SECTION 4

REACTOR

TABLE OF CONTENTS

TABLE OF CONTENTS (Cont)

TABLE OF CONTENTS (Cont)

TABLE OF CONTENTS (Cont)

LIST OF TABLES

*Provided in NEDO-20944, NEDE-20944-P, and/or NEDE-20944-1P.

LIST OF FIGURES

LIST OF FIGURES (Cont)

SECTION 4

REACTOR

This section was prepared using the latest approved revision of the topical reports "General Electric Standard Application for Reactor Fuel" (GESTAR II) including the "United States Supplement," NEDE-24011-P-A and NEDE 24011-P-A-US and the "Reference Safety Report for Boiling Water Reactor Reload Fuel," (CENPD-300-P-A). Applicable sections of these reports are referenced as noted in Sections 4.1 through 4.4. Reference is made to standardized information contained in the topical reports, consistent with the NRC overall standardization philosophy.

4.1 SUMMARY DESCRIPTION

The reactor assembly includes 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 (CRD) housings, and the control rod drives. Figure 3.9-2 shows the arrangement of reactor assembly components. A summary of the important design and performance characteristics is given in Section 1.3. Loading conditions for reactor assembly components are presented 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 (including the fuel, channels, control blades, and in-core 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. Of the preceding components, the fuel assemblies (including fuel rods and channel), control blades, in-core instrumentation, shroud head and steam separator assembly, and steam dryer assembly are removable when the reactor vessel is opened for refueling or maintenance.

4.1.2.1 Reactor Core

4.1.2.1.1 General

The design of the boiling water reactor (BWR) 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.

Important features of the reactor core arrangement are as follows:

1. The bottom entry cruciform control rods consist of boron carbide (B_4C) and/or hafnium as the absorbing material. The control rods have been irradiated for many years and accumulated thousands of hours of service in operating BWRs without significant failure.

2. Fixed, in-core fission chambers provide continuous power range neutron flux monitoring. A guide tube in each in-core assembly provides for a traversing ion chamber for calibration and axial detail. Source and intermediate range detectors are located in-core and are axially retractable. The in-core location of the startup and source range instruments provides coverage of the large reactor core and provides an acceptable signal to noise ratio and neutron to gamma ratio. All in-core instrument

leads enter from the bottom and the instruments are in service during refueling. In-core instrumentation is discussed in Section 7.7.

3. Experience at operating plants has shown that the operator, using the in-core flux monitoring system, can maintain the desired power distribution within a large core by proper control rod scheduling.

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

5. Mechanical reactivity control permits criticality checks during refueling and provides maximum plant safety. At any time in its operating history, the core is designed to be subcritical with any one control rod fully withdrawn.

6. The selected control rod pitch represents a practical value of individual control rod reactivity worth, and allows adequate clearance between control rod drive (CRD) mechanisms below the pressure vessel 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 given in Section 4.3.

4.1.2.1.3 Fuel Assembly Description

Descriptions of the fuel assembly and the fuel rods are referenced in Section 4.2.

4.1.2.1.4 Fuel Assembly Support and Control Rod Location

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

The top guide, mounted inside the shroud, provides lateral support and guidance for each fuel assembly. The reactivity of the core is controlled by cruciform control rods and their associated Mechanical Hydraulic Drive System. The control rods occupy alternate spaces between fuel assemblies. Each independent CRD enters the core from the bottom, accurately positions its associated control rod during normal operation, and yet exerts approximately 10 times the force of gravity to insert the control rod during the scram mode of operation. Bottom entry allows optimum power shaping in the core, ease of refueling, and convenient CRD maintenance.

4.1.2.2 Shroud

Information on the shroud is contained in Section 3.9.5.

4.1.2.3 Shroud Head and Steam Separator Assembly

Information on the shroud head and steam separator assembly is contained in Section 3.9.5.

4.1.2.4 Steam Dryer Assembly

Information on the steam dryer assembly is contained in Section 3.9.5.

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 to counterbalance steam voids in the top of the core and effect significant power flattening. The groups of control elements used for power flattening experience a somewhat higher duty cycle and neutron exposure than the other rods in the control system.

The reactivity control function requires that all rods be available for either reactor scram 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 that 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

A description of the control rods is referenced 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. A description of the supplementary burnable poison is referenced in Section 4.2.

4.1.4 Analysis Techniques

4.1.4.1 Reactor Internal Components

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

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

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

4.1.4.1.1 MASS (Mechanical Analysis of Space Structure)

4.1.4.1.1.1 Program Description

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

4.1.4.1.1.2 Program Version and Computer

GE was using Revision 0 of MASS. The program operated on the Honeywell 6000 computer.

4.1.4.1.1.3 History of Use

Since its development in the early 1960s, the program has been successfully applied to a wide variety of jet engine structural problems, many of which involve extremely complex geometries. The use of the program by the GE nuclear energy component began shortly after its development. The use continued until the late 1970s when the program was retired in favor of newer programs such as ANSYS.

4.1.4.1.1.4 Extent of Application

Besides the jet engine and nuclear energy components, GE's Missile and Space Division, Appliance Division, and Turbine Division have

also applied the program to a wide range of engineering problems. The GE nuclear energy component used it mainly for piping and reactor internals analyses.

4.1.4.1.2 SNAP (MULTISHELL)

4.1.4.1.2.1 Program Description

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

4.1.4.1.2.2 Program Version and Computer

The version maintained by the GE Jet Engine Division at Evandale, Ohio, was used on the Honeywell 6000 computer by the GE nuclear energy component.

4.1.4.1.2.3 History of Use

The initial version of the shell analysis program was completed by the Jet Engine Division in 1961. Considerable modifications and additions were made by the GE nuclear energy component to accommodate a broadening area of application, and the program was used for more than 10 years. In the mid 1970s, the program was replaced by newer finite element programs such as ANSYS.

4.1.4.1.2.4 Extent of Application

The program has been used to analyze jet engine, space vehicle, and nuclear reactor components. Because of its efficiency, economy, and reliability, it has been one of the main shell analysis programs used by GE's nuclear energy component.

4.1.4.1.3 GASP

4.1.4.1.3.1 Program Description

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

4.1.4.1.3.2 Program Version and Computer

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

4.1.4.1.3.3 History of Use

The program was developed by E. L. Wilson in 1965, is found in Reference 4.1-3 and used by the GE nuclear energy component from 1967 until 1975. The program was replaced by other general purpose programs such as ANSYS and SAP.

4.1.4.1.3.4 Extent of Application

The application of GASP by the GE nuclear energy component was mainly for elastic analysis of axisymmetric and plane structures

under thermal and pressure loads. The GE version has been extensively tested and used by GE engineers.

4.1.4.1.4 NOHEAT

4.1.4.1.4.1 Program Description

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

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

4.1.4.1.4.2 Program Version and Computer

The current version of the program is an improvement of the program originally developed by I. Farhoomand and Professor E. L. Wilson; see Reference 4.1-4. The program operates on the Honeywell 6000 computer. It was renamed TASA in the late 1970s.

4.1.4.1.4.3 History of Use

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

4.1.4.1.4.4 Extent of Application

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

4.1.4.1.5 FINITE

4.1.4.1.5.1 Program Description

FINITE is a general purpose, finite element computer program for elastic stress analyses of two dimensional structural problems including plane stress, plane strain, and axisymmetric structures. It has provision for thermal, mechanical, and body force loads. The materials of the structure may be homogeneous or nonhomogeneous, and isotropic or orthotropic. The development of the FINITE program is based on the GASP program.

4.1.4.1.5.2 Program Version and Computer

The version of the program used by the GE nuclear energy component was obtained from the developer, J. E. McConnelee of GE's Gas Turbine Department, in 1969, from Reference 4.1-5, and was used on the Honeywell 6000 computer.

4.1.4.1.5.3 History of Use

Since its completion in 1969, the program has been widely used in GE's Gas Turbine and the Jet Engine Departments for turbine component analyses. The program was used in the GE nuclear energy component until the mid-1970s, when it was replaced by the ANSYS computer program.

4.1.4.1.5.4 Extent of Application

The program was used by the GE nuclear energy component in the analysis of axisymmetric, or nearly axisymmetric, boiling water reactor (BWR) internals.

4.1.4.1.6 DYSEA

4.1.4.1.6.1 Program Description

The DYSEA (Dynamic and Seismic Analysis) program is a GE proprietary program developed specifically for seismic and dynamic analysis of Reactor Pressure Vessel (RPV)/Internals and Building Systems. It calculates the dynamic response of linear structural systems by either temporal modal superposition or response spectrum methods. Fluid structure interaction effects in the RPV are taken into account by way of hydrodynamic mass.

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

4.1.4.1.6.2 Program Version and Computer

The DYSEA version now operating on the Honeywell 6000 computer of the GE nuclear energy component was developed at GE by modifying the SAPIV program. Capability was added to handle the hydrodynamic mass effects due to fluid structure interactions in the reactor. It accommodates three dimensional dynamic problems with beams, trusses, and springs. Both acceleration time histories and response spectra may be used as input.

4.1.4.1.6.3 History of Use

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

4.1.4.1.6.4 Extent of Application

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

4.1.4.1.7 SHELL 5

4.1.4.1.7.1 Program Description

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

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

4.1.4.1.7.2 Program Version and Computer

A copy of the source deck of SHELL 5 is maintained by the GE nuclear energy component. SHELL 5 operated on the UNIVAC 1108 computer.

4.1.4.1.7.3 History of Use

SHELL 5 is a program developed by Gulf General Atomic Incorporated, as mentioned in Reference 4.1-7, in 1969. The program has been in production status at Gulf General Atomic, GE, and at other major computer operating systems since 1970. The program was used in the GE nuclear energy component until the mid 1970s, when it was replaced by the ANSYS program.

4.1.4.1.7.4 Extent of Application

SHELL 5 was used at GE to analyze the reactor shroud support and suppression chamber. Satisfactory results were obtained.

4.1.4.1.8 HEATER

4.1.4.1.8.1 Program Description

HEATER is a computer program used in the hydraulic design of feedwater spargers and their associated delivery header and piping. The program uses test data obtained by GE using full scale mockups of feedwater spargers, combined with a series of models that represent the complex mixing processes obtained in the upper plenum, downcomer, and lower plenum. Mass and energy balances throughout the Nuclear Steam Supply System (NSSS are modeled in detail; see Reference 4.1-8 for more details.

4.1.4.1.8.2 Program Version and Computer

This program was developed in FORTRAN IV by the GE nuclear energy component for the Honeywell 6000 computer.

4.1.4.1.8.3 History of Use

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

4.1.4.1.8.4 Extent of Application

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

4.1.4.1.9 FAP-71 (Fatigue Analysis Program)

4.1.4.1.9.1 Program Description

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

4.1.4.1.9.2 Program Version and Computer

The present version of FAP-71 was completed by L. Young of the GE nuclear energy component in 1971, found in Reference 4.1-9. The program is currently used on the Honeywell 6000 computer.

4.1.4.1.9.3 History of Use

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

4.1.4.1.9.4 Extent of Application

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

4.1.4.1.10 CREEP-PLAST

4.1.4.1.10.1 Program Description

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

The plasticity treatment is based on kinematic hardening and Von Mise's yield criterion. The hardening modulus can be a constant or a function of strain.

4.1.4.1.10.2 Program Version and Computer

This program can be used for elastic plastic analysis, with or without the presence of creep. It can also be used for creep analysis without the presence of instantaneous plasticity. A detailed description of theory is given in Reference 4.1-11. The program operates on the Honeywell 6000 computer.

4.1.4.1.10.3 History of Use

This program was developed by Y. R. Rashid, as discussed in Reference 4.1-11, in 1971. It underwent extensive program testing before it was put on production status.

4.1.4.1.10.4 Extent of Application

This program is used by the GE nuclear energy component in the channel cross section mechanical analysis.

4.1.4.1.11 ANSYS

4.1.4.1.11.1 Program Description

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

The ANSYS program features the following capabilities:

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

2. One dimensional fluid flow analyses

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

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

5. Geometry plotting for all elements in the ANSYS library, including isometric and perspective views of three dimensional structures

6. Restart capability for several analyses types with an option for saving the stiffness matrix, once it is calculated for the structure, and using it for other loading conditions.

4.1.4.1.11.2 Program Version and Computer

The program is maintained by Swanson Analysis Systems, Inc., of Pittsburgh, Pennsylvania and is supplied to GE for use on the Honeywell 6000.

4.1.4.1.11.3 History of Use

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

4.1.4.1.11.4 Extent of Application

ANSYS is used extensively by the GE nuclear energy component for elastic and elastic plastic analysis of the RPV, core support structures, and reactor internals.

4.1.4.2 Fuel Rod Thermal Design Analyses

Fuel rod thermal design analyses are described in Section 2 of GESTAR II (Reference 4.1-1) and section 3 of Reference 4.1-12.

4.1.4.3 Reactor Systems Dynamics

The analysis techniques and computer codes used in reactor systems dynamics are described in Section 4 of References 4.1-10, 4.1-10A,

and 4.1-10B. Section 4.4.4 provide results of the stability analysis for the Reactor Coolant System (RCS).

4.1.4.4 Nuclear Engineering Analysis

The analysis techniques are described and referenced in Section 3 of Reference 4.1-1.

4.1.4.5 Neutron Fluence Calculations

Vessel neutron fluence calculations are described in Section 4.3.2.8.

4.1.4.6 Thermal Hydraulic Calculations

Descriptions of the thermal hydraulic models are given in Section 4 of Reference 4.1-1.

4.1.5 References

4.1-1 "General Electric Standard Application for Reactor Fuel," including the "United States Supplement," NEDE-24011-P-A and NEDE-24011-P-A-US, latest approved revisions.

4.1-2 L. Beitch, "Shell Structures Solved Numerically by Using a Network of Partial Panels," AIAA Journal, Volume 5, No. 3, March 1967.

4.1-3 E. L. Wilson, "A Digital Computer Program For the Finite Element Analysis of Solids With Non-Linear Material Properties," Aerojet General Technical Memo No. 23, Aerojet General, July 1965.

4.1-4 I. Farhoomand and E. L. Wilson, "Non-Linear Heat Transfer Analysis of Axisymmetric Solids," SESM Report SESM71-6, University of California at Berkeley, Berkeley, California, 1971.

4.1-5 J. E. McConnelee, "Finite-Users Manual," General Electric TIS Report DF 69SL206, March 1969.

4.1-6 R. W. Clough and C. P. Johnson, "A Finite Element Approximation For the Analysis of Thin Shells," International Journal of Solid Structures, Vol. 4, 1968.

4.1-7 "A Computer Program For the Structural Analysis of Arbitrary Three-Dimensional Thin Shells," Report No. GA-9952, Gulf General Atomic, 1969.

4.1-8 A. B. Burgess, "User Guide and Engineering Description of HEATER Computer Program," General Electric, NEDE-20731-02 March 1974.

4.1-9 L. J. Young, "FAP-71 (Fatigue Analysis Program) Computer Code," GE/NED Design Analysis Unit R. A. Report No. 49, January 1972.

4.1-10 L. A. Carmichael and G. J. Scatena, Stability and Dynamic Performance of the General Electric Boiling Water Reactor, NEDO-21506, January 1977.

4.1-10A General Electric, "Analytical Methods of Plant Transient Evaluations for the General Electric Boiling Water Reactor," NEDO-10802-A, December 1986.

4.1-10B General Electric, "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," NEDO-24154-A, Volumes 1, 2, and 3, August 1986.

4.1-11 Y. R. Rashid, "Users Manual for CRPSOL Computer Program," NEDO-23538, December 1976.

4.1-12 ABB Combustion Engineering Nuclear Power, "Reference Safety Report for Boiling Water Reactors Reload Fuel," CENPD-300-P-A, July 1966.

4.2 FUEL SYSTEM DESIGN

The format of this section corresponds to Standard Review Plan 4.2 in NUREG-0800. Most of the information is presented by reference to GESTAR II (Ref. 4.2-1) and Reference 4.2-5.

4.2.1 Design Bases

References to design bases are given in Subsection A.4.2.1 of GESTAR II (Ref. 4.2-1) and Reference 4.2-5.

4.2.2 Description and Design Drawings

The fuel system description and design drawings for GE14i, GE14 and GNF2 fuel are given in References 4.2-9, 4.2-10 and 4.2-11.

4.2.2.1 Reactivity Control Assembly (Control Rods)

The control rod descriptions are given in Ref. 4.2-2, Ref. 4.2-3, Ref.4.2-4 and Ref. 4.2-7.

4.2.2.2 Reactivity Control Assembly Evaluation

The control rod evaluations are given in Ref. 4.2-2, Ref. 4.2-3, Ref. 4.2-4, and Ref. 4.2-7.

4.2.3 Design Evaluation

Compliance with the design bases is discussed in Subsection A.4.2.3 of GESTAR II (Ref. 4.2-1) and Reference 4.2-5, with the exception that Paragraphs 4.2.3.2.9 and 4.2.3.3.5 appear as below.

4.2.3.2.9 Mechanical Fracturing Evaluation

All mechanical breaking under normal operation and abnormal operational transients is bounded by the analysis for LOCA plus SSE given in Section 3.9.1.4.10.

4.2.3.3.5 Structural Deformation Evaluation

Results of the Hope Creek specific SSE plus LOCA analysis are documented in Section 3.9.1.4.10.

4.2.4 Testing, Inspection and Surveillance Plans

Descriptions of General Electric fuel assembly testing, inspection, and surveillance are referenced in Subsection A.4.2.4 of GESTAR II (Ref. 4.2-1). Descriptions of ABB fuel assembly testing, inspection, and surveillance are provided in section 9 of reference 4.2-6.

4.2.5 REFERENCES

4.2-1 "General Electric Standard Application for Reactor Fuel," including the "United States Supplement," NEDE-24011-P-A and NEDE-24011-P-A-US, latest revision.

4.2-2 "BWR/4 and BWR/5 Fuel Design", NEDE-20944-P-1 (Proprietary) and NEDO-20944-1, October 1976, and "Amendment 1," January 1977.

4.2-3 "Topical Report - ASEA-ATOM BWR Control Blades for US BWRs" UR 85-225A including Supplement 1A, 2 and 3, ASEA-ATOM, Vasteras, Sweden, October 1985.

