15.0 ACCIDENT ANALYSES

15.0.1 General

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

Certain items of Table 15-1 in the guide warrant comment, as follows:

1. Items 1.3 and 2.1 - There are no pressure regulators in the Nuclear Steam Supply System (NSSS) pressurized water reactor (PWR) design whose malfunction or failure could cause a steam flow transient.

2. Item 6.2 - No instrument lines from the reactor coolant pressure boundary in the NSSS PWR design penetrate the Containment. (For the definition of the Reactor Coolant System boundary, refer to Section 5, ANSI-N18.2, "Nuclear Safety Criteria for the Design of Stationary PWR Plants," 1973.)

15.0.2 Classification of Plant Conditions

Since 1970 the ANS 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 and engineered safeguards functioning is assumed to the extent allowed by considerations, such as the single failure criterion, in fulfilling this principle.

15.0.2.1 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 plant parameter and the value of that parameter which would require either automatic or manual protective action. Inasmuch as Condition I occurrences occur frequently or regularly, they must be considered from the point of view of affecting the consequences of fault conditions (Conditions II, III and IV). In this regard, analysis of each fault condition described is generally based on a conservative set of initial conditions corresponding to adverse conditions which can occur during Condition I operation.

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 - Startup ($K_{eff} \geq 0.99$, $\leq$ 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 ($K_{eff} < 0.99$, $200°F \leq T_{AVG} \leq 350°F$).

e. Mode 5 - Cold Shutdown (K_{eff} < 0.99, T_{avg} < 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 plant Technical Specifications must be considered in conjunction with other operational modes. These include:

a. Operation with components or systems out of service (such as power operation with a reactor coolant pump out of service).

b. Radioactivity in the reactor coolant, due to leakage from fuel with cladding defects and other sources.

1) Fission products
2) Corrosion products
3) Tritium

c. Operation with steam generator primary-to-secondary leakage up to the maximum allowed by the Technical Specifications.

d. Testing as required by the Technical Specifications.

3. Operational transients

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

b. Step load changes (up to $\pm$ 10 percent).

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

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

15.0.2.2 Condition II - Faults of Moderate Frequency

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

The following faults are included in this category:

1. Feedwater system malfunctions causing a reduction in feedwater temperature (Subsection 15.1.1 of RESAR-SP/90 PDA Module 8, "Steam and Power Conversion").

2. Feedwater system malfunctions causing an increase in feedwater flow (Subsection 15.1.2 of RESAR-SP/90 PDA Module 8, "Steam and Power Conversion").

3. Excessive increase in secondary steam flow (Subsection 15.1.3 of RESAR-SP/90 PDA Module 8, "Steam and Power Conversion").

4. Inadvertent opening of a steam generator relief or safety valve causing a depressurization of the main steam system (Subsection 15.1.4 of RESAR-SP/90 PDA Module 6, "Secondary Side Safeguards System").

5. Loss of external load (Subsection 15.2.2 of RESAR-SP/90 PDA Module 8, "Steam and Power Conversion").

6. Turbine trip (Subsection 15.2.3 of RESAR-SP/90 PDA Module 8, "Steam and Power Conversion").

7. Inadvertent closure of main steam isolation valves (Subsection 15.2.4 of RESAR-SP/90 PDA Module 8, "Steam and Power Conversion").

8. Loss of condenser vacuum and other events resulting in turbine trip (Subsection 15.2.5 of RESAR-SP/90 PDA Module 8, "Steam and Power Conversion").

9. Loss of nonemergency A-C power to the station auxiliaries (Subsection 15.2.6 of RESAR-SP/90 PDA Module 6, "Secondary Side Safeguards System").

10. Loss of normal feedwater flow (Subsection 15.2.7 of RESAR-SP/90 PDA Module 6, "Secondary Side Safeguards System").

11. Partial loss of forced reactor coolant flow (Subsection 15.3.1 of RESAR-SP/90 PDA Module 4, "Reactor Coolant System").

12. Uncontrolled rod cluster control assembly bank withdrawal from a subcritical or low power startup condition (Subsection 15.4.1 of this module).

13. Uncontrolled rod cluster control assembly bank withdrawal at power (Subsection 15.4.2 of this module).

14. Control rod misalignment - Dropped full length assembly, dropped full length assembly bank, or statically misaligned full length assembly (Subsection 15.4.3 of this module and RESAR-SP/90 PDA Module 9, "I&C and Electric Power").

15. Startup of an inactive reactor coolant loop at an incorrect temperature (Subsection 15.4.4 of RESAR-SP/90 PDA Module 4, "Reactor Coolant System").

16. Chemical and volume control system malfunction that results in a decrease in the boron concentration in the reactor coolant (Subsection 15.4.6 of RESAR-SP/90 PDA Module 13, "Auxiliary Systems").

17. Inadvertent operation of emergency core cooling system during power operation (Subsection 15.5.1 of RESAR-SP/90 PDA Module 1, "Primary Side Safeguards System").

18. Chemical and volume control system malfunction that increases reactor coolant inventory (Subsection 15.5.2 of RESAR-SP/90 PDA Module 13, "Auxiliary Systems").

19. Inadvertent opening of a pressurizer safety or relief valve (Subsection 15.6.1 of RESAR-SP/90 PDA Module 4, "Reactor Coolant System").

20. Failure of small lines carrying primary coolant outside containment (Subsection 15.6.2 of RESAR-SP/90 PDA Module 1, "Primary Side Safeguards System").

15.0.2.3 Condition III - Infrequent Faults

By definition, Condition III occurrences are faults which may occur very infrequently during the life of the plant. They will be accommodated with the failure of only a small fraction of the fuel rods although sufficient fuel damage might occur to preclude 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 reactor coolant system or containment barriers. The following faults are included in this category:

1. Minor steam system piping failures (Subsection 15.1.5 of RESAR-SP/90 PDA Module 6, "Secondary Side Safeguards System").

2. Complete loss of forced reactor coolant flow (Subsection 15.3.2 of RESAR-SP/90 PDA Module 4, "Reactor Coolant System").

3. Control rod misalignment - Single rod cluster control assembly withdrawal at full power (Subsection 15.4.3 of this module).

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

5. Loss of reactor coolant from small ruptured pipes or from cracks in large pipes, which actuate the emergency core cooling system (Subsection 15.6.4 of RESAR-SP/90 PDA Module 1, "Primary Side Safeguards System").

6. Waste gas system failure (Subsection 15.7.1 of RESAR-SP/90 PDA Module 12, "Waste Management").

7. Radioactive liquid waste system leak or failure (atmospheric release) (Subsection 15.7.2 of RESAR-SP/90 PDA Module 12, "Waste Management").

8. Liquid containing tank failure (Subsection 15.7.3 of RESAR-SP/90 PDA Module 12, "Waste Management").

15.0.2.4 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 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 100. 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 emergency core cooling system and containment. The following faults have been classified in this category:

1. Steam system piping failure (Subsection 15.1.5 of RESAR-SP/90 PDA Module 6, "Secondary Side Safeguards System").

2. Feedwater system pipe break (Subsection 15.2.8 of RESAR-SP/90 PDA Module 6, "Secondary Side Safeguards System").

3. Reactor coolant pump rotor seizure (locked rotor) (Subsection 15.3.3 of RESAR-SP/90 PDA Module 4, "Reactor Coolant System").

4. Reactor coolant pump shaft break (Subsection 15.3.4 of RESAR-SP/90 PDA Module 4, "Reactor Coolant System").

5. Spectrum of rod cluster control assembly ejection accidents (Subsection 15.4.8 of this module).

6. Steam generator tube failure (Subsection 15.6.3 of RESAR-SP/90 PDA Module 6, "Secondary Side Safeguards System").

7. Loss-of-coolant accidents resulting from the spectrum of postulated piping breaks within the reactor coolant pressure boundary (Subsection 15.6.4 of RESAR-SP/90 PDA Module 1, "Primary Side Safeguards System").

8. Fuel handling accident (Subsection 15.7.4 of RESAR-SP/90 PDA Module 12, "Waste Management").

15.0.3 Optimization of Control Systems

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

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

15.0.4 Plant Characteristics and Initial Conditions Assumed in the Accident Analyses

15.0.4.1 Design Plant Conditions

Table 15.0-1 gives the guaranteed nuclear steam supply system thermal power output which is assumed in analyses performed in this report. This power output includes the thermal power generated by the reactor coolant pumps and is consistent with the license application rating described in Chapter 1.0. Allowances for errors in the determination of the steady-state power level are made as described in Subsection 15.0.4.2. The values of pertinent plant parameters utilized in the 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.

15.0.4.2 Initial Conditions

For most accidents which are DNB limited, nominal values of initial conditions are assumed. The allowances on power, temperature, and pressure noted above are determined on a statistical basis and are included in the limit DNBR, as described in WCAP-8567 (Reference 1). This procedure is known as the "Improved Thermal Design Procedure," and is discussed more fully in Section 4.4 of this module.

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

1. Core power	± 2% allowance for calorimetric error
2. Average reactor coolant system temperature	± 4°F allowance for controller deadband and measurement error
3. Pressurizer pressure	± 30 psi allowance for steady-state fluctuations and measurement error.

Table 15.0-3 summarizes initial conditions and computer codes used in the accident analysis, and shows which accidents employed a DNB analysis using the Improved Thermal Design Procedure.

15.0.4.3 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 integrated protection system as described in RESAR-SP/90 PDA Module 9, "I&C and Electric Power" and the Technical Specifications. Audible alarms will be activated in the 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 DNB limited both the radial and axial peaking factors are of importance. The core thermal limits illustrated in Figure 15.0-1 are based on a reference axial power shape. The low DNBR reactor trip setpoint is automatically adjusted for axial shapes differing from the reference shape by the method described in Section 4.4 of this module and also described in RESAR-SP/90 PDA Module 9, "I&C and Electric Power" The radial peaking factor $F_{\Delta H}$ increases with decreasing power and with increasing rod insertion. The increase in $F_{\Delta H}$ resulting from decreasing reactor power and increased rod insertion is accounted for in the low DNBR reactor trip through measurement of power and control rod position.

For transients which may be overpower limited, the total peaking factor F_q is of importance. F_q is continuously monitored through the high Kw/ft reactor trip as described in RESAR-SP/90 PDA Module 9, "I&C and Electric Power" and the Technical Specifications to assure that the limiting overpower conditions are not exceeded.

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 of this module. 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 rod cluster control assembly bank withdrawal from subcritical and rod cluster control assembly ejection incidents, which result in a large power rise over a few seconds), a detailed fuel heat transfer calculation is performed. Although the fuel rod thermal time constant is a function of system conditions, fuel burnup, and rod power, a typical value at beginning-of-life for high power rods is approximately 5 seconds.

15.0.5 Reactivity Coefficients Assumed in the Accident Analyses

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

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 reactor coolant system, 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. The justification for use of conservatively large vs. small reactivity coefficient values are 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.

15.0.6 Rod Cluster Control Assembly Insertion Characteristics

The negative reactivity insertion following a reactor trip is a function of the position vs. time of the rod cluster control assemblies and the variation in rod worth as a function of rod position. With respect to accident analyses, the critical parameter is the time of insertion up to the dashpot entry or approximately 85% of the rod cluster travel. For all accidents the insertion time to dashpot entry is conservatively taken as 3.4 seconds. The normalized rod cluster control assembly position vs. time assumed in accident analyses is shown in Figure 15.0-4.

Figure 15.0-5 shows the fraction of total negative reactivity insertion vs. 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 vs. 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 rod cluster control assembly negative reactivity insertion vs. time is shown in Figure 15.0-6. The curve shown in this figure was obtained from Figures 15.0-4 and 15.0-5. A total negative reactivity insertion following a trip of 4% $\Delta\rho$ 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 of this module.

The normalized rod cluster control assembly negative reactivity insertion vs. time curve for an axial power distribution skewed to the bottom (Figure 15.0-6) is used for 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 by the reactor kinetics code and is not separable from the other reactivity feedback effects. In this case, the rod cluster control assembly position vs. time (Figure 15.0-4) is used as code input.

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

A reactor trip signal acts to open eight trip breakers, two per channel set, feeding power to the control rod drive mechanisms. The loss of power to the mechanism coils causes the mechanisms to release the rod cluster control assemblies which then fall by gravity into the core. There are various instrumentation delays associated with each trip function, including delays in signal actuation, in opening the trip breakers, and in the release of the rods by the mechanism. The total delay to trip is defined as the time delay from the time that trip conditions are reached to the time the rods are free and begin to fall. Limiting trip setpoints assumed in accident analyses and the time delay assumed for each trip function are given in Table 15.0-4.

Reference is made in Table 15.0-4 to the low DNBR trips shown in Figure 15.0-1. These figures present the allowable reactor power as a function of the coolant loop inlet temperature and primary coolant pressure for N loop operation (4-loop operation), for the design flow and power distribution, as described in Section 4.4 of this module.

The boundaries of operation defined by the low DNBR trip are represented as "protection lines" on this diagram. The protection lines are drawn to include all adverse instrumentation and setpoint errors so that under nominal conditions trip would occur well within the area bounded by these lines. The DNB lines represent the locus of conditions for which the DNBR equals the limit value of 1.62. All points below and to the left of a DNB line for a given pressure have DNBR greater than the limit value with the assumed axial and radial power distributions. 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 setpoint); high pressure (fixed setpoint); low pressure (fixed setpoint); low DNBR (variable setpoint); high kw/ft (fixed setpoint).

The limit value, which was used as the DNBR limit for all accidents analyzed with the Improved Thermal Design Procedure (see Table 15.0-3), is conservative compared to the actual design DNBR value required to meet the DNB design basis is discussed in Section 4.4 of this module.

The difference between the limiting trip point assumed for the analysis and the nominal trip point represents an allowance for instrumentation channel error and setpoint error. Nominal trip setpoints are specified in the plant Technical Specifications. During plant startup tests, it 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 determined periodically in accordance with the plant Technical Specifications.

15.0.8 Instrumentation Drift and Calorimetric Errors - Power Range Neutron Flux

The instrumentation drift and calorimetric errors used in establishing the power range high neutron flux setpoint are presented in Table 15.0-5. 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.

15.0.9 Plant Systems and Components Available for Mitigation of Accident Effects

The Westinghouse nuclear steam supply system (NSSS) is designed to afford power protection againts 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 17.0 of the RESAR-SP/90 integrated PDA document will discuss the quality assurance program which is implemented to ensure that the plant 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 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 non-safety-related rod control system, other than the reactor trip portion of the Control rod drive system (CRDS), 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 control system operation to determine the worst case. The pressurizer heaters are not assumed to be energized during any of the Chapter 15 events.

15.0.10 Fission Product Inventories

15.0.10.1 Inventory in the Core

The time dependent fission product inventories in the reactor core are calculated by the ORIGEN code(10) using a data library based on ENDF/B-IV.(11) Core inventories are shown in Table 15.0-7.

The fission product radiation sources considered to be released from the fuel to the containment following a maximum credible accident are based on the assumptions stated in TID-14844(3): 100 percent of the noble gases and 50 percent of the halogens.

15.0.10.2 Inventory in the Fuel Pellet Clad Gap

The radiation sources associated with a gap activity release accident are based on the assumption that the fission products in the space between the fuel pellets and the cladding of all fuel rods in the core are released as a result of cladding failure.

The gap activities were determined using the model suggested in Regulatory Guide 1.25. Specifically, 10 percent of the iodine and noble gas activity (except Kr-85, I-127, and I-129, which are 30 percent) is accumulated in the fuel clad gap. The gap activities are shown in Table 15.0-7.

15.0.10.3 Inventory in the Reactor Coolant

Reactor coolant iodine concentrations for the Technical Specification limit of 1 μCi/gm of dose equivalent (D.E.) I-131 and for the assumed pre-accident iodine spike concentration of 60 μCi/gm of D.E. I-131 are presented in Table 15.0-8. Reactor coolant noble gas concentrations based on 1 percent fuel defects are presented in Table 15.0-9. Iodine appearance rates in the reactor coolant, for normal steady state operation at 1 μCi/gm of D.E. I-131, and for an assumed accident initiated iodine spike are presented in Table 15.0-10.

15.0.11 Residual Decay Heat

15.0.11.1 Total Residual Heat

Residual heat in a subcritical core is calculated for the loss-of-coolant accident per the requirements of Appendix K, 10 CFR 50.46, as described in References 5 and 6. 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.

15.0.12 Computer Codes Utilized

Summaries of some of the principal computer codes used in transient analyses are given below. Other codes, in particular, very specialized codes in which the modeling has been developed to simulate one given accident, such as those used in the analysis of the reactor coolant system pipe 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.

15.0.12.1. FACTRAN

FACTRAN calculates the transient temperature distribution in a cross section of a metal clad UO_2 fuel rod and the transient heat flux at the surface of the cladding using as input the nuclear power and the time-dependent coolant parameters (pressure, flow, temperature, and density). The codes 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 transient: film boiling heat transfer correlations, Zircaloy-water reaction and partial melting of the materials.

FACTRAN is further discussed in Reference 7.

15.0.12.2 LOFTRAN

The LOFTRAN program is used for studies of transient response of a pressurized water reactor 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 reactor protection system is simulated to include reactor trips on neutron flux, low DNBR, high linear power (kW/ft), 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. 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 in Figure 15.0-1. The core limits represent the minimum value of DNBR as calculated for typical, small thimble, large thimble, corner or side cell.

LOFTRAN is further discussed in Reference 8.

15.0.12.3 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 2000 spatial points, and performs its own steady state initialization. Aside from basic cross section data and thermal-hydraulic parameters, the code accepts as input basic driving functions such as inlet temperature, pressure, flow, boron concentration, control rod motion. Various edits are provided, e.g., 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 in Reference 9.

15.0.12.4 THINC

The THINC Code is described in Section 4.4.

15.0.13 REFERENCES

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9. D. H. Risher, Jr., and R. F. Barry, "TWINKLE - A Multi-Dimensional Neutron Kinetics Computer Code": WCAP-7979-P-A (Proprietary) January 1975, and WCAP-8028-A, (Non-Proprietary), January 1975.

10. Bell, M. J., "ORIGEN - The ORNL Isotope Generation and Depletion Code," ORNL-4628, May 1973.

11. "ORIGEN Yields and Cross Sections - Nuclear Transmutation and Decay Data From END F/B-IV", Radiation Shielding Information Center, Oak Ridge National Laboratory, RSIC-DLC-38, September 1975.

TABLE 15.0-1

NUCLEAR STEAM SUPPLY SYSTEM POWER RATINGS

	N-Loop Operation
Reactor core thermal power output (MWt)*	3800
Thermal power generated by the reactor coolant pumps (MWt)	16
Guaranteed nuclear steam supply system thermal power output (MWt)	3816

* Radiological consequences based on 3565 (MWt) power level.

TABLE 15.0-2

VALUES OF PEPTINENT PLANT PARAMETERS
UTILIZED IN ACCIDENT ANALYSES*

	N-Loop Operation
Thermal output of nuclear steam supply system (MWt)	3816
Reactor core thermal power output (MWt)	3800
Core inlet temperature (°F)	560.8
Reactor coolant average temperature (°F)	592.6
Reactor coolant system pressure (psia)	2250
Reactor coolant flow per loop (gpm)	97900
Total reactor coolant flow (10^6 lb/hr)	145.0
Total steam flow from NSSS (10^6 lb/hr)	17.14
Steam pressure at steam generator outlet (psia)	1024
Maximum steam moisture content (%)	0.25
Feedwater temperature at steam generator inlet (°F)	450
Average core heat flux (Btu/hr-ft^2)	162960

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* For accident analyses using the improved thermal design procedure.

TABLE 15.0-2a

VALUES OF PERTINENT PLANT PARAMETERS
UTILIZED IN ACCIDENT ANALYSES*

	N-Loop Operation
Thermal output of nuclear steam supply system (MWt)	3816
Reactor core thermal power output (MWt)	3800
Core inlet temperature (°F)	560.7
Reactor coolant average temperature (°F)	592.9
Reactor coolant system pressure (psia)	2250
Reactor coolant flow per loop (gpm)	96900
Total reactor coolant flow (10^6 lb/hr)	143.5
Total steam flow from NSSS (10^6 lb/hr)	17.14
Steam pressure at steam generator outlet (psia)	1024
Maximum steam moisture content (%)	0.25
Feedwater temperature at steam generator inlet (°F)	450
Average core heat flux (Btu/hr-ft^2)	162960

* For accident analyses not using the improved thermal design procedure.

TABLE 15.0-3

SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED

Faults	Computer Codes Utilized	Kinetic Parameters Assumed: Delayed Neutron Fraction	Moder. Density (Δp/gm/cc)	Doppler	DNB Correlation	Improved Thermal Design Proced.	Initial NSSS Thermal Power Output (MWt)	Reactor Vessel Coolant Flow (gpm)	Vessel Average Temp. (°F)	Press. Pressure (psia)	Press. Water Volume (ft3)	Fe T
15.1 Increase in Heat Removal by the Secondary System												
Feedwater System Malfunction Causing an Increase in Feedwater Flow	(See RESAR-SP/90 PDA Module 8, "Steam and Power Conversion System")											
Excessive Increase in Secondary Steam Flow	(See RESAR-SP/90 PDA Module 8, "Steam and Power Conversion System")											
Accidental Depressurization of the Main Steam System	(See RESAR-SP/90 PDA Module 6, "Secondary Side Safeguards System")											
Steam System Piping Failure	(See RESAR-SP/90 PDA Module 6, "Secondary Side Safeguards System")											
15.2 Decrease in Heat Removal by the Secondary System												
Loss of External Electrical Load and/or Turbine Trip	(See RESAR-SP/90 PDA Module 8, "Steam and Power Conversion System")											
Loss on Non-Emergency A-C Power to the Station Auxiliaries	(See RESAR-SP/90 PDA Module 6, "Secondary Side Safeguards System")											
Loss of Normal Feedwater Flow	(See RESAR-SP/90 PDA Module 6, "Secondary Side Safeguards System")											
Feedwater System Pipe Break	(See RESAR-SP/90 PDA Module 6, "Secondary Side Safeguards System")											

TABLE 15.0-3 (Con't)

Kinetic Parameters Assumed

Faults	Computer Codes Utilized	Delayed Neutron Fraction	Moder. Density (Δρ/gm/cc)	Doppler	DNB Correlation	Improved Thermal Design Proced.	Initial NSSS Thermal Power Output (MWt)	Reactor Vessel Coolant Flow (gpm)	Vessel Average Temp. (°F)	Press. Pressure (psia)	Press. Water Volume (ft3)	Fee Te (
15.3 Decrease in Reactor Coolant System Flow Rate												
Partial and Complete Loss of Forced Reactor Coolant Flow	(See RESAR-SP/90 PDA Module 4, "Reactor Coolant System")											
Reactor Coolant Pump Shaft Seizure (locked rotor	(See RESAR-SP/90 PDA Module 4, "Reactor Coolant System")											
15.4 Reactivity and power distribution anomalies												
Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical or Low Power Startup Condition.	TWINKLE, FACTRAN, THINC	.0075	Refer to Subsection 15.4.1.2	Minimum*	WRB-2	Yes	0	387600	567	2280	NA	N
Uncontrolled Rod Cluster Assembly Bank Withdrawal at Power	LOFTRAN	.0044	0.0 & .409	Minimum Maximum*	WRB-2	Yes	3816/2290	391600	592.6/ 582.4	2250	1545/ 1185	4 3
Control Rod Mis-allignment	THINC	NA	NA	NA	WRB-2	Yes	3816	391600	592.6	2250	NA	N
Startup of an Inactive Reactor Coolant Loop at an Incorrect Temperature	(See RESAR-SP/90 PDA Module 4, "Reactor Coolant System")											

TABLE 15.0-3 (Con't)

Kinetic Parameters Assumed

Faults	Computer Codes Utilized	Delayed Neutron Fraction	Moder. Density (Δp/gm/cc)	Doppler	DNB Correlation	Improved Thermal Design Proced.	Initial NSSS Thermal Power Output (MWt)	Reactor Vessel Coolant Flow (gpm)	Vessel Average Temp. (°F)	Press. Pressure (psia)	Press. Water Volume (ft3)	Feedwat Temp. (°F)
Chemical and Volume Control System Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant	(See RESAR-SP/90 PDA Module 13, "Auxiliary Systems")											
Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position	Refer to Section 4.3	NA	NA	NA	NA	NA	3816	NA	NA	NA	NA	NA
Spectrum of Rod Cluster Control Assembly Ejection Accidents	TWINKLE, FACTRAN	.0055/ .0044	Refer to Subsection 15.4.8 (BOC, EOC)	Minimum*	NA	NA	3816 0	387600	560.5 567	NA	NA	NA
15.5 Increase in Coolant Inventory												
Inadvertent Operation of ECCS During Power Operation	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA
15.6 Decrease in Reactor Coolant Inventory												
Inadvertent Opening of a Pressurizer Safety or Relief Valve	(See RESAR-SP/90 PDA Module 4, "Reactor Coolant System")											

* Reference Figure 15.0-2. Maximum refers to lower curve and minimum refers to upper curve.
NA - Not applicable.
BOC - Beginning of cycle
EOC - End of cycle

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	118%	0.5
Power Range high neutron flux, low setting	35%	0.5
Power range neutron flux, high negative rate	*	*
High neutron flux, P-8	85%	0.5
Low DNBR	Variable, see Figure 15.0-1	6.0**
High pressurizer pressure	2410 psig	2.0
Low pressurizer pressure	1836 psia	2.0
Low reactor coolant flow (from loop flow detectors)	87% loop flow	1.0
RCP underspeed	92% or nominal speed	0.6
Turbine trip	Not applicable	2.0
Safety injection reactor trip	Not applicable	2.0
Low steam generator level	***	***
High steam generator level - produces feedwater isolation and turbine trip	***	***

* See RESAR-SP/90 PDA Module 9, "I&C and Electric Power"

** Total time delay (including RTD time response and trip circuit channel electronics delay) from the time the temperature in the coolant loops exceeds the trip setpoint until the rods are free to fall.

