4.1 SUMMARY DESCRIPTION

Susquehanna Units 1 and 2 are General Electric BWR/4 Boiling Water Reactors. Each reactor contains 764 fuel assemblies and 185 control rods arranged in an upright cylindrical configuration. Light water acts as both moderator and coolant.

The reactor assembly consists of the reactor vessel, its internal components of the core, shroud, steam separator and dryer assemblies, and jet pumps. Also included in the reactor assembly are the control rods, control rod drive housings, and the control rod drives. Figure 3.9-3, shows the arrangement of reactor assembly components. Important design and performance characteristics are discussed in Sections 4.2, 4.3 and 4.4. Loading conditions for reactor assembly components are specified in Section 3.9.

4.1.1 REACTOR VESSEL

The reactor vessel design and description are covered in Section 5.3.

4.1.2 REACTOR INTERNAL COMPONENTS

The major reactor internal components are the core (fuel, channels, control rods, and instrumentation), the core support structure (including the shroud, top guide and core plate), the shroud head and steam separator assembly, the steam dryer assembly, the feedwater spargers, the core spray spargers, and the jet pumps. Except for the Zircaloy in the reactor core, these reactor internals are stainless steel or other corrosion resistant alloys. All major internal components of the vessel can be removed except the jet pump diffusers, the jet pump risers, the shroud, the core spray lines, spargers, and the feedwater sparger. The removal of the steam dryers, shroud head and steam separators, fuel assemblies, in-core instrumentation, control rods, orificed fuel supports, and control rod guide tubes, can be accomplished on a routine basis.

4.1.2.1 Reactor Core

4.1.2.1.1 General

The design of the boiling water reactor core, including fuel, is based on the proper combination of many design variables and operating experience. These factors contribute to the achievement of high reliability.

A number of important features of the boiling water reactor core design are summarized in the following paragraphs:

(1) The BWR core mechanical design is based on conservative application of stress limits, operating experience, and experimental test results. The moderate pressure level characteristics of a direct cycle reactor (approximately 1050 psia) result in moderate cladding temperatures and stress levels.

(2) The low coolant saturation temperature, high heat transfer coefficients, and near-neutral water chemistry of the BWR are significant, advantageous factors in minimizing Zircaloy temperature and associated temperature-dependent corrosion and hydride buildup.

The relatively uniform fuel cladding temperatures throughout the core minimize migration of the hydrides to cold cladding zones and reduce thermal stresses.

(3) The basic thermal and mechanical criteria applied in the design have been proven by irradiation of statistically significant quantities of fuel. The design heat transfer rates and linear heat generation rates are similar to values proven in fuel assembly irradiation.

(4) The design power distribution used in sizing the core represents a worst expected state of operation.

(5) The AREVA critical power methodology for boiling water reactors (References 4.1-12, and 4.1-29) is applied to assure that more than 99.9% of the fuel rods are expected to avoid boiling transition for the most severe abnormal operational transient described in Chapter 15. The possibility of boiling transition occurring during normal reactor operation is insignificant.

(6) Because of the large negative moderator density coefficient of reactivity during normal power operation, the BWR has a number of inherent advantages. These are the uses of coolant flow for power maneuvering, the inherent self-flattening of the radial power distribution, the ease of control, the spatial xenon stability, and the ability to override xenon, in order to follow load.

Boiling water reactors do not have instability problems due to xenon. This has been demonstrated by special tests which have been conducted on operating BWRs in an attempt to force the reactor into xenon instability, and by calculations. No xenon instabilities have ever been observed in the test results. All of these indicators have proven that xenon transients are highly damped in a BWR due to the large negative power coefficient of reactivity (Reference 4.1-1).

Important features of the reactor core arrangement are as follows:

(1) The bottom-entry cruciform control rods consist of B_4C in stainless steel tubes (i.e., B_4C rods) only or a combination of B_4C rods and solid hafnium rods surrounded by stainless steel. Control Rods are further described in subsections 4.1.3.2 and subsequent sections.

(2) The fixed in-core ion chambers provide continuous power range neutron flux monitoring. A probe tube in each in-core assembly provides for a traversing ion chamber for calibration and axial detail. Source and intermediate range monitors are located in-core and are axially retractable. The in-core location of the startup and source range instruments provides coverage of the large reactor core and provides an acceptable signal-to-noise ratio and neutron-to-gamma ratio. All in-core instrument leads enter from the bottom and the instruments are in service during refueling. In-core instrumentation is further discussed in Subsection 7.7.1.6.

(3) As shown by experience, the operator, utilizing the in-core flux monitor system, can maintain the desired power distribution within a large core by proper control rod scheduling.

(4) The channels (Zircaloy-2 or Zircaloy-4) provide a fixed flow path for the boiling coolant, serve as a guiding surface for the control rods, and protect the fuel during handling operations.

(5) The core is designed to be subcritical at any time in its operating history with any one control rod fully withdrawn.

(6) The selected control rod pitch provides the ability to finely control the power distribution in the assemblies contained in the reactor core. The pitch also allows ample clearance below the pressure vessel between control rod drive mechanisms for ease of maintenance and removal.

4.1.2.1.2 Core Configuration

The reactor core is arranged as an upright circular cylinder containing a large number of fuel cells and is located within the reactor vessel. The coolant flows upward through the core. The Susquehanna SES Units utilize a conventional scatter loading with the lowest reactivity bundles placed in the peripheral region of the core. At periodic refueling intervals, each unit will enter an outage. During this time, fuel assemblies are identified to be discharged, new fuel is loaded and fuel assemblies in the reactor core may be "shuffled" to new locations. Therefore, the core loading patterns are both unit and cycle specific. The core configurations for each unit are discussed in Section 4.3.

4.1.2.1.3 Fuel Assembly Description

The fuel assembly is composed of fuel and water rods (or interior water channels), structural components and a fuel channel. The mechanical design of the assembly is described in Section 4.2. The nuclear design of the assembly is described in Section 4.3. Thermal hydraulic design of the assembly is described in Section 4.4.

4.1.2.1.3.1 Fuel Rod

A fuel rod consists of UO_2 pellets and a Zircaloy-2 cladding tube. A fuel rod is made by stacking pellets into a Zircaloy-2 cladding tube which is evacuated and back-filled with helium, and sealed by welding Zircaloy end plugs in each end of the tube.

The BWR fuel rod is designed as a pressure vessel. The ASME Boiler and Pressure Vessel Code, Section III, is used as a guide in the mechanical design and stress analysis of the fuel rod.

The rod is designed to withstand the applied loads, both external and internal. The fuel pellet is sized to provide sufficient volume within the fuel tube to accommodate differential expansion between fuel and clad. Overall fuel rod design is conservative in its accommodation of the mechanisms affecting fuel in a BWR environment. Fuel rod design bases are discussed in more detail in Section 4.2.

4.1.2.1.3.2 Fuel Bundle

The fuel bundle has two important design features:

(1) The bundle design places minimum external forces on a fuel rod; each fuel rod is free to expand in the axial direction.

(2) The unique structural design permits the removal and replacement, if required, of individual fuel rods.

Fuel bundles are designed to meet all the criteria for core performance and to provide ease of handling.

Selected fuel rods in each assembly may differ from the others in initial uranium enrichment, burnable poison content, and fuel rod length. The variation in enrichment and burnable poison distribution produces more uniform power production across the fuel assembly, and thus allows a significant reduction in the amount of heat transfer surface required to satisfy the design thermal limitations. The inclusion of part length fuel rods in the assembly improves the two phase pressure drop, enhances the inherent stability of the bundle, and improves the required shutdown margin of the core design.

Section 4.2 provides a more detailed description of the mechanical design aspects of the fuel bundles in use at Susquehanna.

4.1.2.1.4 Assembly Support and Control Rod Location

All peripheral fuel assemblies are supported by the core plate. Otherwise, individual fuel assemblies in the core rest on fuel support pieces mounted on top of the control rod guide tubes. Each guide tube, with its fuel support piece, bears the weight of four assemblies and is supported by a control rod drive penetration nozzle in the bottom head of the reactor vessel. The core plate provides lateral support and guidance at the top of each control rod guide tube.

The top guide, mounted inside the shroud, provides lateral support and guidance for each fuel assembly. The reactivity of the core is controlled by cruciform control rods which occupy alternate spaces between fuel assemblies. The position of each control rod is controlled by independent mechanical hydraulic drive systems. These systems insert and withdraw the control rod from the bottom of the core and can accurately position its associated control rod during normal operation and yet exert approximately ten times the force of gravity to insert the control rod during the scram mode of operation. Bottom entry allows optimum power shaping in the core, ease of refueling, and convenient drive maintenance.

4.1.2.2 Shroud

The shroud is a cylindrical, stainless steel structure which surrounds the core and provides a barrier to separate the upward flow through the core from the downward flow in the annulus, and also provides a floodable volume in the unlikely event of an accident which tends to drain the reactor pressure vessel. A flange at the top of the shroud mates with a flange on the shroud head and steam separators. The upper cylindrical wall of the shroud and the shroud head form the core discharge plenum. The jet pump diffusers penetrate the shroud support below the core elevation to introduce the coolant to the bottom head volume. The shroud support is designed to support and locate the jet pumps, core support structure, peripheral fuel assemblies and to separate the inlet and outlet flows of the recirculation loops.

Mounted inside the upper shroud cylinder in the space between the top of the core and the upper shroud flange are the core spray spargers with spray nozzles for injection of cooling water. The core spray spargers and nozzles do not interfere with the installation or removal of fuel from the core.

4.1.2.3 Shroud Head and Steam Separators

The shroud head consists of a flange and dome onto which is welded an array of standpipes, with a steam separator located at the top of each standpipe. The shroud head mounts on the flange at the top of the cylinder and forms the cover of the core discharge plenum region. The joint between the shroud head and shroud flange does not require a gasket or other replacement sealing technique. The fixed axial flow-type steam separators have no moving parts and are made of stainless steel.

In each separator, the steam-water mixture rising from the standpipe impinges on vanes which give the mixture a spin to establish a vortex wherein the centrifugal forces separate the steam from the water. Steam leaves the separator at the top and passes into the wet steam plenum below the dryer. The separated water exits from the lower end of the separator and enters the pool that surrounds the standpipes to enter the downcomer annulus.

For ease of removal, the shroud head is bolted to the shroud top flange by long shroud head bolts that extend above the separators for easy access during refueling. The shroud head is guided into position on the shroud via guide rods on the inside of the vessel and locating pins located on the shroud head. The objective of the shroud head bolt design is to provide direct access to the bolts during reactor refueling operations with underwater tool manipulation during the removal and installation of the assemblies.

4.1.2.4 Steam Dryer Assembly

The steam dryer assembly is mounted in the reactor vessel above the shroud head and forms the top and sides of the wet steam plenum. Vertical guide rods on the inside of the vessel provide alignment for the dryer assembly during installation. The dryer assembly is supported by pads extending from the vessel wall and is prevented from lifting during postulated transients by brackets welded to the reactor vessel top head. Steam from the separators flows upward into the dryer assembly. Moisture is removed by the dryer vanes and flows first through a system of troughs and pipes to the pool surrounding the separators and then into the downcomer annulus between the core shroud and reactor vessel wall. The steam leaving the top of the dryer assembly flows into vessel steam outlet nozzles which are located alongside the steam dryer assembly. The schematics of a typical steam dryer panel are shown in Figures 4.1-2 and 4.1-3.

4.1.3 REACTIVITY CONTROL SYSTEMS

4.1.3.1 Operation

The control rods perform dual functions of power distribution shaping and reactivity control. Power distribution in the core is controlled during operation of the reactor by manipulation of selected patterns of rods. The rods, which enter from the bottom of the near-cylindrical reactor core, are positioned in such a manner to counter-balance steam voids in the top of the core and which results in significant power flattening.

These groups of control elements, used for power flattening, experience a somewhat higher duty cycle and neutron exposure than the other rods in the control system.

The reactivity control function requires that all rods be available for either reactor "scram" (prompt shutdown) or reactivity regulation. Because of this, the control elements are mechanically designed to withstand the dynamic forces resulting from a scram. They are connected to bottom-mounted, hydraulically actuated drive mechanisms which allow either axial positioning for reactivity regulation or rapid scram insertion.

The design of the rod-to-drive connection permits each blade to be attached or detached from its drive without disturbing the remainder of the control system. The bottom-mounted drives permit the entire control system to be left intact and operable for tests with the reactor vessel open.

4.1.3.2 Description of Rods

For the original equipment and Duralife D160-C control rods the neutron absorber portion of the control rod is contained in the wings of the cruciform shaped control blades which are inserted in the bypass region between four fuel assemblies. The original equipment control blades contain boron-carbide (B_4C) powder filled stainless steel absorber tubes. Newer generation control blades contain a combination of B_4C filled tubes and solid hafnium rods. The boron-carbide absorber tubes are seal welded with end plugs on either end. Stainless steel balls are used to separate the tubes into individual compartments. The stainless steel balls are held in position by a slight crimp in the tube. The individual tubes act as pressure vessels to contain the helium gas released by the boron-neutron capture reaction. The tubes or rods are held in a cruciform array by a stainless steel sheath extending the full length of the tubes.

A top handle aligns the tubes and provides structural rigidity at the top of the control rod. Rollers, housed in the handle, provide guidance for control rod insertion and withdrawal. A bottom casting is also used to provide structural rigidity and contains positioning rollers and a parachute-shaped velocity limiter. The handle and lower casting are welded into a single structure by means of a small cruciform post located in the center of the control rod.

Replacement Marathon control rods may use a modified handle assembly that eliminates pins and rollers present in the earlier design.

Marathon control blade wings are made up of an array of square tubes welded together. The tube arrays are welded to center tie rods to form the cruciform blade shape. The square tubes are loaded with either B_4C or Hafnium. The B_4C is contained in separate capsules to prevent migration within the tubes. The square tubes are sealed at each end to prevent the neutron poisons from washing out into the coolant. The blade handle and velocity limiter are equivalent to previous control blade designs, (Reference 4.1-24). The Marathon Ultra – HD Control Rod design was introduced in U2C18, (Reference 4.1-30).

Westinghouse CR 99 control rods, introduced in U1C20, are designed similar to the original equipment and newer style Marathon control rod, (Reference 4.1-31). Like the above GE control rods, the Westinghouse CR 99 control rods have a cruciform blade shape, a handle, a B_4C loaded absorber zone and a velocity limiter. Different from the aforementioned GE control rods, the Westinghouse CR 99 control rod has horizontal absorber holes drilled into solid stainless steel wings and uses guide pads (buttons) or no guide pads, rather than upper pins and rollers, to guide control rod motion. Reference 4.1-31 provides additional discussion on the design of the CR 99 control rod.

The control rods can be positioned at 6-in. steps and have a nominal withdrawal and insertion speed of 3 in/sec.

The velocity limiter, an engineered safety feature (ESF), is a device which is an integral part of the control rod and protects against the low probability of a rod drop accident. It is designed to limit the free fall velocity and reactivity insertion rate of a control rod so that minimum fuel damage would occur. It is a one-way device, in that control rod scram time is not significantly affected.

Control rods are cooled by the core leakage (bypass) flow. The core leakage flow is made up of recirculation flow that leaks through the several leakage flow paths, which are:

(1) The area between fuel channel and fuel assembly nosepiece;
(2) The area between fuel assembly nosepiece and fuel support piece;
(3) The area between fuel support piece and core plate;
(4) The area between core plate and shroud; and
(5) The bypass flow holes in the fuel assembly nosepiece.

Further details of the control blade design are provided in Section 4.2.

4.1.4 ANALYSIS TECHNIQUES

4.1.4.1 Reactor Internal Components

The following computer codes were used for initial design of the reactor internal components. Code descriptions are provided for historical purposes only.

Computer codes used for the analysis of the internal components are listed as follows:

(1) MASS
(2) SNAP (MULTISHELL)
(3) GASP
(4) NOHEAT
(5) FINITE
(6) DYSEA
(7) SHELL 5
(8) HEATER
(9) FAP-71
(10) CREEP-PLAST

Detailed descriptions of these programs are given in the following sections:

4.1.4.1.1 MASS (Mechanical Analysis of Space Structure)

4.1.4.1.1.1 Program Description

The program, proprietary of the General Electric Company, is an outgrowth of the PAPA (Plate and Panel Analysis) program originally developed by L. Beitch in the early 1960s. The program is based on the principle of the finite element method. Governing matrix equations are formed in terms of joint displacements using a "stiffness-influence-coefficient" concept originally proposed by L. Beitch (Reference 4.1-2). The program offers curved beam, plate, and shell elements. It can handle mechanical and thermal loads in a static analysis and predict natural frequencies and mode shapes in a dynamic analysis.

4.1.4.1.1.2 Program Version and Computer

The GE Nuclear Energy Division is using a past revision of MASS. This revision is identified as revision "0" in the computer production library. The program operates on the Honeywell 6000 computer.

4.1.4.1.1.3 History of Use

Since its development in the early 60s, the program has been successfully applied to a wide variety of jet-engine structural problems, many of which involve extremely complex geometries. The use of the program in the Nuclear Energy Division also started shortly after its development.

4.1.4.1.1.4 Extent of Application

Besides the Jet Engine and Nuclear Energy Divisions, the Missile and Space Division, the Appliance Division, and the Turbine Division of General Electric have also applied the program to a wide range of engineering problems. The Nuclear Energy Division (NED) uses it mainly for piping and reactor internals analyses.

4.1.4.1.2 SNAP (MULTISHELL)

4.1.4.1.2.1 Program Description

The SNAP Program, which is also called MULTISHELL, is the General Electric Code which determines the loads, deformations, and stresses of axisymmetric shells of revolution (cylinders, cones, discs, toroids, and rings) for axisymmetric thermal boundary and surface load conditions. Thin shell theory is inherent in the solution of E. Peissner's differential equations for each shell's influence coefficients. Surface loading capability includes pressure, average temperature, and linear-through-wall gradients; the latter two may be linearly varied over the shell meridian. The theoretical limitations of this program are the same as those of classical theory.

4.1.4.1.2.2 Program Version and Computer

The current version maintained by the General Electric Jet Engine Division at Evandale, Ohio is being used on the Honeywell 6000 computer in GE/NED.

4.1.4.1.2.3 History of Use

The initial version of the Shell Analysis Program was completed by the Jet Engine Division in 1961. Since then, a considerable amount of modification and addition has been made to accommodate its broadening area of application. Its application in the Nuclear Energy Division has a history longer than ten years.

4.1.4.1.2.4 Extent of Application

The program has been used to analyze jet engine, space vehicle and nuclear reactor components. Because of its efficiency and economy, in addition to reliability, it has been one of the main shell analysis programs in the Nuclear Energy Division of General Electric.

4.1.4.1.3 GASP

4.1.4.1.3.1 Program Description

GASP is a finite element program for the stress analysis of axisymmetric or plane two-dimensional geometries. The element representations can be either quadrilateral or triangular. Axisymmetric or plane structural loads can be input at nodal points. Displacements, temperatures, pressure loads, and axial inertia can be accommodated. Effective plastic stress and strain distributions can be calculated using a bilinear stress-strain relationship by means of an iterative convergence procedure.

4.1.4.1.3.2 Program Version and Computer

The GE version, originally obtained from the developer, Professor E. L. Wilson, operates on the Honeywell 6000 computer.

4.1.4.1.3.3 History of Use

The program was developed by E. L. Wilson in 1965 (Reference 4.1-3). The present version in GE/NED has been in operation since 1967.

4.1.4.1.3.4 Extent of Application

The application of GASP in GE/NED is mainly for elastic analysis of axisymmetric and plane structures under thermal and pressure loads. The GE version has been extensively tested and used by engineers in GE.

4.1.4.1.4 NOHEAT

4.1.4.1.4.1 Program Description

The NOHEAT program is a two-dimensional and axisymmetric transient nonlinear temperature analysis program. An unconditionally stable numerical integration scheme is combined with an iteration procedure to compute temperature distribution within the body subjected to arbitrary time- and temperature-dependent boundary conditions.

This program utilizes the finite element method. Included in the analysis are the three basic forms of heat transfer, conduction, radiation, and convection, as well as internal heat generation. In addition, cooling pipe boundary conditions are also treated. The output includes temperature of all the nodal points for the time instants by the user. The program can handle multitransient temperature input.

4.1.4.1.4.2 Program Version and Computer

The current version of the program is an improvement of the program originally developed by I. Farhoomand and Professor E. L. Wilson of University of California at Berkeley (Reference 4.1-4). The program operates on the Honeywell 6000 computer.

4.1.4.1.4.3 History of Use

The program was developed in 1971 and installed in General Electric Honeywell computer by one of its original developers, I. Farhoomand, in 1972. A number of heat transfer problems related to the reactor pedestal have been satisfactorily solved using the program.

4.1.4.1.4.4 Extent of Application

The program using finite element formulation is compatible with the finite element stress-analysis computer program GASP. Such compatibility simplified the connection of the two analyses and minimizes human error.

4.1.4.1.5 FINITE

4.1.4.1.5.1 Program Description

FINITE is a general-purpose finite element computer program for elastic stress analysis of two-dimensional structural problems including (1) plane stress, (2) plane strain, and (3) axisymmetric structures. It has provision for thermal, mechanical and body force loads. The materials of the structure may be homogeneous or nonhomogeneous and isotropic or orthotopic. The development of the FINITE program is based on the GASP program. (See Subsection 4.1.4.1.3.)

4.1.4.1.5.2 Program Version and Computer

The present version of the program at GE/NED was obtained from the developer J. E. McConnelee of GE/Gas Turbine

Department in 1969 (Reference 4.1-5). The NED version is used on the Honeywell 6000 computer.

4.1.4.1.5.3 History of Use

Since its completion in 1969, the program has been widely used in the Gas Turbine and the Jet Engine Departments of the General Electric Company for the analysis of turbine components.

4.1.4.1.5.4 Extent of Usage

The program is used at GE/NED in the analysis of axisymmetric or nearly axisymmetric BWR internals.

4.1.4.1.6 DYSEA

4.1.4.1.6.1 Program Description

The DYSEA (Dynamic and Seismic Analysis) program is a GE proprietary program developed specifically for seismic and dynamic analysis of RPV and internals/building system. It calculates the dynamic response of linear structural system by either temporal modal superposition or response spectrum method. Fluid-structure interaction effect in the RPV is taken into account by way of hydrodynamic mass.

Program DYSEA was based on program SAPIV with added capability to handle the hydrodynamic mass effect. Structural stiffness and mass matrices are formulated similar to SAPIV. Solution is obtained in time domain by calculating the dynamic response mode by mode. Time integration is performed by using Newmark's b-method. Response spectrum solution is also available as an option.

4.1.4.1.6.2 Program Version and Computer

The DYSEA version now operating on the Honeywell 6000 computer of GE, Nuclear Energy Systems Division, was developed at GE by modifying the SAPIV program. Capability was added to handle the hydrodynamic mass effect due to fluid-structure interaction in the reactor. It can handle 3-Dimensional dynamic problem with beam, trusses, and springs. Both acceleration time histories and response spectra may be used as input.

4.1.4.1.6.3 History of Use

The DYSEA program was developed in the summer of 1976. It has been adopted as a standard production program since 1977 and it has been used extensively in all dynamic and seismic analysis of the RPV and internals/ building system.

4.1.4.1.6.4 Extent of Application

The current version of DYSEA has been used in all dynamic and seismic analysis since its development. Results from test problems were found to be in close agreement with those obtained from either verified programs or analytic solutions.

4.1.4.1.7 SHELL 5

4.1.4.1.7.1 Program Description

SHELL 5 is a finite shell element program used to analyze smoothly curved thin shell structures with any distribution of elastic material properties, boundary constraints, and mechanical thermal and displacement loading conditions. The basic element is triangular whose membrane displacement fields are linear polynomial functions, and whose bending displacement field is a cubic polynomial function (Reference 4.1-6). Five degrees of freedom (three displacements and two bending rotations) are obtained at each nodal point. Output displacements and stresses are in a local (tangent) surface coordinate system.

Due to the approximation of element membrane displacements by linear functions, the in-plane rotation about the surface normal is neglected. Therefore, the only rotations considered are due to bending of the shell cross section and application of the method is not recommended for shell intersection (or discontinuous surface) problems where in-plane rotation can be significant.

4.1.4.1.7.2 Program Version and Computer

A copy of the source deck of SHELL 5 is maintained in GE/NED by Y. R. Rashid, one of the originators of the program. SHELL 5 operates on the UNIVAC 1108 computer.

4.1.4.1.7.3 History of Use

SHELL 5 is a program developed by Gulf General Atomic Incorporated (Reference 4.1-7) in 1969. The program has been in production status at Gulf General Atomic, General Electric, and at other major computer operating systems since 1970.

4.1.4.1.7.4 Extent of Application

SHELL 5 has been used at General Electric to analyze reactor shroud support and torus. Satisfactory results were obtained.

4.1.4.1.8 HEATER

4.1.4.1.8.1 Program Description

HEATER is a computer program used in the hydraulic design of feedwater spargers and their associated delivery header and piping. The program utilizes test data obtained by GE using full scale mockups of feedwater spargers combined with a series of models which represent the complex mixing processes obtained in the upper plenum, downcomer, and lower plenum. Mass and energy balances throughout the nuclear steam supply system are modeled in detail (Reference 4.1-8).

4.1.4.1.8.2 Program Version and Computer

This program was developed at GE/NED in FORTRAN IV for the Honeywell 6000 computer.

4.1.4.1.8.3 History of Use

The program was developed by various individuals in GE/NED beginning in 1970. The present version of the program has been in operation since January 1972.

4.1.4.1.8.4 Extent of Application

The program is used in the hydraulic design of the feedwater spargers for each BWR plant, in the evaluation of design modifications, and the evaluation of unusual operational conditions.

4.1.4.1.9 FAP-71 (Fatigue Analysis Program)

4.1.4.1.9.1 Program Description

The FAP-71 computer code, or Fatigue Analysis Program, is a stress analysis tool used to aid in performing ASME-III Nuclear Vessel Code structural design calculations. Specifically, FAP-71 is used in determining the primary plus secondary stress range and number of allowable fatigue cycles at points of interest. For structural locations at which the 3Sm (P+Q) ASME Code limit is exceeded, the program can perform either (or both) of two elastic-plastic fatigue life evaluations: 1) the method reported in ASME Paper 68-PVP-3, 2) the present method documented in Paragraph NB-3228.3 of the 1971 Edition of the ASME Section III Nuclear Vessel Code. The program can accommodate up to 25 transient stress states of as many as 20 structural locations.

4.1.4.1.9.2 Program Version and Computer

The present version of FAP-71 was completed by L. Young of GE/NED in 1971 (Reference 4.1-9). The program currently is on the NED Honeywell 6000 computer.

4.1.4.1.9.3 History of Use

Since its completion in 1971, the program has been applied to several design analyses of GE BWR vessels.

4.1.4.1.9.4 Extent of Use

The program is used in conjunction with several shell analysis programs in determining the fatigue life of BWR mechanical components subject to thermal transients.

4.1.4.1.10 CREEP/PLAST

4.1.4.1.10.1 Program Description

A finite element program is used for the analysis of two-dimensional (plane and axisymmetric) problems under conditions of creep and plasticity. The creep formulation is based on the memory theory of creep in which the constitutive relations are cast in the form of hereditary integrals. The material creep properties are built into the program and they represent annealed 304 stainless steel. Any other creep properties can be included if required.

The plasticity treatment is based on kinematic hardening and von Mises yield criterion. The hardening modulus can be constant or a function of strain.

4.1.4.1.10.2 Program Version and Computer

The program can be used for elastic-plastic analysis with or without the presence of creep. It can also be used for creep analysis without the presence of instantaneous plasticity. A detailed description of theory is given in Reference 4.1-11. The program is operative on Univac-1108.

4.1.4.1.10.3 History of Use

This program was developed by Y. R. Rashid (Ref. 4.1-11) in 1971. It underwent extensive program testing before it was put on production status.

4.1.4.1.10.4 Extent of Application

The program is used at GE/NED in the channel cross section mechanical analysis.

4.1.4.2 Fuel Rod Thermal Analysis

Fuel Rod Thermal Design Analyses are performed utilizing the classical relationships for heat transfer in cylindrical coordinate geometry with internal heat generation. Steady state fuel rod thermal-mechanical analyses are performed to assure that fuel rod thermal-mechanical limits (e.g., steady state cladding strain and stress, hydrogen absorption, and, corrosion, etc.) are not exceeded. Abnormal operational transients are also evaluated to assure that the damage limit of 1.0% cladding plastic strain is not violated.

Fuel rod analyses were performed with RAMPEX and approved versions of RODEX2, RODEX2A, and COLAPX codes. The fuel rod performance characteristics modeled by the RODEX2 and RODEX2A codes are:

- Gas release
- Radial thermal conduction and gap conductance
- Free rod volume and gas pressure calculations

- Pellet clad interaction (PCI)
- Fuel swelling, densification, cracking, and crack healing
- Cladding creep deformation and irradiation induced growth

RODEX2 determines the initial conditions for fuel rod power ramping analysis, performed using RAMPEX.

RODEX2A (Reference 4.1-28) determines the steady state strain, internal pressure, fuel cladding temperature, corrosion, hydrogen absorption, fuel temperature, and the fuel rod internal pressure for creep collapse analysis. This computer code is used to determine gap conductance for transient analysis.

Creep collapse analysis is performed using the COLAPX code.

Section 4.2 presents the fuel rod mechanical design and associated methodology.

4.1.4.3 Reactor Systems Dynamics

The analysis techniques and computer codes used in reactor systems dynamics are described in section 4 of Reference 4.1-10. Subsection 4.4.4.6 also provides a complete stability analysis for the reactor coolant system.

Channel and core stability analyses were performed by Framatome ANP, Inc. on a fuel design and cycle specific basis using the STAIF code. A description of the methods employed by the STAIF code is provided in Reference 4.1-20. Using the RAMONA 5 code (References 4.1-26 and 27), FANP also performs transient stability analyses in support of the generation of OPRM setpoints.

4.1.4.4 Nuclear Engineering Analysis

A brief summary of principal computer codes used in reactor core design and analysis is provided below.

4.1.4.4.1 CASMO-4

The CASMO-4 computer code (Reference 4.1-25) was developed by STUDSVIK of America to perform steady state modeling of fuel bundles. CASMO-4 uses deterministic transport methods. At the pin cell level it exclusively uses a collision probability method to collapse the energy nuclear data into multi-group data. At the lattice level, it uses a method of characteristics for the neutron equation solution. CASMO-4, as opposed to CASMO-3G, does not need to do pin cell homogenization to perform a 2-D lattice wide transport calculation. The code is used to model each unique fuel lattice in the reactor to calculate few-group cross sections, bundle reactivities and relative fuel rod powers within a fuel bundle. The effects of conditions such as void, control rod presence, moderator temperature, fuel temperature, soluble boron, etc., are included in the model.

4.1.4.4.2 MICROBURN-B2

The MICROBURN-B2 code (Reference 4.1-25) solves a two-group neutron diffusion equation based on an interface current method. It calculates the burnup chain equation for heavy nuclides and burnable poison nuclides, determines the three-dimensional core nodal power distribution, bundle flow, and void distributions. It also determines pin power distributions and thermal margins to technical specification limits. MICROBURN-B2 is used with CASMO-4.

4.1.4.5 Neutron Fluence Calculations

Vessel neutron fluence calculations were performed to determine the azimuthal and axial variation of fluence at the vessel inside surface and at 1/4 T depth. The azimuthal and axial results were synthesized to obtain the fluence profile at the vessel inside surface and 1/4 T depth. The calculations also evaluate vessel fluence at power uprate conditions.

Sections 4.3 and 5.3 provide additional detail regarding reactor pressure vessel irradiation.

4.1.4.6 Thermal Hydraulic Calculations

XCOBRA (References 4.1-21, 4.1-22, and 4.1-23) calculates the steady state thermal hydraulic performance of a BWR. The code determines the flow and local fluid conditions at various axial positions in the core and represents the core as a collection of discrete parallel channels. The only interaction allowed between channels is the equalization of pressure in the inlet and outlet plenums. This is achieved by allowing the core flow to distribute among the various flow channels until the pressure drop in each channel is equalized.

Pressure drop in each channel is determined through the application of two-phase pressure drop correlations and various data which hydraulically characterize the fluid channel. At a given axial position in the core, XCOBRA calculates a core-wide distribution of flow, enthalpy, density, quality, void fraction, and mass velocity.

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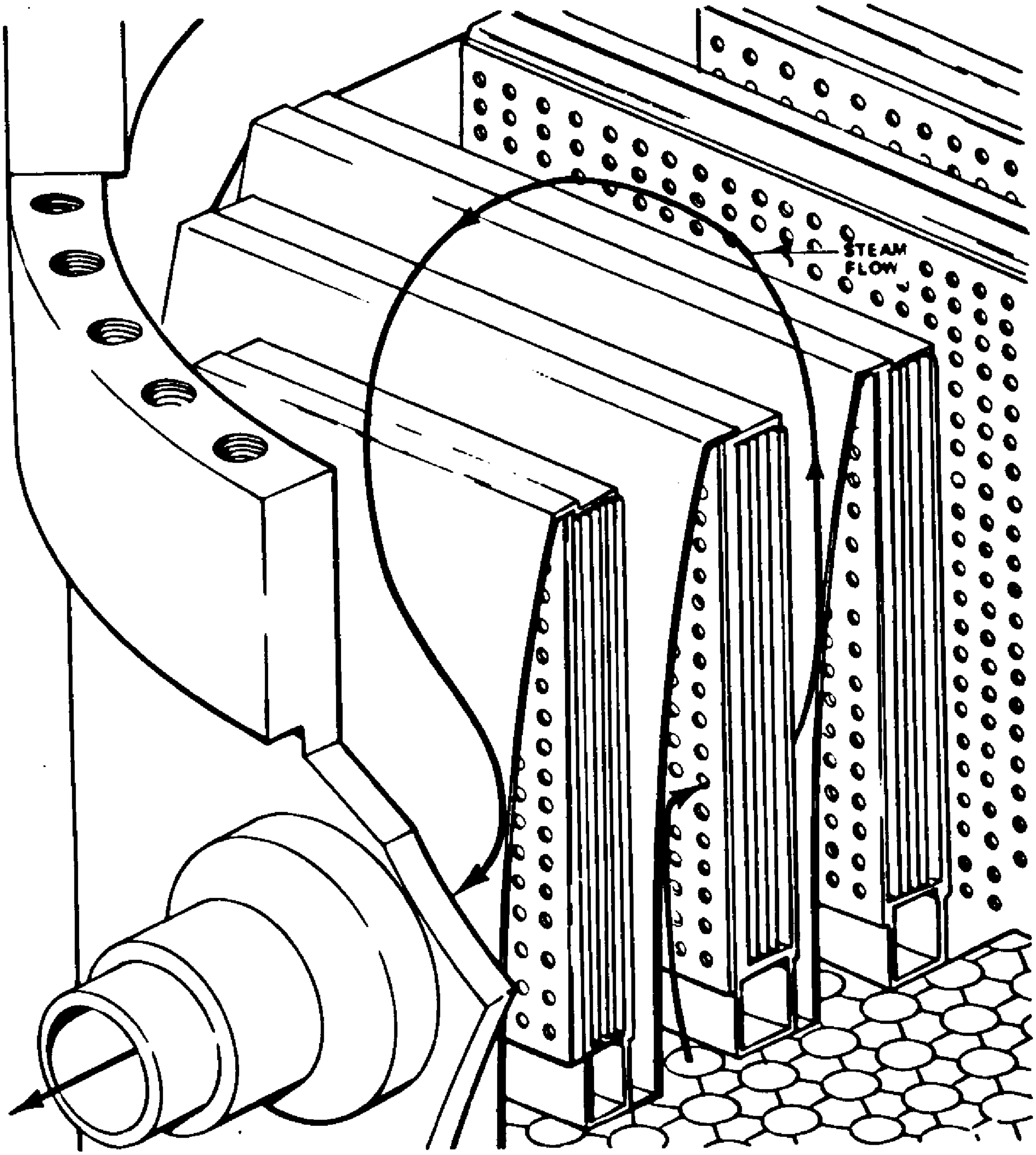

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UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

STEAM DRYER PANEL

FIGURE 4.1-2, Rev. 47

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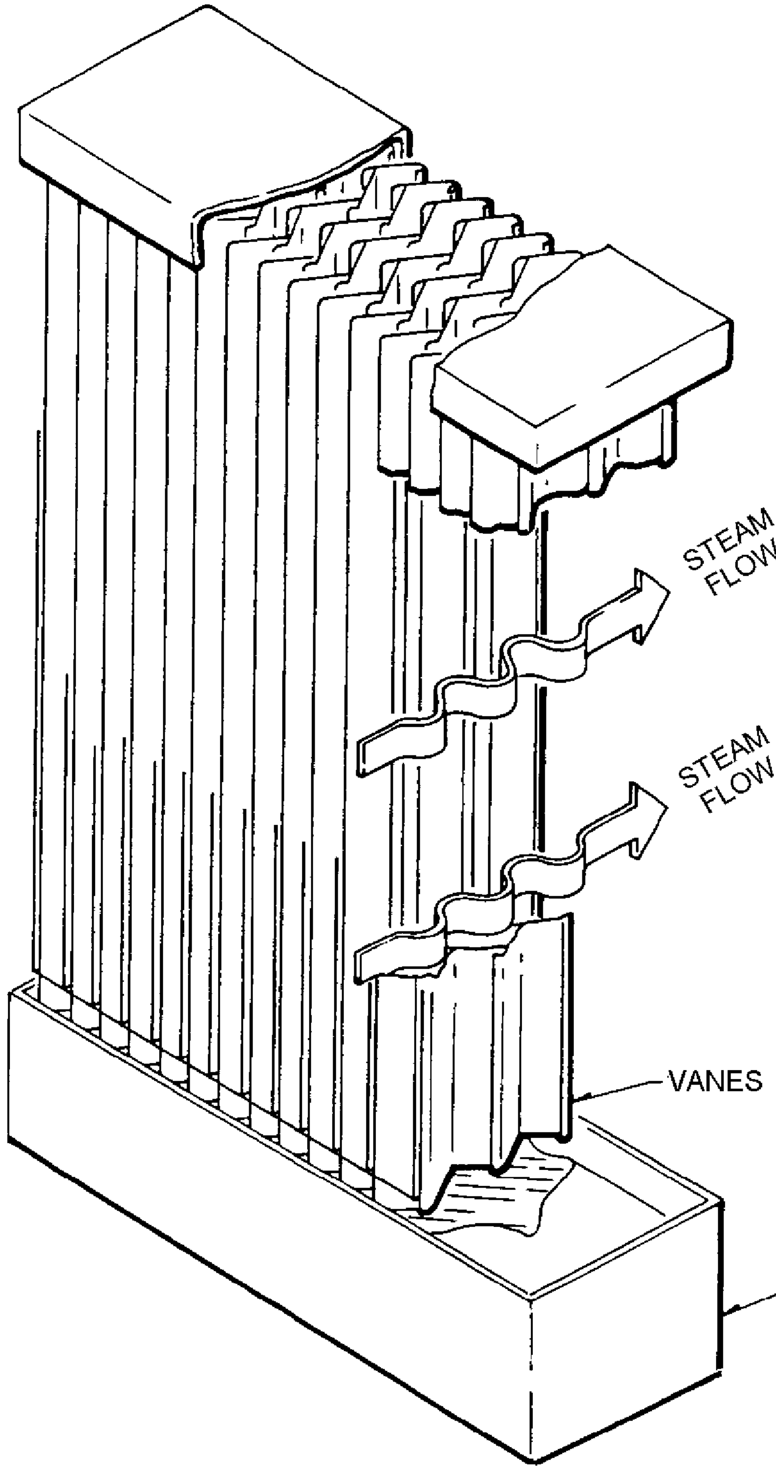

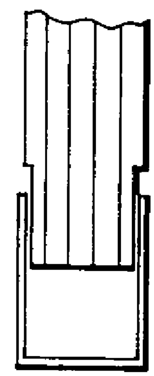

CROSS-SECTION VIEW

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STEAM DRYER

FIGURE 4.1-3, Rev. 54

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4.2 FUEL SYSTEM DESIGN

The fuel system includes the fuel assembly (channeled fuel bundle) and the portion of the control rod assembly which extends above the coupling mechanism on the control rod drive. The following sections discuss the thermal/mechanical design bases, design descriptions, and design evaluations for the fuel system components. Nuclear design is described in Section 4.3 and thermal hydraulic design is described in Section 4.4.

4.2.1 Design Bases

4.2.1.1 Fuel Assembly

The core designs described in Section 4.3 contain one fuel assembly design. The FANP ATRIUM™-10 is the primary fuel type loaded into the core. Occasionally, Lead Use Assemblies (LUAs) are also used in the core to provide operating experience with alternative fuel designs. When used, LUAs are loaded in non-limiting locations in the core.

FANP ATRIUM™-10 Fuel

The mechanical design for the ATRIUM-10™ assembly is based on compliance with generic mechanical design criteria established by FANP and approved by the NRC in Reference 4.2.6-10.

In accordance with the requirements of the approved mechanical design criteria, the ATRIUM™-10 mechanical analyses were performed to provide the following assurances.

1) The fuel assembly shall not fail as a result of normal operation and AOO's.

2) Damage to fuel assemblies shall never prevent control rod insertion when required.

3) The number of fuel rod failures is not underestimated for postulated accidents.

4) Fuel coolability shall always be maintained.

Mechanical design analyses have been performed to evaluate the cladding stress and strain limits, fretting wear, oxidation, hydriding and crud buildup, fuel rod bowing, differential fuel rod growth, internal hydriding, cladding collapse, and cladding and fuel pellet overheat. The RODEX2, RODEX2A, RAMPEX, and COLAPX codes were used in the mechanical design analyses.

Mechanical analyses have also been performed to evaluate the ATRIUM™-10 fuel design for Seismic/LOCA loads and for normal shipping and handling. In addition, FANP ATRIUM™-10 fuel has been evaluated for power uprate conditions.

Results of these analyses and evaluations are discussed in Section 4.2.3.

Lead Use Assemblies (LUA)

Occasionally, LUAs are loaded into the reactor core. The core loading of the LUAs will be such that the assemblies are not loaded into thermally limiting locations. Mechanical design analyses are performed for the LUA to evaluate fuel design parameters similar to that provided for the reload fuel.

4.2.1.2 Original Equipment Control Rod Assembly

The design bases for the original equipment control rod assembly are presented in Reference 4.2.6-3. The End-of-Life evaluation for the original equipment control rod assembly was modified in accordance with PPL's response to IE Bulletin 79-26, Rev. 1. PPL has committed to replacing these control rod assemblies prior to exceeding a limit of 34% B^{10} depletion averaged over the upper one-fourth of the control rod assembly.

4.2.1.3 GE Duralife 160C Control Rod Assembly

The Duralife 160C control rod has been evaluated to assure it has adequate structural margin under loading due to handling, and normal, emergency, and faulted operating modes. The loads evaluated include those due to normal operating transients (scram and jogging), pressure differentials, thermal gradients, seismic deflection, irradiation growth, and all other lateral and vertical loads expected for each condition. The Duralife 160C control rod assembly design bases have been reviewed and approved by the NRC (References 4.2.6-4 and 4.2.6-5).

4.2.1.4 GE Marathon Control Rod Assembly

The Marathon control rod has been evaluated to assure it has adequate structural margin under loading due to handling, and normal, emergency, and faulted operating modes. The loads evaluated include those due to normal operating transients (scram and jogging), pressure differentials, thermal gradients, seismic deflection, irradiation growth, and all other lateral and vertical loads expected for each condition. The Marathon control rod assembly design bases have been reviewed and approved by the NRC (References 4.2.6-12 and 4.2.6-16).

