CHAPTER 4 - REACTOR

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* The listed drawings are included as "General References" only; i.e., refer to the drawings to obtain additional detail or to obtain background information. These drawings are not part of the USAR. They are controlled by the Controlled Documents Program.

DRAWING*	SUBJECT
762E412CA	Control Rod Drive
M05-1078	Control Rod Drive System

CHAPTER 4 - REACTOR

4.1 SUMMARY DESCRIPTION

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 3.9-7, Reactor Vessel Cutaway, shows the arrangement of reactor assembly components. A summary of the important design and performance characteristics is given in section 1.3.1, "Comparison with Similar Facility Designs." Loading conditions for reactor assembly components are specified in section 3.9.5.2.

4.1.1 Reactor Vessel

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

4.1.2 Reactor Internal Components

The major reactor internal components are the core (fuel, channels, control blades, and incore 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. The fuel assemblies (including fuel rods and channel), control blades, incore instrumentation, shroud head and steam separator assembly, and steam dryers are removable when the reactor vessel is opened for refueling or maintenance.

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 levels characteristic of a direct cycle reactor (approximately 1000 psia) result in moderate cladding temperatures and stress levels.

(2) The low coolant saturation temperature, high heat transfer coefficients, and 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 General Electric thermal analysis basis, revised GETAB methodology with reduced uncertainties, is applied to assure that more than 99.9% of the fuel rods in the core are expected to avoid boiling transition for the most severe moderate frequency (Per Regulatory Guide 1.70 Rev 3) 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, the BWR has a number of inherent advantages. These are 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 1).

Important features of the reactor core arrangement are as follows:

(1) The bottom-entry cruciform control rods consist of B_4C contained in four control blade wings.

(2) The fixed in-core fission chambers provide continuous power range neutron flux monitoring. A guide tube in each in-core assembly provides for a traversing ion chamber for calibration and axial detail. Source and intermediate range detectors 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 discussed in Subsection 7.7.1.6.

(3) As shown by experience obtained at Dresden-1 and other plants, 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 Zirconium alloy reusable channels 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 mechanical reactivity control permits criticality checks during refueling and provides maximum plant safety. 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 represents a practical value of individual control rod reactivity worth, and allows adequate 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.

4.1.2.1.3 Fuel Assembly Description

The boiling water reactor core is composed of essentially two components--fuel assemblies and control rods.

4.1.2.1.3.1 Fuel Rod

A fuel rod consists of UO_2 pellets and a Zircaloy cladding tube. A fuel rod is made by stacking pellets into a Zircaloy cladding tube which is evacuated, back-filled with helium, and sealed by welding Zircaloy end plugs in each end of the tube. 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 applied loads, both external and internal. The fuel pellet is sized to provide sufficient clearance within the fuel tube to accommodate axial and radial 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 Subsection 4.2.1.

4.1.2.1.3.2 Fuel Bundle

Each fuel bundle contains fuel rods and water rods which are spaced and supported in a square array by spacers and a lower and upper tie plate. 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.

The fuel assemblies, of which the core is comprised, are designed to meet all the criteria for core performance and to provide ease of handling. Selected fuel rods in each assembly differ from the others in uranium enrichment. This arrangement 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.

A more detailed description of the fuel bundle designs utilized in the current cycle is provided in Appendix 15D, Reload Analysis.

4.1.2.1.4 Assembly Support and Control Rod Location

A few 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 on top of the shroud, provides lateral support and guidance for the top of each fuel assembly. The reactivity of the core is controlled by cruciform control rods, containing boron carbide, and their associated mechanical hydraulic drive system. The control rods occupy alternate spaces between fuel assemblies. Each independent drive enters the core from the bottom, 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 information on the shroud is contained in subsection 3.9.5.1.1.1.

4.1.2.3 Shroud Head and Steam Separators

The information on the shroud head and steam separators is contained in subsection 3.9.5.1.1.3.

4.1.2.4 Steam Dryer Assembly

The information on the steam dryer assembly is contained in subsection 3.9.5.1.1.9.

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 effect 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 Control Rods

The cruciform shaped control rods contain stainless steel wings filled with vibration compacted boron-carbide powder. (Some control rods have boron-carbide filled wings as well as hafnium plates and other control rods have only boron-carbide filled wings).

The hafnium plates and wings are held in a cruciform array.

A top handle, aligns the wings and provides structural rigidity at the top of the control rod. Some control blade handles have rollers or buttons to provide guidance for control rod insertion and withdrawal. Some control blade handles have no rollers or pads. A bottom casting is also used to provide structural rigidity and contains positioning rollers and a parachute-shaped velocity limiter.

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

The velocity limiter 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, the most important of which are:

(1) The area between the fuel channel and the fuel assembly lower tie plate;

(2) Holes in the lower tie plate;

(3) The area between the fuel assembly lower tie plate and the fuel support piece;

(4) The area between the fuel support piece and the control rod guide tube;

(5) The area between the control rod guide tube and the core support plate; and

(6) The area between the core support plate and the shroud.

4.1.3.3 Supplementary Reactivity Control

The initial and reload core control requirements are met by use of the combined effects of the movable control rods, supplementary burnable poison, and variation of reactor coolant flow. The supplementary burnable poison is gadolinia (Gd_2O_3) mixed with UO_2 in selected fuel rods in selected fuel bundle.

4.1.4 Analysis Techniques

4.1.4.1 Reactor Internal Components

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

(11) ANSYS

Detail 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 proprietary program 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 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 Nuclear Energy Division was using a past revision of MASS. This revision is identified as revision "0" in the computer production library. The program operated on the Honeywell 6000 computer and is now retired.

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) used 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. Reissner'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 of the program was obtained from the General Electric Jet Engine Division. It was used on the Honeywell 6000 computer in GE/NED and is now retired.

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, operated on the Honeywell 6000 computer and is now retired.

4.1.4.1.3.3 History of Use

The program was developed by E. L. Wilson in 1965 (Reference 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 General Electric Company.

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 specified 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). The program operated on the Honeywell 6000 computer and is now retired.

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 orthotropic. 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 5). The NED version was used on the Honeywell 6000 computer and is now retired.

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 was 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 systems 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 method. Response spectrum solution is also available as an option.

4.1.4.1.6.2 Program Version and Computer

The DYSEA version was developed at GE by modifying the SAPIV program. The program operated on the Honeywell 6000 computer and is now retired. Capability was added to handle the hydrodynamic mass effect due to fluid-structure interaction in the reactor. It can handle three-dimensional dynamic problems 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 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 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 was maintained in GE/NED by Y. R. Rashid, one of the originators of the program. SHELL 5 operates on the UNIVAC 1108 computer and is now retired.

4.1.4.1.7.3 History of Use

SHELL 5 is a program developed by Gulf General Atomic Incorporated (Reference 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 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 $3S_m$ (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 9). The program operated on the Honeywell 6000 computer and is now retired.

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 11. The program was operative on the Honeywell 6000 which is now retired.

4.1.4.1.10.3 History of Use

This program was developed by Y. R. Rashid (Reference 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 was used at GE/NED in the channel cross section mechanical analysis.

4.1.4.1.11 ANSYS

4.1.4.1.11.1 Program Description

ANSYS is a general-purpose finite element computer program designed to solve a variety of problems in engineering analysis.

The ANSYS program features the following capabilities:

(1) Structural analysis including static elastic, plastic and creep, dynamic, seismic and dynamic plastic, and large deflection and stability analysis.

(2) One-dimensional fluid flow analyses.

(3) Transient heat transfer analysis including conduction, convection, and radiation with direct input to thermal-stress analyses.

(4) An extensive finite element library, including gaps, friction interfaces, springs, cables (tension only), direct interfaces (compression only), curved elbows, etc. Many of the elements contain complete plastic, creep, and swelling capabilities.

(5) Plotting - Geometry plotting is available for all elements in the ANSYS library, including isometric and perspective views of three-dimensional structures.

(6) Restart Capability - The ANSYS program has restart capability for several analyses types. An option is also available for saving the stiffness matrix once it is calculated for the structure, and using it for other loading conditions.

4.1.4.1.11.2 Program Version and Computer

The program is maintained by Swanson Analysis Systems, Inc. of Pittsburgh, Pennsylvania. The program operated on the Honeywell 6000 computer and is now retired.

4.1.4.1.11.3 History of Use

The ANSYS program has been used for productive analysis since early 1970. Users now include the nuclear, pressure vessels and piping, mining, structures, bridge, chemical, and automotive industries, as well as many consulting firms.

4.1.4.1.11.4 Extent of Application

ANSYS is used extensively in GE/NED for elastic and elastic-plastic analysis of the reactor pressure vessel, core support structures, reactor internals and fuel.

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. Conditions of 100% and 116% of rated power are analyzed corresponding to steady-state and short-term transient operation. Abnormal operation transients are also evaluated to assure that the damage limit of 1.0% cladding plastic strain is not violated. The strength theory, terminology, and strain-stress categories presented in the ASME Boiler and Pressure Vessel Code Section III are used as a guide in the mechanical design and stress analysis of the fuel rods.

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 10.

Section 4.4.4 also provides a complete stability analysis for the reactor coolant system.

4.1.4.4 Nuclear Engineering Analysis

The analysis techniques are described and referenced in subsection 4.3.3. The codes used in the analysis are:

Computer Code	Function
Lattice Physics Model	Calculates average few-group cross sections, bundle reactivities, and relative fuel rod powers within the fuel bundle.
BWR Reactor Simulator	Calculates three-dimensional nodal power distributions, exposures and thermal hydraulic characteristics as burnup progresses.

4.1.4.5 Neutron Fluence Determinations

Irradiation of reactor vessel by fast neutrons can be measured by flux wires sealed inside the surveillance capsule. However, the neutron flux level does not always peak at the location of the capsule. A lead factor relating the flux at the flux wires to the peak vessel flux is defined as the ratio of surveillance capsule flux to the peak flux at the vessel inside surface. While the lead factor is a function of core and vessel configurations, it is also dependent on the location of the capsule relative to other reactor internal components. The lead factor can be determined using neutron transport analysis calculations.

Neutron transport analysis can determine the neutron flux distribution in the core and near the reactor vessel by combining the results of two separate two-diminsional calculations. The first of these establishes the azimuthal and radial variation of flux at or around the core midplane. The second analysis determines the relative variation of flux with respect to elevations. The results of these two analyses are combined to provide a synthesized three-dimensional distribution of flux. The ratio of fluxes, or lead factor, between the surveillance capsule location and the peak vessel flux location is determined from this distribution. The methodology of flux calculations will be further discussed in Sections 4.3.2.8 and 4.3.2.9.

4.1.4.6 Thermal Hydraulic Calculations

The digital computer program uses a parallel flow path model to perform the steady-state BWR reactor core thermal-hydraulic analysis. Program input includes the core geometry, operating power, pressure, coolant flow rate and inlet enthalpy, and power distribution within the core. Output from the program includes core pressure drop, coolant flow distribution, critical power ratio, and axial variations of quality, density, and enthalpy for each channel type.

4.1.5 References

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4.2 FUEL SYSTEM DESIGN

Most of the following information is presented by reference to GESTAR II (Reference 1).

The following information was current at the time of initial fuel loading and receipt of the CPS Operating License. Changes in fuel design since that time are not reflected in the Q&R 490.1. However, CPS fuel design and analysis are in accordance with Reference 1.

The proposed reactor fuel design discussed in this section is identical to the Grand Gulf fuel design. Draft SER for Grand Gulf fuel design was written in April 1981 and the final SER was addressed in SER (Feb. 82) and SSER 4 (Feb. 85), affirming approval of the BWR/6 fuel design. Accordingly, that document is directly applicable to CPS without further review or reformatting of Section 4.2.

The letter from R. E. Engel to R. O. Meyer (Reference 10) provides the detailed information requested in Standard Review Plan Section 4.2. The seven issues listed in Question 490.1 are generic concerns applicable to the General Electric fuel design. Recognizing that these issues have been handled in the recent Grand Gulf SER, the same responses are repeated for CPS and added to Section 4.2 for completeness (Q&R 490.1).

Item 1 - Supplemental ECCS Analysis with NUREG-0630

The use of the NUREG-0630 materials models has no impact on the loss-of-coolant accident (LOCA) calculations for CPS since no perforations are calculated using either the NUREG-0630 or General Electric material models. Supplemental calculations using the materials models of NUREG-0630 along with the justification of the current GE position on cladding, swelling, and rupture are contained in References 6, 7, and 11.

Item 2 - Combined Seismic and LOCA Loads Analysis

As discussed in the letter from R. E. Engel to R. O. Meyer (Reference 10), the evaluation of combined seismic and LOCA loads is presented in GE report NEDE-21175-3-P-A, "BWR Fuel Assembly Evaluation of Combined SSE and LOCA Loadings," dated October 1984 (Reference 14).

Item 3 - Enhanced Fission Gas Release Analysis at High Burnups

This subject has been responded generically to the USNRC through GE Operating Reactors Licensing. Reference 8, a letter from R. E. Engel to the USNRC, contains the most recent position by GE on the subject. Plant specific numbers for Grand Gulf are tabulated below to supplement Table 2 of the reference letter to include BWR/6 plants. This table shows that for BWR/6 the impact of enhanced fission gas release can be entirely offset by existing PCT margins without taking credit for recently approved model improvements. Since PCT margins for CPS are expected to be at least as large as those for Grand Gulf, the position stated in the reference letter is also applicable to CPS.

General Electric Assessment of
NRC Fission Gas Correction Factor
for Grand Gulf

Plant	Fuel Type	Exposure (GWD/MT)	GE Evaluation of PCT Increase NRC Correction Factor (ºF)	Plant Margin ot 2200º F (ºF)	Overall Margin (ºF)
Grand Gulf	P8x8R	22	10	115	105
		28	30	186	156
		33	70	318	248
		39	130	436	306
		44	200	508	308

Item 4 - Fuel Rod Bowing

General Electric's fuel surveillance program observations relative to fuel rod bowing are described in the Reference 9 report together with the results of analytical evaluations of the probable extent of fuel rod bowing. Also presented are the results of an extensive thermal-hydraulic test program performed to assess the significance of rod bowing on fuel assembly thermal-hydraulic performance. Based on the presented information, General Electric concludes that fuel rod bowing does not constitute a viable failure mechanism or represent a significant safety concern for General Electric fuel in boiling water reactors.

Item 5 - Fuel Assembly Control Rod Guide Tube Wear Analysis

This appears to be a PWR-related technical issue. "Fuel assembly control rod guide tube" is not a part of standard BWR design.

Item 6 - Fuel Assembly Design Shoulder Gap Analysis

The analysis of fuel rod axial expansion is described in General Electric Licensing Topical Report, "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A-6 (Proprietary), April 1983.

The results of the analysis verify that the fuel rod is designed to accommodate predicted acceptable fuel and cladding differential expansion.

Item 7 - The Analysis of the Fuel Element Internal Pressure at End-of-Core Life

The internal pressure is used in conjunction with other loads on the fuel rod cladding when calculating cladding stresses and comparing these stresses to the design criteria. This analysis is described in General Electric Licensing Topical Report, NEDE-24011-P-A-6 (Proprietary), April 1983. The analysis result shows that the calculated stresses on cladding can be accommodated.

4.2.1 Design Bases

References to design bases are given in Subsection A.4.2.1 of Reference 1.

4.2.2 Description of Fuel Assembly and Associated Components

Reference to the fuel system description and design drawings are given in Subsection A.4.2.2 of Reference 1. The specific fuel system components are addressed in the following paragraphs.

4.2.2.1 Core Cell

Reference to the core cell description and design drawings is given in Subsection A.4.2.2 of Reference 1.

4.2.2.2 Fuel Assembly

Reference to the fuel assembly description and design drawings is given in Subsection A.4.2.2 of Reference 1.

Since the fuel rod cladding ballooning and rupture issue has been resolved for CPS, no further justification of reduction factors used by the General Electric Company will be provided with this response. (Q&R 490.2)

4.2.2.3 Fuel Bundle

Reference to the fuel bundle description and design drawings is given in Subsection A.4.2.2 of Reference 1.

4.2.2.4 Reactivity Control Assembly

4.2.2.4.1 Control Rods

The control rods perform the dual function of power shaping and reactivity control. A typical control rod design is shown in Figure 4.2-6.

Power distribution in the core is controlled during operation of the reactor by manipulating selected patterns of control rods. Control rod displacement tends to counterbalance steam void effects at the top of the core and results in significant power flattening.

The control rod consists of an array of stainless steel wings filled with boron-carbide powder. The control rods are 9.868 inches in total span and are separated uniformly throughout the core on a 12-inch pitch maximum. Each control rod is surrounded by four fuel assemblies.

The main structural components of a control rod are made of stainless steel and consist of a top handle, a bottom casting with a velocity limiter and control rod drive coupling, and four absorber wings.

Some control blade handles have rollers or buttons to provide guidance for control rod insertion and withdrawal. Some control blade handles have no rollers or pads. The control rods are cooled by the core bypass flow.

Operating experience has shown that control rods constructed as described above are not susceptible to dimensional distortions.

The boron-carbide (B_4C) powder in the absorber wings is compacted to about 70 percent of its theoretical density. The boron-carbide contains a minimum of 76.5 percent by weight natural boron. The boron-10 (B-10) minimum content in the boron powder is 18 percent by weight.

4.2.2.4.2 Velocity Limiter

The control rod velocity limiter is an integral part of the bottom assembly of each control rod. This engineered safeguard protects against high reactivity insertion rate by limiting the control rod velocity in the event of a control rod drop accident. It is a one-way device in that the control rod scram velocity is not significantly affected but the control rod dropout velocity is reduced to a permissible limit.

The hydraulic drag forces on a control rod are proportional to approximately the square of the rod velocity and are negligible at normal rod withdrawal or rod insertion speeds. However, during the scram stroke the rod reaches high velocity, and the drag forces must be overcome by the drive mechanism.

To limit control rod velocity during dropout, but not during scram, the velocity limiter is provided with a streamlined profile in the scram (upward) direction.

Thus, when the control rod is scrammed, water flows over the smooth surface of the upper conical element into the annulus between the guide tube and the limiter. In the dropout direction, however, water for the "original equipment" is trapped by the lower conical element and discharged through the annulus between the two conical sections. Because this water is jetted in a partially reversed direction into water flowing upward in the annulus, a severe turbulence is created, thereby slowing the descent of the control rod assembly to less than 3.11 ft/sec.

4.2.3 Design Evaluations

Compliance with the design bases is referenced in Subsection A.4.2.3 of Reference 1. The specific design evaluations are addressed in the following paragraphs. The Duralife-230 specific evaluation is in Reference 12, GE Marathon is in Reference 15, CR82M-1 is in Reference 16 and Marathon Ultra HD is in Reference 18.

Plant-specific and LOCA loading values are provided in Subsection 3.9.1.4.10 and Table 3.9-2(b). These values are bounded by the loading used in Reference 14. The liftoff analysis is provided in Reference 14. (Q&R 490.3)

4.2.3.1 Results of Fuel-Rod Thermal Mechanical Evaluations

Reference to the fuel-rod thermal mechanical evaluation is given in Subsection A.4.2.3 of Reference 1.

4.2.3.2 Results from Fuel Design Evaluations

Reference to the fuel design evaluations is given in Subsection A.4.2.3 of Reference 1.

4.2.3.3 Reactivity Control Assembly Evaluation (Control Rods)

4.2.3.3.1 Materials Adequacy Throughout Design Lifetime

The adequacy of the control rod materials throughout the design life was evaluated in the design of the control rods. The primary materials, B_4C powder, hafnium plate and austenitic stainless steel, have been found to perform adequately for the lifetime of the control rod.

4.2.3.3.2 Dimensional and Tolerance Analysis

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

4.2.3.3.3 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. Mechanical design allows for what little differential thermal growth can exist.

In addition, to further this end, dissimilar metals are avoided.

4.2.3.3.4 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.

If the collet remains open, which is unlikely, calculations indicate that the steady state control rod withdrawal velocity would be 10 ft/sec for a pressure-under line break, the limiting case for rod withdrawal.

4.2.3.3.5 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.

4.2.3.3.6 Effect of Blowdown Loads on Control Rod Channel Clearances

The fuel channel load resulting from an internally applied pressure is evaluated utilizing a fixed beam analytical model under a uniform load. Tests to verify the applicability of the analytical model indicate that the model is conservative. The control blade handle may be plain or have a roller or button. If the gap between channels is less than the blade handle dimension, the handle deflects the channel walls as it makes its way into the core. The friction force is a small percentage of the total force available to the control rod drives for overcoming such friction, and it is concluded that the main steam line break accident does not impede the insertability of the control rod.

4.2.3.3.7 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.

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 analyses were performed considering both a recirculation line break and a steam line break, events which result in the largest hydraulic loading 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.

4.2.3.3.7.1 First Mode of Failure

The first mode of failure is evaluated by the addition of all the stresses resulting from the maximum loads for the faulted condition. This results in the maximum theoretical stress value for that condition. Making a linear supposition of all calculated stresses and comparing this value to the allowable limit defined by the ASME Boiler and Pressure Vessel Code yields a factor of safety of approximately 3. For faulted conditions, the factor of safety is approximately 4.2.

4.2.3.3.7.2 Second Mode of Failure

Evaluation of the second mode of failure is gases based on clearance reduction between the guide tube and the control rod. The minimum allowable clearance is about 0.1 inch. This assumes maximum ovality and minimum diameter of the guide tube and the maximum control rod dimension. The analysis showed that if the approximate 6000 psi for the faulted condition were entirely the result of differential pressure, the clearance between the control rod and the guide tube would reduce by a value of approximately 0.01 inch. This gives a design margin of 10 between the theoretically calculated maximum displacement and the minimum allowable clearance.

4.2.3.3.8 Analysis of Guide Tube Design

Two types of instability were considered in the analysis of guide tube design. The first was the classic instability associated with vertically loaded columns. The second was the diametral collapse when a circular tube experiences external to internal differential pressure.

The limiting axially applied load is approximately 77,500 lb resulting in a material compressive stress of 17,450 psi (code allowable stress). Comparing the actual load to the yield stress level gives a design margin greater than 20 to 1. From these values it can be concluded that the guide tube is not an unstable column.

When a circular tube experiences external to internal differential pressure, two modes of failure are possible depending on whether the tube is "long" or "short". In the analysis here, the guide tube is taken to be an infinitely long tube with the maximum allowable ovality and minimum wall thickness. The conditions will result in the lowest critical pressure calculation for the guide tube (i.e., if the tube was "short," the critical pressure calculation would give a higher number). The critical pressure is approximately 140 psi. However, if the maximum allowable stress is reached at a pressure lower than the critical pressure, then that pressure is limiting. This is the case for a BWR guide tube. The allowable stress of 17,450 psi will be reached at approximately 93 psi. Comparing the maximum possible pressure differential for a steam line break to the limiting pressure of 93 psi gives a design margin greater than 3 to 1. Therefore, the guide tube is not unstable with respect to differential pressure.

4.2.3.3.9 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.

4.2.4 Testing, Inspection and Surveillance

Descriptions of fuel assembly testing, inspection and surveillance are referenced in Subsection A.4.2.4 of Reference 1 and described in References 2 through 5.

4.2.4.1 Fuel, Hardware, and Assembly

Reference to fuel, hardware and assembly inspection and testing is given in Subsection A.4.2.4 of Reference 1.

CPS has stainless steel tubes in its main condenser and feedwater heaters rather than the copper-bearing tubes that have been an issue during the safety review of some other BWR plants. Since there is no source of potentially corrosion-causing copper in the CPS condensate/feedwater systems, the postirradiation surveillance need not be required. (Condenser and feedwater tube materials are identified in Subsection 10.4.7.2.3 and Table 10.4-2.) (Q&R 490.6)

4.2.4.2 Testing and Inspection (Enrichment and Burnable Poison Concentration)

Reference to the testing and inspection of enrichment and burnable poison concentrations is given in Subsection A.4.2.4 of Reference 1.

