with the design basis steam generators moisture carryover fraction.

The condenser is assumed unavailable due to a coincident loss of offsite power. Consequently, the radioactivity release resulting from a locked rotor event is discharged to the environment from the steam generators via the MSSVs and the ADVs. The steam generator releases continue for 8 hours, at which time shutdown cooling is initiated via operation of the RHR system, and environmental releases are terminated.

The activity associated with the release of secondary steam and liquid, and primary to secondary leakage of normal operation RCS, (both at Technical Specification activity limits) via the MSSVs/ADVs is insignificant compared to the failed fuel release, and are therefore not included in this assessment.

The environmental releases for a postulated locked rotor accident are combined with the atmospheric dispersion values presented in Tables 15.0-11, 15.0-14 and 15.0-15 to determine the site boundary and control room doses given in Table 15.0-12 and 15.0-13, respectively. The most limiting atmospheric dispersion factors between the MSSVs/ADVs at each unit relative to the two CR intakes (identified for purposes of assessment as the BVPS-1 MSSVs/ADVs to the BVPS-1 CR intake) is selected to determine a bounding control room dose.

EAB 2-Hour Worst Case Window

AST methodology requires that the worst case dose to an individual located at any point on the boundary at the EAB, for any 2-hr period following the onset of the accident be reported as the EAB dose. For the locked rotor event, the worst two hour period can occur either during the 0-2 hr period when the noble gas release rate is the highest, or during the 6-8 hr period when the iodine and particulate level in the SG liquid peaks (SG releases are terminated at T=8 hrs). Regardless of the starting point of the worst 2 hr window, the 0-2 hr EAB X/Q is utilized.

Accident Specific Control Room Model Assumptions

The control room is conservatively assumed to remain in the normal operation mode. The critical control room parameters utilized in this model are provided in Table 6.4-1a. Section 15.6.5.4 discusses the control room design as related to dose consequences under a sub-section titled "Control Room Habitability."

The methodology used in calculating the offsite doses is discussed in Appendix 15A. The radiological consequences for a locked rotor event do not exceed the dose limits provided in 10 CFR 50.67 as supplemented by SRP 15.0.1 and Regulatory Guide 1.183.

15.3.3.4 Conclusions

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.

Since the peak cladding surface temperature calculated for the hot spot during the worst transient remains considerably less than 2,700°F, the core will remain in place and intact with no loss of core cooling capability. No fuel failures are predicted for the locked rotor accident (Van Houten 1979). It should be noted that the cladding temperature was conservatively calculated assuming that DNB occurs at the initiation of the transient.

15.3.4 Reactor Coolant Pump Shaft Break

15.3.4.1 Identification of Causes and Accident Description

The accident is postulated as an instantaneous failure of an RCP shaft, such as discussed in Section 5.4. Flow through the affected reactor coolant loop is rapidly reduced, though the initial rate of reduction of coolant flow is greater for the RCP rotor seizure event (Section 15.3.3). With a failed shaft, the pump impeller could conceivably be free to spin in the reverse direction instead of being in a fixed position. The effect of such reverse spinning is a slight decrease in the end point (steady-state) core flow.

The analysis presented in Section 15.3.3 represents the limiting condition, assuming a locked rotor for forward flow but a free-spinning shaft for reverse flow in the affected loop.

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

15.3.4.2 Conclusions

The conclusions of Section 15.3.3.4 apply for a reactor coolant pump shaft break accident.

15.3.5 References for Section 15.3

Baldwin, M.S., et al 1975. An Evaluation of Loss of Flow Accident caused by Power System Frequency Transients in Westinghouse PWRs, WCAP-8424, Revision 1.

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

Hargrove, H.G., "FACTRAN-A Fortran Code for Thermal Transients in a UO_2 Fuel Rod," WCAP-7908-A, December 1989.

Van Houten, R. 1979. Fuel Rod as a Consequence of Departure from Nucleate Boiling or Dryout. Office of Nuclear Regulatory Research, USNRC, Washington, D.C. NUREG-0562.

DLC, Calculation "Assessment of the Doses in the Unit 2 Control Room Due to a Locked Rotor Accident at Unit 2 Assuming 18% Failed Fuel (Includes Offsite Dose Added in Rev. 1).

Friedland, A. J. and Ray, S., "Revised Thermal Design Procedure," WCAP-11397-P-A and WCAP-11398-A, April 1989.

Sung, Y. X., et. al., "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," WCAP-14565-P-A (Proprietary) and WCAP-15306-NP-A (non-Proprietary), October 1999.

Tables for Section 15.3

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 Three loops operating, one pump coasting down (with offsite power)	Coastdown begins	0
	Low flow reactor trip	1.8
	Rods begin to drop	2.8
	Minimum DNBR occurs	3.8
Complete Loss of Forced Reactor Coolant Flow (Frequency Decay) (with offsite power)	All operating pumps begin slowing down due to 5 Hz/sec frequency decay rate	0
	Low reactor coolant flow trip point reached	1.7
	Rods begin to drop	2.7
	Minimum DNBR occurs	4.7
Reactor coolant pump shaft seizure (locked rotor) (without offsite power)	Rotor on one pump locks	0
	Low flow trip point reached	0.04
	Rods begin to drop	1.04
	Reactor coolant pumps lose power, coastdown begins	1.04
	Maximum RCS pressure occurs	3.6
	Maximum clad average temperature occurs	3.9

TABLE 15.3-2

SUMMARY OF RESULTS FOR LOCKED ROTOR TRANSIENTS

Maximum reactor coolant system pressure (psia)	2,870
Maximum clad average temperature (°F) core hot spot	1,824
$Zr-H_2O$ reaction at core hot spot (percent by weight)	0.35

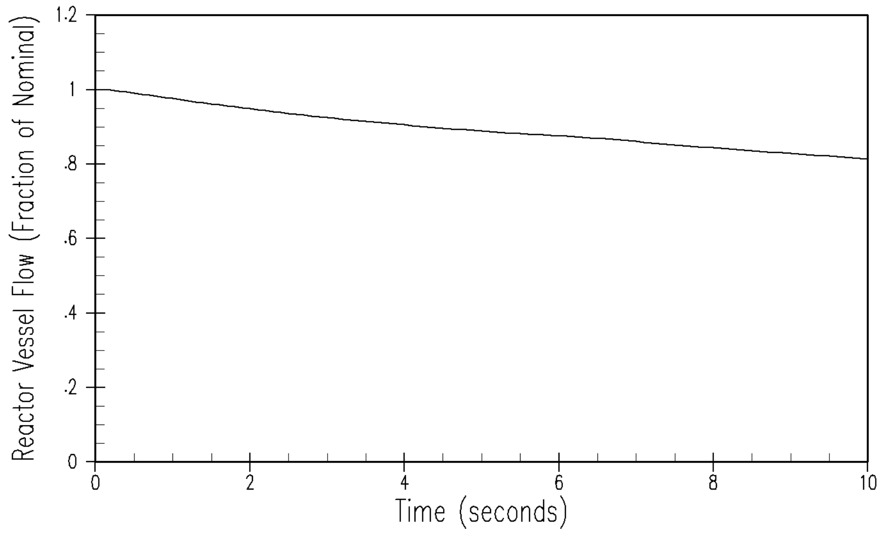
TABLE 15.3-3

PARAMETERS USED FOR THE LOCKED ROTOR ACCIDENT

Core Power Level	2918 MWt
Minimum Reactor Coolant Mass	340,711 lbm
Primary to Secondary SG tube leakage	450 gpd @ STP
Melted Fuel Percentage	0%
Failed Fuel Percentage	20%
Core Activity of Isotopes in Gap	Tables 15.0-7a and 15.0-7b
Radial Peaking Factor	1.75
Fraction of Core Inventory in Fuel gap	I-131 (8%) Kr-85 (10%) Other Noble Gases (5%) Alkali Metals (12%)
Iodine Chemical Form in Gap	4.85% elemental 95% CsI 0.15% Organic
Secondary Side Parameters:	
Minimum Post-Accident SG Liquid Mass	101,799 lbm per SG
Iodine Species released to Environment	97% elemental; 3% organic
Iodine Partition Coefficient in SGs	100 (all tubes submerged)
Particulate Carry-Over Fraction in SGs	0.0025
Steam Releases from SGs	0-2 hr (348,000 lbm) 2-8 hr (773,000 lbm)
Termination of releases from SGs	8 hours
Fraction of Noble Gas Released	1.0 (Released to Environment without holdup)
Environmental Release Point	MSSVs/ADVs
CR emergency Ventilation: Initiation Signal/Timing	CR is maintained under Normal Operation ventilation

Note:

(1) Bounding parameter values are used to encompass an event at either unit.



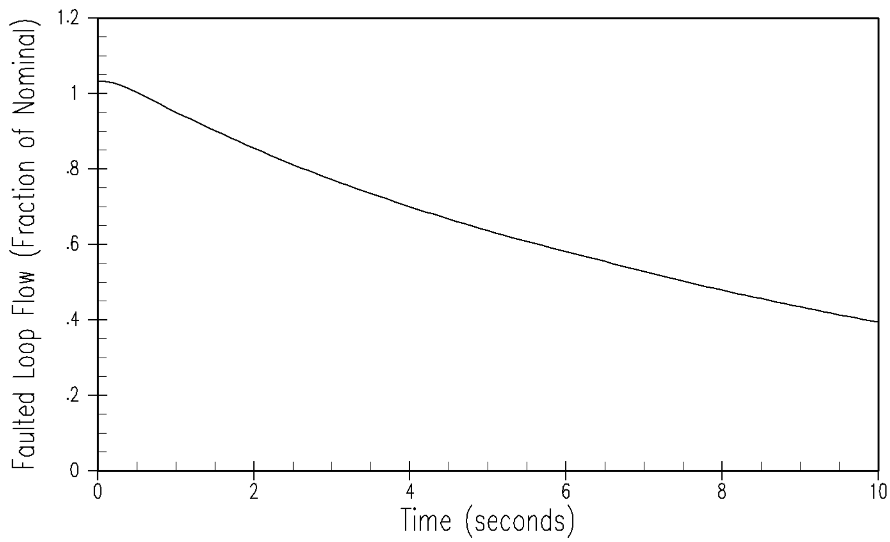
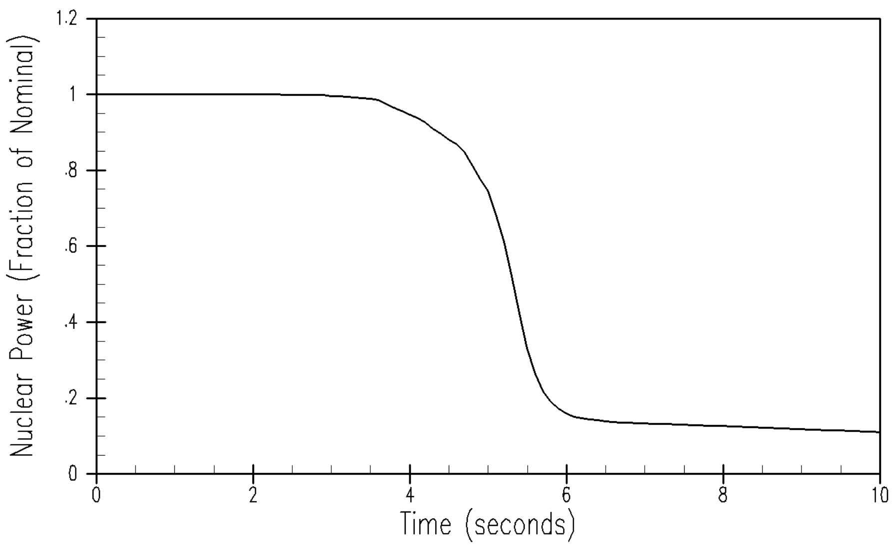


Figure 15.3-1

Reactor Vessel and Faulted Loop Flow Transients for Partial Loss of Flow, One Pump Coasting Down

Beaver Valley Power Station Unit No. 2
Updated Final Safety Analysis Report



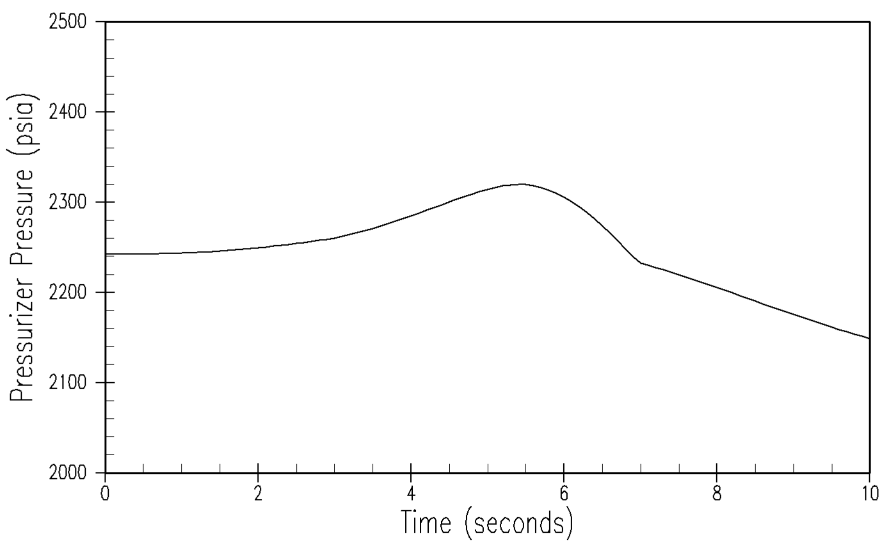


Figure 15.3-2

Nuclear Power and Pressurizer Pressure Transients for Partial Loss of Flow, One Pump Coasting Down

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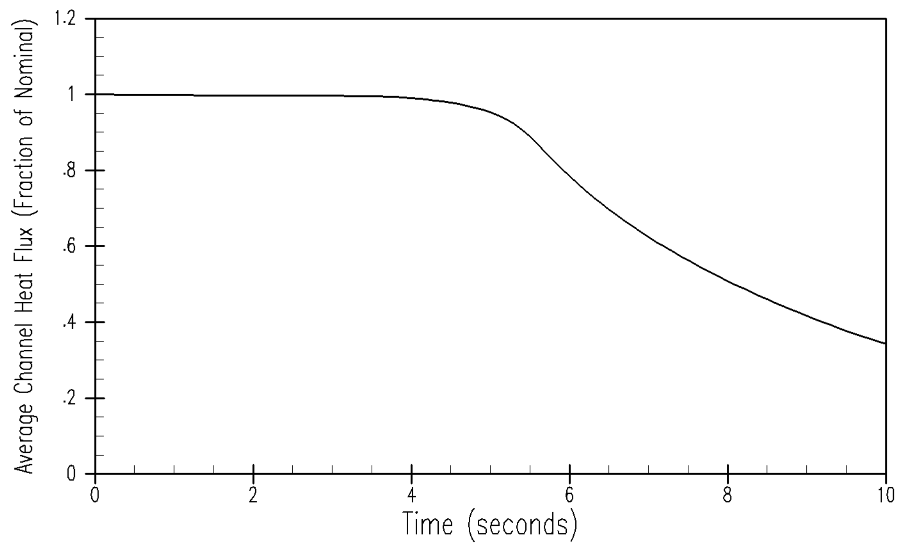

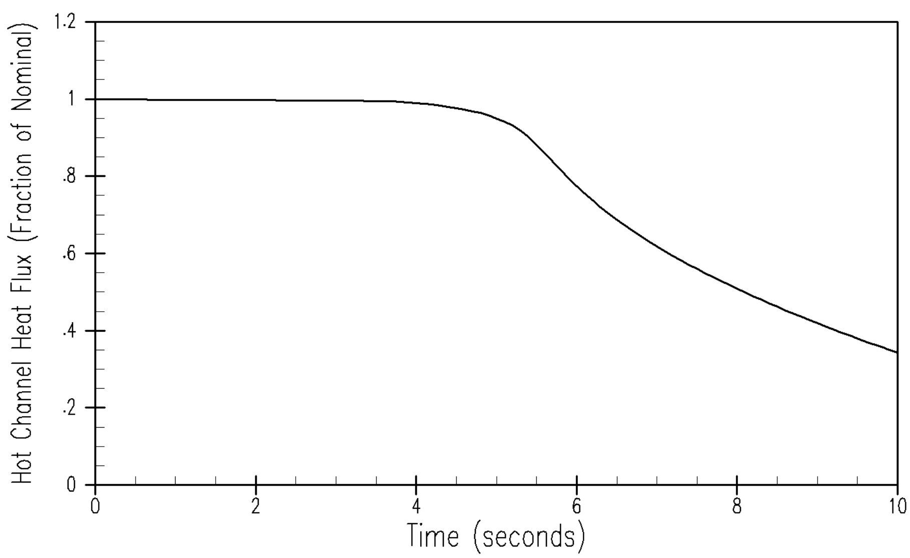

Figure 15.3-3

Average Channel and Hot Channel Heat Flux Transients for Partial Loss of Flow, One Pump Coasting Down

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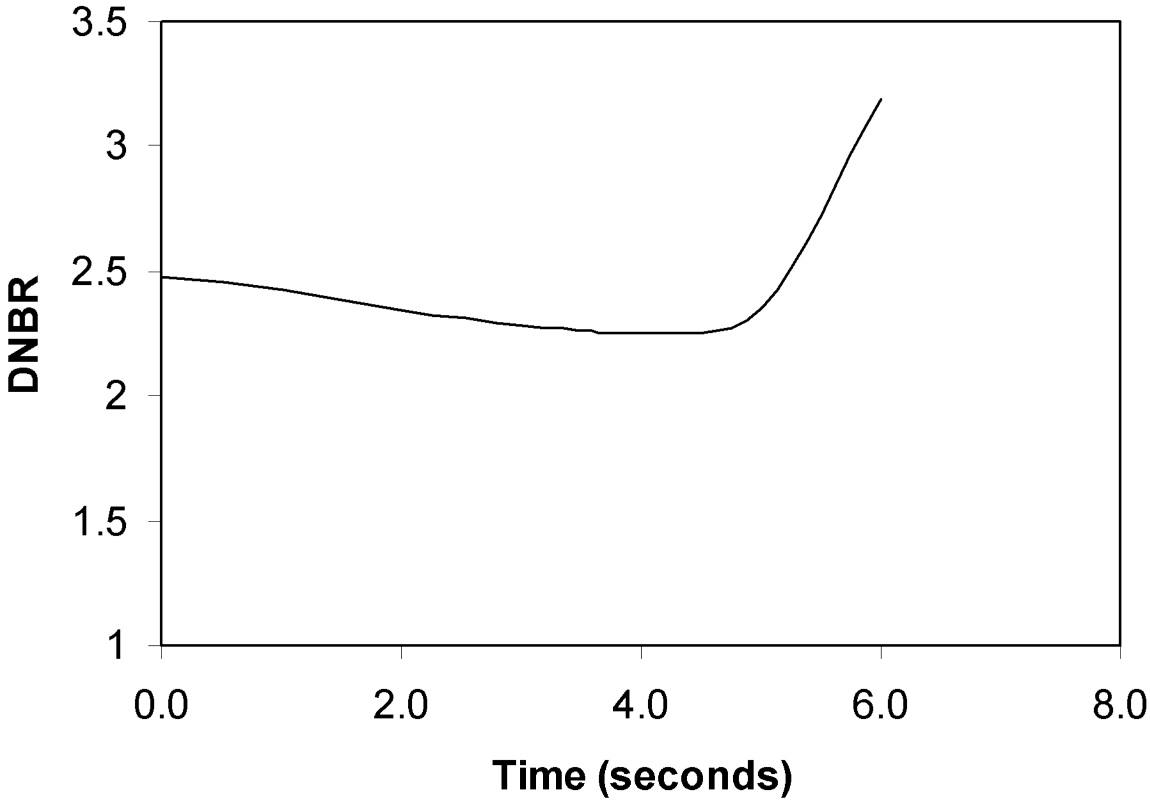

Figure 15.3-4

DNBR vs. Time for Partial Loss of Flow, One Pump Coasting Down

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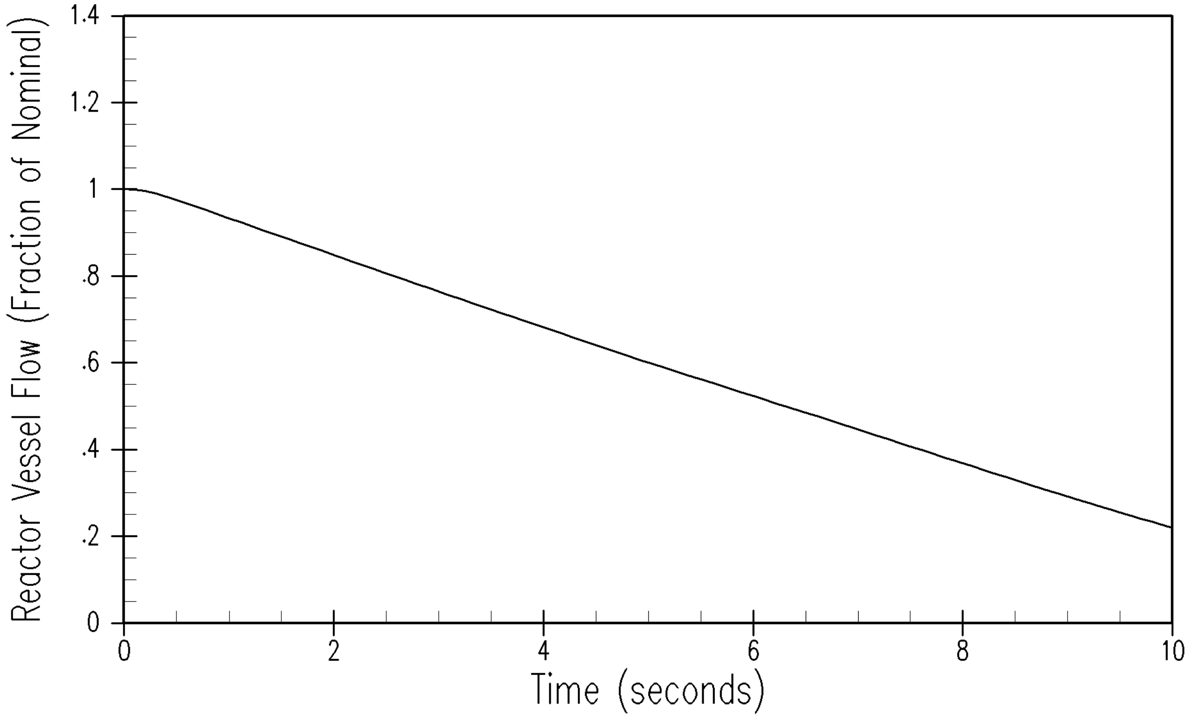

Figure 15.3-9

Core Flow Coastdown vs. Time for Three Loops in Operation, Complete Loss of Flow, Frequency Decay

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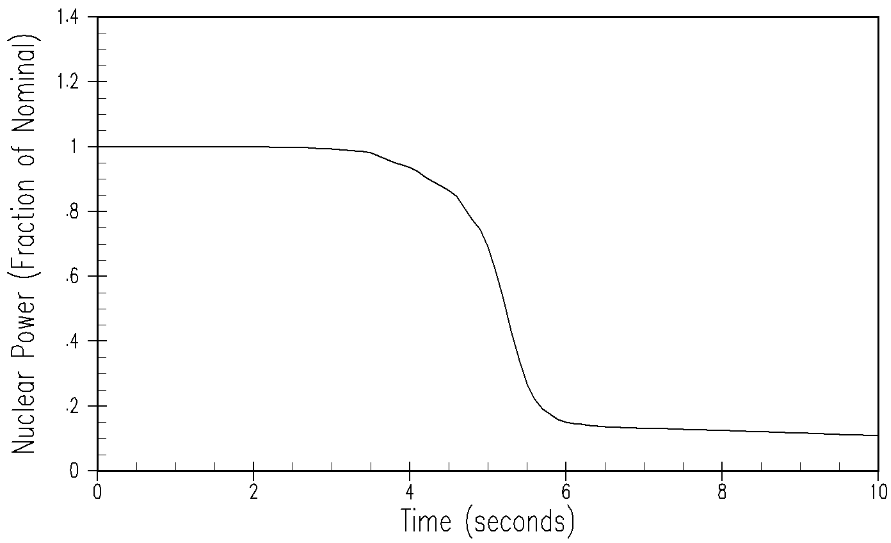

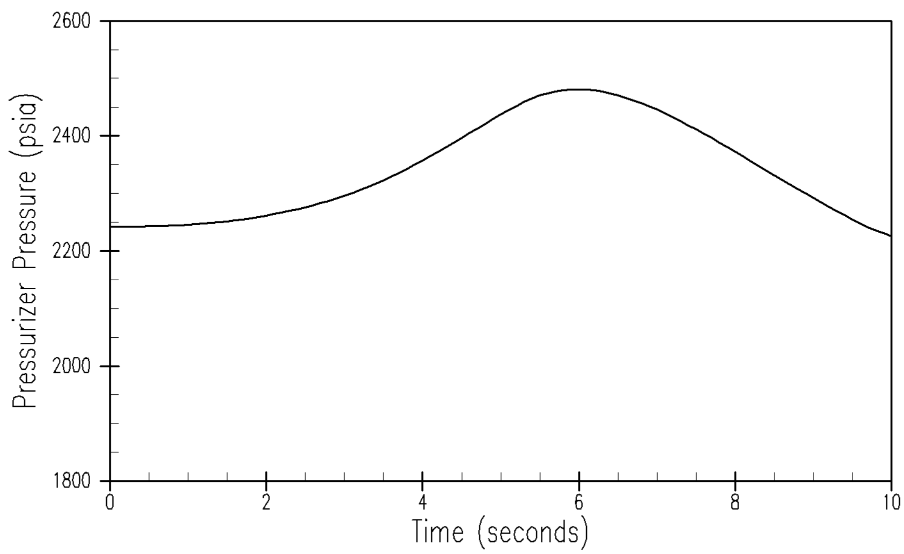

Figure 15.3-10

Nuclear Power and Pressurizer Pressure Transients for Three Loops in Operation, Complete Loss of Flow, Frequency Decay

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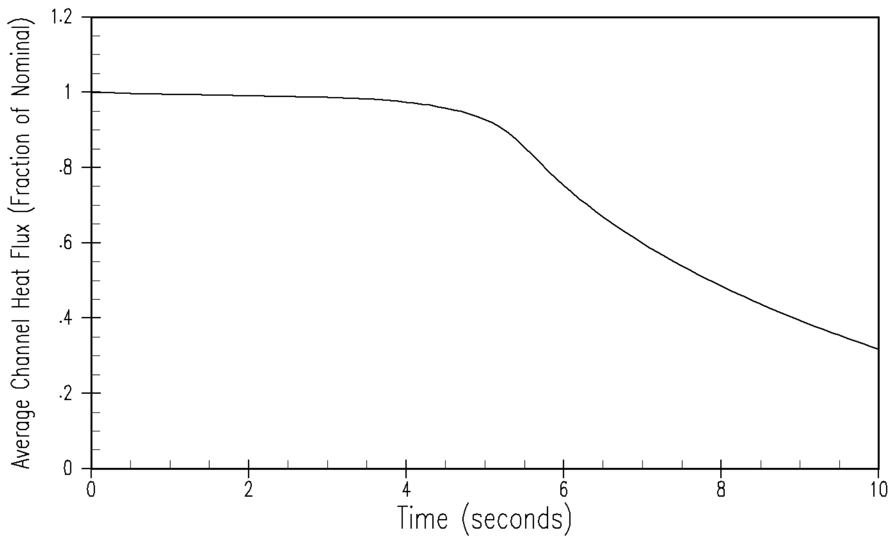

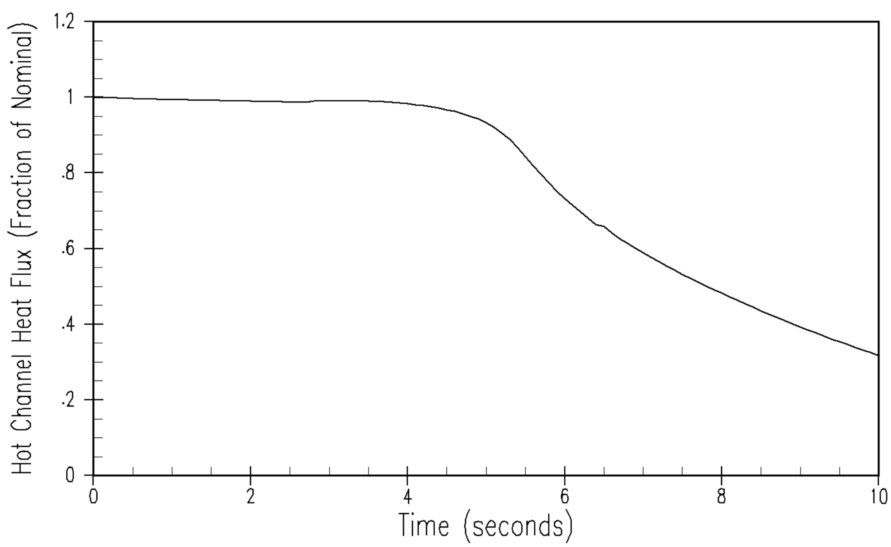

Figure 15.3-11

Average Channel and Hot Channel Heat Flux Transients for Three Loops in Operation, Complete Loss of Flow, Frequency Decay

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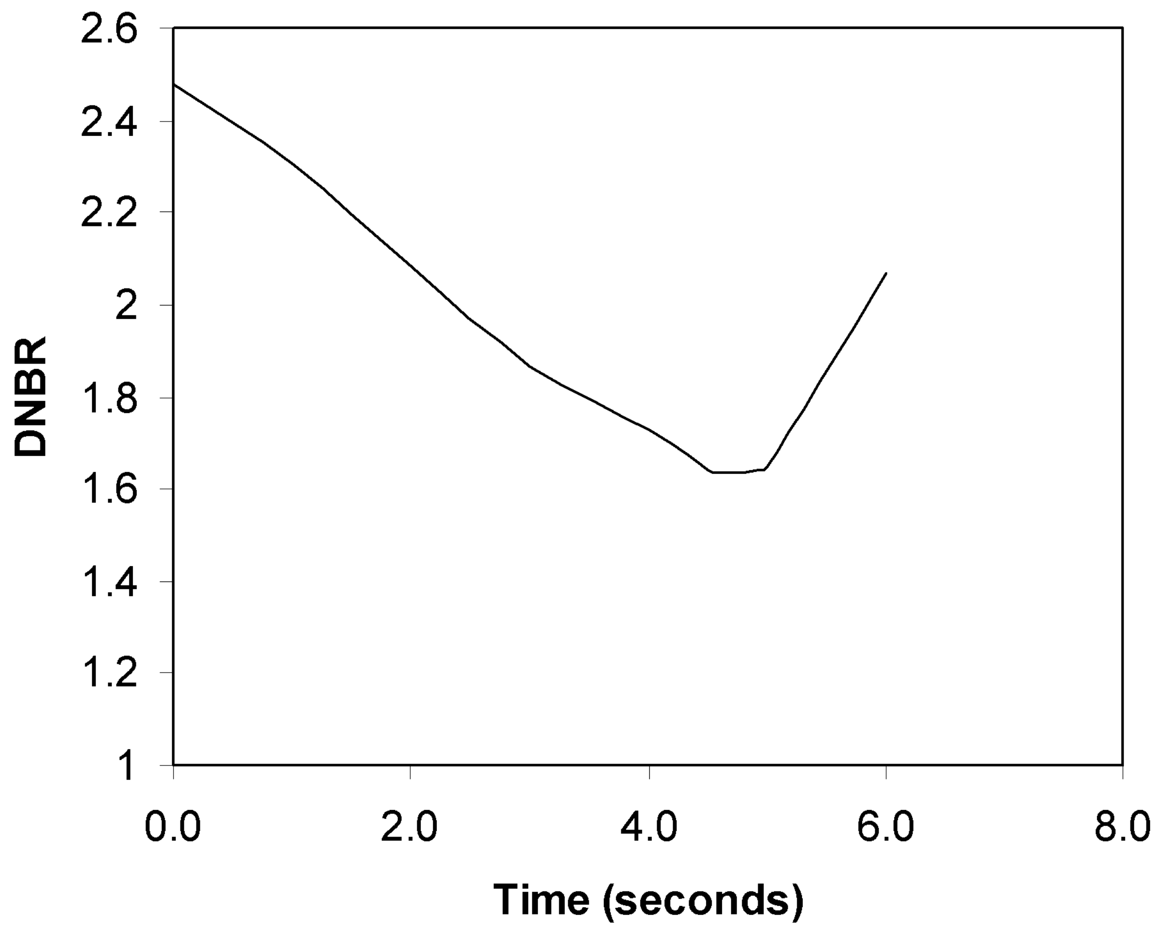

Figure 15.3-12

DNBR versus Time for Three Loops in Operation, Complete Loss of Flow, Frequency Decay

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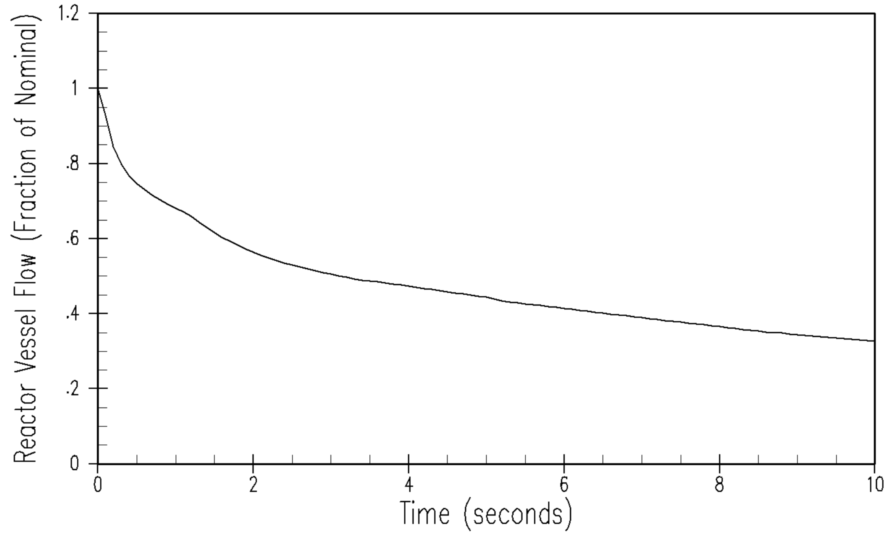

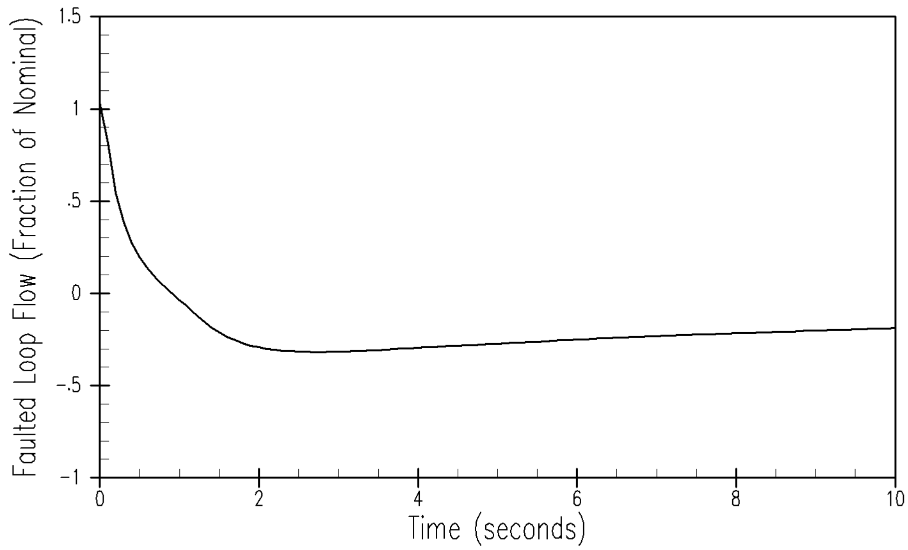

Figure 15.3-17

Reactor Vessel and Faulted Loop Flow Transients for Three Loops in Operation, One Locked Rotor

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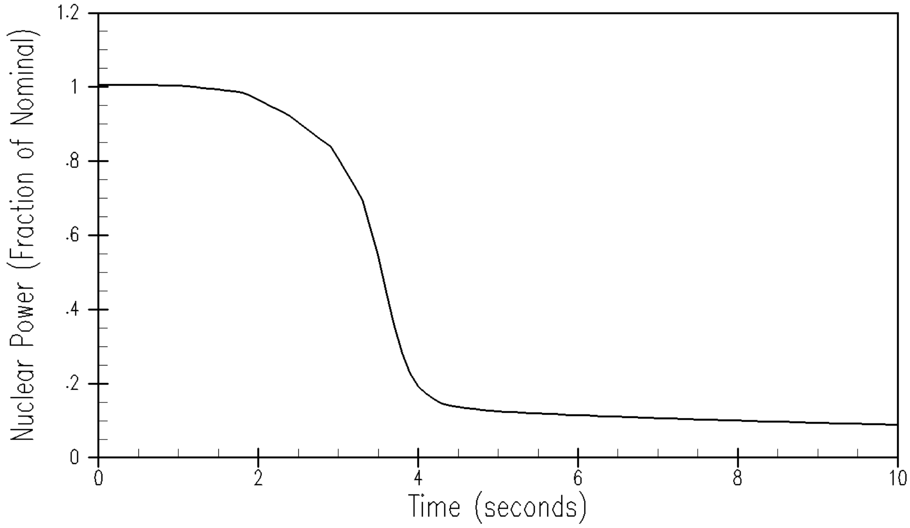

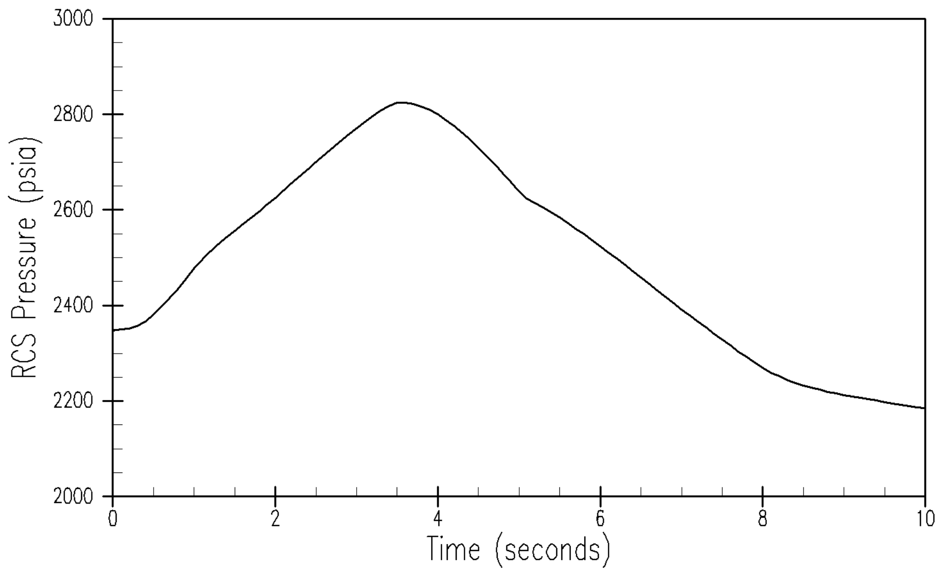

Figure 15.3-18

Nuclear Power and Reactor Coolant System Pressure for Three Loop Operation, One Locked Rotor

Beaver Valley Power Station Unit No. 2
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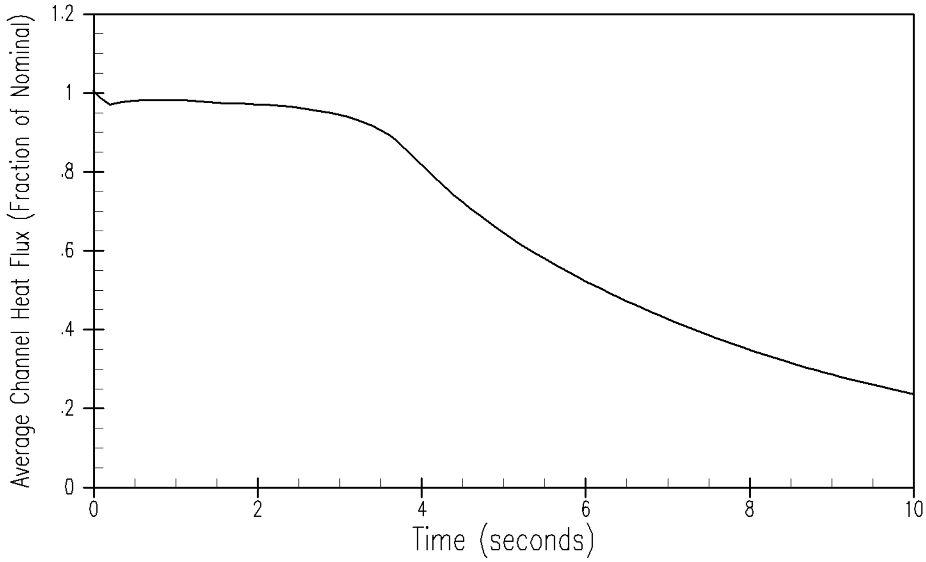

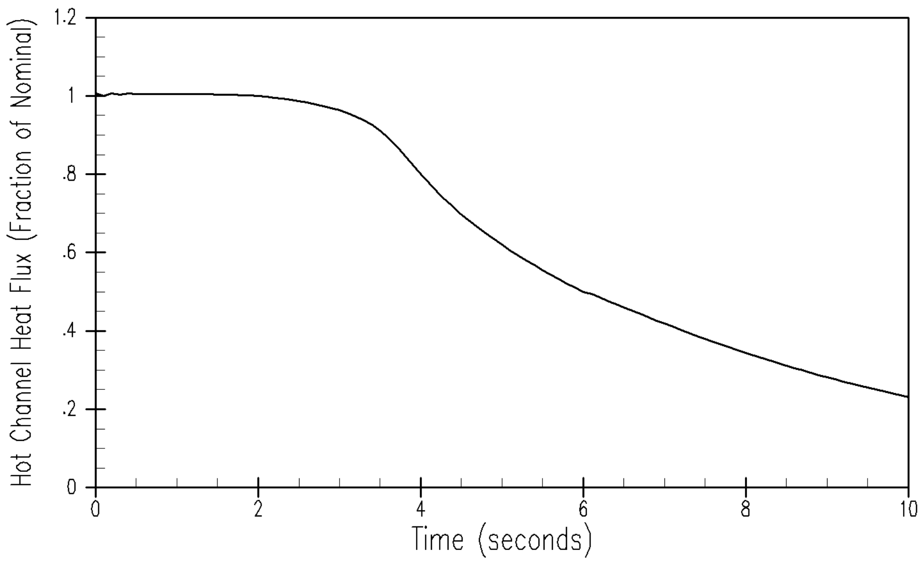

Figure 15.3-19

Average Channel and Hot Channel Heat Flux Transients for Three Loop Operation, One Locked Rotor

Beaver Valley Power Station Unit No. 2 Updated Final Safety Analysis Report

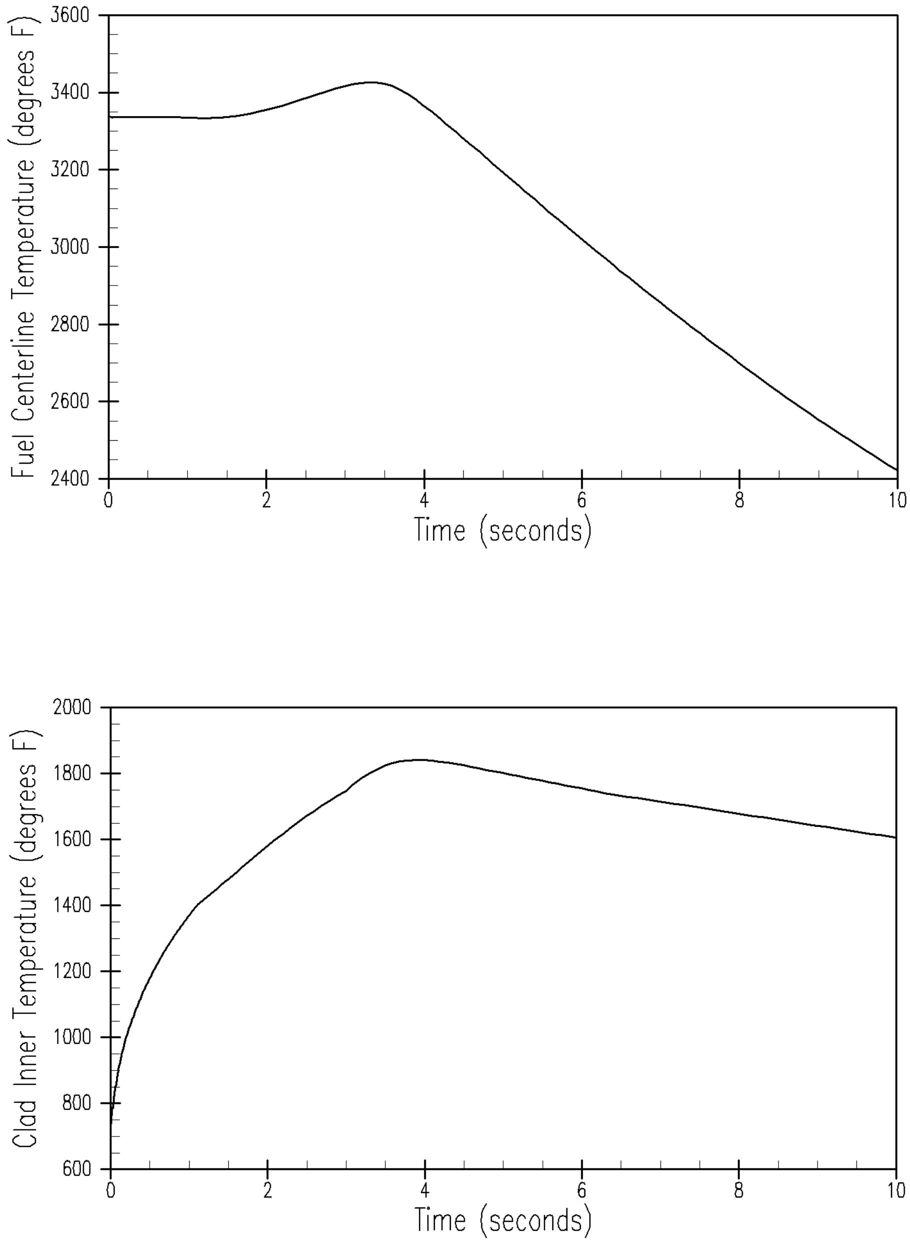

Figure 15.3-20

Maximum Clad and Fuel Centerline Temperatures at Hot Spot for Three Loop Operation, One Locked Rotor

Beaver Valley Power Station Unit No. 2
Updated Final Safety Analysis Report

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 rod cluster control assembly (RCCA) 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.

Discussions of the following incidents are presented in this section:

1. Uncontrolled RCCA bank withdrawal from a subcritical or low power start-up condition (Section 15.4.1).

2. Uncontrolled RCCA bank withdrawal at power (Section 15.4.2).

3. Rod cluster control assembly misoperation (Section 15.4.3).

4. Start-up of an inactive reactor coolant pump (RCP) at an incorrect temperature (Section 15.4.4).

5. Malfunction or failure of the flow controller in a boiling water reactor (BWR) (not applicable) (Section 15.4.5).

6. Chemical and volume control system (CVCS) malfunction that results in a decrease in the boron concentration in the reactor coolant (Section 15.4.6).

7. Inadvertent loading and operation of a fuel assembly in an improper position (Section 15.4.7).

8. Spectrum of RCCA ejection accidents (Section 15.4.8).

9. Spectrum of rod drop accident in a BWR (not applicable) (Section 15.4.9).

Items 1, 2, 4, and 6 are considered to be American Nuclear Society (ANS) Condition II events, Item 7 an ANS Condition III event, and Item 8 an ANS Condition IV event. Item 3 entails both Condition II and III events (Section 15.0.1).

15.4.1 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical or Low Power Start-up Condition

15.4.1.1 Identification of Causes and Accident Description

An RCCA withdrawal accident is defined as the 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 a malfunction of the RCS or rod control system. This could occur with the reactor either subcritical, at hot zero power, or at power. The "at power" case is discussed in Section 15.4.2.

Although the reactor is normally brought to power from a subcritical condition by means of RCCA withdrawal, initial start-up 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 (Section 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 Section 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 (RPS):

1. Source range high neutron flux reactor trip

   Actuated when either of two independent source range channels indicates a neutron flux level above a preselected manually adjustable set point. 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.

2. Intermediate range high neutron flux reactor trip

   Actuated when either of two independent intermediate range channels indicates a flux level above a preselected manually adjustable set point. This trip function may be manually bypassed only after two out of the four power range channels are reading above approximately 10 percent of full power and is automatically reinstated when three out of the four channels indicate a power level below this value.

3. 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 percent of full power. This trip function may be manually bypassed when two out of the four power range channels indicate a power level above approximately 10 percent of full power and is automatically reinstated when three out of the four channels indicate a power level below this value.

4. Power range high neutron flux reactor trip (high setting)

   Actuated when two out of the four power range channels indicate a power level above a preset set point. This trip function is always active.

5. 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 set point. This trip function is always active.

In addition, control rod stops on high intermediate range flux level (one-out-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: 1) an average core nuclear power transient calculation; 2) an average core heat transfer calculation; and 3) the departure from nucleate boiling ratio (DNBR) calculation. The average core nuclear calculation is performed using a spatial neutron kinetics code, TWINKLE (Risher and Barry 1975), to determine the average power generation with time including the various total core feedback effects, that is 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 (Hargrove 1989). The average heat flux is next used in VIPRE (Sung, et al., 1999) (described in Section 4.4 for transient DNBR calculation).

In order to give conservative results for a start-up accident, the following assumptions are made:

1. 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 (least negative) values as a function of power are used (Table 15.0-3).

2. 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 positive value (+5 pcm/°F) is used in the analysis to yield the maximum peak heat flux.

3. The reactor is assumed to be at hot zero power. This assumption is more conservative than that of a lower initial system temperature. The higher initial system temperature yields a larger fuel to 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.

4. Reactor trip is assumed to be initiated by power range high neutron flux (low setting). The most adverse combination of instrument and set point errors, as well as delays for trip signal actuation and RCCA release, is taken into account. A 10 percent increase is assumed for the power range flux trip set point raising it from the nominal value of 25 percent to 35 percent. Since the rise in the neutron flux is so rapid, the effect of errors in the trip set point 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 RCCA is stuck in its fully withdrawn position. Section 15.0.5 further discusses RCCA insertion characteristics.

5. 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 highest combined worth at maximum speed (77 steps/minute). Control rod drive mechanism (CRDM) design is discussed in Section 4.6.

6. The most limiting axial and radial power shapes, associated with having the two highest combined worth sequential control banks in their highest worth position, is assumed in the departure from nucleate boiling (DNB) analysis.

7. The initial power level was assumed to be below the power level expected for any shutdown condition (10^{-9} of nominal power). The combination of highest reactivity insertion rate and lowest initial power produces the highest peak heat flux.

8. Two RCPs are assumed to be in operation. This lowest initial flow minimizes the resulting DNBR.

Plant systems and equipment which are available to mitigate the effects of the accident are discussed in Section 15.0.8 and listed in Table 15.0-6. No single active failure in any of these systems or equipment will adversely affect the consequences of the accident.

Results

Figures 15.4-1, 15.4-2 and 15.4-3 show the transient behavior for the uncontrolled RCCA bank withdrawal, with the accident terminated by reactor trip at 35 percent 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 neutron flux just overshoots the nominal full power value.

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 and cladding temperature. The hot spot fuel average temperature increases to a value lower than the nominal full power hot spot value. The minimum DNBR at all times remains above its limit value.

The calculated sequence of events for this accident is shown in Table 15.4-1. With the reactor tripped, Beaver Valley Power Station - Unit 2 (BVPS-2) returns to a stable condition. The plant may subsequently be cooled down further by following normal plant shutdown procedures.

15.4.1.3 Radiological Consequences

There are no radiological consequences associated with an uncontrolled RCCA bank withdrawal from a subcritical or low power start-up condition event since radioactivity is contained within the fuel rods and the RCS radionuclide concentrations remain within Technical Specification limits. Steam releases to the atmosphere are less severe than those of the loss of non-emergency ac power to the station auxiliaries event described in Section 15.2.6.

15.4.1.4 Conclusions

In the event of a RCCA withdrawal accident from the subcritical condition, the core and the RCS are not adversely affected, since the combination of thermal power and the coolant temperature result in a DNBR which is always greater than the limit value. Thus, the DNB design basis as described in Section 4.4 is met. The radiological consequences of this event are not limiting.

15.4.2 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power

15.4.2.1 Identification of Causes and Accident Description

Uncontrolled 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 set point, 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 cladding, the RPS 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 (a fault of moderate frequency) as defined in Section 15.0.1.

The automatic features of the RPS which prevent core damage following the postulated accident include the following:

1. Power range neutron flux instrumentation actuates a reactor trip if two-out-of-four channels exceed an overpower set point.

2. Reactor trip is actuated if any two-out-of-three ΔT channels exceed an overtemperature ΔT set point. This set point is automatically varied with axial power imbalance, and coolant temperature and pressure to protect against DNB.

3. Reactor trip is actuated if any two-out-of-three ΔT channels exceed an overpower ΔT set point.

4. A high pressurizer pressure reactor trip actuated from any two-out-of-three pressure channels, which is set at a fixed point. This set pressure is less than the set pressure for the pressurizer safety valves.

5. A high pressurizer water level reactor trip actuated from any two-out-of-three level channels when the reactor power is above approximately 10 percent (Permissive-7).

6. Reactor trip is actuated if any two-out-of-four channels exceed a positive neutron flux rate setpoint.

In addition to the preceding reactor trips, there are the following RCCA withdrawal blocks:

1. High neutron flux (one-out-of-four power range).

2. Overtemperature ΔT (two-out-of-three).

3. Overpower ΔT (two-out-of-three).

The manner in which the combination of overpower and overtemperature trips provide protection over the full range of RCS conditions is described in Chapter 7. 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 overtemperature ΔT trip are represented as "protection lines" on this diagram. The protection lines are drawn to include all adverse instrumentation and set point 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 DNBR can be represented as a line. The DNB lines represent the locus of conditions for which the DNBR equals the safety analysis limit value. All points below and to the left of a DNB line for a given pressure have a DNBR greater than the safety analysis limit value. The diagram shows that the DNB design basis is not violated 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 set point); high pressure (fixed set point); low pressure (fixed set point); overpower and overtemperature ΔT (variable set points).

15.4.2.2 Analysis of Effects and Consequences

Method of Analysis

This transient is analyzed by the LOFTRAN Code (Burnett, et. al. 1984). The code computes pertinent plant variables including temperatures, pressures, and power level. The core limits as illustrated on Figure 15.0-1 are used as input to LOFTRAN to determine the minimum DNBR during the transient.

This accident is analyzed with the Revised Thermal Design Procedure (Friedland and Ray, 1989). Plant characteristics and initial conditions are shown in Table 15.0-3. For an uncontrolled rod withdrawal at power accident, the following conservative assumptions are made:

1. Nominal values are assumed for the initial reactor power, pressure, and RCS temperatures (see Table 15.0-3). Uncertainties in initial conditions are included in the limit DNBR as applied by the Revised Thermal Design Procedure (Friedland and Ray, 1989).

2. Reactivity coefficients - two cases are analyzed:

    a. Minimum reactivity feedback - a positive moderator temperature coefficient of reactivity (Table 15.0-3) and a least negative Doppler only power coefficient of reactivity (Figure 15.0-2) are assumed.

    b. Maximum reactivity feedback - a conservatively large negative moderator temperature coefficient and a most negative Doppler only power coefficient are assumed.

3. The reactor trip on high neutron flux is assumed to be actuated at a conservative value of 116 percent of nominal full power. The overtemperature ΔT trip includes all adverse instrumentation and set point errors; the delays for trip actuation are assumed to be the maximum values.

4. The RCCA trip insertion characteristic is based on the assumption that the highest worth assembly is stuck in its fully withdrawn position.

5. The maximum positive reactivity insertion rate is greater than that for the simultaneous withdrawal of the combinations of the two control banks having the maximum combined worth at maximum speed.

The effect of RCCA movement on the axial core power distribution is accounted for by the axial power shape measurement as described in Chapter 7.

Beaver Valley Power Station - Unit 2 systems and equipment which are available to mitigate the effects of the accident are discussed in Section 15.0.8, and listed in Table 15.0-6. No single active failure in any of these systems or equipment will adversely affect the consequences of the accident. A discussion of anticipated transients without trip considerations is presented in Section 15.8.

Results

Figures 15.4-4, 15.4-5 and 15.4-6 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 BVPS-2, small changes in T_{avg} and pressure result, and margin to DNB is maintained.

The transient response for a slow RCCA withdrawal from full power is shown on Figures 15.4-7, 15.4-8 and 15.4-9. 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 15.4-10 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 trip channels. The minimum DNBR is never less than the limit value.

Figures 15.4-11 and 15.4-12 show the minimum DNBR as a function of reactivity insertion rate for RCCA withdrawal incidents starting at 60 percent and 10 percent power, respectively. The results are similar to the 100 percent power case, except as the initial power is decreased, the range over which the overtemperature trip is effective is increased. In either case the DNBR does not fall below the limit value.

The shape of the curves of minimum DNBR 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 the minimum feedback graph in Figure 15.4-11, for example, it is noted that:

1. For high reactivity insertion rates, reactor trip is initiated by the high neutron flux trip. The neutron flux level in the core rises rapidly for these insertion rates while core heat flux and coolant system temperature lag behind due to the thermal capacity of the fuel and coolant system fluid. Thus, the reactor is tripped prior to significant increase in heat flux or water temperature with resultant high minimum DNB ratios during the transient.

2. The overtemperature ΔT reactor trip circuit initiates a reactor trip when measured coolant loop ΔT exceeds a set point based on measured RCS average temperature and pressure. This trip circuit is described fully in Chapter 7; however, it is important in this context to note that the average temperature contribution to the circuit is lead-lag compensated in order to decrease the effect of the thermal capacity of the RCS in response to power increases.

3. For lower reactivity insertion rates, the effectiveness of the overtemperature ΔT trip increases (in terms of increased minimum DNBR) 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.

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 transients are 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 116 percent of its nominal value (that is, the high neutron flux trip point 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 116 percent of its nominal value. Taking into account the effect of the RCCA withdrawal on the axial core power distribution, the peak fuel temperature will remain below the fuel melting temperature.

Since the DNBR limit is not violated 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 in Table 15.4-1. With the reactor tripped, BVPS-2 eventually 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 feedwater system or auxiliary feedwater system (AFWS). Any action required of the operator to maintain BVPS-2 in a stabilized condition will be in a time frame in excess of 10 minutes following reactor trip.

15.4.2.3 Radiological Consequences

There are only minimal radiological consequences associated with an uncontrolled RCCA 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. Since no fuel damage is postulated to occur, the radiological consequences associated with atmospheric steam release from this event are less severe than those of the loss of nonemergency ac power to the station auxiliaries accident described in Section 15.2.6.

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, that is, the minimum value of DNBR is always larger than the limit value. Thus, the DNB design basis as described in Section 4.4 is met. The radiological consequences of this event are not limiting.

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 misoperation accidents include:

1. One or more dropped RCCAs within the same group,
2. A dropped RCCA bank,
3. Statically misaligned RCCA, or
4. Withdrawal of a single RCCA.

Each RCCA has a position indicator channel which displays the 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, moveable gripper, and lift coils of a mechanism) is required to withdraw the RCCA attached to the mechanism. Since the stationary gripper, moveable gripper, and lift coils associated with the RCCAs of a rod group are driven in parallel, any single failure which would cause rod withdrawal would effect 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 assembly events are classified as ANS Condition II incidents (incidents of moderate frequency) as defined in Section 15.0.1. The single RCCA withdrawal incident is classified as an ANS Condition III event, as discussed by the following.

No single electrical or mechanical failure in the rod control system could cause the accidental withdrawal of a single RCCA from the inserted bank at full power operation. The operator could withdraw a single RCCA in the 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 (Section 7.7) or multiple significant operator errors and subsequent and repeated operator disregard of event indication). The probability of such a combination of conditions is considered low such 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 in accordance with General Design Criterion (GDC) 25 which states, "The protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any <u>single</u> malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control <u>rods</u>." 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 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:

1. A sudden drop in the core power level as seen by the nuclear instrumentation system,

2. Asymmetric power distribution as seen on out-of-core neutron detectors or core exit thermocouples,

3. Rod at bottom signal,

4. Rod deviation alarm, or

5. Rod position indication.

Misaligned RCCAs are detected by:

1. Asymmetric power distribution as seen on out-of-core neutron detectors or core exit thermocouples,

2. Rod deviation alarm, or

3. Rod position indicators.

The resolution of the rod position indicator channel is less than ±7.5 inches. Deviation of any RCCA from its group by twice this distance (15 inches) will not cause power distributions worse than 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, the operator is required to take action as required by the Technical Specifications.

If one or more rod position indicator channels should be out of service, detailed operating instructions shall be followed to assure that alignment of the nonindicated 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 assemblies 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 which are available to mitigate the effects of the various control rod misoperations are discussed in Section 15.0.8 and listed in Table 15.0-6. No single active failure in any of these systems or equipment will adversely effect the consequences of the accident.

15.4.3.2 Analysis of Effects and Consequences

1. Dropped RCCAs, dropped RCCA bank, and statically misaligned RCCA.

<u>Method of Analysis</u>

a. One or more dropped RCCAs

For evaluation of the dropped RCCA event, the transient system response is calculated using the LOFTRAN code (Burnett, et. al. 1984).

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 Bishop, Sandberg, and Tong (1965).

b. 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 used as input to the VIPRE code to calculate the DNBR.

Results

a. One of more dropped RCCAs from the same group

Single or multiple RCCAs within the same group result in a negative reactivity insertion. The core is not adversely affected during this period, since power is decreasing rapidly.

Following plant stabilization, normal rod retrieval for shutdown procedures are followed. The operator may manually retrieve the RCCA by following approved operating procedures.

For those dropped RCCAs which do not result in a 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. Figures 15.4-13 and 15.4-14 show 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 Bishop et al (1965). In all cases, the minimum DNBR remains above the limit value.

b. Dropped RCCA Bank

A dropped RCCA bank typically results in a reactivity insertion greater than 500 pcm. The core is not adversely affected during the insertion period, since power is decreasing rapidly. The transient will proceed as described in part a; however, the return to power will be less due to the greater worth of an entire bank. Following plant stabilization, normal rod retrieval or shutdown procedures may subsequently be followed to further cool down the plant.

c. Statically Misaligned RCCA

The most severe misalignment situations with respect to DNBR at significant power levels arise from cases in which one RCCA 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, with bank D inserted to its full power insertion limit and one RCCA fully withdrawn, DNBR does not fall below the limit value. This case is analyzed assuming the initial reactor power, pressure, and RCS temperatures are at their nominal values, including uncertainties (as given in Table 15.0-3) but with the increased radial peaking factor associated with the misaligned RCCA.

Departure from nucleate boiling calculations have not been performed specifically for RCCAs 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 previously, assuming insertion limits on the other banks equivalent to a bank D full-in insertion limit.

For RCCA misalignments with one RCCA fully inserted, the DNBR does not fall below the limit value. This case is analyzed assuming the initial reactor power, pressure, and RCS temperatures are at their nominal values, including uncertainties (as given in Table 15.0-3) but with the increased radial peaking factor associated with the misaligned RCCA.

Departure from nucleate boiling does not occur for the RCCA misalignment incident and thus the ability of the reactor 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 an RCCA group misalignment condition by the operator, the operator is required to take action as required by the plant Technical Specifications and operating instructions.

2. Single RCCA Withdrawal

<u>Method of Analysis</u>

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 THINC to calculate the DNBR for the event. The case of the worst rod withdrawn from 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.

<u>Results</u>

For the single rod withdrawal event, two cases have been considered as follows:

a. 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 Section 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 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 percent.

b. 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 described in Case a.

In cases, such as those described in this section, 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 RCCA misoperation, accidental withdrawal of a single RCCA, is predicted to result in limited fuel damage. The subsequent reactor and turbine trip would result in atmospheric steam dump, assuming the condenser is not available for use. The radiological consequences from this event are less severe than those of the main steam line break (MSLB) event, analyzed in Section 15.1.5.3.

15.4.3.4 Conclusions

For cases of dropped RCCAs or dropped banks for which the reactor is tripped by the power range negative neutron flux rate trip, there is no reduction in the margin to core thermal limits, and consequently the DNB design basis is met.

For all cases of any RCCA fully inserted, or 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 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 Start Up of an Inactive Reactor Coolant Loop

15.4.4.1 Identification of Causes and Accident Description

The plant can be operated in Modes 5 or 6 with an inactive loop in either of two ways. The pump in the inactive loop can be turned off and BVPS-2 operated with the loop isolation valves in the normal fully open position. In this case, there is reverse flow through the inactive loop when a reactor coolant pump in any unisolated loop is operated. The plant can also be operated with the loop isolation valves of a loop closed in order to perform maintenance. In this case, there is no flow from the reactor vessel and active loops to the inactive loop. The plant operates much as if it were a plant without that loop. With the isolation valves closed, the boron concentration of the isolated section of the loop may deviate from the boron concentration of the active loops. The plant may isolate a loop only while the plant is shutdown. Analysis has not been conducted for power operation with a loop isolation valve closed.

Inadvertent opening of an isolated loop is prevented by (1) requiring that any loop isolation valve movement follow strict procedural criteria, and (2) loop isolation valve operators have their power removed while a loop is isolated.

Procedures require that (1) the boron concentration of the isolated loop be verified, and (2) the isolated loop be drained and refilled from the Refueling Water Storage Tank or Reactor Coolant System prior to opening the loop isolation valves, returning the loop to service. An isolated loop will be returned to service within 4 hours of the completion of the refilling to ensure that there is no unacceptable boron stratification in the isolated loop.

15.4.4.2 Assumptions and Method of Analysis

Interlocks are provided to prevent starting a RCP unless:

1. The cold leg loop stop valve in the same loop is fully closed, or

2. Both the hot leg loop stop valve and cold leg loop stop valve are fully open.

The interlocks are a part of the RPS and include the following redundancy:

a. Two independent limit switches to indicate that a valve is fully open.

b. Two independent limit switches to indicate that a valve is fully closed.

The interlocks meet the IEEE Standard 279-1971 criteria and, therefore, cannot be negated by a single failure.

Results

Procedures require that the isolated loop water boron concentration be verified prior to opening loop isolation valves. Procedures also require an isolated loop to be drained and refilled with water supplied from the Refueling Water Storage Tank or Reactor Coolant System within 4 hours prior to opening either the hot or cold leg isolation valves. This prevents several potential concerns. A potential single failure of the blender if the Chemical and Volume Control System was used to fill an isolated loop could lead to unborated primary grade water being injected. Using water from the Refueling Water Storage Tank or Reactor Coolant System ensures that the boron concentration of the isolated loop is sufficient to prevent a dilution of the boron concentration in the active reactor coolant loops which would reduce the shutdown margin to below those values used in safety analyses. Thus, when the isolated loop is returned to service, no single failure could cause an isolated loop to be filled with unborated water. Opening the loop isolation valves within 4 hours of the refill prevents any boron concentration stratification concerns.

15.4.4.3 Radiological Consequences

There are only minimal radiological consequences associated with start-up of an inactive reactor coolant loop. Since no fuel damage is postulated to occur from this transient, the radiological consequences associated with this event are less severe than the loss of nonemergency ac power to the station auxiliaries described in Section 15.2.6.

15.4.4.4 Conclusions

Procedures and interlocks prevent inadvertent opening of loop isolation valves and require that the startup of an isolated loop be performed in a controlled manner. This virtually eliminates any sudden positive reactivity addition from boron dilution. Thus the core cannot be adversely affected by the startup of an isolated loop and fuel design limits are not exceeded. The radiological consequences of this event are not limiting.

15.4.5 A Malfunction or Failure of the Flow Controller in a Boiling Water Reactor Loop that Results in an Increased Reactor Coolant Flow Rate

This section applies only to BWRs and is not applicable to BVPS-2.

15.4.6 Chemical and Volume Control System Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant

15.4.6.1 Identification of Causes and Accident Description

One of the two principal means of positive reactivity insertion to the core is the addition of unborated, primary grade water (PGW) into the RCS through the reactor makeup portion of the CVCS. Boron dilution with these systems is a manually initiated operation under strict administrative controls requiring close operator surveillance with procedures limiting the rate and duration of the dilution. A boric acid blend system is available in the CVCS to allow the operator to match the makeup’s boron concentration to that of the RCS during normal charging.

The principal means of causing an inadvertent boron dilution are the opening of the PGW makeup control valve and failure of the blend system, either by controller or mechanical failure. The reactor makeup portion of the CVCS is 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 to terminate the dilution. An inadvertent dilution from the reactor makeup portion of the CVCS may be terminated by closing the PGW makeup control valve. All expected sources of dilution may be terminated by closing the volume control tank isolation valves 2CHS*LCV115C and E. The lost shutdown margin (SDM) may be regained by opening the refueling water storage tank (RWST) isolation valves 2CHS*LCV115B and D, thus allowing the addition of borated water to the RCS.

Generally, to dilute, the operator must perform two distinct actions:

1) Switch control of the makeup from the automatic makeup mode to the dilute mode, and

2) Turn the control switch.

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 PGW. This requires the operator to determine the concentration of the addition and to set 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 PGW flow rate, to the blended flow rate; i.e., the controller determines the PGW flow rate after the start button is depressed.

The status of the RCS makeup is continuously available to the operator by:

1) Indication of the boric acid and blended flow rates,

2) CVCS and PGW pump status lights,

3) Primary grade water header low pressure alarm,

4) Deviation alarms if the boric acid or blended flow rates deviate by more than 10 percent from the preset values,

5) Source range neutron flux - when reactor is subcritical;

    a) High flux at shutdown alarm,

    b) Indicated source range neutron flux count rates, and

    c) Audible source range neutron flux count rate

6) With the reactor critical;

    a) Axial flux difference alarm (reactor power $\geq$ 50 percent),

    b) Control rod insertion limit low and low-low alarms,

    c) Overtemperature ΔT alarm (at power),

    d) Overtemperature ΔT turbine runback (at power),

    e) Overtemperature ΔT reactor trip, and

    f) Power range neutron flux - high, both high and low set-point reactor trips.

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

15.4.6.2 Analysis of Effects and Consequences

To cover all phases of BVPS-2 operation, boron dilution during the modes of refueling, cold shutdown, hot shutdown, hot standby, start-up, and power is considered in this analysis. Conservative values for necessary parameters were used, (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 response after detection of a dilution transient in progress.

<u>Dilution During Refueling, Cold Shutdown and Hot Shutdown</u>

An uncontrolled boron dilution transient cannot occur during this mode of operation. The primary means for a significant boron dilution is through the injection of unborated water into the Reactor Coolant System. Inadvertent boron dilution is prevented by administrative controls which isolate the primary grade water system isolation valves from the Chemical and Volume Control System, except during planned boron dilution or makeup activities. Thus unborated water cannot be injected into the Reactor Coolant System, making an unplanned boron dilution at these conditions highly improbable, since the source of unborated water to the charging pumps is isolated and the low head safety injection pumps cannot be aligned to the primary grade water supply. This precludes the primary means for an inadvertent boron dilution event in this mode of operation.

The primary grade water system isolation valves may be opened when directed by the control room during this mode of operation only for a planned boron dilution or makeup activity. The primary grade water system isolation valves will be verified to be locked, sealed or otherwise secured in the closed position within 15 minutes after the planned boron dilution or makeup activity is completed. During planned boron dilution events, operator attention will be focused on the boron dilution process and any inappropriate blender operation is unlikely and will be readily identified.

The operator has prompt and definite indication of any boron dilution from the audible count rate instrumentation. High count rate is alarmed in the reactor containment and the control room. In addition a high source range flux level is alarmed in the control room. The count rate increase is proportional to the subcritical multiplication factor.

<u>Dilution During Hot Standby</u>

The Technical Specifications require the reactor to be shutdown by at least 1.77 percent $\Delta k/k$ when in this operating mode. The following conditions were assumed for an inadvertent boron dilution while in this mode:

1) The maximum all rods in critical boron concentration is conservatively estimated to be 1900 ppm. The minimum change from the critical boron concentration assuming all rods in and 1.77% shutdown margin is conservatively estimated to be 182 ppm.

2) The dilution flow rate, with RCS at 2,250 psia, is the maximum calculated capacity of the pumps associated with the primary grade water sources and their flowpaths. The dilution flow rate of 231 gpm is assumed in the boron dilution event safety analysis.

3) A minimum RCS water volume of 6839 ft^3. This is a conservative estimate of the active volume of the RCS with the RCS filled and vented and one RCP operating.

Dilution During Start-up

Start-up is a transitory mode of operation. In this mode BVPS-2 is being taken from one long term mode of operation, hot standby, to another, power. BVPS-2 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, (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 require a SDM of 1.77% $\Delta k/k$. Other conditions assumed are:

1) The dilution flow rate, with RCS at 2,250 psia, is the maximum calculated capacity of the pumps associated with the primary grade water sources and their flowpaths to the suction of the charging/high head safety injection (SI) pumps. The dilution flow rate of 231 gpm is assumed in the boron dilution event safety analysis.

2) A minimum RCS water volume of 7467 cubic feet. This active volume includes the reactor vessel volume, reactor coolant loop piping volumes and the primary steam generator volume. Specifically excluded are the pressurizer and pressurizer surge line volumes.

3) The initial boron concentration is assumed to be 1800 ppm, which is a conservative maximum value for the critical concentration at the condition of hot zero power, rods to the insertion limits and no Xenon.

4) The critical boron concentration following reactor trip is assumed to be 1500 ppm, corresponding to the hot zero power, all rods inserted (minus the most reactive RCCA), no Xenon condition. The 300 ppm change from the initial condition noted above is a conservative minimum value.

Dilution During Full Power Operation

BVPS-2 may be operated at power two ways, automatic T_{avg}/rod control and under operator control. The Technical Specifications require an available trip reactivity of 1.77% $\Delta k/k$.

With BVPS-2 at power and the RCS at pressure, the maximum dilution flow rate is limited by the capacity of the pumps associated with the primary grade water sources and their flowpaths to the suction of the charging/high head safety injection (SI) pumps. Conditions assumed for this mode are:

1) The dilution flow rate, with RCS at 2,250 psia, is the maximum calculated capacity of the pumps associated with the primary grade water sources and their flowpaths when the reactor is in manual control. When in automatic control, the dilution flow rate is the maximum letdown flow. However, 231 gpm is assumed for both the manual and automatic rod control cases.

2) A minimum RCS water volume of 7467 cubic feet. This active volume includes the reactor vessel volume, reactor coolant loop piping volumes and the primary steam generator volume. Specifically excluded are the pressurizer and pressurizer surge line volumes.

3) The initial boron concentration is assumed to be 1800 ppm, which is a conservative maximum value for the critical concentration at the condition of hot zero power, rods to the insertion limits and no Xenon.

4) The critical boron concentration following reactor trip is assumed to be 1500 ppm, corresponding to the hot zero power, all rods inserted (minus the most reactive RCCA), no Xenon condition. The 300 ppm change from the initial condition noted above is a conservative minimum value.

Results

Dilution During Refueling, Cold Shutdown and Hot Shutdown

Dilution during these modes has been precluded through administrative control of valves in the possible dilution flow paths, see Section 15.4.6.2.

Dilution During Hot Standby

In the event that an inadvertent boron dilution transient occurs while in this mode, the operator will be alerted to the transient by the primary grade water header low pressure alarm, by the boric acid or blended flow rate deviation alarms, by increasing audible and indicated count rate on the source range instruments, and by the high flux at shutdown alarm. The time available for operator action during this sequence is at least 15 minutes. Thus, the operator will be able to terminate this accident prior to loss of shutdown margin.

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 BVPS-2 critical. During this mode BVPS-2 is in manual control with the operator required to maintain a very high awareness of 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. Once critical, the power escalation must be sufficiently slow to allow the operator to manually block the source range reactor trip after receiving P-6 from the intermediate range (nominally at 10^5 cps). 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.

