4.4 THERMAL AND HYDRAULIC DESIGN

This section addresses the original plant thermal hydraulic design (number of assemblies, core power and flow, etc), the compatibility of co-resident fuel designs and the relative stability of reload cores.

4.4.1 DESIGN BASES

4.4.1.1 Safety Design Bases

Thermal-hydraulic design of the core shall establish:

(1) Actuation limits for the devices of the nuclear safety systems such that no fuel damage occurs as a result of moderate frequency transient events. For example, the Minimum Critical Power Ratio (MCPR) operating limit is specified such that at least 99.9 percent of the fuel rods in the core are not expected to experience boiling transition during the most severe moderate (Per Regulatory Guide 1.70 Revision 2) frequency transient events.

(2) The thermal-hydraulic safety limits for use in evaluating the safety margin relating the consequences of fuel barrier failure to public safety.

(3) That the nuclear system exhibits no inherent tendency toward divergent or limit cycle oscillations which would compromise the integrity of the fuel or nuclear system process barrier.

4.4.1.2 Power Generation Design Bases

The thermal-hydraulic design of the core shall provide the following operational characteristics:

(1) The ability to achieve rated core power output throughout the design life of the fuel without sustaining premature fuel failure.

(2) Flexibility to adjust core output over the range of plant load and load maneuvering requirements in a stable, predictable manner without sustaining fuel damage.

4.4.1.3 Requirements for Steady-State Conditions

Steady-State Limits

For purposes of maintaining adequate thermal margin during normal steady-state operation, the minimum critical power ratio must not be less than the required MCPR operating limit, and the maximum linear heat generation rate (LHGR) must be maintained below the LHGR limit. This does not specify the operating power nor does it specify peaking factors. These parameters are determined subject to a number of constraints including the thermal limits given previously. The core and fuel design basis for steady-state operation, i.e., MCPR and LHGR limits, have been defined to provide margin between the steady-state operating conditions and any fuel damage condition to accommodate uncertainties and to assure that no fuel damage results even during the worst anticipated transient condition at any time in life.

Steady-state limits also exist on the maximum average planar linear heat generation rate (MAPLHGR). The MAPLHGR limits protect against violation of the ECCS acceptance criteria during a Loss of Coolant Accident and are derived from the LOCA analyses described in Section 6.3.

4.4.1.4 Requirements for Transient Conditions

Transient Limits

The transient thermal limits are established such that no fuel damage is expected to occur during the most severe moderate frequency transient event. Fuel damage is defined as perforation of the cladding that permits release of fission products. Mechanisms that cause fuel damage in reactor transients are:

(1) Severe overheating of fuel cladding caused by inadequate cooling, and

(2) Fracture of the fuel cladding caused by relative expansion of the uranium dioxide pellet inside the fuel cladding.

For design purposes, the transient limit requirement relating to cladding overheating is met if at least 99.9 percent of the fuel rods in the core do not experience boiling transition during any moderate frequency transient event. No fuel damage would be expected to occur even if a fuel rod actually experienced a boiling transition.

A value of 1 percent plastic strain of Zircaloy cladding is conservatively defined as the limit below which fuel damage from overstraining the fuel cladding is not expected to occur. The linear heat generation rate required to cause this amount of cladding strain depends on the fuel type and burnup. The linear heat generation rates are discussed on a fuel type specific basis in Section 4.2.3.

4.4.1.5 Summary of Design Bases

In summary, the steady-state operating limits have been established to assure that the design basis is satisfied for the most severe moderate frequency transient event. There is no steady-state design overpower basis. An overpower which occurs during an incident of a moderate frequency transient event must meet the plant transient MCPR limit and 1% plastic strain limit. Demonstration that the transient limits are not exceeded is sufficient to conclude that the design basis is satisfied.

The MCPR, MAPLHGR, and LHGR limits are sufficiently general so that no other limits need to be stated. For example, cladding surface temperatures will always be maintained within 10 to 15^{o}F of the coolant temperature as long as the boiling process is in the nucleate regime. The cladding and fuel bundle integrity criterion is assured as long as MCPR, MAPLHGR, and LHGR limits are met. There are no additional design criteria on coolant void fraction, core coolant flow-velocities, or flow distribution, nor are they needed. The coolant flow velocities and void fraction become constraints upon the mechanical and physics design of reactor components and are partially constrained by stability and control requirements.

4.4.2 DESCRIPTION OF THERMAL-HYDRAULIC DESIGN OF THE REACTOR CORE

4.4.2.1 Summary Comparison

An evaluation of plant performance from a thermal and hydraulic standpoint is provided in Subsection 4.4.3.

A tabulation of thermal and hydraulic parameters of the core is given in Table 4.4-1.

4.4.2.2 Critical Power Ratio

There are three different types of boiling heat transfer to water in a forced convection system: nucleate boiling, transition boiling, and film boiling. Nucleate boiling, at lower heat transfer rates, is an extremely efficient mode of heat transfer, allowing large quantities of heat to be transferred with a very small temperature rise at the heated wall. As heat transfer rate is increased the boiling heat transfer surface alternates between film and nucleate boiling, leading to fluctuations in heated wall temperatures. The point of departure from the nucleate boiling region into the transition boiling region is called the boiling transition. Transition boiling begins at the critical power and is characterized by fluctuations in cladding surface temperature. Film boiling occurs at the highest heat transfer rates; it begins as transition boiling comes to an end. Film boiling heat transfer is characterized by stable wall temperatures which are higher than those experienced during nucleate boiling.

4.4.2.2.1 Boiling Correlations

The occurrence of boiling transition is a function of the fluid enthalpy, mass flow rate, pressure, flow geometry and assembly power distribution. Framatome Inc. has conducted extensive experimental investigations of these parameters. These parametric studies encompass the entire design range of these variables. The SPCB critical power correlation, Reference 4.4-58, is used for ATRIUM-10 fuel, and the ACE critical power correlation, Reference 4.4-60, is used for ATRIUM-11.

The figure of merit used for reactor design and operation is the Critical Power Ratio (CPR). This is defined as the ratio of the bundle power at which boiling transition occurs to the bundle power at the reactor condition of interest (i.e., the ratio of critical bundle power to operating bundle power). In this definition, the critical power is determined at the same mass flux, inlet temperature, and pressure which exist at the specified reactor condition.

4.4.2.3 Thermal Operating Limits

The limiting constraints in the design of the reactor core are stated in terms of the MCPR, MAPLHGR, and LHGR limits. The design philosophy used to assure that these limits are met involves the selection of one or more power distributions which are more limiting than expected operating conditions and subsequent verification that under these more stringent conditions, the design limits are met. Therefore, the "design power distributions" represent extreme conditions of power. Use of these power distributions in the analyses is a fair and stringent test of the operability of the reactor as designed to comply with the foregoing limits. Expected operating conditions are less severe than those represented by the design power distributions which give the MCPR, MAPLHGR and LHGR limits.

However, it must be established that operation with a less severe power distribution is not a necessary condition for the safety of the reactor. Because there are an infinite number of operating reactor states which can exist (with variations in rod patterns, time in cycle, power level, distribution, flow etc.) which are within the design constraints, it is not possible to determine them all. However, constant monitoring of operating conditions using the available plant measurements can ensure compliance with design objectives.

4.4.2.3.1 Design Power Distribution

Thermal design of the reactor--including the selection of the core size and effective heat transfer area, the design steam quality, the total recirculation flow, the inlet subcooling, and the specification of internal flow distribution -- was performed by the NSSS vendor and is based on the concept and application of a design power distribution. The design power distribution was an appropriately conservative representation of the most limiting thermal operating state at rated conditions and included design allowances for the combined effects (on the fuel rod, and the fuel assembly heat flux and temperature) of the gross and local steady-state power density distributions and adjustments of the control rods.

4.4.2.3.2 Design Linear Heat Generation Rates

The maximum and core average linear heat generation rates are shown in Table 4.4-1. The maximum linear heat generation rate at any location is the average linear heat generation rate at a given axial location multiplied by the total peaking factor of that location.

Fuel type specific LHGR limits and MAPLHGR limits are provided in the Core Operating Limits Report for each unit (see FSAR section 16.3, Technical Requirements Manuals).

4.4.2.4 Void Fraction Distribution

Typical core average and maximum exit void fractions in the core at rated condition are given in Table 4.4-2. The axial distribution of core void fractions for the average radial channel and the maximum radial channel (end of node value) are also given in Table 4.4-2. Similar distributions for steam quality are provided in Table 4.4-3. The core average axial power distributions used to produce these tables are given in Table 4.4-2a.

4.4.2.5 Core Coolant Flow Distribution and Orificing Pattern

Correct distribution of core coolant flow among the fuel assemblies is accomplished by the use of an accurately calibrated fixed orifice at the inlet of each fuel assembly. The orifices are located in the fuel support piece. They serve to control the flow distribution and, hence, the coolant conditions within prescribed bounds throughout the design range of core operation.

The sizing and design of the orifices ensure stable flow in each fuel assembly during all phases of operation at normal operating conditions.

The core is divided into two orificed flow zones. The outer zone is a narrow, reduced-power region around the periphery of the core. The inner zone consists of the core center region. No other control of flow and steam distribution, other than that incidentally supplied by adjusting the power distribution with the control rods, is used or needed. The orifices can be changed during refueling, if necessary.

Design core flow distribution calculations were performed by the NSSS vendor using a design power distribution which consists of a hot and average powered assembly in each of the two orifice zones. The design bundle power and resulting relative flow distribution are given in Table 4.4-4.

The flow distribution to the fuel assemblies is calculated on the assumption that the pressure drop across all fuel assemblies is the same. This assumption has been confirmed by measuring the flow distribution in a modern boiling water reactor as reported in References 4.4-2 and 4.4-36.

There is reasonable assurance, therefore, that the calculated flow distribution throughout the core is in close agreement with the actual flow distribution of an operating reactor.

The use of the design power distribution discussed previously ensures the orificing chosen covers the range of normal operation. The expected shifts in power production during core life are less severe and are bounded by the design power distribution.

4.4.2.6 Core Pressure Drop and Hydraulic Loads

The pressure drop across various core components under the steady state design conditions is included in Table 4.4-1. Analyses for the most limiting conditions, the recirculation line break and the steam line break are reported in Chapter 15.

The components of bundle pressure drop considered are friction, local elevation and acceleration. Reference 4.4-43 presents the methodology and constitutive relationships used by Framatome for the calculation of pressure drop in BWR fuel assemblies. These are implemented in the XCOBRA computer code which is used to perform steady state thermal-hydraulic analyses, Reference 4.4-49.

The thermal hydraulic loads on the fuel rods during steady-state operation, transient, and accident conditions are negligible, primarily because of the channel confinement, thereby resulting in small cross flow between rods (i.e., essentially constant pressure at any given elevation in the fuel bundle).

The loads (i.e. horizontal) across the control blades are minimal or negligible primarily due to the flat interchannel velocity profile as given in Reference 4.4-13.

4.4.2.6.1 Friction Pressure Drop

Friction pressure drop is calculated using the model relation

$$\Delta P_f = \frac{w^2}{2g\rho} \frac{fL}{D_H A^2_{ch}} \phi^2_{TPF}$$

where

ΔP_f	=	friction pressure drop, psi
w	=	mass flow rate,
g	=	acceleration of gravity,
ρ	=	water density,
D_H	=	channel hydraulic diameter,
A_{ch}	=	channel flow area,
L	=	length,
f	=	friction factor, and
ϕ^2_{TPF}	=	two-phase friction multiplier

This basic model is similar to that used throughout the nuclear power industry. The formulation for the two-phase multiplier used by Framatome is the correlation determined by Jones, Reference 4.4-43, which represents a mass velocity correction to the Martinelli-Nelson correlation, Reference 4.4-3. Significant amounts of friction pressure drop data in multirod geometries representative of modern BWR plant fuel bundles have been taken and both the friction factor and two-phase multipliers have been correlated on a best-fit basis using the above pressure drop formulation.

4.4.2.6.2 Local Pressure Drop

The local pressure drop is defined as the irreversible pressure loss associated with an area change such as the orifice, lower tie plates, and spacers of a fuel assembly.

The general local pressure drop model is similar to the friction pressure drop and is:

$$\Delta P_L = \frac{w^2}{2g\rho} \frac{K}{A_2^2} \phi^2_{TPL}$$

where

ΔP_L	=	local pressure drop, psi,
K	=	local pressure drop loss coefficient,
A_2	=	reference area for local loss coefficient, and
ϕ^2_{TPL}	=	two-phase local multiplier,

and w and g are defined the same as for friction. This basic model is similar to that used throughout the nuclear power industry. The two-phase multiplier used by Framatome is given by the ratio of the saturated water and two-phase mixture densities. Tests are performed in both single and two-phase flow to arrive at best-fit design values for spacer and upper tie plate pressure drop. The range of test variables is specified to include the range of interest to boiling water reactors. New data are taken whenever there is a significant design change to ensure the most applicable methods are in use at all times. For ATRIUM-10 fuel, a simple multiplier based on local quality is also used to calculate the spacer pressure drop for the two-phase conditions, Reference 4.4-45.