4.2.4.3 Surveillance Inspection and Testing of Irradiated Fuel Rods

Reference to the surveillance, inspection and testing of irradiated fuel rods is given in Subsection A.4.2.4 of Reference 1.

4.2.5 Operating and Developmental Experience

For the initial core, CPS had 120-mil channels that were expected to have creep deflections of 33% to 35% of the deflections of 80-mil channels having the same operating history. (See NEDE-21354-P, Paragraph 4.2.4.2.) This estimate was based both on analysis and operating experience. In addition, CPS has a S-Lattice reactor core. With this core, control rod drive friction tests will be performed as specified in Clinton's Safety Evaluation Report (NUREG-0853, Supp. 5, pp. 4-1 and 4-2). (Q&R 490.4)

For the current cycle channel design see Appendix 15D, Reload Analysis.

4.2.6 Isotope Test Assemblies (ITAs)

Isotope Test Assemblies (ITAs), also referred to as GE14i assemblies, were installed in Reload 12 for Cycle 13 at specified core locations. The GE14i fuel assemblies are designed for mechanical, nuclear, and thermal-hydraulic compatibility with the GE14 fuel design. The external envelope of the fuel assembly is virtually identical to the GE14 fuel assembly currently supplied to Clinton. The nuclear characteristics of these GE14i ITAs are compatible with those of the current GE14 fuel loaded at Clinton. The GE14i ITAs are evaluated in Reference 17 and reload specific analyses. The GE14i ITAs were removed in Reload 15 for Cycle 16.

Exelon Nuclear has agreed to participate with Global Nuclear Fuel – Americas, LLC (GNF) and GE-Hitachi Nuclear Energy Americas, LLC (GEH) in the bulk generation of cobalt-60 isotope targets. The ITAs will contain natural cobalt (metal), which is an element composed of 100% Co-59, a stable isotope. While in the reactor, the Co-59 atoms absorb neutrons and are converted into Co-60 atoms. After removal from the core, the cobalt isotope rods containing Co-60 are shipped to GEH to process the target rods for various applications. No removal of Co-60 from the target rods will take place at CPS. Separation of the Co-60 sources will only take place at GEH facilities.

4.2.7 References

1. "General Electric Standard Application for Reactor Fuel" NEDE-24011-P-A, latest approved revision.

2. Letter, J. S. Charnley (GE) to C. H. Berlinger (NRC), "Post-Irradiation Fuel Surveillance Program," November 23, 1983.

3. Letter, L. S. Rubenstein (NRC) to R. L. Gridley (GE). "Post-Irradiation Fuel Surveillance," January 18, 1984.

4. Letter, J. S. Charnley (GE) to L. S. Rubenstein (NRC), "Fuel Surveillance Program," February 29, 1984.

5. Letter, J. S. Charnley (GE) to L. S. Rubenstein (NRC), "Addition Details Regarding Fuel Surveillance Program," May 25, 1984.

6. R. H. Buchholz (GE), letter to L. S. Rubenstein (NRC). Impact of Large Rupture Strains on BWR LOCA Analysis, August 14, 1981.

7. G. G. Sherwood (GE), letter to L. S. Rubenstein (NRC), Impact of Large Rupture Strains on BWR LOCA Analysis, August 14, 1981.

8. R. E. Engel, letter to USNRC, Extension of ECCS Performance Limits, May 6, 1981.

9. R. J. Williams, Assessment of Fuel-Rod Bowing in General Electric Boiling Water Reactors, NEDE-24284-P and NEDO24284, August 1980.

10. R. E. Engel (GE), letter to R. 0. Meyer (NRC), Conformance to SRP 4.2, August 11, 1981.

11. J. Quirk (GE), letter to L. S. Rubenstein (NRC), General Electric Analytical Model for Calculation of Local Oxidation and LOCA Analysis, September 14, 1981.

12. Safety Evaluation of the General Electric Duralife-230 Control Rod Assembly, NEDE-22290-P-A, Supplement 3, May 1988.

13. “Generic Reload Fuel Application,” NEDE-24011-P-A-6, April 1983.

14. “BWR Fuel Assembly Evaluation of Combined SSE and LOCA Loadings,” NEDE-21175-3-P-A, October 1984.

15. “GE Marathon Control Rod Assembly,” NEDE-31758P-A, Class III, dated October 1991.

16. NRC SER evaluation of ASEA-ATOM Topical Report on Control Rod Blades (TACS 60093) dated 20 February 1986.

17 "Safety Analysis Report to Support Introduction of GE 14: Isotope Test Assemblies (ITA’s) in Clinton Power Station”, NEDC – 33505P, May 2009 (IP-F-0159)

18 NEDE–33284 Supplement 1P–A, Revision 1, March 2012, Marathon-Ultra Control Rod Assembly Licensing Topical Report

4.3 NUCLEAR DESIGN

Most of the information in Section 4.3 is provided by reference to GESTAR II (Reference 1). Any additions or differences are provided below. Nuclear design information supporting Single Loop Operation is contained in Chapter 15, Appendix B, and design information supporting Maximum Extended Operating Domain and Feedwater Heater Out-of-Service is in Chapter 15, Appendix C. Specific fuel and core design information for reload cycles can be found in Appendix 15D, Reload Analysis.

4.3.1 Design Bases

The nuclear core design bases are discussed in Subsection A.4.3.1 of Reference 1.

4.3.1.1 Safety Design Bases

The safety design bases are discussed in Subsection A.4.3.1.1 of Reference 1.

4.3.1.2 Plant Performance Design Bases

The plant performance design bases are discussed in Subsection A.4.3.1.2 of Reference 1.

4.3.2 Description

The nuclear core description is provided in Subsection A.4.3.2 of Reference 1, with the exception of the subsections below.

4.3.2.1 Nuclear Design Description

The nuclear design description is provided in Subsection A.4.3.2.1 of Reference 1. The reference initial core loading pattern is shown in Figure 4.3-1. For the current cycle core, the reference core loading pattern is shown in Appendix 15D, Reload Analysis.

4.3.2.2 Power Distribution

Power distribution anomalies are discussed in Subsection A.4.3.2.2.4 of Reference 1.

Stringent inspection procedures are planned to ensure the correct assembly of the reactor core. Although a misplacement of a bundle in the core would be a very improbable event, calculations have been performed in order to determine the effects of such accidents on Linear Heat Generation Rate and Critical Power Ratio. These results are presented in Chapter 15.

4.3.2.3 Reactivity Coefficients

Reference to reactivity coefficients is given in Subsection A.4.3.2.3 of Reference 1.

4.3.2.4 Control Requirements

Control requirements are discussed in Subsection A.4.3.2.4 of Reference 1.

4.3.2.4.1 Shutdown Reactivity

Information on shutdown reactivity is provided in Subsection A.4.3.2.4.1 of Reference 1. The cold shutdown margin for the initial core reference core loading pattern is demonstrated in Table 4.3-7. The cold shutdown margin for the current core reference loading pattern is provided in Appendix 15D, Reload Analysis.

4.3.2.4.2 Reactivity Variations

Information on reactivity variations is referenced in Subsection A.4.3.2.4.2 of Reference 1. The combined effects of the individual constituents of reactivity for the initial core are accounted for in each value of k-eff provided in Table 4.3-7. The reactivity variations for the current cycle are provided in Appendix 15D, Reload Analysis.

4.3.2.5 Control Rod Patterns and Reactivity Worths

Initial cycle control rod patterns and reactivity worths are discussed in GE Databooks; 23A1829 Cycle Management Report and 23A1762 Startup Data respectively.

Details on rod motion controls and analyzed control rod motion errors are discussed in Subsections 7.7.1.2 (RCIS), 7.6.1.7 (RPCS) and 15.4 (Reactivity and Power Distribution Anomalies).

For BWR plants, control rod patterns are not uniquely specified in advance; rather, during normal operation, the control rod patterns are selected based on the measured core power distributions. All rod patterns will be such that the core power distribution limits are met throughout the cycle.

Typical control rod patterns and reactivity worths are calculated during the design phase for each operating cycle to insure that all safety and performance criteria are satisfied. However, as stated above, actual control rod patterns used are based on the actual core power distribution. Appendix 15D, Reload Analysis, reports cycle specific core power distribution limits and reactivity worths (see reference 6 in section 15D).

4.3.2.6 Criticality of Reactor During Refueling

A discussion of reactor criticality during refueling is given in Subsection A.4.3.2.6 of Reference 1.

4.3.2.7 Stability

Stability is discussed in Subsection A.4.3.2.7 of Reference 1.

4.3.2.8 Vessel Irradiations

The lead factor was calculated using the two-dimensional discrete ordinates transport code described in Subsection 4.1.4.5. The discrete ordinates code was used in a distributed source mode with cylindrical geometry. The geometry described seven regions with the core modeled as two homogenized regions. The coolant water region between the core and the shroud contained saturated water at 550°F. Subcooled water at 530°F and 1040 psia was used for the coolant between the shroud and the vessel. In the region between the shroud and the vessel, the presence of the jet pumps was ignored. The material compositions for the stainless steel shroud and the carbon steel vessel contained the mixtures by weight as specified in the ASME material specifications for ASME SA240, 304L, and ASME SA533 grade B. A diagram showing the regions and dimensions are shown in Figure 4.3-29.

The distributed source which can be separated in space and energy, was obtained from the core power shape and the neutron spectra. The integral over space and energy was normalized to the total number of neutrons in the core region. The core region is defined as a 1 centimeter thick cross-section of the core with no transverse leakage. The power in this core region is set equal to the maximum power in the axial direction.

Dosimetry located on the inside surface of the vessel was removed after the first fuel cycle and tested to determine the flux at that location. The lead factor relating the dosimeter location to the peak location was used to calculate the peak vessel inside surface flux. Assuming an 80% capacity factor, or 32 effective full power years (EFPY) in 40 years of operation, the fluence for this operating period was estimated. The measured dosimeter flux, and calculated peak flux and fluence are shown in Table 4.3-5. The calculated cycle average neutron flux at the maximum core radius is shown in Table 4.3-6.

4.3.2.9 EPU Flux Calculations

The flux calculations are carried out with the two-dimensional transport code DORT. DORT is a deterministic code using discrete-ordinates method to solve the integro-differential form of the Boltzmann transport equation. The working library for the Clinton DORT calculation is a 26-group cross-section set where the angular dependency of scattering cross-sections is approximated by a third-order Legendre polynomial expansion. As explained in 4.1.4.5, two separate two-dimensional calculations are performed in order to achieve the synthesized three-dimensional flux distributions.

The azimuthal flux distribution is obtained with a calculation model in (R, θ) geometry, assuming quarter-core symmetry with reflective boundary conditions at 0° and 90°. A schematic view of the (R, θ) model is shown in Figure 4.3-30. The model incorporates three core regions, the shroud, water regions inside and outside the shroud, and the vessel wall. In the region between the shroud and the vessel, each jet pump riser or mixer pipe is modeled as a homogenized mixture of steel and water. The power shape and void distribution of a simulated Clinton EPU equilibrium core are used to generate neutron source density and core material compositions. The output of this calculation provides the flux distribution as a function of azimuth and radial distance at core midplane.

The axial flux distribution is calculated with (R, Z) geometry. The core configuration is modeled based on parameters at azimuth 28.3° where the edge of the core is closest to the vessel wall. Output from the (R, Z) calculation provides flux variation as a function of elevation. An axial peaking factor, or the ratio of flux at peak elevation to that at core midplane, is readily assessed. The peak vessel flux is then obtained by multiplying the peak (R, θ) flux at vessel inner radius by the axial peaking factor at the same radius.

Results of these calculations indicate that, for the Clinton EPU equilibrium cycle, the vessel flux peaks at azimuth 65.25°, in an elevation 75.9 inches above the bottom of active fuel (BAF), with a flux level 5.54E9 n/cm^2-sec. Calculated flux level at the 3° capsule location is 5.18E9 n/cm^2-sec. Therefore the lead factor is 5.18E9/5.54E9 or 0.94.

The methodology described above is in accordance with the recommendations of Regulatory Guide (RG) 1.190, which provides state of the art calculation and measurement procedures that are acceptable to the NRC for determining Reactor Pressure Vessel neutron fluence. Future evaluations of RPV fluence will be completed using a method in accordance with the recommendations of RG 1.190 (as noted in Reference 2).

4.3.3 Analytical Methods

The analytical methods and nuclear data used to determine nuclear characteristics are provided in Subsection A.4.3.3 of Reference 1.

4.3.4 Changes

Details of design changes are provided in Subsection A.4.3.4 of Reference 1.

4.3.4.1 Reactor Core

Refer to Subsection A.4.3.4 of Reference 1.

4.3.5 References

1. "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A, latest approved revision.

2. NRC letter from D.V. Pickett approving Amendment 157 to Clinton Facility Operating License No. NPF-62, dated 8-12-03.

Tables 4.3-1 Through 4.3-4

Have Been Deleted Intentionally

Table 4.3-5
DOSIMETER AND VESSEL PEAK FLUXES AND FLUENCES

Time at Power:

EOC1	0.99 EFPY - 3.13×10^7 seconds
32 EFPY	32 EFPY - 1.01×10^9 seconds

Lead Factors:

Inside Surface (I.D.)	0.67

Dosimeter Flux (n/cm^2 - s) 4.6×10^9 (nominal) 5.7×10^9 (upper bound)

FLUENCE (n/cm^2):

	NOMINAL	UPPER BOUND
EOC1 Peak I.D.	2.1×10^{17}	2.7×10^{17}
32 EFPY Peak I.D.	6.9×10^{18}	8.7×10^{18}

Table 4.3-6
CALCULATED CYCLE AVERAGE NEUTRON FLUX AT THE MAXIMUM CORE RADIUS (88.6 in.)

GROUP	LOWER ENERGY BOUND (eV)	FLUX (n/cm^2-sec)
1	10.0×10^6	2.4×10^{10}
2	6.065×10^6	3.3×10^{11}
3	3.679×10^6	1.2×10^{12}
4	2.231×10^6	2.5×10^{12}
5	1.353×10^6	2.7×10^{12}
6	8.208×10^5	2.5×10^{12}
7	4.979×10^5	2.2×10^{12}
8	3.020×10^5	1.5×10^{12}
9	1.832×10^5	1.3×10^{12}
10	6.738×10^4	1.8×10^{12}
11	2.479×10^4	1.3×10^{12}

TABLE 4.3-7
CALCULATED CORE EFFECTIVE MULTIPLICATION
AND CONTROL SYSTEM WORTH - NO VOIDS, 20°C[1] (INITIAL CORE)

Beginning of Cycle, K-effective		
	Uncontrolled	1.106
	Fully Controlled	0.924
	Strongest Control Rod Out	0.969
R,	Maximum Increase in Cold Core	
Reactivity with Exposure into Cycle, ΔK		0.0

Note: (1) This table provides the values for the initial cycle core only. The values for the current cycle are provided in Appendix 15D, Reload Analysis.

4.4 THERMAL AND HYDRAULIC DESIGN

Most of the information of Section 4.4 is provided by reference to GESTAR II (Reference 1). Any additions or differences are provided below. Thermal and hydraulic information supporting Single Loop Operation is contained in Chapter 15, Appendix B, and design information supporting both Maximum Extended Operating Domain and Feedwater Heater Out-of-Service is in Chapter 15, Appendix C. The information given below is the baseline information in support of initial cycle operation. The baseline information in support of the current cycle is provided in Appendix 15D, Reload Analysis.

4.4.1 Design Basis

The thermal and hydraulic design bases are referenced in Sub-section A.4.4.1 of Reference 1. The design steady-state Minimum Critical Power Ratio (MCPR) operating limit and the peak Linear Heat Generation Rate (LHGR) for the initial core are provided in Table 4.4-1. Appendix 15D, Reload Analysis, provides the current cycle operating limit MCPR and LHGR.

4.4.1.1 Safety Design Bases

The safety design bases are discussed in Subsection A.4.4.1 of Reference 1.

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

Requirements for steady-state operating conditions are discussed in Subsection A.4.4.1.2 of Reference 1.

4.4.1.4 Requirements for Anticipated Operational Occurrences (AOOs)

Requirements for transient conditions are defined in Subsection A.4.4.1.4 of Reference 1.

4.4.1.5 Summary of Design Bases

The design bases are summarized in Subsection A.4.4.1.4 of Reference 1.

4.4.2 Description of Thermal-Hydraulic Design of the Reactor Core

A description of the thermal-hydraulic design of the reactor core is referenced in Subsection A.4.4.2 of Reference 1, with the exception of the subsections below.

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

4.4.2.1 Summary Comparison

A tabulation of thermal and hydraulic parameters for the initial core is given in Table 4.4-1. A comparison of this reactor with others of similar design is given in Table 4.4-1.

4.4.2.2 Critical Power Ratio

Reference to the critical power ratio and the model used to calculate this ratio is given in Subsection A.4.4.2.2 of Reference 1.

4.4.2.3 Linear Heat Generation Rate (LHGR)

The LHGR safety limit is referenced in Subsection A.4.4.2.3 of Reference 1.

4.4.2.4 Void Fraction Distribution

The core average and maximum exit void fractions in the initial core at rated condition are given in Table 4.4-1. The axial distribution of core void fractions for the average radial channel and the maximum radial channel (end of node value) for the initial core are given in Table 4.4-2. The core average and maximum exit value is also provided. Similar distributions for steam quality are provided in Table 4.4-3. The core average axial power distribution used to produce these tables is given in Table 4.4-2a.

4.4.2.5 Core Coolant Flow Distribution and Orificing Pattern

The distribution of core coolant flow among the fuel assemblies is described in Subsection A.4.4.2.5 of Reference 1.

4.4.2.6 Core Pressure Drop and Hydraulic Loads

Refer to Subsection A.4.4.2.6 of Reference 1.

4.4.2.7 Correlation and Physical Data

Reference to correlation and physical data is given in Subsection A.4.4.2.7 of Reference 1.

4.4.2.8 Thermal Effects of Operational Transients

The thermal effects of operational transients are referenced in Subsection A.4.4.2.8 of Reference 1.

4.4.2.9 Uncertainties in Estimates

Uncertainties in thermal-hydraulic parameters are discussed in Subsection A.4.4.2.9 of Reference 1.

4.4.2.10 Flux Tilt Considerations

For flux tilt considerations, refer to Subsection A.4.4.2.10 of Reference 1.

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 5.4-1.

4.4.3.1.2 Reactor Coolant System Thermal Hydraulic Data

The steady state distribution of temperature, pressure, and flow rate for each flowpath in the reactor coolant system is shown in Figure 5.1-1.

4.4.3.1.3 Reactor Coolant System Geometric Data

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, 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 section 5.4.1. Section 4.4.3.3 gives the operating 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. The power-flow map for the power range of operation is shown in Figure 4.4-5. 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, A The operating state of the reactor moves along this line for the normal control rod withdrawal sequence in the absence of recirculation pump operation.

The MELLLA upper boundary load line limit or rated thermal power (whichever is less): This load line passes through 100% reactor thermal power (3473 MWth) at 99% core flow. The operating state for the reactor follows this rod line (or similar ones) during recirculation flow changes with a fixed control rod pattern; however, rated power may not be exceeded. EPU/MELLLA upper load line is based on constant xenon concentration at EPU power and 99% core flow.

Cavitation Protection Line This line results from the recirculation pump, flow control valve, and jet pump NPSH requirements.

The Extended Power Uprate (EPU/MELLLA) adds power-flow areas to the standard power-flow operating map. A discussion of the EPU/MELLLA and the supporting analyses is found in Chapter 15, Appendix C.

4.4.3.3.1.1 Performance Characteristics

Other performance characteristics shown on the power-flow operating map are:

Constant Rod Lines These lines show the change in power associated with flow changes, while maintaining constant control rod position.

Constant Position Lines for Flow Control Valve, B, C and D These lines show the change in flow associated with power changes while maintaining flow-control valves at a constant position.

4.4.3.3.2 Regions of the Power Flow Map

Region I This region defines the system operational capability with the recirculation pumps and motors being driven by the low frequency motor-generator set at 25% speed. Flow is controlled by the flow control valve and power changes, during normal startup and shutdown, will be in this region. The normal operating procedure is to start up along curve C - FCV wide open at 25% speed. The switching sequence from the low frequency m-g set to 100% speed will be done in this region with the final reactor core flow resulting on curve D.

Region II This is the low power area of the operating map where cavitation can be expected in the recirculation pumps, jet pumps, or flow control valves. Operation within this region is precluded by system interlocks which trip the main motor from the 100% speed power source to the 25% speed power source.

Region III This represents the normal operating zone of the map where power changes can be made, by either control rod movement or by core flow changes, through the use of the flow control valve.

4.4.3.3.3 Design Features for Power-Flow Control

The following limits and design features are employed to maintain power-flow conditions to the required values shown in Figure 4.4 - 5.

(1) Minimum Power Limits at Intermediate and High Core Flows. To prevent cavitation in the recirculation pumps, jet pumps, and flow control valves, the

recirculation system is provided with an interlock to trip off the 100% speed power source and close the 25% speed power source if the difference between steam line temperature and recirculation pump inlet temperature is less than a preset value (typically 611°F). This differential temperature is measured using high accuracy RTDs with a sensing error of less than 0.2°F at the two standard deviation (2σ) confidence level. This action is initiated electronically through a 15-second time delay. The interlock is active while in both the automatic and manual operation modes.

(2) Minimum Power Limit at Low Core Flow. During low power, low loop flow operations, the temperature differential interlock may not provide sufficient cavitation protection to the flow control valves. Therefore, the system is provided with an interlock to trip off the 100% speed power source and close the 25% speed power source if the feedwater flow falls below a preset level (i.e., 3.13 × 106 lb/hr or 25.2%). The feedwater flow rate is measured by existing process control instruments. The speed change action is electronically initiated. This interlock is active during both automatic and manual modes of operation.

(3) Pump Bearing Limit. For pumps as large as the recirculation pumps, practical limits of pump bearing design require that minimum pump flow be limited to 20% of rated. To assure this minimum flow, the system is designed so that the minimum flow control valve position will allow this rate of flow.

(4) Valve Position. To prevent structural or cavitation damage to the recirculation pump due to pump suction flow starvation, the system is provided with an interlock to prevent starting the pumps, or to trip the pumps if the suction or discharge block valves are at less than 90% open position. This circuit is activated by a position limit switch and is active before the pump is started, during manual operation mode, and during automatic operation mode.

4.4.3.3.3.1 Flow Control

The principal modes of normal operation with valve flow control-Low Frequency Motor Generator (LFMG) set are summarized as follows: the recirculation pumps are started on the 100% speed power source in order to unseat the pump bearings. Suction and discharge block valves are full open and the flow control valve is in the ≤10% position. When the pump is near full speed, the main power source is tripped and the pump allowed to coast down to approximately 25% speed where the LFMG set will power the pump and motor. The flow control valve is then opened to the maximum position at which point reactor heatup and pressurization can commence. When operating pressure has been established, reactor power can be increased. This power-flow increase will follow a line within Region I of the flow control map shown in Figure 4.4-5.

When reactor power is greater than approximately 25.2% of rated, the low feedwater flow interlock is cleared and the main recirculation pumps can be switched to the 100% speed power source. The flow control valve is closed to the ≤10% position before the speed change to prevent large increases in core power and a potential flux scram. A FCV position permissive switch is located on the valve to prevent unexpected speed change without closure first. Administrative controls disable the interlock and close the flow control valve to ≤10% prior to expected speed changes. This operation occurs within Region II of the operating map. The system is then brought to the desired power-flow level within the normal operating area of the map (Region III) by opening the flow control valves and by withdrawing control rods.