After reactor trip with all loops in service there is at least 15 minutes for operator action prior to return to criticality. The required operator action is the opening of valves 2CHS*LCV-ll5B and D to initiate boration and the closing of valves 2CHS*LCV115C and E to terminate 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 trip setpoint resulting in a reactor trip. After reactor trip, with all loops in service, there is at least 15 minutes for operator action prior to return to criticality. The required operator action is the opening of valves 2CHS*LCVl15B and D and the closing of valves 2CHS*LCV115C and E. The boron dilution transient in this case is essentially the equivalent of an uncontrolled rod withdrawal at power. The maximum reactivity insertion rate for a boron dilution transient is conservatively estimated to be 2.8 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 overtemperature ΔT and an overtemperature ΔT turbine runback.

With the reactor in automatic rod control the pressurizer level controller will limit the dilution flow rate to the maximum letdown rate. If a dilution rate in excess of the letdown rate is present, the pressurizer level controller will throttle charging flow down to match the letdown rate. However, a dilution flow rate of 231 gpm is assumed for the automatic rod control case.

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 three alarms to the operator:

1) rod insertion limit - low level alarm,

2) rod insertion limit - low-low-level alarm if insertion continued after, and

3) axial flux difference alarm (outside of the target band).

The rod insertion limit alarms (LOW and LOW-LOW settings) alert the operator at least 15 minutes prior to criticality. This is the amount of time available for the operator to determine the cause of the dilution, isolate the reactor water makeup source, and initiate boration before the available shutdown margin is lost.

The above results demonstrate that in all modes of operation an inadvertent boron dilution is precluded, or responded to by automatic functions, 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.

15.4.6.3 Radiological Consequences

There are only minimal radiological consequences associated with a CVCS malfunction that results in a decrease in boron concentration in the reactor coolant 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. No fuel damage occurs from this transient. The radiological consequences associated with this event are less severe than those of the loss of nonemergency ac power to the station auxiliaries event described in Section 15.2.6.

15.4.6.4 Conclusions

No fuel damage occurs. The radiological consequences of this event are not limiting.

Following termination of the dilution flow, the operator can initiate reboration to recover the shutdown margin using the CVCS. During power operation or start up if the reactor has tripped, operating procedures will also call for operator action to control pressurizer level using the CVCS and to maintain steam generator level through control of the main feedwater system or AFWS.

15.4.7 Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position

15.4.7.1 Identification of Causes and Accident Description

Fuel and core loading errors can arise from the inadvertent loading of one or more fuel assemblies into proper 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 will lead to increased heat fluxes if the error results in placing fuel in core positions calling for fuel of lesser enrichment. Also included among possible core loading errors is the inadvertent loading of one or more fuel assemblies requiring burnable poison rods into a new core without burnable poison rods.

Any error in enrichment, beyond the normal manufacturing tolerances, can cause power shapes which are more peaked than those calculated with the correct enrichments. There is a 5 percent uncertainty margin included in the design value of power peaking factor assumed in the analysis of Condition I and Condition II transients. The incore system of moveable flux detectors, which is used to verify power shapes at the start of life, is capable of revealing any assembly enrichment error or loading error which causes power shapes to be peaked in excess of the design value.

To reduce the probability of core loading errors, each fuel assembly is marked with an identification number and loaded in accordance with a core loading diagram. After core loading, the identification numbers are verified for every assembly in the core.

The power distortion due to any combination of misplaced fuel assemblies would significantly raise peaking factors and would be readily observable with incore flux monitors. In addition to the flux monitors, thermocouples are located at the outlet of about one third of the fuel assemblies in the core. There is a high probability that these thermocouples would also indicate any abnormally high coolant enthalpy rise. Incore flux measurements are taken during the start-up subsequent to every refueling operation.

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

15.4.7.2 Analysis of Effects and Consequences

<u>Method of Analysis</u>

Steady state power distribution in the x-y plane of the core is calculated using computer codes as described in Table 4.1-2. A discrete representation is used wherein each individual fuel rod is described by a mesh interval. The power distributions in the x-y plane for a correctly loaded core assembly are also given in Chapter 4 based on enrichments given in that chapter.

For each core loading error analyzed, the percent deviation (from assembly average power) between the predicted detector readings for a normally loaded core and the perturbed core loadings (Cases A, B, C, and D that follow are for all incore detector locations.

Results

The following core loading error cases have been analyzed:

Case A

Case in which a Region 1 assembly is interchanged with a Region 3 assembly. The particular case considered was the interchange to two adjacent assemblies near the periphery of the core (Figure 15.4-21).

Case B

Case in which a Region 1 assembly is interchanged with a neighboring Region 2 fuel assembly. Two analyses have been performed for this case (Figure 15.4-22 and Figure 15.4-23).

In Case B-1, the interchange is assumed to take place with the burnable poison rods transferred with the Region 2 assembly mistakenly loaded into Region 1. In Case B-2, the interchange is assumed to take place closer to core center and with burnable poison rods located in the correct Region 2 position but with a Region 1 assembly mistakenly loaded in the Region 2 position.

Case C

Enrichment error: Case in which a Region 2 fuel assembly is loaded in the core central position (Figure 15.4-24).

Case D

Case in which a Region 2 fuel assembly instead of a Region 1 assembly is loaded near the core periphery (Figure 15.4-25).

15.4.7.3 Radiological Consequences

There are no radiological consequences associated with inadvertent loading and operation of a fuel assembly in an improper position incident since activity is contained within the fuel rods and RCS within design limits.

15.4.7.4 Conclusions

Fuel assembly enrichment errors would be prevented by administrative procedures implemented in fabrication.

In the event that a single pin or pellet has a higher enrichment than the nominal value, the consequences in terms of reduced DNBR and increased fuel and clad temperatures will be limited to the incorrectly loaded pin or pins and perhaps the immediately adjacent pins.

Fuel assembly loading errors are prevented by administrative procedures implemented during core loading. In the unlikely event that a loading error occurs, analyses in this section confirm that resulting power distribution effects will either be readily detected by the incore moveable detector system or will cause a sufficiently small perturbation to be acceptable within the uncertainties allowed between nominal and design power shapes.

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 an 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 in the Westinghouse pressurized water reactors are intended to preclude the possibility of a 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 RCCAs inserted at high power levels.

<u>Mechanical Design</u>

The mechanical design is discussed in Section 4.6. Mechanical design and quality control procedures intended to preclude the possibility of an RCCA drive mechanism housing failure are listed as follows:

1. Each full length CRDM housing is completely assembled and shop-tested at 4,100 psi.

2. The mechanism housings are individually hydrotested after they are attached to the head adapters in the reactor vessel head, and checked during the hydrotest of the completed RCS.

3. 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.

4. The latch mechanism housing and rod travel housing are each a single length of forged Type 304 stainless steel or equivalent. This material exhibits excellent notch toughness at all temperatures which 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 rod welds.

Nuclear Design

Even if a rupture of an RCCA drive mechanism housing is postulated, the operation 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 an RCCA ejection accident. Therefore, should an 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 required borations at the low level alarm and emergency boration at the low-low alarm.

Reactor Protection

The reactor protection in the event of a rod ejection accident has been described by Burnett (1969). The protection for this accident is provided by high neutron flux trip (high and low setting) and high rate of neutron flux increase trip. These protection functions are described fully in Section 7.2.

Effects on Adjacent Housings

Disregarding the remote possibility of the occurrence of a complete 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. The full length CRDM is described in Section 3.9.4.

Effects of Rod Travel Housing Longitudinal Failures

If a longitudinal failure of the rod travel housing should occur, the region of the position indicator assembly opposite the break would be stressed by the reactor coolant pressure of 2,250 psia. The most probable leakage path would be provided by the radial deformation of the position indicator coil assembly, resulting in the growth of axial flow passages between the rod travel housing and the hollow tube along which the coil assemblies are mounted.

If failure of the position indicator coil assembly should occur, the resulting free radial jet from the failed housing could cause it to bend and contact adjacent rod housings. If the adjacent housings were on the periphery, they might bend outward from their bases. The housing material is quite ductile and plastic hinging without cracking would be expected. Housings adjacent to a failed housing, in locations other than the periphery, would not be bent because of the rigidity of multiple adjacent housings.

Effect of Rod Travel Housing Circumferential Failures

If circumferential failure of a rod travel housing should occur, the broken-off section of the housing would be ejected vertically because the driving force is vertical and the position indicator coil assembly and the drive shaft would tend to guide the broken-off piece upwards during its travel. Travel is limited by the missile shield, thereby limiting the projectile acceleration. When the projectile reached the missile shield it would partially penetrate the shield and dissipate its kinetic energy. The water jet from the break would continue to push the broken-off piece against the missile shield.

If the broken-off piece of the rod travel housing were short enough to clear the break when fully ejected, it would rebound after impact with the missile shield. The top end plates of the position indicator coil assemblies would prevent the broken piece from directly hitting the rod travel housing of a second drive mechanism. Even if a direct hit by the rebounding piece were to occur, the low kinetic energy of the rebounding projectile would not be expected to cause significant damage.

Possible Consequences

From the previous discussion, the probability of damage to an adjacent housing must be considered remote. 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.

Summary

The preceding considerations given lead to the conclusion that failure of a control rod housing, due either to longitudinal or circumferential cracking, would not cause damage to adjacent housings that would increase severity of the initial accident.

15.4.8.1.2 Limiting Criteria

This event is classified as an ANS Condition IV incident (Section 15.0.1). Due to the extremely low probability of an RCCA ejection accident, some fuel damage could be considered an acceptable consequence.

Comprehensive studies of the threshold of fuel failure, and of the threshold of 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 (Taxelius 1970). Extensive tests of UO_2 zirconium-clad fuel rods representative of 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 (Liimataninen and Testa 1966) results, which indicated a failure threshold of 280 cal/gm. Limited results have indicated that this threshold decreases by about 10 percent 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) even for irradiated rods, did not occur below 300 cal/gm.

In view of the preceding 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:

1. Average fuel pallet enthalpy at the hot spot below 200 cal/gm for irradiated fuel.

2. Peak reactor coolant pressure less than that which could cause stresses to exceed the faulted condition stress limits, and

3. Fuel melting will be limited to less than 10 percent of the fuel volume at the hot spot even if the average fuel pellet enthalpy is below the limits of criterion 1 listed previously.

15.4.8.2 Analysis of Effects and Consequences

Method of Analysis

The calculation of the RCCA ejection transient is performed in two stages: 1) an average core channel calculation; and 2) 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, that is, 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 is provided by Risher (1975).

Average Core Analysis

The spatial kinetics computer code, TWINKLE (Risher and Barry 1975) 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 2,000 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 of calculating the rod worth and hot channel factor. TWINKLE is further discussed in Section 15.0.11.

Hot Spot Analysis

In the hot spot analysis, the initial heat flux is equal to the nominal heat flux times the design hot channel factor (F_Q). 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 spot 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 (Hargrove 1989). 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 et al (1965) correlation to determine the film boiling coefficient after DNB. The Bishop et al correlation is conservatively used assuming zero bulk fluid quality. The DNBR 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 Section 15.0.11.

System Overpressure Analysis

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 calculation is conducted to determine the volume surge. Finally, the volume surge is simulated in LOFTRAN (Burnett 1984). This code calculates the pressure transient taking into account fluid transport in the RCS 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 Calculation of Basic Parameters

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 as follows. Table 15.4-2 presents the parameters used in this analysis.

Ejected Rod Worths and Hot Channel Factors

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. The computer codes as described in Table 4.1-2 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 to provide worst case results.

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" are presented by Risher (1975). Experience has shown 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 (Risher 1975).

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 BVPS-2. As discussed previously, no weighting factor is applied to these results.

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 defect used is given in Section 15.0.4. The Doppler weighting factor will increase under accident conditions.

Delayed Neutron Fraction, βeff

Calculations of the effective delayed neutron fraction (β_{eff}) typically yield values no less than 0.70 percent at beginning-of-life and 0.50 percent at end-of-life for the first cycle. The accident is sensitive to a β_{eff} if the ejected rod worth is equal to or greater than a β_{eff} as in zero power transients. In order to allow for future cycles, pessimistic estimates of β_{eff} of 0.55 percent at beginning of cycle and 0.44 percent at end of cycle were used in the analysis.

Trip Reactivity Insertion

The trip reactivity insertion assumed is given in Table 15.4-2 and includes the effect of one stuck RCCA adjacent to the ejected rod. 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 second after the high neutron flux trip point was reached. This delay is assumed to consist of 0.2 second for the instrument channel to produce a signal, 0.15 second for the trip breaker to open, and 0.15 second 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 conservatism is important for hot full power accidents.

The minimum design shutdown margin available for this plant at hot zero power (HZP) may be reached only at end-of-life in the equilibrium cycle. This value includes an allowance for the worst stuck rod, an 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 margin by about an additional 1.0 percent $\Delta k/k$. 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 emergency core cooling system (ECCS) is actuated on low pressurizer pressure or level within one minute after the break. The RCS pressure continues to drop and reaches saturation (approximately 1,200 psi depending on the system temperature) in about eight minutes. Due to the large thermal inertia of primary and secondary system, there has been no significant decrease in the RCS temperature below no-load by this time, and the depressurization itself has caused an increase in shutdown margin by about 0.2 percent $\Delta k/k$ due to the pressure coefficient. Adequate shutdown margin for cooldown is available for more than 10 minutes after the break. The addition of borated water by safety injection flow starting 1 minute after the break is much more than sufficient to ensure that the core remains subcritical during the cooldown.

<u>Reactor Protection</u>

As discussed in Section 15.4.8.1.1, reactor protection for a rod ejection is provided by high neutron flux trip (high and low setting) and high rate of neutron flux increase trip. These protection functions are part of the reactor trip system (RTS). No single failure of the RTS 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 0.20 percent $\Delta\rho$ and an F_Q of 7.10, respectively. The peak fuel enthalpy was 326.8 Btu/lb (181.6 cal/gm). The peak hot spot fuel center temperature reached melting at 4,900°F. However, melting was restricted to less than 10 percent 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 0.7 percent $\Delta\rho$ and a hot channel factor, F_Q of 10.0. The peak fuel enthalpy of 186.1 Btu/lb (103.4 cal/gm). The peak fuel centerline temperature was 3,037°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 0.21 percent $\Delta\rho$ and F_Q of 7.6, respectively. This resulted in a peak fuel enthalpy of 314.5 Btu/lb (174.7 cal/gm). The peak hot spot fuel center temperature reached melting at 4800°F. However, melting was restricted to less than 10 percent 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 with banks B and C at their insertion limits. The results were 0.98 percent $\Delta\rho$ and an F_Q of 25.0, respectively. The peak clad average temperature reached 2,995°F. The peak fuel enthalpy reached 305.7 Btu/lb (169.8 cal/gm) and the peak fuel center temperature reached 4,441°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 previously is given in Table 15.4-2. The nuclear power and hot spot fuel and clad temperature transient for the worst case (beginning-of-life, full power) and also for the end-of-life, zero power case are presented on Figures 15.4-26, 15.4-27, 15.4-28 and 15.4-29.

The calculated sequence of events for the rod ejection accidents, as shown on Figures 15.4-26, 15.4-27, 15.4-28 and 15.4-29, are 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 Section 15.4.8.2.1, the reactor will remain subcritical following reactor trip.

The ejection of an RCCA constitutes a break in the RCS, located in the reactor pressure vessel head. The effects and consequences of loss-of-coolant accidents (LOCAs) are discussed in Section 15.6.5. Following the RCCA ejection, the operator would follow the same emergency instructions as for any other LOCA to recover from the event.

<u>Fission Product Release</u>

It is assumed that fission products are released from the gaps of all rods entering DNB. In all cases considered, less than 10 percent of the rods entered DNB based on a detailed three dimensional THINC analysis (Risher 1975). Although limited fuel melting at the hot spot was predicted for the beginning of cycle full power case, it is highly unlikely that melting will occur since the analysis conservatively assumed that the hot spots before and after ejection were coincident.

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 (Risher 1975). Since the severity of the present analysis does not exceed the "worst case" analysis, the accident for BVPS-2 will not result in an excessive pressure rise or further damage to the RCS.

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 by 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

The site boundary and control room doses due to airborne activity releases following a control rod ejection accident (CREA) are calculated in accordance with the methods described in Regulatory Guide 1.183. The analysis takes into account containment conversion and operation at atmospheric pressure and no longer credits termination of containment leakage at 60 minutes after accident initiation. The analysis is performed at a core power level of 2918 MWt and with the Alternative Source Term (AST) methodology. S&W computer code PERC2 is utilized in the analysis. The dose calculation model is described in Appendix 15A and is consistent with the regulatory guidance.

The worst 2-hour period dose at the Exclusion Area Boundary (EAB), the dose at the Low Population Zone (LPZ) for the duration of the release, and the 0 to 30-day dose to an operator in the control room due to inhalation and submersion are calculated based on postulated airborne radioactivity releases. The environmental releases for a postulated CREA are combined with the atmospheric dispersion values presented in Tables 15.0-11, 15.0-14 and 15.0-15 to determine the site boundary and control room doses given in Tables 15.0-12 and 15.0-13, respectively. Table 15.4-3 lists the key assumptions/parameters utilized to develop the radiological consequences following the control rod ejection accident. The analysis is intended to cover a CREA in either unit of BVPS, so the bounding parameters are listed in the Table. The critical control room parameters utilized in this model are summarized in Table 6.4-1a. Section 15.6.5.4 discusses the control room design as related to dose consequences under a sub-section titled "Control Room Habitability."

In accordance with guidance provided in RG 1.183, two independent release paths to the environment are analyzed:

Scenario 1: The failed/melted fuel resulting from a postulated control rod ejection is released into the RCS, which is released in its entirety into the containment via the ruptured control rod drive mechanism housing, is mixed in the free volume of the containment, and then released at containment technical specification leak rate. Environmental releases are assumed to occur via the containment wall.

Scenario 2: The failed/melted fuel resulting from a postulated control rod ejection is released into the RCS which is then transmitted to the secondary side via steam generator tube leakage. The condenser is assumed to be unavailable due to a loss of offsite power. Environmental releases occur from the steam generators via the main steam safety valves and atmospheric dump valves.

The rod ejection accident produces an adverse core power distribution which results in localized fuel rod damage. The assumed damage includes the breach of the fuel rod, releasing a fraction of the core gap activity, plus melting of a fraction of the core fuel pins which reach or exceed the initial temperature of fuel melting. The quantity of nuclides released from the fuel is based on the assumption that 10 percent of the fuel rods experience clad damage and 0.25 percent of fuel rods experience melting. To account for differences in power level across the core, a design radial peaking factor of 1.75 was applied in determining the inventory of the damaged rods. The equilibrium fuel cycle core inventory at a power level of 2918 MWt, calculated by ORIGENS computer code and listed in Table 15.0-7a, is used to calculate the offsite and control room doses.

In accordance with Regulatory Guide 1.183, the gap activity is assumed to be composed of 10% of the core noble gas and 10% of the core halogens associated with the percentage of fuel that has clad damage. Depending on the release pathway, the composition of the melted fuel is varied. For the containment leakage pathway, the melted fuel activity released is assumed to be composed of 100% of the core noble gas and 25% of the core halogens associated with the percentage of fuel that has melted. For the Secondary System Release pathway the melted fuel activity released is composed of 100% of the core noble gas and 50% of the core halogens associated with the percentage of fuel that has melted.

The chemical composition of the iodine in the gap/melted fuel is assumed to be 95% CsI, 4.85% elemental and 0.15% organic. However, because the sump pH is not controlled following a control rod ejection, it is conservatively assumed that the iodine released via the containment leakage pathway has the same composition as the iodine released via the secondary system release pathway; i.e., it is assumed that for both scenarios, 97% of all halogens available for release to the environment are elemental, while the remaining 3% is organic.

Scenario 1: Transport from the Containment

The failed/melted fuel activity released due to a rod ejection into the RCS is assumed to be instantaneously released into the containment where it mixes homogeneously in the containment free volume. The containment is assumed to leak at the technical specification leak rate of 0.001 day^{-1} for the first 24 hours and at half that value for the remaining 29 days after the event. Except for decay, no credit is taken for depleting the halogen (or noble gas) concentrations airborne in the containment. No credit is taken for processing the containment leakage via the safety related ventilation exhaust and filtration system that services the areas contiguous to containment; i.e., the Supplementary Leak Collection System (SLCRS) filters. To ensure bounding values, the atmospheric dispersion factors utilized for the containment release path reflect the worst value between the containment wall release point and the SLCRS release point for each time period.

Scenario 2: Transport from the Secondary System

The failed or melted fuel activity released due to a rod ejection into the RCS is assumed to be instantaneously and homogeneously mixed in the reactor coolant system and transmitted to the secondary side via primary to secondary steam generator (SG) tube leakage assumed to be at the technical specification value of 150 gpd (@STP) from each steam generator (450 gpd total). The primary to secondary leakage terminates at 2500 seconds after the event when primary pressure is below secondary pressure. At BVPS, the SG tubes remain covered for the duration of the event; therefore, per Regulatory Guide 1.183, the gap/fuel iodines have a partition coefficient of 100 in the SG. The gap noble gases are released freely to the environment without retention in the SG.

The condenser is assumed unavailable due to a coincident loss of offsite power. Consequently, the radioactivity release resulting from a control rod ejection is discharged to the environment from the steam generators via the main steam safety valves and atmospheric dump valves. The releases continue until shutdown cooling is initiated via operation of the RHR system (8 hrs after the accident) and environmental releases are terminated.

Per the regulatory requirement, the 2-hour EAB dose must reflect the "worst case" 2-hour activity release period following the rod ejection event. The worst 2-hr EAB dose will occur during the initial 2 hour period because the primary to secondary leakage stops at 0.6944 hours and the steam release rate is also the highest during this period.

The activity associated with the release of secondary steam/liquid, and primary to secondary leakage at normal technical specification levels is insignificant compared to the failed fuel contribution, and was not quantified in this assessment.

The radiological consequences of the postulated rod ejection accident are presented in Tables 15.0-12 and 15.0-13.

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 stress 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 fission product release, as a result of the number of fuel rods entering DNB, is limited to less than 10 percent of the fuel rods in the core.

The calculated dose values for the rod ejection accident are well within the 10 CFR 50.67 guidelines of 6.3 Rem TEDE.

15.4.9 Spectrum of Rod Drop Accidents in a Boiling Water Reactor

This section applies only to BWRs, and is not applicable to BVPS-2.

15.4.10 References for Section 15.4

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Tables for Section 15.4

TABLE 15.4-1

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH CAUSE REACTIVITY AND POWER DISTRIBUTION ANOMALIES

Accident	Event	Time (sec)
Uncontrolled RCCA bank withdrawal from a sub-critical or low power start-up condition	Initiation of uncontrolled rod withdrawal from 10^{-9} of nominal power	0.0
	Power range high neutron flux low set point reached	10.4
	Peak nuclear power occurs	10.6
	Rods begin to fall into core	10.9
	Minimum DNBR occurs	12.6
	Peak heat flux occurs	12.6
	Peak average clad temperature occurs	13.1
	Peak average fuel temperature occurs	13.3

Accident	Event	Time (sec)
Uncontrolled RCCA Bank Withdrawal at Power		
1. Case A	Initiation of uncontrolled RCCA withdrawal at full power and a high reactivity insertion rate (80 pcm/sec)	0.0
	Power range high neutron flux high trip point reached	1.44
	Rods begin to fall into core	1.94

TABLE 15.4-1 (Cont)

Accident	Event	Time (sec)
	Minimum DNBR occurs	2.90
2. Case B	Initiation of uncontrolled RCCA withdrawal at a small reactivity insertion rate (2 pcm/sec)	0
	Overtemperature ΔT trip point reached	66.43
	Rods beginning to fall into core	68.43
	Minimum DNBR occurs	69.1
Uncontrolled Boron Dilution		
1. Dilution during start-up	Power range-low set point reactor trip due to dilution	0.0
	Shutdown margin lost (if dilution continues after trip)	> 900

TABLE 15.4-1 (Cont)

Accident	Event	Time (sec)
2. Dilution during full power operation		
a. Automatic reactor Control	Operator receives low-low rod insertion limitation due to dilution	0.0
	Shutdown margin lost (if dilution continues after trip)	> 900
b. Manual reactor control	Overtemperature ΔT reactor trip due to dilution	120
	Shutdown margin is lost (if dilution continues after trip)	> 1,020

TABLE 15.4-1 (Cont)

Accident	Event	Time (sec)
Rod Cluster Control Assembly Ejection		
1. Beginning-of-Life, Full Power	Initiation of rod ejection	0.0
	Power range high neutron flux set point reached	0.06
	Peak nuclear power occurs	0.13
	Rods begin to fall into core	0.56
	Peak fuel average temperature occurs	2.39
	Peak clad average temperature occurs	2.46
	Peak heat flux occurs	2.47
2. End-of-Life, Zero Power	Initiation of rod ejection	0.0
	Power range high neutron flux low set point reached	0.17
	Peak nuclear power occurs	0.20
	Rods begin to fall into core	0.67
	Peak heat flux occurs	1.39
	Peak clad average temperature occurs	1.39
	Peak fuel average temperature occurs	1.82

TABLE 15.4-2

PARAMETERS USED IN THE ANALYSIS OF THE ROD CLUSTER CONTROL ASSEMBLY EJECTION ACCIDENT

	Time in Life			
	BOL-HFP Begin	BOL-HZP Begin	EOL-HFP End	EOL-HZP End
Power level (%)	102	0	102	0
Ejected rod worth (% Δk)	0.20	0.70	0.21	0.98
Delayed neutron fraction (%)	0.55	0.55	0.47	0.47
Feedback reactivity weighting	1.500	1.866	1.567	3.620
Doppler-only power defect, pcm	-962	-962	-941	-941
Trip reactivity (% Δk)	4.0	2.0	4.0	2.0
F_Q before rod ejection	2.52		2.52	
F_Q after rod ejection	7.11	10.0	7.6	25.0
Number of operational pumps	3	2	3	2
Maximum fuel pellet average temperature (°F)	4,136	2,568	4,008	3,914
Maximum fuel center temperature (°F)	4,969	3,037	4,869	4,441
Maximum fuel stored energy (cal/gm)	181.6	103.4	174.7	169.8
Fuel melt (%)	<10	0	<10	0

TABLE 15.4-3

PARAMETERS USED IN RADIOLOGICAL ANALYSIS OF THE ROD CONTROL CLUSTER EJECTION ACCIDENT

Containment Pathway Parameters	
Power Level	2918 MWth
Minimum Free Volume	1.75E+6 ft^3
Containment Leakrate (0 –24 hr)	0.1% vol fractions per day
Containment Leakrate (1-30 day)	0.05% vol fractions per day
Failed Fuel Percentage	10%
Percentage of Core Inventory in Fuel Gap	10% (noble gases & halogens)
Melted Fuel Percentage	0.25%
Percentage of Core Inventory in melted fuel released to Containment Atmosphere	100% Noble Gas; 25% Halogens
Chemical Form of Iodine in Failed/Melted fuel	4.85% elemental; 95% CsI 0.15% organic
Radial Peaking Factor	1.75
Core Activity Release Timing	PUFF
Form of Failed/Melted Iodine in the Containment Atmosphere	97% elemental; 3% organic
Equilibrium Core Activity	Table 15.0-7a
Termination of Containment Release	30 days
Environmental Release Point	Containment wall / SLCRS Vent (Containment Top)
Control Room χ/Q Values	Limiting values of Tables 15-0-14 and 15.0-15
Secondary Side Pathway Parameters	
Minimum Reactor Coolant Mass	340,711 lbm
Primary-to-Secondary Leakrate	150 gpd per SG @ STP, 450 gpd total
Termination of Primary-to-Secondary Leakage	2500 secs
Fraction of Failed/Melted Fuel	Same as Containment Pathway

TABLE 15.4-3 (Cont)

PARAMETERS USED IN RADIOLOGICAL ANALYSIS OF THE ROD CONTROL CLUSTER EJECTION ACCIDENT

Percentage of Core Inventory in melted fuel released to Reactor Coolant		100% Noble Gas; 50% Halogens
Iodine Species released to Environment		97% elemental; 3% organic
Iodine Partition Coefficient		100 (all tubes submerged)
Fraction of Noble Gas Released		1.0 (Released to Environ without holdup)
Minimum Post-Accident SG Liquid Mass		99,217 lbm per SG
Steam Releases per SG	0-150 secs:	900 lbs/sec
	150-300 secs	300 lbs/sec
	300-2500 secs	150 lbs/sec
	2500 secs-8 hrs	776,000 lbs
Termination of Release from SGs		8 hours
Environmental Release Point		MSSVs/ADVs
Control Room χ/Q Values		Limiting values of Tables 15-0-14 and 15.0-15
CR Emergency Ventilation:	Initiation	Signal/Timing
Initiation time		30 minutes by manual operation

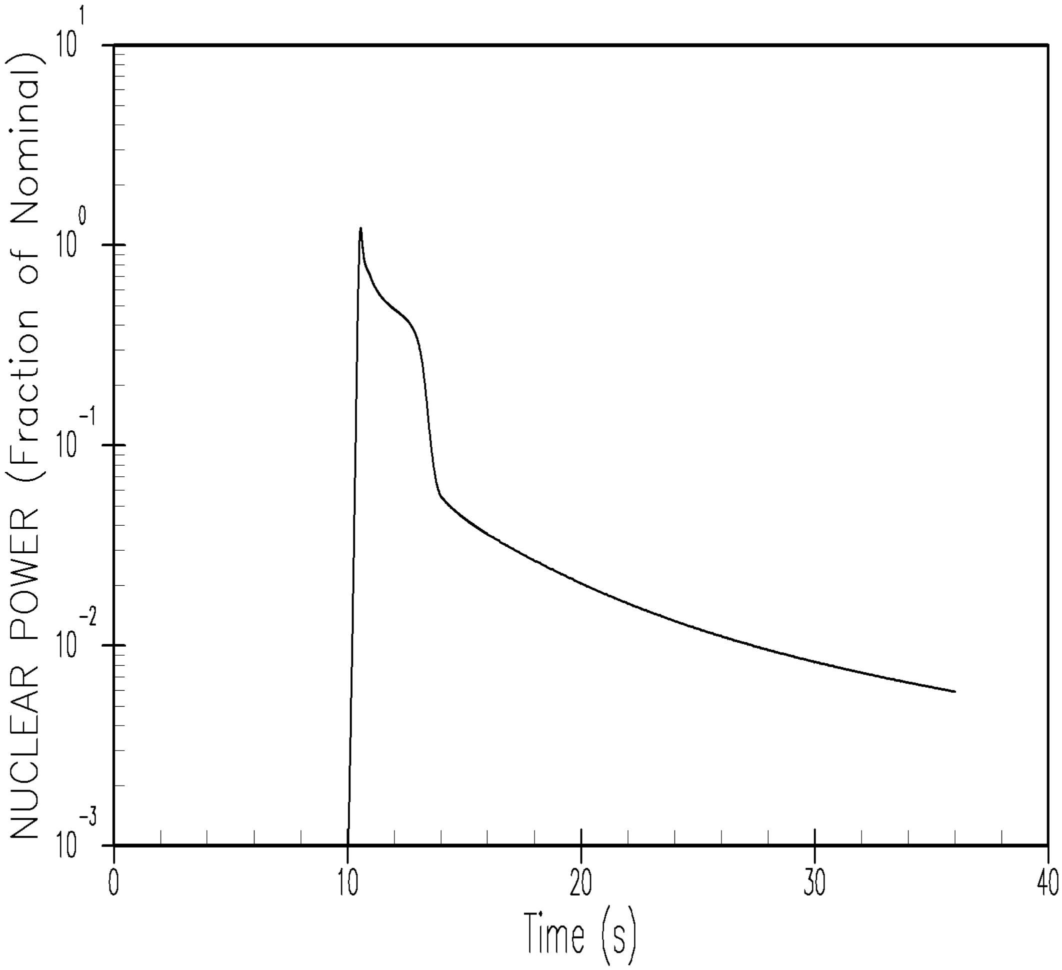


Figure 15.4-1

Nuclear Power Transient for Uncontrolled Rod Withdrawal From a Subcritical Condition

Beaver Valley Power Station Unit No. 2
Updated Final Safety Analysis Report

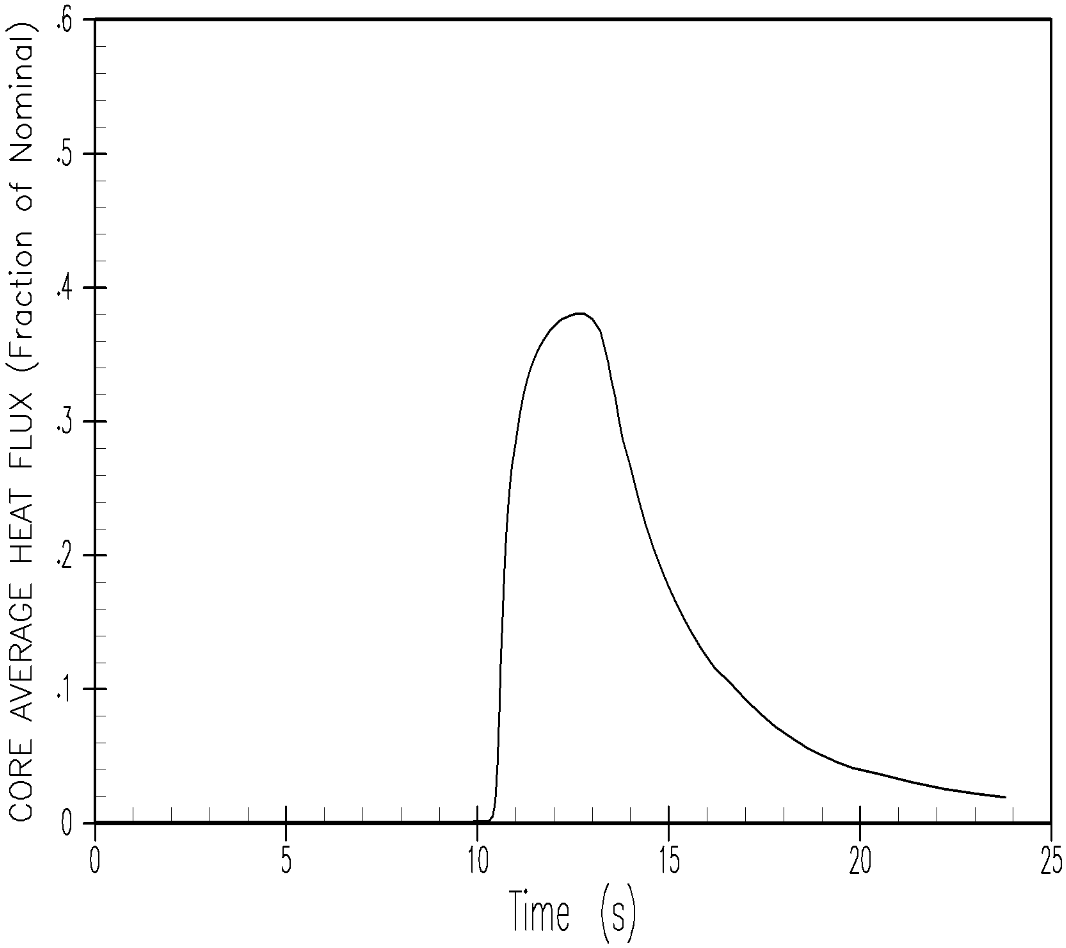


Figure 15.4-2

Thermal Flux Transient for Uncontrolled Rod Withdrawal From a Subcritical Condition

Beaver Valley Power Station Unit No. 2 Updated Final Safety Analysis Report

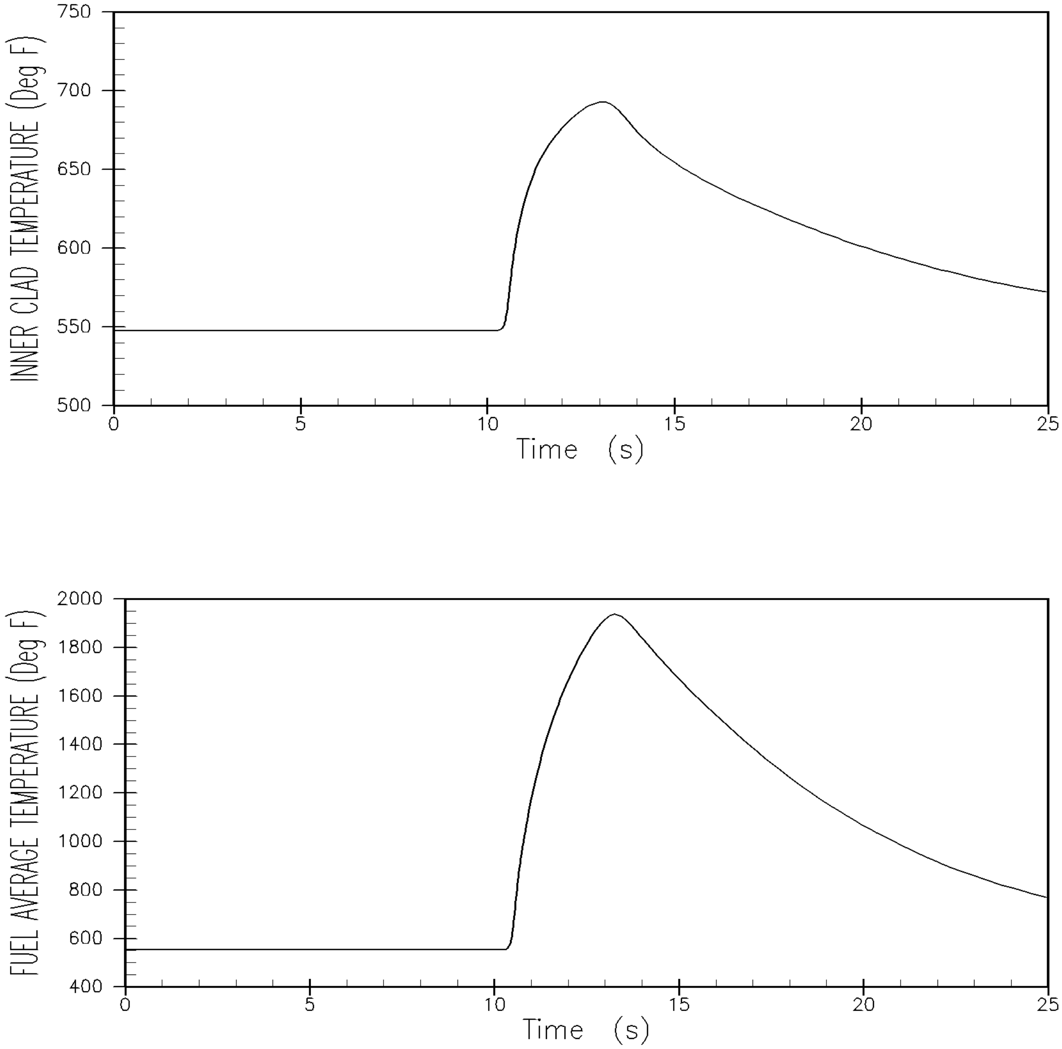


Figure 15.4-3

Fuel and Clad Temperature Transients for Uncontrolled Rod Withdrawal From a Subcritical Condition

Beaver Valley Power Station Unit No. 2 Updated Final Safety Analysis Report

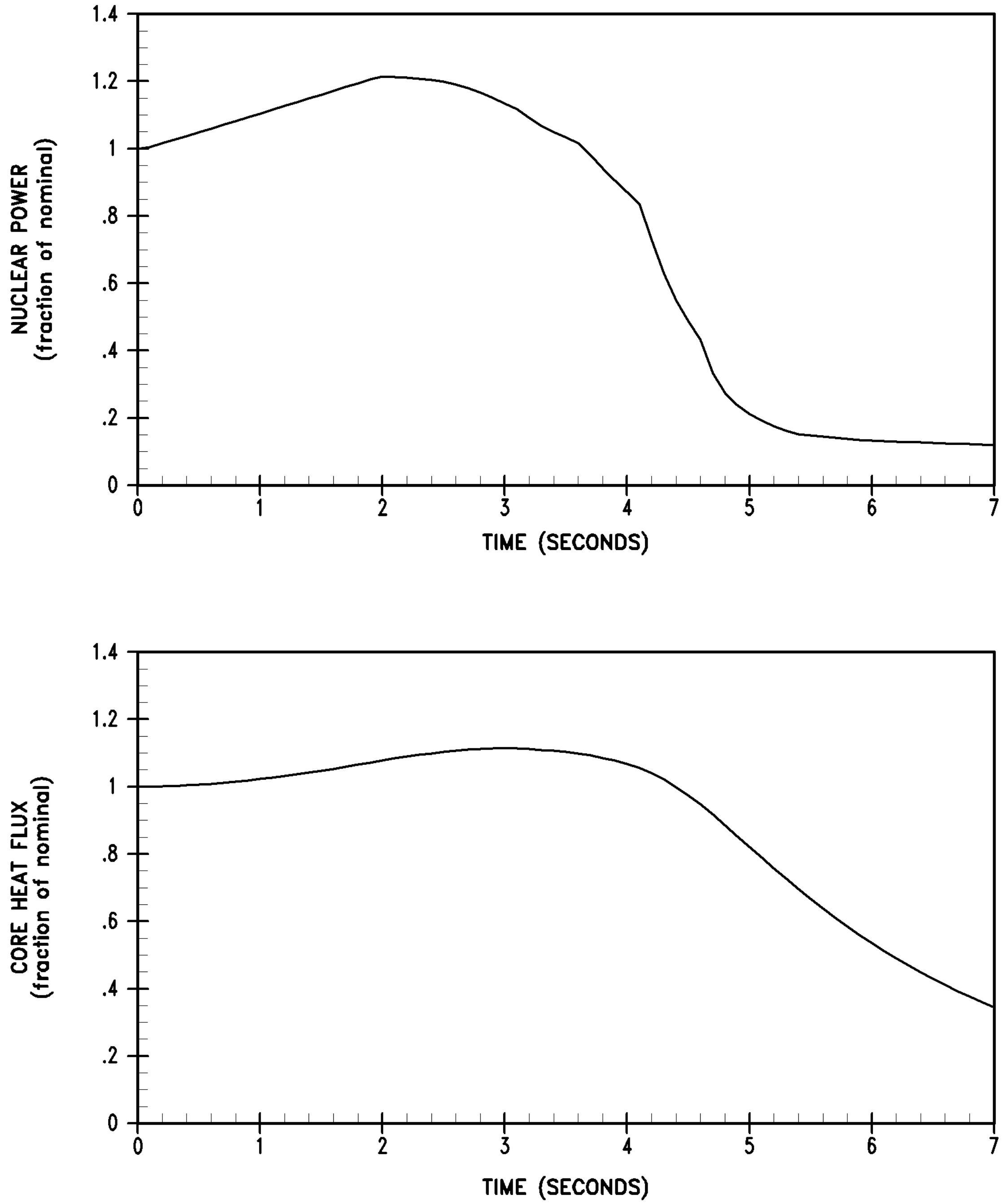
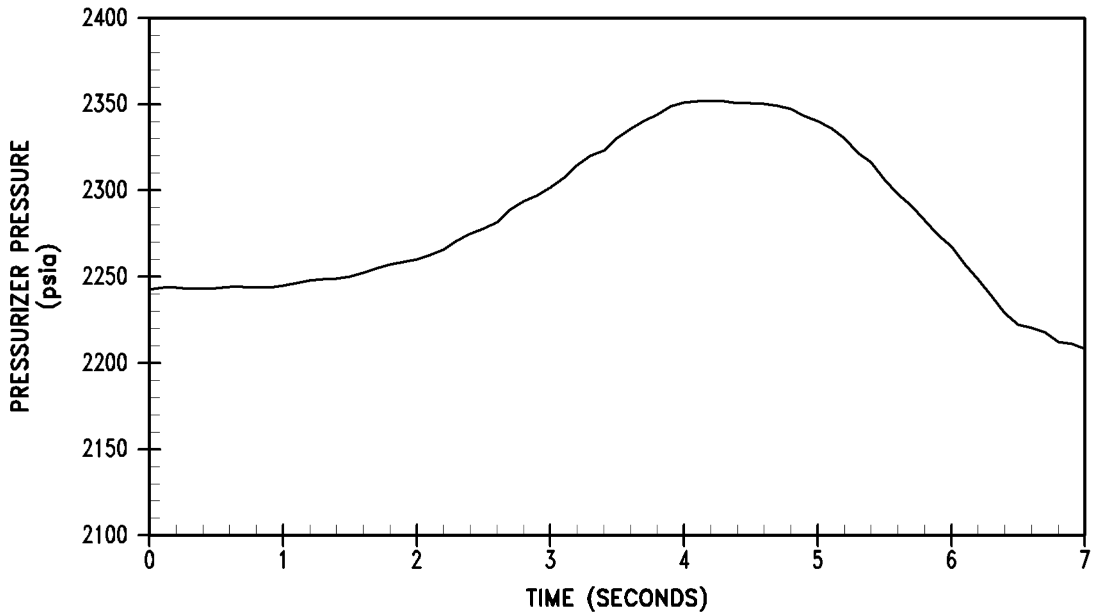


Figure 15.4-4

Nuclear Power and Core Heat Flux Transients for Uncontrolled Rod Withdrawal From Full Power with Minimum Feedback and 80 pcm/sec Withdrawal Rate

Beaver Valley Power Station Unit No. 2
Updated Final Safety Analysis Report



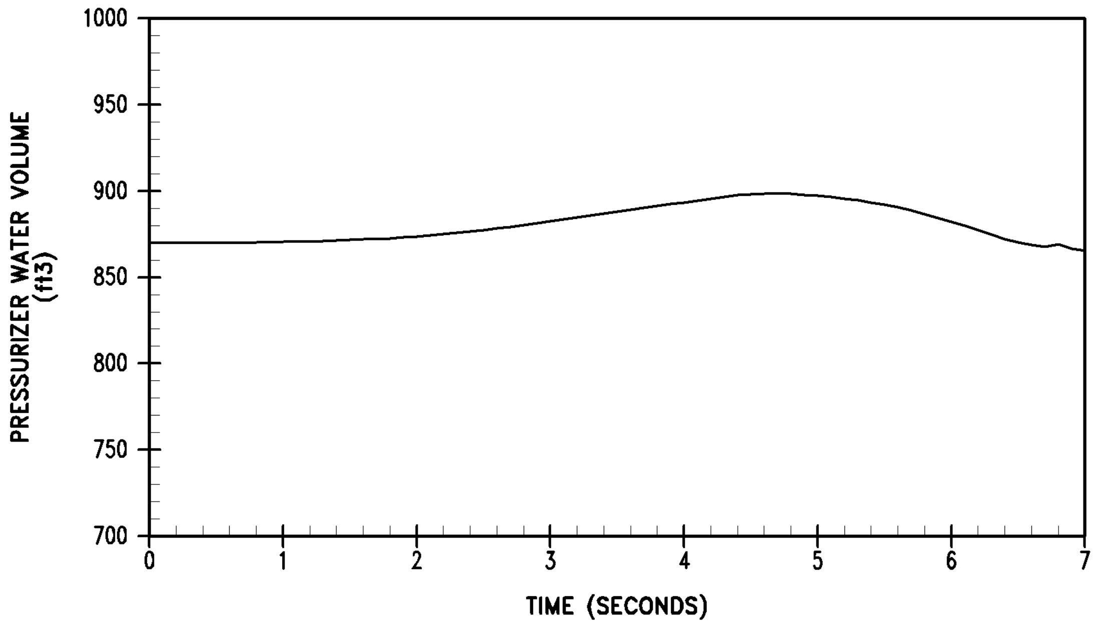


Figure 15.4-5

Pressurizer Pressure and Water Volume Transients for Uncontrolled Rod Withdrawal From Full Power with Minimum Feedback and 80 pcm/sec Withdrawal Rate

Beaver Valley Power Station Unit No. 2
Updated Final Safety Analysis Report

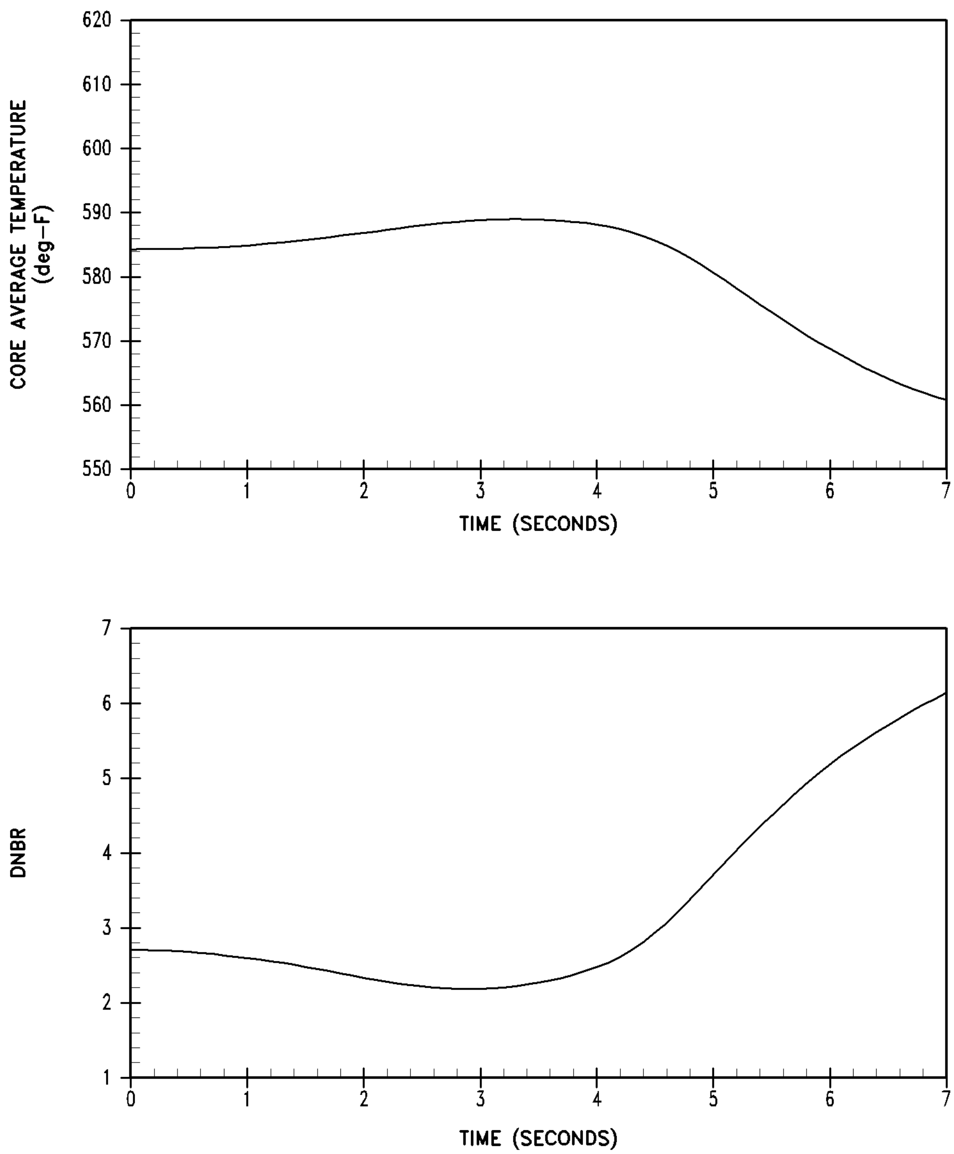


Figure 15.4-6

Core Average Temperature Transient and DNBR vs.Time for Uncontrolled Rod Withdrawal From Full Power with Minimum Feedback and 80 pcm/sec Withdrawal Rate

Beaver Valley Power Station Unit No. 2
Updated Final Safety Analysis Report

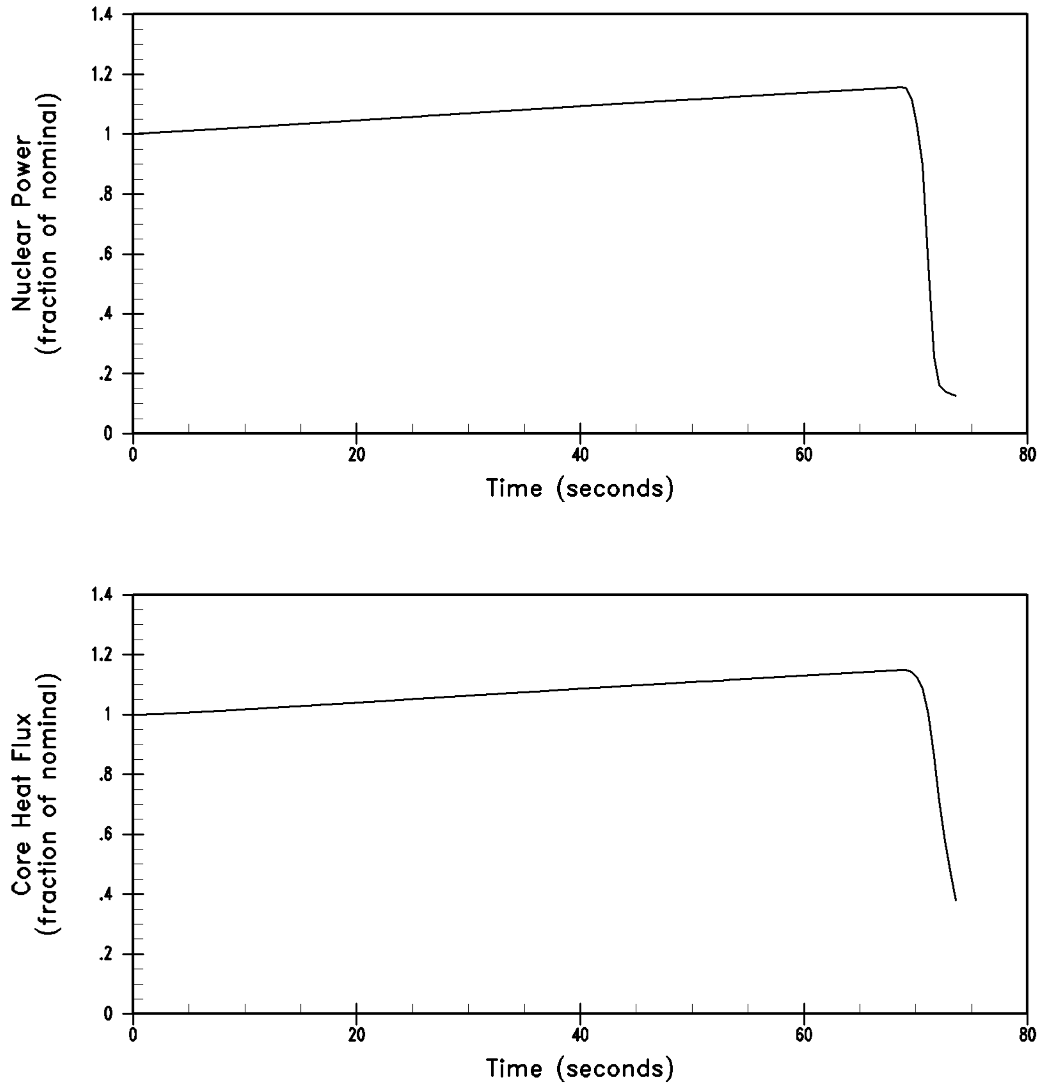


Figure 15.4-7

Nuclear Power and Core Heat Flux Transients for Uncontrolled Rod Withdrawal From Full Power with Minimum Feedback and 2 pcm/sec Withdrawal Rate

Beaver Valley Power Station Unit No. 2
Updated Final Safety Analysis Report

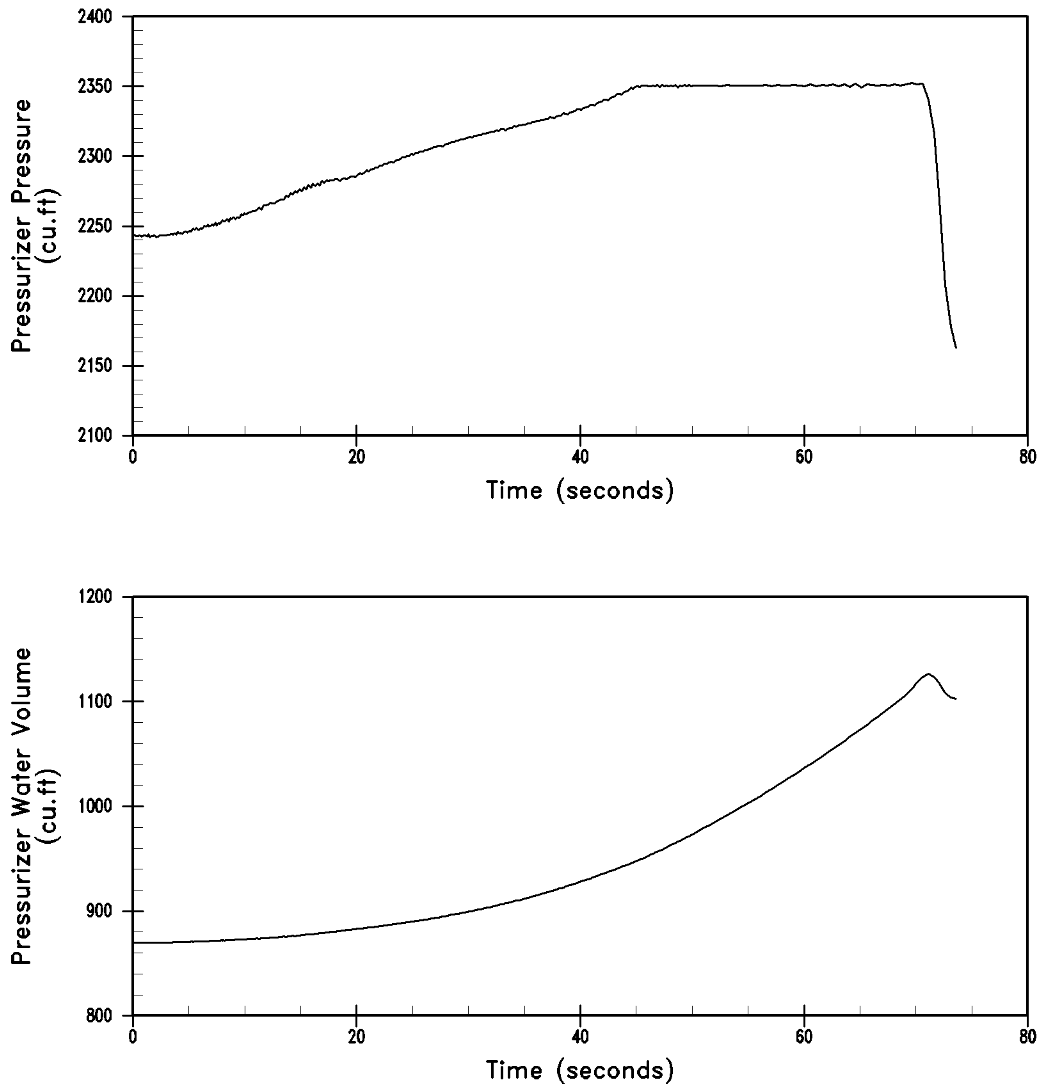
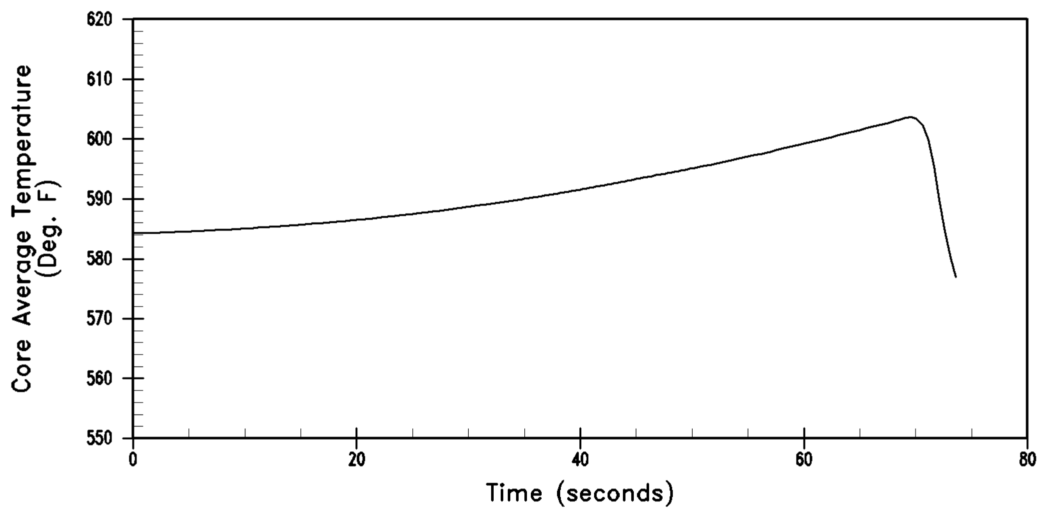


Figure 15.4-8

Pressurizer Pressure and Water Volume Transients for Uncontrolled Rod Withdrawal From Full Power with Minimum Feedback and 2 pcm/sec Withdrawal Rate

Beaver Valley Power Station Unit No. 2
Updated Final Safety Analysis Report



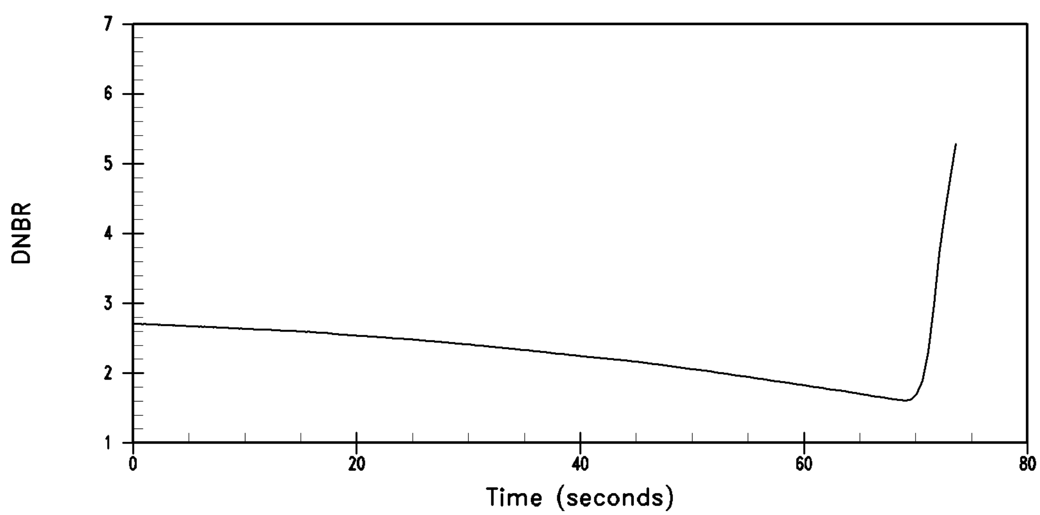


Figure 15.4-9

Core Average Temperature Transient and DNBR vs. Time for Uncontrolled Rod Withdrawal From Full Power with Minimum Feedback and 2 pcm/sec Withdrawal Rate

Beaver Valley Power Station Unit No. 2
Updated Final Safety Analysis Report

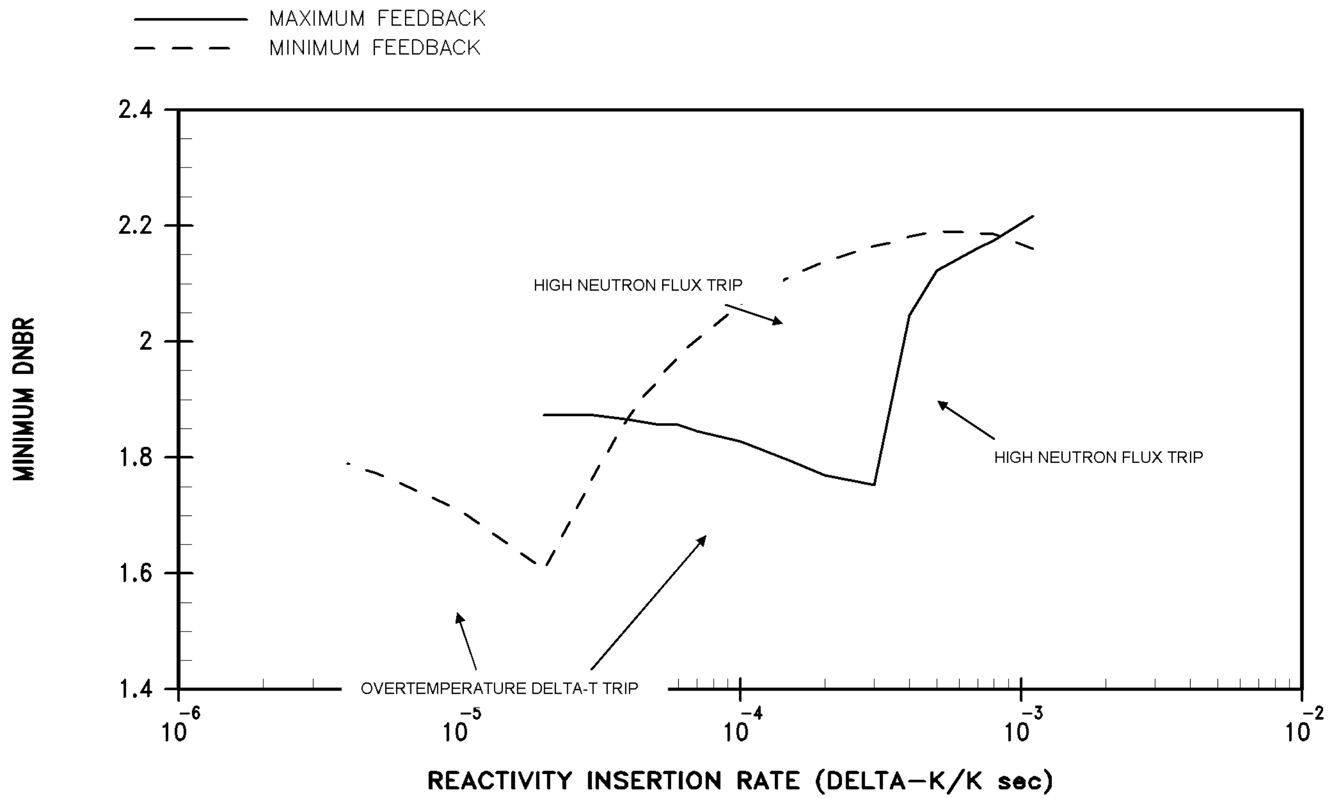


Figure 15.4-10

Uncontrolled Rod Withdrawal at Power (100% power) DNBR vs. Reactivity Insertion Rate

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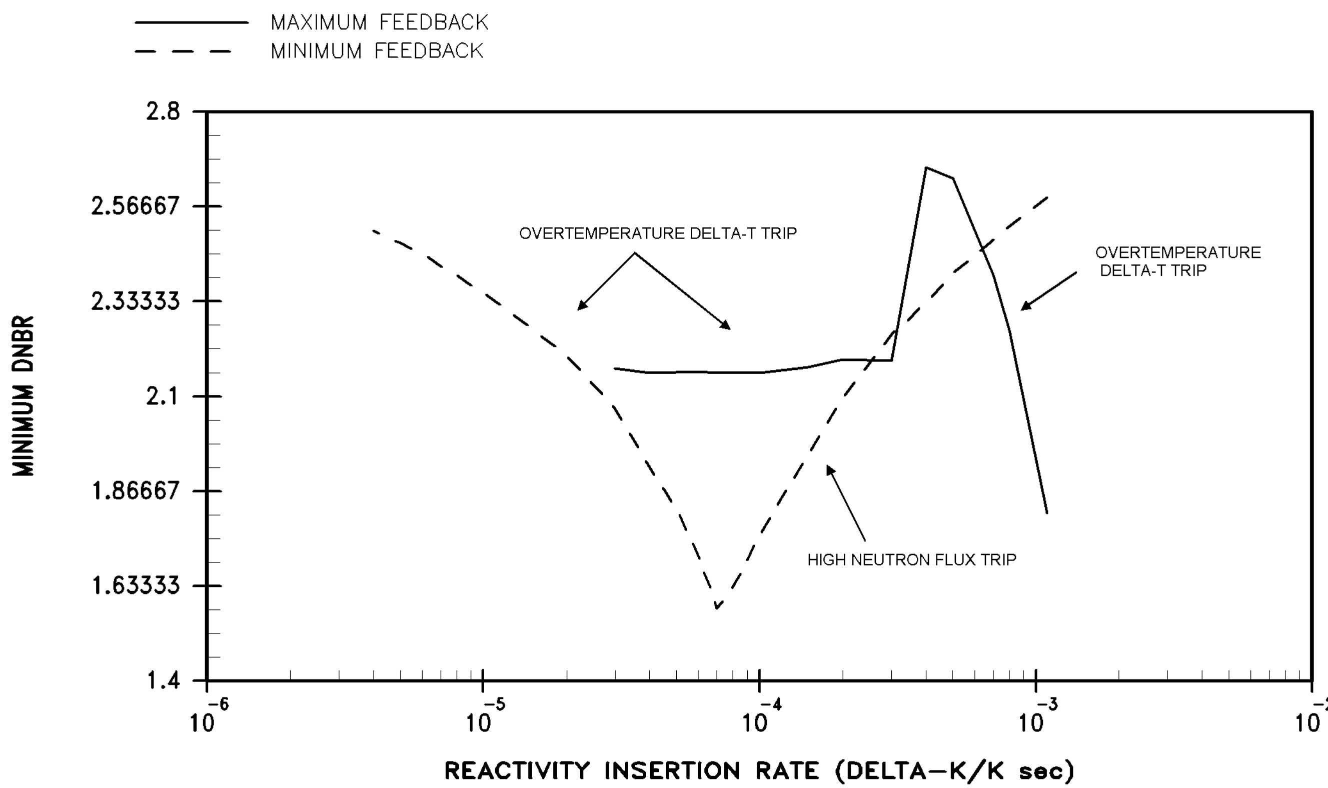

Figure 15.4-11

Uncontrolled Rod Withdrawal at Power (60% power) DNBR vs. Reactivity Insertion Rate

Beaver Valley Power Station Unit No. 2

Updated Final Safety Analysis Report

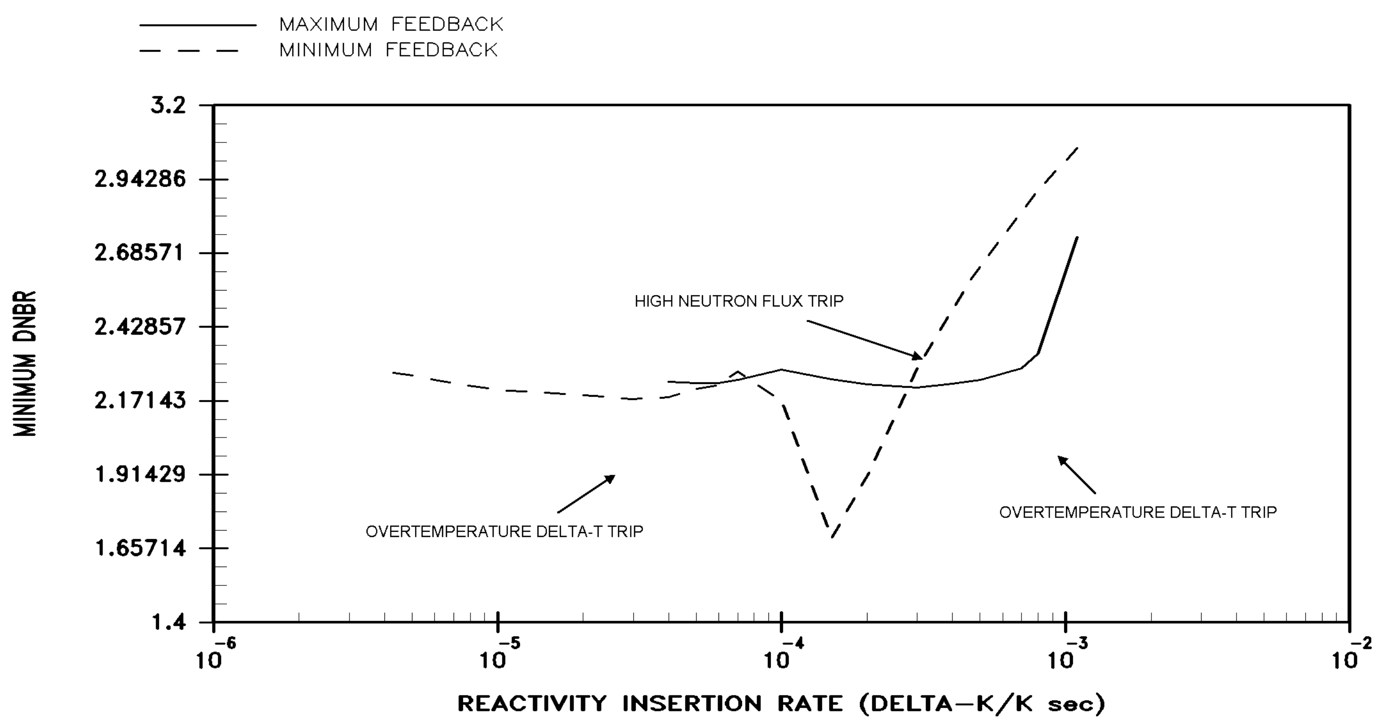

Figure 15.4-12

Uncontrolled Rod Withdrawal at Power (10% power)
DNBR vs. Reactivity Insertion Rate

