SECTION 3.0 - REACTOR

TABLE OF CONTENTS

TABLE OF CONTENTS (cont'd)

TABLE OF CONTENTS (cont'd)

TABLE OF CONTENTS (cont'd)

SECTION 3.0 - REACTOR

LIST OF TABLES

LIST OF FIGURES (cont'd)

FIGURE TITLE

SECTION 3.0 - REACTOR

LIST OF FIGURES

LIST OF FIGURES (cont'd)

SECTION 3.0

REACTOR

3.1 SUMMARY DESCRIPTION

Section 3.0, "Reactor," describes and evaluates those systems most pertinent to the fuel barrier and the control of core reactivity.

Subsection 3.2, "Fuel Mechanical Design," describes the mechanical aspects of the fuel material (uranium dioxide), the fuel cladding, the fuel rods, and the arrangement of fuel rods in bundles. Of particular interest is the ability of the fuel to serve as the initial barrier to the release of radioactive material. The mechanical design of the fuel is sufficient to prevent the escape of significant amounts of radioactive material during normal modes of reactor operation.

Subsection 3.3, "Reactor Vessel Internals Mechanical Design," describes both the arrangements of the supporting structure for the core and the reactor vessel internal components which are provided to properly distribute the coolant delivered to the reactor vessel. In addition to their main function of coolant distribution, the reactor vessel internals separate the moisture from the steam leaving the vessel and provide a floodable inner volume inside the reactor vessel that allows sufficient submergence of the core, under accident conditions, to prevent the gross release of fission products from the fuel. The reactor vessel internals are designed to allow the control rods and CSCS's to perform their safety functions during abnormal operational transients and accidents.

Subsection 3.4, "Reactivity Control Mechanical Design," describes the mechanical aspects of the moveable control rods which are provided to control core reactivity. The CRD hydraulic system is designed so that sufficient energy is available to force the control rods into the core under conditions associated with abnormal operational transients and accidents. Control rod insertion speed is sufficient to prevent fuel damage as a result of any abnormal operation transient.

Control rod housing supports, described in subsection 3.5, are located underneath the reactor vessel near the control rod housings. These supports limit the travel of a control rod in the event that a control rod housing is ruptured.

The supports prevent a nuclear excursion as a result of a housing failure, thus protecting the fuel barrier.

Subsection 3.6, "Nuclear Design," describes the nuclear aspects of the reactor core. The design of the BWR core and fuel is based on a proper combination of design variables, such as moderator-to-fuel volume ratio, core power density, thermal-hydraulic characteristics, fuel exposure level, nuclear characteristics of the core and fuel, heat transfer, flow distribution, void content, heat flux, and operating pressure. All of these conditions are dynamic functions of operating conditions. However, design analyses and calculations, verified by comparison with data from operating plants, are usually performed for specific steady-state conditions. Included in this subsection are summaries of results of these analyses for the fuel cycle, reactivity control, and control rod worths. Also included are discussions of the reactivity coefficients and spatial xenon characteristics of the core.

Subsection 3.7, "Thermal and Hydraulic Design," describes the thermal and hydraulic characteristics of the core. The low coolant saturation temperature, high heat transfer coefficient, and neutral water chemistry of the BWR are significant advantages in minimizing Zircaloy temperatures and associated temperature-dependent hydride pickup. This results in improved fuel cladding performance at long exposures. The relatively uniform fuel cladding temperatures throughout the BWR core minimize migration of the hydrides to cold cladding zones and reduce the thermal stresses. A discussion of fuel failure mechanisms and the parameters associated with fuel damage is included.

The standby liquid control system, described in subsection 3.8, provides a different method of reactor shutdown which is redundant with, but independent of, the control rods. While insertion of only a few of the many independent control rods assures prompt shutdown of the reactor, the standby liquid control system can maintain subcriticality as the reactor cools without reliance upon insertion of any control rods.

3.2 FUEL MECHANICAL DESIGN

3.2.1 Power Generation Objective

The power generation objective of the nuclear fuel is to provide a high integrity assembly of fissionable material which can be arranged in a critical array. The assembly must be capable of efficiently transferring the generated fission heat to the circulating coolant water while maintaining structural integrity and containing the fission products.

3.2.2 Power Generation Design Basis

The nuclear fuel is designed to assure (in conjunction with the core nuclear characteristics, the core thermal and hydraulic characteristics, the plant equipment characteristics, and the capability of the nuclear instrumentation and RPS) that fuel damage limits will not be exceeded during either normal operation or anticipated operational occurrences.

3.2.3 Safety Design Basis

In meeting the power generation objective, the nuclear fuel is utilized as the initial barrier to the release of fission products. The fuel shall be designed so as to comply with the applicable nuclear safety design criteria specified in Sections 1.5.1.4 and 1.5.1.5.

3.2.4 Description

The description of the mechanical aspects of the fuel material, fuel cladding, fuel rods and the arrangement of the fuel rods in the bundles, including the fuel thermal-mechanical and safety analyses for GE fuel products can be found in NEDO-24011-P-A (GESTAR II) and NEDE-31152P (General Electric Fuel Bundle Designs); References 3.2.5.1 and 3.2.5.2 respectively

3.2.5 References

1. "Licensing Topical Report General Electric Standard Application for Reactor Fuel", GE Company Document No. NEDO-24011-P-A, (Latest approved revision).

2. "Global Nuclear Fuels Fuel Bundle Designs", GE Company Document No. NEDE-31152P, (Latest approved revision).

3.3 REACTOR VESSEL INTERNALS MECHANICAL DESIGN

3.3.1 Power Generation Objective

Reactor vessel internals (exclusive of fuel, control rods, and in-core flux monitors) are provided to achieve the following power generation objectives:

1. Maintain partitions between regions within the reactor vessel to provide proper coolant distribution, thereby allowing power operation without fuel damage due to inadequate cooling.

2. Provide positioning and support for the fuel assemblies, control rods, in-core flux monitors, and other vessel internals to assure that control rod movement is not impaired.

3.3.2 Power Generation Design Basis

1. The reactor vessel internals are designed to provide proper coolant distribution during all anticipated normal operating conditions to allow power operation of the core without fuel damage.

2. The reactor vessel internals are arranged to facilitate refueling operations.

3. Adequate working space and access are provided to permit adequate inspection of reactor vessel internals.

3.3.3 Safety Design Basis

The reactor vessel internals mechanical design assures that safety design bases 1 and 2 are satisfied so that the safe shutdown of the plant and removal of decay heat are not impaired.

1. The reactor vessel internals are arranged to provide a floodable volume in which the core can be adequately cooled in the event of a breach in the nuclear system process barrier external to the reactor vessel.

2. Deflections and deformation of reactor vessel internals are limited to assure that the control rods and the Core Standby Cooling System's (CSCS's) can perform their safety functions during abnormal operational transients and accidents.

3.3.4 Description

The reactor vessel internals are installed to properly distribute the flow of coolant delivered to the vessel, to locate and support the fuel assemblies, and to provide an inner volume containing the core (e.g. 2/3 core height) that can be flooded following a break in the nuclear system process barrier external to the reactor vessel. The reactor vessel internals include the following components:

Core shroud and shroud support

Shroud head and steam separator assembly

Core support (core plate)

Top guide

Fuel support pieces

Control rod guide tubes

Jet pump assemblies

Steam dryers

Feedwater spargers

Core spray lines and spargers

Differential pressure and liquid control line

In-core flux monitor guide tubes

Surveillance sample holders

The overall arrangement of the internals within the reactor vessel is shown in Figure 3.3.1. Table 3.3.1 gives detailed design data for the various reactor vessel internals.

The design of the reactor vessel internals is in accordance with the intent of Section III of the ASME Boiler and Pressure Vessel Code. The material used for fabrication of most of the reactor vessel internals is solution heat-treated, unstabilized type 304 austenitic stainless steel conforming to ASTM specifications. The jet pump inlet—mixer (replacement) wedge is fabricated from alloy X-750. The material receives a “high temperature anneal” heat treatment, which reduces its susceptibility to IGSCC. It is also age hardened to avoid galling. The jet pump bolt and keeper nut are fabricated from low carbon type 316 stainless steel. These materials meet the GE requirements for in-reactor use and the requirements of BWRVIP-84. Weld procedures and welders are qualified in accordance with the intent of Section IX of the ASME Boiler and Pressure Vessel Code.

The core structure was fabricated by Rotterdam Dockyard Company in Rotterdam, Holland.

The Rotterdam Dockyard Company is fully qualified to fabricate core structures with adequate machining, handling, and welding equipment. Rotterdam has qualified fabrication and quality control organizations and a system capable of assuring and documenting the required quality level.

These qualifications are supported by Rotterdam's extensive experience in core structure fabrication with such domestic plants as Browns Ferry I, II, and III, Monticello, and Vermont Yankee. Rotterdam has also fabricated parts of the Quad Cities II reactor pressure vessel as well as complete pressure vessels for foreign BWR plants, such as AKM and Nuclenor, and domestic PWR plants.

In addition, 2-in and smaller stainless steel pipe was manufactured by Sandvik Steel, Incorporated (Sweden). The material conforms to the requirements of ASTM A 312/A 379 Type 304.

The floodable inner volume of the reactor vessel is the volume inside the core shroud up to the level of the jet pump nozzles. The boundary of the inner volume consists of the following:

1. The jet pumps from the jet pump nozzles down to the shroud support.

2. The shroud support, which forms a barrier between the outside of the shroud and the inside of the reactor vessel.

3. The reactor vessel wall below the shroud support.

4. The core shroud up to the level of the jet pump nozzles.

3.3.4.1 Core Structure

The core structure surrounds the core of the reactor and consists of the core shroud, shroud head and steam separator assembly, core support, and top guide. This structure is used to form partitions within the reactor vessel, to sustain pressure differentials across the partitions, to direct the flow of the coolant water, and to laterally locate and support the fuel assemblies, control rod guide tubes, and steam separators. Figure 3.3.2 shows the reactor vessel internal flow paths.

3.3.4.1.1 Core Shroud and Shroud Support

The core shroud is a stainless steel cylindrical assembly which provides a partition to separate the upward flow of coolant through the core from the downward recirculation flow. This partition separates the core region from the downcomer annulus. The volume enclosed by the core shroud is characterized by three regions, each with a different shroud diameter. The upper shroud has the largest diameter and surrounds the core discharge plenum which is bounded by the shroud head on top and the top fuel guide below. The central portion of the shroud surrounds the fuel and forms the longest section of the shroud. This section has an intermediate diameter and is bounded at the bottom by the core support assembly. The lower shroud, surrounding part of the lower plenum, has the smallest diameter and, at the bottom, is welded to the reactor vessel shroud support (subsection 4.2, "Reactor Vessel and Appurtenances Mechanical Design").

3.3.4.1.2 Shroud Head and Steam Separator Assembly

The shroud head and steam separator assembly is bolted to the top of the upper shroud to form the top of the core discharge plenum. This plenum provides a mixing chamber for the steam-water mixture before it enters the steam separators. The individual stainless steel axial flow steam separators shown in Figure 3.3.3 are attached to the top of standpipes which are welded into the shroud head. The steam separators have no moving parts. In each separator, the steam-water mixture, rising through the standpipe, passes turning vanes which impart a spin to establish a vortex separating the water from the steam. The steam exits from the top of the separator and rises up to the dryers. The separated water exits from under the separator cap and flows out between the standpipes, draining into the recirculation flow downcomer annulus.

3.3.4.1.3 Core Support (Core Plate)

The core support assembly consists of a circular stainless steel plate stiffened with a rim and beam structure. Perforations in the plate provide lateral support and guidance for the control rod guide tubes, peripheral fuel support pieces, in-core flux monitor guide tubes, and startup neutron sources. The entire assembly is bolted to a support ledge, between the central and lower portions of the core shroud, after proper positioning has been assured by alignment pins which fit into slots in the ledge.

3.3.4.1.4 Top Guide

The top guide is formed by a series of stainless steel beams joined at right angles to form square openings. Each opening provides lateral support and guidance for four fuel assemblies. Holes are provided in the bottom of the beams to anchor the in-core flux monitor guide tubes and startup neutron sources. The top guide is positioned by alignment pins which fit into radial slots of plates which are attached to the support ledge between the upper and central portions of the core shroud.

3.3.4.2 Fuel Support Pieces

The fuel support pieces, shown in Figure 3.3.4, are of two basic types: peripheral and four-lobed. The peripheral fuel support pieces, which are welded to the core support assembly, are located at the outer edge of the core and are not adjacent to control rods. Each peripheral fuel support piece supports one fuel assembly and contains a replaceable orifice assembly designed to assure proper coolant flow to the fuel assembly. The four-lobed fuel support pieces each support four fuel assemblies and are provided with orifice plates to assure proper coolant flow distribution to each fuel assembly. The four-lobed fuel support pieces rest in the top of the control rod guide tubes and are supported laterally by the core support. The control rods pass through slots in the center of the four-lobed fuel support pieces. A control rod and the four fuel assemblies which immediately surround it represent a control cell (subsection 3.6, "Nuclear Design").

3.3.4.3 Control Rod Guide Tubes

The control rod guide tubes, located inside the vessel, extend from the top of the CRD housings through the core support. Each tube is designed as the lateral guide for a control rod and as the vertical support for a four-lobed fuel support piece and the four fuel assemblies surrounding the control rod. The bottom of the guide tube is supported by the CRD housing (subsection 4.2, "Reactor Vessel and Appurtenances Mechanical Design"), which, in turn, transmits the weight of the guide tube, fuel support piece, and fuel assemblies to the reactor vessel bottom head. A thermal sleeve is inserted into the CRD housing from below and is rotated to lock the control rod guide tube in place. A key is inserted into a locking slot in the bottom of the CRD housing to hold the thermal sleeve in position.

3.3.4.4 Jet Pump Assemblies

The jet pump assemblies are located in two semicircular groups in the downcomer annulus between the core shroud and the reactor vessel wall. Each stainless steel jet pump consists of a driving nozzle, suction inlet, throat or mixing section, and diffuser (Figure 3.3.5). The driving nozzle, suction inlet, and throat are joined together as a removable unit and the diffuser is permanently installed. High-pressure water from the recirculation pumps (subsection 4.3, "Reactor Recirculation System") is supplied to each pair of jet pumps through a riser pipe welded to the recirculation inlet nozzle thermal sleeve. A riser brace is welded to cantilever beams extending from pads on the reactor vessel wall.

The jet pump diffuser is a gradual conical section changing to a straight cylindrical section at the lower end. The diffuser is welded to the shroud support. The joint between the throat and the diffuser is a slip/socket fit. A metal-to-metal spherical-to-conical seal joint is used between the nozzle entry section and riser with firm contact maintained by a clamp arrangement which fits under ears on the riser and utilizes a bolt to provide a downward force on a pad on top of the nozzle entry section. The throat section is supported laterally by a bracket attached to the riser. The design permits reflooding the core to the top of the jet pump inlet following a design basis LOCA.

3.3.4.5 Steam Dryers

The steam dryer is a reactor vessel internal component located in the steam dome portion of the RPV. Its function is to dry the steam to a very high quality when it exits the dryer. Although it does not perform a safety function, it must retain its structural integrity to avoid the generation of loose parts that may impact the ability of other structures, systems and components from performing their safety functions.

The original GE parallel vane bank system was not suitable for EPU conditions without modifications. It has been replaced with a Westinghouse 3-ring octagonal shaped vane bank steam dryer during P2R20 for Unit 2 andP3R20 for Unit 3. The Westinghouse Replacement Steam Dryer (RSD) is supported on four brackets attached to the inside wall surface of the reactor pressure vessel. The brackets support the dryer via its support ring. Attached under the support ring is a skirt, which has eight vertical drain channels welded to its inside. At the top of the steam dryer, there are three concentric octagons, each containing eight vane banks. The function of the vane banks is to separate the moisture from the steam flow by letting the steam pass through vertical corrugated plates placed inside the vane banks. Each vane bank has a hood that leads the steam flow into the vane bank. The vane banks stand on troughs (U-shaped channels) that collect and lead the excess water through the girder drain channels and out to the vertical drain channels. A perforated plate is mounted on the inlet side of each vane bank. This ensures an even flow through the vane banks in order to minimize the moisture carryover (MCO) to the main steam system. On top of the vane bank octagons is a web of girders welded to the vane banks for radial support.

3.3.4.6 Feedwater Spargers

The feedwater spargers consist of six stainless steel headers located in the mixing plenum above the downcomer annulus. A separate sparger is fitted to each feedwater nozzle by means of a double seal, triple thermal sleeve assembly and is shaped to conform to the curve of the vessel wall. Sparger end brackets are attached to vessel brackets to support the weight of the spargers. End brackets and wedge blocks position the spargers away from the vessel wall. Feedwater flow enters the center of the spargers and is discharged radially inward through top mounted elbows, each with a converging discharge nozzle, to mix the cooler feedwater with the downcomer flow from the steam separators before it contacts the vessel wall. The feedwater also serves to collapse the steam voids and to subcool the water flowing to the jet pumps and recirculation pumps.

3.3.4.7 Core Spray Lines

The two 100 percent capacity core spray lines separately enter the reactor vessel through the two core spray nozzles located 120° apart (subsection 4.2, "Reactor Vessel and Appurtenances Mechanical Design"). The lines divide immediately inside the reactor vessel. The two halves are routed to opposite sides of the reactor vessel and are supported by clamps attached to the vessel wall. The header halves are then routed downward into the downcomer annulus and pass through the upper shroud immediately below the flange. The flow divides again as it enters the center of the semicircular sparger ring which is routed halfway around the inside of the upper shroud. The ends of the two sparger rings for each line are supported by slip-fit brackets designed to accommodate thermal expansion of the rings. The header routing and supports are designed to accommodate differential movement between the shroud and the vessel. The lines are similar except the sparger rings are at slightly different elevations in the shroud. The proper spray distribution pattern is provided by a combination of distribution nozzles pointed radially inward and downward from the sparger rings (Section 6.0, "Core Standby Cooling Systems").

3.3.4.8 Vessel Head Cooling Spray Nozzle

The vessel head spray function has been permanently disabled. Blind flanges have been installed at the vessel nozzle. The piping is disconnected from vessel head per Mod 1536 for Unit 3 and Mod P00403 for Unit 2.

3.3.4.9 Differential Pressure and Standby Liquid Control Line

The differential pressure and liquid control line serves a dual function within the reactor vessel-- to inject liquid control solution into the coolant stream (subsection 3.8, "Standby Liquid Control System"), and to sense the differential pressure across the core support assembly (subsection 4.2, "Reactor Vessel and Appurtenances Mechanical Design"). The line enters the reactor vessel at a point below the core shroud as two concentric pipes. In the lower plenum, the two pipes separate. The inner pipe terminates near the lower shroud with a perforated length below the core support assembly. It is used to sense the pressure below the core support during normal operation and to inject liquid control solution when required. This location assures that good mixing and dispersion are facilitated. The use of the inner pipe also reduces the thermal shock to the vessel nozzle should the standby liquid control system be actuated. The outer pipe terminates immediately above the core support assembly and senses the pressure in the region outside the fuel assembly channels.

3.3.4.10 In-Core Flux Monitor Guide Tubes

The in-core flux monitor guide tubes extend from the top of the in-core flux monitor housings (subsection 4.2, "Reactor Vessel and Appurtenances Mechanical Design") in the lower plenum to the top of the core support. The power range detectors for the power range monitoring units and the dry tubes for the WRNM detectors are inserted through the guide tubes and are held in place below the top guide by spring tension. A latticework of clamps, tie bars, and spacers gives lateral support and rigidity to the guide tubes. The bolts and clamps are welded after assembly to prevent loosening during reactor operation.

3.3.4.11 Surveillance Sample Holders

The surveillance sample holders are welded baskets containing impact and tensile specimen capsules (subsection 4.2, "Reactor Vessel and Appurtenances Mechanical Design"). The baskets hang from brackets on the inside diameter of the reactor vessel at the mid-height of the core and at radial positions chosen to expose the specimens to the same environment and maximum neutron fluxes experienced by the reactor vessel itself while avoiding jet pump removal interference.

3.3.5 Safety Evaluation

3.3.5.1 Evaluation Methods

To determine that the safety design bases are satisfied, the responses of the reactor vessel internals to loads imposed during normal operation, operational transients, and accidents are examined. Determination of these effects on the ability to insert control rods, cool the core, and flood the inner volume of the reactor vessel are made. Those internals which are required to function for safe shutdown and removal of decay heat are identified, evaluated and designed in accordance with the criteria of Appendix C, structural design criteria.

The ASME Boiler and Pressure Vessel Code, Section III, for Class A vessels is used as a guide to determine limiting stress intensities and cyclic loadings for the reactor vessel internals. For those components for which buckling is not a possible failure mode and stresses are within those stated in the ASME Boiler and Pressure Vessel Code, it is concluded that the safety design bases are satisfied. For those components for which either buckling is a possible failure mode or stresses exceed those presented in the ASME Boiler and Pressure Vessel Code, then either the elastic stability of the structure or the resulting deformation is examined to determine if the safety design bases are satisfied.

