Chapter 4

REACTOR

TABLE OF CONTENTS

Chapter 4

REACTOR

TABLE OF CONTENTS (Continued)

Chapter 4

REACTOR

TABLE OF CONTENTS (Continued)

Chapter 4

REACTOR

TABLE OF CONTENTS (Continued)

Chapter 4

REACTOR

TABLE OF CONTENTS (Continued)

Chapter 4

REACTOR

TABLE OF CONTENTS (Continued)

Section Page

Chapter 4

REACTOR

TABLE OF CONTENTS (Continued)

Chapter 4

REACTOR

TABLE OF CONTENTS (Continued)

Section | Page

Chapter 4

REACTOR

TABLE OF CONTENTS (Continued)

Chapter 4

REACTOR

TABLE OF CONTENTS (Continued)

Chapter 4

REACTOR

LIST OF TABLES

Chapter 4

REACTOR

LIST OF FIGURES

Chapter 4

REACTOR

LIST OF FIGURES (Continued)

Chapter 4

REACTOR

4.1 SUMMARY DESCRIPTION

This section was prepared using the licensing topical report, "General Electric Standard Application for Reactor Fuel" (Reference 4.1-1), "GE14 Amendment 22 Compliance Document" (Reference 4.1-27), the cycle-specific design report (Reference 4.1-2), the "Fuel Bundle Information Report" (Reference 4.1-3), and the "Generic Mechanical Design Criteria for BWR Fuel Designs" (Reference 4.1-4).

The reactor assembly consists of the reactor vessel, 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 5.3-5 shows the arrangement of reactor assembly components. A summary of the important design and performance characteristics is given in Section 1.3. Loading conditions for reactor assembly components are specified in Section 3.9. The core load varies for each cycle and is shown in Reference 4.1-2.

4.1.1 REACTOR VESSEL

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

4.1.2 REACTOR INTERNAL COMPONENTS

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

4.1.2.1 Reactor Core

4.1.2.1.1 General

The reactor core is composed of fuel assemblies manufactured by Global Nuclear Fuel (GNF) and AREVA NP.

A number of important features of the BWR core design are summarized in the following paragraphs:

a. The BWR core mechanical design is based on conservative application of stress limits, operating experience, and experimental test results. The moderate pressure levels characteristics of a direct cycle reactor (approximately 1035 psia) result in moderate cladding temperatures and stress levels;

b. 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;

c. 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;

d. The design power distribution used in sizing the core represents a worst expected state of operation;

e. The AREVA NP thermal margin analyses have been applied to the mixed core to ensure that more than 99.9% of the fuel rods in the core are expected to avoid boiling transition for the most severe anticipated operational occurrences described in Chapter 15. The possibility of boiling transition occurring during normal reactor operation is insignificant; and

f. Because of the large negative moderator density coefficient of reactivity, the BWR has a number of inherent advantages. These are the uses of coolant flow for load following, the inherent self-flattening of the radial power distribution, the ease of control, the spatial xenon stability, and the ability to override xenon to follow load.

Boiling water reactors do not usually have instability problems due to xenon. This has been demonstrated by special tests which were conducted on operating BWRs and by calculations. Xenon transients are highly damped in a BWR due to the large negative power coefficient of reactivity (Reference 4.1-7).

Columbia Generating Station (CGS) has installed a stability monitoring system to ensure hydrodynamic stability while operating in regions susceptible to instability. Stability monitoring system limits are specified in the Technical Specifications for operation in certain regions of the power/flow map to provide assurance of stable core operation for CGS.

Important features of the reactor core arrangement are as follows:

a. The original bottom-entry cruciform control rods consist of B_4C in stainless steel tubes surrounded by a stainless steel sheath;

b. The bottom-entry cruciform Duralife 215 control rods consist of 18 high-purity stainless steel tubes at each wing filled with boron-carbide and three hafnium rods at the edge of each wing and a hafnium plate at the top;

c. The bottom-entry cruciform Marathon control rods consist of 17 high-purity stainless steel tubes in each wing. Eleven of the tubes are filled with boron-carbide, two of the tubes are partially filled with boron-carbide and four tubes are filled with hafnium rods (three at the outer edge of each wing and one at the center of the wing). See Figure 4.2-1.5 for details;

d. The in-core location of the startup and power range instruments provides coverage of the large reactor core and provides an acceptable signal-to-noise ratio and neutron-to-gamma ratio. All in-core instrument leads enter from the bottom and the instruments are in service during refueling. In-core instrumentation is further discussed in Sections 7.6.1.4 and 7.7.1.6;

e. As shown by experience obtained at 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;

f. The Zircaloy-2 and 4 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;

g. 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; and

h. The selected control rod pitch represents a practical value of individual control rod reactivity worth and allows ample clearance below the pressure vessel between control rod drive mechanisms for ease of maintenance and removal.

4.1.2.1.2 Core Configuration

The reactor core is arranged as an upright circular cylinder containing a large number of fuel cells and is located within the reactor vessel. The coolant flows upward through the core. The BWR core is composed of essentially two components: fuel assemblies and control rods. The General Electric Company (GE) control rod mechanical configuration (see Figure 4.2-1) is basically the same as used in all GE BWRs.

4.1.2.1.3 Fuel Assembly Description

The core is loaded with AREVA NP and GNF reload fuel. The AREVA NP reload fuel assemblies are composed of a 10 x 10 array of fuel rods with a single, large, central water channel (References 4.1-20, 4.1-24 and 4.1-26). The GNF reload fuel assemblies are composed of a 10 x 10 array of fuel rods with two large central water rods (References 4.1-27).

4.1.2.1.3.1 Fuel Rod. A fuel rod consists of UO_2 pellets and a Zircaloy-2 cladding tube. A fuel rod is made by stacking pellets into a Zircaloy-2 cladding tube, which is sealed by welding Zircaloy end plugs in each end of the tube. The AREVA NP fuel rods are pressurized to 95 psia (see Reference 4.1-20, 4.1-24 and 4.1-26). The GNF fuel rods are pressurized to 10 atmospheres (References 4.1-27).

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

The rod is designed to withstand the applied loads, both external and internal. The fuel pellet is sized to provide sufficient volume within the fuel tube to accommodate differential expansion between fuel and clad. Overall fuel rod design is conservative in its accommodation of the mechanisms affecting fuel in a BWR environment. Fuel rod design bases are discussed in more detail in Section 4.2.1.

4.1.2.1.3.2 Fuel Bundle. The fuel bundle has two important design features:

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

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

The fuel bundles are designed to meet all the criteria for core performance and to provide ease of handling. Selected fuel rods in each bundle differ from the others in uranium enrichment. This arrangement produces more uniform power production across the fuel bundle and thus allows a significant reduction in the amount of heat transfer surface required to satisfy the design thermal limitations.

The AREVA NP reload fuel bundles contain 91 fuel rods and a single, large, central water channel, all in a 10 x 10 array. The GNF reload bundle contains 92 fuel rods and 2 large central water rods, all in a 10 x 10 array.

4.1.2.1.4 Assembly Support and Control Rod Location

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

The top guide, mounted inside the shroud, provides lateral support and guidance for each fuel assembly. The reactivity of the core is controlled by cruciform control rods 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 insert the control rod in less than 7 sec during the scram mode of operation. Bottom entry allows optimum power shaping in the core, ease of refueling, and convenient drive maintenance.

4.1.2.2 Shroud

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

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

4.1.2.3 Shroud Head and Steam Separators

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

In each separator, the steam-water mixture rising from the standpipe impinges on vanes which give the mixture a spin to establish a vortex wherein the centrifugal forces separate the steam from the water. Steam leaves the separator at the top and passes into the wet steam plenum below the dryer. The separated water exits from the lower end of the separator and enters the pool that surrounds the standpipes to enter the downcomer annulus. An internal steam separator diagram is shown in Figure 4.1-1.

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

4.1.2.4 Steam Dryer Assembly

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

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 Rods

The cruciform shaped control rods contain 76 stainless steel tubes (19 tubes in each wing of the cruciform) filled with vibration compacted boron-carbide powder. The tubes are seal welded with end plugs on either end. Stainless steel balls are used to separate the tubes into individual compartments. The stainless steel balls are held in position by a slight crimp in the tube. The individual tubes act as pressure vessels to contain the helium gas released by the boron-neutron capture reaction.

The tubes are held in a cruciform array by a stainless steel sheath extending the full length of the tubes. A top handle, shown in Figure 4.2-1, aligns the tubes and provides structural rigidity at the top of the control rod. Rollers, housed in the handle, provide guidance for control rod insertion and withdrawal. A bottom casting is also used to provide structural rigidity and contains positioning rollers and a parachute-shaped velocity limiter. The handle and lower casting are welded into a single structure by means of a small cruciform post located in the center of the control rod. A steel stiffener is located approximately at the midspan of each cruciform wing. The control rods can be positioned at 6-in. steps and have a nominal withdrawal and insertion speed of 3 in./sec.

The foregoing description of the control rods applies to the design of the original control rods. There have been two different replacement control rods used, Duralife 215 and Marathon control rods. These replacement rods have design changes that increase the neutron absorption and make other material property improvements. The newer control rods are similar to and are fully interchangeable with the original control rod assemblies and are compatible with the existing nuclear steam supply system hardware.

The Duralife 215 control rods differ from the previous control rods in that a hafnium absorber plate is used at the top of each cruciform section, hafnium absorber rods replace several of the boron carbide absorber rods on the periphery of each cruciform section, and the stainless steel stiffener is removed from each wing. There are 21 absorber rods in each wing, of which 18 are stainless-steel tubes containing boron carbide and three are hafnium rods. The outside diameter remains the same. The length of the absorber column in these rods has been reduced from 143 in. to 137 in. to accommodate the top 6-in.-high hafnium plate. The increased volume of neutron absorber material increases the relative reactivity worth in the cold condition and increases the nuclear lifetime.

The Marathon control rod blades are an improved version of the Duralife 215 control blades and have the absorber and sheath arrangement replaced with an array of square tubes, which results in reduced weight and increased absorber volume. The square tubes each have four lobes to allow adjacent tubes to be welded to each other. The absorber tubes are welded lengthwise to form the four wings of the control rod. Each wing is comprised of 17 absorber tubes. The absorber tubes each act as an individual pressure chamber for the retention of helium. The region between each pair of square tubes is filled with helium and sealed top and bottom by welding. The four wings are then welded to the tie rod to form the cruciform-shaped member of the control rod. The Marathon control rod blade has the full-length tie rod replaced with a segmented tie rod, which also reduces weight.

The square tubes are circular inside and are loaded with either B_4C or hafnium. The B_4C is contained in separate capsules to prevent its migration. The capsules are placed inside the square absorber tubes and are smaller than the absorber tube inside diameter, allowing the B_4C to swell before making contact with the absorber tubes thereby increasing stress corrosion resistance. Empty tubes may be used adjacent to the tie rods to achieve the desired reactivity worth. The combination of absorbers and absorber tubes is based on the needed initial reactivity worth. In addition, empty capsules are used in some absorber tubes to provide a plenum for helium released during B_4C burnup.

The velocity limiter, shown in Figure 4.2-2, 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 of recirculation flow that leaks through the several leakage flow paths, which are as follows:

a. The area between fuel channel and fuel assembly nosepiece,

b. The area between fuel assembly nosepiece and fuel support piece,

c. Holes in the lower tie plate,

d. The area between fuel support piece and core plate,

e. The area between core plate and shroud,

f. Holes in the core plate near power range monitor instrument guide tubes,

g. Various leakage paths around the control rod guide tubes, and

h. Control rod drive cooling water.

4.1.3.3 Supplementary Reactivity Control

Supplemental reactivity control is achieved with burnable poison. The supplementary burnable poison is gadolinia (Gd_2O_3) mixed with UO_2 in selected fuel rods in each fuel bundle.

4.1.4 ANALYSIS TECHNIQUES

4.1.4.1 Reactor Internal Components

Computer codes used for the analysis of the internal components as a basis for the original operating license are listed as follows:

a. MASS
b. SNAP (MULTISHELL)
c. GASP
d. NOHEAT
e. FINITE
f. DYSEA
g. SHELL 5
h. HEATER
i. FAP-71
j. CREEP-PLAST

The following italicized detailed descriptions of these programs are historical and were provided to support the application for an operating license.

4.1.4.1.1 MASS (Mechanical Analysis of Space Structure)

4.1.4.1.1.1 Program Description. This is a proprietary program of GE and is an outgrowth of the PAPA (Plate and Panel Analysis) program originally developed by L. Beitch in the early 1960s. The program is based on the principle of the finite element method. Governing matrix equations are formed in terms of joint displacements using a "stiffness-influence-coefficient" concept originally proposed by L. Beitch (Reference 4.1-9). 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 is using a past revision of MASS. This revision is identified as revision "0" in the computer production library. The program operates on the Honeywell 6000 computer.

4.1.4.1.1.3 History of Use. Since its development in the early 1960s, 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 GE have also applied the program to a wide range of engineering problems. The Nuclear Energy Division uses it mainly for piping and reactor internals analyses.

4.1.4.1.2 SNAP (MULTISHELL)

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

4.1.4.1.2.2 Program Version and Computer. The current version maintained by the GE Jet Engine Division at Evandale, Ohio, is being used on the Honeywell 6000 computer in GE/NED.

4.1.4.1.2.3 History of Use. The initial version of the Shell Analysis Program was completed by the Jet Engine Division in 1961. Since then, a considerable amount of modification and addition has been made to accommodate its broadening area of application. Its application in the Nuclear Energy Division has a history longer than 10 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 GE.

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 from the developer, Professor E. L. Wilson, operates on the Honeywell 6000 computer.

4.1.4.1.3.3 History of Use. The program was developed by Professor E. L. Wilson in 1965 (Reference 4.1-10). 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 the 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 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 histories of all the nodal points established by the user. The program can handle multitransient temperature input.

4.1.4.1.4.2 Program Version and Computer. The current version of the program is an improvement of the program originally developed by I. Farhoomand and Professor E. L. Wilson of University of California at Berkeley (Reference 4.1-11). The program operates on the Honeywell 6000 computer.

4.1.4.1.4.3 History of Use. The program was developed in 1971 and installed in GE 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 provisions for thermal, mechanical, and body force loads. The materials of the structure may be homogeneous or inhomogeneous and isotropic or orthotropic. The development of the FINITE program is based on the GASP program (see Section 4.1.4.1.3).

4.1.4.1.5.2 Program Version and Computer. The present version of the program at GE/NED was obtained from the developer J. E. McConnelee of GE/Gas Turbine Department in 1969 (Reference 4.1-12). The NED version is used on the Honeywell 6000 computer.

4.1.4.1.5.3 History of Use. Since its completion in 1969, the program has been widely used in the Gas Turbine and the Jet Engine Departments of the GE for the analysis of turbine components.

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

4.1.4.1.6 DYSEA

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

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

4.1.4.1.6.2 Program Version and Computer. The DYSEA version now operating on the Honeywell 6000 computer of GE, Nuclear Energy Systems Division, was developed at GE by modifying the SAPIV program. Capability was added to handle the hydrodynamic mass effect due to fluid-structure interaction in the reactor. It can handle 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 all dynamic and seismic analysis since its development. Results from test problems were found to be in close agreement with those obtained from either verified programs or analytic solutions.

4.1.4.1.7 SHELL 5

4.1.4.1.7.1 Program Description. SHELL 5 is a finite shell element program used to analyze smoothly curved thin shell structures with any distribution of elastic material properties, boundary constraints, and mechanical thermal and displacement loading conditions. The basic element is a triangle whose membrane displacement fields are linear polynomial functions, and whose bending displacement field is a cubic polynomial function (Reference 4.1-13). Five degrees of freedom (three displacements and two bending rotations) are obtained at each nodal point. Output displacements and stresses are in a local (tangent) surface coordinate system.

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

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

4.1.4.1.7.3 History of Use. SHELL 5 is a program developed by Gulf General Atomic Incorporated (Reference 4.1-14) in 1969. The program has been in production status at Gulf General Atomic, GE, and at other major computer operating systems since 1970.

4.1.4.1.7.4 Extent of Application. SHELL 5 has been used at GE 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 represents the complex mixing processes obtained in the upper plenum, downcomer, and lower plenum. Mass and energy balances throughout the nuclear steam supply system are modeled in detail (Reference 4.1-15).

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 on as many as 20 structural locations.*

4.1.4.1.9.2 Program Version and Computer. The present version of FAP-71 was completed by L. Young of GE/NED in 1971 (Reference 4.1-16). The program currently is on the NED Honeywell 6000 computer.

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

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

4.1.4.1.10 CREEP/PLAST

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

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

4.1.4.1.10.2 Program Version and Computer. The program can be used for elastic-plastic analysis with or without the presence of creep. It can also be used for creep analysis without the presence of instantaneous plasticity. A detailed description of theory is given in Reference 4.1-17. The program is operative on UNIVAC-1108.

4.1.4.1.10.3 History of Use. This program was developed by Y. R. Rashid (Reference 4.1-17) in 1971. It underwent extensive program testing before it was put on production status.

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

4.1.4.2 Fuel Rod Thermal Analysis

Thermal design analyses of the fuel and core were performed to verify that design criteria are met (see References 4.1-1, 4.1-4, 4.1-20, 4.1-24, 4.1-26 and 4.1-27).

4.1.4.3 Reactor Systems Dynamics

The analysis techniques used in reactor systems dynamics are described in Sections S.1.3 and S.4 of Reference 4.1-1. A complete stability analysis for the reactor coolant system is provided in Section 4.4.4.2.

4.1.4.4 Nuclear Engineering Analysis

The analysis techniques are described in the fuel design reports (see References 4.1-1, 4.1-23 and 4.1-25).

4.1.4.5 Neutron Fluence Calculations

See Section 4.3.2.8.

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, inlet enthalpy, and the 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 fuel type.

4.1.5 REFERENCES

4.1-1 General Electric Standard Application for Reactor Fuel, NEDE-24011-P-A, and Supplement for United States, NEDE-24011-P-A-US (most recent approved version referenced in COLR).

4.1-2 Supplemental Reload Licensing Report for Columbia (most recent version referenced in COLR).

4.1-3 Fuel Bundle Information Report for Columbia (most recent approved version referenced in COLR).

4.1-4 Generic Mechanical Design Criteria for BWR Fuel Designs, ANF-89-98 (P)(A), Revision 1, and Supplement 1, Advanced Nuclear Fuels Corporation, May 1995.

4.1-5 Deleted.

4.1-6 Deleted.

4.1-7 Crowther, R. L., Xenon Considerations in Design of Boiling Water Reactors, APED-5640, June 1968.

4.1-8 Deleted.

4.1-9 Beitch, L., Shell Structures Solved Numerically by Using a Network of Partial Panels, AIAA Journal, Volume 5, No. 3, March 1967.

4.1-10 E. L. Wilson, A Digital Computer Program For the Finite Element Analysis of Solids With Non-Linear Material Properties, Aerojet General Technical, Memo No. 23, Aerojet General, July 1965.

4.1-11 I. Farhoomand and E. L. Wilson, Non-Linear Heat Transfer Analysis of Axisymmetric Solids, SESM Report SESM71-6, University of California at Berkeley, Berkeley, California, 1971.

4.1-12 J. E. McConnelee, Finite-Users Manual, General Electric TIS Report DF 69SL206, March 1969.

4.1-13 R. W. Clough and C. P. Johnson, A Finite Element Approximation For the Analysis of Thin Shells, International Journal Solid Structures, Vol. 4, 1968.

4.1-14 A Computer Program For the Structural Analysis of Arbitrary Three-Dimensional Thin Shells, Report No. GA-9952, Gulf General Atomic.

4.1-15 Burgess, A. B., User Guide and Engineering Description of HEATER Computer Programs, March 1974.

4.1-16 Young, L. J., FAP-71 (Fatigue Analysis Program) Computer Code, GE/NED Design Analysis Unit R. A. Report No. 49, January 1972.

4.1-17 Rashid, Y. R, Theory Report for Creep-Plast Computer Program, GEAP-10546, AEC Research and Development Report, January 1972.

4.1-18 Deleted.

4.1-19 Deleted.

4.1-20 Mechanical and Thermal-Hydraulic Design Report for Columbia Generating Station ATRIUM-10 Fuel Assemblies, EMF-2865(P) Revision 1, Framatome ANP, April 2003.

4.1-21 Deleted.

4.1-22 Deleted.

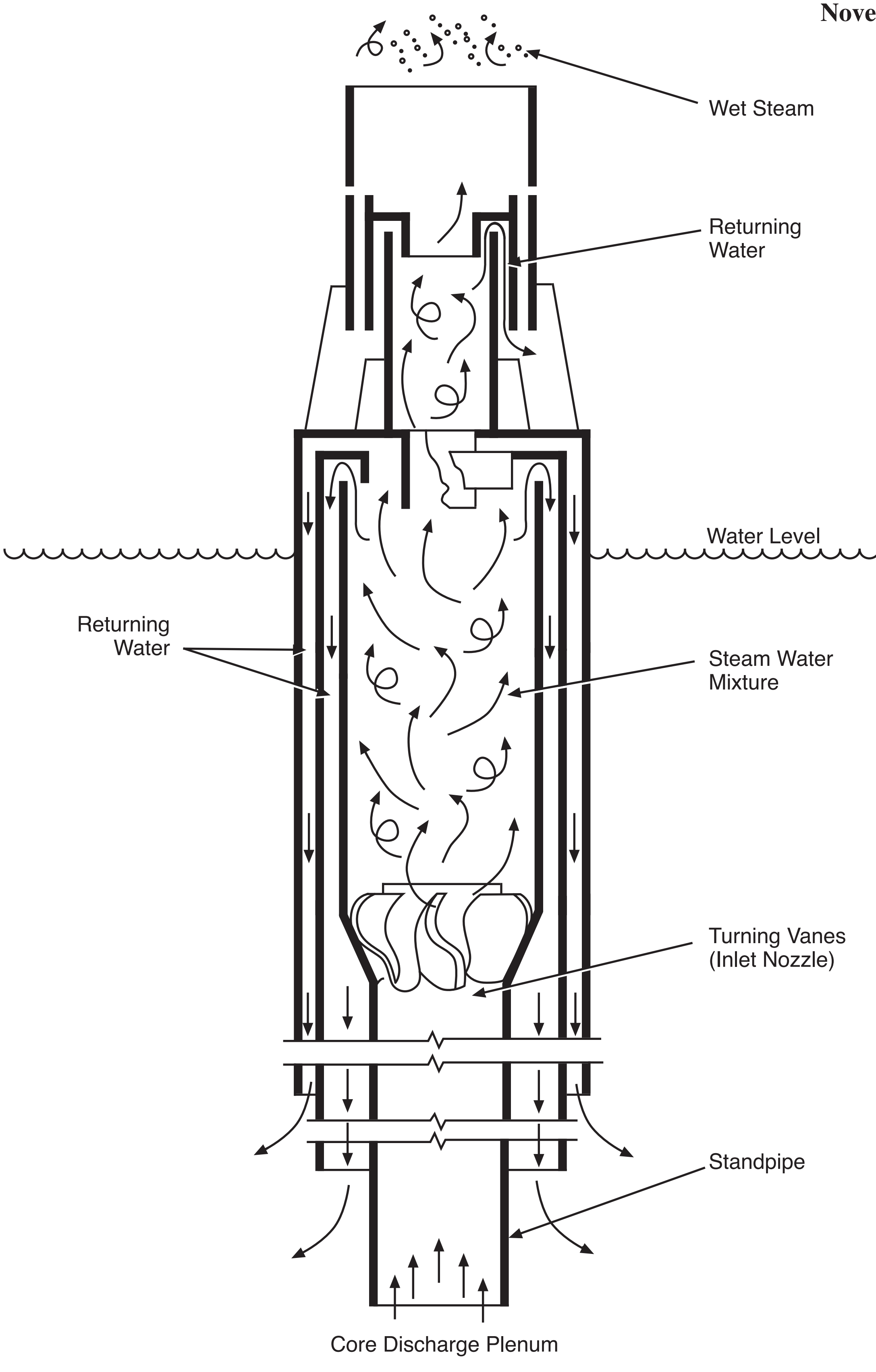
4.1-23 Nuclear Fuel Design Report Columbia Generating Station Fabrication Batch CGS1-18 ATRIUM™-10 Fuel, CGS-FTS-0163, Revision 1, Energy Northwest, October 2004.

4.1-24 Mechanical Design Report for Columbia Generating Station Reload CGS1-18 ATRIUM™-10 Fuel Assemblies, EMF-3152(P), Revision 1, Framatome ANP, Inc., February 2005.

4.1-25 Nuclear Fuel Design Report Columbia Generating Station Fabrication Batch CGS1-19 ATRIUM™-10 Fuel, ANP-2598(P), Revision 0, AREVA NP, January 2007.

