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4.0 Reactor

4.1 Summary Description

The reactor assembly consists of the reactor pressure vessel (RPV), pressure-containing appurtenances (including CRD housings), incore instrumentation housings, and the head vent and spray assembly plus the reactor internal components described in Subsection 4.1.2. Figures 5.3-2a, 5.3-2b, and Table 5.3-2 show the arrangement of the reactor assembly components. A summary of the important design and performance characteristics is given in Subsection 1.3.1. Loading conditions for reactor assembly components are specified in Subsection 3.9.5.2.

4.1.1 Reactor Pressure Vessel

The reactor pressure vessel includes the Reactor Internal Pump (RIP) casing and flow restrictors in each of the steam outlet nozzles, and the shroud support and pump deck which form the partition between the RIP suction and discharge. The RPV design and description are covered in Section 5.3.

4.1.2 Reactor Internal Components

As described in Subsection 3.9.5.1, the major reactor internal components include:

(1) The core (fuel, channels, control blades and instrumentation)

(2) Core support structure (including the shroud, top guide and core plate)

(3) Shroud head and steam separator assembly

(4) Steam dryer assembly

(5) Feedwater spargers

(6) Core spray

(7) Core flooding spargers

Except for the Zircaloy in the reactor core, these reactor internals are stainless steel or other corrosion-resistant alloys. The fuel assemblies (including fuel rods and channel), control blades, shroud head and steam separator assembly, and steam dryers and incore instrumentation dry tubes are removable when the reactor vessel is opened for refueling or maintenance.

4.1.2.1 Reactor Core

Important features of the reactor core are:

(1) The bottom-entry cruciform control rods, which were first introduced in the Dresden-1 reactor in April 1961, have accumulated thousands of hours of service.

(2) Fixed incore fission chambers (LPRMs) provide continuous local power range neutron flux monitoring. A guide tube in each incore assembly provides for a Traversing Incore Probe (TIP) for calibration and axial detail. Startup Range Neutron Monitors (SRNMs) are located at fixed locations between the LPRMs as shown on Figure 4.1-1. The incore location of the startup and source range instruments provides coverage of the large reactor core and provides an acceptable signal-to-noise ratio and neutron-to-gamma ratio. All incore instrument leads enter from the bottom and the instruments are in service during refueling. Incore instrumentation is presented in Subsection 7.6.1.

(3) As shown by experience obtained at Dresden-1 and other BWR plants, utilizing the incore flux monitor system, the desired power distribution can be maintained within a large core by proper control rod scheduling.

(4) The fuel channels (a) provide a fixed flow path for the boiling coolant, (b) serve as a guiding surface for the control rods, and (c) 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 and the other control rods fully inserted.

(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.

(7) The reactor core is arranged as an upright circular cylinder containing a large number of fuel cells and is located within the core shroud inside the reactor vessel.

4.1.2.1.1 Fuel Assembly Description

The fuel assembly description is provided in Section 4.2.

4.1.2.1.2 Assembly Support and Control Rod Location

A few peripheral fuel assemblies are supported by the core plate. Otherwise, individual fuel assemblies in the core rest on fuel support pieces mounted on top of the Control Rod Guide Tubes (CRGTs). Each guide tube, with its orificed fuel support, 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 CRTG and directs the reactor recirculation into the orificed fuel support and through the fuel assemblies. The top guide, mounted on top of the shroud, provides lateral support and guidance for the top of each fuel assembly. The reactivity of the core is controlled by cruciform control rods and their associated mechanical hydraulic drive system. The control rods occupy alternate spaces between fuel assemblies. Each independent drive enters the core from the bottom, and accurately positions its associated control rod during normal operation with an electric motor-driven ball screw. Scram hydraulic pressure acts on the hollow cylinder to exert several 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

Detailed information on the shroud is provided in Subsection 3.9.5.1.1.1.

4.1.2.3 Shroud Head and Steam Separators

Detailed information on the shroud head and separators is presented in Subsection 3.9.5.1.2.1.

4.1.2.4 Steam Dryer Assembly

Detailed information on the steam dryer assembly is presented in Subsection 3.9.5.1.2.3.

4.1.3 Reactivity Control Systems

4.1.3.1 Operation

The control rods perform dual functions of power distribution shaping and reactivity control. Power distribution in the core is controlled during operation of the reactor by manipulation of selected patterns of rods. The rods, which enter from the bottom of the near cylindrical reactor core, are positioned to counterbalance steam voids in the top of the core and effect significant power flattening.

These groups of control rods, 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 rods are mechanically designed to withstand the dynamic forces resulting from a scram. They are connected to bottom mounted, electro-hydraulically actuated drive mechanisms which allow either electric motor-controlled axial positioning for reactivity regulation or hydraulic 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 remain operable for tests with the reactor vessel open.

4.1.3.2 Description of Control Rods

A description of the control rods is provided in Section 4.2.

4.1.3.3 Supplementary Reactivity Control

The 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. Descriptions of the supplementary burnable poison are presented in Sections 4.2 and 4.3.

4.1.4 Analysis Techniques

4.1.4.1 Reactor Internal Components

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

(1) NASTRO4V

(2) SAP4G07

(3) HEATER

(4) USAGE01

(5) ANSYS

(6) CLAPS

(7) ASIST

(8) SEISM03

(9) SASSI

Detail descriptions of these programs are given in the following subsections.

4.1.4.1.1 NASTRO4V

NASTRO4V is a GE in-house version of the MSC/NASTRAN program (Digital VAX Version 64) which is developed and maintained by the MacNeal Schweldler Corporation in Los Angeles. As a general purpose computer program for finite element analysis, its capabilities include (1) static response to concentrated and distributed loads, to thermal expansion and to enforced deformation; (2) dynamic response to transient loads, to steady-state sinusoidal loads and to random excitation; (3) determination of real and complex eigen values for use in vibration analysis, dynamic stability analysis, and elastic stability analysis; (4) nonlinear static and dynamic analysis including material and geometric non-linearities; and (5) steady-state and transient heat conduction.

4.1.4.1.2 SAP4G07

SAP4G07 is a general-purpose finite element computer program used to perform stress, dynamic, and seismic analyses of structural, mechanical and piping components. Dynamic analyses can be done using direct integration or mode superposition. Response spectrum analysis (a mode superposition method) can include multiple support excitation. SAP4G07 is a GE in-house program based on similar programs developed by E. L. Wilson of UC Berkeley and Dr. K.J. Bathe.

4.1.4.1.3 HEATER

HEATER is a computer program used in the hydraulic design of feedwater spargers and their associated delivery header and piping. The program utilizes test data obtained by GE using full-scale mockups of feedwater spargers combined with a series of models which represent the complex mixing processes obtained in the upper plenum, downcomer, and lower plenum. Mass and energy balances throughout the Nuclear Steam Supply System (NSSS) are modeled in detail. The program is used (1) in the hydraulic design of the feedwater spargers for each BWR plant, (2) in the evaluation of design modifications, and (3) the evaluation of unusual operational conditions.

4.1.4.1.4 USAGE01

USAGE01 is a GE proprietary computer program used in performing ASME-III Section NB & NG structural fatigue usage calculations. The program follows Paragraphs NB or NG-3222.4e. For SRV cyclic loading, it automatically accounts for the follow-on cycles.

4.1.4.1.5 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 analysis.

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

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

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

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

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

4.1.4.1.6 CLAPS

CLAPS is a general purpose, two-dimensional finite element program used to perform linear and nonlinear structural mechanics analysis. The program solves plane stress, plane strain, and axisymmetric problems. It may be used to analyze for (1) instantaneous pressure, temperature and flux changes; (2) rapid transients; and (3) steady-state as well as conventional elastic and inelastic buckling analysis of structural components subjected to mechanical loading.

4.1.4.1.7 ASIST

The ASIST program is a GE code which can be used to obtain load distribution, deflections, critical frequencies and mode shapes in the “in-plane” or “normal-to-plane” modes for planar structures of any orientation that (1) are statistically indeterminate, (2) can be represented by straight or curved beams, and (3) are under basically any loading, thermal gradient, or sinusoidal excitation. Deformations and resulting load distributions are compared considering all strain energies (i.e., bending, torsion, shear and direct). ASIST also considers the effects of the deflected shape on loads and provides deflections calculated for the structure. In addition to this beam column (large deflection) capability, the buckling instability of planar structures can also be calculated for the structure.

The ASIST program has been used to determine spring constants, stresses, deflections, critical frequencies and associated mode shapes for frames, shafts, rotors, and other jet engine components. It has been used extensively as a design and analysis tool for various components of nuclear fuel assemblies.

4.1.4.1.8 SEISM03

SEISM03 is a GE proprietary computer program for non-linear dynamic analysis. It is based on the component element method developed by S. Levy and J.P. Wilkinson of GE CR&D. The method uses basic mass, spring, damper, gap, and coupling elements in a direct integration approach to solve non-linear dynamic analysis. This is the main dynamic analysis engineering computer program (ECP). Other programs used in conjunction with SEISM03 are:

(1) SEPRE

This ECP is a pre-processor for SEISM. It takes the output from CRTFI and phases the input time histories of all loads with the basic load time histories. SEPRE also converts all input loads to the format required for input to SEISM.

(2) SEPST

This ECP is the SEISM post-processor. SEPST condenses the SEISM output data into a form which is more practical to interpret. It determines and prints the initial values, the maximum and minimum values for all components, and the times of their occurrence. In addition, it generates the response time history plots of selected components.

(3) CRTFI

The CRTFI program uses, as input, the scaled or composite horizontal acceleration time histories at the mid fuel and end fuel positions to determine (1) the clamping forces to be applied to the analysis model friction elements, (2) the scram uplift forces on a bundle, (3) inertial forces of the fuel in order to obtain reaction forces on both ends of the fuel, and (4) fuel-center deflection and uplift forces due to scram.

4.1.4.1.9 SASSI

SASSI can be used to perform dynamic soil-structure interaction analysis in two or three dimensions. The seismic environment consists of an arbitrary three-dimensional (3-D) superposition of inclined body waves and surface waves. The site consists of semi-infinite elastic of viscoelastic halfspace. The structure and the soil can be modeled using a combination of 3-D solid element, 3-D beam element, four-node quadrilateral plate/shell element, 2-D four-node plane strain element, 3-D spring element, and stiffness/mass matrix element. Seismic load in the form of an acceleration time history can be applied to one of the three global directions at a control point on a soil layer interface. External forces or moments such as impact loads, wave forces, or loads from the rotating machinery can be introduced directly to nodes in the soil-structure system.

SASSI is formulated in the frequency domain using the complex response method and the finite element technique. Although it is strictly a linear program, approximate non-linear analysis can be performed by an iterative scheme called "Equivalent Linear Method."

4.1.4.2 Fuel Design Analysis

The fuel design analysis models are discussed in Section 4.2.

4.1.4.3 Reactor Systems Dynamics

The analysis techniques and computer codes used in reactor systems dynamics are discussed in Reference 4.1-1.

4.1.4.4 Nuclear Analysis

The analysis techniques are discussed in Section 4.3.

4.1.4.5 Neutron Fluence Calculations

Neutron vessel fluence calculations were carried out using a two-dimensional, discrete ordinates, Sn transport code with general anisotropic scattering.

This code is the most widely used two-dimensional, discrete ordinated code for solving various radiation transport problems. The program will solve both fixed source and multiplication problems. Rectangular (XY, or RZ) and polar (R, θ) geometry 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 weighting polynominal expansion matrices for anisotropic scattering but did not include the resonance self-shielding factors.

4.1.4.6 Thermal–Hydraulic Calculations

The thermal-hydraulic models are discussed in Section 4.4.

4.1.5 References

4.1-1 "Stability and Dynamic Performance of the GE BWR", (NEDO-21506, January 1977).

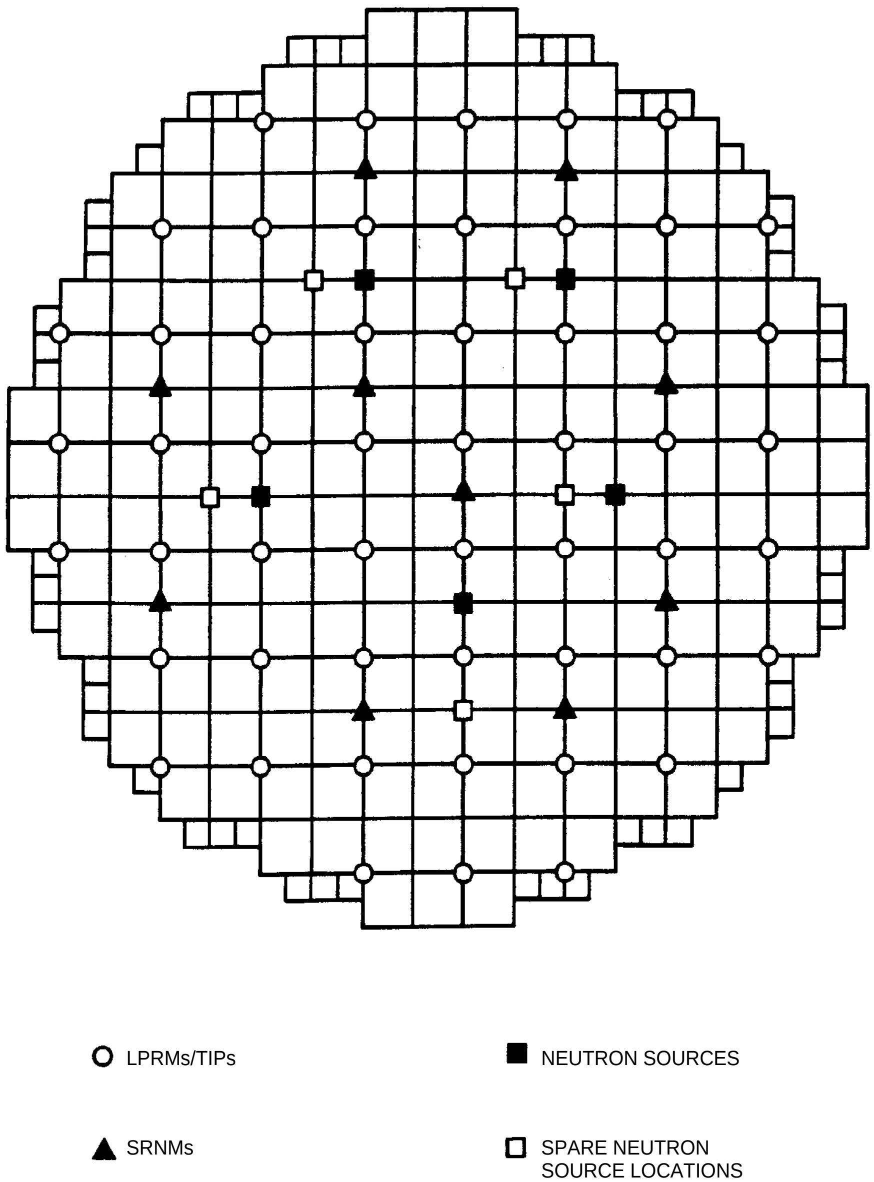


Figure 4.1-1 Core Configuration with Location of Instrumentation

4.2 Fuel System Design

The fuel system is defined as consisting of the fuel assembly and the reactivity control assembly. The fuel assembly is comprised of the fuel bundle, channel and channel fastener. The fuel bundle is comprised of fuel rods, water rods, fuel rods containing burnable neutron absorber, spacers, springs and assembly and fittings.

[*Table 8 of DCD/Introduction identifies the commitments on fuel system design criteria and first cycle design and methods, which, if changed, requires NRC Staff review and approval prior to implementation. The applicable portions of the Tier 2 sections, tables and figures, identified on Table 8 of DCD/Introduction for this restriction, are italicized on the sections, tables and figures themselves.*]*

The fuel to be loaded in an ABWR is any fuel design that is approved by the USNRC or that meets the criteria documented in Appendix 4B. Using these designs will assure that all fuel system design requirements are met.

To demonstrate ABWR system response, a reference core of GE P8x8R fuel is used. This core is shown in Section 4.3. [*Burnup limits will be specified for fuel used in the ABWR. The current maximum exposure limit for this application is 60 GW•d/tU rod average exposure. Any extension to this maximum exposure limit will be submitted to the NRC for review and approval based on the available supporting materials properties vs. exposure information and planned surveillance program.*]† The compliance of this fuel with the Appendix 4B criteria is documented in Appendix 4D. Each COL applicant may have different fuel and core designs which will be provided by the COL applicant to the USNRC for review and approval.

Regarding the reactivity control system, this Section 4.2 addresses only the reactivity control elements that extend from the coupling interface of the control rod drive mechanism (per Regulatory Guide 1.70). The functional design of the reactivity control system is detailed in Section 4.6.

4.2.1 Design Bases

4.2.1.1 Fuel Assembly

The fuel assembly (comprised of the fuel bundle, channel and channel fastener) is designed to ensure that possible fuel damage would not result in the release of

* See Section 3.5 of DCD/Introduction.

† See Section 4.2, second paragraph.

radioactive materials in excess of limits prescribed by 10CFR20, 50 and 100. Evaluations are made in conjunction with the core nuclear characteristics, the core hydraulic characteristics, the plant equipment characteristics, and the instrumentation and protection systems to assure that this requirement is met.

The thermal-mechanical design process emphasizes that:

(1) The fuel assembly provides substantial fission retention capability during all potential operational modes.

(2) The fuel assembly provides sufficient structural integrity to prevent operational impairment of any reactor safety equipment.

The fuel assembly and its components are designed to withstand:

(1) The predicted thermal, pressure and mechanical interaction loadings occurring during startup testing, normal operation, and anticipated operational occurrences

(2) Loading predicted to occur during handling

(3) Incore loading predicted to occur from an operational basis earthquake occurring during normal operating conditions

Operating limits are established to ensure that actual fuel operation is maintained within the fuel rod thermal-mechanical design bases. These operating limits define the maximum allowable fuel pellet operating power level as a function of fuel pellet exposure. Lattice local power and exposure capabilities are applied to transform the maximum allowable fuel pellet power level into Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limits.

The detailed design bases for each of the fuel assembly damage, failure and coolability criteria defined in Section II.A of Standard Review Plan 4.2 (except control rod reactivity; see Subsection 4.2.1.2) are provided in Section 4B.3 of Appendix 4B.

4.2.1.2 Control Rods

The control rod is designed to have:

(1) Sufficient mechanical strength to prevent displacement of its reactivity control material

(2) Sufficient strength to prevent deformation that could inhibit its motion

The detailed design bases for the control rod are provided in Appendix 4C.

4.2.2 Description and Design Drawings

4.2.2.1 [Fuel Assembly

The reference core uses the GE P8x8R fuel design. Information for this fuel design is provided under Tab AY (ABWR fuel design) of Reference 4.2-1. The fuel assembly is shown in Figure 4.2-1, and consists of a fuel bundle and a channel which surrounds the fuel bundle. The fuel bundle contains 62 fuel rods and two water rods. The fuel and water rods are spaced and supported in an eight-by-eight array by upper and lower tieplates and spacers. The lower tieplate has a nosepiece which has the function of supporting the fuel assembly in the reactor. The upper tieplate has a handle for transferring the fuel bundle from one location to another. The identifying fuel assembly serial number is engraved on the top of the handle; no two assemblies bear the same serial number. A boss projects from one side of the handle to ensure proper orientation of the assembly in the core. Finger springs located between the lower tieplate and channel are utilized to control the bypass flow through that flow path.

Fuel assembly parameters are provided in Table 1.3-1.

4.2.2.1.1 Fuel Rods

Two types of fuel rods are used in a fuel bundle; tie rods and standard rods. The tie rods in each fuel bundle have lower end plugs which thread into the lower tieplate and threaded upper end plugs which extend through the upper tieplate. A nut and locking tab are installed on the upper end plug to hold the fuel bundle together. The tie rods support the weight of the assembly only during fuel handling operations. During operation, the assembly is supported by the lower tieplate.

The end plugs of the standard rods have shanks which fit into bosses in the tieplates. An expansion spring is located over the upper end plug shank of each rod in the bundle to keep the rods seated in the lower tieplate.

Each fuel rod contains high density ceramic uranium dioxide fuel pellets stacked within Zircaloy cladding. The fuel rod is evacuated, backfilled with helium, and sealed with end plugs welded into each end. U-235 enrichments may vary from fuel rod to fuel rod within a bundle to reduce local peak-to-average fuel rod power ratios. Selected fuel rods within each bundle may include small amounts of gadolium as a burnable poison.

Adequate free volume is provided within each fuel rod in the form of a pellet-to-cladding gap and a plenum region at the top of each fuel rod. A plenum spring, or retainer, is provided in the plenum space to minimize the movement of the column of fuel pellets inside the fuel rod during shipping and handling. A hydrogen getter is also provided in the plenum space as assurance against chemical attack from inadvertent admission of moisture or hydrogenous impurities into the fuel rod during manufacture.

4.2.2.1.2 Water Rods

Water rods are hollow Zircaloy tubes with several holes around the circumference near each end to allow coolant to flow through. One water rod in each bundle axially positions the spacers.

4.2.2.1.3 Fuel Spacer

The primary function of the spacer is to provide lateral support and spacing of the fuel rods, with consideration of thermal-hydraulic performance, fretting wear, strength, neutron economy, and producibility.

4.2.2.1.4 Finger Springs

Finger springs are employed to control the bypass flow through the channel-to-lower tieplate flow path for some fuel assemblies.

4.2.2.1.5 Channels

The fuel channel is composed of a Zirconium based material or equivalent, and performs the following functions:

(1) *Forms the fuel bundle flow path outer periphery for bundle coolant flow*

(2) *Provides surfaces for control rod guidance in the reactor core*

(3) *Provides structural stiffness to the fuel bundle during lateral loadings applied from fuel rods through the fuel spacers*

(4) *Minimizes, in conjunction with the finger springs and bundle lower tieplate, coolant bypass flow at the channel/lower tieplate interface*

(5) *Transmits fuel assembly seismic loadings to the top guide and fuel support of the core internal structures*

(6) *Provides a heat sink during loss-of-coolant accident (LOCA)*

(7) *Provides a stagnation envelope for incore fuel sipping*

The channel is open at the bottom and makes a sliding seal fit on the lower tieplate surface. The upper end of the fuel assemblies in a four-bundle cell are positioned in the corners of the cell against the top guide beams by the channel fastener springs. At the top of the channel, two diagonally opposite corners have welded tabs which support the weight of the channel on the threaded raised posts of the upper tieplate. One of these raised posts has a threaded hole. The channel is attached to the fuel bundle using the threaded channel fastener assembly, which also includes the fuel assembly positioning spring. Channel-to-channel spacing is assured by to the fuel bundle spacer buttons located on the upper portion of the channel adjacent to the control rod passage area.

4.2.2.2 Control Rods

The control rod assemblies (Figure 4.2-2) perform the functions of power shaping, reactivity control, and scram reactivity insertion for safety shutdown response. Power distribution in the core is *controlled during operation of the reactor by manipulating selected patterns of control rods to counterbalance steam void effects at the top of the core.*

The control rod assembly consists of a sheathed cruciform array of stainless steel tubes filled with boron carbide powder. The main structure of a control rod consists of a top handle, a bottom casting and control rod drive coupling, a vertical center post, and four U-shaped absorber tube sheaths. The top handle, bottom casting and center post are welded into a single skeletal structure. The U-shaped sheaths are welded to the center post, handle and castings to form a rigid housing to contain the absorber tubes. Rollers at the top and bottom of the control rod guide the control rod as it is inserted and withdrawn from the core.

The boron carbide powder in the absorber tubes is compacted to about 70% of its theoretical density and contains a minimum of 76.5% by weight of natural boron. The boron-10 minimum content of the boron is 18% by weight. The absorber tubes are sealed by a plug welded into each end. The boron carbide is longitudinally separated into individual compartments by stainless steel balls at approximately 25 cm intervals. The steel balls are held in place by a slight crimp of the tube.]*

4.2.3 Design Evaluation

4.2.3.1 Fuel Assembly

4.2.3.1.1 [*Evaluation Methods*

The thermal-mechanical evaluations described in Section 4B.3 of Appendix 4B, with the exception of the stress/strain analyses in 4B.3(2)(a), are all performed using the NRC-approved GESTR-MECHANICAL fuel rod thermal-mechanical performance model. The stress/strain methodology is described later in this subsection. Any changes to this methodology must have prior NRC review and approval.

GESTR-MECHANICAL

The GESTR-MECHANICAL fuel rod performance model performs best estimate coupled thermal and mechanical analyses of a fuel rod experiencing a variable operating history. The model explicitly addresses the effects of:

- *Fuel and cladding thermal expansion*
- *Fuel and cladding creep and plasticity*
- *Cladding irradiation growth*
- *Cladding irradiation hardening and thermal annealing of that irradiation hardening*
- *Fuel irradiation swelling*

* See Section 4.2.

- *Fuel irradiation-induced densification*
- *Fuel cracking and relocation*
- *Fuel hot pressing*
- *Fission gas generation and exposure-enhanced fission gas release including fission product helium release*
- *Differential axial expansion of the fuel and cladding reflecting axial slip or lockup of the fuel pellets with the cladding*
- *Fuel phase change volumetric expansion upon melting*

The GESTR-MECHANICAL material properties and component models represent the latest experimental information available.

NRC approval of the GESTR-MECHANICAL model and its application methodology is provided in Reference 4.2-2.

Stress/Strain Analyses

The fuel rod cladding stress analyses are performed using a Monte Carlo statistical method in conjunction with distortion energy theory. Fuel cladding plasticity analyses are also performed when required by the loading conditions.

NRC approval of the stress/strain analyses methodology is provided in Reference 4.2-2.]*

4.2.3.1.2 Evaluation Results

The fuel rod thermal-mechanical evaluations described in Section 4B.3 of Appendix 4B have been completed for the reference fuel design (GE P8x8R design) using the methodologies described in Subsection 4.2.3.1.1. The evaluations demonstrate that the criteria of Appendix 4B are satisfied for the reference fuel design. The NRC previously reviewed and approved this in Reference 4.2-2.

The approval in Reference 4.2-2 contains the following conditions:

(1) [*The license/applicant must provide a plant-specific analysis of combined seismic and LOCA loading using NRC-approved methodology or another acceptable method to demonstrate conformance to the structural acceptance requirements described in Appendix A of Standard Review Plan Section 4.2.*

(2) *The license/applicant must provide an acceptable post-irradiation surveillance program or endorse the approved GE fuel surveillance program.*

* See Section 4.2.

For the reference fuel design, the second condition is satisfied by the fuel surveillance program described in Section 4B.2(3) of Appendix 4B (see also Reference 4.2-3).]*

4.2.3.2 Control Rods

4.2.3.2.1 Evaluation Results

The control rod evaluations described in Section 4C.3 have been completed for the reference control rod. The evaluations demonstrate that the criteria of Appendix 4C are satisfied for the reference B_4C control rod.

4.2.4 Testing, Inspection, and Surveillance Plans

GE has an active program of surveillance of fuel, both production and developmental. [*The NRC has reviewed the GE program and approved it in Reference 4.2-3.*]*

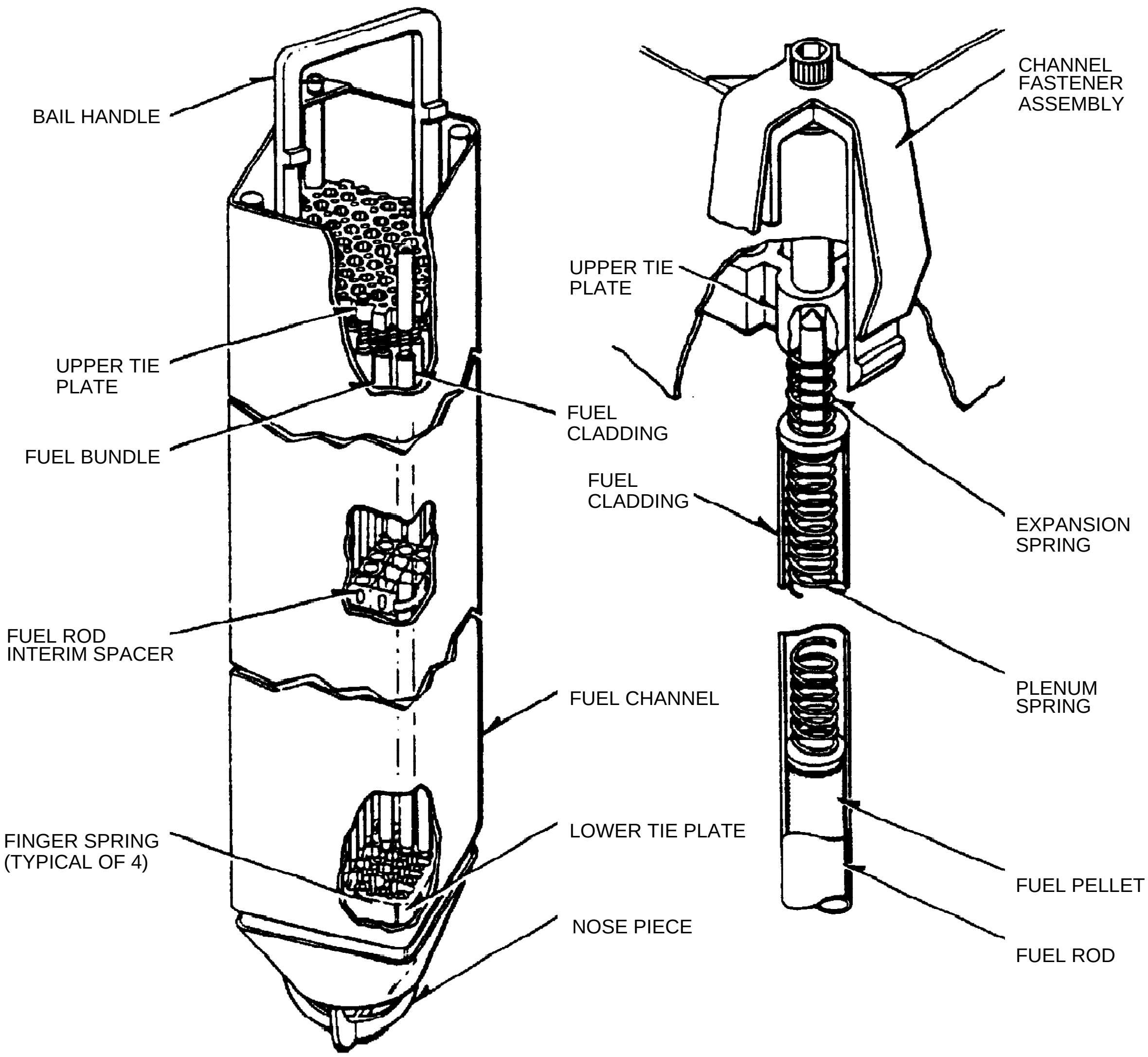
4.2.5 References

4.2-1 [*"GE Fuel Bundle Designs", NEDE-31152P, December 1988.*]*

4.2-2 [*Letter, C. O. Thomas (NRC) to J. S. Charnley (GE), "Acceptance for referencing of Licensing Topical Report NEDE-24011-P Amendment 7 to Revision 6, General Electric Standard Application for Reactor Fuel", March 1, 1985.*]*

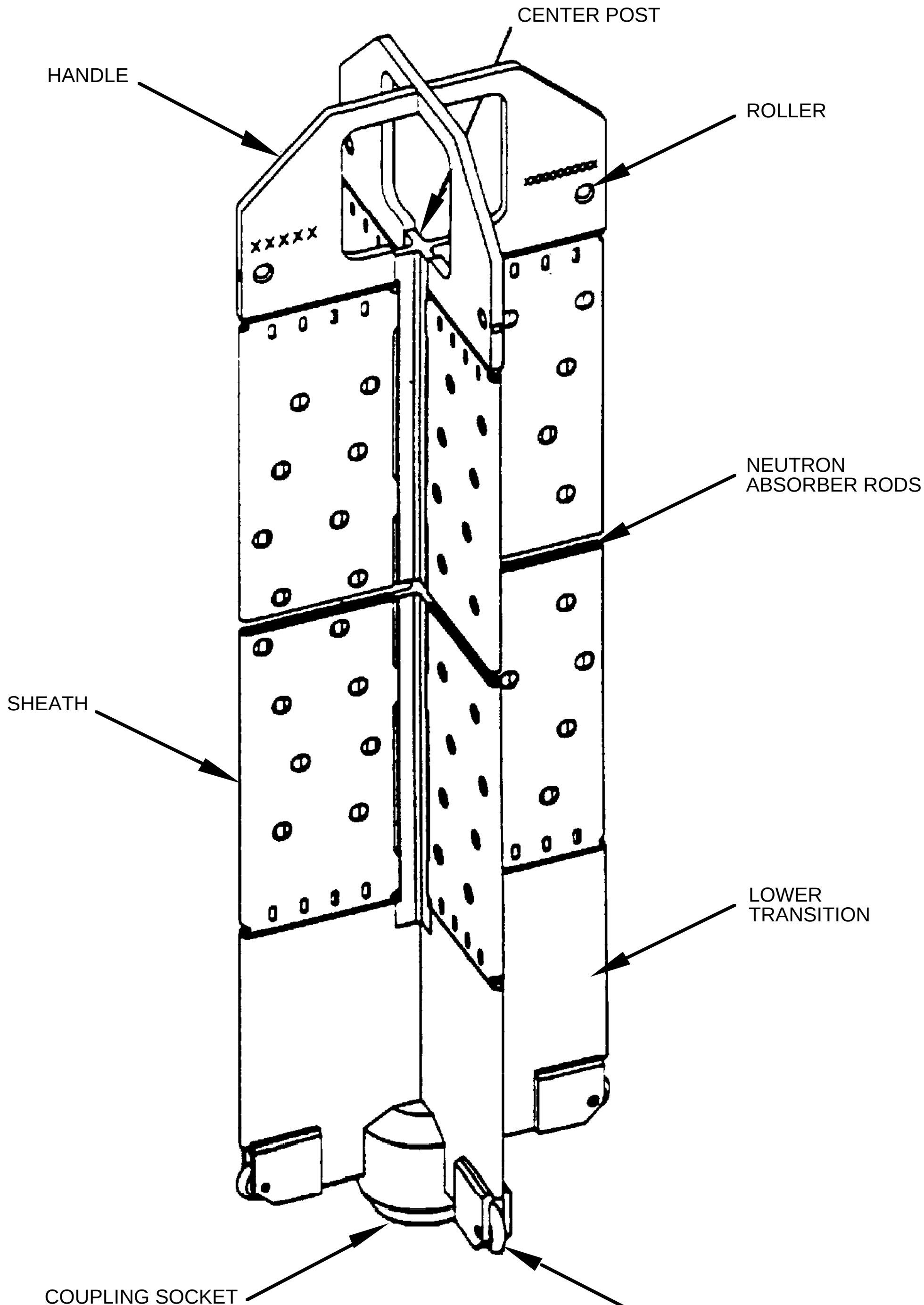
4.2-3 [*Letter, L. S. Rubenstein (NRC) to R. L. Gridley (GE), "Acceptance of GE Proposed Fuel Surveillance Program", June 27, 1984.*]*

* See Section 4.2.



*See Section 4.2 for restriction to change this figure.

[*Figure 4.2-1 Fuel Assembly*]*



*See Section 4.2 for restriction to change this figure.

[*Figure 4.2-2 Control Rod Assembly*]*

4.3 Nuclear Design

This section describes the nuclear core design basis and the models used to analyze the fuel discussed in Section 4.2.

4.3.1 Design Basis

The design bases are those that are required for the plant to operate meeting all safety requirements. Safety design bases fall into two categories:

(1) The reactivity basis, which prevents an uncontrolled positive reactivity excursion

(2) The overpower bases, which prevent the core from operating beyond the fuel integrity limits

4.3.1.1 Reactivity Basis

The nuclear design shall meet the following basis: The core shall be capable of being made subcritical at any time or at any core condition with any control rod pair (with same HCU) fully withdrawn.

4.3.1.2 Overpower Bases

The Technical Specifications limits on Minimum Critical Power Ratio (MCPR) and the Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) are determined such that the fuel will not exceed required licensing limits during abnormal operational occurrences or accidents.

4.3.2 Description

The ABWR core design consists of a light-water moderated reactor, fueled with slightly enriched uranium-dioxide. The use of water as a moderator produces a neutron energy spectrum in which fissions are caused principally by thermal neutrons. At normal operating conditions, the moderator boils, producing a spatially variable distribution of steam voids in the core. The ABWR design provides a system for which is reduced by an increase in the steam void content in the moderator. This void feedback effect is one of the inherent safety features of the ABWR system. Any system input which increases reactor power, either in a local or gross sense, produces additional steam voids which reduce reactivity and thereby reduce the power.

