CHAPTER 15

ACCIDENT ANALYSES

15.0 Accident Analyses

15.0.1 Classification of Plant Conditions

The ANSI 18.2 (Reference 1) classification divides plant conditions into four categories according to anticipated frequency of occurrence and potential radiological consequences to the public. The four categories are as follows:

- Condition I: Normal operation and operational transients
- Condition II: Faults of moderate frequency
- Condition III: Infrequent faults
- Condition IV: Limiting faults

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

15.0.1.1 Condition I: Normal Operation and Operational Transients

Condition I occurrences are those that are expected to occur frequently or regularly in the course of power operation, refueling, maintenance, or maneuvering of the plant. As such, Condition I occurrences are accommodated with margin between a plant parameter and the value of that parameter requiring either automatic or manual protective action.

Because Condition I events occur frequently, they must be considered from the point of view of their effect on 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 that can occur during Condition I operation.

A typical list of Condition I events follows.

Steady-state and Shutdown Operations

See Table 1.1-1 of Chapter 16.

Operation with Permissible Deviations

Various deviations that occur during continued operation as permitted by the plant Technical Specifications are considered in conjunction with other operational modes. These deviations include the following:

- Operation with components or systems out of service (such as an inoperable rod cluster control assembly [RCCA])

- Leakage from fuel with limited cladding defects

- Excessive radioactivity in the reactor coolant:
  - Fission products
  - Corrosion products
  - Tritium

- Operation with steam generator tube leaks

- Testing

Operational Transients

- Plant heatup and cooldown
- Step load changes (up to ±10 percent)
- Ramp load changes (up to 5 percent/minute)
- Load rejection up to and including design full-load rejection transient

15.0.1.2 Condition II: Faults of Moderate Frequency

These faults, at worst, result in 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 (Condition III or IV events). In addition, Condition II events are not expected to result in fuel rod failures, reactor coolant system failures, or secondary system overpressurization. The following faults are included in this category:

- Feedwater system malfunctions that result in a decrease in feedwater temperature (see subsection 15.1.1)

- Feedwater system malfunctions that result in an increase in feedwater flow (see subsection 15.1.2)

- Excessive increase in secondary steam flow (see subsection 15.1.3)

- Inadvertent opening of a steam generator relief or safety valve (see subsection 15.1.4)

- Inadvertent operation of the passive residual heat removal heat exchanger (see subsection 15.1.6)
- Loss of external electrical load (see subsection 15.2.2)
- Turbine trip (see subsection 15.2.3)
- Inadvertent closure of main steam isolation valves (see subsection 15.2.4)
- Loss of condenser vacuum and other events resulting in turbine trip (see subsection 15.2.5)
- Loss of ac power to the station auxiliaries (see subsection 15.2.6)
- Loss of normal feedwater flow (see subsection 15.2.7)
- Partial loss of forced reactor coolant flow (see subsection 15.3.1)
- Uncontrolled RCCA bank withdrawal from a subcritical or low-power startup condition (see subsection 15.4.1)
- Uncontrolled RCCA bank withdrawal at power (see subsection 15.4.2)
- RCCA misalignment (dropped full-length assembly, dropped full-length assembly bank, or statically misaligned assembly) (see subsection 15.4.3)
- Startup of an inactive reactor coolant pump at an incorrect temperature (see subsection 15.4.4)
- Chemical and volume control system malfunction that results in a decrease in the boron concentration in the reactor coolant (see subsection 15.4.6)
- Inadvertent operation of the passive core cooling system during power operation (see subsection 15.5.1)
- Chemical and volume control system malfunction that increased reactor coolant inventory (see subsection 15.5.2)
- Inadvertent opening of a pressurizer safety valve (see subsection 15.6.1)
- Break in instrument line or other lines from the reactor coolant pressure boundary that penetrate containment (see subsection 15.6.2)

15.0.1.3 Condition III: Infrequent Faults

Condition III events are faults that may occur infrequently during the life of the plant. They may result in the failure of only a small fraction of the fuel rods. The release of radioactivity is not sufficient to interrupt or restrict public use of those areas beyond the exclusion area boundary, in accordance with the guidelines of 10 CFR 100. By definition, a Condition III event alone does not generate a Condition IV event or result in a consequential loss of function of the reactor coolant system or containment barriers. The following faults are included in this category:

- Steam system piping failure (minor) (see subsection 15.1.5)
- Complete loss of forced reactor coolant flow (see subsection 15.3.2)
- RCCA misalignment (single RCCA withdrawal at full power) (see subsection 15.4.3)
- Inadvertent loading and operation of a fuel assembly in an improper position (see subsection 15.4.7)
- Inadvertent operation of automatic depressurization system (see subsection 15.6.1)
- Loss-of-coolant accidents (LOCAs) resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary (small break) (see subsection 15.6.5)
- Gas waste management system leak or failure (see subsection 15.7.1)
- Liquid waste management system leak or failure (see subsection 15.7.2)
- Release of radioactivity to the environment due to a liquid tank failure (see subsection 15.7.3)
- Spent fuel cask drop accidents (see subsection 15.7.5)

15.0.1.4 Condition IV: Limiting Faults

Condition IV events are faults that are not expected to take place, but are postulated because their consequences include the potential of the release of significant amounts of radioactive material. They are the faults that must be designed against, and they represent limiting design cases. Condition IV faults are not to cause a fission product release to the environment resulting in doses in excess of the guideline values of 10 CFR 100. A single Condition IV event is not to cause a consequential loss of required functions of systems needed to cope with the fault, including those of the emergency core cooling system and the containment. The following faults are classified in this category:

- Steam system piping failure (major) (see subsection 15.1.5)

- Feedwater system pipe break (see subsection 15.2.8)
- Reactor coolant pump shaft seizure (locked rotor) (see subsection 15.3.3)
- Reactor coolant pump shaft break (see subsection 15.3.4)
- Spectrum of RCCA ejection accidents (see subsection 15.4.8)
- Steam generator tube rupture (see subsection 15.6.3)
- LOCAs resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary (large break) (see subsection 15.6.5)
- Design basis fuel handling accidents (see subsection 15.7.4)

15.0.2 Optimization of Control Systems

A control system setpoint study is performed prior to plant operation to simulate performance of the primary plant control systems and overall plant performance. In this study, emphasis is placed on the development of the overall plant control systems that automatically maintain conditions in the plant within the allowed operating window and with optimum control system response and stability over the entire range of anticipated plant operating conditions. The control system setpoints are developed using the nominal protection and safety monitoring system setpoints implemented in the plant. Where appropriate (such as in margin to reactor trip analyses), instrumentation errors are considered and are applied in an adverse direction with respect to maintaining system stability and transient performance. The accident analysis and plant control system setpoint study in combination show that the plant can be operated and meet both safety and operability requirements throughout the core life and for various levels of power operation.

The plant control system setpoint study is comprised of analyses of the following control systems: plant control, axial offset control, rapid power reduction, steam dump (turbine bypass), steam generator level, pressurizer pressure, and pressurizer level.

15.0.3 Plant Characteristics and Initial Conditions Assumed in the Accident Analyses

15.0.3.1 Design Plant Conditions

Table 15.0-1 lists the principal power rating values assumed in the analyses performed. The thermal power output includes the effective thermal power generated by the reactor coolant pumps. Selected AP600 loop layout elevations are shown in Figure 15.0.3-2 to aid in interpreting plots shown in other Chapter 15 subsections.

The values of other pertinent plant parameters used in the accident analyses are given in Table 15.0-3.

15.0.3.2 Initial Conditions

For most accidents that are departure from nucleate boiling (DNB) limited, nominal values of initial conditions are assumed. The allowances on power, temperature, and pressure are determined on a statistical basis and are included in the departure from nucleate boiling ratio (DNBR) design limit values (see subsection 4.4), as described in WCAP-11397-P-A (Reference 2). This procedure is known as the Revised Thermal Design Procedure (RTDP) and is discussed more fully in Section 4.4.

For accidents that are not DNB limited, or for which the revised thermal design procedure is not used, the initial conditions are obtained by adding the maximum steady-state errors to rated values. The following conservative steady-state errors are assumed in the analysis:

Core power	± 2 percent allowance for calorimetric error
Average reactor coolant system temperature	+6.5 or -7.0°F allowance for controller deadband and measurement errors
Pressurizer pressure	± 50 psi allowance for steady-state fluctuations and measurement errors

Initial values for core power, average reactor coolant system temperature, and pressurizer pressure are selected to minimize the initial DNBR unless otherwise stated in the sections describing the specific accidents. Table 15.0-2 summarizes the initial conditions and computer codes used in the accident analyses.

15.0.3.3 Power Distribution

The transient response of the reactor system is dependent on the initial power distribution. The nuclear design of the reactor core minimizes adverse power distribution through the placement of fuel assemblies and control rods. Power distribution may be characterized by the nuclear enthalpy rise hot channel factor ($F_{\Delta H}$) and the total peaking factor (F_q). Unless specifically noted otherwise, the peaking factors used in the accident analyses are those presented in Chapter 4.

