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10.0 AUXILIARY SYSTEMS

10.1 SUMMARY DESCRIPTION

This section describes the reactor and plant auxiliary systems that are required for operation, but which are not integral portions of the reactor or power conversion equipment. The following systems are described:

New Fuel Storage
Spent Fuel Storage
Tools and Servicing Equipment
Fuel Pool Cooling and Cleanup System
Reactor Building Closed Cooling Water System
Raw Cooling Water System
Raw Service Water System
Residual Heat Removal Service Water System
Emergency Equipment Cooling Water System
Fire Protection Systems
Heating, Ventilating and Air-Conditioning Systems
Demineralized Water System
Control and Service Air Systems
Potable Water and Sanitary Systems
Equipment and Floor Drainage Systems
Process Sampling Systems
Plant Communications System
Lighting System
Auxiliary Boiler System.
Postaccident Sampling System

10.2 NEW FUEL STORAGE

10.2.1 Power Generation Objective

The objective of the new fuel storage arrangement is to provide specially designed dry, clean storage places for the new fuel assemblies when normal storage in the spent fuel pool is not utilized.

10.2.2 Power Generation Design Basis

1. New fuel storage racks shall be supplied to accommodate 30 percent of a full core load of fuel assemblies for each reactor.

2. New fuel storage racks shall be designed and arranged so that the fuel assemblies can be handled efficiently.

10.2.3 Safety Design Basis

1. The new fuel storage racks shall be designed and maintained with sufficient spacing between the fuel assemblies to assure that when the racks are fully loaded, the array shall be subcritical by a substantial margin.

2. The new fuel storage racks loaded with fuel assemblies shall be designed to withstand earthquake loadings to prevent damage to the structure of the racks and minimize distortion of the racks' arrangement.

10.2.4 Description

Each unit is provided with a new fuel storage vault located adjacent to the spent fuel pool as shown in Figures 1.6-2 and 1.6-11. The new fuel storage racks provide a storage place in the new fuel storage vaults for new fuel. Each new fuel storage rack (shown in Figure 10.2-2) holds up to 10 new channeled or unchanneled fuel assemblies in a row spaced 6.625 inch apart center-to-center. The racks are designed so that rack arrangement in rows on an 11-inch center-to-center spacing will limit the effective multiplication factor of the array (k_{eff}) to not more than 0.90. New fuel storage racks are provided for 30 percent of the reactor core load. The fuel assemblies are loaded into the rack through the top. Each hole for a fuel assembly has adequate clearance for inserting or withdrawing the assembly while enclosed in a protective plastic wrapping. Sufficient guidance is provided to preclude damage to the fuel assemblies. Guides are provided to guide the spacers of the fuel elements for the full length of their insertion into the rack. The design of the racks prevents accidental insertion of the fuel assembly in a position not intended for the fuel. The weight of the fuel assembly is supported at the bottom and the rack provides a full horizontal support of the new fuel assembly. Removable

gratings (10-7/8 inch by 6 ft 8 inch) over each fuel rack and permanent gratings at the end spaces of the new fuel storage vault are provided to minimize the number of uncovered assemblies. These gratings can withstand loads of 100 lbs/ft^2. The vaults are provided with steel and concrete covers one foot thick. The vaults are provided with adequate drainage to prevent water collection and flooding. Each new fuel storage rack loaded with fuel is designed as a Seismic Class I structure (see Appendix C) to resist sufficiently the response motion at the installed location within the supporting structure for the design basis earthquake. Information on radiation monitoring of the new fuel storage vaults is provided in Subsection 7.13, "Area Radiation Monitoring System." Each vault is to be provided with neutron dosimeters whenever new fuel is stored there.

10.2.5 Safety Evaluation

BFN has chosen to comply with the criticality requirements specified in 10 CFR 50.68(b). New fuel is not placed in the New Fuel Storage Racks until a criticality analysis of optimum moderator configuration is performed. (Reference TVA to NRC letter dated July 21, 1997, R08970721849)

Stresses in a fully loaded rack are designed not to exceed applicable specification requirements of the American Institute of Steel Construction or the American Society of Civil Engineering when subjected to a horizontal earthquake load of 1.50g applied in any direction. A safety factor of two, based upon the material yield or local critical buckling, is used where these specifications are not applicable.

The storage rack structure is designed to absorb an impact energy of at least 7,000 ft-lbs on an impact surface no larger than 3 inches in diameter. Under this impact force, those members will remain intact whose function it is to physically maintain the abnormal design subcritical spacing to assure that k_{eff} will not exceed 0.95.