4.3.2.1 Nuclear Design Descriptions

The reference loading pattern is the basis for all fuel licensing. It is designed with the intent that it will represent, as closely as possible, the actual core loading pattern; however, there will be occurrences where the number and/or types of bundles in the reference design and the actual core loading do not agree exactly.

For the purpose of this Safety Analysis Report, a reference core loading of 872 fuel bundles was used as the basis for the system dynamic response analyses in Section 6.3 and Chapter 15. [*This reference core loading pattern is provided in Figure 4.3-1.*]* The first core loading in a plant may use blank fuel bundles instead of enriched bundles in parts of the core. Some or all of these blank bundles may be replaced with enriched bundles in later cycles. An example of this type of a loading pattern is shown in Figure 4.3-2.

4.3.2.2 Power Distribution

The core power distribution is a function of fuel bundle design, core loading, control rod pattern, core exposure distributions and core coolant flow rate. The thermal performance parameters, MAPLHGR and MCPR (defined in Table 4.3-1), limit unacceptable core power distribution.

4.3.2.2.1 Power Distribution Measurements

The techniques for measurement of the power distribution within the reactor core, together with instrumentation correlations and operation limits, are discussed in Reference 4.3-1.

4.3.2.2.2 Power Distribution Accuracy

The accuracy of the calculated power distribution is discussed in Reference 4.3-2.

4.3.2.2.3 Power Distribution Anomalies

Stringent inspection procedures are utilized to ensure the correct arrangement of the core following fuel loading. A fuel loading error (a mislocated or a misoriented fuel bundle in the core) would be a very improbable event, but calculations have been performed to determine the effects of such events on CPR. The fuel loading error is discussed further in Chapter 15.

The inherent design characteristics of the BWR are well suited to limit gross power tilting. The stabilizing nature of the large moderator void coefficient effectively reduces the effect of perturbations on the power distribution. In addition, the incore instrumentation system, together with the online computer, provides the operator with prompt information on the power distribution so that he can readily use control rods or other means to limit the undesirable effects of power tilting. Because of these design characteristics, it is not necessary to allocate a specific margin in the peaking factor to account for power tilt. If, for some reason, the power distribution could not be maintained within normal limits using control rods and flow, then the total core power would have to be reduced.

* See Section 4.2.

4.3.2.3 Reactivity Coefficients

Reactivity coefficients, the differential changes in reactivity produced by differential changes in core conditions, are useful in calculating stability and evaluating the response of the core to external disturbances. The base initial condition of the system and the postulated initiating event determine which of the several defined coefficients are significant in evaluating the response of the reactor. The coefficients of interest, relative to ABWR systems, are discussed here individually.

There are two primary reactivity coefficients that characterize the dynamic behavior of boiling water reactors: The Doppler reactivity coefficient and the moderator void reactivity coefficient. Also associated with the ABWR is a power reactivity coefficient and a temperature coefficient. The power coefficient is a combination of the Doppler and void reactivity coefficients in the power operating range, and the temperature coefficient is merely a combination of the Doppler and moderator temperature coefficients. Power and temperature coefficients are not specifically calculated for reload cores.

4.3.2.3.1 Doppler Reactivity Coefficient

The Doppler coefficient is of prime importance in reactor safety. The Doppler coefficient is a measure of the reactivity change associated with an increase in the absorption of resonance-energy neutrons caused by a change in the temperature of the material in question. The Doppler reactivity coefficient provides instantaneous negative reactivity feedback to any rise in fuel temperature, on either a gross or local basis. The magnitude of the Doppler coefficient is inherent in the fuel design and does not vary significantly among BWR designs. For most structural and moderator materials, resonance absorption is not significant, but in U-238 and Pu-240 an increase in temperature produces a comparatively large increase in the effective absorption cross-section. The resulting parasitic absorption of neutrons causes a significant loss in reactivity. In ABWR fuel, in which approximately 97% of the uranium in UO_2 is U-238, the Doppler coefficient provides an immediate negative reactivity response that opposes increased fuel fission rate changes.

Although the reactivity change caused by the Doppler effect is small compared to other power-related reactivity changes during normal operation, it becomes very important during postulated rapid power excursions in which large fuel temperature changes occur. The most severe power excursions are those associated with rod drop accidents. A local Doppler feedback associated with a 1650°C to 2760°C temperature rise is available for terminating the initial excursion.

The Doppler coefficient is determined using the theory and methods described in Reference 4.3-2.

4.3.2.3.2 Moderator Void Coefficient

The moderator void coefficient should be large enough to prevent power oscillation due to spatial xenon changes yet small enough that pressurization transients do not unduly limit plant operation. In addition, the void coefficient in the ABWR has the ability to flatten the radial power distribution and to provide ease of reactor control due to the void feedback mechanism The overall void coefficient is always negative over the complete operating range since the ABWR design is under moderated.

A detailed discussion of the methods used to calculate void reactivity coefficients, their accuracy and their application to plant transient analyses, is presented in Reference 4.3-2.

4.3.2.4 Control Requirements

The General Electric ABWR control rod system is designed to provide adequate control of the maximum excess reactivity anticipated during the plant operation. The shutdown capability is evaluated assuming a cold, xenon-free core.

4.3.2.4.1 Shutdown Reactivity

The core must be capable of being made subcritical, with margin, in the most reactive condition throughout the operating cycle with the most reactive control rod fully withdrawn and all other rods fully inserted. The shutdown margin is determined by using the BWR simulator code (see Section 4.3.3) to calculate the core multiplication at selected exposure points with the strongest rod fully withdrawn. The shutdown margin is calculated based on the carryover of the minimum expected exposure at the end of the previous cycle. The core is assumed to be in the cold, xenon-free condition in order to ensure that the calculated values are conservative. Further discussion of the uncertainty of these calculations is given in Reference 4.3-3.

As exposure accumulates and burnable poison depletes in the lower exposure fuel bundles, an increase in core reactivity may occur. The nature of the increase depends on specifics of fuel loading and control state.

The cold k_{eff} is calculated with the strongest control rod out at various exposures through the cycle. A value R is defined as the difference between the strongest rod out k_{eff} at BOC and the maximum calculated strongest rod out k_{eff} at any exposure point. The strongest rod out k_{eff} at any exposure point in the cycle is equal to or less than:

where:

k_{eff} = k_{eff} (Strongest rod withdrawn)$_{BOC}$ + R,

R is always greater than or equal to 0. The value of R includes equilibrium Sm.

The calculated values of k_{eff} with the strongest rod withdrawn at BOC and of R are reported in Table 4.3-2. For completeness, the uncontrolled k_{eff} and fully controlled k_{eff} values are also reported in Table 4.3-2.

4.3.2.4.2 Reactivity Variations

The excess reactivity designed into the core is controlled by the control rod system supplemented by gadolinia-urania fuel rods. Control rods are used during the cycle partly to compensate for burnup and partly to control the power distribution.

4.3.2.4.3 Standby Liquid Control System

The Standby Liquid Control System (SLCS) is designed to provide the capability of bringing the reactor, at any time in a cycle, from a full power and minimum control rod inventory (which is defined to be at the peak of the xenon transient) to a subcritical condition with the reactor in the most reactive xenon-free state. The requirements of this system are dependent primarily on the reactor power level and on the reactivity effects of voids and temperature between full-power and cold, xenon-free condition. The SLCS is discussed in Subsection 9.3.5.

4.3.2.5 Criticality of Reactor During Refueling

The core is subcritical at all times.

4.3.2.6 Stability

4.3.2.6.1 Xenon Transients

Boiling water reactors do not have instability problems due to xenon. This has been demonstrated by:

(1) Never having observed xenon instabilities in operating BWRs

(2) Special tests which have been conducted on operating BWRs in an attempt to force the reactor into xenon instability

(3) Calculations

All of these indicators have proven that xenon transients are highly damped in a BWR due to the large negative power coefficient.

Analysis and experiments conducted in this area are reported in Reference 4.3-2.

4.3.2.6.2 Thermal Hydraulic Stability

The compliance of GE fuel designs to the criteria set forth in General Design Criterion 12 is demonstrated provided that the following stability compliance criteria are satisfied using approved methods:

(1) Neutron flux limit cycles, which oscillate up to 120% APRM high neutron flux scram setpoint or up to the LPRM upscale alarm trip (without initiating scram) prior to operator mitigating action shall not result in exceeding specified acceptable fuel design limits.

(2) The individual channels shall be designed and operated to be hydrodynamically stable or more stable than the reactor core for all expected operating conditions (analytically demonstrated).

The GE methodology for demonstrating the above has been reviewed and approved by the NRC in Reference 4.3-4. The stability compliance of the fuel designs described in Subsection 4.2.2.1 has been demonstrated on a generic basis and approved by the NRC in Reference 4.3-4. Should a different fuel design be chosen by the COL applicant, the methodology for demonstrating stability will be that approved by the NRC. See Subsection 4.3.5.1 for COL license information. See Subsection 4.4.3.7 for specific thermal hydraulic stability performance information.

4.3.3 Analytical Methods

The nuclear evaluations of all General Electric cores are performed using the analytical tools and methods described in Reference 4.3-2.

The lattice analyses are performed during the bundle design process. The results of these single bundle calculations are reduced to “libraries” of lattice reactivities, relative rod powers, and few group cross-sections as functions of instantaneous void, exposure, exposure-void history, control state, and fuel and moderator temperature, for use in the core analysis. These analyses are dependent upon fuel lattice parameters only and are, therefore, valid for all plants and cycles to which they are applied.

The core analysis is unique for each cycle. It is performed in the months preceding the cycle loading to demonstrate that the core meets all applicable safety limits. The principal tool used in the core analysis is the three-dimensional BWR simulator code, which computes power distributions, exposure, and reactor thermal-hydraulic characteristics, with spatially varying voids, control rods, burnable poisons and other variables.

4.3.4 Changes

Not applicable.

4.3.5 COL License Information

4.3.5.1 Thermal Hydraulic Stability

In the event the COL applicant chooses a fuel design different than that described herein, the methodology for demonstrating stability compliance will be that approved by the NRC (Subsection 4.3.2.6.2).

4.3.6 References

4.3-1 J.F. Carew, "Process Computer Performance Evaluation Accuracy", NEDO-20340, June 1974.

4.3-2 "Steady-State Nuclear Methods", NEDO-30130-AII April 1985.

4.3-3 "BWR/4,5,6 Standard Safety Analysis Report", Revision 2, Chapter 4, June 1977.

4.3-4 Letter, C. O. Thomas (NRC) to H. C. Pfefferlen (GE), "Acceptance for Referencing of Licensing Topical Report" NEDE-24011, Rev. 6, Amendment 8. "Thermal Hydraulic Stability Amendment to GESTAR II", April 24, 1985.

Table 4.3-1 Definition Of Fuel Design Limits

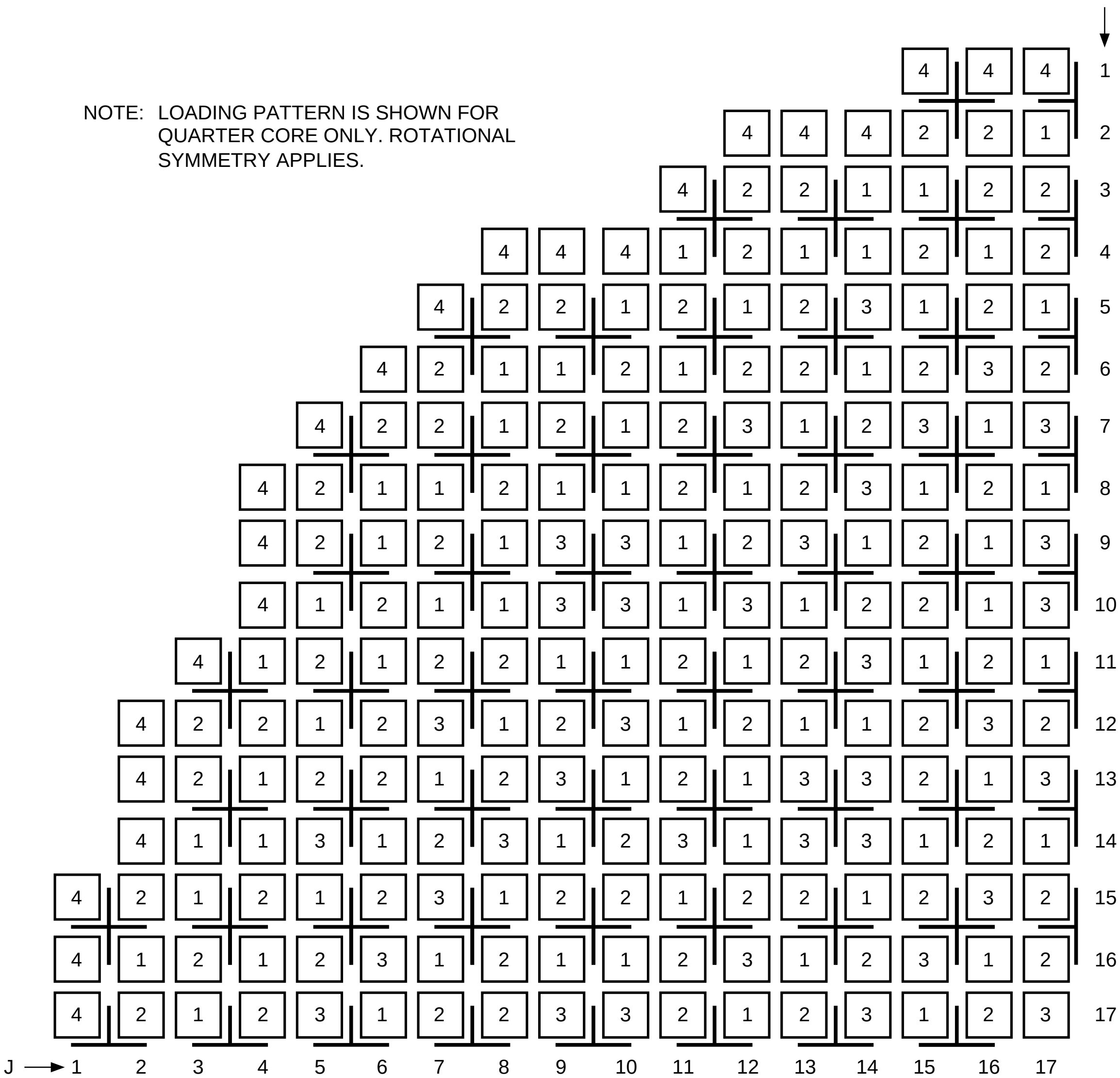
<table>
<tr><th>Maximum Average Planar Heat Generation Rate (MAPLHGR)</th></tr>
<tr><td>The MAPLHGR is the maximum average linear heat generation rate (expressed in kW/m) in any plane of a fuel bundle allowed by the Plant Technical Specifications for that fuel type. This parameter is obtained by averaging the linear heat generation rate over each fuel rod in the plane, and its limiting value is selected such that:
(a) The peak clad temperature during the design basis loss-of-coolant accident will not exceed 1204°C in the plane of interest, and
(b) All fuel design limits specified in Section 4.2 will be met.</td></tr>
<tr><th>Minimum Critical Power Ratio (MCPR)</th></tr>
<tr><td>The MCPR operating limit is the minimum CPR allowed by the Plant Technical Specifications for a given bundle type. The minimum CPR is a function of several parameters, the most important of which are bundle power, bundle flow and bundle R-factor. The R-factor is dependent upon the local power distribution and details of the bundle mechanical design including channel bow considerations. The limiting value of CPR is selected for each bundle type such that, during the most limiting event of moderate frequency, the calculated CPR in that bundle is not less than the safety limit CPR. The MCPR operating limit is attained when the bundle power, R-factor, flow, and other relevant parameters combine to yield the Technical Specification value.</td></tr>
</table>

Table 4.3-2 Calculated Core Effective Multiplication and Control System Worth—No Voids, 20°C

Beginning of Cycle, K-effective*	
Uncontrolled	1.0876
Fully Controlled	0.9145
Strongest Control Rod Pair Out	0.9591
R, Maximum Increase in Cold Core Reactivity with Exposure Cycle, ýk	0.0000

* For the core loading in Figure 4.3-1.

NOTE: LOADING PATTERN IS SHOWN FOR QUARTER CORE ONLY. ROTATIONAL SYMMETRY APPLIES.



BUNDLE TYPE	ENRICHMENT	NUMBER OF BUNDLES
1	HIGH ENRICHMENT, 3.18 WT %	308
2	MEDIUM ENRICHMENT, 2.18 WT %	324
3	LOW ENRICHMENT, 1.23 WT %	148
4	NATURAL URANIUM, 0.71 WT %	92

*See Section 4.2 for restriction to change this figure.

Figure 4.3-1 Core Loading Map Used for Response Analyses]*
[

Figure 4.3-2 [Proprietary information not included in DCD (Refer to SSAR Section 4.3, Amendment 31)]

4.4 Thermal–Hydraulic Design

4.4.1 Design Basis

4.4.1.1 Safety Design Bases

Thermal-hydraulic design of the core shall establish the thermal-hydraulic safety limits for use in evaluating the safety margin relating the consequences of fuel cladding failure to public safety.

4.4.1.2 Requirements for Steady-State Conditions

For purposes of maintaining adequate fuel performance margin during normal steady-state operation, the MCPR must not be less than the required MCPR operating limit, and the APLHGR must be maintained below the required APLHGR limit (MAPLHGR). The steady-state MCPR and MAPLHGR limits are determined by analyses of the most severe moderate frequency anticipated operational occurrences (AOOs) to accommodate uncertainties and provide reasonable assurance that no fuel damage results during moderate frequency AOOs at any time in life. These limits are provided in the Technical Specifications.

4.4.1.3 Requirements for Anticipated Operational Occurrences (AOOs)

The MCPR and MAPLHGR limits are established such that no safety limit is expected to be exceeded during the most severe moderate frequency AOO event as defined in Chapter 15.

4.4.1.4 Summary of Design Bases

In summary, the steady-state operating limits have been established to assure that the design bases are satisfied for the most severe moderate frequency AOO. Demonstration that the steady-state MCPR and MAPLHGR limits are not exceeded is sufficient to conclude that the design bases are satisfied.

4.4.2 Description of Thermal-Hydraulic Design of the Reactor Core

4.4.2.1 Summary Comparison

Typical thermal-hydraulic parameters for the ABWR are compared to those for a typical BWR/6 plant in Table 4.4-1.

4.4.2.2 Critical Power Ratio

A description of the critical power ratio is provided in Subsection 4.4.7.1. Criteria used to calculate the critical power ratio safety limit are given in Appendix 4B, Subsection 4B.8.

4.4.2.3 Average Planar Linear Heat Generation Rate (APLHGR)

Models used to calculate the APLHGR limit are given in Subsection 4.2.3.1.1 as pertaining to the fuel mechanical design limits, and in Subsection 6.3.3.7 as pertaining to 10CFR50 Appendix K limits.

4.4.2.4 Void Fraction Distribution

The axial distribution of core void fractions for the average radial channel and the maximum radial channel (end of node value) for the reference core loading pattern (Figure 4.3-1) are given in Table 4.4-2. The core average and maximum exit value is also provided. Similar distributions for steam quality are given in Table 4.4-3. The core average axial power distribution used to produce these tables is given in Table 4.4-4.

4.4.2.5 Core Coolant Flow Distribution and Orificing Pattern

The flow distribution to the fuel assemblies and bypass flow paths is calculated on the assumption that the pressure drop across all fuel assemblies and bypass flow paths is the same. This assumption has been confirmed by measuring the flow distribution in boiling water reactors (References 4.4-1, 4.4-2, and 4.4-3). The components of bundle pressure drop considered are friction, local, elevation, and acceleration (Subsections 4.4.2.6.1 through 4.4.2.6.4, respectively). Pressure drop measurements made in operating reactors confirm that the total measured core pressure drop and calculated core pressure drop are in good agreement. There is reasonable assurance, therefore, that the calculated flow distribution throughout the core is in close agreement with the actual flow distribution of an operating reactor.

An iteration is performed on flow through each flow path (fuel assemblies and bypass flow paths), which equates the total differential pressure (plenum to plenum) across each path and matches the sum of the flows through each flow path to the total core flow. The total core flow less the control rod cooling flow enters the lower plenum. A fraction of this passes through various bypass flow paths. The remainder passes through the orifice in the fuel support plate (experiencing a pressure loss) where some of the flow exists through the fit-up between the fuel support and the lower tieplate and through the lower tieplate holes into the bypass flow region. All initial and reload core fuel bundles have lower tieplate holes. The majority of the flow continues through the lower tieplate (experiencing a pressure loss) where some flow exists through the flow path defined by the fuel channel and lower tieplate into the bypass region. This bypass flow is lower for those fuel assemblies with finger springs.

Within the fuel assembly, heat balances on the active coolant are performed nodally. Fluid properties are expressed as the bundle average at the particular node of interest and are based on 1967 International Standard Steam-Water Properties. In evaluating fluid properties a constant pressure model is used.

The relative radial and axial power distributions documented in the country-specific supplement are used with the bundle flow to determine the axial coolant property distribution, which gives sufficient information to calculate the pressure drop components within each fuel assembly type. When the equal pressure drop criterion described above is satisfied, the flow distributions are established.

4.4.2.6 Core Pressure Drop and Hydraulic Loads

The components of bundle pressure drop considered are friction, local, elevation and acceleration pressure drops. Pressure drop measurements made in operating reactors confirm that the total measured core pressure drop and calculated core pressure drop are in good agreement.

4.4.2.6.1 Friction Pressure Drop

Friction pressure drop is calculated with a basic model as follows:

$$\Delta P_f = \frac{w^2}{2g_c\rho}\frac{fL}{D_H A_{ch}^2}\phi_{TPF}^2 \qquad (4.4\text{-}1)$$

where

ΔP_f	=	friction pressure drop
w	=	mass flow rate
g_c	=	conversion constant
ρ	=	average nodal liquid density
D_H	=	channel hydraulic diameter
A_{ch}	=	channel flow area
L	=	incremental length
f	=	friction factor
ϕ_{TPF}	=	two-phase friction multiplier

The formulation for the two-phase multiplier is similar to that presented in References 4.4-4 and 4.4-5, and is based on data that is taken from prototypical BWR fuel bundles.

4.4.2.6.2 Local Pressure Drop

The local pressure drop is defined as the irreversible pressure loss associated with an area change, such as the orifice, lower tieplate, and spacers of a fuel assembly.

The general local pressure drop model is similar to the friction pressure drop and is

$$\Delta P_L = \frac{w^2}{2g\rho}\frac{K}{A^2}\phi^2_{TPL} \tag{4.4-2}$$

where

ΔP_L	=	local pressure drop
K	=	local pressure drop loss coefficient
A	=	reference area for local loss coefficient
ϕ_{TPL}	=	two-phase local multiplier

and w, g, and ρ are defined above. The formulation for the two-phase multiplier is similar to that reported in Reference 4.4-5. For advanced spacer designs a quality modifier has been incorporated in the two-phase multiplier to better fit the data. Empirical constants were added to fit the results to data taken for the specific designs of the ABWR fuel assembly. These data were obtained from tests performed in single-phase water to calibrate the orifice, the lower tieplate, and the holes in the lower tieplate, and in both single- and two-phase flow, to derive the best fit design values for spacer and upper tieplate pressure drop. The range of test variables was specified to include the range of interest for the ABWR. New test data are obtained whenever there is a significant design change to ensure the most applicable methods are used.

4.4.2.6.3 Elevation Pressure Drop

The elevation pressure drop is based on the relation:

$$\begin{gathered}\Delta P_E = \bar{\rho}\Delta L; \\ \bar{\rho} = \rho_f(1-\alpha) + \rho_g\alpha\end{gathered} \tag{4.4-3}$$

where

ΔP_E	=	elevation pressure drop
ΔL	=	incremental length

$\bar{\rho}$ = average mixture density

α = nodal average void fraction

ρ_f, ρ_g = saturated water and vapor density, respectively

The void fraction model used is an extension of the Zuber-Findlay model (Reference 4.4-6), and uses an empirical fit constant to predict a large block of steam void fraction data. Checks against new data are made on a continuing basis to ensure the best models are used over the full range of interest.

4.4.2.6.4 Acceleration Pressure Drop

A reversible pressure change occurs when an area change is encountered, and an irreversible loss occurs when the fluid is accelerated through the boiling process. The basic formulation for the reversible pressure change resulting from a flow area change in the case of single-phase flow is given by:

$$\Delta P_{ACC} = \left(1-\sigma_A^2\right)\frac{W^2}{2g_c\rho_1 A_2^2}$$
$$\sigma_A = \frac{A_2}{A_1} = \frac{\text{final flow area}}{\text{initial flow area}} \tag{4.4-4}$$

where

ΔP_{ACC} = acceleration pressure drop

A_2 = final flow area

A_1 = initial flow area

In the case of two-phase flow, the liquid density is replaced by a density ratio so that the reversible pressure change is given by:

$$\Delta P_{ACC} = \left(1-\sigma_A^2\right)\frac{W^2\rho_H}{2g_c\rho^2{}_{KE}A_2^2} \tag{4.4-5}$$

where

$\frac{1}{\rho_H} = \frac{x}{\rho_g} + \frac{(1-x)}{\rho_l}$, homogeneous density

$$\frac{1}{\rho_{KE}^2} = \frac{x^3}{\rho_g^2 \alpha^2} + \frac{(1-x)^3}{\rho_l^2 (1-\alpha)^2}, \tag{4.4-6}$$

α = void fraction at A_2

x = steam quality at A_2

and other terms are as previously defined. The basic formulation for the acceleration pressure change due to density change is:

$$\Delta P_{ACC} = \frac{W^2}{g_c A_{ch}^2}\left(\frac{1}{\rho_{out}} - \frac{1}{\rho_{in}}\right) \tag{4.4-7}$$

where ρ is either the homogeneous density, ρ_H, or the momentum density, ρ_M

$$\frac{1}{\rho_M} = \frac{x^2}{\rho_g \alpha} + \frac{(1-x)^2}{\rho_l (1-\alpha)} \tag{4.4-8}$$

and is evaluated at the inlet and outlet of each axial node. Other terms are as previously defined. The total acceleration pressure drop in the ABWR is on the order of a few percent of the total pressure drop.

4.4.2.7 Correlation and Physical Data

General Electric Company has obtained substantial amounts of physical data in support of the pressure drop and thermal-hydraulic loads discussed in Subsection 4.4.2.6. Correlations have been developed to fit these data to the formulations discussed.

4.4.2.7.1 Pressure Drop Correlations

General Electric Company has taken significant amounts of friction pressure drop data in multi-rod geometries representative of BWR plant fuel bundles and correlated both the friction factor and two-phase multipliers on a best fit basis using the pressure drop formulations reported in Subsection 4.4.2.6.1 and 4.4.2.6.2. Tests are performed in single-phase water to calibrate the orifice and the lower tie-plate, and in both single- and two-phase flow to arrive at best fit design values for spacer and upper tieplate pressure drop. The range of test variables is specified to include the range of test variables is specified to include the range of interest to the ABWR. New data are taken whenever there is a significant design change to ensure the most applicable methods are in use at all times.

Applicability of the single-phase and two-phase hydraulic models discussed in Subsections 4.4.2.6.1 and 4.4.2.6.2 for the fuel designs described in Subsection 4.2.2, was confirmed by full scale prototype flow tests.

4.4.2.7.2 Void Fraction Correlation

The void fraction correlation includes effects of pressure, flow direction, mass velocity, quality, and subcooled boiling.

4.4.2.7.3 Heat Transfer Correlation

The Jens-Lottes (Reference 4.4-7) heat transfer correlation is used in fuel design to determine the cladding-to-coolant heat transfer coefficients for nucleate boiling.

4.4.2.8 Thermal Effects of Anticipated Operational Occurrences

The evaluation of the core's capability to withstand the thermal effects resulting from anticipated operational occurrences is covered in Chapter 15.

4.4.2.9 Uncertainties in Estimates

Uncertainties in thermal-hydraulic parameters are considered in the statistical analysis which is performed to establish the fuel cladding integrity safety limit documented in Subsection 4.4.4.1.

4.4.2.10 Flux Tilt Considerations

For flux tilt considerations, refer to Subsection 4.3.2.2.

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

4.4.3.1 Plant Configuration Data

4.4.3.1.1 Reactor Coolant System Configuration

The Reactor Coolant System is described in Section 5.4.

4.4.3.1.2 Reactor Coolant System Thermal–Hydraulic Data

The steady-state distribution of temperature, pressure and flow rate for each flow path in the Reactor Coolant System is shown in Figure 5.1-1.

4.4.3.1.3 Reactor Coolant System Geometric Data

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

Table 4.4-5 provides the flow path length, height, liquid level, minimum elevations, and minimum flow areas for each major flow path volume within the reactor vessel and recirculation loops of the Reactor Coolant System.

4.4.3.2 Operating Restrictions on Pumps

Expected recirculation pump performance curves are shown in Figure 5.4-3. These curves are valid for all conditions with a normal operation range varying from approximately 20% to 115% of rated pump flow.

Subsection 4.4.3.3 gives the operating limits imposed on the recirculation pumps by cavitation, pump load, bearing design flow starvation, pump speed, and steam separator performance.

It is required that at least 9 out of 10 RIPs are operating for normal operation. For operation with less than 9 RIPs in operation, the COL applicant will provide the necessary supporting analyses.

4.4.3.3 Power/Flow Operating Map

4.4.3.3.1 Limits for Normal Operation

The ABWR must operate with certain restrictions because of pump net positive suction head (NPSH), overall plant control characteristics, core thermal power limits, etc. The power-flow map for 10 RIP operation is shown in Figure 4.4-1, and for 9 RIP operation in Figure 4.4-2. Those power-flow maps for the power range of operation shown were used in the system response analyses documented in Section 6.3 and Chapter 15. See Subsections 4.4.7.1 and 4.4.7.2 for COL license information. The nuclear system equipment, nuclear instrumentation, and the Reactor Protection System, in conjunction with operating procedures, maintain operations within the area of the operating map for normal operating conditions. The boundaries on this map are as follows:

- **Natural Circulation Line, 0:** The operating state of the reactor moves along this line for the normal control rod withdrawal sequence in the absence of recirculation pump operation.
- **102% Power Rod Line or Rated Power (Whichever Is Less):** The 102% power rod line passes through 102% power at 100% flow. The operating state for the reactor follows this rod line (or similar ones) during recirculation flow changes with a fixed control rod pattern; however, rated power may not be exceeded.
- **Steam Separator Limit Line:** This line results from the requirements to have acceptable moisture carryover fraction from the steam separator.

The COL applicant will provide the specific power/flow operating map and thermal limits for the core loading to the NRC for information (Subsection 4.4.7.1).

4.4.3.3.2 Other Performance Characteristics

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

- **Constant Rod Lines A, B, C, D, E, F—**These lines show the change in flow associated with power change while maintaining constant control rod position.
- **Constant Pump Speed Lines 1, 2, 3, 4, 5, 6, 7, 8—**These lines show the change in flow associated with power changes while maintaining constant RIP speeds.

4.4.3.3.3 Regions of the Power/Flow Map

Region I	This region defines the system operational capability with the reactor internal pumps running at their minimum speed (30%). Power changes, during normal startup and shutdown, will be in this region. The normal operating procedure is to startup along Curve 1 shown in Figure 4.4-1.
Region II	This is the low power area of the operating map where the carryover through steam separators is expected to exceed the acceptable value. Operation within this region is precluded by system interlocks.
Region III	This is the high-power/low-flow area of the operating map in which the system is the least damped. Operation within this region is precluded by SCRRI (Selected Control Rods Run-In).
Region IV	This represents the normal operating zone of the map where power changes can be made, by either control rod movement or by core flow changes, through the change of the pump speeds.

4.4.3.3.4 Design Features for Power/Flow Control

The following limits and design features are employed to maintain power/flow conditions shown in Figure 4.4-1:

(1) **Minimum Power Limits at Intermediate and High Core Flows:** To prevent unacceptable separator performance, the recirculation system is provided with an interlock to reduce the RIP speed.

(2) **Pump Minimum Speed Limit:** The reactor internal pumps (RIPs) are equipped with anti-rotation devices (ARD) which prevent a tripped RIP from rotating backwards. The ARD begins operating at 31.4 rad/s decreasing speed. In order to prevent mechanical wear in the ARD, minimum speed is specified at 31.4 rad/s. However, to provide a stable operation, the minimum pump speed is set at 47.1 rad/s (30% of required).

4.4.3.3.5 Flow Control

The normal plant startup procedure requires the startup of all RIPs first and maintain at their minimum pump speed (30% of rated), at which point reactor heatup and pressurization can commence. When operating pressure has been established, reactor power can be increased. This power/flow increase will follow a line within Region I of the flow control map shown in Figure 4.4-1. The system is then brought to the desired power/flow level within the normal operating area of the map (Region IV) by increasing the RIP speeds and by withdrawing control rods.

Control rod withdrawal with constant pump speed will result in power/flow changes along lines of constant pump speed (Curves 1 through 8). Change of pump speeds with constant control rod position will result in power/flow changes along, or nearly parallel to, the rated flow control line (Curves A through F).

4.4.3.4 Temperature-Power Operating Map

Not applicable.

4.4.3.5 Load-Following Characteristics

Not Used.

4.4.3.6 Thermal–Hydraulic Characteristics Summary Table

The thermal-hydraulic characteristics are provided in Table 4.4-1 for the core and tables of Section 5.4 for other portions of the Reactor Coolant System.

4.4.3.7 Thermal Hydraulic Stability Performance

As discussed in the Response to Question 100.1, the ABWR design assures that stability performance in the normal operating region is more stable than current operating BWRs by incorporating the following design features:

(1) Smaller inlet orifices, which increase the inlet single-phase pressure drop, and, consequently, improve the core and channel stability

(2) Wider control rod pitch, which increases flow area, and, reduces the void reactivity coefficient and improves both core and channel stability

(3) More steam separators, which reduce the two-phase pressure drop, and improve stability

In order to reconfirm this conclusion, a stability analysis based on the procedures developed by the BWROG committee on thermal hydraulic stability (Reference 4.4-16) was performed for the ABWR. In this analysis, conservative nuclear conditions, taking into consideration of future core design, were assumed. The results at the most limiting conditions in the normal operating region (i.e., the intercept of 102% rod line with all operating RIPs at their minimum speeds, assuming only 9 out of 10 RIPs are in operation) are as follows:

- Core Decay Ratio 0.72
- Channel Decay Ratio 0.36

These results are shown in Figure 4.4-3 together with the criteria. From Figure 4.4-3, it is confirmed that the ABWR is stable in the normal operating region.

Furthermore, automatic logics (Figure 4.4-4) which prevent plant operation in the region with the least stability margin is also implemented. This design is similar to Option I-A, one of long-term solutions considered by the BWROG. In addition, in order to meet the stability design requirements specified in the ALWR Utility Requirements Document, Option III, LPRM based Oscillation Power Range Monitor (OPRM), which is also one of the long-term solutions considered by the BWROG, has been implemented in the ABWR design.

As for issues that relate to ATWS stability, they are of no concern to the ABWR design, since the ABWR design has logic to automatically initiate the SLCS, including automatic initiation of feedwater run back. Furthermore, the ABWR EPG will incorporate any changes recommended by the BWROG.

In summary, the ABWR stability design is consistent with the licensing methodology proposed by the BWROG committee on thermal hydraulic stability. The ABWR will be stable in the normal operating region.

4.4.4 Loose-Parts Monitoring System

The Loose-Parts Monitoring System (LPMS) is designed to provide detection of loose metallic parts within the reactor pressure vessel. The LPMS detects structure-borne sound that can indicate the presence of loose parts impacting against the reactor pressure vessel or its internals. The LPM detection system can evaluate some aspects of selected signals. However, the system, by itself, cannot diagnose the presence and location of a loose part.

4.4.4.1 Power Generation Design Bases

The LPMS is designed to provide detection and operator warning of loose parts in the RPV to avoid or mitigate damage to or malfunctions of reactor components.

Additional design considerations provide for the inclusion of electronic features to minimize operator interfacing requirements during normal LPMS operation. These electronic features also enhance the LPMS detection and analysis function when operator action is required to investigate potential loose parts.