4.4.2.6.3 Elevation Pressure Drop

The elevation pressure drop is based on the well-known relation

$$\Delta P_E = \rho g L$$

$$\rho = \rho_f (1 - \overline{\alpha}) + \rho_g \overline{\alpha}$$

where

ΔPE	=	elevation pressure drop, psi
L	=	incremental length
ρ	=	average water density
α	=	average void fraction over the length L
ρf, ρg	=	saturated water and vapor density, respectively.
g	=	acceleration of gravity

4.4.2.6.4 Acceleration Pressure Drop

A reversible pressure change occurs when an area change is encountered, and an irreversible loss occurs when the fluid is accelerated through the boiling process. The basic formulation for the reversible pressure change resulting from a flow area change is given by:

$$\Delta P_{ACC} = (1 - \phi^2)\,\frac{W^2}{2 g \rho A_2^2}\,;\;\; \phi = \frac{A_2}{A_1}$$

where

ΔP_{ACC}	=	acceleration pressure drop,
A_2	=	final flow area,
A_1	=	initial flow area,

and other terms are as previously defined. The basic formulation for the acceleration pressure change due to density change is:

$$\Delta P_{ACC} = \frac{W^2}{g A_{ch}^2}\left[\left(\frac{1}{\rho_M}\right)_{out} - \left(\frac{1}{\rho_M}\right)_{in}\right]$$

where

$$\frac{1}{\rho_M} = \frac{x^2}{\alpha \rho_g} + \frac{(1-x)^2}{(1-\alpha)\rho_1}$$

ρ_M	=	momentum density,
x	=	steam quality,
ρ_1	=	saturated liquid density

and other terms are as previously defined. The total acceleration pressure drop in boiling water reactors is on the order of a few percent of the total pressure drop.

4.4.2.7 Correlation and Physical Data

Substantial amounts of physical data support the pressure drop and thermal hydraulic loads discussed in Subsection 4.4.2.6. Correlations have been developed to fit these data to the formulations discussed.

4.4.2.7.1 Pressure Drop Correlations

Pressure drop data in multirod geometries representative of modern BWR plant fuel bundles has been correlated to the friction factor and two-phase multipliers on a best fit basis using the pressure drop formulations reported in Subsections 4.4.2.6.1 and 4.4.2.6.2 Framatome's pressure drop methodology is described in Reference 4.4-43.

New data is taken whenever there is a significant design change. Applicability of the pressure drop correlations is confirmed by full scale prototype flow tests. Pressure drop tests for the ATRIUM-10 fuel designs is reported in Reference 4.4-45. The pressure drop tests for the ATRIUM-11 fuel designs were performed in accordance with the methodology described in Reference 4.4-43.

4.4.2.7.2 Void Fraction Correlation

The void fraction is determined by a Zuber-Findlay model with constitutive relations as supplied by Ohkawa and Lahey, Reference 4.4-43.

4.4.2.7.3 Heat Transfer Correlation

The Jens-Lottes (Reference 4.4-5) wall superheat equation is used in fuel design to determine the cladding-to-coolant heat transfer coefficients for nucleate boiling.

4.4.2.8 Thermal Effects of Operational Transients

The evaluation of the core's capability to withstand the thermal effects resulting from anticipated operational transients is covered in Chapter 15.

4.4.2.9 Uncertainties in Estimates

Uncertainties in thermal-hydraulic parameters are considered in the statistical analysis which is performed to establish the fuel cladding integrity safety MCPR limit such that at least 99.9% of the fuel rods in the core are expected not to experience boiling transition during any moderate frequency transient event. The statistical model and analytical procedure are described in References 4.4-41 and 4.4-42.

The MCPR safety limit is determined by a statistical convolution of the uncertainties associated with the calculation of thermal margin. Some uncertainties are fuel related and others are characteristics of the reactor system. The uncertainties which are considered are shown in Table 4.4-6.

4.4.2.10 Flux Tilt Considerations

The inherent design characteristics of the BWR are particularly well suited to handle perturbations due to flux tilt. The stabilizing nature of the moderator void coefficient effectively damps oscillations in the power distribution. In addition to this damping, the incore instrumentation system and the associated on-line computer provide the operator with prompt and reliable power distribution information. Thus, the operator can readily use control rods or other means to effectively limit the undesirable effects of flux tilting. Because of these features and capabilities, it is not necessary to allocate a specific peaking factor margin to account for flux tilt. If for some reason, the power distribution could not be maintained within normal limits using control rods, then the operating power limits would have to be reduced as prescribed in the Plant Technical Specifications. The power distributions will be maintained such that the operating limits given in the Core Operating Limits Report will not be exceeded.

4.4.2.11 Crud Deposition

In general, the CPR is not affected as crud accumulates on fuel rods (References 4.4-34 and 4.4-35). Therefore, no modifications to the critical power correlation are made to account for crud deposition. The effect of crud deposition on pressure drop and flow is to increase the pressure drop and decrease the flow. An increase in crud deposition for high exposure assemblies would tend to reduce the flow in these assemblies and increase the flow in low exposure, CPR limiting assemblies. No credit is taken for the increase in CPR margin due to crud deposition.

The effects of crud deposition are included in thermal and rod internal pressure calculations, Reference 4.4-47.

4.4.3 DESCRIPTION OF THE THERMAL AND HYDRAULIC DESIGN OF THE REACTOR COOLANT SYSTEM

The thermal and hydraulic design of the reactor coolant system is described in this subsection.

4.4.3.1 Plant Configuration Data

4.4.3.1.1 Reactor Coolant System Configuration

The reactor coolant system is described in Section 5.4 and shown in isometric perspective in Figure 5.4-1. The piping sizes, fittings and valves are listed in Table 5.1-1.

4.4.3.1.2 Reactor Coolant System Thermal Hydraulic Data

Table 5.1-1 provides design temperatures, pressures and flow rates for the reactor coolant system and its components.

4.4.3.1.3 Reactor Coolant System Geometric Data

Coolant volumes of regions and components within the reactor vessel are shown in Figure 5.1-2.

Table 4.4-8 provides the flow path length, height and liquid level, minimum elevations, and minimum flow areas for each major flow path volume within the reactor vessel and recirculation loops of the reactor coolant systems.

Table 4.4-9 provides the lengths and sizes of all safety injection lines to the reactor coolant system.

4.4.3.2 Operating Restrictions on Pumps

Expected recirculation pump performance curves are shown in Figure 5.4-3. These curves are valid for all conditions with a normal operating range varying from approximately 20% to 115% of rated pump flow.

The pump characteristics including considerations of NPSH requirements are the same for the conditions of two pump and one pump operation as described in Subsection 5.4.1. Subsection 4.4.3.3 gives the limits imposed on the recirculation pumps by cavitation, pump loads, bearing design, flow starvation, and pump speed.

4.4.3.3 Power-Flow Operating Map

4.4.3.3.1 Limits for Normal Operation

A boiling water reactor must operate with certain restrictions because of pump net positive suction head (NPSH), overall plant control characteristics, core thermal power limits, etc. A representation of a simplified power-flow map for the power range of operation is shown in Figure 4.4-5. The actual power-flow maps for Units 1 and 2 are found in the respective COLR, FSAR Section 16.3. The nuclear system equipment, nuclear instrumentation, and the reactor protection system, in conjunction with operating procedures, maintain operations within the area of this map for normal operating conditions. The boundaries on this map are as follows:

Natural Circulation Line: The operating state of the reactor moves along this line for the normal control rod withdrawal sequence in the absence of recirculation pump operation.

30 Percent Recirculation Pump Constant Speed Line: Startup operations of the plant are normally carried out with the recirculation pumps operating at approximately 30 percent speed. The operating state for the reactor follows this line for the normal control rod withdrawal sequence.

Rated Flow Control Line: The rated flow control line (100% rod line) passes through 100 percent power at 108 Mlb/hr flow. The operating state for the reactor follows this line for recirculation flow changes with a fixed control rod pattern. The line is based on full power constant xenon concentration.

Cavitation Protection Line: This line (minimum power line) results from the recirculation pump and jet pump NPSH requirements. The recirculation pumps are automatically switched to 30 percent speed when the feedwater flow drops below a preset value.

Note that an actual power-flow map will contain stability related regions. The actual Unit 1 and Unit 2 power-flow maps are included in their respective COLR, FSAR Section16.3.

4.4.3.3.1.1 Performance Characteristics

Other Power Flow Operating Map performance characteristics are:

Recirculation Pump Constant Speed Line: This line shows the change in flow associated with power changes while maintaining constant recirculation pump speed.

Constant Rod Lines: These lines show the change in power associated with flow changes while maintaining constant control rod position (e.g. 80% rod line).

4.4.3.3.2 Regions of the Power Flow Map

For normal operating conditions, the nuclear system equipment, nuclear instrumentation, and the reactor protection system, in conjunction with operating procedures, maintain operation outside the exclusion areas of the power flow map. Main regions of the map are discussed below to clarify operational capabilities.

Region A - This is the transition region between natural circulation operation and 30% pump speed operation. Operation at less than 30% pump speed with two recirculation loops results in flow instabilities (causing flow induced vibrations), therefore the recirculation pumps are not continually operated below 30% pump speed. Normal startup is along the 30% pump speed boundary of this region.

Region B - This region represents the normal operating zone of the map where power changes can be made either by control rod movement or by core flow changes by changing recirculation pump drive speed.

Region C - This is the low power area of the map where cavitation can be expected in the recirculation pumps and in the jet pumps. Operation within this region is precluded by system interlocks which runback the recirculation pumps to 30% speed whenever feedwater flow is less than a preset value (typically 20% of rated).

4.4.3.4 Temperature-Power Operating Map (PWR)

Not Applicable.

4.4.3.5 Load Following Characteristics

The following simple description of boiling water reactor operation with recirculation flow control summarizes the principal modes of normal power range operation. Assuming the plant to be initially hot with the reactor critical, full power operation can be approached by initially moving along the two pump 30% speed line until power is at least above the minimum power line (cavitation interlock) of Region C (see Figure 4.4-5). Note, other low power restrictions may apply as a result of cycle specific transient analyses. This initial sequence may be achieved with control rod withdrawal and manual, individual recirculation pump control. Individual pump startup procedures are provided which achieve 30 percent of full pump speed in each loop. Power, steam flow, and feedwater flow are increased as control rods are manually withdrawn. An interlock prevents low power-high recirculation flow combinations which create recirculation pump and jet pump NPSH problems.

Reactor power increases as the operating state moves to the right on Figure 4.4-5 as the operator manually increases recirculation flow in each loop. Eventually, the operator can switch to simultaneous recirculation pump control. Thermal output can then be increased by either control rod withdrawal or recirculation flow increase. Both combinations are required to achieve full power. The operating map is shown in Figure 4.4-5 with the designated flow control range expected.

The curve labeled “100% Xe Rod Line” (i.e., the “Rated Flow Control Line”) represents a typical steady state power flow characteristic for a fixed rod pattern. It is affected by xenon, core leakage flow assumptions, and reactor vessel pressure variations.

Normal power range operation is along the "Rated Flow Control Line", below the APRM Rod Block Trip Setpoint, and below 100% rated power.

The large negative operating reactivity and power coefficients, which are inherent in the boiling water reactor, provide important advantages as follows:

(1) Good load following with well damped behavior and little undershoot or overshoot in the heat transfer response.

(2) Load following with recirculation flow control.

(3) Strong damping of spatial power disturbances.

The reactor power level can be controlled by flow control over approximately 35 percent of the power level on the rated rod line. Load following is accomplished by varying the recirculation flow to the reactor. This method of power level control takes advantage of the reactor negative void coefficient. To increase reactor power, it is necessary to increase the recirculation flow rate which sweeps some of the voids from the moderator, causing an increase in core reactivity. As the reactor power increases, more steam is formed and the reactor stabilizes at a new power level with the transient excess reactivity balanced by the new void formation. No control rods are moved to accomplish this power level change. Conversely, when a power reduction is required, it is necessary only to reduce the recirculation flow rate. When this is done, more voids in the moderator automatically decrease the reactor power level to that commensurate with the new recirculation flow rate. Again, no control rods are moved to accomplish the power reduction.

Varying the recirculation flow rate (flow control) is more advantageous, relative to load-following, than using control rod positioning. Flow variations perturb the reactor uniformly in the horizontal planes and ensure a flatter power distribution and reduced transient allowances. As flow is varied, the power and void distributions remain approximately constant at the steady state end points for a wide range of flow variations. After adjusting the power distribution by positioning the control rods at a reduced power and flow, and taking into account any effects due to Xe variations, the operator can then bring the reactor to rated conditions by increasing flow, with the assurance that the power distribution will remain approximately constant. Section 7.7 describes how recirculation flow is varied.

4.4.3.6 Thermal and Hydraulic Characteristics Summary Table

The thermal hydraulic characteristics are provided in Table 4.4-1 for the core and tables of Sections 5.1 and 5.4 for other portions of the reactor coolant system.

4.4.4 EVALUATION

The design basis employed for the thermal and hydraulic characteristics incorporated in the core design, in conjunction with the plant equipment characteristics, nuclear instrumentation, and the reactor protection system, is to require that no fuel damage occur during normal operation or

during abnormal operation transients. Demonstration that the applicable thermal-hydraulic limits are not exceeded is given by analyses.

4.4.4.1 Critical Power

The ACE and SPCB critical power correlations are utilized in thermal-hydraulic evaluations. These correlations are discussed in more detail in Subsection 4.4.2.2.1.

4.4.4.2 Core Hydraulics

Core hydraulic models and correlations are discussed in Subsections 4.4.2.6, 4.4.2.7, and 4.4.4.5.

4.4.4.3 Influence of Power Distributions

The influence of power distributions on the thermal-hydraulic design is discussed in Reference 4.4-1, Appendix V for the initial core. The influence of power distribution is included in the cycle specific licensing calculations.

4.4.4.4 Core Thermal Response

The thermal response of the core for accidents and expected transient conditions is discussed in Chapter 15.

4.4.4.5 Analytical Methods

The analytical methods, thermodynamic data, and hydrodynamic data used in determining the thermal and hydraulic characteristics of the core are similar to those used throughout the nuclear power industry.

Core thermal-hydraulic analyses are performed with the aid of a digital computer program. This program models the reactor core through a hydraulic description of orifices, lower tie plates, fuel rods, fuel rod spacers, upper tie plates, fuel channel, and the core bypass flow paths.