For transients that may be DNB limited, the radial peaking factor is important. The radial peaking factor increases with decreasing power level due to control rod insertion. This increase in $F_{\Delta H}$ is included in the core limits illustrated in Figure 15.0.3-1. Transients that may be departure from nucleate boiling limited are assumed to begin with an $F_{\Delta H}$, consistent with the initial power level defined in the Technical Specifications.

The axial power shape used in the DNB calculation is the 1.55 chopped cosine, as discussed in subsection 4.4, for transients analyzed at full power and the most limiting power shape calculated or allowed for accidents initiated at nonfull power or asymmetric RCCA conditions.

The radial and axial power distributions just described are input to the THINC code as described in subsection 4.4.

For transients that may be overpower-limited, the total peaking factor (F_q) is important. Transients that may be overpower-limited are assumed to begin with plant conditions, including power distributions, which are consistent with reactor operation as defined in the Technical Specifications.

For overpower transients that are slow with respect to the fuel rod thermal time constant (for example, the chemical and volume control system malfunction that results in a slow decrease in the boron concentration in the reactor coolant system as well as an excessive increase in secondary steam flow) and that may reach equilibrium without causing a reactor trip, the fuel rod thermal evaluations are performed as discussed in subsection 4.4.

For overpower transients that are fast with respect to the fuel rod thermal time constant (for example, the uncontrolled RCCA bank withdrawal from subcritical or lower power startup and RCCA ejection incident, both of which result in a large power rise over a few seconds), a detailed fuel transient heat transfer calculation is performed.

15.0.4 Reactivity Coefficients Assumed in the Accident Analysis

The transient response of the reactor system is dependent on reactivity feedback effects, in particular, the moderator temperature coefficient and the Doppler power coefficient. These reactivity coefficients are discussed in subsection 4.3.2.3.

In the analysis of certain events, conservatism requires the use of large reactivity coefficient values. The values used are given in Figure 15.0.4-1, 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 versus small reactivity coefficient values is treated on an event-by-event basis. In some cases, conservative combinations of parameters are used to bound the effects of core life, although these combinations may not represent possible realistic situations.

15.0.5 Rod Cluster Control Assembly Insertion Characteristics

The negative reactivity insertion following a reactor trip is a function of the acceleration of the RCCAs as a function of time and the variation in rod worth as a function of rod position. For accident analyses, the critical parameter is the time of insertion up to the dashpot entry, or approximately 85 percent of the rod cluster travel. In analyses where all of the reactor coolant pumps are coasting down prior to, or simultaneous, with RCCA insertion, a time of 1.8 seconds is used for insertion time to dashpot entry.

In Figure 15.0.5-1, the curve labeled "complete loss of flow transients" shows the RCCA position versus time normalized to 1.8 seconds assumed in accident analyses where all reactor coolant pumps are coasting down. In analyses where some or all of the reactor coolant pumps are running, the RCCA insertion time to dashpot is conservatively taken as 2.4 seconds. The RCCA position versus time normalized to 2.4 seconds is also shown in Figure 15.0.5-1.

The use of such a long insertion time provides conservative results for accidents and is intended to apply to all types of RCCAs, which may be used throughout plant life. Drop time testing requirements are specified in the Technical Specifications.

Figure 15.0.5-2 shows the fraction of total negative reactivity insertion versus normalized rod position for a core where the axial distribution is skewed to the lower region of the core. An axial distribution skewed to the lower region of the core can arise from an unbalanced xenon distribution. This curve is used to compute the negative reactivity insertion versus time following a reactor trip, which is input to the point kinetics core models used in transient analyses. The bottom-skewed power distribution itself is not an input into the point kinetics core model.

There is inherent conservatism in the use of Figure 15.0.5-2 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, significantly more negative reactivity is inserted than that shown in the curve, due to the more favorable axial distribution existing prior to trip.

The normalized RCCA negative reactivity insertion versus time is shown in Figure 15.0.5-3. The curves shown in this figure were obtained from Figures 15.0.5-1 and 15.0.5-2. A total negative reactivity insertion following a trip of 4 percent Δk is assumed in the transient analyses except where specifically noted otherwise. This assumption is conservative with respect to the calculated trip reactivity worth available as shown in Table 4.3-3.

The normalized RCCA negative reactivity insertion versus time curve for an axial power distribution skewed to the bottom (Figure 15.0.5-3) is used in those transient analyses for which a point kinetics core model is used. Where special analyses require use of three-dimensional or axial one-dimensional core models, the negative reactivity insertion resulting from the reactor trip is calculated directly by the reactor kinetics code and is not separable from the other reactivity feedback effects. In this case, the RCCA position versus time of Figure 15.0.5-1 is used as code input.

15.0.6 Protection and Safety Monitoring System Setpoints and Time Delays to Trip Assumed in Accident Analyses

A reactor trip signal acts to open two trip breaker sets connected in series, feeding power to the control rod drive mechanisms. The loss of power to the mechanism coils causes the mechanisms to release the RCCAs, which then fall by gravity into the core. There are various instrumentation delays associated with each trip function including delays in signal actuation, in opening the trip breakers, and in the release of the rods by the mechanisms. The total delay to trip is defined as the time delay from the time that trip conditions are reached to the time the rods are free and begin to fall. Limiting trip setpoints assumed in accident analyses and the time delay assumed for each trip function are given in Table 15.0-4a. Reference is made in that table to overtemperature and overpower ΔT trip shown in Figure 15.0.3-1.

Table 15.0-4a also summarizes the setpoints and the instrumentation delay for engineered safety features (ESF) functions used in accident analyses. Time delays associated with equipment actuated (such as valve stroke times) by ESF functions are summarized in Table 15.0-4b.

The difference between the limiting setpoint assumed for the analysis and the nominal setpoint represents an allowance for instrumentation channel error and setpoint error. Nominal 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 are determined periodically in accordance with the plant Technical Specifications.

15.0.7 Instrumentation Drift and Calorimetric Errors, Power Range Neutron Flux

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

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

The secondary power is obtained from measurement of feedwater flow, feedwater inlet temperature to the steam generators, and steam pressure. Installed plant instrumentation is used for these measurements.

15.0.8 Plant Systems and Components Available for Mitigation of Accident Effects

The plant is designed to afford proper protection against the possible effects of natural phenomena, postulated environmental conditions, and dynamic effects of the postulated accidents. In addition, the design incorporates features that minimize the probability and effects of fires and explosions.

Chapter 17 discusses the quality assurance program that is implemented to provide confidence that the plant systems satisfactorily perform their assigned safety functions. The incorporation of these features in the plant, coupled with the reliability of the design, provides confidence that the normally operating systems and components listed in Table 15.0-6 are 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 (Reference 1) is used. The design of safety-related systems (including protection systems) is consistent with IEEE Standard 379-1988 and

Regulatory Guide 1.53 in the application of the single-failure criterion. Conformance to Regulatory Guide 1.53 is summarized in subsection 1.9.1.

Table 15.0-8 summarizes the nonsafety-related systems assumed in the analyses to mitigate the consequences of events. Except for the cases listed in Table 15.0-8, control system action is not used for mitigation of accidents. WCAP-14477 (Reference 11) further reviews the plant control system actions used in the accident analyses.

15.0.9 Fission Product Inventories

The sources of radioactivity for release are dependent on the specific accident. Activity may be released from the primary coolant, from the secondary coolant, and from the reactor core if the accident involves fuel damage. The radiological consequences analyses use the conservative design basis source terms identified in Appendix 15A.

15.0.10 Residual Decay Heat

15.0.10.1 Total Residual Heat

Residual heat in a subcritical core is calculated for the LOCA according to the requirements of 10 CFR 50.46, as described in WCAP-10054-P-A and WCAP-12945-P (References 3 and 4). The large-break LOCA methodology considers uncertainty in the decay power level. The small-break LOCA events and post-LOCA long-term cooling windows use 10 CFR 50, Appendix K, decay heat, which assumes 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 an equilibrium cycle.

15.0.10.2 Distribution of Decay Heat Following a Loss-of-coolant Accident

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

For example, consider the transient resulting from the postulated double-ended break of the largest reactor coolant system pipe; one-half second after the rupture, about 30 percent of the heat generated in the fuel rods is from gamma ray absorption. The gamma power shape is less peaked than the steady-state fission power shape, reducing the energy deposited in the hot rod at the expense of adjacent colder rods. A conservative estimate of this effect on the hot rod is a reduction of 10 percent of the gamma ray contribution or 3 percent of the total heat. Because the water density is considerably reduced at this time, an average of 98 percent of the available heat is deposited in the fuel rods; the remaining 2 percent is absorbed by water, thimbles, sleeves, and grids. Combining the 3 percent total heat reduction from gamma redistribution with this 2 percent absorption produce as the net effect a factor of 0.95, which exceeds the actual heat production in the hot rod. The actual hot rod heat generation is computed during the AP600 large-break LOCA transient as a function of core fluid conditions.

15.0.11 Computer Codes Used

Summaries of some of the principal computer codes used in transient analyses are given as follows. Other codes – in particular, 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 (see subsection 15.6.5) – are summarized in their respective accident analyses sections. The codes used in the analyses of each transient are listed in Table 15.0-2.

