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3.0 REACTOR

3.1 SUMMARY DESCRIPTION

The subsections included in the "Reactor" section describe and evaluate those systems most pertinent to the fuel barrier and the control of core reactivity. The "Fuel Mechanical Design" subsection 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.

The "Reactor Vessel Internals Mechanical Design" subsection 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 additional damage to the fuel and the gross release of fission products from the fuel. The reactor vessel internals are designed to allow the control rods and Core Standby Cooling Systems to perform their safety functions during abnormal operational transients and accidents.

The "Reactivity Control Mechanical Design" subsection describes the mechanical aspects of the moveable control rods. They are provided to control core reactivity. The Control Rod Drive 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 operational transient.

Control Rod Housing Supports 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. They prevent a significant nuclear excursion as a result of the housing failure, thus protecting the fuel barrier and the primary system.

The "Nuclear Design" subsection describes the nuclear aspects of the reactor. The design of the boiling water reactor 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 performed for specific steady state, transient, and accident conditions. Included in the “Nuclear Design” subsection are discussions of operating and shutdown reactivity control requirements. Also included are discussions of the reactivity coefficients and xenon characteristics of the core. Transient and accident analyses are discussed in Chapter 14, “Plant Safety Analysis.” Results of steady state, transient, and accident analyses for current reload core designs are contained in Appendix N.

The "Thermal and Hydraulic Design" subsection describes the thermal and hydraulic characteristics of the core. The low coolant saturation temperature, high heat transfer coefficient, and neutral water chemistry of the boiling water reactor are significant advantages in minimizing Zircaloy temperatures and associated temperature-dependent hydride pickup. This results in improved cladding performance at long exposures. The relatively uniform fuel cladding temperatures throughout the boiling water reactor 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 in the subsection.

The "Standby Liquid Control System" provides a redundant, independent, and different way from the control rods to make the reactor subcritical, even in the cold condition. The Standby Liquid Control System is never expected to be needed because of the large number of control rods available to shut down the reactor. However, in the unlikely event that control rod insertion were to be impaired, the Standby Liquid Control System has the capability of bringing the reactor from rated power to cold shutdown (MODE 4) with the control rods remaining withdrawn in the rated power pattern.

3.2 FUEL MECHANICAL DESIGN

The fuel assembly is comprised of the fuel bundle, channel, and channel fastener. The fuel bundle is comprised of fuel rods, water rods (water channels), spacers, upper and lower tie plates, springs, and fittings.

Fuel licensing acceptance criteria for General Electric (GE) fuel designs are specified in GESTAR II (General Electric Standard Application For Reactor Fuel) (Reference 1). Amendment 22 of GESTAR II established an approved set of licensing acceptance criteria for which fuel design compliance constitutes USNRC acceptance and approval without specific USNRC review. Current GE fuel designs that have received specific USNRC review and approval, or that have been shown to meet the approved fuel licensing acceptance criteria are documented in Reference 2 (General Electric Fuel Bundle Designs). GE designs documented in Reference 2 and approved for use in Browns Ferry reload cores include the GE13 and GE14 fuel product lines. The initial core 7x7 and 8x8 designs, the reload core unpressurized 8x8R design, and the pressurized P8x8R, BP8x8R, GE9B, GE11 designs that are currently in spent fuel storage will not be reinserted in any future reload cores.

Fuel licensing acceptance criteria for AREVA fuel designs are specified in ANF-89-98(P)(A) Revision 1 and Supplement 1. “Generic Mechanical Design Criteria for BWR Designs” (Reference 4). This document contains an approved set of licensing acceptance criteria for which fuel design compliance constitutes USNRC acceptance and approval without specific USNRC review.

For GE fuel, the generic information contained in GESTAR II is supplemented by plant cycle-unique information and analytical results. This cycle-unique information includes a list of the fuel to be loaded in the core and safety analysis results. This information is documented in a separate plant-unique cycle-dependent submittal for each reload. The format for this Supplemental Reload Licensing Report (SRLR) and a description of the transient and accident methods used are given in the country-specific supplement to GESTAR II (Reference 3).

For AREVA fuel, a reload-specific fuel mechanical design report is prepared to document compliance with Reference 4 Generic Design Criteria. The Generic Design Criteria lists the approved methodology documents. The fuel mechanical design report is referenced in the applicable cycle-specific Safety Analysis Report.

3.2.1 Power Generation Objective

The 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 shall be 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 reactor protection system) that fuel damage limits will not be exceeded during either planned operation or abnormal operational transients caused by any single equipment malfunction or single operator error.

Limits are established to assure fuel operation remains within design bases. Three types of thermal limits are used. The first is the Minimum Critical Power Ratio Operating Limit. This limit is generated to protect against the phenomena of dryout, where the liquid film on the fuel cladding surface is boiled/stripped away, thereby creating conditions for a rapid rise in cladding temperature due to the elimination of effective boiling heat transfer. The second limit is the Linear heat Generation Rate. This limit protects the fuel rod thermal/mechanical integrity during normal operation, as well as during anticipated operational occurrences. The third limit is the Maximum Average Planar Linear Heat Generation Rate. This limit protects the fuel from exceeding fuel performance requirements identified by Title 10 Code of Federal Regulations Part 50.46.

Power and Flow dependent multipliers are applied to all three types of thermal limits to protect operation at off-rated conditions. The basic limit types, along with the power/flow multipliers are generated for all fuel types, regardless of fuel vendor or unit.

Power and flow dependent limits (or multipliers) are applied to both the MCPR and thermal-mechanical limits to account for off-rated conditions. The thermal-mechanical limits are protected with off-rated corrections applied directly to the LHGR limits.

3.2.3 Safety Design Bases

In meeting the power generation objectives, the nuclear fuel cladding shall be utilized as the initial barrier to the release of fission products. The fission product retention capability of the nuclear fuel shall be substantial during normal modes of reactor operation so that significant amounts of radioactivity are not released from the reactor fuel barrier.

For GE fuel, the detailed fuel thermal-mechanical design bases and limits are provided in GESTAR II. For AREVA fuel, the thermal-mechanical design bases and limits are provided in the AREVA Generic Design Criteria document (Reference 4). The design bases address each of the fuel system damage, failure, and coolability criteria identified in the Standard Review Plan (NUREG-0800).

3.2.4 Description

A “core cell” is defined as a control rod and the four fuel assemblies which immediately surround it. Each core cell is associated with a 4-lobed fuel support piece. Around the outer edge of the core, certain fuel assemblies are not immediately adjacent to a control rod and are supported by individual peripheral fuel support pieces.

A description of the fuel assembly and various fuel assembly components is provided in the following sub-sections.

3.2.4.1 Fuel Assembly

The fuel assembly consists of a fuel bundle and a channel which surrounds it. The fuel bundle contains fuel rods and water rods (or water channel) which are spaced and supported in a square array by upper and lower tieplates, as well as fuel rod spacers. The lower tieplate has a nosepiece which has the function of supporting the fuel assembly in the reactor. The upper tieplate has a handle for transferring the fuel bundle from one location to another. The identifying fuel assembly serial number is engraved on the top of the handle. No two assemblies bear the same serial number. A boss projects from one side of the handle to aid in ensuring proper orientation of the assembly in the core. Finger springs located between the lower tieplate and channel are utilized to control the bypass flow through that flow path.

3.2.4.1.1 Fuel Rods

Three types of fuel rods are used in a GE fuel bundle; tie rods, standard rods, and (in some designs) part length rods. The tie rods in each fuel bundle have lower end plugs which thread into the lower tieplate and threaded upper end plugs which extend through the upper tieplate. A nut and locking tab are installed on the upper end plug to hold the fuel bundle together. The tie rods support the weight of the assembly during fuel handling operations. All of the standard rods are full length rods. The part length rods are approximately 2/3 of the length of standard fuel rods.

AREVA fuel assemblies contain two basic rod types: standard rods and part length fuel rods (PLFRs). Tie rods are not necessary because the structural tie between the lower and upper tieplate is provided by the water channel. The PLFRs lengths vary by fuel design, relative to standard fuel rods.

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

For the AREVA design, the lower ends of the fuel rods rest on top of the lower tieplate grid. The lower ends of the fuel rods are laterally restrained by an additional spacer grid located just above the lower tieplate. No expansion springs are necessary on each fuel rod because a single, large reaction spring is used on the water channel to hold the upper tieplate in the latched position.

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

Adequate free volume is provided within each fuel rod in the form of a pellet-to-cladding gap and a plenum region. A plenum spring, or retainer, is provided in the plenum space to minimize the movement of the column of fuel pellets inside the fuel rod during shipping and handling. For GE fuel product lines through GE13, a hydrogen getter has historically been provided in the plenum space as assurance against chemical attack from inadvertent admission of moisture or hydrogenous impurities into the fuel rod during manufacture. With enhanced hydrogen controls in place, the optional feature of a reactive getter has been removed for the current GE13 and GE 14 product lines. Likewise, hydrogen and moisture controls eliminate the need for a getter in the AREVA fuel rod design.

3.2.4.1.2 Water Rods or Water Channel

For the GE fuel designs, water rods are hollow Zircaloy tubes with several holes around the circumference near each end to allow coolant to flow through. One water rod in each bundle axially positions the spacers. The AREVA design instead uses one larger internal water channel that has a square cross-section in contrast to round water rods. The water channel displaces 3x3 array of fuel rods in the fuel assembly interior.

3.2.4.1.3 Fuel Spacer

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

3.2.4.1.4 Finger Springs

Finger Springs are employed to control the bypass flow through the channel-to-lower tieplate flow path.

3.2.4.1.5 Debris Filter Lower Tie Plate

Fuel assemblies may include a debris filter as part of the lower tie plate products designed to reduce the probability of foreign material entering the fuel during normal operation. Both GE and AREVA debris filter designs utilize non-line-of-sight coolant flow paths to maximize the ability to stop and retain foreign material. The debris filter does not play a part in the structural performance of the fuel assembly. Designs are explicitly tested to determine hydraulic performance impacts.

3.2.4.1.6 Channels

The BWR Zircaloy fuel channel performs the following functions:

(1) Forms the fuel bundle flow path outer periphery for bundle coolant flow.
(2) Provides surfaces for control rod guidance in the reactor core.
(3) Provides structural stiffness to the fuel bundle during lateral loadings applied from fuel rods through the fuel spacers.
(4) Minimizes, in conjunction with the finger springs and bundle lower tieplate, coolant bypass flow at the channel/lower tieplate interface.
(5) Transmits fuel assembly seismic loadings to the top guide and fuel support of the core internal structures.
(6) Provides a heat sink during loss-of-coolant accident (LOCA).
(7) Provides a stagnation envelope for incore fuel sipping.

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

3.2.5 Safety Evaluation

The GE thermal-mechanical evaluations performed for the fuel are described in GESTAR II. AREVA thermal-mechanical evaluations are described in the Generic Design Criteria (Reference 4). Areas evaluated include:

(1) Fuel System Damage -- stress/strain, fatigue, fretting wear, oxidation, hydriding, corrosion, dimensional changes, internal gas pressure, and hydraulic loads.

(2) Fuel Rod Failure -- hydriding, cladding collapse, fretting wear, overheating of cladding, overheating of pellets, excessive fuel enthalpy, pellet-cladding interaction, bursting, and mechanical fracturing.
(3) Fuel Coolability -- cladding embrittlement, violent expulsion of fuel, generalized cladding melting, fuel rod ballooning, and structural deformation.

3.2.5.1 Evaluation Methods

The GE methods used in performing thermal-mechanical evaluations for the fuel are described in GESTAR II. These evaluations are performed primarily using the NRC-approved GESTR-MECHANICAL fuel rod thermal-mechanical performance model. The GESTR-MECHANICAL fuel rod performance model performs best estimate coupled thermal and mechanical analyses of a fuel rod experiencing a variable operating history. The model explicitly addresses the effects of:

- Fuel and cladding thermal expansion
- Fuel and cladding creep and plasticity
- Cladding irradiation growth
- Cladding irradiation hardening and thermal annealing of that irradiation hardening
- Fuel irradiation swelling
- Fuel irradiation-induced densification
- Fuel cracking and relocation
- Fuel hot pressing
- Fission gas generation and exposure-enhanced fission gas release including fission product helium release
- Differential axial expansion of the fuel and cladding reflecting axial slip or lockup of the fuel pellets with the cladding
- Fuel phase change volumetric expansion upon melting

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

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

AREVA methods used in performing the thermal-mechanical evaluations for the fuel are described in the Generic Design Criteria (Reference 4) and by the approved topical reports referenced thereof. These analyses are performed primarily using the NRC-approved RODEX2A fuel rod thermal-mechanical performance code. This code combines best estimate and conservative models, coupled with a specific input methodology to produce conservative results. The code contains a number of different models to address the following phenomena:

- Fuel densification
- Fuel gaseous and solid swelling
- A physically-based fission gas release model coupled to the swelling model
- Columnar grain growth
- Instantaneous plastic deformation of the fuel
- Fuel cracking, crack volume closure and creep deformation
- Fuel pore migration
- Cladding anisotropic creep deformation, irradiation growth
- Cladding corrosion and hydriding
- Thermal-hydraulic conditions of the fuel rod sub-channel

The code performs interactive calculations on a time incremental basis with conditions updated at each calculated increment. Cladding strain, fuel and cladding temperatures, fission gas release, rod internal pressure are calculated as well as time and burnup-dependent fuel and cladding properties. In addition, the code is used to establish initial conditions for ramping and accident analyses.

Starting with the ATRIUM-10X fuel design, the NRC has approved use of the newer RODEX4 method at BFN. RODEX4 supports thermal mechanical analysis and development of the LHGR limit. RODEX4 is considered a “best estimate” method, utilizing a Monte-Carlo analysis process. The older ROCEX2A methodology is still retained to provide conservative input to transient and LOCA analyses.

3.2.5.2 Evaluation Results

The thermal-mechanical evaluations described above have been completed by GE for all fuel designs included in Reference 2 (General Electric Fuel Bundle Designs). The evaluations demonstrate that these fuel designs meet the required thermal-mechanical licensing criteria documented in GESTAR II.

Similarly, AREVA thermal-mechanical evaluations are described in the applicable fuel mechanical design report prepared for each cycle. These evaluations document compliance of the fuel design to the NRC approved Generic Design Criteria topical report.

3.2.6 Inspection and Testing

The GE fuel quality assurance program is described in GESTAR II. The AREVA fuel quality assurance program is described in FQM U.S. Version, Framatome ANP Fuel Sector Quality Manual. The program covers the quality control areas associated with the manufacture and inspection of new fuel for the areas of:

(1) Material and component procurement.

(2) Fabrication and assembly of components and systems.
(3) Inspection and testing.
(4) Cleaning, packaging, and shipping.

GE and AREVA also have active programs of interim and post-irradiation surveillance of both lead use assemblies and developmental BWR fuel. The GE program and the inspection techniques used are described in GESTAR II. The AREVA program is described in the referenced reports contained in the applicable fuel mechanical design report.

3.2.7 References

1. “General Electric Standard Application for Reactor Fuel,” NEDE-24011-P-A, (See Appendix N for applicable revision).

2. “General Electric Fuel Bundle Designs,” NEDE-31152P, Rev. 7, June 2000.

3. “General Electric Standard Application for Reactor Fuel (Supplement for United States),” NEDE-24011-P-A-US, (See Appendix N for applicable revision).

4. “Generic Mechanical Design Criteria for BWR Fuel Designs,” ANF-89-98(P)(A) Revision 1 and Supplement 1, Advanced Nuclear Fuels Corp., May 1995.

Table 3.2-1

(Deleted by Amendment No. 16)

Table 3.2-2

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Table 3.2-3

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Table 3.2-4

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Table 3.2-5

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Figure 3.2-1
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3.3 REACTOR VESSEL INTERNALS MECHANICAL DESIGN

3.3.1 Power Generation Objective

Reactor vessel internals (exclusive of fuel, control rods and incore flux monitors) are provided to achieve the following objectives:

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

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

c. Provide a source of neutrons to assure meaningful nuclear measurements at reactor low power levels.

3.3.2 Power Generation Design Basis

1. The reactor vessel internals shall be 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 shall be arranged to facilitate refueling operations.

3. The reactor vessel internals shall include devices that permit assessment of the core reactivity condition during periods of low power and subcritical operations.

4. Adequate working space and access shall be provided to permit adequate inspection of reactor vessel internals.

3.3.3 Safety Design Basis

1. The reactor vessel internals shall be 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 shall be limited to assure that the control rods and the Core Standby Cooling Systems can perform their safety functions during abnormal operational transients and accidents.

3. The reactor vessel internals mechanical design shall assure that safety design bases 1 and 2 are satisfied in accordance with the loading criteria of Appendix C, so that the safe shutdown of the plant and removal of decay heat are not impaired.

3.3.4 Description

The reactor vessel internals are installed inside the reactor vessel 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 that can be flooded following a break in the nuclear system process barrier external to the reactor vessel. The reactor vessel internals are the following components:

Core shroud
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
Vessel head cooling spray nozzle
Differential pressure and liquid control line
Incore flux monitor guide tubes
Startup neutron sources
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.

Although not mandatory, 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 most of the fabrication of the reactor vessel internals is solution heat-treated, unstabilized type 304 austenitic stainless steel conforming to ASTM specifications. Weld procedures and welders are qualified in accordance with the intent of Section IX of the ASME Boiler and Pressure Vessel Code. The floodable inner volume of the reactor vessel is shown on Figure 4.3-4. It is the volume inside the core shroud up to the level of the jet pump nozzles. The boundary of the floodable inner volume consists of the following (see Figure 3.3-1):

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

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

c. The reactor vessel wall below the shroud support.

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

3.3.4.1 Core Structure

The core structure surrounds the active 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. The core structure is designed in accordance with the structural loading criteria of Appendix C.

3.3.4.1.1 Core Shroud

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 thus providing a floodable region following a recirculation line break. 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 guide below. The central portion of the shroud surrounds the active fuel and forms the longest section of the shroud. This section has the 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 (see 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. Long holddown bolts are used for easy access during removal. 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 centrifugal type steam separators have no moving parts. In each separator, the steam-water mixture rising through the standpipe passes 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 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, incore 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. Detent sockets are provided beneath the top guide to anchor dry tubes, power range monitor incore detectors, and neutron sources. The top guide is positioned by alignment pins which fit into radial slots in the top of the 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, are located at the outer edge of the active core and are not adjacent to control rods. Each peripheral fuel support piece will support 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 will 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 rod blades 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 core cell (see Subsection 3.2, "Fuel Mechanical Design").

3.3.4.3 Control Rod Guide Tubes

The control rod guide tubes, located inside the vessel, (see Figure 3.3-1) extend from the top of the control rod drive housings up through holes in 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 control rod drive housing (see 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 control rod drive housing from below and is rotated to lock the control rod drive tube in place. A key is inserted into a locking slot in the bottom of the control rod drive 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 (see 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 supported vertically by the shroud support, a flat ring which is welded to the reactor vessel wall and to which is welded the shroud support cylinder. The joint between the throat and the diffuser is a slip 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 posts 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 jet pump diffuser section is welded to the shroud support and provides a positive seal. This permits reflooding the core to the top of the jet pump inlet following a design basis loss-of-coolant accident.[1]

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1 "Design and Performance of GE BWR Jet Pumps," General Electric Co., Atomic Power Equipment Department, July 1968. (APED-5460).

3.3.4.5 <u>Steam Dryers</u>

The steam dryer removes moisture from the wet steam which exits from the steam separator. The wet steam leaving the steam separator flows across the dryer vanes and the moisture flows down through collecting troughs and tubes to the water above the downcomer annulus (see Figure 3.3-6). A skirt extends down into the water to form a seal between the wet steam plenum and the dry steam flowing out the top of the dryer to the steam outlet nozzles. Vertical guide rods facilitate positioning the dryer and shroud head in the vessel. Replacement steam dryers (RSD) have been designed to support Extended Power Uprate (EPU) operation. The RSDs hare curved hood six-bank dryers constructed mainly of type 304L stainless steel. The RSD design incorporates design features that were developed to accommodate flow induced vibration (FIV) acoustic loads that lead to steam dryer failures at a BWR-3 plant. The dryer rests on steam dryer support brackets attached to the reactor vessel wall. The original steam dryers are restricted from lifting by steam dryer holddown brackets which are attached to the reactor vessel closure head over the top of the steam dryer lifting lugs when the head is in place. RSDs are restricted from vertical lifting by latch assemblies which hook underneath two of the steam dryer support brackets.

3.3.4.6 <u>Feedwater Spargers</u>

As a result of cracks discovered in the feedwater nozzle blend radius, nozzle bore regions, and around the sparger flow holes, the General Electric Company (GE) developed an improved nozzle/sparger design which would reduce this cracking. This new design has been installed in all three units. A separate sparger is fitted to each feedwater nozzle (6) with a double piston ring thermal sleeve. Each sparger is shaped to conform to the curve of the vessel wall and is attached to the wall with two end brackets. These end brackets are bolted to the vessel wall brackets which support the weight of the spargers and position the sparger away from the vessel wall. Flow nozzles are welded to the inner radii of the sparger. Feedwater flow enters the center of the sparger and is discharged radially inward and downward through the nozzles 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.

This improved nozzle/sparger design, in conjunction with an ultrasonic testing inspection program, will preclude the possibility of any crack growing to a depth which would endanger the pressure vessel integrity.

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 as shown in Figures 4.2-1 and 4.2-3 (see 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 routings and supports are designed to accommodate differential movement between the shroud and the vessel. The lower portion of Core Spray Downcomer “C” has been replaced with a sectional replacement (Unit 3 only). The other core spray line is identical except that the header enters the opposite side of the vessel and the sparger rings are at a slightly different elevation 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 (see Section 6, "Core Standby Cooling Systems").

3.3.4.8 Vessel Head Cooling Spray Nozzle

The vessel head cooling spray nozzle is mounted to a short length of pipe and a flange, which is bolted to a mating flange above a reactor vessel head nozzle. The vessel head cooling spray nozzles are still installed but are not functional since blind flanges are permanently installed at the mating flange.

3.3.4.9 Differential Pressure and 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 (discussed in Subsection 3.8, "Standby Liquid Control System") and to sense the differential pressure across the core support assembly (see 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 ever be used. 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 Incore Flux Monitor Guide Tubes

The incore flux monitor guide tubes are welded to the top of the incore flux monitor housings (see Subsection 4.2, "Reactor Vessel and Appurtenances Mechanical Design") in the lower plenum and extend up to the top of the core support. The power range detectors for the power range monitoring units and the dry tubes for the source range monitoring/intermediate range monitoring (SRM/IRM) detectors are inserted through the guide tubes and are held in place below the top guide by spring tension. A lattice work of clamps, tie bars, and spacers is bolted around the guide tubes at the approximate level of the reactor vessel shroud support to give 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 Startup Neutron Sources

Startup neutron sources are used to provide a sufficient neutron population to assure that the core neutron flux is detectable by installed neutron monitors and to assure that significant changes in core reactivity are readily detectable by installed neutron flux instrumentation. Antimony-beryllium neutron sources were used for cycle 1. Antimony-beryllium or californium neutron sources may be used in later cycles if spent fuel alone cannot provide the required neutron population. (See Subsection 7.5, "Neutron Monitoring System").

3.3.4.12 Surveillance Sample Holders

The surveillance sample holders are welded baskets containing impact and tensile specimens capsules (see 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 active core and at radial positions of 30°, 120°, and 300°. These locations are chosen to expose the specimens to the same environment and maximum neutron fluxes experienced by the reactor vessel itself while at the same time avoiding jet pump removal interference or damage.

3.3.5 Safety Evaluation

3.3.5.1 Evaluation Methods

To determine that the safety design basis is satisfied, the responses of the reactor vessel internals to loads imposed during normal operation, abnormal operational transients, and accidents were examined. Determination of these effects on the ability to insert control rods, cool the core, and flood the inner volume of the reactor vessel was made. The various structural loading combinations assumed to be imposed on the reactor vessel internals are as described in Appendix C for Class I equipment. These loading combinations include upset loads, emergency loads, and faulted loads.

The ASME Boiler and Pressure 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 Code, it was concluded that the safety design basis is satisfied. For those components, for which either buckling is a possible failure mode or stresses exceed those presented in the ASME Code, then either the elastic stability of the structure or the resulting deformation was examined to determine if the safety design basis was satisfied.

3.3.5.1.1 Specific Events to be Evaluated

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

a. Loss-of-coolant accident. This accident is a break in a recirculation line. The accident results in flow induced loads and acoustic shock loads on some of the reactor vessel internals.

b. Steamline break accident. This accident is a break in one main steamline between the reactor vessel and the flow restrictor. The accident results in significant pressure differentials across the reactor vessel internals.

c. Thermal shock. The most severe thermal shocks to the reactor vessel internals occur when low pressure coolant injection or high pressure coolant injection (LPCI or HPCI) operations reflood the reactor vessel inner volume following either a recirculation line break or a main steamline break (see Section 6, "Core Standby Cooling System").

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

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

3.3.5.1.2 Reactor Internal Pressure Differentials (RIPD)

The core flow, feedwater temperature, reactor pressure, and power level that result in the maximum loads are used as initial conditions.

For GE analyses, the normal condition is analyzed using a digital computer code[2] for the steady-state thermal-hydraulic analysis of a BWR core (ISCOR). The reactor internal pressure differences (RIPDs) are calculated based on the ISCOR results. Appropriate adders or multipliers, which have been conservatively established based on the GE transient analysis methods on a generic basis for GE BWRs, are applied to the normal condition values to determine the upset condition values. For AREVA reload analyses, core and fuel channel RIPDs are evaluated based on XCOBRA analyses.

The locations at which RIPDs are applicable and calculated are shown in Figure 3.3-7. The bounding RIPD values which result from the range of possible reactor power and flow conditions are then determined.

The RIPD values acting on the major components at normal increased core flow conditions are documented in the following reference:

GEH 002N3430, Revision 0, Task T0304: Reactor Internal Pressure Differences and Fuel Lift Evaluation, March 2015, Table 3.3.1.1.

NOTE: The fuel type utilized in the reference is GE13, which is a bounding design for RIPD analysis. AREVA fuel designs in use at BFN are bounded by GE13 as documented in the following reference:

0000-0166-0876-R0, T304 RIPD Assessment for Browns Ferry Control Blade-Channel Interface SC 11-05 Seismic Analysis, GE Hitachi Nuclear Energy, April 2013.

3.3.5.2 Recirculation Line Break

This accident is the same design basis loss-of-coolant accident as described in Section 6, "Core Standby Cooling Systems," and Section 14, "Plant Safety Analysis." It is assumed that an instantaneous, circumferential break occurs in one recirculation loop. The reactor is assumed to be operating at design power with rated recirculation flow at the time of the break.

The recirculation line break LOCA results in short term transient loads which affect those components in the vicinity of the recirculation outlet nozzle. The resulting

2 EPIC/ISCOR, GE Propriety Code

flow-induced and acoustic loads from the recirculation line break LOCA have been determined for a reference BWR plant using a recent three-dimensional thermal-hydraulic transient analysis computer code (TRACG, GE Propriety Code). Browns Ferry specific loads are then determined by scaling the reference plant loads by the effects of the geometric and thermal-hydraulic parameter differences between the reference plant and Browns Ferry. The flow-induced and acoustic loads resulting from the introduction of the TRACG method of calculation are greater than the results determined for past Browns Ferry evaluations.

In order to determine the impact of off-rated power/flow conditions, load multipliers are calculated normalized to a multiplier of 1.0 for the normal power and rated core flow condition. Thus, the effect of off-rated power and flow conditions or increased core flow is illustrated by the normalized multipliers.

The geometric scaling is based on the dimensional differences between the Browns Ferry reactor configuration and the reference BWR plant in the vicinity of the recirculation outlet and shroud annulus region. The thermal-hydraulic scaling is determined by using ISCOR program. The load multipliers for the selected points on the power flow map are also determined by using the ISCOR program.

The acoustic loads are short term transient impulse loads and are conservatively evaluated by determining a static equivalent load. The flow induced loads are more slowly applied transient loads as compared to acoustic loads; therefore, the loads are not added. The higher load governs structural evaluations. Detailed evaluation of these loads indicated that the acoustic loads resulted in higher loads and are the loads used in structural evaluations.

Flow-induced and acoustic loads for a recirculation line break LOCA from operation at minimum recirculation pump speed typically results in the highest acoustic load regardless of the rated power or core flow; however, the probability of a LOCA occurring at this in-frequent operational condition is extremely remote. Therefore, evaluations are based on the more probable LOCA during maximum extended load line limit (MELLL) operation.

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 two possible sources of leakage are:

a. Jet pump throat to diffuser joint.

b. Jet pump nozzle to riser joint.

The jet pump to shroud support joint is welded and therefore is not a possible source of leakage. The throat to diffuser joints for all jet pumps by analysis are shown to leak no more than a total of 225 gpm. The jet pump nozzle to riser joint by analysis is shown to leak no more than 582 gpm for the pumps through which the vessel is being flooded. The summary of maximum leakage is then:

225 gpm Total leakage through all throat-to-diffuser joints
582 gpm Total leakage through all nozzle-to-riser joints
807 gpm Total maximum rate (Units 2 and 3)
157 gpm Additional leakage through bolted design access hole covers (Unit 1 only)
964 gpm Total maximum rate (Unit 1)

LPCI capacity is sized to accommodate 3000 gpm leakage at these locations. It is concluded that the reactor vessel internals retain sufficient integrity during the recirculation line break accident to allow reflooding the inner volume of the reactor vessel.

3.3.5.3 Steamline Break Accident

The analysis of this accident assumes an instantaneous circumferential break of one main steamline between the reactor vessel and the main steamline flow restrictor. This is not the same accident as that described in Chapter 14, "Plant Safety Analysis," (which postulates a break downstream of the flow restrictors) 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. However, mass flow rate is inversely proportional to escape enthalpy, h_e; therefore, depressurization rate is proportional to $1-h_f/h_e$. Consequently, depressurization rate decreases as h decreases, that is, depressurization is less for mixture flow than for steam flow.

The main steam line (MSL) break inside containment is postulated for calculating the RIPDs, except for the steam dryer RIPD for which the steamline break outside containment is postulated, as specified in Section 3.6 of the UFSAR. These faulted loading conditions are analyzed using the LAMB thermal-hydraulic analysis code. The bounding RIPD values for MSL break from the cavitation interlock (20% power/110% core flow) or full power operation is determined and used for reactor internals structural evaluations.

The maximum RIPD values for this faulted condition are identified in the following reference:

GEH 002N3430, Revision 0, Task T0304: Reactor Internal Pressure Differences and Fuel Lift Evaluation, March 2015, Table 3.3.1.1.

Note: The fuel type utilized in the reference is GE13, which is a bounding design for RIPD analysis. AREVA fuel designs in use at BFN are bounded by GE13 as documented in the following reference: 0000-0160-0876-R0, T304 RIPD Assessment for Browns Ferry Control Blade Channel Interface Sc 11-05 Seismic Analysis, GE Hitachi Nuclear Energy, April 2013.

These maximum differential pressures are used, in combination with other assumed structural loads as described in Appendix C, 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 fuel bundle, the core support, the guide tubes, and the metal channels around the fuel bundles.

3.3.5.3.1 Core Support

The two considerations important to the core support evaluation are sliding of the core support and buckling of the supporting beams. Evaluations have determined that the core support will not slide under the postulated accident conditions with preload on the holddown bolts. Additional resistance to sliding is provided by aligners which further stabilize the core support.

The core plate buckling pressure is evaluated by a computer program[3] that calculates core plate stiffener beam-buckling capability. It uses the Rayleigh-Ritz energy method to determine the applied moment to begin yielding, and then to buckle a given tee beam. The tee beam models a segment of a BWR core plate with a stiffener beam. The pressure differential across the plate that would have created this moment is calculated for the longest core plate beam.

The MSL break pressure of 28.5 psid (3952 MWt) is within the permissible load allowed by the Buckling Stability Criteria of Table C.2-3(b) for the safety factors in accordance with Appendix C: Sections C.2 and C.2.6 required emergency and faulted load combinations.

3.3.5.3.2 Guide Tubes

Because the guide tube experiences external-to-internal pressure differentials, the guide tubes were examined for buckling under these conditions. For a guide tube with minimum wall thickness and maximum allowed ovality, the pressure which causes buckling is 105 psi compared to the main steam line break design pressure of 28.5 psid (3952 MWt). This pressure is well within the permissible load allowed

3 PIPST01, GE Proprietary Code

by the Buckling Stability Criteria of Table C.2-3(b) for the safety factors in accordance with Appendix C: Sections C.2 and C.2.6 required emergency and faulted load combinations.

It is concluded that the guide tube will not fail under the assumed accident conditions.

3.3.5.3.3 Fuel Channels

The description of the testing and analysis performed for initial core channels is provided in this section. The NRC approval of the channels used with the current AREVA reload is provided in EMF-93-177(P)(A) Revision 1 with the plant and cycle specific analyses reported in the reload specific fuel mechanical design report.

The fuel channel load due to an internally applied pressure was examined utilizing a fixed-fixed beam analytical model under a uniform load. Tests were 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 guide beam location, etc., that determine the clearance between the control rod blade and fuel channel. If each of these tolerance factors are 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 is a roller or spacer pad to guide the blade as it is inserted. The clearance between channels is 70 mils less than the diameter of the roller or spacer pad, causing it to slide or skid instead of roll. As the rod is inserted about halfway 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 sigma limits, where sigma is the standard deviation in a point distribution of events. Three sigma lies in the 0.995 percentile of probability of nonoccurrence. However, even if interference occurs, the result is negligible. About 1 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 control rod drives.

The above 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 percent 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 roller or spacer pad is at the top of the rod. It is concluded that the main steamline break accident can pose no significant interference to the movement of control rods. Also, the calculated maximum pressure differential across the core is approximately 12 psi below the 42 psi required to lift a fuel bundle.

The AREVA methodology for evaluating the fuel channel deformation due to the internal pressure and irradiation growth makes use of a Monte Carlo analysis to determine the probability of having a stuck control blade condition. A 95/95 statistical statement is made, taking into account variations in core tolerances, fuel channel tolerances, bulge and bow, to demonstrate acceptable interface with the surfaces of the control blades.

Fuel lift is evaluated for AREVA fuel assuming maximum differential pressure conditions. A substantial margin is calculated to exist prior to lift off.

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 loss-of-coolant accident, such as the recirculation line break and the steamline break accidents previously described.

Three specific locations are of particular interest, as shown in Figure 3.3-9. The locations are as follows:

a. Shroud support plate,

b. Shroud-to-shroud support plate discontinuity, and

c. Shroud inner surface at highest irradiation zone.

The peak strain resulting in the shroud support plate is about 6.5 percent. This strain is higher than the 5.0 percent strain permitted by the ASME Code, Section III, for ten cycles, but for one cycle, peak strain corresponds to about six allowable cycles of an extended ASME Code as applied to less than ten cycles.

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 extension of the ASME Code curve represents a similar criteria to that used in the ASME Code, Section III, but applied to fewer than ten cycles of loading. For this type 304 stainless steel material, a 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 Code, Section III, allows 220 cycles of this loading, thus no significant deformations result. The most irradiated point on the inner surface of the shroud is subjected to a total integrated neutron flux of 2.7 x 10^{20} nvt (>1 MeV) by the end of plant life. 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 material does experience some hardening and an apparent loss in uniform elongation, but it does not experience a loss in reduction of area. Because reduction of area is the property which determines tolerable local strain, irradiation effects can be neglected. 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 RPV and RPV internals are determined from dynamic earthquake analysis described in Section 12.2 using the mathematical models of the RPV and internals shown in Figures 12.2-27B and 12.2-27C. The design of the RPV and internals are described in Section C.4 of Appendix C and Appendices J, K, and L.

RPV Internals and Structural Integrity Evaluations were performed with GE fuel per the following Reference:

002N4782, Revision 0, Task T0303: RPV Internals Structural Integrity Evaluation, GE Hitachi Nuclear Energy, April 2015.

For AREVA fuels, the evaluations were performed per the following Reference:

0000-0166-4147, Revision 0, RPV Internals Structural Integrity Evaluations for AREVA ATRIUM-10 and ATRIUM-10XM Fuels, GE Hitachi Nuclear Energy, January 2014.

3.3.5.6 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, 2, and 3 are satisfied.

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 preoperational test program, operational readiness tests are performed on various systems. In the course of these tests such reactor vessel internals as the feedwater spargers, the core spray lines, the vessel head cooling spray nozzle, and the Standby Liquid Control System line are functionally tested.

Steam separator-dryer performance tests were run to determine carryunder and carryover characteristics on the first 1098 MWe boiling water reactor plant to go into operation. Samples were taken from the inlet and outlet of the steam dryers and from the inlet to the main steamlines 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.

Vibration analysis of reactor vessel internals is included in the design to reduce failures due to vibration. When necessary, vibration measurements are made during startup tests to determine the vibration characteristics of the reactor vessel internals and the recirculation loops under forced recirculation flow. Vibratory responses are recorded at various recirculation flow rates using strain gauges 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 are designed to determine any potential, hydraulically induced equipment vibrations and to check that the structures will 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. With this analysis as a guide, the reactor internals were instrumented and tested to ascertain that there were no gross instabilities. The cyclic loadings are evaluated using as a guide the cyclic stress criteria of the ASME Code, Section III. These field tests were performed only on reactor vessel internals that represented a significant departure from design configurations previously tested and found to be acceptable. Field test data were correlated with the analyses to ensure validity of the analytical techniques on a continuing basis[4].

The reactor vessel and internals are designed to assure adequate working space and access for inspection of selected components and locations[5]. The criteria for selecting the components and locations to be inspected are based on the probability of a defect occurring or enlarging at a given location and includes areas of known stress concentrations and locations where cyclic strain or thermal stress might occur. The reactor vessel internals inspection program is detailed in Subsection 4.12, "Inservice Inspection and Testing."

After installation, the RSDs are evaluated during initial power ascension to assess their structural performance. The RSD for the lead unit is assessed for structural integrity through measured on-dryer strains and acceleration. The RSDs for the follow on units are assessed through the analysis of dryer loads projected from main steam line strain gauge measurements. Reactor parameters that have been indicative of past dryer structural failures are also monitored during initial power ascension.