4.2.1.5 Westinghouse CR 99 Control Rod Assembly

The Westinghouse CR 99 control rod has been evaluated to assure it has adequate structural margin under loading due to handling, and normal, emergency, and faulted operating modes. The loads evaluated include those due to normal operating transients (scram and jogging), pressure differentials, thermal gradients, seismic deflection, irradiation growth, and all other lateral and vertical loads expected for each condition. The Westinghouse CR 99 control rod assembly design bases have been reviewed and approved by the NRC (Reference 4.2.6-17).

4.2.2 General Design Description

A summary of fuel design characteristics is provided in Table 4.2-14 for fuel designs loaded in the core at SSES.

4.2.2.1 Core Cell

A core cell consists of a control rod assembly and the four fuel assemblies which immediately surround the control rod. Figure 4.2-15 provides nominal dimensions for a core cell loaded with FANP ATRIUM™-10 fuel and a Duralife 160C control rod. Figures 4.2-15A and 4.2.15B provide nominal dimensions for a core cell loaded with FANP ATRIUM™-10 fuel and a Marathon C+ control rod and a Marathon Ultra – HD control rod, respectively. Figure 4.2-15C provides the nominal dimensions for a core cell loaded with FANP ATRIUM™-10 fuel and a Westinghouse CR 99 control rod. These figures illustrate the general layout of a core cell while providing nominal dimensions for fuel and control rods. A core cell may contain multiple fuel types, regardless of control rod type utilized.

Each core cell is associated with a four-lobed fuel support piece. Around the outer edge of the core, certain fuel assemblies are not immediately adjacent to a control rod and are supported by individual peripheral fuel support pieces.

4.2.2.2 Fuel Bundle

FANP ATRIUM™-10 Fuel

An FANP ATRIUM™-10 fuel bundle contains 83 full length and 8 part length fuel rods. The 8 part length fuel rods are provided to decrease the two phase pressure loss in the top of the bundle thereby providing fuel bundle design that is more stable. A central water channel, which displaces a 3x3 array of fuel rods near the center of the bundle, provides additional moderation within the bundle thereby enhancing fuel utilization. The water channel is also used to fix the spacer locations within the fuel bundle and serves as the main structural member connecting the upper and lower tie plates. A total of 8 spacers are used to maintain fuel rod spacing. Reference 4.2.6-11 provides detailed discussion of the various components of the ATRIUM™-10 fuel bundle. Nominal dimensions for the ATRIUM™-10 fuel bundle are provided in Figure 4.2-15. A schematic of an ATRIUM™-10 fuel bundle is shown in Figure 4.2-17.

4.2.2.3 Fuel Assembly

A fuel assembly is a fuel bundle including the surrounding fuel channel. The fuel assemblies are arranged in the reactor core to approximate a right circular cylinder inside the core shroud. Each fuel assembly is supported by a fuel support piece and the top guide.

The fuel channel enclosing the FANP fuel bundles is fabricated from Zircaloy-4 or Zircaloy-2 and performs these functions: the channel separates flow inside the bundle from the bypass flow between channels; the channel guides the control rod and provides a bearing surface for it; the channel provides rigidity for the fuel bundle. A channel fastener attaches the fuel channel to the fuel bundle using a threaded post on the upper tie plate. Once the channel fastener has been installed and the fuel assembly has been positioned in the core, the spring on the channel fastener helps to hold the assembly against the core grid. A schematic of a fuel channel is shown in Figure 4.2-18. The fuel channel for U2C12 has a slightly different design and is shown in Figure 4.2-18-1.

The fuel channel for U1C14 reload fuel and subsequent reloads for Units 1 and 2 has been changed to a 100 mill wall thickness and is shown in Figure 4.2-18-2, (Reference 4.2.6-9). Co-resident fuel assemblies use the 80-mil fuel channel design. The difference between the 80-mil and 100-mil channel is the thickness of the channel wall. Because the inside of the channel is the same for the two designs, the top 20.1” of the 100-mil channel outer sides that face the upper guide is reduced to 80-mil. This wall thickness will then allow for the bundle to be in the same lateral position as the 80-mil channel. To maintain compatibility with an adjacent 80-mil channel fastener in a control cell, there are also slight modifications to the 100-mil channel fastener to accommodate the 100-mil channel. Starting with U1C15 the fuel channel mechanical design is based on Reference 4.2.6-8.

The Advanced Fuel Channel (AFC) has been introduced on fuel for Units 1 and 2 beginning with the reload for U1C20 and is shown in Figure 4.2-18-3. Co-resident fuel assemblies use the standard 100-mil fuel channel design. The AFC dimensions are the same as the standard 100-mil fuel channels with the difference that the AFC has thinned side-walls in the active core region. The AFC use similar channel fasteners as the standard 100-mil fuel channels. Details regarding the AFC are available in Reference 4.2.6-8.

U1C14 reload fuel uses the Framatome-ANP FUELGUARD lower tie plate design. The FUELGUARD lower tie plate has the same outer envelope dimensions as the small hole lower tie plate, including the same seal/finger springs. The basic difference between the two types of lower tie plates is the grid. The FUELGUARD grid is constructed from wavy plates with support bars for the fuel pins to sit on. The small hole design has machined holes to form a grid for the coolant to flow through and the fuel pins to sit on. Framatome ANP performed flow tests to demonstrate that the FUELGUARD lower tie plate is hydraulically compatibility with the small hole lower tie plate. Both lower tie plates basically have the same loss coefficient. Subsequent reloads for Units 1 and 2 will use the Framatome-ANP FUELGUARD lower tie plate design.

A design change of the upper locking mechanism was incorporated into the ATRIUM-10 fuel assemblies commencing with the Unit 2 Cycle 16 core loading. This design change is referred to as the “Harmonized Advanced Load Chain” or HALC and replaces the previous load chain design. The HALC was made to improve manufacturing reliability and improve the ability to remove the upper tie plate of the fuel assembly. The design change does not affect the thermal, hydraulic or mechanical performance of the fuel assembly. Subsequent reloads for Units 1 and 2 will use the HALC design.

Proper assembly orientation in the core is verified by visual inspection and is assured by verification procedures during core loading. Five visual indications of proper fuel assembly orientation exist. These indications are:

1) The channel fastener assemblies are located at one corner of each fuel assembly adjacent to the center of the control rod.

2) The orientation boss on the fuel assembly handle points toward the adjacent control rod.

3) The channel spacing buttons are adjacent to the control rod passage area.

4) The assembly identification numbers which are located on the fuel assembly handles are all readable from the direction of the center of the cell.

5) There is cell-to-cell replication.

Proper assembly orientation in the core is shown in Figure 4.2-19.

4.2.2.4 Reactivity Control Assembly

4.2.2.4.1 Original Equipment Control Rod Assembly

The control rod consists of a sheathed cruciform array of commercial grade stainless steel tubes filled with B_4C powder. The main structural member of a control rod is made of Type-304 stainless steel and consists of a top handle, a bottom casting with a velocity limiter and control rod drive coupling, a vertical cruciform center post, and four U-shaped absorber tube sheaths. Control rods are cooled by core bypass flow.

Stellite rollers with Haynes Alloy 25 pins located at the top and bottom of the control rod help guide the control rod as it is inserted and withdrawn from the core.

The control rod velocity limiter consists of cast austenitic stainless steel (Grade CF-8) and is an integral part of the control rod bottom casting. The velocity limiter protects against high reactivity insertion rate by limiting the control rod velocity in the event of a control rod drop accident.

Reference 4.2.6-3 provides the general design characteristics of the original equipment control rod assembly. A diagram of this control rod is provided in Figure 4.2-20.

4.2.2.4.2 Duralife 160C Control Rod Assembly

The main differences between the Duralife 160C control rods and the original equipment control rods are:

- the Duralife 160C control rods utilize three solid hafnium rods at each edge of the cruciform to replace the three B_4C rods that are most susceptible to cracking and to increase control rod life;

- the Duralife 160C control rods utilize improved B_4C tube material (i.e. high purity type 304 stainless steel vs. commercial purity stainless steel) to eliminate cracking in the remaining B_4C rods during the lifetime of the control rod;

- the Duralife 160C control rods use GE's crevice-free structure design, which includes additional B_4C tubes in place of the stiffeners, an increased sheath thickness, a full length weld to attach the handle and velocity limiter, and additional coolant holes at the top and bottom of the sheath;

- the Duralife 160C control rods utilize low cobalt-bearing pin and roller materials in place of stellite which was previously used (PH13-8 Mo for pins, Inconel X750 for rollers);

- the Duralife 160C control rod handles are longer by approximately 3.1 inches. The extended handle provides lateral support against the top grid and facilitates fuel moves within the reactor vessel during refueling outages;

- the Duralife 160C control rods are roughly 10% to 15% heavier (depending on the velocity limiter design) as a result of the design changes described above; and

- the Duralife 160C control rod velocity limiter material is cast austenitic stainless steel grade CF-3.

References 4.2.6-4 and 4.2.6-5 provide additional discussion on the design of the various Duralife control rod assemblies, including the features of the Duralife 160C inserted in the SSES Units. A cross section of the Duralife 160C control rod is shown in Figure 4.2-15, and a diagram of this control rod is provided in Figure 4.2-21.

4.2.2.4.3 GE Marathon Control Rod Assembly

The main difference between the Marathon control rod and the Duralife design control rods are:

- the absorber tube and sheath arrangement of the Duralife designs is replaced with an array of square tubes resulting in reduced weight and increased absorber volume;

- the full length center tie rod is replaced with a segmented tie rod which also reduces weight.

References 4.2.6-12 and 4.2.6-16 provide additional discussion on the designs of the Marathon control rod assemblies. A cross section of the Marathon C+ control rod is shown in Figure 4.2-15A. A cross section of the Marathon Ultra HD Control Rod is shown in Figure 4.2-15B. A diagram of the control rod is provided in Figure 4.2-22.

Replacement Marathon control rods may use a modified handle assembly that eliminates pins and rollers present in the earlier design. Figure 4.2-22 is applicable to the rollerless design; the only difference is that the small circles depicting the rollers would be removed from the diagram for the rollerless design. The overall shape and dimensions of the upper handle remains unchanged.

4.2.2.4.4 Westinghouse CR 99 Control Rod Assembly

The Westinghouse BWR CR 99 control rod design is comparable to that of the GE design. Both the Westinghouse and GE control blade designs have a coupling socket, velocity limiter, coupling release handle, 143-inch active absorber zone, B_4C as their main absorber material and a handle. The overall length of the Westinghouse CR 99 control rod is the same as the OE GE control rod design at 173-inches. The main difference in the design between the Westinghouse and GE control rod is that the Westinghouse CR 99 control blade has horizontal absorber holes drilled in solid stainless steel wings. Reference 4.2.6-17 provides additional discussion on the design of the CR 99 control rod. A diagram of the CR 99 control rod is provided in Figure 4.2-23.

4.2.3 Design Evaluations

4.2.3.1 Fuel Design Evaluations

FANP ATRIUM™-10 Fuel

For the FANP ATRIUM™-10 fuel, the design is such that adequate margins to fuel mechanical design limits (e.g., centerline melting temperature, transient strain, etc.) are assured for all anticipated operational occurrences (AOOs) throughout the life of the fuel as demonstrated by the fuel mechanical design analyses (References 4.2.6-10 and 4.2.6-11), provided that the steady state fuel rod power remains within the power history assumed in the analyses. The design steady state power history for the FANP ATRIUM™-10 fuel is shown in Reference 4.2.6-11 and is incorporated into the Unit/Cycle specific Core Operating Limits Report (COLR) as an operating limit. The operating limit may be in terms of planar or pellet exposure. ARTS has been implemented for Unit 1 and Unit 2. The COLR has a flow dependent LHGR multiplier and a power dependent multiplier, which are used to adjust the LHGR limit at off-rated conditions to assure that design limits are not exceeded. The mechanical analyses support a maximum assembly average exposure of 49,400 MWD/MTU for fresh ATRIUM™-10 fuel loaded in Cycles 12 and 13 on Unit 1 and Cycles 10 and 11 on Unit 2. Commencing with Unit 1 Cycle 14 and Unit 2 Cycle 12, the mechanical analyses for fresh ATRIUM™-10 assemblies support a maximum assembly average exposure of 54,000 MWd/MTU.

For U2C14, ARTS has been implemented. With ARTS the need for the FRTP/MFLPD adjustment factor has been eliminated from the U2C14 Technical Specifications and the COLR. For U2C14, the COLR has a flow dependent LHGR multiplier and a power dependent multiplier, which are used to adjust the LHGR limit at off-rated conditions.

FANP has evaluated the performance of ATRIUM™-10 fuel assemblies under Susquehanna Seismic LOCA conditions. For this evaluation, maximum loads and/or stresses were calculated for the fuel components under an acceleration load equivalent to a maximum dynamic load which bounds the allowable bending moment in BWR/4 80 mil and 100 mil fuel channels, (Reference 4.2.6-9). The large margin that resulted from these analyses shows that the ATRIUM™-10 fuel assembly with an 80 mil or 100 mil fuel channel demonstrates adequate structural integrity in the Susquehanna Units under Seismic LOCA conditions. With regard to assembly liftoff, the net force for the ATRIUM™-10 fuel assembly was found to be downwards. Starting with U1C15 the fuel channel mechanical design is based on Reference 4.2.6-8.

4.2.3.2 Results of Control Rod Assembly Design Evaluations

4.2.3.2.1 Original Equipment Control Rod Assembly

The original equipment control rod assembly design evaluations are discussed in Reference 4.2.6-3. Subsequent to the completion of the above referenced evaluations, a new failure mechanism was identified for the original equipment control rod assembly. IE Bulletin 79-26 Rev-1 discusses this failure mechanism and recommends a reduction in the end-of-life criteria for the original equipment control rod assembly. PPL committed to replacing the original equipment control rod assemblies in accordance with this IE bulletin.

4.2.3.2.2 Duralife 160C Control Rod Assembly

The Duralife 160C control rod stresses, strains, and cumulative fatigue have been evaluated and result in an acceptable margin to safety. The control rod insertion capability has been evaluated and found to be acceptable during all modes of plant operation within the limits of plant analyses. The Duralife 160C control rod coupling mechanism is equivalent to the original equipment coupling mechanism, and is therefore fully compatible with the existing control rod drives in the plant. In addition, the materials used in the Duralife 160C are compatible with the reactor environment. The Duralife 160C control rods are roughly 10% to 15% heavier than the original equipment control rod assembly, depending on the velocity limiter design utilized. The impact of the increased weight of the control rods on the seismic and hydrodynamic load evaluation of the reactor vessel and internals has been evaluated and found to be negligible.

With the exception of the crevice-free structure and the extended handle, the Duralife 160C control rod is equivalent to the NRC approved Hybrid I Control Rod Assembly (Reference 4.2.6-4). The mechanical aspects of the crevice-free structure were approved by the NRC for all control rod designs in Reference 4.2.6-5. A neutronics evaluation of the crevice-free structure for the Duralife 160C design was performed by GE using the same NRC approved nuclear interchangeability evaluation methodology as described in Reference 4.2.6-4. These calculations were performed for the original equipment control rods and the Duralife 160C control rods assuming an infinite array of FANP 9x9-2 fuel. The Duralife 160C control rod has a slightly higher worth than the original equipment design, but the increase in worth is within the criterion for nuclear interchangeability. The increase in rod worth has been taken into account in the appropriate reload analyses.

In Reference 4.2.6-4, the NRC approved the Hybrid I (Duralife 160C) control rod which weighs less than the D lattice control rod. The basis of the Control Rod Drop Accident analysis continues to be conservative with respect to control rod drop speed since the Duralife 160C control rod (including the extended handle, crevice free structure, and heavier velocity limiter) weighs less than the D lattice control rod, and the heavier D lattice control rod drop speed is used in the analysis. In addition, GE performed scram time analyses and determined that the Duralife 160C control rod scram times are not significantly different than the original equipment control rod scram times.

Also, the scram speeds are monitored in the plant to assure compliance with safety analysis assumptions and technical specification limits.

IE Bulletin 79-26, Rev. 1 was issued to address B_4C rod cracking and subsequent loss of boron in GE original equipment control rods. The Duralife 160C control rod design contains solid hafnium absorber rods in locations where B_4C tubes have historically failed. The remaining B_4C rods are manufactured with an improved tubing material (high purity stainless steel vs. commercial purity stainless steel), thus, boron loss due to cracking is not expected to occur.

Due to the control rod design, IE Bulletin 79-26, Rev. 1 does not apply to Duralife 160C control rods. However, PPL plans to continue tracking the depletion of each control rod and discharge any control rod prior to a ten percent loss in reactivity worth.

4.2.3.2.3 GE Marathon Control Rod Assembly

The form, fit and function of the Marathon control rod design are equivalent to the original equipment control rods used at Susquehanna. Reference 4.2.6-12 documents NRC acceptance of the GE Marathon control rod mechanical design.

The control rod stresses, strains, and cumulative fatigue were evaluated by GE Nuclear and result in acceptable margins to safety. The control rod insertion capability was evaluated and found to be acceptable during all modes of plant operation within the limits of plant analyses. In addition, the coupling mechanism is fully compatible with the existing control rod drives in the plant. The materials used in the Marathon control rods were also evaluated and are compatible with the reactor environment. The Marathon control rods are approximately the same weight as the original equipment control rods and, therefore, there is no impact on the seismic and hydrodynamic load evaluation for the reactor vessel and internals. With lighter weight than the D160 control rods and envelope dimensions less than or the same as the original equipment, the Marathon design is compatible with existing NSSS hardware and there is no change in scram performance or drop time.

Neutronics evaluations of the Marathon control rods by GE Nuclear using the methodologies described in Reference 4.2.6-12 indicate the C lattice Marathon design for Susquehanna slightly exceeds the +5% beginning-of-life reactivity worth constraint relative to the original equipment all B_4C design. Therefore, the effect of the increased reactivity worth on plant analyses had to be considered. The increased reactivity worth was found to not adversely impact normal operation, and is considered in the analysis of abnormal operational occurrences, infrequent events, or accidents. The Marathon Ultra – HD control rod design satisfies the +5% beginning-of-life reactivity worth constraint, (Reference 4.2.6-16).

The Marathon control rods used improved materials and contain significant design improvements to eliminate cracking and the associated loss of Boron experienced by the original equipment. GE defines the end of life as a 10% reduction in cold reactivity worth in any ¼ axial segment relative to the initial undepleted state of the original equipment control rods. PPL will track the depletion of the Marathon control rods and discharge any control rod prior to reaching the defined end of life, or provide technical justification for its continued use.

4.2.3.2.4 Westinghouse CR 99 Control Rod Assembly

The form, fit and function of the Westinghouse CR 99 control rod design are equivalent to the OE control rods used at Susquehanna. Reference 4.2.6-17 documents NRC acceptance of the CR 99 control rod mechanical design.

The CR 99 control rod stresses, strains, and cumulative fatigue were evaluated by Westinghouse and result in acceptable margins to safety. The CR 99 insertion capability was evaluated and found to be acceptable during all modes of plant operation within the limits of plant analyses. The CR 99 coupling mechanism is fully compatible with the existing control rod drives in the plant. The materials used in the CR 99 control rods were also evaluated and are compatible with the reactor internals and the reactor environment. The CR 99 control rods are similar in nominal weight of the OE control rods and, therefore, there is no impact on the seismic and hydrodynamic load evaluation for the reactor vessel and internals. Scram speeds and settling times in the reactor are not adversely affected by the CR 99 control rods. The CR 99 velocity limiter design is identical to the design of the OE control rods and meets the assumption for the control rod drop accident.

The total worth of the CR 99 control rod is within ±5% of the OE control rod. There is no negative impact to shutdown margin and minimal impact on LPRM detector indications. The nuclear end of life criteria is maintained as 10% reactivity worth decrease relative to the OE control rod (Reference 4.2.6-17).

The CR 99 use of an improved high density absorber material, which is less sensitive to both powder densification and absorber swelling due to neutron absorption reactions, minimizes the possibility of absorber swelling causing contact with the surrounding stainless steel and contributing stress. The CR 99 use of AISI 316L stainless steel, with its better resistance to fast neutron irradiation assisted stress corrosion cracking (IASCC), also reduces the potential for control blade cracking. SSES will track the depletion of the CR 99 control rod and discharge any control rods prior to reaching the defined end of life, or provide technical justification for its continued use.

4.2.4 Testing and Inspection

4.2.4.1 Fuel Hardware and Assembly

Framatome – ANP, Inc (FANP) has developed Quality Control Standards for manufacturing, testing, and inspection of FANP components and fuel bundles. Details regarding FANP manufacturing, testing, and inspection processes are available in Reference 4.2.6-6.

On-site inspection of all new fuel bundles, fuel channels, and control rods is performed prior to installation into a Susquehanna Unit. These inspections are controlled by plant procedures. The procedures were developed based on guidelines provided by the fuel, channel, and control rod suppliers for receipt inspection and include acceptance criteria which are verified for each fuel bundle, fuel channel, or control rod.

The fuel channel management practices in place at Susquehanna are consistent with the recommendations contained in GE SIL 320, 'Recommendations for Mitigation of the Effects of Fuel Channel Bowing'. In addition, PPL will only use fuel channels for only one fuel bundle lifetime and will not reuse them. The fuel channel management practices are continuously reviewed against plant operation and industry practices.

4.2.4.2 Enrichment, Burnable Poison, and Absorber Rod Concentrations

FANP has established adequate measures, in accordance with Reference 4.2.6-6, to assure that nuclear materials of varying enrichment and form are positively identified and physically segregated as required to assure no inadvertent intermixing of enrichment forms. These measures include, as appropriate, identification of storage and processing containers, gamma scan verification of powder, nuclear rod assay, analytical examinations, in-process inspections, cleanouts of processing equipment between enrichments, administrative controls on the handling of materials, and audits of processing and product.

FANP fuel pellets are manufactured in accordance with approved procedures and are controlled by Product Design Specifications which define the allowable concentration tolerances and confidence levels required to verify enrichment and burnable poison concentrations.

General Electric (GE) supplied the original equipment control rods for both Susquehanna Units and is the supplier of the Duralife-160C and Marathon replacement control rods. The absorber materials, boron carbide and hafnium, are certified by GE to meet GE Material Specifications. The isotopic B^{10} content and boron content is verified for each powder lot received by General Electric. All boron carbide absorber rod assemblies are subjected to a leakage test to insure absorber rod integrity. GE performs analysis of hafnium absorber rod lots to insure chemical composition is in conformance with GE Material Specifications.

Westinghouse supplied the CR 99 control rods for use at both Susquehanna Units. The boron carbide absorber material is certified by Westinghouse to meet Westinghouse Material Specifications. The isotopic B^{10} content and boron content is verified for each powder lot received by Westinghouse. Each CR 99 control rod blade is leak tested with helium per the Westinghouse Materials Specifications.

4.2.4.3 Surveillance, Inspection, and Testing

PPL has a fuel reliability program that includes fuel performance monitoring and fuel failure response. On-line fuel performance monitoring is conducted to determine whether there is a fuel failure and may include evaluation of the general location of the failed assembly, the number of fuel assemblies suspected, when the failure occurred, and the approximate exposure of the failed assembly. Determination of this information prior to refueling allows preparation for changes in the following cycle's core design. In addition, control rod sequence exchanges and full power control rod patterns can be developed to minimize the offgas release from the failed rod(s) and stress on the suspect assembly during power maneuvering.

On-line fuel performance monitoring is performed at the Susquehanna station by periodic evaluation of pretreatment offgas activity and/or reactor coolant samples. Verification of failed fuel is made by periodic evaluation of the pretreatment xenon and krypton offgas activity and reactor water cleanup system iodine and cesium activity. The general location of the failed fuel assembly is, typically, identified by control rod motion testing and monitoring of the pretreatment offgas activity. Identification of the exact assembly may be performed by sipping or ultrasonic testing (UT) of the suspect assemblies.

Post-irradiation fuel failure evaluations are, typically, performed to determine the exact fuel rod location within the assembly and the root cause of a fuel rod failure. The exact location of the failed rod may be determined by UT or eddy current testing. Root cause evaluations may include review of manufacturing and inspection records, visual examination of the failure location, and destructive examination of the failed fuel rod.

Post-irradiation inspection programs have been developed by FANP to evaluate fuel design performance. Reference 4.2.6-11 discusses the FANP inspection and surveillance program for irradiated ATRIUM™-10 fuel.

4.2.5 Operating and Developmental Experience

FANP ATRIUM™-10 fuel has been utilized at SSES beginning with Unit 1 Cycle 11 and Unit 2 Cycle 9.

PPL continually tracks the performance of all fuel in the Susquehanna Units in an effort to identify indications of potential fuel rod failures.

Prior to the implementation of a mechanical fuel design into either Susquehanna Unit, that introduces features not currently in other operating plants, a plan will be developed to evaluate the performance of this fuel design in the Susquehanna Units. This plan may include pre and post irradiation fuel assembly characterization, visual inspection, power maneuvering evaluations, fuel clad corrosion evaluations, and UT inspections.

PPL occasionally participates in Lead Use Assembly programs. These programs allow the company to evaluate and gain operating experience with new fuel designs.

4.2.6 References

4.2.6-1 Deleted

4.2.6-2 Deleted

4.2.6-3 "BWR/4 and BWR/5 Fuel Design," NEDE-20944(P), General Electric Company, October 1976, and Letter from Olan D. Parr (NRC) to Dr. G. G. Sherwood (GE), "Review of General Electric Topical Report NEDE-20944-P, BWR/4 and BWR/5 Fuel Design (NEDO-20944 Non-Proprietary Version)", September 30, 1977.

4.2.6-4 "Safety Evaluation of the General Electric Hybrid I Control Rod Assembly for the BWR 4/5 C Lattice, "NEDE-22290-A, General Electric Company, September 1983, and Supplement 1, General Electric Company, July 1985.

4.2.6-5 "Safety Evaluation of the General Electric Duralife 230 Control Rod Assembly, " NEDE-22290-A, Supplement 3, General Electric Company, May 1988.

4.2.6-6 "Nuclear Fuel Business Group Quality Management Manual, "NFQM, Rev. 0, Framatome-ANP,U.S. Version, June 2002.

4.2.6-7 Deleted

4.2.6-8 "Mechanical Design for BWR Fuel Channels", EMF-93-177 (P) (A) Rev. 1, August 2005.

4.2.6-9 "Mechanical Design for BWR Fuel Channels," EMF-93-177 (P)(A) and Supplement 1, Siemens Power Corporation, August 1995.

4.2.6-10 "Generic Mechanical Design Criteria for BWR Fuel Designs," ANF-89-98(P)(A), Rev. 1 and Rev. 1 Supplement 1, Advanced Nuclear Fuels Corporation, May 1995.

4.2.6-11 "Mechanical Design Evaluation for Siemens Power Corporation ATRIUM™ –10-BWR Reload Fuel, " EMF-95-52(P), Rev. 2, Siemens Power Corporation – Nuclear Division, December 1998.

4.2.6-12 "GE Marathon Control Rod Assembly, " NEDE-31758P-A, GE Nuclear Energy, October 1991.

4.2.6-13 Deleted

4.2.6-14 Deleted

4.2.6-15 Deleted

4.2.6-16 NEDE - 33284 Supplement 1 P-A, Revision 1 March 2012, Licensing Topical Report, "Marathon Ultra Control Rod Assembly".

4.2.6-17 WCAP-16182-P-A, Revision 1 October 2009, "Westinghouse BWR Control Rod CR 99 Licensing Report – Update to Mechanical Design Limits".

TABLE 4.2-14

FUEL DESIGN CHARACTERISTICS (Nominal)

PARAMETER	ATRIUM^{TM}-10
Fuel Assembly	
Fuel rod array	10x10
Fuel Rods per Assembly	91
Overall length (in)	176.4
Number of Spacers	8
Fuel Channel Wall Thickness (mils)	80 or 100
Channel inside width (in)	5.278
Full Length Fuel Rod	
Number per assembly	83
Clad O.D. (in)	0.396
Cladding Material	Zircaloy-2
Part Length Fuel Rod	
Number per assembly	8
Cladding Material	Zircaloy-2
Fuel Pellet	
Material	UO_2
Density (% Theoretical) (Bundle Design Exposure 49 GWd/MTU)	95.4
Density (% Theoretical) (Bundle Design Exposure 54 GWd/MTU)	95.8
Diameter (in)	0.3413
Length (in)	0.413
Burnable Absorber	Gd_2O_3
Water Rod/Channel	
# of Water Rods/Channels	1 channel

Security-Related Information
Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
CORE CELL (ATRIUM^{TM}-10 FUEL WITH DURALIFE 160C CONTROL ROD) FIGURE 4.2-15

Security-Related Information
Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
CORE CELL (ATRIUM™-10 FUEL WITH MARATHON CONTROL ROD) FIGURE 4.2-15A

Security-Related Information
Figure Withheld Under 10 CFR 2.390

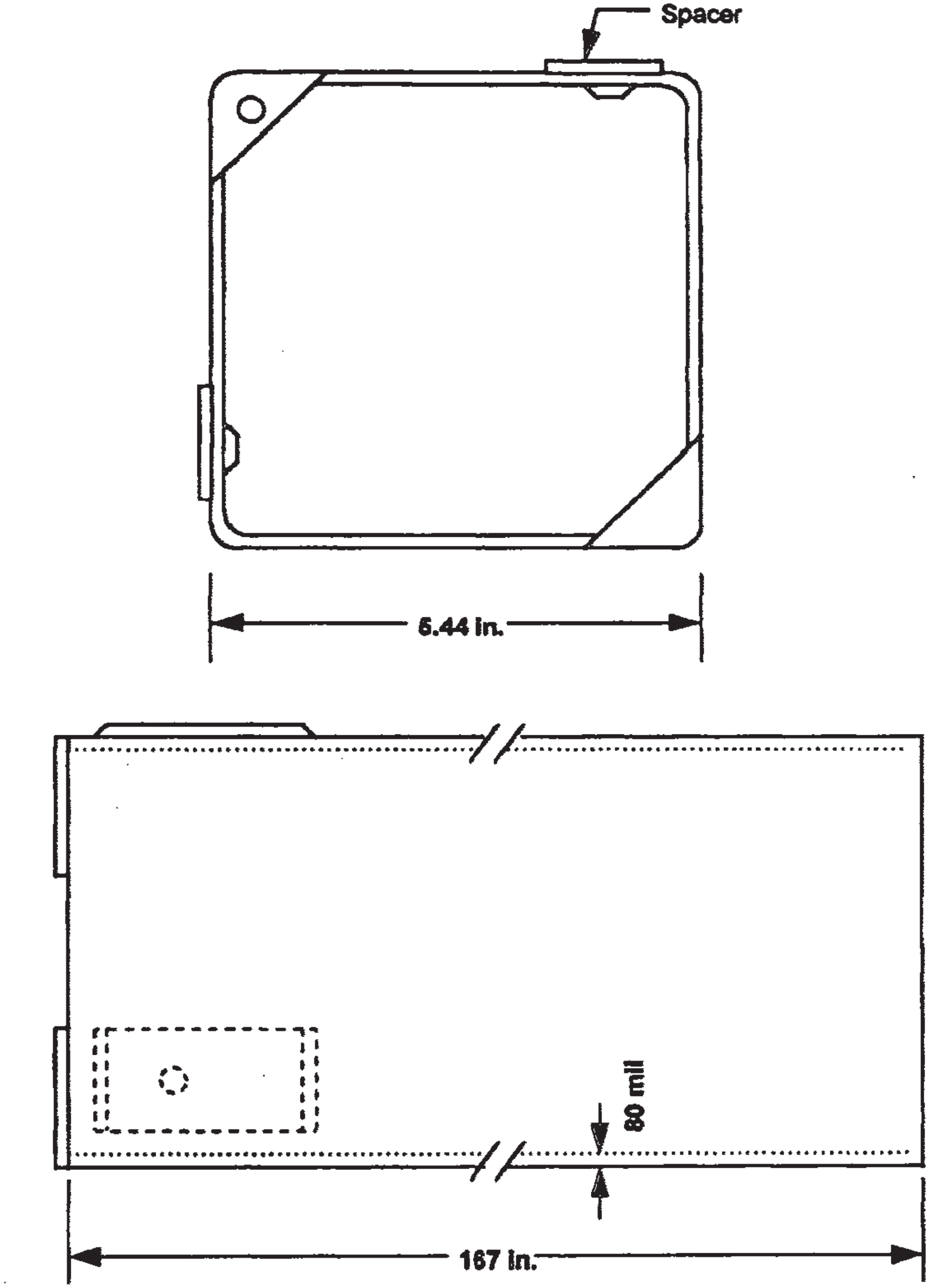
SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
CORE CELL (ATRIUM™-10 FUEL WITH ULTRA-HD CONTROL ROD) FIGURE 4.2-15B

Security-Related Information
Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
CORE CELL (ATRIUM™-10 FUEL WITH WESTINGHOUSE CR 99 CONTROL ROD) FIGURE 4.2-15C

Security-Related Information
Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
FUEL BUNDLE FANP ATRIUM^{TM}-10 FIGURE 4.2-17



FSAR REV. 65

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

80 MIL FUEL CHANNEL

FIGURE 4.2-18, Rev. 54

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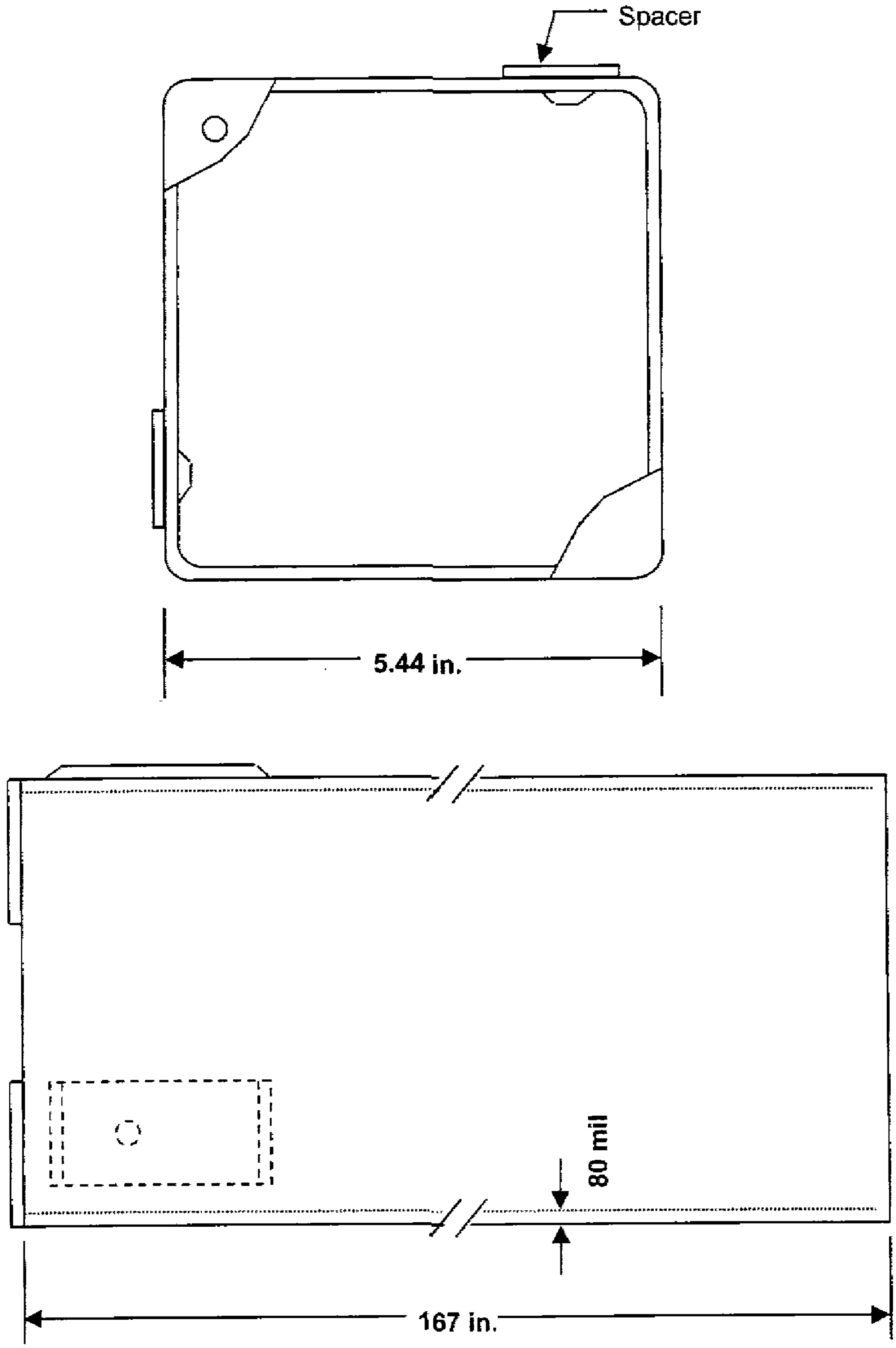

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 AND 2 FINAL SAFETY ANALYSIS REPORT
80 MIL FUEL CHANNEL FIGURE 4.2-18-1, Rev. 1

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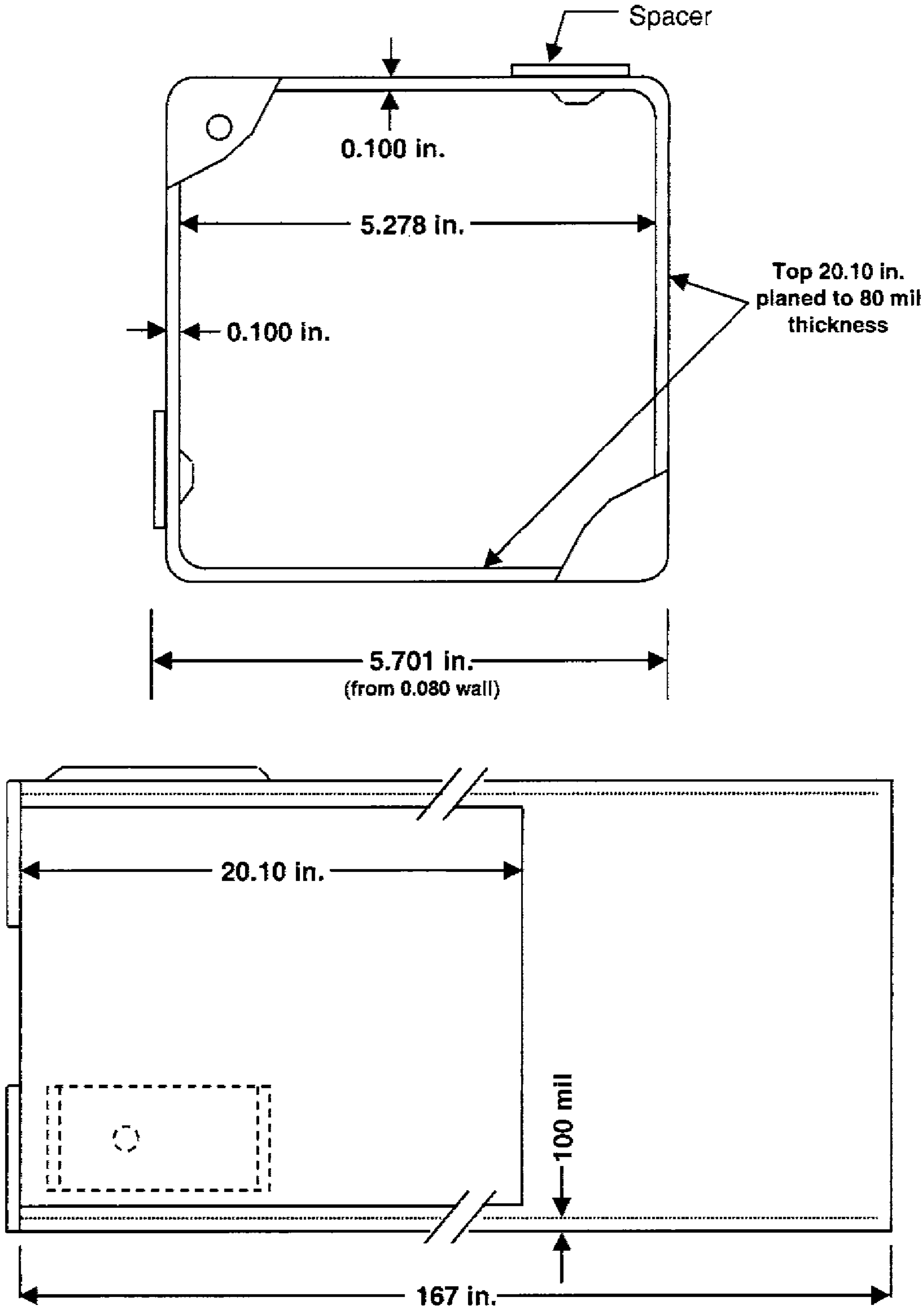

FSAR REV. 65

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

100 MIL FUEL CHANNEL

FIGURE 4.2-18-2, Rev. 1

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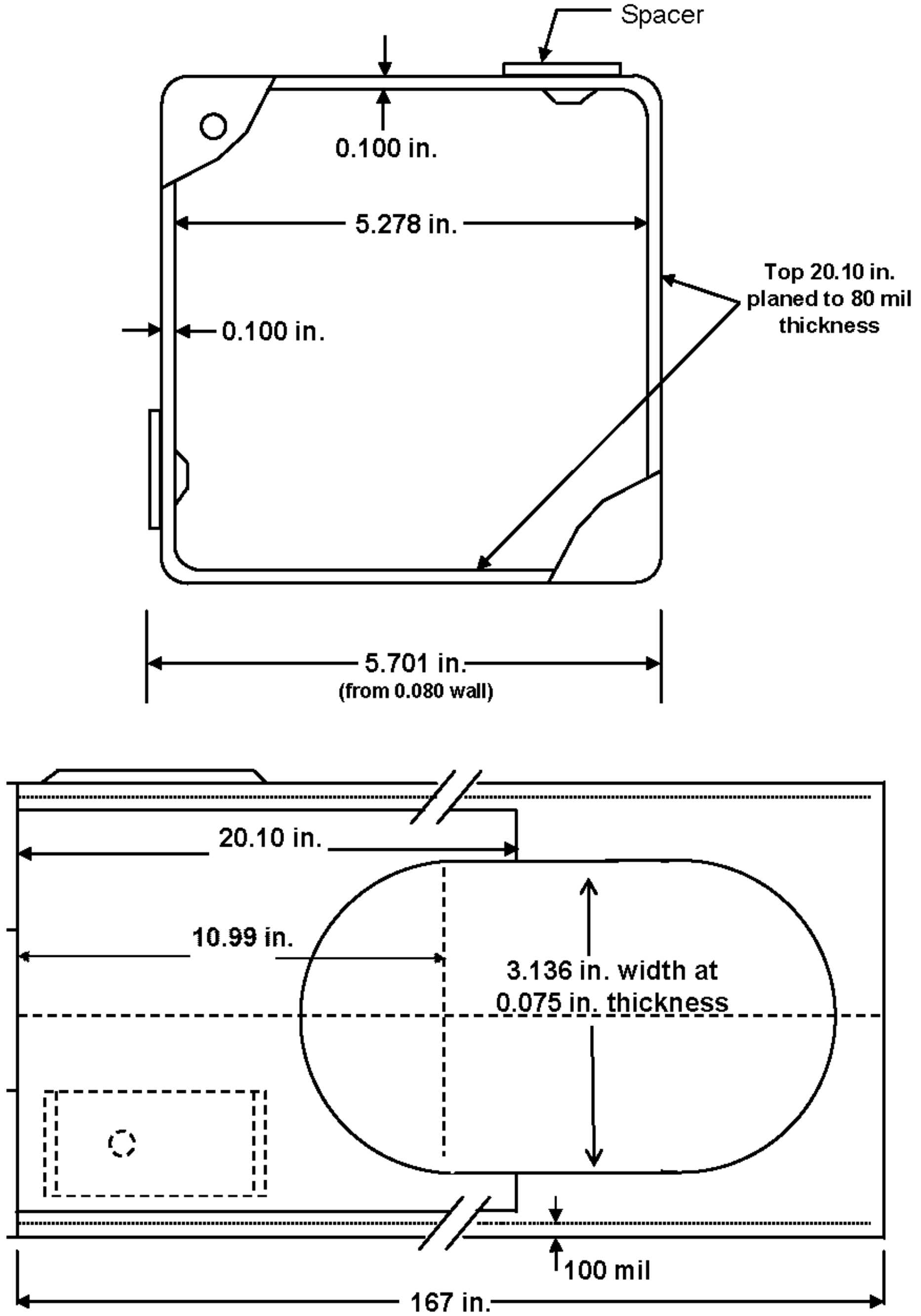

