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 6/8 "Secondary Side Safeguards System/Steam and Power Conversion System").

2. Feedwater system malfunctions causing an increase in feedwater flow (Subsection 15.1.2 of RESAR-SP/90 PDA Module 6/8, "Secondary Side Safeguards System/Steam and Power Conversion System").

3. Excessive increase in secondary steam flow (Subsection 15.1.3 of RESAR-SP/90 PDA Module 6/8, "Secondary Side Safeguards System/Steam and Power Conversion System").

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 6/8, "Secondary Side Safeguards System/Steam and Power Conversion System").

5. Loss of external load (Subsection 15.2.2 of RESAR-SP/90 PDA Module 6/8, "Secondary Side Safeguards System/Steam and Power Conversion System").

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

7. Inadvertent closure of main steam isolation valves (Subsection 15.2.4 of RESAR-SP/90 PDA Module 6/8, "Secondary Side Safeguards System/Steam and Power Conversion System").

8. Loss of condenser vacuum and other events resulting in turbine trip (Subsection 15.2.5 of RESAR-SP/90 PDA Module 6/8, "Secondary Side Safeguards System/Steam and Power Conversion System").

9. Loss of nonemergency A-C power to the station auxiliaries (Subsection 15.2.6 of RESAR-SP/90 PDA Module 6/8, "Secondary Side Safeguards System/Steam and Power Conversion System").

10. Loss of normal feedwater flow (Subsection 15.2.7 of RESAR-SP/90 PDA Module 6/8, "Secondary Side Safeguards System/Steam and Power Conversion System").

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

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

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

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

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

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

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

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

15.0.2.3 Condition III - Infrequent Faults

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

1. Minor steam system piping failures (Subsection 15.1.5 of RESAR-SP/90 PDA Module 6/8, "Secondary Side Safeguards System/Steam and Power Conversion System").

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

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

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 6/8, "Secondary Side Safeguards System/Steam and Power Conversion System").

2. Feedwater system pipe break (Subsection 15.2.8 of RESAR-SP/90 PDA Module 6/8, "Secondary Side Safeguards System/Steam and Power Conversion System").

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

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

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

6. Steam generator tube failure (Subsection 15.6.3 of RESAR-SP/90 PDA Module 6/8, "Secondary Side Safeguards System/Steam and Power Conversion 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 Chapter 7 of this module 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 Chapter 7 of this module. 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 Chapter 7 of this module 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 this module.

In the analysis of certain events, conservatism requires the use of large reactivity coefficient values whereas, in the analysis of other events, conservatism requires the use of small reactivity coefficient values. Some analyses, such as loss of reactor coolant from cracks or ruptures in the reactor coolant system, do not depend highly on reactivity feedback effects. The values used for each accident are given in Table 15.0-3. Reference is made in that table to Figure 15.0-2 which shows the upper and lower bound Doppler power coefficients as a function of power, used in the transient analysis. The justification for use of conservatively large vs. small reactivity coefficient values are treated on an event-by-event basis. Conservative combinations of parameters are used for a given transient to bound the effects of core life, although these combinations may not represent possible realistic situations.

15.0.6 Rod Cluster Control Assembly Insertion Characteristics

The negative reactivity insertion following a reactor trip is a function of the position vs. time of the rod cluster control assemblies and the variation in rod worth as a function of rod position. With respect to accident analyses, the critical parameter is the time of insertion up to the dashpot entry or approximately 85% of the rod cluster travel. For all accidents the insertion time to dashpot entry is conservatively taken as 3.4 seconds. The normalized rod cluster control assembly position vs. time assumed in accident analyses is shown in Figure 15.0-3.

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 following a trip of 4% $\Delta\rho$ is assumed in the transient analyses except where specifically noted otherwise. This assumption is conservative with respect to the calculated trip reactivity worth available as shown in Section 4.3 of 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 limit value of 1.62. All points below and to the left of a DNB line for a given pressure have DNBR greater than the limit value with the assumed axial and radial power distributions. The diagram shows that the DNB design basis is not violated for all cases if the area enclosed with the maximum protection lines is not traversed by the applicable DNBR line at any point.

The area of permissible operation (power, pressure and temperature) is bounded by the combination of reactor trips: high neutron flux (fixed setpoint); high pressure (fixed setpoint); low pressure (fixed setpoint); low DNBR (variable setpoint); high kw/ft (fixed setpoin.).

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 of RESAR-SP/90 PDA Module 6/8, "Secondary Side Safeguards System/Steam and Power Conversion System"), 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

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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.

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

* For accident analyses using the improved thermal design procedure.

