REACTOR COOLANT SYSTEM

TABLE OF CONTENTS

REACTOR COOLANT SYSTEM

TABLE OF CONTENTS (Cont'd)

REACTOR COOLANT SYSTEM

TABLE OF CONTENTS (Cont'd)

REACTOR COOLANT SYSTEM
LIST OF TABLES

REACTOR COOLANT SYSTEM
LIST OF ILLUSTRATIONS

REACTOR COOLANT SYSTEM
LIST OF ILLUSTRATIONS (cont'd)

4.0 REACTOR COOLANT SYSTEM

4.1 SUMMARY DESCRIPTION

The subsections in the "Reactor Coolant System" section describe those systems and components that form the major portions of the nuclear system process barrier. These systems and components contain or transport the fluids coming from or going to the reactor core.

The "Reactor Vessel and Appurtenances Mechanical Design" subsection describes the reactor vessel and the various fittings with which other systems are connected to the vessel. The major safety considerations for the reactor vessel are concerned with the ability of the vessel to function as a radioactive material barrier. Various combinations of structural loading are considered in the vessel design. The vessel meets the requirements of various applicable codes and criteria. The possibility of brittle fracture is considered, and suitable limits are established that avoid conditions where brittle fracture is possible. Periodic, cumulative-fatigue usage evaluations are performed for each reactor vessel to verify that the vessel does not approach usage limits.

The Reactor Recirculation System pumps coolant through the core. Adjustment of the core coolant flow rate changes reactor power output, thus providing a means of following plant load demand or manually changing reactor output without adjusting control rods. The recirculation system is designed with sufficient fluid and pump inertia that fuel thermal limits will not be exceeded as a result of recirculation system malfunctions. The arrangement of the recirculation system is designed so that a piping failure cannot compromise the integrity of the floodable inner volume of the reactor vessel.

The Nuclear System Pressure Relief System is designed to protect the nuclear system process barrier from damage due to overpressure. To accomplish overpressure protection a number of main steam relief valves are provided that can discharge steam from the nuclear system to the primary containment. The Nuclear System Pressure Relief System also acts to automatically depressurize the nuclear system in the event of a loss-of-coolant accidents in which the High Pressure Coolant Injection System (HPCIS) fails to deliver sufficient flow. The depressurization of the nuclear system allows low pressure Emergency Core Cooling Systems to supply enough cooling water to adequately cool the fuel. Six of the main steam relief valves used to provide overpressure protection are arranged to effect automatic depressurization.

The main steam line flow restrictors are venturi-type flow devices. One restrictor is installed in each main steam line close to the reactor vessel but downstream of the main steam relief valves. The restrictors are designed to limit the loss of coolant resulting from a main steam line break outside the primary containment. The

coolant loss is limited so that reactor vessel water level remains above the top of the core during the time required for the main steam isolation valves to close. This action protects the fuel barrier.

Two main steam isolation valves are installed on each main steam line. One valve in each line is located inside the primary containment, the other outside. These valves act automatically to close off the nuclear system process barrier in the event a pipe break occurs downstream of the valves. This action limits the loss of coolant and the release of radioactive materials from the nuclear system. In the event that a main steam line break occurs inside the primary containment, closure of the isolation valve outside the containment acts to seal the primary containment itself.

The Reactor Core Isolation Cooling System (RCICS) includes a turbine-pump driven by reactor vessel steam. Under certain conditions the system automatically starts in time to prevent the core from becoming uncovered without the use of the Core Standby Cooling Systems. The system provides the ability to cool the core during a reactor shutdown in which feedwater flow is not available.

The Residual Heat Removal System (RHRS) includes a number of pumps and heat exchangers that can be used to cool the nuclear system under a variety of situations. During normal shutdown and reactor servicing, the RHRS removes residual and decay heat. One operational mode of the RHRS is low pressure coolant injection (LPCI). LPCI operation is an engineered safeguard for use during a loss-of-coolant accident; this operation is described in Section 6.0, "Emergency Core Cooling Systems." Another mode of RHRS operation allows the removal of heat from the primary containment following a loss-of-coolant accident.

The Reactor Water Cleanup System functions to maintain the required purity of reactor coolant by circulating coolant through a system of filter/demineralizers.

The "Nuclear System Leakage Rate Limits" subsection establishes the limits on nuclear system leakage inside the primary containment so that appropriate action can be taken before the nuclear system process barrier is threatened by a crack large enough to propagate rapidly.

The main steam lines, feedwater piping, and their associated drains are attached to the reactor vessel and provide core coolant flow paths external to it. These lines penetrate the primary containment and specified portions of these lines must provide adequate nuclear system process barrier for normal and accident conditions.

Four steam lines are utilized between the reactor and the turbine which permit turbine stop valve and main steam isolation valve tests during plant operation with a minimum amount of load reduction. In addition, differential pressures on reactor internals under assumed accident conditions of a broken steam line are limited. Feedwater lines provide water to the reactor vessel entering near the top of the

vessel downcomer annulus. Drains are provided at the low point of each main steam line, at the reactor vessel bottom head, and on each side of the recirculation pumps.

The program for preoperational examination and periodic inservice examination of Reactor Coolant System is also defined.

4.2 REACTOR VESSEL AND APPURTENANCES MECHANICAL DESIGN

4.2.1 Power Generation Objective

The reactor vessel power generation design objective is to provide a volume in which the core can be submerged in coolant, thereby allowing power operation of the fuel. The reactor vessel and appurtenances design provides the means for the attachment of pipelines to the reactor vessel and the means for the proper installation of vessel internal components.

4.2.2 Power Generation Design Basis

1. The location and design of the external and internal supports provided as an integral part of the reactor vessel shall be such that stresses in the reactor vessel and supports due to reactions at these supports are within ASME Boiler and Pressure Vessel Code limits.

2. The reactor vessel design lifetime shall be 40 years. Time Limited Aging Analyses (TLAAs) have been identified and evaluated for the reactor vessel 60 year operating life. Summaries of these evaluations for the reactor vessel life are provided in Appendix O, Section O.3.1 and O.3.2.

3. The design of the reactor vessel and appurtenances shall allow for the accomplishment of a suitable program of periodic inspection and surveillance.

4.2.3 Safety Design Basis

1. The reactor vessel and appurtenances shall be designed to withstand adverse combinations of loadings and forces resulting from operation under abnormal and accident conditions.

2. The reactor vessel shall be designed and fabricated to a high standard of quality to provide assurance of an extremely low probability of failure.

3. To minimize the possibility of brittle fracture failure of the nuclear system process barrier, the following shall be required: (1) the initial ductile-brittle transition temperature of materials used in the reactor vessel shall be known by reference or established empirically; (2) expected shifts in transition temperature during design service life due to environmental conditions, such as neutron flux, shall be determined and employed in the reactor vessel design; and (3) operation margins to be observed with regard to the transition temperature shall be designated for each mode of operation.

4. The design shall provide for material surveillance specimens which may be used to verify predicted radiation exposure and to measure the effect of radiation on the vessel material.

4.2.4 Description

4.2.4.1 Reactor Vessel

The reactor vessel is a vertical, cylindrical pressure vessel with hemispherical heads of welded construction. The reactor vessel is designed and fabricated for a useful life of 40 years based upon the specified design and operating conditions. TLAAs have been identified and evaluated for the reactor vessel 60 year operating life. Summaries of these evaluations for the reactor vessel life are provided in Appendix O, Sections O.3.1 and O.3.2. The vessel for each unit is designed, fabricated, inspected, and tested in accordance with the ASME Boiler and Pressure Vessel Code, Section III, 1965 edition, Summer 1965 addenda (Unit 3 vessel - Summer 1966 addenda), code cases 1332-1, 1332-2, 1332-3, 1334, 1335-2 (paragraph 4), 1336, 1339, applicable requirements for Class A vessels as defined therein, and additional GE requirements. The reactor vessel and its supports are designed as Class I equipment in accordance with the loading criteria of Appendix C. The materials used in the design and fabrication of the reactor pressure vessel are shown in Table 4.2-1.

The Browns Ferry Unit 1 vessel was fabricated by B&W. The Browns Ferry Units 2 and 3 vessels were fabricated by Ishikawajima-Harima Heavy Industries Co. (IHI) in Japan, under a contract between B&W and IHI. IHI had previously manufactured the Fukushima I and II vessels. These vessels are built to the ASME Boiler and Pressure Vessel Code and GE specifications. Reactor vessel data is presented in Table 4.2-2.

The cylindrical shell and bottom hemispherical head of the reactor vessel are fabricated of low alloy steel plate which is clad on the interior with weld overlay. The cylindrical shell is clad with stainless steel, and the bottom hemispherical head is clad with Inconel. The plates and forgings are ultrasonically tested and magnetic-particle-tested over 100 percent of their surfaces after forming and heat treatment. Full-penetration welds are used at all joints, including nozzles, throughout the vessel, except for nozzles of less than 3-inch nominal size and the CRD housing-to-stub tube welds. Nozzles of less than 3-inch nominal size are partial-penetration-welded as permitted by ASME Boiler and Pressure Vessel Code, Section III. The electroslag weld process was used on the Browns Ferry pressure vessels. Electroslag welding process variables, quality control procedures and technical details were presented in Appendix F, Dresden 2/3 FSAR, Docket Nos. 50-237 and 50-249.

Although little corrosion of plain carbon or low-alloy steels occurs at temperatures of 500°F to 600°F, higher corrosion rates occur at temperatures around 140°F. The 0.125-inch minimum-thickness cladding provides the necessary corrosion resistance during reactor shutdown and also helps maintain water clarity during refueling operations. Since the vessel head is exposed to a saturated steam environment throughout its operating lifetime, stainless steel cladding is not required over its interior surfaces. Exterior, exposed ferritic surfaces of pressure-containing parts have a minimum corrosion allowance of 1/16 inch. The interior surfaces of the top head and all carbon and low-alloy steel nozzles exposed to the reactor coolant have a corrosion allowance of 1/16 inch. The vessel shape is designed to limit coolant retention pockets and crevices.

The nil-ductility transition (NDT) temperature is defined as the temperature below which ferritic steel breaks in a brittle, rather than ductile, manner. The NDT temperature increases as a function of neutron exposure at integrated neutron exposures greater than about 1 x 10^{17} nvt with neutrons of energies in excess of 1 MeV. Since the material NDT temperature dictates the minimum operating temperature at which the reactor vessel can be pressurized, it is desirable to keep the NDT temperature as low as possible. One way that this is accomplished is by selecting fine-grained steels and by using advanced fabrication techniques to minimize radiation effects. The as-fabricated initial NDT temperature for all carbon and low-alloy steel used in the main closure flanges, closure bolting material, and the shell and head materials connecting to these flanges, including the connecting circumferential weld material, is limited to a maximum of 10°F as determined by ASTM E208. For each main closure flange forging, a minimum of 1 tensile, 3 Charpy V-notch, and 2 drop weight test specimens have been tested from each of two locations about 180° apart on the flange. For all other carbon and low-alloy steel pressure-containing materials, including weld materials and the vessel support skirt material, the initial NDT temperature is no higher than 56°F for Unit 1, and 40°F for Units 2 and 3. A grain size of 5 or finer, as determined by the method in ASTM E112, is maintained.

Another way of minimizing any changes (elevating) to the NDT temperature is by reducing the integrated neutron exposure at the inner surface of the reactor vessel. The coolant annulus between the vessel and core shroud and the core location in the vessel limit the integrated neutron exposure of reactor vessel material to less than 1 x 10^{19} nvt from neutrons with energy levels greater than 1 MeV, within the 40-year design lifetime of the vessel. TLAAs have been identified and evaluated for the reactor vessel 60 year operating life. Summaries of these evaluations for the reactor vessel life are provided in Appendix O, Sections O.3.1 and O.3.2. This is not the expected exposure, nor is it the absolute limit of safe exposure; it is an exposure value that can be demonstrated to be safe and practical to maintain. The maximum calculated exposure for neutrons of 1 MeV or greater is 1.58 X 10^{18} nvt for Unit 1, per GEH Report No. 0000-0166-0632-R0. The maximum calculated exposure for

neutrons of 1 MeV or greater is less than 1 X 10^{19} nvt for Unit 2 and 2.23 X 10^{18} nvt for Unit 3.

The vessel top head is secured to the reactor vessel by studs and nuts which are designed to be tightened with a stud tensioner. The vessel flanges are sealed by two concentric metallic seal-rings designed for no detectable leakage through the inner or outer seal at any operating condition, including: (a) cold hydrostatic pressure test at the hydro-pressure specified in the ASME code, and (b) heating to operating pressure and temperature at a maximum rate of 100°F/hr. To detect lack of seal integrity, a 1-inch vent tap is provided in the area between the two seal-rings, and a monitor line is attached to the tap to provide an indication of leakage from the inner seal-ring seal (see Subsection 7.8). A 1-inch tap is also provided in the area outside the outer seal-ring for use in monitoring leakage. This tap is used only if the inner seal fails and is piped to an accessible place in the drywell and capped.

The head and vessel flanges are low-alloy steel forgings. The sealing surfaces of the reactor vessel head and shell flanges are weld-overlay clad with Inconel 82 (ERNiCr material. The clad thickness is 0.25 inches on both the head flange and shell flange sealing surfaces.

All sensitized austenitic stainless steel has been replaced on the Browns Ferry pressure vessels, except the jet pump riser brace pads on all units. These components have been clad with nonfurnace-sensitized stainless steel weld overlay. Austenitic stainless steel used in other component parts of the reactor coolant pressure boundary, including relief and safety valves, is fully annealed to preclude sensitization.

Welding processes were limited to 110,000 joules per inch and the interpass temperature limited to 350°F to avoid local sensitization of stainless steel. Stainless steel with deliberate additions of nitrogen for enhancing the material strength has not been used.

The vessel nozzles (Figure 4.2-2) are low-alloy steel forgings made in accordance with ASTM A508 CL2 as modified by ASME code case 1332-2, paragraph 5. Nozzles of 3-inch nominal size or larger are full-penetration welded to the vessel. Nozzles of less than 3-inch nominal size may be partial-penetration-welded as permitted by ASME Boiler and Pressure Vessel Code, Section III. Nozzles which are partial-penetration welded are nickel-chromium-iron forgings made in accordance with ASME SB166 as modified by code case 1336.

The vessel top head nozzles are provided with flanges with small groove facing. The drain nozzle is of the full-penetration weld design and extends 16 inches below the bottom outside surface of the vessel. The recirculation inlet nozzles are located as shown in Figures 4.2-1, 4.2-3, and 4.2-4; feedwater inlet nozzles, core spray inlet

nozzles, and the control rod drive hydraulic system return nozzle have thermal sleeves similar to those shown in the detail of Figure 4.2-2.

As a result of cracks discovered in the feedwater nozzle blend and nozzle bore regions of several operating reactors, General Electric and the NRC performed an extensive study of the problem. The program, the solutions, and NRC acceptance of the modifications are fully described in NEDE 21821-A, "Boiling Water Reactor Feedwater Nozzle/Sparger - Final Report," February 1980 (proprietary version). The modifications to the BFNP feedwater nozzles included: (1) removal of the stainless-steel-clad and heat affected zone of the feedwater nozzle bore and nozzle bend radius, and (2) machining the safe end and nozzle bore and inner bend radius to accept the improved double piston ring seal, interference fit spargers with forged tee design, and orificed elbow discharges. Implementing these modifications increased the assurance of maintaining vessel integrity by minimizing the potential for crack initiation due to thermal cycling.

The nozzle for the core differential pressure and standby liquid control pipe is designed with a transition so that the stainless steel outer pipe of the differential pressure and standby liquid control line (see Subsection 3.3, "Reactor Vessel Internals Mechanical Design") can be socket-welded to the inner end of the nozzle and so that the inner pipe passes through the nozzle. This design provides an annular region between the nozzle and the inner standby liquid control line to minimize thermal shock effects on the reactor vessel in the event that use of the Standby Liquid Control System is required.

The jet pump instrumentation penetration seal is welded directly to the outer end of the jet pump instrumentation nozzle. The stainless steel recirculation loop piping (see Subsection 4.3, "Reactor Recirculation System") is welded to the outer end of the recirculation outlet nozzle. The main steam line piping is welded to the outer end of the steam outlet nozzle. The piping attached to the vessel nozzle is designed, installed, and tested in accordance with the requirements of USAS B31.1.0, 1967 edition and the applicable GE design and procurement specifications, which were implemented in lieu of the outdated B31 Nuclear Code Cases-N2, N7, N9, and N10.

Thermocouple pads are located on the exterior of the vessel (see Table 4.2-3). At each thermocouple location, two 3/4-inch-diameter pads are provided: an end pad to hold the end of a 3/16-inch-diameter thermocouple and a clamp pad equipped with a set screw to secure the thermocouple.

The reactor vessel is laterally and vertically supported and braced to make it as rigid as possible without impairing the movements required for thermal expansion. Where thermal requirements prohibit the use of rigid supports, spring anchors or hydraulic snubbers are employed to resist earthquake forces, while allowing sufficient flexibility for thermal expansion.

4.2.4.2 Shroud Support

The reactor vessel shroud support is a radial, cylindrical shell that surrounds the reactor core assembly and is designed so that stresses due to reactions at the shroud support are within ASME code, Section III, requirements for normal, upset, emergency, and faulted loading conditions. The design of the shroud support also takes into account the restraining effect of the components attached to the support, their weight, and earthquake loadings. The vessel shroud support and other internal attachments (jet pump riser support pads, diffuser brackets, guide rod brackets, steam dryer support brackets, dryer holddown brackets, feedwater sparger brackets, and core spray brackets) are as shown in Figures 4.2-1, 4.2-3, and 4.2-4.

4.2.4.3 Reactor Vessel Support Assembly

The reactor vessel support assembly consists of a ring girder, sole plates, and the various bolts, shims, and set screws necessary to position and secure the assembly between the reactor vessel support skirt and the support pedestal. The concrete and steel support pedestal is constructed integrally with the building foundation. Steel anchor bolts are set in the concrete with the threads extending above the surface. The sole plates are set flat and level on the concrete, and the lower flange of the ring girder is set on top of the sole plates and shimmed as necessary to level the ring girder. The anchor bolts extend through both the sole plates and the ring girder bottom flange. High strength bolts are used to bolt the flange of the reactor vessel support skirt to the top flange of the ring girder. The ring girder and sole plates are fabricated of ASTM A36 structural steel according to AISC specifications.

4.2.4.4 Vessel Stabilizers

The vessel stabilizers are connected between the reactor vessel and the top of the shield wall surrounding the vessel to provide lateral stability for the upper part of the vessel. Eight stabilizer brackets are attached by full-penetration welds to the reactor vessel at evenly spaced locations around the vessel below the flange. Each vessel stabilizer consists of a stabilizer rod, threaded at the ends, springs, washers, nut, a plate, and a bumper bracket with tapered shims. The stabilizers are attached to each bracket and apply tension in opposite directions. The stabilizers are evenly preloaded with tensioners to the values of the residual loads. The stabilizers are designed to permit radial and axial vessel expansion, to limit horizontal vibration, and to resist seismic and jet reaction forces.

4.2.4.5 Refueling Bellows

The refueling bellows form a seal between the reactor vessel and the surrounding primary containment drywell to permit flooding of the space (reactor well) above the vessel during refueling operations. The refueling bellows assembly (see Figures

4.2-1, 4.2-3, and 4.2-4) consists of a Type 304 stainless steel bellows, a backing plate, a spring seal, and a removable guard ring. The backing plate surrounds the outer circumference of the bellows to protect it and is equipped with a tap for testing and for monitoring leakage. The self-energizing spring seal is located in the area between the bellows and the backing plate and is designed to limit water loss in the event of a bellows rupture by yielding to make a tight fit to the backing plate when subjected to full hydrostatic pressure. The guard ring attaches to the assembly and protects the inner circumference of the bellows. The guard ring can be removed from above to inspect the bellows. The assembly is welded to the reactor bellows support skirt and the reactor well seal bulkhead plate. The reactor bellows support skirt is welded to the reactor vessel shell flange, and the reactor well seal bulkhead plate bridges the distance to the primary containment drywell wall. Six watertight, hinged covers are bolted in place for normal refueling operation. For normal operation, these covers are opened and removable air supply ducts and air return ducts permit circulation of ventilation air in the region above the reactor well seal.

4.2.4.6 Control Rod Drive Housings

The control rod drive housings are inserted through the control rod drive penetrations in the reactor vessel bottom head and are welded to the stub tubes extending into the reactor vessel[1] (Figure 4.2-2).

Each housing transmits a number of loads to the bottom head of the reactor. These loads include the weight of a control rod and control rod drive, which are bolted to the housing from below (see Subsection 3.4, "Reactivity Control Mechanical Design"), the weight of a control rod guide tube, one four-lobed fuel support piece, and the four fuel assemblies which rest on the top of the fuel support piece (see Subsection 3.3, "Reactor Vessel Internal Mechanical Design"). The housings are fabricated of Type 304 austenitic stainless steel.

4.2.4.7 Control Rod Drive Housing Supports

The control rod drive housing support is designed to prevent a nuclear transient in the unlikely event that there is a control rod drive housing failure. This device consists of a grid structure located below the reactor vessel from which housing supports are suspended. The supports allow only slight movement of the control rod drive or housing in the event of failure. The control rod drive housing support is described in detail in Subsection 3.5, "Control Rod Drive Housing Supports."

---

1 Kobsa, I. R., and Wetzerl, V. R., "Design and Analysis of Control Rod Drive Reactor Vessel Penetrations," General Electric Co., Atomic Power Equipment Department, November 1968 (APED-5703).

4.2.4.8 In-Core Neutron Flux Monitor Housing

The in-core neutron flux monitor housings are inserted up through the in-core penetrations in the bottom head of the reactor vessel and are welded to the inner surface of the bottom head (Figure 4.2-2). An in-core flux monitor guide tube is welded to the top of each housing (see Subsection 3.3, "Reactor Vessel Internals Mechanical Design"), and either a source range monitor/intermediate range monitor (SRM/IRM) drive unit or a local power range monitor (LPRM) is bolted to the seal-ring flange at the bottom of the housing (see Subsection 7.5, "Neutron Monitoring System").

4.2.4.9 Reactor Vessel Insulation

The reactor vessel insulation is an all-metal, reflective insulation having an average maximum heat transfer rate of approximately 80 Btu/hr-ft^2 at the operating conditions of 550°F for the vessel and 135°F for the outside air. The maximum insulation thickness ranges from 4 inches for the upper head to 3-1/2 inches for the cylindrical shell and nozzles and 3 inches for the bottom head. The insulation is designed to permit complete submersion in water without loss of insulating material, contamination from the water, or adverse effect on the insulation efficiency of the insulation assembly after draining and drying. The lower head and cylindrical shell insulation is permanently installed for the 60 year operating life of the vessel. The insulation panels for the cylindrical shell of the vessel are held in place by vessel insulation supports located at two elevations on the vessel. The support brackets for each support are full-penetration-welded to the vessel at 12 evenly spaced locations around the circumference. Provisions are made for removing insulation during inservice inspection.

4.2.4.10 Other Reactor Coolant Pressure Boundary Ferritic Components

The fracture or notch-toughness properties and the operating temperature of ferritic materials are controlled to ensure adequate toughness when the system is pressurized to more than 20 percent of the design pressure. Such assurance is provided by maintaining the lowest service metal temperature, when the system pressure exceeds 20 percent of design pressure, at least 60°F above the nil-ductility transition temperature (NDTT). The lowest service-metal temperature is the lowest temperature which the metal will experience in service while the plant is in operation. It is established by appropriate calculations considering atmosphere ambient temperatures, the insulation or enclosure provided, and the minimum temperature maintained. Further interpretations and requirements are as follows:

A. Charpy V-notch (American Society for Testing and Material Standard A370 Type A) or drop weight (per ASTM E208) tests have been performed to demonstrate that all materials and weld metal meet brittle fracture requirements at test temperature. Test specimens, for the surveillance capsule pulled in 1994, were prepared and tested with minimum impact energy requirements in accordance with Table N-421 and the general provisions of N-313, N-331, N-332, and N-511 of Section III of the ASME Boiler and Pressure Vessel Code. For the surveillance capsule pulled in 2011, per BWRVIP-271/NP, the Charpy impact tests were conducted in accordance with ASTM Standards E185-82 and E23-02. Prior to the Summer 1972 Addenda of the 1971 ASME Section III Boiler and Pressure Vessel Code, impact testing was not required on materials with a nominal section thickness of 1/2 inch or less. However, this 1/2 inch thickness exclusion was increased to 5/8 inch by the ASME Boiler and Pressure Vessel Code, Section III, 1971 Edition, Summer 1972 Addenda. Therefore, after issuance of the Summer 1972 Addenda, impact testing is not required on materials with a nominal section thickness of 5/8 inch or less. The welding procedures used were qualified by impact testing of weld metal and heat affected zone to the same requirements as the base metal in accordance with N-541.

B. Impact tests were not required for the following:

1. Bolting, including nuts, 1-inch nominal diameter or less,

2. Bars with a nominal cross-sectional area not exceeding 1 square inch,

3. Materials with a nominal (section) wall thickness of less than 1/2 inch or 5/8 inch (refer to paragraph 4.2.4.10.A),

4. Components including pumps, valves, piping, and fittings with a nominal inlet pipe size of 6-inch-diameter and less, regardless of thickness, and

5. Consumable insert material, austenitic stainless steel, and nonferrous materials.

C. Impact testing was not required on components or equipment pressure parts having a minimum service temperature of 250°F or more when pressured over 20 percent of the design pressure. Example: Steam line is excluded from brittle fracture test requirement since the steam temperature will be over 250°F when the steam line pressure is at the 20 percent design pressure.

D. Impact testing was not required on components or equipment pressure parts whose rupture could not result in a loss of coolant exceeding the capability of normal makeup systems to maintain adequate core cooling for the duration of a reactor shutdown and orderly cooldown.

E. These criteria apply to components and equipment pressure parts, including flange bolts of the reactor coolant pressure boundary, and do not apply to related components such as anchors, anchor bolts, hangers, suppressors, and restraints.

All components for the Browns Ferry plant were designed and fabricated giving consideration to brittle-fracture control requirements as stated above. However, these specific conditions were not a part of the initial Browns Ferry Units 1 and 2 plant requirements, and due to the status of fabrication on two items, the requirements could not be imposed without scrapping all materials. On Browns Ferry Units 1 and 2 these two items are: (1) feedwater piping through the second containment isolation valve, and (2) the 14-inch HPCI testable check valve (HPCI pump return into feedwater pipe outside the containment). Charpy V-notch impact tests were performed on these items where possible, and results indicate they generally would not meet the conditions under A, above, if they had been imposed.

4.2.5 Safety Evaluation

The reactor vessel design pressure of 1250 psig is determined by an analysis of margins required to provide a reasonable range for maneuvering during operation, with additional allowances to accommodate transients above the operating pressure without causing operation of the safety valves. The design temperature for the reactor vessel (575°F) is based on the saturation temperature of water corresponding to the design pressure.

To withstand external and internal loadings while maintaining a high degree of corrosion resistance, a high-strength, carbon-alloy steel is used as the base metal with an internal cladding applied by weld overlay to the cylindrical shell and bottom head. Use of the ASME Boiler and Pressure Vessel Code, Section III, Class A, pressure vessel code design criteria provides assurance that a vessel designed, built, and operated within its design limits has an extremely low probability of failure due to any known failure mechanism.

The reactor vessel is designed for a 40-year life and will not be exposed to more than 1 X 10^{19} nvt of neutrons with energies exceeding 1 MeV. The reactor vessel is also designed for the transients which could occur during the 40-year life as indicated below. TLAAs have been identified and evaluated for the reactor vessel 60 year operating life. Summaries of these evaluations for the reactor vessel life are provided in Appendix O, Sections O.3.1 and O.3.2.

Type of Cycle	No. of Cycles
Boltup	123
Design hydrostatic test at 1250 psig	130
Startup (100°F/hr heatup rate)	120
Daily reduction to 75 percent power	10,000
Weekly reduction to 50 percent power	2,000
Control rod worth test	400
Loss of feedwater heaters (80 cycles total)	
Turbine trip at 25 percent power	10
Feedwater heater bypass	70
Scram (200 cycles total)	
Loss of feedwater pumps, isolation valves close	10
Turbine trip, feedwater on, isolation valves stay open	40
Reactor overpressure with delayed scram, feedwater stays on, isolation valves stay open	1
Single safety relief valve blowdown	2
All other scrams	147
Improper start of cold recirculation loop	5
Sudden start of pump in cold recirculation loop	5
Shutdown (100°F/hr cooldown rate)	118
Hydrostatic test at 1563 psig	3
Unbolt	123

Stress analysis and load combinations for the reactor vessel are evaluated for the cycles listed above, with the conclusion that ASME code limits are satisfied. The details of assumed loading combinations are described in Appendix C for Class 1 equipment. It is possible that the specified number of cycles for some of the events listed above may be exceeded over the life of the plant. A plant procedure has been implemented at Browns Ferry to maintain surveillance on the number of cycles which have occurred and the resulting fatigue usage factors. When the fatigue usage factor reaches a value of 0.7, the procedure requires a reevaluation to be completed in a timely manner to assure that the allowable fatigue usage factor of 1.0 is not exceeded. Operating limits on pressure and temperature during inservice hydrostatic testing were established using as a guide Appendix G to the ASME Boiler and Pressure Vessel Code, Section III, 1971, which was first added to the code in the summer 1972 addenda. The intent of Appendix G is to set criteria based on fracture toughness to provide a margin of safety against a nonductile failure. The resulting operating limits ensure that a large postulated surface flaw, having a depth of one-quarter of the material thickness and a length of one and one-half of the material thickness, can be safely accommodated in regions of the reactor vessel shell remote from discontinuities. Operating limits on temperature and pressure

when the core is critical were established by using 10 CFR 50, Appendix G, "Fracture Toughness Requirements," paragraph IV.A.2.C. The 1998 Edition of the ASME Section XI Boiler and Pressure Vessel Code including 2000 Addenda was used in the development of the Unit 1 P-T curves. The P-T curve methodology includes the following: 1) the use of K_{1C} from Figure 4200-1 of Appendix A to Section XI and 2) the use of the M_m calculation in the ASME Code paragraph G.2214 of Appendix G to Section XI for a postulated defect normal to the direction of maximum stress. An exemption from specific requirements of 10 CFR Part 50, Appendix G is taken by use of ASME Code Case N-640 for Unit 2 and Unit 3. ASME Code Case N-640 permits the use of an alternative reference fracture curve K_{1c} for RPV materials for use in determining the PT limits. The PT limit curves based on the K_{1c} fracture toughness curve enhance overall plant safety by minimizing challenges to operators since requirements for maintaining a high vessel temperature during pressure testing are lessened. ASME Code Case N-588 methodology was also used as a basis for the PT curves. This code case permits the use of an alternative procedure for calculating applied stress intensity factors during normal operation and pressure test conditions due to pressure and thermal gradients for axial flaws. This methodology is incorporated into the ASME, Section XI Code, 1995 Edition, 1996 Addenda, which is the current code of record for the Unit 2 inservice inspection program. Since Unit 3 uses an earlier code of record for the inservice inspection program, Unit 3 implements the requirements of only the 1995 Edition, 1996 Addenda of ASME Section XI, Appendix G to allow the use of the ASME Code Case N-588 methodology for PT curves. The operating limits are provided in the technical specifications for Browns Ferry. For the purpose of setting these operating limits, the initial RT_{NDT} (nil-ductility reference temperature) was determined from the impact test data taken in accordance with the requirements of the code to which the reactor vessels were designed and manufactured. The maximum NDT temperature allowed by the vessel specifications was 40°F. Although test data on beltline base material show lower NDT temperatures, an assumed RT_{NDT} of 40°F was used in the vessel beltline area, as well as the areas remote from the beltline because the generally accepted NDT temperature for electroslag welds used in the beltline longitudinal seams is 40°F.

The current operating limits on the pressure/temperature (P/T) curves in the technical specifications are based on the following (RT_{NDT}) values. Unit 1 has used 23.1°F for the (RT_{NDT}) value, Unit 2 has used 23.1°F for the (RT_{NDT}) value, and Unit 3 has used 23.1°F for the (RT_{NDT}) value.

For the current P/T curves, fluences were conservatively calculated for licensed operating periods of 38 EFPY for Unit 1, 48 EFPY for Unit 2, and 54 for Unit 3. These periods reflect 60-year reactor pressure vessel operating life and a conservative period of plant operation at 3952 MWt power level. The higher fluence was used to evaluate the vessel against the requirements of 10 CFR 50, Appendix G in accordance with Regulatory Guide 1.99, Revision 2. The end-of-life shelf

energy was evaluated by an equivalent margin analysis (EMA). The results of these evaluations indicated that:

(a) The results of the upper shelf energy EMA for limiting welds and plates for the three vessels remain less than the acceptance criterion in all cases.

(b) The effective full power year (EFPY) shift is slightly increased and, consequently requires a change in the adjusted reference temperature (ART), which is the initial RT_{NDT} plus the shift. The beltline material ART will remain within the 200 degree screening criterion.

In addition to the minimum requirements of the ASME Boiler and Pressure Vessel Code, the following precautions were taken and tests made either to ensure that the initial NDT temperature of the reactor vessel material is low or to reduce the sensitivity of the material to irradiation effects.

a. The material was selected and fabrication procedures were controlled to produce as fine a grain size as practical. It is an objective in fabrication to maintain a grain size of 5 or finer.

b. Drop weight impact tests were performed on each heat and heat treatment charge of all low-alloy steel-plate material in its "as-fabricated" condition.

c. Drop weight impact tests were made on the weld metal, the heat-affected zone of the base metal, and the base metal of the weld test plates simulating seams. If different welding procedures were used for nozzle welds, drop weight tests of similarly prepared coupons were made. The NDT temperature test criteria for the weld and heat-affected zone of the base material are the same as for the unaffected base metal.

d. The actual NDT temperature of the plates opposite the center of the reactor core was determined. In other areas it was sufficient to demonstrate that the two drop weight test specimens did not break at 10°F above the design NDT temperatures. The area of the vessel located opposite the core was fabricated entirely of plate and was not penetrated by nozzles, nor were there any other structural discontinuities in this area which would act as stress risers.

The reactor assembly is designed such that the average annular distance from the outermost fuel assemblies to the inner surface of the reactor vessel is approximately 80 centimeters. This annular volume, which contains the core shroud, the jet pump assemblies, and reactor coolant, serves to attenuate the fast neutron flux incident upon the reactor vessel wall. Using assumptions of plant operation at 3440 Mw(t), 100 percent plant availability, and 40-year plant life, the neutron fluence at the inner

urface of the vessel was calculated to be 3.8 X 10^{17} nvt for neutrons having energies greater than 1 MeV. The results of the analyses of the vessel wall neutron dosimeters which were removed from the Browns Ferry reactor vessels at the end of the first core cycle indicated that the neutron fluence at the inner surfaces of the vessels at the end of 40-year plant life would be 1.56 x 10^{18}, 1.34 x 10^{18}, and 1.31 x 10^{18} nvt for Units 1, 2, and 3, respectively. These results ranged from 3-1/2 to 4 times the calculated fluence of 3.8 x 10^{17} nvt. Thus, additional analyses were required to predict the shifts in RT_{NDT} based on fluence obtained from the analyses of the vessel wall neutron dosimeters. The procedures in Regulatory Guide 1.99, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials," Revision 1, April 1977 were used to predict the RT_{NDT} shifts. Response to Generic Letter 92-01 provides updated fluence data. TLAAs have been identified and evaluated for the reactor vessel 60 year operating life. Summaries of these evaluations for the reactor vessel life are provided in Appendix O, Sections O.3.1 and O.3.2.

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

The fabrication test program was carried out by the reactor vessel vendors on material representative of the formed, heat-treated, and fully fabricated vessel. Tests of base metal and welded joint were performed, and the results were reported during the early stages of vessel construction. Tensile specimens (of 0.505 inch in diameter) from the shell plate material were prepared for various thickness levels of the plate material. These specimens were tested at various temperatures per ASTM Specifications E8 and E21 to determine tensile strength, yield strength, elongation, and reduction of area. Tensile specimens whose gauge diameter is at least 80 percent of the reactor vessel wall thickness were prepared from base metal and weld material. These specimens were tested at room temperature per ASTM Specification E8 to provide stress-strain curves, tensile strength, yield strength, elongation, reduction of area, and macrophotographs of the breaks. Charpy V-notch impact specimens were prepared from base metal and tested per ASTM Specification E23, Type A, to establish curves for determining the transition temperature at which 30 ft-lb of absorbed energy result in ductile fracture for various thickness levels of the plate material.

The Reactor Coolant System was cleaned and flushed before fuel was loaded initially. During the preoperational test program, the reactor vessel and Reactor Coolant System were given a hydrostatic test in accordance with code requirements at 125 percent of design pressure. The vessel temperature is maintained at a minimum of 60°F above the NDT temperature prior to pressurizing the vessel for a hydrostatic test. A hydrostatic test at a pressure not to exceed system operating pressure is made following each removal and replacement of the reactor vessel head. Other preoperational tests included calibrating and testing the reactor vessel

flange seal-ring leakage detection instrumentation, adjusting reactor vessel stabilizers, checking all vessel thermocouples, and checking the operation of the vessel flange stud tensioner.

During the startup test program, the reactor vessel temperatures were monitored during vessel heatup and cooldown to assure that thermal stress on the reactor vessel was not excessive during startup and/or shutdown.

The average rate of reactor coolant temperature change during normal heatup and cooldown is limited to 100°F in any 1-hour period. Only during some postulated events, or in local areas, would this rate of fluid temperature change be exceeded as a result of rapid blowdown, valve operation, or rupture accident.

4.2.6 Inspection and Testing

The inservice inspection and testing program for the reactor vessel and appurtenances is outlined and detailed in Subsection 4.12. Extent and areas of examination, inspection methods, and frequency of examination are established therein.

The surveillance test program provides for the preparation of a series of Charpy V-notch impact specimens and tensile specimens from the base metal of the reactor vessel, weld heat-affected zone metal, and weld metal from a reactor steel joint which simulates a welded joint in the reactor vessel.

The reactor vessel material surveillance program is described in report NEDO-10115, Mechanical Property Surveillance of General Electric BWR Vessels, by J. P. Higgins and F. A. Brant. It describes the specimens, specimen inventory, capsule design, associated equipment, material selection and instructions for handling the specimens. All the requirements of paragraphs 3.1 through 3.3 of ASTM E-185-66 are met. All the requirements of paragraphs 4, 5, 6, 7, and 8 of ASTM E-185-66 are met, except that thermal control specimens discussed in paragraph 4.3 are not used. NEDO-10115, paragraph 5.7 states, "Because the BWR is a constant-temperature device, no special temperature monitoring devices are required." It is felt paragraph 4.3 of E-185-66 is a recommendation rather than a requirement.

The vessel surveillance samples were prepared in accordance with GE purchase specification 21A1111, Rev. No. 9, Attachment B.

The NDT temperatures for the three core region plates were as follows.

Heat No.	Plate No.	NDTT (F)
C2884-2	6-139-19	0
C2868-2	6-139-20	0
C2753-1	6-139-21	-20

The two test plates furnished by Babcock & Wilcox under the requirements of paragraph 3.1.1 of attachment B to specification 21A1111 were fabricated from Heat No. C2884-2 and C2868-2. The two plates were electroslag-welded (B&W Weld Procedure WR-12-4) and heat-treated the same as the core region plates. Tensile and Charpy impact specimen samples were removed as indicated in Figures 3, 4, 5, 6, and 7 of attachment B to 21A1111. (See FSAR Appendices J, K, and L.)

The surveillance test plate 610-0127 was 139 in. long and 60 in. wide, and all excess material is under TVA control in the event that additional material is needed. It is estimated that enough extra material is available for several hundred additional Charpy specimens.

No weak direction specimens were included in the reactor vessel material surveillance program. All Charpy V-notch specimens were taken parallel to the direction of rolling. The majority of developmental work on radiation effects has been with longitudinal specimens. This is considered the best specimen to be used for determination of changes in transition temperature. At the low neutron fluence levels of BWR plants, no change in transverse shelf level is expected and transition temperature changes are minimal.

The specimens and neutron monitor wires were placed near core midheight adjacent to the reactor vessel wall where the neutron exposure is similar to that of the vessel wall (see Subsection 3.3). The specimens were installed at startup or just prior to full-power operation. For Units 1, 2, and 3, Integrated Surveillance Program (ISP) implementation and surveillance specimen schedule withdrawal and testing is governed and controlled by BWRVIP-86 Revision 1-A, the BWRVIP responses to NRC RAIs dated May 30, 2001, December 22, 2001, and January 11, 2005, and the NRC's Safety Evaluation dated February 1, 2002. (NOTE: WRVIP-86, Revision 1-A, was approved by NRC and issues in October 2012, superseding both BWRVIP-86-A and BWRVIP-116.) Surveillance and chemistry data for all representative materials in the BWRVIP ISP have been consolidated into BWRVIP-135 {Integrated Surveillance Program (ISP) Data Source Book and Plant Evaluations.} A test specimen surveillance capsule (the second set of Unit 2 test specimens located at Azimuth 120º) was withdrawn in accordance with the ISP in 2011 during Unit 2 Refueling Outage 16 (U2R16) at approximately 23 EFPY of operation. An additional test specimen surveillance capsule is scheduled for withdrawal during the license

renewal period, this being the third set of Unit 2 test specimens located at Azimuth 300º, which are currently scheduled for removal in the refueling outage closest to without exceeding 40 EFPY of operation. At the present time, this would correspond to Unit 2 Refueling Outage 24 (U2R24) in 2027. Presently, there are no plans to withdraw any capsules from either Unit 1 or Unit 3, as per BWRVIP-135, the BFN Unit 2 capsules provide the best representative plate material for all three units and the best representative weld material for Units 2 and 3. Supplemental Surveillance Program (SSP) Capsules A, B, D, G, E, and I provide the best representative weld material for Unit 1. Test results will provide the necessary data to monitor embrittlement for Units 1, 2, and 3. Since the predicted transition temperature shift of the reactor vessel beltline steel is less than 100°F at end-of-life, the use of the capsules per the ISP meets the requirements of 10 CFR 50, Appendix H, and ASTM E185-82. Revisions to fluence calculations using data obtained from the surveillance capsule specimens will use an NRC approved methodology that meets Regulatory Guide 1.190. {By letter dated August 14, 2008 (EDMS Number L44 080828 014), NRC issued License Amendment 273 for BFN Unit 1, and by letter dated January 28, 2003 (EDMS Number L44 030204 001), NRC issued License Amendment Numbers 279 and 238, for BFN Units 2 and 3 respectively, authorizing adoption of the BWRVIP Integrated Surveillance Program to address the requirements of Appendix H to 10 CFR Part 50.]

TABLE 4.2-1

REACTOR PRESSURE VESSEL MATERIALS

Component	Form	Material	*Spec. (ASTM/ASME)
Heads, Shell	rolled plate	low-allow steel	SA-302 B cc 1339
Closure Flange	forged rings	low-alloy steel	A-508 CL 2 cc 1332-2
Cladding	weld overlay	austenitic stainless steel -inconel	SA-371 type ER309-type ER308 (and carbon content <0.08 w/o)-Inconel 82 and 182
Nozzles	forged shapes	low-alloy steel	A-508-CL2 cc 1332-2
Control Rod Drive Stub Tubes	forged tubes	Inconel	SB-166 cc 1336
Control Rod Drive Housings	pipe	austenitic stainless steel	--
In-Core Housings	pipe	austenitic stainless steel	--
Vessel Supports- External	rolled plate	low-alloy steel	SA-302 Gr.B
Shroud Support- Internal	forging	Inconel	SB-168 Annealed cc 1336
Nozzle Safe Thermal Sleeves	pipe	austenitic stainless steel	SA-312 TP.304
Nozzle Safe Ends	forging	austenitic stainless steel and some low-carbon steel	SA-336-F8/F8M SA-105-2 cc 1332-1
Nozzle for Instrument Penetrations	forging	Inconel	SB-166 cc 1336 para. 1

*cc - Code Case that modifies/augments the material specification.

Table 4.2-2
REACTOR VESSEL DATA

Reactor Vessel	
Inside Diameter, in. (min.)	251 3/8 in.
Inside Length	73 ft 11-1/2 in.
Design Pressure and Temperature, psig @ °F	1250 @ 575
Vessel Nozzles (number and size)	
Recirculation Outlet	2-36 in. to 28 in.
Steam Outlet	4-26 in.
Recirculation Inlet	10-12 in.
Feedwater Inlet	6-12 in.
Core Spray Inlet	2-10 in.
Instrument (one of these is Head Spray)**	2- 6 in.
Control Rod Drive	185- 6 in.
Jet Pump Instrumentation	2- 4 in.
Vent	1- 4 in.
Instrumentation	6- 2 in.
Control Rod Drive Hydraulic System Return *	1- 4 in.
Core Differential Pressure and Liquid Control	1- 2 in.
Drain	1- 2 in.
In-Core Flux Instrumentation	55- 2 in.
Head Seal Leak Detection	2- 1 in.
Approximate Weights (in pounds)	
Bottom Head	207,500
Vessel Shell	842,000
Vessel Flange	106,000
Support Skirt	28,000
Other Vessel Components	65,500
Total Vessel without Top Head	1,249,000
Top Head[1]	252,000
Total Vessel	1,501,000

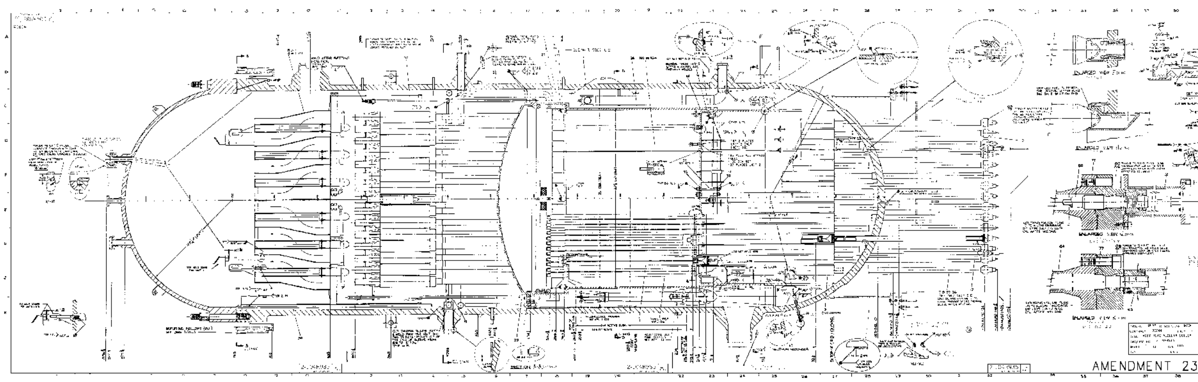
Table 4.2-3
REACTOR VESSEL ATTACHMENTS

	Qty.
Internal Attachments	
Guide Rod Bracket	2
Steam Dryer Support Bracket	4
Dryer Holddown Bracket	4
Feedwater Sparger Bracket	12
Jet Pump Riser Support Pads	20
Jet Pump Diffuser Bracket	20
Core Spray Bracket	4
External Attachments	
Stabilizer Bracket	8
Top Head Lifting Lug	4
Insulation Supports	2
Insulation Support Brackets	12 ea; 2 places
Thermocouple Pad	36

[1] This weight includes 60,095 lbs. which is the weight of the reactor pressure vessel studs, nuts and washers.

* CRD Return Line nozzle is capped.

