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Chapter 4

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4.1 SUMMARY DESCRIPTION

The reactor assembly consists of the reactor vessel; its internal components: the core, shroud, steam separator and dryer assemblies; and jet pumps. Also included in the reactor assembly are the control rods, the control rod drives (CRD) and the CRD housings. Figure 5.3-4 shows the arrangement of reactor assembly components. A summary of the important design and performance characteristics is given in Table 1.3-1. Loading conditions for reactor assembly components are specified in Section 3.9B.5.2.

4.1.1 Reactor Vessel

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

4.1.2 Reactor Internal Components

The major reactor internal components are the core (fuel, channels, control rods, and instrumentation), the core support structure (including the shroud, top guide, and core plate), the shroud head and steam separator assembly, the steam dryer assembly, the feedwater spargers, the core spray spargers, and the jet pumps. Except for the Zircaloy used in the fuel assemblies, reactor internals are stainless steel or other corrosion-resistant alloys. The fuel assemblies (which include fuel rods and channel), control rods, in-core instrumentation, shroud head and steam separator assembly, and steam dryers are removable when the reactor vessel is opened for refueling or maintenance.

4.1.2.1 Reactor Core

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

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

1. The BWR core mechanical design is based on conservative application of stress limits, operating experience, and experimental test results. The moderate pressure level characteristics of a direct-cycle reactor (approximately 1,000 psia) result in moderate cladding temperatures and stress levels.

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

temperature-dependent corrosion and hydride buildup. The relatively uniform fuel cladding temperatures throughout the core minimize migration of the hydrides to cold cladding zones and reduce thermal stresses.

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

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

5. The GE thermal analysis basis (GETAB) is applied to assure that more than 99.9 percent of the fuel rods in the core will avoid boiling transition for the most severe moderate frequency transient described in Chapter 15 and Appendix A. The possibility of boiling transition occurring during normal reactor operation is insignificant.

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

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

Important features of the reactor core arrangement are as follows:

1. The bottom-entry cruciform control rods consist of one of the following absorber section designs:

   a. Boron carbide (B_4C) in stainless steel tubes surrounded by a stainless steel sheath.

   b. B_4C in stainless steel tubes and hafnium bars surrounded by a stainless steel sheath.

   c. B_4C in stainless steel tubes and hafnium rods loaded in stainless steel tubes welded together.

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 source and intermediate range instruments provides coverage of the large reactor core and provides an acceptable signal-to-noise ratio and neutron-to-gamma ratio. All in-core instrument leads enter from the bottom and the instruments are in service during refueling. In-core instrumentation is discussed in Section 7.6.1.4.

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

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

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

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

<u>Core Configuration</u>

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.

<u>Fuel Assembly Description</u>

The fuel assembly description is given in Section 4.2.

<u>Fuel Rod</u>

The fuel rod description is given in Section 4.2.

<u>Assembly Support and Control Rod Location</u>

A few peripheral fuel assemblies are supported by the core plate. Otherwise, individual fuel assemblies in the core rest on fuel

support pieces mounted on top of the control rod guide tubes. Each guide tube, with its fuel support piece, bears the weight of four assemblies and is supported by a 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 on top of the shroud, provides lateral support and guidance for the top of each fuel assembly. The reactivity of the core is controlled by cruciform control rods and their associated mechanical hydraulic drive systems. The control rods occupy alternate spaces between fuel assemblies. Each independent drive enters the core from the bottom and can accurately position its associated control rod during normal operation and 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 drive maintenance.

4.1.2.2 Shroud

The shroud is discussed in Section 3.9B.5.1.

4.1.2.3 Shroud Head and Steam Separators

The shroud head and steam separators are discussed in Section 3.9B.5.1.

4.1.2.4 Steam Dryer Assembly

The steam dryer assembly is discussed in Section 3.9B.5.1.

4.1.3 Reactivity Control Systems

4.1.3.1 Operation

The control rods perform dual functions of power distribution shaping and reactivity control. Power distribution in the core is controlled during operation of the reactor by manipulation of selected patterns of rods. The rods, which enter from the bottom of the near-cylindrical reactor core, are positioned in such a manner as to counterbalance steam voids in the top of the core and effect significant power flattening. These groups of control elements, used for power flattening, experience a somewhat higher duty cycle and neutron exposure than the other rods in the control system.

The reactivity control function requires that all rods be available for either reactor scram (prompt shutdown) or reactivity regulation. Because of this, the control elements are mechanically designed to withstand the dynamic forces resulting from a scram. They are connected to bottom-mounted, hydraulically-actuated drive mechanisms 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 given in Section 4.6.1.1.

4.1.3.3 Supplementary Reactivity Control

The initial and reload core control requirements are met by use of the combined effects of the movable control rods, supplementary burnable poison, and variation of reactor coolant flow. The supplementary burnable absorber description is given 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:

MASS	SHELL 5
SNAP (MULTISHELL)	HEATER
GASP	FAP-71
NOHEAT	CREEP-PLAST
FINITE	ANSYS
DYSEA	

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

4.1.4.1.1 MASS (Mechanical Analysis of Space Structure)

This program, proprietary to General Electric Company (GE), 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[1]. The program offers curved beam, plate, and shell elements. It can handle mechanical and thermal loads in a static analysis and predict natural frequencies and mode shapes in a dynamic analysis.

Program Version and Computer

The GE Nuclear Energy Group (NEG) uses a past revision of MASS. This revision is identified as Revision 0 in the GE computer production library. The program operated on the Honeywell 6000 computer and is now retired.

History of Use

Since its development in the early 1960s, the program has been successfully applied to a wide variety of jet engine structural problems, many of which involve extremely complex geometries. The use of the program in the NEG also started shortly after its development.

Extent of Application

Besides the NEG and jet engine divisions, the missile and space division, appliance division, and turbine division of GE have also applied the program to a wide range of engineering problems. The NEG uses it mainly for piping and reactor internals analyses.

4.1.4.1.2 SNAP (MULTISHELL)

The SNAP program, 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. Reissner's differential equations for each shell's influence coefficients. Surface loading capability includes pressure, average temperature, and linear through-wall gradients; the latter two may be linearly varied over the shell meridian. The theoretical limitations of this program are the same as those of classical theory.

Program Version and Computer

The current version was obtained from the GE Jet Engine Division at Evandale, OH. It is used on the Honeywell 6000 computer in GE/NEG.

History of Use

The initial version of the shell analysis program was completed by the jet engine division in 1961. Since then, a considerable amount of modification and addition has been made to accommodate its broadening area of application. Its application in the NEG has a history longer than 10 yr.

Extent of Application

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

4.1.4.1.3 GASP

GASP is a finite element program for the stress analysis of axisymmetric or plane two-dimensional geometries. The element representations can be either quadrilateral or triangular. Axisymmetric or plane structural loads can be input at nodal

points. Displacements, temperatures, pressure loads, and axial inertia can be accommodated. Effective plastic stress and strain distributions can be calculated using a bilinear stress-strain relationship by means of an iterative convergence procedure.

Program Version and Computer

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

History of Use

The program was developed by E. L. Wilson in 1965[2]. The present version in GE/NEG has been in operation since 1967.

Extent of Application

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

4.1.4.1.4 NOHEAT

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

This program 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. Cooling pipe boundary conditions are also treated. The output includes temperature of all nodal points for the time instants specified by the user. The program can handle multitransient temperature input.

Program Version and Computer

The current version of the program is an improvement of the program originally developed by I. Farhoomand and Professor E. L. Wilson of the University of California at Berkeley[3]. The program operates on the Honeywell 6000 computer.

History of Use

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

Extent of Application

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

4.1.4.1.5 FINITE

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

Program Version and Computer

The present version of the program at GE/NEG was obtained from the developer, J. E. McConnelee of the GE gas turbine department, in 1969[4]. The NEG version was used on the Honeywell 6000 computer and is now retired.

History of Use

Since its completion in 1969, the program has been widely used in the gas turbine and the jet engine departments of GE for the analysis of turbine components.

Extent of Application

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

4.1.4.1.6 DYSEA (Dynamic and Seismic Analysis)

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

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

Program Version and Computer

The DYSEA version now operating on the Honeywell 6000 computer of GE/NEG was developed at GE by modifying the SAPIV program. The capability was added to handle the hydrodynamic mass effect due to fluid-structure interaction in the reactor. It can handle three-dimensional dynamic problems with beams, trusses, and springs. Both acceleration time-histories and response spectra may be used as input.

History of Use

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

Extent of Application

The current version of DYSEA has been used in many 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

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

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

Program Version and Computer

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

History of Use

SHELL 5 is a program developed by Gulf General Atomic Incorporated in 1969(6). The program has been in production status at Gulf General Atomic, GE, and other major computer operating systems since 1970.

Extent of Application

SHELL 5 has been used at GE to analyze reactor shroud support and torus. Results from test problems were found to be in close agreement with those obtained from either verified programs or analytic solutions.

4.1.4.1.8 HEATER

HEATER is a computer program used in the hydraulic design of feedwater spargers and their associated delivery headers 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[7].

Program Version and Computer

This program was developed at GE/NEG for the Honeywell 6000 computer.

History of Use

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

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 unusual operational conditions.

4.1.4.1.9 FAP-71 (Fatigue Analysis Program)

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

Program Version and Computer

The present version of FAP-71 was completed by L. Young of GE/NEG in 1971[8]. The program currently is on the NEG Honeywell 6000 computer.

History of Use

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

Extent of Use

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

4.1.4.1.10 CREEP/PLAST

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

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

Program Version and Computer

The program can be used for elastic-plastic analysis with or without the presence of creep. It can also be used for creep analysis without the presence of instantaneous plasticity. A detailed description of theory is given in NEDO-23538[9].

History of Use

This program, described in NEDO-23538, underwent extensive program testing before it was put on production status.

Extent of Application

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

4.1.4.1.11 ANSYS

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

The ANSYS program features the following capabilities:

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

2. One-dimensional fluid flow analyses.

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

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

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

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

<u>Program Version and Computer</u>

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

<u>History of Use</u>

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

<u>Extent of Application</u>

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

4.1.4.2 Fuel Rod Thermal Analysis

Fuel rod thermal design analyses are described in Section 4 of GESTAR II[10].

4.1.4.3 Reactor Systems Dynamics

The analysis techniques and computer codes used in reactor system dynamics are described in Section A.4.4.4.5 of GESTAR II[10].

Section 3.2.6 also provides a complete 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 GESTAR II[10].

4.1.4.5 Neutron Fluence Calculations

Neutron vessel fluence calculations were originally carried out using a one-dimensional, discrete ordinate, S_N transport code with general anisotropic scattering. This code is a modification of a widely-used discrete ordinate code which solves a wide variety of radiation transport problems. The program solves both fixed source and multiplication problems. Slab, cylinder, and spherical geometries are allowed with various boundary conditions. The fluence calculations incorporate, as an initial starting point, neutron fission distributions prepared from core physics data as a distributed source. Anisotropic scattering was considered for all regions. The cross sections were prepared with 1/E flux weighted, $P_{(3)}$ matrices for anisotropic scattering but did not include resonance self-shielding factors. Fast neutron fluxes at locations other than the core midplane were calculated using a two-dimensional, discrete ordinate code. The two-dimensional code is an extension of the one-dimensional code.

For power uprate conditions, a full two-dimensional calculation of vessel neutron fluences was performed using the DORT[12] discrete ordinates transport code which is an updated version of the DOT code[11]. The core was modeled by specifying two homogeneous regions representing the central and outer portions of the core in each of 25 axial intervals. The Los Alamos National Laboratory 80-group cross sections were used as the source of the microscopic cross sections data.

Subsequent to the above-described initial and power uprate calculations, updated neutron vessel fluence calculations were performed, using methods in compliance with RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," dated March 2001, which have been applied to support P-T limit curves for up to 32 EFPY, applying ASME Code Case N-640. The P-T limits for 32 EFPY reported in Reference 13 were derived using NRC approved methods in Reference 17. The detailed discussion of the NMP2 RG 1.190 methods is documented in References 13 and 14. Future evaluations of reactor vessel fluence will be completed using a method in accordance with the recommendations of RG 1.190 (as noted in Reference 15). NRC approval of the Unit 2 neutron fluence calculational methodology is documented in Reference 16.

4.1.4.6 Thermal-Hydraulic Calculations

The descriptions of the thermal-hydraulic models are given in Section 4 of GESTAR II[10].

4.1.5 References

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

2. Wilson, E. L. 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.

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

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

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

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

7. Burgess, A. B. User Guide and Engineering Description of HEATER Computer Program, March 1974.

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

9. Rashid, Y. R. Users Manual for CRPLS01 Computer Program, NEDO-23538, December 1976.

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

11. RSIC Computer Code Collection - DOT IV Version 4.3 - One and Two-Dimensional Transport Code System, CCC-429, Oak Ridge National Laboratory Report, November 1983.