The storage racks are designed to withstand a pull-up force equal to 4,000 lbs (this is necessary in the event that the fuel assembly or grappling device accidentally becomes fouled during removal). The stress in those members required to maintain the abnormal design subcritical spacing will not exceed 75 percent of the material's yield strength or 75 percent of that stress at which local buckling occurs.

The new fuel racks are restrained by hold-down lugs to assure that rack spacing does not vary under specified earthquake loads. The hold-down bolts restrain the rack in case a stuck fuel assembly is inadvertently hoisted. Each hold-down bolt is designed to withstand 500 lb horizontal shear and uplift force of 5,000 lbs. All materials used in the construction of the new fuel storage racks are specified in accordance with the applicable ASTM specifications, and all welds are in

accordance with the AWS standards for materials used. Materials selected are corrosion resistant or treated to provide the necessary corrosion resistance.

Figure 10.2-1a and 1b

(Deleted by Amendment 17)

SECURITY RELATED INFORMATION
FIGURE WITHHELD UNDER 10 CFR 2.390

UNIT 0

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

NEW FUEL STORAGE RACK
FIGURE 10.2-2

10.3 SPENT FUEL STORAGE

BFN has chosen to comply with the criticality requirements specified in 10 CFR 50.68(b). This section presents a description of the spent fuel storage facilities presently installed at the plant. All of the original low density spent fuel storage racks have been removed from the spent fuel pools. Due to projected shortages of storage space, high density spent fuel storage racks were installed to replace the original low density racks. With the installation of high density racks in the existing spent fuel pools, the storage capacity of each fuel pool was increased from 1080 to 3471 fuel assemblies providing space for storage of approximately four-and-one-half full core loads.

10.3.1 Power Generation Objective

The objective of the spent fuel storage arrangement is to provide specially-designed underwater storage space for the spent fuel assemblies, which require shielding and cooling during storage and handling.

10.3.2 Power Generation Design Basis

1. Spent fuel storage racks for each reactor shall be supplied to accommodate 454 percent of the full core load, or 3,471 fuel assemblies.

2. Spent fuel storage racks shall be designed and arranged so that the fuel assemblies can be efficiently handled during refueling operations.

10.3.3 Safety Design Basis

1. The fuel array in fully loaded spent fuel racks shall be subcritical by a substantial margin.

2. Each spent fuel storage rack containing fuel shall be designed to withstand earthquake loading so that distortion of the spent fuel storage arrangement will be within acceptable bounds.

10.3.4 Description

There are three spent fuel storage pools (SFSPs), one per reactor. The spent fuel storage racks provide a storage place at the bottom of each fuel pool for the spent fuel received from the reactor vessel, as shown in Figures 1.6-2 and 1.6-11 of Subsection 1.6. The racks are full length, top entry, and are designed to maintain the spent fuel in a spatial geometry that precludes the possibility of criticality under normal and abnormal conditions. Normal conditions exist when the spent fuel is stored at the bottom of the fuel pool in design storage position. Abnormal conditions may result from seismic forces or mishandling of equipment.

The high-density storage rack design is free-standing, transferring shear forces to the pool slab through friction resistance provided by the normal force of the weight of the module through the support columns to support pads resting on the pool floor liner.

The high-density storage racks (see Figures 10.3-1, 10.3-2, and 10.3-3) are made up of staggered, stainless-steel container tubes. Thus, there is only one container-tube wall between adjacent spent fuel assemblies. Each tube wall has a core of Boral sandwiched between .036-inch-inside and 0.090-inch-outside stainless steel containers. The tubes are about 14 feet long and have a square cross section with an outer dimension of 6.653 inches and a total wall thickness of .2015 inches. The nominal pitch between fuel assemblies is 6.563 inches.

The Boral core is made up of a central segment of approximately 0.056-inch-thick dispersion of boron carbide in aluminum. This central segment is clad on both sides with 0.010 inches of aluminum. The minimum homogeneous concentration of the boron-10 isotope is 0.013 grams per square centimeter of the Boral plate. The Boral plates are sandwiched between the two stainless steel containers, which are closure-welded. (Vent holes have been added to these storage tubes to prevent the buildup of hydrogen gas between the stainless steel containers.) The completed storage tubes are fastened together by angles welded along the corners and attached to a base plate to form storage modules.

Spent fuel assemblies are stored both within the tubes and in the spaces between the tubes. Two module sizes are used in the Browns Ferry SFSPs, a 13 x 13 module that can store a total of 169 fuel assemblies (85 in tubes and 84 in spaces outside the tubes) and a 13 x 17 module that can store 221 assemblies (111 in tubes and 110 in spaces outside the tubes). Each SFSP contains fourteen of the 13 x 13 modules and five of the 13 x 17 modules.