4.4.4.2 System Description

The LPMS monitors the RPV for indications of loose parts. The alarm setting for each sensor is determined after system installation is complete. The alarm setting is set to discriminate between normal background noises and any actual loose part impact signal to minimize spurious alarms. Each sensor channel is isolated to reduce the possibility of signal ground loop problems and to minimize sensor signal background noise. Background noises are also minimized by use of tuned filters. A disable signal is provided during control rod movement and other plant maneuvers that may initiate a LPM alert-level alarm.

LPMS sensors are usually accelerometers. The array of LPMS accelerometers typically consist of a set of sensors strategically mounted on the external surface of the primary pressure vessel boundary at various elevations and azimuths at natural collection regions for potential loose parts. General mounting locations are at the (1) main steam outlet nozzle, (2) feedwater inlet nozzle, (3) flooded nozzles, and (4) control rod drive housings. The sensors are mounted in such a fashion as to provide good response and sensitivity, even at high frequency.

The online system sensitivity is such that the system can detect a metallic loose part that has a mass between 0.11 kg to 13.6 kg and impacts with kinetic energy of 0.68 joules or more on the inside surface of the RPV within 0.91m of a sensor. The loose parts impact frequency range of interest is typically from 1 to 10 kHz. Frequencies lower than 1 kHz are generally associated with flow-induced vibration (FIV) signals or flow noise.

Physical separation is maintained between the sensor signal cables at each natural collection region to an area where they are grouped and routed through several cable penetrations to a termination area. The termination area is accessible by maintenance personnel during full power operation.

The LPMS includes provisions for both automatic and manual startup of data acquisition equipment with automatic activation in the event that the preset alert level is reached or exceeded. The system also initiates an alarm to the control room personnel when an alert condition is reached. The Data Acquisition System will automatically select the alarmed signal sensor channel plus additional channels for simultaneous recording. The signal analysis equipment will allow immediate visual and audio monitoring of all signals.

Provisions exist for periodic online sensor signal channel check and functional tests and for offline channel calibration during periods of cold shutdown or refueling. The LPMS electronics are designed to facilitate the recognition, location, replacement, repair, and adjustment of malfunctioning LPMS electronic components. The LPMS electronic components located inside the containment have been designed and installed to perform their function following all seismic events that do not require plant shutdown. The plant will shutdown when the recorded motion exceeds the limits specified in Subsection 3.7.4.4. The LPMS electronic components selected for this application are rated to meet the normal operating radiation, vibration, temperature, and humidity environments in which the components are installed.

All LPMS electronic components within the containment are designed for a 60-year design life. In those instances where a 60-year design life is not practicable, a replacement program will be established for those parts that are anticipated to have limited service life.

4.4.4.3 Normal System Operation

The LPMS will be set to alarm for detected signals having characteristics of metal-to-metal impacts.

After installation of the sensor array, the LPMS overall and individual sensor signal channels can be characterized at plant startup before operation monitoring. Each accelerometer channel will exhibit its own particular signature and unique corresponding frequency spectrum. This signature and spectrum results from a combination of both internal and external sources due to normal and transient conditions.

Calibration is an important part of LPMS operation. The alarm level setpoint is determined by using a manual calibration device to simulate the presence of a loose part impact near each sensor. The setpoint is typically based on a percentage of the calibration signal magnitude, and is a function of actual background noise. Additionally, calibrated impacts at various locations near the sensors assist in diagnosing the source of the signal. (e.g LPM sensor signals disabled).

Discrimination logic is typically incorporated in the LPMS to avoid spurious alarms. Discrimination logic rejects events that do not have the characteristics of an impact signal of a loose part. Typical discrimination functions are based on the length of time the signal is above the setpoint, the number of channels alarming, the time between alarms, the repetition of the signal, and the signal waveform and frequency content. False alert signals due to plant maneuvers are avoided by the use of administrative procedures by control room personnel.

Usually, the plant operator makes the preliminary evaluation based on the available information. If the presence of unusual metal impact sound is indicated, then the station engineers perform additional evaluations. LPM experts are required to correctly diagnose the presence and location of a loose part. In order to reach proper conclusions, various factors must be considered such as: plant operating conditions, location of the sensors that alarmed, and comparison of the amplitude and frequency contents of these sensor signals with known normal LPMS operation sensor signal data.

4.4.4.4 Safety Evaluation

The LPMS is intended to be used for information purposes only by the plant operator. The plant operators do not rely on the information provided by the LPMS for the performance of any safety-related action. However, although the LPMS is not classified as a safety-related system, it is designed to meet the seismic and environmental operability recommendations of Regulatory Guide 1.133.

4.4.4.5 Test and Inspection

The LPMS will be calibrated to detect a metallic loose part that has a mass from 0.11 kg to 13.6 kg and impacts with kinetic energy of 0.68 joules within 0.91m of each sensor. Provisions will be made to check the LPM system calibration at the LPMS data acquisition and analysis console at each refueling. The system will be recalibrated as necessary when found to be out of calibration. A test and reset capability will be included for functional test capability.

The manufacturer will provide services of qualified personnel to provide technical guidance for installation, startup, and acceptance testing of the system. In addition, the manufacturer will provide the necessary training of plant personnel for proper system operation and maintenance and planned operating and record-keeping procedures.

4.4.4.6 Instrumentation Application

The LPMS consists of sensors, cables, signal conditioning equipment, alarming monitor, signal analysis and data acquisition equipment, and calibration equipment.

4.4.5 Evaluation

4.4.5.1 Critical Power

The objective for normal and AOOs is to maintain nucleate boiling and thus avoid a transition to film boiling. Operating limits are specified to maintain adequate margin to the onset of the boiling transition. The figure of merit utilized for plant operation is the critical power ration (CPR). This is defined as the ratio of the critical power (bundle power at which some point within the assembly experiences onset of boiling transition) to the operating bundle power. The critical power is determined at the same mass flux level, inlet temperature, and pressure which exists at the specified reactor condition. Thermal margin is stated in terms of the minimum value of the critical power ratio (MCPR) which corresponds to the most limiting fuel assembly in the core. To ensure that adequate margin is maintained, a design requirement based on a statistical analysis was selected as follows:

> Moderate frequency AOOs caused by a single operator error or equipment malfunction shall be limited such that, considering uncertainties in manufacturing and monitoring the core operating state, at least 99.9% of the fuel rods would be expected to avoid boiling transition (Reference 4.4-8).

Both the transient and normal operating thermal limits in terms of MCPR are derived from this basis.

4.4.5.2 Core Hydraulics

Core hydraulics models and correlations are discussed in Subsection 4.4.2.

4.4.5.3 Influence of Power Distributions

The influence of power distributions on the thermal-hydraulic design is discussed in Reference 4.4-8.

4.4.5.4 Core Thermal Response

The thermal response of the core for accidents and expected AOO conditions is given Chapter 15.

4.4.5.5 Analytical Methods

4.4.5.5.1 Fuel Cladding Integrity Safety Limit

The generation of the Minimum Critical Power Ratio (MCPR) limit requires a statistical analysis of the core near the limiting MCPR condition. The MCPR Fuel Cladding Integrity Safety Limit applies not only for core wide AOOs, but is also applied to the localized rod withdrawal error AOO. The safety limit MCPR is derived based on the criteria of Appendix 4B.

4.4.5.5.1.1 Statistical Model

The statistical analysis utilizes a model of the BWR core which simulates the process computer function. This code produces a critical power ratio (CPR) map of the core based on inputs of power distribution, flow and heat balance information. Details of the procedure are documented in Appendix IV of Reference 4.4-8. Random Monte Carlo selections of all operating parameters based on the uncertainty ranges of manufacturing tolerances, uncertainties in measurement of core operating parameters, calculational uncertainties, and statistical uncertainty associated with the critical power correlations are imposed upon the analytical representation of the core and the resulting bundle critical power ratios are calculated.

The minimum allowable critical power ratio is set to correspond to the criterion that 99.9% of the rods are expected to avoid boiling transition by interpolation among the means of the distributions formed by all the trials.

4.4.5.5.1.2 Bounding BWR Statistical Analysis

Statistical analyses have been performed which provide fuel cladding integrity safety limit MCPRs applicable to all GE fuel designs in the ABWR reload and initial core cycles. These safety limit MCPRs were derived based on the criteria in Appendix 4B. The results of the analyses show that at least 99.9% of the fuel rods in the core are expected to avoid boiling transition if the MCPR is equal to or greater than 1.06 for the initial core, and 1.07 for reload cores.

4.4.5.5.2 MCPR Operating Limit Calculational

A plant-unique MCPR operating limit is established to provide adequate assurance that the fuel cladding integrity safety limit for that plant is not exceeded for any moderate frequency AOO. This operating requirement is obtained by addition of the maximum vCPR value for the most limiting AOO (including any imposed adjustment factors) from conditions postulated to occur at the plant to the fuel cladding integrity safety limit.

4.4.5.5.3 Calculational Procedure for AOO Pressurization Events

Core-wide rapid pressurization events are analyzed using the system model (OYDN) documented in References 4.4-9 and 4.4-10. An updated version of OYDN using the advanced physics methods of Reference 4.4-11 is described in Reference 4.4-12. The events analyzed using OYDN are given in the response to Question 440.116 (Chapter 20).

The ABWR version of the OYDN code incorporates minor improvements to model features unique to the ABWR. The changes include an internal pump recirculation model and flow control system, and modifications to the upper plenum model to allow for subcooled liquid.

The time varying axial power shape (TVAPS) is a short time period phenomena that occurs during the control rod scram that terminates an AOO. The analytical procedures used to evaluate the AOOs account for TVAPS either in a bounding manner or explicitly, depending on the AOO and the fuel design.

The ODYN code has been qualified against actual plant data as described in Reference 4.4-12. Comparisons of the results calculated using both ODYN and the ABWR version of ODYN have been made. These comparisons indicate that the ABWR version of ODYN satisfactorily predicts ABWR transient behavior.

4.4.5.5.4 Calculational Procedure for AOO Slow Events

Core-wide non-pressurization AOOs are analyzed using the REDY transient model (References 4.4-13, 4.4-14 and 4.4-15). The events analyzed using REDY are given in the response to Question 440.116 (Chapter 20).

The ABWR version of the REDY code incorporates minor improvements to model features unique to the ABWR, and also improvements have been made to the physical and numerical models in several areas. The core hydraulic model includes a bypass region in addition to the active core region. A more detailed vessel pressure distribution model has been added. Additional changes are the capability to model three separate groupings of internal recirculation pumps, steam quenching in the vessel dome, an option to include momentum balance in the steamline, and the ability to simulate cold water injection into the lower plenum.

Comparisons of results calculated using the ABWR version of REDY have been compared against results calculated using the ABWR version of ODYN, which has been qualified as discussed in Subsection 4.4.5.5.3. The comparisons indicate that the ABWR version of REDY conservatively predicts results relative to ODYN, and therefore it is concluded that the ABWR version of REDY is a satisfactory predictor of ABWR transient behavior.

4.4.6 Testing and Verification

The testing and verification techniques to be used to assure that the planned thermal and hydraulic design characteristics of the core have been provided, and will remain within required limits throughout core lifetime, are discussed in Chapter 14.

4.4.7 COL License Information

4.4.7.1 Power/Flow Operating Map

The COL applicant will provide the specific power/flow operating map to be used at the plant to the USNRC for information (Subsection 4.4.3.3.1).

4.4.7.2 Thermal Limits

The COL applicant will provide the thermal limits for the core loading at the plant to the USNRC for information (Subsection 4.4.3.3.1).

4.4.8 References

4.4-1 “Core Flow Distribution in a Modern Boiling Water Reactor as Measured in Monticello”, NEDO-10299A, October 1976.

4.4-2 H.T. Kim and H. S. Smith, “Core Flow Distribution in a General Electric Boiling Water Reactor as Measured in Quad Cities Unit 1”, NEDO-10722A, August 1976.

4.4-3 “Brunswick Steam Electric Plant Unit 1 Safety Analysis Report for Plant Modifications to Eliminate Significant In-Core Vibrations”, NEDO-21215, March 1976.

4.4-4 R.C. Marinelli and D.E. Nelson, “Prediction of Pressure Drops During Forced Convection Boiling of Water”, ASME Trans., 70, 695-702, 1948.

4.4-5 C.J. Baroozy, “A Systematic Correlation for Two-Phase Pressure Drop”, Heat Transfer Conference (Los Angeles), AICLE, Preprint No. 37, 1966.

4.4-6 N. Zuber and J.A. Findlay, “Average Volumetric Concentration in Two-Phase Flow Systems”, Transactions of the ASME Journal of Heat Transfer, November 1965.

4.4-7 W.H. Jens and P.A. Lottes, “Analysis of Heat Transfer, Burnout, Pressure Drop and Density Data for High Pressure Water”, USAEC Report- 4627, 1972.

4.4-8 ″General Electric BWR Thermal Analysis Basis (GETAB): Data Correlation and Design Application”, NEDO-10958-A, January 1977.

4.4-9 “Qualification of the One-Dimensional Core Transient Model for BWR's”, NEDO-24154, Vol. 1 and 2, October 1978.

4.4-10 ″Qualification of the One-Dimensional Core Transient Model for BWR's”, NEDO-24154-P, Vol. 3, October 1978.

4.4-11 “Steady-State Nuclear Methods”, NEDO-30130-A, May 1985.

4.4-12 Letter from J.S. Charnley (GE) to C.O. Thomas (NRC), “Amendment 11 to GE LTR NEDE-24011-P-A”, February 27, 1985.

4.4-13 R.B. Linford, “Analytical Methods of Plant Transient Evaluations for General Electric Boiling Water Reactor”, NEDO-10802, February 1973.

4.4-14 R.B. Linford, “Analytical Methods of Plant Transient Evaluations for the GE BWR Amendment No. 1”, NEDO-10802-01, June 1975.

4.4-15 R.B. Linford, “Analytical Methods of Plant Transient Evaluations for the GE BWR Amendment No. 2”, NEDO-10802-02, June 1975.

4.4-16 NEDO-31960, “BWR Owners’ Group Long-term Stability Solutions Licensing Methodology”, June 1991.

Table 4.4-1 Typical Thermal–Hydraulic Design Characteristics of the Reactor Core

General Operating Conditions**	BWR/6 238-748	ABWR*
Reference design thermal output (MWt)	3579	3926
Power level for engineered safety features (MWt)	3730	4005
Steam flow rate, at 420°F final feedwater temperature (Mlb/hr)	15.40	16.84
Core coolant flow rate (Mlb/hr)	104.0	115.1
Feedwater flow rate (Mlb/hr)	15.367	16.807
System pressure, nominal in steam dome (psia)	1040	1040
System pressure, nominal core design (psia)	1055	1055
Coolant saturation temperature at core design pressure (°F)	551	551
Average power density (kW/L)	54.1	50.6
Core total heat transfer area (ft^2)	73,303	83,176
Design operating minimum critical power ratio (MCPR)	See Table 15.0-1	
Core inlet enthalpy at 420°F FFWT (Btu/lb)	527.7	527.6
Core inlet temperature at 420°F FFWT (°F)	533	533
Core maximum exit voids within assemblies (%)	79.0	75.1
Core average void fraction, active coolant	0.414	0.408
Active coolant flow area per assembly ($in.^2$)	15.164	15.678
Core average inlet velocity (ft/sec)	6.98	6.43
Maximum inlet velocity (ft/sec)	8.54	7.45
Total core pressure drop (psi)	26.4	24.4
Core support plate pressure drop (psi)	22.0	20.0
Average orifice pressure drop, central region (psi)	5.71	8.75
Average orifice pressure drop, peripheral region (psi)	18.68	17.69
Maximum channel pressure loading (psi)	15.40	10.9
Average-power assembly channel pressure loading (bottom) (psi)	14.1	9.5
Shroud support ring and lower shroud pressure loading	25.7	23.9
Upper shroud pressure loading (psi)	3.7	3.5

* Based on the core loading in Figure 4.3-1.

** English units are utilized in this table since the data obtained in the comparative BWR operating facility is in English units.

Table 4.4-2 Void Distribution for Analyzed Core

Core Average Value - 0.408
Maximum Exit Value - 0.751
Active Fuel Length - 3.71 m

	Node	Core Average (Average Node Value)	Maximum Channel (End of Node Value)
Bottom of Core	1	0	0
	2	0	0.011
	3	0.010	0.069
	4	0.045	0.162
	5	0.104	0.260
	6	0.174	0.346
	7	0.247	0.421
	8	0.315	0.481
	9	0.372	0.529
	10	0.419	0.567
	11	0.458	0.597
	12	0.489	0.623
	13	0.515	0.643
	14	0.536	0.660
	15	0.554	0.675
	16	0.570	0.688
	17	0.585	0.700
	18	0.599	0.712
	19	0.611	0.723
	20	0.623	0.733
	21	0.633	0.740
	22	0.641	0.746
	23	0.646	0.750
Top of Core	24	0.649	0.751

Table 4.4-3 Flow Quality Distribution for Analyzed Core

Core Average Value - 0.145
Maximum Exit Value - 0.258
Active Fuel Length - 3.71 m

	Node	Core Average (Average Node Value)	Maximum Channel (End of Node Value)
Bottom of Core	1	0	0
	2	0	0
	3	0	0.002
	4	0	0.008
	5	0	0.019
	6	0.005	0.033
	7	0.016	0.051
	8	0.028	0.069
	9	0.040	0.087
	10	0.052	0.105
	11	0.064	0.122
	12	0.074	0.137
	13	0.085	0.152
	14	0.094	0.165
	15	0.103	0.177
	16	0.110	0.189
	17	0.118	0.200
	18	0.126	0.212
	19	0.134	0.224
	20	0.142	0.235
	21	0.148	0.245
	22	0.154	0.252
	23	0.158	0.257
Top of Core	24	0.159	0.258

Table 4.4-4 Axial Power Distribution Used to Generate Void and Quality Distributions for Analyzed 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 Reactor Coolant System Geometric Data

		Flow Path Length (m)	Height and Liquid Level (m)	Elevation of Bottom of Each Volume* (m)	Average Flow Areas (m^2)
A.	Lower Plenum	4.65	4.65 4.65	0.0	19.5
B.	Core	4.36	4.36 4.36	4.65	16.1† includes bypass
C.	Upper Plenum and Separators	3.64	3.64 3.64	9.00	16.5
D.	Dome (Above Normal Water Level)	7.80	7.80 0	13.2	30.2
E.	Downcomer Area	12.6	12.6 12.6	1.84	16.2

* Reference point is vessel bottom zero.

† For the core loading given in Figure 4.3-1.

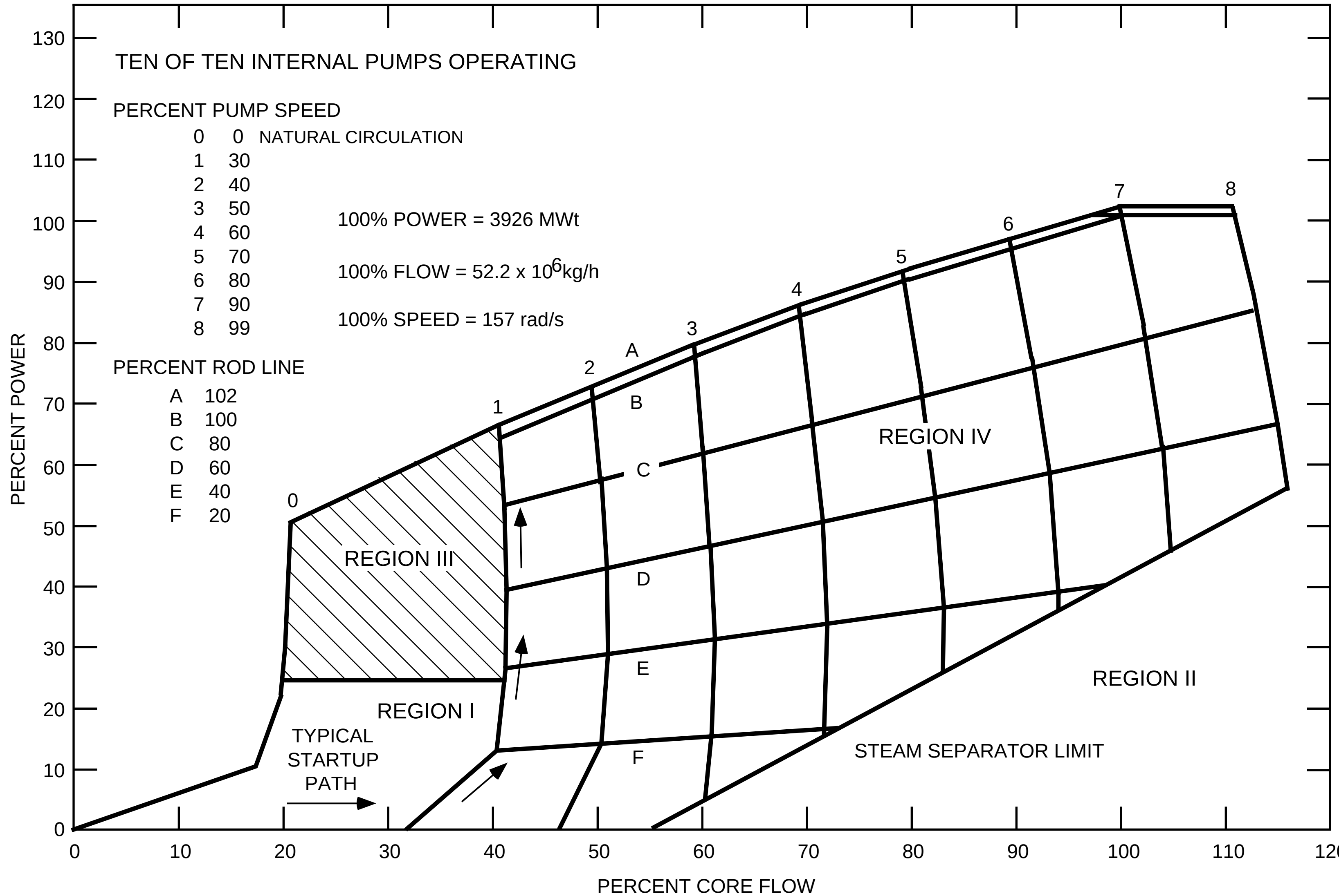


Figure 4.4-1 Power-Flow Operating Map Used for System Response Study

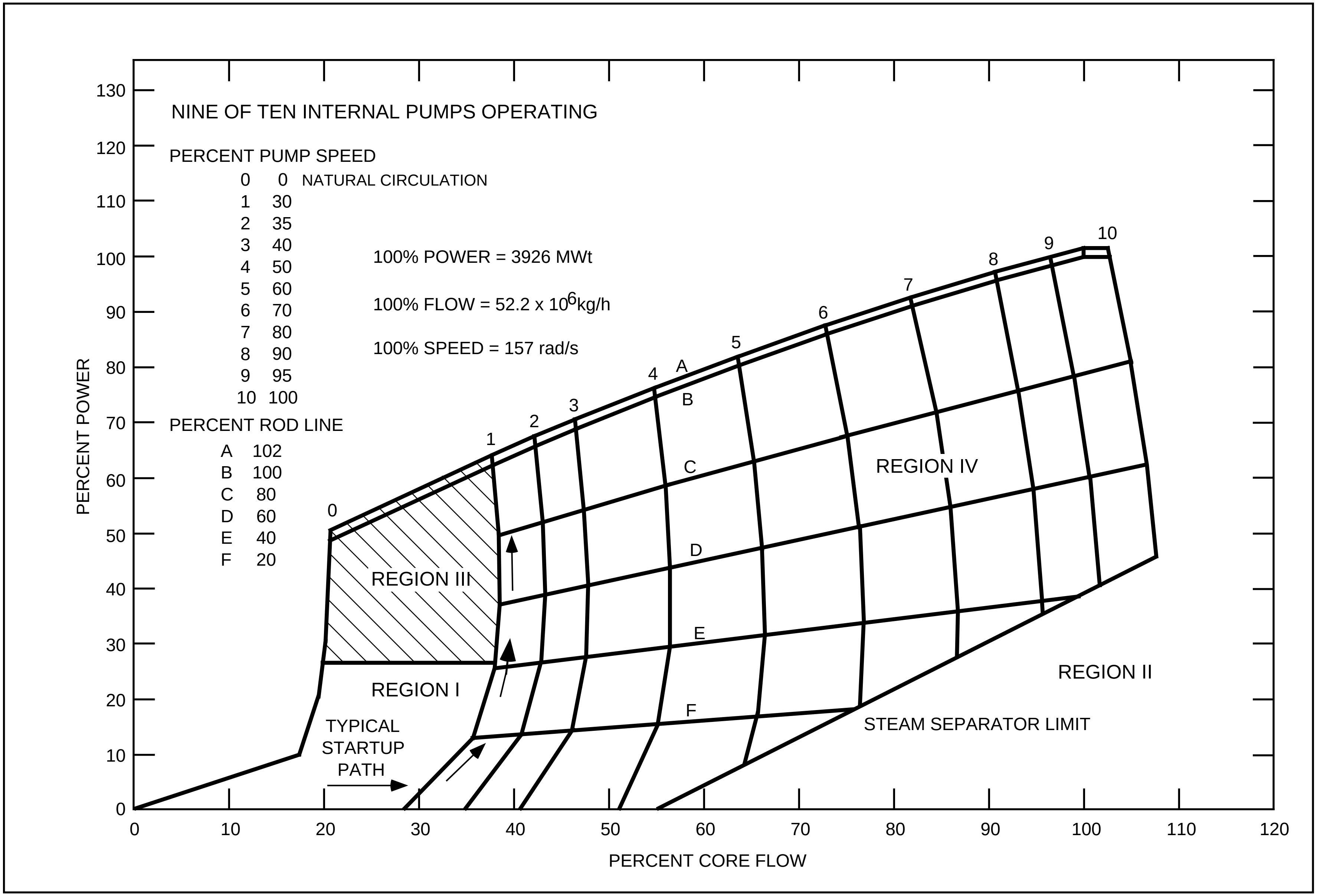


Figure 4.4-2 Power-Flow Operating Map Used for System Response Study (9 RIPs Operation)

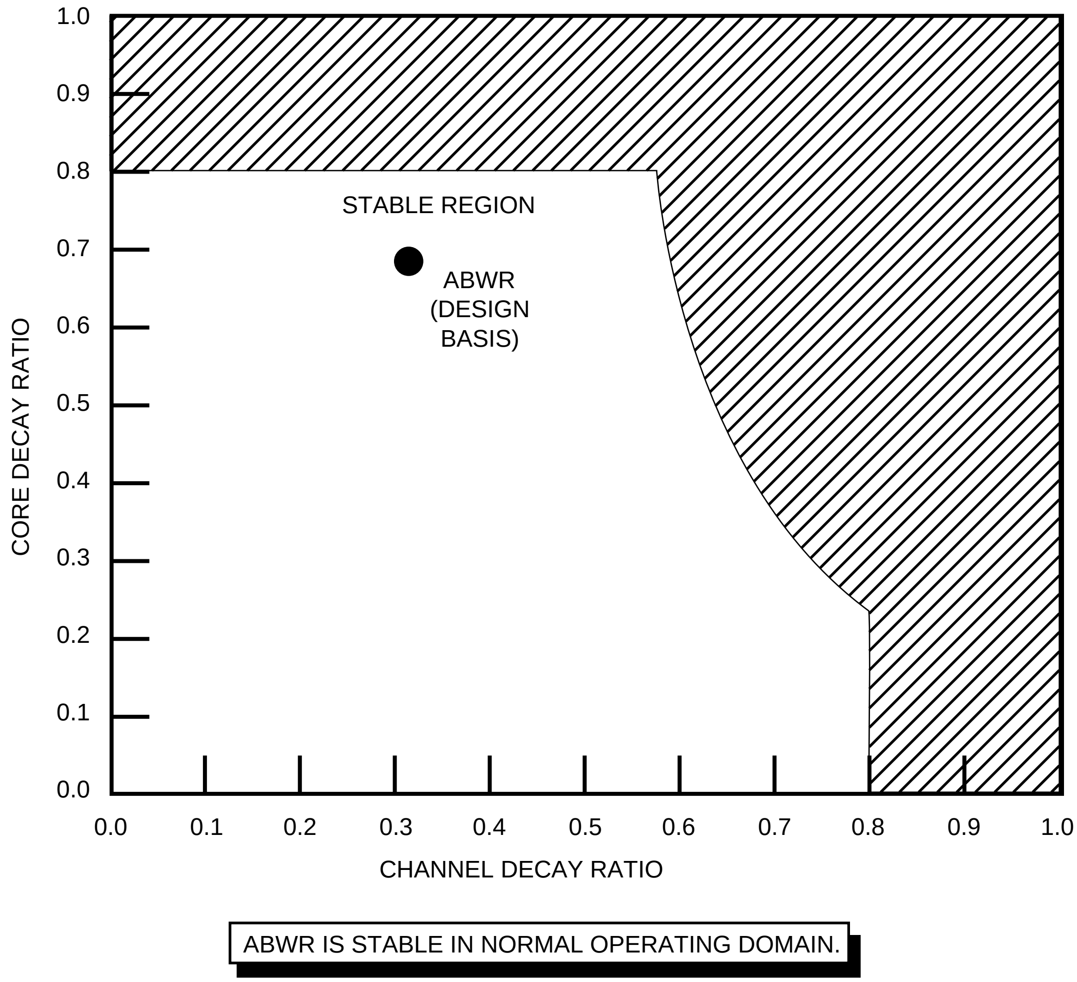


Figure 4.4-3 ABWR Stability

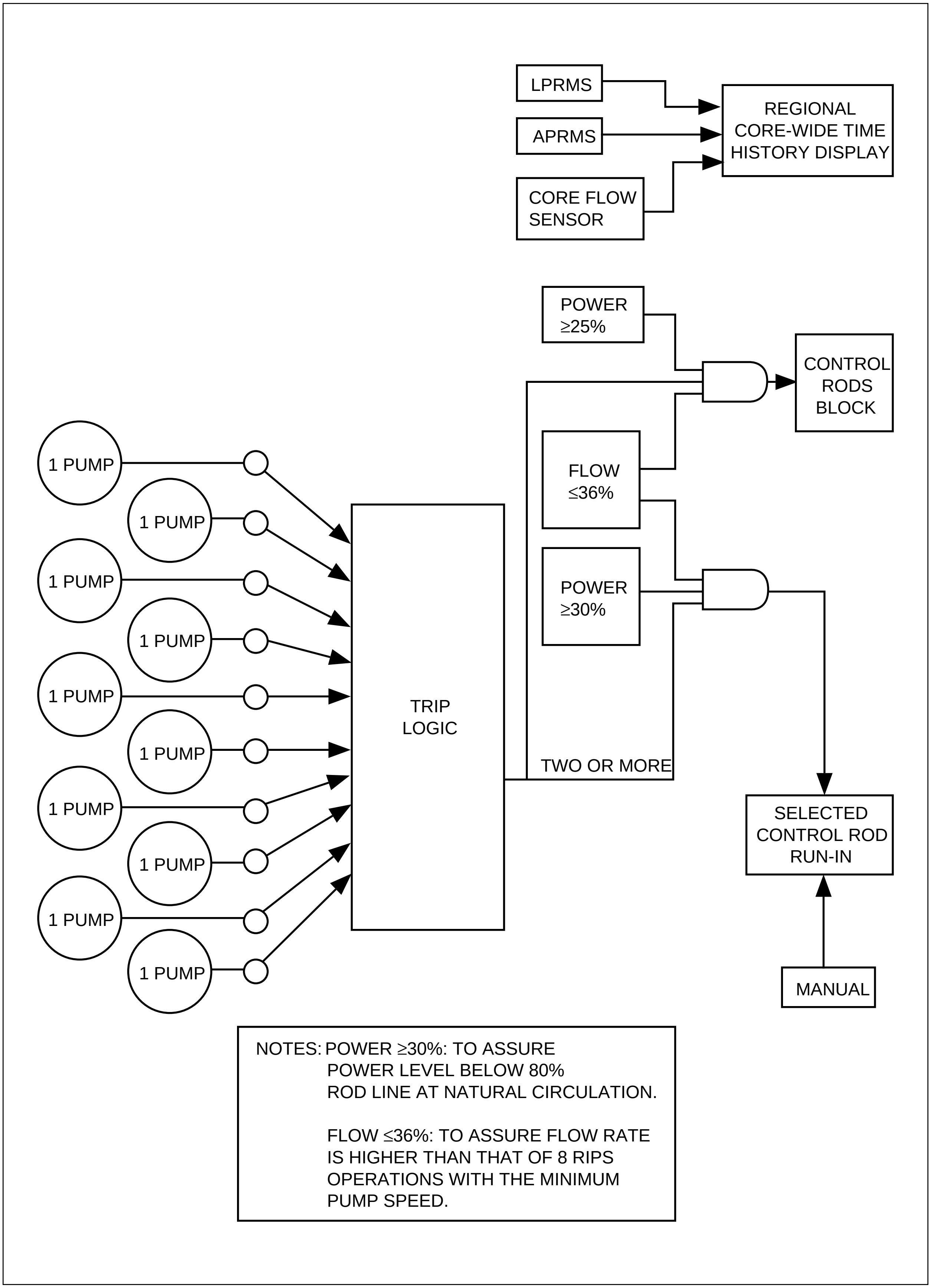


Figure 4.4-4 Stability Controls and Protection Logic

4.5 Reactor Materials

4.5.1 Control Rod Drive System Structural Materials

4.5.1.1 Material Specifications

(1) Material List

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

The properties of the materials selected for the CRD mechanism shall be equivalent to those given in Appendix I to Section III of the ASME Code or Parts A and B of Section II of the ASME Code, or are included in Regulatory Guide 1.85, except that cold-worked austenitic stainless steels shall be controlled by limiting hardness, bend radius, or the amount of induced strain.

(a) Spool Piece Assembly

Spool Piece Housing	ASME 182 Grade F304L
Seal Housing	ASME 182 Grade F304L
Drive Shaft	ASME 479 Grade XM-19 (Hardsurfaced with Colmonoy No. 6)
Ball Bearings	ASTM A756 Type 440C
Gland Packing Spring	Inconel X-750

(b) Ball Spindle

Ball Screw Shaft	ASTM A-564 TP630 (17--4PH) Condition H-1100
Ball Nut	ASTM A-564 TP630 (17-4PH) Condition H-1100
Balls	ASTM A756 Type 440C
Guide Roller	Stellite No. 3
Guide Roller Pin	Haynes Alloy No. 25
Spindle Head Bolt	Stellite No. 6B

	Spindle Head Bushing	Stellite No. 12
	Separation Spring	Inconel X-750
	Separation Magnet	Alnico No. 5
(c)	Buffer Mechanism	
	Buffer Disk Spring	Inconel X-750
	Buffer Sleeve	316L (Hardsurfaced with Colmonoy No. 6)*
	Guide Roller	Stellite No. 3
	Guide Roller Pin	Haynes Alloy No. 25
	Stop Piston	316L (Hardsurfaced with Stellite No. 6)*

* The base material shall be qualified to assure that it is free from sensitization.

(d)	Hollow Piston	
	Piston Tube	XM-19
	Piston Head	316L (Hardsurfaced with Stellite No. 3)*
	Latch	Inconel X-750
	Latch Spring	Inconel X-750
	Bayonet Coupling	Inconel X-750

* The base material shall be qualified to assure that it is free from sensitization.

(e)	Guide Tube	
	Guide Tube	316L

(f) Outer Tube Assembly

Outer Tube	XM-19
Middle Flange	ASME SA182 Grade F304LC

(g) Miscellaneous Parts

Ball for Check Valve	Haynes Stellite No.3
O-Ring Seal (Between CRD Housing and CRD)	321SS Coated with a qualified material
CRD Installation Bolts	ASME SA193 Grade B7

(2) Special Materials

The bayonet coupling, latch and latch spring, separation spring, and gland packing spring are fabricated from Alloy X-750 in the high temperature (1093°C) annealed condition, and aged 20 hours at 704°C to produce a tensile strength of 1137.7 MPa minimum, yield of 724 MPa minimum, and elongation of 20% minimum. The ball screw shaft and ballnut are ASTM A-564, TP 630 (17-4PH) (or its equivalent) in condition H-1100 (aged 4 hours at 593°C), with a tensile strength of 265.3 MPa minimum, yield of 792.92 MPa minimum, and elongation of 15% minimum.

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

All materials for use in this system shall be selected for their compatibility with the reactor coolant as described in Articles NB-2160 and NB-3120 of the ASME Code.

All materials, except SA479 or SA249 Grade XM-19, have been successfully used for the past 15 to 20 years in similar drive mechanisms. Extensive laboratory tests have demonstrated that ASME SA479 or SA249 Grade XM-19 are suitable materials and that they are resistant to stress corrosion in a BWR environment.