12. RSIC Computer Code Collection - TORT Three-Dimensional Discrete Ordinates Transport, CCC-543, Oak Ridge National Laboratory, October 1991.

13. Nine Mile Point Unit 2 Pressure and Temperature Limits Report (PLTR), PTLR-2, Revision 0.

14. Benchmarking of Nine Mile Point Unit 1 and Unit 2 Neutron Transport Calculations, MPM-402781 Revision 1, August 2003.

15. NRC Letter to NMPNS dated November 8, 2004, "Nine Mile Point Nuclear Station Unit Nos. 1 and 2 - Issuance of Amendments RE: Implementation of the Reactor Pressure Vessel Integrated Surveillance Program (TAC Nos. MC1758 and MC1759)."

16. NRC Letter to NMPNS dated October 27, 2003, "Nine Mile Point Nuclear Station, Unit No. 1 - Issuance of Amendment Re: Pressure-Temperature Limit Curves and Tables (TAC No. MB6687)."

17. GEH P-T Curve Licensing Topical Report, NEDC-33178P-A, Revision 1, "GE Hitachi Nuclear Energy Methodology for Development of Reactor Pressure Vessel Pressure-Temperature Curves," (GEH Proprietary), June 2009.

4.2 FUEL SYSTEM DESIGN

The fuel system design for Unit 2 is similar to that reviewed and approved for GESSAR II[1]. Methods and criteria used to evaluate fuel system performance are also identical to those used for GESSAR II. The results of the NRC review of Section 4.2 of GESSAR II, documented in References 2 and 3, are, therefore, applicable to Nine Mile Point - Unit 2 (Unit 2).

The control assembly shall have a nominal axial absorber length of 143 in.

The Unit 2 post-irradiation fuel surveillance program for GE-designed and manufactured fuel assemblies is the program proposed by GE[4-8]. This program had been approved previously as satisfying Standard Review Plan (SRP) 4.2.II.D.3.

4.2.1 References

1. General Electric Standard Safety Analysis Report, Docket No. 50-447.

2. Safety Evaluation Report Related to the Final Design Approval of the GESSAR II BWR/6 Nuclear Island Design, NUREG-0979, April 1983.

3. Safety Evaluation Report Related to the Final Design Approval of the GESSAR II BWR/6 Nuclear Island Design, NUREG-0979 (Supplement No. 1), July 1983.

4. Correspondence from J. S. Charnley (GE) to C. H. Berlinger (NRC) concerning Post-Irradiation Fuel Surveillance Program, November 23, 1983.

5. Correspondence from L. S. Rubenstein (NRC) to R. L. Gridley (GE) concerning Post-Irradiation Fuel Surveillance Program, January 18, 1984.

6. Correspondence from J. S. Charnley (GE) to L. S. Rubenstein (NRC) concerning Fuel Surveillance Program, February 29, 1984.

7. Correspondence from J. S. Charnley (GE) to L. S. Rubenstein (NRC) concerning Additional Details Regarding Fuel Surveillance Program, May 25, 1984.

8. Correspondence from L. S. Rubenstein (NRC) to R. L. Gridley (GE) concerning Acceptance of GE Proposed Fuel Surveillance Program, June 27, 1984.

4.3 NUCLEAR DESIGN

The information in Section 4.3 is provided in the Licensing Topical Report, General Electric Standard Application for Reactor Fuel, GESTAR II[1]. The subsection numbers in Section 4.3 directly correspond to the subsection numbers of Appendix A of GESTAR II[1]. Any additions or differences are given below for each applicable subsection.

4.3.1 Design Bases

See Section A.4.3.1 of GESTAR II[1].

4.3.2 Description

See Section A.4.3.2 of GESTAR II[1].

4.3.2.1 Nuclear Design Description

The information is given in Subsection A.4.3.2.1 of GESTAR II, except for the reference (initial) core loading pattern which is described and shown on Figure 3-1a of NEDE-20944-P-1[2]. The initial core uses three enrichments of fuel bundles, which are shown on Figures 2-2.45, 2-2.49a and b, and 2-2.106a and b of GESTAR II (NEDE-24011-P-A-6, April 1983). Cycle-specific nuclear design description is covered in Appendix A, Section A.4.3.2.1.

4.3.2.2 Power Distribution

4.3.2.2.1 Power Distribution Calculations

A full range of typical calculated power distributions, along with the resultant exposure shapes and the corresponding control rod patterns, are shown in Appendix A of NEDE-20944-P-1[2].

4.3.2.2.2 Power Distribution Measurements

See Section A.4.3.2.2.2 of GESTAR II[1].

4.3.2.2.3 Power Distribution Accuracy

See Section A.4.3.2.2.3 of GESTAR II[1].

4.3.2.2.4 Power Distribution Anomalies

See Section A.4.3.2.2.4 of GESTAR II[1].

4.3.2.3 Reactivity Coefficient

See Section A.4.3.2.3 of GESTAR II[1].

4.3.2.4 Control Requirements

See Section A.4.3.2.4 of GESTAR II[1].

4.3.2.4.1 Shutdown Reactivity

The information is given in Subsection A.4.3.2.4.1 of GESTAR II[1] except for the cold shutdown margin for the reference initial core loading pattern which is given in Table 4.3-1.

4.3.2.4.2 Reactivity Variations

The information is given in Subsection A.4.3.2.4.2 of GESTAR II[1]. The combined effects of the individual constituents of reactivity are accounted for in each K_{eff} in Table 4.3-1.

4.3.2.5 Control Rod Patterns and Reactivity Worths

This information is discussed in Section 3.2.5 of NEDE-20944-P-1[2]. Control rod patterns and the associated power distributions for a typical BWR 5 are presented in Appendix A of NEDE-20944-P-1[2]. These control rod patterns are calculated with the BWR core simulator. Qualification for this model is discussed and referenced in Section 3.3 of GESTAR II[1].

4.3.2.5.1 Scram Reactivity

Scram reactivity is calculated as described in Section S.2 of GESTAR II[1] and discussed in Section 3.2.5.3 of NEDE-20944-P-1[2].

4.3.2.6 Criticality of Reactor During Refueling

See Section A.4.3.2.6 of GESTAR II[1].

4.3.2.7 Stability

See Section A.4.3.2.7 of GESTAR II[1].

4.3.2.7.1 Xenon Transients

See Section A.4.3.2.7.1 of GESTAR II[1].

4.3.2.7.2 Thermal-Hydraulic Stability

See Section 4.4.4.

4.3.2.8 Vessel Irradiations

The neutron fluxes at the vessel were originally calculated using the one-dimensional discrete ordinates transport code described in Section 4.1.4.5. The discrete ordinates code was used in a distributed source mode with cylindrical geometry. The geometry described six regions from the center of the core to a point beyond the vessel. The core region was modeled as a single homogenized cylindrical region. The coolant water region between the fuel channel and the shroud was described as containing saturated water at 550°F and 1,050 psi. The material

compositions for the stainless steel in the shroud and the carbon steel in the vessel contain the mixtures by weight as specified in the ASME material specifications for ASME SA-240, 304L, and ASME SA-533 Grade B. In the region between the shroud and the vessel, the presence of the jet pumps was ignored. A simple diagram showing the regions, dimensions, and weight fractions is shown on Figure 4.3-1.

The distributed source used for this analysis was obtained from the gross radial power description. The distributed source at any point in the core is the product of the power from the power description and the neutron yield from fission. By using the neutron energy spectrum, the distributed source is obtained for position and energy. The integral over position and energy is normalized to the total number of neutrons in the core region. The core region is defined as a 1-cm thick disc with no transverse leakage. The power in this core region is set equal to the average power in the axial direction. The radial power distribution is shown on Figure 4.3-2.

The neutron fluence is determined from the calculated flux by assuming that the plant is operated 90 percent of the time at 90 percent power level for 40 yr, or equivalent to 1×10^9 full power seconds. The neutron fluence will be re-evaluated as required by the plant operating history. The originally calculated fluxes and fluence are listed in Table 4.3-2. The originally calculated neutron flux leaving the cylindrical core is listed in Table 4.3-3.

Vessel neutron fluences were reevaluated for extended power uprate (EPU) conditions using a full two-dimensional calculation method, as described in Section 4.1.4.5. The results indicate that the original neutron fluence basis described above continues to bound operation of the unit at the stretch uprated power level of 3,988 MWt.[6] Vessel neutron fluences were also reevaluated for EPU conditions using a full two-dimensional calculation method, as described in Section 4.1.4.5. The results indicate that the original neutron fluence basis described above continues to bound operation of the unit at the extended uprated power level of 3,988 MWt.[6]

Subsequent to the above-described initial and power uprate evaluations, reactor vessel neutron fluence has been evaluated using a method in accordance with the recommendations of RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," dated March 2001. The 32 EFPY P-T limit curves were derived based on NMP2 RG 1.190 fluence methods. These methods are documented in References 4 and 5. Future evaluations of reactor vessel fluence will be completed using a method in accordance with the recommendations of RG 1.190, as noted in Section 4.1.4.5.

4.3.3 Analytical Methods

See Section A.4.3.3 of GESTAR II[1].

4.3.4 Changes

See Section A.4.3.4 of GESTAR II[1].

4.3.5 References

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

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

3. Licensing Topical Report, Power Uprate Licensing Evaluation for Nine Mile Point Nuclear Power Station, Unit 2, NEDC-31994P, Revision 1, May 1993.

4. Nine Mile Point Unit 2 Pressure and Temperature Limits Report (PTLR), PTLR-2, Revision 0.

5. Benchmarking of Nine Mile Point Unit 1 and Unit 2 Neutron Transport Calculations, MPM-402781 Revision 1, August 2003.

6. Safety Analysis Report for Nine Mile Point Nuclear Power Station, Unit 2, Constant Pressure Power Uprate, NEDC-33351P, Revision 0, May 2009.

TABLE 4.3-1
(Sheet 1 of 1)
CALCULATED CORE EFFECTIVE MULTIPLICATION AND
CONTROL SYSTEM WORTH - NO VOIDS, 20°C

Beginning of Cycle-1, K_{eff}	
Uncontrolled	1.1165
Fully controlled	0.9302
Strongest control rod out (22-55)	0.9710
R, Maximum increase in cold core reactivity with exposure Cycle-1, Δk	0.0047

TABLE 4.3-2
(Sheet 1 of 1)
ORIGINAL CALCULATED NEUTRON FLUXES
USED TO EVALUATE VESSEL IRRADIATION

Neutron Energy (MeV)	Average Flux in Core (n/cm^2-sec)	Flux at Core Boundary (n/cm^2-sec)	Flux at Inside Surface Vessel (n/cm^2-sec)	Flux at 1/4 T Vessel (n/cm^2-sec)
>3.0	1.5×10^{13}	5.4×10^{12}	3.3×10^{8}	1.6×10^{8}
1.0-3.0	3.3×10^{13}	1.2×10^{13}	2.8×10^{8}	2.3×10^{8}
0.1-1.0	5.3×10^{13}	1.7×10^{13}	4.9×10^{8}	6.7×10^{8}

NOTES:

1. The calculated flux is a maximum in the axial direction but average over the azimuthal angle.

2. Maximum fluence >1.0 MeV = 1.1×10^{18} n/cm^2 at 1/4 T of vessel. The maximum fluence is calculated using 1×10^9 full power seconds. The fluence includes an azimuthal peaking factor and a factor to cover analytical uncertainties. The azimuthal peaking factor is derived from the results of a two-dimensional transport calculation. The two-dimensional analysis models the reactor bundle pattern in (R,θ) geometry. Fluxes are calculated at the inside radius of the vessel. The peaking factor used is 1.4.

   In addition to the angular peaking factor, a safety factor of 2 was applied to ensure that the predicted values are conservative.

TABLE 4.3-3
(Sheet 1 of 1)
ORIGINAL CALCULATED NEUTRON FLUX AT CORE EQUIVALENT BOUNDARY

Group	Lower Energy Bound (ev)	Flux (n/cm^2-sec)
1	10.0×10^{6}	4.6×10^{10}
2	6.065×10^{6}	6.1×10^{11}
3	3.679×10^{6}	2.1×10^{12}
4	2.231×10^{6}	4.2×10^{12}
5	1.353×10^{6}	4.4×10^{12}
6	8.208×10^{5}	3.9×10^{12}
7	4.979×10^{5}	4.0×10^{12}
8	3.020×10^{5}	2.8×10^{12}
9	1.832×10^{5}	2.3×10^{12}
10	1.111×10^{5}	1.8×10^{12}
11	6.732×10^{4}	1.4×10^{12}
12	4.087×10^{4}	1.1×10^{12}
13	2.478×10^{4}	1.0×10^{12}
14	1.503×10^{4}	1.0×10^{12}
15	9.119×10^{3}	9.6×10^{11}
16	5.531×10^{3}	9.4×10^{11}
17	3.355×10^{3}	9.4×10^{11}
18	2.034×10^{3}	9.1×10^{11}
19	1.010×10^{3}	1.3×10^{12}
20	2.492×10^{2}	2.5×10^{12}
21	5.560×10^{1}	2.6×10^{12}
22	1.240×10^{1}	2.5×10^{12}
23	0.625	4.0×10^{12}
24	0.0	2.5×10^{13}

4.4 THERMAL-HYDRAULIC DESIGN

4.4.1 Design Bases

The design basis is given in Section A.4.4.1 of GESTAR II[1].