Storage is provided for canned defective fuel and used control rods in each SFSP. Installation of the high-density storage racks provided five extra positions in Unit 2 pool for storage of defective fuel. Control rod storage will be provided by supplying 20 permanent storage locations in the Units 1 and 2 SFSPs and 18 locations in the Unit 3 SFSP, and an aggregate of 370 temporary storage locations.

A transfer canal is provided to join the dual pools of Units 1 and 2. This transfer canal is the same depth as the transfer slot between the reactor well and the fuel pool. The transfer canal has a gate at each end so that the fuel pools can be isolated if necessary. The transfer canal can also be drained if necessary. The canal can be used for maintenance, repairs, etc., and enables fuel assemblies or channels to be transferred between the Unit 1 and Unit 2 spent fuel pools. It also permits storage of fuel and irradiated objects in one pool if it becomes necessary to empty the other pool. A diamond plate walkway is provided for personnel crossing,

and is removable in small, lightweight sections. A further description of the physical characteristics of the reactor well and its relationship to the spent fuel pool is given in Subsection 4.2, "Reactor Vessel and Appurtenances Mechanical Design," and Subsection 5.2, "Primary Containment System."

The spent fuel storage facilities are shared only for Units 1 and 2, and the sharing feature is only the transfer canal that connects the two storage pools. A watertight gate is provided at each end of the transfer canal.

10.3.5 Safety Evaluation

10.3.5.1 High-Density Spent Fuel Storage Racks

A sliding analysis of the free-standing high-density storage racks was performed. A minimum value for the coefficient of friction was used in the sliding analysis, a value that was verified by tests of steel materials, and any variations in the coefficient are covered by the conservatively-low value used. The coefficient of friction used was sufficient to ensure that only small sliding will occur for earthquake motions corresponding to OBE and SSE. An additional non linear analysis for sliding was performed to determine relative displacements if the coefficient of friction were less than the minimum value used. This analysis gives added assurance that there should be no interaction between modules as a consequence of the SSE.

TVA reevaluated the fuel pool structural capacity for the High Density Fuel Storage System (HDFSS) and determined that the existing pool structure is capable of supporting the increased load with a margin of safety within the limits of the ACI 318-71 code.

Structural integrity of the HDFSS has been demonstrated for the design basis load combinations using elastic design methods. The original vendor analysis is documented per Reference 3.

10.3.5.2 Criticality Evaluation for Fuel

Criticality analyses of the spent fuel pools have been performed to accommodate all fuel bundles in the spent fuel storage pool. Calculations were based on a bounding reference assembly defined by U-235 enrichment / Gadolinia concentration zones separated at geometry transitions. Analyses were constructed such that the assumed reactivity levels bound previous fuel products present the spent fuel pool. Requirements for
U-235 enrichment and Gadolinia concentration levels are discussed per References 1 and 2.

The CASMO-4 bundle depletion code is used to calculate k_∞ values for the ATRIUM-10 fuel assembly lattices as a function of exposure and void history for both in-core and in-rack geometries. CASMO-4 is a multigroup, two-dimensional transport theory code with an in-rack geometry option, where typical storage array geometries can be defined.

The spent fuel storage rack assembly calculations were performed with the KENO.Va Monte Carlo code, which is part of the SCALE 4.4 Modular Code System. Cross section data input to KENO.Va were taken from the 44 energy group data library and adjusted using the BONAMI and NITAWL codes to perform resonance corrections, using standard SCALE 4.4 methodology to account for resonance absorption in the uranium.

These computer codes were used to calculate the neutron multiplication factor for an infinite array of fuel assemblies with the following assumptions to ensure that the actual reactivity will always be less than the calculated reactivity:

a) Moderator temperature of 4°C, which gives the highest reactivity for the fuel storage pool.

b) Fuel assemblies are assumed to contain the high reactivity reference bounding lattices for the entire length of the assembly.

c) Fuel is assumed to be commercial uranium.

d) Each lattice in each fuel assembly in the array is assumed to be at the peak reactivity in its lifetime.

e) The most limiting orientation or position of each assembly in its rack cell is accounted for.

f) The analysis takes into account storage with or without fuel channels and channel type.

g) Neutron absorption in fuel structural components is neglected.

h) The maximum reactivity values include all significant manufacturing and calculational uncertainties.

i) The analysis uses a conservative reactivity equivalent beginning of life (REBOL) assembly, defined by U-235 enrichment / Gadolinia concentration zones separated by geometry transitions.

j) A minimum Boron-10 density 0.0130 $B^{10}g/cm^2$ is assumed when modeling the Boral plates.

k) No Boral plates are missing in the normal condition; 1 Boral plate per rack is assumed to be absent in the accident condition, even though no identified scenario can generate such a condition.

