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 ± 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 this module).

12. Uncontrolled rod cluster control assembly bank withdrawal from a subcritical or low power startup condition (Subsection 15.4.1 of RESAR-SP/90 PDA Module 5, "Reactor System").

13. Uncontrolled rod cluster control assembly bank withdrawal at power (Subsection 15.4.2 of RESAR-SP/90 PDA Module 5, "Reactor System").

14. Control rod misalignment - Dropped full length assembly, dropped full length assembly bank, or statically misaligned full length assembly) (Subsection 15.4.3 of RESAR-SP/90 PDA Module 5, "Reactor System").

15. Startup of an inactive reactor coolant loop at an incorrect temperature (Subsection 15.4.4 of this module).

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 this module).

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 this module).

3. Control rod misalignment - Single rod cluster control assembly withdrawal at full power (Subsection 15.4.3 of RESAR-SP/90 PDA Module 5, "Reactor Systems").

4. Inadvertent loading and operation of a fuel assembly in an improper position (Subsection 15.4.7 of RESAR-SP/90 PDA Module 5, "Reactor System").

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 this module).

4. Reactor coolant pump shaft break (Subsection 15.3.4 of this module).

5. Spectrum of rod cluster control assembly ejection accidents (Subsection 15.4.8 of RESAR-SP/90 PDA Module 5, "Reactor System").

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 RESAR-SP/90 PDA Module 5, "Reactor System".

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 RESAR-SP/90 PDA Module 5, "Reactor System" 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 RESAR-SP/90 PDA Module 5, "Reactor System". 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 RESAR-SP/90 PDA Module 5, "Reactor System".

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 (a,c) insertion time to dashpot entry is conservatively taken as [ ] seconds. The normalized rod cluster control assembly position vs. time assumed in accident analyses is shown in Figure 15.0-3.

Figure 15.0-4 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 15.0-4 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 15.0-5. The curve shown in this figure was obtained from Figures 15.0-3 and 15.0-4. A total negative reactivity insertion (a,c) following a trip of [ ] $\Delta\rho$ 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 RESAR-SP/90 PDA Module 5, "Reactor System".

The normalized rod cluster control assembly negative reactivity insertion vs. time curve for an axial power distribution skewed to the bottom (Figure 15.0-5) 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 15.0-3) 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 RESAR-SP/90 PDA Module 5, "Reactor System".

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 [ ] All points below and to the left of a DNB line for a (a,c)

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 RESAR-SP/90 PDA Module 5, "Reactor System".

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 of RESAR-SP/90 PDA Module 5, "Reactor System".

15.0.13 REFERENCES

1. H. Chelemer, et al., "Improved Thermal Design Procedure", WCAP-8567-P (Proprietary), July 1975, and WCAP-8568 (Non-Proprietary) July, 1975.

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7. C. Hunin, "FACTRAN, A FORTRAN IV Code for Thermal Transients in a UO_2 Fuel Rod", WCAP-7908, June 1972.

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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 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)	
Core inlet temperature (°F)	
Reactor coolant average temperature (°F)	
Reactor coolant system pressure (psia)	
Reactor coolant flow per loop (gpm)	
Total reactor coolant flow (10^6 lb/hr)	
Total steam flow from NSSS (10^6 lb/hr)	
Steam pressure at steam generator outlet (psia)	
Maximum steam moisture content (%)	
Feedwater temperature at steam generator inlet (°F)	
Average core heat flux (Btu/hr-ft^2)	

(a,c)

* 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
	(a,c)
Core inlet temperature (°F)	
Reactor coolant average temperature (°F)	
Reactor coolant system pressure (psia)	
Reactor coolant flow per loop (gpm)	
Total reactor coolant flow (10^6 lb/hr)	
Total steam flow from NSSS (10^6 lb/hr)	
Steam pressure at steam generator outlet (psia)	
Maximum steam moisture content (%)	
Feedwater temperature at steam generator inlet (°F)	
Average core heat flux (Btu/hr-ft^2)	

* 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	Kinetic Parameters Assumed: Moder. Density (Δp/gm/cc)	Kinetic Parameters Assumed: 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)	Feedwater Temp. (°F)
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)	Feedwater Temp. (°F)	
15.3 Decrease in Reactor Coolant System Flow Rate													(a,c)
Partial and Complete Loss of Forced Reactor Coolant Flow	LOFTRAN, THINC, FACTRAN	[		]	WRB-2	Yes	3816	[	]	2250	[	]	
Reactor Coolant Pump Shaft Seizure (locked rotor	LOFTRAN, FACTRAN	[		]	N/A	N/A	3892	[	]	2280	[	]	
15.4 Reactivity and power distribution anomalies													
Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical or Low Power Startup Condition.	(See RESAR-SP/90 PDA Module 5, "Reactor System")												
Uncontrolled Rod Cluster Assembly Bank Withdrawal at Power	(See PESAR-SP/90 PDA Module 5, "Reactor System")												
Control Rod Mis-alignment	(See RESAR-SP/90 PDA Module 5, "Reactor System")												(a,c)
Startup of an Inactive Reactor Coolant Loop at an Incorrect Temperature	THINC, LOFTRAN, FACTRAN	[		]	WRB-2	Yes	2671	[	]	2250	[	]	

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)	Feedwater 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	3427	[	]	2250	[	] (a,c)
Spectrum of Rod Cluster Control Assembly Ejection Accidents	TWINKLE, FACTRAN	[		]	NA	NA	3427 0	[	]	NA	NA	NA (a,c)
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	LOFTRAN	[		]	WRB-2	Yes	3816	[	]	2250	[	] (a,c)

* 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		
Power Range high neutron flux, low setting		
Power range neutron flux, high negative rate		
High neutron flux, P-8		
Low DNBR		
High pressurizer pressure		
Low pressurizer pressure		
Low reactor coolant flow (from loop flow detectors)		
RCP underspeed		
Turbine trip		
Safety injection reactor trip		
Low steam generator level		
High steam generator level - produces feedwater isolation and turbine trip		

(a,c)

* See RESAR-SP/90 PDA Module 5, "Reactor System"

** 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			
Feedwater pressure (small correction on enthalpy)			
Steam pressure (small correction on entnalpy			
Feedwater flow			
Assumed calorimetric error (% of rated power)			
Axial power distribution effects on total ion chamber current			
Estimated error (% rated power)			
Assumed error (% of rated power)			
Instrumentation channel drift and setpoint reproducibility			
Estimated error (% or rated power)			
Assumed error (% of rated power)			
Total assumed error in setpoint (a) + (b) + (c)			

(a,

TABLE 15.0-5 (Con't)

	Percent Rated Power
Nominal Setpoint	[ ] (a,c)
Maximum overpower trip point assuming all individual errors are simultaneously in the most adverse direction.	

TABLE 15.0-6

PLANT SYSTEMS AND EQUIPMENT AVAILABLE FOR TRANSIENT AND ACCIDENT CONDITIONS

Incident	Reactor Trip Functions	ESF Actuation Functions	Other Equipment	ESF Equipment
15.1 Increase in Heat Removed by the Secondary System				
Feedwater System Malfunction 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 feed System Safety Injection System
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 feed system; Safety Injection System
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 Equipment
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 feed system
Loss of Normal Feedwater Flow	Steam generator low level, manual	Steam generator low level	Steam generator safety valves	Auxiliary feed 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 feed system, Safety Injection System
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 Equipment
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 Equipment
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 Injection 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 Core Cooling System, Auxiliary Feedwater System, Emergency 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 Core Cooling System, Auxiliary Feedwater System, Containment Heat Removal System, Emergency Power System

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

IODONE 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.