** Unit 1, 2, and 3 head spray nozzle line is capped.



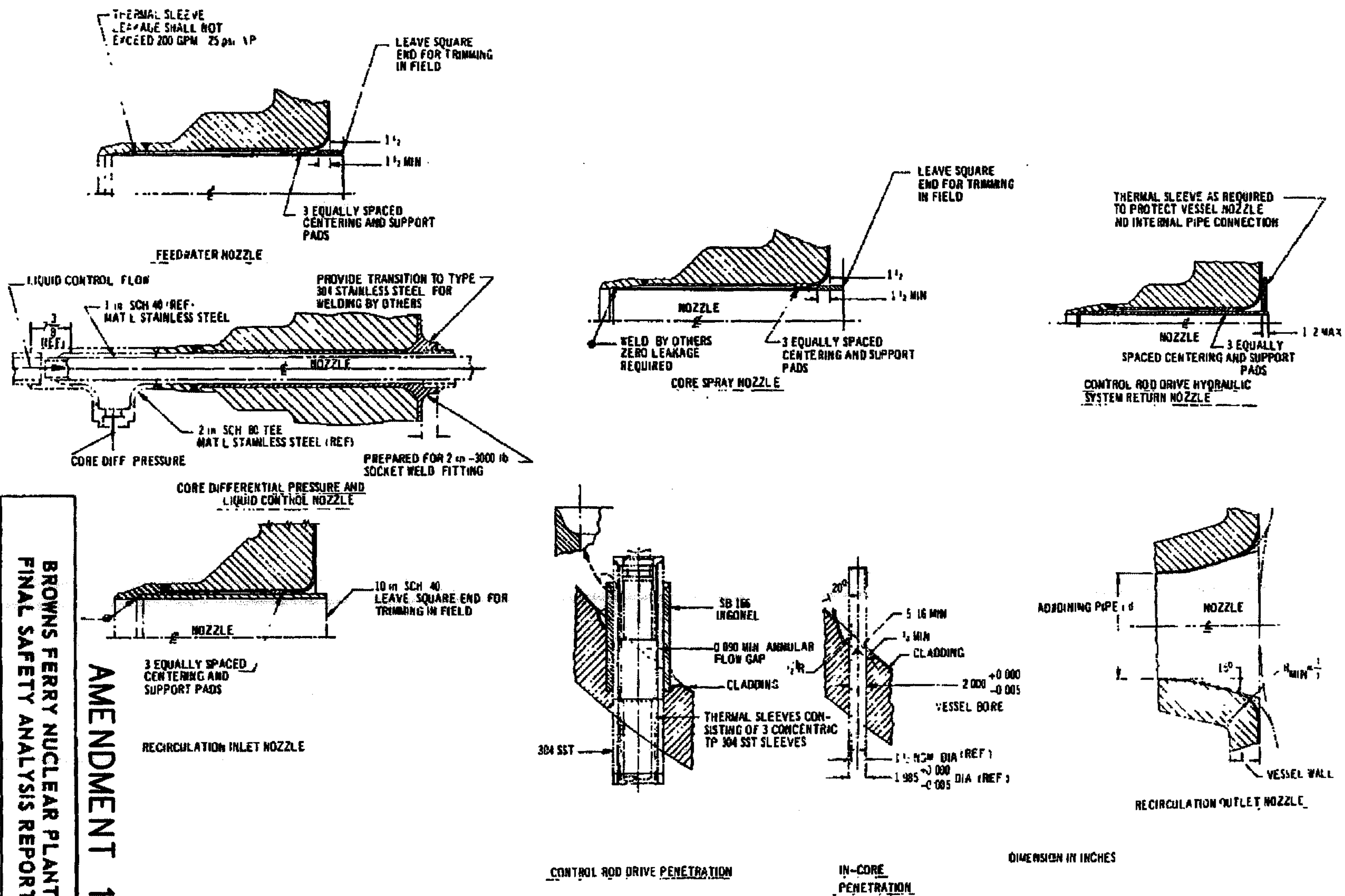
REACTOR BLDG
UNIT 2

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

REACTOR VESSEL

FIGURE 4.2-1

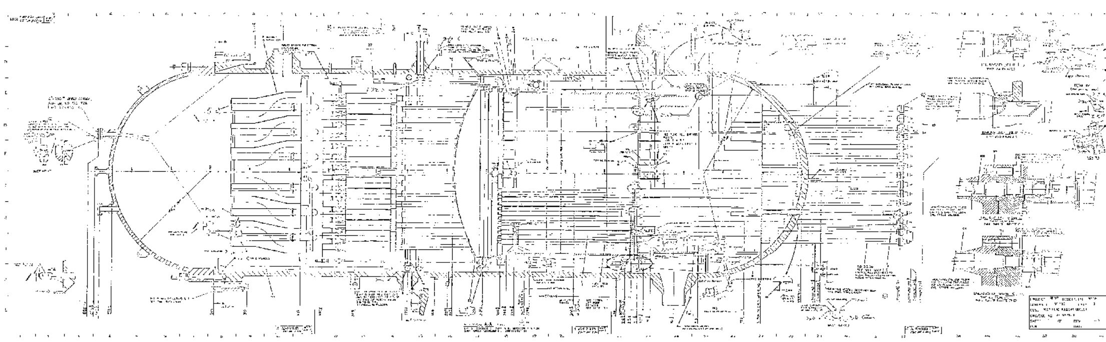
AMENDMENT 23



BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

Reactor Vessel Nozzles and Penetrations
FIGURE 4.2-2

AMENDMENT 16

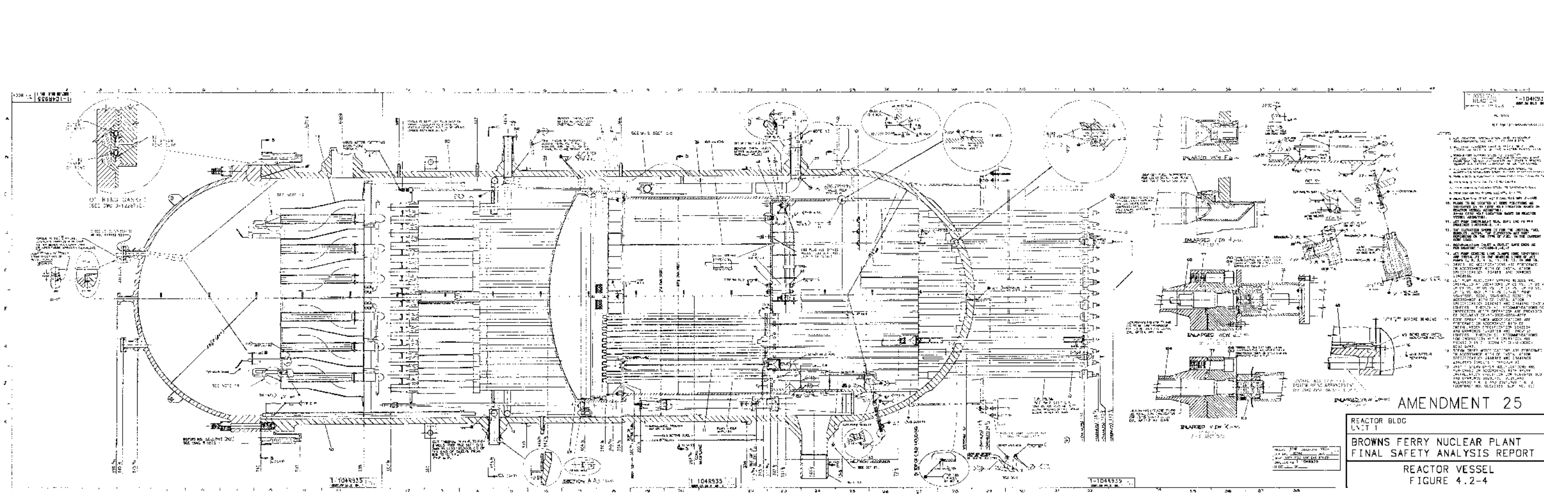


AMENDMENT 24

REACTOR
UNIT 3

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

FIGURE 4.2-3

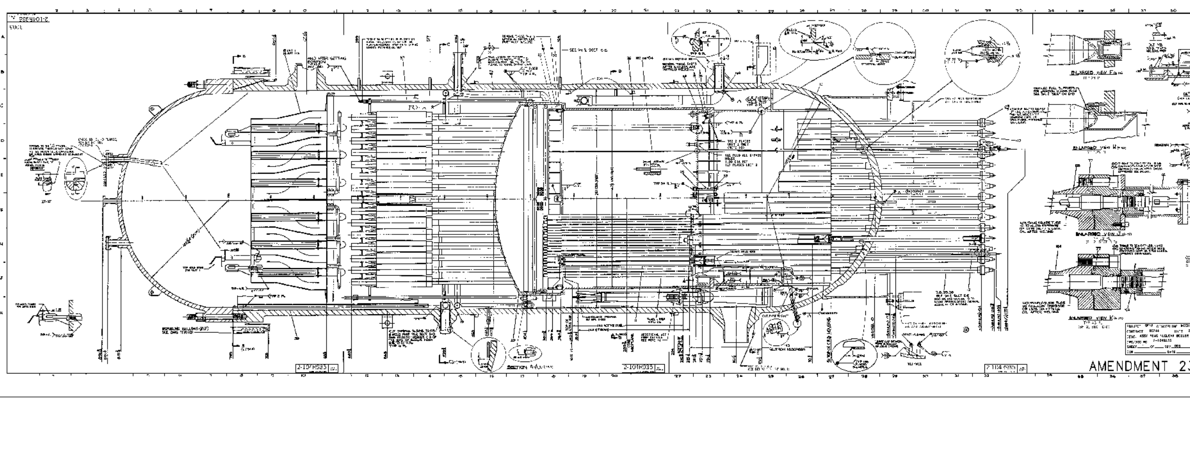


AMENDMENT 25

REACTOR BLDG
UNIT 1

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

REACTOR VESSEL
FIGURE 4.2-4

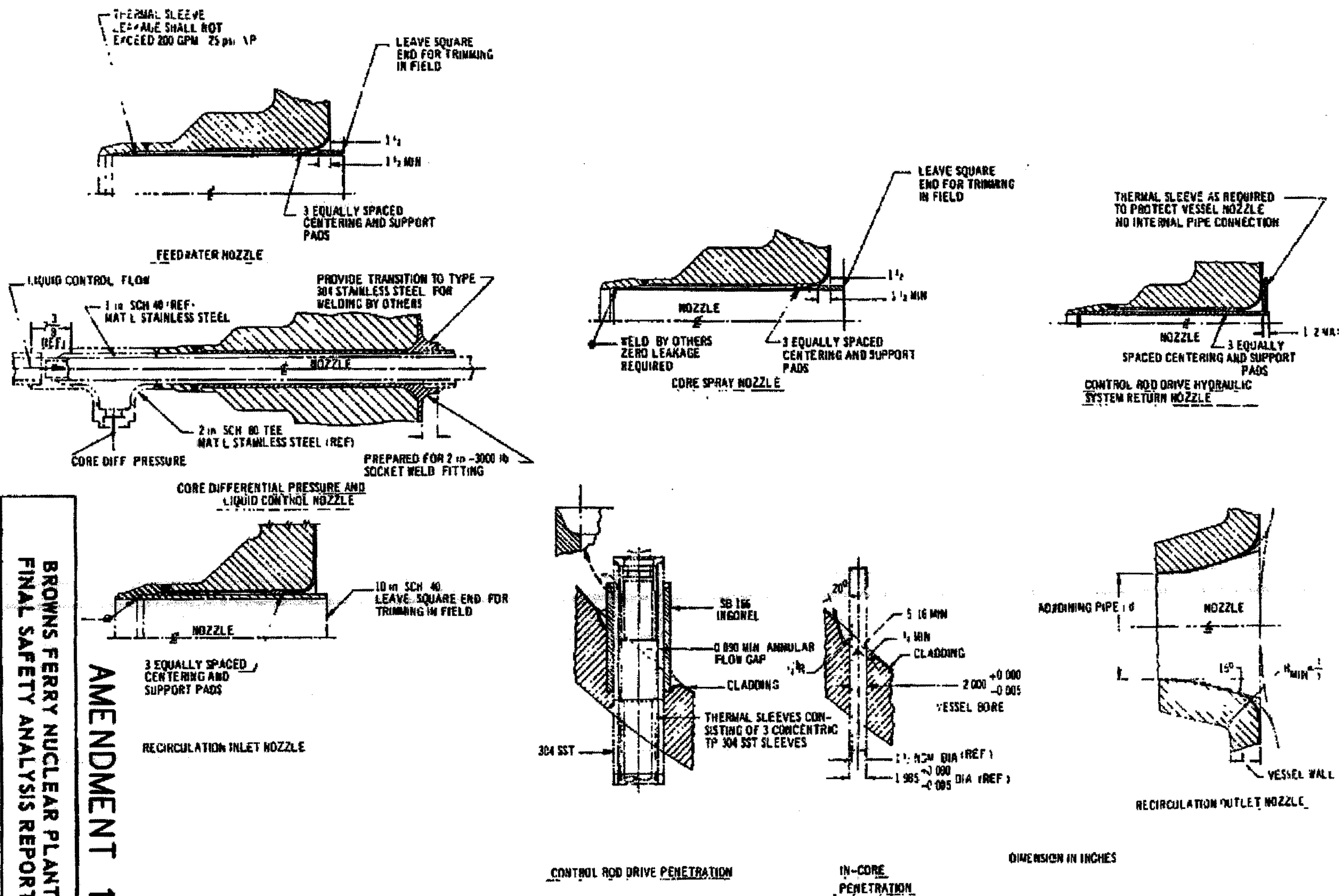


REACTOR BLDG
UNIT 2

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

REACTOR VESSEL
FIGURE 4.2-1

AMENDMENT 23



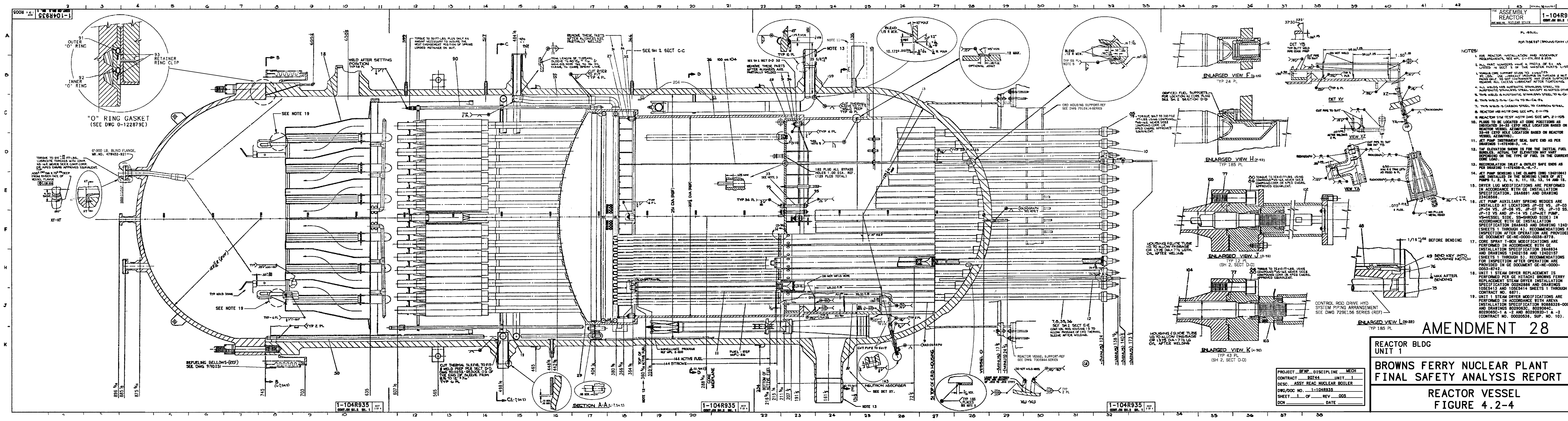
AMENDMENT 16

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

Reactor Vessel Nozzles and Penetrations
FIGURE 4.2-2


NOTES
5. THIS WELD IS AUSTENITIC STAINLESS STEEL TO Ni-Cr-Fe
6. THIS WELD IS Ni-Cr-Fe TO Ni-Cr-Fe
7. THIS WELD IS CARBON STEEL TO CARBON STEEL
10 DELETED
11. THIS CONFIGURATION FOR UNIT 1 ONLY. FOR UNIT 2 SEE TVA DWGS 47W2408-4, -5, -6.
12. TAF ELEVATION SHOWN IS FOR THE INITIAL FUEL BUNDLES. ACTUAL TAF ELEVATION MAY VARY DEPENDING ON THE TYPE OF FUEL IN THE CURRENT CORE LOAD.
13. FOR ADDITIONAL JET PUMP DETAILS REFERENCE THE FOLLOWING DRAWINGS PER GE CONTRACT 00001704:
124D1567-SERIES...SLIP JOINT CLAMPS
124D1243-SERIES...SLIP JOINT CLAMPS
124D2086-SERIES...SLIP JOINT CLAMPS
124D1566-SERIES...AUXILIARY WEDGES
177D4693..........AUXILIARY WEDGES
234C7181-SERIES...INLET-MIXER MAIN WEDGE
234C5141..........INLET-MIXER MAIN WEDGE
177D4867..........INLET-MIXER MAIN WEDGE
124D1029..........INLET-MIXER MAIN WEDGE
ALSO SEE 0-730E883, TABLE 2, FOR JET PUMP MODIFICATION COMPLETION STATUS.
14. JET PUMP SENSING LINE CLAMPS (DWG 124D1064) ARE INSTALLED ON THE SENSING LINES OF JET PUMPS 1,2,3,4,5,11,12,13,14 AND 15.
15. UNIT 3 STEAM DRYER REPLACEMENT IS PERFORMED PER GE HITACHI BROWNS FERRY REPLACEMENT STEAM DRYER INSTALLATION SPECIFICATION 002N2886 AND DRAWINGS 105E5413 AND 105E5414 SHEETS 1 THROUGH 10, CONTRACT NO. 6871.
6"-900# BLIND FLANGE, FOR MK NO SEE TVA DWG 47W452-8
CONTROL ROD DRIVE HYD SYSTEM PIPING ARRANGEMENT SEE DWG 729E156 (REF)
ENLARGED VIEW L
TYP 185 PLACES
49 BEND KEY INTO HOUSING NOTCH
MAX AFTER BENDING
ENLARGED VIEW J
ENLARGED VIEW K
ENLARGED VIEW H
TYP 185 PLACES
ENLARGED VIEW F
TYP 24 PLACES
REFUELING BELLOWS-REF
SECTION A-A
PROJECT BFNP DISCIPLINE MECH
CONTRACT 91750 UNIT 3
DESC. ASSY REAC NUCLEAR BOILER
DWG/DOC NO. 3-104R935
SHEET 1 OF REV 006
DCN DATE
AMENDMENT 28
REACTOR
UNIT 3
BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

FIGURE 4.2-3



AMENDMENT 28

REACTOR BLDG
UNIT 1

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

REACTOR VESSEL
FIGURE 4.2-4

4.3 REACTOR RECIRCULATION SYSTEM

4.3.1 Power Generation Objective

The objective of the Reactor Recirculation System is to provide forced cooling of the core and a variable moderator (coolant) flow to the reactor core for adjusting reactor power level.

4.3.2 Power Generation Design Basis

a. The Reactor Recirculation System shall provide sufficient subcooled water to the core during normal power operation to maintain normal operating temperatures.

b. The Reactor Recirculation System shall operate over a flow control range of 20 percent to 105 percent flow to allow power variation.

c. The Reactor Recirculation System shall be designed to minimize maintenance situations that would require core disassembly and fuel removal.

4.3.3 Safety Design Basis

a. The Reactor Recirculation System, including the recirculation pump trip (RPT) feature, shall be designed so that adequate fuel barrier thermal margin is assured following recirculation pump system malfunctions and postulated transients (such as turbine-generator trip or load rejection).

b. The Reactor Recirculation System shall be designed so that failure of piping integrity does not compromise the ability of the reactor vessel internals to provide a refloodable volume.

4.3.4 Description

The Reactor Recirculation System consists of the two recirculation pump loops external to the reactor vessel which provide the driving flow of water to the reactor vessel jet pumps (see Figures 4.3-1 and 4.3-2a sheets 1, 2, and 3). Each external loop contains one high-capacity motor-driven recirculation pump and two motor-operated gate valves for pump maintenance. Each pump discharge line contains a venturi-type flowmeter nozzle. The recirculation loops are a part of the nuclear system process barrier and are located inside the drywell containment structure. The jet pumps are reactor vessel internals and their location and mechanical design are discussed in Subsection 3.3, "Reactor Vessel Internals Mechanical Design." However, certain operational characteristics of the jet pumps are discussed in this subsection. A summary of the characteristics of the Reactor Recirculation System is presented in Table 4.3-1.

The recirculated coolant consists of saturated water from the steam separators and dryers which has been subcooled by incoming feedwater. This water passes down the annulus between the reactor vessel wall and the core shroud. A portion of the coolant exits from the vessel and passes through the two external recirculation loops to become the driving flow for the jet pumps. The two external recirculation loops each discharge high pressure flow into an external manifold from which individual recirculation inlet lines are routed to the jet pump risers within the reactor vessel. The remaining portion of the coolant mixture in the annulus becomes the driven flow for the jet pumps. This flow enters the jet pumps at the suction inlet and is accelerated by the driving flow. The driving and driven flows are mixed in the jet pump throat section resulting in partial pressure recovery. The balance of recovery is obtained in the jet pump diffusing section (see Figure 4.3-3). The adequacy of the total flow to the core is discussed in Subsection 3.7, "Thermal and Hydraulic Design." Tests have been conducted and documented[1] to show that the jet pump design is sound and that jet pump operation is stable and predictable.

The pump is started at slow speed with the discharge valve closed. Pump speed is not increased until after the discharge valve has been opened utilizing the jogging circuit that opens the valve in steps. There is actually a very low probability that a recirculation loop that has been allowed to cool would need to be placed in service again with the nuclear system hot. A valid reason for closing both the pump discharge valve and the suction valve is to prevent leakage out of that portion of the recirculation loop between the valves, e.g., excessive leakage through the pump mechanical seal. A leak of this nature cannot be repaired without shutting the plant down to permit access to the drywell; the nuclear system would in all probability have been cooled prior to repairing the leak.

Since the removal of Reactor Recirculation System valve internals requires unloading of the nuclear fuel, the valves are provided with high-quality back seats and trim to facilitate stem packing renewal without draining the vessel and to provide adequate leak tightness during normal operation.

The Reactor Recirculation System valves are designed and constructed to meet the requirements of USAS B31.1.0, 1967 edition, with added GE requirements which were implemented in lieu of the outdated B31 Nuclear Code Cases-N2, N7, N9, and N10. The valves are designed to operate under maximum prevailing operating conditions and postulated accident conditions in the drywell.

1 "Design and Performance of G.E. BWR Jet Pumps," General Electric Company, Atomic Power Equipment Department, Sept. 1968 (APED-5460).

The Units 1 and 3 Reactor Recirculation Header equalizer valves were removed during the respective unit recirculation piping replacements in order to reduce the number of welds and, therefore, minimize susceptibility to Intergranular Stress Corrosion Cracking (IGSCC).

Under all operating conditions (Unit 2 only), one equalizer valve in the line between the two pump discharge lines shall be open and the other valve shall be closed (both valves having motive power removed). This is to prevent pressure buildup due to ambient and conduction heating of the water between the equalizer line valves.

The idle pump loop is not completely valved off if it is desired to return the idle loop to service prior to the next reactor cooldown (such as VFD repair for Units 1, 2, and 3). The recirculation pump casing allowable heatup rate is 100°F per hour, the same as the reactor vessel. It is possible to keep the idle loop hot with the equalizer line valved off (Unit 2 only) and the idle loop valves left open, permitting the pressure head created by reverse flow through the idle jet pumps to cause reverse flow through the idle loop. However, it is first necessary to stop the pump rotation by closing either the pump suction or discharge valve until pump rotation stops. Once the oil film is squeezed out of the pump thrust bearing, the pump will not rotate even with both the suction and discharge valves open.

Following one recirculation pump operation, an operational restriction is applied such that the discharge valve of the low speed pump may not be opened unless the speed of the faster pump is less than 50 percent of its rated speed. This limitation provides assurance when going from one-to-two pump operation that excessive vibration of the jet pump risers will not occur.

The feedwater flowing into the reactor vessel annulus during operation provides subcooling for the fluid passing to the recirculation pumps, thus determining the additional net positive suction head (NPSH) available beyond that provided by the pump location below the reactor vessel water level. If feedwater flow is below 17 percent, the recirculation pump speed is automatically limited.

The recirculation pumps can be operated during nuclear system heatup for hydrostatic tests. At this time, they act in conjunction with any contribution from reactor core decay heat to raise nuclear system temperature above the limit imposed on the reactor vessel by nil-ductility transition temperature (NDTT) considerations so that the hydrostatic test can be conducted.

A decontamination connection is provided in the piping on the suction side of the pump to permit flushing and decontamination of the pump and adjacent piping. This connection is arranged for convenient and rapid connection of temporary piping.

The piping low point drain is used during flushing or decontamination to conduct crud away from the piping low point and is also designed for connection of temporary piping.

Each recirculation pump is a single-stage, variable-speed, vertical, centrifugal pump equipped with mechanical shaft seal assemblies. The pump is capable of stable and satisfactory performance while operating continuously at any speed corresponding to a power supply frequency range of 11.5 to 57.5 Hz. For loop startup, each pump operates at a speed corresponding to a power supply frequency of 11.5 Hz with the discharge gate valve closed.

The recirculation pump shaft seal assembly consists of two seals built into a cartridge which can be readily replaced without removing the motor from the pump. The seal assembly is designed to require minimum maintenance over a long period of time, regardless of whether the pump is stopped or operating, and seal over a wide range of pressures and temperatures. The original seal design for Units 1, 2, and 3 has been changed to a seal assembly with an extended design life. Each individual seal in the cartridge is capable of sealing against pump design pressure so that any one seal can adequately limit leakage in the event that the other seal should fail. A breakdown annulus is provided along the pump shaft to reduce leakage in the event of a gross failure of both shaft seals. Provision is made for monitoring the pressure drop across each individual seal as well as the cavity temperature of each seal. Provision is also made for piping the seal leakage to a flow measuring device. Various control room alarms indicate improper seal performance.

The Reactor Building Closed Cooling Water System and the Control Rod Drive Hydraulic System provide cooling to the recirculation pump seals. If either one of these systems is operating, recirculation pump operation without the second cooling system may continue with no harm to the seals. If both seal cooling systems are inoperable (e.g., due to a loss of AC power), the pump seals will overheat approximately 7 minutes after the total loss of cooling and seal deterioration may begin.

Based on fluid loss analysis of extremely degraded seals, the leakage is less than 70 gpm. This amount of leakage will not lead to a safety concern but may degrade the seals such that they would have to be repaired prior to resuming operation.

Each recirculation pump motor is a variable-speed AC, electric motor which can drive the pump over a controlled range of 20 percent to 102 percent of rated pump speed. The motor is designed to operate continuously at any speed within the power supply frequency range of 11.5 Hz to 57.5 Hz. Recirculation pump motors are designed, constructed, and tested in accordance with the applicable sections of the NEMA Standards.

For Units 1, 2, and 3, a variable frequency drive unit located outside the drywell supplies power to each recirculation pump motor. Minimum speed corresponds to a frequency of 11.5 Hz.

For Units 1, 2, and 3, the combined rotating inertia of the recirculation pump and motor are modeled consistent with a coastdown of flow following loss of power to the drive motors, so that the core is adequately cooled during the loss-of-power transient. The effective inertia of these devices are specified in the following form, which takes into account the torque and speed conditions on each rotating shaft.

$$\frac{J\omega}{gTo} = \text{Time}$$

where

- J = inertia (lb-ft^2)
- ω = rated speed (rad/sec)
- g = gravitational constant (32.3 ft/sec^2)
- To = torque at rated speed (ft-lb).

From this equation, the required inertia (J) is calculated.

The recirculation pumps are Classified as machinery, and, as such, are specifically exempt from the jurisdiction of any section of the ASME Boiler and Pressure Vessel Code or of the USA Standard Code for Pressure Piping. The standards of the Hydraulic Institute are the only standards which are applicable; however, they are more pertinent to the testing and performance of the pump and consequently provide little or no guidance in the areas of casing quality and structural integrity.

To assure that the pump casing can withstand a pressure equivalent to that inside the reactor vessel, the pump casing is designed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Class C, as far as this code can be applied. The requirements of Section III of the ASME Boiler and Pressure Vessel Code for Class C vessels (1965 edition) are used as a guide in calculating the thickness of pressure-retaining parts of the recirculation pumps. The casings and forgings are fabricated from austenitic stainless steel. Class C is used because the pump casing does not experience temperature transients as severe as those that portions of the reactor vessel and certain piping connections experience; therefore, it is not necessary to make the cyclic analysis required for Class A equipment.

The design objective for the recirculation pump casing is a useful life of 40 years, accounting for corrosion, erosion, and material fatigue. For the 60 year operating life, the recirculation pump aging effects will be managed using the ASME Section XI Subsections IWB, IWC, and IWD Inservice Inspection Program, Chemistry Control Program, and BWR Stress Corrosion Cracking Program described in Appendix O, Section O.1.4. O.1.5, and O.1.10. Material fatigue for the 60 year operating life has been evaluated as a Time Limited Aging Analysis (TLAA). The summary of this evaluation is provided in Appendix O, Sections O.3.2.3 and O.3.2.4. The pump-drive motor, impeller, and wear rings are designed for as long a life as is practical. The design objective is to provide a unit which will not require removal from the system for rework or overhaul at intervals of less than 5 years.

The recirculation system piping is of stainless steel construction and is designed and constructed to meet the requirements of the USA Standard Code for Pressure Piping, Power Piping, USAS B31.1.0, 1967 edition, and the additional requirements of GE design and procurement specifications which were implemented in lieu of the outdated B31 Nuclear Code Cases-N2, N7, N9, and N10. The suction and discharge pipes are welded to the pump casing.

The coolant in the nuclear process system is at high pressure and contains a large amount of energy. Substantial failure of the nuclear process system could result in a rapid loss of coolant. Although loss of the moderator (coolant) would render the reactor core subcritical, lack of cooling could cause overheating of the reactor core from residual and decay heat, leading to fuel damage and fission product release. The Core Standby Cooling Systems (which adequately cool the reactor core following a design basis loss-of-coolant accident) and the primary containment and containment cooling systems (which control the release of fission products and absorb the energy released by the accident) are not intended to diminish the overall design objective of the entire nuclear system (to design and construct a nuclear system which will not fail). The intent of using Section III of the ASME Boiler and Pressure Vessel Code and USAS B31.1.0, with added GE requirements for the recirculation system, is to design piping systems of high quality.

The Reactor Recirculation System, except for the VFDs on Units 1, 2, and 3, is designed as Class I equipment (see Appendix C) to resist sufficiently the response motion at the installed location within the supporting structure for the Design Basis Earthquake, with the pump assumed filled with water for the analysis. Vibration snubbers located at the top of the motor and at the bottom of the pump casing are designed to resist the horizontal reactions.

The recirculation piping, valves, and pumps are supported by constant support hangers and by sway braces to avoid the use of piping expansion loops which would be required if the pumps were anchored. In addition, the recirculation loops are provided with a system of restraints designed to limit pipe motion so that reaction

forces associated with any split or circumferential break do not jeopardize containment integrity. This restraint system provides adequate clearance for normal thermal expansion movement of the loop. The spacing between limit stops is set on the basis that a split pipe retains its structural load-resisting characteristics. Impact loading is negligible on limit stops, since possible pipe movement is limited to slightly more than the clearance required for thermal expansion movement.

The recirculation system piping, valves, and pump casings are covered with all-metal, reflective, thermal insulation having an average maximum heat transfer rate of 80 Btu/hr-ft^2 with the system at rated operating conditions. The insulation is prefabricated into components for field installation. Removable insulation is provided at various locations to allow for periodic inspection of the insulated equipment.

4.3.5 Safety Evaluation

Reactor Recirculation System pump malfunctions that pose threats of damage to the fuel barrier are described and evaluated in Chapter 14.0, "Plant Safety Analysis." There it is shown that none of the malfunctions results in fuel damage; thus, the recirculation system has sufficient flow coastdown characteristics to maintain fuel thermal margins during abnormal operational transients. In addition, in order to achieve a more rapid core reactivity reduction in the event of a turbine or generator trip (thereby limiting the magnitude of the fuel thermal transient), a recirculation pump trip (RPT) feature has been added. By utilizing the recirculation pump trip in response to a turbine-generator trip or load rejection, the MCPR margin is reduced, allowing normal operation at higher power than without the RPT feature. This satisfies safety design basis a. The RPT feature is described in Subsection 7.9.

The core-flooding capability which is provided by a jet pump design is pictured in Figure 4.3-4. There is no postulated recirculation line break which can prevent reflooding of the core to the level of the jet pump suction inlet. The core-flooding capability of a jet pump design is discussed in detail in the Core Standby Cooling Systems document filed with the AEC as a GE Topical Report.[2] This satisfies safety design basis b.

The Reactor Recirculation System piping and pump design pressures (see Table 4.3-1) are based on peak steam pressure in the reactor dome, plus the static head above the lowest point in the recirculation loop, plus dynamic head due to system operation. Piping and related equipment pressure parts are chosen and analyzed in accordance with applicable codes. Use of the listed code design

---

2 Ianni, P.W., "Core Standby Cooling Systems for Boiling Water Reactors," General Electric Company, Atomic Power Equipment

criteria provides assurance that a system designed, built, and operated within design limits has an extremely low probability of failure due to any known failure mechanism.

4.3.6 Inspection and Testing

Quality control methods were used during the fabrication and assembly of the Reactor Recirculation System to assure that the design specifications were met. Inspection and testing were carried out in accordance with USAS B31.1.0. The reactor coolant system was thoroughly cleaned and flushed before fuel was loaded initially.

During the preoperational test program, the Reactor Recirculation System was given a hydrostatic test at 125 percent of reactor vessel design pressure. A hydrostatic test at a pressure not to exceed system operational pressure is made following each removal and replacement of the reactor vessel head. Other preoperational tests on the Reactor Recirculation System included operating valves and verifying that seal leakage was small enough to permit pump maintenance work, operating pumps and variable frequency drive (Units 1, 2, and 3), and checking flow control transient operation.

During heatup in the startup test program, the horizontal and vertical motions of the Reactor Recirculation System piping and equipment were observed and adjustments of supports were made, as necessary, to assure that components were free to move as designed. Nuclear system responses to recirculation pump trips at rated temperatures and pressure were evaluated during the startup tests, and the plant power response to recirculation flow control was determined.

Inservice inspection is considered in the design of the Reactor Recirculation System to assure adequate working space and access for inspection of selected components. 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, including areas of known stress concentrations and locations where cyclic strain or thermal stress might occur. The recirculation pump casing and valve bodies can be examined when the pump or valve is disassembled for normal maintenance. The piping connection welds can be examined to the extent practical within the limitations of design, geometry, and materials of construction of the components. The inservice inspection and testing program for the recirculation system is detailed in Subsection 4.12.

Table 4.3-1

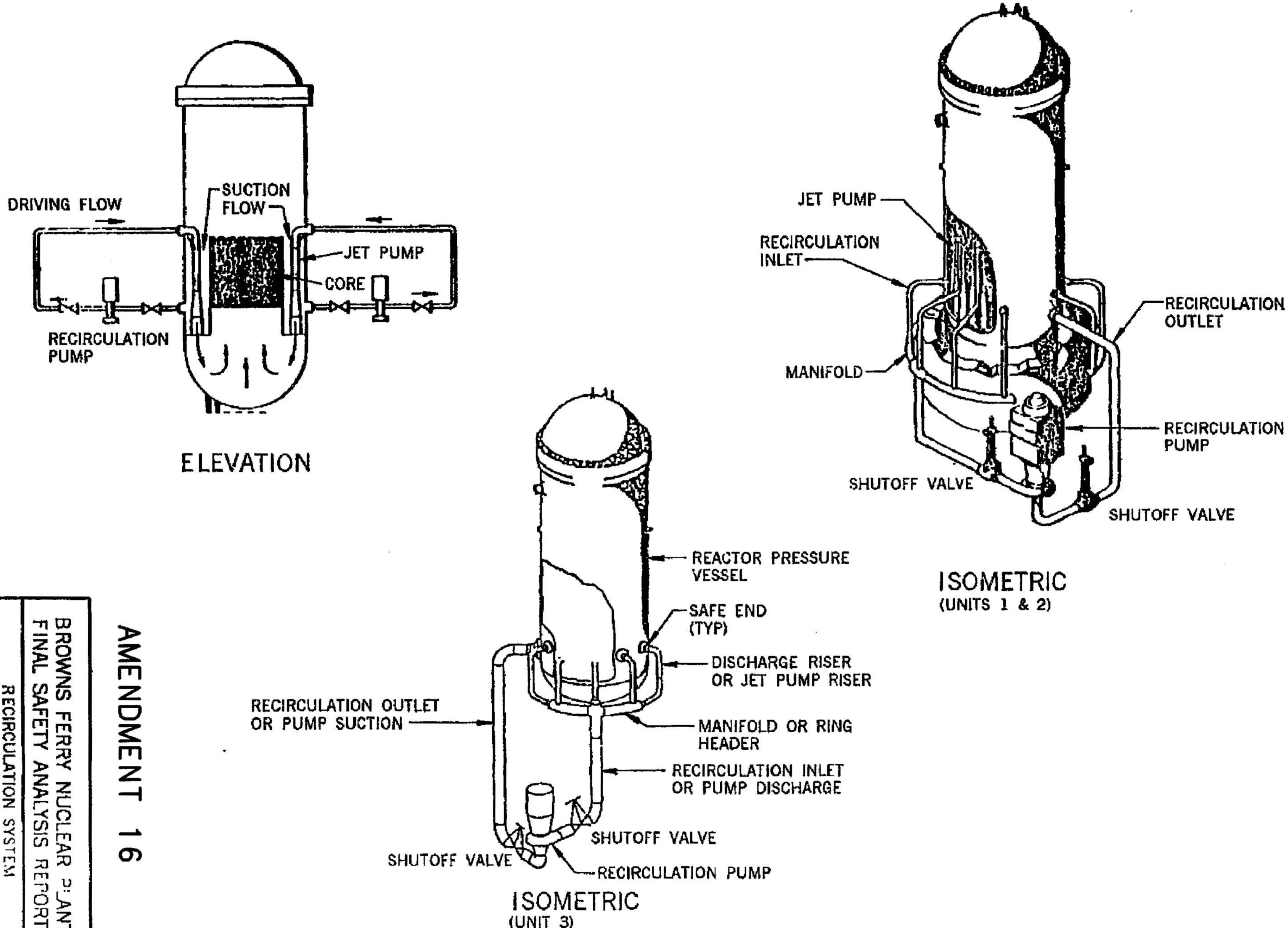
REACTOR RECIRCULATION SYSTEM
DESIGN CHARACTERISTICS
(3952 MWt)

External Loops	
Number of Loops	2
Pipe Sizes (nominal o.d.)	
Pump Suction,in	28
Pump Discharge,in	28
Discharge Manifold,in	22 (Units 1 & 2), 12 & 22 (Unit 3)
Recirculation Inlet Line,in	12
Equalizer Line,in (Unit 2 only)	22
Design Pressure (psig)/Design Temperature(°F)	
Suction Piping	1148/562
Discharge Piping	1326/562
Pumps	1500/575
Operation at 3952 MWt (100% Core Flow)	
Recirculation Pump	
Flow gpm (approximate)	47,400
Flow,lb/hr	17.95×10^6
Total Developed Head,ft	643
Suction Pressure (static),psia	1,056
Available NPSH*(min.),ft	558
Water Temperature (max.),°F	529
Pump Brake HP (min.),hp	6,734
Flow Velocity at Pump Suction, fps (approximate)	30.5
Variable Frequency Drives (Units 1, 2, and 3) and Power Supply	
Frequency (operating range),Hz	11.5-57.5
Total Required Power to Variable Frequency Drive (Units 1, 2, and 3)	
HP/set	7,194
HP total	14,388
Jet Pumps	
Number	20
Total Jet pump flow,lb/hr	102.5×10^6
Throat I.D.,in	8.18
Diffuser I.D.,in	19.0
Nozzle I.D.,in. (representative)	3.14
Diffuser Exit Velocity,ft/sec	15.3

* Includes velocity head.

Table 4.3-1b

(Deleted)

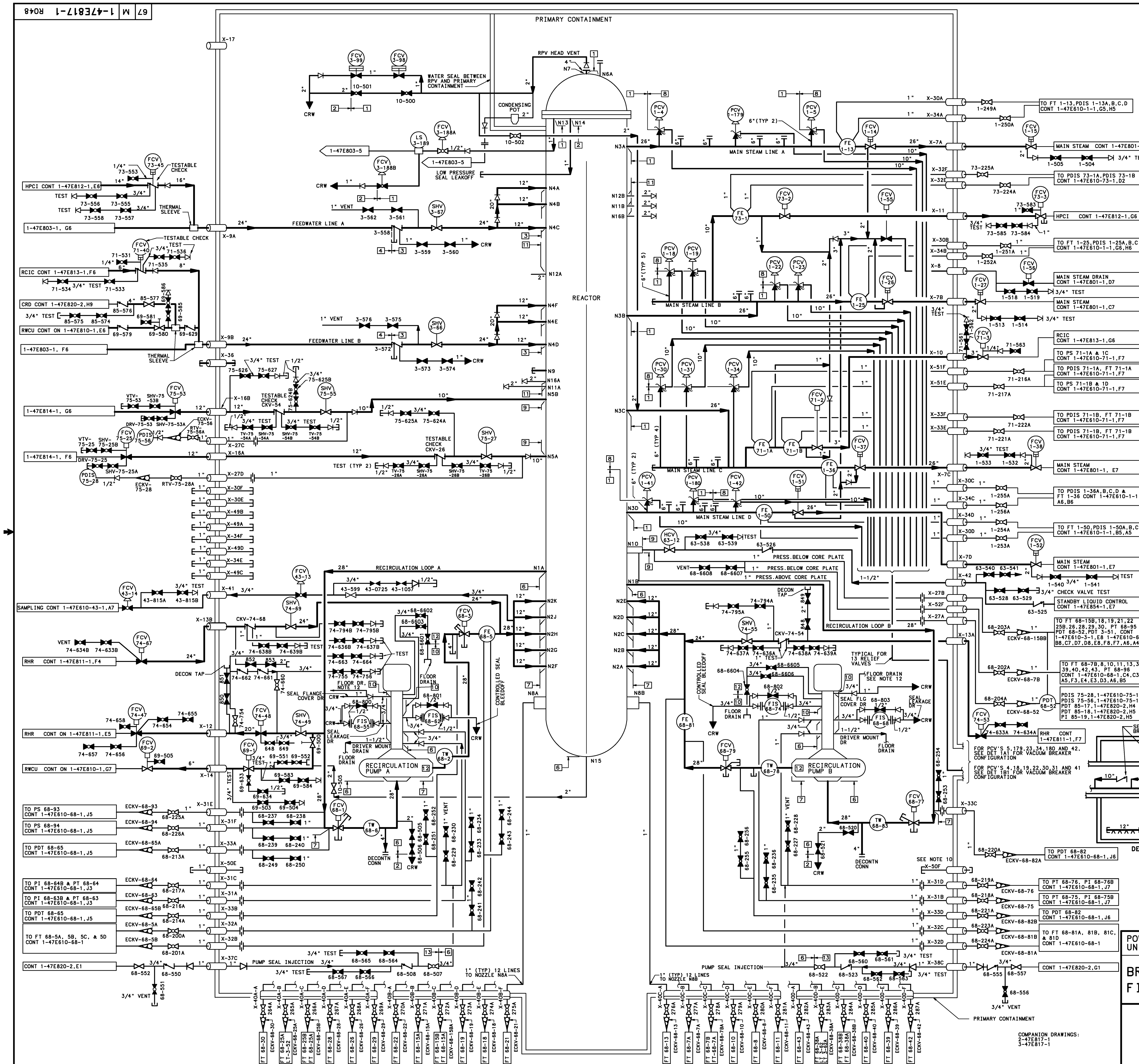


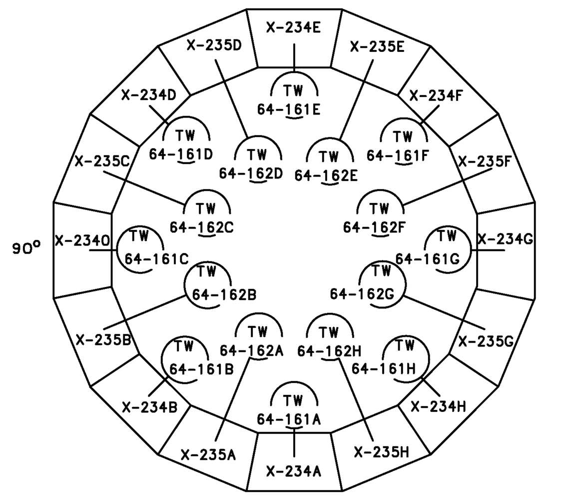
AMENDMENT 16

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

RECIRCULATION SYSTEM
ELEVATION ISOMETRIC

FIGURE 4.3-1





TORUS TEMPERATURE
MONITORING SYSTEM

SYSTEM PRESS.- TEMP DATA		
LINE MK NO.	DESIGN PRESSURE (PSIG)	DESIGN TEMP(°F)
1	1146	562
2	150	300
3	1375	562
4	1375	376
5	1250	376
6	1148	562
7	1326	562
8	625	500
9	1250	575
10	150	212
11	1146	575
12	1500	575
13	1750	150

TABLE A

SYSTEM 1	SYSTEM 10	
	2-1/2"	10"
PCV-1-4	10-506	10-521
PCV-1-5	10-507	10-522
PCV-1-18	10-508	10-523
PCV-1-19	10-509	10-524
PCV-1-22	10-510	10-525
PCV-1-23	10-511	10-526
PCV-1-30	10-512	10-527
PCV-1-31	10-513	10-528
PCV-1-34	10-514	10-529
PCV-1-41	10-515	10-530
PCV-1-42	10-516	10-531
PCV-1-179	10-519	10-532
PCV-1-180	10-520	10-533

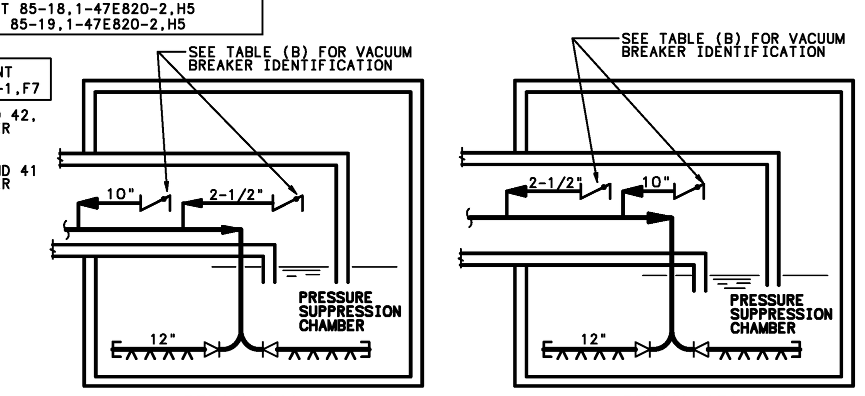
TABLE B
SEE DETAILS 1A1 AND 1B1

NOTES:

1. ALL VALVES ARE THE SAME SIZE AS PIPING UNLESS OTHERWISE NOTED.
2. ALL VALVE BODY DRAINS, BONNET VENTS AND PACKING LEAKOFFS ARE 1/2" UNLESS OTHERWISE NOTED.
3. ALL VALVE NUMBERS SHALL BE PREFIXED WITH THE UNIT NUMBER AND SYSTEM NUMBER I.E. 1-1-5.
4. ALL DRAINS SHOWN , INDICATES CLOSED SYSTEM.
5. CRW DRAINS ARE CONNECTED TO THE REACTOR BUILDING EQUIPMENT DRAIN SUMP.
6. $\boxed{1}$ ETC, DENOTES DESIGN PRESSURE AND TEMPERATURE AS GIVEN IN TABLE (A) SYSTEM PRESS.-TEMP DATA THIS DRAWING.
7. HYDROSTATIC TESTING SHALL BE IN ACCORDANCE WITH THE APPLICABLE CODES.
8. THE DESIGN PRESSURE AND TEMP OF ALL DRAIN AND VENT LINES THROUGH THE LAST ISOLATION VALVE SHALL BE THE SAME AS THE PROCESS LINE.
9. 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.
10. ALL SENSING LINES WITH EXCESS FLOW CHECK VALVES EXCEPT X-40A-A THROUGH X-40D-F HAVE AN ORIFICE COUPLING, (REFER TO DETAIL P20 ON 0-47W600-20).
11. UNIDS ON DRAWING ARE FOR REFERENCE ONLY AND ARE ABBREVIATED TO MEET SPACE CONSTRAINTS. REFER TO MEL FOR COMPLETE UNIDS.
12. SEAL FLANGE COVER IS VENTED THROUGH AN OPEN-ENDED "TEE" CONNECTED THROUGH A DRIP LOOP.

REFERENCE DRAWINGS:
0-47E800-1..........FLOW DIAGRAM-GENERAL PLANT SYSTEMS
0-47E800-2..........MECHANICAL SYMBOLS & FLOW DIAGRAM DRAWING INDEX
1-47E811-1..........FLOW DIAGRAM-RHR SYSTEM
1-47E814-1..........FLOW DIAGRAM-CORE SPRAY SYSTEM
1-47E610-75-1.......CONTROL DIAGRAM-CORE SPRAY SYSTEM
1-47E610-3-1........CONTROL DIAGRAM-REACTOR FEEDWATER SYSTEM
1-47E812-1..........FLOW DIAGRAM-HPCI SYSTEM
1-47E803-1..........FLOW DIAGRAM-REACTOR FEEDWATER SYSTEM
1-47E813-1..........FLOW DIAGRAM-RCIC SYSTEM
1-47E820-2..........FLOW DIAGRAM-CONTROL ROD DRIVE SYSTEM
1-47E810-1..........FLOW DIAGRAM-REACTOR WATER CLEANUP SYSTEM
1-47E610-68-1.......CONTROL DIAGRAM-REACTOR WATER RECIRCN SYSTEM
1-47E610-43-1.......CONTROL DIAGRAM-SAMPLING & WATER QUALITY SYSTEM
1-47E801-1..........FLOW DIAGRAM-MAIN STEAM
1-47E801-2..........FLOW DIAGRAM-MAIN STEAM
1-47E610-1-SERIES...CONTROL DIAGRAM-MAIN STEAM
1-47E610-74-SERIES..CONTROL DIAGRAM-RHR SYSTEM
1-47E610-73-1.......CONTROL DIAGRAM-HPCI SYSTEM
1-47E610-71-1.......CONTROL DIAGRAM-RCIC SYSTEM
1-47E610-69-1.......CONTROL DIAGRAM-REACTOR WATER CLEANUP SYSTEM
1-47E854-1..........FLOW DIAGRAM-STANDBY LIQUID CONTROL SYSTEM
1-47E852-2..........FLOW DIAGRAM-CLEAN RADWASTE SYSTEM
1-47E610-64-1,2.....CONTROL DIAGRAM-PRIMARY CONTAINMENT SYSTEM
1-47E822-1..........FLOW DIAGRAM-CLOSED COOLING WATER SYSTEM
MEL.................INSTRUMENT TABULATIONS
MEL.................TABULATION OF VALVE MARKER TAGS
1-47E852-1..........FLOW DIAGRAM-DIRTY RADWASTE
31000215C-8, -9.....FLOW DIAGRAM-VFD COOLANT

GENERAL ELECTRIC CO DRAWINGS:
021-043090.......SPECIAL TEMPERATURE COMPENSATOR



DETAIL 1A1 DETAIL 1B1

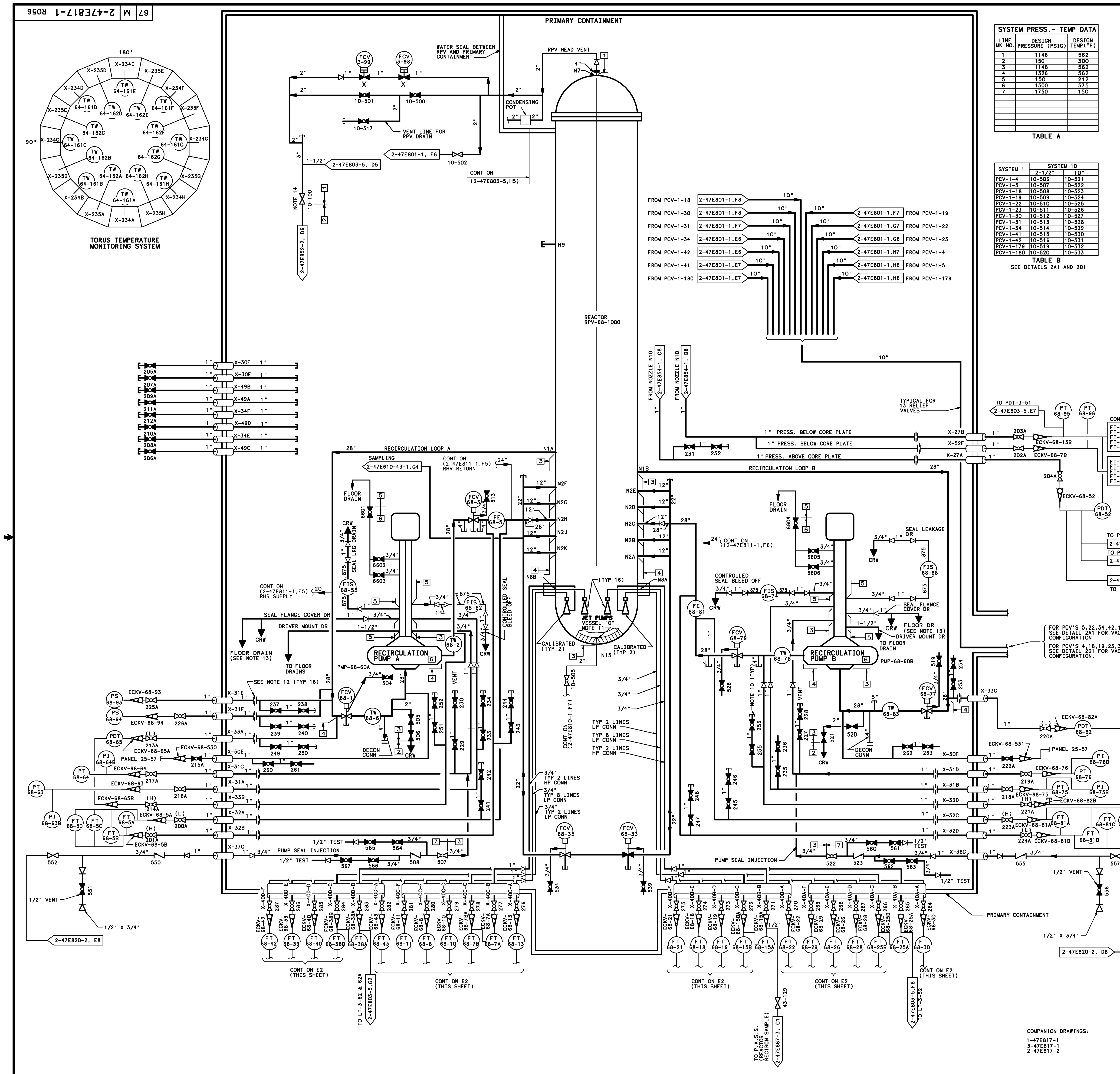
AMENDMENT 26

POWERHOUSE
UNIT 1

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

NUCLEAR BOILER
FLOW DIAGRAM

FIGURE 4.3-2a SH 1



NOTES:

1. ALL VALVES ARE THE SAME SIZE AS PIPING UNLESS OTHERWISE NOTED AND ARE SHOWN IN THEIR NORMAL OPERATING POSITION.
2. COMPONENT UNIDS SHALL BE PREFIXED WITH THE UNIT NUMBER "2-" UNLESS OTHERWISE NOTED. UNIDS ON DRAWINGS ARE FOR REFERENCE ONLY AND ARE ABBREVIATED TO MEET SPACE CONSTRAINTS. REFER TO MEL FOR COMPLETE UNIDS.
3. CRW DRAINS ARE CONNECTED TO THE REACTOR BUILDING EQUIPMENT DRAIN SUMP.
4. $\boxed{1}$ ETC, DENOTES DESIGN PRESSURE AND TEMPERATURE AS GIVEN IN TABLE (A), SYSTEM PRESS-TEMP DATA THIS DRAWING.
5. HYDROSTATIC TESTING SHALL BE IN ACCORDANCE WITH THE APPLICABLE CODES.
6. THE DESIGN PRESSURE AND TEMP OF ALL DRAIN AND VENT LINES THROUGH THE LAST ISOLATION VALVE SHALL BE THE SAME AS THE PROCESS LINE.
7. DRAWING 2-47E817-2 REPRESENTS THE REACTOR WATER RECIRCULATION M-G OIL SET SYSTEM ONLY.
8. DELETED
9. (H) - IS HIGH PRESSURE AND (L) - IS LOW PRESSURE SIDE.
10. VENT, DRAIN AND TEST CONNECTION 1-1/2" AND BELOW WERE PROVIDED WITH PIPE CAPS OR HOSE CONNECTION, BY PLANT PERSONNEL. THIS CONFIGURATION IS SUPPORTED BY ENGINEERING CALCULATION CD-Q0999-923399.
11. VESSEL "0" (VESSEL INVERT) AT 578'-3" BUILDING ELEVATION. SEE CALCULATION CD-Q0303-940391.
12. ALL SENSING LINES WITH EXCESS FLOW CHECK VALVES EXCEPT X-40A-A THRU X-40D-F HAVE AN ORIFICE COUPLING (REFER TO DETAIL P20 ON 0-47W600-20).
13. SEAL FLANGE COVER DRAIN IS VENTED THROUGH AN OPEN-ENDED "TEE" CONNECTED TO A DRIP-LOOP.
14. OPERATIONS MAY CLOSE VALVE (2-SHV-10-100) AS NECESSARY TO REDUCE IN-LEAKAGE.