The original design steady-state minimum critical power ratio (MCPR) limit and the peak LHGR are given in Table 4.4-1. Values for reload cores, including the effects of uprated power operation, are provided in Appendix A.

4.4.2 Description of Thermal-Hydraulic Design of the Reactor Core

Information pertaining to Section 4.4.2 is given in Section A.4.4.2 of GESTAR II[1]. Additions or differences are provided below for each applicable subsection.

4.4.2.1 Summary Comparison

An evaluation of plant performance from a thermal-hydraulic standpoint is provided in Section 4.4.3. A tabulation of thermal-hydraulic parameters of the initial core and a comparison of this reactor with others of similar design are given in Table 4.4-1. Parameters for operation at 3,467 MWt are summarized in the stretch power uprate (SPU) evaluation.[5] Parameters for operation at 3,988 MWt are summarized in the EPU evaluation.[6] Any changes for reload cores are provided in Appendix A.

4.4.2.2 Critical Power Ratio

See Section A.4.4.2.2 of GESTAR II[1].

4.4.2.3 Linear Heat Generation Rate

See Section A.4.4.2.3 of GESTAR II[1].

4.4.2.4 Void Fraction Distribution

See Section A.4.4.2.4 of GESTAR II[1].

The core average and maximum exit void fractions in the core at original 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 core are given in Table 4.4-2. 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. Similar distributions were evaluated for uprated operation[5], and cycle-specific information is used in the reload evaluations in Appendix A.

4.4.2.5 Core Coolant Flow Distribution and Orificing Pattern

See Section A.4.4.2.5 of GESTAR II[1].

4.4.2.6 Core Pressure Drop and Hydraulic Loads

See Section A.4.4.2.6 of GESTAR II[1].

4.4.2.7 Correlation and Physical Data

Substantial amounts of physical data have been obtained in support of the pressure drop and thermal-hydraulic loads. See Section A.4.4.2.7 of GESTAR II.

4.4.2.8 Thermal Effects of Operational Transients

See Section A.4.4.2.8 of GESTAR II[1].

4.4.2.9 Uncertainties in Estimates

See Section A.4.4.2.9 of GESTAR II[1]. In general, the CPR is not affected as crud accumulates on fuel rods[2,3]. Therefore, no modifications to the critical power correlation 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. A more detailed discussion of crud (service-induced variations) and its uncertainty is found in Section III of Reference 4.

4.4.2.10 Flux Tilt Considerations

The inherent design characteristics of the BWR are particularly well suited to handling 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 associated on-line computer provide the Operator with prompt and reliable power distribute information. Thus, the Operator can readily use control rods or other means to effectively limit 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 described in Technical Specifications.

4.4.3 Description of the Thermal-Hydraulic Design of the Reactor Coolant System

4.4.3.1 Plant Configuration Data

The RCS is described in Section 5.4 and shown in isometric perspective on Figure 5.4-1. The piping sizes, fittings, and valves are listed in Table 5.4-1.

4.4.3.1.1 Reactor Coolant System Thermal-Hydraulic Data

The steady state distribution of temperature, pressure, and flow rate for each flow path in the RCS is shown on Figure 5.1-1a.

4.4.3.1.2 Reactor Coolant System Geometric Data

Volumes of regions and components within the reactor vessel are shown on Figure 5.1-1b.

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 Figure 5.4-3. These curves are valid for all conditions with a normal operating range varying from 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, and other requirements. A typical power flow map for the power range of operation is shown on Figure 4.4-1. The Unit 2 Cycle 1 operating power flow map is shown on Figure 15.0-2, the map for EPU/MELLLA+ is shown on Figure 15.0-2a. 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. Rated Flow Control Line This line passes through 100-percent power at 100-percent flow. The operating state for the reactor follows this rod line (or similar ones) during recirculation flow changes with a fixed control rod pattern. The line is based on a constant xenon concentration at rated power and flow.

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

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

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

2. Constant Position Lines for Flow Control Valves B, C, D, and F These lines show the change in flow associated with power changes while maintaining FCVs at a constant position.

4.4.3.3.2 Regions of the Power Flow Map

Region I

This region defines the system operational capability with the recirculation pumps and motors being driven by the low-frequency motor generator (LFMG) set at 25-percent speed. Flow is controlled by the FCV, and power changes during normal startup and shutdown will be in this region. The normal operating procedure is to start up along curve C with the FCV wide open at 25-percent speed.

Region II

This region shows the area where 25-percent pump speed and 100-percent pump speed operating regimes overlap. The switching sequence from the LFMG set to 100-percent speed will be done in this region.

Region III

This is the low-power area of the operating map where cavitation can be expected in the recirculation pumps, jet pumps, or FCVs.

Operation within this region is precluded by system interlocks that trip the main motor from the 100-percent speed power source to the 25-percent speed power source.

Region IV

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

4.4.3.3.3 Design Features for Power Flow Control

The following limits and design features are employed to maintain power flow conditions to the required values shown on Figure 4.4-1.

1. Minimum Power Limits at Intermediate and High Core Flows To prevent cavitation in the recirculation pumps, jet pumps, and FCVs, the recirculation system has an interlock to trip off the 100-percent speed power source and close the 25-percent speed power source if the difference between steam line temperature and recirculation pump inlet temperature is less than a preset value (10.7°F). This differential temperature is measured using high-accuracy resistance temperature detectors (RTDs) with a sensing error of less than 0.2°F at the two standard deviation (two sigma) confidence level. This action is initiated electronically through a 45-sec time delay. The interlock is active in both the automatic and manual operation modes.

2. Minimum Power Limit at Low Core Flow During low-power, low-loop flow operations, the temperature differential interlock may not provide sufficient cavitation protection to the FCVs. Therefore, the system has an interlock to trip off the 100-percent speed power source and close the 25-percent speed power source if the feedwater flow falls below a preset level (about 19 percent of rated). The speed change action is electronically initiated. This interlock is active during both automatic and manual modes of operation.

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

4. Valve Position To prevent structural or cavitation damage to the recirculation pump due to pump suction flow starvation, the system has an interlock to prevent starting the pumps, or to trip the pumps if the suction

or discharge block valves are at less than 90 percent open position. This circuit is activated by a position limit switch and is active before the pump is started, during manual operation mode, and during automatic operation mode.

Flow Control

The principal modes of normal operation with valve flow control-LFMG set are summarized as follows: The recirculation pumps are started on the 100-percent speed power source in order to unseat the pump bearings. Suction and discharge block valves are full open and the FCV is in the minimum position. When the pump is near full speed, the main power source is tripped and the pump allowed to coast down to approximately 25-percent speed where the LFMG set will power the pump and motor. The FCV is then opened to the maximum position at which point reactor heatup and pressurization can begin. When operating pressure has been established, reactor power can be increased. This power flow increase will follow a line within Region I of the flow control map shown on Figure 4.4-1.

When reactor power is greater than approximately 26 to 30 percent of rated, the low feedwater flow interlock is cleared and the main recirculation pumps can be switched to the 100-percent speed power source. The FCV is closed to the minimum position before the speed change to prevent large increases in core power and a potential flux scram. This operation occurs within Region II of the operating map. The system is then brought to the desired power flow level within the normal operating area of the map (Region IV) by opening the FCVs and withdrawing control rods.

Control rod withdrawal with constant FCV position will result in power/flow changes along lines of constant c. FCV movement with constant control rod position will result in power/flow changes along, or nearly parallel to, the rated flow control line.

4.4.3.4 Temperature-Power Operating Map (PWR)

Not applicable.

4.4.3.5 Load-Following Characteristics

Large negative operating reactivity coefficients inherent in the BWR provide the following important advantages:

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 BWR includes the ability to follow load demands over a reasonable range without requiring Operator action. Reactor power can be controlled automatically by flow control over approximately a 35-percent power range at, for example, approximately 1 percent/sec and for a 10-percent step-load change.

4.4.3.6 Thermal-Hydraulic Characteristics Summary Table

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

4.4.4 Evaluation

See Section A.4.4.4 of GESTAR II[1]. Results of the Cycle 1 stability analysis are given in Table 4.4-7 and on Figures 4.4-2, 4.4-3, 4.4-4, and 4.4-5. Evaluation for operation at 3,467 MWt is provided in the power uprate evaluation[5]. Cycle-specific evaluation is covered in Appendix A, Section A.4.4.4.

4.4.4.1 Critical Power

The GEXL critical power correlation is utilized for Cycle 1 fuel (P8X8R) in thermal-hydraulic evaluations. This correlation is discussed in Section 4.4.2.2. Cycle-specific evaluation is covered in Appendix A, Section A.4.4.4.1.

4.4.4.2 Core Hydraulics

Core hydraulic models and correlations are discussed in Sections 4.4.2.6, 4.4.2.7, and 4.4.4.5.

4.4.4.3 Influence of Power Distributions

The influence of power distributions on the thermal-hydraulic design is discussed in GESTAR II Appendix A[1].

4.4.4.4 Core Thermal Response

The thermal response of the core for accidents and expected transient conditions is discussed in Chapter 15 and Appendix A.

4.4.4.5 Analytical Methods

The analytical methods, thermodynamic data, and hydrodynamic data used in determining the thermal-hydraulic characteristics of the core are similar to those used throughout the nuclear power industry. Core thermal-hydraulic analyses are performed with the aid of a digital computer program. This program models the reactor core through a hydraulic description of orifices, lower tie-plates, fuel rods, fuel rod spacers, upper tie-plates, fuel channel, and core bypass flow paths.

4.4.5 Testing and Verification

See Section A.4.4.5 of GESTAR II[1].

4.4.6 Instrumentation Requirements

The reactor vessel instrumentation monitors the key reactor vessel operating parameters during planned operations. This ensures sufficient control of the parameters. The reactor vessel parameters are discussed in Chapter 7.

4.4.6.1 Loose Parts Monitoring System

4.4.6.1.1 Design Basis

1. The loose parts monitoring system (LPMS) is designed to detect loose parts in the RCS.

2. The LPMS is designed to reduce the effects of variations in background noise on system capabilities for the detection of loose parts.

3. The LPMS is designed in conformance with Revision 1 (May 1981) of Regulatory Guide (RG) 1.133 through the alternate approach described in Section 1.8.

4. The LPMS and its integral loose part event analysis computer (LPEAC) system is designed to provide a level of confidence in the validity of any alarm condition.

4.4.6.1.2 System Description

The function of this system is to detect loose parts in the RCS. Alarm functions are provided for reactor vessel internal monitoring channels. Non-RG 1.133 required channels (recirculation loop channels 3, 4, 5 and 6) do not provide automatic alarm functions but still allow on-demand diagnostic capabilities. Loose parts are those metallic objects that can be moved physically by reactor flow. A secondary function of the Unit 2 system is to assist plant personnel in accurately and quickly locating detected loose parts.

Sensing devices mounted within the containment are designed to withstand operating basis earthquake (OBE) and are redundant (ten sensors located on opposite sides of the reactor at five elevations) as described in Table 4.4-8. Separation is maintained between redundant monitoring channel circuits up to and including the main relay room monitors (which contain the alarm circuits). While these precautions have been taken, the system is not considered safety related.

The system has been designed to discriminate between regular noise and signals caused by a loose part.

The LPMS consists of accelerometers (sensors), preamplifiers, loose part monitors, LPEAC, computer display monitor, keyboard and printer. The LPMS is capable of detecting loose part mass from 0.1 to 30 lb with an impact frequency range 1-10 kHz and peak amplitude up to 100 g. The detected signal amplitude is a function of impact mass, energy, and shape.

The LPMS sensors are mounted on the exterior of the reactor vessel primary coolant system. These sensors monitor lower vessel tubes, recirculation pump, recirculation inlet lines, feedwater lines, and instrument nozzles in the upper vessel region. The sensors are strapped or clamped to the measuring region in accordance with Table 4.4-8.

Special low-noise coaxial cables conduct the accelerometer signals to remote-mounted preamplifiers. The preamplifiers condition the signals for transmission over relatively long distances to the detector modules located in the main relay room on the loose part monitoring panel.

The main relay room is a low-radiation area during normal and transient conditions. During accidents, this area is served by the control room special filter train. Most surveillance and maintenance can be performed in this area, which minimizes radiation exposure. Maintenance inside the drywell will be controlled in accordance with the plant radiation protection procedures to minimize exposure.

At the loose part monitoring panel, the signals from the loose parts channel are compared with preset levels to generate alarms. An alarm is generated when a signal exceeds the preset level for a specified period of time.