3.3.5.1.1 Specific Events to be Evaluated

Examination of the spectrum of conditions for which the safety design bases must be satisfied reveals five significant events as follows:

1. LOCA: This accident is an instantaneous circumferential break in a recirculation line. The accident results in some pressure differentials across the reactor vessel internals which exceed normal loads.

2. Steam line break accident: This accident is a break in one main steam line between the reactor vessel and the flow restrictor. The accident results in significant pressure differentials across the reactor vessel internals.

3. Thermal shock: The most severe thermal shocks to the reactor vessel internals occur when LPCI or HPCI operations reflood the reactor vessel inner volume following either a recirculation line break or a main steam line break (Section 6.0, "Core Standby Cooling Systems").

4. Earthquake: This event subjects the reactor vessel internals to significant forces as a result of ground motion.

5. Blowdown hydrodynamic forces: This event subjects the reactor vessel internals to significant forces under the postulated design basis LOCA.

Analysis of other conditions existing during normal operation, operational transients, and accidents shows that the loads affecting the reactor vessel internals are less severe than the five postulated events.

3.3.5.1.2 <u>Pressure Differentials During Rapid Depressurization</u>

A digital computer code[2] is used to analyze the transient conditions in the reactor vessel following the LOCA and the design basis steam line break accident. The analytical model of the vessel consists of nine nodes, which are connected to the necessary adjoining nodes by flow paths having the required resistance and inertial characteristics. The program solves the energy and mass conservation equations for each node to give the depressurization rates in the various regions of the reactor.

The nine nodes are: (1) lower plenum, (2) active core, (3) upper plenum, (4) separation region, (5) downcomer, (6, 7) recirculation pumps, (8) core bypass and guide tube volume, and (9) steam dome.

The flow resistances are evaluated from the irreversible pressure drops associated with known flow rates. If the accident being considered is a rupture in the recirculation loop, an additional flow path exists through the diffusers of the inoperative jet pumps.

Momentum effects are considered for the core inlet, core outlet, separator, and jet pump flows; it is not a significant effect in the other reactor vessel internal flow paths.

Figure 3.3.7 shows the reactor nodes; the normal reactor internals pressure differentials (RIPD's) acting on major components are designated as follows:

RIPDs for Normal Conditions (psid)[1]

Parameter	
Core Plate & Guide Tube	24.94
Shroud Support Ring & Lower Shroud	34.11
Upper Shroud	9.46
Shroud Head	10.08
Shroud Head to Water Level	12.85(Irreversible)
Shroud Head to Water Level (Elevation)	0.85
Top Guide	0.75
Steam Dryer	0.41[2]

Notes

(1) Unit 2 - Conservatively based on 110% core flow and 3951 MWt. The table contains GE14 RIPD values. These values bound GNF2 (Reference 20).

(2) The dryer RIPDs have been calculated at bounding conditions in PEAM-EPU-10 (Reference 20). RIPD for the Replacement Steam Dryer (RSD) was calculated by Westinghouse (PEAM-EPU-130) and resulted in a range of dryer RIPDs from 0.294 to 0.347 psid. The GEH evaluation in PEAM-EPU-10 evaluated the dryer and the other components for normal conditions at a higher dryer RIPD of 0.41 which bounds the Westinghouse evaluation of the RSD and evaluates the rest of the components at the higher dryer RIPD. Therefore, the steam line break analysis performed by GEH for reactor vessel internals loading remains bounding for the RSD (Reference 21).

3.3.5.2 Recirculation Line Break

The postulated break in the recirculation line is not the design basis with respect to internal differential pressure loads. The maximum loads occur following the postulated steam line break accident and are presented in paragraph 3.3.5.3.

3.3.5.2.1 Jet Pump Joints and Access Hole Cover Joints*

An analysis has been performed to evaluate the potential leakage from within the floodable inner volume of the reactor vessel during the recirculation line break and subsequent LPCI reflooding. The possible sources of leakage are:

1. Jet pump throat to diffuser joint
2. Jet pump nozzle to riser joint (in the flowpath to the floodable inner volume)
3. Shroud support plate access hole cover joint (for PBAPS Unit 3 only).
4. Jet pump thermal sleeve to elbow weld.
5. Jet Pump diffuser to adapter weld location AD-3b.
6. Jet pump adapter backing ring upper fillet weld location AD-3a.

*Note: Access hole cover joint leakage is applicable only for the Unit 3 design.

The jet pump diffuser to shroud support joint is welded and therefore is not a possible source of leakage. The jet pump throat to diffuser joint (slip/socket joint) for all jet pump leaks no more than a total of 225 gpm. The jet pump nozzle to riser joint (clamped by the beam bolt) by analysis is shown to leak no more than 582 gpm for the pumps through which the vessel is being flooded. The two welded shroud support plate access hole covers in the PBAPS Unit 3 RPV have been replaced with a bolted access hole cover design. The effect of potential leakage, through the bolted access hole cover joints, on the core flow was considered. It is determined that the total calculated maximum leakage from both the access hole cover joints (neglecting the presence of a seal ring which is installed to minimize leakage) is not more than 500 gpm. Modification P00769 installed clamps on the Unit 3 JP 1/2 (N2E) and JP 13/14 (N2J) RS-1 welds. Leakage at the recirculation inlet nozzle thermal sleeve to jet pump riser elbow welds was evaluated in reference 7 considering two configurations. The first configuration included the two clamps and a crack at JP 9/10 (N2A), the second configuration considered clamps on 360 degree cracks at all RS-1 locations. The results of this evaluation showed:

a. During normal operation, leakage at any single clamped recirculation inlet nozzle thermal sleeve weld will be less than 2% of the original rated recirculation pump flow (45,200 gpm) while operating at the current maximum rated (power rerate) condition, including increased core flow.

b. For an assumed large recirculation line break LOCA, the sum of the LPCI system leakage from the clamped JP 1/2 (N2E) and JP 13/14 (N2J) locations plus the maximum crack length at JP 9/10 (N2A) and other identified system leakages will be limited to a value which has been reconciled with the licensing basis accident analysis (Reference 4), to show that applicable plant safety limits are met.

Engineering evaluations (reference 12 and 13) for a previously evaluated indication on the RS-1 weld of jet pump 9/10, and an indication on the adapter assembly of jet pump 18 (PBAPS, Unit 3 only) calculated postulated leakage rates associated with the evaluated locations. The results of the evaluation showed:

a. During normal operation, leakage from jet pump 9/10 will still be bounded by the previous analysis (reference 8), and leakage from jet pump 18 will be less than 0.05% of the original rated recirculation pump flow (45,200 gpm) while operating at the current maximum rated (power rerate) condition, including increased core flow.

b. For an assumed large recirculation line break LOCA, the sum of the LPCI system leakage from all sources will be limited to a value which has been reconciled with the licensing basis accident analysis (Reference 4), to show that applicable plant safety limits are met.

   Postulated leakage from flaw location on diffuser to adapter welds have been evaluated in Reference 11. This analysis covered known flaw locations and conservatively assumed similar flaws exist in all twenty jet pump diffuser to adapter welds for the purpose of the leakage analysis. The result was that margin relative to the assumed LPCI delivery capacity was still maintained.

   In any event, post RPV internals inspection assessment and corrective actions shall ensure that leakage from flaw locations and other sources within the floodable inner volume combined do not exceed design or licensing basis limits.

   The latest revision of Reference 16 shall be consulted to determine the scope of the inspections performed during each cycle outage and to reference any inspection results and associated evaluations that may have been performed.

3.3.5.3 Steam Line Break Accident

The analysis of this accident assumes an instantaneous circumferential break of one main steam line between the reactor vessel and the main steam line flow restrictor. This is not the same accident as that described in Section 14.0, "Plant Safety Analysis," because greater differential pressures across the reactor vessel internals result from this accident. It is noteworthy that this accident results in greater loading of the reactor vessel internals and a higher depressurization rate than does the recirculation line break. This is because the depressurization rate is proportional to the mass flow rate and the excess of fluid escape enthalpy above saturated water enthalpy (h_f). However, mass flow rate is inversely proportional to escape enthalpy (h_e); therefore, the depressurization rate is proportional to 1 - (h_f/h_e). Consequently, the depressurization rate decreases as h decreases; that is, the depressurization is less for mixture flow than for steam flow.

The reactor is assumed to be at 4030 MWt (1.02 x 120 % of original rated power) with 110 percent of rated recirculation flow at the time of the break. The analysis considers initial conditions of both normal feedwater temperature (NFWT) and reduced feedwater temperature (RFWT) final feedwater temperature reduction (FFWTR) of 90 F.

The initial values of key nuclear system parameters are as follows:

Core Power	4030 MWt
Steam Rate, NFWT	16.565 x 10^6 lbm/hr
Steam Rate, RFWT	14.882 x 10^6 lbm/hr
Core Flow	112.75 x 10^6 lbm/hr
Core inlet enthalpy	526.3 BTU/lbm (NFWT)
Feedwater Temperature	383.5 F (NFWT) 292.9 F (RFWT)

Two conditions were analyzed for faulted condition, High Power and Interlock. The first condition is limiting for certain components because the maximum loads occur at the maximum core flow and maximum void formation in the bundles. The second condition is limiting for certain components because it results in a higher mismatch between the steam flow from the break and the steam generated in the core during a postulated steam line break. At the interlock point with lower thermal power, the core steam flow is much lower than the high power case resulting in a greater difference between the core generated steam flow and the steam exiting through the break.

Figure 3.3.8 shows the variation of differential pressures for various internals as a function of time at high power conditions for the EPU analysis (Reference 20). The differential pressures across the reactor vessel internals resulting from the accident are provided as follows.

RIPD for Faulted Conditions (psid)

Parameter (Ref. Figure 3.3-7)	Interlock	High Power
Core Plate & Guide Tube	29.5	29.5
Shroud Support Ring & Lower Shroud	53	53
Upper Shroud	32	31
Shroud Head	32	32
Shroud Head to Water Level (Irreversible)	33	34
Shroud Head to Water Level (Elevation)	2.4	1.4
Top Guide	2.0	0.76
Steam Dryer*	5.4	3.6

Interlock: This is at the recirculation pump cavitation interlock point, 858.6 MWt (21.7% of rated power). Values are the maximum results from either normal or reduced feedwater temperature with GE14 fuel at 110% rated core flow. The reduced feedwater temperature of 90 F was used. The GE14 fuel is the limiting fuel for RIPD with faulted conditions.

High Power: This is at 102% rated power (4030 MWt). Values are the maximum results from either normal or reduced feedwater temperature with GE14 fuel at 110% rated core flow. Evaluations at these points considered both normal and reduced feedwater temperatures. The reduced feedwater temperature of 90 F was used. The GE14 fuel is the limiting fuel for RIPD with faulted conditions.

* The Steam Dryer values are bounding but are not used in the analysis of the replacement stream dryers.

Note:

The core pressure drop for GNF2 is less than the core pressure drop for GE14 meaning the RIPD results for GNF2 are bounded by GE14. The GNF2 bundle is heavier than the GE14 bundle meaning the fuel lift margin results for GNF2 are bounded by GE14.

These maximum differential pressures are used, in combination with other assumed structural loads, to determine the total loading on the various reactor vessel internals. The various internals are then examined to assess the extent of deformation and collapse, if any. Of particular interest are the responses of the core support, the guide tubes, and the metal channels around the fuel bundles.

3.3.5.3.1 Core Support

The core support sustains the maximum net force, which is an upward force following the steam line break accident, so the effect on the core support holddown bolts must be established. Analysis shows that the applied stresses are about one-half of yield strength for the bolts, indicating that the core support can withstand the effects of the accident.

3.3.5.3.2 Guide Tubes

Because of the externally applied pressure, the guide tube is examined for collapse. As in the case of the lower shroud and core support assembly, a number of formulae are utilized to calculate the collapse pressure. Use of ASME curves indicates the extreme sensitivity to wall thickness.

For the minimum wall thickness for a 10-in Schedule 10 pipe, the ASME curves give a collapse load of 45 psi. Using the average wall thickness, the collapse pressure is increased to over 70 psi. Using empirical relations for tubes over the critical length, the calculated collapse pressure is reached at 54 psi for a wall thickness of 0.150 in, which is 6 mils over the minimum for a 10-in Schedule 10 pipe. The calculated total loading for the guide tubes is considerably below the collapse loading, and it can be concluded that no failure occurs. The analysis also indicates that the control rods are 70 percent to 80 percent inserted at the time the maximum external pressure is applied to the guide tubes.

3.3.5.3.3 Fuel Channels

The fuel channel load due to an internally applied pressure is examined utilizing a fixed-fixed beam analytical model under a uniform load. Tests have been conducted to verify the applicability of the analytical model. The results indicate that the analytical model is conservative. The fuel channels may deform sufficiently outward to cause some interference with movement of the control rod blade. There are about 15 factors, such as fuel channel deformation, core support, hole tolerance, top fuel guide beam location, etc, that determine the clearance between the control rod blade and fuel channel. If each of these tolerance factors is assumed to be at the worst extreme of the tolerance range, then a slight interference would develop under an 18 psi pressure difference across the channel wall. At the top of the control rod there are rollers to guide the blade as it is inserted. The clearance between channels is 70 mils less than the diameter of the roller, causing it to slide or skid instead of roll. As the rod is inserted about half way, there is a tendency for the control rod sheath to push inward on the channel. This is a control rod surface to channel surface contact. A "worst case" study indicates a possibility of a 50-mil interference.

The possibility of a worst case developing is extremely remote. A statistical analysis utilizing a normal distribution for each of the 15 variables indicates that no interference occurs within 3 limits, where 1 is the standard deviation in a point distribution of events. Three lies in the 0.995 percentile of probability of non-occurrence. However, even if interference occurs, the result is negligible. About one pound of lateral force is required to deflect the channel inboard 1 mil. The friction force developed is an extremely small percentage of the total force available to the CRD's.

The previous discussion presupposes the control rod has not moved when the fuel channel experiences the largest magnitude of pressure drop. Analysis indicates that the rod is about 70 to 90 percent inserted. If the rod is beyond 70 percent inserted, then no interference is likely to develop because all the channel deformation is in the lower portion of the fuel channel, whereas the rollers are at the top of the rod. It is concluded that the main steam line break accident poses no significant interference to the movement of control rods.

3.3.5.4 Thermal Shock

The most severe thermal shock effects for the reactor vessel internals result from the reflooding of the reactor vessel inner volume. For some vessel internals, the limiting thermal shock occurs from LPCI operation, and for others HPCI operation is controlling, dependent upon the location of the component. These effects occur as a result of any large LOCA, such as the recirculation line break and the steam line break accidents previously described.

Three specific locations are of particular interest, as shown in Figure 3.3.9. The locations are as follows:

1. Shroud support plate

2. Shroud-to-shroud support plate discontinuity

3. Shroud inner surface at highest irradiation zone.

The peak strain occurring in the shroud support plate is about 6.5 percent. This strain is higher than the 5.0 percent strain equivalent of the stress amplitude permitted by the ASME Boiler and Pressure Vessel Code, Section III, for 10 cycles, but the one cycle peak strain corresponds to about six allowable cycles of an extrapolated ASME code curve.

Figure 3.3.10 illustrates both the ASME code curve and the basic material curves from which it was established (with the safety factor of 2 on strain or 20 on cycles, whichever is more conservative). The extrapolation of the ASME code curve represents a similar criteria to that used in the ASME Boiler and Pressure Vessel Code, Section III, but applied to fewer than 10 cycles of loading. For this type 304 stainless steel material, the stress value of the equivalent 10 percent peak strain corresponds to one allowable cycle of loading. Even a 10 percent strain for a single cycle loading represents a very conservative suggested limit because this has a large safety margin below the point at which even minor cracking is expected to begin. Because the conditions which lead to the calculated peak strain of 6.5 percent are not expected to occur even once during the entire reactor lifetime, the peak strain is considered tolerable.

The results of the analysis of the shroud-to-shroud support plate discontinuity region are as follows:

Amplitude of alternating stress	180,000 psi
Peak strain	1.34 percent

The ASME Boiler and Pressure Vessel Code, Section III, allows 220 cycles of this loading. Thus, no significant deformation results.

The most irradiated point on the inner surface of the shroud is subjected to a total integrated neutron flux of 3.56×10^{21} n/cm^2

(>1 Mev) by the end of plant life (54 EFPY). The peak thermal shock stress is 155,700 psi, corresponding to a peak strain of 0.57 percent. The shroud material is type 304 stainless steel, which is not significantly affected by irradiation. The peak strain resulting from thermal shock at the inside of the shroud represents no loss of integrity of the reactor vessel inner volume.

3.3.5.5 Earthquake

The seismic loads on the reactor pressure vessel and internals due to horizontal motion are based on a dynamic analysis of a reactor pressure vessel and internals model similar to that shown in the Figure 3.3.11. Seismic analysis is performed by coupling this lumped mass model of the reactor pressure vessel and internals with the building-soil structure model to determine the system natural frequencies and mode shapes. The relative displacement, acceleration, and load response of the reactor pressure vessel and internals is then determined by the time-history method. In the time-history method, the dynamic response is determined for each mode of interest and added algebraically for each instant of time. Resulting response time-histories are then examined and the maximum value of displacement, acceleration, shears, and moments are used for design calculations.

The natural frequencies of the reactor internals, reactor vessel, and pedestal system in the vertical direction have been found to be approximately 20 Hz. Examination of the response spectra shows no significant amplification at this frequency. Hence, omitting the vertical motion from seismic analysis to reduce the analytical complexities is acceptable. The effects of vertical excitations are accounted for by increasing or decreasing (whichever causes higher stress) the weight of the various components by a percentage equal to the vertical acceleration expressed in percent "g".

Details of the analysis are presented in Appendix K.

3.3.5.6 Blowdown Hydrodynamic Forces

In order to assure that no significant dynamic amplification of load occurs as a result of the oscillatory nature of the blowdown forces (Figure 3.3.8), a comparison was made of the periods of the applied forces and the natural periods of the reactor internal components being acted upon by the applied forces. These periods are determined from a comprehensive dynamic model of the reactor pressure vessel and internals with 27 degree-of-freedom. Since only motion in the vertical direction is considered here, each structural member (between two lumped masses) can only have an axial load.

Besides the real masses of the reactor pressure vessel and internals, the hydrodynamic mass effects of the water inside the reactor pressure vessel are accounted for.

The smallest period of the applied force (approximately 0.7 sec) is more than 10 times the largest period of the component upon which the force acted (i.e., natural frequency of component is more than 10 times greater than the frequency of the applied load). It is evident that this conclusion would apply for the higher modes, since they would have shorter periods. A typical response curve for a damped, single degree-of-freedom system subjected to a sinusoidal forcing function is shown as Figure 3.3.12. In this figure μ is the amplification factor, which is the ratio of the forced response and the static response, and $r \leq 0.1$ and therefore $\mu \leq 1.01$.

Therefore, it was concluded that no significant load amplification occurred because of the "slowly" changing nature of the applied load and because a statically applied load equal in value to the peak transient load can be used for design purposes.

3.3.5.7 Replacement Steam Dryer

The PBAPS Replacement Steam Dryers (RSDs) were analyzed with the Acoustic Circuit Model Enhanced 2.0 (ACE 2.0), which was benchmarked against the Monticello Nuclear Generating Plant (MNGP), also a Westinghouse octagonal design, for predicting stresses in the hood; and analyzed with the ACE 2.0-SPM (Skirt Protection Model), benchmarked against measured observations in another instrumented dryer, for predicting stresses in the skirt.

A key part of all steam dryer alternating stress evaluations is assessing the effects of acoustic loads induced by flow-induced resonances at the various main steam line valves. The acoustic mode frequencies in the valve standpipes are functions of standpipe dimensions, and are strongly excited when these frequencies coincide with those of flow instability modes across the standpipe openings driven by the main steam flow. There are specific flow rates which drive these acoustic modes, which are usually quite high. The PBAPS EPU main steam flow velocity is generally lower than that of other BWRs that have received NRC-approved EPU license amendments.

The fluctuating acoustic pressure loads were applied to the finite element analysis of the RSDs. Finite element analysis (harmonic analysis in frequency domain) was performed using ANSYS general purpose finite element code. Structural damping with 1% of critical damping was applied for all frequencies and is in accordance with NRC RG 1.20, Rev. 3. Hydrodynamic damping was also used in the structural analysis. Hood and skirt stresses were found to be acceptable.

The RSD fabrication mainly includes full penetration welds with fillet welds joining the perforated plate onto the inlet face of the dryer vane banks. The weld stresses calculated using weld factors bound the stresses calculated according to the ASME Code, Subsection NG. The RSD analysis also accounted for stresses due to the vane passing frequencies (VPFs) of the reactor recirculation pump (RRP), which were determined to be small. When the stresses due to the acoustic loads are added to those due to the RRP VPFs, the resulting alternating stress intensities satisfy the requirement of a minimum stress ratio of 1.0 for the hood.

Subsection NG of Section III of the ASME Code and plant-specific load combinations were used to evaluate stream dryer stresses to establish the acceptability for normal, upset, emergency and faulted conditions. The ratio of allowable stress intensities to maximum computed stress intensities are all greater than 1.0, thus meeting the applicable Code limits.