4.4.4.5.1 Reactor Model

The reactor model includes a hydraulic representation of the orifice, lower tie plate, fuel rods, water rods or inner water channel, spacers, upper tie plate and the fuel channel.

The code can handle a number of fuel channel types and bypass flow paths. Usually there is one fuel assembly representing each of the "hot" fuel types. The average types then make up the balance of the core.

The computer program iterates on flow through each flow path (fuel assemblies and bypass paths) until the total differential pressure (plenum to plenum) across each path is equal, and the sum of the flows through each path equals the total core flow.

For the initial core, orificing was selected to optimize the core flow distribution between orifice regions as discussed in Subsection 4.4.2.5. The core design pressure is determined from the required turbine throttle pressure, the steam line pressure drop, steam dryer pressure drop, and the steam separator pressure drop. The core inlet enthalpy is determined from the reactor and turbine heat balances. The required core flow is then determined by applying the procedures of

this section and specifications such that the applicable thermal limits are satisfied. The results of applying these methods and specifications are:

(1) Flow for each bundle type,
(2) Flow for each bypass path,
(3) Core pressure drop,
(4) Fluid property axial distribution for each bundle type, and
(5) CPR calculations for each bundle type.

For reload cores, the appropriate orificing, core flow and system pressure are used as model input. The same type of calculations that were used for the initial core are performed to calculate the parameters stated in (1)-(5) above.

4.4.4.5.2 System Flow Balances

The basic assumption used by the code in performing the hydraulic analysis is that the flow entering the core will divide itself between the fuel bundles and the bypass flow paths such that each assembly and bypass flow path experience the same pressure drop. The bypass flow paths considered are described in Table 4.4-7 and shown in Figure 4.4-1. Due to the large flow area, the pressure drop in the bypass region above the core plate is essentially all elevation head. Thus, the sum of the core plate differential pressure and the bypass region elevation head is equal to the core differential pressure.

The total core flow less the control rod cooling flow enters the lower plenum through the jet pumps. A fraction of this passes through the various bypass paths. The remainder passes through the orifices in the fuel support (experiencing a pressure loss) where more flow is lost through the fit-up between the fuel support and the lower tie plate and also through the lower tie plate holes into the bypass region. The majority of the flow continues through the lower tie plate (experiencing a pressure loss) where some flow is lost through the flow path defined by the fuel channel and lower tie plate, and restricted by the finger springs, into the bypass region.

Full-scale tests have been performed to establish the flow coefficients for the major flow paths (Reference 4.4-14). The results of these tests were used to support the initial core design. These tests simulate actual plant configurations which have several parallel flow paths and therefore the flow coefficients for the individual paths could not be separated. However, analytical models of the individual flow paths were developed as an independent check of the tests. The models were derived for actual BWR design dimensions and considered the effects of dimensional variations. These models predicted the test results when the "as built" dimensions were applied. When using these models for hydraulic design calculations, nominal drawing dimensions were used. This is done to yield the most accurate prediction of the expected bypass flow. With the large number of components in a typical BWR core, deviations from the nominal dimensions will tend to statistically cancel, resulting in a total bypass flow best represented by that calculated using nominal dimensions.

The bypass and active channel path loss coefficients are based on test data or analytical models. Use of these coefficients produces an accurate prediction of flow through the various flow paths.

The balance of the flow enters the fuel bundle from the lower tie plate and passes through either the fuel rod channel spaces or into a non-fueled water rod or water channel, depending on fuel type. This water rod or water channel flow, remixes with the active coolant channel flow below the

upper tie plate. The uncertainties associated with the calculation of total core flow and assembly flow are considered in the MCPR safety limit calculation, Subsection 4.4.2.9.

4.4.4.5.3 System Heat Balances

Within the fuel assembly, heat balances on the active coolant are performed nodally. Fluid properties are expressed as the bundle average at the particular node of interest. In evaluating fluid properties, a constant pressure model is used. The core power is divided into two parts: an active coolant power and a bypass flow power. The bypass flow is heated by neutron-slowing down and gamma heating transferred to the bypass flow from structures and control elements which are themselves heated by gamma absorption and by the (n, a) reaction in the control material. The fraction of total reactor power deposited in the bypass region is very nearly 2%. A similar phenomenon occurs within the fuel bundle relative to the active coolant and the water rod or inner water channel flows. The net effect is that approximately 96% of the core power is conducted through the fuel cladding and appears as heat flux.

In design analyses the power is allocated to the individual fuel bundles using a relative power factor. The power distribution along the length of the fuel bundle is specified with axial power factors which distribute the bundle's power among the axial nodes. A nodal local peaking factor is used to establish the peak heat flux at each nodal location.
The relative (radial) and axial power distributions, when used with the bundle flow, determine the axial coolant property distribution resulting in sufficient information to calculate the pressure drop components within each fuel type. Once the equal pressure drop criterion has been satisfied, the critical bundle power is determined by an iterative process for each fuel type.

4.4.4.6 Thermal-Hydraulic Stability Analysis

4.4.4.6.1 Introduction

There are many definitions of stability, but for feedback processes and control systems it can be defined as follows: A system is stable if, following a disturbance, the transient settles to a steady, noncyclic state.

A system may also be acceptably safe even if oscillatory, provided that any limit cycle of the oscillations is less than a prescribed magnitude. Instability then, is either a continual departure from a final steady-state value or a greater-than-prescribed limit cycle about the final steady-state value.

The mechanism for instability can be explained in terms of frequency response. Consider a sinusoidal input to a feedback control system which, for the moment, has the feedback disconnected. If there were no time lags or delays between input and output, the output would be in phase with the input. Connecting the output so as to subtract from the input (negative feedback or 180^{o} out-of-phase connection) would result in stable closed loop operation. However, natural laws can cause phase shift between output and input and should the phase shift reach 180 degrees, the feedback signal would be reinforcing the input signal rather than subtracting from it. If the feedback signal were equal to or larger than the input signal (loop gain equal to one or greater), the input signal could be disconnected and the system would continue to oscillate. If the feedback signal were less than the input signal (loop gains less than one), the oscillations would die out.

The design of the BWR is based on the premise that power oscillations can be readily detected and suppressed.

4.4.4.6.2 Description

Three types of stability considered in the design of boiling water reactors are: (1) reactor core (reactivity) stability, (2) channel hydrodynamic stability, and (3) total system stability. Reactivity feedback instability of the reactor core could drive the reactor into power oscillations. Hydrodynamic channel instability could impede heat transfer to the moderator and drive the reactor into power oscillations. The total system stability considers control system dynamics combined with basic process dynamics. A stable system is analytically demonstrated if no inherent limit cycle or divergent oscillation develops within the system as a result of calculated step disturbances of any critical variable, such as steam flow, pressure, neutron flux, and recirculation flow.

The criteria to be considered are stated in terms of two compatible parameters. First is the decay ratio x_2/x_0, designated as the ratio of the magnitude of the second overshoot to the first overshoot resulting from a step perturbation. A plot of the decay ratio is a graphic representation of the physical responsiveness of the system, which is readily evaluated in a time-domain analysis. Second is the damping coefficient ς_n, the definition of which corresponds to the pole pair closest to the $j\omega$, axis in the s-plane for the system closed loop transfer function. This parameter also applies to the frequency-domain interpretation. The damping coefficient is related to the decay ratio as shown in Figure 4.4-2.

4.4.4.6.3 Stability Criteria

The reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable design limits are not possible or can be reliably and readily detected and suppressed.

The assurance that the total plant is stable and, therefore, has significant safety margin shall be demonstrated analytically when the decay ratio, x_2/x_0, is less than 1.0 or, equivalently, when the damping coefficient, ς_n, is greater than zero for each type of stability discussed. Special attention is given to differentiate between inherent system limit cycles and small, acceptable limit cycles that are always present, even in the most stable reactors. The latter are caused by physical nonlinearities (deadband, striction, etc.) in real control systems and are not representative of inherent hydrodynamic or reactivity instabilities in the reactor. The ultimate performance limit criteria for the three types of dynamic performance are summarized below in terms of decay ratio and damping coefficient:

Channel hydrodynamic stability	$x_2/x_0 < 1$, $\varsigma_n > 0$
Reactor core (reactivity) stability	$x_2/x_0 < 1$, $\varsigma_n > 0$
Total system stability	$x_2/x_0 < 1$, $\varsigma_n > 0$

These criteria shall be satisfied for all attainable conditions of the reactor that may be encountered in the course of plant operation. For stability purposes the most severe core power and core flow conditions to which these criteria will be applied correspond to the highest attainable rodline intersection with natural circulation flow.

New Framatome fuel designs are designed to exhibit channel decay ratio characteristics equivalent to existing Framatome fuel designs. Evaluation of the effect of all fuel designs present in the core on the core stability is currently made on a cycle specific basis. In support of these evaluations,

Framatome uses the STAIF computer code for stability calculations, Reference 4.4-48. SSES has implemented Option 3 (oscillation power range monitor system) for the long term stability solution.

4.4.4.6.4 Mathematical Model

For the initial core, the mathematical model representing the core examines the linearized reactivity response of a reactor system with density-dependent reactivity feedback caused by boiling. The core model (References 4.4-27 through 4.4-32), shown in block diagram form in Figure 4.4-3, solves the dynamic equations that represent the reactor core in the frequency domain.

The plant model considers the entire reactor system, neutronics, heat transfer, hydraulics, and the basic processes, as well as associated control systems such as the flow controller, pressure regulator, feedwater controller, etc. Although, the control systems may be stable when analyzed individually, final control system settings must be made in conjunction with the operating reactor so that the entire system is stable. The plant model yields results that are essentially equivalent to those achieved with the core model and allows the addition of the controllers, which have adjustable features permitting the attainment of the desired performance.

The plant model solves the dynamic equations that present the BWR system in the time domain. The variables, such as steam flow and pressure, are represented as a function of time. The extensiveness of this model (Reference 4.4-10, which describes the version of the code used for Susquehanna system stability calculations) is shown in block diagram form in Figure 4.4-3. Many of the blocks are extensive systems in themselves.

For reload cores, the continued applicability of the exclusion region that has been established to assure thermal-hydraulic stability is demonstrated or the exclusion region is redefined. Stability calculations, when required are performed using the STAIF computer code (Reference 4.4-48).

4.4.4.6.5 Analytical Confirmation

References 4.4-37 and 4.4-48 provide a description of the analytical methods used by GE and Framatome as well as model qualification through comparison with test data.

4.4.4.6.6 Analysis Results

Using actual design parameters, the responses of important nuclear system variables for the first core to step disturbances were calculated for three different power/flow conditions. Figures 4.4-7A, 4.4-7B, and 4.4-7C show the responses at 51.5% power and natural circulation. Figures 4.4-8A, 4.4-8B, and 4.4-8C show the responses at rated power/flow conditions. Figures 4.4-9A, 4.4-9B, and 4.4-9C show the responses at the lower end of the automatic power-flow control path. For all of these cases the responses met the stability criterion.

For reload cores, a confirmatory analysis is performed to demonstrate the continued applicability of the core stability regions identified in the COLR. The analysis is based on comparison of core stability performance to previously analyzed cycles. A stability code is used to calculate the variations in decay ratio from cycle to cycle for operating conditions at representative state points near the stability exclusion region.

4.4.5 TESTING AND VERIFICATION

The testing and verification techniques to be used to assure that the planned thermal and hydraulic design characteristics of the core have been provided, and will remain within required limits throughout core lifetime, are discussed in Chapter 14. A summary is as follows:

(1) Preoperational Testing

Tests are performed during the preoperational test program to confirm that construction is complete and that all process and safety equipment is operational. Baseline data are taken to assist in the evaluation of subsequent tests. Heat balance instrumentation, jet pump flow and core temperature instrumentation, is calibrated and set points verified.

(2) Initial Start-Up

Hot functional tests are conducted with the reactor between 5 and 10% power. Core performance is monitored continuously to assure that the reactor is operating within allowable limits (e.g., peaking factors, linear heat generation rate, etc.) and is evaluated periodically to verify the core expected and actual performance margins.

4.4.6 INSTRUMENTATION REQUIREMENTS

The reactor vessel instrumentation monitors the key reactor vessel operating parameters during planned operations. This ensures sufficient control of the parameters. The following reactor vessel sensors are discussed in Subsection 7.7.1.1.

(1) Reactor Vessel Temperature

(2) Reactor Vessel Water Level

(3) Reactor Vessel Coolant Flow Rates and Differential Pressures

(4) Reactor Vessel Internal Pressure

(5) Neutron Monitoring System

4.4.7 REFERENCES

4.4-1 General Electric Thermal Analysis Basis (GETAB): Data, Correlation and Design Application, General Electric Company, January 1977, (NEDO-10958A).

4.4-2 Core Flow Distribution in a Modern Boiling Water Reactor as Measured in Monticello, August 1976, (NEDO-10722A).

4.4-3 R.C. Martinelli and D. E. Nelson, "Prediction of Pressure Drops During Forced Convection Boiling of Water," ASME Trans., 70, pp 695-702, 1948.

4.4-4 Deleted

4.4-5 Jens, W. H., and Lottes, P.A., Analysis of Heat Transfer, Burnout, Pressure Drop, and Density Data for High Pressure Water, USAEC Report-4627, 1972.

4.4-6 Deleted

4.4-7 Deleted

4.4-8 Deleted

4.4-9 Deleted

4.4-10 Analytical Methods of Plant Transient Evaluations for General Electric Boiling Water Reactor, General Electric Company, BWR Systems Department, February 1973, (NEDO-10802).

4.4-11 Deleted

4.4-12 Deleted

4.4-13 Peach Bottom Atomic Power Station Units 2 and 3, Safety Analysis Report for Plant Modifications to Eliminate Significant In-Core Vibration, General Electric Co., NEDO-20994, September, 1975.