FSAR REV. 68

SUSQUEHANNA STEAM ELECTRIC STATION
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FINAL SAFETY ANALYSIS REPORT

ADVANCED FUEL CHANNEL

FIGURE 4.2-18-3, Rev. 0

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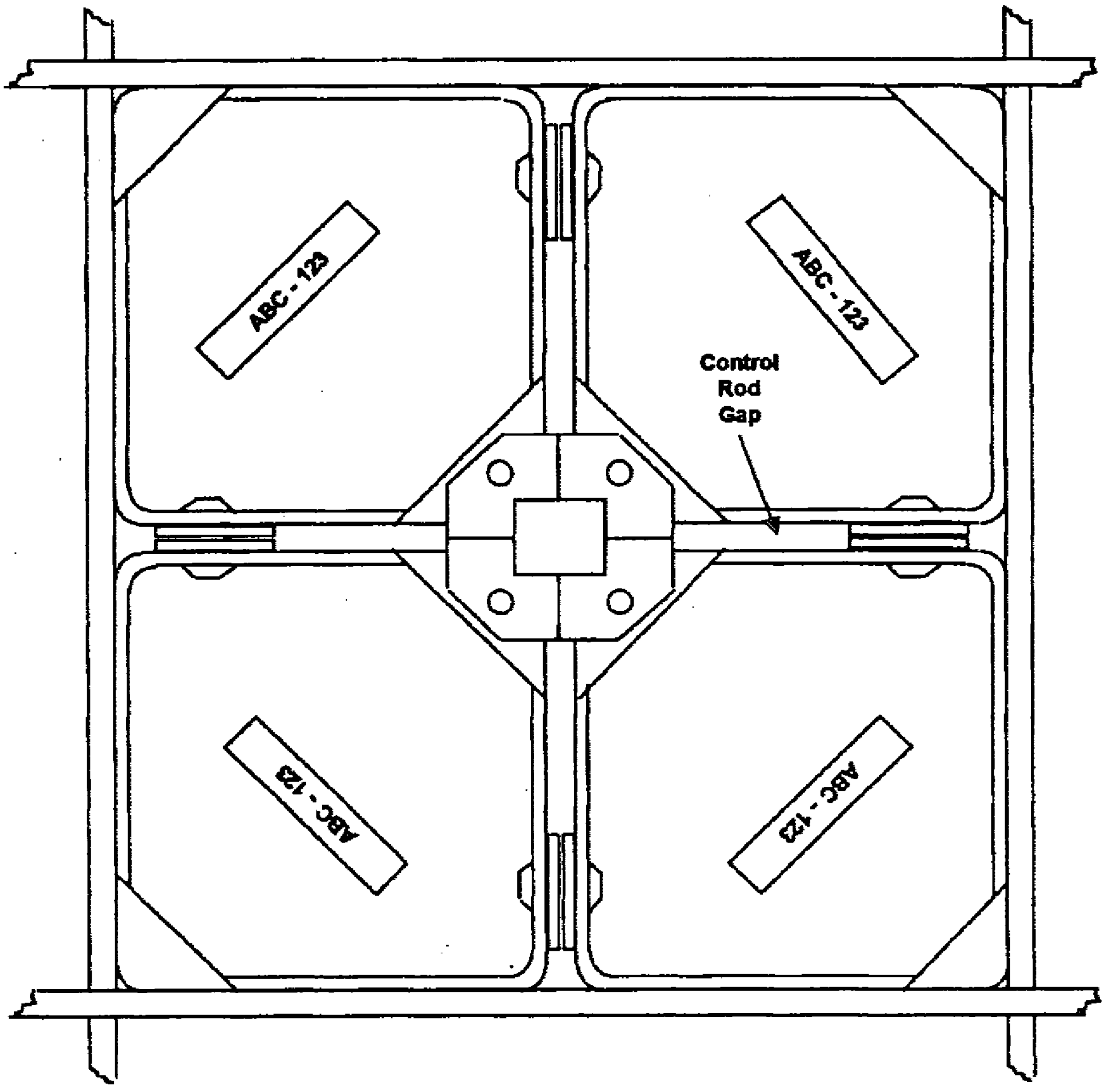

FSAR REV. 65

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

CORRECT FUEL ASSEMBLY
ORIENTATION

FIGURE 4.2-19, Rev. 54

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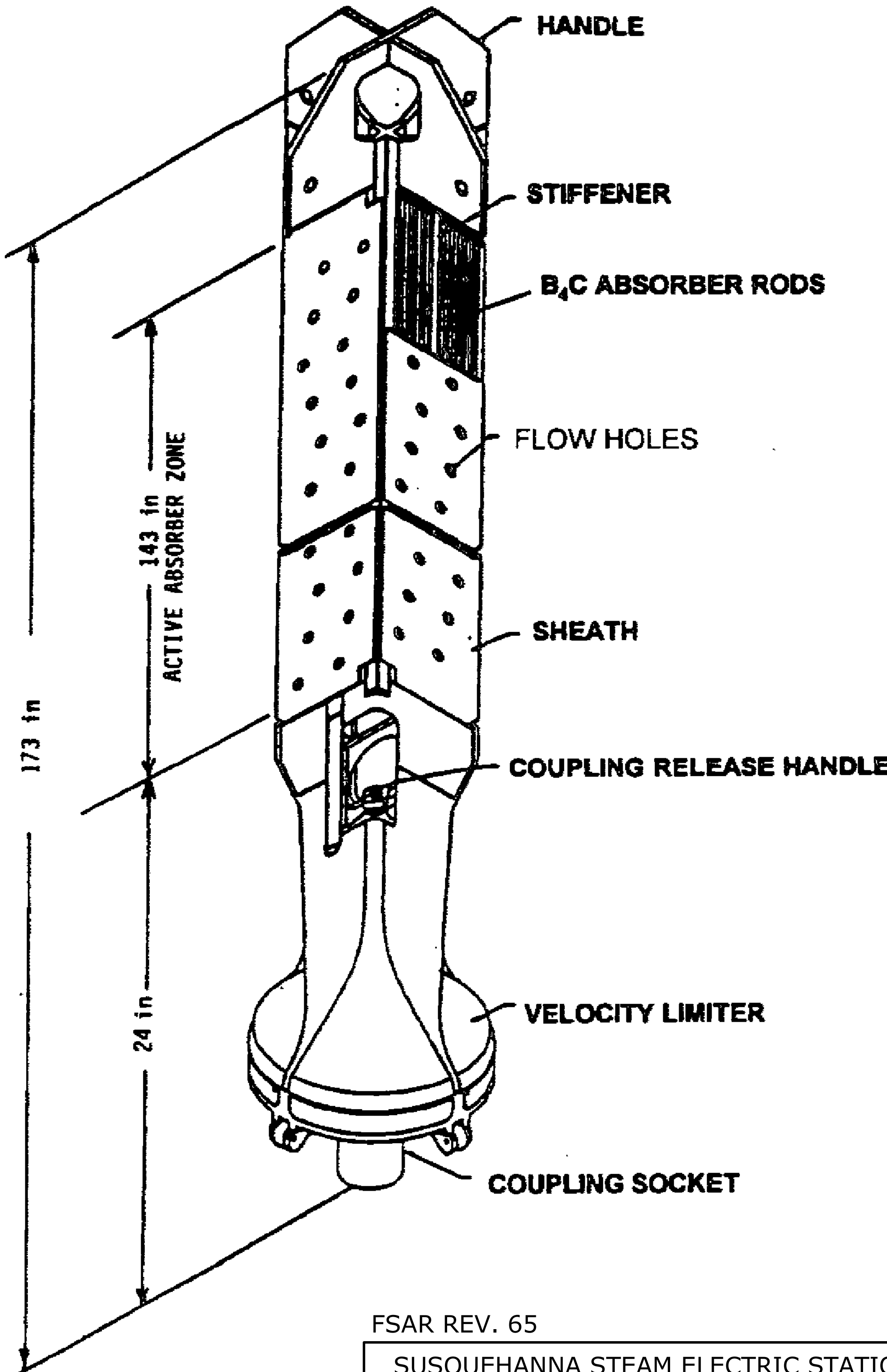

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

CONTROL ROD ASSEMBLY
ORIGINAL EQUIPMENT

FIGURE 4.2-20, Rev. 54

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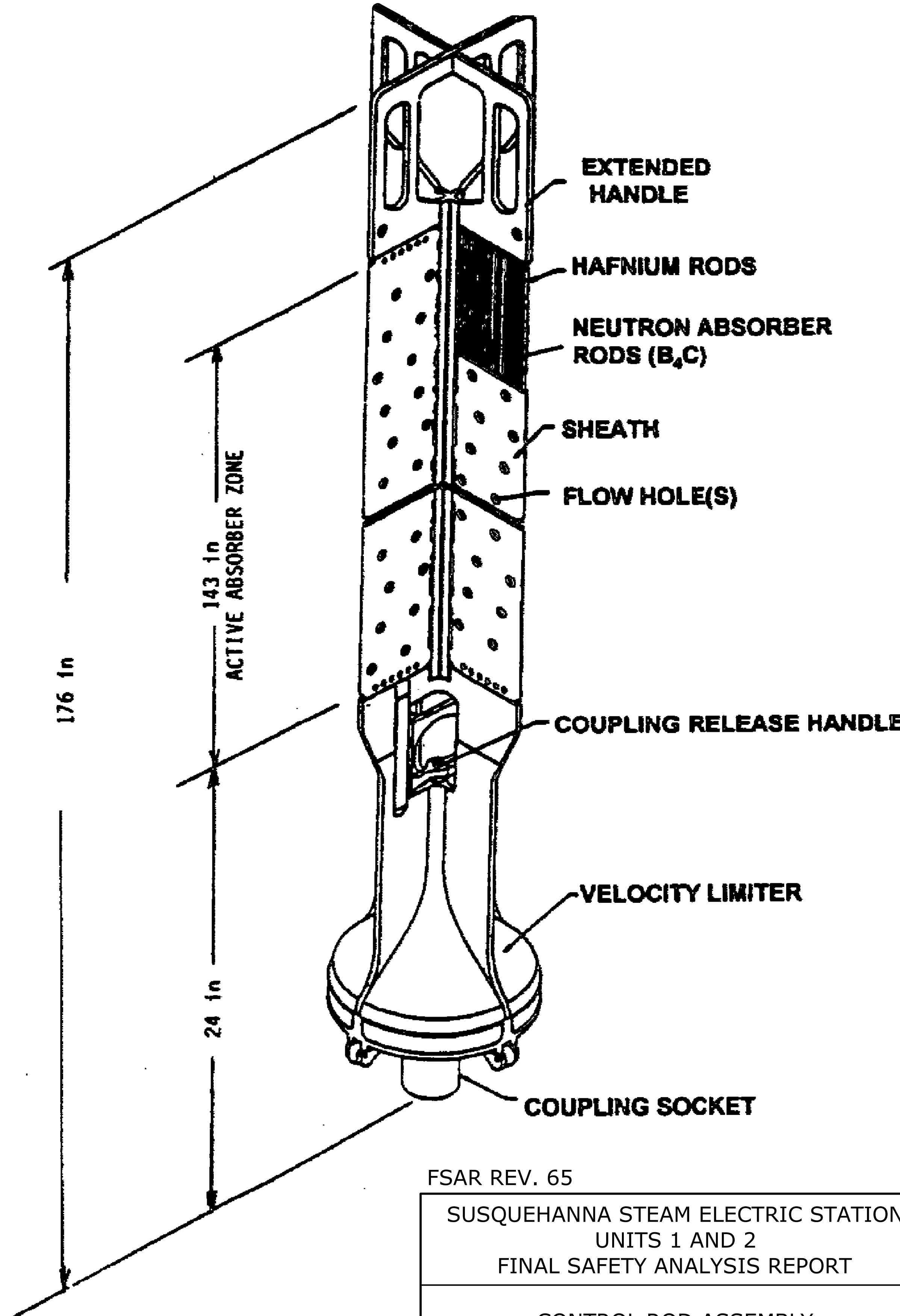

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 AND 2 FINAL SAFETY ANALYSIS REPORT
CONTROL ROD ASSEMBLY DURALIFE 160-C (D-160C) FIGURE 4.2-21, Rev. 54

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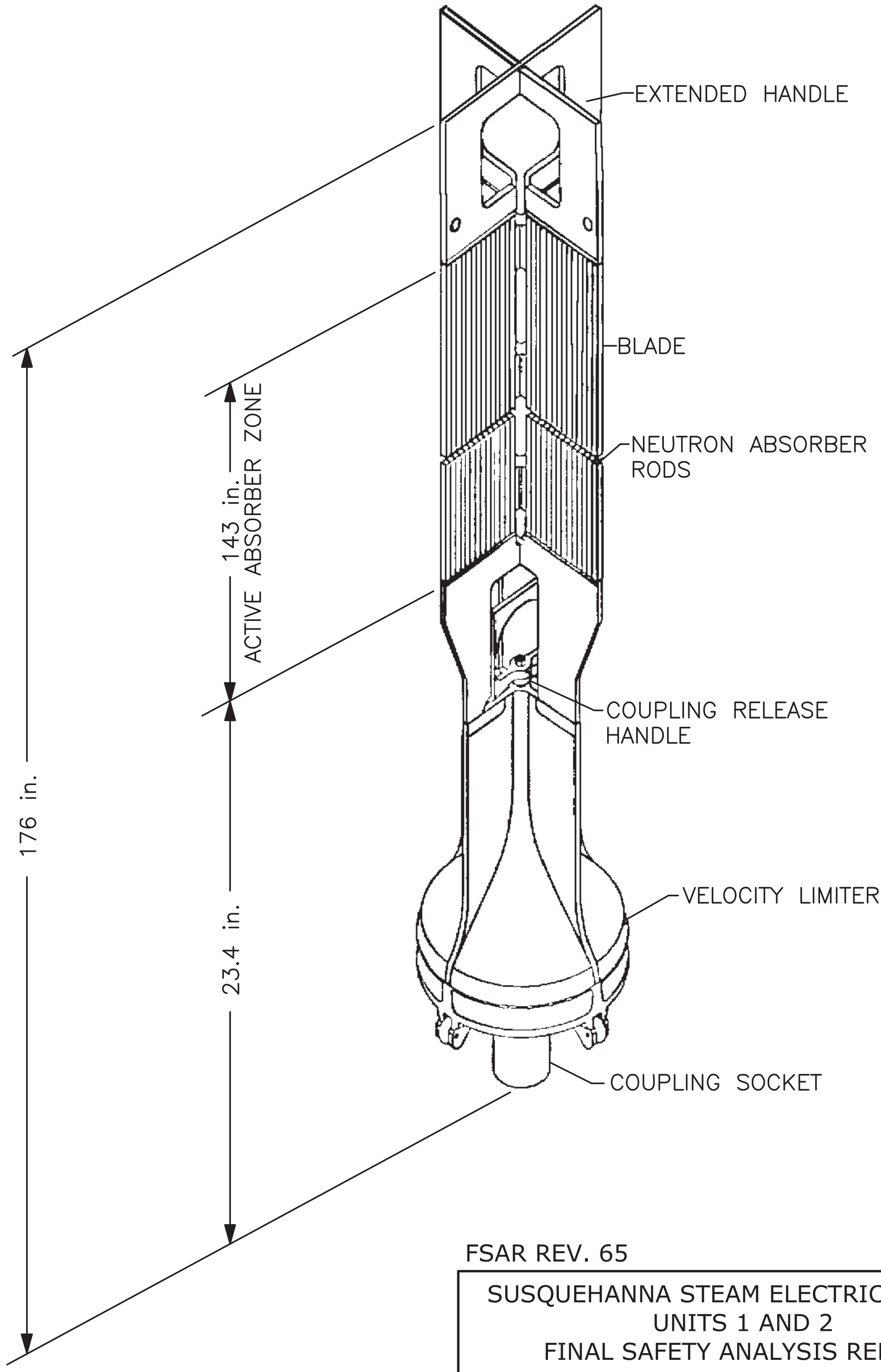

FSAR REV. 65

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

CONTROL ROD ASSEMBLY
MARATHON

FIGURE 4.2-22, Rev. 2

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Security-Related Information
Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
CONTROL ROD ASSEMBLY WESTINGHOUSE CR 99 FIGURE 4.2-23

4.3 NUCLEAR DESIGN

The nuclear design of the initial cores for Susquehanna is described in References 4.3-1, 4.3-2, and 4.3-3. This section incorporates much of the general nuclear design information in Reference 4.3-1 and presents detailed design information for reload cores.

4.3.1 Design Bases

Nuclear design bases fall into two categories: safety design bases and core performance design bases. Safety design bases are required by the General Design Criteria to ensure safe operation of the core. Core performance design bases are required to meet power production objectives.

4.3.1.1 Safety Design Bases

Safety design bases protect the nuclear fuel from damage which would result in a release of radioactivity, representing an undue risk to the health and safety of the public.

Safety design bases are listed below.

1) The core shall be capable of being rendered subcritical at any time or core condition with the highest worth control rod fully withdrawn.

2) The void coefficient shall be negative over the entire operating range.

3) Technical specification limits on Linear Heat Generation Rate (LHGR), Minimum Critical Power Ratio (MCPR), and the Average Planar Linear Heat Generation Rate (APLHGR), shall not be exceeded during steady state operation.

4) The nuclear characteristics of the design shall not exhibit any tendency toward divergent operation.

5) Reload fuel lattice enrichments shall be such that the nuclear design bases are met for the new fuel storage racks (section 9.1.1.1.1.2) and spent fuel storage (section 9.1.2.1.1.2).

4.3.1.2 Plant Performance Design Bases

1) The core design shall have adequate excess reactivity to reach the desired cycle length.

2) The core design shall be capable of operating without exceeding technical specification limits.

3) The core and fuel design and the reactivity control system shall allow continuous, stable regulation of reactivity.

4) The core and fuel design shall have adequate reactivity feedback to facilitate normal operation.

4.3.2 Description

A general description of BWR nuclear characteristics is provided in Reference 4.3-1. A summary of reactor core characteristics for Susquehanna is listed in Table 4.3-1.

4.3.2.1 Nuclear Design Description

The nuclear design of Susquehanna is both unit and cycle specific. A detailed description of the initial core nuclear design is available in Reference 4.3-1. Susquehanna Steam Electric Station Units 1 and 2 operate at power conditions in Table 4.3-1 with increased core flow. Fuel bundle and core reload designs have been developed using NRC approved methods.

4.3.2.1.1 Core Composition

The core contains 764 fuel assemblies arranged in a conventional scatter loaded pattern. Typically, the lowest reactivity fuel assemblies are placed in the peripheral region of the core. SSES uses ATRIUM™-10 fuel as the primary reload fuel mechanical design. In addition, a limited number of Lead Use Assemblies (LUAs) may be loaded to evaluate new fuel designs.

Detailed core compositions and associated core loading patterns are presented in References 4.3-17 and 4.3-18 for Units 1 and 2, respectively. The typical core loading patterns for both units are shown in Figures 4.3-1 and 4.3-2.

4.3.2.1.2 Fuel Bundle Nuclear Design

Reference 4.3-1 describes the first core bundle designs and related fuel nuclear properties. Reload fuel bundle design descriptions are presented below. The burnup dependent behavior of certain nuclear properties is primarily a function of enrichment and does not change significantly with bundle mechanical design. These characteristics include Uranium depletion and Plutonium buildup, fission fraction, delayed neutron fraction, and neutron lifetime. Figures 4.3-3 through 4.3-7 show the typical response of these characteristics with burnup for an enriched reload fuel bundle lattice.

Several ATRIUM™-10 bundle designs are in use at SSES. Each design may have a unique axial enrichment distribution, radial enrichment distribution, or burnable absorber loading. Figure 4.3-8-46 shows the nominal axial zoning for typical fuel bundles used in the reload cores. Figure series 4.3-9 shows the nominal radial enrichment distributions for typical lattice types used in the fuel bundles.

Unit/Cycle-specific reload fresh fuel bundle design descriptions may be found in References 4.3-19 and 4.3-20 for Unit 1 and Unit 2, respectively.

4.3.2.2 Power Distributions

This section presents typical power distributions for SSES reload cores. Typical local, core radial, and core axial power distributions for the initial core are described in Reference 4.3-1.

The core is designed such that the resulting power distributions meet the thermal limits identified in the plant Technical Specifications. The primary criteria for thermal limits are the Maximum Linear Heat Generation Rate (MLHGR) and the Minimum Critical Power Ratio (MCPR). In addition, a Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limit is applied to the plant.

Each of these parameters is a function of the core 3-D power distribution and the local rod-to-rod power distribution. Design calculations are performed to ensure that the core meets thermal limits and to demonstrate that the power distributions comply with the cycle design envelope.

The local peaking factor is defined as the ratio of the power density in the highest power rod in a lattice to the average power density in the lattice. Local effects on Critical Power Ratio are characterized by F-effective. Both the local peaking factor and F-effective have associated target values which typically satisfy the design envelope. Gross power peaking in the core is defined as the ratio of the maximum power density in any axial segment of any bundle in the core to the average core power density. Design allowances are included in the design stage to ensure that thermal limits are met. During plant operation, the power distributions are measured by the in-core instrumentation system and thermal margins are calculated by the core monitoring system.

4.3.2.2.1 Local Power Distribution

The local rod-to-rod power distribution and associated F-effective distribution are a direct function of the lattice enrichment distribution. Near the outside of the lattice where thermal flux peaks due to interbundle water gaps, low enrichment fuel rods are utilized to reduce power peaking. Closer to the center of the bundle, higher enrichment rods are used to increase power peaking and flatten the bundle power distribution. In addition, the water rods (or water channels) in the center of the lattice increase thermal flux and cause more power to be produced in the center of the lattice. The combination of enrichment and water channels results in a relatively flat power distribution.

To control bundle reactivity, Gd_2O_3 is utilized as a burnable absorber. Power is suppressed in gadolinia bearing fuel rods early in bundle life. As gadolinia is depleted, power in these rods initially increases, then decreases as fuel is depleted.
Local power distributions are calculated using licensed methodology described in Section 4.3.3.

Figure 4.3-11-1 shows bundle reactivity (k_∞) as a function of void fraction and burnup for an ATRIUM™-10 fuel assembly dominant lattice. At low exposure, reactivity is higher for lower void fractions. As exposure increases the curves cross, largely due to the effect of void history and the increase in plutonium buildup.

Figures 4.3-11-2 to 4.3-11-4 show typical unrodded local power distributions for an ATRIUM™-10 fuel assembly dominant lattice as a function of burnup with a constant void fraction. Figures 4.3-11-2, 4.3-11-5, and 4.3-11-6 show typical unrodded local power distributions for a fresh ATRIUM™-10 fuel dominant lattice as a function of void fraction at BOC. Figure 4.3-11-7 shows the typical response of the unrodded maximum local peaking factor as a function of void fraction and burnup.

4.3.2.2.2 Radial Power Distribution

The integrated bundle power, commonly referred to as the radial power, is the primary factor for determining MCPR. At rated conditions the MCPR is directly proportional to the radial power. The radial power distribution is a function of the control rod pattern in the core, the fuel bundle type and loading pattern, and void distribution. Radial power is calculated using the licensed methodology described in Section 4.3.3.

4.3.2.2.3 Axial Power Distribution

Axial power distributions in a BWR are a function of control rod position, steam voids, axial gadolinia distribution, and the exposure distribution. Voids tend to skew power toward the bottom of the core; bottom entry control rods reduce the power in the bottom of the core; and the axial gadolinia distribution assists in flattening the power in the bottom of the core. Since the void distribution is primarily determined by the power shape, the two means available for axial power shape optimization are the control rods and gadolinia. Typically, the core axial power shape is bottom peaked at BOC and becomes top peaked at EOC.

Axial power shapes are calculated using the licensed methodology described in Section 4.3.3.

4.3.2.2.4 Power Distribution Measurements

Power distribution measurement methodology and measurement uncertainties are described in References and 4.3-10 and 4.3.13.

4.3.2.2.5 Power Distribution Accuracy

The accuracy of calculated power distributions is discussed in References 4.3-10 and 4.3-13.

4.3.2.3 Reactivity Coefficients

Reactivity coefficients are differential changes in reactivity produced by differential changes in core conditions. These coefficients are useful in calculating the response of the core to varying plant conditions. The initial condition of the core and the postulated initiating event determine which of the coefficients are significant in evaluating core response.

The dynamic behavior of BWRs over all operating states can be characterized by three reactivity coefficients. These coefficients are the Doppler coefficient, the moderator temperature coefficient, and the void coefficient. The Power coefficient is also associated with a BWR; however, this coefficient is the combination of the Doppler and void coefficients in the operating range.

Reactivity coefficients are calculated using the licensed methods described in Section 4.3.3.

4.3.2.3.1 Void Coefficient

The most important reactivity coefficient in a BWR is the void coefficient. The void coefficient must be large enough to prevent power oscillation due to spatial xenon changes, but it must be small enough that pressurization transients do not limit plant operation. The void coefficient inherently flattens the radial power distribution during normal operation and provides enhanced reactor control through the void feedback mechanism. The overall void coefficient is always negative over the complete operating range since the BWR design is typically undermoderated. Void formation changes reactivity by reducing the amount of water available for neutron moderation, thus increasing neutron leakage. Typical values for the void coefficient are listed in Table 4.3-4.

4.3.2.3.2 Moderator Temperature Coefficient

The moderator temperature coefficient (MTC) is the least important of the reactivity coefficients since its effect is limited to a very small portion of the reactor operating range. Once the reactor reaches the power production range, boiling begins and the MTC remains essentially constant. Like the void coefficient, the moderator coefficient is associated with the amount of neutron moderation in the water. The MTC is negative during power operation; however, under cold conditions beginning soon after BOC, the MTC may become slightly positive.

The range of values of MTCs in reload lattices does not include any that are significant from a safety point of view. Typical values for the MTC are listed in Table 4.3-4. The small magnitude of this coefficient, relative to that associated with steam voids, combined with the long time-constant associated with heat transfer from fuel to coolant, makes the reactivity contribution of a change in moderator temperature insignificant during rapid transients.

4.3.2.3.3 Doppler Temperature Coefficient

The Doppler Temperature coefficient (DTC) is the change in reactivity due to a change in fuel temperature. This change in reactivity occurs due to the broadening of the fuel resonance absorption cross sections as temperature increases.

The DTC is primarily a measure of the Doppler broadening of U238 and Pu240 resonance absorption peaks. An increase in fuel temperature increases the effective resonance absorption cross section of the fuel and produces a corresponding reduction in reactivity. The Doppler coefficient changes as a function of core life representing the combined effects of fuel temperature reduction with burnup and the buildup of Pu240. Typical values for the Doppler coefficient are listed in Table 4.3-4.

4.3.2.3.4 Power Coefficient

The power coefficient is determined from the composite of all the significant individual sources of reactivity change associated with a differential change in reactor power. This coefficient assumes constant xenon. Typical values for the power coefficient may be obtained from Reference 4.3-1 for the initial cores.

4.3.2.4 Control Requirements

The core and fuel design in conjunction with the reactivity control system provide a stable system for BWRs. The control rod system is designed to provide adequate control of the maximum excess reactivity anticipated during the equilibrium fuel cycle operation. Since fuel reactivity is a maximum and control rod worth is a minimum at ambient temperature, the shutdown capability is evaluated assuming a cold, xenon free core. The safety design basis requires that the core, in its maximum reactivity condition, shall be subcritical with all control rods inserted except with the highest worth rod completely withdrawn. This limit allows control rod testing at any time in core life and assures that the reactor can be made subcritical by control rods alone.

The typical behavior of hot excess reactivity as a function of cycle exposure for SSES Units 1 and 2 is shown in Figure 4.3-12.

4.3.2.4.1 Shutdown Reactivity

Core Shutdown Margin calculations are performed to assess whether the basic criterion for reactivity control is met by the reload design. This criterion requires that the core, under cold, no xenon conditions, must be subcritical with the highest worth control rod fully withdrawn and all other rods fully inserted. SSES Technical Requirements Manual requires a shutdown margin of at least 0.38% $\Delta k/k$. The shutdown margin requirement is based on the uncertainties associated with the statistical variance of cold criticality calculations at a given exposure, plus a manufacturing uncertainty. The manufacturing uncertainty results from the fact that the calculated highest worth control rod may not be the highest worth rod in reality due to the stackup of manufacturing tolerances in a control cell.

Core Shutdown margin is very dependent on bundle and core designs and is a function of core exposure. Gadolinia loading, enrichment loading, and core loading all significantly affect core and local cell reactivity as a function of exposure. As a result, shutdown margin must be evaluated throughout the expected cycle operation to assure adequate margin to Technical Specification requirements. For design purposes, an additional uncertainty is added to the Technical Specification value to account for prediction uncertainties.

Shutdown margin is calculated as a function of cycle exposure in the following manner:

$$SDM(E) = \frac{1-\left(k_{eff}(E) - bias(E)\right)}{k_{eff}(E) - bias(E)} * 100\%$$

where;

SDM(E) = core shutdown margin (%Δk/k) at cycle exposure E,

$K_{eff}(E)$ = core k-effective at cycle exposure (E) with all rods in except the strongest worth rod (68°F with no xenon),

Bias(E) = core k-effective bias for cold core simulation model at cycle exposure (E). The bias equals the target cold core simulation model critical k-effective minus 1.0.

The Cycle R value is determined from the evaluation of shutdown margin as a function of cycle exposure. The R value is used to determine shutdown margin testing requirements, and it is defined as the difference between the calculated beginning of cycle shutdown margin minus the calculated minimum shutdown margin during the cycle, where shutdown margin is a positive number. The value of R must be either positive or zero and must be determined for each fuel loading cycle.

Typical behavior of shutdown margin as a function of cycle exposure for SSES Units 1 and 2 is shown in Figure 4.3-13.

A description of the methods used to calculate shutdown margin is provided in Section 4.3.3.

4.3.2.4.2 Reactivity Variations

Reference 4.3-1 provides a general discussion of reactivity variations in a BWR/4. The reference provides tables showing typical k-effective values for various power levels, control fractions, and Xenon concentrations. From this data, the general reactivity effect of changing a single core variable can be determined.

4.3.2.5 Control Rod Patterns and Reactivity Worths

4.3.2.5.1 RWM Range

Below the low power setpoint, control rod patterns follow prescribed withdrawal and insertion sequences restricted by the Rod Worth Minimizer (RWM). The sequences are established to assure that the maximum insequence control rod or rod segment reactivity worth would not be sufficient to result in a deposited fuel enthalpy greater than 280 cal/gm in the event of a control rod drop accident. The specified sequences are characterized by homogeneous, scattered patterns of control rod withdrawal or insertion. Further discussion of the RWM and control rod sequence limitations is provided in Section 15.4.9 (Control Rod Drop Accident).

4.3.2.5.2 Operating Range

In the power range, above the low power setpoint, there is no possible rod worth which, if dropped at the design rate of the velocity limiter, could result in a peak fuel enthalpy of 280 cal/gm. Therefore, restrictions on control rod patterns are not required to minimize control rod worths. During power operation the control rod patterns are selected based on the measured core power distributions.

For reload design purposes, optimized control rod patterns are selected for the cycle depletion. The series of design control rod patterns form the Cycle Step Out. Control rod sequence identification (A2, B2, A1, B1) is defined in Reference 4.3-1.

4.3.2.5.3 SCRAM Reactivity

The reactor protection system (RPS) is capable of shutting down the reactor by initiating a SCRAM. The RPS and the control rod drive (CRD) system act quickly enough to prevent the initiating event from driving the fuel beyond transient limits.

During a SCRAM from operating conditions, the control rod worth, reactor power, delayed neutron fraction, and void distributions must be properly accounted for as a function of time. The methodology used to account for these variables and determine SCRAM reactivity is described in Section 4.3.3.

4.3.2.6 Criticality of Reactor During Refueling

Criticality of fuel assemblies in the core during refueling is avoided by assuring that the Technical Specification shutdown margin requirement is met. For core shuffles, a shutdown margin design criterion is defined to account for prediction uncertainties. This criterion helps determine the acceptability of a fuel move for meeting the Technical Specification limit. A description of the methods used to calculate shutdown margin is provided in Section 4.3.3.

4.3.2.7 Stability

Boiling Water Reactors do not have instability problems due to Xenon. Xenon transients are highly damped in a BWR due to the large negative power coefficient. References 4.3-1 and 4.3-3 provide additional discussion of Xenon instability.

Thermal hydraulic stability is discussed in detail in Section 4.4.

4.3.2.8 Vessel Irradiations

The RAMA Fluence Methodology (Reference 4.3-14) is used to evaluate the Reactor Pressure Vessel (RPV) fluence for both units. This methodology has been reviewed and approved by the NRC for RPV fluence evaluations (Reference 4.3-15) and is consistent with applicable regulatory guidance (Reference 4.3-16). Detailed descriptions of the calculations for each unit are provided in References 4.3-7 and 4.3-8. The fast fluence evaluations are based on the RAMA Code Methodology. The Methodology includes a transport code, model builder codes, a fluence calculator code, an uncertainty methodology, and a nuclear data library. The transport code, fluence calculator, and nuclear data library are the primary software components for calculating the neutron flux and fluence. The transport code uses a deterministic, three-dimensional, multigroup nuclear particle transport theory to perform the neutron flux calculations. The transport code couples the nuclear transport method with a general geometry modeling capability to provide a flexible and accurate tool for calculating fluxes in light water reactors. The fluence calculator uses reactor operating history information with isotopic production and decay data to estimate activation and fluence in the reactor components over the operating life of the reactor. The nuclear data library contains nuclear cross-section data and response functions that are needed in the flux, fluence, and reaction rate calculations. The cross sections and response functions are based on the BUGLE-96 nuclear data library. Fluxes are calculated at the inner vessel surface, at ¼ T and ¾ T depths.

The RAMA methodology calculates RPV fluence and uncertainty at all locations in the RPV in the active core region in accordance with applicable regulatory guidance. The results from the vessel fluence coupon analyses are solely used to support the methodology uncertainty analysis. The RAMA methodology directly calculates the fluence at all RPV locations in the active core region. Therefore, lead factors, which were historically used to extrapolate the measured fluence at the coupon locations to the RPV ¼ T depths are no longer used or calculated.

Previous fluence calculations were performed using the DORT computer code, which is described in Section 4.1. The RAMA Fluence Methodology will continue to be used to calculate the fluence for both units and is described in BWRVIP-114 (Reference 4.3-14).

The analytical model for (R,θ) geometry is shown in Figure 4.3-14. The model consists of an inner and outer core region, the shroud, water regions inside and outside the shroud, jet pump components, the vessel wall, inner and outer Cavity, Mirror Insulation and the Biological Shield.

Neutron fluence was determined based on actual and expected operating history for each unit. This included the effects of several power uprates that have occurred during the operating history. Final end of life RPV fluence is calculated for both units at 32 EFPY at the RPV [both inner diameter (ID) and ¼ T (1/4 of the distance from the inside diameter to the outside diameter)] based on actual and expected operating history. Details on the power history assumed in the fluence analysis are provided in footnotes to the data in Table 4.3-5. Table 4.3-5 lists the 32 EFPY maximum fast fluence results and also provides historical results from the original analyses for comparison.

4.3.3 Analytical Methods

Reload design for SSES Units 1 and 2 is performed using NRC approved methodology. The approved methods used for nuclear design are fully described in Reference 4.3-13.

A summary description of several nuclear design codes is provided in Section 4.1.

Reference 4.3-1 describes the methods used for initial core nuclear design.

4.3.4 Changes

Reference 4.3-1 lists several changes made to the initial reactor nuclear design.

Reload core nuclear designs incorporate the following significant changes.

Unit 2, Cycle 9 and Unit 1, Cycle 11 were the first cores to utilize the ATRIUM™-10 fuel design at SSES. ATRIUM™-10 has a 10x10 lattice which is significantly different from the 9x9 lattice utilized in previous cycles. Nuclear characteristics of ATRIUM™-10 fuel are discussed in Section 4.3. Mechanical design of ATRIUM™-10 fuel is discussed in Section 4.2.

The CASMO-3G lattice physics code was first used to support the U1C10 reload design.

Unit 2, Cycle 9 was designed for a 24 month cycle. This cycle length represents a change from the 18 month cycle used for previous core designs. The effects of a 24 month cycle on the U2C9 reload were evaluated in Reference 4.3-11.

Unit 1, Cycle 11 was designed for a 24 month cycle. This cycle length represents a change from the 18 month cycle used for previous core designs. The effects of a 24 month cycle on the U1C11 reload were evaluated in Reference 4.3-12.

The CASMO-4/MICROBURN-B2 code system was first used to support the U1C14 reload design. A summary description of CASMO-4 and MICROBURN-B2 is provided in Section 4.1.

Unit 1 Cycle 14 was the first cycle to utilize 100 mil fuel channels and the Framatome-ANP FUELGUARD Lower Tie Plate design. The 100 mil fuel channel and FUELGUARD Lower Tie Plate are described in Section 4.2.

Unit 1 Cycle 20 was the first cycle to utilize the Advanced Fuel Channel (AFC). The AFC is described in Section 4.2.

4.3.5 References

4.3-1 "BWR/4 and BWR/5 Fuel Design", NEDE-20944(P), General Electric Company, October 1976.

4.3-2 "BWR/4 and BWR/5 Fuel Design", Amendment 1 NEDE-20944-1(P), General Electric Company, January 1977.

4.3-3 Letter from Olan. D. Parr (NRC) to Dr. G. G. Sherwood (GE), "Review of General Electric Topical Report NEDE-20944-P, BWR/4 and BWR/5 Fuel Design (NEDO-20944 Non-Proprietary Version)", September 30, 1977.

4.3-4 Deleted

4.3-5 Deleted

4.3-6 Deleted

4.3-7 "Susquehanna Unit 1 Reactor Pressure Vessel Fluence Evaluation, "PPL-FLU-002-R-002, Rev. 1, TransWare Enterprises, Inc., October 2005.

4.3-8 "Susquehanna Unit 2 Reactor Pressure Vessel Fluence Evaluation,"PPL-FLU-002-R-001, Rev. 0, TransWare Enterprises, Inc., May 2005.

4.3-9 "Power Uprate Engineering Report for Susquehanna Steam Electric Station Units 1 and 2", NEDC-32161P, GE Nuclear Energy, December 1993.

4.3-10 "Advanced Nuclear Fuels Methodology For Boiling Water Reactors", XN-NF-80-19 (P)(A), Volume 1, Supplement 3, Supplement 3 Appendix F, and Supplement 4, Advanced Nuclear Fuels Corporation, November 1990.

4.3-11 "Susquehanna SES Unit 2 Cycle 9 Reload Summary Report", PL-NF-97-003, Rev. 1, PP&L, September 1997.

4.3-12 "Susquehanna SES Unit 1 Cycle 11 Reload Summary Report", PL-NF-98-002, Rev. 1, PP&L, Inc., July 1998.

4.3-13 EMF-2158 (P) (A), Siemens Power Corporation Methodology For Boiling Water Reactors Evaluation and Validation of Casmo-4/Microburn-B2", October 1999.

4.3-14 "BWR vessel and Internals Project RAMA Fluence Methodology Manual," BWRVIP-114, May 2003.

4.3-15 Safety Evaluation of proprietary EPRI Reports, "BWR Vessel And Internals Project, RAMA Fluence Methodology Manual (BWRVIP-114)," "RAMA Fluence Methodology Benchmark Manual – Evaluation of Regulatory Guide 1.190 Benchmark Problems (BWRVIP-115)," "RAMA Fluence Methodology - Susquehanna Unit 2 Surveillance Capsule Fluence Evaluation for Cycles 1-5 (BWRVIP-117)," and "RAMA Fluence Methodology Procedures Manual (BWRVIP-121)," and "Hope Creek Flux Wire Dosimeter Activation Evaluation for

Cycle 1 (TWE-PSE-001-R-001)" (TAC No. MB9765), BWRVIP 2005-208B, William H. Bateman, NRC to Bill Eaton, BWRVIP Chairman, May 13, 2005.

4.3-16 "Calculation and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," Regulatory Guide 1.190, March 2001.