15.0.11.1 FACTRAN Computer Code

FACTRAN (Reference 5) 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 code uses a fuel model which simultaneously exhibits the following features:

- A sufficiently large number of radial space increments to handle fast transients such as rod ejection accidents
- Material properties which are functions of temperature and a sophisticated fuel-to-clad gap heat transfer calculation
- The necessary calculations to handle post-DNB transients: film boiling heat transfer correlations, zircaloy-water reaction, and partial melting of the materials

FACTRAN is further discussed in WCAP-7908 (Reference 5).

15.0.11.2 LOFTRAN Computer Code

The LOFTRAN (Reference 6) 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 generator (tube and shell sides), and pressurizer. The pressurizer heaters, spray, 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 uses a homogeneous, saturated mixture for the thermal transients and a water level correlation for indication and control. The protection and safety monitoring system is simulated to include reactor trips on high neutron flux, overtemperature ΔT, high and low pressure, low flow, and high pressurizer level. Control systems are also simulated, including rod control, steam dump, feedwater control, and pressurizer level and pressure control. The emergency core cooling system, including the accumulators, is also modeled.

LOFTRAN is a versatile program 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 15.0.3-1. The core limits represent the minimum value of DNBR as calculated for typical or thimble cell.

The LOFTRAN code is modified to allow the simulation of the passive residual heat removal (PRHR) heat exchanger, core makeup tanks, and associated protection and safety monitoring system actuation logic. A discussion of these models and additional validation is presented in WCAP-14234 (Reference 10).

LOFTTR2 (Reference 8) is a modified version of LOFTRAN with a more realistic break flow model, a two-region steam generator secondary side, and an improved capability to simulate operator actions during a steam generator tube rupture (SGTR) event.

The LOFTTR2 code is modified to allow the simulation of the PRHR heat exchanger, core makeup tanks, and associated protection system actuation logic. The modifications are identical to those made to the LOFTRAN code. A discussion of these models is presented in WCAP-14234 (Reference 10).

15.0.11.3 TWINKLE Computer Code

The TWINKLE (Reference 7) program is a multidimensional spatial neutron kinetics code, which is patterned after steady-state codes currently used for reactor core design. The code uses an implicit finite-difference method to solve the two-group transient neutron diffusion equations in one, two, and three dimensions. The code uses six delayed neutron groups and contains a detailed multiregion fuel-clad-coolant heat transfer model for calculating pointwise Doppler and moderator feedback effects. The code handles up to 2000 spatial points and performs its own steady-state initialization. Aside from basic cross-section data and thermal-hydraulic parameters, the code accepts as input basic driving functions, such as inlet temperature, pressure, flow, boron concentration, control rod motion, and others. Various edits are provided (for example, channelwise power, axial offset, enthalpy, volumetric surge, point-wise power, and fuel temperatures).

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

15.0.11.4 THINC Computer Code

The THINC code is described in subsection 4.4.

15.0.11.5 WESTAR Computer Code

The WESTAR code is described in subsection 4.4.

15.0.12 Component Failures

15.0.12.1 Active Failures

SECY-77-439 (Reference 9) provides a description of active failures. An active failure results in the inability of a component to perform its intended function.

An active failure is defined differently for different components. For valves, an active failure is the failure of a component to mechanically complete the movement required to perform its function. This includes the failure of a remotely operated valve to change position on demand. The spurious, unintended movement of the valve is also considered as an active failure. Failure of a manual valve to change position under local operator action is included.

Spring-loaded safety or relief valves that are designed for and operate under single-phase fluid conditions are not considered for active failures to close when pressure is reduced below the valve set point. However, when valves designed for single-phase flow are challenged with two-phase flow, such as a steam generator or pressurizer safety valve, the failure to reseat is considered as an active failure.

For other active equipment – such as pumps, fans, and rotating mechanical components – an active failure is the failure of the component to start or to remain operating.

For electrical equipment, the loss of power, such as the loss of offsite power or the loss of a diesel generator, is considered as a single failure. In addition, the failure to generate an actuation signal, either for a single component actuation or for a system-level actuation, is also considered as an active failure.

Spurious actuation of an active component is considered as an active failure for active components in safety-related passive systems. An exception is made for active components if specific design features or operating restrictions are provided that can preclude such failures (such as power lockout, confirmatory open signals, or continuous position alarms).

A single incorrect or omitted operator action in response to an initiating event is also considered as an active failure. The error is limited to manipulation of safety-related equipment and does not include thought-process errors or similar errors that could potentially lead to common cause or multiple errors.

15.0.12.2 Passive Failures

SECY-77-439 also provides a description of passive failures. A passive failure is the structural failure of a static component that limits the effectiveness of the component in carrying out its design function. A passive failure is applied to fluid systems and consists of a breach in the fluid system boundary. Examples include cracking of pipes, sprung flanges, or valve packing leaks.

Passive failures are not assumed to occur until 24 hours after the start of the event. Consequential effects of a pipe leak – such as flooding, jet impingement, and failure of a valve with a packing leak – must be considered.

Where piping is significantly overdesigned or installed in a system where the pressure and temperature conditions are relatively low, passive leakage is not considered a credible failure mechanism. Line blockage is also not considered as a passive failure mechanism.

15.0.12.3 Limiting Single Failures

The most limiting single active failure (where one exists), as described in Section 3.1, of safety-related equipment, is identified in each analysis description. The consequences of this failure are described therein. In some instances, because of redundancy in protection equipment, no single failure that could adversely affect the consequences of the transient is identified. The failure assumed in each analysis is listed in Table 15.0-7.

15.0.13 Operator Actions

For events where the PRHR heat exchanger is actuated, the plant automatically cools down to the safe shutdown condition. Where a stabilized condition is reached automatically following a reactor trip, it is expected that the operator may, following event recognition, take manual control and proceed with orderly shutdown of the reactor in accordance with the normal, abnormal, or emergency operating procedures. The exact actions taken and the time at which these actions occur depend on what systems are available and the plans for further plant operation.

However, for these events, operator actions are not required to maintain the plant in a safe and stable condition. Operator actions typical of normal operation are credited for the inadvertent actuations of equipment in response to a Condition II event.

15.0.14 Loss of Offsite ac Power

As required in GDC 17 of 10 CFR Part 50, Appendix A, anticipated operational occurrences and postulated accidents are analyzed assuming a loss of offsite ac power. The loss of offsite power is not considered as a single failure, and the analysis is performed without changing the event category. In the analyses, the loss of offsite ac power is considered to be a potential consequence of the event.

A loss of offsite ac power will be considered a consequence of an event due to disruption of the grid following a turbine trip during the event. Event analyses that do not result in a possible consequential disruption of offsite ac power do not assume offsite power is lost.

For those events where offsite ac power is lost, an appropriate time delay between turbine trip and the postulated loss of offsite ac power is assumed in the analyses. A time delay of 3 seconds is used. This time delay is based on the inherent stability of the offsite power grid as discussed in Section 8.2. Following the time delay, the effect of the loss of offsite ac power on plant auxiliary equipment – such as reactor coolant pumps, main feedwater pumps, condenser, startup feedwater pumps, and RCCAs – is considered in the analyses.

Design basis LOCA analyses are governed by the GDC-17 requirement to consider the loss of offsite power. For the AP600 design, in which all the safety-grade systems are passive, the availability of offsite power is significant only regarding reactor coolant pump operation for LOCA events. In subsection 15.6.5.4A, a sensitivity case is presented which shows that for large-break LOCAs, assuming the loss of offsite power coincident with the inception of the LOCA event is nonlimiting relative to assuming continued reactor coolant pump operation until the automatic reactor coolant pump trip occurs following an "S" signal less than 20 seconds into the transient. For small-break LOCA events, the AP600 automatic reactor coolant pump trip feature prevents continued operation of the reactor coolant pumps from mixing the liquid and vapor present within a two-phase reactor coolant system inventory to increase the liquid break flow and deplete the reactor coolant system mass inventory rapidly. The automatic reactor coolant pump trip occurs early enough during AP600 small-break LOCA transients that emergency core cooling system performance is not affected by the loss of offsite power assumption because the total break flow is approximately equivalent for reactor coolant pump trip occurring either at time zero or as a result of the "S" signal. Whether a loss of offsite power is postulated at the inception of the LOCA event or occurs automatically later on is unimportant in the subsection 15.6.5.4C long-term cooling analyses because with either assumption, the reactor coolant pumps are tripped long before the long-term cooling windows are analyzed.

The AP600 protection and safety monitoring system and passive safeguards systems are not dependent on offsite power or on any backup diesel generators. Following a loss of ac power, the protection and safety monitoring system and passive safeguards will be able to perform the related safety function and there will be no additional time delays for these functions to be completed.

15.0.15 Combined License Information

This section has no requirement for additional information to be provided in support of the Combined License application.

15.0.16 References

1. American National Standards Institute N18.2, "Nuclear Safety Criteria for the Design of Stationary PWR Plants," 1973.

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

3. Lee, N., "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," WCAP-10054-P-A (Proprietary) and WCAP-10081 (Nonproprietary), August 1985.