4.4-14 "Supplemental Information for Plant Modification to Eliminate Significant In-Core Vibration," NEDE-21156, Class III, January 1976.

4.4-15 Deleted

4.4-16 Deleted

4.4-17 Deleted

4.4-18 Deleted

4.4-19 Deleted

4.4-20 Deleted

4.4-21 Deleted

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4.4-23 Deleted

4.4-24 Deleted

4.4-25 Deleted

4.4-26 Deleted

4.4-27 KAPL-2170 Hydrodynamic Stability of a Boiling Channel, by A. B. Jones; 2 October 1961.

4.4-28 KAPL-2208 Hydrodynamic Stability of a Boiling Channel Part 2, by A. B. Jones; 20 April 1962.

4.4-29 KAPL-2290 Hydrodynamic Stability of a Boiling Channel Part 3, by A. B. Jones and D. G. Dight; 28 June 1963.

4.4-30 KAPL-3070 Hydrodynamic Stability of a Boiling Channel Part 4, by A. B. Jones; 18 August 1964.

4.4-31 KAPL-3072 Reactivity Stability of a Boiling Reactor Part 1, by A. B. Jones and W. M. Yarbrough; 14 September 1964.

4.4-32 KAPL-3093 Reactivity Stability of a Boiling Reactor Part 2, by A. B. Jones, 1 March 1965.

4.4-33 Deleted

4.4-34 McBeth, R.V., R. Trenberth, and R. W. Wood, "An Investigation Into the Effects of Crud Deposits on Surface Temperature, Dry-Out, and Pressure Drop, with Forced Convection Boiling of Water at 69 Bar in an Annular Test Section," AEEW-R-705, 1971.

4.4-35 Green, S.J., B. W. LeTourneau, A.C. Peterson, "Thermal and Hydraulic Effects of Crud Deposited on Electrically Heated Rod Bundles," WAPD-TM-918, Sept. 1970.

4.4-36 H.T. Kim and H.S. Smith, "Core Flow Distribution in a General Electric Boiling Water Reactor as Measured in Quad Cities Unit 1," NEDO-10722A, August, 1976.

4.4-37 Licensing Topical Report, "Stability and Dynamic Performance of the General Electric Boiling Water Reactor," January, 1977 (NEDO-21506).

4.4-38 Deleted

4.4-39 Deleted

4.4-40 Deleted

4.4-41 ANP-10307PA, “AREVA MCPR Safety Limit Methodology for Boiling Water Reactors”.

4.4-42 ANF-524 (P)(A), Revision 2 and Supplements 1 and 2, “Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors,” November 1990.

4.4-43 “Methodology for Calculation of Pressure Drop in BWR Fuel Assemblies”, XN-NF-79-59(P)(A), November 1983.

4.4-44 MICROBURN-B2 Based Impact of Failed / Bypassed LPRMs and TIPs, Extended LPRM Calibration interval on Single Loop Operation on Measured Radial Bundle Power Uncertainty, “EMF-2493(P), Rev. 0, December 2000.

4.4-45 “Thermal-Hydraulic Characteristics of the ATRIUM-10 Fuel Design for Susquehanna”, EMF-95-066(P), June 1995.

4.4-46 “Single Phase Hydraulic Performance of Exxon Nuclear BWR 9x9 Fuel Assembly”, XN-NF-683(P), February 1983.

4.4-47 “Generic Mechanical Design Criteria for BWR Fuel Designs”, ANF-89-98(P)(A) Revision 1, and Revision 1 Supplement 1, May 1995.

4.4-48 EMF-CC-074(P)(A), Volume 4, Revision 0, “BWR Stability Analysis – Assessment of STAIF with input from MICROBURN-B2,” Siemens Power Corporation, August 2000.

4.4-49 “Exxon Nuclear Methodology for Boiling Water Reactors THERMEX: Thermal Limits Methodology Summary Description”, XN-NF-80-19(P)(A) Volume 3 Revision 2, January 1987.

4.4-50 “Impact of Failed/Bypassed LPRMs and TIPs and Extended LPRM Calibration Interval on Radial Bundle Power Uncertainty”, EMF-1903 Revision 2, October 1996.

4.4-51 “Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4 / MICROBURN-B2,” EMF-2158(P)(A), Rev. 0, October 1999.

4.4-52 "Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications," NEDO-32465-A, August 1996

4.4-53 "BWR Owners' Group Long-Term Stability Solutions Licensing Methodology," NEDO-31960-A, June 1991

4.4-54 "BWR Owners' Group Long-Term Stability Solutions Licensing Methodology," NEDO-31960-A Supplement 1, March 1992

4.4-55 "ABB Option III Oscillation Power Range Monitor (OPRM)," CENPD-400-P-A, Revision 1, May 1995

4.4-56 NEDC-32410P-A, “Nuclear Measurement Analysis and Control Power Range Neutron Monitor (NUMAC-PRNM) Retrofit Plus Option III Stability Trip Function”, October 1995.

4.4-57 NEDC-32410P-A Supplement 1, “Nuclear Measurement Analysis and Control Power Range Neutron Monitor (NUMAC-PRNM) Retrofit Plus Option III Stability Trip Function”, November 1997.

4.4-58 EMF-2209 (P) (A), Revision 2, “SPCB Critical Power Correlation, “Framatome ANP, September 2003.

4.4-59 NRC Letter from R. V. Guzman (NRC) to B. T. McKinney (PPL), January 30, 2008, Subject: Susquehanna Steam Electric Station, Units 1 and 2 – Issuance of Amendment Regarding 13-Percent Extended Power Uprate (TAC Nos. MD3309 and MD 3310) [Accession ML 080020182]

4.4-60 ANP-10335P-A, "ACE/ATRIUM-11 Critical Power Correlation."

4.4-61 NRC 2021-0003, "Susquehanna Steam Electric Station, Units 1 and 2 - Issuance of Amendment NOS. 278 and 260 to Allow Application of Advanced Framatome ATRIUM-11 Fuel Methodologies," January 21, 2021.

Table 4.4-1
Typical Thermal and Hydraulic Design Characteristics of the Reactor Core

Security-Related Information
Table Withheld Under 10 CFR 2.390

Table 4.4-2
TYPICAL VOID DISTRIBUTION

Security-Related Information
Table Withheld Under 10 CFR 2.390

Table 4.4-2a

AXIAL POWER DISTRIBUTION USED TO GENERATE
TYPICAL VOID AND QUALITY DISTRIBUTIONS

Security-Related Information
Table Withheld Under 10 CFR 2.390

Table 4.4-3
TYPICAL FLOW QUALITY DISTRIBUTION

Security-Related Information
Table Withheld Under 10 CFR 2.390

Table 4.4-4

TYPICAL CORE FLOW DISTRIBUTION

Security-Related Information
Table Withheld Under 10 CFR 2.390

Table 4.4-6

Uncertainties Considered in MCPR Safety Limit

Reactor System Uncertainties	
Quantity	***Standard Deviation %***
Feedwater Flow	*1.76*
Feedwater Temperature	*0.76*
Reactor Pressure	*0.5*
Total Core Flow	*2.5*

Fuel Related Uncertainties	
Quantity	***Reference***
Assembly flow	*Unit 1: 4.4-42* *Unit 2: 4.4-41*
CPR Correlation Additive Constant	*Unit 1: 4.4-58 (SPCB)* *Unit 2: 4.4-58 (SPCB), 4.4-60 (ACE)*
Assembly radial peaking	*4.4-44*
Rod local peaking	*4.4-51*
Nodal Power	*Unit 1: N/A* *Unit 2: 4.4-44*

Table 4.4-7
BYPASS FLOW PATHS

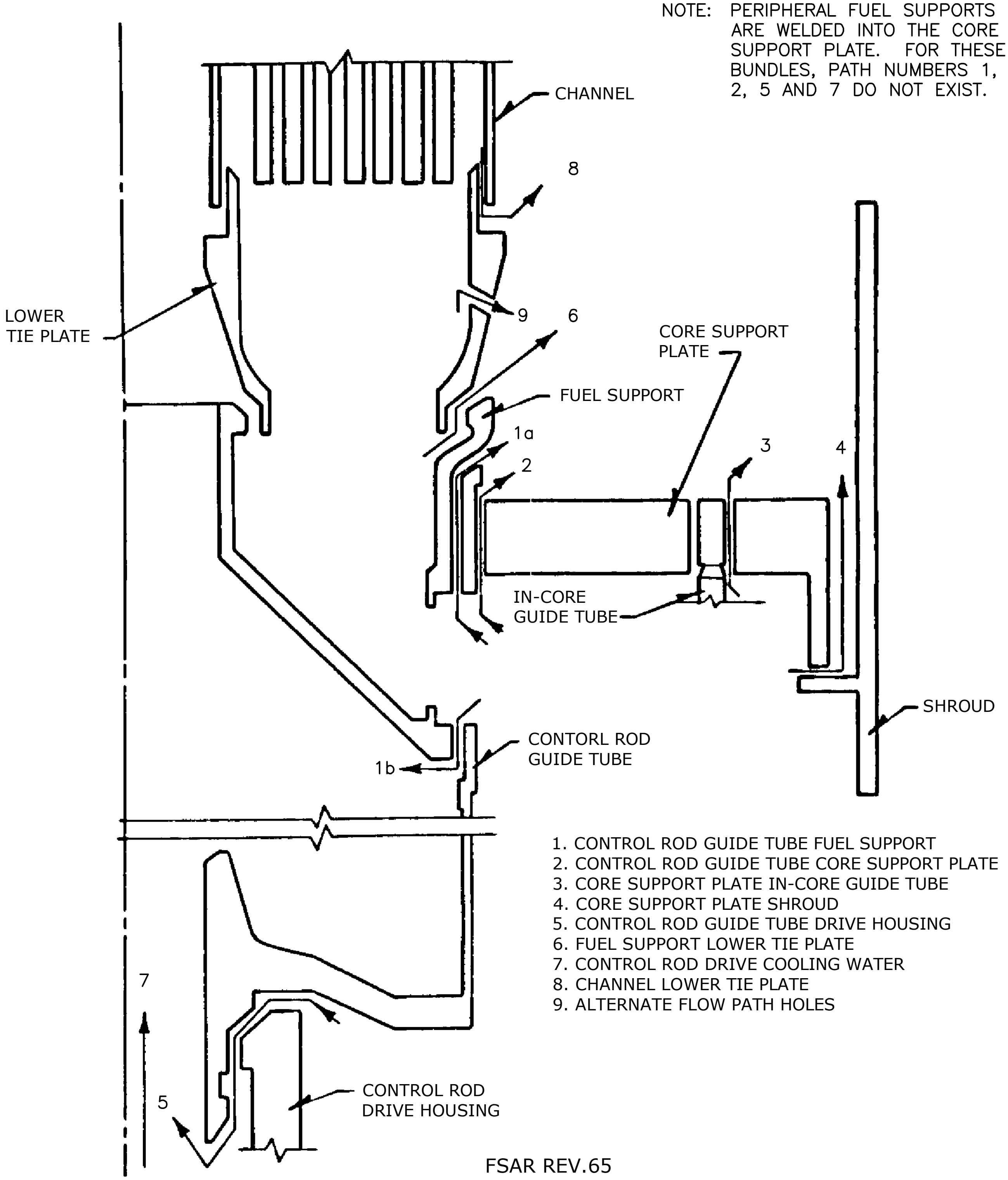
Security-Related Information
Table Withheld Under 10 CFR 2.390

Table 4.4-8
PLANT CONFIGURATION DATA

Security-Related Information
Table Withheld Under 10 CFR 2.390

Table 4.4-9
LENGTHS OF SAFETY INJECTION LINES

Security-Related Information
Table Withheld Under 10 CFR 2.390



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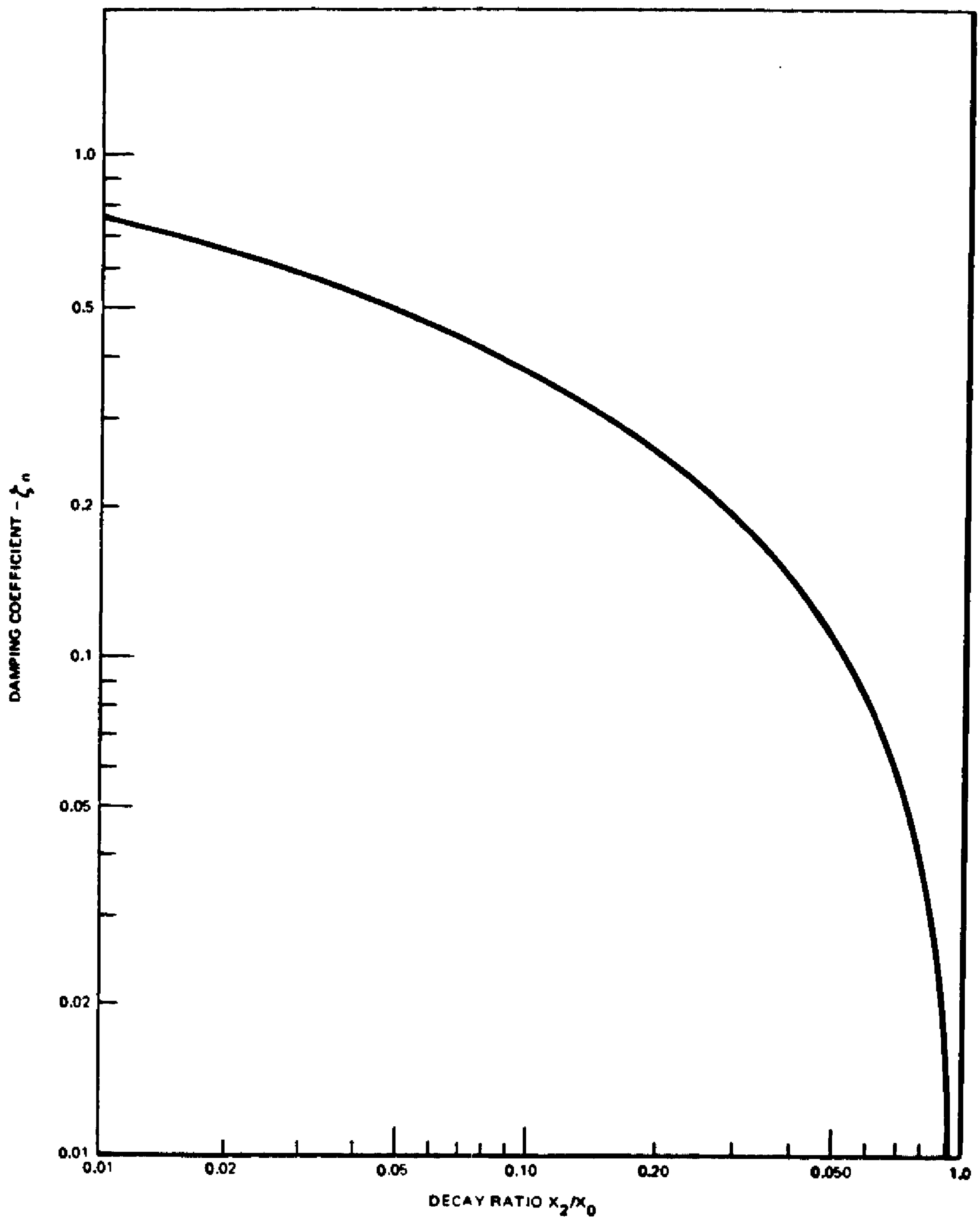
SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
SCHEMATIC OF REACTOR ASSEMBLY SHOWING THE BYPASS FLOW PATHS FIGURE 4.4-1, Rev. 47