4.2-4 "ABB BWR Generic Control Rod Design Methodology," CENPD-290-P, ABB/CE Nuclear Operations, Feb, 1994.

4.2-5 ABB Combustion Engineering Nuclear Power, "Reference Safety Report for Boiling Water Reactor Reload Fuel," CENPD-300-P-A, July 1996.

4.2-6 ABB Combustion Engineering Nuclear Power, "Fuel Assembly Mechanical Design Methodology for Boiling Water Reactors," CENPD-287-P-A, July 1996.

4.2-7 "Westinghouse BWR Control Rod CR99 Licensing Report", WCAP-16182-P, December 2003.

4.2-8 "Fuel Assembly Mechanical Design Methodology for Boiling Water Reactors, Supplement 1 to CENP-287," WCAP-15942-P-A, March 2006.

4.2-9 "GE Hitachi Nuclear Energy Safety Analysis Report to Support Introduction of GE14i Isotope Test Assemblies (ITAs) in Hope Creek Generating Station," NEDC-33529P Revision 0 Class III DRF 0000-0107-3743, December 2009 including Errata and Addenda Number 1 for Hope Creek ITA Safety Analysis Report NEDC-33529P Revision 0, June 10 2010.

4.2-10 "GE14 Compliance With Amendment 22 of NEDE-24011-P-A (GESTAR II)," NEDC-32868P, Revision 6, March 2016.

4.2-11 "GNF2 Advantage Generic Compliance with NEDE-24011-P-A (GESTAR II)," NEDC-33270P, Revision 6, March 2016.

4.3 NUCLEAR DESIGN

Most of the information of Section 4.3 is provided in the licensing topical report, GESTAR II (Reference 4.3-1) and Reference 4.3-3. The subsection numbers in Section 4.3 directly correspond to the subsection numbers of Appendix A of GESTAR II. Any additions or differences are given below for each applicable subsection.

4.3.1 Design Bases

4.3.2 Description

4.3.2.1 Nuclear Design Description

The nuclear design description in GESTAR II is referenced in Subsection A.4.3.2.1 of Reference 4.3-1 except for the reference (initial) core loading pattern, which is shown in Figure 4.3-1. The initial core uses barrier fuel bundles of four different average enrichments with natural uranium, non-barrier type bundles on the periphery of the core. These bundles are described in GESTAR II (Reference 4.3-1).

The ABB nuclear design methodology is described in Reference 4.3-3.

4.3.2.2 Power Distribution

Power distribution is referenced in Subsection A.4.3.2.2 of Reference 4.3-1 and section 4.2-3 of reference 4.3-3.

4.3.2.3 Reactivity Coefficient

4.3.2.4 Control Requirements

4.3.2.4.1 Shutdown Reactivity

Information on shutdown reactivity is referenced in Subsection A.4.3.2.4.1 of Reference 4.3-1, section 4.2-4 of Reference 4.3-3, and section 3.1 of Reference 4.3-7 except for the cold shutdown margin for the reference initial core loading pattern, which is given in Table 4.3-3.

4.3.2.4.1.a Local Critical Tests

One test that can be used to verify Shutdown Margin (SDM) is known as a "local critical test". SDM is the quantity of reactivity needed for a reactor core to reach criticality with the strongest-worth control rod fully withdrawn, all other control rods fully inserted, Xenon free, and with a moderator temperature of greater than or equal to 68°F, corresponding to the most reactive state. A local critical test brings a local region of the reactor core to critical by the withdrawal of several diagonally adjacent control rods. To

adequately assess SDM, each test should consist of the withdrawal of the highest-worth control rod, or one of the highest, in the core, as determined by analysis, in conjunction with adjacent control rods to bring the reactor critical. The method for bringing the reactor to critical should be conducted in a prescribed manner to avoid large reactivity insertions near criticality and to ensure that the enthalpy deposition of a dropped control rod does not exceed the 170 cal/gm requirement set for the Control Rod Drop Accident (CRDA). The CRDA analysis must be done with NRC approved methods.

4.3.2.4.2 Reactivity Variations

Information on reactivity variations is referenced in Subsection A.4.3.2.4.2 of Reference 4.3-1. The combined effects of the individual constituents of reactivity are accounted for in each K_{eff} in Table 4.3-3.

4.3.2.5 Control Rod Patterns and Reactivity Worths

Control rod patterns and reactivity worths are discussed in Reference 4.3-1.

4.3.2.6 Criticality of Reactor During Refueling

4.3.2.7 Stability

4.3.2.7.1 Xenon Transients

4.3.2.7.2 Thermal Hydraulic Stability

4.3.2.8 Vessel Irradiations

The reactor vessel neutron fluence calculations were updated using the NRC-approved General Electric Nuclear Energy (GENE) methodology as documented in Licensing Topical Report NEDC-32983P-A, reference 4.3-5. The methodology is fully described in reference 4.3-5. In general, the methodology is consistent with the guidance in Regulatory Guide 1.190, reference 4.3-6, for neutron flux calculations and is based on a two-dimensional discrete ordinates code.

The results of the fluence calculation are provided in Table 4.3-1. The results are based upon operation for 32 EFPY, with 12 EFPY at 3293 MWt, 3 EFPY at 3339 MWt, and the remaining 17 EFPY at 3952 MWt. The GENE methodology was performed for 3952 MWt. The calculated flux value for 3293 MWt is based on the neutron transport calculation described in reference 4.3-4 in which the jet pumps were not modeled. Results from the GENE methodology demonstrated that the neutron transport calculation in reference 4.3-4 provided sufficient conservatism to permit its use for the 3293 MWt. The peak flux for 3339 MWt includes a 1.4% uprate from 3293 MWt. The peak flux for 3952 MWt bounds the extended power uprate of 3840 MWt. The time period of 32 EFPY is based on 40-year operation at an 80% capacity factor.

Table 4.3-1 provides the ¼ T fluence for the various shell sections of the vessel. These fluence values are used in the reactor vessel fracture toughness analyses described in Section 5.3 and Appendix 5A.

4.3.3 Analytical Methods

4.3.4 Changes

4.3.5 References

4.3-1 "General Electric Standard Application for Reactor Fuel," including the "United States Supplement," NEDE-24011-P-A and NEDE-24011-P-A-US, latest revision.

4.3-2 "BWR/4 and BWR/5 Fuel Design," NEDE-20944-1 (Proprietary) and NEDO-20944-1, October 1976, and "Amendment 1," January 1977.

4.3-3 ABB Combustion Engineering Nuclear Power, "Reference Safety Report for Boiling Water Reactor Reload Fuel," CENPD-300-P-A, July 1996.

4.3-4 "Hope Creek 1 Generating Station RPV Surveillance Materials Testing and Fracture Toughness Analysis," GE-NE-523-A164-1294R1, December 1997.

4.3-5 "Licensing Topical Report, General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluation, "NEDC-32983P-A, Rev 1, December 2001.

4.3-6 NRC Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," Rev 0, March 2001.

4.3-7 "GE Hitachi Nuclear Energy Safety Analysis Report to Support Introduction of GE14i Isotope Test Assemblies (ITAs) in Hope Creek Generating Station," NEDC-33529P Revision 0 Class III DRF 0000-0107-3743, December 2009 including Errata and Addenda Number 1 for Hope Creek ITA Safety Analysis Report NEDC-33529P Revision 0, June 10 2010.

TABLE 4.3-1

CALCULATED NEUTRON FLUXES (USED TO EVALUATE VESSEL IRRADIATION)[(2)]

	Projected 32 EFPY, >1 MeV surface fluence, n/cm^2	Projected 32 EFPY, >1 MeV ¼ T [(1)] fluence, n/cm^2
Peak	$1.1x10^{18}$	$7.63x10^{17}$
Lower-intermediate shell (No. 4)	$1.1x10^{18}$	$7.63x10^{17}$
Lower shell (No. 5)	$1.1x10^{18}$	$7.63x10^{17}$
Intermediate shell (No. 3)	$5.3x10^{17}$	$3.68x10^{17}$
LPCI nozzle	$4.7x10^{17}$	$3.26x10^{17}$

(1) T = 6.10 inches for HCGS vessel.

(2) The projected fluences were calculated using the following reactor power levels:

Cycles 1 to 9	3293 MWt	12 EFPY
Cycles 10 to 12	3339 MWt.	3 EFPY
Cycles 13 to End of Life	3952 MWt.	17 EFPY

TABLE 4.3-2

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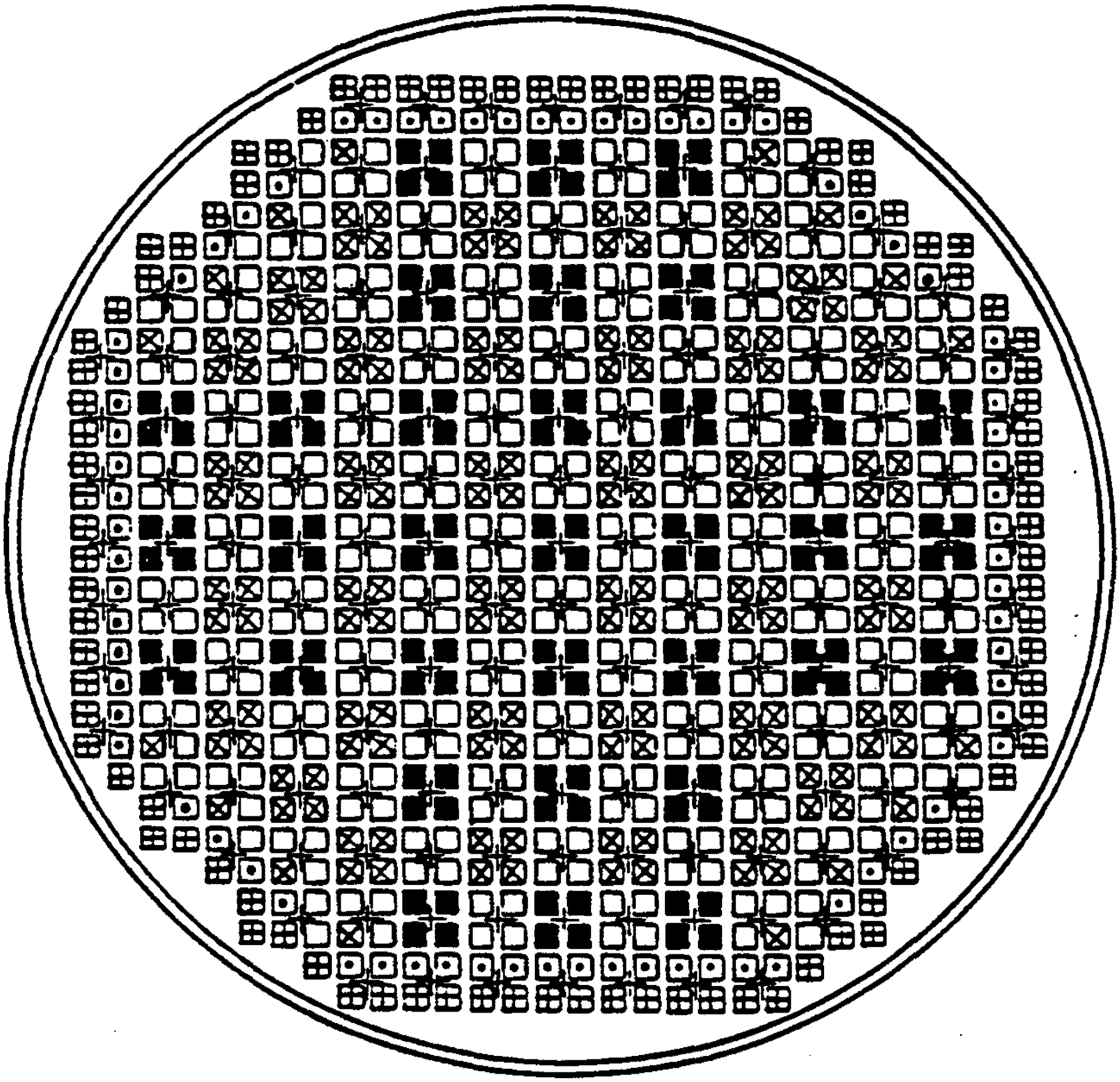
TABLE 4.3-3

CALCULATED CORE EFFECTIVE MULTIPLICATION AND CONTROL SYSTEM WORTH - NO VOIDS, 20°C

Beginning of Cycle-1, K-effective	
Uncontrolled	1.1034
Fully Controlled	0.9233
Strongest Control Rod Out (26-55)	0.9834
R, Maximum Increase in Cold Core Reactivity with Exposure Cycle-1, ΔK	0.0



⊞	= 0.71 wt% U235	(92)*
■	= 0.94 wt% U235	(132)
⊠	= 1.63 wt% U235	(160)
□	= 2.48 wt% U235	(308)
⊡	= 2.78 wt% U235	(72)

* = number of bundles

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APRIL 11, 1988

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

REFERENCE LOADING PATTERN

UPDATED FSAR

FIGURE 4.3-1

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PSEG NUCLEAR L.L.C.
HOPE CREEK GENERATING STATION

THIS FIGURE HAS BEEN DELETED

PSEG NUCLEAR L.L.C.
HOPE CREEK GENERATING STATION

4.4 THERMAL AND HYDRAULIC DESIGN

Most of the information in Section 4.4 for Global Nuclear Fuel is provided in the licensing topical report GESTAR II (Reference 4.4-1). The section numbers in Section 4.4 directly correspond to subsection numbers of Appendix A of GESTAR II. Information on the thermal and hydraulic design for ABB fuel is provided in Reference 4.4-6. The differences are discussed below.

4.4.1 Design Bases

The thermal and hydraulic design bases for GNF fuel are referenced in Section A.4.4.1 of Reference 4.4-1. The thermal and hydraulic design bases for ABB fuel is provided in Section 5.2 of Reference 4.4-6.

4.4.2 Description of Thermal and Hydraulic Design of The Reactor Core

A description of the thermal and hydraulic design of the reactor core for GNF fuel is referenced in Section A.4.4.2 of Reference 4.4-1 and Section 3.1 of Reference 4.4-10. Any additions or differences are given in the appropriate section below. ABB thermal and hydraulic design methods are described in section 5 of Reference 4.4-6.

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

4.4.2.1 Summary

A summary of the thermal and hydraulic design parameters for the initial core is given in Table 4.4-1.

4.4.2.2 Critical Power Ratio

4.4.2.3 Linear Heat Generation Rate

4.4.2.4 Void Fraction Distribution

The core average and maximum exit void fractions in the initial core at rated condition are given in Table 4.4-1. The axial distribution of core void fractions for the average radial channel and the maximum radial channel (end of node value) for the initial core are given in Table 4.4-2. The core average and maximum exit value are also provided. Similar distributions for steam quality are provided in Table 4.4-3. The core average axial power distribution used to produce these tables is given in Table 4.4-4.

4.4.2.5 Core Coolant Flow Distribution and Orificing Pattern

4.4.2.6 Core Pressure Drop and Hydraulic Loads

4.4.2.7 Correlation and Physical Data

GNF has obtained substantial amounts of physical data in support of the pressure drop and thermal hydraulic loads. This information is given in Appendix B of Reference 4.4-1 where responses are provided to NRC questions on Section 4 of GESTAR II.

4.4.2.8 Thermal Effects of Operational Transients

4.4.2.9 Uncertainties in Estimates

4.4.2.10 Flux Tilt Considerations

The inherent design characteristics of the BWR are particularly well suited to handle perturbations due to flux tilt. The stabilizing nature of the moderator void coefficient effectively damps oscillations in the power distribution. In addition to this damping, the in-core instrumentation system and the associated on-line computer provide the operator with prompt and reliable power distribution information. Thus, the operator can readily use control rods or other means to limit effectively the undesirable effects of flux tilting. Because of these features

and capabilities, it is not necessary to allocate a specific peaking factor margin to account for flux tilt. If, for some reason, the power distribution could not be maintained within normal limits using control rods, then the operating power limits would have to be reduced as prescribed in Section 16.

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 (RCS) is described in this section.

4.4.3.1 Plant Configuration Data

4.4.3.1.1 Reactor Coolant System Configuration

The RCS is described in Section 5.4 and is shown in isometric perspective on 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 flow path in the RCS are shown on 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 on 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 RCS.

Table 4.4-6 provides the lengths and sizes of all safety injection lines to the RCS.

4.4.3.2 Operating Restrictions on Pumps

Expected recirculation pump performance curves are shown on Vendor Technical Document PN1-B31-C001-0031. These curves are valid for all conditions with a normal operating range varying from approximately 20 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 BWR must operate with certain restrictions because of pump NPSH, overall plant control characteristics, core thermal power limits, etc. The power flow map for the power range of operation is shown on Figure 4.4-1. The nuclear system equipment, nuclear instrumentation, and the Reactor Protection System (RPS), 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:

1. 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.

2. Recirculation minimum pump constant speed line, B - Startup operations of the plant are normally carried out with the recirculation pumps operating between the Lower Control Limit (≈20% speed), and the speed set by the #1 Speed Limiter (≈30% speed). The operating state for the reactor follows this line for the normal control rod withdrawal sequence.

3. Cavitation protection line - This line (minimum power line) results from the recirculation pump and jet pump NPSH requirements. The recirculation pumps are automatically switched to the #1 Speed Limiter, when the feedwater flow drops below a preset value.

Extended regions have been analyzed to allow flexibility in low xenon startups, fuel depletion compensation at full power, and fuel cycle extension at the end of the normal fuel cycle. These boundaries on the power flow map are as follows:

4. Maximum Extended Load Line Limit Analysis (MELLLA) Region - The MELLLA performed in Reference 4.4-8 allows operation above the 100-percent load line (100-percent rod line) and up to the MELLLA boundary line. During relatively low xenon startups, this will allow control rods to be pulled above the normal 100-percent load line, thus limiting further control rod manipulation near 100-percent power as xenon builds in.

5. Increased Core Flow (ICF) Region - The ICF analysis performed in Reference 4.4-8 allows operation above 100-percent core flow and up to 105-percent core flow. This extra operating domain can be used to compensate for fuel depletion beyond the normal end of cycle. This can be used as an alternative to end-of-cycle power coastdown to extend full-power capability.