Utilizing the REBOL lattice which has a higher reactivity than the maximum reactivity of the bounding bundle and including uncertainties, biases, and the worst accident the resulting k_{eff} is less than or equal to the regulatory limit of ≤ 0.95 at a 95% confidence.

10.3.5.3 Common Safety Evaluation

Each spent fuel storage rack and cell containing fuel is designed as a seismic Class I structure to resist sufficiently the response motion at the installed location within the supporting structures for the Design Basis Earthquakes (OBE and SSE).

Stress in a fully loaded rack will not exceed allowable stresses when subjected to Design Basis Earthquake loads applied in any direction.

All materials used in the construction of the rack are specified in accordance with applicable ASTM specifications, and all welds are in accordance with the AWS standards for materials used. Materials selected are corrosion resistant.

See Section 10.5.5 for discussion on the fuel pool liner and associated piping.

10.3.6 Inspection and Testing

The neutron attenuation of each tube in each rack was tested prior to installation to detect if there were any Boral plates missing from the prescribed locations in the fabricated fuel storage modules.

TVA has committed to install corrosion test specimens in the Browns Ferry Unit 3 SFSP. These specimens will be periodically removed and examined to check the long-term behavior of the rack materials.

REFERENCES

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

2. ANP-2945(P), Revision 1, Browns Ferry Nuclear Plant Units 1, 2, and 3 Spent Fuel Storage Pool Criticality Safety Analysis, AREVA NP, Inc., July 2011.

3. NEDE-24076-P, Design Report and Safety Evaluation for High Density Fuel Storage System, Licensing Topical Report, General Electric, November 1977.

SECURITY RELATED INFORMATION
FIGURE WITHHELD UNDER 10 CFR 2 390

POWERHOUSE
UNIT 1

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

HDFSS ARRANGEMENT POOL #1

FIGURE 10.3-1

SECURITY RELATED INFORMATION
FIGURE WITHHELD UNDER 10 CFR 2.390

POWERHOUSE
UNITS 1-3

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

HIGH DENSITY SPENT FUEL RACK
FIGURE 10.3-2

SECURITY RELATED INFORMATION
FIGURE WITHHELD UNDER 10 CFR 2.390

POWERHOUSE
UNIT 3

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

HDFSS ARRANGEMENT POOL #3

FIGURE 10.3-3

10.4 TOOLS AND SERVICING EQUIPMENT

10.4.1 Introduction

All tools and servicing equipment necessary to meet the reactor general servicing requirements are supplied for efficiency and safe serviceability with a minimum of time. Table 10.4.1 is a listing of tools and servicing equipment supplied with the nuclear system. The following paragraphs describe the use of some of the major tools and servicing equipment.

10.4.2 Fuel Servicing Equipment

Two fuel preparation machines are located in each spent fuel storage pool. These machines are designed to be removed from the pool for servicing.

An equipment support railing is provided around the pool periphery in order to tie off miscellaneous equipment such as the fuel leak detector (sipper) and service tools. The fuel leak detector is used to locate fuel assemblies with perforated fuel pins by sampling water directly from the fuel assembly. Equipment lugs fabricated as part of the pool liner are provided for fixtures that might later be desired by plant operating personnel. In addition, a 4 x 4-inch curb with a 4-inch wide plate of 1-inch thick stainless steel on top is provided around the entire periphery of the refueling volume. Additional equipment may be mounted by welding to, or drilling into the plate. The curb may be used as an additional support or tie-off area. Cable ways are recessed into the floor around the pool periphery with openings to pass cables into the pool from underneath this curbing.

Two new fuel inspection stands are provided near the fuel storage pools on the operating floor to restrain the fuel assembly in vertical position for inspection. The inspection stand can hold two assemblies. The general purpose grapple is a small, hand-actuated tool used generally with fuel. The grapple can be attached to the reactor building auxiliary hoist, jib crane, and the auxiliary hoists on the refueling platform. The general purpose grapple is used to remove new fuel from the shipping containers or the vault, place it in the inspection stand, and transfer it to the fuel pool. It also can be used to shuffle fuel in the pool and to handle fuel during channeling.

A channel handling boom with a spring loaded takeup reel can be used to assist the operator in supporting the channel after it is removed from the fuel assembly and placed in channel storage rack. The boom is set between the two fuel preparation machines. With the channel handling tool attached to the reel, the channel may be conveniently moved between fuel preparation machines.