4 Quad-Cities Station Units 1 and 2 Docket No.'s 50-254 and 265, Amendment 19

5 Brandt, F. A., "Design Provision for In-Service Inspection," General Electric Company. Atomic Power Equipment Department, April 1967 (APED-5450)

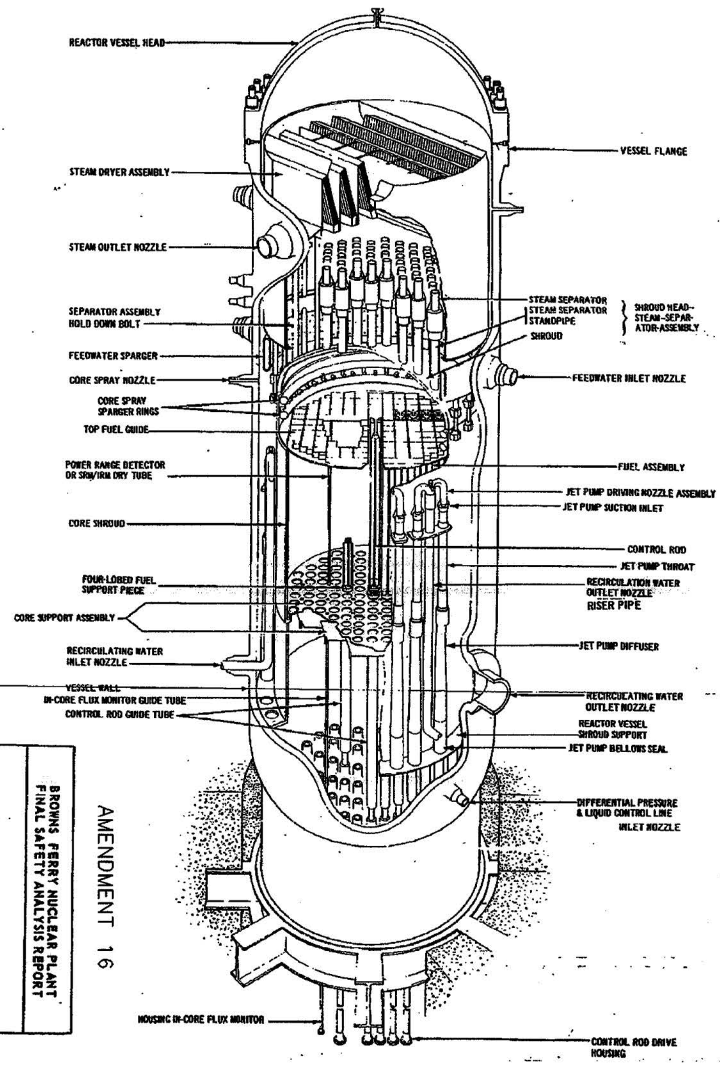
Table 3.3-1

REACTOR VESSEL INTERNALS, DESIGN DATA

Core Shroud	
Upper Portion,o.d.,in.	220
Central Portion,o.d.,in.	207
Lower Portion,o.d.,in.	201
Central Portion, thickness, in.	2
Weight, lb.	116,900
Shroud Head-Steam Separator Assembly	
Head Thickness, in.	2.0
Number of Separators	211
Separator o, d., in.	12.75
Number	6
Standpipe i. d., in.	6.065
Standpipe o, d., in.	6.625
Weight, lb.	139,600
Core Support	
Weight, lb.	20,500
Top Guide	
Weight, lb.	15,200
Fuel Support Pieces	
Number of Preipheral	24
Four Lobe	
Number without Plugs	185
Number with Plugs	0
Weight, lb.	11,300
Control Rod Guide Tubes	
Number	185
Weight, lb.	46,250

Jet Pumps	
Number	20
Throat Diameter, in.	8.18
Weight, lb.	22,700
Original Steam Dryers	
Weight, lb	90,000
Replacement Steam Dryers	
Weight, lb.	119,000
Feedwater Sparger	
Dimensions, in.	6 Sched. 40
Cross Section Area, ft^2	0.2006
Number	6
Core Spray Sparger Rings	
Diameter, in.	4 Sched. 40 S
Cross Section Area, ft^2	0.088
Number of Spray Outlets	260
Weight, lb.	4,317
Vessel Head Cooling Spray Nozzle	
Pipe Size, in.	4
Scheduled	40
Differential Pressure & Liquid Control Line	
Inner Pipe (Liquid Control), in	1 Sched. 40
Outer Pipe, in.	2 Sched. 40
Incore Flux Monitor Guide Tubes	
Number	55
Surveillance Sample Holders	3
Total Weight of Reactor Vessel Internals, pounds (excluding fuel)	477,000

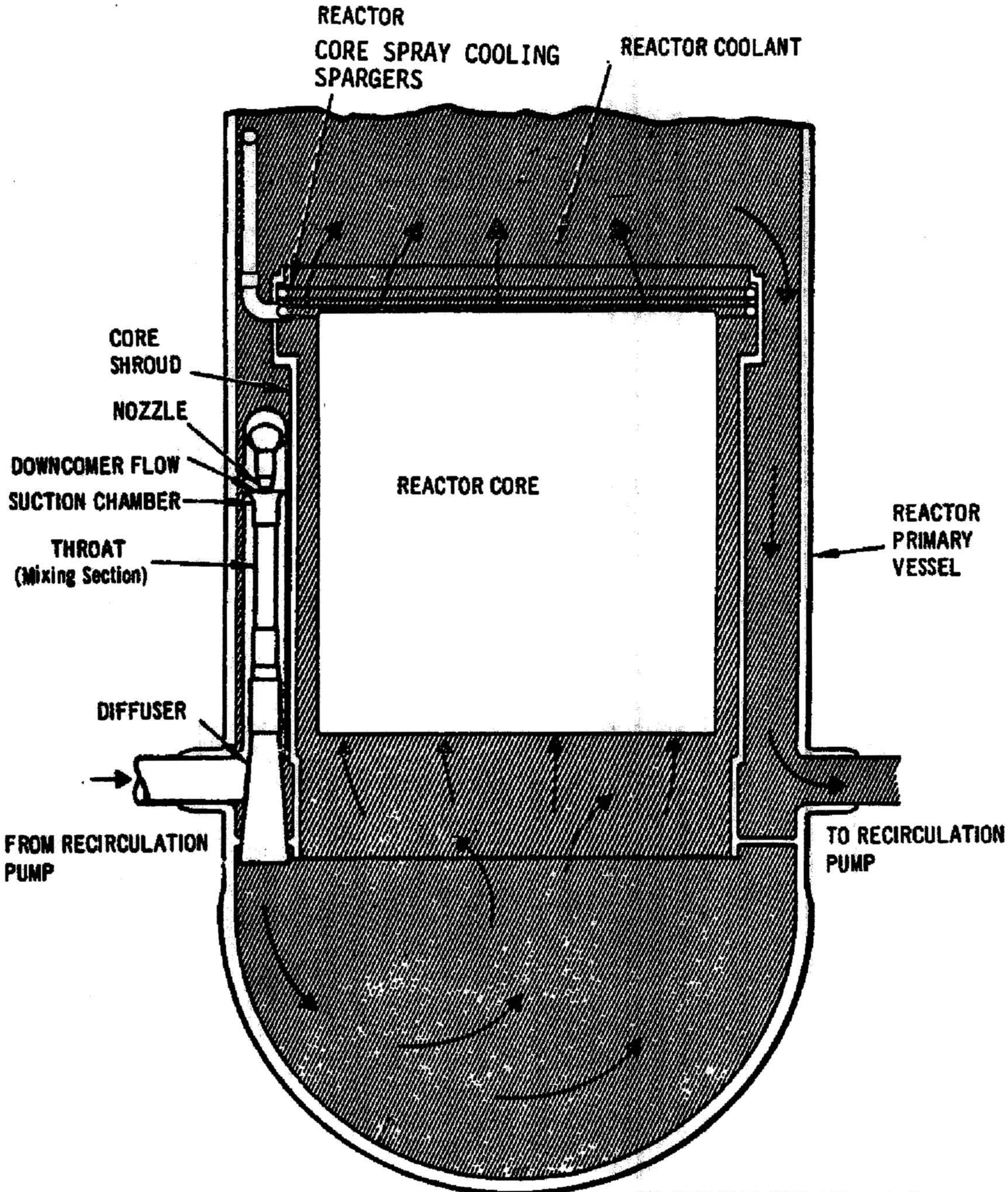
BFNP-



AMENDMENT 16

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

Reactor Vessel Internals
FIGURE 3.3-1



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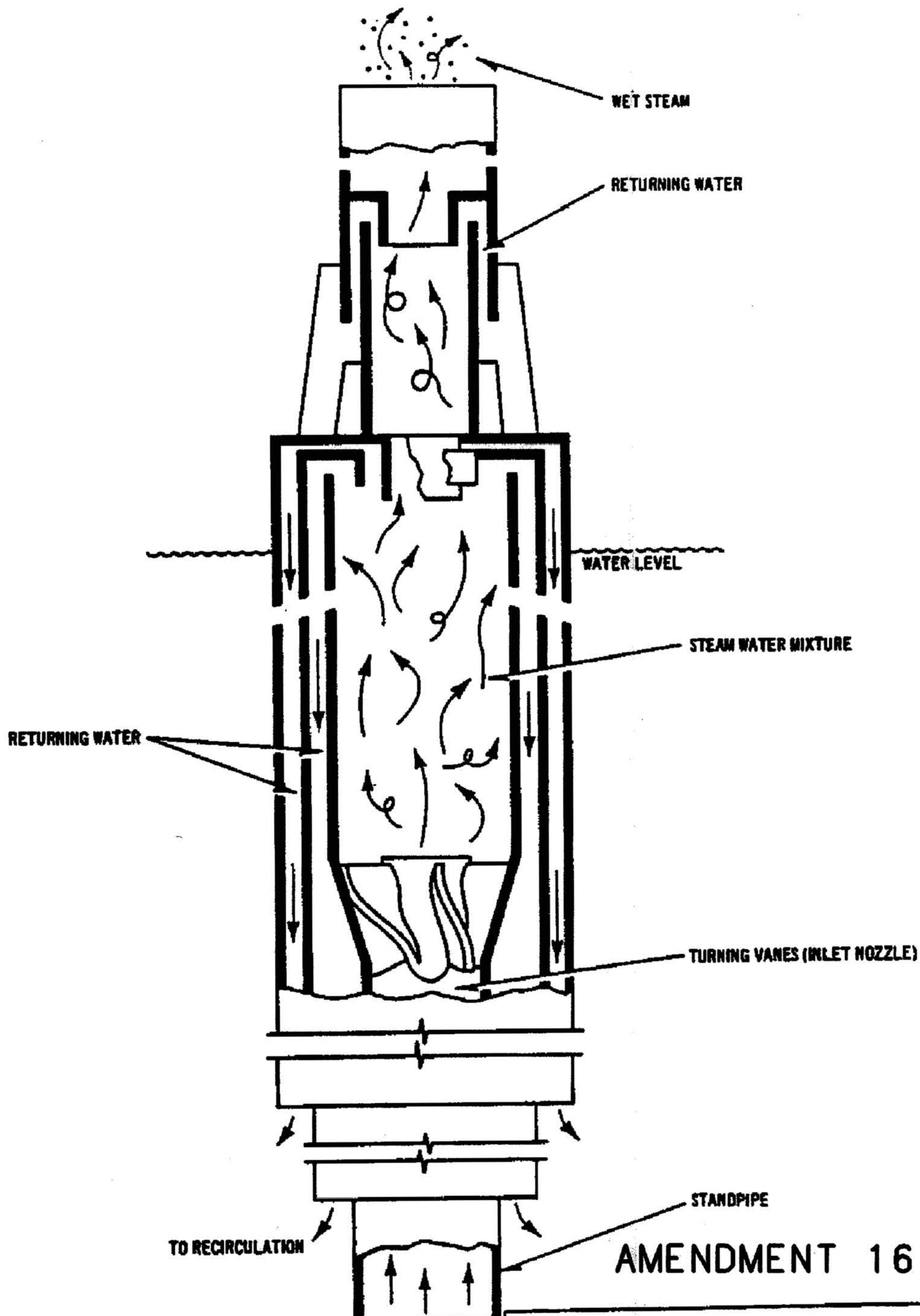
BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

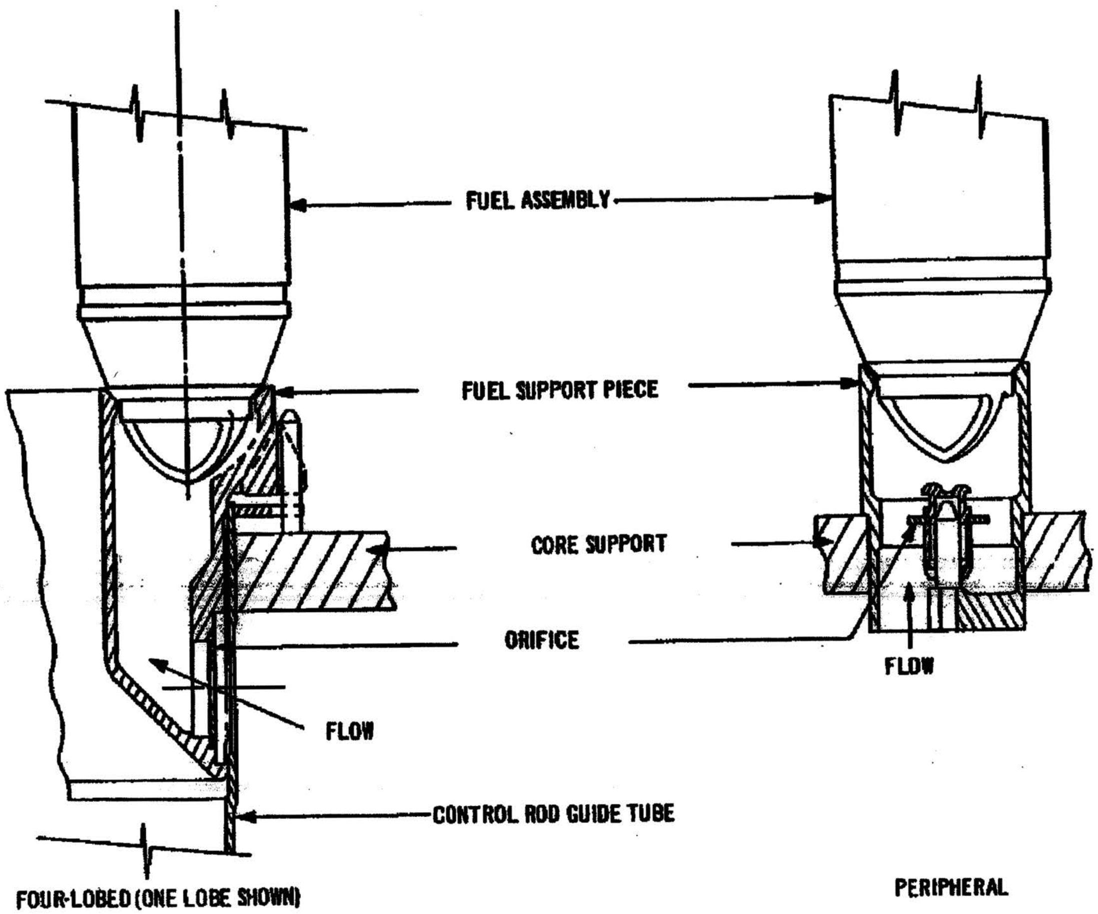
Reactor Vessel Internal Flow – Schematic
FIGURE 3.3-2



AMENDMENT 16

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

Steam Separator
FIGURE 3.3-3



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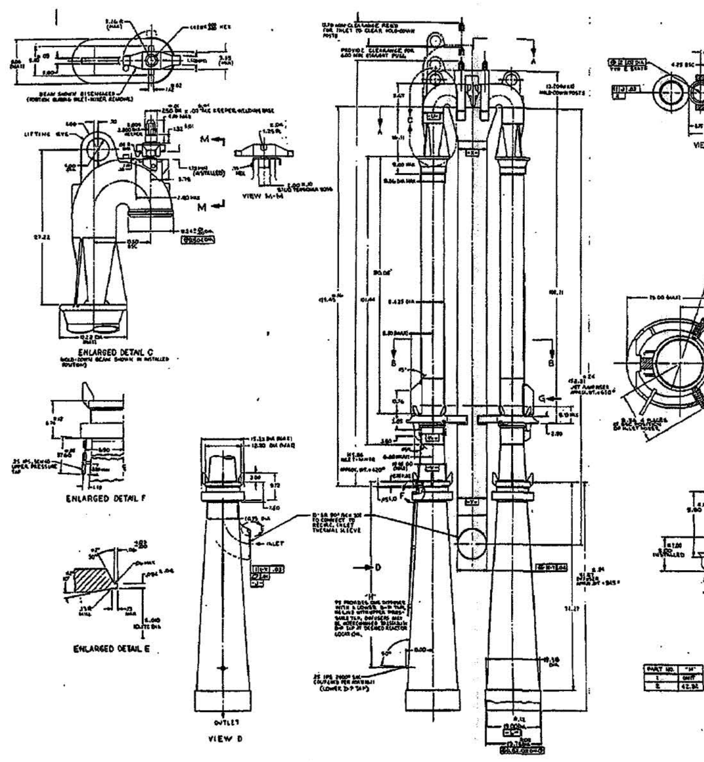
FINAL SAFETY ANALYSIS REPORT

Fuel Support Pieces
FIGURE 3.3-4



BROWNS FERRY NUCLEAR PLANT
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Jet Pump
FIGURE 3.3-6

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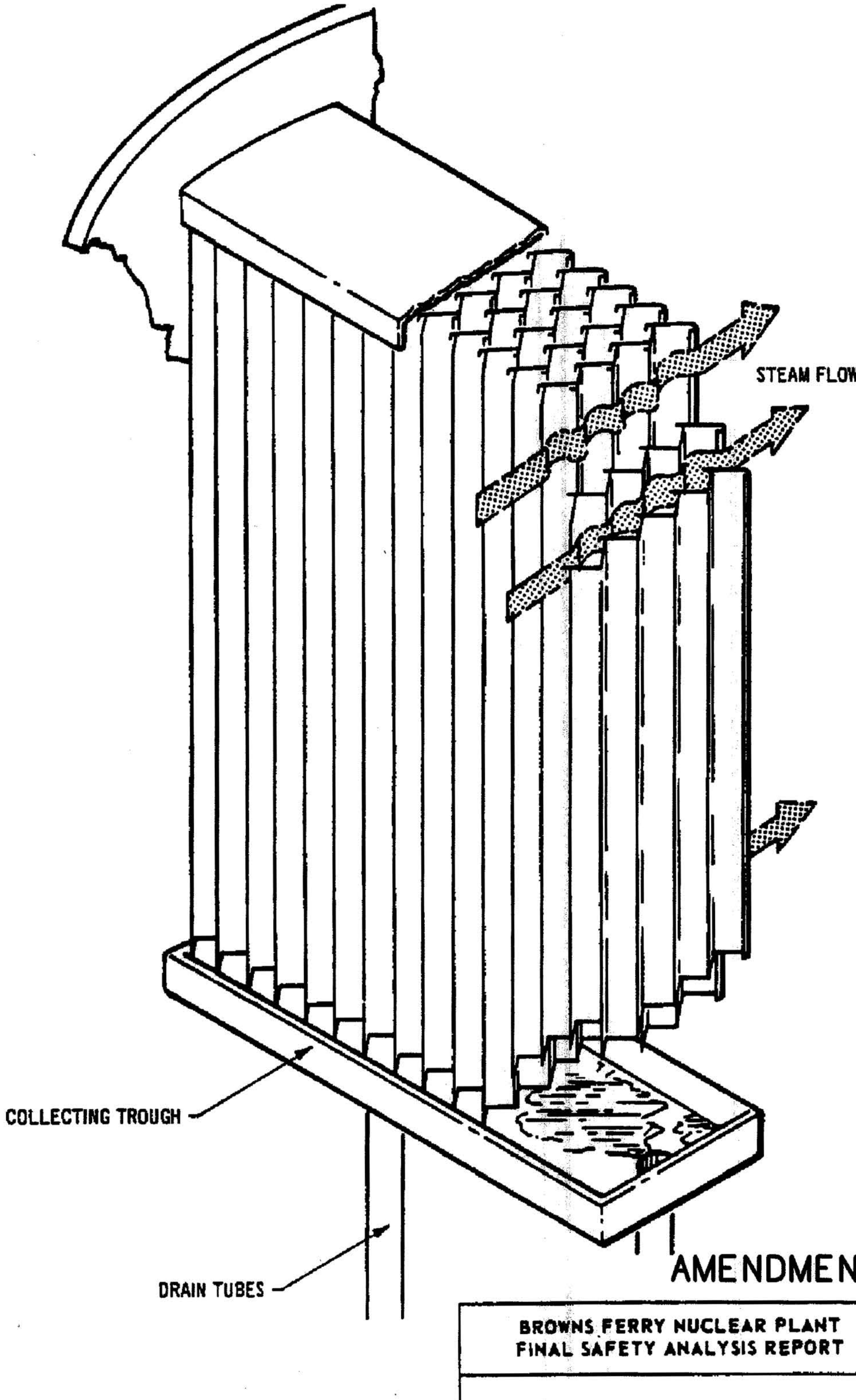

AMENDMENT 16

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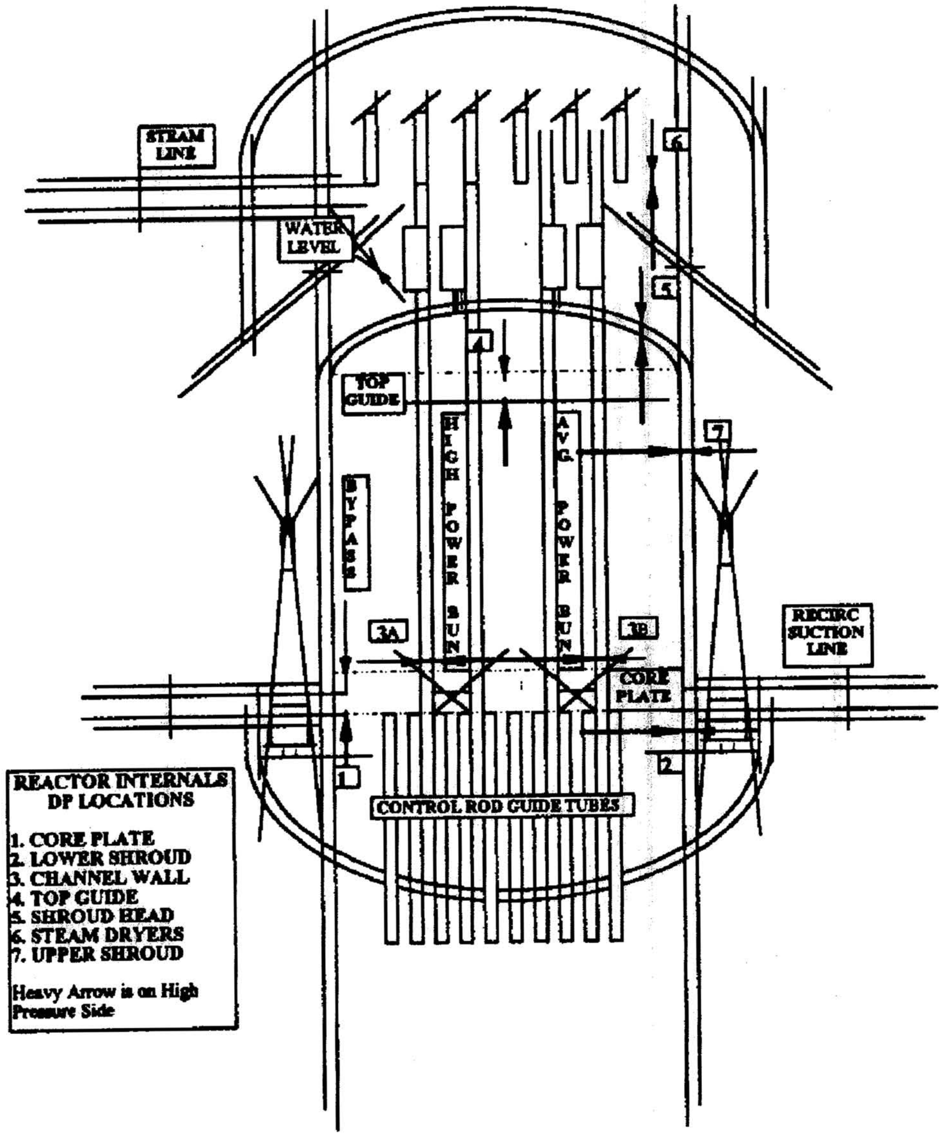
Steam Dryer
FIGURE 3.3-6



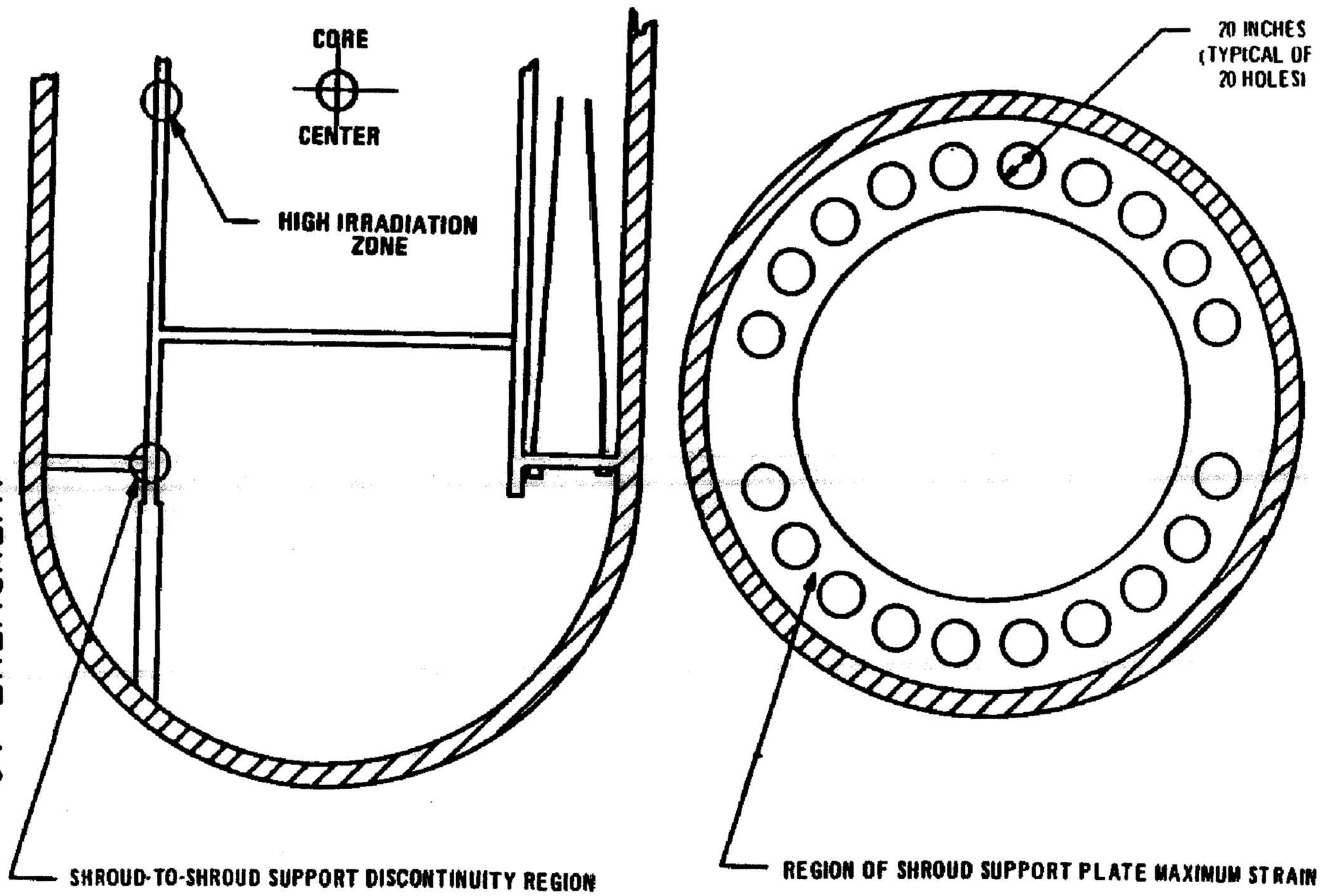
AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

REACTOR INTERNAL PRESSURE
DIFFERENCE (RIPD) LOCATIONS

FIGURE 3.3-7

Figure 3.3-8
(Deleted by Amendment 17)



AMENDMENT 16

BROWNS FERRY NUCLEAR PLANT
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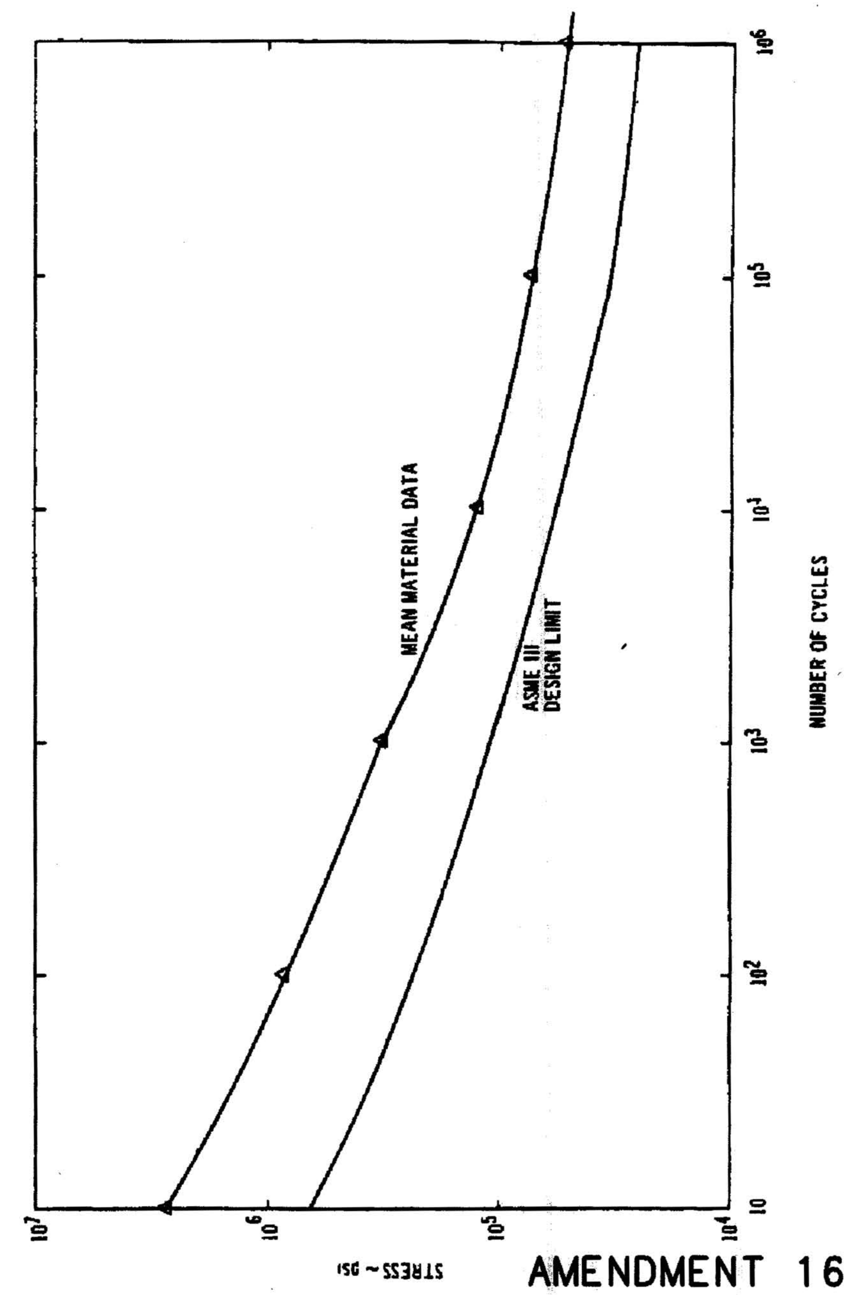
Thermal Shock Transient Analysis Zones
FIGURE 3.3-9



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BROWNS FERRY NUCLEAR PLANT
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Material Behavior Graph -- Cycles vs. Stress
for Stainless Steel
FIGURE 3.3-10

BFN-16

Figure 3.3-11

Deleted by Amendment 10.

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 to quickly terminate the nuclear fission process in the core so that damage to the fuel barrier is limited. The objective is met by inserting reactivity control devices into the reactor core.

3.4.2 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 reactivity control devices in the reactor core.

3.4.3 Safety Design Basis

1. The reactivity control mechanical design shall include control rods.

    a. The control rods shall be so designed and have sufficient mechanical strength to prevent the displacement of their reactivity control material.

    b. The control rods shall have sufficient strength and be of such design as to prevent deformation that could inhibit their motion.

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

2. The reactivity control mechanical design shall provide for a sufficiently rapid insertion of control rods so that no fuel damage results from any abnormal operating transient.

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

4. Each positioning device shall:

    a. Prevent its control rod from withdrawing as a result of a single malfunction.

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

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

d. Be 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. Be locked to its control rod to prevent undesirable separation.

f. Provide positive indication of control rod position to the operator.

3.4.4 Power Generation Design Basis

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

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

3. The reactivity control mechanical design shall provide indication of the CRDM temperature to the operator.

3.4.5 Description

The reactivity control mechanical design consists of control rods which can be positioned in the core, during power operation, by individual control rod drive (CRD) mechanisms.

The CRD mechanisms are part of the CRD System. The CRD System hydraulically operates the CRD mechanisms using water from the condensate storage system as a hydraulic fluid. The CRD mechanisms are used to manually position individual control rods during normal operation but act automatically to rapidly insert all control rods during abnormal (scram) conditions.

The control rods, CRD mechanisms, and that part of the CRD Hydraulic System necessary for scram operation are designed as Class I equipment in accordance with Appendix C, "Structural Qualification of Subsystems and Components."

3.4.5.1 Reactivity Control Devices

3.4.5.1.1 Control Rods

The control rods 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 control rods are positioned in a manner which counterbalances steam void effects at the top of the core and results in significant power flattening.

Five General Electric control rod designs and one Westinghouse design are currently approved for use in BFN reactors: (1) Original Equipment, (2) Modified BWR/6, (3) Hybrid I, (4) Marathon, (5) Ultra, and (6) Westinghouse CR-82M-1. All of these control rods are "Matched Worth" designs. The reactivity worth of the replacement control rods is nearly identical to the Original Equipment control rod design so that all the designs can be used interchangeably without affecting lattice physics and core reload analyses or core monitoring software. These control rods are also designed to be interchangeable considering system performance and mechanical fit. A brief description of each design follows.

Original Equipment Control Rod

The Original Equipment control rod consists of a sheathed cruciform array of neutron absorber rods consisting of stainless steel tubes filled with boron-carbide powder. 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 control rod drive coupling, a vertical cruciform center post, and four U-shaped absorber tube sheaths. The two end castings and the center post are welded into a single skeletal structure. The U-shaped sheaths are resistance welded to the center post and castings to form a rigid housing to contain the neutron absorber rods. Rollers or spacer pads at the top and the rollers at 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 sheaths are perforated to allow the coolant to freely circulate about the absorber tubes. Operating experience has shown that control rods constructed as described above are not susceptible to dimensional distortions, as required by safety design basis 1.b.

The boron-carbide (B_4C) powder in the stainless steel 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 (B-10) content of the boron is 18.0 percent by weight minimum. The absorber tubes are made of type 304 or a high purity type 348 stainless steel. An absorber tube is 0.188 inch in outside diameter and has a 0.025 inch wall thickness (Figure 3.4-2). An 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-inch intervals. The steel balls are held in place by a slight crimp of the tube. 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 end of control blade life occurs when any quarter segment of the control blade reaches a 10-percent reduction in relative reactivity worth. The reduction in blade worth is due to a combination of Boron-10 depletion and boron carbide loss resulting from cracking of the absorber rod tubes. The mechanism identified as causing the tube failures is B_4C swelling resulting in stress corrosion cracking. Given sufficient exposure, the B_4C swelling may initiate small stress corrosion cracks on the tube surface. Examinations of high exposure blades have shown that these surface cracks may exist at average segment Boron-10 depletions greater than 20 percent. Ultimately, the cracks will propagate through the tube wall allowing reactor coolant to enter the tube. In this condition, B_4C can be leached out slowly by the reactor coolant, resulting in a loss of control blade worth. Examinations have shown that the combination of Boron-10 depletion and loss of B_4C result in a 10-percent reduction in relative control blade worth at approximately 34 percent average Boron-10 depletion. This is the defined end of useful blade life for the standard all B_4C control blades.[1]

Modified BWR/6 Control Rod

The Modified BWR/6 control rods differ from the Original Equipment blades by having increased wing thickness, increased neutron absorber, and a double bail handle (Figure 3.4-1). The absorber tubes have a 0.220 inch outside diameter and a 0.027 inch wall thickness. The BWR/6 control rods have been modified with replacement pins and rollers made from low-cobalt materials and sized for BWR/4 application. Cobalt reduction is desirable to reduce activation products within the reactor system and to reduce radiation levels of spent control rods. The design blade life for the BWR/6 control rods is 34 percent average Boron-10 depletion, which is the same as for the Original Equipment blades.

Hybrid I Control Rod

Hybrid I Control Rod (HICR) assemblies contain improved B_4C absorber rod material to eliminate cracking during assembly lifetime.[2,3] Also in the HICR design, the three outermost absorber tubes in each wing are replaced with solid hafnium rods which increase blade lifetime compared to standard all-B_4C control blades. End-of-life boron equivalent depletion for the HICRs is 56 percent for any quarter segment of the control blade.

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2 Safety Evaluation of General Electric Hybride I Control Rod Assembly", NEDE-22290-A, September 1983.

3 "General Electric BWR Control Rod Lifetime," NEDE-30931-P, March 1985.

Marathon Control Rod

The Marathon control rod (Figure 3.4-12) differs from the preceding designs by replacement of the absorber tube and sheath arrangement with an array of square tubes, which results in reduced weight and increased absorber volume[4]. The square tubes are fabricated from a high purity stabilized Type-304 stainless steel that provides high resistance to irradiation-assisted stress corrosion cracking. The absorber tubes are welded lengthwise to form the four wings of the control rod. For the BFN BWR/4 D-lattice design, each wing is comprised of 14 absorber tubes. The absorber tubes each act as an individual pressure chamber for the retention of helium which is produced during neutron absorption reactions. The four wings are welded to a central tie rod to form the cruciform-shaped member of the control rod.

The square tubes are circular inside and are loaded with either B_4C or hafnium. The absorber tubes have an inside diameter of 0.250 inches and a nominal wall thickness of 0.024 inches. The B_4C is contained in separate capsules to prevent its migration. The capsules are placed inside the absorber tubes and are smaller than the absorber tube inside diameter, allowing the B_4C to swell before it makes contact with the absorber tubes thereby providing improved resistance to stress corrosion. The B_4C capsules are fabricated from stainless steel tubing and have stainless steel caps attached by rolling the tubing into grooves in the caps. The capsules are loaded into the individual absorber tubes, which are then sealed at each end by welded end caps. The capsules securely contain the B_4C while allowing the helium to migrate through the absorber tube.

The Marathon design offers increased blade lifetimes due to the increased absorber loading and absorber tube design improvements. The Marathon blades have a quarter segment, end-of-life Boron-10 equivalent depletion limit of 68 percent.

Ultra Control Rod

The GEH manufactured Ultra MD and Ultra HD control rods (formerly known as Marathon-5S and Marathon-Ultra, respectively) are similar to the original Marathon design shown in Figure 3.4-12. The NRC acceptance of the Marathon-5S and Marathon-Ultra Control Rods is documented by Licensing Topical Reports (LTRs). [9,10] .The Ultra MD Control Rod consist of “simplified” absorber tubes, edge welded together to form the control rod wings, and welded to a full-length tie rod to form the cruciform assembly shape. The absorber tubes are filled with a combination of boron carbide (B_4C) capsules, and empty capsules. The Ultra HD

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9 NEDE-33284P-A Rev. 2, “Marathon-5S Control Rod Assembly,” October 2009.

10 NEDE-33284 Supplement 1P-A Rev. 1, “Marathon-Ultra Control Rod Assembly,” March 2012.

Control Rod is a derivative version of the Ultra MD Control Rod in that it uses an identical outage structure. The only differences for the Ultra HD Control Rod is the inclusion of full length hafnium rods in high-depletion absorber tubes, and the use of a then-wall boron carbine capsule, similar in geometry to the previous Marathon control rod design.[4] The original Marathon design was intended to reduce stress/strain associated with B_4C swelling. The Ultra MD and Ultra HD designs were developed to eliminate stress/strain associated with B_4C swelling to prevent end of life cracking issues. The Ultra MD and Ultra HD blades have quarter segment, end-of-life Boron-10 equivalent depletion limit of less than 57 percent to 67 percent, respectively.

Westinghouse CR-82M-1 Control Rod

The Westinghouse CR-82M-1 control rod (Figure 3.4-13) consists of four stainless steel sheets welded together to form the cruciform shaped rod.[5,6] Each sheet has horizontally drilled holes to contain the absorber materials (boron carbide powder and hafnium). The hafnium tip of the CR-82M-1 design protects the control rod from absorber material swelling when operated in the fully withdrawn position. The blade material used in the CR-82M-1 design is AISI 316L stainless steel. AISI 316L stainless steel as structural material is an irradiation resistant steel not readily sensitized to irradiation assisted stress corrosion cracking, and also has an extremely low cobalt content (< 0.02 percent). The design with horizontally drilled absorber holes limits the washout of boron carbide in case of a defect in a wing, thus maintaining full reactivity worth.

The absorber hole geometry for the Westinghouse CR-82M-1 control rod is optimized to provide a matched worth (within 5 percent) to the initial worth of an original equipment control rod. The nuclear end-of-life (10 percent worth decrease from initial original equipment value) depletion limits for the CR-82M-1 design are: 85 percent equivalent Boron-10 depletion for the top quarter segment, and 89 percent equivalent Boron-10 depletion for the other quarter segments. [Note: These are “equivalent” depletion limits developed to allow the utility to monitor the CR-82M-1 control rod as if it were an original equipment control rod.

Conclusion

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

3.4.5.1.2 Control Rod Velocity Limiter (Figures 3.4-3 and 3.4-4)

The control rod velocity limiter is an integral part of the bottom assembly of each control rod. This static engineered safeguard protects against a high reactivity insertion rate by limiting the 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.

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 over the length of the control rod stroke.

It is fabricated of type 304 stainless steel. It has a nominal diameter of approximately 9.2 inches, and is fitted inside the control rod guide tube which has an inside diameter of 10.4 inches. This configuration results in an annulus between the limiter and the guide tube of approximately 0.6 inch. The limiter always remains in the guide tube except when the control rod is removed. Four adjacent fuel assemblies and the fuel assembly support casting must be removed before the control rod can be removed because of the shape of the velocity limiter.

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 through the annulus between the two conical sections. Because this water is jetted in a partially reversed direction into water flowing upward in the annulus, a severe turbulence is created, thereby slowing the descent of the control rod assembly to less than 5 ft/sec at 70 degrees F.[7]

3.4.5.2 Control Rod Drive Mechanisms (Figures 3.4-3, 3.4-5, 3.4-6, and 3.4-9)

The CRD mechanism (drive), used for positioning the control rod in the reactor core, is a double-acting, mechanically latched, hydraulic cylinder using water from the condensate storage system 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 utilization of the water in the reactor as a neutron shield giving the least possible neutron exposure to the drive components. The use of reactor water from the condensate storage system as the operating fluid eliminates the need for special hydraulic fluid. Drives utilize simple piston seals whose leakage does not contaminate the reactor vessel and helps cool the drive mechanisms.

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 six inches of stroke over the twelve foot 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 prevents accidental separation of a control rod from its drive.

Each drive positions its control rod in 6-inch 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 or withdraw travel limit is reached.

An alarm annunciates when withdraw overtravel limit on the drive is reached.

Normally, the control rod seating 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 allows the coupling to be checked.

Individual rod position indicators are grouped together on the control panel in one display and correspond to the relative rod location in the core. Each rod indicator gives continuous rod position indication in digital form. A separate, smaller display is located just below the large display on the vertical part of the bench board. 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 LPRM strings. Rod groups at the periphery of the core may have less than four rods. The small rod display shows the positions in digital form of the rods in the group to which the selected rod belongs. A white light indicates which of the four rods is the one selected for movement.

3.4.5.2.1 Components

Figure 3.4-5 illustrates the principles of operation of a drive. Figures 3.4-6, 3.4-9 and 3.4-9a illustrate the drive in more detail. Currently BWR/6 drives (Figure 3.4-9a) obtained from Hartsville Nuclear Plant and modified by General Electric are acceptable replacements for BWR/4 drives (Figures 3.4-6 and 3.4-9). BWR/6 drives are currently in use at Browns Ferry Nuclear Plant. Throughout the following sections, details for the BWR/4 drives are being maintained for historical purposes.

Following is a description of the main components of the drive and their functions.

Drive Piston and Index Tube

The drive piston is mounted at the lower end of the index tube which 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 and outside 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 square inches versus 4.0 square inches for up-travel 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 (XM-19 for BWR/6 drives). Any index tubes which are found to need replacing during normal CRD maintenance are replaced by index tubes of identical design made of Grade XM-19 stainless steel. This tube has circumferential locking grooves spaced every 6 inches along the outer surface. These grooves transmit the weight of the control rod to the collet assembly.