*** See RESAR-SP/90 PDA Module 6, "Secondary Side Safeguards System"

TABLE 15.0-5

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

Variable	Accuracy of Measurement of Variable (% error)	Effect on Thermal Power Determination (% error) (Estimated)	(Assumed)
Calorimetric errors in the measurement of secondary system thermal power:			
Feedwater temperature	± 0.5		
Feedwater pressure (small correction on enthalpy)	± 0.5	0.3	
Steam pressure (small correction on enthalpy	± 2		
Feedwater flow	± 1.25	1.25	
Assumed calorimetric error (% of rated power)			± 2(a)
Axial power distribution effects on total ion chamber current			
Estimated error (% rated power)		3	
Assumed error (% of rated power)			± 5(b)
Instrumentation channel drift and setpoint reproducibility			
Estimated error (% or rated power)		1	
Assumed error (% of rated power)			± 2 (c)
Total assumed error in setpoint (a) + (b) + (c)			± 9

TABLE 15.0-5 (Con't)

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

TABLE 15.0-6

PLANT SYSTEMS AND EQUIPMENT AVAILABLE FOR TRANSIENT AND ACCIDENT CONDITIONS

Incident	Reactor Trip Functions	ESF Actuation Functions	Other Equipment	ESF Equip
15.1 Increase in Heat Removed by the Secondary System				
Feedwater System Malfunction Causing an Increase in Feed-water Flow	Power range high flux, high steam generator level, manual, low DNBR, high kw/ft	High steam generator level-produced feedwater isolation and turbine trip	Feedwater isolation valves	NA
Excessive Increase Secondary Steam Flow	Power range high flux, manual, low DNBR, high kw/ft	NA	Pressurizer self-actuated safety valves; steam generator safety valves	NA
Accidental Depres-surization of the Main Steam System	Low pressurizer pressure, manual, SIS	Low pressurizer pressure, low compensated steam line pressure, Hi-1 containment pressure, manual, low 4 T_{cold}	Feedwater isolation valves, steamline stop valves	Auxiliary fee System Safety Injection Sys
Steam System Piping Failure	SIS, low pressurizer pressure, manual	Low pressurizer pressure, low compensated steam-line pressure, Hi-1 containment pressure, manual, low 4 T_{cold}	Feedwater isolation valves, steamline stop valves	Auxiliary fee system; Safet Injection Sys
15.2 Decrease in Heat Removal by the Secondary System				
Loss of External Electrical Load/ Turbine Trip	High pressurizer pressure, low DNBR, low steam generator level, manual	Low steam generator level	Pressurizer safety valves, steam generator	Auxiliary feed system

TABLE 15.0-6 (Con't)

Incident	Reactor Trip Functions	ESF Actuation Functions	Other Equipment	ESF Equip
Loss of Non-Emergency A-C Power to the Station Auxiliaries	Steam generator low level, manual	Steam generator low level	Steam generator safety valves	Auxiliary fee system
Loss of Normal Feedwater Flow	Steam generator low level, manual	Steam generator low level	Steam generator safety valves	Auxiliary fee system
Feedwater System Pipe Break	Steam generator low level, high pressurizer pressure, SIS, manual low DNBR	Hi-1 containment pressure, steam generator low level, low compensated steamline pressure	Steamline isolation valves, feedline isolation, pressurizer safety valves steam generator safety valves	Auxiliary fee system, Safet Injection Sys
15.3 Decrease in Reactor Coolant System Flow Rate				
Partial and Complete Loss of Forced Reactor Coolant Flow	Low flow, low RCP speed, manaul	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
15.4 Reactivity and Power Distribution Anomalies				
Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical or low Power Startup Condition	Power range high flux (low s.p.), manual	NA	NA	NA

TABLE 15.0-6 (Con't)

Incident	Reactor Trip Functions	ESF Actuation Functions	Other Equipment	ESF Equip
Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power	Power range high flux, Hi pressurizer pressure, manual, low DNBR	NA	Pressurizer safety valves, steam generator safety valves	NA
Control Rod Misalignment	Power range negative flux rate, manual	NA	NA	NA
Startup of an Inactive Reactor Coolant Loop at an Incorrect Temperature	Power range high flux, P-8, manual	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, low DNBR, high kw/ft	NA	Low insertion limit annunciators for boration, VCT outlet isolation valves	NA
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	NA	NA	NA	NA

TABLE 15.0-6 (Con't)

Incident	Reactor Trip Functions	ESF Actuation Functions	Other Equipment	ESF Equip
15.6 Decrease in Reactor Coolant Inventory				
Inadvertent Opening of a Pressurizer Safety or Relief Valve	Pressurizer low pressure, manual, low DNBR	Low pressurizer pressure	NA	Safety Inject System
Steam Generator Tube Rupture	Reactor trip system	Engineered Safety Features Actuation System	Service Water System, Component Cooling Water System, steam generator safety valves, steam-line stop valves	Emergency Cor Cooling Syste Auxiliary Fee System, Emerg Power Systems
Loss of Coolant Accident from Spectrum of Postulated Piping Breaks within the System	Reactor trip system	Engineered Safety Features Actuation System	Service Water System, Component Cooling Water System, Steam Generator Safety Valves	Emergency Cor Cooling Syste Auxiliary Fee System, Conta Heat Removal Emergency Pow

TABLE 15.0-7

FUEL AND ROD GAP INVENTORIES, CORE (Ci)[a]

Isotope	Core Fuel	Core Gap[b]
I-131	1.0E + 7	1.0E + 6
I-132	1.5E + 8	1.5E + 7
I-133	2.1E + 8	2.1E + 7
I-134	2.3E + 8	2.3E + 7
I-135	2.0E + 8	2.0E + 7
Kr-83m	1.3E + 7	1.3E + 6
Kr-85m	2.9E + 7	2.9E + 6
Kr-85	7.0E + 5	2.1E + 5
Kr-87	5.2E + 7	5.2E + 6
Kr-88	7.5E + 7	7.5E + 6
Kr-89	9.3E + 7	9.3E + 6
Xe-131m	7.5E + 5	7.5E + 4
Xe-133m	3.1E + 7	3.1E + 6
Xe-133	2.0E + 8	2.0E + 7
Xe-135m	4.3E + 7	4.3E + 6
Xe-135	4.5E + 7	4.5E + 6
Xe-138	1.7E + 8	1.7E + 7
I-127	3.0 kg	0.90 kg
I-129	12.2 kg	3.7 kg

a. Three-region equilibrium cycle core at end of life. The three regions have operated at a specific power of 40.03 MWt per metric ton of uranium for 300, 600, and 900 effective full power days, respectively.

b. Gap activity is assumed to be 10 percent of core activity for all isotopes except Kr-85, I-127, and I-129, whose gap activities are assumed to be 30 percent of their core activities (Regulatory Guide 1.25 assumption).

TABLE 15.0-8

REACTOR COOLANT IODINE CONCENTRATIONS FOR

1 μCi/GRAM AND 60 μCI/GRAM OF DOSE EQUIVALENT I-131

Nuclide	Reactor Coolant Concentration (Ci/gm) 1 μCi/gm D.E. I-131	 60 μ Ci/gm D.E. I-131
I-131	0.76	45.6
I-132	0.76	45.6
I-133	1.14	68.4
I-134	0.195	11.7
I-135	0.63	37.8

TABLE 15.0-9

REACTOR COOLANT NOBLE GAS SPECIFIC ACTIVITY
BASED ON ONE PERCENT DEFECTIVE FUEL

Nuclide	Activity (μc/gram)
Kr-85m	2.0
Kr-85	7.3
Kr-87	1.3
Kr-88	3.6
Xe-131m	2.2
Xe-133m	1.7×10^{1}
Xe-133	2.7×10^{2}
Xe-135m	4.8×10^{-1}
Xe-135	7.2
Xe-138	6.4×10^{-1}

TABLE 15.0-10

IODINE APPEARANCE RATES IN THE REACTOR COOLANT (Curies/sec)

	*Equilibrium Appearance Rates Due to Fuel Defects	**Appearance Rates Due to an Accident Initiated Iodine Spike
I-131	3.4×10^{-3}	1.7
I-132	1.8×10^{-2}	9.0
I-133	7.2×10^{-3}	3.6
I-134	1.1×10^{-2}	5.5
I-135	6.8×10^{-3}	3.4

* Based on RCS concentration of 1 µCi/gm of dose equivalent I-131.

** 500 x equilibrium appearance rate.

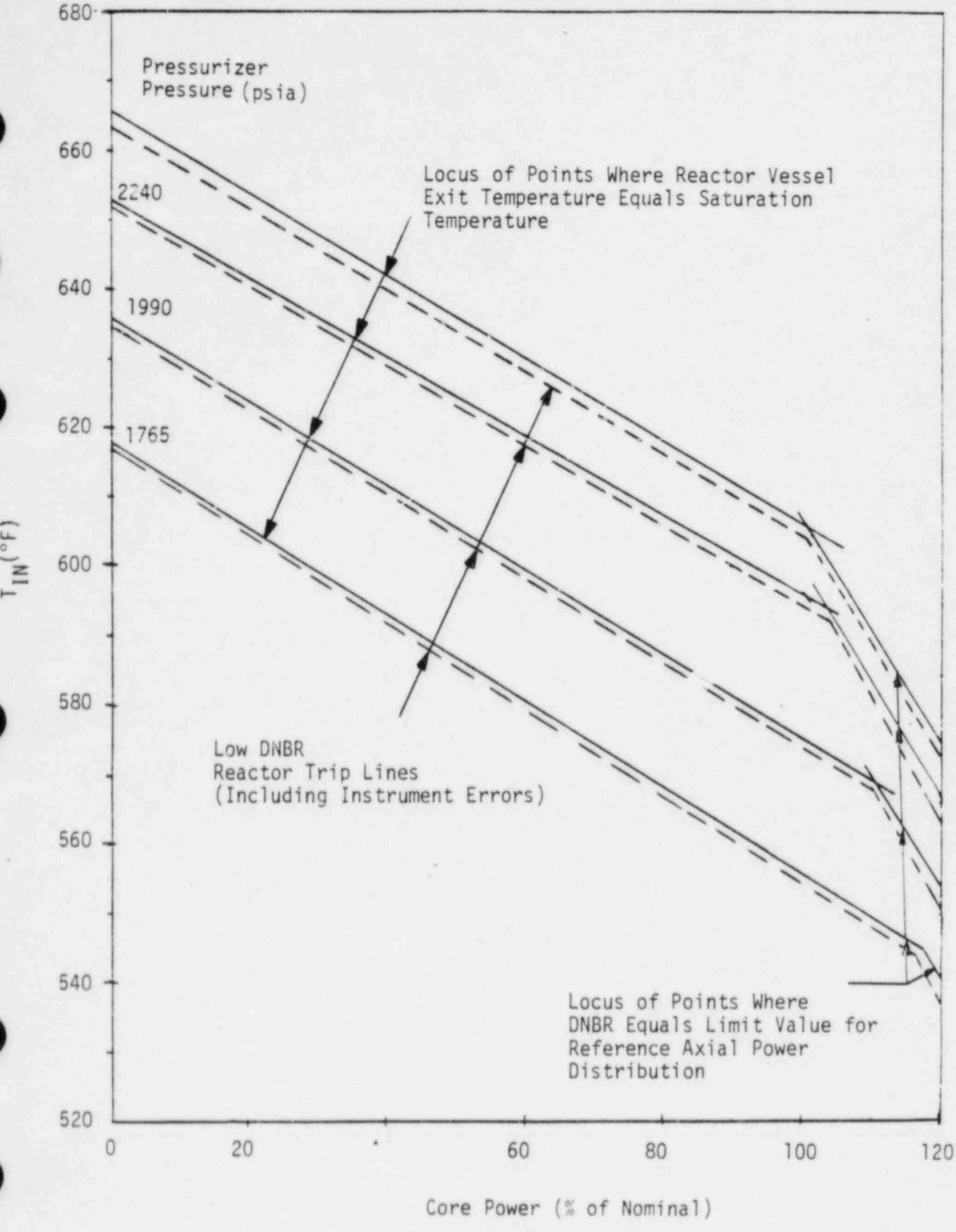
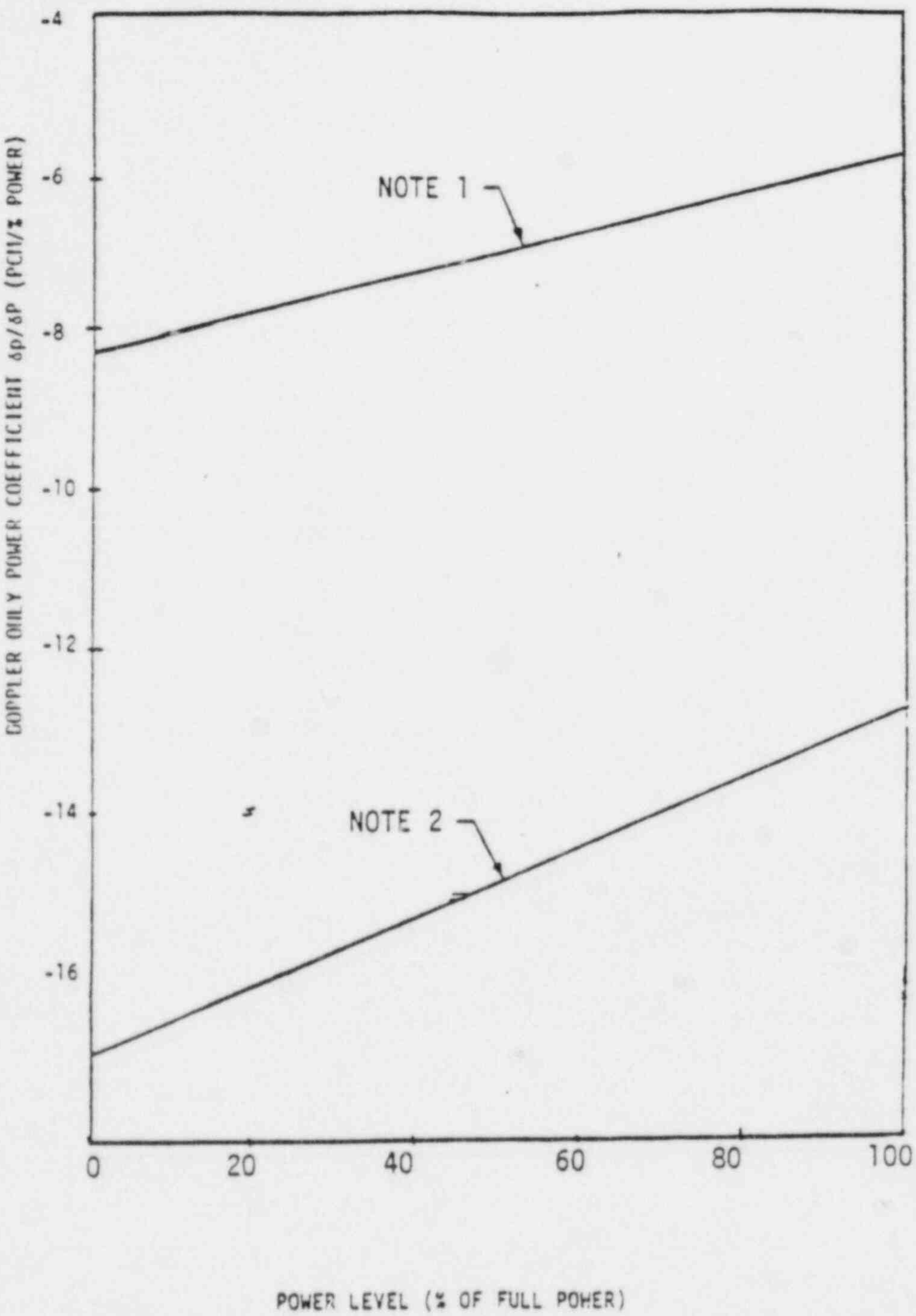


Figure 15.0-1 Illustration of Core Thermal Limits and DNB Protection (N Loop Operation)



Note 1 - Upper Curve, Least Negative Doppler Only Power Defect = -6.95% Δρ (0 to 100% Power)

Note 2 - Lower Curve, Most Negative Doppler Only Power Defect = -1.49% Δρ (0 to 100% Power)