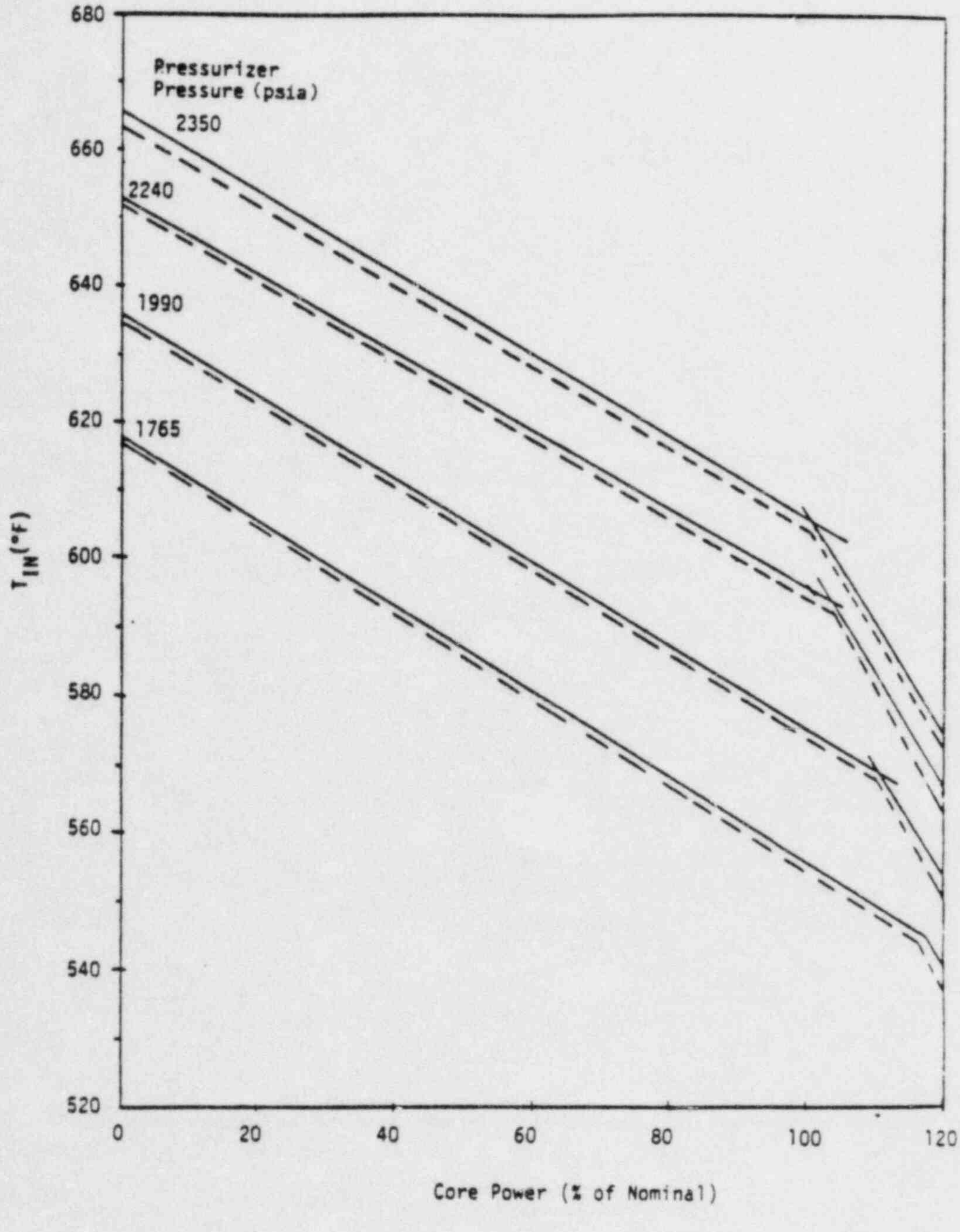
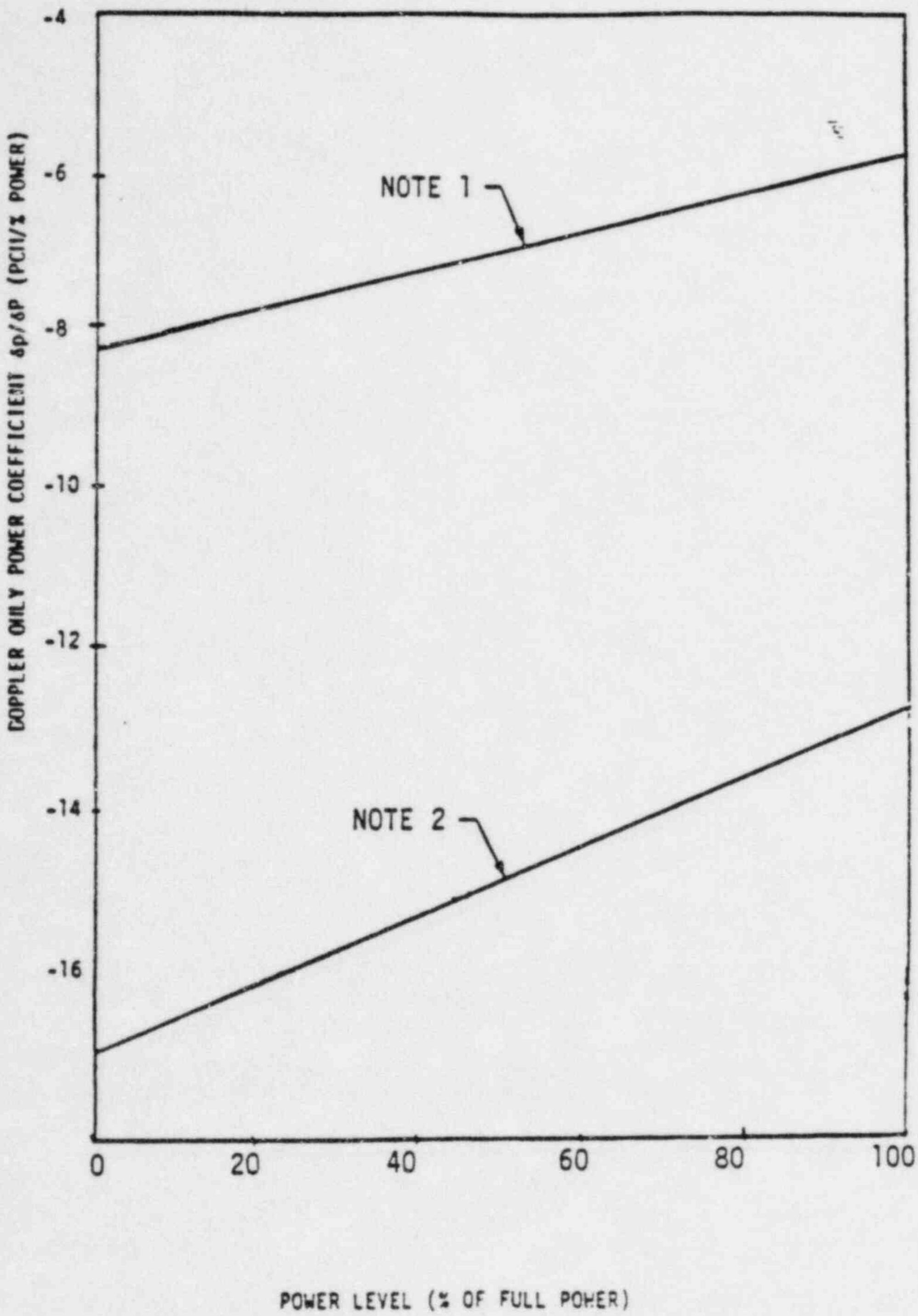


FIGURE 15.0-1 ILLUSTRATION OF CORE THERMAL LIMITS AND DNB PROTECTION (N LOOP OPERATION)