The complete channeling procedure is as follows. Using one of the jib cranes mounted in the area and the general purpose grapple, a spent fuel assembly is lifted into the fuel preparation machine with the carriage lowered. After raising the assembly to its high position, the channel is unbolted from the fuel assembly using the channel bolt wrench. This wrench guides on the channel, and is used to unscrew the bolt and capture it. The channel handling tool is attached to the channel handling boom and lowered to the channel. The tool is attached to the channel dog ears by expanding two fingers on the tool. The channel is then held, and the fuel preparation machine carriage is lowered causing the fuel assembly to slide down out of the channel. The channel will be placed back on the assembly after the inspection or will be replaced with a new channel if found to be damaged. A channel storage rack for accumulating channels is located on the wall between the fuel preparation machines. The channeled fuel is then stored in the pool storage racks ready for insertion in the reactor.

The preceding description is historically accurate and describes the original intent to reuse channels. More recently, Browns Ferry has committed not to use channels for a second bundle lifetime due to channel bow concerns (reference letter to NRC, "Browns Ferry (BFN) - Units 1, 2, and 3 - Response to NRC Bulletin 90-02 - Loss of Thermal Margin Caused by Channel Box Bow," L44900424802).

10.4.3 Servicing Aids

General area underwater lights are provided with a suitable reflector for general downward illumination. Suitable light support brackets are furnished to support the lights in the reactor vessel, allowing the light to be positioned over the area being serviced independent of the platform. Local area underwater lights are small diameter lights for additional downward illumination. Drop lights are lamps with no reflector, and are used for intense radial illumination where needed. These lights are small enough in diameter to fit into fuel channels or control blade guide tubes. A portable underwater television camera and monitor are part of the plant optical aids. The transmitted image can be viewed on the refueling platform. This remote display assists in the inspection of the vessel internals and general underwater surveillance in the reactor vessel and fuel storage pool. A general purpose, clear plastic viewing aid that will float is used to break the water surface for better visibility.

A portable underwater vacuum cleaner is provided to assist in removing crud and miscellaneous objects from the pool floor, or the reactor vessel. The pump and the filter unit are completely submersible for extended periods. Fuel pool tool accessories are also provided to meet the servicing requirements.

10.4.4 Reactor Vessel Servicing Equipment

Reactor vessel servicing equipment is supplied for safe handling of the vessel head and its components, including nuts, studs, bushings, and seals.

A head strongback is used for lifting the drywell head. The strongback is designed to keep the head level during lifting and transport. Cruciform in shape, with four equally spaced lifting points, the strongback is designed so that no single component failure would cause the load to drop or to swing uncontrollably.

The reactor vessel head is serviced and lifted by the reactor head strongback carousel. The carousel is an integrated assembly capable of performing the following functions:

> Lifting of the reactor vessel head - The strongback, when suspended from the reactor building crane main hook, will carry the reactor head plus the carousel, tensioning components, four reactor closure studs, and the removed reactor head nuts and washers stored in the nut storage rack.
>
> Tensioning of reactor head studs - The carousel, when supported on the reactor head lifting lugs, will carry four stud tensioners on a monorail above the reactor/head bolt circle; each tensioner has an electric operated hoist with individual controls. The stud tensioners are hydraulically operated.
>
> Storage with reactor head - The strongback carousel stores with the removed head on the refueling floor pedestals, ready for reinstallation of the head and stud tensioning with nut/washer installation from storage bins in the nut rack.
>
> Storage without the reactor head - When the strongback carousel is not in use, it will be stored on the refueling floor using a special training stand for supporting the lifting rods above the concrete floor.

The head holding pedestals are designed to support the vessel head and strongback/carousel to permit seal replacement and seal surface cleaning and inspection. The mating surface between vessel and pedestal is selected to minimize the possibility of damaging the vessel head.

A reactor servicing platform permits the operator to work at a level just above the reactor vessel flange and permits servicing access for the full core diameter. A service platform support is provided which rests on the vessel flange surface and serves as both a track for the servicing platform, and as a vessel seal surface protector.

A vessel nut handling tool is provided. This tool handles one nut and features a spring loading device to lift the nut and clear the threads.

10.4.5 In Vessel Servicing Equipment

Replacement incore detectors are removed from their shipping container by hand. The instrument handling tool is attached to the incore detector by the operators on the refueling platform. The instrument strongback supports the incore detector until it is in the vessel in a vertical position, then the incore detector is placed in the instrument handling tool mounted on the refueling platform monorail hoist; the strongback is removed; and the LPRM string is lowered into place. Final incore detector insertion is accomplished with the instrument handling tool. The instrument handling tool is used for removing and installing fixed incore detectors as well as handling neutron sources and the Source Range Monitor/Intermediate Range Monitor dry tubes.