Each channel is connected to a selector switch for indication on a digital panel meter and auxiliary readout equipment, such as a LPEAC or audio output.

A primary consideration in the design of the LPMS and its integration with LPEAC is to validate alarm information in terms of waveform capture and spectral contents. In the event the alert level is reached or exceeded, the LPEAC will perform calculation and evaluation automatically on captured data to determine if the loose part alarm is a valid metal impact or a spurious alarm. Should the alarm be judged to be valid, control room annunciator will be energized and the calculated values and waveforms will be recorded in the LPEAC. The recorded data can be displayed at the monitor and printed out from the printer through the keyboard.

The following considerations are addressed to establish the alert level:

1. The alert level considers signals such as normal hydraulic, mechanical, and electric noise due to normal and transient conditions.

2. The alert level considers varying alarm setpoints from sensor to sensor to aid in compensating for specific transducer location noise.

3. Alert levels are evaluated as part of the startup test program to ensure as low of a false alarm rate as practical over normal and transient operating conditions.

4. Alert levels are based on sensor response to the 0.5 ft-lb impact.

The system's on-line sensitivity for detecting a loose part is a loose parts impact energy of 0.5 ft-lb within 3 ft of a sensor.

The system is designed to operate continuously without Operator supervision, except for routine system testing.

The LPMS LPEAC is designed to make preventive maintenance and system monitoring a simple and reliable process. The system can identify degradation in sensor signal condition before the data from a failed sensor is incorrectly used in the evaluation.

4.4.6.1.3 Safety Evaluation

The LPMS is intended to be used for information purposes only and is not a safety-related system. The system conforms with RG 1.133 through the alternate approach described in Section 1.8. The design of the equipment was specifically selected for the application at a nuclear power plant during normal operation (pressure, temperature, humidity, radiation). For example, the LPMS monitoring cabinet is designed for mild environments of the main relay room where it is located. Similarly, the cables, accelerometers, and preamplifiers were designed for their environmental conditions in the primary and secondary containments. The equipment has been "qualified" on the basis of past experience in operating nuclear plants. The plant personnel use the LPMS to assist in the detection of anomalous loose parts. They also use it to assist in determining the location of any anomalous loose parts. The Operators do not rely solely on this system or information provided by this system for the performance of any safety-related action. Any evaluations or actions taken to confirm the presence of a loose part will be handled on a case-by-case basis. Guidance for evaluation is provided on Figures 4.4-7, 4.4-8, and 4.4-9.

4.4.6.1.4 LPMS Training and Calibration

<u>LPMS Training</u>

The scope of training for the onsite LPMS will cover the theory and operation of the LPMS system, including hands-on training. Emphasis will be placed on detection and characterization of loose parts and implementation of diagnostic concepts.

LPMS Calibration

The channel calibration is performed using calibrated impact hammers. Data are taken at various impact levels at various locations relative to each sensor. The data obtained provide information to determine the following system characteristics to be used as a baseline for plant operations:

The system is calibrated to provide alarm sensitivity to a 0.5 ft-lb impact within 3 ft of the sensor.

Additional calibration impacts are included to define:

1. Interchannel arrival order and delay times.

2. Change in amplitude of impact response with distance.

3. Identify changes in the arrival impact leading edge (dispersive effects) as it is received at the sensor from various distances.

4.4.7 References

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

2. McBeth, R. V.; Trenberth, R.; and Wood, R. W. 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.

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

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

5. Licensing Topical Report, Power Uprate Licensing Evaluation for Nine Mile Point Nuclear Power Station, Unit 2, NEDC-31994P, Revision 1, May 1993.

6. Safety Analysis Report for Nine Mile Point Nuclear Power Station, Unit 2, Constant Pressure Power Uprate, NEDC-33351P, Revision 0, May 2009.

TABLE 4.4-1
(Sheet 1 of 2)
THERMAL-HYDRAULIC DESIGN CHARACTERISTICS
OF THE INITIAL REACTOR CORE

General Operating Conditions	Nine Mile Point Unit 2**	LaSalle	WPPSS 2
Reference design thermal output, MWt	3,323	3,293	3,323
Power level for engineered safety features, MWt	3,467	3,458	3,489
Steam flow rate, at 420°F final feedwater temperature, millions lb/hr	14.27	14.166	14.30
Core coolant flow rate, millions lb/hr	108.5	106.5	108.5
Feedwater flow rate, millions lb/hr	14.24	14.127	14.26
System pressure, nominal in steam dome, psia	1,020	1,020	1,020
System pressure, nominal core design, psia	1,035	1,035	1,035
Coolant saturation temperature at core design pressure, °F	549	548.8	549
Average power density, kW/l	49.15	50.0	49.15
Maximum LHGR, kW/ft	13.4	13.4	13.4
Average LHGR, kW/ft	5.4	5.39	5.4
Core total heat transfer area, ft^2	74,871	74,872	74,871
Maximum heat flux, Btu/hr-ft^2	361,600	361,000	361,500
Average heat flux, Btu/hr-ft^2	145,100	145,208	145,100
Operating Limit CPR*			
(Option A)	1.28		1.24
(Option B)	1.24		

TABLE 4.4-1 (Cont'd.)
(Sheet 1 of 2)

General Operating Conditions	Nine Mile Point Unit 2**	LaSalle	WPPSS 2
Core inlet enthalpy at 420°F FFWT, Btu/lb	527.5	532.8	527.6
Core inlet temperature, at 420°F FFWT, °F	533	531.2	533
Core maximum exit voids within assemblies, %	76.2	76.0	76.0
Core average void fraction, active coolant	0.408	0.418	0.418
Maximum fuel temperature, °F	3,435	3,325	3,435
Active coolant flow area per assembly, in^2	15.82	15.67	15.824
Core average inlet velocity, ft/sec	6.91	7.0	6.88
Maximum inlet velocity, ft/sec	8.0	7.5	7.28
Total core pressure drop, psi	24.74	24.8	24.74
Core support plate pressure drop, psi	20.32	20.33	20.32
Average orifice pressure drop			
Central region, psi	6.0	8.44	6.03
Peripheral region, psi	16.0	17.24	16.54
Maximum channel pressure loading, psi	13.28	13.4	13.28

* Extracted from Table 15.0-2.

** This table contains parameters for the original design compared to similar BWR/5 plants. The power uprate evaluation[5] provides information for operation at 3,467 MWt. In addition, any changes in analytical inputs and thermal limits for reload cores are provided in Appendix A.

TABLE 4.4-2
(Sheet 1 of 1)
VOID DISTRIBUTION - INITIAL CORE

Core Average Value = 0.408
Maximum Exit Value = 0.738
Active Fuel Length = 150 Inches

	Node	Core Average (Average Node Value)	Maximum Channel (End of Node Value)
Bottom	1	0.000	0.000
of Core	2	0.000	0.006
	3	0.008	0.062
	4	0.041	0.159
	5	0.103	0.258
	6	0.179	0.342
	7	0.254	0.415
	8	0.322	0.473
	9	0.378	0.519
	10	0.424	0.557
	11	0.461	0.587
	12	0.491	0.611
	13	0.516	0.632
	14	0.536	0.648
	15	0.554	0.663
	16	0.569	0.676
	17	0.583	0.688
	18	0.596	0.700
	19	0.609	0.711
	20	0.620	0.720
	21	0.630	0.728
	22	0.638	0.734
	23	0.643	0.737
Top	24	0.645	0.738
of Core			

TABLE 4.4-3
(Sheet 1 of 1)
FLOW QUALITY DISTRIBUTION - INITIAL CORE

Core Average Value = 0.072
Maximum Exit Value = 0.237
Active Fuel Length = 150 Inches

	Node	Core Average (Average Node Value)	Maximum Channel (End of Node Value)
Bottom of Core	1	0.000	0.000
	2	0.000	0.000
	3	0.000	0.002
	4	0.001	0.008
	5	0.004	0.018
	6	0.010	0.032
	7	0.018	0.048
	8	0.029	0.065
	9	0.040	0.081
	10	0.051	0.097
	11	0.061	0.113
	12	0.071	0.127
	13	0.080	0.140
	14	0.089	0.152
	15	0.097	0.163
	16	0.104	0.174
	17	0.111	0.184
	18	0.118	0.195
	19	0.126	0.206
	20	0.133	0.216
	21	0.139	0.225
	22	0.144	0.231
	23	0.148	0.236
Top of Core	24	0.150	0.237

TABLE 4.4-4
(Sheet 1 of 1)
AXIAL POWER DISTRIBUTION USED TO GENERATE
VOID AND QUALITY DISTRIBUTIONS - INITIAL CORE

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

TABLE 4.4-5
(Sheet 1 of 1)
REACTOR COOLANT SYSTEM GEOMETRIC DATA

	Flow Path Length (in)	Height and Liquid Level (in)	Elevation of Bottom of Each Volume[(1)] (in)	Minimum Flow Areas (sq ft)
Lower plenum	216.5	216.5 216.5	-172.5	71.5
Core	164.5	164.5 164.5	44.0	142.0[(2)]
Upper plenum and separators	178.0	178.0 178.0	208.0	49.5
Dome (above normal water level)	312.0	312.0 0.0	386.0	343.5
Downcomer area	321.0	321.0 321.0	-51.0	79.5
Recirculation loops and jet pumps	108.5 ft[(3)]	403.0 403.0	-394.5	132.5 in^{2}[(3)]

(1) Reference point is recirculation nozzle outlet centerline.
(2) Includes bypass.
(3) One loop.

TABLE 4.4-6
(Sheet 1 of 1)
LENGTHS OF SAFETY INJECTION LINES

System	Pipe Size (in)	Length[1] (± 1 ft)
HPCS	10	3'-0"
	12	60'-10"
LPCS	10	4'-8"
	12	41'-9"
RHR A	12	79'-1"
RHR B	12	80'-6"
RHR C	12	62'-10"

(1) From inside primary containment wall to RPV nozzle safe end.

TABLE 4.4-7
(Sheet 1 of 1)
STABILITY ANALYSIS RESULTS

(Cycle-1 Most Limiting Conditions)

<u>Rod Line Analyzed</u>

Natural circulation		51.5%
Rod pattern		105.0%

<u>Decay Ratio</u>

Total system stability, x_2/x_0	=	See Figures 4.4-3, 4, and 5
Reactor core stability, x_2/x_0	=	0.6 (0.36 Hz resonant frequency)
Channel hydrodynamic performance, x_2/x_0	=	0.49 (0.38 Hz resonant frequency)

TABLE 4.4-8
(Sheet 1 of 1)
SENSOR LOCATION AND MOUNTING INFORMATION

Channel	Instrument Tag No.	Sensor Location	Mounting	Tube/Pipe Diameter - in Inches
1	2LPM-NBE1A	CRD stub tube	Strap	1
2	2LPM-NBE1B		Strap	Tube
3	2LPM-NBE2A	Recirculation pumps	Strap	24
4	2LPM-NBE2B		Strap	Pipe
5	2LPM-NBE3A	Vessel recircu-	Strap	24
6	2LPM-NBE3B	lation inlet lines (180 deg apart)	Strap	Pipe
7	2LPM-NBE4A	Feedwater lines	Strap	12
8	2LPM-NBE4B	(180 deg apart; 90 deg from recirculation line sensors)	Strap	Pipe
9	2LPM-NBE5A	Instrument nozzles	Clamp	1
10	2LPM-NBE5B	upper vessel region	Clamp	Pipe

4.5 REACTOR MATERIALS

4.5.1 Control Rod System Structural Materials

4.5.1.1 Material Specifications

The following material listing applies to the CRD mechanism supplied for Unit 2. The position indicator and minor nonstructural items are omitted.

<u>Cylinder, Tube, and Flange Assembly</u>

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

<u>Piston Tube Assembly</u>

Piston tube	ASME SA-249 or SA-479, Grade XM-19
Stud	ASME SA-479, Grade XM-19
Head	ASME SA-182, Grade F304
Indicator tube	ASME SA-312, Type 316
Cap	ASME SA-182, Grade F304

<u>Drive Line Assembly</u>

Coupling stud	Inconel X-750
Index tube	ASME SA-479 or SA-249, Grade XM-19
Piston head	Armco 17-4 PH
Coupling	ASTM A312, Grade TP 304 or ASTM A269, Grade TP 304
Magnet housing	ASTM A213, A249, or A312, Grade TP 316L, or ASTM A269, A312, Grade 304

<u>Collet Assembly</u>

Collet piston	ASTM A269 or A312, Grade TP 304
Finger	Inconel X-750
Retainer	ASTM A269, Grade TP 304
Guide cap	ASTM A269, Grade TP 304

<u>Miscellaneous Parts</u>

Stop piston	ASTM A276, Type 304
Connector	ASTM A276, Type 304
O-Ring spacer	ASME SA-240, Type 304
Nut	ASME SA-193, Grade B8

Barrel	ASTM A269, Grade TP 304; ASTM A312, Grade TP 304; or ASTM A240, Type 304
Collet spring	Inconel X-750
Ring flange	ASME SA-182, Grade F304

The austenitic material 300 series stainless steels that are listed are all in the annealed condition (with the exception of the outer tube in the cylinder, tube, and flange assembly), and their properties are available published literature. The outer tube is approximately 1/8 hard, and has a tensile of 90,000/125,000 psi, yield of 50,000/85,000 psi, and minimum elongation of 25 percent.