Combined, these extra operating regions allow flexibility in quickly attaining rated power with generally fewer rod pattern adjustments. This is mostly due to the ability to attain the rated power condition at a core flow anywhere from 94.8 to 105 percent of rated core flow. Small compensations for fuel depletion also may be achieved at full power with flow adjustments only. Minimizing control rod movements at or near rated power also is desirable in minimizing fuel pellet-cladding interaction and associated fuel failures. The impact of extended operation in these regions on the transient analyses will need to be evaluated on a cycle-specific basis.

4.4.3.3.1.1 Performance Characteristics

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

1. Recirculation pump constant speed lines, C - These are lines parallel to the minimum pump speed up to the increased core flow condition. These lines represent the change in flow associated with power changes while maintaining constant recirculation pump speed.

2. Constant rod lines - These lines show the change in power associated with flow changes while maintaining constant control rod position (e.g., 86.95 percent rod line, etc). These lines are based on constant xenon concentration.

3. Rated flow control line – The rated flow control line (100 percent rod line) passes through 100 percent power (3840 MWt at 100 percent flow.

4.4.3.3.2 Regions of the Power Flow Map

The power-flow map for the normal power range of operation includes the Extended Power Uprate (EPU), Maximum Extended Load Line Limit Analysis (MELLLA), and Increased Core Flow (ICF) conditions as shown in Figure 4.4-1. The extended operation domain was evaluated in Reference 4.4-8, this extended operating domain includes the maximum upper boundary load line limit (MELLLA) and increased core flow regions as indicated in Figure 4.4-1. The upper load line passes through 100% reactor thermal power (3840 MWt) at 94.8% core flow. The nuclear steam system equipment, nuclear instrumentation and Reactor Protection System (RPS), in conjunction with operating procedures, maintain operations within the area of this map for normal operating conditions. Main regions of the map are discussed below to clarify operational capabilities.

1. Region I - This is the transition region between natural circulation operation and operation at minimum pump speed (~20%). Steady state conditions cannot exist in this area because the recirculation pumps cannot be operated below the minimum pump speed. Normal startup is between the Lower Control Limit (~20%), and the speed set by the #1 Speed Limiter (30%).

2. Region II – MELLLA, MINIMUM PUMP SPEED, CAVITATION INTERLOCK, MAXIMUM CORE FLOW AND RATED CORE POWER LINES: This map region represents the normal operating zone where power changes can be made, either by control rod movement or by core flow changes, by changing recirculation pump drive speed.

3. Region III - This map region represents the low power area where cavitation can be expected in the recirculation pumps and in the jet pumps. Operation within this region is precluded by system interlocks that trip ("Runback") the recirculation pumps to the speed set by the #1 Speed Limiter (30%), whenever feedwater flow is less than a preset value (typically 20 percent of rated flow).

4.4.3.4 Temperature Power Operating Map

This section is not applicable to the Hope Creek Generating Station (HCGS).

4.4.3.5 Load Following Characteristics

The following simple description of BWR operation with recirculation flow control summarizes the principal modes of normal power range operation. Assuming the plant to be initially hot with the reactor critical, full power operation can be approached following the sequence shown as points 1 to 7 on Figure 4.4-1. The first part of the sequence, points 1 to 3, is achieved with control rod withdrawal and manual control of individual recirculation pumps. Individual pump startup procedures are provided that achieve 20% to 30% of full pump speed in each loop. Power, steam flow, and feedwater flow are increased as control rods are manually withdrawn until the feedwater

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flow has reached approximately 20 percent. The #1 Speed Limiter prevents low power, high recirculation flow combinations that create recirculation pump and jet pump NPSH problems.

Reactor power increases as the operating state moves from point 2 to point 3, due to the inherent flow control characteristics of the BWR. Once the feedwater interlock is cleared, the operator can manually increase recirculation flow in each loop until the operating state reaches point 3, the lower limit of the flow control range. At point 3, the operator can switch to simultaneous recirculation pump control. Thermal output can then be increased by either control rod withdrawal or recirculation flow increase. For example, the operator can increase power in the ways indicated by points 4 or 5. With an increase of recirculation flow to rated flow, point 4 can be achieved. If, however, it is desired to maintain low recirculation flow, increased power can be achieved by withdrawing control rods until point 5 is reached. The recirculation system master controller is limited, and these limits establish the operating state, as discussed in Section 7.7. The operating map is shown on Figure 4.4-1 with the designated expected flow control range.

The curves labeled rod lines represent typical steady state power flow characteristics for fixed rod patterns. They are slightly affected by xenon, core leakage flow assumptions (Reference 4.4-1), and reactor vessel pressure variations. However, these effects have been neglected in the rod lines shown.

Normal power range operation is along or below the MELLLA Boundary line. For load increase, plant operation from point 5 to point 6 is controlled by increasing core flow. As xenon builds in or negative reactivity reduces core power, 100% power can be maintained by increasing core flow all the way up to point 7, maximum core flow limit.

The large negative operating reactivity and power coefficients are inherent in the BWR. They provide important advantages as follows:

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

2. Load following with recirculation flow control

3. Strong damping of spatial power disturbances.

Design of the single cycle BWR plant includes the ability to follow load demand over a reasonable range without requiring operator action. This load following capability is accomplished by automatic variation of reactor recirculation flow. The plant responds to ramp load changes at a rate of up to 30 percent per minute without changes in control rod settings.

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

Varying the recirculation flow rate (flow control) is more advantageous, relative to load changes, than using control rod positioning. Flow variations perturb the reactor uniformly in the

horizontal planes and ensure a flatter power distribution and reduced transient allowances. As flow is varied, the power and void distributions remain approximately constant at the steady state end points for a wide range of flow variations. After adjusting the power distribution by positioning the control rods at a reduced power and flow, the operator can then bring the reactor to rated conditions by increasing flow, with the power distribution remaining approximately constant. Section 7.7 describes how recirculation flow is varied.

4.4.3.6 Thermal and Hydraulic Characteristics Summary Table

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

4.4.4 Evaluation

See Section A.4.4.4 of Reference 4.4-1. The results of the cycle 1 stability analysis are given in Table 4.4-7 and Figures 4.4-2 through 4.4-5.

4.4.5 Testing and Verification

See Section A.4.4.5 of Reference 4.4-1 for testing and verification.

4.4.6 Instrumentation Requirements

The reactor vessel instrumentation monitors the key reactor vessel operating parameters during planned operations. This ensures sufficient control of the parameters. The reactor vessel sensors are discussed in Sections 7.5, 7.6, and 7.7.

4.4.7 SRP Rule Review

Acceptance criterion II.8 of SRP Section 4.4 specifies, in part, that the effects of crud should be accounted for in the thermal hydraulic design, and also that process monitoring provisions be capable of detecting a three percent pressure drop in the reactor coolant flow.

In general, the critical power ratio (CPR) is not affected as crud accumulates on fuel rods (References 4.4-2 and 4.4-3). Therefore, no modifications to GEXL are made to account for crud deposition. For pressure drop considerations, the amount of crud assumed to be deposited on the fuel rods and fuel rod spacers is greater than is actually expected at any point in the fuel lifetime. This crud deposition is reflected in a decreased flow area, increased friction factors, and increased spacer loss coefficients, the effect of which is to increase the core pressure drop by approximately 1.7 psi, an amount which is large enough to be detected in monitoring of core pressure drop. It should be noted that assumptions made with respect to crud deposition in core thermal hydraulic analyses are consistent with established water chemistry requirements. More detailed discussion of crud (service induced variations) and its uncertainty is found in Section III of Reference 4.4-4.

4.4.8 References

4.4-1 General Electric Standard Application for Reactor Fuel," including the "United States Supplement," NEDE-24011-P-A and NEDE-24011-P-A-US, latest revision.

4.4-2 R.V. McBeth, R. Trenberth, and R. W. Wood, "An Investigation Into the Effects of Crud Deposits on Surface Temperature, Dry Out, and Pressure Drop, with Forced Convection Boiling of Water at 69 Bar in an Annular Test Section," AEEW-R-705, 1971.

4.4-3 S.J. Green, B.W. LeTourneau, and A.C. Peterson, "Thermal and Hydraulic Effects of Crud Deposited on Electrically Heated Rod Bundles," WAPD-TM-918 September 1970.

4.4-4 General Electric, “General Electric Thermal Analysis Basis (GETAB): Data, Correlation, and Design Application,” NEDO-10958A, January 1977.

4.4-5 General Electric, "Increased Core Flow and Extended Load Line Limit Analysis for Hope Creek Generating Station Unit 1, Cycle 2," NEDC-31487, November 1987.

4.4-6 ABB Combustion Engineering Nuclear Power, “Reference Safety Report for Boiling Water Reactor Reload Fuel, “ CENPD-300-P-A, July 1996.

4.4-7 Deleted

4.4-8 NEDC-33066P, Revision 2, “Hope Creek Generating Station, APRM/RBM/Technical Specifications/ Maximum Extended Load Line Limit Analysis (ARTS/MELLLA),” [Feb. 2005].

4.4-9 NEDC-33076P, Revision 2, “Safety Analysis Report For Hope Creek Constant Pressure Power Uprate”, [August 2006].

4.4-10 “GE Hitachi Nuclear Energy Safety Analysis Report to Support Introduction of GE14i Isotope Test Assemblies (ITAs) in Hope Creek Generating Station,” NEDC-33529P Revision 0 Class III DRF 0000-0107-3743, December 2009 including Errata and Addenda Number 1 for Hope Creek ITA Safety Analysis Report NEDC-33529P Revision 0, June 10 2010.

TABLE 4.4-1

THERMAL AND HYDRAULIC DESIGN CHARACTERISTICS
OF THE REACTOR CORE TYPICAL VALUES BASED ON Cycle 1

General Operating Conditions

Reference design thermal output, MWt	3293
Power level for engineered safety features, MWt	3436
Steam flow rate, at 419.9°F final feedwater temperature (FFWT), millions lb/h	14.159
Core coolant flow rate, millions lb/h	100.0
Feedwater flow rate, millions lb/h	14.127
System pressure, nominal in steam dome, psia	1020
System pressure, nominal core design, psia	1035
Coolant saturation temperature at core design pressure, °F	549

Core inlet enthalpy, at 419.9°F FFWT, Btu/lb	526.1
Core inlet temperature, at 419.9°F FFWT, °F	531.6
Core maximum exit voids within assemblies, percent	77.1
Core average void fraction, active coolant	0.419
Maximum fuel temperature, °F	3435
Active coolant flow area per assembly, in.^{2}	15.824
Core average inlet velocity, ft/s	6.41
Maximum inlet velocity, ft/s	6.803
Total core pressure drop, psi	21.25

Core support plate pressure drop, psi	16.82
Average orifice pressure drop	
Central region, psi	7.16
Peripheral region, psi	14.53
Maximum channel pressure loading, psi	10.88
Average power assembly channel pressure loading (bottom), psi	9.61
Shroud support ring and lower shroud pressure loading, psi	22.87
Upper shroud pressure loading, psi	6.05

TABLE 4.4-2

VOID DISTRIBUTION[1]

TYPICAL VALUES BASED ON CYCLE 1

	Node	Core Average (Average Node Value)	Maximum Channel (End of Node Value)
Bottom of Core	1	0.	0.
	2	0.000	0.013
	3	0.010	0.084
	4	0.047	0.192
	5	0.111	0.295
	6	0.187	0.380
	7	0.264	0.453
	8	0.333	0.510
	9	0.390	0.556
	10	0.436	0.593
	11	0.473	0.622
	12	0.504	0.647
	13	0.529	0.667
	14	0.550	0.683
	15	0.567	0.698
	16	0.583	0.710
	17	0.597	0.722
	18	0.610	0.733
	19	0.622	0.744
	20	0.633	0.753
	21	0.643	0.761
	22	0.651	0.766
	23	0.656	0.770
Top of Core	24	0.659	0.771

(1) Core average value = 0.419

Maximum exit value = 0.771

Active fuel length = 150 inches

TABLE 4.4-3

FLOW QUALITY DISTRIBUTION[1]

TYPICAL VALUES BASED ON CYCLE 1

	Node	Core Average (Average Node Value)	Maximum Channel (End of Node Value)
Bottom of Core	1	0.	0.
	2	0.	0.
	3	0.000	0.004
	4	0.001	0.013
	5	0.005	0.027
	6	0.011	0.045
	7	0.020	0.065
	8	0.031	0;086
	9	0.043	0.107
	10	0.054	0.127
	11	0.066	0.146
	12	0.076	0.164
	13	0.086	0.180
	14	0.095	0.195
	15	0.104	0.209
	16	0.112	0.222
	17	0.120	0.236
	18	0.127	0.249
	19	0.135	0.263
	20	0.143	0.275
	21	0.150	0.286
	22	0.155	0.294
	23	0.159	0.300
Top of Core	24	0.161	0.301

(1) Core average value = 0.077
Maximum exit value = 0.301
Active fuel length = 150 in.

TABLE 4.4-4

AXIAL POWER DISTRIBUTION USED TO GENERATE

VOID AND QUALITY DISTRIBUTIONS

TYPICAL VALUES BASED ON CYCLE 1

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

TABLE 4.4-5

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 (ft^2)
1. Lower plenum	216.5	216.5 216.5	-161.5	92.5
2. Core	163.0	163.0 163.0	55.0	152.0 (includes bypass)
3. Upper plenum and separators	185.0	185.0 185.0	217.5	45.0
4. Dome (above normal water level)	299.5	299.5 0	402.5	352.0
5. Downcomer area	311.0	311.0 311.0	-30.0	118.0
6. Recirculation loops and jet pumps	97.0 ft (one loop)	492.0 492.0	-472.5	77.5 $in.^2$ (one loop)

---

(1) The reference point is the recirculation nozzle outlet centerline.

TABLE 4.4-6

SAFETY INJECTION LINE LENGTHS

Injection Line	Line O.D. (in.)	Line Length
High Pressure Coolant Injection (HPCI)[2]		
Pump discharge to valve F006[1]	14	203 ft
Inside containment to reactor pressure vessel (RPV)	12	104 ft - 6 in.
Low Pressure Coolant Injection (LPCI)[3]		
Loop A:		
Pump discharge to valve F017A[1]	18	208 ft - 9 in.
Inside containment to RPV	12	71 ft - 11 in.
Loop B:		
Pump discharge to valve F017B[1]	18	195 ft - 5 in.
Inside containment to RPV	12	71 ft - 4 in.

Injection Line	Line O.D. (in.)	Line Length
Loop C:		
Pump discharge to valve F017C(1)	18	200 ft - 9 in.
Inside containment to RPV	12	79 ft - 1 in.
Loop D:		
Pump discharge to valve F017D(1)	18	201 ft - 6 in.
Inside containment to RPV	12	94 ft - 8 in.
Core Spray(4)		
Loop A:		
Pump A discharge to common discharge	12	33 ft
Pump C discharge to common discharge	12	27 ft - 2 in.
Common discharge to valve F005A(1)	14	131 ft - 10 in.
Inside containment to RPV	12	79 ft - 7 in.

TABLE 4.4-6 (Cont)

Injection Line	Line O.D. (in.)	Line Length
Loop B:		
Pump B discharge to common discharge	12	29 ft - 4 in.
Pump D discharge to common discharge	12	26 ft - 8 in.
Common discharge to valve F005B[1]	14	159 ft
Inside containment to RPV	12	75 ft - 4 in.

(1) Valve located as close as possible to outside of containment wall

(2) See Figures 6.3-1 and 6.3-2.

(3) See Figure 5.4-13.

(4) See Figure 6.3-7.

TABLE 4.4-7

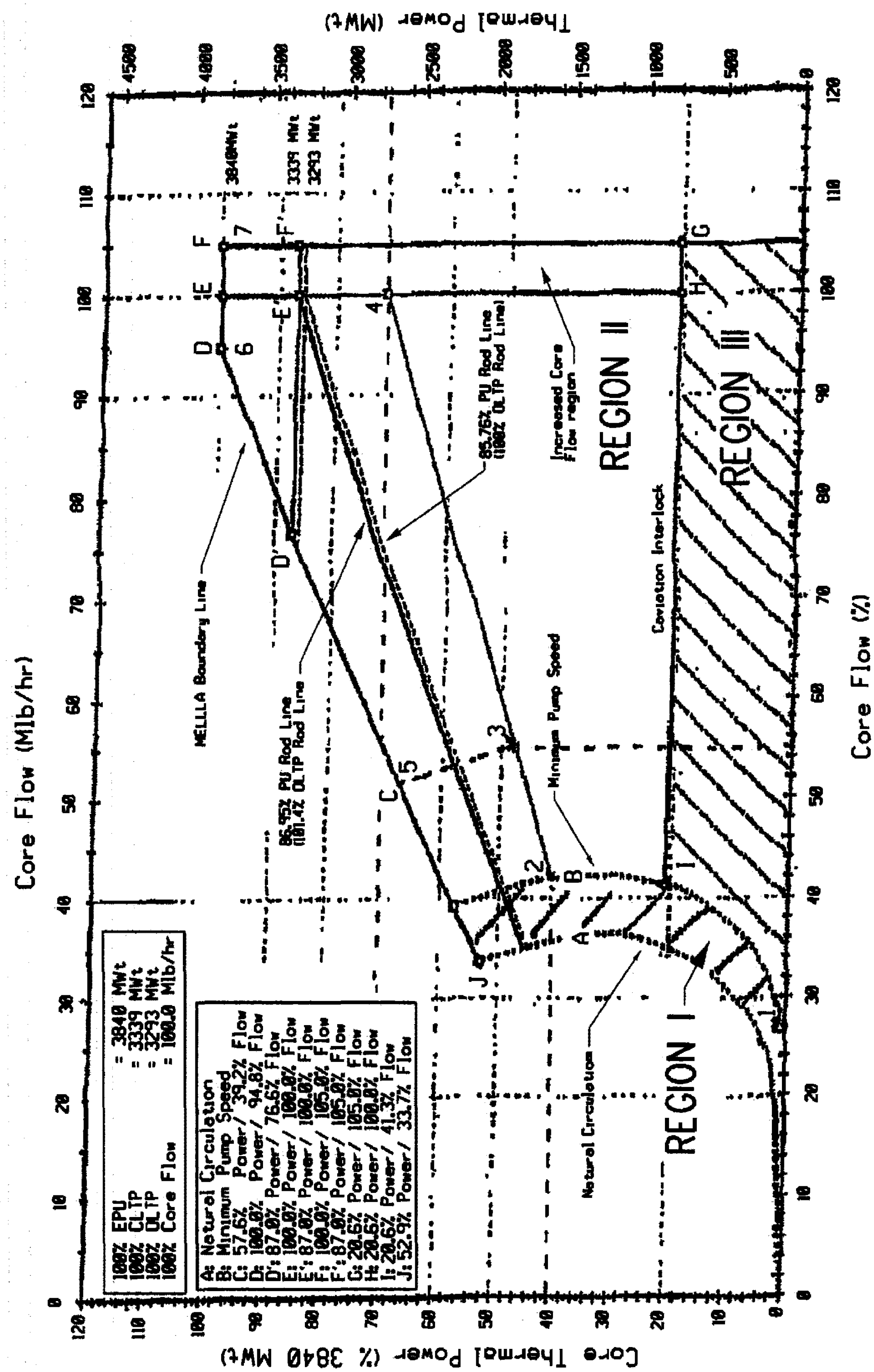
STABILITY ANALYSIS RESULTS
(Cycle-1 Most Limiting Conditions)