In the unlikely event that incore housing flange "O" rings need replacing, an incore guide tube seal and a test plug are provided. The guide tube seal seats when the beveled guide tube enters the vessel from the top. When the drain on the water seal drain assembly is opened, water drains from the incore housing and guide tube; hydrostatic pressure seats the guide tube seal and allows the flange to be removed. The incore guide tube seal contains a bail, similar to the control rod and fuel bail.

10.4.6 Refueling Equipment

One refueling platform is provided for each unit. It is used as the principal means of transporting fuel assemblies back and forth between the reactor well and the storage pool. The platform travels on tracks extending along each side of the reactor well and fuel pool. The platform supports the refueling grapple and auxiliary hoists. The refueling grapple is suspended from a trolley that can traverse the width of the platform.

Platform operations are controlled from a walkway and an operator station on the trolley. The platform contains a Z-axis position-indicating device that indicates the vertical position of the fuel grapple. Horizontal (x and y axis) position indication is available from a manual line-up and sighting system and may be available from a digital display located in the operator's cab. The manual line-up and sighting system involves position marks on the crane that are aligned with positioning marks on a yard stick type device permanently secured to the handrails adjacent to the reactor cavity. The digital display of x and y position, if used, is provided by bridge and trolley position encoders driven by gears which are directly coupled to their respective axis. Where X-Y position is available through encoders, a Programmable Logic Controller (PLC) also maintains three-dimensional operational zone boundary protection. The manual line-up and sighting system and/or the X-Y position display are used to reach the approximate core coordinate locations. Visual inspection by the crane operator and second party verification is used to finalize and actually place the fuel bundle into position.

A single operator is capable of controlling all the motions of the platform required to handle the fuel assemblies during refueling. Interlocks on both the grapple hoist and auxiliary hoists prevent lifting of a fuel assembly over the core with a control rod withdrawn; interlocks also prevent withdrawal of a control rod with a fuel assembly over the core attached to either the fuel grapple or hoists. Interlocks also block travel of the refueling platform over the reactor in the startup mode. The refueling interlocks are described and evaluated in Subsection 7.6, "Refueling Interlocks."

10.4.7 Storage Equipment

In addition to the new and spent fuel storage racks, other storage equipment as listed in Table 10.4-1 is provided.

Defective fuel assemblies may be placed in special fuel cans. Each can is adaptable for individual sipping. For channel removal, the can may be removed from the rack, placed in the fuel preparation machine, the can cover removed, and the channel lifted clear of the fuel assembly. Provisions for dry sipping are provided. This system allows for the detection of leaking fuel rods during refueling and takes place in the fuel pool.

10.4.8 Under Reactor Vessel Servicing Equipment

The necessary equipment to remove control rod drives during a refueling outage is provided. An equipment handling platform with a rectangular open center is provided. This platform is rotatable to provide space under the vessel so the control rod drive can be lowered and removed. Control rod drive handling equipment is used to remove and reinstall control rod drives. A thermal sleeve installation tool is used to rotate the thermal sleeve within the control rod drive housing. Sleeve rotation permits disengagement of the guide tube. A rope and pulley integral with the tool permits complete sleeve removal.

The replacement of fixed incore detectors requires an LPRM nut wrench, spanner tool, wire cutters, and water seal drain assembly. Detector cables are cut as close to the LPRM nut and seal as possible. The nut and seal assembly are removed and a water drain assembly is installed to prevent excessive drainage from the reactor vessel water inventory. The LPRM string assembly is removed from above by refuel floor personnel using the instrument handling tool. The reverse process is used to install a new assembly.

Additional nuclear system tools and servicing equipment are listed in Table 10.4-1.

10.4.9 Storage Pit

Large radioactive components such as the steam dryer and steam separator assembly are stored in the storage pit. The storage pit is separated from the reactor cavity by removable concrete blocks that serve as a shield when the dryer and separator are stored. Other large items, such as the pressure vessel head and drywell head, are stored on the refueling floor.

Because of the relatively low neutron activation dose rates and low crud dose rates, a dry transfer of the dryer is normally expected. To minimize operator exposure during dry transfer of the dryer assembly, the storage pit canal is deep enough so that, if desired, the top of the dryer can be kept at least 2 feet below the operating floor level during transfer (during a dry transfer, portions of the dryer assembly may be above the operating floor level). Wet transfers extend the refueling outage. The storage pit is deep enough below the canal that, with the reactor well drained, a minimum of 6 inches of water shielding can be maintained above the separator plenum dome.