SYSTEM PRESS.- TEMP DATA		
LINE MK NO.	DESIGN PRESSURE (PSIG)	DESIGN TEMP(°F)
1	1146	562
2	150	300
3	1148	562
4	1326	562
5	150	212
6	1500	575
7	1750	150

TABLE A

SYSTEM 1	SYSTEM 10	
	2-1/2"	10"
PCV-1-4	10-506	10-521
PCV-1-5	10-507	10-522
PCV-1-18	10-508	10-523
PCV-1-19	10-509	10-524
PCV-1-22	10-510	10-525
PCV-1-23	10-511	10-526
PCV-1-30	10-512	10-527
PCV-1-31	10-513	10-528
PCV-1-34	10-514	10-529
PCV-1-41	10-515	10-530
PCV-1-42	10-516	10-531
PCV-1-179	10-519	10-532
PCV-1-180	10-520	10-533

TABLE B

SEE DETAILS 2A1 AND 2B1

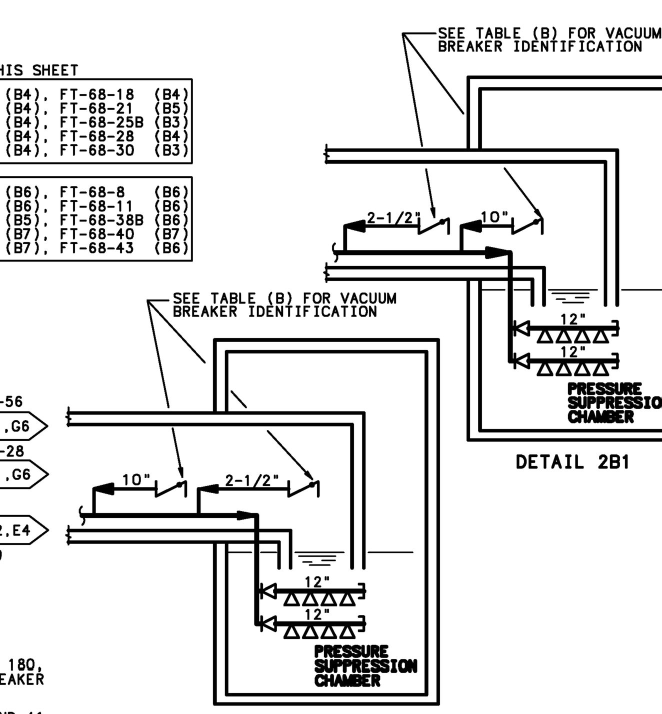
REFERENCE DRAWINGS:

MEL....................VALVE MARKER TAG TABULATIONS
MEL....................INSTRUMENT TABULATIONS FOR REACTOR WATER RECIRCN SYSTEM
47E600-SERIES........MECHANICAL INSTRUMENTS & CONTROLS
2-47E610-3-1.........CONTROL DIAGRAM-REACTOR FEEDWATER SYSTEM
2-47E610-43-1........CONTROL DIAGRAM-SAMPLING & WATER QUALITY SYSTEM
2-47E610-64-1,-2.....CONTROL DIAGRAM-PRIMARY CONTAINMENT SYSTEM
2-47E610-68-1........CONTROL DIAGRAM-REACTOR WATER RECIRCN SYSTEM
0-47E800-1...........FLOW DIAGRAM-GENERAL PLANT SYSTEMS
0-47E800-2...........MECHANICAL-SYMBOLS & FLOW DIAGRAM DRAWING INDEX
2-47E801-1...........FLOW DIAGRAM-MAIN STEAM
2-47E803-5...........FLOW DIAGRAM-REACTOR FEEDWATER SYSTEM
2-47E810-1...........FLOW DIAGRAM-REACTOR WATER CLEANUP SYSTEM
2-47E811-1...........FLOW DIAGRAM-RHR SYSTEM
2-47E812-1...........FLOW DIAGRAM-HPCI SYSTEM
2-47E813-1...........FLOW DIAGRAM-RCIC SYSTEM
2-47E814-1...........FLOW DIAGRAM-CORE SPRAY SYSTEM
2-47E820-2...........FLOW DIAGRAM-CONTROL ROD DRIVE SYSTEM
2-47E844-2...........FLOW DIAGRAM-RAW COOLING WATER SYSTEM
2-47E852-1...........FLOW DIAGRAM-FLOOR AND DIRTY RADWASTE DRAINAGE
2-47E852-2...........FLOW DIAGRAM-CLEAN RADWASTE SYSTEM
2-47E854-1...........FLOW DIAGRAM-STANDBY LIQUID CONTROL SYSTEM
2-47E865-3,-13.......FLOW DIAGRAM-RB HEATING & VENT. AIR FLOW
2-47E867-3...........FLOW DIAGRAM-SAMPLING & WATER QUALITY SYSTEM
2-47E2865-SERIES.....FLOW DIAGRAM-RB HEATING & VENT. AIR FLOW

GENERAL ELECTRIC CO DRAWINGS:
78-CD-3131 AND 3140....SCHEMATIC PIPING DIAGRAM-LH & RH COOLER CONN
531E261 AND 531E267....OUTLINE (PUMP DRIVE SYSTEM)
117C1461-1.............RECIRCULATION PUMP P & ID
719E597-3..............NUCLEAR BOILER P & ID
104R930-1..............CONTROL ROD DRIVE HYDRAULIC SYSTEM P & ID
719E532-SERIES.........PRIMARY CONTAINMENT PENETRATIONS

SYMBOLS:

JET PUMP



DETAIL 2B1

DETAIL 2A1

COMPANION DRAWINGS:
1-47E817-1
3-47E817-1
2-47E817-2

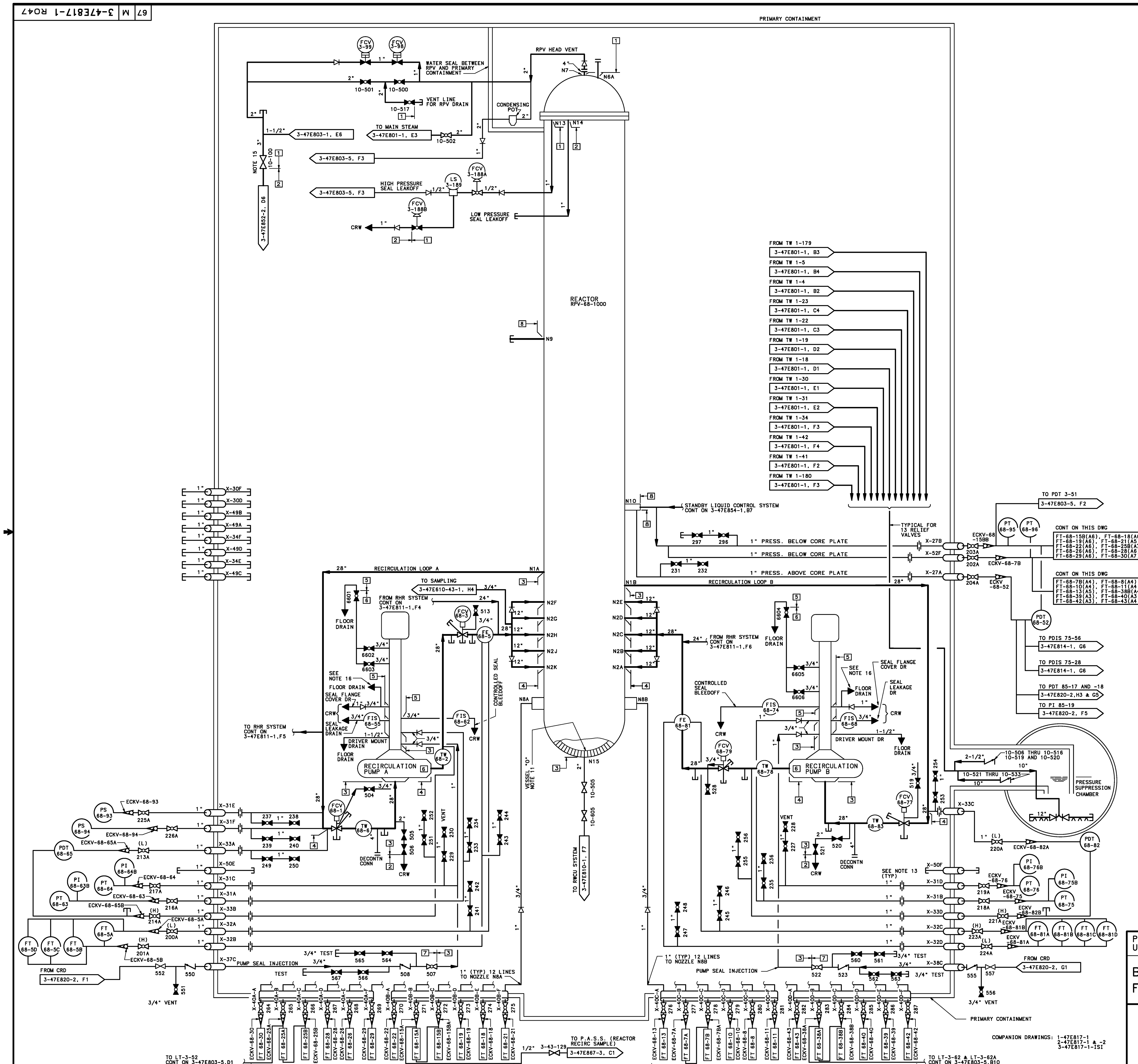
AMENDMENT 26

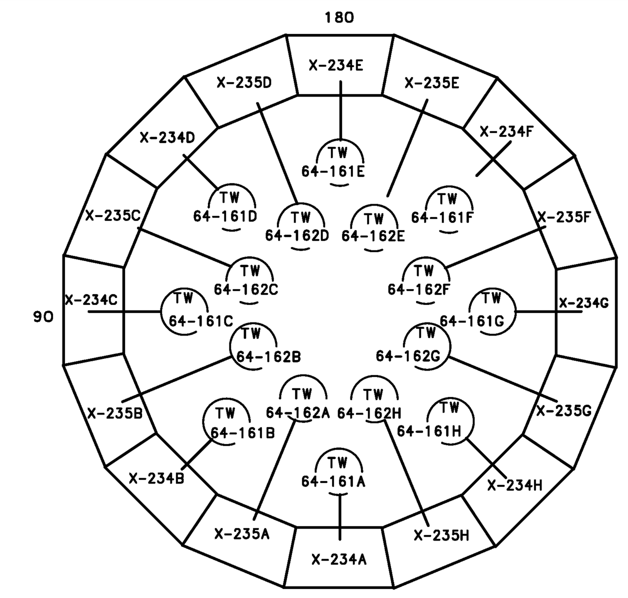
POWERHOUSE
UNIT 2

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

NUCLEAR BOILER
FLOW DIAGRAM

FIGURE 4.3-2a SH 2





TORUS TEMPERATURE MONITORING SYSTEM

SYSTEM PRESS.- TEMP DATA		
LINE MK NO.	DESIGN PRESSURE (PSIG)	DESIGN TEMP(°F)
1	1146	562
2	150	300
3	1148	562
4	1326	562
5	150	212
6	1500	575
7	1750	150
8	1250	575

TABLE A

NOTES:

1. ALL VALVES ARE THE SAME SIZE AS PIPING UNLESS OTHERWISE NOTED.
2. ALL VALVE BODY DRAINS, BONNET VENTS AND PACKING LEAKOFFS ARE 1/2" UNLESS OTHERWISE NOTED.
3. DELETED
4. ALL DRAINS SHOWN INDICATES CLOSED SYSTEM.
5. CRW DRAINS ARE CONNECTED TO THE REACTOR BUILDING EQUIPMENT DRAIN SUMP.
6. $\boxed{1}$ ETC, DENOTES DESIGN PRESSURE AND TEMPERATURE AS GIVEN IN TABLE (A) SYSTEM PRESS-TEMP DATA THIS DRAWING.
7. HYDROSTATIC TESTING SHALL BE IN ACCORDANCE WITH THE APPLICABLE CODES.
8. THE DESIGN PRESSURE AND TEMP OF ALL DRAIN AND VENT LINES THROUGH THE LAST ISOLATION VALVE SHALL BE THE SAME AS THE PROCESS LINE.
9. ALL VALVES ARE PREFIXED "3-68-" AND ALL INSTRUMENTS ARE PREFIXED "3-" UNLESS OTHERWISE NOTED.
10. (H) - IS HIGH PRESSURE AND (L) - IS LOW PRESSURE SIDE.
11. VESSEL "O" (VESSEL INVERT AT 578'-3" BUILDING ELEVATION SEE CALCULATION CD-Q0303-940391.
12. UNIDS ON DRAWING ARE FOR REFERENCE ONLY AND ARE ABBREVIATED TO MEET SPACE CONSTRAINTS, REFER TO MEL FOR COMPLETE UNIDS.
13. ALL SENSING LINES WITH EXCESS FLOW CHECK VALVES EXCEPT X-40A-a THROUGH X-40D-f HAVE AN ORIFICE COUPLING. (REFER TO DETAIL P20 ON 0-47W600-20).
14. 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.
15. OPERATIONS MAY CLOSE VALVE (3-SHV-10-100) AS NECESSARY TO REDUCE INLEAKAGE.
16. SEAL FLANGE COVER DRAIN IS VENTED THROUGH AN OPEN-ENDED "TEE" CONNECTED TO A DRIP-LOOP.

REFERENCE DRAWINGS:
0-47E800-1..........FLOW DIAGRAM-GENERAL PLANT SYSTEMS
0-47E800-2..........MECHANICAL SYMBOLS & FLOW DIAGRAM DRAWING INDEX
3-47E811-1..........FLOW DIAGRAM-RHR SYSTEM
3-47E814-1..........FLOW DIAGRAM-CORE SPRAY SYSTEM
3-47E610-75-1.......MECHANICAL CONTROL DIAGRAM-CORE SPRAY SYSTEM
3-47E610-3-1.......MECHANICAL CONTROL DIAGRAM-
REACTOR FEEDWATER SYSTEM
3-47E812-1..........FLOW DIAGRAM-HPCI SYSTEM
3-47E803-1..........FLOW DIAGRAM-REACTOR FEEDWATER SYSTEM
3-47E813-1..........FLOW DIAGRAM-RCIC SYSTEM
3-47E820-2..........FLOW DIAGRAM-CRD HYDRAULIC SYSTEM
3-47E810-1..........FLOW DIAGRAM-REACTOR WATER CLEANUP SYSTEM
3-47E610-68-1.......MECHANICAL CONTROL DIAGRAM-
REACTOR WATER RECIRN SYSTEM
3-47E610-43-1.......MECHANICAL CONTROL DIAGRAM-
SAMPLING & WATER QUALITY SYSTEM
3-47E610-85-1.......MECHANICAL CONTROL DIAGRAM-
CONTROL ROD DRIVE SYSTEM
3-47E801-1,-2.......FLOW DIAGRAM-MAIN STEAM
3-47E610-1-SERIES...MECHANICAL CONTROL DIAGRAM-MAIN STEAM
3-47E610-74-SERIES..MECHANICAL CONTROL DIAGRAM-RHR SYSTEM
3-47E610-73-1.......MECHANICAL CONTROL DIAGRAM-HPCI SYSTEM
3-47E610-71-1.......MECHANICAL CONTROL DIAGRAM-RCIC SYSTEM
3-47E610-69-1.......MECHANICAL CONTROL DIAGRAM-
REACTOR WATER CLEANUP SYSTEM
3-47E854-1..........FLOW DIAGRAM-STANDBY LIQUID CONTROL SYSTEM
3-47E610-63-1.......MECHANICAL CONTROL DIAGRAM
STANDBY LIQUID CONTROL SYSTEM
3-47E822-1..........FLOW DIAGRAM-CLOSED COOLING WATER SYSTEM
MEL.................INSTRUMENT & VALVE MARKER TAG TABULATIONS
3-47E852-2..........FLOW DIAGRAM - CRW DRAINAGE
3-47E803-5..........MECHANICAL RPV LEVEL SENSING LINES
INSTRUMENTS AND CONTROLS
MEL.................INSTRUMENT TABULATION FOR SYSTEM 3
3-47E867-3..........FLOW DIAGRAM-SAMPLING AND WATER QUALITY SYSTEM

AMENDMENT 27

POWERHOUSE
UNIT 3

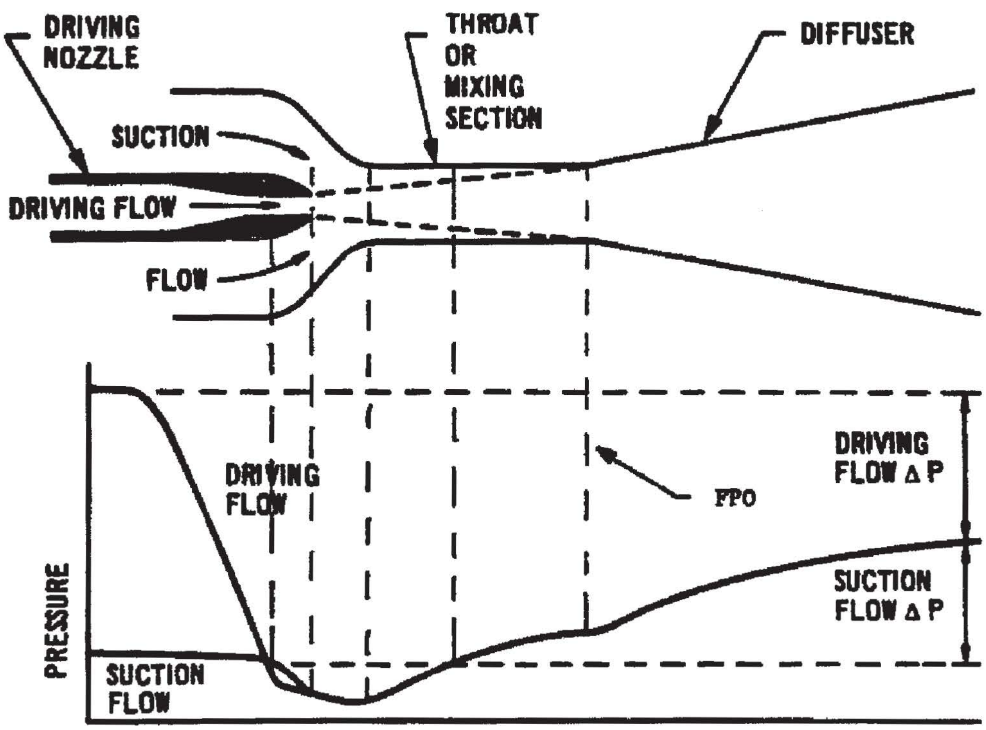
BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

NUCLEAR BOILER
FLOW DIAGRAM

FIGURE 4.3-2a SH 3

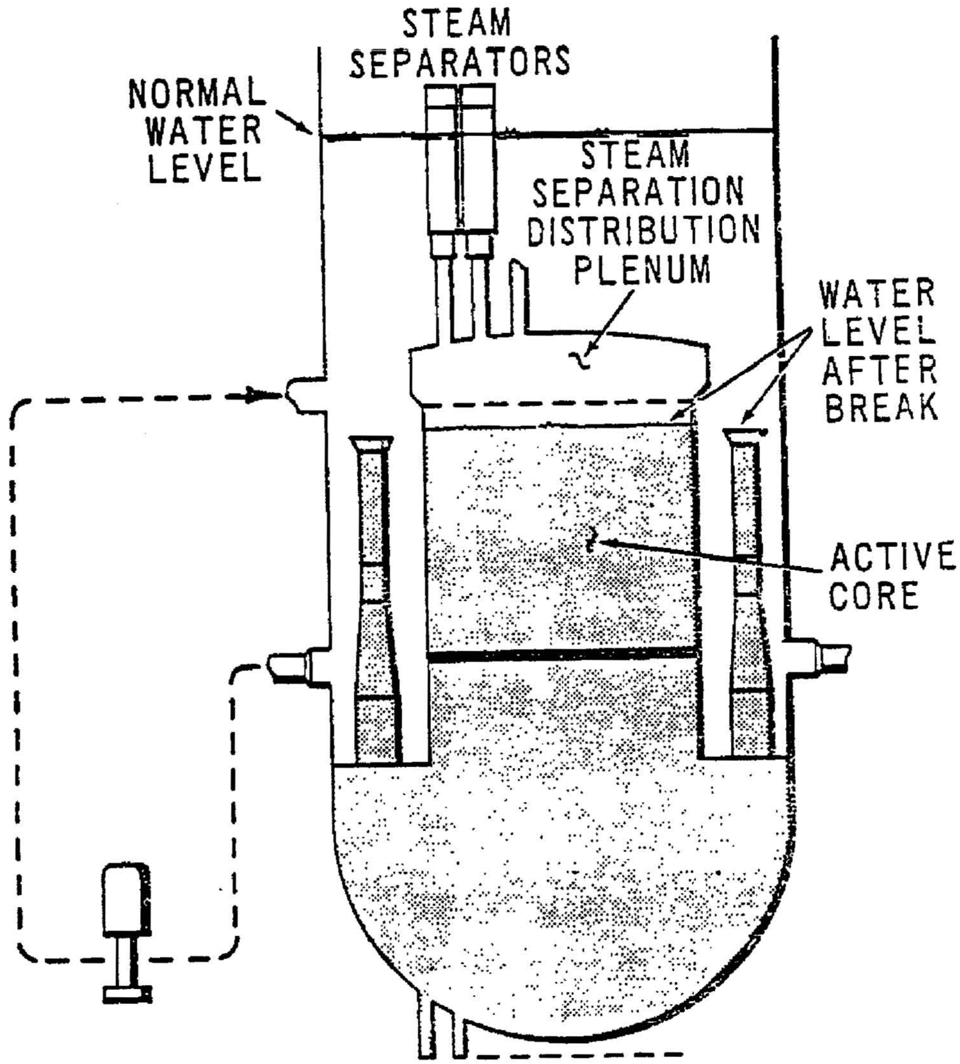
Figure 4.3-2b

Deleted by Amendment 11.



AMENDMENT 16

BROWNS FERRY NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT
Jet Pump—Operating Principle FIGURE 4.3-3



AMENDMENT 16

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Recirculation System - Core Flooding Capability
Figure 4.3-4

4.4 NUCLEAR SYSTEM PRESSURE RELIEF SYSTEM

4.4.1 Safety Objective

The safety objective of the Nuclear System Pressure Relief System is to prevent overpressurization of the nuclear system; this protects the nuclear system process barrier from failure which could result in the uncontrolled release of fission products. In addition, the automatic depressurization feature of the Nuclear System Pressure Relief System acts in conjunction with the Emergency Core Cooling Systems for reflooding the core following breaks in the nuclear system process barrier; this protects the reactor fuel barrier (UO_2 sealed in cladding) from failure due to overheating, which would result in the uncontrolled release of fission products from the reactor fuel barrier.

4.4.2 Power Generation Objective

The power generation objective of the Nuclear System Pressure Relief System is to relieve normal overpressure transients occurring during normal plant isolations and load rejections.

4.4.3 Safety Design Basis

1. The Nuclear System Pressure Relief System shall prevent overpressurization of the nuclear system in order to prevent failure of the nuclear system process barrier.

2. The Nuclear System Pressure Relief System shall provide automatic nuclear system depressurization, if needed, for breaks in the nuclear system so that the Low Pressure Coolant Injection (LPCI) and the Core Spray Systems can operate to protect the fuel barrier. This depressurization is permissive on: (1) concurrent high drywell pressure and low reactor water level, or (2) sustained reactor low water level, and (3) availability of one of the RHR pumps in the LPCI mode or two of the appropriate core spray pumps.

3. The main steam relief valve (MSRV) discharge piping shall be designed to accommodate forces resulting from relief action and shall be supported for reactions due to flow at maximum MSRV discharge capacity so that system integrity is maintained. The MSRV discharge piping shall be routed to the pressure suppression pool.

4. The Nuclear System Pressure Relief System shall be designed for testing prior to nuclear system operation and for periodic verification of the operability of the Nuclear System Pressure Relief System.

4.4.4 Power Generation Design Basis

1. The nuclear system main steam relief valves shall not discharge to the primary containment drywell.

2. The main steam relief valves shall properly reclose following a plant isolation or load rejection, so that normal operation can be resumed as soon as possible.

3. The capacity of the main steam relief valves shall be sufficient to prevent reactor pressure from exceeding the allowable overpressure of ASME Boiler and Pressure Vessel Code, Section III, during an isolation transient with indirect scram.

4.4.5 Description

The Nuclear System Pressure Relief System includes 13 main steam relief valves, all of which are located on the main steam lines within the drywell between the reactor vessel and the flow restrictors.

The main steam relief valves provide three main protection functions:

1. Overpressure relief operation. All 13 main steam relief valves can be opened manually from the main control room or are self-actuated to limit the pressure rise.

2. Overpressure safety operation. The valves are opened (self-actuated) to prevent exceeding the design allowable stress limits on the reactor vessel and associated piping.

3. Depressurization operation. Six of the 13 valves are available to be opened automatically as part of the Emergency Core Cooling System (ECCS).

The main steam lines, in which the main steam relief valves are installed, are designed, installed, and tested in accordance with USAS B31.1.0, 1967 edition, and the applicable GE design and procurement specifications, which were implemented in lieu of the outdated B31 Nuclear Code Cases-N2, N7, N9, and N10. The main steam relief valves are distributed among the four main steam lines so that an accident cannot completely disable a safety, relief, or automatic depressurization function. (See Figure 4.3-2a sheet 1 of Subsection 4.3 and Figures 11.1-1a, 11.1-1c, and 11.1-1e of Subsection 11.1 for schematic location, and Figures 4.5-1, 4.5-2, and 4.5-3 of Subsection 4.5 for layout details of the valves and piping.)

The design and installation of the main steam relief valves include the following:

a. Clearance of at least 6 in. is provided between valves and other equipment (excluding MSRV pilot solenoid valves),

b. Space is provided between all welds on the header for inspection greater than 2t + 2" (where t is minimum wall thickness),

c. Clearance is provided between header and bottom of flange for bolt removal when valve is installed,

d. A flange rating of 1500 lb. was provided for structural stability instead of a 900 lb.-rated flange required for pressure-temperature rating,

e. An inlet pipe Schedule 160 was used for structural stability instead of Schedule 80 required for pressure-temperature rating, and

f. The discharge piping provides for equalization of discharge thrust forces.

For analysis, the special loadings listed below are considered in addition to the usual design loads such as weight, pressure, temperature, and earthquake:

1. The jet force exerted on the main steam relief valves during the first millisecond when the valve is open and steady-state flow has not yet been established. (With steady-state flow, the dynamic flow reaction forces will be self-equilibrated by the discharge piping.)

2. The dynamic effects of the kinetic energy of the piston disc assembly when it impacts on the internals of the valve.

All code-allowable stresses are met with these special loads acting concurrently with other design loads. The highest stress is at the branch connection to the header. The results of this analysis are contained in Appendix C, Table C.4-2.

The main steam relief valves are designed, constructed, and marked with data in accordance with the ASME Boiler and Pressure Vessel Code, Section III, 1968 edition and addenda through summer 1970 for the two-stage valves. Setpoint tolerance (pressure at which valve "pops" wide open) is in accordance with ASME Boiler and Pressure Vessel Code, Section I, paragraph PG-72(c).
Pressure-containing parts of the valve body are fabricated of ASTM A216, Grade WCB. The main steam relief valve is designed for operation with saturated steam containing less than 1 percent moisture. The relieving pressures for overpressure

relief and safety operating modes are adjustable between 1025 and 1190 psig, with a maximum backpressure of 40 percent of the set pressure. The lowest MSRV setpoint has been raised to 1135 psig. This serves to alleviate the "simmering" problems that contribute to valve failures. Also, the bore size of the valves has been increased slightly (from 4.94 or 5.03 inches to 5.125 inches) to accommodate more relief capacity. The delay time (maximum elapsed time between overpressure signal and actual valve motion) and the response time (maximum valve stroke time) are less than 0.5 second total.

Each valve is self-actuating at the set relieving pressure, but may also be actuated by remotely-operated devices to permit remote-manual or automatic opening at lower pressures. The remote air actuators are controlled by DC powered solenoid valves. The power actuated device is capable of opening the valve at any steam pressure above 50 psig and is capable of holding the valve open until the steam pressure decreased to about 20 psig. The solenoid valves are normally closed, fail-closed valves, and a power or valve malfunction will prevent the main steam relief valve from operating for Automatic Depressurization System (ADS). Abnormal solenoid-valve operation would be detected during the operational tests of the main steam relief valve. A complete rupture of the solenoid valve would result in a low air pressure/accumulator alarm.

Each of the six main steam relief valves provided for automatic depressurization is equipped with an air accumulator and check valve arrangement. These accumulators are provided to assure that the valves can be held open following failure of the air supply to the accumulators, and they are sized to contain sufficient air for a minimum of five valve operations. To ensure an emergency supply of air is available to provide for the five valve operations under accident conditions, an accumulator leak test is performed once per operating cycle. The first and second actuations are assumed to occur with drywell pressure at 35 psig and subsequent actuations with the drywell at 0 psig. Redundant sources of pneumatic pressure are provided by the Drywell Control Air (DCA) and Containment Atmospheric Dilution (CAD) systems. Accumulators are not required for the main steam relief valves not used for automatic depressurization. The main steam relief valves which are a part of the ADS normally receive their motive air from the drywell control air system. The air pressure in each accumulator is continuously monitored by a pressure switch which annunciates in the control room on low air pressure. The pressure switch, in order to ensure operability, is calibrated and functionally tested once per operating cycle. The drywell control air system is also continuously monitored for low air pressure by means of a pressure switch located in the system downstream of the receivers and which annunciates in the control room. A manual transfer can also be made to the plant control air system as another backup for control air. The main steam relief valves are designed to operate under maximum prevailing operating conditions and postulated accident conditions in the drywell. In addition, the ADS

accumulators and piping up to and including the isolation check valves are seismically qualified and capable of performing their functions during and following an accident.

The automatic depressurization feature of the Nuclear System Pressure Relief System serves as a backup to the High Pressure Coolant Injection (HPCI) System under loss-of-coolant accident conditions. If high drywell pressure and low water level persist and one of the low pressure coolant injection (LPCI) pumps or two of the appropriate core spray pumps are available, the nuclear system is depressurized sufficiently to permit the LPCI and Core Spray Systems to operate to protect the fuel barrier. Depressurization is accomplished through automatic opening of some of the main steam relief valves to vent steam to the pressure suppression pool. For small line breaks, if the HPCI system fails, the nuclear system is depressurized in sufficient time to allow the Core Spray or LPCI Systems to provide core cooling to prevent excessive fuel clad temperatures. When HPCI is considered to be the single failure, six ADS valves are required to meet the requirements for ADS. As shown in Table 6.5-3, dependent upon the recognized single failure, between four and six valves remain available and the results of LOCA analyses confirm that the requirements for ADS continue to be met. For large breaks, the vessel depressurizes rapidly through the break without assistance from ADS. Discharge pressure indication of one LPCI pump or two core spray pumps combined with one of the following initiation paths will cause the main steam relief valves to open: (l) reactor vessel low water level and primary containment (drywell) high pressure in conjunction with a 120 seconds timer timed out; or (2) sustained reactor low water level for 360 seconds. Further descriptions of the operation of the automatic depressurization feature are found in Section 6.0, "Emergency Core Cooling Systems," and Subsection 7.4, "Emergency Core Cooling System Control and Instrumentation." The Automatic Depressurization System is designed as seismic Class I equipment in accordance with Appendix C.

A manual depressurization of the nuclear system can be effected in the event the main condenser is not available as a heat sink after reactor shutdown. The steam generated by core decay heat is discharged to the pressure suppression pool. The main steam relief valves are operated by remote manual controls from the Main Control Room to control nuclear system pressure.

The number, set pressures, and capacities of the main steam relief valves are shown in Table 4.4-1a (it should be noted that the ± three percent tolerance is for analytical purposes only). Actual MSRV opening setpoints following testing must still be set at nominal values ± one percent.

The original three-stage Target Rock valves (Model 67F) have been changed to two-stage valves (Target Rock MSRV model No. 7567F) to minimize spurious openings and to respond to NUREG 0737, Item II.K.3.16.

Two-Stage Valve Operation

The Target Rock pilot-operated main steam relief valve (Model 7567F) consists of two principal assemblies: a pilot stage assembly and the main stage assembly (refer to Figure 4.4-1). These two assemblies are directly coupled to provide a unitized, self-actuated safety/relief valve. The pilot stage assembly is the pressure sensing and control element and the main stage assembly is a hydraulically (system fluid) actuated follower valve which provides the pressure relief function.
Self-actuation of the pilot assembly at set pressure vents the main piston chamber, permitting the system pressure to fully open the main assembly. The pilot assembly consists of two relatively small, low-flow, pressure-sensing elements. The spring loaded pilot disc senses the set pressure, and the pressure-loaded stabilizer disc senses the reseat pressure. Spring force (preload force) is applied to the pilot disc by means of the pilot rod. Thus, the adjustment of the spring preload force will determine the set pressure of the valve.

The main assembly of the Target Rock main steam relief valve is a reverse-seated, hydraulically-actuated angle globe valve. Actuation of the main assembly permits discharge of fluid from the protected system at the valve's rated flow capacity and provides the system pressure-relief function of the valve. The major components of the main stage are the valve body, disc/piston assembly, and preload spring.

A typical sequence of operation for overpressure relief self-actuation can be described as follows (refer to Figures 4.4-1 and 4.4-2).

1. In its normally closed position, the main stage disc is tightly seated by the combined forces exerted by the system internal pressure acting on the area of the disc and the preload spring. Note that in the closed, no-flow position, the static pressures will be equal in the valve inlet nozzle and in the chamber over the main stage piston. This pressure equalization is made possible by leakage past the piston, via the ring gap and drain and vent grooves.

2. When system pressure increases to the valve set pressure, pilot stage operation will vent the chamber over the main stage piston to downstream of the valve via internal porting. This venting action creates a differential pressure across the main stage piston in a direction tending to open the valve. The main stage piston is sized such that the resultant opening force is greater than the combined spring preload and hydraulic seating force.

3. Once the main stage disc starts to open, the hydraulic seating force is reduced, causing a significant increase in opening force and the characteristic full opening or "popping" action.

4. When system pressure has been reduced sufficiently, the pilot disc reseats and precludes depressurization of the main piston chamber. Leakage of system fluid, past the main stage piston and stabilizer seat, repressurizes the chamber over the piston, canceling the hydraulic opening force and permitting the preload spring and flow forces to close the main stage. Once closed, the additional hydraulic seating force, due to system pressure acting on the main stage disc, seats the main stage tightly and prevents leakage.

A remotely-controlled air operator is fitted to the pilot stage assembly to provide selective operation of the valve at system pressure other than set pressure. This is a diaphragm-type, pneumatic actuator which must be actuated to open the valve. It is actuated by means of a solenoid control valve which admits drywell control air to the air-operator piston chamber and strokes the air operator stem, in turn stroking the pilot disc via the pilot rod. The main stage then opens as described in previous paragraphs. Deenergizing the solenoid vents the air operator and permits the pilot disc to reseat. The main stage then reseats as previously described.

Main Steam Relief Valve Position Indication

The main steam relief valve position is monitored by two systems. A single-train acoustic monitoring system has been installed on all the main steam relief valves to provide unambiguous Main Control Room indication (and alarm) of valve position. The system responds to NRC requirements of NUREG 0578, item 2.1.3.a. The system is qualified as seismic Class I and is powered by a Class 1E power supply. There also exists a temperature sensor in the discharge piping of each valve which can be used to determine individual valve positions. Temperature indications are also provided in the control room. The acoustic monitor satisfies the valve position alarm and annunciation requirements. Refer to Paragraph 7.4.3.3.4 for additional details. (High-temperature alarm and annunciation is removed for Units 1, 2, and 3)

Non Safety Related Alternate Automatic Means of Opening the MSRVs Upon Overpressurization:

During inservice pressure transient events in the relief mode, safety grade pressure sensors (found in Section 7.4.3, "Automatic Depressurization System") actuate the MSRVs. This method of automatically opening the MSRV permits application of the full main steam line pressure to break the corrosion bonds that may have developed between the pilot/disc interface. When the relief mode is actuated, the setpoint spring preload is removed from the pilot disc, and a rapidly applied full differential pressure is seen across the pilot disc. This alternate means of actuation is capable of opening the MSRVs. This non-safety related automatic means of opening the MSRV is applicable for Units 1, 2, and 3.

Main Steam Relief Valve (MSRV) Discharge

The main steam relief valves are installed so that each valve discharge is piped through its own uniform-diameter discharge line to a point below the minimum water level in the primary containment pressure suppression pool to permit the steam to condense in the pool. Thermal mixing in the pool during main steam relief valve blowdown is enhanced by T-quencher discharge devices at the pressure suppression pool end of the main steam relief valve discharge lines. Water in the line above pressure suppression pool water level would cause excessive pressure at the valve discharge when it is again opened. For this reason, one small check valve and one large check valve venting to the drywell are provided on each main steam relief valve discharge line to prevent drawing water up into the line, due to steam condensation, following termination of main steam relief valve operation. The main steam relief valves are located on the main steamline piping, rather than on the reactor vessel top head, primarily to simplify the discharge piping to the pool and to avoid the necessity for removing sections of this piping when the reactor head is removed for refueling. In addition, the main steam relief valves are more accessible during a quick shutdown to correct possible valve malfunctions when located on the steam lines.

The discharge piping has been modified as part of the torus integrity program. This modification has been described in a letter from L. M. Mills to Harold R. Denton dated May 22, 1981. A submittal by GE (NEDO-21888, "Mark I Containment Program Load Definition Report," December, 1980), on behalf of TVA, describes the reassessment of the torus design to include pressure suppression pool hydrodynamic loads due to MSRV discharge and pressure suppression pool response.

The reassessment of Mark I containments was precipitated from the large-scale testing of the Mark III containment system. Pressure suppression pool hydrodynamic loads resulting from the effect of drywell air and steam being rapidly forced into the pressure suppression pool during a postulated LOCA and/or MSRV discharge were identified, which had not been considered in the original design. The Mark I Owners Group, of which TVA is a member, and GE responded by submitting the Mark I Containment Program Load Definition Report (described above) and the Mark I Containment Program Structural Acceptance Criteria Plant Unique Analysis Application Guide (NEDO-24538-1). These reports describe the generic pressure suppression-pool hydrodynamic-load definition and assessment procedures for use in plant-unique pressure suppression-chamber design analyses. TVA has applied the load definitions and approved structural acceptance criteria to the entire torus, torus internals, MSRV piping, and attached piping of Browns Ferry Nuclear Plant.

A plant-unique main steam relief valve discharge test was performed as part of the Browns Ferry Nuclear Plant Unit 2 unique analysis, as requested by the NRC in NUREG-0661. This test did confirm the methods used to calculate containment loads from the various MSRV discharge cases. For the results of this test to be completely acceptable, all modifications which significantly influence torus motion had to be in their final configuration.

The magnitude of the MSRV discharge related loads is a function of the type of discharge device used. The device found to substantially reduce the hydrodynamic discharge loads, compared to other devices, is the T-quencher developed specifically for the Mark I torus (see Figures 4.4-6, 4.4-7, and 4.4-8). The devices have been added to the MSRV discharge lines at Browns Ferry Nuclear Plant. Discharge piping and relief valves were analyzed for deadweight, thermal, seismic and relief valve blowdown loads. The support locations, orientation, and design loads satisfy ASME Boiler and Pressure Vessel Code, Section III, Class 2, equations and stress allowables. One main steam safety valve (1-501 from line B) and one safety valve (1-537 from line C) were removed and the connections blanked off with blind flanges and a relief valve was added to Main Steam Line A and Main Steam Line D. Additionally, the valve throat diameters were increased from 5" nominal size to 5.125" nominal size. These modifications increased the installed relief capacity per valve to 870,000 lbm/hr at 1090 psig. The addition of these MSRVs do not adversely affect the stresses imposed on the headers to which the valves are attached. The torus shell is also adequate for the larger discharge loads.

4.4.6 Safety Evaluation

The ASME Boiler and Pressure Vessel Code requires that each vessel designed to meet Section III be protected from pressure in excess of the vessel design pressure. A peak-allowable pressure of 110 percent of the vessel design pressure is allowed by the code.

The main steam relief valves are set to open by self-actuation (overpressure safety mode) in the range from 1135 to 1155 psig. This satisfies the ASME code specifications for safety valves, since the lowest-set valve opens below the 1250 psig nuclear system design pressure, and the highest-set valve opens below 1313 psig (105 percent of nuclear system design pressure). The setpoints are also set high enough to avoid MSRV simmering problems.

Main steam relief valve capacity to support operation at the original licensed thermal power level of 3293 MWt was determined by analyzing the pressure rise accompanying the main steam flow stoppage resulting from a 3-second main steam isolation valve closure initiated from turbine-generator design operating conditions. The analysis hypothetically assumed the reactor is shut down by an indirect scram. The analysis indicated that the main steam

relief valve capacities provide sufficient flow to maintain an adequate margin below the peak ASME code-allowable pressure in the nuclear system (1375 psig). Figure 4.4-3 is representative of the nuclear system response which might be expected during such a transient. The required main steam relief valve capacity is currently determined for each core reload by analyzing the pressure rise accompanying the main steam flow stoppage resulting from a three second main steam isolation valve closure initiated at an initial dome pressure of 1055 psig (corresponding to the Improved Technical Specifications LCO value of 1050 psig plus 5 psi margin). The analysis hypothetically assumed the reactor is shutdown by a high neutron flux scram signal (i.e., failure of the MSIV position direct scram signal). For the analysis, the self-actuated setpoints of the 12 main steam relief valves were assumed to be as shown in Table 4.4-1a (one MSRV with the lowest opening setpoint is assumed inoperable). The analysis indicated that the main steam relief valve capacities shown in Table 4.4-1a provide sufficient flow to maintain an adequate margin below the peak ASME code-allowable pressure in the nuclear system (1375 psig). Additional discussion and results of this overpressurization analysis are documented in Chapter 14.

The results of the specific analysis for each unit can be found in the current reload licensing analysis for that unit (Appendix N). The sequence of events assumed in this analysis was investigated only to meet code requirements for pressure-relief-system evaluation

Evaluations of the automatic depressurization capability of the Nuclear System Pressure Relief System are presented in Section 6.0, "Emergency Core Cooling Systems" and Subsection 7.4, "Emergency Core Cooling System Controls and Instrumentation."

The piping attached to the main steam relief valve discharges was initially designed, installed, and tested in accordance with USAS B31.1.0, 1967 edition and the applicable GE design and procurement specifications, which were implemented in lieu of the outdated B31 Nuclear Code Cases-N2, N7, N9, and N10. New analyses of the main steam system and MSRV discharge piping have been performed in accordance with ANSI B31.1, 1973 edition, with Addenda up to Summer 1975. This analysis included deadweight, thermal, seismic, and main steam relief valve blowdown loadings. Snubbers have been added to reduce stresses in the main steam and main steam relief valve piping.

4.4.7 Inspection and Testing

The main steam relief valves were tested in accordance with the manufacturer's quality control procedures to detect defects and prove operability prior to installation. The following final tests were witnessed by a representative of the purchaser:

a. Test at USAS-specified hydrotest pressure using nitrogen, and

b. Nitrogen leakage test at design pressure with a maximum permitted leakage of 2cc per inch of seat diameter per hour.

The main steam relief valves were installed as received from the factory. The setpoints were adjusted, verified, and indicated on the valves by the vendor prior to shipment. Proper manual and automatic actuation of the main steam relief valves was verified during the preoperational test program.

It is recognized that it is not feasible to test the main steam relief valve setpoints while the valves are in place or during normal plant operation. The valves are mounted on 6-inch-diameter, 1500-pound, primary service rating flanges so that they may be removed for maintenance or bench checks and reinstalled during normal plant shutdowns. The external surface and seating surface of all main steam relief valves are 100 percent visually inspected when the valves are removed for maintenance or bench checks.

Operational tests of the main steam relief valves are performed once per operating cycle by means of an automatic actuation of the ADS valve logic circuitry from a simulated or actual initiation signal and by means of a manual actuation of all the main steam relief valves until thermocouples or acoustic monitors downstream of the valves indicate steam is flowing from the valve. This can also be demonstrated by the response of the turbine control valves or bypass valves, by a change in measured steam flow, or by any other method suitable to verify steam flow.

Main steam relief valves are removed and bench-tested following each operating cycle. The testing procedures include criteria for set pressure and seat leakage to determine valve acceptability. Monitoring and recording of valve stroke time, disc lift, and blowdown reseat pressure are included in the test to determine proper valve operation. Bench-testing is also required following any activity that will affect valve operability or set pressure prior to installing the valve.

During unit operation, discharge tailpipe temperatures and acoustic monitors are monitored and evaluated to determine if the valves are leaking excessively.

In response to NUREG-0578, Item 2.1.2, "Performance Testing for Relief and Safety Valves," TVA elected to participate in the BWR Owners Group Test Program of the safety/relief valves. The test program addressed those conditions that could result in single-phase liquid or two-phase flow through the safety/relief valves at low-pressure conditions.

The results of the tests are summarized in the BWR Owners Group S/RV Test Program Final Report, entitled "Analysis of Generic BWR Safety/Relief Valve Operability Test Results," NEDO-24988, submitted to D. G. Eisenhut by T. J. Dente, September 25, 1981.

The tested valves satisfy the acceptance criteria for operability; and therefore, the operational adequacy of the Browns Ferry MSRVs has been demonstrated.