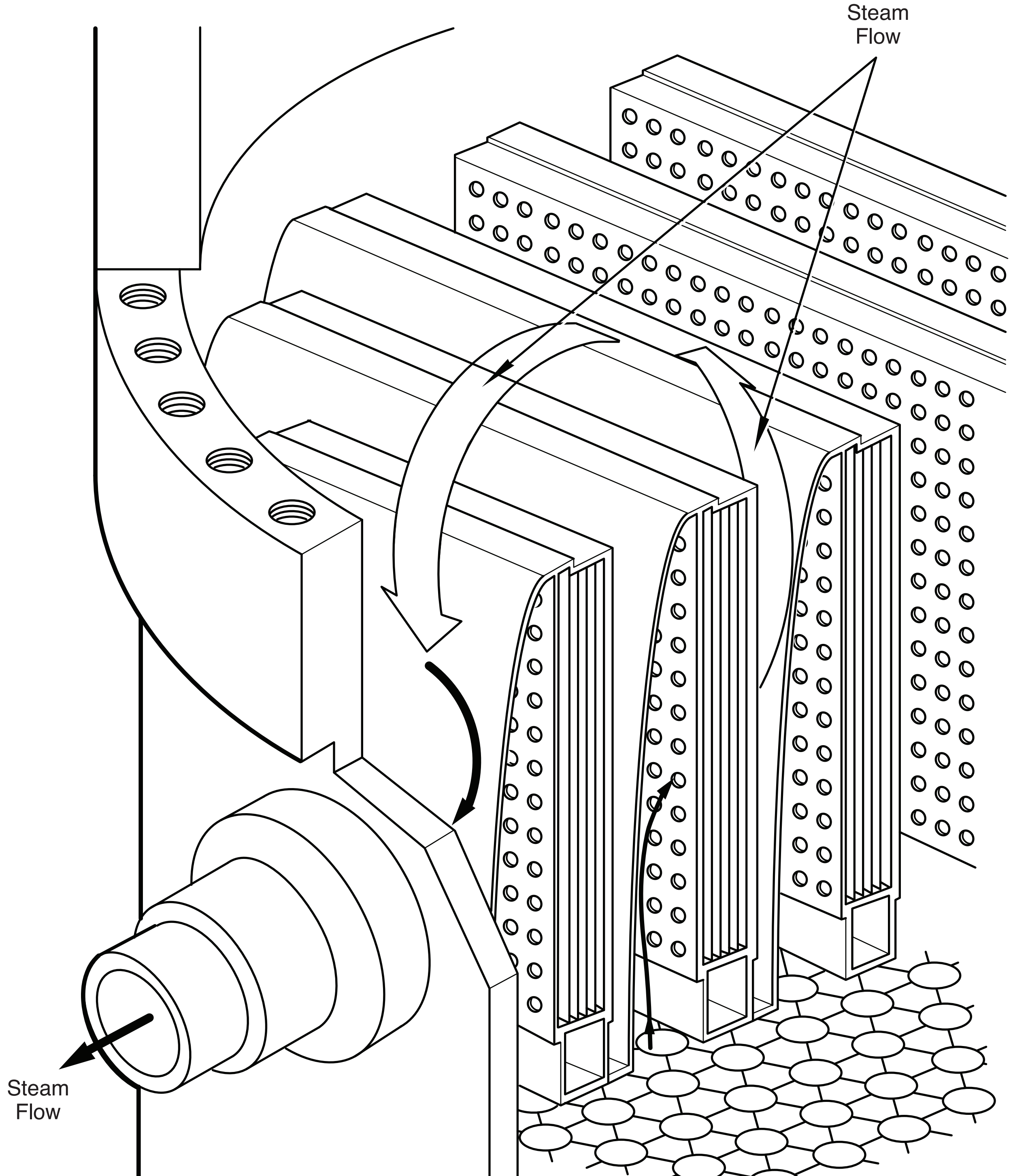
4.1-26 Mechanical Design Report for Columbia Generating Station Reload CGS1-19 ATRIUM™-10 Fuel Assemblies, ANP-2595(P), Revision 0, AREVA NP, January 2007.

4.1-27 GE14 Compliance With Amendment 22 of NEDE-24011-P-A (GESTAR II), NEDC-32868P, Revision 2, September 2007.



Steam Separator

Draw. No. 960690.95 | Rev. | Figure 4.1-1



Columbia Generating Station
Final Safety Analysis Report

Steam Dryer Panel

Draw. No. 960690.98 | Rev. | Figure 4.1-2

DELETED

Columbia Generating Station
Final Safety Analysis Report

Draw. No. 910402.33 | Rev. | Figure 4.1-3

4.2 FUEL SYSTEM DESIGN

See Appendix A, Section A.4.2 of Reference 4.2-1 and Reference 4.2-2.

4.2.1 DESIGN BASES

General Electric BWR fuel assembly and channel design bases, analytical methods, and evaluation results are described in Reference 4.2-1 (Appendix A, subsection A.4.2.1), Reference 4.2-6 and Reference 4.2-25. The AREVA NP fuel assembly and channel design bases, analytical methods, and evaluation results are described in Reference 4.2-3, Reference 4.2-4, Reference 4.2-21, Reference 4.2-23, and Reference 4.2-24.

4.2.1.1 Fuel System Damage Limits

4.2.1.1.1 Stress/Strain Limits

See subsection 2.2.1.1.2 of Reference 4.2-1 and Reference 4.2-4.

4.2.1.1.2 Fatigue Limits

See subsections 2.2.1.2.2 of Reference 4.2-1 and Reference 4.2-4.

4.2.1.1.3 Fretting Wear Limits

Fretting wear is considered in the mechanical design analysis of the assembly. See subsection 2.2.1.3.2 of Reference 4.2-1 and Reference 4.2-4.

4.2.1.1.4 Oxidation, Hydriding, and Corrosion Limits

See subsection 2.2.1.4.2.2 of Reference 4.2-1 for the hydriding limit. Oxidation and corrosion are considered in the mechanical design analysis. See subsection 2.2.1.4.1.2 of Reference 4.2-1 and Reference 4.2-4.

4.2.1.1.5 Dimensional Change Limits

See Reference 4.2-6, subsection 2.2.1.5.2 of Reference 4.2-1, and Reference 4.2-4.

4.2.1.1.6 Internal Gas Pressure Limit

See subsection 2.2.1.6.2 of Reference 4.2-1 and Reference 4.2-4.

4.2.1.1.7 Hydraulic Loads Limits

See subsection 2.2.1.7.2 of Reference 4.2-1 and Reference 4.2-4.

4.2.1.1.8 Control Rod Reactivity Limits

See Section 3.2 and 3.3 of Reference 4.2-1 and Reference 4.2-7.

4.2.1.2 Fuel Rod Failure Limits

4.2.1.2.1 Hydriding Limits

See subsection 2.2.2.1 of Reference 4.2-1 and Reference 4.2-4.

4.2.1.2.2 Cladding Collapse Limits

See subsection 2.2.2.2.2 of Reference 4.2-1 and Reference 4.2-4.

4.2.1.2.3 Fretting Wear Limits

See subsection 2.2.1.3.2 of Reference 4.2-1 and Reference 4.2-4.

4.2.1.2.4 Overheating of Cladding Limits

See subsections 2.2.2.4 and 4.3.1 of Reference 4.2-1 and Reference 4.2-4.

4.2.1.2.5 Overheating of Pellet Limits

See subsection 2.2.2.5.2 of Reference 4.2-1 and Reference 4.2-4.

4.2.1.2.6 Excessive Fuel Enthalpy Limits

See Reference 4.2-4 and subsection 2.2.2.6 of Reference 4.2-1.

4.2.1.2.7 Pellet-Cladding Interaction Limits

See subsection 2.2.2.7.2 of Reference 4.2-1 and Reference 4.2-4.

4.2.1.2.8 Bursting Limits

See subsections 2.2.2.8 and 2.2.3.4 of Reference 4.2-1 and Reference 4.2-4.

4.2.1.2.9 Mechanical Fracturing Limits

See subsection 2.2.2.9.2 of Reference 4.2-1 and Reference 4.2-4.

4.2.1.3 Fuel Coolability Limits

4.2.1.3.1 Cladding Embrittlement Limits

See Reference 4.2-10, subsection 2.2.3.1 of Reference 4.2-1, and Reference 4.2-4.

4.2.1.3.2 Violent Expulsion of Fuel Limits

See subsection 2.2.3.2 of Reference 4.2-1 and Reference 4.2-4.

4.2.1.3.3 Generalized Cladding Melt Limits

Same as Section 4.2.1.3.1 and subsection 2.2.3.3 of Reference 4.2-1.

4.2.1.3.4 Fuel Rod Ballooning Limits

Same as Section 4.2.1.2.8.

4.2.1.3.5 Structural Deformation Limits

See subsection 2.2.3.5 of Reference 4.2-1 and Reference 4.2-4.

4.2.2 DESCRIPTION AND DESIGN DRAWINGS

See Reference 4.2-26, Reference 4.2-2, Reference 4.2-3, Reference 4.2-21, and Reference 4.2-23.

4.2.2.1 Control Rods

The control rods (typical configuration shown in Figures 4.2-1.1, 4.2-1.2, and 4.2-1.3) perform the dual function of power shaping and reactivity control. Power distribution in the core is controlled during operation of the reactor by manipulating selected patterns of control rods. Control rod withdrawal tends to counterbalance steam void effects at the top of the core and results in significant axial power flattening.

The original control rods consists of a sheathed cruciform array of stainless steel tubes filled with boron-carbide (B_4C) powder. The control rods are 9.88 in. in total span and are separated uniformly throughout the core on a 12-in. pitch maximum. Each control rod is surrounded by four fuel assemblies.

The main structural member of a control rod is made of type 304 stainless steel and consists of a top handle, an original bottom casting with a velocity limiter and control rod drive coupling, a vertical cruciform center post, and four U-shaped absorber tube sheaths. The top handle, bottom casting, and center post are welded into a single skeletal structure. The U-shaped sheaths are resistance welded to the center post, handle, and castings to form a rigid housing to contain the boron-carbide-filled absorber rods. Rollers at the top and bottom of the control rod guide the control rod as it is inserted and withdrawn from the core. The control rods are cooled by the core bypass flow. The U-shaped sheaths are perforated to allow the coolant to circulate freely about the absorber tubes. 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 tubes is compacted to about 70% of its theoretical density. The boron-carbide contains a minimum of 76.5% by weight natural boron. The boron-10 (B-10) minimum content of the boron is 18% by weight. Absorber tubes are made of type 304 stainless steel. Each absorber tube is 0.188-in. O.D. and has a 0.025-in. wall thickness. Absorber tubes are sealed by a plug welded into each end. The boron-carbide is longitudinally separated into individual compartments by stainless steel balls at approximately 16-in. intervals. The steel balls are held in place by a slight crimp of the tube. Should boron-carbide tend to sinter in service, the steel balls will keep the resulting void spaces distributed over the length of the absorber tube.

Some of the control rods have been replaced with Duralife 215 or Marathon control rods. The main structural member of the Duralife 215 control rod design is made of stainless steel and consists of a top handle, a tie rod, a bottom control rod drive coupling, and four sheaths containing the neutron absorber. The top handle, tie rod, velocity limiter, and sheaths are welded into a single structure. The neutron absorber in each wing of the sheath consists of 18 high-purity stainless-steel tubes filled with boron-carbide, three hafnium rods at the edge of the wing, and a hafnium plate at the top.

The sheaths of the Duralife 215 blades are attached to the structure with full fusion corner welds to the handle, tie rod, and velocity limiter to form a rigid housing. Inconel X750 rollers at the top and bottom of the control rod guide the control rod as it is inserted and withdrawn from the core. These rollers rotate on PH 13-8 Mo pins. The sheaths are perforated and the hafnium absorber plate has coolant grooves to allow the coolant to circulate freely about the absorber and flush the joint between the sheath and handle.

The number of boron carbide absorber rods in each wing has been changed from 19 rods with an I.D. of 0.138 in. to 18 rods with an I.D. of 0.148 in. The outside diameter remains the same. The length of the absorber column in these rods has been reduced from 143 in. to 137 in. to accommodate a top 6-in.-high hafnium plate. In addition, three 0.188 in. O.D. hafnium rods have been added to the edge of each wing.

The Marathon control rod blade consists of a top handle, a segmented tie rod (for weight savings), a bottom control rod coupling/velocity limiter and four wings consisting of an array of square tubes. The square tubes each have four lobes to allow adjacent tubes to be welded to each other. The absorber tubes are welded lengthwise to form the four wings of the control rod. Each wing is comprised of 17 absorber tubes. The four wings are then welded to the tie rod to form the cruciform-shaped member of the control rod.

The square tubes are circular inside and are loaded with either B_4C or hafnium. The combination of absorbers and absorber tubes is based on the needed initial reactivity worth. In addition, empty capsules are used in some absorber tubes to provide a plenum for helium released during B_4C burnup.

A comparison of the original, the Duralife 215 and the Marathon control rod dimensions and materials is given in Table 4.2-1.

4.2.2.2 Velocity Limiter

The control rod velocity limiter (see Figure 4.2-2) is an integral part of the bottom assembly of each control rod. This feature protects against a 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 velocity limiter is in the form of two nearly mated conical elements that act as a large clearance piston inside the control rod guide tube. The lower conical element is separated from the upper conical element by four radial spacers 90 degrees apart.

The hydraulic drag forces on a control rod are approximately proportional to 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 is trapped by the lower conical element and discharged through the annulus between the two conical sections. Because this water is forced in a partially reversed direction into water flowing upward in the annulus, a severe turbulence is created, and this slows the descent of the control rod assembly to less than 5 ft/sec.

4.2.3 DESIGN EVALUATION

See Appendix A, subsection A.4.2.3 of Reference 4.2-1 and Reference 4.2-2.

4.2.3.1 Fuel System Damage Evaluation

4.2.3.1.1 Stress/Strain Evaluation

See Reference 4.2-3, Reference 4.2-21 and Reference 4.2-23. Fuel rod internal pressure has been shown to remain below system pressure for rod peak burnups well beyond anticipated achieved burnup. For GNF fuel see section 2.2.1.1.3 of Reference 4.2-1.

4.2.3.1.2 Fatigue Evaluation

See subsection 2.2.1.2.3 of Reference 4.2-1, Reference 4.2-3, Reference 4.2-21, and Reference 4.2-23.

4.2.3.1.3 Fretting Wear Evaluation

See Reference 4.2-12, subsection 2.2.1.3.3 of Reference 4.2-1, Reference 4.2-3, Reference 4.2-21, and Reference 4.2-23.

4.2.3.1.4 Oxidation, Hydriding, and Corrosion Evaluation

See subsections 2.2.1.4.1.3 and 2.2.1.4.2.3 of Reference 4.2-1, Reference 4.2-3, Reference 4.2-21, and Reference 4.2-23.

4.2.3.1.5 Dimensional Change Evaluation

See Reference 4.2-6, subsection 2.2.1.5.3 of Reference 4.2-1, Reference 4.2-3, Reference 4.2-5, Reference 4.2-21, and Reference 4.2-23.

4.2.3.1.6 Internal Gas Pressure Evaluation

See subsection 2.2.6 of Reference 4.2-26, Reference 4.2-3, Reference 4.2-21, and Reference 4.2-23. 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. The analysis results show that the calculated cladding stresses are below allowable limits even with internal gas pressure and other loads at end of life normal and transient conditions, respectively.

4.2.3.1.7 Hydraulic Load Evaluation

See subsection 2.2.1.7.3 of Reference 4.2-1, Reference 4.2-2, Reference 4.2-3, Reference 4.2-12, Reference 4.2-21, Reference 4.2-23, and Section 3.9.

4.2.3.1.8 Control Rod Reactivity Evaluation

See Appendix A, subsection A.4.3.2 of Reference 4.2-1. Energy Northwest calculates the fluence of each control blade using an appropriate conversion factor for fuel exposures adjacent to the control blade. Control blade shuffling or replacement is based on the calculated blade fluence as compared to vendor allowed values (Reference 4.2-22). The vendor allowed values account for the reduction in control blade worth due to a combination of boron-10 depletion and boron loss resulting from cracking of the absorber tubes.

4.2.3.2 Fuel Rod Failure

4.2.3.2.1 Hydriding Evaluation

See Section 4.2.3.1.4.

4.2.3.2.2 Cladding Collapse Evaluation

See subsection 2.2.8 of Reference 4.2-26, Reference 4.2-3, Reference 4.2-21, and Reference 4.2-23.

4.2.3.2.3 Fretting Wear Evaluation

See Section 4.2.3.1.3.

4.2.3.2.4 Overheating of Cladding Evaluation

See section 2.6 of Reference 4.2-26, Reference 4.2-3, Reference 4.2-21, and Reference 4.2-23.

4.2.3.2.5 Overheating of Pellet Limits

See subsection 2.2.9 of Reference 4.2-26, Reference 4.2-3, Reference 4.2-21, and Reference 4.2-23.

4.2.3.2.6 Excessive Fuel Enthalpy Evaluation

See Section 15.4.9, Reference 4.2-2, and subsection 2.12 of Reference 4.2-26.

4.2.3.2.7 Pellet-Cladding Interaction Evaluation

Calculated results do not exceed the 1% plastic strain or minimum critical power ratio (MCPR) fuel cladding integrity safety limits; thus fuel pellet melting does not occur. These are the most applicable general design criteria for pellet cladding interaction (PCI) phenomena. While PCI-induced fuel failures remain a commercially undesirable problem, they are not a safety concern. Boiling water reactors (BWRs) have been designed and licensed with provisions to accommodate operating with fuel cladding perforation, and field experience confirms that plants do indeed operate within radiological release limits.

Operation below the thermal-mechanical limits historically has resulted in very few pellet-cladding interaction (PCI) fuel failures. Furthermore power maneuvering guidelines have been developed that have further reduced fuel failures due to the PCI mechanism.

4.2.3.2.8 Bursting Evaluation

An adaptation of the NUREG-0630 perforation strain versus temperature curve used for the AREVA NP LOCA analysis is described in Reference 4.2-17. See subsection 2.11.1 of Reference 4.2-26.

4.2.3.2.9 Mechanical Fracturing Evaluation

See Reference 4.2-12, subsection 2.2.2.9.3 of Reference 4.2-1, Reference 4.2-3, Reference 4.2-21, and Reference 4.2-23.

4.2.3.3 Fuel Coolability Evaluation

4.2.3.3.1 Cladding Embrittlement Evaluation

See Section 4.2.3.2.8 and subsection 2.2.3.1 of Reference 4.2-1.

4.2.3.3.2 Violent Expulsion of Fuel Evaluation

See Section 15.4.9 and subsection 2.12 of Reference 4.2-26.

4.2.3.3.3 Generalized Cladding Melt Evaluation

See Section 4.2.3.2.8 and subsection 2.11.2 of Reference 4.2-26.

4.2.3.3.4 Fuel Rod Ballooning Evaluation

See Section 4.2.3.2.8 and subsection 2.11.1 of Reference 4.2-26.

4.2.3.3.5 Structural Deformation Evaluation

See Section 4.2.3.2.9.

4.2.4 TESTING, INSPECTION, AND SURVEILLANCE PLANS

4.2.4.1 Fuel Testing, Inspection, and Surveillance

See Appendix A, subsection A.4.2.4 of Reference 4.2-1.

4.2.4.2 Online Fuel System Monitoring

Columbia Generating Station (CGS) has two independent radiation detection systems that are directly capable of detecting fission product releases from failed fuel rods in an online manner. The main steam line radiation (MSLR) monitors are described in Section 11.5.2.1.1. Because the MSLR monitors are located relatively close to the reactor core, they are capable of sensing gross fission product releases in a few seconds.

The offgas system radiation (OGSR) monitors are capable of detecting low-level emissions of noble gases in 2 to 3 minutes after the gases leave the fuel. The OGSR monitors are described in more detail in Section 11.5.2.2.1.

4.2.4.3 Post-Irradiation Surveillance

The following fuel surveillance will be conducted after the refueling outage for the CGS unit on fuel discharged during the refueling outage that has given indication of gross cladding defects or anomalies during plant operation.

Scope

The fuel surveillance program, developed to provide verification of the reliable performance of the CGS fuel design, will consist of the following inspections and measurements:

a. Visual inspection of the peripheral rods will be performed on discharged fuel, that has given indication of gross cladding defects or anomalies during plant operation, after each refueling outage. The examination will be capable of detecting and characterizing generic gross cladding defects or anomalies; and

b. If anomalous behavior of the fuel cladding, components of the fuel assembly, or significant rod bow are detected by visual examination, further investigation, and measurements of such significant anomalies will be conducted after the refueling outages.

Implementation

a. Onsite receiving inspection of all the initial core fuel assemblies and subsequent reloads will be documented. Any significant anomalies detected will be documented;

b. Fuel performance history and related plant operational data will be monitored and analyzed during operation;

c. Assemblies discharged during each refueling outage that have given indication of gross cladding defects or anomalies during plant operation will be selected for visual inspection. The visual examination of the peripheral rods will include observations for cladding defects, fretting, rod bowing, missing components, corrosion, crud deposition, and geometric distortions. The defects or anomalies on the cladding surface area examined will be either videotaped or photographed to document and characterize the anomaly;

d. In the event that significant anomalies are observed during the refueling examination, all other discharged assemblies may also be visually inspected. The results will be analyzed to determine fuel utilization strategy and possible safety implications in accordance with the operating procedures and applicable licensing requirements;

e. If unusual defects are observed, the fuel with the defects and the applicable operational data will be investigated and further appropriate tests and examination of the defected fuel will be performed; and

f. If defects of an unusual nature are detected, an oral report will be made to the NRC after the completion of the inspection activities. Under normal conditions, the report will contain visual examination summaries confirming the reliable performance of the fuel assemblies. In the event that significant anomalies or unusual defects are observed, the report will contain the description and related data of onsite receiving inspection and operational conditions. Evaluation and studies to identify causes for any encountered anomalies or defects will be assessed and the results will be reported to the NRC as they become available.

4.2.4.4 Channel Management Program

Fuel channels are subject to bulge, bow, and elongation when irradiated in reactors. Excessive deformations (bow and bulge) could produce traversing in-core probe asymmetries and control blade frictional resistance.

All new reload fuel will be loaded with new channels. Energy Northwest has in the past reinserted requalified channels in CGS but has transitioned away from channel reuse (References 4.2-13, 4.2-14, and 4.2-15).

The Safety Evaluation Report Related to the Operation of WPPSS Nuclear Project No. 2, Supplement 3 (Reference 4.2-16), discusses measurement of selected discharged fuel channels for deflection. The intent of this deflection measurement was to qualify channels for reuse. Because Energy Northwest no longer reuses channels, the qualification of channel reuse has been discontinued.

In addition to the above channel management program, Energy Northwest is taking a number of operational actions to monitor channel distortion in the core. These include Technical Specifications requirements for periodic scram testing and rod notch testing, which would provide an indication of pending driveline friction between control rod and bowed channels. Should either of these tests suggest a driveline friction problem, the tests described in NEDE-21354-P, Reference 4.2-6, and Safety Communication SC08-05, Reference 4.2-27, would then be used to isolate the cause.

4.2.5 REFERENCES

4.2-1 General Electric Company, General Electric Standard Application for Reactor Fuel (NEDE-24011-P-A), and Supplement for United States (NEDE-24011-P-A-US) (most recent approved version referenced in COLR).

4.2-2 AREVA NP, Columbia Generating Station Cycle 19 Reload Analysis, ANP-2602, Revision 0, March 2007.

4.2-3 Framatome ANP, Mechanical and Thermal-Hydraulic Design Report for Columbia Generating Station ATRIUM-10 Fuel Assemblies, EMF-2865(P), Revision 1, April 2003.

4.2-4 Advanced Nuclear Fuels Corporation, Generic Mechanical Design Criteria for BWR Fuel Designs, ANF-89-98(P)(A), Revision 1 and Supplement 1, May 1995.

4.2-5 Deleted.

4.2-6 General Electric Company, BWR Fuel Channel Mechanical Design and Deflection, NEDE-21354-P, September 1976.

4.2-7 General Electric Company, Control Blade Lifetime with Potential B_4C Loss, NEDO-24226 and Supplement 1.

4.2-8 Deleted.

4.2-9 Deleted.

4.2-10 General Electric Company, Analytical Model for Loss-of-Coolant Analyses in Accordance with 10 CFR 50 Appendix K, NEDO-20566P, January 1976.

4.2-11 Deleted.

4.2-12 General Electric Company, GE Duralife 215 Control Rod Safety Evaluation, GENE-778-028-0790, Revision 2, July 1992.

4.2-13 Letter and Attachment from G. C. Sorensen, Manager, Regulatory Programs, Supply System to NRC, Subject: Nuclear Plant No. 2, Operating License NPF-21, Modification to WNP-2 Cycle Reload Submittal and Response to NRC Bulletin 90-02: Loss of Thermal Margin Caused by Channel Box Bow, GO2-90-075, April 13, 1990.

4.2-14 Letter from G. C. Sorensen, Manager, Regulatory Programs, Supply System to NRC, Subject: Nuclear Plant No. 2, Operating License NPF-21, Final Response to NRC Bulletin 90-02; Loss of Thermal Margins Caused by Channel Box Bow, GO2-90-162, September 28, 1990.

4.2-15 Letter and Attachments from P. L. Eng., Project Manager, NRC to G. C. Sorensen, Manager, Regulatory Programs, Supply System, Evaluation of Response to NRC Bulletin 90-92; Loss of Thermal Margins Caused by Channel Box Bow (TAC No. 76354), April 22, 1991.

4.2-16 Nuclear Regulatory Commission, Safety Evaluation Report Related to the Operation of WPPSS Nuclear Project No. 2, NUREG-0892, Supplement 3, Washington, D.C., May 1983.

4.2-17 Exxon Nuclear Company, Exxon Nuclear Company ECCS Cladding Swelling and Rupture Model, XN-NF-82-07(P)(A), Revision 1, November 1982.

4.2-18 Deleted.

4.2-19 Deleted.

4.2-20 General Electric Company, GE Marathon Control Rod Assembly (NEDE-31758P-A), October 1991.

4.2-21 Framatome ANP, Mechanical Design Report for Columbia Generating Station Reload CGS1-18 ATRIUM™-10 Fuel Assemblies, EMF-3152(P), Revision 1, February 2005.

4.2-22 General Electric Company, GE BWR Control Rod Lifetime, NEDE-30931 (most recent revision specified in CVI 768-00,91).

4.2-23 AREVA NP, Mechanical Design Report for Columbia Generating Station Reload CGS1-19 ATRIUM™-10 Fuel Assemblies, ANP-2595(P), Revision 0, January 2007.

4.2-24 Framatome ANP, Inc., Mechanical Design for BWR Fuel Channels, Revision 1, EMF-93-177(P)(A), August 2005.

4.2-25 General Electric Company, Fuel Assembly Evaluation of Combined Safe Shutdown Earthquake (SSE) and Loss-of-Coolant Accident (LOCA) Loadings, NEDE-21175-3-P, July 1982.