4.3-17 EC-FUEL-1857, Rev. 0, "Unit 1 Cycle 21 Core Design" (General Reference per NEI 98-03)

4.3-18 EC-FUEL-1875, Rev. 3, "Unit 2 Cycle 20 Core Design" (General Reference per NEI 98-03)

4.3-19 EC-FUEL-1856, Rev. 0, "Unit 1 Cycle 21 Fresh Fuel Design and Cross Section Generation" (General Reference per NEI 98-03)

4.3-20 EC-FUEL-1874, Rev. 0, "Unit 2 Cycle 20 Fresh Fuel Design and Cross Section Generation" (General Reference per NEI 98-03)

TABLE 4.3-1

REACTOR CORE CHARACTERISTICS

Reactor Type/Configuration	BWR-4/2 Loop Jet Pump Recirculation System, C-Lattice
Rated Thermal Power, Unit 1	3952 MWt
Rated Thermal Power, Unit 2	3952 Mwt
Number of Fuel Assemblies	764
Number of Control Rods	185
Number of Traversing In-core Probe Locations	43
Active Core Height, ft	12.45
Control Rod Pitch, inches	12.0
Fuel Assembly Pitch, inches	6.0

Table 4.3-4
TYPICAL CORE REACTIVITY COEFFICIENTS

Security-Related Information
Table Withheld Under 10 CFR 2.390

Table 4.3-5
FAST NEUTRON FLUENCES > 1 Mev
(32 EFPY Fluence)[1]

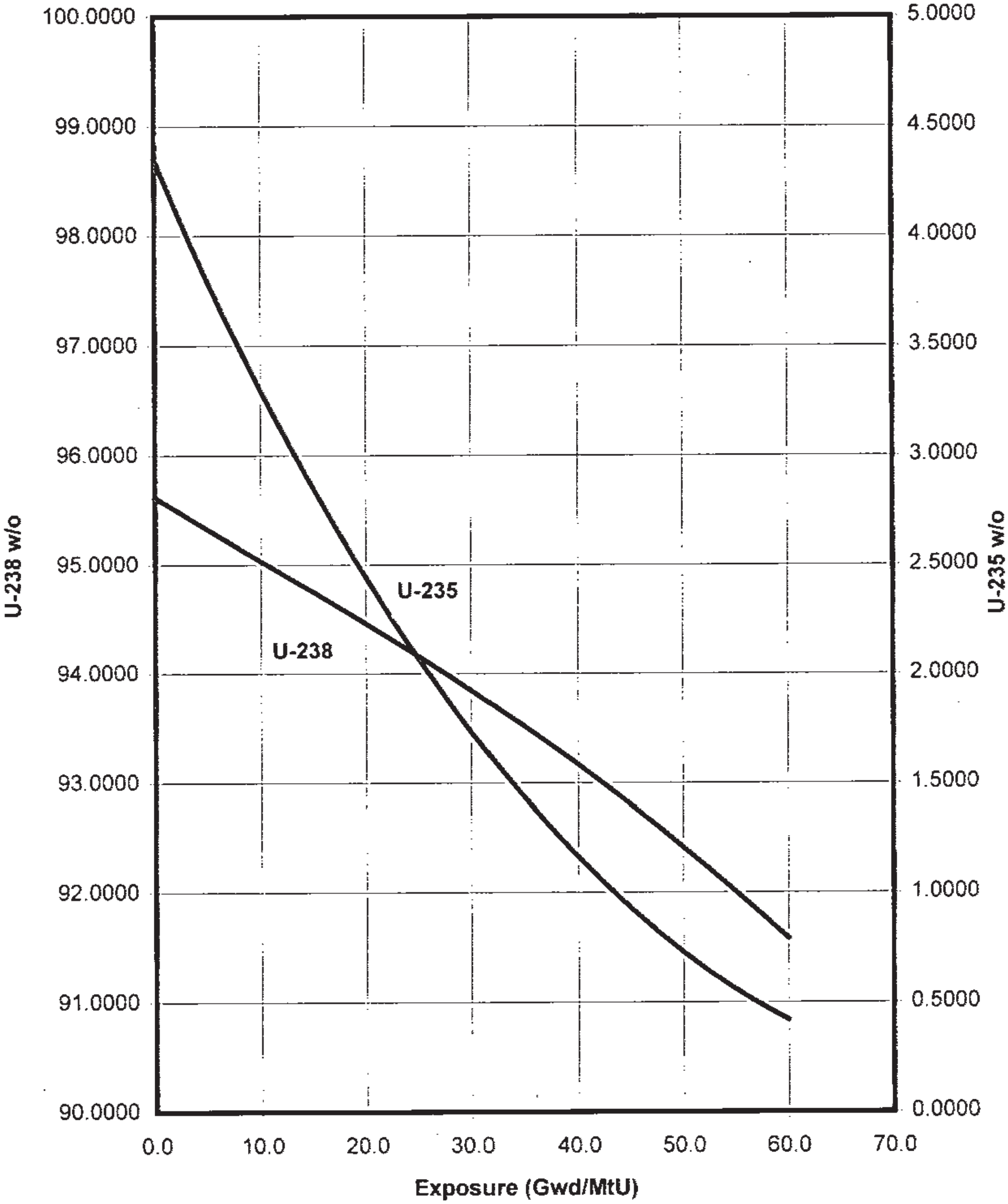
Security-Related Information
Table Withheld Under 10 CFR 2.390

Security-Related Information
Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
CORE LOADING MAP TYPICAL OF UNIT 1
FIGURE 4.3-1

Security-Related Information
Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
CORE LOADING MAP TYPICAL OF UNIT 2 FIGURE 4.3-2



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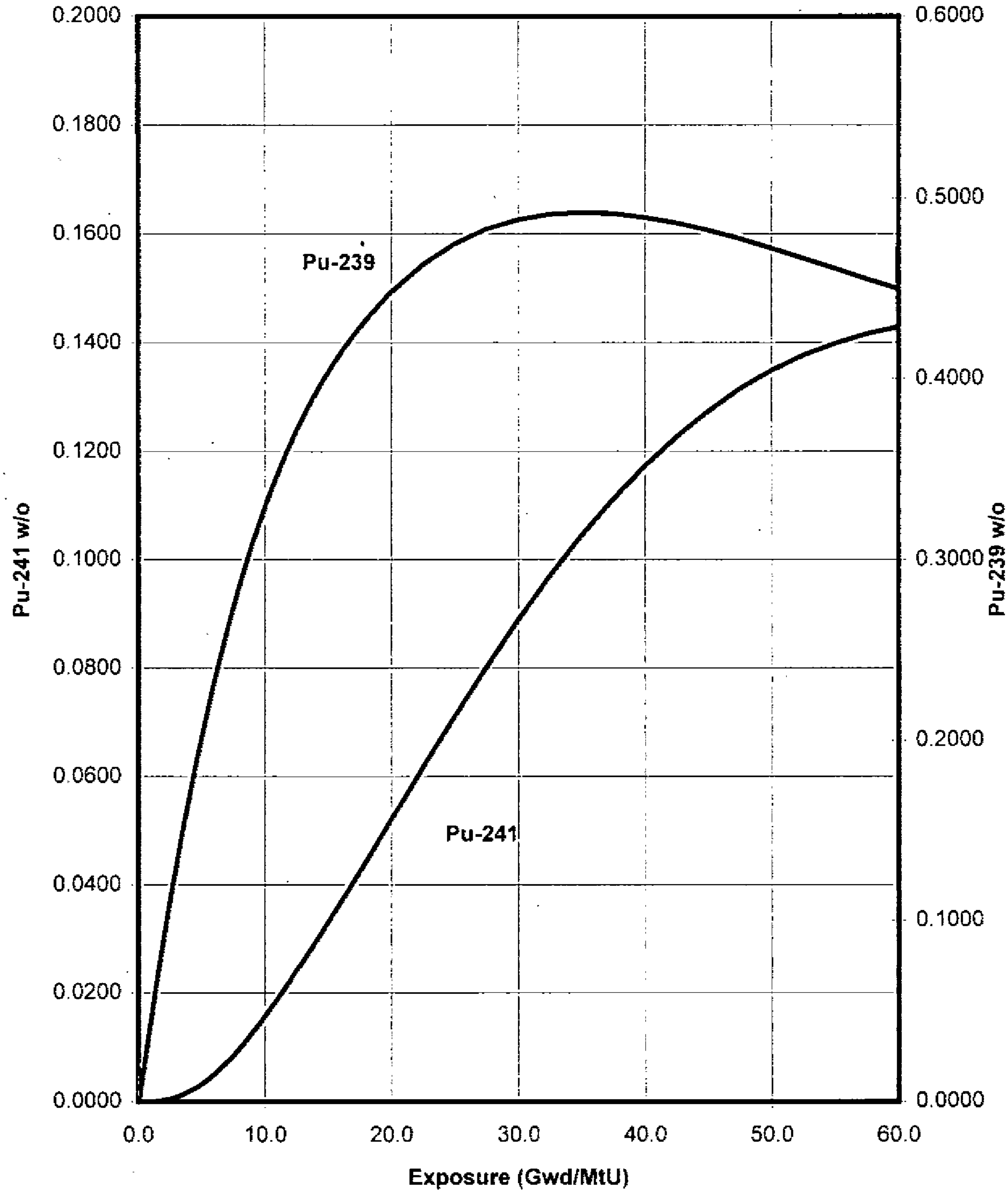
SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 AND 2 FINAL SAFETY ANALYSIS REPORT
URANIUM DEPLETION AS A FUNCTION OF EXPOSURE, 40% VOIDS (TYPICAL) FIGURE 4.3-3, Rev. 54

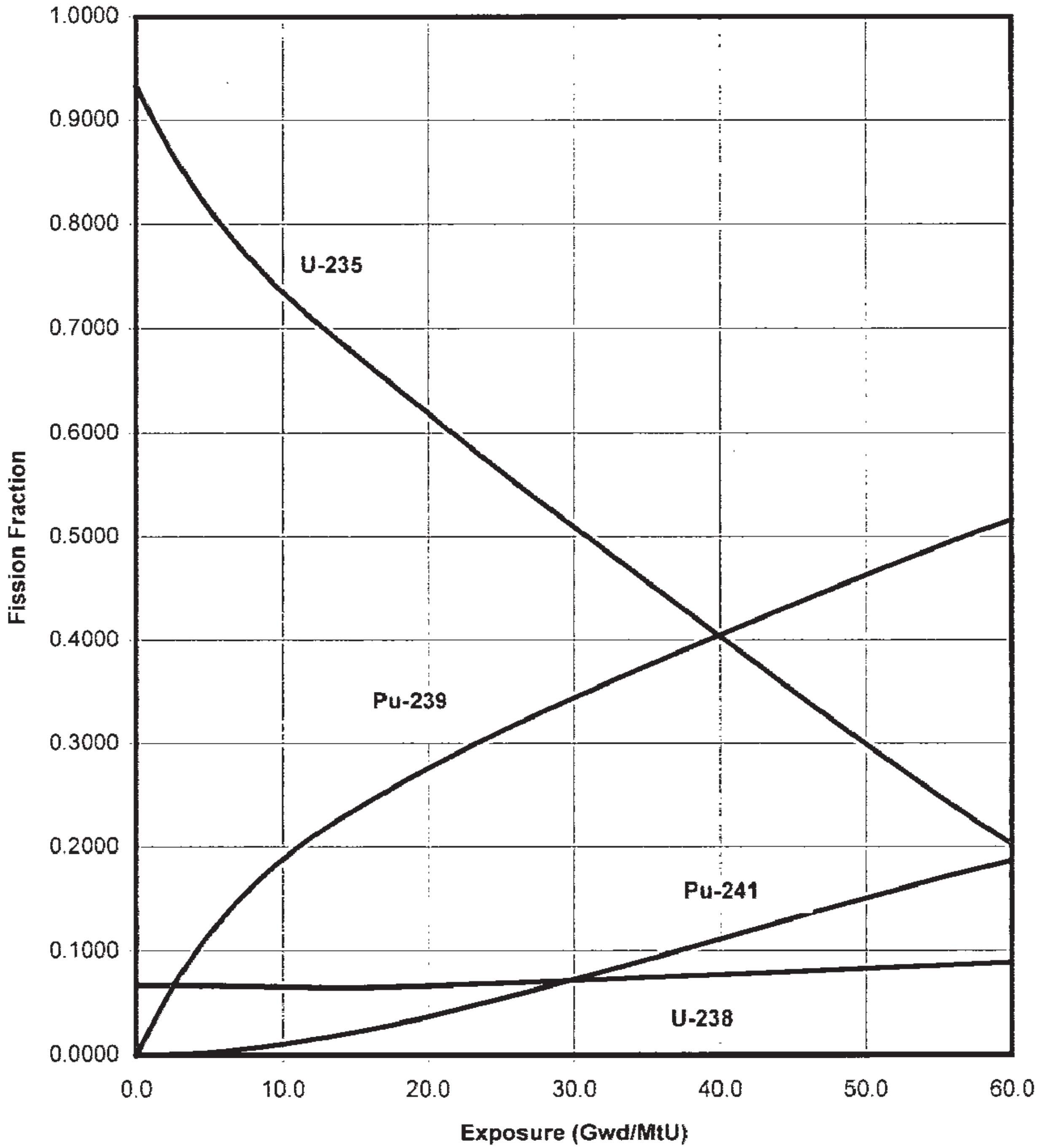
Auto Cad: Figure Fsar 4_3_3.dwg



FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 AND 2 FINAL SAFETY ANALYSIS REPORT
PLUTONIUM BUILDUP AS A FUNCTION OF EXPOSURE, 40% VOIDS (TYPICAL) FIGURE 4.3-4, Rev. 54

Auto Cad: Figure Fsar 4_3_4.dwg



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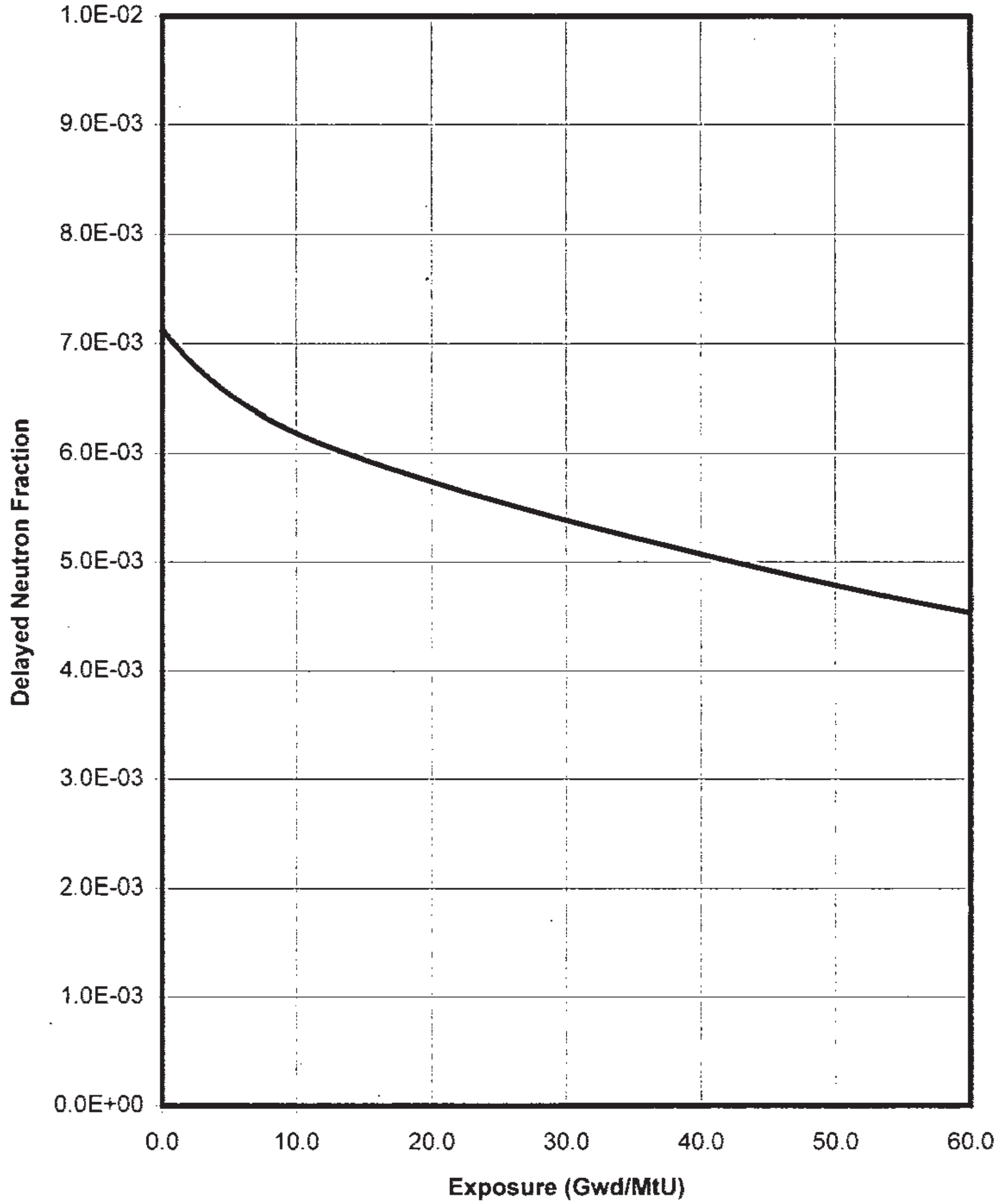
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

FISSION FRACTION AS A FUNCTION OF
EXPOSURE, 40% VOIDS
(TYPICAL)

FIGURE 4.3-5, Rev. 54

Auto Cad: Figure Fsar 4_3_5.dwg



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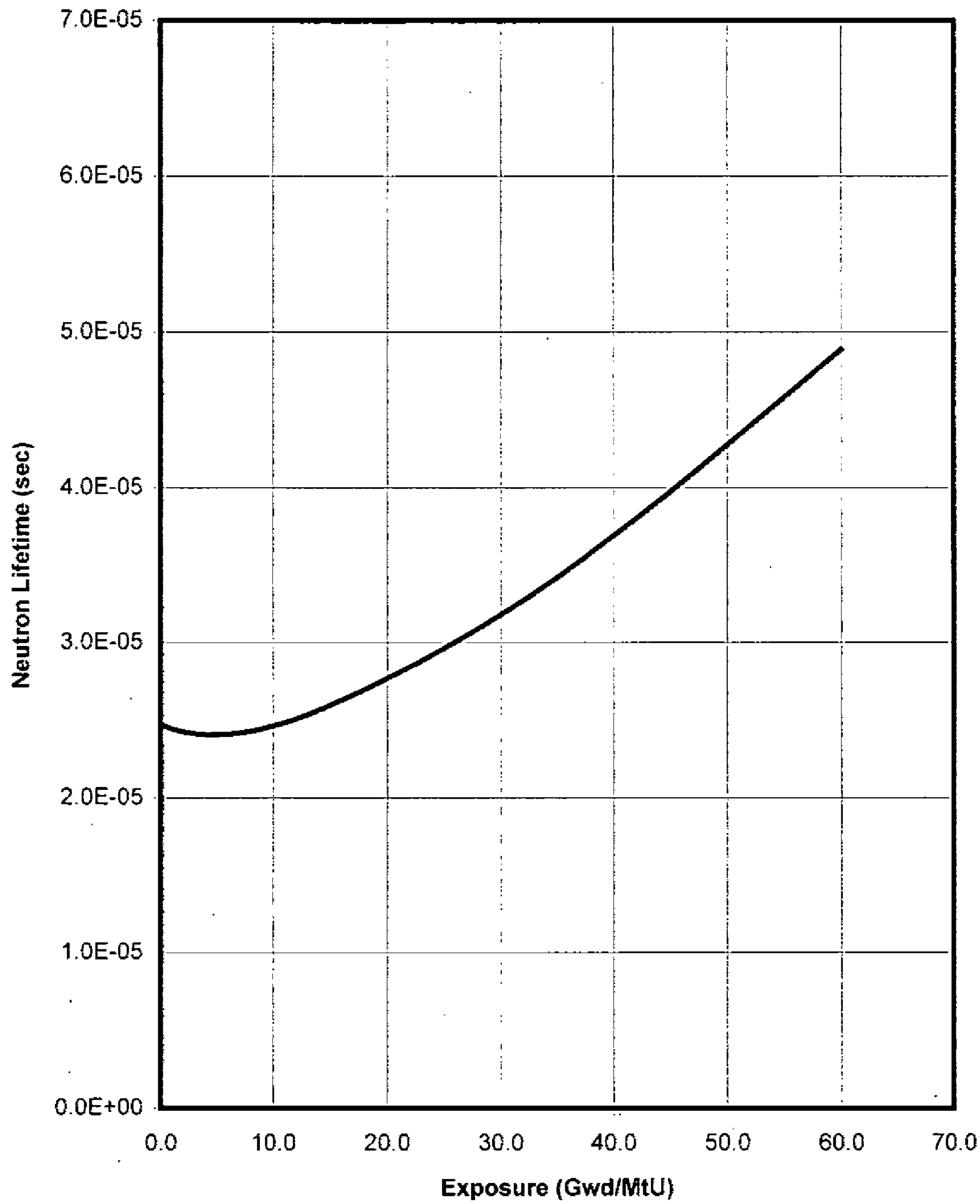
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

DELAYED NEUTRON FRACTION AS A
FUNCTION OF EXPOSURE
40% VOIDS
(TYPICAL)

FIGURE 4.3-6, Rev. 54

Auto Cad: Figure Fsar 4_3_6.dwg



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

NEUTRON LIFETIME AS A FUNCTION OF
EXPOSURE, 40% VOIDS
(TYPICAL)

FIGURE 4.3-7, Rev. 54

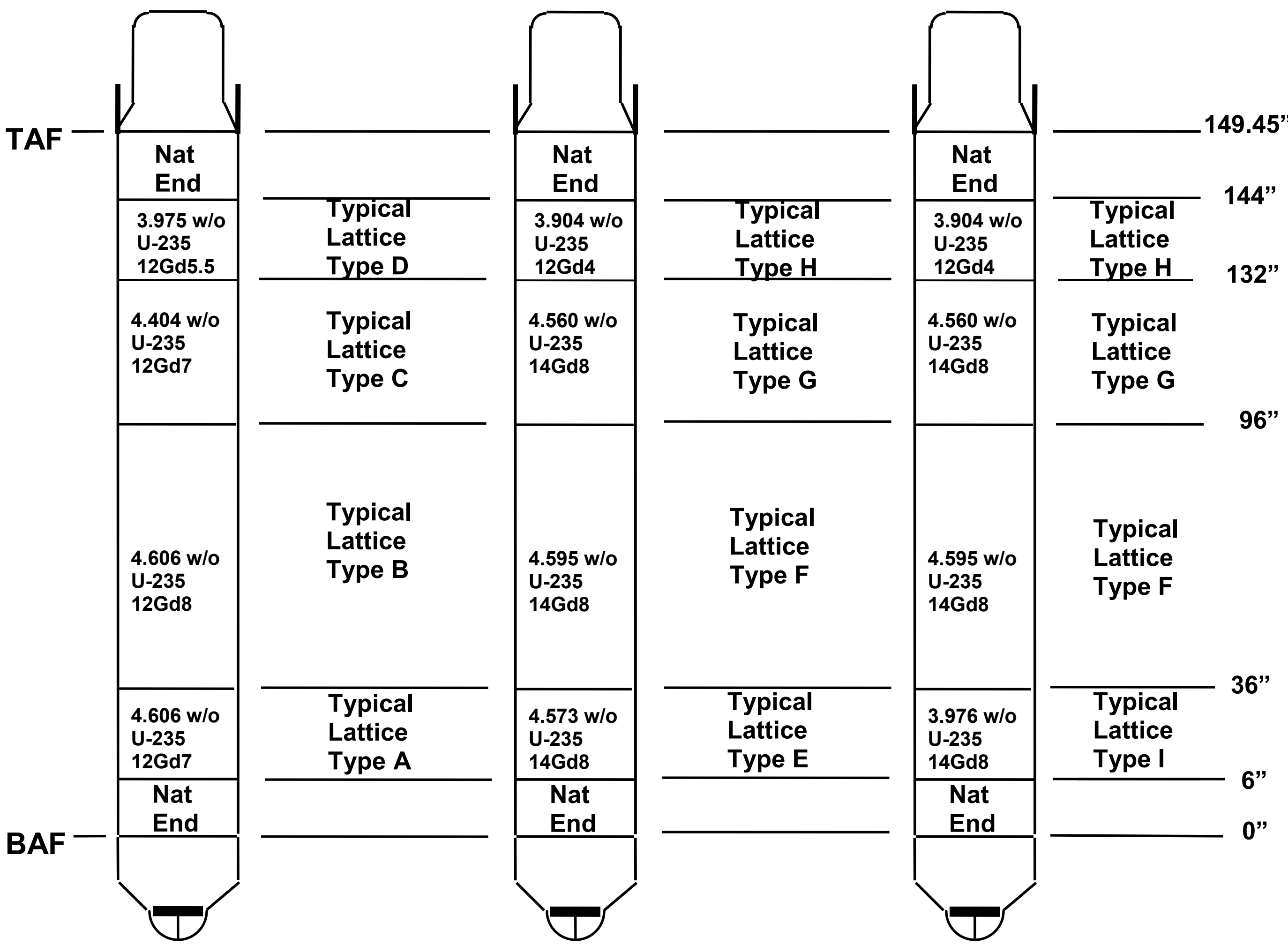
Auto Cad: Figure Fsar 4_3_7.dwg

Typical Assembly Types

Reload Bundle Description

(ATRIUM-10 with Standard or Advanced Fuel Channel)

3.7% < Bundle Average Enrichment < 4.3%



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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
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ATRIUM™-10 FUEL AXIAL ENRICHMENT
(NOMINAL) TYPICAL ASSEMBLIES

FIGURE 4.3-8-46, Rev 0

E1 2.40	E2 3.60	E3 4.45	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E3 4.45	E2 3.60	E1 2.40
E2 3.60	E5 4.95	G1 4.45 7.00	E5 4.95	E5 4.95	E5 4.95	G1 4.45 7.00	E5 4.95	E5 4.95	E2 3.60
E3 4.45	G1 4.45 7.00	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95	G1 4.45 7.00	E3 4.45
E5 4.95	E5 4.95	E5 4.95	G1 4.45 7.00	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95
E5 4.95	E5 4.95	E5 4.95	E5 4.95				E5 4.95	G1 4.45 7.00	E5 4.95
E5 4.95	E5 4.95	E5 4.95	E5 4.95		Water Channel		E5 4.95	E5 4.95	E5 4.95
E5 4.95	G1 4.45 7.00	E5 4.95	E5 4.95				E5 4.95	G1 4.45 7.00	E4 4.70
E3 4.45	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95	G1 4.45 7.00	E5 4.95	E3 4.45
E2 3.60	E5 4.95	G1 4.45 7.00	E5 4.95	G1 4.45 7.00	E5 4.95	G1 4.45 7.00	E5 4.95	E5 4.95	E2 3.60
E1 2.40	E2 3.60	E3 4.45	E5 4.95	E5 4.95	E5 4.95	E4 4.70	E3 4.45	E2 3.60	E1 2.40

Type
U-235

Type
U-235
Gd_2O_3

Pellet Type	Quantity	U-235 + Gd_2O_3 Concentration (wt%)
E1	4	2.40
E2	8	3.60
E3	8	4.45
G1	12	4.45 + 7.00
E4	2	4.70
E5	57	4.95

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ATRIUM™-10 FUEL RADIAL ENRICHMENT (NOMINAL) TYPICAL LATTICE A

Figure 4.3-9-123, Rev. 0

E1 2.40	E2 3.60	E3 4.45	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E3 4.45	E2 3.60	E1 2.40
E2 3.60	E5 4.95	G1 4.45 8.00	E5 4.95	E5 4.95	E5 4.95	G1 4.45 8.00	E5 4.95	E5 4.95	E2 3.60
E3 4.45	G1 4.45 8.00	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95	G1 4.45 8.00	E3 4.45
E5 4.95	E5 4.95	E5 4.95	G1 4.45 8.00	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95
E5 4.95	E5 4.95	E5 4.95	E5 4.95				E5 4.95	G1 4.45 8.00	E5 4.95
E5 4.95	E5 4.95	E5 4.95	E5 4.95		Water Channel		E5 4.95	E5 4.95	E5 4.95
E5 4.95	G1 4.45 8.00	E5 4.95	E5 4.95				E5 4.95	G1 4.45 8.00	E4 4.70
E3 4.45	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95	G1 4.45 8.00	E5 4.95	E3 4.45
E2 3.60	E5 4.95	G1 4.45 8.00	E5 4.95	G1 4.45 8.00	E5 4.95	G1 4.45 8.00	E5 4.95	E5 4.95	E2 3.60
E1 2.40	E2 3.60	E3 4.45	E5 4.95	E5 4.95	E5 4.95	E4 4.70	E3 4.45	E2 3.60	E1 2.40

Type U-235	Type U-235 Gd_2O_3

Pellet Type	Quantity	U-235 + Gd_2O_3 Concentration (wt%)
E1	4	2.40
E2	8	3.60
E3	8	4.45
G1	12	4.45 + 8.00
E4	2	4.70
E5	57	4.95

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ATRIUM™-10 FUEL RADIAL ENRICHMENT (NOMINAL) TYPICAL LATTICE B

Figure 4.3-9-124, Rev. 0

E1 2.40	E2 3.20	E3 4.00	E4 4.45	E4 4.45	E4 4.45	E4 4.45	E3 4.00	E2 3.20	E1 2.40
E2 3.20		G1 4.45 7.00	E6 4.95		E6 4.95	G1 4.45 7.00	E6 4.95		E2 3.20
E3 4.00	G1 4.45 7.00	E6 4.95	E6 4.95	E6 4.95	E6 4.95	E6 4.95	E6 4.95	G1 4.45 7.00	E3 4.00
E4 4.45	E6 4.95	E6 4.95	G1 4.45 7.00	E6 4.95	E6 4.95	E6 4.95	E6 4.95	E6 4.95	E5 4.70
E4 4.45		E6 4.95	E6 4.95				E6 4.95	G1 4.45 7.00	E5 4.70
E4 4.45	E6 4.95	E6 4.95	E6 4.95		Water Channel		E5 4.70		E5 4.70
E4 4.45	G1 4.45 7.00	E6 4.95	E6 4.95				E4 4.45	G1 4.45 7.00	E5 4.70
E3 4.00	E6 4.95	E6 4.95	E6 4.95	E6 4.95	E5 4.70	E4 4.45	G1 4.45 7.00	E6 4.95	E3 4.00
E2 3.20		G1 4.45 7.00	E6 4.95	G1 4.45 7.00		G1 4.45 7.00	E6 4.95		E2 3.20
E1 2.40	E2 3.20	E3 4.00	E5 4.70	E5 4.70	E5 4.70	E5 4.70	E3 4.00	E2 3.20	E1 2.40

Type U-235	Type U-235 Gd_2O_3

Pellet Type	Quantity	U-235 + Gd_2O_3 Concentration (wt%)
E1	4	2.40
E2	8	3.20
E3	8	4.00
E4	10	4.45
G1	12	4.45 + 7.00
E5	10	4.70
E6	31	4.95

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UNITS 1 AND 2
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ATRIUM™-10 FUEL RADIAL ENRICHMENT
(NOMINAL)
TYPICAL LATTICE C

Figure 4.3-9-125, Rev. 0

E1 2.40	E2 3.20	E3 3.60	E4 4.00	E4 4.00	E4 4.00	E4 4.00	E3 3.60	E2 3.20	E1 2.40
E2 3.20		G1 4.00 5.50	E5 4.45		E5 4.45	G1 4.00 5.50	E5 4.45		E2 3.20
E3 3.60	G1 4.00 5.50	E5 4.45	E5 4.45	E5 4.45	E5 4.45	E5 4.45	E5 4.45	G1 4.00 5.50	E3 3.60
E4 4.00	E5 4.45	E5 4.45	G1 4.00 5.50	E5 4.45	E5 4.45	E5 4.45	E5 4.45	E5 4.45	E4 4.00
E4 4.00		E5 4.45	E5 4.45	Water Channel			E5 4.45	G1 4.00 5.50	E4 4.00
E4 4.00	E5 4.45	E5 4.45	E5 4.45				E4 4.00		E4 4.00
E4 4.00	G1 4.00 5.50	E5 4.45	E5 4.45				E4 4.00	G1 4.00 5.50	E4 4.00
E3 3.60	E5 4.45	E5 4.45	E5 4.45	E5 4.45	E4 4.00	E4 4.00	G1 4.00 5.50	E5 4.45	E3 3.60
E2 3.20		G1 4.00 5.50	E5 4.45	G1 4.00 5.50		G1 4.00 5.50	E5 4.45		E2 3.20
E1 2.40	E2 3.20	E3 3.60	E4 4.00	E4 4.00	E4 4.00	E4 4.00	E3 3.60	E2 3.20	E1 2.40

Type U-235	Type U-235 Gd_2O_3

Pellet Type	Quantity	U-235 + Gd_2O_3 Concentration (wt%)
E1	4	2.40
E2	8	3.20
E3	8	3.60
E4	20	4.00
G1	12	4.00 + 5.50
E5	31	4.45

A10T-3975L-12G55-FS10 Enrichment Distribution

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

ATRIUM™-10 FUEL RADIAL ENRICHMENT (NOMINAL) TYPICAL LATTICE D

Figure 4.3-9-126, Rev. 0

E1 2.40	E2 3.60	E3 4.45	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E3 4.45	E2 3.60	E1 2.40
E2 3.60	E5 4.95	G1 4.45 8.00	E5 4.95	E5 4.95	E5 4.95	G1 4.45 8.00	E5 4.95	E5 4.95	E2 3.60
E3 4.45	G1 4.45 8.00	E5 4.95	E5 4.95	G1 4.45 8.00	E5 4.95	E5 4.95	E5 4.95	G1 4.45 8.00	E3 4.45
E5 4.95	E5 4.95	E5 4.95	G1 4.45 8.00	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E4 4.70
E5 4.95	E5 4.95	G1 4.45 8.00	E5 4.95				E5 4.95	G1 4.45 8.00	E4 4.70
E5 4.95	E5 4.95	E5 4.95	E5 4.95		Water Channel		E5 4.95	E5 4.95	E4 4.70
E5 4.95	G1 4.45 8.00	E5 4.95	E5 4.95				E5 4.95	G1 4.45 8.00	E3 4.45
E3 4.45	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95	G1 4.45 8.00	E5 4.95	E3 4.45
E2 3.60	E5 4.95	G1 4.45 8.00	E5 4.95	G1 4.45 8.00	E5 4.95	G1 4.45 8.00	E5 4.95	E5 4.95	E2 3.60
E1 2.40	E2 3.60	E3 4.45	E4 4.70	E4 4.70	E4 4.70	E3 4.45	E3 4.45	E2 3.60	E1 2.40

Type U-235	Type U-235 Gd_2O_3

Pellet Type	Quantity	U-235 + Gd_2O_3 Concentration (wt%)
E1	4	2.40
E2	8	3.60
E3	10	4.45
G1	14	4.45 + 8.00
E4	6	4.70
E5	49	4.95

A10B-4573L-14G80-FS10 Enrichment Distribution

FSAR Rev. 69

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

ATRIUM™-10 FUEL RADIAL ENRICHMENT
(NOMINAL)
TYPICAL LATTICE E

Figure 4.3-9-127, Rev. 0

E1 2.40	E2 3.60	E3 4.45	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E3 4.45	E2 3.60	E1 2.40
E2 3.60	E5 4.95	G1 4.45 8.00	E5 4.95	E5 4.95	E5 4.95	G1 4.45 8.00	E5 4.95	E5 4.95	E2 3.60
E3 4.45	G1 4.45 8.00	E5 4.95	E5 4.95	G1 4.45 8.00	E5 4.95	E5 4.95	E5 4.95	G1 4.45 8.00	E3 4.45
E5 4.95	E5 4.95	E5 4.95	G1 4.45 8.00	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95
E5 4.95	E5 4.95	G1 4.45 8.00	E5 4.95				E5 4.95	G1 4.45 8.00	E5 4.95
E5 4.95	E5 4.95	E5 4.95	E5 4.95		Water Channel		E5 4.95	E5 4.95	E5 4.95
E5 4.95	G1 4.45 8.00	E5 4.95	E5 4.95				E5 4.95	G1 4.45 8.00	E4 4.70
E3 4.45	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95	G1 4.45 8.00	E5 4.95	E3 4.45
E2 3.60	E5 4.95	G1 4.45 8.00	E5 4.95	G1 4.45 8.00	E5 4.95	G1 4.45 8.00	E5 4.95	E5 4.95	E2 3.60
E1 2.40	E2 3.60	E3 4.45	E5 4.95	E5 4.95	E5 4.95	E4 4.70	E3 4.45	E2 3.60	E1 2.40

Type U-235	Type U-235 Gd_2O_3

Pellet Type	Quantity	U-235 + Gd_2O_3 Concentration (wt%)
E1	4	2.40
E2	8	3.60
E3	8	4.45
G1	14	4.45 + 8.00
E4	2	4.70
E5	55	4.95

A10B-4595L-14G80-FS10 Enrichment Distribution

FSAR Rev. 69

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

ATRIUM™-10 FUEL RADIAL ENRICHMENT (NOMINAL)
TYPICAL LATTICE F

Figure 4.3-9-128, Rev. 0

E1 2.40	E2 3.60	E3 4.45	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E3 4.45	E2 3.60	E1 2.40
E2 3.60		G1 4.45 8.00	E5 4.95		E5 4.95	G1 4.45 8.00	E5 4.95		E2 3.60
E3 4.45	G1 4.45 8.00	E5 4.95	E5 4.95	G1 4.45 8.00	E5 4.95	E5 4.95	E5 4.95	G1 4.45 8.00	E3 4.45
E5 4.95	E5 4.95	E5 4.95	G1 4.45 8.00	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95
E5 4.95		G1 4.45 8.00	E5 4.95				E5 4.95	G1 4.45 8.00	E5 4.95
E5 4.95	E5 4.95	E5 4.95	E5 4.95	Water Channel			E5 4.95		E5 4.95
E5 4.95	G1 4.45 8.00	E5 4.95	E5 4.95				E5 4.95	G1 4.45 8.00	E4 4.70
E3 4.45	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95	G1 4.45 8.00	E5 4.95	E3 4.45
E2 3.60		G1 4.45 8.00	E5 4.95	G1 4.45 8.00		G1 4.45 8.00	E5 4.95		E2 3.60
E1 2.40	E2 3.60	E3 4.45	E5 4.95	E5 4.95	E5 4.95	E4 4.70	E3 4.45	E2 3.60	E1 2.40

Type U-235	Type U-235 Gd_2O_3

Pellet Type	Quantity	U-235 + Gd_2O_3 Concentration (wt%)
E1	4	2.40
E2	8	3.60
E3	8	4.45
G1	14	4.45 + 8.00
E4	2	4.70
E5	47	4.95

A10T-4560L-14G80-FS10 Enrichment Distribution

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**SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT**

**ATRIUM™-10 FUEL RADIAL ENRICHMENT (NOMINAL)
TYPICAL LATTICE G**

Figure 4.3-9-129, Rev. 0

E1 2.40	E2 3.20	E3 3.60	E4 4.00	E4 4.00	E4 4.00	E4 4.00	E3 3.60	E2 3.20	E1 2.40
E2 3.20		G1 3.80 4.00	E5 4.45		E5 4.45	G1 3.80 4.00	E5 4.45		E2 3.20
E3 3.60	G1 3.80 4.00	E5 4.45	E5 4.45	E5 4.45	E5 4.45	E5 4.45	E5 4.45	G1 3.80 4.00	E3 3.60
E4 4.00	E5 4.45	E5 4.45	G1 3.80 4.00	E5 4.45	E5 4.45	E5 4.45	E4 4.00	E4 4.00	E4 4.00
E4 4.00		E5 4.45	E5 4.45				E4 4.00	G1 3.80 4.00	E4 4.00
E4 4.00	E5 4.45	E5 4.45	E5 4.45		Water Channel		E4 4.00		E4 4.00
E4 4.00	G1 3.80 4.00	E5 4.45	E5 4.45				E4 4.00	G1 3.80 4.00	E4 4.00
E3 3.60	E5 4.45	E5 4.45	E4 4.00	E4 4.00	E4 4.00	E4 4.00	G1 3.80 4.00	E4 4.00	E3 3.60
E2 3.20		G1 3.80 4.00	E4 4.00	G1 3.80 4.00		G1 3.80 4.00	E4 4.00		E2 3.20
E1 2.40	E2 3.20	E3 3.60	E4 4.00	E4 4.00	E4 4.00	E4 4.00	E3 3.60	E2 3.20	E1 2.40

Type U-235	Type U-235 Gd_2O_3

Pellet Type	Quantity	U-235 + Gd_2O_3 Concentration (wt%)
E1	4	2.40
E2	8	3.20
E3	8	3.60
G1	12	3.80 + 4.00
E4	28	4.00
E5	23	4.45

A10T-3904L-12G40-FS10 Enrichment Distribution

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ATRIUM™-10 FUEL RADIAL ENRICHMENT
(NOMINAL)
TYPICAL LATTICE H

Figure 4.3-9-130, Rev. 0

E1 2.40	E2 3.20	E3 3.60	E4 4.00	E4 4.00	E4 4.00	E4 4.00	E3 3.60	E2 3.20	E1 2.40
E2 3.20	E5 4.45	G1 3.80 8.00	E5 4.45	E5 4.45	E5 4.45	G1 3.80 8.00	E5 4.45	E5 4.45	E2 3.20
E3 3.60	G1 3.80 8.00	E5 4.45	E5 4.45	G1 3.80 8.00	E5 4.45	E5 4.45	E5 4.45	G1 3.80 8.00	E3 3.60
E4 4.00	E5 4.45	E5 4.45	G1 3.80 8.00	E5 4.45	E5 4.45	E5 4.45	E5 4.45	E5 4.45	E4 4.00
E4 4.00	E5 4.45	G1 3.80 8.00	E5 4.45				E5 4.45	G1 3.80 8.00	E4 4.00
E4 4.00	E5 4.45	E5 4.45	E5 4.45		Water Channel		E4 4.00	E5 4.45	E4 4.00
E4 4.00	G1 3.80 8.00	E5 4.45	E5 4.45				E4 4.00	G1 3.80 8.00	E4 4.00
E3 3.60	E5 4.45	E5 4.45	E5 4.45	E5 4.45	E4 4.00	E4 4.00	G1 3.80 8.00	E5 4.45	E3 3.60
E2 3.20	E5 4.45	G1 3.80 8.00	E5 4.45	G1 3.80 8.00	E5 4.45	G1 3.80 8.00	E5 4.45	E5 4.45	E2 3.20
E1 2.40	E2 3.20	E3 3.60	E4 4.00	E4 4.00	E4 4.00	E4 4.00	E3 3.60	E2 3.20	E1 2.40

Type U-235	Type U-235 Gd_2O_3

Pellet Type	Quantity	U-235 + Gd_2O_3 Concentration (wt%)
E1	4	2.40
E2	8	3.20
E3	8	3.60
G1	14	3.80 + 8.00
E4	20	4.00
E5	37	4.45

A10B-3976L-14G80 Enrichment Distribution

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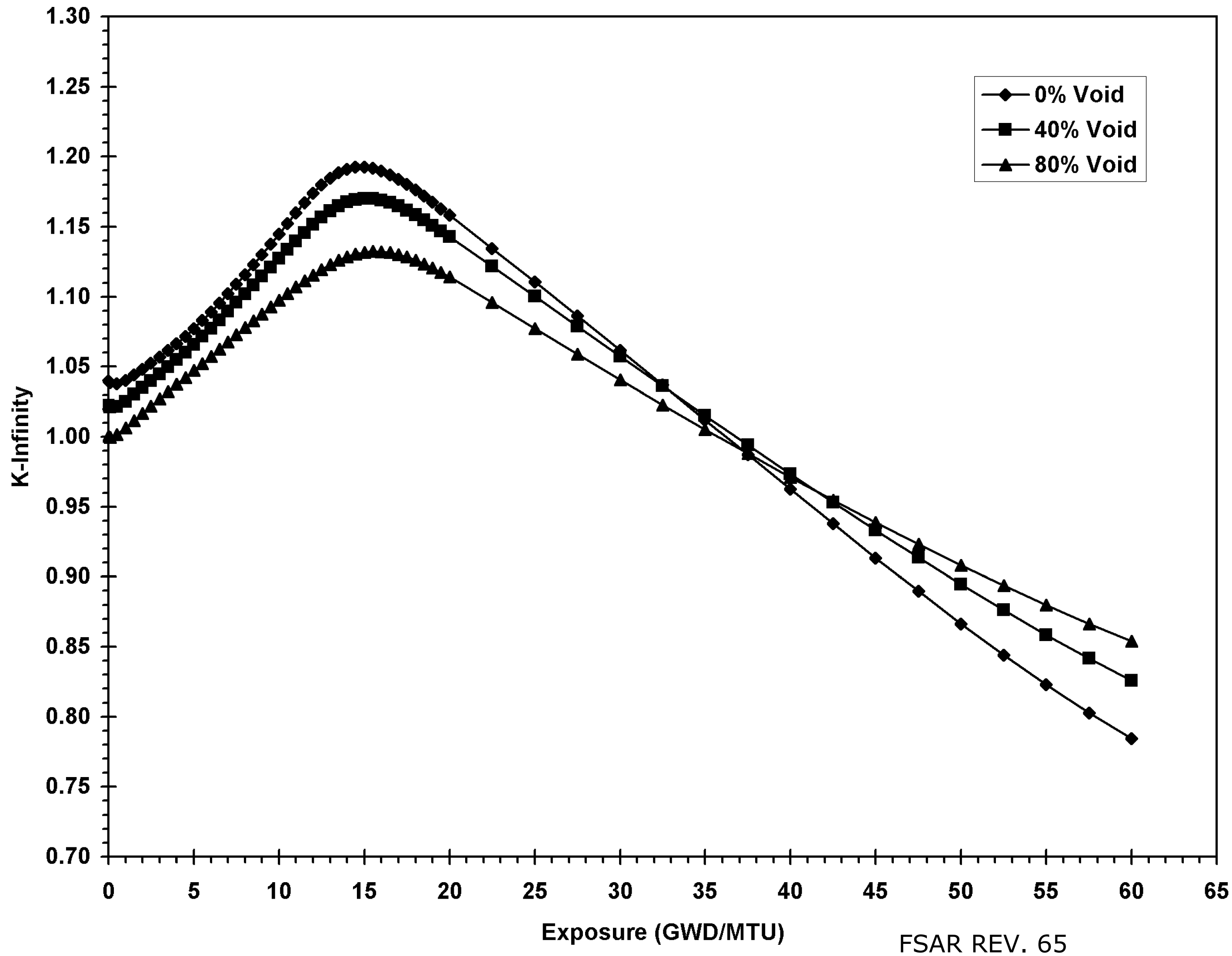
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