Both units contain main steam line strain gauges. Since Unit 2 is the prototype for Unit 3, it was also fitted with on-dryer instrumentation for the initial EPU power ascension in order to validate the RSD evaluation methodology and correlate the results to the main steam line gauge measurements.

3.3.5.8 Conclusions

The analyses of the responses of the reactor vessel internals to situations imposing various loading combinations on the internals show that deformations are sufficiently limited to allow both adequate control rod insertion and proper operation of the core standby cooling systems. Sufficient integrity of the internals is retained in such situations to allow successful reflooding of the reactor vessel inner volume. The analyses considered various loading combinations, including loads imposed by external forces. Thus, safety design bases 1 and 2 are satisfied (Ref. Section 3.3.3).

3.3.6 Inspection and Testing

Quality control methods were used during the fabrication and assembly of reactor vessel internals to assure that the design specifications were met.

The reactor coolant system, which includes the reactor vessel internals, was thoroughly cleaned and flushed before fuel was loaded initially.

During the pre-operational test program, operational readiness tests were performed on various systems. In the course of these tests, such reactor vessel internals as the feedwater spargers, the core spray lines, and the standby liquid control system line were functionally tested.

Steam separator-dryer performance tests were made during the startup test program to determine carryunder and carryover characteristics. Steam samples were taken from the inlet and outlet of the steam dryers and from the inlet to the main steam lines at various reactor power levels, water levels, and recirculation flow rates. Moisture carryover was determined from sodium-24 activity in these samples and in reactor water samples. Carryunder was determined from measured flows and temperatures determined by heat balances.

A vibration analysis of reactor vessel internals was performed in the design. In the event that the design of the reactor vessel internals represented a significant departure from design configuration previously tested and found acceptable, vibration measurements were taken during startup tests. The measurements were used to determine the vibration characteristics of the reactor vessel internals and the recirculation loops under forced recirculation flow. Vibratory responses were recorded at various recirculation flow rates using strain gages on fuel channels and control rod guide tubes, accelerometers on the shroud support plate and recirculation loops, and linear differential transducers on the upper shroud and shroud head steam separator assembly. The vibration analyses and tests were designed to determine any potential, hydraulically induced equipment vibrations and to check that the structures do not fail due to fatigue. The structures were analyzed for natural frequencies, mode shapes, and vibrational magnitudes that could lead to fatigue at these frequencies. The cyclic loadings were evaluated using as a guide the cyclic stress criteria of the ASME Boiler and Pressure Vessel Code, Section III.

Adequate working space was provided inside the reactor vessel to allow access for inspection. Examinations are performed to satisfy the ASME XI code requirements as specified in 10CFR50.55A (Ref. Appendix I, "In-Service Inspection Program").

3.3 REACTOR VESSEL INTERNALS MECHANICAL DESIGN

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19. NEDC-33566P, Safety Analysis Report for Exelon Peach Bottom Atomic Power Station, Units 2 and 3, Constant Pressure Power Uprate.

20. PEAM-EPU-10, T0304 Reactor Internal Pressure Differences and Fuel Lift Evaluation, Peach Bottom Atomic Power Station, Units 2 and 3.

21. PEAM-EPU-130, Replacement Steam Dryer Disposition on GEH Task Evaluations.

TABLE 3.3.1

REACTOR VESSEL INTERNALS DESIGN DATA

Core Shroud	
Upper Portion, od, in	220
Central Portion, od, in	207.125
Central Portion, Thickness, in	2
Weight, lb	116,900
Shroud Head-Steam Separator Assembly	
Head Thickness, in	2.0
Number of Separators	211
Separator od, in	12.75
Standpipe, id, in	6.065
Standpipe, od, in	6.625
Weight, lb	139,600
Core Support	
Weight, lb	20,500
Top Guide	
Weight, lb	15,200
Fuel Support Pieces	
Number of Peripheral	24
Four-Lobe	
Number Without Plugs	185
Number With Plugs	0
Weight, lb	11,300
Control Rod Guide Tubes	
Number	185
Weight, lb	46,350
Jet Pumps	
Number	20
Throat Diameter, in	8.18
Weight, lb	22,700

TABLE 3.3.1 (Continued)

Steam Dryers	
Weight, lb	126,000
Feedwater Sparger	
Diameter, in	6-Sched. 40
Cross-Section Area, sq ft	0.2006
Number	6
Core Spray Sparger	
Diameter, in	4-Sched. 40S
Cross-Section Area, sq ft	0.0884
Number of Spray Outlets/Sparger	260
Weight, lb	4,317
*Vessel Head Cooling Spray Nozzle	
Pipe Size, in	4-Sched. 40
Differential Pressure & Liquid Control Line	
Inner Pipe (Liquid Control), in	1-Sched. 40
Outer Pipe, in	2-Sched. 40
In-Core Flux Monitor Guide Tubes	
Number	55
Surveillance Sample Holders	3
Total Weight of Reactor Vessel Internals, lb (excluding fuel, control rods, feedwater spargers, vessel head cooling spray nozzles, in-core guide tubes, startup neutron sources)	498,000

* Function is deleted (See Section 3.3.4.8)

3.4 REACTIVITY CONTROL MECHANICAL DESIGN

3.4.1 Safety Objective

The safety objective of the reactivity control mechanical design is to provide a means of rapid reactor shutdown so that damage to the fuel barrier is limited or prevented. The objective is met by inserting neutron-absorbing material into the reactor core.

3.4.2 Safety Design Basis

1. The reactivity control mechanical design includes control rods.

    a. The control rods have sufficient mechanical strength to prevent the displacement of their reactivity control material.

    b. The control rods have sufficient strength and are designed to prevent deformation that could inhibit their motion.

    c. Each control rod includes a device to limit its free fall velocity to such a rate that the nuclear system process barrier is not damaged due to pressure increase caused by the rapid reactivity increase resulting from the free fall of one control rod from its fully inserted position.

2. The reactivity control mechanical design provides for a sufficiently rapid insertion of control rods so that no fuel damage results from any abnormal operating transient and limits fuel damage under accident conditions.

3. The reactivity control mechanical design includes positioning devices each of which individually support and position a control rod.

4. Each positioning device:

    a. Prevents gross withdrawal of its control rod as a result of a single malfunction of the positioning device.

    b. Avoids conditions which could prevent its control rod from being inserted.

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

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

e. Is locked to its control rod to prevent undesirable separation.

3.4.3 Power Generation Objective

The power generation objective of the reactivity control mechanical design is to provide a means to control power generation in the fuel. This objective is met by positioning neutron-absorbing material in the reactor core.

3.4.4 Power Generation Design Basis

1. The reactivity control mechanical design includes reactivity control devices (control rods) which contain and hold the reactivity control material necessary to control the excess reactivity in the core.

2. The reactivity control mechanical design includes provisions for adjustment of the control rods to permit control of power generation in the core.

3.4.5 Description

The reactivity control mechanical design consists of control rods which can be positioned in the core by individual CRDS mechanisms.

The CRD mechanisms are part of the CRDS. The CRDS hydraulically operates the CRD mechanisms using water from the condensate system as a hydraulic fluid. The CRD pump takes suction from the condensate system on the discharge side of the condensate demineralizers in order to provide high purity deaerated water to the CRDS. A flow control station is installed downstream of the tap from the condensate system, and ties into the CRD pump suction line before the CRD suction filter (Drawing M-356). The flow control station will divert approximately 250 gpm from the condensate system, which will supply the CRD and the remainder will be passed on to the condensate storage tank. The flow will ensure an adequate supply for recharging the accumulators after a scram and a deaerated water supply to the CRDS at all times. In the event that the flow from the condensate system is interrupted, the condensate storage tank provides a backup source to ensure CRDS operability without operator action being required when the condensate storage tank level is within its normal operating range. The CRD mechanisms manually position the control rods during normal operation and act automatically to rapidly insert the control rods when required. The control rods, CRD mechanisms, and that part of the CRD hydraulic system necessary for scram are designed to seismic Class I criteria.

3.4.5.1 Reactivity Control Devices

3.4.5.1.1 Control Rods

The control rods (Figures 3.4.1.A, 3.4.1.B and 3.4.1.C) perform the dual function of power shaping and reactivity control. Power distribution in the core is controlled during operation of the reactor by manipulation of selected patterns of control rods. The bottom entry design of the control rods counterbalances steam void effects at the top of the core.

The control rod design originally supplied at PBAPS consisted of a sheathed cruciform array of stainless steel tubes filled with boron-carbide powder (all-B_4C rods). Several new design longer life control rods are currently in use at PBAPS. They contain a mixture of boron-carbide absorber rods and solid hafnium absorber rods (see Sections 3.4.8, 3.4.9, and 3.4.10). The following discussion applies to the originally supplied all-B_4C control rods. Section 3.4.8 discusses the longer life control rod designs, Section 3.4.9 discusses the Marathon Control Rod Assembly, and 3.4.10 discusses the Westinghouse Atom Control Rod Assembly. The control rods are 9.75 inches in total span and are located uniformly through the core on a 12-inch pitch. Each control rod is surrounded by four fuel assemblies.

The main structural member of a control rod is made of type 304 stainless steel and consists of a top casting which incorporates a handle, a bottom casting which incorporates a velocity limiter and a CRD coupling, a vertical cruciform center post, and four U-shaped absorber tube sheathes. The two end castings and the center post are welded into a singly skeletal structure. The U-shaped sheathes are resistance welded to the center post and castings to form a rigid housing to contain the boron-carbide filled absorber tubes. Rollers at the top and the bottom of the control rod provide guidance for the control rod as it is inserted and withdrawn from the core. The control rods are cooled by the fuel assembly bypass flow. The U-shaped sheathes are perforated to allow the coolant to freely circulate about the absorber tubes. Operating experience has shown that control rods constructed as described are not susceptible to dimensional distortions, thus satisfying safety design basis lb.

The boron-carbide (B_4C) powder in the absorber tubes is compacted to about 70 percent of its theoretical density; the boron-carbide contains a minimum of 76.5 percent by weight natural boron. The boron-10 content of the boron is 18.0 percent by weight minimum. The absorber tubes are made of type 304 stainless steel, are 0.188-in. in outside diameter and have a 0.025-in wall thickness. Each absorber tube is sealed by a plug welded into each end. The boron-carbide is separated longitudinally into individual compartments by stainless steel balls at approximately 16-in intervals. The steel balls are prevented from settling by a slight crimp in the tube wall below each ball. Should the boron-carbide tend to compact further in service, the steel balls will distribute the resulting voids over the length of the absorber tube.

The operational lifetime of the control rods is determined by the burnup of boron-10 from neutron absorption. The nuclear lifetime limit is reached when the peak boron depletion results in a 10 percent loss in relative control worth.

The mechanical lifetime limit is defined as the time at which the internal helium pressure from the boron-10 (neutron, alpha) reaction results in stresses in any absorber tube of the control rod reaching the most restrictive design limit.

Based on experimental data, a helium release analytical model is used to correlate the fraction of generated helium which is released from the boron-carbide with the boron-10 burnup fraction. This model predicts a release fraction which starts at 4 percent for zero boron-10 burnup and increases to approximately 20 percent release at 100 percent boron-10 burnup.

Since the control rods enter from the bottom of the core, the neutron exposure of the control rods is skewed toward the top half of the control rod. The absorber tube at the outer edge of each blade of the control rod receives more neutron irradiation than any other tube in the blade. Neutron irradiation is significantly less for each absorber tube located closer to the center of the control rod. The absorber tubing at the lower end of the control rods undergoes negligible fast flux irradiation and, as a result, retains its initial annealed material properties throughout the lifetime of the control rods. Thus, the allowable design stress for all absorber tubes which extend into the bottom end of the control rod is based upon the mechanical properties of fully annealed type 304 stainless steel.

The average mechanical lifetime of the control rods is calculated to be approximately 18 yr of full power operation. The actual lifetime of a control rod is strongly dependent on where it is used in the core and on its mechanical design. The actual replacement of control rods depends on the loss of reactivity control capability and gas pressure buildup and varies among control rods. The average expected service life of control rods is approximately 15 yr.

The control rods and absorber tubes meet the requirements of safety design basis 1a.

3.4.5.1.2 <u>Control Rod Velocity Limiter</u>

The control rod velocity limiter is an integral part of the bottom assembly of each control rod. This engineered safeguard protects against a high reactivity insertion rate by limiting control rod velocity in the event of a control rod drop accident. It is a one way device, in that the control rod scram velocity is not significantly affected but the control rod dropout velocity is reduced to a permissible limit (Figures 3.4.2 and 3.4.3).

A new lightweight velocity limiter (see Section 3.4.8 and Figure 3.4.1.B) was designed and incorporated on the longer life control rod assemblies. The new velocity limiter was designed to the same design specifications as the original velocity limiter and meets or exceeds all of the design requirements, e.g., rod drop velocity, scram performance, and structural integrity.

The velocity limiter is in the form of two nearly mated conical elements that act as a large clearance piston and baffle inside the control rod guide tube and is effective for the length of the control rod stroke.

The hydraulic drag forces on a control rod are approximately proportional to the square of the rod velocity and are negligible during normal rod withdrawal or rod insertion. However, during the scram stroke the rod reaches high velocity and the drag forces could become appreciable.

In order to limit control rod velocity during dropout but not during scram, the velocity limiter is provided with a streamlined profile in the scram (upward) direction. Thus, when the control rod is scrammed, the velocity limiter assembly offers little resistance to the flow of water over the smooth surface of the upper conical element into the annulus between the guide tube and the limiter. In the dropout direction, however, water is trapped by the lower conical element and discharged though the annulus between the two conical sections. Because this water is jetted in a partially reversed direction into water flowing upward in the annulus, a severe turbulence is created. The resultant hydrodynamic forces slow the descent of the control rod assembly to less than 5 ft/sec at 70°F.

3.4.5.2 Control Rod Drive Mechanisms

The CRD mechanism used for positioning the control rod consists of a double-acting, mechanically latched, hydraulic cylinder using water from the condensate system or condensate storage tank (CST) as its operating fluid. The individual drives are mounted on the bottom head of the reactor pressure vessel. Each drive is an integral unit contained in a housing extending below the reactor vessel. The lower end of each drive housing terminates in a flange to which the drive is bolted. The drives do not interfere with refueling and are operative even when the head is removed from the reactor vessel. The bottom location makes maximum use of the water in the reactor as a neutron shield minimizing neutron exposure to the drive components. The use of high quality condensate as the operating fluid contributes to the simplicity and reliability of the design. For example, simple piston seals are used, since leakage does not contaminate the reactor vessel and helps cool the drive mechanisms (Figures 3.4.2, 3.4.4, 3.4.5, and 3.4.9).

The drives are capable of inserting or withdrawing a control rod at a slow, controlled rate for reactor power level adjustment, as well as providing rapid insertion when required. A locking mechanism on the drive allows the control rod to be locked at every 6 in of stroke over the 12-ft length of the core.

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

Each drive positions its control rod in 6-in increments of stroke, and holds it in these distinct latch positions until actuated by the hydraulic system for movement to a new position. Indication is provided for each rod that shows when the insert travel limit and withdraw travel limit is reached. An alarm annunciates when the withdraw overtravel limit on the drive is reached. Normally, the seating of the control rod at the lower end of its stroke prevents the drive withdraw overtravel limit from being reached. If the drive can reach the withdrawal overtravel limit, it indicates that the control rod is uncoupled from its drive.

The overtravel limit alarm permits the operator to confirm that the rod is coupled to the drive.

Individual rod position indicators are grouped together on the control panel in one display and correspond to the relative rod locations in the core. Each rod indicator gives continuous rod position indication in digital form. Color indication is provided at the fully-in and fully-out positions (green for in, red for out) for emphasis. A separate, smaller, four-rod display is located on the reactor operator's console. This display presents the positions of the control rod selected for movement and the other rods in the rod group. For display purposes, the control rods are considered in groups of four adjacent rods centered around a common core volume monitored by four local power range monitor strings (subsection 7.5, "Neutron Monitoring System"). Rod groups at the periphery of the core may have less than four rods. The four rod display shows the positions of the rods in the group to which the selected rod belongs. A white light indicates which of the four rods is the one selected for movement.

3.4.5.2.1 Components

Figure 3.4.4 illustrates the principle of operation of a drive. Figures 3.4.5 and 3.4.9 illustrate the drive in more detail. Following is a description of the main components of the drive and their functions:

Drive Piston and Index Tube

The drive piston, mounted at the lower end of the index tube, functions as a piston rod. The drive piston and index tube make up the main moving assembly in the drive. The drive piston operates between positive end stops, with a hydraulic cushion provided at the upper end only. The piston has both inside (contracting) and outside (expanding) seal rings, and operates in an annular space between an inner cylinder (fixed piston tube) and an outer cylinder (drive cylinder).

The effective piston area for down-travel or withdraw is about 1.2 sq in versus 4.0 sq in for uptravel or insertion. This difference in driving area tends to balance out the control rod weight and makes it possible to always have a higher insertion force than withdrawal force.

The index tube is a long hollow shaft made of nitrided type 304 stainless steel. This tube has circumferential locking grooves spaced every 6 in along the outer surface. These grooves transmit the weight of the control rod to the collet assembly which locks the rod at each 6-in step. A double tapered groove is provided at position 48 to allow uncoupling during outages.

Collet Assembly

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

Locking is accomplished by six fingers mounted on the collet piston. In the locked or latched position the fingers engage a locking groove in the index tube.

The collet piston is normally held in the latched position by a return spring force of approximately 150 lb. Metal piston rings are used to seal the collet piston from reactor vessel pressure. The collet assembly will not unlatch until the collet fingers are unloaded by a short, automatically sequenced, drive in signal. A pressure approximately 180 psi above reactor vessel pressure acting on the collet piston is required to overcome spring force, slide the collet up against the conical surface in the guide cap, and spread the fingers out so that they do not engage a locking groove. The collet piston is nitrided to minimize wear due to rubbing against the surrounding cylinder surfaces.

Fixed in the upper end of the drive assembly is a guide cap. This member provides the unlocking cam surface for the collet fingers. It also serves as the upper bushing for the index tube and is nitrided to provide a compatible bearing surface for the index tube.

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

Piston Tube and Stop Piston

Extending upward inside the drive piston and index tube is an inner cylinder or column called the piston tube. The piston tube is fixed to the bottom flange of the drive and remains stationary. Water is brought to the upper side of the drive piston through this tube. A series of orifices at the top of the tube provides progressive water shutoff to cushion the drive piston at the end of its scram stroke.

A stationary piston, called the stop piston, is mounted on the upper end of the piston tube. This piston provides the seal between reactor vessel pressure and the space above the drive piston. It also functions as a positive end stop at the upper limit of control rod travel. A stack of spring washers just below the stop piston helps absorb the final mechanical shock at the end of control rod travel. The piston rings are similar to the outer drive piston rings. A bleed-off passage to the center of the piston tube is located between the two pairs of rings. This arrangement allows seal leakage from the reactor vessel (during a scram) to be bled directly to the discharge line, rather than to the space above the drive piston. The lower pair of seals is used only during the cushioning of the drive piston at the upper end of the stroke.

Position Indicator

The center tube of the drive mechanism forms a well to contain the position indicator probe. The position indicator probe is an aluminum extrusion attached to a cast aluminum housing. Mounted on the extrusion is a series of hermetically sealed, magnetically operated, position indicator switches. Each switch is sheathed in a braided glass sleeve, and the entire probe assembly is protected by a thin-walled stainless steel tube. The switches are actuated by a ring magnet attached at the bottom of the drive piston. The drive piston, piston tube, and indicator tube are all of nonmagnetic stainless steel, allowing the individual switches to be operated by the magnet as the piston passes. One switch is located at each position corresponding to an index tube groove, thus allowing indication at each latching point. An additional switch is located at each midpoint between latching points, allowing indication of the intermediate positions during drive motion. Thus, indication is provided for each 3 in of travel. Switches are provided for the full-in and full-out positions. One additional switch (an overtravel switch) is located at a position below the normal full-out positions. Because the limit of down-travel is normally provided by the control rod itself as it reaches the backseat position, the index tube can pass this position and actuate the overtravel switch only if it is uncoupled from its control rod. A convenient means is thus provided to verify that the drive and control rod are coupled after installation of a drive or at any time during plant operation.

Flange and Cylinder Assembly

The fixed components of the drive mechanism (inner cylinder and center tube) are welded to the drive flange. A sealing surface on the upper face of this flange is used in making the seal to the drive housing flange. Teflon-coated, stainless steel O-rings are used for the seals. In addition to the reactor vessel seal, the two hydraulic control lines to the drive are sealed at this face. A drive can thus be replaced without removing the control lines, which are permanently welded into the housing flange. The drive flange contains the integral ball or two-way check (shuttle) valve. This valve directs reactor vessel pressure or driving pressure, whichever is higher, to the underside of the drive piston. Reactor vessel pressure is admitted to this valve from the annular space between the drive and drive housing through passages in the flange. A screen is provided to intercept foreign material in the water. Water used to operate the collet piston passes between the outer tube and cylinder tube. The inside of the cylinder tube is honed to provide the seating surface required for the drive piston seals.

