TABLE OF CONTENTS

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

LIST OF TABLES

4.0 REACTOR

This chapter was prepared using the latest approved version of the licensing topical report “General Electric Standard Application for Reactor Fuel” (GESTAR) NEDE-24011-P-A including the “United States Supplement,” NEDE-24011-P-A-US. Applicable sections of this report are referenced as noted in <Section 4.1>, <Section 4.2>, <Section 4.3>, and <Section 4.4>. Reference is made to standardized information contained in the topical report, consistent with the NRC overall standardization philosophy.

<Appendix 15B>, Reload Safety Analysis provides a summary description of the fuel designs, corresponding nuclear and thermal-hydraulic characteristics, stability considerations, etc. for the current cycle reload core.

4.1 SUMMARY DESCRIPTION

The reactor assembly consists of the reactor vessel, its internal components of the core, shroud, steam separator and dryer assemblies, and jet pumps. Also included in the reactor assembly are the control rods, control rod drive housing and the control rod drives. <Figure 3.9-19>, Reactor Vessel Cutaway, shows the arrangement of reactor assembly components. A summary of the important design and performance characteristics is given in <Section 1.3.1.1>. Loading conditions for reactor assembly components are specified in <Section 3.9>.

4.1.1 REACTOR VESSEL

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

4.1.2 REACTOR INTERNAL COMPONENTS

The major reactor internal components are the core (fuel, channels, control 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, and the core spray lines.

The steam dryers, shroud head and steam separators, fuel assemblies, in-core assemblies, control rods, orificed fuel supports, feedwater spargers, core spray spargers, and control rod guide tubes, can be removed.

4.1.2.1 Reactor Core

4.1.2.1.1 General

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

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

a. The BWR core mechanical design is based on conservative application of stress limits, operating experience and experimental test results. The moderate pressure level characteristic of a direct cycle reactor (approximately 1,000 psia) results 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 General Electric thermal analysis basis, GETAB, is applied to assure that more than 99.9% of the fuel rods in the core are expected to avoid boiling transition for the most severe moderate frequency per <Regulatory Guide 1.70>, (Revision 3) transient described in <Chapter 15>. The possibility of boiling transition occurring during normal reactor operation is insignificant.

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, in order to follow load.

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

Important features of the reactor core arrangement are as follows:

a. The bottom-entry cruciform control rod designs consist of several absorber tubes filled with neutron absorbing material such as B_4C and/or hafnium.

Control rods typical of the Original Equipment control rod design have been irradiated for more than eight years in the Dresden-1 reactor and have accumulated thousands of hours of service without significant failure in operating BWRs.

The lead Marathon control rod was loaded in Oyster Creek in November 1988. Inspection after one cycle indicated the integrity of the overall assembly as well as that of the absorber tubes and welds was maintained.

b. The fixed in-core fission chambers provide continuous power range neutron flux monitoring. A guide tube in each in-core assembly provides for a traversing ion chamber for calibration and axial detail. Source and intermediate range detectors are located in-core and are axially retractable. The in-core location of the source and intermediate range instruments provides coverage of the large reactor core and provides an acceptable signal-to-noise ratio and neutron-to-gamma ratio. All in-core instrument leads enter from the bottom and the instruments are in service during refueling. In-core instrumentation is discussed in <Section 7.6>.

c. As shown by experience obtained at Dresden-1 and other plants, the operator, utilizing the in-core flux monitor system, can maintain the desired power distribution within a large core by proper control rod scheduling.

d. The Zircaloy 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 of the assembly.

e. 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.

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

4.1.2.1.2 Core Configuration

The reactor core is arranged as an upright circular cylinder containing a large number of fuel cells and is located within the reactor vessel. The coolant flows upward through the core. The core arrangement (plan view) and the lattice configuration are described in <Section 4.3>.

4.1.2.1.3 Fuel Assembly Description

The boiling water reactor core is composed of essentially two components--fuel assemblies and control rods. The fuel assembly <Section 4.2> and control rod mechanical configurations <Figure 4.2-1>, <Figure 4.2-2>, and <Figure 4.2-3>, are basically the same as used in Dresden-1 and in all subsequent General Electric boiling water reactors. A description of the fuel assembly including fuel rods, water rods, other fuel assembly components, and channels are given in <Section 4.2> which references Section 2.1 of GESTAR (Reference 5). A discussion of the fuel designs utilized for the current cycle is contained in <Appendix 15B>, Reload Safety Analysis. A general description of the fuel rods and bundle is given below.

4.1.2.1.3.1 Fuel Rod

A fuel rod consists of UO_2 pellets and a Zircaloy cladding tube. Barrier fuel bundles consist of fuel rods with a thin, high purity zirconium liner, i.e., barrier, mechanically bonded to the cladding tube. A fuel rod is made by stacking pellets into the Zircaloy cladding tube which is evacuated, back-filled with helium and sealed by welding

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

4.1.2.1.3.2 Fuel Bundle

Each fuel bundle contains fuel rods and water rods which are spaced and supported in a square (nxn) array by spacers and a lower and upper tie plate. Fuel bundle design descriptions are contained in GESTAR (Reference 5). 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.

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

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

4.1.2.1.4 Assembly Support and Control Rod Location

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

The top guide, mounted on top of the shroud, provides lateral support and guidance for the top of each fuel assembly. The reactivity of the core is controlled by cruciform control rods and their associated mechanical hydraulic drive system. The control rods occupy alternate spaces between fuel assemblies. Each independent drive enters the core from the bottom, and can accurately position its associated control rod during normal operation and yet exert approximately ten times the force of gravity to insert the control rod during the scram mode of operation.

4.1.2.2 Shroud

The information on the shroud is contained in <Section 3.9.5.1>.

4.1.2.3 Shroud Head and Steam Separators

The information on the shroud head and steam separators is contained in <Section 3.9.5.1>.

4.1.2.4 Steam Dryer Assembly

The information on the steam dryer assembly is contained in <Section 3.9.5.1>.

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.

The reactivity control function requires that all rods be available for either reactor scram (prompt shutdown) or reactivity regulation. Because of this, the control elements are mechanically designed to withstand the dynamic forces resulting from a scram. They are connected to bottom-mounted, hydraulically actuated drive mechanisms which allow either axial positioning for reactivity regulation or rapid scram insertion. The design of the rod-to-drive connection permits each blade to be attached or detached from its drive without disturbing the remainder of the control system. The bottom-mounted drives permit the entire control system to be left intact and operable for tests with the reactor vessel open.

4.1.3.2 Description of Control Rods

The description for control rod assembly designs applicable to Perry are given in <Section 4.2.2.1>.

4.1.3.3 Supplementary Reactivity Control

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

4.1.4 ANALYSIS TECHNIQUES

4.1.4.1 Reactor Internal Components

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

a. MASS

b. DYSEA

c. FAP-71

d. ANSYS

Detailed descriptions of these programs are given in the sections that follow.

4.1.4.1.1 MASS (Mechanical Analysis of Space Structure)

4.1.4.1.1.1 Program Description

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

4.1.4.1.1.2 (Deleted)

4.1.4.1.1.3 (Deleted)

4.1.4.1.1.4 (Deleted)

4.1.4.1.2 DYSEA

4.1.4.1.2.1 Program Description

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

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

4.1.4.1.2.2 (Deleted)

4.1.4.1.2.3 (Deleted)

4.1.4.1.2.4 (Deleted)

4.1.4.1.3 FAP-71 (Fatigue Analysis Program)

4.1.4.1.3.1 Program Description

The FAP-71 computer code, or Fatigue Analysis Program (Reference 3), 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, or (2) the present method documented in Paragraph NB-3228.3 of the 1971 Edition of the ASME Section III Nuclear Vessel Code. The program can accommodate up to 25 transient stress states of as many as 20 structural locations.

4.1.4.1.3.2 (Deleted)

4.1.4.1.3.3 (Deleted)

4.1.4.1.3.4 (Deleted)

4.1.4.1.4 ANSYS

4.1.4.1.4.1 Program Description

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

The ANSYS program features the following capabilities:

a. Structural analysis including static, elastic, plastic and creep, dynamic, seismic and dynamic plastic, and large deflection and stability analysis.

b. One-dimensional fluid flow analyses.

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

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

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

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

4.1.4.1.4.2 (Deleted)

4.1.4.1.4.3 (Deleted)

4.1.4.1.4.4 (Deleted)

4.1.4.2 Fuel Rod Thermal Analysis

Fuel rod thermal analyses are described in Section 2 of GESTAR II (Reference 5).

4.1.4.3 Reactor Systems Dynamics

The analysis techniques and computer codes used in reactor systems dynamics are described in Section 4 (Reference 4).

4.1.4.4 Nuclear Engineering Analysis

The analysis techniques are described in Subsection A.4.3.3 of GESTAR II (Reference 5). The codes used in the analysis are:

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

4.1.4.5 Neutron Fluence Calculations

Neutron flux at the reactor vessel ID was calculated using the transport codes and assumptions described below.

4.1.4.5.1 Unit 1 Neutron Fluence Calculations

Unit 1 neutron vessel fluence calculations were performed with DORT, which is the two-dimensional module of the TORT (Reference 6) three-dimensional, discrete ordinates, Sn transport code system. This code will solve a wide variety of radiation transport problems including both fixed source and multiplication problems. Slab, cylinder, and spherical geometries are allowed with various boundary conditions.

The fluence calculations incorporate, as an initial starting point, a neutron source distribution prepared from core power distribution data. Anisotropic scattering was considered for all regions. The cross sections were prepared with 1/E flux weighted, P_L matrices for anisotropic scattering. A two-dimensional transport calculation in (R,θ) coordinates was performed to obtain fast neutron fluxes at core midplane. Fast neutron fluxes at locations other than the core mid-plane were calculated using a second two-dimensional calculation in (R,Z) coordinates.

The fast neutron flux calculations are used to establish the lead factor, which is the ratio of flux between the surveillance capsule locations and the location of peak vessel inside surface flux. Use of the lead factor is discussed in <Section 4.3.2.8.1>.

4.1.4.5.2 (Deleted)

4.1.4.6 Thermal Hydraulic Calculations

The digital computer program uses a parallel flow path model to perform the steady-state BWR reactor core thermal-hydraulic analysis. Program input includes the core geometry, operating power, pressure, coolant flow rate and inlet enthalpy, and power distribution within the core. Output from the program includes core pressure drop, coolant flow distribution, critical power ratio, and axial variations of quality, density and enthalpy for each channel type. A description of the thermal-hydraulic models is given in Section 4 of GESTAR (Reference 5).

4.1.5 REFERENCES FOR SECTION 4.1

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

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

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

4. Carmichael, L. A. and Scatena, G. J., "Stability and Dynamic Performance of the General Electric Boiling Water Reactor," APED-5652.

5. General Electric Company “General Electric Standard Application for Reactor Fuel” including the “United States Supplement,” NEDE-24011-P-A and NEDE-24011-P-A-US, (latest approved revision).

6. CCC-543, “TORT-DORT Two- and Three- Dimensional Discrete Ordinates Transport Version 2.8.14” Radiation Shielding Information Center, Oak Ridge National Laboratory.

4.2 FUEL SYSTEM DESIGN

The format of this section corresponds to Standard Review Plan Section 4.2 in <NUREG-0800>. Most of the information is presented by reference to the licensing topical report GESTAR (Reference 1). The subsection numbers in <Section 4.2> generally correspond to the subsection numbers of Appendix A of GESTAR. Any additional information or differences are given for the applicable subsection.

<Appendix 15B>, Reload Safety Analysis provides summary information on the fuel system design.

4.2.1 DESIGN BASES

Information on fuel system design bases is provided in (Reference 1) (Subsection A.4.2.1).

4.2.2 DESCRIPTION AND DESIGN DRAWINGS

Information on fuel system description and design drawings is provided in (Reference 12) and (Reference 13) for the GE14 fuel product line and in (Reference 14) for the GNF2 fuel product line, except for the reactivity control assembly description, which is described below.

4.2.2.1 Reactivity Control Assembly

The main structural member of a control rod is made of stainless steel and consists of a top handle, a bottom casting with a velocity limiter and a control rod drive coupling, and four wings attached to a vertical cruciform center post. The top handle, bottom casting and center post are welded into a single skeletal structure. The top handle provides structural rigidity at the top of the control rod. The bottom casting also provides for structural rigidity and contains positioning rollers and a parachute-shaped velocity limiter. Rollers, housed in the top handle and bottom casting of the control rod, provide guidance for control rod insertion and withdrawal. Marathon Control Rods supplied after 2007 do not contain rollers in the top hands.