Collet Assembly

The collet assembly serves as the index tube locking mechanism. It is located in the upper part of the drive unit. The collet assembly prevents the index tube from accidentally 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 assembly), and the collet piston seals.

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

The collet piston is normally held in the latched position by a return spring force of approximately 150 pounds. 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 of 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.

As 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 combined with drive seals and bushings 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 on the BWR/4 design. In the BWR/6 design a buffer piston is included between the drive piston and the stop piston. This isolates the higher pressures from the drive piston seals during the deceleration phase of the scram stroke. 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 are 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 of Alnico S carried 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 inches of travel. Duplicate switches are provided for the full-in and full-out positions. One additional switch (an overtravel switch) is located at a position below the normal full-out position. Because the limit of 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. A thermocouple is located in each position indicator to indicate drive mechanism temperature in the control room. This satisfies safety design basis 4.f.

Cylinder, Tube, and Flange Assembly

The cylinder, tube, and flange assembly consists of an inner cylinder, outer tube, and a flange. 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.

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 these 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 is so situated as to direct 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 at this point. A cooling water orifice, adjacent to the insert port of the flange, permits cooling water flow from the CRD through the annulus formed by the CRD outer tube and the thermal sleeve in the CRD housing.

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

Coupling Spud, Plug, and Unlocking Tube

The upper end of the index tube is threaded to receive a coupling spud. The coupling (Figure 3.4-3) 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 (approximately 250 pounds) 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. However, with the lock plug in place, a force in excess of 50,000 pounds 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 pounds by a rod extending up through the center of the control rod to an unlocking handle located above the control rod velocity limiter. The control rod, with the lock plug raised, can then be separated from the drive. The lock plug may also be pushed up from below to uncouple a drive without removing the reactor pressure vessel head for access to change a CRD drive or perform maintenance or inspections. 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 4.e.

3.4.5.2.2 Materials of Construction

Factors determining the choice of materials are listed below:

a. 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. Index tubes for BWR/6 drives and replacement index tubes are made of XM-19.

b. The coupling spud is made of Inconel 750 which is aged to produce maximum physical strength and also provide the required corrosion resistance. As misalignment tends to produce a chafing in the semispherical contact area, the entire part is protected by a thin vapor-deposited chromium plating (Electrolyzing). This plating also serves to prevent galling of the threads attaching the coupling spud to the index tube.

c. Inconel 750 is used for the collet fingers, which must 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.

d. For BWR/4 CRDs, Graphitar 14 is selected for seals and bushings on the drive piston and stop piston. The material is inert and has a low friction coefficient when water lubricated.

   For BWR/6 CRDs, a composite material of nickel, chrome, graphite and resin is selected for seals and bushings on the drive piston and stop piston. The material reduces the leakage due to excessive wear or premature breakage. The drive is supplied with cooling water to normally hold temperatures below 250 degree F. The CRD high temperature alarm set point is 350 degrees F based on the BWR/6 seals and bushings being more temperature resistant than the BWR/4 seals.

   The Graphitar is relatively soft, which is advantageous when an occasional particle of foreign matter reaches a seal. The resulting scratches in the seal reduce sealing efficiency until worn smooth, but the drive design can tolerate considerable water leakage past the seals into the reactor vessel. These seals determine the service life of the CRDM.

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

a. Seals and bushings on the drive piston and stop piston are Graphitar 14. For later model BWR/6 CRDs, seals and bushings on the drive piston and stop piston are a composite material of nickel, chrome, graphite and resin.

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

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

d. Elastometric 0-ring seals are ethylene propylene.

e. Collet piston rings are Haynes 25 alloy.

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

g. Nitriding by a proprietary New Malcomizing process, Electrolyzing (a vapor deposition of chromium), and chromium plating are used in certain areas where resistance to abrasion is necessary.

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

i. Replacement index tubes and piston tubes are made of grade XM-19 stainless steel.

j. BWR/6 drives also use XM-19 for the index tubes and piston tubes.

k. The buffer assembly for BWR/6 CRDs consists of the stop piston, buffer piston, seal ring, nut, locking cup and the buffer shaft which is secured to the top of the piston tube assembly. The materials used to fabricate these components are Inconel X-750, Inconel-600, Armco 17-4PH and Haynes 25.

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 (Figures 3.4-8a Sheets 1, 2, 3, 4 and 5, 3.4-8b, 3.4-8c, 3.4-8d, 3.4-8e, 3.4-8f, 3.4-8g, and 3.4-8h)

The Control Rod Drive Hydraulic System supplies and controls the pressure and flow requirements to the drives.

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. The water discharged from a drive, during a normal control rod positioning operation, flows through its HCU and into the exhaust header. The discharged water then backflows through the other 184 CRD exhaust valves into the reactor vessel via the cooling water header.

3.4.5.3.1 CRD Hydraulic Supply and Discharge Subsystems (Figures 3.4-7, 3.4-8a Sheets 1, 2, 3, 4 and 5, 3.4-8c, 3.4-8d, 3.4-8e, 3.4-8f, 3.4-8g and 3.4-8h)

The CRD hydraulic supply and discharge subsystems control the pressure and flows required for the operation of the control rod drive mechanisms. These hydraulic requirements identified by the function they perform are as follows:

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

b. Drive pressure of about 260 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.

c. Cooling water to the drives is normally supplied at pressures greater than reactor pressure and at adequate flow rate to prevent drive component degradation due to elevated temperatures.

d. The exhaust water header is maintained at a pressure about 20 psig above vessel pressure to direct the flow of the water displaced during normal control operation of the drives back into the reactor vessel by backflowing through the other 184 CRD exhaust valves.

e. A scram discharge volume of approximately 3.3 gallons per drive to receive the water displaced from the drives during a scram is required. The scram discharge volume is vented and drained except during scram when it is isolated and filled with scram water until the scram signal is cleared and the scram reset. The scram discharge volume will reach reactor pressure following a scram.

The CRD hydraulic supply and discharge subsystems provide the required functions with the pumps, filters, valves, instrumentation, and piping shown in Figures 3.4-8a sheets 1, 2, 3, and 4, 3.4-8b, 3.4-8c, 3.4-8d, 3.4-8e, and 3.4-8f and described in the following paragraphs. Duplicate components are included, where necessary, to assure continuous system operation if an in-service component requires maintenance.

<u>Pumps</u>

One 100 percent capacity supply pump is provided for each unit to pressurize the system with water from the condensate storage system. One common 100 percent capacity spare pump is provided for Units 1 and 2. It can supply water to either control rod drive hydraulic systems. Unit 3's system is separate and has one spare 100 percent capacity pump. Change over (or selection) of the pumps is performed manually, either locally or from the main control room. Each pump is installed with a suction strainer and a discharge check valve to prevent bypassing flow backwards through the nonoperating pump.

A minimum flow bypass connection between the discharge of the pumps and the condensate storage tank prevents overheating of the pumps in the event that the pump discharge is inadvertently closed. In addition, a portion of the CRD flow is directed to the recirculation and the reactor water cleanup pump bearing seals for pump seal cooling and Reactor Vessel Level Instrumentation System (RVLIS) reference leg back filler. Pump discharge pressure is indicated locally at the inlet to the drive water filters by a pressure indicator.

<u>Filters</u>

Two parallel filters remove foreign material larger than 50 microns absolute (25 microns nominal) from the hydraulic supply subsystem water. The isolated filter can be drained, cleaned, and vented for reuse while the other is in service. A differential pressure indicator monitors 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.

Accumulator Charging Pressure

The accumulator charging pressure is maintained automatically by a flow-sensing element, controller, and an air-operated flow control valve. During normal operation, the accumulator charging pressure is established upstream from the flow control valve by the restriction of the flow control valve. During scram, the flow-sensing system upstream of the accumulator charging header detects high flow in the charging header and closes the flow control valve. The flow control valve is closed so that the proper flow to recharge the accumulators is diverted from the hydraulic supply header to the accumulator charging header. The parallel spare valve is provided with isolation valves to permit maintenance of the noncontrolling valve.

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 established through the flow control valve is the sum of the maximum water required to cool all the drives and that amount of water needed to provide a stable hydraulic system for insertion and withdrawal of the mechanism.

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) and then goes into the line downstream from the cooling pressure control valve. The two solenoid-operated stabilizing valves also have identical, backup function, solenoid-operated stabilizing valves which are selectable from the main control room in the event that the normal path valves become inoperative or require online maintenance. One stabilizing valve passes flow equal to the nominal 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 passes the flow from the drive water pressure control valve through the control rod drives and into the vessel. At normal flow rates, the cooling water header pressure will be approximately 10 to 15 psi above reactor vessel pressure.

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, the life of their seals is shortened by exposure to reactor temperatures.

Exhaust Water Header

The exhaust water header takes water discharged during a normal control rod positioning operation and directs it through the other CRD exhaust valves into the reactor vessel. If necessary, the exhaust water may be directed into the reactor vessel via the RWCU system by opening a normally closed valve.

Scram Discharge System

The scram discharge system is used to contain the reactor vessel water from all the drives during a scram. This system is provided in the two scram discharge volumes (SDVs) which each drain to a scram discharge instrument volume (SDIV). Water level monitors on the SDIVs provide an alarm if water is retained in the system. During normal plant operation, the volumes are empty with their drain and vent valves open.

Upon receipt of a scram signal, the drain and vent valves close. Position indicator switches on the drain and vent valves indicate valve position by lights in the control room.

During a scram, the scram discharge volume partially fills with water which is discharged from above the drive pistons. While scrammed, the control rod drive seal leakage continues to flow to the discharge volume until the discharge volume pressure equals reactor vessel pressure. There is a check valve in each HCU which prevents reverse flow from the scram discharge volume to the drive. When the initial scram signal is cleared from the reactor protection system, the scram discharge volume scram signal may be overridden with the override switch and the scram discharge system drained. A control system interlock will not allow the drives to be withdrawn until the discharge system is emptied to a safe level.

A test pilot valve allows the discharge volume valves to 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. As an alternative to the test pilot valve in Units 1, 2 and 3, a key-lock test switch is provided to de-energize each of the SDV Drain and Vent Pilot Solenoid valves. This will allow stroke time testing of SDV Drain and Vent Valves to be performed.

Six level switches on the scram discharge instrument 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 reactor protection system trip system) initiate a scram to shut down the reactor while sufficient free volume is still present to receive the scram discharge. After a scram, these same level switches must be cleared by draining the scram discharge volume before reactor operation can be resumed.

The piping and equipment pressure parts in the CRD hydraulic supply and discharge subsystems are designed in accordance with USAS B 31.1.0.

3.4.5.3.2 Hydraulic Control Units (Figures 3.4-7, 3.4-8a Sheets 2 and 4, 3.4-8c, 3.4-8e, and 3.4-11)

Each hydraulic control unit controls a single CRD. The basic components in each hydraulic control unit are manual, pneumatic and electrically operated valves, an accumulator, filters, related piping, and electrical connections.

Each hydraulic control unit furnishes pressurized water, upon signal, to a control rod drive. The drive then 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 hydraulic control unit and their functions are as follows:

Insert Drive Valve

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

Insert Exhaust Valve

The insert exhaust valve is a solenoid-operated valve which opens on an insert or withdrawal 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 drive 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 Reactor Protection System Trip System. Two scram pilot valves control both the scram inlet valve and the scram exhaust valve. The scram pilot valves are identical, 3-way, solenoid-operated, normally energized valves. On loss of electrical signal to the pilot valves, the pressure ports are closed and the exhaust ports are opened on both pilot valves. The pilot valves are arranged as shown in Figures 3.4-7, 3.4-8a sheets 2 and 4, 3.4-8c, and 3.4-8e so that the trip system signal must be removed from both valves before air pressure is discharged from the scram valve operators.

Scram Inlet Valve

The scram inlet valve is opened to supply scram water pressure to the bottom of the drive piston. The scram inlet valve is a globe 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 pounds. The valve opening time is approximately 0.1 second from start to full open.

Scram Exhaust Valve

The scram exhaust valve opens slightly before the scram inlet valve, exhausting water from above the drive piston during a scram. Quicker opening times are achieved because of a greater spring force in the valve operator. Otherwise this valve is similar to the scram inlet valve.

The scram inlet and scram exhaust valves have a position indicator switch which energizes a light in the control room as soon as both valves open.

Scram Accumulator

The scram accumulator stores sufficient energy to insert a control rod to the fully inserted position during a scram independent of any other source of energy. The accumulator consists of a water volume pressurized by a volume of 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 retains the water in the accumulator in the event supply pressure is lost.

During normal plant operation, the accumulator piston has a differential pressure across it which is equal to the difference in the charging water pressure and the nitrogen cylinder pressure.

Loss of nitrogen causes a decrease in the nitrogen pressure which 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 4.d.

3.4.5.4 Control Rod Drive System Operation

The control rod drive system performs three operational functions: rod insertion, row withdrawal, and scram. The functions are described below.

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 allows water from above the drive piston to discharge to the exhaust header.

As is illustrated in Figure 3.4-6, 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 approximately 4 gpm for a shim speed (nonscram operation) of 3 inches per second. 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 drive water 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 1000 pounds.

Rod Withdrawal

Drive withdrawal is, by design, more involved. First the collet fingers (latch) must be raised to reach the unlocked position as in Figure 3.4-5. The notches in the index tube and the collet fingers are shaped so that the downward force on the index tube holds the collet fingers in place. The index tube must be lifted before the collet fingers can be released. This is done by opening the drive insert valves (in the manner described in the preceding paragraph) for approximately 1 second using an automatic sequence timer. The withdraw valves are then opened (by the sequence timer mechanism), 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 raises, the collet fingers are cammed outward, away from the index tube, by the guide cap.

The pressure required to release the latch is set and maintained high enough to overcome the force of the latch return spring plus the force of reactor pressure opposing movement of the collet piston. When this occurs, the index tube is unlatched and free to move in the withdrawal direction. Water displaced by the drive piston flows out through the withdrawal speed control valve which is set to give the control rod a shim withdrawal of approximately 3 inches per second. The entire valving sequence is automatically controlled and is initiated by a single operation of the rod withdraw 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 1400 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 any possible friction. The characteristics of the hydraulic system are such that, after the initial acceleration is achieved (approximately 30 milliseconds after start of motion), the drive continues at a fairly constant velocity of approximately 5 feet per second. 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. In the BWR/6 design a buffer piston is included between the drive piston and the stop piston. This isolates the higher pressures from the drive piston seals during the deceleration phase of the scram stroke.

Each drive requires about 2.5 gallons of water during the scram stroke. There is adequate water capacity in each drive's 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 1000 psig and the scram discharge volume at atmospheric pressure, the scram force without an accumulator is over 1000 pounds.

NOTCH POSITION	SCRAM TIMES (a)(b) (seconds) REACTOR STEAM DOME PRESSURE $\geq$ 800 psig
46	0.45
36	1.08
26	1.84
06	3.36

(a) Maximum scram time from fully withdrawn position, based on de-energization of scram pilot valve solenoids at time zero.

(b) Scram times as a function of reactor steam dome pressure, when <800 psig are within established limits.

3.4.6 Safety Evaluation

3.4.6.1 Evaluation of Control Rods

As discussed above, it has been determined 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, as required by safety design basis 1.c. This velocity is evaluated by the rod drop accident analysis in Section 14, "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 control rod drive 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 Control Rod Drive System 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, "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 control rod, the weight of the housing below the attachment weld to the vessel nozzle, and reactor pressure acting on the 6-inch 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 above, 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 and by the amount the support structure deflects under load. In the current design, maximum deflection is approximately 3 inches. If the collet were to remain latched, no further control rod ejection would occur.[8] The housing would not drop far enough to clear the vessel penetration. Reactor water would leak through the 0.06-inch diametral clearance between the housing o.d. and the vessel penetration i.d. 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 control rod drive housing support and calculations indicate that the steady state rod withdrawal velocity would be 0.3 ft/sec. During withdraw, 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. Failure is more likely to occur after another basic failure wherein the drive housing, or housing flange, separates from the reactor vessel. Failure of the housing, however, does not necessarily lead directly to failure of the hydraulic lines.

If the pressure-under 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.

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 2 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) Pressure-Over Line Breaks

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 1000 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 above. 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, high drive temperature printed out and alarmed by a recorder in the control room, and by operation of the drywell sump pump.

c. All Drive Flange Bolts Fail in Tension

Each control rod drive is bolted by eight cap screws to a flange at the bottom of a drive housing which is welded to the reactor vessel. Bolts are made of AISI-4140 steel. Replacement cap screws are made of AISI-4340 or SA-540 B23 steel which is more resistant to stress corrosion cracking.

In the event that progressive or simultaneous failure of all 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 1435 pounds 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 1650 pounds return force, would latch, stopping rod withdrawal.

d. Weld Joining Flange to Housing Fails in Tension

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 a forged type 304 stainless steel and the housing material is seamless type 304 stainless steel pipe. 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 1250 psig and the design temperature is 575F. 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 inches. Downward drive movement would be small; therefore, most of the drive would remain inside the housing. The pressure-under and pressure-over lines are flexible enough to withstand the small downward displacement and remain attached to the flange. Reactor water would follow the same leakage path described in c, above, except that the exit to the atmosphere would be through the gap between the lower end of the housing and the top of the flange. Water would flash to steam in the annulus surrounding the drive. The leakage rate would be approximately 840 gpm.

If the basic 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 pounds. 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. <u>Housing Wall Ruptures</u>

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.

If the housing wall rupture described above were to occur, reactor water would flash to steam and leak to the atmosphere at approximately 1030 gpm through the hole in the housing. Choke flow conditions described in c, above would exist. In this case, however, the leakage flow would be greater because the flow resistance is less; that is, the leaking water and 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 basic 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 1030 gpm reduces the pressure in the annulus outside the drive 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-inch-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-inch-diameter plug 0.250 inch 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 blow-out 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 were 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 basic 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 pounds 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. Pressure Regulator and Bypass Valves Fail Closed (Reactor Pressure O psig)

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 260 psig in the header just upstream from the valve. The other is a hand-operated valve that is normally closed but that can be valved in and 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 cooling water header. The two solenoid-operated stabilizing valves also have identical, backup function, solenoid-operated stabilizing valves which are selectable from the main control room in the event that the normal path valves become inoperative or require online maintenance. 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 1700 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 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 1670-psi pressure differential across the collet piston. Flow would be upward past the velocity limiter piston, but retarding force would be negligible.

h. <u>Ball Check Valve Fails to Close Off Passage to Vessel Ports</u>

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 job 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 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 pounds).

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 Control Rod Drive System prevents rod withdrawal as required by safety design basis 4.a. It is shown above 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 CRD System, such as the following:

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

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

c. The Reactor Protection System and HCU's are designed so that the scram signal and mode of operation override all others.

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

e. 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 basis 4.b and 4.c.

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 5000 scrams and approximately 100,000 latching cycles during 5000 hours of exposure to simulated operating conditions. These tests have demonstrated the following:

a. That the drive withstands the forces, pressures, and temperatures imposed without difficulty.

b. That wear, abrasion, and corrosion of the nitrided type 304 stainless parts are negligible. That mechanical performance of the nitrided surface is superior to materials used in earlier operating reactors.

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

d. That usable seal lifetimes greater than 1000 scram cycles may be expected.

3.4.7.2 Factory Quality Control Tests

Quality control of welding, heat treatment, dimensional tolerances, material verification, etc., was 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, control rod drive mechanisms, and hydraulic control units were as follows:

Control Rod Absorber Tube Tests

a. The tubing and end plug material integrity was verified by ultrasonic inspection.

b. Boron content of the Boron-10 fraction of each lot of boron-carbide was verified.

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

CRD Mechanism Tests

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

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

c. All drive parts which could not be visually inspected for dirt were flushed with filtered water at high velocity. No significant foreign material was permissible in effluent water.

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

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

f. 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:

a. All hydraulic systems were hydrostatically tested in accordance with USAS B31.1.0.

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

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

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

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

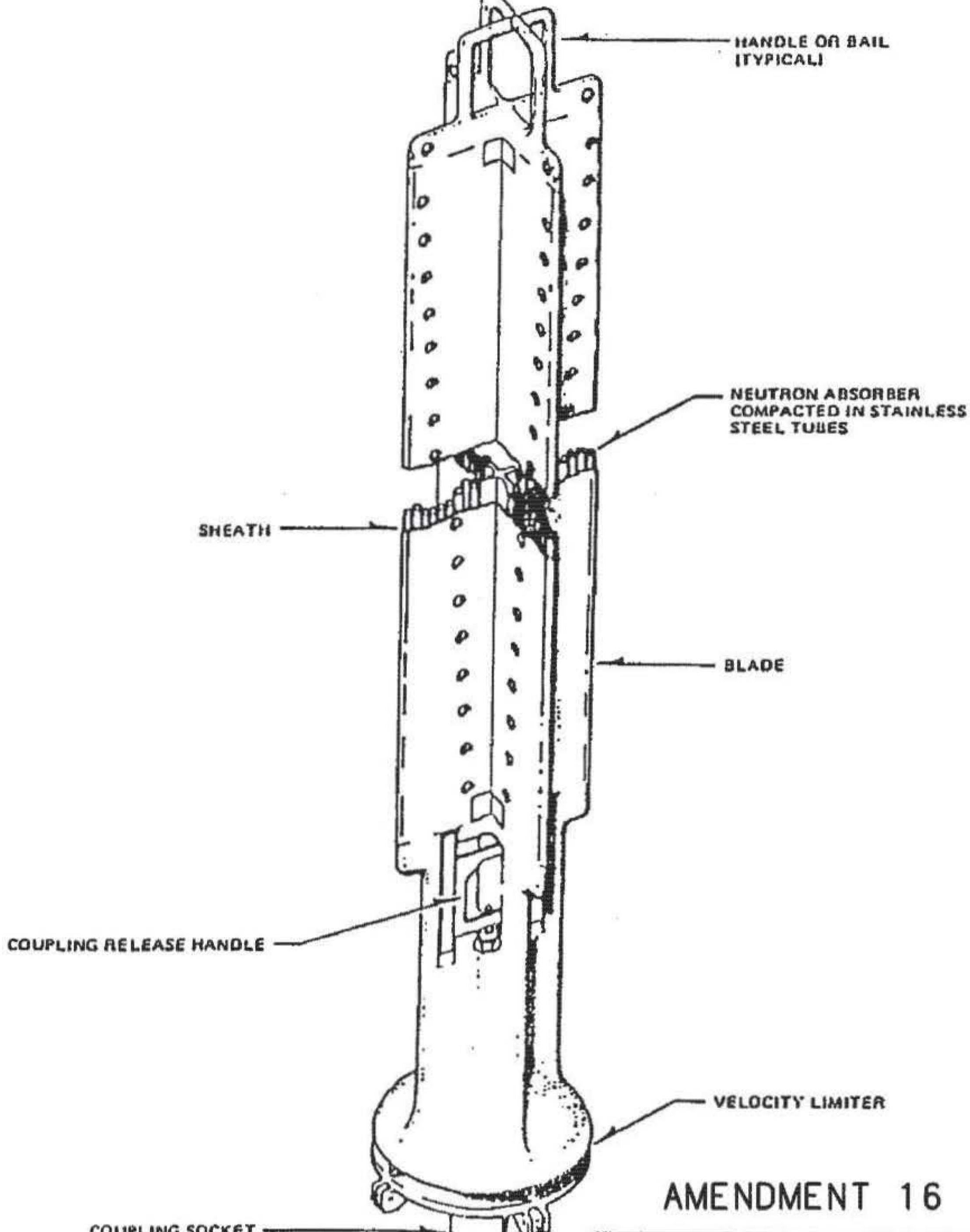
3.4.7.3 Operational Tests

After installation, all rods, hydraulic control units, and drive mechanisms were tested through their full range for operability. Details of the preoperational test are given in Subsection 13.4.

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

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

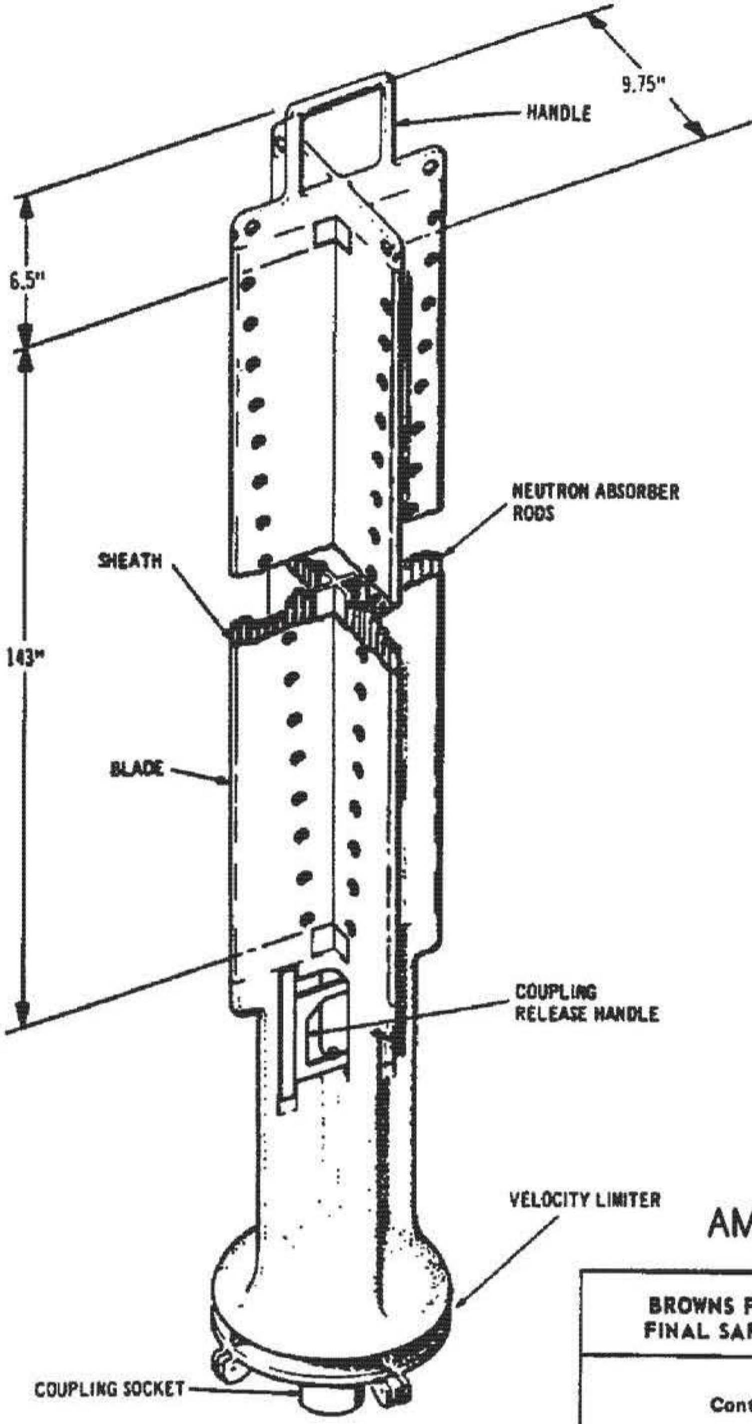
Hydraulic supply subsystem pressures can be observed from instrumentation in the control room. Scram accumulator pressures can be observed locally on the nitrogen pressure gages.



AMENDMENT 16

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

Control Rod — Isometric
FIGURE 3.4-1



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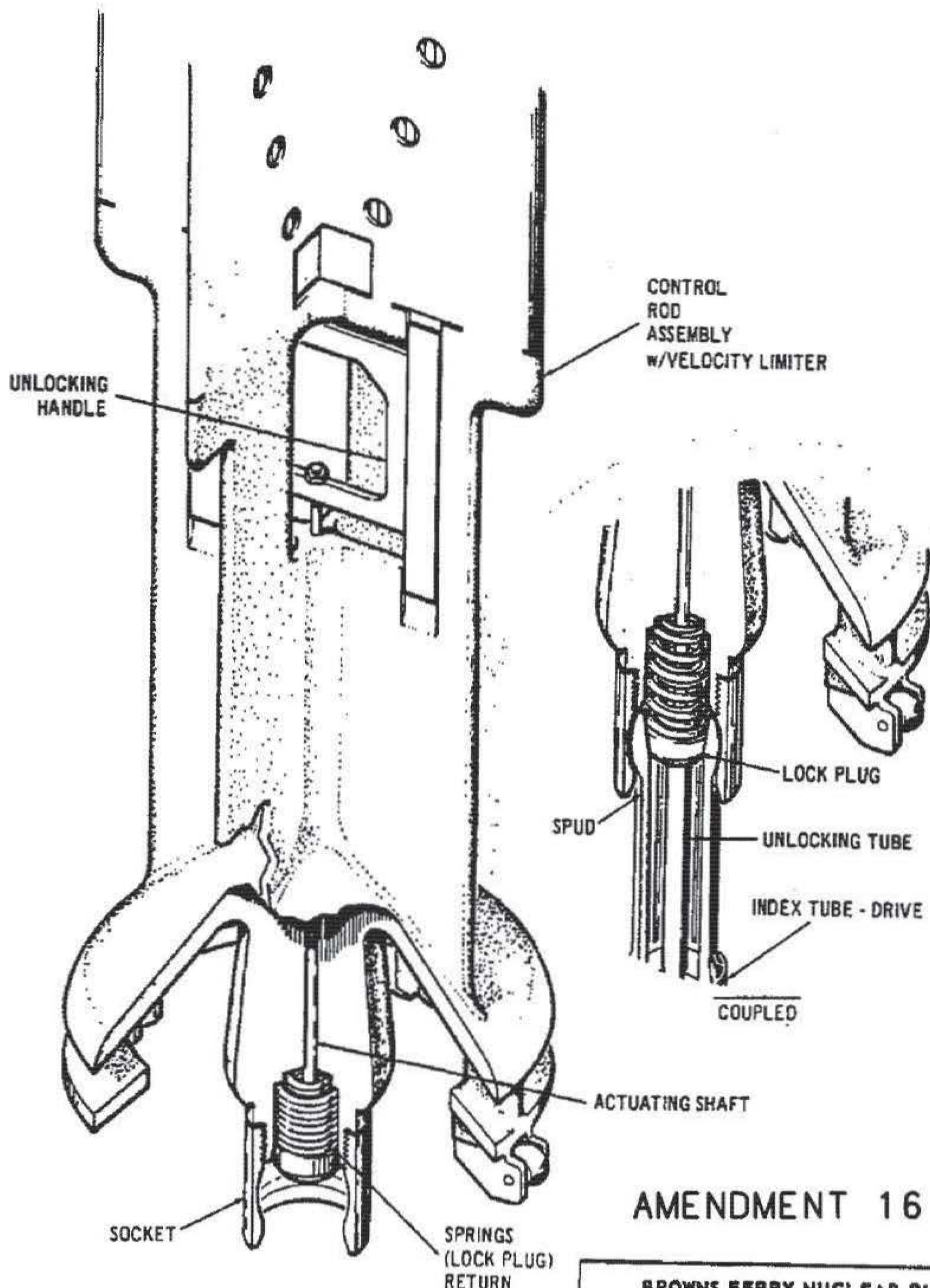
BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

Control Rod – Isometric
FIGURE 3.4-2



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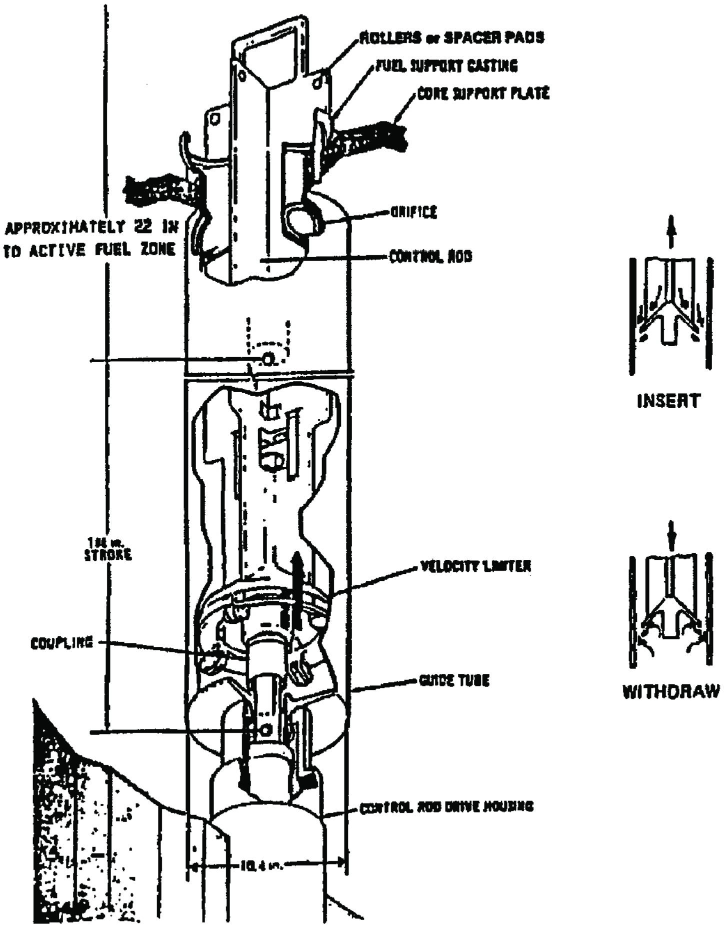
BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

Control Rod to Control Rod Drive Coupling – Isometric

FIGURE 3.4-3



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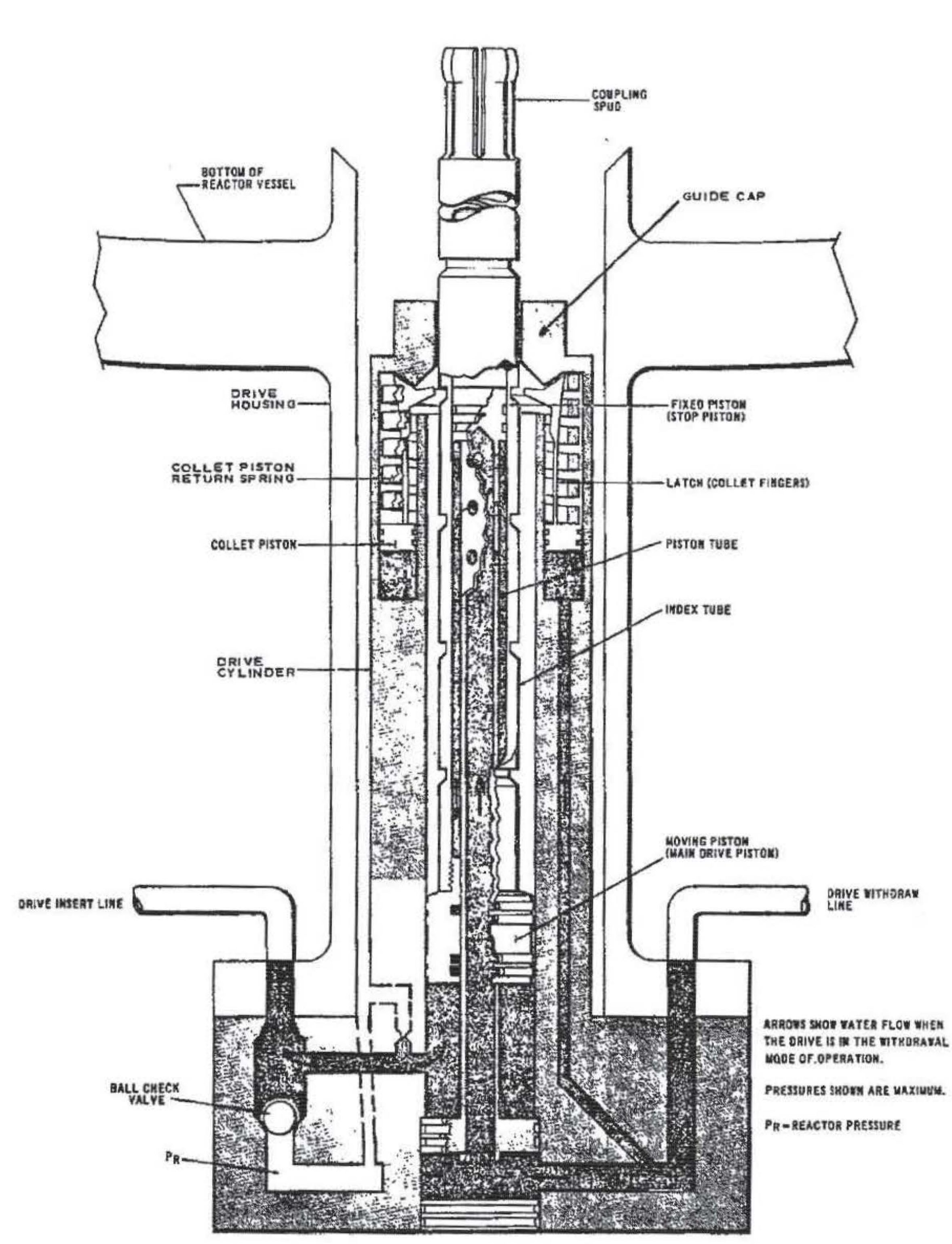
BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

CONTROL ROD VOLOCITY LIMITOR-
ISOMETRIC

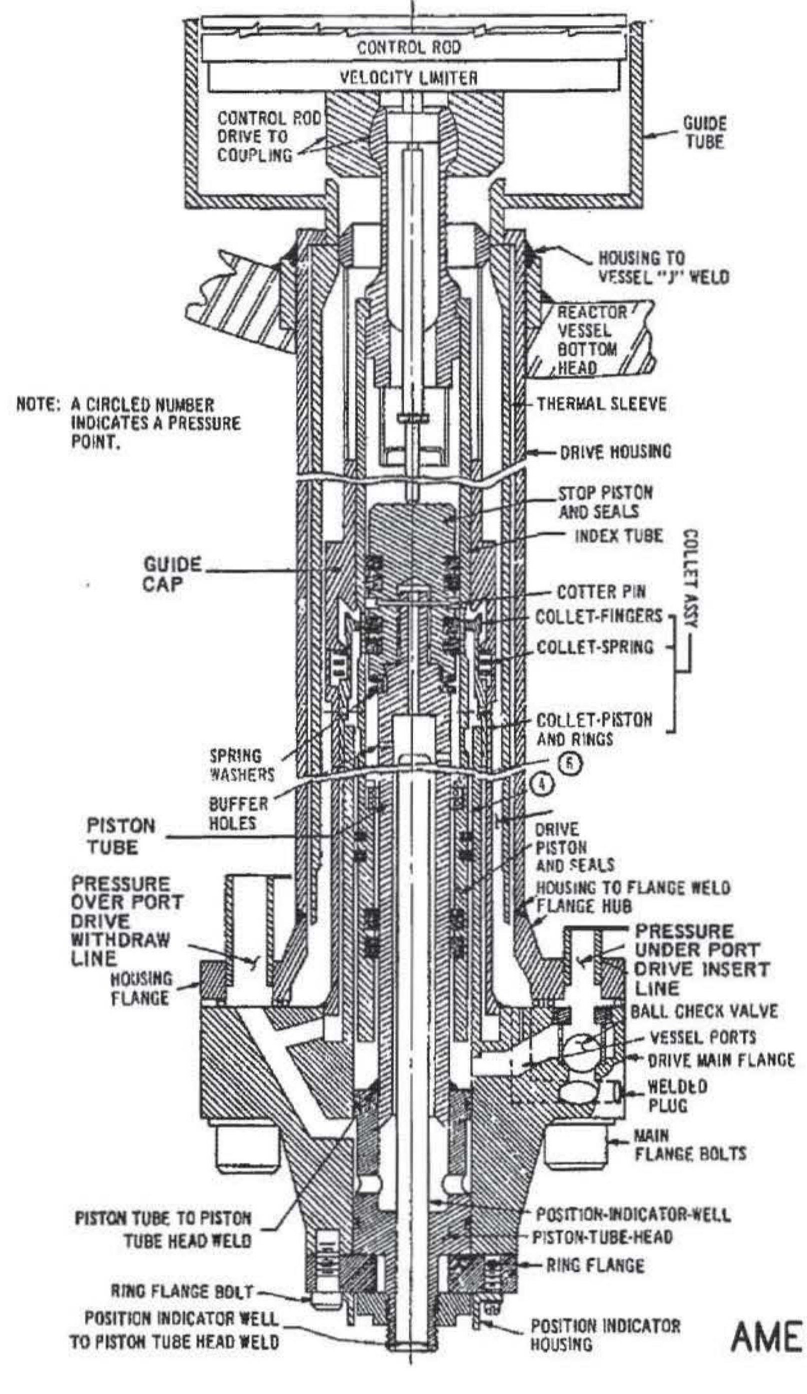
FIGURE 3.4-4



BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

Control Rod Drive, Simplified Component Illustration
FIGURE 3.4-5

AMENDMENT 16



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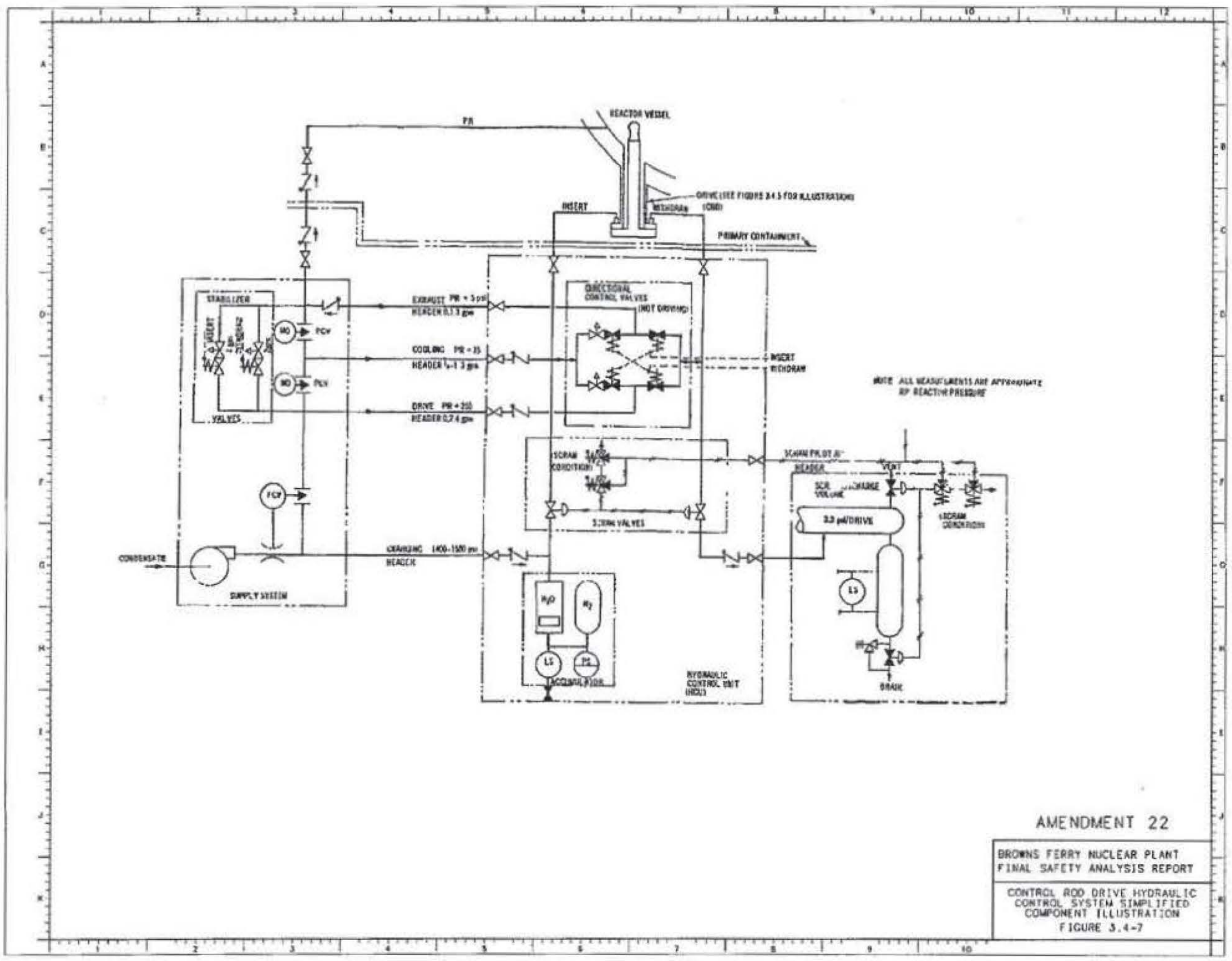
BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