FIGURE 15.0-2 DOPPLER POWER COEFFICIENT USED IN ACCIDENT ANALYSIS

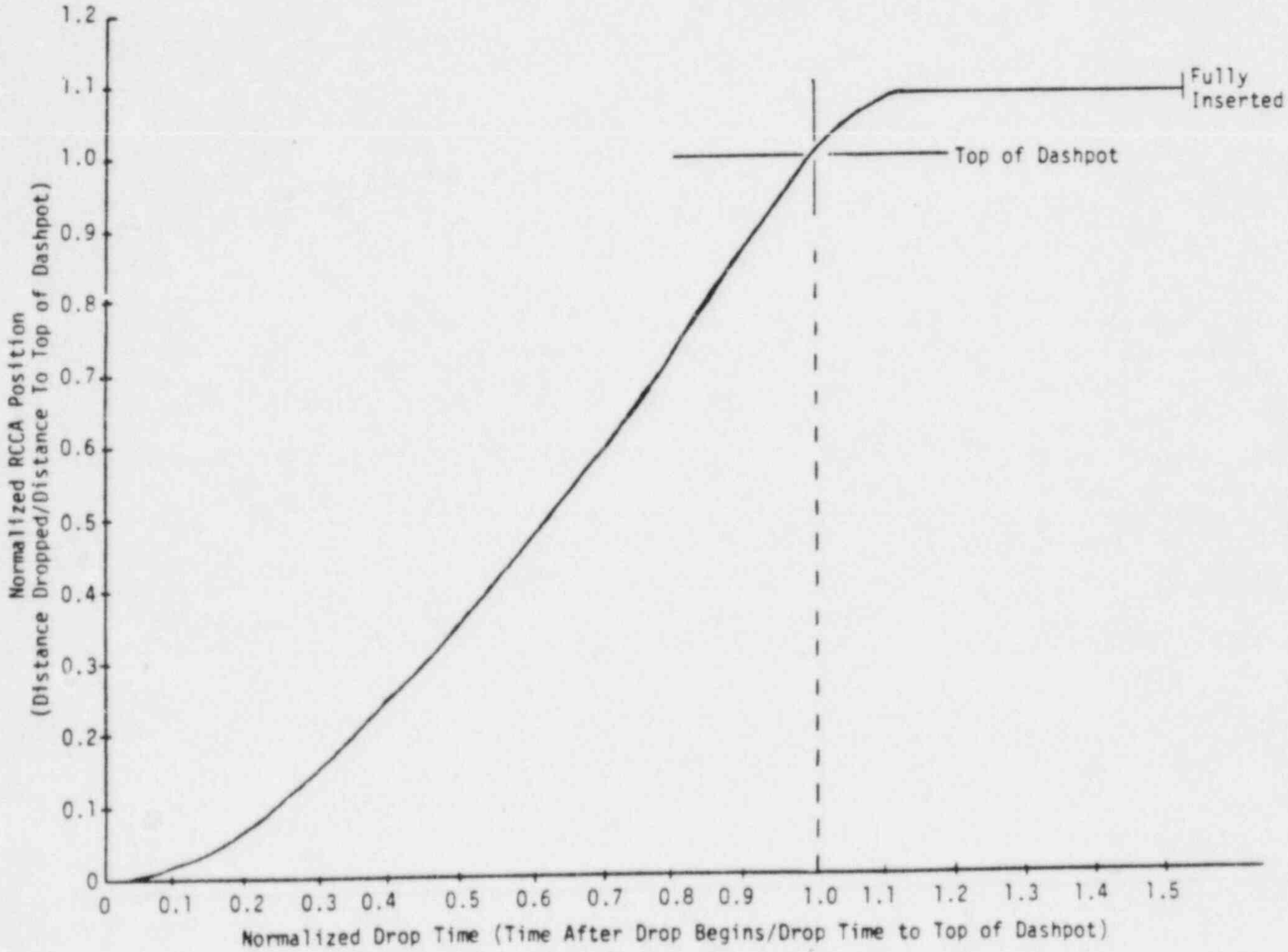


FIGURE 15.0-3 RCCA POSITION VS. TIME TO DASHPOT

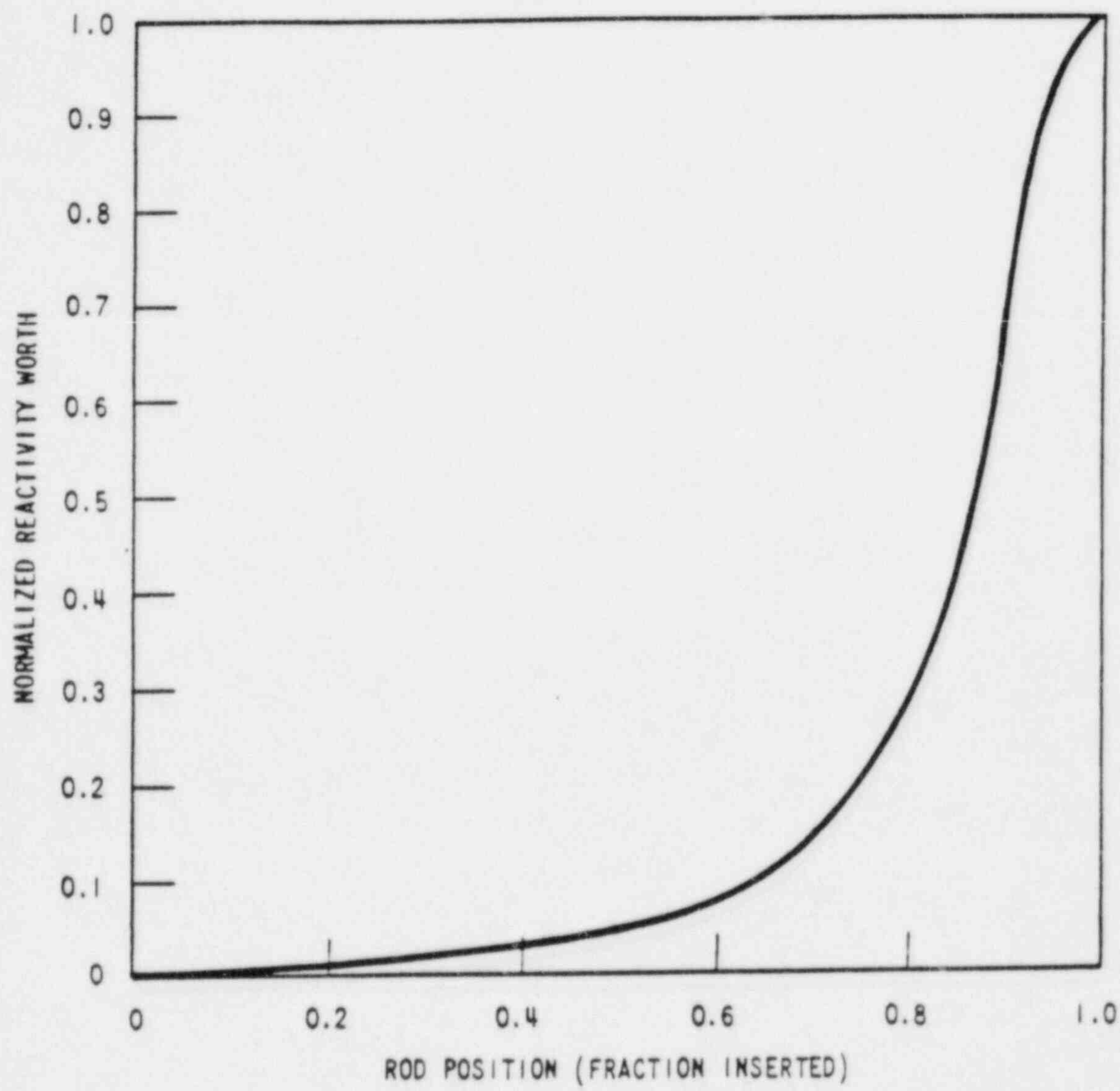


FIGURE 15.0-4 NORMALIZED RCCA REACTIVITY WORTH VS. FRACTION INSERTION

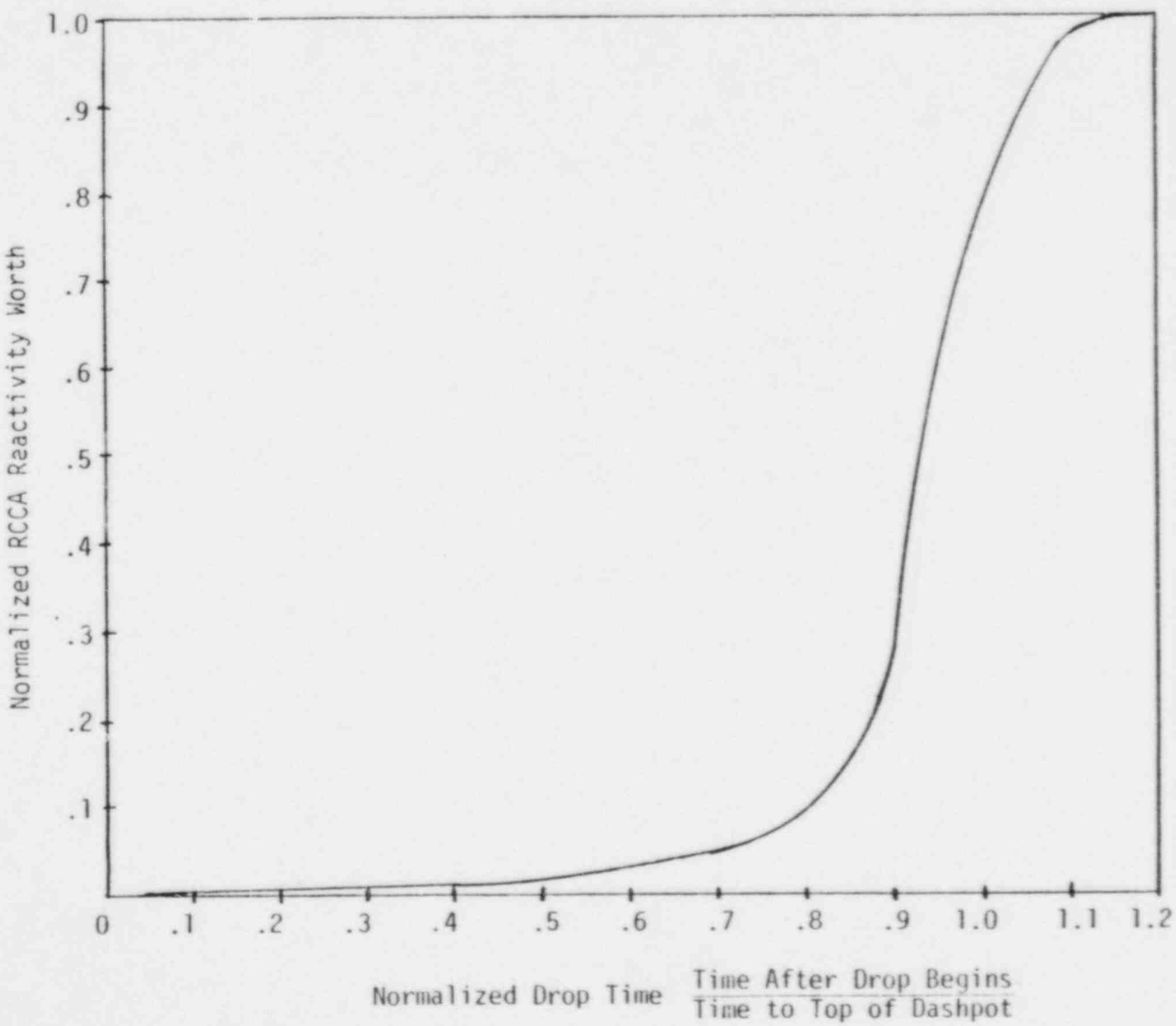


FIGURE 15.0-5 NORMALIZED RCCA BANK REACTIVITY WORTH VS. NORMALIZED DROP TIME

15.4 REACTIVITY AND POWER DISTRIBUTION ANOMALIES

A number of faults have been postulated which could result in reactivity and power distribution anomalies. Reactivity changes could be caused by control rod motion or ejection, boron concentration changes, or addition of cold water to the reactor coolant system (RCS). Power distribution changes could be caused by control rod motion, misalignment, or ejection; or by static means such as fuel assembly mislocation. These events are discussed in this section. Detailed analyses are presented for the most limiting of these events.

Discussions of the following accidents pertinent to the reactor system are presented in this section:

A. Uncontrolled rod cluster control assembly bank withdrawal from a subcritical or low-power startup condition.

B. Uncontrolled rod cluster control assembly (RCCA) bank withdrawal at power.

C. RCCA misalignment. (See RESAR-SP/90 PDA Module 9, "I&C and Electric Power" for a discussion of dropped RCCAs and dropped RCCA banks).

D. Inadvertent loading and operation of a fuel assembly in an improper position.

E. Spectrum of RCCA ejection accidents.

Items A and B above are considered to be American Nuclear Society (ANS) Condition II events, item D an ANS Condition III event, and item E an ANS Condition IV event. Item C entails both Conditions II and III events.

All of the accidents in this section have been analyzed. It has been determined that the most severe radiological consequences result from the complete rupture of a control rod drive mechanism housing provided in Subsection 15.4.8. Therefore, radiological consequences are reported only for that limiting case.

15.4.1 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal From a Subcritical Or Low-Power Startup Condition

15.4.1.1 Identification of Causes and Accident Description

An RCCA withdrawal accident is defined as an uncontrolled addition of reactivity to the reactor core caused by withdrawal of RCCAs, resulting in a power excursion. Such a transient could be caused by a malfunction of the reactor control or rod control systems. This could occur with the reactor subcritical, at hot zero power or at power. The "at power" case is discussed in Subsection 15.4.2.

Although the reactor is normally brought to power from a subcritical condition by means of RCCA withdrawal, initial startup procedures with a clean core call for boron dilution. The maximum rate of reactivity increase in the case of boron dilution is less than that assumed in this analysis (See Subsection 15.4.6 of RESAR-SP/90 PDA Module 13 "Auxiliary Systems").

The RCCA drive mechanisms are wired into preselected bank configurations which are not expected to be 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 (a fault of moderate frequency) as defined in Subsection 15.0.2.

The neutron flux response to a continuous reactivity insertion is characterized by a fast rise terminated by the reactivity feedback effect of the negative Doppler coefficient. This self-limitation of the power excursion is of primary importance, since it limits the power to a tolerable level during the delay time for protective action. Should a continuous RCCA withdrawal accident occur, the transient will be terminated by the following automatic features of the reactor protection system:

A. Source Range High Neutron Flux Reactor Trip

This trip function is actuated when two out of the four independent source range channels indicate a neutron flux level above a preselected manually adjustable setpoint. Each channel may be manually bypassed only after the corresponding intermediate range flux channel indicates a flux level above a specified level. It is automatically reinstated when the corresponding intermediate range channel indicates a flux level below a specified level.

B. Intermediate Range High Neutron Flux Reactor Trip

This trip function is actuated when two out of the four independent intermediate range channels indicate a flux level above a preselected manually adjustable setpoint. Each channel may be manually bypassed only after the corresponding power range channel is reading above approximately 10 percent of full power, and it is automatically reinstated when that channel indicates a power level below this value.

C. Power Range High Neutron Flux Reactor Trip (Low Setting)

This trip function is actuated when two out of the four power range channels indicate a power level above approximately 25 percent of full power. Each channel may be manually bypassed when it indicates a power level above approximately 10 percent of full power and is automatically reinstated only after it indicates a power level below this value.

D. Power Range High Neutron Flux Reactor Trip (High Setting)

The trip function is actuated when two out of the four power range channels indicate a power level above a preset setpoint. It is always active.

E. High Nuclear Flux Rate Reactor Trip

This trip function is actuated when the positive rate of change of neutron flux on two out of four nuclear power range channels indicates a rate above the preset setpoint. It 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

15.4.1.2.1 Method of Analysis

The analysis of the uncontrolled RCCA bank withdrawal from a subcritical accident is performed in three stages: first an average core nuclear power transient calculation, then an average core heat transfer calculation, and finally the departure from nucleate boiling ratio (DNBR) calculation. The average core nuclear calculation is performed using spatial neutron kinetics methods, TWINKLE[4], to determine the average power generation with time, including the various total core feedback effects; i.e., Doppler reactivity and moderator reactivity. In the second stage, the average heat flux and temperature transients are determined by performing a fuel rod transient heat transfer calculation in FACTRAN[6]. In the final stage, the average heat flux is next used in THINC (described in Section 4.4) for the transient DNBR calculation.

Plant characteristics and initial conditions are discussed in Subsection 15.0.4. To give conservative results for a startup accident, the following assumptions are made:

A. Since the magnitude of the power peak reached during the initial part of the transient for any given rate of reactivity insertion is strongly dependent on the Doppler coefficient, conservatively low values as a function of power are used. See Subsection 15.0.4 and Table 15.0-3.

B. Contribution of the moderator reactivity coefficient is negligible during the initial part of the transient because the heat transfer time between the fuel and the moderator is much longer than the neutron flux response time. However, after the initial neutron flux peak, the succeeding rate of power increase is affected by the moderator reactivity coefficient. A conservative value is used in the analysis to yield the maximum peak heat flux.

C. 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-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 (K_{eff}) is assumed to be 1.0, since this results in the worst nuclear power transient.

D. Reactor trip is assumed to be initiated by power range high neutron flux (low setting). The most adverse combination of instrument and setpoint errors, as well as delays from trip signal actuation and RCCA release, is taken into account. A 10-percent error increase is assumed for the power range flux trip setpoint 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 setpoint on the actual time at which the rods are released is negligible. In addition, the reactor trip insertion characteristic is based on the assumption that the highest worth RCCA is stuck in its fully withdrawn position. See Subsection 15.0.6 for RCCA insertion characteristics.

E. The maximum positive reactivity insertion rate assumed is greater than that for the simultaneous withdrawal of the combination of the two sequential control banks having the greatest combined worth at maximum speed (45 in./min). Control rod drive mechanism design is discussed in Section 4.6.

F. The most limiting axial and radial power shapes, associated with having the two highest combined worth banks in their high-worth position, are assumed in the departure from nucleate boiling (DNB) analysis.

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

H. Two reactor coolant pumps are assumed to be in operation. This is conservative with respect to DNB.

Plant systems and equipment which are available to mitigate the effects of the accident are discussed in Subsection 15.0.9 and listed in Table 15.0-6. No single active failure in any of these systems or components will adversely affect the consequences of the accident.

15.4.1.2.2 Results

Figures 15.4-1 through 15.4-3 show the transient behavior for the uncontrolled RCCA bank withdrawal incident, with the accident terminated by reactor trip at 35 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 average neutron flux transient.

The energy release and the fuel temperature increases are relatively small. The thermal flux response, of interest for DNB considerations, is shown on Figure 15.4-2. The beneficial effect of the inherent thermal lag in the fuel is evidenced by a peak heat flux much less than the full-power nominal value. There is a large margin to DNB during the transient since the rod surface heat flux remains below the design value; there is a high-degree of subcooling at all times in the core. Figure 15.4-3 shows the response of the average fuel and cladding temperature. The average fuel temperature increases to a value lower than the nominal full-power value. The minimum DNBR at all times remains above the limiting value.

The calculated sequence of events for this accident is shown on Table 15.4-1. With the reactor tripped, the plant may subsequently be cooled down further by following normal plant shutdown procedures.

15.4.1.3 Conclusions

In the event of an 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 results in a DNBR greater than the limiting value. The DNBR design basis is described in Section 4.4; applicable acceptance criteria have been met.

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 setpoint, there is a net increase in the reactor coolant temperature. Unless terminated by manual or automatic action, the power mismatch and resultant coolant temperature rise could eventually result in DNB. Therefore, in order to avert damage to the fuel cladding the reactor protection system is designed to terminate any such transient before the DNBR falls below the limit value.

This event is classified as an ANS Condition II incident (a fault of moderate frequency) as defined in Subsection 15.0.2 of this module.

The automatic features of the reactor protection system 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 setpoint.

2. Reactor trip is actuated if any two out of four N-16 power signals exceed a low DNBR setpoint. This setpoint is automatically varied with axial power shape, $F_{\Delta H}$, coolant temperature and pressure to protect against DNB.

3. Reactor trip is actuated if any two out of four high kw/ft channels exceed high kw/ft setpoint. The kw/ft calculation is automatically varied with axial power shape and F_{xy} to ensure that the allowable heat generation rate (kw/ft) is not exceeded.

4. A high pressurizer pressure reactor trip actuated from any two-out-of-four 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-four level channels when the reactor power is above approximately 10% (Permissive-7).

6. Intermediate range neutron flux instrumentation actuates a reactor trip; if one-out-of-two channels exceed an overpower setpoint.

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

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

2. Low DNBR (two-out-of-four).

3. High kw/ft (two-out-of-four).

The manner in which the low DNBR trip provides protection over the full range of reactor coolant system conditions is described in RESAR-SP/90 PDA Module 9, "I&C and Electric Power". Figure 15.0-1 presents allowable reactor power as a function of coolant loop inlet temperature and power as a function of primary coolant pressure for the design flow and power distribution as described in Section 4.4. The boundaries of operation defined by the low DNBR trip are represented as "protection lines" on this diagram. The protection lines are drawn to include all adverse instrumentation and setpoint errors so that under nominal conditions trip would occur well within the area bounded by these lines. The DNB lines represent the locus of conditions for which the DNBR equals the limit value. All points below and to the left of a DNB line for a given pressure have a DNBR greater than the limit value with the assumed axial and radial power distributions. 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 setpoint); hign pressure (fixed setpoint); low pressure (fixed setpoint); low DNBR (variable setpoint); high kw/ft (fixed setpoint).

15.4.2.2 Analysis of Effects and Consequences

15.4.2.2.1 Method of Analysis

This transient is analyzed by the LOFTRAN code (Reference 9). This code simulates the neutron kinetics, reactor coolant system, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level. The core limits as illustrated in Figure 15.0-1 are used as input to LOFTRAN to determine the minimum DNBR during the transient.

This accident is analyzed with the Improved Thermal Design Procedure as described in WCAP-8567. Initial operating conditions are assumed at values consistent with steady-state operation. Plant characteristics and initial conditions are discussed in Subsection 15.0.4 of this module.

In order to obtain conservative results for an uncontrolled rod withdrawal at power accident, the following assumptions are made:

1. Initial reactor power, pressure, and RCS temperatures are assumed to be at their nominal values. Uncertainties in initial conditions are included in the limit DNBR as described in WCAP-8567.

2. Reactivity Coefficients - two cases are analyzed:

   a. Minimum Reactivity Feedback. A least negative moderator temperature coefficient of reactivity 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 or 118% of nominal full power. The low DNBR trip includes all adverse instrumentation and setpoint 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 combined with boron dilution occurring at the maximum dilution rate.

6. Cases are analyzed for operation with four loops in service.

The effect of RCCA movement on the axial core power distribution is accounted for by the axial power shape measurement as described in RESAR-SP/90 PDA Module 9, "I&C and Electric Power".

Plant systems and equipment which are available to mitigate the effects of the accident are discussed in Subsection 15.0.9, and listed in Table 15.0-6. No single active failure in any of these systems or equipment will adversely offset the consequences of the accident. A discussion of ATWT considerations is presented in Reference 10.

15.4.2.2.2 Results

Figures 15.4-4 through 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 the plant, 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 in Figures 15.4-7 through 15.4-9. Reactor trip on low DNBR 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 low DNBR trip channels. The minimum DNBR is never less than the limit value.

Figure 15.4-11 shows the minimum DNBR as a function of reactivity insertion rate for RCCA withdrawal incidents starting at 60% power. The results are similar to the 100 percent power case, except as the initial power is decreased, the range over which the low DNBR trip is effective is increased. In neither case does the DNBR 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 Figure 15.4-10, for example, it is noted that:

1. For high reactivity insertion rates (i.e., above approximately 2.0×10^{-4} ΔK/sec) with maximum reactivity feedback, 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 a significant increase in heat flux or water temperature. As the reactivity insertion rate decreases, core heat flux and coolant temperatures can remain more nearly in equilibrium with the neutron flux.

2. The low DNBR reactor trip initiates a trip when measured reactor coolant system parameters approach the DNB lines shown in Figure 15.0-1. As the reactivity insertion rate decreases below ~ 2×10^{-4} ΔK/sec, the rise in the reactor coolant temperature results in a trip initiated by the low DNBR reactor trip.

Since the RCCA withdrawal at power incident is an overpower transient, the fuel temperatures rise during the transient until after reactor trip occurs. For high reactivity insertion rates, the overpower transient is fast with respect to the fuel rod thermal time constant, and the core heat flux lags behind the neutron flux response. Due to this lag, the peak core heat flux does not exceed 118% of its nominal value (i.e., the high neutron flux trip 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 low DNBR reactor trip before a DNB condition is reached. The peak heat flux again is maintained below 118% of its nominal value. Taking into account the effect of the RCCA withdrawal on the axial core power distribution, the peak fuel temperature will remain below the fuel melting temperature.

Since DNB does not occur at any time during the RCCA withdrawal at power transient, the ability of the primary coolant to remove heat from the fuel rod is not reduced. Thus, the fuel cladding temperature does not rise significantly above its initial value during the transient.

The calculated sequence of events for this accident is shown on Table 15.4-1. With the reactor tripped, the plant eventually returns to a stable condition. The plant may subsequently be cooled down further by following normal plant shutdown procedures.

The operating procedures would call for operator action to control RCS boron concentration and pressurizer level using the CVCS, and to maintain steam generator level through control of the main or auxiliary feedwater system. Any action required of the operator to maintain the plant in a stabilized condition will be in a time frame in excess of ten minutes following reactor trip.

15.4.2.3 Radiological Consequences

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 would be less severe than the steamline break accident analyzed in Subsection 15.1.5 of RESAR-SP/90 PDA Module 6, "Secondary Side Safeguards System".

15.4.2.4 Conclusions

The high neutron flux and low DNBR trip channels provide adequate protection over the entire range of possible reactivity insertion rates, i.e., the minimum value of DNBR is always larger than the limit value. Thus, the DNB design-basis as described in Section 4.4 is met. The radiological consequences would be less severe than the steamline break accident analyzed in Subsection 15.1.5 of RESAR-SP/90 PDA Module 6, "Secondary Side Safeguards System".

15.4.3 Rod Cluster Control Assembly Misoperation (System Malfunction or Operator Error)

15.4.3.1 Identification of Causes and Accident Description

Rod cluster control assembly (RCCA) misoperation accidents include:

1. A dropped RCCA (see RESAR-SP/90 PDA Module 9, "I&C and Electric Power" for analysis results),
2. A dropped RCCA bank (see RESAR-SP/90 PDA Module 9, "I&C and Electric Power" for analysis results),
3. Statically misaligned RCCA (see the following discussion and Table 15.4-2 for analysis results), and
4. Withdrawal of a single RCCA (see the following discussion for analysis results).

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. Group demand position is also indicated.

Full length RCCA's are always moved in preselected banks, and the banks are always moved in the same preselected sequence. Each bank of RCCA's is divided into two groups. The rods comprising a group operate in parallel through multiplexing thyristors. The two groups in a bank move sequentially such that the first group is always within one step of the second group in the bank. A definite schedule of actuation (or deactuation of the stationary gripper, movable gripper, an lift coils of a mechanism) is required to withdraw the RCCA attached to the mechanism. Since the stationary gripper, movable gripper, and lift coils associated with the RCCA's of a rod group are driven in parallel, any single failure which would cause rod withdrawal would affect a minimum of one group. Mechanical failures are in the direction of insertion, or immobility.

The dropped assembly, dropped assembly bank, and statically misaligned assembly events are classified as ANS Condition II incidents (incidents of moderator frequency) as defined in Subsection 15.0.2 of this module. However, the single RCCA withdrawal incident is classified as an ANS Condition III event, as discussed below.

No single electrical or mechanical failure in the rod control system could cause the accidental withdrawal of a single RCCA from the inserted bank at full power operation. A single RCCA in the control bank could be withdrawn 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), or multiple serious operator errors and subsequent and repeated operator disregard of event indication. The probability of such a combination of conditions is low. The limiting consequences may include slight fuel damage. Since this is 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 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>." (Emphasis has been added). It has been shown that single failures resulting in RCCA bank withdrawals do not violate specified fuel design limits. Moreover, no single malfunction can result in the withdrawal of a single RCCA. Thus, it is concluded that criterion established for the single rod withdrawal at power is appropriate and in accordance with General Design Criterion 25.

Misaligned RCCA's are detected by:

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

2. Rod deviation alarm, or

3. Rod position indicators.

The resolution of the rod position indicator system 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 deviations with respect to group demand 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.