Rod Line Analyzed

Natural circulation	51.5 percent rated power
Rod pattern	105.0 percent rated thermal power

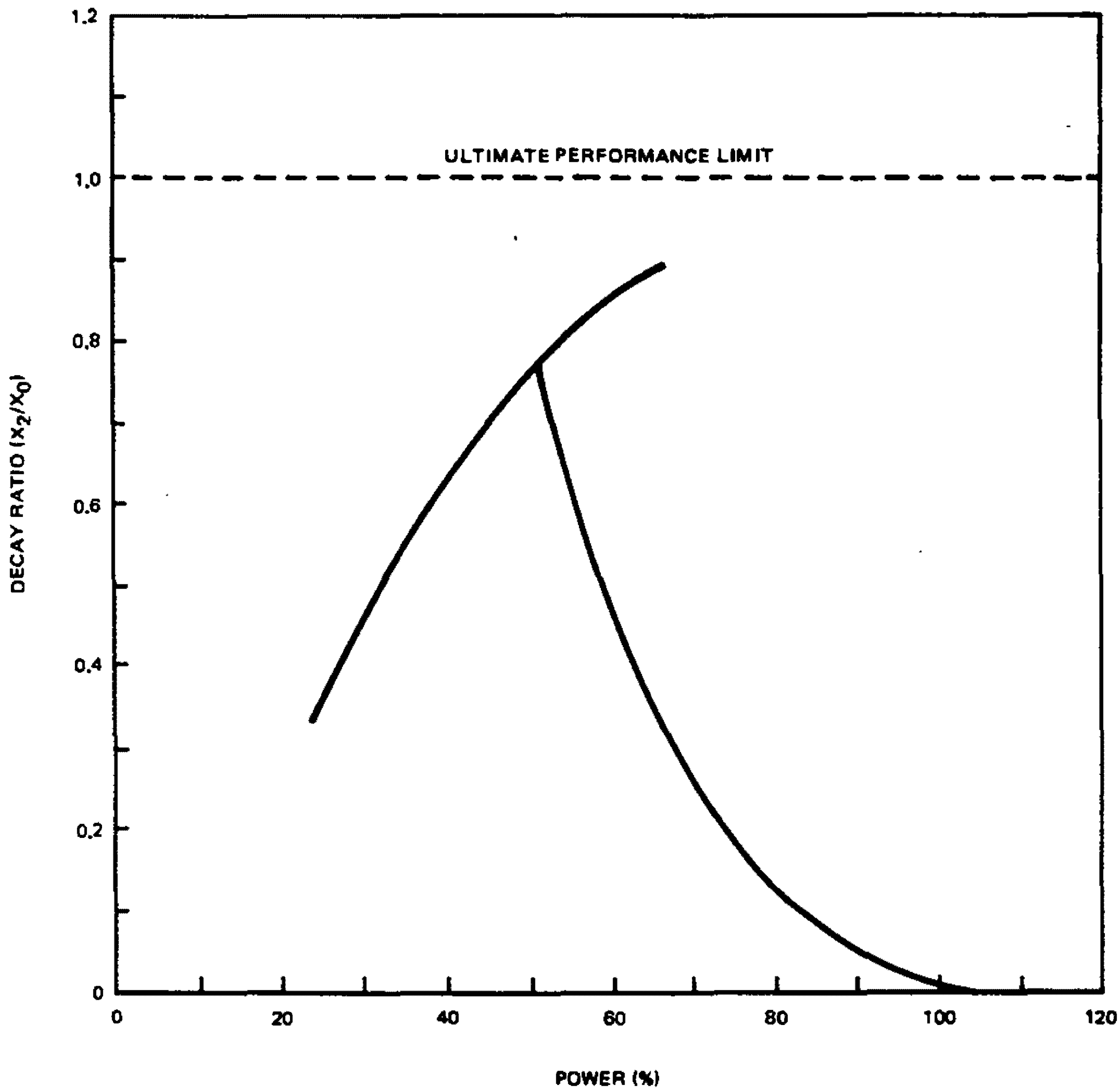
Decay Ratio

Total system stability, x_2/x_0	- See Figures 4.4-8 through 4.4-10
Reactor core stability, x_2/x_0	- 0.78 (0.32 Hz resonant frequency)
Channel hydrodynamic performance, x_2/x_0	- 0.54 (0.36 Hz resonant frequency)



Revision 17, June 23, 2009

PSEG Nuclear, LLC HOPE CREEK NUCLEAR GENERATING STATION	Hope Creek Nuclear Generating Station POWER - FLOW OPERATING MAP Updated FSAR Figure 4.4-1



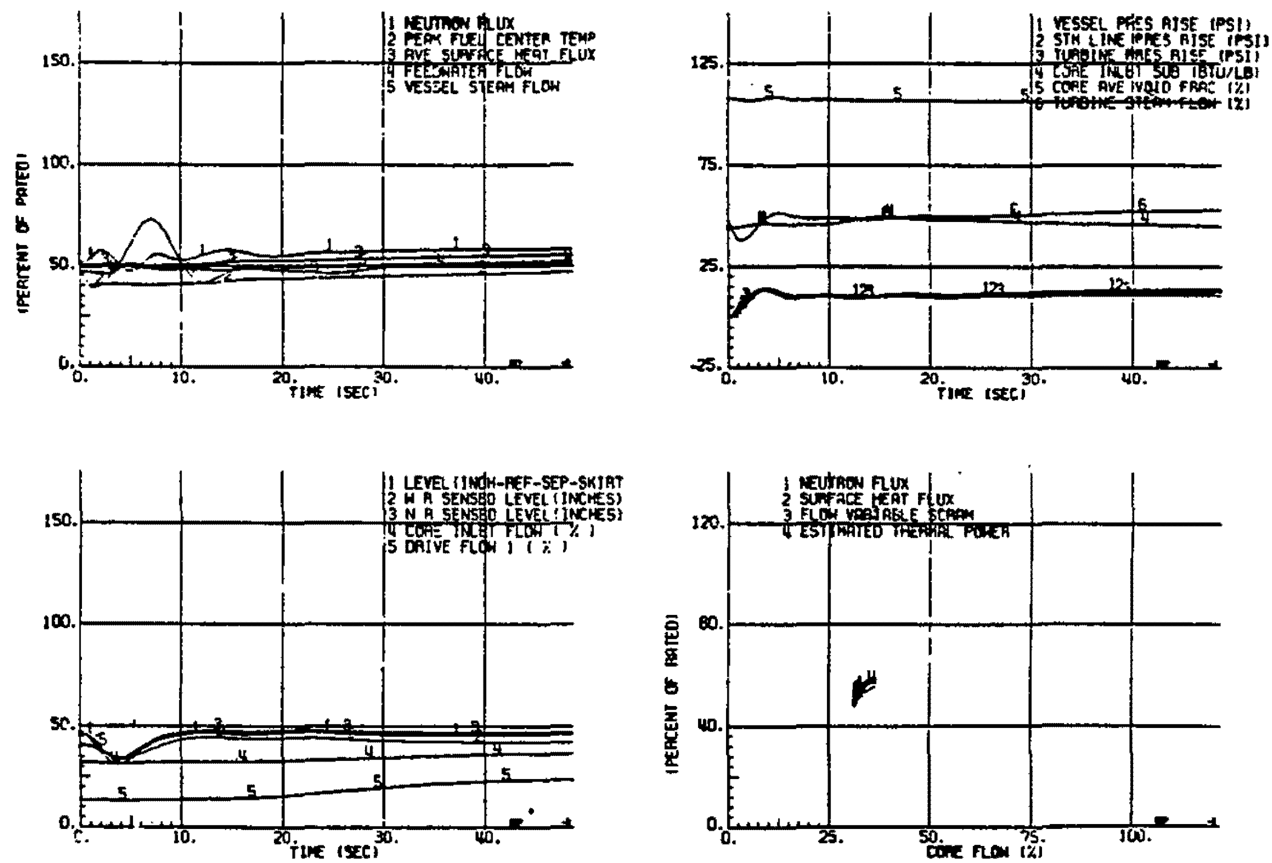
REVISION 0
APRIL 11, 1988

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

TOTAL CORE STABILITY

UPDATED FSAR

FIGURE 4.4-2



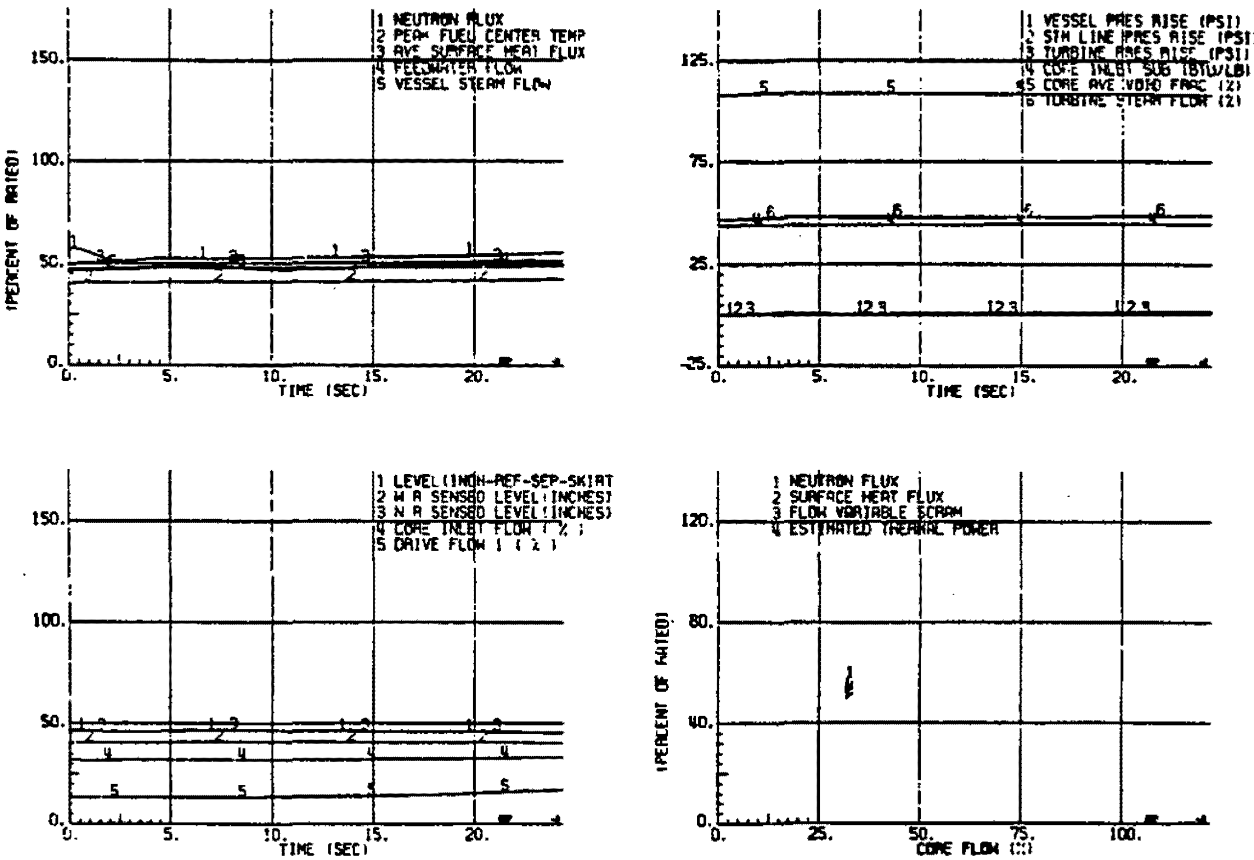
REVISION 0
APRIL 11, 1988

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

10 PSI PRESSURE REGULATOR SETPOINT STEP AT 51.5% OF RATED POWER (NATURAL CIRCULATION)

UPDATED FSAR

FIGURE 4.4-3



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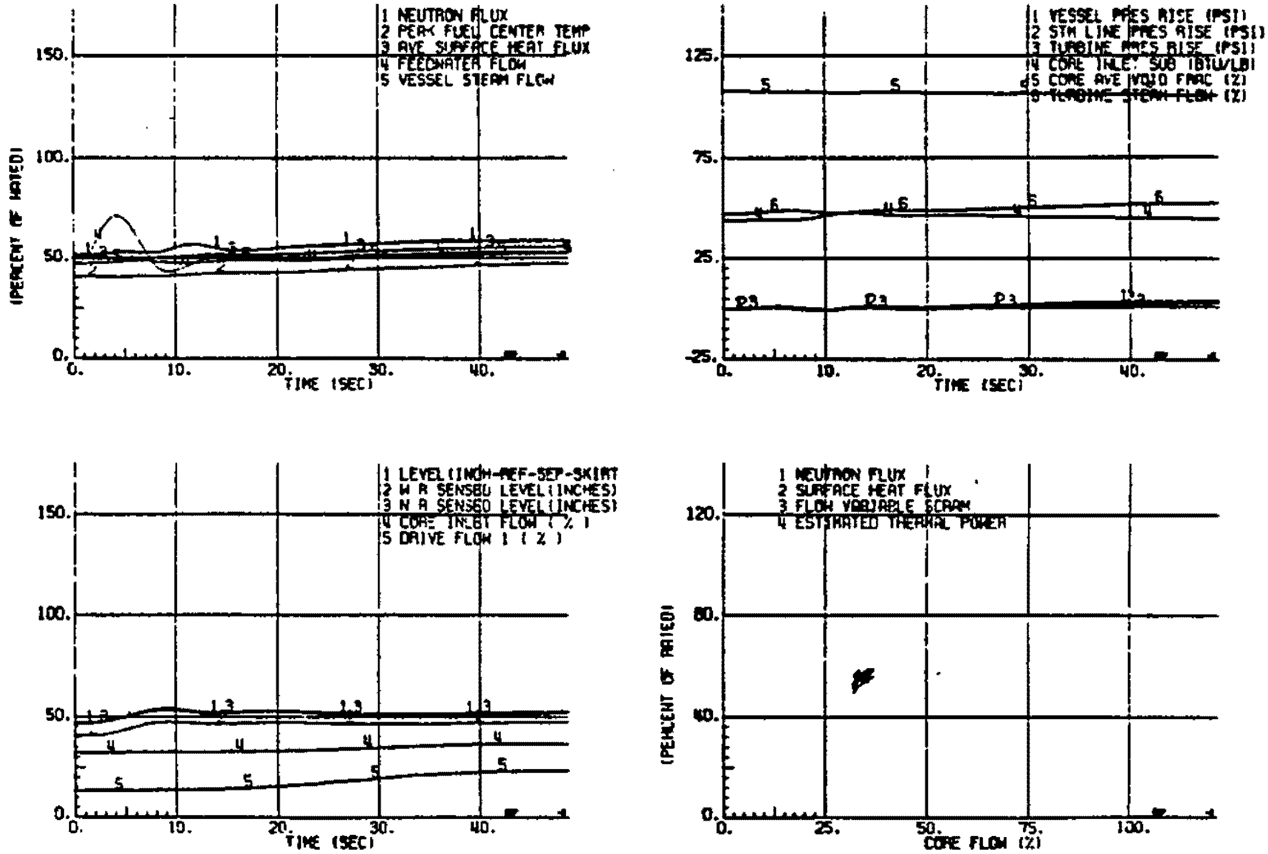
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

10 CENT REACTIVITY STEP
AT 51.5% OF RATED POWER
(NATURAL CIRCULATION)

UPDATED FSAR

FIGURE 4.4-4



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APRIL 11, 1988

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

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

UPDATED FSAR

FIGURE 4.4-5

4.5 REACTOR MATERIALS

4.5.1 Control Rod Drive System Structural Materials

4.5.1.1 Materials Specifications

4.5.1.1.1 Material List

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

1. Cylinder, tube, and flange assembly:

Flange	ASME SA182 Grade F304
Plugs	ASME SA182 Grade F304
Cylinder	ASTM A269 Grade TP 304
Outer tube	ASTM A269 Grade TP 304
Tube	ASME SA351 Grade CF3
Spacer	ASME SA351 Grade CF3

2. Piston tube assembly:

Piston tube	ASME SA249 Grade XM-19, or ASME SA479 Grade XM-19
Stud	ASME SA479 Grade XM-19
Head	ASME SA182 Grade F304

Indicator tube	ASME SA312 Grade TP 316
Cap	ASME SA182 Grade F304

3. Drive line assembly:

Coupling spud	Inconel X-750
Index tube	ASME SA249 Grade XM-19, or ASME SA479 Grade XM-19
Piston head	ARMCO 17-4 PH
Coupling	ASME SA312 Grade TP 304, or ASTM A511 Grade MT 304
Magnet housing	ASME SA312 Grade TP 304, or ASTM A511 Grade MT 304

4. Collet assembly:

Collet piston	ASTM A269 Grade TP 304, or ASME SA312 Grade TP 304
Finger	Inconel X-750
Retainer	ASTM A269 Grade TP 304, or ASTM A511 Grade MT 304
Guide cap	ASTM A269 Grade TP 304

5. Miscellaneous parts:

Stop piston	ASTM A276 Type 304
Connector	ASTM A276 Type 304

O-Ring spacer	ASME SA240 Type 304
Nut	ASME SA193 Grade B8
Barrel	ASTM A269 Grade TP 304, ASME SA312 Grade TP 304, or ASME SA240 Type 304
Collet spring	Inconel X-750
Ring flange	ASME SA182 Grade F304

The austenitic material 300 series stainless steels listed above 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 to 125,000 psi, a yield strength of 50,000 to 85,000 psi, and a minimum elongation of 25 percent.