The control rods are separated uniformly throughout the core on a 12-inch pitch maximum. Each control rod is surrounded by four fuel assemblies. The control rods can be positioned at 6-inch steps and have a nominal withdrawal and insertion speed of 3 in./sec.

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

a. The area between the fuel channel and the fuel assembly lower tie plate;

b. Holes in the lower tie plate;

c. The area between the fuel assembly lower tie plate and the fuel support piece;

d. The area between the fuel support piece and the control rod guide tube;

e. The area between the control rod guide tube and the core support plate; and

f. The area between the core support plate and the shroud.

4.2.2.1.1 Original Equipment Control Rods

The original equipment control rod consists of a sheathed cruciform array of 72 Type-304 stainless steel absorber tubes (18 tubes in each wing of the cruciform) filled with vibration compacted boron carbide powder shown in <Figure 4.2-1>. The boron carbide (B_4C) powder in the absorber tubes is compacted to about 70 percent of its theoretical density. The B_4C contains a minimum of 76.5 percent by weight natural boron. The Boron-10 (B-10) minimum content of the boron is 18 percent by weight. The top handle aligns the absorber tubes which are seal welded with end plugs on either end. Each absorber tube is 0.22 inch outside diameter and has a 0.027 inch wall thickness. The B_4C is longitudinally separated into individual compartments by stainless steel balls at approximately 17-inch intervals. The balls are held in position by a slight crimp in the tube. The individual tubes provide containment of the helium gas released by the boron-neutron capture reaction. Should the B_4C compact further in service, the steel balls will distribute the resulting voids over the length of the absorber.

The absorber tubes are held in a cruciform array by a stainless steel U-shaped sheath extending the full length of the tubes. The 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.

4.2.2.1.2 Marathon Control Rods

The General Electric Marathon control rod consists of a cruciform array of externally-square absorber tubes that are welded full length to each other to form a straight line array called a wing, shown in <Figure 4.2-4>. Each wing is comprised of 14 absorber tubes with each tube acting as an individual chamber to hold the helium released from the boron carbide (B_4C). The four wings are welded to tie rod segments to form the cruciform-shaped member of the control rod. The square absorber tubes are circular inside and are loaded with either B_4C capsules or hafnium metal rods. The B_4C powder is compacted to about 70 percent of its theoretical density into thin-walled, stainless steel capsules with stainless steel end caps to prevent B_4C migration and to allow helium release from the capsules into the absorber tube. The capsules are smaller than the absorber tube inside diameter, allowing B_4C to swell before it makes contact with the absorber tube to prevent excessive absorber tube strains. The B_4C contains a minimum of 76.5 percent by weight natural boron. The Boron-10 (B-10) minimum content of the boron is 18 percent by weight. The B_4C capsules are either 3.00 or 11.41 inch minimum length.

Two hafnium rods each 71.40 inch minimum length are located in the three edge absorber tubes. Hafnium does not emit gases during its depletion. However, hafnium has demonstrated swelling due to hydriding when clad with stainless steel resulting in high strains and cracking of the control rods. The hafnium metal is sized smaller than the absorber tube inside diameter to accommodate the swelling and to prevent excessive absorber tube strains.

The Marathon design uses an enhanced grade of high purity Type-304 stainless steel referred to as RAD RESIST 304S which provides a high resistance to irradiation-assisted corrosion cracking. Niobium and Tantalum are added to provide greater protection against stress corrosion cracking. Material hardening characteristics are the same as the Type-304 stainless steel used in previously approved designs.

The mechanical design for the GE Marathon control rods are described in (Reference 8) and accepted by the NRC for licensing applications in the Safety Evaluation Report in (Reference 8).

The GE Marathon Ultra HD control rods are considered to be direct replacements for the GE Marathon control rods, with respect to fit, form and function, from both a mechanical and nuclear perspective. The minor differences between the GE Marathon control rods and the GE Marathon Ultra HD control rods are shown in Table 2-1 of (Reference 10). These differences are justified as acceptable and as having a negligible impact on the function of the control rods. The mechanical design for the GE Marathon Ultra HD control rods are described in (Reference 10) and accepted by the NRC for licensing applications in the Safety Evaluation Report in (Reference 10).

An additional GE-provided equivalency evaluation comparing the GE Marathon Ultra HD control rods to the GE Marathon control rods is contained in (Reference 11).

4.2.2.1.3 Velocity Limiter

The control rod velocity limiter <Figure 4.2-3> is an integral part of the bottom assembly of each control rod. This engineered safeguard protects against high reactivity insertion rate by limiting the control rod free fall velocity in the event of a control rod drop accident. It is a one-way device in that the control rod scram velocity (control rod scram time) 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 and is at a 15 degree angle relative to the upper conical element, with the peripheral separation less than the central separation.

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

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

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

4.2.3 DESIGN EVALUATIONS

Information on the fuel system evaluation for compliance with the design bases is provided in (Reference 1) (Subsection A.4.2.3).

4.2.4 TESTING, INSPECTION AND SURVEILLANCE PLANS

Information on testing, inspection and surveillance is provided in (Reference 1) (Subsection A.4.2.4). Fuel assembly surveillance plans are further described in (Reference 2), (Reference 3), (Reference 4), and (Reference 5).

4.2.5 OPERATING AND DEVELOPMENTAL EXPERIENCE

For a discussion of fuel experience, see (Reference 6) and (Reference 7).

4.2.6 REFERENCES FOR SECTION 4.2

1. Global Nuclear Fuel "General Electric Standard Application for Reactor Fuel," including the United States Supplement, NEDE-24011-P-A, and NEDE-24011-P-A-US, (latest approved revision).

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

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

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

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

6. "Experience with BWR Fuel through January 1981," NEDE-24343, May 1981.

7. J. S. Charnley (GE) to L. S. Rubenstein (NRC), “1985 Fuel Experience Report,” August 13, 1986.

8. GE Marathon Control Rod Assembly, NEDE-31758P-A, October 1991.

9. (Deleted)

10. GE Marathon-Ultra Control Rod Assembly, NEDE-33284 Supplement 1P-A Revision 1, March 2012.

11. GEH Nuclear Energy Part Equivalency Evaluation Number 112-0495, Revision 0, May 29, 2012.

12. General Electric Company, “Global Nuclear Fuels Fuel Bundle Designs,” NEDC-31152P, (latest approved revision).

13. Global Nuclear Fuel, “GE14 Compliance with Amendment 22 of NEDE-24011-P-A (GESTAR II),” NEDC-32868P, Revision 5, May 2013.

14. Global Nuclear Fuel, “GNF2 Advantage Generic Compliance with NEDE-24011-P-A (GESTAR II),” NEDC-33270P, Revision 5, May 2013.

4.3 NUCLEAR DESIGN

Most of the information in <Section 4.3> is provided in the licensing topical report GESTAR (Reference 1). The subsection numbers in <Section 4.3> generally correspond to the subsection numbers of Appendix A of GESTAR. Additional information or differences are given for each applicable subsection below.

<Appendix 15B>, Reload Safety Analysis provides information on the current cycle nuclear design.

4.3.1 DESIGN BASES

Information on nuclear design bases is provided in (Reference 1) (Subsection A.4.3).

4.3.1.1 (Deleted)

4.3.1.1.1 Reactivity Bases

Information on reactivity bases is provided in (Reference 1) (Subsection A.4.3.1.1).

4.3.1.1.2 Overpower Bases

Information on overpower bases is provided in (Reference 1) (Subsection A.4.3.1.2).

4.3.1.2 (Deleted)

4.3.2 DESCRIPTION

Information on the nuclear design description is provided in (Reference 1) (Subsection A.4.3.2).

4.3.2.1 Nuclear Design Description

The nuclear design description is provided in (Reference 1) (Subsection A.4.3.2.1) with the exception of the core loading pattern. The loading pattern for the current reload cycle is given in <Appendix 15B>, Reload Safety Analysis. The fuel bundle description is provided in (Reference 1) (Subsection A.4.2.2) and the applicable bundle types for the current reload cycle are given in <Appendix 15B>, Reload Safety Analysis.

4.3.2.1.1 (Deleted)

4.3.2.1.2 (Deleted)

4.3.2.2 Power Distribution

Information on power distribution is provided in (Reference 1) (Subsection A.4.3.2.2).

4.3.2.2.1 (Deleted)

4.3.2.2.2 (Deleted)

4.3.2.2.3 (Deleted)

4.3.2.2.4 Power Distribution Calculations

Information on power distribution calculations is provided by the Perry nuclear fuel vendor in the cycle management report for each reload cycle, and discussed further in <Section 4.3.2.5>.

A full range of calculated power distributions along with the resultant exposure shapes and corresponding control rod patterns are also shown in Appendix 4A of (Reference 7), for a typical BWR/6.

4.3.2.2.5 Power Distribution Measurements

Information on power distribution measurements is provided in (Reference 1) (Subsection A.4.3.2.2.2).

4.3.2.2.6 Power Distribution Accuracy

Information on power distribution accuracy is provided in (Reference 1) (Subsection A.4.3.2.2.3).

4.3.2.2.7 Power Distribution Anomalies

Information on power distribution anomalies is provided in (Reference 1) (Subsection A.4.3.2.2.4).

4.3.2.3 Reactivity Coefficients

Information on reactivity coefficients including void, moderator, temperature, doppler and power coefficients is provided in (Reference 1) (Subsection A.4.3.2.3).

4.3.2.4 Control Requirements

Information on control requirements is provided in (Reference 1) (Subsection A.4.3.2.4). Further information is provided below.

The control rod system is designed to provide adequate control of the maximum excess reactivity anticipated during the equilibrium fuel cycle operation.

Thus, the basis for design of the burnable poison loading is that it shall compensate for the reactivity difference between the control rod system capability and the core fuel. Because fuel reactivity is at a maximum and control at a minimum at ambient temperature, the shutdown capability is evaluated assuming a cold, xenon free core.

The safety design basis requires that the core, in its maximum reactivity condition, be subcritical with the control rod of the highest worth fully withdrawn and all others fully inserted. This limit allows control rod testing at any time in core life and assures that the reactor can be made subcritical by control rods alone.

4.3.2.4.1 Shutdown Reactivity

Information on shutdown reactivity is provided in (Reference 1) (Subsection A.4.3.2.4.1). See <Appendix 15B>, Reload Safety Analysis for the cold shutdown margin for the current cycle reference loading pattern.

4.3.2.4.2 Reactivity Variations

Information on reactivity variations is provided in (Reference 1) (Subsection A.4.3.2.4.2). The combined effects of the individual constituents of reactivity for the current reload cycle are accounted for in each K_{eff} in the Reload Safety Analysis <Appendix 15B>.

The excess reactivity designed into the core is controlled by a control rod system supplemented by gadolinia-urania fuel rods. The average fuel enrichment for the core load is chosen to provide excess reactivity in the fuel assemblies sufficient to overcome the neutron losses caused by core neutron leakage, moderator heating and boiling, fuel temperature rise, equilibrium xenon and samarium poisoning, plus an allowance for fuel depletion.

4.3.2.5 Control Rod Patterns and Reactivity Worths

Information on control rod patterns and reactivity worths is provided to the Perry staff by the Perry nuclear fuel vendor in the cycle management report and the beginning of cycle cold startup report for each reload cycle.

4.3.2.5.1 Rod Control and Information System

Control rod patterns and associated control rod reactivity worths are regulated by the Rod Control and Information System (RCIS). This system utilizes redundant inputs to provide rod pattern control over the complete range of reactor operations. The control rod worths are limited to such an extent that the Rod Drop Accident (RDA) and the Power Range Rod Withdrawal Error (RWE) become unimportant. The RCIS provides for stable control of core reactivity in both the single rod or rod gang mode of operation. The Rod Pattern Controller (RPC) mode of RCIS provides protection from an RDA from startup to about 19 percent of rated power. The Rod Withdrawal Limiter (RWL) provides protection from the RWE for all conditions above the low power setpoint (LPSP). Each of these modes is described in the following sections.