Control rod withdrawal with constant flow control valve position will result in power-flow changes along lines of constant c sub (v) (constant position). Flow control valve movement with constant control rod position will result in power-flow changes along, or nearly parallel to, the rated flow control line.

4.4.3.4 Temperature-Power Operating Map (PWR)

Not applicable.

4.4.3.5 Load-Following Characteristics

Deleted

4.4.3.6 Thermal and Hydraulic Characteristics Summary Table

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

4.4.4 Evaluation

Refer to Subsection A.4.4.4 of Reference 1. The results of the stability analysis for the initial core are given in Table 4.4-11 and Figures 4.4-6 through 4.4-9. The results of the stability analysis for the current cycle are provided in Appendix 15D, Reload Analysis.

4.4.4.1 Critical Power

Reference to the GEXL critical power correlation is given in Subsection A.4.4.4.1 of Reference 1.

4.4.4.2 Core Hydraulics

Core hydraulic models and correlations are discussed in Subsection A.4.4.4.2 of Reference 1.

4.4.4.3 Influence of Power Distributions

The influence of power distributions on the thermal-hydraulic design is referenced in Subsection A.4.4.4.3 of Reference 1.

4.4.4.4 Core Thermal Response

The thermal response of the core is referenced in Subsection A.4.4.4.4 of Reference 1.

4.4.4.5 Analytical Methods

Analytical methods used in determining the thermal and hydraulic characteristics of the core are discussed in Subsection A.4.4.4.5 of Reference 1.

4.4.4.6 Thermal-Hydraulic Stability Analysis

Reference to the thermal-hydraulic stability analysis is given in Subsection A.4.4.4.6 of Reference 1.

4.4.5 Testing and Verification

Refer to Subsection A.4.4.5 of Reference 1.

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 reactor vessel sensors are discussed in Subsections 7.7.1.1 and 7.6.1.5.

4.4.7 References

1. "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A, latest approved revision

TABLE 4.4-1
THERMAL AND HYDRAULIC DESIGN CHARACTERISTICS
OF THE INITIAL REACTOR CORE(1)

General Operating Conditions	CLINTON (218-624)	PERRY (238-748)	GRAND GULF (251-800)
Reference design thermal output, Mwt	2894	3579	3833
Power level for engineered safety features, Mwt	3016	3758	4025
Steam flow rate, at 420° final feedwater temperature, millions lb/hr	12.453	15.400	16.49
Core coolant flow rate, millions lb/hr	84.5	104.0	112.5
Feedwater flow rate, millions lb/hr	12.428	15.367	16.46
System pressure, nominal in steam dome, psia	1040	1040	1040
System pressure, nominal core design, psia	1055	1055	1055
Coolant saturation temperature at core design pressure, °F	551	551	551
Average power density, kW/liter	52.4	54.1	54.1
Maximum Linear Heat Generation Rate, kW/ft	13.4	13.4	13.4
Average Linear Heat Generation Rate, kW/ft	5.7	5.9	5.9
Core total heat transfer area, ft^2 78,398	61,151	73,303	
Maximum heat flux, Btu/hr-sq ft 361,600	361,600	361,600	
Average heat flux, Btu/hr-sq ft 159,800	154,600	159,500	
Design operating minimum critical power ratio	1.20	1.20	1.20
Core inlet enthalpy at 420°F FFWT, Btu/b	527.8	527.7	527.9
Core inlet temperature, at 420°F FFWT, °F	533	533	533
Core maximum exit voids within assemblies, %	76.0	79.0	76
Core average void fraction, active coolant	0.411	0.4140	0.412
Maximum fuel temperature, °F	3435	3435	3435
Active coolant flow area per assembly, $in.^2$	15.164	15.164	5.164
Core average inlet velocity, ft/sec	6.82	6.98	7.07
Maximum inlet velocity, ft/sec	7.90	8.54	8.57
Total core pressure drop, psi	25.26	26.4	26.74
Core support plate pressure drop, psi	20.84	22.0	22.32
Average orifice pressure drop			
Central region, psi	5.41	5.71	5.78

TABLE 4.4-1 (Cont'd)
THERMAL AND HYDRAULIC DESIGN CHARACTERISTICS
OF THE INITIAL REACTOR CORE[1]

General Operating Conditions	CLINTON (218-624)	PERRY (238-748)	GRAND GULF (251-800)
Peripheral region, psi	17.95	18.68	19.16
Maximum channel pressure loading, psi	14.52	15.40	15.59
Average-power assembly channel pressure loading (bottom), psi	13.28	14.1	14.22
Shroud support ring and lower shroud pressure loading	24.84	25.7	25.12
Upper shroud pressure loading, psi	4.0	3.7	2 .8

Note: (1) This table provides an historical comparison of the Clinton Power Station reactor core design with other BWR/6 reactors, based on the initial core design. This table is not maintained current.

TABLE 4.4-2
VOID DISTRIBUTION(Initial Core)

CORE AVERAGE VALUE	=	0.411
MAXIMUM EXIT VALUE	=	0.759
ACCTIVE FUEL LENGTH	=	150 INCHES

	Node	Core Average (Average Node Value)	Maximum Channel (End of Node Value)
Bottom	1	0.000	0.000
of Core	2	0.000	0.005
	3	0.008	0.063
	4	0.040	0.165
	5	0.101	0.269
	6	0.174	0.357
	7	0.249	0.433
	8	0.319	0.492
	9	0.377	0.540
	10	0.425	0.577
	11	0.463	0.607
	12	0.494	0.632
	13	0.520	0.653
	14	0.541	0.669
	15	0.559	0.684
	16	0.575	0.697
	17	0.589	0.709
	18	0.603	0.721
	19	0.615	0.731
	20	0.627	0.741
	21	0.637	0.749
	22	0.645	0.754
	23	0.650	0.758
Top of	24	0.653	0.759
Core			

TABLE 4.4-2a

AXIAL POWER DISTRIBUTION USED TO GENERATE VOID AND QUALITY DISTRIBUTIONS (Initial Core)

	Node	Axial Power Factor
Bottom of Core		
	1	0.38
	2	0.69
	3	0.93
	4	1.10
	5	1.21
	6	1.30
	7	1.47
	8	1.51
	9	1.49
	10	1.44
	11	1.36
	12	1.28
	13	1.16
	14	1.06
	15	1.01
	16	0.97
	17	0.94
	18	0.97
	19	0.96
	20	0.91
	21	0.77
	22	0.59
	23	0.38
Top of Core	24	0.12

TABLE 4.4-3
FLOW QUALITY DISTRIBUTION (Initial Core)

CORE AVERAGE VALUE	=	0.077
MAXIMUM EXIT VALUE	=	0.268
ACTIVE FUEL LENGTH	=	150 INCHES

	Node	Core Average (Average Node Value)	Maximum Channel (End of Node Value)
Bottom	1	0.000	0.000
of Core	2	0.000	0.000
	3	0.000	0.002
	4	0.001	0.009
	5	0.004	0.020
	6	0.010	0.036
	7	0.019	0.054
	8	0.030	0.073
	9	0.042	0.092
	10	0.053	0.110
	11	0.065	0.127
	12	0.076	0.143
	13	0.086	0.158
	14	0.095	0.171
	15	0.103	0.184
	16	0.112	0.197
	17	0.120	0.208
	18	0.128	0.221
	19	0.136	0.233
	20	0.144	0.244
	21	0.151	0.254
	22	0.156	0.262
	23	0.160	0.266
Top	24	0.162	0.268
of Core			

Tables 4.4-4 Through 4.4-7

Have Been Deleted Intentionally

TABLE 4.4-8
REACTOR COOLANT SYSTEM GEOMETRIC DATA

		Flow Path Length (in.)	Height and Liquid Level (in.)	Elevation of Bottom of Each Volume* (in.)	Minimum Flow Areas (sq ft)
A.	Lower Plenum	208.5	208.5 208.5	-166.5	75.0
B.	Core	164.5	164.5 164.5	42.0	121.5 includes bypass (Initial Core)
C.	Upper Plenum and Separators	174.5	174.5 174.5	206.5	46.5
D.	Dome (Above Normal Water Level)	284.0	284.0	381.0	259.0
E.	Downcomer Area	314.0	314.0 314.0	-33.0	53.5
F.	Recirculation Loops and Jet Pumps	110.0ft (one loop)	383.0 383.0	-378	100.5in^2

*Reference Point is recirculation nozzle outlet centerline.

TABLE 4.4-9
LENGTHS AND SIZES OF SAFETY INJECTION LINES

SYSTEM LINES	SIZE (inches)	LENGTH (ft)
HPCS- Pump discharge to RPV	16	4
	14	106
	12	5
	10	108
RHR-"A" - Pump discharge to RPV	14	334
	12	86
	10	5
RHR-"B" - Pump discharge to RPV	14	187
	12	240
	10	5
RHR-"C" - Pump discharge to RPV	14	133
	12	240
	10	5
LPCS - Pump discharge to RPV	12	176
	10	115
	14	2

Table 4.4-10 has been deleted.

TABLE 4.4-11
STABILITY ANALYSIS RESULTS[1]

Rod Line Analyzed	
Natural Circulation	51.5% rated power
Rod Pattern	105.0% rated power
Decay Ratio	
Total System Stability, X_2/X_0	See Figures 4.4-7a through 4.4-9d
Reactor Core Stability, X_2/X_0	0.98 (Also see Figure 4.4-6)
Channel Hydrodynamic Performance, X_2/X_0	0.98

Note: (1) This analysis was performed based on the initial core design. The results of the stability analysis for the current cycle are provided in Appendix 15D, Reload Analysis.

4.5 REACTOR MATERIALS

4.5.1 Control Rod System Structural Materials

4.5.1.1 Material Specifications

a. Material List

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
Plugs	ASME SA 182 Grade F304
Cylinder	ASTM A269 Grade TP 304
Outer Tube	ASTM A269 Grade TP 304
Tube	ASME SA 351 Grade CF-3
Spacer	ASME SA 351 Grade CF-3

(2) Piston Tube Assembly

Piston Tube	ASME SA 479 Grade XM-19 or ASME SA 249 Grade XM-19
Nose	ASME SA 479 Grade XM-19
Base	ASME SA 479 Grade XM-19
Ind. Tube	ASME SA 312 Type 316
Cap	ASME SA 182 Grade F316

(3) Drive Line Assembly

Coupling Spud	Inconel X-750
Compression Cylinder	ASME SA 479 Grade XM-19 or ASME SA 249 Grade XM-19
Index Tube	ASME SA 479 Grade XM-19 or ASME SA 249 Grade XM-19
Piston Head	Armco 17-4 PH
Piston Coupling	ASME SA 312 Grade TP 304 or ASTM A269 Grade TP 304
Magnet Housing	ASME SA 312 Grade TP 304 or ASTM A269 Grade TP304 or ASME SA 312, Grade TP316L or ASTM A249, Type 316L or ASTM A213, Type 316L

(4) Collet Assembly

Collet Piston	ASTM A269, Grade TP304 or ASTM A312, Grade TP304
Finger	Inconel X-750
Retainer	ASTM A269 Grade TP304
Guide Cap	ASTM A269 Grade TP304

(5) Miscellaneous Parts

Stop Piston	Armco 17-4 PH
O-Ring Spacer	ASME SA 240 Type 304
Nut	ASME SA 479 Grade 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
Buffer Shaft	Armco 17-4 PH
Buffer Piston	Armco 17-4 PH
Buffer Spring	Inconel X-750
Nut (hex)	Inconel X-750

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, nut (hex), and collet spring are fabricated from Inconel X-750 in the annealed or equalized condition, and aged 20 hours at 1300° F to produce a tensile of 165,000 psi minimum, yield of 105,000 psi minimum, and elongation of 20% minimum. The piston head, stop piston, buffer shaft, and buffer piston are Armco 17-4 PH in condition H-ll00 (aged 6 hours at 1100° F), with a tensile of 140,000 psi minimum, yield of 115,000 psi minimum, and elongation of 15% minimum. The collet and buffer springs are fabricated from alloy X-750 wire in the spring temper condition and aged 4 1/2 hours at 1200° F to produce a tensile of 200,000 psi minimum.

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

All materials, except SA 479 Grade XM-19, have been successfully used for the past 10 to 15 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.

b. Special Materials

No cold worked austenitic stainless steels with a yield strength greater than 90,000 psi are employed in the Control Rod Drive system. Hardenable martensitic stainless steels are not used. Armco 17-4 PH (precipitation hardened stainless steel) is used for the piston head, stop piston, buffer shaft, and buffer piston. This material is aged to the H-II00 condition to produce resistance to stress corrosion cracking in the BWR environments. Armco 17-4 PH (H-II00) has been successfully used for the past 10 to 15 years in BWR drive mechanisms.

4.5.1.2 Austenitic Stainless Steel Components Processes, Inspections and Tests

All austenitic stainless steel used in the Control Rod Drive is solution annealed material with one exception, the outer tube in the cylinder, tube, and flange assembly. See Paragraph 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 300 Series stainless steel components to temperatures in the sensitization range.

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

(2) The collet piston and guide cap (collet assembly) are nitrided to provide a wear resistant surface.

Colmonoy hard surfacing is applied by the flame spray process. Parts are preheated to 550-800° F and then sprayed with Colmonoy. The sprayed coating is fused at about 2000° F using an oxyacetylene torch followed by air cooling.

Nitriding is accomplished using a proprietary process called New Malcomizing. Components are exposed to a temperature of about 1080° F for about 20 hours during the nitriding cycle.

Colmonoy hard surfaced components have performed successfully for the past 10 to 15 years in drive mechanisms. Nitrided components have been used in Control Rod Drives since 1967. 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.

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 and comply with the intent of Regulatory Guide 1.44.

A. Regulatory Guide 1.44
General Compliance or Alternate Approach Assessment: For Commitment, Revision Number, and Scope, see Section 1.8.

B. Control of Delta Ferrite Content

Control rod drive parts were fabricated after the issuance of Rev. 2 to Reg. Guide 1.31.

All type 308 weld metal was purchased to a specification which required a minimum of 5% delta ferrite. Ferrite measurements were made with a calibrated magnetic instrument on un-diluted weld pads for each lot and heat of weld filler metal. For the submerged arc welding process, measurements were made for each wire-flux combination.

These procedures comply with the requirements of Rev. 2 to Reg. Guide 1.31.

A. Regulatory Guide 1.31
General Compliance or Alternate Approach Assessment: For Commitment, Revision Number, and Scope, see Section 1.8.

4.5.1.3 Other Materials

These are discussed in Subsection 4.5.1.1.b.

4.5.1.4 Cleaning and Cleanliness Control

4.5.1.4.1 Protection of Materials During Fabrication, Shipping, and Storage

All the Control Rod Drive parts listed above (Paragraph 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 dessicant. 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 four 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. This inspection must be performed with a GE representative present.

A. Regulatory Guide 1.37
General Compliance or Alternate Approach Assessment: For Commitment, Revision Number, and Scope, see Section 1.8.

4.5.2 Reactor Internal Materials

4.5.2.1 Material Specifications

Materials used for the Core Support Structure:

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

Shroud, core plate, and top guide ASME: SA240, SA182, SA479, SA312, SA249, or SA213 (all Type 304L).

Shroud Stabilizer Assemblies – The stabilizer assemblies are fabricated from 316 SST (SA-240 or SA-182) with carbon content less than 0.02%, XM-19 SST (SA-240 or SA-182) with carbon content less than 0.04%, and Alloy X-750 (Ni-Cr-Fe) per SB637, Grade UNS N07750, Type 3.

Peripheral fuel supports - ASME: SA312 Grade TP304L or TP316L

Core Plate and Top Guide Hardware:

Core Plate Studs - ASTM: A479, TP304
ASME: SA-193, Grade B8A
ASME: SA-479, TPXM-19

Core Plate Nuts - ASTM: A479, TP304
ASME: SA-193, Grade 8A
ASME: SA-479, TPXM-19

Core Plate Wedges - ASME: SA479 TP304

Top Guide Studs/Nuts - ASME: SA479 TPXM-19

Top Guide Sleeves - ASME: SA182 Grade F304L or F316L,
ASME: SA213 TP304L, 316, or 316L
ASME: SA249 TP304L, 316; or 316L,or
ASME: SA479 TP304, 304L, 316L or XM-19

Control rod guide tube - ASME: SA358 Grade 304, SA312 Grade TP304, SA249 Type 304, SA351 Grade CF8; ASTM: A276 Type 304, A240 Type 304, A351 Grade CF.

Orificed fuel support - ASME: SA351 Grade CF8, SA479 Type 316L; ASTM: A240 Type 304 or 316L, A276 Type 304.

Materials Employed in Other Reactor Internal Structures.

(1) Steam Separator and Steam Dryer

All materials are Type 304, 304L or 316L stainless steel.

Plate, Sheet and Strip	ASTM A240, Type 304, 304L or 316L
Forgings	ASTM A182, Grade F304 or 304L

Bars	ASTM A276 Type 304, 304L or 316L ASME SA479 Type 316L
Pipe	ASTM A312 Grade TP 304 or 316L
Tube	ASTM A269 Grade TP 304 or ASTM A249 Type 316L or ASTM A213 Type 316L
Castings	ASTM A351 Grade CF8 ASTM A403 WPVV-304 or WP-304

(2) Jet Pump Assemblies

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

Castings	ASTM A351 Grade CF8 and ASME SA 351 Grade CF3
Bars	ASTM A276 Type 304 and ASTM A637 Grade 688 ASME A479, Type 304L ASME SA479, XM-19
Bolts	ASTM A193 Grade B8 or B8M ASME SA479 Type 316L
Sheet and Plate	ASTM A240 Type 304, 304L, 316L, and ASME SA 240 Type 316L, XM-19
Tubing	ASTM A269 Grade TP 304
Pipe	ASTM A358 Type 304, 316L and ASME SA312 Grade TP 304, 316L ASTM A312 Type 304
Welded Fittings	ASTM A403 Grade WP304
Forged or Rolled Parts	ASME SA182 or ASTM A182 Grade F304, F316L ASTM B166, and ASTM A637 Grade 688 ASTM A182 Grade F304, F316L

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

a. The inlet mixer adaptor casting, the wedge casting, bracket casting adjusting screw casting, and the diffuser collar casting are Type 304 hard surfaced with Stellite 6 for slip fit joints.

b. The diffuser is a bimetallic component made by welding a Type 304 forged ring to a forged Inconel 600 ring, made to Specification ASTM B166.

c. The inlet-mixer contains a pin, insert, and beam made of Inconel X-750 to Specification ASTM A637 Grade 688.

All core support structures are fabricated from ASME specified materials, and designed in accordance with requirements of ASME Code, Section III, Subsection NG. The other reactor internals are non-coded, and they are fabricated from ASTM or ASME specification materials. Material requirements in the ASTM specifications are identical to requirements in corresponding ASME material specifications.

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. Requirements of ASME Section IX B&PV Code, are followed in fabrication of core support structures and other internals.

4.5.2.3 Nondestructive Examination of Tubular Products

Wrought seamless tubular products for CRD guide tubes, CRD housings, and peripheral fuel supports, were supplied in accordance with ASME Section III, Class CS, which require examination of the tubular products by radiographic and/or ultrasonic methods according to paragraph NG-2550.

Wrought seamless tubular products for other internals were supplied in accordance with the applicable ASTM or ASME 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.

Reactor internals were fabricated prior to the issuance of Rev. 2 to Reg. Guide 1.31.

Ferrite measurements were made in accordance with the requirements of the ASME code in effect at that time. This code required the use of the chemical composition in conjunction with the Shaeffler diagram to verify that weld filler metal contained a minimum of 5 percent delta ferrite.

An extensive test program performed by General Electric Company, with the concurrence of the Regulatory Staff, demonstrated that the use of the Shaeffler diagram to control weld filler metal ferrite at 5 percent minimum was adequate to produce satisfactory production welds. The 400 production welds evaluated in this program were fabricated with filler metal controlled in accordance with the Shaeffler diagram to contain a minimum of 5 percent ferrite. All these production welds met the requirements of the Interim Regulatory Position to Reg. Guide 1.31 which was in effect at that time.

Regulatory Guide 1.34, Control of Electroslag Weld Properties

Electroslag welding is not employed for any reactor internals.

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. For 304 steel with carbon content in excess of 0.035 percent carbon, 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 and comply with the intent of Regulatory Guide 1.44.

Regulatory Guide 1.71, Welder Qualification for Areas of Limited

There are few restrictive welds involved in the fabrication of items described in this section. Mock-up welding was performed on the welds with most difficult access. Mock-ups were examined with radiography or by sectioning.

Regulatory Guide 1.37, Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants

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.

A. Regulatory Guide 1.37
General Compliance or Alternate Approach Assessment: For Commitment, Revision Number, and Scope, See Section 1.8.

4.5.2.5 Other Materials

Materials, other than Type 300 stainless steel, employed in vessel internals are:

SA 479 Type XM-19 stainless steel
SB 166, 167, and 168 Nickel-Chrome-Iron (Inconel 600)
SA 637 Grade 688 Inconel X-750

Inconel 600 tubing plate, and sheet are used in the annealed conditions. Bar may be in the annealed or cold-drawn condition.

Inconel X-750 components are fabricated in the annealed or equalized condition and aged 20 hours at 1300° F.

Stellite 6 hard surfacing is applied to austenitic stainless steel castings using the gas tungsten arc welding or plasma arc surfacing processes.

All materials, except SA 479 Grade XM-19, have been successfully used for the past 10 to 15 years in BWR applications. Extensive laboratory tests have demonstrated that XM-19 is a suitable material and that it is resistant to stress corrosion in a BWR environment.

4.5.3 Control Rod Drive Housing Supports

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

	Material
Grid	ASTM-A-441
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

For further control rod drive housing support information refer to Subsection 4.6.1.2.

4.6 FUNCTIONAL DESIGN OF REACTIVITY CONTROL SYSTEMS

The reactivity control systems consist of control rods and control rod drives, supplementary reactivity control for the initial core (Subsection 4.3), and the Standby Liquid Control System, (described in Subsection 9.3.5).

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) The design shall provide for a sufficiently rapid control rod insertion such that no fuel damage results from any abnormal operating transient.

(2) The 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 System (CRD) 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 treatment system, and/or condensate storage tanks 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 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, and also present the positions of the control rod selected for movement and 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, "Neutron Monitoring System"). 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. The function of the index tube is similar to that of a piston rod in a conventional hydraulic cylinder. 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 thinwalled 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 piston and drive piston seals.

The center tube of the drive mechanism forms a well to contain the position indicator probe. The probe is an aluminum extrusion attached to a cast aluminum housing. Mounted on the extrusion are hermetically sealed, magnetically operated, reed switches. The entire probe assembly is protected by a thin-walled stainless 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. Two switches are located at each position corresponding to an index tube groove, thus allowing redundant indication at each latching point. Two additional switches are located at each midpoint between latching points to indicate the intermediate positions during drive motion. Thus, indication is provided for each 3 inches of travel. Duplicate switches are provided for the full-in and full-out positions. Redundant over-travel 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 over-travel 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.

4.6.1.1.2.2.7 Lock Plug

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.

If it is desired to uncouple a drive without removing the reactor pressure vessel head for access, the lock plug can also be pushed up from below. In this case, the piston tube assembly 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.

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. Large tensile and column loads are applied during scram. The reactor environment limits the choice of materials suitable for corrosion resistance. To meet these varied requirements, the index tube is made from 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 X-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 (Electrolized). 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 X-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

Graphitar 14 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 Graphitar 14.