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

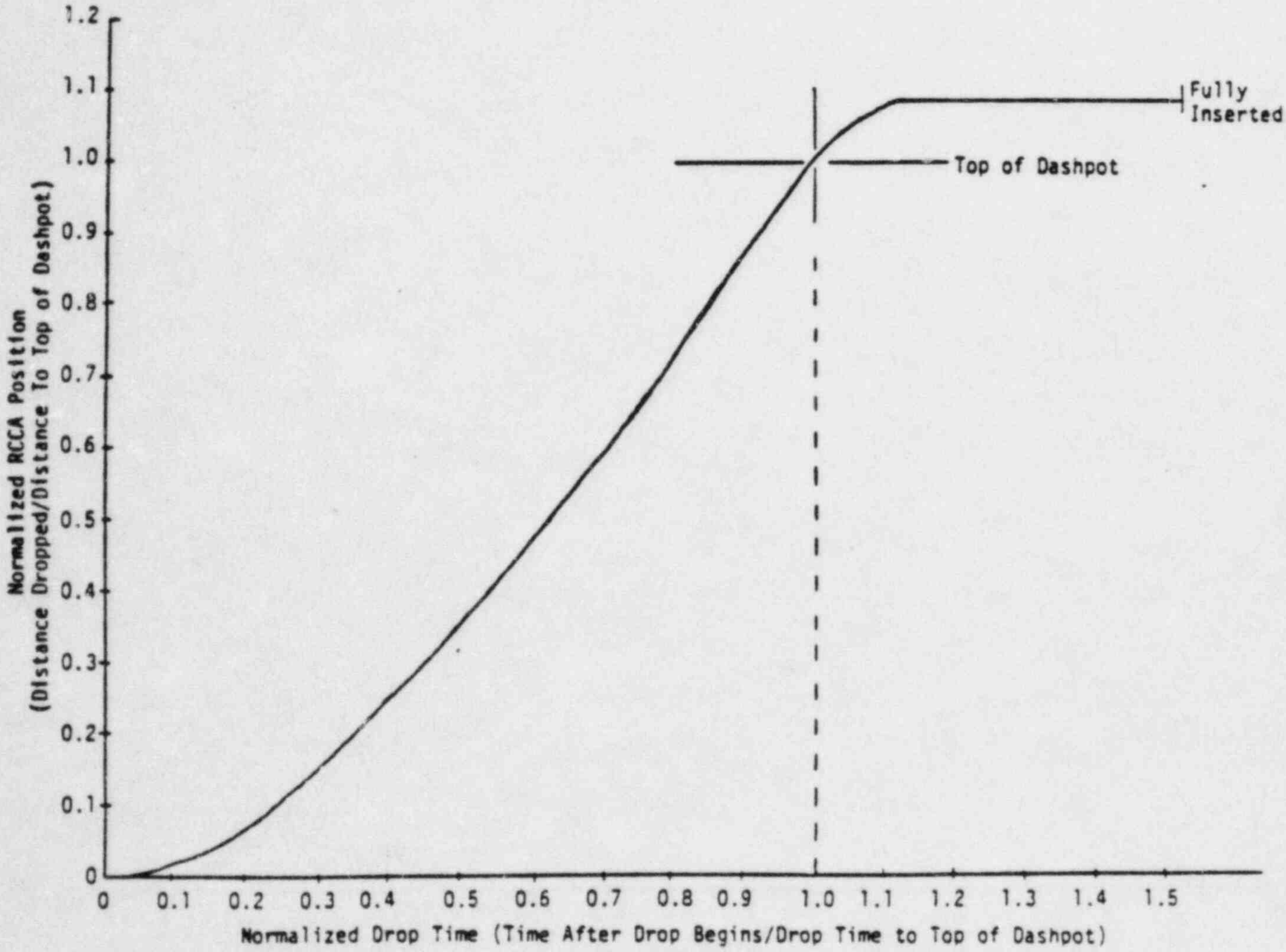


FIGURE 15.0-3 RCCA POSITION VS. TIME TO DASHPOT

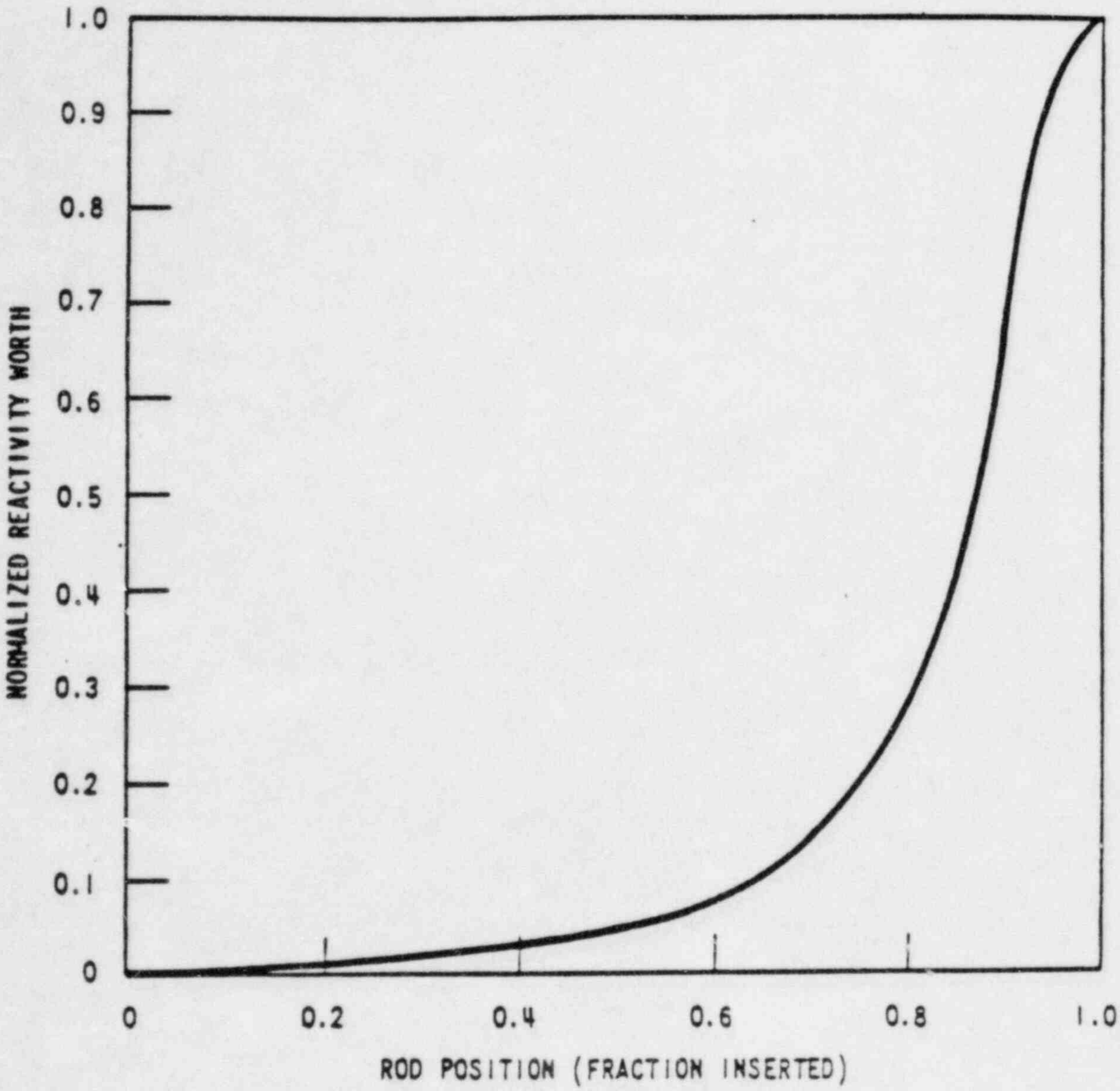


FIGURE 15.0-4 NORMALIZED RCCA REACTIVITY WORTH VS. FRACTION INSERTION

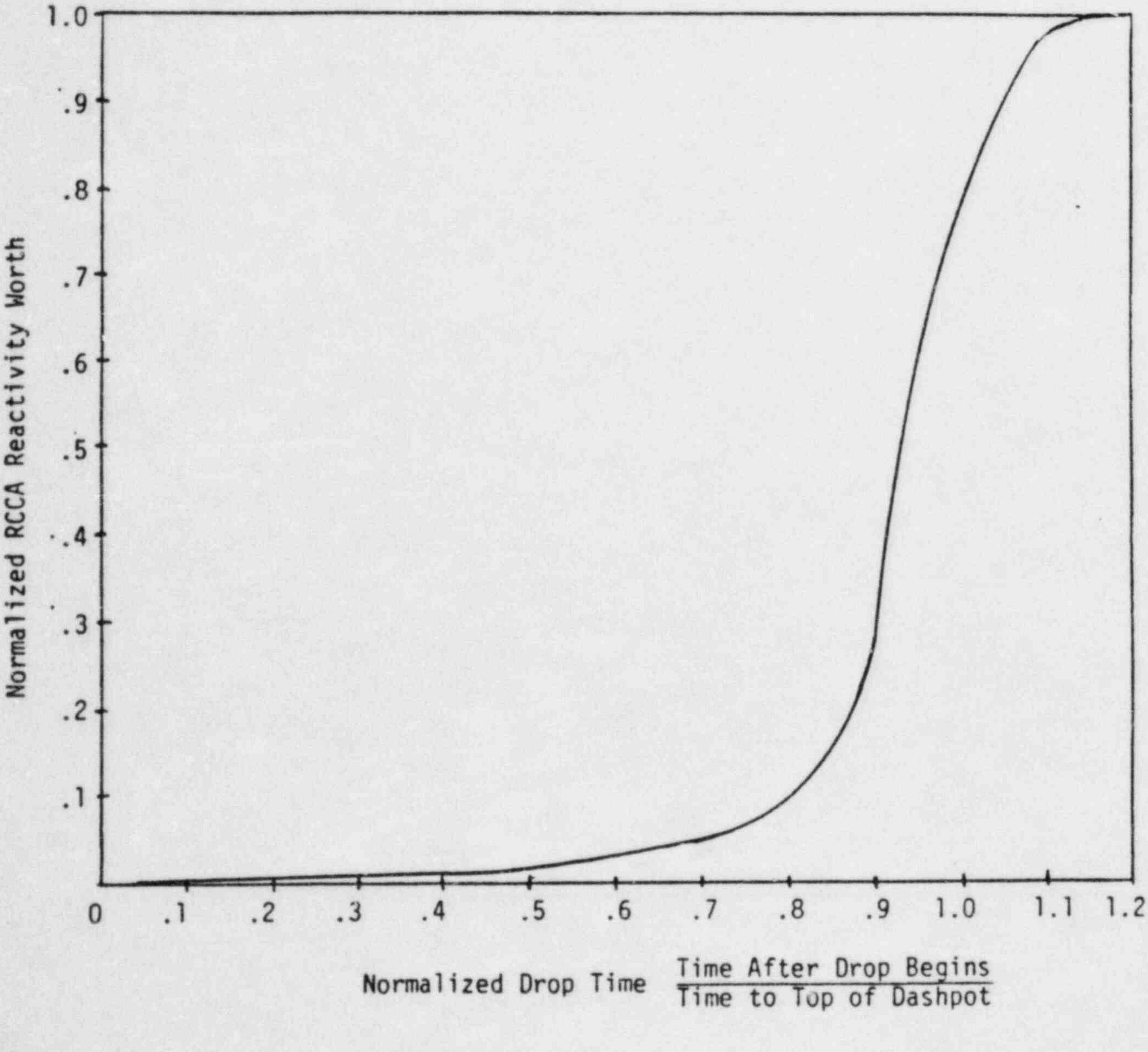


FIGURE 15.0-5 NORMALIZED RCCA BANK REACTIVITY WORTH VS. NORMALIZED DROP TIME

15.3 DECREASE IN REACTOR COOLANT SYSTEM FLOWRATE

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

A. Partial loss of forced reactor coolant flow.

B. Complete loss of forced reactor coolant flow.

C. Reactor coolant pump shaft seizure (locked rotor).

D. Reactor coolant pump shaft break.

Item A above is considered to be an American Nuclear Society (ANS) Condition II event, item B an ANS Condition III event, and items C and D ANS Condition IV event.

All of the accidents in this section have been analyzed. It has been determined that the most severe radiological consequences will result from the reactor coolant pump shaft seizure accident discussed in Subsection 15.3.3. Therefore, doses are reported only for that limiting case.

15.3.1 Partial Loss of Forced Reactor Coolant Flow

15.3.1.1 Identification of Causes and Accident Description

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

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

A partial loss-of-coolant flow is classified as an ANS Condition II incident (a fault of moderate frequency), as defined in Subsection 15.0.1.

The necessary protection against this event is provided by the low primary coolant flow reactor trip signal, which is actuated in any reactor coolant loop by two out of four low-flow signals. Above permissive P8, low flow in any loop will actuate a reactor trip. Between approximately 10-percent power (permissive P7) and the power level corresponding to permissive P8, low flow in any two loops will actuate a reactor trip.

15.3.1.2 Analysis of Effects and Consequences

A. Method of Analysis

Partial loss of flow involving loss of two pumps with four loops in operation has been analyzed.

This transient is analyzed by three digital computer codes. First, the LOFTRAN code [1] is used to calculate the loop and core flow during the transient, the time of reactor trip based on the calculated flows, the nuclear power transient, and the primary system pressure and temperature transients. The FACTRAN code [2] is then used to calculate the heat flux transient based on the nuclear power and flow from LOFTRAN. Finally, the THINC code (Section 4.4 of RESAR-SP/90 PDA Module 5, "Reactor System") is used to calculate the departure from

nucleate boiling ratio (DNBR) during the transient, based on the heat flux from FACTRAN and the flow from LOFTRAN. The DNBR transients presented represent the minimum of the typical cell, small thimble cell, large thimble cell, corner cell or the side cell.

B. Initial Conditions

Plant characteristics and initial conditions are discussed in Subsection 15.0.4. Initial operating conditions for this event are assumed at values consistent with normal steady state full power. This accident is analyzed with the improved thermal design procedure as described in WCAP 8567.

C. Reactivity Coefficients

A conservatively large absolute value of the Doppler-only power coefficient is used. (See Figure 15.0-2.) This is equivalent to a total integrated Doppler reactivity from 0-to 100-percent power of 0.0149 Δk.

The least negative moderator temperature coefficient (minimum moderator density coefficient) is assumed, since this results in the maximum core power during the initial part of the transient, when the minimum DNBR is reached. (See Figure 15.0-1.)

For these analyses, the curve of trip reactivity insertion versus time (Figure 15.0-5) was used.

D. Flow Coastdown

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

Plant systems and equipment which are necessary to mitigate the effects of the accident are discussed in 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.

E. Results

Figures 15.3-1 through 15.3-4 show the transient response for the loss of two reactor coolant pumps with four loops in operation. Figure 15.3-4 shows the DNBR to be always greater than the limit value.

The plant is tripped by the low-flow trip rapidly enough to ensure that the ability of the reactor coolant to remove heat from the fuel rod is not greatly reduced. Thus, the average fuel and clad temperatures do not increase significantly above their respective initial values.

The calculated sequence of events for the case analyzed is shown on Table 15.3-1. The affected reactor coolant pumps will continue to coast down, and the core flow will reach a new equilibrium value.

With the reactor tripped, a stable plant condition will eventually be attained. Normal plant shutdown may then proceed.

15.3.1.3 Conclusions

The analysis shows that the DNBR will not decrease below the limit value at any time during the transient. The DNBR design basis is described in Section 4.4 of RESAR-SP/90 PDA Module 5, "Reactor System". All applicable acceptance criteria are met.

15.3.2 Complete Loss of Forced Reactor Coolant Flow

15.3.2.1 Identification of Causes and Accident Description

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

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

A complete loss-of-flow accident is classified as an American Nuclear Society Condition III event (an infrequent fault), as defined in Subsection 15.0.1. The following signals provide protection against this event:

- Low reactor coolant pump speed reactor trip.
- Low reactor coolant loop flow reactor trip.

The reactor trip on reactor coolant pump low speed is provided to protect against conditions which can cause a loss of voltage to all reactor coolant pumps, i.e., station blackout. This function is blocked below approximately 10-percent power (permissive P7).

The reactor trip on reactor coolant pump underfrequency is provided to trip the reactor for an underfrequency condition, resulting from frequency disturbances on the power grid. If the maximum grid frequency decay rate is less than approximately 5 HZ/SEC, this trip will protect the core from under-frequency events. Reference 3 provides analyses of grid frequency disturbances.

The reactor trip on low primary coolant loop flow is provided to protect against loss-of-flow conditions which affect only one reactor coolant loop. This function is generated by two out of four low flow signals per reactor coolant loop. Above permissive P8, low flow in any loop will actuate a reactor trip. Between approximately 10-percent power (permissive P7) and the power level corresponding to permissive P8, low flow in any two loops will actuate a reactor trip. If the maximum grid frequency decay rate is low enough, this trip function will protect the core from underfrequency events. This effect is fully described in reference 3.

15.3.2.2 Analysis of Effects and Consequences

A. Method of Analysis

The complete loss-of-flow transient has been analyzed for a loss of four pumps with four loops in operation.