Auto-Cad Figure Fsar 4_4_1.dwg



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UNITS 1 & 2
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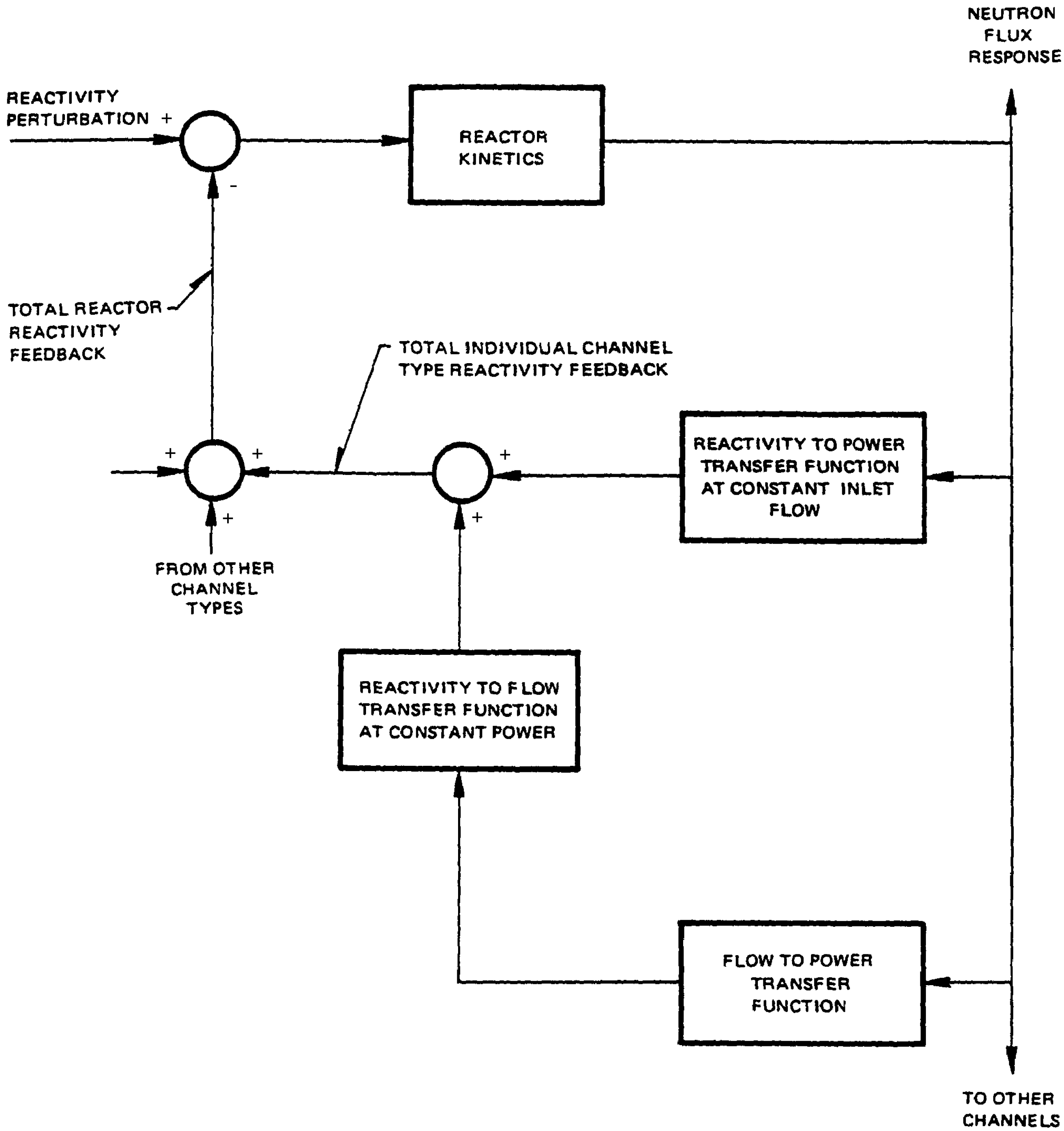
DAMPING COEFFICIENT VERSUS
DECAY RATIO
(SECOND ORDER SYSTEMS)

FIGURE 4.4-2, Rev. 47

Auto-Cad Figure Fsar 4_4_2.dwg



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UNITS 1 & 2
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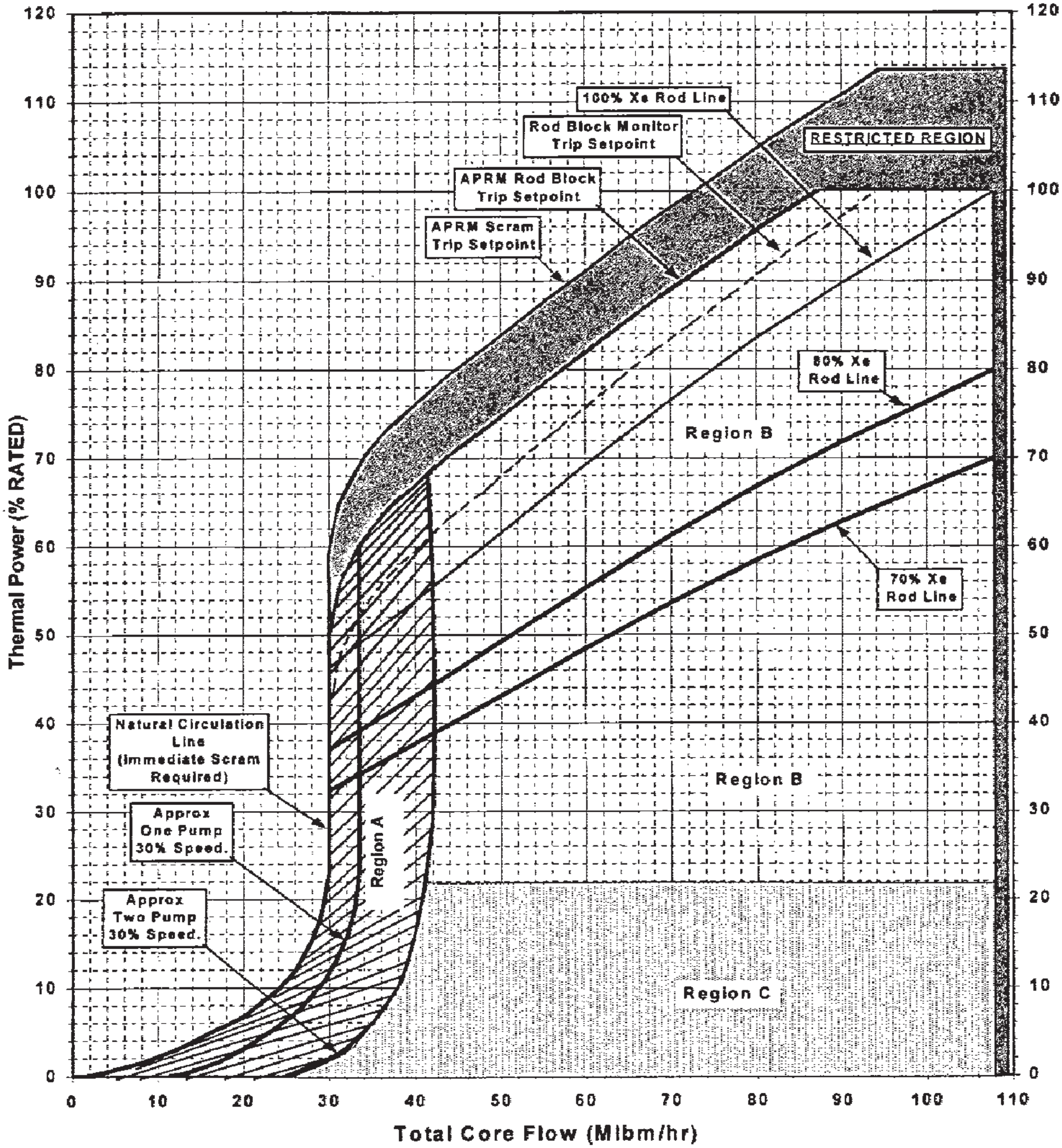
HYDRODYNAMIC AND
CORE STABILITY MODEL
FOR INITIAL CORE

FIGURE 4.4-3, Rev. 54

Auto-Cad Figure Fsar 4_4_3.dwg



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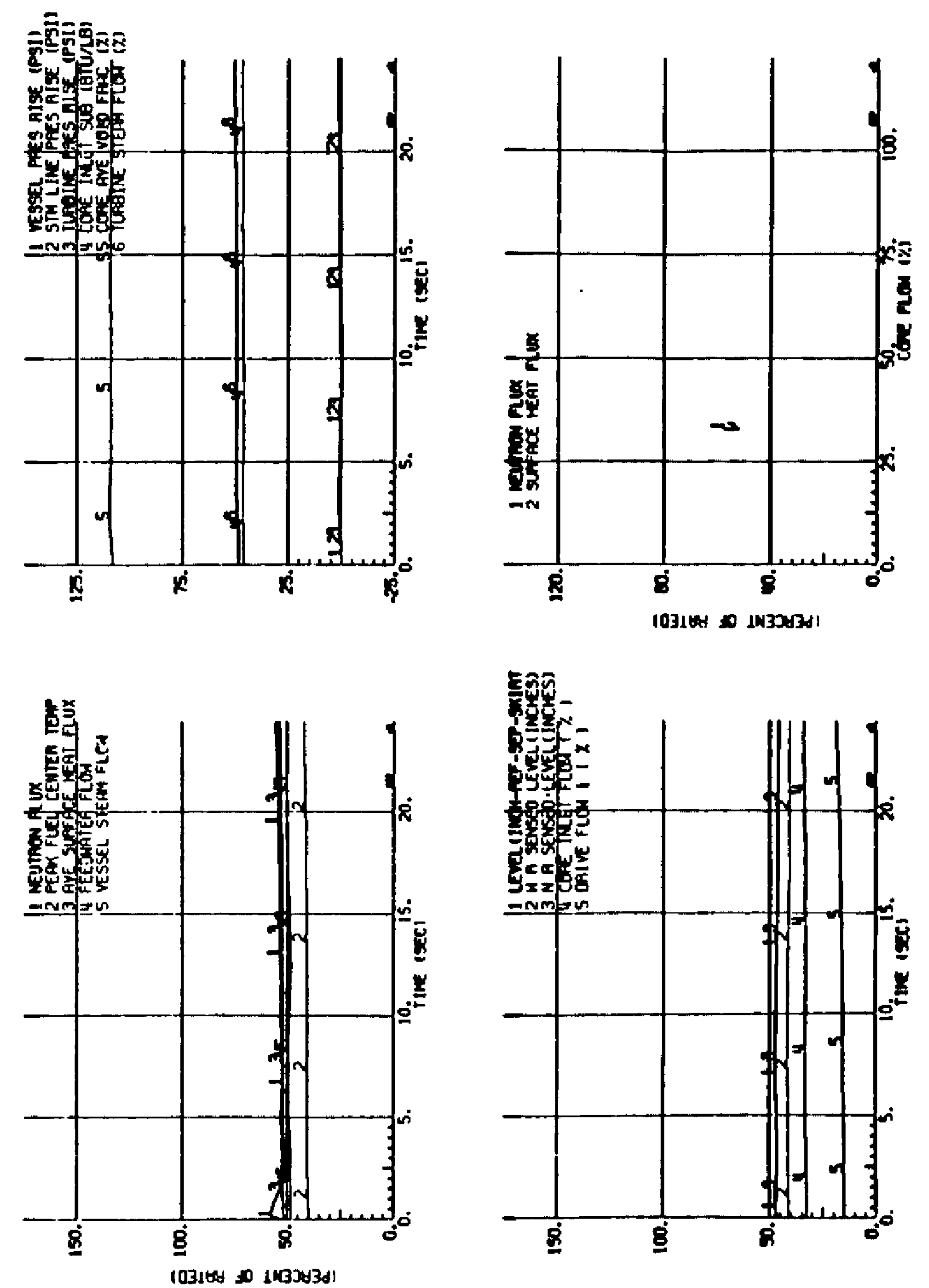
SIMPLIFIED
POWER-FLOW OPERATING MAP