4.3.2.5.2 Rod Pattern Controller (RPC) Mode

The RPC mode restricts control rod patterns to prescribed withdrawal sequences from the all-rods-inserted startup condition to about 19 percent of rated power. This mode minimizes control rod worths to the extent that they are not an important concern in the operation of a BWR. The consequences of an RDA or an RWE in this range are significantly less severe than that required to violate fuel safety limits. This system is described in detail in (Reference 4). Exception to (Reference 4) may be taken for "Alternate" control rod scram time testing provided that the exception does not result in exceeding the bounding analysis criteria used in (Reference 4). The supporting documents are provided in (Reference 8). Above 19 percent of rated power, control rod worths are very small due to the formation of voids in the moderator. Therefore, restrictions on control rod patterns are not required to minimize control rod worths.

The RPC Mode restrictions are also applied during a reactor shutdown, except that the RPC Mode restrictions may be bypassed for a reactor shutdown using the Improved BPWS Control Rod Insertion Process (Reference 12) and (Reference 13) provided:

- Withdrawn Control Rods have a confirmed coupling check.
- Control Rods, which do not have a confirmed coupling check, are fully inserted before bypassing the RPC Mode restrictions.

A coupling check is considered to be "confirmed" if no Single Operator Error can result in an incorrect coupling check, i.e., the coupling confirmation is performed once with two operators involved who both verify the rod is coupled, or the coupling confirmation is performed on two separate occasions. For purposes of this shutdown process, the method for confirming that control rods are coupled varies depending on the position of the control rod in the core. Details on this coupling confirmation requirement are provided in <Section 4> and <Section 5> of (Reference 13).

Once the above conditions are met, the RPC Mode restrictions may be bypassed. Operable control rod insertions may continue by continuously inserting the control rods to position 00.

If control rods without a confirmed coupling check can not be inserted before reducing power below the low power setpoint, then control rod insertions must be performed using the RPC Mode restrictions. Once all rods without a confirmed coupling check are inserted then the RPC Mode restrictions may be bypassed. Operable control rod insertions may continue by continuously inserting the control rods to position 00.

Normally, following bypassing of the RPC Mode restrictions, control rods are continuously inserted from their current position to the full in position in approximately the reverse order of the RPC Mode restrictions. During a shutdown, it may be necessary to bank a group of rods or to insert other control rods to control Thermal Limits. Other restrictions for unique situations such as for shutdowns with one stuck rod are provided in (Reference 13) and in plant procedures.

Once the RPC Mode restrictions have been bypassed and control rod insertions have begun using the Improved BPWS Control Rod Insertion Process, control rod withdrawals are not permitted unless compliance with the RPC requirements are re-established.

4.3.2.5.3 Rod Withdrawal Limiter (RWL) Mode

Above the low power set point the RCIS relies on the RWL mode to provide regulation of control rod withdrawals in order to prevent the occurrence of a rod withdrawal error. This mode limits the withdrawal of a single control rod or a gang of control rods to a predetermined increment depending on the power level. The system senses the location of the rod or rods and automatically blocks withdrawal when the preset increment is reached. The preset limit is determined by generic analyses such that the ΔMCPR and ΔLHGR are less than the limiting transient. At rated conditions (above the high power set point) the rod will block at a 12-inch withdrawal. Between the low power and high power set points, the increment is allowed to increase, to 24-inches. Below the low power set point the RWL mode does not apply.

4.3.2.5.4 Control Rod Operation

The control rods can be operated either individually or in a gang composed of up to four rods. The purpose of the ganged rods is to reduce the time required for plant startup or recovery from a scram. The RCIS provides regulation of control rod operation regardless of whether rods are being moved in single or ganged mode. The assignment of control rods to RCIS groups is shown in <Figure 4.3-4>, <Figure 4.3-5>, <Figure 4.3-6>, and <Figure 4.3-7>, for the A and B patterns respectively. Also shown in these figures is the division of the groups into gangs of 1 to 4 rods which can be moved simultaneously.

4.3.2.5.5 Scram Reactivity

The Reactor Protection System (RPS) responds to some abnormal operational transients by initiating a scram. The RPS and the Control Rod Drive (CRD) System act quickly enough to prevent the initiating disturbance from driving the fuel beyond transient limits. Additional information on scram reactivity is provided in (Reference 1) (Subsection S.5.1.5.2).

4.3.2.6 Criticality of Reactor During Refueling

Information on criticality of the reactor during refueling is provided in (Reference 1) (Subsection A.4.3.2.6).

The maximum allowable value of k-effective is less than 1.000 at any time.

4.3.2.7 Stability

4.3.2.7.1 Xenon Transients

Information on xenon transients is provided in (Reference 1) (Subsection A.4.3.2.7.1).

4.3.2.7.2 Thermal-Hydraulic Stability

Information on thermal-hydraulic stability is provided in (Reference 1) (Subsection A.4.3.2.7.2) and is also covered in <Section 4.4.4.6>. Thermal-hydraulic stability for the current reload cycle core is discussed in <Appendix 15B>, Reload Safety Analysis.

4.3.2.8 Vessel Irradiation

Neutron fluence at the reactor vessel is calculated as described below.

4.3.2.8.1 Unit 1 Vessel Irradiation

The lead factor for the RPV inside wall was determined by using a combination of two separate two-dimensional neutron transport computer analyses. The first of these established the azimuthal and radial variation of flux in the vessel at the fuel midplane elevation. The second analysis determined the relative variation of flux with elevation. The azimuthal and axial distribution results were combined to provide a simulation of the three-dimensional distribution of flux. The ratio of fluxes, or lead factor, between the surveillance capsule location and the peak flux locations was obtained from this distribution.

The DORT computer program <Section 4.1.4.5.1>, which utilizes the discrete ordinates method to solve the Boltzmann transport equation in two dimensions, was used to calculate the spatial flux distribution produced by a fixed source of neutrons in the core region. The azimuthal distribution was obtained with a model specified in (R,θ) geometry. A schematic illustration of the (R,θ) vessel model is shown in <Figure 4.3-9> for 1/4-core geometry. The actual calculation utilized a 1/4-core model with reflective boundary conditions at 0° and 90°. The model incorporates inner and outer core regions, the shroud, water regions inside and outside the shroud, jet pump components, and the vessel wall. A spatial mesh consisting of 194 radial intervals and 181 azimuthal intervals was used. The core region material compositions and neutron source densities were representative of values at the core midplane elevation (75 inches above the bottom of active fuel), which is near the elevation of the wires. The distributed source, which is assumed to be separate in space and energy, was obtained from the core power distribution and fission neutron spectra.

The integral over position and energy is normalized to the total fission neutron source in the region. Neutron cross-sections were specified for a 26 energy group set, with angular dependence of the scattering cross-sections approximated by a third-order Legendre polynomial expansion. The output of this calculation provided the distribution of flux as a function of azimuth and radius at reactor midplane. The azimuth of the peak flux and its magnitude relative to the flux at the 3° azimuth, which is the azimuth of the flux wires, were determined from this distribution.

The calculation of the axial flux distribution was performed in (R,Z) geometry, using a simplified cylindrical representation of the core configuration and realistic simulations of the axial variations of power density and coolant mass density. The core description was based on conditions near the azimuth angle of 21.8° where the edge of the core is closest to the vessel wall. The elevation of the peak flux was determined, as well as its magnitude relative to the flux at the surveillance capsule elevation.

The two-dimensional transport calculations indicate that flux maxima occur at azimuthal locations which are displaced by 25.5° from the RPV quadrant references (0°, 90°, etc.), at an elevation about 101 inches above the bottom of the active fuel. Calculated fluxes were obtained for the capsule position at the 3° azimuth and at the peak flux location on the vessel inside surface by combining the (R,θ) and (R,Z) flux distributions. The lead factor, as determined from the ratio of the calculated fluxes at these locations, is 0.52.

Dosimetry located on the inside surface of the vessel was removed after the first fuel cycle and tested to determine the flux at that location. The lead factor relating the dosimeter location to the peak location was used to calculate the peak vessel inside surface flux. Assuming an 80% capacity factor, or 32 effective full power years (EFPY) in 40 years of operation, the fluence for this operating period was estimated (Reference 9). Results are shown in <Table 4.3-2>. Dosimetry measurements were repeated after 5.5 EFPY after removing the surveillance capsule at the 3° azimuth (Reference 10) and (Reference 11). Results are provided in <Table 4.3-2>.

4.3.2.8.2 (Deleted)

4.3.3 ANALYTICAL METHODS

Information on analytical methods is provided in (Reference 1) (Subsection A.4.3.3).

4.3.4 CHANGES

Information on changes relative to the previous design documented in (Reference 7), is provided in (Reference 1) (Subsection A.4.3.4).

4.3.5 REFERENCES FOR SECTION 4.3

1. Global Nuclear Fuel "General Electric Standard Application for Reactor Fuel," including the United States Supplement, NEDE-24011-P-A and NEDE-24011-P-A-US, (latest approved revision).

2. J. A. Woolley, "3D BWR Core Simulator," May 1976 (NEDO-20953).

3. G. R. Parkos, "BWR Simulator Methods Verification," January 1977.

4. C. J. Paone, "Banked Position Withdrawal Sequence," January 1977 (NEDO-21231).

5. (Deleted)

6. (Deleted)

7. General Electric Standard Safety Analysis Report (GESSAR).

8. T. C. Lee (GE) to K. Donovan/P. Gilles (CEI), "Control Rod Scram Time Testing Procedure," TCL-88039, TCL-8905, TCL-8910, TCL-9022.

9. T. A. Caine, "Implementation of <Regulatory Guide 1.99>, Revision 2 for Perry Nuclear Power Plant Unit 1," November 1989 (SASR 89-76/DRF 137-0010).

10. L. J. Tilly, "Perry Unit 1 RPV Surveillance Materials Testing and Analysis," November 1996 (GE-NE-B1301793-01, Revision 0).

11. M. O'Connor, "Pressure-Temperature Curves for FirstEnergy Corporation, Using the KI_c Methodology Perry Unit 1," April 2002 (GE-NE-0000-0000-8763-01, Revision 0).

12. License Amendment 150, Perry Nuclear Power Plant, Unit No. 1 – Issuance of Amendment RE: TSTF-476, “Improved Banked Position Withdrawal Sequence Control Rod Insertion Process”, Per The Consolidated Line Item Improvement Process (TAC No. MD8184), August 28, 2008.

13. NEDO-33091-A, Revision 2, July 2004, Improved BPWS Control Rod Insertion Process.

<TABLE 4.3-1>

DELETED

TABLE 4.3-2

NEUTRON CALCULATION AND DOSIMETRY RESULTS USED TO EVALUATE VESSEL IRRADIATION

UNIT 1

Time of Measurement	Neutron Energy (MeV)	Capsule Fluence at time of Measurement (n/cm^2)	End-of-Life Fluence at Vessel ID (n/cm^2)	End-of-Life Fluence at 1/4 T Vessel Wall (n/cm^2)
1.09 EFPY	>1.0	1.47 x 10^{17}	4.3 x 10^{18} (1)	3.0 x 10^{18}
5.5 EFPY (3° azimuth capsule)	>1.0	3.53 x 10^{17}	4.0 x 10^{18} (2)	2.8 x 10^{18}
EOL as a result of power uprate	>1.0	N/A	4.1 x 10^{18} (3)	2.9 x 10^{18}

NOTES:

(1) Peak end-of-life fluence is based on flux wire test results and <Regulatory Guide 1.99>, Revision 2 calculated per (Reference 9) (1.09 EFPY), for 32 effective full-power years.

(2) Peak end-of-life fluence is based on flux wire test results and <Regulatory Guide 1.99>, Revision 2 calculated per (Reference 11) (5.5 EFPY), for 32 effective full-power years.

(3) Peak end-of-life fluence is based on flux wire test results and <Regulatory Guide 1.99>, Revision 2 calculated per (Reference 11) (5.5 EFPY), for 32 effective full-power years. The fluence is assumed to increase proportionally to the uprate (e.g., a 5% increase in flux for a 5% power uprate).

UNIT 2

(Deleted)

4.4 THERMAL AND HYDRAULIC DESIGN

Most of the information in <Section 4.4> is provided in the licensing topical report GESTAR (Reference 1). The subsection numbers in <Section 4.4> generally correspond to the subsection numbers of Appendix A of GESTAR. Any additional information or differences are given for each applicable subsection.

<Appendix 15B>, Reload Safety Analysis provides information on the current cycle thermal-hydraulic design.