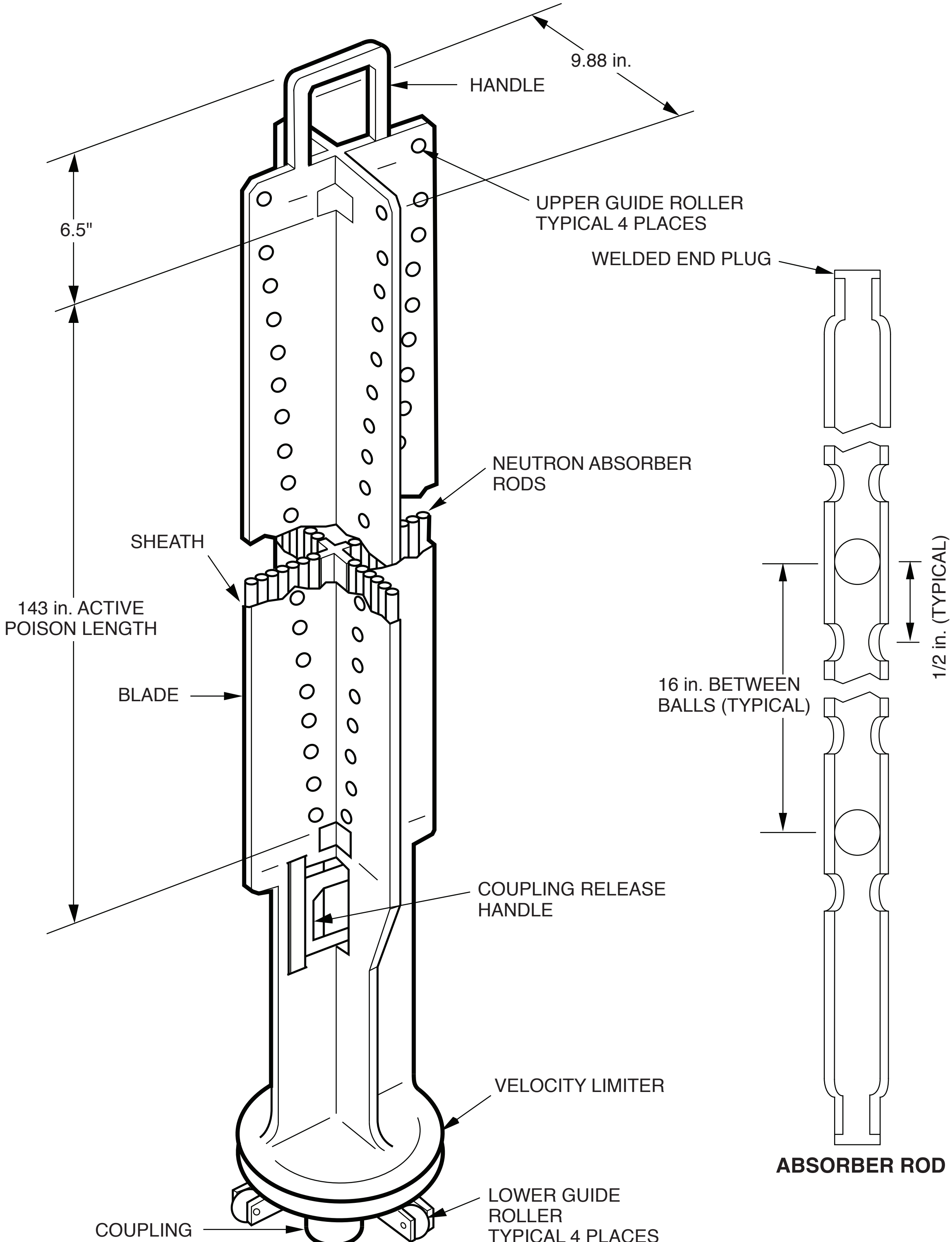
4.2-26 General Electric Company, GE14 Compliance With Amendment 22 of NEDE-24011-P-A (GESTAR II), NEDC-32868P, Revision 2, September 2007.

4.2-27 GE Hitachi Nuclear Energy, Updated Surveillance Program for Channel-Control Blade Interference Monitoring, Safety Communication (SC) 08-05, Revision 1, December 17, 2008.

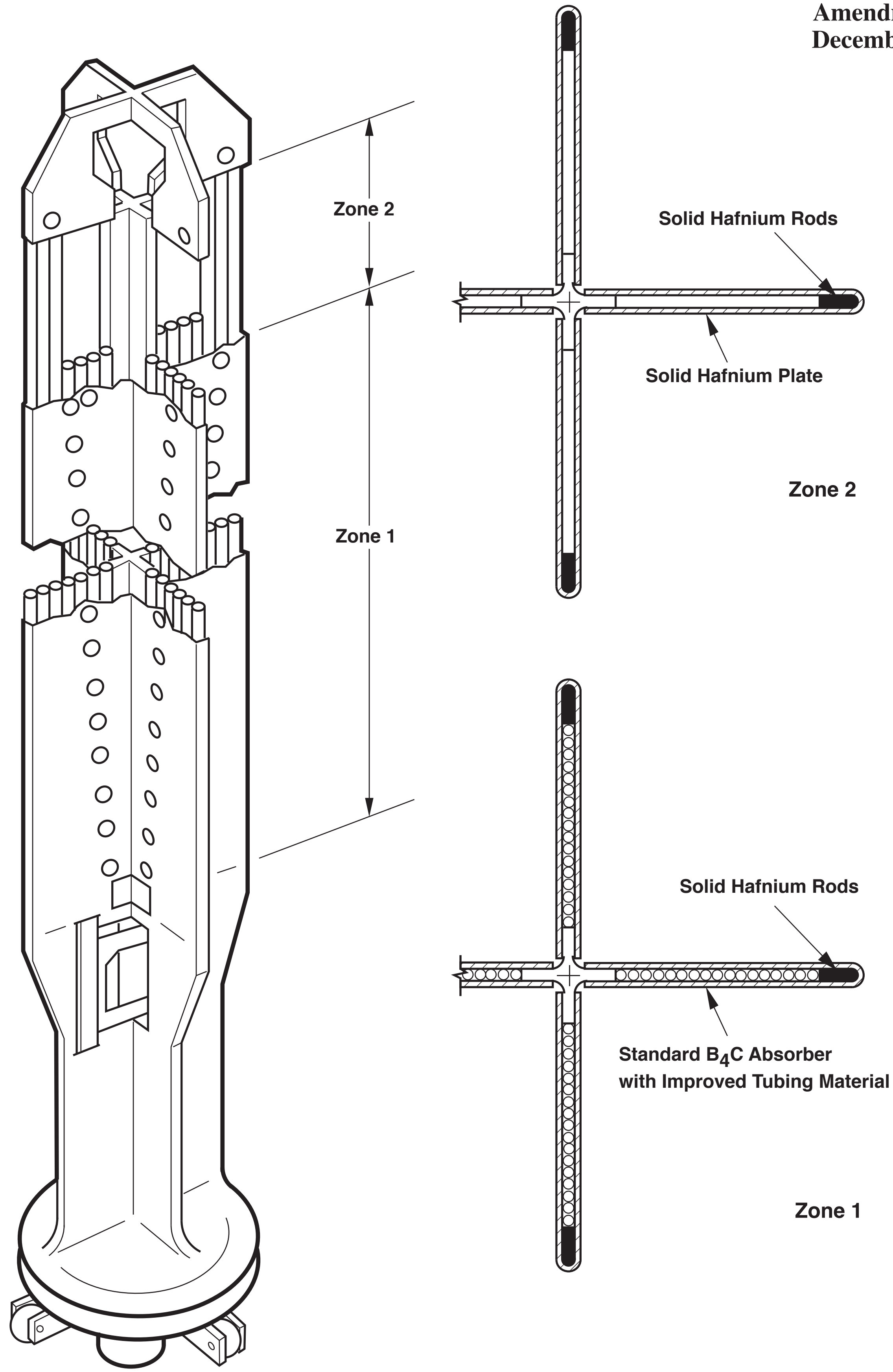
Table 4.2-1

Control Rod Parameters

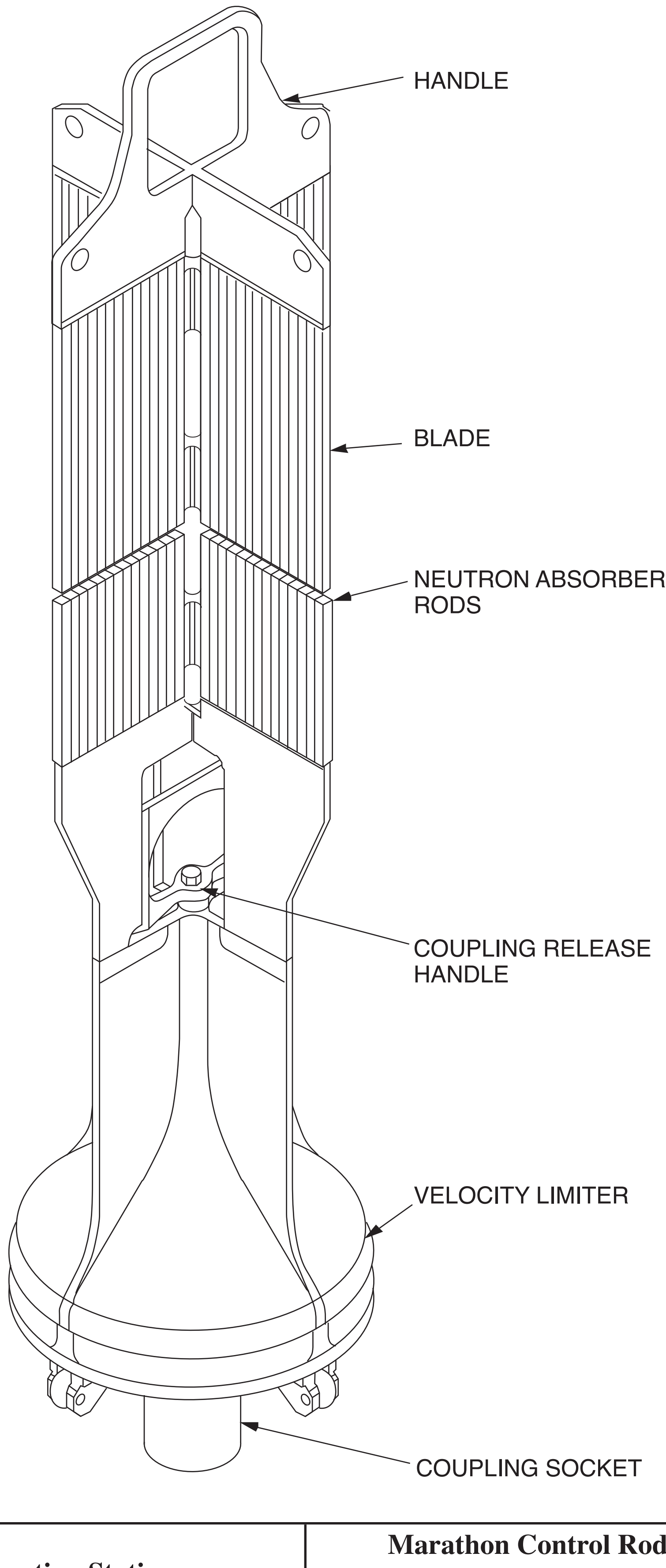
	Original Equipment	Duralife 215	Marathon
Control rod weight, lb (kg)	186 (84.4)	204 (92.5)	197 (89.4)
Absorber rod - boron-carbide			
Number per control rod	76	72	52
Length, in. (mm)	143 (3632)	137 (3480)	143.7 (3650)
Inside diameter, in. (mm)	0.138 (3.51)	0.148 (3.76)	0.189 (4.80)
Density, grams/cm^3	1.76 (Nominal)	1.76 (Nominal)	1.76 (70% Theoretical)
Absorber tube			
Cladding material	Stainless steel	High purity stainless steel	304S
O.D., in. (mm)	0.188 (4.78)	0.188 (4.78)	0.246 (6.248)
Wall thickness, in. (mm)	0.025 (0.635)	0.020 (0.508)	0.021 (0.533)
Absorber rods - hafnium			
Number per control rod	-	12	16
Length, in. (mm)	-	143 (3632)	143.4 (3642)
Diameter, in. (mm)	-	0.188 (4.78)	0.188 (4.78)
Density, grams/cm^3		13.1	13.0
Absorber plate - hafnium			
Number per control rod	-	4	-
Length, in. (mm)	-	6 (152)	-
Width, in. (mm)	-	3.42 (86.87)	-
Thickness, in. (mm)	-	0.188 (4.78)	-
Density, grams/cm^3	-	13.1	-
Sheath thickness, in. (mm)	0.030 (0.762)	0.034 (0.864)	-
Stiffener	Yes	No	-
Pin material	Haynes Alloy 25	PH 13-8 MO	PH 13-8 MO
Roller material	Stellite 3	Inconel X750	Inconel X750



Columbia Generating Station Final Safety Analysis Report	Original Equipment (OEM) Control Rod Blade Assembly
	Draw. No. 960690.96 / Rev. / Figure 4.2-1.1



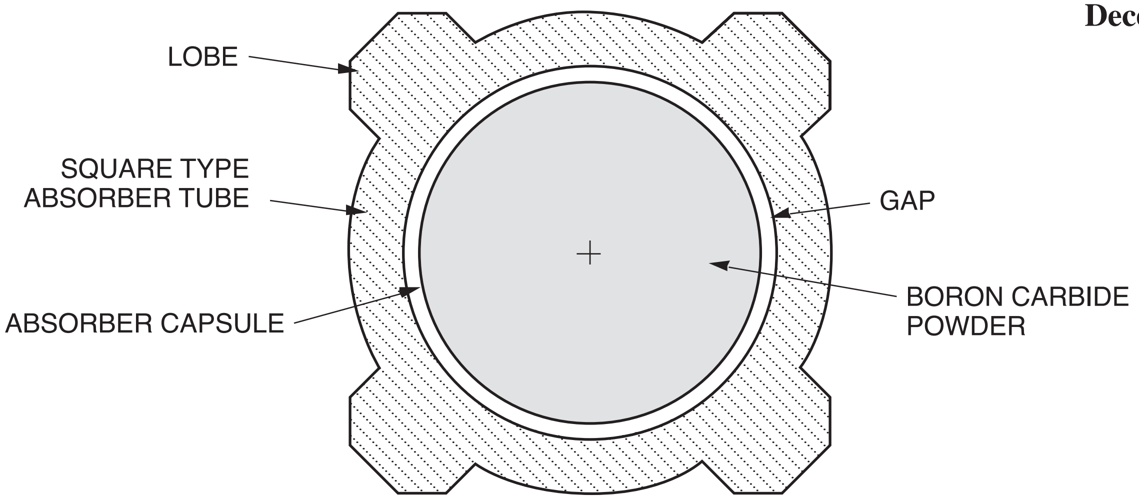
Columbia Generating Station Final Safety Analysis Report	DuraLife 215 Control Rod Blade Assembly
	Draw. No. 010126.54 | Rev. | Figure 4.2-1.2



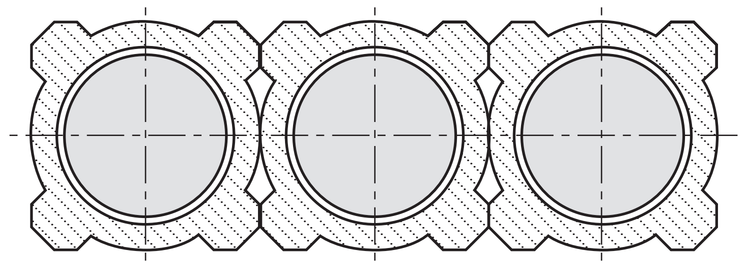
Columbia Generating Station
Final Safety Analysis Report

Marathon Control Rod Blade Assembly

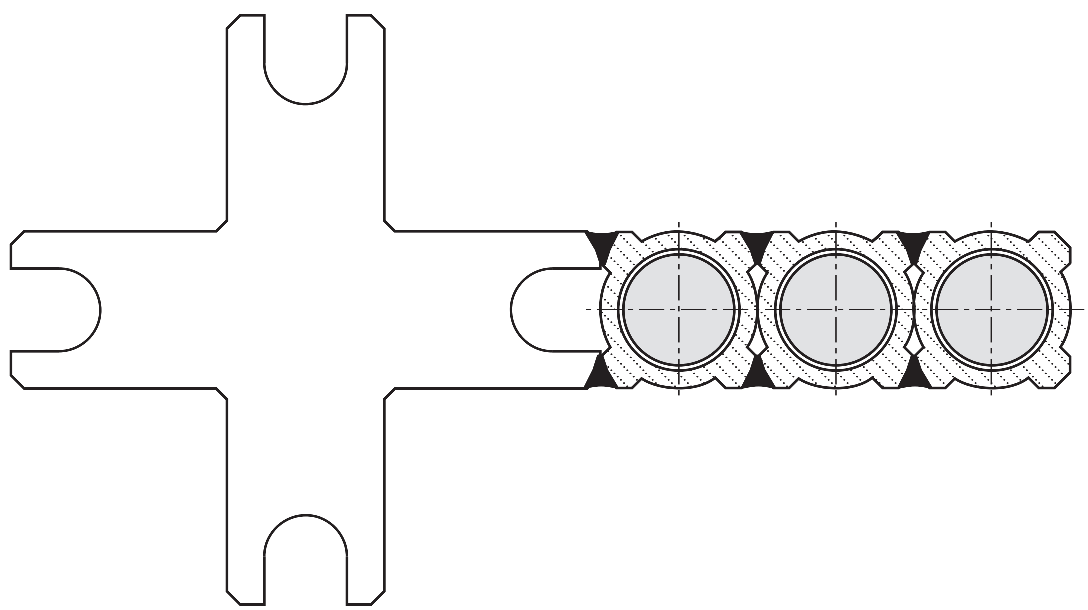
Draw. No. 010126.55 | Rev. | Figure 4.2-1.3



B_4C Placement in Capsules and Absorber Tubes



Absorber Tubes
(Before Welding)



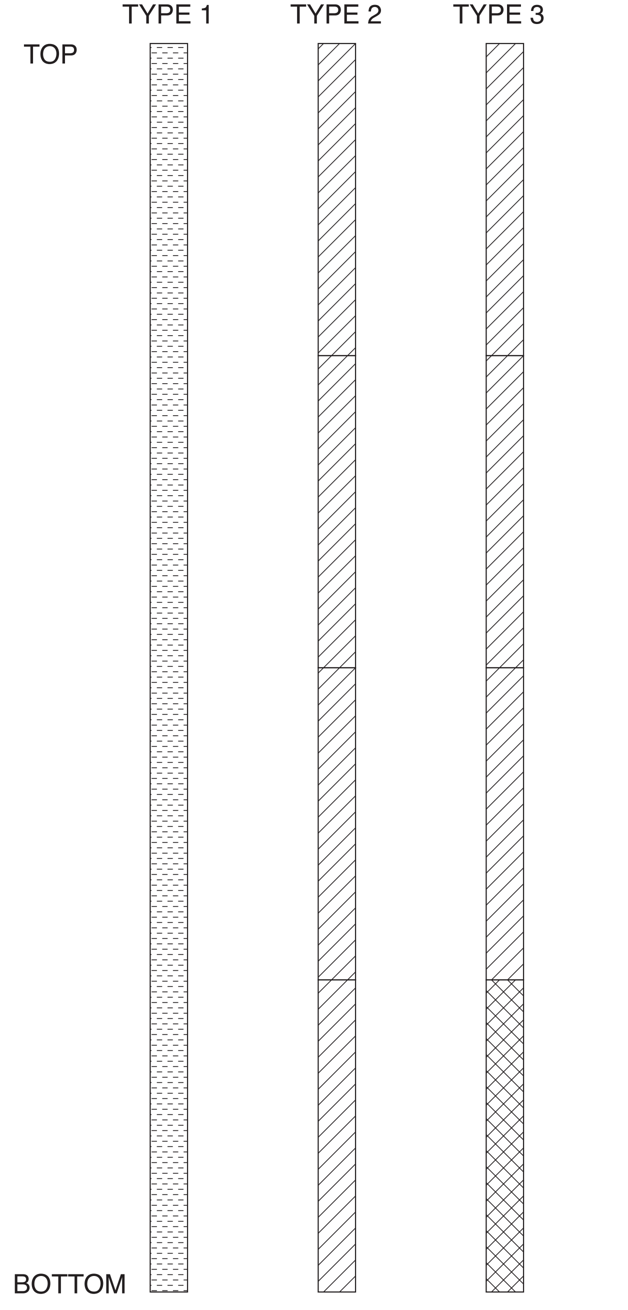
Absorber Tubes Welded to Tie Rods

TYPE	ABSORBER LOADING
1	ONE 143.1” LONG HAFNIUM ROD
2	FOUR 35.77” LONG B4C CAPSULES
3	THREE 35.77” LONG B4C CAPSULES ONE 23.85” LONG B4C CAPSULE ONE 11.925” LONG EMPTY CAPSULE

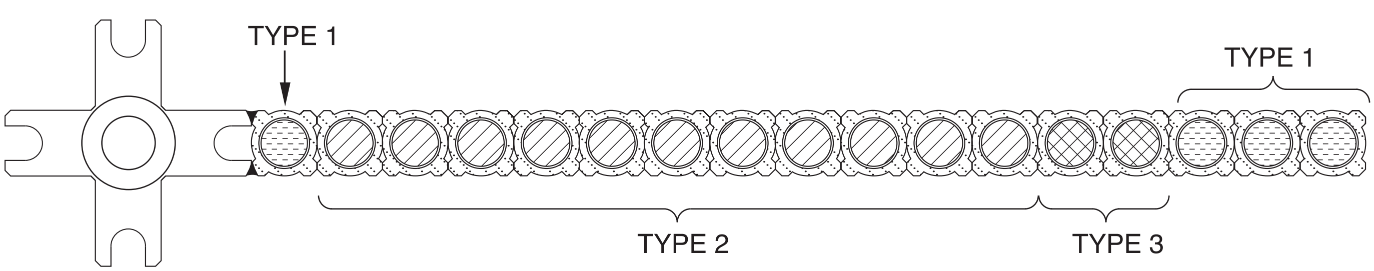
HAFNIUM

B4C ABSORBER CAPSULE

EMPTY CAPSULE



ABSORBER MATERIAL CONTAINED IN TUBES

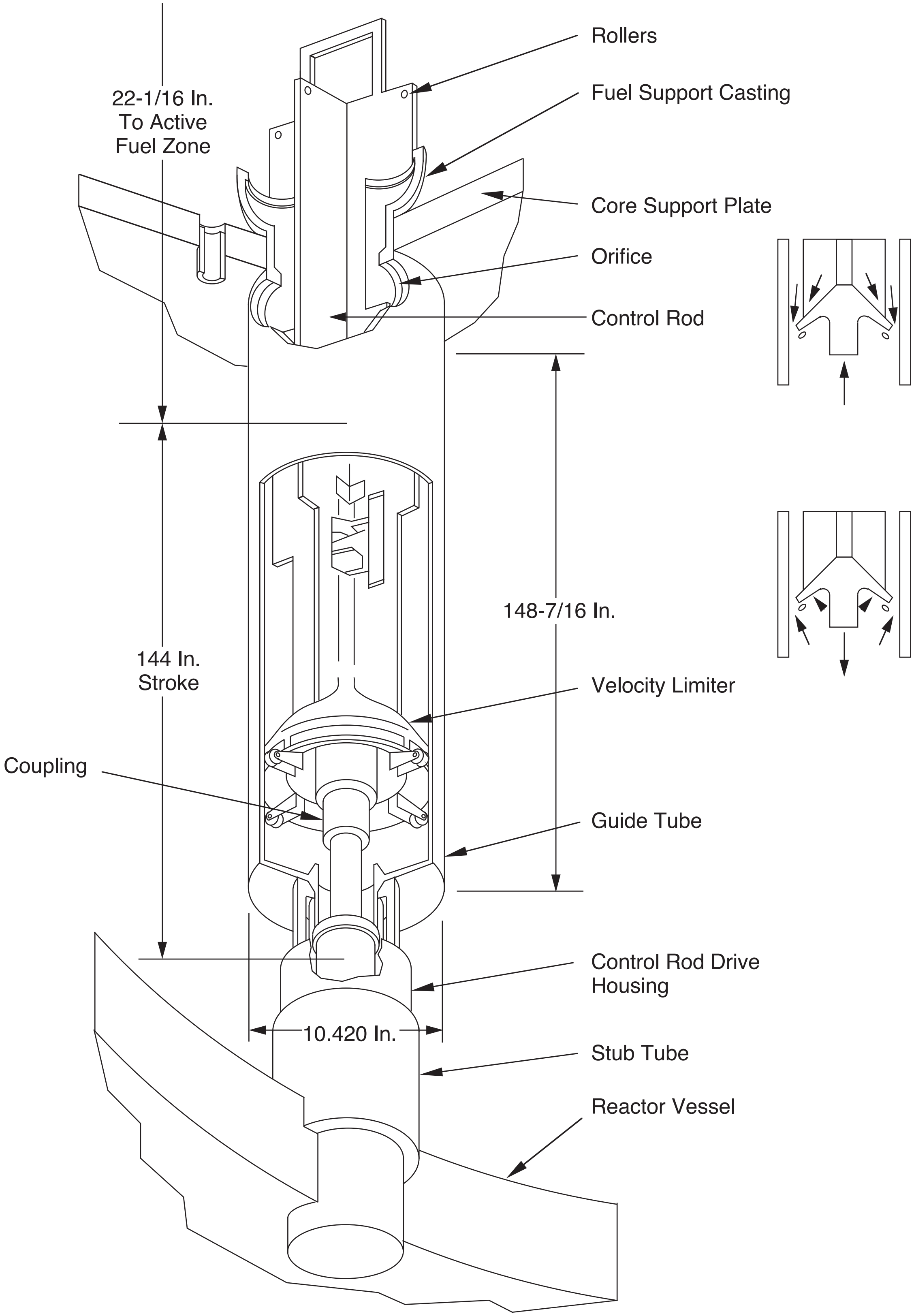


CONTROL ROD CROSS PARTIAL SECTION

Columbia Generating Station
Final Safety Analysis Report

Marathon Control Rod Blade Absorber Placement

Draw. No. **010126.57** | Rev. | Figure **4.2-1.5**



Control Rod Velocity Limiter

Draw. No. 960690.97 | Rev. | Figure 4.2-2

4.3 NUCLEAR DESIGN

4.3.1 DESIGN BASES

See Reference 4.3-2 and Appendix A, subsection A.4.3.1 of Reference 4.3-4.