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 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 low DNBR 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 Subsection 15.0.9 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.

15.4.3.2 Analysis of Effects and Consequences

A statically misaligned RCCA, and a single RCCA withdrawal are analyzed in the following paragraphs.

15.4.3.2.1 Method of Analysis for Dropped or 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 by the THINC code to calculate the DNBR.

15.4.3.2.2 Statically Misaligned RCCA Results

The most severe misalignment situations with respect to DNBR at significant power levels arise from cases in which 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 shown in Table 15.4-2, 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 was analyzed assuming the initial reactor power, pressure, and RCS temperatures are at their nominal values, but with the increased radial peaking factor associated with the misaligned RCCA. Uncertainties in initial conditions are included in the limit DNBR as described in WCAP-8567.

DNB calculations have not been performed specifically for assemblies missing from other banks; however, power shape calculations have been done as required for the RCCA ejection analysis. Inspection of the power shapes shows that the DNB and peak kw/ft situation is less severe than the bank D case discussed above assuming insertion limits on the other banks equivalent to a bank D full-in insertion limit.

For the RCCA misalignments shown in Table 15.4-2 with one RCCA fully inserted, the DNBR does not fall below the limit value. This case was analyzed assuming the initial reactor power, pressure, and RCS temperatures are at their nominal values, but with the increased radial peaking factor associated with the misaligned RCCA. Uncertainties in the initial conditions are included in the limit DNBR as described in WCAP-8567.

DNB does not occur for the RCCA misalignment incident and thus the ability of the primary coolant to remove heat from the fuel rod is not reduced. The peak fuel temperature corresponds to a linear heat generation rate based on the radial peaking factor penalty associated with the misaligned RCCA (as noted in Table 15.4-2) and the design axial power distribution. The resulting linear heat generation is well below that which would cause fuel melting.

Following the identification of a RCCA group misalignment condition the operator realigns the RCCA group following approved procedures.

15.4.3.2.3 Single RCCA Withdrawal Method of Analysis

Power distributions within the core are calculated using the computer codes as described in Table 4.1-2. The peaking factors are then used by THINC to calculate the minimum 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.

15.4.3.2.4 Single RCCA Withdrawal Results

For the single rod withdrawal event, two cases have been considered as follows:

1. If the reactor is in the manual control mode, continuous withdrawal of a single RCCA results in both an increase in core power and coolant temperature, and an increase in the local hot channel factor in the area of the withdrawing RCCA. In terms of the overall system response, this case is similar to those presented in Subsection 15.4.2; however, the increased local power peaking in the area of the withdrawn RCCA results in lower minimum DNBR's than for the withdrawn bank cases. Depending on initial bank insertion and location of the withdrawn RCCA, automatic reactor trip may not occur sufficiently fast to prevent the minimum core DNB from falling below the limit value. Evaluation of this case at the power and coolant conditions at which the low DNBR 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.

2. If the reactor is in the automatic control mode, the multiple failures that result in the withdrawal of a single RCCA will result in the immobility of the other RCCA's in the controlling bank. The transient will then proceed in the same manner as Case 1 described above.

For such cases as above, a reactor trip will ultimately ensue, although not sufficiently fast in all cases to prevent a minimum DNBR in the core of less than the limit value. Following reactor trip, normal shutdown procedures are followed.

15.4.3.3 Radiological Consequences

The most limiting rod cluster control assembly misoperation, accidental withdrawal of a single RCCA, is predicted to result in less than 1% fuel damage. The subsequent reactor and turbine trip would result in atmospheric steam dump, assuming the condenser was not available for use. The radiological consequences from this event would be no greater than the main steamline break event, analyzed in Subsection 15.1.5 of RESAR-SP/90 PDA Module 6, "Secondary Side Safeguards System".

15.4.3.4 Conclusions

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. Thus, the DNB design basis as described in Section 4.4 is met.

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. The radiological consequences from these events would be no greater than the main steamline break accident analyzed in Subsection 15.1.5 of RESAR-SP/90 PDA Module 6, "Secondary Side Safeguards System".

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 such as can arise from the inadvertent loading of one or more fuel assemblies into improper positions, loading a fuel rod during manufacture with one or more pellets of the wrong enrichment or the loading of a full fuel assembly during manufacture with pellets of the wrong enrichment 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 integral fuel burnable absorber (IFBA) rods into a new core without IFBA's.

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 startup subsequent to every refueling operation.

This event is classified as an ANS Condition III incident (an infrequent incident) as defined in Subsection 15.0.2.

15.4.7.2 Analysis of Effects and Consequences

15.4.7.2.1 Method of Analysis

Steady state power distribution in the x-y plane of the core are 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 4.0 based on enrichments given in that section.

As an example of the impact of the core loading error on power distribution, several cases were examined assuming a loading error was present in a typical standard 4-loop first core design. While not performed on the WAPWR first core design specifically, these cases serve to illustrate a power distribution deviation associated with loading errors would be readily observable by the incore movable detector system.

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) are shown for all incore detector locations (see Figures 15.4-12 through 15.4-16, inclusive).

15.4.7.2.2 Results

The following core loading error cases have been analyzed:

Case A:

This is 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 (see Figure 15.4-12).

Case B:

This is a case in which a Region 1 assembly is interchanged with a neighboring Region 2 fuel assembly. Two analyses have been performed for this case (see Figures 15.4-13 and 15.4-14).

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 in a Region 1 assembly mistakenly loaded in the Region 2 position.

Case C:

Enrichment error: This is a case in which a Region 2 fuel assembly is loaded in the core central position (see Figure 15.4-15).

Case D:

This is a case in which a Region 2 fuel assembly instead of a Region 1 assembly is loaded near the core periphery (see Figure 15.4-16).

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 since activity is contained within the fuel rods and reactor coolant system, within design limits.

15.4.7.4 Conclusions

Fuel assembly enrichment errors are 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 confirms 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 a rod cluster control assembly (RCCA) and drive shaft. The consequence of this mechanical failure is a rapid positive reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage.

15.4.8.1.1 Design Precautions and Protection

Certain features in the WAPWR are intended to preclude the possibility of a rod ejection accident or to limit the consequences if the accident occurs. These include a sound, conservative mechanical design of the rod housings, together with a thorough quality control (testing) program during assembly, and a nuclear design which lessens the potential ejection worth of RCCAs and minimizes the number of assemblies inserted at high power levels.

15.4.8.1.1.1 Mechanical Design

The mechanical design is discussed in Subsection 3.9.4 and Section 4.6. Mechanical design and quality control procedures intended to preclude the possibility of an RCCA drive mechanism housing failure are listed below:

A. Each control rod drive mechanism housing is completely assembled and shop tested at 4100 psi.

B. The mechanism housings are individually hydrotested after they are attached to the head adapters in the reactor vessel head and checked during the hydrotest of the completed reactor coolant system (RCS).

C. Stress levels in the mechanism are not affected by anticipated system transients at power or by the thermal movement of the coolant loops.

Moments induced by the design earthquake can be accepted within the allowable primary working stress range specified by the American Society of Mechanical Engineers Code, Section III, for Class 1 components.

D. The latch mechanism housing and rod travel housing are each a single length of forged type 304 stainless steel. 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 which are subject to periodic inspections.

15.4.8.1.1.2 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 RCCA inserted only far enough to permit load follow. Reactivity changes caused by core depletion and xenon transients are compensated for 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 require boration at low level alarm and emergency boration at the low-low alarm.

15.4.8.1.1.3 Reactor Protection

The reactor protection in the event of a rod ejection accident has been described in reference 1. 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 in RESAR-SP/90 PDA Module 9, "I&C and Electric Power".

15.4.8.1.1.4 Effects on Adjacent Housings

Disregarding the remote possibility of the occurrence of an 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 control rod drive mechanism is described in Subsection 3.9.4.

15.4.8.1.1.5 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 2250 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.

In the WAPWR, the design of the Integrated Head Package (IHP) is such that all of the RCCA pressure housings, including those on the periphery, are supported. Two horizontal support points are provided: one at the top of the latch housing (i.e., the intermediate seismic support) and near the top of the rod travel housing (i.e., upper seismic support). Thus the adjacent assembly failure mechanism associated with the longitudinal failure of an RCCA cannot occur in the WAPWR design.

15.4.8.1.1.6 Effects of Rod Travel Housing Circumferential Failures

If circumferential failure of the 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 (sufficient to cause failure of an adjacent housing).

15.4.8.1.1.7 Possible Consequences

From the above 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 sufficient to prevent that RCCA from falling upon receipt of a trip signal is not expected.

15.4.8.1.1.8 Summary

The considerations given above 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 American Nuclear Society (ANS) Condition IV incident. See Subsection 15.0.2 for a discussion of ANS classifications. Due to the extremely low probability of an RCCA ejection accident, some fuel damage would 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(2). Extensive tests of UO_2 zirconium-clad fuel rods representative of those in pressurized water reactor type cores such as WAPWR have demonstrated failure thresholds in the range of 240 to 257 cal/g. However, other rods of a slightly different design have exhibited failure as low as 225 cal/g. These results differ significantly from the TREAT(3) results, which indicated a failure threshold of 280 cal/g. 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/g for unirradiated rods and 200 cal/g for irradiated rods; catastrophic failure (large fuel dispersal, large pressure rise), even for irradiated rods, did not occur below 300 cal/g.

In view of the above experimental results and conformance with Regulatory Guide 1.77, criteria are applied to ensure that there is little or no possibility of fuel dispersal in the coolant, gross lattice distortion, or severe shock waves. These criteria are:

A. Average fuel pellet enthalpy at the hot spot will be below 225 cal/g for unirradiated fuel and 200 cal/g for irradiated fuel.

B. Average clad temperature at the hot spot will be below the temperature at which clad embrittlement may be expected (2700°F).

C. Peak reactor coolant pressure will be less than that which could cause stresses to exceed the faulted condition stress limits.

D. 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 A, above.

15.4.8.2 Analysis of Effects and Consequences

A. Method of Analysis

The calculation of the RCCA ejection transient is performed in two stages, first an average core channel calculation and then a hot region calculation. The average core calculation is performed using spatial neutron kinetics methods to determine the average power generation with time including the various total core feedback effects, i.e., Doppler reactivity and moderator reactivity. Enthalpy and temperature transients at 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 conservatively assumed to persist throughout the transient.

A detailed discussion of the method of analysis can be found in reference 1.

B. Average Core Analysis

The spatial kinetics computer code, TWINKLE[4], is used for the average core transient analysis. This code uses cross sections generated by LEOPARD[5] to solve the two-group neutron diffusion theory kinetic equation in one, two, or three spatial dimensions (rectangular coordinates) for six delayed neutron groups and up to 2000 spatial points. The computer code includes a detailed multiregion, transient fuel clad coolant heat transfer model for calculation of pointwise Doppler and moderator feedback effects. In this analysis, the code is used as a one-dimensional axial kinetics code since it allows a more realistic representation of the spatial effects of axial moderator feedback and RCCA movement. However, since the radial dimension is missing, it is still necessary to employ conservative methods (described below) of calculating the ejected rod worth and hot channel factor. Further description of TWINKLE appears in Subsection 15.0.12.

C. Hot Spot Analysis

In the hot spot analysis the initial heat flux is equal to the nominal value multiplied by the design hot channel factor. During the transient, the heat flux hot channel factor is linearly increased to the transient value in 0.1 seconds, the time for full ejection of the rod. Therefore, the assumption is made that the hot spots before and after ejection are coincident. This is conservative, since the peak after ejection will occur in, or adjacent to, the assembly with the ejected rod. Prior to ejection the power in this region will necessarily be depressed.

The hot spot analysis is performed using the detailed fuel and clad transient heat transfer computer code, FACTRAN(6). 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 parabolic 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 departure from nucleate boiling (DNB), and the Bishop-Sandburg-Tong correlation(7) to determine the film boiling coefficient after DNB. The Bishop-Sandburg-Tong correlation is conservatively used, assuming zero bulk fluid quality. The departure from nucleate boiling ratio is not calculated; instead the code is forced into DNB by specifying a conservative DNB heat flux. The gap heat transfer coefficient can be calculated by the code; however, it is adjusted in order to force the full power, steady state temperature distribution to agree with the fuel heat transfer design codes. Further description of FACTRAN appears in Subsection 15.0.12.

D. 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 absorption by the coolant.

The pressure surge is calculated by first performing the fuel heat transfer calculation to determine the average and hot spot heat flux versus time. Using this heat flux data, a THINC (Section 4.4) calculation is conducted to determine the volume surge. Finally, the volume surge is simulated in a plant transient computer code. This code calculates the pressure transient taking into account fluid transport in the 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 below. Table 15.4-3 presents the parameters used in this analysis.

15.4.8.2.1.1 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. Standard nuclear design codes are used in the analysis. No credit is taken for the flux flattening effects of reactivity feedback. The calculation is performed for the maximum allowed bank insertion at a given power level, as determined by the rod insertion limits. Adverse xenon distributions are considered in the calculation.

Appropriate margins are added to the ejected rod worth and hot channel factors to account for any calculational uncertainties, including an allowance for nuclear peaking due to densification.

Power distributions before and after ejection for a worst case can be found in reference 1. During plant startup physics testing, ejected rod worths and power distributions are measured in the zero and full power configurations and compared to values used in the analysis. It has been found that the ejected rod worth and power peaking factors are consistently overpredicted in the analysis.

15.4.8.2.1.2 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[1].

15.4.8.2.1.3 Moderator and Doppler Coefficient

The critical boron concentrations at the beginning of life and end of life are adjusted in the nuclear code in order to obtain moderator density coefficient curves which are conservative compared to actual design conditions for the plant. As discussed above, no weighting factor is applied to these results.

The Doppler reactivity defect is determined as 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 Subsection 15.0.5. The Doppler weighting factor will increase under accident conditions, as discussed above.

15.4.8.2.1.4 Delayed Neutron Fraction, β_{eff}

Calculations of the effective delayed neutron fractron (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 β_{eff} if the ejected rod worth is equal to or greater than 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.

15.4 8.2.1.5 Trip Reactivity Insertion.

The trip reactivity insertion assumed is given in Table 15.4-3. These values are reduced by the ejected rod reactivity. The shutdown reactivity was simulated by dropping a rod of the required worth into the core. The start of rod motion occurred 0.5 seconds after the high neutron flux trip point was reached. This delay is assumed to consist of 0.2 seconds for the instrument channel to produce a signal, 0.15 seconds for the trip breaker to open, and 0.15 seconds for the coil to release the rods. A curve of trip rod insertion versus time was used which assumed that insertion to the dashpot does not occur until 3.4 seconds after the start of fall. The choice of such a conservative insertion rate means that there is over 1 second after the trip point is reached before significant shutdown reactivity is inserted into the core. This is particularly important conservatism for hot full power (HFP) accidents.

The minimum design shutdown margin available for this plant at hot zero power (HZP) may be reached only at the end of the voided portion of the cycle, in the equilibrium cycle. This value includes adverse xenon distribution, conservative Doppler and moderator defects, and an allowance for calculational uncertainties.

Depressurization calculations have been performed for a typical four-loop plant, assuming the maximum possible size break (2.75 in. diameter) located in the reactor pressure vessel. The results show a rapid pressure drop and a decrease in system water mass due to the break. The safety injection system is actuated on low pressurizer pressure within 1 minute after the break. The RCS pressure continues to drop and reaches saturation (1200 psi) in about 2 to 3 minutes. Due to the large thermal inertia of primary and secondary systems, 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 Δk due to the pressure coefficient. The cooldown transient could not absorb the available shutdown margin until more than 10 minutes after the break. The addition of borated (2000 ppm) safety injection flow starting 1 minute after the break is much more than sufficient to ensure that the core remains subcritical during the cooldown.

15.4.8.2.1.6 Reactor Protection.

As discussed in Subsection 15.4.8.1.1.3, 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. No single failure of the reactor trip system will negate the protection functions required for the rod ejection accident or adversely affect the consequences of the accident.

15.4.8.2.1.7 Results.

Cases are presented for both beginning and end of life voided and flooded portions of cycle at zero and full power.

A. Beginning of Cycle, Voided, Full Power

Control bank D was assumed to be inserted to its insertion limit. The water displacer rods and gray rods were assumed to be inserted. The worst ejected rod worth and hot channel factor were conservatively calculated to be 0.20-percent Δk and 7.10, respectively. The peak hot spot clad average temperature was 2234°F. The peak hot spot fuel center temperature reached melting at 4900°F. However, melting was restricted to less than 10 percent of the pellet.

B. Beginning of Cycle, Voided, 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 water displacer rods and gray rods were assumed to be inserted. The worst ejected rod is located in control bank D and has a worth of 0.86 percent Δk and a hot channel factor of 13.0. The peak hot spot clad temperature reached 2119°F; the fuel center temperature was 3131.°F.

C. End of Cycle, Voided, Full Power

Control bank D was assumed to be inserted to its insertion limit. This case represents middle of cycle core conditions with water displacer rods and gray rods inserted and just prior to WDR withdrawal. The ejected rod worth and hot channel factors were conservatively calculated to be 0.20 percent Δk and 7.1, respectively. This resulted in a peak clad temperature of 2078°F. The peak hot spot fuel center temperature reached melting at 4800°F. However, melting was restricted to less than 10 percent of the pellet.

D. End of Cycle, Voided, 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 C and B at their insertion limits. This case represents middle of cycle conditions with the water displacer rods and gray rods inserted and just prior, to WDR withdrawal. The results were 1.04 percent ΔK and 24.0 respectively. The peak clad and fuel center temperatures were 2048°F and 3239°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.

E. Beginning of Cycle, Flooded, Full Power

Control bank D was assumed to be inserted to its insertion limit. Middle of life core conditions were assumed along with a 1-step withdrawal of all water displacer rods and gray rods. The worst ejected rod worth and hot channel factor were conservatively calculated to be 0.20 percent Δk and 7.10, respectively. The peak hot spot clad average temperature was 2065°F. The peak hot spot fuel center temperature reached melting at 4900°F. However, melting was restricted to less than 10% of the pellet.

F. Beginning of Cycle, Flooded, Zero Power

For this condition, control bank D was assumed to be fully inserted and banks B and C were at their insertion limits. Middle of life core conditions were assumed along with a 1-step withdrawal of all water displacer rods and gray rods. The worst ejected rod worth and hot channel factor were conservatively calculated to be 0.86 percent Δk and a hot channel factor of 18.54. The peak hot spot clad temperature reached was 2223°F; the fuel center temperature was 3401°F.

G. End of Cycle, Flooded, Full Power

Control bank D was assumed to be inserted to its insertion limit. End of life core conditions were assumed with the water displacer rods and gray rods withdrawn. The ejected rod worth and hot channel factor were conservatively calculated to be 0.20 percent Δk and 7.1, respectively. This resulted in a peak clad temperature of 2101°F. The peak hot spot fuel center temperature reached melting at 4800°F. However, melting was restricted to less than 10 percent of the pellet.

H. End of Cycle, Flooded, 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 C and B at their insertion limits. End of life core conditions were assumed with the water displacer rods and gray rods withdrawn. The ejected rod worth and hot channel factor were conservatively calculated to be 0.90 percent Δk and 24.0, respectively. The peak clad and fuel center temperatures were 2499°F and 3682°F. The Doppler weighting factor for this case is significantly higher than for the other cases due to the very large transient hot channel factor.

A summary of the cases presented above is given in Table 15.4-3. The nuclear power and hot spot fuel and clad temperature transients for the worst cases (beginning of life, voided, full power; and end of life, flooded, zero power) are presented in Figures 15.4-17 through 15.4-20.

The calculated sequence of events for the worst case rod ejection accidents, as shown in Figures 15.4-17 through 15.4-20, is presented in Table 15.4-1. For all cases, reactor trip occurs early in the transient, after which the nuclear power excursion is terminated. As discussed previously in Subsection 15.4.8.2.1, the reactor remains 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 Subsection 15.6.5 of RESAR-SP/90 PDA Module 1, "Primary Side Safeguards System". Following the RCCA ejection, the operator would follow the same emergency instructions as for any other LOCA to recover from the event.

15.4.8.2.1.8 Fission Product Release

It is assumed that fission products are released from the gaps of all rods entering DNB. In all cases considered, less than 10 percent of the rods entered DNB based on a detailed three-dimensional THINC analysis(1). Although limited fuel melting at the hot spot was predicted for the full-power cases, in practice melting is not expected since the analysis conservatively assumed that the hot spots before and after ejection were coincident.

15.4.8.2.1.9 Pressure Surge

A detailed calculation of the pressure surge for an ejection worth of one dollar at beginning of life, hot full power, for a typical 12-foot core, Westinghouse PWR indicates that the peak pressure does not exceed that which would cause stress to exceed the faulted condition stress limits(1). Since the severity of the present analysis is much lower than the worst-case analysis, the accident for this plant will not result in an excessive pressure rise or further damage to the RCS.

15.4.8.2.1.10 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 undermoderated, and bowing will tend to increase the undermoderation at the hot spot. 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 crossflow 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.

A. The iodine activity in the reactor coolant prior to the accident is based upon an iodine spike which has raised the reactor coolant concentration to 60 μCi/g of dose equivalent (DE) I-131.

B. The noble gas concentrations in the reactor coolant are based upon 1-percent defective fuel.

C. Following the rod ejection accident, 10 percent of the fuel rods in the core undergo DNB. Hence, 10 percent of the core iodine and noble gas gap inventory is released to the reactor coolant. In addition, 0.25 percent of the fuel in the core is assumed to melt and release 0.00125 of the core iodines and 0.0025 of the core noble gases to the reactor coolant.

D. The secondary coolant iodine activity is based on the DE of 0.1 μCi/g of I-131.

15.4.8.3.1.2 Mathematical Models Used in the Analysis

Mathematical models used in the analysis are described in the following sections:

A. The mathematical models used to analyze the activity released during the course of the accident are described in Appendix 15A.

B. The atmospheric dispersion factors used in the analysis were calculated based on onsite meteorological measurement programs.

C. The thyroid inhalation dose and total-body gamma immersion doses to a receptor at the exclusion area boundary and outer boundary of the low population zone were analyzed using the models described in Appendix 15A.

15.4.8.3.1.3 Identification of Leakage Pathways and Resultant Leakage Activity

Radionuclides carried from the primary coolant to the steam generators via leaking tubes are released to the environment via the steam line safety or power-operated relief valves. Iodines are assumed to mix with the secondary coolant and partition between the generator liquid and steam before release to the environment. Noble gases are assumed to be directly released.

Forty-five percent of the iodines and one hundred percent of the noble gases carried by the primary coolant spill are released to the containment vapor space and are leaked to the environment at the containment design leak rate. For the iodine release, 39 percent of the break flow is assumed to initially flash to vapor and 10 percent of the nonflashed portion is assumed to become airborne; i.e., 0.39 plus 10 percent of 0.61 for a total of 0.45.

All activity is released to the environment with no consideration given to radioactive decay or to cloud depletion by ground deposition during transport to the exclusion area boundary and low population zone. Hence, the resultant radiological consequences represent the most conservative estimate of the potential integrated dose due to the postulated rod ejection accident.

15.4.8.3.2 Identification of Uncertainties and Conservative Elements in the Analysis

A. The initial reactor coolant iodine activity is based on the Technical Specification limit of 1.0 μCi/g of DE I-131 which is further increased by a large preaccident iodine spike to 60 μCi/g, resulting in equivalent concentrations many times greater than the reactor coolant activities based on 0.12 percent defective fuel and expected iodine spiking values associated with normal operating conditions.