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

<u>Coupling Spud, Plug, and Unlocking Tube</u>

The upper end of the index tube is threaded to receive a coupling spud. The coupling (Figure 3.4.2) is designed to accommodate a small amount of angular misalignment between the drive and the control rod. Six spring fingers allow the coupling spud to enter the mating socket on the control rod. The control rod weight of approximately 250 lb is sufficient to force the spud fingers to enter the socket and push the lock plug up, allowing the spud to enter the socket completely and the plug to snap back into place.

With the lock plug in place, a force in excess of 50,000 lb is required to pull the coupling apart. Two means of uncoupling are provided. With the reactor vessel head removed, the lock plug may be raised against the spring force of approximately 50 lb by a rod extending up through the center of the control rod to an unlocking handle located above the control rod velocity limiter. The control rod, with the lock plug raised, can then be separated from the drive. The lock plug may also be pushed up from below if it is desired to uncouple a drive without removing the reactor pressure vessel head for access. In this case, the central portion of the drive mechanism is pushed up against the uncoupling rod assembly which raises the lock plug and allows the coupling spud to disengage the socket as the drive piston and index tube are driven down.

The coupling spud and locking tube meet the requirements of safety design basis 4e.

3.4.5.2.2 <u>Materials of Construction</u>

Factors determining the choice of materials are listed below:

1. The index tube must withstand the locking and unlocking action of the collet fingers. A compatible bearing combination must be provided which is able to withstand moderate misalignment forces. The reactor environment

limits the choice of materials suitable for corrosion resistance. The column and tensile loads can be satisfied by an annealed 300 series stainless steel. The wear and bearing requirements are provided by Malcomizing the completed tube. To obtain suitable corrosion resistance, a carefully controlled process of surface preparation is employed.

2. The coupling spud is made of Inconel-750 which is aged to produce maximum physical strength and also provides the required corrosion resistance. Because misalignment tends to produce a chafing in the semi-spherical contact area, the entire part is protected by thin vapor-deposited chromium plating (electrolizing). This plating also prevents galling of the threads attaching the coupling spud to the index tube.

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

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

All drive components exposed to reactor vessel water are made of AISI 300 series stainless steel except the following:

1. Seals and bushings on the drive piston and stop piston are Graphitar-14 or Graphitar-3030.

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

3. The ball check valve is a Haynes, Deloro, or equivalent Stellite cobalt-base alloy.

4. Elastomeric O-ring seals are ethylene propylene.

5. Collet piston rings are Haynes-25 alloy.

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

7. Nitriding, electrolizing (a vapor deposition of chromium), and chromium plating are used in areas where resistance to abrasion is necessary.

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

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

3.4.5.3 Control Rod Drive Hydraulic System

The CRD hydraulic system supplies and controls the pressure and flow requirements to the drives (Drawings M-356 and M-357).

There is one supply subsystem which supplies water at the proper pressures and sufficient flow to the hydraulic control units (HCU's). Each HCU controls the flow to and from a drive. The water discharged from the drives during a scram flows through the HCU's to the scram discharge volume. Water is added to the reactor pressure vessel through the CRD's themselves using the cooling flow path.

3.4.5.3.1 Control Rod Drive Hydraulic Supply and Discharge Subsystems

The CRD hydraulic supply and discharge subsystems control the pressure and flows required for the operation of the CRD mechanisms. These hydraulic requirements identified by the function they perform are as follows (Figures 3.4.6, 3.4.10, and Drawings M-356 and M-357):

1. An accumulator charging pressure of approximately 1,400 to 1,500 psig is required. Flow is required only during scram reset or during system startup.

2. Drive pressure of about 250 psi above reactor vessel pressure is required at a flow rate of approximately 4 gpm to insert a control rod, and 2 gpm to withdraw a control rod during normal operation.

3. Cooling water to the drives is required at approximately 20 psi above reactor vessel pressure and at a flow rate of 0.25 to 0.33 gpm per drive unit.

(Cooling water may be interrupted for short periods without drive damage.)

4. A scram discharge volume of approximately 3.34 gal per drive is required. The scram discharge volume is required to contain air at atmospheric pressure, except during scram when it is filled with water until the scram signal is cleared and the system reset. The scram discharge volume will reach reactor pressure following a scram.

The CRD hydraulic supply and discharge systems provide the required functions with the pumps, filters, valves, instrumentation, and piping shown in Drawing M-356 and described in the following paragraphs. Duplicate components are included, where necessary, to assure continuous system operation if an in-service component requires maintenance.

The CRD system also provides a purge water supply to the RWCU recirculation pump motors. The purge supply branches off the CRD pump discharge header downstream of the CRD pump water filters. The purpose of this purge water is to prevent the buildup of contamination in the RWCU pump motor internals.

The CRD hydraulic system provides a source of water for the backfill system utilized to maintain a continuous purge of the reactor water level instrument reference leg.

The CRD hydraulic system provides a source of water to purge the recirculation pump seals.

<u>Pumps</u>

One supply pump is provided to pressurize the system with water from the condensate system or condensate storage tank. One spare pump is provided for standby. Each pump is installed with a suction strainer and a discharge check valve to prevent bypassing flow backwards through the non-operating pump.

A minimum flow bypass connection between the discharge of the pump and the condensate storage tank prevents overheating of the pump in the event that the pump discharge is inadvertently closed.

<u>Filters</u>

The CRD drive water filters remove foreign material larger than 50 microns absolute (25 microns normal) from the hydraulic supply subsystem water. A differential pressure indicator and alarm monitor the filter element as it collects foreign material. A strainer in the filter discharge line guards the hydraulic system in the event of filter element failure.

The exhaust water filters provide protection to the CRDs from carbon steel corrosion product carryover from the stabilizer loop.

The CRD pump suction filters remove particulates from the CRD supply water.
The HCU manifold filters are installed in the directional control manifold to protect the directional control valves from damage due to rust or scale which could enter from the CRD hydraulic system water.

The stabilizing valve filters are installed to protect the stabilizing valves from damage due to rust or scale which could enter from the CRD hydraulic system water.

Accumulator Charging Pressure

The accumulator charging pressure is automatically controlled by the design of the system. The maximum charging water pressure is governed by a combination of the CRD water pump head versus flow characteristic, the total CRD water pump flow and the head losses between the CRD water pump and the charging water header. During normal operation, the accumulator charging pressure is established by the flow control valve. During scram, the flow sensing system upstream at the accumulator charging header detects high flow in the charging header and partly closes the flow control valve. The flow control valve is closed enough so that the proper flow to recharge the accumulators is diverted from the hydraulic supply header to the accumulator charging header.

The pressure in the charging header is monitored in the control room with a pressure indicator and high-pressure alarm.

During normal operation, the constant flow through the flow control valve is the water flow required to cool all the drives.

Drive Water Pressure

The drive water pressure control valve, which is manually adjusted from the control room, maintains the required pressure in the drive water header.

A flow rate of approximately 6 gpm (the sum of the flow rates required to insert and to withdraw a control rod) normally passes from the drive water pressure header through two solenoid-operated stabilizing valves (arranged in parallel). One stabilizing valve passes flow equal to the drive insert flow; the other passes flow equal to the drive withdrawal flow. The appropriate stabilizing valve is closed when operating a drive to divert the required flow to the drive. Thus, the flow through the drive pressure control valve is always constant.

Flow indicators are provided in the drive water header and in the line downstream from the stabilizing valves so that flow rate through the stabilizing valves can be adjusted.

Differential pressure between the reactor vessel and the drive water pressure header is indicated in the control room.

Cooling Water Pressure

The cooling water header is located downstream of the drive water pressure control valve. A differential pressure indicator in the control room indicates the difference between reactor vessel pressure and the drive cooling water pressure. Although the drives can function without cooling water, seal life is shortened by exposure to reactor temperatures.

Exhaust Water Header

The exhaust water header distributes the exhaust water from an individual rod movement to the other drives by backflowing the SV 13-121 valve.

Equalizer Valves

The purpose of the equalizer valves is to repressurize the exhaust water header after a scram and prevent excessively high CRD operating differential pressure during subsequent operation of a selected CRD.

Scram Discharge Volume

The scram discharge volume is used to limit the loss of and contain the reactor vessel water from all the drives during a scram. This volume is provided in the scram discharge header. During normal plant operation, each discharge header is empty and the drain and vent valves are open. Upon receipt of a scram signal, the drain and vent valves close. Position indicator switches on the drain and vent valves actuate valve position lights in the control room.

During a scram, the scram discharge volume partly fills with water which is discharged from above the drive pistons. While scrammed, the CRD seal leakage continues to flow to the discharge volume until the pressure equals reactor vessel pressure. A check valve in each HCU prevents reverse flow from the scram discharge header volume to the drive. After the scram initiating signal is cleared or bypassed, the scram discharge volume scram signal is overridden with the key lock override switch, the RPS is reset and the scram discharge volume is drained.

The scram discharge volume valves can be tested without disturbing the reactor protection system. Closing the discharge volume valves allows the outlet scram valve seats to be leak tested by timing the accumulation of leakage inside the scram discharge volume.

Six level switches on the scram discharge volume, set at three different water levels, guard against operation of the reactor without sufficient free volume present in the scram discharge volume to receive the scram discharge water in the event of a scram. At the first (lowest) level, one level switch initiates an alarm for operator action. At the second level, one level switch initiates a rod withdrawal block to prevent further withdrawal of any control rod. At the third (highest) level, the four level switches (two for each RPS trip system) initiate a scram to shut down the reactor while sufficient free volume is available to receive the scram discharge.

3.4.5.3.2 Hydraulic Control Units

Each HCU serves a single drive unit. The basic components in each HCU are manual, pneumatic, and electrically operated valves, and accumulator, filters, relating piping, and electrical connections (Figures 3.4.6, 3.4.11, and Drawing M-357).

Each HCU furnishes pressurized water upon signal to a CRD which positions its control rod as required. Operation of the electrical system which supplies scram and normal control rod positioning signals to the hydraulic control unit is described in subsection 7.7 "Reactor Manual Control System." The basic components contained in each HCU and their functions are as follows:

Insert Drive Valve

The insert drive valve is a solenoid-operated valve which opens on an insert signal to supply drive water to the bottom of the main drive piston.

Insert Exhaust Valve

The insert exhaust valve is a solenoid-operated valve which opens on an insert signal to discharge water from above the drive piston to the exhaust header.

Withdrawal Drive Valve

The withdrawal drive valve is a solenoid-operated valve which opens on a withdrawal signal to supply water to the top side of the drive piston.

Withdrawal Exhaust Valve

The withdrawal exhaust valve is a solenoid-operated valve which opens on a withdrawal signal to discharge water from below the main drive piston to the exhaust header.

Speed Control Valves

The speed control valves, which regulate the control rod insertion and withdrawal rates during normal operation, are manually adjustable flow control valves used to regulate the water flow to and from the volume beneath the main drive piston. Once a speed control valve is properly adjusted, it is not necessary to readjust the valve except to compensate for changes in piston seal leakage.

Scram Pilot Valves

The scram pilot valves are operated from the RPS trip system. The scram pilot valves control both the scram inlet valve and the scram exhaust valve. The scram pilot valves are three-way, solenoid-operated, normally energized valves. Two normally energized solenoids maintain air pressure to scram inlet and outlet valves. The pilot valves are arranged as shown in Figure 3.4.6 and Drawing M-357 so that the RPS system power must be removed from both solenoids before air pressure is discharged from the scram valve operators.

Scram Inlet Valve

The scram inlet valve opens to supply pressurized water to the bottom of the drive piston. The valve is a globe type valve which is opened by the force of an internal spring and system pressure and closed by air pressure applied to the top of its diaphragm operator. The opening force of the spring is approximately 700 lb. The valve opening time is approximately 0.1 sec from start to full open. The valve has a position indicator switch which energizes a light in the control room as soon as both the inlet and outlet valves start to open.

Scram Outlet Valve

The scram outlet valve opens slightly before the scram inlet valve exhausting water from above the drive piston. A quicker opening time is achieved because of a larger spring in the valve operator. Otherwise, this valve is similar to the scram inlet valve.

Scram Accumulator

The scram accumulator stores sufficient energy to fully insert a control rod independent of any other source of energy. The accumulator consists of a water volume pressurized by nitrogen. The accumulator has a piston separating the water on top from the nitrogen below. A check valve in the charging line to each accumulator prevents loss of water in the event supply pressure is lost.

During normal plant operation, the pressure on the water side of the accumulator is a function of the charging water header pressure. The nitrogen side of the accumulator is manually charged and the pressure is maintained according to the plant Technical Specifications. Decrease in nitrogen pressure below a specified setpoint actuates the pressure switch and sounds an alarm in the control room.

Also to ensure that the accumulator is always capable of producing a scram, it is continuously monitored for water leakage. A float-type level switch actuates an alarm if water leaks past the barrier and collects in the accumulator instrumentation block. The accumulator instrumentation block is located below the accumulator (nitrogen side) in such a way that it will receive any water which leaks past the accumulator piston.

The scram accumulator meets the requirements of safety design basis 4d.

3.4.5.4 Control Rod Drive System Operation

The CRDS performs three operational functions: rod insertion, rod withdrawal, and scram. The functions are described as follows.

Rod Insertion

Rod insertion is initiated by a signal from the operator to the insert valve solenoids which opens both insert valves. The insert drive valve applies reactor pressure plus approximately 90 psig to the bottom of the drive piston.

The insert exhaust valve opens to allow water from the drive piston to discharge to the exhaust header.

As illustrated in Figure 3.4.5, the locking mechanism is a ratchet type device and does not interfere with rod insertion. The speed at which the drive moves is determined by the pressure drop through the insert speed control valve which is set for about 4 gpm for a speed of 3 in/sec (nominal). During normal insertion, the pressure on the downstream side of the speed control valve is 90 to 100 psi above reactor vessel pressure. However, if the drive slows down for any reason, the flow through and pressure drop across the insert speed control valve will decrease and the full 250 psi (min) differential pressure will be available to cause continued insertion. With 250 psi differential pressure acting on the drive piston, the piston exerts an upward force of 1,000 lb.

Rod Withdrawal

Drive withdrawal is, by design, more involved. The collet fingers (latch) must be raised to reach the unlocked position as in Figure 3.4.4. The notches in the index tube hold the collet fingers in place. The index tube must be lifted before the collet fingers can be released. This is done by opening the drive insert valves (in the manner described in the preceding paragraph) for approximately 1 sec. The withdraw valves are then opened, applying driving pressure above the drive piston and opening the area below the piston to the exhaust header. Pressure is simultaneously applied to the collet piston. As the collet piston rises, the collet fingers are cammed outward away from the index tube by the guide cap.

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

Rod Scram

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

The large differential pressure (initially about 1,400 psi and always several hundred psi depending on reactor vessel pressure) produces a large upward force on the index tube and control rod, giving the rod a high initial acceleration and providing a large margin of force to overcome friction. The characteristics of the hydraulic system are such that, after the initial acceleration is achieved, the drive continues at a fairly constant velocity. This characteristic provides a high initial rod insertion rate. As the drive piston nears the top of its stroke, the piston seals close off the large passage in the stop piston tube and the drive slows down.

Each drive requires about 2.5 gal of water during the scram stroke. There is adequate water capacity in each drive accumulator to complete a scram in the required time at low reactor vessel pressure. At higher reactor vessel pressures, the accumulator is assisted on the upper end of the stroke by reactor vessel pressure acting on the drive via the ball check (shuttle) valve. As water is forced from the accumulator, the accumulator discharge pressure falls below reactor vessel pressure. This causes the check valve to shift its position to admit reactor pressure under the drive piston. Thus, reactor vessel pressure furnishes the force needed to complete the scram stroke at higher reactor vessel pressures. When the reactor vessel is up to full operating pressure, the accumulator is actually not needed to meet scram time requirements. With the reactor at 1,000 psig and the scram discharge volume at atmospheric pressure, the scram force without an accumulator is over 1,000 lb.

The CRDS, with accumulators, provides the following maximum scram time performance when reactor steam dome pressure $\geq$800 psig.

Notch Position	Scram Time* (sec)
46	0.44
36	1.08
26	1.83
06	3.35

* Based on de-energization of scram pilot valve solenoids at time zero. When reactor steam dome pressure <800 psig, established scram time limits apply.

3.4.6 Safety Evaluation

3.4.6.1 Evaluation of Control Rods

It is apparent from the foregoing description that the control rods meet the design basis requirements. The description also indicates how the control rod-to-drive coupling unit meets design basis requirements.

3.4.6.2 Evaluation of Control Rod Velocity Limiter

The control rod velocity limiter limits the free fall velocity of the control rod to a value which cannot result in nuclear system process barrier damage[1] as required by safety design basis 1c. This velocity is evaluated by the rod drop accident analysis in Section 14.0, "Plant Safety Analysis."

The following sequence of events is necessary to postulate an accident in which the control rod velocity limiter is required:

1. The rod-to-drive coupling fails.

2. The control rod sticks near the top of the core.

3. The drive is withdrawn and the control rod does not follow.

4. The operator fails to notice the lack of plant response as the CRD is withdrawn.

5. The control rod later becomes loose and falls freely to the fully withdrawn position.

3.4.6.3 Evaluation of Scram Time

The rod scram function of the CRDS provides the negative reactivity insertion which is required by safety design basis 2. The scram time shown in the description is adequate as shown by the transient analyses of Section 14.0, "Plant Safety Analysis."

3.4.6.4 Analysis of Malfunctions Relating to Rod Withdrawal

There are no known single malfunctions which could cause even a single rod to withdraw. The following malfunctions have been postulated and the results analyzed.

a. Drive Housing Fails at Attachment Weld

   The bottom head of the reactor vessel has a penetration with an internal nozzle for each control rod drive location. A drive housing is raised into position inside each penetration and fastened to the top of the internal nozzle with a J-weld. The drive is raised into the drive housing and bolted to a flange at the bottom of the housing. The basic failure considered is a complete circumferential crack through the housing wall at an elevation just below the J-weld. The housing material is seamless type 304 stainless steel pipe with a minimum tensile strength of 75,000 psi.

Static loads on the housing wall include the weight of the drive and the control rod, the weight of the housing below the attachment weld to the vessel nozzle, and reactor pressure acting on the 6-in. diameter cross-sectional area of the housing and the drive. Dynamic loading is due to the reaction force during drive operation.

If the housing were to fail as described, the following sequence of events is foreseen. The housing would separate from the vessel and the control rod; the drive and the housing would be blown downward against the support structure by reactor pressure acting on the cross-sectional area of the housing and the drive. The amount of downward motion of the drive and associated parts would be determined by the gap between the bottom of the drive and the support structure deflection under load. In the current design, maximum deflection is approximately 3 in. If the collet were to remain latched, no further control rod ejection would occur(2). The housing would not drop far enough to clear the vessel penetration. Reactor water would leak through the 0.06-in. diametral clearance between the housing outer diameter and the vessel penetration inner diameter at a rate of approximately 440 gpm.

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

b. Rupture of Either or Both Hydraulic Lines to a Drive Housing Flange

1. Pressure-Under Line Breaks

In this case, a partial or complete circumferential opening is postulated at or near the point where the line enters the housing flange.

If the pressure-under line were to fail, and if the collet were latched, no control rod withdrawal would occur. There would be no pressure differential across the collet piston in this case, and therefore no tendency to unlatch the collet. Consequently, it would not be possible to either insert or withdraw the control rod involved.

If reactor pressure were to shift the drive ball check valve against its upper seat, the broken pressure-under line would be sealed off. If the ball check valve were to be prevented from seating, reactor water would leak to the atmosphere. Cooling water could not be supplied to the drive involved because of the broken line. Loss of cooling water would cause no immediate damage to the drive. However, prolonged drive exposure to temperatures at or near reactor temperature could lead to deterioration of material in the seals. High temperature would be indicated to the operator by the thermocouple in the position indicator probe and leakage by operation of the drywell sump pump.

If the basic line failure were to occur at the same time the control rod is being withdrawn, and if the collet were to remain open, calculations indicate that the steady-state control rod withdrawal velocity would be ft/sec. In this case, however, there would not be sufficient hydraulic force to hold the collet open and spring force would normally cause the collet to latch, stopping rod withdrawal.

2. <u>Pressure-Over Line Breaks</u>

The failure considered is complete breakage of the pressure-over line at or near the point where the line enters the housing flange. If the line were to break, pressure over the drive piston would drop from reactor pressure to atmospheric pressure. If there were any significant reactor pressure (approximately 500 psig or greater) it would act on the bottom of the drive piston, and the drive would insert to the fully inserted position. Drive insertion would occur regardless of the operational mode at the time of the failure. After full insertion, reactor water would leak past the stop piston seals, the contracting seals on the drive piston and the collet piston seals. This leakage would exhaust to atmosphere through the broken pressure-over line. In an experiment to simulate this failure, a leakage rate of 80 gpm has been measured with reactor pressure at 1,000 psi. If the reactor were hot, drive temperature would increase. The reactor operator would be apprised of the situation by indication of the fully inserted drive, by high drive temperature indicated and printed out on a recorder in the control room, and by operation of the drywell sump pump.