Control Rod Drive, Schematic Diagram
FIGURE 3.4-6



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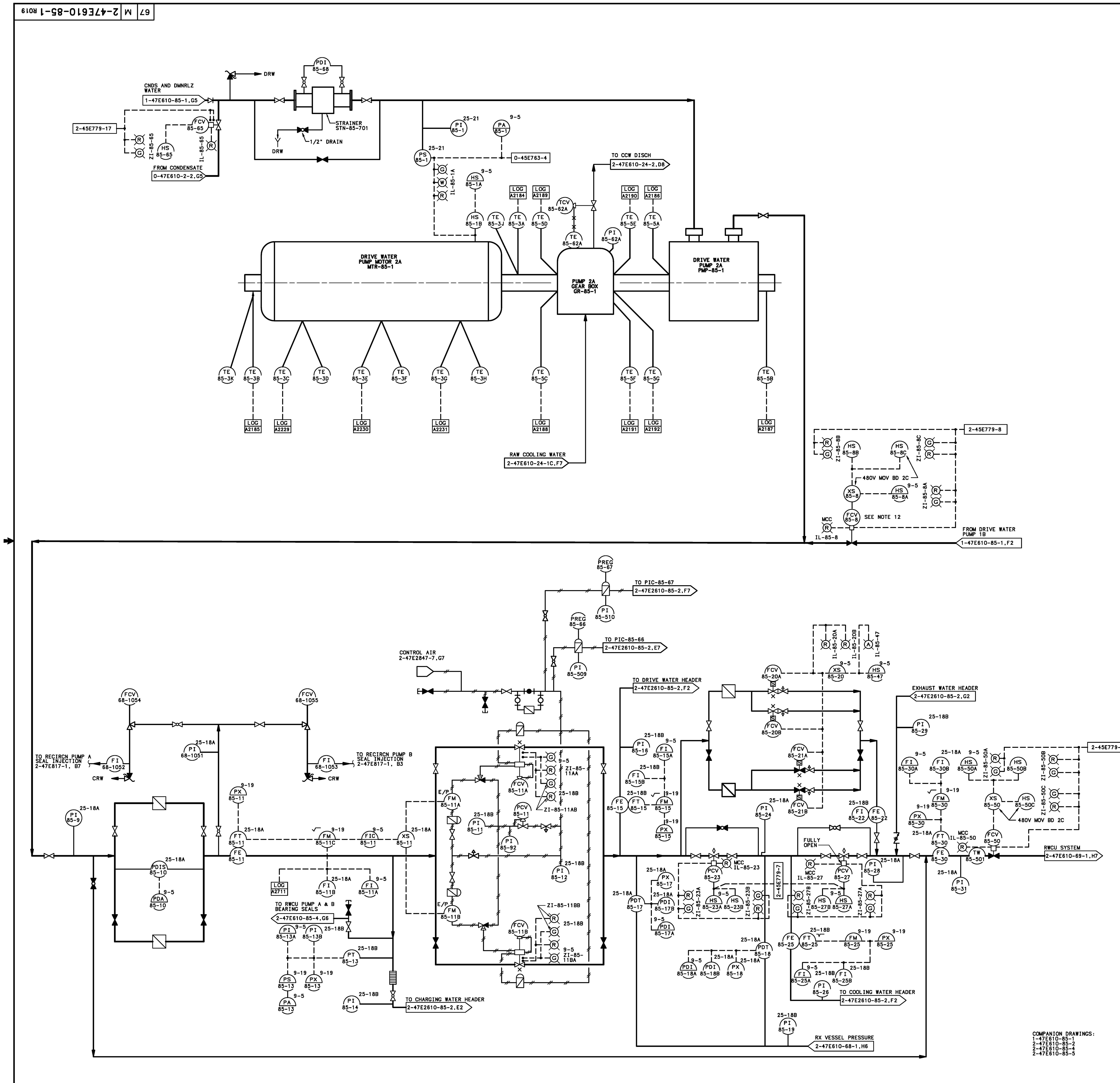
BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

CONTROL ROD DRIVE HYDRAULIC
CONTROL SYSTEM SIMPLIFIED
COMPONENT ILLUSTRATION
FIGURE 3.4-7



NOTES:

1. THE CONTROL ROD DRIVE (CRD) SYSTEM IS DESIGNED TO PROVIDE THE REACTOR WITH POSITIVE REACTIVITY CONTROL WHILE MAINTAINING A CONSTANT QUANTITY OF CONDENSATE TO THE REACTOR VESSEL IN THE FORM OF DRIVE SEAL LEAKAGE, DRIVE COOLING, AND DIRECT FLOW TO THE REACTOR. THE PUMP SUCTION IS FROM THE CONDENSATE STORAGE SYSTEM AND IS RAISED TO A PRESSURE ABOVE THE REACTOR OPERATING PRESSURE BY THE DRIVE WATER PUMPS. THE PRESSURE IS THEN REDUCED TO THE THREE PRESSURES NEEDED FOR DRIVING, CHARGING, AND COOLING.
2. UNITS 1 AND 2 SHARE THE SPARE DRIVE WATER PUMP (1B) WHILE UNIT 3 HAS ITS OWN SPARE. EACH PUMP IS CAPABLE OF SUPPLYING 100 GPM AT 1500 PSIG.
3. THE CRD HYDRAULIC SYSTEM PROVIDES CRD OPERATING FLUID, CRD COOLANT, ACCUMULATOR CHARGING AND EXHAUST WATER TO THE REACTOR.
4. THREE STAGES OF WATER PRESSURE ARE DERIVED BY THREE SEPARATE REGULATING SYSTEMS. THE SCRAM ACCUMULATOR CHARGING WATER IS AUTOMATICALLY REGULATED TO 1400 PSIG. THE FLOW CONTROL SYSTEM USING FCV-85-11A OR B AND FE-85-11 MAINTAINS 100 GPM TO THE SYSTEM. CRD DRIVE WATER IS MAINTAINED AT REACTOR PRESS. (PR) + 250 PSIG BY CONTROLLING PCV-85-23 AND THE AUTOMATIC CLOSING OF FCV-85-20A OR 21A DURING ROD MOVEMENT. THE COOLING WATER HEADER PRESSURE IS REGULATED TO PR + 20 PSIG BY POSITIONING PCV-85-27. ONE HUNDRED GALLONS PER MINUTE (100 GPM) FLOWS INTO THE REACTOR WHETHER OR NOT THE CONTROL RODS ARE MOVING OR AT REST.
5. EACH CRD IS EQUIPPED WITH A SEPARATE SCRAM ACCUMULATOR WHICH IS CHARGED WITH NITROGEN. THE NITROGEN CHARGING HEADERS ARE MAINTAINED AT 1400 PSIG. ACCUMULATOR CHARGE PRESSURES ARE MONITORED BY LOCAL PRESSURE GAUGES PI-85-34 AND PS-85-34 WHICH ALARM LOW PRESSURE. ACCUMULATOR DIAPHRAGM INTEGRITY IS ALARMED WITH THE USE OF LS-85-34.
6. CRD POSITIONING IS ACCOMPLISHED BY ELECTRICAL CONTROL OF THE INSERT VALVES FCV-85-40A AND D OR THE WITHDRAW VALVES FCV-85-40B AND C.
7. EACH CONTROL ROD, AS IT TRAVELS UP (INSERT) OR DOWN (WITHDRAW) PASSES A NUMBER OF SWITCHES. THE TOP TWO POSITION SWITCHES ZS-85-41A AND B ARE CALLED "OVER TRAVEL" AND THE BOTTOM TWO POSITION SWITCHES ARE "WITHDRAW BACKSEAT" ZS-85-42 AND "WITHDRAW DISCONNECT" ZS-85-43. THERE ARE 48 SWITCHES BETWEEN THESE LIMITS ZS-85-44 (-1 THRU -48). THEY ARE DIVIDED INTO TWO SETS. ODD NUMBERED SWITCHES (EXAMPLE FOR CRD NUMBER 1; ZS-85-44-1, -3, -5 ETC.), ARE THE ROD DRIFTING SWITCHES AND THE EVEN NUMBERED SWITCHES ARE THE CRD LOCKED POSITION SWITCHES. THE ABOVE SWITCHES ARE ACTUATED BY A MAGNET IN THE POSITION PROBE. THE ROD POSITION IS DISPLAYED IN THE MAIN CONTROL ROOM ON PANEL 9-5 REACTOR PANEL.
8. INSTRUMENTATION ASSOCIATED WITH BACKUP CONTROLS IS LOCATED AS SHOWN ON THE DIAGRAM. IF A LOCATION IS NOT SHOWN THE DEVICES WILL BE LOCATED ON THE APPROPRIATE 4160V AC BOARD, 480V AC BOARD, OR 250V DC BOARD. (REF GE DESIGN SPEC. 22A1470)
9. THIS CONTROL LOOP HEADER IS PIPED ONLY TO CRD MODULES 26-55, 30-43, 30-07 AND 30-23, PI-85-71A AND B ARE ASSOCIATED ONLY WITH CRD MODULES 30-43 AND 30-23 RESPECTIVELY.
10. ALL COMPONENTS SHOWN HAVE UNIT 2- PREFIX, UNLESS OTHERWISE NOTED.
11. PARTIAL LOGIC FOR CONTROL ROD DRIVE SYSTEM IS SHOWN ON THE MECHANICAL LOGIC DIAGRAM LISTED IN REFERENCE DRAWINGS.
12. 2-FCV-85-8 MAY BE OPENED OR CLOSED TO SUPPORT PLANT OPERATIONS EXCEPT WHEN CRD PUMP 1B IS NEEDED TO SUPPLY WATER TO THE UNIT 1 CRD HYDRAULIC CONTROL SYSTEM WHERE 1-FCV-85-8 VALVE IS OPENED. THIS CONDITION REQUIRES THAT 2-FCV-85-8 BE CLOSED.

REFERENCE DRAWINGS:

0-45E763-4..............4160V SHUTDOWN AUX POWER SCHEMATIC DIAGRAM
2-45E763-6..............4160V SHUTDOWN AUX POWER SCHEMATIC DIAGRAM
2-45E779-7,-8,-17.......480V SHUTDOWN AUX POWER SCHEMATIC DIAGRAM
2-47W600-224............RPS-CRD CONTROL AIR PRESSURE PANELS
0-47W600-224A...........RPS-CRD CONTROL AIR PRESSURE PANELS
MEL.....................INSTRUMENT TABULATIONS-CRD
2-47E610-1-2............MECHANICAL CONTROL DIAGRAM-MAIN STEAM SYSTEM
0-47E610-2-2............MECHANICAL CONTROL DIAGRAM-CONDENSATE & DEMINERALIZED WATER STORAGE SYSTEM
2-47E610-68-1...........MECHANICAL CONTROL DIAGRAM-REACTOR WATER RECIRCN SYSTEM
2-47E610-69-1...........MECHANICAL CONTROL DIAGRAM-REACTOR WATER CLEANUP SYSTEM
2-47E2610-74-1..........MECHANICAL CONTROL DIAGRAM-RESIDUAL HEAT REMOVAL SYSTEM
2-47E610-85-1, -2, -4...CONTROL DIAGRAM-CRD HYDRAULIC SYSTEM
2-47E2610-85-2..........MECHANICAL CONTROL DIAGRAM-CRD HYDRAULIC SYSTEM
0-47E800-1..............FLOW DIAGRAM-GENERAL PLANT SYSTEMS
0-47E800-2..............MECHANICAL SYMBOLS & FLOW DIAGRAM DRAWING INDEX
2-47E810-1..............FLOW DIAGRAM-REACTOR WATER CLEANUP SYSTEM
0,1,2,3-47E820-SERIES...FLOW DIAGRAM-CRD
2-47E852-2..............FLOW DIAGRAM-DRAINAGE-REACTOR BUILDING
2-47E2847-1 & -7........FLOW DIAGRAM-CONTROL AIR

GE DRAWINGS:

2-791E201-2 & -3.......CRD HYDRAULIC INSTRUMENT CONTROL SYSTEM
2-730E321-14...........REACTOR MANUAL CONTROL SYSTEM
730E145................CRD HYDRAULIC CONTROL UNIT, PIPING AND INSTRUMENT DIAGRAM
729E499................CRD FUNCTIONAL CONTROL DIAGRAM

COMPANION DRAWINGS:
1-47E610-85-1
2-47E610-85-2
2-47E610-85-4
2-47E610-85-5

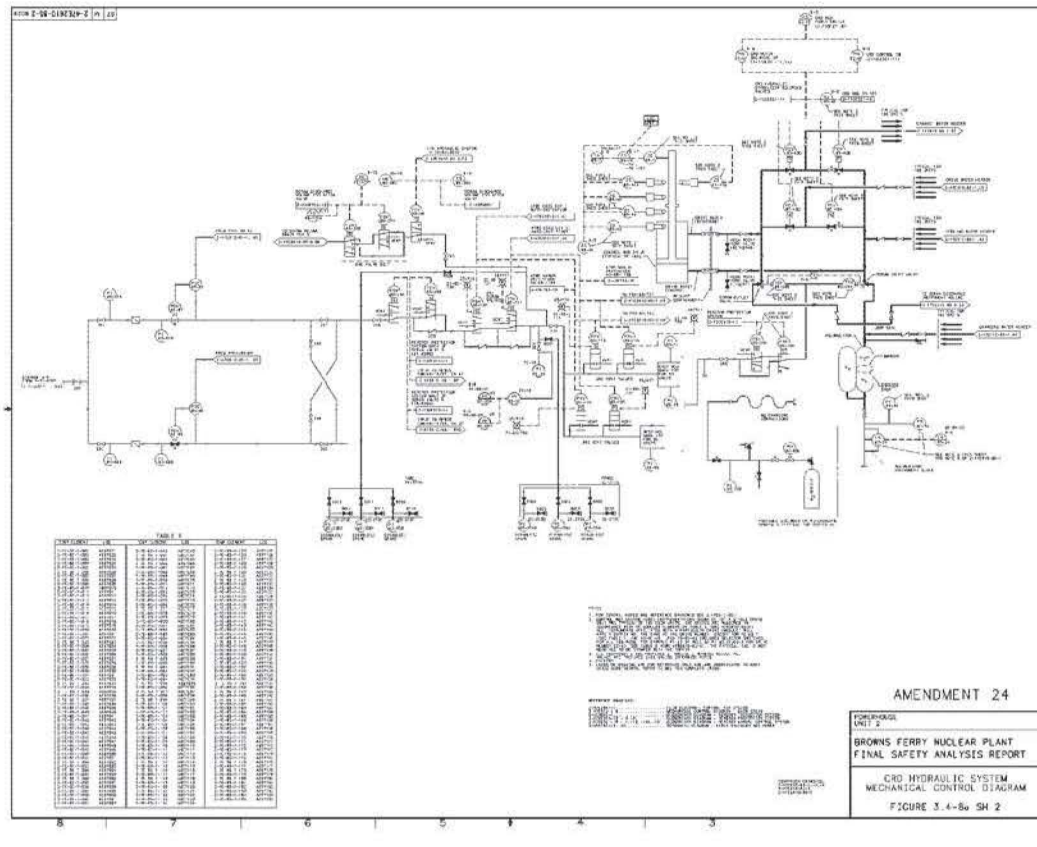
AMENDMENT 28

REACTOR BUILDING
UNIT 2

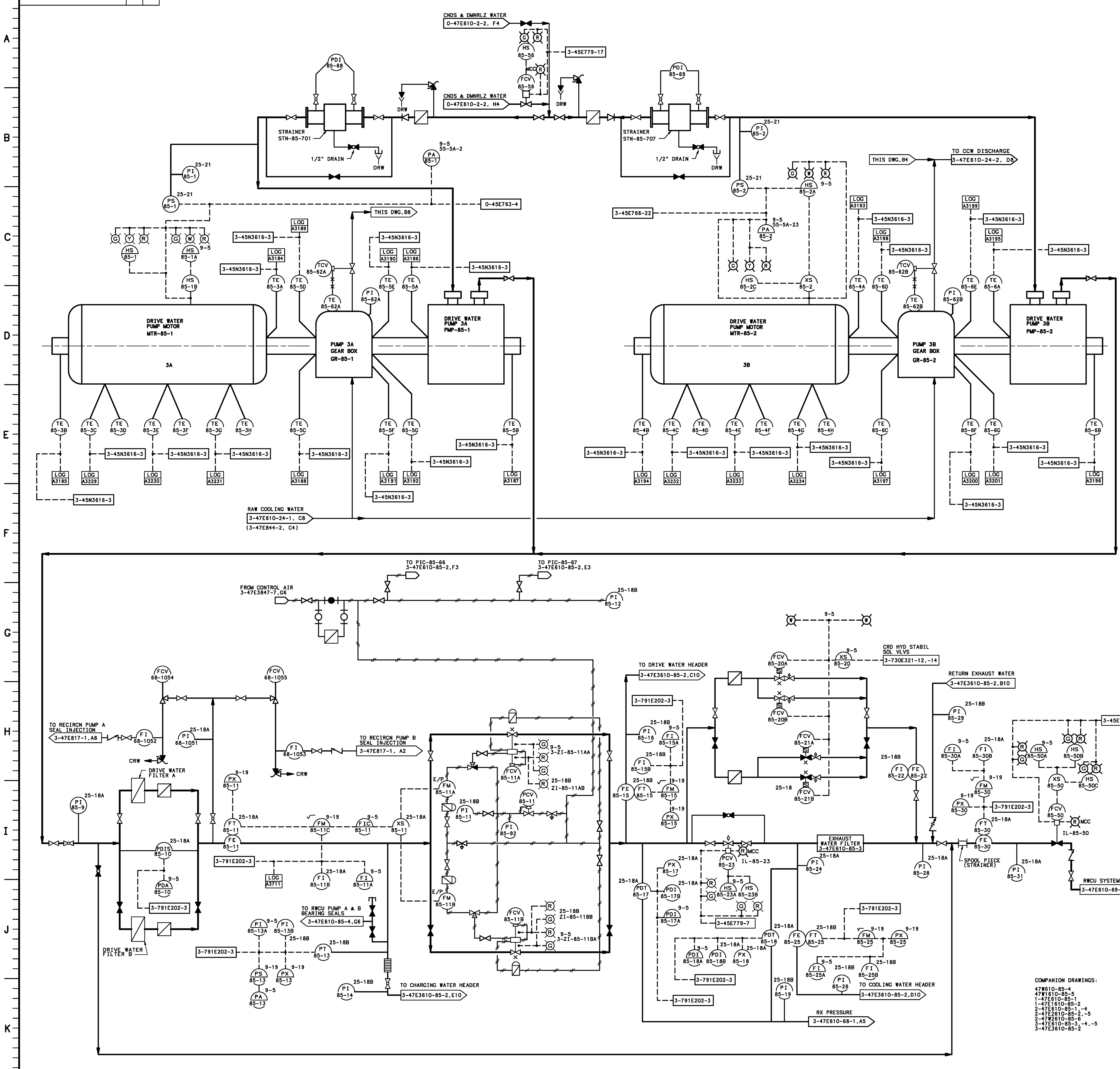
BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

CRD HYDRAULIC SYSTEM
MECHANICAL CONTROL DIAGRAM

FIGURE 3.4-8a SH 1


AMENDMENT 24
POWERHOUSE
UNIT 2
BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT
CRD HYDRAULIC SYSTEM
MECHANICAL CONTROL DIAGRAM
FIGURE 3.4-8a SH 2

3-47E610-85-1 R014 | M | 67



GENERAL NOTE:
THE CONTROL ROD DRIVE (CRD) SYSTEM IS DESIGNED TO PROVIDE THE REACTOR WITH POSITIVE REACTIVITY CONTROL WHILE MAINTAINING A CONSTANT QUANTITY OF CONDENSATE TO THE REACTOR VESSEL IN THE FORM OF DRIVE SEAL LEAKAGE DRIVE COOLING, AND FLOW TO THE REACTOR VIA RWCU RETURN LINE. THE PUMP SUCTION IS FROM THE CONDENSATE STORAGE SYSTEM AND IS RAISED TO A PRESSURE ABOVE THE REACTOR OPERATING PRESSURE BY THE DRIVE WATER PUMPS. THE PRESSURE IS THEN REDUCED TO THE THREE PRESSURES NEEDED FOR DRIVING, CHARGING AND COOLING.

NOTES:
1. HYDRAULIC CONTROL UNIT (HCU) DEVICES AND CONTROL ROD DRIVE UNIT (CRDU) DEVICES ARE SHOWN FOR A SINGLE UNIT AND ARE TYPICAL OF 185 SUCH UNITS. THE DRIVES ARE NUMBERED IN ACCORDANCE WITH TABLE 3 ON 3-730E321-4. UNIDS OF ALL COMPONENTS AFFILIATED WITH A PARTICULAR DRIVE WILL HAVE A SUFFIX THE SAME AS THE DRIVE COORDINATE. FOR EXAMPLE, THE UNID FOR FLOW CONTROL VALVE 39B ASSOCIATED WITH DRIVE 2-43 WOULD BE 3-FCV-85-39B/243, AND FOR CYLINDER 718 WOULD BE 3-CYL-85-718/243.
2. UNITS 1 AND 2 SHARE THE SPARE DRIVE WATER PUMP (1B) WHILE UNIT 3 HAS IT'S OWN SPARE. EACH PUMP IS CAPABLE OF SUPPLYING 100 GPM AT 1500 PSIG.
3. THE CRD HYDRAULIC SYSTEM PROVIDES CRD OPERATING FLUID, CRD COOLANT, ACCUMULATOR CHARGING AND EXHAUST WATER TO THE REACTOR.
4. THREE STAGES OF WATER PRESSURE ARE DERIVED BY TWO SEPARATE REGULATING SYSTEMS. THE SCRAM ACCUMULATOR CHARGING WATER IS AUTOMATICALLY REGULATED TO 1400 PSIG. THE FLOW CONTROL SYSTEM USING FCV-85-11A OR B AND FE-85-11 MAINTAINS APPROXIMATELY 65 GPM TO THE SYSTEM CRD DRIVE WATER IS MAINTAINED AT REACTOR PRESSURE. (Pr) + 250 TO 270 PSIG BY CONTROLLING PCV-85-23 AND THE AUTOMATIC CLOSING OF FCV-85-20A OR 21A DURING ROD MOVEMENT. THE COOLING WATER HEADER PRESSURE IS APPROXIMATELY PR + 20 PSIG. SIXTY-FIVE GALLONS PER MINUTE (65 GPM) FLOWS INTO THE REACTOR WHETHER OR NOT THE CONTROL RODS ARE MOVING OR AT REST.
5. EACH CRD IS EQUIPPED WITH A SEPARATE SCRAM ACCUMULATOR WHICH IS CHARGED WITH NITROGEN. ACCUMULATOR CHARGE PRESSURES ARE MONITORED BY LOCAL PRESSURE GAUGES PI-85-34/219 THRU 5843 AND PS-85-34/219 THRU 5843 WHICH ALARM ON LOW PRESSURE. ACCUMULATOR PISTON INTEGRITY IS ALARMED WITH THE USE OF LS-85-34/218 THRU 5843.
6. CRD POSITIONING IS ACCOMPLISHED BY ELECTRICAL CONTROL OF THE INSERT VALVES FCV-85-40A/219 THRU 5843 AND FCV-85-40D/219 THRU 5843 OR THE WITHDRAWAL VALVES FCV-85-40B/219 THRU 5843 AND FCV-85-40C/219 THRU 5843.
7. EACH CONTROL ROD, AS IT TRAVELS UP (INSERT) OR DOWN (WITHDRAW) PASSES A NUMBER OF SWITCHES. THE TOP TWO POSITION SWITCHES ZS-85-41A/219 THRU 5843 AND ZS-85-41B/219 THRU 5843 ARE CALLED "OVER TRAVEL". THE BOTTOM TWO POSITION SWITCHES ARE "WITHDRAW BACKSEAT", ZS-85-42/219 THRU 5843 AND "WITHDRAW DISCONNECT", ZS-85-43/219 THRU 5843. THERE ARE 48 SWITCHES BETWEEN THESE LIMITS ZS-85-44/219 THRU 5843 (-1 THRU -48). THEY ARE DIVIDED INTO TWO SETS, THE ODD NUMBERED SWITCHES ARE THE "ROD DRIFTING SWITCHES" AND THE EVEN NUMBERED SWITCHES ARE THE "LOCKED POSITION SWITCHES". THE ABOVE SWITCHES ARE ACTUATED BY A MAGNET IN THE POSITION PROBE. THE ROD POSITION IS DISPLAYED IN THE MAIN CONTROL ROOM ON REACTOR PANEL 9-5.
8. INSTRUMENTATION ASSOCIATED WITH BACKUP CONTROLS IS LOCATED AS SHOWN ON THE DIAGRAM. IF A LOCATION IS NOT SHOWN, THE DEVICES WILL BE LOCATED ON THE APPROPRIATE 4160V AC BOARD, 480V AC BOARD OR 250V DC BOARD. (REF GE DESIGN SPEC 22A1470)
9. LEADING ZEROS SHOWN IN MEL AS PART OF THE UNID ARE NOT DEPICTED. FOR ADDITIONAL GUIDANCE, REFER TO NEDP-4.

SYMBOL:

CONTROL AIR

REFERENCE DRAWINGS:

0-45E763-4..........WIRING DIAGRAM - 4160V UNIT AUX POWER SCHEMATIC DIAGRAM
3-45E766-22.........WIRING DIAGRAM - 4160V SHUTDOWN AUX POWER SCHEMATIC DIAGRAM
3-45N3616-3.........WIRING DIAGRAM - THERMOCOUPLES & RTD'S CONNECTION DIAGRAM
3-45E779-7,-8.......WIRING DIAGRAM - 480V SHUTDOWN AUX POWER SCHEMATIC DIAGRAM
MEL.................INSTRUMENT TABULATION CONTROL ROD DRIVE-SYSTEM 85
0-47E610-2-2........CONTROL DIAGRAM - CONDENSATE & DEMINERALIZED WATER STORAGE SYSTEM
3-47E610-24-1,-2....CONTROL DIAGRAM - RAW COOLING WATER SYSTEM
3-47E610-68-1.......CONTROL DIAGRAM - REACTOR RECIRCULATION SYSTEM
3-47E610-69-1.......CONTROL DIAGRAM - REACTOR WATER CLEANUP SYSTEM

47E800-1............FLOW DIAGRAM - GENERAL PLANT SYSTEMS
0-47E800-2..........MECH SYMBOLS & FLOW DIAGRAM DRAWING INDEX
3-47E817-1..........FLOW DIAGRAM - NUCLEAR BOILER
0-47E820-1..........FLOW DIAGRAM - CONTROL ROD DRIVE HYDRAULIC SYSTEM
3-47E844-2..........FLOW DIAGRAM - RAW COOLING WATER
3-47E3847-7.........FLOW DIAGRAM - CONTROL AIR SYSTEM

GENERAL ELECTRIC REFERENCE DRAWINGS:

729E499.............FUNCTIONAL CONTROL DIAGRAM - CONTROL ROD DRIVE
730E145.............PIPING & INSTRUMENT DIAGRAM - CRD HYD CONTROL UNIT
3-730E321-14........ELEMENTARY DIAGRAM - REACTOR MANUAL CONTROL SYSTEM
3-730E915-17........ELEMENTARY DIAGRAM - REACTOR PROTECTION SYSTEM
3-791E202...........ELEMENTARY DIAGRAM - CRD HYD INSTR SYSTEM

COMPANION DRAWINGS:

47W610-85-4
47W1610-85-5
1-47E610-85-1
1-47E1610-85-2
2-47E610-85-1,-4
2-47E2610-85-2,-5
2-47W2610-85-6
3-47E610-85-3,-4,-5
3-47E3610-85-2

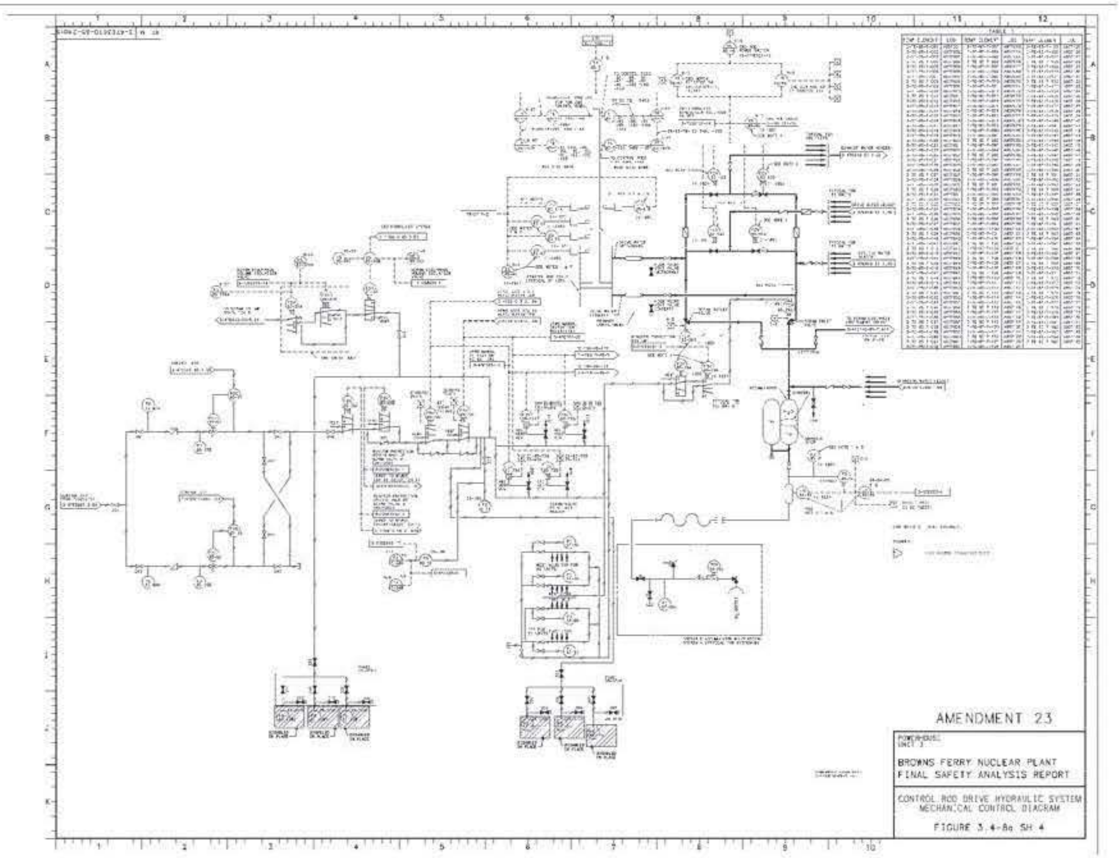
AMENDMENT 28

POWERHOUSE
UNIT 3

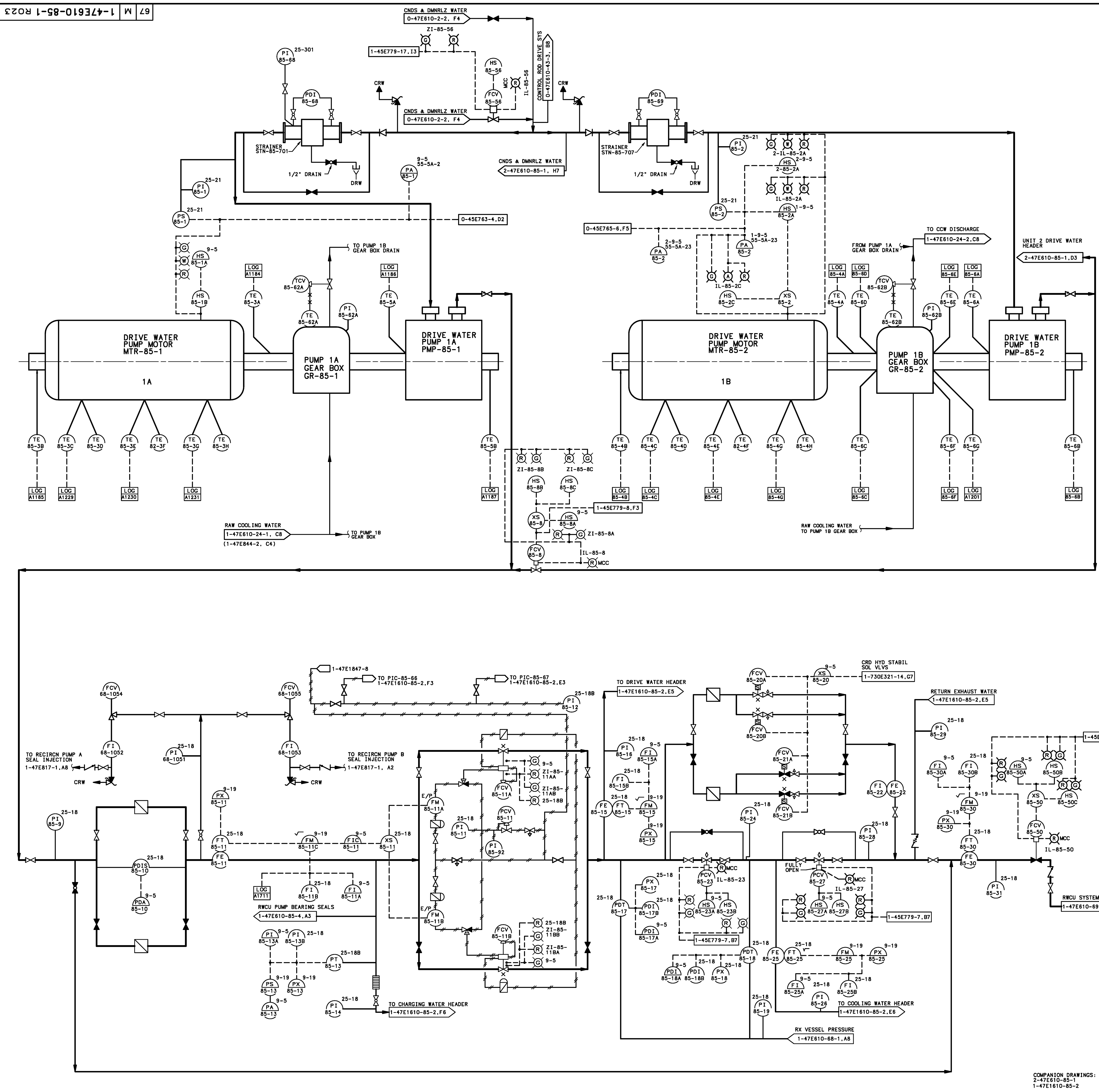
BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

CONTROL ROD DRIVE HYDRAULIC SYSTEM
MECHANICAL CONTROL DIAGRAM

FIGURE 3.4-8a SH 3



AMENDMENT 23
POWERHOUSE UNIT 3
BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT
CONTROL ROD DRIVE HYDRAULIC SYSTEM
MECHANICAL CONTROL DIAGRAM
FIGURE 3.4-8a SH 4



GENERAL NOTE:
THE CONTROL ROD DRIVE (CRD) SYSTEM IS DESIGNED TO PROVIDE THE REACTOR WITH POSITIVE REACTIVITY CONTROL WHILE MAINTAINING A CONSTANT QUANTITY OF CONDENSATE TO THE REACTOR VESSEL IN THE FORM OF DRIVE SEAL LEAKAGE DRIVE COOLING, AND FLOW TO THE REACTOR VIA RWCU RETURN LINE. THE PUMP SUCTION IS FROM THE CONDENSATE STORAGE SYSTEM AND IS RAISED TO A PRESSURE ABOVE THE REACTOR OPERATING PRESSURE BY THE DRIVE WATER PUMPS. THE PRESSURE IS THEN REDUCED TO THE THREE PRESSURES NEEDED FOR DRIVING, CHARGING AND COOLING.

NOTES:
1. HYDRAULIC CONTROL UNIT (HCU) DEVICES AND CONTROL ROD DRIVE UNIT (CRDU) DEVICES ARE SHOWN FOR A SINGLE UNIT AND ARE TYPICAL OF 185 SUCH UNITS. THE DRIVES ARE NUMBERED IN ACCORDANCE WITH TABLE 3 ON 1-730E321-4. UNIDS OF ALL COMPONENTS AFFILIATED WITH A PARTICULAR DRIVE WILL HAVE A SUFFIX THE SAME AS THE DRIVE COORDINATE. FOR EXAMPLE, THE UNID FOR FLOW CONTROL VALVE 39B ASSOCIATED WITH DRIVE 2-43 WOULD BE 1-FCV-85-39B/243, AND FOR CYLINDER 718 WOULD BE 1-CYL-85-718/243.
2. UNITS 1 AND 2 SHARE THE SPARE DRIVE WATER PUMP (1B) WHILE UNIT 3 HAS ITS OWN SPARE. EACH PUMP IS CAPABLE OF SUPPLYING 100 GPM AT 1500 PSIG.
3. THE CRD HYDRAULIC SYSTEM PROVIDES CRD OPERATING FLUID, CRD COOLANT, ACCUMULATOR CHARGING AND EXHAUST WATER TO THE REACTOR.
4. THREE STAGES OF WATER PRESSURE ARE DERIVED BY TWO SEPARATE REGULATING SYSTEMS. THE SCRAM ACCUMULATOR CHARGING WATER IS AUTOMATICALLY REGULATED TO 1400 PSIG. THE FLOW CONTROL SYSTEM USING FCV-85-11A OR B AND FE-85-11 MAINTAINS APPROXIMATELY 65 GPM TO THE SYSTEM CRD DRIVE WATER IS MAINTAINED AT REACTOR PRESSURE. (Pr) + 250 TO 270 PSIG BY CONTROLLING PCV-85-23 AND THE AUTOMATIC CLOSING OF FCV-85-20A OR 21A DURING ROD MOVEMENT. THE COOLING WATER HEADER PRESSURE IS APPROXIMATELY Pr + 20 PSIG. SIXTY-FIVE GALLONS PER MINUTE (65 GPM) FLOWS INTO THE REACTOR WHETHER OR NOT THE CONTROL RODS ARE MOVING OR AT REST.
5. EACH CRD IS EQUIPPED WITH A SEPARATE SCRAM ACCUMULATOR WHICH IS CHARGED WITH NITROGEN. ACCUMULATOR CHARGE PRESSURES ARE MONITORED BY LOCAL PRESSURE GAUGES PI-85-34/219 THRU 5843 AND PS-85-34/219 THRU 5843 WHICH ALARM ON LOW PRESSURE. ACCUMULATOR PISTON INTEGRITY IS ALARMED WITH THE USE OF LS-85-34/219 THRU 5843.
6. CRD POSITIONING IS ACCOMPLISHED BY ELECTRICAL CONTROL OF THE INSERT VALVES FCV-85-40A/219 THRU 5843 AND FCV-85-40D/219 THRU 5843 OR THE WITHDRAWAL VALVES FCV-85-40B/219 THRU 5843 AND FCV-85-40C/219 THRU 5843.
7. EACH CONTROL ROD, AS IT TRAVELS UP (INSERT) OR DOWN (WITHDRAW) PASSES A NUMBER OF SWITCHES. THE TOP TWO POSITION SWITCHES ZS-85-41A/219 THRU 5843 AND ZS-85-41B/219 THRU 5843 ARE CALLED "OVER TRAVEL". THE BOTTOM TWO POSITION SWITCHES ARE "WITHDRAW BACKSEAT", ZS-85-42/219 THRU 5843 AND "WITHDRAW DISCONNECT", ZS-85-43/219 THRU 5843. THERE ARE 48 SWITCHES BETWEEN THESE LIMITS ZS-85-44/219 THRU 5843 (-1 THRU -48). THEY ARE DIVIDED INTO TWO SETS, THE ODD NUMBERED SWITCHES ARE THE "ROD DRIFTING SWITCHES" AND THE EVEN NUMBERED SWITCHES ARE THE "LOCKED POSITION SWITCHES".
THE ABOVE SWITCHES ARE ACTUATED BY A MAGNET IN THE POSITION PROBE. THE ROD POSITION IS DISPLAYED IN THE MAIN CONTROL ROOM ON REACTOR PANEL 9-5.
8. INSTRUMENTATION ASSOCIATED WITH BACKUP CONTROLS IS LOCATED AS SHOWN ON THE DIAGRAM. IF A LOCATION IS NOT SHOWN, THE DEVICES WILL BE LOCATED ON THE APPROPRIATE 4160V AC BOARD, 480V AC BOARD OR 250V DC BOARD. (REF GE DESIGN SPEC 22A1470)
9. UNIDS ON DRAWING ARE FOR REFERENCE ONLY AND ARE ABBREVIATED AS SHOWN IN THE EXAMPLE TO MEET SPACE CONSTRAINTS. REFER TO MEL FOR COMPLETE UNIDS. ALL UNIDS ARE IN UNIT 1 UNLESS OTHERWISE NOTED. LEADING ZEROS SHOWN IN MEL AS PART OF THE UNID ARE NOT DEPICTED. FOR ADDITIONAL GUIDANCE REFER TO NEDP-4.