4.3.1.1 Reactivity Basis

See Appendix A, subsection A.4.3.1.1 of Reference 4.3-4.

4.3.1.2 Overpower Bases

See Appendix A, subsection A.4.3.1.2 of Reference 4.3-4.

4.3.2 DESCRIPTION

See Appendix A, subsection A.4.3.2 of Reference 4.3-4.

4.3.2.1 Nuclear Design Description

See Appendix A, subsection A.4.3.2.1 of Reference 4.3-4. The reference core loading pattern is provided in Reference 4.3-7. See Table 4.3-2, Table 4.3-3 and Reference 4.3-19.

4.3.2.2 Power Distribution

See Appendix A, subsection A.4.3.2.2 of Reference 4.3-4.

4.3.2.2.1 Power Distribution Calculations

See References 4.3-7 and 4.3-17.

4.3.2.2.2 Power Distribution Measurements

See Appendix A, subsection A.4.3.2.2.2 of Reference 4.3-4.

4.3.2.2.3 Power Distribution Accuracy

See Appendix A, subsection A.4.3.2.2.3 of Reference 4.3-4.

4.3.2.2.4 Power Distribution Anomalies

See Appendix A, subsection A.4.3.2.2.4 of Reference 4.3-4.

4.3.2.3 Reactivity Coefficients

See Appendix A, subsection A.4.3.2.3 of Reference 4.3-4.

4.3.2.4 Control Requirements

See Appendix A, subsection A.4.3.2.4 of References 4.3-4.

4.3.2.4.1 Shutdown Reactivity

See Appendix A, subsection A.4.3.2.4.1 of Reference 4.3-4.

The cold shutdown margin for the reference core loading pattern is provided in Reference 4.3-7.

As discussed in Section 4.6.3.1.1.5, the shutdown margin with the highest worth control rod withdrawn shall be analytically determined to be at least 0.38% Δk/k or shall be determined by test to be at least 0.28% Δk/k. To ensure that the safety design basis for shutdown margin is satisfied, additional design margin is adopted during design development so that a shutdown margin of at least 1.00% Δk/k is calculated with the highest worth control rod fully withdrawn.

4.3.2.4.2 Reactivity Variations

See Appendix A, subsection A.4.3.2.6 of Reference 4.3-4.

The excess reactivity designed into the core is controlled by the control rod system supplemented by gadolinia-urania fuel rods (Reference 4.3-17).

Control rods are used during the cycle partly to compensate for burnup and partly to control the power distribution.

Reactivity balances are not used in describing BWR behavior because of the strong interdependence of the individual constituents of reactivity. Therefore, the design process does not produce components of a reactivity balance at the conditions of interest. Instead, it gives the k_{eff} representing all effects combined. Further, any listing of components of a reactivity balance is quite ambiguous unless the sequence of the changes is clearly defined.

4.3.2.5 Control Rod Patterns and Reactivity Worths

See References 4.3-17 and 4.3-18.

4.3.2.6 Criticality of Reactor During Refueling

See Appendix A, subsection A.4.3.2.6 of Reference 4.3-4.

Compliance with Technical Specification shutdown margin requirements is demonstrated through plant procedures and reactivity analyses performed for reload specific refueling activities.

4.3.2.7 Stability

See Appendix A, subsection A.4.3.2.7 of Reference 4.3-4.

4.3.2.7.1 Xenon Transients

See Appendix A, subsection A.4.3.2.7.1 of Reference 4.3-4.

4.3.2.7.2 Thermal Hydraulic Stability

See Appendix A, subsection A.4.3.2.7.2 of Reference 4.3-4.

4.3.2.8 Vessel Irradiations

The reactor pressure vessel (RPV) irradiation calculation provides a best-estimate prediction of the fluence rather than a conservative prediction as was the case with earlier methods. The methodology for the neutron flux calculation conforms to Licensing Topical Report (LTR) NEDC-32983-P-A (Reference 4.3-20). In general, the methodology described in the LTR adheres to the guidance in Regulatory Guide 1.190 for neutron flux evaluation and was approved by the U.S. NRC in the Safety Evaluation Report (SER) for referencing in licensing actions.

The fluence calculations are performed with the DORTG01V discrete ordinates transport code. The LTR provides a description of the DORT calculation used to determine the RPV fluence, as well as the calculations used to predict the measured dosimetry and validate the transport model. The calculational model includes a representation of the peripheral fuel assemblies and the core-internals, downcomer and vessel geometry. Calculations are performed to determine the bundle-average power distribution in the peripheral fuel bundles for input to the DORT core neutron source. Calculations employ a relatively fine (r, θ, z) spatial mesh and are carried out using an S_{12} angular quadrature set. The eighty-group MATXS cross section library is the basic nuclear data set. The cross section data used in these calculations is based on the ENDF/B-V nuclear data except for iron, hydrogen and oxygen. Since the cross sections for these elements have changed significantly in the more recent ENDF/B-VI data set, ENDF/B-VI cross sections were used for oxygen, hydrogen, and individual iron isotopes. The cross section library is used in performing the energy and spatial self-shielding and removal calculations. The scattering cross sections are represented using a P_3 Legendre expansion. The calculations are performed in azimuthal (r, θ) and axial (r, z) geometries. A synthesis technique is used to determine the three-dimensional fluence distribution.

Figure 4.3-1 shows a quadrant of the core and the vessel internal components that are relevant to the flux calculation (Reference 4.3-5). The reactor core is divided into three radial zones, based on the geometric layout of the bundles and their relative contribution to the shroud and RPV flux. The (r, θ) analysis used the polar coordinates to define the calculation model as a planar sector between pre-selected reactor azimuths (typically 0° and 90°). Since the surveillance capsule is centered close to the midplane elevation of the core, core midplane data is assumed for the analysis. The model includes several material regions radially: three in-core regions, the bypass water region, shroud, downcomer water, jet-pump riser, jet-pump inlet mixer, surveillance capsule holder/bracket, and the RPV cladding and base metal.

The core model for the axial (r,z) calculation is a cylinder simulating the cross-sectional area of the core at a pre-selected azimuth. For the capsule flux calculation, the (r,z) calculation was performed at the 300° azimuth, where the capsule is located. For the shroud/RPV flux calculation, the azimuth of 24° was selected because it is near the peak shroud flux and peak RPV flux. The core cylinder contains the afore-mentioned three radial zones for each of the 25 axial fuel nodes. Each axial fuel node is sub-divided into bundle-dependent radial regions so that each core region is modeled with its respective water density, structure material density, and actinide concentration. Similar to the (r, θ) model, there are bypass water, shroud, downcomer water, and RPV regions beyond the core.

Table 4.3-1 summarizes the neutron fluence results (Reference 4.3-5). Two sets of fluence data are presented: at the end of 40 years (33.1 EFPY), and at the end of 60 years (51.6 EFPY). Note EFPY is defined as 3323 MWt based effective full power years. The calculation of 33.1 EFPY factors in the uprated power (3486 MWt) from Cycle 11 through end of life (Reference 4.3-5). Fluence projections after Cycle 17 include a 10% adder to bound potential variation in future cycles.

The RPV peak fluence (at 33.1 EFPY) given in Table 4.3-1 is used for development of the P-T limit curves. The peak 1/4 T fluence values (n/cm^2) used for P-T curve development are: 1.75E+17 for lower shell #1, 5.11E+17 for lower-intermediate shell #2, 2.81E+17 for N6 nozzle and 2.13E+17 for girth weld between shell #1 and shell #2 (Reference 4.3-6). The 1/4 T fluences were calculated in accordance with RG 1.99, Revision 2.

4.3.3 ANALYTICAL METHODS

See Appendix A, subsection A.4.3.3 of Reference 4.3-4.

4.3.4 CHANGES

See Appendix A, subsection A.4.3.4 of Reference 4.3-4.

4.3.5 REFERENCES

4.3-1 Deleted.

4.3-2 Advanced Nuclear Fuels Corporation, "Generic Mechanical Design Criteria for BWR Fuel Designs," ANF-89-98 (P)(A), Revision 1 and Supplement 1, May 1995.

4.3-3 Deleted.

4.3-4 General Electric Standard Application for Reactor Fuel, NEDE-24011-P-A, and Supplement for United States, NEDE-24011-P-A-US (most recent approved version referenced in COLR).

4.3-5 GE Nuclear Energy, Washington Public Power Supply System WNP-2 RPV Surveillance Materials Testing and Analysis, Document No. GE-NE-B1301809-01, March 1997.

GE Nuclear Energy, "Energy Northwest Columbia Generating Station Neutron Flux Evaluation," GE-NE-0000-0023-5057-R0, April 2004.

4.3-6 GE Nuclear Energy, "Pressure-Temperature Curves for Energy Northwest Columbia," NEDC-33144-P (CVI CAL 1012-00,3).

4.3-7 Supplemental Reload Licensing Report for Columbia (most recent version referenced in COLR).

4.3-8 Fuel Bundle Information Report for Columbia (most recent version referenced in COLR).

4.3-9 Deleted.

4.3-10 Framatome ANP, "Nuclear Fuel Design Report Columbia Generating Station - 1 Fabrication Batch CGS-1 ATRIUM-10 Fuel," EMF-2866(P), Revision 0, December 2002.

4.3-11 Deleted.

4.3-12 GE Nuclear Energy, Washington Public Power Supply System Nuclear Project 2, "WNP-2 Power Uprate Transient Analysis Task Report," GE-NE-208-08-0393, September 1993.

4.3-13 GE Nuclear Energy, "Licensing Topical Report, General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluations," NEDC-32983-P-A, December 2001.

4.3-14 Energy Northwest, "Nuclear Fuel Design Report, Columbia Generating Station Fabrication Batch CGS1-18, ATRIUM™-10 Fuel," CGS-FTS-0163, Revision 1, October 2004.

4.3-15 AREVA NP, "Nuclear Fuel Design Report Columbia Generating Station Fabrication Batch CGS1-19 ATRIUM™-10 Fuel," ANP-2598(P), Revision 0, January 2007.

4.3-16 Letter from N. J. Carr, AREVA NP, to J. L. Lewis, Energy Northwest, "CGS1-19 Operation with FFTR at 355 Degrees," AEN-07-029, Revision 1, April 16, 2007.

4.3-17 Reference Loading Pattern (most recent version referenced in COLR).

4.3-18 GE14 Compliance With Amendment 22 of NEDE-24011-P-A (GESTAR II), NEDC-32868P, Revision 2, September 2007.

4.3-19 "Global Nuclear Fuels Fuel Bundle Designs," NEDE-31152P, Revision 9, May 2007.

4.3-20 GE Nuclear Energy, "Licensing Topical Report, General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluations," NEDC-32983-P-A, Revision 2, January 2006.

Table 4.3-1

Summary of Neutron Fluence Results

	Flux (n/cm^2-s)		Fluence (n/cm^2)	
	Cycle 10	Representative Future Cycle	40-year (33.1 EFPY)*	60-year (51.6 EFPY)*
RPV				
At Midplane	6.92E+08	5.75E+08	6.77E+17	1.03E+18
At Peak Elevation	7.60E+08	6.27E+08	7.41E+17	1.12E+18
Peak/Midplane	1.10	1.09	1.09	1.09
Elevation for 10^{17} fluence (inches above BAF)				
Bottom			-3.3	-7.0
Top			156.2	160.0
Shroud				
At Midplane	1.81E+12	1.54E+12	1.80E+21	2.75E+21
At Peak Elevation	2.07E+12	1.73E+12	2.02E+21	3.06E+21
Peak/Midplane	1.15	1.12	1.12	1.11
Top Guide	2.08E+13	1.91E+13	2.15E+22	3.31E+22
Core Plate	3.39E+11	3.04E+11	3.46E+20	5.31E+20

* EFPY is defined as 3323 MWt based effective full power years. The calculation of 33.1 and 51.6 EFPY factors in the uprated power (3486 MWt) from Cycle 11 through End of Life (Reference 4.3-5).

Table 4.3-2

Reload Fuel Neutronic Design Values

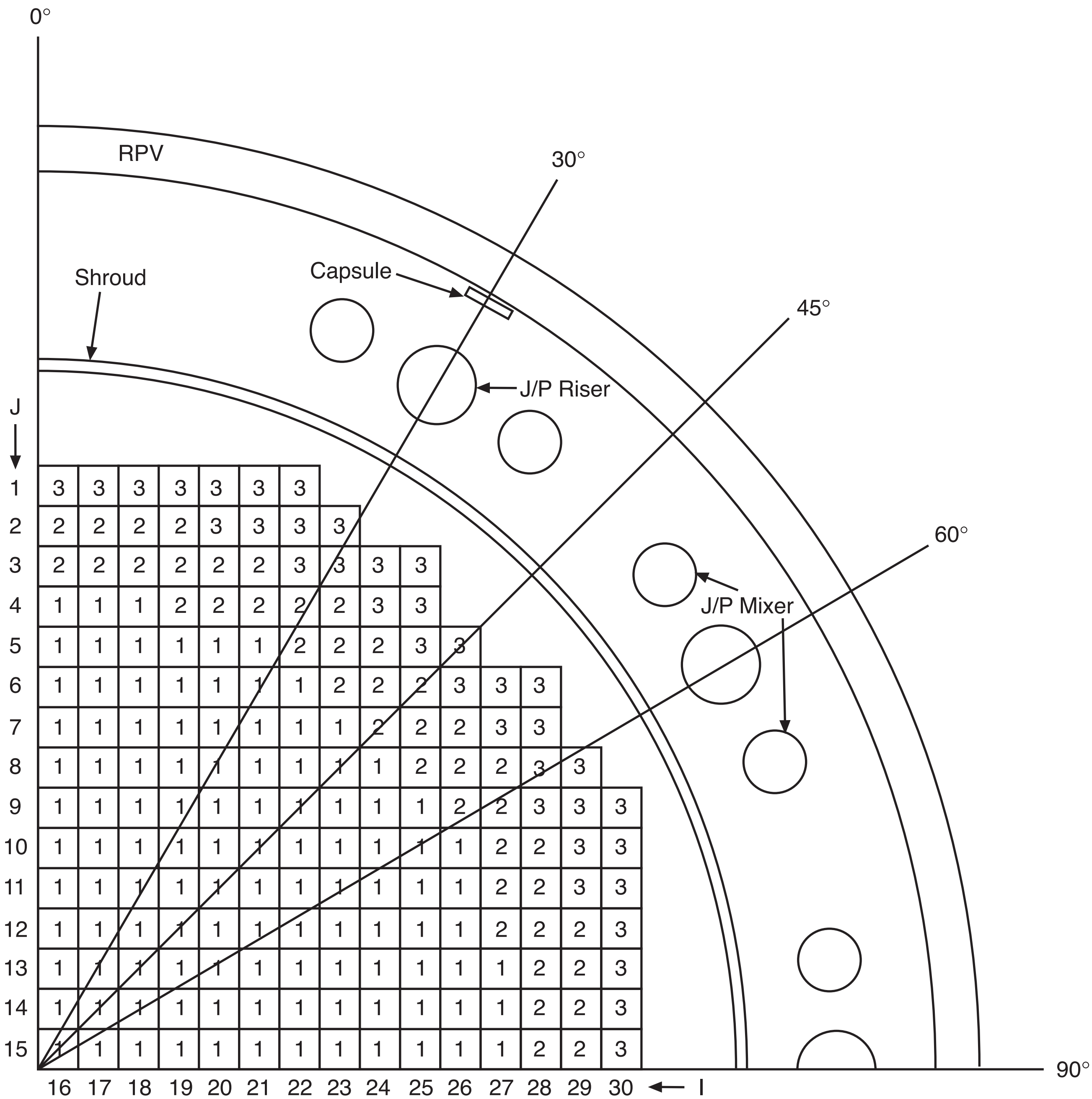
	GE14	ATRIUM-10
Fuel pellet		
Fuel material	UO_2 sintered pellets	UO_2 sintered pellets
Density, g/cm^3 % of T.D.	10.503 97.0	10.51 95.85
Diameter		
Enriched fuel	0.345	0.3413
Natural fuel	0.345	0.3413
Fuel rod		
Fuel length, full, in.	150	149.45
Fuel length, partial, in.	84	90.00
Cladding material	Zirconium/Zircaloy-2	Zircaloy-2
Clad I.D., in.	0.352	0.3480
Clad O.D., in.	0.404	0.3957
Fuel assembly		
Number of fuel rods, full length	78	83
Number of fuel rods, partial length	14	8
Number of inert water rods	2 water rods occupying 8 fuel rod locations	1 water channel
Fuel rod enrichments	Reference 4.3-8	References 4.3-10, 4.3-14, and 4.3-15
Fuel rod pitch, in.	0.510	0.510
Fuel assembly loading, kg uranium	Reference 4.3-8	177.6 to 177.9

Table 4.3-3

Neutronic Design Values

Parameter	Value
Core data	
Number of fuel assemblies	764
Rated power, MWt	3486
Rated core flow, Mlbm/hr	108.5
Core inlet enthalpy, Btu/lbm	528.7
Reactor dome pressure, psia	1035
Fuel assembly pitch, in.	6.00
Water gap thickness for ATRIUM-10 fuel, in.	0.522
Control rod data[a]	
Absorber material	B_4C
Total blade span, in.	9.75
Total blade support span, in.	1.58
Blade thickness	0.260
Blade face-to-face internal dimension, in.	0.200
Absorber rods per blade	76
Absorber rods outside diameter, in.	0.188
Absorber rods inside diameter, in.	0.138
Absorber density, % of theoretical	70.0

[a] Original equipment control rods. Some of the control blades are replaced with Duralife 215 and Marathon control blades.



Columbia Generating Station
Final Safety Analysis Report

Core Layout and Vessel Internal Components

Draw. No. 960690.10 | Rev. | Figure 4.3-1

4.4 THERMAL-HYDRAULIC DESIGN

4.4.1 DESIGN BASES

4.4.1.1 Safety Design Bases

See Appendix A, subsection A.4.4.1.1 of Reference 4.4-1, and Reference 4.4-2.

4.4.1.2 Requirements for Steady-State Conditions

See Appendix A, subsection A.4.4.1.2 of Reference 4.4-1, and Reference 4.4-5.

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

4.4.1.3 Requirements for Anticipated Operational Occurrences (AOOs)

See Appendix A, subsection A.4.4.1.3 of Reference 4.4-1, and Reference 4.4-2.

4.4.1.4 Summary of Design Bases

See Appendix A, subsection A.4.4.1.4 of Reference 4.4-1, and Reference 4.4-4.

4.4.2 DESCRIPTION OF THERMAL-HYDRAULIC DESIGN OF REACTOR CORE

See Appendix A, subsection A.4.4.2 of Reference 4.4-1 and Reference 4.4-8.

4.4.2.1 Summary Comparison

An evaluation of plant performance from a thermal and hydraulic standpoint is provided in Section 4.4.3. A tabulation of thermal and hydraulic parameters of the core is given in Table 4.4-1.

4.4.2.2 Critical Power Ratio

See Reference 4.4-7.

4.4.2.2.1 Crud Buildup Effect

No modifications to the critical power correlation are required to account for crud deposition.

4.4.2.2.2 GEXL14 Correlation Applied to GE14 Reload Fuel

The GEXL14 critical power correlation was developed using extensive test data based on the GE14 fuel design (see Reference 4.4-7). The approved GEXL14 correlation has been shown to accurately predict core thermal-hydraulic behavior in plant transient conditions.

4.4.2.2.3 GEXL97 Correlation Applied to ATRIUM-10 Reload Fuel

The GEXL97 correlation is applied to ATRIUM-10 fuel. This is accomplished using the procedures described in Reference 4.4-9.

4.4.2.3 Linear Heat Generation Rate

See Reference 4.4-12 for GE14 fuel and Reference 4.4-6 for AREVA NP fuel.

4.4.2.4 Void Fraction Distribution

The void fraction exit values are provided in Table 4.4-2.

4.4.2.5 Core Coolant Flow Distribution and Orificing Pattern

Correct distribution of core coolant flow among the fuel assemblies is accomplished by the orifices fixed at the inlet of each fuel assembly in the fuel support pieces. The orifices control the flow distribution and, hence, the coolant conditions within prescribed bounds throughout the design range of core operation. The sizing and design of the orifices ensure stable flow in each fuel assembly during normal operating conditions.

The core is divided into two orifice flow zones. The outer zone is a narrow, reduced-power region around the core periphery. The inner zone is the core central region. No other flow or steam distribution, other than that provided by adjusting power distribution with control rods, is used or needed.

4.4.2.5.1 Flow Distribution Data Comparison

Design core flow calculations were made using the design power distributions. The flow distribution to the fuel assemblies was calculated based on the assumption that the pressure drop across all of the fuel assemblies is the same. This assumption has been confirmed by measuring the flow distribution in BWRs. Therefore, there is a reasonable assurance that the calculated flow distribution throughout the core is in close agreement with the actual flow distribution (Reference 4.4-8). The GE14 fuel is hydraulically compatible with the ATRIUM-10 fuel differing only a few percent in key parameters (Reference 4.4-8). Therefore, there is a reasonable assurance that the Columbia Generating Station (CGS) flow distribution calculations with GE14 and ATRIUM-10 fuels agree with the actual data.

4.4.2.5.2 Effect of Channel Flow Uncertainties on the MCPR Uncertainty

The channel flow uncertainty has been inherently considered in its contribution to the MCPR uncertainty when evaluating the probability of a fuel rod subject to a boiling transition in establishing the safety limit MCPR (Reference 4.4-10).

The channel flow uncertainty is not an independent parameter contributing to the MCPR uncertainty. Its effect has been included in evaluating the probability of a boiling transition during a core wide power and flow calculation.

4.4.2.6 Core Pressure Drop and Hydraulic Loads

See References 4.4-1 and 4.4-8 (Global Nuclear Fuel [GNF]) and 4.4-5 (AREVA).

For pressure drop considerations, GNF models crud on fuel rods. This consideration is not included in the AREVA NP methodology.

4.4.2.7 Correlation and Physical Data

See References 4.4-7 and 4.4-9.

4.4.2.8 Thermal Effects of Operational Transients

See References 4.4-6 and 4.4-4.

4.4.2.9 Uncertainties in Estimates

See References 4.4-7, 4.4-10, and 4.4-11.

4.4.2.10 Flux Tilt Considerations

See Appendix A, subsection A.4.4.2.10 of Reference 4.4-1.

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

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 5.4-2.

4.4.3.1.2 Reactor Coolant System Thermal Hydraulic Data

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

4.4.3.1.3 Reactor Coolant System Geometric Data

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

Table 4.4-3 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 4.4-4 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 Figures 5.4-2 and 5.4-7. These curves are valid for all conditions with a normal operating range varying from approximately 25% to 105% of rated pump flow.

The pump characteristics, including considerations of net positive suction head (NPSH) requirements, are the same for the conditions of two-pump and one-pump operation as described in Section 5.4.1. Subsection 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

The power-flow operating map for the power range of operation is shown in Figure 4.4-1. The boundaries of this map are as follows.

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

b. Extended load line limit (ELLL): The line passes through 100% power at 88% core flow and is the 108% rod line,

c. Rated power line: Constant 100% power line,

d. ICF line: Constant 106% increased core flow line, and

e. Pump cavitation interlock line: This line is required to protect either the recirculation pumps or the jet pumps from cavitation damage.

4.4.3.3.2 Regions of the Power-Flow Map

a. Region I — This region defines the system startup operational capability with the recirculation pumps and motors being driven by the adjustable speed drives (ASDs). Flow is controlled by the variable speed pump, and power changes during normal startup and shutdown will be in this region;

b. Region II — This is the low power area of the operating map where cavitation can be expected in the recirculation pumps and jet pumps. Operation within this region is precluded by system interlocks that run back the pumps to minimum speed; and

c. 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 use of the variable speed pumps.

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 4.4-1:

a. Minimum power limits at intermediate and high core flows. To prevent cavitation in the recirculation pumps and jet pumps, the recirculating system is provided with an interlock to run back the pump speed to 15 Hz if the difference between steam line temperature and recirculation pump inlet temperature is less than a preset value (10.7°F). This action is initiated electronically through a time delay.

b. Minimum power limit at low core flow. During low power, low loop flow operations, the temperature differential interlock provides cavitation protection. Activation of the temperature differential interlock will run back the pump speed to 15 Hz. The ASD output speed/frequency is measured by instrumentation provided for monitoring the ASD. The speed change action is electronically initiated.

c. 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 25% of rated. To ensure this minimum flow, the system is designed so that the minimum pump speed will allow this rate of flow.

d. 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 individual loop manual control mode, or during ganged loop manual control.

The cavitation limits are established for two-pump operation, but will not protect the jet pumps and recirculation pumps on one-pump operation. Therefore, additional procedural operational limits are established to prevent cavitation damage during the single loop operation. The procedural operational limits are shown in Figure 4.4-2.

Flow Control. The principal modes of normal operation with ASD flow control are summarized as follows: The recirculation pumps are started when the suction and discharge block valves are full open; with the pump speed at 15 Hz the 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 4.4-1.

When reactor power is greater than approximately 20% of rated, the steam line to recirculation pump inlet differential temperature low feedwater flow interlock is cleared and the main recirculation pump speed can be manually increased from 15 Hz. The system is then brought to the desired power-flow level within the normal operating area of the map (Region III) by individual loop control or manual ganged control of the ASD system output frequency and by withdrawing control rods.

Recirculation pump speed increases resulting from ASD system output frequency increases toward 63 Hz with constant control rod position will result in power/flow changes along, or nearly parallel to, the 100% rod line.

4.4.3.4 Temperature-Power Operating Map

Not applicable.