3. Coincident Breakage of both Pressure-Over and Pressure-Under Lines

This failure would require simultaneous occurrence of the failures described. Pressures above and below the drive piston would drop to zero and the ball check valve would shift to close off the broken pressure-under line. Reactor water would flow from the annulus outside of the drive through the vessel ports to the space below the drive piston. As in the pressure-over line break case, the drive would then insert at a speed dependent on reactor pressure. Full insertion would occur regardless of the operational mode at the time of failure. Reactor water would leak past the drive seals and out of the broken pressure-over line to the atmosphere as described above. Drive temperature would increase. The reactor operator would be apprised of the situation by indication of the fully inserted drive, by high drive temperature printed out by a recorder in the control room, and by operation of the drywell sump pump.

c. All Drive Flange Bolts Fail in Tension

Each CRD is bolted to a flange at the bottom of a drive housing which is welded to the reactor vessel. Bolts are made of AISI-4140 steel or AISI-4340 steel with a minimum tensile strength of 125,000 psi. Each bolt has an allowable load capacity of at least 15,200 lb.

Capacity of the eight bolts is at least 121,500 lb. The major load on all eight bolts, due to reactor design pressure of 1,280 psi, is 30,400 lb.

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

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

If the bolt failure were to occur while the control rod is being withdrawn, pressure below the collet piston would drop to zero and the collet, with 1,650 lb return force, would latch, stopping rod withdrawal.

d. <u>Weld Joining Flange to Housing Fails in Tension</u>

The failure considered is a crack in or near the weld joining the flange to the housing that extends through the wall and completely around the circumference of the housing so that the flange can separate from the housing. The flange material is forged type 304 stainless steel with a minimum tensile strength of

75,000 psi. The housing material is seamless type 304 stainless steel pipe with a minimum tensile strength of 75,000 psi. A conventional full penetration weld of type 308 stainless steel is used to join the flange to the housing. Minimum tensile strength is approximately the same as the parent metal. The design pressure is 1,250 psig and the design temperature is 575°F. A combination of reactor pressure acting downward on the cross-sectional area of the drive; the weight of the control rod, drive, and flange; and the dynamic reaction force during drive operation result in a maximum tensile stress at the weld of approximately 6,000 psi.

In the event that the basic failure described above were to occur, the flange and the attached drive would be blown downward against the support structure. The support structure loading would be slightly less severe than in drive housing failure, since reactor pressure would act only on the drive cross-sectional area. Since there would be no differential pressure across the collet piston, the collet would remain latched and control rod motion would be limited to approximately 3 in. Downward drive movement would be small; therefore, most of the drive would remain inside the housing. The pressure-under and pressure-over lines are flexible enough to withstand the small downward displacement and remain attached to the flange. Reactor water would follow the same leakage path as previously described in 3.4.6.4c, except that the exit to the atmosphere would be through the tap between the lower end of the housing and the top of the flange. Water would flash to steam in the annulus surrounding the drive. The leakage rate would be approximately 840 gpm.

If the flange-to-housing joint failure were to occur at the same time the control rod is being withdrawn (a small fraction of the total operating time), and if the collet were held unlatched, the flange would separate from the housing, the drive and flange would be blown downward against the support structure, and the calculated steady-state rod withdrawal velocity would be 0.13 ft/sec. Since the pressure-under and pressure-over lines remain intact, driving water pressure would continue to be supplied to the drive and the normal exhaust line restriction would exist. The pressure below the velocity limiter piston would decrease below normal due to leakage out of the gap between the housing and the flange to the atmosphere. This differential pressure across the velocity limiter piston would result in a net downward force of approximately 70 lb. However, leakage out of the housing would greatly reduce the pressure in the annulus surrounding the drive so that the net downward force on the drive piston would be less than normal. The overall effect would be a reduction of rod withdrawal speed to a value approximately one-half of normal speed. The collet would remain unlatched with a 560-psi differential across the collet piston, but should relatch as soon as the drive signal is removed.

e. Housing Wall Ruptures

The failure considered in this case is a vertical split in the drive housing wall just below the bottom head of the reactor vessel. The hole was considered to have a flow area equivalent to the annular area between the drive and the thermal sleeve so that flow through this annular area, rather than flow through the hole in the housing, would govern leakage flow. The housing is made from type 304 stainless steel seamless pipe having a minimum tensile strength of 75,000 psi. The maximum hoop stress of 11,900 psi is due primarily to reactor design pressure of 1,250 psig acting on the inside of the housing.

If the housing wall rupture described above were to occur, reactor water would flash to steam and leak to the atmosphere at approximately 1,030 gpm through the hole in the housing. Choke-flow conditions described in 3.4.6.4c would exist. In this case, however, the leakage flow would be greater because the flow resistance is less; that is, the leaking water and the steam would not have to flow down the length of the housing to reach the atmosphere. Critical pressure at which the water would flash to steam is 350 psi.

There would be no pressure differential across the collet piston tending to cause collet unlatching, but the drive would insert due to loss of pressure in the drive housing, and therefore, in the space above the drive piston.

If the housing wall failure were to occur at the same time the control rod is being withdrawn (a small fraction of the total operating time), the drive would stop withdrawing, but the collet would remain unlatched. The drive stoppage would be caused by a reduction in the net downward force acting on the drive line. This would occur when the leakage flow of 1,030 gpm reduces the pressure in the annulus outside the drive to approximately 540 psig and therefore reduces the pressure acting on the top of the drive piston to this value. There would be a pressure differential of approximately 710 psi across the collet piston, holding the collet unlatched as long as the operator held the withdraw signal.

f. Flange Plug Blows Out

A 3/4-in diameter hole is drilled in the drive flange to connect the vessel ports with the bottom of the ball check valve. The outer end of this hole is sealed with an 0.812-in diameter plug 0.250 in thick. The plug is held in place with a full-penetration weld of type 308 stainless steel. The failure considered is a full circumferential crack in this weld and subsequent blowout of the plug.

If the weld were to fail and the plug were to blow out, there would be no control rod motion provided the collet remains latched. There would be no pressure differential across the collet piston tending to cause collet unlatching. Reactor water would leak past the velocity limiter piston, down the annulus between the drive and the thermal sleeve through the vessel ports and drilled passage, and out the open plug hole to the atmosphere at approximately 320 gpm. This leakage calculation is based on liquid only exhausting from the flange as a worst case. Actually, hot reactor water would flash to steam, and choke-flow conditions would exist, so that the expected leakage rate would be lower than the calculated value. Drive temperature would rise, and the alarm would signal the operator.

If the plug weld failure were to occur at the same time the control rod is being withdrawn (a small percentage of the total operating time), and if the collet were to stay unlatched, calculations indicate that control rod withdrawal speed would be approximately 0.24 ft/sec. Leakage out of the open plug hole in the flange would cause reactor water to flow downward past the velocity limiter piston. The small differential pressure across the piston would result in an insignificant driving force of approximately 10 lb tending to increase withdraw velocity.

The collet would be held unlatched by a 295-psi pressure differential across the collet piston as long as the driving signal was maintained.

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

g. <u>Pressure Regulator and Bypass Valves Fail Closed (Reactor Pressure 0 psig)</u>

Pressure in the drive water header supplying all drives is controlled by regulating the amount of water from the supply pump that is bypassed back to the reactor. This is accomplished primarily with the drive water control valves, and secondarily with the pressure stabilizing valves. There are two drive water control valves arranged in parallel. One is a motor-operated valve that can be adjusted from the control room. This valve is normally in service and is partially open to maintain a pressure of reactor pressure plus 250 psig in the header just upstream from the valve. The other is a hand-operated valve that is normally closed but that can be operated locally whenever the motor-operated valve is out of service.

The pressure stabilizing valves are solenoid-operated and have built-in needle valves for adjusting flow. The two valves are arranged in parallel between the drive water header and the return line to the reactor. One valve is set to bypass 2 gpm and closes when any drive is given a withdraw signal, so that flow is diverted to the drive being operated rather than back to the reactor. Relatively constant header pressure is thus maintained. Similarly, the other valve is set to bypass 4 gpm and closes when any drive is given an insert signal.

The failure considered is when all of these valves are closed so that maximum supply pump head of 1,700 psi builds up in the drive water header. The major portion of the bypass flow normally passes through the motor-operated valve; therefore, closure of this valve is most critical.

Since the lowest exhaust line pressure exists when reactor pressure is zero, this reactor condition is also assumed.

If the valve closure failure described above were to occur at the same time the control rod is being withdrawn, calculations indicate that steady-state withdrawal speed would be approximately 0.5 ft/sec or twice normal velocity. The collet would be held unlatched by a 1,670-psi pressure differential across the collet piston. Flow would be upward past the velocity limiter piston, but retarding force would be negligible.

h. Ball Check Valve Fails to Close Off Passage to Vessel Ports

The failure considered in this case depends upon the following sequence of events. If the ball check valve were to seal off the passage to the vessel ports during the up signal portion of the jog withdraw cycle, the collet would be unlatched. This is the normal withdrawal sequence. Then if the ball were to move up and become jammed in the ball cage by foreign material or prevented from reseating at the bottom by foreign material that settles out on the seat surface, water from below the drive piston would return to the reactor through the vessel ports and the annulus between the drive and the housing. Since this return path would have lower than normal flow resistance, the calculated withdrawal speed would be 2 ft/sec. During withdrawal, there would be a differential pressure across the collet piston of approximately 40 psi. Therefore, the collet would tend to latch and would have to stick open before continuous withdrawal at 2 ft/sec could occur. Water would flow upward past the velocity limiter piston and a small retarding force would be generated (approximately 120 lb).

i. Hydraulic Control Unit Valve Failures

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

j. Failure of the Collet Fingers to Latch

The drive continues to withdraw (after removal of the signal) at a fraction of its normal withdrawal speed. There is no known means for the collet fingers to become unlocked without some initiating signal. Failure of the withdrawal drive valve to close following a rod withdrawal has the same effect as failure of the collet fingers to latch in the index tube and is immediately apparent to the operator. Accidental opening of the withdrawal drive valve normally does not unlock the collet fingers because of the characteristic of the collet fingers to remain locked until unloaded.

k. Withdrawal Speed Control Valve Failure

Normal withdrawal speed is determined by differential pressures at the drive and set for a nominal value at 3 in/sec. The characteristics of the pressure regulating system are such that withdrawal speed is maintained independent of reactor vessel pressure. Tests have determined that accidental opening of the speed control valve to the full open position produces a velocity of approximately 6 in/sec.

The CRDS prevents rod withdrawal as required by safety design basis 4a. It has been shown that only multiple failures in a drive unit and its control unit could cause an unplanned rod withdrawal.

3.4.6.5 Scram Reliability

High scram reliability is the result of a number of features of the CRDS, such as the following:

1. There are two sources of scram energy to insert each control rod when the reactor is operating: accumulator pressure and reactor vessel pressure.

2. Each drive mechanism has its own scram and pilot valves so that only one drive can be affected by failure of a scram valve to open. Two pilot valve solenoids are provided for each drive. Both solenoids must be de-energized to initiate a scram.

3. The RPS and HCU's are designed so that the scram signal and mode of operation override all others.

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

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

The scram reliability meets the requirements of safety design bases 4b and 4c.

3.4.6.6 Control Rod Support and Operation

As shown in the description, each control rod is independently supported and controlled as required by safety design basis 3.

3.4.7 Inspection and Testing

3.4.7.1 Development Tests

The development drive (one prototype) testing included over 5,000 scrams and approximately 100,000 latching cycles during 5,000 hr of exposure to simulated operating conditions. These tests have demonstrated the following:

1. The drive withstands the forces, pressures, and temperatures imposed without difficulty.

2. Wear, abrasion, and corrosion of the nitrided type 304 stainless parts are negligible. Mechanical performance of the nitrided surface is superior to materials used in earlier operating reactors.

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

4. Usable seal lifetimes greater than 1,000 scram cycles may be expected.

3.4.7.2 Factory Quality Control Tests

Quality control of welding, heat treatment, dimensional tolerances, material verification, etc, were maintained throughout the manufacturing process to assure reliable performance of the mechanical reactivity control components. Some of the quality control tests on the control rods, CRD mechanisms, and HCU's are as follows:

Control Rod Absorber Tube Tests

1. The integrity of the tubing and end plug material was verified by ultrasonic inspection.

2. Boron contents of the boron-10 fraction of each lot of boron-carbide was verified.

3. The weld integrity of the finished absorber tubes was verified by helium leak testing.

CRD Mechanism Tests

1. Hydrostatic testing of the drives to check pressure welds was in accordance with ASME codes.

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

3. All drive parts which cannot be visually inspected for dirt are flushed with filtered water at high velocity. No significant foreign material was permissible in effluent water.

4. Seal leakage tests were performed to demonstrate proper seal operation.

5. Each drive was tested for motion, latching, and control rod position indicating.

6. Each drive was subjected to cold scram tests at various reactor pressures to verify proper scram performance.

Hydraulic Control Unit Tests

Each HCU received the following tests:

1. All hydraulic systems were hydrostatically tested in accordance with ANSI B31.1.

2. All electrical components and systems were tested for electrical continuity and resistance to ground.

3. The correct operation of the accumulator pressure and level switches was verified.

4. The unit's ability to perform its part of a scram was demonstrated.

5. Proper operation and adjustment of the insert and withdrawal valves was demonstrated.

3.4.7.3 Operational Tests

After installation, all rods, HCU's, and drive mechanisms were tested through their full range for operability.
During normal operation, each time a control rod is withdrawn a notch, the operator can observe the in-core monitor indications for proper neutron response 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, withdrawing the rod one notch, and returning it to its original position while the operator observes the in-core monitor indications.

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

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

Control rod scram time testing is performed as required by Technical Specification (TS) 3.1.4, Control Rod Scram Times and is further described in TS 3.1.4 Bases. Control rod scram time testing is required to verify the continued performance of the scram function during the operating cycle by testing a representative sample of control rods. A representative sample contains at least 10% of the control rods. As a result of the implementation of TS amendment 262/266 (Control Rod Time Testing Frequency), the acceptance criteria for the number of control rods that fail to insert within the time limitations of TS Table 3.1.4-1 was revised to 7.5% of the total rods tested during that surveillance. The frequency of the testing is based on operating experience that has shown control rod scram times do not significantly change over an operating cycle.

3.4.8 Longer Life Control Rods Assembly

PBAPS currently uses a combination of control rods originally supplied with the plant (all-B_4C rods) and several of the longer life control rods (Duralife series D-120, D-160, D-190, and/or D-230, References 3, 4, 5, and 7). The Duralife series of control rods were designed to be direct replacement rods for the original all-B_4C blades. The new rods were designed to increase the lifetime expectancy of the control rods by replacing some of the B_4C absorber rods with solid hafnium absorber material and/or by using an improved B_4C absorber rod tube material.

The essential differences between the Duralife control rods and the original all-B_4C control rods are as follows:

1. Improved B_4C absorber rod tube material is used to eliminate cracking due to IGSCC (D-120, D-160, D-190, and D-230),

2. Some B_4C absorber rods are replaced with solid hafnium material to eliminate mechanical strain due to internal gas swelling (D-160, D-190, and D-230),

3. A 6-inch hafnium absorber plate is added to the top of each wing to increase blade lifetime (D-190 and D-230),

4. A lighter weight velocity limiter is used to compensate for the heavier hafnium material (D-190 and D-230),

5. Improved pin and roller materials are used to reduce future radiation exposure problems (D-120, D-160, D-190, and D-230),

6. Increased volume of neutron absorber material (B_4C and hafnium) to increase blade life (D-230).

In addition, new pin and roller material (PH 13-Mo and Inconel X-750) were used to replace the cobalt material which becomes very radioactive during rod exposure.

A lightweight velocity limiter was designed and incorporated in the D-190 and D-230 control rod assemblies to offset the increased weight of the hafnium absorber material. The new velocity limiter is lighter than the previous velocity limiter while maintaining the rod drop velocity below the design basis limits. The velocity limiter has been subjected to extensive testing to confirm its ability to meet all performance and design requirements. The velocity limiter dimensions are within the envelope of the original limiter and, thus, are compatible with all NSSS hardware.

The performance of the Duralife series of control rods has been compared with the performance of the current all-B_4C control rods in regards to the LHGR, MCPR, and MAPHGR thermal limits. The D-120, D-160, D-190, and D-230 control rod weights and rod worths are comparable to the values used in the original all-B_4C control rods. Therefore, the scram speed and scram reactivity are also comparable. It follows then that the LHGR, MCPR, and Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) thermal limits are not affected by the Duralife series of control rods (References 3, 4, and 5).

3.4.9 Marathon Control Rod Assembly

PBAPS currently uses a combination of control rods originally supplied with the plant (all B_4C rods), several longer life rods (Duralife series D-120, D-160, D-190, and/or D-230, References 3, 4, 5, and 7), and Marathon (Marathon and Marathon-5S, References 6 and 10) control rods. The Marathon control rods are designed to be direct replacements for the existing rods and were designed to increase the resistance to irradiation stress corrosion cracking (IASCC) and increase control rod performance by:

a) Using a structural tube configuration to perform the function of the sheath and the absorber rods, which reduces the weight and increases the absorber volume.

b) Using a segmented tie rod instead of a full length tie rod, which reduces weight.

c) Replaces the stainless steel balls and associated crimps of B_4C absorber with separate B_4C filled capsules which dramatically improves resistance to IASCC and adds design flexibility.

d) Uses material which is highly resistant to IASCC to fabricate the unique tube configuration, which increases lifetime.

The essential differences between the Marathon control rod and the Duralife-230 design is the replacement of the absorber tube and sheath arrangement with an array of square tubes, which reduces the weight and increases the absorber volume (Reference 6). In addition, the full length tie rods are replaced with a segmented tie rod which reduces weight.

Pin and roller material (PH 13-Mo and Inconel X-750), used on some Marathon control rods, is consistent with the material used in the Duralife series control rods. Pad material (316 stainless steel), used on later Marathon control rods, is consistent with the other materials used in the Marathon control rod.

The velocity limiter can be either the original velocity limiter design (used with the all-B_4C rods) or the lighter weight velocity limiter design (used in the Duralife series design). The selection is based on the control rod assembly weight requirements. Each type of velocity limiter is compatible with all NSSS hardware.

The comparison confirmed that the reactivity worth of the control rods are within +/- 5% of the all B_4C rod, thus the Marathon does not need special treatment in the core analysis (Reference 6).

A high reactivity worth version of the Marathon control rod is available as an option. If the reactivity worth of this option exceeds the original B_4C rod by +/- 5% then the plant core analysis will account for the increase.

3.4.10 Westinghouse Atom Control Rod Assembly

The Westinghouse Atom CR 82M-1 rod is an approved control rod for use PBAPS Units 2 and 3. The CR 82M-1 rods were designed to be a direct replacement for the currently used D-120 rods. The new rods will be inserted in subsequent cycles as the D-130 rods are depleted. (Reference 8)

3.4.11 GEH Ultra Control Rod Assembly

General Electric-Hitachi (GEH) has developed the GEH Ultra MD (also known as Marathon-5S or Marathon Ultra MD) control rod is a derivative of the Marathon design. The primary difference between the Ultra MD and the original Marathon design is a simpler absorber tube geometry. The new simplified absorber tubes use the same crack resistant stainless steel as the original Marathon design.

GEH has also developed the Ultra HD (also known as Marathon-Ultra) control rod design. The only difference between the Ultra HD and the Ultra MD design is the absorber section load pattern. Where the Ultra MD is an all-boron carbide capsule design, the Ultra HD incorporates full-length hafnium rods in outer edge, high depletion tube locations. The geometry and composition of these hafnium rods is identical to those used in the Marathon design. In addition, to maximize the neutron absorber mass, thin-wall capsules are used, with a similar wall thickness to the capsules in the Marathon design.

A nuclear evaluation of the Ultra HD and Ultra MD control rod shows that the initial cold and hot reactivity worths are within ± 5% of the original equipment control rod. Therefore, the Ultra HD and Ultra MD is a direct nuclear replacement for previous control rod designs.

The structure of the Ultra HD and Ultra MD control rod has been evaluated during all normal and upset conditions, and has been found to be mechanically acceptable. The fatigue usage of the control rod has also been found to be well below lifetime limits. (References 10 and 11)

3.4 REACTIVITY CONTROL MECHANICAL DESIGN

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3.5 CONTROL ROD DRIVE HOUSING SUPPORTS

3.5.1 Safety Objective

The safety objective of the CRD housing supports is to protect against additional damage to the nuclear system process barrier, or damage to the fuel barrier, by preventing any significant nuclear transient in the event a drive housing breaks or separates from the bottom of the reactor vessel.

3.5.2 Safety Design Basis

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

2. Clearance is provided between the housings and the supports to prevent vertical contact stresses due to thermal expansion during plant operation.

3.5.3 Description

The control rod housing supports are illustrated in Figure 3.5.1. Horizontal beams are installed immediately below the bottom head of the reactor vessel, between the rows of control rod housings, and are bolted to brackets welded to the steel form liner of the drive room in the reactor support pedestal.

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

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

Individual grids rest on the support bars between adjacent beams. Because a single-piece grid would be difficult to handle in the limited workspace and because it is necessary that CRD's, position indicators, and in-core instrumentation components are accessible for inspection and maintenance, each grid is designed to be assembled or disassembled in place. Each grid assembly is made from two grid plates, a clamp, and a bolt. The top part of the clamp acts as a guide to assure that each grid is correctly positioned directly below the respective CRD housing which it would support in the postulated accident.