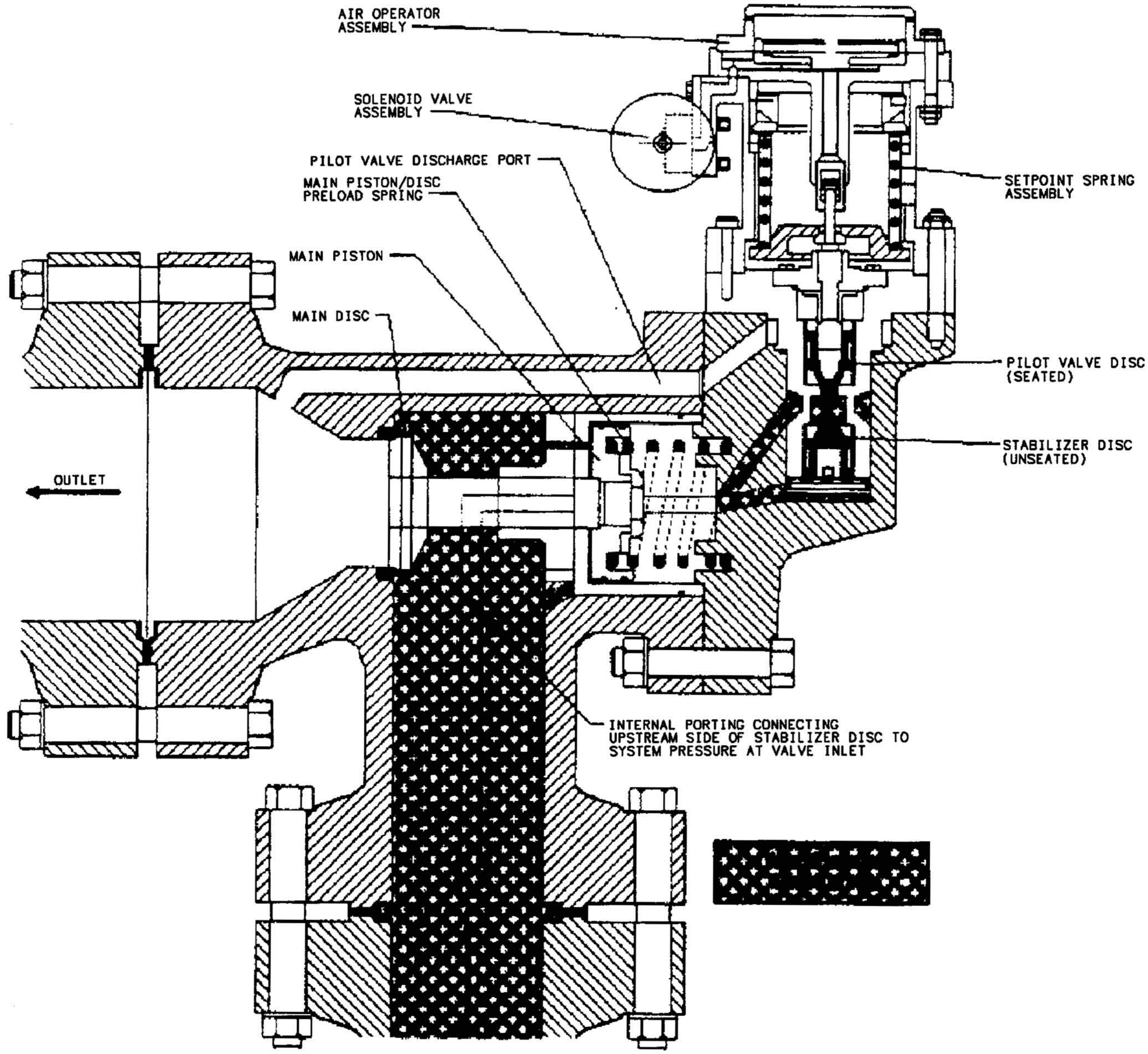
TABLE 4.4-1
(Deleted by Amendment 22)

TABLE 4.4-1A

NUCLEAR SYSTEM MAIN STEAM RELIEF VALVES

Units 1, 2, and 3

	Number of Valves	Set Pressure (psig)	Capacity at Set Pressure (each),(lb/hr)
Main Steam Relief	4	1135 (+3%)	905,000
Valves	4	1145 (+3%)	913,000
	5	1155 (+3%)	921,000

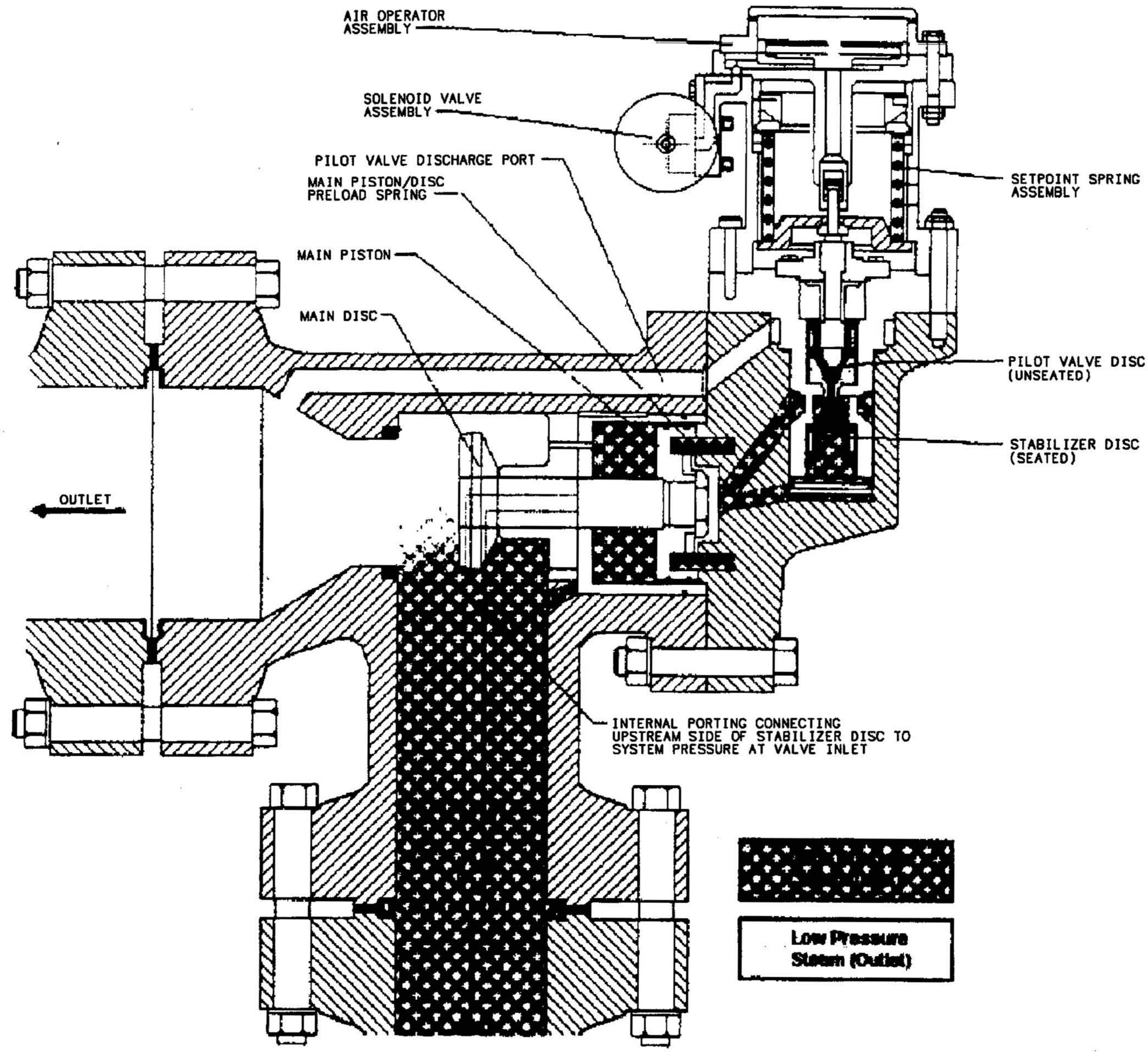


AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

2-STAGE SAFETY/RELIEF
VALVES
SCHEMATIC (CLOSED POSITION)

FIGURE 4.4-1

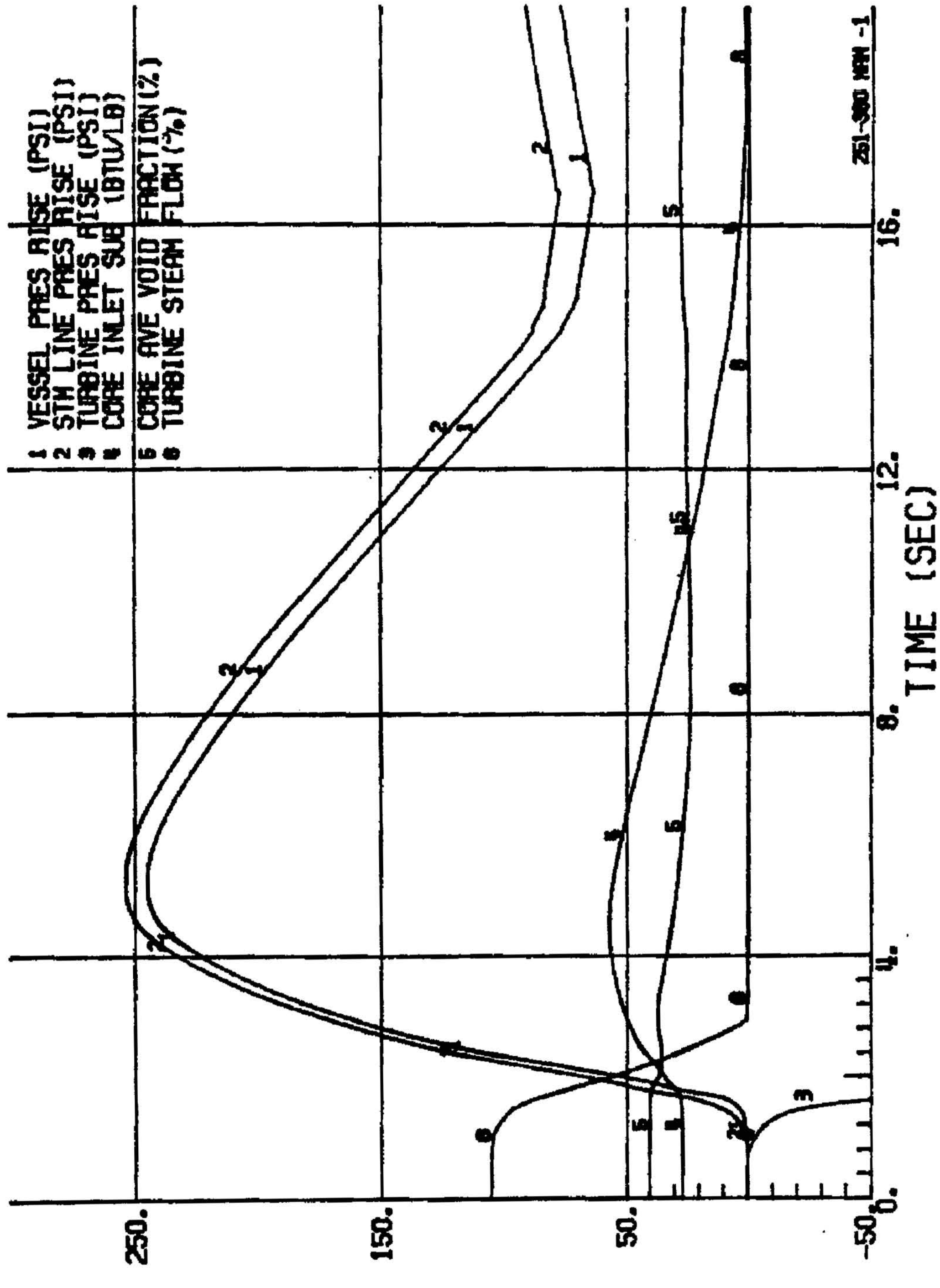


AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

2-STAGE SAFETY/RELIEF
VALVES
SCHEMATIC (OPEN POSITION)

FIGURE 4.4-2



NOTE: This figure is representative of the nuclear system response. See current reload amendment for up-to-date system response.

AMENDMENT 16

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

Safety Valve Sizing Analysis
FIGURE 4.4-3

Figure 4.4-4

Deleted by Amendment 13.

Figure 4.4-5

Deleted by Amendment 13.

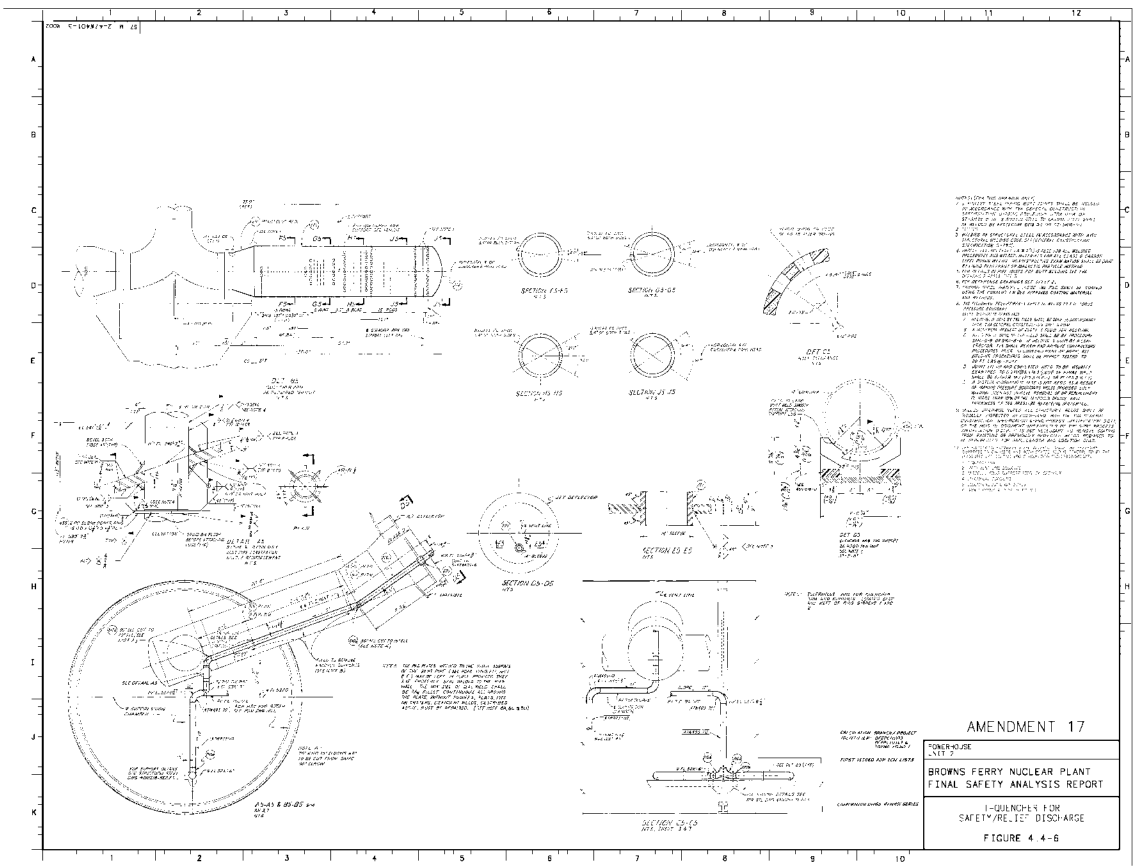
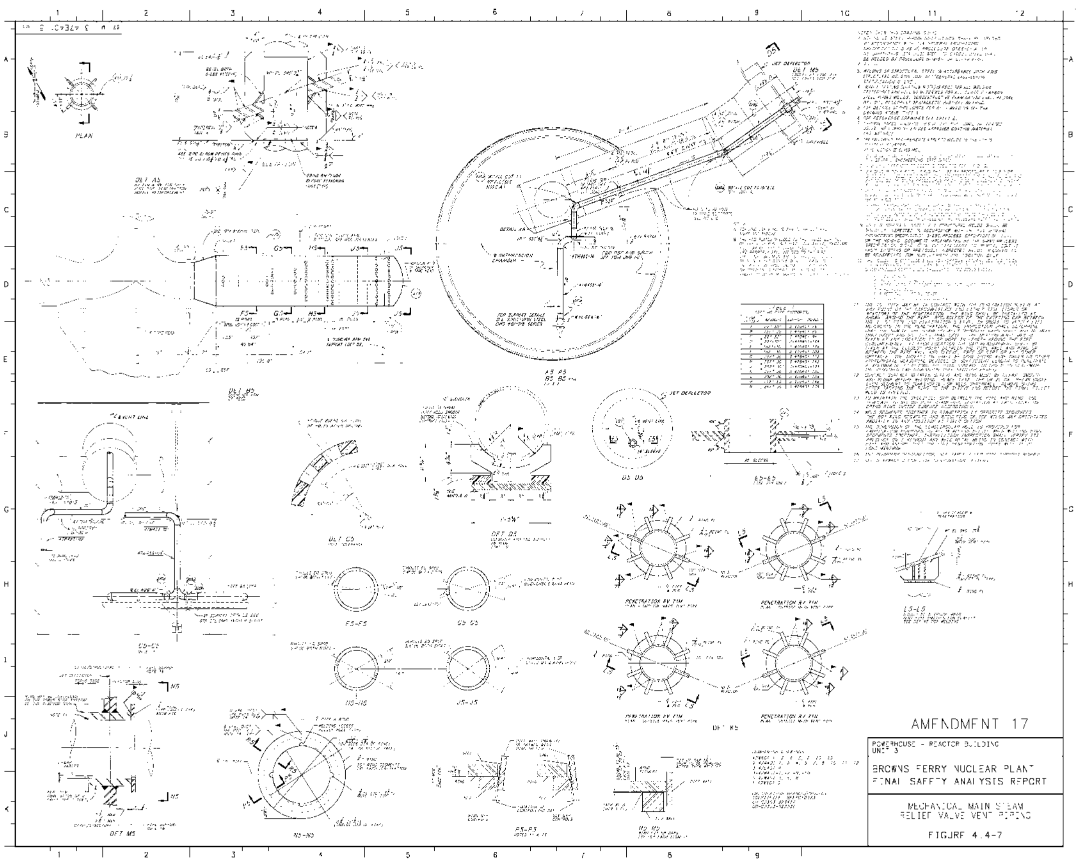

SECTION F5-F5
SECTION G5-G5
SECTION H5-H5
SECTION J5-J5
SECTION D5-D5
SECTION E5-E5
SECTION C5-C5
DET C5
DET G5
AMENDMENT 17
POWERHOUSE
UNIT 2
BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT
T-QUENCHER FOR
SAFETY/RELIEF DISCHARGE

FIGURE 4.4-6



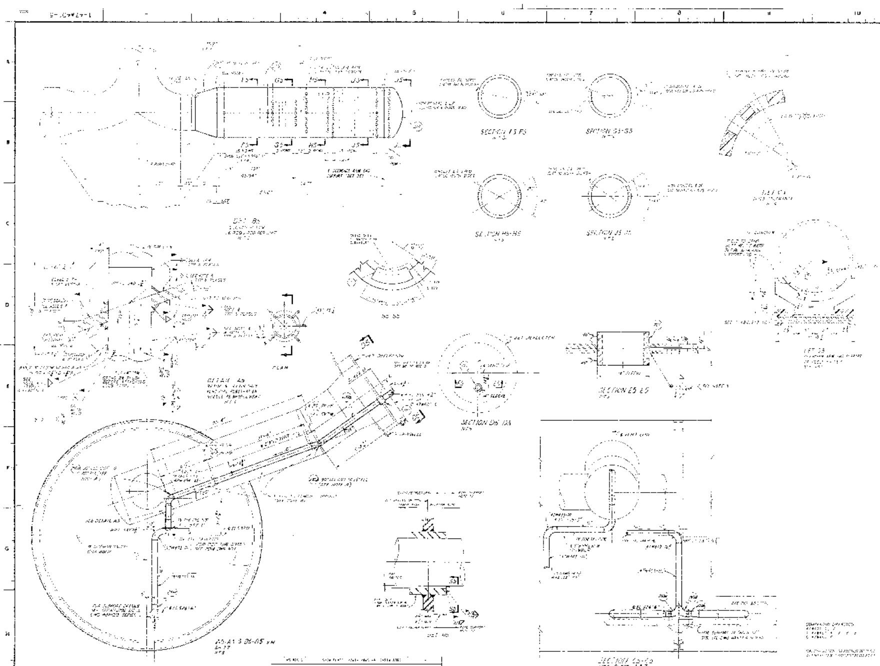
AMENDMENT 17

POWERHOUSE - REACTOR BUILDING
UNIT 3

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

MECHANICAL MAIN STEAM
RELIEF VALVE VENT PIPING

FIGURE 4.4-7



AMENDMENT 25

POWERHOUSE
REACTOR BLDG UNIT 1

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

MECHANICAL
MAIN STEAM RELIEF
VALVE VENT PIPING

FIGURE 4.4-8

4.5 MAIN STEAM LINE FLOW RESTRICTOR

4.5.1 Safety Objective

To protect the fuel barrier, the main steam line flow restrictors limit the loss of water from the reactor vessel before main steam isolation valve closure in case of a main steam line rupture outside the primary containment.

4.5.2 Safety Design Basis

1. The main steam line flow restrictor shall be designed to limit the loss of coolant from the reactor vessel following a steam line rupture outside the primary containment to the extent that the reactor vessel water level does not fall below the top of the core within the time required to close the main steam isolation valves.

2. The main steam line flow restrictor shall be designed to withstand the maximum pressure difference expected across the restrictor following complete severance of a main steam line.

4.5.3 Description

A main steam line flow restrictor is provided for each of the four main steam lines. The restrictor is a complete assembly welded into a vertical section of the main steam line between reactor vessel and the first main steam isolation valve, and downstream of the main steam relief valves. The restrictor limits the coolant flow rate from the reactor vessel in the event of a main steam line break outside the primary containment to the maximum (choke) flow. The restrictor assembly consists of a venturi-type nozzle insert welded into a carbon steel pipe. The venturi-type nozzle insert is constructed utilizing all austenitic stainless steel and is held in place with a full circumferential fillet weld. The restrictor assembly is self draining (low point pockets are internally drained to steam line).

The flow restrictor is designed and fabricated in accordance with USAS B31.1.0, 1967 edition and the applicable GE design and procurement specifications, which were implemented in lieu of the outdated B31 Nuclear Code Cases-N2, N7, N9, and N10. Preinstallation inspection and testing are in accordance with the ASME Boiler and Pressure Vessel Code, Sections I, III, and IX, 1965 edition. The container pipe is also designed and fabricated in accordance with USAS B31.1.0, 1967 edition, the applicable GE design and procurement specifications, and with the ASME Boiler and Pressure Vessel Code, Sections I, III, and IX, 1965 edition. The flow restrictor has no moving parts, and the mechanical structure of the restrictor is capable of withstanding the velocities and forces under main steam line break conditions where maximum differential pressure is approximately 1375 psi.

The ratio of the venturi throat diameter to a steam line diameter is approximately 0.6. This results in less than a 9 psi pressure difference at rated flow. This design limits the steam flow in a severed line to about 200 percent of its rated flow, yet it results in negligible increase in steam moisture content during normal operation. The restrictor is also used in the measurement of steam flow to provide indication in the control room, to provide input to the feedwater level control system, and to initiate closure of the main steam isolation valves when steam flow exceeds preselected operational limits.

4.5.4 Safety Evaluation

In the event of a main steam line break outside the primary containment, steam flow rate is restricted in the venturi throat by a two-phase mechanism similar to the critical flow phenomenon in gas dynamics. This limits the steam quantity flow rate, thereby reducing the reactor vessel coolant blowdown and the fuel clad temperature increase subsequent to the blowdown. The probability of fuel failure and its consequences are therefore decreased.

Analysis of the steam line rupture accident (see Section 14.0, "Plant Safety Analysis") shows that the core remains covered and that the amount of radioactive materials released to the environs through the main steam line break does not exceed the values of 10 CFR 50.67.

Pressure surges caused by a two-phase mixture impinging on the flow restrictor result in stresses which do not exceed code-allowable limits. There is adequate margin in the code for withstanding the pressure load due to impact pressure from the possible oncoming two-phase mixture predicted during main steam line break accident conditions.

Tests were conducted on a scale model to determine final design and performance characteristics of the flow restrictor, including maximum flow rate of the restrictor corresponding to the accident conditions, irreversible losses under normal plant operating conditions, and discharge moisture level. The tests showed that the flow restrictor operation at critical throat velocities is stable and predictable. Unrecovered differential pressure across scale model restrictor is consistently about 10 percent of the total nozzle pressure differentials, and the restrictor performance is in agreement with existing ASME correlation. Full size restrictors have slightly different hydraulic shape and a differential pressure loss of approximately 15 percent.

4.5.5 Inspection and Testing

Because the flow restrictor forms a permanent part of the main steam line piping and has no moving components, no testing program is planned. Only very slow erosion will occur with time, and such a slight enlargement will not have safety significance.

4.5 MAIN STEAM LINE FLOW RESTRICTOR

4.5.1 Safety Objective

To protect the fuel barrier, the main steam line flow restrictors limit the loss of water from the reactor vessel before main steam isolation valve closure in case of a main steam line rupture outside the primary containment.

4.5.2 Safety Design Basis

1. The main steam line flow restrictor shall be designed to limit the loss of coolant from the reactor vessel following a steam line rupture outside the primary containment to the extent that the reactor vessel water level does not fall below the top of the core within the time required to close the main steam isolation valves.

2. The main steam line flow restrictor shall be designed to withstand the maximum pressure difference expected across the restrictor following complete severance of a main steam line.

4.5.3 Description

A main steam line flow restrictor is provided for each of the four main steam lines. The restrictor is a complete assembly welded into a vertical section of the main steam line between reactor vessel and the first main steam isolation valve, and downstream of the main steam relief valves. The restrictor limits the coolant flow rate from the reactor vessel in the event of a main steam line break outside the primary containment to the maximum (choke) flow. The restrictor assembly consists of a venturi-type nozzle insert welded into a carbon steel pipe. The venturi-type nozzle insert is constructed utilizing all austenitic stainless steel and is held in place with a full circumferential fillet weld. The restrictor assembly is self draining (low point pockets are internally drained to steam line).

The flow restrictor is designed and fabricated in accordance with USAS B31.1.0, 1967 edition and the applicable GE design and procurement specifications, which were implemented in lieu of the outdated B31 Nuclear Code Cases-N2, N7, N9, and N10. Preinstallation inspection and testing are in accordance with the ASME Boiler and Pressure Vessel Code, Sections I, III, and IX, 1965 edition. The container pipe is also designed and fabricated in accordance with USAS B31.1.0, 1967 edition, the applicable GE design and procurement specifications, and with the ASME Boiler and Pressure Vessel Code, Sections I, III, and IX, 1965 edition. The flow restrictor has no moving parts, and the mechanical structure of the restrictor is capable of withstanding the velocities and forces under main steam line break conditions where maximum differential pressure is approximately 1375 psi.

The ratio of the venturi throat diameter to a steam line diameter is approximately 0.6. This results in less than approximately 12 psi pressure difference at rated flow. This design limits the steam flow in a severed line to about 200 percent of its rated flow, yet it results in negligible increase in steam moisture content during normal operation. The restrictor is also used in the measurement of steam flow to provide indication in the control room, to provide input to the feedwater level control system, and to initiate closure of the main steam isolation valves when steam flow exceeds preselected operational limits.

4.5.4 Safety Evaluation

In the event of a main steam line break outside the primary containment, steam flow rate is restricted in the venturi throat by a two-phase mechanism similar to the critical flow phenomenon in gas dynamics. This limits the steam quantity flow rate, thereby reducing the reactor vessel coolant blowdown and the fuel clad temperature increase subsequent to the blowdown. The probability of fuel failure and its consequences are therefore decreased.

Analysis of the steam line rupture accident (see Section 14.0, "Plant Safety Analysis") shows that the core remains covered and that the amount of radioactive materials released to the environs through the main steam line break does not exceed the values of 10 CFR 50.67.

Pressure surges caused by a two-phase mixture impinging on the flow restrictor result in stresses which do not exceed code-allowable limits. There is adequate margin in the code for withstanding the pressure load due to impact pressure from the possible oncoming two-phase mixture predicted during main steam line break accident conditions.

Tests were conducted on a scale model to determine final design and performance characteristics of the flow restrictor, including maximum flow rate of the restrictor corresponding to the accident conditions, irreversible losses under normal plant operating conditions, and discharge moisture level. The tests showed that the flow restrictor operation at critical throat velocities is stable and predictable. Unrecovered differential pressure across scale model restrictor is consistently about 10 percent of the total nozzle pressure differentials, and the restrictor performance is in agreement with existing ASME correlation. Full size restrictors have slightly different hydraulic shape and a differential pressure loss of approximately 15 percent.

4.5.5 Inspection and Testing

Because the flow restrictor forms a permanent part of the main steam line piping and has no moving components, no testing program is planned. Only very slow erosion will occur with time, and such a slight enlargement will not have safety significance.

4.6 MAIN STEAM ISOLATION VALVES

4.6.1 Safety Objectives

Two main steam isolation valves (MSIVs), one on each side of the primary containment barrier, in each of the main steam lines close automatically to:

a. Prevent damage to the fuel barrier by limiting the loss of reactor coolant water in case of a major leak from the steam piping outside the primary containment, and

b. Limit release of radioactive materials by closing the primary containment barrier in case of a major leak from the nuclear system inside the primary containment.

4.6.2 Safety Design Basis

The main steam isolation valves, individually or collectively, shall:

a. Close the steam lines within the time established by design basis accidents to limit the release of reactor coolant or radioactive materials,

b. Close the steam lines at a speed slow enough so that simultaneous (inadvertent) closure of all steam lines will not induce a more severe transient on the nuclear system than closure of the turbine stop valves while the bypass valves remain closed,

c. Close the steam lines when required despite single failure in either valve or the attached controls, to provide a high level of reliability for the safety function,

d. Use separate energy sources, as the motive force, to independently close the redundant main steam isolation valves in an individual steam line,

e. Use local stored energy (compressed air and springs) to close at least one main steam isolation valve in each steam line without relying on continuity of any variety of electrical power for the motive force to achieve closure,

f. Be able to close the steam lines during or after seismic loadings to ensure isolation if the nuclear system is breached by an earthquake, and

g. Be testable during normal operating conditions, to demonstrate that the valves will function.

4.6.3 Description

Two main steam isolation valves (MSIVs) are welded in a horizontal run of each of the four main steam lines, with one valve inside the primary containment barrier and the other as close as practical to the outside of the primary containment barrier (see Figures 4.5-1, 4.5-2, and 4.5-3 of Subsection 4.5). The valves, when closed, form part of the nuclear system process barrier for openings outside the primary containment, and part of the primary containment barrier for nuclear system breaks inside the containment.

The description and testing of the controls for the main steam isolation valves are included in Subsection 7.3, "Primary Containment and Reactor Vessel Isolation Control Systems."

A drawing of a main steam isolation valve is shown in Figure 4.6-1.

Each valve is a "Y"-pattern, 26-inch globe valve connected to matching 26-inch, schedule 80 (nominal I.D. 23.647 in.) pipe. A nominal rate of steam flow for Extended Power Uprate (3952 MWt) is 4.1 x 10^6 lb/hr at 1050 psia RPV dome pressure. The main disc or poppet is attached to the lower end of the stem and moves in guides at a 45-degree angle from the inlet pipe. Normal steam flow tends to close the valve and higher inlet pressure tends to hold the valve closed. The bottom end of the valve stem closes a small pressure-balancing hole in the poppet; when open, it acts as a pilot valve to relieve differential pressure forces on the poppet. The valve stroke for a 26-inch valve has approximately a 14-inch stem travel; the main poppet travels approximately 13 inches with approximately the last inch of valve travel closing the pilot hole. A helical spring between the stem and the poppet keeps the pilot hole open when the poppet is off its seat, but failure of the spring will not prevent closure of the valve. The air cylinder can open the poppet with a maximum of 200 psi differential pressure across the isolation valve in a direction tending to hold the valve closed.

The diameter of the poppet seat is approximately the same size as the inside diameter of the pipe, and the 45-degree angle permits stream lining of the inlet and outlet passage to minimize pressure drop during normal steam flow and to avoid blockage by debris. The valve stem penetrates the valve bonnet through a stuffing box having a set of replaceable packing. The poppet backseats on the bonnet cover in the fully open position, and leakage is prevented by the stem packing. The bonnet has provisions for seal welding in case leaks develop after the valve has extensive service.

The upper end of the stem is attached to a combination air cylinder and hydraulic dashpot that are used for opening and closing the valve and for speed control, respectively. Speed is adjusted by a valve in the hydraulic return line alongside the dashpot; the valve closing time is adjustable to meet the required Technical Specification limits (a minimum of three seconds and a maximum of five seconds).

The cylinder is supported on large shafts screwed and pinned into the valve bonnet. The shafts are also used as guides for the helical springs used to assist the valve to close. The springs exert downward force on the spring seat member which is attached to the stem. Spring guides prevent scoring in normal operation and prevent binding if a spring breaks. The spring seat member is also closely guided on the support shafts and rigidly attached to the stem to control any eccentric force in case of a broken spring.

On each MSIV, switches located at approximately 90 percent open, 85 percent open, and up to 7 percent open positions are actuated by the motion of the spring seat member. On each MSIV, the 90 percent open switches initiate reactor scram if several MSIVs close simultaneously (see Subsection 7.2, “Reactor Protection System”), the 85 percent open switch turns on the closed lights valve position, and the switch set at up to 7 percent open position indicates the valve is closed.

The MSIV is operated by pneumatic pressure and action of compressed springs. The control unit is attached to the air cylinder, and contains the pneumatic, AC and DC control valves for opening, closing, and slow-speed exercising of the main valve. The control power available is 120-V AC at 60 cycles and 250-V DC. Both the AC and DC control valve solenoids use approximately 0.5 amps of control power for each solenoid. Remote manual switches in the control room enable the operator to operate or close each valve either at fast speed for primary containment isolation or at the slow speed (approximately 45 to 60 seconds) for exercising or testing.

MSIV operating air is supplied at approximately 81 psig to 105 psig for the outboard valves and approximately 90 to 105 psig for the inboard valves from the various plant air systems through a check valve. An air accumulator between the control valve and the check valve provides a source of backup operating air.

This accumulator is designed to provide for one closing actuation following loss of air supply. Once closed, the valve is held closed by the springs.

The valve is designed for saturated steam flow at 1250 psig and 575°F, with a moisture content of approximately 0.23 percent.

In the event that the main steam line should rupture downstream from the valve, the steam flow quickly increases to no more than 200 percent of rated, flow being limited from further increase by the venturi flow restrictor upstream of the valves.

During valve closure, the MSIV initially has little effect in reducing flow because the flow is choked by the venturi restrictor upstream from the valves. After the main valve poppet enters the flow stream, flow is reduced as a function of the MSIV cross-sectional flow area versus travel characteristic.

The design objective for the valve is a minimum of 40 years of service at the specified operating conditions. The estimated operating cycles per year is 100 cycles during the first year and 50 cycles per year thereafter. In addition to minimum wall thickness required by applicable codes, a corrosion allowance of 0.120-inch minimum is added to provide for 40 years of service. For the 60 year operating life, the Technical Specification Surveillance Requirements will assure the MSIVs are capable of performing their design functions and the MSIV aging effects will be managed using the ASME Section XI Subsections IWB, IWC, and IWD Inservice Inspection Program, Chemistry Control Program, BWR Stress Corrosion Cracking Program and One-Time Inspection Program described in Appendix O, Sections O.1.4, O.1.5, O.1.10, and O.1.26.

Design specification normal and maximum ambient operating conditions for the MSIVs are tabulated in drawing 47E225-110 for each unit. See FSAR, Appendix M, Subsection M.8. However, the inside valves are not exposed to maximum conditions continuously, particularly during reactor shutdown, and the valves outside the primary containment and shielding are in much less severe ambient conditions.

The main steam isolation valve installations are designed as seismic Class I equipment to resist sufficiently the response motion at the installed location within the reactor building from the Design Basis Earthquake (see Appendix C). The valve assembly is manufactured to withstand the design basis seismic forces. The stresses caused by horizontal and vertical seismic forces are considered to act simultaneously and are added directly. The seismic coefficients are specified as 0.73g horizontal and 0.07g vertical. The stresses in the actuator supports caused by seismic loads are combined with the stresses caused by other live and dead loads including the operating loads. The allowable stress for this combination of loads is based on the ordinary allowable stress as set forth in the applicable codes. The parts of the main steam isolation valves which constitute a process fluid boundary are designed, fabricated, inspected, and tested as required by USAS B31.1.0, 1967 edition and the applicable GE design and procurement specifications, which were implemented in lieu of the outdated B31 Nuclear Code Cases-N2, N7, N9, and N10. The control valves and other equipment provided in the valve assembly were designed, manufactured, and shop-tested in accordance with the then-current revision of the following codes and standards, where applicable:

USA Standards Institute B31.1 and B16.5,
American Society for Testing and Materials (ASTM),
American Society of Mechanical Engineers (ASME),

ASME Boiler and Pressure Vessel Code, Sections I, III, and VIII,
American Institute of Electrical Engineers,
Pipe Fabrication Institute, and
National Electrical Manufacturers Association.

4.6.4 Safety Evaluation

In a direct cycle nuclear power plant, the reactor steam goes to the turbine and other equipment outside the reactor containments. The analysis of a complete sudden steam line break outside the primary containment is described in Section 14.0, "Plant Safety Analysis." It shows that the fuel barrier is protected against loss of cooling if main steam isolation valve closure takes as long as 5.5 seconds (includes up to 0.5 seconds for the instrumentation to initiate valve closure after the break and the maximum allowable valve stroke time). For the LOCA inside of containment, the inboard main steam isolation valve closure can take as long as 2 minutes, which is before any radiation releases occur as described in Section 14.6. The calculated radiological effects of the radioactive material assumed released with the steam are shown to be well within the guideline values for such an accident. Thus, safety design basis "a" is shown to be satisfied with considerable margin.

The shortest closing time (approximately 3 seconds) of the main steam isolation valves is also shown to be satisfactory by Chapter 14.0, "Plant Safety Analysis." The switches on the valves initiate reactor scram when several valves are ≤ 90 percent open. The pressure rise in the system, from stored and decay heat, may cause the nuclear system main steam relief valves to open briefly, but the rise in fuel cladding temperature will be insignificant. The transient is less than that from sudden closure of the turbine stop valves (in approximately 0.1 second), coincident with postulated failure of the turbine bypass valves to open. No fuel damage results. Thus, safety design basis "b" is shown to be satisfied with considerable margin.

The ability of this 45°, Y-design globe valve to close in a few seconds after a steam line break, under conditions of high pressure differentials and fluid flows, with fluid mixtures ranging from mostly steam to mostly water, has been demonstrated in a series of tests in dynamic test facilities. Dynamic tests with a 1-inch valve show that the analytical method is valid. A large size, 20-inch valve was tested in a range of steam/water blowdown conditions simulating postulated accident conditions.*

The following specified hydrostatic, leakage, and stroking tests, as a minimum, were performed by the valve manufacturer in shop tests.

---

* E. Van Zylstra, W. Sutherland, and D. Rockwell, "Design and Performance of GE BWR Main Steam Isolation Valves," General Electric Co., Atomic Power Equipment Department, March 1969 (APED-5750).

a. Each valve was tested at rated pressure (1,000 psig) and no flow to verify capability to close between 3 and 10 seconds. The valve was stroked several times and the closing time recorded. The valve is closed by the air cylinder and springs, and may also be closed by the springs only. The closing time is usually slightly greater when closed by springs only.

b. At least the first valve of each size was tested to demonstrate that the valve will close at rated pressure and no flow in the specified time after the valve had been held open (energized) for 1 week.

c. Leakage with the valve seated and backseated was measured. Seat leakage was measured by pressurizing the upstream side of the valve to 1250 psig. The specified maximum seat leakage, using cold water at design pressure, was 2cc per hour per inch of seat diameter. In addition, an air seat leakage test was conducted using 50 psi pressure upstream. Maximum permissible leak was 1/10 SCFH per inch of seat diameter. No visible leakage from the stem packing at design pressure was allowed. The valve stem was operated a minimum of three times from the closed to open position, and the packing leakage was verified to still be zero by visual examination.

d. Each valve was hydrostatically tested at USAS B16.5-specified test pressure (2,380 psig) with cold water.

e. During valve fabrication, extensive nondestructive tests and examinations were made, including radiographic, liquid penetrant, or magnetic particle examinations of castings, forgings, welds, hardfacings, and bolts.

The spring guides, the guiding of the spring seat member on the support shafts, and rigid attachment of the seat member ensure proper alignment of actuating components. Binding of the valve poppet in the internal guides is prevented by making the poppet in the form of a cylinder longer than its diameter, and by applying the stem force near the bottom of the poppet. Clearance is provided between the poppet and its guides so that some cocking of the poppet or warpage of the seat can be tolerated and still achieve a seal.

After the MSIVs were installed in the nuclear system, each valve was tested several times in accordance with the extensive "Preoperational Test Procedures," and "Startup Test Procedures." The startup tests were performed at several reactor operating conditions.

During the initial plant startup tests, the MSIV leak tightness was determined. When nuclear system pressure had reached approximately 800 psig, the leak tightness was checked by closing the MSIVs, evacuating the steam lines downstream and the turbine steam chest to the condenser, closing the steam chest valves, and recording the steam chest pressure. No pressure rise meant the valves were tight. If leakage

is indicated, each valve may be checked individually by opening the other valve in the same steam line with all other MSIVs closed, evacuating and closing the steam chest, and checking for pressure rise.

Redundancy is provided by two MSIVs in each steam line so that either can perform the isolation function, and either can be tested for leakage after closing the other. The inside valve, the outside valve and their control systems are physically separated. Considering the redundancy, the mechanical strength, the closing forces, and the leakage tests discussed above, the main steam isolation valves satisfy safety design bases "c", "d", and "e" to limit the release of reactor coolant or radioactive materials, within the margins evaluated in Section 14.0, "Plant Safety Analysis."

The MSIVs and their installation are designed as seismic Class I equipment for inclusion of seismic loadings, as delineated in Appendix C.

The design of the MSIVs for seismic loadings is discussed in paragraph 4.6.3 above. These loads are small compared with the pressure and operating loads the valve components are designed to withstand. The cantilevered support of the air cylinder, hydraulic cylinder, springs, and controls is the key area. The increase in loading at the joints between the support shafts and the valve bonnet caused by the specified earthquake loading is negligible. Therefore, the seismic loading requirement of design basis "f" is met.

Electrical equipment, associated with the MSIVs, that operates in an accident environment is limited to the wiring, solenoid valves, and position switches on the MSIVs. The design and purchase specifications for the wiring, solenoid valves, and position switches for accident environmental conditions are contained in the BFN 10 CFR 50.49 program. Under the accident conditions, ambient pressure and temperature increase to approximately 50 psig and 337°F; each valve is required to close within a 2 minute exposure to these conditions. The valve closing is completed during this two minute time frame.

Operation of the valves in the normal operating conditions and postulated accident environments is ensured by the requirements of the purchase specifications, review and approval of equipment design and vendor drawings, extensive control of materials, fabrication procedures, fabrication tests, nondestructive examinations, shop tests, preoperational and startup tests of the installed valves, and prescribed periodic inspections and tests during the plant life.

Safety design basis "g" is met, as described in paragraph 4.6.5.

4.6.5 Inspection and Testing

The main steam isolation valves may be tested during plant operation, and tested and inspected during refueling outages. The test operations are listed below.

The main steam isolation valves may be tested and exercised individually to the 85-percent-open position in the following manner. A minimum amount of load reduction may be required during testing.

a. Press the test pushbutton until the closed light goes on (85-percent-open position). The valve moves at the slow speed.

b. Release the test pushbutton and the valve will automatically reopen, turning off the closed light.

c. Repeat the test on each MSIV.

The main steam isolation valves may be tested and exercised individually to the fully closed position in the following manner.

a. Reduce reactor power to approximately 75 percent of full power.

b. Turn the MSIV control switch to the closed position, observing the time interval between switch closure and the open light going off. The closing time should be within the established Technical Specification limits.

c. Return the MSIV control switch to the open position.

d. Repeat the test on each MSIV.

e. After all the MSIVs have been tested, reactor power may be returned to the normal level.

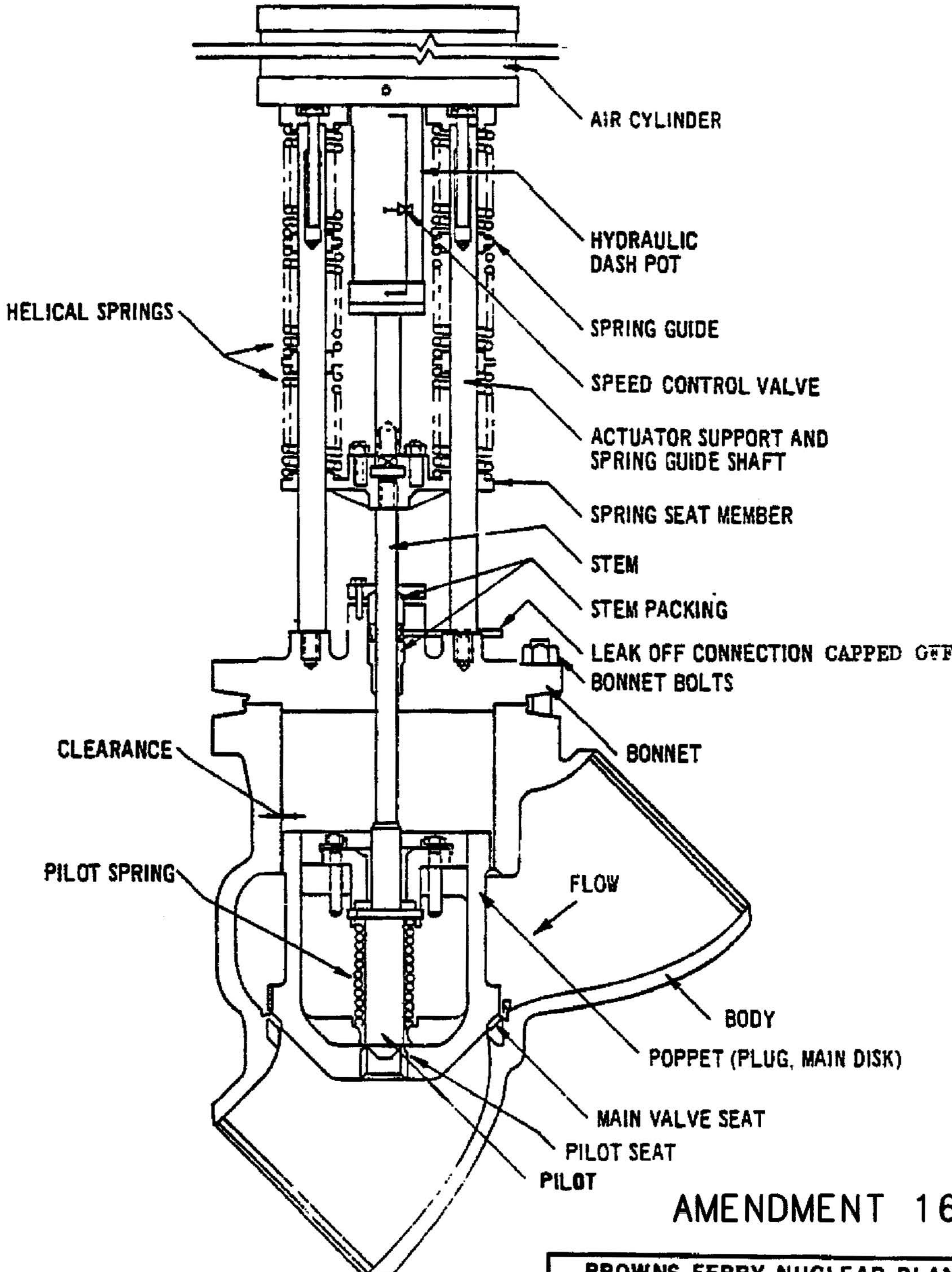
During reactor shutdowns for refueling, the main steam isolation valves are tested and visually inspected as necessary.

Leakage from the valve stem packing may become suspect, during reactor operation, from measurements of leakage into the primary containment or from observations or similar measurements in the secondary containment. During shutdown, while the nuclear system is pressurized, the leak rate through the packing can be observed by visual inspection.

The leak rate through the MSIV seats (pilot and poppet seats) can be measured accurately during shutdown by pressurizing between the closed valves with compressed gas.

During pre-startup tests following a refueling outage or MSIV disassembly, the valves will receive the same hydrostatic or inservice leakage tests which are imposed on the primary system.

This test and leakage measurement program will ensure that the valves are operating properly, and that a leakage trend is detected.



AMENDMENT 16

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Main Steam Isolation Valve
Figure 4.6-1

4.7 REACTOR CORE ISOLATION COOLING SYSTEM

4.7.1 Power Generation Objective

The Reactor Core Isolation Cooling System (RCICS) provides makeup water to the reactor vessel during shutdown and isolation from the main heat sink to supplement or replace the normal makeup sources and operates automatically in time to obviate any requirement for the Core Standby Cooling Systems (see Chapter 6, “Emergency Core Cooling Systems”).

4.7.2 [Deleted]

4.7.3 Power Generation Design Basis

1. The system shall operate automatically in time to maintain sufficient coolant in the reactor vessel so that the Core Standby Cooling Systems are not required.

2. Provision shall be made for remote-manual operation of the system by an operator.

3. The power supply for the system shall be provided by immediately-available energy sources of high reliability in order to provide a high degree of assurance that the system shall operate when necessary.

4. Provision shall be made so that periodic testing can be performed during plant operation, in order to provide a high degree of assurance that the system shall operate when necessary.

4.7.4 Safety Design Basis

Piping and equipment, including support structures, shall be designed to withstand the effects of an earthquake without a failure which could lead to a release of radioactivity in excess of the guideline values given in 10 CFR 50.67.

4.7.5 Description

The RCICS consists of a steam-driven, turbine-pump unit and associated valves and piping capable of delivering makeup water to the reactor vessel. A summary of the design requirements of the turbine-pump unit is shown on Table 4.7-1. The transient analyses are based on a RCIC flow rate of 600 GPM. A system diagram is shown in Figures 4.7-1a, 4.7-1c, and 4.7-1e.

The steam supply to the turbine comes from the main steam line from the reactor vessel. The steam exhaust from the turbine dumps to the pressure suppression pool. The pump takes suction from the condensate header, or from the pressure suppression pool header, via a core spray pump supply header. The pump discharges either to the feedwater line or to a full-flow return test line to the condensate storage tanks. A minimum-flow bypass line to the pressure suppression pool is provided for pump protection. The makeup water is delivered into the reactor vessel through a connection to the feedwater line and is distributed within the reactor vessel through the feedwater sparger. The connection to the feedwater line is provided with a thermal sleeve. Cooling water for the RCICS turbine lube-oil cooler and gland-seal condenser is supplied from the discharge of the pump (see Figures 4.7-1b, 4.7-1d, and 4.7-1f). Whenever RCIC is lined up to take suction from the condensate storage tank, the discharge piping of the RCIC is periodically vented from the high point of the system and water flow observed in accordance with Technical Specifications surveillance frequency requirements for system operability.

Following any reactor shutdown, steam generation continues due to heat produced by the radioactive decay of fission products. Initially the rate of steam generation can be as much as approximately 6 percent of rated flow, and is augmented during the first few seconds by delayed neutrons and some of the residual energy stored in the fuel. The steam normally flows to the main condenser through the turbine bypass or, if the condenser is isolated, through the main steam relief valves to the pressure suppression pool. The fluid removed from the reactor vessel can be entirely made up by the feedwater pumps if the main steam line isolation valves are open or partially made up from the Control Rod Drive System which is supplied by the control rod drive feed pumps. If makeup water is required to supplement these sources of water, the RCICS turbine-pump unit either starts automatically upon receipt of a Reactor Vessel Water Level - Low Low, Level 2 signal or is started by the operator from the control room by remote-manual controls. The RCICS delivers its design flow within 30 seconds after actuation. To limit the amount of fluid leaving the reactor vessel, a Reactor Vessel Water Level - Low Low Low, Level 1 signal also actuates the closure of the main steam isolation valves.

For events other than pipe breaks, RCICS has a makeup capacity sufficient to prevent the reactor vessel water level from decreasing to the level where the core is uncovered without the use of Core Standby Cooling Systems (see Section 14.0, "Plant Safety Analysis"). The pump suction is normally lined up to the condensate storage tanks through the condensate supply header. Other systems which use the same tanks for condensate, and could jeopardize the availability of this reserve quantity, are restricted by a standpipe to the use of water in the upper portion of the tanks. About 135,000 gallons are below the standpipe in each condensate tank. This quantity represents the conservatively calculated amount of water required to maintain reactor vessel level for at least 5.5 hours in hot shutdown conditions (MODE 3).

The backup supply of cooling water for the RCICS is the pressure suppression pool Ring Header. The turbine-pump assembly is located below the level of the condensate storage tank and below the minimum water level in the pressure suppression pool to ensure positive suction head to the pump. Pump NPSH requirements are met by providing adequate suction head and adequate suction line size.