TABLE 15.0-2a

VALUES OF PERTINENT PLANT PARAMETERS
UTILIZED IN ACCIDENT ANALYSES*

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

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

TABLE 15.0-3

SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED

Faults	Computer Codes Utilized	Kinetic Parameters Assumed: Delayed Neutron Fraction	Moder. Density (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)
15.1 Increase in Heat Removal by the Secondary System												
Feedwater System Malfunction Causing an Increase in Feedwater Flow	(See RESAR-SP/90 PDA Module 6/8, "Secondary Side Safeguards System/Steam and Power Conversion System")											
Excessive Increase in Secondary Steam Flow	(See RESAR-SP/90 PDA Module 6/8, "Secondary Side Safeguards System/Steam and Power Conversion System")											
Accidental Depressurization of the Main Steam System	(See RESAR-SP/90 PDA Module 6/8, "Secondary Side Safeguards System/Steam and Power Conversion System")											
Steam System Piping Failure	(See RESAR-SP/90 PDA Module 6/8, "Secondary Side Safeguards System/Steam and Power Conversion 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 6/8, "Secondary Side Safeguards System/Steam and Power Conversion System")											
Loss on Non-Emergency A-C Power to the Station Auxiliaries	(See RESAR-SP/90 PDA Module 6/8, "Secondary Side Safeguards System/Steam and Power Conversion System")											
Loss of Normal Feedwater Flow	(See RESAR-SP/90 PDA Module 6/8, "Secondary Side Safeguards System/Steam and Power Conversion System")											
Feedwater System Pipe Break	(See RESAR-SP/90 PDA Module 6/8, "Secondary Side Safeguards System/Steam and Power Conversion System")											

TABLE 15.0-3 (Con't)

Kinetic Parameters Assumed

Faults	Computer Codes Utilized	Delayed Neutron Fraction	Moder. Density (Δp/gm/cc)	Doppler	DNB Correlation	Improved Thermal Design Proced.	Initial NSSS Thermal Power Output (MWt)	Reactor Vessel Coolant Flow (gpm)	Vessel Average Temp. (°F)	Press. Pressure (psia)	Press. Water Volume (ft3)	Feedwater Temp. (°F)
15.3 Decrease in Reactor Coolant System Flow Rate												
Partial and Complete Loss of Forced Reactor Coolant Flow	(See RESAR-SP/90 PDA Module 4, "Reactor Coolant System")											
Reactor Coolant Pump Shaft Seizure (locked rotor	(See RESAR-SP/90 PDA Module 4, "Reactor Coolant System")											
15.4 Reactivity and power distribution anomalies												
Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical or Low Power Startup Condition.	(See RESAR-SP/90 PDA Module 5, "Reactor System")											
Uncontrolled Rod Cluster Assembly Bank Withdrawal at Power	(See RESAR-SP/90 PDA Module 5, "Reactor System")											
Control Rod Mis-alignment	(Later)											
Startup of an Inactive Reactor Coolant Loop at an Incorrect Temperature	(See RESAR-SP/90 PDA Module 4, "Reactor Coolant System")											

TABLE 15.0-3 (Con't)

Kinetic Parameters Assumed

Faults	Computer Codes Utilized	Delayed Neutron Fraction	Moder. Density (Δp/gm/cc)	Doppler	DNB Correlation	Improved Thermal Design Proced.	Initial NSSS Thermal Power Output (MWt)	Reactor Vessel Coolant Flow (gpm)	Vessel Average Temp. (°F)	Press. Pressure (psia)	Press. Water Volume (ft3)	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	(See RESAR-SP/90 PDA Module 5, "Reactor System")											
Spectrum of Rod Cluster Control Assembly Ejection Accidents	(See RESAR-SP/90 PDA Module 5, "Reactor System")											
15.5 Increase in Coolant Inventory												
Inadvertent Operation of ECCS During Power Operation	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA
15.6 Decrease in Reactor Coolant Inventory												
Inadvertent Opening of a Pressurizer Safety or Relief Valve	(See RESAR-SP/90 PDA Module 4, "Reactor Coolant System")											

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

TABLE 15.0-4

TRIP POINTS AND TIME DELAYS TO TRIP ASSUMED IN ACCIDENT ANALYSES

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

** 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.

TABLE 15.0-5

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

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

TABLE 15.0-5 (Con't)