ATRIUM™-10 FUEL RADIAL ENRICHMENT
(NOMINAL)
TYPICAL LATTICE I

Figure 4.3-9-131, Rev. 0



FSAR REV. 65

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

FANP ATRIUM^{TM}-10 FUEL DOMINANT LATTICE
HOT - UNCONTROLLED
K-INFINITY VS. EXPOSURE
(TYPICAL)

FIGURE 4.3-11-1, Rev. 55

Auto Cad: Figure Fsar 4_3_11_1.dwg

1.203	1.223	1.209	1.220	1.201	1.198	1.211	1.289	1.041	1.241
1.223	1.070	0.403	0.952	0.885	0.959	0.403	1.080	1.172	1.262
1.209	0.403	0.809	0.844	0.388	0.909	0.884	0.858	0.412	1.232
1.220	0.952	0.844	0.935	1.058	1.170	1.130	0.990	0.953	1.220
1.201	0.885	0.388	1.058	W	W	W	1.081	0.403	1.160
1.198	0.959	0.909	1.170	W	W	W	1.164	0.898	1.196
1.211	0.403	0.884	1.130	W	W	W	1.002	0.407	1.222
1.289	1.080	0.858	0.990	1.081	1.164	1.002	0.465	1.075	1.123
1.041	1.172	0.412	0.953	0.403	0.898	0.407	1.075	1.226	1.292
1.241	1.262	1.232	1.220	1.160	1.196	1.222	1.123	1.292	1.249

Maximum Local Power = 1.292

FSAR REV. 65

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

FANP ATRIUM™-10 FUEL DOMINANT LATTICE
LOCAL PEAKING, UNRODDED, 40% VOIDS
0 MWD/MTU
(TYPICAL)

FIGURE 4.3-11-2, Rev. 55

Auto Cad: Figure Fsar 4_3_11_2.dwg

0.978	1.018	1.060	1.055	1.036	1.040	1.067	1.071	0.903	0.988
1.018	1.028	0.897	0.960	0.885	0.952	0.868	1.030	1.048	1.034
1.060	0.897	0.927	0.894	0.783	0.918	0.913	0.910	0.939	1.087
1.055	0.960	0.894	0.922	1.007	1.052	1.036	0.982	1.020	1.092
1.036	0.885	0.783	1.007	W	W	W	1.065	0.913	1.084
1.040	0.952	0.918	1.052	W	W	W	1.109	0.979	1.089
1.067	0.868	0.913	1.036	W	W	W	1.071	0.949	1.113
1.071	1.030	0.910	0.982	1.065	1.109	1.071	0.973	1.088	0.998
0.903	1.048	0.939	1.020	0.913	0.979	0.949	1.088	1.074	1.049
0.988	1.034	1.087	1.092	1.084	1.089	1.113	0.998	1.049	0.999

Maximum Local Power = 1.113

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

FANP ATRIUM^{TM}-10 FUEL DOMINANT LATTICE
LOCAL PEAKING, UNRODDED, 40% VOIDS
15000 MWD/MTU
(TYPICAL)

FIGURE 4.3-11-3, Rev. 55

Auto Cad: Figure Fsar 4_3_11_3.dwg

0.936	0.954	1.006	1.023	1.020	1.021	1.028	1.003	0.899	0.937
0.954	0.995	0.951	1.013	0.961	1.010	0.943	1.032	0.993	0.955
1.006	0.951	1.009	0.992	0.912	1.000	0.988	0.975	0.964	1.011
1.023	1.013	0.992	1.009	1.044	1.057	1.052	1.030	1.040	1.036
1.020	0.961	0.912	1.044	W	W	W	1.060	0.963	1.040
1.021	1.010	1.000	1.057	W	W	W	1.073	1.001	1.039
1.028	0.943	0.988	1.052	W	W	W	1.053	0.974	1.041
1.003	1.032	0.975	1.030	1.060	1.073	1.053	0.991	1.051	0.960
0.899	0.993	0.964	1.040	0.963	1.001	0.974	1.051	0.999	0.958
0.937	0.955	1.011	1.036	1.040	1.039	1.041	0.960	0.958	0.941

Maximum Local Power = 1.073

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
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FANP ATRIUM^{TM}-10 FUEL DOMINANT LATTICE
LOCAL PEAKING, UNRODDED, 40% VOIDS
40000 MWD/MTU
(TYPICAL)

FIGURE 4.3-11-4, Rev. 55

Auto Cad: Figure Fsar 4_3_11_4.dwg

1.225	1.242	1.225	1.254	1.243	1.235	1.234	1.314	1.058	1.262
1.242	1.052	0.355	0.953	0.893	0.963	0.356	1.068	1.163	1.275
1.225	0.355	0.796	0.843	0.343	0.912	0.887	0.849	0.362	1.237
1.254	0.953	0.843	0.951	1.079	1.197	1.156	0.996	0.933	1.234
1.243	0.893	0.343	1.079	W	W	W	1.088	0.354	1.172
1.235	0.963	0.912	1.197	W	W	W	1.170	0.879	1.210
1.234	0.356	0.887	1.156	W	W	W	0.990	0.355	1.229
1.314	1.068	0.849	0.996	1.088	1.170	0.990	0.409	1.044	1.134
1.058	1.163	0.362	0.933	0.354	0.879	0.355	1.044	1.216	1.304
1.262	1.275	1.237	1.234	1.172	1.210	1.229	1.134	1.304	1.260

Maximum Local Power = 1.314

FSAR REV. 65

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

FANP ATRIUM™-10 FUEL DOMINANT LATTICE
LOCAL PEAKING, UNRODDED, 0% VOIDS
0 MWD/MTU
(TYPICAL)

FIGURE 4.3-11-5, Rev. 55

Auto Cad: Figure Fsar 4_3_11_5.dwg

1.154	1.187	1.178	1.171	1.141	1.144	1.172	1.247	1.006	1.194
1.187	1.085	0.473	0.955	0.878	0.958	0.471	1.095	1.175	1.231
1.178	0.473	0.837	0.857	0.453	0.915	0.892	0.877	0.485	1.217
1.171	0.955	0.857	0.928	1.043	1.152	1.113	0.994	0.980	1.197
1.141	0.878	0.453	1.043	W	W	W	1.083	0.474	1.139
1.144	0.958	0.915	1.152	W	W	W	1.171	0.925	1.171
1.172	0.471	0.892	1.113	W	W	W	1.028	0.481	1.205
1.247	1.095	0.877	0.994	1.083	1.171	1.028	0.545	1.112	1.103
1.006	1.175	0.485	0.980	0.474	0.925	0.481	1.112	1.231	1.264
1.194	1.231	1.217	1.197	1.139	1.171	1.205	1.103	1.264	1.212

Maximum Local Power = 1.264

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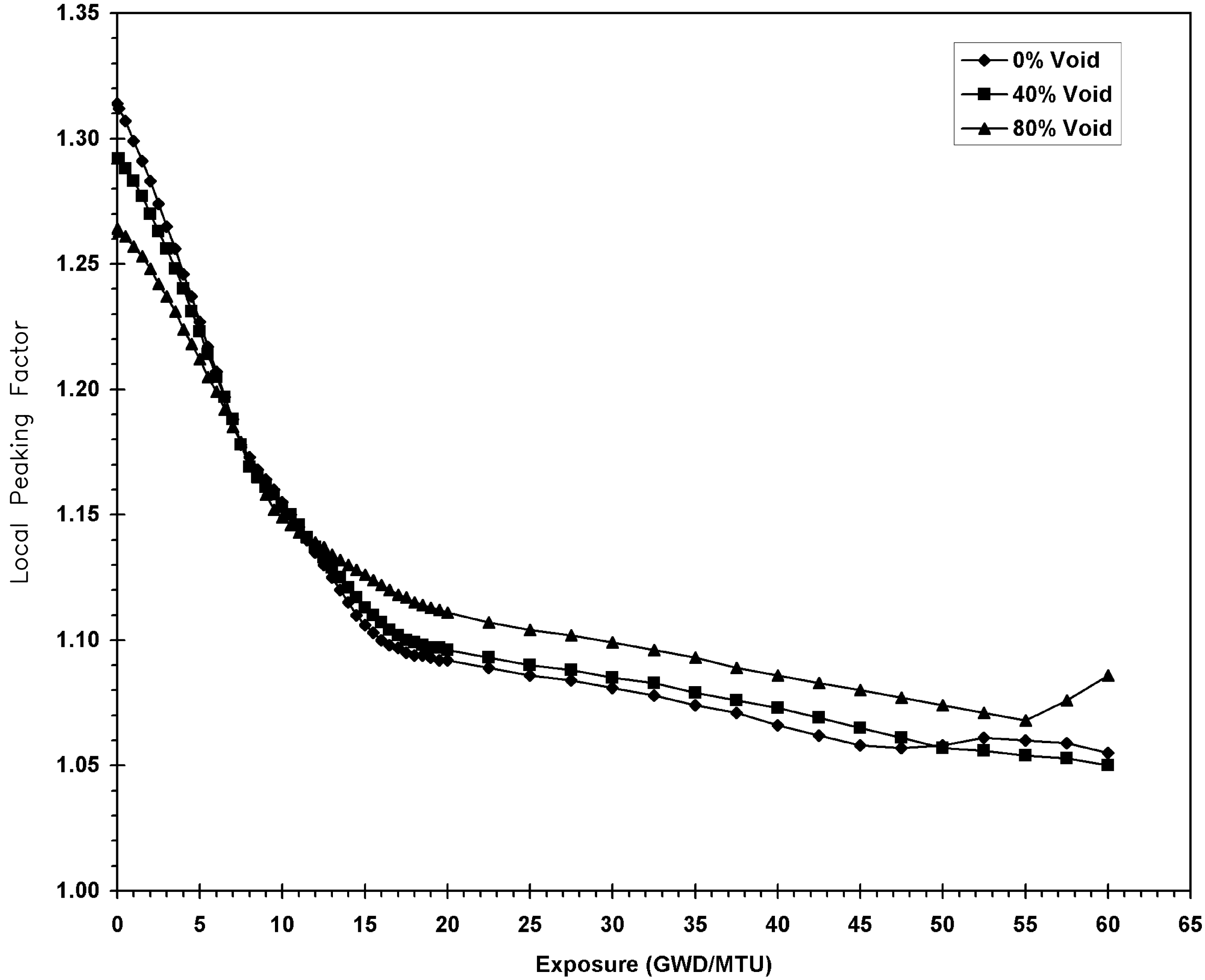
FANP ATRIUM™-10 FUEL DOMINANT LATTICE
LOCAL PEAKING, UNRODDED, 80% VOIDS
0 MWD/MTU
(TYPICAL)

FIGURE 4.3-11-6, Rev. 55

Auto Cad: Figure Fsar 4_3_11_6.dwg



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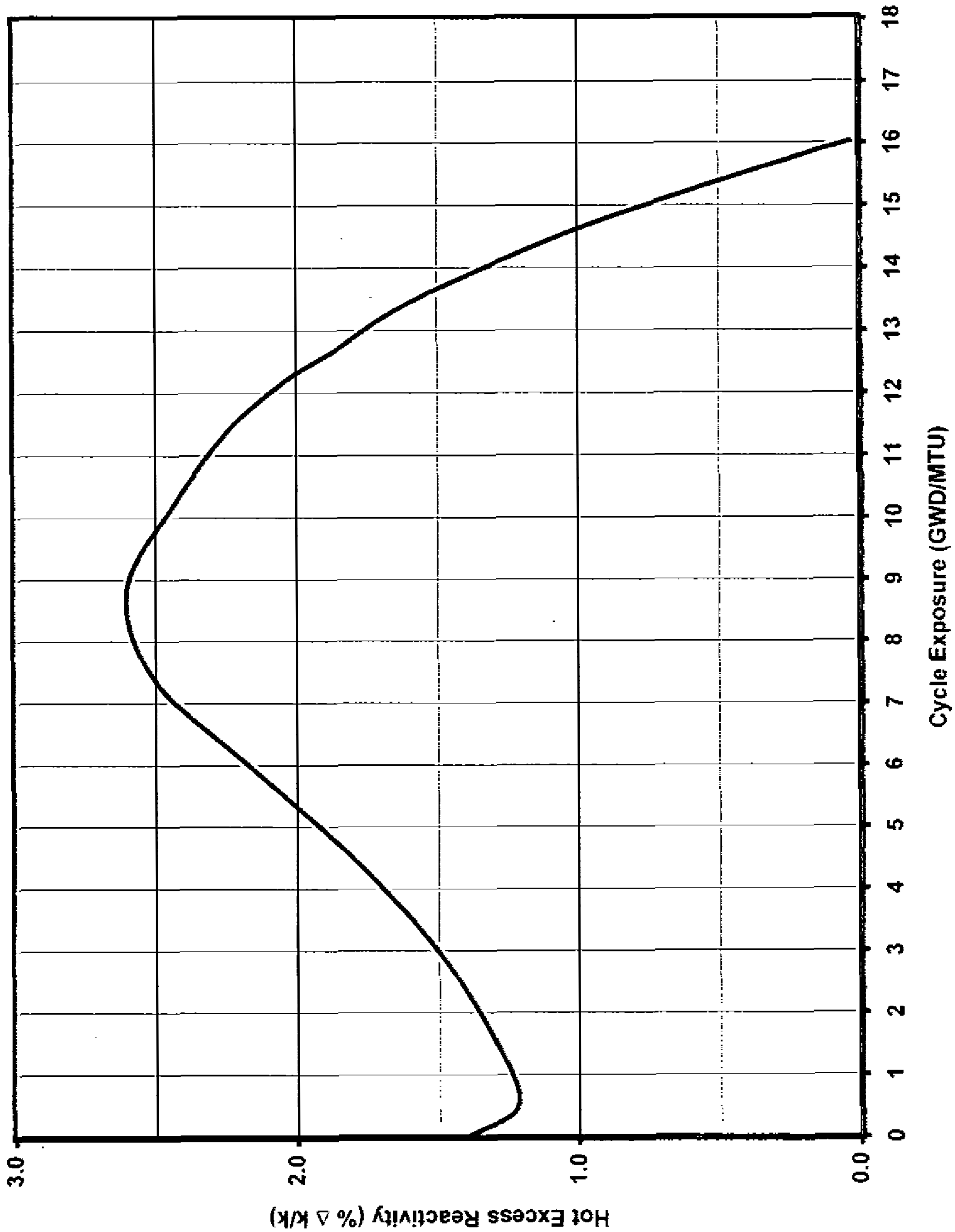
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

FANP ATRIUM™-10 FUEL DOMINANT LATTICE
MAXIMUM HOT-UNCONTROLLED
LOCAL PEAKING FACTOR VS. EXPOSURE
(TYPICAL)

FIGURE 4.3-11-7, Rev. 55

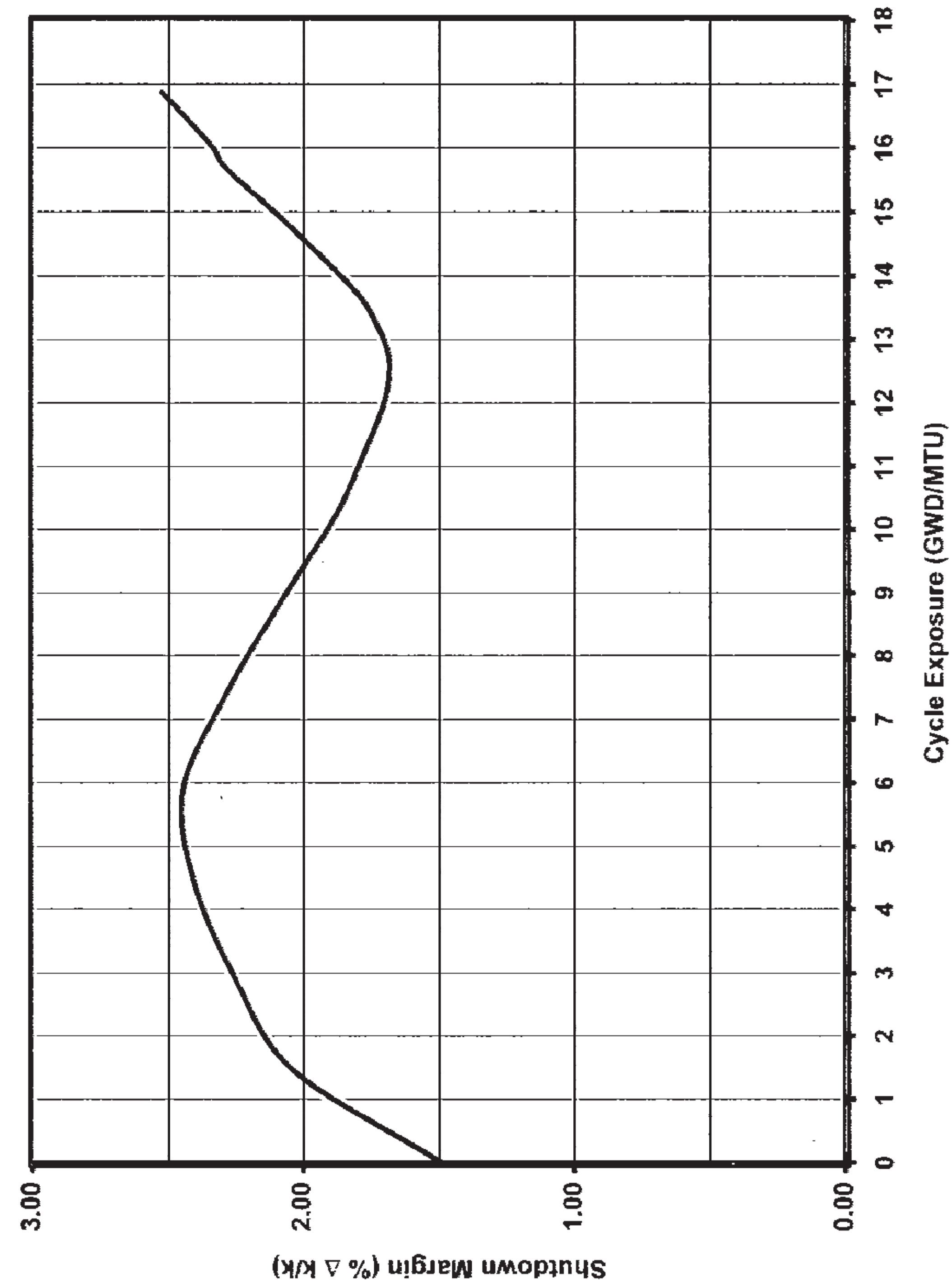
Auto Cad: Figure Fsar 4_3_11_7.dwg



FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 AND 2 FINAL SAFETY ANALYSIS REPORT
HOT EXCESS REACTIVITY VS. EXPOSURE (TYPICAL) FIGURE 4.3-12, Rev. 54

Auto Cad: Figure Fsar 4_3_12.dwg



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

SHUTDOWN MARGIN VS. EXPOSURE
(TYPICAL)

FIGURE 4.3-13, Rev. 54

Auto Cad: Figure Fsar 4_3_13.dwg

Security-Related Information
Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
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VESSEL FLUENCE (R.θ) MODEL FOR
AZIMUTHAL FLUX DISTRIBUTION

FIGURE 4.3-14

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 ANP, Inc. (FANP) 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. This correlation is based on accurate test data of full-scale prototypic simulations of reactor fuel assemblies operating under conditions typical of those in actual reactor designs. The correlation is a "best fit" to the data and is used together with a statistical analysis to assure adequate reactor thermal margins (Reference 4.4-42).

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 FANP 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 FANP 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 FANP 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 FANP's pressure drop methodology is described in Reference 4.4-43.

New data are 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 FANP ATRIUM-10 fuel designs is reported in Reference 4.4-45. The range of tests variables is specified to include the range of interest to boiling water reactors.

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 Reference 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. Examples of fuel related uncertainties are those introduced by the critical power correlation, the calculation of core wide power peaking and the calculation of core wide flow distribution. Examples of uncertainties which are characteristics of the reactor system are the measurement uncertainties associated with reactor pressure, total core flow, feedwater flow and feedwater temperature. 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 SPCB critical power correlation is utilized in thermal-hydraulic evaluations. This correlation is 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 FANP fuel designs are designed to exhibit channel decay ratio characteristics equivalent to existing FANP 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, FANP 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 FANP 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

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4.4-18 Deleted

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4.4-22 Deleted

4.4-23 Deleted

4.4-24 Deleted

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4.4-26 Deleted

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Table 4.4-1
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(1)

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***
	ATRIUM-10
Assembly flow	*4.4-42*
CPR Correlation Additive Constant	*4.4-58 (SPCB)*
Assembly radial peaking	*4.4-44*
Rod local peaking	*4.4-51*

1. Additional uncertainty penalties are given in Reference 4.4-59 for EPU conditions.

Table 4.4-7
BYPASS FLOW PATHS

Security-Related Information
Table Withheld Under 10 CFR 2.390

Table 4.4-8
PLANT CONFIGURATION DATA

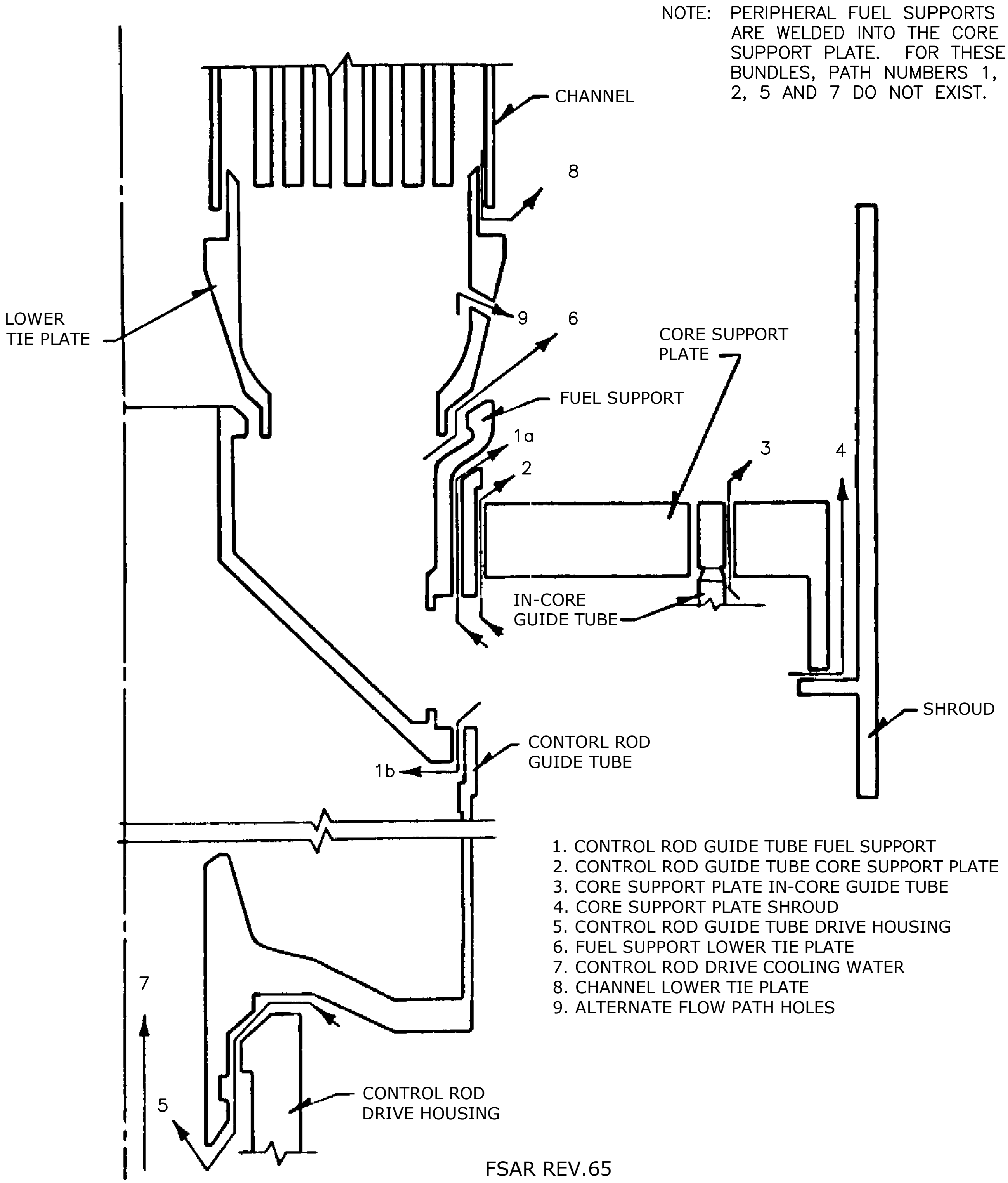
Security-Related Information
Table Withheld Under 10 CFR 2.390

Table 4.4-9

LENGTHS OF SAFETY INJECTION LINES

Security-Related Information

Table Withheld Under 10 CFR 2.390



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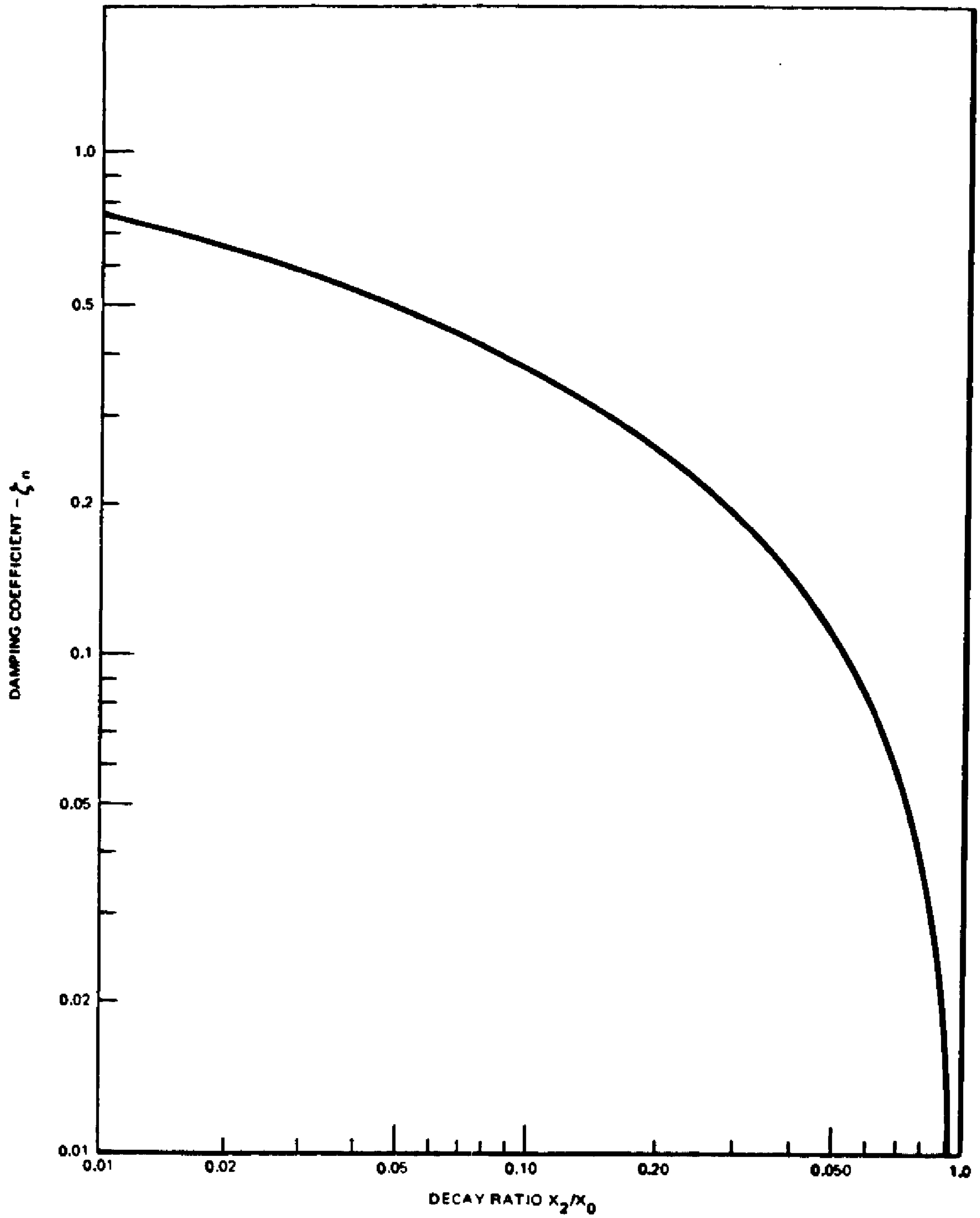
FINAL SAFETY ANALYSIS REPORT

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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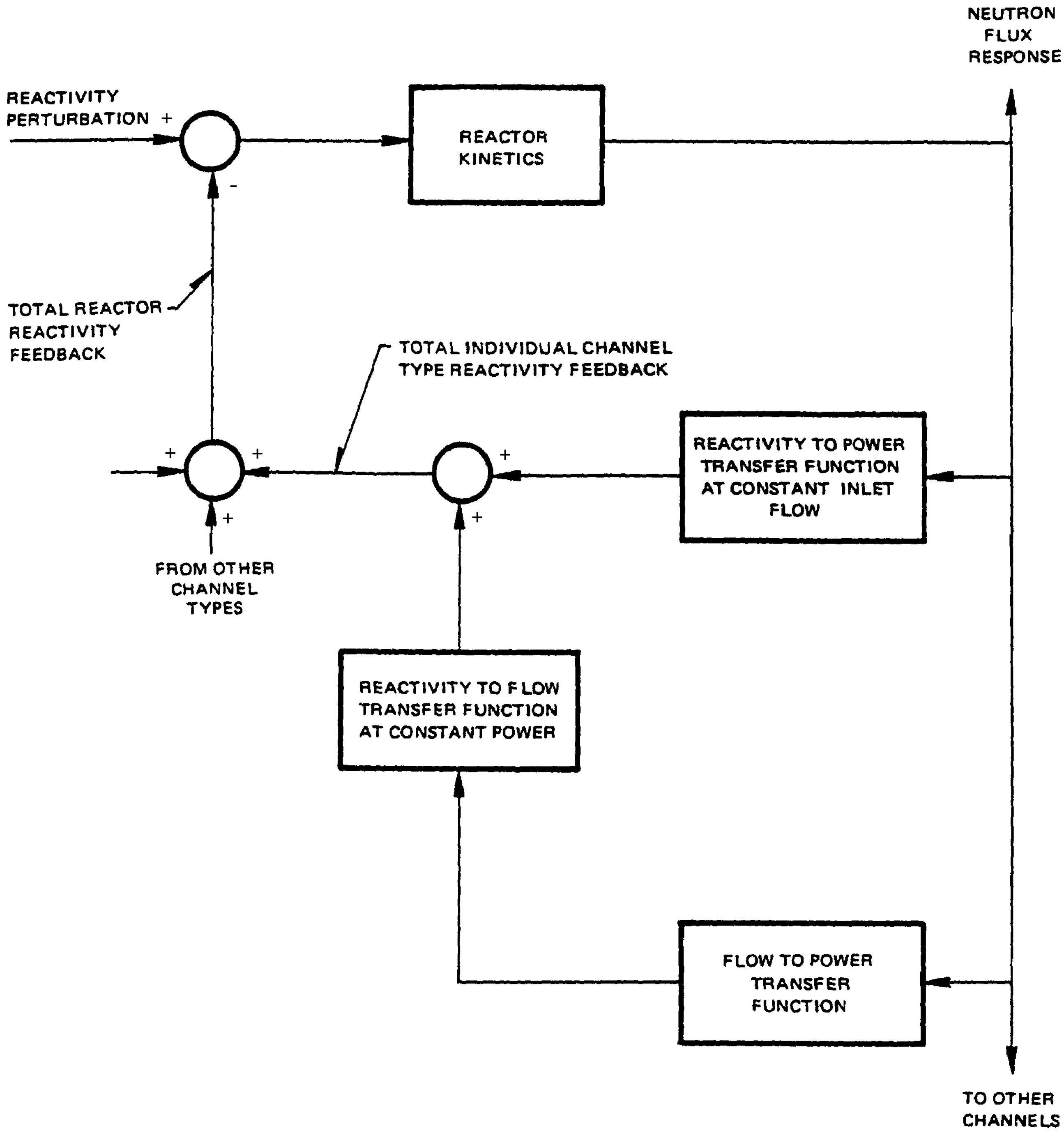
DAMPING COEFFICIENT VERSUS
DECAY RATIO
(SECOND ORDER SYSTEMS)

FIGURE 4.4-2, Rev. 47

Auto-Cad Figure Fsar 4_4_2.dwg



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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
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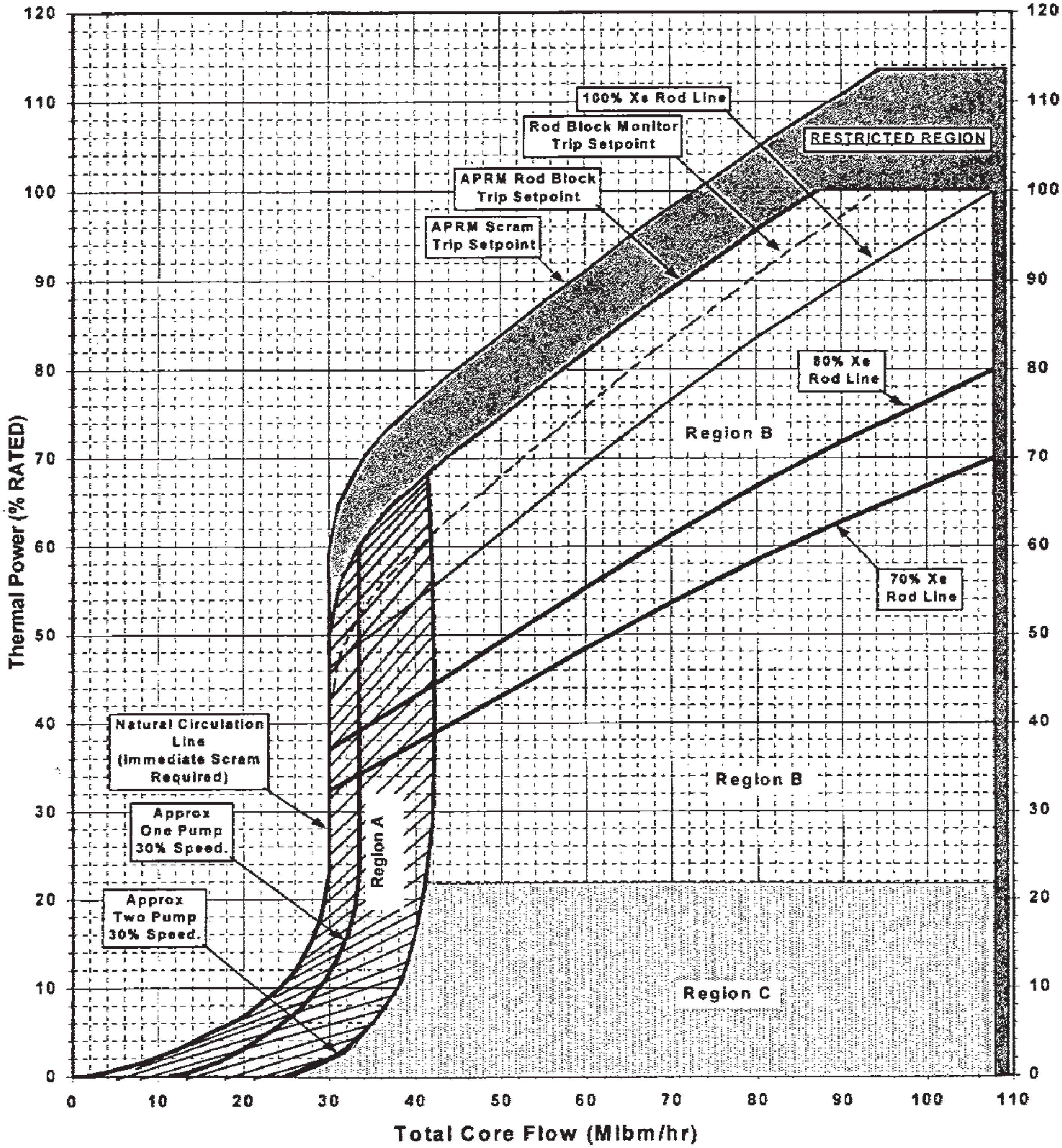
HYDRODYNAMIC AND
CORE STABILITY MODEL
FOR INITIAL CORE

FIGURE 4.4-3, Rev. 54

Auto-Cad Figure Fsar 4_4_3.dwg



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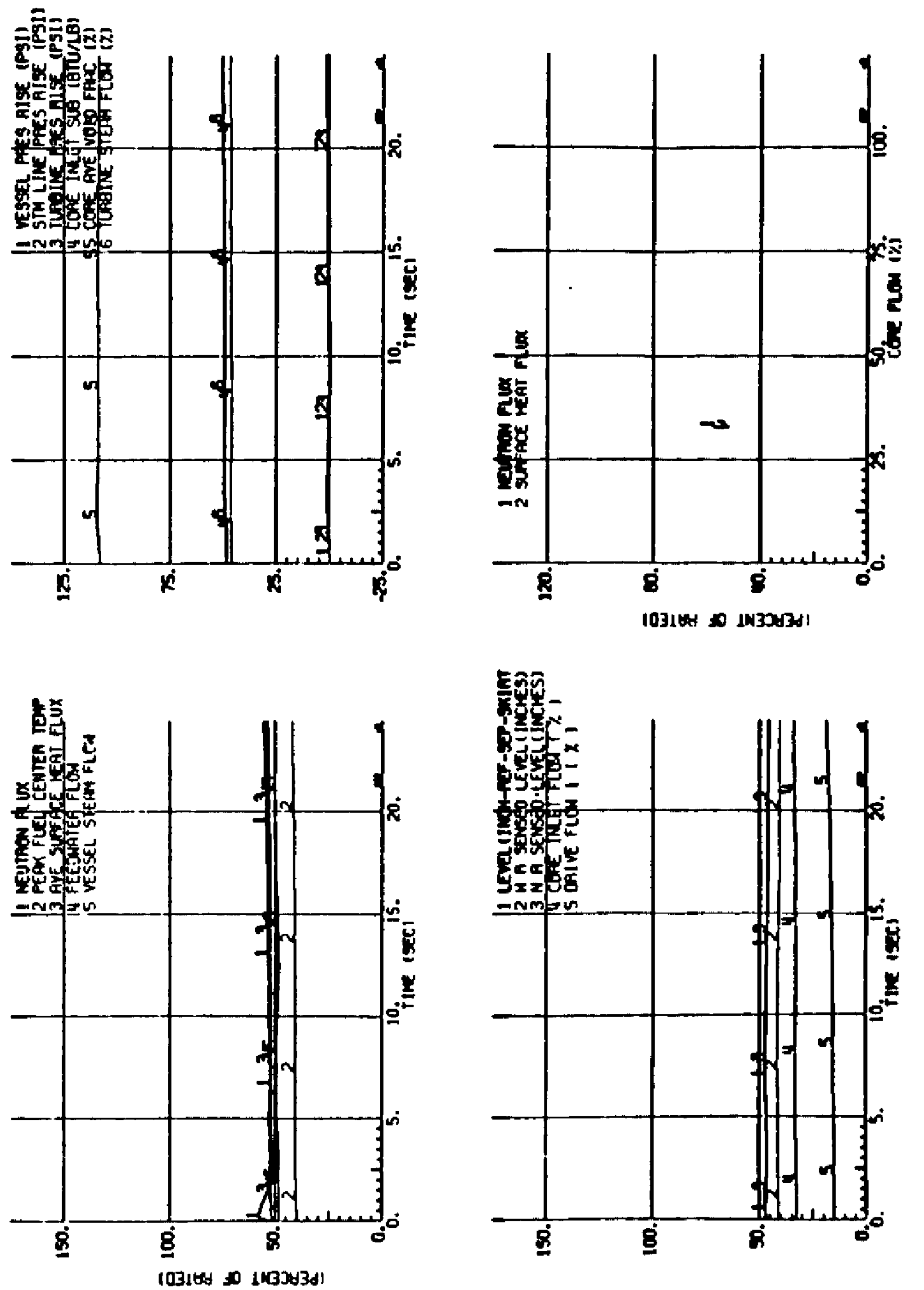
SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
SIMPLIFIED POWER-FLOW OPERATING MAP FIGURE 4.4-5, Rev. 55

Auto-Cad Figure Fsar 4_4_5.dwg



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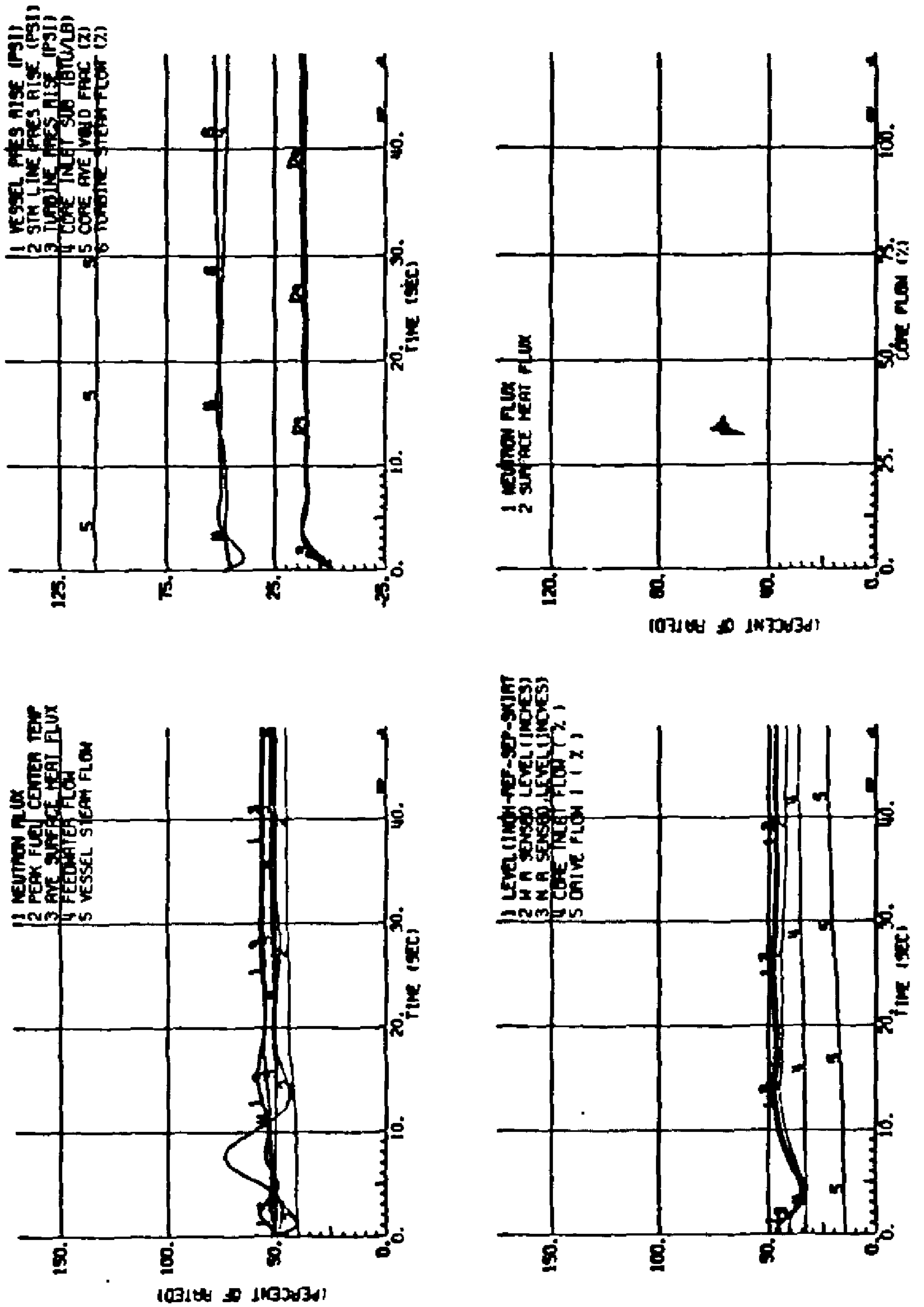
10 CENT ROD REACTIVITY STEP
AT 51.5% RATED POWER
(NATURAL CIRCULATION)