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

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

The American Institute of Steel Construction (AISC) Specification for the Design, Fabrication, and Erection of Structural Steel for Building is used in the design of the CRD housing support system. However, to provide a structure that absorbs as much energy as practical without yielding, the allowable tension and bending stresses are taken as 90 percent of yield, and the shear stress as 60 percent of yield. These are 1.5 times the corresponding AISC allowable stresses of 60 percent and 40 percent of yield. This stress criterion is considered desirable for this application and adequate for the "once in a lifetime" loading condition.

For mechanical design purposes, the postulated failure resulting in the highest forces is an instantaneous circumferential separation of the CRD housing from the reactor vessel, with an internal pressure of 1,250 psig (reactor vessel design pressure) acting on the area of the separated housing. The weight of the separated housing, CRD, and blade, plus the pressure force of 1,250 psig acting on the area of the separated housing give a force of approximately 35,000 lb. This force is multiplied by a factor of 3 for impact, conservatively assuming the housing travels through a 1-in gap before contacting the supports. The total force (10^5 lb) is then treated as a static load in design. The CRD housing supports are designed to seismic Class I criteria.

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

grid	- ASTM-A-441
disc springs	- Schnorr Type BS-125-71-8
hex bolts and nuts	- ASTM-A-307

3.5.4 Safety Evaluation

Downward travel of CRD housing and its control rod following the postulated housing failure is the sum of the compression of the disc springs under dynamic loading and the initial gap between the grid and the bottom contact surface of the CRD flange. If the reactor were cold and pressurized, the downward motion of the control rod would be limited to the approximate 2-in spring compression plus approximately a 1-in gap. If the reactor were hot and pressurized, the gap would be approximately 1/4 in and the spring compression slightly less than in the cold condition. In either case, the control rod movement following a housing failure is limited substantially below one drive notch" movement (6 in). The nuclear transient from sudden withdrawal of any control rod through a distance of one drive notch at any position in the cores does not result in a transient sufficent to cause damage to any radioactive material barrier. This meets the fuel damage limitation of safety design basis 1.

The CRD housing supports are in place during power operation when the nuclear system is pressurized. The housing supports may be removed when the reactor is in the shutdown condition even when the reactor is pressurized, because all control rods are then inserted. Even if a control rod is ejected during shutdown, the reactor remains subcritical because it is designed to remain subcritical.

At plant operating temperature a gap of approximately 1/4 in exists between the CRD housing and the supports; at lower temperatures the gap is greater. Because the supports do not come in contact with any of the CRD housings, except during the postulated accident condition, vertical contact stresses are prevented as required by safety design basis 2.

3.5.5 Inspection and Testing

When the reactor is in the shutdown mode, the CRD housing supports may be removed for inspection and maintenance of the CRD's. When the support structure is reinstalled, it is inspected for proper assembly, particular attention being given to assure that the correct gap between the CRD lower contact surface and the grid is reestablished.

3.6 NUCLEAR DESIGN

Peach Bottom Units 2 and 3 are BWR/4's with 251 inch vessels, 764 fuel assemblies. For the most part, tables and figures are not contained in this chapter. However, applicable tables and figures from the referenced documents are listed. Much of the information of Section 3.6 is provided in the licensing topical report, GESTAR II (Reference 8). Any additions or differences are given below for each applicable subsection.

3.6.1 Power Generation Objective

1. Attain rated power generation from the nuclear fuel for a given period of time.

2. Attain reactor nuclear stability throughout core life.

3. Allow normal power operation of the nuclear fuel without sustaining fuel damage.

3.6.2 Plant Performance Design Bases

The core and fuel design meets the following bases:

1. The design has adequate excess reactivity to attain the desired cycle length.

2. The design is capable of operating at rated conditions without exceeding technical specification limits.

3. The core and fuel design and the reactivity control system allow continuous, stable regulation of reactivity.

4. The core and fuel design have adequate reactivity feedback to facilitate normal operation.

3.6.3 Safety Design Bases

Information on the Design Basis is referenced in subsection 3.1 of GESTAR II (Reference 8).

3.6.4 Nuclear Requirements

The following nuclear requirements are established for systems and equipment other than the fuel itself. The fuel nuclear design is compatible with these requirements.

3.6.4.1 Control Rods

The control rods are of such number and reactivity worths that the insertion of all but the control rod of highest worth is sufficient to make the fuel subcritical under the most reactive condition of the nuclear system. The control rod of highest worth is considered fully withdrawn.

3.6.4.2 Reactor Manual Control System

1. Control rod operating patterns and withdrawal sequences are specified so that control rod worths are low enough to prevent damage to the nuclear system process barrier as a result of any single control rod dropping from the fully inserted position to the fully withdrawn position.

2. The methodology of GE Topical Report "General Electric Standard Application for Reactor Fuel (GESTAR II, Main and U.S. Supplement)", NEDE-24011-P-A, Revision 19, dated May 2012 (ADAMS Accession No. ML121390274) shall be used to analyze control rod startup/shutdown sequences and control rod patterns. Cycle specific control rod patterns during startup and shutdown conditions shall continue to be controlled by the operator and the rod worth minimizer so that only specified control rod sequences and relative positions are allowed over the operating range of all control rods inserted to 10 percent of rated power.

3. Control rod withdrawal increment (notch) sizes are limited so that rod movement of one notch does not result in less than a 20-sec reactor period.

3.6.4.3 Standby Liquid Control System

The standby liquid control system has sufficient reactivity characteristics that it is capable of bringing the reactor from full power to a cold shutdown condition at any time in core life (subsection 3.8, "Standby Liquid Control System").

3.6.5 Fuel Nuclear Characteristics

The plant utilizes a light water moderated reactor, fueled with slightly enriched uranium dioxide. At operating conditions the moderator boils, producing a spatially variable density of steam voids within the core. The use of a water moderator produces a neutron energy spectrum from which the fissions are produced principally by thermal neutrons. The BWR design provides a system for which reactivity changes are inversely proportional to the steam void content in the moderator. This void feedback effect is one of the inherent safety features of the BWR 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.

The fuel for the BWR is uranium dioxide enriched to approximately 2 to 5 weight percent U-235 with the remaining 95 to 98 weight percent U-238. Early in the fuel life the fissioning of the U-235 produces the majority of the energy. The presence of U-238 in uranium dioxide fuel leads to the production of significant quantities of plutonium during core operation. This plutonium contributes to both fuel reactivity and reactor power production, i.e., approximately 50 percent at end-of-life. In addition, direct fissioning of U-238 by fast neutrons yields approximately 7 to 10 percent of the total power and contributes to an increase of delayed neutrons in the core. Finally, the U-238 also has a strong negative Doppler reactivity coefficient that resists power increases during excursions.

3.6.5.1 Nuclear Design Description

The BWR core design consists of a light-water moderated reactor fueled with slightly enriched uranium-dioxide. There are 764 fuel bundles in each of the two cores with each bundle consisting of a matrix of zircaloy fuel rods. Selected fuel rods within each bundle incorporate small amounts of gadolinia as burnable poison. Gd_2O_3 is uniformly distributed in the UO_2 pellet and forms a solid solution. Details of the UO_2-Gd_2O_3 fuel are given in Reference 1.

In the nuclear design of a core, a set of system parameters must be determined which yield safe, reliable and economical reactor operation over the desired core lifetime. The nuclear design analysis consists of a number of models of neutron behavior in the reactor that are implemented by computer programs to simulate nuclear behavior of the reactor core. The nuclear analysis of the core interacts with other aspects of core design, including but not limited to, thermal-hydraulic analysis of core cooling, structural analysis of core components and economic performance. The nuclear analysis associated with the design of a core can be grouped into the following general areas: determination of core criticality and power distributions, reactivity control analysis, depletion analysis, fuel loading and core arrangement and reactor safety analysis.

The reference loading patterns for current and past cycles are documented in the Supplemental Reload Licensing Report (SRLR) for that cycle. The reference loading pattern is the basis for all fuel licensing. The actual as-loaded core may be different than the reference loading pattern. Any differences between the reference loading pattern and the actual as-loaded core are evaluated, resolved and documented in the appropriate design record file for the cycle in question. The as-loaded core can be found in either the unit and cycle specific core design report, the cycle management report or their equivalents.

3.6.5.1.1 Fuel Nuclear Properties

The bundle reactivity is a complex function of several physical properties. The important properties are average bundle enrichment, gadolinia rod location and gadolinia concentration, void fraction, and accumulated exposure. At low exposure the reactivity effect due to void formation is readily apparent; however, at higher exposure, due to the effect of void history, the curves cross. The primary reason for this behavior is the greater rate of plutonium formation at the higher void fraction. Early in the fuel bundle life approximately 93 percent of the power is produced by fissions in U-235 with the remainder coming from fast fissions in U-238. At high exposures typical of discharge, the power production due to plutonium exceeds that of the U-235.

3.6.5.2 Power Distributions

Information on the Power Distribution is referenced in Subsection 3.2.2 of GESTAR II (Reference 8).

3.6.5.2.1 Local Power Distribution

The local rod-to-rod power distribution and the associated R-Factor distribution are direct functions of the lattice fuel rod enrichment distribution. Near the outside of the lattice where the thermal flux peaks due to interbundle water gaps, low enrichment fuel rods are utilized to minimize power peaking. Closer to the center of the bundle, higher enrichment fuel rods (either full or partial length) are used to increase the power generation and flatten the power distribution. In addition, water channels containing unvoided water are at the center of the lattice in order to increase the thermal flux and produce more power in the center of the lattice (fuel design dependent). The combination of these factors results in the relatively flat local power distribution. The fuel rods which contain gadolinia produce relatively little power early in bundle life; however, as the gadolinia is depleted, the power in these rods increases to approximately 90 percent of the lattice average.

The high power rods deplete at a greater rate and the local peaking factor decreases with exposure. The local power distribution tends to flatten with increasing void fraction. The presence of a control blade adjacent to the bundle significantly perturbs the local power distribution. Although the local peaking factor is quite large in this case, the gross power in a controlled bundle is sufficiently low such that a controlled lattice is never limiting.

3.6.5.2.2 Radial Power Distribution

The integrated bundle power, commonly referred to as the radial power, is the primary factor for determining the Minimum Critical Power Ration (MCPR). At rated conditions the MCPR is directly proportional to the radial power peaking. The radial power distribution is a complex function of the fuel bundle type and distribution, the control rod pattern in the core, and the void condition for that bundle and power. A three-dimensional BWR simulator is used to calculate the three-dimensional power distribution in the core, and the power is axially integrated to determine average bundle power.
The radial power distribution is controlled by both the fuel bundle distribution, as dictated by the loading, and the control rod sequence. The control rods are used to optimize radial power peaking throughout the cycle.

3.6.5.2.3 Axial Power Distribution

The axial power distribution is a function of exposure distribution, enrichment distribution, void history, control rod pattern, and the various steady-state void distributions resulting from different recirculation flow rates. The exposure shape and the void histories existing in the bundles which remain from previous cycles provide much of the fixed power shaping. The enrichment distribution provides further shaping since the power level in the top and bottom of the core is significantly reduced by the use of 6 to 12 inches (fuel type dependent) of natural uranium (non-enriched) in the top and 6 inches (fuel type dependent) in the bottom of the fuel rods. The power distribution resulting from these fixed parameters can be further optimized by the use of the two variable parameters: steady-state voids and control rods. The effect of voids is to skew the power toward the bottom of the core, and the effect of the bottom entry control rods is to reduce the power in the bottom of the core and skew the power upwards. The void distribution is determined primarily by the power shape and the recirculation flow rate. These have a much stronger influence than the fixed conditions and are the two mechanisms available for optimizing the axial power shape. Hence, the combination of the exposure distribution, void distribution, and use of natural uranium, modified as needed by the selection of control rod patterns and recirculation flow rate, enable the core to achieve the desired end of cycle exposure distribution which, with all control rods withdrawn, results in optimal uranium utilization.

3.6.5.2.4 Power Distribution Measurements

The measurement of the power distribution within the reactor core together with instrumentation correlations and operations limits are discussed in reference 5.

3.6.5.2.5 Power Distribution Accuracy

The accuracy of the calculated local rod-to-rod power distribution is discussed in reference 6. The accuracy of the radial, axial, and the gross three-dimensional power distribution calculations is discussed in reference 7.

3.6.5.3 Analytical Methods

The nuclear evaluations of all General Electric reload cores are performed using the analytical tools and methods described in Sections 3.3 and 3.4 of GESTAR II (Reference 8 and 15).

3.6.5.4 Reactivity Control

The core and fuel design in conjunction with the reactivity control system provide an inherently stable system for BWR's. The control rod system is designed to provide adequate control of the maximum excess reactivity anticipated during fuel cycle operation. Gadolinia loaded in carefully selected fuel rods of the high enrichment reload fuel bundles compensates for the high reactivity that these bundles would otherwise have early in life. This burnable poison is designed to burn out when the bundle reactivity can be controlled with rods alone. Hence, the reactor can always be brought subcritical by control rod insertion.

Fuel reactivity is influenced by factors such as moderator temperature, xenon concentration, and burnable poisons. When the ratio of the moderator to fuel in the core is relatively small, the reactor operates in an under-moderated condition. When under-moderated, an increase in moderator temperature results in a decrease in moderator density, a resulting decrease in thermal neutrons, and a decrease in power. As the ratio of moderator to fuel increases, the reactor enters an over-moderated condition. With an increase in temperature in this condition, there is also a reduction in neutrons reaching thermal energies, but this is outweighed by a reduction in the number of neutrons absorbed in the moderator. As a result, there is a net increase in neutrons available for fission, and an increase in power.

The strong local absorption effects of burnable poisons in fresh fuel make the core under-moderated. As burnable poisons are depleted during the fuel cycle, the core becomes less under-moderated, potentially, depending on fuel type, leading to a slightly over-moderated condition. As a result, the maximum core reactivity may occur later in the fuel cycle and at a temperature greater than the minimum assumed temperature of 68°F.

The safety design basis requires that the core, in its maximum reactivity condition, be subcritical with the control rod of the highest worth fully withdrawn and all others fully inserted. This allows control rod testing to be performed at any time in core life and assures that the reactor can be made subcritical by control rods alone. Shutdown capability and margin are evaluated assuming a xenon-free core, at the most limiting time during the operating cycle, and at the most limiting temperature at or above 68°F.

3.6.5.4.1 Shutdown Reactivity

Information of Shutdown Reactivity is referenced in Subsection 3.2.4.1 of GESTAR II (Reference 8).

3.6.5.4.2 Reactivity Variations

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

Control rods are used during the fuel burnup, partly to balance the power distribution effect of steam voids as indicated by the in-core flux monitors. The core loading design provides an inherent control of the gross power distribution with control rods and void distributions supplying additional flexibility. This permits control of fuel burnup and isotopic composition throughout the core to the extent necessary to counteract the effects of voids on axial power distribution at the end of a fuel cycle, when a few or no control rods remain in the core.

Reactivity balances have not normally been used in describing BWR behavior because of the strong dependence of, for example, rod worth on temperature and void fraction; therefore, the design process does not produce components of a reactivity balance at the conditions of interest. Instead, it gives the k_{eff} representing all effects combined. Further, any listing of components of a reactivity balance is quite ambiguous unless the sequence of the changes is clearly defined.

Consider, for example, the reactivity effect of control rods and burnable poison. The combined worth of these two absorbers would be considerably different than the sum of their individual worths. Even this combined worth would be of questionable significance unless the path and conditions of other parameters (i.e., temperature, void, xenon, etc) were completely specified. Many other illustrations could be presented showing that the reactivity balance approach, which may be appropriate in some types of reactors, is completely inappropriate in a BWR. This is related to the large potential excess reactivity in a BWR combined with the dependence of interaction (shadowing) factors on reactor state.

3.6.5.5 Refueling Cycle

Refueling is performed approximately every 24 months on a partial, roughly 1/3, batch basis. The replacement fuel, described earlier, is designed to achieve an average exposure of approximately 50 GWD/T. This is roughly 16 to 17 GWD/T per cycle.

3.6.5.5.1 Criticality of the Reactor During Refueling

The reactor is maintained subcritical during refueling by refueling interlocks. These interlocks, designed to back up procedural core reactivity controls during refueling, are described in Section 3.9 of Technical Specifications.

3.6.5.5.2 Criticality of Fuel Assemblies

The criticality of fuel assemblies during refueling operations is addressed in paragraph 3.2.5 of Reference 8.

3.6.5.6 Control Rod Patterns and Reactivity Worths

3.6.5.6.1 Rod Worth Minimizer System Range

Below 10 percent of rated power the control rod patterns are restricted to prescribed withdrawal sequences enforced by the rod worth minimizer system. This system minimizes control rod worths to the extent that they are not an important concern with the operation of a BWR. The consequences of a rod drop accident or a rod withdrawal error in this range are significantly less severe than that required to violate fuel integrity limits.

3.6.5.6.2 Operating Range

In the power range, above 10 percent of rated power, control rod worths are very small due to the formation of voids in the moderator. Therefore, restrictions on control rod patterns are not required to minimize control rod worths. Below 10 percent of rated power the control rod patterns are selected as described in paragraph 3.6.5.1.

3.6.5.6.3 Scram Reactivity

The RPS responds to some abnormal operational transients by initiating a scram. The RPS and the CRDS act quickly enough to prevent the initiating disturbance from driving the fuel beyond transient limits. The scram reactivity curves for the initial cores at specified exposure points are shown for Unit 2 in Figures 3.6.2 and 3.6.3 and for Unit 3 in Figure 3.6.4.

At the hot-operating condition the control rod, power, delayed neutron, and void distributions must all be properly accounted for as a function of time. Therefore, the scram reactivity is calculated using coupled neutronic/thermal-hydraulic space-time models utilizing neutron diffusion theory and including six delayed neutron groups. The codes that perform these calculations are described in Reference 8. The coupled neutronics and thermal-hydraulics properly accounts for the redistribution of the power, neutron flux, and voids during scram.

3.6.5.7 Reactivity Coefficients

The following important reactor core characteristics are discussed and derived in Reference 8:

Topic	Reference 8 Subsection
Reactivity Coefficients	3.2.3
Moderator Temperature Coefficient	3.2.3
Doppler Reactivity Coefficient	3.2.3.1
Moderator Void Coefficient	3.2.3.2

3.6.5.8 Stability

3.6.5.8.1 Xenon Transients

BWR's do not have instability problems due to xenon. This has been demonstrated by operating BWR's for which xenon instabilities have never been observed (such instabilities would readily be detected by the LPRM's), by special tests which have been conducted on operating BWR's in an attempt to force the reactor into xenon instability, and by 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 9.

3.6.5.8.2 Thermal-Hydraulic Stability

This subject is covered in Section 3.2.6.2 of Reference 8.

3.6.6 Changes to the Original GE-BWR/4 Nuclear Design

Relative to the original core fuel design (7x7), several design changes have occurred and either have been or are currently being used in Units 2 and 3. A description of all bundle designs currently in use can be found in Reference 4.

3.6.7 Nuclear Evaluations

The analyses presented in paragraph 3.6.5 show that the safety design basis is satisfied in conjunction with the nuclear design requirements of paragraph 3.6.4. Adequate protection is provided for the cladding and nuclear system process barrier. The nuclear requirements for reactivity control systems and the settings of the reactor protection system are primarily associated with limitations on the levels and rates of change of reactivity, power, and temperature. Normal plant operation is conducted at rates and values of these parameters, such that reactor transients are readily observable and controllable by plant personnel.

The reactor protection system responds to some abnormal operational transients by initiating a scram. The reactor protection system and the CRD system act quickly enough to prevent the initiating disturbance from causing fuel damage. The scram reactivity curves used in the reactivity excursion analyses are included as Figures 3.6.2, 3.6.3, and 3.6.4. Abnormal operational transients are evaluated in Section 14, "Plant Safety Analysis." No fuel damage results from any abnormal operational transient.

The specified rod withdrawal sequences and the rod worth minimizer maintain rod worth at acceptably low values to minimize the consequence of a rod reactivity accident. At any specified reactor state, peak enthalpies for rod removal accidents vary proportionately with rod worths. Peak enthalpy provides the best index for determining the consequences of a reactivity accident when correlated to experimental measurements. Analyses and a survey of pertinent experimental data (Reference 12) indicate that prompt dispersal of finely fragmented fuel into the coolant with subsequent large pressure rise rates does not occur at excursion energy densities below 425 cal/g. Excursion energies above this level can cause pressure surges which may endanger the nuclear system process barrier.

In order to provide margin below the 425 cal/g, a limit on peak fuel enthalpy of 280 cal/g is selected. At this point the uranium dioxide vapor pressure is insignificant. This fuel enthalpy limit is supported by a careful study of all available SPERT, TREAT, KIWI, and PULSTAR tests (Reference 13). These tests indicate fairly rapid pressure rise rates above a fuel enthalpy of 400 cal/g. These pressure rise rates increase with increasing fuel enthalpy. At 280 cal/g, the pressure rise rates become very low, less than 50 psi/sec. Pressure rise rates of this order of magnitude pose no threat to the nuclear system process barrier. The specified control rod withdrawal sequences to be used are designed to limit rod worth, so that the drop of any control rod from the core to the position of its drive results in a peak fuel enthalpy of not more than 280 cal/g. A velocity limiter, which is an integral part of the control rod, limits the maximum rod velocity to 5 ft/sec. The velocity limiter is described in subsection 3.4, "Reactivity Control Mechanical Design."