No cold-worked austenitic stainless steels except those with controlled hardness or strain are employed in the Control Rod Drive (CRD) System. During fabrication and installation, special controls are used to limit the induced strain, and the bend radii are kept above a minimum value.

4.5.1.2 Austenitic Stainless Steel Components

(1) Processes, Inspections and Tests

All austenitic stainless steels are used in the solution heat treated condition. In all welded components which are exposed to service temperature exceeding 93°C, the carbon content is limited not to exceed 0.020%. On qualification, there is a special process employed which subjects selected 300 Series stainless steel components to temperatures in the sensitization range. The drive shaft, buffer sleeve, piston head and buffer are hard surfaced with Colmonoy 6 (or its equivalent). Colmonoy (or its equivalent) hard-surfaced components have performed successfully for the past 15 to 20 years in drive mechanisms. It is normal practice to remove some CRDs at each refueling outage. At this time, the Colmonoy (or its equivalent) hard-surfaced parts are accessible for visual examination. This inspection program is adequate to detect any incipient defects before they could become serious enough to cause operating problems (see Subsection 4.5.3.1 for COL license information). The degree of conformance to Regulatory Guide 1.44 is presented in Subsection 4.5.2.4.

(2) Control of Delta Ferrite Content

Discussion of this subject and the degree of conformance to Regulatory Guide 1.31 is presented in Subsection 4.5.2.4.

4.5.1.3 Other Materials

These are presented in Subsection 4.5.1.1(2).

4.5.1.4 Cleaning and Cleanliness Control

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

(1) Any processing which increases part temperature above 93°C

(2) Assembly which results in decrease of accessiblity for cleaning

(3) Release of parts for shipment

The specification for packaging and shipping the control rod drive provides the following:

The CRD is rinsed in hot deionized water and dried in preparation for shipment. The ends of the drive are then covered with a vapor-tight barrier with dessicant. Packaging is designed to protect the drive and prevent damage to the vapor barrier. Audits have indicated satisfactory protection.

Semiannual examination of 10% of the units humidity indicators is required to verify that the units are dry and in satisfactory condition. This inspection shall be performed with a GE-Engineering designated representative present. The position indicator probes are not subject to this inspection.

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

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

4.5.2 Reactor Internal Materials

4.5.2.1 Material Specifications

Materials Used for the Core Support Structure:

- **Shroud Support—**Nickel-Chrome-Iron-Alloy, ASME SB166 or SB168
- **Shroud, Core Plate, and Grid—**ASME SA240, SA182, SA479, SA312, SA249, or SA213 (all Type 304L or 316L)
- **Peripheral Fuel Supports—**ASME SA312 Grade Type-304L or 316L
- **Core Plate and Top Guide Studs, Nuts, and Sleeves—**ASME SA-479 (Type 304, 316, or XM-19) (all parts); or SA-193 Grade B8 Type 304 (studs); or SA-194 Grade 8 (Type 304) (nuts); or SA-479 (Type 304L or 316L), SA-182 (Grade F304L or F316L), SA-213 (Type 304L, 316 or 316L), SA-249 (Type 304L, 316, or 316L) (sleeves)
- **Control Rod Drive Housing—**ASME SA-312 Grade TP304L or 316L SA-182 Grade F304L or F316L, and ASME SA-351 Type CF3 (Type 304L) or Type CF3M (Type 316L)
- **Control Rod Guide Tube—**ASME SA-351 Type CF3 or CF3M, or SA-358, SA-312, or SA-249 (Type 304L or 316L)
- **Orificed Fuel Support—**ASME SA-351 Type CF3 (Type 304L) or CF3M (Type 316L)

Materials Employed in Shroud Head and Separator Assembly and Steam Dryer Assembly:

All materials are 304L or 316L stainless steel:

- **Plate, Sheet—**ASTM A240 Type 304L or 316L and Strip
- **Forgings—**ASTM A182 Grade 304L or 316L
- **Bars—**ASTM A276 Type 316L or 304L
- **Pipe—**ASTM A312 Grade TP-304L or 316L
- **Tube—**ASTM A269 Grade TP-304L or 316L
- **Castings—**ASTM A351 Grade CF8, CF8M

All core support structures are fabricated from ASME specified materials, and designed in accordance with requirements of ASME Code Section III, Subsection NG. The other reactor internals are noncoded, and they are fabricated from ASTM or ASME specification materials.

4.5.2.2 Controls on Welding

Core support structures are fabricated in accordance with requirements of ASME Code Section III, Subsection NG-4000, and the examination and acceptance criteria shown in NG-5000. Other internals are not required to meet ASME Code requirements. ASME Section IX B&PV Code requirements are followed in fabrication of core support structures.

4.5.2.3 Non-Destructive Examination of Wrought Seamless Tubular Products

Wrought seamless tubular products for CRD housings and peripheral fuel supports are supplied in accordance with ASME Section III, Class CS, which requires examination of the tubular products by radiographic and/or ultrasonic methods according to Paragraph NG-2550. The examination will satisfy the requirements of NG-5000.

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

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

Significantly cold-worked stainless steels are not used in the reactor internals except for vanes in the steam dryers; cold work is controlled by applying limits on hardness, bend radii and surface finished on ground surfaces. Furnace sensitized material are not allowed. Electroslag welding is not applied for structural welds. The delta ferrite content for weld materials used in welding austenitic stainless steel assemblies is verified on undiluted weld deposits for each heat or lot of filler metal and electrodes. The delta ferrite content is defined for weld materials as 5.0 Ferrite Number (FN) minimum, 8.0FN average and 20FN maximum. This ferrite content is considered adequate to prevent any micro-fissuring (Hot Cracking) in austenitic stainless steel welds. This procedure complies with the requirements of Regulatory Guide 1.31.

The limitation placed upon the delta ferrite in austenitic stainless steel castings is 8FN (ferrite number) minimum and a maximum value of 20FN. The maximum limit is used for those castings designed for a 60 year life such as the fuel support pieces, in order to limit the effects of thermal aging degradation. Short in-reactor lifetime components such as the fuel tie plates do not require such a limit.

Proper solution annealing of the 300 series austenitic stainless steel is verified by testing per ASTM-A262, "Recommended Practices for Detecting Susceptibility to Intergranular Attack in Stainless Steels." Welding of austenitic stainless steel parts is performed in accordance with Section IX (Welding and Brazing Qualification) and Section II Part C (Welding Rod Electrode and Filler Metals) of the ASME B&PV. Welded austenitic stainless steel assemblies require solution annealing to minimize the possibility of the sensitizing. However, welded assemblies are dispensed from this requirement when there is documentation that welds are not subject to significant sustained loads and assemblies have been free of service failure. Other reasons, in line with Regulatory Guide 1.44, for dispensing with the solution annealing are that (1) assemblies are exposed to reactor coolant during normal operation service which is below 93.3°C temperature or (2) assemblies are of material of low carbon content (less than 0.020%). These controls are employed in order to comply with the intent of Regulatory Guide 1.44.

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

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

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

4.5.2.5 Other Materials

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

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

(1) SA479 Type XM-19 stainless steel

(2) SB166, 167, and 168, Nickel-Chrome-Iron (Alloy 600)

(3) SA637 Grade 688 Alloy X-750

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

Alloy X-750 components are fabricated in the annealed or equalized condition and aged when required. Where maximum resistance to stress corrosion is required, the material is used in the high temperature (2000°C) annealed plus single aged condition.

Stellite 6 (or its equivalent) hard surfacing is applied to the austenitic stainless steel HPCF couplings using the gas tungsten arc welding or plasma arc surfacing processes.

All materials used for reactor internals shall be selected for their compatibility with the reactor coolant as shown in ASME Code Section III, NG-2160 and NG-3120. The fabrication and cleaning controls will preclude contamination of nickel-based alloys by chloride ions, fluoride ions, sulfur, or lead.

All materials, except SA479 Grade XM-19, have been successfully used for the past 15 to 20 years in BWR applications. Extensive laboratory tests have demonstrated that XM-19 is a suitable material and that it is resistant to stress corrosion in a BWR environment.

4.5.3 COL License Information

4.5.3.1 CRD Inspection Program

The CRD inspection program shall include provisions to detect incipient defects before they become serious enough to cause operating problems. The CRD nozzle and bolting are included in the inservice inspection program (Table 5.2-8, System Number B11/B12). CRD bolting is accessible for inservice examinations during normally scheduled CRD maintenance (Subsection 4.5.1.2(1)).

4.6 Functional Design of Reactivity Control System

The Reactivity Control System consists of (1) control rods and Control Rod Drive (CRD) System, (2) supplementary reactivity control in the form of a gadolinia-urania fuel rods (Section 4.3), and (3) the Standby Liquid Control System (Subsection 9.3.5).

Evaluations of the reactivity control systems against the applicable General Design Criteria (GDC) are contained in the following subsections:

GDC	Subsection
23	3.1.2.3.4
25	3.1.2.3.6
26	3.1.2.3.7
27	3.1.2.3.8
28	3.1.2.3.9
29	3.1.2.3.10

4.6.1 Information for Control Rod Drive System

4.6.1.1 Design Bases

4.6.1.1.1 Safety Design Bases

The CRD System shall meet the following safety design bases:

(1) The design shall provide for rapid control rod insertion (scram) so that no fuel damage results from any moderately frequent event (Chapter 15).

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

(3) Each positioning device shall be capable of holding the control rod in position and preventing it from inadvertently withdrawing outward during any non-accident, accident, post-accident and seismic condition.

(4) Each positioning device shall be capable of detecting the separation of the control rod from the drive mechanism to prevent a rod drop accident.

(5) Each positioning device shall provide a means to prevent or limit the rate of control rod ejection from the core due to a break in the drive mechanism pressure boundary. This is to prevent fuel damage resulting from rapid insertion of reactivity.

4.6.1.1.2 Power Generation Design Basis

The CRD System design shall meet the following power generation design bases:

(1) The design shall provide for controlling changes in core reactivity by positioning neutron-absorbing control rods within the core.

(2) The design shall provide for movement and positioning of control rods in increments to enable optimized power control and core power shaping.

4.6.1.2 Description

The CRD System is composed of three major elements:

(1) Electro-hydraulic fine motion control rod drive (FMCRD) mechanisms,

(2) Hydraulic control units (HCU), and

(3) Control rod drive hydraulic subsystem (CRDHS).

The FMCRDs provide electric-motor-driven positioning for normal insertion and withdrawal of the control rods and hydraulic-powered rapid insertion (scram) of control rods during abnormal operating conditions. There are a total of 205 FMCRDs mounted in housings welded into the reactor vessel bottom head.

The hydraulic power required for scram is provided by high pressure water stored in 103 individual HCUs. Each HCU contains a nitrogen-water accumulator charged to high pressure and the necessary valves and components to scram two FMCRDs. Additionally, during normal operation, the HCUs provide a flow path for purge water to the associated FMCRDs.

The CRDHS supplies clean, demineralized water which is regulated and distributed to provide charging of the HCU scram accumulators and purge water flow to the FMCRDs during normal operation. The CRDHS is also the source of pressurized water for purging the Reactor Internal Pumps (RIPs) and the Reactor Water Cleanup(CUW) System pumps.

The CRD System performs the following functions:

(1) Controls changes in core reactivity by positioning neutron-absorbing control rods within the core in response to control signals from the Rod Control and Information System (RCIS).

(2) Provides movement and positioning of control rods in increments to enable optimized power control and core power shape in response to control signals from the RCIS.

(3) Provides the ability to position large groups of rods simultaneously in response to control signals from the RCIS.

(4) Provides rapid control rod insertion (scram) in response to manual or automatic signals from the Reactor Protection System (RPS) so that no fuel damage results from any plant transient.

(5) In conjunction with the RCIS, provides automatic electric motor-driven insertion of the control rods simultaneously with hydraulic scram initiation. This provides an additional, diverse means of fully inserting a control rod.

(6) Supplies rod status and rod position data for rod pattern control, performance monitoring, operator display and scram time testing by the RCIS.

(7) In conjunction with the RCIS, prevents undesirable rod pattern or rod motions by imposing rod motion blocks in order to protect the fuel.

(8) In conjunction with the RCIS, prevents the rod drop accident by detecting rod separation and imposing rod motion block.

(9) Provides alternate rod insertion (ARI), an alternate means of actuating hydraulic scram, should an anticipated transient without scram (ATWS) occur.

(10) In conjunction with the RCIS, provides for selected control rod run-in (SCRRI) for core thermal-hydraulic stability control.

(11) Prevents rod ejection by means of a passive brake mechanism for the FMCRD motor and a scram line inlet check valve.

(12) Supplies purge water for the RIPs and CUW pumps.

The design bases and further discussion of both the RCIS and RPS, and their control interfaces with the CRD System, are presented in Chapter 7.

4.6.1.2.1 Fine Motion Control Rod Drive Mechanism

The FMCRD used for positioning the control rod in the reactor core is a mechanical/hydraulic actuated mechanism (Figures 4.6-1, 4.6-2 and 4.6-3). An electric motor-driven ball-nut and spindle assembly is capable of positioning the drive at a minimum of 18.3 mm increments. Hydraulic pressure is used for scrams. The FMCRD penetrates the bottom head of the reactor pressure vessel. The FMCRD does not interfere with refueling and is operative even when the head is removed from the reactor vessel.

The fine motion capability is achieved with a ball-nut and spindle arrangement driven by an electric motor. The ball-nut is keyed to the guide tube (roller key) to prevent its rotation and traverses axially as the spindle rotates. A hollow piston rests on the ball-nut and upward motion of the ball-nut drives this piston and the control rod into the core. The weight of the control rod keeps the hollow piston and ball-nut in contact during withdrawal.

A single HCU powers the scram action of two FMCRDs. Upon scram valve initiation, high pressure nitrogen from the HCU raises the piston within the accumulator, forcing water through the scram piping. This water is directed to each FMCRD connected to the HCU. Inside each FMCRD, high-pressure water lifts the hollow piston off the ball-nut and drives the control rod into the core. A spring washer buffer assembly stops the hollow piston at the end of its stroke. Departure from the ball-nut releases spring-loaded latches in the hollow piston that engage slots in the guide tube. These latches support the control rod in the inserted position. The control rod cannot be withdrawn until the ball-nut is driven up and engaged with the hollow piston. Stationary fingers on the ball-nut then cam the latches out of the slots and hold them in the retracted position. A scram action is complete when every FMCRD has reached their fully inserted position.

The use of the FMCRD mechanisms in the CRD System provides several features which enhance both the system reliability and plant operations. Some of these features are listed and discussed briefly as follows:

(1) Diverse Means of Rod Insertion

The FMCRDs can be inserted either hydraulically or electrically. In response to a scram signal, the FMCRD is inserted hydraulically via the stored energy in the scram accumulators. A signal is also given simultaneously to insert the FMCRD electrically via its motor drive. This diversity provides a high degree of assurance of rod insertion on demand.

(2) Absence of FMCRD Piston Seals

The FMCRD pistons have no seals that require periodic drive removal for maintenance; the FMCRD internals can remain in place for their full design life. Only a sample of two or three complete FMCRDs are planned to be removed for inspection each refueling outage to document drive condition. This is an order of magnitude reduction compared to previous BWR product lines in which 20 to 30 complete drives are removed for piston seal replacement each refueling outage.

(3) FMCRD Discharge

The water which scrams the control rod discharges into the reactor vessel and does not require a scram discharge volume, thus eliminating a potential source for common mode scram failure.

(4) Plant Maneuverability

The fine motion capability of the FMCRD allows rod pattern optimization in response to fuel burnup or load-following demands. Such a feature complements the ability to load follow with core flow rate adjustments. Combining this with Reactor Recirculation System flow control, further improves plant maneuverability.

(5) Plant Automation

The relatively simple logic of the FMCRD permits plant automation. This feature is utilized for automatic reactor startup and shutdown and for automatic load following.

(6) Reactor Startup Time

The FMCRDs can be moved in large groups. Movements of large groups of control rods (called gangs) are utilized to reduce the time for reactor startup.

(7) Rod Drop Accident Prevention

The control rod separation detection feature of the FMCRD virtually eliminates the possibility of a Rod Drop Accident (RDA) by preventing rod withdrawal when control rod separation is detected. Additionally, movement of rods in large groups during reactor startup greatly reduces the maximum relative rod worth to levels lower than current rod pattern controls. Rod pattern controls are retained in order to verify proper automatic rod movements and to mitigate the consequences of a rod withdrawal error

The drives are readily accessible for inspection and servicing. The bottom location makes maximum utilization of the water in the reactor as a neutron shield and gives the least possible neutron exposure to the drive components. Using water from the condensate treatment system and / or condensate storage tanks as the operating fluid eliminates the need for special hydraulic fluid.

4.6.1.2.2 FMCRD Components

Figure 4.6-1 provides a simplified schematic of the FMCRD operating principles. Figure 4.6-2 illustrates the drive in more detail.

The basic elements of the FMCRD are as follows:

(1) Components of the FMCRD required for electrical rod positioning or fine motion control (including the motor, brake release, associated connector, ball screw shaft, ball-nut and hollow piston).

(2) Components of the FMCRD required for hydraulic scram (including hollow piston and buffer).

(3) Components of the FMCRD required for pressure integrity (including the middle flange, installation bolts and spool piece).

(4) Rod position indication (position synchronizing signal generators).

(5) Reed position switches for scram surveillance.

(6) Control rod separation detection devices (dual Class 1E CRD separation switches).

(7) Bayonet coupling between the drive and control rod.

(8) Brake mechanism to prevent rod ejection in the event of a break in the FMCRD primary pressure boundary, and ball check valve to prevent rod ejection in the event of a failure of the scram insert line.

(9) Integral internal blowout support (to prevent CRD blowout).

(10) FMCRD seal leak detection system.

These features and functions of the FMCRD are described below.

4.6.1.2.2.1 Components for Fine Motion Control

The fine motion capability is achieved with a ball-nut and spindle arrangement driven by an electric stepping motor. The ball-nut is keyed to the guide tube (roller key) to prevent its rotation, and it traverses axially as the spindle rotates. A hollow piston rests on the ball-nut and upward motion of the ball-nut drives the control rod into the core. The weight of the control rod keeps the hollow piston and ball-nut in contact during withdrawal.

The drive motor, located outside the pressure boundary, is connected to the spindle by a drive shaft. The drive shaft penetrates the pressure boundary and is sealed by conventional packings. A splined coupling connects the drive shaft to the spindle. The lower half of the splined coupling is keyed to the drive shaft and the upper half keyed to the spindle. The tapered end of the drive shaft fits into a conical seat on the end of the spindle to keep the two axially aligned. The entire weight of the control rod and drive internals is carried by a drive shaft thrust bearing located outside the pressure boundary.

The axially moving parts are centered and guided by radial rollers. The ball-nut and bottom of the hollow piston include radial rollers bearing against the guide tube. Radially adjustable rollers at both ends of the labyrinth seal keep the hollow piston precisely centered in this region.

The top of the rotating spindle is supported against the inside of the hollow piston by a stationary guide. A hardened bushing provides the circumferential bearing between the rotating spindle and stationary guide. Rollers of the guide run in axial grooves in the hollow piston to prevent the guide from rotating with the spindle.

4.6.1.2.2.2 Components for Scram

The scram action is initiated by the HCU. High-pressure water lifts the hollow piston off the ball-nut and drives the control rod into the core. A spring washer buffer assembly stops the hollow piston at the end of its stroke. Departure from the ball-nut releases spring-loaded latches in the hollow piston that engage slots in the guide tube. These latches support the control rod in the inserted position.

The control rod cannot be withdrawn until the ball-nut is driven up and engaged with the hollow piston. Stationary fingers on the ball-nut cam the latches in the hollow piston out of the slots in the guide tube and hold them in the retracted position when the ball-nut and hollow piston are re-engaged.

Re-engagement of the ball-nut with the hollow piston following scram is automatic. Simultaneous with the initiation of the hydraulic scram, each FMCRD motor is signaled to start in order to cause movement of the ball-nut upward until it is in contact with the hollow piston. This action completes the rod full-in insertion and leaves the drives in a condition ready for restarting the reactor. With the latches in the hollow piston retracted, the permanent magnets in the stepping motor provide the holding torque to maintain the control rods fully inserted in the core. When the motor and brake are de-energized, the passive holding torque from the brake keeps the rods fully inserted.

The automatic run-in of the ball-nut, using the electric motor drive, following the hydraulic scram provides a diverse means of rod insertion as a backup to the accumulator scram.

The components for scram that are classified as safety-related within the drive are the hollow piston, latches, guide tube and brake.

4.6.1.2.2.3 FMCRD Pressure Boundary

The CRD housing (attached to the RPV) and the CRD middle flange and lower housing (spool piece) which enclose the lower part of the drive are a part of the reactor pressure boundary (Figure 4.6-1). The middle housing is attached to the CRD housing by four threaded bolts. The spool piece is, in turn, held to the middle housing and secured to the CRD housing by a separate set of eight main mounting bolts which become a part of the reactor pressure boundary. This arrangement permits removing the lower housing, drive shaft and seal assembly without disturbing the rest of the drive. Removing the lower housing transfers the weight of the driveline from the drive shaft to the seat in the middle housing. Both the spindle and drive shaft are locked to prevent rotation while the two are separated.

The part of the drive inserted into the CRD housing is contained within the outer tube. The outer tube is the drive hydraulic scram pressure boundary, eliminating the need for designing the CRD housing for the scram pressure. The outer tube is welded to the middle flange at the bottom and is attached at the top with the CRD blowout support, which bears against the CRD housing. The blowout support and outer tube are attached by slip-type connection that accounts for any slight variation in length between the drive and the drive housing.

Purge water continually flows through the drive. The water enters through the ball check valve in the middle housing and flows around the hollow piston into the reactor. Conventional packing seals the drive shaft and O-rings seal the lower housing. A labyrinth seal near the top of the drive restricts the flow into the reactor. During a scram, the labyrinth seals the high-pressure scram water from the reactor vessel without adversely affecting the movement of the hollow piston.

4.6.1.2.2.4 Rod Position Indication

Control rod position indication is provided by the FMCRDs to the control system by a position detection system, which consists of position detectors and position signal converters.

Each FMCRD provides two position detectors, one for each control system channel, in the form of synchronizing signal generators directly coupled to the stepping motor shaft through gearing. The output signals from these generators are analog. The analog signals are converted to digital signals by position signal converters. This configuration provides continuous detection of rod position during normal operation.

4.6.1.2.2.5 Scram Position Indication

Scram position indication is provided by a series of magnetic reed switches to allow for measurement of adequate drive performance during scram. The magnetic switches are located at intermediate intervals over 60% of the drive stroke. They are mounted in a probe exterior to the drive housing. A magnet in the hollow piston trips each reed switch in turn as it passes by.

As the bottom of the hollow piston contacts and enters the buffer, a magnet is lifted which operates a reed switch, indicating scram completion. This continuous full-in indicating switch is shown conceptually in Figure 4.6-3. It provides indication whenever the drive is at the full-in latched position or above.

4.6.1.2.2.6 Control Rod Separation Detection

Two redundant and separate Class 1E switches are provided to detect the separation of the hollow piston from the ball-nut. This means two sets of reed switches physically separated from one another with their cabling run through separate conduits. The separation switch is classified Class 1E, because its function detects a detached control rod and causes a rod block, thereby preventing a rod drop accident. Actuation of either switch also initiates an alarm in the control room

The principle of operation of the control rod separation mechanism is illustrated in Figure 4.6-4. During normal operation, the weight of the control rod and hollow piston resting on the ball-nut causes the spindle assembly to compress a spring on which the lower half of the splined coupling between the drive shaft and spindle assembly rests (the lower half of the splined coupling is also known as the “weighing table”). When the hollow piston separates from the ball-nut, or when the control rod separates from the hollow piston, the spring is unloaded and pushes the weighing table and spindle assembly upward. This action causes a magnet in the weighing table to operate the Class 1E reed switches located in a probe outside the lower housing.

4.6.1.2.2.7 Bayonet Couplings

There are two bayonet couplings associated with the FMCRD. The first is at the FMCRD / control rod guide tube / housing interface as illustrated in Figure 4.6-7. This bayonet locks the FMCRD and the base of the control rod guide tube to the CRD housing and functions to retain the control rod guide tube during normal operation and dynamic loading events. The bayonet also holds the FMCRD against ejection in the event of a hypothetical failure of the CRD housing weld. The locating pin on the core plate that engages the flange of the control rod guide tube and the bolt pattern on the FMCRD / housing flange assure proper orientation between the control rod guide tube and FMCRD to assure that the bayonet is properly engaged.

The second bayonet coupling is located between the control rod and FMCRD, as shown on Figure 4.6-5. The coupling spud at the top end of the FMCRD hollow piston engages and locks into a mating socket at the base of the control rod. The coupling requires a 45° rotation for engaging or disengaging. Once locked, the drive and rod form an integral unit that can only be unlocked manually by specific procedures before the components can be separated.

The FMCRD design allows the coupling integrity of this second bayonet to be checked by driving the ball-nut down into an overtravel-out position. After the weighing spring has raised the spindle to the limit of its travel, further rotation of the spindle in the withdraw direction will drive the ball-nut down away from the hollow piston (assuming the coupling is engaged). Piston movement, if any, can then be detected by a reed switch at the overtravel position. If the hollow piston and control rod are properly coupled the overtravel reed switch will not be activated, thus confirming the coupling integrity. If the hollow piston is uncoupled from the control rod the piston will follow the ball-nut to the overtravel position. The overtravel reed switch will be then be actuated by a magnet in the hollow piston, thereby indicating an uncoupled condition.

4.6.1.2.2.8 FMCRD Brake and Ball Check Valve

The FMCRD design incorporates an electromechanical brake (Figure 4.6-6) keyed to the motor shaft. The brake is normally engaged by spring force when the FMCRD is stationary. It is disengaged for normal rod movements by signals from the RCIS. Disengagement is caused by the energized magnetic force overcoming the spring load force. The braking torque of 49 N·m (minimum)between the motor shaft and the CRD spool piece is sufficient to prevent control rod ejection in the event of failure in the pressure-retaining parts of the drive mechanism. The brake is designed so that its failure will not prevent the control rod from rapid insertion (scram).

The electromechanical brake is located between the stepping motor and the synchronizing signal generators. The stationary spring-loaded disk and coil assembly are contained within the brake mounting bolted to the bottom of the stepping motor. The rotating disk is keyed to the stepping motor shaft and synchro shaft.

The brake is classified as passive safety-related because it performs its holding function when it is in its normally de-energized condition.

A ball check valve is located in the middle flange of the drive at the scram inlet port. The check valve is classified as safety-related because it actuates to close the scram inlet port under conditions of reverse flow caused by a break of the scram line. This prevents the loss of pressure to the underside of the hollow piston and the generation of loads on the drive that could cause a rod ejection.

4.6.1.2.2.9 Integral Internal Blowout Support

An internal CRD blowout support replaces the support structure of beams, hanger rods, grids and support bars used in BWR/6 and product lines before that. The internal support concept is illustrated schematically in Figure 4.6-7. This system utilizes the CRD outer tube integral with the internal support to provide the anti-ejection support. The outer tube is locked at top via the internal support to the control rod guide tube (CRGT) base by a bayonet coupling, which is described above. The outer tube is bolted to the CRD housing flange via the middle flange welded to it at the bottom, as described above in a discussion on FMCRD pressure boundary.

The CRD blowout support is designed to prevent ejection of the CRD and the attached control rod considering failures of two types at the weld (Point A in Figure 4.6-7) between the CRD housing and the stub tube penetration of the RPV bottom head: (1) a failure through the housing along the fusion line – just below the weld with the weld and the housing extension inside the vessel remaining intact, or (2) a failure of the weld itself with the entire housing remaining intact but without support at the penetration.

With a housing failure, the weight plus pressure load acting on the drive and housing would tend to eject the drive. In this event, the CRGT base remains supported by the intact housing extension inside the vessel; therefore, the CRD locked with the CRGT base remains supported, and thereby also restricts the coolant leakage through the small area of the annulus between the CRD outer tube and the inside of the CRD housing. In the event of total failure of the weld itself leaving the entire housing intact, the housing would tend to be driven downward by the total weight plus vessel pressure. However, after the interconnected assembly of the housing, CRD and CRGT moves down a short distance, the flange at the top of the CRGT contacts the core plate, stopping further movement of the assembly. Since the CRD is positively locked to the CRGT base, it cannot eject. In this case, the housing which bears on top of the blowout support, is also prevented from leaving the penetration, thereby restricting the coolant leak path to the small area of the annulus between the outside of CRD housing and the inside of the penetration stub tube.

An orderly shutdown would result if any of the two failures were to occur, since the restricted coolant leakage would be less than the supply from the normal make up systems. The safety-related components that provide the anti-ejection function are the (1) internal CRD blowout support, (2) CRD outer tube and middle flange, (3) entire CRD housing, (4) CRGT and (5) core plate. The materials of these components are specified to meet quality requirements consistent with that function.

If a total failure of all the flange bolts attaching the spool piece flange and also the middle flange with the CRD housing flange (Point B on Figure 4.6-7) were to occur, the drive would be prevented from moving downward by the middle flange seat provided for the spindle adapter as part of the anti-rotation gear (see Subsection 4.6.2.3.3.1.3).

4.6.1.2.2.10 FMCRD Seal Leak Detection

An FMCRD seal leak detection subsystem is located in the lower drywell underneath the drive mechanisms. It is provided to permit monitoring and collection of leakage flow past the drive shaft seal assemblies in the lower drive housings (spool pieces). By this means, seal performance can be observed during plant operation to facilitate maintenance planning for drive seal refurbishment during plant outages. The seal leak detection subsystem also functions to contain the drive leakage within a closed system where it can be routed to the drywell equipment drain sump as identified leakage.

The seal leak detection subsystem is composed of small diameter piping, flow sight glass boxes and leakage flow meters arranged into multiple leak detection groups. Each leak detection group consists of leak-off piping from multiple drives routed to a common flow sight glass box. The leak-off piping is connected to the flow sight glasses in such a way as to allow visual confirmation of leakage flow from the individual pipes and identification of the leaking FMCRD. Visual observation of leakage in this manner can only be made during plant outages when the lower drywell is accessible to plant personnel.

During plant operation, the leakage water is collected in the individual flow sight glass boxes and detected by the flow meters installed in the drain piping from each box. The flow meters are integral type meters which can sense very small quantities of leakage from each box. This method is used to monitor drive leakage during plant operation when the lower drywell is inaccessible to personnel. It allows identification of excessive leakage from any particular leak detection group.

The leakage water from all the leak detection groups is collected in a common drain header pipe and routed to the lower drywell equipment drain sump, where it contributes to containment identified leakage.

4.6.1.2.2.11 Materials of Construction

The materials of construction for the FMCRD are discussed in Subsection 4.5.1.

4.6.1.2.3 Hydraulic Control Units

Each hydraulic control unit (HCU) furnishes pressurized water for hydraulic scram, on signal from the RPS, to two drive units. Additionally, each HCU provides the capability to adjust purge flow to the two drives. A test port is provided on the HCU for connection of a portable test station to allow controlled venting of the scram insert line to test the FMCRD ball check valve during plant shutdown. Operation of the electrical system that supplies scram signals to the HCU is described in Chapter 7.

The basic components of each HCU are described in the following paragraphs. The HCU configuration is shown on the CRD System P&ID (Figure 4.6-8).

(1) Scram Pilot Valve Assembly

The scram pilot valve assembly is operated from the RPS. The scram pilot valve assembly, with two solenoids, controls the scram inlet valve. The scram pilot valve assembly is solenoid-operated and is normally energized. Upon loss of electrical signal to the solenoids (such as the loss of external AC power), the inlet port closes and the exhaust port opens. The pilot valve assembly (Figure 4.6-8) is designed so that the trip system signal must be removed from both solenoids before air pressure can be discharged from the scram valve operators. This prevents the inadvertent scram of both drives associated with a given HCU in the event of a failure of one of the pilot valve solenoids.

(2) Scram Inlet Valve

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

(3) Scram Accumulator

The scram accumulator stores sufficient energy to fully insert two control rods at any reactor 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 that 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 in the control room if water leaks past the piston barrier and collects in the accumulator instrumentation block.

(4) Purge Water Orifice and Makeup Valve

Each HCU has a restricting orifice in the purge water line to control the purge flow rate to the two associated FMCRDs. This orifice maintains the flow at a constant value while the drives are stationary. A bypass line containing a solenoid-operated valve is provided around this orifice. The valve is signaled to open and increase the purge water flow whenever either of the two associated FMCRDs is commanded to insert by the Rod Control and Information System (RCIS). During FMCRD insertion cycles, the hollow piston moves upward, leaving an increased volume for water within the drive. Opening of the purge water makeup valve increases the purge flow to offset this volumetric increase and precludes the backflow of reactor water into the drive, thereby preventing long-term drive contamination.

(5) Test Connection for FMCRD Ball Check Valve Testing and Friction Testing

Contained within the HCU is a test port to allow connection of temporary test equipment for the conduct of FMCRD ball check valve testing and drive friction testing. This test port, which has a quick-connect type coupling, is located downstream of the restricting orifice and check valve in the purge water line.

FMCRD ball check valve testing is performed by attaching the check valve test fixture to the HCU test port. The test fixture exercises the check valve by generating a controlled backflow through the check valve housing, causing the valve to backseat. The backflow is contained within a controlled volume inside the test fixture.

FMCRD friction testing also utilizes a special test fixture connected to the HCU test port. The test fixture contains a small pump and associated hydraulic controls to pressurize the underside of the hollow piston. When the pressure under the hollow piston is high enough to overcome both the combined hollow piston and control rod weight and the drive line friction, the hollow piston will separate from the ball-nut and drift the control rod into the core. Instrumentation measures the pressure under the hollow piston as it is being inserted. The measured pressure is a direct indication of the drive line friction. Water for the test fixture pump is supplied from the CRD pump suction line via piped connections to test ports located in the HCU rooms.

4.6.1.2.4 Control Rod Drive Hydraulic Subsystem

The Control Rod Drive Hydraulic Subsystem (CRDHS) supplies water under high pressure to charge the accumulators, to purge the FMCRDs and to purge the Reactor Internal Pumps (RIPs) and Reactor Water Cleanup (CUW) System pumps. The CRDHS provides the required functions with the pumps, valves, filters, piping, instrumentation and controls shown on the CRD System P&ID (Figure 4.6-8). Duplicate components are included where necessary to assure continuous system operation if an inservice component should require maintenance. For system and component classification, see Section 3.2.

The CRDHS hydraulic requirements and components are described in the following paragraphs.

4.6.1.2.4.1 Hydraulic Requirements

The CRDHS process conditions are shown in Figure 4.6-9. The hydraulic requirements, identified by the function they perform, are:

(1) An accumulator hydraulic charging pressure of approximately 14.71 MPaG is required. Flow to the accumulators is required only during scram reset or system startup.

(2) Purge water to the drives is required at a flow rate of approximately 1.3 L/min per drive unit.

(3) Approximately 10 L/min purge flow is provided to the RIPs and 20 L/min to the CUW pumps. This flow is provided to both systems at approximately CRD pump discharge pressure. Each system provides its own pressure breakdown equipment to satisfy its individual hydraulic requirements.

4.6.1.2.4.2 CRD Supply Pump

One supply pump pressurizes the CRD System with water from the condensate treatment system and/or condensate storage tanks. One spare pump is provided for standby. A discharge check valve prevents backflow through the nonoperating pump. A portion of the pump discharge flow is diverted through a minimum flow bypass line to the condensate storage tank. This flow is controlled by an orifice and is sufficient to prevent pump damage if the pump discharge is inadvertently closed.

Condensate water is processed by disposable element type pump suction filters with a 25-micrometer absolute rating. The drive water filter, downstream of the pump, is a cleanable element type with 50-micrometer absolute rating. A differential pressure indicator and control room alarm monitor each filter element as they collect foreign materials.

4.6.1.2.4.3 Accumulator Charging Water Header

Accumulator charging pressure is established by precharging the nitrogen accumulator to a precisely controlled pressure at known temperature. During scram, the scram valves open and permit the stored energy in the accumulators to discharge into the drives. The resulting pressure decrease in the charging water header allows the CRD supply pump to “run out” (i.e., flow rate to increase substantially) into the control rod drives via the charging water header. The flow element upstream of the charging water header senses high flow and provides a signal to the manual/auto flow control station which, in turn, closes the system flow control valve. This action effectively blocks the flow to the purge water header so that the runout flow is confined to the charging water header.