(2) All springs and members requiring spring action (collet fingers, coupling spud, and spring washers) are made of Inconel X-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, stop piston, buffer shaft and buffer piston are made of Armco 17-4 PH.

(9) Certain fasteners and locking devices are made of Inconel X-750 or 600.

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 Drawings M05-1078 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, and is returned to the reactor vessel via the HCUs of non-moving drives. 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 Drawing M05-1078 and 768E412CA. The hydraulic requirements, identified by the function they perform, are as follows:

(1) An accumulator hydraulic charging pressure of approximately 1750 to 2000 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 each control rod and 2 gpm to withdraw each control rod is required.

(3) Cooling water to the drives is required at approximately 15 psi above reactor vessel pressure and at a flow rate of approximately 0.34 gpm per drive unit.

(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 (excluding the instrument volume).

4.6.1.1.2.4.2 System Description

The CRD hydraulic system provides the required functions with the pumps, filters, valves, instrumentation, and piping shown in Drawing M05-1078 and described in the following paragraphs. A secondary function of the CRD system is to provide a source of water for the keep-fill system of the reactor water level instrumentation system.

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

4.6.1.1.2.4.2.1 Supply Pump

One supply pump pressurizes the system with water from the condensate treatment system and/or condensate storage tanks. 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 pump damage if the pump discharge is inadvertently closed.

A third pump (1C11-C300) powered from the Division II Class 1E power distribution system, is used to provide water to the reactor recirculation pump seals in the event the CRD supply pumps are not available (e.g., due to loss of normal station power, corrective maintenance, etc).

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. A 250-micron strainer in the filter bypass line protects the pump when the filters are being serviced. The drive water filter, downstream of the pump, is a cleanable element type with a 50 or 15-micron absolute rating. A differential pressure indicator and control room alarm monitor the filter element as it collects foreign materials.

Refer to Appendix D, response II.K.3.25 for additional discussion and licensing basis for the third pump.

4.6.1.1.2.4.2.2 Accumulator Charging Pressure

Accumulator charging pressure is established by precharging the nitrogen accumulator to a precisely controlled pressure at known temperature. Accumulator pressure is established by first pre-charging to approximately 1200 psig with nitrogen and then turning on the water supply pump and charging to approximately 1750 psig. 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 element upstream of the accumulator charging header senses high flow and provides a signal to the manual auto-flow control station which in turn closes the system flow control valve. This action maintains increased flow through the charging water header, while avoiding prolonged pump operation at "run-out" conditions. 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 and drive cooling.

4.6.1.1.2.4.2.3 Drive Water Pressure

Drive water pressure required in the drive header is maintained by the drive pressure control valve, which is manually adjusted from the control room. A flow rate of approximately 16 gpm (the sum of the flow rate required to insert 4 control rods) normally passes from the drive water pressure stage through eight solenoid operated stabilizing valves (arranged in parallel) into the cooling water header. The flow through two stabilizing valves equals the drive insert flow for one drive; that of one stabilizing valve equals the drive withdrawal flow for one drive. When operating a drive(s), the required flow is diverted to the drives by closing the appropriate stabilizing valves, at the same time opening the drive directional control and exhaust solenoid valves. 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

The cooling water header is located downstream from the drive/cooling pressure valve. The drive/cooling pressure control valve is manually adjusted from the control room to produce the required drive/cooling water pressure balance.

The flow through the flow control valve is virtually constant. Therefore, once adjusted, the drive/cooling pressure control valve will maintain the correct drive pressure and cooling water pressure, independent of reactor vessel pressure. Changes in setting of the pressure control valves are required only to adjust for changes in the cooling requirements of the drives, as the drive 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 indicated and recorded, 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.

During normal plant operation the scram discharge volume is empty, and vented to the atmosphere through its two open vent and two open drain valves. When a scram occurs, upon a signal from the safety circuit, these vent and drain valves are closed to conserve reactor water. 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 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.

Four liquid-level float switches and six transmitters are connected to the instrument volume to monitor the volume for abnormal water level. They are set at three different levels. The lowest level indicates 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. At the second level, a rod withdrawal block is actuated to prevent further withdrawal of any control rod when leakage accumulates to half the capacity of the instrument volume. At the third level, the remaining switches are interconnected with the trip channels of the Reactor Protection System and will initiate a reactor scram should water accumulation fill the instrument volume.

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, "Rod Control and Information System."

The basic components in each HCU are manual, pneumatic, and electrical valves; an accumulator; related piping; electrical connections; filters; and instrumentation (See Figure 4.6-8). 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 Valve Assembly

The scram pilot valve assembly is operated from the Reactor Protection System. The scram pilot valve assembly, with two solenoids, controls both the scram inlet valve and the scram exhaust valve. The scram pilot valve assembly is solenoid-operated and is normally energized. On loss of electrical signal to the solenoids, such as the loss of external a-c power, the inlet port closes and the exhaust port opens. The pilot valve assembly (Figure 4.6-8) is 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 position indicator switch on this valve energizes a light in the control room as soon as the valve starts to 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 any 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 in the following sections.

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 (nonscram 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 pressures 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 valve assembly 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 (approximately 1750 psi, initially 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 diminishing velocity. This characteristic provides a high initial rod insertion rate. As the drive piston nears the top of its stroke, the piston reaches the buffer and the driveline is brought to a stop at the full-in position.

Prior to a scram signal the accumulator in the Hydraulic Control Unit has 1750-2000 psig on the water side and approximately 1750 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 inch area. As CRD motion begins, this pressure drops to the gas side pressure less line losses between the accumulator and the CRD. When the drive reaches the full-in position, the accumulator completely discharges with a resulting gas side pressure of approximately 1200 psig.

The control rod drive accumulators are necessary to scram the control rods within the required time. Each drive, however, has an internal ballcheck valve which allows reactor pressure to be admitted under the drive piston. If the reactor is above 600 psi this valve ensures rod insertion in the event the accumulator is not charged or the inlet scram valve fails to open. The insertion time, however, will be slower than the scram time with a properly functioning scram system.

The Control Rod Drive System, with accumulators, provides the following scram performances 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 listed. Maximum scram time (seconds) from notch position 48 to notch position:

Reactor Dome Pressure (psig)	43	29	13
1050	0.32	0.86	1.57
950	0.31	0.81	1.44

The scram times include an assumed maximum delay of 0.100 seconds from tripping of the scram load drivers to start of CRD motion.

The scram time from position 48 is a function of reactor pressure. As reactor pressure increases, scram times will increase. The CPS Technical Specifications specify the scram times required to ensure that the scram reactivity assumed in the design basis accident and transient analyses are met. To account for single failure and slow scramming rods with one control cell separation, the scram times specified in the Technical Specifications are faster than those assumed in the design basis analysis.

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 rod control and information system. The objective of the rod control and information 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, "Instrumentation and Control System".

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 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 4.6-9. Horizontal beams are installed immediately below the bottom head of the reactor vessel, between the rows of CRD housings.

The beams are welded to brackets which are 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 inch 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 3/4 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 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 l-inch 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. Loading conditions and examples of stress analysis results and limits are shown in Table 3.9-2. Safety evaluation is discussed in Subsection 4.6.2.3.3.

4.6.2 Evaluations of the CRDS

4.6.2.1 Failure Mode and Effects Analysis

This subject is covered in section 15A "NSOA"

4.6.2.2 Protection from Common Mode Failures

The applicant's position on this subject is covered in section 15A "NSOA"

4.6.2.3 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.

4.6.2.3.1 Control Rods

4.6.2.3.1.1 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, hafnium and 304 austenitic stainless steel, have been found suitable in meeting the demands of the BWR environment.

4.6.2.3.1.2 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.

4.6.2.3.1.3 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.

In addition, to further this end, dissimilar metals are avoided.

4.6.2.3.1.4 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 under "Rupture of Hydraulic Line(s) to Drive Housing Flange." In summary, if the collet were to remain open, which is unlikely, calculations indicate that the steady-state control rod withdrawal velocity would be 10 ft/sec for a pressure under line break, the limiting case for rod withdrawal.

4.6.2.3.1.5 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, "Analysis of Malfunction Relating to Rod Withdrawal."

4.6.2.3.1.6 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 and the effects of probable control rod failures (see Subsection 4.6.2.3.2.2, "Analysis of Malfunction Relating to Rod Withdrawal.)"

4.6.2.3.1.7 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.

4.6.2.3.1.8 Mechanical Damage

In addition to the analyses performed on the control rod drive (Subsection 4.6.2.3.2.2, "Analysis of Malfunction Relating to Rod Withdrawal and Subsection 4.6.2.3.2.3, "Scram Reliability") and the control rod blade, analyses were performed on the control rod guide tube, reference Subsections 4.2.3.3.7 through 4.2.3.3.8 for these analyses.

4.6.2.3.1.9 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, "Accident Analyses."

4.6.2.3.2 Control Rod Drives

4.6.2.3.2.1 Evaluation of Scram Time

The rod scram function of the control rod drive system provides the negative reactivity insertion required by safety design basis 4.6.1.1.1.1.1.1(1). The scram time shown in the description is adequate as shown by the transient analyses of Chapter 15, "Accident Analyses."

4.6.2.3.2.2 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, "Accident Analyses". Therefore, the physical and radiological consequences of such rod withdrawals are less than those analyzed in the rod drop accident.

4.6.2.3.2.2.1 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 CRD housing material at the vessel penetration is seamless, type Iconel 600, tubing with a minimum tensile strength of 80,000 psi, and type 304 stainless steel pipe below the vessel with a minimum 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 1); the housing would not drop far enough to clear the vessel penetration; reactor water would leak at a rate of approximately 180 gpm through the 0.03-inch diametral clearance between the housing and the vessel penetration.

If the basis 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.

4.6.2.3.2.2.2 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 (insert) line break: (2) pressure-over (withdrawn) line break; and (3) coincident breakage of both of these lines.

The NRC staff's acceptance (as given in NUREG 0619) of the CRDRL deletion is based on completion of the following four requirements:

1. Demonstration by test of CRD flow to the reactor vessel is equal to or greater than the boil-off rate discussed in NUREG 0619.

2. Installation of equalizing valves between the cooling water header and the exhaust water header.

3. Installation of flush ports on the exhaust header if carbon steel piping is utilized.

4. Rerouting the flow stabilizer loop to the cooling water header with stainless steel piping.

The NRC Staff has further determined that the CRD System make-up flow test recommended in NUREG 0619 (requirement #1 above) is no longer necessary for LRG-I projects (NUREG 0771) and LRG-II projects (NUREG 0887) since the intent of this requirement is met in other ways such as:

a. Plant fire prevention/protection and separation enhancements.

b. Development of systems-oriented emergency procedure guidelines.

c. Post-TMI emergency core cooling system modifications.

The CRD system specifications (i.e., Design Specifications, Design Specification Data Sheets, P&ID, etc.) comply to the latter three requirements. The CRD system P&ID (see Drawing M05-1078) shows 1) the pressure equalizing valve as communicating with the cooling water header and the exhaust water header, 2) the flow stabilizer loop routed to the cooling water header, and 3) both the exhaust header and the flow stabilizer loop as requiring the use of stainless steel piping (Q&R 410.5).

4.6.2.3.2.2.2.1 <u>Pressure-under (Insert) Line Break</u>

For the case of a pressure-under (insert) 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 (insert) 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 pressures 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 containment. 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 10 ft/sec.

4.6.2.3.2.2.2.2 Pressure-over (Withdrawn) Line Break

The case of the pressure-over (withdrawn) 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 containment through the broken pressure-over line. The leakage rate at 1000 psi reactor pressure is estimated to be 1 to 3 gpm, however with the graphitar seals of the stop piston removed, the leakage rate could be as high as 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 annunciated in the control room, and by operation of the drywell sump pump.

4.6.2.3.2.2.2.3 Simultaneous Breakage of the Pressure-over (Withdrawn) and Pressure-under (Insert) Lines

For the simultaneous breakage of the pressure-over (withdrawn) and pressure-under (insert) 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 (at reactor pressure approximately 600 psig or greater) 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 containment, 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 annunciated in the control room, and operation of the drywell sump pump.

4.6.2.3.2.2.3 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 steel, with a minimum tensile strength of 125,000 psi. Each bolt has an allowable load capacity of 15,200 pounds. Capacity of the 8 bolts is 121,600 pounds. As a result of the reactor design pressure of 1250 psig, the major load on all 8 bolts is 30,400 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 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.

4.6.2.3.2.2.4 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 575oF. 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.

4.6.2.3.2.2.5 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 CRD housing is made of Inconel 600 seamless tubing (at the penetration to the vessel), with a minimum tensile strength of 80,000 psi, and of Type 304 stainless steel seamless pipe below the vessel with a minimum tensile strength of 75,000 psi. The maximum hoop stress of 9,047 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.

There would be no pressure differential acting across the collet piston 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.

4.6.2.3.2.2.6 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 acting across the collet piston 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.

4.6.2.3.2.2.7 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 less than 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.

4.6.2.3.2.2.8 Drive Cooling Water 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/cooling water 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 is set to develop a pressure approximately 260 psi in excess of reactor pressure.

If the flow through the drive/cooling water 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 approximately 260 psig to no more than 2000 psig. Calculations indicate that the drive would accelerate from 3 inch/sec to approximately 5 inch/sec. A pressure differential of 1970 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.

The function of the F003 pressure control valve (PCV) is to provide a means of adjusting the drive water header and cooling water header pressures. The F003 PCV is a manually controlled motor operated valve which is controllable from the main control room. Indicating lights are provided in the control room for the valve full-open and full-closed positions. Adjustment of the F003 PCV in conjunction with adjustments to the F002 flow control valve permit adjustment of the drive water header pressure to approximately 260 psi above vessel pressure while, at the same time, maintaining the drive cooling water header pressure at approximately 20 psi above vessel pressure.

If the F003 PCV were to fail to a full-open position, the cooling water pressure would increase and the drive water pressure would decrease. The resulting cooling water pressure increase could cause control rods to drive inward. The existence of rod drifts would be alarmed to the control room operator for appropriate action. The resulting drop in drive water pressure would make normal control and notch movements impossible but would not affect the ability of the scram function.

Conversely, if the F003 PCV were to fail to a full-closed position, the cooling water pressure would decrease while the drive water pressure should increase. The reduction in cooling water pressure (and flow) would eventually lead to high CRD temperatures being alarmed in the control room. The CRD system's scram function would not be affected by the increase in drive water pressure. In the limiting case, the resulting increase in drive water pressure would reach up to the shutoff pressure of the supply pump. The occurrence of this condition during withdrawal of a drive at zero reactor pressure will result in a drive pressure increase from approximately 260 psig to no more than 2000 psig. Calculations and tests indicate that the drive would accelerate from 3 inch/sec to no more than 7 inch/sec. The rod movement would stop after the driving signal is removed or a rod block is enforced by the Rod Control and Information System (RC&IS). The RC&IS will preclude continuous rod withdrawal and hence safety evaluations show that the integrity of the fuel is not compromised. These evaluations are discussed in Chapter 15 of the USAR.

In both of the cases described above, the manually operated bypass PCV (F004) in conjunction with isolation gate valves located upstream and downstream of the F003 PCV would enable the operators to take corrective action. Other hydraulic system features such as in line pressure relief valve (F040) and the countering responses of the flow control valve (F002) which would reduce the impact of the above cases were not considered.

In conclusion, although the failure to the full-open or full-closed position of the drive/cooling water PCV would cause perturbation in the CRD system operation it does not affect the scram capability of the CRD system. (Q&R 410.4)

4.6.2.3.2.2.9 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.

4.6.2.3.2.2.10 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.

4.6.2.3.2.2.11 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.

4.6.2.3.2.2.12 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.

4.6.2.3.2.3 Scram Reliability

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

(1) An individual accumulator is provided for each control rod drive with sufficient stored energy to scram at any reactor pressure. The reactor vessel itself, at pressures above 600 psi, will supply the necessary force to insert a drive if its accumulator is unavailable.

(2) Each drive mechanism has its own scram valves and a dual solenoid scram pilot valve therefore only one drive can be affected if a scram valve fails to open. Both pilot 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.

4.6.2.3.2.4 Control Rod Support and Operation

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

4.6.2.3.3 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 3/4 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 3/4 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. Inspection and testing of CRD housing supports is discussed in 4.6.3.2.1.

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 initial development drive (prototype of the standard locking piston design) 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 over-travel position. Failure of the drive to over-travel 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 mechanism and the associated control and protection systems was 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 defined criteria and acceptable ranges of such characteristics as seal leakage, friction and scram performance under fixed test conditions which had to be met before the component was shipped.

The after installation, prestartup tests (Chapter 14) included normal and scram motion and were primarily intended to verify that piping, valves, electrical components and instrumentation were properly installed. The test specifications included criteria and acceptable ranges for drive speed, timer settings, scram valve response times, and control pressures. These tests were intended more to document system condition rather than to be tests of performance.

As fuel was placed in the reactor, the startup test procedure (Chapter 14) was followed. The tests in this procedure were intended to demonstrate that the initial operational characteristics met the limits of the specifications over the range of primary coolant temperatures and pressures from ambient to operating. The detailed specifications and procedures followed the general pattern established for such specifications and procedures in BWRs that were under construction and in operation at the time.

4.6.3.1.1.5 Surveillance Tests

The surveillance requirements (SR) 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 0.38% delta k/k with the highest worth rod analytically determined or 0.28% delta k/k with the highest worth rod determined by test 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 rods fully inserted.

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

The 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.

(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 over-travel 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 over-travel position feature provides a positive check on the coupling integrity, for only an uncoupled drive can reach the over-travel 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.

In addition to operability testing, functional testing of the accumulator water leak detectors and calibration of the accumulator pressure detectors, including alarm setpoint $\geq$ 1520 psig on decreasing pressure are performed every 24 months.

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 paragraph 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 paragraphs 4.6.2.3.2.2.1 through 4.6.2.3.2.2.12.

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 Control Systems

4.6.4.1 Vulnerability to Common Mode Failures

The system is located such that it is protected from common mode failures due to missiles and failures of moderate and high energy piping and fire. Sections 3.5 and 3.6, and Subsection 9.5.1 discuss protection of essential systems against missiles, pipe ruptures and fire.

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

4.6.6 References

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.