Control rod drop excursion analysis is described in Reference 8, Section S.2.2.3.1. This analysis, whose results are presented in item 15 of the Reload Licensing Submittals, shows that peak fuel enthalpies of 280 cal/g are not reached.

Rod patterns permitted by operating procedures and supplemented by the rod worth minimizer restrict most rod worths to less than 0.01 Δk, although larger values are acceptable within the 280 cal/g limit. Above 10 percent power, it is impossible to obtain a rod with worth high enough to produce peak enthalpy of 280 cal/g if the rod were removed at 5 ft/sec. Planned patterns, therefore, are not needed to limit the consequences of the rod drop accident when the reactor is above 10 percent power.

3.6.8 Verification and Testing

The shutdown reactivity requirement is verified any time core loading changes are made. Nuclear limitations for components other than the fuel are verified by testing the individual system. The test capabilities are described in other subsections.

Correct fuel bundle loading in the reactor core is readily verified by visual observation and assured by verification procedures during core loading. Testing is performed to ensure technical specification requirements are verified.

3.6 NUCLEAR DESIGN

REFERENCES

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14. NEDC-33566P, “Safety Analysis Report for Exelon Peach Bottom Atomic Power Station Units 2 and 3, Constant Pressure Power Uprate,” September 2012.

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TABLES 3.6.1 THROUGH 3.6.6

THESE TABLES HAVE BEEN DELETED

3.7 THERMAL AND HYDRAULIC DESIGN

3.7.1 Power Generation Objective

The objective of the thermal and hydraulic design of the core is to achieve power operation of the fuel over the life of the core without sustaining fuel damage.

3.7.2 Power Generation Design Basis

1. The thermal-hydraulic characteristics of the core provide the ability to achieve rated core power output throughout the design lifetime of the fuel without sustaining fuel damage.

2. The thermal-hydraulic characteristics of the core provide the flexibility to adjust core power output over the range of plant load and load maneuvering requirements without sustaining fuel damage.

3.7.3 Safety Design Basis

Thermal-hydraulic design of the core establishes:

1. Actuation limits for the devices of the nuclear safety systems so that no fuel damage occurs as a result of moderate frequency transient events. Specifically, the minimum critical power ratio (MCPR) operating limit is specified so that at least 99.9 percent of the fuel rods in the core are not expected to experience boiling transition during the most severe moderate frequency transient events.

2. The thermal-hydraulic safety limits for use in evaluating the safety margin relating the consequences of fuel barrier failure to public safety.

3. That the nuclear system exhibits no inherent tendency toward divergent or limit cycle oscillations which would compromise the integrity of the fuel or nuclear system process barrier.

3.7.4 Thermal and Hydraulic Design Limits

3.7.4.1 Steady-State Limits

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

3.7.4.2 Transient Limits

The transient thermal limits are established so that no fuel damage is expected to occur during the most severe moderate frequency transient event. Fuel damage is defined as perforation of the cladding that permits release of fission products. Mechanisms that could cause fuel damage in reactor transients are:

1. Severe overheating of the fuel cladding caused by inadequate cooling.

2. Fracture of the fuel cladding caused by relative expansion of the uranium dioxide pellet inside the fuel cladding.

For design purposes, the transient limit requirement is met if at least 99.9 percent of the fuel rods in the core do not experience boiling transition during any moderate frequency transient event. No fuel damage would be expected to occur even if a fuel rod actually experiences a boiling transition during any moderate frequency transient event.

A value of 1 percent strain of Zircaloy cladding is conservatively defined in Reference 2 as the limit below which fuel damage from overstraining the fuel cladding is not expected to occur. Available data indicate that the threshold for damage is in excess of this value. See UFSAR Section 3.2 for information pertaining to the mechanical aspects of fuel materials.

3.7.4.3 Summary of Design Limits

In summary, the steady-state operating limits have been established to ensure that the design basis is satisfied for the most severe moderate frequency AOO. There is no steady-state design overpower basis. An overpower that occurs during an incident of a moderate frequency transient event must meet the plant transient MCPR limit. Demonstration that the transient limits are not exceeded is sufficient to conclude that the design basis is satisfied.

The MCPR and LHGR limits are sufficiently general so that no other limits need to be stated. For example, cladding surface temperatures are always maintained within 10 to 15°F of the coolant temperature as long as the boiling process is in the nucleate regime. The cladding and fuel bundle integrity criterion is ensured as long as the MCPR and LHGR limits are met. There are no additional design criteria on coolant void fraction, core coolant flow velocities, or flow distribution, nor are they needed. The flow distribution is controlled by the MCPR requirement. The coolant flow velocities and void fraction become constraints upon the mechanical and physics design of reactor components and are partially constrained by stability and control requirements.

Further discussion of reactor reload thermal-hydraulic design limits is found in Section 4.0 of reference 2. This section includes bases and methods of calculation. The cycle-specific results of these limit determinations are presented in the Supplemental Reload Licensing Submittal Report for the cycle in question of Peach Bottom Units 2 and 3, respectively.

3.7.5 Thermal and Hydraulic Characteristics

3.7.5.1 Application of Thermal-Hydraulic Limits to Core Design

The design basis employed for the thermal and hydraulic characteristics incorporated in the core design, in conjunction with the plant equipment characteristics, nuclear instrumentation, and the reactor protection system, is to require that no fuel damage occurs during normal operation or during abnormal operational transients. Demonstration that the applicable thermal-hydraulic limits are not exceeded is given by analyses.

3.7.5.2 Description of Thermal-Hydraulic Design of the Reactor Core

The information for selected paragraphs of this subsection is not presented herein, but is provided in Section 4.2 of reference 2 (see specific references as follows).

3.7.5.2.1 Critical Power Ratio

See Section 4.2.1 of reference 2.

3.7.5.2.2 Linear Heat Generation Rate

See Section 2.2 of reference 2.

3.7.5.2.3 Core Coolant Flow Distribution and Orificing Pattern

See Section 4.2.3 of reference 2.

3.7.5.2.4 Core Pressure Drop and Hydraulic Loads

See Section 4.2.4 of reference 2. Analysis for the most limiting conditions, the recirculation line break and the steam line break, are reported in Section 14, "Plant Safety Analysis."

3.7.5.2.5 Correlation and Physical Data

See Section 4.2.5 of reference 2.

3.7.5.2.6 Thermal Effects of Operational Transients

The evaluation of the core's capability to withstand the thermal effects resulting from an anticipated operational ocurrence is covered in Section 14, "Plant Safety Analysis."

3.7.5.2.7 Uncertainties in Estimates

See Section 4.2.7 of reference 2.

3.7.5.2.8 Flux Tilt Considerations

The inherent design characteristics of the BWR are particularly well suited to handle perturbations due to flux tilt. The stabilizing nature of the moderator void coefficient effectively damps oscillations in the power distribution. In addition to this damping, the in-core instrumentation system and the associated on-line computer provide the operator with prompt and reliable power distribution information. Thus, the operator can readily use control rods or other means to effectively limit the undesirable effects of flux tilting. Because of these features and capabilities, it is not necessary to allocate a specific peaking factor margin to account for flux tilt. If for some reason the power distribution could not be maintained within normal limits using control rods, then core thermal power would have to be reduced as prescribed in Technical Specifications.

3.7.5.3 Analytical Methods

The analytical methods employed in the thermal-hydraulic evaluations of each reload core design are detailed in Sections 4.3 and the Country Specific Supplement of reference 2. Applicable subsections are included in this document only by reference.

3.7.5.3.1 <u>Reactor Limits Determination</u>

See the Supplement for the United States section of reference 2.

3.7.5.3.1.1 <u>Fuel Cladding Integrity Safety Limit</u>

See Section 4.3.1.1 of reference 2.

3.7.5.3.1.2 <u>MCPR Operating Limit Calculational Procedure</u>

See Section 4.3.1.2 of reference 2.

3.7.5.3.1.3 <u>Vessel Pressure ASME Code Compliance Model</u>

See Section S.1.2 of reference 2.

3.7.5.3.1.4 <u>Stability Analysis Method</u>

See Section S.1.3 of reference 2.

3.7.5.3.2 <u>Steady-State Hydraulic Models</u>

See Section 4 of reference 2.

3.7.5.4 <u>Performance Range for Normal Operations</u>

A BWR must operate within certain restrictions due to pump net positive suction head (NPSH), overall plant control characteristics, core thermal power limits, etc. The power-flow map for the power range of operation is shown in Figure 3.7.1. The nuclear system equipment, nuclear instrumentation, and the RPS, in conjunction with operating procedures, maintain operations within the allowable operating domain of this map. The boundaries on this map are as follows.

<u>Approximate Natural Circulation Line</u>

Reactor power moves along this line in the absence of recirculation pump operation, however, this is not a normal operating state for plant operations. The natural circulation line show on Figure 3.7.1 is approximate.

<u>Approximate 30% Pump Speed Lower Limit Line</u>

Startup operations of the plant are normally carried out with the recirculation pumps operating at approximately 28 to 30 percent speed. The operating state for the reactor follows this line for the normal control rod withdrawal sequence.

100% Rod Line

The 100% rod line passes through 100 percent power at 100 percent flow and forms the lower boundary of the Maximum Extended Load Line Limit (MELLL) region.

Increased Core Flow (ICF) Region

Core flow can be increased to 110% of the original rated flow rate. The increased core flow region (Reference 6) provides an expanded operating envelope to compensate for the reduction in reactivity as exposure increases during the fuel cycle. This is used to support longer fuel cycles by maintaining full power after control rods are fully withdrawn. A smoothed average value of core flow may be used to demonstrate compliance with the 110% maximum core flow limit.

Maximum Extended Load Line Limit (MELLL)

The MELLL load line represents the load line for which the plant is analyzed and licensed to operate (Reference 5). The MELLL load line allows operation at high power at reduced recirculation flow rates, which is utilized at beginning of cycle. This extension of the operating domain provides added flexibility during plant startups and maneuvers, reduced recirculation pump power usage, and better fuel cycle economies resulting from the hardened neutron energy spectrum.

The minimum core flow along the MELLL load line is approximately 99% of rated core flow at 100% core thermal power.

Maximum Extended Load Line Limit Analysis Plus (MELLLA+)

MELLLA+ was implemented following the increase in rated thermal power to 3951 MWt in order to expand the core flow range at 100% rated thermal power. All lines on the power/flow map in Figure 3.7.1 other than those associated with the MELLLA+ operating domain, are unchanged by MELLLA+. The MELLLA+ domain extends from 55% rated core flow at 78.8% of rated thermal power to 83% rated core flow at 100% rated thermal power. Due to stability considerations at high power and low core flow, the MELLLA+ domain was not extended below 55% rated core flow (Reference 17).

APRM Rod Block Line

The line shown on the graph provides a barrier to inadvertently reaching the APRM scram set point by means of control rod withdrawal.

Feedwater Temperature Reduction

Operation with reduced feedwater temperature is permitted up to 55°F for applications to accommodate equipment out of service conditions (FWHOOS) including operation along the MELLL load line (Reference 9).

A feedwater temperature reduction (FWTR) up to 90°F is permitted for fuel cycle extension. Feedwater temperature reduction is limited to 10°F when operating in the MELLLA+ domain.

Asymmetric Feedwater Temperature Operation (AFTO)

Under normal conditions the feedwater temperatures in the two main feedwater lines entering the primary containment from the reactor feedwater pump area closely match each other. The specific configuration of the PBAPS feedwater lines at the discharge of the reactor feedwater pumps make it possible to have asymmetric (unequal) feedwater temperatures, when specific heater strings or individual heaters are not in service. The impact of operating with asymmetric feedwater heating has been evaluated (References 11, 15, and 18).

Asymmetric feedwater temperatures translate into uneven core inlet temperatures and result in greater uncertainty in the calculation of local power distribution and core thermal performance monitoring.

Thermal limit penalties must be applied to account for this increased core monitoring uncertainty, when operating with a feedwater temperature asymmetry (going into the RPV) greater than a predetermined threshold value. This threshold value varies with power and core flow. The relationship of the threshold AFTO value is incorporated in the station procedures that govern AFTO. The thermal limit penalties are applicable for asymmetric temperature differences up to 55°F.

Equipment Out-of-Service (EOOS)

Equipment out-of-service (EOOS) features such as the Turbine Bypass Valve out-of-service (TBVOOS) contingency mode of operation, the End-of-Cycle Recirculation Pump Trip out-of-service (EOC-RPTOOS) contingency mode of operation and Single Loop Operation (SLO) are also evaluated within the performance range for normal operation. Transient analyses described in Section 14.0, "Plant Safety Analysis," demonstrate that adequate fuel thermal limits can be established for these modes of operation such that damage to the fuel barrier or pressure in excess of the nuclear system pressure limits is avoided, as required by the safety design basis.

3.7.5.5 Flow Control

The following simplified description of BWR operation summarizes the principle modes of normal power range operation. Prior to startup the recirculation pumps are started one at a time and typically held at a pump speed of 30 percent or less of rated speed. The first part of the startup sequence is achieved by withdrawing control rods with the recirculation pumps at a pump speed of 30 percent or less of rated speed. Core power, steam flow, and feedwater flow increase as control rods are withdrawn by the operator, until feedwater flow increases to a point above the feedwater flow interlock. The low feedwater flow interlock (approximately 20% feedwater flow) prevents low power-high recirculation flow combinations which may create recirculation system NPSH problems. The natural circulation characteristics of the BWR are still very influential in this part of the power flow region.

Once the feedwater interlock has been cleared the recirculation flow in each loop can be increased to increase power. The operator then can achieve full power by a combination of control rod withdrawals and pump speed increases, depending on operating and core management strategies. A typical strategy for plant startup is to increase core flow to a mid range value. Then control rods are withdrawn to a point just below the MELLL load line. Core flow can then be increased until the desired high power condition is reached. The normal power range operation is bounded by the MELLL region maximum rod line and 100% power.

The large negative operating coefficients, which are inherent in the BWR, provide the following important advantages:

1. Stable load change response following with well damped behavior and little undershoot or overshoot in the heat transfer response.
2. Load changes with recirculation flow control.
3. Strong damping of spatial power disturbances.

To increase reactor power, it is necessary only to increase the recirculation flow rate which reduces core average void content, causing an increase in core reactivity. As the reactor power increases, more steam is formed and the reactor stabilizes at a new power level with the transient excess reactivity balanced by the new void formation. No control rods are moved to accomplish this power level change. Conversely, when a power reduction is required, it is necessary only to reduce the recirculation flow rate. When this is done, more voids are formed in the moderator, and the reactor power output automatically decreases to a new power level commensurate with the new recirculation flow rate. No control rods are moved to accomplish the power reduction.

Varying the power level by varying the recirculation flow rate (flow control) is more advantageous than using control rod positioning. Flow variations perturb the reactor uniformly in the horizontal planes and thus, allow operation with flatter power distribution and reduced transient allowances. As the flow is varied, the power and void distributions remain approximately constant at the steady-state end points for a wide range of flow variations. These constant distributions provide the important advantage that the operator can adjust the power distribution at a reduced power and flow by movement of control rods and then bring the reactor to rated conditions by increasing flow, with the assurance that the power distribution will remain approximately constant. Subsection 7.9, "Recirculation Flow Control System," describes the means by which recirculation flow is varied.

3.7.5.6 Core Power Distribution

Thermal design of the reactor--including the selection of the core size and effective heat transfer area, the design steam quality, the total recirculation flow, the inlet subcooling, and the specification of internal flow distribution--is based on the concept and application of a design power distribution. The design power distribution is an appropriately conservative representation of the most limiting thermal operating state at rated conditions and includes design allowances for the combined effects (on the fuel rod, and the fuel assembly heat flux and temperature) of the gross and local steady-state power density distributions and adjustments of the control rods.

The design power distribution is used in conjunction with flow and pressure drop distribution computations to determine the thermal conditions of the fuel and the enthalpy conditions of the coolant throughout the core. The design power distribution is based on detailed calculations of the neutron flux distribution as discussed in reference 2.

Current cycle-specific results of these calculations are presented in the Supplemental Reload Licensing Submittal Report of the cycle in question for Peach Bottom Units 2 and 3, respectively.

3.7.6 Thermal and Hydraulic Evaluation

3.7.6.1 Design Minimum Critical Heat Flux Ratio Limit

The objective for normal operation and transient events is to maintain nucleate boiling and thus avoid a transition to film boiling. Previously, the operating limit utilized to maintain adequate margin to the onset of nucleate boiling was the minimum critical heat flux ratio (MCHFR). However, this has been replaced by the more limiting minimum critical power ratio (MCPR). Therefore, further discussion of MCHFR is omitted, and MCPR is detailed in paragraph 3.7.6.2.

3.7.6.2 Design Minimum Critical Power Ratio

The design basis employed for the thermal and hydraulic characteristics incorporated in the core design, in conjunction with the plant equipment characteristics, nuclear instrumentation, and the reactor protection system, is to require that no fuel damage occur during normal operation or during abnormal operational transients. Demonstration that the applicable thermal-hydraulic limits are not exceeded is given by analyses. The methods employed in these analyses are described in reference 2 and are included in this document only by reference.

3.7.6.2.1 Critical Power

See Section 4.3.1 of reference 2.

3.7.6.2.2 Core Hydraulics

See Section 4.3.2 of reference 2.

3.7.6.2.3 Influence of Power Distribution

See Section 4.3.3 of reference 2.

3.7.6.2.4 Core Thermal Response

The thermal response of the core to accidents and expected transient conditions is discussed in Section 14, "Plant Safety Analysis."

3.7.6.2.5 Analytical Methods

See Section 4.3.5 of reference 2.

3.7.6.2.6 Thermal-Hydraulic Stability Analysis

See Section S.4 of reference 2.

3.7.6.3 Fuel Damage Analysis

Fuel damage is perforation of the fuel cladding. Defects in the fuel cladding should be minimized for two reasons:

1. Defects permit the release of fission products to the reactor coolant. This release involves a portion of those fission products that have diffused out of uranium dioxide matrix.

2. Water which enters the fuel rods through defects can cause progressive clad corrosion and further deterioration of the cladding in the fuel rod leading eventually to water and steam leaching of fission products and uranium from the fuel pellets. If this progressive failure persists, the reactor coolant activity level increases, and it becomes necessary to replace or repair the fuel assembly.

Predictions of the amount of fuel damage associated with a specific operation involves complex functions interrelating design methods and material properties, manufacturing methods and assembly tolerances, material specifications and quality control, operating variables and the effectiveness of reactor protection equipment, the initial starting conditions, and the amount of change in these conditions that constitute fuel damage. The interrelationships between these variables are continually evaluated by design, production, operating, and safety engineers. The only practical method of interrelating these variables is through use of probability functions and uncertainty analyses. The uncertainty analyses must yield an acceptably low fuel failure rate to achieve safe use of nuclear power. Quality control procedures, including 100 percent ultrasonic inspection of Zircaloy tubing and 100 percent helium leak check of fuel rods, assure an extremely low probability that fuel rods have leaks prior to operation.

Reactor assembly procedures are carefully planned to ensure proper core assembly and proper inspection of the assembled core. Analysis has been made to determine the effect of nonconformance with these procedures, for example, fuel misorientation. Fueling procedures and mechanical features have been developed to avoid errors in fuel element orientation in the core. The handle employed in lifting the fuel assembly has an integral lug or pointer to indicate the orientation of the corner containing the lowest enrichments. Procedures require that this corner be adjacent to the control rod, and the proper orientation is verified visually after all loading operations are complete. If an assembly is inserted in the core incorrectly, the operator will observe a mismatch in the channel fasteners at the top of the channel which mate with those on the adjacent channels when properly oriented. This mismatch is a second indication that the orientation is in error.

The orientation error is more fully described and analyzed in reference 2, paragraphs S.2.2.1.9 and S.2.2.2.1.

Rigid quality control assembly and inspection procedures reduce the probability of fuel failures resulting from such causes to a negligible value.

3.7.7 Changes to the Original Thermal-Hydraulic BWR/4 Design

3.7.7.1 Modifications to Eliminate Significant In-Core Vibration

During an outage at a GE-designed BWR in early 1975, it was determined that some fuel assembly channels exhibited corner wear adjacent to in-core neutron monitor and startup source locations (in-cores). Subsequent inspections at plants of similar design revealed similar corner wear on the fuel assembly channels. The most severe wear corresponded to the location of the LPRM. Less severe channel wear was found at areas which correspond to the SRM, IRM, and startup source locations. (Note: Peach Bottom uses a WRNM system in place of the original SRM/IRM system. However, because the dry tube for WRNM replacement interface is the same as the original SRM/IRM dry tube, the basis for the modifications described here is not affected.)

It was postulated, and subsequently confirmed by out-of-reactor testing, that the wear was caused by vibration of the in-core tubes due primarily to a high-velocity jet of water flowing through the bypass flow holes in the lower core plate. This caused the tubes to wear against the channel corner. The wear has been observed to penetrate the channel wall at two BWR's.