Beaver Valley Power Station Unit No. 2
Updated Final Safety Analysis Report

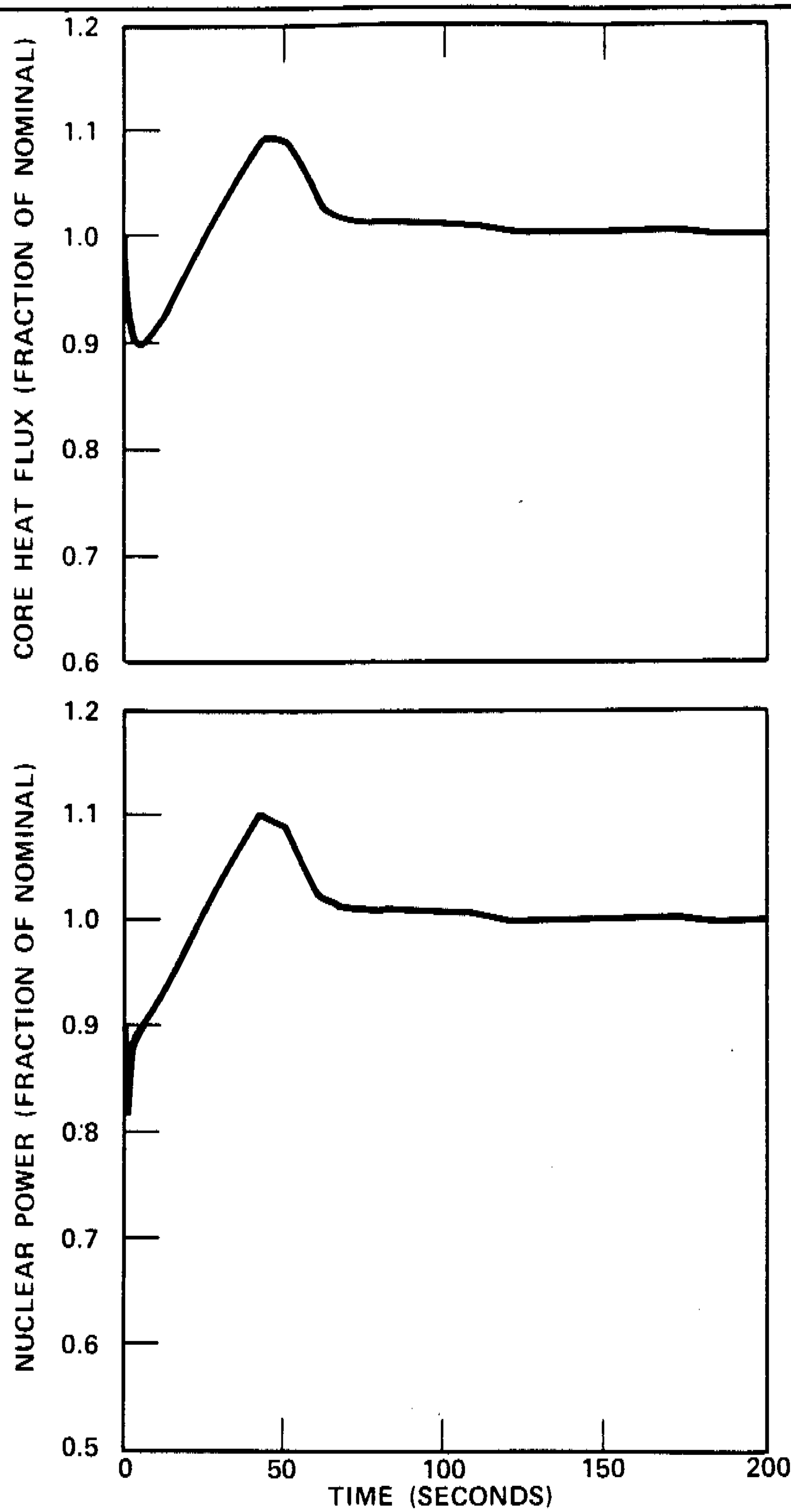

FIGURE 15.4-13

NUCLEAR POWER TRANSIENT AND CORE HEAT FLUX TRANSIENT FOR DROPPED ROD CLUSTER CONTROL ASSEMBLY

BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

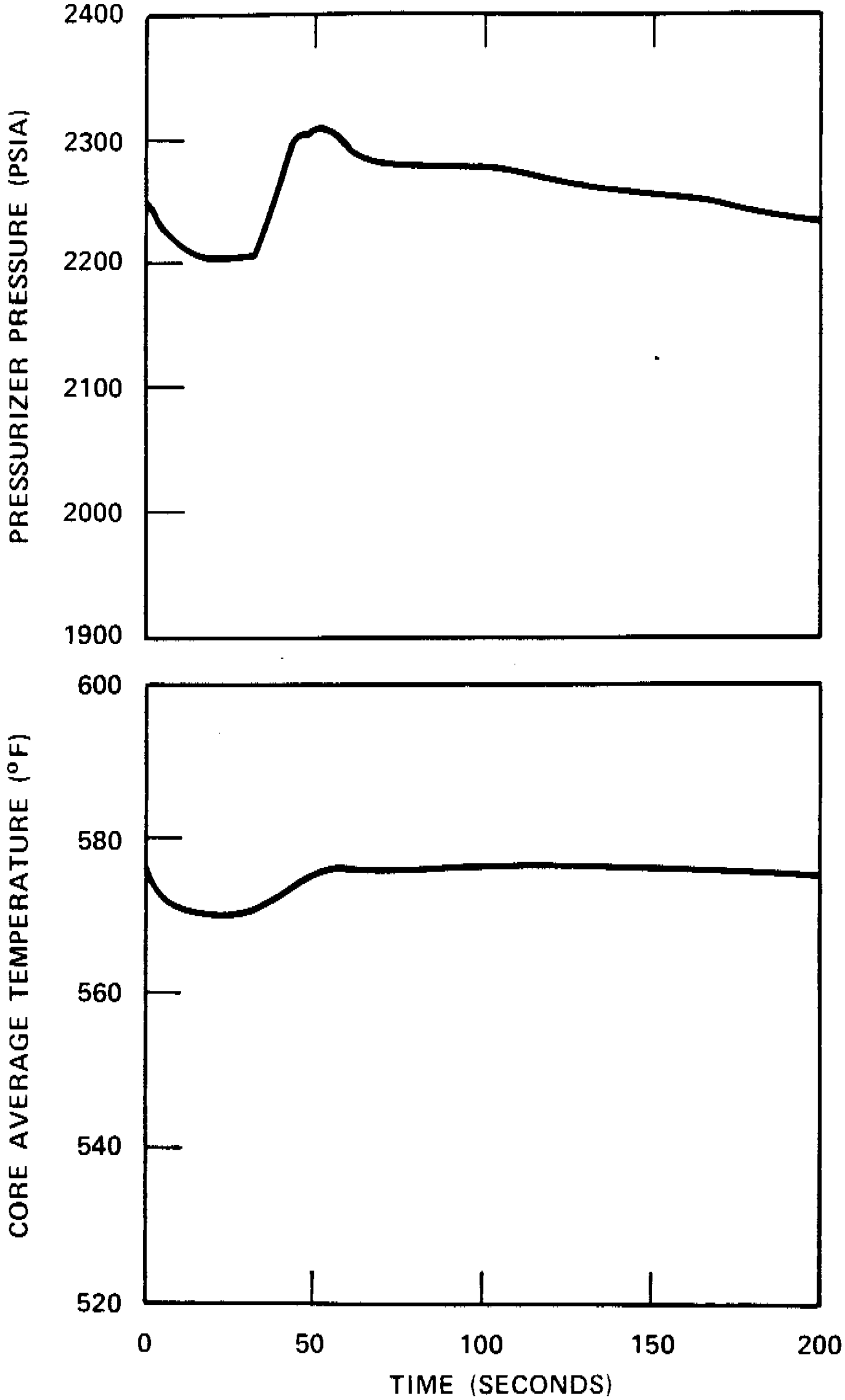

FIGURE 15.4-14
PRESSURIZER PRESSURE TRANSIENT AND CORE AVERAGE TEMPERATURE TRANSIENT FOR DROPED ROD CLUSTER CONTROL ASSEMBLY

BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

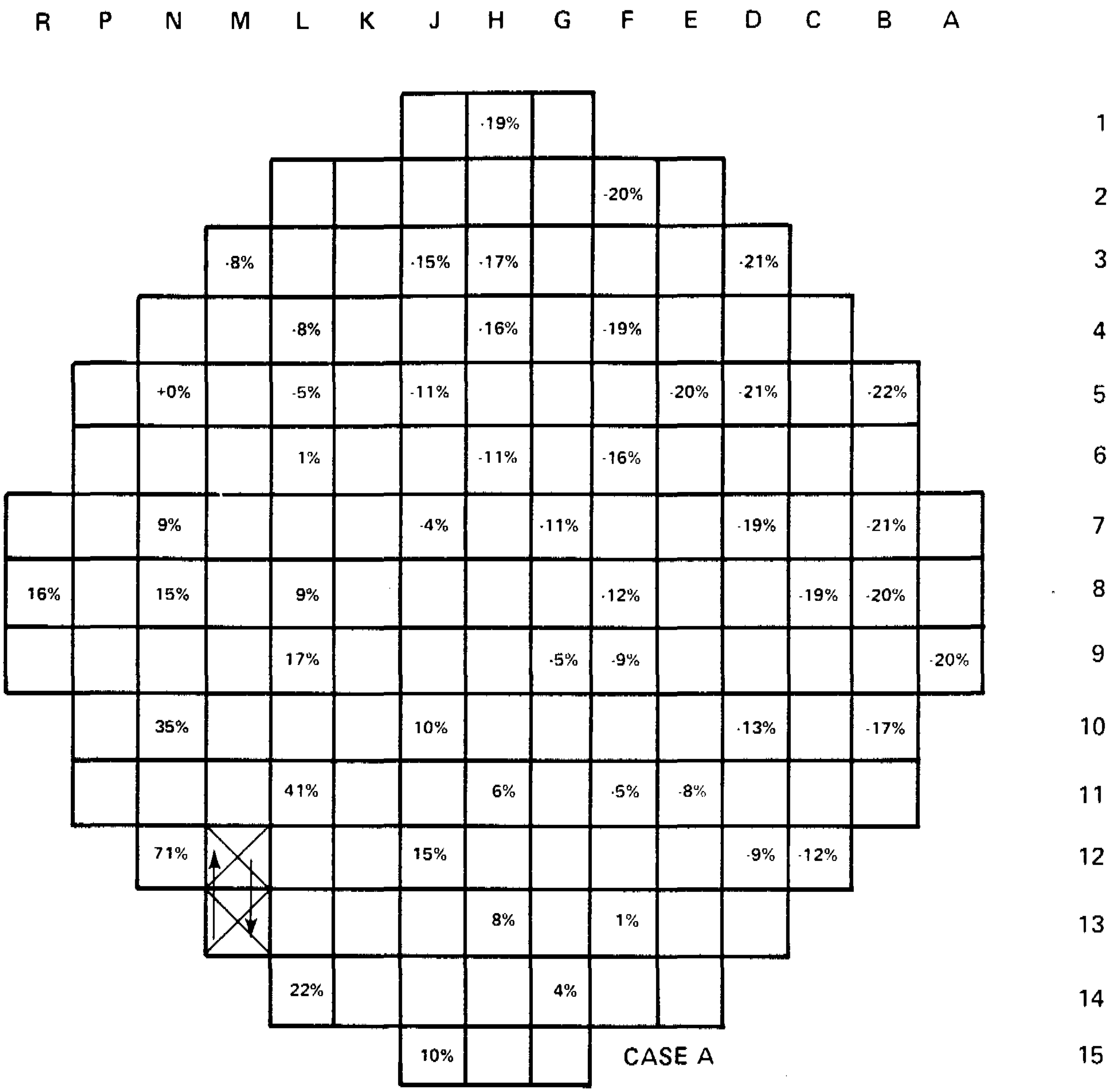

FIGURE 15.4-21
INTERCHANGE BETWEEN REGION 1 AND REGION 3 ASSEMBLY-PERCENT FLUX DEVIATION
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1								1%							
2										2%					
3				0%			1%	1%				2%			
4					0%			1%		2%					
5			-0%		0%		1%				3%	3%		3%	
6					0%			1%		3%					
7			0%				0%		2%			4%		2%	
8	-1%		-1%	-0%	-0%					4%			2%	1%	
9					-1%				-1%	-3%					+1%
10			-1%				-1%					-3%		-1%	
11					-1%			-2%		-3%	-3%				
12			-1%				-2%					-2%	-2%		
13								-2%		-2%					
14					-2%				-2%						
15							-2%								

CASE B-1

FIGURE 15.4-22

INTERCHANGE BETWEEN REGION 1 AND REGION 2 ASSEMBLY, BURNABLE POISON RODS BEING RETAINED BY THE REGION 2 ASSEMBLY-PERCENT FLUX DEVIATION

BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

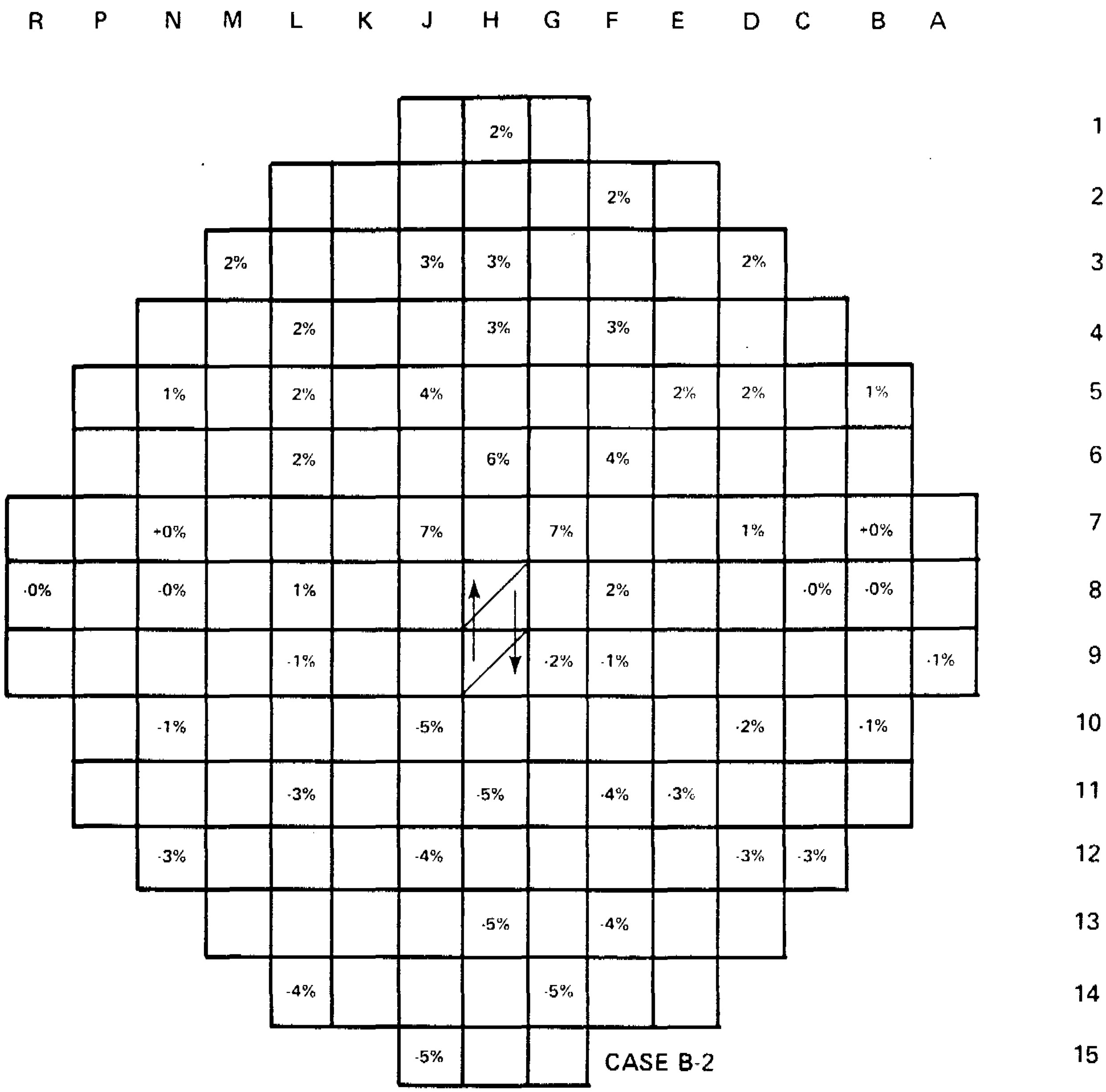

FIGURE 15.4-23
INTERCHANGE BETWEEN REGION 1 AND REGION 2 ASSEMBLY, BURNABLE POISON ROD BEING TRANSFERRED TO THE REGION 1 ASSEMBLY-PERCENT FLUX DEVIATION
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1								-4%							
2										-4%					
3				-3%			-3%	-3%				-3%			
4					-2%			-1%		-2%					
5			-3%		-1%		2%				-1%	-2%		-4%	
6					0%			7%		3%					
7			-3%				12%		12%			-1%		-4%	
8	-4%		-3%		2%					7%			-3%	-4%	
9					2%				12%	6%					-4%
10			-3%				6%					-2%		-4%	
11					-1%			2%		0%	-1%				
12			-3%				-1%					-3%	-3%		
13								-3%		-3%					
14					-4%				-4%						
15							-4%								

CASE C

FIGURE 15.4-24
ENRICHMENT ERROR A REGION 2 ASSEMBLY LOADED INTO THE CORE CENTRAL POSITION - PERCENT FLUX DEVIATION

BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

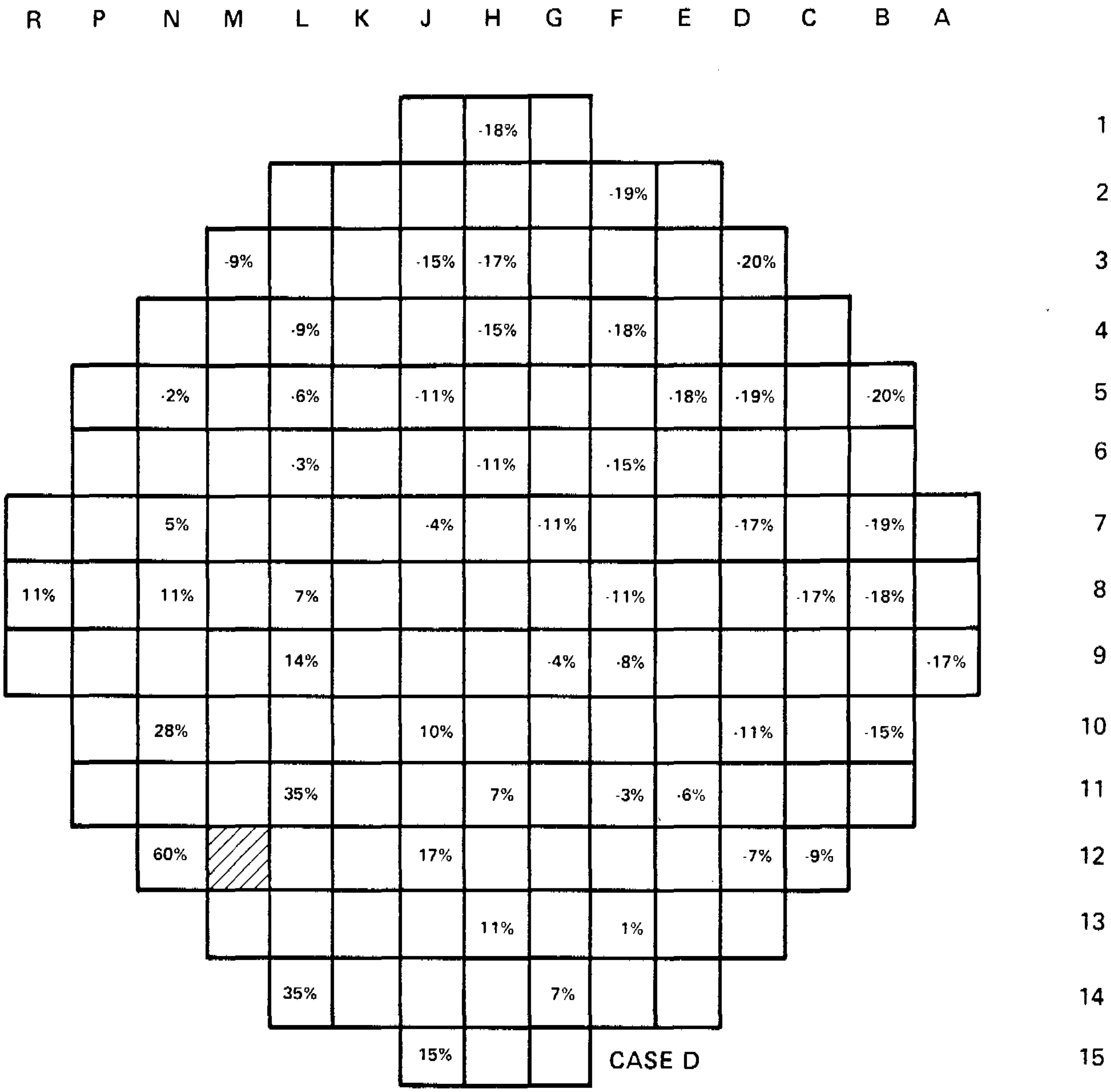

FIGURE 15.4-25

LOADING A REGION 2 ASSEMBLY INTO A REGION 1 POSITION NEAR CORE PERIPHERY-PERCENT FLUX DEVIATION

BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

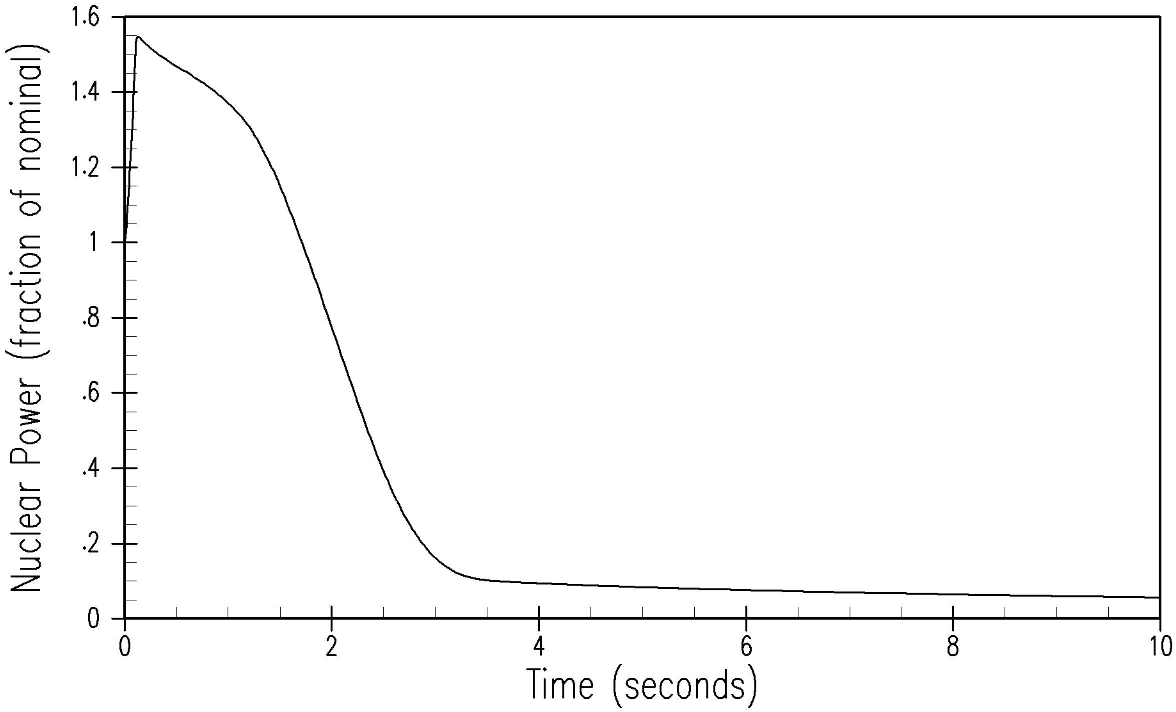

Figure 15.4-26

Nuclear Power Transient
BOL – HFP Rod Ejection Accident

Beaver Valley Power Station Unit No. 2
Updated Final Safety Analysis Report

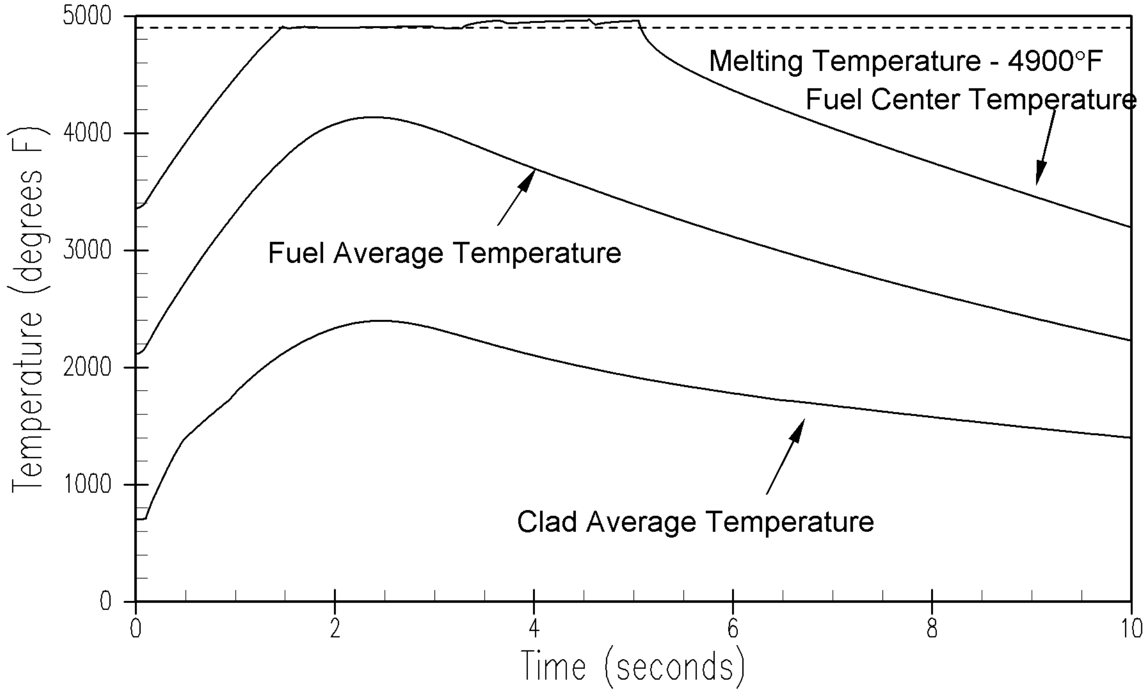

Figure 15.4-27

Hot Spot Fuel and Clad Average Temperature versus Time BOL – HFP Rod Ejection Accident

Beaver Valley Power Station Unit No. 2
Updated Final Safety Analysis Report

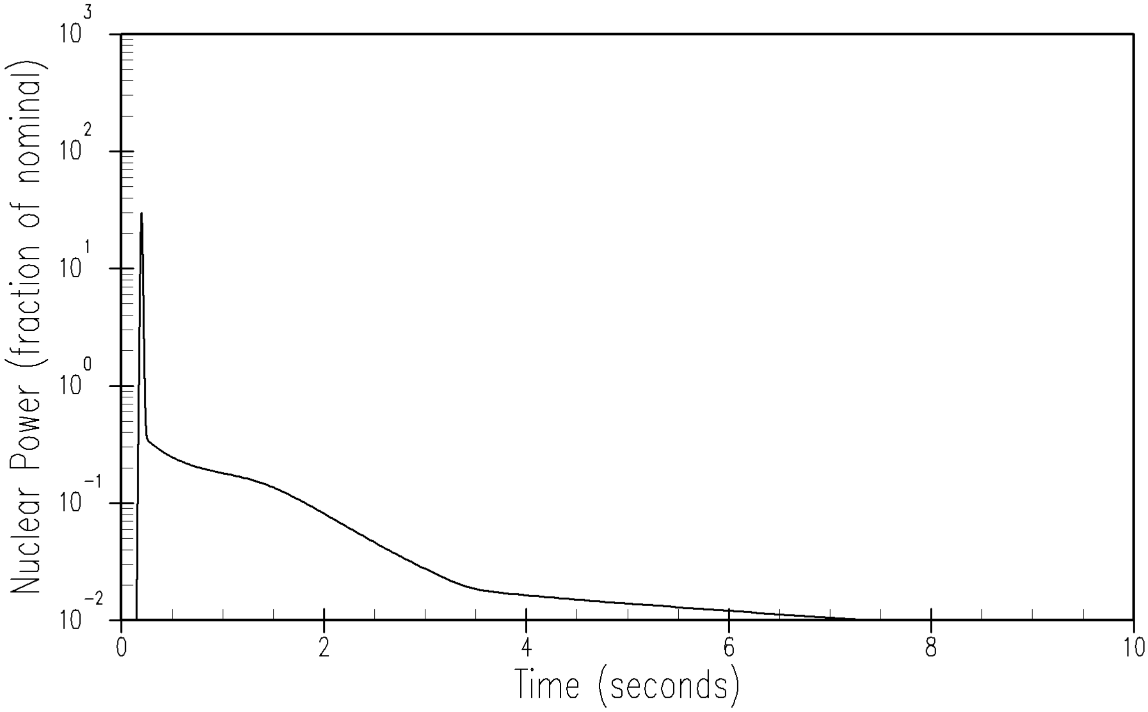

Figure 15.4-28

Nuclear Power Transient
EOL – HZP Rod Ejection Accident

Beaver Valley Power Station Unit No. 2
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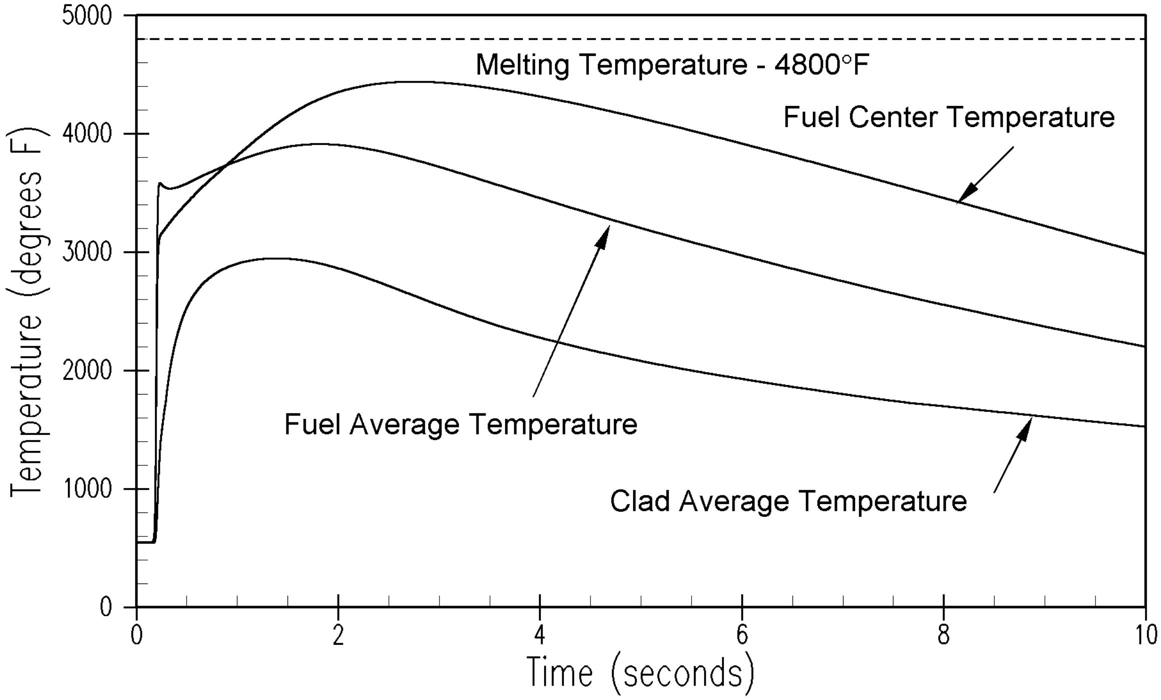

Figure 15.4-29

Hot Spot Fuel and Clad Average Temperature vs. Time EOL – HZP Rod Ejection Accident

Beaver Valley Power Station Unit No. 2
Updated Final Safety Analysis Report

15.5 INCREASE IN REACTOR COOLANT INVENTORY

Several events have been postulated which could cause an increase in reactor coolant inventory. Discussion of the following events are presented in this section:

1. Inadvertent Operation of Emergency Core Cooling System During Power Operation.

2. Chemical and Volume Control System Malfunction or Operator Error that Increases Reactor Coolant Inventory.

3. Boiling Water Reactor Transients (not applicable to BVPS-2).

The discussions are considered to be American Nuclear Standards (ANS) Condition II events. Section 15.0.1 provides 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

Spurious emergency core cooling system (ECCS) operation at power could be caused by operator error or a false electrical actuation signal. A spurious signal may originate from any of the safety injection actuation channels as described in Section 7.3.

Following the actuation signal, the suction of the charging/high head SI pumps is diverted from the volume control tank to the refueling water storage tank (RWST). The charging pumps are assumed to then force the concentration of (2,600 ppm maximum) boric acid solution from the RWST through the boron injection tank bypass line through the header and injection line and into the cold leg of each loop. The low head safety injection (LHSI) pumps also start automatically but provide no injection flow when the reactor coolant system (RCS) is at normal pressure.

A safety injection (SI) signal normally results in a reactor trip followed by a turbine trip. However, it cannot be assumed that any single fault that actuates the ECCS will also produce a reactor trip. If a reactor trip is generated by the spurious SI signal, the operator should determine if the spurious signal was transient or steady state in nature. The operator must also determine if the Safety Injection signal should be blocked. For a spurious occurrence, the operator would stop the safety injection, after verifying the required criteria for terminating safety injection, and maintain the plant in the hot shutdown condition. If the ECCS actuation instrumentation must be repaired, future plant operation will be in accordance with the Technical Specifications.

If the reactor protection system (RPS) does not produce an immediate trip as a result of the spurious SI signal, the reactor still experiences a negative reactivity excursion due to the injected boron causing a decrease in reactor power. The power mismatch causes a drop in reactor coolant average temperature (T_{avg}) and consequent coolant shrinkage, pressurizer pressure, and level drop. Turbine load will decrease due to the effect of reduced steam pressure after the turbine throttle valve is fully open. If automatic rod control is used, these effects will be lessened until the rods have moved out of the core. The transient is eventually terminated by the RPS low pressure trip or by manual trip.

The time to trip is affected by initial operating conditions including core burnup history which affects initial boron concentration, rate of change of boron concentration, Doppler, and moderator coefficients. At lower loads coolant contraction will be slower resulting in a longer time to trip.

Recovery from this second case is made in the same manner as described for the case where the SI signal results directly in a reactor trip. The only difference is the lower T_{avg} and pressure associated with the power mismatch during the transient. The time at which reactor trip occurs is of no concern for this occurrence.

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

As indicated in Section 15.0.1.5, it is also necessary to assess the effect of the event on pressurizer safety valve (PSV) operability. Should the transient result in pressurizer overfill prior to the passing of an acceptable amount of time which allows the operator to diagnose and terminate the event, water could be passed through the pressurizer safety valves. If the number of water relief events is excessive and/or the fluid temperature during the relief is too low, the potential for damage to the PSVs exists. Such damage, potentially leading to a failure of a valve to reclose, must be avoided to preclude the possibility of the Condition II Inadvertent Initiation ECCS event to progress to a Condition III Small Break LOCA. As such, separate cases are considered for an assessment of the PSV operability. These cases are discussed in Section 15.5.1.2.

15.5.1.2 Analysis of Effects and Consequences

Based on historical precedence, this event does not lead to a serious challenge of the DNB design basis. The decrease in core power and RCS average temperature more than offset the decrease in RCS pressure such that the minimum calculated DNBR occurs at the start of the transient. As such, no explicit reanalysis of this event has been performed to address DNB concerns. The discussion and results in this section correspond to a typical DNBR analysis of the Inadvertent ECCS Actuation event except for Section 15.5.1.2.1, which discusses the PSV operability/ pressurizer overfill case. The results for the DNB case are being maintained for historical purposes.

Method of Analysis

The spurious operation of the ECCS is analyzed by employing the detailed digital computer program LOFTRAN (Burnett et al 1984).

Because of the power and temperature reduction during the transient, operating conditions do not approach the core limits. Analysis of several cases shows the results are relatively independent of time to trip.

A typical transient is presented for historical purposes, representing minimum reactivity feedback. Results with maximum reactivity feedback are similar except that the transient is slower. Initial operating conditions are assumed at values consistent with steady state operation. Beaver Valley Power Station - Unit 2 characteristics and initial conditions are further discussed in Section 15.0.3.

The assumptions are as follows:

1. Initial operating conditions

   Initial reactor power and RCS temperatures are assumed to be at their maximum values consistent with the steady state full power operation including allowances for calibration and instrument errors. The initial RCS pressure is assumed at a minimum value consistent with steady state full power operation including allowances for calibration and instrument errors. This results in the minimum margin to core protection limits at the initiation of the accident. Cases with three loops in operation are considered.

2. Moderator and Doppler coefficients of reactivity

   A least negative moderator temperature coefficient is used. A least negative Doppler power coefficient was assumed (Figure 15.0-2).

3. Reactor control

   The reactor is assumed to be in manual control.

4. Pressurizer heaters

   Pressurizer heaters are assumed to be inoperable in order to increase the rate of pressure drop.

5. Boron injection

   At time zero two charging/high head SI pumps inject borated water into the cold legs of each loop.

6. Turbine load

   Turbine load is assumed constant until the governor drives the throttle valve wide open. Then turbine load drops as steam pressure drops.

7. Reactor trip

Reactor trip is initiated by low pressurizer pressure.

Plant systems and equipment which are necessary to mitigate the effects of the accident are discussed in Section 15.0.8 and listed in Table 15.0-6. No single active failure in any of these systems or equipment will adversely affect the consequences of the accident.

Results

Figures 15.5-1, 15.5-2 and 15.5-3 (for three loops in operation) show the transient response to inadvertent operation of ECCS during power operation. Neutron flux starts decreasing immediately due to the injection of boron but steam flow does not decrease until well into the transient when the turbine throttle valve goes wide open. The mismatch between load and nuclear power causes T_{avg}, pressurizer water level, and pressurizer pressure drop. When the low pressure trip set point is reached, the reactor trips and control rods start moving into the core. Departure from nucleate boiling ratio (DNBR) increases throughout the transient.

The calculated sequence of events is shown in Table 15.5-1. After reactor trip, pressure and temperature slowly rise since the turbine is tripped and the reactor is producing some power due to delayed neutron fissions, and decay heat. Recovery from this accident is discussed in Section 15.5.1.1.

15.5.1.2.1 Pressurizer Safety Valve (PSV) Operability Assessment

Since the potential for overfilling the pressurizer exists during an Inadvertent Initiation of the ECCS event, water could be discharged from the pressurizer. Under certain conditions, the BVPS-2 Technical Specifications allow the pressurizer relief valves to be blocked. Should the event occur when the relief valves are blocked, water could be discharged through the pressurizer safety valves (PSVs) and the PSVs are not designed to pass water. As such, some additional cases are considered to address PSV operability concerns.

These additional cases are similar to the cases discussed earlier except for the following:

1. The initial reactor coolant average temperature is assumed to be 95°F below its nominal value and the initial pressurizer pressure is assumed to be 45 psi below its nominal value.

2. Reactor trip is assumed to occur from the SI signal coincident with the start of the transient. This assumption exacerbates the pressurizer water volume transient.

3. The initial pressurizer water level is assumed to be at its nominal full power level plus 7% span to account for uncertainties.

4. The pressurizer sprays are modeled at their full capacity. Operation of the pressurizer sprays tends to fill the pressurizer faster.

5. Operation of the pressurizer heaters will tend to fill the pressurizer faster but operation of the heaters will also result in more favorable fluid conditions at the time of filling. Thus, cases are evaluated both with and without the pressurizer heaters.

Both cases were analyzed and both predicted pressurizer overfill prior to 10 minutes. The case with pressurizer heaters and the case without the heaters predicted five PSV openings prior to 10 minutes. The water relief temperature for the case without pressurizer heaters is slightly lower than the case with pressurizer heaters assumed. An assessment of the resulting fluid conditions was conducted and it was concluded that PSV operability was maintained. The fluid conditions do not challenge the integrity of the PSVs.

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 from this transient, the radiological consequences associated with atmosphere steam release from this event are less severe than those of the loss of nonemergency ac power to the station auxiliaries event described in Section 15.2.6.

15.5.1.4 Conclusions

Results of the analysis show that spurious safety injection without immediate reactor trip presents no hazard to the integrity of the RCS or violation to the DNB limit.

If the reactor does not trip immediately, the low pressurizer pressure reactor trip will be actuated. This trips the turbine and prevents excess cooldown thereby expediting recovery from the incident. The radiological consequences of this event are not limiting.

15.5.2 Chemical and Volume Control System Malfunction or Operator Error Increases Reactor Coolant Inventory

An increase in reactor coolant inventory which results from the addition of cold, unborated water to the RCS is analyzed in Section 15.4.6. An increase in reactor coolant inventory which results from the injection of borated water into the RCS is analyzed in Section 15.5.1.

15.5.3 BWR Transients

This section is not applicable to BVPS-2.

15.5.4 References for Section 15.5

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

Tables for Section 15.5

TABLE 15.5-1

TIME SEQUENCE OF EVENTS FOR AN INCIDENT WHICH RESULTS IN AN INCREASE IN REACTOR COOLANT INVENTORY

Accident	Event	Time (sec)
Inadvertent operation of ECCS during power operation*	Spurious safety injection signal generated. Charging pumps begin injecting borated water.	0.0
	Turbine throttle control valve wide open	37
	Low pressurizer pressure reactor trip point reached	74.7
	Control rod motion begins	76.7

* The results presented here correspond to a typical analysis of the Inadvertent ECCS Actuation event. The results are being maintained for historical purposes.

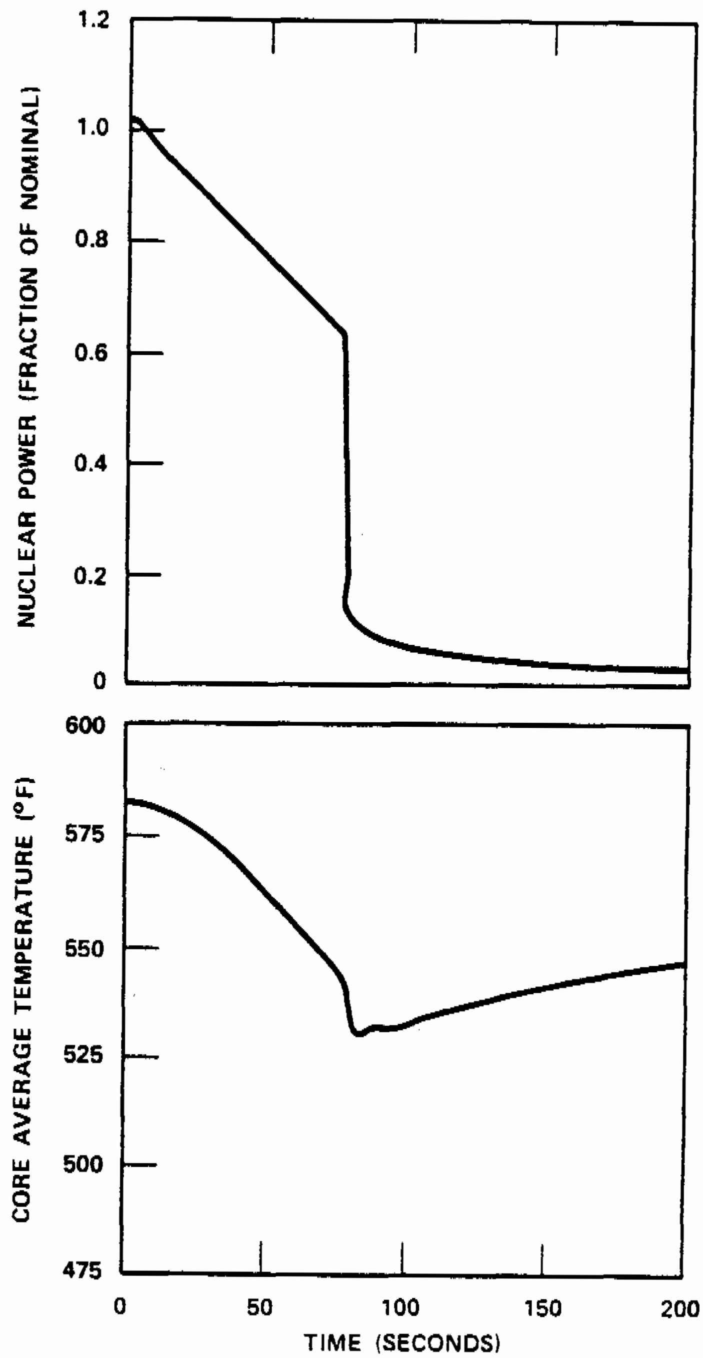

FIGURE 15.5-1
NUCLEAR POWER TRANSIENT AND CORE AVERAGE TEMPERATURE TRANSIENT FOR INADVERTENT OPERATION OF ECCS DURING N-LOOP OPERATION
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

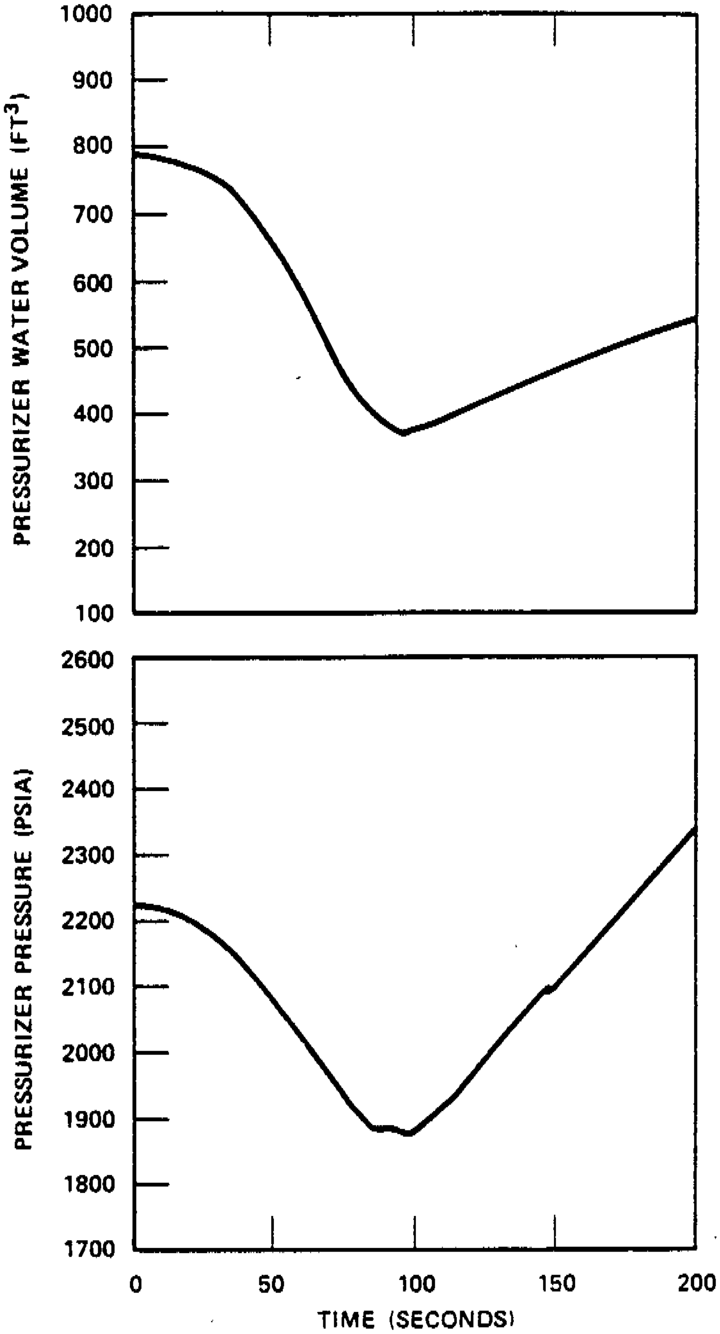

FIGURE 15.5-2
PRESSURIZER PRESSURE TRANSIENT AND PRESSURIZER WATER VOLUME TRANSIENT FOR INADVERTENT OPERATION OF ECCS DURING N-LOOP OPERATION
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

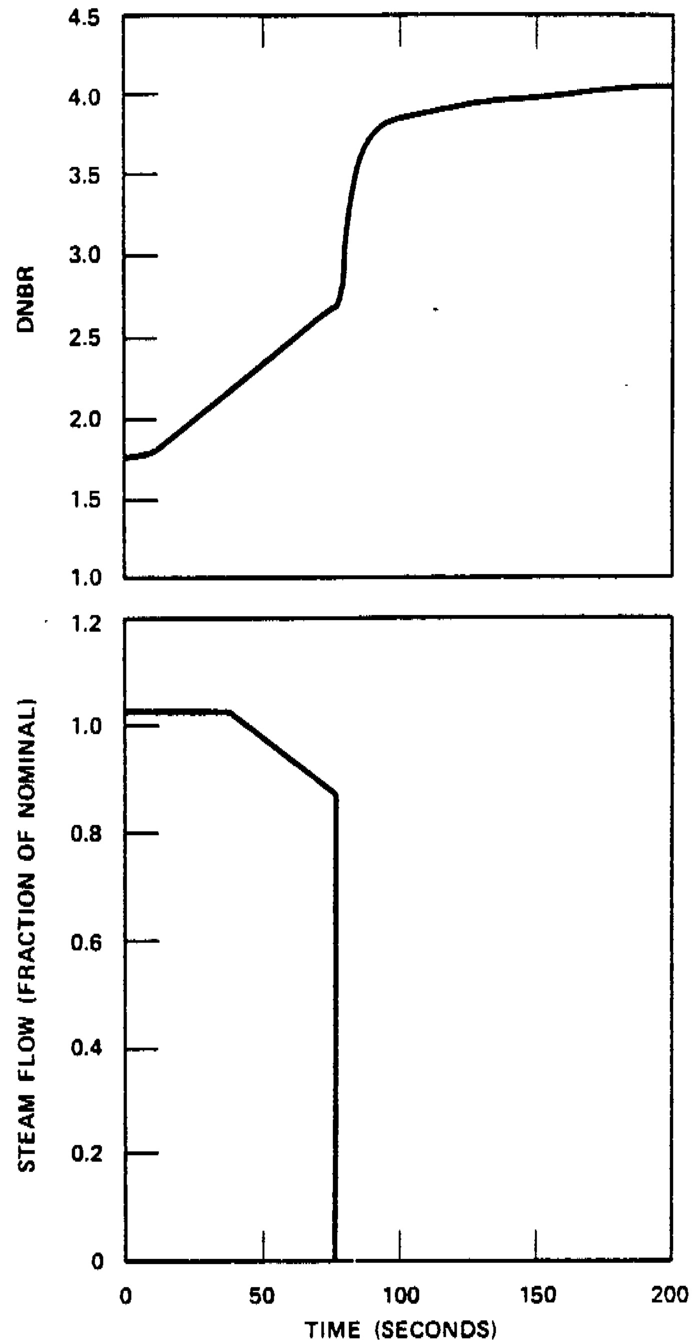

FIGURE 15.5-3
DNBR AND STEAM FLOW TRANSIENTS FOR INADVERTENT OPERATION OF ECCS DURING N-LOOP OPERATION
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

15.6 DECREASE IN REACTOR COOLANT INVENTORY

Events which result in a decrease in reactor coolant inventory as discussed in this section are as follows:

1. Inadvertent opening of a pressurizer relief valve (Section 15.6.1).
2. Failure of small lines carrying primary coolant outside containment (Section 15.6.2).
3. Steam generator tube failure (Section 15.6.3).
4. Boiling water reactor piping failure outside containment (not applicable) (Section 15.6.4).
5. Loss-of-coolant accident (LOCA) resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary (RCPB) (Section 15.6.5).
6. Boiling water reactor transients is not applicable to Beaver Valley Power Station - Unit 2 (BVPS-2) (Section 15.6.6).

Items 1 and 2 are considered to be American Nuclear Society (ANS) Condition II events. Items 3 and 5 are considered to be ANS Condition IV events (Section 15.0.1).

15.6.1 Inadvertent Opening of a Pressurizer Relief Valve

15.6.1.1 Identification of Causes and Accident Description

An accidental depressurization of the reactor coolant system (RCS) could occur as a result of an inadvertent opening of a pressurizer relief valve. Initially, the event results in a rapidly decreasing RCS pressure which could reach hot leg saturation conditions without reactor protection system intervention. At that time, the pressure decrease is slowed considerably. The pressure continues to decrease throughout the transient. The effect of the pressure decrease would be to increase power via the moderator density feedback, but the reactor control system (if in the automatic mode) functions to maintain the power essentially constant throughout the initial stage of the transient. The average coolant temperature decreases slowly, but the pressurizer level increases due to voiding from the pressure decrease, prior to the reactor trip on low pressurizer pressure.

The reactor may be tripped by the following reactor protection system (RPS) signals:

1. Overtemperature ΔT, and
2. Pressurizer low pressure.

An inadvertent opening of a pressurizer relief valve is classified as an ANS Condition II event, a fault of moderate frequency. Section 15.0.1 discusses 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).

Initial operating conditions are assumed at values consistent with steady-state operation. Plant characteristics and initial conditions are discussed in Section 15.0.3. In order to give conservative results in calculating the departure from nucleate boiling ratio (DNBR) during the transient, the following assumptions are made:

1. Initial reactor power, pressure, and RCS temperatures are assumed to be at their nominal values consistent with the steady state full power operation consistent with the revised thermal design procedure (Reference 29). This results in the minimum margin to core protection limits at the initiation of the accident.

2. A most positive moderator temperature coefficient of reactivity 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.

3. A least negative Doppler only power coefficient is assumed (Figure 15.0-2) in order to limit the amount of negative feedback as power increases due to moderator feedback.

4. One case is analyzed considering three loops in operation.

Plant systems and equipment which are available to mitigate the effects of RCS depressurization caused by an inadvertent relief valve opening are discussed in Section 15.0-8 and listed in Table 15.0-6.

Normal reactor control systems are not required to function. The rod control system is assumed to be in the automatic mode to hold the core at full power longer and thus delay the trip. This is a worst-case assumption; if the reactor were in manual control, an earlier trip could occur on low pressurizer pressure. The RPS functions to trip the reactor on the appropriate signal. No single active failure will prevent the RPS from functioning properly.

Results

The system response to an inadvertent opening of a pressurizer relief valve is shown on Figures 15.6-1, 15.6-2 and 15.6-3. 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 pressurizer pressure. The pressure decay transient following the accident is given on Figure 15.6-2. Pressure drops more rapidly after core heat generation is reduced via the trip, and then slows once saturation temperature is reached in the hot leg. The DNBR decreases initially, but increases rapidly following the trip, as shown on Figure 15.6-3. The DNBR remains above the limit value throughout the transient.

Following reactor trip, RCS pressure will continue to fall until flow through the inadvertently opened valve is terminated. Automatic actuation of the safety injection system may occur if the pressure falls to the low pressurizer pressure safety injection set point. The RCS pressure will stabilize following operator action to terminate flow to the inadvertently opened valve; normal operating procedures may then be followed. The operating procedures would call for operator action to control RCS boron concentration and pressurizer level using the chemical and volume control system (CVCS) and to maintain steam generator level through control of the main or auxiliary feedwater system. Any action required of the operator to stabilize BVPS-2 will be in a time frame in excess of ten minutes following reactor trip.

The calculated sequence of events for the inadvertent opening of a pressurizer relief valve incident is shown in Table 15.6-1.

15.6.1.3 Radiological Consequences

An inadvertent opening of a pressurizer relief valve releases primary coolant to the pressurizer relief tank. However, assuming a direct release to the containment atmosphere, the radiological consequences of this event would be substantially less than those of a LOCA (Section 15.6.5). This is 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 RPS signals provide adequate protection against the RCS depressurization event. The DNBR remains above the limit value throughout the transient; thus, the departure from nucleate boiling (DNB) design-basis as described in Section 4.4 is met.

15.6.2 Failure of Small Lines Carrying Primary Coolant Outside Containment

15.6.2.1 Identification of Causes and Accident Description

Lines connected to the RCS and penetrating the containment, as well as isolation provisions are identified in Table 6.2-60.

There are no instrument lines connected to the RCS that penetrate the containment. There are, however, the sample lines from the hot and cold legs of reactor coolant loops and the steam and liquid space of the pressurizer, and the CVCS letdown and excess letdown lines that penetrate the containment. The sample lines and the CVCS letdown and excess letdown lines are all provided with normally open containment isolation valves on both sides of the containment wall. In all cases, the containment isolation valves are designed in accordance with the containment isolation requirements of General Design Criterion 55 (Section 6.2.4).

The most severe small line rupture with regard to radioactivity release during normal BVPS-2 operation is a complete severance of the 2-inch letdown line at a location outside containment, downstream of the letdown heat exchanger and with a coincident loss of heat exchanger cooling. This event would result in a loss-of-reactor coolant at the rate of approximately 16.0 lbm/s based on a density of 44.13 lbs/ft^3 and on the flow restriction provided by two of the three letdown line orifices in service (the 45-gpm orifice and one of the 60-gpm orifices), shown on Table 9.3-8.

The time required for the operator to identify the accident and isolate the rupture is expected to be less than 15 minutes. Diverse instrumentation in the form of letdown line pressure and flow downstream of the postulated break location, volume control tank level and pressurizer level with indication at the main control board will allow detection of the failure by the operator. In addition, a control room operator can determine specific plant areas which are experiencing high radiation after receiving plant high radiation annunciation. The operator would isolate the letdown line rupture by closing the letdown orifice isolation valves, 2CHS*AOV200A, B, and C or the pressurizer low level isolation valves, 2CHS*LCV460A and 2CHS*LCV460B. All valves are provided with control switches with indicating lights at the main control board and at the emergency shutdown panel. All valves are air-operated and designed to fail close on loss of air or electrical power. There are no single failures that would prevent isolation of the letdown line rupture.

15.6.2.2 Analysis of Effects and Consequences

Method of Analysis

The amount of primary coolant released is conservatively estimated by assuming critical flows in the ruptured letdown line. The mass of fluid released from the postulated break was calculated using the Zaloudek correlation in WCAP-8312-A (Reference 2) for subcooled liquids and the theoretical model developed by Moody for saturated conditions. Immediately after the rupture, the Moody model is used for a saturated liquid until the liquid in the letdown line between the orifices and rupture point is depleted. After the liquid is depleted, Zaloudek's subcooled correlation is used at the orifice and continues until isolation occurs at 15 minutes after the break. These critical flow correlations are in accordance with WCAP-8312-A.

15.6.2.3 Radiological Consequences

The failure outside the containment of small lines carrying primary coolant is postulated to occur in the letdown line after the letdown heat exchanger. The rupture of this line will result in the loss of primary coolant, with isolation occurring within 15 minutes. The rupture will result in the discharge of primary coolant directly into the auxiliary building or into the contiguous areas, with the radioactivity released to the environment at ground level. All potential locations for the small line break in the auxiliary building are within ventilation zones of the supplementary leak collection and release system (SLCRS). A small line break in the contiguous areas would be serviced by the SLCRS after receipt of a high radiation signal from a ventilation monitor that is not safety-related. However, the conservative analysis does not take credit for SLCRS operation.

The assumptions for evaluating the radiological consequences of the postulated small line failure are summarized in Table 15.6-2. The conservative analysis assumes primary coolant Technical Specification equilibrium activities as presented in Table 15.0-8c.

Additionally, a concurrent iodine spike is postulated to occur with iodine release rates into the primary coolant at a rate calculated using the methodology provided in Table 15.0-10a. The resulting releases to the environment are based on the stated assumptions.

Computer program PERC2 is used to calculate the control room and site boundary doses due to airborne radioactivity releases following a small line break outside Containment at BVPS. Bounding parameter values are used to encompass an event at either unit.

Regulatory Guide 1.183 does not address a small line break outside Containment. The dose assessment herein follows the current BVPS licensing basis model, but for purposes of consistency, uses the most limiting dose limits set by Regulatory Guide 1.183 for accident evaluations.

The SLB outside containment postulates the break of the 2-inch RCS letdown line in the Auxiliary Building resulting in a maximum break flow of 16.79 lbm/sec. Thirty-seven percent of the break flow is calculated to flash. The iodine activity in the break flow is assumed to become airborne in proportion to the flash fraction, whereas the noble gases are assumed to be airborne and discharged to the environment without decontamination or holdup.

The activity in the Auxiliary Building is released to the environment via the Ventilation Vent. The most limiting atmospheric dispersion factors between the ventilation vent release point at each unit relative to the two CR intakes (identified for purposes of assessment as the BVPS-1 Ventilation Vent to the BVPS-1 CR intake) is selected to determine a bounding control room dose. No credit is taken for Auxiliary building holdup or filtration.

The radiological consequences resulting from a postulated failure of a small line carrying primary coolant outside containment are presented in Tables 15.0-12 and 15.0-13. The offsite doses are determined using the calculated environmental releases for this accident and the atmospheric dispersion values given in Tables 15.0-11, 15.0-14 and 15.0-15. The methodology for calculating the offsite doses is discussed in Appendix 15A.

<u>EAB 2 hr Worst Case Window</u>

AST methodology requires that the worst case dose to an individual located at any point on the boundary at the EAB, for any 2-hr period following the onset of the accident be reported as the EAB dose. Since the event is based on a 15-minute release, the worst 2-hour period for the EAB is the 0 to 2-hour period.

<u>Accident-Specific Control Room Model Assumptions</u>

The control room is assumed to remain in the normal operation mode. The critical control room parameters utilized in this model are summarized in Table 6.4-1a. Section 15.6.5.4 discusses the control room design as related to dose consequences under a sub-section titled “Control Room Habitability.”

The radiological consequences for this event are a small fraction of the guidelines of 10 CFR 50.67.

15.6.3 STEAM GENERATOR TUBE RUPTURE (SGTR)

15.6.3.1 Identification of Cause and Accident Description

The accident examined is the complete severance of a single steam generator tube. The accident is assumed to take place at power with the reactor coolant contaminated with fission products corresponding to continuous operation with a limited number of defective fuel rods. The accident leads to an increase in contamination of the secondary system due to leakage of radioactive coolant from the reactor coolant system (RCS). In the event of a coincident loss of offsite power, or failure of the Condenser Steam Dump System, discharge of radioactivity to the atmosphere takes place via the steam generator atmospheric steam dump valves (and safety valves if their setpoint is reached).

The steam generator tube material is Inconel-600 and is a highly ductile material; thus, it is considered that the assumption of a complete severance of a tube is conservative. The more probable mode of tube failure would be one or more smaller leaks of undetermined origin. Activity in the steam and power conversion system is subject to continual surveillance, and, an accumulation of such leaks which exceeds the limits established in the Technical Specifications is not permitted during Unit operation.

Due to the series of alarms as described below, the operator will readily determine that a steam generator tube rupture has occurred, identify and isolate the ruptured steam generator, and complete the required recovery actions to stabilize the plant and terminate the primary to secondary break flow. The recovery procedure can be completed on a time scale which ensures that break flow to the secondary system is terminated before water level in the affected steam generator rises into the main steam pipe. Sufficient indications and controls are provided to enable the operator to carry out these functions.

Assuming normal operation of the various plant control systems, the following sequence of events is initiated by a tube rupture:

1. Pressurizer low pressure and low level alarms are actuated and charging pump flow increases in an attempt to maintain pressurizer level. On the secondary side there is a steam flow/feedwater flow mismatch before the trip as feedwater flow to the affected steam generator is reduced due to the break flow which is now being supplied to that steam generator from the primary side.

2. The air ejector discharge radiation monitor, steam generator blowdown sample radiation monitor, and/or main steamline radiation monitor will alarm, indicating an increase in radioactivity in the secondary system.

3. Continued loss of reactor coolant inventory leads to a reactor trip signal generated by low pressurizer pressure or by overtemperature ΔT. Resultant plant cooldown following reactor trip leads to a rapid decrease in RCS pressure and pressurizer level, and a safety injection (SI) signal, initiated by low pressurizer pressure, follows soon after the reactor trip. The SI signal automatically terminates normal feedwater supply and initiates auxiliary feedwater (AFW) addition via the AFW pumps.

4. The reactor trip automatically trips the turbine and if offsite power is available, the steam dump valves open permitting steam dump to the condenser. In the event of a coincident loss of offsite power, the steam dump valves would automatically close to protect the condenser. The steam generator pressure would rapidly increase resulting in steam discharge to the atmosphere through the steam line atmospheric dump valves (and safety valves if their setpoint is reached).

5. Following reactor trip and SI actuation, the continued action of the AFW supply and borated SI flow (supplied from the refueling water storage tank) provide a heat sink which absorbs some of the decay heat. This reduces the amount of steam bypass to the condenser, or in the case of loss of offsite power, steam relief to the atmosphere.

6. SI flow results in stabilization of the RCS pressure and pressurizer water level, and the RCS pressure trends toward an equilibrium value where the SI flow rate equals the break flow rate.

In the event of an SGTR, the plant operators must diagnose the SGTR and perform the required recovery actions to stabilize the plant and terminate the primary to secondary leakage. The operator actions for SGTR recovery are provided in the plant Emergency Operating Procedures. The major operator actions include identification and isolation of the ruptured steam generator, cooldown and depressurization of the RCS to restore inventory, and termination of SI to stop primary to secondary leakage. These operator actions are described below.

1. Identify the ruptured steam generator.

   High secondary side activity, as indicated by the air ejector discharge radiation monitor, steam generator blowdown sample radiation monitor, or main steamline radiation monitor, typically will provide the first indication of an SGTR event. The ruptured steam generator can be identified by an unexpected increase in steam generator narrow range level, high radiation from a steam generator water sample, a high radiation indication on a main steamline radiation monitor, or high radiation from a steam generator blowdown line. For an SGTR that results in a reactor trip at high power, the steam generator water level will decrease to near the bottom of the narrow range scale for all of the steam generators. The AFW flow will begin to refill the steam generators, distributing flow to each of the steam generators. Since primary to secondary leakage adds additional liquid inventory to the ruptured steam generator, the water level will increase more rapidly in that steam generator. This response, as displayed by the steam generator water level instrumentation, provides confirmation of an SGTR event and also identifies the ruptured steam generator.

2. Isolate the ruptured steam generator from the intact steam generators and isolate feedwater to the ruptured steam generator.

   Once the steam generator with a tube rupture has been identified, recovery actions begin by isolating steam flow from and stopping feedwater flow to the ruptured steam generator. In addition to minimizing radiological releases, this also reduces the possibility of overfilling the ruptured steam generator with water by 1) minimizing the accumulation of feedwater flow and 2) enabling the operator to establish a pressure differential between the ruptured and intact steam generators as a necessary step toward terminating primary to secondary leakage.

3. Cool down the RCS using the intact steam generators.

   After isolation of the ruptured steam generator, the RCS is cooled as rapidly as possible to less than the saturation temperature corresponding to the ruptured steam generator pressure by dumping steam from only the intact steam generators. This ensures adequate subcooling will exist in the RCS after depressurization of the RCS to the ruptured steam generator pressure in subsequent actions. If offsite power is available, the normal steam dump system to the condenser can be used to perform this cooldown. However, if offsite power is lost, the RCS is cooled using the atmospheric steam dump valves or the residual heat release valve to release steam from the intact steam generators.

4. Depressurize the RCS to restore reactor coolant inventory.

   When the cooldown is completed, SI flow will increase RCS pressure until break flow matches SI flow. Consequently, SI flow must be terminated to stop primary to secondary leakage. However, adequate reactor coolant inventory must first be assured. This includes both sufficient reactor coolant subcooling and pressurizer inventory to maintain a reliable pressurizer level indication after SI flow is stopped. Since leakage from the primary side will continue after SI flow is stopped until RCS and ruptured steam generator pressures equalize, an "excess" amount of inventory is needed to ensure pressurizer level remains on span. The "excess" amount required depends on RCS pressure and reduces to zero when RCS pressure equals the pressure in the ruptured steam generator.

The RCS depressurization is performed using normal pressurizer spray if the reactor coolant pumps (RCPs) are running. However, if offsite power is lost or the RCPs are not running for some other reason, normal pressurizer spray is not available. In this event, RCS depressurization can be performed using the pressurizer power operated relief valves (PORVs) or auxiliary pressurizer spray.

5. Terminate SI to stop primary to secondary leakage.

   The previous actions will have established adequate RCS subcooling, a secondary side heat sink, and sufficient reactor coolant inventory to ensure that SI flow is no longer needed. When these actions have been completed, SI flow must be stopped to terminate primary to secondary leakage. Primary to secondary leakage will continue after SI flow is stopped until RCS and ruptured steam generator pressures equalize. Charging flow, letdown, and pressurizer heaters will then be controlled to prevent repressurization of the RCS and reinitiation of leakage into the ruptured steam generator.

Following SI termination, the plant conditions will be stabilized, the primary to secondary break flow will be terminated, and all immediate safety concerns will have been addressed. At this time a series of operator actions are performed to prepare the plant for cooldown to cold shutdown conditions. Subsequently, actions are performed to cool down and depressurize the RCS to cold shutdown conditions and to depressurize the ruptured steam generator.

15.6.3.2 Analysis of Effects and Consequences

An SGTR results in the leakage of contaminated reactor coolant into the secondary system and subsequent release of a portion of the activity to the atmosphere. Therefore, an analysis must be performed to assure that the offsite radiological consequences resulting from an SGTR are within the allowable guidelines. One of the major concerns for an SGTR is the possibility of steam generator overfill since this could potentially result in a significant increase in the offsite radiological consequences. Therefore, an analysis was performed to demonstrate margin to steam generator overfill, assuming the limiting single failure relative to overfill. The results of this analysis demonstrated that there is margin to steam generator overfill for BVPS Unit 2. An analysis was also performed to determine the offsite radiological consequences, assuming the limiting single failure relative to offsite doses without steam generator overfill. Since steam generator overfill does not occur, the results of this analysis represent the limiting consequences for an SGTR for BVPS Unit 2. The analyses to demonstrate margin to overfill for a design basis SGTR for BVPS Unit 2 are presented in the Extended Power Uprate Licensing Report, and the results of the offsite radiological consequences analysis are discussed below.

A thermal and hydraulic analysis was performed to determine the plant response for a design basis SGTR, and to determine the integrated primary to secondary break flow and the mass releases from the ruptured and intact steam generators to the condenser and to the atmosphere. This information was then used to calculate the quantity of radioactivity released to the environment and the resulting radiological consequences.

15.6.3.3 Thermal and Hydraulic Analysis

The plant response following an SGTR was analyzed with the LOFTTR2 program until the primary to secondary break flow is terminated. The reactor protection system and the automatic actuation of the engineered safeguards systems were modeled in the analysis. The major operator actions which are required to terminate the break flow for an SGTR were also simulated in the analysis.

Analysis Assumptions

The accident modeled is a double-ended break of one steam generator tube located at the top of the tube sheet on the outlet (cold leg) side of the steam generator. It was assumed that the reactor is operating at full power at the time of the accident and the initial secondary mass was assumed to correspond to operation at nominal steam generator mass minus an allowance for uncertainties. It was also assumed that a loss of offsite power occurs at the time of reactor trip and the highest worth control assembly was assumed to be stuck in its fully withdrawn position at reactor trip.

The limiting single failure was assumed to be the failure of the atmospheric steam dump valve on the ruptured steam generator. Failure of this valve in the open position will cause an uncontrolled depressurization of the ruptured steam generator which will increase primary to secondary leakage and the mass release to the atmosphere. It was assumed that the ruptured steam generator atmospheric steam dump valve fails open when the ruptured steam generator is isolated, and that the valve was subsequently isolated by locally closing the associated block valve.

The major operator actions required for the recovery from an SGTR are discussed in Section 15.6.3.1 and these operator actions were simulated in the analysis. The operator action times which were used for the analysis are presented in Table 15.6-5. It is noted that the atmospheric steam dump valve on the ruptured steam generator was assumed to fail open at the time the ruptured steam generator was isolated. Before proceeding with the recovery operations, the failed open atmospheric steam dump valve was assumed to be isolated by locally closing the associated block valve. It was assumed that the ruptured steam generator atmospheric steam dump valve is isolated at 10.0 minutes after the valve was assumed to fail open. After the ruptured steam generator atmospheric steam dump valve was isolated, the additional delay time of 2.4 minutes (Table 15.6-5) was assumed for the operator action time to initiate the RCS cooldown.

Transient Description

The LOFTTR2 analysis results are described below. The sequence of events for this transient is presented in Table 15.6-4.

Following the tube rupture, reactor coolant flows from the primary into the secondary side of the ruptured steam generator since the primary pressure is greater than the steam generator pressure. In response to this loss of reactor coolant, pressurizer level decreases as shown in Figure 15.6-58. The RCS pressure also decreases as shown in Figure 15.6-59 as the steam bubble in the pressurizer expands. As the RCS pressure decreases due to the continued primary to secondary leakage, automatic reactor trip occurs at 116.4 seconds on an overtemperature ΔT trip signal.

After reactor trip, core power rapidly decreases to decay heat levels. The turbine stop valves close and steam flow to the turbine is terminated. The steam dump system is designed to actuate following reactor trip to limit the increase in secondary pressure, but the steam dump valves remain closed due to the loss of condenser vacuum resulting from the assumed loss of offsite power at the time of reactor trip. Thus, the energy transfer from the primary system causes the secondary side pressure to increase rapidly after reactor trip until the steam generator atmospheric steam dump valves (and safety valves if their setpoints are reached) lift to dissipate the energy, as shown in Figure 15.6-60.

The pressurizer level and RCS pressure decrease more rapidly after reactor trip as energy transfer to the secondary shrinks the reactor coolant and the leak flow continues to deplete primary inventory. The decrease in RCS inventory results in a low pressurizer pressure SI signal at 141.2 seconds. The main feedwater flow will be terminated and AFW flow will be automatically initiated following SI actuation. After SI actuation, the RCS pressure and pressurizer level tend to stabilize until the ruptured steam generator atmospheric steam dump valve is assumed to fail open.

Since offsite power is assumed lost at reactor trip, the RCPs trip and a gradual transition to natural circulation flow occurs. Immediately following reactor trip the temperature differential across the core decreases as core power decays (see Figures 15.6-61 and 15.6-62); however, the temperature differential subsequently increases as the reactor coolant pumps coast down and natural circulation flow develops. The cold leg temperatures trend toward the steam generator temperature as the fluid residence time in the tube region increases. The intact steam generator loop temperatures continue to slowly decrease due to the continued AFW flow until operator actions are taken to perform the RCS cooldown. The ruptured steam generator loop temperatures also continue to slowly decrease until the ruptured steam generator is isolated and the atmospheric steam dump valve was assumed to fail open.

Major Operator Actions

1. Identify and Isolate the Ruptured Steam Generator

   AFW to the ruptured steam generator was assumed to be isolated at 800 seconds after the initiation of the SGTR. It was assumed that the ruptured steam generator is isolated at 1018 seconds. The ruptured steam generator atmospheric steam dump valve was also assumed to fail open at this time, and the failure was simulated at 1020 seconds because of the computer program limitations. The failure causes the ruptured steam generator to rapidly depressurize as shown in Figure 15.6-60, which results in an increase in primary to secondary leakage.

   The depressurization of the ruptured steam generator increases the break flow and energy transfer from primary to secondary which results in a decrease in the ruptured loop temperatures as shown in Figure 15.6-62. As noted previously, the intact steam generator loop temperatures also slowly decrease, as shown in Figure 15.6-61, until the RCS cooldown is initiated. The shrinkage of the reactor coolant due to the decrease in the RCS temperatures results in a decrease in the pressurizer level and RCS pressure as shown in Figures 15.6-58 and 15.6-59. It was assumed that the time required for the operator to identify that the ruptured steam generator atmospheric steam dump valve is open and to locally close the associated block valve is 10.0 minutes. At 1620 seconds the depressurization of the ruptured steam generator was terminated and the ruptured steam generator pressure begins to increase as shown in Figure 15.6-60.

2. Cool Down the RCS to establish Subcooling Margin

   After the block valve for the ruptured steam generator atmospheric steam dump valve was closed, there is a 24 minute operator action time imposed prior to initiation of cooldown. Thus, the RCS cooldown was initiated at 1764 seconds. By this time, the ruptured steam generator pressure has increased to the intact steam generator pressure and stabilized at that value. The RCS cooldown target temperature is determined based on the ruptured steam generator pressure at that time. Since offsite power was lost, the RCS was cooled by dumping steam to the atmosphere using the intact steam generator atmospheric steam dump valves. The cooldown was continued until RCS subcooling at the ruptured steam generator pressure is 20°F plus an allowance for instrument uncertainty. Because the ruptured steam generator pressure has increased to the intact steam generator pressure prior to performing the cooldown, the associated temperature the RCS must be cooled to is not as low, which has the net effect of reducing the time required for cooldown. The cooldown was initiated at 1764 seconds and was completed at 2968 seconds.

The reduction in the intact steam generator pressures required to accomplish the cooldown is shown in Figure 15.6-60, and the effect of the cooldown on the RCS temperature is shown in Figure 15.6-61. The pressurizer level and RCS pressure also decrease during this cooldown process due to shrinkage of the reactor coolant, as shown in Figures 15.6-58 and 15.6-59.

3. Depressurize RCS to Restore Inventory

After the RCS cooldown, a 4.0 minute operator action time is included prior to the RCS depressurization. The RCS is depressurized to assure adequate coolant inventory prior to terminating SI flow. With the RCPs stopped, normal pressurizer spray is not available and thus the RCS is depressurized by opening a pressurizer PORV. The depressurization is initiated at 3210 seconds and continued until the criteria in the Emergency Operating Procedures are satisfied. The RCS depressurization reduces the break flow as shown in Figure 15.6-64, and increases SI flow to refill the pressurizer as shown in Figure 15.6-58.

4. Terminate SI to Stop Primary to Secondary Leakage

The previous actions establish adequate RCS subcooling, a secondary side heat sink, and sufficient reactor coolant inventory to ensure that SI flow is no longer needed. When these actions have been completed, the SI flow must be stopped to prevent repressurization of the RCS and to terminate primary to secondary leakage. The SI flow is terminated at this time if the SI termination criteria in the Emergency Operating Procedures are satisfied.

After depressurization was completed, an operator action time of 3.0 minutes was assumed prior to initiation of SI termination. Since the SI termination requirements are satisfied, SI termination actions were performed at this time by closing off the SI flow path. After SI termination, the RCS pressure begins to decrease as shown in Figure 15.6-59. The intact steam generator atmospheric steam dump valves are also opened to dump steam to maintain the prescribed RCS temperature to ensure that subcooling is maintained. When the atmospheric steam dump valves are opened, the increased energy transfer from primary to secondary also aids in the depressurization of the RCS to the ruptured steam generator pressure. The differential pressure between the RCS and the ruptured steam generator is shown in Figure 15.6-63. Figure 15.6-64 shows that the primary to secondary leakage continues after the SI flow is stopped until the RCS and ruptured steam generator pressures equalize.