FIGURES 4.2-1 THROUGH 4.2-5

HAVE BEEN DELETED

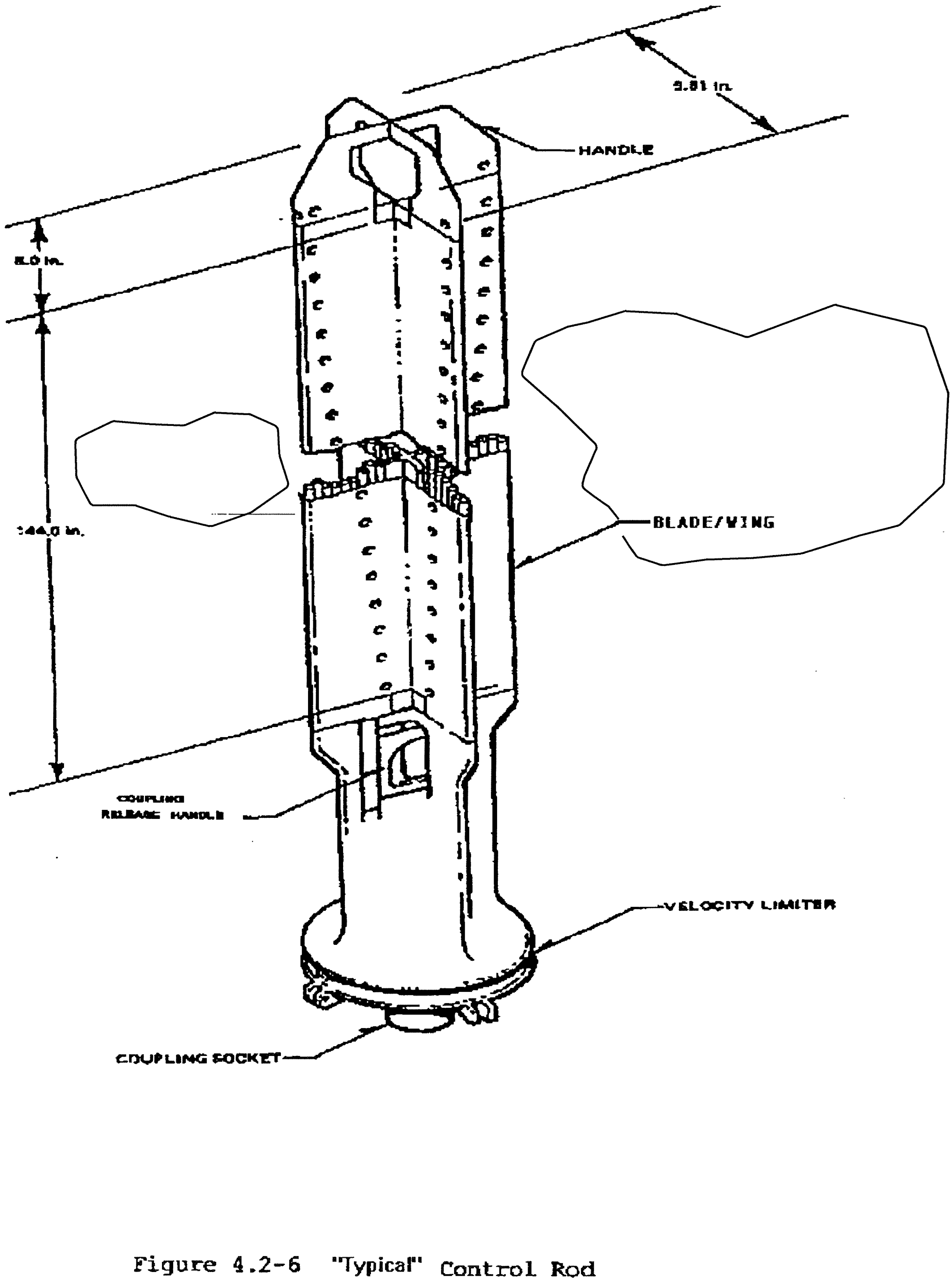


Figure 4.2-6 "Typical" Control Rod

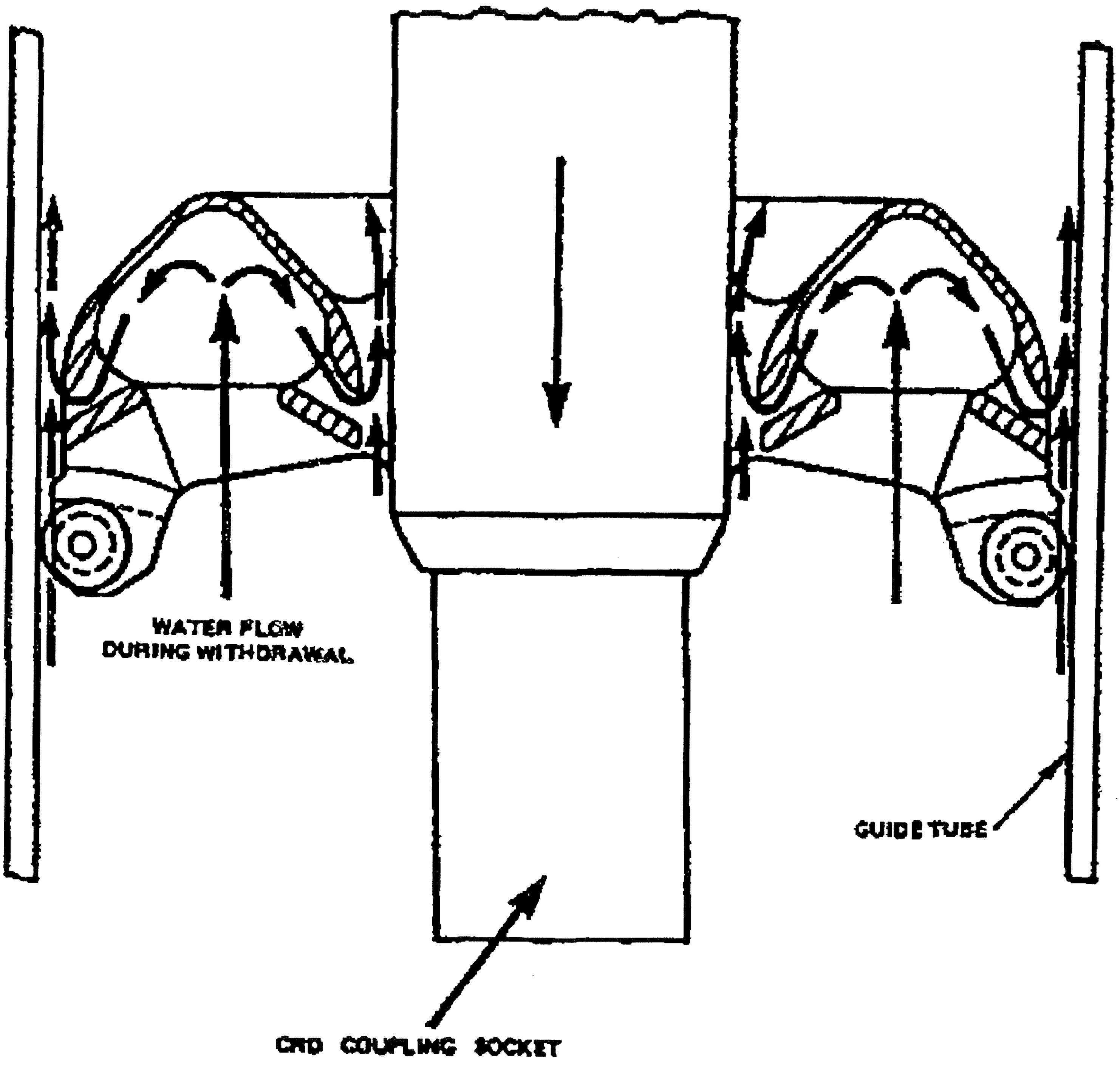
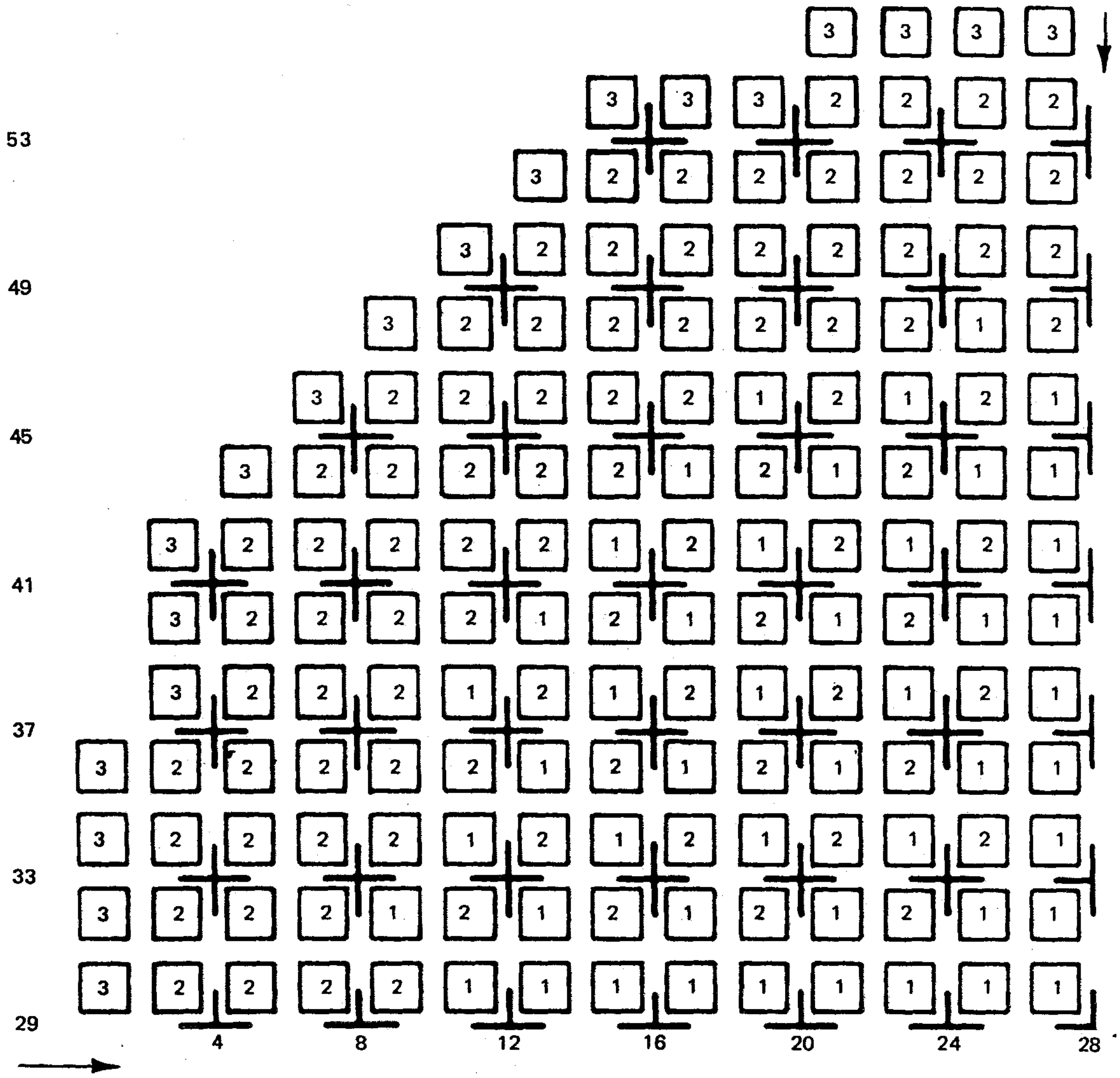


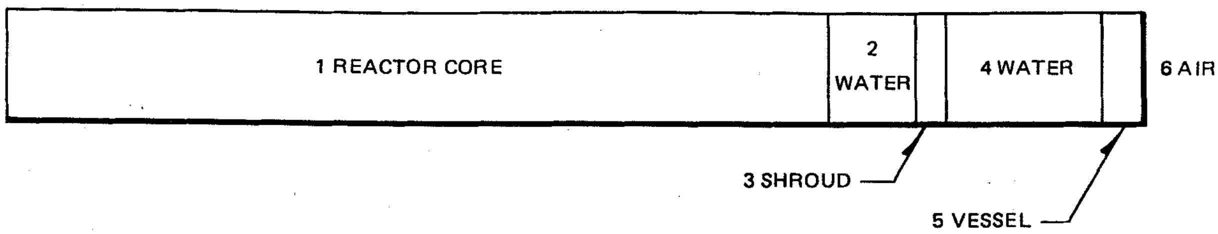
Figure 4.2-7 "Typical" Velocity Limiter

NOTE: LOADING PATTERN IS PRESENTED FOR QUARTER CORE ONLY. REFLECTIVE SYMMETRY APPLIES TO REMAINDER OF CORE



BUNDLE TYPE	NO. OF BUNDLES	BUNDLE AVERAGE ENRICHMENT
1 MEDIUM ENRICHMENT	188	1.54
2 HIGH ENRICHMENT	360	2.00
3 NATURAL URANIUM	76	0.711

Figure 4.3-1. Initial Core Loading Map



MATERIAL		RADIUS (in.)	MATERIAL	MATERIAL DENSITY
NO.	NAME			
			WATER	0.318 g/cm^3
			UO_2	2.334 g/cm^3
1	REACTOR CORE	84.6	ZIRCONIUM	0.978 g/cm^3
			304L STAINLESS STEEL	0.74 g/cm^3
2	WATER	91.0	WATER	0.74 g/cm^3
3	SHROUD	93.0	304L STAINLESS STEEL	FROM ASME SA 240
4	WATER	109.0	WATER	0.74 g/cm^3
5	VESSEL	114.41	CARBON STEEL	FROM ASME SA 240
6	AIR		AIR	1.3×10^{-3} g/cc

Figure 4.3-2. Model for One-Dimensional Transport Analysis of Vessel Fluence (218-624)

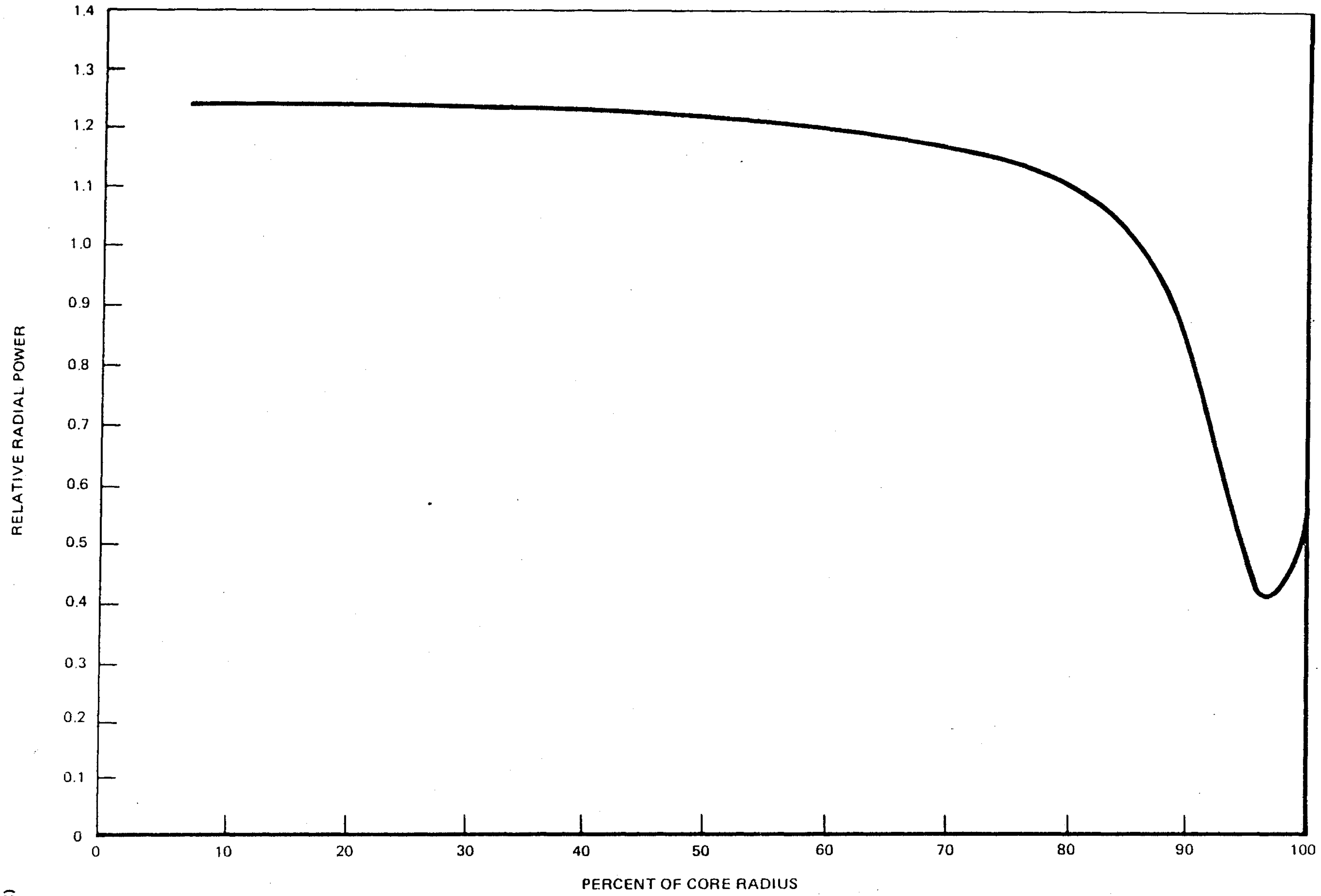


Figure 4.3-3. Radial Power Distributions Used in the Vessel Fluence Calculation.

(218-624)

FIGURES 4.3-4 THROUGH 4.3-28

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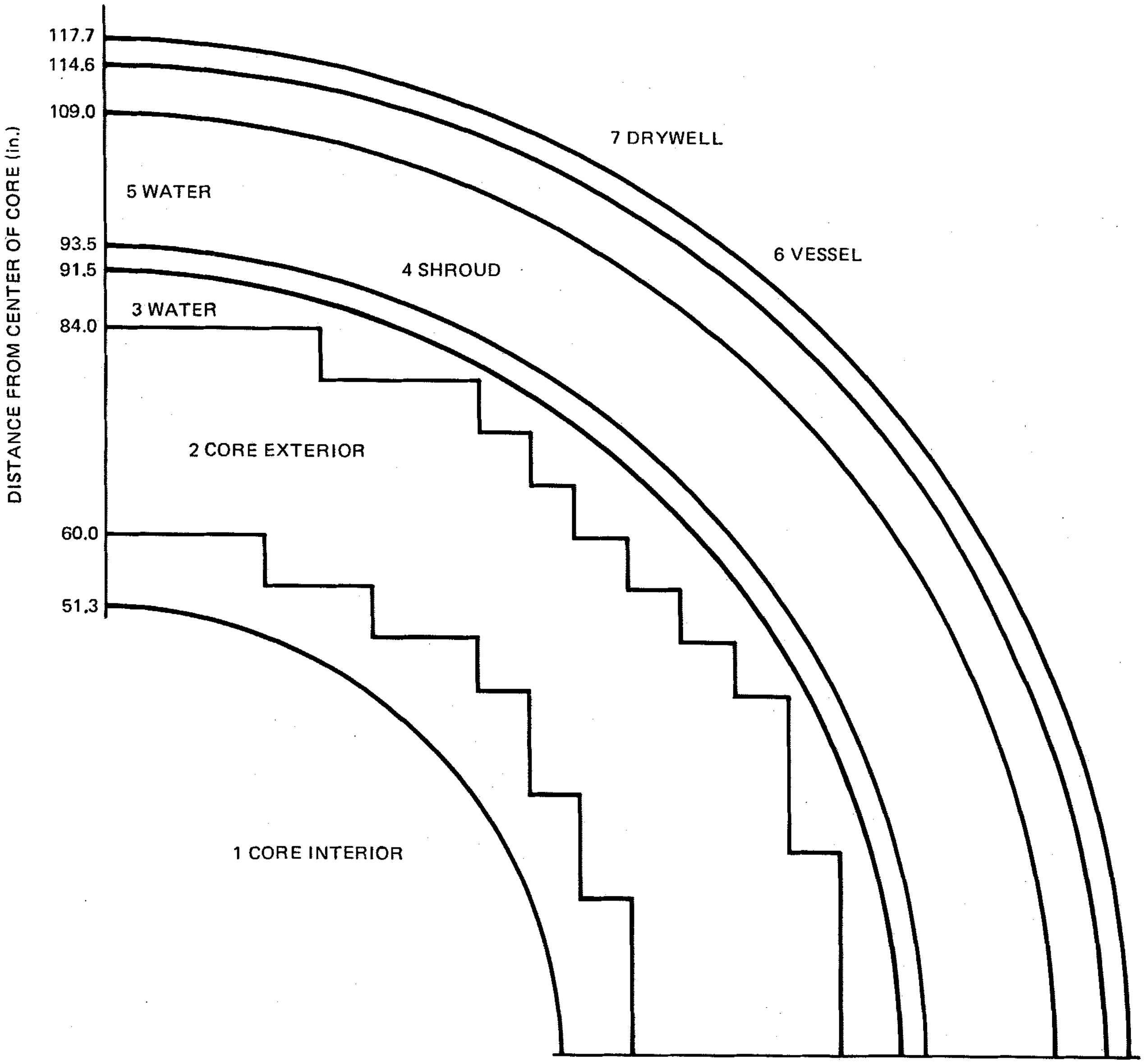

Figure 4.3-29 MODEL FOR TWO-DIMENSIONAL TRANSPORT ANALYSIS OF VESSEL FLUENCE

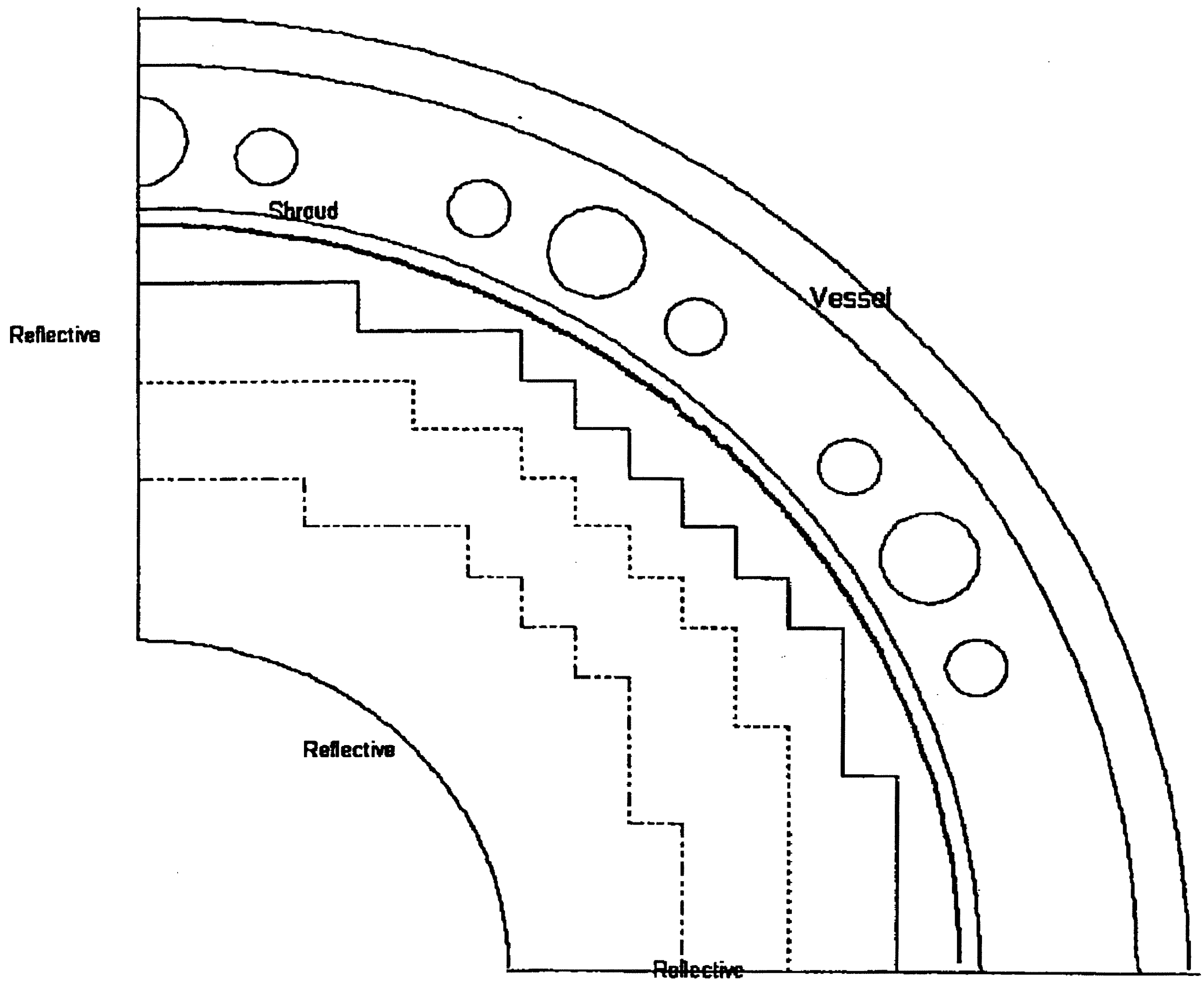

Figure 4.3-30 EPU (R,Θ) CALCULATION MODEL

FIGURES 4.4-1 THROUGH 4.4-4

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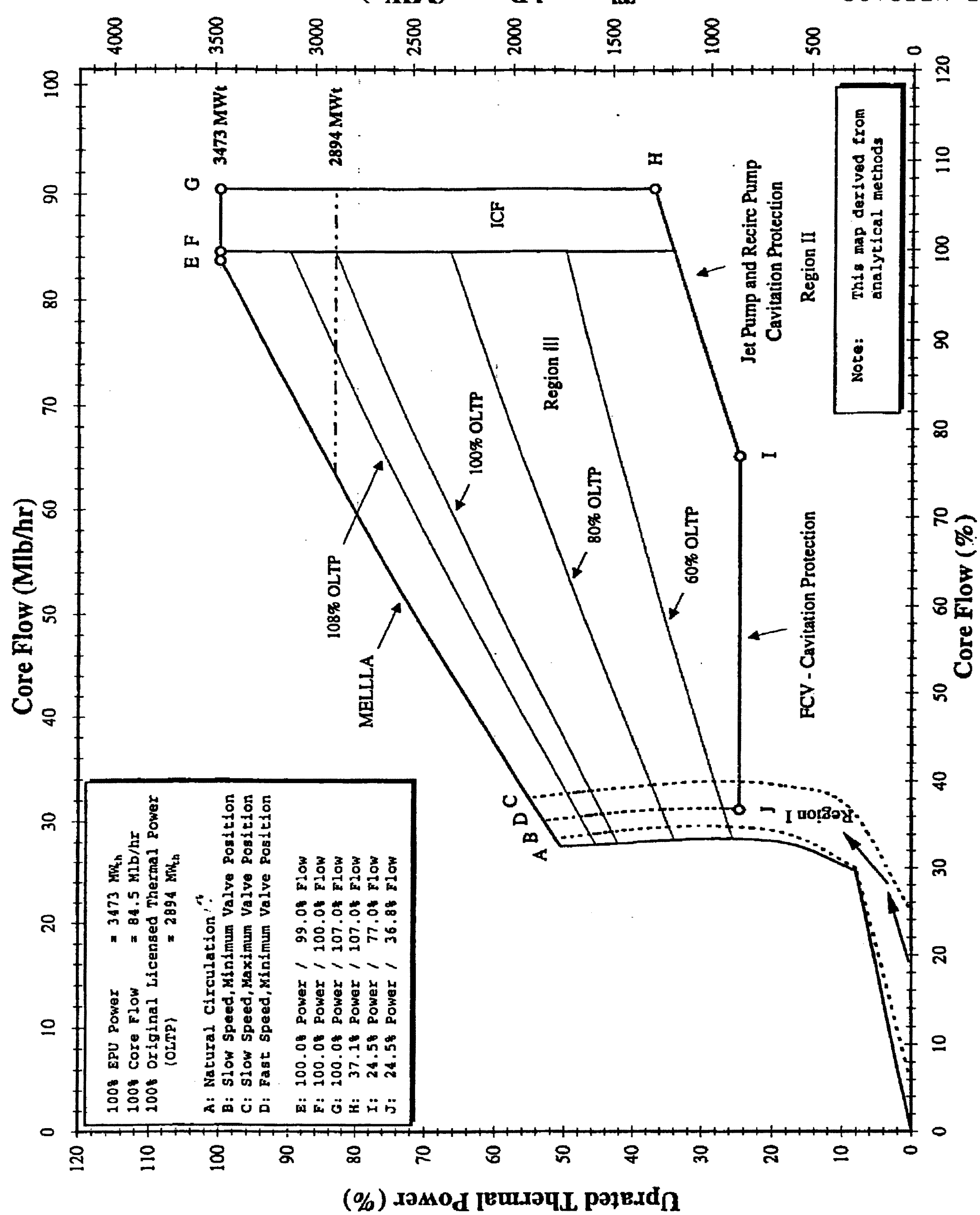