Peach Bottom is one of the product line which incorporates bypass flow holes in the lower core support plate. This product line is commonly referred to as a BWR/4. References 3 and 4 describe the actions that were taken at Peach Bottom to make permanent plant modifications to eliminate the recurrence of significant channel wear, and present the complete safety analysis that was performed. These modifications include plugging of the bypass flow holes in the core support plate and providing an alternate flow path through the fuel assembly lower tie plates. Since this modification maintains the same flow distribution in the core region as in the original design, the normal power, stability, and abnormal transient thermal-hydraulic margins remain effectively the same. For further discussion of this modification see references 3 and 4.

Subsequent to the initial plug design, effective plug life was reevaluated. References 7 and 8 provide further details on these subsequent evaluations. A reevaluation of effective plug life was performed based on testing and analysis of four sample plugs extracted from Unit 3 during 3R13. Reference 14 provides further details on the evaluation. The evaluation is applicable to the original plug design only.

Modification P-00916 replaces the original core support plate plugs used to plug the bypass flow holes with an extended life plug design. The modification is performed for PBAPS Unit 3. References 12 and 13 provide further details on the modification.

3.7.8 Verification and Testing

The detailed core power and MCPR distribution is calculated periodically. The plant is operated as necessary to maintain MCPR and the LHGR within the design values.

3.7 THERMAL AND HYDRAULIC DESIGN

REFERENCES

1. General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application, NEDE-10958-PA and 10958-A January 1977.

2. Licensing Topical Report, "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A February 1991 and all subsequent revisions.

3. Peach Bottom Atomic Power Station Units 2 and 3, Safety Analysis Report for Plant Modifications to Eliminate Significant In-Core Vibrations, NEDC-20994, September 1975.

4. "Supplemental Information for Plant Modification to Eliminate Significant In-Core Vibration," NEDE-21156, January 1976.

5. "General Electric Boiling Water Reactor Maximum Load Line Limit and ARTS Improvement Program Analyses for Peach Bottom Units 2 and 3, " NEDC-32162P, Rev 2, March 1995.

6. Safety Review for Peach Bottom Atomic Power Station Units 2 & 3, 110% Increased Core Flow Operation and Final Feedwater Temperature. NEDC-32174P, May 1994.

7. "Revision of Core Support Plate Plug Service Life," NEDC-32120P, July 1992.

8. "Core Support Plate Plug Service Life Evaluation for Peach Bottom Units 2 and 3," GENE B11-00647-01, March 1995 and GENE B11-00647-0IRI, May 1996

9. "Safety Review for Peach Bottom Atomic Power Station Units 2 & 3, Feedwater Heater Out-of-Service", General Electric Co. Report NEDC-32401P, October 1994.

10. "Peach Bottom Atomic Power Station Evaluation of Extended Final Feedwater Temperature Reduction", General Electric Co. Report NEDC-32707P, Supplement 1", May 1998.

11. "Safety Review for Peach Bottom Atomic Power Station, Units 2 & 3, Asymmetric Feedwater Temperature Operation," General Electric Co. Report NEDC-32691P, May 1997.

12. "Safety Evaluation for Extended Life Core Support Plugs for Exelon Nuclear Peach Bottom Atomic Power Station Units 2 & 3," GE-NE-B13-02100-00-01, DRF No. B13-02100-00, Class 3, May 2001.

13. "Core Support Plug Design Specification, Peach Bottom 2, 3," GE Nuclear Energy Document No. 26A5703, Revision 0.

14. "Evaluation of Core Support Plate Plugs, Peach Bottom Units 2 and 3," GE Nuclear Energy Report No. 0000-0004-8846, August 2002.

15. "Peach Bottom Asymmetric Feedwater Temperature Operation Fuel Thermal Limits Evaluation Update," GE-NE-0000-0057-0522-R0, January 2007.

16. NEDC-33566P, "Safety Analysis Report for Exelon Peach Bottom Atomic Power Station Units 2 and 3, Constant Pressure Power Uprate," September 2012.

17. NEDC-33720P, "Safety Analysis Report for Peach Bottom Atomic Power Station Units 2 & 3 Maximum Extended Load Line Limit Analysis Plus," (M+SAR), September, 2014.

18. "Asymmetric Feedwater Temperature Operation for EPU/MELLLA+," GEH Report 001N6733-R1, September 2014.

3.8 STANDBY LIQUID CONTROL SYSTEM

3.8.1 Safety Objective

The safety objective of the standby liquid control system is to provide a backup method, which is redundant with, but independent of, the control rods, to establish and maintain the reactor subcritical as the nuclear system cools. Maintaining subcriticality as the nuclear system cools assures that the fuel barrier is not threatened by overheating in the event that not enough of the control rods can be inserted to counteract the positive reactivity effects of a decrease in moderator temperature.

Tto mitigate an Anticipated Transient Without SCRAM (ATWS) event, the system provides the means to rapidly shutdown the reactor in order to maintain suppression pool temperature to ≤180°F.

In addition, the LOCA analysis using the Alternative Source Term (AST) methodology, as presented in Section 14.9.2, credits the SLCS for injecting sodium penetaborate to maintain the suppression pool pH greater than 7 throughout the accident duration. Maintaining suppression pool pH levels at or above 7 following an accident ensures that sufficient iodine will be retained in the suppression pool water, ensuring offsite doses remain within 10CFR50.67 limits.

3.8.2 Safety Design Basis

1. Backup capability for reactivity control is provided, independent of normal reactivity control provisions in the nuclear reactor, to shut down the reactor if the normal control ever becomes inoperative.

2. The backup system has the capacity for controlling the reactivity difference between the steady-state, rated operating condition of the reactor with voids and the cold shutdown condition, including shutdown margin, to assure complete shutdown from the most reactive condition, at any time in the core life.

3. The time required for actuation and effectiveness of the backup control is consistent with the nuclear reactivity rate of change predicted between rated operating and cold shutdown conditions. A fast scram of the reactor or operational control of fast reactivity transients is not specified to be accomplished by this system. The system meets the performance requirements of 10CFR50.62.

4. Means are provided by which the functional performance capability of the backup control system components can be verified under conditions approaching actual use requirements. A substitute solution, rather than the actual neutron absorber solution, may be injected into the reactor to test the operation of all components of the redundant control system.

5. The neutron absorber is dispersed within the reactor core in sufficient quantity to provide a reasonable margin for leakage or imperfect mixing.

6. The system is reliable to a degree consistent with an engineered safeguard system; the possibility of unintentional or accidental shutdown of the reactor by this system is minimized.

3.8.3 Description

The standby liquid control system is manually initiated from the control room to pump a boron neutron absorber solution into the reactor if the operator believes the reactor cannot be shut down or kept shut down with the control rods. However, insertion of control rods is expected always to assure prompt shutdown of the reactor should it be required. The SLC system injects borated water into the reactor vessel to add negative reactivity to compensate for the various reactivity effects of plant operation thus shutting down the reactor. To meet this objective, it is necessary to inject a quantity of boron, which produces a concentration equivalent to 660 ppm of natural boron in the reactor coolant at 68°F. The primary components of the standby liquid control system are the solution tank, the test tank, two 100% capacity positive displacement pumps with their associated relief valves and accumulators, two explosive valves and associated local controls and instrumentation. These components are all located in the reactor building outside the primary containment (Drawings M-358 and M-1-DD-3).

The standby liquid control system has a minimum storage tank volume, solution concentration, and enrichment sufficient to maintain suppression pool pH greater than 7.0 throughout a LOCA event.

In accordance with 10CFR50.62, which was promulgated to reduce the risk associated with anticipated transient without scram (ATWS) events, the standby liquid control system has a minimum flow capacity and boron content equivalent in control capacity to 86 gallons per minute of 13 weight percent natural sodium pentaborate solution. This requirement has been satisfied by using a lower concentration sodium pentaborate solution which has been enriched in boron-10, the isotope having a high neutron absorption cross-section.

For operation at 3,951 MWt, the SLC system uses enriched boron (i.e., $\geq$ 92 atom-% boron-10) to achieve a faster rate of negative reactivity insertion. The SLC system is capable of injecting a borated water solution into the reactor vessel at a boron concentration and enrichment, and flow rate exceeds the ATWS rule requirement (10 CFR 50.62). The design ensures rapid shutdown of the reactor in order to maintain suppression pool temperature to $\leq$ 180°F during an ATWS event. To meet this objective, SLC system parameters are controlled by specific surveillance requirements in the Technical Specifications as follows:

1. The SLC pump must achieve a flow rate of $\geq$ 49.1 gpm at a discharge pressure of $\geq$ 1275 psig,
2. The SLC storage tank minimum solution volume must be maintained $\geq$ 52%, and
3. The SLC storage tank solution must have a SPB concentration between $\geq$ 8.32 and $\leq$ 9.82 weight percent that is enriched with boron-10 to $\geq$ 92 atom-%.

The neutron absorber solution is pumped into the reactor vessel and discharged near the bottom of the core shroud where it mixes with the cooling water rising through the core (subsection 4.2, "Reactor Vessel and Appurtenances Mechanical Design," and subsection 3.3, "Reactor Vessel Internals Mechanical Design").

The boron in the solution absorbs thermal neutrons and thereby terminates the nuclear fission chain reaction in the uranium fuel.

The specified neutron absorber solution is sodium pentaborate ($Na_2B_{10}O_{16} \cdot 10H_2O$). It is normally prepared by dissolving dry enriched sodium pentaborate in demineralized water. An air sparger is provided in the tank for mixing.

The minimum tank solution volume is calculated based on the concentration of boron-10 required in the reactor coolant to achieve and maintain cold shutdown (subsection 3.8.4, "Safety Evaluation") and the minimum solution concentration and Boron-10 enrichment levels. A maximum solution concentration limit of 9.82% has been established to ensure that the saturation temperature does not exceed 43°F. Since the ambient temperature in the vicinity of the system is expected to always be greater than 53°F, precipitation of sodium pentaborate is not a concern.

Although it is no longer required to prevent solution precipitation, heat tracing has been maintained on the pump suction piping, with a controller setpoint near normal ambient temperature. Similarly, the solution tank heater controls are set near normal ambient temperature.

Solution tank level indication is provided in the control room. Instrumentation also exists to initiate control room alarms for high-low solution tank level, high-low solution tank temperature, tank heater short to solution, high-low pump suction line solution temperature and high-low pump casing temperature. The pump heater is disconnected so that the pump temperature element will sense ambient temperature and initiate an alarm in the event of unusually low room temperature.

The two 100% capacity positive displacement reciprocating pumps each have a capacity of about 50 gallons per minute while pumping into the reactor vessel at the reactor vessel maximum operating pressure. The pump design pressure is 1,500 psig.

The two relief valves are set at approximately 1,450 psig to exceed the reactor operating pressure by a sufficient margin to avoid leakage through the relief valves, and to lift below the pump design pressure of 1500 psig. The relief valves are installed with the discharge flooded to prevent evaporation and precipitation within the valve. To prevent bypass flow from one pump in case of relief valve failure in the line from the other pump, a check valve is installed downstream of each relief valve line in the pump discharge pipe.

A bladder type pneumatic-hydraulic accumulator is installed on the piping near each relief valve to dampen pulsations from the pumps to protect the system.

The two explosive-actuated injection valves provide high assurance of opening when needed and that boron will not leak into the reactor even when the pumps are being tested. The valves have a firing reliability in excess of 99 percent. Each explosive valve is closed by a plug in the inlet chamber. The plug is circumscribed with a deep groove so the end will readily shear off when pushed by the valve plunger. This opens the inlet hole through the plug. The sheared end is pushed out of the way in the chamber, and is shaped so it will not block the ports after release.

The shearing plunger is actuated by an explosive charge having dual ignition primers. The charge is inserted in the side chamber of the valve. Ignition circuit continuity is monitored by a trickle current, and an alarm occurs in the control room if either explosive valve's ignition circuit opens. Indicator lights show which ignition circuit opened. To service a valve after firing, a 6-in spool piece is removed immediately upstream of the valve to gain access to the shear plug.

The standby liquid control system is actuated by a three-position keylock switch on the control room console. This assures that switching from the "off" position is a deliberate act. Switching to either side starts one injection pump, opens both explosive valves, and closes the reactor cleanup system isolation valves to prevent loss or dilution of the boron.

A green light in the control room indicates that the pump motor contactor is open (pump not running). A red light indicates that the contactor is closed (pump running).

If the pump lights, pump discharge pressure, SLCS tank level or reactor power indicates that the liquid may not be flowing, the operator can immediately turn the keylock switch to the other side, which actuates the other injection pump and sends a redundant signal to open both explosive valves and closes the reactor water cleanup system isolation valves. Cross-piping and check valves assure a flow path through either pump and either explosive valve. The chosen pump will start even though its local switch is in the "stop" position. Pump discharge pressure indication is also provided in the control room.

Equipment drains and tank overflow are piped to separate containers (such as 55-gal drums) that can be removed and disposed of independently to prevent any trace of boron from inadvertently reaching the reactor.

Instrumentation is provided locally at the standby liquid control tank consisting of solution temperature indication and control, tank level, and tank heater status. Instrumentation and control logic is presented in Drawing M-1-CC-5.

3.8.4 Safety Evaluation

The standby liquid control system is a special safety system and an engineered safeguard system. Per its original design function as an alternate means of reactor shut down, the system was not required for plant operation and not needed to respond to or mitigate a design basis event because of the large number of independent control rods available to shut down the reactor. However, the adoption of Alternate Source Term (AST) methodology requires operation of the standby liquid vontrol system in order to provide post-LOCA suppression pool pH control.

The system is designed to bring the reactor from rated power to a cold shutdown at any time in core life. The reactivity compensation provided will reduce reactor power from rated to zero and allow cooling the nuclear system to normal room temperature, with the control rods remaining withdrawn in the rated power pattern. It includes the reactivity gains due to complete decay of the rated power xenon inventory. It also includes the positive reactivity effects from eliminating steam voids, changing water density from hot to cold, reduced Doppler effect in uranium, reduction of neutron leakage from boiling to cold, and decreasing control rod worth as the moderator cools. The specified minimum final concentration of boron in the reactor core assures a substantial shutdown margin.

Cooldown of the nuclear system will take a minimum of several hours, to remove the thermal energy stored in the reactor, cooling water, and associated equipment and to remove most of the radioactive decay heat. The controlled limit for the reactor vessel cooldown is 100°F/hr. Normal operating temperature is about 550 F. Usually, shutting down the plant with the main condenser and various shutdown cooling systems will take 10 to 35 hours before the reactor vessel is opened, and much longer to reach room temperature (68°F), which is the condition of maximum reactivity and, therefore, the condition which requires the maximum boron concentration.

The specified minimum average concentration of boron-10 in the reactor to provide the specified shutdown margin, after operation of the standby liquid control system, is 121 ppm (Figure 3.8.4). The minimum quantity of sodium pentaborate to be injected into the reactor is calculated based on the required average concentration in the reactor coolant and the quantity of reactor coolant in the reactor vessel (water level at high level trip setpoint), recirculation loops and the single longest shutdown cooling loop at 68°F. The result is increased by 25 percent to allow for imperfect mixing, leakage, and volume in other small piping connected to the reactor. This minimum concentration will be achieved if the solution is prepared in the concentration defined in section 3.8.3 and maintained above saturation temperature. The minimum boron-10 injection rate is specified by the ATWS rule (10CFR50.62) to be equivalent in control capacity to 86 gallons per minute of 13 weight percent natural sodium penetaborate solution. This rule is satisfied by maintaining system parameters in accordance with the following equivalency equation:

$$\frac{Q}{86\text{ gpm}} \times \frac{C}{13\text{ wt.\%}} \times \frac{E}{19.8\text{ atom \%}} \geq 1$$

where,

Q = Single Pump flow rate (gallons per minute)

C = Sodium pentaborate concentration (% by weight)

E = Boron-10 enrichment (atom %)

At 3,951 MWt, the SLC system parameters previously controlled and verified by using the equivalency equation, including SPB concentration, SLC system pump flow rate, and the SPB boron-10 enrichment, are all controlled by a specific surveillance requirement in the Technical Specifications (i.e., TS 3.1.7). These surveillance requirements are more specific than the equivalency equation, which continues to equate to ≥ 1.0.

At 3,951 MWt, actuation of the SLC system injects a SPB solution into the RPV at ≥ 49.1 gpm and a discharge pressure of ≥ 1275 psig with a SPB concentration between ≥ 8.32 and ≤ 9.82 weight percent and boron enrichment with ≥ 92 atom-% boron-10. The SLC system design uses highly enriched boron that achieves a rapid rate of negative reactivity insertion and a rapid shutdown of the reactor core. A rapid shutdown of the reactor is required to limit the heat generated in the reactor that is ultimately transferred to the containment structure and suppression pool during an ATWS event in order to meet the following ATWS acceptance criteria:

1. Maintain the peak vessel bottom pressure in the reactor pressure vessel less than the ASME Service Level C limit of 1,500 psig.

2. Maintain containment pressure and temperature less than the design pressure (56 psig) and temperature (281°F) of the containment structure.

3. Limit the peak clad temperature and cladding oxidation to within the acceptance criteria of 10CFR50.46.

4. Limit the suppression pool temperature to ≤ 180°F.

The ATWS and containment analyses for operation at 3,951 MWt confirm that the SLC system design effectively mitigates an ATWS event because the suppression pool temperature does not exceed 180°F ensuring adequate NPSH is available for the RHR pumps aligned to take suction from the suppression pool with no credit for containment accident pressure (see Section 6.4.5).

Boron mixing studies performed by General Electric (NEDE-24222 & NEDC-30921) confirmed that the mixing of boron with the reactor coolant will be adequate to stop the fission chain reaction.

The standby liquid control system is designed to seismic Class I criteria. The system piping is designed as described in Appendix A.

Since the standby liquid control system is required to be operable in the event of a loss of offsite power, the pumps, valves, and controls are powered from the standby ac power supply in the absence of normal power. The pumps and valves are powered and controlled from separate Class 1E buses and circuits so that a single power failure will not prevent system operation. The essential instruments and lights are powered from the 120-V ac instrument power supply.

The standby liquid control system and pumps have sufficient pressure margin, up to the system relief valve setting of approximately 1,450 psig, to assure solution injection into the reactor above the normal pressure. The nuclear system relief and safety valves begin to relieve pressure at or above about 1135 psig; therefore, the standby liquid control system positive displacement pumps cannot overpressurize the nuclear system.

Only one of the two liquid control pumps is needed for proper system operation. If one pump is inoperable, there is no immediate threat to shutdown capability, and reactor operation may continue while repairs are being made. The period while one redundant component (upstream of the explosive valves) may be out of operation is consistent with the very small probability of failure of both the control rod shutdown capability and the alternate component in the standby liquid control system.

Consistent with the LOCA analysis using AST methodology, calculations demonstrate that at the Technical Specification minimum acceptance criteria for the SLC System storage tank volume, and SPB solution concentration and enrichment, injection of the SLC System solution post-LOCA will maintain suppression pool pH greater than 7.0 throughout the accident duration.

3.8.5 Inspection and Testing

Operational testing of the standby liquid control system is performed in at least two parts to avoid injecting boron into the reactor inadvertently. Functional testing of each pump is accomplished by recirculating solution to the solution tank or by pumping it to the test tank. Pump capacity is measured by the rate of level rise in the test tank. The test tank can also be filled with demineralized water and used as a pump suction source for recirculation testing or for flushing the system prior to vessel injection testing. The test tank can hold demineralized water for about 3 minutes of operation. Demineralized water from the makeup system is available for refilling or flushing the system. For recirculation testing and flushing operations the pumps are operated using the local control switches.

Functional testing of the injection portion of the system is accomplished by closing the locked open valve from the solution tank, opening the locked closed valve from the test tank (containing demineralized water), and actuating the keylock switch in the control room to either the A or B circuit. Normally one explosive valve is disabled for the test, and the other is actuated. This starts the pump and blows open the non-test disabled injection valve(s). The lights and alarms in the control room indicate that the system is operating.

After the functional tests, the injection valve(s) and explosive charge(s) are replaced and all valves are returned to their normal positions as shown in Drawing M-358.

By closing a local locked open valve to the reactor in the containment, leakage through the injection valves can be detected at a test connection in the line between the containment isolation check valves. (Position indicator lights in the control room indicate that the local valve is closed for test, or open and ready for operation.)

Should the solution ever be injected into the reactor, either intentionally or inadvertently, the boron may be removed from the system by flushing for gross dilution followed by operation of the reactor water cleanup system. There is practically no effect on reactivity when the boron-10 concentration has been reduced below about 10 ppm.

The concentration of sodium pentaborate in the solution is determined by chemical analysis.

The boron-10 enrichment of the sodium pentaborate solution is predetermined by the specified enrichment of the dry chemicals used to prepare the solution. The enrichment is verified by analysis during the procurement process, and any time boron is added to the SLCS tank. Periodically thereafter, samples of the solution are taken, and the boron-10 enrichment is calculated.

The gas pressure in the two accumulators is measured to detect leakage. A pressure gage and portable nitrogen supply are required to test and recharge the accumulators.