The ruptured steam generator water volume is shown in Figure 15.6-65. The water volume in the ruptured steam generator when the break flow is terminated is less than the volume for the margin to overfill case in the licensing or Engineering Report and is significantly less than the total steam generator volume of 5730 ft^3. The mass of water in the ruptured steam generator is also shown as a function of time in Figure 15.6-66.

Mass Releases

The mass releases were determined for use in evaluating the exclusion area boundary and low population zone radiation exposure. The steam releases from the ruptured and intact steam generators, the feedwater flows to the ruptured and intact steam generators, and primary to secondary break flow into the ruptured steam generator were determined for the period from accident initiation until 2 hours after the accident and from 2 to 8 hours after the accident. The releases for 0-2 hours were used to calculate the radiation doses at the exclusion area boundary for a 2 hour exposure, and the releases for 0-8 hours were used to calculate the radiation doses at the low population zone for the duration of the accident.

The operator actions for the SGTR recovery up to the termination of primary to secondary leakage were simulated in the LOFTTR2 analysis. Thus, the steam releases from the ruptured and intact steam generators, the feedwater flows to the ruptured and intact steam generators, and the primary to secondary leakage into the ruptured steam generator were determined from the LOFTTR2 results for the period from the initiation of the accident until the leakage is terminated.

Following the termination of leakage, it was assumed that the actions are taken to cool down the plant to cold shutdown conditions. The atmospheric steam dump valves for the intact steam generators were assumed to be used to cool down the RCS to the RHR system operating temperature of 350°F, at the maximum allowable cooldown rate of 100°F/hr. The steam releases and the feedwater flows for the intact steam generators for the period from leakage termination until 2 hours were determined from a mass and energy balance using the calculated RCS and intact steam generator conditions at the time of leakage termination and at 2 hours. The RCS cooldown was assumed to be continued after 2 hours until the RHR system in-service temperature of 350°F is reached. Depressurization of the ruptured steam generator was then assumed to be performed to the RHR in-service pressure of 375 psia via steam release from the ruptured steam generator atmospheric steam dump valve. The RCS pressure was also assumed to be reduced concurrently as the ruptured steam generator is depressurized. It was assumed that the continuation of the RCS cooldown and depressurization to RHR operating conditions are completed within 8 hours after the accident since there is ample time to complete the operations during this time period. The steam releases and feedwater flows from 2 to 8 hours were determined for the intact and ruptured steam generators from a mass and energy balance using the conditions at 2 hours and at the RHR system in-service conditions.

After 8 hours, it was assumed that further plant cooldown to cold shutdown as well as long-term cooling is provided by the RHR system. Therefore, the steam releases to the atmosphere were terminated after RHR in-service conditions were assumed to be reached at 8 hours.

For the time period from initiation of the accident until leakage termination, the releases were determined from the LOFTTR2 results for the time prior to reactor trip and following reactor trip. Since the condenser is in service until reactor trip, any radioactivity released to the atmosphere prior to reactor trip would be through the air ejector discharge. After reactor trip, the releases to the atmosphere were assumed to be via the steam generator atmospheric steam dump valves. The mass release rates to the atmosphere from the LOFTTR2 analysis are presented in Figures 15.6-67 and 15.6-68 for the ruptured and intact steam generators, respectively, for the time period until leakage termination. The mass releases calculated from the time of leakage termination until 2 hours and from 2-8 hours were also assumed to be released to the atmosphere via the steam generator atmospheric steam dump valves. The mass releases for the SGTR event for the 0-2 hour and 2-8 hour time intervals considered are presented in Table 15.6-5a.

15.6.3.4 Radiological Consequences

Computer program PERC2 is used to calculate the control room and site boundary doses due to airborne radioactivity releases following a steam generator tube rupture (SGTR) at BVPS-2.

The dose assessments follow the guidance provided in Regulatory Guide 1.183. Table 15.6-5b lists the key parameters utilized to develop the radiological consequences following this accident.

The SGTR results in a reactor trip and a simultaneous loss of offsite power at 225 seconds after the event. Due to the tube rupture the primary coolant with elevated iodine concentrations (pre-accident or concurrent iodine spike) flows into the faulted steam generator and the associated activities are released to the environment via secondary side steam releases. Before the reactor trip, the activities are released from the air ejector. After the reactor trip the steam release is via the MSSVs/ADVs.

The spiking primary coolant activities leaked into the intact steam generator at the maximum allowable primary-to-secondary leakage value are also released to the environment via secondary steam releases.

The environmental releases for a postulated SGTR are combined with the atmospheric dispersion values presented in Tables 15.0-11 and 15.0-15 to determine the site boundary and control room doses given in Table 15.0-12 and 15.0-13, respectively. The most limiting atmospheric dispersion factors for each of the release points relative to the two CR intakes (identified for purposes of assessment as the BVPS-2 MSSVs/ADVs to the BVPS-2 CR intake, and the BVPS-2 air ejector to the BVPS-2 Intake) are selected to determine a bounding control room dose.

Since there is no fuel damage associated with this event, the radiological consequences are determined assuming each of the following occurrences which results in an increased inventory of iodine in the primary coolant:

1. Pre-accident iodine spike, and

2. Accident-initiated concurrent iodine spike.

The pre-accident iodine spike is the result of a primary plant transient which will increase the primary system iodine concentrations to the levels discussed in Section 15.0.9.4 and shown in Table 15.0-9. The accident-initiated or concurrent iodine spike is modeled by assuming that the iodine release rates from the fuel rods into the primary coolant are 335 times the Technical Specification equilibrium release rates. The iodine release rates for the concurrent iodine spiking conditions are calculated for the SGTR as detailed in Section 15.0.9.4 and Tables 15.0.10 and 15.0-10a.

The initial secondary side liquid and steam activity is relatively small and its contribution to the total dose is small compared to that contributed by the rupture flow. However, the release of the secondary side liquid activity and the resultant doses are also included in this analysis. The initial secondary side iodine activity is assumed to be at the Technical Specification limit of 0.1 μCi/gm DE I-131.

<u>Ruptured SG Release</u>

A postulated SGTR will result in a large amount of primary coolant being released into the ruptured steam generator via the break location with a significant portion of it flashed to the steam space. The noble gases in the break flow and the iodine in the flashed flow are assumed immediately available for release from the steam generator without retention. The iodine in the non-flashed portion of the break flow mixes uniformly with the steam generator liquid mass and is released into the steam space in proportion to the steaming rate and partition factor. Before the reactor trip at 225 seconds, the activities in the steam are released to the environment from the main condenser air ejector. All steam noble gases and organic iodine are released directly to the environment. Only a portion of the elemental iodine carried with the steam is partitioned to the air ejector and released to the environment. The rest is partitioned to the condensate, returned to all three steam generators and assumed to be available for future steaming release. After the reactor trip, the break flow continues until the primary system is fully depressurized. No credit is taken for the condenser, since a LOOP is assumed to occur simultaneously with the reactor trip. The steam is released from the MSSVs/ADVs. The release from the faulted SG includes a short period release between 2 and 8 hrs when the faulted SG is manually depressurized in preparation for RHR operation.

Intact SG release

The activity release from the intact steam generator is due to normal primary-to-secondary leakage and steam release from the secondary side. The Primary-to-Secondary leak rate is assumed to be 150 gpd per SG. All of the iodine activity in the referenced leakage is assumed to mix uniformly with the steam generator liquid and released in proportion to the steaming rate and the partition factor. Before the reactor trip at 225 seconds, the steam is released from the main condenser air ejector. After the reactor trip, the steam is released from the MSSVs/ADVs. The reactor coolant noble gases that enter the intact steam generator are released directly to the environment without holdup. The steam release from the intact steam generator continues until initiation of shutdown cooling 8 hours after the accident.

Release of Initial SG Liquid Activity

The initial iodine inventory in the steam generator liquid is assumed to be at Technical Specification levels and is released to the environment, due to steam releases, via the condenser/air ejector before reactor trip, and via the MSSVs/ADVs after reactor trip.

EAB 2 hr Worst Case Window

AST methodology requires that the worst case dose to an individual located at any point on the boundary at the EAB, for any 2-hr period following the onset of the accident be reported as the EAB dose. The major source for the SGTR is the flashed portion of the RCS break flow which is terminated before T=2 hrs. Therefore, the worst 2-hr window dose for both the pre-accident and accident initiated spike case occurs during T=0 hr to T=2 hrs after the accident.

Accident Specific Control Room Model Assumptions

No credit is taken for initiation of the control room emergency ventilation system following a SGTR. Following termination of the environmental release, the control room is purged, at T=8 hrs, at a rate of 16,200 cfm, for a period of 30 mins. The critical control room parameters utilized in this model are summarized in Table 6.4-1a. Section 15.6.5.4 discusses the control room design as related to dose consequences under a sub-section titled "Control Room Habitability."

The methodology used in calculating the offsite doses is discussed in Appendix 15A. The radiological consequences for a SGTR do not exceed the dose limits provided in 10 CFR 50.67 as supplemented by SRP 15.0.1 and Regulatory Guide 1.183.

15.6.4 Spectrum of Boiling Water Reactor Steam System Piping Failures outside of Containment

Not applicable to BVPS-2.

15.6.5 Loss-of-Coolant Accidents Resulting from a Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary

15.6.5.1 Identification of Causes and Frequency Classification

A LOCA is the result of a pipe rupture of the RCPB (Section 5.2). For the analyses reported here, a major pipe break (large break) is defined as a rupture 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, in that it is not expected to occur during the lifetime of BVPS-2 but is postulated as a conservative design basis (Section 15.0.1).

A minor pipe break (small break), as considered here, is defined as a single or multiple aperture rupture of the RCPB with a total cross-sectional area less than 1.0 ft^2 in which the normally operating charging system flow is not sufficient to sustain pressurizer level and pressure. This is considered a Condition III event, in that it is an infrequent fault which may occur during the life of BVPS-2.

The acceptance criteria for the LOCA are described in 10 CFR 50.46 (Reference 3) as follows:

1. The calculated maximum fuel element cladding temperature shall not exceed 2200°F.

2. The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.

3. 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.

4. Calculated changes in core geometry shall be such that the core remains amenable to cooling.

5. After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

These criteria were established to provide significant margin in emergency core cooling system (ECCS) performance following a LOCA.

15.6.5.2 Major Reactor Coolant System Pipe Ruptures (Loss-Of-Coolant-Accident) (Best Estimate LOCA) (BELOCA)).

The analysis specified by 10 CFR 50.46 (Reference 3), "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Power Reactors," is presented in this section. The results of the Best-Estimate large break loss-of-coolant accident (LOCA) analysis are summarized in Table 15.6-8d, and show compliance with the acceptance criteria.

For the purpose of ECCS analyses, Westinghouse (W) defines a large break loss-of-coolant accident (LOCA) as a rupture 1.0 ft^2 or larger of the reactor coolant system piping including the double ended rupture of the largest pipe in the reactor coolant system or of any line connected to that system.

Should a major break occur, rapid depressurization of the Reactor Coolant System (RCS) to a pressure nearly equal to the containment pressure occurs in approximately 40 seconds, with a nearly complete loss of system inventory. Rapid voiding in the core shuts down reactor power. A safety injection system signal is actuated when the low pressurizer pressure setpoint is reached. These countermeasures will limit the consequences of the accident in two ways:

1. Borated water injection complements void formation in causing rapid reduction of power to a residual level corresponding to fission product decay heat. An average RCS/sump mixed boron concentration is calculated to ensure that the post-LOCA core remains subcritical. However, no credit is taken for the insertion of control rods to shut down the reactor in the large break analysis.

2. Injection of borated water provides heat transfer from the core and prevents excessive cladding temperatures.

Before the break occurs, the reactor is assumed to be in a full power equilibrium condition, i.e., the heat generated in the core is being removed through the steam generator secondary system. At the beginning of the blowdown phase, the entire RCS contains sub-cooled liquid which transfers heat from the core by forced convection with some fully developed nucleate boiling. During blowdown, heat from fission product decay, hot internals and the vessel, continues to be transferred to the reactor coolant. After the break develops, the time to departure from nucleate boiling is calculated. Thereafter, the core heat transfer is unstable, with both nucleate boiling and film boiling occurring. As the core becomes voided, both transition boiling and forced convection are considered as the dominant core heat transfer mechanisms. Heat transfer due to radiation is also considered.

The heat transfer between the RCS and the secondary system may be in either direction, depending on the relative temperatures. In the case of the large break LOCA, the primary pressure rapidly decreases below the secondary system pressure and the steam generators are an additional heat source. In the Beaver Valley Unit 2 Power Station Large Break LOCA analysis using the WCOBRA/TRAC methodology (Reference 46), the steam generator secondary is conservatively assumed to be isolated (main feedwater and steam line) at the initiation of the event to maximize the secondary side heat load.

15.6.5.2.1 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 cooling system cold leg pipe. The reactor core and internals together with the Emergency Core Cooling System (ECCS) are designed so that the reactor can be safely shut down and the essential heat transfer geometry of the core preserved following the accident. Long-term coolability is maintained.

When the RCS depressurizes to approximately 640 psia, the accumulators begin to inject borated water into the reactor coolant loops. Borated water from the accumulator in the broken loop is assumed to spill to containment and be unavailable for core cooling for breaks in the cold leg of the RCS. Flow from the accumulators in the intact loops may not reach the core during depressurization of the RCS due to the fluid dynamics present during the ECCS bypass period. ECCS bypass results from the momentum of the fluid flow up the downcomer due to a break in the cold leg, which entrains ECCS flow out toward the break. Bypass of the ECCS diminishes as mechanisms responsible for the bypassing are calculated to be no longer effective.

The blowdown phase of the transient ends when the liquid level in the lower plenum reaches its minimum. After the end of the blowdown, refill of the reactor vessel lower plenum begins. Refill is completed when emergency core cooling water has filled the lower plenum of the reactor vessel, which is bounded by the bottom of the active fuel region of the fuel rods (called bottom of core (BOC) recovery time).

The reflood phase of the transient is defined as the time period lasting from BOC recovery until the reactor vessel has been filled with water to the extent that the core temperature rise has been terminated. From the latter stage of blowdown and on into the beginning of reflood, the intact loop accumulator tanks rapidly discharge borated cooling water into the RCS. Although a portion injected prior to end of bypass is lost out the cold leg break, the accumulators eventually contribute to the filling of the reactor vessel downcomer. The downcomer water elevation head provides the driving force required for the reflooding of the reactor core. The high head safety injection (HHSI) pump aids in the filling of the downcomer and core and subsequently supply water to help maintain a full downcomer and complete the reflooding process. The low head safety injection (LHSI) also aids the reflooding process by providing water to the core.

Continued operation of the ECCS pumps supplies water during long-term cooling. Core temperatures have been reduced to long-term steady state levels associated with dissipation of residual heat generation. After the water level of the refueling water storage tank (RWST) reaches a minimum allowable value, coolant for long-term cooling of the core is obtained by switching from the injection mode to the sump recirculation mode of ECCS operation. Spilled borated water is drawn from the engineered safety features (ESF) containment sumps by two of the four Recirculation Spray System pumps and returned to the RCS cold legs. Figure 15.6-8a contains a schematic of the bounding sequence of events for the Beaver Valley Unit 2 Best-Estimate large break LOCA transient.

For the Best-Estimate large break LOCA analysis, one ECCS train, including one high head safety injection (HHSI) pump and one RHR (low-head) pump, starts and delivers flow through the injection lines. The accumulator and safety injection flows from the broken loop were assumed to spilled to containment. Both emergency diesel generators (EDGs) are assumed to start in the modeling of the containment spray pumps. Modeling full containment heat removal systems operation is required by Branch Technical Position CSB 6-1 (Reference 48) and is conservative for the large break LOCA.

To minimize delivery to the reactor, the HHSI and LHSI branch line chosen to spill is selected as the one with the minimum resistance.

15.6.5.2.2 Large Break LOCA Analytical Model

In 1988, as a result of the improved understanding of LOCA thermal-hydraulic phenomena gained by extensive research programs, the NRC staff amended the requirements of 10 CFR 50.46 and Appendix K, "ECCS Evaluation Models," so that a realistic evaluation model may be used to analyze the performance of the ECCS during a hypothetical LOCA (Reference 42). 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 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 limits. Further guidance for the use of best-estimate codes was provided in Regulatory Guide 1.157 (Reference 43).

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 44). 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 PWR plants based on the revised 10 CFR 50.46 rules was developed by Westinghouse with the support of EPRI and Consolidated Edison and was approved by the NRC (Reference 45). The methodology is documented in WCAP-12945, "Code Qualification Document (CQD) for Best Estimate LOCA Analysis" (Reference 46).

The thermal-hydraulic computer code which was reviewed and approved for the calculation of fluid and thermal conditions in the PWR during a large break LOCA is WCOBRA/TRAC Version MOD7A (Reference 46).

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:

- Ability to model transient three-dimensional flows in different geometries inside the vessel

- Ability to model thermal and mechanical non-equilibrium between phases

- Ability to mechanistically represent interfacial heat, mass, and momentum transfer in different flow regimes

- Ability to represent important reactor components such as fuel rods, steam generators, reactor coolant pumps, etc.

The reactor vessel is modeled with the three-dimensional, three-field fluid model, while the loop, major loop components, and safety injection points are modeled with the one-dimensional fluid model.

The basic building block for the vessel is the channel, a vertical stack of single mesh cells. Several channels can be connected together by gaps to model a region of the reactor vessel. Regions that occupy the same level form a section of the vessel. Vessel sections are connected axially to complete the vessel mesh by specifying channel connections between sections. Heat transfer surfaces and solid structures that interact significantly with the fluid can be modeled with rods and unheated conductors. The fuel parameters are generated using the Westinghouse fuel performance code (PAD 4.0, Reference 41).

One-dimensional components are connected to the vessel. Special purpose components exist to model specific components such as the steam generator and pump.

A typical calculation using WCOBRA/TRAC begins with the establishment of a steady-state initial condition with all loops intact. The input parameters and initial conditions for this steady-state calculation are discussed in the next section.

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 follows 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 12) and mass and energy releases from the WCOBRA/TRAC calculation. The parameters used in the containment analysis to determine this pressure curve are presented in Tables 6.2-50 through 6.2-53.

The methods used in the application of WCOBRA/TRAC to the large break LOCA are described in References 45 through 46. 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 (Reference 47). The final step of the best-estimate methodology is to combine all the uncertainties related to the code and plant parameters and estimate the PCT at the 95th percentile ($PCT^{95\%}$). The steps taken to derive the PCT uncertainty estimate are summarized below:

1. Plant Model Development

   In this step, a WCOBRA/TRAC model of the Beaver Valley Unit 2 Power Station (BVPS-2) is developed. A high level of noding detail is used, in order to provide an accurate simulation of the transient. However, specific guidelines are followed to assure 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 range of the plant operating conditions 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." 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. The results of these calculations for BVPS-2 are discussed in Section 5 of Reference 47. The most limiting input conditions, based on these confirmatory runs, are then combined into a single transient, which is then called the "reference transient."

3. PWR Sensitivity Calculations

A series of PWR transients are performed in which the initial fluid conditions and boundary conditions are ranged around the nominal conditions used in the reference transient. The results of these calculations for BVPS-2 form the basis for the determination of the initial condition bias and uncertainty discussed in Section 6 of Reference 47.

Next, a series of transients are performed which vary the power distribution, taking into account all possible power distributions during normal plant operation. The results of these calculations for BVPS-2 form the basis for the determination of the power distribution bias and uncertainty (response surface) discussed in Section 7 of Reference 47.

Finally, a series of transients are performed which vary parameters that affect the overall system response ("global" parameters) and local fuel rod response ("local" parameters). The results of these calculations for BVPS-2 form the basis for the determination of the model bias and uncertainty (response surface) discussed in Section 8 of Reference 47.

4. Response Surface Calculations

The results from the power distribution and global model WCOBRA/TRAC runs performed in Step 3 are fit by regression analyses into equations known as response surfaces. The results of the initial conditions run matrix are used to generate a PCT uncertainty distribution.

5. Uncertainty Evaluation

The total PCT uncertainty from the initial conditions, power distribution, and model calculations is derived using the approved methodology (Reference 46). The uncertainty calculations assume certain plant operating ranges which may be varied depending on the results obtained. These uncertainties are then combined to determine the initial estimate of the total PCT uncertainty distribution for the guillotine and limiting split breaks. The results of these initial estimates of the total PCT uncertainty are compared to determine the limiting break type. If the split break is limiting, an additional set of split transients are performed which vary overall system response ("global" parameters) and local fuel rod response ("local" parameters). The results of these calculations form the basis for the determination of the model bias and uncertainty discussed in Section 9 of Reference 47. Finally, an additional series of runs is made to quantify the bias and uncertainty due to assuming that the above three uncertainty categories are independent. The final PCT uncertainty distribution is then calculated for the limiting break type, and the 95th percentile PCT ($PCT^{95\%}$) is determined, as described later under Uncertainty Evaluation.

6. 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 established in step 2, or may be narrower for some parameters to gain additional margin.

There are three major uncertainty categories or elements:

- Initial condition bias and uncertainty
- Power distribution bias and uncertainty
- Model bias and uncertainty

Conceptually, these elements may be assumed to affect the reference transient PCT as shown below

$$PCT_i = PCT_{REF,i} + \Delta PCT_{IC,i} + \Delta PCT_{PD,i} + \Delta PCT_{MOD,i} \quad (15.6.5.2\text{-}1)$$

where,

$PCT_{REF,i}$ = Reference transient PCT: The reference transient PCT is calculated using WCOBRA/TRAC at the nominal conditions identified in Table 15.6-8a, for the blowdown, first reflood and second reflood periods.

$\Delta PCT_{IC,i}$ = Initial condition bias and uncertainty: This bias is the difference between the reference transient PCT, which assumes several nominal or average initial conditions, and the average PCT taking into account all possible values of the initial conditions. This bias takes into account plant variations which have a relatively small effect on PCT. The elements which make up this bias and its uncertainty are plant-specific.

$\Delta PCT_{PD,i}$ = Power distribution bias and uncertainty: This bias is the difference between the reference transient PCT, which assumes a nominal power distribution, and the average PCT taking into account all possible power distributions during normal plant operation. Elements which contribute to the uncertainty of this bias are calculational uncertainties, and variations due to transient operation of the reactor.

$\Delta PCT_{MOD,i}$ = Model bias and uncertainty: This component accounts for uncertainties in the ability of the WCOBRA/TRAC code to accurately predict important phenomena which affect the overall system response ("global" parameters) and the local fuel rod response ("local" parameters). The code and model bias is the difference between the reference transient PCT, which assumes nominal values for the global and local parameters, and the average PCT taking into account all possible values of global and local parameters.

The separability of the bias and uncertainty components in the manner described above is an approximation, since the parameters in each element may be affected by parameters in other elements. The bias and uncertainty associated with this assumption is quantified as part of the overall uncertainty methodology and included in the final estimates of $PCT^{95\%}$.

15.6.5.2.3 Large Break LOCA Analysis Results

A series of WCOBRA/TRAC calculations were performed using the Beaver Valley Unit 2 Power Station (BVPS-2) input model, to determine the effect of variations in several key LOCA parameters on peak cladding temperature (PCT). From these studies, an assessment was made of the parameters that had a significant effect as will be described in the following sections.

15.6.5.2.3.1 LOCA Reference Transient Description

The plant-specific analysis performed for the Beaver Valley Unit 2 Nuclear Station indicated that the double-ended cold leg guillotine (DECLG) break is more limiting than the split break. The plant conditions used in the reference transient are listed in Table 15.6-8a. The following is a description of the final reference transient.

The LOCA transient can be conveniently divided into a number of time periods in which specific phenomena are occurring. 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, early reflood, late reflood, and long term cooling phases. The important phenomena occurring during each of these phases are discussed for the reference transient. The results are shown in Figures 15.6-8b through 15.6-8o.

Critical Heat Flux (CHF) Phase (0 - 2 seconds)

Immediately following the cold leg rupture, the break discharge rate is subcooled and high, the core flow reverses, the fuel rods go through departure from nucleate boiling (DNB) and the cladding rapidly heats up while core power shuts down. Figure 15.6-8b shows the maximum cladding temperature in the core, as a function of time. The hot water in the core and upper plenum flashes to steam during this period. This phase is terminated when the water in the lower plenum and downcomer begins to flash. The mixture swells and the intact loop pumps, still rotating in single-phase liquid, push this two-phase mixture into the core.

Upward Core Flow Phase (2 - 6 seconds)

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, and the break discharge rate is low because the fluid is saturated at the break. Figures 15.6-8c and 15.6-8d show the break flowrate from the vessel and loop sides of the break. This phase ends as lower plenum mass is depleted, the loops become two-phase, and the pump head degrades. If pumps are highly degraded or the break flow is large, the cooling effect due to upward flow may not be significant. Figure 15.6-8e shows the void fraction for one intact loop pump and the broken loop pump. The intact loop pump remains in single-phase liquid flow for several seconds, while the broken loop pump is in two-phase and steam flow soon after the break.

Downward Core Flow Phase (6 - 25 seconds)

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. Figures 15.6-8f and 15.6-8g show the vapor flow at the mid-core of channels 11 and 13. While liquid and entrained liquid flows also provide core cooling, the vapor flow entering the core best illustrates this phase of core cooling. This period is enhanced by flow from the upper head. As the system pressure continues to fall, the break flow and consequently the core flow, are reduced. The core begins to heat up as the system reaches containment pressure and the vessel begins to fill with Emergency Core Cooling System (ECCS) water.

Refill Phase (25 - 35 seconds)

The core experiences a nearly adiabatic heatup as the lower plenum fills with ECCS water. Figure 15.6-8h shows the lower plenum liquid level. This phase ends when the ECCS water enters the core and entrainment begins, with a resulting improvement in heat transfer. Figures 15.6-8i and 15.6-8j show the liquid flows from the accumulator and the safety injection from an intact loop (Loop 1).

Early Reflood Phase (35 - 50 seconds)

The accumulators begin to empty and nitrogen enters the system. This forces water into the core which then boils as the lower core region begins to quench, causing repressurization. The repressurization is best illustrated by the reduction in pumped SI flow (~40 sec). During this time, core cooling may be increased. The system then settles into a gravity driven reflood which exhibits lower core heat transfer. Figures 15.6-8k and 15.6-8l show the core and downcomer liquid levels. Figure 15.6-8m shows the vessel fluid mass. As the quench front progresses further into the core, the peak cladding temperature (PCT) location moves higher in the top core region. Figure 15.6-8n shows the movement of the PCT location over the duration of the transient. As the vessel continues to fill, the PCT location is cooled and the heatup PCT transient is terminated.

Late Reflood Phase (50 - 200 seconds)

The late reflood phase is characterized by boiling in the downcomer. The mixing of ECC water with hot water and steam from the core, in addition to the continued heat transfer from the hot vessel metal, reduces the subcooling of ECC in the lower plenum and downcomer. For Beaver Valley Unit 2, a significant late reflood peak does not exist (Figure 15.6-8b). Therefore, boiling in the downcomer does not impact the transient to a large degree. The effect of the reflooding transient on the cladding temperature is provided in Figures 15.6-8b and 15.6-8o. The reference transient resulted in a blowdown PCT of 1566°F, a first reflood PCT of 1616°F and a late second reflood PCT of 1753°F.

Long Term Core Cooling

At the end of the WCOBRA/TRAC calculation, the core and downcomer levels are increasing as the pumped safety injection flow exceeds the break flow. The core and downcomer levels would be expected to continue to rise, until the downcomer mixture level approaches the loop elevation. At that point, the break flow would increase, until it roughly matches the injection flowrate. The core would continue to be cooled until the entire core is eventually quenched.

15.6.5.2.3.2 Confirmatory Sensitivity Studies

A number of sensitivity calculations were carried out to investigate the effect of the key LOCA parameters, and to develop the required data for the uncertainty evaluation. In the sensitivity studies performed, LOCA parameters were varied one at a time. For each sensitivity study, a comparison between the base case and the sensitivity case transient results was made.

The results of the sensitivity studies are summarized in Tables 15.6-8b and 15.6-8c. A full report on the results for all sensitivity study results is included in Sections 5 and 8 of Reference 47. The results of these analyses lead to the following conclusions:

1. Modeling maximum steam generator tube plugging (22%) results in a higher PCT than minimum steam generator tube plugging (0%).

2. Modeling offsite power available results in a higher PCT than loss-of-offsite power (LOOP).

3. Modeling the maximum value of vessel average temperature (T_{avg} = 580°F) results in a higher PCT than the minimum value of vessel average temperature (T_{avg} = 566.2°F).

4. Modeling the minimum power fraction (P_{LOW} = 0.2) in the low power/periphery channel of the core results in a higher PCT than the maximum power fraction (P_{LOW} = 0.6).

5. The limiting break type is a double ended cold leg guillotine (DECLG) break. This transient then becomes the reference transient for the determination of uncertainties.

15.6.5.2.3.3 Initial Conditions Sensitivity Studies

Several calculations were performed to evaluate the effect of change in the initial conditions on the calculated LOCA transient. These calculations analyzed key initial plant conditions over their expected range of operation. These studies included effects of ranging RCS conditions (pressure and temperature), safety injection temperature, and accumulator conditions (pressure, temperature, volume, and line resistance). The results of these studies are presented in Section 6 of Reference 47.

The calculated results were used to develop initial condition uncertainty distributions for the blowdown and reflood peaks. These distributions are then used in the uncertainty evaluation to predict the PCT uncertainty component resulting from initial conditions uncertainty ($\Delta PCT_{IC,i}$).

15.6.5.2.3.4 Power Distribution Sensitivity Studies

Several calculations were performed to evaluate the effect of power distribution on the calculated LOCA transient. The power distribution attributes which were analyzed are the peak linear heat rate relative to the core average, the maximum relative rod power, the relative power in the bottom third of the core (P_{BOT}), and the relative power in the middle third of the core (P_{MID}). The choice of these variables and their ranges are based on the expected range of plant operation. The power distribution parameters used for the reference transient are biased to yield a relatively high PCT. The reference transient uses the maximum $F_{\Delta H}$, a skewed to the top power distribution, and a F_Q at the midpoint of the sample range. A run matrix was developed in order to vary the power distribution attributes singly and in combination. The calculated results are presented in Section 7 of Reference 47. The sensitivity results indicated that power distributions with peak powers shifted towards the top of the core produced higher PCTs.

The calculated results were used to develop response surfaces, as described in Step 4 of Section 15.6.5.2.2, which could be used to predict the change in PCT for various changes in the power distributions for the blowdown and reflood peaks. These were then used in the uncertainty evaluation, to predict the PCT uncertainty component resulting from uncertainties in power distribution parameters, ($\Delta PCT_{PD,i}$).

15.6.5.2.3.5 Global Model Sensitivity Studies

Several calculations were performed to evaluate the effect of broken loop resistance, break discharge coefficient, and condensation rate on the PCT for the guillotine break. As in the power distribution study, these parameters were varied singly and in combination in order to obtain a data base which could be used for response surface generation. The run matrix and ranges of the break flow parameters are described in Reference 46 (generic methodology). The limiting split break was also identified using the methodology described in Reference 46 (generic methodology). The plant specific calculated results are presented in Section 8 of Reference 47. The results of these studies indicated that the double ended cold leg guillotine break resulted in the highest PCT.

The calculated results were used to develop response surfaces as described in Section 15.6.5.2.2, which could be used to predict the change in PCT for various changes in the flow conditions. These were then used in the uncertainty evaluation to predict the PCT uncertainty component resulting from uncertainties in global model parameters ($\Delta PCT_{MOD,i}$).

15.6.5.2.3.6 Uncertainty Evaluation and Results

The PCT equation was presented in Section 15.6.5.2.2. Each element of uncertainty is initially considered to be independent of the other. Each bias component is considered a random variable, whose uncertainty and distribution is obtained directly, or is obtained from the uncertainty of the parameters of which the bias is a function. For example, $\Delta PCT_{PD,i}$ is a function of F_Q, $F_{\Delta H}$, P_{BOT}, and P_{MID}. Its distribution is obtained by sampling the plant F_Q, $F_{\Delta H}$, P_{BOT}, and P_{MID} distributions and using a response surface to calculate $\Delta PCT_{PD,i}$. Since ΔPCT_i is the sum of these biases, it also becomes a random variable. Separate initial PCT frequency distributions are constructed as follows for the guillotine break and the limiting split break size:

1. Generate a random value of each ΔPCT element.

2. Calculate the resulting PCT using Equation 15.6.5.2-1.

3. Repeat the process many times to generate a histogram of PCTs.

For Beaver Valley Unit 2, the results of this assessment showed the split break to be non-limiting.

A final verification step is performed in which additional calculations (known as "superposition" calculations) are made with WCOBRA/TRAC, simultaneously varying several parameters which were previously assumed independent (for example, power distributions and models). Predictions using Equation 15.6.5.2-1 are compared to this data, and additional biases and uncertainties are applied.

The estimate of the PCT at 95 percent probability is determined by finding that PCT below which 95 percent of the calculated PCTs reside. This estimate is the licensing basis PCT, under the revised ECCS rule.

The results for the Beaver Valley Unit 2 Power Station are given in Table 15.6-8d, which shows the second reflood 95th percentile PCT ($PCT^{95\%}$) of 1976°F. As expected, the difference between the 95 percent value and the average value increases with increasing time, as more parameter uncertainties come into play.

15.6.5.2.3.7 Evaluations

The Beaver Valley Unit 2 Power Station will be transitioning from the V5H fuel without IFM to the RFA fuel with IFMs. An additional calculation was performed with the Reference Transient conditions to determine the effects of the mixed core. The calculation modeled a fresh RFA fuel assembly in the Hot Assembly channel surrounded by burned V5H fuel in the core OH/SC/FSM average channel and LP channel. The RFA fuel is also modeled in the average channel under the GT. A minimum burnup of 10,000 MWD/MTU was assumed for all of the V5H assemblies. The analysis results show a 6°F, and 15°F in PCT penalty when compared to the Reference Transient first and second reflood PCTs, respectively.

In addition, two additional calculations were performed to assess IFBA fuel and ZIRC-4 clad fuel. The base analysis discussed in Sections 15.6.5.2.3.1 to 15.6.5.2.3.6 is for non-IFBA and ZIRLO™ clad fuel. An analysis of IFBA fuel and ZIRC-4 clad fuel was performed independently, utilizing the HOTSPOT code and the high PCT case identified in Section 10.2 of Reference 47. The analysis results indicated that IFBA fuel and ZIRC-4 clad fuel are bounded by non-IFBA fuel with ZIRLO™ clad fuel.

15.6.5.2.4 Large Break LOCA Conclusions

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 for the Beaver Valley Power Station Unit 2 is as follows:

1) There is a high level of probability that the peak cladding temperature (PCT) shall not exceed 2200°F. The results presented in Table 15.6-8d indicate that this regulatory limit has been met with a reflood $PCT^{95\%}$ of 1976°F.

2) The maximum calculated local oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation. The approved Best-Estimate LOCA methodology assesses this requirement using a plant-specific transient which has a PCT in excess of the estimated 95 percentile PCT ($PCT^{95\%}$). Based on this conservative calculation, a maximum local oxidation of 6.7 percent is calculated, which meets the regulatory limit.

3) 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 were to react. The total amount of hydrogen generated, based on this conservative assessment, is 0.0089 times the maximum theoretical amount, which meets the regulatory limit.

4) Calculated changes in core geometry shall be such that the core remains amenable to cooling. This requirement is met by demonstrating that the PCT does not exceed 2200°F, the maximum local oxidation does not exceed 17%, and the seismic and LOCA forces are not sufficient to distort the fuel assemblies to the extent that the core cannot be cooled. The BE methodology (Reference 46) specifies that the effects of LOCA and seismic loads on core geometry do not need to be considered unless grid crush extends to in-board assemblies. Fuel assembly structural analyses performed for Beaver Valley Unit 2 indicate that this condition does not occur. Therefore, this regulatory limit is met.

5) After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long lived radioactivity remaining in the core. The conditions at the end of the WCOBRA/TRAC calculations indicates that the transition to long term cooling is underway even before the entire core is quenched.

15.6.5.2.5 SER Requirements

The generic SER requirements for three- and four-loop plants (Reference 45) have been met for this BVPS-2 analysis.

15.6.5.2.6 Plant Operating Range

The expected PCT and its uncertainty developed above is valid for a range of plant operating conditions. In contrast to current Appendix K calculations, many parameters in the base case calculation are at nominal values. The range of variation of the operating parameters has been accounted for in the estimated PCT uncertainty. Table 15.6-8e summarizes the operating ranges for the Beaver Valley Unit 2 Power Station. If operation is maintained within these ranges, the LOCA analysis developed in Reference 47 is considered to be valid.

15.6.5.2.7 Post Analysis of Record Evaluations

In addition to the analysis presented in this section, evaluation and assessments may be performed as needed to address computer code errors and emergent issues, or to support plant changes. The issues or changes are evaluated, and the impact on the Peak Cladding Temperature (PCT) is determined. The resultant increase or decrease in PCT is applied to the analysis or record PCT. The PCTs, including all penalties and benefits, are presented in Table 15.6-9 for the large break LOCA. The resultant PCT is demonstrated to be less than the 10 CFR 50.46(b) requirement of 2200°F.

In addition, 10 CFR 50.46 requires that licensees assess and report the effect of changes to or error in the evaluation model used in the LOCA analyses. These reports constitute addenda to the analysis of record provided in the UFSAR until the overall changes become significant as defined by 10 CFR 50.46. If the assessed changes or errors in the evaluation model result in significant changes in calculated PCT, a schedule for formal reanalysis or other action as needed to show compliance will be addressed in the report to the NRC.

Finally, the criteria of 10 CFR 50.46 require that holders and users of the evaluation models establish a number of definitions and processes for assessing changes in the models or their use. Westinghouse, in consultation with the Westinghouse Owner's Group, developed an approach for compliance with the reporting requirements. This approach has been documented in WCAP-13451, *Westinghouse Methodology for Implementation of 10 CFR 50.46 Reporting*. Beaver Valley provides the NRC with annual and 30-day reports, as applicable, for the Beaver Valley Unit 2 Power Station. Beaver Valley intends to provide future reports required by 10 CFR 50.46 consistent with the approach described in WCAP-13451.

15.6.5.3 Small Break Loss-of-Coolant Accident

15.6.5.3.1 Sequence of Events and Systems Operations

Should a break smaller than 1.0 ft^2 occur, depressurization of the RCS causes fluid to flow to the RCS from the pressurizer, resulting in a pressure and level decrease in the pressurizer. Reactor trip occurs when the pressurizer low pressure trip setpoint is reached. The Safety Injection System (SIS) is actuated when the appropriate setpoint is reached. The consequences of the accident are limited in two ways:

1. Reactor trip and borated water injection complement void formation in causing rapid reduction of nuclear power to a residual level corresponding to the fission product decay heat. However, no credit is taken in the Small Break LOCA analysis for boron content in the injection water.

2. Injection of borated water provides for heat transfer from the core and prevents excessive cladding temperatures.

The time sequence of events for the small break LOCA analysis is shown in Table 15.6-1c.

Before the break occurs, Beaver Valley Power Station Unit-2 (BVPS-2) is in an equilibrium condition, i.e., the heat generated in the core is being removed via the Secondary System. During blowdown, heat from radioactive decay, hot internals and the vessel continues to be transferred to the reactor coolant. At the beginning of the blowdown phase, the entire RCS contains subcooled liquid which transfers heat from the core by forced convection with some fully developed nucleate boiling. Thereafter, the core heat transfer is based on local conditions with transition boiling and forced convection to steam as the major heat transfer mechanisms.

The heat transfer between the RCS and the Secondary System may be in either direction, depending on the relative temperatures. In the case of continued heat addition to the Secondary System, secondary system pressure increases and steam release through the Main Steam Safety Valves may occur. Makeup water to the secondary side is automatically provided by the auxiliary feedwater system. The safety injection signal actuates a feedwater isolation signal which isolates normal feedwater flow by closing the main feedwater line isolation valves, the feedwater control and bypass valves, and also initiates auxiliary feedwater flow by starting the auxiliary feedwater pumps. The secondary flow aids in the reduction of RCS pressure.

When the RCS depressurizes to 625 psia, the accumulators begin to inject borated water into the reactor coolant loops. Since the LOOP is assumed coincident with reactor trip signal, the reactor coolant pumps are assumed to trip at the initiation of the accident. The effects of pump coastdown are included in the blowdown analysis.

Following this gradual blowdown phase of the transient, a period of core recovery is followed by long-term recirculation.

15.6.5.3.2 Small Break LOCA Thermal Analysis

The requirements of an acceptable ECCS evaluation model are presented in Appendix K of 10 CFR 50 (Federal Register 1988).

15.6.5.3.2.1 Small Break LOCA Evaluation Model

For small breaks (less than 1.0 ft^2) the NOTRUMP computer code (Reference 23, 24 and 39) is employed to calculate the transient depressurization of the Reactor Coolant System as well as to describe the mass and energy of the fluid flow through the break. The NOTRUMP computer code is a state-of-the-art one-dimensional general network code incorporating a number of advanced features. Among these are calculation of thermal non-equilibrium in all fluid volumes, flow regime-dependent drift flux calculations with counter-current flooding limitations, mixture level tracking logic in multiple-stacked fluid nodes and regime-dependent heat transfer correlations. The NOTRUMP small break LOCA emergency core cooling system (ECCS) evaluation model (Reference 24) was developed to determine the RCS response to design basis small break LOCAs, and to address NRC concerns expressed in Reference 25.

In NOTRUMP, the reactor coolant system model is nodalized into volumes interconnected by flowpaths. The broken loop is modeled explicitly, while the intact loops are lumped into a second loop. Transient behavior of the system is determined from the governing conservation equations of mass, energy, and momentum. The multinode capability of the program enables explicit, detailed spatial representation of various system components which, among other capabilities, enables a proper calculation of the behavior of the loop seal during a loss-of-coolant accident. The reactor core is represented as heated control volumes with associated phase separation models to permit transient mixture height calculations. Detailed descriptions of the NOTRUMP code and the evaluation model are provided in References 23, 24, and 39.

Safety Injection flow rate to the Reactor Coolant System as a function of system pressure is used as part of the input (see Figures 15.6-42a and 15.6-42b). The Safety Injection System (SIS) was assumed to be delivering to the RCS 27 seconds after the generation of a safety injection signal. The flow from one high head safety injection pump was assumed.

Peak cladding temperature calculations are performed with the LOCTA- IV code (Reference 26) using the NOTRUMP calculated core pressure, fuel rod power history, uncovered core steam flow and mixture heights as boundary conditions. A schematic representation of the computer code interface is given on Figure 15.6-40. Figure 15.6-41 depicts the hot rod axial power shape used to perform the small break LOCA analysis. This shape was chosen because it represents a distribution with power concentrated in the upper regions of the core (the axial offset is +13%). This power shape is skewed to the top of the core with the peak local power occurring at the 9.5-foot core elevation. Such a distribution is limiting for small-break LOCAs because of the core uncovery process for small breaks. As the core uncovers, the cladding in the upper elevation of the core heats up and is sensitive to the local power at that elevation. The cladding temperatures at the lower elevations of the core, below the two-phase mixture height, remains low. The peak cladding temperature occurs above 9.5 feet.

NOTRUMP evaluation model analyses are performed assuming loss of offsite power coincident with reactor trip, and a limiting single active failure (i.e., loss of one ECCS train on a failure to start of one diesel generator). Diesel generator failure is presumed to render inoperable one motor driven auxiliary feedwater pump which results in one motor driven auxiliary feedwater pump being credited in the analyses. The diesel generator failure does not render the turbine driven auxiliary feedwater pump inoperable, but it was conservatively not credited in the analysis.

If the single failure postulated is loss of the turbine driven auxiliary feedwater pump during a small break LOCA, the results are bounded by the NOTRUMP evaluation model single failure described above. The two motor driven auxiliary feedwater pumps and two high head safety injection pumps would be credited if this single failure were postulated.

The small break analysis was performed with the Westinghouse ECCS Small Break Evaluation Model (References 23, 24 and 39). The model was approved for this use by the Nuclear Regulatory Commission in May, 1985.

15.6.5.3.2.2 Input Parameters and Initial Conditions

Table 15.6-10a lists important input parameters and initial conditions used in the small break analysis.

The analysis presented in this section was performed with a reactor vessel upper head fluid temperature equal to the RCS hot leg temperature.

The requirements of Appendix K regarding specific model features were met by selecting models which provide a significant overall conservatism in the analysis. The assumptions made pertain to the conditions of the reactor and associated safety system equipment at the time that the LOCA occurs and include such items as the core peaking factors, and the performance of the ECCS. Decay heat generated throughout the transient is also conservatively calculated.

15.6.5.3.2.3 Small Break LOCA Results

A full spectrum of breaks was analyzed at the beginning-of-life (BOL) fuel rod conditions to determine the limiting break size for peak clad temperature (PCT) and transient oxidation. The limiting PCT and oxidation cases were each analyzed to determine the limiting time-in-life. A break spectrum of 1.5-inch, 2-inch, 2.25-inch, 2.5-inch, 2.75-inch, 3-inch, 3.25-inch, 4-inch and 6-inch breaks was considered. The 1.5-inch case was found to be non-limiting in NOTRUMP and, therefore, peak clad temperature (PCT) information was not calculated. The 3-inch break was found to be limiting for PCT at a limiting time-in-life of 6,500 MWD/MTU, using ZIRLO^{TM} fuel with annular pellets modeled. A summary of the results can be found in Table 15.6-1c and Table 15.6-10b.

The limiting maximum local oxidation case was the 2.5-inch break case. The maximum local transient oxidation is 13.42% at a limiting time-in-life of 15,000 MWD/MTU. The limiting transient oxidation occurs at the burst elevation and includes both outside and post-rupture inside oxidation. Pre-existing (pre-transient) oxidation was also considered and the sum of the pre-transient and transient oxidation remains below 17% at all times in life.

Figures 15.6-49A through 15.6-49G present the Reactor Coolant System pressure transient for the 2-inch, 2.25-inch, 2.5-inch, 2.75-inch, 3.25-inch, 4-inch, and 6-inch breaks, respectively. Figures 15.6-50A through 15.6-50G present the volume history (mixture height) plots for the breaks. The peak clad temperatures for all cases are less than the peak clad temperature of the 3-inch break. The peak clad temperatures are given in Figures 15.6-51A through 15.6-51G.

The limiting maximum clad temperature calculated of the analysis of record (AOR) is 1917°F for a 3-inch break. Figure 15.6-43 shows the RCS pressure response transient, and Figure 15.6-44 provides the core mixture height, each for the limiting PCT break case in the AOR. The peak clad temperature transient is shown in Figure 15.6-45 for the limiting PCT break size. The steam flow rate for the limiting PCT break is shown in Figure 15.6-46. When the mixture level drops below the top of the core, the steam flow computed by NOTRUMP provides cooling to the upper portion of the core. The heat transfer coefficients for this phase of the transient are given in Figure 15.6-47. The hot spot fluid temperature for the limiting PCT break is shown in Figure 15.6-48.

The peak clad temperature may be affected by analysis model revisions and operating conditions, as reported under the provisions of 10 CFR 50.46. (See 10 CFR 50.46 report for current PCTs.)

15.6.5.3.2.4 Conclusions

The analysis presented in this section shows that the high head portion of the ECCS provides sufficient core flooding to meet the 10 CFR 50.46 acceptance criteria of:

1. The calculated peak fuel element cladding temperature does not exceed 2200°F.

2. The local cladding oxidation limit of 17 percent is not exceeded during or after quenching.

3. The amount of fuel element cladding that reacts chemically with water or steam to generate hydrogen does not exceed 1 percent of the total amount of Zircaloy in the fuel rod cladding.

4. The core remains amenable to cooling during and after the break.

5. The core temperature is reduced and decay heat is removed for an extended period of time, as required by the long-lived radioactivity remaining in the core.

Hence, adequate protection is afforded by the ECCS in the event of a small break LOCA.

15.6.5.3.2.5 Post Analysis of Record Evaluations

In addition to the analysis presented in this section, evaluation and assessments may be performed as needed to address computer code errors and emergent issue, or to support plant changes. The issues or changes are evaluated, and the impact on the Peak Cladding Temperature (PCT) is determined. The resultant increase or decrease in PCT is applied to the analysis or record PCT. The PCTs, including all penalties and benefits, are presented in Table 15.6-10c for the small break LOCA. The resultant PCT is demonstrated to be less than the 10 CFR 50.46(b) requirement of 2200°F.

As discussed in Section 15.6.5.2.7, 10 CFR 50.46 requires that licensees assess and report the effect of changes to or errors in the evaluation model used in the LOCA analyses. The requirements for the large break LOCA analysis are also applicable to the small break LOCA analysis.

15.6.5.4 Radiological Consequences

The DBA LOCA dose analysis supporting BVPS-2 utilizes input parameter values that are bounding for an event at either Unit 1 or Unit 2, Alternative Source Terms (AST) methodology as outlined in 10 CFR 50.67 and Regulatory Guide 1.183, and a core power level of 2918 MWt. The analysis assumes containment leakage for 30 days.

A LOCA would increase the pressure in the containment, initiating containment isolation auxiliary feedwater, emergency core cooling, and containment spray. Normal ventilation in the auxiliary and contiguous buildings is realigned and the engineered safety features (ESF) areas are aligned and exhausted by the supplementary leak collection and release system (SLCRS). However, no credit is taken for filtration of the containment and ESF leakage prior to release to the environment.

Due to the rapid pressure transient expected following a LOCA, the Containment Isolation Phase B (CIB) signal, which initiates the control room isolation and emergency ventilation system, is assumed to occur at T=0 hours.

The analysis assumes a Loss of Offsite Power (LOOP) at T=0 hours. The impact of a LOOP at a more unfavorable time following the accident, such as during the fuel release phase, is not addressed per NRC Information Notice (IN) 93-17. The need to evaluate a design basis event assuming a simultaneous or subsequent LOOP is based on the cause/effect relationship between the two events (an example illustrated in IN 93-17 is that a LOCA results in a turbine trip and a loss of power generation to the grid, thus causing grid instability and a LOOP a few seconds later, i.e., a reactor trip could result in a LOOP). IN 93-17 concludes that plant design should reflect all credible sequences of the LOCA/LOOP, but states that a sequence of a LOCA and an unrelated LOOP (which would be the case if a LOOP was assumed to occur 1 to 2 hours after the event) is of very low probability and is not a concern.

The doses to personnel in the control room following the DBA are provided in Table 15.0-13 while the control room design and operation is described in Section 6.4. Input parameters used for the LOCA dose assessment are provided in Table 15.6-11. The estimated doses to the population at the exclusion area boundary (EAB) and at the low population zone (LPZ) outer boundary are provided in Table 15.0-12. These doses are due to releases from the containment vacuum release system prior to containment isolation, containment leakage, ESF leakage, and back-leakage to the RWST.

Containment Vacuum System Release Source

It is assumed that the containment Vacuum System is operating at the initiation of the LOCA and that the release is terminated as part of containment isolation. In accordance with Regulatory Guide 1.183 the entire RCS inventory, assumed to be at technical specification levels (see Table 15.0-8c) is released to the containment at T=0 hours. It is conservatively assumed that 100% of the volatiles are instantaneously and homogeneously mixed in containment atmosphere. Containment pressurization (due to the RCS mass and energy release), combined with the relief line cross-sectional area, results in a 1600 scfm release of containment atmosphere to the environment over a period of 5 seconds (i.e., prior to containment isolation). A 2200 scfm release is assumed in the dose analysis to bound BVPS-1 conditions. Since the release is isolated within 5 seconds after the LOCA, i.e., before the onset of the gap phase release assumed to be at 30 seconds, no fuel damage releases are postulated.

Per Regulatory Guide 1.183, the chemical form of the iodine released from the RCS is assumed to be 97% elemental and 3% organic. The containment Vacuum System line is routed to the Process Vent which is located on top of the BVPS-1 Cooling Tower. However, since the associated piping is non-seismic, it is conservatively assumed that the release occurs at the containment wall.

No credit is taken for processing this release via the safety related ventilation exhaust and filtration system that services the areas contiguous to containment; i.e., the supplementary leak collection system (SLCRS) filters. To ensure bounding values, the atmospheric dispersion factors utilized for this release reflect the worst value between the containment wall release point and the SLCRS release point for 0-2 hour time period.

Table 15.6-11 tabulates the significant input parameters and assumptions used in determining the radiological consequence due to the containment vacuum release pathway. An assessment of the activity release via this pathway demonstrates that its contribution to the site boundary and control room dose is negligible.

Containment Leakage Source

The inventory of fission products in the reactor core available for release via containment leakage following a LOCA is based on Table 15.0-7a which represents a conservative equilibrium reactor core inventory of dose significant isotopes, assuming maximum full power operation at a core power level of 2918 MWt, and taking into consideration fuel enrichment and burnup.

The fission products released from the fuel are assumed to mix instantaneously and homogeneously throughout the free air volume of the primary containment as it is released from the core. Containment sprays are utilized as one of the primary means of fission product cleanup following a LOCA. BVPS design includes a containment quench spray and a containment re-circulation spray system at each of the units. Following post LOCA containment pressurization, the quench spray system is automatically initiated by the CIB signal, and injects cooling water from the refueling water storage tank (RWST), into the containment, via the quench spray system spray headers. Based on an assumption of a LOOP coincident with the LOCA, the quench spray is assumed to be initiated, at either unit, by approximately T=120 seconds, and is available until depletion of the RWST inventory based on maximum ESF. The recirculation spray system which is assumed to be initiated at T=3870 seconds takes suction from the containment sump and provides recirculation spray inside containment via the recirculation spray headers. Credit for recirculation spray is taken up to 4 days post-LOCA.

In accordance with Regulatory Guide 1.183, two fuel release phases are considered for DBA analyses: (a) the gap release, which begins 30 seconds after the LOCA and continues for 30 minutes and (b) the early In-Vessel release phase which begins 30 minutes into the accident and continues for 1.3 hours.

Since the BVPS long term sump pH is controlled to values of 7 and greater, the chemical form of the radioiodine released from the fuel is assumed to be 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodine. With the exception of noble gases, elemental and organic iodine, all fission products released are assumed to be in particulate form.

The activity released from the core during each release phase is modeled as increasing in a linear fashion over the duration of the phase. The release into the containment is assumed to terminate at the end of the early in-vessel phase, approximately 1.8 hours after the LOCA.

In the “effectively” sprayed region the activity transport model takes credit for aerosol removal due to steam condensation and via containment recirculation and quench sprays based on spray flowrates associated with minimum ESF. It considers mixing between the sprayed and unsprayed regions of the containment, reduction in airborne radioactivity in the containment by concentration dependent aerosol removal lambdas, and isotopic in-growth due to decay.

Since, using SRP 6.5.2 methodology, the calculated elemental iodine spray removal lambdas are greater than 20 hr^{-1}, it is conservatively assumed that the sprays remove the elemental iodine at the same rate as the aerosols when the aerosol removal rates are less than 20 hr^{-1}, and at 20 hr^{-1} when the aerosol removal rate is greater than 20 hr^{-1}. In the effectively sprayed region, a minimum plateout coefficient of 2 hr^{-1} is calculated for BVPS. This allows a maximum elemental iodine removal rate in the effectively sprayed region, during the spray period, of 22 hr^{-1}.

In the unsprayed region, the aerosol removal lambdas reflect gravitational settling. No credit is taken for elemental iodine removal in the unsprayed region.

Since the spray removal coefficients are based on calculated time dependent airborne aerosol mass, there is no restriction on the DF for particulate iodine. The maximum DF for elemental iodine is based on SRP 6.5.2 and is limited to a DF of 200.

Mixing between the "effectively" sprayed and unsprayed regions of the containment is assumed for the duration of the accident. Though higher mixing rates are expected, the dose analysis conservatively assumes a mixing rate of 2 unsprayed volumes per hour in accordance with SRP 6.5.2.

Current BVPS design includes a containment sump pH control system which ensures a long term sump pH equal or greater than 7.0. Consequently, iodine re-evolution is not addressed. The definition of long term as it relates to sump pH and iodine re-evolution post LOCA is based on NUREG/CR 5732.

Radioactivity is assumed to leak from both the sprayed and unsprayed region to the environment at the containment technical specification leak rate for the first day, and half that leakage rate for the remaining duration of the accident (i.e., 29 days). No credit is taken for processing the containment leakage via the safety related ventilation exhaust and filtration system that services the areas contiguous to containment; i.e., the supplementary leak collection system (SLCRS) filters. To ensure bounding values, the atmospheric dispersion factors utilized for the containment release path reflects the worst value between the containment wall release point and the SLCRS release point for each time period.

Table 15.6-11 tabulates the significant input parameters and assumptions used in determining the radiological consequence due to containment leakage.

<u>ESF and RWST back-leakage</u>

With the exception of noble gases, all the fission products released from the core in the gap and early in-vessel release phases are assumed to be instantaneously and homogeneously mixed in the primary containment sump water at the time of release from the fuel. The minimum sump volume increases to a steady state minimum value of 480,750 gallons two hours after the LOCA. Three sump volume values are utilized in the transport model. Up to the first half hour after the LOCA, the sump volume is about 14% of the final value. For the next one and half-hours the sump volume is about 43% of the final value. For the remainder of the accident the steady state minimum sump volume is utilized. In accordance with Regulatory Guide 1.183, with the exception of iodine, all radioactive materials in the recirculating liquid are assumed to be retained in the liquid phase. The subsequent environmental radioactivity release is discussed below:

- <u>ESF leakage</u>: Equipment carrying sump fluids and located outside containment are postulated to leak at twice the expected value of 5700 cc/hr into the auxiliary building. This rate is used in the dose analysis to bound BVPS-1 conditions. The BVPS-2 value is 2134 cc/hr. ESF leakage is expected starting at initiation of the recirculation spray which at BVPS-1 is 1200 seconds (start time based on maximum ESF). Note that due to the long-term nature of this release, minor variations in the start time of this release will not significantly impact the resultant doses. The dose analysis assumes that recirculation is initiated at 300 secs. The peak sump water temperature after 300 seconds is 250°F. As noted in Regulatory Guide 1.183, the fraction of total iodine in the liquid that becomes airborne should be assumed to be equal to the fraction of the leakage that flashes to vapor. The flash fraction, (using Regulatory Guide 1.183 methodology,) associated with this temperature is calculated to be less than 10%. Consequently, in accordance with Regulatory Guide 1.183, 10% of the halogens associated with this leakage is assumed to become airborne and are exhausted (without mixing and without holdup) to the environment via the SLCRS vent located on top of Containment. In accordance with Regulatory Guide 1.183, the chemical form of the iodine released from the sump water is 97% elemental and 3% organic. No credit is taken for the SLCRS filters.

- <u>RWST Back-leakage</u>: Sump water back-leakage into the RWST (located in the Yard) is postulated to occur at twice the expected leakrate of 1 gpm, to be released directly to the environment via the RWST vent. Sump water begins to leak into the RWST at 1782 seconds after the LOCA. At 3055 seconds, the iodine begins to flow out of the RWST and disperses to the environment. A significant portion of the iodine associated with the RWST back-leakage is retained within the tank due to equilibrium iodine distribution balance between the RWST gas and liquid phases (i.e., a time dependent iodine partition coefficient). Environmental airborne iodine activity resulting from RWST leakage is assumed to be 97% elemental and 3% organic. In the dose model, this phenomenon is modeled using a series of effective environmental release rate lambdas from the RWST vent. The analysis of RWST back-leakage envelops the RWST design modifications for ECCS switchover level setpoint and maximum temperature.

Table 15.6-11 tabulates all significant input parameters and assumptions used in determining the radiological consequence due to ESF and RWST back-leakage.

Direct Dose

The direct dose is due to activity in the atmosphere and sump of the containment building and from contained sources in systems carrying radioactive LOCA fluid outside containment. The shielding provided by the containment structure and other buildings, plus the distance factor are considered in the evaluation of the direct doses to the control room and offsite locations. Contained sources that contribute to the direct shine dose in the control room are discussed in Section 6.4.2.5. Note that contained sources are an insignificant contributor to the dose at the EAB and LPZ.

Control Room Habitability

Beaver Valley Power Station is served by a single control room that supports both units. The joint control room is serviced by two ventilation intakes, one assigned to BVPS-1 and the other to BVPS-2. These air intakes are utilized for both the normal as well as the accident mode.

During normal plant operation, both ventilation intakes are operable providing a total supply of 500 cfm of unfiltered outside air makeup which includes all potential inleakage and uncertainties.

Upon receipt of a containment isolation Phase B signal, or a high radiation signal from the control room area monitors, the normal outside air supply dampers automatically close, thus isolating the control room envelope. This signal also initiates the Unit 2 redundant control room emergency ventilation systems (CREVS). On detection of failure of one train, the second train is automatically initiated after a short time delay.

In the unlikely event that neither of the BVPS-2 trains can be put in service, operator action may be utilized to initiate the BVPS-1 control room emergency ventilation system. This unlikely scenario is utilized in accident analysis to allow flexibility in taking out a BVPS-2 CREVS train for maintenance.

The control room emergency ventilation system is capable of maintaining the ambient control room envelope pressure slightly above atmospheric pressure for an indefinite period of time after the accident, thereby limiting inleakage. Periodic control room envelope unfiltered air inleakage tests are performed to confirm that the control room envelope is operable. Each control room emergency ventilation system can draw outside air through an emergency supply filtration unit which consists of a HEPA filter and carbon adsorber with removal efficiency of 99% for the particulate aerosols and 98% for the radioiodines. These emergency supply filtration units and associated air handling equipment are designed to Seismic Category I and Safety Class 3 requirements.

The control room emergency ventilation system recirculation flow is not filtered, but remains in service during emergency conditions to maintain the control room within design temperature limits.

The BVPS control areas are contained in a single control room envelope, which is therefore modeled as a single region. Isotopic concentrations in areas outside the control room envelope are assumed to be comparable to the isotopic concentrations at the control room intake locations.

The atmospheric dispersion factors for the various combinations of release point/receptor are provided in Tables 15.0-14 and 15.0-15.

The atmospheric dispersion factors associated with control room envelope inleakage are assumed to be the same as those utilized for the control room intake. Control room envelope tracer gas tests have indicated that potential sources of unfiltered inleakage into the control room envelope during the post accident pressurization mode are the normal operation dampers associated with the control room ventilation system to which it is reasonable to assign the same X/Q as that of the Control Room air intake.

The other source of inleakage is potentially that associated with door seals. This inleakage, plus an allowance of 10 cfm for ingress and egress, is assigned to the door leading into the control room that is considered the point of primary access. This door is located at grade level and in-between the BVPS-1 and BVPS-2 control room air intakes. It is located close enough to the referenced air intakes to allow the assumption that the X/Q associated with this source of inleakage would be reasonably similar to that associated with the air intakes.

Conservative estimates of control room unfiltered inleakage that envelope the results of recent tracer gas testing performed in the year 2001 were used and provide margin for surveillance tests.

Taking into account loss of offsite power, the maximum estimated delay following a LOCA in attaining control room isolation after receipt of a CIB signal to switch from control room normal operation to emergency ventilation mode, is 77 seconds, which accounts for delays due to Emergency Diesel Generator start, sequencing and damper movement/re-alignment. CREVS Train A fan is expected to get a start signal at T=90 seconds. Considering the CREVS time delay relay setting for Train B fan start, plus fan acceleration time, the total auto start delay is estimated to be 137 seconds.

Since the LOCA analysis is intended to be bounding for an event at either unit, no credit is taken for automatic initiation of the BVPS control room emergency ventilation system. Rather it is assumed that operator action will be necessary to initiate the control room emergency ventilation system, and that a pressurized control room will be available within T=30 minutes. As discussed above, in the event one of the BVPS-2 trains is out of service, and the second train fails to start, operator action will be utilized to initiate the BVPS-1 control room emergency ventilation system.

To support development of bounding control room doses, the most limiting X/Q associated with the release point/receptor for an event in either unit is utilized. The control room post-accident ventilation model utilized in the LOCA dose analysis corresponds to an assumed "single intake" which utilizes the worst case atmospheric dispersion factor (X/Q) from release points associated with accidents at either unit, to the limiting control room intake.

A control room unfiltered inleakage of 300 cfm is conservatively assumed during the time it is isolated, (i.e., between T=77 seconds to T=30 minutes). This value is based on the results of tracer gas testing in the isolated mode, and includes a 10 cfm unfiltered inleakage due to ingress/egress as well as margin. The analysis takes into account measured inleakage using mean values of the tracer gas test measurements.

The control room emergency filtered ventilation intake flow varies between 600 to 1030 cfm, which includes allowance for measurement uncertainties. The control room unfiltered inleakage during the emergency pressurization mode is conservatively assumed to be 30 cfm (includes 10 cfm unfiltered inleakage due to ingress/egress) to reflect the results of tracer gas testing in the pressurized mode, and to also accommodate margin for potential future deterioration. The analysis takes into account measured inleakage using mean values of the tracer gas test measurements.

For reasons outlined below, the dose model uses the minimum intake flow rate of 600 cfm in the pressurized mode as it is considered to be more limiting. Although the intake of radioisotopes is higher at the larger intake rate of 1030 cfm, it is small compared to the radioactivity entering the control room, in both cases, due to unfiltered inleakage. Consequently, the depletion of airborne activity in the control room via the higher exhaust rate of 1030 cfm make the lower intake rate of 600 cfm more limiting from a dose consequence perspective. This argument holds true because the committed effective dose equivalent (CEDE) from inhalation is far more limiting than the deep dose equivalent (DDE) from immersion which is principally from noble gases.

The control room operator doses include contributions due to cloud immersion, external plume shine, airborne activity shine through penetrations in adjoining areas, direct shine from sources in the RWST and from the buildup of activity on the control room intake filters. The direct dose from sources inside the containment and RWST were found to be an insignificant dose contributor.

Control room envelope ventilation design parameters used for the LOCA analysis are presented in Table 6.4-1a. The radiation doses to a control room operator due to various postulated DBAs are summarized in Table 15.0-13.

Dose Model

The radiological consequences from a postulated LOCA are analyzed in accordance with the guidance provided in Regulatory Guide 1.183. S&W computer code PERC2 is utilized in the analysis. PERC2 is a multiple compartment activity transport code which calculates the committed effective dose equivalent from inhalation and the deep dose equivalent from submersion due to halogens, noble gases and other nuclides at the offsite locations and in the control room. The total effective dose equivalent (TEDE) is the sum of CEDE and DDE. The dose calculation model is described in Appendix 15A and is consistent with the regulatory guidance. Table 15.6-11 tabulates all significant input parameters and assumptions used in determining the radiological consequences of a LOCA.

The environmental releases resulting from the LOCA are used in conjunction with the atmospheric dispersion values given in Tables 15.0-14 and 15.0-15, whichever is more limiting.

The estimated post LOCA TEDE dose at the EAB and LPZ is presented in Table 15.0-12 and is within the guidelines of 10 CFR Part 50.67. The estimated dose to the BVPS-2 control room operator due to a LOCA at BVPS is presented in Table 15.0-13 and is within the guidelines of 10 CFR Part 50.67.

15.6.6 Boiling Water Reactor Transients

Not applicable to BVPS-2.

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49. WCAP-13451, "Westinghouse Methodology for Implementation of 10 CFR 50.46 Reporting," October 1992.

Tables for Section 15.6

TABLE 15.6-1

TIME SEQUENCE OF EVENTS FOR THE INADVERTENT OPENING
OF A PRESSURIZER RELIEF VALVE

Event	Time (sec)
Relief valve opens	0.0
Low pressurizer pressure reactor trip set point reached	20.4
Rods begin to drop	22.4
Minimum DNBR occurs	23.2

TABLE 15.6-1c

TIME SEQUENCE OF EVENTS FOR A SMALL BREAK LOCA
(2900 MWt Core Power / Westinghouse Model 51M Steam Generators)

Break Size	1.5-inch	2-inch	2.25-inch	2.5-inch	2.75-inch	3-inch	3.25-inch	4-inch	6-inch
Break Initiation	0	0	0	0	0	0	0	0	0
Reactor Trip Signal	57.7	30.6	23.8	18.9	15.3	12.6	10.9	7.5	4.4
S-Signal	74.7	42.1	33.8	28.1	24.1	20.9	18.9	15.3	11.1
Accumulator Injection	N/A	3494	2278	1714	1340	1082	915	557	261
PCT Time (3)	(1)	3220.2	2317.6	2118.9	1680.8	1316.6	1151.1	723.5	3274.6
Core Recovery	N/A	5687	5599	(2)	(2)	(2)	(2)	(2)	3564

Notes:

(1) It has been judged that no core uncovery of any consequence will take place and the 1.5-inch case is non-limiting. Therefore, no PCT calculations were performed.

(2) For the cases where core recovery is greater than the transient time, basis for transient termination can be concluded based on some or all of the following: (1) The RCS system pressure is decreasing which will increase SI flow, (2) Total RCS system mass is increasing due to SI flow exceeding break flow, and (3) Core mixture level has begun to increase and is expected to continue for the remainder of the accident.

(3) The limiting time-in-life for the 3-inch break case for PCT was determined to be at 6,500 MWD/MTU. All other PCT times are for beginning-of-life (BOL).

TABLE 15.6-2

PARAMETERS USED FOR THE
SMALL LINE CARRYING
PRIMARY COOLANT FAILURE

Core Power Level	2918 MWt
Minimum Reactor Coolant Mass	340,711 lbm
CVCS letdown line break - mass flow rate	16.79 lbm/s
Break Flow Flash Fraction	37%
Time to isolate break	15 minutes
Melted Fuel Percentage	0%
Failed Fuel Percentage	0%
RCS Tech Spec NG & Iodine Concentration	Table 15.0-8c (0.35 µCi/gm DE-I131)
RCS Equilibrium Iodine Appearance Rates	Table 15.0-10
Accident Initiated Spike Appearance Rate	500 times equilibrium
Duration of Accident Initiated Spike	4 hours
Iodine Species released to Environment	97% elemental; 3% organic
SLCRS Filter Efficiency	0%
Environmental Release Point	Ventilation Vent
CR Emergency Ventilation: Initiation Signal/Timing	CR is maintained under Normal Operation ventilation

TABLE 15.6-4

SEQUENCE OF EVENTS

EVENT		TIME (sec)
SG Tube Rupture		0
Reactor Trip		116
Safety Injection		141
Ruptured SG Isolated	(AFW)	800
	(Steam Line)	1018
Ruptured SG Atmospheric Steam Dump Valves Fails Open		1020
Ruptured SG Atmospheric Steam Dump Valve Block Valve Closed		1620
RCS Cooldown Initiated		1764
RCS Cooldown Terminated		2968
RCS Depressurization Initiated		3210
RCS Depressurization Terminated		3298
SI Terminated		3478
Break Flow Terminated		4076

TABLE 15.6-5

OPERATOR ACTION TIMES FOR DESIGN BASIS SGTR ANALYSIS

Action	Time Intervals
Identify and isolate ruptured SG	Isolation of auxiliary feedwater flow to the ruptured steam generator is assumed to occur 5.5 minutes after reactor trip or when the narrow range level reaches 27.5% whichever time is greater. Steam line isolation of the ruptured steam generator is assumed to occur 15 minutes after reactor trip or when the narrow range level reaches 27.5%, whichever time is greater.
Operator action time to initiate cooldown	2.4 min
Cooldown	Calculated by LOFTTR2
Operator action time to initiate depressurization	4 min
Depressurization	Calculated by LOFTTR2
Operator action time to initiate SI termination	3 min
SI termination and pressure equalization	Calculated by LOFTTR2

TABLE 15.6-5a

STEAM GENERATOR TUBE RUPTURE MASS RELEASE RESULTS

	TABLE MASS FLOW (POUNDS)	
	0 - 2 HRS	2 - 8 HRS
Ruptured SG		
- Condenser	142,300	0
- Atmosphere	67,300	46,800
- Feedwater	169,800	0
Intact SGs		
- Condenser	281,900	0
- Atmosphere	380,300	798,500
- Feedwater	844,300	820,300
Break Flow	206,600	0

TABLE 15.6-5b

PARAMETERS USED IN EVALUATING RADIOLOGICAL CONSEQUENCES OF A STEAM GENERATOR TUBE RUPTURE

Core Power Level	2918 MWt
Reactor Coolant Mass	368,000 lbm
Break Flow to Faulted Steam Generator	(0-116.4 sec) 9,200 lbm (116.4-3932 sec) 190,300 lbm
Time of Reactor Trip	116.4 sec
Termination of Break Flow to Faulted SG	3932 sec
Amount of Break Flow that Flashes	(0-116.4 sec) 1,730.2 lbm (116.4-1968.5 sec) 7496 lbm
Leakage Rate to Intact Steam Generators	150 gpd @ STP for each steam generator
Failed/Melted Fuel Percentage	0%
RCS Tech Spec Iodine & NG Concentration	Table 15.0.8c (0.35 µCi/gm DE-I131)
RCS Equilibrium Iodine Appearance Rates	Table 15.0-10
Pre-Accident Iodine Spike Activity	Table 15.0-9
Accident Initiated Spike Appearance Rate	335 times equilibrium
Duration of Accident Initiated Spike	4 hours
Secondary System Release Parameters	
Intact SG Liquid Mass (min)	95,150 lbm
Faulted SG Liquid Mass (min)	95,150 lbm
Initial Mass in Steam Generators	95,150 lbm
Tech Spec Activity in SG liquid	Table 15.0.8c (0.1 µCi/gm DE-I131)
Form of All Iodine Released to the Environment via Steam Generators	97% elemental; 3% organic
Iodine Partition Coefficient (unflashed portion)	100 (all tubes submerged)
Fraction of Iodine Released (flashed portion))	1.0 (Released without holdup)
Fraction of Noble Gas Released from any SG	1.0 (Released without holdup)
Partition Factor in Condenser	100 elemental iodine 1 organic iodine / Noble Gases
Steam Flow to Condenser before Reactor Trip	142,300 lbm (Faulted SG) 281,900 lbm (Intact SGs)
Faulted SG Steam Releases via MSSV/ADVs	(116.4-3932 sec) 74,200 lbm (3932-7200 sec) 0 lbm (7200-28,800 sec) 43,600 lbm
Intact SG Steam Releases via MSSV/ADVs	(116.4-3932 sec) 172,400 lbm (3932-7200 sec) 230,000 lbm (7200-28,800 sec) 775,600 lbm
Termination of Release from SGs	8 hours
Environmental Release Points	(0-116.4 sec) Main Condenser Air Ejector (116.4 sec-8 hr) MSSVs/ADVs
CR Emergency Ventilation: Initiation Signal/Timing	8 hours after DBA
CR is maintained in normal ventilation mode	
CR Purge Initiation (Manual)Time and Rate	16,200 cfm (min) for 30 min

TABLE 15.6-8a

Key LOCA Parameters and Reference Transient Assumptions

	Parameter	Reference Transient	Uncertainty or Bias
1.0	Plant Physical Description		
a.	Dimensions	Nominal	ΔPCT_{MOD}[1]
b.	Flow resistance	Nominal	ΔPCT_{MOD}[1]
c.	Pressurizer location	Intact loop	Bounded
d.	Hot assembly location	Under limiting location	Bounded
e.	Hot assembly type	17x17 RFA with IFMs and ZIRLO™ clad	Bounded
f.	SG tube plugging level	High (22%)	Bounded*
2.0	Plant Initial Operating Conditions		
2.1	Reactor Power		
a.	Core average linear heat rate (AFLUX)	Nominal - Based on 100% of power (2900 MWt)	ΔPCT_{PD}[2]
b.	Hot Rod Peak linear heat rate (PLHR)	Derived from desired Tech Spec (TS) limit FQ = 2.52 and maximum baseload FQ = 2.1	ΔPCT_{PD}[2]
c.	Hot rod average linear heat rate (HRFLUX)	Derived from TS $F_{\Delta H}$ = 1.75	ΔPCT_{PD}[2]
d.	Hot assembly average heat rate (HAFLUX)	HRFLUX/1.04	ΔPCT_{PD}[2]
e.	Hot assembly peak heat rate (HAPHR)	PLHR/1.04	ΔPCT_{PD}[2]
f.	Axial power distribution (PBOT, PMID)	Figure 15.6-8p	ΔPCT_{PD}[2]
g.	Low power region relative power (PLOW)	0.2	Bounded*
h.	Hot assembly burnup	BOL	Bounded
i.	Prior operating history	Equilibrium decay heat	Bounded
j.	Moderator Temperature Coefficient (MTC)	Tech Spec Maximum (0)	Bounded
k.	HFP boron	800 ppm	Generic

TABLE 15.6-8a (cont.)