The coupling spud, collet fingers, and collet spring are fabricated from Inconel X-750 in the annealed or equalized condition, and aged 20 hours at 1300°F to produce a tensile strength of 165,000 psi minimum, a yield strength of 105,000 psi minimum, and an elongation of 20 percent minimum. The piston head is ARMCO 17-4 PH in condition H-1100 (aged 4 hours at 1100°F), with a tensile strength of 140,000 psi minimum, a yield strength of 115,000 psi minimum, and an elongation of 15 percent minimum.

These are widely used materials, whose properties are well known. The parts are readily accessible for inspection and replacement if necessary. All materials have been successfully used for the past 10 to 15 years in similar drive mechanisms.

4.5.1.1.2 Special Materials

No cold worked austenitic stainless steels with a yield strength greater than 90,000 psi are employed in the CRD system. Hardenable martensitic stainless steels are not used. ARMCO 17-4 PH (precipitation hardened stainless steel) is used for the piston head. This material is aged to the H-1100 condition to produce resistance to stress corrosion cracking in the boiler water reactor (BWR) environment. ARMCO 17-4 PH (H-1100) has been successfully used for the past 20 years in BWR drive mechanisms.

4.5.1.2 <u>Austenitic Stainless Steel Components</u>

4.5.1.2.1 Processes, Inspections, and Tests

Solution annealed 300 series stainless steel material used in the fabrication of CRD parts is verified as being correctly solution annealed by testing in accordance with ASTM A262, Recommended Practices for Detecting Susceptibility to Intergranular Attack in Stainless Steels.

Two special processes are employed that subject selected 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 following components are nitrided to provide a wear resistant surface:

    a. Piston tube (piston tube assembly)

    b. Index tube (drive line assembly)

    c. Collet piston and guide cap (collet assembly).

Colmonoy hard surfacing is applied on the cylinder, spacer, and retainer by the flame spray process.

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

Colmonoy hard surfaced components have performed successfully for the past 20 years in drive mechanisms. Nitrided components have been used in CRDs since 1967. It is normal practice to remove some CRDs at each refueling outage. At this time, both the Colmonoy hard surfaced parts and the 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 can become serious enough to cause operating problems.

Welding is performed in accordance with Section IX of the ASME B&PV Code. Heat input for stainless steel welds is restricted to a maximum of 50,000 Joules per inch and an interpass temperature of 350°F. These controls are employed to avoid severe sensitization and comply with the intent of Regulatory Guide 1.44.

4.5.1.2.2 Control of Delta Ferrite Content

All type 308 weld metal is purchased to a specification that 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.

An extensive test program performed by GE, with the concurrence of the regulatory staff, has demonstrated that controlling weld filler metal ferrite at 5 percent minimum produces production welds that meet the requirements of Regulatory Guide 1.31, Control of Stainless Steel Welding. A total of approximately 400 production welds in five BWR plants were measured and all welds met the

requirements of the interim regulatory position on, Regulatory Guide 1.31. For general compliance or alternate approach assessment for Regulatory Guide 1.31, see Section 4.5.2.4.1.

4.5.1.3 <u>Other Materials</u>

These are discussed in Section 4.5.1.1.2.

4.5.1.4 <u>Cleaning and Cleanliness Control</u>

4.5.1.4.1 Protection of Materials During Fabrication, Shipping, and Storage

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

1. Any processing that increases part temperature above 200°F

2. Assembly that results in a decrease of accessibility for cleaning

3. Release of parts for shipment.

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

1. The drive is rinsed in hot deionized water and dried in preparation for shipment. The ends of the drive are then covered with a vapor tight barrier with desiccant. Packaging is designed to protect the drive and prevent damage to the vapor barrier. The planned storage period

considered in the design of the container and packaging is 2 years. This packaging has been qualified and in use for a number of years. Periodic audits have indicated satisfactory protection.

2. The degree of surface cleanliness obtained by these procedures meets the requirements of Regulatory Guide 1.37. For general compliance or alternate approach assessment for Regulatory Guide 1.37, see Section 4.5.2.4.6.

3. Site or warehouse storage specifications require inside heated storage comparable to level B of ANSI N45.2.2.

4. After the second year, a yearly inspection of 2 percent of the units, or a semi-annual examination of the humidity indicators of 10 percent of the units is required to verify that the units are dry and in satisfactory condition. This inspection is performed with a GE representative present. Position indicator probes are not subject to this inspection.

4.5.1.5 SRP Rule Review

Acceptance criterion II.1 of SRP Section 4.5.1 specifies, in part, that the properties of the materials selected for the CRD mechanism must be equivalent to those given in Appendix I to Section III of the ASME B&PV Code, or Parts A and B of Section III of the Code, or are included in Regulatory Guide 1.85, Code Case Acceptability ASME Section III Materials.

Only those parts of the CRD forming part of the primary pressure boundary are Code material (i.e., require ASME specification for material certification). Jurisdiction of Section III of the Code does not extend to the nonpressurized parts of the CRD. ASME materials are identified in the material list of Section 4.5.1.1.

4.5.2 Reactor Internals Materials

4.5.2.1 Materials Specifications

Materials used for the core support structure are as follows:

1. Shroud support - All material is plate: if nickel chromium iron alloy, ASME SB168; if clad carbon steel, ASME SA516; if clad low alloy steel, ASME SA533 or SA508, Class 2

2. Shroud, core plate, and top guide - ASME SA240, SA193, SA194, SA182, SA479, SA312; ASTM A276, A249, A213 (Types 304L or 304)

3. Peripheral fuel support - ASME SA312 Grade TP 304

4. Core plate studs and nuts - ASME SA182, ASTM A276 (both Type 304)

5. Control rod drive (CRD) housing - ASME SA312 Grade TP 304, SA182 Grade F304

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

7. Orificed fuel support - ASME SA351 Type CF8.

Materials used in other reactor internal structures are as follows:

1. Steam separator and steam dryer - All materials are Type 304 or 316L stainless steel:

   Plate, sheet, and strip - ASTM A240 Type 304 or 316L

   Forgings - ASTM A182 Grade F304

Bars	-	ASTM A276 Type 304
Pipe	-	ASME SA312 Grade TP 304
Tubing	-	ASTM A269 Grade TP 304
Castings	-	ASTM A351 Grade CF8

2. Jet pump assemblies - The components in the jet pump assemblies are a riser, riser brace, inlet mixer, diffuser, adaptor, and brackets. Materials used for these components are to the following specifications:

Castings	-	ASTM A351 Grade CF8
Bars	-	ASTM A276 Type 304
Bolts	-	ASTM A193 Grade B8 or B8M
Nuts	-	ASTM A194 Grade 8 or 8M
Sheet and plate	-	ASTM A240 Type 304 and 304L
Tubing	-	ASTM A269 Grade TP 304
Pipe	-	ASTM A358 Type 304 and ASTM A312 Grade TP 304
Weld Coupling	-	ASTM A403 Grade WP 304
Forging	-	ASTM A182 Grade F304
Inconel forgings	-	ASTM B166

Materials in the jet pump assemblies that are not Type 304 stainless steel are listed below:

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

b. The adaptor 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. The Beam is made to Specification ASTM A637 Grade 688.

d. The jet pump beam bolt is ASTM A276 Type 304, or ASME SA479 Type 316L SHT.

e. The jet pump sensing line clamps are Type 316L and Type 308L stainless steels.

3. Core spray spargers and core spray lines - Materials used for these components are as follows:

ASME SA240, SA182, SA312, SA351, SA479, SA193, SA194, or ASTM A276, A249, A213, A269, A403, Types 304, 304L, 316, 316L, CF3, or CF8.

4. Feedwater sparger - Materials used in this component are 304, 3041, and 316L stainless steel.

All core support structures are fabricated form ASME and ASTM specified materials, and are designed using the ASME B&PV Code, Section III as a guide. The other reactor internals are non-coded and are fabricated from ASTM or ASME specification materials. Material requirements in the ASTM specifications are identical to requirements in corresponding ASME material specifications.

4.5.2.2 Controls on Welding

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

4.5.2.3 Nondestructive Examination of Wrought Seamless Tubular Products

Wrought, seamless, tubular products used for the control rod guide tubes, the CRD housings, and the peripheral fuel supports, are supplied in accordance with applicable ASME material specifications. These specifications require a hydrostatic test on each length of tubing. No specific nondestructive testing is performed on the tubes.

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

4.5.2.4.1 Conformance with Regulatory Guide 1.31, Control of Stainless Steel Welding

Cold worked stainless steels are not used in the reactor internals. All austenitic stainless steel weld filler materials are 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.

Reactor internals were fabricated prior to the issuance of Revision 2 of Regulatory Guide 1.31. Ferrite measurements were made in accordance with the requirements of the ASME B&PV Code in effect at that time. This Code required the use of the chemical composition in conjunction with the Schaeffler diagram to verify that the weld filler metal contained an average of 5 percent delta ferrite minimum.

An extensive test program performed by GE, with the concurrence of the regulatory staff, has demonstrated that the use of the Schaeffler diagram to control weld filler metal ferrite at 5 percent minimum was adequate to produce satisfactory production welds. Four hundred production welds were evaluated in this program. These welds were formed with the filler metal controlled in accordance with the Schaeffler diagram to contain an average of 5 percent

ferrite minimum. All of these production welds met the requirements of the interim regulatory position on Regulatory Guide 1.31 in effect at that time.

4.5.2.4.2 Conformance with Regulatory Guide 1.34, Control of Electroslag Weld Properties

Electroslag welding is not employed for any reactor internals.

4.5.2.4.3 Conformance with Regulatory Guide 1.36, Nonmetallic Thermal Insulation for Austenitic Stainless Steel

For external applications, all nonmetallic insulation meets the requirements of Regulatory Guide 1.36

4.5.2.4.4 Conformance with Regulatory Guide 1.44, Control of the Use of Sensitized Stainless Steel

All wrought austenitic stainless steel is purchased in the solution heat treated condition. Heating above 800°F is prohibited (except for welding) unless the stainless steel with carbon content in excess of 0.035 percent carbon, purchase specifications restrict the maximum weld heat input to 100,000 Joules per inch, and the weld interpass temperature to 350°F maximum. Welding is performed in accordance with Section IX of the ASME B&PV Code. These controls are employed to avoid severe sensitization, and comply with the intent of Regulatory Guide 1.44.

4.5.2.4.5 Conformance with 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 is performed on the welds with most difficult access. Mock-ups are examined with radiography or by sectioning.

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

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

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

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

4.5.2.5 Other Materials

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

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

1. ASME SB166, 167, and 168 Nickel-Chrome-Iron (Inconel 600)

2. ASME SA637 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 1300°F.

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

All materials have been successfully used for the past 10 to 15 years in boiling water reactor (BWR) applications.

4.5.2.6 SRP Rule Review

4.5.2.6.1 Acceptance Criterion II.2

Acceptance criterion II.2 of SRP Section 4.5.2 states that, for acceptance, the welds of components for core support structures and reactor internals are to be fabricated in accordance with the Code, Section III, NG-4000, and must meet the examination and acceptance criteria shown in NG-5000.

Actual welding of these materials and components was done prior to the inclusion of NG-designated requirements in the Code. The requirements of ASME Code Section IX are followed in the fabrication of core support structures and other internals. Details for compliance with or alternate approaches to Regulatory Guides 1.31, 1.34, 1.37, 1.44, and 1.71 are presented in Section 4.5.2.4.

4.5.2.6.2 Acceptance Criterion II.3

Acceptance criterion II.3 of SRP Section 4.5.2 states that, for acceptance of nondestructive examination of wrought seamless tubular products and fittings, examination shall be in accordance with the requirements of ASME Code Section III. NG-2500. The acceptance criteria shall be in accordance with the requirements of ASME Code Section III, NG-5300.

Wrought seamless tubular products for the control rod guide tubes, the CRD housings, and the peripheral fuel supports have been supplied in accordance with ASME specifications applicable at the time of purchase. This was prior to the inclusion of NG-designated

requirements in the Code. The specifications used required hydrostatic testing on each length of tubing. No specific nondestructive testing was performed on the tubes (see Section 4.5.2.3).

4.5.2.6.3 Acceptance Criterion II.4

Acceptance criterion II.4 of SRP Section 4.5.2 states, in part, that Regulatory Guide 1.31, Control of Ferrite Content in Stainless Steel Weld Metal, describes acceptable criteria for ensuring the integrity of welds in stainless steel components. As pointed out in Section 4.5.2.4, the reactor internals were fabricated prior to Revision 2 of Regulatory Guide 1.31, but are in compliance with Revision 1 of Regulatory Guide 1.31 and the provisions of Regulatory Guide 1.44.

4.5.3 Control Rod Drive Housing Supports

All control rod drive (CRD) housing support subassemblies are fabricated of ASTM A36 structural steel, except for the following items:

1.	Grid	ASTM A441
2.	Disc springs	Schnoor, Types BS-125-71-8
3.	Hex bolts and nuts	ASTM A307
4.	6 x 4 x 3/8 tubes	ASTM A500 Grade B

For further CRD housing support information, refer to Section 4.6.1.3.

4.6 FUNCTIONAL DESIGN OF REACTIVITY CONTROL SYSTEMS

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

4.6.1 Information for the CRD System

4.6.1.1 Design Bases

4.6.1.1.1 Safety Design Bases

The CRD mechanical system meets the following safety design bases:

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

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

3. Each positioning device:

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

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

    c. Is energized individually 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.

4.6.1.1.2 Power Generation Design Basis

The CRD system is designed to position the control rods to control power generation in the reactor core.

4.6.1.2 Description

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

4.6.1.2.1 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, as shown on Figures 4.6-1, 4.6-2, 4.6-3, and 4.6-4. The individual drives are mounted on the bottom head of the reactor pressure vessel (RPV). The drives do not interfere with refueling and are operative even when the top head is removed from the reactor vessel.

The drives are also readily accessible for inspection and servicing. The bottom location makes maximum use 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 the condensate storage tank (CST) as the operating fluid eliminates the need for special hydraulic fluid. The drives use simple piston seals, leakage from which 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, in addition to 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 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 control 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 control 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 control rod and drive are uncoupled. Withdrawal to the over travel limit is annunciated by an alarm.

Individual control rod indicators, grouped in one control panel display, correspond to relative rod locations in the core. A separate, smaller display, located just to the right of the large display, presents the positions of the control rod selected for movement and of the other rods in the affected rod group.

For display purposes, the control rods are considered in groups of four adjacent rods centered around a common core volume. Each group is monitored by four local power range monitor (LPRM) strings, as discussed in Section 7.6.1.4. Rod groups at the periphery of the core may have less than four rods. The small rod display shows the positions, in digital form, of the rods in the group to which the selected rod belongs. A white light indicates which of the rods in the group is the one selected for movement.

4.6.1.2.2 Drive Components

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

4.6.1.2.2.1 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 outer cylinder wall. A pair of split bushings on the outside of the piston prevents piston contact with the outer cylinder wall. The effective piston area for downtravel, or withdrawal, is approximately 1.2 square inches vs. 4.1 square inches for uptravel, or insertion. This difference in driving area tends to balance the control rod weight and ensures a higher force for insertion than for withdrawal.

4.6.1.2.2.2 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.

4.6.1.2.2.3 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 flange and cylinder assembly), 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.

4.6.1.2.2.4 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 series of orifices at the top of the tube provides progressive water shutoff to cushion the drive piston at the end of its scram stroke.

4.6.1.2.2.5 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. A stack of spring washers just below the stop piston helps absorb the final mechanical shock at the end of control rod travel. The piston rings are similar to the drive piston outer rings. A bleed-off passage to the center of the piston tube is located between the two pairs of rings. During a scram, this arrangement allows seal leakage from the reactor vessel to be bled directly to the discharge line. The lower pair of seals is used only during the cushioning of the drive piston at the upper end of the stroke.

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. Each switch is sheathed in a braided glass sleeve, and 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. One switch is located

at each position corresponding to an index tube groove, thus allowing indication at each latching point. An additional switch is 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. One additional switch (an over travel switch) is located at a position below the normal full out position. Because the limit of down travel is normally provided by the control rod itself as it reaches the backseat position, the drive can pass this position and actuate the over travel switch 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.

4.6.1.2.2.6 Flange and Cylinder Assembly

The 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 shown on 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.

4.6.1.2.2.7 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 RPV 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.

4.6.1.2.3 Drive Materials of Construction

Factors that determine the choice of construction materials for the drives are discussed below.

4.6.1.2.3.1 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 operation. The reactor environment limits the choice of materials suitable for corrosion resistance. To meet these varied requirements, the index tube is made from an annealed, single phase, nitrogen strengthened, austenitic stainless steel. The wear and bearing requirements are provided by Malcomizing the completed tube. To obtain suitable corrosion resistance, a carefully controlled process of surface preparation is employed.

4.6.1.2.3.2 Coupling Spud

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

4.6.1.2.3.3 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, for the area contacting the index tube and unlocking cam surface of the guide cap.

4.6.1.2.3.4 Seals and Bushings

Graphitar 14 is selected for seals and bushings on the drive piston and stop piston. The material is inert and has a low friction coefficient when water lubricated. Because some loss of Graphitar 14 strength is experienced at higher temperatures, the

drive is supplied with cooling water to hold temperatures below 250°F. The Graphitar 14 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.

4.6.1.2.3.5 Summary

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

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

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

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

4. Elastomeric O-ring seals are ethylene propylene.

5. Metal piston rings are Haynes 25 alloy.

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

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

8. The drive piston head is made of ARMCO 17-4 PH.

Pressure containing portions of the drives are designed and fabricated in accordance with requirements of the ASME B&PV Code, Section III.

4.6.1.2.4 Control Rod Drive Hydraulic System

The CRD hydraulic system shown on Plant Drawings M-46-1 and M-47-1 supplies and controls the pressure and flow to and from the drives through hydraulic control units (HCU). The water discharged from the drives during a scram flows through the HCUs to the scram discharge volume. The water discharged from a drive during a normal control rod positioning operation flows through the HCU to the exhaust header, and is returned to the reactor vessel via the HCUs of nonmoving drives. There are as many HCUs as there are CRDs.