The coupling spud, collet fingers, and collet spring are fabricated from Inconel X-750 in the annealed or equalized condition and aged 20 hr at 1,300°F to produce a tensile of 165,000 psi minimum, yield of 105,000 psi minimum, and elongation of 20 percent minimum. The piston head is Armco 17-4 PH in condition H-1100 (aged 4 hr at 1,100°F), with a tensile of 140,000 psi minimum, yield of 115,000 psi minimum, and 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 steel materials, except SA-479 or SA-249, Grade XM-19, have been successfully used for the past 10 to 15 yr in similar drive mechanisms. Extensive laboratory tests have demonstrated that ASME SA-479 or SA-249, Grade XM-19, is a suitable material and that it is resistant to stress corrosion in a BWR environment.

Special Materials

No cold-worked austenitic stainless steels with a yield strength greater than 90,000 psi are employed in the CRD system. Martensitic precipitation-hardened stainless steel, Armco 17-4 PH, is used for the piston head, stop piston buffer shaft, and buffer piston. This material is aged to the H-1100 condition to produce resistance to stress corrosion cracking in BWR environments. Armco 17-4 PH (H-1100) has been successfully used for the past 10 to 15 yr in BWR drive mechanisms.

4.5.1.2 Austenitic Stainless Steel Components

Processes, Inspections, and Tests

Solution-annealed 300 series stainless steel material used in fabricating CRD parts is verified as being correctly solution-annealed by testing in accordance with ATMS-A262.

Two special processes are employed that subject selected 300 series stainless steel components to temperatures in the sensitization range:

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

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

Nitriding is accomplished using a proprietary process called New Malcomizing. Components are exposed to a temperature of about 1,080°F for approximately 20 hr during the nitriding cycle.

Colmonoy hard-surfaced components have performed successfully for the past 10 to 15 yr 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 nitrided surfaces are accessible for visual examination.

Welding is performed in accordance with Section IX of the ASME Boiler and Pressure Vessel Code. Heat input for stainless steel welds is restricted to a maximum of 50,000 joules/in and interpass temperature to 350°F. These controls are employed to avoid severe sensitization and are assessed to meet the intent of RG 1.44.

Control of Delta Ferrite Content

CRD parts were fabricated after the issuance of RG 1.31. All Type 308 weld metal was purchased to a specification which requires a minimum of 5-percent delta ferrite. Ferrite measurements were made with a calibrated magnetic instrument on undiluted weld pads for each lot and heat of weld filler metal. For the submerged arc welding process, measurements were made for each wire flux combination.

These procedures comply with RG 1.31.

4.5.1.3 Other Materials

These are discussed in Section 4.5.1.1.

4.5.1.4 Protection of Materials During Fabrication, Shipping, and Storage

All 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 which are completely removable by the applied cleaning

process. All contaminants are then required to be removed by the appropriate cleaning process prior to any of the following:

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

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

3. Release of parts for shipment.

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

The degree of surface cleanliness obtained by these procedures is assessed against and meets the requirements of RG 1.37.

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

After the second year, a yearly inspection of 2 percent of the units or a semiannual examination of the humidity indicators, where present, of 10 percent of the units is required to verify that the units are dry and in satisfactory condition. Position indicator probes are not subject to this inspection.

4.5.2 Reactor Internal Materials

4.5.2.1 Material Specifications

Materials used for the core support structure include:

Shroud support	Nickel chrome-iron-alloy, ASME SB-168
Shroud, core plate, and top guide	ASME SA-240, SA-193, SA-194, SA-182, SA-479, SA-312, ASTM A276, A249, A213, (Type 304L or 304)
Peripheral fuel supports	SA-312 Type 304
Core plate studs and nuts	ASME A-276, Type 304, SA-182 Grade F304
Control rod drive housing	ASME SA-312, Type 304 and ASME SA-182, Type 304

Control rod guide tube	ASTM A351 or ASME SA-351, Type CF8, ASME SA-358, SA-312, SA-249, and ASTM A312, A240, A276, A269, A358 (Type 304)
Orificed fuel support	ASME SA-351, Type CF8

Materials employed in other reactor internal structures include:

1. Steam Separator and Steam Dryer All materials are 300 series stainless steel:

Plate, sheet, and strip	ASTM A240, Type 304 or 316L
Forgings	ASTM A182, Grade F304
Bars	ASTM A276, Type 304 or 316L
Pipe	ASTM A312, Grade TP 304
Tube	ASTM A269, Grade TP 304
Bolting material	ASTM A193, Grade B8
Nuts	ASTM A194, Grade 8
Castings	ASTM A351, Grade CF8 (Type 304)

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

Castings	ASTM A351, Grade CF8 and ASME SA-351, Grade CF3 ASTM A194, Grade 8 or 8M
Bars	ASTM A276, Type 304 ASME SA-479, 316L ASTM A479, 304L
Bolts	ASTM A193, Grade B8 or B8M
Sheet and plate	ASTM A240, Type 304, and ASTM A240, Type 304L
Tubing	ASTM A269, Grade TP 304
Pipe	ASTM A358, Type 304, ASTM A312, Type 304, and ASME SA312, Grade TP 304

Welded fittings	ASTM A403, Grade WP304
Forging	ASME SA-182, Grade F304, ASTM B166, and ASTM A637, Grade 688

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

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

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

c. The inlet-mixer contains a pin and insert made of Inconel X-750 to Specification ASTM A637, Grade 688, or ASTM B637, Grade UNS N07750, Type 3.

d. The jet pump beam is fabricated from Modified Alloy 718 material, specified as Grade 718 Type 2 in ASME Code Case N-60-6. Use of this material for the jet pump beams was approved in License Amendment No. 141 (Reference 1).

3. <u>Core Spray Spargers and Core Spray Lines</u> Materials used for these components are:

ASME SA-312 Type 304L for core spray spargers.

ASME SA-376 Type 316L for core spray lines.

All core support structures are fabricated from ASME- and ASTM-specified materials and designed using ASME Section III as a guide. The other reactor internals are noncoded 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 ASME Boiler and Pressure Vessel 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 for CRD guide tubes, CRD housings, and peripheral fuel supports were supplied in accordance with applicable ASME material specifications. These specifications require a hydrostatic test on each length of

tubing. No specific nondestructive testing was performed on the tubes.

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

Regulatory Guide 1.31 Cold-worked stainless steels are not used in the reactor internals.

Regulatory Guide 1.31 All austenitic stainless steel weld filler materials were supplied with a minimum of 5 percent delta ferrite. This amount of ferrite is considered adequate to prevent microfissuring in austenitic stainless steel welds.

Purchase orders for the reactor internals were placed prior to the issuance of Revision 2 to RG 1.31. Ferrite measurements were made in accordance with the requirements of the ASME Code that require the use of the chemical composition in conjunction with the Schaeffler diagram to verify that weld filler metal contained a minimum of 5 percent delta ferrite.

An extensive test program performed by GE, with the concurrence of the Nuclear Regulatory Commission (NRC) Staff, 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. The 400 production welds evaluated in this program were fabricated with filler metal controlled in accordance with the Schaeffler diagram to contain a minimum of 5 percent ferrite. All of these production welds were assessed against and met the requirements of the interim regulatory position of RG 1.31, which was in effect at that time.

Regulatory Guide 1.34 Electroslag welding is not employed for any reactor internals.

Regulatory Guide 1.36 For external applications, all nonmetallic insulation is assessed against and meets the requirements of RG 1.36.

Regulatory Guide 1.37 Exposure to contaminants was avoided by carefully controlling all cleaning and processing materials that contact stainless steel during manufacture and construction. Any inadvertent surface contamination was removed to avoid potential detrimental effects.

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

The degree of cleanliness obtained by these procedures is assessed to meet the requirements of RG 1.37.

Regulatory Guide 1.44 All wrought austenitic stainless steel was purchased in the solution heat-treated condition. Heating above

800°F was prohibited (except for welding) unless the stainless steel was subsequently solution annealed. For Type 304 steel with carbon content in excess of 0.035 percent carbon, purchase specifications restricted the maximum weld heat input to 110,000 joules/in, and the weld interpass temperature to 350°F maximum. Welding was performed in accordance with Section IX of the ASME Boiler and Pressure Vessel Code. These controls were employed to avoid severe sensitization and are assessed to meet the intent of RG 1.44.

Regulatory Guide 1.71 There are few restrictive welds involved in the fabrication of items described in this section. Mockup welding was performed on the welds with most difficult access. Mockups were examined with radiography or by sectioning.

4.5.2.5 Other Materials

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

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

SB-166, SB-167, and SB-168 nickel-chrome-iron (Inconel 600)

SA-637, Grade 688 (Inconel X-750)

SB-637, Modified Alloy 718

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 hr at 1300°F. Modified Alloy 718 components are solution annealed and precipitation hardened for 6 hr 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.

4.5.3 Control Rod Drive Housing Supports

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

Item	Material
Grid	ASTM A-441
Disc springs	Schnorr, Type BS-125-71-8
Hex bolts and nuts	ASTM A-307
6 x 4 x 3/8 tubes	ASTM A-500, Grade B

For further CRD housing support information, refer to Section 4.6.1.2.

4.5.4 References

1. Nine Mile Point Nuclear Station - Unit 2, License Amendment No. 141 issued by NRC letter from R. V. Guzman to K. Langdon, dated April 13, 2012.

4.6 FUNCTIONAL DESIGN OF REACTIVITY CONTROL SYSTEMS

The reactivity control systems consist of control rods and CRDs, supplementary reactivity control for the core (Section 4.3) and the standby liquid control system (SLCS) (Section 9.3.5).

4.6.1 Control Rod Drive System

4.6.1.1 Control Rod Drive System Design

4.6.1.1.1 Design Bases

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. Each positioning device:

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

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

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

Power Generation Design Basis

The CRD system drive design provides for positioning the control rods to control power generation in the core.

4.6.1.1.2 Description

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

hydraulic control units (HCUs), interconnecting piping, instrumentation, and electrical controls).

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 (Figures 4.6-1 through 4.6-4). The individual drives are mounted on the bottom head of the RPV. The drives do not interfere with refueling and are operative even when the head is removed from the reactor vessel.

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

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

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

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

The individual rod indicators, grouped in one control panel display, correspond to relative rod locations in the core. A separate, smaller display is located just below the large display on the vertical part of the benchboard. This display presents the positions of the control rod selected for movement and the other rods in the affected rod group.

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

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.

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

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

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

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

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

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

A guide cap is fixed in the upper end of the drive assembly. This member provides the unlocking cam surface for the collet fingers and serves as the upper bushing for the index tube. If reactor water is used during a scram to supplement accumulator pressure, it is drawn through a filter on the guide cap.

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.

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 outer drive piston rings. A bleed-off passage to the center of the piston tube is located between the two pairs of rings. This arrangement allows seal leakage from the reactor vessel (during a scram) 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. 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 in of travel. Duplicate switches are provided for the full-in and full-out positions. One additional switch (an overtravel switch) is located at a position below the normal full-out position. Because the limit of downtravel is normally provided by the control rod itself as it reaches the backseat position, the drive can pass this position and actuate the overtravel switches only if it is uncoupled from its control rod. A convenient means is thus provided to verify that the drive and control rod are coupled after installation of a drive or at any time during plant operation.

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

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

Both the cylinder tube and outer tube are welded to the drive flange. The upper ends of these tubes have a sliding fit to allow for differential expansion. The upper end of the index tube is threaded to receive a coupling spud. The coupling (Figure 4.6-1) accommodates a small amount of angular misalignment between the drive and the control rod. Six spring fingers allow the coupling spud to enter the mating socket on the control rod. A plug then enters the spud and prevents uncoupling.

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

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.

Materials of Construction

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

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

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

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

Seals and Bushings Graphitar 14 or Graphitar 3030 is used for seals and bushings on the drive piston and stop piston. The material is inert and has a low friction coefficient when water-lubricated. Because some loss of Graphitar strength is experienced at higher temperatures, the drive is supplied with cooling water to hold temperatures below 250°F. Graphitar is relatively soft, which is advantageous when an occasional particle of foreign matter reaches a seal. The resulting scratches in the seal reduce sealing efficiency until worn smooth, but the drive design can tolerate considerable water leakage past the seals into the reactor vessel.

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 or Graphitar 3030.

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 Section III of the ASME Boiler and Pressure Vessel Code.