EXAMPLE:	MEL UNID	DRAWING UNID
	BFN-1-STN-085-0707	STN-85-707
	BFN-1-ZI-085-008A	ZI-85-8A

10. PARTIAL LOGIC FOR CONTROL ROD DRIVE SYSTEM IS SHOWN ON THE MECHANICAL LOGIC DIAGRAM LISTED IN REFERENCE DRAWINGS.

REFERENCE DRAWINGS:

MEL....................CRD INSTRUMENT TABULATION
0-47W600-224&-224A...RPS-CRD CONTROL AIR PRESSURE PANELS
0-47E610-2-2.........CONTROL DIAGRAM-CONDENSATE
2-47E610-24-1D.......CONTROL DIAGRAM-RAW COOLING WATER
1-47E610-68-1........CONTROL DIAGRAM-RECIRCULATION SYSTEM
1-47E610-69-1........CONTROL DIAGRAM-RWCU SYSTEM
2-47E610-85-1........CONTROL DIAGRAM-CRD HYD
1-47E610-85-2........CONTROL DIAGRAM-CRD HYD
2-47E610-85-4........CONTROL DIAGRAM-CRD HYD
0-47E800-1...........FLOW DIAGRAM-GENERAL PLANT SYSTEMS
0-47E800-2...........MECHANICAL SYMBOLS & FLOW DIAGRAM DRAWING INDEX
0-47E847-1 THRU -3...FLOW DIAGRAM-CONTROL AIR
0-45E763-4...........4160V UNIT AUX POWER SCHEMATIC DIAGRAM
2-45E779-7...........480V SHUTDOWN AUX POWER
0-47E610-43-3........MECHANICAL CONTROL DIAGRAM SAMPLING & WATER QUALITY SYSTEM

GE DRAWINGS:

1-730E321-14.........REACTOR MANUAL CONTROL SYSTEM
1-791E201-2,-3.......CRD HYD INSTR SYSTEM

SYMBOL:

CONTROL AIR

COMPANION DRAWINGS:
2-47E610-85-1
1-47E1610-85-2

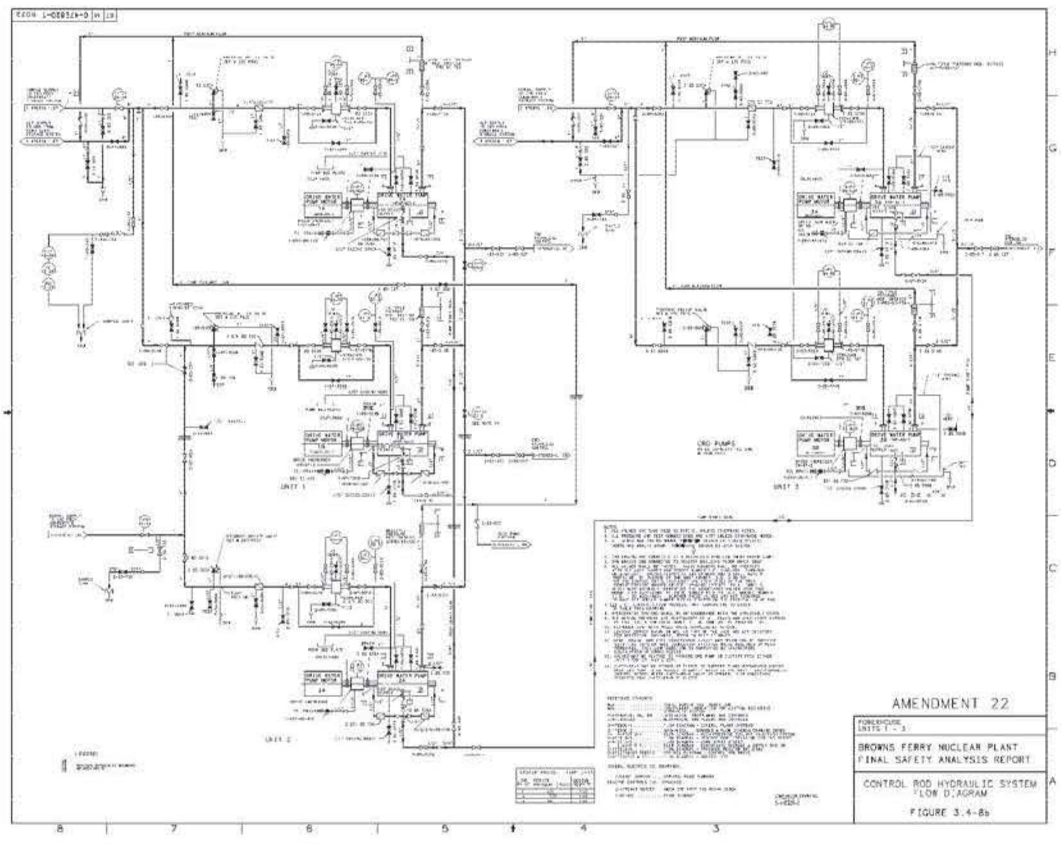
AMENDMENT 28

REACTOR BUILDING
UNIT 1

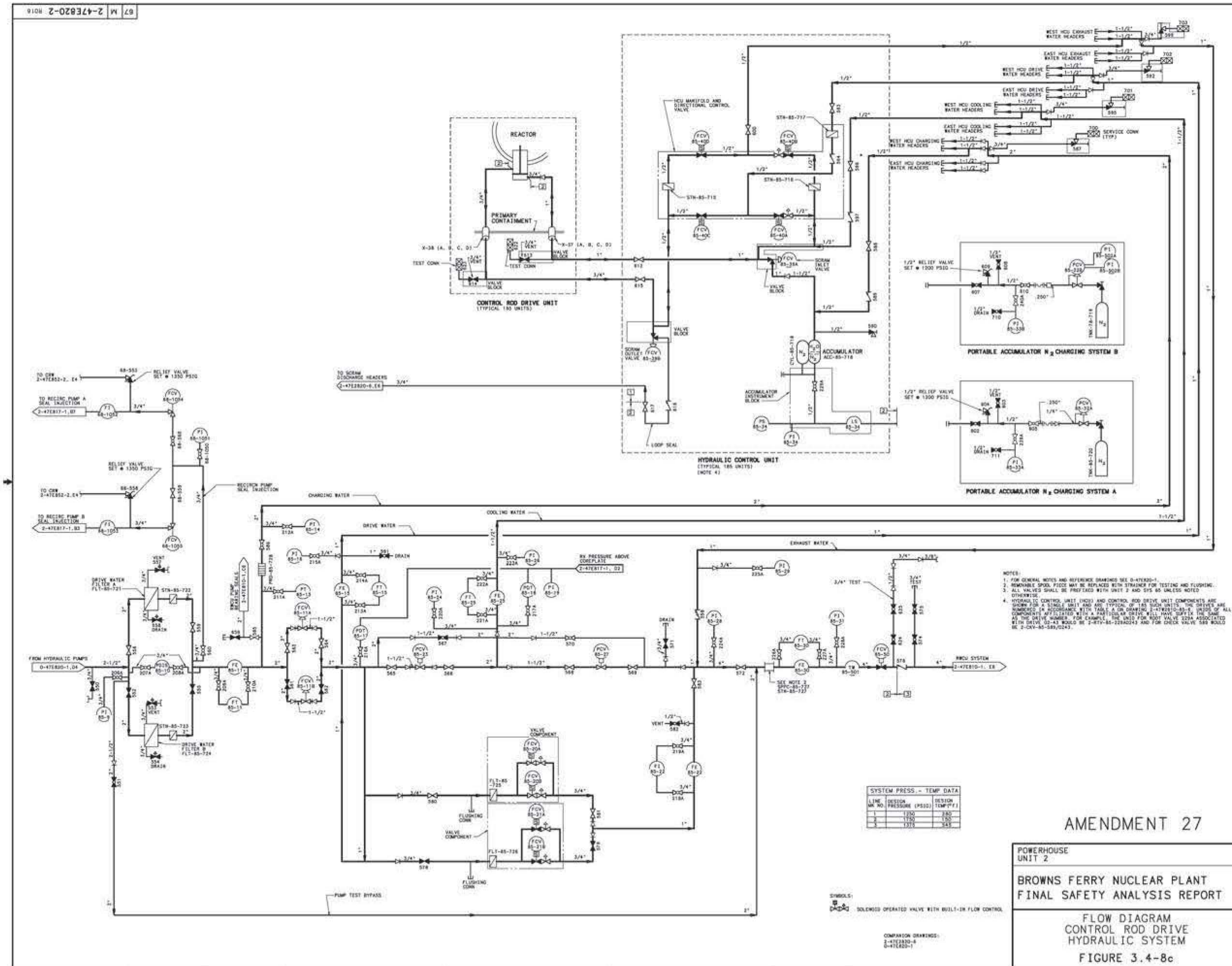
BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

CRD HYDRAULIC SYSTEM
MECHANICAL CONTROL DIAGRAM

FIGURE 3.4-8a SH 5



CRD PUMPS
UNIT 1
UNIT 2
UNIT 3
DRIVE WATER PUMP
DRIVE WATER PUMP MOTOR
AMENDMENT 22
BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT
CONTROL ROD HYDRAULIC SYSTEM
FLOW DIAGRAM
FIGURE 3.4-8b



NOTES:

1. FOR GENERAL NOTES AND REFERENCE DRAWINGS SEE 0-47E820-1.
2. REMOVABLE SPOOL PIECE MAY BE REPLACED WITH STRAINER FOR TESTING AND FLUSHING.
3. ALL VALVES SHALL BE PREFIXED WITH UNIT 2 AND SYS 85 UNLESS NOTED OTHERWISE.
4. HYDRAULIC CONTROL UNIT (HCU) AND CONTROL ROD DRIVE UNIT COMPONENTS ARE SHOWN FOR A SINGLE UNIT AND ARE TYPICAL OF 185 SUCH UNITS. THE DRIVES ARE NUMBERED IN ACCORDANCE WITH TABLE A ON DRAWING 2-47W2610-85-6. UNIDS OF ALL COMPONENTS AFFILIATED WITH A PARTICULAR DRIVE WILL HAVE SUFFIX THE SAME AS THE DRIVE NUMBER. FOR EXAMPLE, THE UNID FOR ROOT VALVE 229A ASSOCIATED WITH DRIVE 02-43 WOULD BE 2-RTV-85-229A0243 AND FOR CHECK VALVE 589 WOULD BE 2-CKV-85-589/0243.

SYSTEM PRESS.- TEMP DATA

LINE MK NO.	DESIGN PRESSURE (PSIG)	DESIGN TEMP(°F)
1	1250	280
2	1750	150
3	1375	545

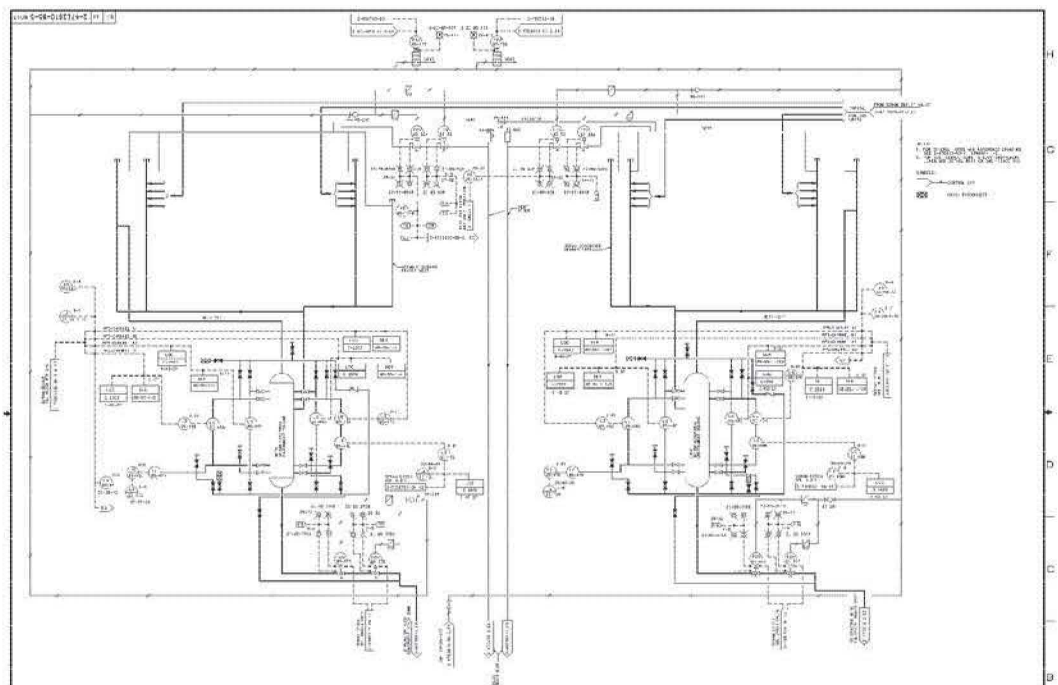
AMENDMENT 27

POWERHOUSE
UNIT 2

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

FLOW DIAGRAM
CONTROL ROD DRIVE
HYDRAULIC SYSTEM

FIGURE 3.4-8c



AMENDMENT 22

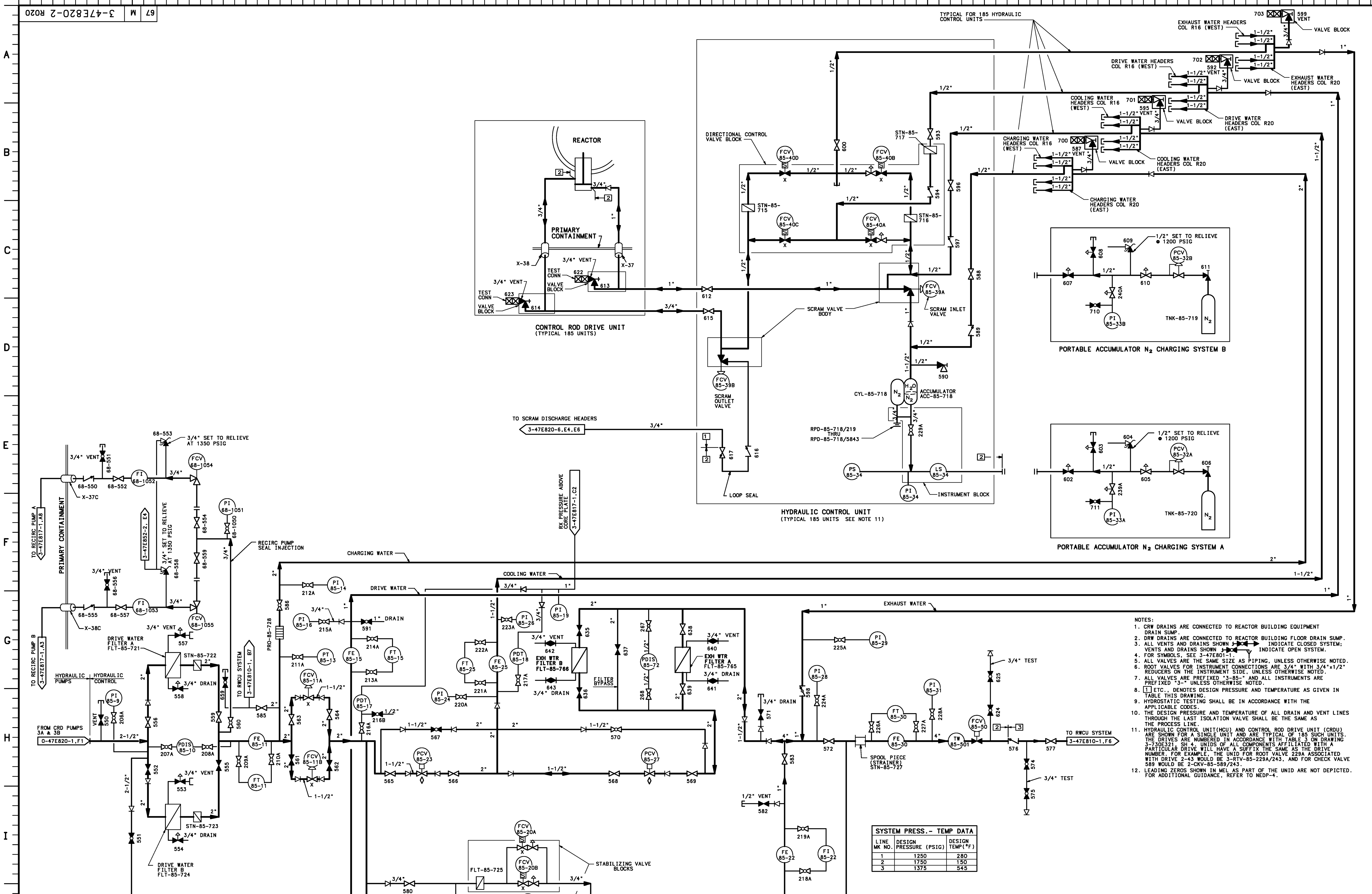
REACTOR BUILDING
UNIT 2

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

CRD HYDRAULIC SYSTEM
MECHANICAL CONTRO. DIAGRAM

FIGURE 3.4-8d

3-47E820-2 R020 | M | 67



NOTES:

1. CRW DRAINS ARE CONNECTED TO REACTOR BUILDING EQUIPMENT DRAIN SUMP.
2. DRW DRAINS ARE CONNECTED TO REACTOR BUILDING FLOOR DRAIN SUMP.
3. ALL VENTS AND DRAINS SHOWN INDICATE CLOSED SYSTEM; VENTS AND DRAINS SHOWN INDICATE OPEN SYSTEM.
4. FOR SYMBOLS, SEE 3-47E801-1.
5. ALL VALVES ARE THE SAME SIZE AS PIPING, UNLESS OTHERWISE NOTED.
6. ROOT VALVES FOR INSTRUMENT CONNECTIONS ARE 3/4" WITH 3/4"x1/2" REDUCERS ON THE INSTRUMENT SIDE, UNLESS OTHERWISE NOTED.
7. ALL VALVES ARE PREFIXED "3-85-" AND ALL INSTRUMENTS ARE PREFIXED "3-" UNLESS OTHERWISE NOTED.
8. $\boxed{1}$ ETC., DENOTES DESIGN PRESSURE AND TEMPERATURE AS GIVEN IN TABLE THIS DRAWING.
9. HYDROSTATIC TESTING SHALL BE IN ACCORDANCE WITH THE APPLICABLE CODES.
10. THE DESIGN PRESSURE AND TEMPERATURE OF ALL DRAIN AND VENT LINES THROUGH THE LAST ISOLATION VALVE SHALL BE THE SAME AS THE PROCESS LINE.
11. HYDRAULIC CONTROL UNIT(HCU) AND CONTROL ROD DRIVE UNIT (CRDU) ARE SHOWN FOR A SINGLE UNIT AND ARE TYPICAL OF 185 SUCH UNITS. THE DRIVES ARE NUMBERED IN ACCORDANCE WITH TABLE 3 ON DRAWING 3-730E321, SH 4. UNIDS OF ALL COMPONENTS AFFILIATED WITH A PARTICULAR DRIVE WILL HAVE A SUFFIX THE SAME AS THE DRIVE NUMBER. FOR EXAMPLE, THE UNID FOR ROOT VALVE 229A ASSOCIATED WITH DRIVE 2-43 WOULD BE 3-RTV-85-229A/243, AND FOR CHECK VALVE 589 WOULD BE 2-CKV-85-589/243.
12. LEADING ZEROS SHOWN IN MEL AS PART OF THE UNID ARE NOT DEPICTED. FOR ADDITIONAL GUIDANCE, REFER TO NEDP-4.

SYSTEM PRESS.- TEMP DATA

LINE MK NO.	DESIGN PRESSURE (PSIG)	DESIGN TEMP(°F)
1	1250	280
2	1750	150
3	1375	545

REFERENCE DRAWINGS:

MEL..........................INSTRUMENT TABULATION FOR SYSTEM 85
MEL..........................VALVE MARKER TAG TABULATION
3-47E610-85-1...............MECHANICAL CONTROL DIAGRAM - CRD HYDRAULIC SYSTEM
3-47E810-1..................FLOW DIAGRAM - REACTOR WATER CLEANUP SYSTEM
3-47E817-1..................FLOW DIAGRAM - NUCLEAR BOILER
3-47E852-2..................FLOW DIAGRAM - CLEAN RADWASTE & DECONTAMINATION DRAINAGE

GENERAL ELECTRIC CO. DRAWINGS:

730E145...........PIPING AND INSTRUMENT DIAGRAM
3-730E321-4.......ELEMENTARY DIAGRAM

REACTOR CONTROLS, INC. DRAWINGS:

BF-001.........FLOW DIAGRAM

COMPANION DRAWINGS:

0-47E820-1
1-47E820-2,-6,-7
2-47E820-2,-6,-7
3-47E820-6,-7

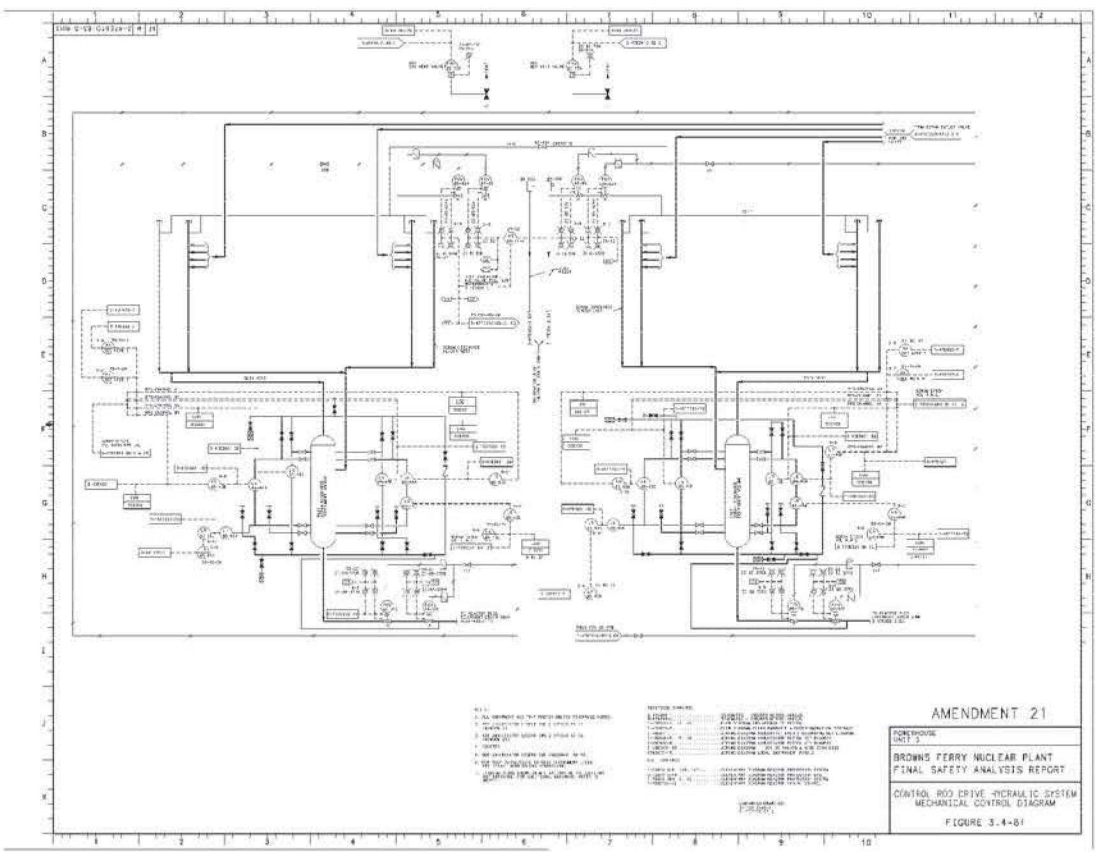
AMENDMENT 28

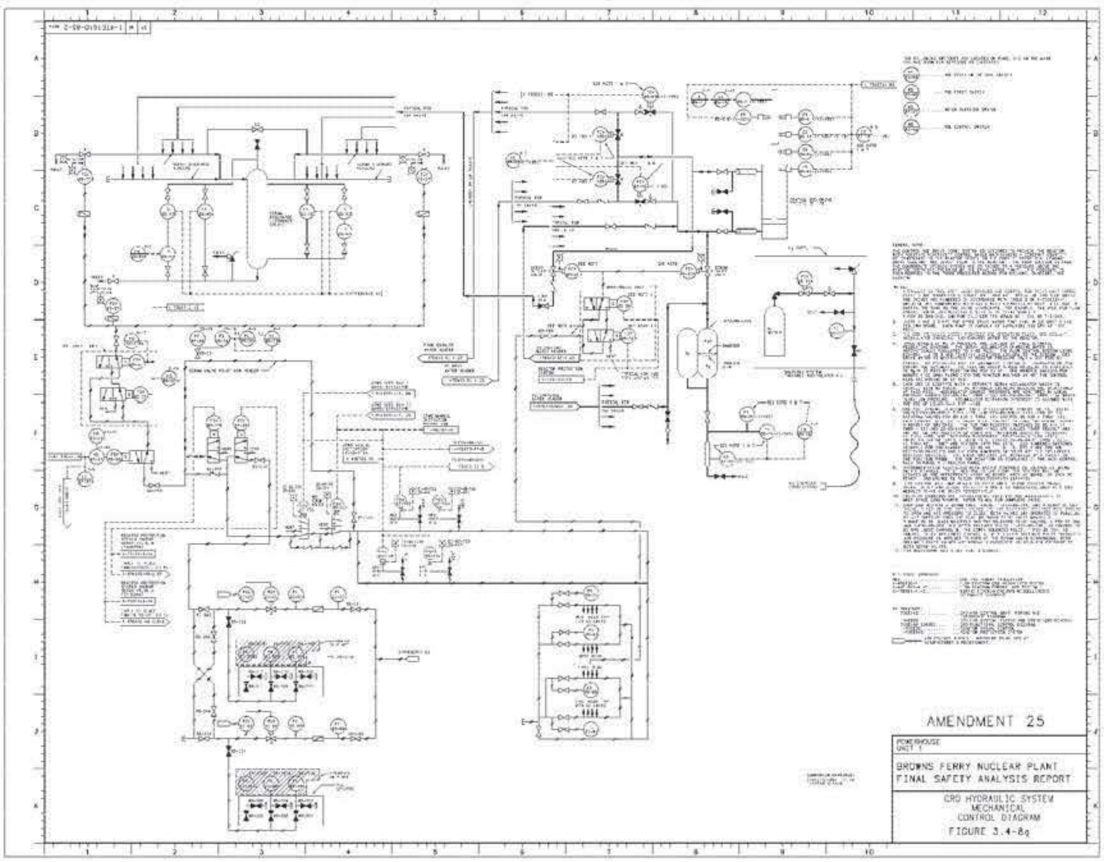
POWERHOUSE
UNIT 3

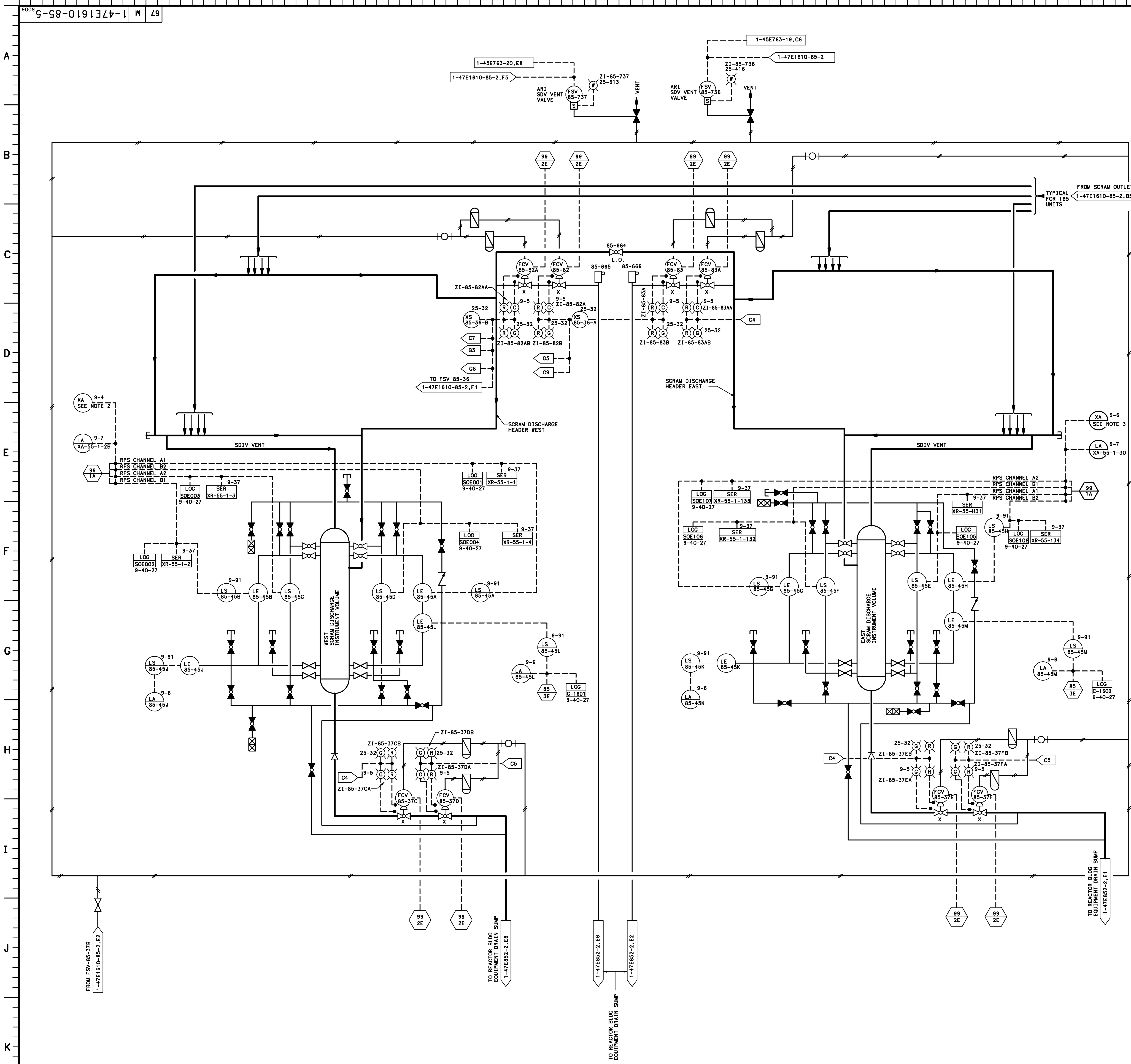
BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

CONTROL ROD DRIVE HYDRAULIC SYSTEM
FLOW DIAGRAM

FIGURE 3.4-8e


AMENDMENT 21
POWERHOUSE
UNIT 3
BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT
CONTROL ROD DRIVE HYDRAULIC SYSTEM
MECHANICAL CONTROL DIAGRAM
FIGURE 3.4-8I


AMENDMENT 25
POWERHOUSE
UNIT 1
BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT
CRD HYDRAULIC SYSTEM
MECHANICAL
CONTROL DIAGRAM
FIGURE 3.4-8q



REFERENCE DRAWINGS:

BF-001, -002 & -008............REACTOR CONTROLS INC. (CONTR 69C60-91121)
1-47E1610-85-1 & -2............MECH CONTROL DIAGRAM CRD HYDRAULIC SYSTEM
1-47E852-2.....................FLOW DIAGRAM CLEAN RADWASTE & DECON DRAINAGE
1-47E820-6.....................FLOW DIAGRAM CONTROL ROD DRIVE HYDRAULIC SYSTEM
1-47E605-28....................MECHANICAL LAYOUT OF CONTROL BOARDS

1-730E915......................RPS ELEM
1-730E321-SERIES...............RMCS ELEM
1-729E136-3....................RPS IED

NOTES:

1. THIS DRAWING SUPERSEDES 47W610-85-5 FOR UNIT 1.
2. SEE ANNUNCIATOR LEGEND DRAWING 1-47E610-55-14 (WINDOW 1).
3. SEE ANNUNCIATOR LEGEND DRAWING 1-136B2649 SHT 14 (WINDOW 21).
4. FOR TEST CONNECTIONS TO SDIV INSTRUMENT LINES, SEE DETAIL B199 ON DRAWING 0-47W600-199.
5. LEADING ZEROS SHOWN IN MEL AS PART OF THE UNID ARE NOT DEPICTED. FOR ADDITIONAL GUIDANCE, REFER TO NEDP-4.

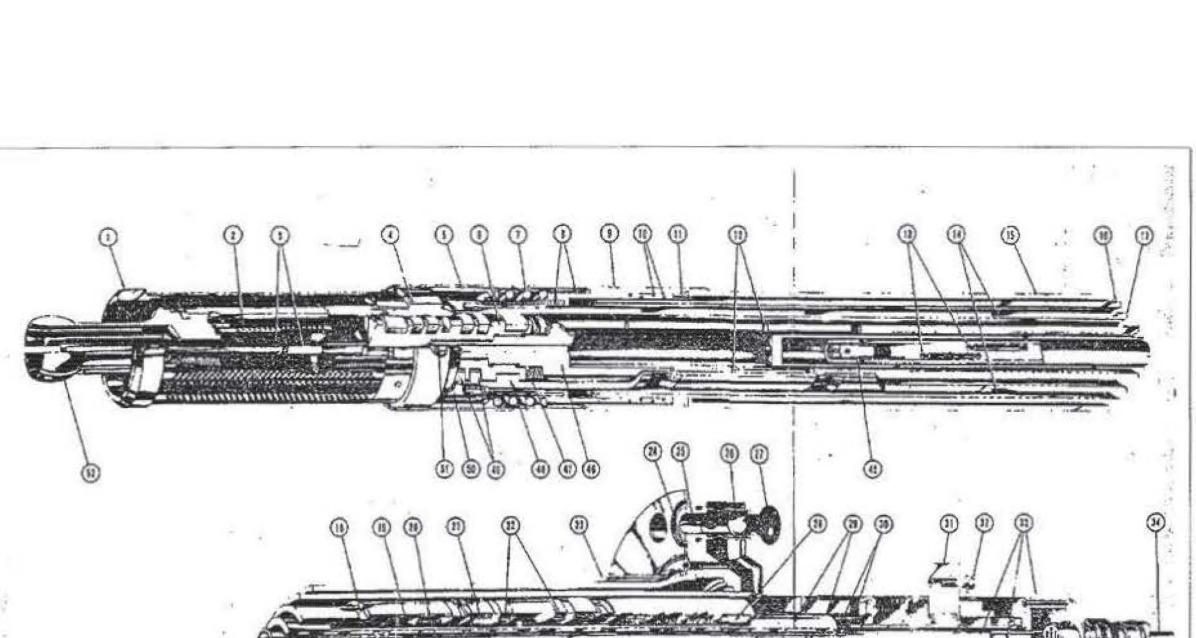
AMENDMENT 28

REACTOR BUILDING
UNIT 1

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

CRD HYDRAULIC SYSTEM
MECHANICAL CONTROL DIAGRAM

FIGURE 3.4-8h

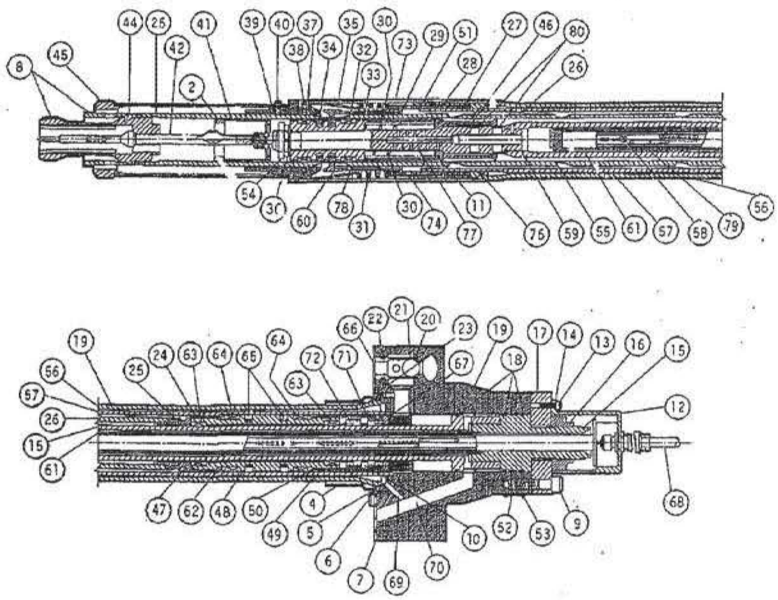


1. EXTERNAL FILTER ASSEMBLY
2. INTERNAL FILTER ASSEMBLY
3. UNCOUPLING ROD ASSEMBLY
4. GUIDE CAP
5. BARREL
6. STOP PISTON
7. COLLET SPRING
8. COLLET AND COLLET PISTON
9. COLLET HOUSING (Part of cylinder, tube and flange)
10. COLLET PISTON SEALS
11. SPACER (Part of cylinder, tube, and flange)
12. BUFFER ORIFICES (Typical)
13. POSITION INDICATOR SWITCHES (Typical)
14. LOCKING GROOVE (Typical)
15. OUTER TUBE (Part of cylinder, tube, and flange)
16. CYLINDER TUBE
17. INDEX TUBE
18. LOCKING BAND (Typical)
19. INTERNAL PISTON SEAL RINGS (Typical)
20. INTERNAL PISTON BUSHINGS (Typical)
21. EXTERNAL PISTON BUSHINGS
22. EXTERNAL PISTON SEALS
23. STRAINER
24. COOLING WATER ORIFICE
25. DRIVE-INSERT WATER INLET (Scram and insert; Drive-withdrawal outlet)
26. BALL-SHUTTLE VALVE
27. REACTOR WATER INLET (Through drive housing)
28. SWITCH-ACTUATING MAGNET (Part of drive piston)
29. PISTON TUBE ASSEMBLY
30. DRIVE-WITHDRAW PORTS AND ANNULUS (Also scram outlet)
31. RING FLANGE
32. BACKING SCREW (Typical)
33. POSITION INDICATOR PROBE
34. POSITION INDICATOR CABLE
35. PISTON TUBE NUT
36. CAP SCREW (Typical)
37. O-RING SEALS
38. DRIVE-INSERT PORTS AND ANNULUS
39. DRIVE FLANGE (Part of cylinder, tube, and flange)
40. UNLOCKING PORT AND ANNULUS
41. DRIVE-WITHDRAW WATER INLET (Also outlet for scram water)
42. METAL O-RING SEAL (Drive to housing)
43. DRIVE PISTON
44. INDICATOR TUBE (Part of piston tube)
45. THERMOCOUPLE (Part of position indicator probe)
46. STUD (Part of piston tube)
47. SPRING WASHERS
48. STOP PISTON BUSHINGS (Typical)
49. STOP PISTON SEAL RINGS (Typical)
50. COLLET FINGER (Typical)
51. COTTER PIN
52. COUPLING SPUD

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

Control Rod Unit Drive – Cutaway Illustration
FIGURE 3.4-9

AMENDMENT 16



2 - SEAL RING (INNER FILTER)
3 - FLAT HEAD SCREW (O-RING SPACER MOUNTING)
4 - STRAINER
5 - FLAT HEAD SCREW
6 - DOWEL PIN
7 - O-RING SPACER
8 - SPUD & CYLINDER
9 - SOCKET-HEAD CAP SCREW (RING FLANGE MOUNTING)
10 - SEAL CUP (PART OF DRIVE PISTON ASSEMBLY)
11 - SEAL RING (BUFFER PISTON)
12 - POSITION INDICATOR PROBE
13 - FILLISTER-HEAD SCREW (POSITION INDICATOR PROBE MOUNTING)
14 - LOCKWASHER (FOR PART 13)
15 - PISTON TUBE ASSEMBLY
16 - NUT (PISTON TUBE)
17 - RING FLANGE
18 - O-RING (PISTON TUBE)
19 - CYLINDER, TUBE, AND FLANGE
20 - BALL (CHECK VALVE)
21 - BALL RETAINER
22 - O-RING (BALL RETAINER)
23 - COOLING WATER ORIFICE

24 - DRIVE PISTON ASSEMBLY
25 - LOCKING BAND
26 - INDEX TUBE
27 - SEAL RING (COLLET PISTON - INTERNAL)
28 - SEAL RING (COLLET PISTON - EXTERNAL)
29 - COLLET & PISTON
30 - BUFFER SPRING
31 - COLLET SPRING
32 - SPLIT BUSHING (STOP PISTON)
33 - STOP PISTON
34 - SEAL RING (STOP PISTON)
35 - BARREL
36 - NUT (STOP PISTON)
37 - PLUG (GUIDE CAP)
38 - FILLISTER-HEAD SCREW (GUIDE CAP PLUG MOUNTING)
39 - GUIDE CAP
40 - DRILLED FILLISTER-HEAD SCREW (OUTER FILTER MOUNTING)
41 - INNER FILTER
42 - UNCOUPLING ROD
44 - LOCKING BAND
45 - OUTER FILTER
46 - BUFFER PISTON
47 - PISTON HEAD (PART OF DRIVE PISTON)
48 - PISTON COUPLING (PART OF DRIVE PISTON)
49 - MAGNET HOUSING (PART OF DRIVE PISTON)
50 - LOCKING BAND (ON DRIVE PISTON)
51 - COLLET HOUSING (PORTION OF OUTER TUBE)
52 - NAMEPLATE
53 - DRIVE SCREW
54 - LOCKING CUP
55 - INDEX TUBE NOTCH
56 - OUTER TUBE (PART OF CYLINDER, TUBE, AND FLANGE)
57 - INNER TUBE (PART OF CYLINDER, TUBE, AND FLANGE)
58 - THERMOCOUPLE (PART OF POSITION INDICATOR PROBE)
59 - BUFFER SHAFT
60 - COLLET FINGER (PART OF COLLET RETAINER, AND PISTON)
61 - INDICATOR TUBE (PART OF PISTON TUBE)
62 - LOCKING BAND (ON DRIVE PISTON)
63 - INTERNAL BUSHING (DRIVE PISTON)
64 - EXTERNAL BUSHING (DRIVE PISTON)
65 - OUTER SEALS (DRIVE PISTON)
66 - INSERT PORT (INSERT & SCRAM INLET/WITHDRAW OUTLET)
67 - RING MAGNET (PART OF DRIVE PISTON)
68 - CABLES (POSITION INDICATOR)
69 - PORT TO COLLET PISTON (WITHDRAW PRESSURE TO COLLET PISTON)
70 - WITHDRAW PORT (WITHDRAW INLET/ INSERT OUTLET & SCRAM DISCHARGE)
71 - INNER SEALS (DRIVE PISTON - DRIVE DOWN SEALS)
72 - INNER SEALS (DRIVE PISTON - DRIVE-UP SEALS)
73 - WATER PORTS IN COLLET HOUSING
74 - WATER PORTS IN COLLET PISTON
75 - VESSEL-WATER PORTS
76 - SHOULDER (PART OF CYLINDER, TUBE AND FLANGE)
77 - BUFFER ORIFICES IN PISTON TUBE ASSEMBLY
78 - COLLET RETAINER (HOLDS COLLET FINGERS)
79 - POSITION INDICATOR SWITCHES
80 - BUFFER SHAFT PORTS

AMENDMENT 16

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

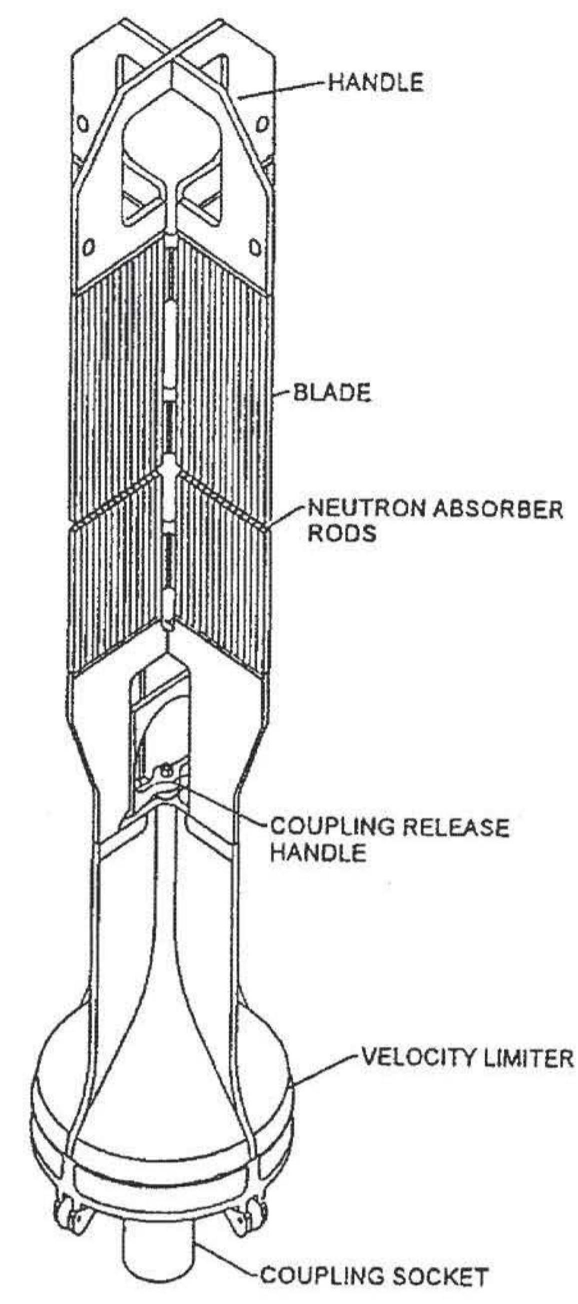
MODIFIED CONTROL ROD DRIVE (BWR/6)

FIGURE 3.4-9a

Figure 3.4-10

Deleted by Amendment 7.