FIGURE 4.4-7A, Rev. 47

Auto-Cad Figure Fsar 4_4_7A.dwg



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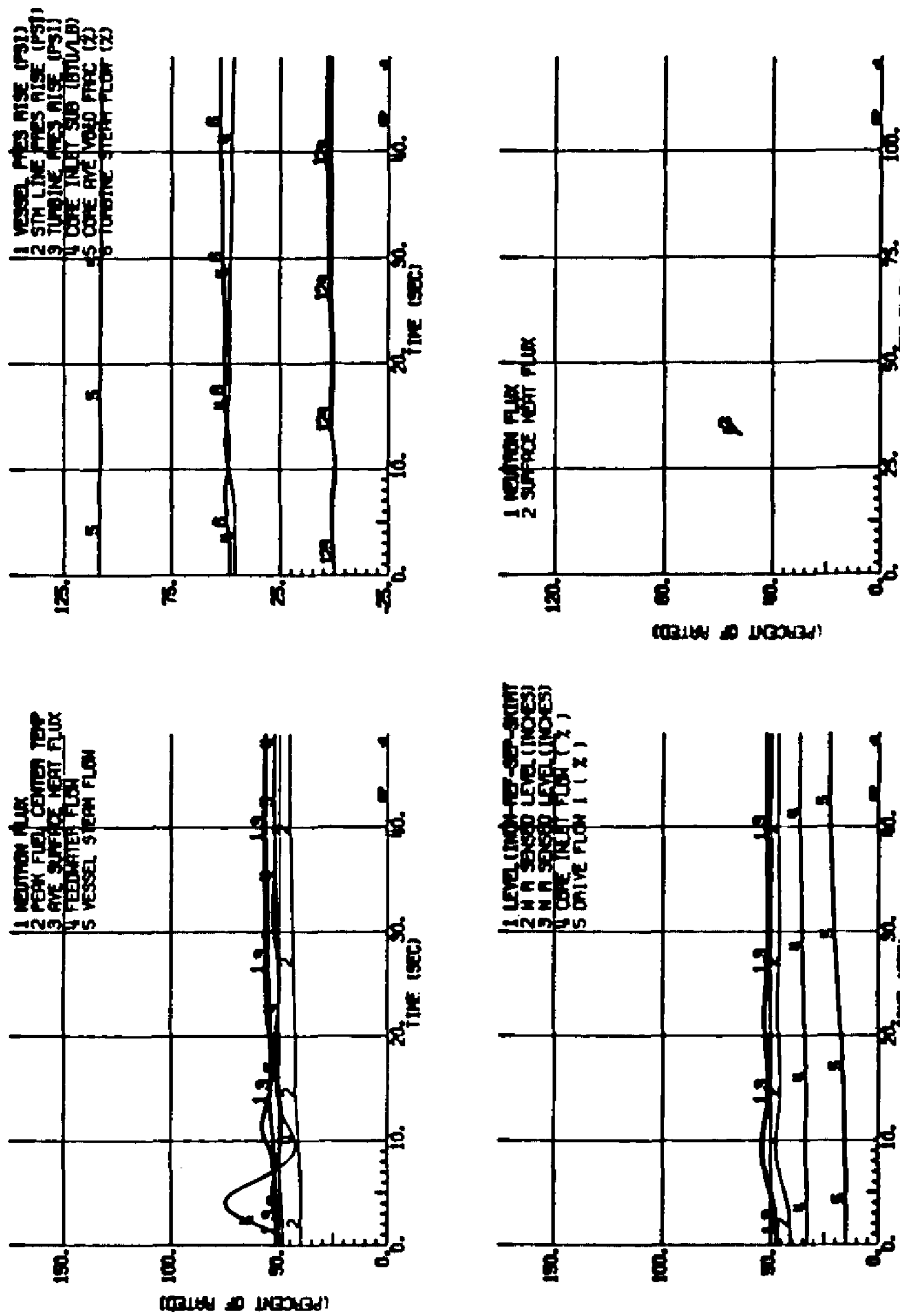
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

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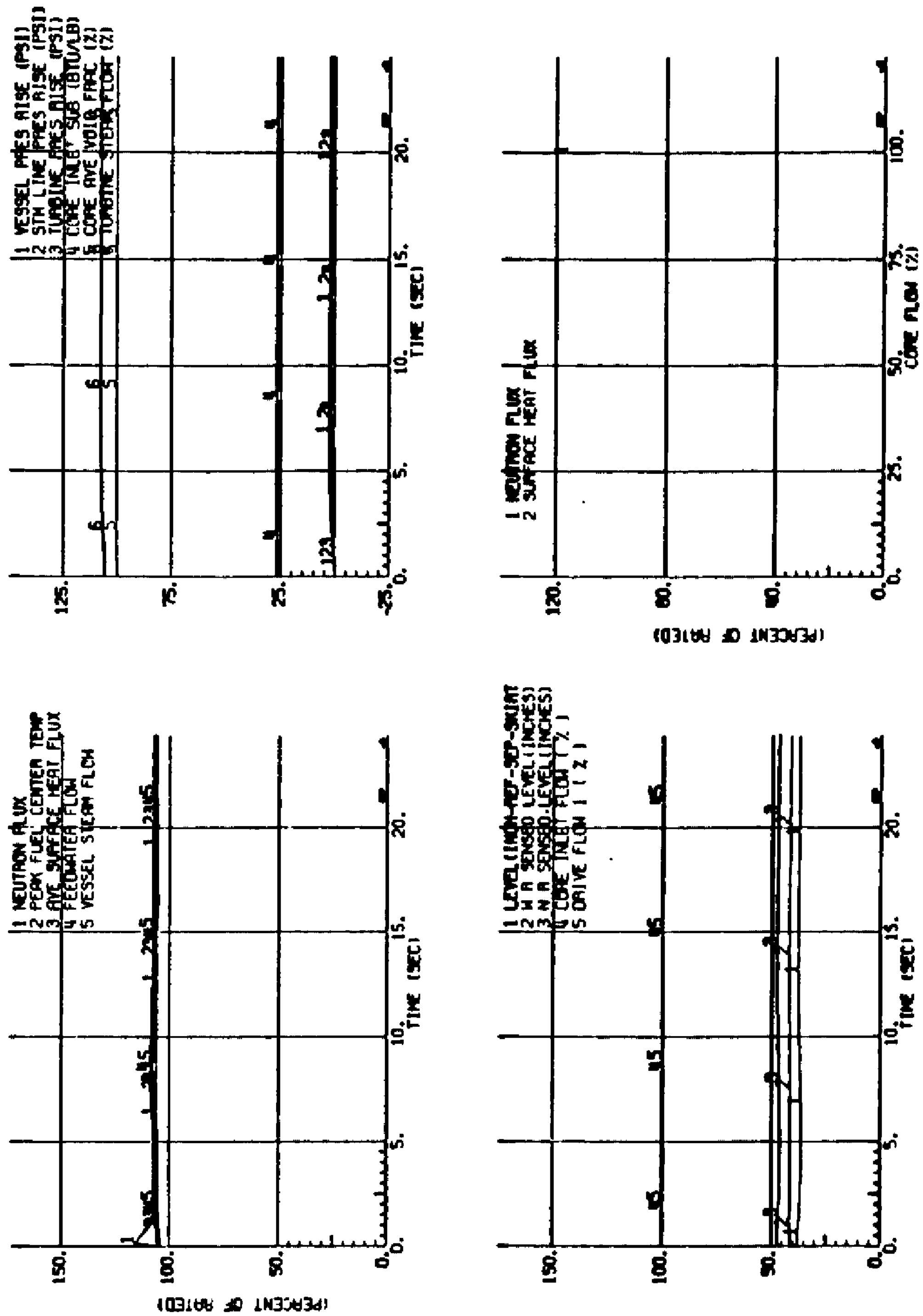
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

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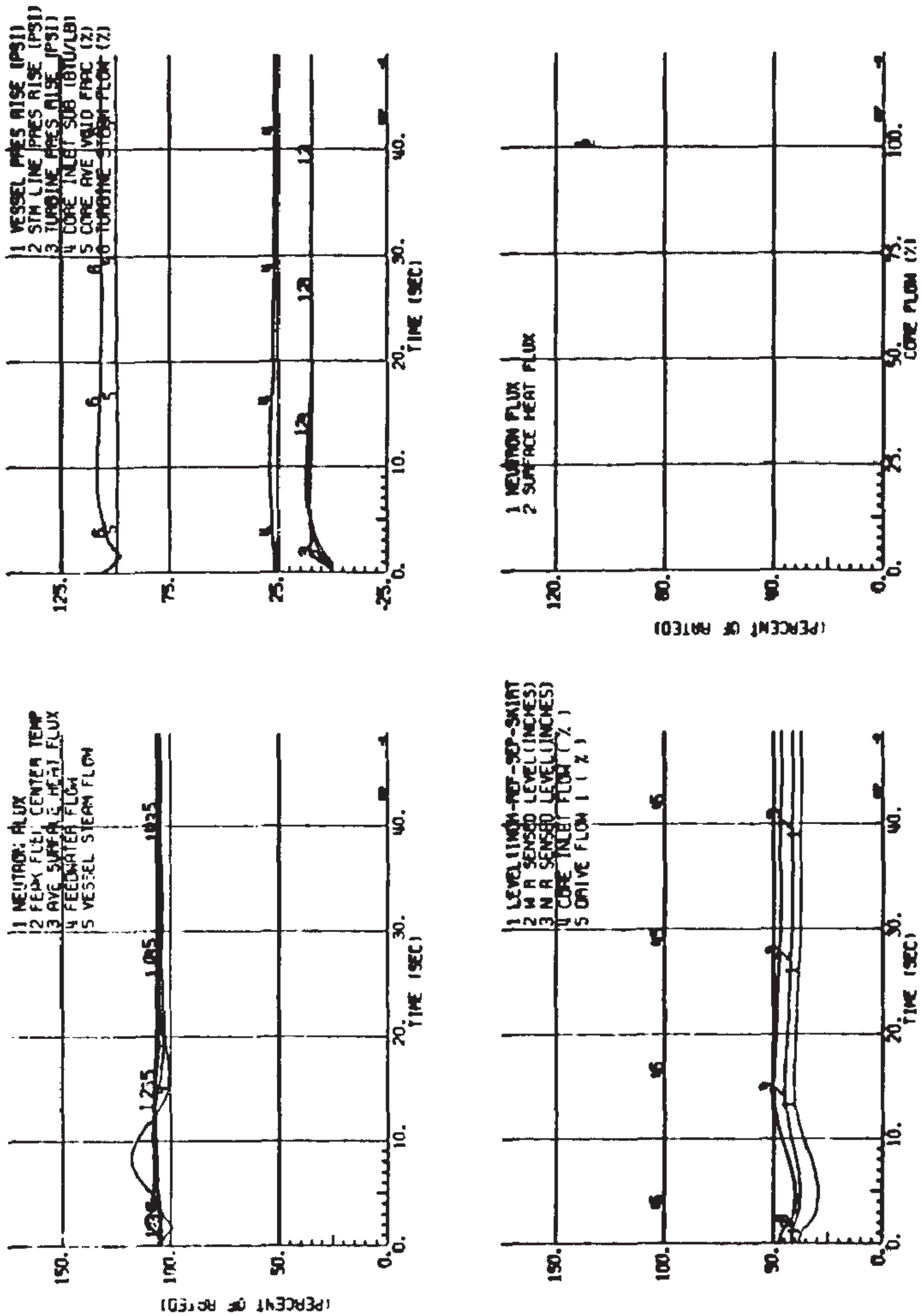
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

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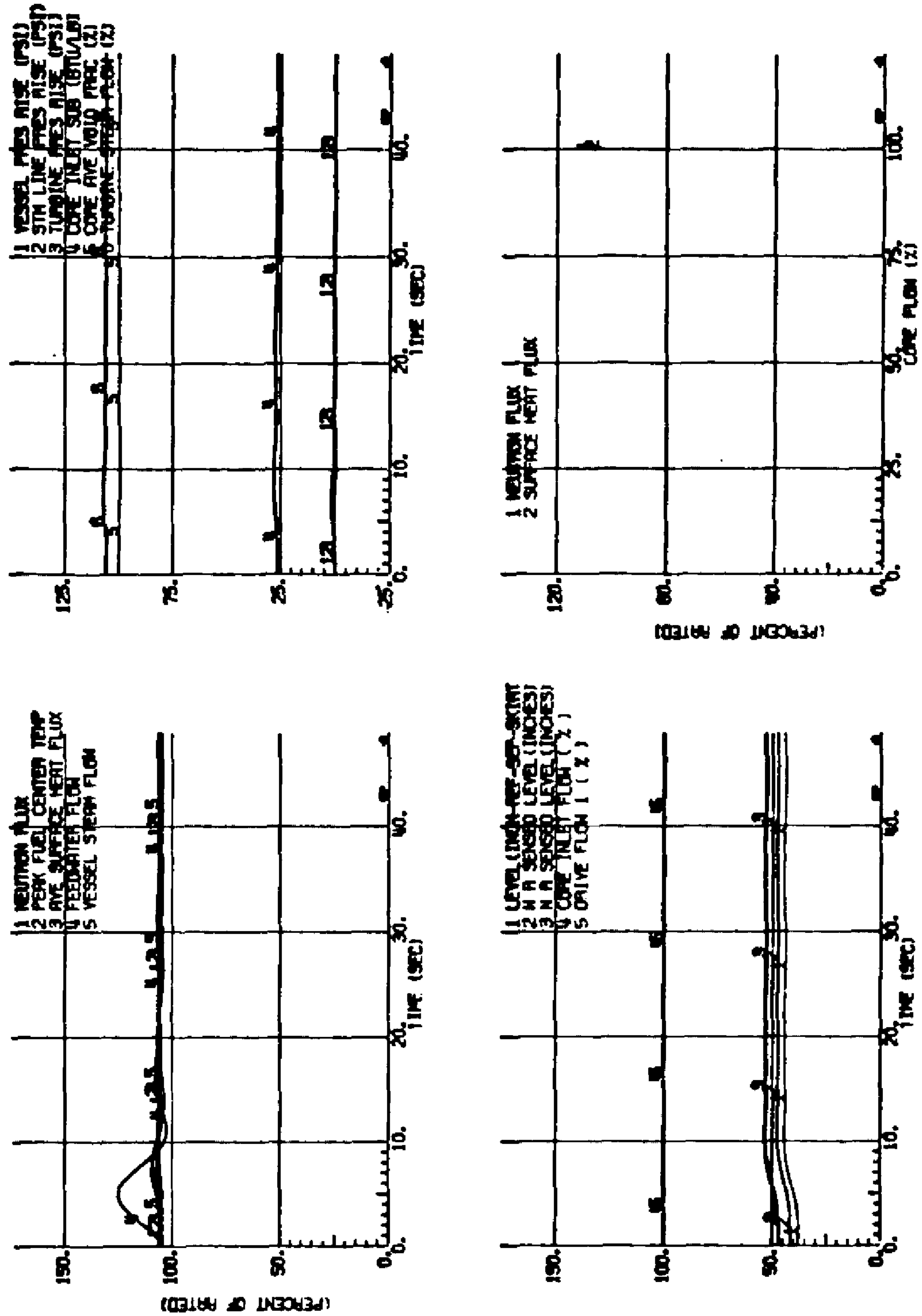
10 PSI PRESSURE REGULATOR
SETPOINT STEP AT 105%
RATED POWER AND 100%
RATED FLOW

FIGURE 4.4-8B, Rev. 47

Auto-Cad Figure Fsar 4_4_8B.dwg



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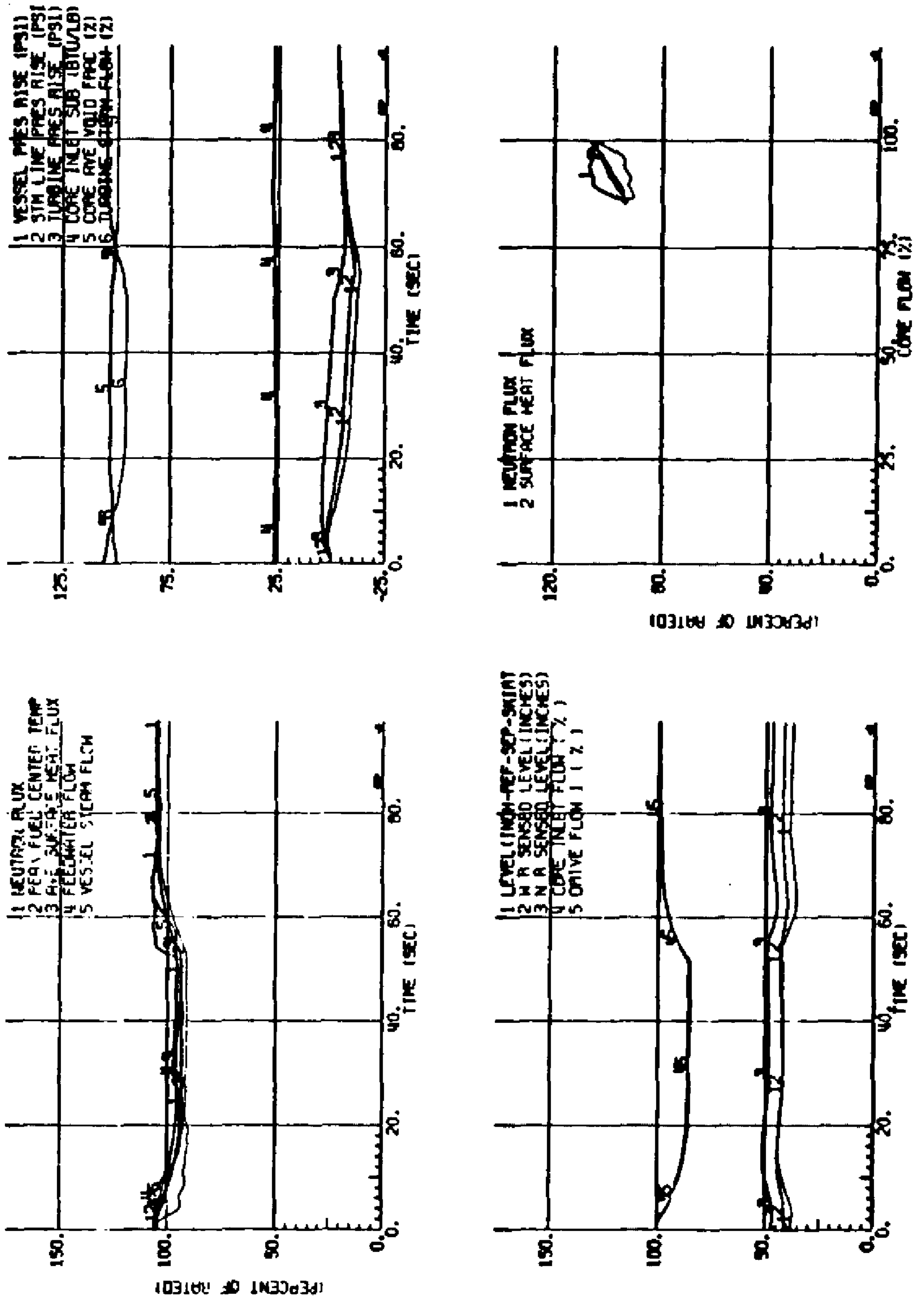
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

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



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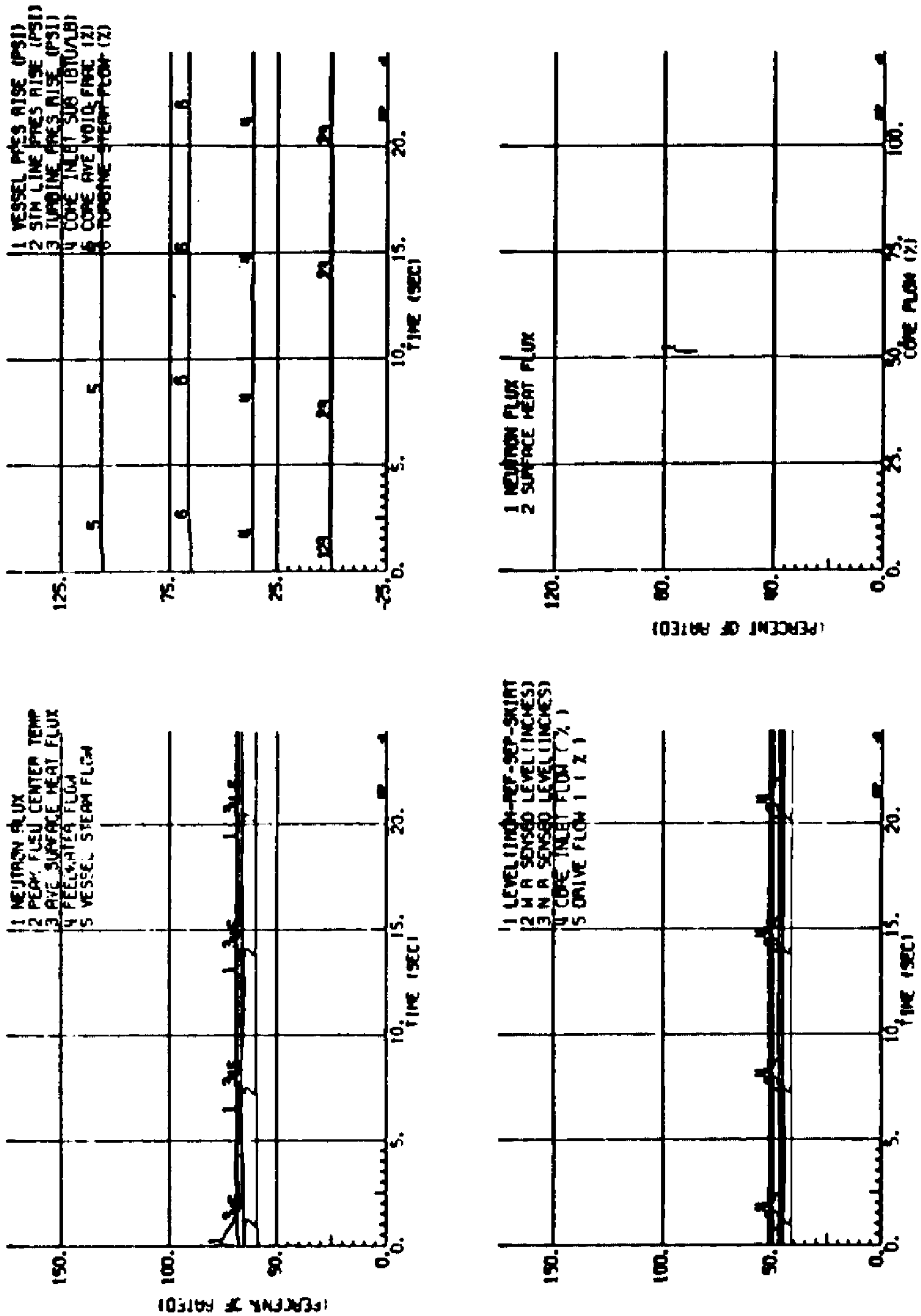
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

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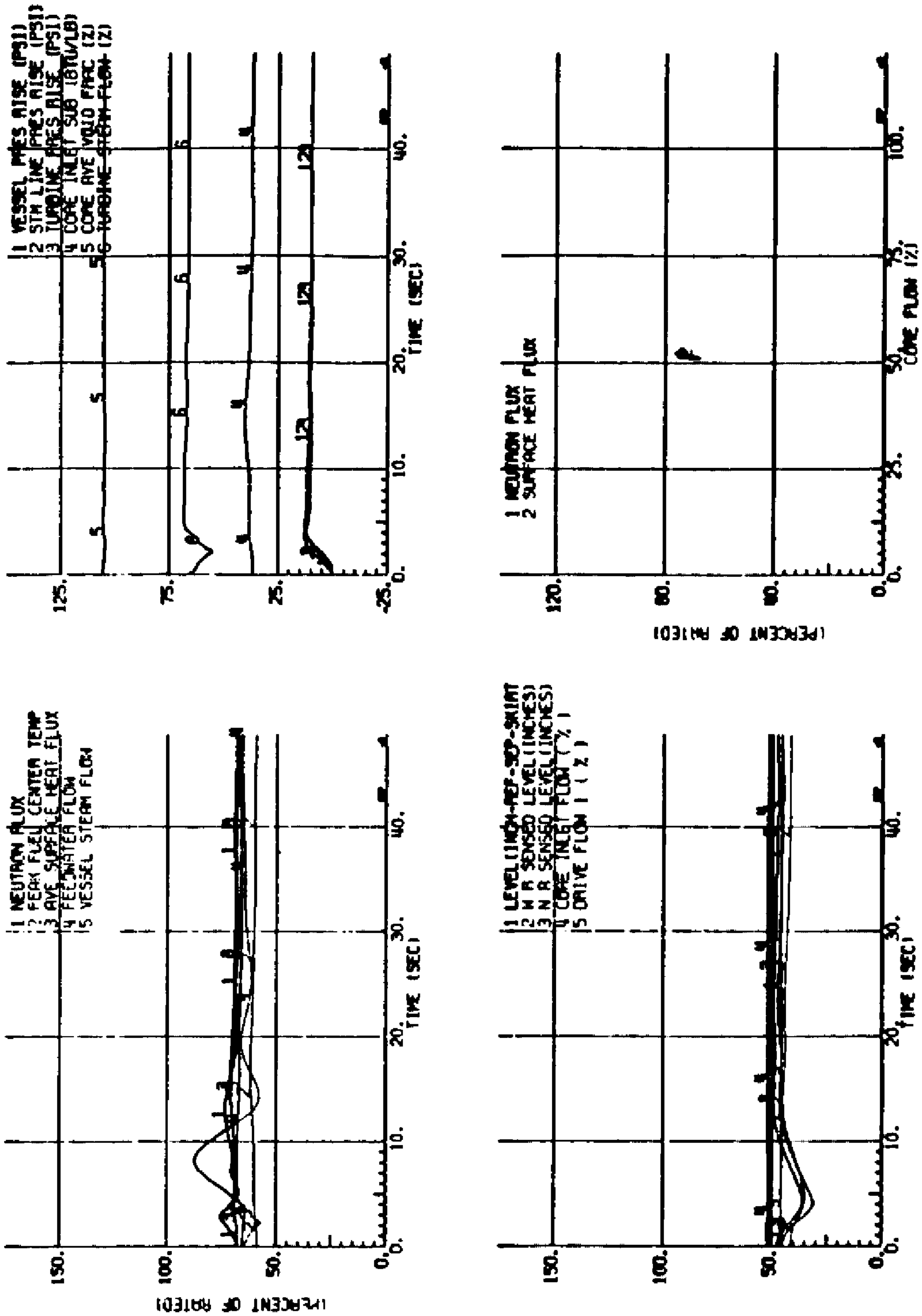
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

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



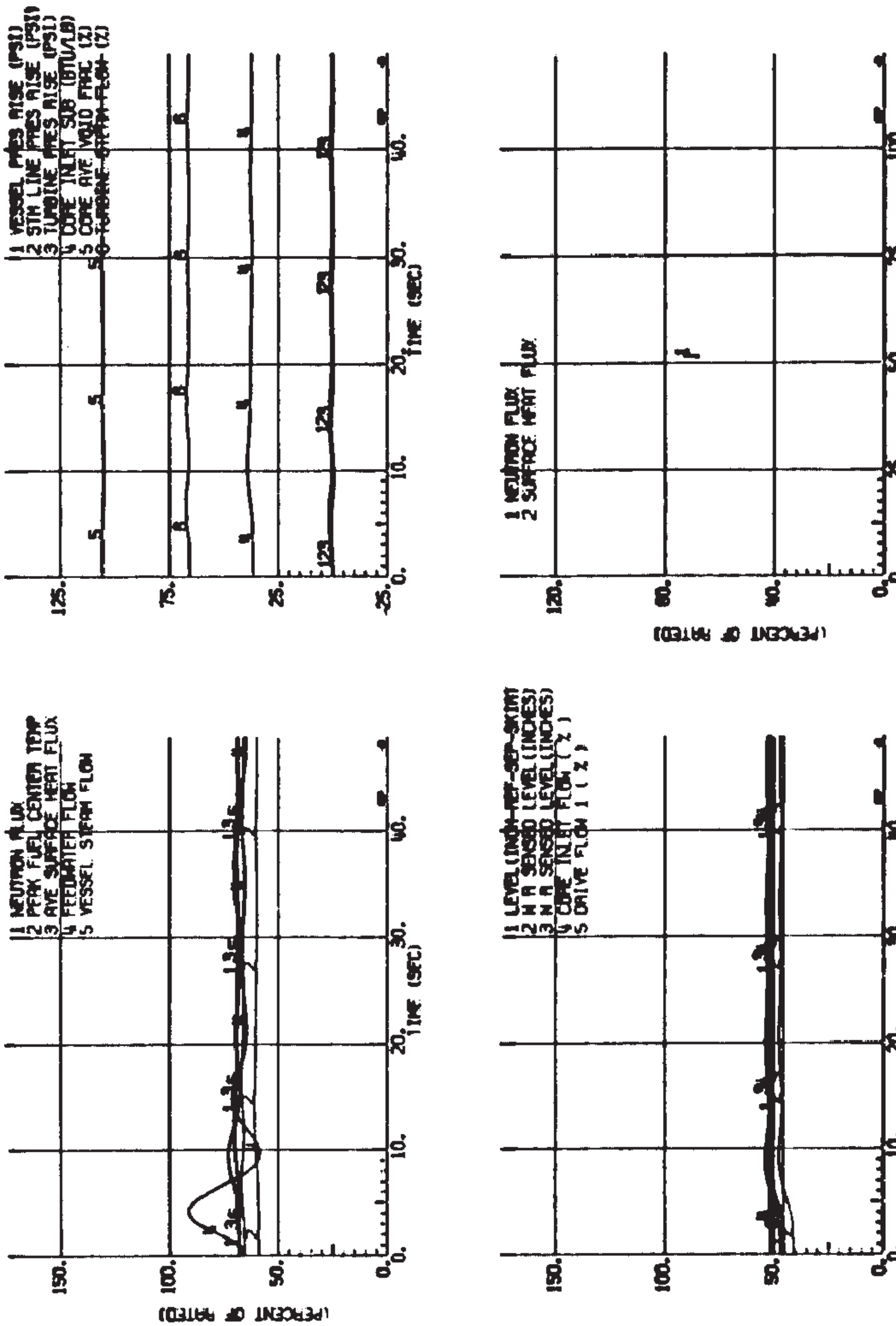
FSAR REV.65

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



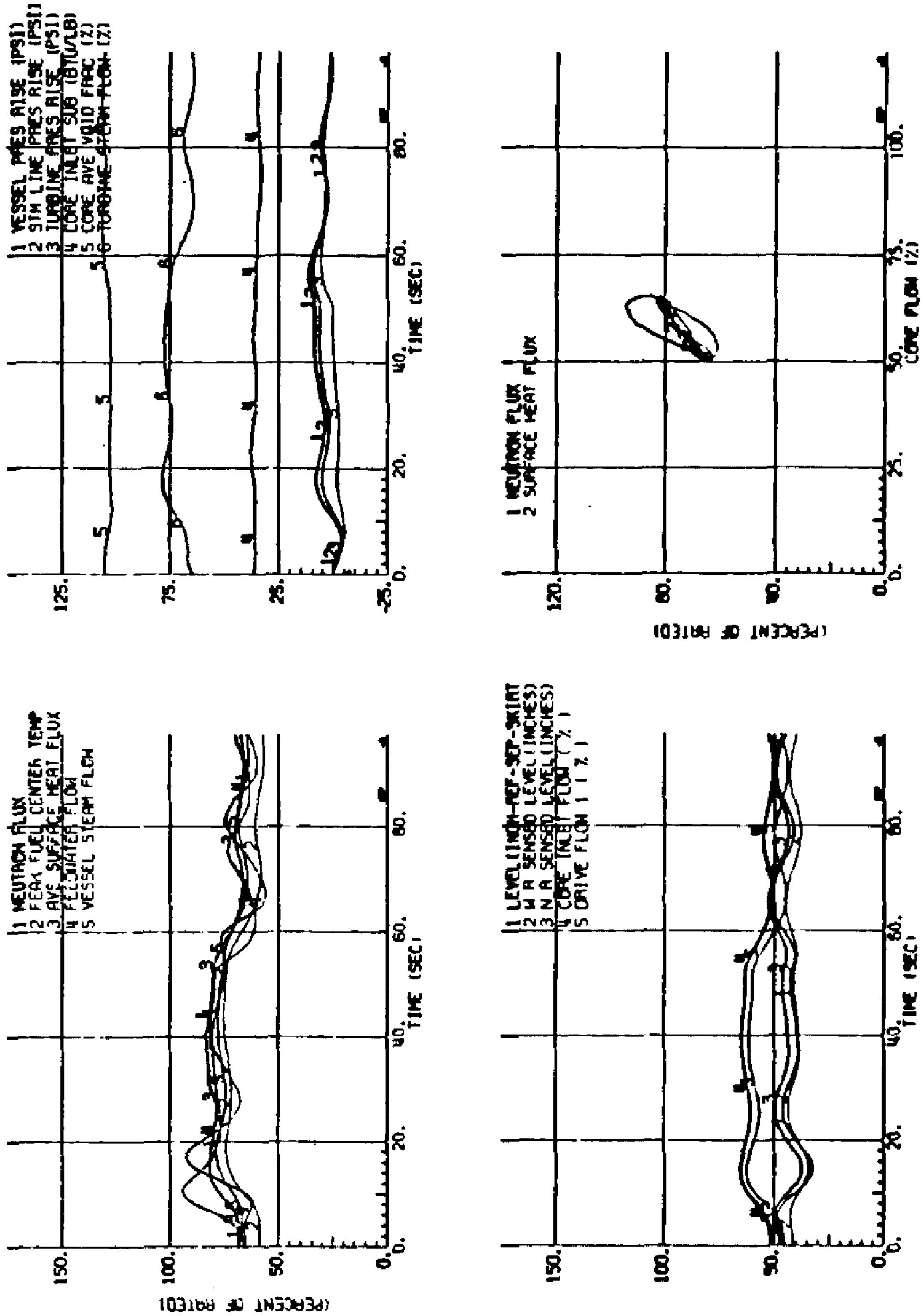
FSAR REV.65

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

4.5 REACTOR MATERIALS

4.5.1 CONTROL ROD SYSTEM STRUCTURAL MATERIALS

4.5.1.1 Material Specifications

The following material listing applies to the control rod drive mechanism supplied for this application. The position indicator, and minor non-structural items are omitted.

(1) Cylinder, Tube and Flange Assembly

Flange	ASME SA 182 Grade F304, F304L
Plugs	ASME SA 182 Grade F304
Cylinder	ASTM A269 Grade TP 304
Outer Tube	ASTM A269 Grade TP 304
Tube Upper	ASME SA 351 Grade CF3
Spacer	ASME SA 351 Grade CF3

(2) Piston Tube Assembly

Piston Tube	ASME SA 249 Grade XM-19
Stud	ASME SA 479 Grade XM-19
Head	ASME SA 182 Grade F304
Ind. Tube	ASME SA 312 Type 316
Cap	ASME SA 182 Grade F3

(3) Drive Assembly

Coupling Spud	Inconel X-750
Index Tube	ASME SA 249 Grade XM-19
Piston Head	Armco 17-4 PH ASTM A564 Grade 630
Piston Coupling	ASME SA 312 Grade TP 304 or ASTM A511 Grade MT 304
Magnet Housing	ASME SA 312 Grade TP 304 or

(4) Collet Assembly

Collet Piston	ASTM A269 Grade TP 304 or ASME SA 312 Grade TP 304
Finger	Inconel X-750
Retainer	ASTM A269 Grade TP 304 o ASTM A511 Grade MT 304
Guide Cap	ASTM A269 Grade TP 304

(5) Miscellaneous Parts

Stop Piston	ASTM A276 Type 304
Connector	ASTM A276 Type 304
O-Ring Spacer	ASME SA 240 Type 304
Nut	ASME SA 193 Grade B8 ASME SA479 XM-19
Barrel	ASTM A269 Grade TP 304 or ASME SA 312 Grade TP 304 or ASME SA 240 Type 304
Collet Spring	Inconel X-750
Ring Flange	ASME SA 182 Grade F304

The materials listed under ASTM specification number are all in the annealed condition (with the exception of the outer tube in the cylinder, tube and flange assembly), and their properties are readily available. The outer tube is approximately 1/8 hard, and has a tensile of 90,000/125,000 psi, yield of 50,000/85,000 psi, and minimum elongation of 25%.

The coupling spud, collet fingers and collet spring are fabricated from Inconel X-750 in the annealed or equalized condition, and heat treated to produce a tensile of 165,000 psi minimum, yield of 105,000 psi minimum and elongation of 20% minimum. The piston head is Armco 17-4 PH (ASTM A564 Grade 630) in condition H 1100, with a tensile of 140,000 psi minimum, yield of 115,000 psi minimum and elongation of 15% minimum.

These are widely used materials, whose properties are well known. The parts are readily accessible for inspection and replacement if necessary.

All materials, including SA 479 Grade XM-19, have been successfully used for years in similar drive mechanisms. Extensive laboratory tests have demonstrated that ASME SA 479 Grade XM-19 is a suitable material and that it is resistant to stress corrosion in a BWR environment.

4.5.1.2 Special Materials

No cold worked austenitic stainless steels with a yield strength greater than 90,000 psi are employed in the Control Rod Drive mechanism. Hardenable martensitic stainless steels are not used. Armco 17-4 PH (ASTM A564 Grade 630 precipitation hardened stainless steel) is used for the piston head. This material is aged to the H-1100 condition to produce resistance to stress corrosion cracking in the BWR environments. Armco 17-4 PH (ASTM A564 Grade 630), (H-1100) has been successfully used for years in BWR drive mechanisms.

4.5.1.3 Processes, Inspections and Tests

All austenitic stainless steel used in the Control Rod Drive mechanisms are solution annealed material with one exception, the outer tube in the cylinder, tube, and flange assembly. See Subsection 4.5.1.1. Proper solution annealing is verified by testing per ASTM-A262 "Recommended Practices for Detecting Susceptibility to Intergranular Attack in Stainless Steels."

Two special processes are employed which subject selected components to temperatures in the sensitization range.

a. The cylinder (cylinder, tube and flange assembly) and the retainer (collet assembly) are hard surfaced with Colmonoy 6.

b. The following components are nitrided to provide a wear resistant surface:

   - Piston tube (piston tube assembly)
   - Index tube (drive line assembly)
   - Collet piston and guide cap (collet assembly)

Colmonoy hard-surfaced components have performed successfully for years in drive mechanisms. Nitrided components have also accumulated many years of BWR service. It is normal practice to remove some Control Rod Drives at each refueling outage. At this time, both the Colmonoy hard surfaced parts and nitrided surfaces are accessible for visual examination. In addition, dye penetrant examinations have been performed on nitrided surfaces of the longest service drives. This inspection program is adequate to detect any incipient defects before they could become serious enough to cause operating problems.

All austenitic stainless steel is purchased in the solution heat treated condition. Welding is performed in accordance with Section IX of the ASME Boiler and Pressure Vessel Code. Heat input for stainless steel welds is restricted to a maximum of 50,000 Joules per inch and interpass temperature to 350°F. Heating above 800°F (except for welding) is prohibited unless the welds are subsequently solution annealed. These controls are employed to avoid severe sensitization. Compliance with Regulatory Guide 1.44 is discussed in Section 3.13.

4.5.1.4 Control of Delta Ferrite Content

All type 308 weld metal is purchased to a specification which requires a minimum of 5% delta ferrite. This amount of ferrite is adequate to prevent any micro-fissuring (hot cracking) in austenitic stainless steel welds.

START HISTORICAL

An extensive test program performed by General Electric Company, with the concurrence of the Regulatory Staff, has demonstrated that controlling weld filler metal ferrite at 5% minimum produces production welds which meet the requirements of Regulatory Guide 1.31. A total of approximately 400 production welds in five BWR plants were measured and all welds met the requirements of the Interim Regulatory Position to Regulatory Guide 1.31.

END HISTORICAL

4.5.1.5 Protection of Materials During Fabrication, Shipping, and Storage

All the Control Rod Drive parts listed above (Subsection 4.5.1.1) are fabricated under a process specification which limits contaminants in cutting, grinding and tapping coolants and lubricants. It also restricts all other processing materials (marking inks, tape etc.) to those which are completely removable by the applied cleaning process. All contaminants are then required to be removed by the appropriate cleaning process prior to any of the following:

(1) Any processing which increases part temperature above 200°F.
(2) Assembly which results in decrease of accessibility for cleaning.
(3) Release of parts for shipment.

The specification for packaging and shipping the Control Rod Drive provides the following:

The drive is rinsed in hot deionized water and dried in preparation for shipment. The ends of the drive are then covered with a vapor tight barrier with desiccant. Packaging is designed to protect the drive and prevent damage to the vapor barrier. The planned storage period considered in the design of the container and packaging is two years. This packaging has been qualified and in use for a number of years.

Periodic audits have indicated satisfactory protection.

The degree of surface cleanliness obtained by these procedures meets the requirements of Regulatory Guide 1.37.

Site or warehouse storage specifications require inside heated storage comparable to level B of ANSI 45.2.2. After the second year, a yearly inspection of 10% of the humidity indicators (packaged with the drives) is required to verify that the units are dry.

4.5.2 REACTOR INTERNAL MATERIALS

4.5.2.1 Material Specifications

Materials used for the Core Support Structure:

Shroud Support - Nickel Chrome Iron Alloy, ASME SB166 or SB168.

Shroud, core plate, top guide, and internal structures welded to these components, ASME SA240, SA182, SA479, SA312, SA249, or SA213 (Type 304 and 304L).

Peripheral fuel supports - SA312 Type 304.

Core plate and top guide studs and nuts. ASME SA479, SA193 Grade B8, SA194 Grade 8, ASTM A276 (all Type 304).