<u>Control Rod Drive Hydraulic System</u> The CRD hydraulic system (Figure 4.6-5) supplies and controls the pressure and flow to and from the drives through HCUs. The water discharged from the drives during a scram flows through the HCUs to the scram discharge volume (SDV). The water discharged from a drive during a normal control rod positioning operation flows through the HCU and 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.

The CRD hydraulic system normally takes suction from the main condensate system, downstream of the condensate demineralizers (CNDs). All piping is located within the turbine building or reactor building. The secondary source of water is the CST if the main condensate system is not available. Both sources of water are protected from cold weather.

The control rod drive hydraulic system also provides water to the nuclear boiler instrumentation system reference leg backfill injection lines.

<u>Hydraulic Requirements</u> The CRD hydraulic system design is shown on Figures 4.6-5 through 4.6-7. The hydraulic requirements, identified by the function they perform, are as follows:

1. An accumulator hydraulic charging pressure of approximately 1,400 to 1,500 psig is required. Flow to

the accumulators is required only during scram reset or system startup.

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

3. The range of acceptable cooling water flow under normal operations is identified in Figure 4.6-7, Note 3D.

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

System Description The CRD hydraulic system provides the required functions with the pumps, filters, valves, instrumentation, and piping shown on Figure 4.6-5 and described as follows:

Duplicate components are included to assure continuous system operation if an inservice component requires maintenance.

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

The pump suction line contains two suction filters. The pump suction filter is a disposable element type with a 25-micron absolute rating. Normally, one filter is in service and one filter is in standby. When filter replacement is required, the standby filter is placed in service. Alternatively, if both filters require replacement, a 250-micron strainer in the filter bypass line protects the pump. The drive water filter, downstream of the pump, is a cleanable element type with a 50-micron absolute rating. A differential pressure indicator and control room alarm monitor the filter element as it collects foreign materials.

The pumps also provide water for the reactor vessel level instrumentation reference leg backfill injection lines.

These lines inject a small stream of water into four level instrument reference legs to prevent the accumulation of noncondensable gases. The gases can cause erroneous reactor water level measurements during RPV depressurization.

The supply is taken from the CRD water pumps discharge header located on el 215'. The supply lines are tapped off the side of the horizontal CRD header to mitigate the possibility of air that may be trapped in the CRD system from being introduced into the nuclear boiler instrumentation sensing lines. The supply header splits into divisionally separated and seismically supported supply lines which supply injection water to the backfill injection flow control metering stations. The metering stations consisting of in-line filters, parallel flow meters, metering valves, dual in-series check valves, isolation valves, vent valves and drain valves are physically separated divisionally and provide injection flow to their respective reference leg. The dual in-series check valves prevent draindown of the reference legs with loss of CRD pressure. The check valves are classified as safety related with the first check valve forming the boundary between the nonsafety-related CRD system and the safety-related nuclear boiler instrumentation system. The metering valves used to adjust the required flow rate are located downstream of the check valves and provide dampening of CRD pressure perturbation.

<ins>Accumulator Charging Pressure</ins> Accumulator charging pressure is established by precharging the nitrogen accumulator to a precisely-controlled pressure at known temperature. During scram, the scram inlet (and outlet) valves open and permit the stored energy in the accumulators to discharge into the drives. The resulting pressure decrease in the charging water header allows the CRD supply pump flow rate to increase into the CRDs via the charging water header. The flow element upstream of the accumulator charging header senses high flow and provides a signal to the manual auto-flow control station which in turn closes the system FCV. This action maintains increased flow through the charging water header, while avoiding prolonged pump operation at increased flow conditions. Pressure in the charging header is monitored in the main control room with a pressure indicator and high pressure alarm. During normal operation the FCV maintains a constant system flow rate. This flow is used for drive flow and drive cooling.

<ins>Drive Water Pressure</ins> Drive water pressure required in the drive header is maintained by the drive/cooling pressure control valve, which is manually adjusted from the control room. A flow rate of approximately 6 gpm (the sum of the flow rate required to insert and withdraw a control rod) normally passes from the drive water pressure stage through two solenoid-operated stabilizing valves (arranged in parallel) into the cooling water header. The flow through one stabilizing valve equals the drive insert flow; that of one 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, at the same time opening the drive directional control and exhaust solenoid valves. Thus, flow through the drive pressure control valve is always constant.

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

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

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

Scram Discharge Volume The SDV consists of header piping that connects to each HCU and drains into an instrument volume. The header piping is sized to receive and contain all the water discharged by the drives during a scram, independent of the instrument volume. There are two sets of headers, each with its own directly connected scram discharge instrument volume (SDIV) attached to the low point of the header piping. The large-diameter pipe of the instrument volume serves as a vertical extension of the SDV; however, no credit is taken for this pipe in determining SDV capacity.

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

During a scram, the SDV partly fills with water discharged from above the drive pistons. After scram is completed, the CRD seal leakage from the reactor continues to flow into the SDV until the discharge volume pressure equals the reactor vessel pressure. A check valve in each HCU prevents reverse flow from the scram discharge header volume to the drive. When the initial scram signal is cleared from the 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 discharge volume vent and drain valves to be tested without disturbing the RPS. Closing the SDV valves allows the outlet scram valve seats to be leak tested by timing the accumulation of leakage inside the SDV.

Six liquid-level switches and two level transmitters are connected to each SDIV to monitor the volume for abnormal water level. They are set at three different levels. They provide redundant and diverse inputs to the RPS scram function, inputs to the main control room annunciation function, and inputs control rod withdrawal block function (Figure 4.6-5). At the lowest level, a level switch actuates to indicate that the volume is not completely empty during postscram draining or to indicate that the volume starts to fill through leakage accumulation at other times during reactor operation. At the second level, one level switch produces a rod withdrawal block to prevent further withdrawal of any control rod when leakage accumulates to half the capacity of the instrument volume. The remaining four switches are interconnected with the trip channels of the RPS and will initiate a reactor scram on high water level in the SDIV. Two of these switches are actuated by level transmitters to provide diversity of signals to the RPS.

The Unit 2 design implements the modifications to the scram discharge system to comply with the criteria enumerated in the Generic Safety Evaluation Report - BWR Scram Discharge System.[3]

The criteria given in the referenced safety evaluation report (SER) are organized according to (1) functional, (2) safety, (3) operational, and (4) design and surveillance criterion. A summary of each criterion is given below along with a discussion of how the scram discharge system complies.

1. Functional Criteria

    a. Functional Criterion 1

        The SDV shall be of sufficient capacity to receive and contain water exhausted by a full reactor scram without adversely affecting CRD scram performance.

        Nine Mile Point Unit 2 Compliance

        Provisions are made to accept a minimum SDV of 3.34 gal per drive, as specified through the system design specifications. This minimum SDV is based on conservative assumptions as to the performance of the scram system. In the event of a coolant leak into the SDV, an automatic scram will occur before the required SDV capacity is threatened.

2. Safety Criteria

   a. Safety Criterion 1

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

      Nine Mile Point Unit 2 Compliance

      No single active failure in the scram system design will prevent a reactor scram. The scram discharge system design meets the NRC acceptance criterion for Safety Criterion 1. Partial or full loss of service functions will not adversely affect the scram system function or will result in a full reactor scram. There is no reduction in the pipe size of the header piping going from the HCUs to and including the SDIV. This hydraulic coupling permits operability of the scram level instrumentation prior to loss of system function. The scram level instrumentation is redundant and diverse to assure no single failure or common mode failure prevents a reactor scram.

   b. Safety Criterion 2

      No single active failure results in uncontrolled loss of reactor coolant.

      Nine Mile Point Unit 2 Compliance

      Redundant SDV vent and drain valves are provided as part of the SDV modifications done for the Unit 2 plant. The redundant SDV valve configuration assures that no single active failure can result in an uncontrolled loss of reactor coolant. An additional solenoid-operated pilot valve controls the redundant vent and drain valve. The vent and drain system is therefore sufficiently redundant to avoid a failure to isolate the SDV due to solenoid failure. The vent and drain valve's opening and closing sequences are controlled to minimize excessive hydrodynamic forces.

   c. Safety Criterion 3

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

Nine Mile Point Unit 2 Compliance

Diverse and redundant level-sensing instrumentation on the SDIV is provided for the automatic scram function. SDIV water level is measured by utilization of both float-sensing and pressure-sensing devices. Instrument taps have been relocated from the vent and drain piping to the SDIV to protect the level-sensing instrumentation from the flow dynamics in the scram discharge system. Each SDIV has a redundant instrument loop. A one-out-of-two twice logic is employed for the automatic scram function. This instrumentation arrangement ensures the automatic scram function on high SDIV water level in the event of a single active or passive failure. These SDV modifications will be implemented at Unit 2.

d. Safety Criterion 4

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

Nine Mile Point Unit 2 Compliance

See response to Safety Criterion 3.

e. Safety Criterion 5

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

Nine Mile Point Unit 2 Compliance

The SDIV scram level instrumentation arrangement and trip logic allows instrument adjustment or surveillance without bypassing the scram function or directly causing a scram. Each level instrument can be individually isolated without bypassing the scram function. A one-out-of-two twice trip logic is employed. Unit 2 plant Technical Specifications will ensure that the scram function is not bypassed during repair, replacement, adjustment, or surveillance of any system component.

3. Operational Criteria

a. Operational Criterion 1

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

Nine Mile Point Unit 2 Compliance

See response to Safety Criterion 5.

b. Operational Criterion 2

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

Nine Mile Point Unit 2 Compliance

Supervisory instrumentation and alarms such as accumulator trouble, scram valve air supply low pressure, and scram discharge instrument volume drain alarms, are adequate and permit surveillance of the scram system's readiness.

c. Operational Criterion 3

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

Nine Mile Point Unit 2 Compliance

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

d. Operational Criterion 4

Vent paths shall be provided to ensure adequate drainage in preparation for scram reset.

Nine Mile Point Unit 2 Compliance

A vent line is provided as part of the scram discharge system to ensure proper drainage in preparation for scram reset. The Unit 2 position is to provide a dedicated vent line with a nonsubmerged discharge during normal operating conditions. Furthermore, the vent line vacuum breaker provides additional vent capability under normal operating conditions and ensures adequate vent capability should the SDV vent line become submerged (i.e., following a LOOP). The vacuum breaker is required to open at a differential pressure no greater than 5 in of water.

e. Operational Criterion 5

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

Nine Mile Point Unit 2 Compliance

The SDV vent line and drain lines are dedicated lines. The SDV vent discharges into a reactor building equipment drain tank (2DER-TK2B). A vacuum breaker on the SDV vent line precludes water from siphoning back into the SDIV.

4. Design Criteria

a. Design Criterion 1

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

Nine Mile Point Unit 2 Compliance

As discussed in response to Functional Criterion 1, a minimum SDV of 3.34 gal per drive is provided. Furthermore, there is no reduction in the pipe size of the header piping going from the HCUs to and including the SDIV. The SDIV is directly connected to the SDV at the low point of the scram discharge header piping. These requirements satisfy the NRC's acceptance criteria for Design Criterion 1.

b. Design Criterion 2

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

Nine Mile Point Unit 2 Compliance

See response to Functional Criterion 1 and Design Criterion 1.

c. Design Criterion 3

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

Nine Mile Point Unit 2 Compliance

See response to Safety Criterion 3.

d. Design Criterion 4

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

Nine Mile Point Unit 2 Compliance

See response to Safety Criterion 3.

e. Design Criterion 5

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

Nine Mile Point Unit 2 Compliance

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

f. Design Criterion 6

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

Nine Mile Point Unit 2 Compliance

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

g. Design Criterion 7

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

Nine Mile Point Unit 2 Compliance

See response to Design Criterion 1.

h. Design Criterion 8

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

Nine Mile Point Unit 2 Compliance

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

i. Design Criterion 9

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

Nine Mile Point Unit 2 Compliance

The present alarm and rod block instrumentation meets this criterion.

j. Design Criterion 10

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

Nine Mile Point Unit 2 Compliance

See response to Safety Criterion 2 and Operational Criterion 3.

5. Surveillance Criteria

a. Surveillance Criterion 1

Vent and drain valves shall be periodically tested.

b. Surveillance Criterion 2

Verifying level detection instrumentation shall be periodically tested in place.

c. Surveillance Criterion 3

The operability of the entire system as an integrated whole shall be demonstrated in accordance with plant Technical Specifications.

Nine Mile Point Unit 2 Compliance

The Unit 2 Technical Specifications and periodic testing requirements will address these criteria.