4. Bajorek, S. M., "Code Qualification Document for Best Estimate LOCA Analysis," WCAP-12945-P (Proprietary), Revision 1, March 1998.

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Table 15.0-1

NUCLEAR STEAM SUPPLY SYSTEM POWER RATINGS

Thermal power output (MWt)	1940
Effective thermal power generated by the reactor coolant pumps (MWt)	7
Core thermal power (MWt)	1933

Table 15.0-2 (Sheet 1 of 4)

SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED

			Reactivity Coefficients Assumed			
Section	**Faults**	**Computer Codes Used**	**Moderator Density ($\Delta k/gm/cm^3$)**	**Moderator Temperature (pcm/°F)**	**Doppler**	**Initial Thermal Power Output Assumed (MWt)**
15.1	Increase in heat removal from the primary system					
	Feedwater system mal-functions that result in an increase in feedwater flow	LOFTRAN	0.374	–	Upper curve of Figure 15.0.4-1	0 and 1940
	Excessive increase in secondary steam flow	LOFTRAN	0.0 and 0.374	–	Upper and lower curves of Figure 15.0.4-1	1940
	Inadvertent opening of a steam generator relief or safety valve	LOFTRAN	Function of moderator density (see Figure 15.1.4-1)	–	See subsection 15.1.4	0 (subcritical)
	Steam system piping failure	LOFTRAN, THINC	Function of moderator density (see Figure 15.1.4-1)	–	See subsection 15.1.5	0 (subcritical)
	Inadvertent operation of the PRHR	LOFTRAN	See subsection 15.1.6.2.1	–	Upper curve of Figure 15.0.4-1	1940

Table 15.0-2 (Sheet 2 of 4)

SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED

Section	Faults	Computer Codes Used	Reactivity Coefficients Assumed: Moderator Density ($\Delta k/gm/cm^3$)	Reactivity Coefficients Assumed: Moderator Temperature (pcm/°F)	Reactivity Coefficients Assumed: Doppler	Initial Thermal Power Output Assumed (MWt)
15.2	Decrease in heat removal by the secondary system					
	Loss of external electrical load and/or turbine trip	LOFTRAN	0.0 and 0.374	–	Lower and upper curves of Figure 15.0.4-1	1940
	Loss of nonemergency ac power to the station auxiliaries	LOFTRAN	0.0	–	Lower curve of Figure 15.0.4-1	1978.8[(a)]
	Loss of normal feedwater flow	LOFTRAN	0.0	–	Lower curve of Figure 15.0.4-1	1978.8[(a)]
	Feedwater system pipe break	LOFTRAN	0.374	–	Lower curve of Figure 15.0.4-1	1978.8[(a)]
15.3	Decrease in reactor coolant system flow rate					
	Partial and complete loss of forced reactor coolant flow	LOFTRAN, FACTRAN, WESTAR	0.0	–	Lower curve of Figure 15.0.4-1	1940
	Reactor coolant pump shaft seizure (locked rotor)	LOFTRAN, FACTRAN	0.0	–	Lower curve of Figure 15.0.4-1	1978.8[(a)]

Table 15.0-2 (Sheet 3 of 4)

SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED

Section	Faults	Computer Codes Used	Reactivity Coefficients Assumed: Moderator Density ($\Delta k/gm/cm^3$)	Reactivity Coefficients Assumed: Moderator Temperature (pcm/°F)	Reactivity Coefficients Assumed: Doppler	Initial Thermal Power Output Assumed (MWt)
15.4	Reactivity and power distribution anomalies					
	Uncontrolled RCCA bank withdrawal from a subcritical or low power startup condition	TWINKLE, FACTRAN, THINC	–	0.0	Coefficient is consistent with a Doppler defect of -0.67%Δk	0
	Uncontrolled RCCA bank withdrawal at power	LOFTRAN	0.0 and 0.374	–	Upper and lower curves of Figure 15.0.4-1	10%, 60%, and 100% of 1940
	RCCA misalignment	See Section 4.3	NA	–	Not applicable	1940
	Startup of an inactive reactor coolant pump at an incorrect temperature	LOFTRAN, FACTRAN THINC	0.374	–	Upper curve of Figure 15.0.4-1	1358
	Chemical and volume control system malfunction that results in a decrease in the boron concentration in the reactor coolant	NA	NA	–	Not applicable	0 and 1940
	Inadvertent loading and operation of a fuel assembly in an improper position	See Section 4.3	NA	–	Not applicable	1940

Table 15.0-2 (Sheet 4 of 4)

SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED

Section	Faults	Computer Codes Used	Reactivity Coefficients Assumed: Moderator Density ($\Delta k/gm/cm^3$)	Reactivity Coefficients Assumed: Moderator Temperature (pcm/°F)	Reactivity Coefficients Assumed: Doppler	Initial Thermal Power Output Assumed (MWt)
15.4	Spectrum of RCCA ejection accidents	TWINKLE, FACTRAN	Refer to sub-section 15.4.8	Refer to subsection 15.4.8	Coefficient consistent with a Doppler defect of -0.67% ΔK at BOC[b] and -0.63% ΔK at EOC	0 and 1978.8[a]
15.5	Increase in reactor coolant inventory					
	Inadvertent operation of the emergency core cooling system during power operation	LOFTRAN	0 and 0.374	–	Upper and lower curves of Figure 15.0.4-1	1940
15.6	Decrease in reactor coolant inventory					
	Inadvertent opening of a pressurizer safety valve and inadvertent operation of ADS	LOFTRAN	0.0	–	Lower curve of Figure 15.0.4-1	1940
	Steam generator tube failure	LOFTTR2	0.0	–	Lower curve of Figure 15.0.4-1	1978.8[a]
	LOCAs resulting from the spectrum of postulated piping breaks within the reactor coolant pressure boundary	NOTRUMP WCOBRA/TRAC	See subsection 15.6.5 references	–	See subsection 15.6.5 references	1971.7

Notes:

a. 102% of rated thermal power
b. BOC – Beginning of core cycle
EOC – End of core cycle

Table 15.0-3

NOMINAL VALUES OF PERTINENT PLANT PARAMETERS USED IN ACCIDENT ANALYSES

		Without RTDP[a]	
	RTDP With 10% Steam Generator Tube Plugging	**Without Steam Generator Tube Plugging**	**With 10% Steam Generator Tube Plugging**
Thermal output of NSSS (MWt)	1940.0	1940.0	1940.0
Core inlet temperature (°F)	533.40	531.9	532.8
Vessel average temperature (°F)	567.6	565.9	567.6
Reactor coolant system pressure (psia)	2250.0	2250.0	2250.0
Reactor coolant flow per loop (gpm)	9.66 E+04	9.71 E+04	9.48 E+04
Steam flow from NSSS (lbm/hr)	8.43 E+06	8.43 E+06	8.43 E+06
Steam pressure at steam generator outlet (psia)	794.0	801.0	794.0
Maximum steam moisture content (%)	0.10	0.10	0.10
Assumed feedwater temperature at steam generator inlet (°F)	435.0	435.0	435.0
Average core heat flux (Btu/-hr-ft^2)	1.43 E+05	1.43 E+05	1.43 E+05

Note:

a. Steady-state errors discussed in subsection 15.0.3 are added to these values to obtain initial conditions for transient analyses.

Table 15.0-4a (Sheet 1 of 2)

PROTECTION AND SAFETY MONITORING SYSTEM SETPOINTS AND TIME DELAY ASSUMED IN ACCIDENT ANALYSES

Function	Limiting Setpoint Assumed in Analyses	Time Delays (seconds)
Reactor trip on power range high neutron flux, high setting	118%	0.5
Reactor trip on power range high neutron flux, low setting	35%	0.5
High neutron flux, P-8	84%	0.9
Reactor trip on source range neutron flux reactor trip	Not applicable	0.5
Overtemperature ΔT	Variable (see Figure 15.0.3-1)	2.0
Overpower ΔT	Variable (see Figure 15.0.3-1)	2.0
Reactor trip on high pressurizer pressure	2460 psia	2.0
Reactor trip on low pressurizer pressure	1800 psia	1.2
Reactor trip on low reactor coolant flow in any cold leg	87% loop flow	1.45
Reactor trip on reactor coolant pump under speed	90%	0.767
Reactor trip on low steam generator narrow range level	45,000 lbm	2.0
High-2 steam generator level	100% of narrow range level span	2.0 (reactor trip) 0.0 (turbine trip) 2.0 (feedwater isolation)
Reactor trip on high-3 pressurizer water level	87% of span	2.0
PRHR actuation on low steam generator wide range level	25,000 lbm	2.0
"S" signal and steamline isolation on low T_{cold}	490°F	2.0

Table 15.0-4a (Sheet 2 of 2)