Information pertaining to single recirculation loop operation is contained in <Appendix 15F>.

4.4.1 DESIGN BASIS

4.4.1.1 Safety Design Bases

Information on safety design bases is provided in (Reference 1) (Subsection A.4.4.1.1).

4.4.1.2 (Deleted)

4.4.1.3 Requirements for Steady-State Conditions

Information on requirements for steady-state conditions is provided in (Reference 1) (Subsection A.4.4.1.2).

<Appendix 15B>, Reload Safety Analysis provides the current cycle operating limit MCPR and LHGR.

4.4.1.4 Requirements for Transient Conditions

Information on requirements for transient conditions is provided in (Reference 1) (Subsection A.4.4.1.3). If Exposure-Dependent MCPR Limits are used, information on Exposure-Dependent Limits are in (Reference 1) (Subsection S.5.1.4).

4.4.1.5 Summary of Design Bases

A summary of the design bases is provided in (Reference 1) (Subsection A.4.4.1.4).

4.4.2 DESCRIPTION OF THERMAL-HYDRAULIC DESIGN OF THE REACTOR CORE

4.4.2.1 Summary Comparison

An evaluation of plant performance from a thermal and hydraulic standpoint is discussed in <Section 4.4.3>.

4.4.2.2 Critical Power Ratio

Information on the critical power ratio including boiling correlations is provided in (Reference 1) (Subsection A.4.4.2.2). The current boiling correlation used is provided in <Appendix 15B>, Reload Safety Analysis.

4.4.2.3 Linear Heat Generation Rate (LHGR)

Information on linear heat generation rate is provided in (Reference 1) (Subsection A.4.4.2.3).

4.4.2.4 Void Fraction Distribution

Void fraction distributions are calculated by the Perry nuclear fuel vendor for each reload cycle.

4.4.2.5 Core Coolant Flow Distribution and Orificing Pattern

Information on core coolant flow distribution and orificing pattern is provided in (Reference 1) (Subsection A.4.4.2.5).

4.4.2.6 Core Pressure Drop and Hydraulic Loads

Information on the core pressure drop and hydraulic loads is provided in (Reference 1) (Subsection A.4.4.2.6).

4.4.2.7 Correlation and Physical Data

Information on the correlation and physical data is provided in (Reference 1) (Subsection A.4.4.2.7).

4.4.2.8 Thermal Effects of Operational Transients

Information on thermal effects of operational transients is provided in (Reference 1) (Subsection A.4.4.2.8).

4.4.2.9 Uncertainties in Estimates

Information on uncertainties in estimates is provided in (Reference 1) (Subsection A.4.4.2.9).

4.4.2.10 Flux Tilt Considerations

Information on flux tilt considerations is provided in (Reference 1) (Subsection A.4.4.2.10) and in <Section 4.3.2.2.7>.

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

The thermal and hydraulic design of the reactor coolant system is provided in (Reference 1) (Subsection A.4.4.3).

4.4.3.1 Plant Configuration Data

4.4.3.1.1 Reactor Coolant System Configuration

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

4.4.3.1.2 Reactor Coolant System Thermal Hydraulic Data

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

4.4.3.1.3 Reactor Coolant System Geometric Data

Volumes of regions and components within the reactor vessel are shown in <Figure 5.1-2>.

<Table 4.4-5> 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 system.

<Table 4.4-6> provides the lengths and sizes of all safety injection lines to the reactor coolant system.

4.4.3.2 Operating Restrictions on Pumps

Expected recirculation pump performance curves are shown in <Figure 5.4-3>. These curves are valid for all conditions with a normal operating range varying from approximately 20 percent to 115 percent 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>. <Section 4.4.3.3> gives the operating limits imposed on the recirculation pumps by cavitation, pump loads, bearing design flow starvation, and pump speed.

4.4.3.3 Power-Flow Operating Map

4.4.3.3.1 Limits for Normal Operation

A boiling water reactor must operate with certain restrictions because of pump NPSH, overall plant control characteristics, core thermal power limits, etc. The power-flow map for the standard power range of operation is shown in <Figure 4.4-2>. The cycle specific power flowmap is provided in <Figure 15E.2-1>. The nuclear system equipment, nuclear instrumentation and the reactor protection system, in conjunction with operating procedures, maintain operations within the area of this map for normal operating conditions. The boundaries on this map are as follows:

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

b. Maximum Extended Operating Domain (MEOD) Boundary Line, or Rated Thermal Power (whichever is less): This boundary line passes through 100% reactor thermal power at 81% reactor core flow and is defined in USAR 15E.2. The operating state for the reactor roughly follows the rod line (or similar ones) during recirculation flow changes with a fixed control rod pattern. The slope of the power response to recirculation flow changes can vary where the power could cross the boundary on a flow change. The reactor power may continue to change due to slower affects such as final feedwater temperature and xenon. This boundary Line may not be exceeded.

c. Maximum Thermal Power Line: The maximum thermal power line is bounded by 81% core flow and 105% core flow. Rated power may not be exceeded.

d. Maximum Core Flow Line: The maximum core flow line is bounded by 100% thermal power and the Jet Pump and Recirc Pump Cavitation Protection line.

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

MEOD extends additional power/flow areas to the standard power-flow operating map. A discussion of MEOD and the supporting analyses are found in <Appendix 15E>.

Information pertaining to single recirculation loop operation is contained in <Appendix 15F>.

4.4.3.3.1.1 Performance Characteristics

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

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

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

4.4.3.3.2 Regions of the Power-Flow Map

Region I — This region defines the system operational capability with the recirculation pumps and motors being driven by the low frequency motor-generator set at 25 percent speed. Flow is controlled by the flow control valve and power changes, during normal startup and shutdown, will be in this region. The normal operating procedure is to startup along line B - FCV wide open at 25 percent speed.

Region II — This region shows the area where the switching sequence from the low frequency motor-generator set to 100 percent speed will be done.

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

Region IV — 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 flow control valve.

4.4.3.3.3 Design Features for Power-Flow Control

The following limits and design features are employed to maintain power-flow conditions within the required values shown in <Figure 4.4-2>. The cycle specific power-flow map is provided in <Figure 15E.2-1>.

a. Minimum power limits at intermediate and high core flows. To prevent cavitation in the recirculation pumps and jet pumps, the recirculation system is provided with an interlock to trip off the 100 percent speed power source and close the 25 percent speed power source if the difference between steam dome temperature and recirculation pump inlet temperature is less than a preset value (typically 6-11°F). The capability exists to bypass the cavitation interlock above the 70% Rod Line. The differential temperature is obtained using high accuracy Resistance Temperature Detectors (RTD's) with a sensing error of less than 0.2°F at the two standard deviation (2σ) confidence level, for the recirculation pump inlet temperature. Steam dome pressure is converted to a temperature measurement for the steam dome temperature. This action is initiated electronically through a 15-second time delay. The interlock is active while in the manual operation mode.

b. Minimum power limit at low core flow. During low power, low loop flow operations, the temperature differential interlock may not provide sufficient cavitation protection to the flow control valves. Therefore, the system is provided with an interlock to trip off the 100 percent speed power source and close the 25 percent speed power source if the feedwater flow falls below a preset level (approximately 22 percent of rated). The feedwater flow rate is measured by existing process control instruments. The speed change action is electronically initiated. This interlock is active during the manual mode of operation.

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 20 percent of rated. To assure this minimum flow, the system is designed so that the minimum flow control valve position will allow this rate of flow.

d. Valve Position. To prevent structural or cavitation damage to the recirculation pump due to pump suction flow starvation, the system is provided with pump trips if the suction or discharge block valves are at less than 90 percent open position. This circuit is activated by position limit switches and is active before the pump is started during the manual operation mode.

4.4.3.3.3.1 Flow Control

The principal modes of normal operation with valve flow control-low frequency motor generator (LFMG) set are summarized as follows: the recirculation pumps are started on the 100 percent speed power source to supply the necessary break away torque. Suction and discharge block valves are full open and the flow control valve is in the minimum position. When the pump is near full speed, the main power source is tripped and the pump allowed to coast down to approximately 25 percent speed where the LFMG set will power the pump and motor. The flow control valve is then opened to the maximum position at which point reactor heatup and pressurization can commence. When operating pressure has been established, reactor power can be increased. This power-flow increase will follow a line within Region I of the flow control map shown in <Figure 4.4-2>. The cycle specific power-flow map is provided in <Figure 15E.2-1>.

When reactor power is greater than approximately 20-28 percent of rated, the low feedwater flow interlock is cleared and the main recirculation pumps can be switched to the 100 percent speed power source. The flow control valve is closed to the minimum position before the speed change to prevent large increases in core power and a potential flux scram. This operation occurs within Region II of the operating map. The system is then brought to the desired power-flow level within the normal operating area of the map (Region IV) by opening the flow control valves and by withdrawing control rods.

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

4.4.3.4 Temperature-Power Operating Map (PWR)

This section is not applicable to PNPP.

4.4.3.5 Load Following Characteristics

This function is not available at Perry. The load following circuits have been disabled.

4.4.3.6 (Deleted)

4.4.4 EVALUATION

Information on the evaluation of the thermal-hydraulic design is provided in (Reference 1) (Subsection A.4.4.4).

4.4.4.1 Critical Power

Information on critical power is provided in (Reference 1) (Subsection A.4.4.4.1). The current boiling correlation used is provided in <Appendix 15B>, Reload Safety Analysis.

4.4.4.2 Core Hydraulics

Information on core hydraulics is provided in (Reference 1) (Subsection A.4.4.4.2).

4.4.4.3 Influence of Power Distributions

Information on influence of power distributions is provided in (Reference 1) (Subsection A.4.4.4.3).

4.4.4.4 Core Thermal Response

Information on core thermal response is provided in (Reference 1) (Subsection A.4.4.4.4). Information on core thermal response for the current cycle limiting transients is provided in <Appendix 15B>, Reload Safety Analysis.

4.4.4.5 Analytical Methods

Information on analytical methods is provided in (Reference 1) (Subsection A.4.4.4.5). Current General Electric design methodologies are found in (Reference 1).

4.4.4.6 Thermal-Hydraulic Stability Analysis

Information on thermal-hydraulic stability analysis is provided in (Reference 1) (Subsection A.4.4.4.6). Thermal-hydraulic stability for the current reload cycle core is discussed in <Appendix 15B>, Reload Safety Analysis.

4.4.5 TESTING AND VERIFICATION

Information on testing and verification is provided in (Reference 1) (Subsection A.4.4.5).

4.4.6 INSTRUMENTATION REQUIREMENTS

The reactor vessel instrumentation monitors the key reactor vessel operating parameters during planned operations. This ensures sufficient control of the parameters. The following reactor vessel sensors are discussed in <Section 7.6.1>, <Section 7.2.1>, and <Section 7.5.1>.

a. (Deleted)

b. Reactor Vessel Water Level

c. (Deleted)

d. Reactor Vessel Internal Pressure

e. Neutron Monitoring Systems

The Reactor Vessel Coolant Temperature and the Reactor Vessel Coolant Flow Rates and Differential Pressures are included in the scope of the thermal hydraulic information discussed in <Section 7.5.1>.

4.4.6.1 Loose Parts Monitoring

Not applicable to PNPP design, Loose Parts Monitoring System was eliminated/abandoned in place.

<TABLE 4.4-1>

<TABLE 4.4-2>

<TABLE 4.4-3>

<TABLE 4.4-4>

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TABLE 4.4-5

REACTOR COOLANT SYSTEM GEOMETRIC DATA

		Flow Path Length (in.)	Height and Liquid Level (in.)	Elevation of Bottom of Each Volume[1] (in.)	Minimum Flow Areas (sq ft)
A.	Lower Plenum	213.5	213.5 213.5	-170.5	84.0
B.	Core	164.5	164.5 164.5	43.0	146.5 Includes bypass
C.	Upper Plenum and Separators	179.0	179.0 179.0	207.5	57.5
D.	Dome (Above Normal Water Level)	289.5	289.5 0	386.0	309.0
E.	Downcomer Area	311.5	311.5 311.5	-27.5	66.0
F.	Recirculation Loops and Jet Pumps	114.0 ft (one loop)	398.0 398.0	-392.0	132.5 in.^{2}

NOTE:

(1) Reference Point is recirculation nozzle outlet centerline.