Safety concerns associated with postulated pipe breaks in the CRD system, as identified in NUREG-0803, have been considered in the Unit 2 design. Specific responses to the recommendations noted in Section 5 of NUREG-0803 on piping integrity are addressed below. With regard to pipe break mitigation and effects, a break in the SDV system and the consequent environmental effects are not postulated to occur based on the upgraded Unit 2 design. All portions of the SDV are designed, fabricated, inspected, installed, and tested in accordance with the requirements of the ASME Code Section III, Subsection NC. In addition, the SDV has been designed for the most severe conditions expected during a scram, including the effects of high temperature (450°F) and water hammer. Redundant SDV vent and drain valves have been installed so that no single active failure of these valves following a scram could result in an uncontrolled release of reactor coolant. In accordance with NUREG-0803, the SDV system is classified as a moderate energy system. Based on the guidelines of SRP 3.6.2, a postulated crack is considered under normal plant conditions of startup, operation at power and hot standby, and shutdown. For these conditions, the SDV system is completely depressurized with open vents and drains and, therefore, no severe environmental effects would result should a break occur.

Responses to NUREG-0803 recommendations on piping integrity are as follows:

1. <u>Periodic In-service Inspection and Surveillance for the SDV System</u>

   A program of in-service inspection (ISI) and surveillance for the CRD lines and SDV piping in accordance with ASME XI requirements will be incorporated at Unit 2.

2. <u>Threaded Joint Integrity</u>

   Threaded connections in the SDV system are not used.

3. <u>Seismic Design Verification</u>

The insert/withdrawal lines and SDV piping are designed in accordance with the requirements for ASME III Class 2 piping. Seismic Category I loads and hydrodynamic forces (i.e., water hammer) have been included in the design.

4. <u>HCU-SDV Equipment Procedures Review</u>

   Plant operation, maintenance, and surveillance procedures will provide sufficient guidance to ensure the system integrity.

5. <u>As-Built Inspection of SDV Piping and Supports</u>

   As-built inspection of SDV piping and its supports will be conducted prior to startup.

As stated previously, a break in the SDV is not postulated to occur based on the upgraded Unit 2 design. Additional means are also available to address other safety concerns of NUREG-0803 associated with an unlikely SDV system failure. First, the layout of essential emergency core cooling system (ECCS) equipment (i.e., pumps, motors) located in flood-tight areas in the reactor building would prevent the equipment from being sprayed with cascading water. Second, Unit 2 has implemented emergency operating procedures (EOPs), based upon the BWR Owners' Group (BWROG) emergency procedure guideline and severe accident guideline (EPG/SAG), that address potential leaks in secondary containment. In addition, reactor water iodine concentrations will be limited as described in the Technical Specification to ensure acceptable radiation levels in the reactor building.

<u>Hydraulic Control Units</u> 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 7.7.1.1.

The basic components in each HCU are manual, pneumatic, and electrical valves; an accumulator; related piping; electrical connections; filters; and instrumentation (Figures 4.6-5 and 4.6-8). The components and their functions are described as follows:

<u>Insert Drive Valve</u> The insert drive valve is solenoid operated and opens on an insert signal. The valve supplies drive water to the bottom side of the main drive piston.

<u>Insert Exhaust Valve</u> 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.

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.

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

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

Scram Pilot Valve Assembly The scram pilot valve assembly is operated from the RPS. The scram pilot valve assembly, with two solenoids, controls both the scram inlet valve and the scram exhaust valve. The scram pilot valve assembly is solenoid operated and is normally energized. On loss of electrical signal to the solenoids, the inlet port closes and the exhaust port opens. The pilot valve assembly (Figures 4.6-5 and 4.6-8) is designed so that both solenoids must be deenergized before air pressure can be discharged from the scram valve operators. This prevents inadvertent scram of a single drive in the event of a failure of one of the pilot valve solenoids.

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 switch on this valve operates a light in the main control room as soon as the valve starts to open.

Scram Exhaust Valve The scram exhaust valve opens slightly before the scram inlet valve, exhausting water from above the drive piston. The exhaust valve opens faster than the inlet valve because of the higher air pressure spring setting in the valve operator. Scram exhaust valves are equipped with position indicators.

Scram Accumulator The scram accumulator stores sufficient energy to fully insert a control rod at lower reactor vessel pressures. At higher vessel pressures, the accumulator pressure is assisted or supplanted by reactor vessel pressure. The accumulator is a hydraulic cylinder with a free-floating piston. The piston separates the water on top from the nitrogen below. A

check valve in the accumulator charging line prevents loss of water pressure in the event supply pressure is lost.

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

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

<u>Control Rod Drive System Operation</u>

The CRD system performs rod insertion, rod withdrawal, and scram. These operational functions are described in the following sections.

<u>Rod Insertion</u> Rod insertion is initiated by a signal from the Operator, which causes both insert solenoid valves to open. The insert drive valve applies reactor pressure plus approximately 90 psi to the bottom of the drive piston. The insert exhaust valve allows water from above the drive piston to discharge to the exhaust header.

As 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 shim speed (nonscram operation) of 3 in/sec. During normal insertion, the pressure on the downstream side of the speed control valve is 90 to 100 psi above reactor vessel pressure. However, if the drive slows for any reason, the flow through and pressure drop across the insert speed control valve decreases. The full differential pressure (260 psi) is then available to cause continued insertion. With 260-psi differential pressure acting on the drive piston, the piston exerts an upward force of 1,040 lb.

<u>Rod Withdrawal</u> Rod withdrawal is, by design, more involved than insertion. The collet finger (latch) must be raised to reach the unlocked position (Figure 4.6-3). The notches in the index tube and the collet fingers are shaped so that the downward force on the index tube holds the collet fingers in place. The index tube must be lifted before the collet fingers can be released. This is done by opening the drive insert valves (in the manner described in the preceding section) for approximately 1 sec. The withdraw valves are then opened, applying driving pressure above the drive piston and opening the area below the piston to the exhaust header. Pressure is simultaneously applied to the collet piston. As the piston 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 withdrawal direction. Water displaced by the drive piston flows out through the withdrawal speed control valve, which is set to give the control rod a shim speed of 3 in/sec. The entire valving sequence is automatically controlled and is initiated by a single operation of the rod withdrawal switch.

For Cycle 7, a cycle-specific analysis has been completed for rod withdrawal rates up to 6.0 in per second. For all other cycles, a cycle-generic analysis has been completed for rod withdrawal rates up to 5.0 in per second.

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 SDV.

The large differential pressure (approximately 1,500 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 1,450 to 1,510 psig on the water side and 1,050 to 1,100 psig on the nitrogen side. As the inlet scram valve opens, the full water-side pressure is available at the CRD acting on a 4.1-sq in area. As CRD motion begins, this pressure drops to the gas-side pressure less line losses between the accumulator and the CRD. At low vessel pressures the accumulator completely discharges with a 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.

Limits on scram insertion times are contained in Technical Specification 3.1.4.

Alternate Rod Insertion The alternate rod insertion (ARI) feature is designed to increase the reliability of the CRD system (Section 4.6.1). ARI provides for insertion of reactor control rods by depressurizing the scram discharge air header through valves which are redundant and diverse from the RPS scram function.

A signal to insert control rods results in energizing the eight ARI solenoid valves. Two solenoid valves in series with the backup scram valves also have parallel functioning check valves to assure venting of air from the air supply line in the event an ARI valve fails. Four solenoid valves provide for venting of the A and B HCU scram headers to the atmosphere to depressurize the headers and scram all rods. Two solenoid valves vent the air header to the SDV drain and vent lines, closing the vent and drain valves and isolating the SDV.

Instrumentation

The instrumentation for both the control rods and CRDs is defined by that given for the manual control system. The objective of the reactor manual control system (RMCS) 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 instrumentation and controls for the RMCS are described in Section 7.7.1.1.

4.6.1.2 Control Rod Drive Housing Supports

4.6.1.2.1 Safety Objective

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

4.6.1.2.2 Safety Design Bases

The CRD housing supports meet the following safety design bases:

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

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

4.6.1.2.3 Description

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

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

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

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

When the support bars and grids are installed, a gap of 1 in (+0.50/-0.25 in, at a temperature of 110°F or less) 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 1/2 in to 1 1/4 in.

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

The American Institute of Steel Construction (AISC) Manual of Steel Construction, Specification for the Design, Fabrication and Erection of Structural Steel for Buildings, was used as a guide in designing the CRD housing support system. The CRD housing support system is a mechanical system; therefore, RG 1.94 does not apply. However, to provide a structure that absorbs as much energy as practical without yielding, the allowable tension and bending stresses used were 90 percent of yield and the shear stress used was 60 percent of yield. These design stresses are 1.5 times the AISC allowable stresses (60 and 40 percent of yield, respectively).

For purposes of mechanical design, the postulated failure resulting in the highest forces is an instantaneous circumferential separation of the CRD housing from the reactor vessel, with the reactor at an operating pressure of 1,086 psig (at the bottom of the vessel) acting on the area of the separated housing. The weight of the separated housing, CRD, and blade, plus the pressure of 1,086 psig acting on the area of the separated housing, gives a force of approximately 32,000 lb. This force is used to calculate the impact force, conservatively assuming that the housing travels through a 1 1/2-in gap before it contacts the supports. The impact force (124,000 lb for the maximum gap of 1 1/2 in) is then treated as a static load in design. The CRD housing supports are designed as Category I equipment in accordance with Section 3.2. Loading conditions and examples of stress analysis results and limits are shown in Table 3.9B-2. Safety evaluation is discussed in Section 4.6.2.3.3.

4.6.2 Evaluations of the CRDs

4.6.2.1 Failure Modes and Effects Analysis

The evaluation of CRD system failure is discussed under Nuclear Safety Operational Analysis (NSOA) in Appendix 15A (Section 15A.6.6.3) and on Figures 15A-51 through 15A-53.

4.6.2.2 Protection from Common Mode Failures

See Section 4.6.2.1.

4.6.2.3 Safety Evaluation

Safety evaluation of the control rods, CRDs, and CRD housing supports is described in the following sections. Further description of control rods is contained in Section 4.2.

4.6.2.3.1 Control Rods

<u>Materials Adequacy Throughout Design Lifetime</u>

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

<u>Dimensional and Tolerance Analysis</u>

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

Thermal Analysis of the Tendency to Warp

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

Forces for Expulsion

An analysis has been performed which 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. In summary, if the collet were to remain open, which is unlikely, calculations indicate that the steady state control rod withdrawal velocity would be 2 ft/sec for a pressure-under line break, the limiting case for rod withdrawal.

Functional Failure of Critical Components

The consequences of a functional failure of critical components have been evaluated and the results are covered in Section 4.6.2.3.2.

Precluding Excessive Rates of Reactivity Addition

In order to preclude excessive rates of reactivity addition, an analysis has been performed both on the velocity limiter device and the effect of probable control rod failures (Section 4.6.2.3.2).

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.

Procedural controls are established during the core design process and during the process of determining the final core configuration, that mitigate the consequences of channel bowing.

Fuel channel deflection measurements may be used to justify the use of the fuel channels.

In the future, analytical channel lifetime prediction methods, benchmarked and backed up by periodic deflection measurements of a sample of the highest duty fuel channels, may be used to ensure clearance between control rod blades and fuel channels without additional testing.

Mechanical Damage

In addition to the analysis performed on the CRD (Section 4.6.2.3.2) and the control rod, analyses were performed on the control rod guide tube (Sections 3.9B.1.4.2 and 3.9B.5.3.3).

Evaluation of Control Rod Velocity Limiter

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

4.6.2.3.2 Control Rod Drives

Evaluation of Scram Time

The rod scram function of the CRD system provides the negative reactivity insertion required by safety design bases in Section 4.6.1.1.1. The scram time shown in the description is adequate as shown by the transient analyses of Chapter 15 and Appendix A.

Analysis of Malfunction Relating to Rod Withdrawal

There are no known single malfunctions that cause the unplanned withdrawal of even a single control rod. However, if multiple malfunctions are postulated, studies show that an unplanned rod withdrawal can occur at withdrawal speeds that vary with the combination of malfunctions postulated.

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 fastened by welding. The drive is raised into the drive housing and bolted to a flange at the bottom of the housing.

The housing material is seamless, Type 304 stainless steel pipe with a minimum tensile strength of 75,000 psi. The basic failure considered here is a complete circumferential crack through the housing wall at an elevation just below the J-weld. Static loads on the housing wall include the weight of the drive and the control rod, the weight of the housing below the J-weld, and the reactor pressure acting on the 6-in diameter cross-sectional area of the housing and the drive. Dynamic loading results from the reaction force during drive operation.

If the housing were to fail as described, the following sequence of events is foreseen. The housing would separate from the vessel. The CRD and housing would be blown downward against the support structure by reactor pressure acting on the cross-sectional area of the housing and the drive. The downward motion of the drive and associated parts would be determined by

the gap between the bottom of the drive and the support structure, and by the deflection of the support structure under load. In the current design, maximum deflection is approximately 3 in. If the collet were to remain latched, no further control rod ejection would occur[2], the housing would not drop far enough to clear the vessel penetration, and reactor water would leak at a rate of approximately 180 gpm through the 0.03-in diametral clearance between the housing and the vessel penetration.