Special liner considerations account for the abrasion and high unit loadings that occur on areas where the dryer and separator assemblies are placed. Provisions are made to control airborne contamination from storage pit walls and equipment during reactor well draindown.

10.4.10 Spent Fuel Shipping Cask Decontamination

Decontamination of spent fuel shipping casks is carried out in a facility located on the El. 664 floor adjacent to the Unit 2 fuel storage pit (Figure 1.6-2). The facility, which is shown in Figure 10.4-7, consists of a housing, a spray nozzle system, a ventilation system, an inspection lift, and means for rotating the cask. This facility serves all three units.

After a loaded shipping cask has been removed from a spent fuel pit with the overhead crane, the cask is moved into position in front of the housing, and the safety cables are removed from between the crane lower block and the cask yoke. The cask is then moved into the housing, the housing door closed, rubber seals placed around the rotator link, the rotator power supply cable connected, and the ventilation system started.

Water for the spray system is delivered by a hydraulic jet unit in which steam and cold water are mixed to produce a hot water stream at elevated pressure. Approximately 50 gpm is delivered to the spray nozzles at a temperature of about 200°F and a pressure of about 330 psig. The system is put into operation at a control panel outside the housing. As the cask rotates, water is sprayed from the top nozzles onto the top of the cask. Thereafter, the water flow is diverted to the

vertical nozzle bank, which is positioned at the top of the cask. After remaining stationary through one or more rotations of the cask, the nozzle bank is lowered about 6 inches and the process is repeated. This continues until the nozzle bank reaches the lower edge of the cask. The water flow then is diverted to the bottom nozzles, which spray the bottom of the cask through a number of cask rotations.

After the spray nozzles have been shut off, an operator enters the housing. Working from the inspection lift, he hand-cleans areas not reached by the spray nozzles. Smears are taken to locate areas that need additional hand cleaning.

A temporary ventilation system is provided to maintain a slight negative pressure within the housing, thereby minimizing outleakage of spray. The temporary system consists of a moisture separator, a HEPA filter and a blower mounted on the top of the housing. The blower has a capacity of approximately 1000 cfm.

Water is drained from the housing to a 15,000-gal cask decontamination tank located in the radwaste building (see paragraph 9.2.4.4). It is expected that about 10,000 gallons will be utilized in the decontamination of a cask.

When decontamination is completed, the cask is moved out of the housing, and the safety cables are replaced.

Table 10.4-1

TOOLS AND SERVICING EQUIPMENT***

Fuel Servicing Equipment
Fuel Prep Machine
New Fuel Inspection Stand
Channel Bolt Wrench
Channel Handling Tool
Fuel Inspection Fixture
General Purpose Grapple
Channel Transfer Grapple

Servicing Aids
Pool Tool Accessories
Actuating Pole
General Area Underwater Light
Local Area Underwater Light
Drop Light
Underwater TV System
Underwater Vacuum Cleaner
Light Support Bracket
In-Core Cutter
In-Core Manipulator

Reactor Vessel Servicing
*Reactor Vessel Servicing Tools
Steam Line Plugs
Shroud Head Bolt Wrench
Vessel Nut Handling Tool
Head Holding Pedestal
Head Nut and Washer Rack
Head Stud Rack
Dryer and Separator Sling
Drywell Head Strongback
Service Platform
Service Platform Support
Power Wrench
Stud Tensioner Assembly
Strongback Carousel

In-Vessel Servicing
Instrument Strongback
Control Rod Grapple

In-Vessel Servicing (Continued)
Control Rod Guide Tube Grapple
Fuel Support Grapple
Control Rod Latch Tool
Instrument Handling Tool
Orifice Grapple (Peripheral)
Control Rod Guide Tube Seal
Incore Guide Tube Seal
Orifice Holder (Peripheral)
Blade Guide
Grid Guide
Jet Pump Servicing Tools

Refueling Equipment
Refueling Equipment Servicing Tools
**Refueling Platform Equipment Assembly
Jib Crane

Storage Equipment
Spent Fuel Storage Rack
Channel Storage Rack
Storage Rack (Control Rod and Defective Fuel)
New Fuel Storage Rack
Defective Fuel Storage Container

Under-Reactor Vessel Servicing Equipment
Neutron Monitoring System Servicing
  LPRM nut wrench
  Spanner Tool
  Water Seal Cap/Drain Assembly
  Water Seal Drain and Pusher Assembly
CRD Handling Equipment
Equipment Handling Equipment
Thermal Sleeve Installation Tool
Incore Flange Seal Test Plug

Control Rod Drive Hydraulic System Servicing Equipment
CRD Servicing Equipment
CRD Hydraulic System Tools

* (Double Stud Thread Protectors and Guide Caps. Share Other Equipment.)