CLINTON POWER STATION
UPDATED SAFETY ANALYSIS REPORT

Figure 4.4-5

Reactor Operating Map
Normal Operation

Figure 4.4-6
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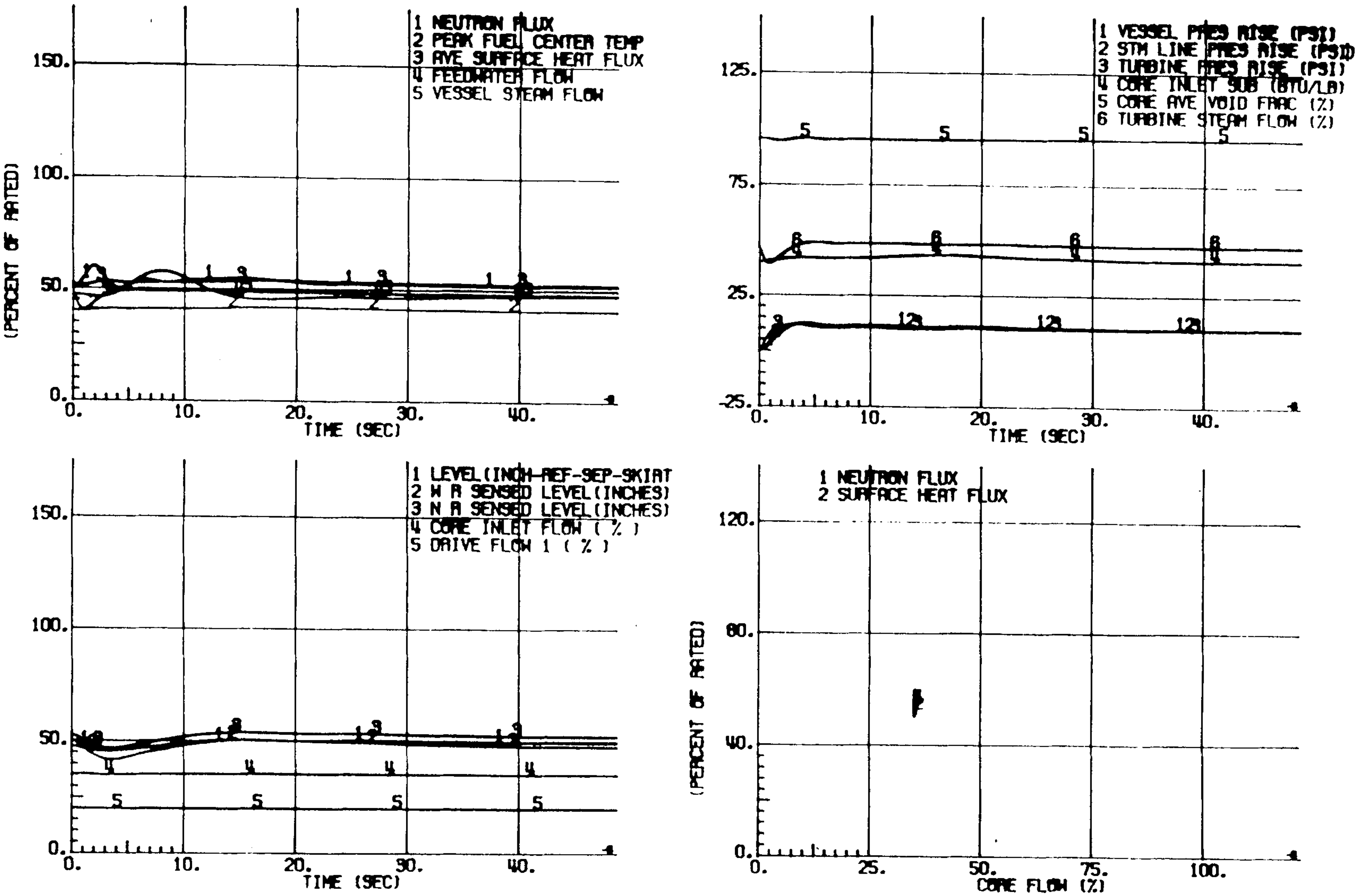


Figure 4.4-7a. 10 psi Pressure Regulator Setpoint Step Change at 51.5% Power / 33% Flow (Natural Circulation) - Based on Initial Core Design

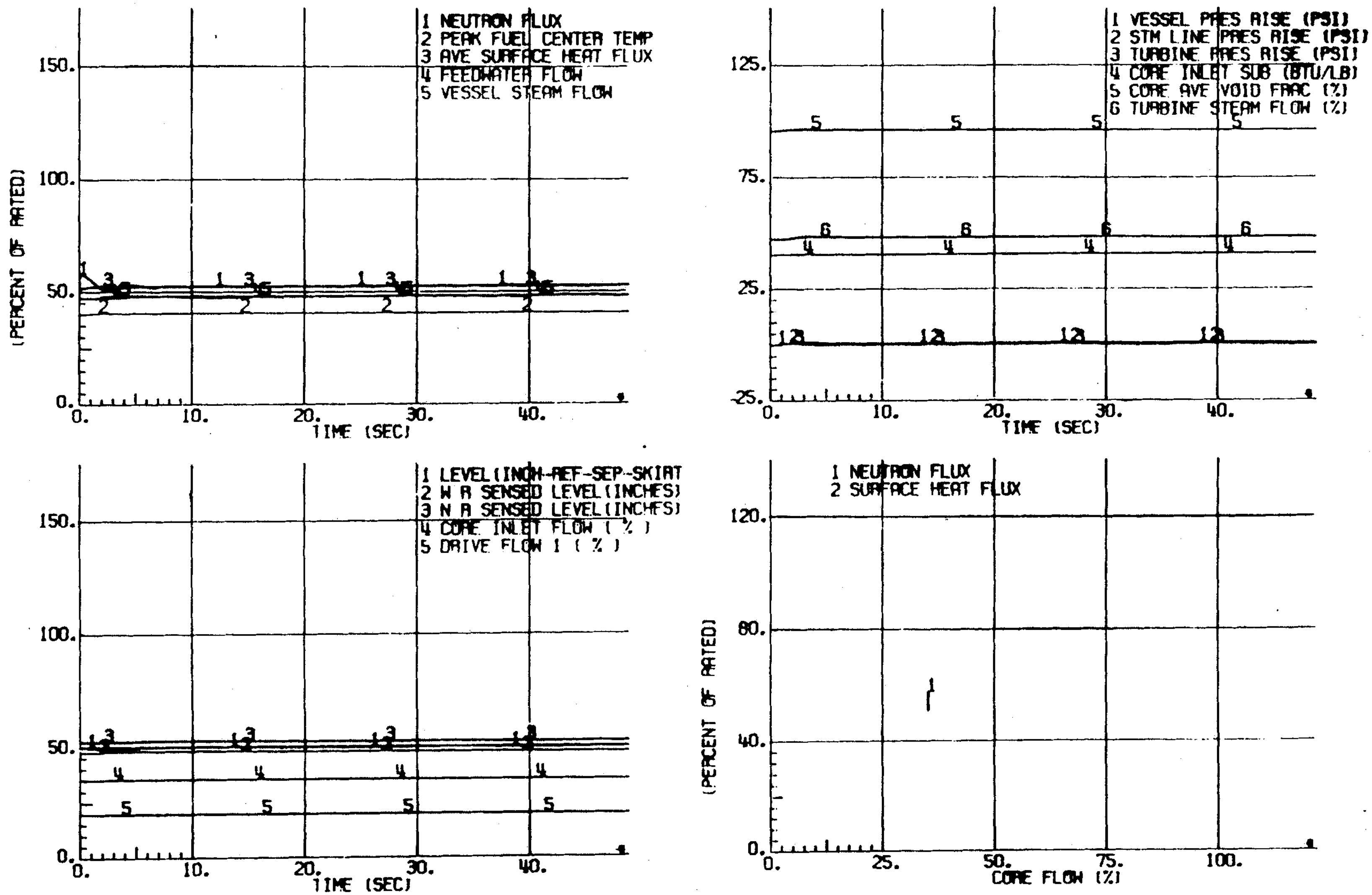


Figure 4.4-7b. 10 Cent Reactivity Step Change at 51.5% Power / 33% Flow (Natural Circulation) - Based on Initial Core Design

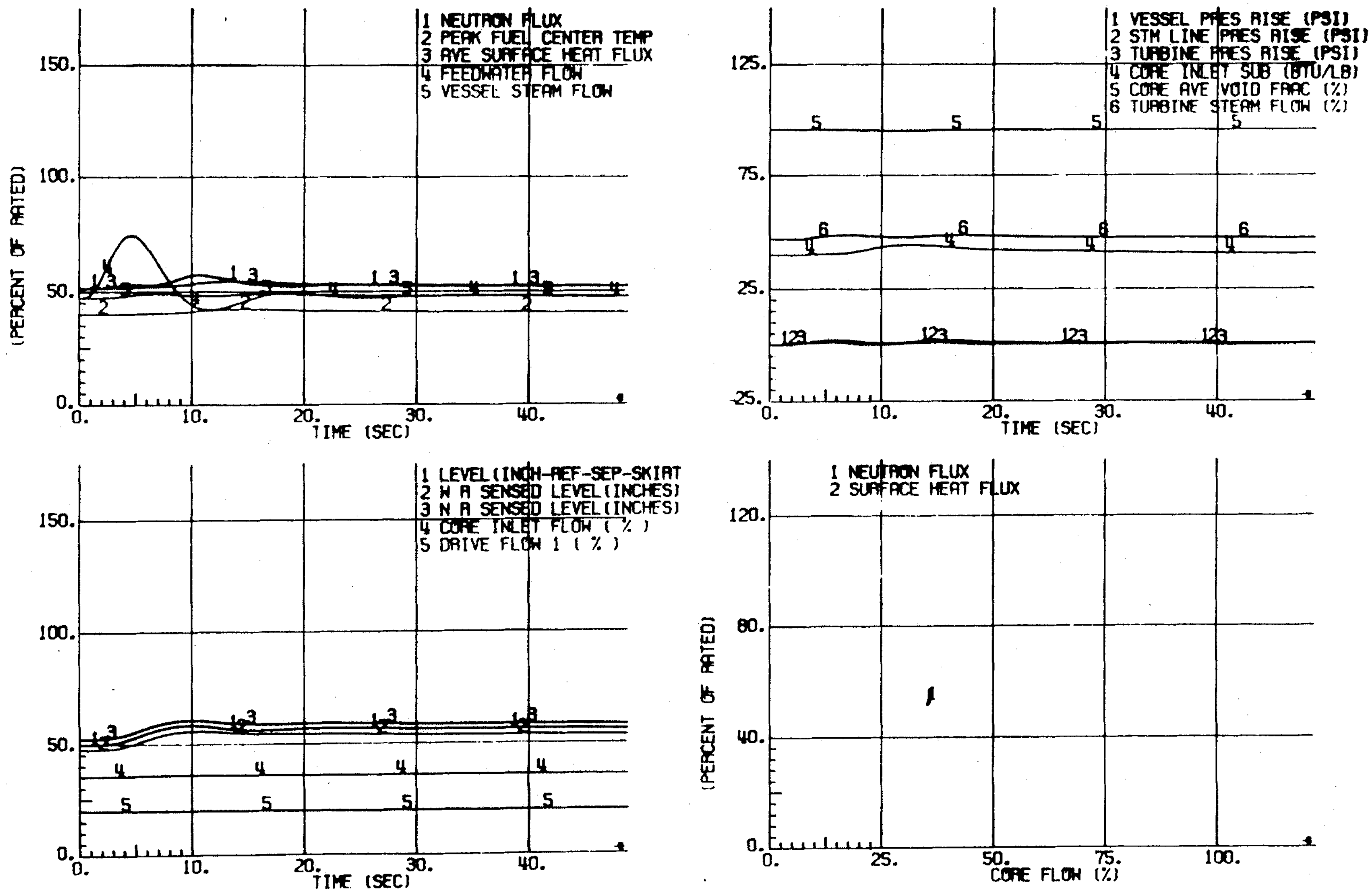


Figure 4.4-7c. 6-Inch Water Level Setpoint Step Change at 51.5% Power / 33% Flow (Natural Circulation) - Based on Initial Core Design

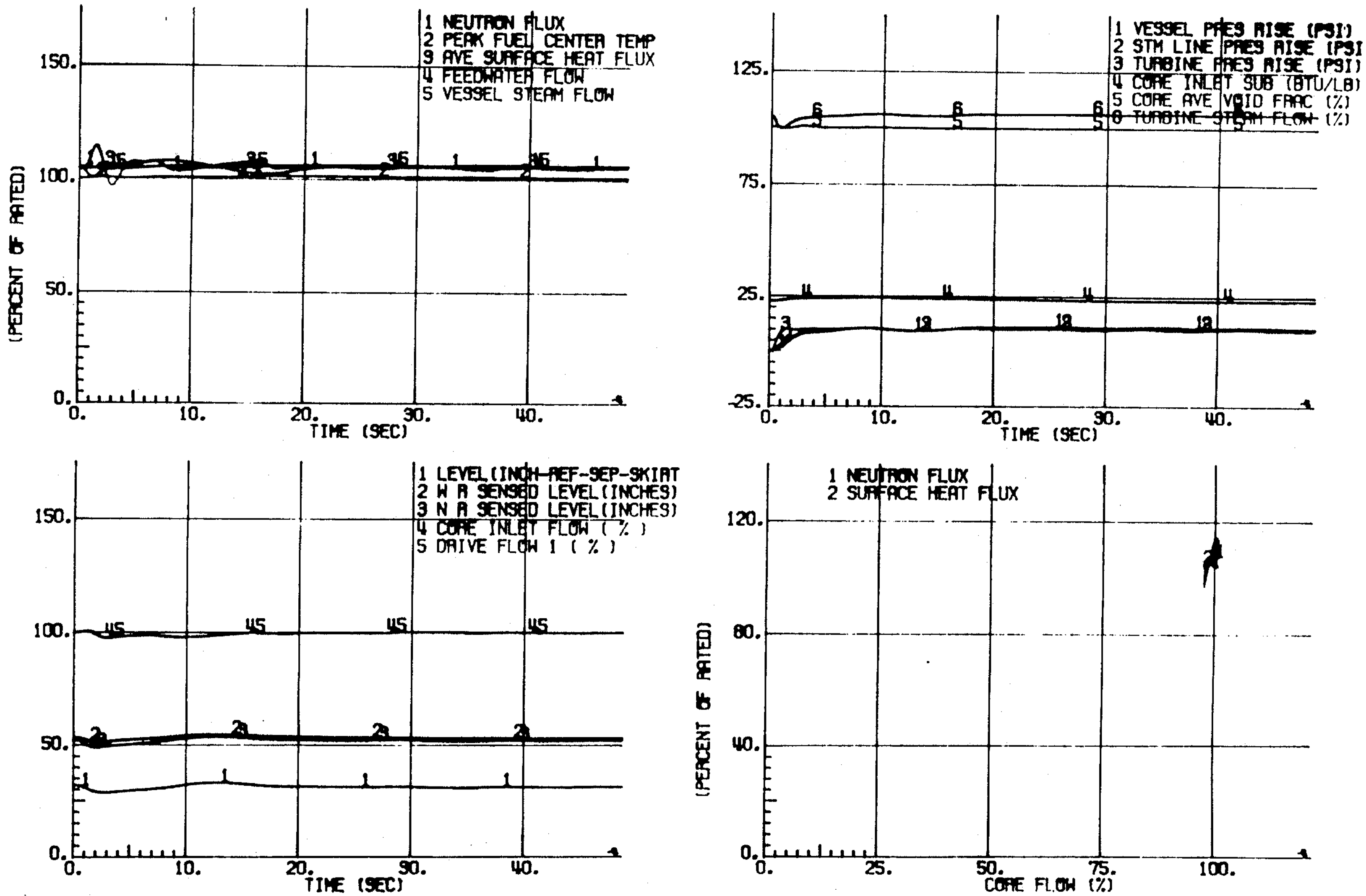


Figure 4.4-8a. 10 psi Pressure Regulator Setpoint Step Change at 104.2% Power / 100% Flow - Based on Initial Core Design

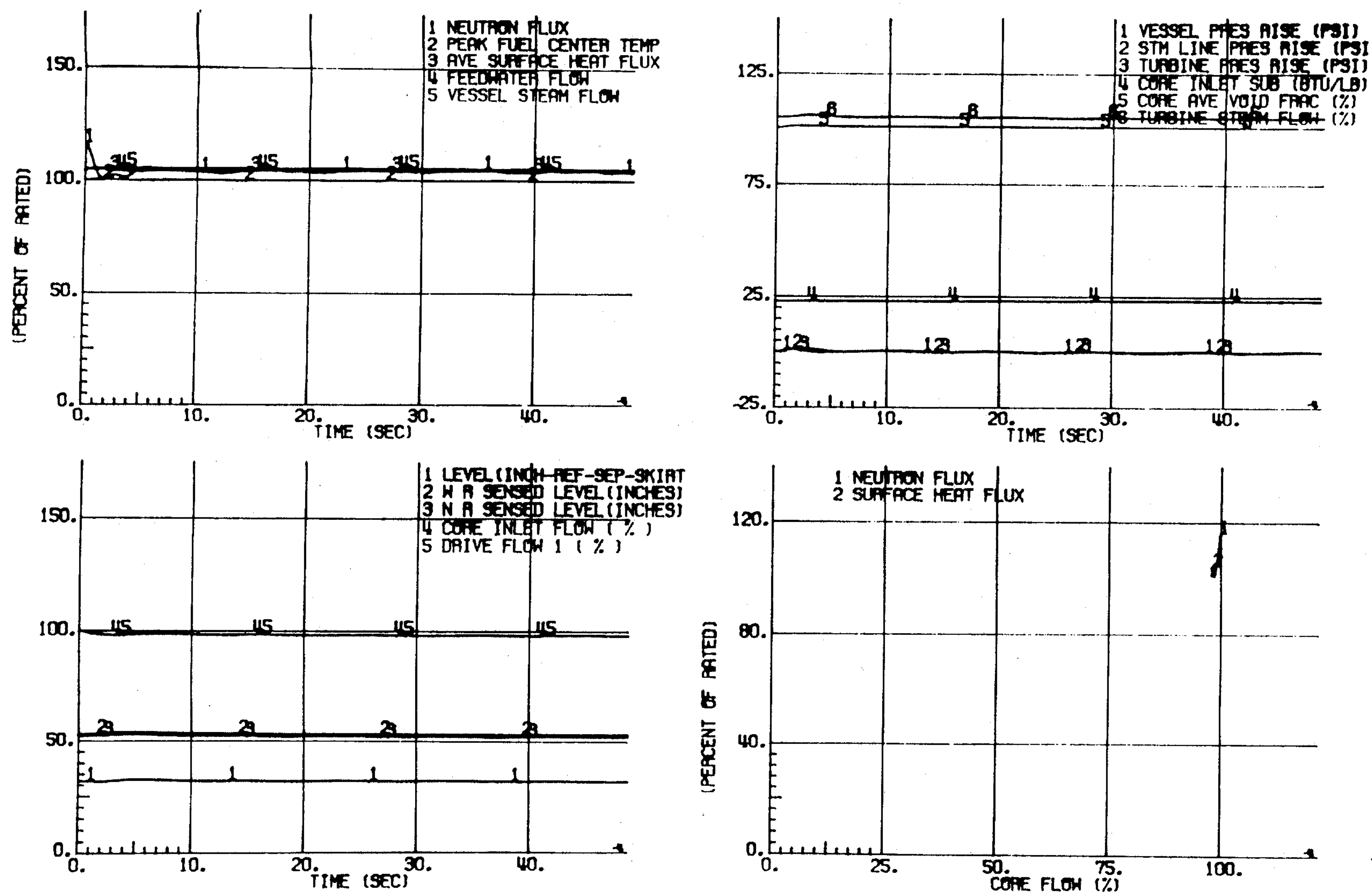


Figure 4.4-8b. 10 Cent Reactivity Step Change at 104.2% Power / 100% Flow - Based on Initial Core Design

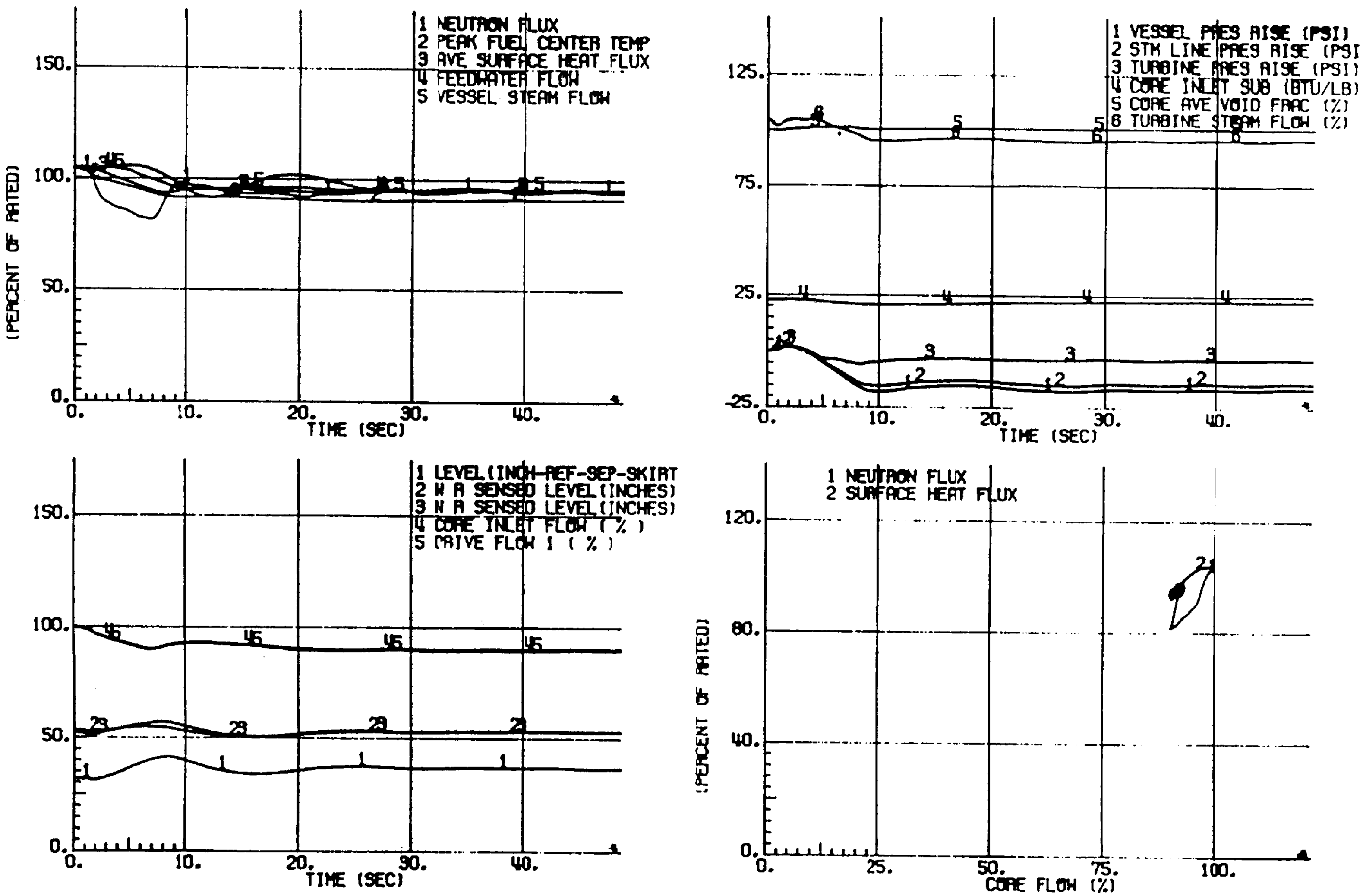


Figure 4.4-8c. 10 Percent Load Demand Step Change at 104.2% Power / 100% Flow Based on Initial Core Design