4.6.1.2.4.1 Hydraulic Requirements

The CRD hydraulic system design is shown on Plant Drawings M-46-1 and M-47-1. The CRD system process diagram and data is given on Vendor Technical Document PN1-C11-1020-0009. The hydraulic requirements, identified by the function they perform, are as follows:

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

2. Drive pressure of 260 psi (minimum) above reactor vessel pressure is required. A flow rate of approximately 4 gpm to insert a control rod and 2 gpm to withdraw a control rod is required.

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

4. The scram discharge volume is sized to receive and contain all the water discharged by the drives during a scram; a minimum volume of 3.34 gallons per drive is required, excluding the instrument volume.

4.6.1.2.4.2 System Description

The CRD hydraulic system provides the required functions with the pumps, filters, valves, instrumentation, and piping shown on Plant Drawings M-46-1 and M-47-1 and described in the following paragraphs.

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

1. Drive water pump - The drive water pump pressurizes the system with water from the condensate treatment system and/or the condensate storage tank (CST). One spare pump is provided for standby. A discharge check valve prevents backflow through the standby pump. A portion of the pump discharge flow is diverted through a minimum flow bypass line to the CST. This flow is controlled by an orifice and is sufficient to prevent 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 disposable element type with a 50-micron absolute rating. A 250-micron strainer is provided at the inlet of each suction filter to reduce the debris loading on the suction filter elements. A 250-micron strainer in the suction filter bypass line protects the pump when both suction filters are out of service. The drive water filters downstream of the pump are cleanable element type with a 50-micron absolute rating. A 250-micron strainer in each drive water filter discharge line protects the hydraulic system if there is a filter element failure. Local differential pressure indicators and main control room alarms monitor the filter elements as they collect foreign material.

2. Accumulator charging pressure - Accumulator charging pressure is established by precharging the nitrogen accumulator to a precisely controlled pressure at a known temperature. During a 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 drive water pump to run out, i.e., allows the flow rate to increase substantially, into the CRDs via the charging water header. To prevent prolonged pump operation at run out conditions, a flow element upstream of the accumulator charging header senses the high flow and provides a signal to the manual/auto flow control station. The flow controller allows for automatic or manual closing of the of the flow control valve downstream of the accumulator charging header.

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

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

3. Drive water pressure - Drive water pressure required in the drive water header is maintained by the drive/cooling pressure control valve, which is manually adjusted from the main control room. A flow of approximately 6 gpm (the sum of the flow rate required to insert and withdraw a control rod) normally passes from the drive water header through two solenoid operated stabilizing valves (arranged in parallel) into the cooling water header. The flow through one stabilizing valve equals the drive insert flow; that through the other stabilizing valve equals the drive withdrawal flow. When operating a drive, the required flow is diverted to that drive by closing the

appropriate stabilizing valve, and at the same time opening the drive direction control and exhaust solenoid valves. Thus, flow through the drive/cooling pressure control valve is always constant.

Flow indicators for the drive water header and 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 water header is indicated in the main control room.

4. Cooling water header - The cooling water header is located downstream from the drive/cooling pressure control valve. The drive/cooling pressure control valve is manually adjusted from the main 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 maintains the correct drive water pressure and cooling water pressure, independent of reactor vessel pressure. Changes in the setting of the pressure control valve are required only to adjust for changes in the cooling requirements of the drives, as the drive seal characteristics change with time. Cooling water flow is indicated in the main control room. Differential pressure between the reactor vessel and the cooling water header is indicated in the main control room. 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 main control room.

5. Exhaust water header - The exhaust water header connects to each HCU and provides a low pressure plenum and

discharge path for the fluid expelled from the drives during control rod insert and withdraw operations. The fluid injected into the exhaust water header during rod movements is discharged back up to the RPV via reverse flow through the insert exhaust directional solenoid valves of adjoining HCUs. The pressure in the exhaust water header is, therefore, maintained at essentially reactor pressure. To ensure that the pressure in the exhaust water header is maintained near reactor pressure during the period of vessel pressurization, redundant pressure equalizing valves connect the exhaust water header to the cooling water header.

6. Scram discharge volume - The scram discharge volume (SDV) consists of two sets of 12 in. diameter header piping, each of which connects to one-half of the HCUs and drains into a 12 in. diameter scram discharge instrument volume (SDIV). Each set of header piping is sized to receive and contain all the water discharged by one-half of the drives during a scram, independent of the SDIV.

   The header piping slopes to a low point with a minimum pitch of 1/8 in. per foot as shown on Plant Drawing 1-P-BF-03. The SDIV for each header set is directly connected to the low point of the header piping. The large diameter pipe of each SDIV thus serves as a vertical extension of the SDV. A 2 in. piping connection at the bottom of the SDIV provides drainage of the SDIV and SDV via sloped drain lines with a minimum 1/8 in. per foot slope.

   During normal plant operation, the SDV is empty and is vented to the atmosphere through its open vent and drain valves. When a scram occurs, upon a signal from the safety circuit, these vent and drain valves are closed to conserve reactor water. Redundant vent and drain valves are provided to ensure against loss of reactor coolant

from the SDV following a scram. Lights in the main control room indicate the position of these valves.

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

Remote manual switches in the pilot valve solenoid circuits allow the SDV vent and drain valves to be tested without disturbing the RPS. Closing the SDV valves allows the outlet scram valve seats to be leak tested, by timing the accumulation of leakage inside the SDV.

Four liquid level switches and two level transmitters connected to each SDIV monitor the SDIV for abnormal water level. Each transmitter provides an input signal to a level indicating switch (trip unit). The level switches and trip units provide redundant and diverse inputs to the RPS scram function and inputs to the main control room annunciation and control rod block functions, as shown on Plant Drawing M-47-1. Three different levels setpoints are used. At the lowest level, a level switch actuates to indicate that the SDIV is not completely empty during post scram draining, or to indicate that the SDIV starts to fill through leakage accumulation at other times during reactor operation. At the second level, a level switch produces a rod withdrawal block to prevent further withdrawal of any control rod when leakage accumulates to half the capacity of the SDIV. The remaining two level switches and two transmitter actuated trip units are interconnected with

the trip channels of the RPS and initiate a reactor scram on high water level in the SDIV.

4.6.1.2.4.3 Hydraulic Control Units

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

The basic components in each HCU, as shown on Plant Drawings M-46-1 and M-47-1, and Figure 4.6-8, are manual, pneumatic, and electrical valves; an accumulator; related piping; electrical connections; filters; and instrumentation. The components and their functions are described in the following paragraphs.

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 drive piston.

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.

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

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 drive piston. A correctly adjusted unit does not require readjustment, except to compensate for changes in drive seal leakage.

6. Scram pilot valve assembly - The scram pilot valve assembly is operated from the RPS. The scram pilot valve assembly, with two normally energized solenoids, controls both the scram inlet valve and the scram outlet valve. Upon loss of electrical signal to the solenoids, such as the loss of external ac power, the pilot valve inlet port closes and the exhaust port opens. The pilot valve assembly 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 if one of the pilot valve solenoids fails.

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 main control room as soon as the valve starts to open.

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

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

   During normal plant operation, the accumulator piston is seated at the bottom of the cylinder. Loss of nitrogen decreases the nitrogen pressure, actuating a pressure switch and sounding an alarm in the main control room.

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

4.6.1.2.5 Control Rod Drive System Operation

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

4.6.1.2.5.1 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 water header.

As is illustrated on 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 nominal shim speed (non-scram operation) of 3 in./s. During normal insertion, the pressure on the downstream side of the speed control valve is 90 to 100 psi above reactor vessel pressure. However, if the drive slows for any reason, the flow through and pressure drop across the insert speed control valve decrease; the full differential pressure (260 psi) is then available to cause continued insertion. With a 260-psi differential pressure acting on the drive piston, the piston exerts an upward force of 1040 pounds.

The insert speed is checked and adjusted if required at least once per refueling cycle. Industry experience has shown that shim speed can decrease from plant shutdown (cold) conditions to normal operating conditions (hot). Calibration ranges were established to compensate for this anticipated change for both hot and cold conditions to optimize drive performance and are used in the adjustment procedure. Acceptable values were also determined in reference 4.6-2. These values are as follows:
Calibration range: 3.0 to 3.61 ips cold; 3.0 to 3.3 ips hot.
Acceptable range: 2.78 to 4.04 ips cold: 2.4 to 4.04 ips hot

4.6.1.2.5.2 Rod Withdrawal

Rod withdrawal is, by design, more involved than insertion. The collet fingers (latch) must be raised to reach the unlocked position, as shown on 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 drive piston to the exhaust water header. Pressure is simultaneously applied to the collet piston. As the collet piston rises, 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 nominal shim speed of 3 in./s. The entire valving sequence is automatically controlled and is initiated by a single operation of the rod withdraw switch.

The withdraw speed is checked and adjusted if required at least once per refueling cycle. Industry experience has shown that shim speed can increase from plant shutdown (cold) conditions to normal operating conditions (hot). Calibration ranges were established to compensate for this anticipated change for both hot and cold conditions to optimize drive performance. These values are used in the adjustment procedure. .Acceptable values were also determined in reference 4.6-2. These values are as follows:
Calibration range: 2.39 to 2.76 ips cold; 2.7 to 3.0 ips hot.
Acceptable range: 1.84 to 3.6 ips cold: 1.84 to 4.16 ips hot

4.6.1.2.5.3 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 1500 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 nearly constant velocity. This characteristic provides a high initial rod insertion rate. As the drive piston nears the top of its stroke, the piston seals close off the large passage (buffer orifices) in the stop piston tube, providing a hydraulic cushion at the end of travel.

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

The ball check valve in the drive flange allows reactor pressure to supply the scram force whenever reactor pressure exceeds the supply pressure at the drive. This occurs, due to accumulator pressure decay and inlet line losses, during all scrams at higher vessel pressures. When the reactor is close to, or at, full operating

pressure, reactor pressure alone will insert the control rod in the required time, although the accumulator does provide additional margin at the beginning of the stroke.

The CRD system, with accumulators, provides the following scram performances at full power operation, in terms of average elapsed time after the opening of the RPS trip actuator (scram signal). Times for drives to attain the percentages of scram stroke are listed below:

Percent of full stroke	5	20	50	90
Stroke in inches	7.2	28.8	72.0	129.6
Average time in seconds	0.375	0.90	2.0	3.5

4.6.1.2.5.4 Alternate Rod Insertion

The alternate rod insertion (ARI) feature of the Redundant Reactivity Control System (RRCS) is designed to increase the reliability of the CRD system, as discussed in Section 4.6.1.1. ARI provides for insertion of reactor control rods by depressurizing the scram discharge air header through valves that are redundant and diverse from the RPS scram valves.

The RRCS signal to insert control rods results in the energizing of the eight ARI valves shown on Plant Drawing M-47-1. Two valves in series with the backup scram valves also have parallel functioning check valves to ensure the venting of air from the air supply line, in the event one or more of the ARI valves fail. Four valves provide for venting to atmosphere the A and B HCU scram headers, thereby depressurizing the headers and scramming all rods. Two additional valves vent the scram air header leading to the SDV drain and vent valves, closing these valves and isolating the SDV.

4.6.1.2.6 Instrumentation

The instrumentation for both the control rods and CRDs is defined by that given for the Reactor Manual Control System. The objective of the Reactor Manual Control System is to provide the operator with the means to make changes in nuclear reactivity, so that reactor power level and power distribution can be controlled. The system allows the operator to manipulate control rods. The design bases and further discussion are provided in Section 7.

4.6.1.3 <u>Control Rod Drive Housing Supports</u>

4.6.1.3.1 Safety Objective

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

4.6.1.3.2 Safety Design Bases

The CRD housing supports meet the following safety design bases:

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

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

4.6.1.3.3 Description

The CRD housing supports are shown on 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 that are welded to the steel form liner of the drive room in the reactor support pedestal.

Hanger rods, approximately 10 feet long and 1-3/4 inches 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 assembly or disassembly in place. 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 CRD flange. During system heatup, this gap is reduced by a net downward expansion of the housings with respect to the supports. In the hot operating condition, the gap is approximately 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, is 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 are 90 percent of yield and the allowable shear stress used is 60 percent of yield. These design stresses are 1.5 times the AISC allowable stresses (60 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 operating pressure of 1086 psig (at the bottom of the vessel) acting on the area of the separated housing. The weight of the separated housing, the CRD, and the control blade, plus the pressure of 1086 psig acting on the area of the separated housing, gives a force of approximately 32,000 pounds. 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 pounds) is then treated as a static load in design. The CRD housing supports are designed as Seismic Category I equipment, as discussed in Section 3.2. Loading conditions and examples of stress analysis results and limits are shown in Table 3.9-5. A safety evaluation is provided in Section 4.6.2.3.3.

4.6.2 Evaluations of the CRD System

4.6.2.1 Failure Mode and Effects Analysis

This subject is discussed in Section 15.9.

4.6.2.2 Protection from Common Mode Failures

This subject is discussed in Section 15.9.

4.6.2.3 Safety Evaluation

Safety evaluation of the control rods, control rod drives (CRDs), and CRD housing supports is described below. Further description of the 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 is evaluated in the mechanical design of the control rods. The primary materials, boron carbide (B_4C) powder, Hafnium, and 304 or 316 austenitic stainless steel, have been found suitable in meeting the demands of the boiling water reactor (BWR) environment.

4.6.2.3.1.2 Dimensional and Tolerance Analysis

Layout studies are done to ensure that, given the worst combination of part tolerance ranges at assembly, no interference exists that 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 that 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 and in addition, dissimilar metals are avoided.

4.6.2.3.1.4 Forces for Expulsion

An analysis has been performed that evaluates the maximum pressure forces that 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, under Rupture of Hydraulic Line(s) to the 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/s 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 discussed in Section 4.6.2.3.2.2.

4.6.2.3.1.6 Precluding Excessive Rates of Reactivity Addition

To preclude excessive rates of reactivity addition, analysis has been performed on the velocity limiter device, as well as the effect of probable control rod failures, as discussed in Section 4.6.2.3.2.2.

4.6.2.3.1.7 Effect of Fuel Rod Failure on Control Rod Channel Clearances

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

4.6.2.3.1.8 Mechanical Damage

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

In addition to the analysis performed on the CRDs, discussed in Sections 4.6.2.3.2.2 and 4.6.2.3.2.3, and the control rod blade, the following discussion summarizes the analysis performed on the control rod guide tube.

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

1. Inward load due to pressure differential
2. Lateral loads due to flow across the guide tube
3. Dead weight
4. Seismic (vertical and horizontal)
5. Vibration.

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

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

4.6.2.3.1.9 Evaluation of Control Rod Velocity Limiter

The control rod velocity limiter limits the free fall velocity of the control rod to a value that cannot result in nuclear system process barrier damage. This velocity is evaluated by the rod drop accident analysis in Section 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 CRD system provides the negative reactivity insertion required by the safety design bases as

discussed in Section 4.6.1.1. The scram time shown in the description from Section 4.6.1.2 is adequate, as shown by the transient analyses of Section 15.

4.6.2.3.2.2 Analysis of Malfunctions 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. Potential malfunctions relating to the unplanned withdrawal of a control rod are discussed in the following paragraphs.

1. Drive housing failure at attachment weld - The bottom head of the reactor vessel has a penetration for each CRD location. A drive housing is raised into position inside each penetration and is fastened by welding. The drive is raised into the drive housing and is bolted to a flange at the bottom of the housing.

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

   Static loads on the housing wall include the weight of the drive and the control rod, the weight of the housing below the J-weld, and the reactor pressure acting on the 6-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:

a. The housing would separate from the reactor vessel.

b. The CRD with coupled control rod and the housing would be blown downward against the support structure by reactor pressure acting on the cross-sectional area of the housing and the drive.

c. The downward motion of the drive and associated parts would be determined by the gap between the bottom of the drive and the top of the support structure, and by the deflection of the support structure under the load. In the current design, maximum deflection is approximately 3 inches.

d. If the collet remains latched, no further control rod ejection occurs, as mentioned in Reference 4.6-1; the housing does not drop far enough to clear the vessel penetration.

e. Reactor water would leak at a rate of approximately 180 gpm through the 0.03-inch diametric 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:

a. The housing would separate from the reactor vessel.

b. The CRD with coupled control rod and the housing would be blown downward against the CRD housing support.

c. Calculations indicate that the steady state rod withdrawal velocity would be 0.3 ft/s.

d. 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.

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

a. 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 drywell. 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 CRD position indicator probe. A broken line would also be indicated by high cooling water flow, and by operation of the drywell floor drain sump pump.

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/s.

b. Pressure over (withdraw) line break - The case of the pressure over (withdraw) 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 control blade. 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 drywell through the broken pressure over line. The leakage rate at 1000 psi reactor pressure is estimated to be 1 to 3 gpm;

however, with the Graphitar 14 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 a fully inserted drive, by a high drive temperature (annunciated in the main control room), and by operation of the drywell floor drain sump pump.

c. Simultaneous breakage of the pressure-over (withdraw) and pressure under (insert) lines - For the simultaneous breakage of the pressure over (withdraw) and pressure under (insert) lines, pressures above and below the drive piston would drop to atmospheric pressure, 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 drywell as described above. Drive temperature would increase. Indication in the main control room would include the drift alarm, the fully inserted drive, the high drive temperature annunciation, and operation of the drywell floor drain sump pump.

3. All drive flange bolts fail in tension - Each CRD is bolted to a flange at the bottom of a drive housing. The flange is welded to the drive housing. Bolts are made of AISI-4140 steel, with a minimum tensile strength of 125,000 psi. Each bolt has an allowable load capacity of

15,200 pounds. Capacity of the eight bolts is 121,600 pounds. As a result of the reactor design pressure of 1250 psig, the major load on all eight 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 only on the drive cross-sectional area 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 atmospheric pressure. Pressure forces, in fact, exert 1435 pounds to hold the collet in the latched position.

If the bolts failed during control rod withdrawal, pressure below the collet piston would drop to atmospheric

pressure. The collet, with 1650 pounds return force, would latch and stop rod withdrawal.

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

   If the basic flange to housing joint failure occurred, the flange and the attached drive with coupled control rod 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 with coupled control rod and the flange would be blown downward against the support structure. The calculated steady state rod withdrawal velocity would be 0.13 ft/s. Because the 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.