FIGURE 4.4-5, Rev. 55

Auto-Cad Figure Fsar 4_4_5.dwg



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SUSQUEHANNA STEAM ELECTRIC STATION
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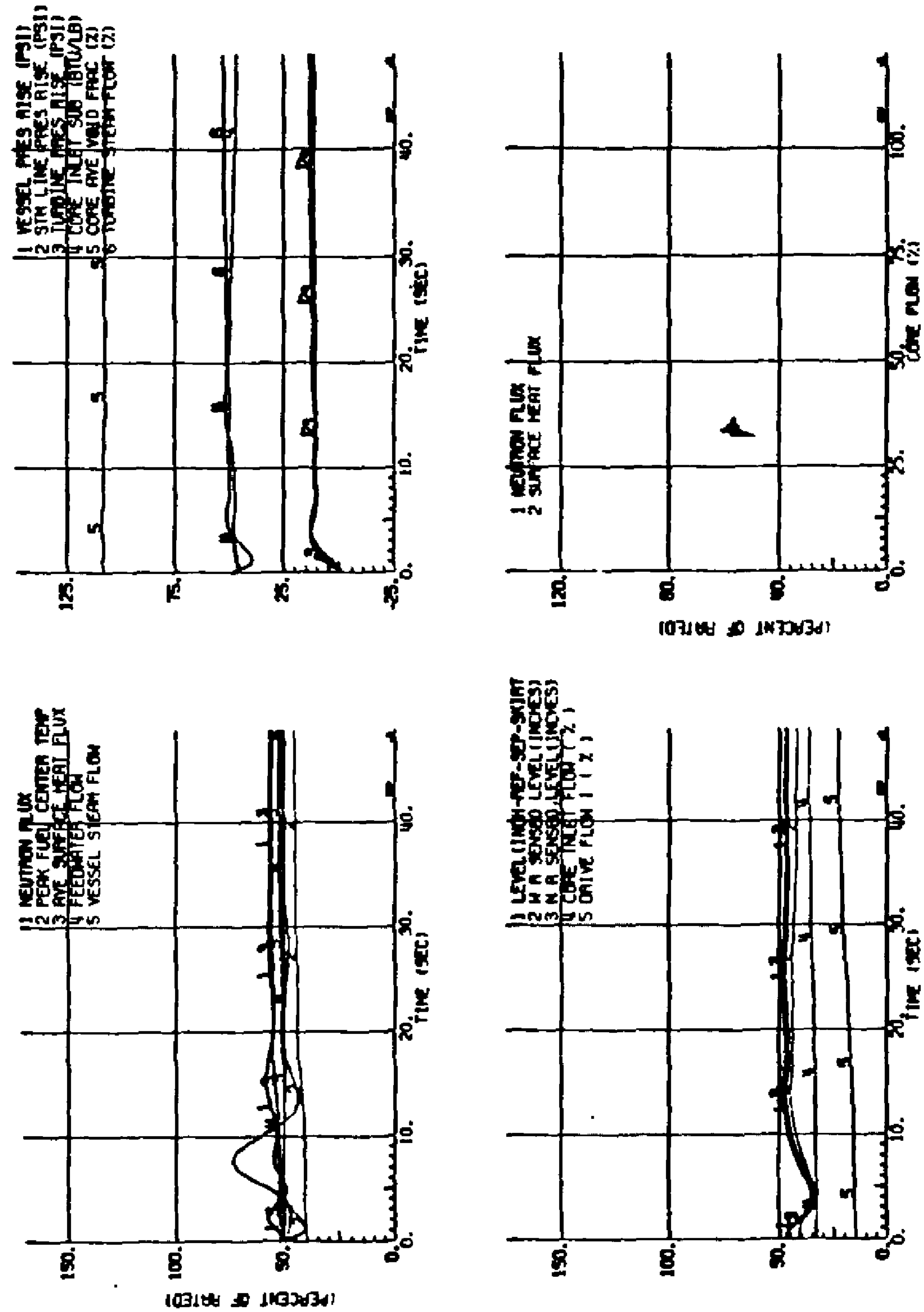
10 CENT ROD REACTIVITY STEP
AT 51.5% RATED POWER
(NATURAL CIRCULATION)

FIGURE 4.4-7A, Rev. 47

Auto-Cad Figure Fsar 4_4_7A.dwg



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SUSQUEHANNA STEAM ELECTRIC STATION
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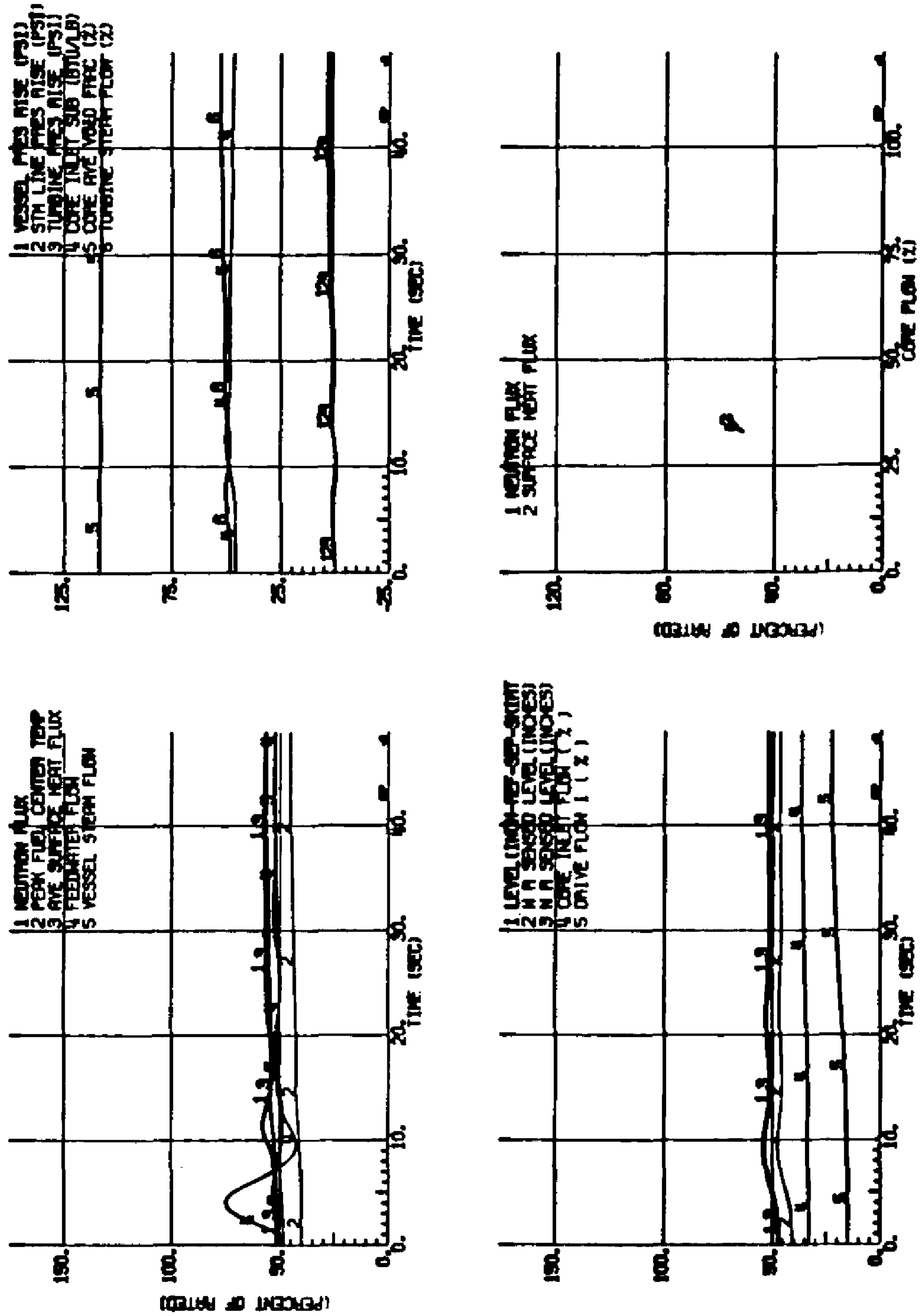
10 PSI PRESSURE REGULATOR
SETPOINT STEP AT 51.5%
RATED POWER
(NATURAL CIRCULATION)

FIGURE 4.4-7B, Rev. 47

Auto-Cad Figure Fsar 4_4_7B.dwg



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SUSQUEHANNA STEAM ELECTRIC STATION
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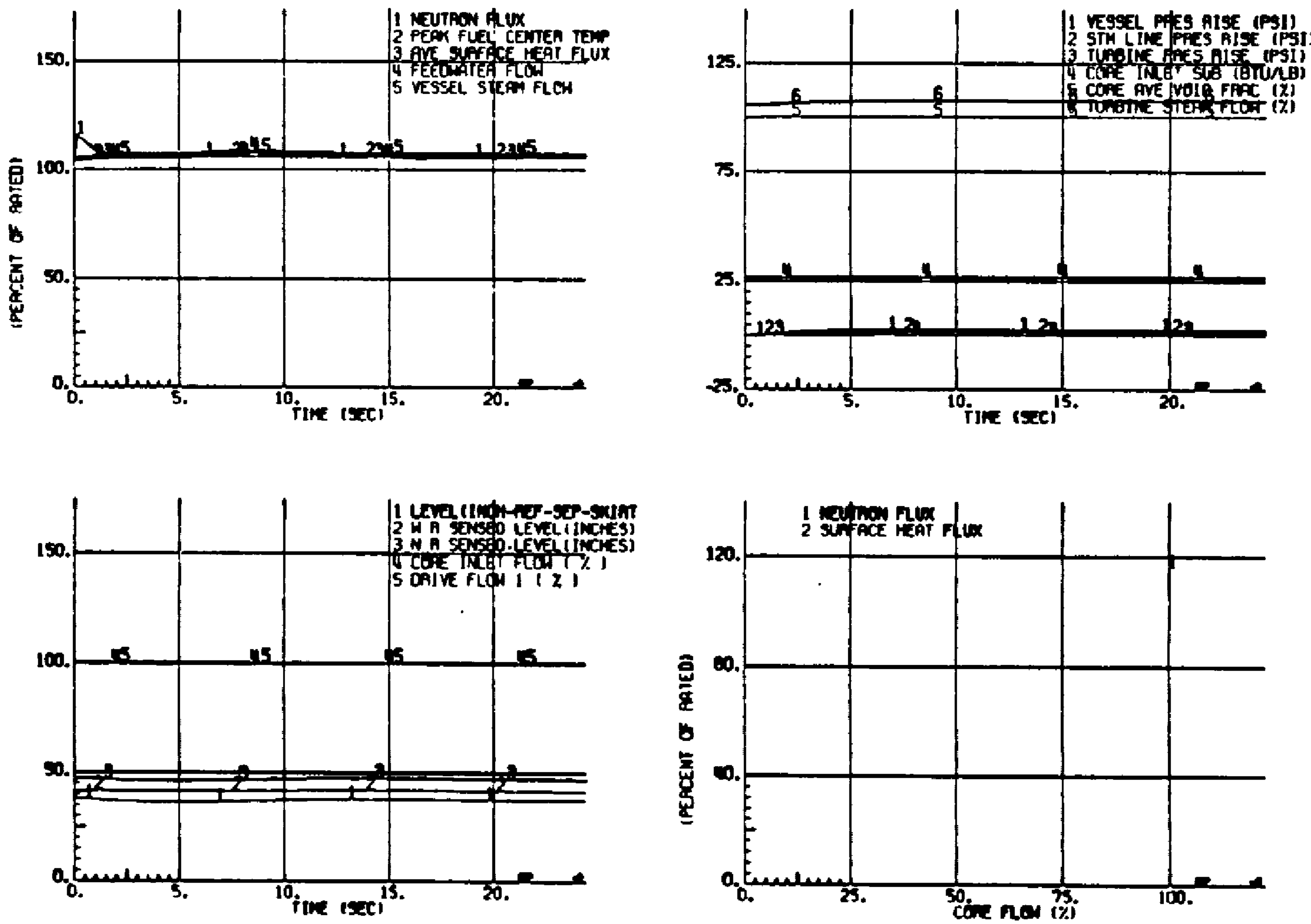
FINAL SAFETY ANALYSIS REPORT

6-INCH LEVEL SETPOINT STEP
AT 51.5% RATED POWER
(NATURAL CIRCULATION)