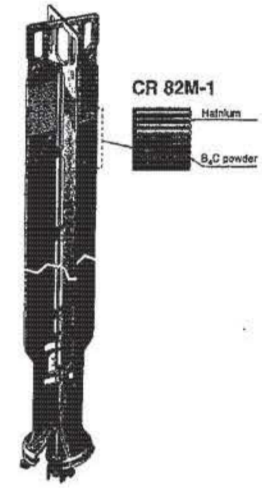
Figure 3.4-11
(Deleted by Amendment 22)



AMENDMENT 16

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

Marathon Control Rod — Isometric
FIGURE 3.4-12



ADMENDMENT 20

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

WESTINGHOUSE
CR-82M-1
CONTROL ROD

FIGURE 3.4-13

3.5 CONTROL ROD DRIVE HOUSING SUPPORTS

3.5.1 Safety Objective

The control rod drive housing supports 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 shall be limited, following a postulated control rod drive (CRD) housing failure, so that any resulting nuclear transient could not be sufficient to cause fuel damage or additional damage to the process barrier.

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

3.5.3 Description

The control rod drive 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 reactor support pedestal.

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

The support bars are bolted between the bottom ends of the hanger rods. The spring pivots at the top and the beveled loose-fitting ends on the support bars prevent substantial bending moment in the hanger rods if the support bars are 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 work space and because it is necessary that control rod drive, position indicators, and in-core instrumentation components be 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 1″ ± .25″ gap is provided between the grid and the bottom contact surface of the control rod drive flange. During system heatup this gap is reduced by a net downward expansion of the housings with respect to the supports. In the hot operating condition, a minimum gap of 7/16″ is maintained.

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," was 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 were 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 1250 psig (reactor vessel design pressure) acting on the area of the separated housing. The weight of the separated housing, control rod drive, and blade, plus the force of 1250 psig pressure acting on the area of the separated housing, gives a force of approximately 35,000 pounds. This force is multiplied by a factor of 3 for impact, conservatively assuming the housing travels through a 1-inch gap before contacting the supports. The total force (10.5×10^4 pounds) is then treated as a static load in design formulas. The control rod drive housing supports are designed as Class I equipment in accordance with Appendix C, "Structural Qualification of Subsystems and Components".

All control rod drive 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

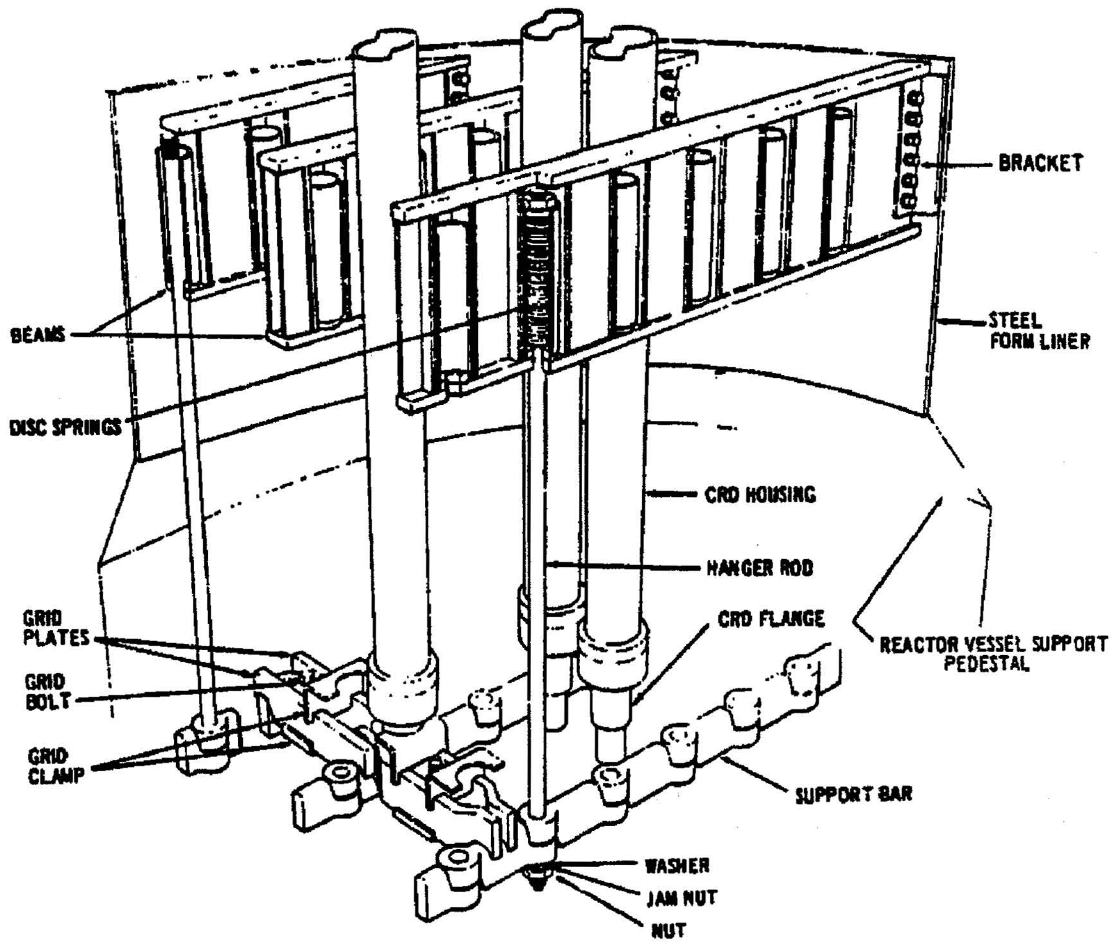
Downward travel of a CRD housing and its control rod following the postulated housing failure is the sum of the compression of the disc springs under dynamic loading and the initial gap between the grid and the bottom contact surface of the CRD flange. If the reactor were cold and pressurized, the downward motion of the control rod would be limited to the approximate 2-inch spring compression plus the installed 1″ ± .25″ gap. If the reactor were hot and pressurized, the gap would be greater than a 7/16″ 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 inches). The nuclear transient from sudden withdrawal of any control rod through a distance of one drive notch at any position in the core does not result in a transient sufficient to cause damage to any radioactive material barrier. This meets the fuel damage limitation of safety design basis 1.

The control rod drive housing supports are in place any time the reactor is to be operated.

At plant operating temperature, a minimum gap of 7/16″ is maintained 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

The control rod drive housing supports were inspected after initial installation. When the reactor is in the shutdown mode (MODE 4 or MODE 5), the control rod drive housing supports may be removed to permit inspection and maintenance of the control rod drives. 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 flange lower contact surface and the grid is maintained.



AMENDMENT 16

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

Control Rod Drive Housing Support
Isometric
FIGURE 3.5-1

3.6 NUCLEAR DESIGN

This section describes the nuclear core design basis and the models used to analyze the fuel discussed in Subsection 3.2, “Fuel Mechanical Design.” The nuclear design criteria of AREVA fuel is described in Reference 15 with the detailed design provided in a reload specific fuel nuclear design report.

3.6.1 Power Generation Objective

The objectives of the fuel nuclear design are as follows:

a. To attain rated power generation from the nuclear fuel for a given period of time,

b. To attain reactor nuclear stability throughout core life,

c. To allow normal power operation of the nuclear fuel without sustaining fuel damage.

3.6.2 Power Generation Design Basis

1. Fuel nuclear design shall provide sufficient excess reactivity during power operation to achieve the core design burnup.

2. Fuel nuclear design shall provide sufficient negative reactivity feedback to facilitate normal maneuvering and control.

3. Fuel nuclear design shall, in combination with reactivity control systems, allow continuous, stable regulation of core excess reactivity.

3.6.3 Safety Design Basis

The design bases are those that are required for the plant to operate, meeting all safety requirements. Safety design bases fall into two categories: (1) the reactivity basis, which prevents an uncontrolled positive reactivity excursion, and (2) the overpower bases, which prevent the core from operating beyond the fuel integrity limits.

3.6.3.1 Reactivity Basis

The nuclear design shall meet the following basis: The core shall be capable of being made subcritical at any time or at any core condition with the highest worth control rod fully withdrawn.

3.6.3.2 <u>Overpower Bases</u>

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

3.6.4 <u>Description</u>

The BWR core design consists of a light-water moderated reactor, fueled with slightly enriched uranium-dioxide. The use of water as a moderator produces a neutron energy spectrum in which fissions are caused principally by thermal neutrons. At normal operating conditions, the moderator boils, producing a spatially variable distribution of steam voids in the core. The BWR design provides a system for which reactivity is reduced by an increase in the steam void content in the moderator. Void feedback effects are one inherent safety feature of the BWR. 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.

3.6.4.1 <u>Nuclear Design Description</u>

The reference loading pattern for each cycle is documented by the vendor specific reload licensing analysis report. The current licensing report for each BFN unit is included in Appendix N of the FSAR.

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

Any differences between the reference loading pattern and the actual loading pattern are evaluated as described in Reference 12 for AREVA analyzed reload cores. To assure that licensing calculations performed on the reference core are applicable to the as-loaded core, certain key parameters, which affect the licensing calculations, are examined to assure that there is no adverse impact. If the final loading plan does not meet the necessary criteria, a re-examination of the parameters that determine the operating limits is performed. Only when this examination has been completed and it has been established that the as-loaded core satisfies the licensing basis will the core be operated.

3.6.4.2 Power Distribution

The core power distribution is a function of fuel bundle design, core loading, control rod pattern, core exposure distribution, and core coolant flow rate. The thermal performance parameters MAPLHGR, MLHGR, and MCPR (defined in Table 3.6-1) limit unacceptable core power distributions.

3.6.4.2.1 Power Distribution Measurements

The techniques for measurement of the power distribution within the reactor core, together with instrumentation correlations and operation limits, are discussed in Reference 11 for AREVA analyzed cores.

3.6.4.2.2 Power Distribution Accuracy

The accuracy of the calculated power distribution is discussed in References 11 and 13 for AREVA analyzed reload cores.

3.6.4.2.3 Power Distribution Anomalies

Stringent inspection procedures are utilized to ensure the correct arrangement of the core following fuel loading. A fuel loading error (a mislocated or a misoriented fuel bundle in the core) would be a very improbable event, but calculations have been performed to determine the effects of such events on CPR and LHGR. The fuel loading error is discussed further References 12 and 17.

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

3.6.4.3 Reactivity Coefficients

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

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

3.6.4.3.1 Doppler Reactivity Coefficient

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

Although the reactivity change caused by the Doppler effect is small compared to other power-related reactivity changes during normal operation, it becomes very important during postulated rapid power excursions in which large fuel temperature changes occur. The most severe power excursions are those associated with rod drop accidents. A local Doppler feedback associated with a 3000°F to 5000°F temperature rise is available for terminating the initial excursion.

The application of the Doppler coefficient to the AREVA analysis of the rod drop accident is discussed in Reference 12.

3.6.4.3.2 Moderator Void Coefficient

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

A detailed discussion of the methods used to calculate void reactivity coefficients, their accuracy, and their application to plant transient analyses is presented in References 11 and 12 for AREVA analyzed cores.

3.6.4.4 Control Requirements

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

3.6.4.4.1 Shutdown Reactivity

The core must be capable of being made subcritical, with margin, in the most reactive condition throughout the operating cycle with the most reactive control rod fully withdrawn and all other rods fully inserted. The shutdown margin is determined by using a BWR simulator code (see Subsection 3.6.5, "Analytical Methods") to calculate the core multiplication at selected exposure points with the strongest rod fully withdrawn. The shutdown margin is calculated based on the carryover of the minimum expected exposure at the end of the previous cycle. The core is assumed to be in the cold, xenon-free condition in order to ensure that the calculated values are conservative. Further discussion of the uncertainty of these calculations is given in References 11 and 12 for AREVA analyses.

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

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

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

where,

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

3.6.4.4.2 Reactivity Variations

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

3.6.4.4.3 Standby Liquid Control System

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

3.6.4.5 Criticality of Reactor During Refueling

The core is subcritical at all times during refueling. This is ensured by a combination of refueling interlocks and analytical verification of shutdown margin. Shutdown margin is determined by using a BWR simulator code (see Subsection 3.6.5, “Analytical Methods”) to calculate the core multiplication with the strongest rod fully withdrawn for the final reload core configuration and for limiting interim core configurations in the case of an incore shuffle.

3.6.4.6 Stability

3.6.4.6.1 Xenon Transients

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

(1) Never having observed xenon instabilities in operating BWRs
(2) Special tests which have been conducted on operating BWRs in an attempt to force the reactor into xenon instability
(3) Calculations

All of these indicators have proven that xenon transients are highly damped in a BWR due to the large negative power coefficient.
Analyses and experiments conducted in this area are reported in Reference 8.

3.6.4.6.2 Thermal Hydraulic Stability

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

(1) Neutron flux limit cycles, which oscillate up to 120% APRM high neutron flux scram setpoint or up to the LPRM upscale alarm trip (without initiating scram) prior to operator mitigating action shall not result in exceeding specified acceptable fuel design limits.
(2) The individual channels shall be designed and operated to be hydrodynamically stable or more stable than the reactor core for all expected operating conditions (analytically demonstrated).

The AREVA for demonstrating the above has been reviewed and approved by the NRC in Reference 14. See Subsection 3.7.6.2, “Thermal Hydraulic Stability Performance,” for additional information regarding core thermal-hydraulic stability.

3.6.5 Analytical Methods

AREVA nuclear evaluations are performed using the analytical tools and methods described in References 11 and 12.

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

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

3.6.6 Reactivity of Fuel in Storage

The NRC amended its regulations in December 1998 to give licensees the option of either meeting the criticality accident requirements of 10 CFR 70.24 paragraphs (a) through (c) in handling and storage areas for Special Nuclear Material (SNM), or electing to comply with certain requirements in a new section, 50.68, in 10 CFR part 50. Browns Ferry has chosen to comply with the new requirements of 10 CFR 50.68(b).

To meet these new requirements the quantity of SNM, other than nuclear fuel stored onsite, shall be less than the quantity necessary for a critical mass. The quantity of SNM specified to be enough for a critical mass in Section 1.1 of Regulatory Guide 10.3, “Guide for the Preparation of Applications for Special Nuclear Material Licenses of Less than Critical Mass Quantities” is 350 gram of U-235, 200 grams of U-233, and 200 grams of Pu-239. The combined total of non-fuel SNM maintained at BFN is far less than this quantity. The maximum nominal U-235 enrichment of the fresh fuel assemblies is limited to five percent by weight. Requirements for new fuel storage are described in Section 10.2 of the FSAR. Requirements for spent fuel storage are described in FSAR, Section 10.3.5. Requirements for the reactor building ventilation radiation monitors are described in FSAR, Section 7.12.5. The existing radiation monitors on the refuel zone are considered to meet 10 CFR 50.68(b) requirements. The remaining 50.68 (b) requirements for fuel handling outside of approved storage areas are contained in plant procedures.

The basic criterion associated with the storage of irradiated (spent) and new fuel is that the effective multiplication factor of fuel stored under normal conditions will be ≤0.95 for high density racks. For storage of new fuel in the new fuel storage vaults, the effective multiplication factor will be ≤0.90 for dry conditions and ≤0.95 for flooded conditions. [Note: Placement of fuel in the new fuel storage vaults is currently prohibited at Browns Ferry. This restriction is administratively controlled by BFN Site Procedures.]

The current and legacy fuel products have been assessed and shown to meet the criteria per Reference 10.

3.6.7 References

1. (Deleted)

2. (Deleted)

3. (Deleted)

4. (Deleted)

5. (Deleted)

6. (Deleted)

7. (Deleted)

8. R. L. Crowther, “Xenon Considerations in Design of Boiling Water Reactors,” APED-5640, June 1968.

9. (Deleted)

10. “Browns Ferry Nuclear Plant Units 1, 2, and 3 Spent Fuel Storage Pool Criticality Safety Analysis for ATRIUM™ 10XM (ANP-3160(P) Revision 0), October 2012.

11. EMF-2158(P)(A) Revision 0, “Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICROBURN-B2,” Siemens Power Corporation, October 1999.

12. XN-NF-80-19(P)(A) Volume 1 and Supplements 1 and 2, “Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis,” Exxon Nuclear Company, March 1983.

13. EMF-2493(P) Revision 0, “MICROBURN-B2 Based Impact of Failed/Bypassed LPRMs and TIPs, Extended LPRM Calibration Interval, and Single Loop Operation on Measured Radial Bundle Power Uncertainty,” Siemens Power Corporation, December 2000.

14. EMF-CC-074(P)(A) Volume 4 Revision 0, “BWR Stability Analysis - Assessment of STAIF with Input from MICROBURN-B2,” Siemens Power Corporation, August 2000.

15. ANF-89-98(P)(A) Revision 1 and Supplement 1, “Generic Mechanical Design Criteria for BWR Fuel Designs,” Advanced Nuclear Fuels Corporation, May 1995.

16. ANP-10307PA Revision 0 AREVA MCPR Safety Limit Methodology for Boiling Water Reactors, AREVA NP, June 2011.

17. XN-NF-80-19(P)(A) Volume 4 Revision 1, "Exxon Nuclear Methodology for Boiling Water Reactors Application of the ENC Methodology to BWR Reloads," Exxon Nuclear Company, April 1986.

18. (Deleted)

19. (Deleted)

20. (Deleted)

21. ANP-10298PA Revision 0, ACE/ATRIUM 10XM Critical Power Correlation, AREVA NP, March 2010.

22. ANP-3140(P) Revision 0, Browns Ferry Units 1, 2, and 3 Improved K-Factor Model for ACE/ATRIUM 10XM Critical Power Correlation, AREVA NP, August 2012.

23. EMF-2209(P)(A) Revision 3, SPCB Critical Power Correlation, AREVA NP, September 2009.

Table 3.6-1

DEFINITION OF FUEL DESIGN LIMITS

Maximum Linear Heat Generation Rate (MLHGR) The MLHGR is the maximum linear heat generation rate expressed in kW/ft for the fuel rod with the highest surface heat flux at a given nodal plane in the bundle. The MLHGR operating limit is bundle type dependent. The MLHGR is monitored to assure that all mechanical design requirements will be met.
Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) The MAPLHGR is the maximum average linear heat generation rate (expressed in kW/ft) in any plane of a fuel bundle allowed by the plant Technical Specifications for that fuel type. This parameter is obtained by averaging the linear heat generation rate over each fuel rod in the plane, and its limiting value is selected such that: (a) The peak clad temperature during the design basis loss-of-coolant accident will not exceed 2200°F in the plane of interest, and (b) All fuel design limits specified in Subsection 3.2, “Fuel Mechanical Design,” will be met if the MLHGR is not monitored for that purpose.
Minimum Critical Power Ratio (MCPR) The critical power ratio is defined as the ratio of the critical power (bundle power at which some point within the assembly experiences onset of boiling transition) to the operating bundle power. The critical power is determined at the same mass flux, inlet temperature, and pressure which exists at the specified reactor condition. Thermal margin is stated in terms of the minimum value of the critical power ratio, MCPR, which corresponds to the most limiting fuel assembly in the core.
Operating Limit MCPR The MCPR operating limit is the minimum CPR specified in the Core Operating Limits Report for a given bundle type. The minimum CPR is a function of several parameters, the most important of which are bundle power, bundle flow, and empirically derived correlation coefficients (R-factors in GE terminology; f_{eff} or K-factor for AREVA). The empirical correlation coefficients are dependent upon the local power distribution and details of the bundle mechanical design including channel bow considerations. The limiting value of CPR is selected for each bundle type such that, during the most limiting event of moderate frequency, the calculated CPR in that bundle is not less than the safety limit CPR. The MCPR safety limit is attained when parameters (e.g., bundle power, flow, pressure, subcooling, etc.) are applied to approved licensing correlations, supporting the Technical Specification value. [*Note: The K-factor is used for AREVA reload analyses and monitoring with the ACE critical power correlation, per References 21 and 22. Fuel designs monitored using the AREVA SPCB correlation, a similar term, $F_{eff,}$ is developed to account for local power distribution and details of the bundle mechanical design, per Reference 23. Channel bow considerations are accounted during MCPR Safety Limit analyses, per (Reference 16)]

Figures 3.6-1 through 3.6-13

(Deleted by Amendment No. 16)

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

The thermal hydraulic design of the core shall provide the following characteristics:

a. The ability to achieve rated core power output throughout the design lifetime of the fuel without sustaining fuel damage.

b. 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

1. The thermal hydraulic design of the core shall establish limits for use in setting devices of the nuclear safety systems so that no fuel damage occurs as a result of abnormal operational transients (see Chapter 14, "Plant Safety Analysis").

2. The thermal hydraulic design of the core shall establish a thermal hydraulic safety limit for use in evaluating the safety margin relating the consequences of fuel barrier failure to public safety.

3.7.4 Thermal and Hydraulic Limits

3.7.4.1 Requirements for Steady-State Conditions

For purposes of maintaining adequate fuel performance margin during normal steady-state operation, the Minimum Critical Power Ratio (MCPR) must not be less than the required MCPR operating limit, the Average Planar Linear Heat Generation Rate (APLHGR) must be maintained below the required Maximum APLHGR limit (MAPLHGR) and the Linear Heat Generation Rate (LHGR) must be maintained below the required Maximum LHGR limit (MLHGR). The steady-state MCPR, MAPLHGR, and MLHGR limits are determined by analysis of the most severe moderate frequency Abnormal Operational Transients (AOTs) to accommodate uncertainties and provide reasonable assurance that no fuel damage results during moderate frequency AOTs at any time in life.

3.7.4.2 Requirements for Abnormal Operational Transients (AOTs)

The MCPR, MAPLHGR, and MLHGR limits are established such that no safety limit is expected to be exceeded during the most severe moderate frequency AOT event as defined in Chapter 14, "Plant Safety Analysis."

3.7.4.3 Summary of Design Bases

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

3.7.5 Description of Thermal - Hydraulic Design of the Reactor Core

3.7.5.1 Critical Power Ratio

A description of the critical power ratio is provided in Subsection 3.7.7.1, "Critical Power." Criteria used to calculate the critical power ratio safety limit are given in References 32 and 42 for AREVA reload analyses.

3.7.5.2 Average Planar Linear Heat Generation Rate (APLHGR)

Models used to calculate the APLHGR limit are given in Subsection 3.2.5.1, "Evaluation Methods," as pertaining to the fuel mechanical design limits, and in Subsection 6.5.2.1, "Analysis Model," as pertaining to 10 CFR 50, Appendix K limits.

3.7.5.3 Core Coolant Flow Distribution and Orificing Pattern

The flow distribution to the fuel assemblies and bypass flow paths is calculated on the assumption that the pressure drop across all fuel assemblies and bypass flow paths is the same. This assumption has been confirmed by measuring the flow distribution in boiling water reactors (References 2, 3, and 4). The components of bundle pressure drop considered are friction, local, elevation, and acceleration (Subsections 3.7.5.4.1 through 3.7.5.4.4, respectively). Pressure drop measurements made in operating reactors confirm that the total measured core pressure drop and calculated core pressure drop are in good agreement. There is reasonable assurance, therefore, that the calculated flow distribution throughout the core is in close agreement with the actual flow distribution of an operating reactor.

An iteration is performed on flow through each flow path (fuel assemblies and bypass flow paths), which equates the total differential pressure (plenum to plenum) across each path and matches the sum of the flows through each path to the total core flow. The total core flow less the control rod cooling flow enters the lower plenum. A fraction of this passes through various bypass flow paths. The remainder passes through the orifice in the fuel support plate (experiencing a pressure loss) where some of the flow exits through the fit-up between the fuel support and the lower tieplate and through the lower tieplate holes into the bypass flow region. All reload core fuel bundles have lower tieplate holes. The majority of the flow continues through the lower tieplate (experiencing a pressure loss) where some flow exits through the flow path defined by the fuel channel and lower tieplate into the bypass region. This bypass flow is lower for those fuel assemblies with finger springs. The bypass flow paths considered in the analysis and typical values of the fraction of bypass flow through each flow path are given in Reference 5.

Within the fuel assembly, heat balances on the active coolant are performed nodally. Fluid properties are expressed as the bundle average at the particular node of interest. In evaluating fluid properties a constant pressure model is used.

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

3.7.5.4 Core Pressure Drop and Hydraulic Loads

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

3.7.5.4.1 Friction Pressure Drop

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

$$\Delta P_f = \frac{w^2}{2 g_c \rho} \frac{fL}{D_H A_{ch}^2} \Phi_{TPF}^2$$

where

ΔP_f = friction pressure drop, psi

w = mass flow rate

g_c = conversion factor

ρ = average nodal liquid density

D_H = channel hydraulic diameter

A_{ch} = channel flow area

L = incremental length

f = friction factor

Φ_{TPF} = two-phase friction multiplier

AREVA pressure drop methodology is described in References 33, 34, and 35.

3.7.5.4.2 Local Pressure Drop

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

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

$$\Delta P_L = \frac{w^2}{2 g_c \rho} \frac{K}{A^2} \Phi_{TPL}^2$$

where

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

and w, g, and ρ are defined above. For AREVA analyses the Reference 33, 34, and 35 methodologies are used. For advanced spacer designs a quality modifier has been incorporated in the two-phase multiplier to better fit the data. Empirical constants were added to fit the results to data taken for the specific designs of the BWR fuel assembly. These data were obtained from tests performed in single-phase water to calibrate the orifice, the lower tieplate, and the holes in the lower tieplate, and in both single- and two-phase flow, to derive the best fit design values for spacer and upper tieplate pressure drop. The range of test variables was specified to include the range of interest for boiling water reactors. New test data are obtained whenever there is a significant design change to ensure the most applicable methods are used.

3.7.5.4.3 Elevation Pressure Drop

The elevation pressure drop is based on the relation:

$$\Delta P_E = \bar{\rho}\Delta L \frac{g}{g_c} ;$$

$$\bar{\rho} = \rho_f (1-\alpha) + \rho_g \alpha$$

where

ΔP_E = elevation pressure drop, psi

ΔL = incremental length

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

g = acceleration of gravity

α = nodal average void fraction

ρ_f, ρ_g = saturated water and vapor density, respectively

AREVA void fraction models are described in References 33, 34, and 35.

3.7.5.4.4 Acceleration Pressure Drop

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

$$\Delta P_{ACC} = (1-\sigma_A^2)\frac{w^2}{2g_c\rho_f A_2^2}$$

$$\sigma_A = \frac{A_2}{A_1}$$

where

ΔP_{ACC} = acceleration pressure drop

A_2 = final flow area

A_1 = initial flow area

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

$$\Delta P_{ACC} = (1-\sigma_A^2)\frac{w^2\rho_H}{2g_c\rho_{KE}^2 A_2^2}$$

where

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

$$\frac{1}{\rho_{KE}^2} = \frac{x^3}{\rho_g^2\alpha^2} + \frac{(1-x)^3}{\rho_f^2(1-\alpha)^2}$$, kinetic energy density

α = void fraction at A_2

x = steam quality at A_2

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

$$\Delta P_{ACC} = \frac{w^2}{g_c A_{ch}^2}\left[\frac{1}{\rho_{OUT}} - \frac{1}{\rho_{IN}}\right]$$

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

$$\frac{1}{\rho_M} = \frac{x^2}{\rho_g \alpha} + \frac{(1-x)^2}{\rho_f (1-\alpha)}$$

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

3.7.5.5 Correlation and Physical Data

AREVA has obtained substantial amounts of physical data in support of the pressure drop and thermal-hydraulic loads discussed in Subsection 3.7.5.4, "Core Pressure Drop and Hydraulic Loads." Correlations have been developed to fit these data to the formulations discussed.

3.7.5.5.1 Pressure Drop Correlations

AREVA has taken significant amounts of friction pressure drop data in multi-rod geometries representative of BWR plant fuel bundles and correlated both the friction factor and two-phase multipliers on a best fit basis using the pressure drop formulations reported in Subsections 3.7.5.4.1 and 3.7.5.4.2. Tests are performed in single-phase water to calibrate the orifice and the lower tie-plate, and in both single-phase and two-phase flow to arrive at best fit design values for spacer and upper tie-plate pressure drop. The range of test variables is specified to include the range of interest to boiling water reactors. New data are taken whenever there is a significant design change to ensure the most applicable methods are in use at all times.

Applicability of the single-phase and two-phase hydraulic models discussed in Subsections 3.7.5.4.1 and 3.7.5.4.2 have been confirmed by full scale prototype flow testing.

3.7.5.5.2 Void Fraction Correlation

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

3.7.5.5.3 Heat Transfer Correlation

The fuel heat transfer correlations for AREVA reload analyses are described in Reference 37.

3.7.5.6 Thermal Effects of Abnormal Operational Transients

The evaluation of the core's capability to withstand the thermal effects resulting from abnormal operational transients is covered in Chapter 14, “Plant Safety Analysis”.

3.7.5.7 Uncertainties in Estimates

Uncertainties in thermal-hydraulic parameters are considered in the statistical analysis which is performed to establish the fuel cladding integrity safety limit documented in Subsection 3.7.7.1.1, “Fuel Cladding Integrity Safety Limit.”

3.7.5.8 Flux Tilt Considerations

For flux tilt considerations, refer to Subsection 3.6.4.2, “Power Distribution.”

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

3.7.6.1 Power/Flow Operating Map

3.7.6.1.1 Performance Range for Normal Operations

A boiling water reactor must operate within certain restrictions due to pump net positive suction head (NPSH) requirements, overall plant control characteristics, core thermal power limits, etc. A typical operating power-flow map for BFN is shown in Figure 3.7-1. The boundaries on the maps are as follows:

Natural Circulation Line (Line A in Figure 3.7-1)

The operating state of the reactor moves along this line for the normal control rod withdrawal sequence in the absence of recirculation pump operation.

20 Percent Pump Speed Line (Line B in Figure 3.7-1)

The operating state for the reactor follows this line for the normal control rod withdrawal sequence with recirculation pumps operating at approximately 20 percent speed.

100 Percent Rod Line (Line which runs through point E in Figure 3.7-1)

The 100% rod line passes through 100 percent power at 100 percent flow. The operating state for the reactor follows this line (or one roughly parallel to it) for recirculation flow changes with a fixed control rod pattern. The line is based on constant xenon concentration.

APRM Rod Block Line (Shown in Figure 3.7-1)

The line shown on the graph limits control rod withdrawal to within the constraint of the control rod block line.

Pump Constant Speed Line

The lines passing through Points H and I and Points G and J show the change in flow associated with power reduction by control rod insertion from 3458 MWt, 105% core flow and 3458 MWt, 100% core flow, respectively, while maintaining constant recirculation pump speed.

Minimum Expected Flow Control Line (Shown in Figure 3.7-1)

This line approximates the expected flow control line that the plant would follow if core flow were rapidly reduced from 107.7% of rated core flow at 35.4% power by reducing recirculation pump speed to 20% with a constant control rod pattern.

Increased Core Flow (ICF) Region (The region bounded by Points E, G, J, I, H, and F in Figure 3.7-1)

The plant is licensed for Increased Core Flow (ICF) operation up to a maximum of 105% of rated core flow at 100% rated power. At core thermal powers less than rated but greater than or equal to 3458 MWt, the maximum allowable core flow is limited to 105% of rated core flow. At thermal powers less than 3458 MWt, the maximum core flow is set by the constant recirculation pump speed line that passes through Point H, up to a maximum core flow of 112.6% of rated flow at 35.4% rated power on the Power/Flow operating map. ICF can be used to extend full power operation beyond the point where all rods are out at rated power and flow conditions (End Of Full Power Life - EOFPL). ICF may be used prior to reaching EOFPL for operating flexibility.

Maximum Extended Load Line Limit Analysis (MELLLA) Region

The plant is licensed for Maximum Extended Load Line Limit Analysis (MELLLA) which allows operation at 3458 MWt down to 81% rated flow conditions. As shown in Figure 3.7-1, with 120% power uprate to 3952 MWt, MELLLA allows operation at full power down to 99% rated flow conditions. The MELLLA region may be used to set target rod patterns on a higher rod line to accommodate xenon accumulation. The MELLLA region also increases the allowance for flow window operation such as compensating for small core reactivity changes with burnup by adjusting core flow. This reduces the need to continually adjust rod patterns. With the implementation of EPU to allow power operation to a power level of 3952 MWt, the minimum core flow along the MELLL load line is approximately 99% of rated core flow at 100% core thermal power.

3.7.6.1.2 Flow Control

The following simplified description of BWR operation summarizes the principle modes of normal power range operation. Prior to unit startup the recirculation pumps are started one at a time and typically held at a pump speed of 28 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 28 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 19% 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.

3.7.6.2 <u>Thermal-Hydraulic Stability Performance</u>

The AREVA analytical methodology for demonstrating stability compliance for AREVA fuel designs is described in Subsection 3.6.4.6, “Stability.” To provide additional assurance that regional instabilities will not occur, Browns Ferry has implemented the long-term stability solution designated as Option III in NEDO-31960, Supplement 1, “BWR Owners’ Group Long-Term Stability Solution Licensing Methodology.”

For the Option III long-term stability solution, the Oscillation Power Range Monitor (OPRM) Upscale Trip function of the Power Range Neutron Monitoring (PRNM) system is enabled. [Note: See Section 7.5.7.3.5 for a detailed description of the OPRM system.] The OPRM Upscale Trip function provides protection from exceeding the fuel MCPR Safety Limit in the event of thermal-hydraulic power oscillations. The OPRM receives input signals from the Local Power Range Monitors (LPRMs) within the reactor core. An Upscale Trip is issued if oscillatory changes in the neutron flux are detected. The OPRM Upscale Trip function is required to be operable when the plant is in a region of power-flow operation where actual thermal-hydraulic oscillations might occur (Tech Spec enabled region - greater than 23% rated thermal power and less than 60% recirculation drive flow).

A cycle specific Option III stability analysis is performed for each reload core to determine the appropriate OPRM setpoint. The analysis considers both steady state startup operation and the case of a two recirculation pump trip from rated power. The resulting stability based Operating Limit MCPRs as a function of OPRM setpoint are reported in the AREVA Reload Analysis Report (included in Appendix N of the FSAR). The actual OPRM setpoint is selected such that required margin to the MCPR Safety Limit is provided without stability being a limiting event.

If the OPRM trip function should become inoperable, alternate methods to detect and suppress oscillations may be implemented in accordance with the Technical Specifications.

3.7.7 Evaluation

The thermal-hydraulic design of the reactor core and reactor coolant system is based upon an objective of no fuel damage during normal operation or during abnormal operational transients. This design objective is demonstrated by analysis as described in the following sections.

3.7.7.1 Critical Power

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

> Moderate frequency AOTs caused by a single operator error or equipment malfunction shall be limited such that, considering uncertainties in manufacturing and monitoring the core operating state, at least 99.9% of the fuel rods would be expected to avoid boiling transition (Reference 32).

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

3.7.7.1.1 Fuel Cladding Integrity Safety Limit

The generation of the Minimum Critical Power Ratio (MCPR) limit requires a statistical analysis of each reload core near the limiting MCPR condition. The MCPR Fuel Cladding Integrity Safety Limit applies not only for core wide AOTs, but is also applied to the localized rod withdrawal error AOT. The cycle-specific Safety Limit MCPR is derived based on methodology documented in References 32 and 42 for AREVA reload analyses. For AREVA reload analyses, the Reference 43, 44, and 46-47 approved critical power correlations are used as appropriate to specific fuel types.

The resulting safety limit MCPR for each cycle is given in the AREVA Reload Analysis Report for each BFN unit (included in Appendix N of the FSAR).

Statistical Model

The statistical analysis utilizes a model of the BWR core which simulates the process computer function. This code produces a critical power ratio (CPR) map of the core based on inputs of power distribution, flow, and heat balance information. Details of the procedure are documented in References 32 and 42 for AREVA reload analyses. Random Monte Carlo selections of all operating parameters based on the uncertainty ranges of manufacturing tolerances, uncertainties in measurement of core operating parameters, calculational uncertainties, and statistical uncertainty associated with the critical power correlations are imposed upon the analytical representation of the core and the resulting bundle critical power ratios are calculated. Uncertainties typical of AREVA cycle-specific analyses are provided in Reference 39.

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

BWR Statistical Analysis

Statistical analyses are performed for each operating cycle that provide the fuel cladding integrity safety limit MCPR. This safety limit MCPR is derived based on methodology documented in Reference 42.

3.7.7.1.2 MCPR Operating Limit Calculational Procedure

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

Calculational Procedure for AOT Pressurization Events

Core-wide rapid pressurization events (turbine trip w/o bypass, load rejection w/o bypass, feedwater controller failure) are analyzed using the system model COTRANSA2 (Reference 40) for AREVA reload analyses.