4.4.3.5 Load-Following Characteristics

The automatic load following feature has been deleted from the system. All load increases or decreases on CGS are manually controlled by the operator.

4.4.3.6 Thermal and Hydraulic Characteristics Summary Table

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

4.4.4 EVALUATION

4.4.4.1 Bypass Flow

Table 4.4-5 shows the bypass flows for the two cases of the GE14 and AREVA NP ATRIUM-10 core.

4.4.4.2 Thermal Hydraulic Stability Analysis

Core thermal-hydraulic analyses support stability regions specified in the Technical Specifications. The Technical Specifications define regions on the power-flow map that preclude operation and regions when operation is allowed with surveillance provided by the stability monitoring system.

4.4.5 TESTING AND VERIFICATION

See Appendix A, subsection A.4.4.5 of Reference 4.4-1.

4.4.6 INSTRUMENTATION REQUIREMENTS

See Appendix A, subsection A.4.4.6 of Reference 4.4-1.

4.4.6.1 Loose Parts

The instrumentation for online monitoring for loose parts in the reactor vessel has been deactivated.

See Section 7.7.1.12 for further information.

4.4.7 REFERENCES

4.4-1 General Electric Company, General Electric Standard Application for Reactor Fuel, NEDE-24011-P-A, and Supplement for United States, NEDE-24011-P-A-US (most recent approved version referenced in COLR).

4.4-2 Advanced Nuclear Fuels Corporation, Generic Mechanical Design Criteria for BWR Fuel Designs, ANF-89-98(P)(A), Revision 1 and Supplement 1, May 1995.

4.4-3 Deleted.

4.4-4 Supplemental Reload Licensing Report for Columbia (most recent version referenced in COLR).

4.4-5 Mechanical and Thermal-Hydraulic Design Report for Columbia Generating Station ATRIUM-10 Fuel Assemblies, EMF-2865(P), Revision 1, April 2003.

4.4-6 AREVA NP, Columbia Generating Station Cycle 19 Reload Analysis, ANP-2602, Revision 0, March 2007.

4.4-7 GEXL14 Correlation for GE14 Fuel, NEDC-32851P-A, Revision 4, September 2007.

4.4-8 GE14 Thermal-Hydraulic Compatibility with Columbia Legacy Fuel, GNF S-0000-0092-8136, Revision 0, October 2008.

4.4-9 GEXL97 Correlation Applicable to ATRIUM-10 Fuel, NEDC-33419P, Revision 0, June 2008.

4.4-10 Methodology and Uncertainties for Safety Limit MCPR Evaluations, NEDC-32601P-A, August 1999.

4.4-11 Power Distribution Uncertainties for Safety Limit MCPR Evaluations, NEDC-32694P-A, August 1999.

4.4-12 Fuel Bundle Information Report for Columbia (most recent version referenced in COLR).

Table 4.4-1

Thermal and Hydraulic Design Characteristics
of the Reactor Core

General Operating Conditions	Parameter
Reference design thermal output, MWt	3486
Power level for engineered safety features, MWt	3716
Steam flow rate, at 421.2°F final feedwater temperature, millions lb/hr	15.01
Core coolant flow rate range, millions lb/hr	95.5-115
Feedwater flow rate, millions lb/hr	14.98
System pressure, nominal in steam dome, psia	1035
Core exit pressure, nominal, psia	1047
Coolant saturation temperature at core design pressure, °F	550
Average power density, kW/liter	51.56
Average linear heat generation rate, kW/ft	4.05
Core total heat transfer area, ft^2	86,099
Average heat flux, Btu/hr-ft^2	132,790
Design operating minimum critical power ratio (MCPR)	(see COLR)[a]
Core inlet enthalpy at 421.2°F FFWT, Btu/lb	528.7
Core inlet temperature, at 421.2°F FFWT, °F	533.9
Power assembly exit void fraction, % (RPF=1.0)	71.2
Assembly flow, klbm/hr	120.4[b]
Core pressure drop, psid	23.437[b]

[a] Core Operating Limits Report.

[b] Based on full core of GE14, (1035 psia dome pressure, 3486 MWt (100%) power and 108.5 Mlbm/hr (100% of rated core flow).

Table 4.4-2

Mixed Core Thermal Hydraulic Analysis Results[a]

	GE14	ATRIUM-10
Assembly flow (Klb/hr)	114.38	116.07
Exit quality (active region)	0.244	0.242
Exit void fraction	0.813	0.811
Critical power ratio[b]	1.531	1.439

[a] Core 1/3 GE14 fuel and 2/3 ATRIUM-10 fuel
3486 MWt core flow: 108.5 Mlbm/hr high power assembly
1.40 Radial peaking factor.

[b] Estimates obtained using the GEXL critical power correlation (References 4.4-7 and 4.4-9).

Table 4.4-3

Reactor Coolant System Geometric Data

	Flow Path Length (in.)	Height and Liquid Level (in.)	Elevation of Bottom of Each Volume[a]	Minimum Flow Areas (ft^2)
Lower plenum	216	216 216	-172.5	71.5
Core	164	164 164	44.0	142.0
Upper plenum and separators	178	178 178	208.0	49.5
Dome (above normal water level)	312	312 0	386.0	343.5
Downcomer area	321	321 321	-51.0	79.5
Recirculation loops and jet pumps (one loop)	108.5 ft	403 403	-394.5	132.5 in^2

[a] Reference point is recirculation nozzle outlet centerline.

Table 4.4-4

Lengths and Sizes of Safety Injection Lines

	Line O.D. (in.)	Line Length (ft)
HPCS line		
Pump discharge to valve[a]	16	319
From HPCS-V-4 inside containment to RPV	12.75	108
Total		427
LPCI lines		
Loop A		
1. Pump discharge to reducer	18	421
2. Reducer to injection valve,[a] RHR-V-42A	14	6
3. From RHR-V-42A to RPV	14	94
Total		521
Loop B		
1. Pump discharge to reducer	18	394
2. Reducer to injection valve RHR-V-42B	14	6
3. Inside containment to RPV	14	93
Total		493
Loop C		
1. Pump discharge to reducer	18	71
2. Reducer to injection valve RHR-V-42C	14	138
3. Inside containment to RPV	14	99
Total		308
LPCS line		
Pump discharge to valve[a]	16	222
Inside containment to RPV	12.75	117
Total		339

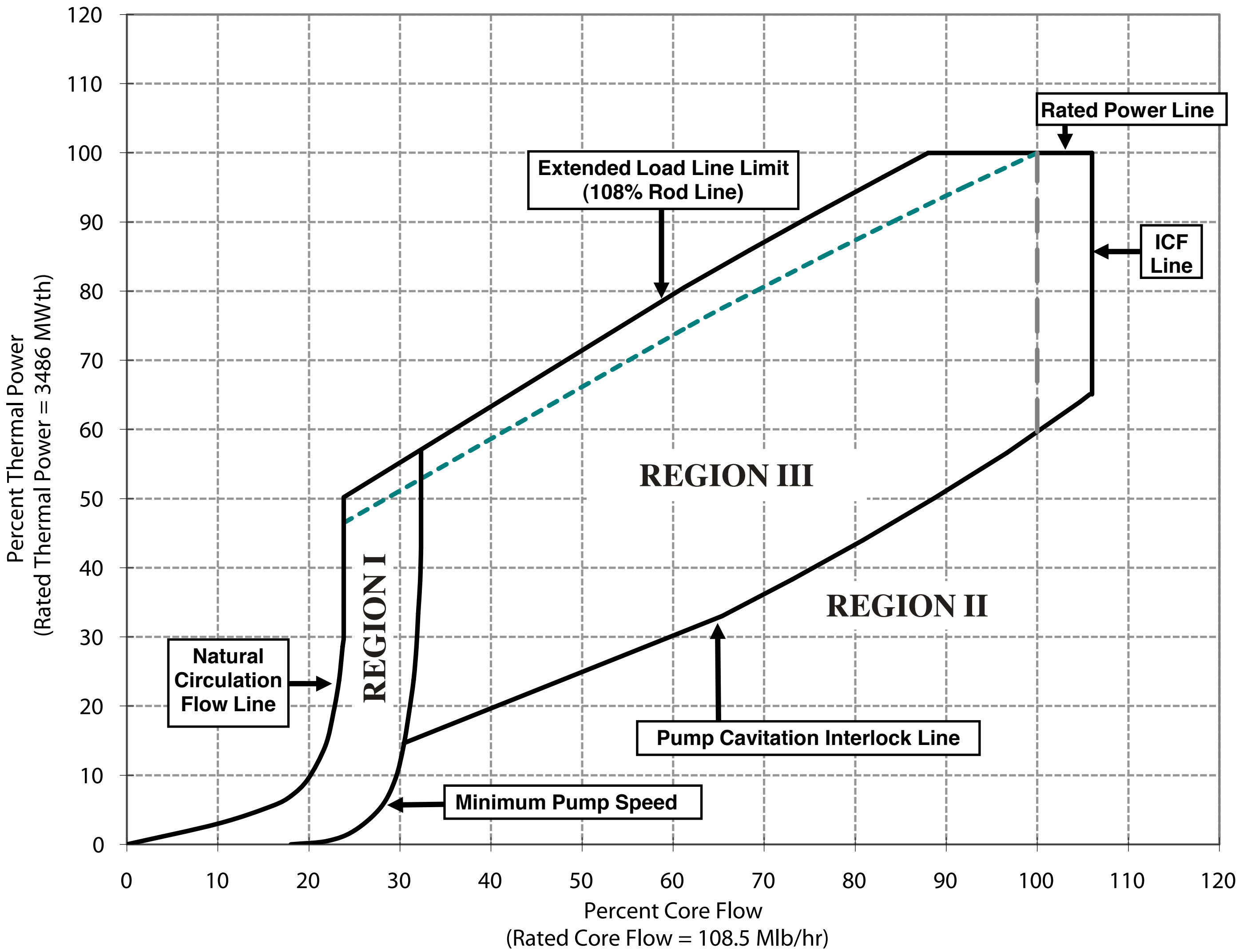
[a] Injection valve located as near as possible to outside of containment wall.

Table 4.4-5

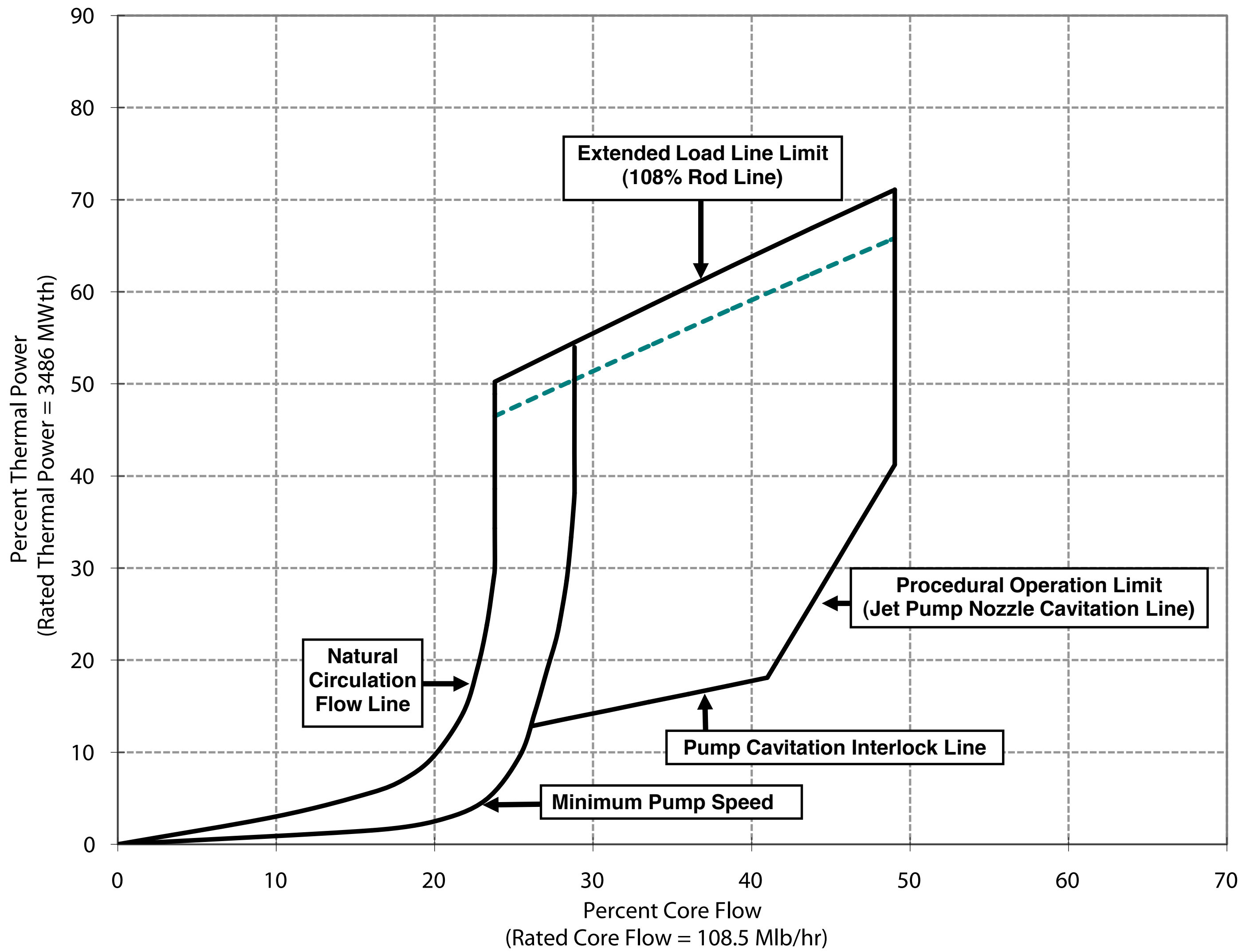
Core Pressure Drop and Leakage Flow
Results for Core Configurations

Case	Core Pressure Drop (psid)	Core Bypass Flow (%)
1: All GE14 core	23.437	14.86[b]
2: All AREVA NP ATRIUM-10 core	21.8	10.8

[a] Core power: 3486 MWt; core flow: 108.5 Mlb/hr.
[b] Including both leakage flow and water rod flow.



Power-Flow Operating Map
Two Loop Operation



Columbia Generating Station Final Safety Analysis Report	Power-Flow Operating Map Single Loop Operation
	Draw. No. 060108.05 / Rev. / Figure 4.4-2

4.5 REACTOR MATERIALS

4.5.1 CONTROL ROD SYSTEM STRUCTURAL MATERIALS

4.5.1.1 Material Specifications

The following material listing applies to the control rod drive (CRD) mechanism supplied for this application. The position indicator and minor nonstructural items are omitted.

a. Cylinder, tube, and flange assembly

Flange	ASME SA 182 grade F304
Housing cap screws	ASME SA 540 grade B23, CL4 or SA 193 grade B7
Plugs	ASME SA 182 grade F304
Cylinder	ASTM A269 grade TP 304
Outer tube	ASTM A269 grade TP 304
Tube	ASTM A269 grade TP 304
Spacer	ASTM A269 grade TP 304 or ASTM A511 grade MT 304

b. Piston tube assembly

Piston tube	ASTM A269 grade TP 304 or ASTM A479 grade XM-19
Stud	ASTM A276 type 304
Head/base	ASME SA 182 grade F304
Indicator tube	ASME SA 312 type 316
Cap	ASME SA 182 grade F304 or TP 316

c. Drive assembly

Coupling spud	Inconel X-750
Index tube	ASTM A269 grade TP 304 or ASTM A479 grade XM-19

Piston head	Armco 17-4 PH
Coupling	ASME SA 312 grade TP 304 or ASTM A511 grade MT 304
Magnet housing	ASME SA 312 grade TP 304 or ASTM A511 grade MT 304

d. Collet assembly

Collet piston	ASTM A269 grade TP 304 or ASME SA 312 grade TP 304
Finger	Inconel X-750
Retainer	ASTM A269 grade TP 304 or ASTM A511 grade MT 304
Guide cap	ASTM A269 grade TP 304

e. Miscellaneous parts

Stop piston	ASTM A276 type 304
Connector	ASTM A276 type 304
O-ring spacer	ASME SA 240 type 304
Piston tube nut	ASME SA 194 grade B8 or B8A or 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
Ring flange cap screws	ASME SA 193 grade B6

The materials listed under ASTM specification numbers 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 strength of 90,000/125,000 psi, yield strength of 50,000/85,000 psi, and minimum elongation of 25%.

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

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

4.5.1.2 Special Materials

No cold-worked austenitic stainless steels with a yield strength greater than 90,000 psi are employed in the CRD system. Armco 17-4 PH (precipitation hardened stainless steel) is used for the piston head. This material is aged to the H-1100 condition to produce resistance to stress corrosion cracking in the BWR environments. Armco 17-4 PH (H-1100) has been successfully used in the past in BWR drive mechanisms. The only hardenable martensitic stainless steel used is the ring flange cap screws. The material is TP 410 in the H-1100 condition.

4.5.1.3 Processes, Inspections, and Tests

All austenitic stainless steel used in the CRD system is solution annealed material with one exception, the outer tube in the cylinder, tube, and flange assembly (see Section 4.5.1.1). Proper solution annealing is verified by testing per ASTM A262, "Recommended Practices for Detecting Susceptibility to Intergranular Attack in Stainless Steels."

Two special processes are employed which subject selected components to temperatures in the sensitization range. These processes are performed on austenitic stainless steel, including XM-19.

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

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

    1. Tube (cylinder, tube, and flange assembly)
    2. Piston tube (piston tube assembly)
    3. Index tube (drive line assembly)
    4. Collet piston and guide cap (collet assembly)

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

All austenitic stainless steel is required to be in the solution heat treated condition. Welding is performed in accordance with Section IX of the ASME Boiler and Pressure Vessel (B&PV) Code. Heat input for stainless-steel welds is restricted to a maximum of 50,000 joules/in. 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.

4.5.1.4 Control of Delta Ferrite Content

All type 308 weld metal is required to comply with a specification which requires a minimum of 5% delta ferrite. This amount of ferrite is adequate to prevent any micro-fissuring (hot cracking) in austenitic stainless steel welds. (See Section 4.5.2.4.)

4.5.1.5 Protection of Materials During Fabrication, Shipping, and Storage

All the CRD parts listed in Section 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:

a. Any processing which increases part temperature above 200°F,
b. Assembly which results in decrease of accessibility for cleaning, and
c. Release of parts for shipment.

The specification for packaging and shipping the CRD provides for the following:

The drive is rinsed in hot deionized water and dried in preparation for shipment. The ends of the drive are then covered with a vapor tight barrier with desiccant. Packaging is designed to protect the drive and prevent damage to the vapor barrier. The planned storage period considered in the design of the container and packaging is 2 years. This packaging has been in use for a number of years. Periodic audits have indicated satisfactory protection.

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

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

4.5.2 REACTOR INTERNAL MATERIALS

4.5.2.1 Material Specifications

Materials used for the core support structure:

a. Shroud support - Nickel chrome iron alloy, ASME SB166 or SB168,

b. Shroud, core plate (and aligners), top guide (and aligners), and internal structures welded to these components, ASME SA240, SA182, SA479, SA312, SA249, or SA213 (all type 304, except the shroud which is 304L),

c. Peripheral fuel supports - SA312 type 304,

d. Core plate and top guide studs and nuts, and core plate wedges. ASME SA479, SA193 grade B8, SA194 grade 8 (all type 304),

e. Control rod drive housing. ASME SA312 type 304, SA182 type 304,

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

g. Orificed fuel support. ASME SA351 type CF8.

Materials used in the steam separators and steam dryers:

a.	All materials are type 304 stainless steel,	
b.	Plate, sheet, and strip	ASTM A240, type 304,
c.	Forgings	ASTM A182, grade F304,
d.	Bars	ASTM A479, type 304,
e.	Pipe	ASTM A312, grade TP 304,
f.	Tube	ASTM A269, grade TP 304,
g.	Bolting material	ASTM A193, grade B8,
h.	Nuts	ASTM A194, grade 8, and
i.	Castings	ASTM A351, grade CF8.

Materials used in the jet pump assemblies:

The components in the jet pump assemblies are a riser, restrainer brackets, inlet-mixers, slip joint clamps, diffusers, and a riser brace. Materials used for these components are to the following specifications:

a.	Castings	ASTM A351 grade CF 8 and ASME SA351 grade CF3,
b.	Bars	ASTM A276 type 304 and ASTM A370 grade E38 and E55,
c.	Bolts	ASTM A193 grade B8 or B8M,
d.	Sheet and plate	ASTM A240 type 304, ASTM A276 type 304, ASTM A358, and ASME SA240 type 304L,
e.	Tubing	ASTM A269 grade TP 304,
f.	Pipe	ASTM A358 type 304 and ASME SA312 grade TP 304,
g.	Welded fittings	ASTM A403 grade WP304, and
h.	Forgings	ASME SA182 grade F304, ASTM B166, and ASTM A637 grade 688.

Materials in the jet pump assemblies which are not type 304 stainless steel are listed below:

a. The inlet mixer adapter 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, inserts and beam made of Inconel X-750 to Specification ASTM B637 grade UNS N07750 (Beam), and ASTM A370 grade E38 and E55 (pin and insert);

d. The jet pump beam bolt is type 316L stainless steel;

e. The jet pump slip joint clamp body is fabricated from solution heat-treated ASTM A-182/ASME SA-182 Grade F XM-19 stainless steel with a maximum of 0.04% carbon;

f. The jet pump slip joint clamp adjustable bolt, bolt retainer, pins, and ratchet lock spring are fabricated from ASTM B-637/ASME SB-637 UNS N07750 Type 3 (Alloy X-750); and

g. All components of jet pump restrainer bracket auxiliary wedge assemblies are fabricated from ASTM B-637 UNS N07750 Type 3 (Alloy X-750), except for the frame.

All core support structures are fabricated from ASME specified materials and designed using ASME Code Section III, Appendix I allowable stresses, and ASME Code Section III, Class I, reactor vessel design rules as guides. The other reactor internals are fabricated from ASTM specification materials. Material requirements in the ASTM specifications are identical to requirements in corresponding ASME material specifications. The allowable stress levels specified in ASME Code Section III, Appendix I, are used as a guide in the design of all internal structures in the reactor.

4.5.2.2 Controls on Welding

For core support structures and other internals, weld procedures and welders are qualified in accordance with the ASME B&PV Code, Section IX.

4.5.2.3 Nondestructive Examination of Wrought Seamless Tubular Products

Wrought seamless tubular products are used in the fabrication of the CRD housing. This ASME Code Section III component is designed to the rules of Subsection NB, and the material specified is ASME SA-312 supplemented by GE specifications which invoke Subsection NB requirements. This material meets the requirements of NB-2550 and meets the intent of Regulatory Guide 1.66. The CRD housings are built to the 1971 Edition, Summer 1971 Addenda of the code.

Other internal non-code safety and non-safety components are optionally fabricated from wrought seamless tubular products. This material is supplied in accordance with the applicable ASTM material specifications and is nondestructively examined to the extent specified therein. In addition, the specification for tubular products employed for CRD housings external to the reactor pressure vessel (RPV) meet requirements of paragraph NB-2550 which meets the intent of Regulatory Guide 1.66.

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

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

Regulatory Guide 1.31, Control of Stainless Steel Welding

All austenitic stainless steel weld filler materials were supplied with a minimum of 5% delta ferrite. This amount of ferrite is considered adequate to prevent micro-fissuring in austenitic stainless steel welds. An extensive test program performed by General Electric Company, with the concurrence of the Regulatory Staff, has demonstrated that controlling weld filler metal ferrite at 5% minimum produces production welds which meet the requirements of Regulatory Guide 1.31. A total of approximately 400 production welds in five BWR plants were measured and all welds met the requirements of the Interim Regulatory Position to Regulatory Guide 1.31.

Regulatory Guide 1.34, Control of Electroslag Weld Properties.

Electroslag welding is not employed for any reactor internals.

Regulatory Guide 1.36, Nonmetallic Thermal Insulation for Austenitic Stainless Steel.

Nonmetallic thermal insulation is not employed for any components in the reactor vessel. For external applications, all nonmetallic insulation meets the requirements of Regulatory Guide 1.36.

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

All wrought austenitic stainless steel was solution heat treated. Heating above 800°F was prohibited (except for welding) unless the stainless steel was subsequently solution annealed. Purchase specifications restricted the maximum weld heat input to 110,000 joules per in., and the weld interpass temperature to 350°F maximum. Welding was performed in accordance with Section IX of the ASME B&PV Code Section IX. 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 Accessibility

Welder qualification for areas of limited accessibility is discussed in Sections 1.8.2 and 1.8.3.

4.5.2.5 Contamination, Protection, and Cleaning of Austenitic Stainless Steel

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

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

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

4.5.3 CONTROL ROD DRIVE HOUSING SUPPORTS

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

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

All CRD housing support subassemblies are fabricated of ASTM A36 structural steel, except for the following items:

a.	Grid	ASTM A441,
b.	Disc springs	Schnerr, type BS-125-71-8,
c.	Hex bolts and nuts	ASTM A307, and
d.	6 x 4 x 3/8 tubes	ASTM A500 grade B.

4.6 FUNCTIONAL DESIGN OF REACTIVITY CONTROL SYSTEMS

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

4.6.1 INFORMATION FOR THE CONTROL ROD DRIVE SYSTEM

4.6.1.1 Control Rod Drive System Design

4.6.1.1.1 Design Bases

4.6.1.1.1.1 Safety Design Bases. The CRD mechanical system meets the following safety design bases:

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

b. The design includes positioning devices, each of which individually supports and positions a control rod.

c. Each positioning device

   1. Prevents its control rod from initiating withdrawal as a result of a single malfunction,

   2. Is individually operated so that a failure in one positioning device does not affect the operation of any other positioning device, and

   3. Is 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.

4.6.1.1.1.2 Power Generation Design Basis. The CRD system design provides for positioning the control rods to control power generation in the core.