Key LOCA Parameters and Reference Transient Assumptions

Parameter	Reference Transient	Uncertainty or Bias
2.2 Fluid Conditions		
a. T_{avg}	Nominal (580.0°F)	ΔPCT_{IC}^{3} *
b. Pressurizer pressure	Nominal (2250.0 psia)	ΔPCT_{IC}^{3}
c. Loop flow	87200 gpm	ΔPCT_{MOD}^{1}**
d. T_{UH}	THOT	0
e. Pressurizer level	Nominal (60%)	0
f. Accumulator temperature	Nominal (87.5°F)	ΔPCT_{IC}^{3}
g. Accumulator pressure	Nominal (640 psia)	ΔPCT_{IC}^{3}
h. Accumulator liquid volume	Nominal (997 ft^3)	ΔPCT_{IC}^{3}
i. Accumulator line resistance	Nominal	ΔPCT_{IC}^{3}
j. Accumulator boron	Minimum (1900 ppm)	Bounded
3.0 Accident Boundary Conditions		
a. Break location	Cold leg	Bounded
b. Break type	Guillotine	ΔPCT_{MOD}
c. Break Size	Nominal (cold leg area)	ΔPCT_{MOD}
d. Offsite power	On (RCS pumps running)	Bounded*
e. Safety injection flow	Minimum	Bounded
f. Safety injection temperature	Nominal (75.0°F)	ΔPCT_{IC}
g. Safety injection delay	Max delay (17.0 sec)	Bounded
h. Containment pressure	Bounded - Based on initial pressure of 14.3 psia. Bounding pressure curve (Figure 6.2-119) is based on COCO containment calculation using conditions supplied in Tables 6.2-50, 51, 52 & 53.	Bounded Bounded
i. Single failure	ECCS: Loss of 1 SI train Containment press: all trains operating	Bounded
j. Control rod drop time	No control rods	

TABLE 15.6-8a (cont.)

Key LOCA Parameters and Reference Transient Assumptions

Parameter	Reference Transient	Uncertainty or Bias
4.0 Model Parameters		
a. Critical flow	Nominal (C_D = 1.0)	ΔPCT_{MOD}[1]
b. Resistance uncertainties in broken loop	Nominal (as coded)	ΔPCT_{MOD}[1]
c. Initial stored energy/fuel rod behavior	Nominal (as coded)	ΔPCT_{MOD}[1]
d. Core heat transfer	Nominal (as coded)	ΔPCT_{MOD}[1]
e. Delivery and bypassing of ECC	Nominal (as coded)	Conservative
f. Steam binding/entrainment	Nominal (as coded)	Conservative
g. Non-condensible bases/accumulator nitrogen	Nominal (as coded)	Conservative
h. Condensation	Nominal (as coded)	ΔPCT_{MOD}[1]

Notes:

1. PCT_{MOD} indicates this uncertainty is part of code and global model uncertainty
2. PCT_{PD} indicates this uncertainty is part of power distribution uncertainty
3. PCT_{IC} indicates this uncertainty is part of initial condition uncertainty

* Confirmed to be limiting
** Assumed to be result of loop resistance uncertainty

Table 15.6-8b

Confirmatory Cases PCT Results Summary

Case	Blowdown	PCT (°F) 1st Reflood	2nd Reflood
Initial Transient	1554	1596	1679
Loss-Of-Offsite-Power (LOOP)	1539	1582	1667
Reduced SGTP (0%)	1565	1571	1672
Decreased PLOW (0.2)	1566	1616	1753
Low Nominal RCS T_{avg} (566.2°F)	1533	1641	1749
Final Reference Transient	1566	1616	1753

Table 15.6-8c

Confirmatory Split Cases PCT Results Summary

Case	PCT (°F) 1st Reflood	PCT (°F) 2nd Reflood
CD = 0.8	1042	~1100
CD = 1.0	1526	1645
CD = 1.2 (Limiting)	1642	1662
CD = 1.4	1617	1621
Reference Guillotine Transient	1616	1753

Table 15.6-8d

Beaver Valley Unit 2 Large Break LOCA Results

Component	Blowdown	First Reflood	Second Reflood	Criteria
50th Percentile PCT (°F)	<1544	<1584	<1676	N/A
95th Percentile PCT (°F)	<1772	<1860	<1976	<2200
Maximum Local Oxidation (%)		<6.7		<17.0
Maximum Total Hydrogen Generation (%)		<0.89		<1.0

Table 15.6-8e

Plant Operating Range Allowed by the Best-Estimate Large Break LOCA Analysis

Parameter			Operating Range
1.0	Plant Physical Description		
	a) Dimensions		No in-board assembly grid deformation during LOCA + SSE
	b) Flow resistance		N/A
	c) Pressurizer location		N/A
	d) Hot assembly location		Anywhere in core interior*
	e) Hot assembly type		Fresh 17X17 RFA with IFMs, ZIRC-4 or ZIRLO™ clad and IFBA or Non-IFBA**
	f) SG tube plugging level		≤ 22%, SG Model 51M
2.0	Plant Initial Operating Conditions		
	2.1 Reactor Power		
		a) Core avg linear heat rate	Core power ≤ 102% of 2900 MWt @ 2% Calorimetric
		b) Peak linear heat rate	$F_Q \leq 2.52$
		c) Hot rod average linear heat rate	$F_{\Delta H} \leq 1.75$
		d) Hot assembly average linear heat rate	$\overline{P}_{HA} \leq 1.75/1.04$
		e) Hot assembly peak linear heat rate	$F_{QHA} \leq 2.52/1.04$
		f) Axial power dist (PBOT, PMID)	Figure 15.6.8p
		g) 28 assembly peripheral region relative power (PLOW)	0.2 ≤ PLOW ≤ 0.6
		h) Hot assembly burnup	≤ 75000 MWD/MTU, lead rod
		i) Prior operating history	All normal operating histories
		j) MTC	≤ 0 at HFP
		k) HFP boron (minimum)	Normal letdown (800 ppm)
		l) Rod power census	See Table 15.6-8f

* Peripheral locations will not physically be lead power assembly.

** Analysis models thimble plugs removed which is judged to bound thimble plug installed. Hence any combination of thimble plugs installed/removed is supported.

Table 15.6-8e (cont.)
Plant Operating Range Allowed by the Best-Estimate Large Break LOCA Analysis

Parameter	Operating Range
2.2 Fluid Conditions	
a) Tavg	$566.2 \pm 4.0 \leq T_{avg} \leq 580.0 \pm 4.0°F$
b) Pressurizer pressure	$P_{RCS} = 2250 \pm 50$ psia
c) Loop flow	≥ 87,200 gpm/loop
d) T_{UH}	Current upper internals, T_{hot} UH
e) Pressurizer level	Normal level, automatic control
f) Accumulator temperature	$70 \leq T_{ACC} \leq 105°F$
g) Accumulator pressure	$575 \leq P_{ACC} \leq 716$ psia
h) Accumulator volume	$922 \leq V_{ACC} \leq 1072$ ft^3
i) Accumulator fL/D	Current line configuration
j) Minimum accumulator boron	≥ 1900 ppm
3.0 Accident Boundary Conditions	
a) Break location	N/A
b) Break type	N/A
c) Break size	N/A
d) Offsite power	Available or LOOP
e) Safety injection flow	Table 15.6-8g
f) Safety injection temperature	45°F ≤ SI Temp ≤ 105°F
g) Safety injection delay	≤ 17 seconds (with offsite power) ≤ 27 seconds (with LOOP)
h) Containment pressure	Bounded see Figure 6.2-119 & Tables 6.2-50, 51, 52 & 53
i) Single failure	Loss of one train of pumped ECCS
j) Control rod drop time	N/A

Table 15.6-8f

Rod Census Used in Best-Estimate Large Break LOCA Analysis

Rod Group	Power Ratio (Relative to HA Rod Power)	% of Core
1	1.0	10
2	0.912	10
3	0.853	10
4	0.794	10
5	0.735	10
6	0.676	10
7	< 0.65	40

Table 15.6-8g

Beaver Valley Unit 2 Best Estimate Large Break LOCA Total Minimum Injected Flow (HHSI and LHSI from 2 Intact Loops)

RCS Pressure (psig)	Flow Rate (GPM)
0	2719.5
10	2556.5
20	2385.5
50	1807.6
90	441.3
100	251.5
150	245.2
200	239.1
400	215.0
600	189.1

Table 15.6-9

Peak Clad Temperature Including All Penalties and Benefits
Best Estimate Large Break LOCA

	Blowdown	First Reflood	Second Reflood
Analysis of Record PCT (°F)	1772	1860	1976
PCT Assessment Allocated to AOR			
a. MONTECF Version 2.4 (°F)	+36	N/A	N/A
b. Revised Blowdown Heatup Uncertainty Distribution (°F)	+49	+5	+5
c. HOTSPOT Fuel Relocation Error (°F)	0	0	+40
d. RAOC Evaluation (°F)	+4	+12	0
e. Accumulator Pressure Range Evaluation (°F)	0	0	-4
f. Design Input Changes with Respect to Plant Operation (°F)	-100	-105	-190
g. Evaluation of Pellet Thermal Conductivity and Peaking Factor Burdown (°F)	0	+25	+10
h. Revised Heat Transfer Multiplier Distributions (°F)	-5	5	-35
i. Error in Burst Strain Application (°F)	0	+20	+30
Best Estimate Large Break LOCA Resultant PCT for Comparison to 10 CFR 50.46 Requirements (°F)	1756	1822	1832

The maximum fuel element cladding temperature shall not exceed 2200°F per 10 CFR 50.46(b)(1).

Evaluation Basis

FQ = 2.4 $F^N_{\Delta H}$ = 1.62   SGTP = 22%

TABLE 15.6-10a

SMALL BREAK LOCA ANALYSIS INPUT PARAMETERS AND RESULTS

<u>Input Parameter</u>

Reactor core power[1], (MWt)	2900
Peak linear power[1,2], (kw/ft)	13.17
Total peaking factor (FQ^T) at peak[2]	40
Power shape	See Figure 15.6-41
Fuel	17 x 17 RFA
Accumulator water volume, nominal (ft^3/accumulator)	997
Accumulator gas pressure, minimum (psia)	625
Pumped safety injection flow	See Figures 15.6-42a and 15.6-42b
Steam generator tube plugging level (%)	22
Thermal Design Flow/loop, (gpm)	82,840
Vessel inlet temperature, (°F)	540.95
Vessel outlet temperature, (°F)	623.72
Reactor coolant pressure, (psia)	2300
Steam pressure (psia)	726.50

[1] 0.6 percent is added to this power to account for calorimetric error. Reactor coolant pump heat is not modeled in the SBLOCA analyses.

[2] This represents a power shape corresponding to a peaking factor envelope ($K_{(Z)}$) based on F_Q=2.40.

Table 15.6-10b

SMALL BREAK LOCA[1] RESULTS
FUEL CLADDING DATA
(2900 MWt Core Power / Westinghouse Model 51M Steam Generators)

Break Size (in)	2	2.25	2.5[2]	2.75	3[3]	3.25	4	6
PCT (°F)	1753	1846	1845	1829	1917	1712	1456	900
PCT Elevation (ft)	12.0	12.0	12.0	12.0	12.0	11.75	11.5	11.25
Hot Rod Burst Time (sec)	N/A	N/A	1725.2	N/A	1314.4	N/A	N/A	N/A
Hot Rod Burst Elevation (ft)	N/A	N/A	11.75	N/A	12.0	N/A	N/A	N/A
Max. Local ZrO_2 Reaction (%)	3.83	4.3	13.42	4.74	7.79	1.9	0.37	0.01
Max. Local ZrO_2 Elev. (ft)	12.0	12.0	11.75	12.0	12.0	11.75	11.25	11.25
Core-Wide Avg. ZrO_2 (%)	<1.0	<1.0	<1.0	<1.0	<1.0	<1.0	<1.0	<1.0

1. A 1.5-inch break size NOTRUMP case was also analyzed, but because it resulted in minimal core uncovery, a PCT for that break size was not calculated.

2. The limiting time-in-life for the 2.5-inch break case for transient oxidation was determined to be at 15,000 MWD/MTU.

3. The limiting time-in-life for the 3-inch break case for PCT was determined to be at 6,500 MWD/MTU.

Table 15.6-10c

Peak Clad Temperature Including All Penalties and Benefits
Small Break LOCA

Analysis of Record PCT (°F)	1917
PCT Assessments Allocate to AOR	
a. None	N/A
Small Break LOCA Resultant PCT for Comparison to 10 CFR 50.46 Requirements (°F)	1917

The maximum fuel element cladding temperature shall not exceed 2200°F per 10 CFR 50.46(b)(1).

TABLE 15.6-11

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A LOSS-OF-COOLANT ACCIDENT

A. Core Parameters

1.	Core Power (MWt)	2918
2.	Fuel Cycle Length (days)	548

B. Radiation Source Terms

1. Puff release of 100% of the Reactor Coolant Inventory to the containment atmosphere. RCS activity concentrations are listed in Table 15.0-8c and are the same as BVPS-1.

2. Core Inventory release timing

gap phase	Onset: 30 sec Duration: 30 min.
early-in-vessel phase	Onset: 30.5 min Duration: 1.3 hours

3. Elements in each group and Release Fractions Released from the Core to Containment Atm Following LOCA (BVPS Unit 2 core activity is listed in Table 15.0-7a and is the same as BVPS-1)

Group	Gap Release phase	Early In-Vessel Release phase	Nuclides
Noble Gas	0.05	0.95	Xe, Kr, Rn, H
Halogens	0.05	0.35	I, Br
Alkali Metals	0.05	0.25	Cs, Rb
Tellurium Group		0.05	Te, Sb, Se, Sn, In, Ge, Ga, Cd, As, Ag
Barium, Strontium		0.02	Ba, Sr, Ra
Noble Metals		0.0025	Ru, Rh, Pd, Mo, Tc, Co
Cerium Group		0.0005	Ce, Pu, Np, Th, U, Pa, Cf, Ac
Lanthanides		0.0002	La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am, Gd, Ho, Tb, Dy

4. Elements in each group and Release Fractions Released from the Core to Sump Following LOCA (BVPS Unit 2 core activity is listed in Table 15.0-7a and is the same as BVPS-1)

Group	Gap Release phase	Early In-Vessel Release phase	Nuclides
Noble Gas	0	0	Xe, Kr, Rn, H
Halogens	0.05	0.35	I, Br
Alkali Metals	0.05	0.25	Cs, Rb
Tellurium Group		0.05	Te, Sb, Se, Sn, In, Ge, Ga, Cd, As, Ag
Barium, Strontium		0.02	Ba, Sr, Ra
Noble Metals		0.0025	Ru, Rh, Pd, Mo, Tc, Co
Cerium Group		0.0005	Ce, Pu, Np, Th, U, Pa, Cf, Ac
Lanthanides		0.0002	La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am, Gd, Ho, Tb, Dy

TABLE 15.6-11 (Continued)

PARAMETERS USED IN EVALUATING THE
RADIOLOGICAL CONSEQUENCES OF A LOSS-OF-COOLANT ACCIDENT

5. Iodine form of core activity released

From the Containment atmosphere due to melted and failed fuel	95% cesium iodide 4.85% elemental 0.15% organic
From the Sump water due to melted and failed fuel	97% elemental 3% organic

C. <u>Data and Assumptions Used to Estimate Containment Airborne Activity Released</u>

1. Containment Vacuum Relief Line release	2200 scfm for 5 seconds
2. Containment leakage rate (%/day)	
a. From t=0 to 24 hours	0.1
b. From t=1 to 30 days	0.05
3. Containment leakage duration (day)	30
4. Containment minimum free volume (ft^3)	1,750,000
5. Containment spray coverage (%)	63
6. Spray deposition effectiveness and timing aerosols	
a. beginning of spray effectiveness	120 seconds
b. ending of spray effectiveness	96 hours
elemental iodine	
c. beginning of spray effectiveness	120 seconds
d. ending of spray effectiveness	8 hours
7. Natural deposition mechanisms and timing	
<u>Sprayed region aerosols</u>	
a. beginning of effectiveness	30 seconds
b. ending of effectiveness	96 hours
elemental iodine	
c. beginning of effectiveness	30 seconds
d. ending of effectiveness	8 hours
<u>unsprayed region aerosols</u>	
e. beginning of effectiveness	30 seconds
f. ending of effectiveness	8 hours
elemental iodine	Not credited

TABLE 15.6-11 (Continued)

PARAMETERS USED IN EVALUATING THE
RADIOLOGICAL CONSEQUENCES OF A LOSS-OF-COOLANT ACCIDENT

8. Aerosol and Elemental Iodine Removal Coefficients used in LOCA dose analysis

Period		Sprayed Region		Unsprayed Region	
From: (hour)	To: (hour)	Aerosol (hr^{-1})	Elem I (hr^{-1})	Aerosol (hr^{-1})	Elem I (hr^{-1})
0	0.00833	----	----	----	----
0.00833	0.02139	5.96	2	0.003	0
0.02139	0.02372	5.96	2	0.003	0
0.02372	0.073	3.872	3.872	0.003	0
0.073	0.14096	2.954	2.954	0.003	0
0.14096	0.20056	2.687	2.687	0.003	0
0.20056	0.2686	2.213	2.213	0.004	0
0.2686	0.39841	2.058	2.058	0.004	0
0.39841	0.5	1.971	1.971	0.005	0
0.5	0.50833	1.969	1.969	0.006	0
0.50833	0.51416	1.971	1.971	0.006	0
0.51416	0.54541	7.455	7.455	0.006	0
0.54541	0.62038	8.194	8.194	0.021	0
0.62038	0.71383	13.128	13.128	0.050	0
0.71383	0.82628	14.802	14.802	0.056	0
0.82628	0.98233	15.168	15.168	0.061	0
0.98233	1.0823	15.593	15.593	0.065	0
1.0823	1.33802	25.567	22	0.067	0
1.33802	1.45671	28.641	22	0.073	0
1.45671	1.55529	28.899	22	0.075	0
1.55529	1.79435	29.137	22	0.077	0
1.79435	1.80833	29.34	22	0.079	0
1.80833	1.81353	29.359	22	0.080	0
1.81353	1.83451	16.589	16.589	0.080	0
1.83451	1.88966	9.873	9.873	0.080	0
1.88966	1.99366	6.654	6.654	0.084	0
1.99366	2	6.402	6.402	0.084	0
2	2.07982	5.452	5.452	0.084	0
2.07982	2.18606	4.507	4.507	0.089	0
2.18606	2.34274	3.818	3.818	0.089	0
2.34274	2.49782	3.3	3.3	0.089	0
2.49782	2.66091	2.933	2.933	0.095	0
2.66091	3.00886	2.478	2.478	0.095	0
3.00886	3.47714	1.375	1.375	0.096	0
3.47714	5	1.2	1.2	0.091	0
5	6.4	1.085	1.085	0.086	0
6.4	8	1	0	0.082	0
8	10	0.924	0	0.077	0
10	24	0.7173	0	0	0
24	62.4	0.5766	0	0	0
62.4	96	0.4	0	0	0
96	720	0	0	0	0

TABLE 15.6-11 (Continued)

PARAMETERS USED IN EVALUATING THE
RADIOLOGICAL CONSEQUENCES OF A LOSS-OF-COOLANT ACCIDENT

9.	Containment mixing rate (unsprayed volumes per hour)	2
10.	Long term pH	>7
11.	Max DF for aerosols	No Restriction
12.	Max DF for elemental iodine	200
13.	SLCRS filter efficiency (%)	Not Credited

D. Data and Assumptions Used to Estimate Sump Activity Releases

1. ECCS Leakage Assumptions

a.	Leak Initiation Time (sec)	1200
b.	Leak Rate (cc/hr - doubled in analysis)	5700
c.	Sump water Volume	
	300 sec to 30 min (ft^3)	19,111
	30 min to 2 hours (ft^3)	25,333
	2 hours to 30 days (ft^3)	43,577
d.	Leakage Fraction	0.1
e.	Sump Temp after 300 seconds (°F)	250

2. RWST Back-leakage Assumptions

a.	Beginning of back-leakage post accident (sec)	1782
b.	Sump water iodine leakage	elemental
c.	Beginning of RWST Release post accident(sec)	3055
d.	End of RWST release post accident (day)	30
e.	Rate of back-leakage to RWST (gpm- doubled in analysis)	1
f.	Iodine release fraction from RWST used in LOCA analysis (values tabulated below)	

Period From	Period To	RWST Vent Iodine Release Rate (unit of day^{-1})	RWST Vent Gas Release Rate (unit of day^{-1})
0.84861	1.66667	1.0E-02	0.78
1.66667	1.75	8.0E-03	0.78
1.75	2	6.0E-03	0.78
2	3	4.0E-03	0.78
3	5	2.0E-03	0.48
5	9	1.1E-03	0.48
9	11	1.1E-03	0.098
11	48	2.4E-04	0.098
48	72	1.1E-04	0.05
72	96	3.0E-05	0.05
96	120	1.0E-05	0.05
120	144	6.0E-06	0.042
144	168	2.0E-06	0.042
168	192	1.0E-06	0.042
192	216	8.0E-07	0.042
216	264	7.0E-07	0.042
264	312	6.0E-07	0.042
312	384	5.0E-07	0.042
384	480	4.0E-07	0.042
480	576	3.0E-07	0.042
576	672	2.4E-07	0.042
672	720	2.0E-07	0.042

TABLE 15.6-11 (Continued)

PARAMETERS USED IN EVALUATING THE
RADIOLOGICAL CONSEQUENCES OF A LOSS-OF-COOLANT ACCIDENT

E. Control Room Parameters

1.	Control Room Volume (ft^3)	1.73E+5
2.	Control Room Normal Intake including Inleakage (cfm)	500
3.	Time when CR is isolated (sec)	77
4.	Infiltration during Isolation mode (cfm)	300
5.	Time when CR Emergency Vent is manually initiated (min)	30
6.	Emergency vent flowrate (cfm)	600 to 1030
7.	Control Room intake filter removal efficiency	
	a. aerosols (%)	99
	b. elemental/organic iodine(%)	98
8.	In-leakage during Emergency Vent mode (scfm)	30

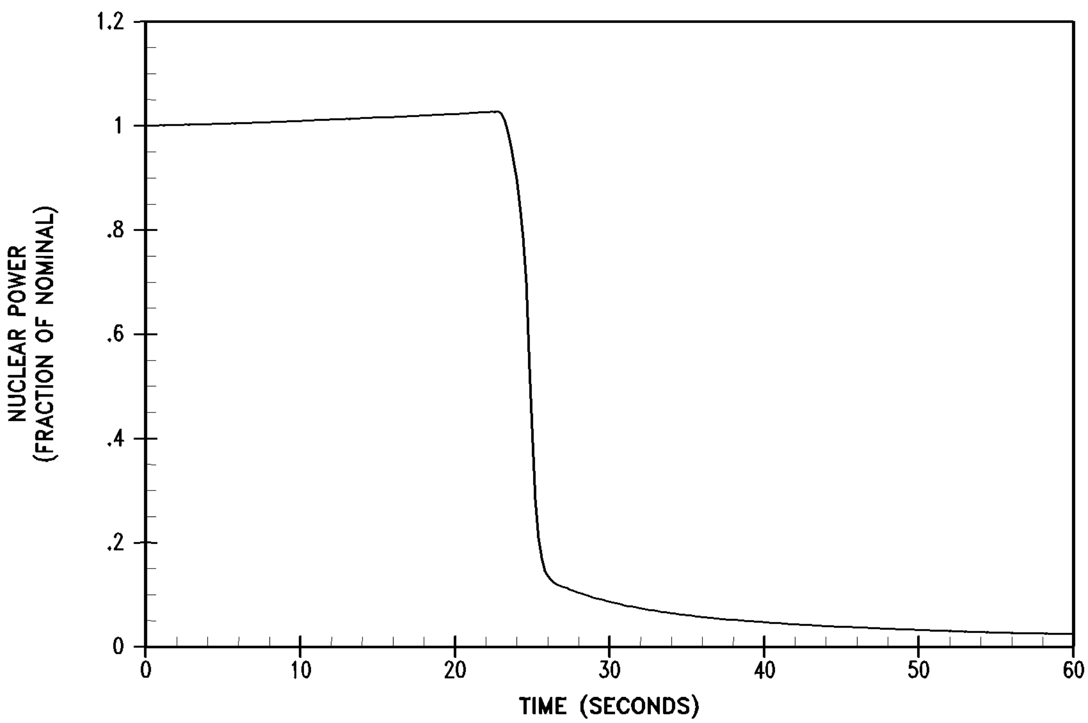

Figure 15.6-1

Nuclear Power Transient for Inadvertent Opening of a Pressurizer Relief Valve

Beaver Valley Power Station Unit No. 2
Updated Final Safety Analysis Report

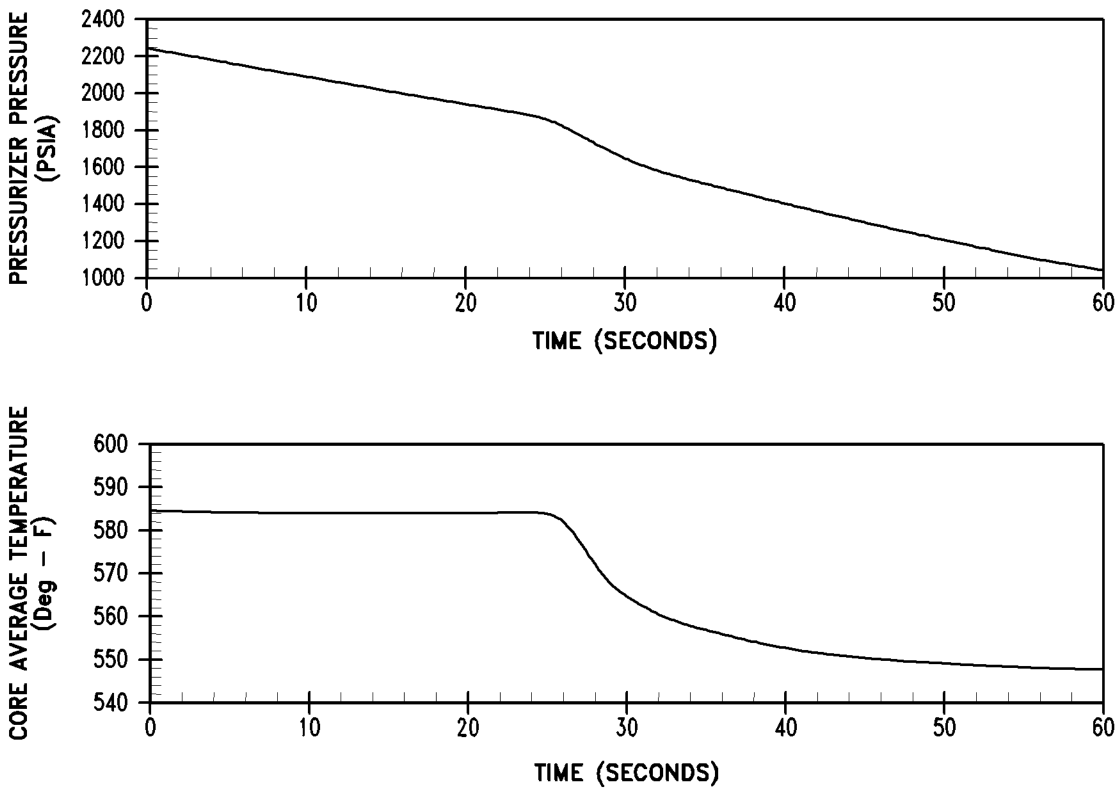

Figure 15.6-2

Pressurizer Pressure and Core Temperature Transients for Inadvertent Opening of a Pressurizer Relief Valve

Beaver Valley Power Station Unit No. 2 Updated Final Safety Analysis Report

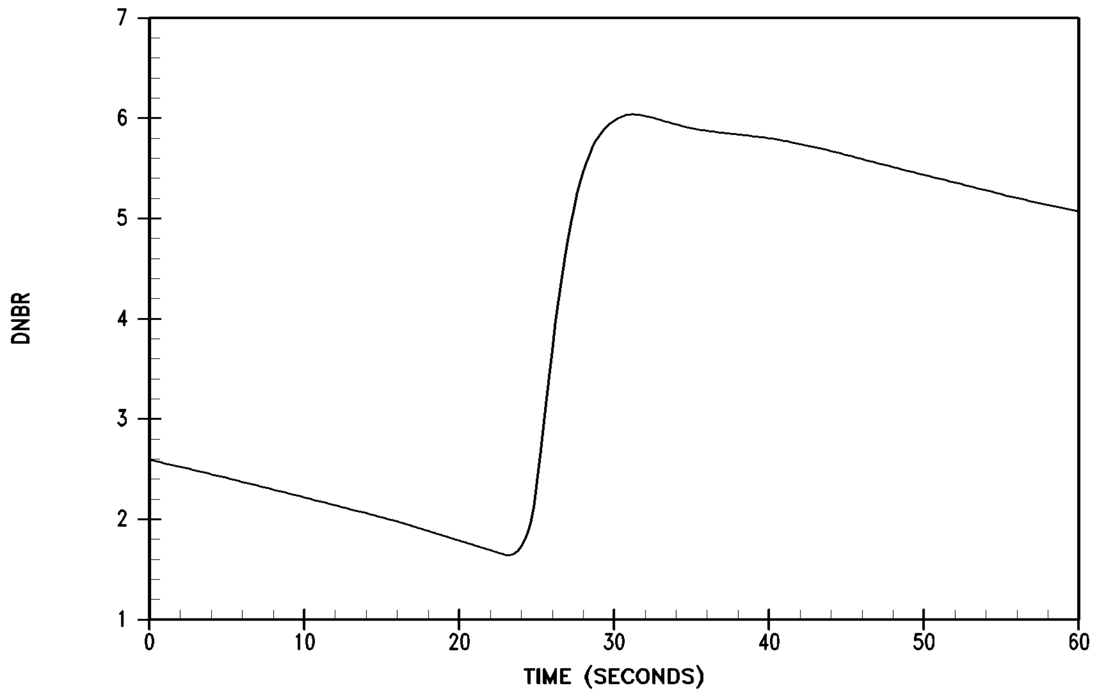

Figure 15.6-3

DNBR Transient for Inadvertent Opening of a Pressurizer Relief Valve

**Beaver Valley Power Station Unit No. 2
Updated Final Safety Analysis Report**

BLOWDOWN	0 second	BREAK OCCURS
		REACTOR TRIP
		PUMPED SI SIGNAL (PRESSURIZER PRESSURE)
		ACCUMULATOR INJECTION BEGINS
		PUMPED ECCS INJECTION BEGINS (ASSUMING OFFSITE POWER AVAILABLE)
		CONTAINMENT HEAT REMOVAL SYSTEM STARTS (ASSUMING OFFSITE POWER AVAILABLE)
	20-30 seconds	END OF BYPASS
	30-40 seconds	END OF BLOWDOWN
REFILL		BOTTOM OF CORE RECOVERY
REFLOOD	10 minutes	ACCUMULATOR EMPTY
		CORE QUENCHED
LONG TERM COOLING	24 hours	SWITCH TO COLD LEG RECIRCULATION ON RWST EXTREME LOW LEVEL ALARM
		SWITCH TO HOT LEG/COLD LEG RECIRCULATION