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

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

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

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

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

position indicator probe. A second indication would be high cooling water flow.

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

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

<u>Simultaneous Breakage of the Pressure-Over (Withdrawal) and Pressure-Under (Insert) Lines</u> For the simultaneous breakage of the pressure-over (withdrawal) and pressure-under (insert) lines, pressures above and below the drive piston would drop to zero, and the ball check valve would close the broken pressure-under line. Reactor water would flow from the annulus outside the drive, through the vessel ports, and to the space below the drive piston. As in the case of pressure-over line breakage, the drive would then insert (at reactor pressure approximately 600 psig or greater) at a speed dependent on reactor pressure. Full insertion would occur regardless of the operational mode at the time of failure. Reactor water would leak past the drive seals and cut the broken pressure-over line to the containment, as previously described. Drive temperature would increase. Indication in the main control room would include the drift alarm, the fully-inserted drive, the high drive temperature annunciated in the main control room, and operation of the drywell floor drain pump.

<u>All Drive Flange Bolts Fail in Tension</u> Each CRD is bolted to a flange at the bottom of a drive housing. The flange is welded to the drive housing. Bolts are made of AISI-4140 steel with a minimum tensile strength of 125,000 psi. Each bolt has an

allowable load capacity of 15,200 lb. Capacity of the eight bolts is 121,600 lb. As a result of the reactor design pressure of 1,250 psig, the major load on all eight bolts is 30,400 lb.

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

If the collet were latched, control rod ejection would be limited to the distance the drive can drop before coming to rest on the support structure. There would be no tendency for the collet to unlatch, because pressure below the collet piston would drop to zero. Pressure forces, in fact, exert 1,435 lb to hold the collet in the latched position.

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

<u>Weld Joining Flange to Housing Fails in Tension</u> The failure considered is a crack in or near the weld that joins the flange to the housing. This crack extends through the wall and completely around the housing. The flange material is forged, Type 304 stainless steel, with a minimum tensile strength of 75,000 psi. The housing material is seamless, Type 304 stainless steel pipe, with a minimum tensile strength of 75,000 psi. The conventional, full penetration weld of Type 308 stainless steel has a minimum tensile strength approximately the same as that for the parent metal. The design pressure and temperature are 1,250 psig and 575°F. Reactor pressure acting on the cross-sectional area of the drive; the weight of the control rod, drive, and flange; and the dynamic reaction force during drive operation result in a maximum tensile stress at the weld of approximately 6,000 psi.

If the basic flange-to-housing joint failure occurred, the flange and the attached drive would be blown downward against the support structure. The support structure loading would be

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

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

Housing Wall Ruptures This failure is a vertical split in the drive housing wall just below the bottom head of the reactor vessel. The flow area of the hole is considered equivalent to the annular area between the drive and the thermal sleeve. Thus, flow through this annular area, rather than flow through the hole in the housing, would govern leakage flow. The CRD housing is made of 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 (1,250 psig) acting on the inside of the housing.

If such a rupture were to occur, reactor water would flash to steam and leak through the hole in the housing to the drywell at approximately 1,030 gpm. Choke-flow conditions would exist, as described previously for the flange-bolt failure. However, leakage flow would be greater because flow resistance would be less; i.e., the leaking water and steam would not have to flow

down the length of the housing to reach the drywell. A critical pressure of 350 psi causes the water to flash to steam.

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

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

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

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

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

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

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

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

If the plug were to blow out while the drive was latched, there would be no control rod motion. No pressure differential would exist across the collet piston to unlatch the collet. As in the previous failure, reactor water would flow past the velocity limiter, down the annulus between the drive and thermal sleeve, through the vessel ports and drilled passage, through the ball check valve cage, and out the open plug hole to the drywell. The leakage calculations indicate the flow rate is less than 350 gpm. This calculation assumes liquid flow, but flashing of the hot reactor water to steam would reduce this rate to a lower value. Drive temperature would rapidly increase and initiate an alarm in the 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/sec. There would be a large retarding force exerted by the velocity limiter due to a 35-psi pressure differential across the velocity limiter piston.

Drive/Cooling Water Pressure Control Valve Closure (Reactor Pressure, 0 psig) The pressure to move a drive is generated by the pressure drop of practically the full system flow through the drive/cooling water pressure control valve. This valve is either a motor-operated valve (MOV) or a standby manual valve; either one is adjusted to a fixed opening. The normal pressure drop across this valve develops a pressure 260 psi in excess of reactor pressure.

If the flow through the drive/cooling water pressure control valve were to be stopped, as by a valve closure or flow blockage, the drive pressure would increase to the shutoff pressure of the supply pump. The occurrence of this condition during withdrawal of a drive at zero vessel pressure will result in a drive pressure increase from 260 psig to no more than 1,750 psig. Calculations indicate that the drive would accelerate from 3 in/sec to approximately 6.5 in/sec. A pressure differential of 1,670 psi across the collet piston would hold the collet unlatched. Flow would be upward, past the velocity limiter

piston, but retarding force would be negligible. Rod movement would stop as soon as the driving signal was removed.

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

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

Collet Fingers Fail to Latch This 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 would continue to move at a reduced speed.

Withdrawal Speed Control Valve Failure Normal withdrawal speed is determined by differential pressures in the drive and is set for a nominal value of 3 in/sec. For Cycle 7, a cycle-specific analysis has been completed for rod withdrawal rates up to 6.0 in per second. For all other cycles, a cycle-generic analysis has been completed for rod withdrawal rates up to 5.0 in per second. Withdrawal speed is maintained by the pressure regulating system and is independent of reactor vessel pressure. Tests have shown that accidental opening of the speed control valve to the full-open position produces a velocity of approximately 6 in/sec.

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

Scram Reliability

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

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

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

3. The 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 will not restrain or prevent control rod insertion during scram.

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

Control Rod Support and Operation

As described in the preceding sections, each control rod is independently supported and controlled as required by safety design bases.

4.6.2.3.3 Control Rod Drive Housing Supports

Downward travel of the CRD housing and its control rod following the postulated housing failure equals the sum of these distances: 1) the compression of the disc springs under dynamic loading, and 2) the initial gap between the grid and the bottom contact surface of the CRD flange. If the reactor were cold and pressurized, the downward motion of the control rod would be limited to the spring compression (approximately 2 in) plus a gap of 1 in (+0.50/-0.25 in). If the reactor were hot and pressurized, the gap would be approximately 1/2 in to 1 1/4 in and the spring compression would be slightly less than in the cold condition. In either case, the control rod movement following a housing failure is substantially limited below one drive "notch" movement (6 in). Sudden withdrawal of any control rod through a distance of one drive notch at any position in the

core does not produce a transient sufficient to damage any radioactive material barrier.

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

At plant operating temperature, a gap of approximately 1/2 in to 1 1/4 in exists between the CRD housing and the supports. At lower temperatures the gap is greater. Because the supports do not contact any of the CRD housing except during the postulated accident condition, vertical contact stresses are prevented. Inspection and testing of CRD housing supports is discussed in Section 4.6.3.2.

4.6.3 Testing and Verification of the CRDs

4.6.3.1 Control Rod Drives

4.6.3.1.1 Testing and Inspection

Development Tests

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

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

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

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

4. Usable seal lifetimes in excess of 1,000 scram cycles can be expected.

Factory Quality Control Tests

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

1. CRD 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 scram tests at various reactor pressures to verify correct scram performance.

2. 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 withdrawal valves is demonstrated.

<u>Operational Tests</u>

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

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

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

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

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: 1) preinstallation, 2) after installation prior to startup, and 3) during startup testing.

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

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

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

Surveillance Tests

The surveillance requirements for the CRD system are described in the Technical Specifications.

Functional Tests

The functional testing program of the CRDs consists of the 5-yr maintenance life and the 1.5 times design life test programs described in Section 3.9B.4.4.

There are a number of failures that can be postulated for the CRD. 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 drive down inlet FCV (V122) failure.
4. HCU drive down outlet FCV (V120) failure.
5. CRD scram performance with V120 malfunction.
6. HCU drive up outlet control valve (V121) failure.
7. HCU drive up inlet control valve (V123) failure.
8. Cooling water check valve (V138) leakage.
9. CRD flange 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.

4.6.3.2 Control Rod Drive Housing Supports

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

4.6.4 Information for Combined Performance of Reactivity Control Systems

4.6.4.1 Vulnerability to Common Mode Failures

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

4.6.4.2 Accidents Taking Credit for Multiple Reactivity Control 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

1. Rawlings, S. C., et al. BWR4 and BWR5 Fuel Design, October 1976 NEDE-20944-P and NEDO-20944, nonproprietary.

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

3. Check, P. S. BWR Scram Discharge System Safety Evaluation, Nuclear Regulatory Commission, December 1980.

APPENDIX 4A

INTERGRANULAR STRESS CORROSION CRACKING

APPENDIX 4A

INTERGRANULAR STRESS CORROSION CRACKING

Welded austenitic stainless steel reactor internals are relied upon to permit adequate core cooling for any mode of normal operation or under credible postulated accident conditions. This appendix demonstrates that components which may have carbon content at the maximum limit are not subject to intergranular stress corrosion cracking (IGSCC).

Carbon content of austenitic stainless steels is used as an indication for grain boundary chromium depletion. Chromium depletion, also known as sensitization, can lead to IGSCC in BWR components. IGSCC due to sensitization was the primary cause of cracking in BWR piping components. This appendix addresses the potential for this phenomenon manifesting in BWR internal components. For sensitization-induced IGSCC to occur in reactor internals fabricated from wrought austenitic stainless steel, the following three conditions are all necessary:

1. Sensitized condition.

2. Corrosive environment.

3. Significant tensile stress.

Since carbon content in the wrought austenitic stainless steel is a major factor leading to material sensitization from welding, the level of sensitization can be minimized by keeping the carbon content of the stainless steel below 0.035 percent.

Oxygenated BWR water is considered a corrosive environment when the water temperature exceeds 200°F for sustained periods of time (1,000 hr during the life of the plant). Parts subjected to oxygenated BWR water below this temperature are not susceptible to IGSCC.

Significant tensile stress on a particular component or part is dependent on its residual stress from fabrication and the applied load. The criteria for designating a part's tensile stress as not significant are:

1. The calculated stress levels are judged to be low.

2. No previous IGSCC problems based on long-term GE field experience.

Austenitic stainless steel components associated with the jet pump, low-pressure coolant injection (LPCI), core spray, and feedwater component of the reactor internals permit core cooling for normal and accident conditions.

For jet pump parts, a Type 304 grade (maximum carbon content 0.055 percent) stainless steel was used. Welded parts (riser elbow, riser pipe, and riser support) are subjected to low applied tensile stress, and reactor experience on those parts indicates no IGSCC.

For LPCI parts, a Type 304 grade (maximum carbon content 0.055 percent) stainless steel was used. These welded parts (ring and flange) are subjected to low applied tensile stresses during normal plant operation, and IGSCC is not expected.

For core coolant parts, Type 316L grade and Type 304L grade (maximum carbon content 0.035 percent) stainless steels meet the maximum carbon limits for preventing sensitization by welding. Therefore, IGSCC due to sensitization is not expected.

For feedwater parts, a Type 316L grade (maximum carbon content 0.035 percent) stainless steel meets the maximum carbon limits for preventing sensitization of the material by welding. Therefore, IGSCC is not expected.

In general, IGSCC due to sensitization is not a problem for Unit 2 internal components since applied stresses are low and, in addition, for many parts, low carbon stainless steels that are not expected to experience sensitization were utilized. Continued operation with high-purity coolant will further reduce the potential for IGSCC.

Industry-wide efforts are underway to address the potential for IGSCC in all vessel internal and attachment components.* Ongoing and proposed studies will provide additional information regarding the relative IGSCC susceptibility and expected lifetime of internals and attachment welds over the next several years. The recent studies have shown benefits of operating with high-purity water under HWC conditions.

HWC at Unit 2 utilizes hydrogen injection and noble metal chemical addition (NMCA or NobleChem™) to reduce the potential for IGSCC of the stainless steel reactor vessel components and recirculation piping. Hydrogen injection is provided through the condensate system; and NobleChem™ is periodically added through the recirculation pump differential pressure transmitter lines (off-line), or through the feedwater system (on-line). The noble

* BWROG/NRC meeting held in Rockville, MD, on March 22, 1989, to review the BWR industry program for RPV and internals inspection.

metals adhere to the internal surface oxide layer and have no long-term impact on coolant chemistry.

A crack arrest verification (CAV) system is installed in Unit 2 to provide real-time indication of the performance of plant materials in the BWR environment. The CAV system is discussed in Section 5.2.3.2.5.

A durability monitor (DM) skid is installed in Unit 2 to provide a means of ensuring that adequate noble metals coating is available on the plant components being protected from IGSCC with low level of hydrogen injection. A stream of sample water from upstream of the regenerative heat exchanger in the RWCU system is routed through the DM skid.