4.6.1.1.2 Description

The CRD system controls gross changes in core reactivity by incrementally positioning neutron absorbing control rods within the reactor core in response to manual control signals. It is also required to quickly shut down the reactor (scram) in emergency situations by rapidly inserting withdrawn control rods into the core in response to a manual or automatic signal. The CRD system consists of locking piston CRD mechanisms, and the CRD hydraulic system (including power supply and regulation, hydraulic control units (HCUs), 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 Figures 4.6-1 through 4.6-4). The individual drives are mounted on the bottom head of the reactor pressure vessel (RPV). The drives do not interfere with refueling and are operative even when the head is removed from the RPV.

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-in. increments of stroke over the length of the core.

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

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

The individual rod indicators, grouped in one control panel display, correspond to relative rod locations in the core. A separate, smaller display is located just below the large display on the vertical part of the benchboard. This display presents 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 local power range monitor (LPRM) strings (see Section 7.6.1.4). Rod groups at the periphery of the core may have less than four rods. The small rod display shows the positions, in digital form, of the rods in the group to which the selected rod belongs. A white light indicates which of the four rods is the one selected for movement.

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

4.6.1.1.2.2.1 Drive Piston. The drive piston is mounted at the lower end of the index tube. This tube functions as a piston. 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 down-travel or withdrawal is approximately 1.2 in.^{2} versus 4.1 in.^{2} for up-travel or insertion. This difference in driving area tends to balance the control rod weight and ensures 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 in. 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 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 end of the stop piston is threaded on to the top of the piston tube forming a space for a set of spring washers which serve to protect both the drive piston and the stop piston from damage as the drive piston reaches its end of travel. The upper end of the piston tube has a series of orifices which hydraulically dampen the drive piston motion as the inner seals (or buffer seals) slide past them, effectively cutting off the exhaust path for the over-piston water. 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, position indicator 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 in. 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 down-travel 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.

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

Two means of uncoupling are provided. With the reactor vessel head removed, the lock plug can be raised against the spring force of approximately 50 lb 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.

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

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

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

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

4.6.1.1.2.3.2 Coupling Spud. The coupling spud is made of Inconel 750 that is aged for maximum physical strength and the required corrosion resistance. Because misalignment tends to cause chafing in the semispherical contact area, the part is protected by a thin chromium plating (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 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.

Experience at some operating boiling water reactors (BWR) indicates that failures can occur in the collet fingers of the CRD mechanism. To resolve this problem, some BWR facilities installed a revised collet retainer design. However, CGS does not have the revised collet retainer design. General Electric (GE) has demonstrated by testing and operating experience that the existing CRDs meet all safety and licensing requirements and are expected to give full life performances. However, as a result of examining operating drives, GE has discovered evidence of intergranular stress corrosion cracking (IGSCC) in some CRD drive components and has made design improvements to preclude IGSCC in the future. The spare parts for CRD components purchased by Energy Northwest incorporate this revised design. Along with the other parts of the drive, the collet retainer tube, piston tube, and index tube will be routinely checked and changed out, if necessary, with the parts incorporating the revised design.

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:

a. Seals and bushings on the drive piston and stop piston are Graphitar 14,

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

c. The ball check valve is a Haynes Stellite cobalt-base alloy,

d. Elastomeric O-ring seals are ethylene propylene,

e. Metal piston rings are Haynes 25 alloy,

f. Certain wear surfaces are hard faced with Colmonoy 6,

g. Nitriding by a proprietary new malcomizing process and chromium plating are used in certain areas where resistance to abrasion is necessary, and

h. The drive piston head is made of Armco 17-4 PH.

Pressure containing portions of the drives are designed and fabricated in accordance with requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section III.

4.6.1.1.2.4 Control Rod Drive Hydraulic System. The CRD hydraulic system (Figure 4.6-5) 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 HCU to the scram discharge volume (SDV). 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 nonmoving drives. Each CRD has an associated HCU.

4.6.1.1.2.4.1 Hydraulic Requirements. The CRD hydraulic system design is shown in Figures 4.6-5 and 4.6-6. The hydraulic requirements, identified by the function they perform, are as follows:

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

b. Drive water header pressure of approximately 260 psi above reactor vessel pressure is required. A flow rate of approximately 4 gpm to insert a control rod and 2 gpm to withdraw a control rod is required;

c. Cooling water to the drives is required at approximately 20 psi above reactor vessel pressure and at a flow rate of approximately 0.34 gpm per drive unit.

(Cooling water to a drive can be interrupted for short periods without damaging the drive);

d. The SDV 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);

e. Purge water flow to the RPV level instrumentation reference leg backfill system at a flow rate from 0.6 gal/hr to 2.4 gal/hr.

4.6.1.1.2.4.2 System Description. The CRD hydraulic systems provide the required functions with the pumps, filter, valves, instrumentation, and piping shown in Figure 4.6-5 and described in the following.

Duplicate components are included, where necessary, to ensure continuous system operation if an inservice component requires maintenance.

4.6.1.1.2.4.2.1 Supply Pump. One supply pump pressurizes the system with water from a condensate supply header, which takes suction from the condensate treatment system and/or condensate storage tanks depending on plant operation. One installed spare pump is provided for standby. A discharge check valve prevents backflow through the nonoperating pump. A portion of the pump discharge flow is diverted through a minimum flow bypass line to the condensate storage tank. This flow is controlled by an orifice and is sufficient to prevent immediate pump damage if the pump discharge is inadvertently closed.

Condensate water is processed by two filters in the system. The normal CRD pump suction flow path includes a 25-µ filter with a 250-µ Y-strainer upstream of the filter. The filtration capacity of these two in-series elements is limited by and therefore characterized by, the 25-µ filter. In addition, another 250-µ Y-strainer is provided on the pump suction bypass line (see Figure 4.6-5).

The filters used on the CRD system are of a rugged design and failure of the filters are not considered likely. Alarms are provided to give an early warning to the operator that maintenance is required.

The only known mode of failure of the filter element is for it to collapse due to high differential pressure. The CRD pump suction filter can withstand a maximum differential pressure of 20 psi and an alarm indicates in the control room high suction filter differential pressure at 8 psi. The filter element is additionally protected and strengthened by a stainless steel, perforated center tube. The CRD pump discharge filter can withstand a maximum differential pressure of 300 psi and an alarm indicates in the control room high differential pressure at 20 psi. The filter element is constructed entirely of stainless steel.

If the CRD systems pump suction and discharge filters were bypassed completely, possible presence of corrosion particles would not affect the reliability of the scram function of the CRD system. The presence of corrosion particles may accelerate wear of the drive components over a period of time. However, such wear is not a safety concern since this degradation in drive performance already occurs during normal rod operations and is detectable.

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

Pressure in the charging header is monitored in the control room with a pressure indicator and low pressure alarm. Charging water header pressure is not essential to successfully scram the plant. Each of the accumulators are prevented from leaking back to the charging water header by a check valve. Therefore, the pressure required to scram each rod is maintained. The integrity and leaktightness of these check valves are routinely tested as part of the surveillance test program. In addition, when the reactor is at rated pressure, no accumulator pressure is necessary to scram the plant.

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

4.6.1.1.2.4.2.3 Drive Water Pressure. Drive water pressure required in the drive header is maintained by the drive/cooling water pressure control valve, which is manually adjusted from the control room. A flow rate of approximately 6 gpm (the sum of the flow rate required to insert and withdraw a control rod) normally passes from the drive water pressure stage through two solenoid-operated stabilizing valves (arranged in parallel) and then goes into the cooling water header. The flow through one stabilizing valve equals the drive insert flow; that of the other stabilizing valve equals the drive withdrawal flow. When operating a drive, the required flow is diverted to that drive by closing the appropriate stabilizing valve while at the same time opening the drive directional control and exhaust solenoid valves. Thus, flow through the drive/cooling water 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 water pressure valve. The drive/cooling water 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 water 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 valve are required only to adjust for changes in the cooling requirements of the drives, as 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 monitored by a temperature recorder.

4.6.1.1.2.4.2.5 Scram Discharge Volume. The CGS SDV header system is designed as a continually expanding path from the 185 individual 0.75-in. scram discharge (withdrawal) lines to one of two integrated scram discharge volume/instrument volume (SDV/IV) systems (one system per approximately half the drives). Each integrated SDV/IV system consists of a continuously downsloping piping run expanding from the SDV (consisting of seven 6-in. Return headers from the individual HCU banks to an 8-in. combined return header) to the 12-in. vertically oriented IV. The only location where blockage need be assumed (piping less than 2-in. diameter) is in the 0.75 in. discharge line from the individual HCU. Blockage here would only cause failure of one control rod to insert. This is an acceptable consequence for a single failure and has been evaluated as part of the plant design basis. The header piping is sized to receive and contain all the water discharged by the drives during a full scram (3.34 gal per drive) independent of the IV.

During normal plant operation each SDV is empty and vented to the atmosphere through its open vent and drain valve. When a scram occurs on a signal from the safety circuit, these vent and drain valves are closed to conserve reactor water. Redundant vent and drain valves are incorporated in the design of the SDV to ensure that no single failure can result in uncontrolled loss of reactor coolant. Lights in the control room indicate the position of these valves.

During a scram, the SDV partly fills with water discharged from above the drive pistons. After scram is completed, the CRD seal leakage from the reactor continues to flow into the SDV 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 (RPS), the SDV signal is overridden with a key lock override switch, and the SDV 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 RPS. Closing the SDV valves allows the outlet scram valve seats to be leak tested by timing the accumulation of leakage inside the SDV.

Six liquid-level switches and two level transmitters directly connected to each instrument volume monitor the volume for abnormal water level. They provide redundant and diverse input to the RPS scram function and control room annunciation and control rod withdrawal block function. They are set at three different levels. At the lowest level, a level switch actuates to indicate that the volume is not completely empty during post-scram draining or to indicate that the volume starts to fill through leakage accumulation at other times during reactor operation. At the second level, one level switch produces a rod withdrawal block to prevent further withdrawal of any control rod, when leakage accumulates to half the capacity of the instrument volume. The remaining four switches are interconnected with trip channels of the RPS and will initiate a reactor scram on high water level while sufficient volume for a full scram still exists within the SDV. Two of these switches are actuated by level transmitters to provide diversity of signals to the RPS.

In the event of a slow or partial loss of air pressure, the high-level scram setpoint and the SDV/IV system capacity ensure that scram capability is maintained even in the event of maximum inleakage into the SDV prior to a scram. Analysis, assuming the maximum inleakage of 5 gpm and using the actual calculation piston-over area to determine the scram volume requirements, shows that adequate scram discharge volume will remain in the SDV system at the time that a scram is initiated.

A partial loss of air pressure does not result in the uncontrolled release of reactor coolant to the reactor building should all or most of the scram discharge valves lift. When the water buildup reaches scram initiation level in the IV, a scram signal is produced. This will cause the air supply to the vent and drain valves to vent, thereby ensuring that the vent and drain valves close and isolate. For leakage rates that do not result in buildup in the IV, the leak will drain to the reactor building equipment drain system. The drain system will alarm for leakage rates greater than 5 gal/minute. The operator can then take appropriate action, e.g., isolate the leak, scram the reactor, increase air pressure, etc., as required.

4.6.1.1.2.4.3 Hydraulic Control Units. Each 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 Section 7.7.1.2.

The basic components in each HCU are manual, pneumatic, and electrical valves; an accumulator, related piping, electrical connections, filters, and instrumentation (see Figures 4.6-5, 4.6-6, and 4.6-7). The components and their functions are described in the following.

4.6.1.1.2.4.3.1 Insert Drive Valve. The insert drive valve 123 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 121 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 122 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 120 opens on a withdraw signal and discharges water from below the main drive piston to the exhaust header. It also serves as the settle valve, which opens following any normal drive movement (insert or withdraw) to allow the control rod and its drive to settle back into the nearest latch position.

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

4.6.1.1.2.4.3.6 Scram Pilot Valves. The scram pilot valves are operated from the RPS. Two scram pilot valves control both the scram inlet valve and the scram exhaust valve. The scram pilot valves are identical, three-way, solenoid-operated, normally energized valves. On loss of electrical signal to the pilot valves, such as the loss of external ac power, the inlet ports close and the exhaust ports open on both valves. The pilot valves (Figure 4.6-5) are arranged so that the trip system signal must be removed from both valves before air pressure can be discharged from the scram valve operators.

This prevents the 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 toe 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. A position indicator switch on this valve energizes a light in the control room as soon as the valve starts to open.

4.6.1.1.2.4.3.9 Scram Accumulator. The scram accumulator stores sufficient energy to fully insert a control rod at lower vessel pressures. At higher vessel pressures the accumulator pressure is assisted or supplanted by reactor vessel pressure. The accumulator is a hydraulic cylinder with a free-floating piston. The piston separates the water on top from the nitrogen below. A check valve in the accumulator charging water 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 CRD system performs rod insertion, rod withdrawal, and scram. These operational functions are described below.

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

As is illustrated in Figure 4.6-3, the locking mechanism is a ratchet-type device and does not interfere with rod insertion. The speed at which the drive moves is determined by the flow through the insert speed control valve, which is set for approximately 4 gpm for a shim speed (nonscram operation) of 3 in./sec. During normal insertion, the pressure on the downstream side of the speed control valve is 90 psi to 100 psi above reactor vessel pressure. However, if the drive slows for any reason, the flow through, and pressure drop across, the insert speed control valve will decrease; the full differential pressure (260 psi) will then be available to cause continued insertion. With 260 psi differential pressure acting on the drive piston, the piston exerts an upward force of 1040 lb.

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

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

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

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

Prior to a scram signal the accumulator in the HCU has approximately 1450-1510 psig on the water side and 1050-1100 psig on the nitrogen side. As the inlet scram valve opens, the full water-side pressure is available at the CRD acting on a 4.1 in.^{2} area. As CRD motion begins, this pressure drops to the gas-side pressure less line losses between the accumulator and the CRD system; at low vessel pressures the accumulator completely discharges with a resulting gas-side pressure of approximately 575 psi. The CRD accumulators are required to scram the control rods when the reactor pressure is low, and the accumulators retain sufficient stored energy to ensure the complete insertion of the control rods in the required time.

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

The CRD system provides the following performance at full power operation and with accumulators. The scram insertion time is measured from the instant the scram pilot valve solenoids are deenergized.

Position inserted from fully withdrawn	45	39	25	5
Tech Spec scram insertion time (sec)	0.528	0.866	1.917	3.437

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

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

4.6.1.2 Control Rod Drive Housing Supports

4.6.1.2.1 Safety Objective

The CRD housing supports prevent any significant nuclear transient if 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:

a. 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, and

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

4.6.1.2.3 Description

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

Hanger rods, approximately 10 ft long and 1.75 in. in diameter, are supported from the beams on stacks of disc springs. These springs compress approximately 2 in. 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 CRDs, 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 in. at room temperature (approximately 70°F) is provided between the grid and the bottom contact surface of the CRD 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 0.25 in.

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.

For purposes of mechanical design, the postulated failure resulting in the highest forces is an instantaneous circumferential separation of the CRD housing from the reactor vessel, with an internal pressure of 1250 psig (reactor vessel design pressure) acting on the area of the separated housing. The weight of the separated housing, CRD, and blade, plus the pressure of 1250 psig acting on the area of the separated housing, gives a force of approximately 35,000 lb. This force is multiplied by a factor of three for impact, conservatively assuming that the housing travels through a 1-in. gap before it contacts the supports. The total force (105,000 lb) is then treated as a static load in design.

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

4.6.2 EVALUATION OF THE CONTROL ROD DRIVES

Safety evaluation of the control rods, CRDs, and CRD housing supports is described below. Further description of control rods is contained in Section 4.2. The evaluation of the effects of pipe breaks on the CRDs may be found in Section 3.6.

4.6.2.1 Control Rods

4.6.2.1.1 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 type 304 austenitic stainless steel, have been found suitable in meeting the demands of the BWR environment.

4.6.2.1.2 Dimensional and Tolerance Analysis

Layout studies are done to ensure that, given the worst combination of extreme detail part tolerance ranges at assembly, no interference exists which will restrict the passage of control rods.

The italicized information is historical and was provided to support the application for an operating license.

In addition, during initial preoperational testing, an observer who is in direct communication with the control room will observe the operation of each individual control rod and verify that there is no binding or restriction to rod motion and will listen for any scraping or binding noises which may signify rod misalignment. In addition, the function of each CRD line will be measured as indicated by the differential pressure developed across the CRD piston during notch withdrawal. These differential pressure traces will be compared to reference traces to proper operation and the absence of abnormal friction.

4.6.2.1.3 Thermal Analysis of the Tendency to Warp

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

4.6.2.1.4 Forces for Expulsion

An analysis has been performed that evaluates the maximum pressure forces which could tend to eject a control rod from the core. The results of this analysis are given in Section 4.6.2.2.2.2.2. In summary, if the collet were to remain open, which is unlikely, calculations indicate that the steady-state control rod withdrawal velocity would be 2 ft/sec for a pressure-under line break, the limiting case for rod withdrawal.

4.6.2.1.5 Functional Failure of Critical Components

The consequences of a functional failure of critical components have been evaluated and the results are discussed in Section 4.6.2.2.2.

4.6.2.1.6 Precluding Excessive Rates of Reactivity Addition

To preclude excessive rates of reactivity addition, analysis has been performed both on the velocity limiter device and the effect of probable control rod failures (see Section 4.6.2.2.2).

4.6.2.1.7 Effect of Fuel Rod Failure on Control Rod Channel Clearances

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

4.6.2.1.8 Mechanical Damage

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

In addition to the analysis performed on the CRD (see Sections 4.6.2.2.2 and 4.6.2.2.3) and the control rod blade, the following discussion summarizes the analysis performed on the control rod guide tube.

The guide tube can be subjected to any or all of the following loads:

a. Inward load due to pressure differential,
b. Lateral loads due to flow across the guide tube,
c. Dead weight,
d. Seismic (vertical and horizontal), and
e. Vibration.

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

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

4.6.2.1.9 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.

4.6.2.2 Control Rod Drives

4.6.2.2.1 Evaluation of Scram Time

The rod scram function of the CRD system provides the negative reactivity insertion required by safety design basis Section 4.6.1.1.1.1. The scram time shown in the description is adequate as shown by the transient analyses in Chapter 15.

4.6.2.2.2 Analysis of Malfunction Relating to Rod Withdrawal

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

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

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

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

If the basic housing failure were to occur while the control rod is being withdrawn (this is a small fraction of the total drive operating time) and if the collet were to stay unlatched, the following sequence of events is foreseen. The housing would separate from the vessel. The drive and housing would be blown downward against the CRD 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.

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

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

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

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

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

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

4.6.2.2.2.2.3 Simultaneous Breakage of the Pressure-Over and Pressure-Under Lines. For the simultaneous breakage of the pressure-over and pressure-under lines, pressures above and below the drive piston would drop to zero, and the ball check valve would close the broken pressure-under line. Reactor water would flow from the annulus outside the drive, through the vessel ports, and to the space below the drive piston. As in the case of pressure-over line breakage, the drive would then insert (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 atmosphere, as described above. Drive temperature would increase. Indication in the control room would include the drift alarm, the fully inserted drive, and operation of the drywell sump pump.

4.6.2.2.2.3 All Drive Flange Bolts Fail in Tension. Each CRD 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 lb. Capacity of the eight bolts is 121,600 lb. As a result of the reactor design pressure of 1250 psig, the major load on all eight bolts is 30,400 lb.

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

If the collet were latched, control rod ejection would be limited to the distance the drive can drop before coming to rest on the support structure. There would be no tendency for the collet to unlatch, because pressure below the collet piston would drop to zero. Pressure forces, in fact, exert 1435 lb 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-lb return force, would latch and stop rod withdrawal.

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

If the basic flange-to-housing joint failure occurred, the flange and the attached drive would be blown downward against the support structure. The support structure loading would be slightly less than that for drive housing failure because reactor pressure would act only on the drive cross-sectional area. Lack of differential pressure across the collet piston would cause the collet to remain latched and limit control rod motion to approximately 3 in. 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 lb. 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.

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

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

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

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

4.6.2.2.2.6 Flange Plug Blows Out. To connect the vessel ports with the bottom of the ball check valve, a hole of 0.75-in. diameter is drilled in the drive flange. The outer end of this hole is sealed with a plug of 0.812-in. diameter and 0.25-in. thickness. A full-penetration, type 308 stainless steel weld holds the plug in place. The postulated failure is a full circumferential crack in this weld and subsequent blowout of the plug.

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

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.

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

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

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 ft/sec. There would be a large retarding force exerted by the velocity limiter due to a 35 psi pressure differential across the velocity limiter piston.

4.6.2.2.2.8 Drive/Cooling Water Pressure Control Valve Failure. 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 develops a pressure 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 260 psig to no more than 1750 psig. Calculations indicate that the drive would accelerate from 3 in./sec to approximately 6.5 in./sec. A pressure differential of 1670 psi across the collet piston would hold the collet unlatched. Flow would be upward past the velocity limiter piston, but retarding force would be negligible. Rod movement would stop as soon as the driving signal was removed.

Conversely, if the 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 drift 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 rod notch movements impossible but would not affect the ability of the scram function.

In both of the cases described above, the manually operated bypass PCV in conjunction with the isolation gate valves located upstream and downstream of the PCV would enable the operators to take corrective action.

In conclusion, although the failure to the full open or full closed position of the drive/cooling water PCV will cause perturbation in the CRD system operation, it does not present a safety problem to affect the scram capability of the CRD system.

4.6.2.2.2.9 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 lb.

4.6.2.2.2.10 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 SDV 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.

4.6.2.2.2.11 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.

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

4.6.2.2.2.13 Slow or Partial Loss of Air to the Scram Discharge Valves. The CGS IV is adequately hydraulically coupled to the SDV, i.e., the IV is connected directly to the SDV with piping of a diameter equal to or greater than the diameter of the SDV headers. This allows for direct detection of liquid buildup so that the ability to scram is ensured.

The basis of the instrument volume high level scram setpoint and the SDV/IV physical arrangement provides for scram action before significant SDV reduction occurs which could affect scram capability.

The high-level scram setpoint and the SDV/IV system capacity ensure that scram capability is maintained even in the event of maximum inleakage into the SDV prior to a scram. Analysis, assuming the maximum inleakage of 5 gpm and using the actual calculated piston-over area to determine the scram volume requirements, shows that adequate SDV will remain in the SDV system at the time that a scram is initiated.

The partial loss of air pressure does not result in the uncontrolled release of reactor coolant to the reactor building. The vent and drain valves tends are spring to close-held open by air. Flow through the valve tends to close it. As air pressure decreases the valves will begin to close to limit coolant inventory loss. When the water buildup reaches scram initiation level in the IV, a scram signal is produced.

This will cause the air supply to the vent and drain valves to vent, thereby ensuring that the vent and drain valves close and isolate. For leakage rates which do not result in buildup in the IV, the leak will drain to the reactor building equipment drain system. The drain system will alarm for leakage rates greater than 5 gal/minute. The operator can then take appropriate action, e.g., isolate the leak, scram the reactor, increase air pressure, etc., as required.

4.6.2.2.3 Scram Reliability

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

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

b. Each drive mechanism has its own scram and pilot valves so only one drive can be affected if a scram valve fails to open. Two pilot valves are provided for each drive. Both pilot valves must be deenergized to initiate a scram;

c. The RPS and the HCUs are designed so that the scram signal and mode of operation override all others;

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

e. The SDV is monitored for accumulated water and the reactor will scram before the volume is reduced to a point that could interfere with a scram.

4.6.2.2.4 Control Rod Support and Operation

Each control rod is independently supported and controlled as required by the safety design bases.

4.6.2.3 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 0.25 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 reactor coolant 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 0.25 in. exists between the CRD housing and the supports. At lower temperatures the gap is greater. Because the supports do not contact any of the CRD housing except during the postulated accident condition, vertical contact stresses are prevented.

4.6.3 TESTING AND VERIFICATION OF THE CONTROL ROD DRIVES

4.6.3.1 Control Rod Drives

4.6.3.1.1 Testing and Inspection

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

a. The drive easily withstands the forces, pressures, and temperatures imposed;

b. 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;

c. The basic scram speed of the drive has a satisfactory margin above minimum plant requirements at any reactor vessel pressure; and

d. 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 ensure reliable performance of the mechanical reactivity control components. Some of the quality control tests performed on the control rods, CRD mechanisms, and HCU are listed below:

a. Control rod drive mechanism tests

    1. Pressure welds on the drives are hydrostatically tested in accordance with ASME codes;

2. Electrical components are checked for electrical continuity and resistance to ground;

3. 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;

4. Seals are tested for leakage to demonstrate correct seal operation;

5. Each drive is tested for shim motion, latching, and control rod position indication; and

6. Each drive is subjected to cold scram tests at various reactor pressures to verify correct scram performance.

b. Hydraulic control unit tests

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

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

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

4. The unit's ability to perform its part of a scram is demonstrated; and

5. 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 CRD 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 CRD mechanisms and the associated control and protection systems were 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 must be met before the component was shipped.

The after installation, prestartup tests (Section 14.2) 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, time settings, scram valve response times, and control pressures. These tests were intended more to document system condition than as 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 meet the limits of the specifications over the range of primary coolant temperatures and pressures from ambient to operating.

4.6.3.1.1.5 Surveillance Tests. The surveillance requirements for the CRD system are as follows:

a. Prior to each in-vessel fuel movement during fuel loading sequence, the shutdown margin with the highest worth control rod withdrawn shall be analytically determined to be at least 0.38% $\Delta k/k$ or shall be determined by test to be at least 0.28% $\Delta k/k$;

b. Once within 4 hr after criticality following fuel movement within the RPV or control rod replacement, the shutdown margin with the highest worth control rod withdrawn shall be analytically determined to be at least 0.38% $\Delta k/k$ or shall be determined by test to be at least 0.28% $\Delta k/k$;

c. Each withdrawn control rod shall be exercised one notch (i.e., inserted at least one notch and then may be returned to its original position) at least once every 31 days.