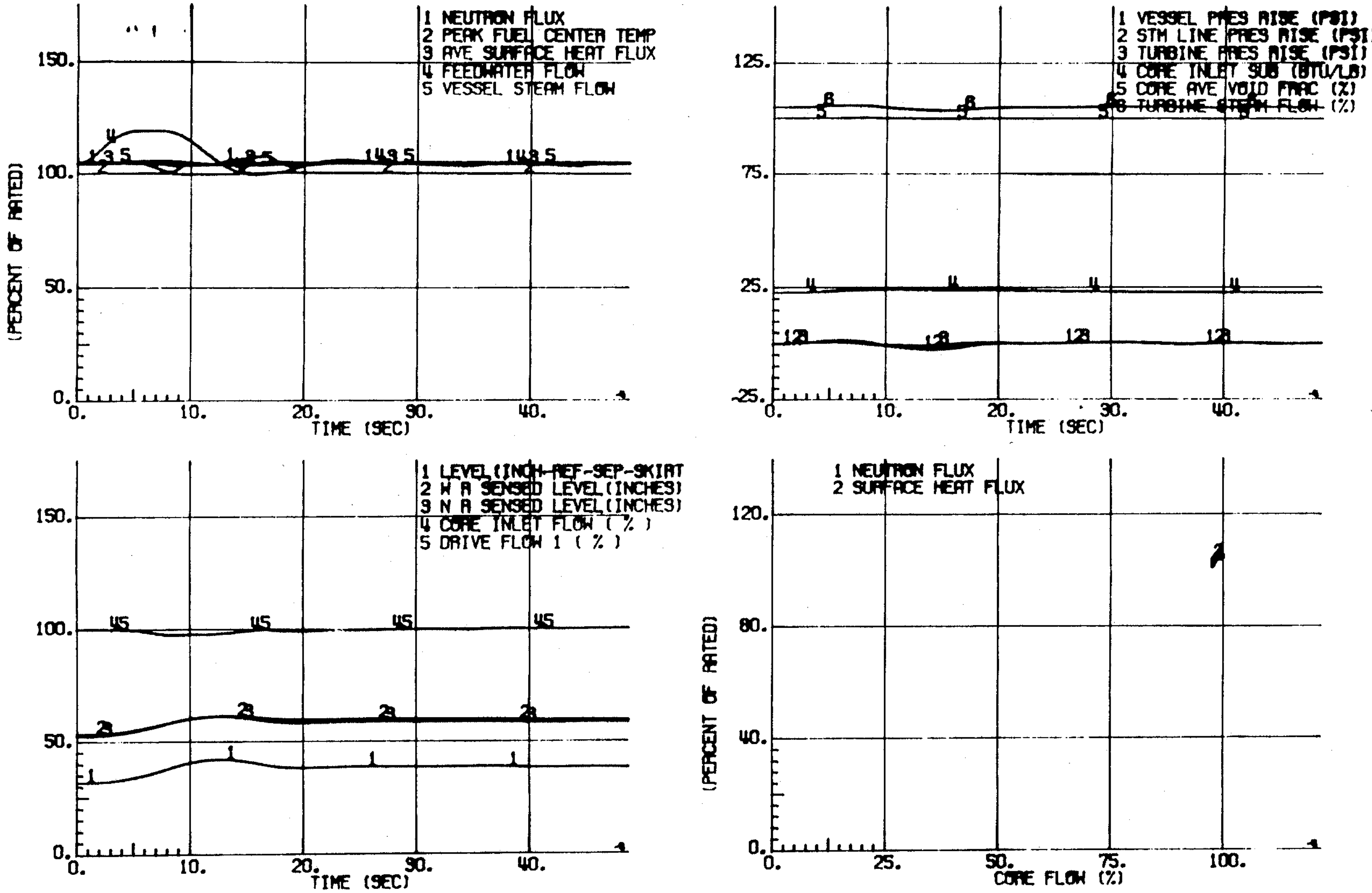


Figure 4.4-8d. 6-Inch Water Level Setpoint Step Change at 104.2% Power / 100% Flow - Based on Initial Core Design

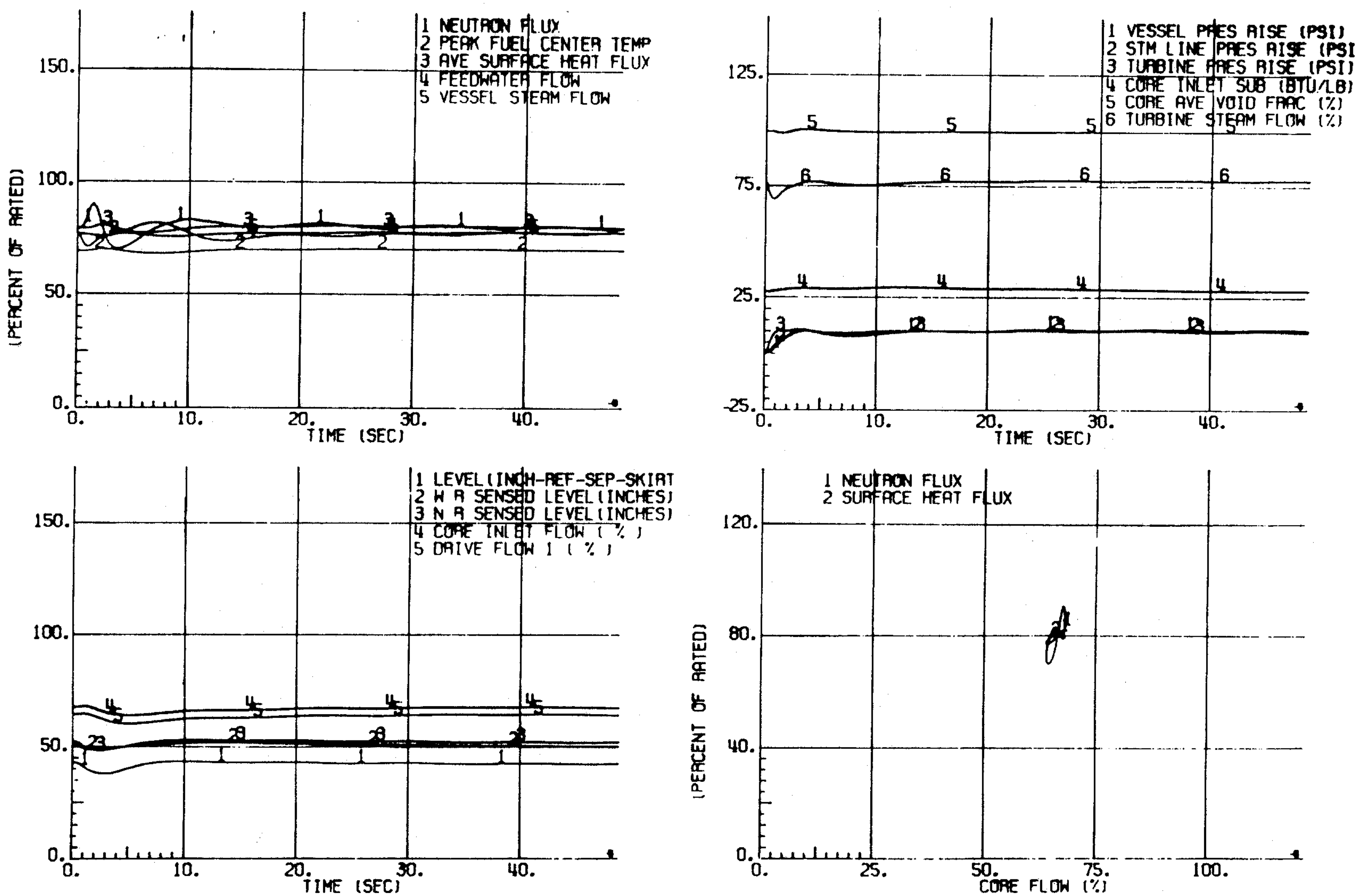


Figure 4.4-9a. 10 psi Pressure Regulator Setpoint Step Change at 79% Power / 63% Flow - Based on Initial Core Design

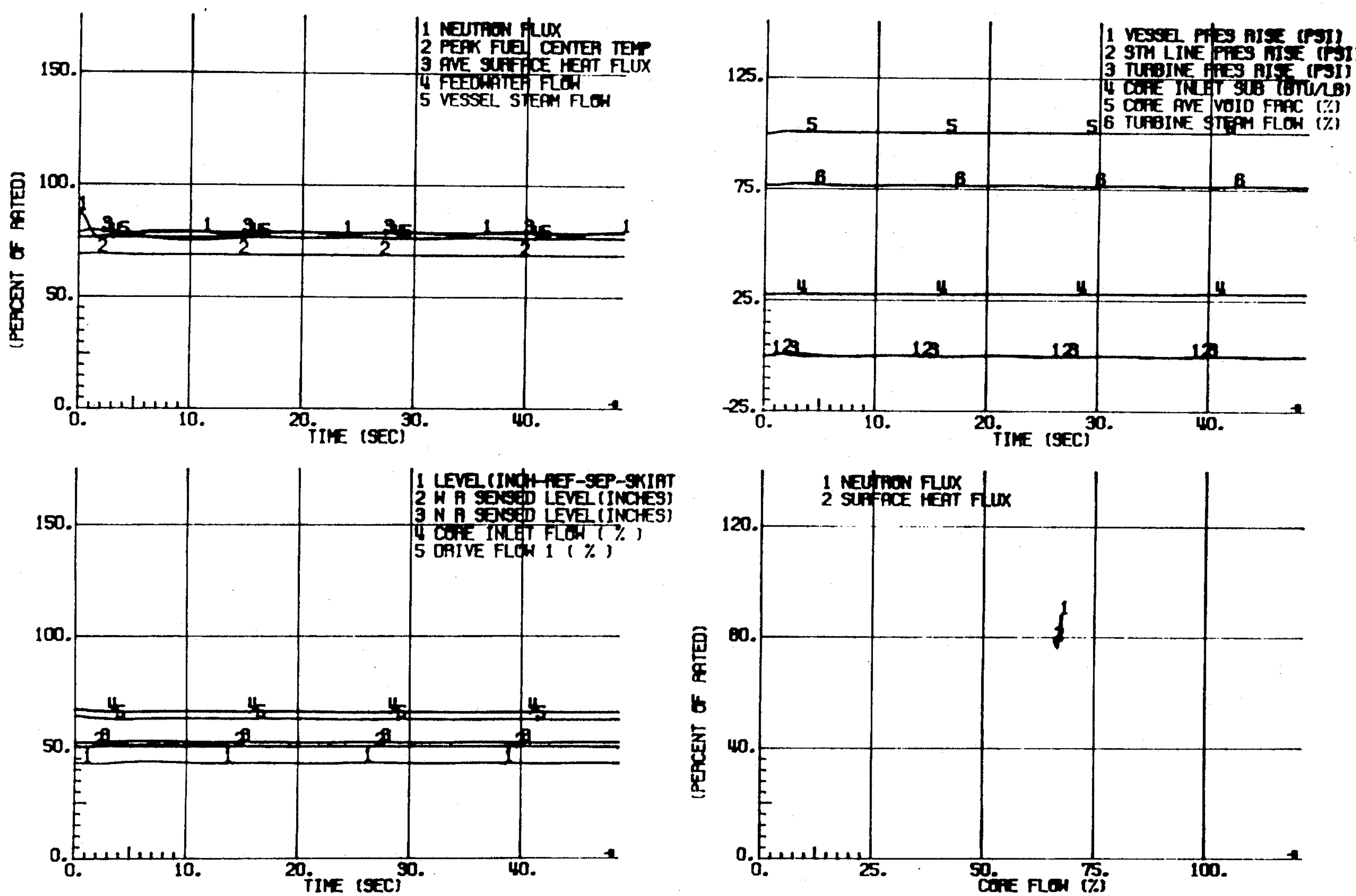


Figure 4.4-9b. 10 Cent Reactivity Step Change at 79% Power / 63% Flow - Based on Initial Core Design

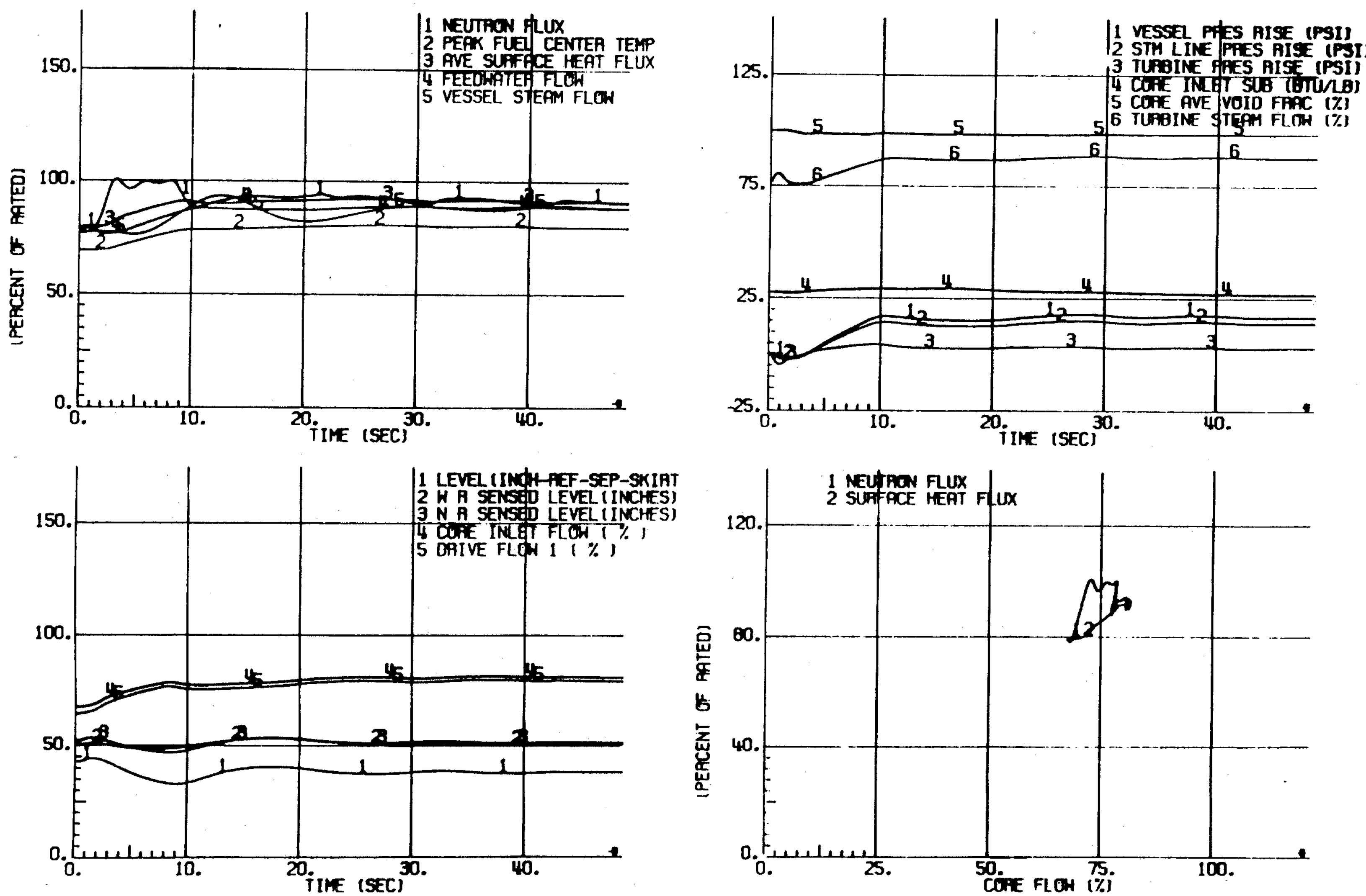


Figure 4.4-9c. 10 Percent Load Demand Step Change at 79% Power / 63% Flow
Based on Initial Core Design

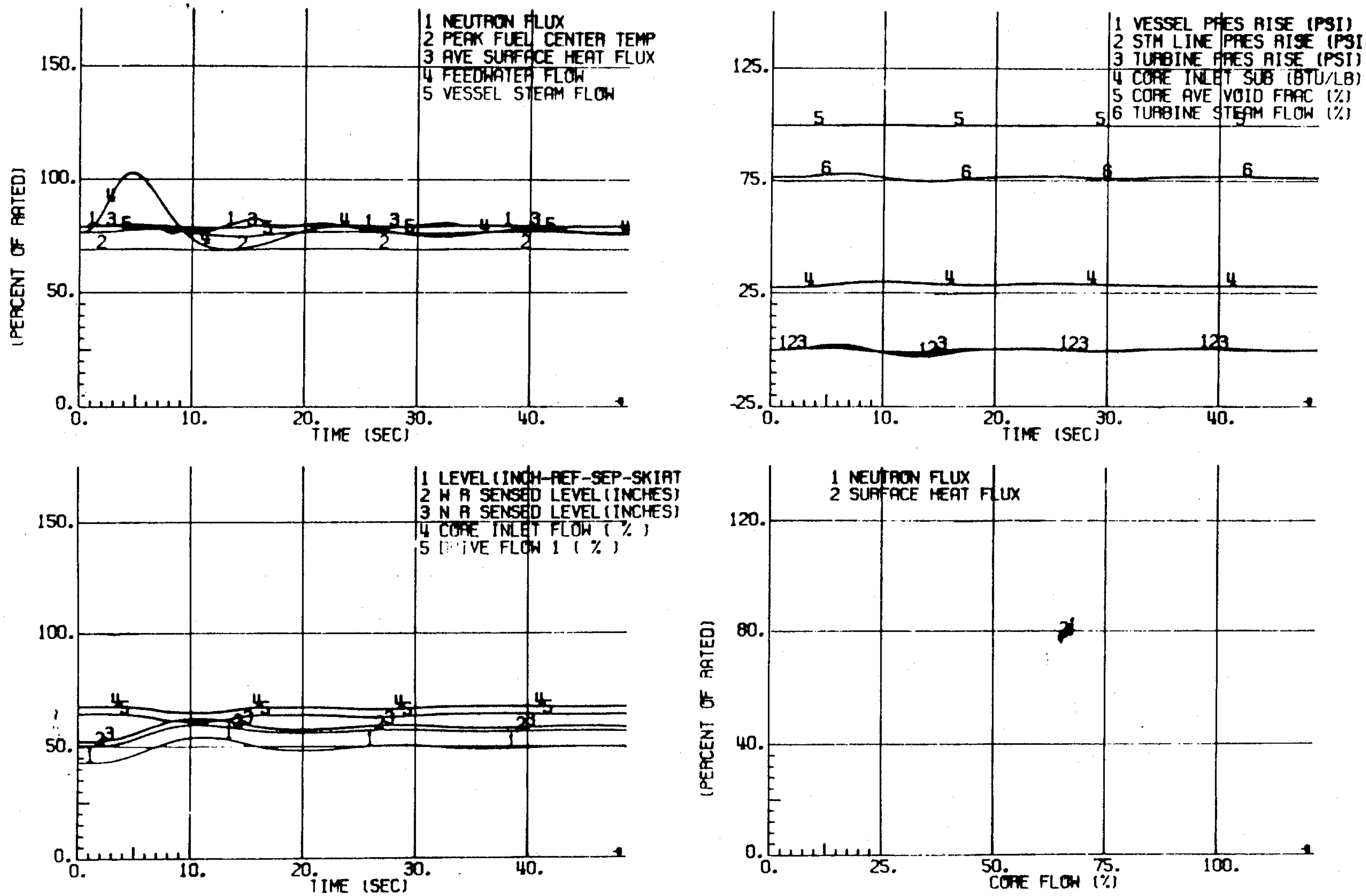


Figure 4.4-9d. 6-Inch Water Level Setpoint Step Change at 79% Power / 63% Flow - Based on Initial Core Design