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

TABLE 15.0-6

PLANT SYSTEMS AND EQUIPMENT AVAILABLE FOR TRANSIENT AND ACCIDENT CONDITIONS

Incident	Reactor Trip Functions	ESF Actuation Functions	Other Equipment	ESF Equipmen
15.1 Increase in Heat Removed by the Secondary System				
Feedwater System Malfunction Causing an Increase in Feedwater 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 Depressurization 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	Emergency feed system; Safety Injection Syst
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	Emergency feed system; Safety Injection Syst
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	Emergency feedwater 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	Emergency feedwa system
Loss of Normal Feedwater Flow	Steam generator low level, manual	Steam generator low level	Steam generator safety valves	Emergency feedwa 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	Emergency feedw system, Safety Injection Syste
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, PORVs, SG overfill control valves and steamline stop valves	Emergency Core Cooling System, Emergency Feedwa System, Emergenc 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, Emergency Feedw System, Contair Heat Removal Sy Emergency Power

TABLE 15.0-7

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

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

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

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

TABLE 15.0-8

REACTOR COOLANT IODINE CONCENTRATIONS FOR

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

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

TABLE 15.0-9

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

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

TABLE 15.0-10

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

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

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

** 500 x equilibrium appearance rate.

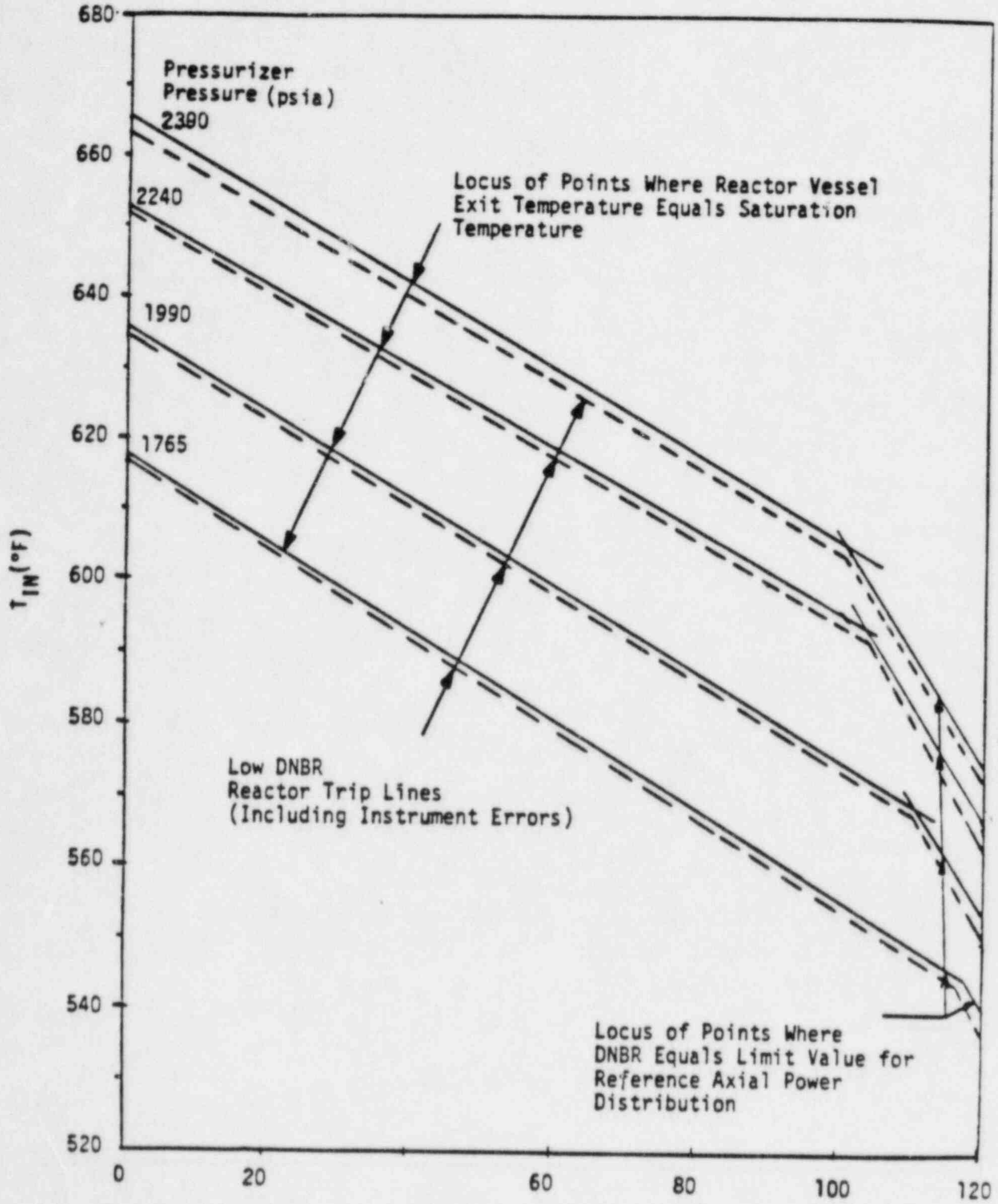

Figure 15.0-1 Illustration of Core Thermal Limits and DNB Protection (N Loop Operation)