All components normally required for initiating operation of the RCICS are completely independent of auxiliary AC power, plant service air, and external cooling water systems, requiring only DC power from a unit battery to operate the valves, vacuum pump, and condensate pumps. The power source for the turbine-pump unit is the steam generated in the reactor pressure vessel by the decay heat in the core. The steam is piped directly to the turbine, and the turbine exhaust is piped to the pressure suppression pool.

If for any reason the reactor vessel is isolated from the main condenser, pressure in the reactor vessel increases but is limited by automatic or remote-manual actuation of the main steam relief valves. Main steam relief valve discharge is piped to the pressure suppression pool. Throughout the period of RCICS operation, the exhaust from the RCICS turbine and main steam relief valve discharge being condensed in the pressure suppression pool results in a temperature rise in the pool. During this period RHR heat exchangers are used to control pool water temperature, if normal AC power is available for operation of the RHR system. The results of a conservative isolation scenario at an initial power level of 102% of 3952 MWt, where 1) it is assumed that only a single RHR pump and RHR heat exchanger are available for pool cooling, 2) pool cooling is delayed for 10-minutes following the isolation and 3) the RCICS suction is from the condensate storage tank, show the maximum pool temperature of (approximately) 184ºF would be reached at about 4 hours into the event.

The RCICS turbine-pump unit is located in a shielded area to ensure that personnel access areas are not restricted during RCICS operation. An analysis of the possibility of the failure of the RCIC turbine has been performed. Stresses in the turbines are sufficiently low, such that wheel failure is not predicted, even at the theoretical run-away condition of twice rated speed. Even though similar results were obtained for the analysis of the HPCI turbine, the HPCI and RCIC turbines are located in separate concrete rooms within the Reactor Building. An assumed failure of either turbine could not cause sufficient damage to prevent safe shutdown of the plant. The turbine controls provide for automatic trip of the RCICS turbine upon receiving any of the following signals:

a. Turbine overspeed--to prevent damage to the turbine and turbine casing,

b. Pump low-suction pressure--to prevent damage to the turbine-pump unit due to loss of cooling water,

c. Turbine high-exhaust pressure--indicating turbine or turbine control malfunction, and

d. Automatic isolation signal-indicating RCIC steam line rupture.

Since the steam supply line to the RCICS turbine is a primary containment boundary, certain signals automatically isolate this line causing shutdown of the RCICS turbine. Automatic shutdown of the steam supply is described in Subsection 7.3, "Primary Containment and Reactor Vessel Isolation Control System."

The turbine control system is positioned by the demand signal from a flow controller, and satisfies a twofold purpose:

a. Limit the turbine pump speed to its maximum normal operating value, and

b. Position the turbine governor valve(s) as required to maintain constant pump discharge flow over the pressure range of system operation.

The RCICS piping within the drywell up to and including the outer isolation valve is designed in accordance with the USA Standard Code for Pressure Piping, USAS B31.1.0, 1967 edition, and the applicable GE design and procurement specifications, which were implemented in lieu of the out dated B31 Nuclear Code Cases-N2, N7, N9, and N10, plus ASME Boiler and Pressure Vessel Code, Section I, 1965 edition. Other piping is designed in accordance with the USAS B31.1.0, 1967 edition, as applicable. The thermal sleeve (liner) in the feedwater line is designed as a nonpressure-containing liner and is provided to protect the pressure-containing piping tee from excessive thermal stress.

4.7.6 Safety Evaluation

The safety design basis is satisfied by design of the RCICS containment function to seismic Class I specifications (see Appendix C).

4.7.7 Inspection and Testing

A design flow functional test of the RCICS is performed during plant operation by taking suction from the condensate header and discharging through the full flow test return line back to the condensate storage tank. The discharge valve to the feed line remains closed during the test and reactor operation is undisturbed. Testing of the pump discharge valve is accomplished in accordance with Subsection 4.12, Inservice Inspection and Testing. Control system design provides automatic return from test to operating mode if system initiation is required during testing. Periodic inspection and maintenance of the turbine-pump unit are based on manufacturer's recommendations and sound maintenance practices. Valve position indication, as well as instrumentation alarms, is displayed in the control room.

Table 4.7-1

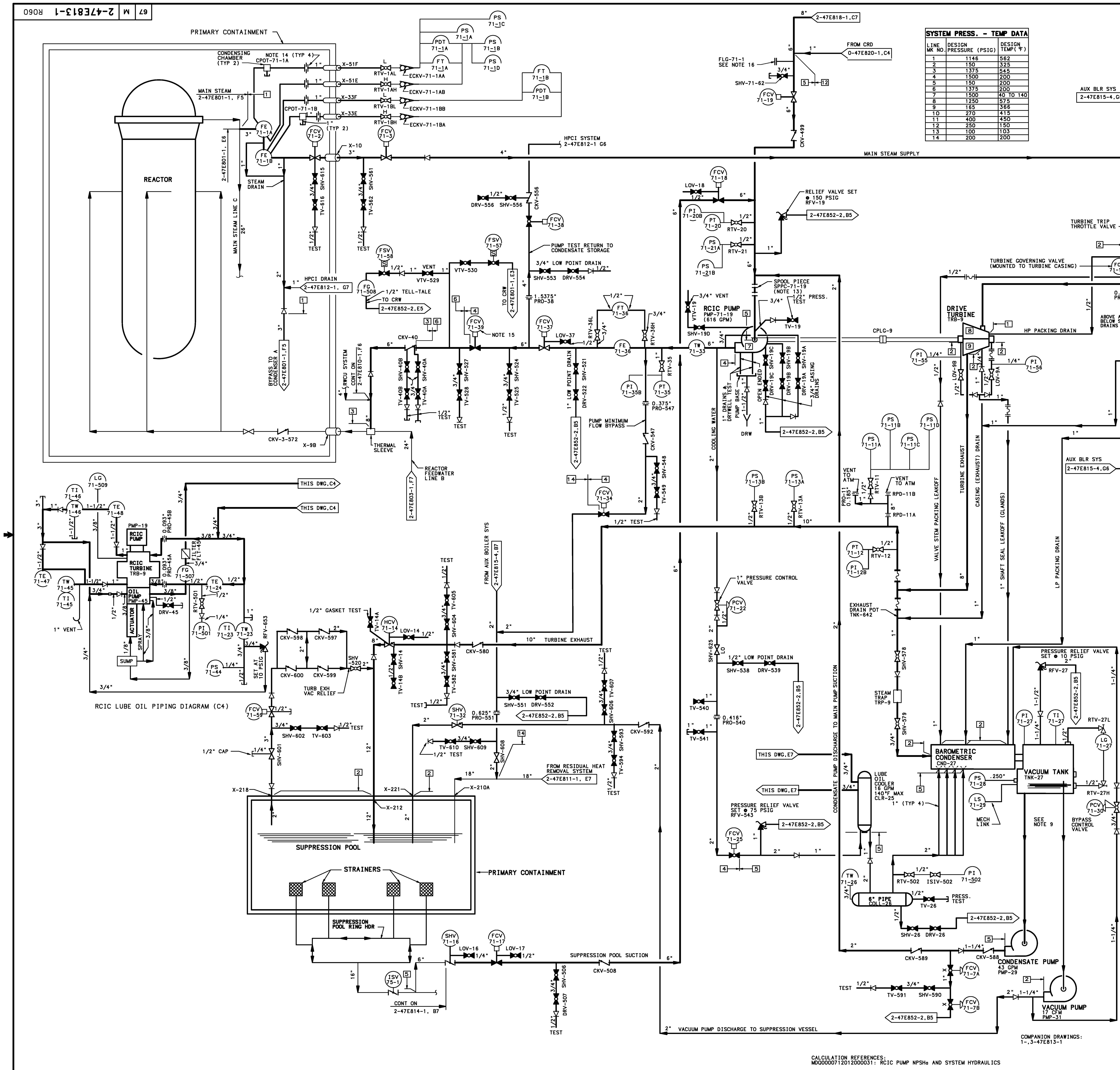
REACTOR CORE ISOLATION COOLING SYSTEM TURBINE - PUMP DESIGN DATA

PUMP

Number required - 1	Design Temperature - 40°F to 140°F
Capacity - 100%	Design Pressure - 1500 psig NPSH - 20 ft (minimum)
Developed Head -	
2930 ft	@ 1189 psia reactor pressure
525 ft	@ 165 psia reactor pressure
Flow Rate	Injection Flow 600 gpm Cooling Water Flow 16 gpm Total Pump Discharge 616 gpm

TURBINE

Number required - 1	
Capacity - 100%	
Steam Inlet Pressure range (psig)	150 to 1174 (saturated)
Steam Exhaust Pressure (psia)	25 (Unit 1) 32 (Unit 2), 29 (Unit 3)



SYSTEM PRESS. - TEMP DATA		
LINE MK NO.	DESIGN PRESSURE (PSIG)	DESIGN TEMP(°F)
1	1146	562
2	150	325
3	1375	545
4	1500	200
5	150	200
6	1375	200
7	1500	40 TO 140
8	1250	575
9	165	366
10	270	415
11	400	450
12	250	150
13	100	103
14	200	200

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 VALVES ARE THE SAME SIZE AS PIPING, UNLESS OTHERWISE NOTED.
4. ALL VALVE DRAINS AND VENTS ARE PIPED TO CRW.
5. 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, SYS 71 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-502
	BFN-2-PT-003-0204C	PT-204C

6. [1] ETC, DENOTES DESIGN PRESSURE AND TEMPERATURE AS GIVEN IN TABLE, THIS DRAWING.
7. DELETED
8. THE DESIGN PRESSURE AND TEMPERATURE OF ALL DRAIN AND VENT LINES THROUGH THE LAST ISOLATION VALVE SHALL BE THE SAME AS THE PROCESS LINE.
9. CONDENSATE PUMP AND VACUUM PUMP SUCTION LINES ARE INTERNAL TO TANK AND SHOWN ONLY FOR CLARITY.
10. HYDROTESTING OF THERMOWELLS AND FITTINGS SHALL BE PERFORMED IN SHOP AT 30 PSIG. SYSTEM LEAK TEST PER ANSI B31.10 POWER PIPING CODE.
11. FOR DESIGN PRESSURE AND DESIGN TEMPERATURE SEE TERRY STEAM TURBINE MANUAL, CONTROL # 0467 (CONTRACT # 90744).
12. 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.
13. SPOOL PIECE MAY BE REPLACED BY STRAINER DURING MAINTENANCE. THE SPOOL PIECE MUST BE INSTALLED PRIOR TO RETURNING THE SYSTEM TO SERVICE. SEE 0-47E456-1, -2 FOR STRAINER DETAIL.
14. ORIFICE COUPLING, SEE 0-47W600-20, DETAIL P20.
15. VALVE 2-FCV-071-0039 WILL SEAL LEAK TIGHT ONLY IN THE DIRECTION OF FEEDWATER TO RCIC SYSTEM DUE TO A PRESSURE LOCKING MODIFICATION WHICH DRILLED A 1/4" HOLE IN THE FEEDWATER SIDE FACE OF THE VALVE. (DCN T41301)
16. THE PIPING COMPONENT IDENTIFIED IS A LINE STOP FITTING WITH AN INTERNAL PLUG AND BLIND FLANGE. NORMAL CONFIGURATION DOES NOT ISOLATE FLOW AND FITTING, PLUG AND BLIND FLANGE ARE CONSIDERED PRESSURE RETAINING.

REFERENCE DRAWINGS:

MEL......................VALVE MARKER TAG TABULATION
2-47E456-408..........MECHANICAL-RCIC OIL PIPING INSTRUMENTS
2-47E852-1,-2.........FLOW DIAGRAM - DRAINAGE INSTALLATION SECTIONS AND DETAILS
0-47E820-1............FLOW DIAGRAM - CONTROL ROD DRIVE
0-47E800-1............FLOW DIAGRAM - GENERAL PLANT SYSTEMS
0-47E800-2............MECHANICAL - SYMBOLS & FLOW DIAGRAM DRAWING INDEX
2-47E815-4............FLOW DIAGRAM AUXILIARY BOILER SYSTEM
2-47E801-1............FLOW DIAGRAM - MAIN STEAM
2-47E811-1............FLOW DIAGRAM - RHR SYSTEM
2-47E812-1............FLOW DIAGRAM - HPCI SYSTEM
0-47E815-1............FLOW DIAGRAM - AUX BOILER SYSTEM
2-47E818-1............FLOW DIAGRAM - CNDS STORAGE & SUPPLY SYSTEM
MEL......................INSTRUMENT TABULATION
2-47E610-71-1,2.......CONTROL DIAGRAM - RCIC SYSTEM
2-47E803-1............FLOW DIAGRAM - REACTOR FEEDWATER
2-47E810-1............FLOW DIAGRAM - REACTOR WATER CLEANUP
2-47E814-1............FLOW DIAGRAM - CORE SPRAY

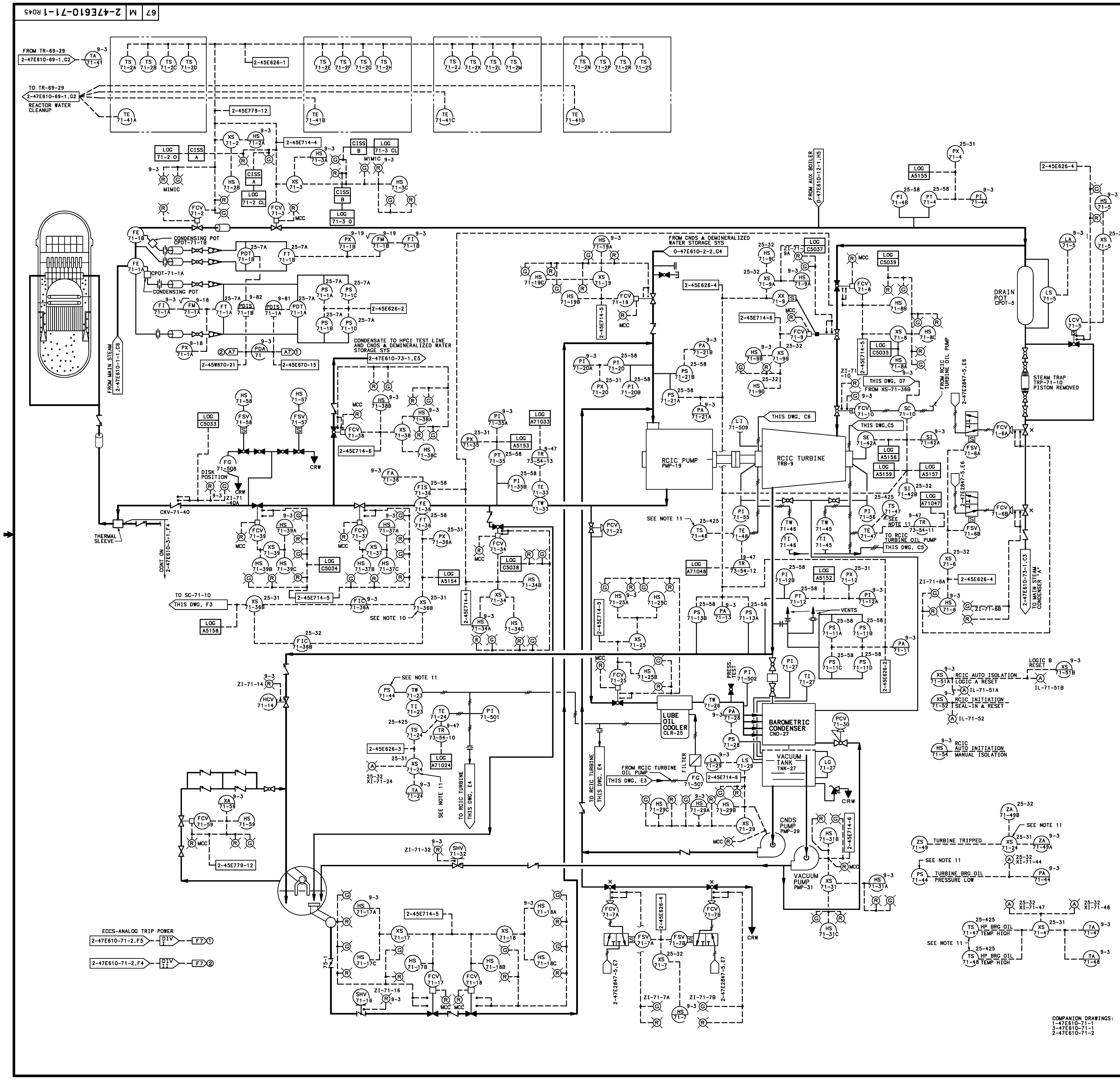
AMENDMENT 26

POWERHOUSE
UNIT 2

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

REACTOR CORE ISOLATION COOLING SYSTEM FLOW DIAGRAM

FIGURE 4.7-1a



NOTES:

1. THE REACTOR CORE ISOLATION COOLING SYSTEM IS DESIGNED TO COOL THE CORE WHEN THE REACTOR VESSEL IS ISOLATED AND REACTOR FEEDWATER IS UNAVAILABLE. WHEN THE WATER LEVEL IN THE REACTOR VESSEL REACHES SETPOINT, THE RCIC SYSTEM IS AUTOMATICALLY STARTED BY ACTIVATION OF REACTOR LEVEL SWITCHES LIS-3-58A, LIS-3-58B, LIS-3-58C AND LIS-3-58D (2-47E610-3-1), ARRANGED IN ONE-OUT-OF-TWO-TWICE LOGIC TO OPEN VALVE FCV-71-8. THE SAME SIGNAL CLOSES DRAIN LINE VALVES FCV-71-6A AND FCV-71-6B.

2. A MECHANICAL SPEED GOVERNOR PREVENTS TURBINE OVERSPEED BY TRIPPING THROTTLE TRIP VALVE FCV-71-9 WHEN A TURBINE OVERSPEED OCCURS. A CONTROL GOVERNOR, FIC-71-36, EQUIPPED WITH AUTOMATIC AND MANUAL MEANS FOR SPEED SET POINT CONTROL REGULATES TURBINE SPEED VIA "TURBINE SPEED CHANGER" TO CONTROL RCIC MAIN PUMP DISCHARGE FLOW.

3A. A RCIC STANDBY MODE SIGNAL SHALL BE INITIATED AT REACTOR VESSEL WATER LEVEL 8 (HIGH LEVEL). THE REACTOR VESSEL WATER LEVEL 8 SIGNAL SHALL BE INITIATED BY TWO LEVEL SWITCHES ARRANGED IN A TWO-OUT-OF-TWO LOGIC CONFIGURATION. THE REACTOR VESSEL WATER LEVEL 8 SIGNAL SHALL: (A) CLOSE THE STEAM ADMISSION VALVE FCV-71-8 AND (B) CLOSE THE MINI-FLOW ISOLATION VALVE FCV-71-34. IN THE EVENT AN INITIATION SIGNAL IS RECEIVED AFTER THE SYSTEM HAS BEEN SHUTDOWN ON HIGH RV LEVEL, THE SYSTEM SHALL BE CAPABLE OF AUTOMATIC RESTART.

3B. THE RCIC TURBINE TRIP VALVE IS TRIPPED BY:
A. TURBINE OVERSPEED VIA THE SPEED GOVERNOR.

C. LOW RCIC PUMP SUCTION PRESSURE SENSED BY PS-71-21.

D. HIGH RCIC TURBINE EXHAUST PRESSURE SENSED BY EITHER PS-71-13A OR PS-71-13B.
E. MANUALLY FROM THE MAIN CONTROL ROOM BY HS-71-9A OR BY HS-71-9C ON PANEL 25-32.

A TURBINE TRIP:

A. CLOSES FCV-71-9.
B. CLOSES FCV-71-34.
C. ANNUNCIATES ON ZA-71-49A OR 49B.
D. ACTIVATES THE ALARM ASSOCIATED WITH THE CONDITION INITIATING THE TRIP.

4. THE TURBINE CASING IS PROTECTED FROM OVERPRESSURE BY TWO RUPTURE DISCS. PS-71-11A THRU D DETECTS DISC RUPTURES AND ANNUNCIATES ON PA-71-11 IN THE MAIN CONTROL ROOM. PROVIDES 1 OUT OF 2 TWICE LOGIC TO INITIATE AUTO ISOLATION OF TURBINE (SEE NOTE 5).

5. THE RCIC TURBINE WILL BE AUTOMATICALLY ISOLATED FROM THE STEAM SUPPLY BY CLOSING ISOLATION VALVES FCV-71-2 AND FCV-71-3 IF:

A. A HIGH PRESSURE DROP OCCURS ACROSS THE RCIC ELBOW TYPE FLOW ELEMENT FE-73-1A & 1B WHEN THE STEAM LINE RUPTURES. THE HIGH FLOW IS SENSED BY EITHER PDT-71-1A OR PDT-71-1B AND ANNUNCIATED ON PDA-71-1.
B. HIGH AREA TEMPERATURES INDICATING A STEAM LINE BREAK ARE DETECTED BY ANY OF FOUR SETS OF TEMPERATURE SWITCHES (TS-71-2A THROUGH TS-71-2S). EACH SET CONNECTED IN ONE-OUT-OF-TWO-TWICE LOGIC. FOUR INDEPENDENT TEMPERATURE SENSING ELEMENTS (TE-71-41A THROUGH TE-71-41D) ACTIVATE THE MAIN CONTROL ROOM ANNUNCIATOR TA-71-41.

C. LOW REACTOR PRESSURE IS DETECTED BY PS-71-1A THROUGH PS-71-1D ARRANGED IN ONE-OUT-OF-TWO-TWICE LOGIC.
D. HIGH PRESSURE BETWEEN THE TURBINE EXHAUST RUPTURE DISCS (SEE NOTE 4).

6. NORMALLY MAKEUP WATER FOR THE RCIC PUMP SUCTION IS SUPPLIED BY THE CONDENSATE STORAGE TANK THROUGH FLOW CONTROL VALVE FCV-71-19. AN ALTERNATE SOURCE OF WATER IS OBTAINED FROM THE SUPPRESSION POOL THROUGH ISOLATION VALVES FCV-71-17 AND FCV-71-18. PUMP DISCHARGE FLOW ROUTES ARE AS FOLLOWS:

A. NORMALLY VIA THE FEEDWATER LINE TO THE REACTOR THROUGH ISOLATION VALVES FCV-71-37, FCV-71-39, AND CHECK VALVE FCV-71-40.
B. DURING TESTING VIA TEST LINE TO CONDENSATE STORAGE THROUGH FLOW CONTROL VALVE FCV-71-38.
C. MINIMUM FLOW BYPASS TO THE SUPRESSION POOL THROUGH FLOW CONTROL VALVE FCV-71-34.

7. IF RCIC SYSTEM OPERATION IS REQUIRED WHEN IT IS IN THE TEST MODE, THE CONTROL SYSTEM IS DESIGNED TO AUTOMATICALLY RETURN THE SYSTEM TO THE OPERATING MODE.

8. COOLING WATER FOR THE PUMP, TURBINE, AND LUBE OIL COOLER IS SUPPLIED FROM THE PUMP DISCHARGE LINE THROUGH PRESSURE CONTROL VALVE PCV-71-22 SET AT 75 PSIA.

9. TRANSFER AND CONTROL SWITCHES ASSOCIATED WITH BACKUP CONTROL WILL BE ON THE DESIGNATED PANELS OR LOCATED ON THE APPROPRIATE 250V DC MOV BOARDS.

10. XS-71-36B IS SHOWN TWICE TO SHOW HOW THIS TRANSFER SWITCH DISCONNECTS FIC-71-36A AND CONNECTS FIC-71-36B LOCATED AT THE BACKUP CONTROL CENTER.

11. XS-71-24, PS-71-44, TS-71-47 & TS-71-48 ARE SHOWN TWICE TO SHOW HOW THESE SWITCHES TRANSFER INDICATION FROM THE MAIN CONTROL PANEL 9-3 TO THE BACKUP CONTROL PANEL 25-32.

12. 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 SYSTEM 71, 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

13. COMPLETE LOGIC FOR REACTOR CORE ISOLATION SYSTEM IS SHOWN ON MECHANICAL LOGIC DIAGRAM LISTED IN REFERENCE DRAWINGS.

REFERENCE DRAWINGS:

0-47E800-1..................FLOW DIAGRAM-GENERAL PLANT SYSTEMS
0-47E800-2..................MECHANICAL SYMBOLS AND FLOW DIAG DWG INDEX
2-47E2847-5.................FLOW DIAGRAM-CONTROL AIR SUPPLY
2-47E813-1..................FLOW DIAGRAM-RCIC
2-47E610-73-1...............CONTROL DIAGRAM-HPCI
2-47E610-3-1................CONTROL DIAGRAM-REACTOR FEEDWATER
2-47E610-1-1................CONTROL DIAGRAM-MAIN STEAM
2-47E610-69-1...............CONTROL DIAGRAM-REACTOR WATER CLEANUP
0-47E610-2-2................CONTROL DIAGRAM-CONDENSATE SYSTEM
0-47E610-12-1...............CONTROL DIAGRAM-AUXILIARY BOILER SYSTEM
0-47E456-1,-2...............RCIC SYSTEM MECHANICAL DRAWINGS
0-47W456-3,-4...............RCIC SYSTEM MECHANICAL DRAWINGS
MEL.........................INSTRUMENT TABULATIONS FOR SYSTEM 71
2-45E670-SERIES.............WIRING DIAGRAM-ECCS
2-45E626-1,-2,-3,-4.........WIRING DIAG REACTOR CORE ISOLATION COOLING
2-45W670-21.................WIRING DIAG EMERG CORE COOLING SYS DIV I
2-45E714-5,-6...............WIRING DIA 250V REAC MOV BD 2C
2-45E779-5,-12..............WIRING DIAGRAM 480V SHUTDOWN AUX POWER SCH DIA
2-47E611-64-SERIES..........MECHANICAL LOGIC DIAGRAM PRIMARY CONTAINMENT ISOLATION SYSTEM
2-47E611-71-SERIES..........MECHANICAL LOGIC DIAGRAM REACTOR CORE ISOLATION COOLING SYSTEM

GE DRAWINGS:

729E652-1.............. FUNCTIONAL CONTROL DIAGRAMS-RCIC
2-729E652-4.............FUNCTIONAL CONTROL DIAGRAMS-RCIC

TERRY TURBINE CO.:
C-883-X .... RCIC TURBINE GOVERNOR CONTROL SYSTEM

NASH ENGR CO.:
D-14-1365 .... RCIC BAROMETRIC CONDENSER

SYMBOLS:

☒ QUICK DISCONNECT WITH SINGLE END SHUTOFF

☒ QUICK DISCONNECT

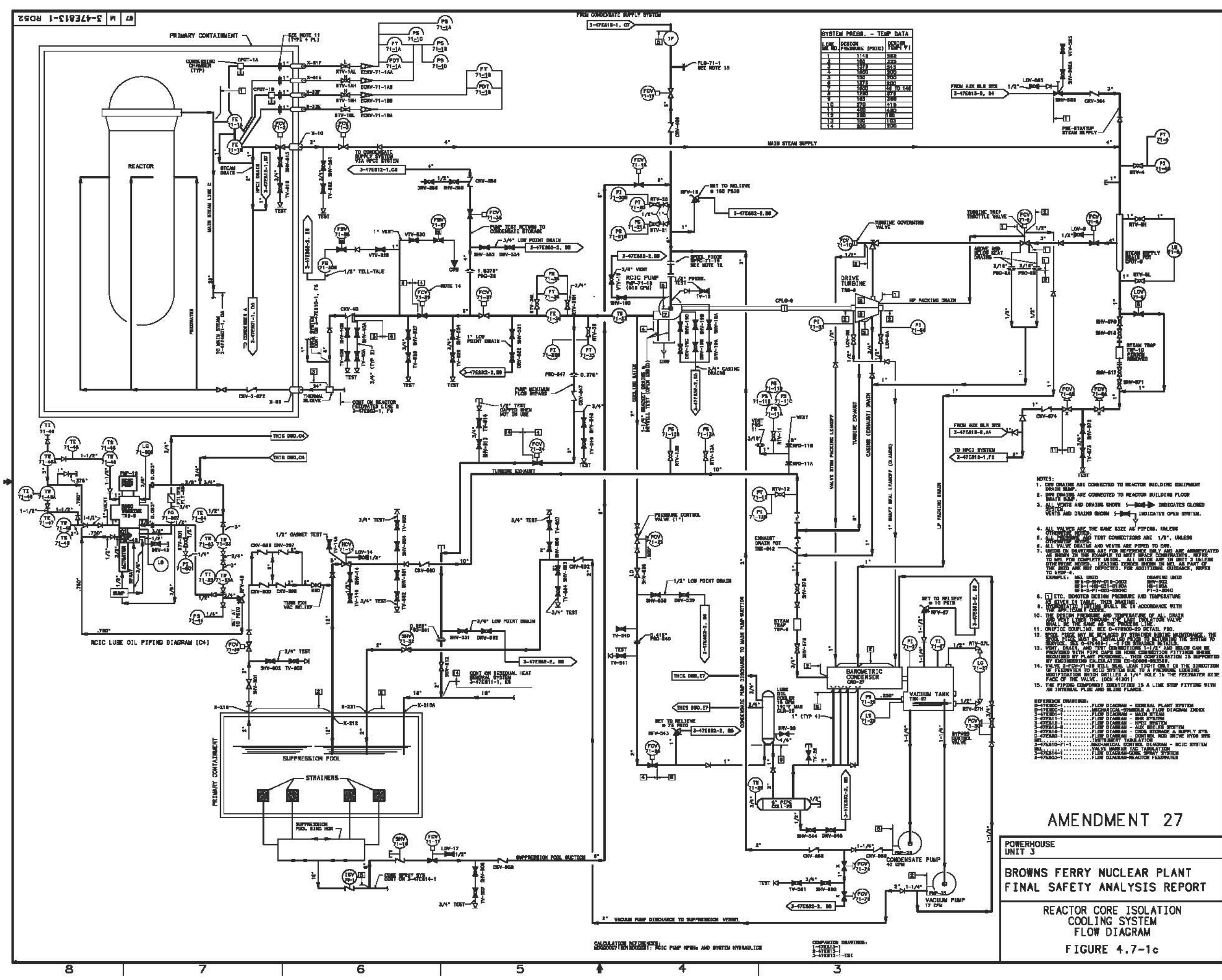
AMENDMENT 28

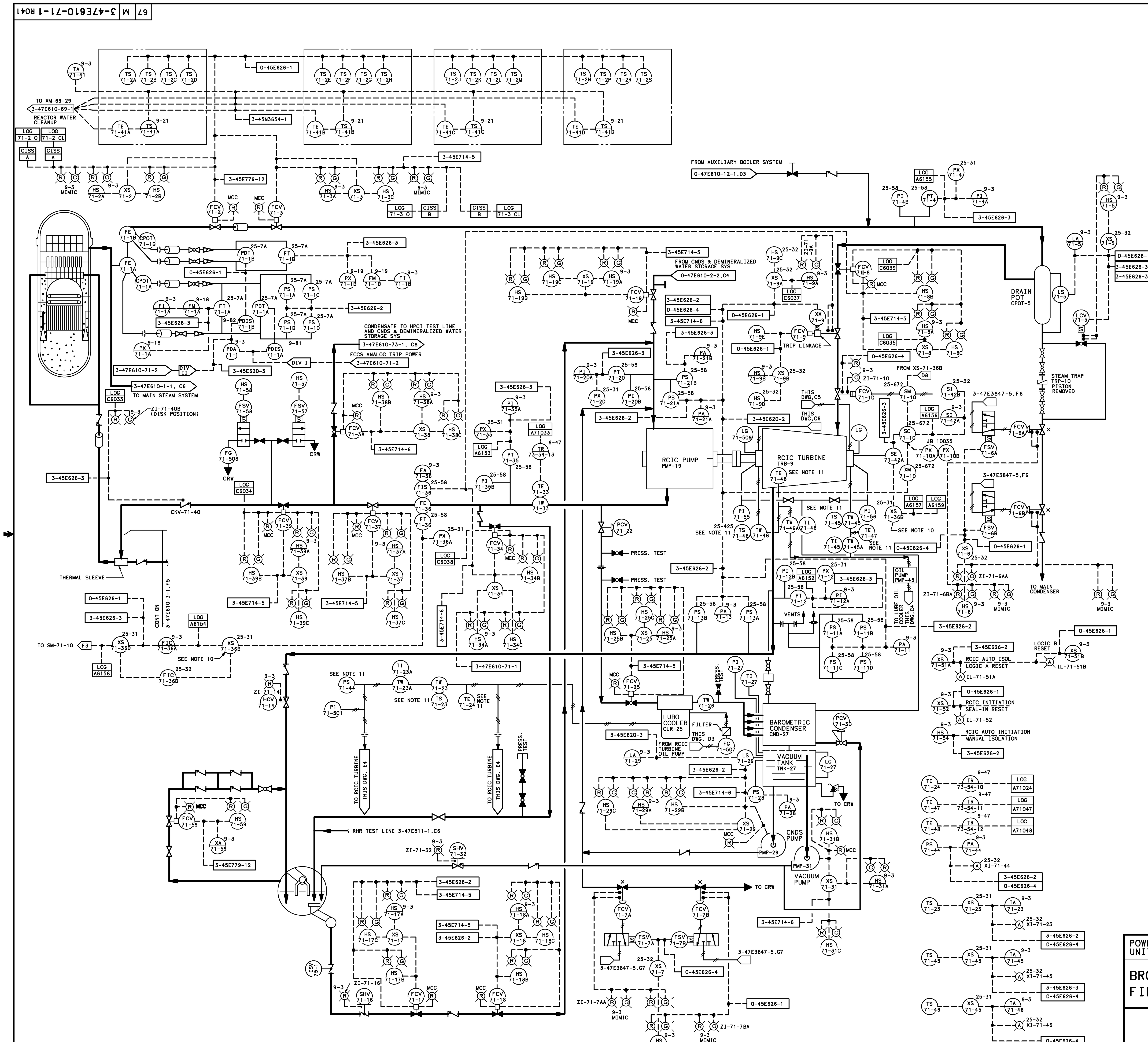
POWERHOUSE
UNIT 2

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

REACTOR CORE ISOLATION
COOLING SYSTEM
MECHANICAL CONTROL DIAGRAM

FIGURE 4.7-1b


AMENDMENT 27
POWERHOUSE
UNIT 3
BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT
REACTOR CORE ISOLATION
COOLING SYSTEM
FLOW DIAGRAM
FIGURE 4.7-1c



COMPANION DRAWINGS:
1-47E610-71-1
2-47E610-71-1

NOTES:

1. THE REACTOR CORE ISOLATION COOLING SYSTEM IS DESIGNED TO COOL THE CORE WHEN THE REACTOR VESSEL IS ISOLATED AND REACTOR FEEDWATER IS UNAVAILABLE. WHEN THE WATER LEVEL IN THE REACTOR VESSEL REACHES 31.5" BELOW REACTOR FEEDWATER ZERO, THE RCIC SYSTEM IS AUTOMATICALLY STARTED BY ACTIVATION OF REACTOR LEVEL SWITCHES LIS-3-58A, LIS-3-58B, LIS-3-58C, AND LIS-3-58D (SEE 3-47E610-3-1), ARRANGED IN ONE-OUT-OF-TWO-TWICE LOGIC TO OPEN VALVE FCV-71-8. THE SAME SIGNAL CLOSES DRAIN LINE VALVES FCV-71-6A AND FCV-71-6B.
2. A MECHANICAL SPEED GOVERNOR PREVENTS TURBINE OVERSPEED BY TRIPPING THROTTLE TRIP VALVE FCV-71-9 WHEN A TURBINE OVERSPEED OCCURS. A CONTROL GOVERNOR, FIC-71-36, EQUIPPED WITH AUTOMATIC AND MANUAL MEANS FOR SPEED SET POINT CONTROL REGULATES TURBINE SPEED VIA "TURBINE SPEED CHANGER" TO CONTROL RCIC MAIN PUMP DISCHARGE FLOW.

3A. A RCIC STANDBY MODE SIGNAL SHALL BE INITIATED AT REACTOR VESSEL WATER LEVEL 8 (HIGH LEVEL). THE REACTOR VESSEL WATER LEVEL 8 SIGNAL SHALL BE INITIATED BY TWO LEVEL SWITCHES ARRANGED IN A TWO-OUT-OF-TWO LOGIC CONFIGURATION. THE REACTOR VESSEL WATER LEVEL 8 SIGNAL SHALL: (A) CLOSE THE STEAM ADMISSION VALVE FCV-71-8 AND (B) CLOSE THE MINI FLOW ISOLATION VALVE FCV-71-34. IN THE EVENT AN INITIATION SIGNAL IS RECEIVED AFTER THE SYSTEM HAS BEEN SHUTDOWN ON HIGH RV LEVEL, THE SYSTEM SHALL BE CAPABLE OF AUTOMATIC RESTART.

3B. THE RCIC TURBINE TRIP VALVE IS TRIPPED BY:
   A. TURBINE OVERSPEED VIA THE SPEED GOVERNOR.
   B. DELETED.
   C. LOW RCIC PUMP SUCTION PRESSURE SENSED BY PS-71-21A.
   D. HIGH RCIC TURBINE EXHAUST PRESSURE SENSED BY EITHER PS-71-13A OR PS-71-13B.
   E. MANUALLY FROM THE MAIN CONTROL ROOM BY HS-71-9A OR BY HS-71-9C ON PANEL 25-32.

A TURBINE TRIP:
   A. CLOSES FCV-71-9.
   B. CLOSES FCV-71-34.
   C. ANNUNCIATES ON ZA-71-49A OR 49B.
   D. ACTIVATES THE ALARM ASSOCIATED WITH THE CONDITION INITIATING THE TRIP.

4. THE TURBINE CASING IS PROTECTED FROM OVERPRESSURE BY TWO RUPTURE DISCS. PS-71-11A THRU D SET AT 10 PSIG DETECTS DISC RUPTURES AND ANNUNCIATES ON PA-71-11 IN THE MAIN CONTROL ROOM.
5. THE RCIC TURBINE WILL BE AUTOMATICALLY ISOLATED FROM THE STEAM SUPPLY BY CLOSING ISOLATION VALVES FCV-71-2 AND FCV-71-3 IF:
   A. A HIGH PRESSURE DROP OCCURS ACROSS THE RCIC ELBOW TYPE FLOW ELEMENT FE-71-1A & 1B WHEN THE STEAM LINE RUPTURES. THE HIGH FLOW IS SENSED BY EITHER PD1S-71-1A OR PD1S-71-1B AND ANNUNCIATED ON PDA-71-1.
   B. HIGH AREA TEMPERATURES INDICATING A STEAM LINE BREAK ARE DETECTED BY ANY OF FOUR SETS OF TEMPERATURE SWITCHES (TS-71-2A THROUGH TS-71-2S SET AT 194°F). EACH SET CONNECTED IN ONE-OUT-OF-TWO-TWICE LOGIC. FOUR INDEPENDENT TEMPERATURE SENSING ELEMENTS AND ASSOCIATED TEMPERATURE SWITCHES TE-71-41A AND TS-71-41A THROUGH TE-71-41D AND TS-71-41D SET AT 175 F ACTIVATE THE MAIN CONTROL ROOM ANNUNCIATOR TA-71-41.
   C. LOW REACTOR PRESSURE IS DETECTED BY PS-71-1A THROUGH PS-71-1D EACH SET AT 50 PSIG AND ARRANGED IN ONE-OUT-OF-TWO-TWICE LOGIC.
6. NORMALLY MAKEUP WATER FOR THE RCIC PUMP SUCTION IS SUPPLIED BY THE CONDENSATE STORAGE TANK THROUGH FLOW CONTROL VALVE FCV-71-19. AN ALTERNATE SOURCE OF WATER IS OBTAINED FROM THE SUPPRESSION POOL THROUGH ISOLATION VALVES FCV-71-17 AND FCV-71-18. PUMP DISCHARGE FLOW ROUTES ARE AS FOLLOWS:
   A. NORMALLY VIA THE FEEDWATER LINE TO THE REACTOR THROUGH ISOLATION VALVES FCV-71-37, FCV-71-39, AND CHECK VALVE FCV-71-40.
   B. DURING TESTING VIA TEST LINE TO CONDENSATE STORAGE THROUGH FLOW CONTROL VALVE FCV-71-38.
   C. MINIMUM FLOW BYPASS TO THE SUPPRESSION POOL THROUGH FLOW CONTROL VALVE FCV-71-34.
7. IF RCIC SYSTEM OPERATION IS REQUIRED WHEN IT IS IN THE TEST MODE, THE CONTROL SYSTEM IS DESIGNED TO AUTOMATICALLY RETURN THE SYSTEM TO THE OPERATING MODE.
8. COOLING WATER FOR THE PUMP, TURBINE, AND LUBE OIL COOLER IS SUPPLIED FROM THE PUMP DISCHARGE LINE THROUGH PRESSURE CONTROL VALVE PCV-71-22 SET AT 75 PSIA.
9. TRANSFER AND CONTROL SWITCHES ASSOCIATED WITH BACKUP CONTROL WILL BE ON THE DESIGNATED PANELS OR LOCATED ON THE APPROPRIATE 250V DC MOV BOARDS.
10. XS-71-36B IS SHOWN THREE TIMES TO SHOW HOW THIS TRANSFER SWITCH DISCONNECTS FIC-71-36A AND CONNECTS FIC-71-36B LOCATED AT THE BACKUP CONTROL CENTER.
11. XS-71-23, XS-71-45, TS-71-45 & -46, XS-71-24 AND PS-71-44 ARE EACH SHOWN TWICE TO SHOW HOW THESE SWITCHES TRANSFER INDICATION FROM THE MAIN CONTROL PANEL 9-3 TO THE BACKUP CONTROL PANEL 25-32.
12. 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

13. FLEX HOSE ( ) TO BE INSTALLED FOR TESTING PURPOSES ONLY.

REFERENCE DRAWINGS:

MEL............................INSTRUMENT TABULATIONS
3-47E610-1-1A...............CONTROL DIAGRAM-MAIN SYSTEM
0-47E610-2-2.................CONTROL DIAGRAM-CONDENSATE SYSTEM
3-47E610-3-1.................CONTROL DIAGRAM-REACTOR FEEDWATER
3-47E610-69-1...............CONTROL DIAGRAM-REACTOR WATER CLEANUP
2-47E611-71-4...............LOGIC DIAGRAM-RCIC
0-47E800-1....................FLOW DIAGRAM-GENERAL PLANT SYSTEM
0-47E800-2....................MECHANICAL SYMBOLS & FLOW DIAGRAM INDEX
3-47E610-73-1...............HPCI SYSTEM
3-47E813-1....................FLOW DIAGRAM-RCIC SYSTEM
3-45E779-12..................WIRING DIAGRAM-480V SHUTDOWN AUXILIARY POWER SCHEMATIC DIAGRAM
3-45E714-5,-6...............WIRING DIAGRAM-250V MOV BOARD 1C SCHEMATIC DIAGRAM
0-45E626-1,-4...............WIRING DIAGRAM-RCIC SCHEMATIC DIAGRAM
3-45E626-2,-3...............WIRING DIAGRAM-RCIC SCHEMATIC DIAGRAM
0-47E610-12-1...............CONTROL DIAGRAM-AUXILIARY BOILER
3-45E620-2,-3...............KEY DIAGRAM-ANNUNCIATOR SYSTEM
3-47E811-1....................FLOW DIAGRAM-RHR SYSTEM
3-45N3654-1..................WIRING DIAGRAM-UNIT CONTROL BOARDS PANEL 9-21, SH 1
3-45W670-SERIES.............WIRING DIAGRAM ECCS

GE DRAWINGS:

729E652, SH 1 THRU 4....FUNCTIONAL CONTROL DIAGRAMS-RCIC

TERRY TURBINE CO.
C-883-X.................RCIC TURBINE GOVERNOR CONTROL SYSTEM

NASH ENGR CO.
D-14-1365..............RCIC BAROMETRIC CONDENSER

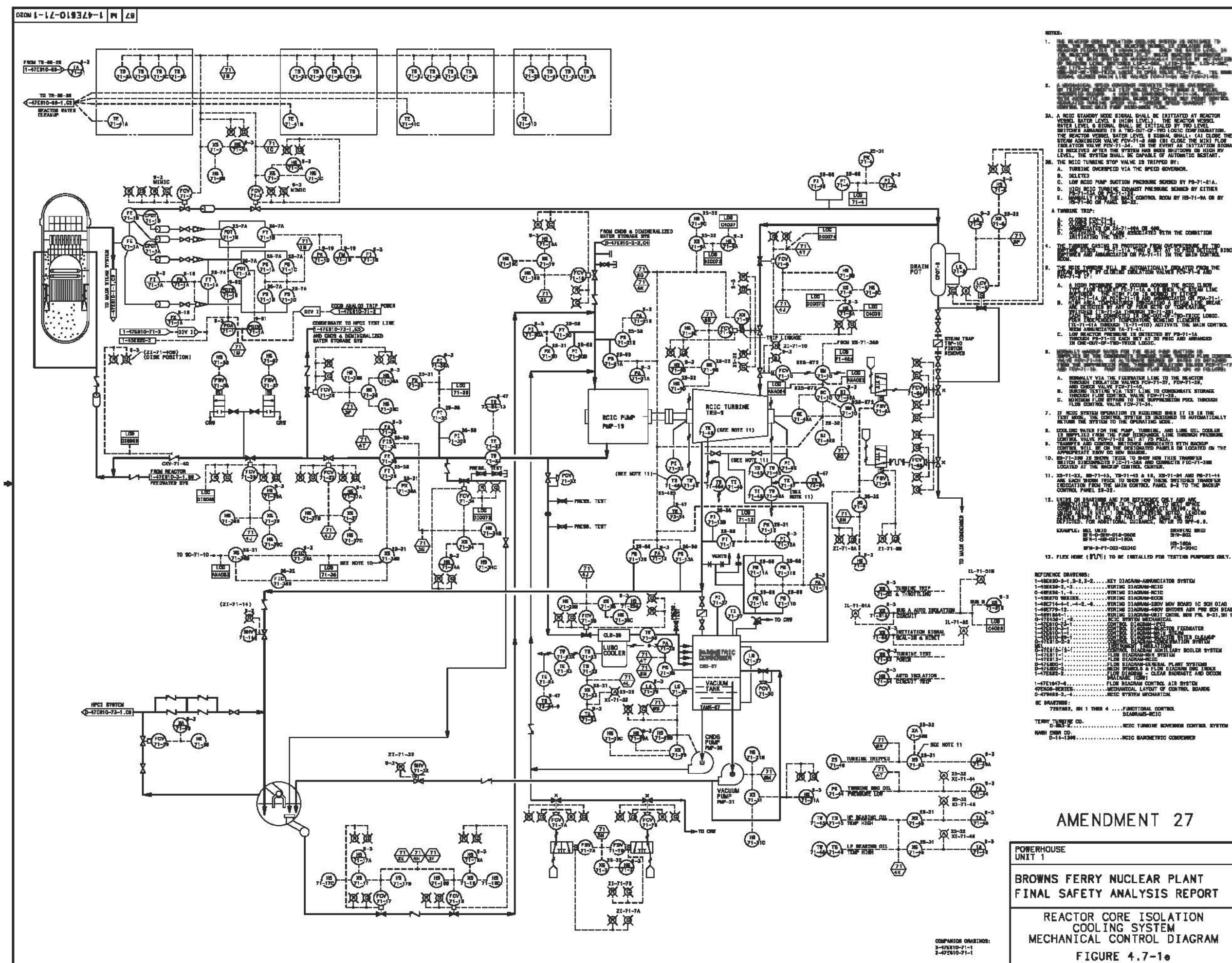
AMENDMENT 26

POWERHOUSE
UNIT 3

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

REACTOR CORE ISOLATION
COOLING SYSTEM
MECHANICAL CONTROL DIAGRAM

FIGURE 4.7-1d



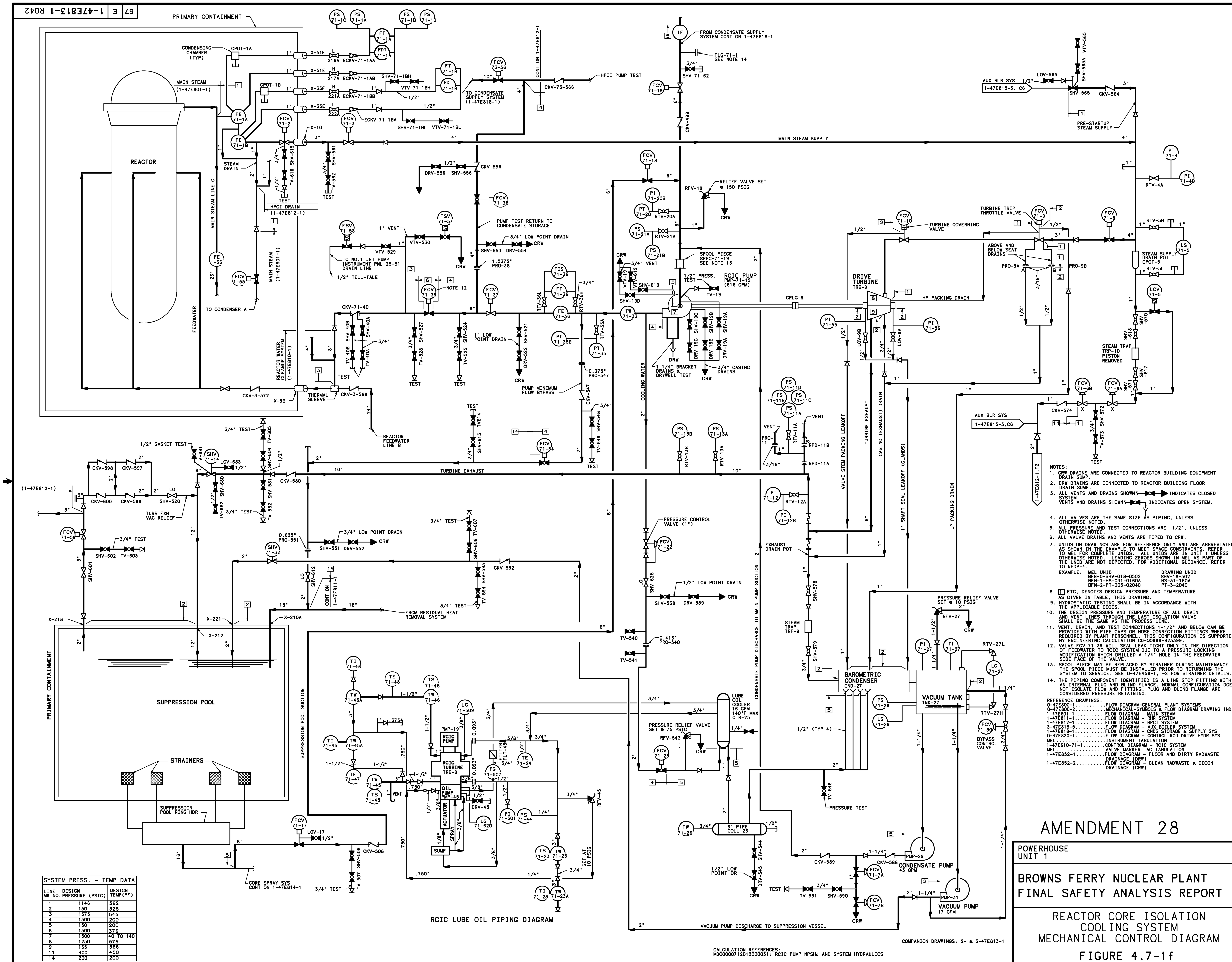
AMENDMENT 27

POWERHOUSE
UNIT 1

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

REACTOR CORE ISOLATION
COOLING SYSTEM
MECHANICAL CONTROL DIAGRAM

FIGURE 4.7-1a



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 INDICATES CLOSED SYSTEM.
   VENTS AND DRAINS SHOWN INDICATES OPEN SYSTEM.
4. ALL VALVES ARE THE SAME SIZE AS PIPING, UNLESS OTHERWISE NOTED.
5. ALL PRESSURE AND TEST CONNECTIONS ARE 1/2", UNLESS OTHERWISE NOTED.
6. ALL VALVE DRAINS AND VENTS ARE PIPED TO CRW.
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 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

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. 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.
12. VALVE FCV-71-39 WILL SEAL LEAK TIGHT ONLY IN THE DIRECTION OF FEEDWATER TO RCIC SYSTEM DUE TO A PRESSURE LOCKING MODIFICATION WHICH DRILLED A 1/4" HOLE IN THE FEEDWATER SIDE FACE OF THE VALVE.
13. SPOOL PIECE MAY BE REPLACED BY STRAINER DURING MAINTENANCE. THE SPOOL PIECE MUST BE INSTALLED PRIOR TO RETURNING THE SYSTEM TO SERVICE. SEE 0-47E456-1, -2 FOR STRAINER DETAILS.
14. THE PIPING COMPONENT IDENTIFIED IS A LINE STOP FITTING WITH AN INTERNAL PLUG AND BLIND FLANGE. NORMAL CONFIGURATION DOES NOT ISOLATE FLOW AND FITTING. PLUG AND BLIND FLANGE ARE CONSIDERED PRESSURE RETAINING.