The Time Varying Axial Power Shape (TVAPS) calculation is performed by COTRANSA2 (Reference 40) for AREVA reload analyses. The TVAPS is a short time period phenomena that occurs during the control rod scram that terminates an AOT. The analytical procedures used to evaluate the AOTs account for TVAPS either in a bounding manner or explicitly, depending on the AOT and the fuel design.

Calculational Procedure for AOT Slow Events

For AREVA reload analyses, the MICROBURN-B2 (Reference 35) 3-D simulator code is used for quasi-steady-state loss of feedwater heating (LFWH) transients; for transient events that cannot be handled in a quasi-steady-state manner, COTRANSA2 (Reference 40) is used. Inadvertent HPCI startup is not analyzed due to the fact that the core enthalpy change for the event is similar to the loss of feedwater heating event and the severity of the inadvertent HPCI startup is generally bounded by the load reject no bypass and feedwater controller failure events. The loss of feedwater heating event is analyzed each cycle to demonstrate that the loss of feedwater heating event and inadvertent HPCI startup events remain non-limiting. When necessary, it is analyzed using COTRANSA2 for AREVA analyses.

Rod Withdrawal Error Calculational Procedure

The reactor core behavior during the rod withdrawal error transient is calculated by doing a series of steady-state three-dimensional coupled nuclear-thermal-hydraulic calculations using the 3-D BWR Simulator (Reference 35).

Event Descriptions

For AREVA analyses, the AOT descriptions in Chapter 14 of this UFSAR are utilized with appropriate cycle-specific input parameter updates.

MCPR Operating Limit Calculation

For AREVA reload analyses, the ΔCPR for rapid AOTs is calculated using XCOBRA (Reference 41) for the initial steady-state analysis and XCOBRA-T (Reference 37) for the transient thermal margin analysis of the limiting fuel assembly.

MCPR Uncertainty Considerations for AREVA Reload Analyses

For fast transient events, the one-dimensional kinetic thermal-hydraulic COTRANSA2 code is used for the reactor system analysis, with the XCOBRA/XCOBRA-T codes evaluating the initial and transient hot channel hydraulics and ΔCPR. The NRC approved application methodology of References 40, 41, and 42 provides adequate conservatism by accounting for uncertainties in the computed ΔCPR results. Therefore, for a given transient event, the required operating limit MCPR (OLMCPR) is simply the XCOBRA-T calculated ΔCPR added to the safety limit MCPR (SLMCPR).

The results of the system pressurization transients are sensitive to the control rod scram speed used in the calculations. To take advantage of average scram speeds faster than those associated with the technical specifications surveillance times, scram speed-dependent MCPR limits are provided. If the control rod scram time performance is equal or better than the nominal scram speed (NSS) insertion times specified in the core operating limits report (COLR), the NSS-based MCPR operating limits apply. If the control rod scram time performance is equal or better than the optimal scram speed (OSS) insertion times specified in the COLR, the OSS-based MCPR operating limits apply. Otherwise, MCPR operating limits are applied that are based on the technical specification scram speed (TSSS) control rod insertion times. The plant technical specifications allow for operation with a certain number and arrangement of “slow” control rods as well as one stuck control rod. Conservative adjustments to the OSS, NSS, and TSSS scram speeds are input to the reload transient analyses to account for these slow and stuck rod effects on scram reactivity.

TSSS, NSS, and OSS-based power-dependent MCPR operating limits are reported in the reload licensing analysis report.

Low Flow and Low Power Effects on MCPR

The operating limit MCPR must be increased for low flow and, for plants with ARTS, low power conditions. This is because, in the BWR, power increases as core flow increases which results in a corresponding lower MCPR. If the MCPR at a reduced flow condition were at the 100% power and flow MCPR operating limit, a sufficiently large inadvertent flow increase could cause the MCPR to decrease below the Fuel Cladding Integrity Safety Limit MCPR.

The Average Power Range Monitor, Rod Block Monitor, and Technical Specification (ARTS) Improvement Program imposes both power- and flow-dependent limits imposed on the operating limit MCPR (OLMCPR). The flow-dependent required OLMCPR, $MCPR_f$, is defined as a function of the core flow rate and maximum rated power core flow capability. For powers between 100% of rated and the bypass point for the turbine stop valve/turbine control valve fast closure scram signal (about 26% of rated), the power-dependent OLMCPR, $MCPR_p$, is directly supplied. For powers between 23% rated and the bypass point, the $MCPR_p$ limits are absolute values and are defined separately for high core flows (>50% of rated flow) and for low core flows (≤50% of rated flow) conditions. There is no thermal limits monitoring required below 23% of rated power. The OLMCPR to be used at powers less than 100% becomes the most limiting value of either $MCPR_f$ or $MCPR_p$.

<u>End-of-Cycle Coastdown Considerations</u>

AOT analyses are performed at the full power, EOC, all-rods-out condition. Once an individual plant reaches this condition, it may shutdown for refueling or it may be placed in a coastdown mode of operation. In the end-of-cycle coastdown type of operation, if coastdown is initiated from a reactor power level greater than 3458 MWt, recirculation pump speed must be decreased as reactor power coasts down in order to maintain core flow less than or equal to 105% of rated. Once reactor power is less than or equal to 3458 MWt, the plant is allowed to coastdown to a lower percent of rated power while maintaining recirculation pump speed less than or equal to the constant pump speed line corresponding to 105% of rated core flow at 3458 MWt.

For GE methods, in Reference 29, evaluations were made at 90%, 80%, and 70% power level points on the linear curve. The results show that the pressure and MCPR from the limiting pressurization AOT exhibit a larger margin for each of these points than the EOC full power, full flow case. MLHGR limits for the full power, rated flow case are conservative for the coastdown period, since the power will be decreasing and rated core flow will be maintained. Therefore, it can be concluded that the coastdown operation beyond full power operation is conservatively bounded by the analysis at the EOC conditions. In Reference 30, this conclusion is confirmed for coastdown operation down to 40% power and is shown to hold for analyses performed with ODYN.

For AREVA methods, the nominal start of coastdown cycle exposure is conservatively extended. Coastdown limits are then determined at the final cycle exposure bounding of anticipated operation, forming the licensing basis maximum core average exposure (CAVEX).

3.7.7.2 Analysis Options

3.7.7.2.1 MCPR Margin Improvement Options

Several MCPR margin improvement options have been developed for operating BWRs. The following options are utilized at Browns Ferry:

(1) Recirculation Pump Trip
(2) Thermal Power Monitor
(3) Exposure-Dependent Limits
(4) Improved Scram Times

The exposure-dependent limits option is used on an as-needed basis. AREVA Reload Analysis Report, for each unit indicates which options are currently analyzed.

Recirculation Pump Trip

Due to operator concerns associated with the difficulty of restarting the recirculation pumps following a reactor scram, the recirculation pump trip (RPT) feature is maintained in a bypass condition.

For many of the plant operating cycles, the limiting AOTs are the turbine trip, generator load rejection, or other AOTs which result in a turbine trip. A significant improvement in thermal margin can be realized if the severity of these transients is reduced. The RPT feature accomplishes this by cutting off power to the recirculation pump motors anytime that the turbine control valve or turbine stop valve fast closure occurs.

This rapid reduction in recirculation flow increases the core void content during the AOT, thereby reducing the peak AOT power and heat flux.

Basically, the RPT consists of switches installed in both the turbine control valves and the turbine stop valves. When these valves close, breakers are tripped which releases the recirculation pumps to coast down under their own inertia.

Thermal Power Monitor

The APRM simulated thermal power trip (APRM thermal power monitor) is a minor modification to the APRM system. The modified APRM system generates two upscale trips. On one, the APRM signal (which is proportional to the thermal neutron flux) is compared with a reference which is not dependent on flow rate.

During normal reactor operations, neutron flux spikes may occur due to conditions such as transients in the recirculation system, transients during large flow control load maneuvers, or transients during turbine stop valve tests. The neutron flux leads the heat flux during transients because of the fuel time constant. And the neutron flux for these transients trips upscale before the heat flux increases significantly. (High heat flux is the precursor of fuel damage.) Thus, increased availability can be achieved without fuel jeopardy by adding a trip dependent on heat flux (thermal power).

For this trip, the APRM signal is passed through a low pass RC filter. It is compared with a recirculation flow rate dependent reference which decreases approximately parallel to the flow control lines.

In addition to increased availability, another benefit is that with the minor operational spikes filtered out, the heat flux trip setpoint is lower than the neutron flux trip setpoint. For long, slow AOTs such as the loss-of-feedwater heater, the heat flux and neutron flux are almost in equilibrium. For these AOTs, the lower trip setpoint results in an earlier scram with a smaller increase in heat flux and a corresponding reduction in the consequences.

The APRM Simulated Thermal Power Trip at Browns Ferry is non-safety grade and is not taken credit for in any of the licensing transient analyses.

Exposure-Dependent Limits

The severity of any plant AOT pressurization event is worst at the end of the cycle primarily because the EOC all-rods-out scram curve gives the worst possible scram response. It follows that some limits relief may be obtained by analyzing the AOTs at other interim points in the cycle and administering the resulting limits on an “exposure dependent” basis.

This technique is straightforward and consists merely of repeating certain elements of the AOT analyses for selected midcycle exposures. Because the scram reactivity function monotonically deteriorates with exposure (after the reactivity peak), the limit determined for an exposure E_i is administered for all exposures in the interval $E_{i-1} < E \leq E_i$ where E_{i-1} is the next lower exposure point for which a limit was determined. This results in a table of MCPR limits to be applied through different exposure intervals of the cycle.

Improved Scram Times

As described in Section 3.7.7.1.2, subsection titled “MCPR Uncertainty Considerations for AREVA Reload Analyses”, for AREVA reload analyses power-dependent MCPR limits are provided for OSS, NSS, and TSSS bases. OSS and NSS bases limits may be used depending on scram speed measurements. Otherwise, the TSSS MCPR limits are applied. The TSSS and NSS-based power-dependent MCPR operating limits are reported in the reload licensing analysis report.

3.7.7.2.2 Operating Flexibility Options

A number of operating flexibility options have been developed for BWRs. The following options are utilized at Browns Ferry:

(1) Maximum Extended Load Line Limit
(2) Increased Core Flow
(3) Feedwater Temperature Reduction
(4) Turbine Bypass Out of Service
(5) ARTS Program
(6) Recirculation Coolant Pump Out of Service (Single-Loop Operation)
(7) End-of-Cycle Recirculation Pump Trip Out of Service
(8) Power Load Unbalance Out of Service

The AREVA Reload Licensing Report for each unit indicates which options are currently analyzed (included in Appendix N of the FSAR).

Maximum Extended Load Line Limit

The Maximum Extended Load Line Limit Analysis (MELLLA) expands the operating domain to allow operation at 3458 MWt down to 81% rated flow conditions and at 3952 MWt down to 99% rated flow conditions. Addition of the MELLLA region provides improved power ascension capability to full power and additional flow range at rated power. Evaluations performed for MELLLA conditions include normal and AOTs, LOCA analysis, containment responses, and stability. The reload fuel dependent results of these analyses are re-evaluated each cycle.

Increased Core Flow Operation

Analyses are performed in order to justify operation at core flow rates in excess of the 100% rated flow condition. The limiting AOTs that are analyzed at rated flow as part of a standard supplemental reload licensing report are reanalyzed for increased core flow operation. In addition, the loss-of-coolant accident (LOCA), fuel loading error, rod drop accident, and rod withdrawal error are also re-evaluated for increased flow operation.

The effects of the increased pressure differences on the reactor internal components, fuel channels, and fuel bundles as a result of the increased flow are analyzed in order to ensure that the design limits will not be exceeded.

The thermal-hydraulic stability is re-evaluated for increased core flow operation, and the effects of flow-induced vibration are also evaluated to assure that the vibration criteria will not be exceeded.

Feedwater Temperature Reduction

Analyses are performed in order to justify operation at a reduced feedwater temperature at rated thermal power. Usually, the analyses are performed for end-of-cycle operation with the last-stage feedwater heaters valved out of service. However, throughout cycle operation, an additional feedwater temperature reduction can be justified by analyses at the appropriate operating conditions.

The limiting AOTs are reanalyzed for operation at a reduced feedwater temperature. In addition, the loss-of-coolant accident (LOCA), fuel loading error, rod drop accident, and rod withdrawal error are also re-evaluated for operation at a reduced feedwater temperature.

Turbine Bypass Out of Service

Operation of the turbine bypass system is assumed in the analysis of the Feedwater Controller Failure (FWCF)-maximum demand event. If this event is limiting or near limiting, the operating limit MCPR basis may be invalid if the bypass system cannot be demonstrated as fully functional. Reload specific evaluations may incorporate a FWCF without credit for bypass operation calculation as a provision when temporary factors render the system unavailable. Additionally, for extended operation with degraded bypass system operation, evaluations in support of this condition are augmented with the appropriate limiting events, such as the FWCF, for the applicable cycle.

ARTS Program

The ARTS program is a comprehensive project involving the Average Power Range Monitor (APRM), the Rod Block Monitor (RBM), and Technical Specification improvements.

Implementing the ARTS program provides for the following improvements which enhance the flexibility of the BWR during power level monitoring.

(1) The Average Power Range Monitor (APRM) trip setdown requirement is replaced by power-dependent and flow-dependent MCPR operating limits to reduce the need for manual setpoint adjustments. In addition, another set of power- and flow-dependent limits (LHGR and/or MAPLHGR) are also specified for more rigorous fuel thermal protection during postulated transients at off-rated conditions. These power- and flow-dependent limits are verified for plant-specific application during the initial ARTS licensing implementation. For AREVA reload analyses, the power-and flow-dependent limits are reviewed and updated as needed each cycle.

(2) The RBM system is modified from flow-biased to power-dependent trips to allow the use of a new generic non-limiting analysis for the Rod Withdrawal Error (RWE) and to improve response predictability to reduce the frequency of nonessential alarms. For AREVA reload analyses, the RWE analysis with ARTS is re-evaluated each cycle.

The resulting improvements in the flexibility of the BWR provided by ARTS are designed to significantly minimize the time to achieve full power from startup conditions.

<u>Recirculation Coolant Pump Out of Service</u>

The plant is licensed to allow extended Single-Loop Operation (SLO). The capability of operating at reduced power with a single recirculation loop is highly desirable, from a plant availability/outage planning standpoint, in the event maintenance of a recirculation pump or other components renders one loop inoperative. SLO analyses evaluate the plant for continuous operation at a maximum expected power output.

To justify SLO, safety analyses have to be reviewed for one-pump operation. The MCPR fuel cladding integrity safety limit, AOT analyses, operating limit MCPR, and non-LOCA accidents are evaluated. Increased uncertainties in the total core flow and traversing incore probe (TIP) readings result in a small increase in the fuel cladding integrity safety limit MCPR.

SLO can also result in changes to plant response during a LOCA. These changes are accommodated by the application of reduction factors to the two-loop operation LHGR and/or MAPLHGR limits, if required. Reduction factors are evaluated on a plant and fuel type dependent basis. In each subsequent reload, reduction factors are checked for validity and, if new fuel types are added, new reduction factors may be needed in order to maintain the validity of the SLO analysis.

End-of-Cycle Recirculation Pump Trip Out of Service

The EOC-RPT-OOS contingency mode of operation eliminates the automatic recirculation pump trip signal when turbine trip or load rejection occurs. As such, the core flow decreases at a slower rate following the recirculation pump trip due to the anticipated transient without scram (ATWS) high pressure recirculation system trip, thus, increasing the severity of the transient responses.

Power Load Unbalance Out of Service

The load rejection event scenario depends on whether the initial power level is sufficient for the PLU feature to operate. The PLU causes fast closure of the TCV. If the PLU does not operate as the result of a load rejection, the TCV closes at the maximum demand rate for speed control (servo mode). For the PLUOOS analysis, the PLU is assumed inoperable at any power level and does not initiate fast TCV closure.

3.7.7.3 Core Hydraulics

Core hydraulics models and correlations are discussed in Subsection 3.7.5, “Description of Thermal-Hydraulic Design of the Reactor Core”.

3.7.7.4 Influence of Power Distributions

The influence of power distributions on the thermal-hydraulic design is discussed in Reference 41.

3.7.7.5 Core Thermal Response

The thermal response of the core for accidents and expected AOT conditions is given in Chapter 14, “Plant Safety Analysis”.

3.7.7.6 Analytical Methods

The analytical methods, thermodynamic data, and hydrodynamic data used in determining the thermal and hydraulic characteristics of the core are documented in Subsection 3.7.7.1.2, “MCPR Operating Limit Calculational Procedure.”

3.7.8 References

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BFN-28

Figure 3.7-1

OPERATING MAP
BFN

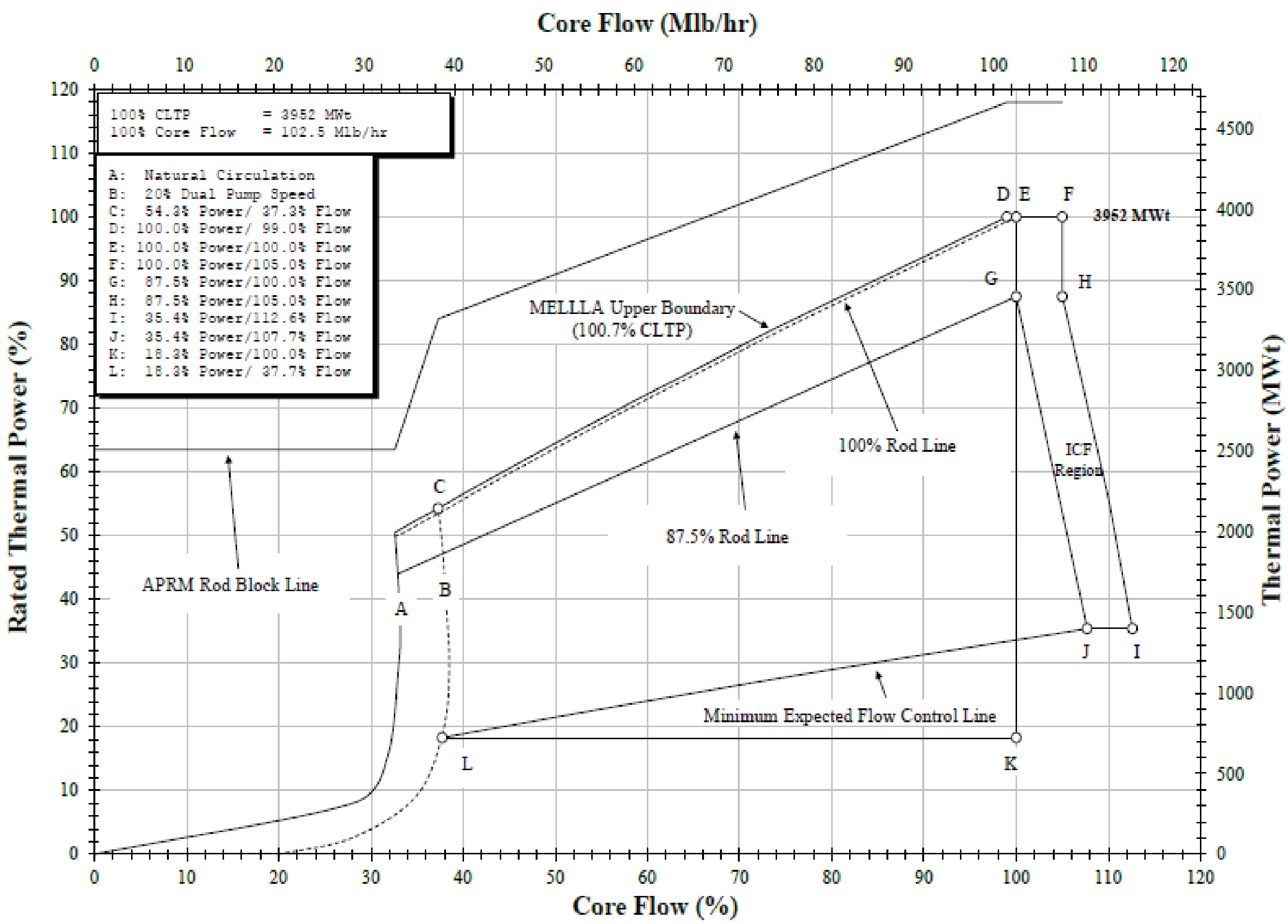

Figure 3.7-2

OPERATING MAP
BFN UNIT 2

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BFN-28

Figure 3.7-3

OPERATING MAP
BFN UNIT 3

DELETED BY AMENDMENT 28

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 independent of the control rods, to make the reactor subcritical over its full range of operating conditions and provide sufficient buffering agent to maintain the suppression pool pH at or above 7.0 following a DBA LOCA involving fuel damage (see Section 14.6.3.5). Making the reactor subcritical is essential to permit the nuclear system to cool to the point where corrective actions can be carried out. Maintaining the suppression pool pH at or above 7.0 following a LOCA involving fuel damage supports the LOCA radiological dose analyses that do not consider the re-evolution of iodine to the containment atmosphere. The Standby Liquid Control System is classified as a special safety system.

3.8.2 Safety Design Basis

1. Backup capability for reactivity control shall be provided, independent of normal reactivity control provisions in the nuclear reactor, to shut down the reactor if the normal control is impaired so that cold shutdown (MODE 4) cannot be obtained with control rods alone.

2. The backup system shall have the capacity for controlling the reactivity difference between the steady-state rated operating condition of the reactor and the cold shutdown condition (MODE 4), 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 reactivity control shall be consistent with the nuclear reactivity rate of change predicted between rated operating and cold shutdown conditions (MODE 4). A scram of the reactor or operational control of fast reactivity transients is not specified to be accomplished by this system.

4. Means shall be provided by which the functional performance capability of the system components can be verified periodically under conditions approaching actual use requirements. Demineralized water, rather than the actual neutron absorber solution, is injected into the reactor to test the operation of all components of the redundant control system.

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

6. The system shall be reliable to a degree consistent with its role as a special safety system.

7. The possibility of unintentional or accidental shutdown of the reactor by this system shall be minimized.

8. The system shall be capable of supplying buffering agent to the suppression pool in the event of a large recirculation break. Sufficient buffering agent shall be provided to ensure that the pH of the suppression pool for DBA post-LOCA events involving fuel damage remains at or above 7.0 for 30 days.

3.8.3 Description (Figures 3.8-1, 3.8-2, 3.8-3, 3.8-5, and 3.8-6)

The Standby Liquid Control System is manually initiated from the Main Control Room to pump a boron neutron absorber solution into the reactor if:

1. The operator determines the reactor cannot be shut down or kept shut down with the control rods; or

2. Fuel damage occurs post-LOCA.

The Standby Liquid Control System is required to shut down the reactor at a steady rate within the capacity of the shutdown cooling systems and to keep the reactor from going critical again as it cools.

The Standby Liquid Control System is needed in the improbable event that not enough control rods can be inserted in the reactor core to accomplish subcriticality in the normal manner.

The Standby Liquid Control System is also required to supply sodium pentaborate solution for post-LOCA events that involve fuel damage to maintain the suppression pool pH at or above 7.0. The radiological dose analyses for the DBA LOCA assumes concentrations of iodine species consistent with a suppression pool pH at or above 7.0 (i.e., re-evolution of iodine to the containment atmosphere is not considered). The sodium pentaborate solution is credited as a buffering agent to offset the post-LOCA production of acids (e.g., radiolysis products).

The system consists of a boron solution tank, a test water tank, two positive-displacement pumps, two explosive-actuated valves, and associated local valves and controls. They are mounted in the Reactor Building outside the primary containment. The liquid is piped into the reactor vessel via the differential pressure and liquid control line and discharged near the bottom of the core lower support plate through a standpipe so it mixes with the cooling water rising through the core (see Sections 4.2, "Reactor Vessel and Appurtenances Mechanical Design," and 3.3, "Reactor Vessel Internals Mechanical Design").

The Boron-10 isotope absorbs thermal neutrons and thereby terminates the nuclear fission chain reaction in the uranium fuel.

The specified neutron absorber solution is enriched sodium pentaborate ($Na_2B_{10}0_{16}$-$10H_20$). It consists of a mixture of borax, enriched boric acid, and demineralized water prepared in accordance with approved plant procedures to ensure the proper volume and enriched sodium pentaborate concentration is present in the standby liquid control tank. A sparger is provided in the tank for mixing, using air. To prevent system plugging, the tank outlet is raised above the bottom of the tank and is fitted with a strainer.

At all times when it is possible to make the reactor critical, the configuration of the Standby Liquid Control System shall satisfy the following equation:

$$\frac{(C)\ (Q)\ (E)}{(8.7\ \text{WT\%})\ (50\ \text{GPM})\ (94\ \text{ATOM\%}\ )} \geq 1.0$$

C = sodium pentaborate solution weight percent concentration
Q = SLCS pump flow rate in gpm
E = Boron-10 atom percent enrichment in the sodium pentaborate solution

The SLC system is used to control the Suppression Pool pH in the event of a DBA LOCA by injecting sodium pentaborate into the reactor vessel. The solution is then transported to the suppression pool by mixing with the ECCS flow circulating through the reactor and flowing out of the recirculation break into the suppression chamber. The amount of sodium pentaborate solution that must be available for injection following a DBA LOCA is determined as part of the DBA LOCA radiological dose analysis. This quantity is maintained in the storage tank as specified in the Technical Specifications.

The solution concentration is normally limited to a maximum of 9.2 weight percent to preclude unwanted precipitation of the sodium pentaborate. The saturation temperature of the 9.2 percent solution is 40°F which provides a 10°F thermal margin below the lowest temperature predicted for the SLCS equipment area. Tank heating components provide backup assurance that the sodium pentaborate solution temperature will never fall below 50°. The sodium pentaborate solution concentration is allowed to be >9.2 weight percent provided the concentration and temperature of the solution are within the limits permitted by the technical specifications. High or low temperature, high or low liquid level, or a shorted heater causes an alarm in the control room. Tank level indication is also provided in the control room.

Each positive displacement pump was originally sized to inject sodium pentaborate into the reactor in 50 to 125 minutes (approximately 50 gpm), depending on the amount of solution in the tank, at the reactor vessel maximum operating pressure. The minimum quantity of enriched sodium pentaborate is injected when required in less than 2 hours. The pump and system design pressure is 1500 psig. The two relief valves are set at approximately 1425 psig to exceed the reactor operating pressure by a sufficient margin to avoid valve leakage. 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 discharge piping near each relief valve to dampen pulsations from the pumps to protect the system.

Unit 1 is equipped with a maintenance-free suction accumulator at the SLC pump-inlet flange to provide suction stabilization and protect the system.

Unit 2 is equipped with a maintenance-free suction accumulator at the SLC pump-inlet flange to provide suction stabilization and protect the system.

Unit 3 is equipped with a maintenance-free suction accumulator at the SLC pump-inlet flange to provide suction stabilization and protect the system.

The two explosive-actuated injection valves provide high assurance of opening when needed and ensure that the boron solution will not leak into the reactor even when the pumps are being tested. The valves have a demonstrated firing reliability in excess of 99.99 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 with 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 with dual ignition primers 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 circuit opens. Indicator lights show which primer circuit is opened. To service a valve after firing, a 6-inch length of pipe (spool piece) must be removed immediately upstream of the valve to gain access to the shear plug.

The Standby Liquid Control System is actuated by a five-position spring return to "normal" keylock switch located on the control room console. The keylock feature ensures that switching from the "stop" position is a deliberate act (safety design basis 7). Momentarily placing the switch to either "start A" or "start B" position starts the respective injection pump, opens both explosive valves, and closes the Reactor Water Cleanup System isolation valves to prevent loss or dilution of the boron solution.

A green light in the control room indicates that power is available to the pump motor contactor, but that the contactor is open (pump not running). A red light indicates the contactor is closed (pump running). A white light indicates that the motor has tripped or the local handswitch is in the test position.

A red light beside the switch turns on when liquid is flowing through an elbow style flow meter and associated flow indicating switch downstream of the explosive valves. If the flow light or pump lights indicate that the liquid may not be flowing, the operator can immediately turn the switch to the other side, which actuates the alternate pump. Crosspiping and check valves assure a flow path through either pump and either explosive valve. The chosen pump will start even though its local switch at the pump is in the "stop" position for test or maintenance. Pump discharge pressure indication is also provided in the control room.

Equipment drains and tank overflow are piped not to the waste system but to separate containers (such as 55-gallon drums) that can be removed and disposed of independently to prevent any trace of the boron solution 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 heater status.

3.8.4 Safety Evaluation

3.8.4.1 Reactivity Control

The Standby Liquid Control System is a special safety system not required for normal plant operation, and is never expected to be needed for reactor shutdown because of the large number of control rods available to shut down the reactor.

The system is designed to make the reactor subcritical from rated power to a cold shutdown (MODE 4) at any time in core life. The reactivity compensation provided will reduce reactor power from rated to the after-heat level and allow cooling the nuclear system to normal 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 the boiling to cold condition, and decreasing control rod worth as the moderator cools. A licensing analysis is performed each cycle to verify adequate SLCS shutdown capacity. The analysis assumes the specified minimum final concentration of boron in the reactor core and allows for calculational uncertainties. The SLCS shutdown capacity is reported in Appendix N.

The specified minimum average concentration of natural boron in the reactor to provide the specified shutdown margin, after operation of the Standby Liquid Control System, is 720 ppm (parts per million). The minimum quantity of sodium pentaborate to be injected into the reactor is calculated based on the required 720 ppm average concentration in the reactor coolant, Boron-10 enrichment, the quantity of reactor coolant in the reactor vessel, recirculation loops, and the entire RHR system in the shutdown cooling mode, at 70°F and reactor normal water level. The result is increased by 25 percent to allow for imperfect mixing, leakage, and volume in other piping connected to the reactor. This minimum concentration is achieved by preparing the solution as defined in paragraph 3.8.3 and maintaining it above saturation temperature. This satisfies safety design basis 5.

Cooldown of the nuclear system will take several hours, at a minimum, 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 coolant temperature cooldown is 100°F per hour. 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 24 hours before the reactor vessel is opened, and much longer to reach room temperature (70°F). The addition of RHR shutdown cooling volume results in the dilution of the dissolved boron. Therefore, the pressure at which RHR shutdown cooling is activated represents the point of maximum reactivity, when the control rods are still withdrawn, and is the point which requires the maximum boron. Analyses are performed to bound the saturation temperature at this pressure using the equivalent of 720 ppm at 70° F. Analyses demonstrating that adequate shutdown capability exists under these conditions ensure that safety design basis 2 is met.

The specified boron injection rate is limited to the range of 7 to 40 ppm per minute change of boron concentration in the reactor pressure vessel and recirculation loop piping water volumes. The lower rate ensures that the boron is injected into the reactor in less than 2 hours, which is considerably faster than the cooldown rate. The upper limit injection rate insures that there is sufficient mixing such that the boron does not recirculate through the core in uneven concentrations which could possibly cause asymmetric power oscillations in the core. This satisfies safety design basis 3.

3.8.4.2 Suppression Pool pH Control

The Standby Liquid Control System is required to supply sodium pentaborate solution for post-LOCA events that involve fuel damage to maintain the suppression pool pH at or above 7.0. The radiological dose analysis for the DBA LOCA assumes concentrations of iodine species consistent with a suppression pool pH at or above 7.0 (i.e., re-evolution of iodine to the containment atmosphere is not considered).

The quantity of sodium pentaborate necessary to offset the post-LOCA production of acid and maintain the suppression pool pH at or above 7.0 has been documented as part of the LOCA radiological dose analysis. This quantity is maintained in the storage tank as specified in the technical specifications. Maintaining the suppression pool pH at or above 7.0 is a concern following a DBA LOCA involving fuel damage. With a LOCA involving a recirculation pipe break, there will be sufficient flow from the ECCS systems through the reactor vessel and out of the break to transport the buffering agent to the suppression pool. The calculation methodology for suppression pool pH control was based on the approach outlined in NUREG-1465 and NUREG/CR-5950. The design inputs were conservatively established to maximize the post-LOCA production of acids and to minimize the post-LOCA production and/or addition of bases. Other design input values such as initial suppression pool volume and pH were selected to minimize the calculated pH.

It is expected that the initial effects on post-accident suppression pool pH will come from rapid fission product transport and the formation of cesium compounds, which would result in increasing the suppression pool pH. However, cesium compounds are not credited in the long-term pH analyses and the determination of the final (30 day) pH value. As radiolytic production of nitric acid and hydrochloric acid proceeds, and these acids are transported to the pool over the first days of the event, the pH would become more acidic.

The buffering effect of sodium pentaborate solution injection within several hours is sufficient to offset the affects of these acids that are transported to the suppression pool. In these events, the addition of a buffering agent to the suppression pool offsets the radiolysis production of acids. This satisfies safety design basis 8.

3.8.4.3 System Safety Evaluation

The Standby Liquid Control System is classified as a special safety system. To assure the availability of the Standby Liquid Control System, two sets of the components required to actuate the pumps and explosive valves are provided in parallel for redundancy (safety design basis 6).

The SLC components required for the performance of the safety-related suppression pool pH control function are qualified for the post-LOCA environmental conditions they will be subjected to during the performance of this function.

The Standby Liquid Control System is designed as a Class I system for withstanding the specified earthquake loadings (see Appendix C). Nonprocess equipment such as the test tank is designed as Class II. The system piping and equipment are designed, installed, and tested in accordance with USAS B31.1.0, Section I.

For the reactivity control function, the Standby Liquid Control System is not required to be designed to meet the single failure criterion because it serves as a backup to the control rods. System reliability is enhanced by providing redundancy of pumps and valves. Hence, redundancy is not required for the tank heater or the heating cable.

For the suppression pool pH control function, the SLC system does not completely meet the single failure criteria with regard to the containment isolation check valves and the main control room selector switch. Although a single failure to open one of the two check valves could prevent SLC injection, the potential for failure is very low based on the quality as established by its procurement as an ASME, Section III, Class 2 safety-related valve, periodic testing and inspection, and historical performance of the component. Also, although a failure of the selector switch in the main control room could prevent either train or both trains of injection from functioning, the switch is a highly reliable component at an accessible location. The switch could be easily replaced or bypassed to start one of the SLC trains if it were to fail.

The Standby Liquid Control System is required to be operable in the event of a station power failure so 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 buses and circuits so that a single failure affecting power supply 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 nominal setting of 1425 psig, to assure solution injection into the reactor at a pressure of at least three percent above the lowest setpoint of the main steam relief valves (1174 psig). The nuclear system is protected from overpressurization during operation of the Standby Liquid Control System positive displacements pumps by the nuclear system main steam relief valves.

For the Standby Liquid Control System suppression pool pH function, this operating condition is consistent with other two train systems. Although the Standby Liquid Control System is not strictly a two train system, active components most susceptible to failure (e.g., pumps and squib valves) are redundant which provides additional assurance that most single failures will not impede the ability of the system to perform its function. Only one of the two standby 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 system pumps are powered by a diesel backed source and are not load shed. The period during which one redundant component upstream of the explosive valves may be out of operation will be 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, together with the fact that nuclear system cooldown takes 10 or more hours while liquid control solution injection takes less than 2 hours. This indicates the considerable time available for testing and restoring the Standby Liquid Control System to operable condition after testing while reactor operation continues. Assurance that the system will still fulfill its function during repairs is obtained by demonstrating operation of the operable pump.

It can be seen that the Standby Liquid Control System satisfies safety design basis 1.

3.8.4.4 Quality Assurance

The equipment that performs the special safety functions of the Standby Liquid Control System (provide a backup method to make the reactor subcritical and provide sufficient buffering agent to maintain the suppression pool pH at or above 7.0 following a DBA LOCA involving fuel damage) are classified as quality related. As delineated by condition of the Units 1, 2, and 3 BFN Operating Licenses, the Augmented Quality Program for the Standby Liquid Control System provides the quality control elements to ensure component reliability for the required alternative source term function as governed by the BFN Quality Assurance Program.

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. By opening two closed valves to the solution tank, the boron solution may be recirculated by turning on either pump with its local switch. With the valves to and from the solution tank closed and the three valves opened to and from the test tank, the demineralized water in the test tank can be recirculated by turning on either pump locally. After pumping boron solution, demineralized water is pumped to flush out the pumps and pipes. Functional testing of the injection portion of the system is accomplished by closing the open valve from the solution tank, opening the closed valve from the test tank, and actuating the switch in the control room to either the A or B circuit. This starts one pump and ignites one of the explosive actuated injection valves to open.

The lights and alarms in the control room indicate that the system is functioning. This satisfies safety design basis 4.

After the functional test, the affected injection valve and explosive charge must be replaced and all the valves returned to their normal positions as indicated in Figures 3.8-1, 3.8-2, 3.8-3, 3.8-5, and 3.8-6.

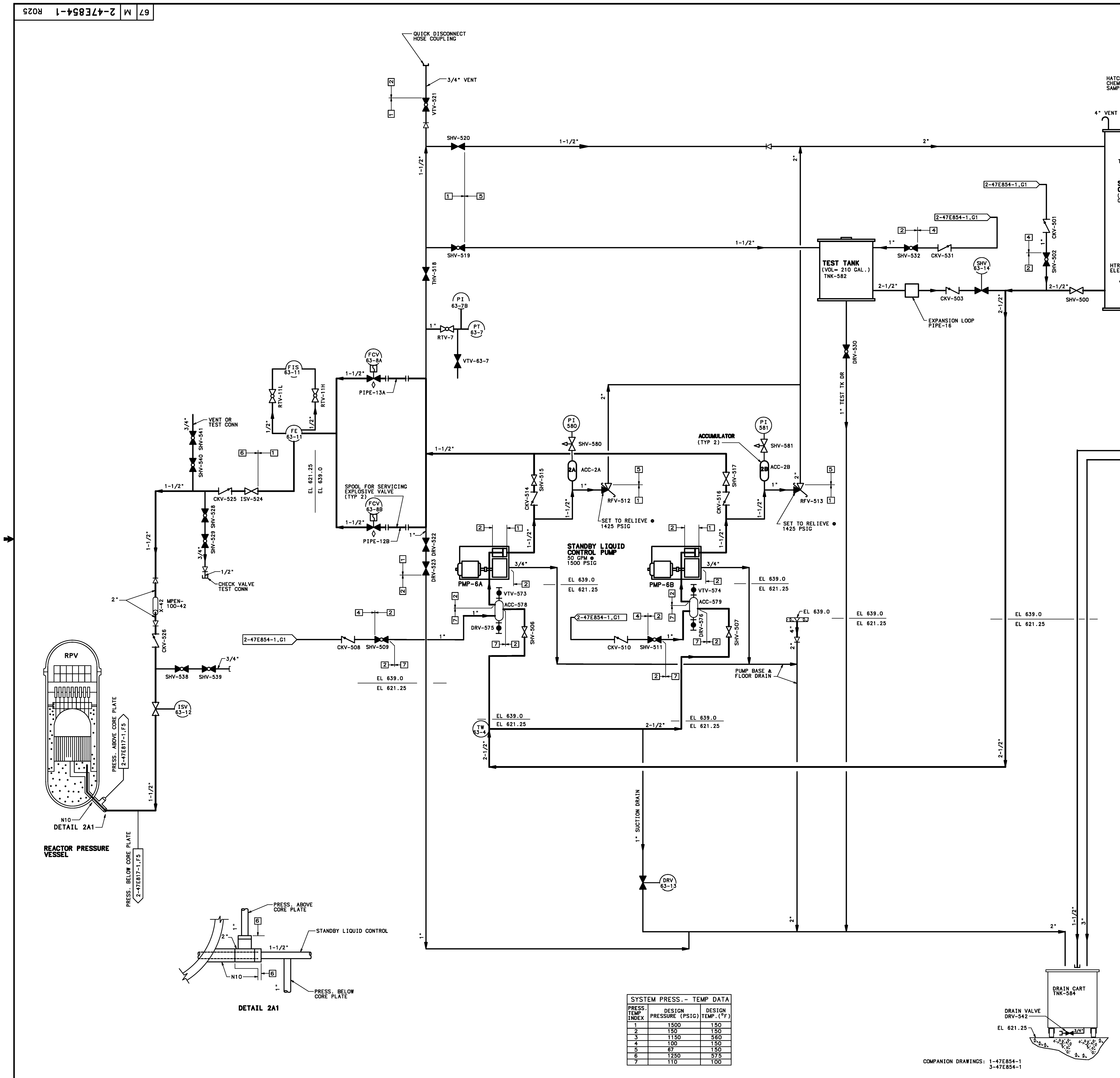
By closing a local normally open valve to the reactor in the containment, leakage through the injection valves can be detected at a test connection in the line between the containment isolation check valves. (A position indicator light in the control room indicates when the local valve is full open and ready for operation.) Leakage from the reactor through the first check valve can be detected by opening the same test connection whenever the reactor is pressurized.

The test tank contains sufficient demineralized water for testing pump operation. Demineralized water from the makeup or condensate storage system is available at 30 gpm for refilling or flushing the system.

Should the boron solution ever be injected into the reactor, either intentionally or inadvertently, then after making certain that the normal reactivity controls will keep the reactor subcritical, the boron is removed from the reactor coolant system by flushing for gross dilution followed by operation of the reactor cleanup system. There is practically no effect on reactor operations when the boron concentration has been reduced below about 50 ppm.

The sodium pentaborate solution weight percent in the SLCS storage tank is periodically determined by titration or equivalent chemical analysis. The Boron-10 isotopic atom percent concentration of the solution is also determined periodically, utilizing mass spectrometry or equivalent technology.

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



NOTES:

1. UNIDS ON DRAWINGS ARE FOR REFERENCE ONLY AND ARE ABBREVIATED AS SHOWN IN THE EXAMPLE TO MEET SPACE CONSTRAINTS. REFER TO MEL FOR COMPLETE UNIDS. ALL UNIDS ARE IN UNIT 2 AND SYSTEM 63 UNLESS OTHERWISE NOTED. LEADING ZEROES SHOWN IN MEL AS PART OF THE UNID ARE NOT DEPICTED. FOR ADDITIONAL GUIDANCE, REFER TO NEDP-4.