TABLE 4.4-6

LENGTHS OF SAFETY INJECTION LINES[1]

	Pipe Diameter (in.)		Pipe Length (ft)
LPCS System	14		106
	12		149
		Total	255
HPCS System	16		139
	12		120
		Total	259
RHR-A	18		112
	12		106
		Total	218
RHR-B	18		113
	12		152
		Total	265
RHR-C	18		120
	12		254
		Total	374

NOTE:

[1] Lengths are from pump discharge to RPV nozzle.

4.5 REACTOR MATERIALS

4.5.1 CONTROL ROD DRIVE SYSTEM STRUCTURAL MATERIALS

4.5.1.1 (Deleted)

(Deleted)

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

The coupling spud, collet fingers, buffer spring, nut (hex), and collet spring are fabricated from Alloy X-750 in the annealed or equalized condition, and aged 20 hours at 1,300°F to produce a tensile strength of 165,000 psi minimum, a yield strength of 105,000 psi minimum, and elongation of 20 percent minimum. The piston head, stop piston, buffer shaft, and buffer piston are Armco 17-4 PH in condition H-1100, aged 6 hours at 1,100°F with a tensile strength of 140,000 psi minimum, a yield strength of 115,000 psi minimum, and elongation of 14 percent minimum.

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

All materials, except SA 479 or SA 249 Grade XM-19, have been successfully used for the past 10 to 15 years in similar drive mechanisms. Extensive laboratory tests have demonstrated that ASME SA 479 or SA 249 Grade XM-19 is a suitable material and that it is resistant to stress corrosion in a BWR environment.

b. Special Materials

No cold worked austenitic stainless steels with a yield strength greater than 90,000 psi are employed in the Control Rod Drive system. Hardenable martensitic stainless steels are not used. Armco 17-4 PH (precipitation hardened stainless steel) is used for the piston head, stop piston, buffer shaft, and buffer piston.

This material is aged to the H-1100 condition to produce resistance to stress corrosion cracking in the BWR environments.
Armco 17-4 PH (H-1100) has been successfully used for the past 10 to 15 years in BWR drive mechanisms.

4.5.1.2 Austenitic Stainless Steel Components

a. Processes, Inspections and Tests

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

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

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

Colmonoy hard surfacing is applied by the flame spray or the TIG welding process. Parts are preheated to 550-800°F and then sprayed with Colmonoy. The sprayed coating is fused at about 2,000°F using an oxyacetylene torch followed by air cooling.

Nitrided components are exposed to a temperature of about 1,080°F for about 20 hours during nitriding cycle.

Colmonoy hard surfaced components have performed successfully for the past 10 to 15 years in drive mechanisms. Nitrided components have been used in control rod drives since 1967. It is normal practice to remove some Control Rod Drives at each refueling outage. At this time, both the Colmonoy hard surfaced parts and nitrided surfaces are accessible for visual examination. In addition, dye penetrant examinations have been performed on nitrided surfaces of the longest service drives. This inspection program is adequate to detect any incipient defects before they could become serious enough to cause operating problems.

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

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

General Compliance or Alternate Approach Assessment: For Commitment, Revision Number, and Scope, see <Section 1.8>.

b. Control of Delta Ferrite Content

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

<Regulatory Guide 1.31>, Control of Stainless Steel Welding

General Compliance or Alternate Approach Assessment: For Commitment, Revision Number and Scope, see <Section 1.8>.

4.5.1.3 Other Materials

These are discussed in <Section 4.5.1.1.b>.

4.5.1.4 Cleaning and Cleanliness Control

4.5.1.4.1 Protection of Materials During Fabrication, Shipping and Storage

All the Control Rod Drive parts listed 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.

c. Release of parts for shipment.

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

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

The degree of surface cleanliness obtained by these procedures meets the regulatory positions of <Regulatory Guide 1.37>.

Semiannual examination of the humidity indicators of a minimum of 10 percent of the units during inside heated warehouse storage is required to verify that the units are dry and in satisfactory condition.

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

General Compliance or Alternate Approach Assessment: For Commitment, Revision Number and Scope, see <Section 1.8>.

4.5.2 REACTOR INTERNAL MATERIALS

4.5.2.1 Material Specifications

Materials used for the Core Support Structure:

a. Shroud Support - Nickel-Chrome-Iron-Alloy, ASME SB 166 or SB 168.

b. Shroud, core plate and top guide - ASME SA 240, SA 182, SA 479, SA 312, SA 249, or SA 213 (all Type 304L).

c. Peripheral fuel supports - ASTM: SA 312 Gr TP 304, and Type 304L.

d. Core plate and top guide studs and nuts, and core plate wedges - ASME SA 479 (Type 304 and XM-19), SA 193 Grade B8A, SA 194 Grade 8A (all Type 304) and ASTM A-276.

e. Top guide pins - ASME SA 479 (Type XM-19) and ASME Code Case N-207-1.

f. Control rod drive housing - ASME SA 312 Type 304, SA 182 Type 304 and ASME SB 167 Type Inconel 600.

g. Control rod guide tube - ASTM: A 358 Gr 304, A 312 Gr TP 304, A 351 Gr CF8, A 249 TP 304; ASME: SA 358 Gr 304, SA 312 Gr TP 304, SA 351 TP GR CF3.

h. Orificed fuel support - ASTM: A 249 TP 304, A 240 TP 316L, A 479 TP 316L. ASME: SA-351 Type CF8.

Materials Employed in Other Reactor Internal Structures.

a. Shroud Head/Steam Separators and Steam Dryer

All materials are Type 304 or 304L stainless steel. Shroud Head/Steam Separators and the Steam Dryer are fabricated in accordance with the following ASME and ASTM specifications respectively:

Plate, Sheet and Strip	ASME SA 240 and ASTM A240
Forgings	ASME SA 182 or SA 479 and ASTM A182
Bars	ASME SA 479 and ASTM A479
Pipe	ASME SA 312 and ASTM A312
Tube	ASME SA 213 or SA 249 and ASTM A269
Castings	ASME SA 351 and ASTM A351

b. Jet Pump Assemblies

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

Castings	ASTM A351 Grade CF8 and ASME SA 351 Grade CF3
Bars	ASTM A276 Type 304 and ASTM A637 Grade 688 ASTM A479 Type 316L ASME SB637 N07750 Type 3 (Alloy X-750)
Bolts	ASTM A193 Grade B8 or B8M ASME SA 479 Type 316L
Sheet and Plate	ASTM A240 Type 304, 304L, 316L ASME SA 240 Type 304L, 316L

Tubing	ASTM A269 Grade TP 304
Pipe	ASTM A358 Type 304, and 316L ASME SA 312 Grade TP 304, 316L
Welded Fittings	ASTM A403 Grade WP304
Forged or Rolled Parts	ASME SA 182 or ASTM A182 Grade F304, F316L 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 Adaptor casting, the wedge casting, bracket casting adjusting screw casting, and the Diffuser collar casting are Type 304 hard surfaced with Stellite 6 for slip fit joints.

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

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

d. The Inlet-Mixer for jet pumps 15, 16, 17 and 18 contain a wedge made of Alloy X-750 to Specification ASME SB637 N07750 Type 3.

All core support structures are fabricated from ASME specified materials, and designed in accordance with the requirements of ASME Code, Section III, Subsection NG. Other reactor internals are non-coded, and they are fabricated from ASTM or ASME specification materials. Material requirements in the ASTM specifications are identical to requirements in corresponding ASME material specifications. Note, ASTM A637 and ASME SA637 materials are now designated as ASTM B637 and ASME SB637 materials.

4.5.2.2 Controls on Welding

Core support structures are fabricated in accordance with the requirements of ASME Code Section III, Subsection NG. Other internals are not required to meet ASME Code requirements. Requirements of ASME Section IX are followed in the fabrication of core support structures and other internals.

4.5.2.3 Nondestructive Examination of Wrought Seamless Tubular Products

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

Wrought seamless tubular products for other internals were supplied in accordance with the applicable ASTM or ASME material specifications. These specifications require a hydrostatic test on each length of tubing.

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

<Regulatory Guide 1.31>, Control of Ferrite Content in Stainless Steel Weld Metal

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

<Regulatory Guide 1.34>, Control of Electroslag Weld Properties

Electroslag welding is not employed for any reactor internals.

<Regulatory Guide 1.36>, Non-Metallic Thermal Insulation for Austenitic Stainless Steel

For external applications, all nonmetallic insulation meets the regulatory positions of <Regulatory Guide 1.36>.

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

All wrought austenitic stainless steel was purchased in the solution heat treated condition. Heating above 800°F was prohibited (except for welding) unless the stainless steel was subsequently solution annealed. For 304 steel with carbon content in excess of 0.035 percent carbon, purchase specifications restricted the maximum weld heat input to 110,000 Joules per inch, and the weld interpass temperature to 350°F maximum. Welding was performed in accordance with Section IX of the ASME Boiler and Pressure Vessel Code. These controls were employed to avoid severe sensitization and comply with the intent of <Regulatory Guide 1.44>.

<Regulatory Guide 1.71>, Welder Qualification for Areas of Limited Accessibility

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

<Regulatory Guide 1.37>, Quality Assurance Requirements for Cleaning of Fluid Systems and associated components of water-cooled Nuclear Power Plants.

Exposure to contaminants 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 obtained by these procedures meets the regulatory positions of <Regulatory Guide 1.37>.

<Regulatory Guide 1.31>, <Regulatory Guide 1.34>, <Regulatory Guide 1.36>, <Regulatory Guide 1.44>, <Regulatory Guide 1.37>, and <Regulatory Guide 1.71>

General Compliance or Alternate Approach Assessment: For Commitment, Revision Number and Scope, See <Section 1.8>.

4.5.2.5 Other Materials

Cold worked stainless steels are not used in the reactor internals.

Hardenable martensitic stainless steels and precipitation hardening stainless steels are not used in the reactor internals.

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

SA 479 Type XM-19 stainless steel

SB 166, 167, and 168 Nickel-Chrome-Iron (Inconel 600)

SA 637 Grade 688 Inconel X-750

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

Inconel X-750 components are fabricated in the annealed or equalized condition and aged 20 hours at 1,300°F. Tube may be hot finished, while sheet may be as rolled.

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

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

4.5.3 CONTROL ROD DRIVE HOUSING SUPPORTS

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

	Material
Grid	ASTM A441
Disc springs	Schnorr, Type BS-125-71-8
Hex bolts and nuts	ASTM A307
6 x 4 x 3/8 tubes	ASTM A 500 Grade B

For further control rod drive housing support information refer to <Section 4.6.1.2>.

4.6 FUNCTIONAL DESIGN OF REACTIVITY CONTROL SYSTEMS

The reactivity control systems consist of control rods and control rod drives, supplementary reactivity control for the initial core <Section 4.3> and the standby liquid control system described in <Section 9.3.5>.

4.6.1 INFORMATION FOR CRDS

4.6.1.1 Control Rod Drive System Design

4.6.1.1.1 Design Bases

4.6.1.1.1.1 General Design Bases

4.6.1.1.1.1.1 Safety Design Bases

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

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

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

c. Each positioning device shall:

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

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

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

4.6.1.1.1.1.2 Power Generation Design Bases

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

4.6.1.1.2 Description

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

4.6.1.1.2.1 Control Rod Drive Mechanisms

The CRD mechanism (drive) used for positioning the control rod in the reactor core is a double-acting, mechanically latched, hydraulic cylinder using water as its operating fluid <Figure 4.6-1>, <Figure 4.6-2>, <Figure 4.6-3>, and <Figure 4.6-4>. The individual drives are mounted on the bottom head of the reactor pressure vessel. The drives do not interfere with refueling and are operative even when the head is removed from the reactor vessel.

The drives are also readily accessible for inspection and servicing. The bottom location makes maximum utilization of the water in the reactor as a neutron shield and gives the least possible neutron exposure to the drive components. Using water from the condensate treatment system, and/or condensate storage tanks as the operating fluid eliminates the need for special hydraulic fluid. Drives are able to utilize simple piston seals whose leakage does not contaminate the reactor water but provides cooling for the drive mechanisms and their seals.

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

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

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

The individual rod indicators, grouped in one control panel display, correspond to relative rod locations in the core.

Changes in local flux during control rod motion at power may be observed by monitoring the readings of the appropriate local power range monitor (LPRM) string. To facilitate this when a control rod is selected, the output of an appropriate LPRM string is displayed along with the position of the selected control rod. Except for certain peripheral control rods, the LPRM string used is diagonally adjacent to the selected control rod.