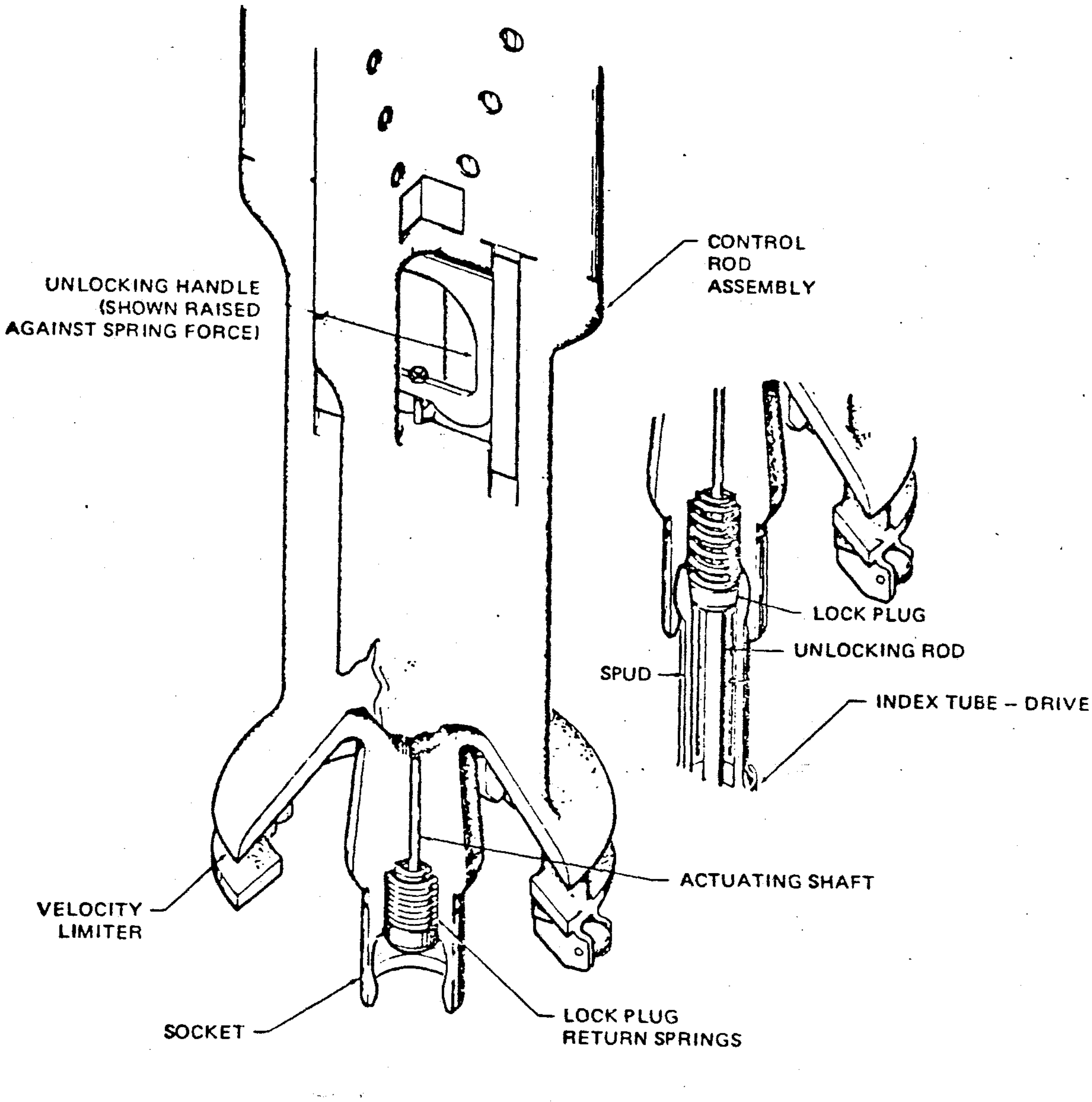


Figure 4.6-1. Control Rod to Control Rod Drive Coupling

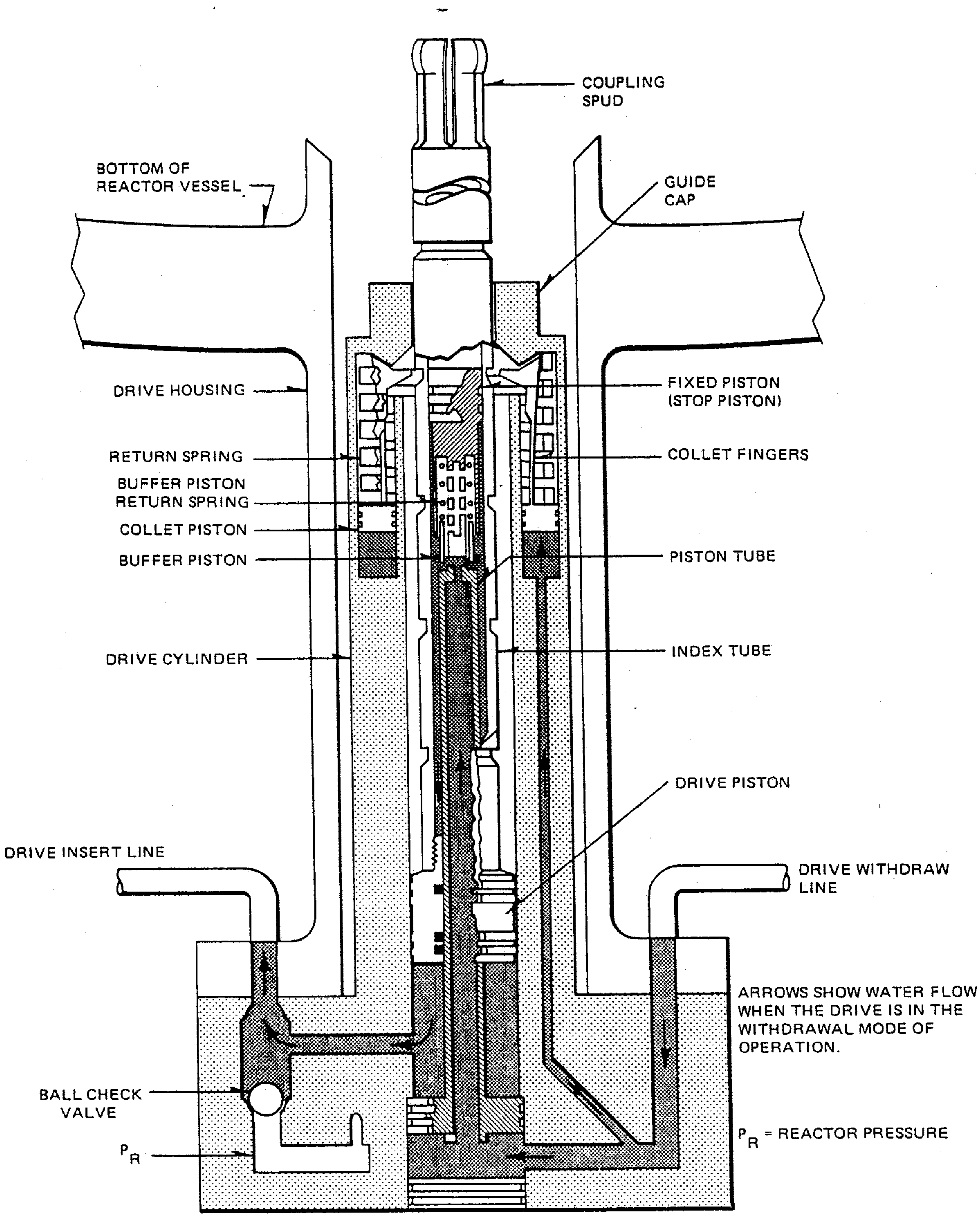


Figure 4.6-2. Control Rod Drive Unit

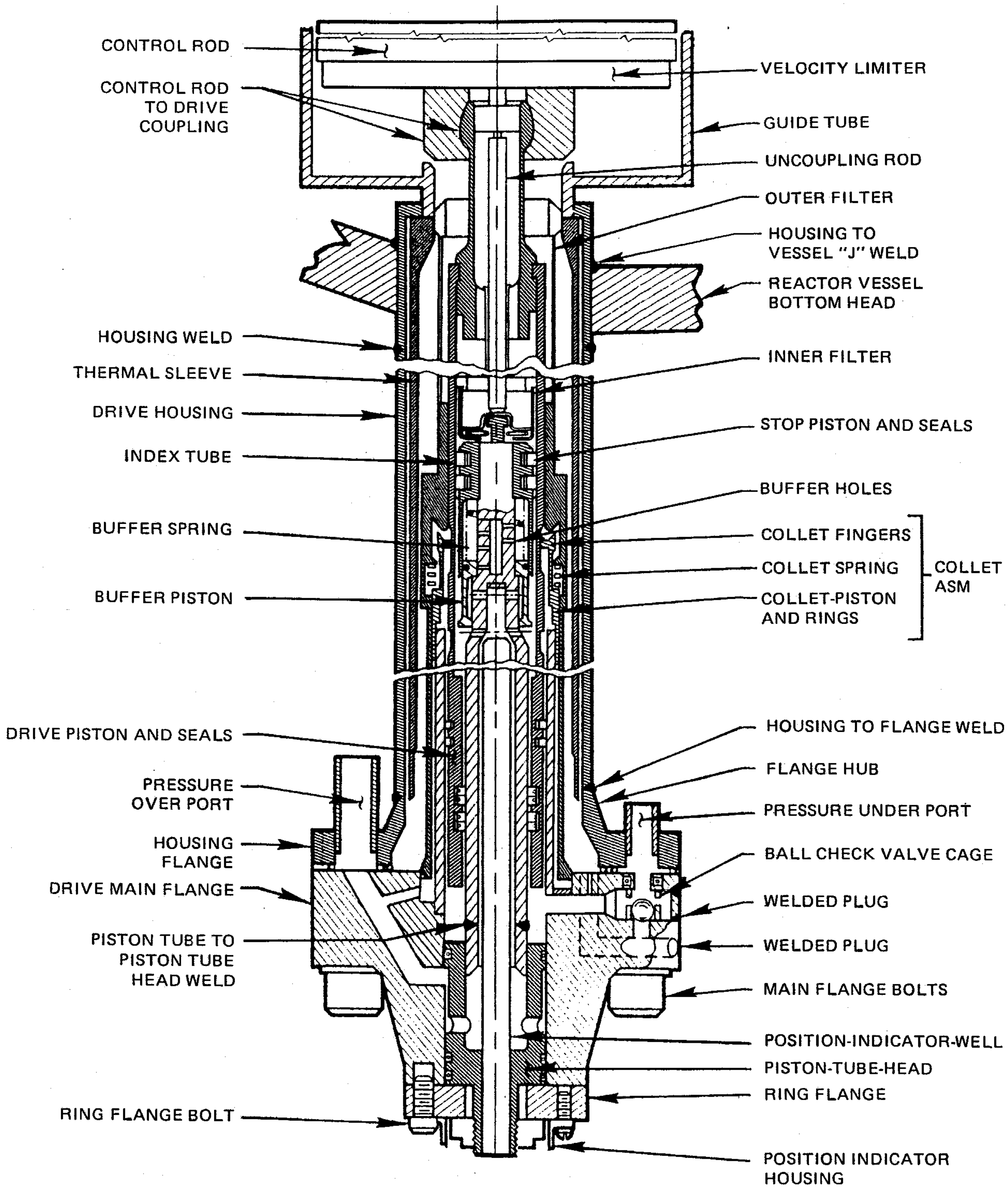
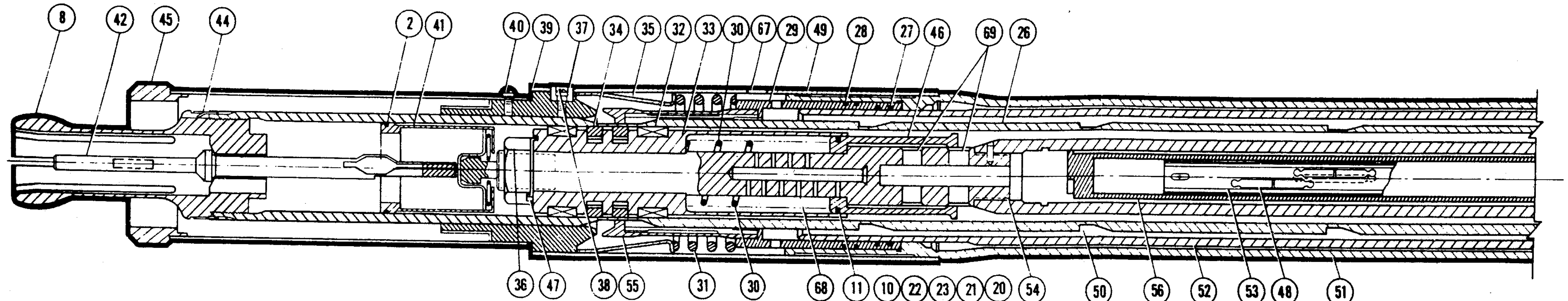
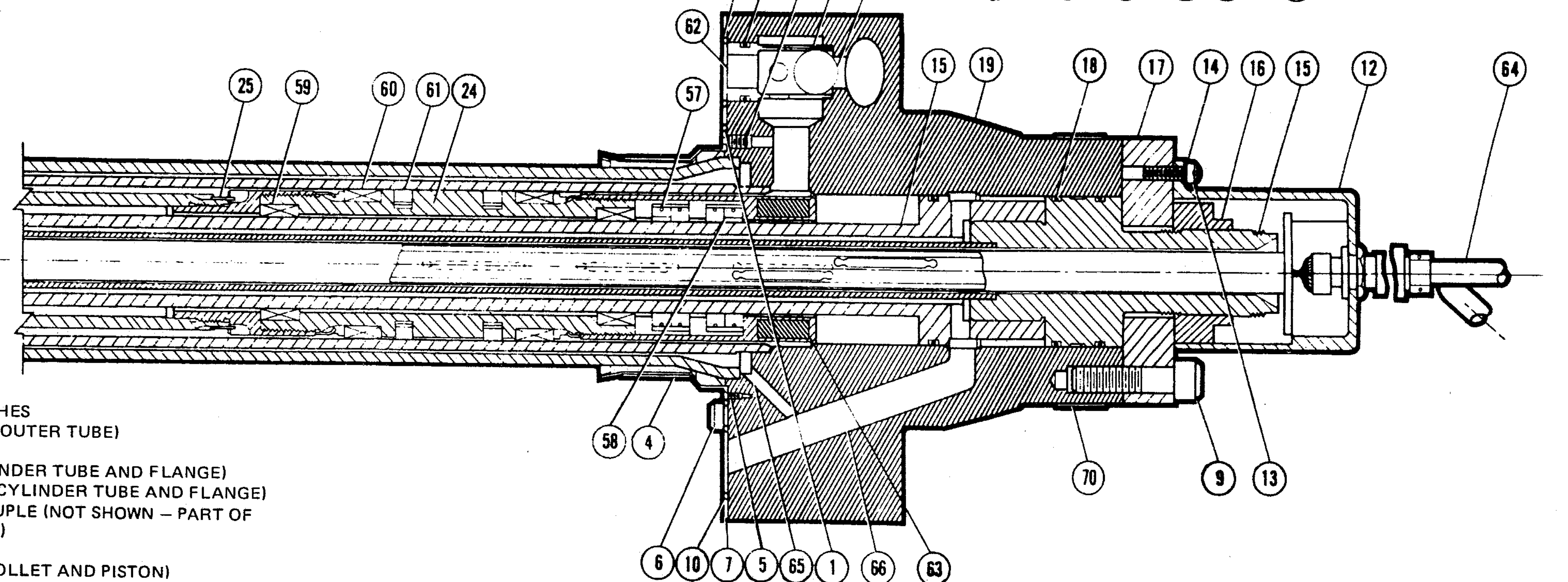


Figure 4.6-3 CONTROL ROD DRIVE SCHEMATIC





1 O-RING (FLANGE FACE)
2 SEAL RING (INNER FILTER)
3 SCREW, FLAT HEAD (O-RING SPACER MOUNTING NOT SHOWN)
4 STRAINER
5 SCREW, FLAT HEAD (STRAINER)
6 DOWEL PIN (DRIVE ALIGNMENT)
7 O-RING SPACER
8 SPUD (COUPLING)
9 CAP SCREW (RING FLANGE)
10 "O" RING (INSERT AND WITHDRAW PORTS)
11 SEAL RING (BUFFER PISTON)
12 POSITION INDICATOR PROBE
13 SCREW (POSITION PROBE)
14 WASHER (FOR P13)
15 PISTON TUBE
16 NUT (PISTON TUBE RETAINER)
17 RING FLANGE
18 O-RING (PISTON TUBE HEAD)
19 CYLINDER, TUBE AND FLANGE
20 BALL (CHECK VALVE)
21 BALL RETAINER
22 O-RING (BALL RETAINER)
23 SET SCREW PLUG (COOLING ORIFICE)
24 DRIVE PISTON
25 BAND
26 INDEX TUBE
27 SEAL RING (COLLET PISTON INTERNAL)
28 SEAL RING (COLLET PISTON EXTERNAL)
29 COLLET AND PISTON
30 BUFFER SPRING
31 COLLET SPRING
32 SPLIT BUSHING (STOP PISTON)
33 STOP PISTON
34 SEAL RING (STOP PISTON)
35 BARREL
36 NUT (STOP PISTON RETAINER)
37 PLUG (GUIDE CAP MOUNTING)
38 SCREW, FILLISTER HEAD (GUIDE CAP)
39 GUIDE CAP
40 DRILLED FIL HD SCREW (OUTER FILTER)
41 INNER FILTER
42 ROD (UNCOUPLING)
44 BAND (SPUD)
45 FILTER (OUTER)
46 BUFFER PISTON
47 WASHER, LOCK
48 POSITION INDICATOR SWITCHES
49 COLLET HOUSING (PART OF OUTER TUBE)
50 INDEX TUBE NOTCH
51 OUTER TUBE (PART OF CYLINDER TUBE AND FLANGE)
52 INNER CYLINDER (PART OF CYLINDER TUBE AND FLANGE)
53 LOCATION OF THERMAL COUPLE (NOT SHOWN – PART OF POSITION INDICATOR PROBE)
54 BUFFER SHAFT
55 COLLET FINGER (PART OF COLLET AND PISTON)
56 INDICATOR TUBE (PART OF PISTON TUBE)
57 INNER SEALS (DRIVE PISTON – DRIVE UP SEALS)
58 INNER SEALS (DRIVE PISTON – DRIVE DOWN SEALS)
59 INTERNAL BUSHING (DRIVE PISTON)
60 EXTERNAL BUSHING (DRIVE PISTON)
61 OUTER SEALS (DRIVE PISTON)
62 INSERT PORT (INSERT AND SCRAM INLET/WITHDRAW OUTLET)
63 RING MAGNET (PART OF DRIVE PISTON)
64 CABLE (POSITION INDICATOR)
65 PORT TO COLLET PISTON (WITHDRAW PRESSURE TO COLLET PISTON)
66 WITHDRAW PORT (WITHDRAW INLET INSERT OUTLET AND SCRAM DISCHARGE)
67 WATER PORTS IN COLLET HOUSING
68 BUFFER ORIFICES IN BUFFER SHAFT
69 FLOW PORTS IN BUFFER SHAFT
70 NAMEPLATE

Figure 4.6-4 CONTROL ROD DRIVE UNIT (CUTAWAY)

Figures 4.6-5 through 4.6-7 have been deleted.

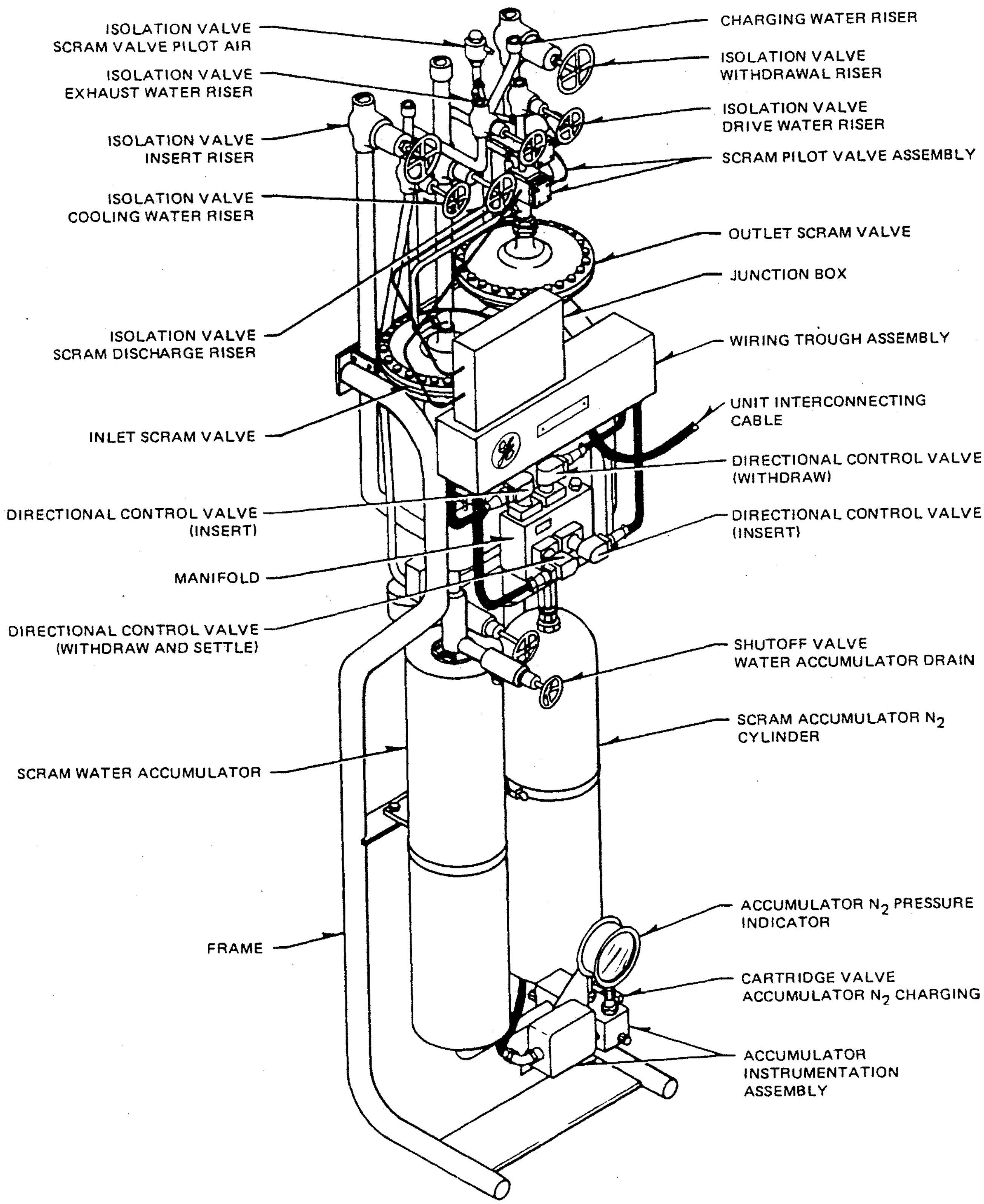


Figure 4.6-8. Control Rod Drive Hydraulic Control Unit

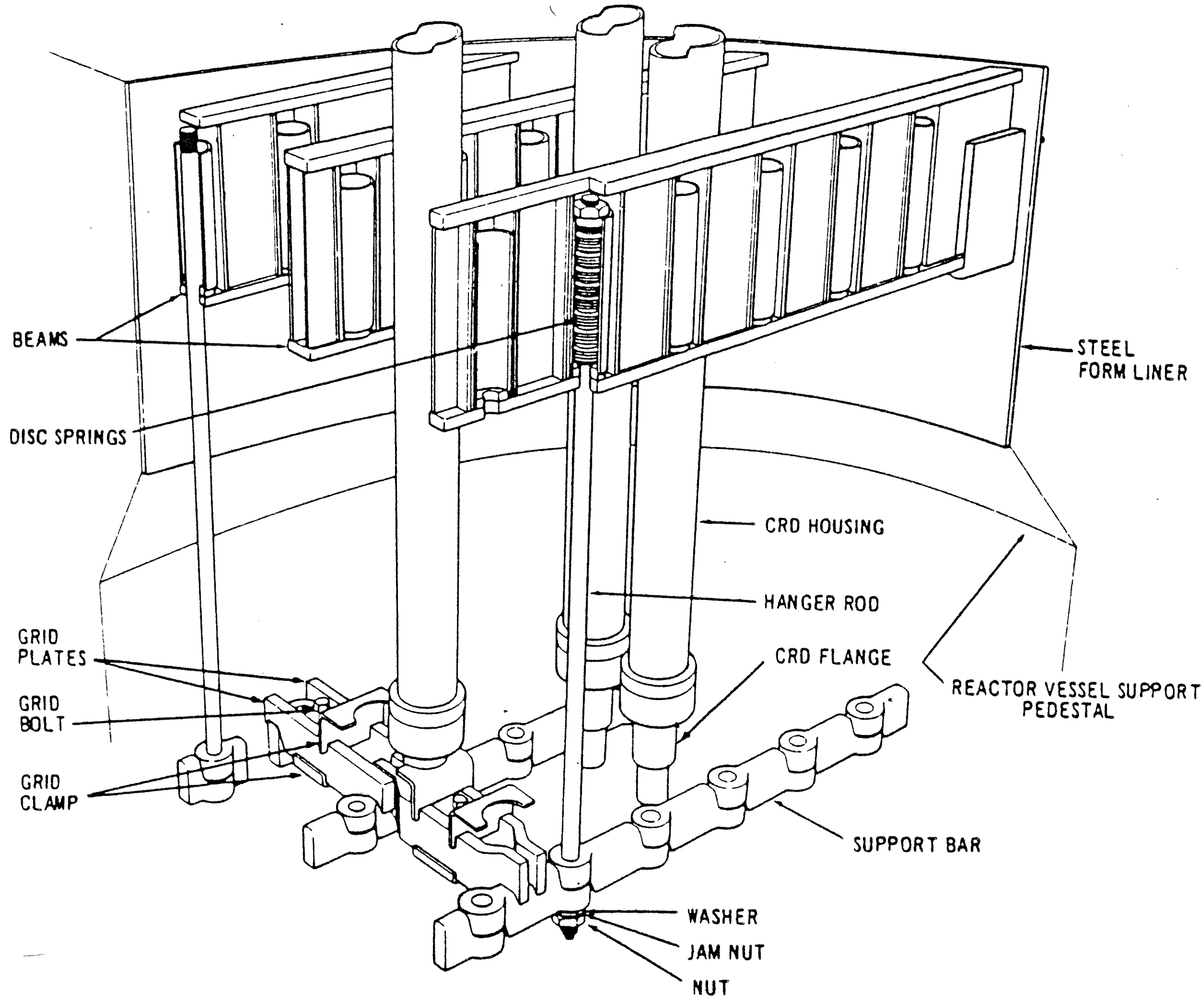


Figure 4.6-9. Control Rod Drive Housing Support