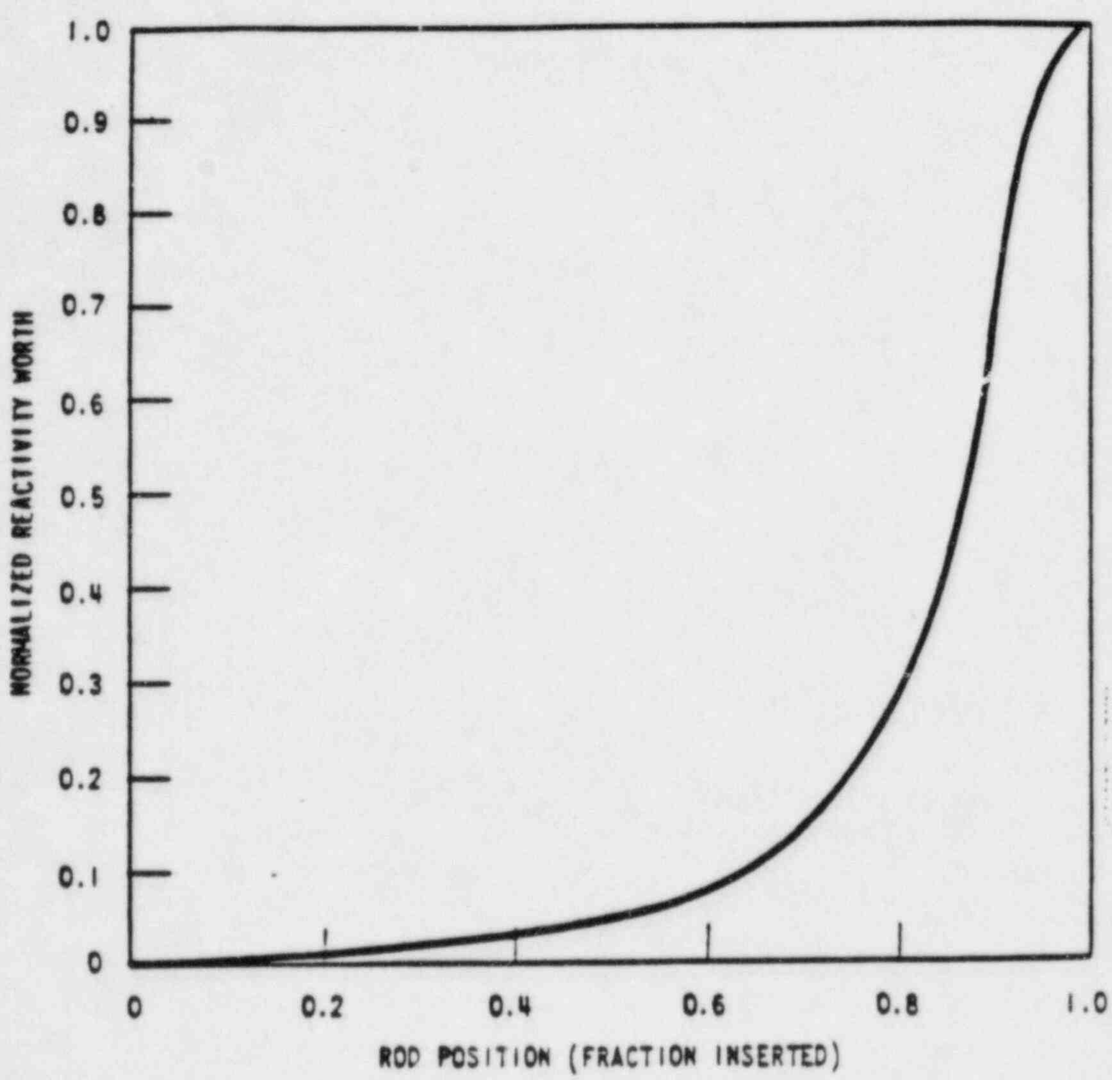

FIGURE 15.0-4 NORMALIZED RCCA REACTIVITY WORTH VS. FRACTION INSERTION

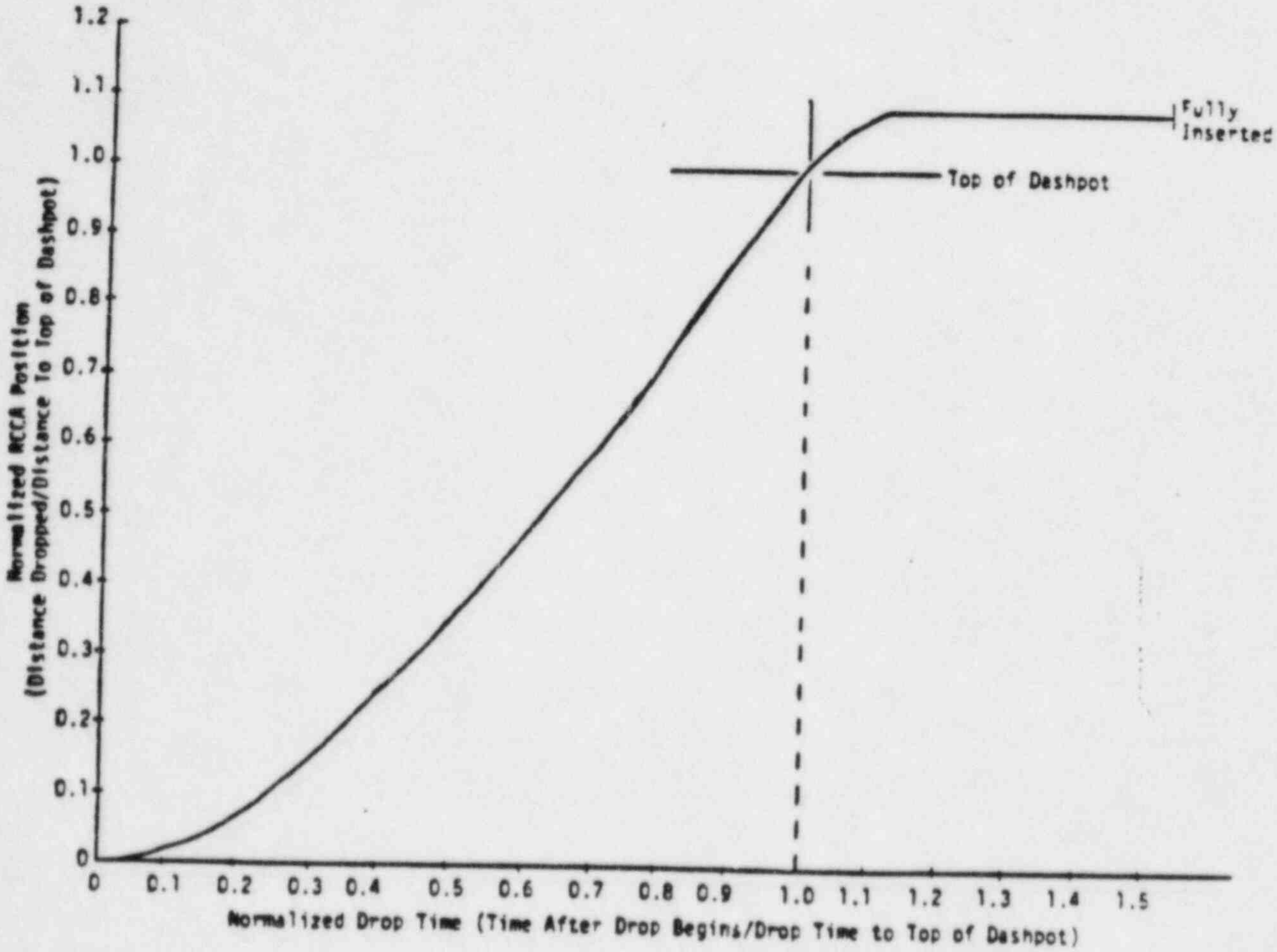

FIGURE 15.0-3 RCCA POSITION VS. TIME TO DASHPOT

15.4 REACTIVITY AND POWER DISTRIBUTION ANOMALIES

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

15.4.3.1 Identification of Causes and Accident Description

Rod cluster control assembly (RCCA) misoperation accidents include:

- A dropped RCCA, discussed in this module (to be provided later in the integrated PDA document).
- A dropped RCCA bank, discussed in this module (to be provided later in the integrated PDA document).
- Statically misaligned RCCA (discussed in RESAR-SP/90 PDA Module 5, "Reactor System").
- Withdrawal of a single RCCA (discussed in RESAR-SP/90 PDA Module 5, "Reactor System").