B. The noble gas activities are based on 1 percent defective fuel which cannot exist simultaneously with 1.0 μCi/g I-131. For iodines, 1 percent defects would be approximately three times the Technical Specification limit.

C. The fraction of failed fuel is assumed to be equal to the fraction of fuel rods experiencing DNB, without consideration given to the extent of the zirc-water reaction. Based on experimental data[8] no oxidation related fuel rod clad failure is predicted. Likewise, the small amount of melted fuel assumed (0.25 percent) is not predicted.

D. A 1-gal/min steam generator primary-to-secondary leakage is assumed, which is significantly greater than that anticipated during normal operation.

E. The meteorological conditions which may be present at the site during the course of the accident are uncertain. However, it is highly unlikely that the assumed meteorological conditions would be present during the course of the accident for any extended period of time. Therefore, the radiological consequences evaluated, based on the meteorological conditions assumed, are conservative.

15.4.8.3.3 Conclusions

15.4.8.3.3.1 Filter Loadings

The only engineered safety feature filtration system considered in the analysis which limits the consequences of the rod ejection accident is the control room filtration system.

Integrated activity on the control room filters have been evaluated for the more limiting LOCA analysis as discussed in Subsection 15.6.5 of RESAR-SP/90 PDA Module 1, "Primary Side Safeguards Systems". Since the control room filters are capable of accommodating the potential design basis LOCA fission product iodine loadings, there will be sufficient capacity to accommodate any fission product loading due to a postulated rod ejection accident.

15.4.8.3.3.2 Dose to Receptor at the Exclusion Area Boundary and Low Population Zone Outer Boundary

The potential radiological consequences resulting from the occurrence of a postulated rod ejection accident have been conservatively analyzed for a typical 4-loop, 12-foot core Westinghouse PWR using assumptions and models described. The total-body gamma dose due to immersion from direct radiation and the thyroid dose due to inhalation have been analyzed for the 0 to 2 hours dose at the exclusion area boundary and for the duration of the accident (0 to 2.8 hours) at the low population zone outer boundary. The results are listed in Table 15.4-5. The resultant doses are well within the guideline values of 10 CFR 100.

15.4.9 REFERENCES

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7. Bishop, A. A., Sandburg, R. O., and Tong, L. S., "Forced Convection Heat Transfer at High Pressure After the Critical Heat Flux," ASME 65-HT-31, August 1965.

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9. Burnett, T. W. T., et al., "LOFTRAN Code Description," WCAP-7907-P-A, April, 1984.

10. "Westinghouse Anticipated Transients Without Reactor Trip Analysis", WCAP-8330, August, 1974.

TABLE 15.4-1 (SHEET 1 OF 4)

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH RESULT IN REACTIVITY AND POWER DISTRIBUTION ANOMALIES

Accident	Event	Time (sec)
Uncontrolled RCCA bank withdrawal from a subcritical or low-power startup condition	Initiation of uncontrolled rod withdrawal from 10^{-9} of nominal power	0.0
	Power range high neutron flux low setpoint reached	10.5
	Peak nuclear power occurs	10.6
	Rods begin to fall into core	11.0
	Minimum DNBR occurs	13.2
	Peak heat flux occurs	13.2
	Peak average clad temperature occurs	13.6
	Peak average fuel temperature occurs	13.7

TABLE 15.4-1 (SHEET 2 of 4)

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH CAUSE REACTIVITY AND POWER DISTRIBUTION ANOMALIES

Accident	Event	Time (sec)
Uncontrolled RCCA Bank Withdrawal at Power		
1. Case A	Initiation of uncontrolled RCCA withdrawal at a high reactivity insertion rate (75 pcm/sec)	0.0
	Power range high neutron flux high trip point reached	1.04
	Rods begin to fall into core	1.54
	Minimum DNBR occurs	2.6
2. Case B	Initiation of uncontrolled RCCA withdrawal at a small reactivity insertion rate (1 pcm/sec)	0
	Low DNBR reactor trip signal initiated	111.2
	Rods begin to fall into the core	113.7
	Minimum DNBR occurs	114.6

TABLE 15.4-1 (SHEET 3 OF 4)

Accident	Event	Time (sec)
RCCA ejection accident		
1. Beginning of life, full power	Initiation of rod ejection	0.0
	Power range high neutron flux setpoint reached	0.045
	Peak nuclear power occurs	0.14
	Rods begin to fall into core	0.54
	Peak fuel average temperature occurs	2.02
	Peak clad temperature occurs	2.15
	Peak heat flux occurs	2.15
2. End of life, zero power	Initiation of rod ejection	0.0
	Power range high neutron flux low setpoint reached	0.205
	Peak nuclear power occurs	0.246
	Rods begin to fall into core	0.705
	Peak clad temperature occurs	1.89

TABLE 15.4-1 (SHEET 4 OF 4)

Accident	Event	Time (sec)
	Peak heat flux occurs	1.95
	Peak fuel average temperature occurs	2.08

TABLE 15.4-2

MINIMUM CALCULATED DNBR FOR ROD CLUSTER CONTROL ASSEMBLY MISALIGNMENT

CASES ANALYZED	RADIAL POWER* PEAKING FACTOR ($F_{\Delta H}$)	MINIMUM DNBR
Bank D at insertion limit, D-12** fully withdrawn	1.63	***
Rod Cluster Control Assembly G-13 fully inserted	1.68	***
Rod Cluster Control Assembly D-12 fully inserted	1.67	***
Rod Cluster Control Assembly H-12 fully inserted	1.68	***
Rod Cluster Control Assembly F-10 fully inserted	1.66	***

*Values include 15% uncertainty allowance in $F_{\Delta H}$.

**Designations such as D-12 specify a core location; see Chapter 4.0.

***Minimum value greater than limit value [ ]; see Section 4.4. (a,c)

TABLE 15.4-3

PARAMETERS USED IN THE ANALYSIS OF THE ROD CLUSTER
CONTROL ASSEMBLY EJECTION ACCIDENT

	VOIDED PORTION OF CYCLE				FLOODED PORTION OF CYCLE			
Time in Life	BOL HZP	BOL HFP	EOL HZP	EOL HFP	BOL HZP	BOL HFP	EOL HZP	EOL HFP
Power level (%)	0	102	0	102	0	102	0	102
Ejected rod worth (%ΔK)	.86	.20	1.04	0.2	.86	.20	0.90	0.2
Delayed neutron fraction	0.55	0.55	0.44	0.44	0.55	0.55	0.44	0.44
Feedback activity weighting	2.071	1.30	3.19	1.30	2.071	1.30	3.19	1.30
Trip reactivity (%Δk)	2.0	4.0	2.0	4.0	2.0	4.0	2.0	4.0
Fq before rod ejection		2.6		2.6		2.6		2.6
Fq after rod ejection	13.0	7.1	24.0	7.1	18.54	7.1	24.0	7.1
Number of operational pumps	2	4	2	4	2	4	2	4
Maximum fuel pellet average temperature (°F)	2784.	3814.	2748.	3606.	2969.	3585.	3296.	3654.
Maximum fuel pellet center temperature (°F)	3131.	4938.	3239.	4866.	3401.	4945.	3682.	4900.
Maximum clad average temperature (°F)	2119.	2234.	2048.	2078.	2223.	2065.	2499.	2101.
Maximum fuel stored energy (cal/gm)	113.6	164.7	111.9	154.	122.4	152.8	135.7	156.5
Percent of fuel melted	0	<10	0	<0	0	<10	0	<10

TABLE 15.4-4 (SHEET 1 OF 3)

PARAMETERS USED IN EVALUATING
THE RADIOLOGICAL CONSEQUENCES OF A
CONTROL ROD EJECTION ACCIDENT
(For a Typical Four-Loop Westinghouse PWR)

I. Source Data	
A. Core power level (MWt)	3565
B. Total steam generator tube leakage (gal/min)	1
C. Reactor coolant iodine activity prior to accident	An assumed preaccident iodine spike, which has resulted in the DE of 60 μCi/g of I-131 in the reactor coolant. See Table 15A-6.
D. Gap activity released to reactor coolant from failed fuel	10 percent. See Table 15A-3.
E. Melted fuel	0.25 percent of core (0.00125 of core iodines, 0.0025 of core noble gases)
F. Reactor coolant noble gas activity	Based on 1 percent defective fuel.
G. Secondary system initial activity	DE of 0.1 μCi/g of I-131.
H. Reactor coolant mass (g)	2.3×10^{8}
I. Secondary coolant mass, 4 generators (g)	1.9×10^{8}
J. Offsite power	Lost after trip
II. Atmospheric Dispersion Factors	See Table 15A-2.

TABLE 15.4-4 (SHEET 2 OF 3)

III. Activity Release Data

A. Containment

1. Leak rate (percent/day)	0.2
2 Mass of primary coolant discharged to containment (lb)	
0 to 1600 s	9.3×10^4
1600 to 4700 s	3.4×10^5
4700 to 10000 s	6.9×10^5
3. Fraction of activity carried by reactor coolant spill that is assumed to be airborne	
Iodines	0.45
Noble gases	1.0

B. Steam generators

1. Primary-to-secondary leak rate (gal/min)[a]	1.0
2. Mass of steam released (lb)	
0 - 214 s	4.9×10^4
3. Iodine partition factor	100

a. Based on water at 590°F, 2250 psia.

TABLE 15.4-4 (SHEET 3 of 3)

IV. Activity Released to the Environment

Isotope	0 to 2 h (Ci)	2 to 2.8 h (Ci)
I-131	74.0	64.0
I-132	73.0	44.0
I-133	141.0	119.0
I-134	587.0	189.0
I-135	1096.0	862.0
Xe-131m	5.2	6.3
Xe-133m	203.5	241.0
Xe-133	1368.0	1630.0
Xe-135m	4.0	0.68
Xe-135	262.0	296.0
Xe-138	22.0	4.4
Kr-85m	147.0	156.0
Kr-85	15.0	18.0
Kr-87	141.0	113.0
Kr-88	328.0	327.0

TABLE 15.4-5

RADIOLOGICAL CONSEQUENCES OF A
CONTROL ROD EJECTION ACCIDENT
(For a Typical Four-Loop Westinghouse PWR)

	Doses (rem)
Exclusion Area Boundary (0 to 2 h)	
Thyroid	13.3
Whole-body gamma	0.16
Low Population Zone Outer Boundary (2.8 h)	
Thyroid	36.4
Whole-body gamma	0.1

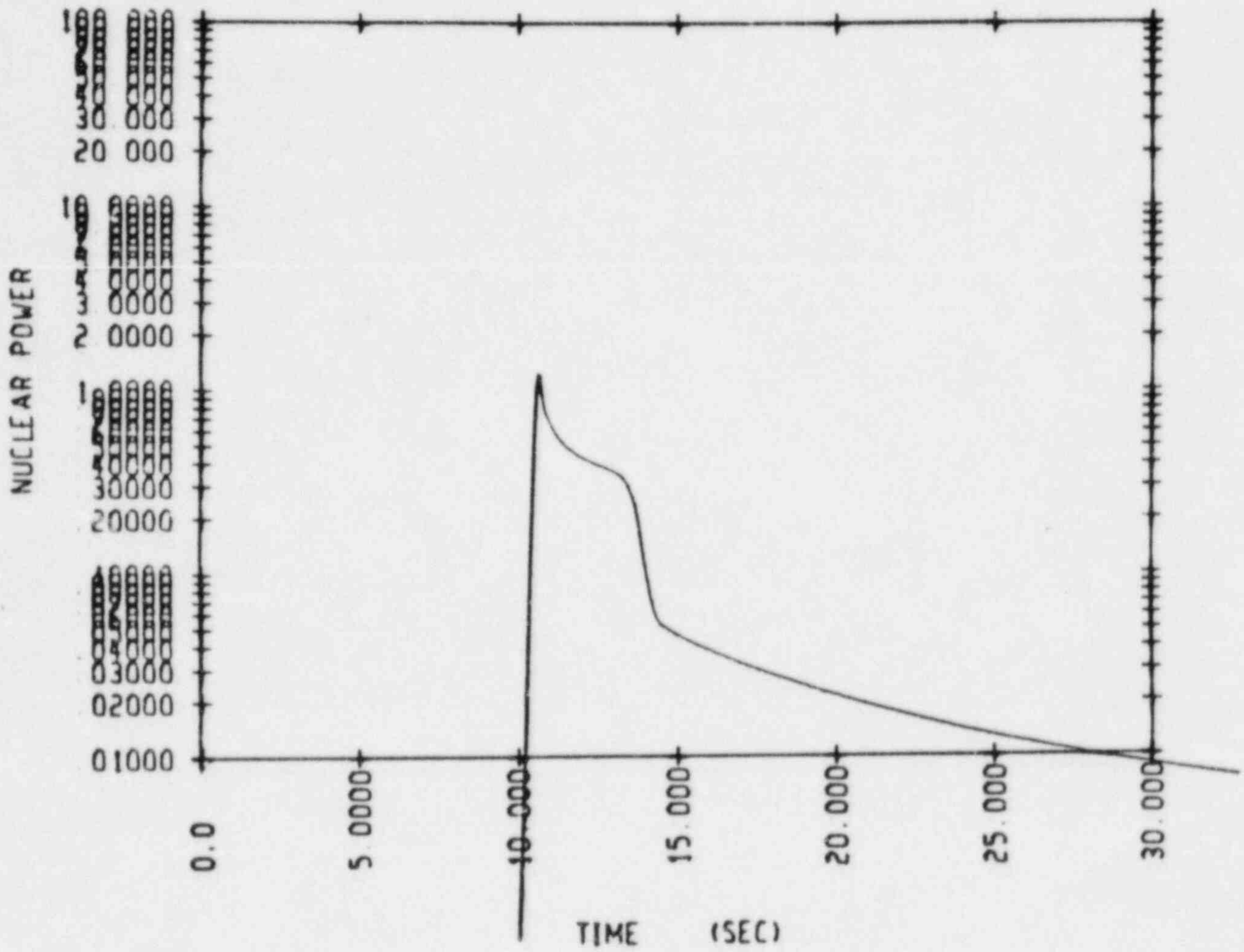


Figure 15.4-1 Neutron Flux Transient for Uncontrolled Rod Withdrawal from a Subcritical Condition

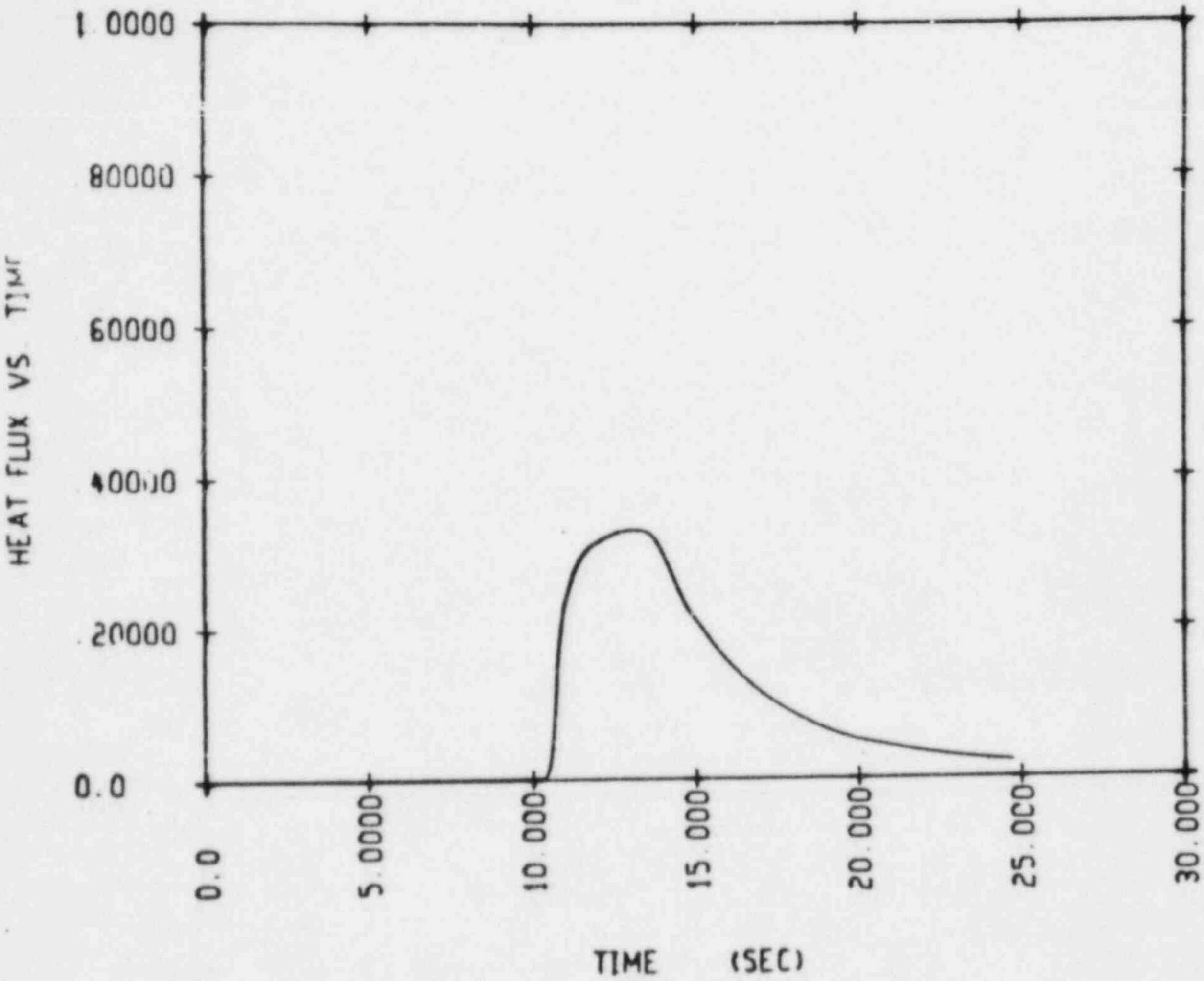


Figure 15.4-2 Thermal Flux Transient for Uncontrolled Rod Withdrawal from a Subcritical Condition

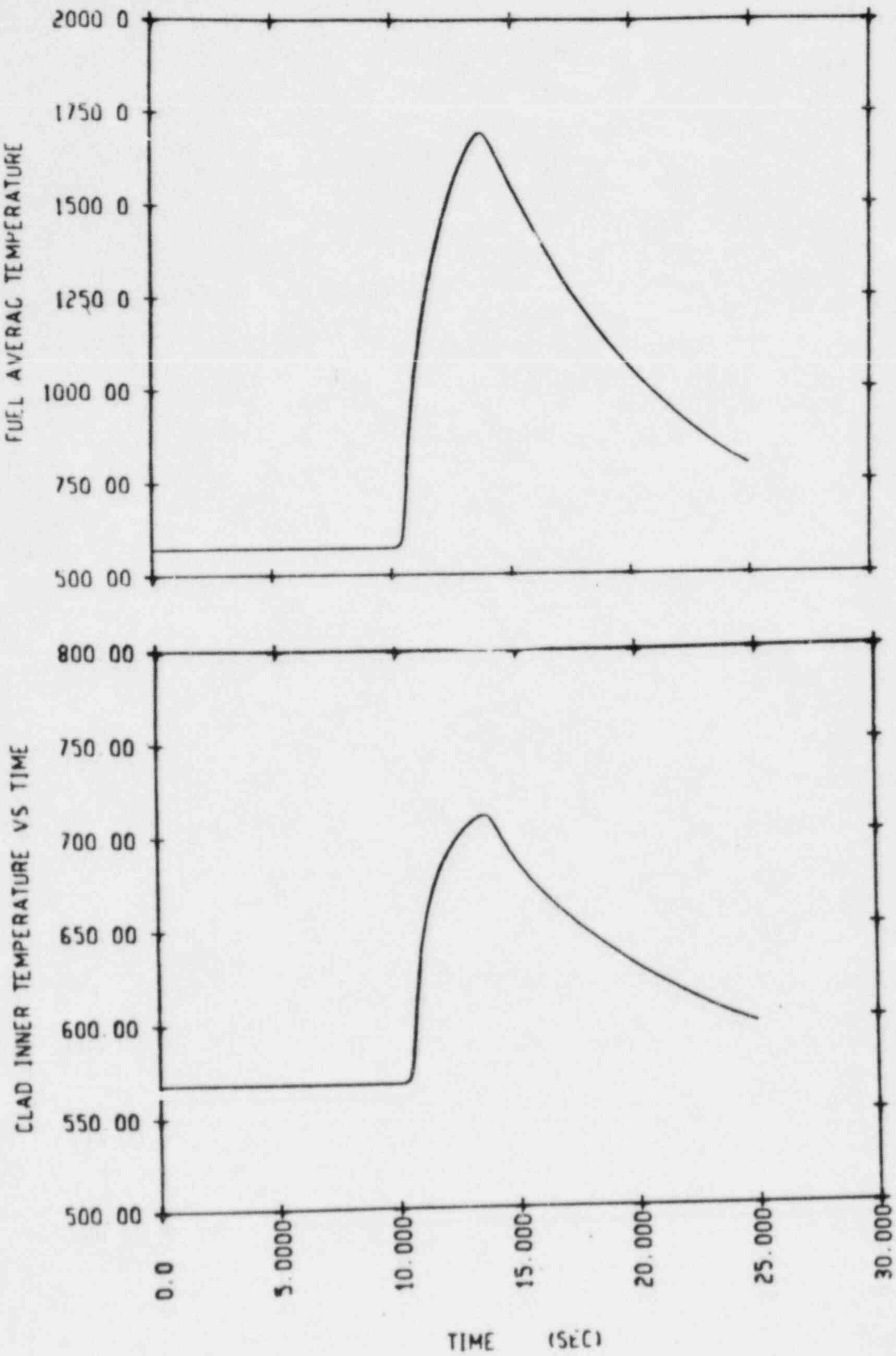


Figure 15.4-3 Fuel and Clad Temperature for Uncontrolled Rod Withdrawal from a Subcritical Condition