FIGURE 4.4-7C, Rev. 47

Auto-Cad Figure Fsar 4_4_7C.dwg



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SUSQUEHANNA STEAM ELECTRIC STATION
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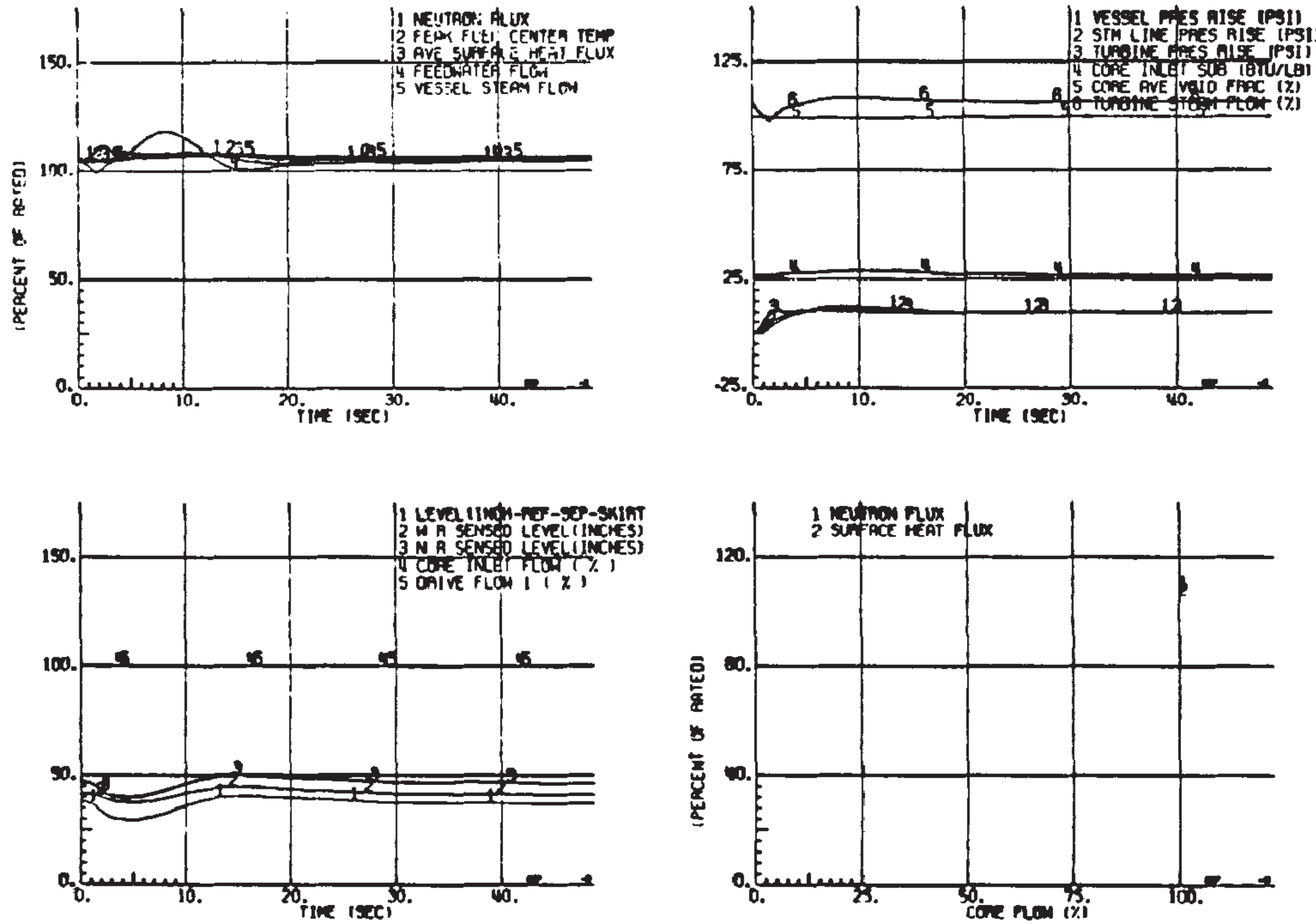
FINAL SAFETY ANALYSIS REPORT

10 CENT ROD REACTIVITY STEP
AT 105% RATED POWER AND
100% RATED FLOW

FIGURE 4.4-8A, Rev. 47

Auto-Cad Figure Fsar 4_4_8A.dwg



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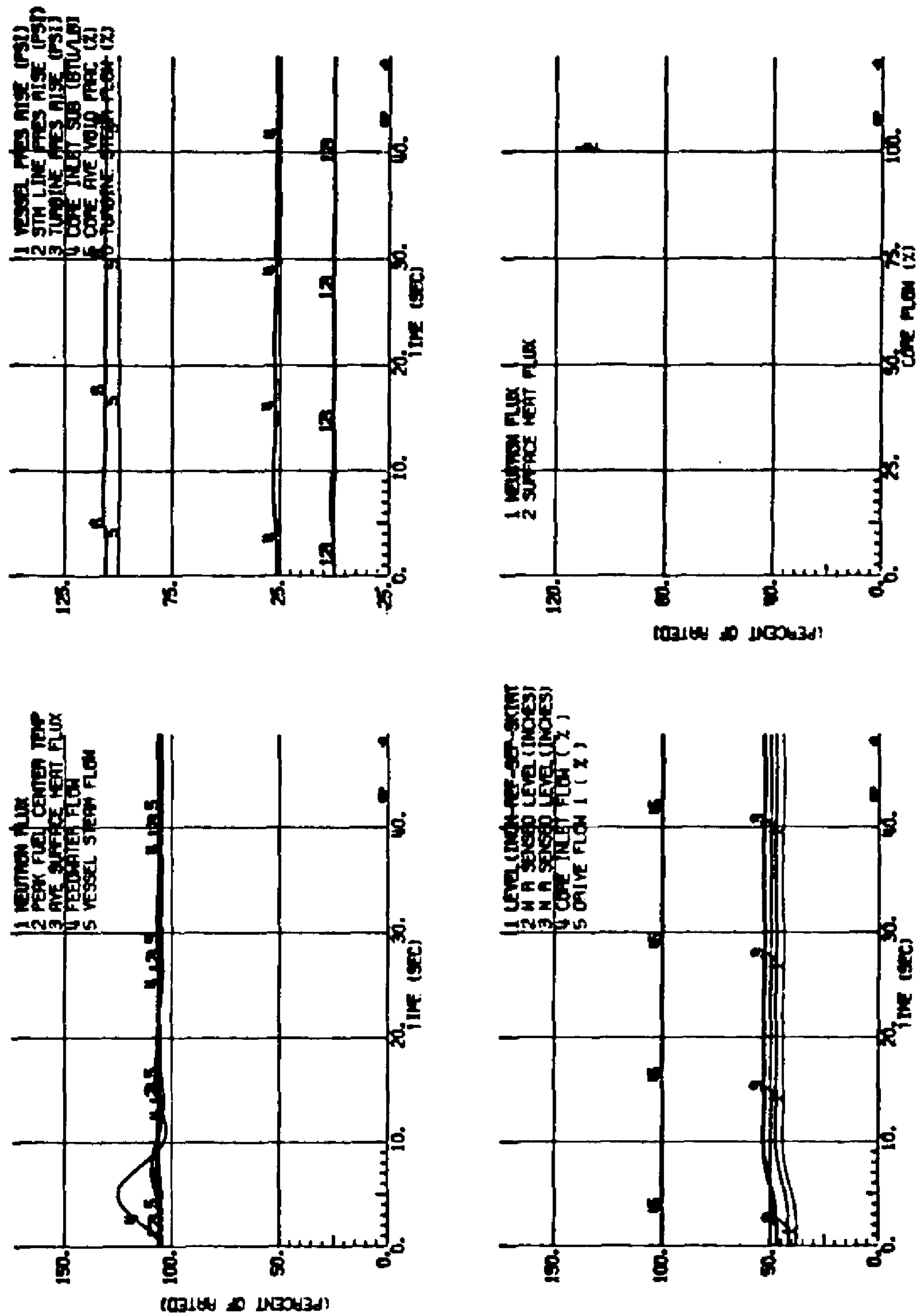
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

10 PSI PRESSURE REGULATOR
SETPOINT STEP AT 105%
RATED POWER AND 100%
RATED FLOW

FIGURE 4.4-8B, Rev. 47

Auto-Cad Figure Fsar 4_4_8B.dwg



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UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

6-INCH LEVEL SETPOINT STEP
AT 105% RATED POWER AND
100% RATED FLOW

FIGURE 4.4-8C, Rev. 47

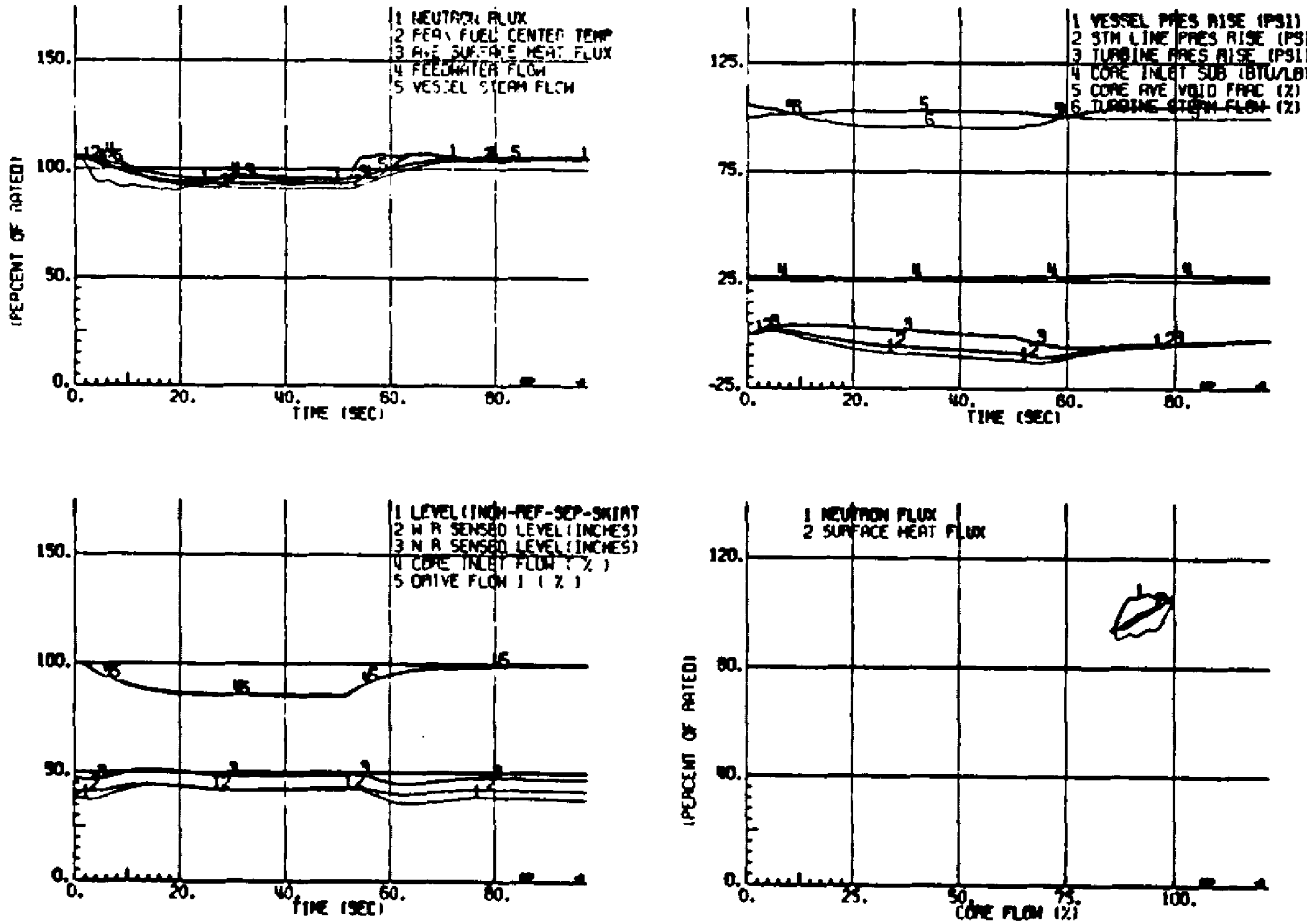
Auto-Cad Figure Fsar 4_4_8C.dwg

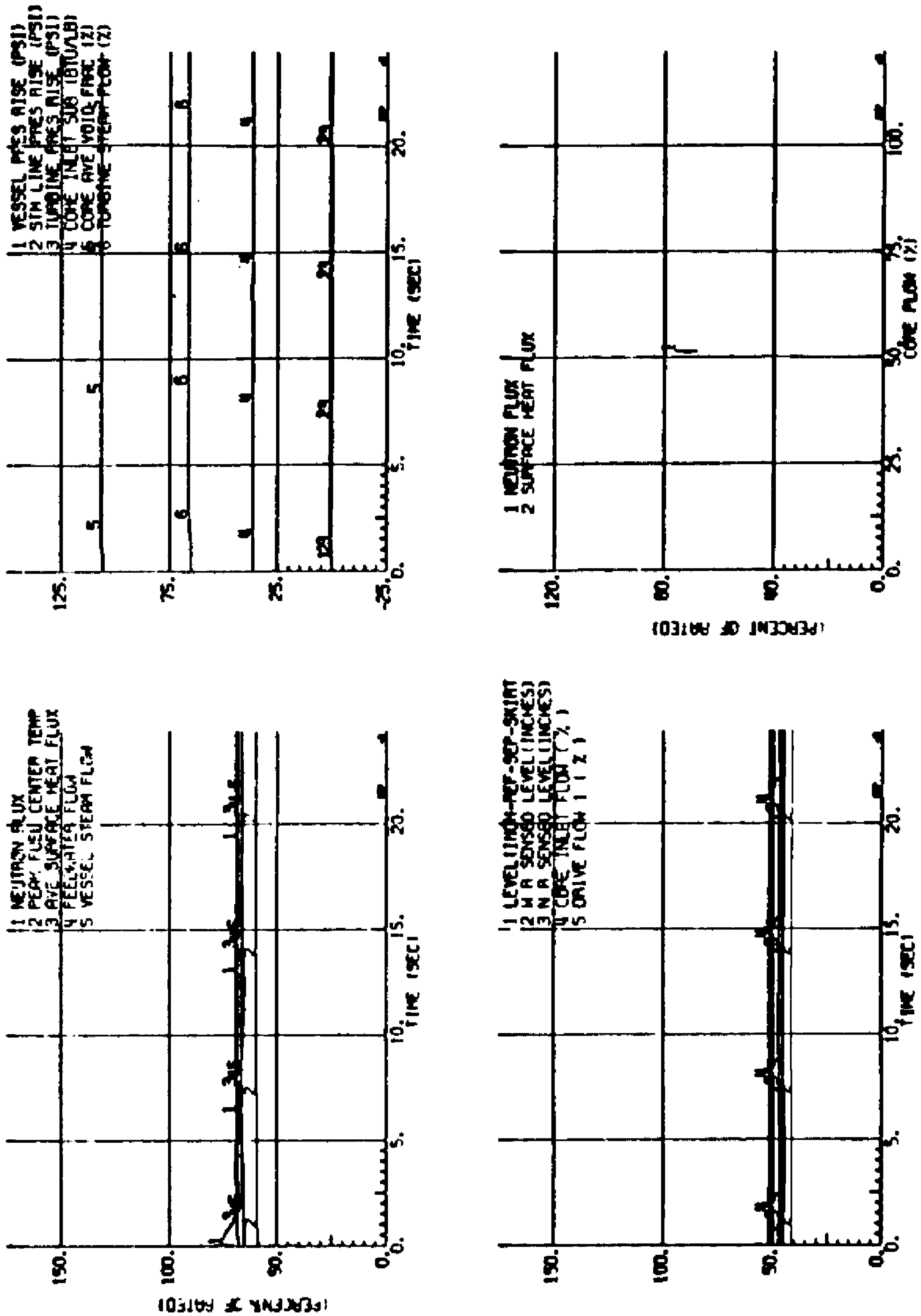
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