Safety-related pressure instrumentation is provided in the charging water header to monitor header performance. The pressure signal from this instrumentation is provided to both the RCIS and RPS. If charging water header pressure degrades, the RCIS will initiate a rod block and alarm at a predetermined low pressure setpoint. If pressure degrades even further, the RPS will initiate a scram at a predetermined low-low pressure setpoint. This assures the capability to scram and safely shut down the reactor before the HCU accumulator pressure can degrade to the level where scram performance is adversely affected following the loss of charging header pressure.

The charging water header contains a check valve and a bladder type accumulator. The accumulator is located downstream of the check valve in the vicinity of the low header pressure instrumentation. It is sized to maintain the header pressure downstream of the check valve above the scram setpoint until the standby CRD pump starts automatically, following a trip or failure of the operating CRD pump. Pressure instrumentation installed on the pump discharge header downstream of the CRD pump drive water filters monitors system pressure and generates the actuation signals for startup of the standby pump if the pressure drops below a predetermined value that indicates a failure of the operating pump.

4.6.1.2.4.4 Purge Water Header

The purge water header is located downstream from the flow control valve. The FCV adjusts automatically to maintain constant flow to the FMCRDs as reactor vessel pressure changes. Because flow is constant, the differential pressure between the reactor vessel and CRDHS is maintained constant independent of reactor vessel pressure. A flow indicator in the control room monitors purge water flow. A differential pressure indicator is provided in the control room to indicate the difference between reactor vessel pressure and purge water pressure.

4.6.1.2.5 Control Rod Drive System Operation

The operating modes of the CRD System are described in the following sections.

4.6.1.2.5.1 Normal Operation

Normal operation is defined as those periods of time when no control rod drives are in motion. Under this condition, the CRD System provides charging pressure to the HCUs and supplies purge water to the control rod drives, RIPs and CUW pumps.

A multi-stage centrifugal pump (C001) supplies the system with water from the condensate and feedwater system and/or CST. A constant portion of the pump discharge is continuously bypassed back to the CST in order to maintain a minimum flow through the pump. This prevents overheating of the pump if the discharge line is blocked. The total pump flow during normal operation is the sum of the bypass flow, the FMCRD purge water flow through the flow control valve (F010), the RIP purge flow, and the CUW pump purge flow. The standby pump provides a full capacity backup capability to the operating pump. It will start automatically if failure of the operating pump is detected by pressure instrumentation located in the common discharge piping downstream of the drive water filters.

The system water is processed by redundant filters in both the pump suction and discharge lines. One suction filter (D001) and one drive water filter (D002) are normally in operation, while the backup filters are on standby and valved out of service. Differential pressure instrumentation and control room alarms monitor the filter elements as they collect foreign material.

The purge water for each drive is provided by the purge water header. The purge water flow control valve (F010) automatically regulates the purge water flow to the drive mechanisms. The purge water flow rate is indicated in the control room.

In order to maintain the ability to scram, the charging water header maintains the accumulators at a high pressure. The scram valves remain closed except during and after scram, so during normal operation no flow passes through the charging water header. Pressure in the charging water header is monitored continuously. A significant degradation in the charging header pressure causes a low pressure warning alarm and rod withdrawal block by the RCIS. Further degradation, if occurring, causes a reactor scram by the RPS.

Pressure in the pump discharge header downstream of the drive water filters is also monitored continuously. Low pressure in this line is used to indicate that the operating pump has failed or tripped. If it should occur, automatic startup of the standby pump is initiated and the system is quickly repressurized. This prevents the malfunctioning of the operating pump from causing a reactor scram on low charging water header pressure, an event which would otherwise be a direct consequence of the malfunction.

4.6.1.2.5.2 Control Rod Insertion and Withdrawal

The FMCRD design provides the capability to move a control rod up and down both in fine steps of 18.3 mm and continuously over its entire range at a speed of 30 mm/s ±10%. Normal control rod movement is under the control of the RCIS. The RCIS controls the input of actuation power to the FMCRD motor from the electrical power supply (via the stepping motor driver module) in order to complete a rod motion command. The FMCRD motor rotates a screw shaft which, in turn, causes the vertical translation of a ball-nut on the screw shaft. This motion is transferred to the control rod via a hollow piston which rests on the ball-nut. Thus, the piston with the control rod is raised or lowered, depending on the direction of rotation of the FMCRD motor and screw shaft.

During a drive insertion, the purge water flow to the drive is increased by opening the solenoid-operated purge water makeup valve within the associated HCU. The increased flow offsets the volumetric displacement within the drive as the hollow piston is inserted into the core and prevents reactor water from being drawn back into the drive.

4.6.1.2.5.3 Scram

Upon loss of electric power to both scram pilot valve solenoids, the scram valve in the associated HCU opens to apply the hydraulic insert forces to its respective FMCRDs using high pressure water stored within the precharged accumulator (the nitrogen-water accumulator having previously been pressurized with charging water from the CRDHS). Once the hydraulic force is applied, the hollow piston disengages from the ball-nut and inserts the control rod rapidly. The water displaced from the drive is discharged into the reactor vessel. Indication that the scram has been successfully completed (all rods full-in position) is displayed to the operator.

The CRD System provides the following scram performance with vessel pressure below 7.48 MPaG (as measured at the vessel bottom), in terms of the average maximum elapsed time to attain the listed scram position (percent insertion) after loss of signal to the scram solenoid pilot valves (time zero):

Percent Insertion	Time (s)
Start of Motion	≤ 0.20
10	≤ 0.42
40	≤ 1.00
60	≤ 1.44
100	≤ 2.80

The start of motion is the time delay between loss of signal to the scram solenoid pilot valve and actuation of the 0% reed switch.

Simultaneous with the hydraulic scram, each FMCRD motor is started in order to cause electric-driven run-in of the ball-nut until it reengages with the hollow piston at the full-in position. This action is known as the scram follow function. It completes the rod full-in insertion and prepares the drives for subsequent withdrawal to restart the reactor.

After reset of the RPS logic, each scram valve recloses and allows the CRDHS to recharge the accumulators.

4.6.1.2.5.4 Alternate Rod Insertion

The alternate rod insertion (ARI) function of the CRD System provides an alternate means for actuating hydraulic scram that is diverse and independent from the RPS. The signals to initiate the ARI are high reactor dome pressure or low reactor vessel water Level 2 or manual operator action. Following receipt of any of these signals, solenoid-operated valves (F043, F044, F047, F048 and F049) on the scram air header open to reduce pressure in the header, allowing the HCU scram valves to open. The FMCRDs then insert the control rods hydraulically in the same manner as the RPS initiated scram. The same signals that initiate ARI will simultaneously actuate the FMCRD motors to insert the control rods electrically.

4.6.1.2.6 Instrumentation

The instrumentation for the CRD System is defined on the system P&ID (Figure 4.6-8). Supervisory instrumentation and alarms such as accumulator trouble and low charging water header pressure are adequate and permit surveillance of the CRD System’s readiness.

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

4.6.2 Evaluations of the CRD System

4.6.2.1 Failure Mode and Effects Analysis

This subject is covered in Appendix 15B.

4.6.2.2 Protection from Common Mode Failures

The position on this subject is covered in Appendix 15B.

4.6.2.3 Safety Evaluation

The safety evaluation of the control rod drives is given below.

4.6.2.3.1 Evaluation of Scram Time

The rod scram function of the CRD System provides the negative reactivity insertion required by Safety Design Basis 4.6.1.1.1(1). The scram time shown in the description is reflected in plant transient analyses (Chapter 15).

4.6.2.3.2 Scram Reliability

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

(1) Each accumulator provides sufficient stored energy to scram two CRDs at any reactor pressure.

(2) Each pair of drive mechanisms has its own scram valve and dual solenoid scram pilot valve; therefore, only a single scram valve needs to open for scram to be initiated. 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 FMCRD hollow piston and guide tube are designed so they will not restrain or prevent control rod insertion during scram.

(5) Each FMCRD mechanism initiates electric motor-driven insertion of its control rod simultaneous with the initiation of hydraulic scram. This provides a diverse means to assure control rod insertion.

4.6.2.3.3 Precluding Excessive Rate of Reactivity Addition

Excessive rates of reactivity addition are precluded in the design of the FMCRD. Prevention of rod ejection due to FMCRD pressure boundary failure and prevention of control rod drop are described below.

4.6.2.3.3.1 Control Rod Ejection Prevention

A failure of the CRD System pressure boundary will generate differential pressure forces across the drive, which will tend to eject the CRD and its attached control rod. The design of the FMCRD includes features that preclude rod ejection from occurring in these hypothetical circumstances. The following subsections describe how these features function for pressure boundary failures at various locations.

4.6.2.3.3.1.1 Failures at Drive Housing 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.

In the event of a failure of the housing just below the housing to penetration weld, or a failure of the weld itself with the housing remaining intact, ejection of the CRD and attached control rod is prevented by the integral internal CRD blowout support. The details of this internal blowout support structure are contained in Subsection 4.6.1.2.2.9.

4.6.2.3.3.1.2 Rupture of Hydraulic Line to Drive Housing Flange

For the case of a scram insert line break, a partial or complete circumferential opening is postulated at or near the point where the line enters the housing flange. This failure, if not mitigated by special design features, could result in rod ejection at speeds exceeding maximum allowable limits of 10 cm/s (assuming rod pattern control) or 15 cm maximum travel distance before full stop. Failure of the scram insert line would cause loss of pressure to the underside of the hollow piston. The force resulting from full reactor pressure acting on the cross-sectional area of the hollow piston, plus the weights of the control rod and hollow piston, is imposed on the ball-nut. The ball-nut, in turn, translates this resultant force into a torque acting on the spindle. When this torque exceeds the motor residual torque and seal friction, reverse rotation of the spindle will occur, permitting rod withdrawal. Analyses show that the forces generated during this postulated event can result in rod ejection speeds which exceed the maximum allowable limits.

The FMCRD design provides two diverse means of protection against the results of a postulated scram insert line failure. The first means of protection is a ball check valve located in the middle flange of the drive at the scram port. Reverse flow during a line break will cause the ball to move to the closed position. This will prevent loss of pressure to the underside of the hollow piston, which, in turn, will prevent the generation of loads on the drive which could cause rod ejection.

The second means of protection is the FMCRD brake described in Subsection 4.6.1.2.2.8. In the event of the failure of the check valve, the passive brake will prevent the ball spindle rotation and rod ejection.

4.6.2.3.3.1.3 Total Failure of All Drive Flange Bolts

The FMCRD design provides an anti-rotation device which engages when the lower housing (spool piece) is removed for maintenance. This device prevents rotation of the spindle and hence control rod motion when the spool piece is removed. The two components of the anti-rotation device are (1) the upper half of the coupling between the lower housing drive shaft and ball spindle, and (2) the back seat of the middle flange (Figure 4.6-1). The coupling of the lower housing drive shaft to the ball spindle is splined to permit removal of the lower housing. The underside of the upper coupling piece has a circumferentially splined surface which engages with a mating surface on the middle flange back seat when the spindle is lowered during spool piece removal. When engaged, spindle rotation is prevented. In addition to preventing rotation, this device also provides sealing of leakage from the drive while the spool piece is removed. In the unlikely event of the total failure of all the drive flange bolts attaching the spool piece flange and the middle flange of the drive to the housing flange, the anti-rotation device will be engaged when the spool piece falls and the middle flange / outer tube / CRD blowout support will be restrained by the control rod guide tube base bayonet coupling, thus preventing rod ejection.

4.6.2.3.3.2 Control Rod Drop Prevention

Control rod drop is prevented by the following features:

(1) Two redundant Class 1E switches in the FMCRD sense separation of the hollow piston, which positions the control rod, from the ball-nut. These switches sense either separation of the piston from the nut or separation of the control rod from the piston, and block further lowering of the nut, thereby preventing drop of either the control rod or the control rod and hollow piston as an assembly (See Subsection 4.6.1.2.2.6 for further details).

(2) Two redundant spring-loaded latches on the hollow piston open to engage in openings in the guide tube within the FMCRD to catch the hollow piston if separation from the ball-nut were to occur. These latches open to support the hollow piston (and control rod) following every scram until the ball-nut is run-in to provide the normal support for the hollow piston (and control rod).

(3) The control-rod to hollow-piston coupling is a bayonet type coupling. Coupling is verified by pull test for the control rod upon initial coupling at refueling and again each time an attempt is made to drive beyond the “full out” position during reactor operation. The control rod can only be uncoupled from the FMCRD by relative rotation, which is not possible during operation. The control rod cannot rotate, since it is always constrained between four fuel assemblies, and the hollow piston / CRD bayonet coupling cannot rotate, since the hollow piston has rollers which operate in a track within the FMCRD. Only structural failure would permit or result in control rod to FMCRD uncoupling, which, in turn, could only result in rod drop if the redundant switches failed to sense separation. In such failure scenarios, the rate of rod drop may exceed acceptable reactivity addition rates; however, the number of failures involved in the scenario are so numerous that the probability of occurrence for the event is low enough to be categorized as incredible.

4.6.2.3.4 CRD Maintenance

The procedure for removal of the FMCRD for maintenance or replacement is similar to previous BWR product lines. The control rod is first withdrawn until it backseats onto the control rod guide tube. This metal-to-metal contact provides the seal that prevents draining of reactor water when the FMCRD is subsequently lowered out of the CRD housing. The control rod normally remains in this backseated condition at all times with the FMCRD out; however, in the unlikely event it also has to be removed, a temporary blind flange is first installed on the end of the CRD housing to prevent draining of reactor water.

If the operator inadvertently removes the control rod after FMCRD is out without first installing the temporary blind flange, or conversely, inadvertently removes the FMCRD after first removing the control rod, an unisolable opening in the bottom of the reactor will be created, resulting in drainage of reactor water. The possibility of inadvertent reactor draindown by this means is considered remote for the following reasons:

(1) Procedural controls similar to those of current BWRs will provide the primary means for prevention. Current BWR operating experience demonstrates this to be an acceptable approach. There has been no instance of an inadvertent draindown of reactor water due to simultaneous CRD and control rod removal.

(2) During drive removal operations, personnel will be required to monitor under the RPV for water leakage out of the CRD housing. Abnormal or excessive leakage occurring after only a partial lowering of the FMCRD within its housing will indicate the absence of the full metal-to-metal seal between the control rod and control rod guide tube required for full drive removal. In this event, the FMCRD can then be raised back into its installed position to stop the leakage and allow corrective action.

See Subsection 4.6.6.1 for COL applicant license information.

The FMCRD design also allows for separate removal of the stepping motor, position indicator probe (PIP) and spool piece for maintenance during plant outrages without disturbing the upper assembly of the drive. While these FMCRD components are removed for servicing, the associated control rod is maintained in the fully inserted position by one of two mechanical locking devices that prevent rotation of the ball spindle and drive shaft.

The first anti-rotation device (Detail A in Figure 4.6-10) is engaged when the motor assembly consisting of the stepping motor, brake and synchro is removed. It is a horizontally acting spring-actuated sliding pin located on the bottom of the spool piece. When the motor assembly is lowered away from the spool piece, the sliding pin is released from its normally retracted position and engaged by spring force with gear teeth on the spool piece drive shaft, thereby locking the shaft in place. This design is similar to that of an anti-rotation device that has been successful for many years in the same application by a European FMCRD design.

With the motor assembly removed, the sliding pin can be visually checked from below the drive to verify that it is properly engaged. When the vessel head is removed, another means of verification of proper locking is for the operator to view the top of the control rod from over the reactor vessel. If the top of the control rod is visible at its normal full-in position, it provides both direct indication that the control rod remains fully inserted and additional assurance that the ball spindle is restrained from reverse rotation. The drive shaft remains locked in this manner until the motor assembly is reattached to the spool piece. During motor installation, a pin-and-roller device on the top of the motor engages with a lever attached to the sliding pin as the motor is raised in to contact with the spool piece. The pin-and-roller forces the lever and sliding pin away from the drive shaft and into the normally retracted, unlocked position.

The second anti-rotation device (Detail B in Figure 4.6-10) is engaged when the spool piece is removed from the FMCRD. As described in Subsection 4.6.2.3.3.1.3, this device is a spline arrangement between the ball spindle lower portion and the middle flange backseat. When removing and lowering the spool piece, the weight of the ball spindle, hollow piston and control rod provides a vertical force in the downward direction that brings the two splines together. This locks the ball spindle into the backseat and prevents reverse rotation. As with the first anti-rotation device, proper engagement of this device can be visually checked from below the drive. If the splines did not completely lock together, there will be indication of this because the ball spindle will not seat against the backseat and there will be a small gap for leakage of water. If this should occur, removal of the spool piece can be discontinued and corrective action taken. If there is no leakage, it confirms that the splines are properly locked together. Also as in the case of the first anti-rotation device, visual observation of the top of the control rod from over the reactor vessel provides another means for verifying proper locking of the ball spindle. The ball spindle remains locked in this position until the spool piece is reattached to the FMCRD. During spool piece installation, the end of the drive shaft fits into a seat on the end of the ball spindle. As the spool piece is raised off the middle flange backseat, the anti-rotation splines disengage and the weight of the ball spindle, hollow piston and control rod is transferred to the spool piece assembly.

4.6.3 Testing and Verification of the CRDs

4.6.3.1 Development Tests

The initial development of the FMCRD involved testing of a prototype based on a European drive design. Testing of this prototype included more than 600 scrams and 67,000 motor-driven cycles. A subsequent prototype was developed for installation in an operating BWR for the purpose of demonstrating FMCRD performance under actual BWR operating conditions. This in-plant FMCRD prototype was tested extensively prior to installation at the operating plant, including over 500 scrams and 63,000 step cycles. The inplant FMCRD was installed at LaSalle Unit 2, where it was tested for one complete operating cycle.

A reference FMCRD prototype design, based on refinements of initial development prototypes described above, has been developed and tested. To date, testing of this reference prototype has included over 1,000 scrams and 150,000 step cycles. These tests have demonstrated the following:

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

(2) No abnormal distortion or deformation was found. Wear, abrasion and corrosion were negligible.

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

4.6.3.2 Factory Quality Control Tests

The quality control specifications and procedures will follow the general pattern established for such specifications and procedures in BWRs presently in operation.

Quality control of welding, heat treatment, dimensional tolerances, material verification and similar factors will be maintained throughout the manufacturing process to assure reliable performance of the mechanical reactivity control components. Some of the quality control tests performed on the 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) Drive shaft seals are tested for leakage to demonstrate correct seal operation.

   (e) Each drive is tested for shim (drive-in and -out) motion and control rod position indication.

   (f) Each drive is subjected to cold 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 HCU's ability to perform its part of a scram is demonstrated.

4.6.3.3 Functional Tests

These tests evaluate drive performance under conditions of crud / contamination, seismic misalignment, channel bulge, failed buffer, rod drop (to test hollow piston latch functionality), and rod ejection (to test FMCRD brake functionality).

4.6.3.4 Operational Tests

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

The switches which detect separation will provide indication and automatic rod withdrawal block should a control rod separate from the drive mechanism during rod withdrawal. Additionally, the operator can observe the incore monitor indications to verify that the control rod is following the drive mechanism. All control rods that are partially withdrawn from the core can be tested for rod-following by inserting or withdrawing the rod one or two steps and returning it to its original position, while the operator observes the incore 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 hollow piston to overtravel-out demonstrates the integrity of the rod-to-drive coupling.

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

4.6.3.5 Acceptance Tests

Criteria for acceptance of the CRD system and the associated control and protection systems will be incorporated in specifications and test procedures covering the preoperational test phase and the startup test phase.

The preoperational 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, scram valve response times, and control pressures. These are tests intended more to document system condition rather than tests of performance.

As fuel is placed in the reactor, the startup test procedure (Chapter 14) is followed. The tests in this procedure are intended to demonstrate that the initial operational characteristics meet the limits of the specifications over the range of primary coolant temperatures and pressures from ambient to operating. The detailed specifications and procedures are similar to those in BWRs presently in operation.

In the preoperational and startup test phases, the drive insertion times measured during scram tests are compared with scram performance criteria derived from data taken during the development testing of the reference FMCRD prototype design (see Subsection 4.6.3.1). Similar to current BWRs, the performance criteria specifies the acceptable range of drive insertion times at reactor vessel pressures extending from ambient to full operating pressure. In the preoperational test phase, the scram tests are typically performed with the reactor vessel at ambient pressure. If a drive is operating properly, it will insert within the specified time limits corresponding to this low pressure condition. Given there is no significant change to any other condition which can affect scram performance (e.g., accumulator pressure, scram lines losses, driveline friction), the drive can then be expected to also operate within the limits specified for high reactor pressure. Projection of the drive performance to the high pressure condition in this manner provides confidence for proceeding with the high pressure scram testing of the startup test phase.

4.6.3.6 Surveillance Tests

The surveillance requirements for the CRD System are described below. While these requirements have not yet been formalized, the intent is to follow the general pattern established for surveillance testing in BWRs presently in operation.

(1) Each fully withdrawn control rod is exercised at least once each week. Each partially withdrawn control rod is exercised at least once each month.

(2) The coupling integrity is verified for each withdrawn control rod when the rod is fully withdrawn the first time. The procedure, as described in Section 4.6.1.2.2.7, is to withdraw the drive into the overtravel condition and observe the operation of the overtravel reed switch. If the drive is properly coupled to the control rod the overtravel reed switch will not actuate. If the reed switch actuates, it indicates the drive is uncoupled from the control rod.

(3) During operation, accumulator pressure and level at the normal operating value are verified.

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

(4) At the time of each major refueling outage, each operable control rod is subjected to scram time tests from the fully withdrawn position.

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

4.6.4 Information for Combined Performance of Reactivity Control Systems

4.6.4.1 Vulnerability to Common Mode Failures

The Reactivity Control System is located such that it is protected from common mode failures due to missiles, failures of moderate and high energy piping, and fire. Sections 3.5, 3.6 and 3.7, and Subsection 9.5.1 discuss protection of essential systems against missiles, pipe breaks, seismic and fire, respectively.

4.6.4.2 Accidents Taking Credit for Multiple Reactivity Systems

There are no postulated accidents documented 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 Subsection 4.6.4.2, credit is not taken for multiple reactivity control systems for any postulated accidents documented in Chapter 15. (See Subsection 4.6.2.3.4)

4.6.6 COL License Information

4.6.6.1 CRD and FMCRD Maintenance Procedures During Maintenance

The COL applicant shall develop procedures to ensure that maintenance procedures have provisions to prohibit coincident removal of the CRD blade and drive of the same assembly. In addition, the COL applicant shall develop contingency procedures to provide core and spent fuel cooling capability and mitigative actions during CRD replacement with fuel in the vessel.

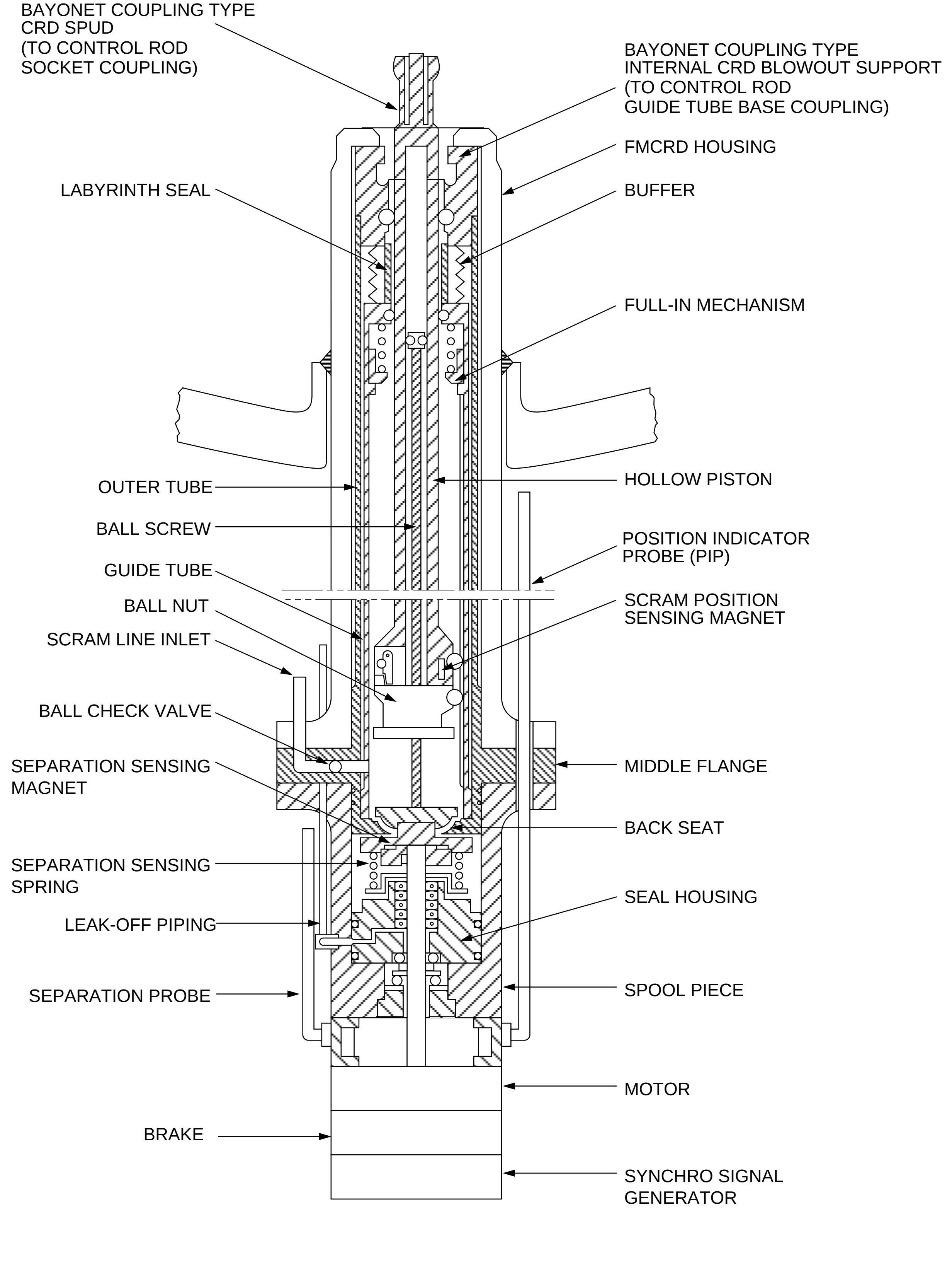


Figure 4.6-1 Fine Motion Control Rod Drive Schematic

NO.	MAIN PART NAME
1	SEAL HOUSING
2	DRIVE SHAFT
3	BEARING
4	GLAND PACKING
5	SPRING (FOR GLAND PACKING)
6	SPOOL PIECE
7	BALL SCREW SHAFT
8	BALL NUT AND BALLS
9	GUIDE ROLLER AND PIN (FOR BALL NUT)
10	SPINDLE HEAD BOLT
11	SPINDLE HEAD BUSHING
12	SEPARATION SPRING
13	MAGNET
14	BUFFER SPRING
15	BUFFER SLEEVE
16	GUIDE ROLLER, PIN
17	STOP PISTON
18	PISTON TUBE
19	PISTON HEAD
20	LATCH
21	LATCH SPRING
22	BAYONET COUPLING
23	COMPRESSION ROD
24	GUIDE TUBE
25	OUTER TUBE
26	MIDDLE FLANGE
27	CHECK VALVE BALL
28	RUBBER SEAL
29	O RING SEAL (BETWEEN CRD HOUSING AND CRD)
30	CRD INSTALLATION BOLT
31	UPPER GUIDE (CRD BLOWOUT SUPPORT
32	ANTI-ROTATION SPLINES
33	ANTI-ROTATION PIN

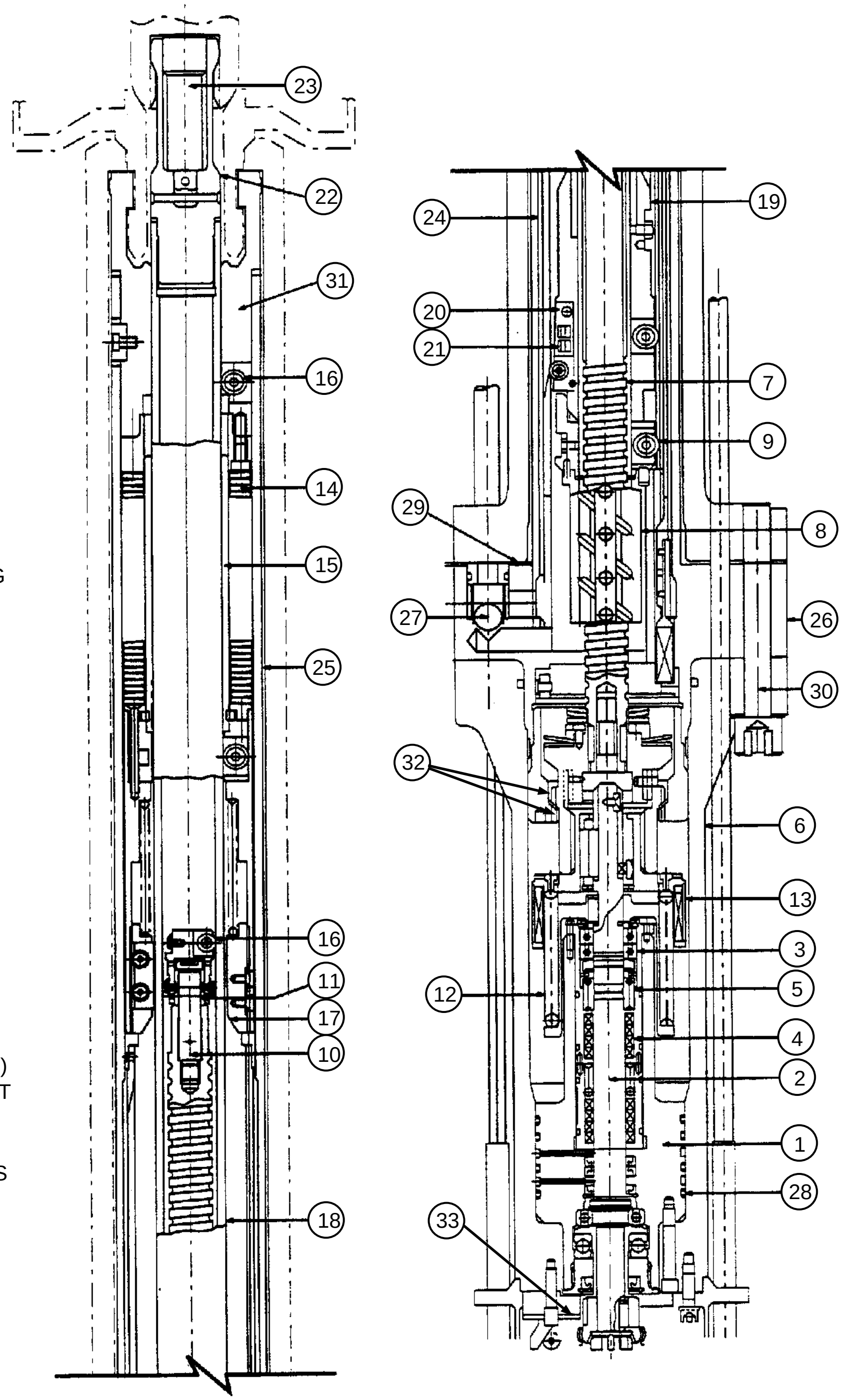


Figure 4.6-2 Fine Motion Control Rod Drive Unit (Cutaway)