The method of analysis and the assumptions made regarding initial operating conditions and reactivity coefficients are identical to those discussed in Subsection 15.3.1.2, except that following the loss of power supply to all pumps at power, a reactor trip is actuated by reactor coolant pump low speed.

B. Results

Figures 15.3-5 through 15.3-8 show the transient response for the loss of power to all reactor coolant pumps with four loops in operation. The reactor is assumed to be tripped on a low reactor coolant pump

speed signal. Figure 15.3-8 shows the departure from nucleate boiling ratio (DNBR) to be always greater than the limit value.

The plant is tripped by the low reactor coolant pump speed trip sufficiently fast to ensure that the ability of the primary coolant to remove heat from the fuel rod is not greatly reduced. Thus, the average fuel and clad temperatures do not increase significantly above their respective initial values.

The calculated sequence of events for the case analyzed is shown on Table 15.3-1. The reactor coolant pumps will continue to coastdown, and natural circulation flow will eventually be established, as demonstrated in Subsection 15.2.6 of RESAR-SP/90 PDA Module 6, "Secondary Side Safeguards System". With the reactor tripped, a stable plant condition will be attained. Normal plant shutdown may then proceed.

15.3.2.3 Conclusions

The analysis performed has demonstrated that, for the complete loss of forced reactor coolant flow, the DNBR does not decrease below the limit value at any time during the transient. The design basis for the DNBR is described in Section 4.4 of RESAR-SP/90 PDA Module 5, "Reactor System". All applicable acceptance criteria are met.

15.3.3 Reactor Coolant Pump Shaft Seizure (Locked Rotor)

15.3.3.1 Identification of Causes and Accident Description

The accident postulated is an instantaneous seizure of a reactor coolant pump rotor, as discussed in Section 5.4. Flow through the affected reactor coolant loop is rapidly reduced, leading to an initiation of a reactor trip on a low-flow signal.

Following initiation of the reactor trip, heat stored in the fuel rods continues to be transferred to the coolant, causing the coolant to expand. At the same time, heat transfer to the shell side of the steam generators is reduced: first, because the reduced flow results in a decreased tube side film coefficient; second, because the reactor coolant in the tubes cools down while the shell-side temperature increases. (Turbine steam flow is reduced to zero upon plant trip.) The rapid expansion of the coolant in the reactor core, combined with reduced heat transfer in the steam generators, causes an insurge into the pressurizer and a pressure increase throughout the reactor coolant system. (RCS). The insurge into the pressurizer compresses the steam volume, actuates the automatic spray system, opens the power-operated relief valves, and opens the pressurizer safety valves, in that sequence. The two power-operated relief valves are safety grade and would be expected to function properly during the accident. However, for conservatism, their pressure-reducing effect, as well as the pressure-reducing effect of the spray, is not included in the analysis.

This event is classified as an American Nuclear Society Condition IV incident (a limiting fault), as defined in Subsection 15.0.2.

15.3.3.2 Analysis of Effects and Consequences

15.3.3.2.1 Method of Analysis

Two digital computer codes are used to analyze this transient. The LOFTRAN code[1] is used to calculate the resulting loop and core flow transients following the pump seizure, the time of reactor trip based on the loop flow transients, and the nuclear power following reactor trip and to determine the peak pressure. The thermal behavior of the fuel located at the core hot spot is investigated by using the FACTRAN code,[2] which uses the core flow and the nuclear power calculated by LOFTRAN.

The FACTRAN code includes a film boiling heat transfer coefficient.

At the beginning of the postulated locked rotor accident (at the time the shaft in one of the reactor coolant pumps is assumed to seize), the plant is assumed to be in operation under the most adverse steady-state operating conditions, i.e., maximum guaranteed steady-state thermal power, maximum steady-state pressure, and maximum steady-state coolant average temperature. Plant characteristics and initial conditions are further discussed in Subsection 15.0.4. The accident is evaluated with and without offsite power available.

For the case without offsite power available, power is lost to the unaffected pumps 0. seconds after reactor trip.

For the peak pressure evaluation, the initial pressure is conservatively estimated as 30 psi above nominal pressure (2250 psia) to allow for errors in the pressurizer pressure measurement and control channels. This is done to obtain the highest possible rise in the coolant pressure during the transient. To obtain the maximum pressure in the primary side, conservatively high loop pressure drops are added to the calculated pressurizer pressure.

15.3.3.2.2 Evaluation of the Pressure Transient

After pump seizure, the neutron flux is rapidly reduced by control rod insertion. Rod motion is assumed to begin 1 second after the flow in the affected loop reaches 87 percent of nominal flow. No credit is taken for the pressure-reducing effect of the pressurizer relief valves, pressurizer spray, steam dump, or controlled feedwater flow after plant trip. Although these operations are expected to occur and would result in a lower peak pressure, an additional degree of conservatism is provided by ignoring their effect.

The pressurizer safety valves are fully open at 2575 psia, and their capacity for steam relief is described in Section 5.4.

15.3.3.2.3 Evaluation of Departure from Nucleate Boiling (DNB) in the Core During the Accident

For this accident, DNB is calculated to occur in the core; therefore, an evaluation of the consequences with respect to fuel rod thermal transients is performed. Results obtained from analysis of this "hot spot" condition represent the upper limit with respect to clad temperature and zirconium-water reaction. Although DNB is predicted to occur, no fuel rod failures are postulated as discussed in Subsection 15.3.3.3 and reference 4.

In the evaluation, the rod power at the hot spot is conservatively assumed to (a,c) be [ ] times the average rod power (i.e., F_Q = [ ]) at the initial core power level.

15.3.3.2.4 Film Boiling Coefficient

The film boiling coefficient is calculated in the FACTRAN code(2) using the Bishop-Sandberg-Tong film-boiling correlation. The fluid properties are evaluated at film temperature (average between wall and bulk temperatures). The program calculates the film coefficient at every time step, based upon the actual heat transfer conditions at the time. The nuclear power, system pressure, bulk density, and mass flowrate as a function of time are used as program input.

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

15.3.3.2.5 Fuel Clad Gap Coefficient

The magnitude and time dependence of the heat transfer coefficient between fuel and clad (gap coefficient) have a pronounced influence on the thermal results. The larger the value of the gap coefficient, the more heat is transferred between pellet and clad. Based on investigations on the effect of the gap coefficient upon the maximum clad temperature during the transient,

the gap coefficient was assumed to increase from a steady-state value consistent with initial fuel temperature to 10,000 Btu/h-ft^2-°F at the initiation of the transient. Thus, the large amount of energy stored in the fuel because of the small initial value of the gap coefficient is released to the clad at the initiation of the transient.

15.3.3.2.6 Zirconium-Steam Reaction

The zirconium-steam reaction can become significant above 1800°F (clad temperature). The Baker-Just parabolic rate equation shown below is used to define the rate of the zirconium-steam reaction.

$$\frac{d\,(w^2)}{dt} = 33.3 \times 10^6 \exp \quad - \frac{45,500}{1.986T}$$

where:

W = amount reacted (mg/cm^2).

t = time (s).

T = temperature (°F).

The reaction heat is 1510 cal/g.

The effect of zirconium-steam reaction is included in the calulation of the hot spot clad temperature transient.

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 affect the consequences of the accident.

15.3.3.2.7 Results

Figures 15.3-9 through 15.3-12 show the transient results for one locked rotor with four loops in operation (with and without offsite power available). The results of these calculations are also summarized in Table 15.3-1. The peak RCS pressure reached during the transient is less than that which would cause stresses to exceed the faulted condition stress limits of the American Society of Mechanical Engineers Code, Section III. Also, the peak clad surface temperature is considerably less than 2700°F. It should be noted that the clad temperature was conservatively calculated, assuming that DNB occurs at the initiation of the transient. These results represent the most limiting conditions with respect to the locked rotor event or the pump shaft break.

The calculated sequence of events for the case analyzed is shown in Table 15.3-1. Figure 15.3-9 shows that the core flow rapidly reaches a new equilibrium value (for the case with offsite power available). With the reactor tripped, a stable plant condition will eventually be attained. Normal plant shutdown may then proceed.

15.3.3.3 Radiological Consequences

The evaluation of the radiological consequences of a postulated seizure of a reactor coolant pump rotor, i.e., locked rotor accident (LRA), assumes that the reactor has been operating with a small percent of defective fuel and leaking steam generator tubes for sufficient time to establish equilibrium concentrations of radionuclides in the reactor coolant and in the secondary coolant.

As a result of the accident, a fraction of the fuel rods will undergo DNB. However, no fuel clad failure is predicted [4]. Radionuclides carried by the primary coolant to the steam generator via leaking tubes are released to the environment via the steam line safety or power-operated relief valves.

15.3.3.3.1 Analytical Assumptions

The major assumptions and parameters used in the analysis are itemized in Table 15.3-3. The following is a more detailed discussion of the source term.

15.3.3.3.1.1 Source Term Calculations

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

A. The iodine concentrations in the reactor coolant will be based upon accident intiated and preaccident iodine spikes.

   1. Accident Initiated Spike - The initial primary coolant iodine concentration is 1μ Ci/gm of Dose Equivalent (D.E.) I-131. Following the reactor trip, associated with the LRA, an iodine spike is created in the primary system which increases the iodine release rate from the fuel to the primary coolant to a valve 500 times greater than the release rate corresponding to the equilibrium primary system iodine concentration of 1μ Ci/gm of D.E. I-131. The duration of the spike, 2.5 hours, is sufficient to increase the initial primary system I-131 inventory by a factor of 100.

   2. Pre accident Spike - A reactor transient has occurred prior to the LRA and has raised the primary coolant iodine concentration from 1μ Ci/gm to 60 μ Ci/gm of D.E. I-131.

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

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

15.3.3.3.1.2 General Parameters Used in the Analysis

The general parameters and mathematical models used in the analysis are described in Appendix 15A.

15.3.3.3.1.3 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. Noble gases are assumed to be directly released.

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 intergrated dose due to the postulated LRA.

15.3.3.3.2 Identification of Uncertainties and Conservative Elements in the Analysis

A. The initial reactor coolant iodine concentrations are based on the Technical Specification limit of 1.0μ Ci/gm of D.E. I-131 with large iodine spike values, resulting in equivalent concentrations many times greater than the concentrations found in typical operating plants.

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

D. 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.

15.3.3.3.3 Conclusions

15.3.3.3.3.1 Filter Loadings

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

Integrated activity on the control room filters have been evaluated for the more limiting loss-of-coolant accident (LOCA) analysis, as discussed in Subsection 15.6.5.4.6 of RESAR-SP/90 PDA Module 1, "Primary Side Safeguards System". 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 LRA.