The control rod exercise tests serve as a periodic check against deterioration of the control rod system and also verifies the ability of the CRD to scram. If a rod can be moved with drive pressure, it may be expected to scram since higher pressure is applied during scram;

d. The coupling integrity shall be verified for each withdrawn control rod as follows:

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

   2. When the rod is fully withdrawn each 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;

e. During operation, accumulator pressure and level at the normal operating value shall be verified.

   Experience with CRD systems of the same type indicates that weekly verification of accumulator pressure and level is sufficient to ensure operability of the accumulator portion of the CRD system;

f. 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; and

g. A channel functional test of the accumulator leak detectors and a channel calibration of the accumulator pressure detectors, which verifies an alarm setpoint ≥940 psig on decreasing pressure, is performed at least once per 30 months.

4.6.3.1.1.6 Functional Tests. The functional testing program of the CRDs consists of the 5-year maintenance life and the 1.5X design life test programs as described in Section 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:

a. Simulated ruptured scram line test,
b. Stuck ball check valve in CRD flange,
c. HCU drive down inlet flow control valve (V122) failure,
d. HCU drive down outlet flow control valve (V120) failure,
e. CRD scram performance with V120 malfunction,
f. HCU drive up outlet control valve (V121) failure,
g. HCU drive up inlet control valve (V123) failure,
h. Cooling water check valve (V138) leakage,
i. CRD flange check valve leakage,
j. CRD stabilization circuit failure,
k. HCU filter restriction,
l. Air trapped in CRD hydraulic system,
m. CRD collet drop test, and
n. CR qualification velocity limiter drop test.

Additional postulated CRD failures are discussed in Sections 4.6.2.2.2.1 through 4.6.2.2.2.12.

4.6.3.2 Control Rod Drive Housing Supports

CRD housing supports are removed for inspection and maintenance of the CRDs. 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 two reactivity control systems, the CRD and SLC systems, do not share any instrumentation or components. Thus, a common mode failure of the reactivity systems would be limited to an accident event which could damage essential equipment in the two independent systems.

A seismic event or the postulated accident environments (see Section 3.11) are not considered potential common mode failures since the essential (scram) portions of the CRD system are designed to Seismic Category I standards and to operate as required under postulated accident environmental conditions. The SLC system is also designed to Seismic Category I standards.

No common mode power failure is considered credible. The scram function of the CRD system is "fail-safe" on a loss of power and is designed to override any other CRD function. The SLC system has two independent power supplies to its essential redundant pumps and valves. The power supplies to the SLC system are considered vital and as such are switched to the onsite standby diesels on a loss of normal power sources.

Essential components (including cabling and piping) for the SLC system are separated from essential CRD components in the secondary containment by physical barriers and/or by at least 40 ft of physical separation. The various safety studies performed by the architect-engineer verified that this separation is sufficient to prevent simultaneous failure of the reactivity systems due to pipe break and whip, credible fires, and all potential missiles. The location of the primary components of these systems is shown in Figures 1.2-7 through 1.2-12. The CRD insert and withdrawal lines penetrate at the bottom of the RPV whereas the SLC lines connect to the HPCS line which penetrates the RPV. Protection of the reactivity control systems from postulated events, such as pipe breaks, is discussed in Section 3.6.

A fault tree analysis was completed for both of these systems, and the calculated unreliability is less than 10^{-7}/reactor year. This unreliability is an estimate of the failure to fully insert the control rods into the core, combined with a failure to inject boron into the vessel by the SLC. Failure to insert control rods is defined to be noninsertion of the CRDs in the following manner: 50% in a "checkerboard pattern," 31% in a random pattern, or 4% in a cluster.

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

4.6.6 ALTERNATE ROD INSERTION SYSTEM

4.6.6.1 System Description

The alternate rod insertion (ARI) system provides an alternate means to scram the control rods which is diverse and independent from the RPS. The ARI system may be actuated either manually or automatically. The automatic signal to initiate ARI comes from high reactor vessel pressure or low reactor water level. The setpoints for ARI automatic initiation have been chosen such that a normal scram should already have been initiated by the above parameters prior to ARI initiation. The ARI system causes a scram by relieving the scram air header through four sets of solenoid valves. This, in turn, causes the scram inlet and outlet valves to open. The CRD units then insert the control blades to shutdown the reactor.

The ARI system has been designed to ensure that rod motion begins within sufficient time to ensure the ARI design objectives of Reference 4.6-2 are satisfied. These rod movement times are based on plant unique conditions and compliance with ARI design objectives to ensure that plant safety considerations will be met.

4.6.6.2 Alternate Rod Insertion Redundancy

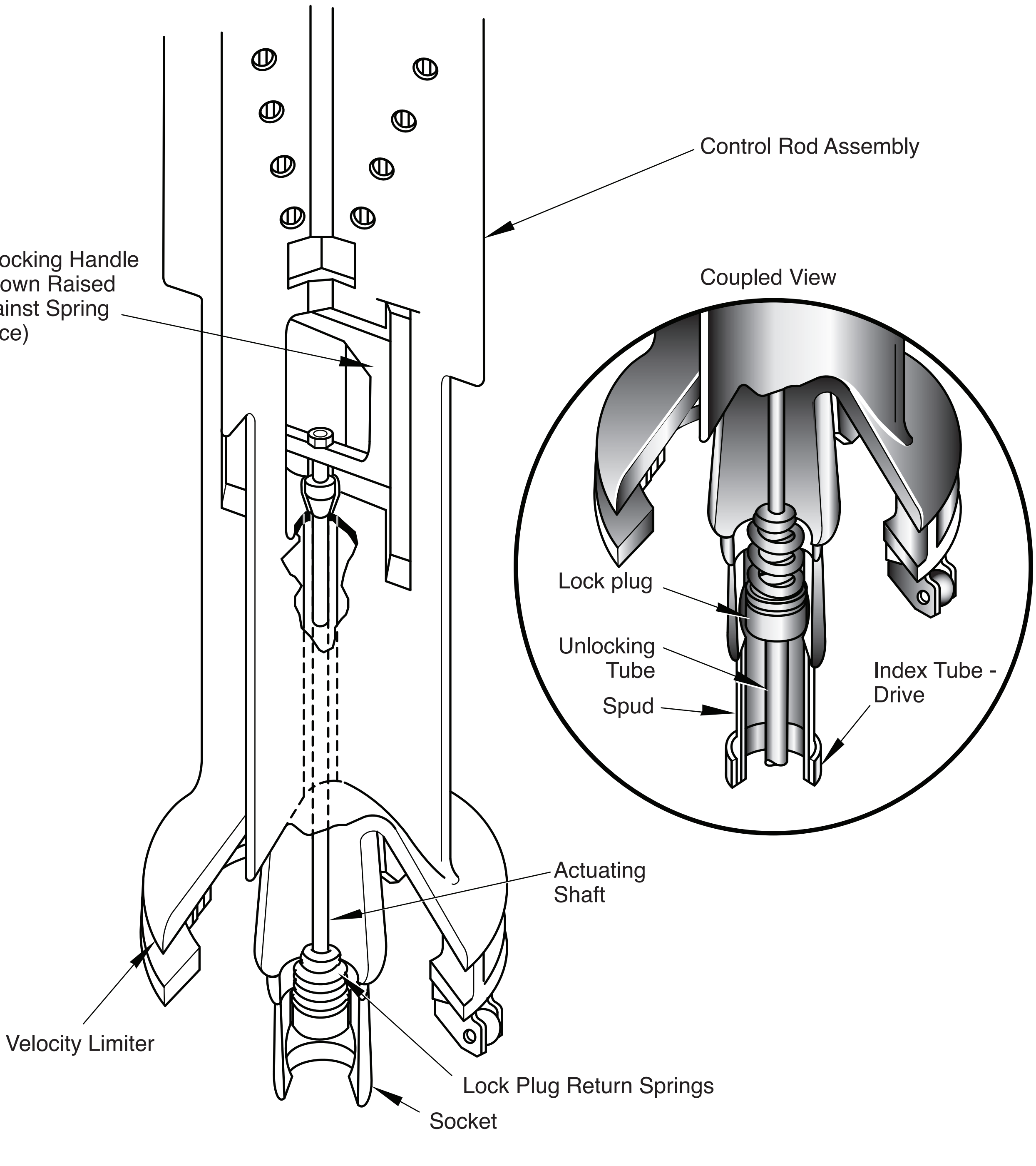
The ARI system constitutes a redundant back-up to the normal scram system and is, therefore, not redundant in itself. That is, the ARI system is only one system with two divisions. Both divisions must function properly for the design basis rod insertion times to be met.

The ARI system is, however, redundant in the aspect of preventing spurious scrams. Each vent point for ARI in the scram air header consists of two valves in series (see Figure 4.6-5). The valves must be energized to vent the air header. This design is intended to prevent spurious scrams and unnecessary cycling of the power plant.

4.6.7 REFERENCES

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

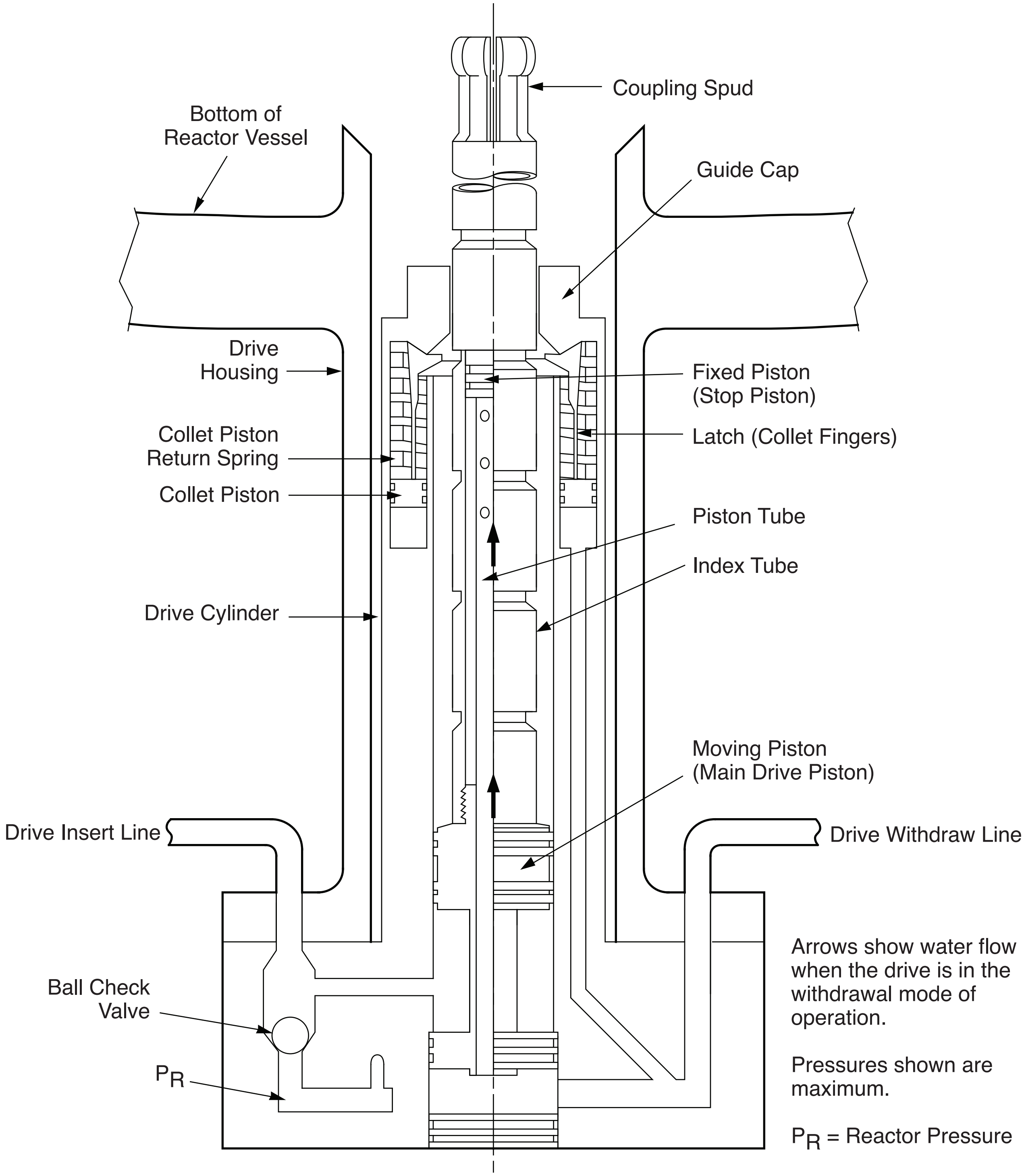
4.6-2 NEDE-31096-P, "Licensing Topical Report, Anticipated Transient Without Scram," Response to NRC ATWS Rule 10 CRF 50.62, February 1987.



Columbia Generating Station
Final Safety Analysis Report

Control Rod to Control Rod Drive Coupling

Draw. No. 960690.99	Rev.	Figure 4.6-1

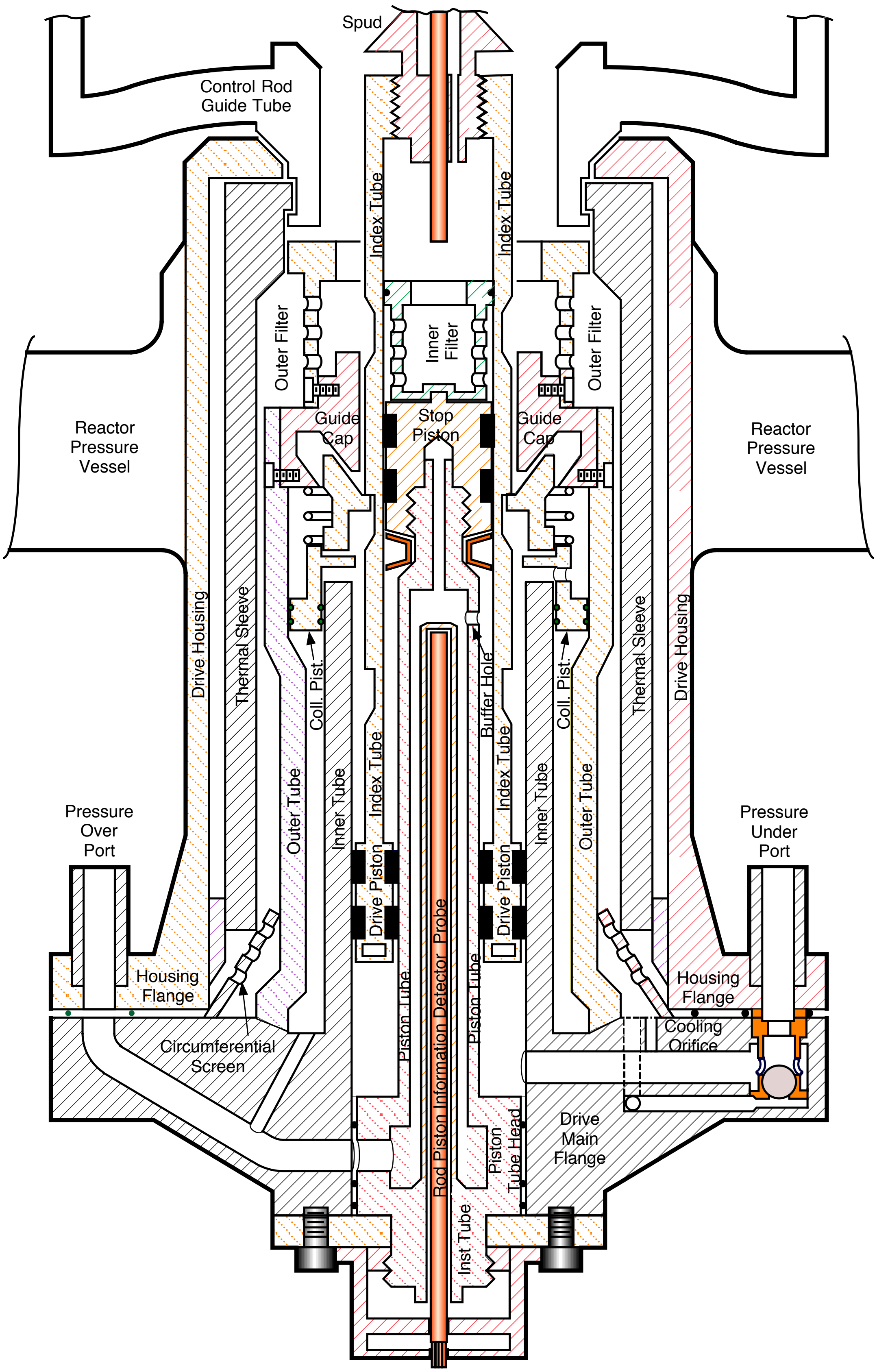


Arrows show water flow when the drive is in the withdrawal mode of operation.

Pressures shown are maximum.

P_R = Reactor Pressure

Columbia Generating Station Final Safety Analysis Report	Control Rod Drive Unit
	Draw. No. 960222.99 Rev. Figure 4.6-2

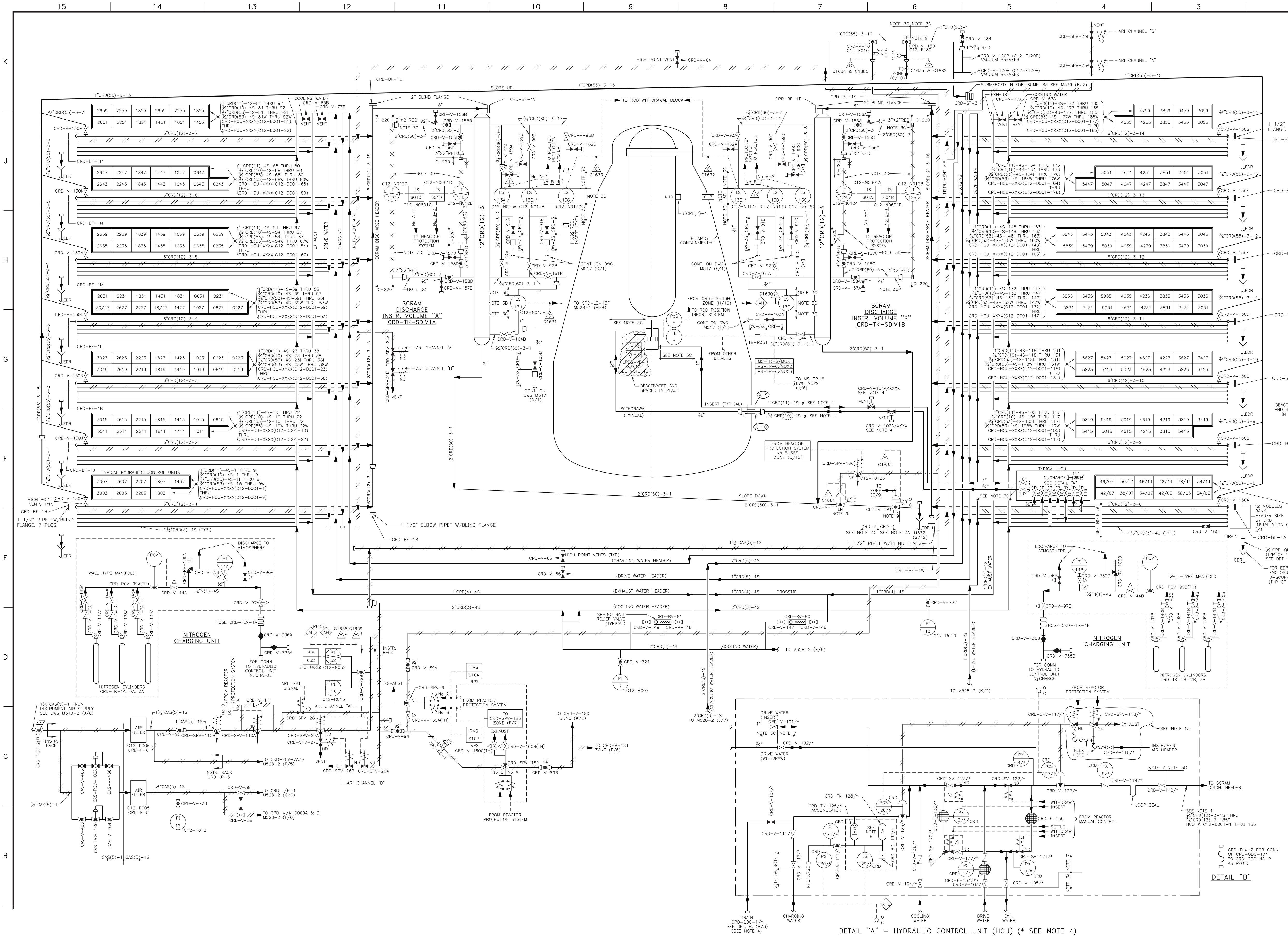


Columbia Generating Station
Final Safety Analysis Report

Control Rod Drive Unit (Schematic)

Draw. No. 960222.96	Rev.	Figure 4.6-3

Figure Not Available For Public Viewing



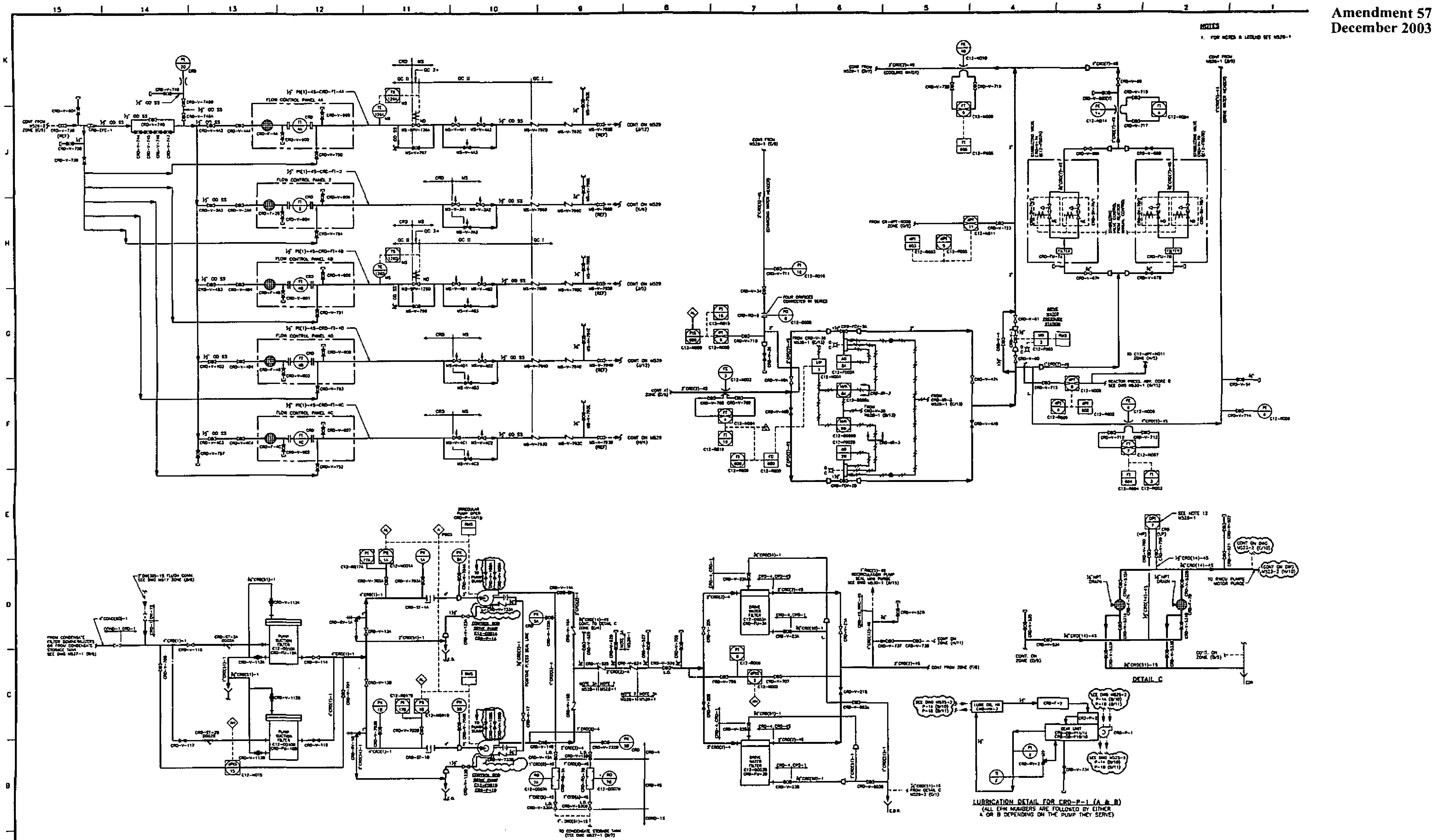
NOTES:

1. ALL PRESSURE AND FLOW INSTRUMENT ROOT VALVES NOT LABELED WILL BE 3/4" GLOBE UNLESS SPECIFICALLY NOTED OTHERWISE.
2. THESE COMPONENTS ARE SUPPLIED WITH THE ASSOCIATED EQUIPMENT
3. PIPING VALVES AND ASSOCIATED COMPONENTS ON THIS DRAWING SHALL BE CLASSIFIED AS FOLLOWS (BREAK POINTS ARE INDICATED ON THE FLOW DIAGRAM)
   - A. ALL PIPING AND VALVES, EXCEPT AS NOTED
     SEISMIC CATEGORY II⊕
     QUALITY CLASS II
     CODE GROUP D
     ⊕HANGERS TO BE DESIGNED TO SEISMIC CATEGORY I LOADS
   - C. PIPING AND VALVES OF SUBSYSTEM (10), (11), (12), (50), (53), (55), AND (60) AS NOTED
     SEISMIC CATEGORY I
     QUALITY CLASS I
     CODE GROUP B
   - D. INSTRUMENTS AS NOTED
     SEISMIC CATEGORY I
     QUALITY CLASS I
     CODE GROUP E
   - E. INSTRUMENT IMPULSE LINES AND VALVES AS NOTED
     SEISMIC CATEGORY I
     QUALITY CLASS I
     CODE GROUP B
4. # UNIQUE LINE NUMBERS 1 THRU 185 ARE DIRECTLY RELATED TO THE HYDRAULIC CONTROL UNIT (HCU) MPL NUMBERS ASSIGNED BY G.E. THE 185 UNIQUE EPN's ARE DENOTED BY "*" (CRD-V-105/XXXX).
5. ALL INSTRUMENTATION ON THIS DRAWING TO BE IDENTIFIED BY THE PREFIX "CRD" UNLESS OTHERWISE NOTED.
6. CODE BREAK DEFINITIONS FOR THERMOWELLS ARE SHOWN ON M610, "INSTALLATION OF THERMOWELLS AND SAMPLE PROBES".
7. HCU PIPING AND VALVES AS NOTED
   SEISMIC CATEGORY I
   QUALITY CLASS I
   CODE GROUP D
8. HCU ACCUMULATORS AND NITROGEN BOTTLES DESIGNED TO ASME VIII STANDARDS
9. MANUAL HANDWHEEL LOCKED IN THE NEUTRAL POSITION.
10. ~~THE LOOSE PART DETECTION SENSORS (LPDS) ARE ON FOUR (4) CRD DRIVE ASSEMBLIES ONLY AS FOLLOWS~~
    ~~LPDS-VE-7 CRD-DRVE-1859~~
    ~~LPDS-VE-8 CRD-DRVE-4259~~
    ~~LPDS-VE-9 CRD-DRVE-4203~~
    ~~LPDS-VE-10 CRD-DRVE-1803~~
    DEACTIVATED AND SPARED IN PLACE
11. THE POSITION DETECTOR AND TEMPERATURE ELEMENT ARE PART OF THE DRIVE UNIT AND DO NOT HAVE SEPARATE EPNs.
12. THIS IS ACTUALLY A DPIS BUT IS USED AS A DPI, WIRES AND SWITCH ARE SPARED IN PLACE.
13. FLEX HOSE IS SHOWN BETWEEN CRD-SPV-117/*, CRD-SPV-118/*, & CRD-V-116/*. COPPER TUBING IS AN ACCEPTABLE ALTERNATIVE.