4.6.1.1.2.2 Drive Components

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

4.6.1.1.2.2.1 Drive Piston

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

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

This difference in driving area tends to balance the control rod weight and assures a higher force for insertion than for withdrawal.

4.6.1.1.2.2.2 Index Tube

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

4.6.1.1.2.2.3 Collet Assembly

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

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

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

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

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

4.6.1.1.2.2.4 Piston Tube

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

4.6.1.1.2.2.5 Stop Piston

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

The center tube of the drive mechanism forms a well to contain the position indicator probe. The probe is an aluminum extrusion attached to a cast aluminum housing. Mounted on the extrusion are hermetically sealed, magnetically operated, reed switches. The entire probe assembly is protected by a thin-walled stainless steel tube. The switches are actuated by a ring magnet located at the bottom of the drive piston.

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

4.6.1.1.2.2.6 Flange and Cylinder Assembly

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

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

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

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

4.6.1.1.2.2.7 Lock Plug

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

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

The control rod is heavy enough to force the spud fingers to enter the socket and push the lock plug up, allowing the spud to enter the socket completely and the plug to snap back into place. Therefore, the drive can be coupled to the control rod using only the weight of the control rod.

4.6.1.1.2.3 Materials of Construction

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

4.6.1.1.2.3.1 Index Tube

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

4.6.1.1.2.3.2 Coupling Spud

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

4.6.1.1.2.3.3 Collet Fingers

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

4.6.1.1.2.3.4 Seals and Bushings

Graphite carbon seal material 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 Graphite carbon seal material strength is experienced at higher temperatures, the drive is supplied with cooling water to hold temperatures below 250°F. The Graphite carbon seal material 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 Graphite carbon seal material.

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

h. The drive piston head, stop piston, buffer shaft, and buffer piston are made of Armco 17-4 PH.

i. Certain fasteners and locking devices are made of Inconel X-750 or 600.

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

The CRD return line is capped to avoid potential nozzle cracking as required by <NUREG-0619>.

4.6.1.1.2.4 Control Rod Drive Hydraulic System

The control rod drive 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's to the scram discharge volume. The water discharged from a drive during a normal control rod positioning operation flows through the HCU, the exhaust header and is returned to the reactor vessel via the HCU's of non-moving drives. There is one HCU for each control rod drive.

4.6.1.1.2.4.1 Hydraulic Requirements

The CRD hydraulic system design is shown in <Figure 4.6-5> and <Figure 4.6-7>. The hydraulic requirements, identified by the function they perform, are as follows:

a. An accumulator hydraulic charging pressure of approximately 1,750 to 2,000 psig is required. Flow to the accumulators is required only during scram reset or system startup.

b. Drive pressure of approximately 260 psi above reactor vessel pressure measured at a point immediately above the core plate is required. A flow rate of approximately 4 gpm to insert each control rod and 2 gpm to withdraw each control rod is required.

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

d. The scram discharge volume is sized to receive, and contain, all the water discharged by the drives during a scram while maintaining a pressure less than 65 psig; a minimum volume of 3.34 gallons per drive is required (excluding the instrument volume).

e. Charging water header supplies approximately .008 gpm to level control instruments, used to prevent buildup of non-condensible gases.

4.6.1.1.2.4.2 System Description

The CRD hydraulic systems provide the required functions with the pumps, filters, valves, instrumentation, and piping shown in <Figure 4.6-5> and described in the following paragraphs.

4.6.1.1.2.4.2.1 Supply Pump

One supply pump pressurizes the system with water from the condensate treatment system and/or condensate storage tanks. One spare pump is provided for standby. A discharge check valve prevents backflow through the non-operating 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 sets of filters in the system. The pump suction filters are a disposable element type designed to remove particulate which could reduce the operating life of the drive water filters. A 250-micron strainer in the filter bypass line protects the pump when these filters are being serviced. The drive water filters, downstream of the pump, are cleanable element types with a 50-micron absolute rating. A differential pressure indicator and control room alarm monitor the operating filter element as it collects foreign materials.

4.6.1.1.2.4.2.2 Accumulator Charging Pressure

Accumulator charging pressure is established by precharging the nitrogen accumulator and then opening the charging water isolation valve. During scram, the scram inlet (and outlet) valves open and permit the stored energy in the accumulators to discharge into the drives. The resulting pressure decrease in the charging water header allows the CRD supply pump to “run out” (i.e., flow rate to increase substantially) into the control rod drives via the charging water header. The flow element upstream of the accumulator charging header senses high flow and provides a signal to the manual auto-flow control station which in turn closes the system flow control valve. This action diverts increased flow to the charging water header.

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

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

4.6.1.1.2.4.2.3 Drive Water Pressure

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

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

4.6.1.1.2.4.2.4 Cooling Water Header

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

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

4.6.1.1.2.4.2.5 Scram Discharge Volume (SDV)

The scram discharge volume consists of header piping which connects to each HCU and drains into an instrument volume. The header piping is sized to receive and contain all the water discharged by the drives during a scram, independent of the instrument volume. Each of the two sets of headers has its own directly-connected scram discharge instrument volume (SDIV) attached to the low point of the header piping. The large diameter pipe of the instrument volume serves as a vertical extension of the SDV (though no credit is taken for it in determining SDV sizing requirements).

During normal plant operation the scram discharge volume is empty, and vented to atmosphere through its open vent and drain valve. When a scram occurs, upon a signal from the safety circuit these vent and drain valves are closed to conserve reactor water. Redundant vent and drain valves are provided to assure against loss of reactor coolant from the SDV following a scram. Lights in the control room indicate the position of these valves.

During a scram, the scram discharge volume partly fills with water discharged from above the drive pistons. After scram is completed, the control rod drive seal leakage from the reactor continues to flow into the scram discharge volume until the discharge volume pressure equals the reactor vessel pressure. A check valve in each HCU prevents reverse flow from the scram discharge header volume to the drive. When the initial scram signal is cleared from the reactor protection system, the scram discharge volume signal is overridden with a keylock override switch, and the scram discharge volume is drained and returned to atmospheric pressure.

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

Each instrument volume is monitored by level switches and by transmitter activated trip units <Figure 4.6-5>. One level switch and trip unit (contacts) in series constitutes one trip logic for input to the RPS. Each RPS trip system receives two logic trip inputs both from one instrument volume. Two level switches and two transmitter/trip units in a one-out-of-two twice logic will provide redundant and diverse inputs to the RPS to initiate a reactor scram when water in each instrument volume exceeds that preset high water level. Furthermore, alarms and rod blocks will also provide warnings at lower water levels to control room operators if the instrument volume is not completely empty.

4.6.1.1.2.4.3 Hydraulic Control Units

Each hydraulic control unit (HCU) furnishes pressurized water, on signal, to a drive unit. The drive then positions its control rod as required. Operation of the electrical system that supplies scram and normal control rod positioning signals to the HCU is described in <Section 7.7.1>.

The basic components in each HCU are manual, pneumatic and electrical valves; an accumulator; related piping; electrical connections; filters; and instrumentation <Figure 4.6-5>, <Figure 4.6-7> and <Figure 4.6-8>. The components and their functions are described in the following paragraphs.

4.6.1.1.2.4.3.1 Insert Drive Valve

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

4.6.1.1.2.4.3.2 Insert Exhaust Valve

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

4.6.1.1.2.4.3.3 Withdraw Drive Valve

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

4.6.1.1.2.4.3.4 Withdraw Exhaust Valve

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

4.6.1.1.2.4.3.5 Speed Control Units

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

4.6.1.1.2.4.3.6 Scram Pilot Valve Assembly

The scram pilot valve assembly is operated from the reactor protection system. The scram pilot valve assembly, with two solenoids, controls both the scram inlet valve and the scram exhaust valve. The scram pilot valve assembly is solenoid-operated and is normally energized. On loss of electrical signal to the solenoids, such as the loss of external ac power, the inlet port closes and the exhaust port opens. The pilot valve assembly <Figure 4.6-5> is designed so that the trip system signal must be removed from both solenoids before air pressure can be discharged from the scram valve operators. This prevents inadvertent scram of a single drive in the event of a failure of one of the pilot valve solenoids.

4.6.1.1.2.4.3.7 Scram Inlet Valve

The scram inlet valve opens to supply pressurized water to the bottom of the drive piston. This quick opening globe valve is operated by an internal spring and system pressure. It is closed by air pressure applied to the top of its diaphragm operator. A position indicator switch on this valve energizes a light in the control room as soon as the valve starts to open.

4.6.1.1.2.4.3.8 Scram Exhaust Valve

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

4.6.1.1.2.4.3.9 Scram Accumulator

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

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

A float type level switch actuates an alarm if water leaks past the piston barrier and collects in the accumulator instrumentation block.

4.6.1.1.2.5 Control Rod Drive System Operation

The control rod drive system performs rod insertion, rod withdrawal and scram. These operational functions are described in the sections that follow.

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 travel speed (nonscram operation) of 3 in./sec. During normal insertion, the pressure on the downstream side of the speed control valve is approximately 90 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 1,040 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 <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 second. 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 travel speed of 3 in./sec. The entire valving sequence is automatically controlled and is initiated by a single operation of the rod withdraw switch.

4.6.1.1.2.5.3 Scram

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

The large differential pressure (approximately 1,750 psi, initially and always several hundred psi, depending on reactor vessel pressure) produces a large upward force on the index tube and control rod. This force gives the rod a high initial acceleration and provides a large margin of force to overcome friction. After the initial acceleration is achieved, the drive continues at a diminishing velocity. This characteristic provides a high initial rod insertion rate. As the drive piston nears the top of its stroke, the piston reaches the buffer and the driveline is brought to a stop at the full-in position.

Prior to a scram signal the accumulator in the Hydraulic Control Unit has 1,750-2,000 psig on the water side and approximately 1,750 psig on the nitrogen side. As the inlet scram valve opens, the full water side pressure is available at the control rod drive acting on a 4.1 sq inch area. As CRD motion begins, this pressure drops to the gas side pressure less line losses between the accumulator and the CRD. When the drive reaches the full-in position, the accumulator completely discharges with a resulting gas side pressure of approximately 1,200 psig.

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

The maximum scram insertion time for each control rod from the fully withdrawn position, based on de-energization of the scram pilot valve solenoid as time zero is contained in the Technical Specifications.

4.6.1.1.2.5.4 Alternate Rod Insertion (ARI)

The Alternate Rod Insertion feature is designed to increase the reliability of the Control Rod Drive system scram function. ARI provides for insertion of reactor control rods by depressurizing the scram air header through valves which are redundant and diverse from the reactor protection system scram valves.

The Redundant Reactivity Control System (RRCS) <Section 7.6.1.12>, signal to insert control rods results in energizing the eight ARI valves shown on <Figure 4.6-5>. Two valves in series with the backup scram valves assure venting of air from the air supply line in the event one or more of the ARI valves fails. Four valves provide for venting of the A and B HCU scram valve pilot air headers to atmosphere to depressurize the headers and scram all rods. Two additional valves vent the scram air header which serves the scram discharge volume drain and vent lines, closing those valves and isolating the SDV.

4.6.1.1.2.6 Instrumentation

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

The design bases and further discussion are covered in <Chapter 7>, "Instrumentation and Control Systems."

4.6.1.2 Control Rod Drive Housing Supports

4.6.1.2.1 Safety Objective

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

4.6.1.2.2 Safety Design Bases

The CRD housing supports shall meet the following safety design bases:

a. Following a postulated CRD housing failure, control rod downward motion shall be limited so that any resulting power excursion could not be sufficient to cause fuel damage.

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 4.6-9>. Horizontal beams are installed immediately below the bottom head of the reactor vessel, between the rows of CRD housings. The beams are welded to brackets which are welded to the steel form liner of the drive room in the reactor support pedestal.

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

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

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

When the support bars and grids are installed, a gap of approximately 1 inch at room temperature (approximately 70°F) is provided between the grid and the bottom contact surface of the control rod drive flange. During system heatup, this gap is reduced by a net downward expansion of the housings with respect to the supports. In the hot operating condition, the gap is approximately 3/4 inch.