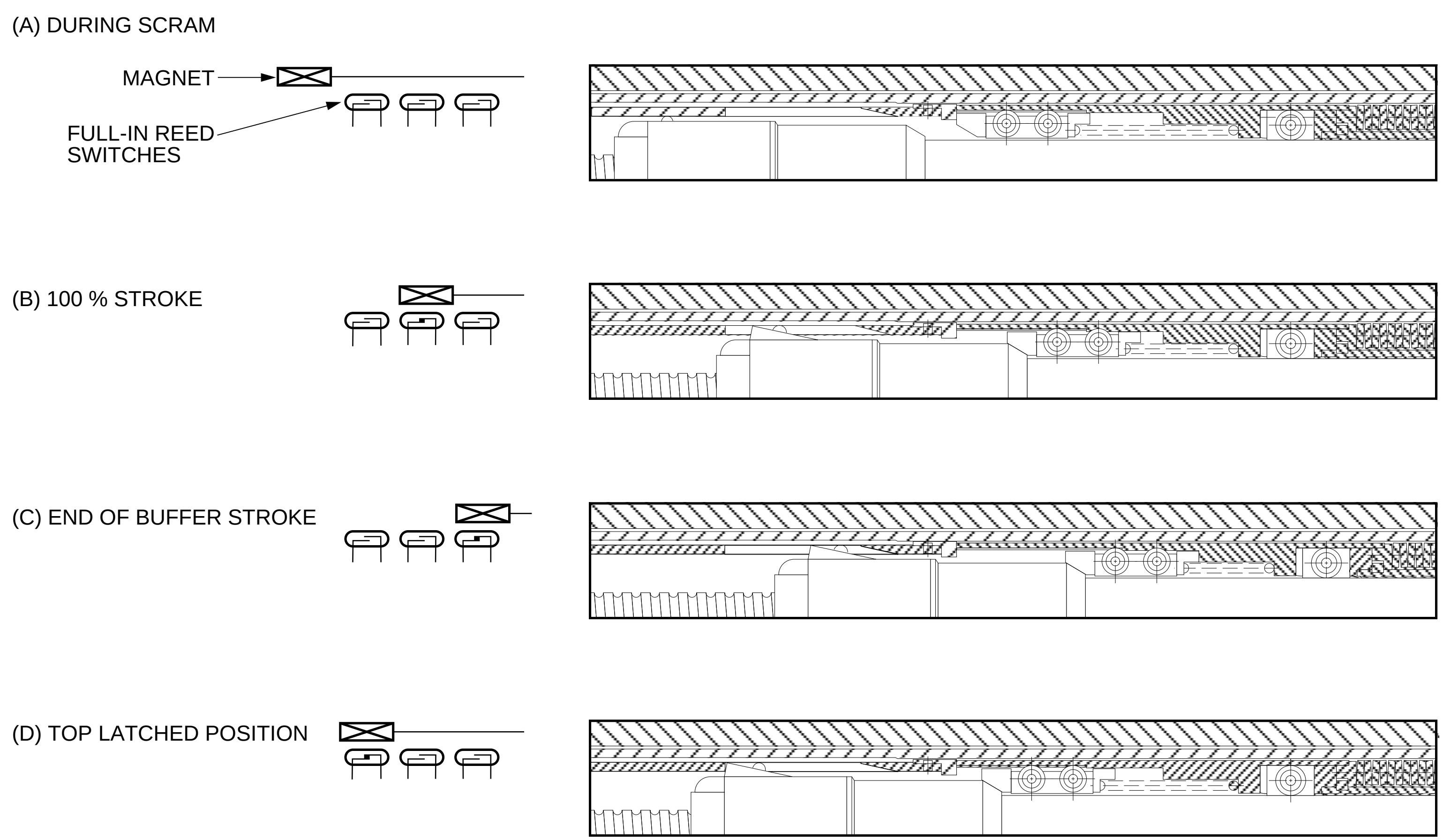


Figure 4.6-3 Continuous Full-in Indicating Device

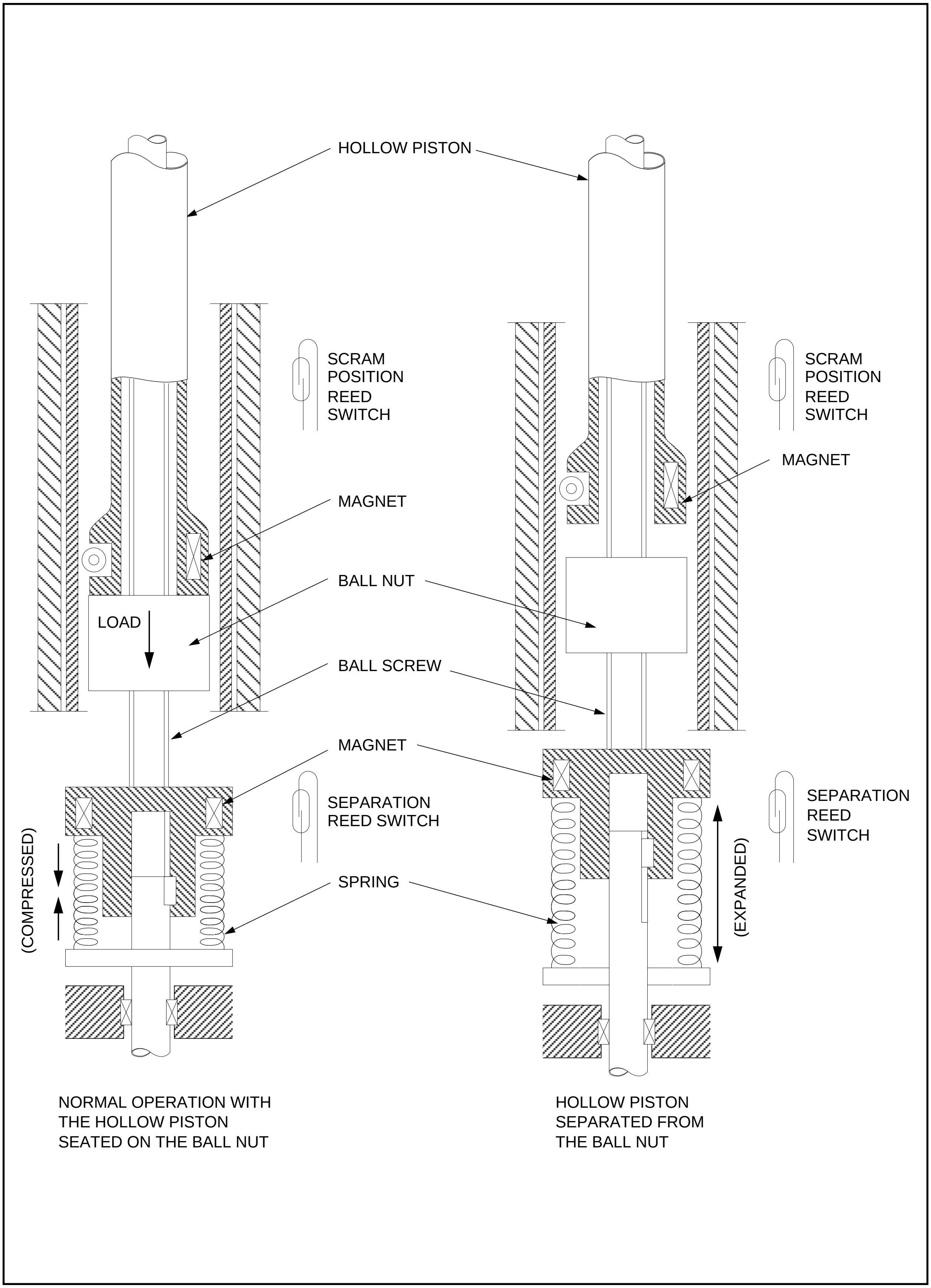


Figure 4.6-4 Control Rod Separation Detection

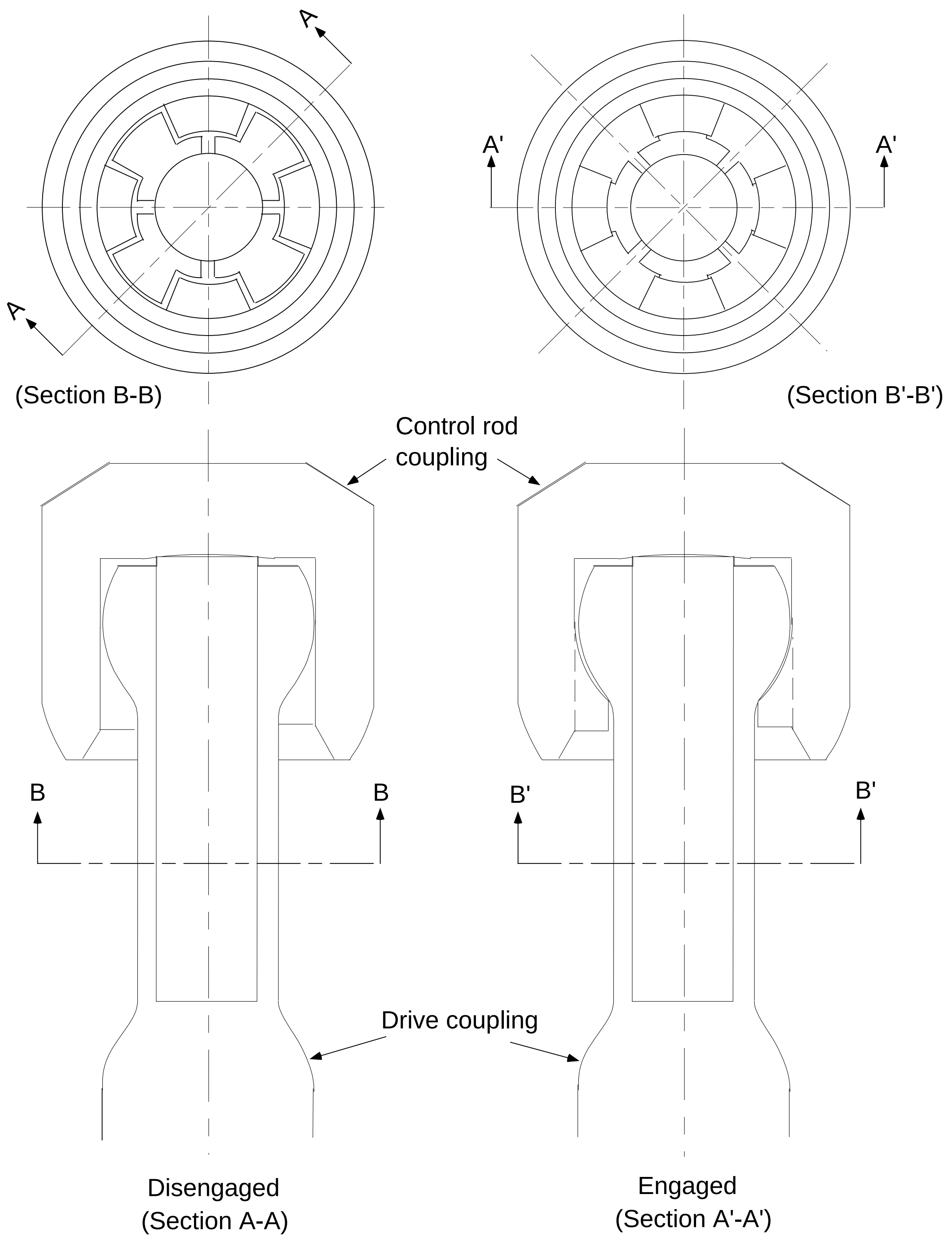


Figure 4.6-5 Control Rod to Control Rod Drive Coupling

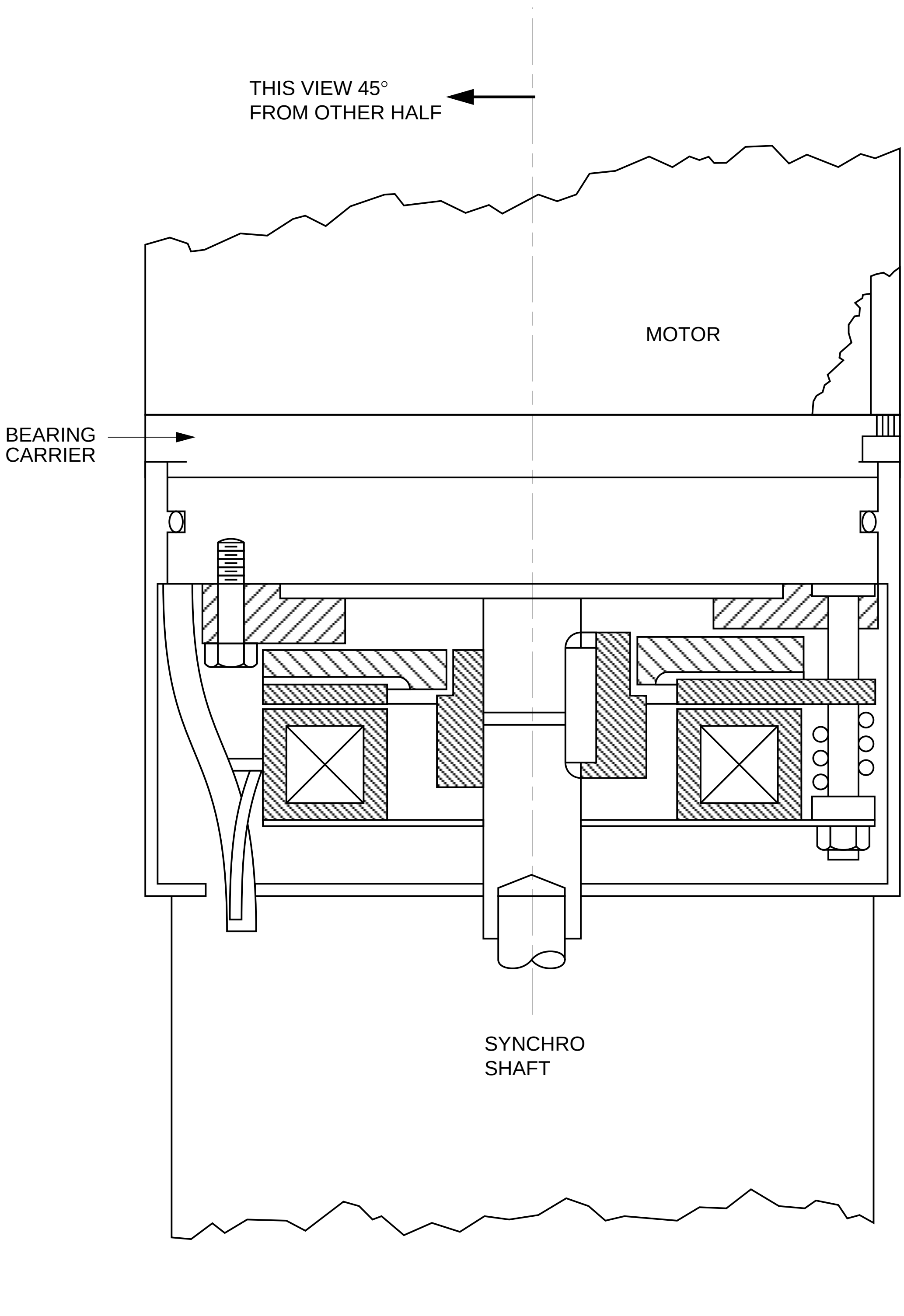


Figure 4.6-6 FMCRD Electro-mechanical Brake

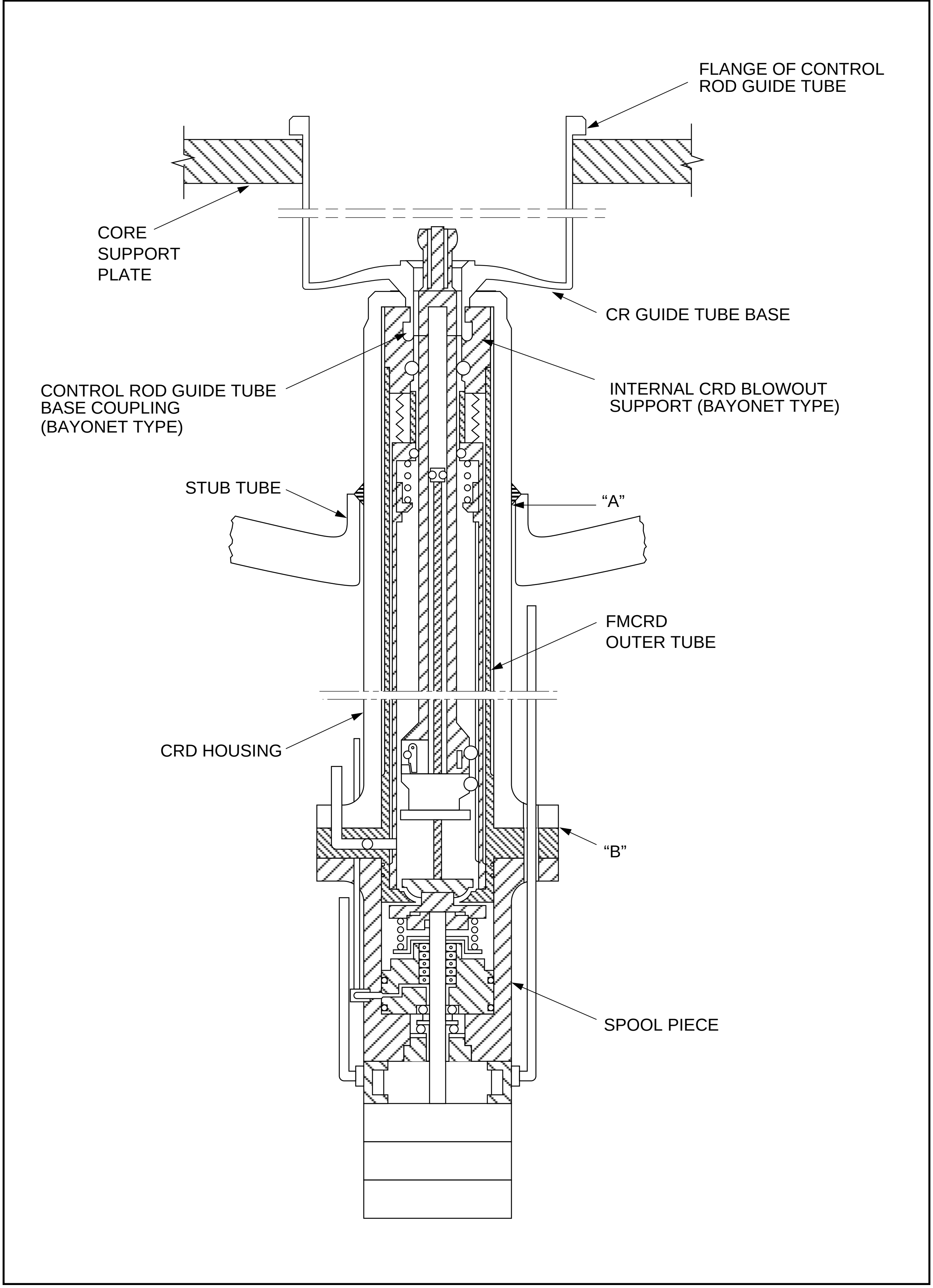


Figure 4.6-7 Internal Blowout Support Schematic

The following figures are located in Chapter 21:

Figure 4.6-8 Control Rod Drive System P&ID (Sheets 1-3)

Figure 4.6-9 Control Rod Drive System PFD

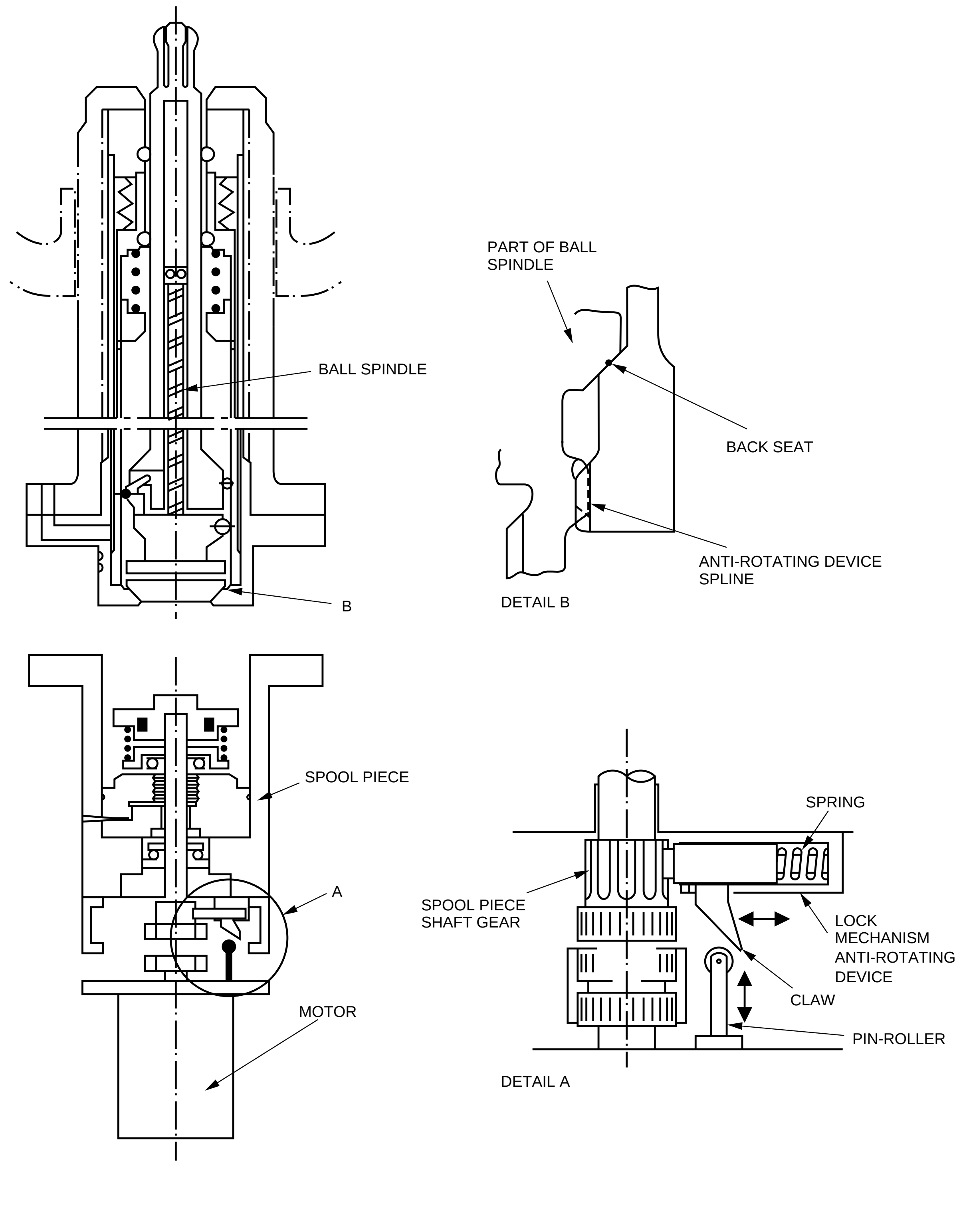


Figure 4.6-10 FMCRD Anti-Rotation Devices

4A Typical Control Rod Patterns and Associated Power Distribution for ABWR

4A.1 Introduction

This appendix contains a typical simulation of an equilibrium cycle. The control rod patterns used are just one example of a set of control rod patterns which could be used to provide the radial and axial power shaping needed to meet the Technical Specifications.

[*The basic control rod strategy for this case is provided in Table 4A-1.*

4A.2 Power Distribution Strategy

A basic operating principle (Haling) used to minimize power peaking throughout an operating cycle has been developed and is applied to boiling water reactors (Reference 4A-1). The main concept is that "for any given set of end-of-cycle conditions, the power peaking factor is maintained at a minimum value when the power shape does not change during the operating cycle."

4A.3 Results of Core Simulation Studies

Table 4A-2 itemizes the exposure steps and their related figure numbers. The detailed data presented demonstrates that this design can be operated throughout this cycle with adequate margins to allow for operating flexibility. Significant margin exists relative to the MAPLHGR limit (Section 6.3). The variation of the minimum critical power ratio (MCPR) with cycle exposure is shown in Figure 4A-14. Similarly, a large margin is indicated with respect to the expected MCPR operating limit.]*

4A.4 References

4A-1 J.A. Woolley, "Three-Dimensional BWR Core Simulator", January 1977 (NEDO-20953A).

* See Section 4.2.

Table 4A-1 Basic Control Strategy for Typical ABWR

The basic control rod strategy for this case consists of a single control rod pattern using only deeply inserted rods to compensate for excess reactivity located adjacent to low reactivity fuel. Axially zoned fuel flattens the core axial power distribution, eliminating the need for shallow control rods. Insertion of the deep blades next to low reactivity fuel, combined with a slowly varying hot excess reactivity, minimizes control rod motion and eliminates the need for any control rod pattern exchanges. The core is operated at 100% rated core flow from the beginning of the cycle until the core is critical with all control rods withdrawn. Then the core flow is gradually increased to 111% rated core flow, maintaining criticality while extending the cycle burnup.

Table 4A-2 Incremental Exposure Steps and Related Figures Numbers

Incremental Exposure (GWd/t)	Simulation	Figure Numbers
8.5	All-Rods-Out Haling 100% Flow	4A-1a through 4A-1e
9.0	All-Rods-Out Haling 111% Flow EOC	4A-2a through 4A-2e
0.2	Stepwise Rod Pattern	4A-3a through 4A-3e
1.1	Stepwise Rod Pattern	4A-4a through 4A-4e
2.2	Stepwise Rod Pattern	4A-5a through 4A-5e
3.3	Stepwise Rod Pattern	4A-6a through 4A-6e
4.4	Stepwise Rod Pattern	4A-7a through 4A-7e
5.5	Stepwise Rod Pattern	4A-8a through 4A-8e
6.6	Stepwise Rod Pattern	4A-9a through 4A-9e
7.7	Stepwise Rod Pattern	4A-10a through 4A-10e
8.0	Stepwise Rod Pattern	4A-11a through 4A-11e
8.4	All-Rods-Out 100% Flow Stepwise Rod Pattern	4A-12a through 4A-12e
9.0	All-Rods-Out 111% Flow Stepwise Rod Pattern EOC	4A-13a through 4A-13e

Figure 4A-1a [Proprietary information not included DCD (Refer to SSAR Appendix 4A, Amendment 34)]

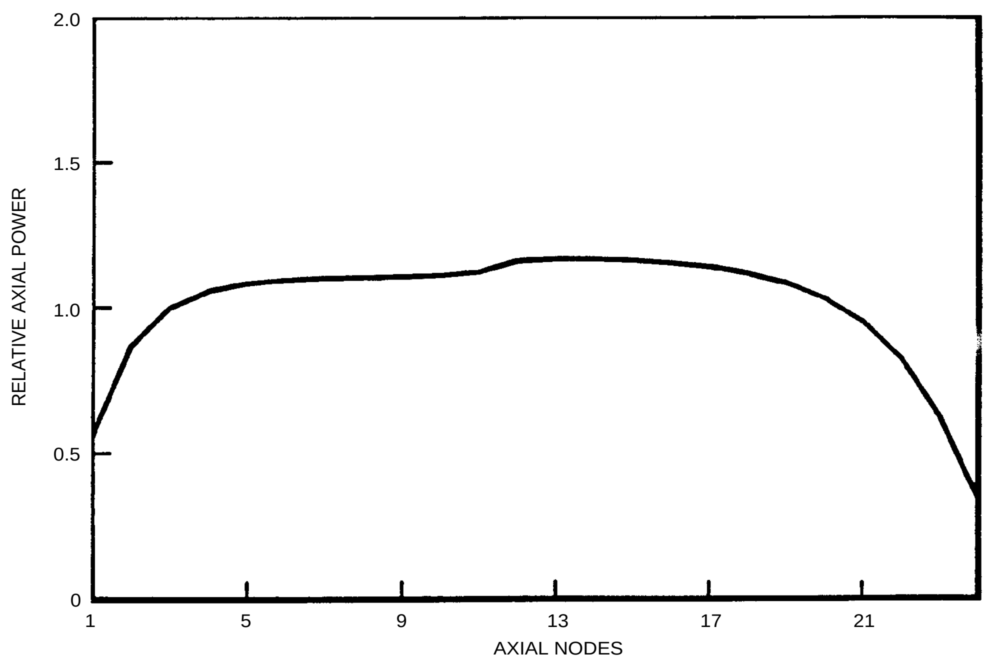


Figure 4A-1b Relative Axial Power at 8.5 GWd/t Cycle Exposure (Haling)

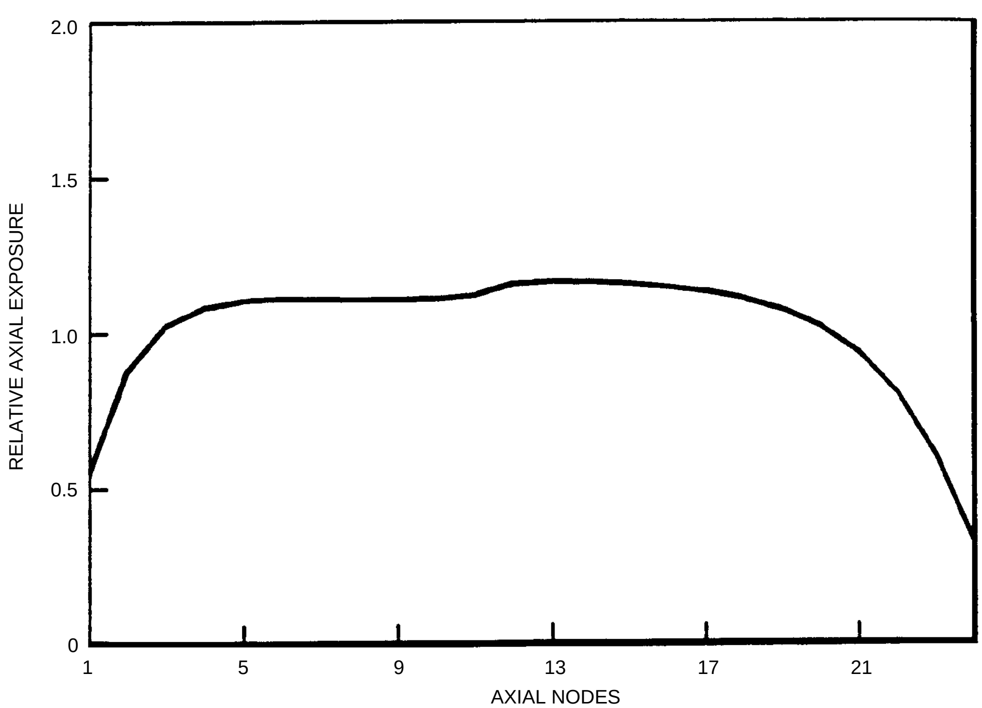


Figure 4A-1c Relative Axial Exposure at 8.5 GWd/t Cycle Exposure (Haling)

Figure 4A-1d [Proprietary information not included DCD (Refer to SSAR Appendix 4A, Amendment 34)]

Figure 4A-1e [Proprietary information not included DCD (Refer to SSAR Appendix 4A, Amendment 34)]

Figure 4A-2a [Proprietary information not included DCD (Refer to SSAR Appendix 4A, Amendment 34)]

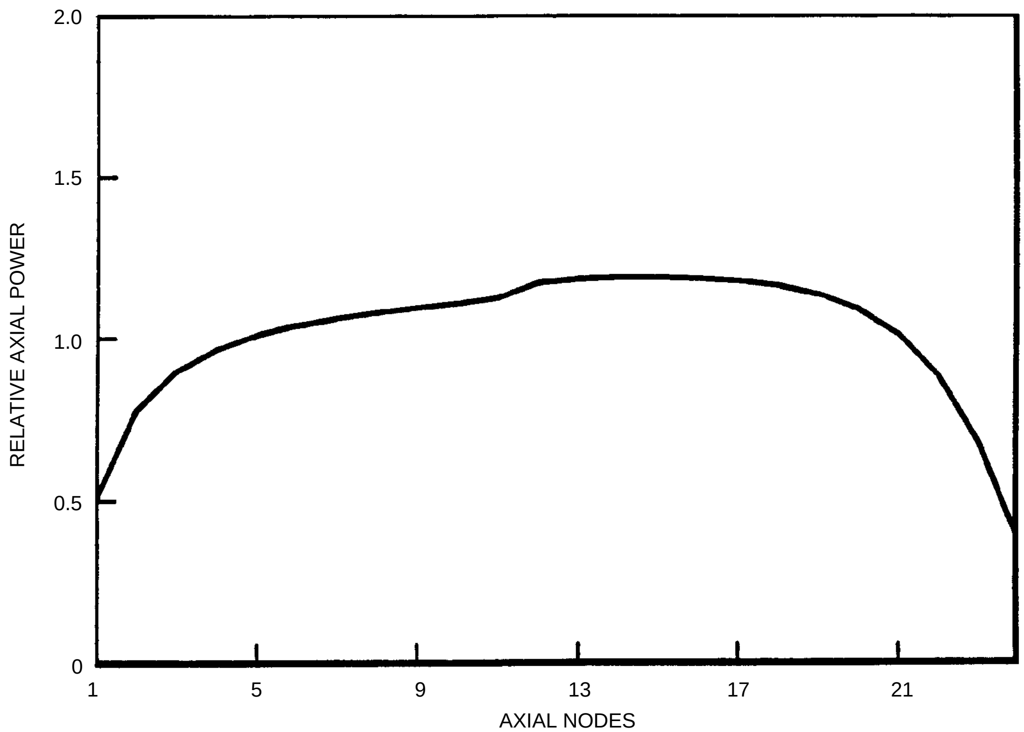


Figure 4A-2b Relative Axial Power at 9.0 GWd/t Cycle Exposure (Haling)

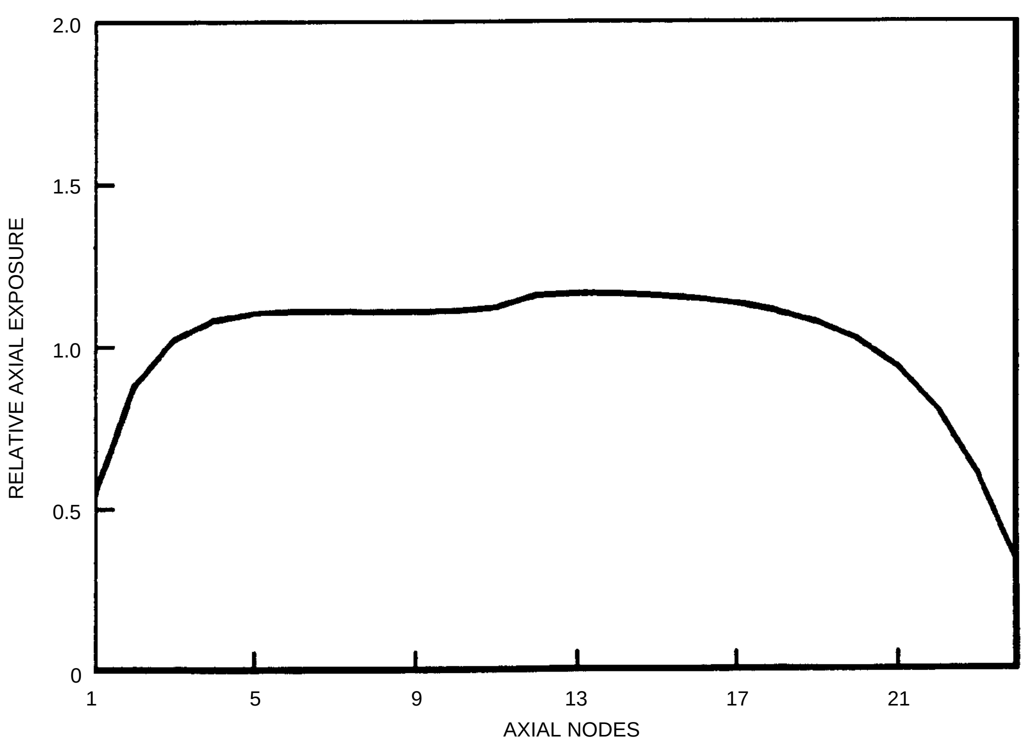


Figure 4A-2c Relative Axial Exposure at 9.0 GWd/t Cycle Exposure (Haling)

Figure 4A-2d [Proprietary information not included DCD (Refer to SSAR Appendix 4A, Amendment 34)]

Figure 4A-2e [Proprietary information not included DCD (Refer to SSAR Appendix 4A, Amendment 34)]

Figure 4A-3a [Proprietary information not included DCD (Refer to SSAR Appendix 4A, Amendment 34)]

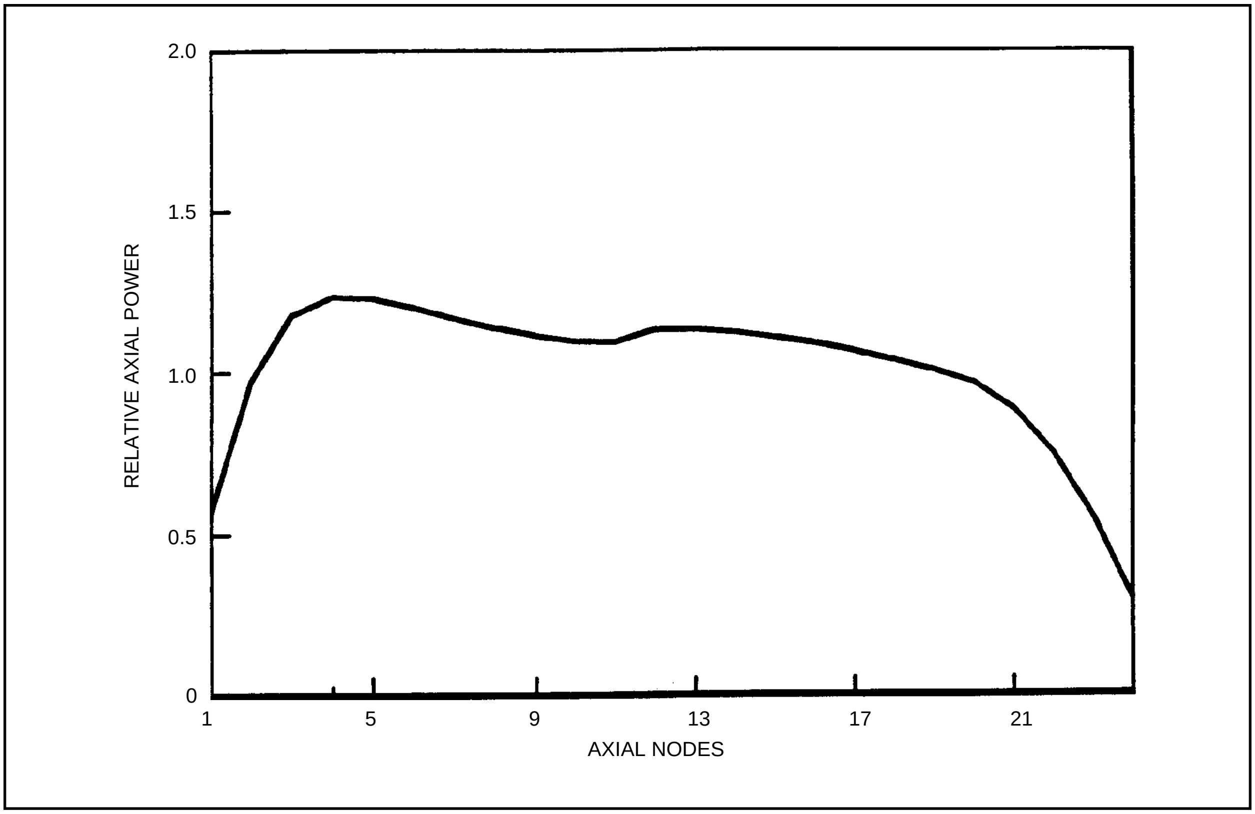


Figure 4A-3b Relative Axial Power at 0.2 GWd/t Cycle Exposure

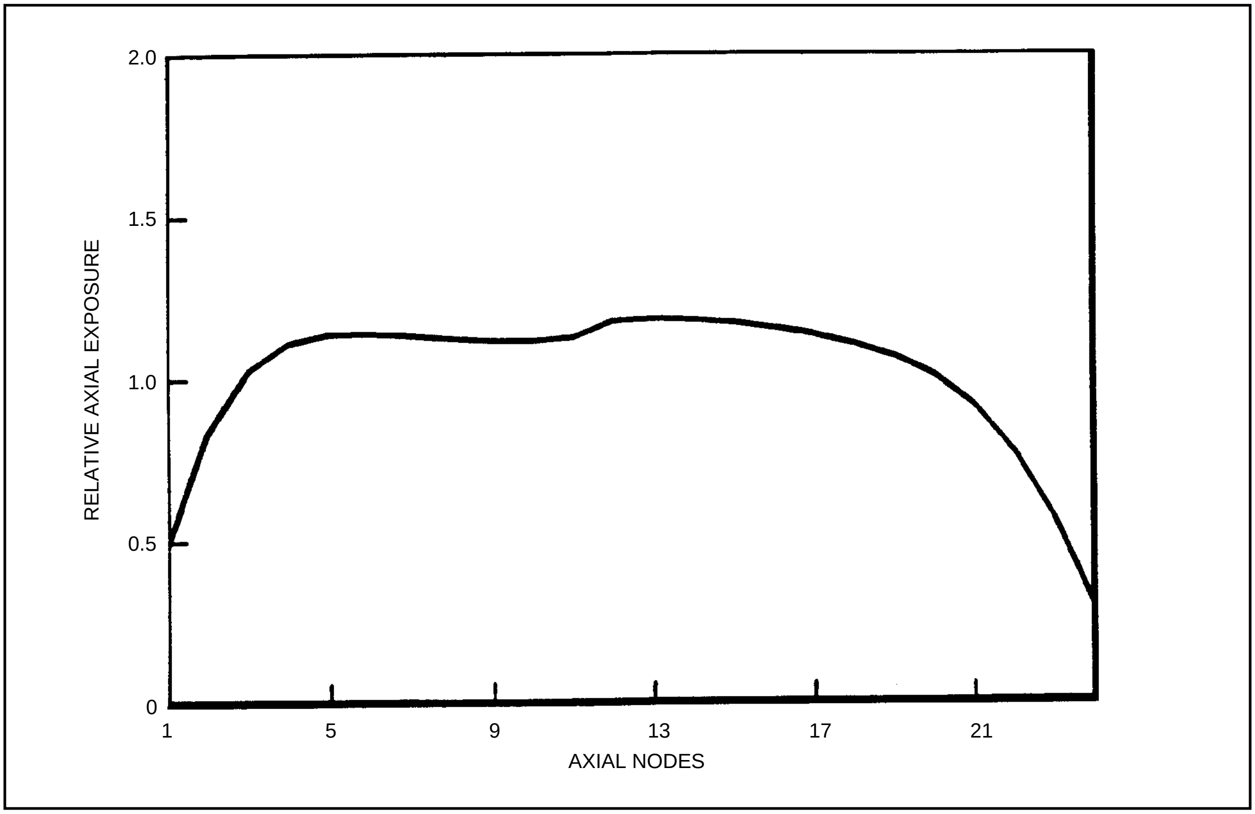


Figure 4A-3c Relative Axial Exposure at 0.2 GWd/t Cycle Exposure

Figure 4A-3d [Proprietary information not included DCD (Refer to SSAR Appendix 4A, Amendment 34)]