Figure 15.6-8a

Typical Time Sequence of Events for the Beaver Valley Unit 2 BELOCA Analysis

Beaver Valley Power Station Unit No. 2
Updated Final Safety Analysis Report

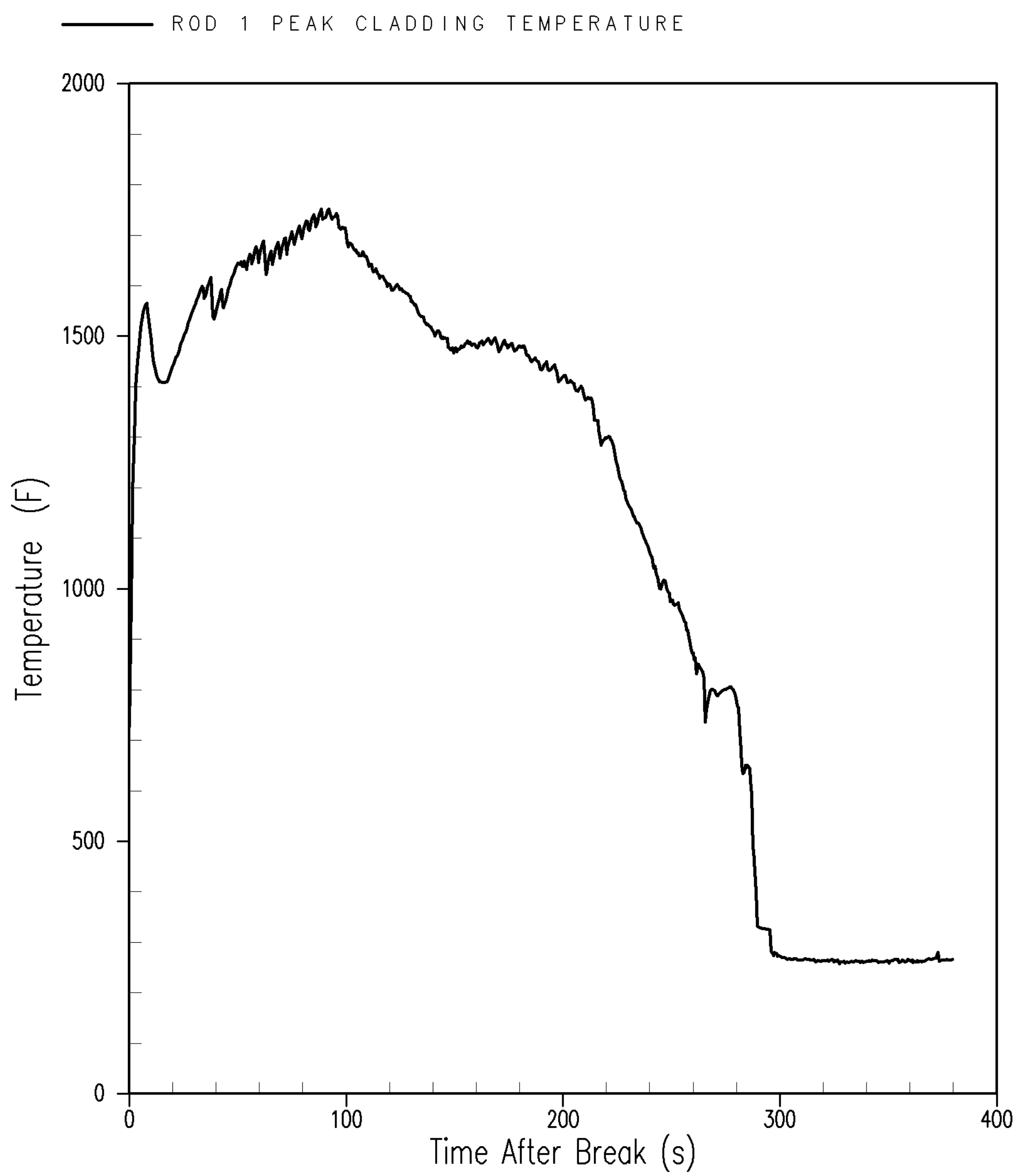

Figure 15.6-8b

Peak Cladding Temperature for Reference Transient

Beaver Valley Power Station Unit No. 2
Updated Final Safety Analysis Report

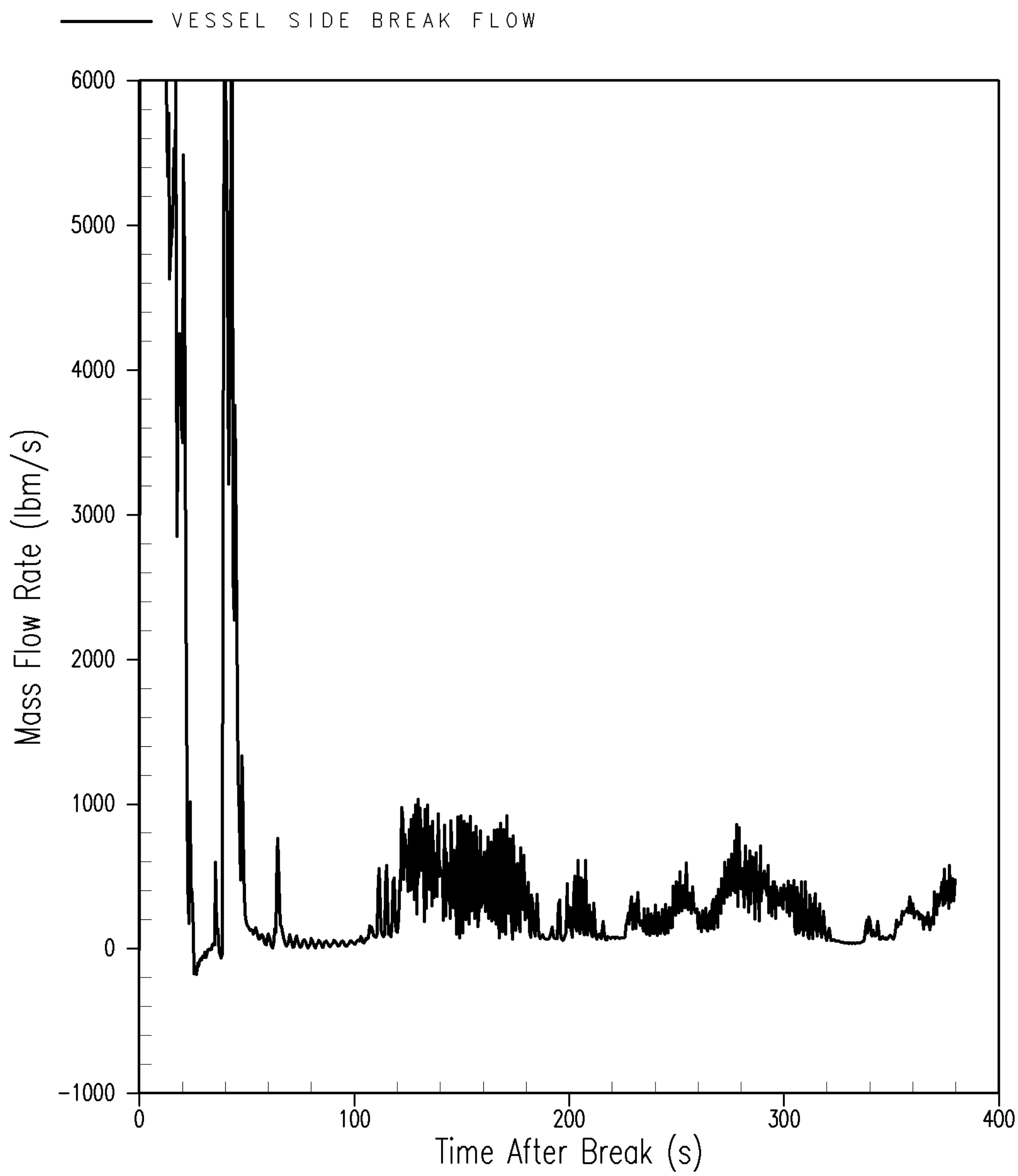

Figure 15.6-8c

Break Flow on Vessel Side of Broken Cold Leg for Reference Transient

Beaver Valley Power Station Unit No. 2
Updated Final Safety Analysis Report

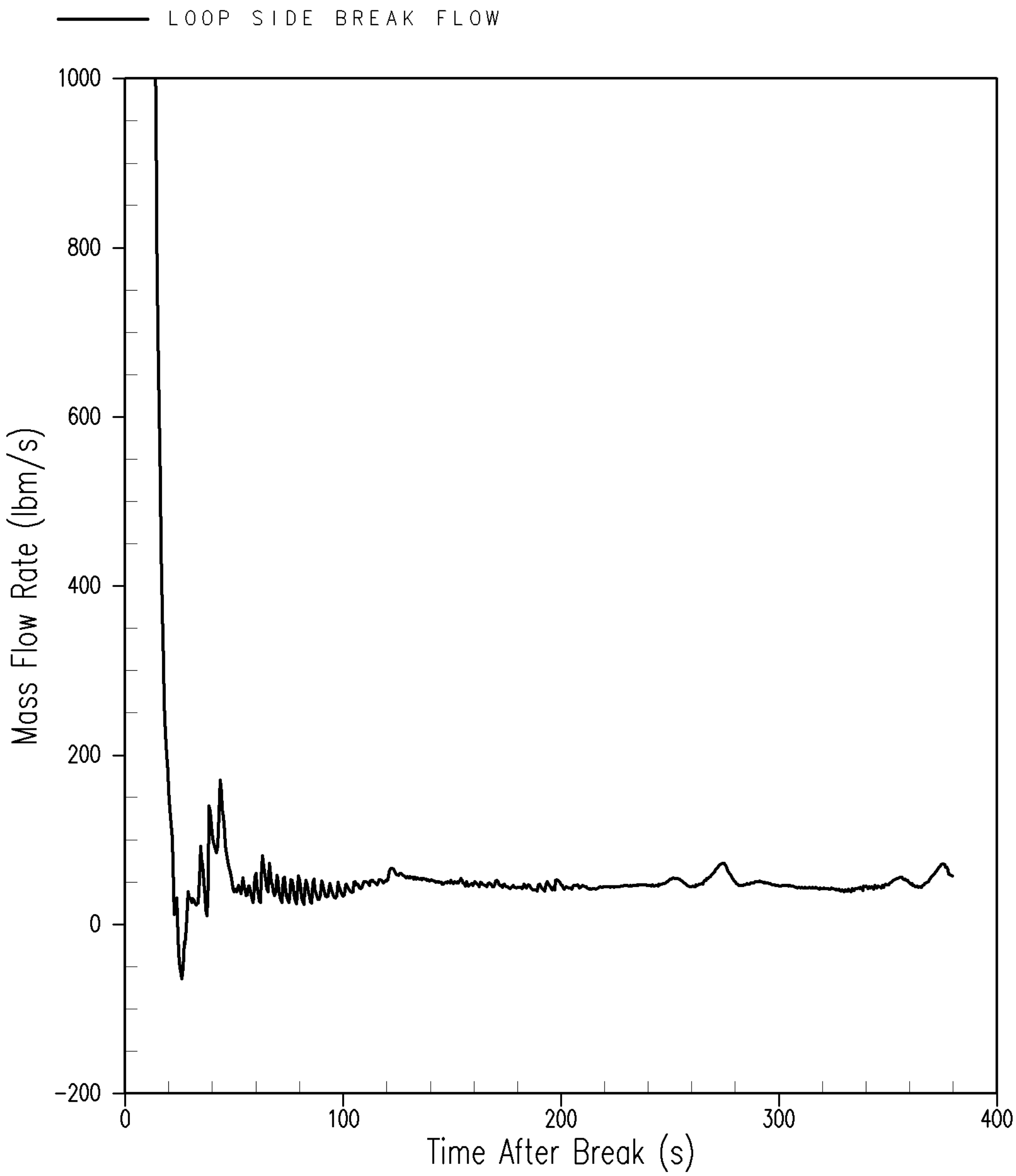

Figure 15.6-8d

Break Flow on Loop Side of Broken Cold Leg for Reference Transient

Beaver Valley Power Station Unit No. 2 Updated Final Safety Analysis Report

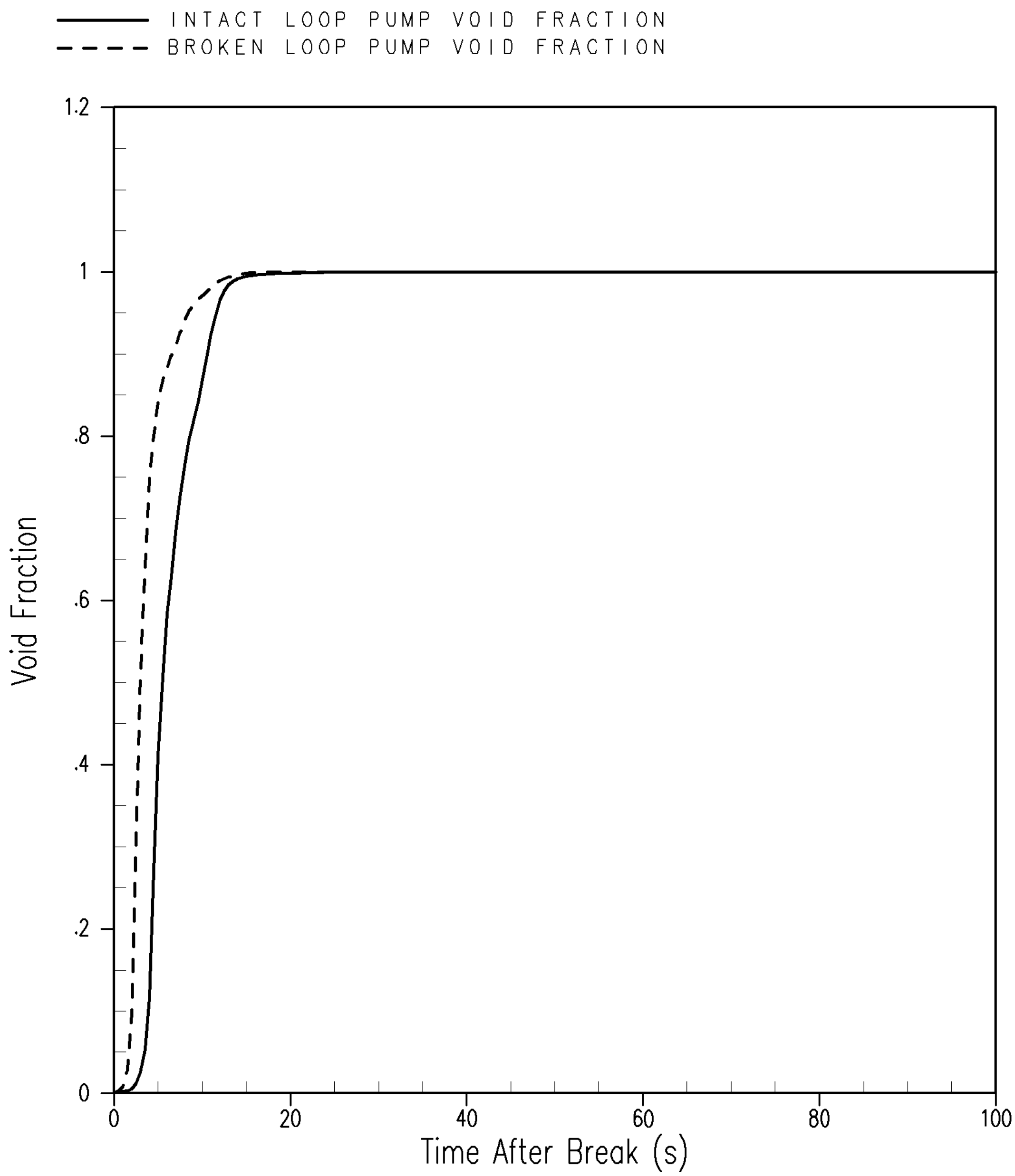

Figure 15.6-8e

Void Fraction at the Intact and Broken Loop Pump Inlet for Reference Transient

Beaver Valley Power Station Unit No. 2
Updated Final Safety Analysis Report

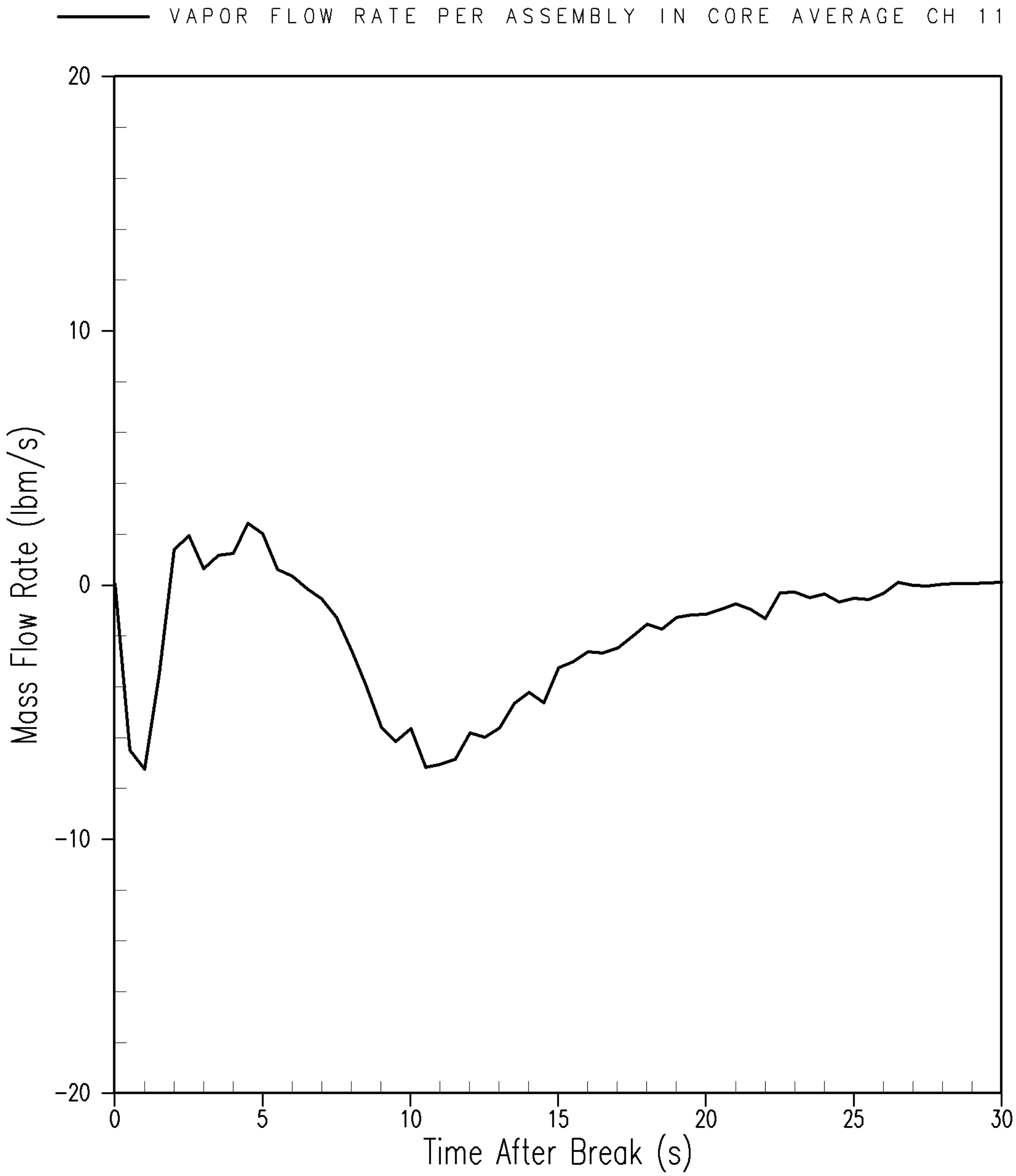

Figure 15.6-8f

Vapor Flow Rate at Midcore in Channel 11 During Blowdown for Reference Transient

Beaver Valley Power Station Unit No. 2 Updated Final Safety Analysis Report

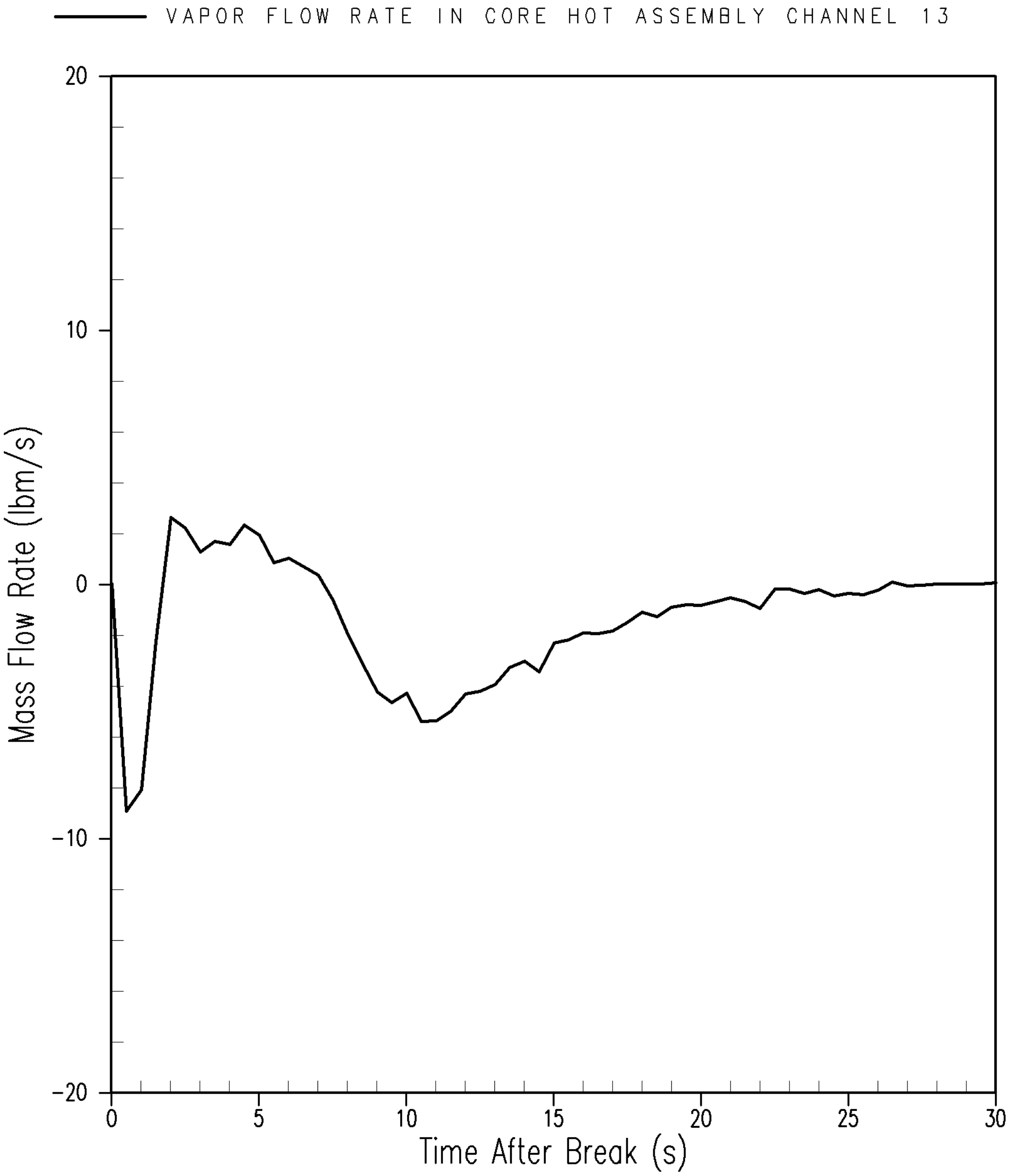

Figure 15.6-8g

Vapor Flow Rate at Midcore in Channel 13 During Blowdown for Reference Transient

Beaver Valley Power Station Unit No. 2
Updated Final Safety Analysis Report

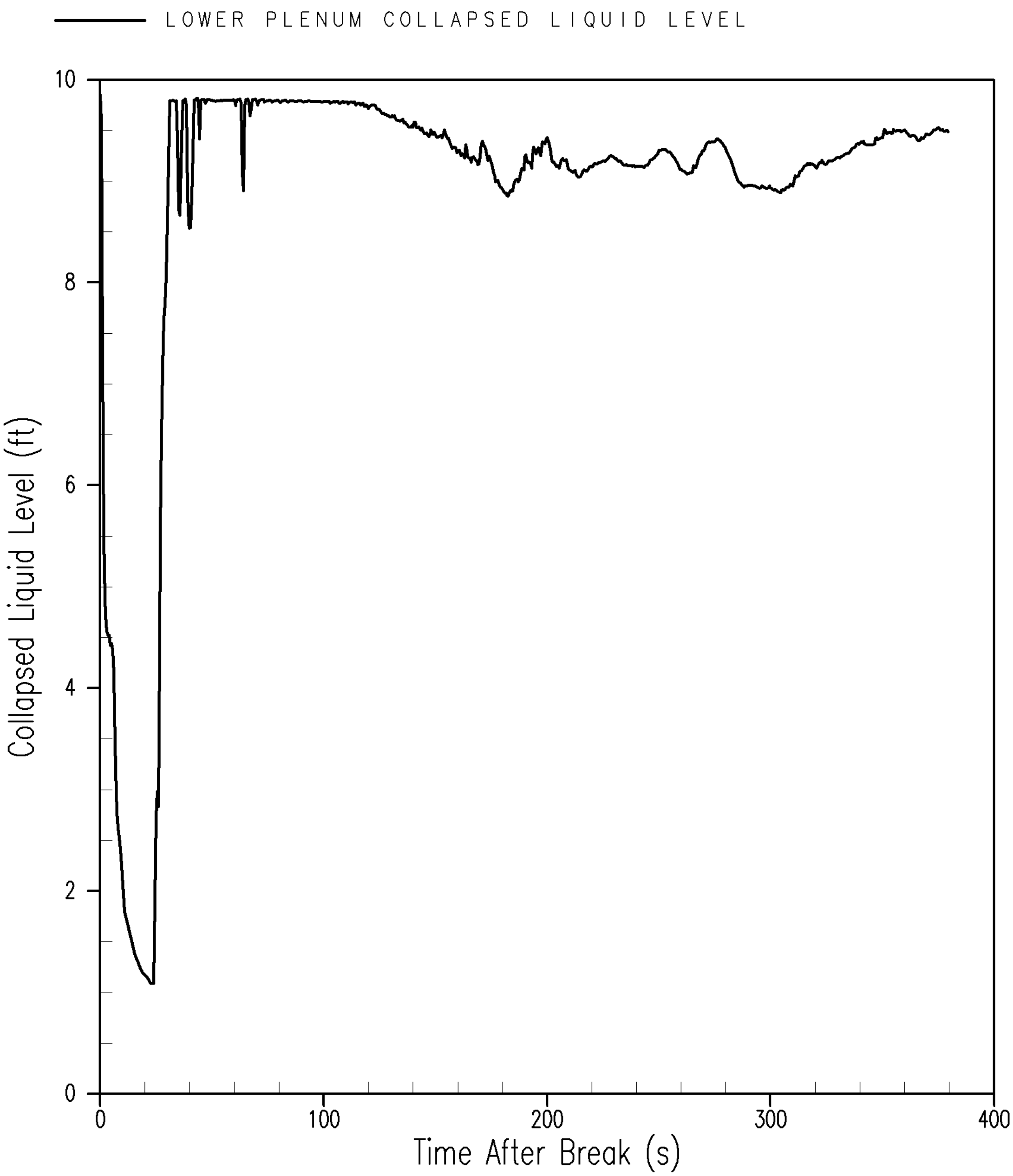

Figure 15.6-8h

Collapsed Liquid Level in Lower Plenum for Reference Transient

Beaver Valley Power Station Unit No. 2
Updated Final Safety Analysis Report

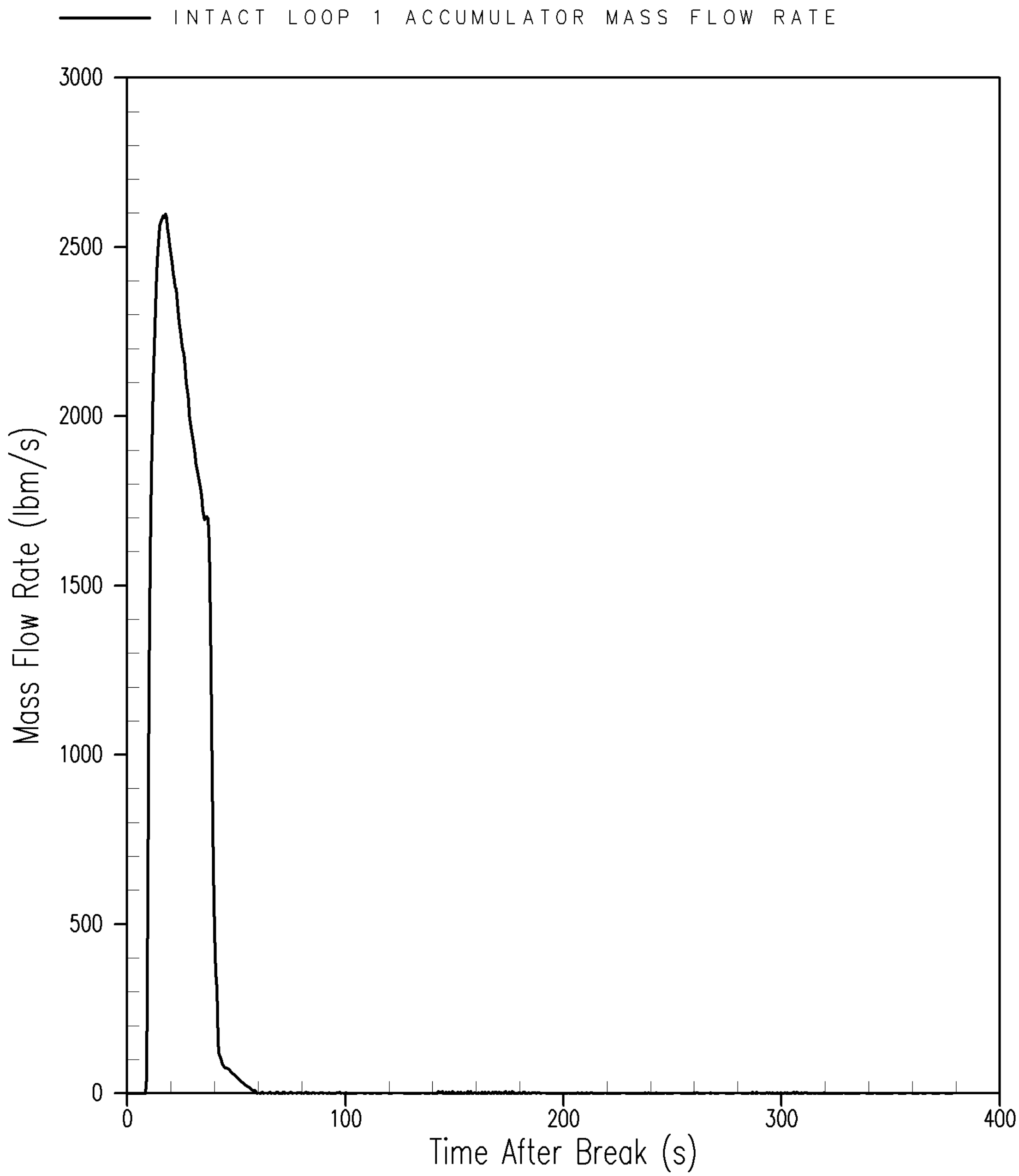

Figure 15.6-8i

Accumulator Mass Flow Rate for Reference Transient

Beaver Valley Power Station Unit No. 2
Updated Final Safety Analysis Report

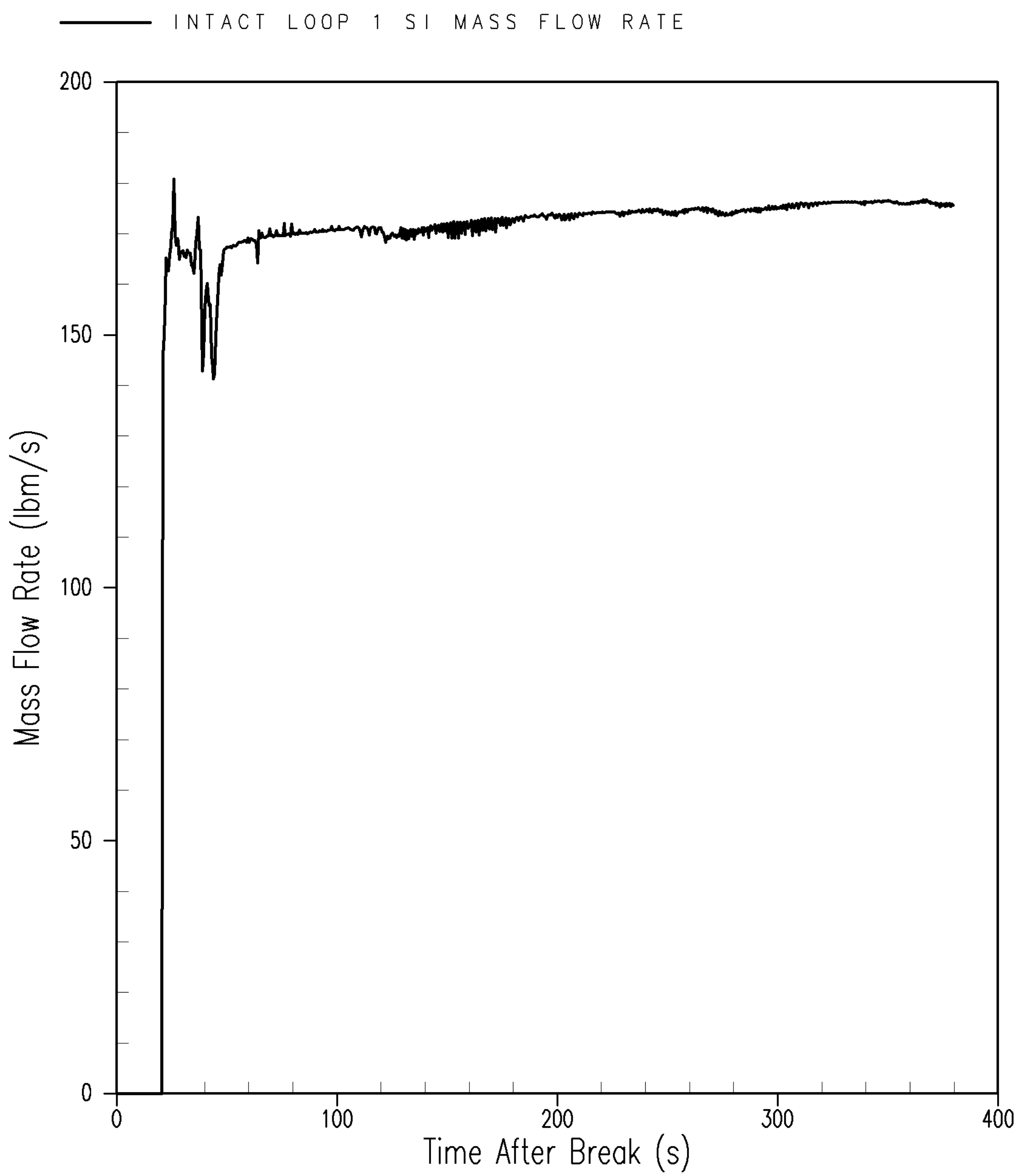

Figure 15.6-8j

SI Mass Flow Rate for Reference Transient

Beaver Valley Power Station Unit No. 2
Updated Final Safety Analysis Report

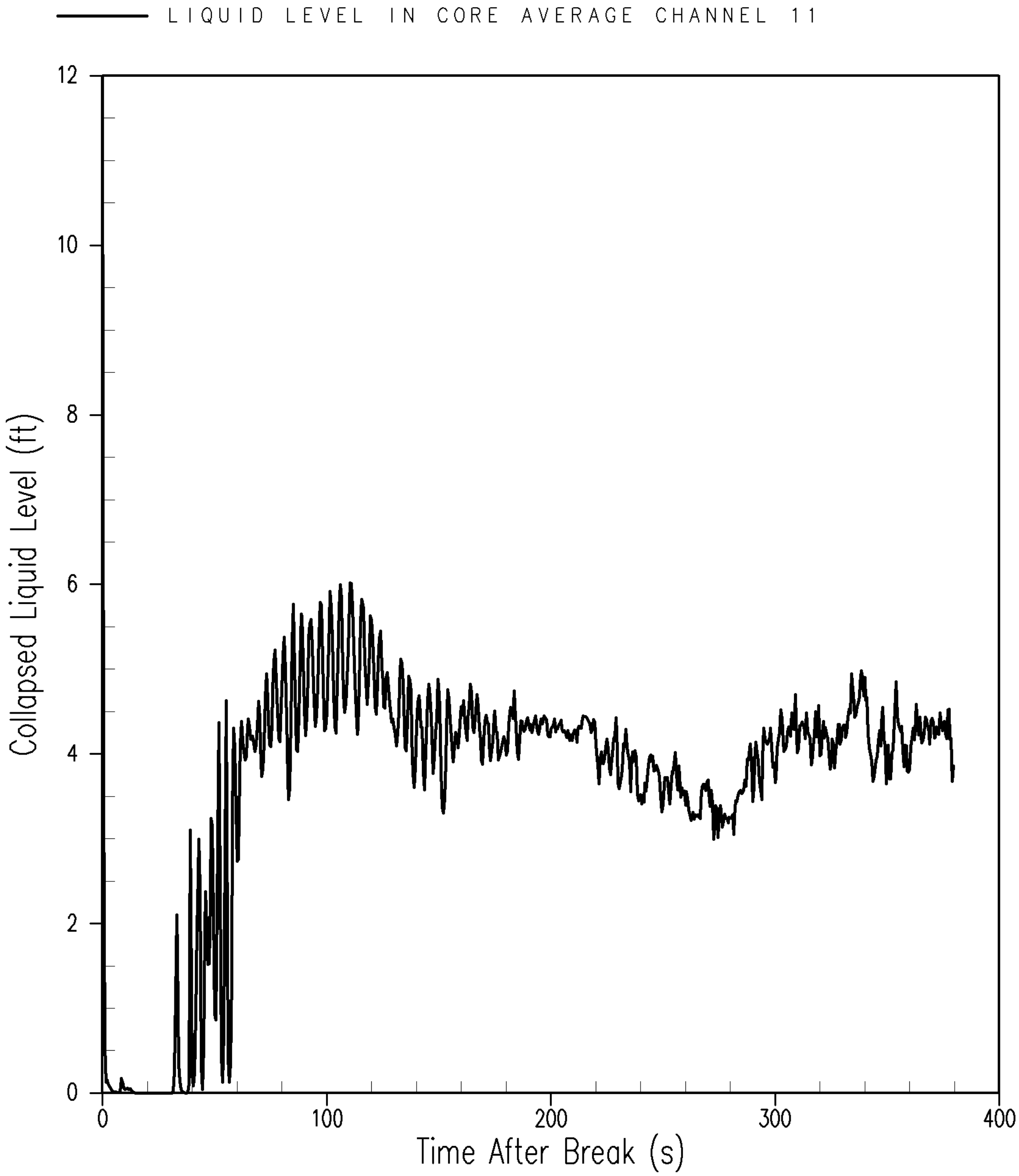

Figure 15.6-8k

Collapsed Liquid Level in Core for Reference Transient

Beaver Valley Power Station Unit No. 2
Updated Final Safety Analysis Report

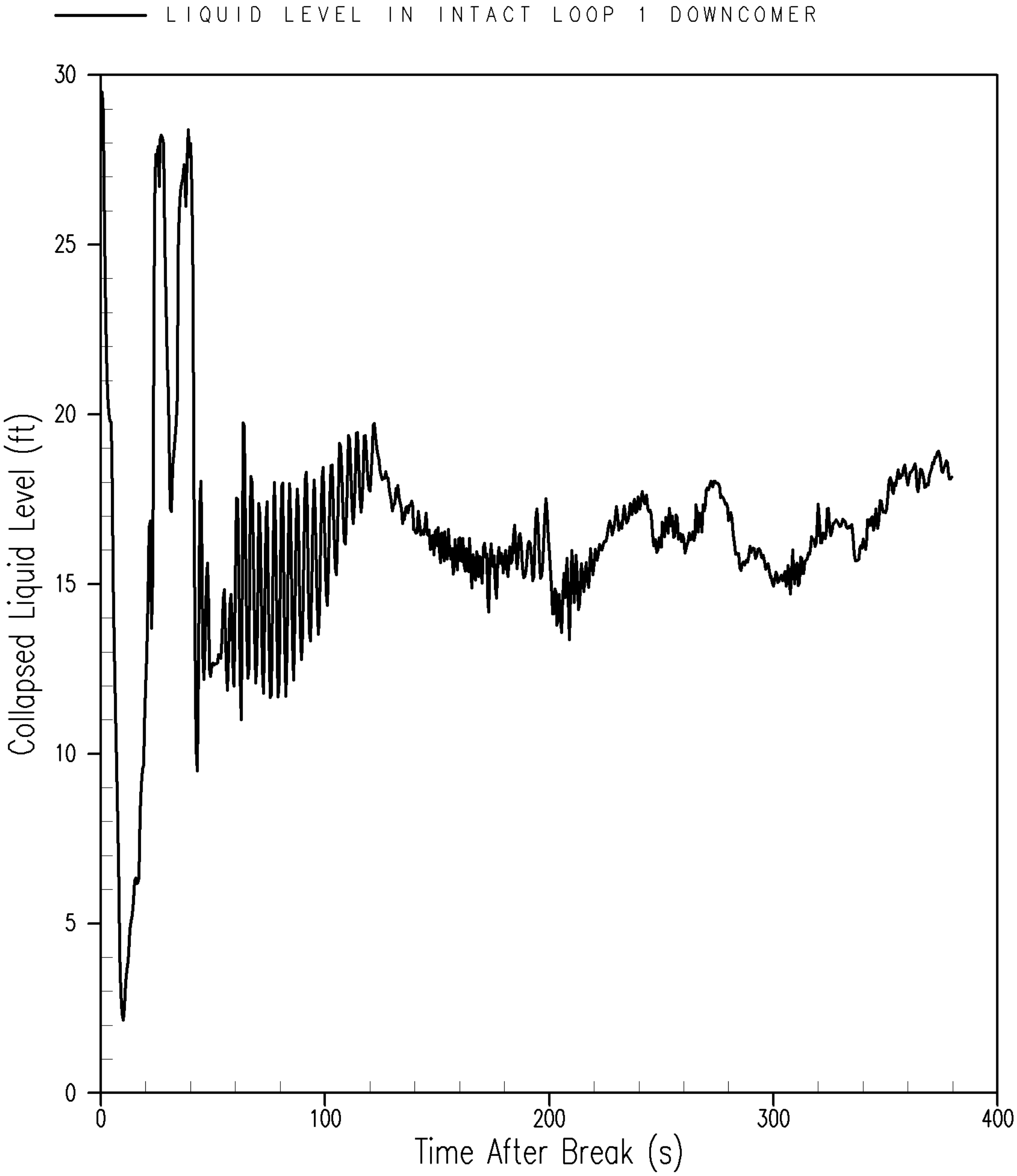

Figure 15.6-8I

Collapsed Liquid Level in Downcomer for Reference Transient

Beaver Valley Power Station Unit No. 2
Updated Final Safety Analysis Report

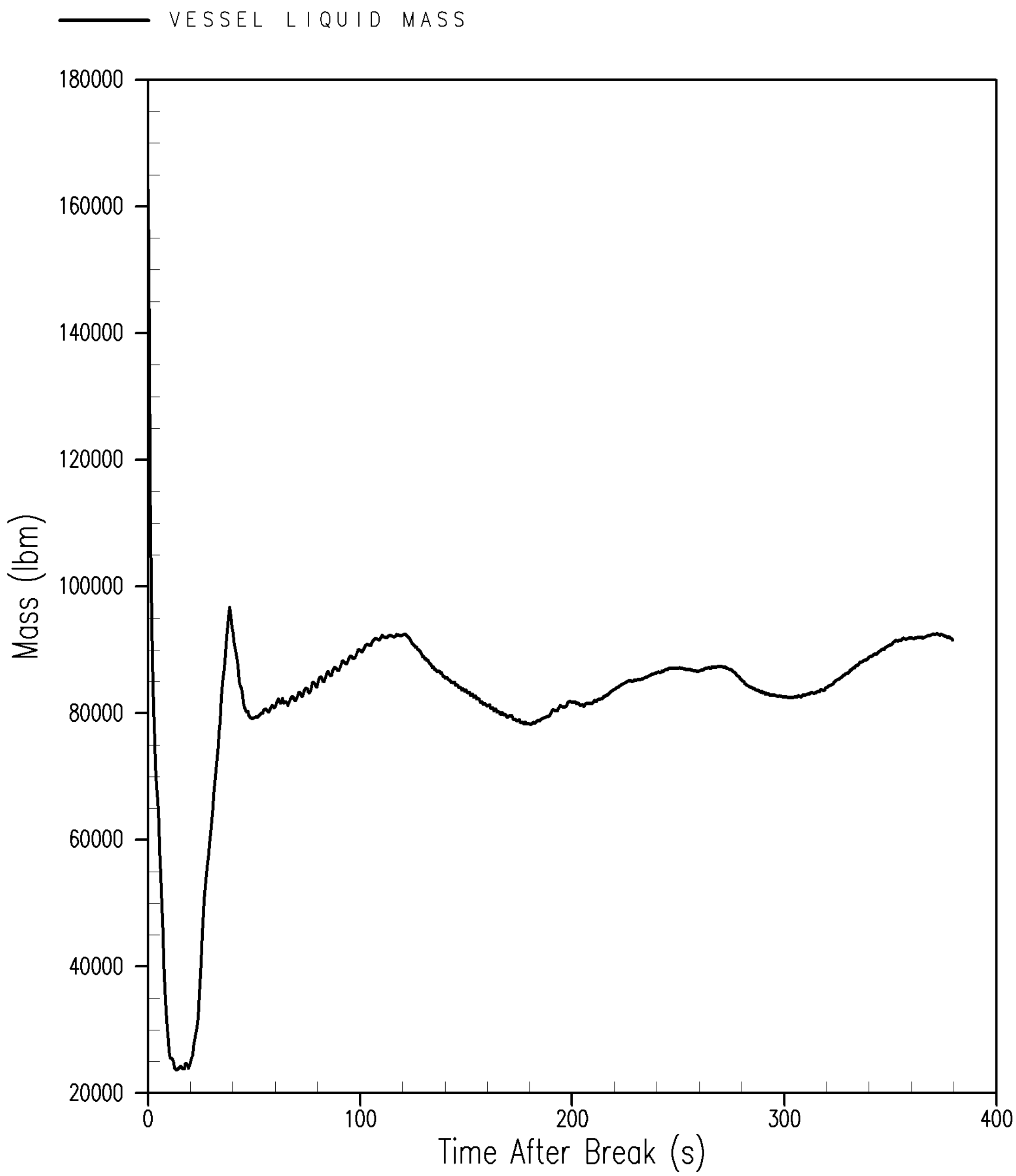

Figure 15.6-8m

Vessel Fluid Mass for Reference Transient

Beaver Valley Power Station Unit No. 2 Updated Final Safety Analysis Report

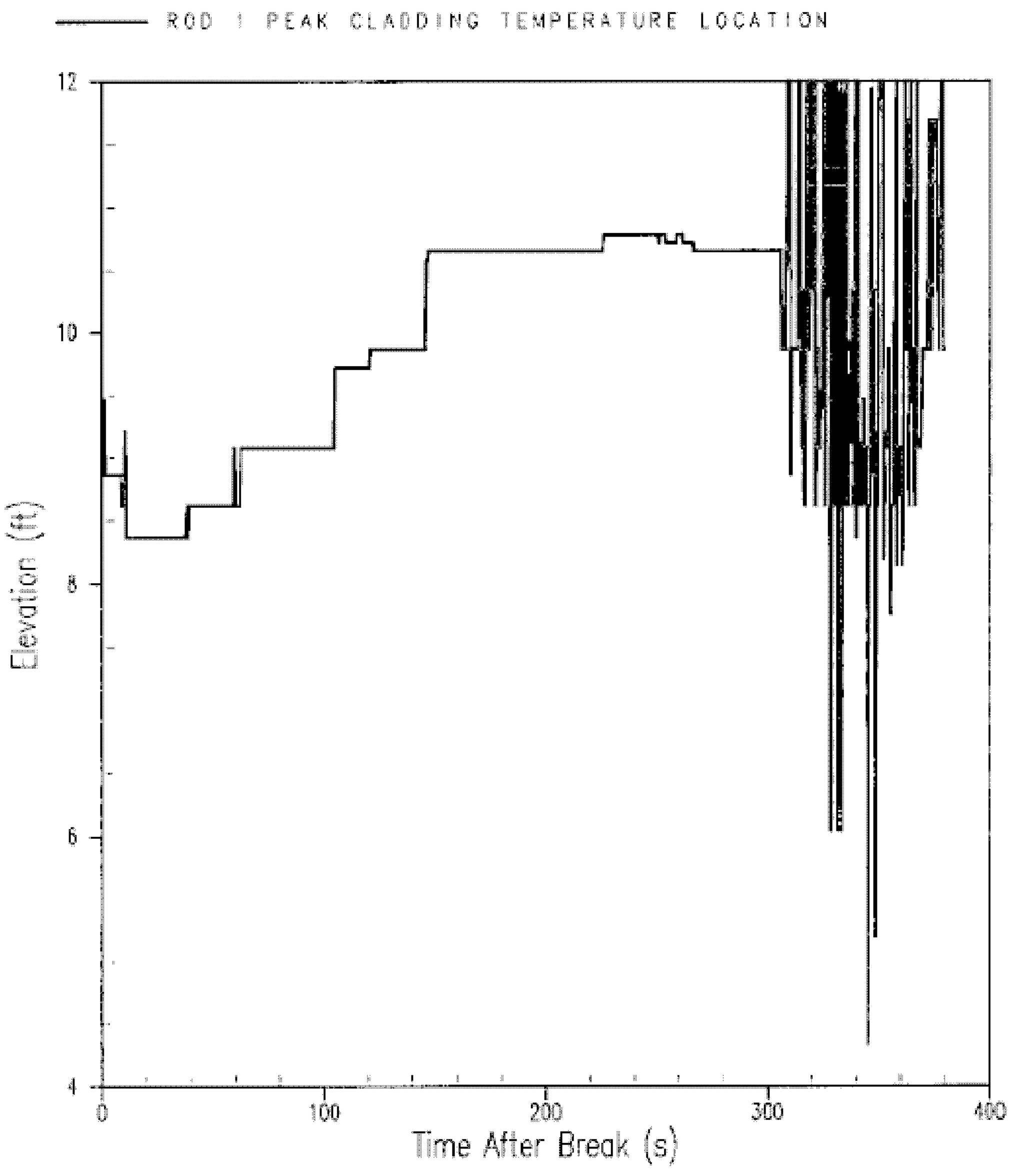

Figure 15.6-8n

Peak Cladding Temperature Location for Reference Transient

Beaver Valley Power Station Unit No. 2 Updated Final Safety Analysis Report

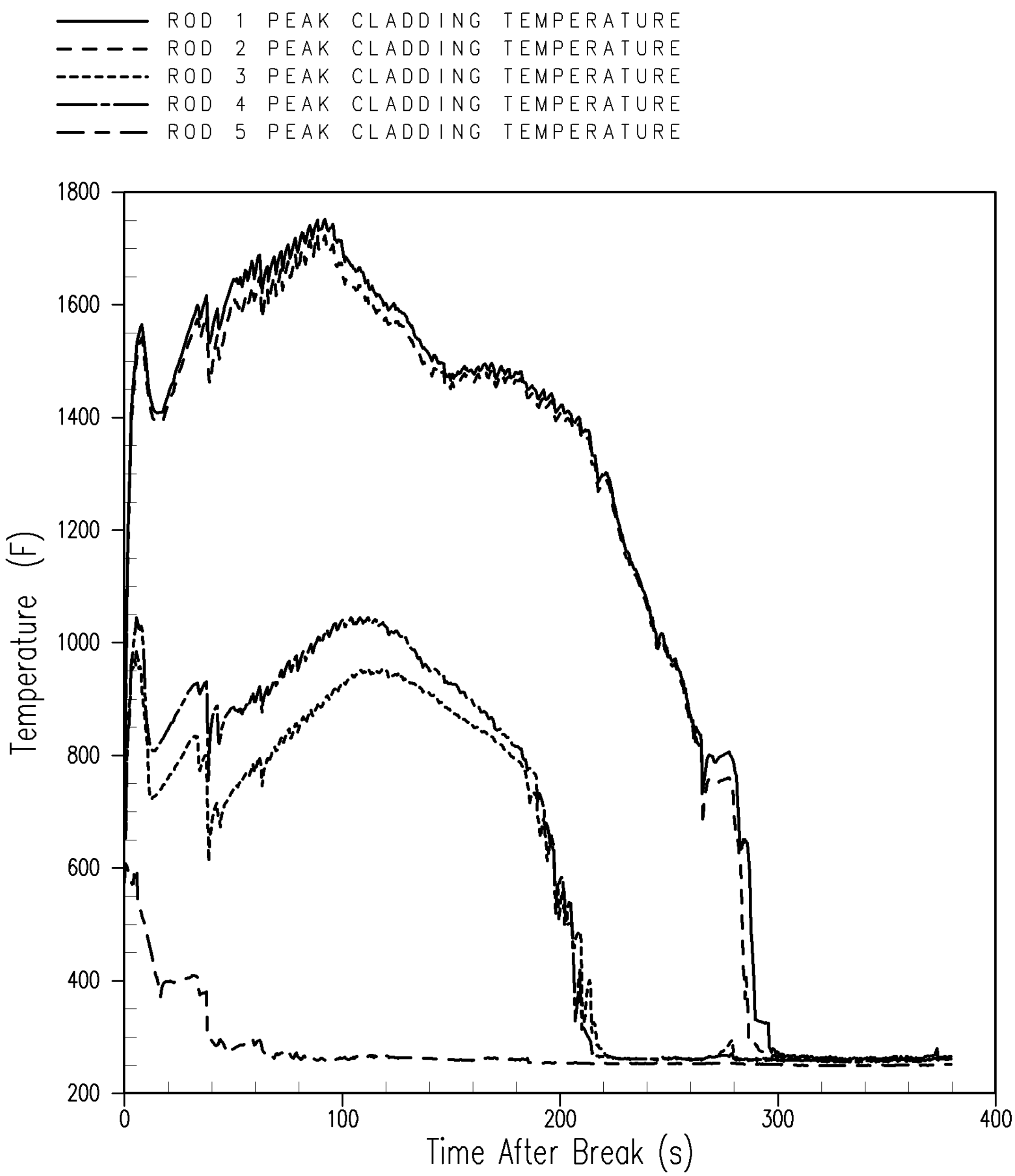

Figure 15.6-8o

Peak Cladding Temperature Comparison for Five Rods for Reference Transient

Beaver Valley Power Station Unit No. 2 Updated Final Safety Analysis Report

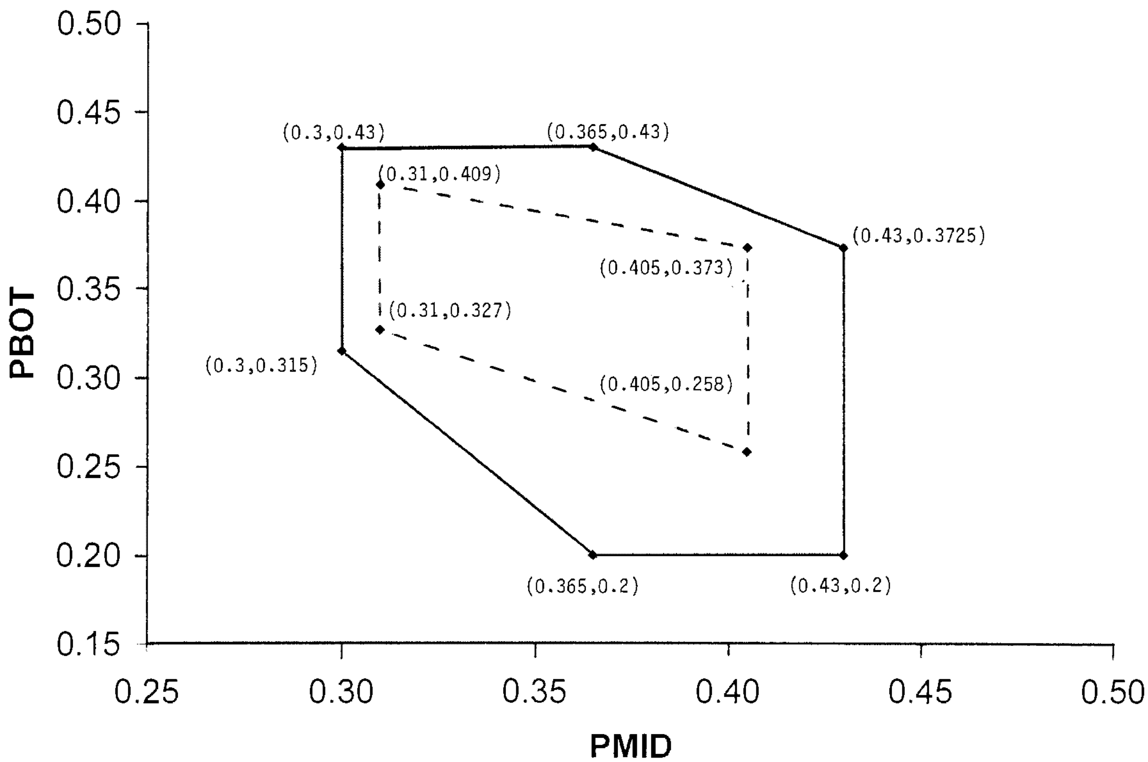

Figure 15.6-8p

Beaver Valley Unit 2 PBOT/PMID Sampling Limits (Plant Operating range indicated by dashed line; WCOBRA/TRAC response surface range indicated by solid line.)

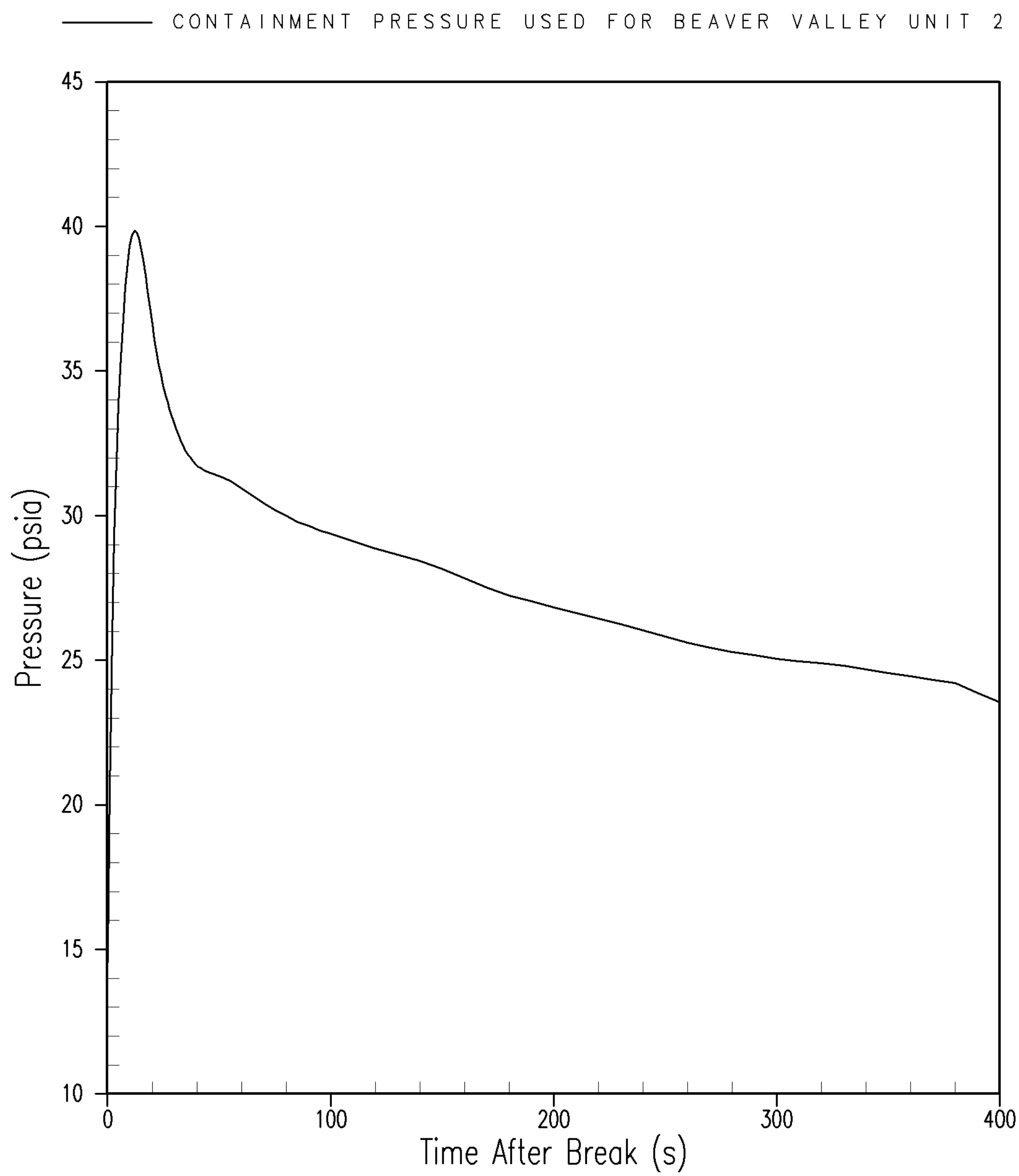

Figure 15.6-8q

Containment Pressure used in Reference Transient

Beaver Valley Power Station Unit No. 2
Updated Final Safety Analysis Report

Rev. 6

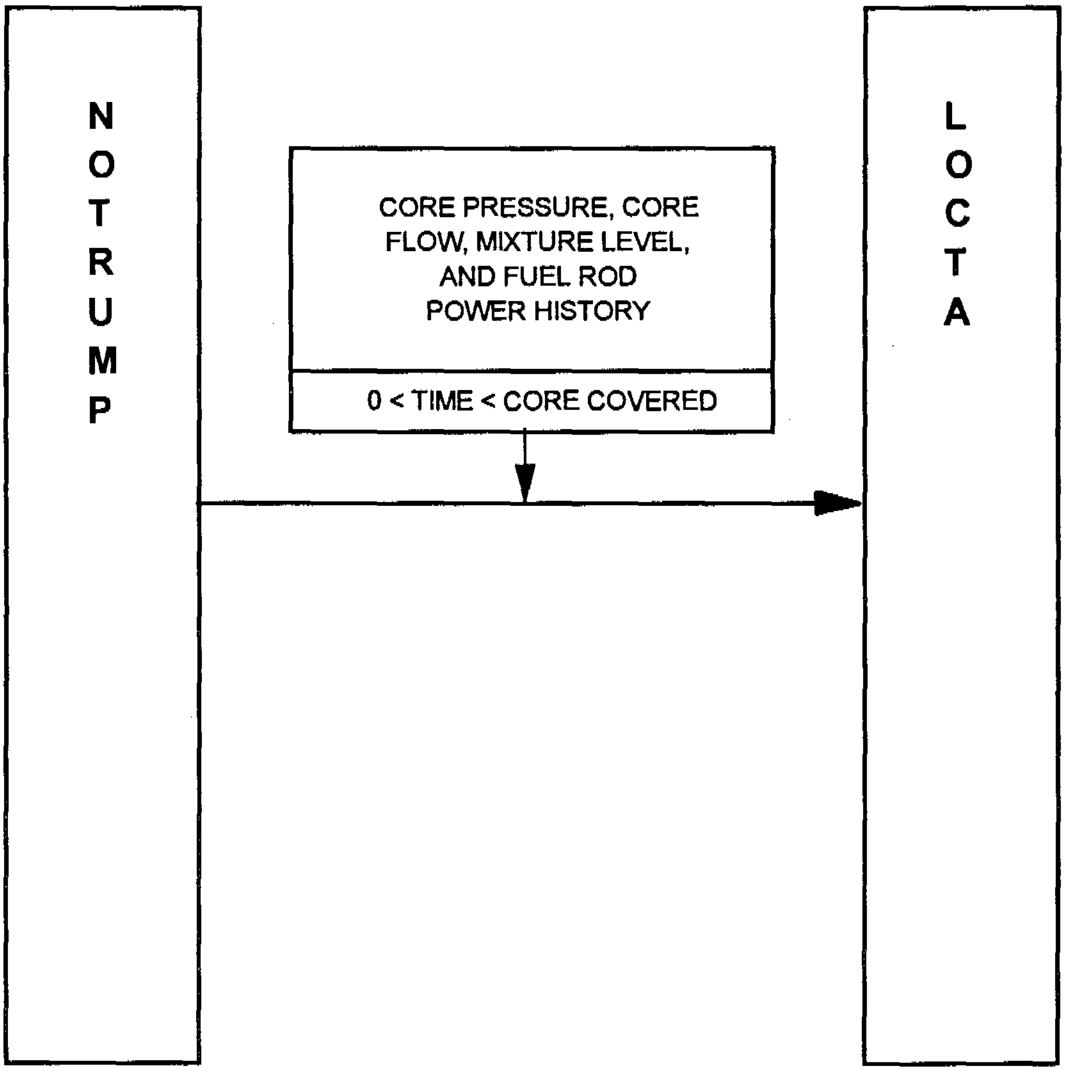

FIGURE 15.6-40
CODE INTERFACE DESCRIPTION FOR SMALL BREAK MODEL
BEAVER VALLEY POWER STATION - UNIT 2
UPDATED FINAL SAFETY ANALYSIS REPORT

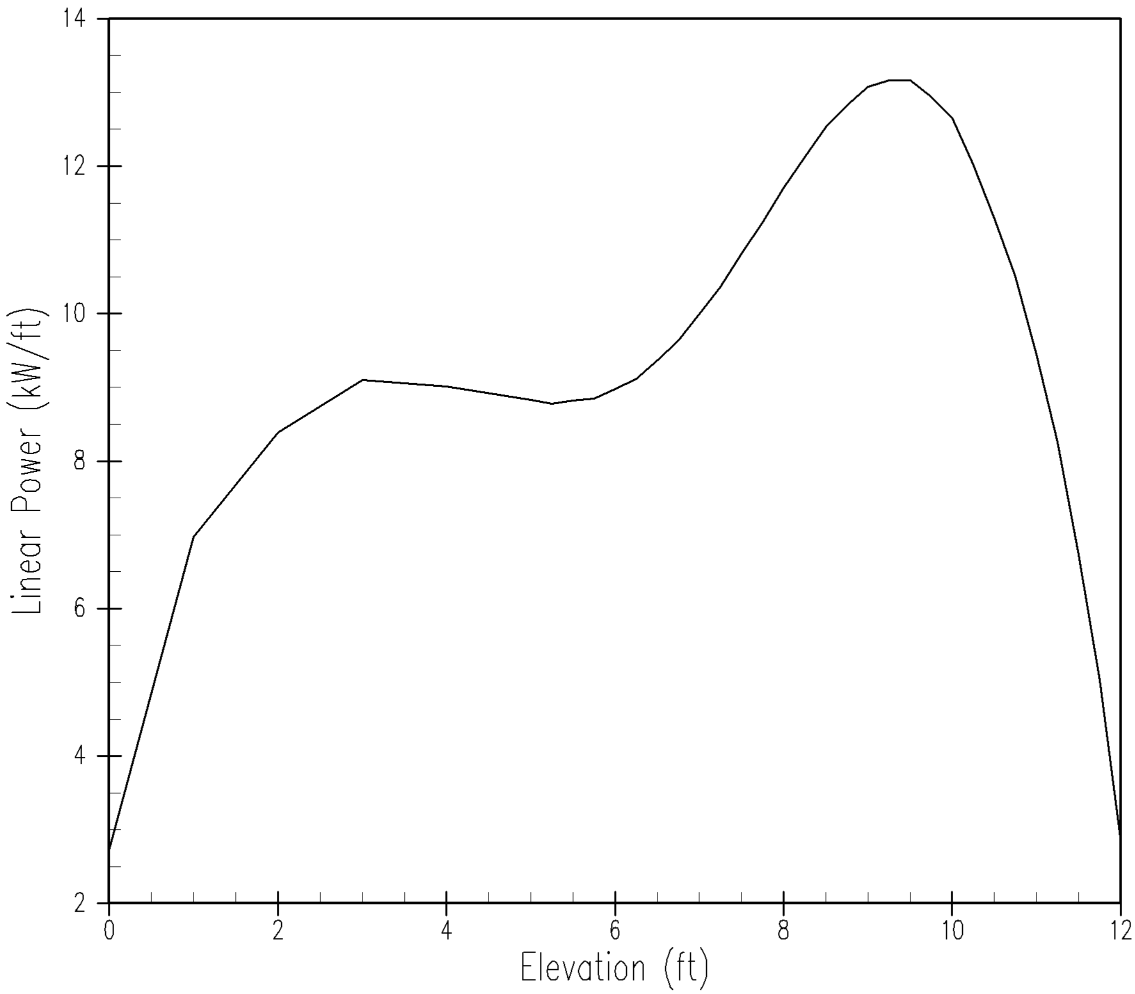

Figure 15.6-41

Small Break Hot Rod Power

Beaver Valley Power Station Unit No. 2
Updated Final Safety Analysis Report

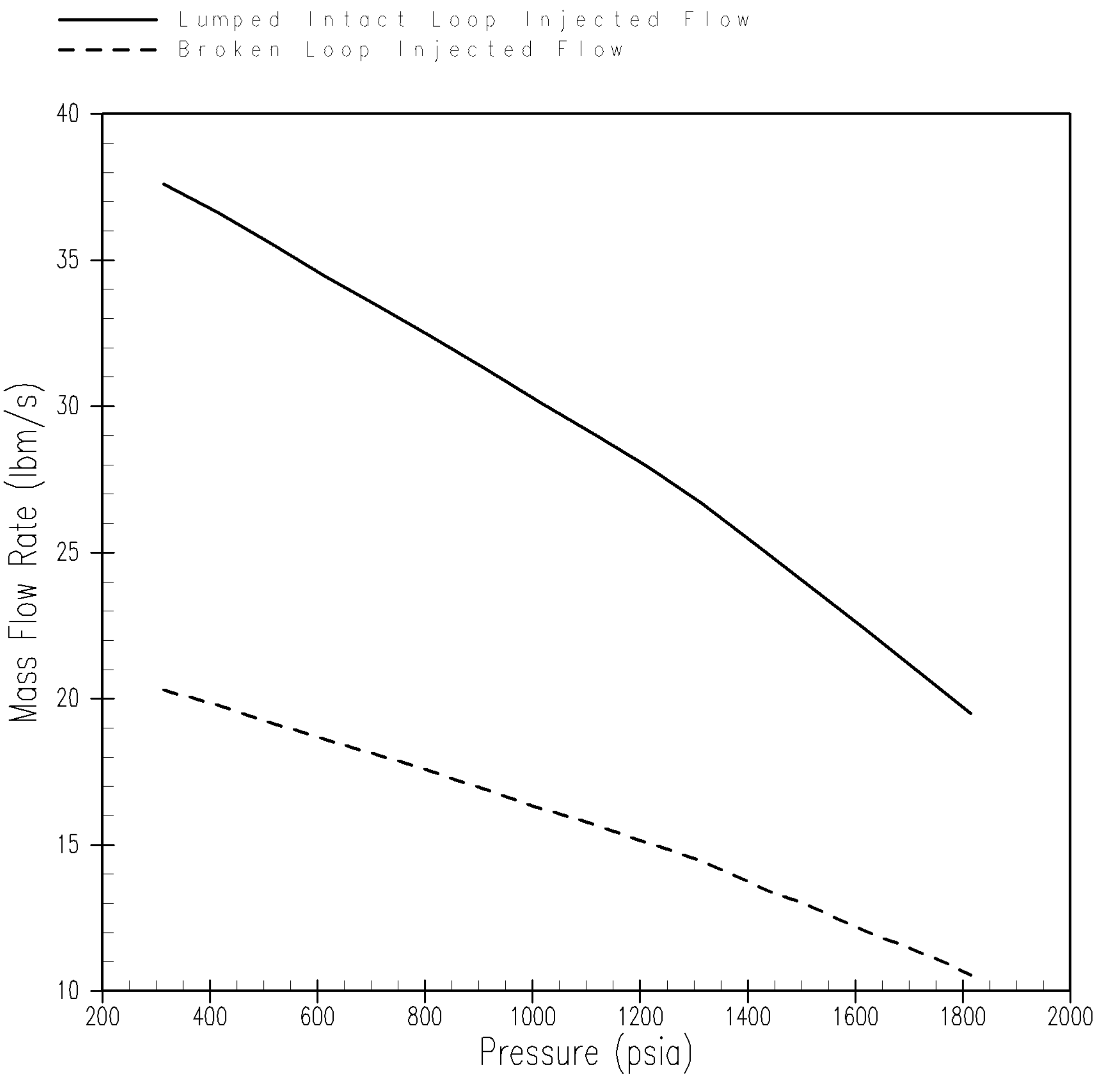

Figure 15.6-42a

Small Break LOCA Safety Injection Flows (1 HHSI pump, faulted loop injects to RCS pressure – 1.5 inch – 4-inch breaks)

Beaver Valley Power Station Unit No. 2 Updated Final Safety Analysis Report

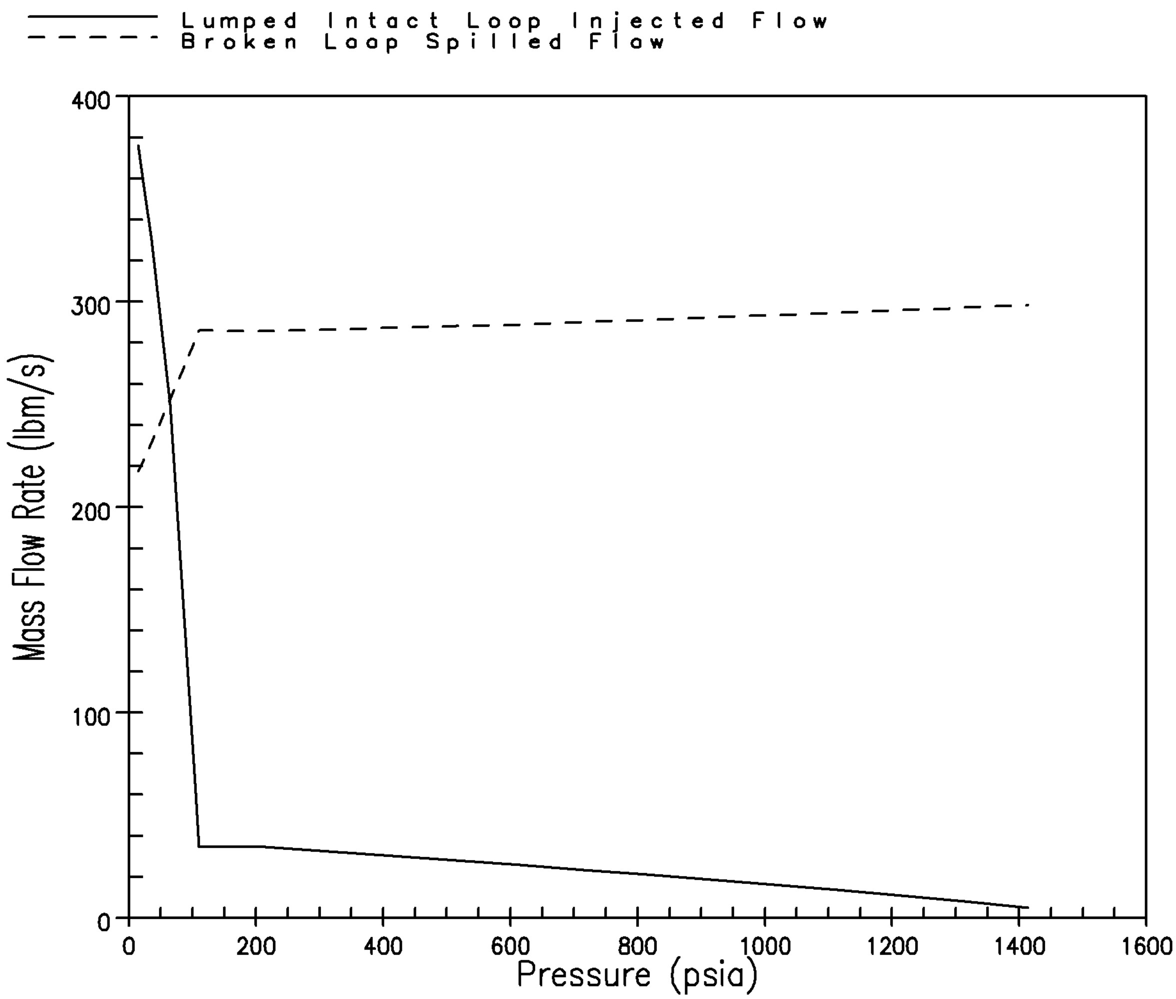

Figure 15.6-42b

Small Break LOCA Safety Injection Flows (1 HHSI pump, 1 LHSI pump, faulted loop spills to containment pressure (0 psig) because the break is postulated along the HHSI line – 6-inch break)

Beaver Valley Power Station Unit No. 2
Updated Final Safety Analysis Report

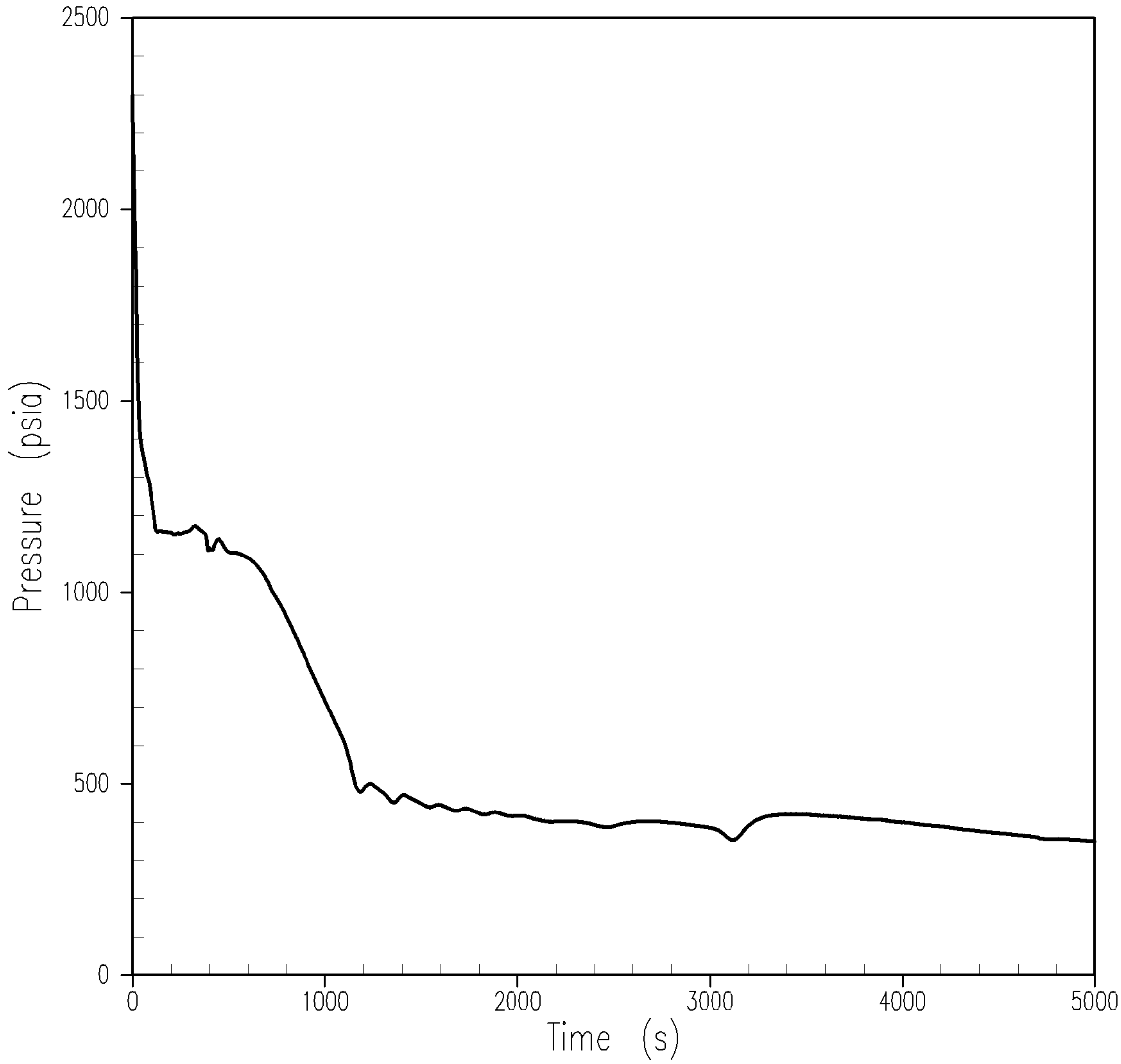

Figure 15.6-43

RCS Pressure (3-inch Break)

Beaver Valley Power Station Unit No. 2
Updated Final Safety Analysis Report

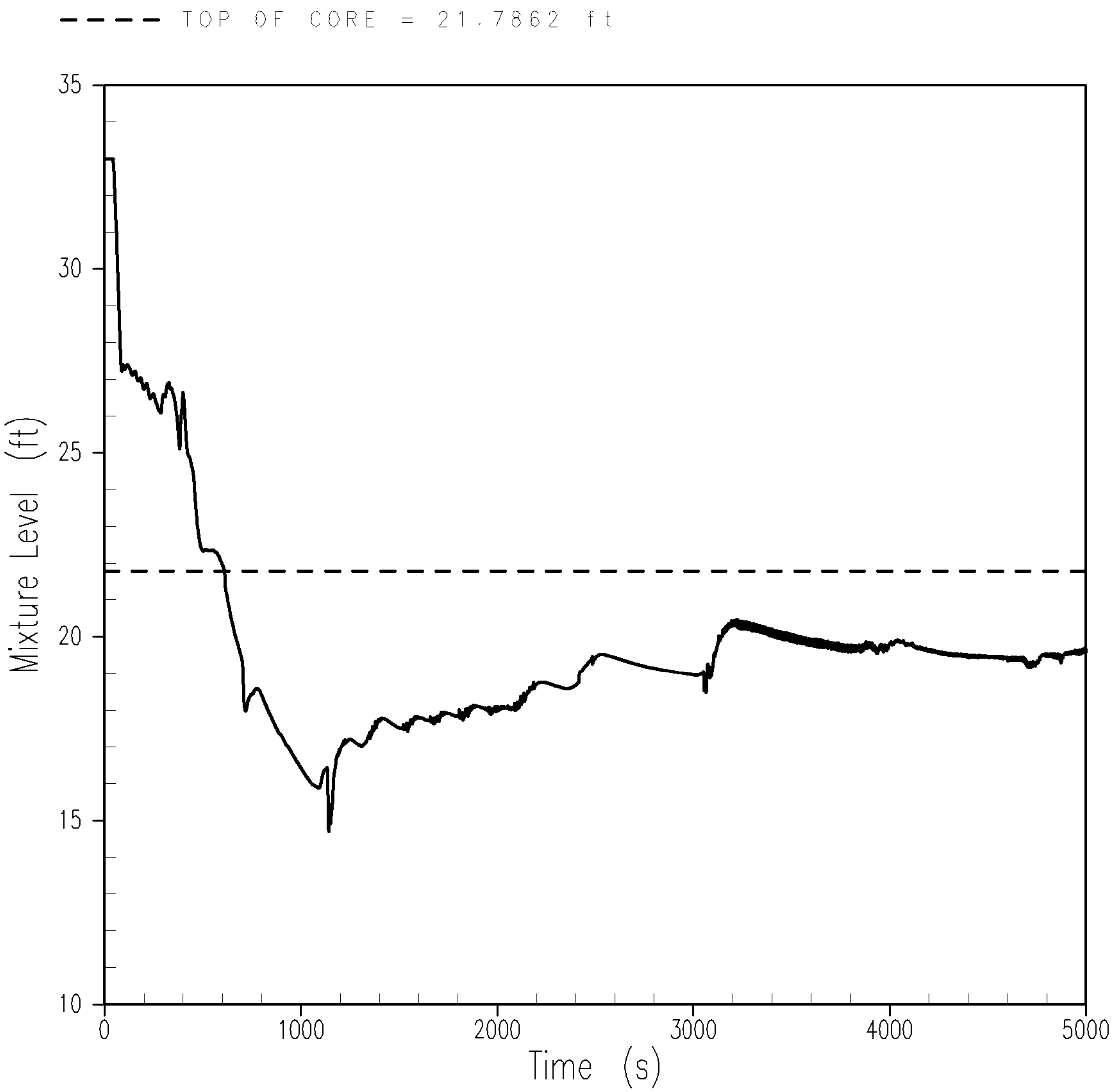

Figure 15.6-44

Core Mixture Height (3-inch Break)

Beaver Valley Power Station Unit No. 2
Updated Final Safety Analysis Report

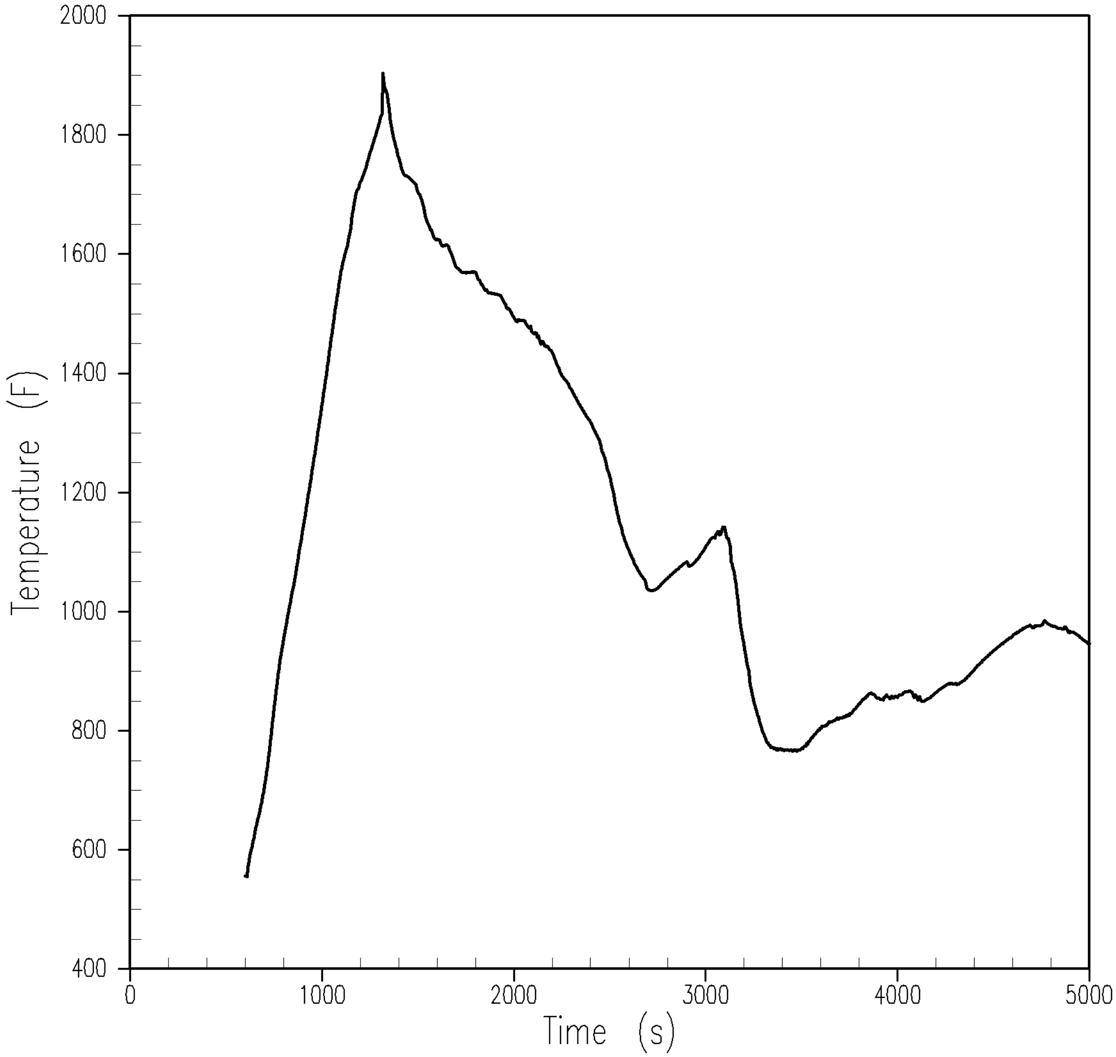

Figure 15.6-45

Peak Clad Temperature (3-inch Break)

Beaver Valley Power Station Unit No. 2
Updated Final Safety Analysis Report

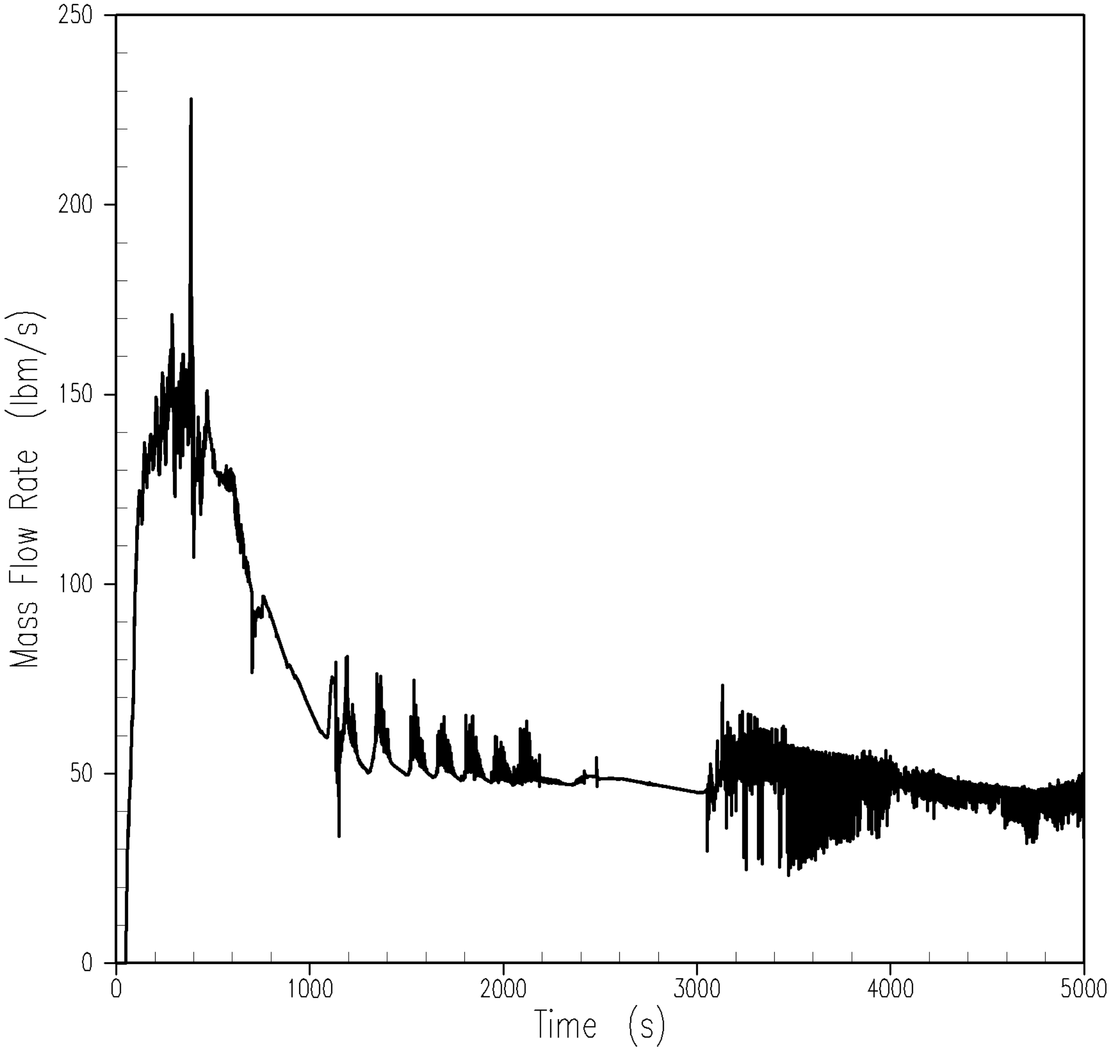

Figure 15.6-46

Core Steam Flowrate (3-inch Break)

Beaver Valley Power Station Unit No. 2
Updated Final Safety Analysis Report

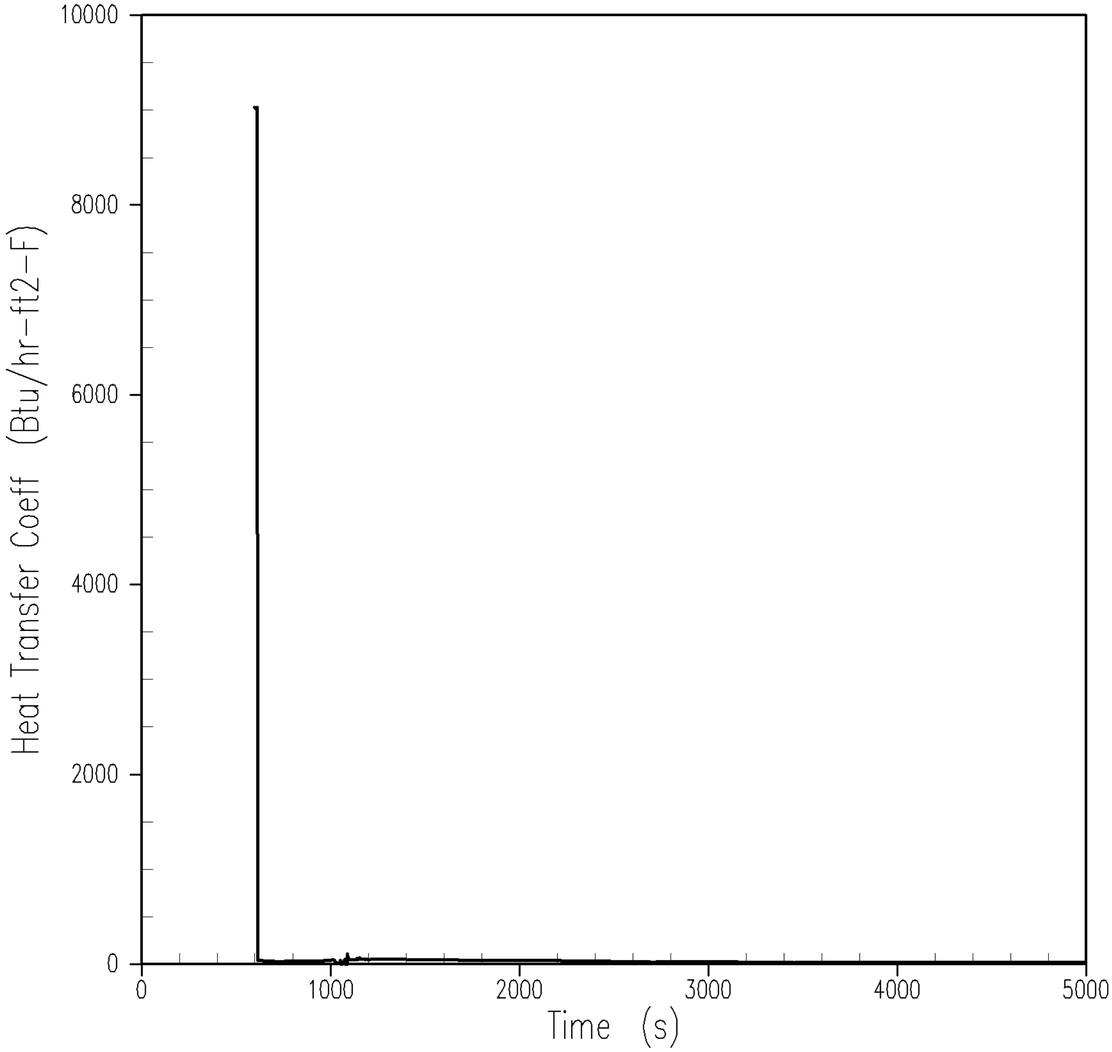

Figure 15.6-47

Hot Spot Heat Transfer Coefficient (3-inch Break)

Beaver Valley Power Station Unit No. 2
Updated Final Safety Analysis Report

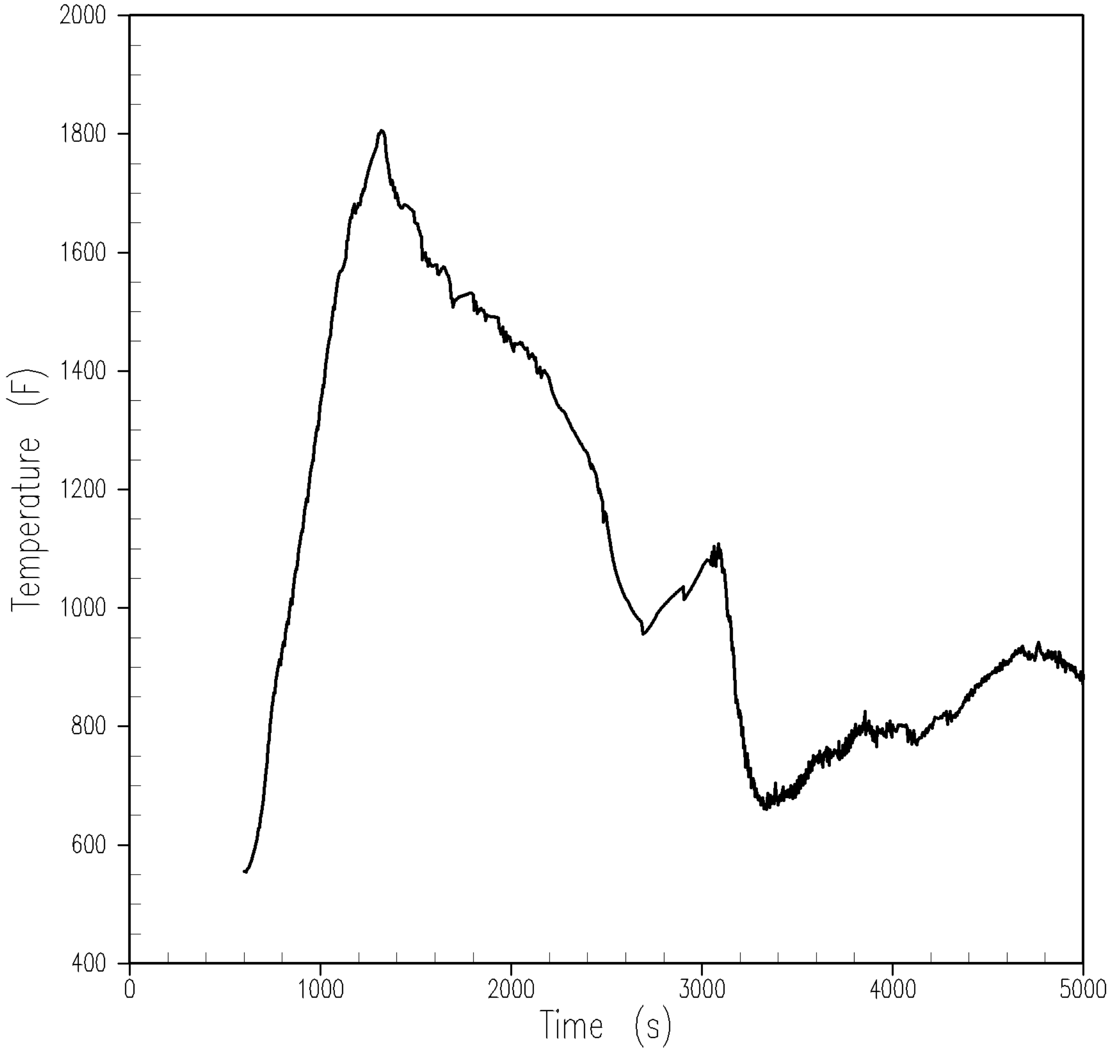

Figure 15.6-48

Hot Spot Fluid Temperature (3-inch Break)

Beaver Valley Power Station Unit No. 2 Updated Final Safety Analysis Report

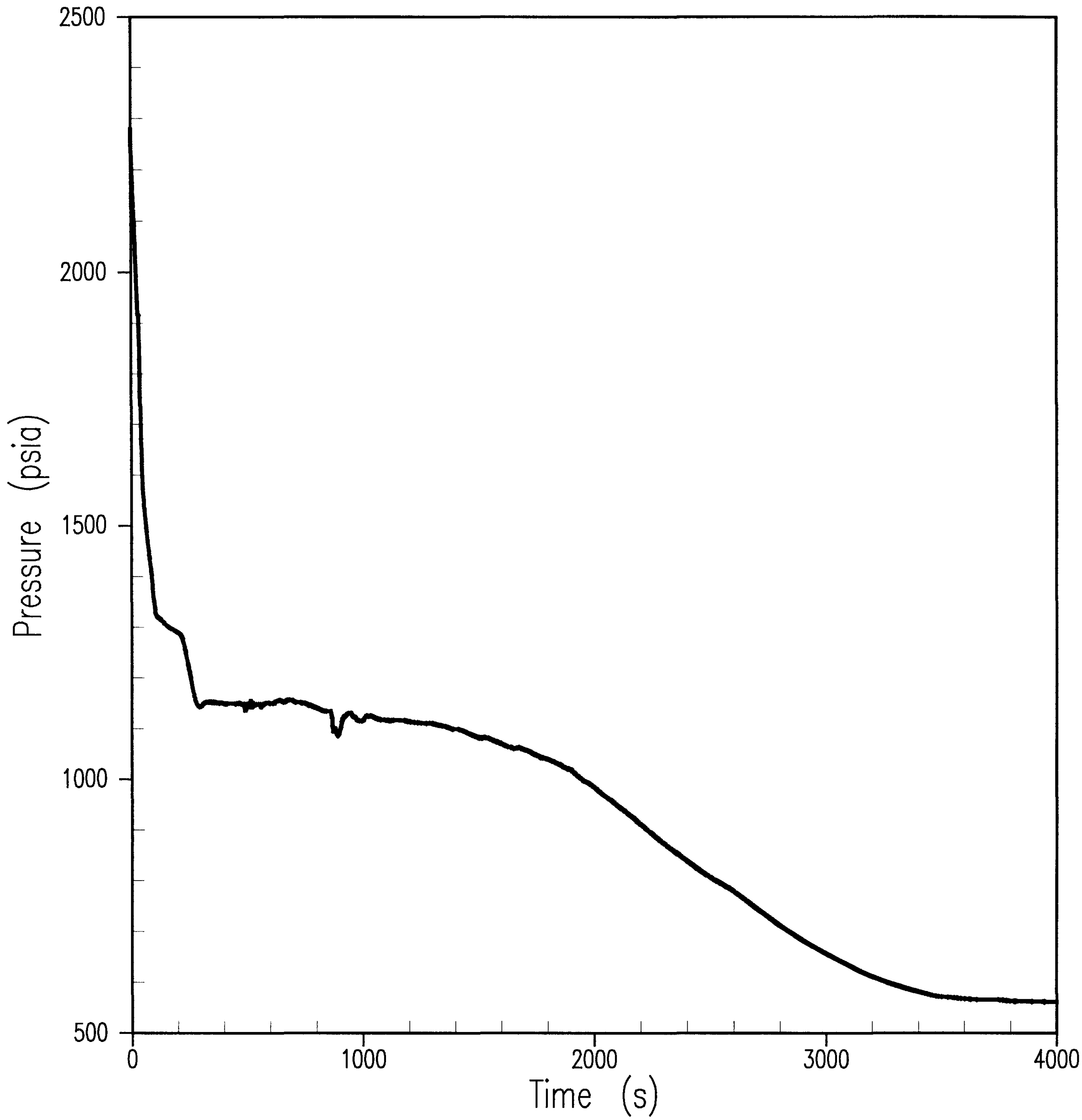

Figure 15.6-49
RCS Pressure (2-Inch Break)
BEAVER VALLEY POWER STATION-UNIT 2
UPDATED FINAL SAFETY ANALYSIS REPORT

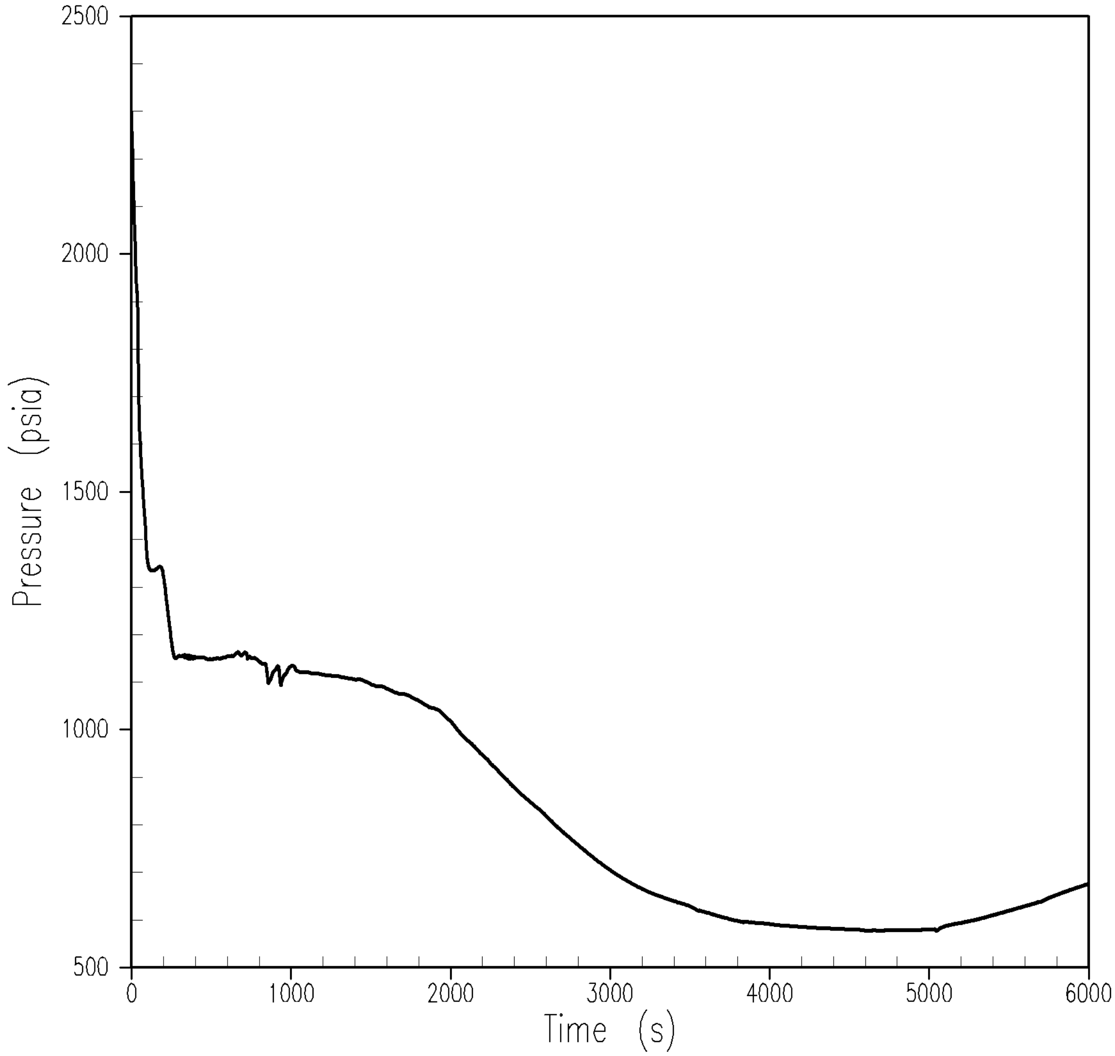

Figure 15.6-49A

RCS Pressure (2-inch Break)

Beaver Valley Power Station Unit No. 2
Updated Final Safety Analysis Report

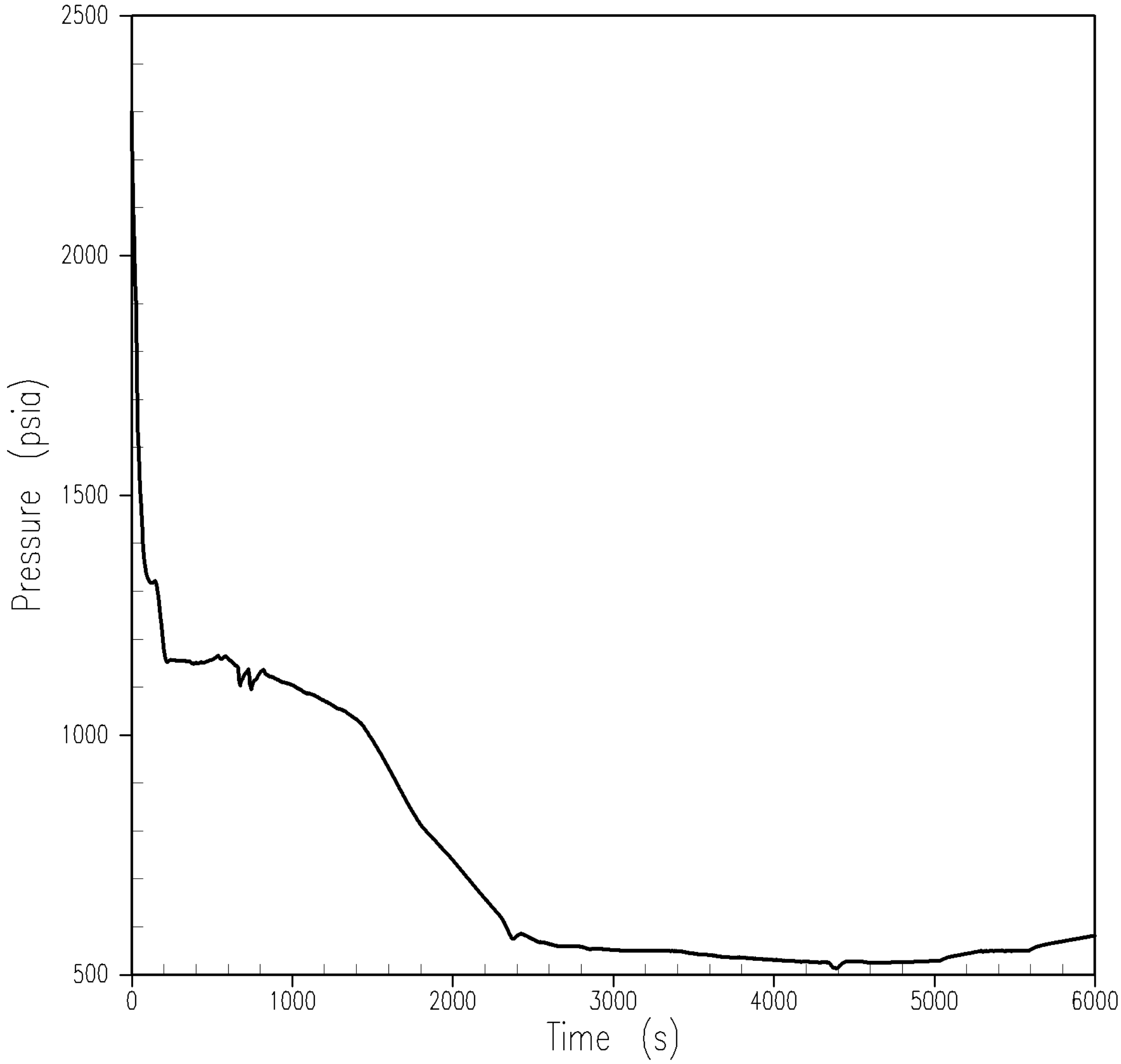

Figure 15.6-49B

RCS Pressure (2.25-inch Break)

Beaver Valley Power Station Unit No. 2
Updated Final Safety Analysis Report

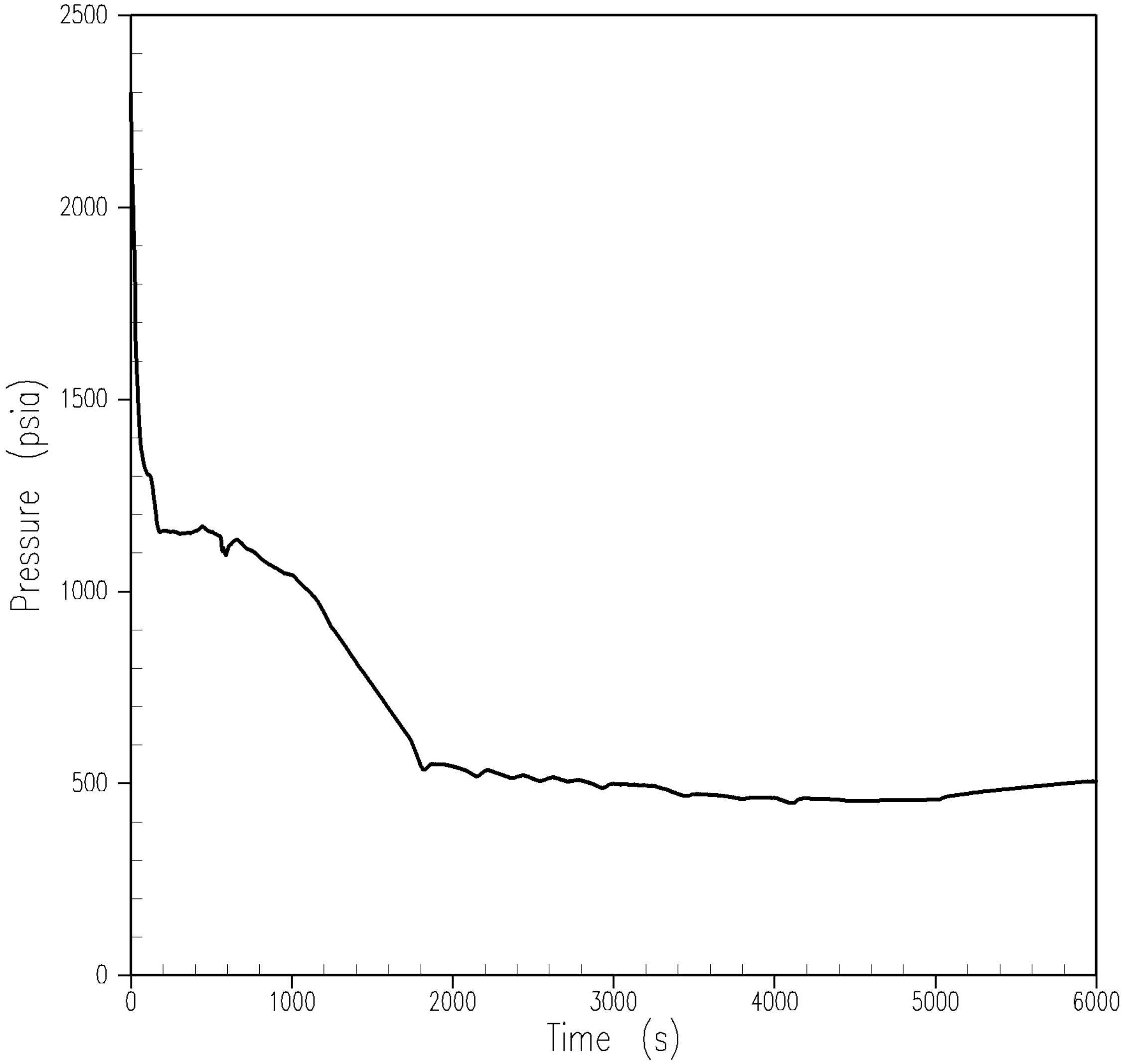

Figure 15.6-49C

RCS Pressure (2.5-inch Break)

Beaver Valley Power Station Unit No. 2
Updated Final Safety Analysis Report

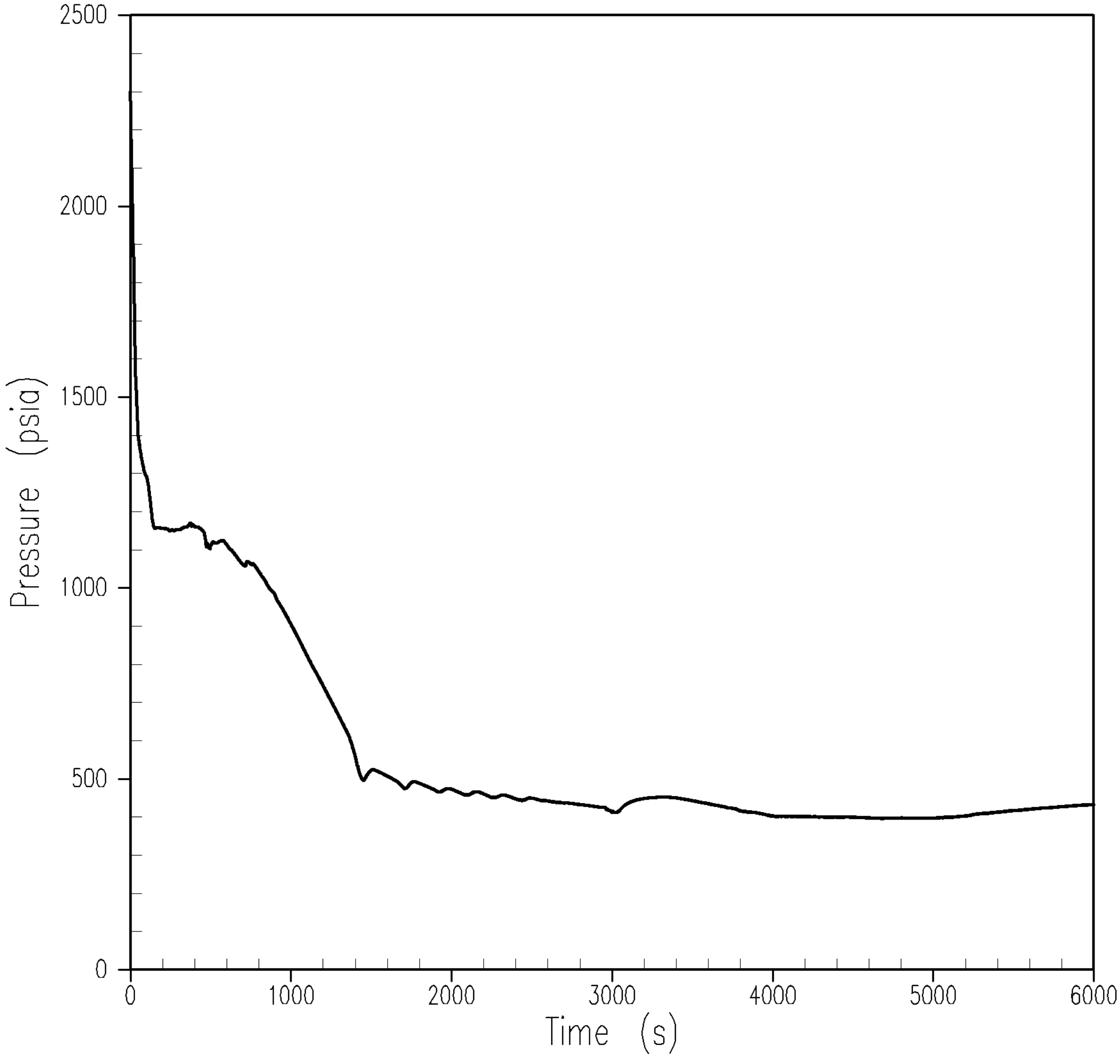

Figure 15.6-49D

RCS Pressure (2.75-inch Break)

Beaver Valley Power Station Unit No. 2
Updated Final Safety Analysis Report

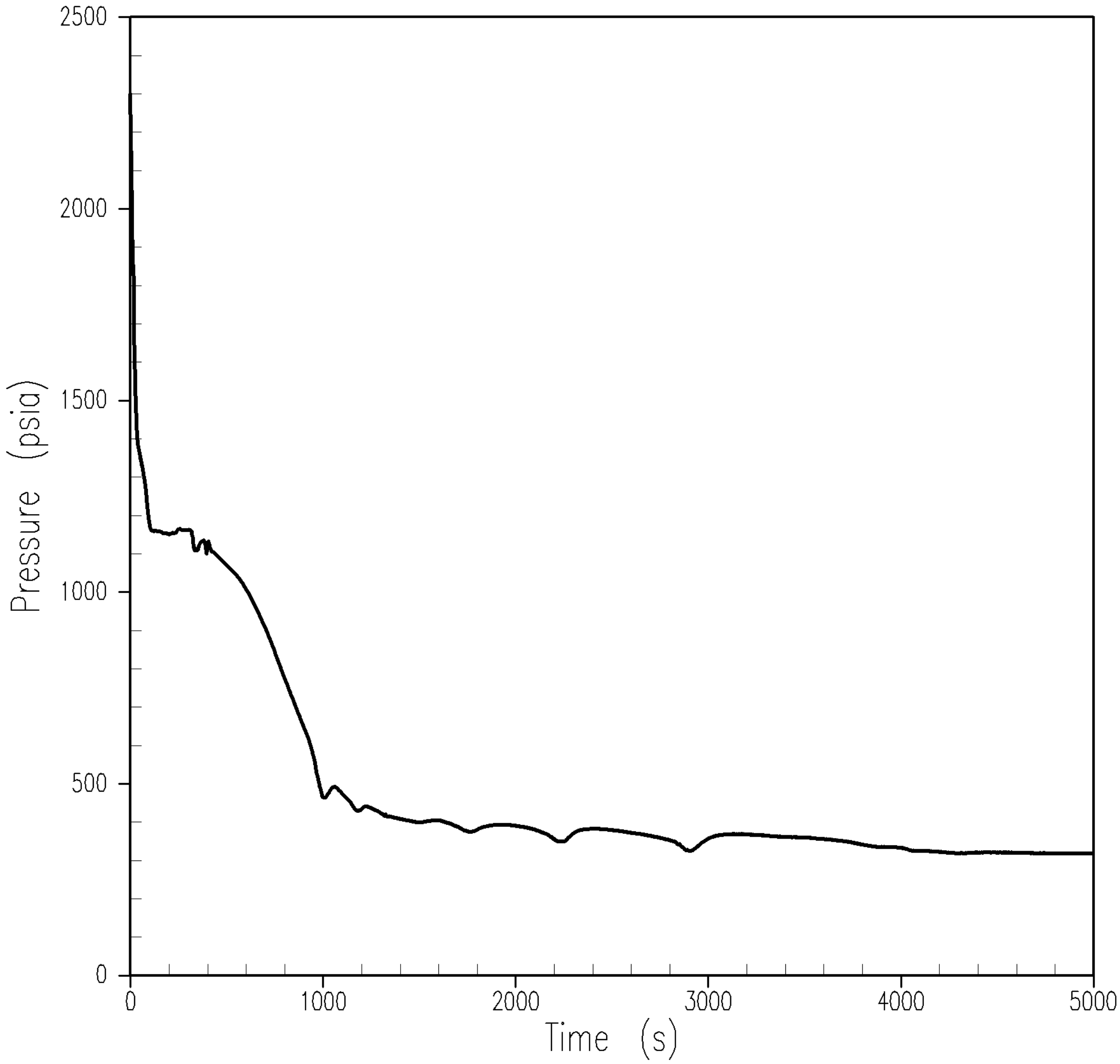

Figure 15.6-49E

RCS Pressure (3.25-inch Break)