5. Housing wall ruptures - This failure is a vertical split in the drive housing wall just below the bottom head of the reactor vessel. The flow area of the hole is considered equivalent to the annular area between the drive and the thermal sleeve. Thus, flow through this annular area, rather than flow through the hole in the housing, would govern leakage flow. The housing is made of Type 304 stainless steel seamless pipe, with a minimum tensile strength of 75,000 psi. The maximum hoop stress

of 11,900 psi results primarily from the reactor design pressure (1250 psig) acting on the inside of the housing.

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

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 acting on the drive piston. The net force reduction would occur when the leakage flow of 1030 gpm reduces the pressure in the annulus outside the drive to approximately 540 psig, thereby reducing the pressure acting on top of the drive piston to the same value. A pressure differential of approximately 710 psi would exist across the collet piston and hold the collet unlatched as long as the withdraw signal is maintained.

6. Flange plug blows out - To connect the vessel ports with the bottom of the ball check valve, a hole of 0.75-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 main 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/s. 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 is 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.

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 drive flange forging. This hole is sealed with a plug of 1.31-inch diameter and 0.38-inch thickness. A full-penetration weld, using Type 308 stainless steel filler, holds the plug in place. The failure postulated is a circumferential crack in this weld leading to a blowout of the plug.

   If the plug were to blow out while the drive was latched, there would be no control rod motion. No pressure differential would exist across the collet piston to unlatch the collet. As in the previous failure, reactor water would flow past the velocity limiter, down the annulus between the drive and thermal sleeve, through the vessel ports and drilled passage, through the ball check valve cage and out the open plug hole to the drywell. The leakage calculation indicates 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 main 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 ft/s. There would be a large retarding force exerted by the velocity limiter due to a 35 psi pressure differential across the velocity limiter piston.

8. Drive/cooling water pressure control valve closure (reactor pressure, 0 psig) - The pressure to move a drive is generated by the pressure drop of total 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 drive water pump. The occurrence of this condition during withdrawal of a drive at zero vessel pressure would result in a drive pressure increase from 260 psig to no more than 1750 psig. Calculations indicate that the drive would accelerate from 3 in./s to approximately 6.5 in./s. A pressure differential of 1670 psi across the collet piston would hold the collet unlatched. Flow would be upward, past the velocity limiter piston, but retarding force would be negligible. Rod movement would stop as soon as the driving signal is removed.

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/s. 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/s could occur. Water would flow upward past the velocity limiter piston, generating a small retarding force of approximately 120 pounds.

10. Hydraulic control unit (HCU) 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.

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.

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./s. Withdrawal speed is maintained by the pressure regulating system and is independent of reactor vessel pressure. Tests performed at a GE facility under controlled conditions (i.e., no other concurrent conditions existing that could cause rod withdrawal speed to be greater) have shown that accidental opening of the speed control valve to the fully open position produces a velocity of approximately 6 in./s.

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

4.6.2.3.2.3 Scram Reliability

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

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

2. Each drive mechanism has its own scram valves and scram pilot valve; therefore, only one drive can be affected if a scram valve fails to open. Both pilot valve solenoids must be deenergized to initiate a scram.

3. The Reactor Protection System (RPS) and the HCUs are designed so that the scram signal and mode of operation override all others.

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

5. The scram discharge volume is monitored for accumulated water and the reactor is scrammed 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 the 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: the compression of the disc springs under dynamic loading, and the initial gap between the grid and the bottom contact surface of the CRD flange. If the reactor is cold and pressurized, the downward motion of the control rod would be limited to the spring compression (approximately 2 inches) plus a gap of approximately 1 inch. If the reactor is hot and pressurized, the gap would be approximately 3/4 inches 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 limited to less than one drive notch movement (6 inches). 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 inches 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 housings, except during the postulated accident condition, vertical contact stresses are prevented. Inspection and testing of the CRD housing supports are discussed in Section 4.6.3.2.

4.6.3 Testing and Verification of the CRD System

4.6.3.1 Control Rod Drives

4.6.3.1.1 Development Tests

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

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

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

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

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

4.6.3.1.2 Factory Quality Control Tests

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

1. Control rod tests:

   a. Absorber rods are examined by nondestructive methods to ensure correct B_4C density, Hafnium content, and weld integrity.

   b. The absorber rod process control rod is inspected to ensure correct assembly.

   c. The envelope of each control rod is inspected to ensure correct assembly.

2. Control rod drive mechanism tests:

   a. Pressure welds on the drives are hydrostatically tested in accordance with ASME Codes.

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

   c. Drive parts that cannot be visually inspected for dirt are flushed with filtered water at high velocity. No significant foreign material is permitted in effluent water.

   d. Seals are tested for leakage to demonstrate correct seal operation.

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

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

3. HCU tests:

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

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

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

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

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

4.6.3.1.3 Operational Tests

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

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

To make a positive test of control rod to CRD coupling integrity, the operator withdraws a control rod to the end of its travel and then attempts to withdraw the drive to the over-travel position. Failure of the drive to over travel demonstrates rod to drive coupling integrity.

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

4.6.3.1.4 Acceptance Tests

Criteria for acceptance of the individual CRD mechanisms and the associated control and protection systems are incorporated in specifications and test procedures covering three distinct phases: preinstallation, after installation and prior to startup, and during startup testing.

The preinstallation 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 discussed in 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 are tests intended more to document system condition rather than to test system performance.

As fuel is placed in the reactor, the startup test procedure discussed in Section 14 is followed. The tests in this procedure are intended to demonstrate that the initial operational characteristics meet the limits of the specifications over the range of primary coolant temperatures and pressures from ambient to operating. The detailed specifications and procedures follow the general pattern established for such specifications and procedures in BWRs presently under construction and in operation.

4.6.3.1.5 Surveillance Tests

The surveillance requirements for the CRD system are described below:

1. Sufficient control rods are withdrawn, following core alterations (fuel movement, control rod replacement, control rod shuffling) to measure Shutdown Margin (SDM) and ensure that the core can be made subcritical at any time in the subsequent fuel cycle with the most reactive operable control rod fully withdrawn and all other operable rods fully inserted.

2. Each partially or fully withdrawn control rod is exercised one notch at least once each month. In the event that operation is continuing with one immovable control rod, this test is performed at least once each day.

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

3. The coupling integrity is verified for each withdrawn control rod as follows:

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

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

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

4. During operation, accumulator pressure and level at the normal operating value is verified.

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

5. At each refueling outage, each operable control rod is 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.

6. Prior to startup after the first refueling outage, PSE&G shall:

   a. confirm that the leak rates, loading conditions and material properties for the scram discharge volume piping system are bounded by the limiting values for those parameters identified in the May 10, 1984, BWR Owners Group letter.

   b. comply with BWR Owners Group recommendations for leak detection capability.

c. comply with the applicable generic secondary containment Emergency Procedure Guidelines,

d. provide assurance that the expected radiation fields and contact exposure levels at the scram discharge volume piping systems in the facility will not impair the performance of routine tests, inspections, and post-scram reset walkdowns.

4.6.3.1.6 Functional Tests

The functional testing program of the CRDs consists of the 5-year maintenance life test program and the 1.5X design life test program, as described in Section 3.9.4.

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

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

1. Simulated ruptured scram line test

2. Stuck ball check valve in CRD flange

3. HCU withdraw drive valve (SV122) failure

4. HCU withdraw exhaust valve (SV120) failure

5. CRD scram performance with SV120 malfunction

6. HCU insert exhaust valve (SV121) failure

7. HCU insert drive valve (SV123) failure

8. Cooling water check valve (V138) leakage

9. CRD flange ball check valve leakage

10. CRD stabilization circuit failure

11. HCU filter restriction

12. Air trapped in CRD hydraulic system

13. CRD collet drop test

14. Control rod qualification velocity limiter drop test.

Additional postulated CRD failures are discussed in Section 4.6.2.3.2.2.

4.6.3.2 Control Rod Drive Housing Supports

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

4.6.4 Information for Combined Performance of Reactivity Control Systems

4.6.4.1 Vulnerability to Common Mode Failures

The two reactivity control systems, Control Rod Drive (CRD) and Standby Liquid Control (SLC), do not share any instrumentation or components. Thus, a common mode failure of the reactivity systems would be limited to an accident event, which could damage essential equipment in the two independent systems.

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

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

4.6.4.2 Accidents Taking Credit for Multiple Reactivity Systems

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

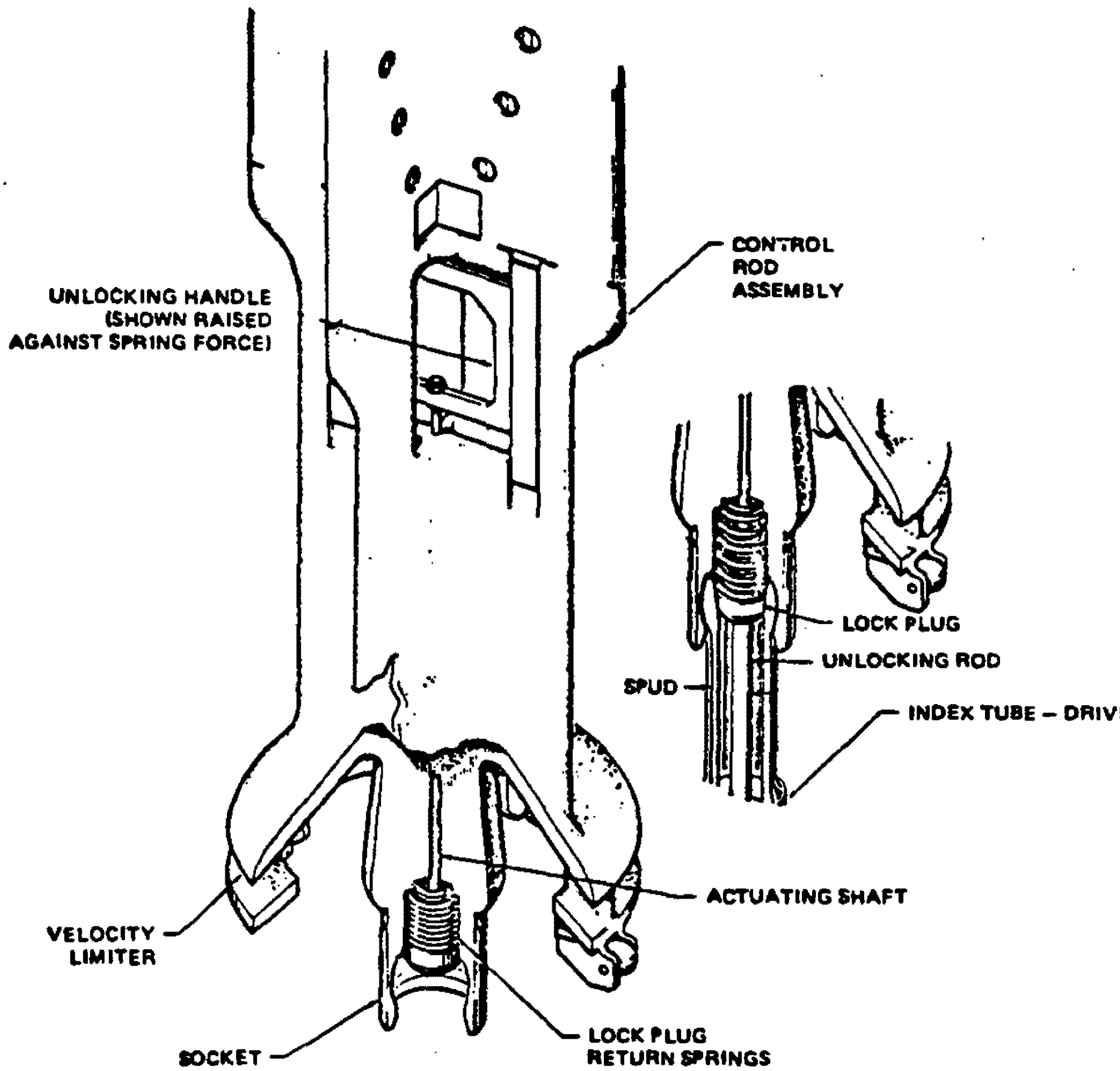
4.6.5 Evaluation of Combined Performance

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

4.6.6 References

4.6-1 J.E. Benecki, "Impact Testing on Collet Assembly for Control Rod Drive Mechanism 7RDB144A," APED-5555, General Electric Atomic Power Equipment Department, November 1967.

4.6-2 Engineering Evaluation H1-BF-NEE-1826 Control Rod Notch Speed Requirements.



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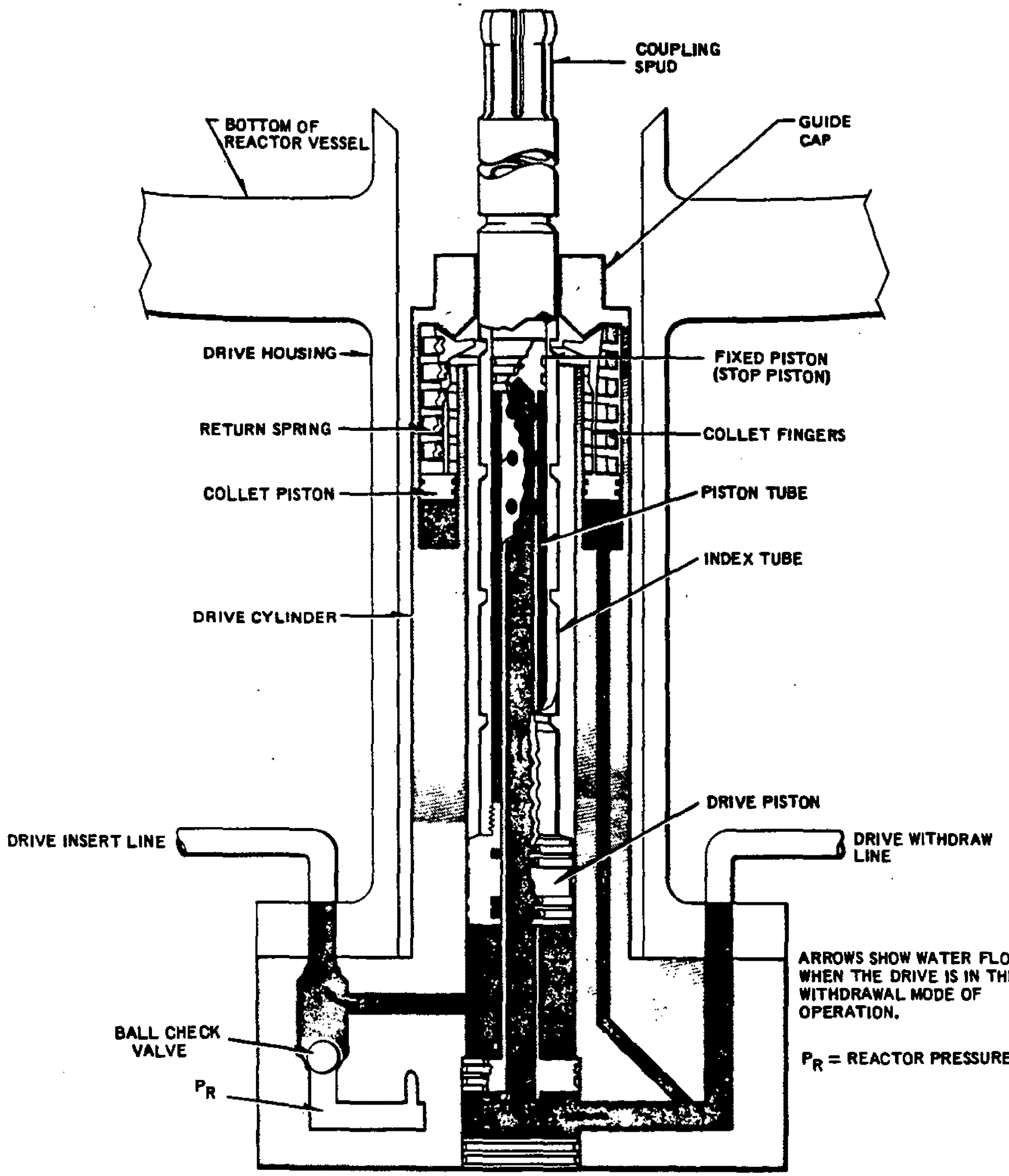
CONTROL ROD TO
CONTROL ROD DRIVE COUPLING

UPDATED FSAR

FIGURE 4.6-1



ARROWS SHOW WATER FLOW WHEN THE DRIVE IS IN THE WITHDRAWAL MODE OF OPERATION.