Figure 4A-3e [Proprietary information not included DCD (Refer to SSAR Appendix 4A, Amendment 34)]

Figure 4A-4a [Proprietary information not included DCD (Refer to SSAR Appendix 4A, Amendment 34)]

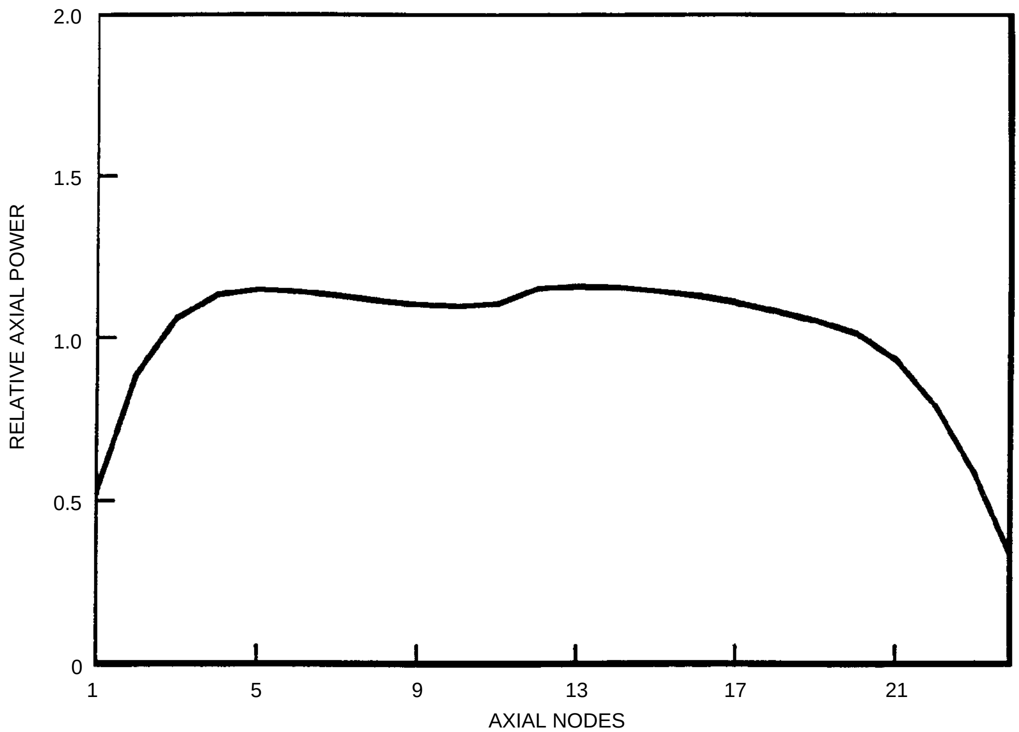


Figure 4A-4b Relative Axial Power at 1.1 GWd/t Cycle Exposure

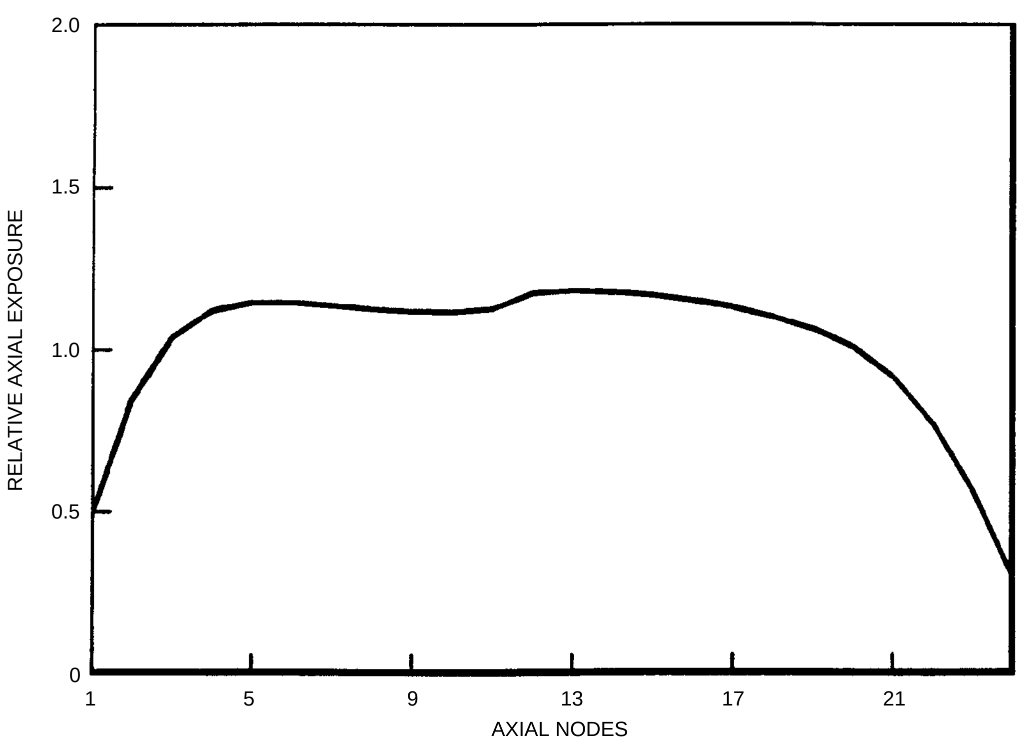


Figure 4A-4c Relative Axial Exposure at 1.1 GWd/t Cycle Exposure

Figure 4A-4d [Proprietary information not included DCD (Refer to SSAR Appendix 4A, Amendment 34)]

Figure 4A-4e [Proprietary information not included DCD (Refer to SSAR Appendix 4A, Amendment 34)]

Figure 4A-5a [Proprietary information not included DCD (Refer to SSAR Appendix 4A, Amendment 34)]

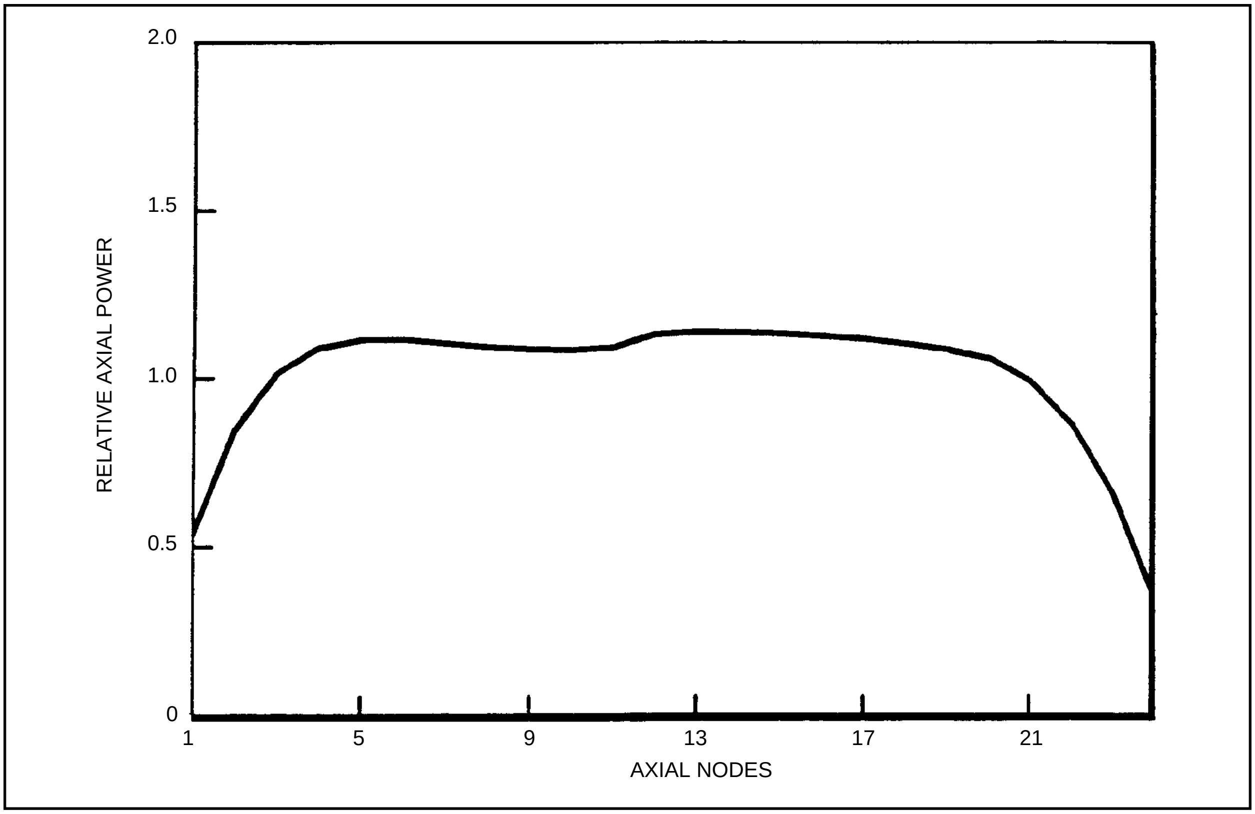


Figure 4A-5b Relative Axial Power at 2.2 GWd/t Cycle Exposure

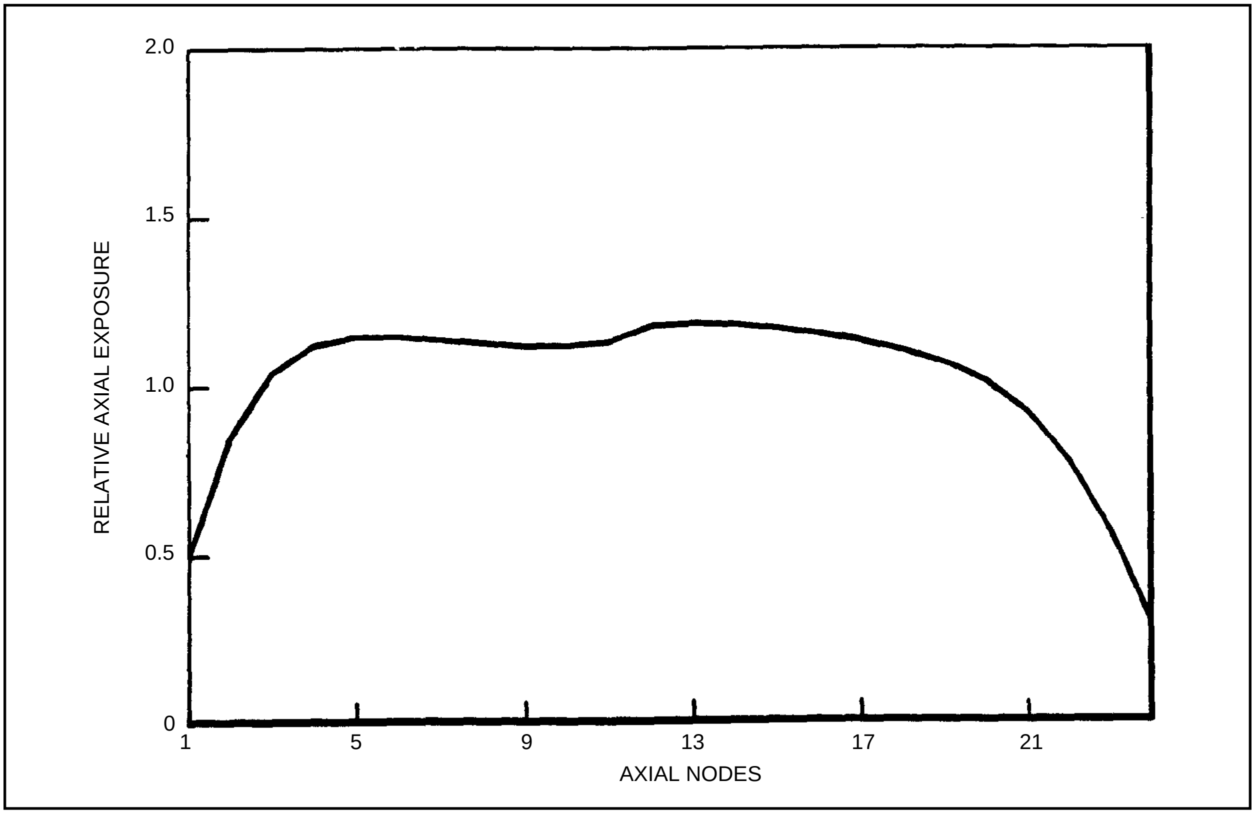


Figure 4A-5c Relative Axial Exposure at 2.2 GWd/t Cycle Exposure

Figure 4A-5d [Proprietary information not included DCD (Refer to SSAR Appendix 4A, Amendment 34)]

Figure 4A-5e [Proprietary information not included DCD (Refer to SSAR Appendix 4A, Amendment 34)]

Figure 4A-6a [Proprietary information not included DCD (Refer to SSAR Appendix 4A, Amendment 34)]

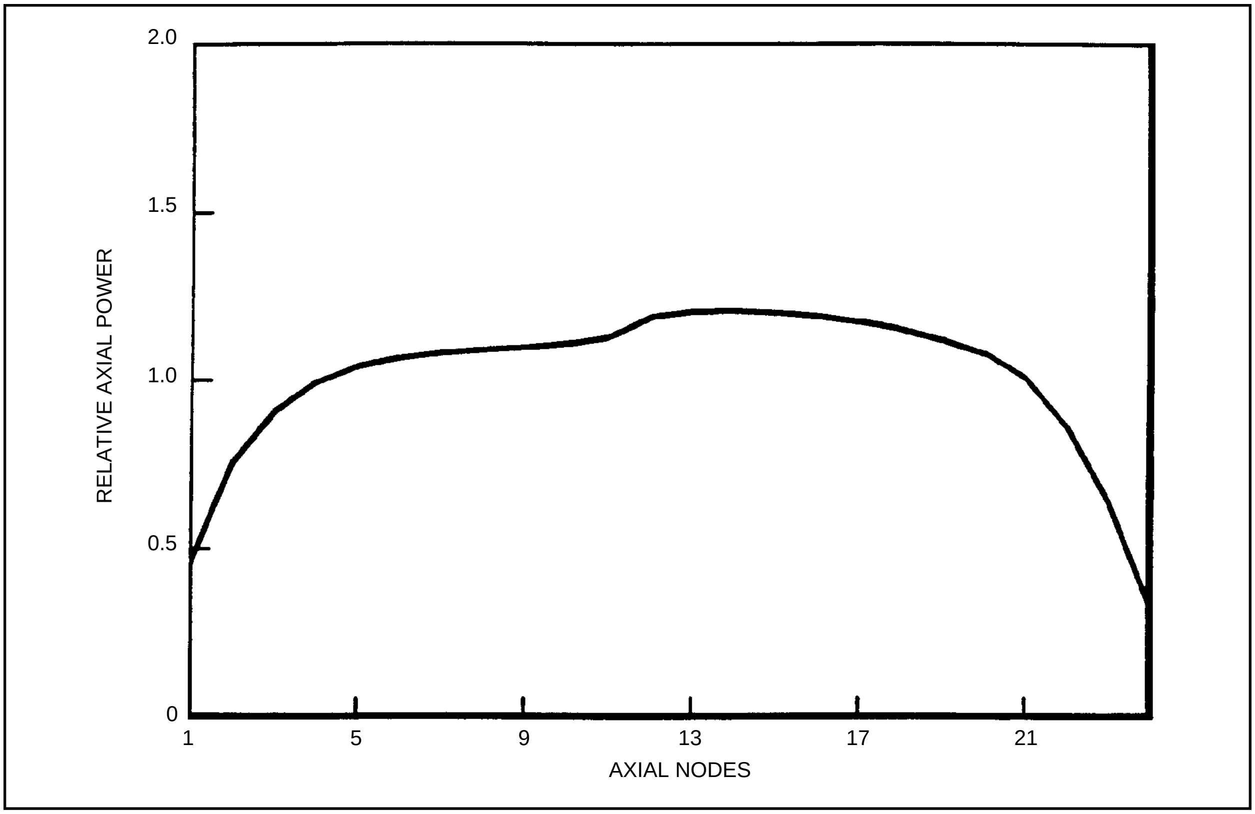


Figure 4A-6b Relative Axial Power at 3.3 GWd/t Cycle Exposure

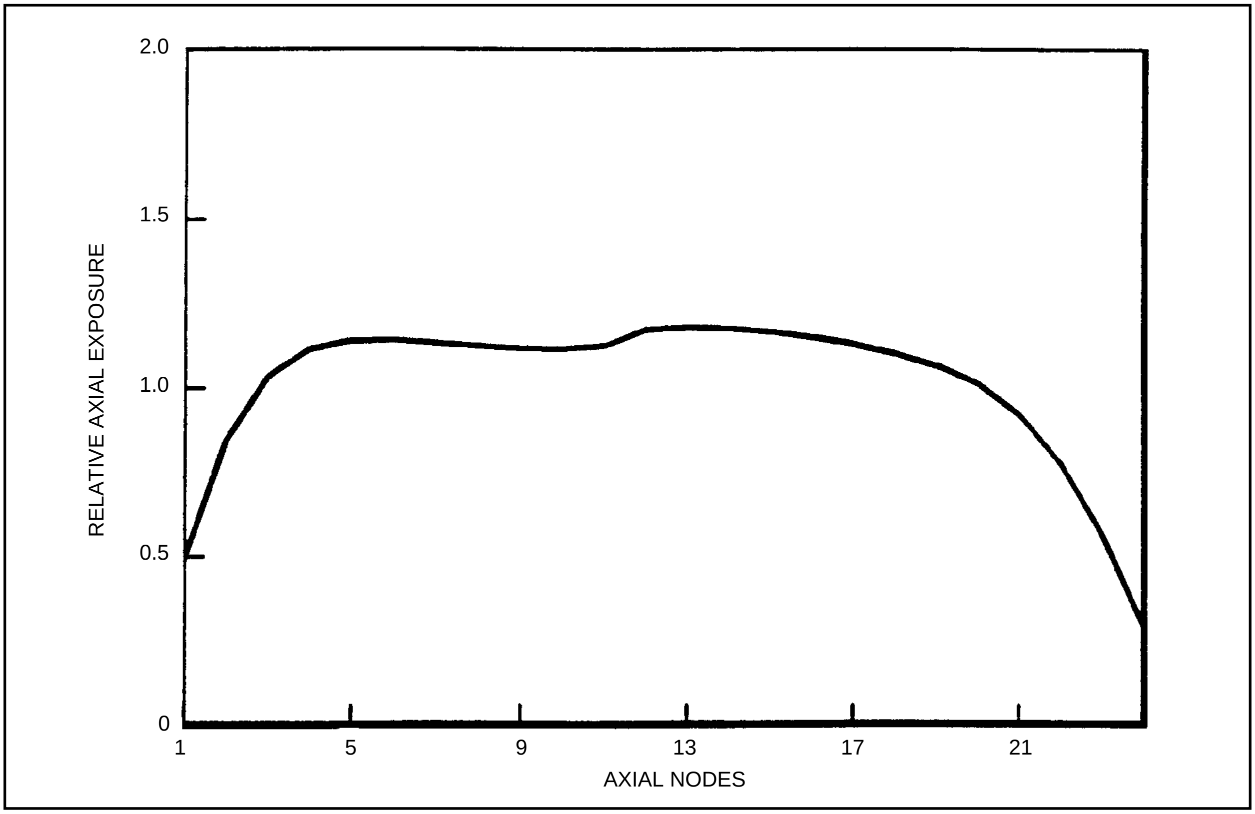


Figure 4A-6c Relative Axial Exposure at 3.3 GWd/t Cycle Exposure

Figure 4A-6d [Proprietary information not included DCD (Refer to SSAR Appendix 4A, Amendment 34)]

Figure 4A-6e [Proprietary information not included DCD (Refer to SSAR Appendix 4A, Amendment 34)]

Figure 4A-7a [Proprietary information not included DCD (Refer to SSAR Appendix 4A, Amendment 34)]

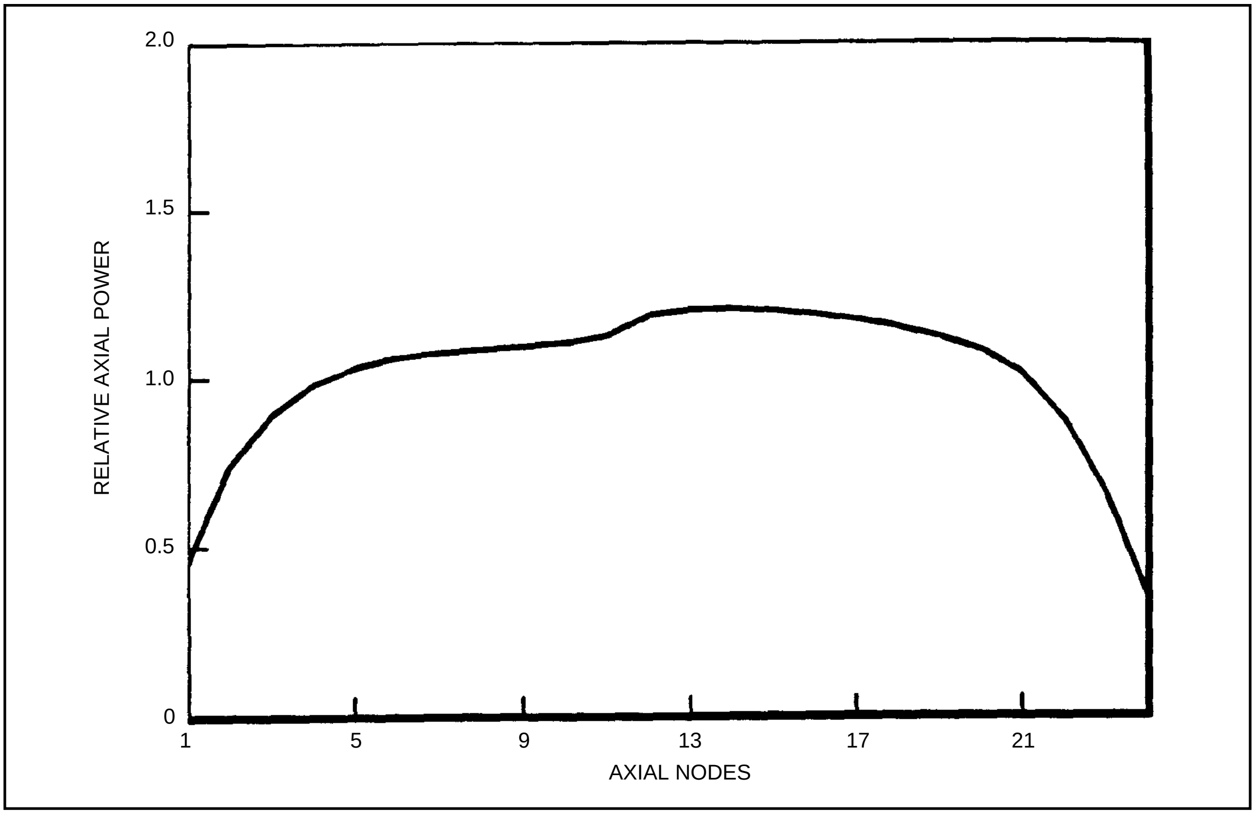


Figure 4A-7b Relative Axial Power at 4.4 GWd/t Cycle Exposure

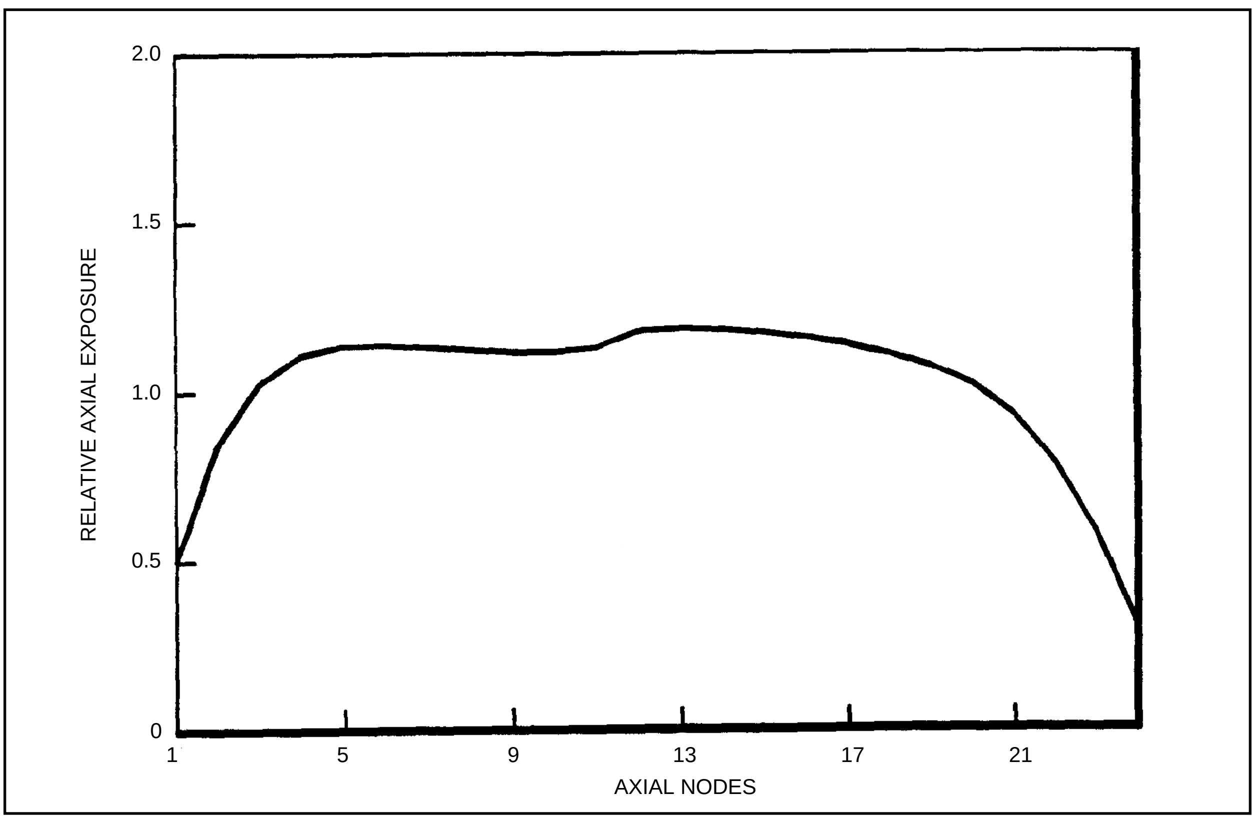


Figure 4A-7c Relative Axial Exposure at 4.4 GWd/t Cycle Exposure

Figure 4A-7d [Proprietary information not included DCD (Refer to SSAR Appendix 4A, Amendment 34)]

Figure 4A-7e [Proprietary information not included DCD (Refer to SSAR Appendix 4A, Amendment 34)]

Figure 4A-8a [Proprietary information not included DCD (Refer to SSAR Appendix 4A, Amendment 34)]

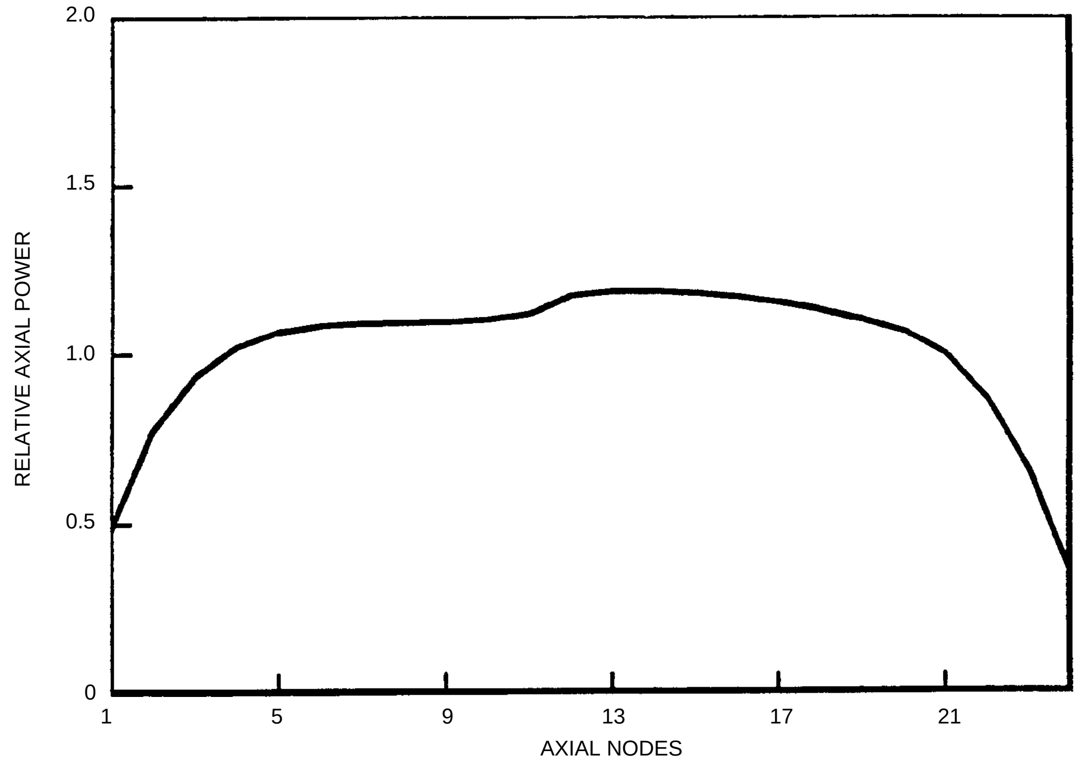


Figure 4A-8b Relative Axial Power at 5.5 GWd/t Cycle Exposure

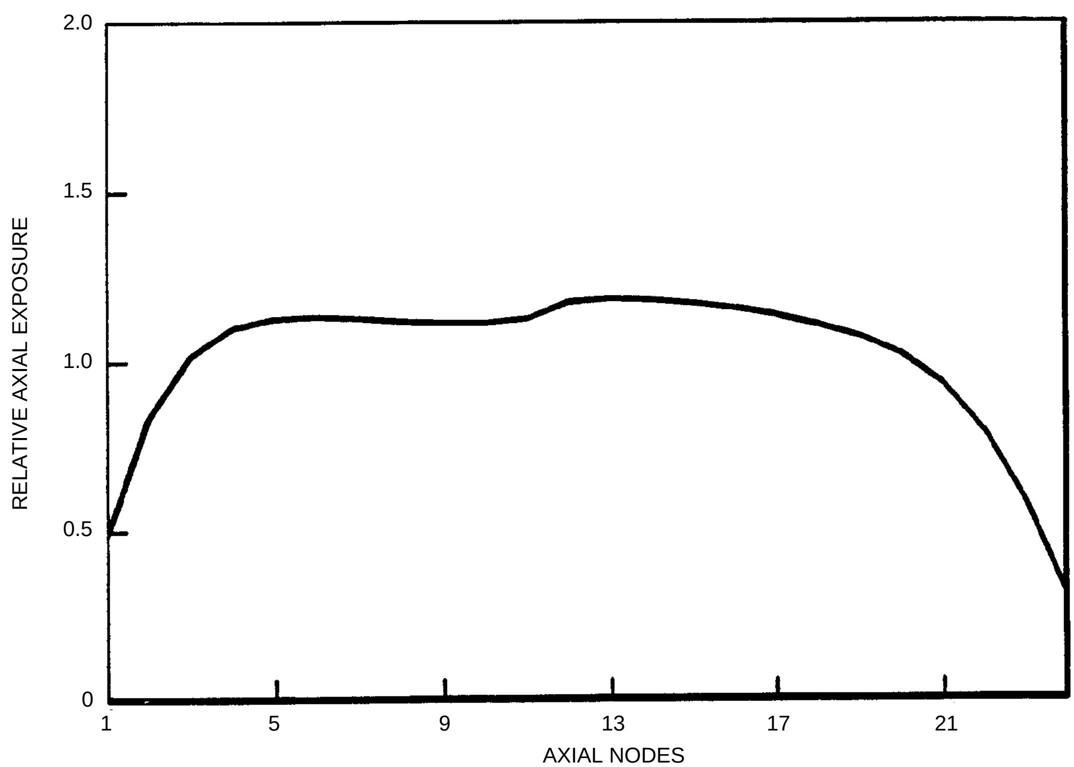


Figure 4A-8c Relative Axial Exposure at 5.5 GWd/t Cycle Exposure

Figure 4A-8d [Proprietary information not included DCD (Refer to SSAR Appendix 4A, Amendment 34)]

Figure 4A-8e [Proprietary information not included DCD (Refer to SSAR Appendix 4A, Amendment 34)]

Figure 4A-9a [Proprietary information not included DCD (Refer to SSAR Appendix 4A, Amendment 34)]