15.3.3.3.3.2 Doses to Receptor at the Exclusion Area Boundary and Low Population Zone Outer Boundary

The potential radiological consequences resulting from the occurrence of a postulated LRA have been conservatively analyzed using assumptions and models described. The total-body gamma dose due to inhalation have been analyzed for the 0- to 2-h dose at the exclusion area boundary and for the duration of the

accident (0 to 8 h) at the low population zone outer boundary. The results are listed in Table 15.3-4. The resultant doses are well within the guideline values of 10 CFR 100.

15.3.4 Reactor Coolant Pump Shaft Break

15.3.4.1 Identification of Causes and Accident Description

The accident is postulated as an instantaneous failure of a reactor coolant pump shaft, as discussed in Section 5.4. Flow through the affected reactor coolant loop is rapidly reduced, though the initial rate of reduction of coolant flow is greater for the reactor coolant pump rotor seizure event. Reactor trip is initiated on a low-flow signal in the affected loop.

Following initiation of the reactor trip, heat stored in the fuel rods continues to be transferred to the coolant, causing the coolant to expand. At the same time, heat transfer to the shell side of the steam generators is reduced-first, because the reduced flow results in a decreased tube-side film coefficient; second, because the reactor coolant in the tubes cools down while the shell-side temperature increases. (Turbine steam flow is reduced to zero upon plant trip.) The rapid expansion of the coolant in the reactor core, combined with reduced heat transfer in the steam generators, causes an insurge into the pressurizer and a pressure increase throughout the reactor coolant system. The insurge into the pressurizer compresses the steam volume, actuates the automatic spray system, opens the pressurizer safety valves, in that sequence. The two power-operated relief valves are designed for reliable operation and would be expected to function properly during the accident, However, for conservatism, their pressure-reducing effect, as well as the pressure-reducing effect of the spray, is not included in the analysis.

This event is classified as an American Nuclear Society Condition IV incident (limiting fault), as defined in Subsection 15.0.1.

15.3.4.2 Conclusion

The consequences of a reactor coolant pump shaft break are similar to those calculated for the locked rotor (see Subsection 15.3.3). The bounding results for the locked rotor transients presented in Figures 15.3-9 thru 15.3-12 and summarized in Table 15.3-2 are also applicable for the reactor coolant pump shaft break. With a failed shaft, the impeller could conceivably be free to spin in a reverse direction as opposed to being fixed in position as assumed in the locked-rotor analysis. However, the net effect on core flow is negligible, resulting in only a slight decrease in the end point (steady-state) core flow. For both the shaft break and locked-rotor incidents, reactor trip occurs very early in the transient. In addition, the locked-rotor analysis conservatively assumes that departure from nucleate boiling occurs at the beginning of the transient.

15.3.5 REFERENCES

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

2. Hargrove, H. G., "FACTRAN - A FORTRAN-IV Code for Thermal Transients in a UO_2 Fuel Rod," WCAP-7908, June, 1972.

3. Baldwin, M. S., et al., "An Evaluation of Loss of Flow Accidents Caused by Power System Frequency Transients in Westinghouse PWRs," WCAP-8424, Revision 1, May, 1975.

4. Van Houten, R., "Fuel Rod Failure as a Consequence of Departure from Nucleate Boiling or Dryout", NUREG-0562, June 1979.

TABLE 15.3-1 (Sheet 1 of 2)

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH RESULT IN A DECREASE IN REACTOR COOLANT SYSTEM FLOWRATE

Accident	Event	Time (s)
Partial loss of forced reactor coolant flow		
Loss of two pumps with four loops in operation	Coastdown begins	0.0
	Low-flow reactor trip	1.56
	Rods begin to drop	2.56
	Minimum DNBR occurs	4.7
Complete loss of forced reactor coolant flow		
Loss of four pumps with four loops in operation	All operating pumps lose power and begin coasting down	0.0
	Reactor coolant pump low speed reactor trip setpoint reached	1.11
	Rods begin to drop	1.71
	Minimum DNBR occurs	4.4
Reactor coolant pump shaft seizure (locked rotor)		
One locked rotor with four loops in operation with offsite power available	Rotor on one pump locks	0.0
	Low-flow trip point reached	0.06
	Rods begin to drop	1.06
	Maximum clad temperature occurs	4.1
	Maximum reactor coolant system pressure occur	4.1

TABLE 15.3-1 (Sheet 2 of 2)

Accident	Event	Time (s)
One locked rotor with four loops in operation without offsite power available	Rotor on one pump locks	0.0
	Low-flow trip point reached	0.6
	Rods begin to drop	1.06
	Maximum clad temperature occurs	4.3
	Maximum reactor coolant system pressure occurs	4.3

TABLE 15.3-2

SUMMARY OF RESULTS FOR LOCKED ROTOR TRANSIENTS

(FOUR LOOPS OPERATING INITIALLY)

	With Offsite Power Available	Without Offsite Power Available
Maximum RCS pressure (psia)	2561	2642
Maximum clad temperature, core hot spot (°F)	1828	1907
Zr-H_2O reaction, core hot spot (percent by weight)	.3	.4

TABLE 15.3-3 (Sheet 1 of 2)

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A LOCKED ROTOR ACCIDENT

I.	Source Data		
	A.	Core power level (MWt)	3565
	B.	Total steam generator tube leakage (gal/min)	1
	C.	Reactor coolant iodine activity:	
		1. Accident Initiated Spike	Initial activity equal to the dose equivalent of 1.0μ Ci/gm of I-131 with an assumed iodine release into the reactor coolant by a factor of 500. See Table 15.0-10
		2. Pre-Accident Spike	An assumed pre-accident iodine spike, which has resulted in the dose equivalent of 60μ Ci/gm of I-131 in the reactor coolant. See Table 15.0-8
	D.	Reactor coolant noble gas activity	Based on 1-percent defective fuel. See Table 15.0-9.
	E.	Secondary system initial activity	Based on 0.1 μCi/g of D.E. I-131.
	F.	Reactor coolant mass (g)	2.3×10^8
	G.	Secondary coolant mass, 4 generators (g)	1.9×10^8
	H.	Offsite power	Lost after trip
	I.	Primary-to-secondary leakage duration (h)	8
	J.	Species of iodine	100-percent elemental
II.	Atmospheric Dispersion Factors		See Table 15A-2

TABLE 15.3-3 (Sheet 2 of 2)

III. Activity Release Data

A. Primary-to-secondary leakrate (gal/min))(a) — 1.0

B. Steam Released (lb)

0 to 2 h	555,000
2 to 8 h	1,365,000

C. Iodine partition factor — 100

IV. Activity Released to the Environment

a. Accident Initiated Spike

Isotope	0-2 h (Ci)	2-8 h (Ci)
I-131	0.22	1.64
I-132	0.25	2.2
I-133	0.34	2.9
I-134	0.07	0.3
I-135	0.19	1.8

b. Pre-Accident Spike

Isotope	0-2 h (Ci)	2-8 h (Ci)
I-131	0.28	1.58
I-132	0.21	0.36
I-133	0.41	2.0
I-134	0.03	0.01
I-135	0.21	0.77

c. Noble gases - both cases

Isotope	0-2 h (Ci)	2-8 h (Ci)
Xe-131m	0.5	2.1
Xe-133m	5.5	16.0
Xe-133	87.0	254.0
Xe-135m	0.03	1.0 x 10-4
Xe-135	2.0	5.0
Xe-138	0.04	3 x 10-4
Kr-85m	0.4	0.9
Kr-85	1.8	7.0
Kr-87	0.2	0.1
Kr-88	0.6	1.1

TABLE 15.3-4

RADIOLOGICAL CONSEQUENCES OF A LOCKED ROTOR ACCIDENT

Case 1. Accident Initiated Iodine Spike	
Exclusion area boundary (0-2 h) Thyroid (rem)	0.027
Low-population outer boundary (8 h) Thyroid (rem)	0.05
Case 2. Pre-Accident Iodine Spike	
Exclusion area boundary (0-2 h) Thyroid (rem)	0.034
Low-population zone outer boundary (8h) Thyroid (rem)	0.047
Both Cases: Whole Body Gamma (rem)	
Exclusion area boundary (0-2 h)	2.7 x 10-4
Low-population zone outer boundary (8 h)	2.2 x 10-4

(a,c)

CORE FLOW (FRACTION OF INITIAL)

LOOP 1 FLOW (FRACTION OF NOMINAL)

TIME (SEC)

FIGURE 15.3-1

FLOW TRANSIENTS FOR 4 LOOPS IN OPERATION,
2 PUMPS COASTING DOWN

(a,c)

PRESSURIZER PRESSURE (PSIA) NUCLEAR POWER (FRACTION OF NOMINAL)

TIME (SEC)

FIGURE 15.3-2

NUCLEAR POWER AND PRESSURIZER PRESSURE TRANSIENTS FOR 4 LOOPS IN OPERATION, 2 PUMPS COASTING DOWN

(a,c)

HEAT FLUX (FRACTION OF NOMINAL)

HEAT FLUX (FRACTION OF NOMINAL)

TIME (SEC)

FIGURE 15.3-3

AVERAGE AND HOT CHANNEL HEAT FLUX TRANSIENTS FOR 4 LOOPS IN OPERATION, 2 PUMPS COASTING DOWN

FIGURE 15.3-4

DNBR VERSUS TIME FOR 4 LOOPS IN OPERATION, 2 PUMPS COASTING DOWN

(a,c)

CORE FLOW (FRACTION OF INITIAL)

LOOP 1 FLOW (FRACTION OF NOMINAL)

TIME (SEC)