Beaver Valley Power Station Unit No. 2
Updated Final Safety Analysis Report

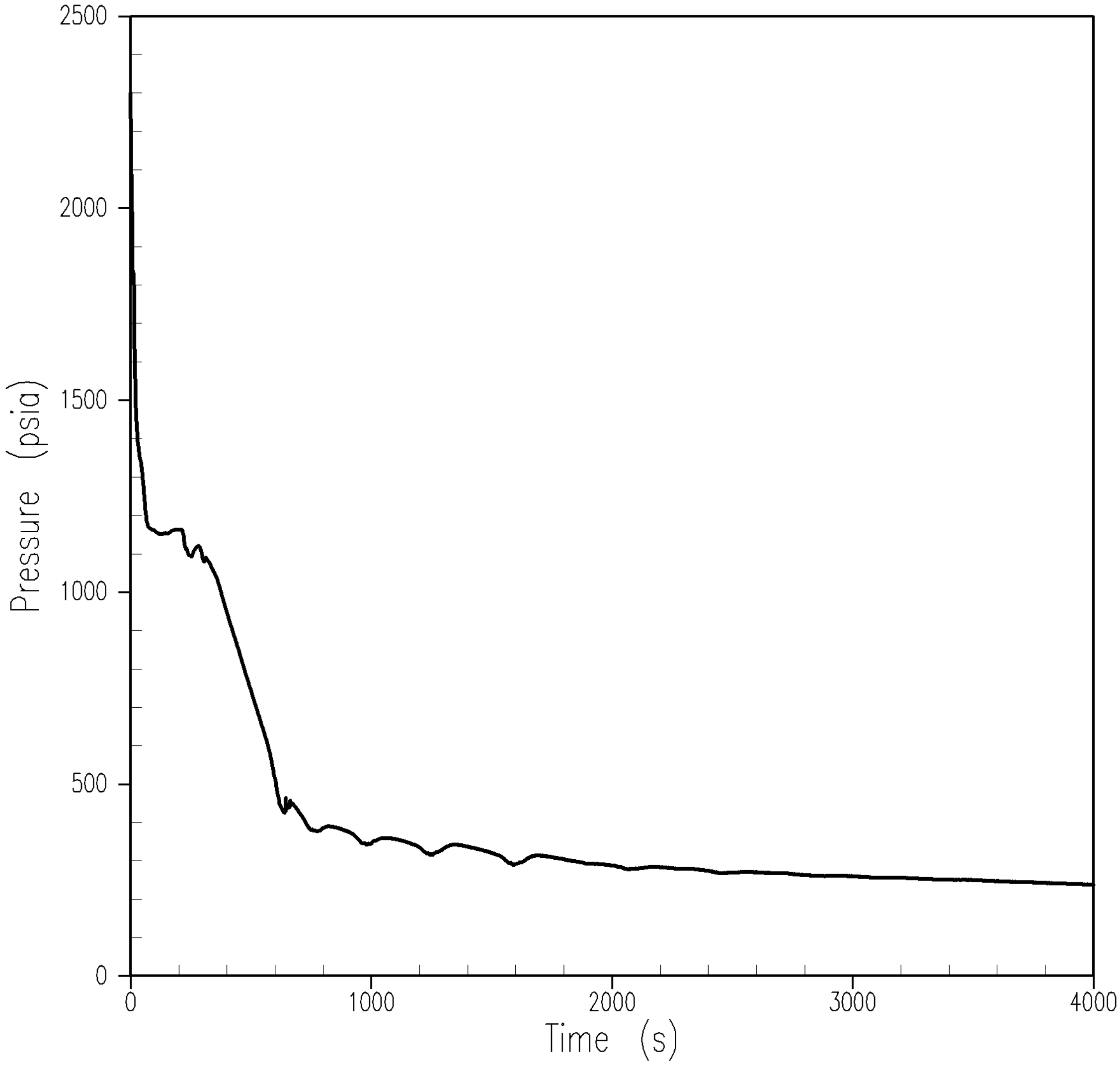

Figure 15.6-49F

RCS Pressure (4-inch Break)

Beaver Valley Power Station Unit No. 2
Updated Final Safety Analysis Report

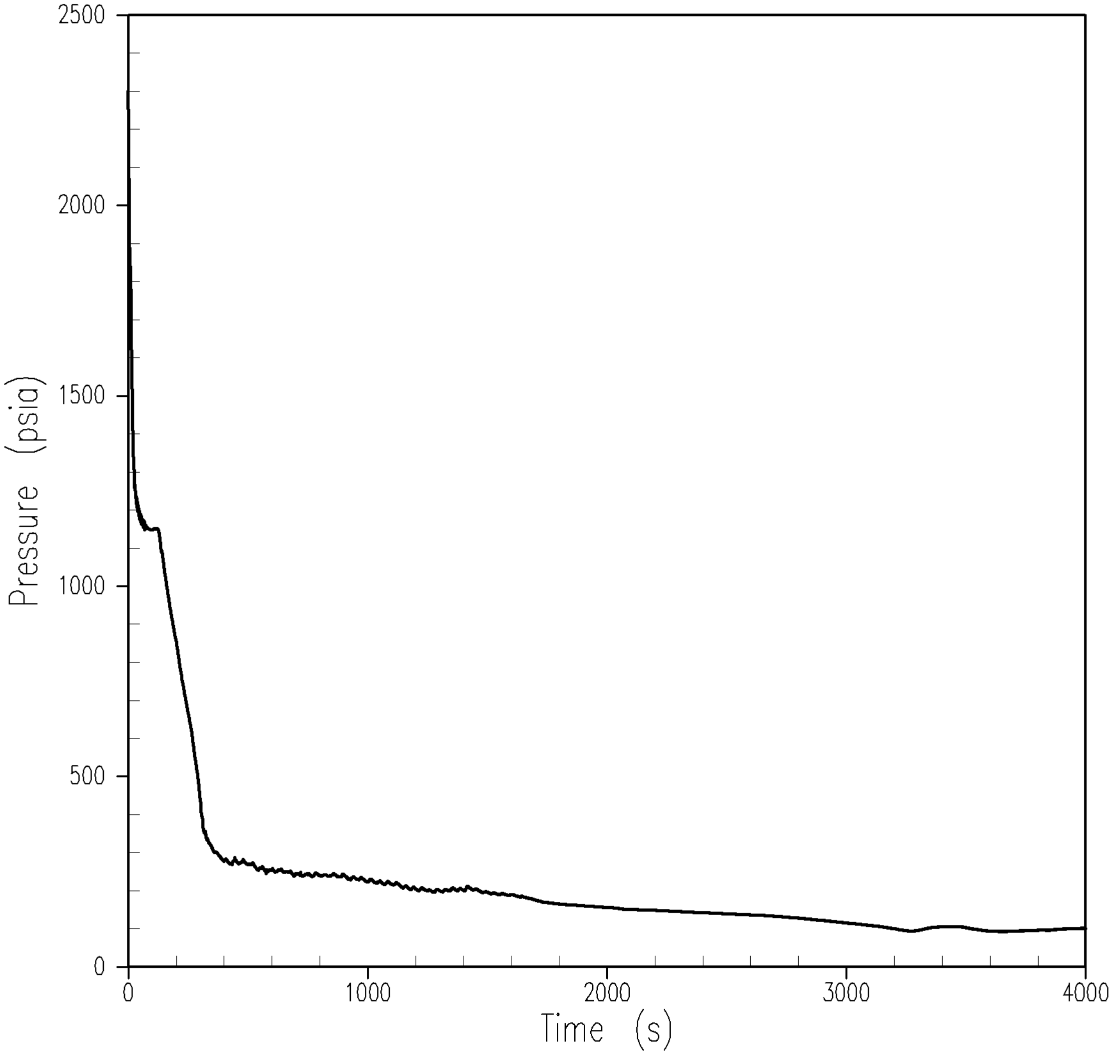

Figure 15.6-49G

RCS Pressure (6-inch Break)

Beaver Valley Power Station Unit No. 2
Updated Final Safety Analysis Report

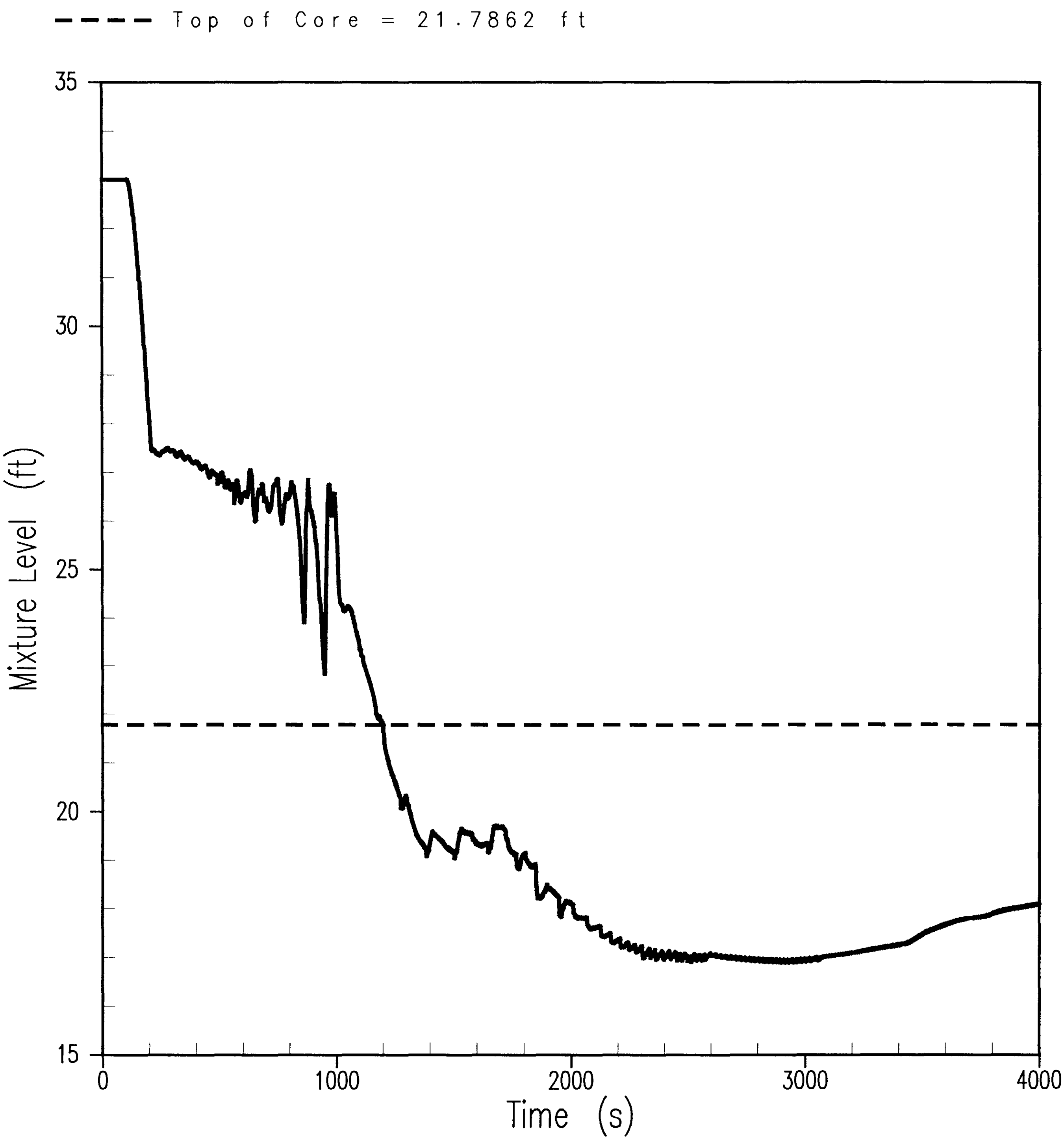

Figure 15.6-50
Core Mixture Height (2-Inch Break)
BEAVER VALLEY POWER STATION-UNIT 2
UPDATED FINAL SAFETY ANALYSIS REPORT

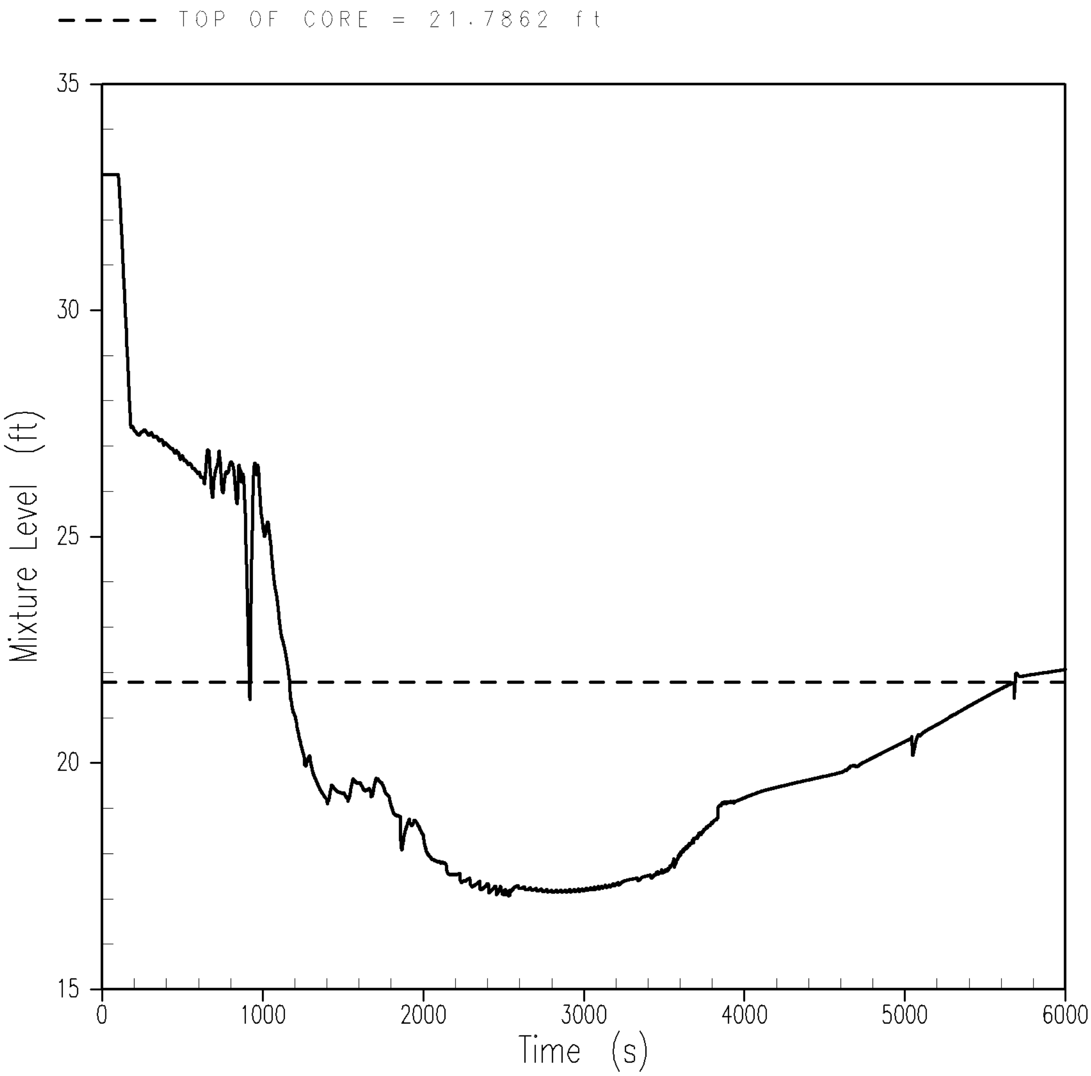

Figure 15.6-50A

Core Mixture Height (2-inch Break)

Beaver Valley Power Station Unit No. 2
Updated Final Safety Analysis Report

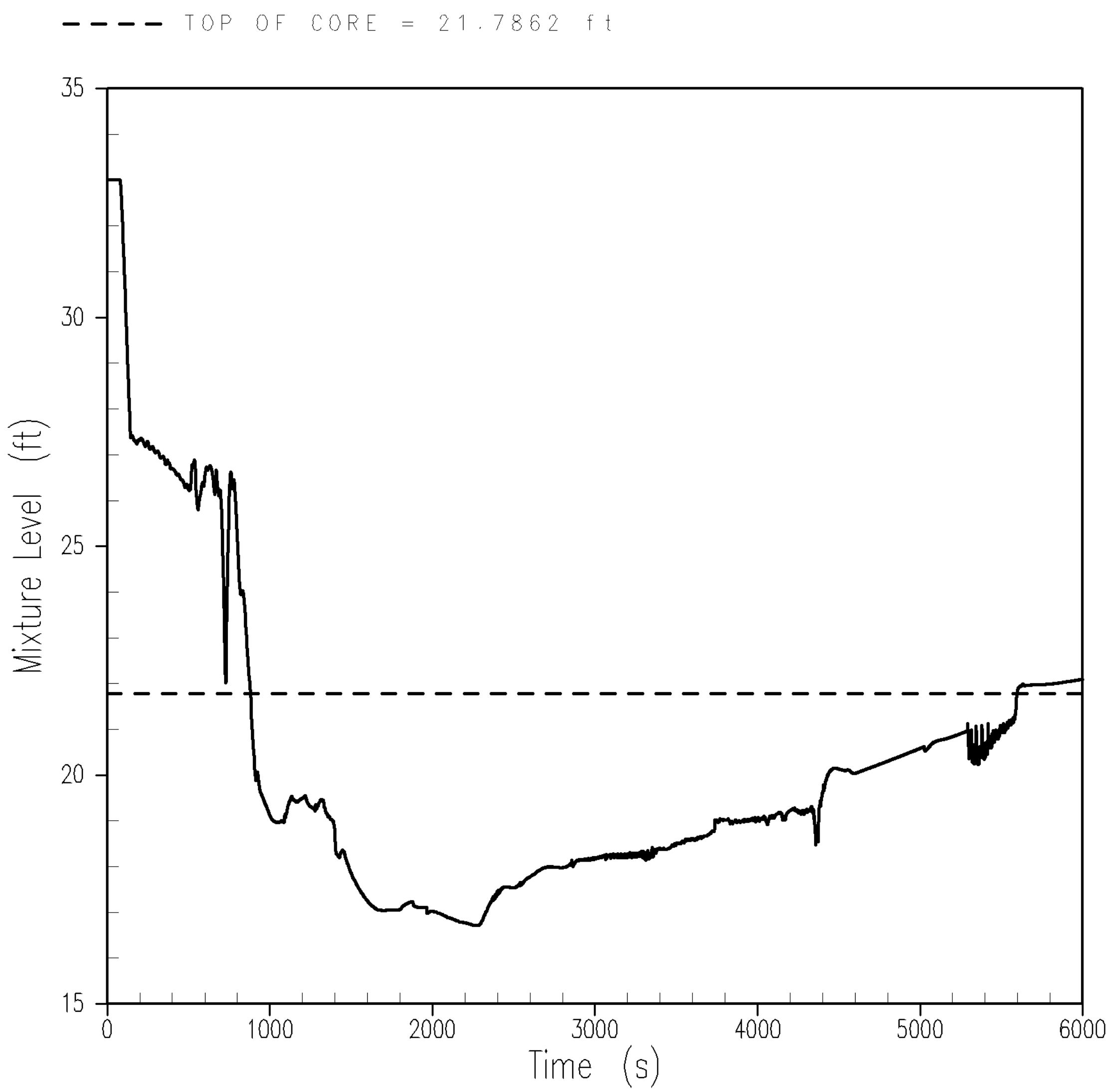

Figure 15.6-50B

Core Mixture Height (2.25-inch Break)

Beaver Valley Power Station Unit No. 2
Updated Final Safety Analysis Report

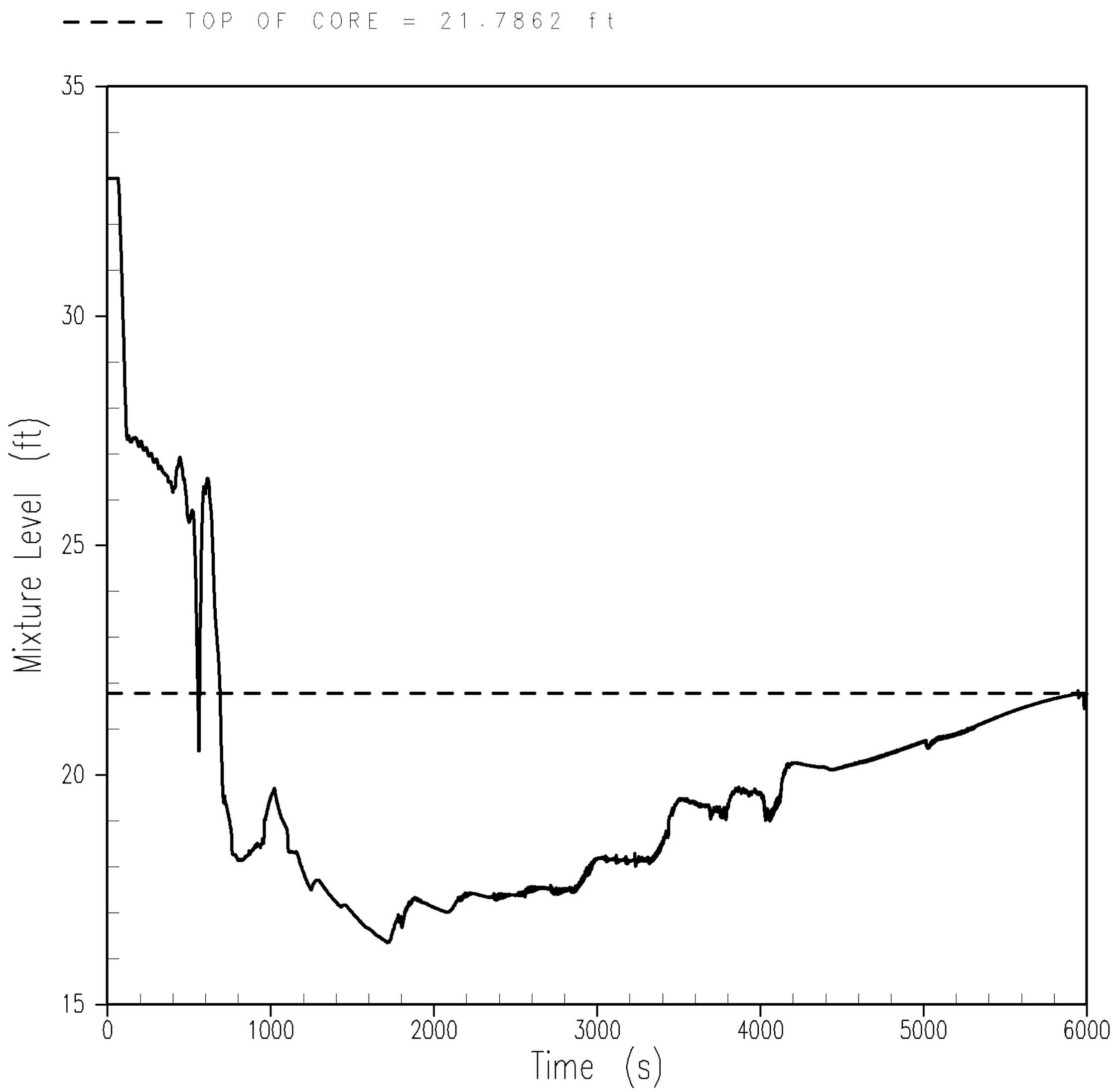

Figure 15.6-50C

Core Mixture Height (2.5-inch Break)

Beaver Valley Power Station Unit No. 2
Updated Final Safety Analysis Report

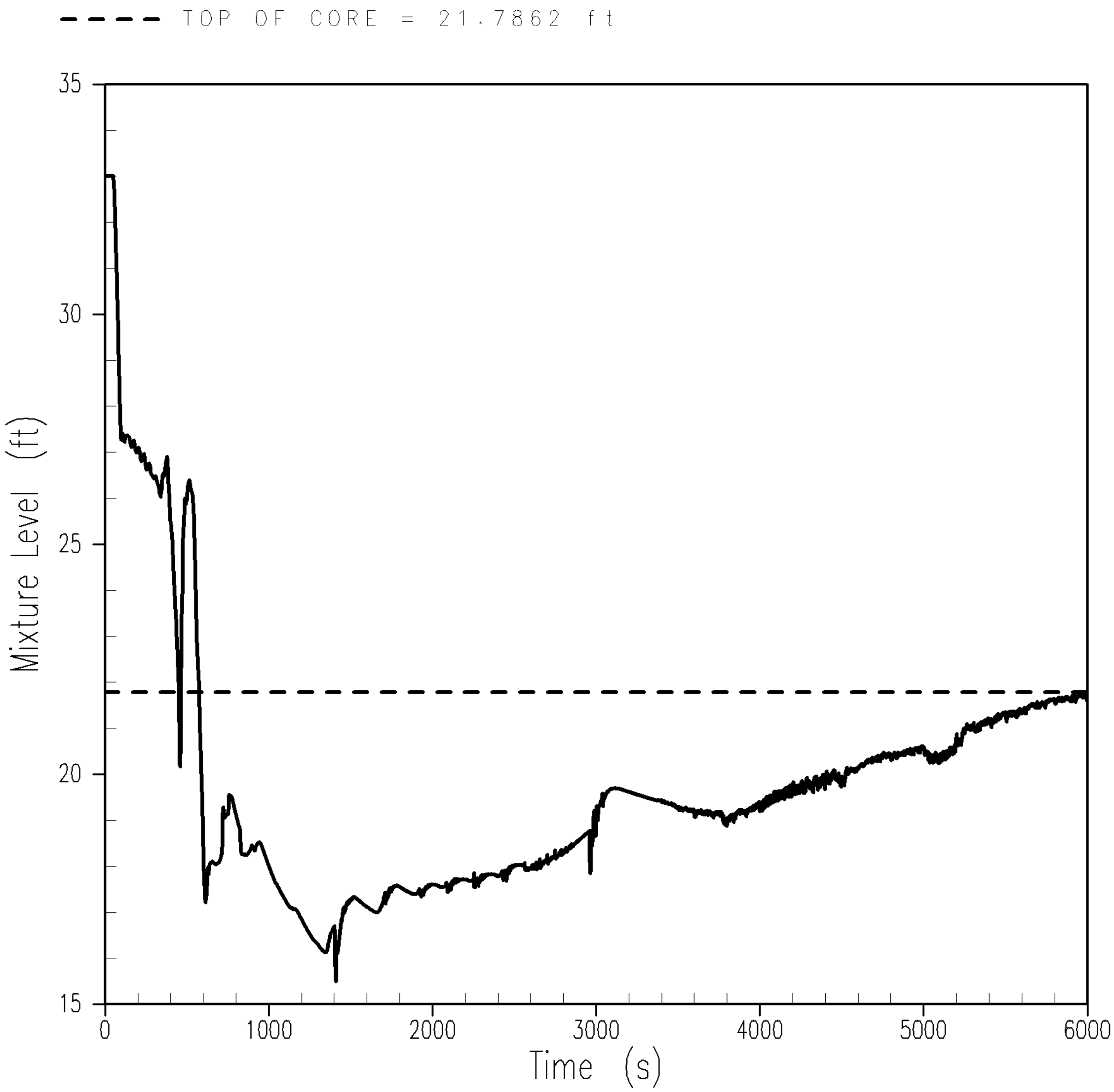

Figure 15.6-50D

Core Mixture Height (2.75-inch Break)

Beaver Valley Power Station Unit No. 2
Updated Final Safety Analysis Report

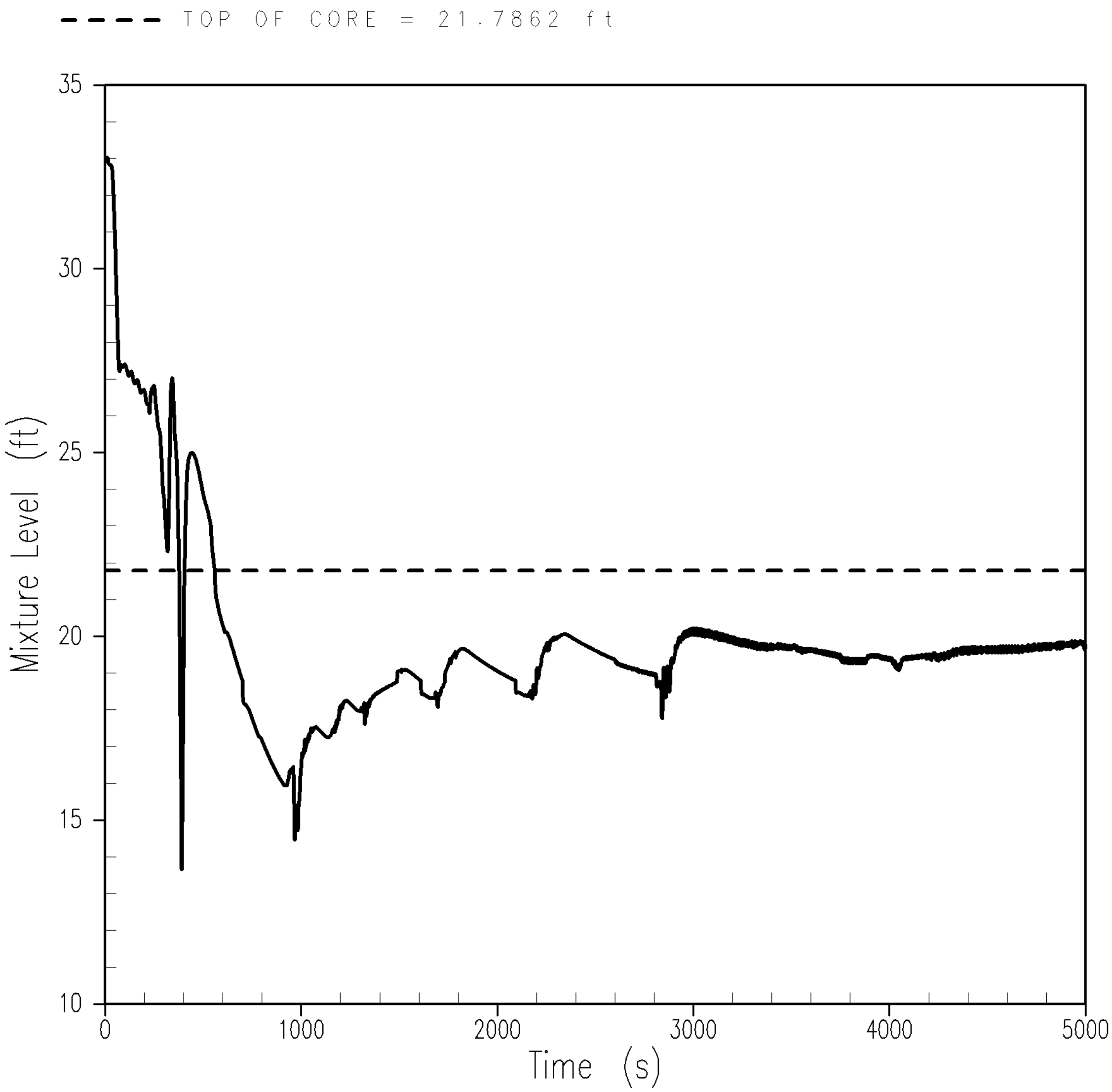

Figure 15.6-50E

Core Mixture Height (3.25-inch Break)

Beaver Valley Power Station Unit No. 2
Updated Final Safety Analysis Report

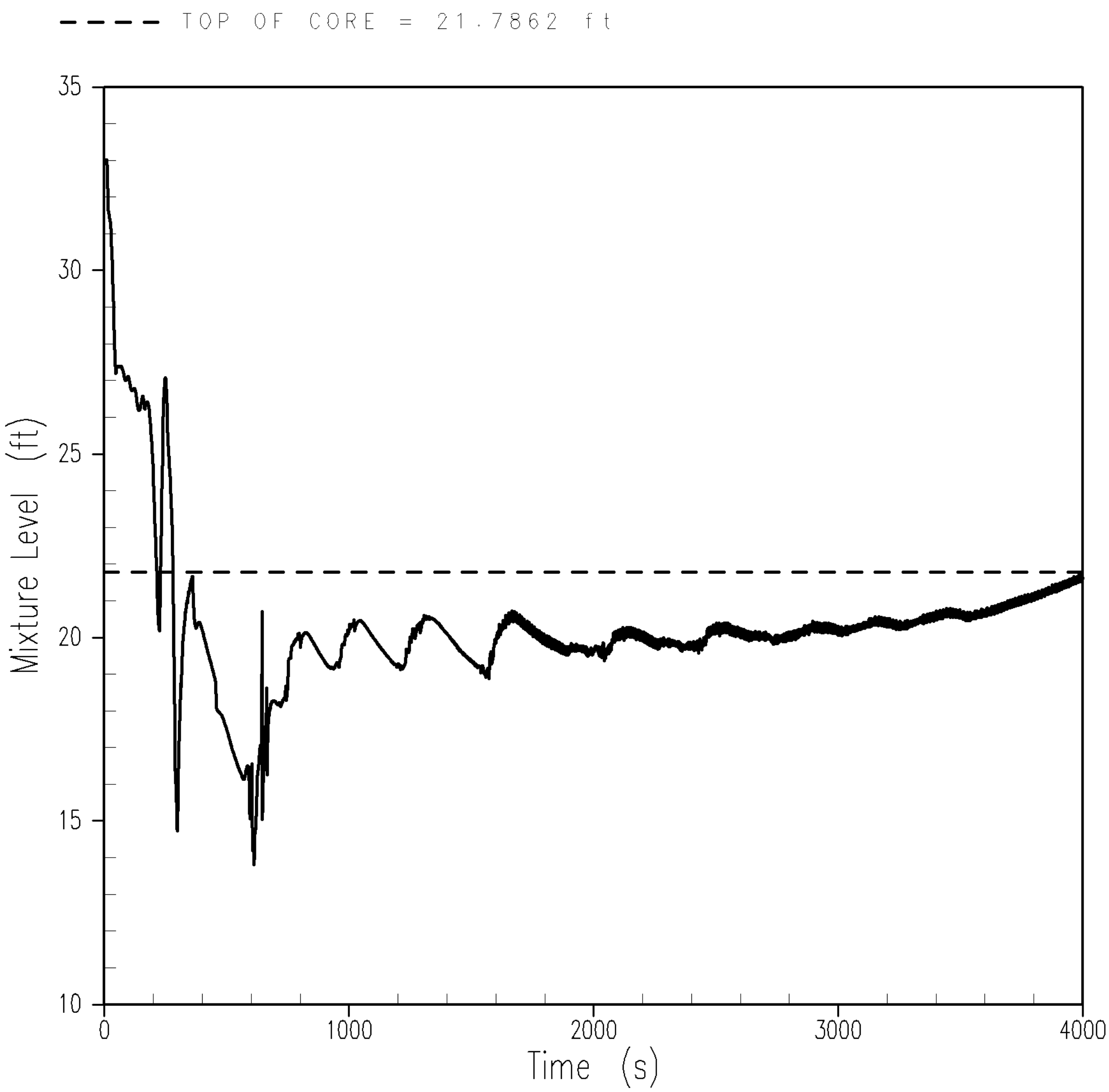

Figure 15.6-50F

Core Mixture Height (4-inch Break)

Beaver Valley Power Station Unit No. 2
Updated Final Safety Analysis Report

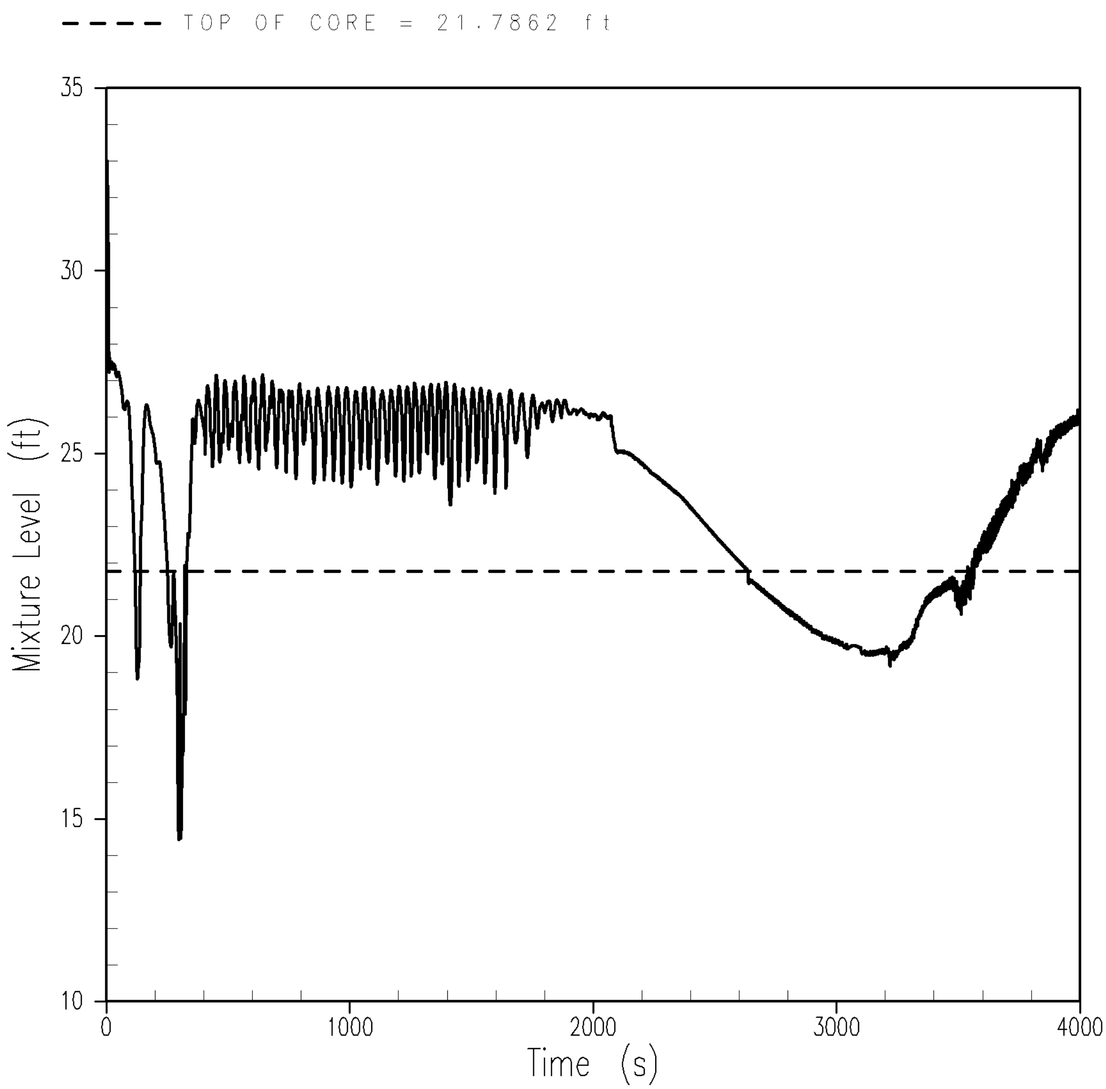

Figure 15.6-50G

Core Mixture Height (6-inch Break)

Beaver Valley Power Station Unit No. 2
Updated Final Safety Analysis Report

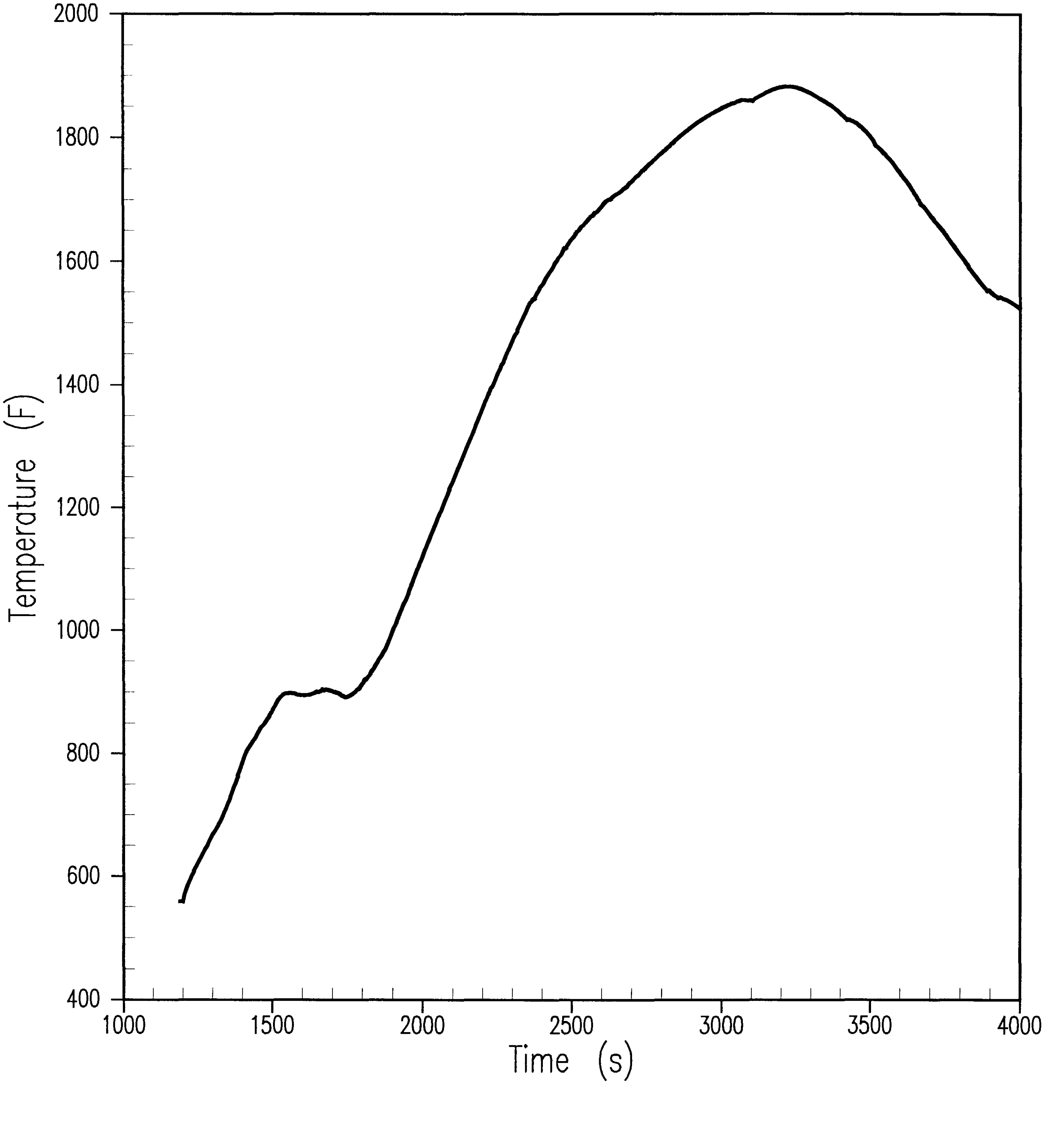

Figure 15.6-51
Hot Spot Clad Temperature (2-Inch Break)
BEAVER VALLEY POWER STATION-UNIT 2
UPDATED FINAL SAFETY ANALYSIS REPORT

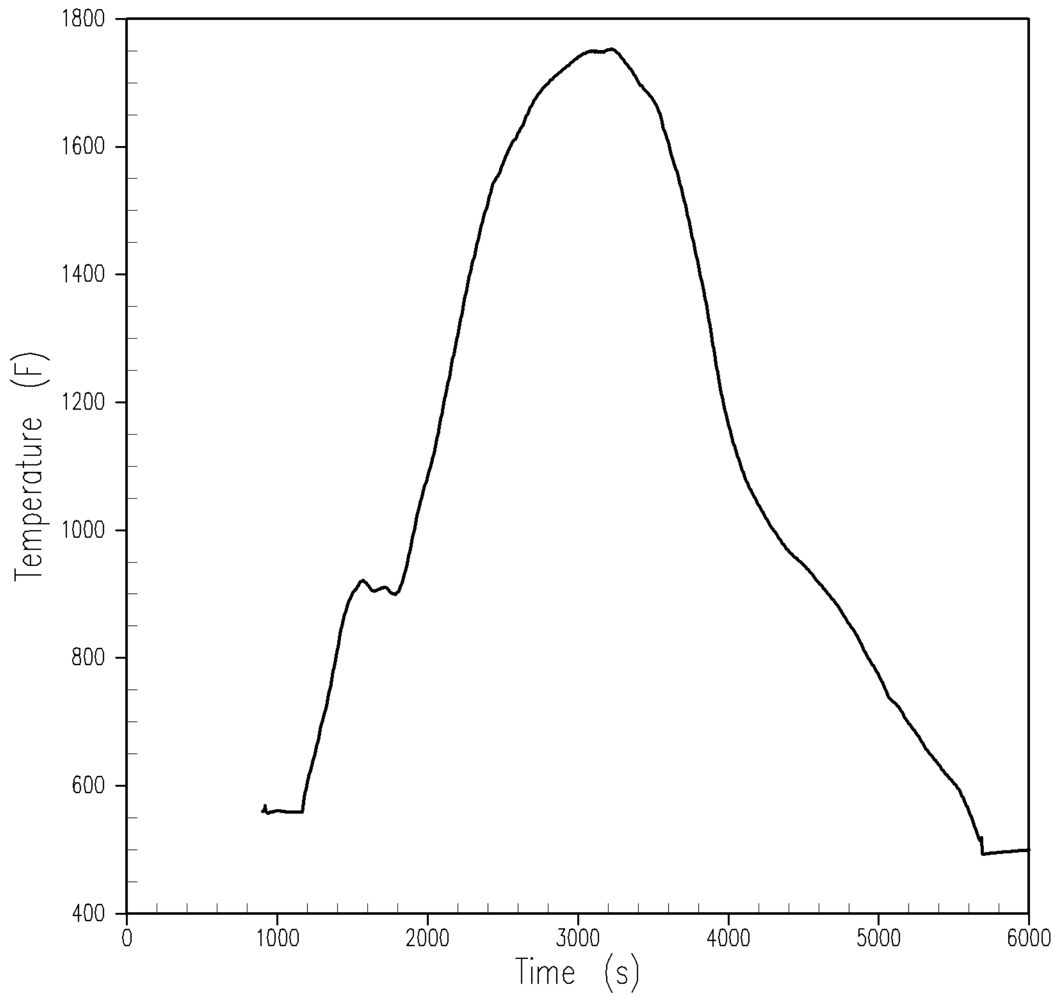

Figure 15.6-51A

Peak Clad Temperature (2-inch Break)

Beaver Valley Power Station Unit No. 2
Updated Final Safety Analysis Report

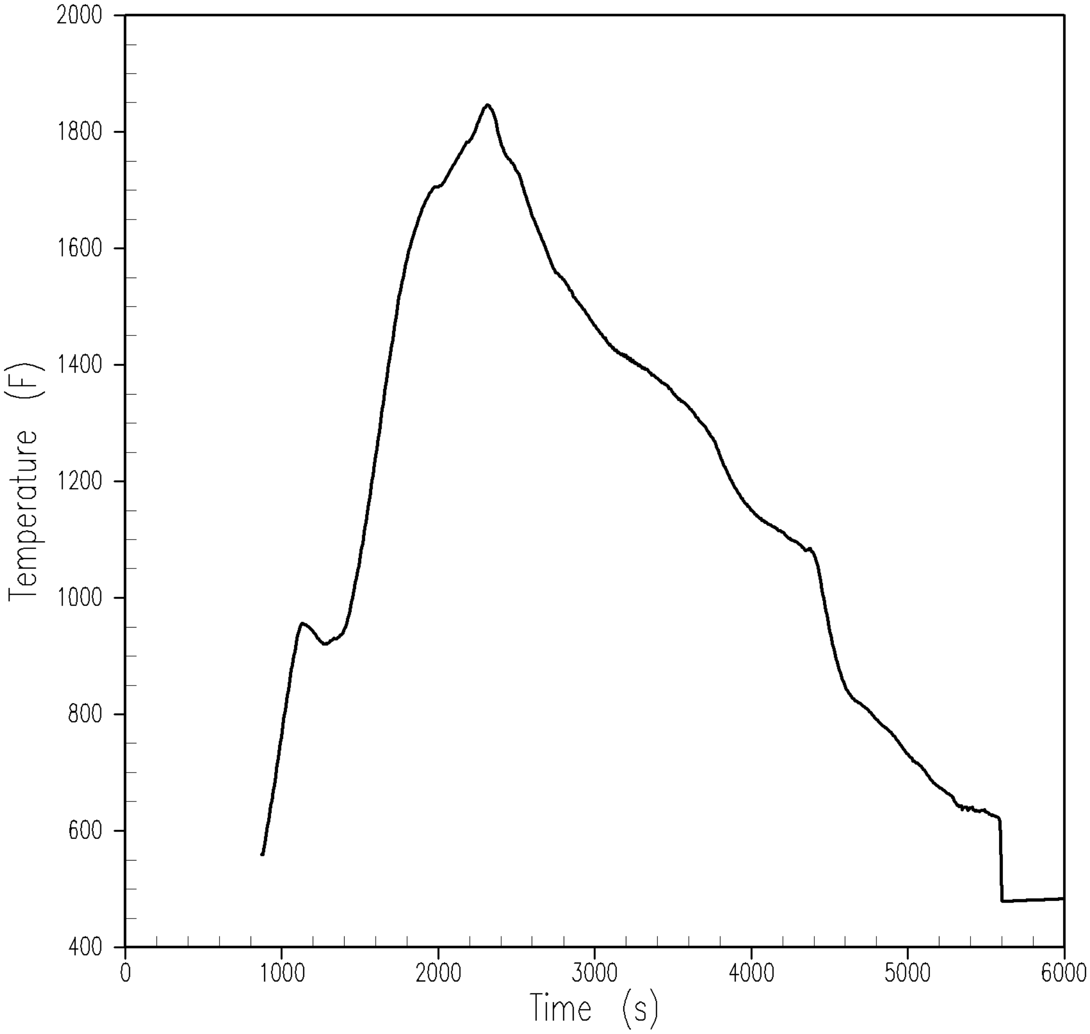

Figure 15.6-51B

Peak Clad Temperature (2.25-inch Break)

Beaver Valley Power Station Unit No. 2
Updated Final Safety Analysis Report

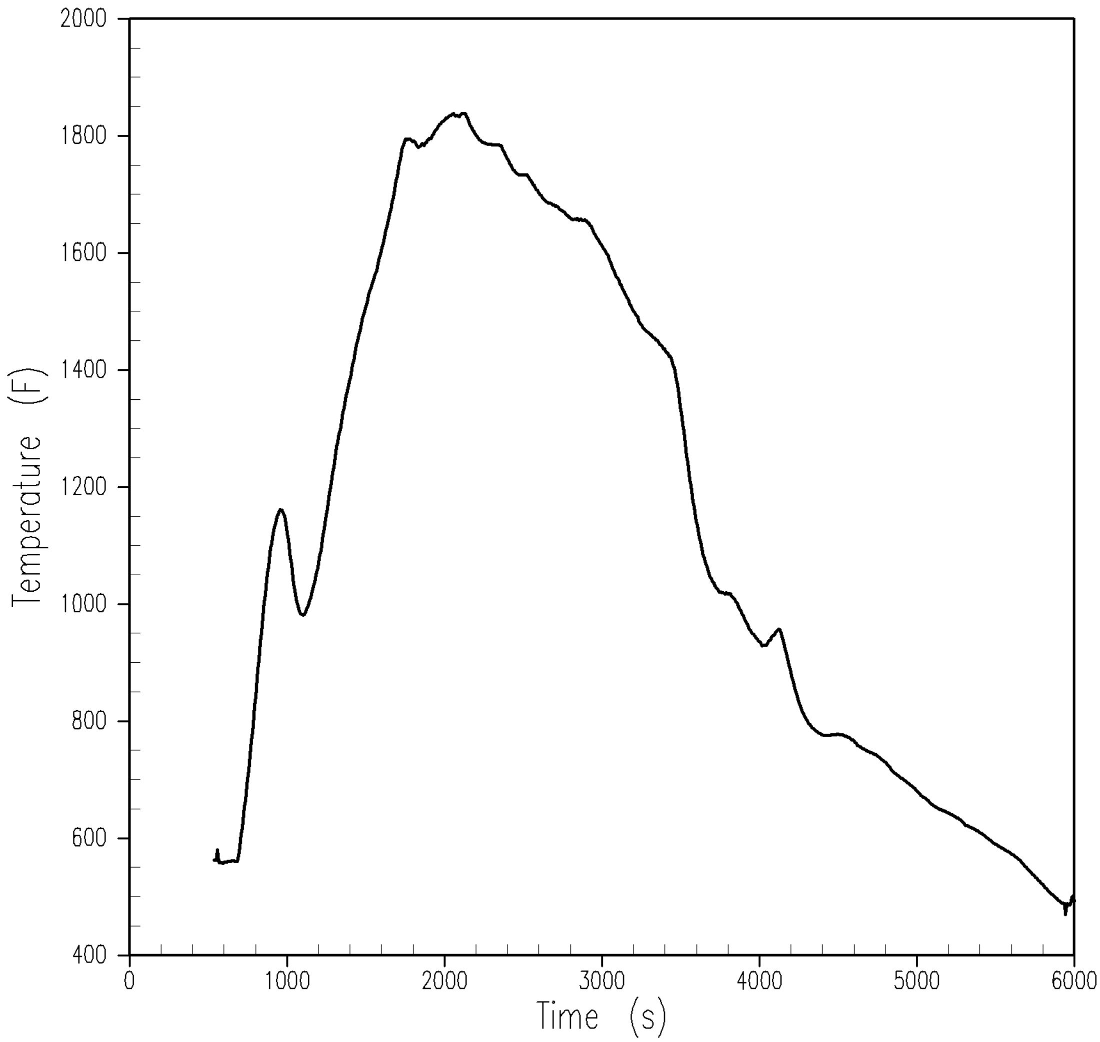

Figure 15.6-51C

Peak Clad Temperature (2.5-inch Break)

Beaver Valley Power Station Unit No. 2
Updated Final Safety Analysis Report

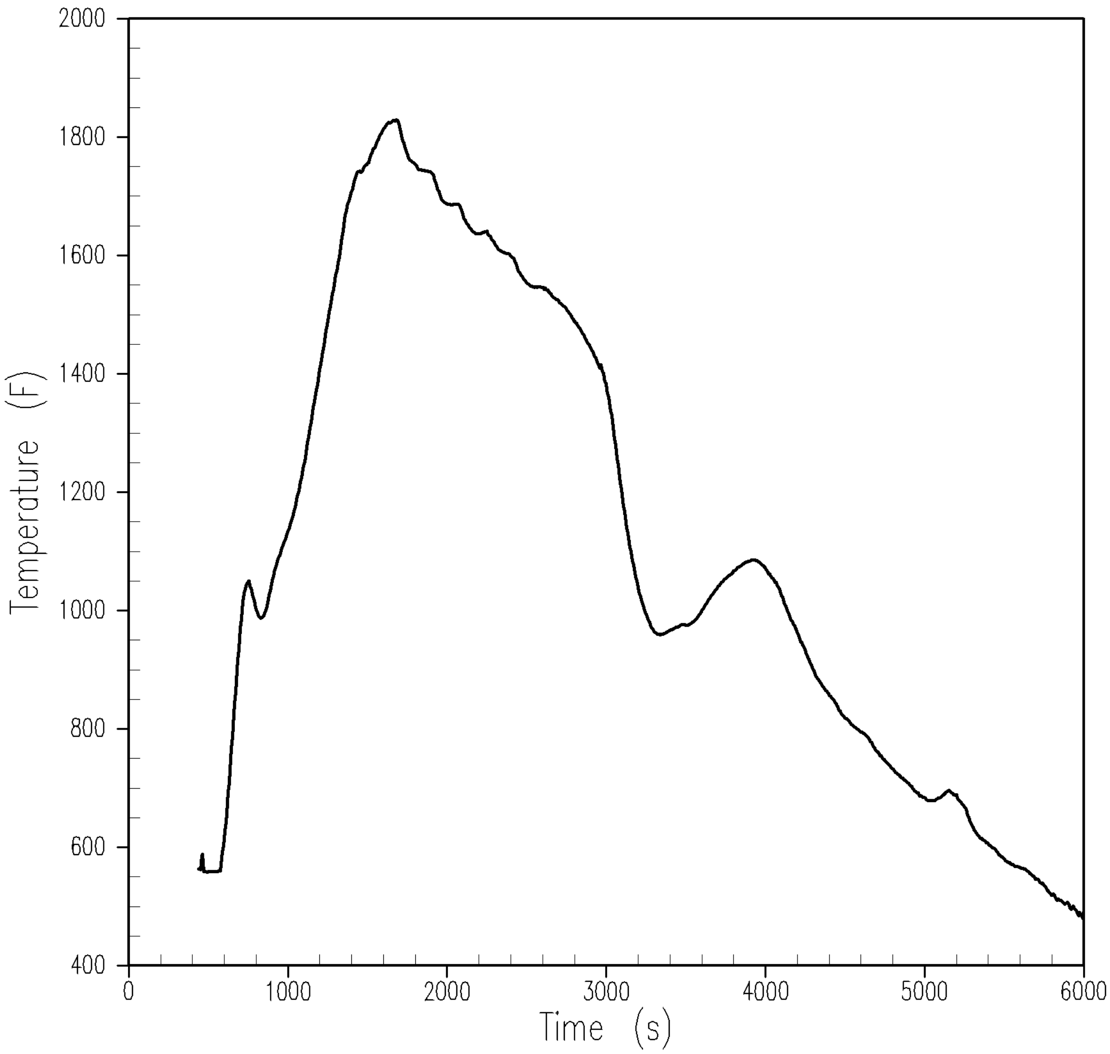

Figure 15.6-51D

Peak Clad Temperature (2.75-inch Break)

Beaver Valley Power Station Unit No. 2
Updated Final Safety Analysis Report

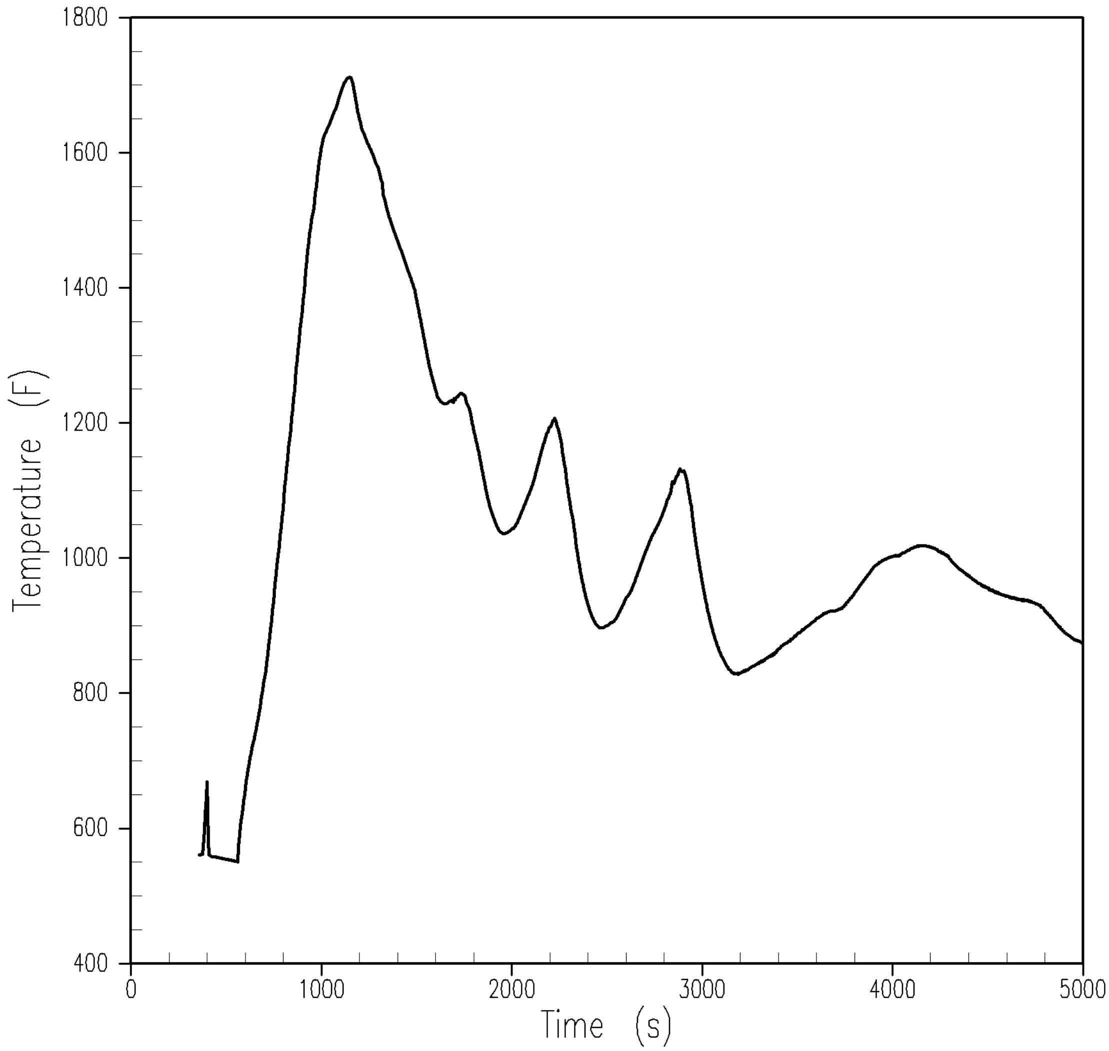

Figure 15.6-51E

Peak Clad Temperature (3.25-inch Break)

Beaver Valley Power Station Unit No. 2
Updated Final Safety Analysis Report

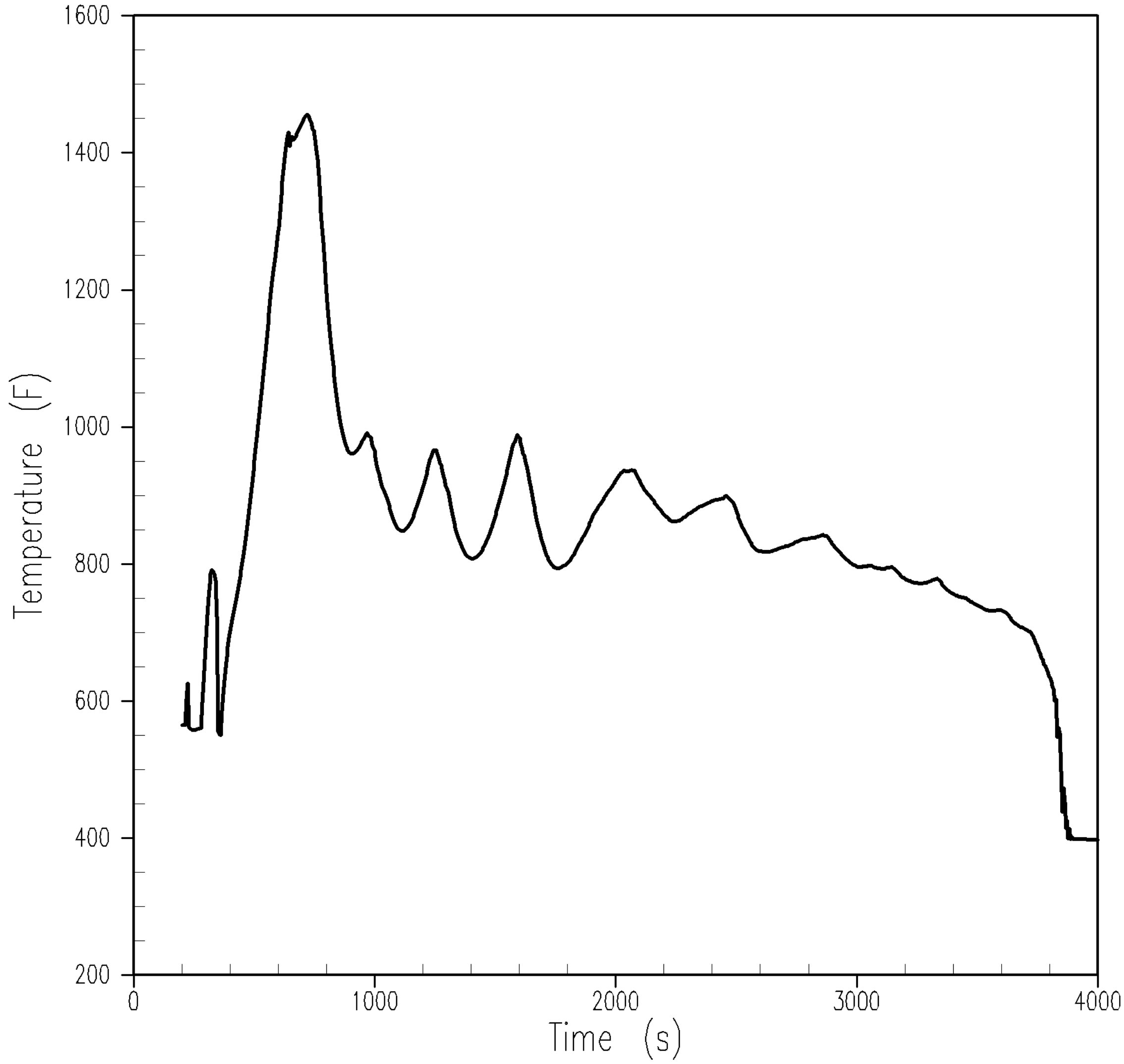

Figure 15.6-51F

Peak Clad Temperature (4-inch Break)

Beaver Valley Power Station Unit No. 2
Updated Final Safety Analysis Report

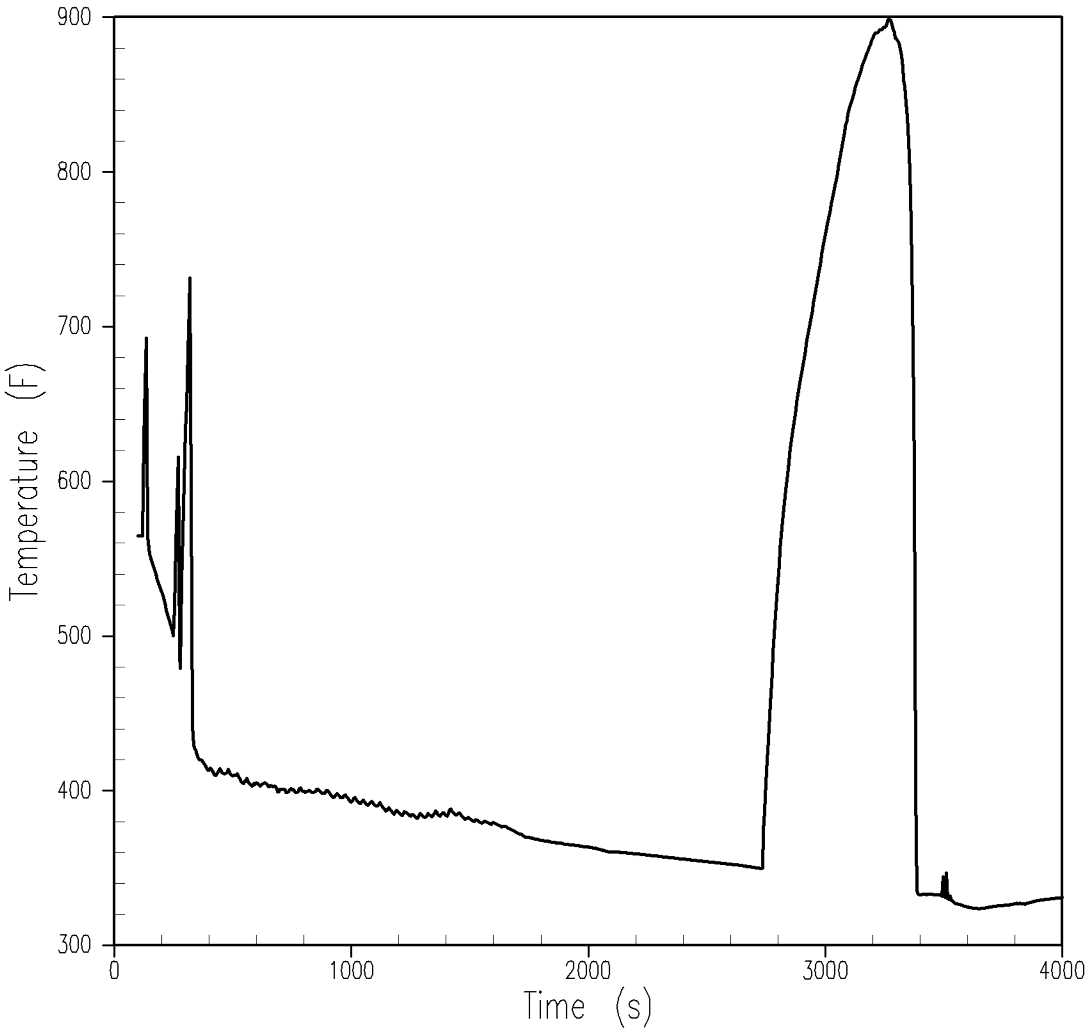

Figure 15.6-51G

Peak Clad Temperature (6-inch Break)

Beaver Valley Power Station Unit No. 2
Updated Final Safety Analysis Report

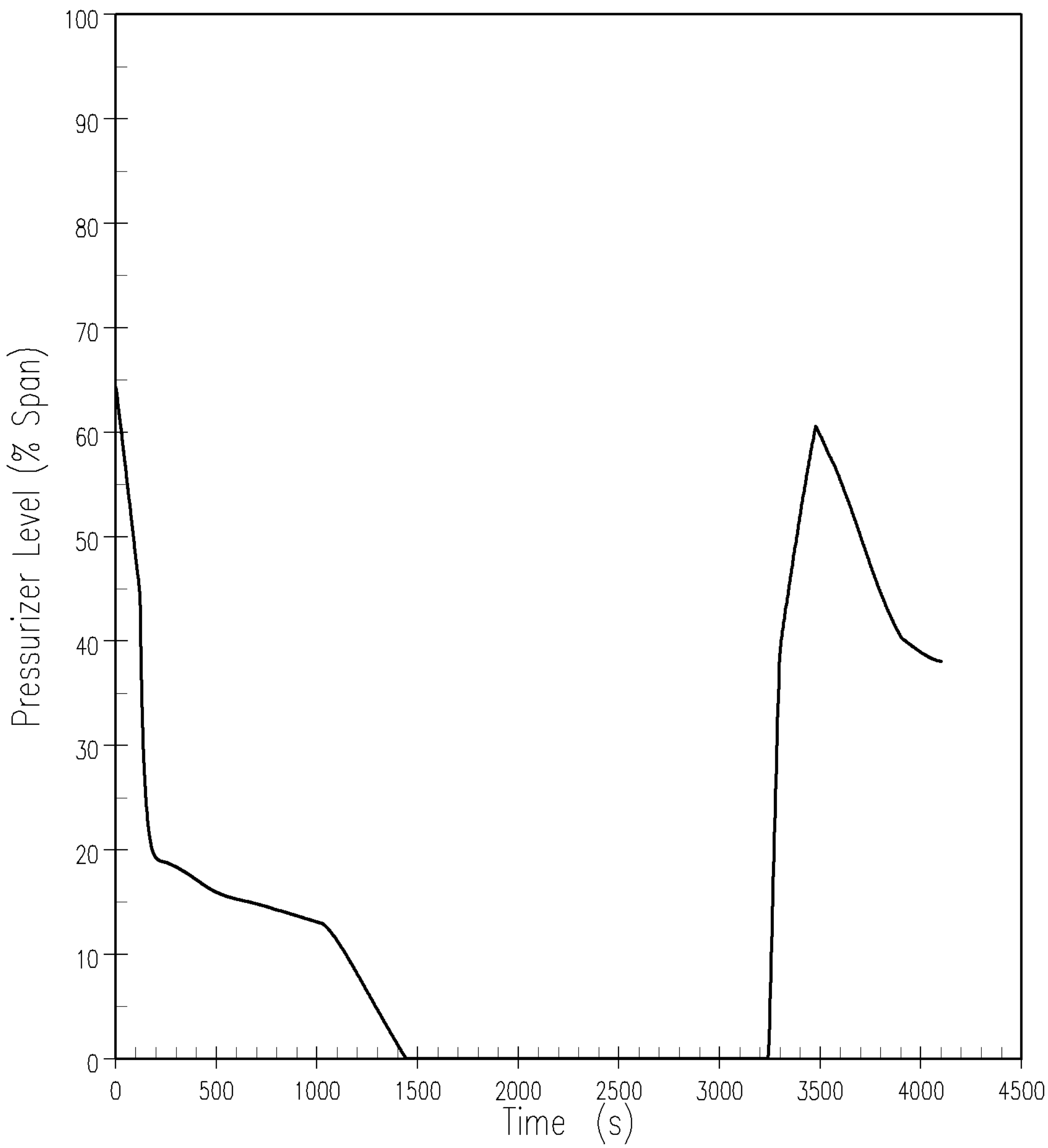

Figure 15.6-58

Pressurizer Level (SGTR)