10% LOAD DEMAND STEP AT
105% RATED POWER AND 100%
RATED FLOW

FIGURE 4.4-8D, Rev. 47

Auto-Cad Figure Fsar 4_4_8D.dwg





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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

10 CENT ROD REACTIVITY STEP AT
68% RATED POWER AND 51.5%
RATED FLOW

FIGURE 4.4-9A, Rev. 47

Auto-Cad Figure Fsar 4_4_9A.dwg

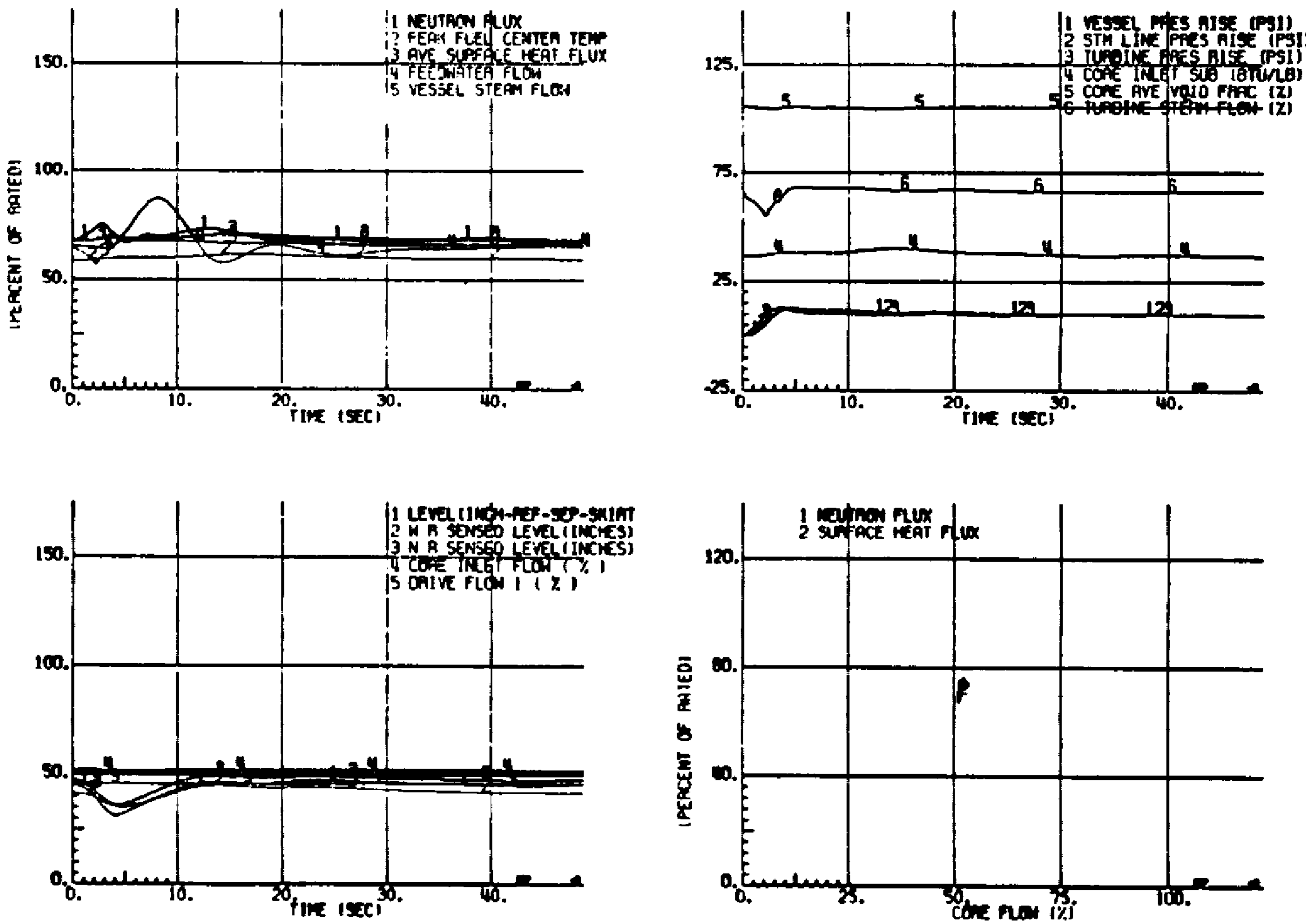
FSAR REV.65

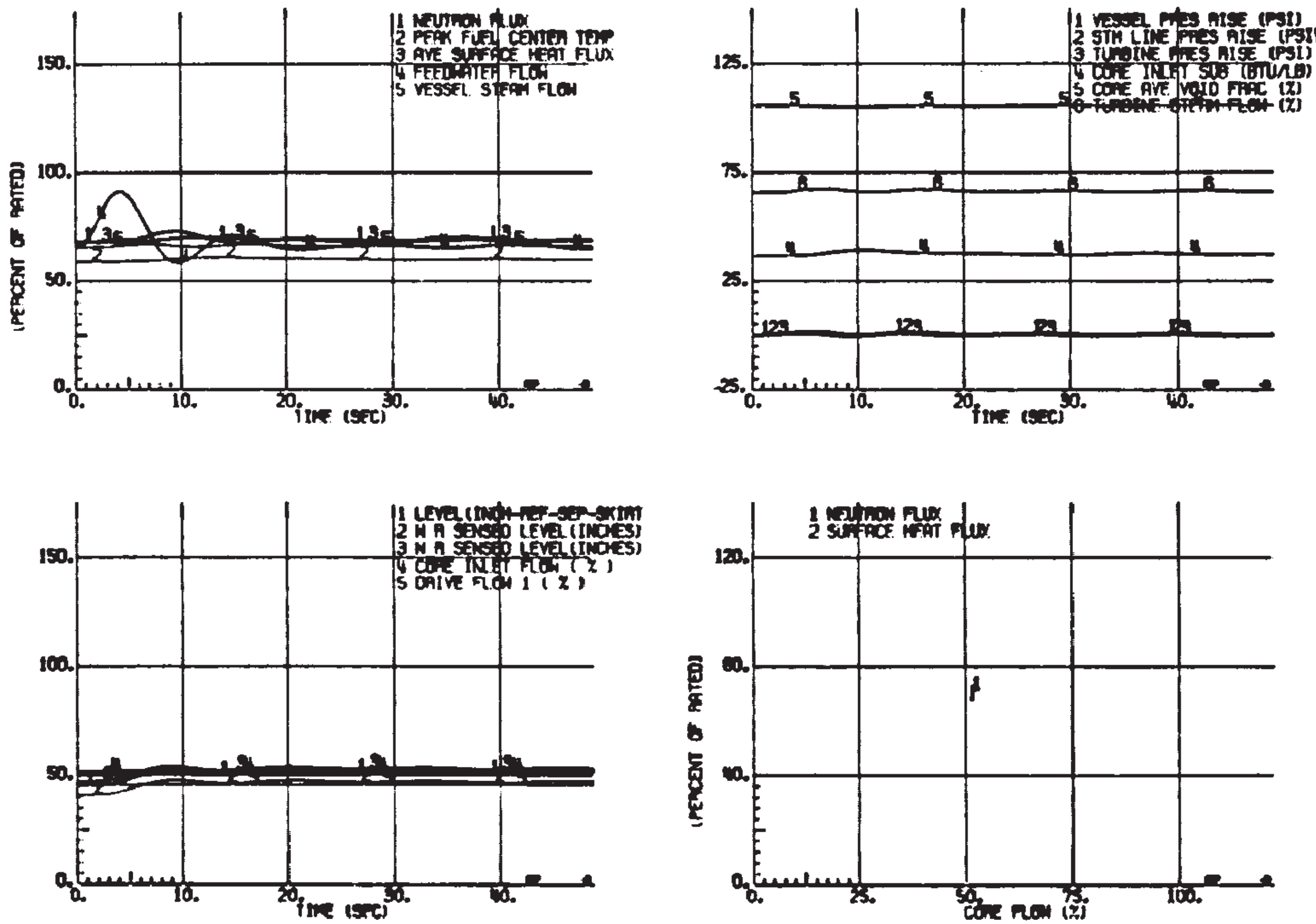
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

10 PSI PRESSURE REGULATOR STEP
AT 68% RATED POWER AND
51.5% RATED FLOW

FIGURE 4.4-9B, Rev. 47

Auto-Cad Figure Fsar 4_4_9B.dwg





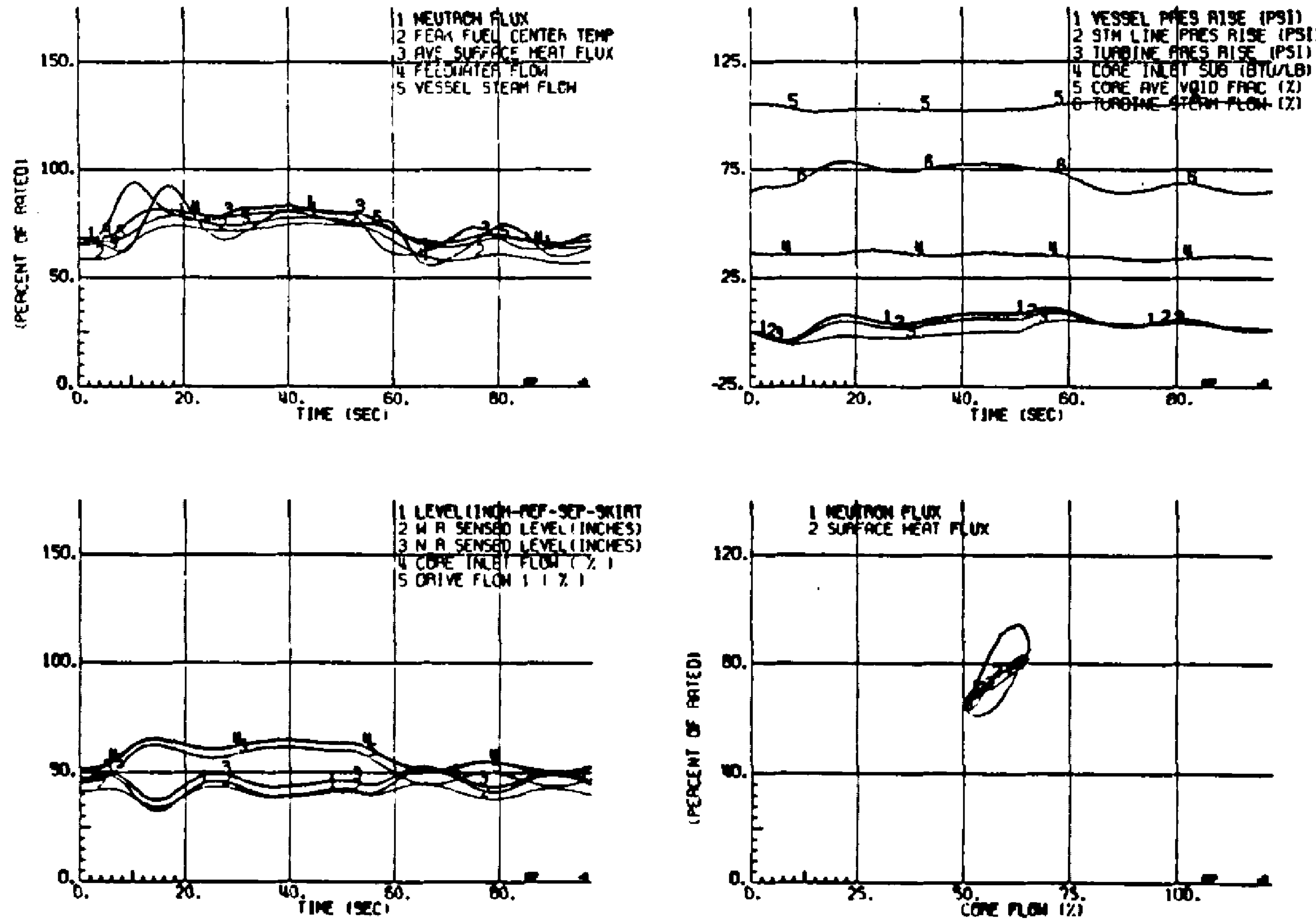
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

6-INCH WATER LEVEL SETPOINT
STEP AT 60% RATED POWER AND
51.5% RATED FLOW

FIGURE 4.4-9C, Rev. 47

Auto-Cad Figure Fsar 4_4_9C.dwg



FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

10% LOAD DEMAND STEP AT
68% RATED POWER AND 51.5%
RATED FLOW

FIGURE 4.4-9D, Rev. 47

Auto-Cad Figure Fsar 4_4_9D.dwg