FIGURE 15.3-5

FLOW TRANSIENTS FOR 4 LOOPS IN OPERATION, 4 PUMPS COASTING DOWN

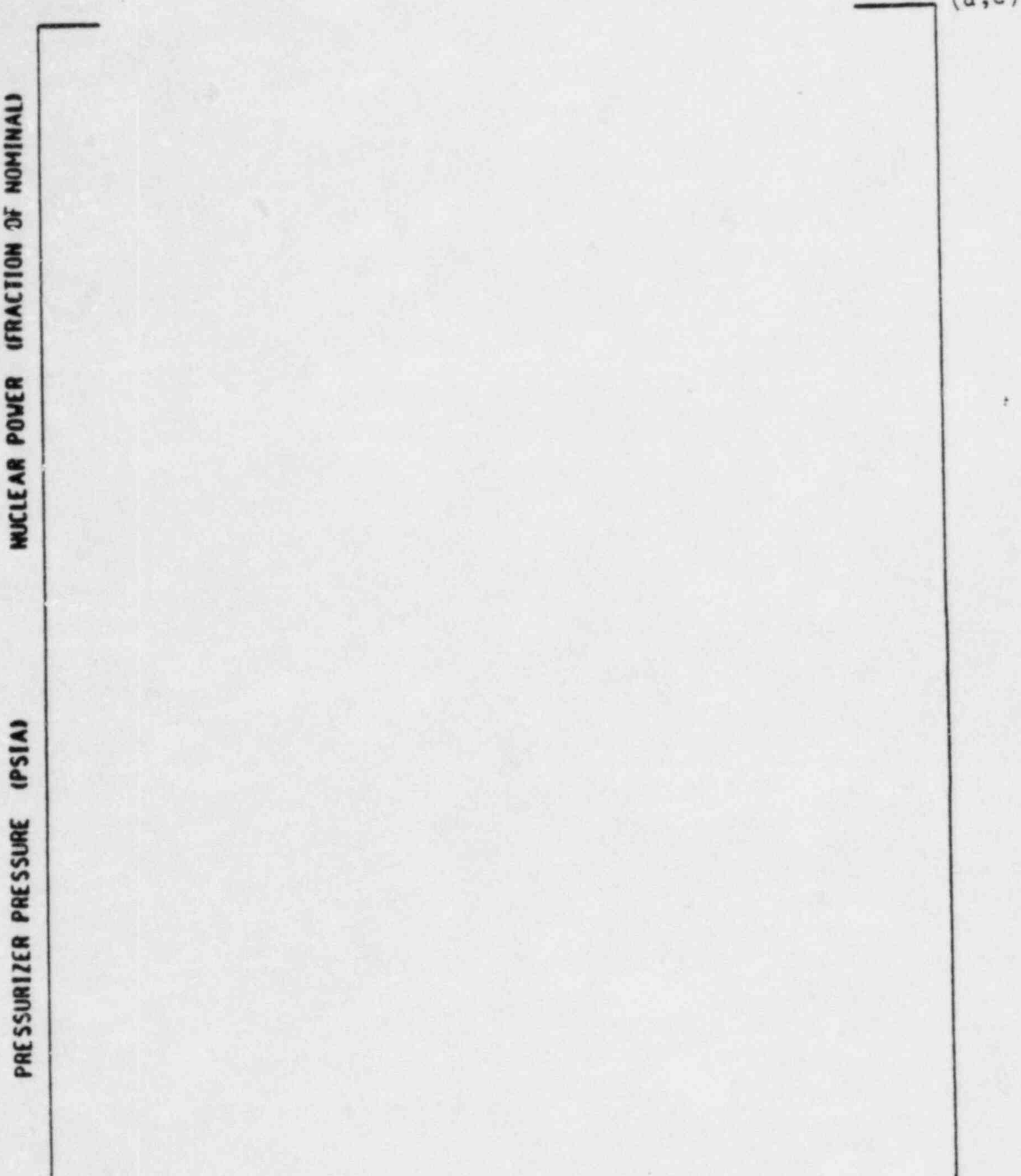


FIGURE 15.3-6

NUCLEAR POWER AND PRESSURIZER PRESSURE TRANSIENTS FOR 4 LOOPS IN OPERATION, 4 PUMPS COASTING DOWN

(a,c)

HEAT FLUX (FRACTION OF NOMINAL)

HEAT FLUX (FRACTION OF NOMINAL)

TIME (SEC)

FIGURE 15.3-7

AVERAGE AND HOT CHANNEL HEAT FLUX TRANSIENTS FOR 4 LOOPS IN OPERATION, 4 PUMPS COASTING DOWN

FIGURE 15.3-8
DNBR VS. TIME FOR 4 LOOPS IN OPERATION, 4 PUMPS COASTING DOWN

(a,c)

CORE FLOW (FRAC. OF NOMINAL)

LOOP 1 FLOW (FRAC. OF NOMINAL)

TIME (SEC)

FIGURE 15.3-9

FLOW TRANSIENTS FOR 4 LOOPS IN OPERATION, 1 LOCKED ROTOR

(a,c)

PRESSURE

TIME

FIGURE 15.3-10

PEAK REACTOR COOLANT PRESSURE FOR 4 LOOPS IN OPERATION, 1 LOCKED ROTOR

(a,c)

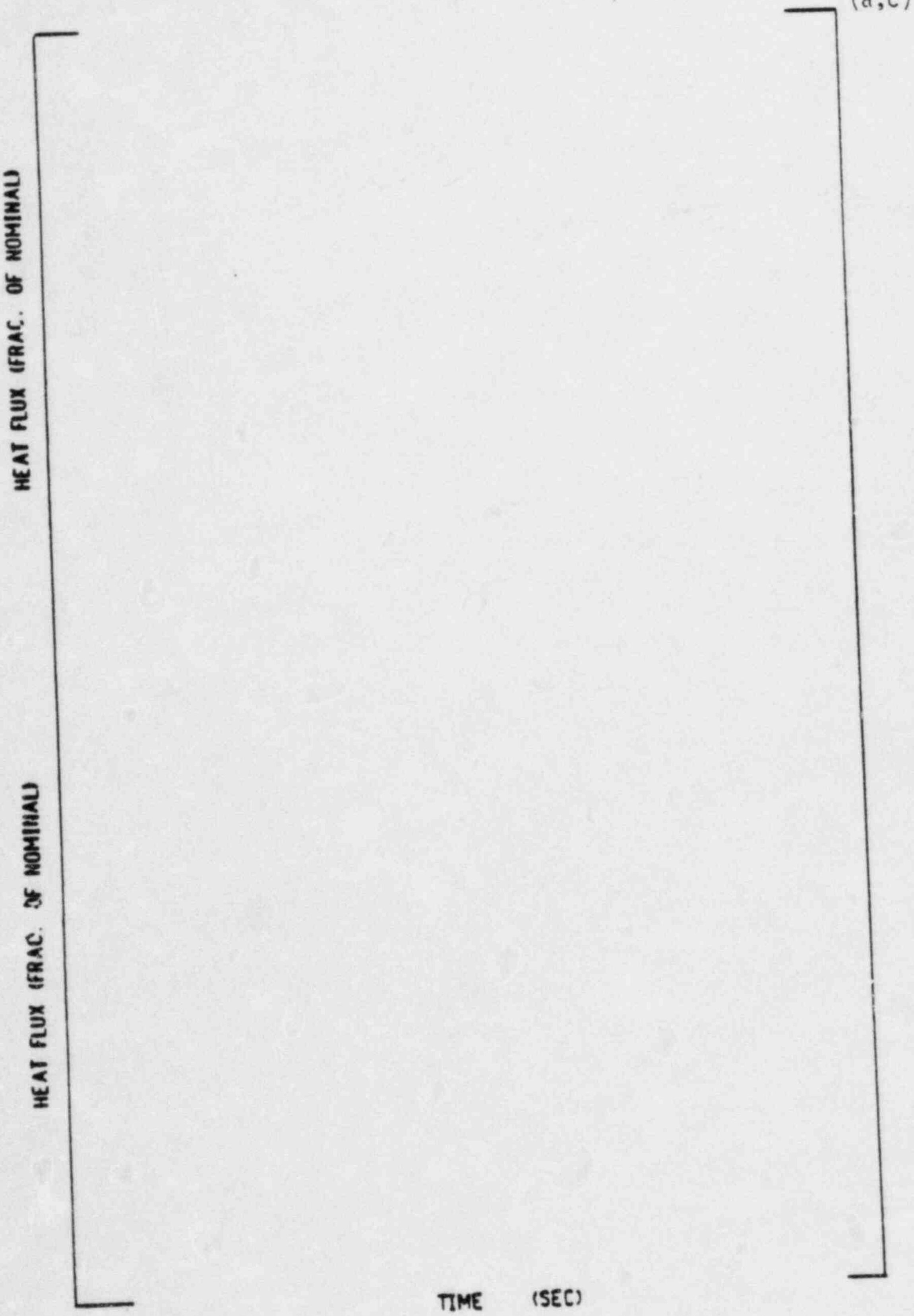


FIGURE 15.3-11

AVERAGE AND HOT CHANNEL HEAT FLUX TRANSIENTS FOR 4 LOOPS IN OPERATION, 1 LOCKED ROTOR

(a,c)

NUCLEAR POWER (FRAC. OF NOMINAL)

CLAD INNER TEMP (DEG. F)

TIME (SEC)

FIGURE 15.3-12

NUCLEAR POWER AND MAXIMUM CLAD TEMPERATURE AT HOT SPOT TRANSIENTS FOR 4 LOOPS IN OPERATION, 1 LOCKED ROTOR

15.4 REACTIVITY AND POWER DISTRIBUTION ANOMALIES

15.4.4 Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature

15.4.4.1 Identification of Causes and Accident Description

If the plant is operating with one reactor coolant pump out of service, there is reverse flow through the inactive loop due to the pressure difference across the reactor vessel. The cold leg temperature in an inactive loop is identical to the cold leg temperature of the active loops (the reactor core inlet temperature). If the reactor is operated at power and assuming the secondary side of the steam generator in the inactive loop is not isolated, there is a temperature drop across the steam generator in the inactive loop and, with the reverse flow, the hot leg temperature of the inactive loop is lower than the reactor core inlet temperature.

Administrative procedures require that the unit be brought to a load of less than 25 percent of full power prior to starting the pump in an inactive loop in order to bring the inactive loop hot leg temperature closer to the core inlet temperature. Starting an idle reactor coolant pump without bringing the inactive loop hot leg temperature close to the core inlet temperature would result in the injection of cold water into the core, which would cause a reactivity insertion and subsequent power increase.

This event is classified as an American Nuclear Society Condition II incident (a fault of moderate frequency) as defined in Subsection 15.0.1.

If the startup of an inactive reactor coolant pump accident occurs, the transient is terminated automatically by a reactor trip on low coolant loop flow when the power range neutron flux (two out of four channels) exceeds the P-8 setpoint, which has been previously reset for three-loop operation.

15.4.4.2 Analysis of Effects and Consequences

15.4.4.2.1 Method of Analysis

This transient is analyzed using three digital computer codes. The LOFTRAN code[1] is used to calculate the loop and core flow, nuclear power, and core pressure and temperature transients following the startup of an idle pump. FACTRAN[2] is used to calculate the core heat flux transient based on core flow and nuclear power from LOFTRAN. The THINC code (Section 4.4 of RESAR-SP/90 PDA Module 5, "Reactor System") is then used to calculate the departure from nucleate boiling ratio (DNBR) during the transient based on system conditions (pressure, temperature, and flow) calculated by LOFTRAN and heat flux as calculated by FACTRAN.

Plant characteristics and initial conditions are discussed in Subsection 15.0.3. In order to obtain conservative results for the startup of an inactive pump accident, the following assumptions are made:

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

B. Following initiation of startup of the idle pump, the inactive loop flow reverses and accelerates to its nominal full-flow value in approximately 20 seconds.

C. A conservatively large negative moderator temperature coefficient is assumed.

D. A least negative Doppler-only power coefficient is used (See Figure 15.0-2).

E. The initial reactor coolant loop flows are at the appropriate values for one pump out of service.

F. The reactor trip is assumed to occur on low coolant loop flow when the power range neutron flux exceeds the P-8 setpoint. The P-8 setpoint is conservatively assumed to be 84 percent of rated power, which corresponds to the nominal setpoint plus 9 percent for nuclear instrumentation errors.

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 affect the consequences of the accident.