Beaver Valley Power Station Unit No. 2
Updated Final Safety Analysis Report

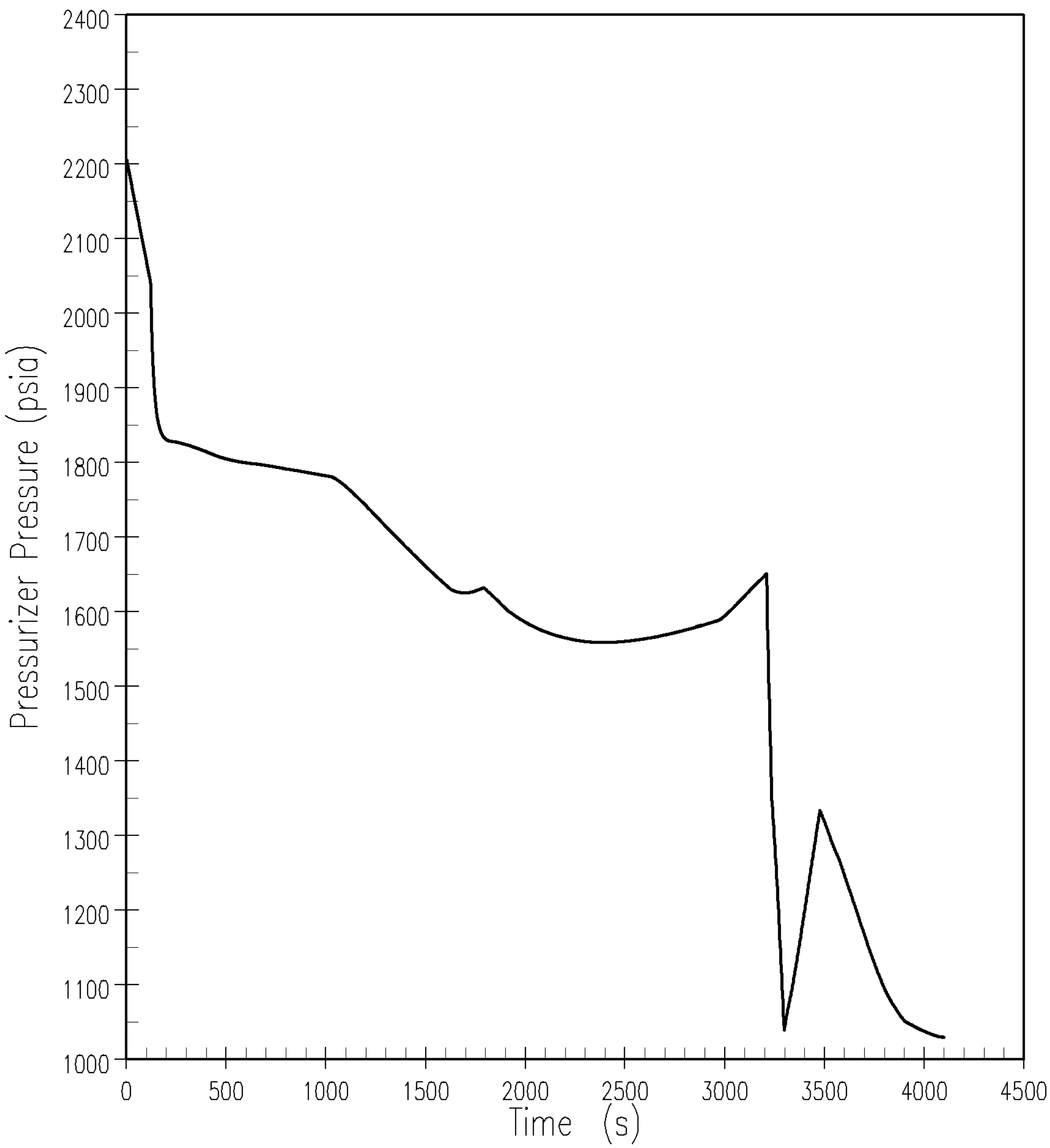

Figure 15.6-59

RCS Pressure (SGTR)

Beaver Valley Power Station Unit No. 2
Updated Final Safety Analysis Report

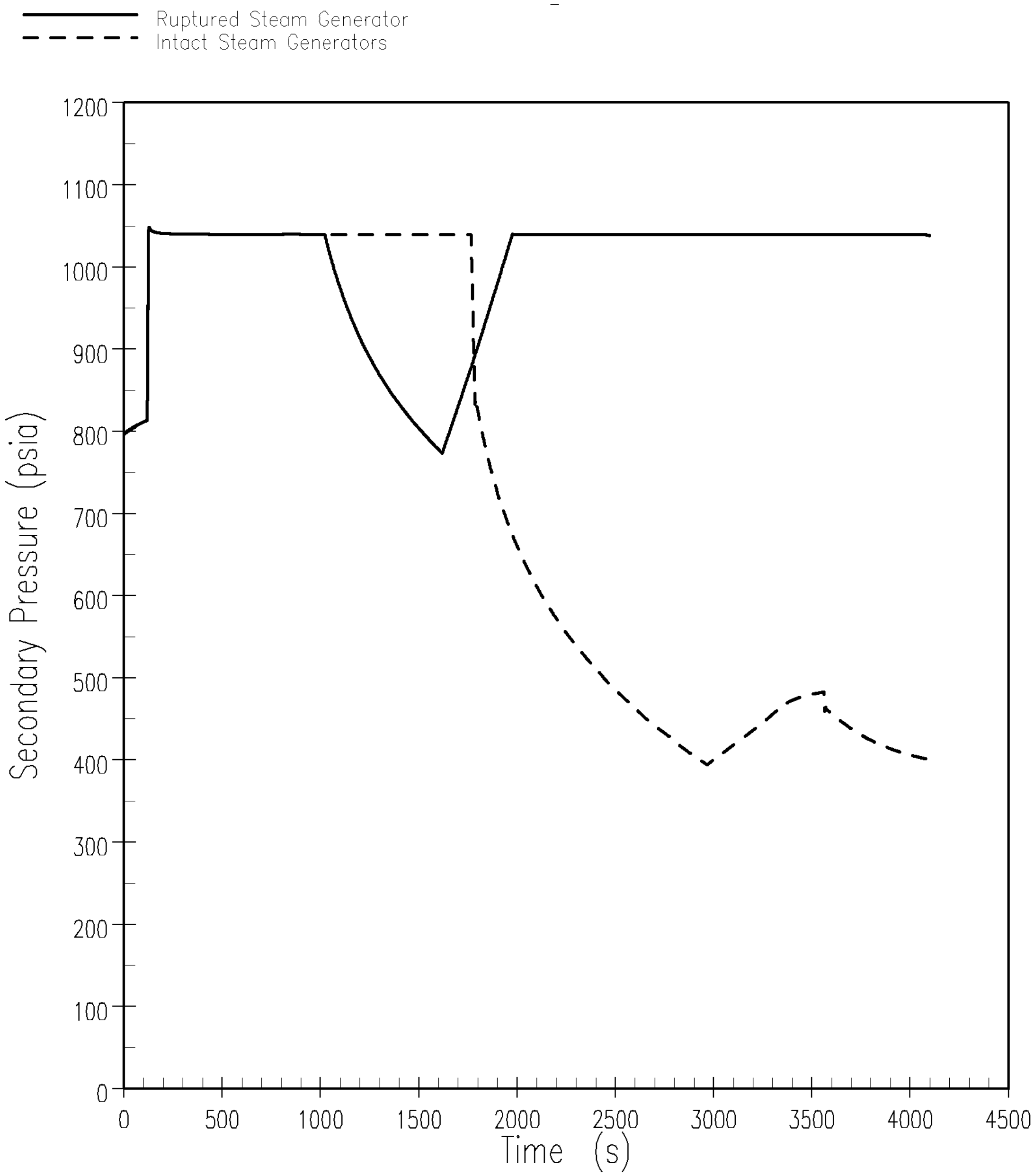

Figure 15.6-60

Secondary Pressure (SGTR)

Beaver Valley Power Station Unit No. 2
Updated Final Safety Analysis Report

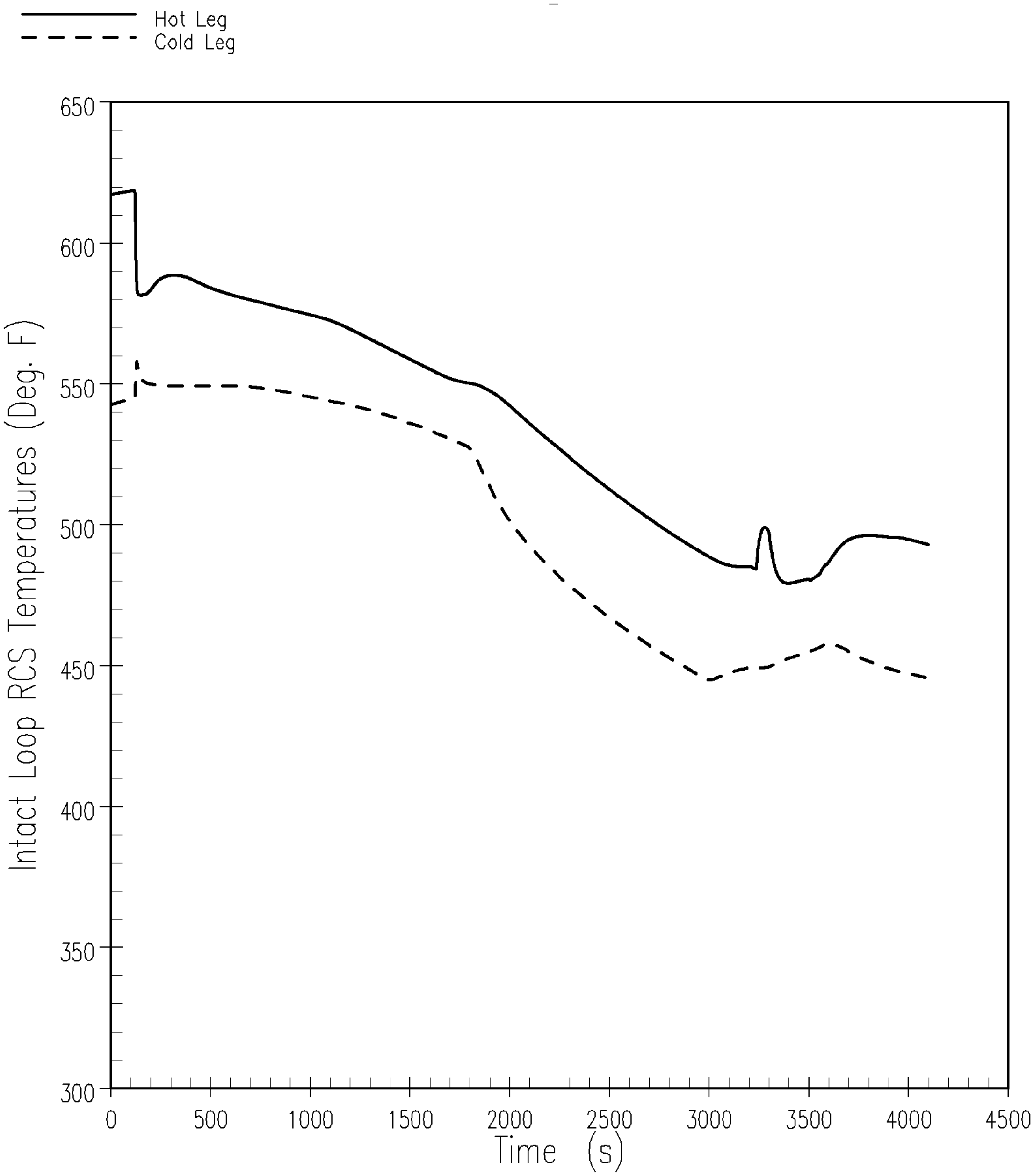

Figure 15.6-61

Intact Loop Hot and Cold Leg RCS Temperatures (SGTR)

Beaver Valley Power Station Unit No. 2
Updated Final Safety Analysis Report

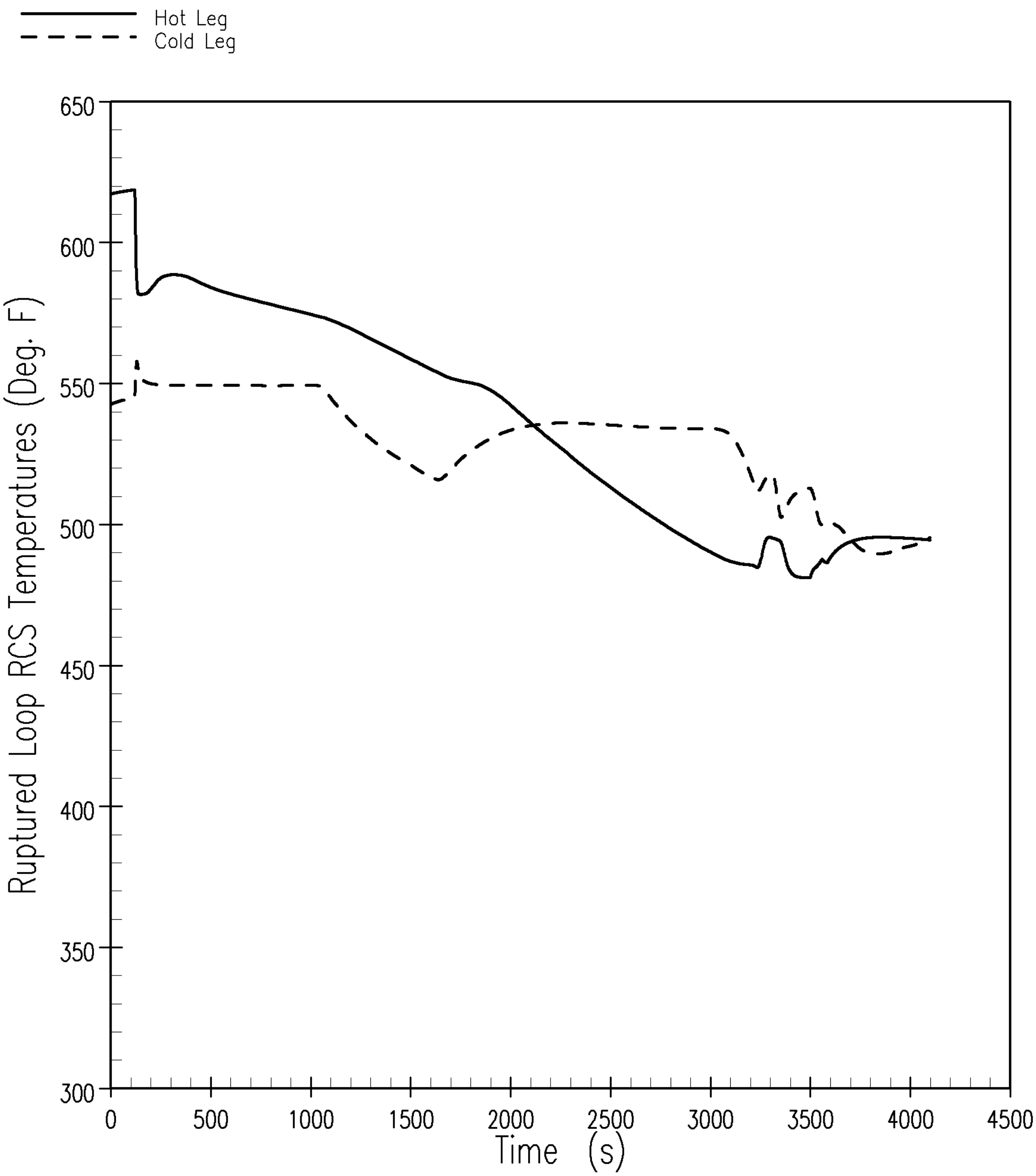

Figure 15.6-62

Ruptured Loop Hot and Cold Leg RCS Temperatures (SGTR)

Beaver Valley Power Station Unit No. 2
Updated Final Safety Analysis Report

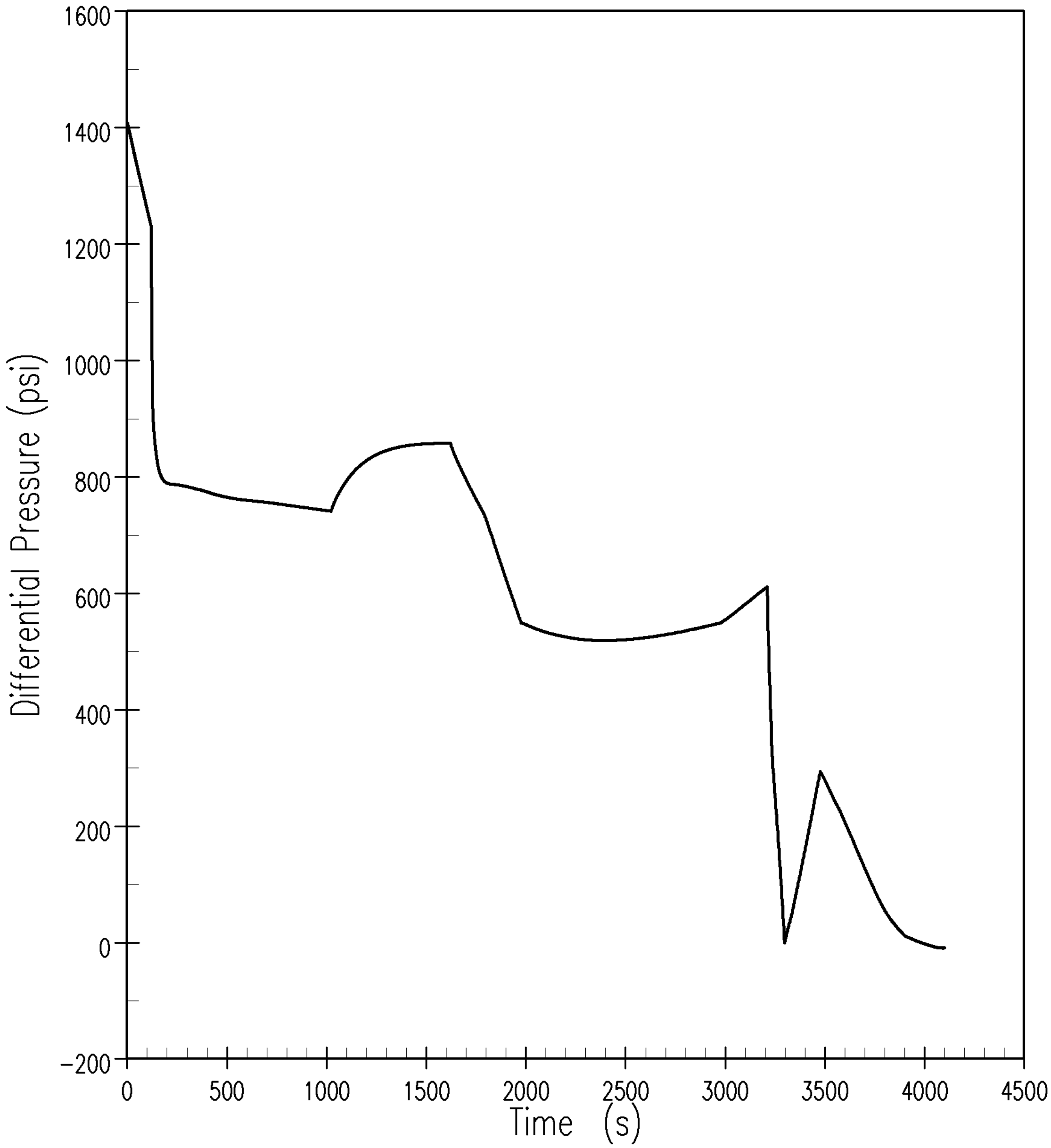

Figure 15.6-63

Differential Pressure Between RCS and Ruptured SG (SGTR)

Beaver Valley Power Station Unit No. 2
Updated Final Safety Analysis Report

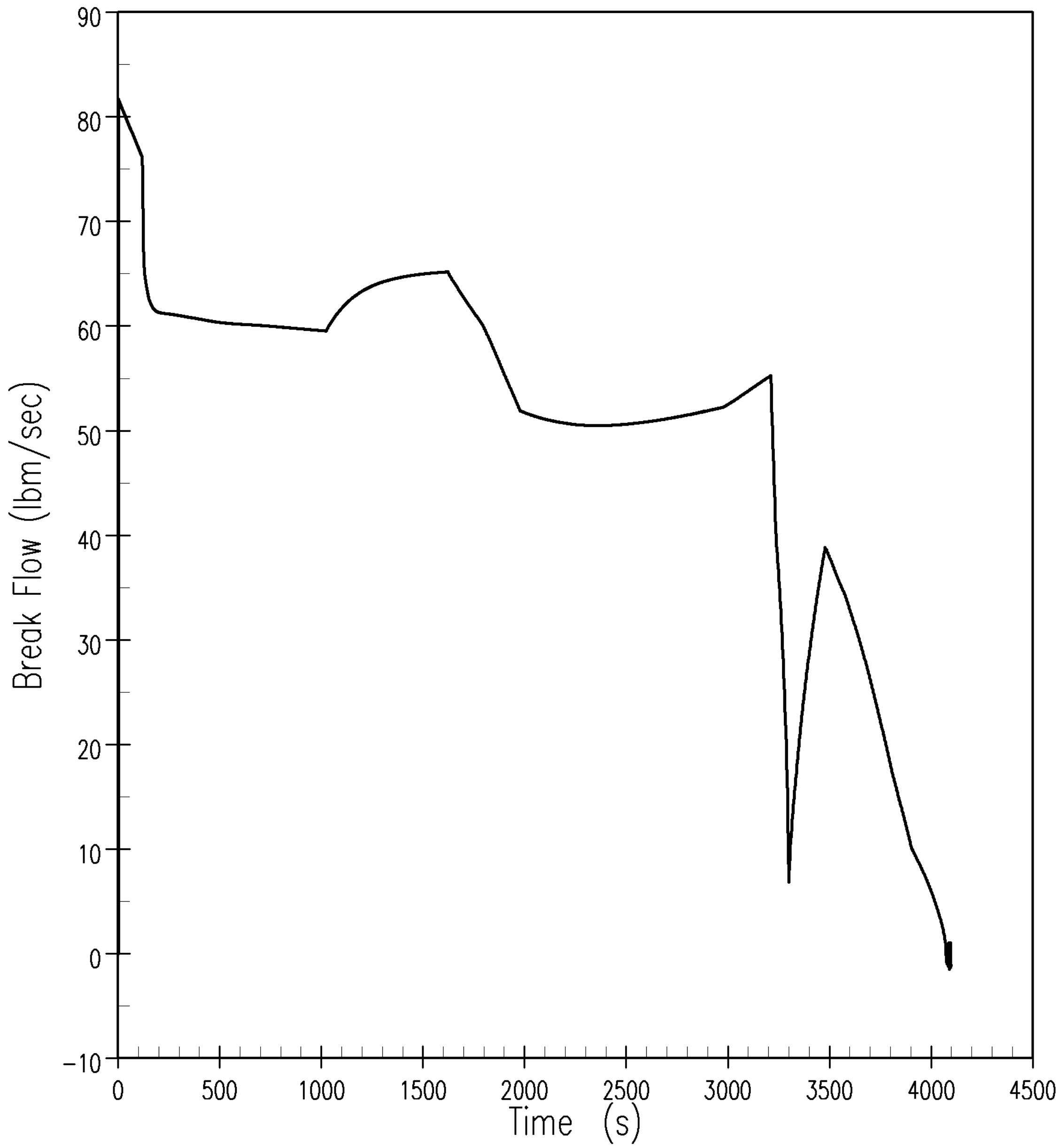

Figure 15.6-64

Primary to Secondary Break Flow Rate (SGTR)

Beaver Valley Power Station Unit No. 2 Updated Final Safety Analysis Report

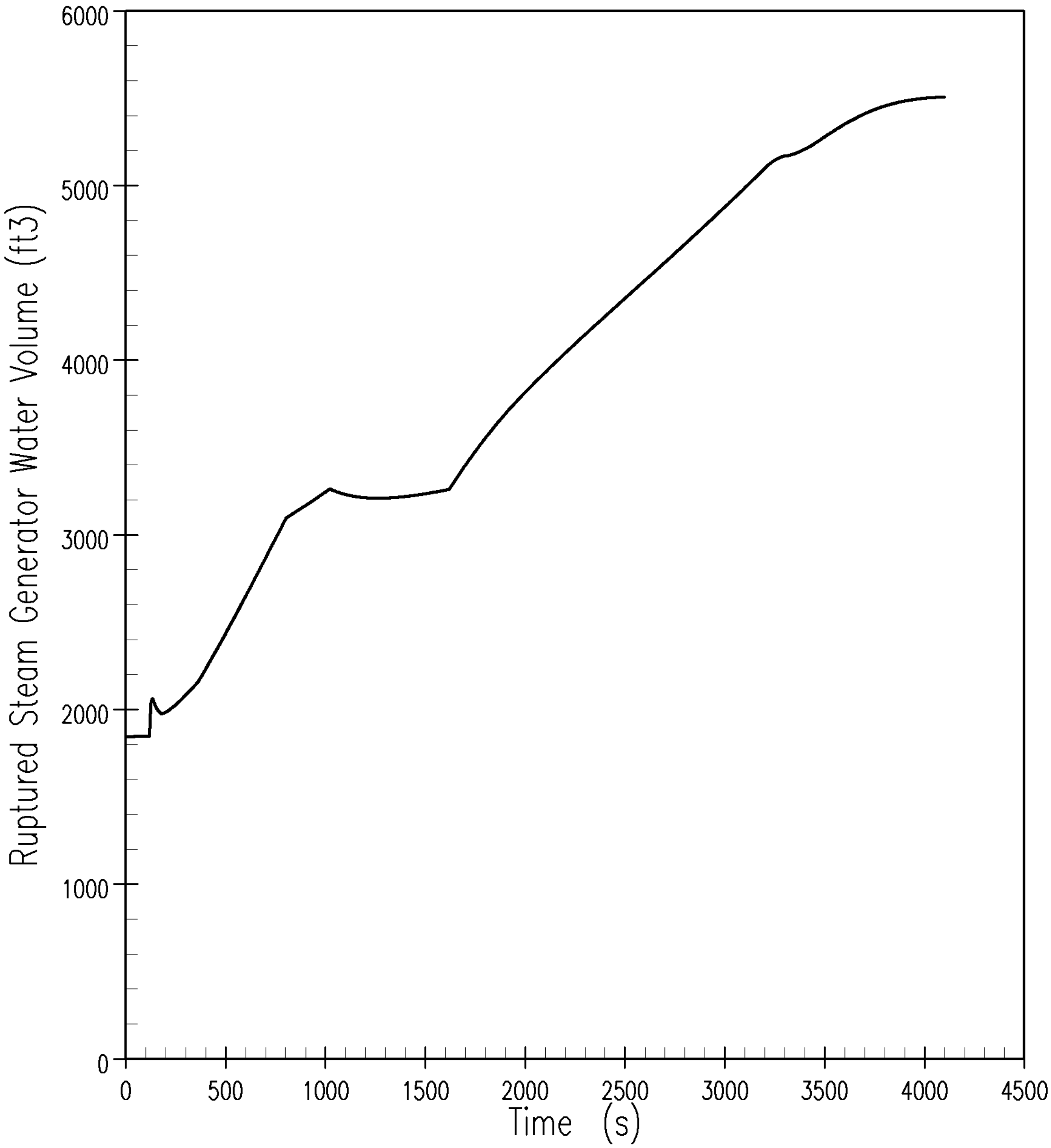

Figure 15.6-65

Ruptured SG Water Volume (SGTR)

Beaver Valley Power Station Unit No. 2
Updated Final Safety Analysis Report

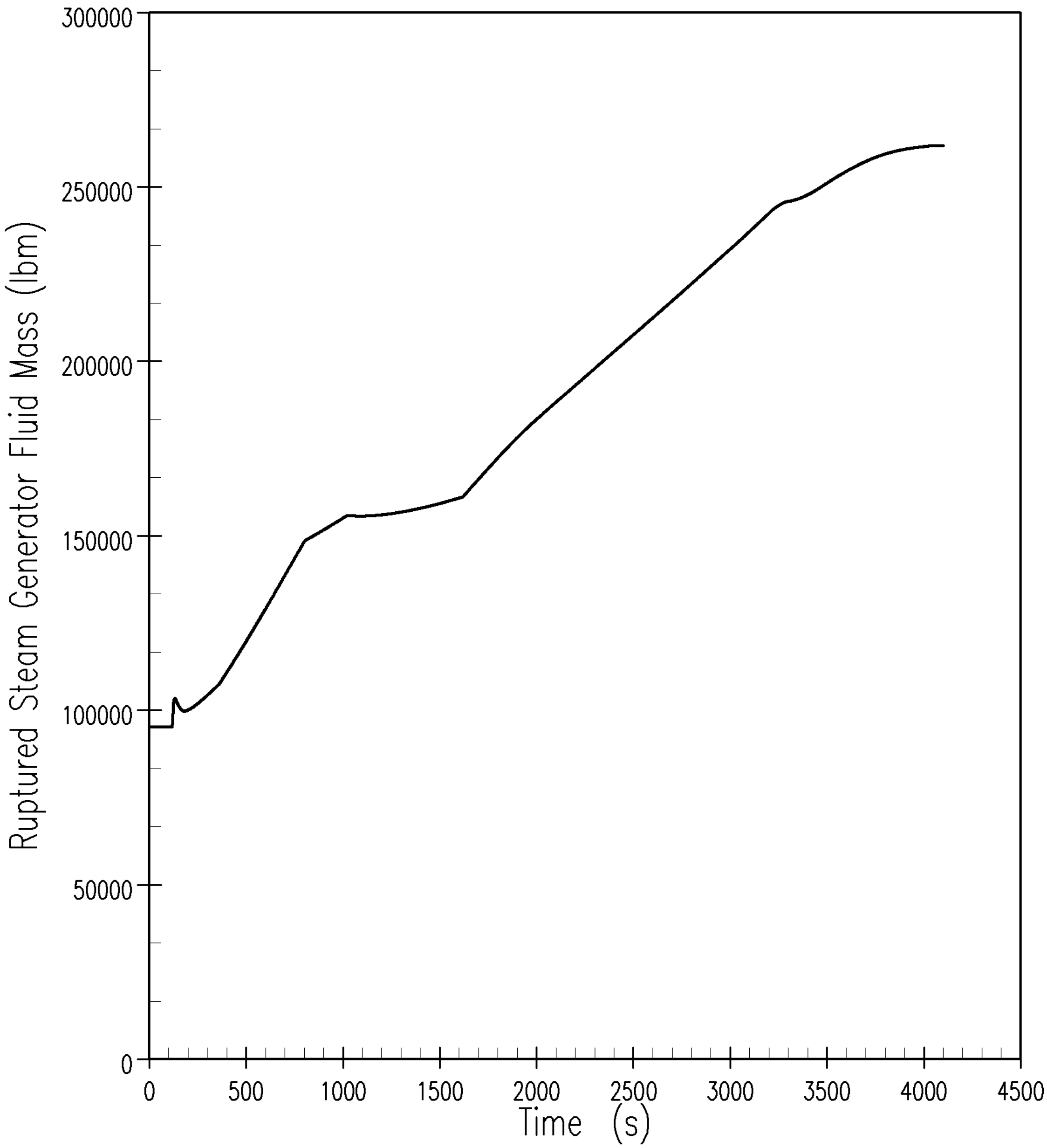

Figure 15.6-66

Ruptured SG Water Mass (SGTR)

Beaver Valley Power Station Unit No. 2
Updated Final Safety Analysis Report

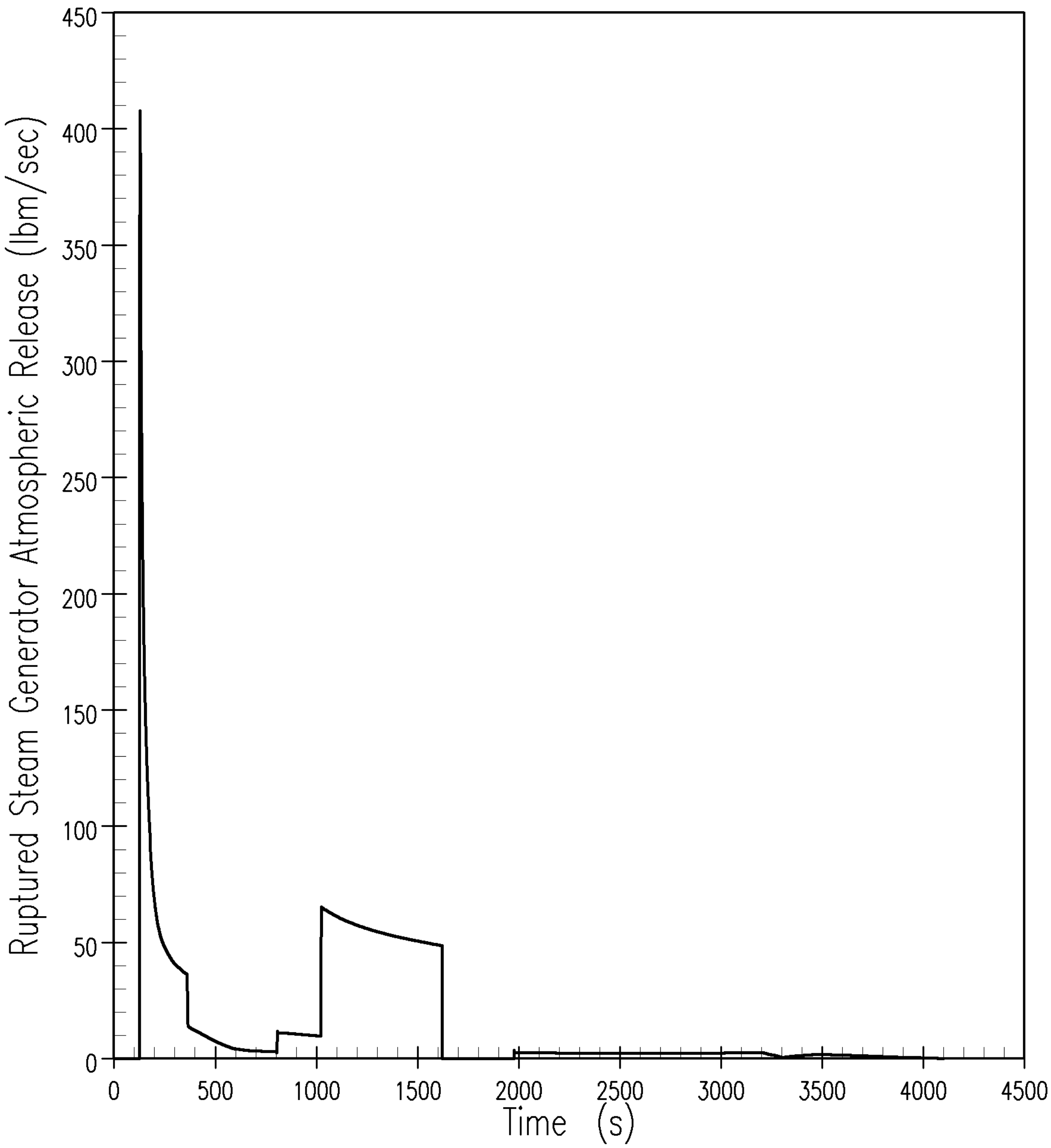

Figure 15.6-67

Ruptured SG Mass Release Rate to the Atmosphere (SGTR)

Beaver Valley Power Station Unit No. 2
Updated Final Safety Analysis Report

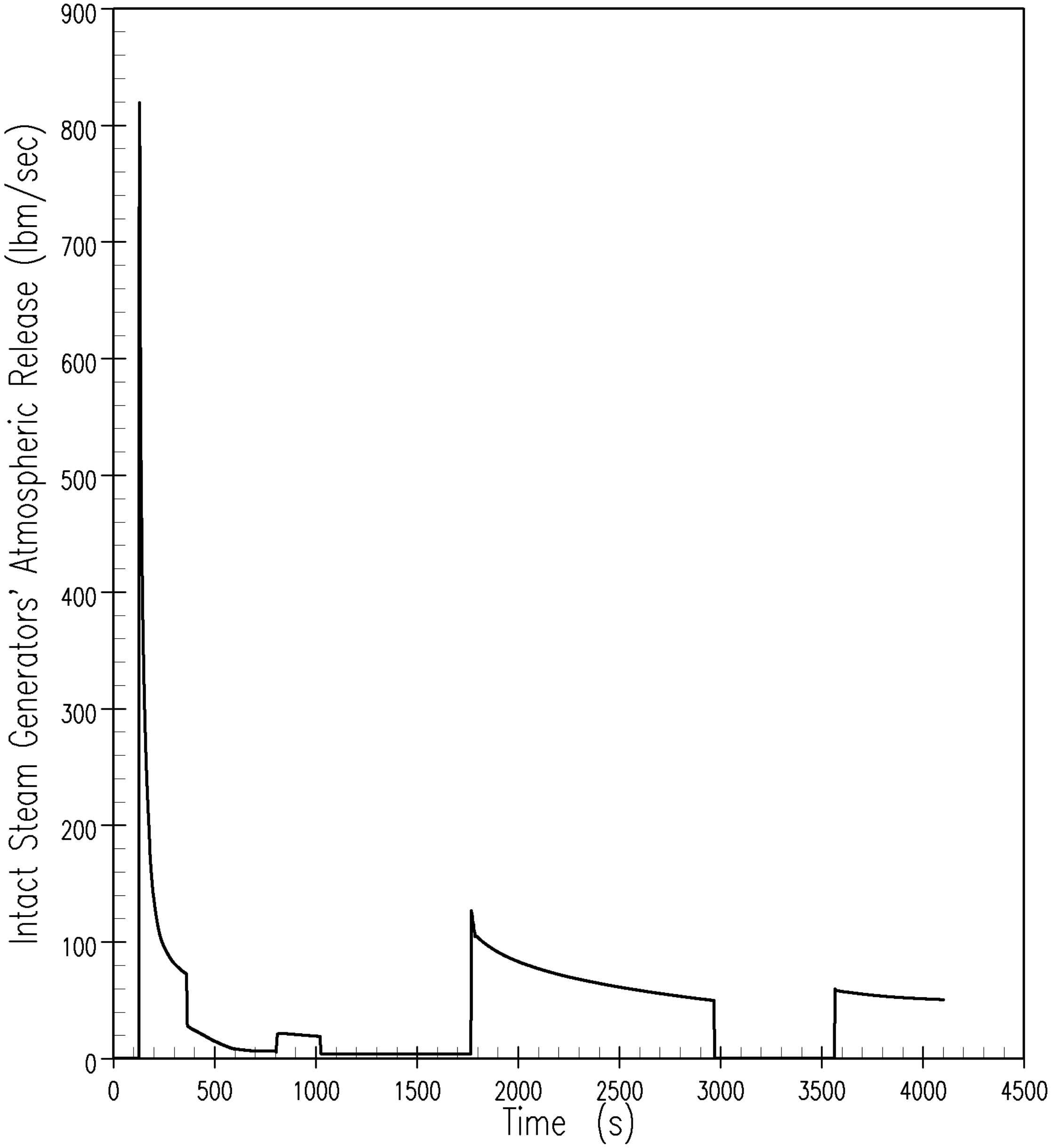

Figure 15.6-68

Intact SGs Mass Release Rate to the Atmosphere (SGTR)

Beaver Valley Power Station Unit No. 2
Updated Final Safety Analysis Report

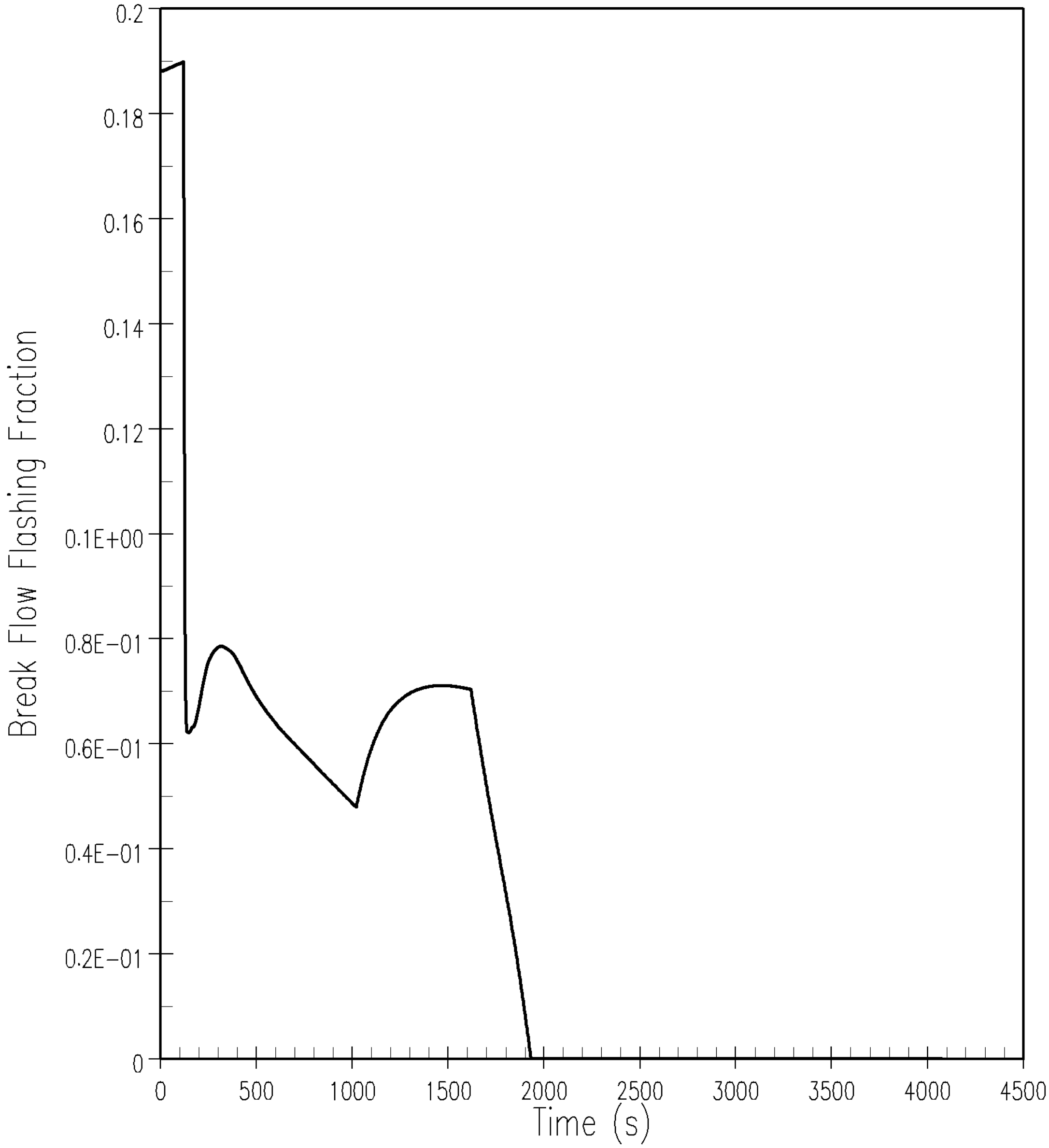

Figure 15.6-70

Break Flow Flashing Fraction (SGTR)

Beaver Valley Power Station Unit No. 2
Updated Final Safety Analysis Report

15.7 RADIOACTIVE RELEASE FROM A SYSTEM OR COMPONENT

15.7.1 Waste Gas System Failure

This section has been relocated to Section 11.3.4.

15.7.2 Radioactive Liquid Waste System Leak or Failure (Atmospheric Release)

This section of the Standard Review Plan (NUREG-0800) has been deleted.

15.7.3 Postulated Radioactive Releases Due to Liquid Containing Tank Failures

15.7.3.1 Identification of Causes and Accident Description

The postulated radioactive release due to liquid-containing tank failures is classified as an ANS Condition III event, an incident which may occur during the lifetime of a plant.

All tanks have been qualitatively considered for radioactive releases. The failure and subsequent release of the contents of the tanks with the largest inventory of radioactivity most likely to infiltrate the nearest potable water supply in an unrestricted area were evaluated. The following tanks were considered for release: coolant recovery tank - located on Beaver Valley Power Station - Unit 1 (BVPS-1) and utilized by both BVPS-1 and Beaver Valley Power Station - Unit 2 (BVPS-2), refueling water storage tank (RWST) (Section 6.2.2), waste drain tank (Section 11.2), and the steam generator blowdown hold tank (Section 11.2). The most limiting liquid-containing tank failure postulated is the RWST. Although the coolant recovery tank and the steam generator blowdown hold tank have larger radionuclide inventories than the RWST, a rupture of either of these tanks would not be as limiting since the liquid pathway to the river is through ground water.

The RWST is located outside, in a partially enclosed structure for biological shielding purposes only, east of the safeguards area. Eighty percent of the volume of the RWST is assumed released as a result of the tank rupture. The total released volume is postulated to reach the river with none of the released liquid permeating into the ground. The inventory of the RWST is based on 40 years of cumulative buildup and decay of nuclides resulting from reactor coolant purification and subsequent mixing with refueling cavity water during refueling operations. The radionuclide inventory of the reactor coolant is based on a 100-hour decay after shutdown and a failed fuel fraction consistent with NUREG-0017 (USNRC 1976). This purified and diluted liquid is then transferred to the RWST.

The design provisions provided to control the release of radioactive materials from the RWST meet GDC 60 and are described in Section 11.2.

15.7.3.2 Analysis of the Effects and Consequences

15.7.3.2.1 Method of Analysis

The determination of radionuclide inventory in the RWST is based on the techniques of NUREG-0133 (USNRC 1978) with the exception of the expected failed fuel fraction, which will be in accordance with NUREG-0017 (USNRC 1976).

15.7.3.3 Radiological Consequences

The radionuclide concentrations in the downstream potable water supply resulting from a liquid-containing tank rupture are determined for the postulated rupture of the RWST. For the RWST rupture, 80 percent of the total volume of the tank drains into the river. Since the time required for liquid transit from the yard to the river is less than 2 hours, it is conservatively assumed that none of the liquid permeates into the ground. The maximum plume concentrations at the potable water supply occur at Chester, West Virginia, 7.1 miles downstream from BVPS-2.

The radionuclide concentrations at the potable water supply resulting from a rupture of the RWST are based on the assumptions given in Table 15.7-4. The inventory in the tank is derived from coolant which has been purified and diluted during the refueling process. The primary coolant activities are based on an expected failed fuel fraction of 0.0012 and are presented in Table 11.1-2. The primary coolant concentrations are purified at the rate of 120 gpm for 40 hours after shutdown of the reactor, and for an additional 60 hours at 60 gpm diluted by the refueling cavity liquid volume, and decayed for 15.6 days before being transferred to the RWST. The concentrations in the RWST are assumed to accumulate during each refueling outage and will decay for the life of the plant.

For the postulated rupture, credit is taken for the travel time required for the plume to reach the potable water supply intake and the dilution of the RWST inventory in the river. The resulting radionuclide concentrations at the water supply intake are presented in Table 15.7-5. The individual and the integrated prorated radionuclide concentrations are below the limits listed in Appendix B, Table II, Column 2, of 10 CFR 20.

15.7.4 Radiological Consequence of Fuel Handling Accidents

15.7.4.1 Identification of Causes and Accident Description

The fuel handling accident is classified as an ANS Condition IV event, faults that are not expected to occur but are postulated because their consequences include the potential for the release of significant amounts of radioactive material.

The fuel handling accident is postulated to occur in the fuel building and in the containment.

The fuel handling accident sequence of events consists of the dropping of one fuel assembly on or near other fuel assemblies resulting in maximum cladding damage to the fuel rods with subsequent instantaneous release of all the gap radionuclide inventory.

The gap radionuclide inventory is based on the minimum time after refueling shutdown of 100 hours and peak inventories for the damaged fuel assemblies. The pool water provides retention capabilities for radioiodines as described in Table 15.7-6.

The radioactivity released from the pool into the building atmosphere will be released via one or a combination of three pathways. These are: the containment equipment hatch, the auxiliary building ventilation system or the SLCRS.

The radioactivity control features of the fuel storage and handling systems inside containment and in the fuel building meet the requirements of GDC 61 (Section 9.1).

15.7.4.2 Analysis of the Effects and Consequences

15.7.4.2.1 Method of Analysis

The assumptions applied to the evaluation of the release of radioactivity from the fuel and the fuel building are based on Regulatory Guide 1.183.

Activity may be released to the environment following a fuel handling accident in the containment building through either the open personnel airlock, equipment hatch, or other open penetrations. Controls have been established to close any open penetration following a fuel handling accident. Although this provision will further minimize the activity release, the accident radiological consequence analysis does not take credit for closure.

15.7.4.3 Radiological Consequences

The BVPS-2 plant Technical Specifications prohibit initiation of fuel handling activities in the Fuel Pool or in the Containment until 100 hours after reactor shutdown. Analyses of the deflections and resulting stresses on the dropped fuel results in the damage of 137 of the 264 rods in a fuel assembly. All of the fuel gap activity associated with the damaged rods is assumed to be released. A radial peaking factor of 1.75 is applied to the core average gap activity. The activity (consisting of noble gases, halogens, and alkali metals) is released in a “puff” to the fuel pool or reactor cavity.

The radioiodine released from the fuel gap is assumed to be 95% CsI, 4.85% elemental, and 0.15% organic. Due to the acidic nature of the water in the reactor cavity (pH less than 7), the CsI will immediately disassociate, thus changing the chemical form of iodine in the water to 99.85% elemental and 0.15% organic. The minimum depth of water in the fuel pool and reactor cavity is 23 ft over the top of the damaged fuel assembly. Therefore, per RG 1.183, the pool provides an overall effective decontamination factor for elemental and organic iodines of 200. Per RG 1.183, the chemical form of the iodines above the reactor cavity is 57% elemental and 43% organic.

Noble gas and unscrubbed iodines rise to the water surface whereas all of the alkali metals released from gap are retained in the reactor cavity water. Since the fuel pool area and containment are assumed to be open, and there is no means of isolating the accident release, all of the airborne activity resulting from the FHA is exhausted out of the building in a period of 2 hours. The analysis assumes that during refueling, the ventilation is operational above the spent fuel pool area.

The exhaust flows from the containment and Fuel Pool Area may be directed out of the SLCRS release point. However, since the containment and fuel buildings are "open", releases could also occur from anywhere along the containment wall (e.g., via the equipment or personnel hatch) or via the fuel building normal operation release point, i.e., the ventilation vent. Because the location of the release is unknown, the worst case dispersion factor (identified for purposes of assessment as the BVPS-2 ventilation vent to the BVPS-2 CR intake) is used without taking any credit for SLCRS flows or filtration in this analysis.

EAB 2 hr Worst Case Window

AST methodology requires that the worst case dose to an individual located at any point on the boundary at the EAB, for any 2-hr period following the onset of the accident be reported as the EAB dose. Since the event is based on a 2-hour release, the worst 2-hour period for the EAB is the 0 to 2-hour period.

Accident-Specific Control Room Model Assumptions

The control room is assumed to remain in the normal operation mode. The critical control room parameters utilized in this model are summarized in Table 6.4-1a. Section 15.6.5.4 discusses the control room design as related to dose consequences under a sub-section titled "Control Room Habitability."

The radiological consequences of the postulated fuel handling accident are analyzed based on the assumptions and initial maximum assembly gap activities listed in Tables 15.7-6 and 15.7-6a. Offsite doses and control room doses are calculated using the releases in combination with the atmospheric dispersion values given in Tables 15.0-11 and 15.0-15 and the methodology described in Appendix 15A.

The radiological consequences of the postulated fuel handling accident presented in Table 15.0-12 and Table 15.0-13, are well within the guidelines of 10 CFR 50.67, that is, less than 25 rem TEDE offsite and 5 rem TEDE for the control room. Additionally, the offsite doses are within the criteria of Regulatory Guide 1.183 and NUREG-0800 of 6.3 rem TEDE.

15.7.5 Spent Fuel Cask Drop Accidents

15.7.5.1 Identification of Causes and Description

Cask handling procedures ensure that a postulated spent fuel cask drop height of 30 feet is not exceeded. If the spent fuel cask trolley limiting devices are removed during cask handling within the plant, the 30-foot drop height is still not exceeded.

15.7.5.2 Analysis of Effects and Consequences

The details of spent fuel cask handling are provided in Section 9.1.5.

15.7.5.3 Radiological Consequences

Since a spent fuel cask drop exceeding 30 feet cannot occur, no radiological analysis need be performed for a spent fuel cask drop accident.

15.7.6 References for Section 15.7

Underhill, D.W. 1972. Effects of Rupture in a Pressurized Noble Gas Adsorption Bed; Nuclear Safety Volume 13 Number 6.

U.S. Nuclear Regulatory Commission (USNRC 1976). Calculations of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors (PWR-GALE Code). NUREG-0017.

USNRC 1978. Preparation of Radiological Effluent Technical Specification for Nuclear Power Plants. NUREG-0133.

USNRC 1981. Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (formerly issued as NUREG 75/087). NUREG-0800.

USNRC 2000 Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors. Regulatory Guide 1.183.

Tables for Section 15.7

TABLE 15.7-4 (HISTORICAL)

ASSUMPTIONS USED FOR THE LIQUID-CONTAINING TANK FAILURE

Power (MWt)	2,766
Fraction of fuel with defects	0.0012
Tank	RWST
Tank volume (gal)	910,000
Release fraction	0.8
Source stream	Primary coolant (Table 11.1-2)
Primary coolant purification and decay time after reactor shutdown (hrs)	
at 120 gpm	40
at 60 gpm	60
Dilution volume (ft^3)	34,900
Decay time during refueling (days)	15.6
Feed rate to RWST (gpm)	139
Feed time to RWST (hrs)	31.3
Number of refueling	40
Plume travel time to potable water supply (hrs)	65
Dilution factor in river	443

The information presented in the above table was developed in support of the original License and is considered Historical.

TABLE 15.7-5 (HISTORICAL)

RADIONUCLUDE CONCENTRATIONS IN POTABLE WATER SUPPLY
RESULTING FROM THE RWST RUPTURE

Nuclide	Concentration* (μCi/cc)
I-131	$6.1x10^{-9}$
I-132	$3.4x10^{-11}$
Rb-86	$1.0x10^{-11}$
Sr-89	$4.2x10^{-11}$
Sr-90	$4.5x10^{-11}$
Y-90	$2.4x10^{-11}$
Y-91	$8.7x10^{-12}$
Zr-95	$7.7x10^{-12}$
Nb-95m	$5.6x10^{-14}$
Nb-95	$8.6x10^{-12}$
Mo-99	$4.1x10^{-11}$
Tc-99m	$4.0x10^{-11}$
Ru-103	$4.8x10^{-12}$
Ru-106	$3.3x10^{-12}$
Rh-103m	$4.4x10^{-12}$
Rh-106	$3.3x10^{-12}$
Te-125m	$3.7x10^{-12}$
Te-127m	$4.4x10^{-11}$
Te-127	$4.4x10^{-11}$
Te-129m	$1.4x10^{-10}$
Te-129	$9.4x10^{-11}$
Te-131m	$1.2x10^{-15}$
Te-132	$3.3x10^{-11}$
Cs-134	$2.5x10^{-7}$
Cs-136	$1.1x10^{-8}$
Cs-137	$1.4x10^{-6}$
Ba-140	$1.0x10^{-11}$
La-140	$1.2x10^{-11}$
Ce-141	$7.0x10^{-12}$
Ce-144	$8.7x10^{-12}$
Pr-143	$2.8x10^{-12}$
Pr-144	$8.7x10^{-12}$
Cr-51	$1.7x10^{-10}$
Mn-54	$8.7x10^{-11}$
Fe-55	$1.1x10^{-9}$
Fe-59	$1.2x10^{-10}$
Co-58	$2.2x10^{-9}$
Co-60	$2.7x10^{-9}$
Np-239	$2.2x10^{-13}$

NOTE:

*Nuclide concentrations of less that $1.0x10^{-15}$ μCi/cc are excluded.

The information presented in the above table was developed in support of the original License and is considered Historical.

TABLE 15.7-6

ASSUMPTIONS USED FOR THE
FUEL HANDLING ACCIDENT ANALYSIS

Core Power Level	2918 MWt
Number of Rods in Fuel Assemblies	264
Total Number of Fuel Assemblies	157
Number of Damaged Rods	137
Decay Time Prior to Fuel Movement	100 hours
Radial Peaking Factor	1.75
Fraction of Core Inventory in gap	I-131 (8%) Kr-85 (10%) Other Noble Gases (5%) Other Halides (5%) Alkali Metals (12%)
Core Activity of Isotopes in Gap	Table 15.7-6a
Iodine Form of gap release before scrubbing	99.85% elemental 0.15% Organic
Min depth of water in Fuel Pool or Reactor Cavity	23 ft
Scrubbing Decontamination Factors	Iodine (200) Noble Gas (1) Particulates (∞)
Rate of Release from Fuel:	PUFF
Environmental Release Rate (unfiltered) within a 2-hour period	All airborne activity
Environmental Release Points:	
Accident in Fuel Pool Area	More Restrictive of Ventilation Vent or SLCRS
Accident in Containment	More Restrictive of Equipment Hatch or SLCRS
CR Emergency Ventilation: Initiation Signal/Timing	Control room is maintained in normal ventilation mode

TABLE 15.7-6a

ACTIVITIES USED FOR THE
FUEL HANDLING ACCIDENT ANALYSIS

Nuclide	Core Activity with 100 hour decay (Ci)	Gap Fraction	Fuel Gap Activity (Ci)
KR-85	8.27E+05	0.1	4.78E+02
KR-85M	3.77E+00	0.05	1.09E-03
XE-127	9.50E+00	0.05	2.75E-03
XE-129M	4.49E+03	0.05	1.30E+00
XE-131M	1.00E+06	0.05	2.89E+02
XE-133	1.11E+08	0.05	3.21E+04
XE-133M	2.07E+06	0.05	5.99E+02
XE-135	2.13E+05	0.05	6.16E+01
XE-135M	6.51E+02	0.05	1.88E-01
BR- 82	4.25E+04	0.05	1.23E+01
I-129	2.86E+00	0.05	8.27E-04
I-130	7.64E+03	0.05	2.21E+00
I-131	5.62E+07	0.08	2.60E+04
I-132	4.74E+07	0.05	1.37E+04
I-133	5.86E+06	0.05	1.69E+03
I-135	3.98E+03	0.05	1.15E+00

Where:

$$\boxed{\mathrm{GapActivity} = \frac{\mathrm{cpa} \times \mathrm{gf} \times 137 \times 1.75}{264 \times 157}}$$

Where

cpa = 100 hr decayed core activity,
gf = gap fraction,
1.75 = radial peaking factor,
137 = # of ruptured rods,
264 = # of rods per assembly,
157 = total # of assemblies in the core.

15.8 ANTICIPATED TRANSIENTS WITHOUT SCRAM

An anticipated transient without SCRAM (ATWS) is an anticipated operational occurrence (such as 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[1,2] on ATWS showed that acceptable consequences would result provided that the turbine trips and auxiliary feedwater flow is initiated in a timely manner.

The effects of ATWS are not considered as part of the design basis for transients analyzed in Chapter 15. The final USNRC ATWS rule[3] requires that all U.S. Westinghouse-designed plants install ATWS mitigation system actuation circuitry (AMSAC) to initiate a turbine trip and actuate auxiliary feedwater independent of the reactor trip system.

15.8.1 References

1. Burnett, T. W. T., et al., "Westinghouse Anticipated Transients Without Trip Analysis," WCAP-8330, August 1974.

2. Letter from T. M. Anderson (Westinghouse) to S. H. Hanauer (USNRC), "ATWS Submittal," NS-TMA-2182, December 1979.

3. ATWS Final Rule, Code of Federal Regulations 10 CFR 50.62, "Requirements for Reduction of Risk from Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants."

APPENDIX 15A

DOSE METHODOLOGY

BEAVER VALLEY POWER STATION

APPENDIX 15A

DOSE METHODOLOGY

The radiological consequences of the design basis accidents (DBAs) are represented by the calculated results of thyroid doses, whole-body gamma doses, and beta skin doses, or alternatively by the calculated total effective dose equivalent (TEDE), at the exclusion area boundary (EAB), the low population zone (LPZ), and in the main control room. The doses at the EAB are based on the release of radionuclides over the worst 2-hr period following the occurrence of a postulated accident. For accidents lasting beyond 2 hours, doses are calculated for the LPZ based on releases over the duration of the accident, up to 30 days following the occurrence of an accident. The control room dose is based on releases over a 30-day period following the loss-of-coolant accident (LOCA).

Except as noted, the dose estimates to the operator in the control room or to the public following a design basis accident are calculated in terms of TEDE and in accordance with the guidance provided in Regulatory Guide 1.183. The acceptance criteria for Waste Gas System Rupture at the site boundary is 500 mrem whole body in accordance with BTP ETSB 11-5, whereas the acceptance criteria for the control room dose is 5 rem whole body or its equivalent in accordance with 10 CFR 50 GDC 19.

15A.1 Original Licensing Basis

Thyroid doses are calculated based on Regulatory Guide 1.4, June 1974 and the following equation:

$$D_{thy} = \sum_i Q_i \, (\chi / Q)(BR)(C_{thy_i}) \tag{15A-1}$$

where:

D_{thy} = thyroid dose (Rem)

Q_i = activity of iodine isotope i released (Ci)

χ/Q = atmospheric dispersion factor (sec/m^3)

BR = breathing rate (m^3/sec)

C_{thy_i} = thyroid dose conversion factor for iodine isotope i (Rem/Ci) (DiNunno et al 1962)

The χ/Q values presented in Table 15.0-14 were calculated using the methodology described in Section 2.3. For persons offsite, the breathing rates are assumed to be:

3.47×10^{-4} m^3/sec, 0 to 8 hours

1.75×10^{-4} m^3/sec, 8 to 24 hours

2.32×10^{-4} m^3/sec, >24 hours

These values are taken from Regulatory Guide 1.4.

External whole-body gamma doses and beta skin doses are calculated using Equations 15A-2 and 15A-3 derived from equations in Regulatory Guide 1.4.

$$D_\gamma = 0.25 \sum_i \bar{E}_{\gamma i} Q_i (\chi / Q) \tag{15A-2}$$

and

$$D_\beta = 0.23 \sum_i \bar{E}_{\beta i} Q_i (\chi / Q) \tag{15A-3}$$

where:

D_γ = gamma dose from a semi-infinite cloud (Rem)

D_β = beta skin dose from a semi-infinite cloud (Rem)

$\bar{E}_{\gamma i}$ = average gamma energy per disintegration of nuclide i, (Mev/dis)

$\bar{E}_{\beta i}$ = average beta energy per disintegration of nuclide i (Mev/dis)

Q_i = activity of gamma or beta emitting nuclide i released to the environment (Ci)

χ/Q = atmospheric dispersion factor (sec/m^3)

The thyroid and beta skin doses to control room personnel due to inleakage are calculated as described previously. The gamma whole-body dose due to inleakage is calculated using Equation 15A-4:

$$D_\gamma = \frac{R}{2} t \sum_i A_i K_i \tag{15A-4}$$

where:

D_γ = gamma whole-body dose (Rem)

R = radius of an equivalent hemispherical control room (m)

t = time (hr)

A_i = concentration of nuclide i (Ci/m^3)

K_i = conversion factor for nuclide i (Rem-m^2/Ci-hr), defined as follows:

$$K_i = 3.7 \times 10^6 \sum_i S_{ij} C_{ij} \tag{15A-5}$$

where:

S_{ij} = gamma energy per disintegration of nuclide i at energy E_j (Mev/dis)

C_{ij} = flux-to-dose conversion factor for nuclide i at energy E_j (Rem-cm^2-sec/Mev-hr)

and the constant 3.7 x 10^6 has the units Dis-m^2/sec-Ci-cm^2. The gamma dose to control room personnel due to a cloud external to the control room is determined with Equation 15A-6:

$$D_\gamma = (\chi / Q)_{CR} \sum_i Q_i \, CF_i \qquad (15A\text{-}6)$$

where:

D_γ = gamma whole-body dose (Rem)

$(\chi/Q)_{CR}$ = atmospheric dispersion value for the main control room (sec/m^3)

Q_i = total activity in external cloud for nuclide i (Ci)

CF_i = dose conversion factor for nuclide i, given in Table 15A-1 (Rem-m^3/Ci-sec)

15A.2 Updated Dose Calculation Models

Commencing with analyses performed in 1996, the whole body gamma dose, beta skin dose, and thyroid dose commitments described in Section 15A.1 have been calculated using the dose quantities described in this section.

Effective Dose Equivalent (EDE) as described in ICRP-26, replaces the traditional whole body gamma dose quantity. Like the whole body dose it replaces, the EDE model assumes that the receptor is immersed in a semi-infinite cloud. The EDE model estimates the dose to each organ in the body due to this cloud, applies a weighting factor to each organ, and sums the weighted doses to obtain the EDE. This quantity is applicable to the WGSR.

$$D_{EDE} = \chi / Q \bullet \sum_i (Q_i \bullet C_{EDE_i}) \qquad (15A\text{-}7)$$

where:

D_{EDE} = Effective Dose Equivalent (EDE)

Q_i = Activity of nuclide i released

χ/Q = Atmospheric dispersion factor

C_{EDE_i} = Dose conversion factor for nuclide i (DOE/EH-0070, 1988)

For the control room dose analyses, the EDE is corrected to account for the finite volume of the control room using the method of Murphy-Campe:

$$D_{EDE_{CR}} = \chi / Q \bullet \frac{V^{0.338}}{1173} \bullet \sum_i (Q_i \bullet C_{EDE_i}) \qquad (15A\text{-}8)$$

where:

$D_{EDE_{CR}}$ = Effective Dose Equivalent (EDE) for control room

V = volume of control room, ft^3

Skin Dose Equivalent (skin DE) as described in ICRP-26, replaces the traditional beta skin dose quantity. Based on immersion in a semi-infinite cloud. This quantity is applicable to the WGSR.

$$D_{SKIN} = \chi / Q \bullet \sum_i (Q_i \bullet C_{SKIN_i}) \qquad (15A\text{-}9)$$

where:

D_{SKIN} = Effective Dose Equivalent (skin DE)

Q_i = Activity of nuclide i released

χ/Q = Atmospheric dispersion factor

C_{SKIN_i} = Dose conversion factor for nuclide i (DOE/EH-0070, 1988)

Thyroid Committed Dose Equivalent (thyroid CDE) as described in ICRP-26 and ICRP-30, replaces the traditional thyroid dose quantity based on the critical organ model of ICRP-2 used in TID-14844. This quantity is applicable to the WGSR.

$$D_{CDE_{thy}} = \chi / Q \bullet \sum_i (Q_i \bullet C_{CDE_i} \bullet BR) \qquad (15A\text{-}10)$$

where:

$D_{CDE_{thy}}$ = Thyroid Committed Dose Equivalent (CDE)

Q_i = Activity of iodine isotope i released

χ/Q = Atmospheric dispersion factor

BR = Breathing rate

= 3.47E-4 m^3/sec, 0-8 hours

= 1.75E-4 m^3/sec, 0-24 hours

= 2.32E-4 m^3/sec, > 24 hours

= 3.47E-4 m^3/sec, 0-30 days control room analysis

C_{CDE_i} = Dose conversion factor for nuclide i (USEPA FGR11, 1988)

Committed Effective Dose Equivalent (CEDE) as described in ICRP-26 and ICRP-30. This value represents the dose due to intake of radioactive material. This dose quantity is calculated and used only when analyses are performed pursuant to Regulatory Guide 1.183. This quantity is not applicable to the WGSR.

$$D_{CEDE} = \chi/Q \bullet \sum_i (Q_i \bullet C_{CEDE_i} \bullet BR) \qquad (15A\text{-}11)$$

where:

D_{CEDE} = Committed Effective Dose Equivalent

Q_i = Activity of nuclide i released

χ/Q = Atmospheric dispersion factor

BR = Breathing rate

3.5E-04 m^3/sec, 0-8 hours (Offsite)
1.8E-04 m^3/sec, 8-24 hours (Offsite)
2.3E-04 m^3/sec, >24 hours (Offsite)
3.5E-04 m^3/sec, 0-30 days (Control Room)

C_{CEDE_i} = Dose conversion factor for nuclide i (from USEPA FGR 11, 1988)

Total Effective Dose Equivalent (TEDE) as described in ICRP-26 and ICRP-30. This dose quantity is calculated and used only when analyses are performed pursuant to Regulatory Guide 1.183. This quantity is not applicable to the WGSR.

TEDE = EDE + CEDE

Determination of dose inside the control room from cloud external to the control room.

The gamma dose to control room personnel due to a cloud external to the control room is determined with Equation 15A-6. The difference between gamma effective dose equivalent (gamma EDE) and the gamma dose calculated with Equation 15A-6 is assumed to be negligible.

15A.2a Dose Calculation Models for Accidents other than the WGSR

The Total Effective Dose Equivalent (TEDE) at the site boundary and in the control room is calculated by computer code PERC2. PERC2 is a multiple compartment activity transport code which calculates the Committed Effective Dose Equivalent (CEDE) from inhalation and the Deep Dose Equivalent (DDE) from submersion due to halogens, noble gases and other nuclides at the offsite locations and in the control room. The TEDE is the sum of CEDE and DDE.

Committed Effective Dose Equivalent (CEDE) Inhalation Dose - The CEDE dose is calculated by the same equation given in section 15A.2 (Equation 15A-11). The dose conversion factors are taken from USEPA Federal Guidance Report No. 11, 1988. The control room dose is adjusted by the appropriate occupancy factors.

Deep Dose Equivalent (DDE) from External Exposure - According to the guidance provided in Section 4.1.4 and Section 4.2.7 of RG 1.183, the Effective Dose Equivalent (EDE) may be used in lieu of DDE in determining the contribution of external dose to the TEDE if the whole body is irradiated uniformly. The EDE in the control room is based on a finite cloud model that addresses buildup and attenuation in air. The dose equation is based on the assumption that the dose point is at the center of a hemisphere of the same volume as the control room. The dose rate at that point is calculated as the sum of typical differential shell elements at a radius R. The dose rate equation is the same as Eq.15A-4, except that the flux to dose rate conversion factors are taken from ANSI/ANS 6.1.1-1991. The flux to dose rate conversion factors are presented in Table 15A-1a. The Deep Dose Equivalent at the EAB and LPZ locations is calculated using the semi-infinite cloud model as described by Eq.15A-2.

15A.3 Dose Calculations Computer Programs

The following is a list of computer programs that are used to calculate source terms and radiological consequences for the DBAs discussed in Section 15.

ACTIVITY 2

Program ACTIVITY 2 calculates the concentration of fission products in the fuel, coolant, waste gas decay tanks, ion exchangers, miscellaneous tanks, and release lines to the atmosphere for a pressurized water reactor system. The program uses a library of properties of more than 100 significant fission products and may be modified to include as many as 200 nuclides. The program output presents the activity and energy spectrum at the selected part of the system for any specified operating time.

RADIOISOTOPE

Program RADIOISOTOPE calculates the activity of nuclides in a closed system by solving the appropriate decay-purification equations. Based on the activity of any nuclide in the system at some initial time, the program calculates the activity of that nuclide and its products at any later time. The program also calculates the specific gamma activity for each of seven fixed energy groups.

ION EXCHANGER

Program ION EXCHANGER calculates the activity of nuclides in an ion exchanger or tank of a nuclear reactor plant by solving the appropriate growth-decay-purification equations. Based on a known feed rate of primary coolant or other fluid with known radionuclide activities, it calculates the activity of each nuclide and its products in the ion exchanger or tank at some later time. The program also calculates the specific gamma activity for each of the seven fixed energy groups.

REM123 [HISTORICAL]

Program REM123 calculates the gamma, beta, and thyroid doses at the site boundary and LPZ according to Regulatory Guides 1.3 and 1.4. The program considers various isotopes of krypton, xenon, and iodine. Built-in constants include average beta and gamma energies, breathing rates, time intervals, and thyroid dose conversion factors.

DRAGON 4 [HISTORICAL]

Program DRAGON 4 evaluates the activities, dose rates, and integrated doses in the containment building and main control room of a nuclear facility or at a vicinal site following the release of halogens and noble gases from some control volume. The fission product release to the atmosphere, together with the activities and integrated activities of the halogens that are accumulated in the system, are also computed. DRAGON 4 has capabilities which include analysis of accident (instantaneous release) conditions and long-term continuous source leakage. Site dose calculations use the semi-infinite cloud models suggested by Regulatory Guides 1.3 and 1.4. The gamma dose in the main control room is computed based on the finite cloud model (Equation 15A-4). Average beta and gamma energies are used in all dose calculations.

QADMOD

Program QADMOD calculates dose rates at a series of detector locations with shielding for a number of different source points representing volumetric sources. The program is a modified version of the QAD P-5 program written at the Los Alamos Scientific Laboratory by R. E. Malenfant. This program has been upgraded to include: 1) the FASTER geometry routines, 2) a point source option, 3) a translated cylindrical source volume option, and 4) internal library data for conversion factors, build-up factor coefficients, and mass attenuation factors for several materials and compositions.

TRAILS_PC

Program TRAILS_PC performs calculations involving transport of radioactive species between compartments that are related by first order linear processes. It is specifically structured to evaluate the transport of radioactivity in design basis accidents, and for calculating dose rates and doses at a user defined offsite location and in the control room. This code was developed and tested at BVPS and has been benchmarked against the SWEC DRAGON code.

PERC2

Program PERC2 is identical to DRAGON in terms of the environmental transport and dose conversion, but it includes the following:

- Provision of time-dependent releases from the reactor coolant system to the containment atmosphere.
- Provision for airborne radionuclides other than noble gas and iodine, including daughter ingrowth.
- Provision for calculating organ doses other than thyroid.
- Provisions for tracking time-dependent inventories of all radionuclides in all control regions of the plant model.
- Provision for calculating energies as well as activities for the inventoried radionuclides to permit direct equipment qualification and vital area access assessment.

ASCOT_PC [HISTORICAL]

Program ASCOT_PC models the transport of radioactivity released through the containment to the environment and main control room. ASCOT_PC is based on the program TRAILS_PC, but provides for a two-region (sprayed and unsprayed) modeling of the containment. ASCOT, an earlier variant, has similar capabilities with the exception of decay progeny ingrowth.

QADCG and QAD/CGGP_PC

These programs are variants of the QAD point kernel shielding program written at the Los Alamos Scientific Laboratory by R. E. Malenfant. The QADCG version implements a combinatorial geometry method of describing problem geometry. The QAD/CGGP_PC version implements combinatorial geometry and the geometric progression build-up factor algorithm. At BVPS, these codes are used only for proton shielding calculations.

ORIGEN-S

Program ORIGEN-S calculates fuel depletion, actinide transmutation, fission product buildup and decay and associated radiation source terms. At BVPS, ORIGEN-S has been used to develop reactor core inventory, and decayed inventories after various cool down times. These values are used in design basis radiological consequence analyses. This code was developed for the NRC at the Oak Ridge National Laboratory. This code is documented as part of the SCALE package in NUREG/CR-0200.

15A.4 References for Section 15A

DiNunno, J. J.; Anderson, F.D.; Baker, R.E.; and Waterfield, R.L. 1962. Calculation of Distance Factors for Power and Test Reactor Sites, TID 14844.

U.S. Atomic Energy Commission (USAEC) 1974. Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors. Regulatory Guide 1.4, Revision 2.

Kocher, D. C., External Dose-Rate Conversion Factors for Calculation of Dose to the Public, DOE/EH-0070, 1988

Eckerman, K. F., et al, Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion, EPA-520/1-88-020, 1988

ICRP, Recommendations of the International Commission of Radiological Protection, ICRP Publication 26, 1977

ICRP, Limits for Intakes of Radionuclides by Workers, ICRP Publication 30, 1979

Murphy, K. G. and Campe, K. W., Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19, published in proceedings of 13th AEC Air Cleaning Conference

US NRC NUREG/CR-0200, ORIGEN-S: Scale System Module to Calculate Fuel depletion, Actinide Transmutation, Fission Product Buildup and Decay, and associated Radiation Source Terms

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

K. R. Eckerman and J. C. Ryan, External Exposure to Radionuclides in Air, Water and Soil (Federal Guidance Report 12), EPA 402-R-93-081, 1993.

ANSI/ANS 6.1.1-1991, "Neutron and Gamma-ray Fluence-to-dose Factors"

Tables for Appendix 15A

TABLE 15A-1

DOSE CONVERSION FACTORS FOR CONTROL ROOM
DOSE DUE TO EXTERNAL CLOUD

Nuclide	Conversion Factor $\left[\frac{(Rem/sec)}{(Ci/m^3)}\right]$
Kr-83m	0.0
Kr-85m	$2.9x10^{-7}$
Kr-85	$2.1x10^{-7}$
Kr-87	$4.2x10^{-3}$
Kr-88	$5.4x10^{-3}$
Kr-89	$3.3x10^{-3}$
Xe-131m	$1.2x10^{-6}$
Xe-133m	$1.7x10^{-6}$
Xe-133	$6.0x10^{-7}$
Xe-135m	$4.8x10^{-5}$
Xe-135	$3.4x10^{-6}$
Xe-137	$2.8x10^{-5}$
Xe-138	$8.1x10^{-4}$
I-131	$9.5x10^{-6}$
I-132	$8.2x10^{-4}$
I-133	$9.7x10^{-5}$
I-134	$8.2x10^{-4}$
I-135	$1.7x10^{-3}$

TABLE 15A-1a

FLUX-TO-DOSE CONVERSION FACTORS
(Used in SWEC PERC2 Code for the Re-analysis of LOCA and CREA)

Energy Mean MeV	(rem/hr)/ $(MeV/cm^2\text{-}s)$
0.01	7.8E-07
0.025	1.7E-06
0.0375	1.5E-06
0.0575	1.2E-06
0.085	1.1E-06
0.125	1.0E-06
0.225	1.1E-06
0.375	1.2E-06
0.575	1.2E-06
0.85	1.2E-06
1.25	1.2E-06
1.75	1.1E-06
2.25	1.1E-06
2.75	1.0E-06
3.5	9.5E-07
5.0	8.7E-07
7.0	8.2E-07
9.5	7.8E-07