REFERENCE DRAWINGS:

- 0-47E800-1..........FLOW DIAGRAM-GENERAL PLANT SYSTEMS
- 0-47E800-2..........MECHANICAL-SYMBOLS & FLOW DIAGRAM DRAWING INDEX
- 1-47E801-1..........FLOW DIAGRAM - MAIN STEAM
- 1-47E811-1..........FLOW DIAGRAM - RHR SYSTEM
- 1-47E812-1..........FLOW DIAGRAM - HPCI SYSTEM
- 3-47E815-5..........FLOW DIAGRAM - AUX BOILER SYSTEM
- 1-47E818-1..........FLOW DIAGRAM - CNDS STORAGE & SUPPLY SYS
- 0-47E820-1..........FLOW DIAGRAM - CONTROL ROD DRIVE HYDR SYS
- MEL.................INSTRUMENT TABULATION
- 1-47E610-71-1.......CONTROL DIAGRAM - RCIC SYSTEM
- MEL.................VALVE MARKER TAG TABULATION
- 1-47E852-1..........FLOW DIAGRAM - FLOOR AND DIRTY RADWASTE DRAINAGE (DRW)
- 1-47E852-2..........FLOW DIAGRAM - CLEAN RADWASTE & DECON DRAINAGE (CRW)

SYSTEM PRESS. - TEMP DATA

LINE MK NO.	DESIGN PRESSURE (PSIG)	DESIGN TEMP(°F)
1	1146	562
2	150	325
3	1375	545
4	1500	200
5	150	200
6	1500	376
7	1500	40 TO 140
8	1250	575
9	165	366
11	400	450
14	200	200

AMENDMENT 28

POWERHOUSE
UNIT 1

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

REACTOR CORE ISOLATION COOLING SYSTEM
MECHANICAL CONTROL DIAGRAM

FIGURE 4.7-1f

Figures 4.7-2a through 4.7-2h
(Deleted by Amendment 22)

4.8 RESIDUAL HEAT REMOVAL SYSTEM (RHRS)

4.8.1 Safety Objective

The safety objectives of the Residual Heat Removal System (RHRS) are as follows:

a. To restore and maintain the coolant inventory in the reactor vessel so that the core is adequately cooled after a loss-of-coolant accident. The Residual Heat Removal System also provides cooling for the pressure suppression pool so that condensation of the steam resulting from the blowdown due to the design basis loss-of-coolant accident is ensured.

b. The Residual Heat Removal System further extends the redundancy of the Core Standby Cooling Systems by providing for containment cooling.

4.8.2 Power Generation Objective

The Residual Heat Removal System provides the means to meet the following power generation objectives:

a. Remove decay heat and residual heat from the nuclear system so that refueling and nuclear system servicing can be performed.

b. Supplement the Fuel Pool Cooling and Cleanup System capacity when necessary to provide additional pool cooling capacity.

4.8.3 Safety Design Basis

1. The RHRS shall act automatically (except when in the shutdown cooling mode), in combination with other Core Standby Cooling Systems, to restore and maintain the coolant inventory in the reactor vessel such that the core is adequately cooled to preclude fuel clad temperatures in excess of 2200°F following a design basis loss-of-coolant accident.

2. The RHRS, in conjunction with other Core Standby Cooling Systems, shall have such diversity and redundancy that only a highly improbable combination of events could result in their failure to provide adequate core cooling.

3. The source of water for restoration of reactor vessel coolant inventory shall be located within the primary containment in such a manner that a closed cooling water path is established.

4. To provide a high degree of assurance that the RHRS operates satisfactorily during a loss-of-coolant accident, each active component shall be capable of being tested during operation of the nuclear system. The inboard isolation check valve can only be tested during cold shutdown (MODE 4 or MODE 5).

5. To provide an additional source of water for post-accident containment flooding a crosstie shall be provided between the RHR Service Water System and RHRS. (This long term capability is not credited in the mitigation of design basis accidents and does not perform an active safety related function.)

4.8.4 Power Generation Design Basis

1. The RHRS shall be designed with enough capacity that the service water outlet temperature can be limited during shutdown conditions to minimize fouling.

4.8.5 Summary Description

The RHRS is designed for five modes of operation to satisfy all the objectives and bases:

1. Shutdown cooling (Units 1, 2, and 3),

2. Containment spray and pool cooling,

3. Low pressure coolant injection, and

4. Standby cooling.

5. Supplemental fuel pool cooling.

To provide clarity to the information presented herein, each mode of operation is defined as a subsystem of the RHRS and is discussed separately. It is shown how each subsystem contributes toward satisfying all the objectives and bases of the RHRS.

The major equipment of the RHRS consists of four heat exchangers and four pumps for each unit. There are twelve RHR service water pumps for the plant (see Section 10.9, "RHR Service Water System"), eight of which can be used for RHRSW purposes. The equipment is connected by associated valves and piping, and the controls and instrumentation are provided for proper system operation. A process diagram of the RHRS is shown in Figures 7.4-6a sheets 1, 2, and 3 of Section 7.4.

A description of the controls and instrumentation is presented in Section 7.4, "Core Standby Cooling Control and Instrumentation." A description of how operation of the equipment in the RHRS in conjunction with other Core Standby Cooling Systems protects the core in case of a loss-of-coolant accident is presented in Chapter 6.0, "Core Standby Cooling Systems."

The RHRS pumps are sized on the basis of the flow required during the low pressure coolant injection (LPCI) mode of operation, which is the mode requiring the maximum flow rate. In addition, the system pumps are equipped with discharge flow limiting orifice plates to prevent pump operation in "runout" conditions and to prevent any damage that might occur in the case of a recirculation line break. The heat exchangers are sized on the basis of their required duty for the pressure suppression pool cooling function. It is concluded that the power generation design objective is met. A summary of the design requirements of the RHRS pumps and the heat exchangers is presented in Table 4.8-1. See Section 6.5 for system requirements utilized in the Emergency Core Cooling System analysis.

Permanent connections with normally closed valves are provided on the shutdown cooling piping circuit for supplying cooling water to the Fuel Pool Cooling and Cleanup System (see Figures 7.4-6a sheets 1, 2, and 3). This permits the RHRS heat exchangers to be used to assist fuel pool cooling when required (see Section 10.5, "Fuel Pool Cooling and Cleanup System").

One of the RHRS loops, consisting of two heat exchangers, two pumps in parallel, and associated piping, is located in one area of the Reactor Building. The other heat exchangers, pumps, and piping, forming a second loop, are located in another area of the Reactor Building to minimize the possibility of a single physical event causing the loss of the entire system. This arrangement satisfies the safety design basis 2. In addition, the pump suction and heat exchanger discharge lines of one loop in Unit 1 (Loop II) are cross-connected to the pump suction and heat exchanger discharge lines of one loop in Unit 2. Unit 2 and Unit 3 systems are cross-connected in a similar manner. Two normally closed isolation valves are provided in each heat exchanger discharge cross-connection, and four normally closed isolation valves are provided in each suction cross-connection (one at each pump suction), as shown in Figure 4.8-1.

RHRS equipment is designed in accordance with Class I seismic criteria (see Appendix C) to resist sufficiently the response motion at the installed location within the supporting building from the Design Basis Earthquake.

The system piping and pumps are designed in accordance with the requirements of USAS B31.1.0, 1967 edition, as augmented by GE specifications which were implemented in lieu of the outdated B31 Nuclear Code Cases-N2, N7, N9, and N10. The system is constructed and tested in accordance with TVA construction

specifications. The pumps are also designed and constructed in accordance with the standards of the Hydraulic Institute. The shell side of the heat exchangers is designed in accordance with the ASME Boiler and Pressure Vessel Code, 1965 edition, Section III, Class C vessels, and TEMA Class C; and the tube side is designed in accordance with Section VIII and TEMA Class C. The provisions of the ASME Boiler and Pressure Vessel Code, Section III, Winter Addenda of 1966, paragraph N2113, apply.

4.8.6 Description

4.8.6.1 Shutdown Cooling

The shutdown cooling subsystem is an integral part of the RHRS and is placed in operation during a normal shutdown and cooldown. The initial phase of nuclear system cooldown is accomplished by dumping steam from the reactor vessel to the main condenser with the main condenser acting as the heat sink. The RHRS is typically placed in the shutdown cooling mode of operation when reactor vessel pressure has decreased sufficiently to clear the interlocks associated with the shutdown cooling suction valves. The shutdown cooling subsystem alone is capable of completing cooldown to 125°F in less than 34 hours and maintaining the nuclear system at 125°F so that the reactor can be refueled and serviced.

Reactor coolant is pumped by the RHRS pumps from one of the recirculation loops through the RHRS heat exchangers, where cooling takes place by transferring heat to the RHR service water system. Reactor coolant is returned to the reactor vessel via either recirculation loop.

During a nuclear system shutdown and cooldown, any one of the four RHR shutdown cooling subsystems can provide the required decay heat removal function and maintain or reduce the reactor coolant temperature as required.

The RHRS is normally flushed with water of condensate quality or better in preparation for shutdown cooling operation during the steam dumping phase of plant cooldown. This flush is not required if 1) there is an immediate need for RHR shutdown cooling to control reactor vessel level, temperature, or pressure, or 2) RHR shutdown cooling is removed from and returned to service during an outage and no activities have occurred which could result in water quality degradation below acceptable limits for reactor vessel injection.

4.8.6.2 Containment Cooling

The containment cooling subsystem is an integral part of the RHRS and is placed in operation to limit the temperature of the water in the pressure suppression pool so that immediately after the design basis loss-of-coolant accident (guillotine break of a recirculation system suction line) has occurred, the maximum bulk pool temperature does not exceed 179ºF. The maximum permissible bulk pool temperature is limited by the potential for stable and complete condensation of steam discharged from the main steam relief valves as well as the design analyses of the torus attached piping (see Sections 5.2.3.3.2 and 5.2.4.3).

With the RHRS in the suppression pool cooling mode of operation, the RHRS pumps are aligned to pump water from the pressure suppression pool through the RHRS heat exchangers where cooling takes place by transferring heat to the RHR service water. For adequate containment cooling, a minimum of two RHR pumps and associated heat exchangers must remain available for several hours after a design basis loss-of-coolant accident. The flow returns to the pressure suppression pool via the flow test line (see Figures 7.4-6a sheets 1, 2, and 3). Pressure suppression pool temperature operational limits are provided in Technical Specification, Section 3.6.2.1.

The pressure suppression pool cooling mode of RHRS is initiated to restore pressure suppression pool temperature to within allowable limits during plant operation. IN 87-10 Supplement 1 identifies the potential for the RHRS to be damaged and unable to perform its Low Pressure Coolant Injection function should a LOCA and LOOP occur while RHRS is in the SPC mode of operation. The safety design basis for RHRS requires that only a highly improbable combination of events can result in RHRS being rendered unable to perform its core cooling function (see Section 4.8.3). To meet this requirements, PRA analyses have established a time limit for RHRS operation in the SPC mode and the time RHRS is in SPC mode is tracked to ensure this time limit is not exceeded.

The containment spray cooling mode of operation provides additional redundancy to the Core Standby Cooling Systems for post-accident conditions. The water pumped through the RHRS heat exchangers may be diverted to spray headers in the drywell and above the pressure suppression pool. The spray headers in the drywell condense any steam that may exist in the drywell, thereby lowering containment pressure. The spray collects in the bottom of the drywell until the water level rises to the level of the pressure suppression vent lines, where it overflows and drains back to the pressure suppression pool. Approximately 5 percent of this flow may be directed to the pressure suppression chamber spray ring to cool any noncondensable gases collected in the free volume above the pressure suppression pool.

The spray headers of the RHRS cannot be placed in operation unless the core cooling requirements of the low pressure coolant injection subsystem have been satisfied. These requirements may be bypassed by the operator using a keylock switch in the control room (see Section 7.4, "Core Standby Cooling Control and Instrumentation").

4.8.6.3 Low Pressure Coolant Injection

The low pressure coolant injection (LPCI) subsystem is an integral part of the RHRS. It operates to restore and, if necessary, maintain the coolant inventory in the reactor vessel after a loss-of-coolant accident so that the core is sufficiently cooled to preclude fuel clad temperatures in excess of 2200°F and subsequent energy release due to a metal-water reaction. A detailed discussion of the requirements and response of the equipment which operates during LPCI for a loss-of-coolant accident may be found in Chapter 6.0, "Core Standby Cooling Systems." A detailed discussion of the requirements and response of the controls and instrumentation of LPCI during a loss-of-coolant accident may be found in Section 7.4, "Core Standby Cooling Control and Instrumentation."

In general, LPCI operation involves restoring the water level in the reactor vessel to a sufficient height for adequate cooling after a loss-of-coolant accident. The LPCI subsystem operates in conjunction with the High Pressure Coolant Injection System (HPCIS), the Auto Depressurization System and the Core Spray System to achieve this goal (see Chapter 6.0, "Core Standby Cooling Systems"). This capability satisfies safety design basis 1.

The HPCIS is a high-head, low-flow system and pumps water into the reactor vessel when the nuclear system is at high pressure. If the HPCIS fails to maintain the required level of water in the reactor vessel, the automatic depressurization feature of the Nuclear System Pressure Relief System functions to reduce nuclear system pressure so that LPCI operates to inject water into the pressure vessel. LPCI is a low-head, high-flow subsystem and delivers rated flow of ≥ 9000 gpm for each pump to the reactor vessel against an indicated pressure of ≥ 125 psig. All these operations are carried out automatically. LPCI is designed to reflood the reactor vessel to at least two-thirds core height and to maintain this level. After the core has been flooded to this height, the capacity of one RHR pump is more than sufficient to maintain the level.

During LPCI operation, the RHRS pumps take suction from the pressure suppression pool and discharge to the reactor vessel into the core region through both of the recirculation loops. Two pumps discharge to each injection header, assuring flooding of the vessel through at least one loop. Any spillage through a break in the lines within the primary containment returns to the pressure suppression

pool through the pressure suppression vent lines. A bypass line to the pressure suppression pool is provided so that the pumps are not damaged if operating with the discharge valves shut.

Added resistance in the pump discharge lines prevents insufficient NPSH in the LPCI mode of operation. It is concluded that safety design basis 3 is satisfied.

Service water flow to the RHRS heat exchangers is not required immediately after a loss-of-coolant accident because heat rejection from the containment is not necessary during the time it takes to flood the reactor.

Power for the RHRS pumps and the RHR service water pumps comes from the 4-kV AC power shutdown boards. Power for these boards normally comes from the auxiliary supply, but if this source is not available, power is available from the standby (diesel) AC power source.

4.8.6.4 Standby Cooling

Standby coolant supply connection and RHR crossties are provided to maintain a long-term reactor core and primary containment cooling capability irrespective of primary containment integrity or operability of the Residual Heat Removal System associated with a given unit. The standby coolant supply connection and RHR crossties provide added long-term redundancy to the other emergency core and containment cooling systems and are designed to accommodate certain situations which, although unlikely to occur, could jeopardize the functioning of these systems.

By proper valve alignment (see Figure 4.8-1), the network created by the RHR crossties permits the B (or D) RHR pumps on Unit 1 to circulate Unit 2 pressure suppression pool or reactor vessel water through the B (or D) heat exchangers on Unit 1 in the unlikely event that the Unit 2 RHR pumps are unavailable. The crosstie network is sized for a minimum flow of 5,000 gpm, which will achieve about 91 percent of full flow heat transfer capability of the RHR heat exchangers.

In a like fashion, the A (or C) RHR pumps on Unit 2 can be used to circulate Unit 1 pressure suppression pool or reactor vessel water through the A (or C) heat exchangers on Unit 2. The B (or D) RHR pumps on Unit 2 and the A (or C) RHR pumps on Unit 3 can be similarly utilized.

Pressure suppression pool water which has been circulated through the RHR heat exchangers on one unit can be used to flood the reactor core, spray the drywell and pressure suppression chamber, or returned to the pressure suppression chamber of the adjacent unit. In this way, decay heat and residual heat can be removed from the reactor core and primary containment of the adjacent unit on a long-term basis. By proper valve alignment (see Figure 4.8-1), the network created by the standby

coolant supply connection and RHR crossties permits the D2 (or D1) RHR service water pump and header to supply raw water directly to the reactor core of Units 1 or 2 as the reactor pressure approaches 50 psig. The service water pump and header can also be valved to supply raw water to the drywell or pressure suppression chamber spray headers or directly to the pressure suppression chamber of either unit. In a similar fashion, the B2 (or B1) RHR service water pump and header can supply raw water to the reactor core of Units 2 or 3 or into the respective drywell/pressure suppression chamber spray headers or directly to the pressure suppression chambers.

The Standby Coolant Supply System is sized to supply a minimum raw water flow of 3,250 gpm, against a reactor pressure of 65 psig with a drywell pressure of 15 psig. It is concluded that safety design basis 5 is satisfied.

4.8.6.5 Supplemental Fuel Pool Cooling

A description of how the RHRS heat exchangers can be used to assist fuel pool cooling when required is contained in Section 10.5, “Fuel Pool Cooling and Cleanup System.”

4.8.7 Safety Evaluation

Since the LPCI and containment cooling subsystems act with other Core Standby Cooling Systems to satisfy the safety objective, they are properly evaluated in conjunction with the other Core Standby Cooling Systems. This safety evaluation is in Chapter 6.0, "Core Standby Cooling Systems." The safety evaluation of the controls and instrumentation of the LPCI subsystem is in Section 7.4, "Core Standby Cooling Control and Instrumentation."

4.8.8 Inspection and Testing

A design flow functional test of the RHRS pumps is performed during normal plant operation by taking suction from the pressure suppression pool and discharging through the test lines back to the pressure suppression pool. The discharge valves to the reactor recirculation loops remain closed during this test and reactor operation is undisturbed.

An operational test of these discharge valves is performed by shutting the downstream valve after it has been satisfactorily tested and then operating the upstream valve. The discharge valves to the containment spray headers are checked in a similar manner by operating the upstream and downstream valves individually. All these valves can be actuated from the control room using remote manual switches. Control system design provides automatic return from test to

operating mode if LPCI initiation is required during testing. It is concluded that safety design basis 4 is satisfied.

Periodic inspection and maintenance of the RHRS pumps, pump motors, valves and valve motors, and heat exchangers are based on manufacturer's recommendations and sound maintenance practices.

A discussion of the availability of engineered safeguards and frequency of testing of equipment is presented in Chapter 6.0, "Core Standby Cooling Systems."

4.8.8.1 RHR Heat Exchanger Performance Monitoring Program Requirements

To ensure that the RHR heat exchangers are maintained in a condition that meets or exceeds the minimum performance capability assumed in the containment analyses, which support not taking credit for containment accident pressure in the NPSH analyses, the following program requirements are established. These program requirements are established to satisfy Technical Specification 5.5.1.4, Residual Heat Removal (RHR) Heat Exchanger Performance Monitoring Program.

1. The DBA-LOCA minimum required heat removal rate is 80,136,000 Btu/hour (hr) per heat exchanger with two heat exchangers in service. The EPU fire event minimum required heat removal rate is 124,966,800 Btu/hr with one heat exchanger in service.

2. The thermal performance test acceptance criteria for an RHR heat exchanger is less than or equal to 0.001562 hr-ft^2-F/Btu with no more than 77 tubes (4.57% of 1700 tubes) plugged.

3. The nominal (measured) test result (fouling resistance) including the test and measurement uncertainty will be used for comparison to the thermal performance acceptance criteria.

4. The program includes the following requirements:

   a. Each RHR heat exchanger is performance tested at a nominal interval of four years, not to exceed five years.

   b. The cooling water side of the RHR heat exchangers is inspected and cleaned periodically as determined by the preventative maintenance (PM) program. The maximum interval for performing RHR heat exchanger inspection and cleaning is limited to five years (four years + 25%). Any increase in the inspect and clean interval beyond five years will be evaluated in accordance with PM program procedures, GL 89-13 program implementing procedures and TVA programmatic procedure change

requirements. Inspection results and performance testing results will be used to technically justify extending the inspect and clean interval. The as-found inspection acceptance criteria is less than 77 tubes obstructed (sum of the number of tubes mechanically plugged and the number of tubes obstructed by macrofouling).

5. The following aspects of the RHR Heat Exchanger performance monitoring program meet the guidance provided in EPRI 3002005340, Service Water Heat Exchanger Test Guidelines, May 2015:

    a. The Heat Transfer Test Method will be used
    b. Temporary surface mounted temperature instrumentation
    c. Temporary differential pressure (DP) instrumentation
    d. Temporary data acquisition system (DAS) including the associated software
    e. Test data analysis - the analysis determines the overall fouling resistance for the heat exchanger and also determines the associated uncertainty in the test result (fouling resistance)
    f. The uncertainty analysis methodology
    g. Data reduction

6. Temporary instruments are calibrated against standards traceable to the National Institute of Standards and Technology or compared to nationally or internationally recognized consensus standards.

7. Computer programs used in the thermal performance analysis are required to meet the 10 CFR 50 Appendix B, and 10 CFR 21 requirements. PROTO-HX is the computer program used for thermal performance analyses in the RHR Heat Exchanger Performance Monitoring Program.

8. Compensatory measures include entering the condition into the TVA corrective action program, cleaning of the heat exchangers after inspections, determining if more frequent inspections and cleaning are required, and evaluating past operability/functionality when tube plugging and macrofouling acceptance criteria from inspection procedures are not met. The methodology for performing as-found and as-left inspections are provided in TVA Standard Programs and Processes procedures. As-left inspections are procedurally required and ensure that tubes found blocked/obstructed by macrofouling will not be left in the as-found condition.

9. Changes to the program requirements above will be controlled in accordance with 10 CFR 50.59, “Changes, tests, and experiments.” Change to the program requirements may be made without prior NRC approval provided the changes do not require a change to the Technical Specification requirements and the changes do not require NRC approval pursuant to 10 CFR 50.59.

TABLE 4.8-1

RESIDUAL HEAT REMOVAL SYSTEM EQUIPMENT DESIGN DATA

RHRS PUMPS

Number Installed per Unit - 4	Design Temperature - 350°F
Capacity/Pump - 50% (LPCI)	Design pressure - 450 psig
	Shutoff Head - 780 ft

Design Conditions/Pump 0 psid*	
Discharge Flow (gpm)	20,000 (2 in one loop) 10,800 (1 in one loop) Units 1, 2, 3
Rated Pump Capacity	10,000 gpm at 560 ft Total Dynamic Head
NPSH Required at 90°F (ft)	30
Operating Conditions/Pump	
Discharge Flow (gpm)	0-12,000
Discharge Head (ft)	780-420
Differential Pressure (psid)	295-0
NPSH Required at 90°F (ft)	30-34

RHRS HEAT EXCHANGERS

Number Installed per Unit - 4
Shell Side Fluid - Reactor Water or Pressure Suppression Pool Water
Tube Side Fluid - RHR Service Water (River Water)
Shell and Tube Side Design Pressure - 450 psig and Design Temperature 40-350°F
Pressure Drop Design Conditions - shell side 10 psi
- tube side 6 psi

Suppression Pool Cooling Analysis (3952 MWt) - ANS/ANSI 5.1 (with 2σ uncertainty)

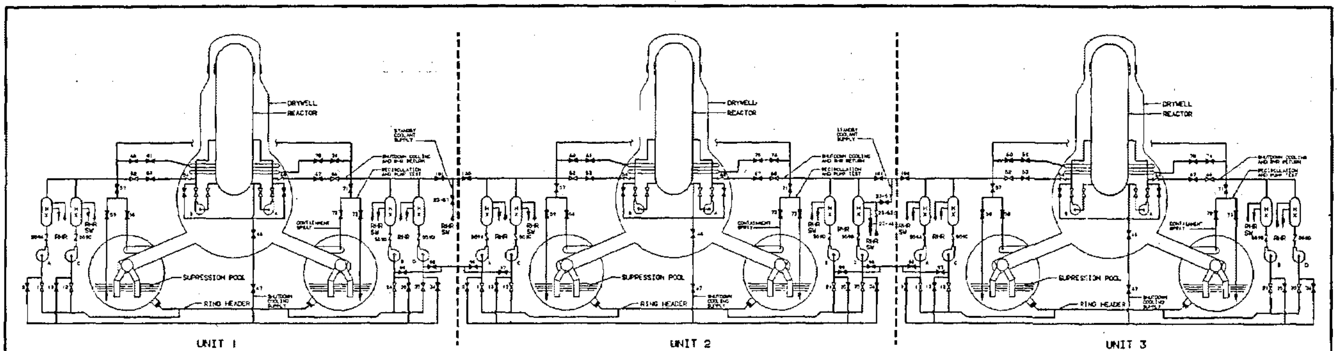
DBA LOCA Analysis:

Shell Side Flow (gpm)	6500
Inlet Temperature Shell Side (°F)	179.0
Heat Exchanger Duty (Btu/hr)	80,136,000
Tube Side Flow (gpm)	4000
Inlet Temperature Tube Side (°F)	95ºF
Heat Exchanger K-factor	265 BTU/sec-ºF

Limiting NFPA 805 (fire) event

Shell Side Flow (gpm)	7500
Inlet Temperature Shell Side (°F)	207.7
Heat Exchanger Duty (Btu/hr)	124,966,800
Tube Side Flow (gpm)	4500
Inlet Temperature Tube Side (°F)	88.0ºF
Heat Exchanger K-factor	290 BTU/sec-ºF

*psid - pounds per square inch difference between reactor vessel and drywell.



POWERHOUSE
UNITS 1,2 & 3

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

RESIDUAL HEAT REMOVAL SYSTEM
UNIT CROSS CONNECTIONS AND
STANDBY COOLANT SUPPLY
FIGURE 4.8-1

AMENDMENT 16

4.9 REACTOR WATER CLEANUP SYSTEM

4.9.1 Power Generation Objective

The Reactor Water Cleanup System maintains high reactor-water purity to limit chemical and corrosive action, thereby limiting fouling and deposition on heat transfer surfaces. The Reactor Water Cleanup System also removes corrosion products to limit impurities available for activation by neutron flux and resultant radiation from deposition of corrosion products. The system also provides a means for removal of reactor water.

4.9.2 Power Generation Design Basis

1. Provision shall be made for the continuous mechanical and chemical filtration and demineralization of reactor water to quality specifications.

2. Provision shall be made for discharge of reactor water at reduced activity during startup and shutdown.

3. Provisions shall be made to limit the heat loss and the fluid loss from the nuclear system.

4.9.3 Description (Figures 4.9-1, 4.9-2, 4.9-3, 4.9-5, 4.9-6, 4.9-7, 4.9-8, 4.9-9, and 4.9-10)

The Reactor Water Cleanup System provides continuous purification of a portion of the recirculation flow. The processed fluid is returned to the reactor vessel, to radwaste, or to the main condenser. Regenerative heat exchangers are provided to limit heat loss from the nuclear system. The system can be placed in service at any time during normal reactor operation or shutdown conditions.

The major equipment of the Reactor Water Cleanup System is located in the Reactor Building and consists of two pumps, regenerative and nonregenerative heat exchangers, and two filter/demineralizers with supporting equipment. The entire system is connected by associated valves and piping, and controls and instrumentation are provided for proper system operation.

Design and construction of pressure-retaining piping and components of the Reactor Water Cleanup System was initially in accordance with the requirements of USAS B31.1.0, 1967 Edition, as supplemented by the requirements of the applicable GE specifications, which were implemented in lieu of the outdated B31 Nuclear Code Cases-N2, N7, N9, and N10. Design data for the major pieces of equipment are presented in Table 4.9-1.

Reactor coolant is continuously removed from the Reactor Coolant Recirculation System, cooled in the regenerative and nonregenerative heat exchangers, filtered and demineralized, and returned to the feedwater system through the shell side of the regenerative heat exchanger. The RWCU System has one return line through reactor feedwater line B.

Because the ion exchange resins used in the filter-demineralizer are temperature-limited (Table 4.9-1), the reactor coolant must be cooled prior to processing in the units. The regenerative heat exchanger transfers heat from the influent water to the effluent water. The effluent returns to the feedwater system. The nonregenerative heat exchanger cools the influent water further by transferring heat to the Reactor Building Closed Cooling Water System. During startup and shutdown, excess water in the primary system is sent to the main condenser or to radwaste by diverting part or all of the filter-demineralizer effluent. This reduces the effectiveness of the regenerative heat exchanger. The nonregenerative heat exchanger has the capability of reducing the filter-demineralizer influent temperature to the required level, while maintaining an adequate diversion flow rate.

The filter-demineralizer units (Figures 4.9-2, 4.9-3, 4.9-6, 4.9-7, 4.9-9, and 4.9-10) are pressure precoat-type filters which use either finely ground mixed ion exchange resins or a mixture of powdered resins and fibrous material as a precoat medium, they serve as a combination filter-ion exchange medium. Spent resins are not regenerable, but are sluiced from a filter-demineralizer unit to a backwash receiver tank, (from which they are pumped to the cleanup phase separators for dewatering, decay, and disposal). A strainer is installed on the outlet of each filter-demineralizer unit to prevent resins from entering the Reactor Coolant Recirculation System in the event of a resin support failure. Each strainer is provided with an alarm which is energized by high differential pressure (20 psi). A bypass line is provided around the filter-demineralizer units for bypassing the units when necessary. Each unit has a holding pump which starts in the event of low flow, in order to hold the resin in place on the support elements.

Relief valves and instrumentation are provided to protect the equipment against overpressurization and the resins against overheating. The system is automatically isolated when signaled by any of the following occurrences.

a. High temperature downstream of the nonregenerative heat exchanger. To protect the ion exchange resin from damage due to high temperature.

b. Reactor Vessel Water Level - Low, Level 3. To protect the core in case of a possible break in the Reactor Water Cleanup System piping and equipment (see Subsection 7.3, "Primary Containment and Reactor Vessel Isolation Control System").

c. Standby Liquid Control System actuation. To prevent removal of the boron by the ion exchange resin.

d. High temperature indicative of a RWCU pipe break/critical crack in any of the following areas: main steam valve vault, RWCU pipe trench, RWCU pump rooms, or the RWCU heat exchanger room to isolate the system (see subsection 7.3).

Sample points are provided upstream and downstream of each filter-demineralizer unit. The sample analysis provides an indication of the effectiveness of the filter-demineralizer units. The influent sample point is also used as the normal source of reactor coolant samples for analysis of coolant system activity required by Technical Specifications, Section 3.4.6 and for coolant chemistry requirements specified in Section 3.4.1 of the Technical Requirements Manual. Reactor Coolant System activity analysis includes a determination of dose equivalent I-131 concentration which includes quantitative measurements for I-131, I-132, I-133, I-134, and I-135.

Operation of the Reactor Water Cleanup System is controlled from the Main Control Room. Resin-changing operations, which include backwashing and precoating, are controlled from a local control panel in the Reactor Building.

4.9.4 Inspection and Testing

The Reactor Water Cleanup System is normally in service. Satisfactory operation is demonstrated without the need for special testing. Periodic inspection and maintenance are carried out based on manufacturer's recommendations and sound maintenance practices.

TABLE 4.9-1

REACTOR WATER CLEANUP SYSTEM EQUIPMENT DESIGN DATA

MAIN CLEANUP RECIRCULATION PUMPS

Number Required:	2	Design Temperature (°F):	150
Capacity (each):	50%	Design Pressure (psig):	1300
Discharge Flow (gpm/pump):	180	Discharge Head at Rated Flow (ft):	500

HEAT EXCHANGERS

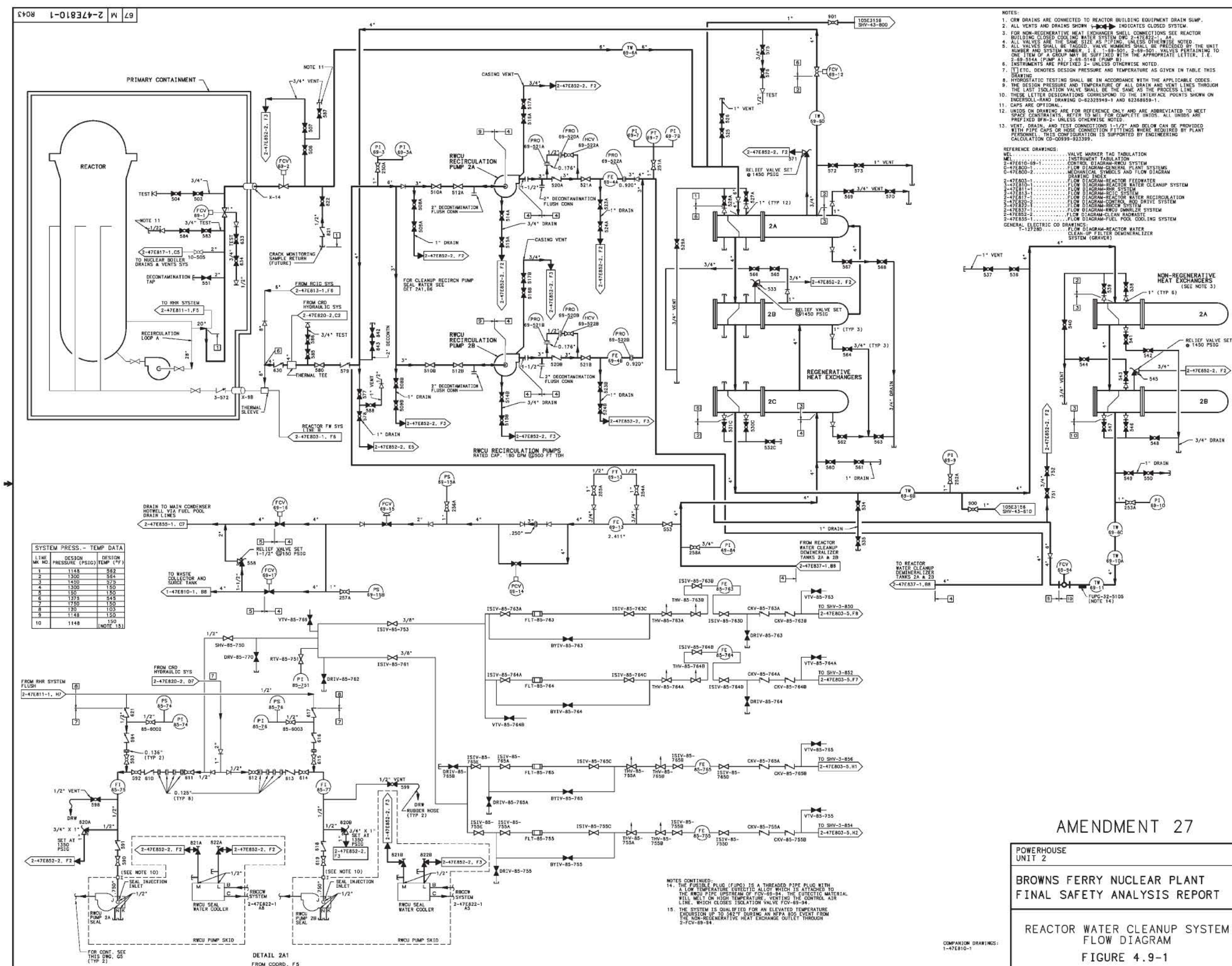
	Regenerative	Nonregenerative
Reactor Coolant Flow Rate (lb/hr)	133,000/187,530*	133,000/187,530*
Shell Side Pressure (psig)	1,450	150
Shell Side Temperature (°F)	575	370
Tube Side Pressure (psig)	1,450	1,450
Tube Side Temperature (°F)	575	575

FILTER-DEMINERALIZERS

Number Required:	2	Design Temperature (°F):	150
Capacity (each):	50%	Design Pressure (psig):	1450
Flow Rate/Unit (lb/hr):	66,650/93,765*		

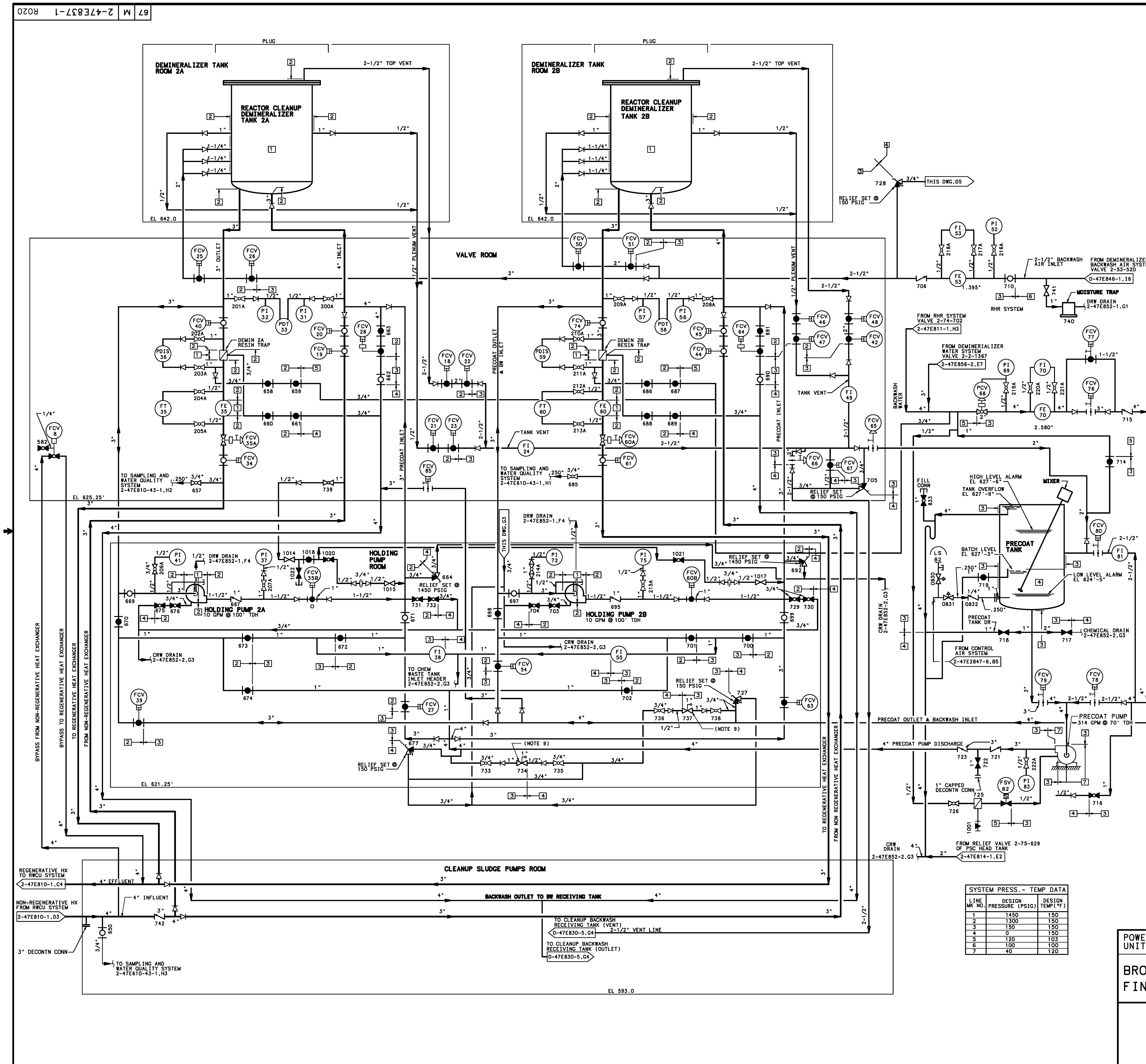
Effluent Conductivity (mmho max):	0.1
Effluent pH:	6.5 to 7.5
Effluent Insolubles (ppb-measured as residue on 0.45 micron filter paper):	<10

*Lower mass flow rate corresponds to the maximum flow of 270 gpm. Higher mass flow rate corresponds to the maximum flow of 340 gpm. Operation of RWCU above 270 gpm depends upon the ability of RBCCW to accommodate the added heat load.



NOTES:
1. CRW DRAINS ARE CONNECTED TO REACTOR BUILDING EQUIPMENT DRAIN SUMP.
2. ALL VENTS AND DRAINS SHOWN INDICATES CLOSED SYSTEM.
3. FOR NON-REGENERATIVE HEAT EXCHANGER SHELL CONNECTIONS SEE REACTOR BUILDING CLOSED COOLING WATER SYSTEM DWG 2-47E822-1, A4.
4. ALL VALVES ARE THE SAME SIZE AS PIPING, UNLESS OTHERWISE NOTED.
5. ALL VALVES SHALL BE TAGGED. VALVE NUMBERS SHALL BE PRECEDED BY THE UNIT NUMBER AND SYSTEM NUMBER, I.E. 1-69-501, 2-69-501. VALVES PERTAINING TO ONE ITEM OF A GROUP MAY BE SUFFIXED WITH THE APPROPRIATE LETTER, I.E. 2-69-514A (PUMP A), 2-69-514B (PUMP B).
6. INSTRUMENTS ARE PREFIXED 2- UNLESS OTHERWISE NOTED.
7. 1 ETC. DENOTES DESIGN PRESSURE AND TEMPERATURE AS GIVEN IN TABLE THIS DRAWING
8. HYDROSTATIC TESTING SHALL BE IN ACCORDANCE WITH THE APPLICABLE CODES.
9. THE DESIGN PRESSURE AND TEMPERATURE OF ALL DRAIN AND VENT LINES THROUGH THE LAST ISOLATION VALVE SHALL BE THE SAME AS THE PROCESS LINE.
10. THESE LETTER DESIGNATIONS CORRESPOND TO THE INTERFACE POINTS SHOWN ON INGERSOLL-RAND DRAWING D-62325949-1 AND 62368659-1.
11. CAPS ARE OPTIONAL.
12. UNIDS ON DRAWING ARE FOR REFERENCE ONLY AND ARE ABBREVIATED TO MEET SPACE CONSTRAINTS. REFER TO MEL FOR COMPLETE UNIDS. ALL UNIDS ARE PREFIXED BFN-2- UNLESS OTHERWISE NOTED.
13. 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
2-47E610-69-1....CONTROL DIAGRAM-RWCU SYSTEM
0-47E800-1....FLOW DIAGRAM-GENERAL PLANT SYSTEMS
0-47E800-2....MECHANICAL SYMBOLS AND FLOW DIAGRAM DRAWING INDEX
2-47E803-1....FLOW DIAGRAM-REACTOR FEEDWATER
3-47E810-1....FLOW DIAGRAM-REACTOR WATER CLEANUP SYSTEM
2-47E811-1....FLOW DIAGRAM-RHR SYSTEM
2-47E813-1....FLOW DIAGRAM-RCIC SYSTEM
2-47E817-1....FLOW DIAGRAM-REACTOR WATER RECIRCULATION
2-47E820-2....FLOW DIAGRAM-CONTROL ROD DRIVE SYSTEM
2-47E822-1....FLOW DIAGRAM-RBCCW SYSTEM
2-47E837-1....FLOW DIAGRAM-RWCU DMNRLZR SYSTEM
2-47E852-2....FLOW DIAGRAM-CLEAN RADWASTE
2-47E855-1....FLOW DIAGRAM-FUEL POOL COOLING SYSTEM
GENERAL ELECTRIC CO DRAWINGS:
T-12728D....FLOW DIAGRAM-REACTOR WATER CLEAN-UP FILTER DEMINERALIZER SYSTEM (GRAVER)
NOTES CONTINUED:
14. THE FUSIBLE PLUG (FUPG) IS A THREADED PIPE PLUG WITH A LOW TEMPERATURE EUTECTIC ALLOY WHICH IS ATTACHED TO THE RWCU PIPE UPSTREAM OF FCV-69-94. THE EUTECTIC MATERIAL WILL MELT ON HIGH TEMPERATURE, VENTING THE CONTROL AIR LINE, WHICH CLOSES ISOLATION VALVE FCV-69-94.
15. THE SYSTEM IS QUALIFIED FOR AN ELEVATED TEMPERATURE EXCURSION UP TO 562°F DURING AN NFPA 805 EVENT FROM THE NON-REGENERATIVE HEAT EXCHANGE OUTLET THROUGH 2-FCV-69-94.
PRIMARY CONTAINMENT
REACTOR
RWCU RECIRCULATION PUMP 2A
RWCU RECIRCULATION PUMP 2B
RWCU RECIRCULATION PUMPS
REGENERATIVE HEAT EXCHANGERS
NON-REGENERATIVE HEAT EXCHANGERS
DETAIL 2A1
COMPANION DRAWINGS: 1-47E810-1
67 M 2-47E810-1 R043
AMENDMENT 27
POWERHOUSE UNIT 2
BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT
REACTOR WATER CLEANUP SYSTEM FLOW DIAGRAM

FIGURE 4.9-1



DEMINERALIZER DATA CLEANUP
DEMINERALIZER RATE (PER UNIT)
DESIGN 340 GPM (MAX)........1.08 GPM PER SQ FT
QUANTITY OF BACKWASH AIR....236 SCFM AT 25 PSIG
BACKWASH WATER..............NORMAL-157 GPM AT 25 PSIG
..............MAXIMUM-236 GPM AT 25 PSIG
PRESSURE....................DESIGN-1300 PSIG
....................OPERATING-1155 PSIG
TEMPERATURE.................DESIGN-150°F
.................OPERATING-130°F

NOTES:
1. FOR DETAILED OPERATING INSTRUCTION, BACKWASH RATES, ETC, SEE MANUFACTURER'S INSTRUCTION MANUAL.
2. ALL VALVES ARE SAME SIZE AS PIPE UNLESS OTHERWISE NOTED.
3. HEAVY LINES SHOW WATER FLOW THROUGH SYSTEM DURING NORMAL DEMINERALIZATION.
4. OPERATIONAL VALVES ARE SHOWN IN THEIR NORMAL OPERATING POSITION.
5. ALL VALVE TAGS SHALL BE PRECEEDED BY THE NUMBER 2-69-, UNLESS OTHERWISE NOTED.
6. 1 ECT, DENOTES DESIGN PRESSURE AND TEMPERATURE AS GIVEN IN TABLE THIS DRAWING.
7. HYDROSTATIC TESTING SHALL BE IN ACCORDANCE WITH THE APPLICABLE CODES.
8. THE DESIGN PRESSURE AND TEMPERATURE OF ALL DRAIN AND VENT LINES THROUGH THE LAST ISOLATION VALVE SHALL BE THE SAME AS THE PROCESS PIPE.
9. THE AIR RELEASE VALVES 69-734 & 737 HAVE A 1" SUPPLY CONNECTION AND 1/2" VENT CONNECTION.
10. 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
0-47B601-69 SERIES......INSTRUMENT TABULATION-REACTOR WATER CLEANUP SYSTEM
2-47E610-43-1...........MECHANICAL CONTROL DIAGRAM-SAMPLING & WATER QUALITY SYSTEM
2-47E610-69-1...........MECHANICAL CONTROL DIAGRAM-REACTOR WATER CLEANUP SYSTEM
47E800-1................MECHANICAL FLOW DIAGRAM-GENERAL PLANT SYSTEMS
0-47E800-2..............MECHANICAL SYMBOLS & FLOW DIAGRAM DRAWING INDEX
2-47E810-1..............MECHANICAL FLOW DIAGRAM-REACTOR WATER CLEANUP SYSTEM
2-47E811-1..............MECHANICAL FLOW DIAGRAM-RESIDUAL HEAT REMOVAL SYSTEM
2-47E814-1..............FLOW DIAGRAM-CORE SPRAY SYSTEM
0-47E830-5..............MECHANICAL FLOW DIAGRAM-RADWASTE
0-47E846-1..............MECHANICAL FLOW DIAGRAM-DEMINERALIZER BACKWASH AIR
2-47E852-1 & -2.........MECHANICAL FLOW DIAGRAM-CLEAN RADWASTE & DECONTAMINATED DRAINAGE
2-47E856-2..............MECHANICAL FLOW DIAGRAM-DEMINERALIZED WATER
T-12728.................MECHANICAL FLOW DIAGRAM-REACTOR WATER CLEANUP FILTER DEMINERALIZER SYSTEM (GRAVER)
2-47E2847-6.............MECHANICAL I/C FLOW DIAGRAM CONTROL AIR SYSTEM.