PROTECTION AND SAFETY MONITORING SYSTEM SETPOINTS AND TIME DELAY ASSUMED IN ACCIDENT ANALYSES

Function	Limiting Setpoint (Assumed in Analyses)	Time Delays (seconds)
"S" signal and steamline isolation on low steamline pressure	420 psia (with an adverse environment assumed) 540 psia (without an adverse environment assumed)	2.0
"S" signal on low pressurizer pressure	1700 psia	1.2
"S" signal on high-1 containment pressure	8 psig	1.2
Reactor coolant pump trip following "S"	—	15.0
PRHR actuation of high-3 pressurizer water level	87% of span	2.0 (plus 15.0-second timer delay)
Chemical and volume control system isolation on high-2 pressurizer water level	74% of span	2.0
Chemical and volume control system isolation on high-1 pressurizer water level coincident with "S" signal	30% of span	2.0
Boron dilution block on source range flux multiplication	1.6 over 50 minutes	10.0
ADS Stage 1 actuation on core makeup tank low level signal	67.5% of tank volume	60.0 (plus 20.0 seconds for isolation to open)
ADS Stage 4 actuation on core makeup tank low-low level signal	20% of tank volume	60.0 (plus 30.0 seconds for isolation valve to open)

Table 15.0-4b

LIMITING DELAY TIMES FOR EQUIPMENT ASSUMED IN ACCIDENT ANALYSES

Component	Time Delays (seconds)
Feedwater isolation valve closure, feedwater control valve closure, or feedwater pump trip	10 (maximum value for non-LOCA) 5 (maximum value for mass/energy)
Steamline isolation valve closure	5
Core makeup tank discharge valve opening time	20 (maximum) 10 (nominal value for best-estimate LOCA) 40 seconds (small-break LOCA value: follows a 20-second interval of no valve movement)
Chemical and volume control system isolation valve closure	10
PRHR discharge valve opening time	20 (maximum) 10 (nominal value for best-estimate LOCA) 1.0 second (small-break LOCA value: follows a 20-second interval of no valve movement)
Demineralized water transfer and storage system isolation valve closure time	20
Steam generator power-operated relief valve block valve closure	44
Automatic depressurization system (ADS) valve opening times	See Table 15.6.5-11

Table 15.0-5

DETERMINATION OF MAXIMUM POWER RANGE NEUTRON FLUX CHANNEL TRIP SETPOINT, BASED ON NOMINAL SETPOINT AND INHERENT INSTRUMENTATION UNCERTAINTIES

Nominal setpoint (% of rated power)		109
Calorimetric errors in the measurement of secondary system thermal power:		
Variable	**Accuracy of Measurement of Variable**	**Effect on Thermal Power Determination (% of Rated Power)**
Feedwater temperature	±3°F	
Steam pressure (small correction on enthalpy)	±6 psi	
Feedwater flow	±0.5% ΔP instrument span (two channels per steam generator)	
Assumed calorimetric error		2.0 (a)*
Radial power distribution effects on total ion chamber current		7.8 (b)*
Allowed mismatch between power range neutron flux channel and calorimetric measurement		2.0 (c)*
Instrumentation channel drift and setpoint reproducibility	0.4% of instrument span (120% power span)	0.84(d)*
Instrumentation channel temperature effects		0.48(e)*
*Total assumed error in setpoint (% of rated power): $[(a)^2 + (b)^2 + (c)^2 + (d)^2 + (e)^2]^{1/2}$		±8.4
Maximum power range neutron flux trip setpoint assuming a statistical combination of individual uncertainties (% of rated power)		118

Table 15.0-6 (Sheet 1 of 5)

PLANT SYSTEMS AND EQUIPMENT AVAILABLE FOR TRANSIENT AND ACCIDENT CONDITIONS

Incident	Reactor Trip Functions	ESF Actuation Functions	ESF and Other Equipment
Section 15.1			
Increase in heat removal from the primary system			
Feedwater system malfunctions that result in an increase in feedwater flow	Power range high flux, overtemperature ΔT, overpower ΔT, manual	High-2 steam generator level produced feedwater isolation and turbine trip	Feedwater isolation valves
Excessive increase in secondary steam flow	Power range high flux, overtemperature ΔT, overpower ΔT, manual	–	–
Inadvertent opening of a steam generator safety valve	Low pressurizer pressure, manual "S"	Low pressurizer pressure, low T_{cold}, low-2 pressurizer level	Core makeup tank, feedwater isolation valves, steam line stop valves
Steam system piping failure	"S," low pressurizer pressure, manual	Low pressurizer pressure, low compensated steam line pressure, high-1 containment pressure, low T_{cold}, manual	Core makeup tank, feedwater isolation valves, main steam line isolation valves (MSIVs), accumulators
Inadvertent operation of the PRHR	Overpower ΔT, power range high neutron flux, low pressurizer pressure, "S," manual	Low pressurizer pressure, low T_{cold}, low-2 pressurizer level	Core makeup tank

Table 15.0-6 (Sheet 2 of 5)

PLANT SYSTEMS AND EQUIPMENT AVAILABLE FOR TRANSIENT AND ACCIDENT CONDITIONS

Incident	Reactor Trip Functions	ESF Actuation Functions	ESF and Other Equipment
Section 15.2			
Decrease in heat removal by the secondary system			
Loss of external load/turbine trip	High pressurizer pressure overtemperature ΔT, overpower ΔT, manual	–	Pressurizer safety valves, steam generator safety valves
Loss of nonemergency ac power to the station auxiliaries	Steam generator low narrow range level, high pressurizer pressure, high pressurizer level, manual	Steam generator low narrow range level coincident with low startup water flow, steam generator low wide range level	PRHR, steam generator safety valves, pressurizer safety valves
Loss of normal feedwater flow	Steam generator low narrow range level, high pressurizer pressure, high pressurizer level, manual	Steam generator low narrow range level coincident with low startup water flow, steam generator low wide range level	PRHR, steam generator safety valves, pressurizer safety valves
Feedwater system pipe break	Steam generator low narrow range level, high pressurizer pressure, manual	Steam generator low wide range level, low steam line pressure, high-1 containment pressure	PRHR, core makeup tank, MSIVs, feedline isolation, pressurizer safety valves, steam generator safety valves

Table 15.0-6 (Sheet 3 of 5)

**PLANT SYSTEMS AND EQUIPMENT
AVAILABLE FOR TRANSIENT AND ACCIDENT CONDITIONS**

Incident	Reactor Trip Functions	ESF Actuation Functions	ESF and Other Equipment
Section 15.3			
Decrease in reactor coolant system flow rate			
Partial and complete loss of forced reactor coolant flow	Low flow, underspeed, manual	–	Steam generator safety valves, pressurizer safety valves
Reactor coolant pump shaft seizure (locked rotor)	Low flow, manual, high pressurizer pressure	–	Pressurizer safety valves, steam generator safety valves
Section 15.4			
Reactivity and power distribution anomalies			
Uncontrolled RCCA bank withdrawal from a subcritical or low power startup condition	Power range high flux (low setpoint), source range high flux, intermediate range high flux, manual	–	–
Uncontrolled RCCA bank withdrawal at power	Power range high flux, overtemperature ΔT, high pressurizer pressure, manual	–	Pressurizer safety valves, steam generator safety valves
RCCA misalignment	Overtemperature ΔT, manual	–	–
Startup of an inactive reactor coolant pump at an incorrect temperature	Power range high flux, low flow (P-8 interlock), manual	–	–

Table 15.0-6 (Sheet 4 of 5)

PLANT SYSTEMS AND EQUIPMENT AVAILABLE FOR TRANSIENT AND ACCIDENT CONDITIONS

Incident	Reactor Trip Functions	ESF Actuation Functions	ESF and Other Equipment
Section 15.4 (cont)			
Chemical and volume control system malfunction that results in a decrease in boron concentration in the reactor coolant	Source range high flux, overtemperature ΔT, manual	Source range flux doubling	Low insertion limit annunciators
Spectrum of RCCA ejection accidents	Power range high flux, high positive flux rate, manual	–	Pressurizer safety valves
Section 15.5			
Increase in reactor coolant inventory			
Inadvertent operation of the ECCS during power operation	High pressurizer pressure, manual, "safeguards" trip, high pressurizer level	High pressurizer level, low T_{cold}	Core makeup tank, pressurizer safety valves, chemical and volume control system isolation, PRHR
Section 15.6			
Decrease in reactor coolant inventory			
Inadvertent opening of a pressurizer safety valve or ADS path	Low pressurizer pressure, overtemperature ΔT, manual	Low pressurizer pressure	Core makeup tank, ADS, accumulator

Table 15.0-6 (Sheet 5 of 5)

PLANT SYSTEMS AND EQUIPMENT AVAILABLE FOR TRANSIENT AND ACCIDENT CONDITIONS

Incident	Reactor Trip Functions	ESF Actuation Functions	ESF and Other Equipment
Section 15.6 (cont)			
Steam generator tube rupture	Low pressurizer pressure, overtemperature ΔT, safeguards ("S"), manual	Low pressurizer pressure, high steam generator level, low steam line pressure	Core makeup tank, PRHR, steam generator safety and/or relief valves, MSIVs, radiation monitors (air removal, steamline, and steam generator blowdown), startup feedwater isolation, chemical and volume control system pump isolation, pressurizer heater isolation, steam generator power-operated relief valve isolation
LOCAs resulting from the spectrum of postulated piping breaks within the reactor coolant pressure boundary	Low pressurizer pressure, safeguards ("S"), manual	High-1 containment pressure, low pressurizer pressure	Core makeup tank, accumulator, ADS, steam generator safety and/or relief valves, PRHR, in-containment water storage tank (IRWST)