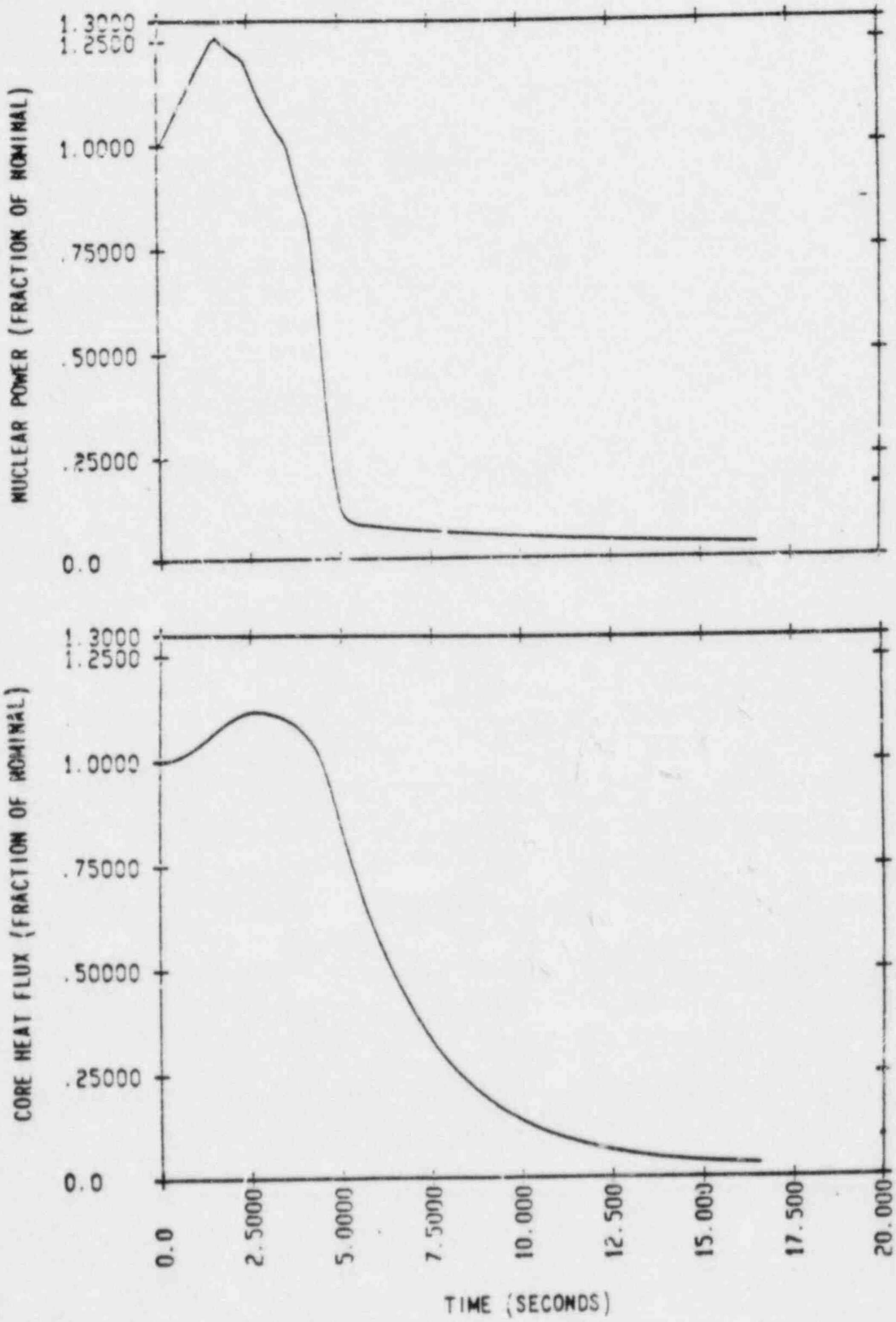


Figure 15.4-4 Nuclear Power Transient and Heat Flux Transient for Uncontrolled Rod Withdrawal from Full Power With Minimum Feedback and 75 PCM/SEC Withdrawal Rate

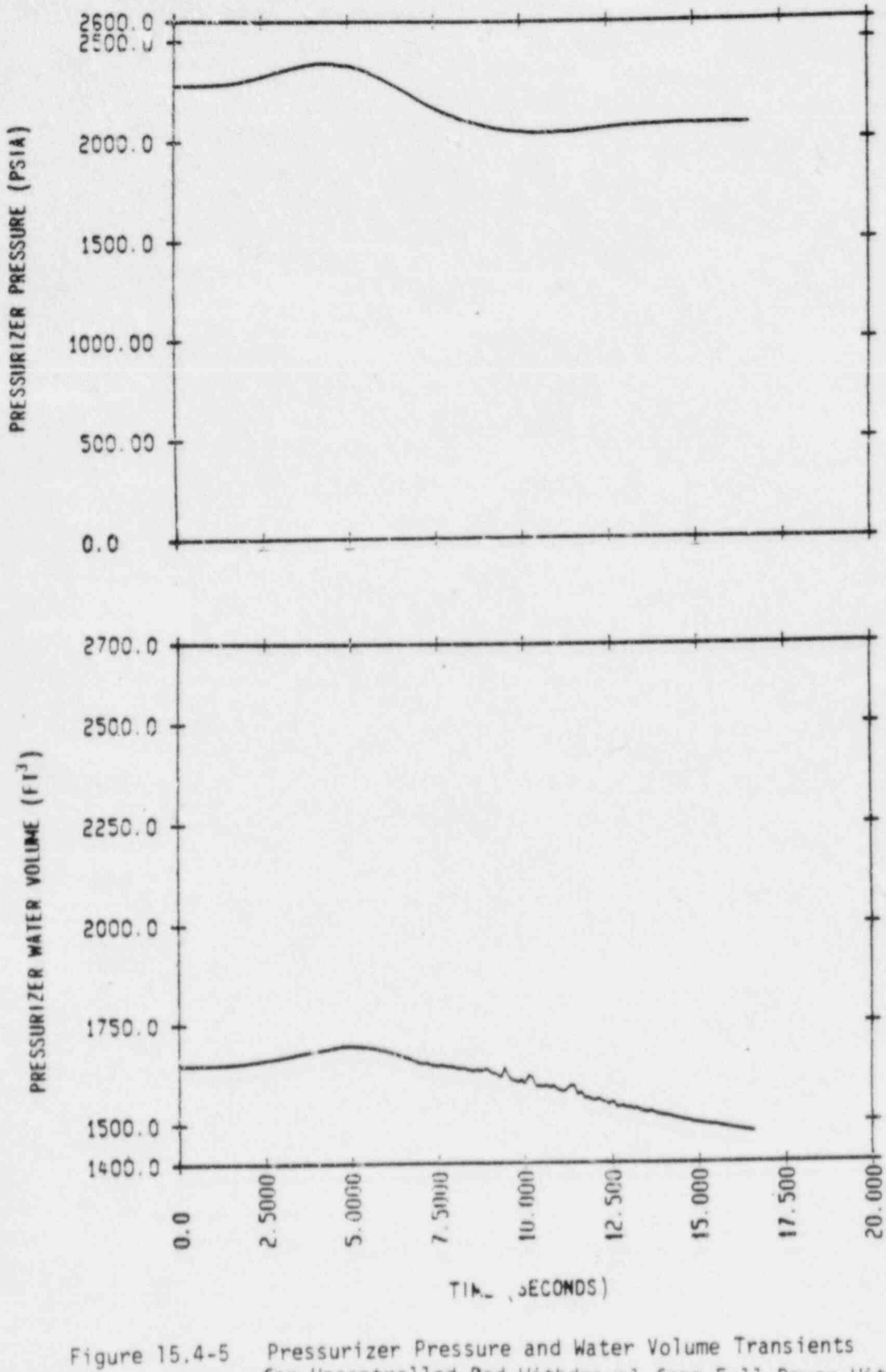


Figure 15.4-5 Pressurizer Pressure and Water Volume Transients for Uncontrolled Rod Withdrawal from Full Power With Minimum Feedback and 75 PCM/SEC Withdrawal Rate

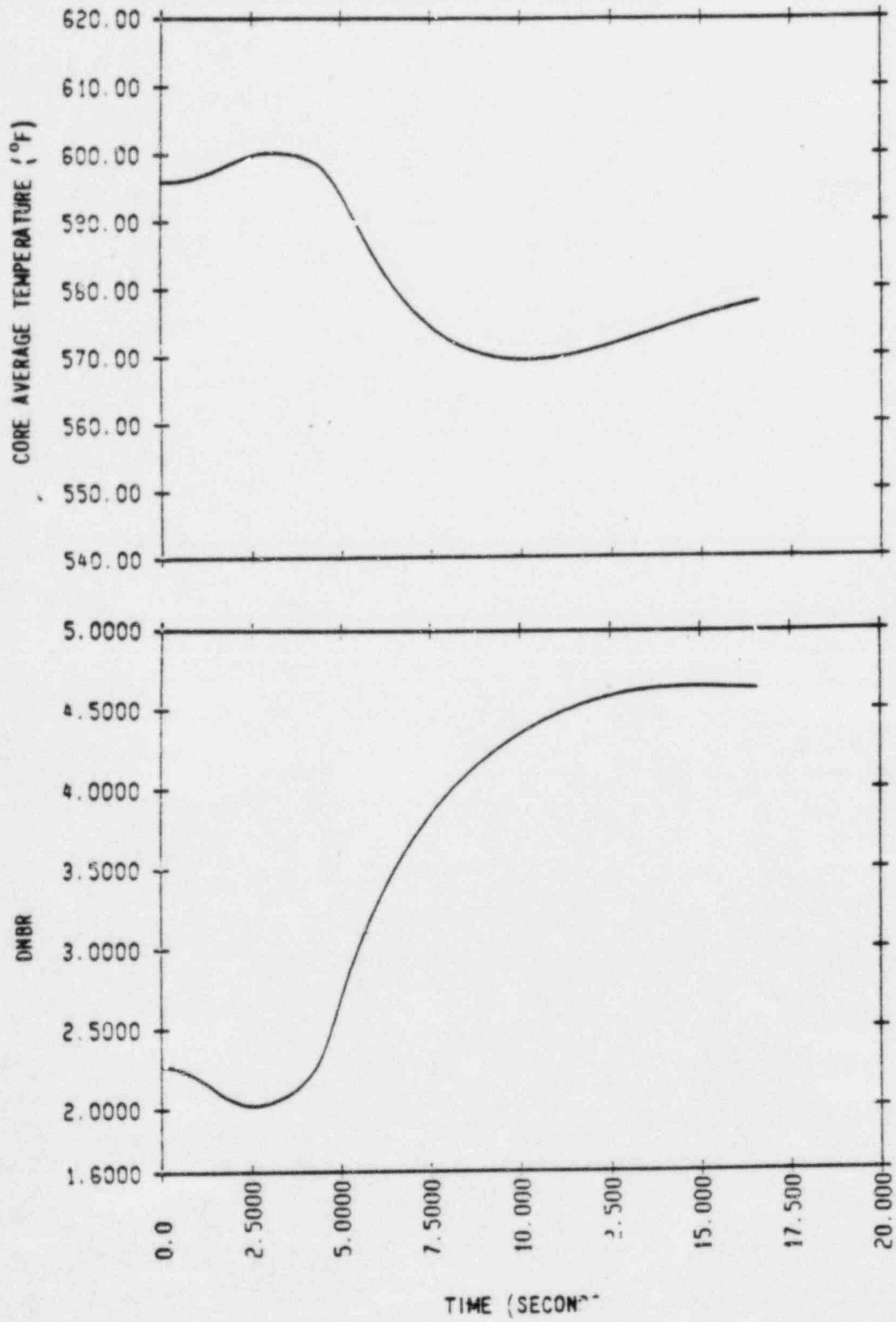


Figure 15.4-6 Core Average Temperature Transient and DNBR vs Time for Uncontrolled Rod Withdrawal from Full Power With Minimum Feedback and 75 PCM/SEC Withdrawal Rate

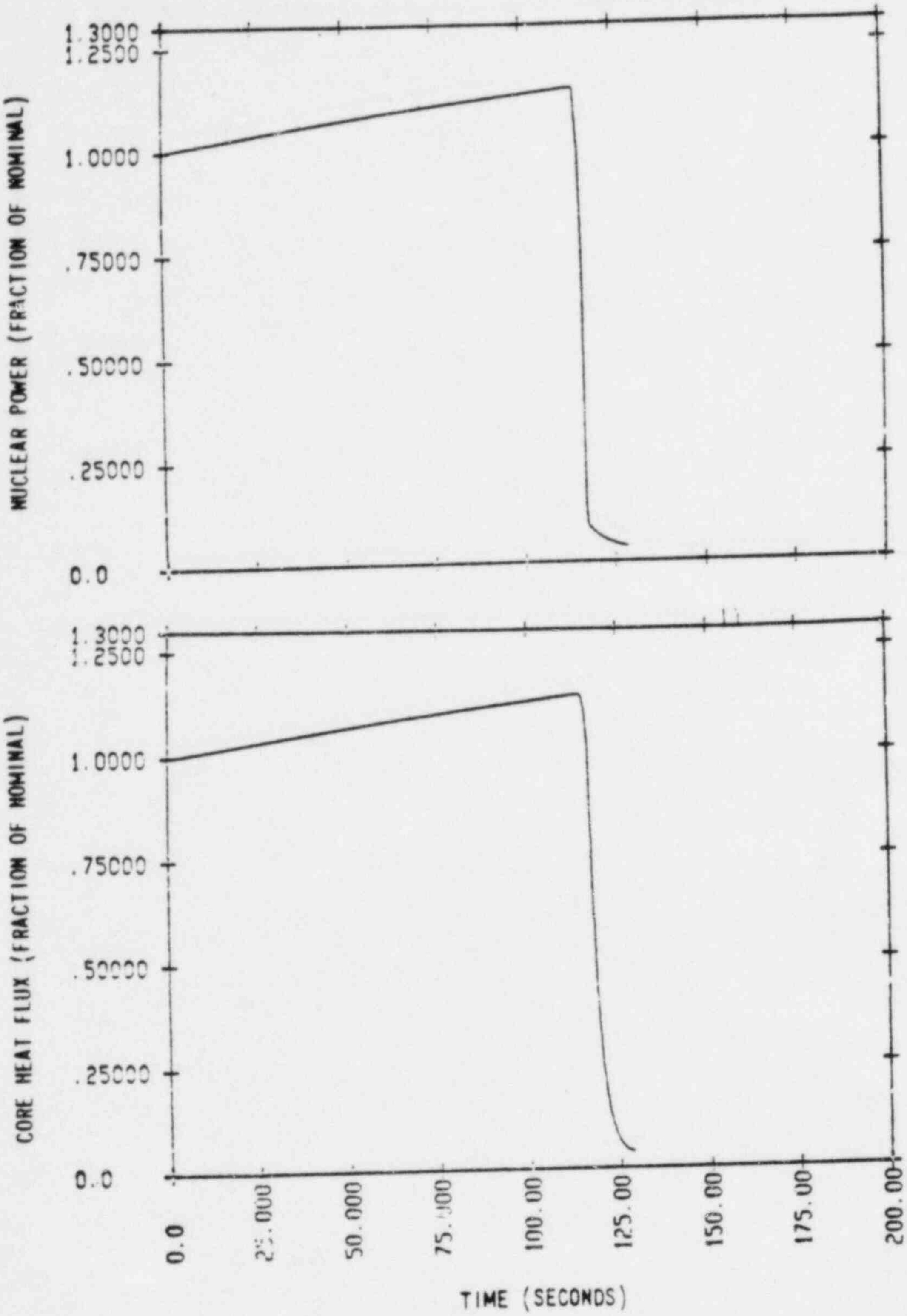


Figure 15.4-7 Nuclear Power Transient and Heat Flux Transients for Uncontrolled Rod Withdrawal from Full Power With Minimum Feedback and 1 PCM/SEC Withdrawal Rate

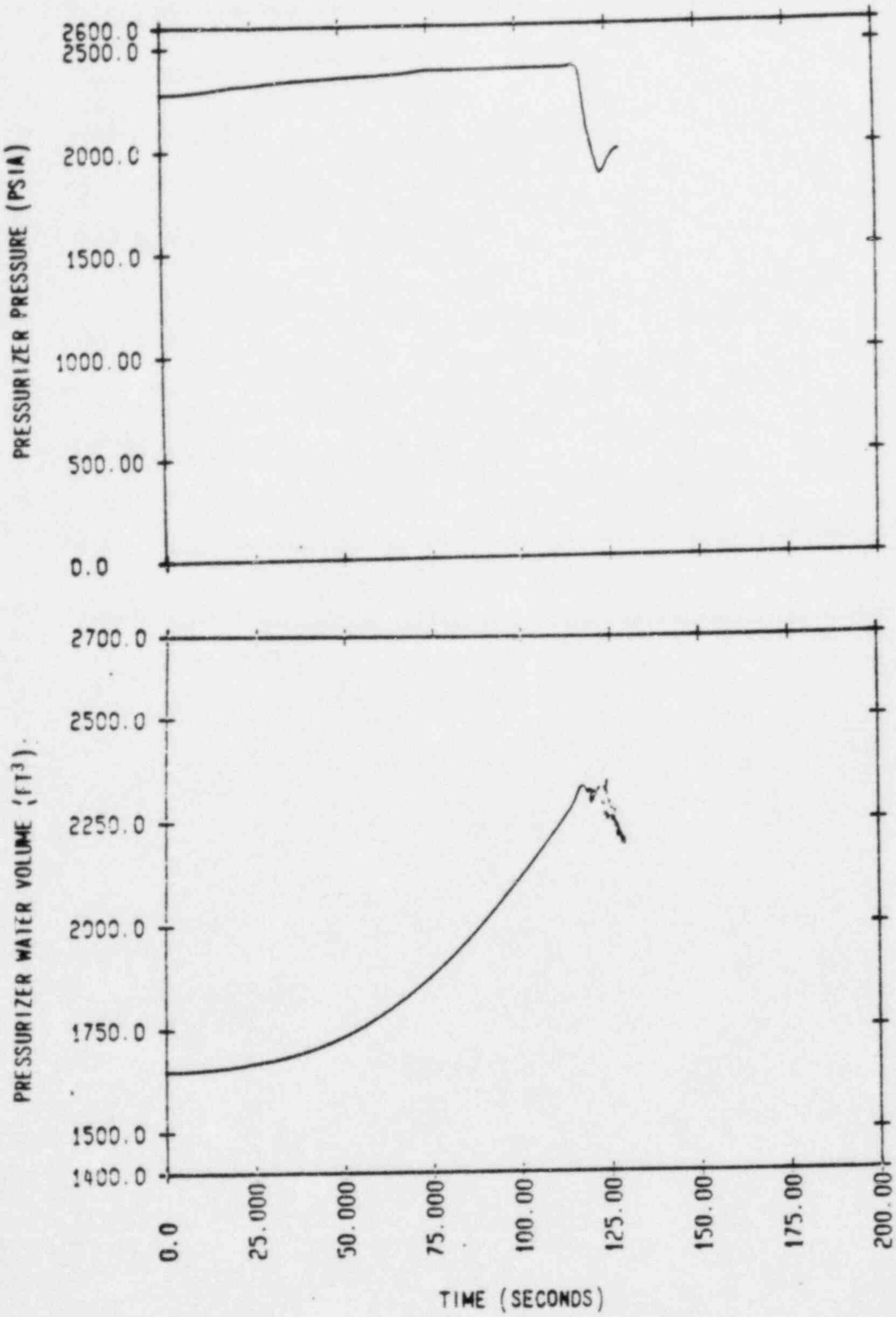


Figure 15.4-8 Pressurizer Pressure and Water Volume Transients for Uncontrolled Rod Withdrawal from Full Power With Minimum Feedback and 1 PCM/SEC Withdrawal Rate

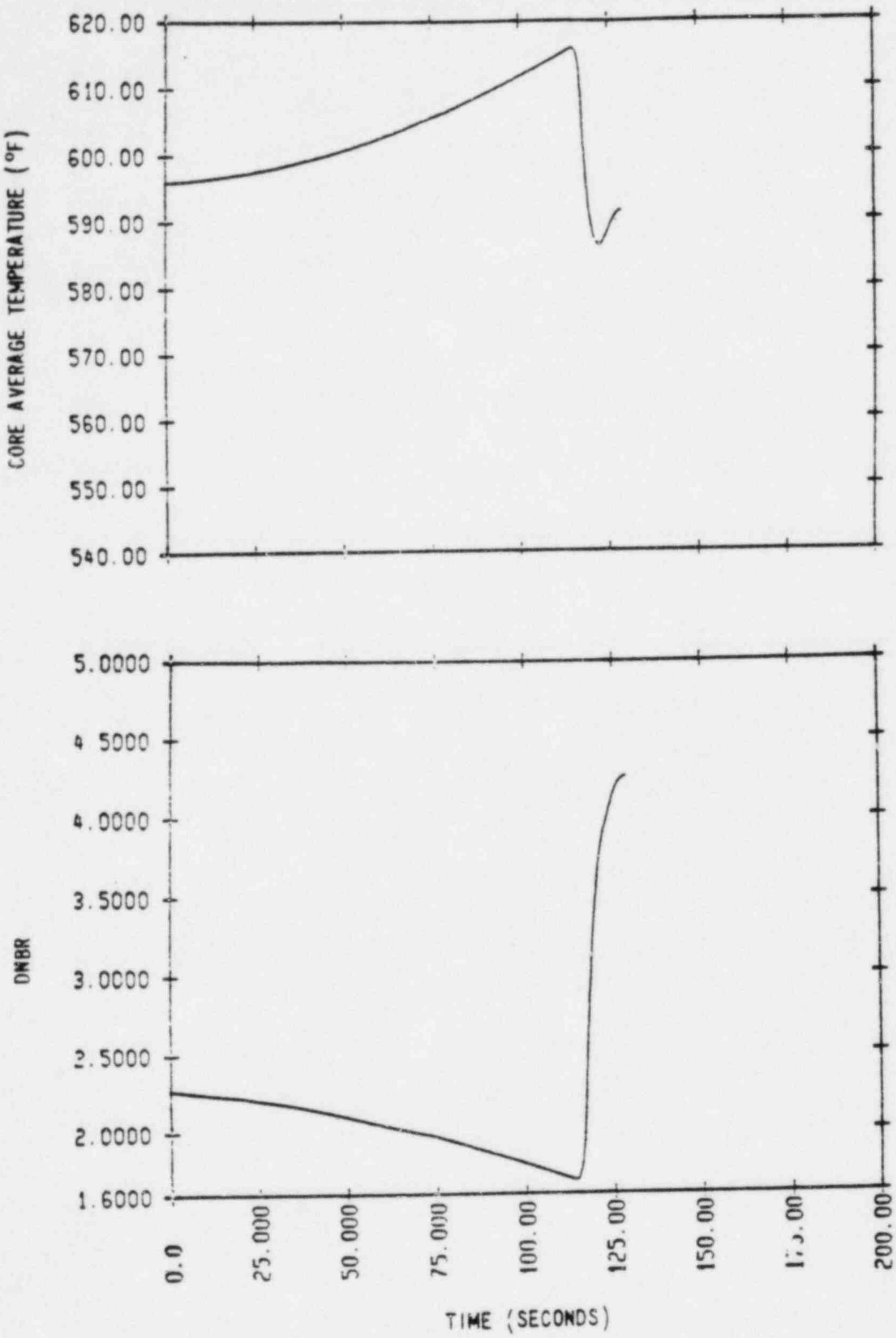


Figure 15.4-9 Core Average Temperature Transient and DNBR vs Time for Uncontrolled Rod Withdrawal from Full Power With Minimum Feedback and 1 PCM/SEC Withdrawal Rate