SYSTEM PRESS.- TEMP DATA

LINE MK NO.	DESIGN PRESSURE (PSIG)	DESIGN TEMP(°F)
1	1450	150
2	1300	150
3	150	150
4	0	150
5	120	103
6	100	100
7	40	120

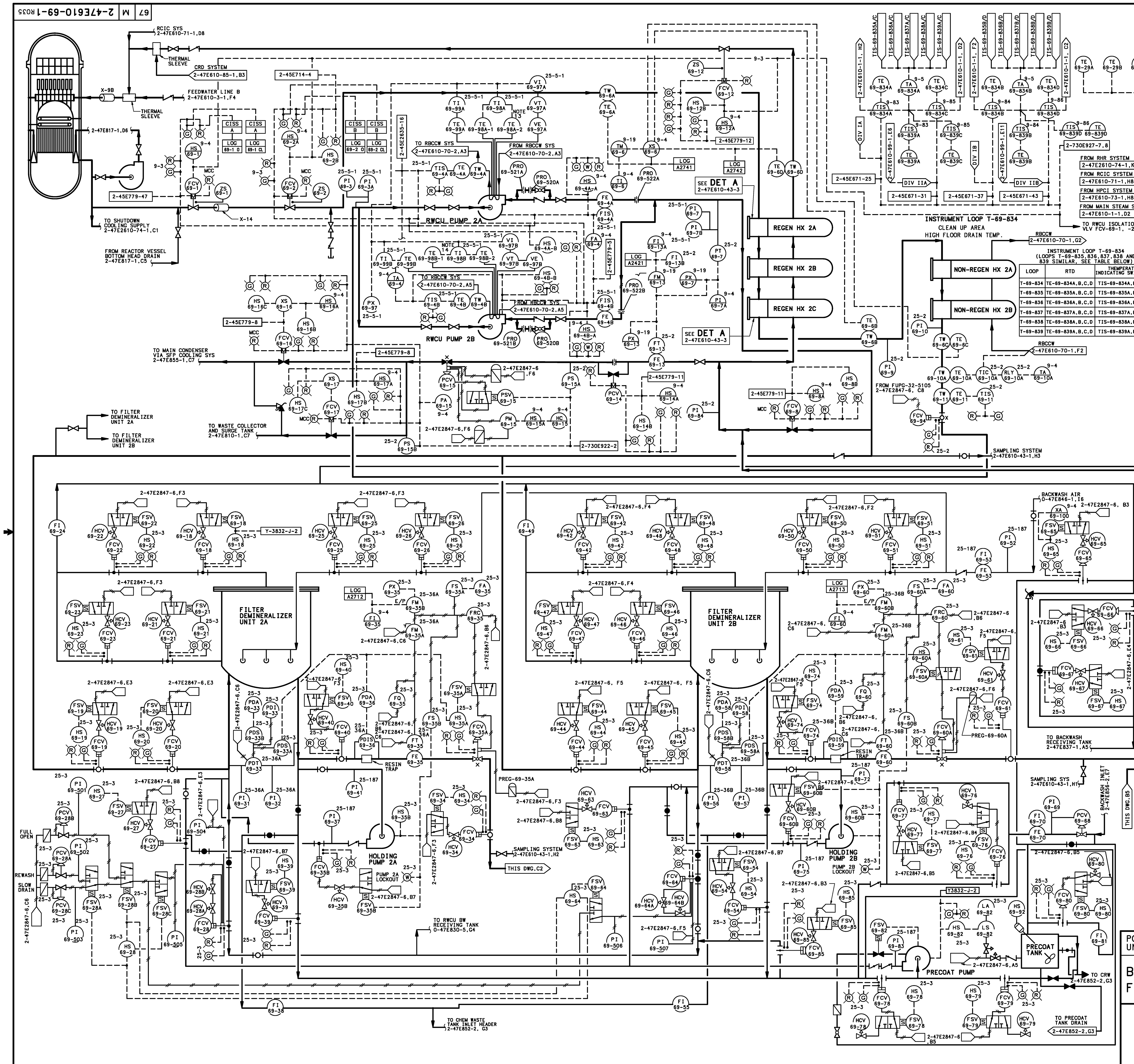
AMENDMENT 28

POWERHOUSE
UNIT 2

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

REACTOR WATER CLEANUP
DEMINERALIZER
FLOW DIAGRAM

FIGURE 4.9-2



INSTRUMENT LOOP T-69-834
(LOOPS T-69-835,836,837,838 AND
839 SIMILAR, SEE TABLE BELOW)

LOOP	RTD	THEMPERATURE INDICATING SWITCHES
T-69-834	TE-69-834A,B,C,D	TIS-69-834A,B,C,D
T-69-835	TE-69-835A,B,C,D	TIS-69-835A,B,C,D
T-69-836	TE-69-836A,B,C,D	TIS-69-836A,B,C,D
T-69-837	TE-69-837A,B,C,D	TIS-69-837A,B,C,D
T-69-838	TE-69-838A,B,C,D	TIS-69-838A,B,C,D
T-69-839	TE-69-839A,B,C,D	TIS-69-839A,B,C,D

NOTES:

1. THE REACTOR WATER CLEANUP SYSTEM PROVIDES FOR CONTINUOUS MECHANICAL AND CHEMICAL FILTRATION AND DEMINERALIZATION OF A PORTION OF THE REACTOR WATER TO QUALITY SPECIFICATION. IMPURITIES ARE REMOVED BY FILTRATION, ABSORPTION, AND ION EXCHANGE.
2. WATER FLOWS FROM THE REACTOR VESSEL TO THE REGENERATIVE HEAT EXCHANGER, THE NONREGENERATIVE THROUGH THE RWCU RECIRCULATING PUMPS TO THE DEMINERALIZERS . THE WATER THEN FLOWS TO THE REGENERATIVE HEAT EXCHANGER BACK TO THE REACTOR VESSEL.
3. BACKWASH AND PRECOAT OPERATIONS ARE MANUALLY INITIATED AND AUTOMATICALLY CONTINUED. DURATION OF EACH OPERATION WILL BE ADJUSTED IN THE FIELD BASED UPON SYSTEM REQUIREMENTS.
4. FCV-69-1, -2 & -12 ARE ISOLATION VALVES THAT MAY BE OPERATED MANUALLY FROM THE MAIN CONTROL ROOM. THE VALVES ARE AUTOMATICALLY CLOSED BY THE FOLLOWING SIGNALS:
   (A) ACTUATION OF STANDBY LIQUID CONTROL SYSTEM.
   (B) HIGH TEMPERATURE UPSTREAM OF THE FILTER-DEMINERALIZER.
   (C) HIGH TEMPERATURE IN AREAS OCCUPIED BY RWCU EQUIPMENT (RWCU HEAT EXCHANGER ROOM OR RWCU PUMP ROOMS 2A & 2B), OR HIGH TEMPERATURE IN THE RWCU PIPE TRENCH OR HIGH TEMPERATURE FROM RWCU PIPING IN THE MAIN STEAM VALVE VAULT.
   (D) REACTOR WATER LEVEL LOW.
5. THE RECIRCULATION PUMPS ARE PROTECTED FROM OVERHEATING BY TEMPERATURE SWITCHES IN THE COOLING WATER DISCHARGE (TIS-69-4A AND -4B) WHICH TRIP THE PUMP AND ALARM ON HIGH COOLING WATER TEMPERATURE. FLOW SWITCHES AT EACH PUMP DISCHARGE (FIS-69-4A AND -4B) MONITOR FLOW TO TRIP THE PUMPS UPON LOW FLOW AND ALARM (FA-69-4).
6. TEMPERATURE ACROSS THE HEAT EXCHANGERS IS MONITORED BY TI-69-6.
7. PRESSURE DROP ACROSS THE FILTER DEMINERALIZERS IS MONITORED BY (PDT-69-33 AND -58) WHICH ACTUATES AN ALARM (PDA-69-33 & -58) AND CLOSES THE DISCHARGE VALVE (FCV-69-35A & -60A) WHEN THE DIFFERENTIAL PRESSURE EXCEEDS 25 PSIG. A MANUAL SWITCH (HS-69-35A & -60A) IS ALSO PROVIDED FOR MANUAL ISOLATION OF THE FILTER DEMINERALIZER VESSEL.
8. LOW FLOW FROM A & B DEMINERALIZER (FE-69-35 & -60) WILL START THE HOLDING PUMP BY ACTUATING (FS-69-35B & -60B) OPEN VALVE FCV-69-35B & -60B CLOSE (FCV-69-35A & -60A) AND SOUND ALARM FA-69-35 & -60.
9. ONE CLEANUP PRECOAT SYSTEM SERVES BOTH DEMINERALIZER UNITS. A LEVEL SWITCH (LS-69-82) PROVIDES HIGH AND LOW ALARM AND LOW LEVEL PUMP INTERLOCK PROTECTION.
10. TIS-69-11 WILL ISOLATE THE SYSTEM ON HIGH DISCHARGE TEMPERATURE. THE COMPARTMENT FLOOR DRAIN LEAK DETECTION ELEMENTS (TS-69-30A THROUGH H) CAN BE SET TO ISOLATE THE SYSTEM AT APPROXIMATELY 20° F ABOVE AMBIENT TEMPERATURE.
11. FOR PANEL LOCATIONS OF PILOT LIGHTS FOR CONTROL VALVES, SEE THE PANEL NUMBER FOR THE ASSOCIATED HAND SWITCH.
12. 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 4160-V AC BOARD, 480-V AC BOARD, OR 250-V DC BOARD. (REFERENCE GE DESIGN SPEC 22A1470.)
13. TE-69-98A-1 PUMP 2A INBOARD BEARING T/C,
    TE-69-98A-2 PUMP 2A OUTBOARD BEARING T/C,
    TE-69-99A PUMP 2A CASING T/C.
14, TE-69-98B-1 PUMP 2B INBOARD BEARING T/C,
    TE-69-98B-2 PUMP 2B OUTBOARD BEARING T/C,
    TE-69-99B PUMP 2B CASING T/C.
15. CONTROL AIR TO INSTRUMENT SUPPLIED BY CONTROL AIR HEADER IN CABINET 25-3.
16. CONTROL AIR TO INSTRUMENTS SUPPLIED BY CONTROL AIR HEADER ON PANEL 25-36A & B.
17. ALL COMPONENTS ARE PREFIXED WITH UNIT 2 DESIGNATOR UNLESS OTHERWISE NOTED.
18. COMPLETE LOGIC FOR REACTOR WATER CLEANUP SYSTEM IS SHOWN ON MECHANICAL LOGIC DIAGRAM LISTED IN REFERENCE DRAWINGS.

REFERENCE DRAWINGS:

1,2,3-45E779-5,-8,-11,-12..SCHEMATIC DIAGRAM
2-45E614-SERIES............WIRING DIAGRAMS & MISC SCHEMATIC
2-45E2641-1,-7.............CONNECTION DIAGRAM
2-45E2642-1,-4.............CONNECTION DIAGRAM
2-45E2643-1,-5.............CONNECTION DIAGRAM
2-45N2664-1,-2.............CONNECTION DIAGRAM
2-45N2654-1,-2.............CONNECTION DIAGRAM
2-45N2635-16,-18...........CONNECTION DIAGRAM
2-45E2756-9................CONNECTION DIAGRAM
MEL........................INSTRUMENT TABULATION
2-47E406-1,2 & 3...........REACTOR WATER CLEANUP SYSTEM PIPING

47W552-SERIES..............MECHANICAL DIAGRAM
47W600-96..................EQUIPMENT LOCATION
0-47E600-112...............EQUIPMENT LOCATION
2-47E600-58................EQUIPMENT LOCATION
2-47E610-1-1...............CONTROL DIAGRAM - MAIN STEAM SYSTEM
1,2,3-47E610-3-1...........CONTROL DIAGRAM - REACTOR FEEDWATER
1,2,3-47E610-43-1..........CONTROL DIAGRAM - SAMPLING AND WATER QUALITY SYSTEM
2-47E610-68-1..............CONTROL DIAGRAM - REACTOR RECIRCN SYS
1,2-47E610-70-1............CONTROL DIAGRAM - REACTOR BLDG CLOSED COOLING WATER SYSTEM
1,2,3-47E610-71-1..........CONTROL DIAGRAM - RCIC SYSTEM
2-47E610-73-1..............CONTROL DIAGRAM - HPCI SYSTEM
2-47E2610-74-1, -2.........CONTROL DIAGRAM - RHR SYSTEM
0,1,2,3-47E610-78-1........CONTROL DIAGRAM - FUEL POOL COOLING & DEMINERALIZER
0-47E800-1.................FLOW DIAGRAM - GENERAL PLANT SYSTEMS
0-47E800-2.................MECHANICAL - SYMBOLS AND FLOW DIAGRAM DRAWING INDEX
2-47E810-1.................FLOW DIAGRAM - REACTOR WTR CLEANUP SYS
2-47E817-1.................FLOW DIAGRAM - REACTOR WTR RECIRCN SYS
0-47E830-2.................FLOW DIAGRAM - RADWASTE
0-47E830-5.................FLOW DIAGRAM - RADWASTE
2-47E837-1.................FLOW DIAGRAM - REACTOR WATER CLEANUP DEMINERALIZER
0-47E846-1.................FLOW DIAGRAM - DMNRLZR BACKWASH AIR SYS
1,2,3-47E852-2.............FLOW DIAGRAM - CLEAN RADWASTE AND DECON DRAINAGE
2-47E855-1.................FLOW DIAGRAM - FUEL POOL COOLING SYS
0-47E856-1.................FLOW DIAGRAM - DMNRLZR BACKWASH SYS
2-47E856-2.................FLOW DIAGRAM - DEMINERALIZED WATER
2-47E2847 SERIES...........FLOW DIAGRAM - CONTROL AIR SYSTEM
2-47E610-73-1..............CONTROL DIAGRAM - HPCI SYSTEM
3-47E610-1-1A..............CONTROL DIAGRAM - MAIN STEAM SYSTEM
2-47E917-1.................FLOW DIAGRAM - REACTOR WTR RECIRCN SYS
2-47E610-70-1..............CONTROL DIAGRAM - REACTOR BLDG CLOSED COOLING WATER SYSTEM
0-47E610-77-5, -4..........CONTROL DIAGRAM - RADWASTE SYSTEM
2-47E610-68-1..............CONTROL DIAGRAM - REACTOR RECIRCN SYS
2-47E2610-74-1.............CONTROL DIAGRAM - RHR SYSTEM
2-47E611-63-SERIES.........MECHANICAL LOGIC DIAGRAM STANDBY LIQUID CONTROL SYSTEM
2-47E611-64-SERIES.........MECHANICAL LOGIC DIAGRAM - PRIMARY CONTAINMENT ISOLATION SYSTEM
2-47E611-69-1,-2...........MECHANICAL LOGIC DIAGRAM - REACTOR WATER CLEANUP SYSTEM

GRAVER WATER CONDITIONING CO.
T-12728..............FLOW DIAGRAM

GE DRAWINGS:
728E900 ....REACTOR WATER CLEANUP SYSTEM
730E719 ....FUNCTIONAL CONTROL DIAGRAM
0-730E163 ....PROCESS DIAGRAM
728E905 ....CLEANUP FILTER DEMINERALIZER SYSTEM

AMENDMENT 28

POWERHOUSE
UNIT 2

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

REACTOR WATER CLEANUP SYSTEM
MECHANICAL CONTROL DIAGRAM

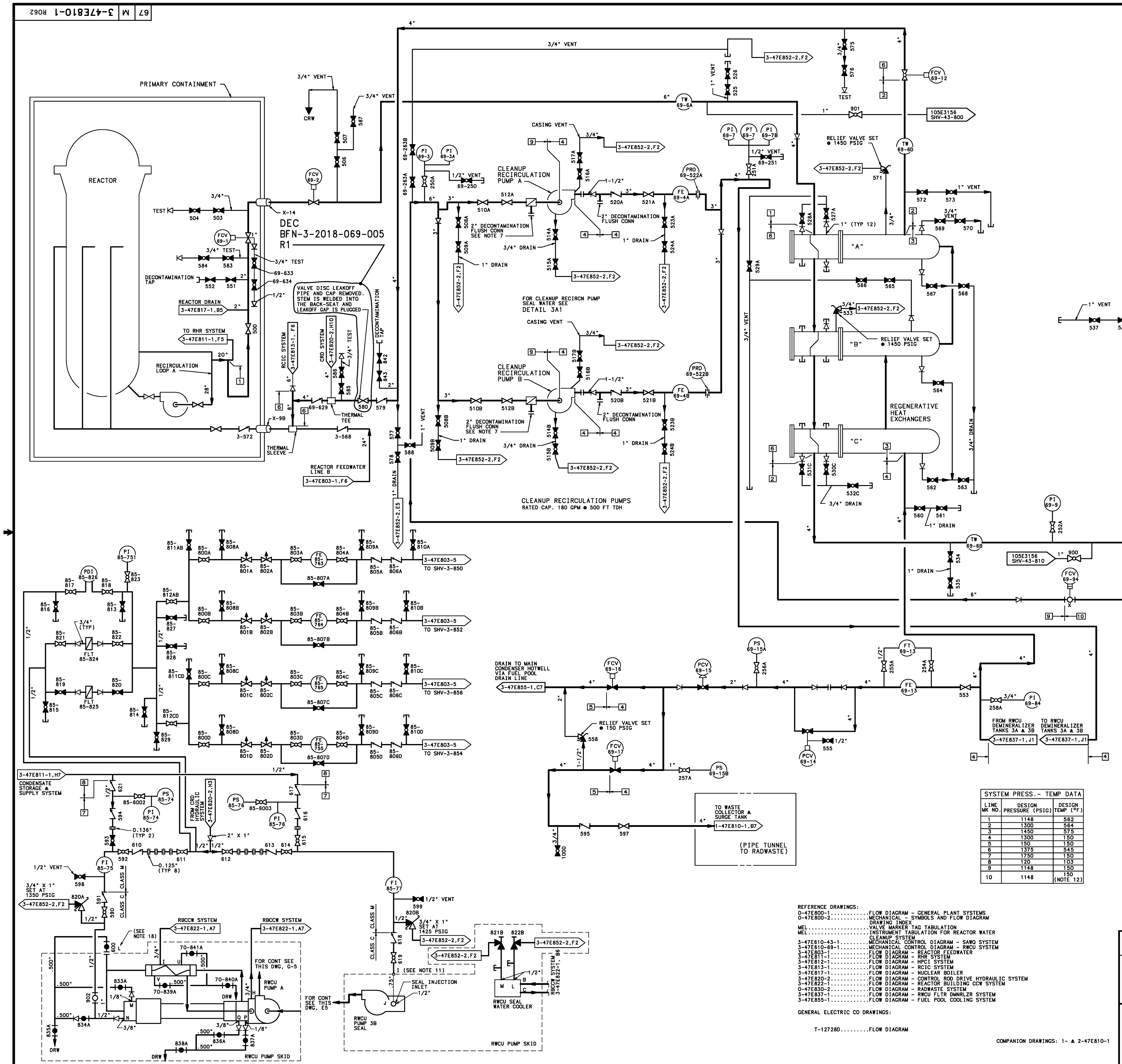
FIGURE 4.9-3

BFN-16

Figure 4.9-4

Deleted by Amendment 9

Figures 4.9-4a through 4.9-4d
(Deleted by Amendment 22)



NOTES:

1. CRW DRAINS ARE CONNECTED TO REACTOR BUILDING EQUIPMENT DRAIN SUMP.
2. ALL VENTS AND DRAINS SHOWN INDICATES CLOSED SYSTEM.
3. FOR NON-REGENERATIVE HEAT EXCHANGER SHELL CONNECTIONS SEE REACTOR BUILDING CLOSED COOLING WATER SYSTEM DWG 3-47E822-1.
4. ALL VALVES ARE THE SAME SIZE AS PIPING, UNLESS OTHERWISE NOTED.
5. ALL TEST CONNECTIONS ARE 1/2" AND PRESSURE CONNECTIONS ARE 1", UNLESS OTHERWISE NOTED.
6. ALL VALVES ARE PREFIXED 3-69 UNLESS OTHERWISE NOTED.
7. STRAINER INTERNAL PARTS REMOVED AFTER STARTUP.
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. THESE LETTER DESIGNATIONS CORRESPOND TO THE INTERFACE POINTS SHOWN ON INGERSOLL-RAND DRAWING 0-62325949 AND 62368659. (CONTRACT NO. 74113A)
12. THE SYSTEM SHALL BE QUALIFIED FOR AN ELEVATED TEMPERATURE EXCURSION UP TO 562° F DURING AN NFPA 805 EVENT FROM THE NON-REGENERATIVE HEAT EXCHANGER OUTLET THROUGH 3-FCV-069-0094.
13. THE FUSIBLE PLUG IS A THREADED PIPE END WITH LOW TEMPERATURE ALLOY EUTECTIC MATERIAL WHICH IS ATTACHED TO THE RWCU PIPE UPSTREAM OF FCV-69-94. THE EUTECTIC MATERIAL WILL MELT ON HIGH TEMPERATURE, VENTING THE CONTROL AIR LINE, WHICH CLOSES ISOLATION VALVE FCV-69-94.
14. UNIDS ON DRAWING ARE FOR REFERENCE ONLY AND ARE ABBREVIATED TO MEET SPACE CONSTRAINTS, REFER TO MEL FOR COMPLETE UNIDS.
15. 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.
16. DELETED.
17. IF NEW FLOWSERVE CARTRIDGE SEAL IS INSTALLED, THERE IS NOT A CONN N TO CAP/PLUG. NOT APPLICABLE TO PUMP 3B.
18. THESE LETTER DESIGNATIONS CORRESPOND TO THE INTERFACE POINTS SHOWN ON HAYWARD TYLER, INC DRAWING 01-500-868 (CONTRACT NO. 2885991).

SYSTEM PRESS.- TEMP DATA

LINE MK NO.	DESIGN PRESSURE (PSIG)	DESIGN TEMP (°F)
1	1148	562
2	1300	564
3	1450	575
4	1300	150
5	150	150
6	1375	545
7	1750	150
8	120	103
9	1148	150
10	1148	150 (NOTE 12)

REFERENCE DRAWINGS:

0-47E800-1............FLOW DIAGRAM - GENERAL PLANT SYSTEMS
0-47E800-2............MECHANICAL - SYMBOLS AND FLOW DIAGRAM DRAWING INDEX
MEL...................VALVE MARKER TAG TABULATION
MEL...................INSTRUMENT TABULATION FOR REACTOR WATER CLEANUP SYSTEM
3-47E610-43-1.........MECHANICAL CONTROL DIAGRAM - SAWQ SYSTEM
3-47E610-69-1.........MECHANICAL CONTROL DIAGRAM - RWCU SYSTEM
3-47E803-1............FLOW DIAGRAM - REACTOR FEEDWATER
3-47E811-1............FLOW DIAGRAM - RHR SYSTEM
3-47E812-1............FLOW DIAGRAM - HPCI SYSTEM
3-47E813-1............FLOW DIAGRAM - RCIC SYSTEM
3-47E817-1............FLOW DIAGRAM - NUCLEAR BOILER
3-47E820-2............FLOW DIAGRAM - CONTROL ROD DRIVE HYDRAULIC SYSTEM
3-47E822-1............FLOW DIAGRAM - REACTOR BUILDING CCW SYSTEM
0-47E830-2............FLOW DIAGRAM - RADWASTE SYSTEM
3-47E837-1............FLOW DIAGRAM - RWCU FLTR DMNRLZR SYSTEM
3-47E855-1............FLOW DIAGRAM - FUEL POOL COOLING SYSTEM

GENERAL ELECTRIC CO DRAWINGS:

T-12728D.........FLOW DIAGRAM

COMPANION DRAWINGS: 1- & 2-47E810-1

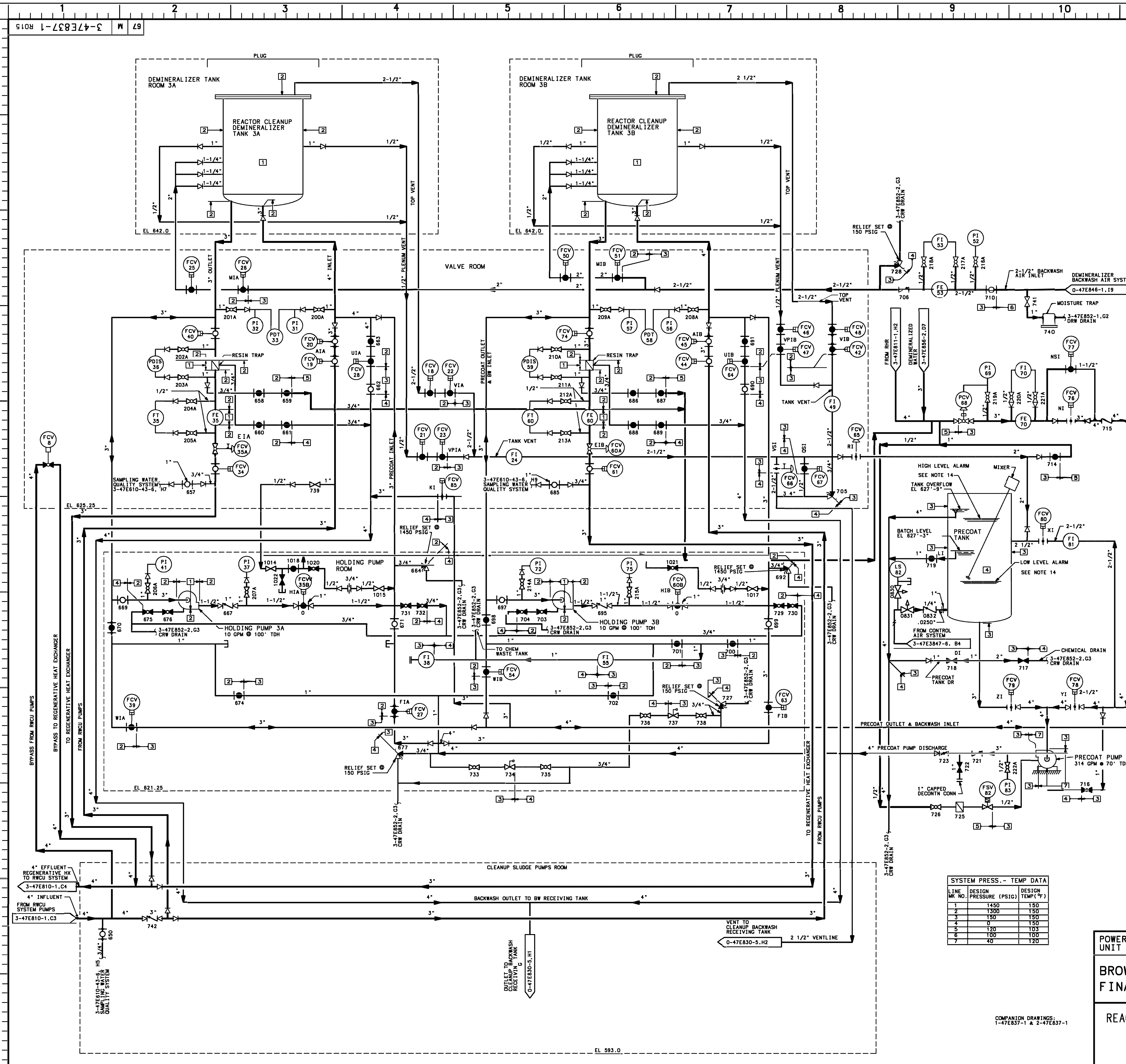
AMENDMENT 28

POWERHOUSE
UNIT 3

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

REACTOR WATER CLEANUP SYSTEM
FLOW DIAGRAM

FIGURE 4.9-5



DEMINERALIZER DATA CLEANUP

DEMINERALIZER RATE (PER UNIT)

DESIGN 340 GPM (MAX).......1.08 GPM PER SQ FT

QUANTITY OF BACKWASH AIR...236 SCFM AT 25 PSIG

BACKWASH WATER.............NORMAL - 157 GPM AT 25 PSIG
.............MAXIMUM - 236 GPM AT 25 PSIG

PRESSURE...................DESIGN - 1300 PSIG
...................OPERATING - 1155 PSIG

TEMPERATURE................DESIGN - 150°F
................OPERATING - 130°F

NOTES:

1. FOR DETAILED OPERATING INSTRUCTION, BACKWASH RATES, ETC, SEE MANUFACTURER'S INSTRUCTION MANUAL.
2. ALL VALVES ARE SAME SIZE AS PIPE UNLESS OTHERWISE NOTED.
3. HEAVY LINES SHOW WATER FLOW THROUGH SYSTEM DURING NORMAL DEMINERALIZATION.
4. OPERATIONAL VALVES ARE SHOWN IN THEIR NORMAL OPERATING POSITION.
5. VALVE MARKER TAG NUMBERS CARRY THE PREFIX 3-69 AS IN 3-69-650, ETC. THE SYSTEM IDENTIFICATION NUMBER FOR RWCU SYSTEM IS 69.
6. FLOW CONTROL VALVES SHALL BE DESIGNATED 3-FCV-69-33 AND NOTED ON THIS DRAWING AS FCV 33.
7. EIA, ETC. INDICATES VALVE DESIGNATIONS AS SHOWN ON MANUFACTURERS FLOW DIAGRAM T-12728.
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. DELETED.
12. CRW = CLEAN RADWASTE
ORW = DIRTY RADWASTE
13. ALL CONTROL VALVES FAIL CLOSED UNLESS NOTED OTHERWISE.
14. FOR SET POINT ELEVATIONS OF PRECOAT TANK HIGH/LOW LEVEL ALARMS REFER TO MASTER EQUIPMENT LIST (MEL).
15. 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:

0-47E800-2......MECHANICAL SYMBOLS & FLOW DIAGRAM INDEX
0-47E830-5......MECHANICAL FLOW DIAGRAM-RADWASTE
47E800-1........MECHANICAL FLOW DIAGRAM-GENERAL PLANT SYSTEM
0-47E846-1......MECHANICAL FLOW DIAGRAM-DEMINERALIZER BACKWASH AIR SYSTEM
3-47E810-1......MECHANICAL FLOW DIAGRAM-REACTOR WATER CLEANUP SYSTEM
3-47E811-1......MECHANICAL FLOW DIAGRAM-RESIDUAL HEAT REMOVAL SYSTEM
3-47E852-1......MECHANICAL FLOW DIAGRAM-FLOOR & DIRTY RADWASTE DRAINAGE
3-47E852-2......MECHANICAL FLOW DIAGRAM-CLEAN RADWASTE & DECONTAMINATED DRAINAGE
3-47E856-2......MECHANICAL FLOW DIAGRAM-DEMINERALIZED WATER
3-47E610-69-1...MECHANICAL CONTROL DIAGRAM-REACTOR WATER CLEANUP SYSTEM
3-47E610-43-1...MECHANICAL CONTROL DIAGRAM-SAMPLING & WATER QUALITY SYSTEM

VENDOR DRAWING:

T-12728.........MECHANICAL FLOW DIAGRAM-REACTOR WATER CLEANUP FILTER DEMINERALIZER SYSTEM (GRAVER)

SYSTEM PRESS.- TEMP DATA

LINE MK NO.	DESIGN PRESSURE (PSIG)	DESIGN TEMP(°F)
1	1450	150
2	1300	150
3	150	150
4	0	150
5	120	103
6	100	100
7	40	120

COMPANION DRAWINGS:
1-47E837-1 & 2-47E837-1

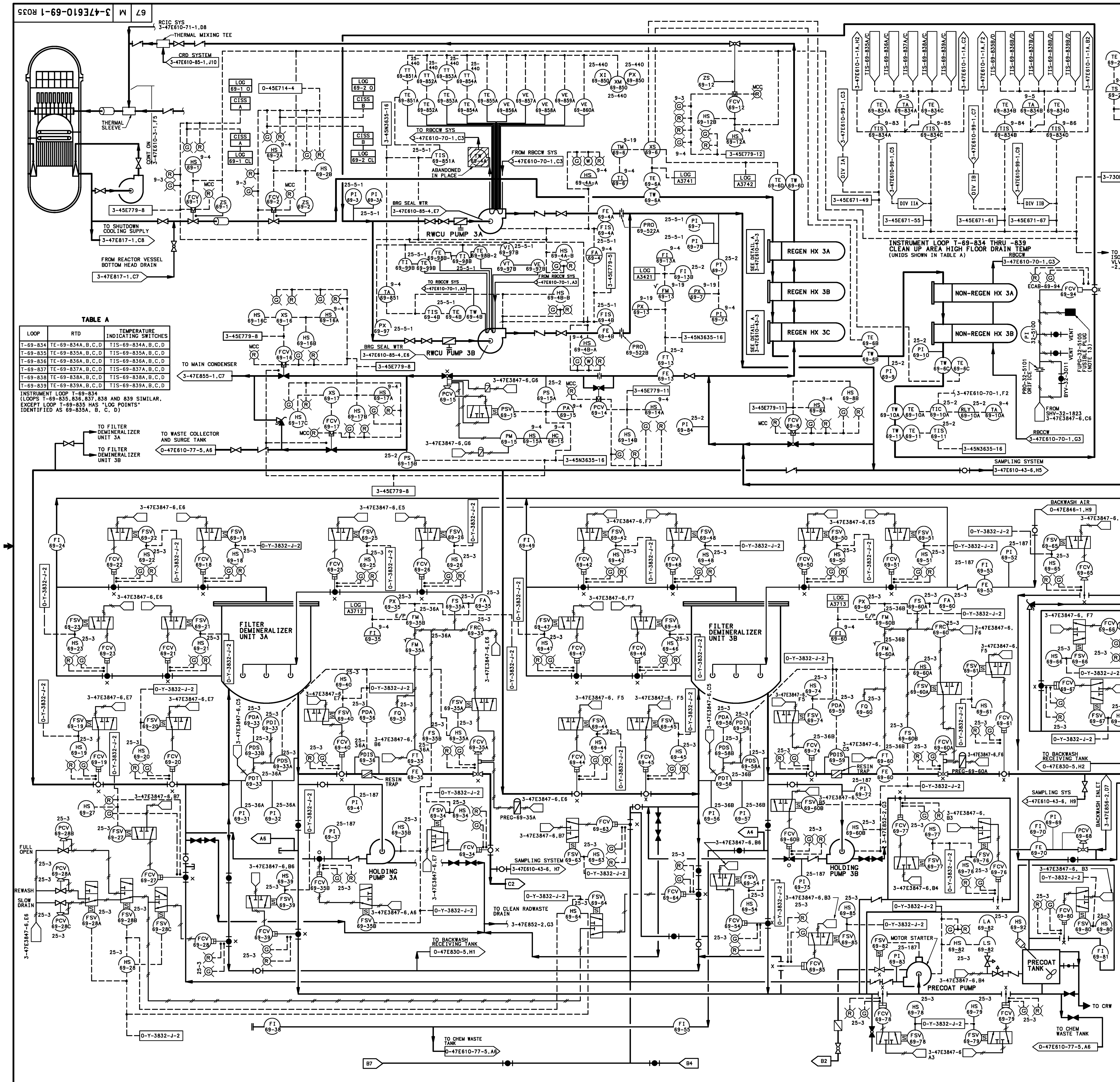
AMENDMENT 28

POWERHOUSE
UNIT 3

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

REACTOR WATER CLEANUP DEMINERALIZER
FLOW DIAGRAM

FIGURE 4.9-6



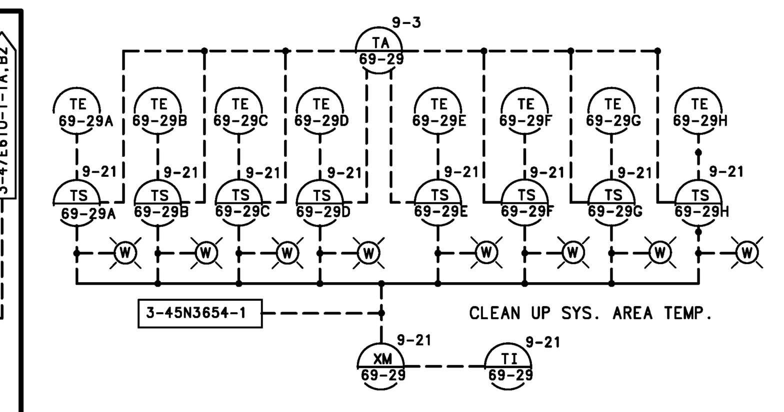


TABLE A

LOOP	RTD	TEMPERATURE INDICATING SWITCHES
T-69-834	TE-69-834A,B,C,D	TIS-69-834A,B,C,D
T-69-835	TE-69-835A,B,C,D	TIS-69-835A,B,C,D
T-69-836	TE-69-836A,B,C,D	TIS-69-836A,B,C,D
T-69-837	TE-69-837A,B,C,D	TIS-69-837A,B,C,D
T-69-838	TE-69-838A,B,C,D	TIS-69-838A,B,C,D
T-69-839	TE-69-839A,B,C,D	TIS-69-839A,B,C,D

INSTRUMENT LOOP T-69-834
(LOOPS T-69-835,836,837,838 AND 839 SIMILAR, EXCEPT LOOP T-69-835 HAS "LOG POINTS" IDENTIFIED AS 69-835A, B, C, D)

NOTES:

1. THE REACTOR WATER CLEANUP SYSTEM PROVIDES FOR CONTINUOUS MECHANICAL AND CHEMICAL FILTRATION AND DEMINERALIZATION OF A PORTION OF THE REACTOR WATER TO QUALITY SPECIFICATION. IMPURITIES ARE REMOVED BY FILTRATION, ABSORPTION, AND ION EXCHANGE.
2. WATER FLOWS FROM THE REACTOR VESSEL TO THE REGENERATIVE HEAT EXCHANGER, THE NONREGENERATIVE THROUGH THE RWCU RECIRCULATING PUMPS TO THE DEMINERALIZERS. THE WATER THEN FLOWS TO THE REGENERATIVE HEAT EXCHANGER BACK TO THE REACTOR VESSEL.
3. BACKWASH AND PRECOAT OPERATIONS ARE MANUALLY INITIATED AND AUTOMATICALLY CONTINUED. DURATION OF EACH OPERATION WILL BE ADJUSTED IN THE FIELD BASED UPON SYSTEM REQUIREMENTS.
4. FCV-69-1,AND -2 ARE ISOLATION VALVES THAT MAY BE OPERATED MANUALLY FROM THE MAIN CONTROL ROOM. THE VALVES ARE AUTOMATICALLY CLOSED BY THE FOLLOWING SIGNALS:
   (A) ACTUATION OF STANDBY LIQUID CONTROL SYSTEM.
   (B) HIGH TEMPERATURE UPSTREAM OF THE FILTER-DEMINERALIZER.
   (C) HIGH TEMPERATURE IN AREAS OCCUPIED BY CLEANUP SYSTEM EQUIPMENT. ACTUATED BY TS-69-29J, K, L OR M.
   (D) HIGH TEMPERATURE IN FLOOR DRAINS IN AREAS OCCUPIED BY CLEANUP SYSTEM EQUIPMENT. ACTUATED BY TS-69-30A, B, C, D, E, F, G, OR H.
   (E) REACTOR WATER LOW LEVEL.
5. THE RECIRCULATION PUMPS ARE PROTECTED FROM OVERHEATING BY TEMPERATURE SWITCHES (TIS-69-851A AND -4B) WHICH TRIP THE PUMP AND ALARM ON HIGH TEMPERATURE. FLOW SWITCHES AT EACH PUMP DISCHARGE (FIS-69-4A AND -4B) MONITOR FLOW TO TRIP THE PUMPS UPON LOW FLOW AND ALARM (FA-69-4A AND -4B).
6. TEMPERATURE ACROSS THE HEAT EXCHANGERS IS MONITORED BY TI-69-6.
7. PRESSURE DROP ACROSS THE FILTER DEMINERALIZERS IS MONITORED BY (PDT-69-33 AND -58) WHICH ACTUATES AN ALARM (PDA-69-33 & -58) AND CLOSES THE DISCHARGE VALVE (FCV-69-35A & -60A) WHEN THE DIFFERENTIAL PRESSURE EXCEEDS 25 PSIG. A MANUAL SWITCH (HS-69-35A & -60A) IS ALSO PROVIDED FOR MANUAL ISOLATION OF THE FILTER DEMINERALIZER VESSEL.
8. LOW FLOW FROM A DEMINERALIZER (FE-69-35) WILL START HOLDING PUMP 3A BY ACTUATING FS-69-35B, OPENING VALVE FCV-69-35B, CLOSING FCV-69-35A, AND SOUND ALARM FA-69-35. THE OPERATION OF HOLDING PUMP 38 IS SIMILIAR.
9. ONE CLEANUP PRECOAT SYSTEM SERVES BOTH DEMINERALIZER UNITS. A LEVEL SWITCH (LS-69-82) PROVIDES HIGH AND LOW ALARM AND LOW LEVEL PUMP INTERLOCK PROTECTION.
10. TIS-69-11 WILL ISOLATE THE SYSTEM ON HIGH DISCHARGE TEMPERATURE. THE COMPARTMENT FLOOR DRAIN LEAK DETECTION ELEMENTS (TS-69-30A THROUGH H) CAN BE SET TO ISOLATE THE SYSTEM AT APPROXIMATELY 20° F ABOVE AMBIENT.
11. FOR PANEL LOCATIONS OF PILOT LIGHTS FOR CONTROL VALVES, SEE THE PANEL NUMBER FOR THE ASSOCIATED HAND SWITCH.
12. 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 4160-V AC BOARD, 480-V AC BOARD, OR 250-V DC BOARD. (REFERENCE GE DESIGN SPEC 22A1470.)
13. THE FUSIBLE PLUG IS A THREADED PIPE END WITH A LOW TEMPERATURE ALLOY EUTECTIC MATERIAL WHICH IS ATTACHED TO THE RWCU PIPE UPSTREAM OF FCV-69-94. THE EUTECTIC MATERIAL WILL MELT ON HIGH TEMPERATURE, VENTING THE CONTROL AIR LINE, WHICH CLOSES ISOLATION VALVE FCV-69-94.

REFERENCE DRAWINGS:
3-45E714-4..........250V DC SCHEMATIC DIAGRAM
0-47E610-77-4.......MECHANICAL CONTROL DIAGRAM RADWATER SYSTEM
0-47E610-77-5.......MECHANICAL CONTROL DIAGRAM RADWATER SYSTEM
0-47E800-2..........MECHANICAL SYMBOLS AND FLOW DIAGRAM INDEX
3-45E779-5,8,11,12...480V SCHEMATIC DIAGRAM
3-45N3635-16........CONNECTION DIAGRAM
3-45N3654-1.........WIRING DIAGRAM PANEL 9-21
3-47E810-1..........FLOW DIAGRAM - REACTOR WATER CLEANUP SYSTEM
3-47E3847-6.........FLOW DIAGRAM CONTROL AIR SYSTEM
MEL.................INSTRUMENT TABULATION FOR REACTOR WATER CLEANUP SYSTEM
0-47E800-1..........FLOW DIAGRAM - GENERAL PLANT SYSTEMS
3-47E837-1..........FLOW DIAGRAM REACTOR WATER CLEANUP DEMINERALIZER
0-47E846-1..........FLOW DIAGRAM - DEMINERALIZER BACKWASH AIR
3-47W406-1..........REACTOR WATER CLEANUP SYSTEM PIPING
0-Y-3832-J-2........REACTOR CLEANUP SCHEMATIC DRAWING
3-47E610-71-1.......MECHANICAL CONTROL DIAGRAM RCIC SYSTEM
3-47E610-3-1........MECHANICAL CONTROL DIAGRAM FEEDWATER SYSTEM
3-47E610-70-1.......MECHANICAL CONTROL DIAGRAM RBCCW SYSTEM
3-47E856-2..........FLOW DIAGRAM DEMINERALIZER WATER
0-47E830-5..........FLOW DIAGRAM RADWASTE
3-47E852-2..........FLOW DIAGRAM CLEAN RADWASTE & DECONTAMINATED DRAINAGE
3-47E406-2..........REACTOR WATER CLEANUP SYSTEM PIPING
0-47W406-3..........REACTOR WATER CLEANUP SYSTEM PIPING
3-47E610-1-1A.......MECHANICAL CONTROL DIAGRAM MAIN STEAM SYSTEM

GE DRAWINGS
730E719........FUNCTIONAL CONTROL DIAGRAM

GRAVER WATER CONDITIONING CO
T-12728........FLOW DIAGRAM

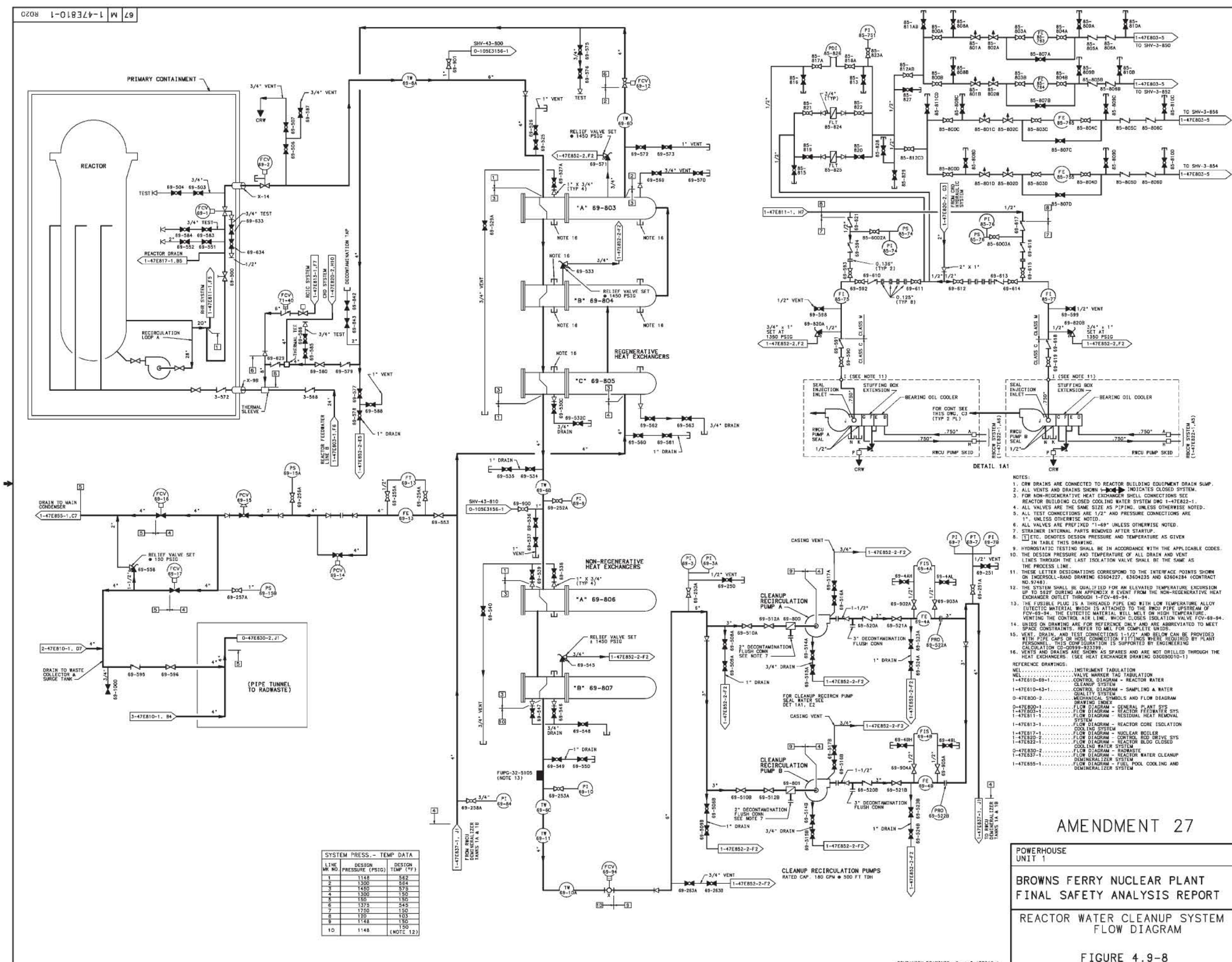
AMENDMENT 28

POWERHOUSE
UNIT 3

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

REACTOR WATER CLEANUP SYSTEM
MECHANICAL CONTROL DIAGRAM

FIGURE 4.9-7



SYSTEM PRESS.- TEMP DATA		
LINE MK NO.	DESIGN PRESSURE (PSIG)	DESIGN TEMP (°F)
1	1148	562
2	1300	564
3	1450	575
4	1300	150
5	150	150
6	1375	545
7	1750	150
8	120	103
9	1148	150
10	1148	150 (NOTE 12)

NOTES:

1. CRW DRAINS ARE CONNECTED TO REACTOR BUILDING EQUIPMENT DRAIN SUMP.
2. ALL VENTS AND DRAINS SHOWN INDICATES CLOSED SYSTEM.
3. FOR NON-REGENERATIVE HEAT EXCHANGER SHELL CONNECTIONS SEE REACTOR BUILDING CLOSED COOLING WATER SYSTEM DWG 1-47E822-1.
4. ALL VALVES ARE THE SAME SIZE AS PIPING, UNLESS OTHERWISE NOTED.
5. ALL TEST CONNECTIONS ARE 1/2" AND PRESSURE CONNECTIONS ARE 1", UNLESS OTHERWISE NOTED.
6. ALL VALVES ARE PREFIXED "1-69" UNLESS OTHERWISE NOTED.
7. STRAINER INTERNAL PARTS REMOVED AFTER STARTUP.
8. 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. THESE LETTER DESIGNATIONS CORRESPOND TO THE INTERFACE POINTS SHOWN ON INGERSOLL-RAND DRAWING 63604227, 63604235 AND 63604284 (CONTRACT NO.9748).
12. THE SYSTEM SHALL BE QUALIFIED FOR AN ELEVATED TEMPERATURE EXCURSION UP TO 562F DURING AN APPENDIX R EVENT FROM THE NON-REGENERATIVE HEAT EXCHANGER OUTLET THROUGH 1-FCV-69-94.
13. THE FUSIBLE PLUG IS A THREADED PIPE END WITH LOW TEMPERATURE ALLOY EUTECTIC MATERIAL WHICH IS ATTACHED TO THE RWCU PIPE UPSTREAM OF FCV-69-94. THE EUTECTIC MATERIAL WILL MELT ON HIGH TEMPERATURE, VENTING THE CONTROL AIR LINE, WHICH CLOSES ISOLATION VALVE FCV-69-94.
14. UNIDS ON DRAWING ARE FOR REFERENCE ONLY AND ARE ABBREVIATED TO MEET SPACE CONSTRAINTS. REFER TO MEL FOR COMPLETE UNIDS.
15. 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.
16. VENTS AND DRAINS ARE SHOWN AS SPARES AND ARE NOT DRILLED THROUGH THE HEAT EXCHANGERS. (SEE HEAT EXCHANGER DRAWING 03G050D10-1)

REFERENCE DRAWINGS:

- MEL....................INSTRUMENT TABULATION
- MEL....................VALVE MARKER TAG TABULATION
- 1-47E610-69-1........CONTROL DIAGRAM - REACTOR WATER CLEANUP SYSTEM
- 1-47E610-43-1........CONTROL DIAGRAM - SAMPLING & WATER QUALITY SYSTEM
- 0-47E800-2...........MECHANICAL SYMBOLS AND FLOW DIAGRAM DRAWING INDEX
- 0-47E800-1...........FLOW DIAGRAM - GENERAL PLANT SYS
- 1-47E803-1...........FLOW DIAGRAM - REACTOR FEEDWATER SYS
- 1-47E811-1...........FLOW DIAGRAM - RESIDUAL HEAT REMOVAL SYSTEM
- 1-47E813-1...........FLOW DIAGRAM - REACTOR CORE ISOLATION COOLING SYSTEM
- 1-47E817-1...........FLOW DIAGRAM - NUCLEAR BOILER
- 1-47E820-2...........FLOW DIAGRAM - CONTROL ROD DRIVE SYS
- 1-47E822-1...........FLOW DIAGRAM - REACTOR BLDG CLOSED COOLING WATER SYSTEM
- 0-47E830-2...........FLOW DIAGRAM - RADWASTE
- 1-47E837-1...........FLOW DIAGRAM - REACTOR WATER CLEANUP DEMINERALIZER SYSTEM
- 1-47E855-1...........FLOW DIAGRAM - FUEL POOL COOLING AND DEMINERALIZER SYSTEM