   EXAMPLES:

MEL UNID	DRAWING UNID
BFN-0-SHV-018-0502	SHV-18-502
BFN-1-HS-031-0160A	HS-31-160A
BFN-2-PT-003-0204C	PT-3-204C

2. ALL VALVES ARE SAME SIZE AS PIPE UNLESS OTHERWISE NOTED.
3. OPERATIONAL VALVES ARE SHOWN IN THEIR NORMAL OPERATING POSITION.
4. 1, 2, ETC, DENOTE DESIGN PRESSURE AND TEMPERATURE AS GIVEN IN DATA TABLE ON THIS DRAWING.
5. DELETED.
6. DELETED.
7. VENT, DRAIN, AND TEST CONNECTIONS 1-1/2" AND BELOW CAN BE PROVIDED WITH PIPE CAPS OR HOSE CONNECTION FITTINGS WHERE REQUIRED BY PLANT PERSONNEL. THIS CONFIGURATION IS SUPPORTED BY ENGINEERING CALCULATION CD-Q0999-923399.

REFERENCE DRAWINGS:

MEL....................TABULATION VALVE MARKER TAGS
MEL....................TABULATION OF INSTRUMENTS
0-47W462-1, -3........PIPING DRAWINGS
2-47W462-2............PIPING DRAWINGS
0-47W600-56...........MECHANICAL - INSTRUMENTS AND CONTROLS
2-47E610-3-SERIES.....CONTROL DIAGRAM - REACTOR FEEDWATER SYSTEM
2-47E610-63-1.........CONTROL DIAGRAM - STANDBY LIQUID CONTROL SYSTEM
2-47E610-68-SERIES....CONTROL DIAGRAM - REACTOR WATER RECIRCULATION SYSTEM
2-47E610-75-SERIES....CONTROL DIAGRAM - CORE SPRAY SYSTEM
2-47E610-85-SERIES....CONTROL DIAGRAM - CONTROL ROD DRIVE SYSTEM
47E800-1..............FLOW DIAGRAM - GENERAL PLANT SYSTEMS
0-47E800-2............MECHANICAL SYMBOLS AND FLOW DIAGRAM DRAWING INDEX
2-47E817-1............FLOW DIAGRAM - REACTOR WATER RECIRCULATION SYSTEM
0-47E845-2............FLOW DIAGRAM - COMPRESSED AIR-STATION SERVICE
2-47E856-2............FLOW DIAGRAM - DEMINERALIZED WATER

GE DRAWINGS:

919D926...............FUNCTIONAL CONTROL SYSTEM - SLC SYSTEM
22A1315...............DESIGN SPECIFICATION - SLC SYSTEM
22A1315AB.............DESIGN SPECIFICATION - SLC SYSTEM

SYMBOL:

HEATING TAPE

◊ FAIL AS IS

SYSTEM PRESS.- TEMP DATA

PRESS. TEMP INDEX	DESIGN PRESSURE (PSIG)	DESIGN TEMP.(°F)
1	1500	150
2	150	150
3	1150	560
4	100	150
5	67	150
6	1250	575
7	110	100

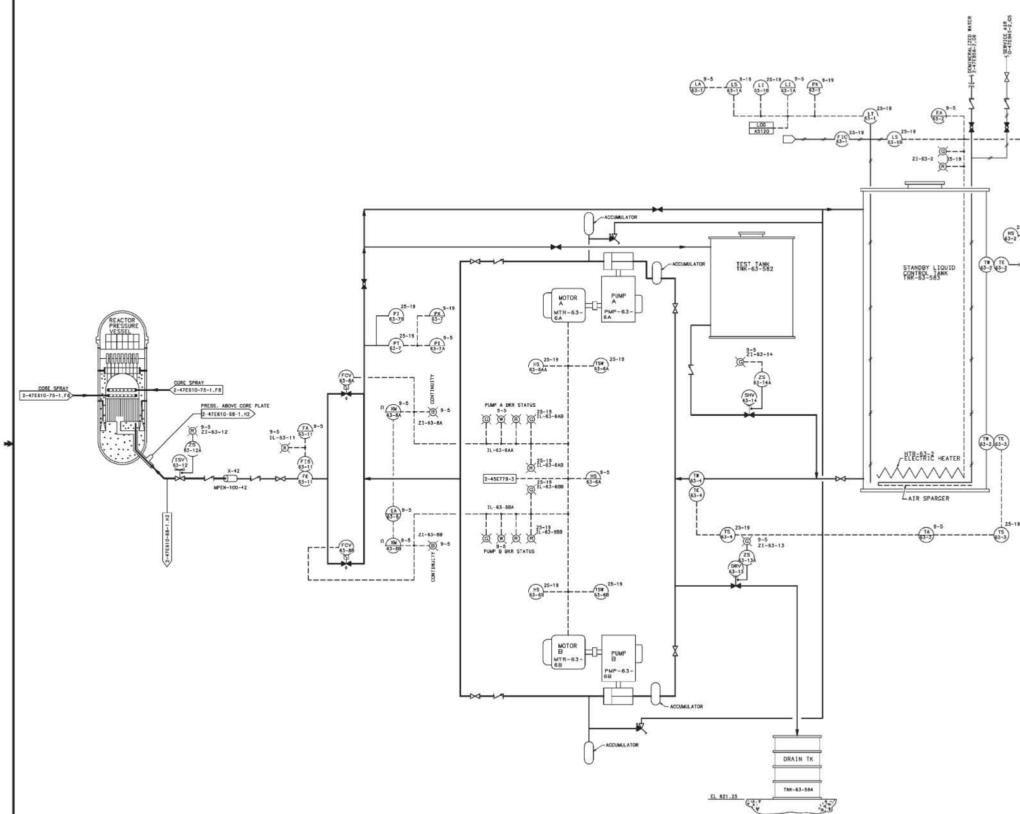
AMENDMENT 28

POWERHOUSE
UNIT 2

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

STANDBY LIQUID CONTROL SYSTEM
FLOW DIAGRAM

FIGURE 3.8-1



NOTES:

1. THE STANDBY LIQUID CONTROL SYSTEM IS USED AS A REDUNDANT, INDEPENDENT BACKUP SYSTEM CAPABLE OF SHUTTING DOWN THE REACTOR AND MAINTAINING IT IN A SHUTDOWN CONDITION IF THE CONTROL ROD DRIVE SYSTEM IS TOTALLY INOPERATIVE. THE SLC SYSTEM ALSO SERVES AS BACKUP REACTIVITY CONTROL DURING REFUELING.
2. A NEUTRON POISON, B10 ENRICHED SODIUM PENTABORATE (EPSB), IS STORED IN THE STANDBY LIQUID CONTROL SYSTEM (SLCS) STORAGE TANK. TIC-63-2 CONTROLS THE TANK HEATER WHICH ACTS AS A BACKUP MEANS FOR ENSURING THAT THE TANK SOLUTION IS MAINTAINED ABOVE THE ESPB'S SATURATION TEMPERATURE. LEVEL SWITCH LS-63-1B, IS INTERLOCKED WITH THE HEATER CONTACTOR TO DEENERGIZE THE HEATERS ON SLCS TANK LOW LEVEL. TEMPERATURE SWITCHES TS-63-3 AND TS-63-4 ACTIVATE TA-63-3 IN THE MAIN CONTROL ROOM (MCR) IF THE SOLUTION APPROACHES THE SATURATION TEMPERATURE OF THE ESPB OR TO INDICATE HEATER FAILURE MODES. THE LEVEL OF THE ESPB SOLUTION IN THE SLCS STORAGE TANK IS MONITORED BY LT-63-1 AND WITH INDICATION IN THE MCR, LI-63-1A AND LOCALLY ON LI-63-1B. THE LEVEL INDICATION IS UTILIZED TO VERIFY OPERATION OF THE SLCS DURING ACCIDENT AND NORMAL OPERATION. LEVEL ALARM LA-63-1 ACTUATES ON A LOW LEVEL AND HIGH LEVEL FOR INDICATION OF GROSS FAILURE MODES.
3. EXPLOSIVE-TYPE VALVES, FCV-63-8A AND FCV-63-8B, ARE ACTUATED BY THE PUMP CONTROL SWITCH, HS-63-6A, IN THE MAIN CONTROL ROOM. THE EXPLOSIVE VALVES ARE NOT ACTUATED WHEN EITHER LOCAL SWITCH HS-63-6B OR HS-63-6AA IS USED TO START ITS RESPECTIVE PUMP.
4. ACTUATING THE EXPLOSIVE VALVES CLOSES THE REACTOR WATER CLEANUP SYSTEM ISOLATION VALVES VIA AN INTERLOCK
5. THE EXPLOSIVE VALVE AND PUMP CONTROL CIRCUITS ARE CONTINUOUSLY MONITORED FOR CONTINUITY, AND ANY MALFUNCTION ACTUATES ALARM EA-63-8.
6. DELETED.
7. UNIDS ON DRAWINGS ARE FOR REFERENCE ONLY AND ARE ABBREVIATED AS SHOWN IN THE EXAMPLE TO MEET SPACE CONSTRAINTS. REFER TO MEL FOR COMPLETE UNIDS. ALL UNIDS ARE IN UNIT 2 UNLESS OTHERWISE NOTED. LEADING ZEROES SHOWN IN MEL AS PART OF THE UNID ARE NOT DEPICTED. FOR ADDITIONAL GUIDANCE, REFER TO NEDP-4.

EXAMPLE: MEL UNID	DRAWING UNID
BFN-0-SHV-018-0502	SHV-18-502
BFN-1-HS-031-0160A	HS-31-160A
BFN-2-PT-003-0204C	PT-3-204C

8. COMPLETE LOGIC FOR STANDBY LIQUID CONTROL SYSTEM IS SHOWN ON MECHANICAL LOGIC DIAGRAM LISTED IN REFERENCE DRAWINGS.

REFERENCE DRAWINGS:

MEL....................INSTRUMENT TABULATION-STANDBY LIQUID CONTROL SYSTEM
2-47E854-1............FLOW DIAGRAM-STANDBY LIQUID CONTROL SYSTEM
2-47E817-1............FLOW DIAGRAM-NUCLEAR BOILER
47W462-SERIES.........PIPING DRAWINGS
2-47E600-139..........MECH & INSTR CONTROL
47W600-56.............MECH & INSTR CONTROL
2-47E856-2............FLOW DIAGRAM-DEMINERALIZED WATER SYSTEM
0-47E845-2............FLOW DIAGRAM-COMPRESSED AIR-STATION SERVICE
2-47E610-75-1.........CONTROL DIAGRAM-CORE SPRAY SYSTEM
2-47E610-68-1.........CONTROL DIAGRAM-REACTOR RECIRCULATION SYSTEM
0-47E800-2............MECHANICAL SYMBOLS AND FLOW DIAGRAM DRAWING INDEX
47E800-1..............FLOW DIAGRAM-GENERAL PLANT SYSTEMS
2-45E779-3............WIRING DIAGRAMS 480V SD AUX PWR SCHEMATIC
2-47E611-63-1.........MECHANICAL LOGIC DIAGRAM STANDBY LIQUID CONTROL SYSTEM
2-47E611-69-1,-2......MECHANICAL LOGIC DIAGRAM REACTOR WATER CLEANUP SYSTEM

GE DRAWINGS
719E599........PIPING AND INSTRUMENT DIAGRAM-STANDBY LIQUID CONTROL
919D926........FUNCTIONAL CONTROL DIAGRAM-STANDBY LIQUID CONTROL
2-161F282-1....PROCESS DIAGRAM-STANDBY LIQUID CONTROL SYSTEM
719E599........PARTS LIST-STANDBY LIQUID CONTROL SYSTEM
22A1315........DESIGN SPECIFICATION-STANDBY LIQUID CONTROL SYSTEM
22A1315AB......DESIGN SPECIFICATION DATA SHEETS-STANDBY LIQUID CONTROL
2-729E854-1....STANDBY LIQUID CONTROL ELEM DIAGRAM

REFERENCE CALCULATIONS:
BFNAPS7-007........B-10 ENRICHMENT DETERMINATION

SYMBOL:

◇ - FAIL AS IS

67 M 2-47E610-63-1R021

COMPANION DRAWING: 47E610-63-1

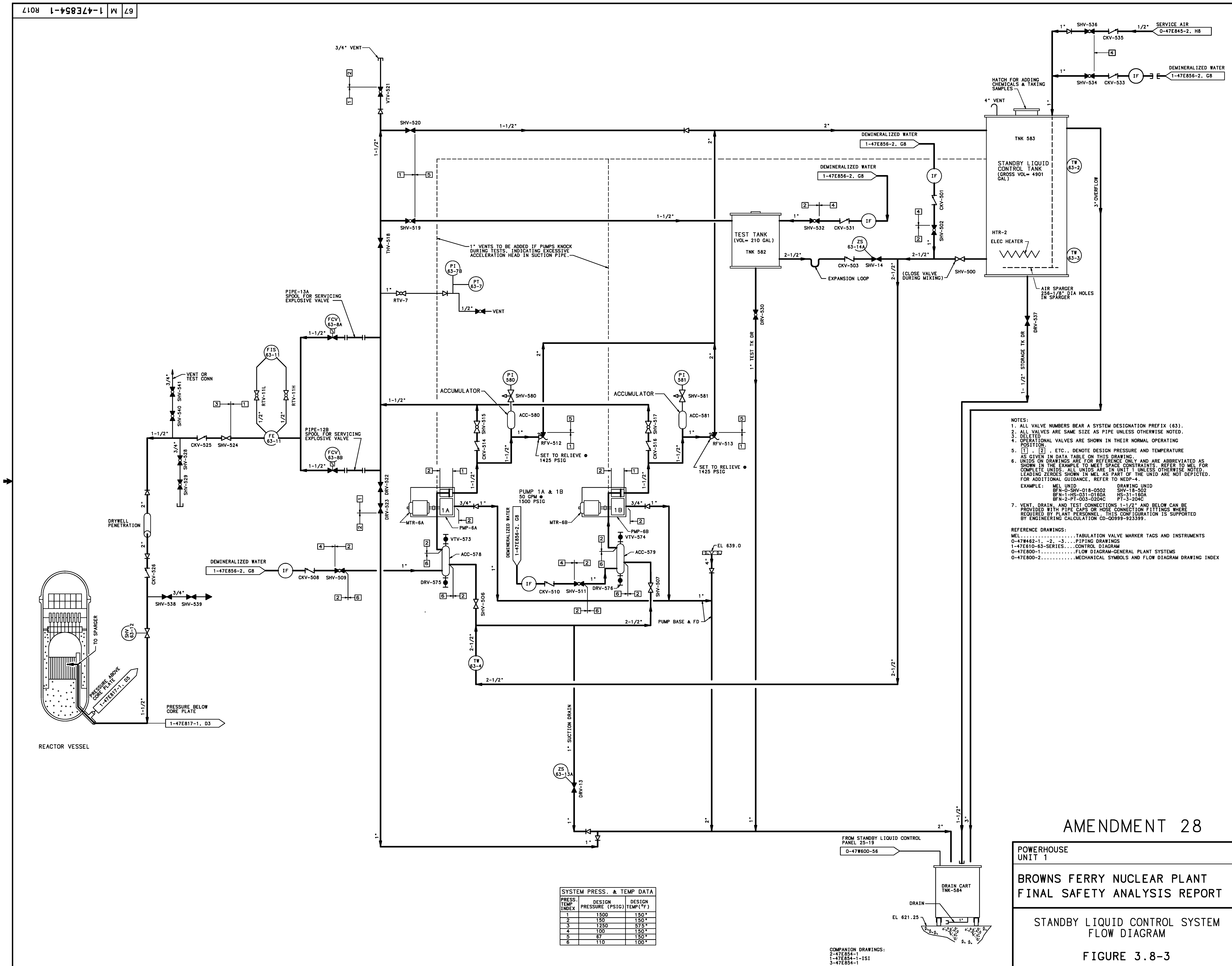
AMENDMENT 27

POWERHOUSE
UNIT 2

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

STANDBY LIQUID CONTROL SYSTEM
MECHANICAL CONTROL DIAGRAM

FIGURE 3.8-2



NOTES:

1. ALL VALVE NUMBERS BEAR A SYSTEM DESIGNATION PREFIX (63).
2. ALL VALVES ARE SAME SIZE AS PIPE UNLESS OTHERWISE NOTED.
3. DELETED
4. OPERATIONAL VALVES ARE SHOWN IN THEIR NORMAL OPERATING POSITION.
5. 1 , 2 , ETC., DENOTE DESIGN PRESSURE AND TEMPERATURE AS GIVEN IN DATA TABLE ON THIS DRAWING.
6. UNIDS ON DRAWINGS ARE FOR REFERENCE ONLY AND ARE ABBREVIATED AS SHOWN IN THE EXAMPLE TO MEET SPACE CONSTRAINTS. REFER TO MEL FOR COMPLETE UNIDS. ALL UNIDS ARE IN UNIT 1 UNLESS OTHERWISE NOTED. LEADING ZEROES SHOWN IN MEL AS PART OF THE UNID ARE NOT DEPICTED. FOR ADDITIONAL GUIDANCE, REFER TO NEDP-4.

EXAMPLE:

MEL UNID	DRAWING UNID
BFN-0-SHV-018-0502	SHV-18-502
BFN-1-HS-031-0160A	HS-31-160A
BFN-2-PT-003-0204C	PT-3-204C

7. VENT, DRAIN, AND TEST CONNECTIONS 1-1/2" AND BELOW CAN BE PROVIDED WITH PIPE CAPS OR HOSE CONNECTION FITTINGS WHERE REQUIRED BY PLANT PERSONNEL. THIS CONFIGURATION IS SUPPORTED BY ENGINEERING CALCULATION CD-Q0999-923399.

REFERENCE DRAWINGS:

MEL....................TABULATION VALVE MARKER TAGS AND INSTRUMENTS
0-47W462-1, -2, -3....PIPING DRAWINGS
1-47E610-63-SERIES....CONTROL DIAGRAM
0-47E800-1............FLOW DIAGRAM-GENERAL PLANT SYSTEMS
0-47E800-2............MECHANICAL SYMBOLS AND FLOW DIAGRAM DRAWING INDEX

SYSTEM PRESS. & TEMP DATA

PRESS. TEMP INDEX	DESIGN PRESSURE (PSIG)	DESIGN TEMP(°F)
1	1500	150°
2	150	150°
3	1250	575°
4	100	150°
5	67	150°
6	110	100°

AMENDMENT 28

POWERHOUSE
UNIT 1

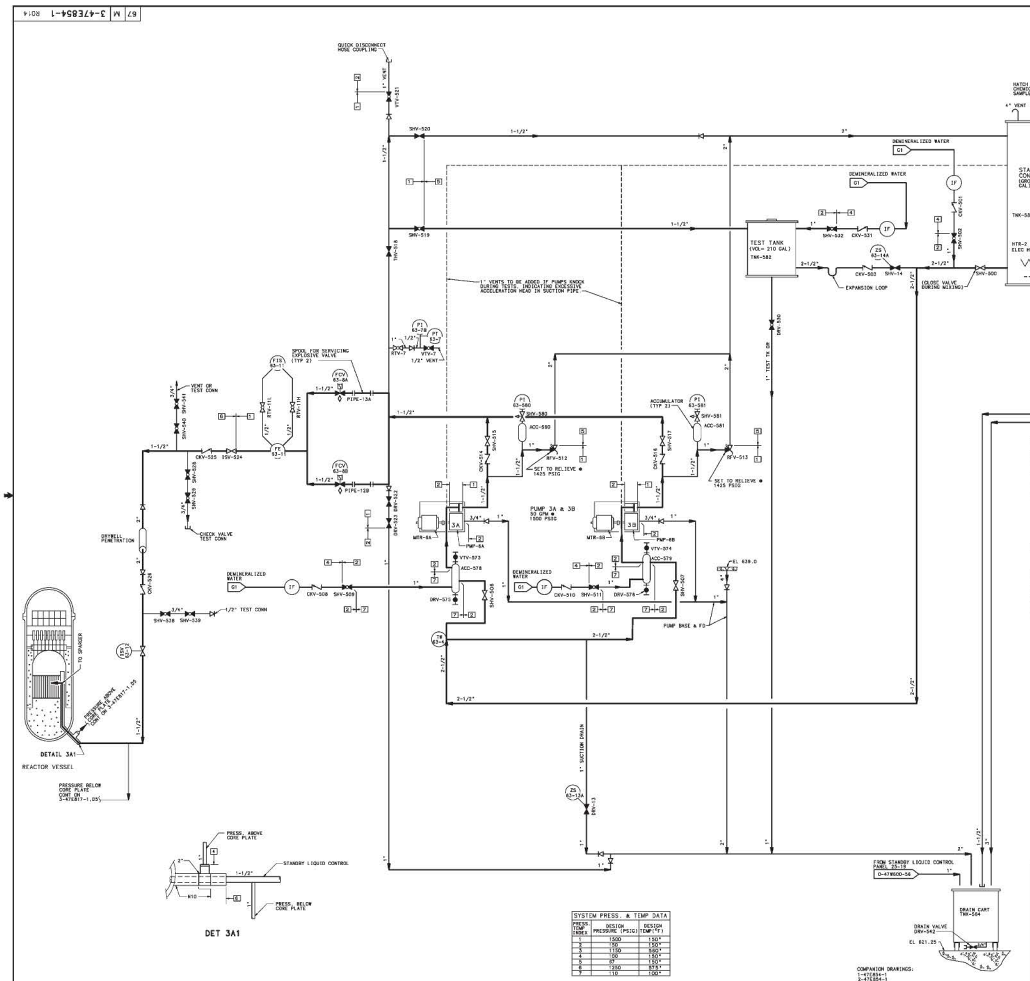
BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

STANDBY LIQUID CONTROL SYSTEM
FLOW DIAGRAM

FIGURE 3.8-3

BFN-22

Figure 3.8-4
(Deleted by Amendment 22)



NOTES:

1. ALL VALVES ARE PREFIXED "3-63" AND ALL INSTRUMENTS ARE PREFIXED "3-" UNLESS OTHERWISE NOTED.
2. ALL VALVES ARE SAME SIZE AS PIPE UNLESS OTHERWISE NOTED.
3. DELETED
4. OPERATIONAL VALVES ARE SHOWN IN THEIR NORMAL OPERATING POSITION.
5. $\boxed{1}$, $\boxed{2}$, ETC., DENOTE DESIGN PRESSURE AND TEMPERATURE AS GIVEN IN DATA TABLE ON THIS DRAWING.
6. DELETED.
7. UNIDS ON DRAWINGS ARE FOR REFERENCE ONLY AND ARE ABBREVIATED AS SHOWN IN THE EXAMPLE TO MEET SPACE CONSTRAINTS. REFER TO MEL FOR COMPLETE UNIDS. ALL UNIDS ARE IN UNIT 3 UNLESS OTHERWISE NOTED. LEADING ZEROES SHOWN IN MEL AS PART OF THE UNID ARE NOT DEPICTED. FOR ADDITIONAL GUIDANCE, REFER TO NEDP-4.

EXAMPLE: MEL UNID	DRAWING UNID
BFN-0-SHV-018-0502	SHV-18-502
BFN-1-HS-031-0160A	HS-31-160A
BFN-2-PT-003-0204C	PT-3-204C

8. VENT, DRAIN, AND TEST CONNECTIONS 1-1/2" AND BELOW CAN BE PROVIDED WITH PIPE CAPS OR HOSE CONNECTION FITTINGS WHERE REQUIRED BY PLANT PERSONNEL. THIS CONFIGURATION IS SUPPORTED BY ENGINEERING CALCULATION CD-Q0999-923399.

REFERENCE DRAWINGS:

MEL......................VALVE MARKER TAG TABULATION
MEL......................INSTRUMENT TABULATION
0-47W462-1, -2, -3....PIPING DRAWINGS
3-47E610-63-1.........MECHANICAL CONTROL DIAGRAM - STANDBY LIQUID CONTROL SYS
47E800-1..............FLOW DIAGRAM - GENERAL PLANT SYSTEMS
0-47E800-2............MECHANICAL SYMBOLS AND FLOW DIAGRAM DRAWING INDEX
0-47W600-56...........MECHANICAL - INSTRUMENTS AND CONTROLS
3-47E817-1............FLOW DIAGRAM - NUCLEAR BOILER
0-47E845-2............FLOW DIAGRAM - COMPRESSED AIR STATION SERVICE
3-47E856-2............FLOW DIAGRAM - DEMINERALIZED WATER

SYSTEM PRESS. & TEMP DATA

PRESS. TEMP INDEX	DESIGN PRESSURE (PSIG)	DESIGN TEMP(°F)
1	1500	150°
2	150	150°
3	1150	560°
4	100	150°
5	67	150°
6	1250	575°
7	110	100°

COMPANION DRAWINGS:
1-47E854-1
2-47E854-1

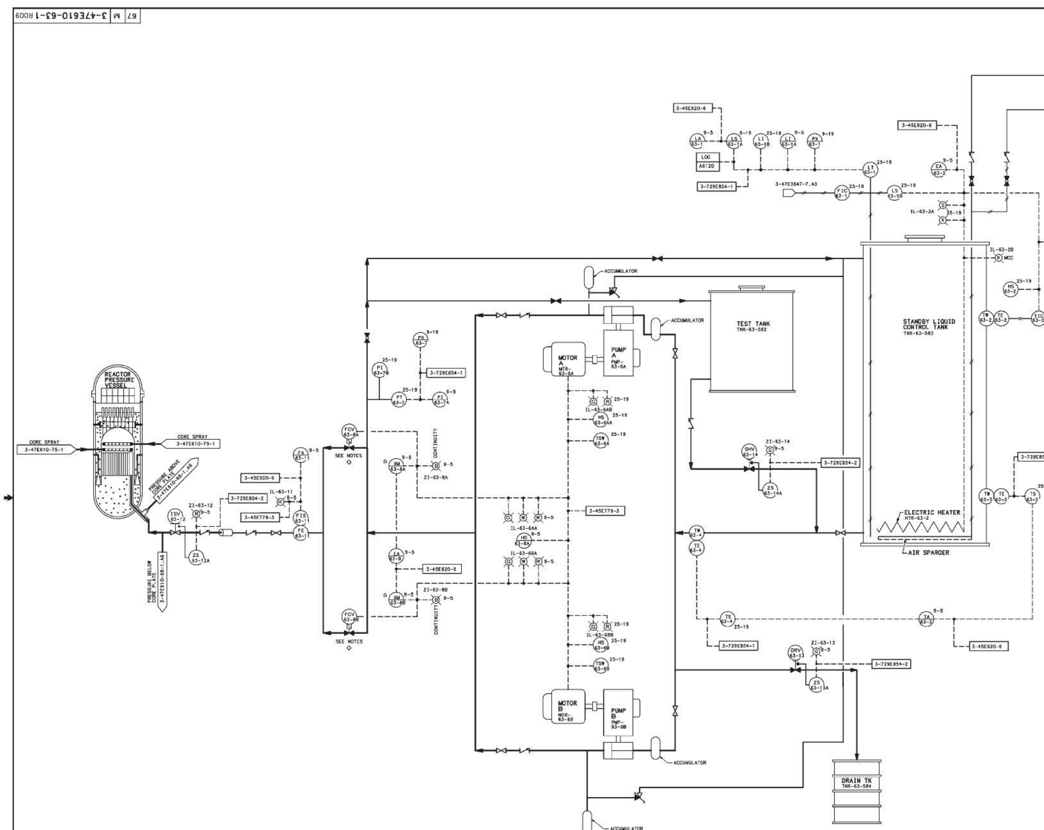
AMENDMENT 27

POWERHOUSE
UNIT 3

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

STANDBY LIQUID CONTROL SYSTEM
FLOW DIAGRAM

FIGURE 3.8-5



NOTES:

1. THE STANDBY LIQUID CONTROL SYSTEM IS USED AS A REDUNDANT, INDEPENDENT BACKUP SYSTEM CAPABLE OF SHUTTING DOWN THE REACTOR AND MAINTAINING IT IN A SHUTDOWN CONDITION IF THE CONTROL ROD DRIVE SYSTEM IS TOTALLY INOPERATIVE. THE SLC SYSTEM ALSO SERVES AS BACKUP REACTIVITY CONTROL DURING REFUELING.
2. A NEUTRON POISON, B-10 ENRICHED SODIUM PENTABORATE (ESPB), IS STORED IN THE STANDBY LIQUID CONTROL SYSTEM (SLCS) STORAGE TANK. TIC-63-2 CONTROLS THE TANK HEATER WHICH ACTS AS A BACKUP MEANS FOR ENSURING THAT THE TANK SOLUTION IS MAINTAINED ABOVE THE ESPB'S SATURATION TEMPERATURE. LEVEL SWITCH LS-63-1B, IS INTERLOCKED WITH THE HEATER CONTACTOR TO DEENERGIZE THE HEATERS ON SLCS TANK LOW LEVEL TEMPERATURE SWITCHES TS-63-3 AND TS-63-4 ACTIVATE TA-63-3 IN THE MAIN CONTROL ROOM (MCR) IF THE SOLUTION APPROACHES THE SATURATION TEMPERATURE IN THE ESPB OR TO INDICATE HEATER FAILURE MODES. THE LEVEL OF THE ESPB SOLUTION IN THE SLCS STORAGE TANK IS MONITORED BY LT-63-1 AND WITH INDICATION IN THE (MCR), LI-63-1A AND LOCALLY ON LI-63-1B. THE LEVEL INDICATION IS UTILIZED TO VERIFY OPERATION OF THE SLCS DURING ACCIDENT AND NORMAL OPERATION. LEVEL ALARM LA-63-1 ACTUATES ON A LOW LEVEL AND HIGH LEVEL FOR INDICATION OF GROSS FAILURE MODES.
3. EXPLOSIVE-TYPE VALVES, FCV-63-8A AND FCV-63-8B, ARE ACTUATED BY THE PUMP CONTROL SWITCH, HS-63-6A, IN THE MAIN CONTROL ROOM. THE EXPLOSIVE VALVES ARE NOT ACTUATED WHEN EITHER LOCAL SWITCH HS-63-6B OR HS-63-6AA IS USED TO START ITS RESPECTIVE PUMP.
4. ACTUATING THE EXPLOSIVE VALVES CLOSES THE REACTOR WATER CLEANUP SYSTEM ISOLATION VALVES VIA AN INTERLOCK (SEE 47W611-69-1, LOGIC MATRIX 1A).
5. THE EXPLOSIVE VALVE AND PUMP CONTROL CIRCUITS ARE CONTINUOUSLY MONITORED FOR CONTINUITY, AND ANY MALFUNCTION ACTUATES ALARM EA-63-8.
6. DELETED.
7. UNIDS ON DRAWINGS ARE FOR REFERENCE ONLY AND ARE ABBREVIATED AS SHOWN IN THE EXAMPLE TO MEET SPACE CONSTRAINTS. REFER TO MEL FOR COMPLETE UNIDS. ALL UNIDS ARE IN UNIT 3 UNLESS OTHERWISE NOTED. CABLES ARE IN THE SAME SYSTEM AS THEIR END DEVICE. LEADING ZEROES SHOWN IN MEL AS PART OF THE UNID ARE NOT DEPICTED. FOR ADDITIONAL GUIDANCE, REFER TO NEDP-4.

EXAMPLE:

MEL UNID	DRAWING UNID
BFN-0-SHV-018-0502	SHV-18-502
BFN-1-HS-031-0160A	HS-31-160A
BFN-2-PT-003-0204C	PT-3-204C

REFERENCE DRAWINGS:

MEL....................INSTRUMENT TABULATION - SLC SYSTEM
0-47E610-75-1........CONTROL DIAGRAM - CORE SPRAY SYSTEM
3-47E854-1...........FLOW DIAGRAM - SLC SYSTEM
0-47E800-2...........MECHANICAL SYMBOLS & FLOW DIAGRAM DRAWING INDEX
3-47E3847-7..........MECHANICAL FLOW DIAGRAM - CONTROL AIR SYSTEM
3-45E620-6...........WIRING DIAGRAM - ANNUNCIATOR KEY DIAGRAM
3-45E779-3,-5........WIRING DIAGRAM - 480V SHUTDOWN AUXILIARY POWER, SCHEMATIC DIAGRAM
0-47E845-2...........MECHANNICAL FLOW DIAGRAM - COMPRESSED AIR STATION SERVICE
3-47E610-68-1........MECHANICAL CONTROL DIAGRAM - REACTOR CIRCULATION SYSTEM

GE DRAWINGS:

719E599..............PIPING AND INSTRUMENT DIAGRAM - SLC SYSTEM
919D926..............FUNCTIONAL CONTROL DIAGRAM - SLC SYSTEM
161F292..............PROCESS DIAGRAM - SLC SYSTEM

GE DESIGN SPECIFICATION:

22A1315..............DESIGN SPECIFICATION - SLC SYSTEM
22A1315AB............DESIGN SPECIFICATION - SLC SYSTEM DATA SHEETS
3-729E854-1..........STANDBY LIQUID CONTROL - ELEMENTARY DIAGRAM
3-729E854-2..........STANDBY LIQUID CONTROL - ELEMENTARY DIAGRAM

SYMBOL:

◇ FAIL AS IS

COMPANION DRAWING:
1-47E610-63-1
2-47E610-63-1

AMENDMENT 27

POWERHOUSE
UNIT 3

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

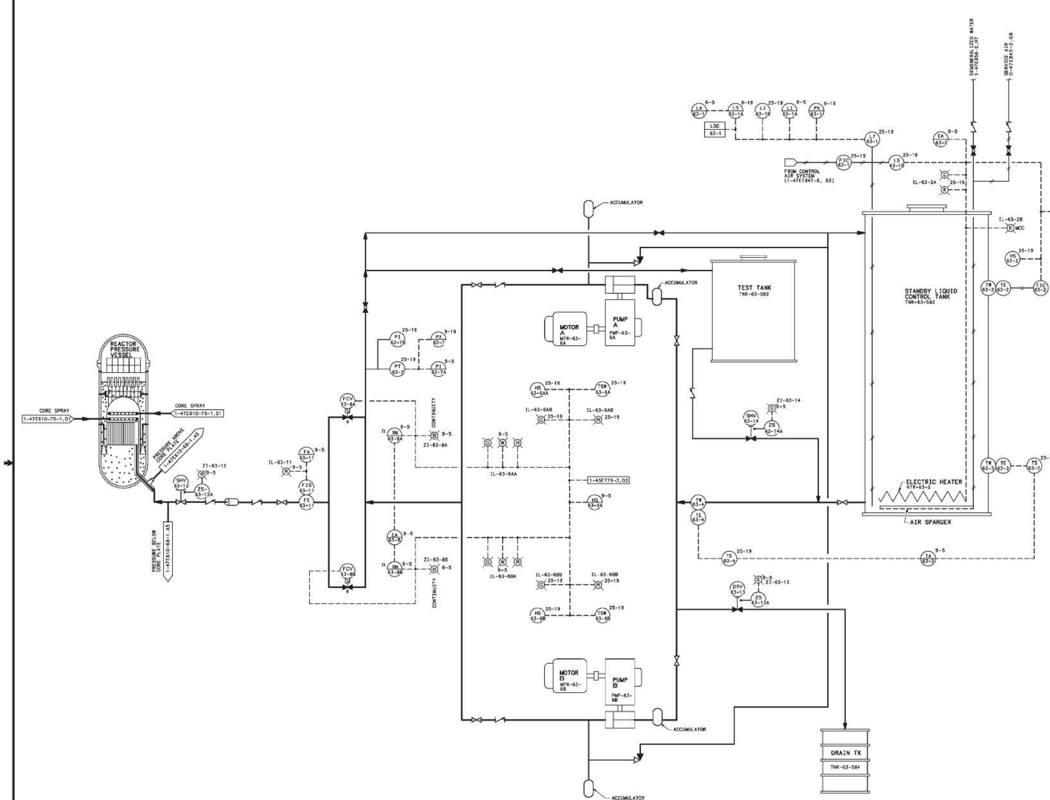
STANDBY LIQUID CONTROL SYSTEM
MECHANICAL CONTROL DIAGRAM

FIGURE 3.8-6

BFN-25

Figure 3.8-7

(Deleted by Amendment 25)



NOTES:

1. THE STANDBY LIQUID CONTROL SYSTEM IS USED AS A REDUNDANT, INDEPENDENT BACKUP SYSTEM CAPABLE OF SHUTTING DOWN THE REACTOR AND MAINTAINING IT IN A SHUTDOWN CONDITION IF THE CONTROL ROD DRIVE SYSTEM IS TOTALLY INOPERATIVE. THE SLC SYSTEM ALSO SERVES AS BACKUP REACTIVITY CONTROL DURING REFUELING.
2. A NEUTRON POISON, B-10 ENRICHED SODIUM PENTABORATE (ESPB) IS STORED IN THE STANDBY LIQUID CONTROL SYSTEM (SLCS) STORAGE TANK. TIC-63-2 CONTROLS THE TANK HEATER WHICH ACTS AS A BACKUP MEANS FOR ENSURING THAT THE TANK SOLUTION IS MAINTAINED ABOVE THE ESPB'S SATURATION TEMPERATURE. LEVEL SWITCH LS-63-1B, IS INTERLOCKED WITH THE HEATER CONTACTOR TO DEENERGIZE THE HEATERS ON SLCS TANK LOW LEVEL. TEMPERATURE SWITCHES TS-63-3 AND TS-63-4 ACTIVATE TA-63-3 IN THE MAIN CONTROL ROOM (MCR) IF THE SOLUTION APPROACHES THE SATURATION TEMPERATURE OF THE ESPB OR TO INDICATE HEATER FAILURE MODES. THE LEVEL OF THE ESPB SOLUTION IN THE SLCS STORAGE TANK IS MONITORED BY LT-63-1 AND WITH INDICATION IN THE (MCR), LI-63-1A AND LOCALLY ON LI-63-1B. THE LEVEL INDICATION IS UTILIZED TO VERIFY OPERATION OF THE SLCS DURING ACCIDENT AND NORMAL OPERATION. LEVEL ALARM LA-63-1 ACTUATES ON A LOW LEVEL AND HIGH LEVEL FOR INDICATION OF GROSS FAILURE MODES.
3. EXPLOSIVE-TYPE VALVES, FCV-63-8A AND FCV-63-8B, ARE ACTUATED BY THE PUMP CONTROL SWITCH, HS-63-6A, IN THE MAIN CONTROL ROOM. THE EXPLOSIVE VALVES ARE NOT ACTUATED WHEN EITHER LOCAL SWITCH HS-63-6B OR HS-63-6AA IS USED TO START ITS RESPECTIVE PUMP.
4. ACTUATING THE EXPLOSIVE VALVES CLOSES THE REACTOR WATER CLEANUP SYSTEM ISOLATION VALVES VIA AN INTERLOCK (SEE 47W611-69-1, LOGIC MATRIX 1A).
5. THE EXPLOSIVE VALVE AND PUMP CONTROL CIRCUITS ARE CONTINUOUSLY MONITORED FOR CONTINUITY, AND ANY MALFUNCTION ACTUATES ALARM EA-63-8.
6. UNIDS ON DRAWING ARE FOR REFERENCE ONLY AND ARE ABBREVIATED AS SHOWN IN THE EXAMPLE TO MEET SPACE CONSTRAINTS. REFER TO MEL FOR COMPLETE UNIDS. LEADING ZEROES SHOWN IN MEL AS PART OF THE UNID ARE NOT DEPICTED. FOR ADDITIONAL GUIDANCE, REFER TO NEDP-4.

EXAMPLE: MEL UNID	DRAWING UNID
BFN-0-SHV-018-0502	SHV-18-502
BFN-1-HS-031-0160A	HS-31-160A
BFN-2-PT-003-0204C	PT-3-204C

7. THE USE OF B-10 ESPB PRECLUDES THE NEED FOR USING HEAT TRACE UNDER NORMAL AND ABNORMAL AMBIENT TEMPERATURE CONDITIONS EXISTING IN THE SLCS EQUIPMENT AREA. SINCE THE WORST CASE AMBIENT AREA TEMPERATURE PREDICTED IS ABOVE THE SATURATION TEMPERATURE OF THE ESPB, THE NEED FOR HEAT TRACE IS ELIMINATED.

REFERENCE DRAWINGS:

MEL....................INSTRUMENT TABULATION-SLC SYSTEM
1-47E610-75-1........CONTROL DIAGRAM-CORE SPRAY SYSTEM
1-47E854-1...........FLOW DIAGRAM-SLC SYSTEM
0-47E800-2...........MECHANICAL SYMBOLS & FLOW DIAGRAM DRAWING INDEX
0-47E845-2...........MECHANICAL FLOW DIAGRAM-STATION SERVICE AIR
1-45E620-6-2.........WIRING DIAGRAM-ANNUNCIATOR KEY DIAGRAM
1-45E779-3, -5.......WIRING DIAGRAM-480V S/D AUX PWR SCHEMATIC
1-47E610-68-1........MECHANICAL CONTROL DIAGRAM-REACTOR RECIRC
1-47E856-2...........MECHANICAL FLOW DIAGRAM-DEMINERALIZED WATER
1-47E1847-6..........MECHANICAL FLOW DIAGRAM-CONTROL AIR
1-729E854-1, -2......ELEMENTARY DIAGRAM-STANDBY LIQUID CONTROL

GE DESIGN SPECIFICATION:
22A1315.........DESIGN SPECIFICATION-SLC SYSTEM
22A1315AB.......DESIGN SPECIFICATION-SLC SYSTEM DATA SHEETS

SYMBOL:
◊ FAIL AS IS

COMPANION DRAWINGS:
2-47E610-63-1
3-47E610-63-1

AMENDMENT 27

POWERHOUSE
UNIT 1

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

STANDBY LIQUID CONTROL SYS
MECHANICAL CONTROL DIAGRAM

FIGURE 3.8-8