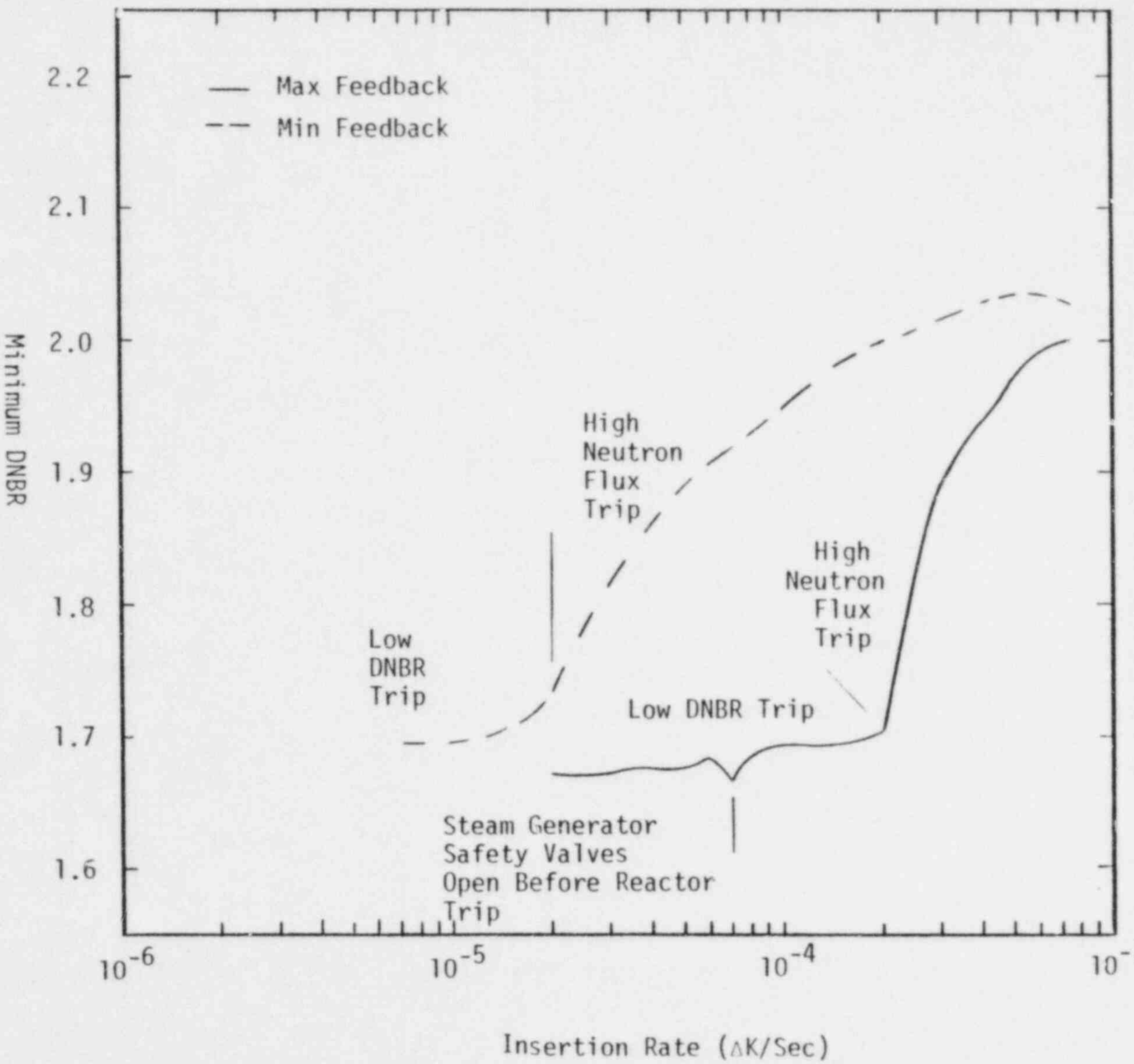


Figure 15.4-10 Rod Withdrawal at Power (100% Power)

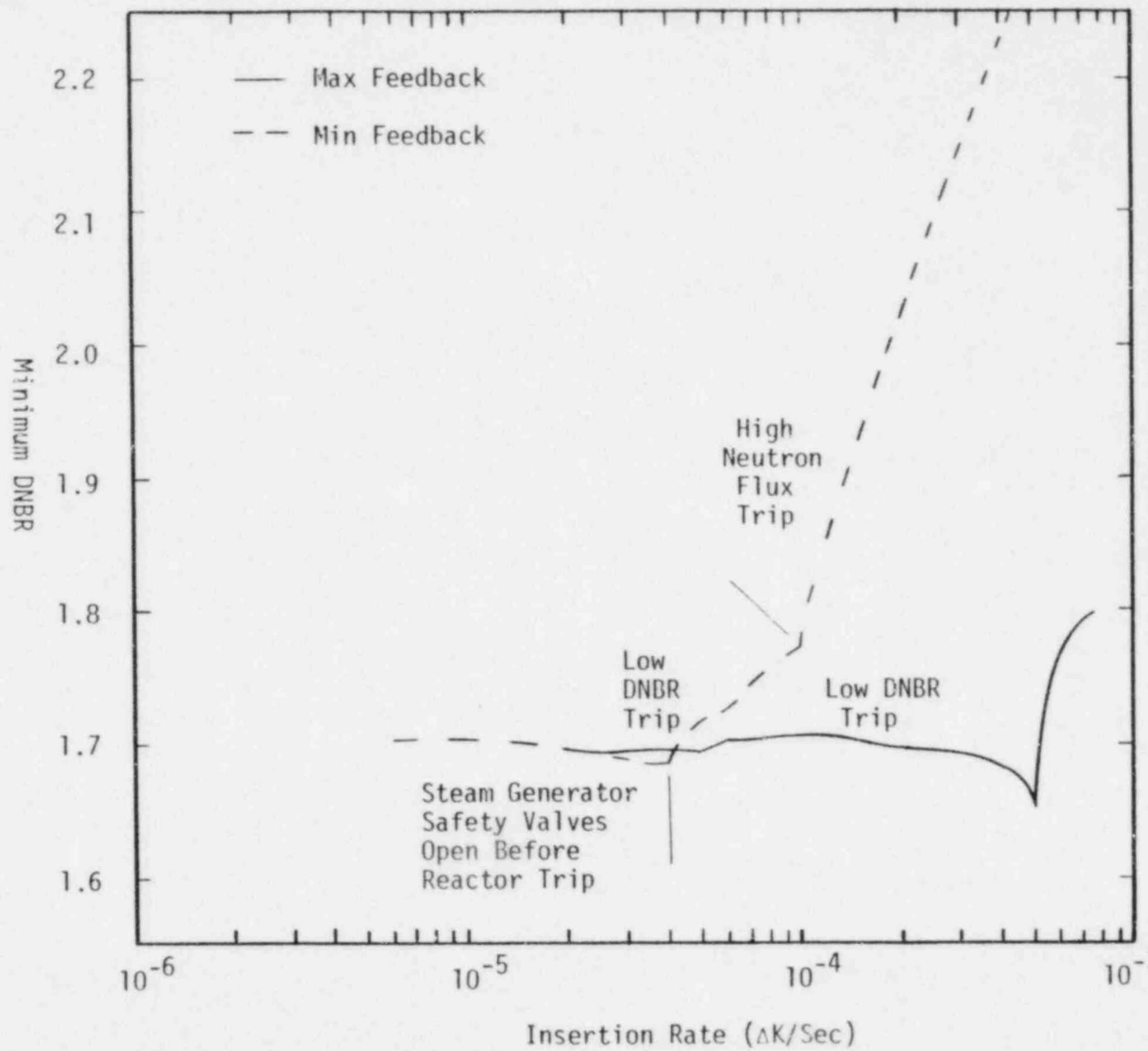


Figure 15.4-11 Rod Withdrawal at Power (60% Power)

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							-8.9			-7.4					
2			-5.6			-9.1		-8.5							
3								-8.2		-6.8		-4.1		0.2	
4		-7.9	-8.2					-7.7							
5					-8.4				-6.0		-3.8		-1.8		
6	-8.5		-8.4			-7.4		-5.5						-0.3	
7				-7.7			-5.0			-1.2			-1.0		
8	-7.7		-7.3		-5.9		-3.2			1.5		3.2	3.4	3.6	
9		-6.9							2.7		5.9				6.0
10					-3.4		0.7					10.6			
11	-5.3				-1.8			5.9			17.1				11.4
12						1.3			12.3			24.6			
13			0.1		0.7			7.7						23.6	
14			2.5				4.7			11.1		17.6			
15					2.1			6.5							

CASE A

FIGURE 15.4-12 REPRESENTATIVE % CHANGE IN LOCAL ASSY. AVG. POWER FOR INTERCHANGE BETWEEN REGION 1 AND REGION 3 ASSY.

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							-4.8			-4.5					
2			-0.4			-4.8		-4.8							
3								-4.8		-4.4		-2.6		1.4	
4		-2.8	-3.1					-4.5							
5					-3.9				-4.3		-4.6		-1.5		
6	-3.8		-3.8			-3.6		-2.9						0.5	
7				-3.6			-2.0			-2.3			2.2		
8	-3.5		-3.4		-2.6		-0.7			11.4		11.3	5.8	4.4	
9		-3.2							5.2		16.7				5.4
10					-1.7		0.5					8.8			
11	-2.2				-1.0			2.2			6.9				6.6
12						0.0			2.9			6.5			
13			1.2		0.0			1.6						10.3	
14			3.2			0.8				3.2		6.0			
15					0.3			1.5							

CASE B-1

FIGURE 15.4-13 REPRESENTATIVE % CHANGE IN LOCAL ASSY. AVG. POWER FOR INTERCHANGE BETWEEN REGION 1 AND REGION 2 ASSY. WITH BP RODS RETAINED BY THE REGION 2 ASSY.

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							1.0			1.1					
2			5.1			1.0		1.0							
3								1.1		1.1		1.9		4.9	
4		1.7	1.7					1.4							
5					1.1				1.8		1.1		0.7		
6	0.0		0.2			1.8		3.9						4.0	
7				0.0			5.2			2.2			-0.3		
8	-0.7		-0.6		0.3		5.1			1.5		-0.3	-0.6	-0.7	
9		-1.0							-1.1		-0.8				-0.9
10					-1.4		-3.1					-1.3			
11	-0.9				-1.7						-1.7				-0.9
12						-2.5			-2.9			-1.1			
13			0.7		-1.9			-2.9						-2.5	
14			2.3				-2.8			-2.4		-0.8			
15					-2.1			-2.8							

CASE B-2

FIGURE 15.4-14 REPRESENTATIVE % CHANGE IN LOCAL ASSY. AVG. POWER FOR INTERCHANGE BETWEEN REGION 1 AND REGION 2 ASSY. WITH THE BP RODS TRANSFERRED TO REGION 1 ASSY.

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							-2.2			-2.1					
2			2.0			-2.0		-2.1							
3								-1.5		-1.6		-1.0		2.0	
4		-0.9	-1.0					-0.4							
5					-0.4				1.2		-0.5		-1.4		
6	-2.1		-1.6			2.3		5.7						-2.0	
7				-3.2			9.7			4.4			-1.7		
8	-2.3		-1.6		1.8		13.6	X		5.6		-0.4	-1.6	-2.1	
9		-2.2							9.7		1.1				-2.2
10					0.3		4.5					-0.9			
11	-1.9				-0.4			1.8			-0.5				-1.9
12						-0.9			-0.6			-1.1			
13			0.4		-1.4			-1.5						2.0	
14			2.0				-2.1			-2.0		-0.9			
15					-1.9			-2.2							

CASE C

FIGURE 15.4-15 REPRESENTATIVE % CHANGE IN LOCAL ASSY. AVG. POWER FOR ENRICHMENT ERROR (REGION 2 ASSY. LOADED INTO CORE CENTRAL POSITION)

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							-11			-14					
2			0.4			-9.2		-12							
3								-12		-14		-15		-13	
4		3.2	1.2					-11							
5					-1.5				-12		-15		-16		
6	9.8		7.1			-1.6		-8.0						-16	
7				9.2			-2.3			-12			-14		
8	20.0		17.8		10.8		0.8			-10		-14	-15	-16	
9		27.2							-5.5		-11				-15
10					20.7		5.8					-12			
11	42.0		X		23.6			1.9			-8.6				-13
12						14.0			-1.7			-8.9			
13			38.6		20.4			2.8						-7.0	
14			35.9				7.0			-3.3		-6.3			
15					15.3			2.9							

CASE D

FIGURE 15.4-16 REPRESENTATIVE % CHANGE IN LOCAL ASSY. AVG. POWER FOR LOADING REGION 2 ASSY. INTO REGION 1 POSITION NEAR CORE PERIPHERY

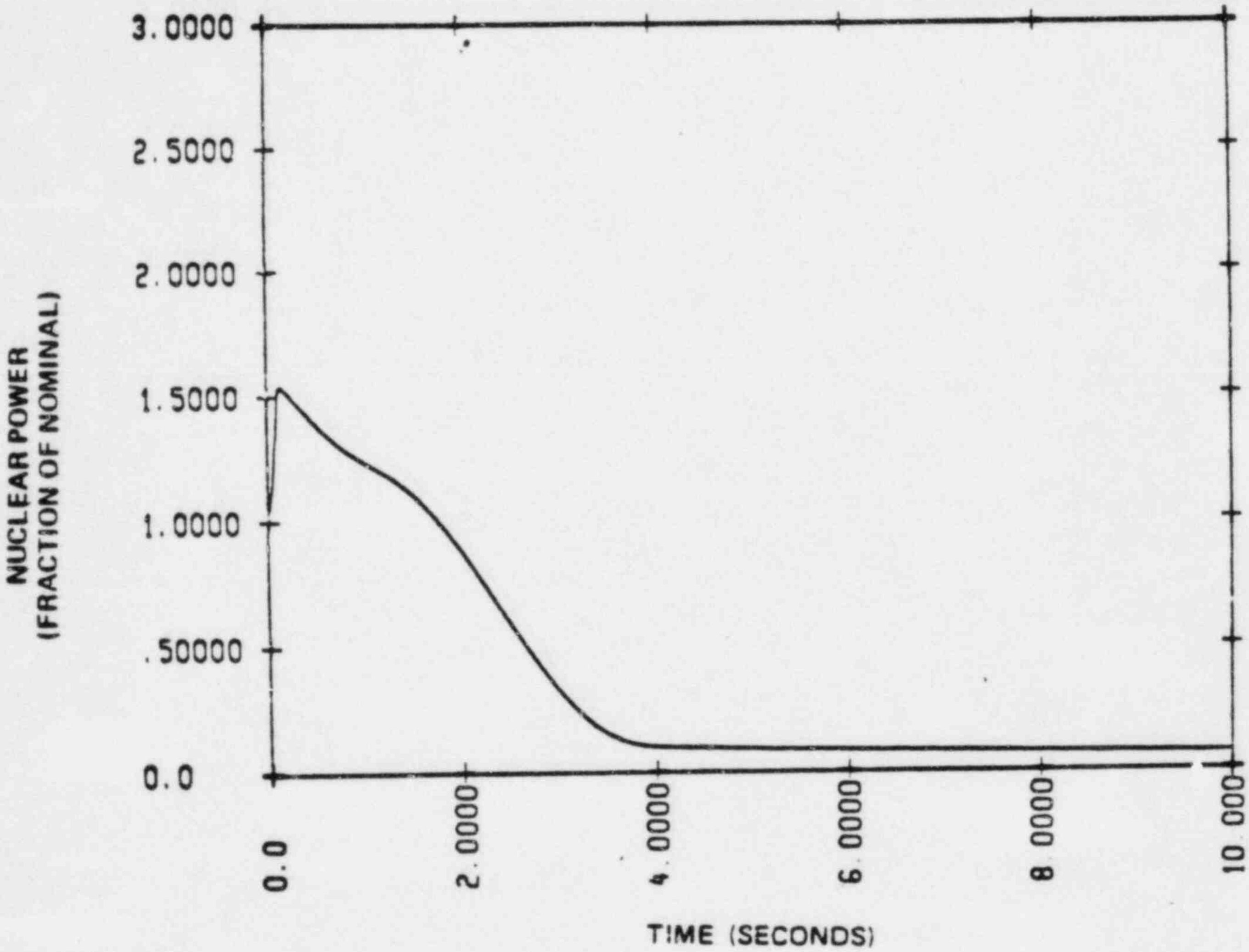


Figure 15.4-17 Nuclear Power Transient
BOL, Voided, Full Power

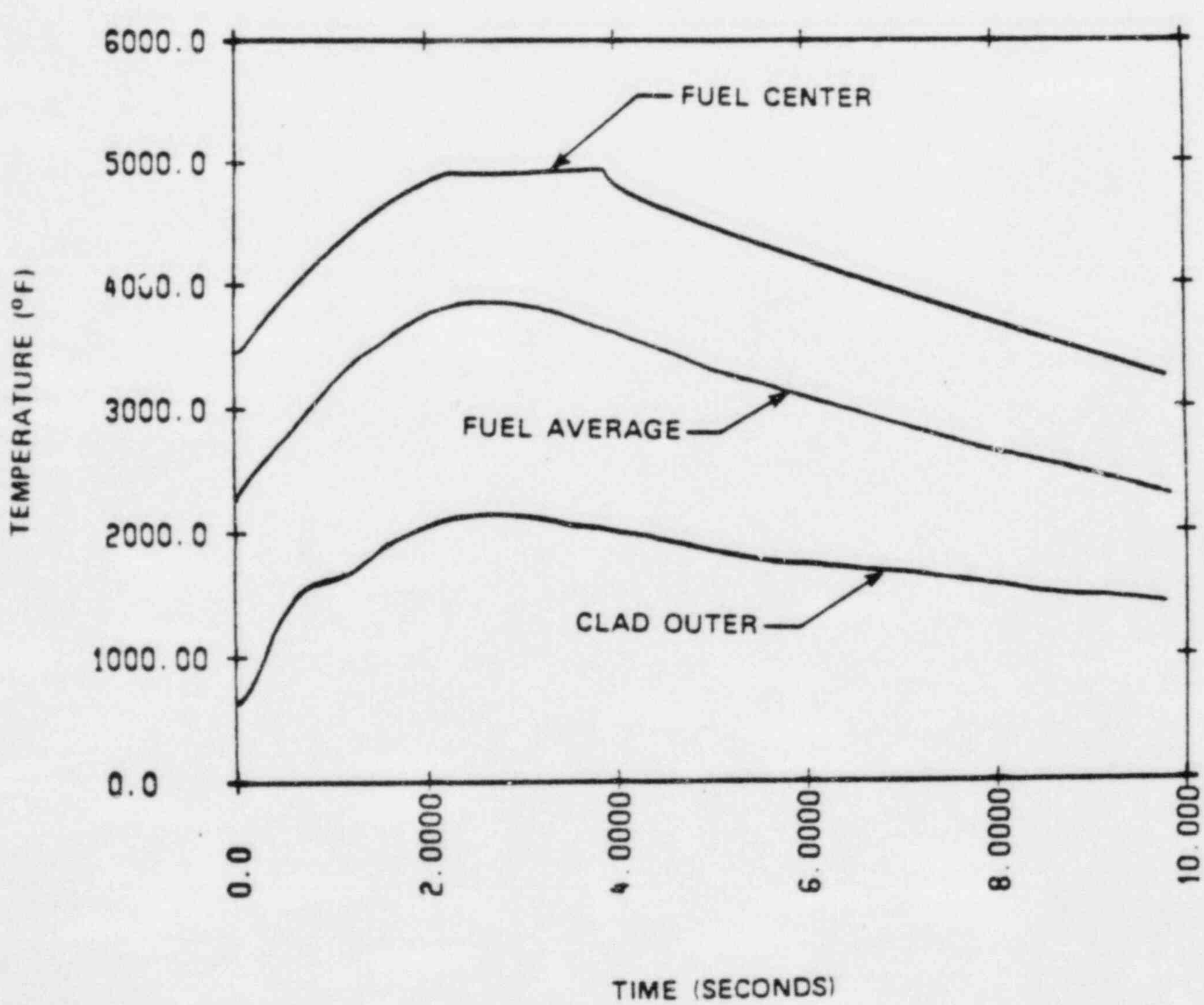


Figure 15.4-18 Hot Spot Fuel and Clad Temperature vs Time BOL, Voided, Full Power

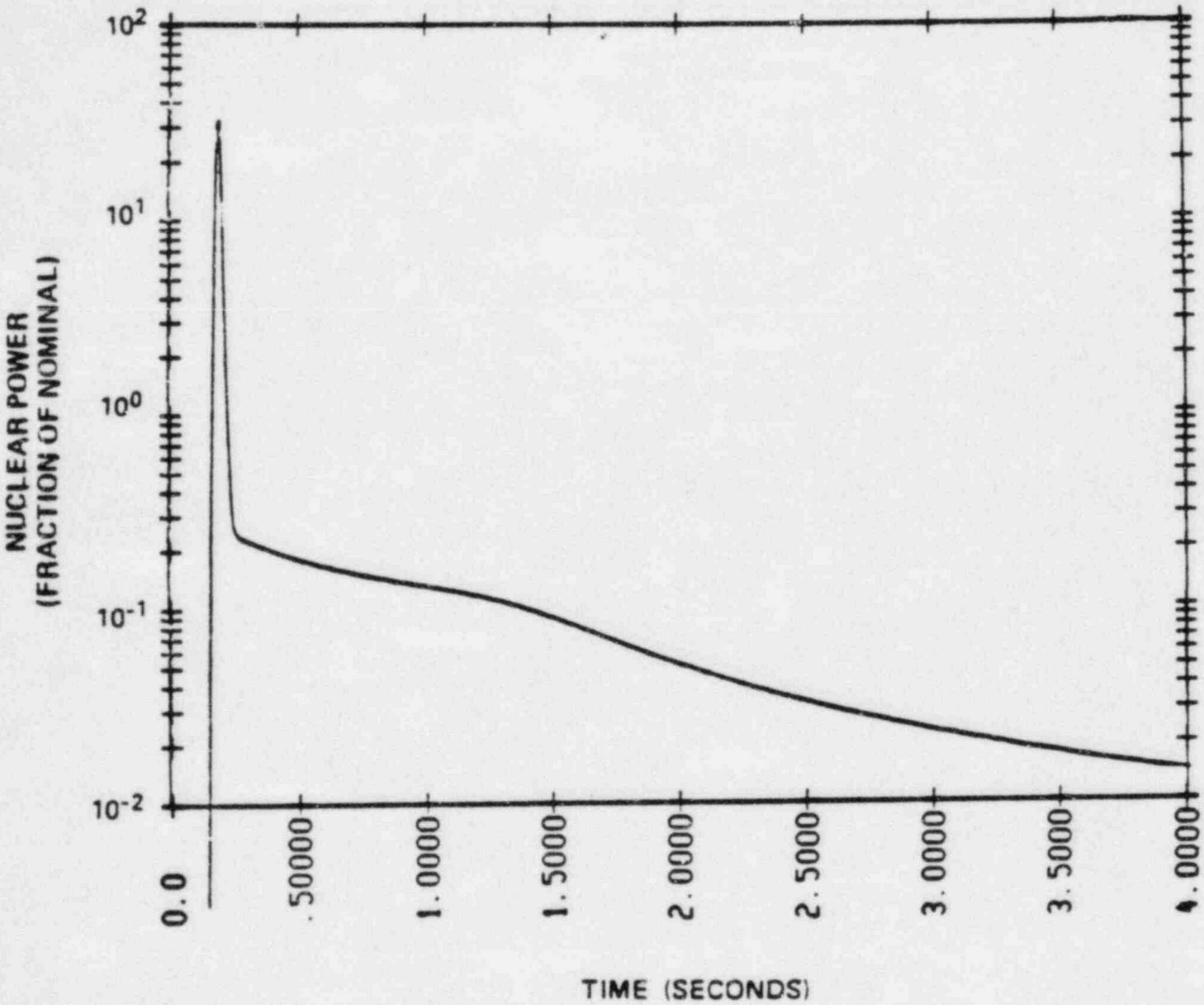


Figure 15.4-19 Nuclear Power Transient
EOL, Flooded, Zero Power

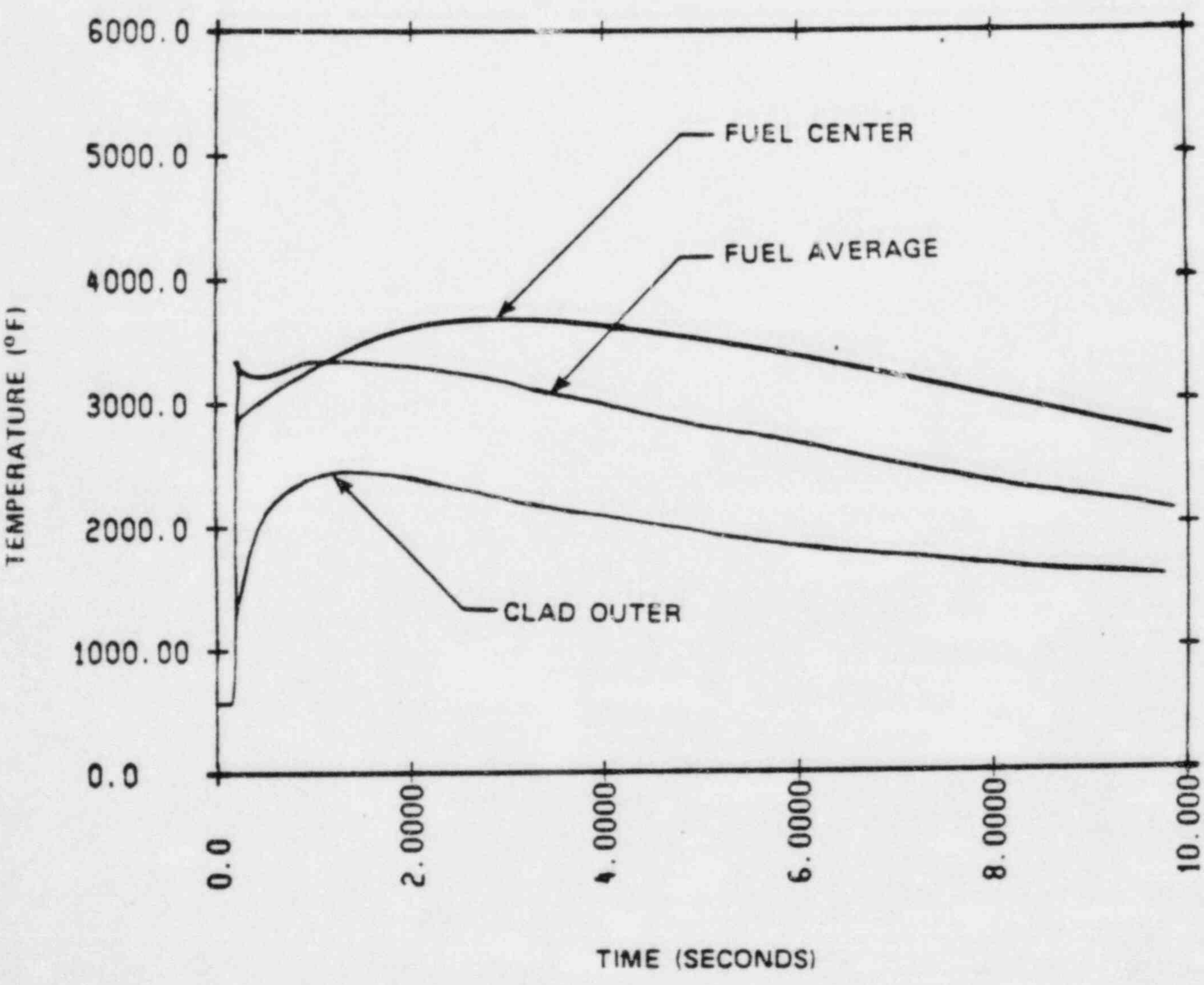


Figure 15.4-20 Hot Spot Fuel and Clad Temperatures vs Time
EOL, Flooded, Zero Power