Table 15.0-7 (Sheet 1 of 2)

SINGLE FAILURES ASSUMED IN ACCIDENT ANALYSES

Event Description	Failure
Feedwater temperature reduction[a]	–
Excessive feedwater flow	One protection division
Excessive steam flow	One protection division
Inadvertent secondary depressurization	One core makeup tank discharge valve
Steam system piping failure	One core makeup tank discharge valve
Inadvertent operation of the PRHR	One protection division
Steam pressure regulator malfunction[b]	–
Loss of external load	One protection division
Turbine trip	One protection division
Inadvertent closure of main steam isolation valve	One protection division
Loss of condenser vacuum	One protection division
Loss of ac power	One PRHR discharge valve
Loss of normal feedwater	One PRHR discharge valve
Feedwater system pipe break	One PRHR discharge valve
Partial loss of forced reactor coolant flow	One protection division
Complete loss of forced reactor coolant flow	One protection division
Reactor coolant pump locked rotor	One protection division
Reactor coolant pump shaft break	One protection division
RCCA bank withdrawal from subcritical	One protection division
RCCA bank withdrawal at power	One protection division
Dropped RCCA, dropped RCCA bank	One protection division
Statically misaligned RCCA[c]	–
Single RCCA withdrawal	One protection division

Notes:

a. No protection action required
b. Not applicable to AP600
c. No transient analysis

Table 15.0-7 (Sheet 2 of 2)

SINGLE FAILURES ASSUMED IN ACCIDENT ANALYSES

Event Description	Failure
Inactive reactor coolant pump startup	One protection division
Flow controller malfunction[b]	–
Uncontrolled boron dilution	One protection division
Improper fuel loading[c]	–
RCCA ejection	One protection division
Inadvertent emergency core cooling system operation at power	One PRHR discharge valve
Increase in reactor coolant system inventory	One PRHR discharge valve
Inadvertent reactor coolant system depressurization	One protection division
Failure of small lines carrying primary coolant outside containment[c]	–
Steam generator tube rupture	Faulted steam generator power-operated relief valve fails open
Spectrum of LOCA	
Small breaks	One ADS Stage 4 valve
Large breaks	One core makeup tank discharge valve
Long-term cooling	One ADS Stage 4 valve

Notes:
a. No protection action required
b. Not applicable to AP600
c. No transient analysis

Table 15.0-8

NONSAFETY-RELATED SYSTEM AND EQUIPMENT USED FOR MITIGATION OF ACCIDENTS

	Event	Nonsafety-related System and Equipment
15.1.5	Feedwater system malfunctions that result in an increase in feedwater flow	Main feedwater pump trip
15.1.4	Inadvertent opening of a steam generator relief or safety valve	MSIV backup valves[1] Main steam branch isolation valves
15.1.5	Steam system piping failure	MSIV backup valves[1] Main steam branch isolation valves
15.2.7	Loss of normal feedwater	Pressurizer heater block
15.5.1	Inadvertent operation of the core makeup tanks during power operation	Pressurizer heater block
15.5.2	Chemical and volume control system malfunction that increases reactor coolant inventory	Pressurizer heater block
15.6.3	Steam generator tube rupture	Pressurizer heater block MSIV backup valves[1] Main steam branch isolation valves
15.6.5	Small-break LOCA	Pressurizer heater block

Note:
1. These include the turbine stop or control valves, the turbine bypass valves, and the moisture separator reheat steam supply control valve.

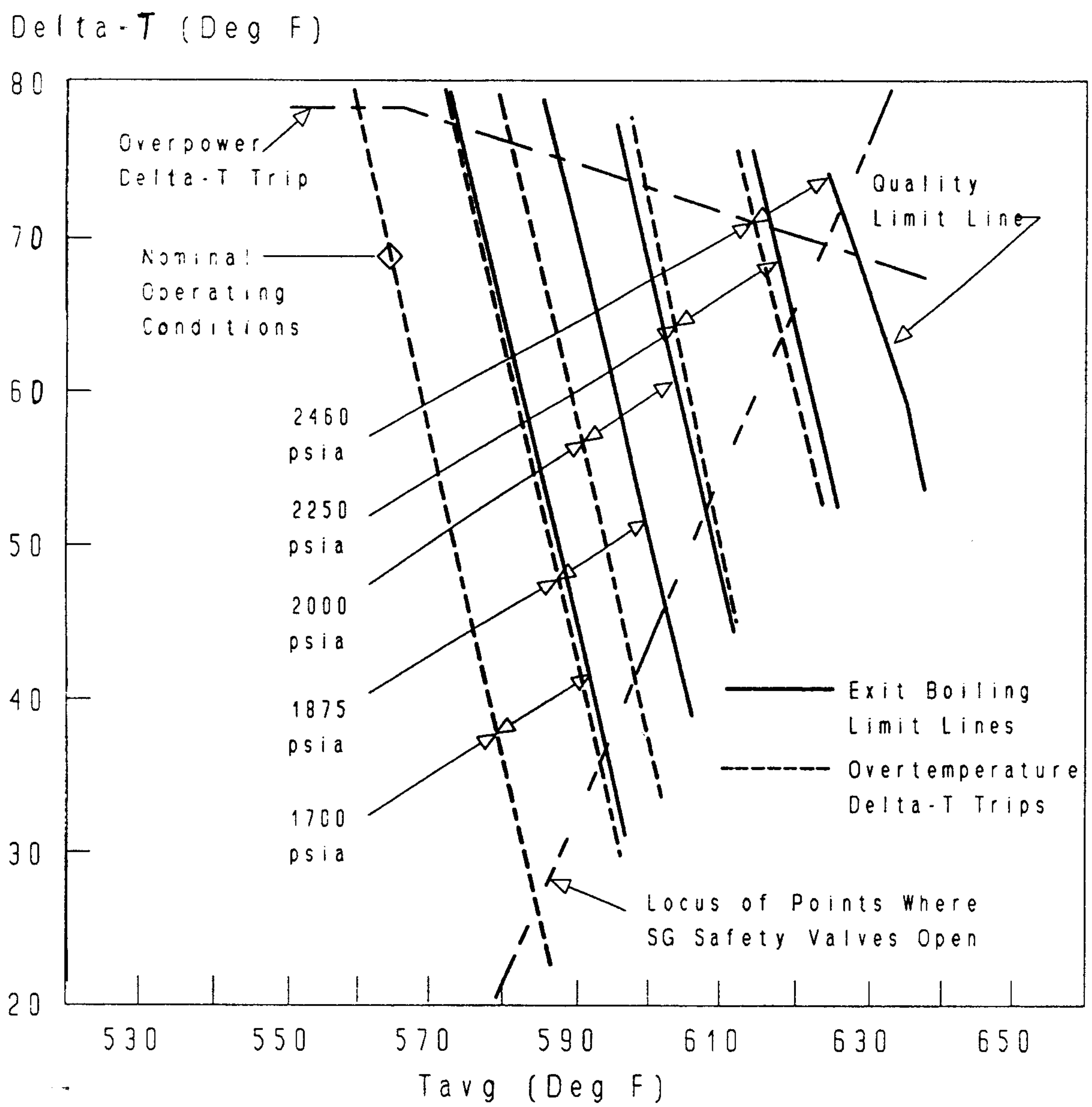
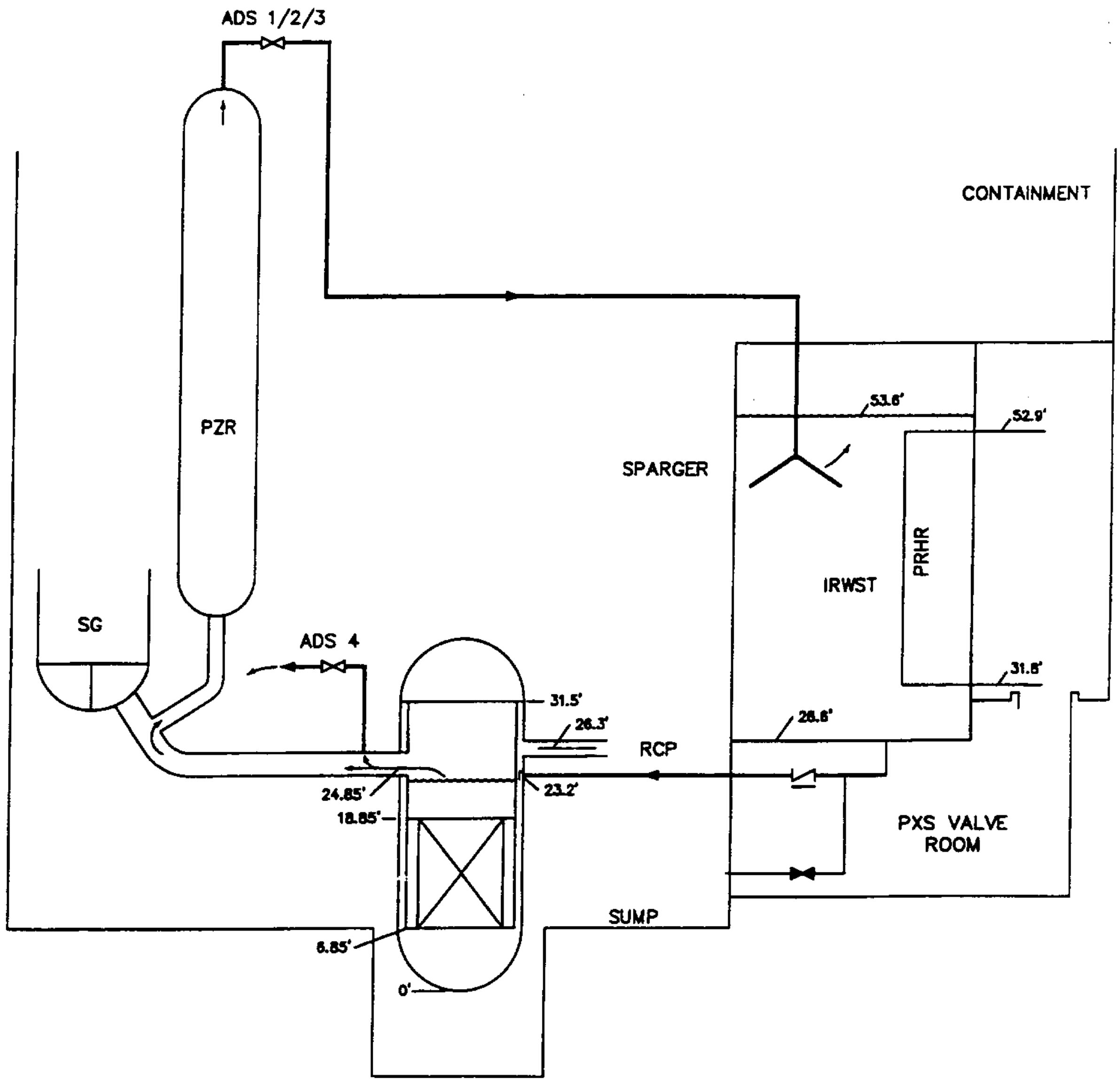