LEGEND:

1. ALL VALVES SUFFIXED WITH A (TH) DENOTE A THROTTLED VALVE.

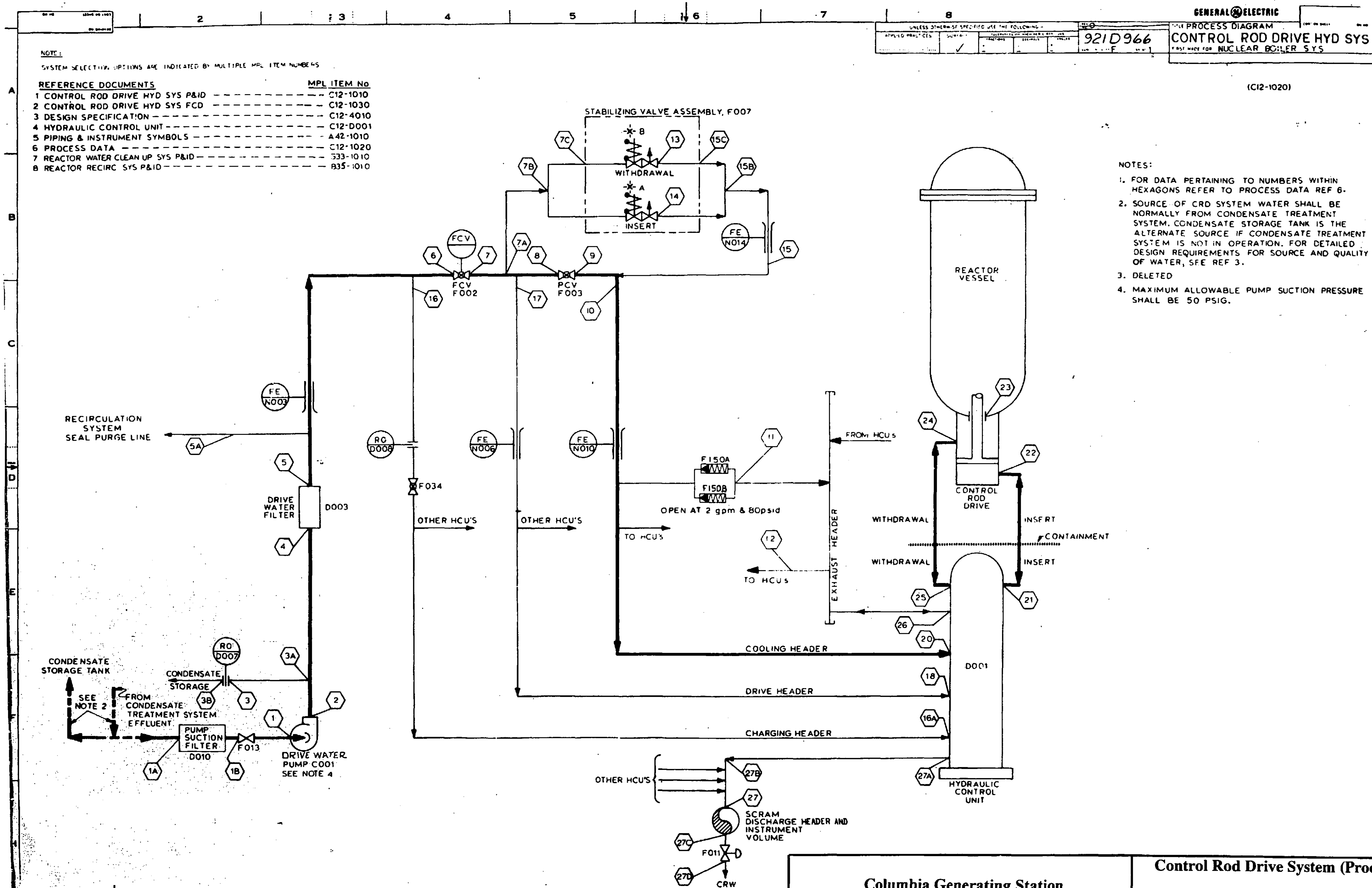
Columbia Generating Station Final Safety Analysis Report	Control Rod Drive Hydraulic System	
	Draw. No. M528-1	Rev. 75
	Figure 04.6-5.1	



Columbia Generating Station
Final Safety Analysis Report

Control Rod Drive Hydraulic System

Draw. No. M528-2 | Rev. 7 | Figure 4.6-5.2



Columbia Generating Station
Final Safety Analysis Report

Control Rod Drive System (Process Diagram)

Draw. No. 02C12-04,25,1	Rev. 1	Figure 4.6-6.1

GENERAL ELECTRIC

PROCESS DIAG DATA
CONTROL ROD DRIVE HYD SYSTEM

921D966AA

MPL NO. C12-1020

MODE A NORMAL OPERATION

LOCATION	1A	1	2	3	4	5	5A	6	7	8	9	10	11	12	13
FLOW . GPM	93	93	93	20	73	73	10	63	63	57	57	63	0	0	2
PRESSURE PSIG	21	19	1487	1487	1476	1462	1462	1455	PR • 260	PR • 260	PR • 30	PR • 30	PR	PR	PR • 30

LOCATION	14	15	16	17	18		20	21	22	23	24	25	26	27	
FLOW . GPM	4	6	0	0	0		.34 MAX	.34 MAX	.34 MAX	.34 MAX	0	0	0	0	
PRESSURE PSIG	PR • 30	PR • 30	1455				PR • 15	PR • 14	PR • 14	PR	PR		PR	0	

CONDITIONS:

1. DRIVES LATCHED
2. PRESSURE OF REACTOR (PR) AT 1000 PSIG.
3. MAXIMUM COOLING FLOW TO DRIVES. MINIMUM REQUIRED PRESSURE AT POSITION 1A IS 20 FEET OF WATER AT 200 GPM.

MODE A SIZES THE COOLING WATER HEADERS.

LINE LOSS FROM LOCATION 10 TO LOCATION 20 SHALL NOT EXCEED 3 PSIG.

MODE B ROD INSERTION

LOCATION	1A	1	2	3	4	5	5A	6	7	8	9	10	11	12	13
FLOW . GPM	93	93	93	20	73	73	10	63	63	57	57	59	0	.7	2
PRESSURE PSIG	21	19	1487	1487	1476	1462	1462	1455	PR • 260	PR • 260	PR • 30	PR • 30	<PR • 8	<PR • 8	PR • 30

LOCATION	14	15	16	17	18		20	21	22	23	24	25	26	27	
FLOW . GPM	0	2	0	4	4		0	4	4	1.3	.7	.7	.7	0	
PRESSURE PSIG	PR • 30	PR • 30	1455	PR • 260	PR • 250		PR • 15	PR • 91	PR • 90	PR	PR • 20 MAX	PR • 20 MAX	PR • 8 MAX	0	

CONDITIONS:

1. DRIVE INSERTING
2. PRESSURE OF REACTOR (PR) AT 1000 PSIG.
3. MAXIMUM DRIVING FLOW TO DRIVES

MODE B SIZES THE DRIVE WATER HEADERS.

MODE C SCRAM

LOCATION	1A	1	2	3	4	5	5A	6	7	8	9	10	11	12	13
FLOW . GPM	45	45	45	20	25	25	10	15	15	15	15	15	15	14.9	0
PRESSURE PSIG	21	21	1550	1550									SEE NOTE 9	SEE NOTE 9	

LOCATION	14	15	16	17	18		20	21	22	23	24	25	26	27	
FLOW . GPM	0	0	0	0	0		0	90	90	-3.6	· 30	30	0.1 • SEE NOTE 9	APPROX 5565	
PRESSURE PSIG								1167 MIN	731 MIN	PR	256 MAX	94		65 MAX	
														SEE NOTE 10	

CONDITIONS:

1. DRIVES SCRAMMING
2. PRESSURE OF REACTOR (PR) AT 1000 PSIG.
3. FLOWS BASED ON MAXIMUM ROD VELOCITY OF 85 INCHES PER SECOND.

MODE C SIZES THE INSERT AND WITHDRAW LINES.

MODE D SCRAM COMPLETED

LOCATION	1A	1	2	3	4	5	5A	6	7	8	9	10	11	12	13
FLOW . GPM	200	200	200	20	180	180	10	15	15	15	15	15	15	14.9	0
PRESSURE PSIG	21	19	1210						>PR	>PR	>PR	>PR	>PR	>PR	

LOCATION	14	15	16	17	18		20	21	22	23	24	25	26	27	
FLOW . GPM	0	0	155	0	0		0	0.92	0.92	0.92	SEE NOTE 9	SEE NOTE 9	0.1	0	
PRESSURE PSIG			988					76	76	PR	65 MAX	65 MAX		65 MAX	
										SEE NOTE 10					

CONDITIONS:

1. SCRAMMING OF DRIVES COMPLETED
2. PRESSURE OF REACTOR (PR) AT 0 PSIG.
3. MAXIMUM CRD SUPPLY PUMP FLOW.

MODE D SIZES THE PUMP SUCTION LINE.

NOTE: MINIMUM ACCUMULATOR PRECHARGE PRESSURE IS 565 PSIG.

Columbia Generating Station
Final Safety Analysis Report

Control Rod Drive System (Process Diagram)

Draw. No. 02C12-04,26,1 | Rev. 1 | Figure 4.6-6.2

Form No. 960690

TABLE I

LOCATION	1A-1B	1B--1	2---6	3A-3B	6--9	7A-7B	7B-7C
DESIGN PRESS. (PSIG.)	150	150	1750	1750	1750	1750	1750
DESIGN TEMP. (DEG F)	150	150	150	150	150	150	150
ESTIMATED LINE SIZE (INCHES)	4	4	2	1	1.5	1	0.75

LOCATION	10-20	11-12	15B-15C	15-15B	16-16A	17-18	12-26
DESIGN PRESS. (PSIG.)	1750	1750	1750	1750	1750	1750	1750
DESIGN TEMP. (DEG F)	150	150	150	150	150	150	150
ESTIMATED LINE SIZE (INCHES)	2**	1	0.75	1	2	1	1

LOCATION	21-22 "SEE NOTE 13"	24-25 "SEE NOTE 13"	27A-27B "SEE NOTE 13"	27B-27 "SEE NOTE 13"	27-27C "SEE NOTE 14"	27C-27D "SEE NOTE 14"	5A
DESIGN PRESS. (PSIG.)	1750	1750	1250	1250	1250	1250	1750
DESIGN TEMP. (DEG F)	150	500 (PEAK)	450 (PEAK)	450 (PEAK)	450 (PEAK)	450 PEAK)	150
ESTIMATED LINE SIZE (INCHES)	1	0.75	0.75	*	10	2	.75

* SEE CRD SYSTEM DESIGN SPECIFICATION.

** 2 INCH HEADER TO EACH HALF OF THE TOTAL QUANTITY OF HCU'S.

NOTES:

1. DEFINITION OF SYMBOLS
   PR - INDICATES PRESSURE OF THE REACTOR
2. MAXIMUM OPERATING TEMPERATURES
   THE MAXIMUM SYSTEM OPERATING TEMPERATURE WILL NOT EXCEED 150 DEG. F. FROM LOCATION 1 THROUGH 27 WITH THE FOLLOWING EXCEPTIONS.

	LOCATION	MAXIMUM TEMP. (DEG. F.)
MODE A -	23	200
MODE A -	23	546
MODE A - (LEAKING SCRAM DISCHARGE VALVE)	24	500
	25	500
	27	280
MODE D -	23	475
	24	475
	25	475
	27	450

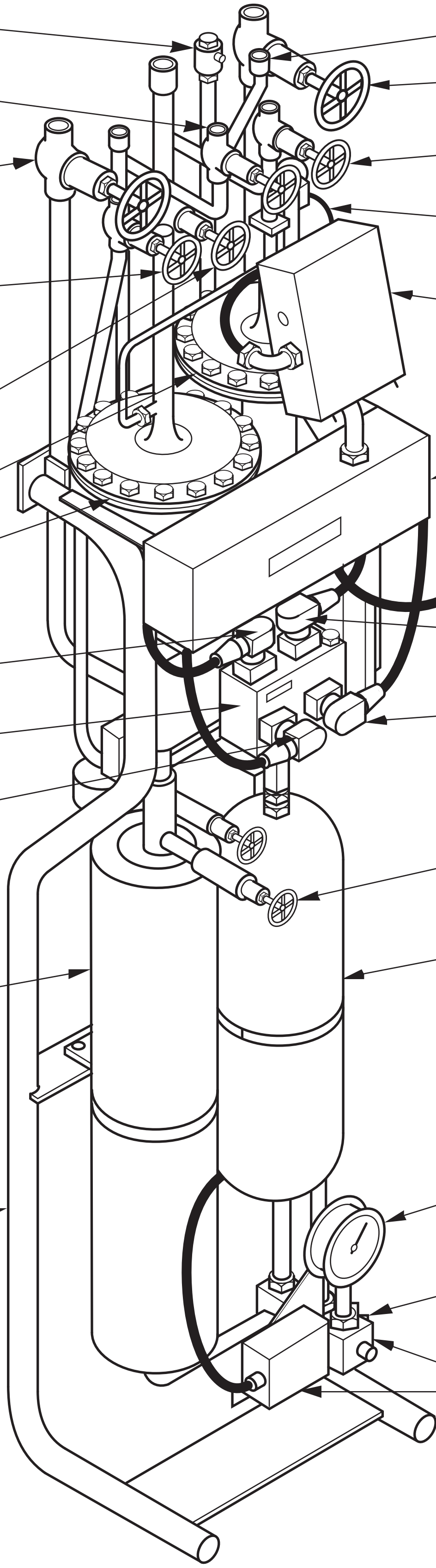
3. MODE A -
   A. MAXIMUM CHARGING WATER PRESSURE SHALL BE 1600 PSIG NOMINAL, ACCUMULATOR PRECHARGE PRESSURE SHALL BE 575 PSIG NOMINAL, 580 PSIG MAXIMUM, AT 70° F.
   B. DELETED
   C. LOCATION 20 - THE CRD COOLING WATER PRESSURE SHALL BE DETERMINED BY THE COOLING WATER FLOW.
   D. LOCATION 23 - MAXIMUM DRIVE COOLING REQUIREMENTS WILL NOT EXCEED 0.34 GPM/DRIVE FOR THE CONDITIONS LISTED. MINIMUM DRIVE COOLING REQUIREMENTS WILL NOT BE LESS THAN 0.20 GPM/DRIVE.
4. MODE B -
   A. LOCATION 13 AND 14 - INSERT VALVE F007-A CLOSES ON DRIVE INSERT SIGNAL. WITHDRAW VALVE F007-B CLOSES ON DRIVE WITHDRAW SIGNAL BUT DOES NOT STAY CLOSED DURING SETTLING.
   B. LOCATION 18 - THE CRD DRIVE WATER PRESSURE SHALL NOT BE LESS THAN PR + 250 PSIG. FOR THE CONDITIONS INDICATED.
5. MODE C -
   A. DELETED
   B. THE TEMPERATURES LISTED IN NOTE 2 FOR POSITION 24, 25 AND 27 MAY BE ASSUMED TO OCCUR LESS THAN 1 PERCENT OF THE OPERATING LIFE OF THE SYSTEM. SEE THE CRD HYDRAULIC SYSTEM DESIGN SPECIFICATION TO DETERMINE CYCLIC STRESS DUE TO THERMAL EXPANSION.
   C. LOCATION 21 TO 22 - THE PRESSURE DROP FROM LOCATION 21 TO 22 SHALL NOT EXCEED 435 PSI AT 90 GPM FOR ANY CRD.
   D. LOCATION 23 - A NEGATIVE FLOW RATE INDICATES FLOW FROM THE REACTOR THROUGH THE DRIVE SEAL, INTO THE CRD. THE MAXIMUM LEAK RATE FROM THE THE REACTOR CAN REACH 10 GPM PER DRIVE.
   E. LOCATION 24 TO 25 - THE PRESSURE DROP FROM LOCATION 24 TO 25 SHALL NOT EXCEED 162 PSI AT 30 GPM FOR ANY CRD.
   F. RESPONSE TIME OF FCV-F002 IS SUCH THAT SCRAM IS COMPLETED BEFORE FCV-F002 STARTS TO CLOSE.
   G. SCRAM DRAIN VALVE F011 AND VENT VALVE F010 CLOSE WITH A SCRAM SIGNAL.
6. MODE D -
   A. DELETED
   B. LOCATION 27 - THE SCRAM DISCHARGE VOLUME SHALL BE SIZED SO THAT THE RESULTING PRESSURE AFTER 100 PERCENT STROKE IS LESS THAN 65 PSIG.
7. MAXIMUM ALLOWABLE PUMP SUCTION PRESSURE SHALL BE 50 PSIG.
8. PROCESS DIAGRAM 921D966 SHALL BE USED WITH AND FORM PART OF THIS PROCESS DATA. IF THERE ARE ANY CONFLICTS BETWEEN THE PROCESS DIAGRAM AND THIS PROCESS DATA, THE PROCESS DATA SHALL GOVERN.
9. DURING SCRAM, THIS FLOW WILL BE DIRECTED INTO THE SCRAM DISCHARGE VOLUME. FOLLOWING SCRAM, THIS FLOW WILL DECLINE AS VALVE F002 CLOSES AND AS THE SCRAM DISCHARGE VOLUME PRESSURIZES TO EQUAL THE REACTOR PRESSURE. AFTER THE SCRAM DISCHARGE VOLUME AND THE REACTOR VESSEL PRESSURE HAVE EQUALIZED, FLOW WILL BE DIVERTED TO THE REACTOR VESSEL VIA THE CRD WITHDRAW LINES AT A FLOW RATE DEPENDENT ON THE REACTOR PRESSURE:
   I.E. (A.) APPROX. 15 GPM AT "0" PSIG. REACTOR PRESSURE.
   (B.) APPROX. 6 GPM AT "1000" PSIG. REACTOR PRESSURE.
10. THIS VALUE APPLIES IMMEDIATELY FOLLOWING COMPLETION OF SCRAM. PRESSURE WILL SUBSEQUENTLY EQUALIZE WITH REACTOR PRESSURE.
11. DESIGN PRESSURE AND TEMPERATURE SHOWN IN "TABLE I" IS FOR INFORMATION ONLY AND IS THE BASIS FOR DESIGN OF BWRS SUPPLIED EQUIPMENT. ESTIMATED LINE SIZES ARE FOR INFORMATION ONLY. ACTUAL LINE SIZES AS DETERMINED BY THE PIPING DESIGNER SHALL MEET THE PROCESS DATA HYDRAULIC REQUIREMENTS.
12. ALL VALUES SHOWN IN MODES A, B, C, AND D ARE NOMINAL UNLESS-OTHERWISE NOTED.
13. INSERT AND WITHDRAWAL PIPING SHALL BE DESIGNED FOR HYDRODYNAMIC LOADS AS A RESULT OF A NORMAL SCRAM AT ZERO & NORMAL REACTOR PRESSURES SHORT STROKE AND FULL STROKE SCRAM AND A SCRAM WITH FAILED CRD BUFFER. PLANT LOAD COMBINATIONS SHOULD INCLUDE CONSIDERATION OF THOSE SYSTEM HYDRODYNAMIC LOADS.
14. THE SCRAM DISCHARGE VOLUME (SDV) AND ITS VENT AND DRAIN PIPING DESIGN SHALL CONSIDER THE HYDRODYNAMIC LOADS WHICH MAY OCCUR DUE TO 1) SDV ISOLATION AND 2) SDV VENTING AND DRAINING FOLLOWING A SCRAM COMPLETION AT REACTOR OPERATING PRESSURE.

Columbia Generating Station
Final Safety Analysis Report

Control Rod Drive System (Process Diagram)

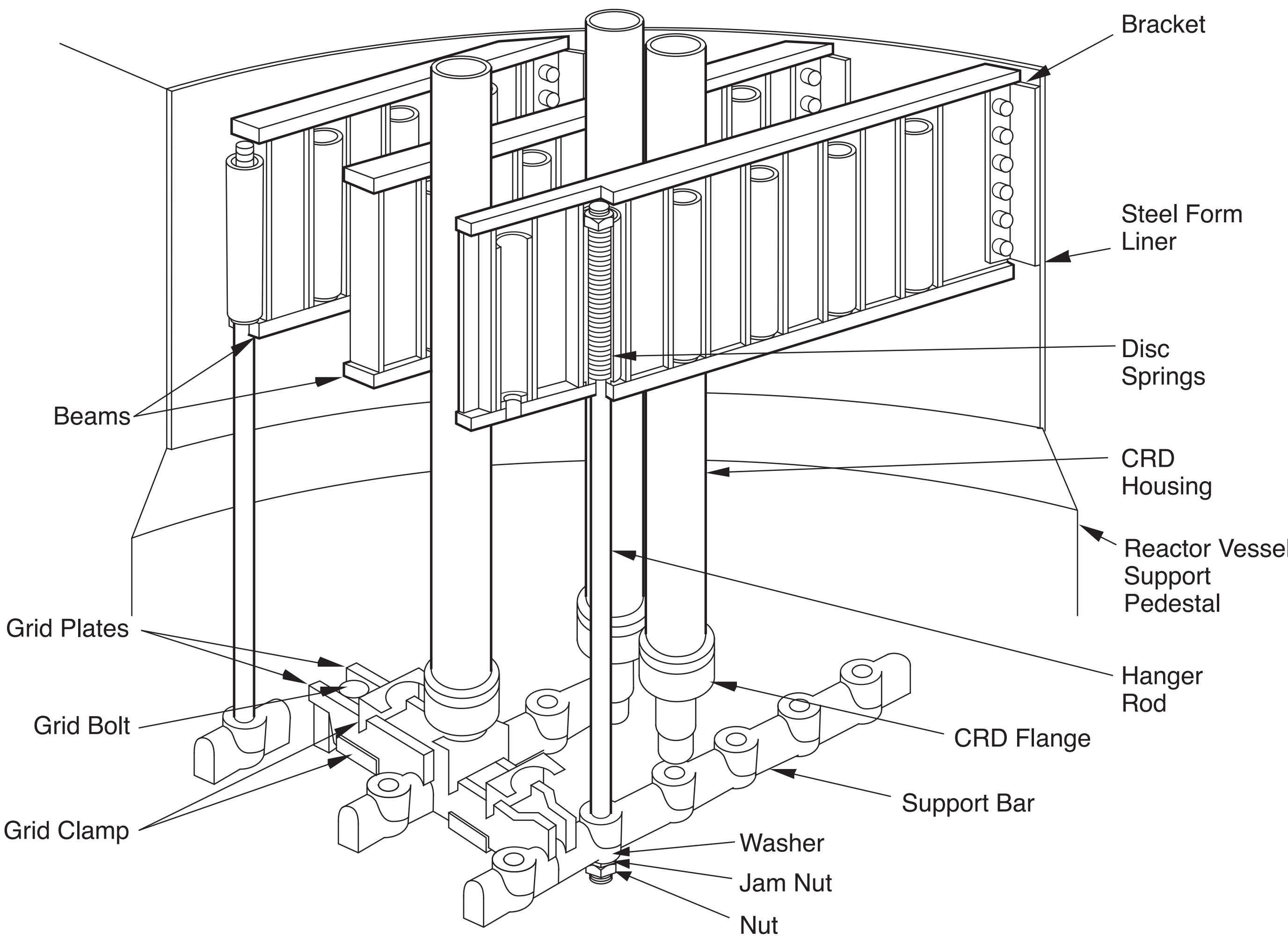
Draw. No. 02C12-04,26,2 | Rev. 2 | Figure 4.6-6.3

Amendment 53
November 1998



Columbia Generating Station Final Safety Analysis Report	Control Rod Drive Hydraulic Control Unit		
	Draw. No. 900547.25	Rev.	Figure 4.6-7

Form No. 960690



Columbia Generating Station
Final Safety Analysis Report

Control Rod Drive Housing Support

Draw. No. 900547.26 | Rev. | Figure 4.6-8