15.4.4.2.2 Results

The results following the startup of an idle pump with the above listed assumptions are shown in Figures 15.4-1 through 15.4-5. As shown in these curves, during the first part of the transient the increase in core flow with cooler water results in an increase in nuclear power and a decrease in core average temperature. The minimum DNBR during the transient is considerably greater than [ ] (See Section 4.4 of RESAR-SP/90 PDA Module 5, "Reactor System" for a description of the DNBR design basis.) (a,c)

Reactivity addition for the inactive loop startup accident is due to the decrease in core water temperature. During the transient this decrease is due both to the increase in reactor coolant flow and, as the inactive loop flow reverses, to the colder water entering the core from the hot leg side (colder temperature side prior to the start of the transient) of the steam generator in the inactive loop. Thus, the reactivity insertion rate for this transient changes with time. The resultant core nuclear power transient, computed with consideration of both moderator and Doppler reactivity feedback effects, is shown in Figure 15.4-1.

The calculated sequence of events for this accident is shown in Table 15.4-1. The transient results illustrated in Figures 15.4-1 through 15.4-5 indicate that a stabilized plant condition, with the reactor tripped, is rapidly approached. Plant cooldown may subsequently be achieved by following normal shutdown procedures.

15.4.4.3 Conclusions

The transient results show that the core is not adversely affected. There is considerable margin to the DNBR limit. Thus, the DNB design basis as described in Section 4.4 of RESAR-SP/90 PDA Module 5, "Reactor System" is met.

15.4.5 REFERENCES

1. Burnett, T. W. T., et al., "LOFTRAN Code Description," WCAP-7907, October 1972.

2. Hargrove, H. G., "FACTRAN-A FORTRAN-IV Code for Thermal Transients in UO_2 Fuel Rod," WCAP-7908, June 1972.

TABLE 15.4-1

Time Sequence of Events for Incidents which Cause Reactivity and Power Distribution Anomalies

Accident	Event	Time (sec)
Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature	Initiation of pump startup	0.0
	Power reaches p-8 setpoint	11.321
	Rods begin to drop	11.821
	Minimum DNBR occurs	13.0

(a,c)

NUCLEAR POWER (FRACTION OF NOMINAL)

TIME (SEC)

FIGURE 15.4-1 IMPROPER STARTUP OF AN INACTIVE REACTOR COOLANT PUMP

(a,

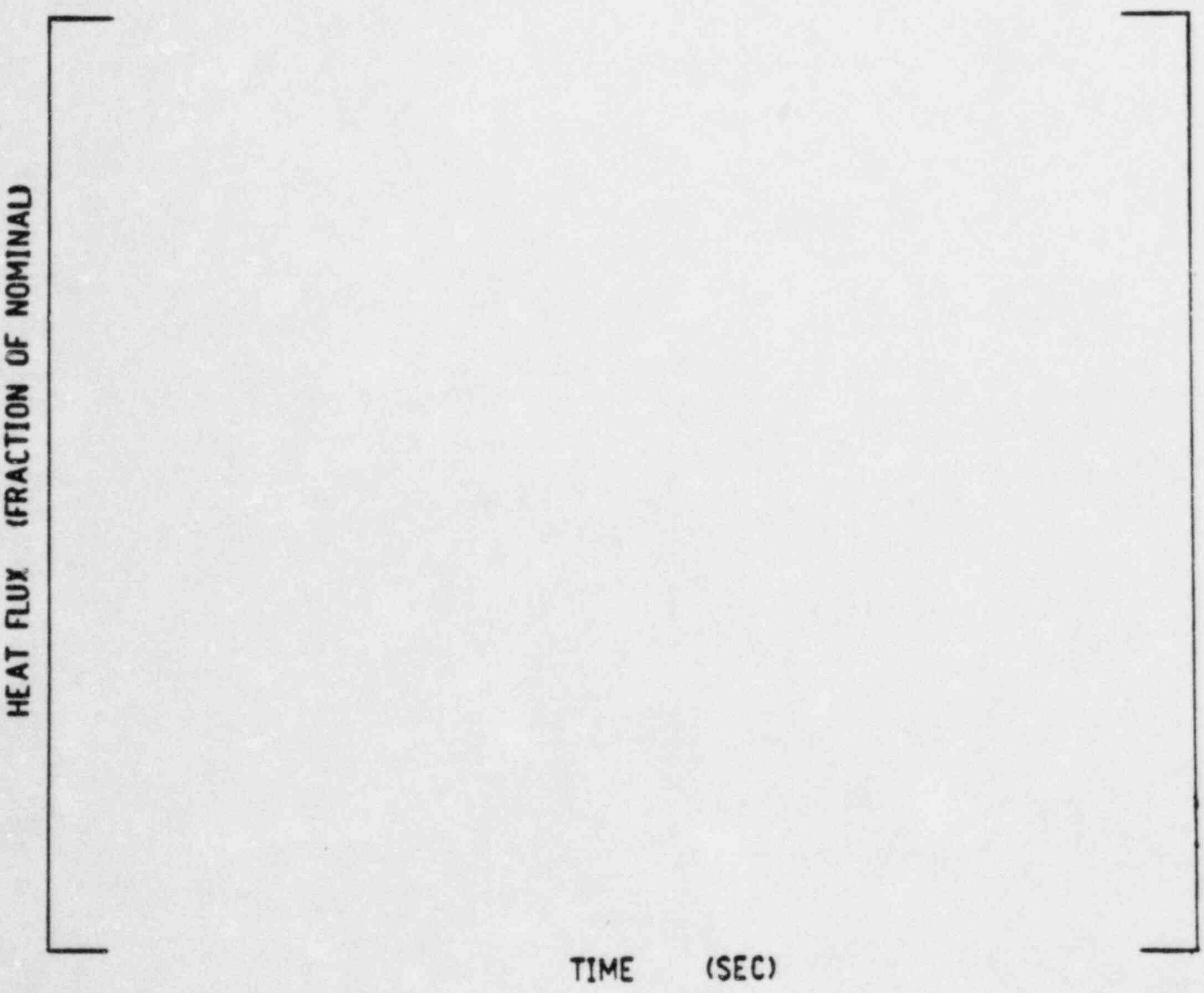


FIGURE 15.4-2 IMPROPER STARTUP OF AN INACTIVE REACTOR COOLANT PUMP

(a,

CORE FLOW (FRACTION OF INITIAL)

TIME (SEC)

FIGURE 15.4-3 IMRPOPER STARTUP OF AN INACTIVE REACTOR COOLANT PUMP

(a,c)

REACTOR COOLANT SYSTEM PRESSURE (PSIA)

CORE AVERAGE TEMPERATURE (DEG-F)

TIME (SEC)

FIGURE 15.4-4 IMPROPER STARTUP OF AN INACTIVE REACTOR COOLANT PUMP

(a,c)

DNBR

TIME (SECONDS)

FIGURE 15.4-5 IMPROPER STARTUP OF AN INACTIVE REACTOR COOLANT PUMP

15.5 INCREASE IN REACTOR COOLANT INVENTORY

Several events have been postulated which could cause an increase in reactor coolant inventory or a change in a reactor coolant boron concentration. Discussion of the following events is presented in this section:

1. Inadvertent operation of Emergency Core Cooling System (ECCS) during power operation.

2. Chemical and Volume Control System (CVCS) malfunction that increases reactor coolant inventory.

These event are considered to be American Nuclear Society (ANS) Condition II transients (see Subsection 15.0.1).

15.5.1 Inadvertent Operation of ECCS During Power Operation

15.5.1.1 Identification of Causes and Accident Description

Spurious Safety Injection System (SIS) operation at power could be caused by error or a false electrical actuation signal. A spurious signal may originate from any of the safety injection actuation channels as described in Section 7.3 of RESAR-SP/90 PDA Module 9, "I&C and Electric Power".

Following the actuation signal the safety injection pumps will start, and the valves isolating the high concentration boron source will open as described in Section 7.3 of RESAR-SP/90 PDA Module 9, "I&C and Electric Power". The safety injection pumps have a shutoff head of about 1800 psi and consequently provide no flow at normal Reactor Coolant System (RCS) pressure.

An SIS signal normally results in a reactor trip followed by a turbine trip. However, it cannot be assumed that any signal fault that actuates SIS will also produce a reactor trip. If a reactor trip is generated by the spurious SIS signal, the plant is automatically brought to the hot shutdown condition. Since the safety injection pumps have a shutoff head less than operating pressure in the RCS their actuation has no effect on the transient.

If the Reactor Protection System (RPS) does not produce an immediate trip as a result of the spurious SIS signal, the reactor will continue to operate at power. Since the safety injection pumps have a shutoff head of about 1800 psi they will not provide borated water following their actuation at normal RCS pressure.

15.5.1.2 Conclusions

Spurious safety injection without immediate reactor trip has no effect on the RCS.

If a reactor trip is generated by the spurious safety injection signal, a normal shutdown can be commenced without boration from the safety injection pumps because of the shutoff head of about 1800 psi.

15.5.2 Chemical and Volume Control System Malfunction That Increases Reactor Coolant Inventory

An increase in reactor coolant inventory which results from the addition of cold, unborated water to the reactor coolant system (RCS) is analyzed in Subsection 15.4.6 of RESAR-SP/90 PDA Module 13, "Auxiliary Systems". An increase in reactor coolant inventory which results from the injection of highly borated water into the RCS is analyzed in Subsection 15.5.1, Inadvertent Operation of the Emergency Core Cooling System During Power Operation.

15.6 DECREASE IN REACTOR COOLANT INVENTORY

Events which result in a decrease in reactor coolant inventory, as discussed in this section, are as follows:

A. Inadvertent opening of a pressurizer safety or relief valve.

B. Break in instrument line or other lines from reactor coolant pressure boundary that penetrate the containment.

C. Steam generator tube failure.

D. Spectrum of boiling water reactor (BWR) steam system piping failures outside of containment (not applicable to the WAPWR).

E. Loss-of-coolant accident (LOCA) resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary.

F. A number of BWR transients (not applicable to WAPWR).

All of the applicable accidents in this category have been analyzed. It has been determined that the most severe radiological consequences will result from the major LOCA of Subsection 15.6.5. Therefore, the LOCA is the design basis accident. The LOCA chemical and volume control system (CVCS) letdown line break outside the containment and the steam generator tube rupture accident have been analyzed radiologically. All other accidents in this section are bounded by these accidents.

15.6.1 Inadvertent Opening of a Pressurizer Safety or Relief Valve

15.6.1.1 Identification of Causes and Accident Description

An accidental depressurization of the reactor coolant system (RCS) could occur as a result of an inadvertent opening of a pressurizer relief or safety valve. Since a pressurizer safety valve is sized to relieve approximately

twice the steam flowrate of a relief valve and will therefore allow a much more rapid depressurization upon opening, the most severe core conditions resulting from an accidental depressurization of the RCS are associated with an inadvertent opening of a pressurizer safety valve. Initially the event results in a rapidly decreasing RCS pressure until this pressure reaches a value corresponding to the hot leg saturation pressure. At this time, the pressure decrease is slowed considerably. The pressure continues to decrease throughout the transient. The effect of the pressure decrease is to decrease power via the moderator density feedback, but the reactor control system (if in the automatic mode) functions to maintain the power essentially constant throughout the initial stage of the transient. The average coolant temperature decreases slowly, but the pressurizer level increases until reactor trip.