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

The American Institute of Steel Construction (AISC) Manual of Steel Construction, "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings," was used in designing the CRD housing support system. However, to provide a structure that absorbs as much energy as practical without yielding, the allowable tension and bending stresses used were 90 percent of yield and the shear stress used was 60 percent of yield. These design stresses are 1.5 times the AISC allowable stresses (60 percent and 40 percent 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 1,086 psig (at the bottom of the vessel) acting on the area of the separated housing. The weight of the separated housing, control rod drive and blade, plus the pressure of 1,086 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-inch gap before it contacts the supports. The impact force (109,000 lb) is then treated as a static load in design. The CRD housing supports are designed as Category I (seismic) equipment in accordance with <Section 3.2>. Loading conditions and examples of stress analysis results and limits are shown in <Table 3.9-3>. Safety evaluation is discussed in <Section 4.6.2.3.3>.

4.6.2 EVALUATIONS OF THE CRDS

4.6.2.1 <u>Failure Mode and Effects Analysis</u>

This subject is discussed in <Appendix 15A>.

4.6.2.2 <u>Protection from Common Mode Failures</u>

This subject is discussed in <Appendix 15A>.

4.6.2.3 Safety Evaluation

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

4.6.2.3.1 Control Rods

4.6.2.3.1.1 Materials Adequacy Throughout Design Lifetime

The adequacy of the materials throughout the design life was evaluated in the mechanical design of the control rods. The primary materials, B_4C powder, hafnium metal, 304 austenitic stainless steel, and RAD RESIST 304S as supplied by General Electric have been found suitable in meeting the demands of the BWR environment.

4.6.2.3.1.2 Dimensional and Tolerance Analysis

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

4.6.2.3.1.3 Thermal Analysis of the Tendency to Warp

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

4.6.2.3.1.4 Forces for Expulsion

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

4.6.2.3.1.5 Functional Failure of Critical Components

The consequences of a functional failure of critical components have been evaluated and the results are covered in <Section 4.6.2.3.2.2>, "Analysis of Malfunction Relating to Rod Withdrawal."

4.6.2.3.1.6 Precluding Excessive Rates of Reactivity Addition

In order to preclude excessive rates of reactivity addition, analysis has been performed both on the velocity limiter device and the effect of probable control rod failures <Section 4.6.2.3.2.2>.

4.6.2.3.1.7 Effect of Fuel Rod Failure on Control Rod Channel Clearances

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

4.6.2.3.1.8 Mechanical Damage

In addition to the analysis performed on the control rod drive <Section 4.6.2.3.2.2> and <Section 4.6.2.3.2.3> and the control rod blade, analyses were performed on the control rod guide tube; refer to <Section 4.2.3.3.7>, and <Section 4.2.3.3.8> for these analyses.

4.6.2.3.1.9 Evaluation of Control Rod Velocity Limiter

The control rod velocity limiter limits the free fall velocity of the control rod to a value that cannot result in nuclear system process barrier damage. The rod drop accident is evaluated in <Chapter 15>.

4.6.2.3.2 Control Rod Drives

4.6.2.3.2.1 Evaluation of Scram Time

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

4.6.2.3.2.2 Analysis of Malfunction Relating to Rod Withdrawal

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

4.6.2.3.2.2.1 Drive Housing Fails at Attachment Weld

The bottom head of the reactor vessel has a penetration for each control rod drive location. A drive housing is raised into position inside each penetration and fastened by welding. The drive is raised into the drive housing and bolted to a flange at the bottom of the housing.

The CRD housing material at the vessel penetration is seamless, Inconel 600 tubing with a minimum tensile strength of 80,000 psi, and Type 304 stainless steel pipe below the vessel with a minimum strength of 75,000 psi. The basic failure considered here is a complete circumferential crack through the housing wall at an elevation just below the J-weld.

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

If the housing were to fail as described, the following sequence of events is foreseen. The housing would separate from the vessel. The control rod drive and housing would be blown downward against the support structure, by reactor pressure acting on the cross-sectional area of the housing and the drive. The downward motion of the drive and associated parts would be determined by the gap between the bottom of the drive and the support structure and by the deflection of the support structure under load. In the current design, maximum deflection is approximately 3 inches. If the collet were to remain latched, no further control rod ejection would occur (Reference 1); the housing would not drop far enough to clear the vessel penetration; reactor water would leak at a rate of approximately 180 gpm through the 0.03-inch diametral clearance between the housing and the vessel penetration.

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

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

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

4.6.2.3.2.2.2.1 Pressure-under (Insert) Line Break

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

If the pressure-under (insert) line were to fail and if the collet were latched, no control rod withdrawal would occur. There would be no pressure differential across the collet piston and, therefore, no tendency to unlatch the collet. Consequently, the associated control rod could not be withdrawn, but if reactor 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 containment. Because of the broken line, cooling water could not be supplied to the drive involved. Loss of cooling water would cause no immediate damage to the drive. However, prolonged exposure of the drive to temperatures at or near reactor temperature could lead to deterioration of material in the seals. High temperature would be indicated to the operator by the thermocouple in the position indicator probe. A second indication would be high cooling water flow.

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

4.6.2.3.2.2.2.2 Pressure-over (Withdrawn) Line Break

The case of the pressure-over (withdrawn) line breakage considers the complete breakage of the line at or near the point where it enters the housing flange. If the line were to break, pressure over the drive piston would drop from reactor pressure to atmospheric pressure. Any significant reactor pressure (approximately 600 psig or greater) would act on the bottom of the drive piston and fully insert the drive. Insertion would occur regardless of the operational mode at the time of the failure. After full insertion, reactor water would leak past the stop piston seals. This leakage would exhaust to the containment through the broken pressure-over line. The leakage rate at 1,000 psi reactor pressure is estimated to be 1 to 3 gpm, however with the graphitar seals of the stop piston removed, the leakage rate could be as high as 10 gpm, based on experimental measurements. If the reactor were hot, drive temperature would increase. This situation would be indicated to the reactor operator by the drift alarm, by the fully inserted drive, by a high drive temperature annunciated in the control room, and by operation of the drywell sump pump.

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

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

4.6.2.3.2.2.3 All Drive Flange Bolts Fail in Tension

Each control rod drive is bolted to a flange at the bottom of a drive housing. The flange is welded to the drive housing. Bolts are made of either AISI-4140 or E-434OH steel, with a minimum tensile strength of 125,000 psi. Each bolt has an allowable load capacity of at least 15,200 pounds. Minimum capacity of the 8 bolts is 121,600 pounds. As a result of the reactor design pressure of 1,250 psig, the major load on all 8 bolts is 30,400 pounds.

If a progressive or simultaneous failure of all bolts were to occur, the drive would separate from the housing. The control rod and the drive would be blown downward against the support structure. Impact velocity and support structure loading would be slightly less than that for drive housing failure, because reactor pressure would act on the drive cross-sectional area only and the housing would remain attached to the reactor vessel. The drive would be isolated from 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 1,435 pounds to hold the collet in the latched position.

If the bolts failed during control rod withdrawal, pressure below the collet piston would drop to zero. The collet, with 1,650 pounds return force, would latch and stop rod withdrawal.

4.6.2.3.2.2.4 Weld Joining Flange to Housing Fails in Tension

The failure considered is a crack in or near the weld that joins the flange to the housing. This crack extends through the wall and completely around the housing. The flange material is forged, Type 304 stainless steel, with a minimum tensile strength of 75,000 psi. The housing material is seamless, Type 304 stainless steel pipe, with a minimum tensile strength of 75,000 psi. The conventional, full penetration weld of Type 308 stainless steel has a minimum tensile strength approximately the same as that for the parent metal. The design pressure and temperature are 1,250 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 5,100 psi.

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

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

4.6.2.3.2.2.5 Housing Wall Ruptures

This failure is a vertical split in the drive housing wall just below the bottom head of the reactor vessel. The flow area of the hole is considered equivalent to the annular area between the drive and the thermal sleeve. Thus, flow through this annular area, rather than flow through the hole in the housing, would govern leakage flow. The CRD housing is made of Inconel 600 seamless tubing (at the penetration to the vessel), with a minimum tensile strength of 80,000 psi, and of Type 304 stainless steel seamless pipe below the vessel with a minimum tensile strength of 75,000 psi. The maximum hoop stress of 9,000 psi results primarily from the reactor design pressure (1,250 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 1,030 gpm. Choke-flow conditions would exist, as described previously for the flange-bolt failure. However, leakage flow would be greater because flow resistance would be less, that is, the leaking water and steam would not have to flow down the length of the housing to reach the drywell. A critical pressure of 350 psi causes the water to flash to steam.

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

If this failure occurred during control rod withdrawal, drive withdrawal would stop, but the collet would remain unlatched. The drive would be stopped by a reduction of the net downward force action on the drive line. The net force reduction would occur when the leakage flow of 1,030 gpm reduces the pressure in the annulus outside the drive to approximately 540 psig, thereby reducing the pressure acting on top of the drive piston to the same value. A pressure differential of approximately 710 psi would exist across the collet piston and hold the collet unlatched as long as the operator held the withdraw signal.

4.6.2.3.2.2.6 Flange Plug Blows Out

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

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

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

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

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

4.6.2.3.2.2.7 Ball Check Valve Plug Blows Out

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

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

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

4.6.2.3.2.2.8 Drive/Cooling Water Pressure Control Valve Closure (Reactor Pressure, 0 psig)

The drive/cooling water pressure control valve in the failed closed or open condition can only effect the velocity of the control rod during insert or withdrawal. The scram pressure source is independent of the drive water pressure.

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 2,000 psig. Calculations indicate that the drive would accelerate from 3 inch/sec to approximately 7 inch/sec. A pressure differential of 1,970 psi across the collet piston would hold the collet unlatched. Flow would be upward, past the velocity limiter piston, but retarding force would be negligible. Rod movement would stop as soon as the driving signal was removed.

4.6.2.3.2.2.9 Ball Check Valve Fails to Close Passage to Vessel Ports

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

4.6.2.3.2.2.10 Hydraulic Control Unit Valve Failures

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

4.6.2.3.2.2.11 Collet Fingers Fail to Latch

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

4.6.2.3.2.2.12 Withdrawal Speed Control Valve Failure

Normal withdrawal speed is determined by differential pressures in the drive and is set for a nominal value of 3 in./sec. Withdrawal speed is maintained by the pressure regulating system and is independent of reactor vessel pressure. Tests have shown that accidental opening of the speed control valve to the full-open position produces a velocity of approximately 5 in./sec. The control rod drive system prevents unplanned rod withdrawal and it has been shown above that only multiple failures in a drive unit and in its control unit could cause an unplanned rod withdrawal.

4.6.2.3.2.3 Scram Reliability

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

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

b. Each drive mechanism has its own scram valves and a dual solenoid scram pilot valve therefore only one drive can be affected if a scram valve fails to open. Both pilot valve solenoids must be de-energized to initiate a scram.

c. The reactor protection system and the HCU’s 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.

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

4.6.2.3.2.4 Control Rod Support and Operation

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

4.6.2.3.3 Control Rod Drive Housing Supports

Downward travel of the CRD housing and its control rod following the postulated housing failure equals the sum of these distances: (1) the compression of the disc springs under dynamic loading, and (2) the initial gap between the grid and the bottom contact surface of the CRD flange. If the reactor were cold and pressurized, the downward motion of the control rod would be limited to the spring compression (approximately 2 in.) plus a gap of approximately 1 in. If the reactor were hot and pressurized, the gap would be approximately 3/4 in. and the spring compression would be slightly less than in the cold condition. In either case, the control rod movement following a housing failure is substantially limited below one drive "notch" movement (6 in.). Sudden withdrawal of any control rod through a distance of one drive notch at any position in the core does not produce a transient sufficient to damage any radioactive material barrier.

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

At plant operating temperature, a gap of approximately 3/4 in. exists between the CRD housing and the supports. At lower temperatures the gap is greater. Because the supports do not contact any of the CRD housing except during the postulated accident condition, vertical contact stresses are prevented. Inspection and testing of CRD housing supports is discussed in <Section 4.6.3.2.1>.

4.6.3 TESTING AND VERIFICATION OF THE CRDS

4.6.3.1 Control Rod Drives

4.6.3.1.1 Testing and Inspection

4.6.3.1.1.1 Development Tests

The initial development drive (prototype of the standard locking piston design) testing included more than 5,000 scrams and approximately 100,000 latching cycles. One prototype was exposed to simulated operating conditions for 5,000 hours. 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.

d. Usable seal lifetimes in excess of 1,000 scram cycles can be expected. See (Reference 2) for more information on Fast Scram Qualification Program.