Figure 15.0.3-1

Overpower and Overtemperature ΔT Protection



Note: All elevations are relative to the bottom inside surface of the reactor vessel

Figure 15.0.3-2

AP600 Loop Layout

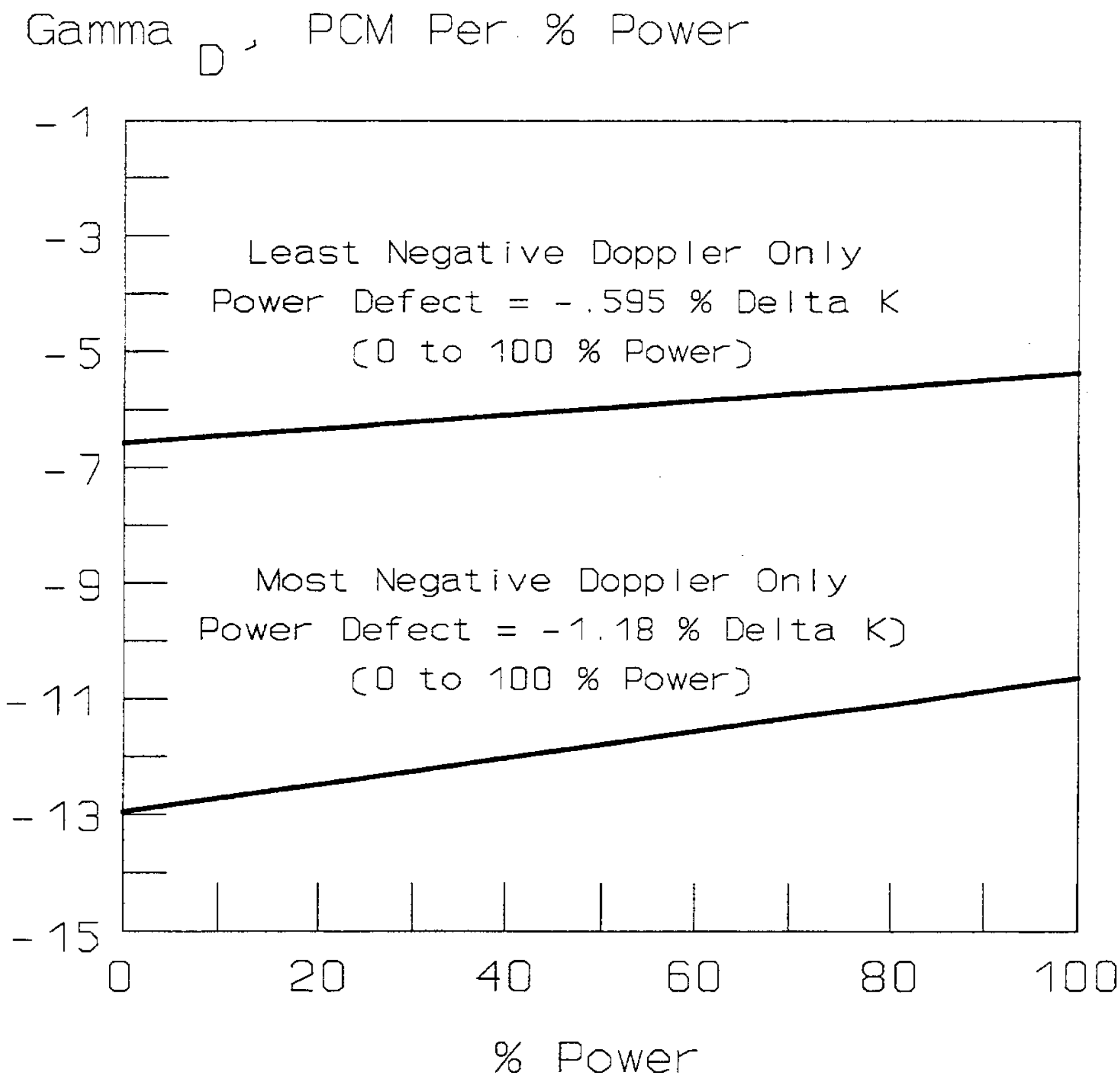


Figure 15.0.4-1

Doppler Power Coefficient used in Accident Analysis