The reactor may be tripped by the following reactor protection system signals:

- o Low DNBR.
- o Pressurizer low pressure.

An inadvertent opening of a pressurizer safety valve is classified as an American Nuclear Society (ANS) Condition II event, a fault of moderate frequency.

15.6.1.2 Analysis of Effects and Consequences

15.6.1.2.1 Method of Analysis

The accidental depressurization transient is analyzed by employing the detailed digital computer code LOFTRAN.(1) The code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief valves and main steam isolation valves, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level.

Plant characteristics and initial conditions are discussed in Subsection 15.0.3. In order to give conservative results in calculating the departure from nucleate boiling ratio (DNBR) during the transient, the following assumptions are made:

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

B. A least negative moderator coefficient of reactivity is assumed. The spatial effect of voids resulting from local or subcooled boiling is not considered in the analysis with respect to reactivity feedback or core power shape.

C. A large (absolute value) Doppler coefficient of reactivity is used such that the resultant amount of positive feedback is conservatively high to retard any power decrease resulting from moderator reactivity feedback.

Plant systems and equipment which are necessary to mitigate the effects of RCS depressurization caused by an inadvertent safety valve opening are discussed in Subsection 15.0.9 and listed in Table 15.0-6.

Normal reactor control systems are not required to function. The rod control system is assumed to be in the automatic mode to hold the core at full power longer and thus delay the trip. The reactor protection system functions to trip the reactor on the appropriate signal. No single active failure will prevent the reactor protection system from functioning properly.

15.6.1.2.2 Results

The system response to an inadvertent opening of a pressurizer safety valve is shown in Figures 15.6-1 and 15.6-2. Figure 15.6-1 illustrates the nuclear power transient following the depressurization. Nuclear power is maintained at the initial value until reactor trip occurs on low pressurizer pressure. The pressure decay transient and average temperature transient following the

accident are given in Figure 15.6-2. Pressure drops more rapidly while core heat generation is reduced via the trip and then slows once saturation temperature is reached in the hot leg. The DNBR decreases initially but increases rapidly following the trip, as shown in Figure 15.6-1. The DNBR remains above the limiting value throughout the transient. The DNBR design basis is described in Section 4.4 of RESAR-SP/90 PDA Module 5, "Reactor System".

The calculated sequence of the events for the inadvertent opening of a pressurizer safety valve incident is shown in Table 15.6-1.

15.6.1.3 Conclusion

The results of the analysis show that the pressurizer low pressure and the low DNBR reactor protection system signals provide adequate protection against the RCS depressurization event.

15.6.2 REFERENCE

1. Burnett, T. W. T., et al., "LOFTRAN Code Description," WCAP-7907, October 1972.

TABLE 15.6-1

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH CAUSE A DECREASE IN REACTOR COOLANT INVENTORY

Accident	Event	Time (s)
Inadvertent opening of a pressurizer safety valve	Pressurizer safety valve opens fully	0.0
	Low pressurizer pressure reactor trip setpoint reached	56.379
	Minimum DNBR occurs	59.0
	Rods begin to drop	58.379

(a,c)

NUCLEAR POWER (FRACTION OF NOMINAL)

DNBR

TIME (SEC)

FIGURE 15.6-1 NUCLEAR POWER AND DNBR TRANSIENTS FOR INADVERTANT OPENING OF A PRESSURIZER SAFETY VALVE

(a,c)

PRESSURIZER PRESSURE (psia)

CORE AVERAGE TEMPERATURE (°F)

TIME (SEC)

FIGURE 15.6-2 PRESSURIZER PRESSURE TRANSIENTS AND CORE A[illegible] TEMP. TRANSIENT FOR INADVERTANT OPENING OF A PRESSURIZER SAFETY VALVE

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.

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(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_o, 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

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

15A.2.4 <u>OFFSITE THYROID DOSE CALCULATION MODEL</u>

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:

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

where

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

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

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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

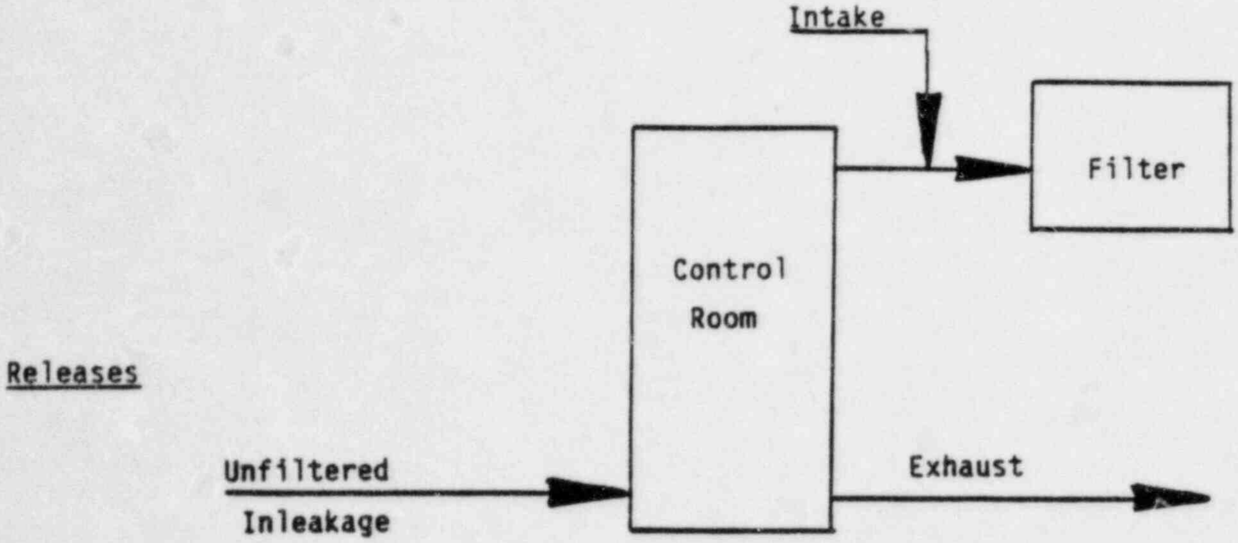
$D_{\gamma B}$ = gamma-body dose in rem

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.

$$\frac{dA_{CR}(t)}{dt} = [F_2 R_{FIN} + R_{UIN} \ \frac{X}{Q} \ R(t) - \lambda_3 A_{CR}(t)$$

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-m^3/Ci-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 (χ/Q), s/m^3	Table 15A-2	
Dose conversion factors		
Gamma-body and beta skin, rem-m^3/Ci-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)	(χ/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 $\frac{Rem-m^3}{Ci-s}$	Beta-Skin $\frac{Rem-m^3}{Ci-s}$	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

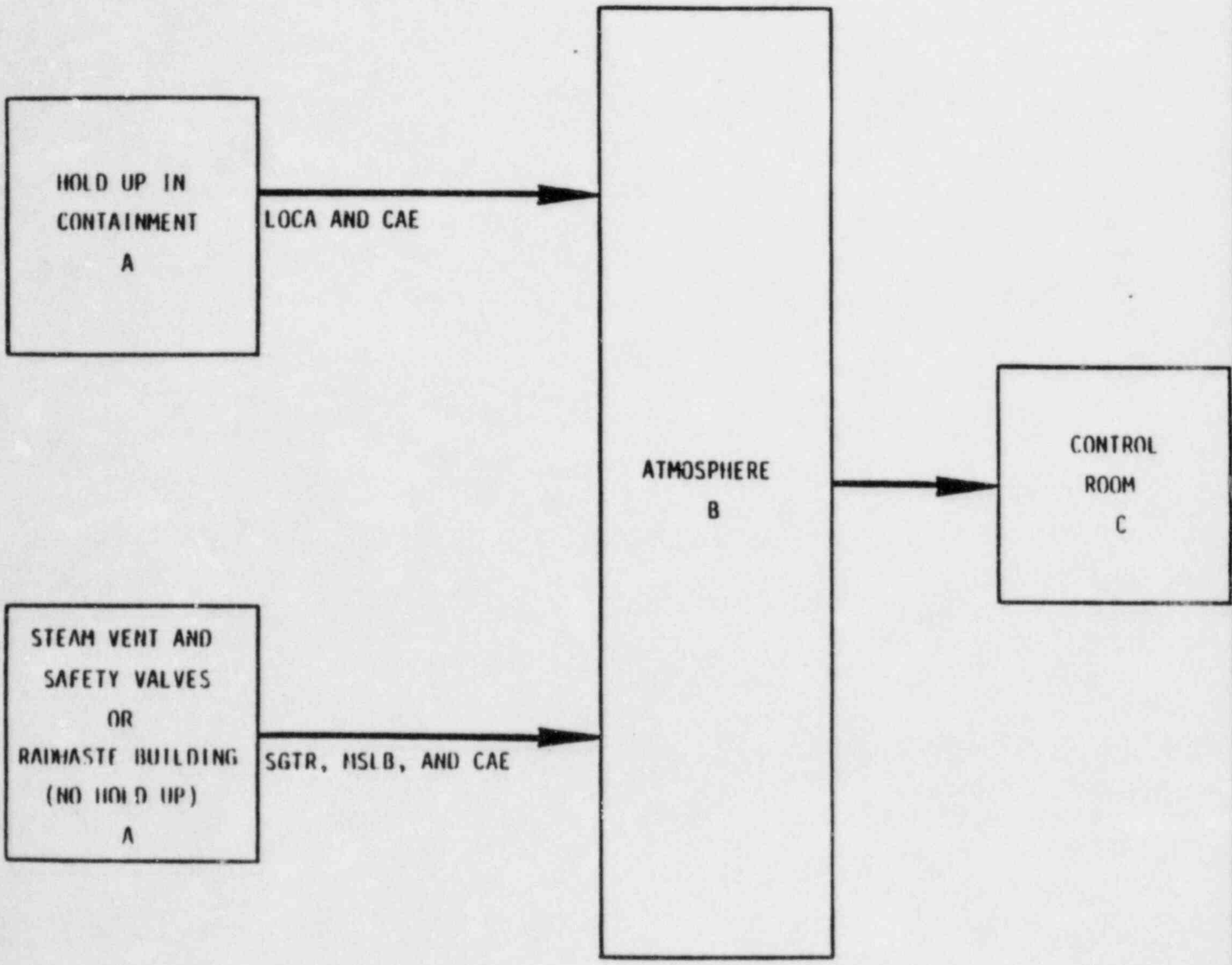


Figure 15.A-1 Release Pathways