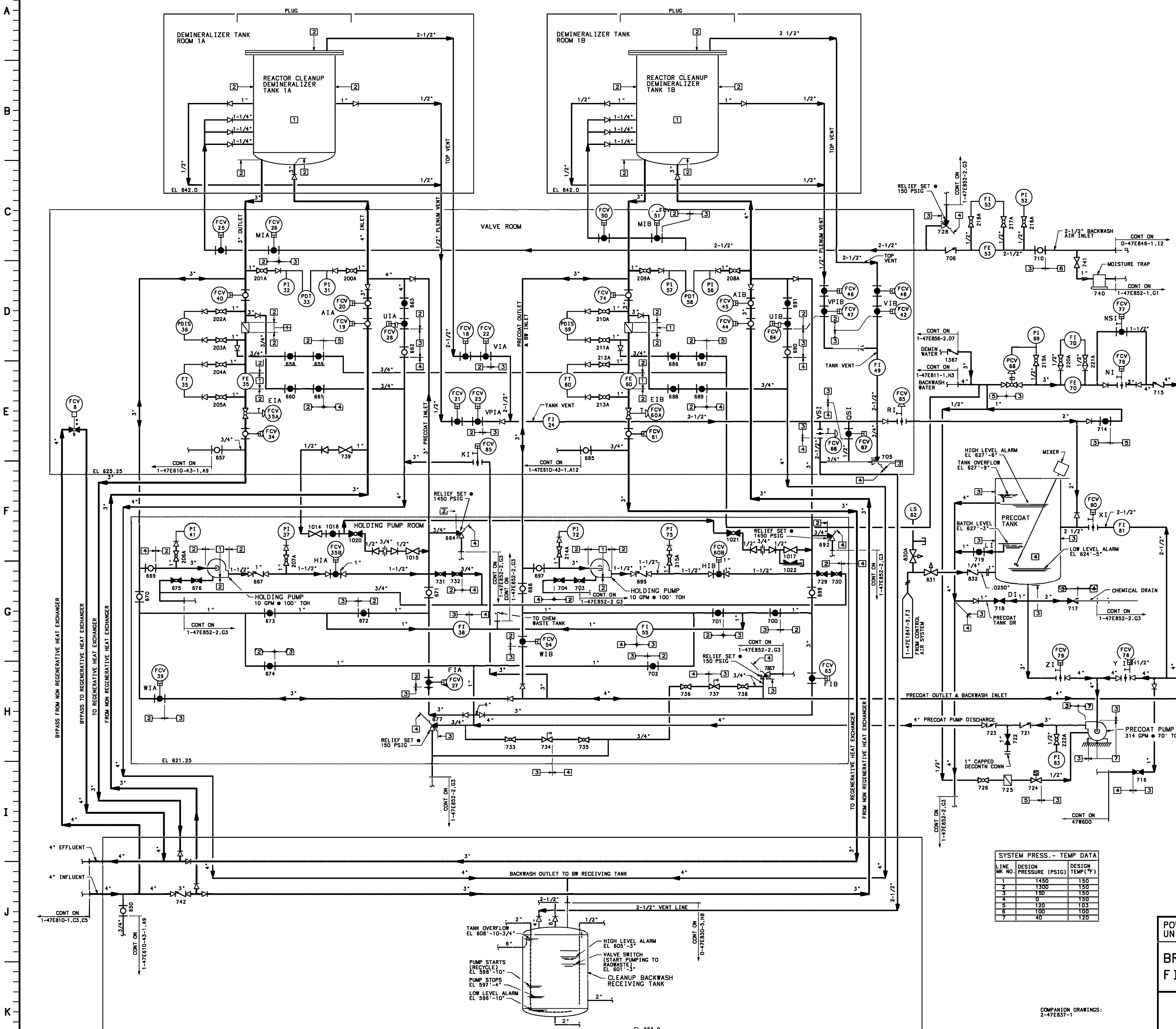
AMENDMENT 27

POWERHOUSE
UNIT 1

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

REACTOR WATER CLEANUP SYSTEM
FLOW DIAGRAM

FIGURE 4.9-8



DEMINERALIZER DATA CLEANUP

DEMINERALIZER RATE (PER UNIT)

DESIGN 340 GPM (MAX)......1.08 GPM PER SQ FT

QUANTITY OF BACKWASH AIR...236 SCFM AT 25 PSIG
BACKWASH WATER.............NORMAL - 157 GPM AT 25 PSIG
.............MAXIMUM - 236 GPM AT 25 PSIG
PRESSURE...................DESIGN - 1300 PSIG
...................OPERATING - 1155 PSIG
TEMPERATURE................DESIGN - 150°F
................OPERATING - 130°F

NOTES:

1. FOR DETAILED OPERATING INSTRUCTION, BACKWASH RATES, ETC. SEE MANUFACTURER'S INSTRUCTION MANUAL.
2. ALL VALVES ARE SAME SIZE AS PIPE UNLESS OTHERWISE NOTED.
3. HEAVY LINES SHOW WATER FLOW THROUGH SYSTEM DURING NORMAL DEMINERALIZATION.
4. OPERATIONAL VALVES ARE SHOWN IN THEIR NORMAL OPERATING POSITION.
5. VALVE MARKER TAG NUMBERS ON THIS DRAWING WHICH ARE NUMBERED FOR UNIT 1 CARRY THE PREFIX 1-69 AS IN 1-69-650, ETC. THE SYSTEM IDENTIFICATION NUMBER FOR RWCU SYSTEM IS 69.
6. FLOW CONTROL VALVES SHALL BE DESIGNATED 1-FCV-69-33 AND NOTED ON THIS DRAWING AS FCV 33.
7. E1A, ETC. INDICATES VALVE DESIGNATIONS AS SHOWN ON MANUFACTURER'S FLOW DIAGRAM T-12728A.
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. OPERATION OF 1-69-691 REQUIRES ENTRY INTO THE RWCU VALVE ROOM, THEREFORE THIS VALVE SHALL BE OPEN DURING UNIT 1 RECOVERY PROCESS WITH THE REACTOR DEPRESSURIZED.
12. CRW = CLEAN RADWASTE
DRW = DIRTY RADWASTE
13. ALL CONTROL VALVES FAIL CLOSED UNLESS NOTED OTHERWISE.
14. FOR SET POINT ELEVATIONS OF PRECOAT TANK HIGH/LOW LEVEL ALARMS REFER TO MASTER EQUIPMENT LIST (MEL).
15. 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:

0-47E800-1.......MECHANICAL FLOW DIAGRAM-GENERAL PLANT SYSTEM
0-47E800-2.......MECHANICAL SYMBOLS & FLOW DIAGRAM INDEX
0-47E830-5.......MECHANICAL FLOW DIAGRAM-RADWASTE
0-47E846-1.......MECHANICAL FLOW DIAGRAM-DEMINERALIZER BACKWASH AIR SYSTEM
1-47E810-1.......MECHANICAL FLOW DIAGRAM-REACTOR WATER CLEANUP SYSTEM
1-47E811-1.......MECHANICAL FLOW DIAGRAM-RESIDUAL HEAT REMOVAL SYSTEM
1-47E852-1.......MECHANICAL FLOW DIAGRAM-FLOOR & DIRTY RADWASTE DRAINAGE
1-47E852-2.......MECHANICAL FLOW DIAGRAM-CLEAN RADWASTE & DECONTAMINATED DRAINAGE
1-47E856-2.......MECHANICAL FLOW DIAGRAM-DEMINERALIZED WATER
1-47E610-69-1....MECHANICAL CONTROL DIAGRAM-REACTOR WATER CLEANUP SYSTEM
1-47E610-43-1....MECHANICAL CONTROL DIAGRAM-SAMPLING & WATER QUALITY SYSTEM

SYSTEM PRESS.- TEMP DATA

LINE MK NO.	DESIGN PRESSURE (PSIG)	DESIGN TEMP(°F)
1	1450	150
2	1300	150
3	150	150
4	0	150
5	120	103
6	100	100
7	40	120

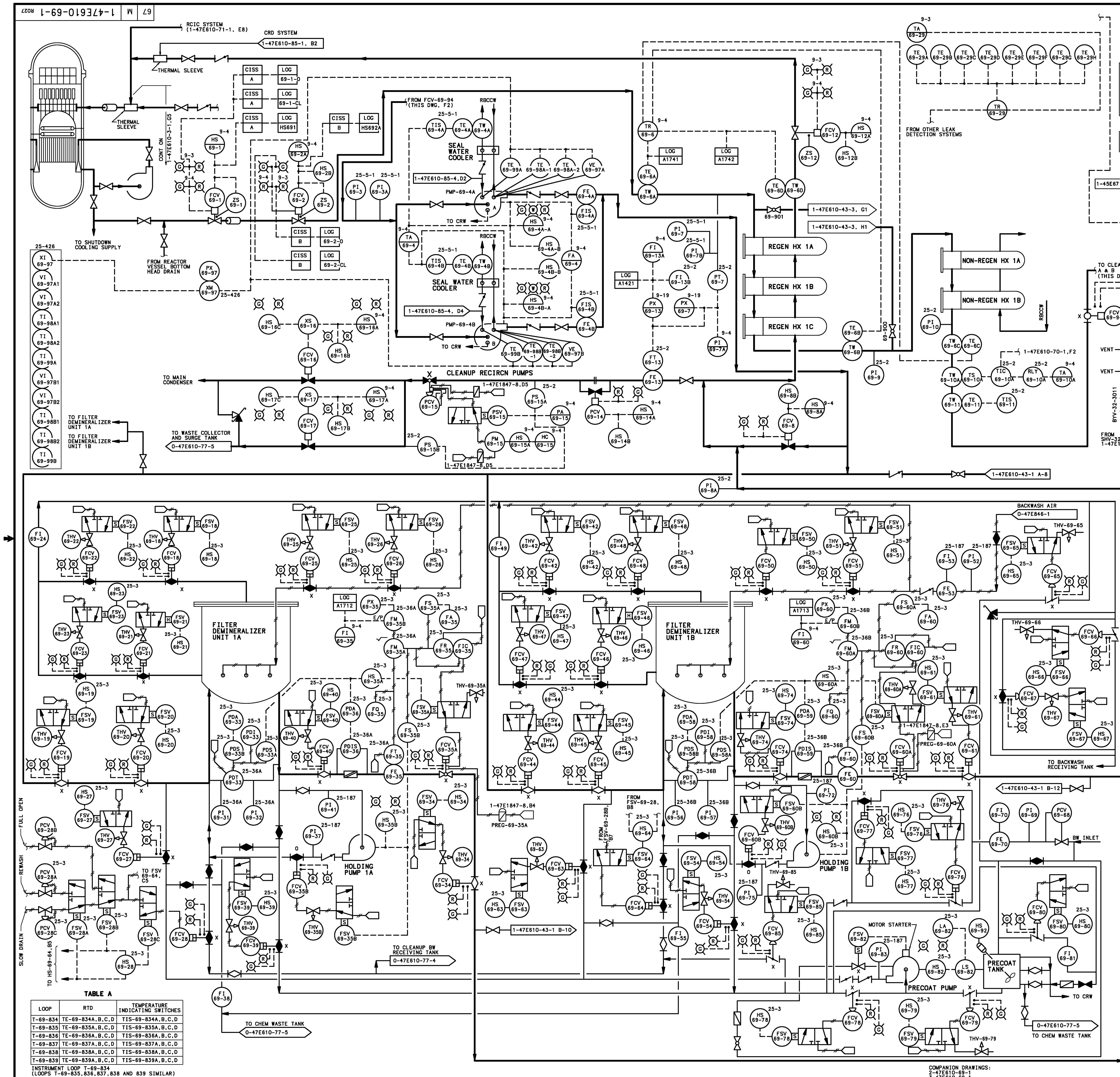
AMENDMENT 28

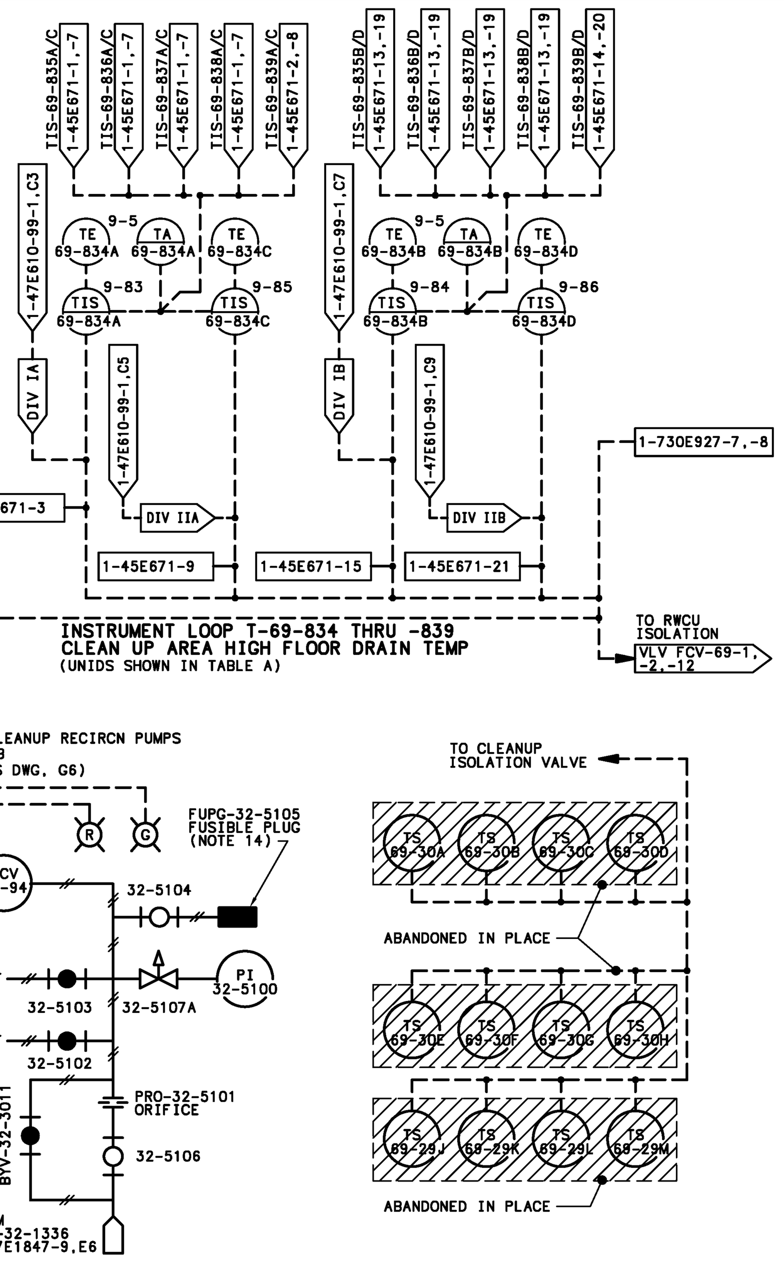
POWERHOUSE
UNIT 1

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

REACTOR WATER CLEANUP
DEMINERALIZER
FLOW DIAGRAM

FIGURE 4.9-9





NOTES:

1. THE REACTOR WATER CLEANUP SYSTEM PROVIDES FOR CONTINUOUS MECHANICAL AND CHEMICAL FILTRATION AND DEMINERALIZATION OF A PORTION OF THE REACTOR WATER TO QUALITY SPECIFICATION. IMPURITIES ARE REMOVED BY FILTRATION, ABSORPTION, AND ION EXCHANGE.
2. WATER FLOWS FROM THE REACTOR THROUGH THE CLEANUP RECIRCULATING PUMPS, REGENERATIVE HEAT EXCHANGERS, NONREGENERATIVE HEAT EXCHANGERS, FILTER DEMINERALIZER, THEN BACK THROUGH THE REGENERATIVE HEAT EXCHANGERS TO THE REACTOR VESSEL.
3. BACKWASH AND PRECOAT OPERATIONS ARE MANUALLY INITIATED AND AUTOMATICALLY CONTINUED. DURATION OF EACH OPERATION WILL BE ADJUSTED IN THE FIELD BASED UPON SYSTEM REQUIREMENTS.
4. FCV-69-1 AND -2 ARE ISOLATION VALVES THAT MAY BE OPERATED MANUALLY FROM THE MAIN CONTROL ROOM. THE VALVES ARE AUTOMATICALLY CLOSED BY THE FOLLOWING SIGNALS:
   (A) ACTUATION OF STANDBY LIQUID CONTROL SYSTEM
   (B) HIGH TEMPERATURE UPSTREAM OF THE FILTER-DEMINERALIZER
   (C) HIGH TEMPERATURE IN AREAS OCCUPIED BY CLEANUP SYSTEM EQUIPMENT ACTUATED BY TS-69-29J, K, L OR M
   (D) HIGH TEMPERATURE IN FLOOR DRAINS IN AREAS OCCUPIED BY CLEANUP SYSTEM EQUIPMENT ACTUATED BY TS-69-30A, B, D, E, F, G OR H
   (E) REACTOR WATER LOW LEVEL
5. THE RECIRCULATION PUMPS ARE PROTECTED FROM OVERHEATING BY TEMPERATURE SWITCHES IN THE COOLING WATER DISCHARGE (TIS-69-4A AND -4B) WHICH TRIP THE PUMP AND ALARM ON HIGH COOLING WATER TEMPERATURE. FLOW SWITCHES AT EACH PUMP DISCHARGE (FS-69-4A AND 4B) MONITOR FLOW TO TRIP THE PUMPS UPON LOW FLOW AND ALARM (FA-69-4A AND -4B).
6. TEMPERATURE ACROSS THE HEAT EXCHANGERS IS MONITORED BY TI-69-6.
7. PRESSURE DROP ACROSS THE FILTER DEMINERALIZERS IS MONITORED BY (PDT-69-33 AND -58) WHICH ACTUATES AN ALARM (PDA-69-33 AND -58) AND CLOSES THE DISCHARGE VALVES (FCV-69-35A AND -60A) WHEN THE DIFFERENTIAL PRESSURE EXCEEDS 25 PSIG. A MANUAL SWITCH (HS-69-35A AND -60A) IS ALSO PROVIDED FOR MANUAL ISOLATION OF THE FILTER DEMINERALIZER VESSEL.
8. LOW FLOW FROM A DEMINERALIZER (FE-69-35) WILL START HOLDING PUMP 1A BY ACTUATING FS-69-35B, OPENING VALVE FCV-69-35B, CLOSING FCV-69-35A, AND SOUNDING ALARM FA-69-35. THE OPERATION OF HOLDING PUMP 1B IS SIMILAR.
9. ONE CLEANUP PRECOAT SYSTEM SERVES BOTH DEMINERALIZER UNITS. A LEVEL SWITCH (LS-69-82) PROVIDES HIGH AND LOW ALARM AND LOW LEVEL PUMP INTERLOCK PROTECTION.
10. TIS-69-11 WILL ISOLATE THE SYSTEM ON HIGH DISCHARGE TEMPERATURE. THE COMPARTMENT FLOOR DRAIN LEAK DETECTION ELEMENTS TS-69-30A THROUGH -30H CAN BE SET TO ISOLATE THE SYSTEM AT APPROXIMATELY 20° ABOVE AMBIENT.
11. FOR PANEL LOCATIONS OF PILOT LIGHTS FOR CONTROL VALVES, SEE THE PANEL NUMBER FOR THE ASSOCIATED HAND SWITCH.
12. 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 (REFERENCE GE DESIGN SPEC 22A1470).
13. UNIDS ON DRAWING ARE FOR REFERENCE ONLY AND ARE ABBREVIATED TO MEET SPACE CONSTRAINTS. REFER TO MEL FOR COMPLETE UNIDS.
14. THE FUSIBLE PLUG IS A THREADED PIPE END WITH A LOW TEMPERATURE ALLOY EUTECTIC MATERIAL WHICH IS ATTACHED TO THE RWCU PIPE UPSTREAM OF FCV-69-94. THE EUTECTIC MATERIAL WILL MELT ON HIGH TEMPERATURE, VENTING THE CONTROL AIR LINE WHICH CLOSES ISOLATION VALVE FCV-69-94.

REFERENCE DRAWINGS:
1-47W406-1..................REACTOR WATER CLEANUP SYSTEM PIPING
47W406-2..................REACTOR WATER CLEANUP SYSTEM PIPING
0-47W406-3..................REACTOR WATER CLEANUP SYSTEM PIPING
MEL......................INSTRUMENT TABULATION (SYSTEM 69)
0-47E800-1................FLOW DIAGRAM - GENERAL PLANT SYSTEMS
0-47E800-2................MECHANICAL SYMBOLS AND FLOW DIAGRAM DRAWING INDEX
1-47E810-1................FLOW DIAGRAM - REACTOR WATER CLEANUP SYSTEM
0-47E837-1................FLOW DIAGRAM - REACTOR WATER CLEANUP DEMINERALIZER
0-47E846-1................FLOW DIAGRAM - DEMINERALIZER BACKWASH AIR.
1-45E615 SERIES..........CONTAINMENT ISOLATION STATUS SYSTEM (CISS)

GRAVER WATER CONDITIONING CO.
T-12728..............FLOW DIAGRAM

GE DRAWINGS:
730E719..............FUNCTIONAL CONTROL DIAGRAM

TABLE A

LOOP	RTD	TEMPERATURE INDICATING SWITCHES
T-69-834	TE-69-834A,B,C,D	TIS-69-834A,B,C,D
T-69-835	TE-69-835A,B,C,D	TIS-69-835A,B,C,D
T-69-836	TE-69-836A,B,C,D	TIS-69-836A,B,C,D
T-69-837	TE-69-837A,B,C,D	TIS-69-837A,B,C,D
T-69-838	TE-69-838A,B,C,D	TIS-69-838A,B,C,D
T-69-839	TE-69-839A,B,C,D	TIS-69-839A,B,C,D

INSTRUMENT LOOP T-69-834
(LOOPS T-69-835,836,837,838 AND 839 SIMILAR)

AMENDMENT 28

POWERHOUSE
UNIT 1

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

REACTOR WATER CLEANUP
SYSTEM
MECHANICAL CONTROL DIAGRAM

FIGURE 4.9-10

4.10 NUCLEAR SYSTEM LEAKAGE RATE LIMITS

4.10.1 Safety Objective

Nuclear system leakage rate limits are established so that appropriate action can be taken before the integrity of the nuclear system process barrier is unduly compromised.

4.10.2 Safety Design Basis

1. The nuclear system leakage rate limit shall be set so that corrective action can be taken:

    a. Before the nuclear system process barrier is threatened with significant compromise,

    b. Before the rate of leakage exceeds the coolant makeup capability, and

    c. Before the total leakage rate within the drywell exceeds the capability for leakage removal from the drywell.

2. Means shall be provided for the detection of leakage rates so that corrective action can be taken before the integrity of the nuclear system process barrier is unduly compromised.

4.10.3 Description

This subsection describes the leakage detection systems which are provided to detect abnormal leakage from the nuclear system process barrier inside the primary containment. Also discussed in this subsection are nuclear system leakage rate limits and how they are established.

The systems that detect gross leakage resulting from a pipe rupture outside the primary containment (such as in the main steam lines, HPCI steam line, RCIC steam line, and reactor water cleanup lines) and initiate automatic isolation are considered as part of the Primary Containment and Reactor Vessel Isolation Control System; and they are discussed in Section 7.3, "Primary Containment Isolation System."

Collection and processing of leakage outside the primary containment in the Reactor Building and in all other buildings are discussed in Section 10.16, "Equipment and Floor Drainage Systems," and Section 9.2, "Liquid Radwaste System."

Figures 4.10-1 and 9.2-3j are diagrams of the drywell leak detection system (except for the drywell leak detection radiation monitoring system) and drywell sumps, respectively. As shown in the figures, there are two drywell sumps. One

sump (drywell equipment drain sump) receives drainage from the pump seal leak-off, reactor vessel head flange vent drain, and other equipment drains. The second sump (floor drain collector sump) receives control rod drive, valve stem, and flange leakages, floor drains, closed cooling water system drains and drywell cooling unit condensation. Collection of leakage in excess of normal background amounts is indicative of a process system leak. For anticipated leakage rates of equipment and specific piping paths, see Section 9.2, "Liquid Radwaste System."

Leaks within the primary containment are detected by: (a) increased pressure and temperature in the primary containment, (b) monitoring the flow in the equipment drain sump and floor drain sump, (c) monitoring the cooling water temperature to and from the drywell coolers, and (d) monitoring the drywell for airborne activity (7.14).

The drywell cooling system recirculates the drywell atmosphere through heat exchangers to maintain the drywell at its design operating temperature. With the drywell atmospheric coolers operating inside a sealed drywell, an abnormal temperature rise inside the drywell would indicate a coolant and/or steam leak.

The drywell leak detection radiation monitoring system consists of four sample points, two near the top of the spherical portion of the drywell, 180° apart, and two near the recirculation pumps, 180° apart. The two top samples are manifolded together and routed through one line in an instrument drywell penetration. The two lower samples are manifolded and routed through another line in the same instrument drywell penetration. Two automatic isolation valves are provided in series in each line outside the drywell. The two lines are manifolded together and routed to a radiation monitor. The sample return from the radiation monitor is provided with two automatic isolation valves and routed through another line in an instrument drywell penetration. The inlet and return automatic isolation valves close on primary containment isolation and are provided with override switches in the main control room.

Detection, identification, and measurement of leakage in the drywell have been separated into identified and unidentified leakage. Limits have been established for unidentified and total leakage inside the drywell. Total leakage is defined as the sum of the identified and unidentified leakage.

4.10.3.1 Identified Leakage Rate

The identified leakage rate is the sum of all component leakage rates that input into the drywell equipment drain sump.

The pump packing glands and other seals in systems that are part of the nuclear system process barrier, and from which normal design leakage is expected, are provided with drains or auxiliary sealing systems. The valves in Units 2 and 3 and

pumps in the Reactor Recirculation System inside the drywell are equipped with double seals. The pump suction and discharge valves of Reactor Recirculation System in Unit 1 are equipped with live load packing. Leakage from the primary recirculation pump seals is piped to the equipment drain sump as described in Section 4.3, "Reactor Recirculation System." Leakage from the main steam relief valves is identified by temperature sensors which transmit to the Main Control Room. Any temperature increase detected by these sensors above the drywell ambient temperature indicates valve leakage. Unambiguous Main Control Room indication and alarm of valve position is provided by use of an acoustic monitoring system on the main steam relief valve tailpipes. Leakage from the reactor vessel head flange gasket is piped to a collection chamber and then to the equipment drain sump. A more detailed discussion is presented in Section 7.8, "Reactor Vessel Instrumentation."

Thus, the leakage rates from pumps, valve seals, and the reactor vessel head seal are measurable during operation of the plant. These leakage rates, plus any other leakage rates that input into the drywell equipment drain sump, are defined as identified leakage rates.

4.10.3.2 Unidentified Leakage Rate

The unidentified leakage rate is the rate at which leakage enters the drywell floor drain sumps. A threat of significant compromise to the nuclear system process barrier exists if the barrier contains a crack that is large enough to propagate rapidly. The unidentified leakage rate is limited because of the possibility that most of the unidentified leakage rate might be emitted from a single crack in the nuclear system process barrier.

A leakage rate of 150 gpm has been conservatively calculated to be the minimum liquid leakage from a crack large enough to propagate rapidly. An allowance for reasonable leakage that does not compromise barrier integrity, and is not identifiable, is made for normal plant operation.

The unidentified leakage rate limit is established at 5 gpm, which is far enough below the 150 gpm leakage rate to allow time for corrective action to be taken before the process barrier is significantly compromised.

Both the GE (GEAP-5260, Failure Behavior in ASTM A106B Pipes Containing Axial Through-wall Flows, by M. B. Reynolds, April 1968) and the BMI (Recent Work on Flow Behavior in Pressure Vessels, by A. R. Duffy, R. J. Eiber, and W. A. Maxey, April 1969); also, Quarterly Progress Reports, "Investigation of the Initiation and Extent of Ductile Pipe Rupture," by Eiber, et al, for the period May 1966 through 1969) test results indicate that theoretical fracture mechanics formulas do not predict critical crack length, but that satisfactory empirical expressions may be developed to

fit test results. A simple equation which fits the data in the range of normal design stresses (for carbon steel pipe) is as follows.

(1) Crack Length.

$$lc = \frac{15{,}000 \quad D}{\sigma_h}$$

(see data correlation on Figure 4.10-3),

where:

lc = critical crack length (inches)
D = mean pipe diameter (inches)
σ_h = nominal hoop stress (psi).

(2) Crack Opening Displacement. The theory of elasticity predicts a crack opening displacement of

$$\omega = \frac{2 \quad 1\sigma}{E}$$

where:

l = crack length
σ = applied nominal stress
E = Young's Modulus.

Measurements of crack opening displacement made by BMI show that local yielding greatly increases the crack opening displacement as the applied stress approaches the failure stress σf. A suitable correction factor for plasticity effects is

$$c = \sec\left(\frac{\pi}{2} \bullet \frac{\sigma}{\sigma_f}\right)$$

The crack opening area is given by

$$A = C \frac{\pi}{4} \omega 1 = \frac{\pi 1^2 \sigma}{2E} \sec\left(\frac{\pi}{2} \times \frac{\sigma}{\sigma_f}\right)$$

For a given crack length l, sf = 15,000 D/l.

(3) Leakage Flow Rate. The maximum flow rate for blowdown of saturated water at 1000 psi is 55 lb/sec-in.^{2} and for saturated steam the rate is 14.6 lb/sec-in.^{2} (APED-4827, Maximum Two-Phase Vessel Blowdown from Pipes, by F. J. Moody, April 1965). Friction in the flow passage reduces this rate, but for cracks leaking at 15 gpm (2.08 lb/sec), the effect of friction is small. The required leak size for 15 gpm flow is

A = 0.038 in.^{2} (saturated water)

A = 0.143 in.^{2} (saturated steam).

From this mathematical model, the critical crack length and the 15 gpm crack length have been calculated for representative BWR pipe sizes (Schedule 80) and pressure (1050 psi). The lengths of through-wall cracks that would leak at the rate of 15 gpm as a function of nominal pipe size are as follows.

Nominal Pipe Size (Sch 80)	Critical Crack Length (inches)	15 gpm Crack Steam Line	Length(inches) Water Line
4	9.6	8.4	6.7
12	19.6	12.1	7.5
24	34.8	14.0	7.8

It is important to recognize that the failure of ductile piping with a long, through-wall crack is characterized by large crack opening displacements which precede unstable rupture. Judging from observed crack behavior in the GE and BMI experimental programs, involving both circumferential and axial cracks, it is estimated that leak rates of hundreds of gpm will precede crack instability. Measured crack-opening displacements for the BMI experiments were in the range of 0.1 to 0.2 inches at the time of incipient rupture, corresponding to leaks of the order of one square inch in size for plain carbon steel piping. For austenitic stainless steel piping, even larger leaks are expected to precede crack instability, although there is insufficient data to permit quantitative prediction.

4.10.3.3 Total Leakage Rate

Total leakage rate consists of all leakage, identified and unidentified, which flows to the drywell floor drain and equipment drain sumps.

The criterion for establishing the total leakage rate limit is based on the makeup capability of the Control Rod Drive (CRD) and the RCIC systems which are independent of the feedwater system, normal AC power for two of the five CRD pumps, and the Core Standby Cooling Systems. The CRD system supplies makeup into the reactor vessel; the RCIC system can supply 600 gpm through the feedwater sparger to the reactor vessel. The total leakage rate limit is established at 30 gpm, which is substantially below the minimum normal inflow of the CRD System.

The total leakage rate is also set low enough to prevent overflow of the drywell sumps. The equipment drain sump (capacity 1,000 gallons) and the floor drain sump (capacity 1,000 gallons), which collect all leakage, are each drained, when required, by operation of a single pump throttled to operate at approximately 50 gpm. Dual sump pumps are available in each sump for redundancy. The total leakage rate limit is set below the removal capacity of a single pump in each sump because of the possibility that most of the total leakage could flow into one sump.

Each sump has an alarm system and automatic pump-starting sequence as follows. At the first high-water-level setting, the preselected pump is automatically started. If the water level continues to rise, a higher water-level setting starts the standby pump and actuates an alarm. The pumps are alternately selected for operation by an automatic pump-selector switch. The alarm indicates that leakage into that sump is equal to, or is exceeding, the capacity of one pump or that the preferred pump failed to start.

PCIS Isolation Valves: 1/2/3-FCV-77-2B, Drywell Floor Drain Sump Outboard Isolation Valves, and 1/2/3-FCV-77-15B, Drywell Equipment Drain Sump Outboard Isolation Valve are maintained in the closed position. With these valves now closed, the Sump Pumps auto start will be inhibited. There will not be an automatic high sump level start of the DW Floor Drain Sump pumps or the Unit 3 DW Equipment Drain Sump Pumps. Operator action will be required on high level and temperature to initiate start of pumps.

The flow integrators are combined with the flow recorder and presented as separate output channels on the flow recorder. Total leakage rate is periodically calculated from these flow integrators. A flow recorder continually plots time-versus-discharge flow rate from each sump; an increase in leakage rate is detectable by an increase in sump-discharge flow time and an increased frequency in discharge flow cycles. Increases in total leakage rate are also detectable from records kept of flow integrator readings.

A pump running timer records the actual amount of time each sump pump runs. By utilizing the known capacity of the sump pump and a pump-run-time, real time comparison on average leakage rate is established. If this average leakage rate exceeds a pre-established limit, an alarm sounds in the control room. The drywell equipment drain sump timer does not perform this function when there is a high water level coupled with persistent high temperature. The pump stays in recirculation mode when not discharging and the alarm does not indicate excessive leakage rate.

4.10.4 Safety Evaluation

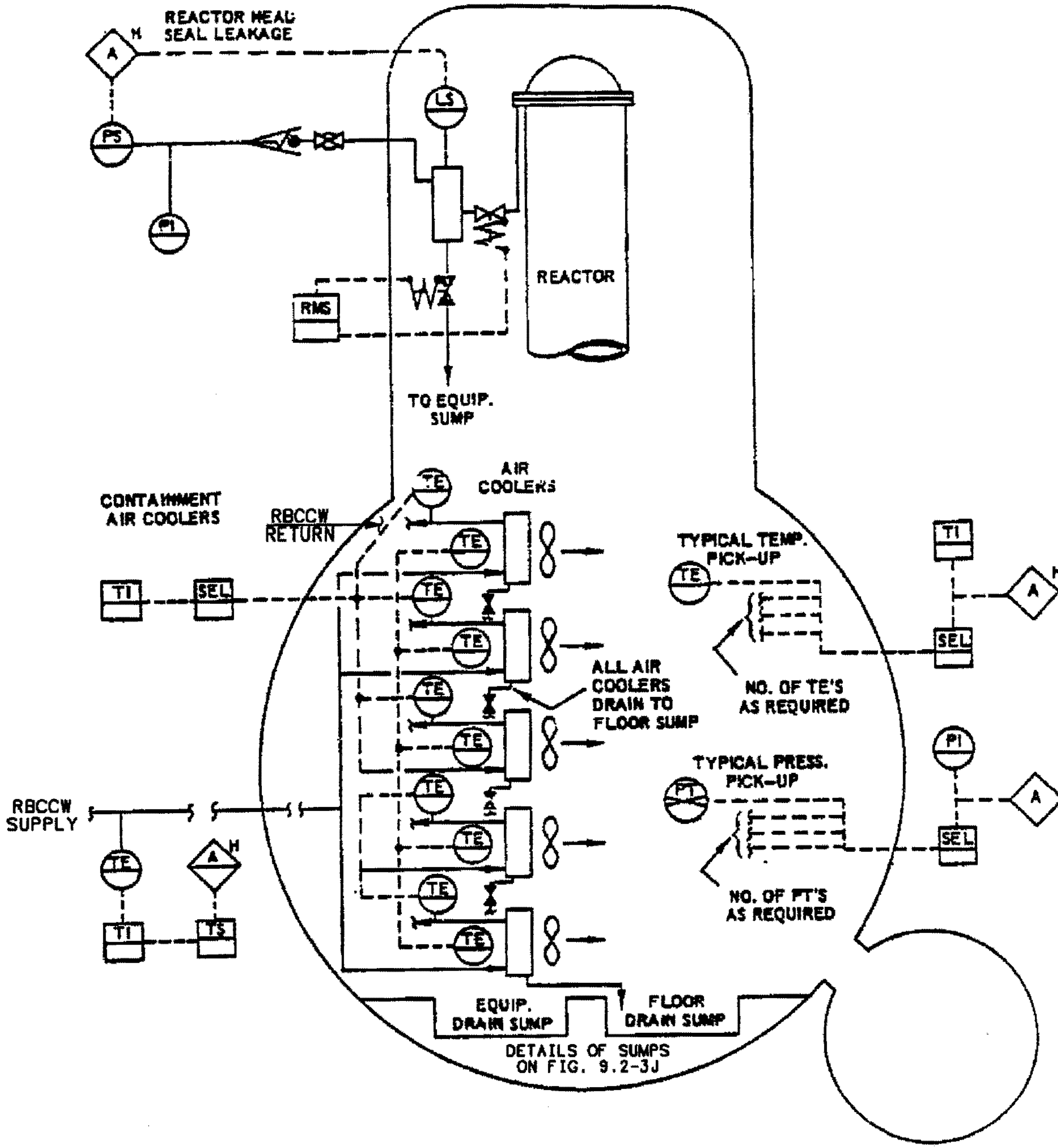
The unidentified leakage rate limit is based, with an adequate margin for contingencies, on the crack size large enough to propagate rapidly. The established limit is sufficiently low so that, even if the entire unidentified leakage rate were coming from a single crack in the nuclear system process barrier, corrective action could be taken before the integrity of the barrier is threatened with significant compromise.

The limit on total leakage rate is established so that in the absence of normal AC power and feedwater, and without using the Core Standby Cooling Systems, the leakage loss from the nuclear system could be replaced. The CRD system furnishes normal makeup and the RCIC system can furnish 600 gpm to the reactor vessel, both of which are independent of feedwater. The RCIC and two of five CRD pumps for the plant are independent of normal AC power. The limit on total leakage also allows a reasonable margin below the discharge capability of either the floor drain or equipment drain sump pumps. Thus, the established, total-leakage rate limit allows sufficient time for corrective action to be taken before either the nuclear system coolant makeup or the drywell sump removal capabilities are exceeded. Safety design basis 1 is therefore satisfied.

A discussion of the leakage detection instrumentation is provided in the description. This information shows that means are provided for the detection of leakage so that corrective action can be taken before the integrity of the nuclear system process barrier is unduly compromised. This provision satisfies safety design basis 2.

4.10.5 Inspection and Testing

The pumps and controls are periodically inspected and tested to verify proper operation and instrument operability. Readings from the drywell sump and radiation monitoring systems are checked and recorded as appropriate based on requirements of Technical Specifications, Sections 3.4.4 and 3.4.5.



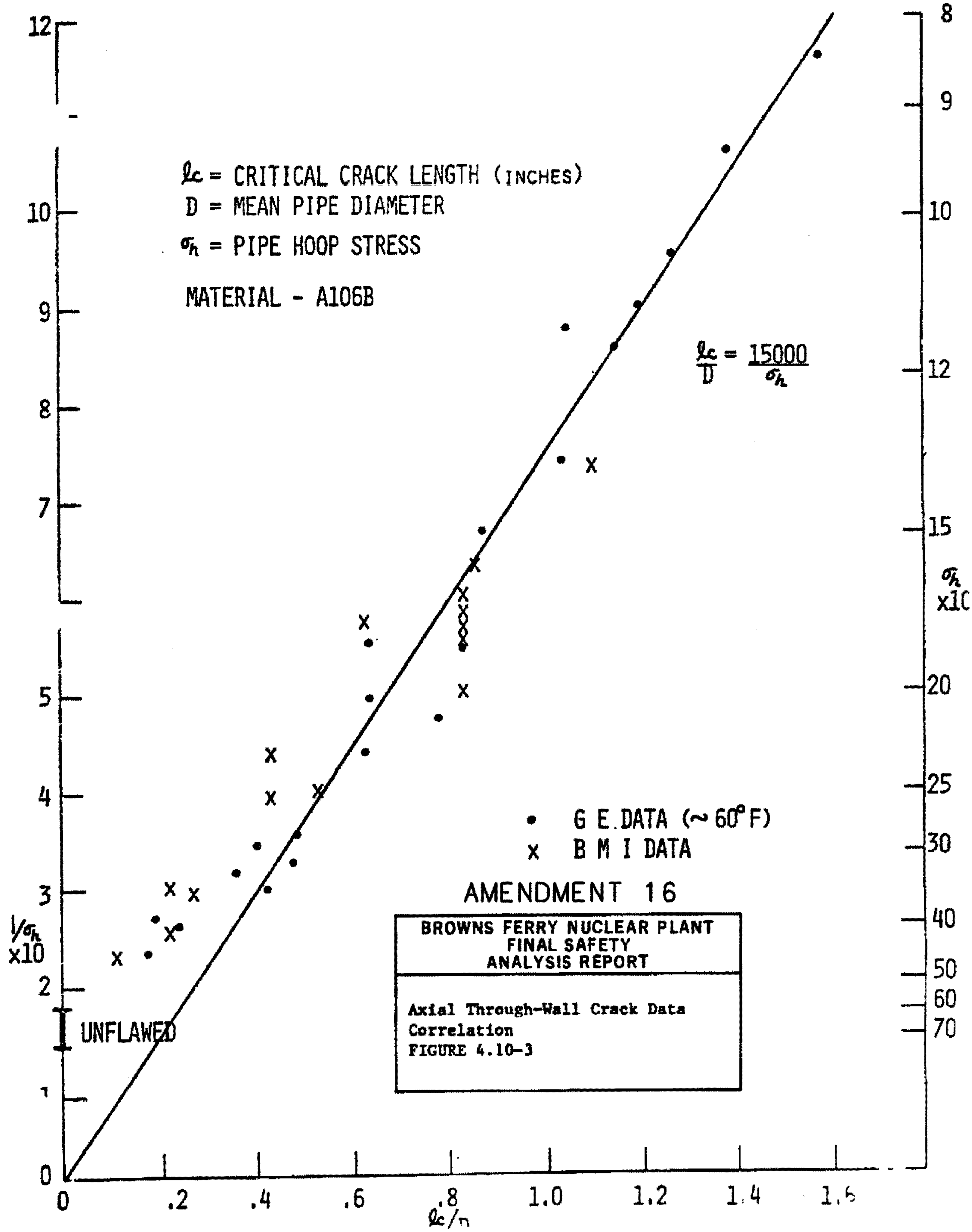
AMENDMENT 17

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

DRYWELL LEAK DETECTION SYSTEM DIAGRAM

FIGURE 4.10-1

Figure 4.10-2
(Deleted by Amendment 16)



AMENDMENT 16

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Axial Through-Wall Crack Data Correlation
FIGURE 4.10-3

4.11 Main Steam Lines, Feedwater Piping, and Drains

4.11.1 Power Generation Objective

The power generation objective of the main steamlines is to conduct steam from the reactor vessel through the primary containment to the steam turbine. The power generation objective of the feedwater lines is to provide a piping path for delivery of water through the primary containment to the reactor vessel.

4.11.2 Safety Design Basis

1. The main steam and feedwater lines shall be designed to accommodate operational stresses, such as internal pressures, without a failure which could lead to a release of radioactivity in excess of the guideline values in 10 CFR 50.67.

2. The main steam and feedwater lines within the primary containment shall be designed to withstand the effects of an earthquake without a failure which could lead to a release of radioactivity in excess of the guideline values in 10 CFR 50.67.

4.11.3 Power Generation Design Basis

1. The main steam and feedwater lines shall be designed to allow inservice testing and inspections.

2. The main steam lines shall be designed to conduct steam from the reactor vessel over the full range of reactor power operation.

3. The feedwater piping shall be designed to conduct water to the reactor vessel over the full range of reactor power operation.

4.11.4 Description

The feedwater piping is designed to conduct water from sources outside the primary containment to the reactor vessel. The general requirements of the feedwater system are covered in Subsection 7.10, "Feedwater Control System," and 11.8, "Condensate and Reactor Feedwater Systems." All main steam and feedwater piping are classified according to service and location. A diagram of the feedwater piping is shown in Figure 4.11-1.

The main steam piping is designed to conduct steam from the reactor vessel through the primary containment to the steam turbine. Four steam lines are utilized between the reactor and the turbine. The use of these multiple lines permits turbine stop valve and main steam isolation valve tests during plant operation with a

minimum amount of load reduction. To fully achieve this objective, the four steam lines are connected to a header upstream of the turbine stop valves. This header placement also ensures that the turbine bypass system is connected to the used steam lines and not to idle lines. A diagram of the main steam piping is shown in Figures 4.5-1, 4.5-2, and 4.5-3 of Subsection 4.5, "Main Steam Line Flow Restrictors."

Acoustic vibration suppressors are installed in 6-inch blind-flanged branch lines on the main steam piping in the drywell to minimize accoustic loading on the steam dryer and other components. The suppressor locations are shown in Figures 4.5-1 and 11.1-1e for Unit 1, Figures 4.5-2 and 11.1-1a for Unit 2, and Figures 4.5-3 and 11.1-1c for Unit 3.

Design and construction of pressure retaining piping and components of the Main Steam System and Feedwater System were initially in accordance with the requirements of USAS B31.1.0, 1967 Edition, as supplemented by the requirements of the applicable GE design and procurement specifications, which were implemented in lieu of the outdated B31 Nuclear Code Cases-N2, N7, N9, and N10. Quality control methods were used during the fabrication and assembly of main steam and feedwater piping to ensure that the design specifications were met.

A drain line is connected to the low points of each main steam line, both inside and outside the drywell. Both sets of drains are connected to a header and are connected by valving to permit drainage to the main condenser hotwell. A vent line is provided around the final valve to the condenser hotwell to permit continuous draining of the steam line low points. The inside steam line drains slope downward from the steam line low point to the orifice outside the drywell. The drain line from the orifice to the condenser hotwell slopes down to the main condenser. An additional drain is provided from the low point of the drains to clean-radwaste to permit purging the lines for maintenance. During operations only the outside drain valve is open allowing continuous drainage to the condenser through the orifice.

The inside and outside steam line drains are capable of being utilized to equalize pressure across the main steam isolation valves prior to restart following a steam line isolation. Assuming all steam line isolation valves have closed and the steam lines outside the drywell have been depressurized, the isolation valves outside the drywell are opened first; the drain lines are then used to warm up and pressurize the outside steam lines. Finally, the main steam isolation valves inside the drywell are opened.

Feedwater line breaks are isolated on the vessel inlet side by closure of the check valves. The break continues to be fed by feedwater until low level in the reactor vessel is detected and the low level instrumentation actuates the main steam isolation valves and the valves then close. Temperature sensors are strategically located in the steam tunnel near the feedwater lines in order to sense any rise in

ambient temperature. These sensors also initiate closure of the main steam isolation valves if the ambient temperature in the steam tunnel rises too high.

4.11.5 Safety Evaluation

Differential pressures on reactor internals under the assumed accident conditions of a ruptured steam line are limited by both the utilization of flow restrictors and the utilization of four main steam lines. Main steam and feedwater piping are designed in accordance with the USAS B31.1.0, 1967 edition, Code for Power Piping which describes the primary and secondary allowable stresses associated with the main steam and feedwater piping. Design of piping in accordance with these requirements ensures the meeting of safety design basis 1. Safety design basis 2 is met by design of main steam and feedwater piping from the pressure vessel to the outside isolation valve to Class I specifications in accordance with the loading criteria of Appendix C.

4.11.6 Inspection and Testing

Prior to initial operation, the main steam and feedwater piping were inspected and tested in accordance with USAS B31.1.0, 1967 edition and the applicable GE design and procurement specifications, which were implemented in lieu of the outdated B31 Nuclear Code Cases-N2, N7, N9 and N10. Inservice inspection is considered in the design of the main steam and feedwater piping. This consideration assures adequate working space and access for inspection of selected components. Access requirements for inservice inspection are in accordance with the requirements of APED-5450, "Design Provisions for Inservice Inspection." Subsection 4.12 describes the inservice inspection and testing program.

Figure 14.11-1 through Figure 14.11-18

(Deleted)

4.12 INSERVICE INSPECTION AND TESTING

4.12.1 Introduction

The preservice examinations at Browns Ferry were conducted using the 1971 Edition, Summer 1971 Addenda, of ASME Section XI as a guideline. The inservice examinations performed during the first 40-month cycle were in accordance with the 1971 Edition, Summer 1971 Addenda, of ASME Section XI for Units 1 and 2; and the 1974 Edition, Summer 1975 Addenda, of ASME Section XI for Unit 3.

The three units at Browns Ferry were put on a concurrent inspection and pressure test cycle beginning July 1, 1980; therefore, all inservice examinations performed during the 80-month cycle and those exams to be performed during the 120-month cycle shall be performed in accordance with the 1974 Edition, Summer 1975 Addenda of ASME Section XI or as stated in 1-, 2-, and 3-SI-4.6.G, Browns Ferry Nuclear Plant Surveillance Instruction, Inservice Inspection Program, Codes of Record.

The inservice inspection and testing programs for vessels, piping, pumps, valves, and pressure tests shall be performed in accordance with Section XI of ASME Boiler and Pressure Vessel Code and the ASME Code for Operation and Maintenance of Nuclear Power Plants (ASME OM Code) and applicable addenda as required by 10 CFR 50, Section 50.55a except where specific written relief has been identified. Several ASME Code Cases have been approved for use in the inservice inspection program and utilization of code cases accepted by NRC Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability ASME Section XI, Division 1," as applicable to Browns Ferry.

The inservice pressure tests program for the first interval was prepared in accordance with the 1974 Edition, Summer 1975 Addenda of ASME Section XI for all units. The Inservice Pump and Valve Testing Program for the first interval was prepared in accordance with the 1980 Edition, Winter 1980 Addenda ASME Section XI for all units.

The inservice pressure test program, inservice examination program, and inservice pump and valve test program for subsequent intervals will be prepared in accordance with the edition of the ASME Section XI Code as specified by the 10 CFR 50.55a requirements at the time of the update, except where specific relief has been granted.

The examinations are performed using TVA and/or contractor personnel. Nondestructive examination personnel shall be certified in accordance with a program meeting the requirements of SNT-TC-1A and/or ANSI N45.2.6 and/or ANSI/ASNT CP-189, except where specific relief has been granted.

4.12.2 Scope

Periodic inservice examinations for Browns Ferry ASME Class 1, Class 2, and Class 3 components, and reactor vessel internals are defined in plant procedures. Periodic pump, valve and pressure testing in accordance with the applicable subsections of ASME Section XI and ASME OM Codes are defined in Plant Procedures.

The requirement for Inservice Inspection Examinations for Browns Ferry equivalent ASME Class MC became effective September 9, 1996, being incorporated as a result of ASME Section XI, Subsection IWE into 10 CFR 50.55(a).

4.12.3 Responsibility

TVA is responsible for the performance of the inservice inspections and tests outlined under the scope (4.12.2) of this program.

4.12.4 Area and Extent of Examination

The area and extent of examination are given in plant procedures.