P_R = REACTOR PRESSURE

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CONTROL ROD DRIVE UNIT

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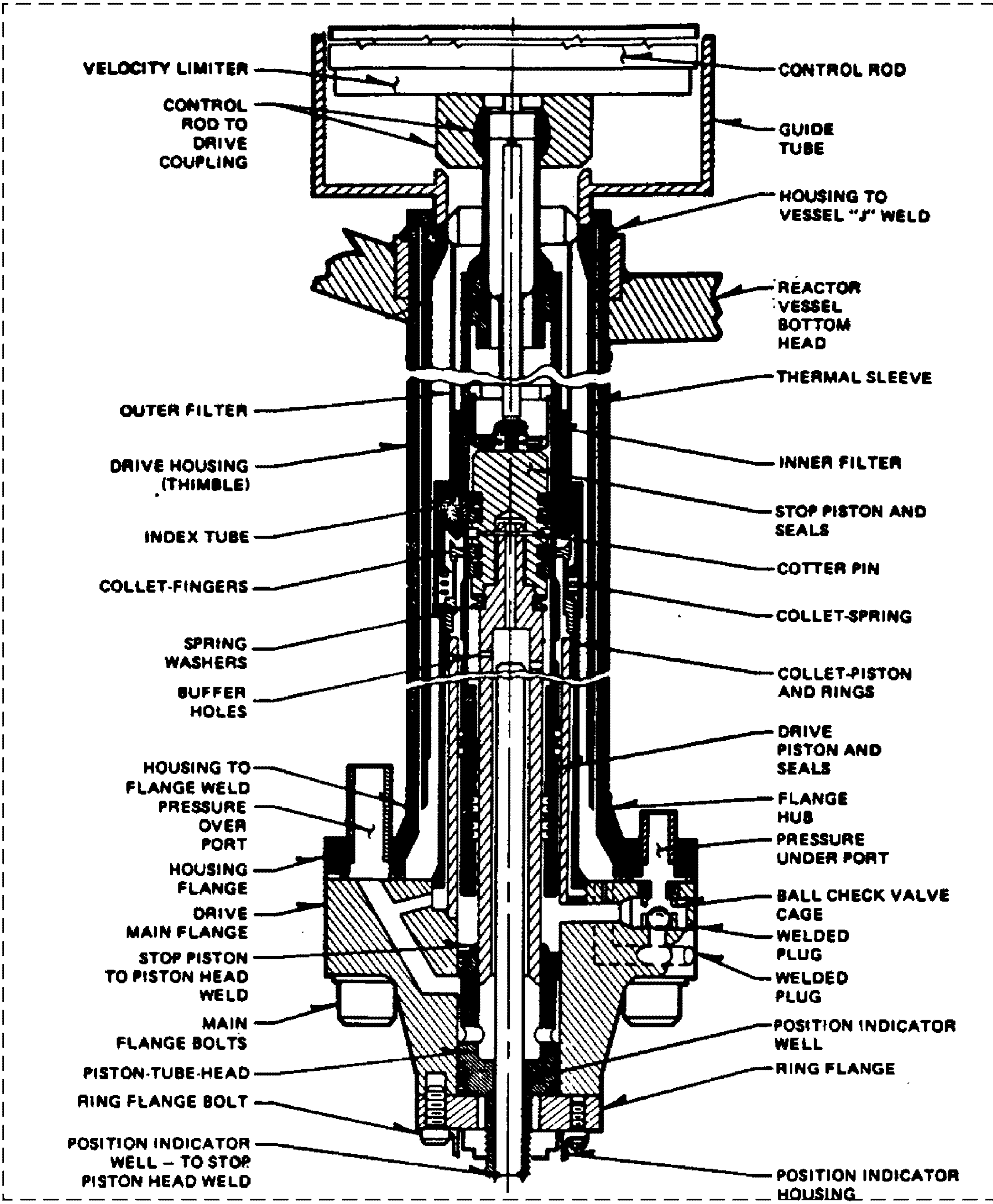
FIGURE 4.6-2



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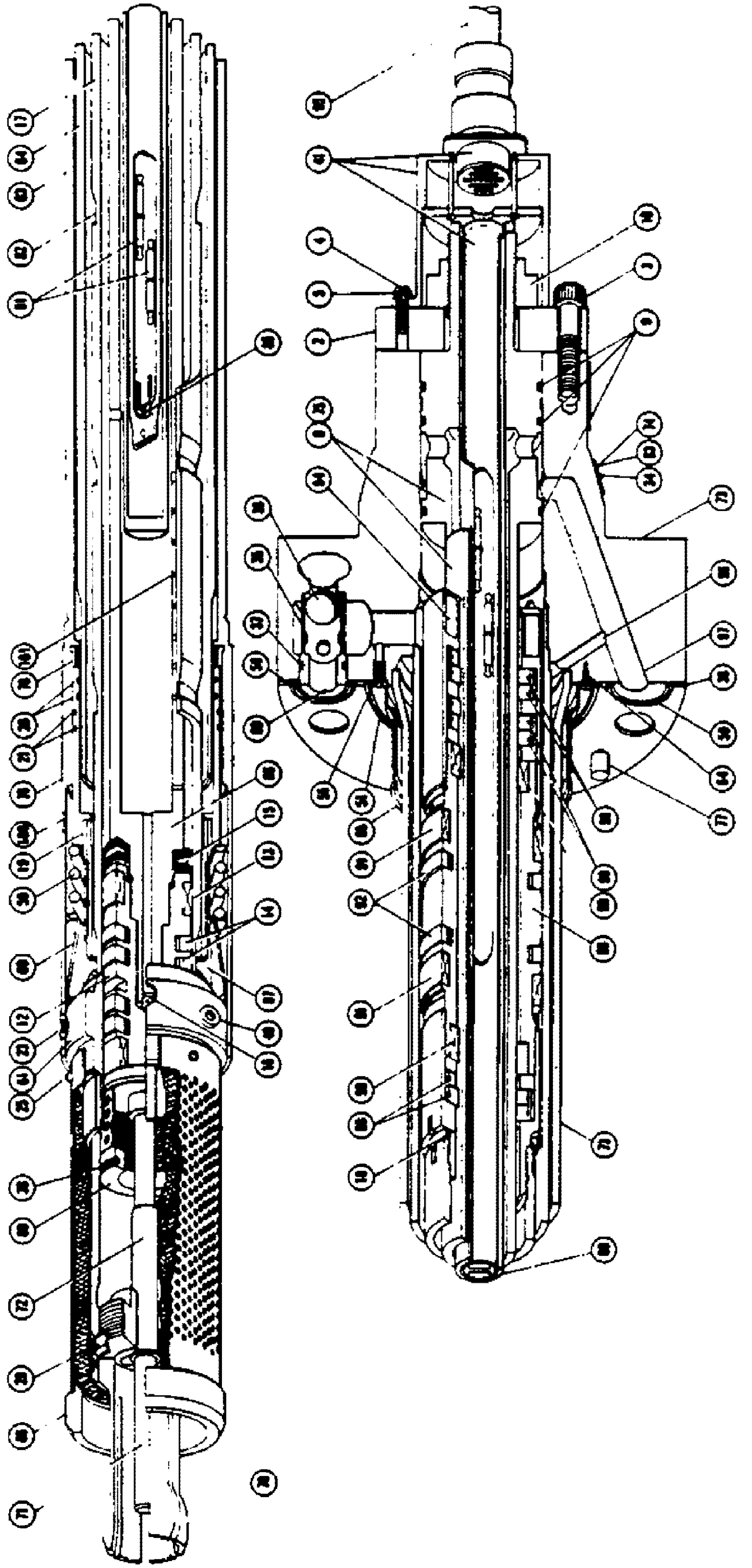
PSEG Nuclear, LLC HOPE CREEK NUCLEAR GENERATING STATION	Hope Creek Nuclear Generating Station CONTROL ROD DRIVE SCHEMATIC
	Updated FSAR Figure 4.6-3

2	RING FLANGE
3	SOCKET HEAD CAP SCREW (RING FLANGE MOUNTING)
4	FILLISTER HEAD SCREW (POSITION INDICATOR PROBE MOUNTING)
5	LOCKWASHER (FOR PART 4)
8 75	PISTON TUBE
9	O RING (PISTON TUBE)
10	NUT (PISTON TUBE)
12	STOP PISTON
13	SPLIT BUSHING (STOP PISTON)
14	SEAL RING (STOP PISTON)
15	SPRING WASHERS
16	COTTER PIN (STOP PISTON)
17	INDEX TUBE
18	BAND
19	COLLET AND PISTON
20	SEAL RING (COLLET PISTON - INTERNAL)
21	SEAL RING (COLLET PISTON - EXTERNAL)
22	FILLISTER HEAD SCREW (GUIDE CAP PLUG MOUNTING)
25	DRILLED FILISTER HEAD SCREW (OUTER FILTER MOUNTING)
28	SEAL RING (INNER FILTER)
29	BAND
34	DRIVE SCREW (NAMEPLATE MOUNTING)
35	BALL RETAINER
36	BALL (CHECK VALVE)
37	O-RING (BALL RETAINER)
38	O-RING SPACER
41	POSITION INDICATOR PROBE
49	PLUG (GUIDE CAP)
50	O-RING (INSERT AND WITHDRAW PORTS)
51	O RING (CRD FLANGE FACE)
56	SET SCREW PLUG (COOLING WATER ORIFICE)
57	SCREW (O-RING SPACER MOUNTING) (NOT SHOWN)
59	COLLET SPRING
60	BARREL
61	GUIDE CAP
63, 74	NAMEPLATE
64	FLAT HEAD SCREW (STRAINER MOUNTING)
65	STRAINER
66	OUTER FILTER
69	DRIVE PISTON
70	SPUD
71	ROD (UNCOUPLING)
72	TUBE
73	CYLINDER TUBE AND FLANGE
77	DOWEL (ALIGNMENT) PIN
78	COLLET HOUSING (PORTION OF OUTER TUBE)
79	SPACER (PART OF CYLINDER, TUBE, AND FLANGE)
80	INNER FILTER
81	POSITION INDICATOR SWITCHES
82	INDEX TUBE NOTCH
83	OUTER TUBE (PART OF CYLINDER TUBE AND FLANGE)
84	INNER CYLINDER (PART OF CYLINDER TUBE, AND FLANGE)
85	THERMOCOUPLE (PART OF POSITION INDICATOR PROBE)
86	STUD (PORTION OF PISTON TUBE)
87	COLLET FINGER (PART OF COLLET AND PISTON)
88	INDICATOR TUBE (PART OF PISTON TUBE)
89	INNER SEALS (DRIVE PISTON - BUFFER SEALS)
90	INTERNAL BUSHING (DRIVE PISTON)
91	EXTERNAL BUSHING (DRIVE PISTON)
92	OUTER SEALS (DRIVE PISTON)
93	INSERT PORT (INSERT AND SCRAM INLET/WITHDRAW OUTLET)
94	RING MAGNET (PART OF DRIVE PISTON)
95	CABLE (POSITION INDICATOR)
96	PORT TO COLLET PISTON (WITHDRAW PRESSURE TO COLLET PISTON)
97	WITHDRAW PORT (WITHDRAW INLET/INSERT OUTLET AND SCRAM DISCHARGE)
98	INNER SEALS (DRIVE PISTON - DRIVE DOWN SEALS)
99	INNER SEALS (DRIVE PISTON - DRIVE UP SEALS)
100	WATER PORTS IN COLLET HOUSING
101	BUFFER ORIFICES IN PISTON TUBE (TYPICAL)



Revision 15, October 27, 2006

PSEG Nuclear, LLC HOPE CREEK NUCLEAR GENERATING STATION	Hope Creek Nuclear Generating Station CONTROL ROD DRIVE (CUTAWAY)
	Updated FSAR — Figure 4.6-4

Figure F4.6-5 intentionally deleted.

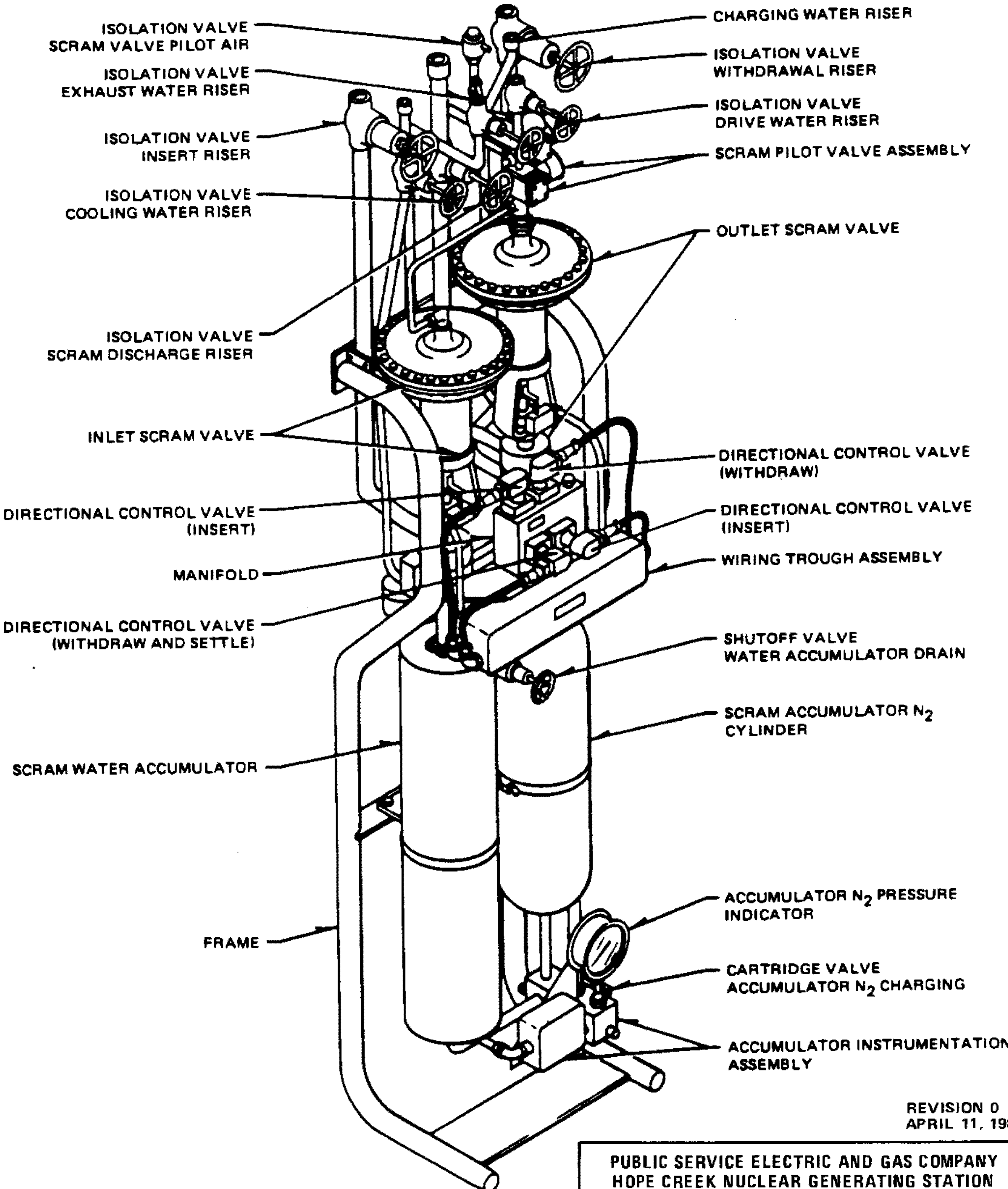
Refer to Plant Drawing M-46-1 in DCRMS

Figure F4.6-6 SH 1-2 intentionally deleted.

Refer to Plant Drawing M-47-1 for both sheets in DCRMS

Figure F4.6-7 SH 1-4 intentionally deleted.

Refer to Vendor Technical Document PN1-C11-1020-0009 for all sheets in DCRMS



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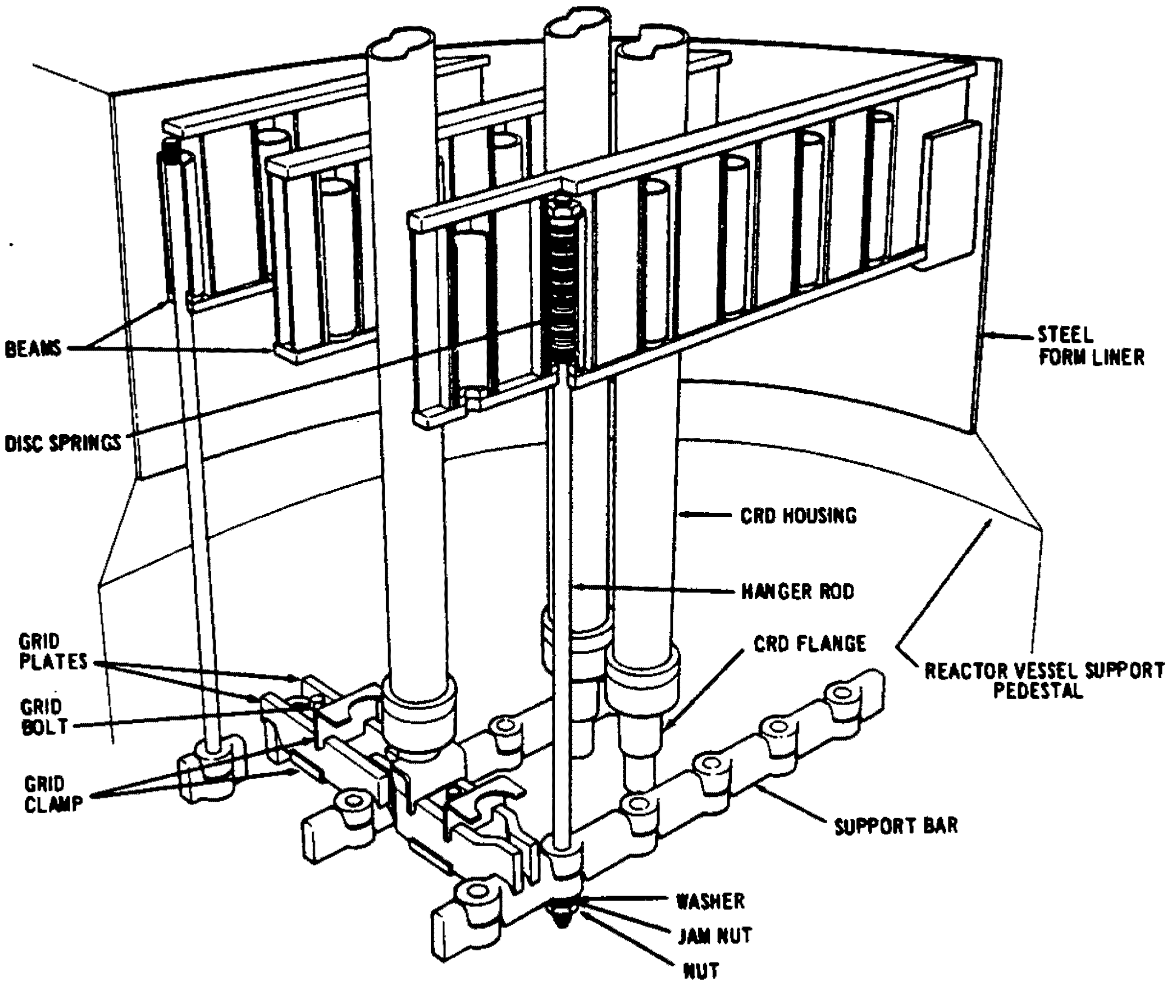
CONTROL ROD DRIVE
HYDRAULIC CONTROL UNIT

UPDATED FSAR

FIGURE 4.6-8



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HOPE CREEK NUCLEAR GENERATING STATION

CONTROL ROD DRIVE
HOUSING SUPPORT

UPDATED FSAR

FIGURE 4.6-9

Figure F4.6-10 intentionally deleted.

Refer to Plant Drawing 1-P-BF-03 in DCRMS