Top guide pins, ASME SA 479 (Type 316 or XM-19), ASTM A276 T304.

Control rod drive housing. ASME SA312 Type 304, SA182 Type 304.

Control rod drive guide tube. ASME SA351 Type CF8, SA358, SA312, SA249 (Type 304).

Orificed fuel support. ASME SA351 Type CF8.

Materials Employed in Other Reactor Internal Structures.

(1) Steam Separator

All materials are Type 304 stainless steel.

Plate, Sheet and Strip	ASTM A240, Type 304
Forgings	ASTM A182, Grade F304
Bars	ASTM A479 Type 304
Pipe	ASTM A312 Grade TP 304
Tube	ASTM A269 Grade TP 304
Castings	ASTM A351 Grade CF8

Replacement materials for the steam separator may be Type 316 stainless steel.

Replacement Steam Dryer

All materials are Type 304, 304L, 316, 316L, stainless steel, CF3, XM-19, or X-750

Plate, Sheet and Strip	ASTM A240, Type 304, 304L, 316, 316L, XM-19
Forgings	ASTM A182, Grade F304, F304L, F316, F316L, FXM-19 ASTM A965, Grade F304, F304L, F316, F316L, XM-19
Bars	ASTM A479, Type 304, 304L, 316, 316L, XM-19
Pipe	ASTM A312, Grade TP304, TP304L, TP316, TP316L ASTM A358, Grade 304, 304L, 316, 316L ASTM A376, Grade TP304, TP316
Castings	ASTM A351, Grade CF3
Bar, Forging	ASTM B637 – UNSN07750
Wire	AMS 5698, 5699G

(2) Jet Pump Assemblies

The components in the Jet Pump Assemblies are a Riser, Inlet, Mixer, Diffuser, Adaptor, and Brackets. Materials used for these components are to the following specifications.

Castings	ASTM A351 Grade CF8
Bars	ASTM A276 Type 304
Bolts	ASTM A193 Grade B8 or B8M
	ASTM A194 Grade B8
Nut	ASTM A240 Type 304 and 304L
Sheet and Plate	ASTM A269 Grade TP 304
Tubing	ASTM A358 Type 304 and
Pipe	ASTM A312 Grade TP 304
Weld Coupling	ASTM A403 Grade WP304
Forgings	ASTM A182 Grade F304
Inconel Forgings	ASTM B166

Materials in the Jet Pump Assemblies which are not Type 304 stainless steel are listed below:

(1) The Inlet Mixer Adaptor casting, the wedge casting, bracket casting adjusting screw, and the Diffuser collar casting are Type 304 hard surfaced with Stellite 6 for slip fit joints.

(2) The Adaptor is a bi-metallic component made by welding a Type 304 forged ring to a forged Inconel 600 ring, made to Specification ASTM B166.

(3) The Inlet-Mixer contains a pin, insert, and beam made of Inconel X-750. The pin and insert are made to General Electric Specification B50YP44A1 and the beam is made to ASTM B637 UNS N07750 to resist IGSCC. The Jet Pump Beam Bolt is SS316L that is nitrided.

(4) Jet Pump Beam Bolt assemblies with rachet-style keepers use a beam, keeper, lock plate, pins, and lock pins made of Alloy X-750 ASTM B-637. Additionally, they use machine screws made of XM-19 stainless steel. They use a bolt that is Type 316L stainless steel.

(5) The Auxiliary Spring Wedge Assemblies installed on the Jet Pumps in those locations necessary to ensure three point contact at the restrainer bracket set screws are made of Inconel X-750 to Specification ASTM B637.

(6) The replacement jet pump Mixer Wedge Assemblies installed on Jet Pumps in those locations necessary to ensure three point contact at the restrainer bracket are made of Inconel X-750 to Specification ASTM B637. The replacement wedges are not stellite coated. The Guide Rod and Keeper Nut are made of Type 316 stainless steel.

(7) Slip Joint Clamps are installed on select jet pumps. The slip joint clamp body is fabricated from solution heat treated ASTM A-182 Grade F XM-19 stainless steel. The adjustable bolt and ratchet lock spring are fabricated from ASME SB-637 or ASTM B-637 UNS N07750 Type 3 (Alloy X-750).

(8) Anti-Vibration Solution (AVS) hardware is installed on select jet pumps. Materials used include XM-19, X-750, 316 stainless steel, and Nitronic 60.

(9) Slip Joint Diffuser Rings are installed on select Unit 1 and Unit 2 jet pumps. The Slip Joint Diffuser Ring is fabricated from solution heat treated 300 series austenitic stainless steel with a maximum of 0.02% carbon, XM-19 (22Cr-13Ni-5Mn) stainless steel, Alloy X-750 (Ni-Cr-Fe), or Nitronic-60 (18Cr-8Ni-4Si-N).

All core support structures are fabricated from ASME and ASTM equivalent specified materials, and designed in accordance with the requirements of ASME Code Section III, Appendix I. The other reactor internals are non-coded, and they are fabricated from ASTM specification materials. Material requirements in the ASTM specifications are identical to requirements in corresponding ASME material specifications. The allowable stress levels specified in ASME Code Section III, Appendix I, are used as a guide in the design of all non-coded internal structures in the BWR.

4.5.2.2 Controls on Welding

Core support structures are fabricated in accordance with requirements of ASME Code Section III, Subsection NG. Other internals are not required to meet ASME Code requirements; however, they are fabricated to the requirements of ASME Section IX.

4.5.2.3 Nondestructive Examination of Wrought Seamless Tubular Products

For core support structures, wrought seamless tubular products were supplied in accordance with applicable ASME material specifications. These specifications require examination of the tubular product by radiographic and/or ultrasonic methods according to paragraph NG-2550 of ASME Code Section III. In addition, the specification for tubular products employed for CRD housings external to the RPV meet requirements of paragraph NB-2550. Compliance with Regulatory Guide 1.66 is discussed in Section 3.13.

Other internals are non-coded, and wrought seamless tubular products were supplied in accordance with the applicable ASTM material specifications. These specifications require a hydrostatic test on each length of tubing.

4.5.2.4 Fabrication and Processing of Austenitic Stainless Steel - Regulatory Guide Conformance

Regulatory Guide 1.31, Control of Stainless Steel Welding

All austenitic stainless steel weld filler materials were supplied with a minimum of 5% delta ferrite. This amount of ferrite is considered adequate to prevent micro-fissuring in austenitic stainless steel welds.

START HISTORICAL

An extensive test program performed by General Electric Company, with the concurrence of the Regulatory Staff, has demonstrated that controlling weld filler metal ferrite at 5% minimum produces production welds which meet the requirements of Regulatory Guide 1.31, "Control of Stainless Steel Welding". A total of approximately 400 production welds in five BWR plants were measured and all welds met the requirements of the Interim Regulatory Position to Regulatory Guide 1.31.

END HISTORICAL

Regulatory Guide 1.34, Control of Electroslag Weld Properties

Electroslag welding is not employed for any reactor internals.

Regulatory Guide 1.36, Non-metallic Thermal Insulation for Austenitic Stainless Steel

Non-metallic thermal insulation is not employed for any components in the reactor vessel.

Regulatory Guide 1.44, Control of the Use of Sensitized Stainless Steel

All wrought austenitic stainless steel was purchased in the solution heat treated condition. Heating above 800°F was prohibited (except for welding) unless the stainless steel was subsequently solution annealed. Purchase specifications restricted the maximum weld heat input to 110,000 Joules per inch, and the weld interpass temperature to 350°F maximum. Welding was performed in accordance with Section IX of the ASME Boiler and Pressure Vessel Code. These controls were employed to avoid severe sensitization. Compliance with Regulatory Guide 1.44 is discussed in Section 3.13.

Regulatory Guide 1.71, Welder Qualification for Areas of Limited Accessibility

All weld procedures and welders were qualified in accordance with ASME Section IX and Section III. Areas of restricted access required qualification test assemblies welded under simulated access conditions. Prior to performing welding on any assembly joints requiring a mockup, qualified welders were required to perform one weld joint or a minimum of 12 inches of weld mockup under simulated space, accessibility, and adjacent component configuration. The mockup joint was welded in strict accordance with the approved welding procedure. Nondestructive examination of the mockup was by radiography or by sectioning and subject to established acceptance critieria.

4.5.2.5 Contamination, Protection, and Cleaning of Austenitic Stainless Steel

Exposure to contaminant was avoided by carefully controlling all cleaning and processing materials which contact stainless steel during manufacture and construction. Any inadvertent surface contamination was removed to avoid potential detrimental effects.

Special care was exercised to insure removal of surface contaminants prior to any heating operation. Water quality for rinsing, flushing, and testing was controlled and monitored.

The degree of cleanliness obtained by these procedures meets the requirements of Regulatory Guide 1.37.

4.5.3 Control Rod Drive Housing Supports

The American Institute of Steel Construction (AISC) Manual of Steel Construction, "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings," was used in designing the CRD housing support system. However, to provide a structure that absorbs as much energy as practical without yielding, the allowable tension and bending stresses used were 90% of yield and the shear stress used was 60% of yield. These design stresses are 1.5 times the AISC allowable stresses (60% and 40% of yield, respectively).

For purposes of mechanical design, the postulated failure resulting in the highest forces is an instantaneous circumferential separation of the CRD housing from the reactor vessel, with the reactor at an operating pressure of 1086 psig (at the bottom of the vessel) acting on the area of the separated housing. The weight of the separated housing, control rod drive, and blade, plus the pressure of 1086 psig acting on the area of the separated housing, gives a force of approximately 32,000 lb. This force is used to calculate the impact force, conservatively assuming that the housing travels through a 1-in. gap before it contacts the supports. The impact force (109,000 lb) is then treated as a static load in design. The CRD housing supports are designed as Category I (seismic) equipment in accordance with Section 3.2.

All CRD housing support subassemblies are fabricated of ASTM-A-36 structural steel, except for the following items:

	Material
Grid	ASTM-A-441
Support Bars	AISI MT1015 and A36
Spring Housings	ASTM A513 Type 5
Spacers	AISI MT1015
Disc springs	Schnorr, Type BS-125-71-8
Hex bolts and nuts	ASTM-A-307
6 x 4 x 3/8 tubes	ASTM-A-500 Grade B

4.6 FUNCTIONAL DESIGN OF REACTIVITY CONTROL SYSTEMS

Functional design of the control rod drive system (CRD) is discussed below. Functional design of the recirculation flow control system and standby liquid control system are described in subsections 5.4.1 and 9.3.5, respectively.

4.6.1 Information for CRDS

4.6.1.1 Control Rod Drive System Design

4.6.1.1.1 Design Bases

4.6.1.1.1.1 General Design Bases

4.6.1.1.1.1.1 Safety Design Bases

The control rod drive mechanical system shall meet the following safety design bases:

(1) Design shall provide for a sufficiently rapid control rod insertion that no fuel damage results from any abnormal operating transient.

(2) Design shall include positioning devices, each of which individually supports and positions a control rod.

(3) Each positioning device shall:

   a. Prevent its control rod from initiating withdrawal as a result of a single malfunction.

   b. Be individually operated so that a failure in one positioning device does not affect the operation of any other positioning device.

   c. Be individually energized when rapid control rod insertion (scram) is signaled so that failure of power sources external to the positioning device does not prevent other positioning devices' control rods from being inserted.

4.6.1.1.1.1.2 Power Generation Design Basis

The control rod system drive design shall provide for positioning the control rods to control power generation in the core.

4.6.1.1.2 Description

The control rod drive (CRD) system controls gross changes in core reactivity by incrementally positioning neutron absorbing control rods within the reactor core in response to manual control signals. It is also required to quickly shut down the reactor (scram) in emergency situations by rapidly inserting withdrawn control rods into the core in response to a manual or automatic signal. The control rod drive system consists of locking piston control rod drive mechanisms, and the CRD hydraulic system (including power supply and regulation, hydraulic control units, interconnecting piping, instrumentation and electrical controls).

4.6.1.1.2.1 Control Rod Drive Mechanisms

The CRD mechanism (drive) used for positioning the control rod in the reactor core is a double-acting, mechanically latched, hydraulic cylinder using water as its operating fluid. (See Figure 4.6-1, 4.6-2, 4.6-3, and 4.6-4.) The individual drives are mounted on the bottom head of the reactor pressure vessel.

The drives do not interfere with refueling and are operative even when the head is removed from the reactor vessel. The drives are also readily accessible for inspection and servicing. The bottom location makes maximum utilization of the water in the reactor as a neutron shield and gives the least possible neutron exposure to the drive components. Using water from the condensate system as the operating fluid eliminates the need for special hydraulic fluid. Drives are able to utilize simple piston seals whose leakage does not contaminate the reactor water but provides cooling for the drive mechanisms and their seals.

The drives are capable of inserting or withdrawing a control rod at a slow, controlled rate, as well as providing rapid insertion when required. A mechanism on the drive locks the control rod at 6-inch increments of stroke over the length of the core.

A coupling spud at the top end of the drive index tube (piston rod) engages and locks into a mating socket at the base of the control rod. The weight of the control rod is sufficient to engage and lock this coupling. Once locked, the drive and rod form an integral unit that must be manually unlocked by specific procedures before the components can be separated.

The drive holds its control rod in distinct latch positions until the hydraulic system actuates movement to a new position. Withdrawal of each rod is limited by the seating of the rod in its guide tube. Withdrawal beyond this position to the over-travel limit can be accomplished only if the rod and drive are uncoupled. Withdrawal to the over-travel limit is annunciated by an alarm.

The individual rod indicators, grouped in one control panel display, correspond to relative rod locations in the core. A separate, smaller hardwired display is located on the standby information panel. A presentation is available on the unit operating benchboard. This latter display presents the position of the control rod selected for movement as well as the other rods in the affected rod group.

For display purposes the control rods are considered in groups of four adjacent rods centered around a common core volume. Each group is monitored by four LPRM strings (see Subsection 7.6.1.5). Rod groups at the periphery of the core may have less than four rods. The small rod display shows the positions, in digital form, of the rods in the group to which the selected rod belongs. A white light indicates which of the four rods is the one selected for movement.

4.6.1.1.2.2 Drive Components

Figure 4.6-2 illustrates the operating principle of a drive. Figures 4.6-3 and 4.6-4 illustrate the drive in more detail. The main components of the drive and their functions are described below.

4.6.1.1.2.2.1 Drive Piston

The drive piston is mounted at the lower end of the index tube. This tube functions as a piston rod. The drive piston and index tube make up the main moving assembly in the drive. The drive piston operates between positive end stops, with a hydraulic cushion provided at the upper end only. The piston has both inside and outside seal rings and operates in an annular space between an inner cylinder (fixed piston tube) and an outer cylinder (drive cylinder). Because the type of inner seal used is effective in only one direction, the lower sets of seal rings are mounted with one set sealing in each direction.

A pair of nonmetallic bushings prevents metal-to-metal contact between the piston assembly and the inner cylinder surface. The outer piston rings are segmented step-cut seals with expander springs holding the segments against the cylinder wall. A pair of split bushings on the outside of the piston prevents piston contact with the cylinder wall. The effective piston area for downtravel, or withdrawal, is approximately 1.2 sq. in. versus 4.1 sq. in. for uptravel, or insertion. This difference in driving area tends to balance the control rod weight and assures a higher force for insertion than for withdrawal.

4.6.1.1.2.2.2 Index Tube

The index tube is a long hollow shaft made of nitrided stainless steel. Circumferential locking grooves, spaced every 6 inches along the outer surface, transmit the weight of the control rod to the collet assembly.

4.6.1.1.2.2.3 Collet Assembly

The collet assembly serves as the index tube locking mechanism. It is located in the upper part of the drive unit. This assembly prevents the index tube from accidentally moving downward. The assembly consists of the collet fingers, a return spring, a guide cap, a collet housing (part of the cylinder, tube, and flange), and the collet piston.

Locking is accomplished by fingers mounted on the collet piston at the top of the drive cylinder. In the locked or latched position the fingers engage a locking groove in the index tube.

The collet piston is normally held in the latched position by a force of approximately 150 lb. supplied by a spring. Metal piston rings are used to seal the collet piston from reactor vessel pressure. The collet assembly will not unlatch until the collet fingers are unloaded by a short, automatically sequenced, drive-in signal. A pressure, approximately 180 psi above reactor vessel pressure, must then be applied to the collet piston to overcome spring force, slide the collet up against the conical surface in the guide cap, and spread the fingers out so they do not engage a locking groove.

A guide cap is fixed in the upper end of the drive assembly. This member provides the unlocking cam surface for the collet fingers and serves as the upper bushing for the index tube.

If reactor water is used during a scram to supplement accumulator pressure, it is drawn through a filter on the guide cap.

4.6.1.1.2.2.4 Piston Tube

The piston tube is an inner cylinder, or column, extending upward inside the drive piston and index tube. The piston tube is fixed to the bottom flange of the drive and remains stationary. Water is brought to the upper side of the drive piston through this tube. A buffer shaft, at the upper end of the piston tube, supports the stop piston and buffer components.

4.6.1.1.2.2.5 Stop Piston

A stationary piston, called the stop piston, is mounted on the upper end of the piston tube. This piston provides the seal between reactor vessel pressure and the space above the drive piston. It also functions as a positive end stop at the upper limit of control rod travel. Piston rings and bushings, similar to those used on the drive piston, are mounted on the upper portion of the stop piston. The lower portion of the stop piston forms a thin-walled cylinder containing the buffer piston, its metal seal ring, and the buffer piston return spring. As the drive piston reaches the upper end of the scram stroke it strikes the buffer piston. A series of orifices in the buffer shaft provides a progressive water shutoff to cushion the buffer piston as it is driven to its limit of travel. The high pressures generated in the buffer are confined to the cylinder portion of the stop piston, and are not applied to the stop s steel tube. The switches are actuated by a ring magnet located at the bottom of the drive piston.

The drive piston, piston tube, and indicator tube are all of nonmagnetic stainless steel, allowing the individual switches to be operated by the magnet as the piston passes. One switch is located at each position corresponding to an index tube groove and one switch is located at the midpoint between each latching point. Thus, indication is provided for each 3 inches of travel. Duplicate switches are provided for the full-in and full-out positions. Redundant overtravel switches are located at a position below the normal full-out position. Because the limit of downtravel is normally provided by the control rod itself as it reaches the backseat position, the drive can pass this position and actuate the overtravel switches only if it is uncoupled from its control rod. A convenient means is thus provided to verify that the drive and control rod are coupled after installation of a drive or at any time during plant operation.

4.6.1.1.2.2.6 Flange and Cylinder Assembly

A flange and cylinder assembly is made up of a heavy flange welded to the drive cylinder. A sealing surface on the upper face of this flange forms the seal to the drive housing flange. The seals contain reactor pressure and the two hydraulic control pressures. Teflon coated, stainless steel rings are used for these seals. The drive flange contains the integral ball, or two-way, check (ball-shuttle) valve. This valve directs either the reactor vessel pressure or the driving pressure, whichever is higher, to the underside of the drive piston. Reactor vessel pressure is admitted to this valve from the annular space between the drive and drive housing through passages in the flange.

Water used to operate the collet piston passes between the outer tube and the cylinder tube. The inside of the cylinder tube is honed to provide the surface required for the drive piston seals.

Both the cylinder tube and outer tube are welded to the drive flange. The upper ends of these tubes have a sliding fit to allow for differential expansion.

The upper end of the index tube is threaded to receive a coupling spud. The coupling (see Figure 4.6-1) accommodates a small amount of angular misalignment between the drive and the control rod. Six spring fingers allow the coupling spud to enter the mating socket on the control rod. A plug then enters the spud and prevents uncoupling.

Two means of uncoupling are provided. With the reactor vessel head removed, the lock plug can be raised against the spring force of approximately 50 pounds by a rod extending up through the center of the control rod to an unlocking handle located above the control rod velocity limiter. The control rod, with the lock plug raised, can then be lifted from the drive.

4.6.1.1.2.2.7 Lock Plug

The lock plug can also be pushed up from below, if it is desired to uncouple a drive without removing the reactor pressure vessel head for access. In this case, the central portion of the drive mechanism is pushed up against the uncoupling rod assembly, which raises the lock plug and allows the coupling spud to disengage the socket as the drive piston and index tube are driven down.

The control rod is heavy enough to force the spud fingers to enter the socket and push the lock plug up, allowing the spud to enter the socket completely and the plug to snap back into place. Therefore, the drive can be coupled to the control rod using only the weight of the control rod. However, with the lock plug in place, a force in excess of 50,000 lb. is required to pull the coupling apart.

4.6.1.1.2.3 Materials of Construction

Factors that determine the choice of construction materials are discussed in the following subsections.

4.6.1.1.2.3.1 Index Tube

The index tube must withstand the locking and unlocking action of the collet fingers. A compatible bearing combination must be provided that is able to withstand moderate misalignment forces. The reactor environment limits the choice of materials suitable for corrosion resistance. The column and tensile loads can be satisfied by an annealed, single phase, nitrogen strengthened, austenitic stainless steel. The wear and bearing requirements are provided by Malcomizing the complete tube. To obtain suitable corrosion resistance, a carefully controlled process of surface preparation is employed.

4.6.1.1.2.3.2 Coupling Spud

The coupling spud is made of Inconel 750 that is aged for maximum physical strength and the required corrosion resistance. Because misalignment tends to cause chafing in the semispherical contact area, the part is protected by a thin chromium plating (Electrolyzed). This plating also prevents galling of the threads attaching the coupling spud to the index tube.

4.6.1.1.2.3.3 Collet Fingers

Inconel 750 is used for the collet fingers, which must function as leaf springs when cammed open to the unlocked position. Colmonoy 6 hard facing provides a long wearing surface, adequate for design life, to the area contacting the index tube and unlocking cam surface of the guide cap.

4.6.1.1.2.3.4 Seals and Bushings

Graphite Composite is selected for seals and bushings on the drive piston and stop piston. The material is inert and has a low friction coefficient when water-lubricated. Because some loss of Graphitar strength is experienced at higher temperatures, the drive is supplied with cooling water to hold temperatures below 250°F. The Graphitar is relatively soft, which is advantageous when an occasional particle of foreign matter reaches a seal. The resulting scratches in the seal reduce sealing efficiency until worn smooth, but the drive design can tolerate considerable water leakage past the seals into the reactor vessel.

4.6.1.1.2.3.5 Summary

All drive components exposed to reactor vessel water are made of austenitic stainless steel except the following:

(1) Seals and bushings on the drive piston and stop piston are Graphite Composite.

(2) All springs and members requiring spring action (collet fingers, coupling spud, and spring washers) are made of Inconel-750.

(3) The ball check valve is a Haynes Stellite cobalt-base alloy.

(4) Elastomeric O-ring seals are ethylene propylene.

(5) Metal piston rings are Haynes 25 alloy.

(6) Certain wear surfaces are hard-faced with Colmonoy 6.

(7) Nitriding by a proprietary new Malcomizing process and chromium plating are used in certain areas where resistance to abrasion is necessary.

(8) The drive piston head is made of Armco 17-4Ph.

Pressure-containing portions of the drives are designed and fabricated in accordance with requirements of Section III of the ASME Boiler and Pressure Vessel Code.

4.6.1.1.2.4 Control Rod Drive Hydraulic System

The control rod drive hydraulic system (Dwgs. M-146, Sh. 1 and M-147, Sh. 1) supplies and controls the pressure and flow to and from the drives through hydraulic control units (HCU). The water discharged from the drives during a scram flows through the HCUs to the scram discharge volume. The water discharged from a drive during a normal control rod positioning operation flows through the HCU, the exhaust header, through the other HCUs to combine with the cooling water flow at the CRD's, and into the reactor vessel. There are as many HCUs as the number of control rod drives.

4.6.1.1.2.4.1 Hydraulic Requirements

The CRD hydraulic system design is shown in Dwgs. M-146, Sh. 1, M-147, Sh. 1, and M1-C12-85, Sh. 1. The hydraulic requirements, identified by the function they perform, are as follows:

(1) An accumulator hydraulic charging pressure of approximately 1400 to 1500 psig is required. Flow to the accumulators is required only during scram reset or system startup.

(2) Drive pressure of approximately 250 psi above reactor vessel pressure is required. A flow rate of approximately 4 gpm to insert a control rod and 2 gpm to withdraw a control rod is required.

(3) Cooling water to the drives is required at approximately 30 psi above reactor vessel pressure and at a flow rate of 0.20 to 0.34 gpm per drive unit. (Cooling water to a drive can be interrupted for short periods without damaging the drive.)

(4) The scram discharge volume is sized to receive and contain all the water discharged by the drives during a scram; a minimum volume of 3.34 gal. per drive is required.

4.6.1.1.2.4.2 System Description

The CRD hydraulic systems provide the required functions with the pumps, filter, valves, instrumentation, and piping shown in Dwgs. M-146, Sh. 1 and M-147, Sh. 1 and described in the following subsection.

Duplicate components are included, where necessary, to assure continuous system operation if an in-service component requires maintenance.

Leakage inside containment from the CRD Hydraulic System can be detected by the leakage detection system described in Subsection 5.2.5. Again referencing the CRD Hydraulic System, Dwgs. M-146, Sh. 1, M-147, Sh. 1 and M1-C12-85, Sh. 1, it may be seen that the CRD system piping which is downstream of the drivewater pumps is maintained above vessel pressure. Thus, any leakage from the CRD system between the drive water pumps and the HCU will be the relatively clean water, low temperature fluid of the CRD Hydraulic System. If the leakage were to be so large as to cause a significant pressure decrease in the CRD system piping there could be flow from the primary system out the leakage path. If the leak occurred in the piping between the pumps and the HCU, an additional failure of a hydraulic control unit check valve would be necessary for there to be a loss of primary coolant. In any case, a large leak of the CRD system fluid would cause one or more of the following depending on the location of the leak: a decrease in scram accumulator pressure (which alarms an annunciator); or a decrease in drive water pressure or cooling water flow (both of which are displayed on the control room panel).

In the case of leakage from the primary system, there would be reverse flow from the reactor pressure vessel to the drives. This induction of hot primary system coolant into the drive would cause the drive to heat and alarm a high temperature annunciator.

4.6.1.1.2.4.2.1 Supply Pump

One supply pump pressurizes the system with water from the condensate system. One spare pump is provided for standby. A discharge check valve prevents backflow through the nonoperating pump. A portion of the pump discharge flow is diverted through a minimum flow bypass line to the condensate storage tank. This flow is controlled by an orifice and is sufficient to prevent immediate pump damage if the pump discharge is inadvertently closed.

Condensate water is processed by two filters in the system. The pump suction filter is a disposable element type with a 25-micron absolute rating. The drive water filter down-stream of the pump is a replaceable element type with a 50-micron absolute rating. Differential pressure indicators and control room alarms monitor the filter elements as they collect foreign materials.

4.6.1.1.2.4.2.2 Accumulator Charging Pressure

Initially, the accumulator is precharged to 575 psig nominal at 70°F (580 psig max. at 70°F) with nitrogen.

Accumulator charging pressure is established by the discharge pressure of the system supply pump. During scram the scram inlet (and outlet) valves open and permit the stored energy in the accumulators to discharge into the drives. The resulting pressure decrease in the charging water header allows the CRD supply pump to "run out" (i.e., flow rate to increase substantially) into the control rod drives via the charging water header. The flow sensing system upstream of the accumulator charging header detects high flow and decreases flow returning to the RPV. This action maintains increased flow through the charging water header.

Pressure in the charging header is monitored in the control room with a pressure indicator and low pressure alarm.

During normal operation the flow control valve maintains a constant system flow rate. This flow is used for drive flow, drive cooling, and system stability.

4.6.1.1.2.4.2.3 Drive Water Pressure

Drive water pressure required in the drive header is maintained by the pressure control valve, which is manually adjusted from the control room.

A flow rate of approximately 6 gpm (the sum of the flow rate required to insert and withdraw a control rod) normally bypasses the drive water pressure control station through two solenoid operated stabilizing valves (arranged in parallel). The flow through one stabilizing valve equals the drive insert flow; that of the other stabilizing valve equals the drive withdrawal flow. When operating a drive, the required flow is diverted to that drive by closing the appropriate stabilizing valve. Thus, flow through the drive pressure control valve is always constant.

Flow indicators in the drive water header and in the line downstream from the stabilizing valves allow the flow rate through the stabilizing valves to be adjusted when necessary. Differential pressure between the reactor vessel and the drive pressure stage is indicated in the control room.

4.6.1.1.2.4.2.4 Cooling Water Header

All water passing through the pressure control valve and the stabilizing valves is routed to the reactor via the cooling water header. Without flow in the drive and charging water headers the cooling water flow is equal to the flow passing through the flow control valve.

The flow through the flow control valve is virtually constant. Therefore, once adjusted, the drive pressure control valve can maintain the required pressure independent of reactor pressure. Changes in setting of the pressure control valve is required only to adjust for changes in the cooling requirements of the drives, as their seal characteristics change with time. A flow indicator in the control room monitors cooling water flow. A differential pressure indicator in the control room indicates the difference between reactor vessel pressure and drive cooling water pressure. Although the drives can function without cooling water, seal life is shortened by long-term exposure to reactor temperatures. The temperature of each drive is recorded in the reactor building, and excessive temperatures are annunciated in the control room.

4.6.1.1.2.4.2.5 Scram Discharge Volume

The scram discharge volume consists of header piping which connects to each HCU and drains into an instrument volume. The header piping is sized to receive and contain all the water discharged by the drives during a scram, independent of the instrument volume.

Each of the two sets of headers has its own directly connected scram discharge instrument volume (SDIV) attached to the low point of the header piping. The large diameter pipe of the instrument volume thus serves as a vertical extension of the SDV (though no credit is taken for it in determining SDV requirements).

During normal plant operation the scram discharge volume is empty, and vented to atmosphere through its open vent and drain valves. When a scram occurs, upon a signal from the safety circuit these vent and drain valves are closed to conserve reactor water. Redundant vent and drain valves are provided to assure against loss of reactor coolant from the SDV following a scram. Lights in the control room indicate the position of these valves.
During a scram, the scram discharge volume partly fills with water discharged from above the drive pistons. After scram is completed, the control rod drive seal leakage from the reactor continues to flow into the scram discharge volume until the discharge volume pressure equals the reactor vessel pressure. A check valve in each HCU prevents reverse flow from the scram discharge header volume to the drive. When the initial scram signal is cleared from the reactor protection system, the scram discharge volume isolation signal is overridden with a keylock override switch, and the scram discharge volume is drained and returned to atmospheric pressure.

Remote manual switches in the pilot valve solenoid circuits allow the discharge volume vent and drain valves to be tested without disturbing the reactor protection system. Closing the scram discharge volume valves allows the outlet scram valve seats to be leak-tested by timing the accumulation of leakage inside the scram discharge volume.

Twelve liquid-level instruments connected to the instrument volume, monitor the volume for abnormal water level. They are set at three different levels. Two level switches are set at the lowest level. These level switches actuate to indicate that the volume is not completely empty during post-scram draining or to indicate that the volume starts to fill through leakage accumulation at other times during reactor operation. Two level switches are set at the second level. These level switches produce a rod withdrawal block to prevent further withdrawal of any control rod, when leakage accumulates to half the capacity of the instrument volume. The remaining 8 instruments (see Subsection 7.2.1.1.4.2(g)) are interconnected with the trip channels of the Reactor Protection System (RPS) and will initiate a reactor scram should water accumulation fill the instrument volume.

To assure more reliable and safer operation of the Scram Discharge Volume System, the following modifications have been made:

1. Added redundant vent and drain valves to ensure that an uncontrolled loss of reactor coolant would not result in the event of a single active failure.

2. Redundant and diverse sensors are used such that no single component failure or service condition will prevent scram, alarm and rod block functions.

3. Instrument piping has been designed to minimize the transient hydrodynamic effects. Instrumentation taps are provided on the vertical scram discharge instrument volume.

4. The scram discharge volume vent line is a dedicated, non-submerged line routed to the Reactor Building sump. A vacuum breaker is installed on the high point of the vent line and will open on a differential pressure of no greater than five inches of water.

5. The air-operated vent and drain valves will close under loss of air; valve position is provided in the main control room.

6. System piping geometry is designed such that the system drains continuously during normal plant operation. The drain line is a dedicated line routed to the Reactor Building sump.

7. The level instrumentation is designed to be maintained, tested and calibrated during plant operation without causing a scram.

4.6.1.1.2.4.3 Hydraulic Control Units

Each hydraulic control unit (HCU) furnishes pressurized water, on signal, to a drive unit. The drive then positions its control rod as required. Operation of the electrical system that supplies scram and normal control rod positioning signals to the HCU is described in Subsection 7.7.1.2. The basic components in each HCU are manual, pneumatic, and electrical valves; an accumulator; related piping; electrical connections; filters; and instrumentation (see Dwgs. M-147, Sh. 1, M1-C12-85, Sh. 1, and Figure 4.6-7). The components and their functions are described in the following paragraphs.

4.6.1.1.2.4.3.1 Insert Drive Valve

The insert drive valve is solenoid-operated and opens on an insert signal. The valve supplies drive water to the bottom side of the main drive piston.

4.6.1.1.2.4.3.2 Insert Exhaust Valve

The insert exhaust solenoid valve also opens on an insert signal. The valve discharges water from above the drive piston to the exhaust water header.

4.6.1.1.2.4.3.3 Withdraw Drive Valve

The withdraw drive valve is solenoid-operated and opens on a withdraw signal. The valve supplies drive water to the top of the drive piston.

4.6.1.1.2.4.3.4 Withdraw Exhaust Valve

The solenoid-operated withdraw exhaust valve opens on a withdraw signal and discharges water from below the main drive piston to the exhaust header. It also serves as the settle valve, which opens following any normal drive movement (insert or withdraw) to allow the control rod and its drive to settle back into the nearest latch position.

4.6.1.1.2.4.3.5 Speed Control Units

The insert drive valve and withdraw exhaust valve have a speed control unit. The speed control unit regulates the control rod insertion and withdrawal rates during normal operation. The manually adjustable flow control unit is used to regulate the water flow to and from the volume beneath the main drive piston. A correctly adjusted unit does not require readjustment except to compensate for changes in drive seal leakage.

4.6.1.1.2.4.3.6 Scram Pilot Valves

The scram pilot valves are operated from the reactor protection system. A scram pilot valve with two solenoids controls both the scram inlet valve and the scram exhaust valve. The scram pilot valves are three-way, solenoid-operated, normally energized valves. On loss of electrical signal to the solenoids, such as the loss of external AC power, the inlet port closes and the exhaust port opens. The pilot valves (Dwgs. M-146, Sh. 1 and M-147, Sh. 1 are designed so that the trip system signal must be removed from both solenoids before air pressure can be discharged from the scram valve operators. This prevents inadvertent scram of a single drive in the event of a failure of one of the pilot valve solenoids.

4.6.1.1.2.4.3.7 Scram Inlet Valve

The scram inlet valve opens to supply pressurized water to the bottom of the drive piston. This quick opening globe valve is operated by an internal spring and system pressure. It is closed by air pressure applied to the top of its diaphragm operator. A main control room indicating light is energized when both the scram inlet valve and the scram exhaust valve are fully open.

4.6.1.1.2.4.3.8 Scram Exhaust Valve

The scram exhaust valve opens slightly before the scram inlet valve, exhausting water from above the drive piston. The exhaust valve opens faster than the inlet valve because of the higher air pressure spring setting in the valve operator.

4.6.1.1.2.4.3.9 Scram Accumulator

The scram accumulator stores sufficient energy to fully insert a control rod at lower vessel pressures. At higher vessel pressures the accumulator pressure is assisted or supplanted by reactor vessel pressure. The accumulator is a hydraulic cylinder with a free-floating piston. The piston separates the water on top from the nitrogen below. A check valve in the accumulator charging line prevents loss of water pressure in the event supply pressure is lost.

During normal plant operation, the accumulator piston is seated at the bottom of its cylinder. Loss of nitrogen decreases the nitrogen pressure, which actuates a pressure switch and sounds an alarm in the control room.

To ensure that the accumulator is always able to produce a scram, it is continuously monitored for water leakage. A float type level switch actuates an alarm if water leaks past the piston barrier and collects in the accumulator instrumentation block.

4.6.1.1.2.5 Control Rod Drive System Operation

The control rod drive system performs rod insertion, rod withdrawal, and scram. These operational functions are described below.

4.6.1.1.2.5.1 Rod Insertion

Rod insertion is initiated by a signal from the operator to the insert valve solenoids. This signal causes both insert valves to open. The insert drive valve applies reactor pressure plus approximately 90 psi to the bottom of the drive piston. The insert exhaust valve allows water from above the drive piston to discharge to the exhaust header.

As is illustrated in Figure 4.6-3, the locking mechanism is a ratchet-type device and does not interfere with rod insertion. The speed at which the drive moves is determined by the flow through the insert speed control valve, which is set for approximately 4 gpm for a shim speed (non-scram operation) of 3 in./sec. During normal insertion, the pressure on the downstream side of the speed control valve is 90 to 100 psi above reactor vessel pressure. However, if the drive slows for any reason, the flow through, and pressure drop across, the insert speed control valve will decrease; the full differential pressure (250 psi) will then be available to cause continued insertion. With 250-psi differential pressure acting on the drive piston, the piston exerts an upward force of 1040 lb.

4.6.1.1.2.5.2 Rod Withdrawal

Rod withdrawal is, by design, more involved than insertion. The collet finger (latch) must be raised to reach the unlocked position (see Figure 4.6-3). The notches in the index tube and the collet fingers are shaped so that the downward force on the index tube holds the collet fingers in place. The index tube must be lifted before the collet fingers can be released. This is done by opening the drive insert valves (in the manner described in the preceding paragraph) for approximately 1 sec. The withdraw valves are then opened, applying driving pressure above the drive piston and opening the area below the piston to the exhaust header. Pressure is simultaneously applied to the collet piston. As the piston raises, the collet fingers are cammed outward, away from the index tube, by the guide cap.

The pressure required to release the latch is set and maintained at a level high enough to overcome the force of the latch return spring plus the force of reactor pressure opposing movement of the collet piston. When this occurs, the index tube is unlatched and free to move in the withdraw direction. Water displaced by the drive piston flows out through the withdraw speed control valve, which is set to give the control rod a shim speed of 3 in./sec. The entire valving sequence is automatically controlled and is initiated by a single operation of the rod withdraw switch.

4.6.1.1.2.5.3 Scram

During a scram the scram pilot valves and scram valves are operated as previously described. With the scram valves open, accumulator pressure is admitted under the drive piston, and the area over the drive piston is vented to the scram discharge volume.

The large differential pressure (initially approximately 1500 psi and always several hundred psi, depending on reactor vessel pressure) produces a large upward force on the index tube and control rod. This force gives the rod a high initial acceleration and provides a large margin of force to overcome friction. After the initial acceleration is achieved, the drive continues at a nearly constant velocity. This characteristic provides a high initial rod insertion rate. As the drive piston nears the top of its stroke the piston seals close off the large passage (buffer orifices) in the stop piston tube, providing a hydraulic cushion at the end of travel.

Prior to a scram signal the accumulator in the Hydraulic Control Unit has approximately 1450-1510 psig on the water side and 1050-1100 psig on the nitrogen side. As the inlet scram valve opens, the full water side pressure is available at the control rod drive acting on a 4.1 sq. in. area. As CRD motion begins, this pressure drops to the gas side pressure less line losses between the accumulator and the CRD. At low vessel pressures the accumulator completely discharges with a vaulting gas side pressure of approximately 575 psi. The control rod drive accumulators are required to scram the control rods when the reactor pressure is low, and the accumulators retain sufficient stored energy to ensure the complete insertion of the control rods in the required time.

The ball check valve in the drive flange allows reactor pressure to supply the scram force whenever reactor pressure exceeds the supply pressure at the drive. This occurs, due to accumulator pressure decay and inlet line losses, during all scrams at higher vessel pressures. When the reactor is close to or at fully operating pressure, reactor pressure alone will insert the control rod in the required time, although the accumulator does provide additional margin at the beginning of the stroke.

The control rod drive system, with accumulators, provides scram performance at full power operation, in terms of average elapsed time after the opening of the reactor protection system trip actuator (scram signal) for the drives to attain the scram strokes.

4.6.1.1.2.6 Instrumentation

The instrumentation for both the control rods and control rod drives is defined by that given for the manual control system. The objective of the reactor manual control system is to provide the operator with the means to make changes in nuclear reactivity so that reactor power level and power distribution can be controlled. The system allows the operator to manipulate control rods.

The design bases and further discussion are covered in Chapter 7.

4.6.1.2 Control Rod Drive Housing Supports

4.6.1.2.1 Safety Objective

The control rod drive (CRD) housing supports prevent any significant nuclear transient in the event a drive housing breaks or separates from the bottom of the reactor vessel.

4.6.1.2.2 Safety Design Bases

The CRD housing supports are engineered safety features and shall meet the following safety design bases:

(1) Following a postulated CRD housing failure, control rod downward motion shall be limited so that any resulting nuclear transient could not be sufficient to cause fuel damage.

(2) The clearance between the CRD housings and the supports shall be sufficient to prevent vertical contact stresses caused by thermal expansion during plant operation.

4.6.1.2.3 Description

The CRD housing supports are shown in Figure 4.6-8. Horizontal beams are installed immediately below the bottom head of the reactor vessel, between the rows of CRD housings. The beams are supported by brackets welded to the steel form liner of the drive room in the reactor support pedestal.

Hanger rods, approximately 10 ft. long and 1-3/4 in. in diameter, are supported from the beams on stacks of disc springs. These springs compress approximately 2 inches under the design load.

The support bars are bolted between the bottom ends of the hanger rods. The spring pivots at the top, and the beveled, loose fitting ends on the support bars prevent substantial bending moment in the hanger rods if the support bars are overloaded.

Individual grids rest on the support bars between adjacent beams. Because a single piece grid would be difficult to handle in the limited work space and because it is necessary that control rod drives, position indicators, and in-core instrumentation components be accessible for inspection and maintenance, each grid is designed for in-place assembly or disassembly. Each grid assembly is made from two grid plates, a clamp, and a bolt. The top part of the clamp guides the grid to its correct position directly below the respective CRD housing that it would support in the postulated accident.

When the support bars and grids are installed, a gap of approximately 1 inch at room temperature (approximately 70°F) is provided between the grid and the bottom contact surface of the control rod drive flange. During system heatup, this gap is reduced by a net downward expansion of the housings with respect to the supports. In the hot operating condition, the gap is approximately 1/4 inch.

In the postulated CRD housing failure, the CRD housing supports are loaded when the lower contact surface of the CRD flange contacts the grid. The resulting load is then carried by two grid plates, two support bars, four hanger rods, their disc springs, and two adjacent beams.

The American Institute of Steel Construction (AISC) Manual of Steel Construction, "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings," was used in designing the CRD housing support system. However, to provide a structure that absorbs as much energy as practical without yielding, the allowable tension and bending stresses used were 90% of yield and the shear stress used was 60% of yield. These design stresses are 1.5 times the AISC allowable stresses (60% and 40% of yield, respectively).

For purposes of mechanical design, the postulated failure resulting in the highest forces is an instantaneous circumferential separation of the CRD housing from the reactor vessel, with an internal pressure of 1086 psig acting on the area of the separated housing. The weight of the separated housing, control rod drive, and blade, plus the pressure of 1086 psig acting on the area of the separated housing, gives a force of approximately 32,000 lb. This force is used to calculate the impact force, conservatively assuming that the housing travels through a 1-in. gap before it contacts the supports. The impact force (109,000 lb.) is then treated as a static load in design.

All CRD housing support subassemblies are fabricated of commonly available structural steel, except for the disc springs, which are Schnorr, Type BS-125-71-8.