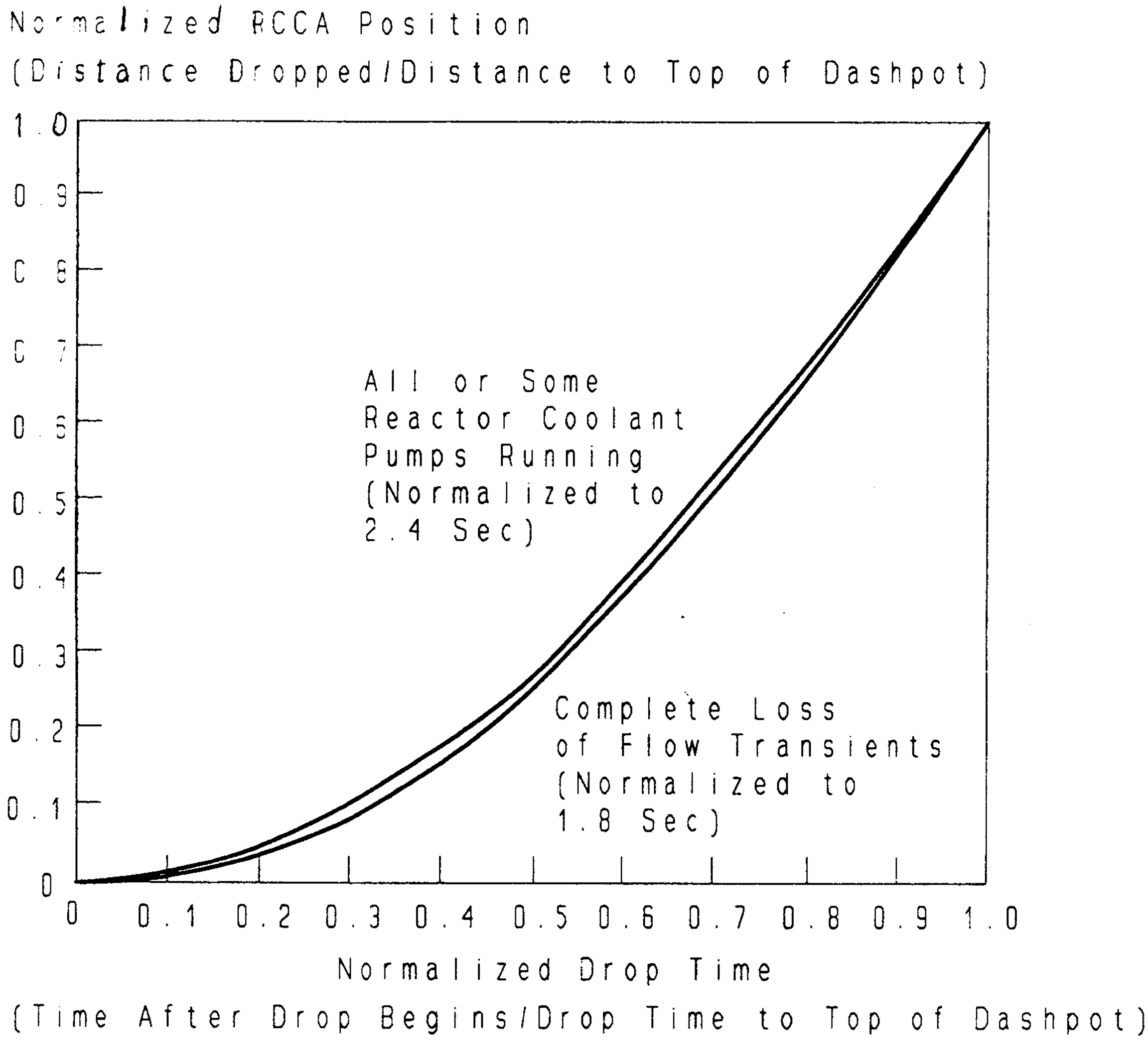


Figure 15.0.5-1

RCCA Position Versus Time to Dashpot

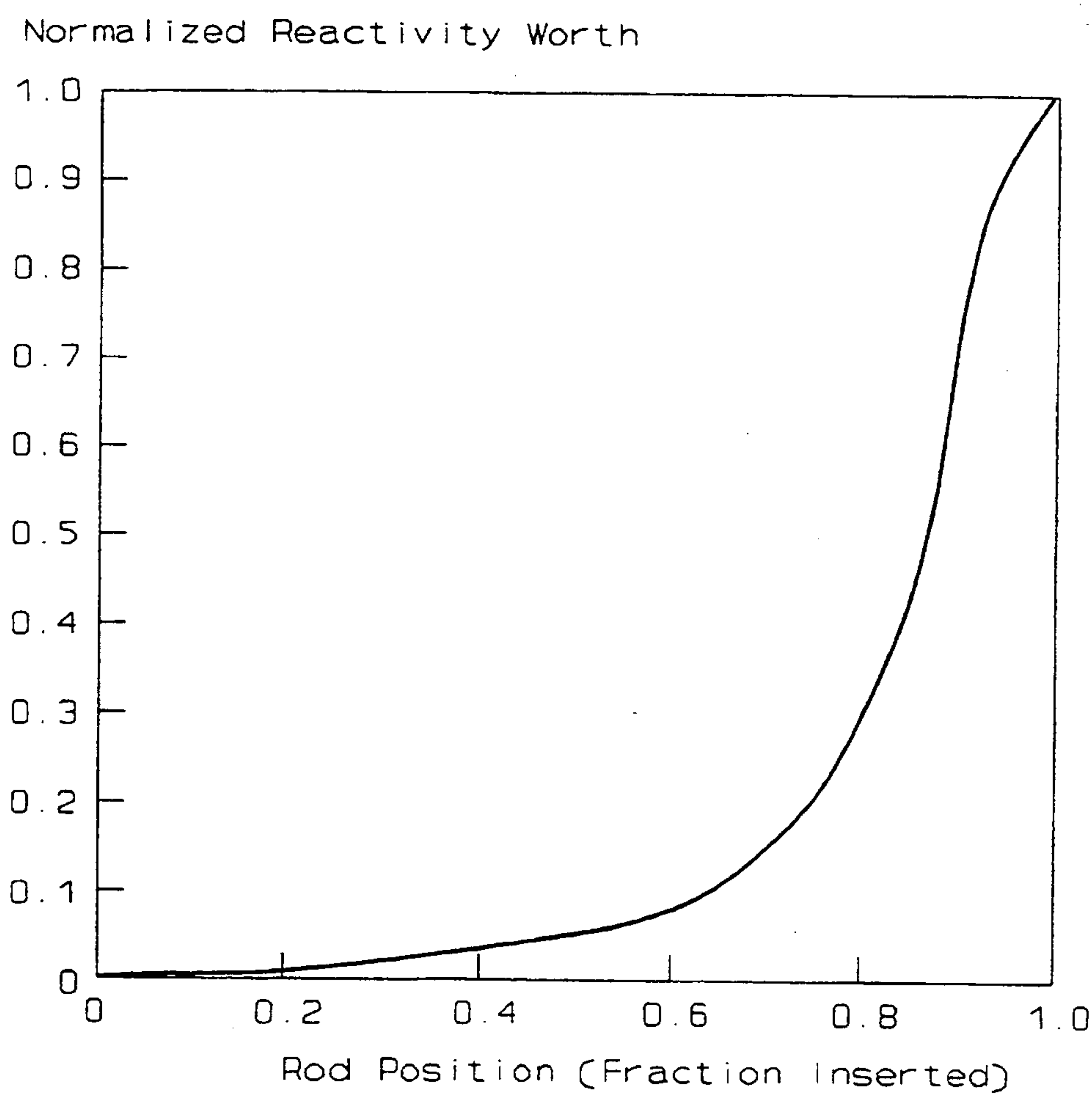


Figure 15.0.5-2

Normalized Rod Worth Versus Position

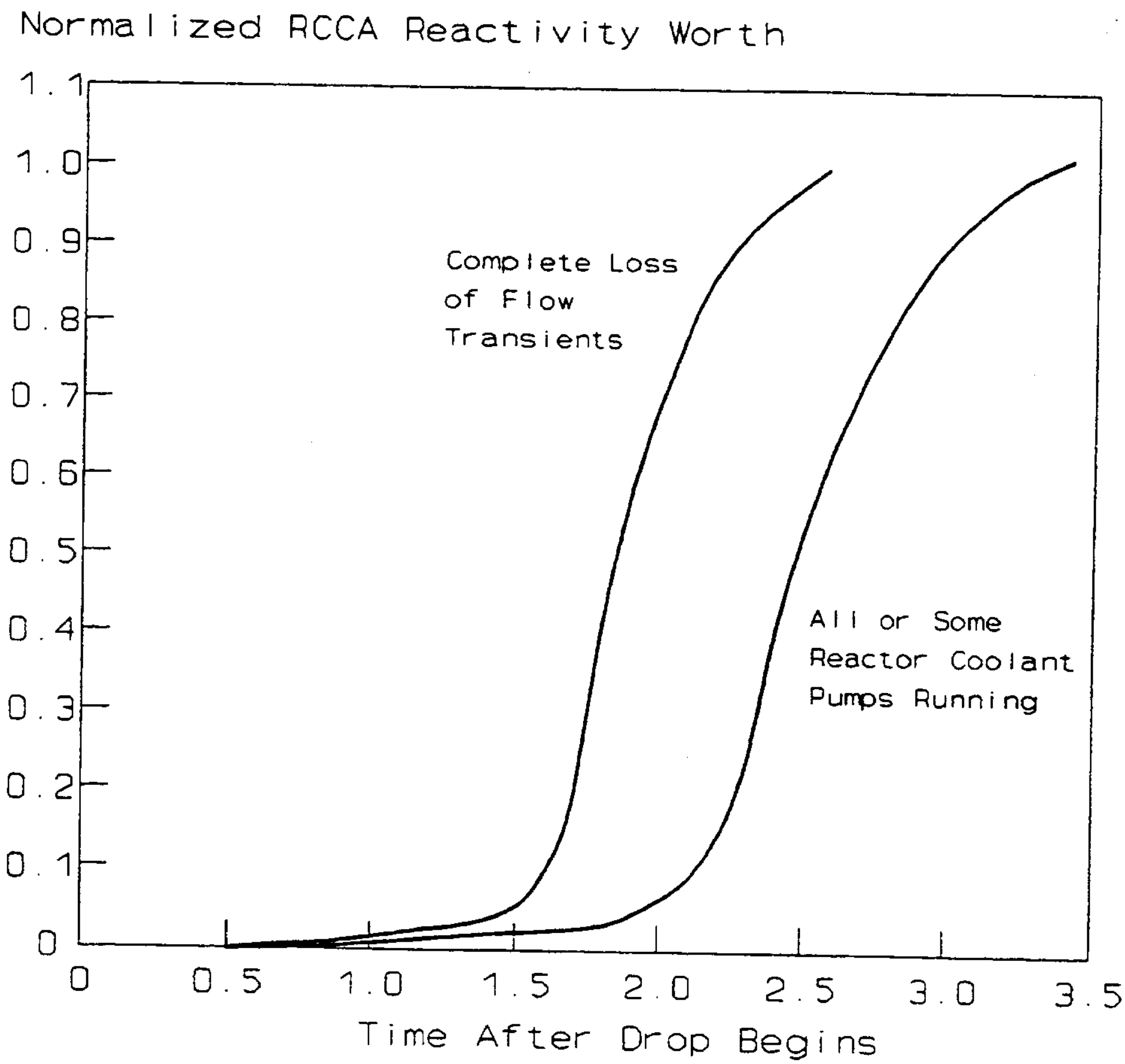


Figure 15.0.5-3

Normalized RCCA Bank
Reactivity Worth Versus Drop Time