APPENDIX 15A[1]

ACCIDENT ANALYSIS RADIOLOGICAL CONSEQUENCES EVALUATION MODELS AND PARAMETERS

15A.1 GENERAL ACCIDENT PARAMETERS

This appendix contains the parameters used in analyzing the radiological consequences of postulated accidents. Table 15A-1 contains the general parameters used in all the accident analyses. For parameters specific only to particular accidents, refer to that accident parameter section. The site specific, ground-level release, short-term dispersion factors (For accidents, ground-level releases are assumed.) are based on Regulatory Guide 1.145 (reference 1) methodology and represent the 0.5-percent worst-sector meteorology and these are given in Table 15A-2. The thyroid (via inhalation pathway), beta-skin, and gamma body (via immersion pathway) dose factors based on reference 2 are given in Table 15A-3.

15A.2 OFFSITE RADIOLOGICAL CONSEQUENCES CALCULATIONAL MODELS

This section presents the models and equations used for calculating the integrated activity released to the environment, the accident flowpaths, and the equations for dose calculations. Two major release models are considered:

1. A single holdup system with no internal cleanup.

2. A holdup system wherein a two-region spray model is used for internal cleanup.

---

(1) This Appendix is included in all RESAR-SP/90 Modules for which there are transients with potential radiological consequences.

15A.2.1 ACCIDENT RELEASE PATHWAYS

The release pathways for the major accidents are given in Figure 15A-1. The accident and their pathways are as follows:

A. Loss-of-Coolant Accident (LOCA)

Immediately following a postulated LOCA, the release of radioactivity from the containment in to the environment with the Integrated Safeguards System (ISS) in full operation. The release in this case is calculated using equations 6a and 6b which take into account a two-region spray model within the containment.

B. Control Assembly Ejection (CAE)

Radioactivity release to the environment due to the CAE accident is direct and unfiltered. The releases from the primary system are calculated using equation 5 which considers holdup in the single-region primary system (the spray removal is not assumed); the secondary (steam) releases via the relief valves are calculated without any holdup. The pathways for these releases are A-B and A'-B.

15A.2.2 SINGLE-REGION RELEASE MODEL

It is assumed that any activity released to the holdup system instantaneously diffuses to uniformly occupy the system volume.

The following equations are used to calculate the integrated activity released from postulated accidents.

$$A_1(t) = A_1(0)e^{-\lambda t} \qquad (1)$$

where $A_1(0)$ = initial source activity at time t_0, Ci

$A_1(t)$ = source activity at time t, Ci

λ_1 = total removal constant from primary holdup system, S^{-1}

$$\lambda_1 = \lambda_d + \lambda_{1\ell} + \lambda_r \qquad (2)$$

where

λ_d = decay removal constant, S^{-1}

$\lambda_{1\ell}$ = primary holdup leak or release rate, S^{-1}

λ_r = internal removal constant, i.e., sprays, plateout, etc.; S^{-1}

Thus, the direct release rate to the atmosphere from the primary holdup system

$$R_U(t) = \lambda_{1\ell}\ [A_1(t)] \qquad (3)$$

where:

$R_u(t)$ = unfiltered release rate (Ci/s)

The integrated activity release is the integral of the above equation.

$$IAR\ (t) = \int_o^t R_u(t)\ dt = \int_o^t \lambda_{1\ell}\ A_1(o)e^{-\lambda_1 t}\ dt \qquad (4)$$

This yields:

$$IAR(t) = (\lambda_{1\ell}\ A_1(o)/\lambda_1)\ (1 - e^{-\lambda_1 t}) \qquad (5)$$

15A.2.3 TWO-REGION SPRAY MODEL IN CONTAINMENT (LOCA)

A two-region spray model is used to calculate the integrated activity released to the environment. The model consists of sprayed and unsprayed regions in containment and a constant mixing rate between them.

As it is assumed that there are no sources after initial release of the fission products, the remaining processes are removal and transfer so that the multivolume containment is described by a system of coupled first-order differential equations.

For a two-region model, the above system reduces to

$$\frac{dA_1}{dt} = -\sum_{j=1}^{K_1} \lambda_{1j} A_1 - Q_{12} \frac{A_1}{V_1} + Q_{21} \frac{A_2}{V_2} \qquad (6a)$$

$$\frac{dA_2}{dt} = -\sum_{j=1}^{K_2} \lambda_{2j} A_2 - Q_{21} \frac{A_2}{V_2} + Q_{12} \frac{A_1}{V_1} \qquad (6b)$$

where

A_i = fission product activity in volume i, Ci

$Q_{i\ell}$ = transfer rate from volume i to volume ℓ, cc/s

V_i = volume of the ith compartment, cc

λ_{ij} = removal rate of the jth removal process in volume i, s^{-1}

K_i = total number of removal processes in the volume i

To calculate the integrated activity released to the atmosphere, the release rate of activity is first calculated. This is found from

$$R(t) = \sum_{i=1}^{2} \lambda_{1\ell} \; A(t) \qquad (7)$$

The integrated activity released from time $t_0 - t_1$ is then

$$IAR = \int_{t_0}^{t_1} R\,(t)\,dt$$

15A.2.4 OFFSITE THYROID DOSE CALCULATION MODEL

Offsite thyroid doses are calculated using the equation:

$$D_{TH} = \sum_i DCF_{THi} \sum_j (IAR)_{ij} \; (BR)_j \; (x/Q)_j \qquad (8)$$

where

$(IAR)_{ij}$ = integrated activity of isotope i released[a] during the time interval j, Ci

$(BR)_j$ = breathing rate during time interval j, m^3/s

$(\chi/Q)_j$ = offsite atmospheric dispersion factor during time interval j, s/m^3

DCF_{THi} = thyroid dose conversion factor via inhalation for isotope i rem/Ci

D_{TH} = thyroid dose via inhalation, rems

15A.2.5 OFFSITE BETA-SKIN DOSE CALCULATIONAL MODEL

Assuming a semi-infinite cloud of beta emitters, off-site beta-skin doses are calculated using the equation:

$$D_{\beta S} = {}_{i}\ DCF_{\beta i}\ {}_{j}\ (IAR)_{ij}\ (\chi/Q)_j$$

where

$D_{\beta S}$ = beta-skin dose in rem

$DCF_{\beta i}$ = beta-skin dose conversion factor for the ith isotope in $rem\text{-}m^3/Ci\text{-}s$

and $(IAR)_{ij}$ and $(\chi/Q)_j$ are defined in Subsection 15A.2.4.

15A.2.6 OFFSITE GAMMA-BODY DOSE CALCULATIONAL MODEL

Assuming a semi-infinite cloud of gamma emitters, offsite gamma-body doses are calculated using the equation:

---

a. No credit is taken for cloud depletion by ground deposition and radioactive decay during transport to the exclusion area boundary or the outer boundary of the low-population zone.

$$D_{\gamma B} = \sum_i DCF_{\gamma i} \sum_j (IAR)_{ij} \ (\chi/Q)_j$$

where

$(IAR)_{ij}$ and $(\chi Q)_j$ are defined in Section 15A.2.4.

and

$DCF_{\gamma i}$ = gamma-body dose conversion factor for the ith isotope in rem-m^3/Ci-s

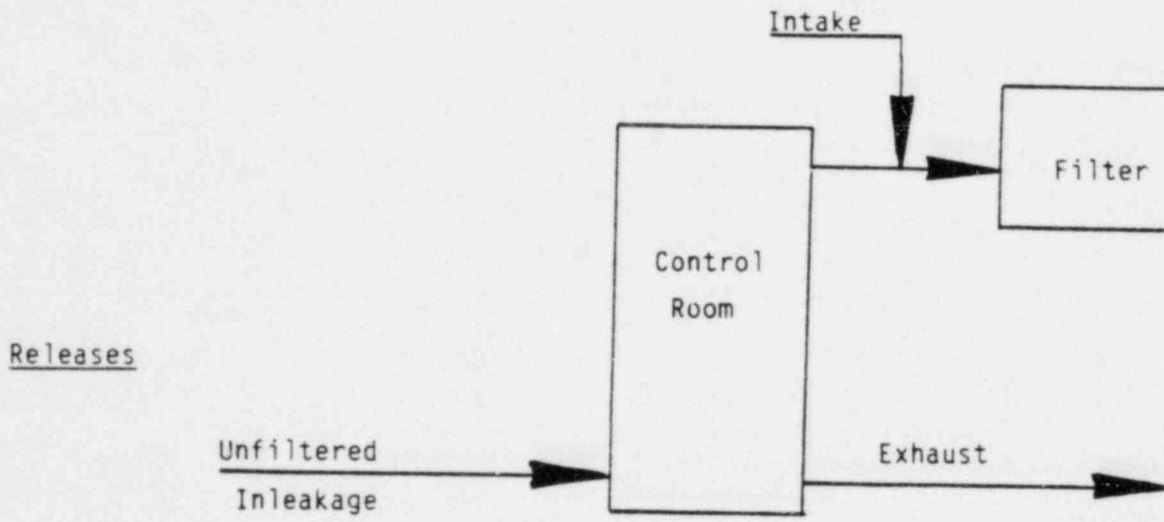
$D_{\gamma B}$ = gamma-body dose in rem

15A.3 CONTROL ROOM RADIOLOGICAL CONSEQUENCES CALCULATIONAL MODELS

Radiation doses to a control room operator as a result of a postulated LOCA are presented in this chapter. (A study of the radiological consequences in the control room due to various postulated accidents indicate that the LOCA is the limiting case.)

15A.3.1 INTEGRATED ACTIVITY IN CONTROL ROOM

The integrated activity in the control room during each time interval is found by multiplying the release by the appropriate χ/Q to give the concentration at the control room intake. This activity is brought into the control room through the filtered intake and by unfiltered inleakage. The control room ventilation system recirculates control room air through charcoal filters and exhausts a portion to the atmosphere.



From this we can calculate the total integrated activity in the control room during any time interval.

The activity in the control room can be calculated by the same method used to calculate activity in the containment.

15A.3.2 INTEGRATED ACTIVITY CONCENTRATION IN CONTROL ROOM FROM SINGLE-REGION SYSTEM

To calculate the integrated activity concentration in the control room we must first calculate the activity in the control room at any time t, and then integrate again to find the integrated activity.

$$\frac{dA_{CR}(t)}{dt} = [F_2R_{FIN} + R_{UIN} \frac{X}{Q} R(t) - \lambda_3 A_{CR}(t)$$

where:

$A_{CR}(t)$ = activity in the control room at any time t, Ci

F_2 = filter nonremoval fraction on intake

R_{FIN} = filtered intake rate in m^3/s

R_{UIN} = unfiltered intake rate in m^3/s

R(t) = activity of release in Ci/s given in equation 3 of subsection 15A.2.2

$\lambda_3 = \lambda_{3\ell} + \lambda_d + \lambda_r$

where

λ_3 = total removal rate from control room in s^{-1}

$\lambda_{3\ell}$ = exhaust rate from control room in s^{-1}

λ_d = isotopic decay constant in s^{-1}

λ_r = recirculation removal rate in s^{-1}

The integrated activity in the control room (IA_{CR}) is determined by the expression

$$IA_{CR}(t) = \frac{1}{V_{CR}} \int_o^t A_{CR}(t)\, dt$$

Where: V_{CR} = control room volume

This $IA_{CR}(t)$ is used to calculate the doses to the operator in the control room. This activity is multiplied by an occupancy factor which accounts for the time fraction the operator is in the control room.

15A.3.3 CONTROL ROOM THYROID DOSE CALCULATIONAL MODEL

Control room thyroid doses via inhalation pathway are calculated using the following equation:

$$D_{TH-CR} = BR \sum_{i} DCF_{THi} \sum_{j} (IA_{CRij})\,(O_j)$$

where

D_{TH-CR} = control room thyroid dose in rem

BR = breathing rate assumed to be always 3.47×10^{-4} m^3/s

DCF_{THi} = thyroid dose conversion factor for adult via inhalation in rem/Ci for isotope i

IA_{CRij} = integrated activity concentration in control room, Ci-s/m^3 for isotope i during time interval j

o_j = control room occupancy fraction during time interval j

15A.3.4 CONTROL ROOM BETA-SKIN DOSE CALCULATIONAL MODEL

The beta-skin doses to a control room operator are calculated using the following equation:

$$D_{\beta-CR} = {}_{i}\; DCF_{\beta i}\; {}_{j}\; (IA_{CRij}) \times o_j$$

$D_{\beta-CR}$ = beta skin dose in the control room (rem).

$DCF_{\beta i}$ = beta skin dose conversion factor for isotope i (rem-m^3/Ci-s)

IA_{CRij} = integrated activity concentration in the control room, Ci-s for isotope i during time interval jm^3.

o_j = control room occupancy fraction during time interval j.

15A.3.5 CONTROL ROOM GAMMA-BODY DOSE CALCULATION

Due to the finite structure of the control room, the gamma-body doses to a control room operator will be substantially less than what they would be due to immersion in an infinite cloud of gamma emitters. The finite cloud gamma doses are calculated using Murphy's method (reference 3) which models the control room at a hemisphere. The following equation is used:

$$D_{B-CR} = \frac{1}{GF} \sum_i DCF_i \sum_j (IA_{CRij}) (O_j)$$

where

GF = dose reduction due to control room geometry factor

GF = $1173/V_1^{0.338}$

V_1 = volume of the control room, ft^3

DCF_i = gamma-body dose conversion factor for isotope i, $rem\text{-}m^3/Ci\text{-}s$

D_{B-CR} = gamma-body dose in the control room, rem

Other symbols have been defined in Subsections 15A.2.5 and 15A.3.3.

15A.3.5.1 Model for Radiological Consequences Due to Radioactive Cloud External to the Control Room

This dose is calculated based on the semi-infinite cloud model which is modified using the protection factors described in Subsection 7.5.4 of reference 4 to account for the control room walls.

15A.4 REFERENCES

1. USNRC Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," August 1979.

2. USNRC Regulatory Guide 1.109, Rev. 1, "Calculation of Annual Doses to Man From Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50 Appendix I," October 1977.

3. Murphy, K. G., and Campe, K. M., "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19," Paper presented at the 13th AEC Air Cleaning Conference.

4. "Meteorology and Atomic Energy 1968," D. H. Slade (ed.), USAEC Report, TID 24190, 1968.

TABLE 15A-1

PARAMETERS USED IN ACCIDENT ANALYSIS

General		(a,c)
Core power level, MWt		
Full-power operation, effective full-power days (EFPD)		
Maximum radial peaking factor		
Steam generator tube leak rate, gal/min		
Sources		
Activity Release Parameters		
Free volume of containment, ft^3		
Containment leak rate		
0-24 h, percent per day		
After 24 h, percent per day		
Control room		
Free volume, ft^3		
Unfiltered infiltration rate, ft^3/min		
Filtered intake rate, ft^3/min		
Internal recirculation rate through filters, ft^3/min		
Iodine removal efficiency for recirculation filters (all forms of iodine), percent	95	
Iodine removal efficiency for intake filters (all forms of iodine), percent	95	
High efficiency particulate air filter efficiency, percent	99	
Miscellaneous		
Atmospheric dispersion factors (x/Q), s/m^3	Table 15A-2	
Dose conversion factors		
Gamma-body and beta skin, $rem\text{-}m^3/Ci\text{-}s$	Table 15A-4	
Thyroid, rem/Ci	Table 15A-4	

TABLE 15A-2

LIMITING SHORT-TERM ATMOSPHERIC DISPERSION FACTORS FOR ACCIDENT ANALYSIS (s/m^3)*

Location Type/ Time Interval (h)	(x/Q)
Site boundary	
0-2	2.0E-4
Low-population zone	
0-2	7.0E-5
2-8	3.5E-5
8-24	2.0E-5
24-96	9.0E-6
96-720	3.0E-6
Control room	
0-2	4.0E-3
2-8	3.0E-3
8-24	2.8E-3
24-96	2.0E-3
96-720	1.5E-3

* For the A. W. Vogtle Site.

TABLE 15A-3
DOSE CONVERSION FACTORS USED IN ACCIDENT ANALYSIS

Nuclide	Gamma-Body $\frac{Rem\text{-}m^3}{Ci\text{-}s}$	Beta-Skin $\frac{Rem\text{-}m^3}{Ci\text{-}s}$	Thyroid (Rem/Ci)
I-131	NA	NA	1.49E+6
I-132	NA	NA	1.43E+4
I-133	NA	NA	2.69E+5
I-134	NA	NA	3.73E+3
I-135	NA	NA	5.60E+4
Kr-85m	3.71E-2	4.63E-2	NA
Kr-85	5.11E-4	4.25E-2	NA
Kr-87	1.88E-1	3.09E-1	NA
Kr-88	4.67E-1	7.52E-2	NA
Xe-131m	2.91E-3	1.51E-2	NA
Xe-133m	7.97E-3	3.15E-2	NA
Xe-133	9.33E-3	9.70E-3	NA
Xe-135m	9.91E-2	2.25E-2	NA
Xe-135	5.75E-2	5.90E-2	NA
Xe-138	2.80E-1	1.31E-1	NA

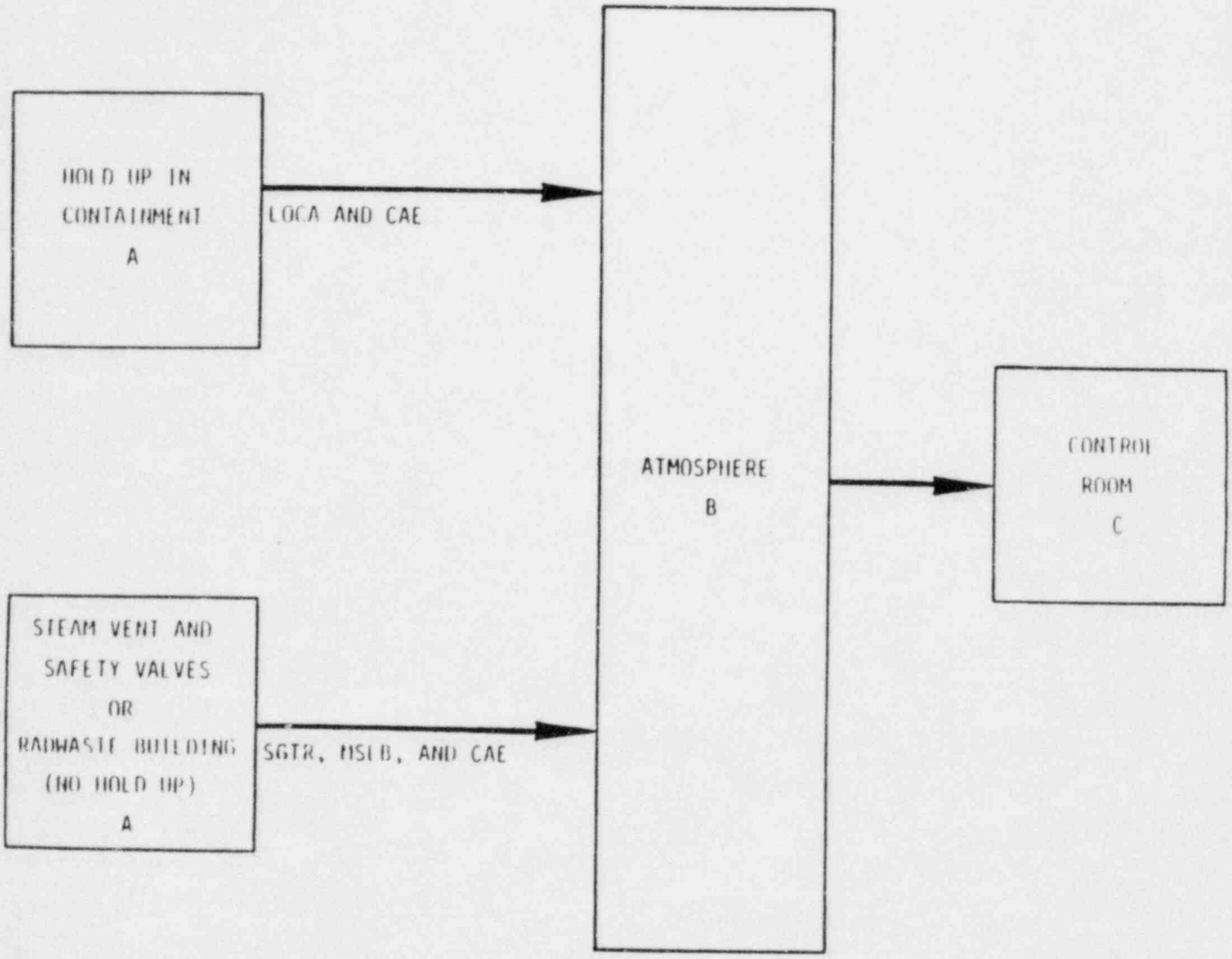


Figure 15.A-1 Release Pathways