4.6.2 Evaluations of the CRDS

This subject is covered under nuclear safety and operational analysis (NSOA) in Appendix 15A, Subsection 15A.6.5.3.

4.6.2.1 Safety Evaluation

Safety evaluation of the control rods, CRDS, and control rod drive housing supports is described below. Further description of control rods is contained in Section 4.2.

4.6.2.1.1 Control Rods

4.6.2.1.1.1 Materials Adequacy Throughout Design Lifetime

The adequacy of the materials throughout the design life was evaluated in the mechanical design of the control rods. The primary materials, B_4C powder and 304 austenitic stainless steel, have been found suitable in meeting the demands of the BWR environment.

4.6.2.1.1.2 Dimensional and Tolerance Analysis

Layout studies are done to assure that, given the worst combination of part tolerance ranges at assembly, no interference exists which will restrict the passage of control rods. In addition, preoperational verification is made on each control blade system to show that the acceptable levels of operational performance are met.

4.6.2.1.1.3 Thermal Analysis of the Tendency to Warp

The various parts of the control rod assembly remain at approximately the same temperature during reactor operation, negating the problem of distortion or warpage. What little differential thermal growth could exist is allowed for in the mechanical design. A minimum axial gap is maintained between absorber rod tubes and the control rod frame assembly for the purpose. In addition, dissimilar metals are avoided to further this end.

4.6.2.1.1.4 Forces for Expulsion

An analysis has been performed which evaluates the maximum pressure forces which could tend to eject a control rod from the core. The results of this analysis are given in Subsection 4.6.2.3.2.2.2. In summary, if the collet were to remain open, which is unlikely, calculations indicate that the steady-state control rod withdrawal velocity would be 2 ft./sec. for a pressure-under line break, the limiting case for rod withdrawal.

4.6.2.1.1.5 Functional Failure of Critical Components

The consequences of a functional failure of critical components have been evaluated and the results are covered in Subsection 4.6.2.3.2.2.

4.6.2.1.1.6 Precluding Excessive Rates of Reactivity Addition

In order to preclude excessive rates of reactivity addition, analysis has been performed both on the velocity limiter device, an engineered safety feature, and the effect of probable control rod failures (see Subsection 4.6.2.3.2.2).

4.6.2.1.1.7 Effect of Fuel Rod Failure on Control Rod Channel Clearances

The control rod drive mechanical design ensures a sufficiently rapid insertion of control rods to preclude the occurrence of fuel rod failures which could hinder reactor shutdown by causing significant distortions in channel clearances.

4.6.2.1.1.8 Mechanical Damage

Analysis has been performed for all areas of the control system showing that system mechanical damage does not affect the capability to continuously provide reactivity control.

In addition to the analysis performed on the control rod drive (Subsection 4.6.2.3.2.2 and Subsection 4.6.2.3.2.3) and the control rod blade, the following discussion summarizes the analysis performed on the control rod guide tube. The guide tube can be subjected to any or all of the following loads:

(1) Inward load due to pressure differential
(2) Lateral loads due to flow across the guide tube
(3) Dead Weight
(4) Seismic (Vertical and Horizontal)
(5) Vibration

In all cases analysis was performed considering both a recirculation line break and a steam line break. These events result in the largest hydraulic loadings on a control rod guide tube.

Two primary modes of failure were considered in the guide tube analysis; exceeding allowable stress and excessive elastic deformation. It was found that the allowable stress limit will not be exceeded and that the elastic deformations of the guide tube never are great enough to cause the free movement of the control rod to be jeopardized.

4.6.2.1.1.9 Evaluation of Control Rod Velocity Limiter

The control rod velocity limiter limits the free fall velocity of the control rod to a value that cannot result in nuclear system process barrier damage. This velocity is evaluated by the rod drop accident analysis in Chapter 15. The control rod velocity limiter is an engineered safety feature.

4.6.2.1.2 Control Rod Drives

4.6.2.1.2.1 Evaluation of Scram Time

The rod scram function of the control rod drive system provides the negative reactivity insertion required by safety design basis in Subsection 4.6.1.1.1.1.1. The scram time shown in the description is adequate as shown by the transient analyses of Chapter 15.

4.6.2.1.2.2 Analysis of Malfunction Relating to Rod Withdrawal

There are no known single malfunctions that cause the unplanned withdrawal of even a single control rod. However, if multiple malfunctions are postulated, studies show that an unplanned rod withdrawal can occur at withdrawal speeds that vary with the combination of malfunctions postulated. In all cases the subsequent withdrawal speeds are less than that assumed in the rod drop accident analysis as discussed in Chapter 15. Therefore, the physical and radiological consequences of such rod withdrawals are less than those analyzed in the rod drop accident.

4.6.2.1.2.2.1 Drive Housing Fails at Attachment Weld

The bottom head of the reactor vessel has a penetration for each control rod drive location. A drive housing is raised into position inside each penetration and fastened by welding. The drive is raised into the drive housing and bolted to a flange at the bottom of the housing. The housing material is seamless, Type 304 stainless steel pipe with a minimum tensile strength of 75,000 psi. The basic failure considered here is a complete circumferential crack through the housing wall at an elevation just below the J-weld.

Static loads on the housing wall include the weight of the drive and the control rod, the weight of the housing below the J-weld, and the reactor pressure acting on the 6-in. diameter cross-sectional area of the housing and the drive. Dynamic loading results from the reaction force during drive operation.

If the housing were to fail as described, the following sequence of events is foreseen. The housing would separate from the vessel. The control rod drive and housing would be blown downward against the support structure by reactor pressure acting on the cross-sectional area of the housing and the drive. The downward motion of the drive and associated parts would be determined by the gap between the bottom of the drive and the support structure and by the deflection of the support structure under load. In the current design, maximum deflection is approximately 3 in. If the collet were to remain latched, no further control rod ejection would occur (Reference 4.6-1); the housing would not drop far enough to clear the vessel penetration. Reactor water would leak at a rate of approximately 220 gpm through the 0.03-inch diametral clearance between the housing and the vessel penetration.

If the basic housing failure were to occur while the control rod is being withdrawn (this is a small fraction of the total drive operating time) and if the collet were to stay unlatched, the following sequence of events is foreseen. The housing would separate from the vessel. The drive and housing would be blown downward against the control rod drive housing support. Calculations indicate that the steady-state rod withdrawal velocity would be 0.3 ft./sec. During withdrawal, pressure under the collet piston would be approximately 250 psi greater than the pressure over it. Therefore, the collet would be held in the unlatched position until driving pressure was removed from the pressure-over port.

4.6.2.1.2.2.2 Rupture of Hydraulic Line(s) to Drive Housing Flange

There are three types of possible rupture of hydraulic lines to the drive housing flange: (1) pressure-under line break; (2) pressure-over line break; and (3) coincident breakage of both of these lines.

4.6.2.1.2.2.2.1 Pressure-under Line Break

For the case of a pressure-under line break, a partial or complete circumferential opening is postulated at or near the point where the line enters the housing flange. Failure is more likely to occur after another basic failure wherein the drive housing or housing flange separates from the reactor vessel. Failure of the housing, however, does not necessarily lead directly to failure of the hydraulic lines.

If the pressure-under line were to fail and if the collet were latched, no control rod withdrawal would occur. There would be no pressure differential across the collet piston and, therefore, no tendency to unlatch the collet. Consequently, the associated control rod could not be withdrawn, but if reactor pressure is greater than 600 psig, it will insert on a scram signal.

The ball check valve is designed to seal off a broken pressure-under line by using reactor pressure to shift the check ball to its upper seat. If the ball check valve were prevented from seating, reactor water would leak to the atmosphere. Because of the broken line, cooling water could not be supplied to the drive involved. Loss of cooling water would cause no immediate damage to the drive. However, prolonged exposure of the drive to temperatures at or near reactor temperature could lead to deterioration of material in the seals. High temperature would be indicated to the operator by the thermocouple in the position indicator probe. A second indication would be high cooling water flow.

If the basic line failure were to occur while the control rod is being withdrawn the hydraulic force would not be sufficient to hold the collet open, and spring force normally would cause the collet to latch and stop rod withdrawal. However, if the collet were to remain open, calculations indicate that the steady-state control rod withdrawal velocity would be 2 ft./sec.

4.6.2.1.2.2.2.2 Pressure-over Line Break

The case of the pressure-over line breakage considers the complete breakage of the line at or near the point where it enters the housing flange. If the line were to break, pressure over the drive piston would drop from reactor pressure to atmospheric pressure. Any significant reactor pressure (approximately 600 psig or greater) would act on the bottom of the drive piston and fully insert the drive. Insertion would occur regardless of the operational mode at the time of the failure. After full insertion, reactor water would leak past the stop piston seals. This leakage would exhaust to the atmosphere through the broken pressure-over line. The leakage rate at 1000 psi reactor pressure is estimated to be 4 gpm nominal but not more than 10 gpm, based on experimental measurements. If the reactor were hot, drive temperature would increase. This situation would be indicated to the reactor operator by the drift alarm, by the fully inserted drive, by a high drive temperature (indicated and printed out on a recorder in the control room), and by operation of the drywell sump pump.

For the simultaneous breakage of the pressure-over and pressure-under lines, pressures above and below the drive piston would drop to zero, and the ball check valve would close the broken pressure-under line. Reactor water would flow from the annulus outside the drive, through the vessel ports, and to the space below the drive piston. As in the case of pressure-over line breakage, the drive would then insert (if the reactor were above 600 psi) at a speed dependent on reactor pressure. Full insertion would occur regardless of the operational mode at the time of failure. Reactor water would leak past the drive seals and out the broken pressure-over line to the atmosphere, as described above. Drive temperature would increase. Indication in the control room would include the drift alarm, the fully inserted drive, the high drive temperature printed out on a recorder in the control room, and operation of the drywell sump pump.

4.6.2.1.2.2.3 All Drive Flange Bolts Fail in Tension

Each control rod drive is bolted to a flange at the bottom of a drive housing. The flange is welded to the drive housing. Bolts are made of AISI-4140 (or AISI-4340) steel, with a minimum tensile strength of 125,000 (or 135,000 for AISI-4340) psi. Each bolt has an allowable load capacity of 14,800 (or 28,800 for AISI-4340) pounds. Capacity of the 8 bolts is 118,400 (or 230,400 for AISI-4340) pounds. As a result of the reactor design pressure of 1250 psig, the design basis load on all 8 bolts is 45,015 pounds.

If a progressive or simultaneous failure of all bolts were to occur, the drive would separate from the housing. The control rod and the drive would be blown downward against the support structure. Impact velocity and support structure loading would be slightly less than that for drive housing failure, because reactor pressure would act on the drive cross-sectional area only and the housing would remain attached to the reactor vessel. The drive would be isolated from the cooling water supply. Reactor water would flow downward past the velocity limiter piston, through the large drive filter, and into the annular space between the thermal sleeve and the drive. For worst-case leakage calculations, the large filter is assumed to be deformed or swept out of the way so it would offer no significant flow restriction. At a point near the top of the annulus, where pressure would have dropped to 350 psi, the water would flash to steam and cause choke-flow conditions. Steam would flow down the annulus and out the space between the housing and the drive flanges to the drywell. Steam formation would limit the leakage rate to approximately 840 gpm.

If the collet were latched, control rod ejection would be limited to the distance the drive can drop before coming to rest on the support structure. There would be no tendency for the collet to unlatch, because pressure below the collet piston would drop to zero. Pressure forces, in fact, exert 1435 pounds to hold the collet in the latched position.

If the bolts failed during control rod withdrawal, pressure below the collet piston would drop to zero. The collet, with 1650 pounds return force, would latch and stop rod withdrawal.

4.6.2.1.2.2.4 Weld Joining Flange to Housing Fails in Tension

The failure considered is a crack in or near the weld that joins the flange to the housing. This crack extends through the wall and completely around the housing. The flange material is forged, Type 304 stainless steel, with a minimum tensile strength of 75,000 psi. The housing material is seamless, Type 304 stainless steel pipe, with a minimum tensile strength of 75,000 psi. The conventional, full penetration weld of Type 308 stainless steel has a minimum tensile strength approximately the same as that for the parent metal. The design pressure and temperature are 1250 psig and 575°F. Reactor pressure acting on the cross-sectional area of the drive; the weight of the control rod, drive, and flange; and the dynamic reaction force during drive operation result in a maximum tensile stress at the weld of approximately 6000 psi.

If the basic flange-to-housing joint failure occurred, the flange and the attached drive would be blown downward against the support structure. The support structure loading would be slightly less than that for drive housing failure, because reactor pressure would act only on the drive cross-sectional area. Lack of differential pressure across the collet piston would cause the collet to remain latched and limit control rod motion to approximately 3 inches. Downward drive movement would be small, therefore, most of the drive would remain inside the housing. The pressure-under and pressure-over lines are flexible enough to withstand the small displacement and remain attached to the flange. Reactor water would follow the same leakage path described above for the flange bolt failure, except that exit to the drywell would be through the gap between the lower end of the housing and the top of the flange. Water would flash to steam in the annulus surrounding the drive. The leakage rate would be approximately 840 gpm.

If the basic failure were to occur during control rod withdrawal (a small fraction of the total operating time) and if the collet were held unlatched, the flange would separate from the housing. The drive and flange would be blown downward against the support structure. The calculated steady-state rod withdrawal velocity would be 0.13 ft./sec. Because pressure-under and pressure-over lines remain intact, driving water pressure would continue to the drive, and the normal exhaust line restriction would exist. The pressure below the velocity limiter piston would drop below normal as a result of leakage from the gap between the housing and the flange. This differential pressure across the velocity limiter piston would result in a net downward force of approximately 70 pounds. Leakage out of the housing would greatly reduce the pressure in the annulus surrounding the drive. Thus, the net downward force on the drive piston would be less than normal. The overall effect of these events would be to reduce rod withdrawal to approximately one-half of normal speed. With a 560-psi differential across the collet piston, the collet would remain unlatched; however, it should relatch as soon as the drive signal is removed.

4.6.2.1.2.2.5 Housing Wall Ruptures

This failure is a vertical split in the drive housing wall just below the bottom head of the reactor vessel. The flow area of the hole is considered equivalent to the annular area between the drive and the thermal sleeve. Thus, flow through this annular area, rather than flow through the hole in the housing, would govern leakage flow. The housing is made of Type 304 stainless steel seamless pipe, with a minimum tensile strength of 75,000 psi. The maximum hoop stress of 11,900 psi results primarily from the reactor design pressure (1250 psig) acting on the inside of the housing.

If such a rupture were to occur, reactor water would flash to steam and leak through the hole in the housing to the drywell at approximately 1030 gpm. Choke-flow conditions would exist, as described previously for the flange-bolt failure. However, leakage flow would be greater because flow resistance would be less, that is, the leaking water and steam would not have to flow down the length of the housing to reach the drywell. A critical pressure of 350 psi causes the water to flash to steam.

No pressure differential across the collet piston would tend to unlatch the collet; but the drive would insert as a result of loss of pressure in the drive housing causing a pressure drop in the space above the drive piston.

If this failure occurred during control rod withdrawal, drive withdrawal would stop, but the collet would remain unlatched. The drive would be stopped by a reduction of the net downward force action on the drive line. The net force reduction would occur when the leakage flow of 1030 gpm reduces the pressure in the annulus outside the drive to approximately 540 psig, thereby reducing the pressure acting on top of the drive piston to the same value. A pressure differential of approximately 710 psi would exist across the collet piston and hold the collet unlatched as long as the operator held the withdraw signal.

4.6.2.1.2.2.6 Flange Plug Blows Out

To connect the vessel ports with the bottom of the ball check valve, a hole of 3/4-inch diameter is drilled in the drive flange. The outer end of this hole is sealed with a plug of 0.812 inch diameter and 0.25 inch thickness. A full-penetration, Type 308 stainless steel weld holds the plug in place. The postulated failure is a full circumferential crack in this weld and subsequent blowout of the plug.

If the weld were to fail, the plug were to blow out, and the collet remained latched, there would be no control rod motion. There would be no pressure differential across the collet piston acting to unlatch the collet. Reactor water would leak past the velocity limiter piston, down the annulus between the drive and the thermal sleeve, through the vessel ports and drilled passage, and out the open plug hole to the drywell at approximately 320 gpm. Leakage calculations assume only liquid flows from the flange. Actually, hot reactor water would flash to steam, and choke-flow conditions would exist. Thus, the expected leakage rate would be lower than the calculated value. Drive temperature would increase and initiate an alarm in the control room.

If this failure were to occur during control rod withdrawal and if the collet were to stay unlatched, calculations indicate that control rod withdrawal speed would be approximately 0.24 ft./sec. Leakage from the open plug hole in the flange would cause reactor water to flow downward past the velocity limiter piston. A small differential pressure across the piston would result in an insignificant driving force of approximately 10 lb., tending to increase withdraw velocity.

A pressure differential of 295 psi across the collet piston would hold the collet unlatched as long as the driving signal was maintained.

Flow resistance of the exhaust path from the drive would be normal because the ball check valve would be seated at the lower end of its travel by pressure under the drive piston.

4.6.2.1.2.2.7 Ball Check Valve Plug Blows Out

As a means of access for machining the ball check valve cavity, a 1.25 inch diameter hole has been drilled in the flange forging. This hole is sealed with a plug of 1.31 inch diameter and 0.38 inch thickness. A full-penetration weld, utilizing Type 308 stainless steel filler, holds the plug in place. The failure postulated is a circumferential crack in this weld leading to a blowout of the plug.

If the plug were to blow out while the drive was latched, there would be no control rod motion. No pressure differential would exist across the collet piston to unlatch the collet. As in the previous failure, reactor water would flow past the velocity limiter, down the annulus between the drive and thermal sleeve, through the vessel ports and drilled passage, through the ball check valve cage and out the open plug hole to the drywell. The leakage calculations indicate the flow rate would be 350 gpm. This calculation assumes liquid flow, but flashing of the hot reactor water to steam would reduce this rate to a lower value. Drive temperature would rapidly increase and initiate an alarm in the control room.

If the plug failure were to occur during control rod withdrawal, (it would not be possible to unlatch the drive after such a failure) the collet would relatch at the first locking groove. If the collet were to stick, calculations indicate the control rod withdrawal speed would be 11.8 feet per second. There would be a large retarding force exerted by the velocity limiter due to a 35 psi pressure differential across the velocity limiter piston.

4.6.2.1.2.2.8 Drive Pressure Control Valve Closure (Reactor Pressure, 0 psig)

The pressure to move a drive is generated by the pressure drop of practically the full system flow through the drive pressure control valve. This valve is either a motor operated valve or a standby manual valve; either one is adjusted to a fixed opening. The normal pressure drop across this valve develops a pressure 250 psi in excess of reactor pressure.

If the flow through the drive pressure control valve were to be stopped, as by a valve closure or flow blockage, the drive pressure would increase to the shutoff pressure of the supply pump. The occurrence of this condition during withdrawal of a drive at zero vessel pressure will result in a drive pressure increase from 250 psig to no more than 1750 psig. Calculations indicate that the drive would accelerate from 3 in./sec. to approximately 6.5 in./sec. A pressure differential of 1670 psi across the collet piston would hold the collet unlatched. Flow would be upward, past the velocity limiter piston, but retarding force would be negligible. Rod movement would stop as soon as the driving signal was removed.

4.6.2.1.2.2.9 Ball Check Valve Fails to Close Passage to Vessel Ports

Should the ball check valve sealing the passage to the vessel ports be dislodged and prevented from reseating following the insert portion of a drive withdrawal sequence, water below the drive piston would return to the reactor through the vessel ports and the annulus between the drive and the housing rather than through the speed control valve. Because the flow resistance of this return path would be lower than normal, the calculated withdrawal speed would be 2 ft./sec. During withdrawal, differential pressure across the collet piston would be approximately 40 psi. Therefore, the collet would tend to latch and would have to stick open before continuous withdrawal at 2 ft./sec, could occur. Water would flow upward past the velocity limiter piston, generating a small retarding force of approximately 120 pounds.

4.6.2.1.2.2.10 Hydraulic Control Unit Valve Failures

Various failures of the valves in the HCU can be postulated, but none could produce differential pressures approaching those described in the preceding paragraphs and none alone could produce a high velocity withdrawal. Leakage through either one or both of the scram valves produces a pressure that tends to insert the control rod rather than to withdraw it. If the pressure in the scram discharge volume should exceed reactor pressure following a scram, a check valve in the line to the scram discharge header prevents this pressure from operating the drive mechanisms.

4.6.2.1.2.2.11 Collet Fingers Fail to Latch

The failure is presumed to occur when the drive withdraw signal is removed. If the collet fails to latch, the drive continues to withdraw at a fraction of the normal speed. This assumption is made because there is no known means for the collet fingers to become unlocked without some initiating signal. Because the collet fingers will not cam open under a load, accidental application of a down signal does not unlock them. (The drive must be given a short insert signal to unload the fingers and cam them open before the collet can be driven to the unlock position.) If the drive withdrawal valve fails to close following a rod withdrawal, the collet would remain open and the drive continue to move at a reduced speed.

4.6.2.1.2.2.12 Withdrawal Speed Control Valve Failure

Normal withdrawal speed is determined by differential pressures in the drive and is set for a nominal value of 3 in./sec. Withdrawal speed is maintained by the pressure regulating system and is independent of reactor vessel pressure. Tests have shown that accidental opening of the speed control valve to the full-open position produces a velocity of approximately 6 in./sec.

The control rod drive system prevents unplanned rod withdrawal and it has been shown above that only multiple failures in a drive unit and in its control unit could cause an unplanned rod withdrawal.

4.6.2.1.2.3 Scram Reliability

High scram reliability is the result of a number of features of the CRD system. For example:

(1) Two reliable sources of scram energy are used to insert each control rod: individual accumulators at low reactor pressure, and the reactor vessel pressure itself at power.

(2) Each drive mechanism has its own scram and a dual solenoid scram pilot valve so only one drive can be affected if a scram valve fails to open. Both valve solenoids must be deenergized to initiate a scram.

(3) The reactor protection system and the HCUs are designed so that the scram signal and mode of operation override all others.

(4) The collet assembly and index tube are designed so they will not restrain or prevent control rod insertion during scram.

(5) The scram discharge volume is monitored for accumulated water and the reactor will scram before the volume is reduced to a point that could interfere with a scram.

4.6.2.1.2.4 Control Rod Support and Operation

As described above, each control rod is independently supported and controlled as required by safety design bases.

4.6.2.1.3 Control Rod Drive Housing Supports

Downward travel of the CRD housing and its control rod following the postulated housing failure equals the sum of these distances: (1) the compression of the disc springs under dynamic loading, and (2) the initial gap between the grid and the bottom contact surface of the CRD flange. If the reactor were cold and pressurized, the downward motion of the control rod would be limited to the spring compression (approximately 2 in.) plus a gap of approximately 1 in. If the reactor were hot and pressurized, the gap would be approximately 1/4 in. and the spring compression would be slightly less than in the cold condition. In either case, the control rod movement following a housing failure is substantially limited below one drive "notch" movement (6 in.). Sudden withdrawal of any control rod through a distance of one drive notch at any position in the core does not produce a transient sufficient to damage any radioactive material barrier.

The CRD housing supports are in place during power operation and when the nuclear system is pressurized. If a control rod is ejected during shutdown, the reactor remains subcritical because it is designed to remain subcritical with any one control rod fully withdrawn at any time.

At plant operating temperature, a gap of approximately 1/4 in. exists between the CRD housing and the supports. At lower temperatures the gap is greater. Because the supports do not contact any of the CRD housing except during the postulated accident condition, vertical contact stresses are prevented.

4.6.3 Testing and Verification of the CRDs

4.6.3.1 Control Rod Drives

4.6.3.1.1 Testing and Inspection

4.6.3.1.1.1 Development Tests

The development drive (prototype) testing included more than 5000 scrams and approximately 100,000 latching cycles. One prototype was exposed to simulated operating conditions for 5000 hours. These tests demonstrated the following:

(1) The drive easily withstands the forces, pressures, and temperatures imposed.

(2) Wear, abrasion, and corrosion of the nitrided stainless parts are negligible. Mechanical performance of the nitrided surface is superior to that of materials used in earlier operating reactors.

(3) The basic scram speed of the drive has a satisfactory margin above minimum plant requirements at any reactor vessel pressure.

(4) Usable seal lifetimes in excess of 1000 scram cycles can be expected.

4.6.3.1.1.2 Factory Quality Control Tests

Quality control of welding, heat treatment, dimensional tolerances, material verification, and similar factors is maintained throughout the manufacturing process to assure reliable performance of the mechanical reactivity control components. Some of the quality control tests performed on the control rods, control rod drive mechanisms, and hydraulic control units are listed below:

(1) Control rod drive mechanism tests:

- a. Pressure welds on the drives are hydrostatically tested in accordance with ASME codes.
- b. Electrical components are checked for electrical continuity and resistance to ground.
- c. Drive parts that cannot be visually inspected for dirt are flushed with filtered water at high velocity. No significant foreign material is permitted in effluent water.
- d. Seals are tested for leakage to demonstrate correct seal operation.
- e. Each drive is tested for shim motion, latching, and control rod position indication.
- f. Each drive is subjected to cold scram tests at various reactor pressures to verify correct scram performance.

(2) Hydraulic control unit tests:

a. Hydraulic systems are hydrostatically tested in accordance with the applicable code.

b. Electrical components and systems are tested for electrical continuity and resistance to ground.

c. Correct operation of the accumulator pressure and level switches is verified.

d. The unit's ability to perform its part of a scram is demonstrated.

e. Correct operation and adjustment of the insert and withdrawal valves is demonstrated.

4.6.3.1.1.3 Operational Tests

After installation, all rods and drive mechanisms can be tested through their full stroke for operability.

During normal operation, each time a control rod is withdrawn a notch, the operator can observe the in-core monitor indications to verify that the control rod is following the drive mechanism. All control rods that are partially withdrawn from the core can be tested for rod-following by inserting or withdrawing the rod one notch and returning it to its original position, while the operator observes the in-core monitor indications.

To make a positive test of control rod to control rod drive coupling integrity, the operator can withdraw a control rod to the end of its travel and then attempt to withdraw the drive to the overtravel position. Failure of the drive to overtravel demonstrates rod-to-drive coupling integrity.

Hydraulic supply subsystem pressures can be observed from instrumentation in the control room. Scram accumulator pressures can be observed on the nitrogen pressure gages.

4.6.3.1.1.4 Acceptance Tests

Criteria for acceptance of the individual control rod drive mechanisms and the associated control and protection systems will be incorporated in specifications and test procedures covering three distinct phases: (1) pre-installation, (2) after installation prior to startup, and (3) during startup testing.

The pre-installation specification will define criteria and acceptable ranges of such characteristics as seal leakage, friction and scram performance under fixed test conditions which must be met before the component can be shipped.

Then after installation, prestartup tests will be performed as outlined in Chapter 14.

As fuel is placed in the reactor, the power test procedure will be performed as outlined in Chapter 14.

4.6.3.1.1.5 Surveillance Tests

The surveillance requirements for the control rod drive system are described below.

(1) Sufficient control rods shall be withdrawn, following a refueling outage when core alterations are performed, to demonstrate with a margin of 0.25% k that the core can be made subcritical at any time in the subsequent fuel cycle with the strongest operable control rod fully withdrawn and all other operable rods fully inserted.

(2) Each partially or fully withdrawn control rod shall be exercised one notch at least once each 31 days.

In the event that operation is continuing with three or more rods valved out of service, this test shall be performed at least once each day.

The weekly control rod exercise test serves as a periodic check against deterioration of the control rod system and also verifies the ability of the control rod drive to scram. If a rod can be moved with drive pressure, it may be expected to scram since higher pressure is applied during scram. The frequency of exercising the control rods under the conditions of three or more control rods valved out of service provides even further assurance of the reliability of the remaining control rods.

(3) The coupling integrity shall be verified for each withdrawn control rod as follows:

    a. When the rod is first withdrawn, observe discernible response of the nuclear instrumentation; and

    b. When the rod is fully withdrawn the first time, observe that the drive will not go to the overtravel position.

Observation of a response from the nuclear instrumentation during an attempt to withdraw a control rod indicates indirectly that the rod and drive are coupled. The overtravel position feature provides a positive check on the coupling integrity, for only an uncoupled drive can reach the overtravel position.

(4) During operation, accumulator pressure and level at the normal operating value shall be verified. Experience with control rod drive systems of the same type indicates that weekly verification of accumulator pressure and level is sufficient to assure operability of the accumulator portion of the control rod drive system.

(5) At the time of each major refueling outage, each operable control rod shall be subjected to scram time tests from the fully withdrawn position.

Experience indicates that the scram times of the control rods do not significantly change over the time interval between refueling outages. A test of the scram times at each refueling outage is sufficient to identify any significant lengthening of the scram times.

4.6.3.1.1.6 Functional Tests

The functional testing program of the control rod drives consists of the 5 year maintenance life and the 1.5X design life test programs as described in Subsection 3.9.4.4.

There are a number of failures that can be postulated on the CRD but it would be very difficult to test all possible failures. A partial test program with postulated accident conditions and imposed single failures is available.

The following tests with imposed single failures have been performed to evaluate the performance of the CRDs under these conditions.

Simulated Ruptured Scram Line Test
Stuck Ball Check Valve in CRD Flange
HCU Drive Down Inlet Flow Control Valve (V122) Failure
HCU Drive Down Outlet Flow Control Valve (V120) Failure
CRD Scram Performance with V120 Malfunction
HCU Drive Up Outlet Control Valve (V121) Failure
HCU Drive Up Inlet Control Valve (V123) Failure
Cooling Water Check Valve (V138) Leakage
CRD Flange Check Valve Leakage
CRD Stabilization Circuit Failure
HCU Filter Restriction
Air Trapped in CRD Hydraulic System
CRD Collet Drop Test
CR Qualification Velocity Limiter Drop Test

Additional postulated CRD failures are discussed in Subsections 4.6.2.1.2.2.1 through4.6.2.1.2.2.11.

4.6.3.2 Control Rod Drive Housing Supports

4.6.3.2.1 Testing and Inspection

CRD housing supports are removed for inspection and maintenance of the control rod drives. The supports for one control rod can be removed during reactor shutdown, even when the reactor is pressurized, because all control rods are then inserted. When the support structure is reinstalled, it is inspected for correct assembly with particular attention to maintaining the correct gap between the CRD flange lower contact surface and the grid.

4.6.4 Information for Combined Performance of Reactivity Systems

4.6.4.1 Vulnerability to Common Mode Failures

The reactivity control systems have been located in accordance with the separation criteria described in Section 3.12. The locations of the equipment for these systems are shown on the figures in Section 1.2.

4.6.4.2 Accidents Taking Credit for Multiple Reactivity Systems

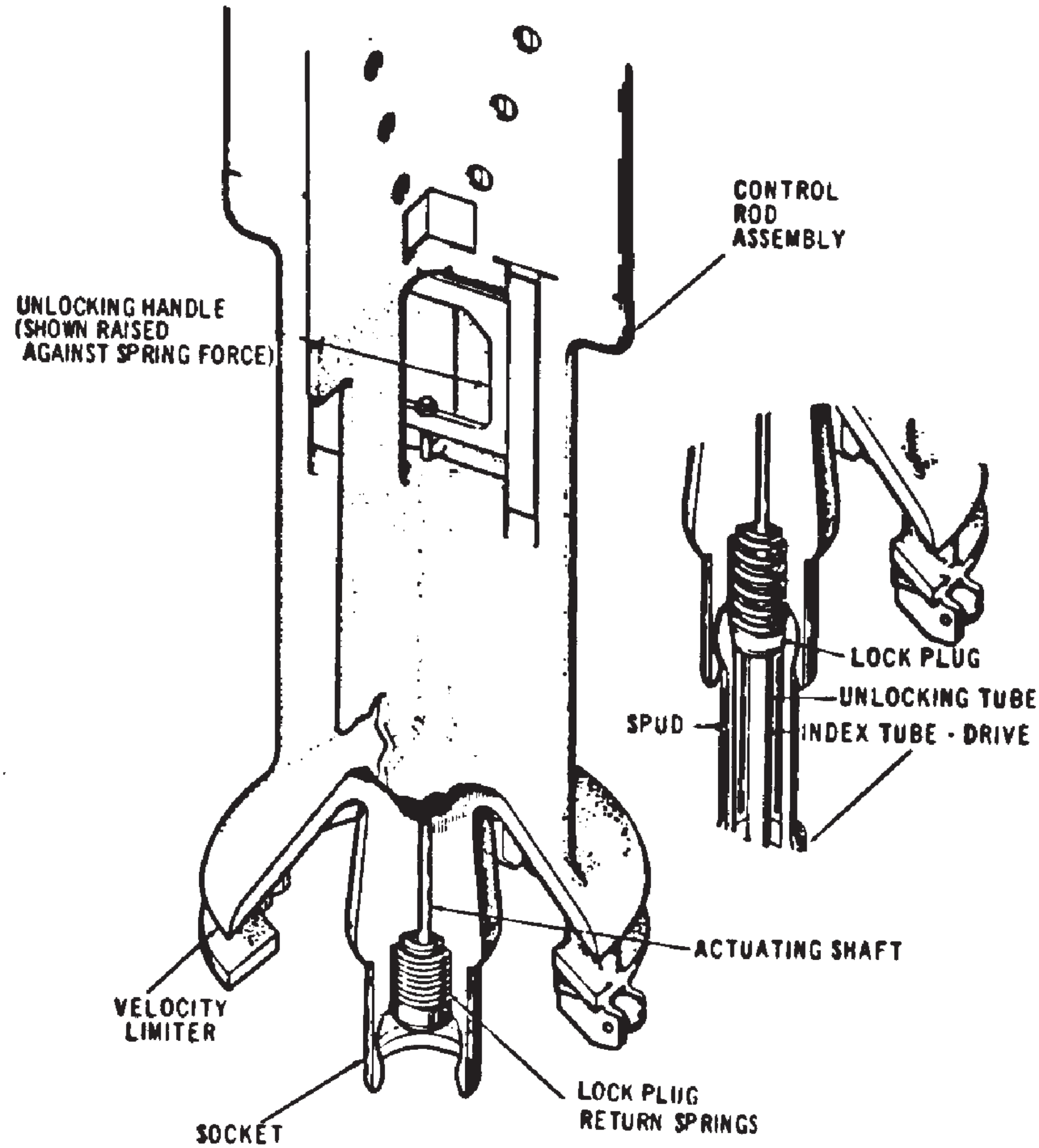
There are no postulated accidents evaluated in Chapter 15 that take credit for two or more reactivity control systems preventing or mitigating each accident.

4.6.5 EVALUATION OF COMBINED PERFORMANCE

As indicated in Subsection 4.6.4.2, credit is not taken for multiple reactivity control systems for any postulated accidents in Chapter 15.

4.6.6 REFERENCES

4.6-1 Benecki, J. E., "Impact Testing on Collet Assembly for Control Rod Drive Mechanism 7RD B144A," General Electric Company, Atomic Power Equipment Department, APED-5555, November 1967.



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CONTROL ROD TO
CONTROL ROD DRIVE
COUPLING

FIGURE 4.6-1, Rev. 47

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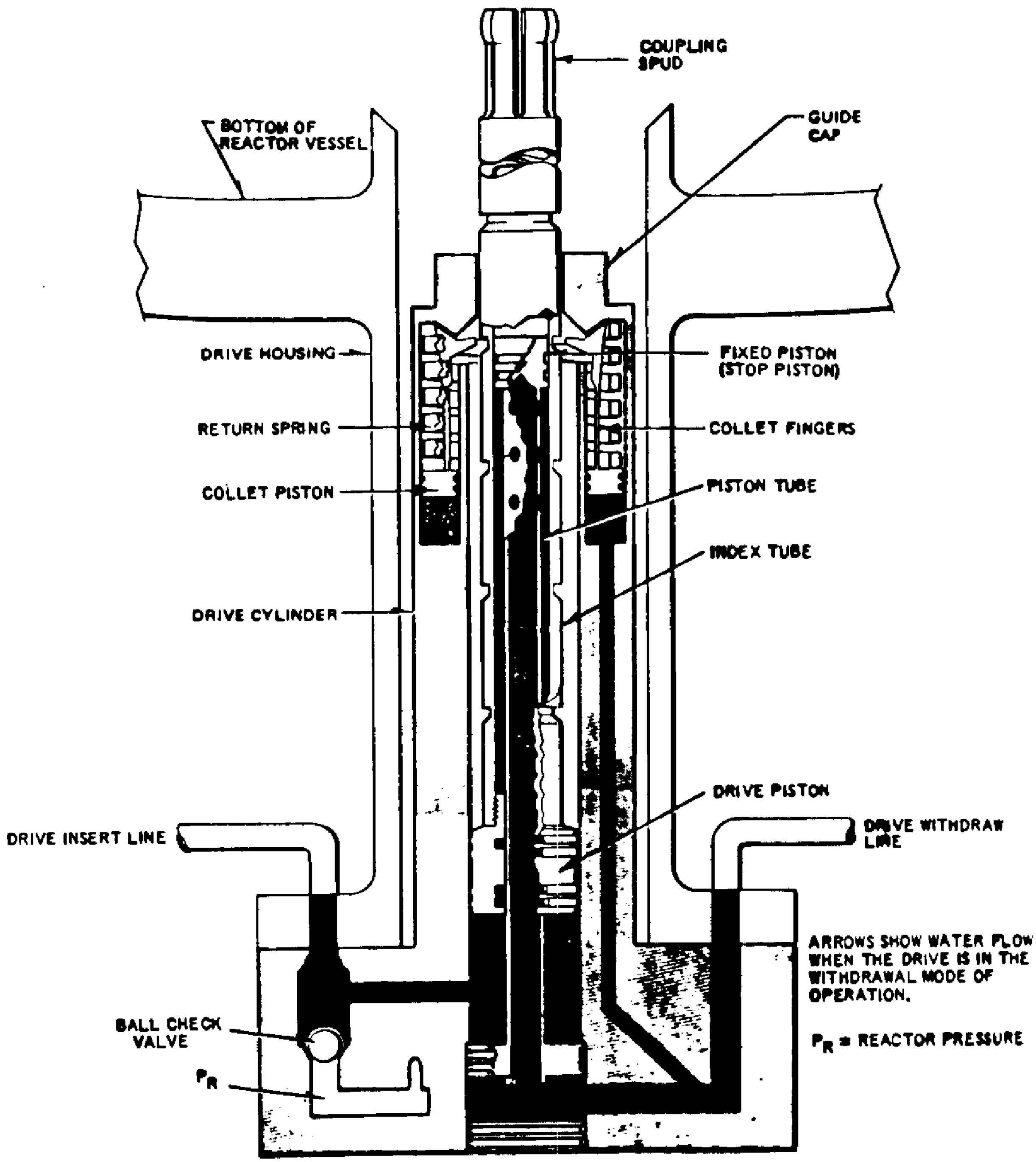

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CONTROL ROD DRIVE UNIT

FIGURE 4.6-2, Rev. 47

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Security-Related Information
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CONTROL ROD DRIVE SCHEMATIC FIGURE 4.6-3

Security-Related Information
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SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
CONTROL ROD DRIVE (CUTAWAY) FIGURE 4.6-4

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-146, Sh. 1

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SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 4.6-5A replaced by dwg. M-146, Sh. 1 FIGURE 4.6-5A, Rev. 50

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THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-147, Sh. 1

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 4.6-5B replaced by dwg. M-147, Sh. 1 FIGURE 4.6-5B, Rev. 55

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M1-C12-8, Sh. 1

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Figure 4.6-6 replaced by dwg.
M1-C12-8, Sh. 1

FIGURE 4.6-6, Rev. 49

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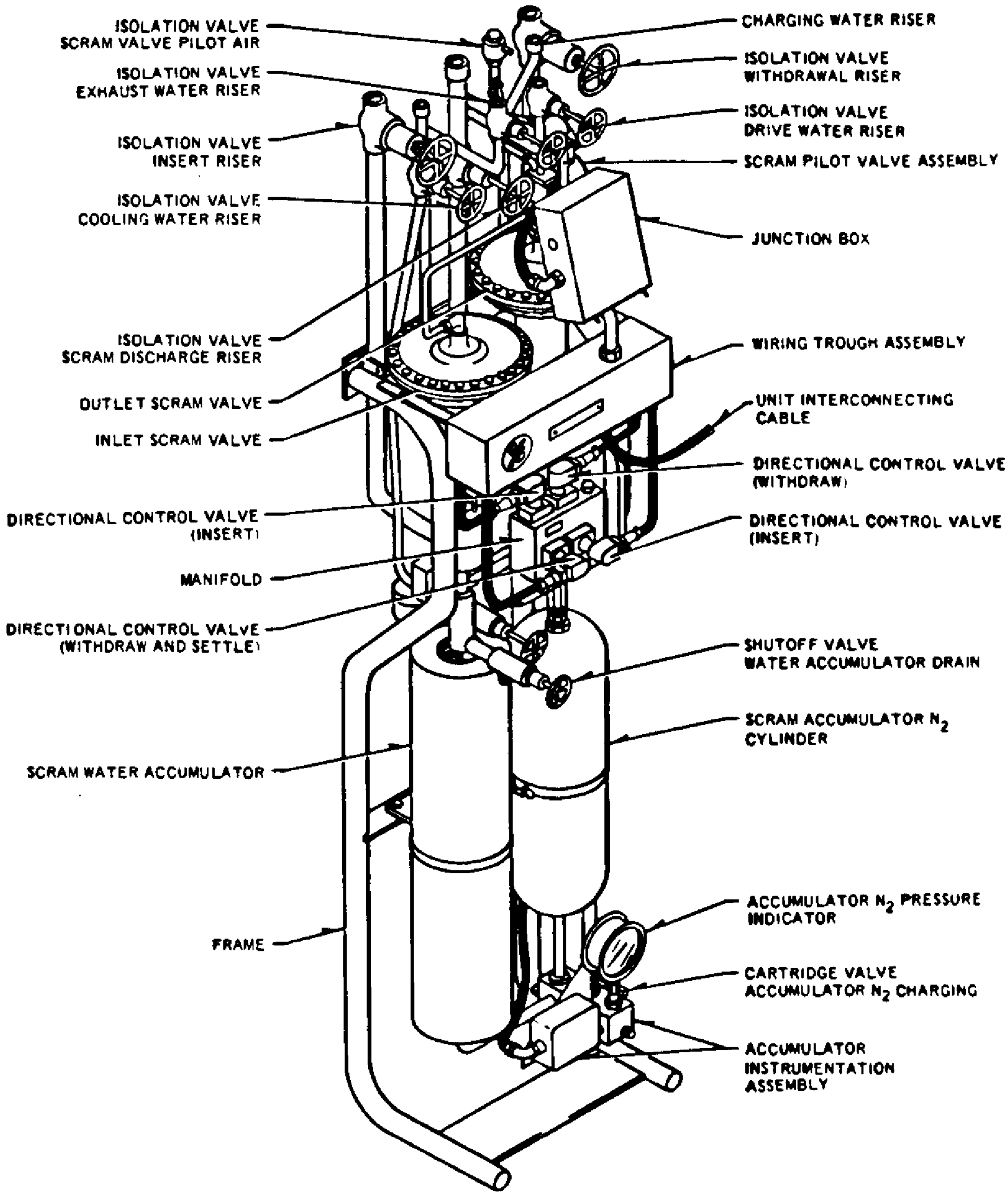

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CONTROL ROD DRIVE
HYDRAULIC
CONTROL UNIT

FIGURE 4.6-7, Rev. 47

Auto-Cad Figure Fsar 4_6_7.dwg

Security-Related Information
Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
CONTROL ROD DRIVE HOUSING SUPPORT FIGURE 4.6-8