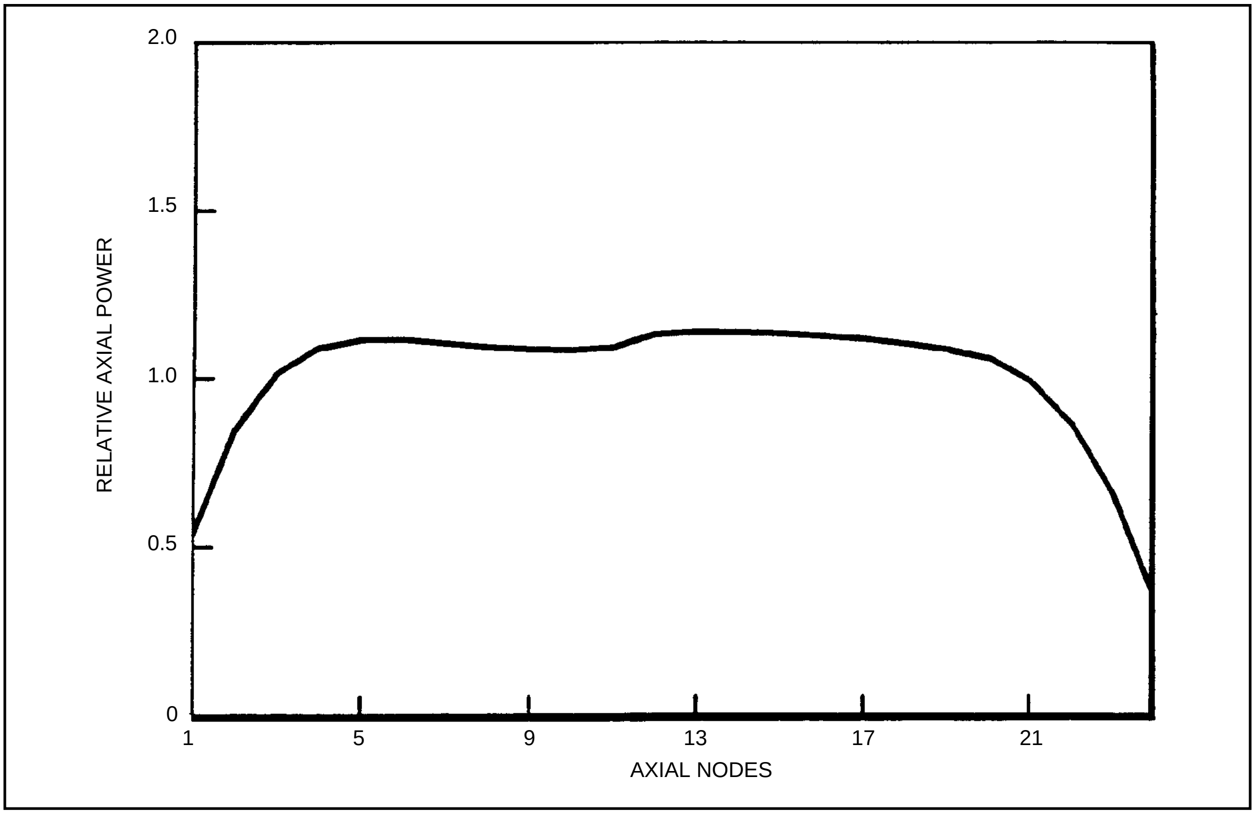


Figure 4A-9b Relative Axial Power at 6.6 GWd/t Cycle Exposure

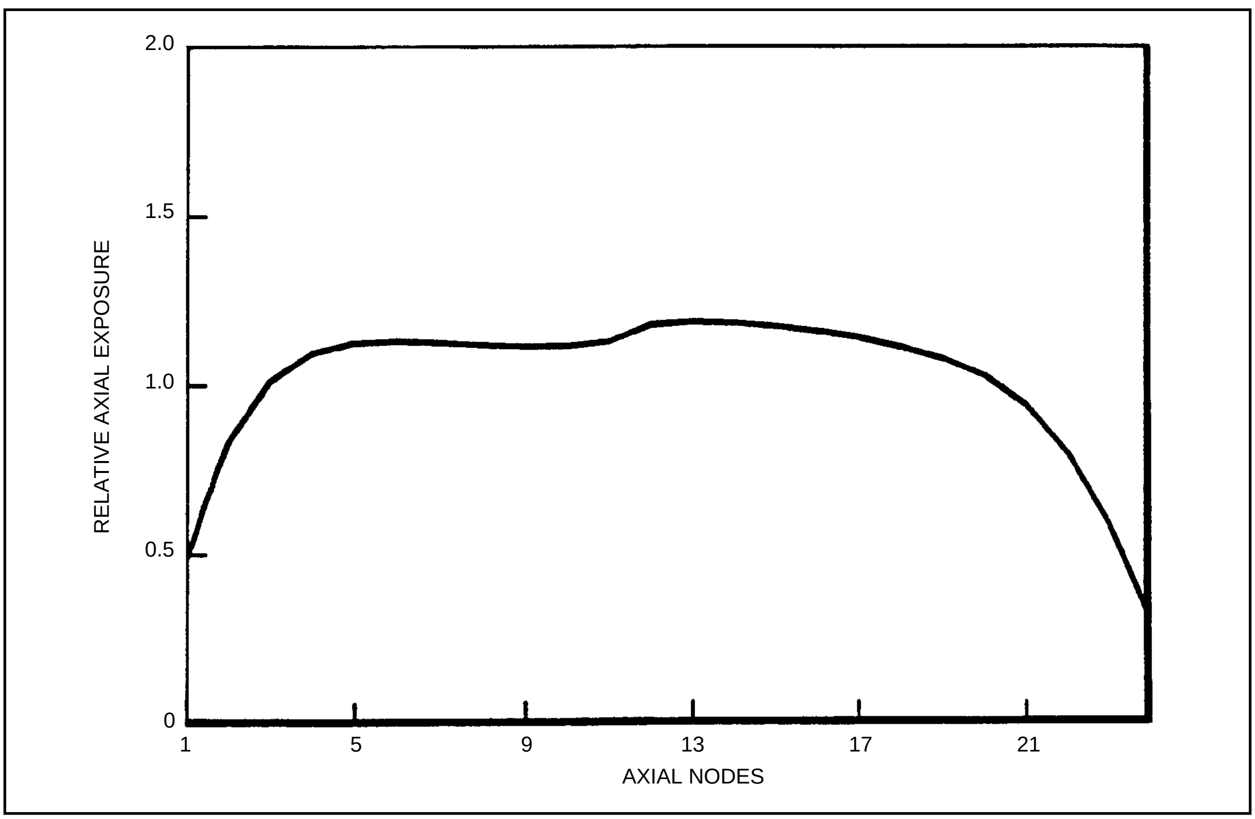


Figure 4A-9c Relative Axial Exposure at 6.6 GWd/t Cycle Exposure

Figure 4A-9d [Proprietary information not included DCD (Refer to SSAR Appendix 4A, Amendment 34)]

Figure 4A-9e [Proprietary information not included DCD (Refer to SSAR Appendix 4A, Amendment 34)]

Figure 4A-10a [Proprietary information not included DCD (Refer to SSAR Appendix 4A, Amendment 34)]

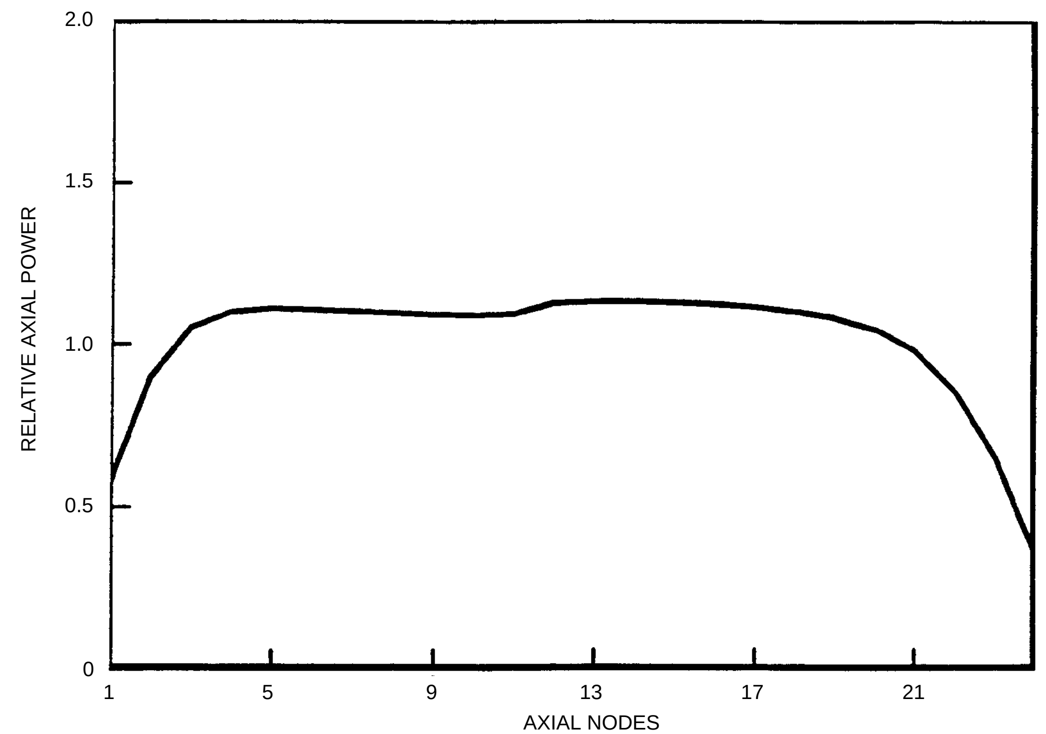


Figure 4A-10b Relative Axial Power at 7.7 GWd/t Cycle Exposure

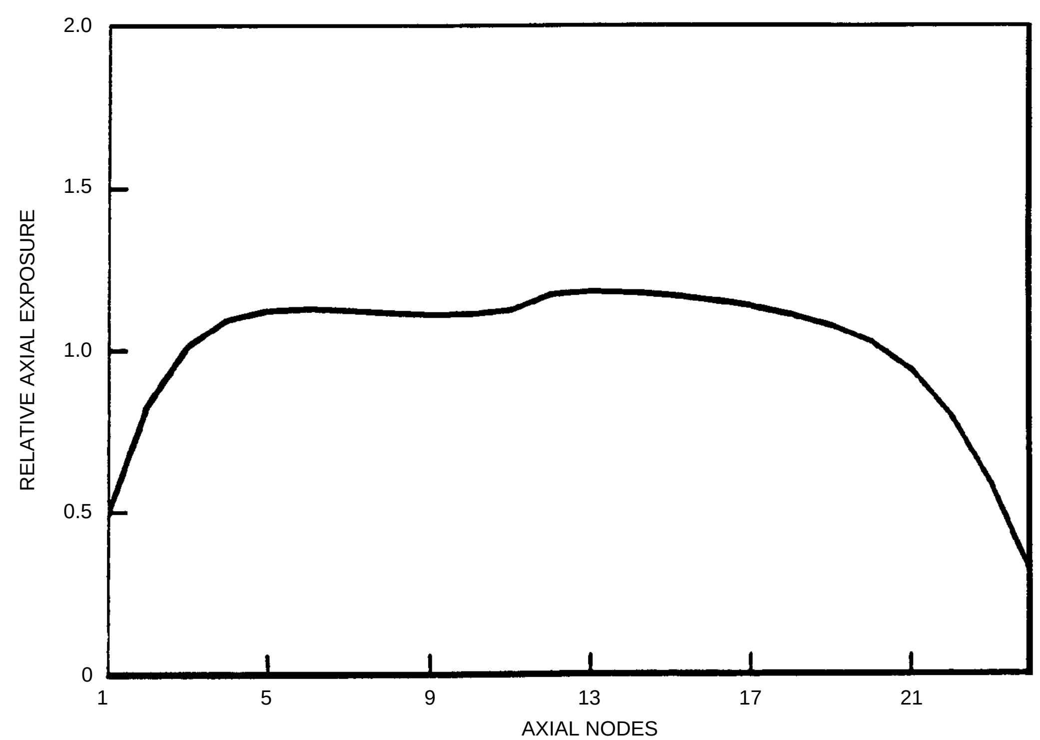


Figure 4A-10c Relative Axial Exposure at 7.7 GWd/t Cycle Exposure

Figure 4A-10d [Proprietary information not included DCD (Refer to SSAR Appendix 4A, Amendment 34)]

Figure 4A-10e [Proprietary information not included DCD (Refer to SSAR Appendix 4A, Amendment 34)]

Figure 4A-11a [Proprietary information not included DCD (Refer to SSAR Appendix 4A, Amendment 34)]

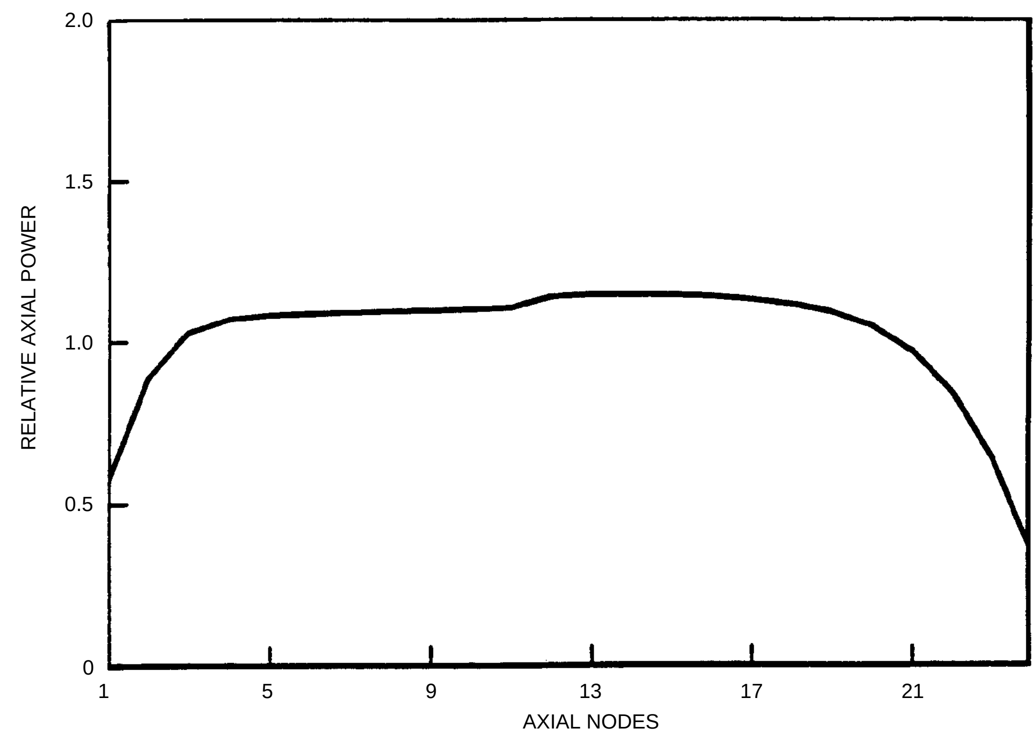


Figure 4A-11b Relative Axial Power at 8.0 GWd/t Cycle Exposure

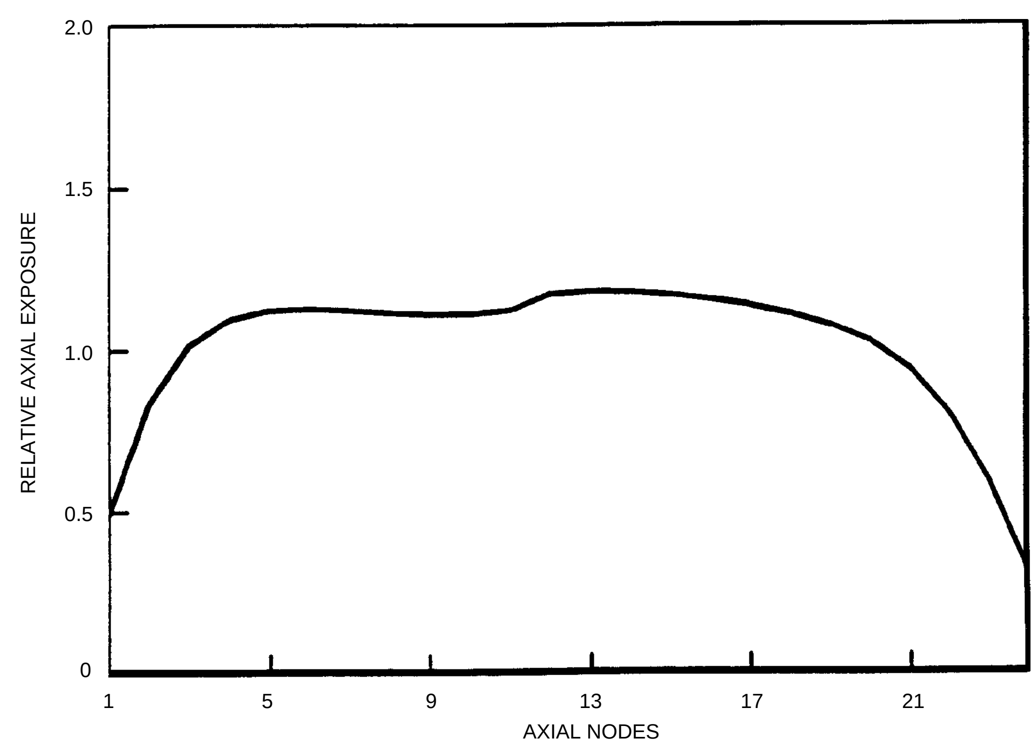


Figure 4A-11c Relative Axial Exposure at 8.0 GWd/t Cycle Exposure

Figure 4A-11d [Proprietary information not included DCD (Refer to SSAR Appendix 4A, Amendment 34)]

Figure 4A-11e [Proprietary information not included DCD (Refer to SSAR Appendix 4A, Amendment 34)]

Figure 4A-12a [Proprietary information not included DCD (Refer to SSAR Appendix 4A, Amendment 34)]

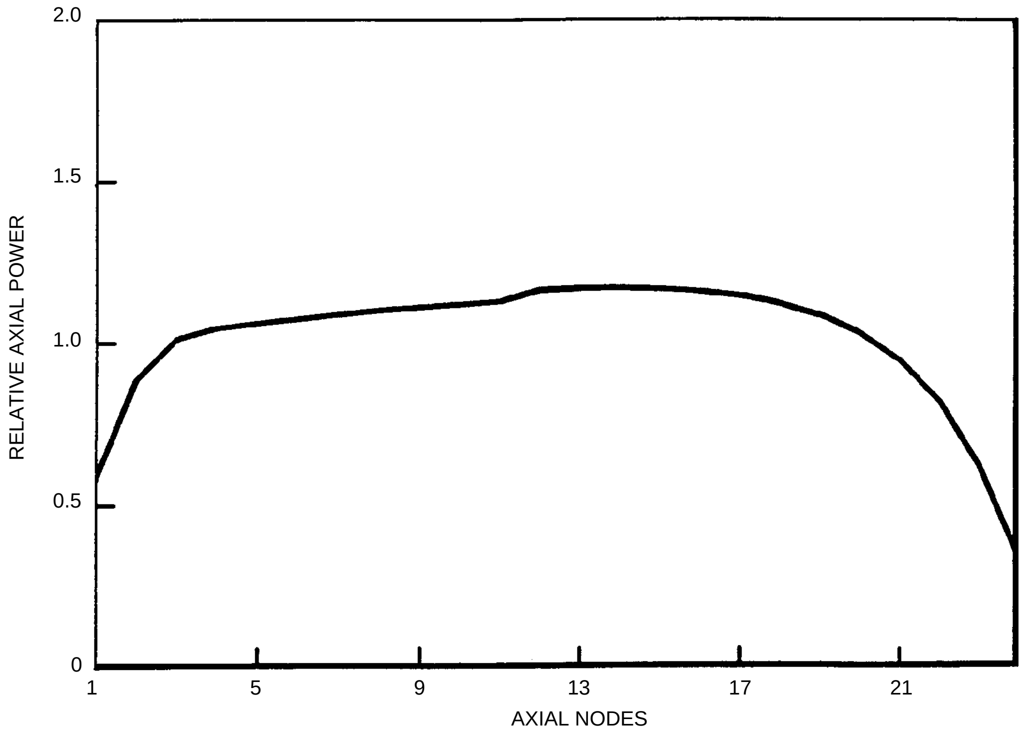


Figure 4A-12b Relative Axial Power at 8.4 GWd/t Cycle Exposure

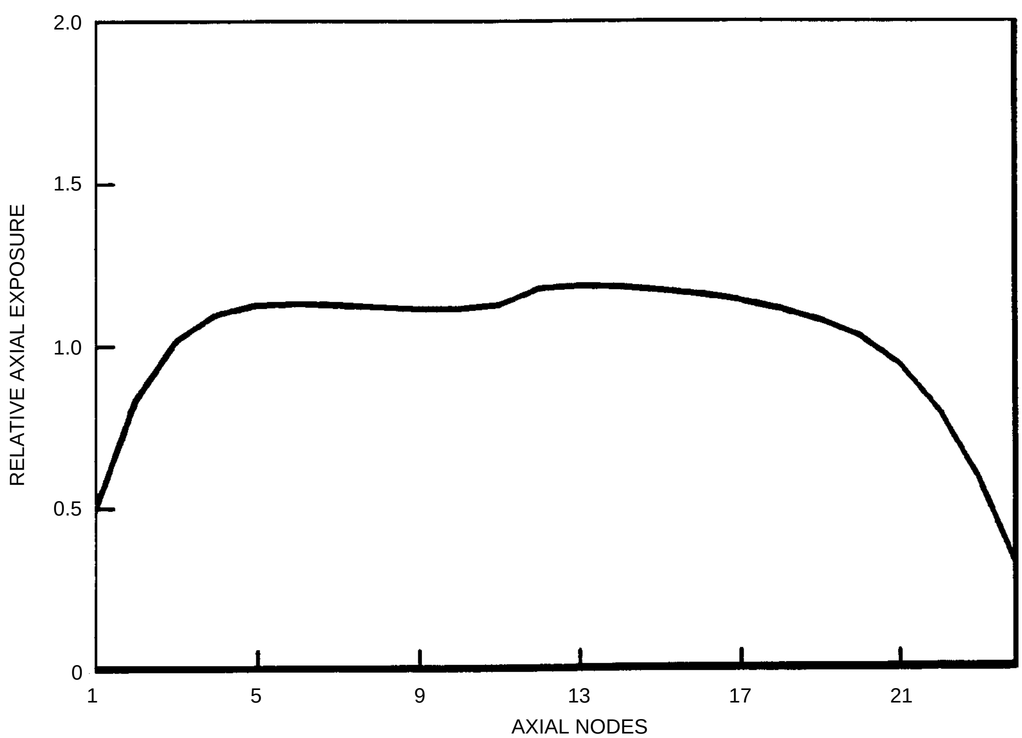


Figure 4A-12c Relative Axial Exposure at 8.4 GWd/t Cycle Exposure

Figure 4A-12d [Proprietary information not included DCD (Refer to SSAR Appendix 4A, Amendment 34)]

Figure 4A-12e [Proprietary information not included DCD (Refer to SSAR Appendix 4A, Amendment 34)]

Figure 4A-13a [Proprietary information not included DCD (Refer to SSAR Appendix 4A, Amendment 34)]

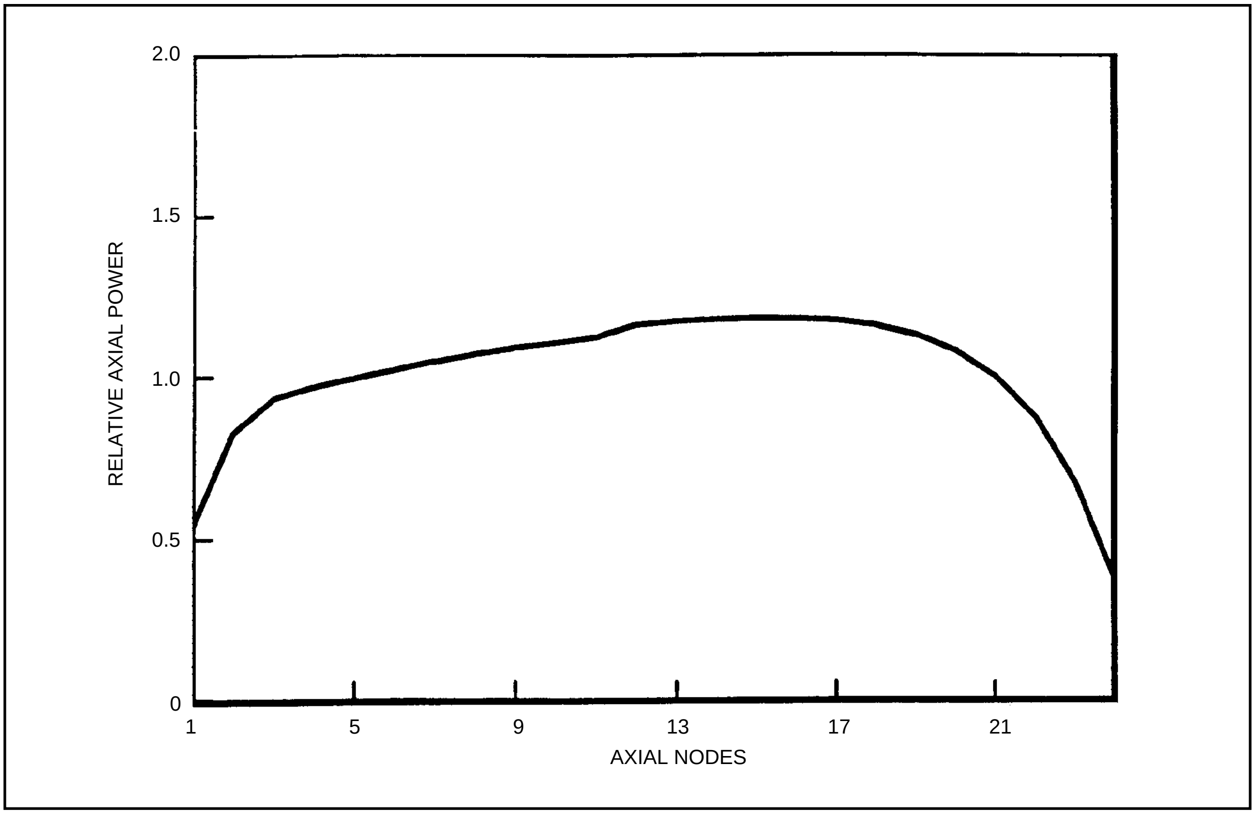


Figure 4A-13b Relative Axial Power at 9.0 GWd/t Cycle Exposure

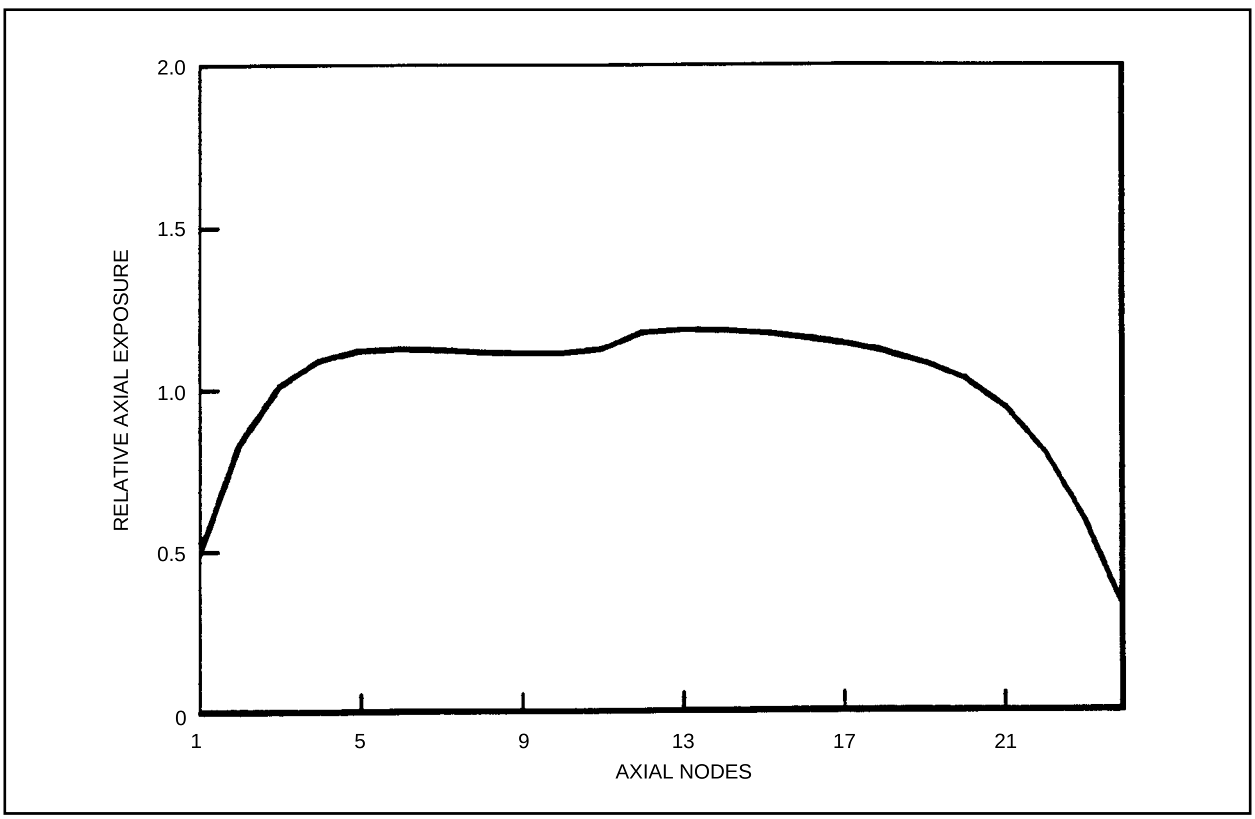


Figure 4A-13c Relative Axial Exposure at 9.0 GWd/t Exposure

Figure 4A-13d [Proprietary information not included DCD (Refer to SSAR Appendix 4A, Amendment 34)]

Figure 4A-13e [Proprietary information not included DCD (Refer to SSAR Appendix 4A, Amendment 34)]

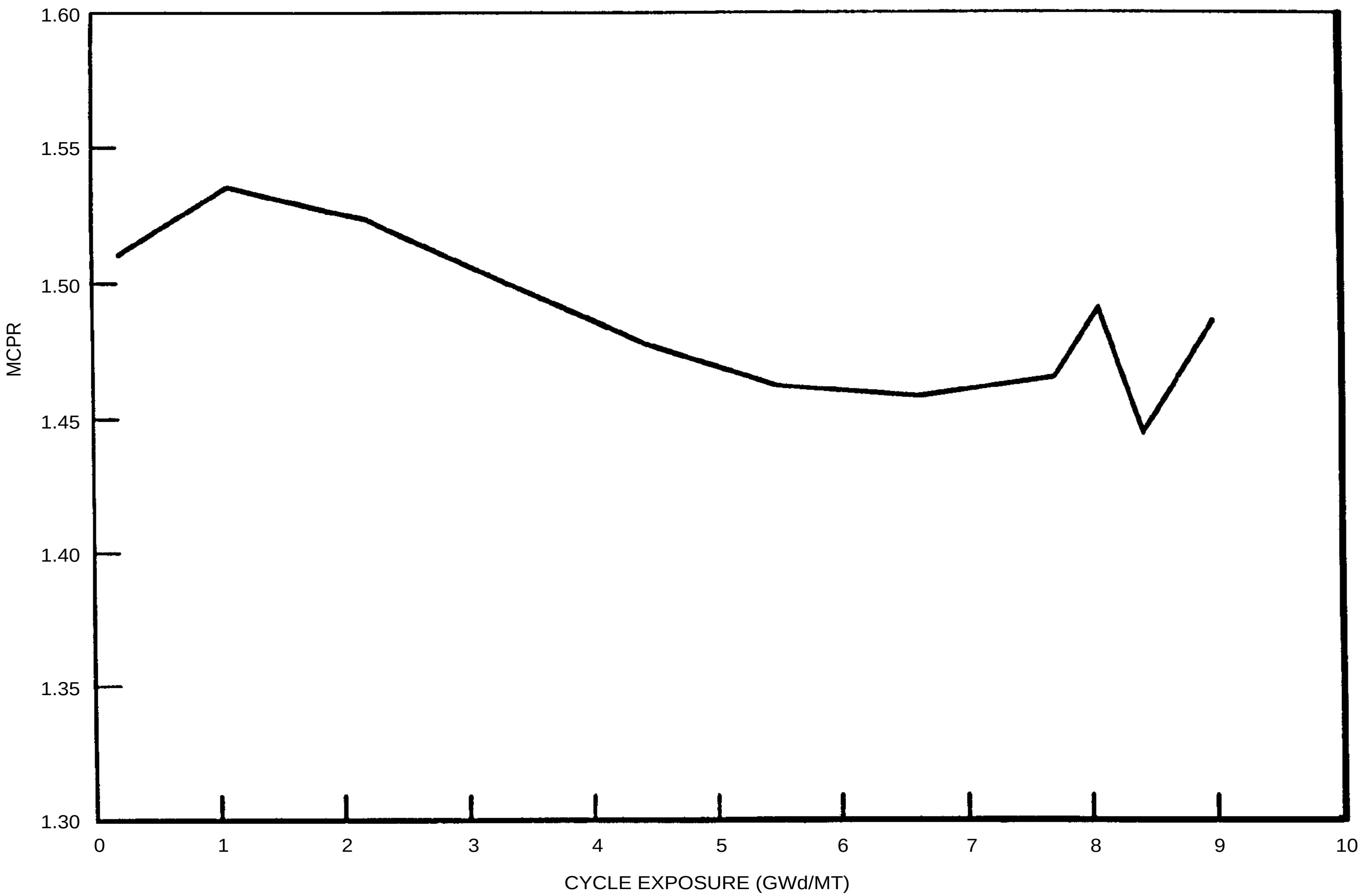


Figure 4A-14 Minimum Critical Power Ratio (MCPR) as a Function of Cycle Exposure

4B Fuel Licensing Acceptance Criteria

4B.1 Introduction

A set of fuel licensing acceptance criteria has been established for evaluating fuel designs and for determining the applicability of generic analyses to these designs. Fuel design compliance with the fuel licensing acceptance criteria constitutes USNRC acceptance and approval of the fuel design for initial core and reload applications without specific USNRC review. [*The fuel licensing acceptance criteria are presented in Reference 4B-1. Any change to these criteria must have prior NRC review and approval.*]*

4B.2 References

4B-1 [*SSAR Sections 4B.2 through 4B.16, Proprietary Information, Amendment 35*]*

* See Section 4.2.

4C Control Rod Licensing Acceptance Criteria

4C.1 Introduction

A set of acceptance criteria has been established for evaluating new control rod designs. Control rod compliance with these criteria constitutes the basis for USNRC acceptance and approval of the design. [*The control rod licensing acceptance criteria and their bases are provided below. Any change to these criteria must have prior NRC review and approval.*]*

4C.2 [*General Criteria*

Control rod designs must meet the following acceptance criteria:

(1) *The control rod stresses, strains, and cumulative fatigue shall be evaluated to not exceed the ultimate stress or strain of the material.*

(2) *The control rod shall be evaluated to be capable of insertion into the core during all modes of plant operation within the limits assumed in the plant analyses.*

(3) *The material of the control rod shall be shown to be compatible with the reactor environment.*

(4) *The reactivity worth of the control rod shall be included in the plant core analyses.*

(5) *A surveillance program shall be implemented if a change in design features such as new absorber material or structural material not previously used in reactor cores could impact the function of the control rod.*

4C.3 Basis for Acceptance Criteria

The following comprise the basis for the licensing acceptance criteria given in Section 4C.2.

4C.3.1 Stress, Strain and Fatigue

The control rod is evaluated to assure that it does not fail because of loads due to shipping, handling, and normal, abnormal, emergency, and faulted operating modes. To assure that the control rod does not fail, these loads must not exceed the ultimate stress and strain limit of the material including irradiation effects for its design life. Fatigue must not exceed a fatigue usage factor of 1.0.

The loads evaluated include those due to normal operational transients (scram and jogging), pressure differentials, thermal gradients, flow- and system-induced vibration, and irradiation growth in addition to the lateral and vertical loads expected for each condition. Fatigue usage is

* See Section 4.2.

based upon the cumulative effect of the cyclic loadings. The analyses include corrosion and crud deposition as a function of time as appropriate.

Conservatism is included in the analyses by including margin to the limit or by assuming loads greater than expected for each condition. Higher loads can be incorporated into the analyses by increasing the load itself or by statistically considering the uncertainties in the value of the load.

4C.3.2 Control Rod Insertion

The control rod is evaluated to be sure that it can be inserted during normal, abnormal, emergency and faulted modes of operation, to include the safe shutdown earthquake event combined with LOCA event within the limits assumed in the plant analyses and throughout its design life. These evaluations include a combination of analyses of the geometrical clearance and actual testing. The analyses consider the effects of manufacturing tolerances, swelling and irradiating growth. Tests may be performed to demonstrate control rod insertion capability for conditions such as control rod or fuel channel deformation and vibrations due to safe shutdown earthquakes.

4C.3.3 Control Rod Material

The external control rod materials must be capable of withstanding the reactor coolant environment for the design life of the control rod. Effects of crudding, crevices, stress corrosion and irradiation upon the material must be included in the control rod and core evaluations. Irradiation effects to be considered include material hardening and absorber depletion and swelling.

4C.3.4 Reactivity

The reactivity worth of the control rod is determined by the initial amount and type of absorber material and irradiation depletion. Scram time insertion performance and control rod drop times affect the total reactivity inserted into the core. All of these effected must be included in the plant core analyses including nuclear, abnormal operational occurrences, infrequent events, and accidents. The reactivity worth of the rod must provide, under conditions of normal operation (including abnormal operational occurrences), appropriate margin for malfunctions, such as two stuck control rods or accidental control rod withdrawal, without exceeding specified acceptable fuel design limits.

4C.3.5 Surveillance Criteria

Visual inspection of the lead depletion control rod design possessing the new design feature and three additional control rods of such design that are within 15% of the estimated fast fluence of the lead control rod shall be performed. If fewer than three control rods are within 15% of the estimated fast fluence of the lead control rod, only those within 15% shall be inspected. Should evidence of a problem arise, arrangements will be made to inspect additional control rods to the extent necessary to identify the root cause of the problem.]*

* See Section 4.2.

4D Reference Fuel Design Compliance with Acceptance Criteria

[Proprietary information not included in DCD (Refer to SSAR Appendix 4D, Amendment 35).]