4.6.3.1.1.2 Factory Quality Control Tests

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

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.

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.

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

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

4.6.3.1.1.4 Acceptance Tests

Criteria for acceptance of the individual control rod drive mechanisms and the associated control and protection systems are 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 defines criteria and acceptable ranges of such characteristics as seal leakage, friction and scram performance under fixed test conditions which must be met before the component can be shipped.

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

As fuel is placed in the reactor, the startup test procedures <Chapter 14> are followed. The tests in these procedures 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. The detailed specifications and procedures follow the general pattern established for such specifications and procedures in BWRs presently under construction and in operation.

4.6.3.1.1.5 Surveillance Tests

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

a. Sufficient control rods shall be withdrawn, following a refueling outage following fuel movement within the reactor pressure vessel or control rod replacement, to demonstrate with a margin of 0.28% $\Delta k/k$ (with the highest worth control rod determined by test) that the core can be made subcritical at any time in the subsequent fuel cycle with the strongest operable control rod fully withdrawn and all other operable rods fully inserted. With the highest worth control rod analytically determined, a margin of 0.38% $\Delta k/k$ must be demonstrated.

b. When above the low power setpoint of the Rod Pattern Control System, each withdrawn control rod shall be inserted one notch at least once every 31 days.

In the event that operation is continuing with any control rod immovable as a result of excessive friction or mechanical interference, this test shall be performed within 24 hours from discovery of the stuck control rod.

The control rod exercise tests serve as a periodic check against deterioration of the control rod system and also verifies the ability of the control rod drive to scram. If a rod can be moved with drive pressure, it may be expected to scram since higher pressure is applied during scram. The frequency of exercising the control rods under the conditions of any control rod immovable as a result of excessive friction or mechanical interference provides even further assurance of the reliability of the remaining control rods.

c. 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. Each time the rod is fully withdrawn, observe that the drive will not go to the overtravel position.

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

d. During operation, accumulator pressure at the normal operating value shall be verified.

   Experience with control rod drive systems of the same type indicates that weekly verification of accumulator pressure is sufficient to assure operability of the accumulator portion of the control rod drive system.

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

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

4.6.3.1.1.6 Functional Tests

The functional testing program of the control rod drives consists of the 5 year maintenance life and the 1.5X design life test programs as described in <Section 3.9>.

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 CRD's under these conditions.

Simulated Ruptured Scram Line Test
Stuck Ball Check Valve in CRD Flange
HCU Drive Down Inlet Flow Control Valve (V122) Failure
HCU Drive Down Outlet Flow Control Valve (V120) Failure
CRD Scram Performance with Vl20 Malfunction
HCU Drive Up Outlet Control Valve (V121) Failure
HCU Drive Up Inlet Control Valve (V123) Failure
Cooling Water Check Valve (V138) Leakage
CRD Flange Check Valve Leakage
CRD Stabilization Circuit Failure
HCU Filter Restriction
Air Trapped in CRD Hydraulic System
CRD Collet Drop Test
CR Qualification Velocity Limiter Drop Test

Additional postulated CRD failures are discussed in <Section 4.6.2.3.2.2.1>, <Section 4.6.2.3.2.2.2>, <Section 4.6.2.3.2.2.3>, <Section 4.6.2.3.2.2.4>, <Section 4.6.2.3.2.2.5>, <Section 4.6.2.3.2.2.6>, <Section 4.6.2.3.2.2.7>, <Section 4.6.2.3.2.2.8>,

<Section 4.6.2.3.2.2.9>, <Section 4.6.2.3.2.2.10>, and <Section 4.6.2.3.2.2.11>.

4.6.3.2 Control Rod Drive Housing Supports

4.6.3.2.1 Testing and Inspection

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

4.6.4 INFORMATION FOR COMBINED PERFORMANCE OF REACTIVITY CONTROL SYSTEMS

4.6.4.1 Vulnerability to Common Mode Failures

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

4.6.4.2 Accidents Taking Credit for Multiple Reactivity Systems

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

4.6.5 EVALUATION OF COMBINED PERFORMANCE

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

Except for the Standby Liquid Control System, Perry's control rod scram system meets all the criteria enumerated in Section 4 of the NRC Staff generic safety evaluation report BWR Scram Discharge System, dated 12/1/80. A summary of each criterion is given below along with a discussion of how the scram discharge system complies.

a. Functional Criteria

Functional Criterion 1

The scram discharge volume shall have sufficient capacity to receive and contain water exhausted by a full reactor scram without adversely affecting control rod drive scram performance.

Perry Compliance:

A minimum scram discharge volume of 3.34 gallons per drive is specified through the system design specifications. This minimum scram discharge volume is based on conservative assumptions as to the performance of the scram system. In the event of a coolant leak into the SDV an automatic scram will occur before the required SDV available volume is threatened.

b. Safety Criteria

Safety Criterion 1

No single active failure of a component or service function shall prevent a reactor scram, under the most degraded conditions that are operationally acceptable.

Perry Compliance:

No single active failure in the scram system design will prevent a reactor scram. The Perry scram discharge system design meets the NRC acceptance criterion for Safety Criterion 1. Partial loss or full loss of service functions will result in either not adversely affecting the scram system function or a full reactor scram. The Perry system has no reduction in the pipe size of the header piping going from the HCU's to and including the Scram Discharge Instrument Volume (SCIV). This hydraulic coupling permits operability of the scram level instrumentation prior to loss of system function. The scram level instrumentation are redundant and diverse to assure no single active failure or common mode failure prevents a reactor scram.

Safety Criterion 2

No single active failure shall result in an uncontrolled loss of reactor coolant.

Perry Compliance:

Redundant Scram Discharge Volume (SDV) vent and drain valves are provided as part of the SDV modifications. The redundant SDV valve configuration assures that no single failure can result in an uncontrolled loss of reactor coolant. An additional solenoid operated pilot valve controls the redundant vent and drain valve. The vent and drain system is therefore sufficiently redundant to avoid a failure to isolate the SDV due to solenoid failure. The vent and drain valve’s opening and closing sequences are controlled to minimize excessive hydrodynamic forces.

Safety Criterion 3

The scram discharge system instrumentation shall be designed to provide redundancy, to operate reliably under all conditions and shall not be adversely affected by hydrodynamic forces or flow characteristics.

Perry Compliance:

Diverse, and redundant level sensing instrumentation on the Scram Discharge Instrument Volume (SDIV) is provided for the automatic scram function. SDIV water level is measured by utilization of both float sensing and pressure sensing devices. Instrument taps have been relocated from the vent and drain piping to the SDIV to protect the level sensing instrumentation from the flow dynamics in the scram discharge system. Each SDIV has a redundant instrument loop. A one-out-of-two twice logic is employed for the automatic scram function. This instrumentation arrangement assures the automatic scram function on high SDIV water level in the event of a single active or passive failure.

Safety Criterion 4

System operation conditions which are required for scram shall be continuously monitored.

Perry Compliance:

See response to Safety Criterion 3.

Safety Criterion 5

Repair, replacement, adjustment, or surveillance of any system component shall not require the scram function to be bypassed.

Perry Compliance:

The SDIV scram level instrumentation arrangement and trip logic allows instrument adjustment or surveillance without bypassing the scram function or directly causing a scram. Each level instrument can be individually isolated without bypassing the scram function. A one-out-of-two twice trip logic is employed. Plant procedures will insure that the scram function is not bypassed during surveillance, repair or replacement of any system component.

c. Operational Criteria

Operational Criterion 1:

Level instrumentation shall be designed to be maintained, tested or calibrated during plant operation without causing a scram.

Perry Compliance:

See response to Safety Criterion 5.

Operational Criterion 2:

The system shall include sufficient supervisory instrumentation and alarms to permit surveillance of system operation.

Perry Compliance:

Supervisory instrumentation and alarms such as accumulator trouble, scram valve air supply low and high pressure and scram discharge volume not drained alarms, are adequate and permit surveillance of the scram system's readiness.

Operational Criterion 3:

The system shall be designed to minimize the exposure of operating personnel to radiation.

Perry Compliance:

Minimizing the exposure of operating personnel to radiation is a consideration in equipment design and location.

Operational Criterion 4:

Vent paths shall be provided to assure adequate drainage of the SDV in preparation for scram reset.

Perry Compliance:

A vent line is provided as part of the scram discharge system to assure proper drainage in preparation for scram reset. GE specifications require the vent to be provided by a dedicated vent line with a non-submerged discharge to the atmosphere. Furthermore, additional vent capability is provided by the vent line vacuum breakers. The vacuum breakers are required to start to open at no greater than 1.0 psi differential.

Operational Criterion 5:

Vent and drain functions shall not be adversely affected by other system interfaces. The objective of this requirement is to preclude water backup in the scram instrument volume which could cause spurious scram.

Perry Compliance:

The SDV vent and drain lines are required to be dedicated lines. The vent line discharge is directed into the drywell and the drain line discharge is directed into the suppression pool. Vacuum breakers on the SDV vent line and shutoff valves on the SDV vent and drain lines preclude water from siphoning back into the SDIV from their respective discharge systems.

d. Design Criteria

Design Criterion 1:

The scram discharge headers shall be sized in accordance with GE OER-54 and shall be hydraulically coupled to the instrument volume(s) in a manner to permit operability of the scram level instrumentation prior to loss of system function. Each system shall be analyzed based on a plant-specific maximum in-leakage to ensure that the system function is not lost prior to initiation of automatic scram. Maximum in-leakage is the maximum flow rate through the scram discharge line without control-rod motion, summed over all control rods. The analysis should show no need for vents or drains.

Perry Compliance:

As discussed in response to Functional Criterion 1, a minimum scram discharge volume of 3.34 gallons per drive is specified through the system design specifications. Furthermore, the Perry system has no reduction in the pipe size of the header piping going from the HCU's to the scram discharge volume including the SDIV. The SDIV is directly connected to the scram discharge volume at the low point of the scram discharge header piping. These requirements satisfy the NRC's acceptance criteria for Design Criterion 1.

Design Criterion 2:

Level instrumentation shall be provided for automatic scram initiation while sufficient volume exists in the scram discharge volume.

Perry Compliance:

See response to Functional Criterion 1 and Design Criterion 1.

Design Criterion 3:

Instrumentation taps shall be provided on the vertical instrument volume and not on the connected piping.

Perry Compliance:

See response to Safety Criterion 3.

Design Criterion 4:

The scram instrumentation shall be capable of detecting water accumulation in the instrumented volume(s) assuming a single active failure in the instrumentation system or the plugging of an instrument line.

Perry Compliance:

See response to Safety Criterion 3.

Design Criterion 5:

Structural and component design shall consider loads and conditions including those due to fluid dynamics, thermal expansion, internal pressure, seismic considerations, and adverse environments.

Perry Compliance:

The SDV and associated vent and drain piping is classified as important to safety and required to meet the ASME Section III Class 2 and Seismic Category I requirements.

Design Criterion 6:

The power-operated vent and drain valves shall close under loss of air and/or electric power. Valve position indication shall be provided in the control room.

Perry Compliance:

The present vent and drain valve design operation meets this criterion.

Design Criterion 7:

Any reductions in the system piping flow path shall be analyzed to assure system reliability and operability under all modes of operation.

Perry Compliance:

See response to Design Criterion 1.

Design Criterion 8:

System piping geometry (i.e., pitch, line size, orientation) shall be such that the system drains continuously during normal plant operation.

Perry Compliance:

All SDV piping is required to be continuously sloped from its high point to its low point.

Design Criterion 9:

Instrumentation shall be provided to aid the operator in the detection of water accumulation in the instrumented volume(s) prior to scram initiation.

Perry Compliance:

The present alarm and rod block instrumentation meets this criterion.

Design Criterion 10:

Vent and drain line valves shall be provided to contain the scram discharge water, with a single active failure and to minimize operational exposure.

Perry Compliance:

See response to Safety Criterion 2.

e. Surveillance Criteria

Implementation of surveillance procedures to comply with the following surveillance criteria are included in the plant surveillance program. (The Technical Specifications comply with the intent of the Safety Evaluation Report's Surveillance Criteria.)

Surveillance Criterion 1:

Vent and drain valves shall be periodically tested.

Surveillance Criterion 2:

Verifying level detection instrumentation shall be periodically tested in place.

4.6.6 REFERENCES FOR SECTION 4.6

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

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