**Refueling Platform Equipment Assembly includes the refueling platform and the main fuel grapple in its contents.

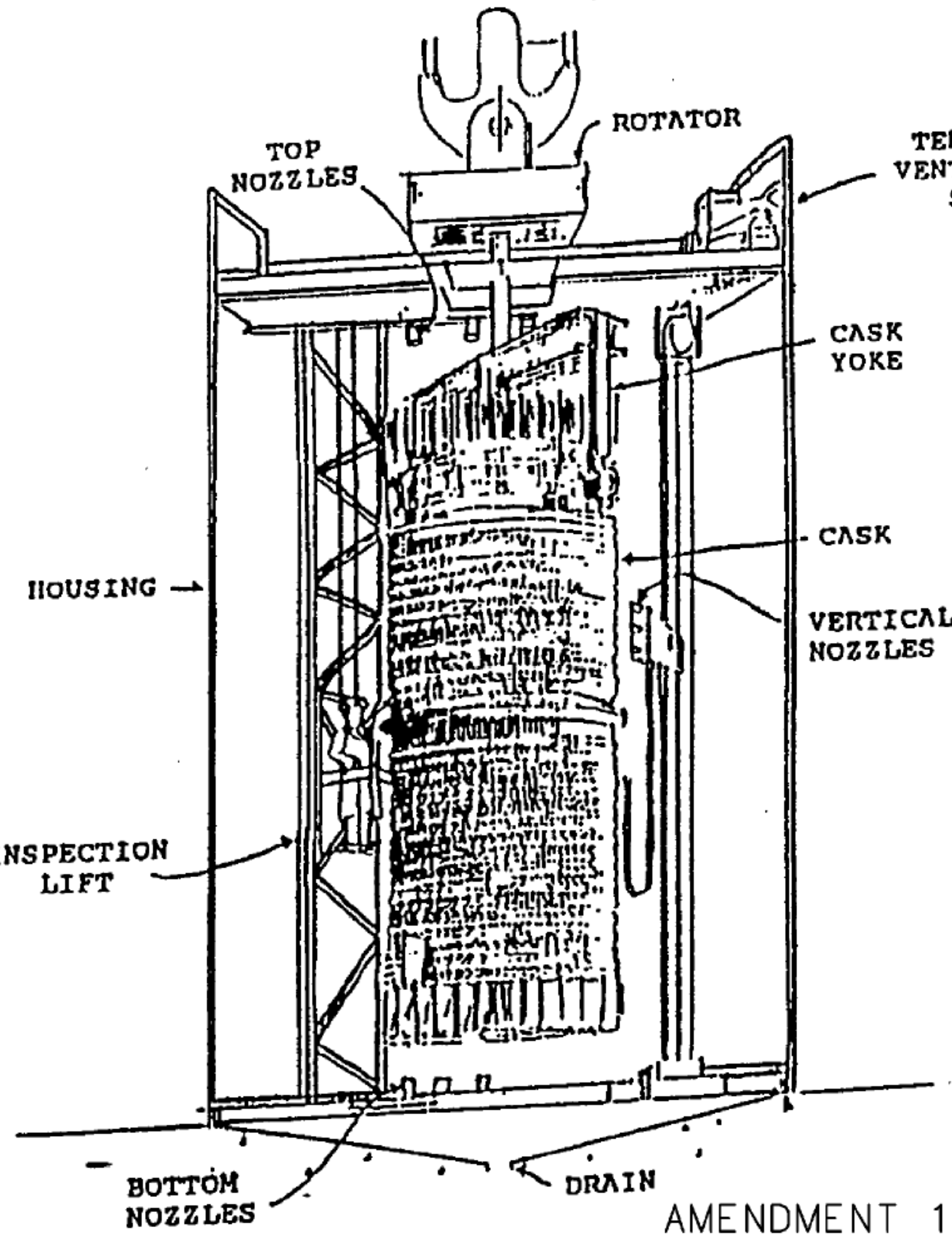
***TVA and/or an offsite vendor may employ other tools or equipment not listed in Table 10.4-1 to conduct fuel inspection, component inspection, testing, or maintenance/modification activities in conjunction with refuel floor (outage and non-outage) work items.

Figure 10.4-1

Deleted by Amendment 13.

Figures 10.4-2 through 10.4-6

Deleted by Amendment 9.



AMENDMENT 16

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

Spent Fuel Shipping Cask
Decontamination Facility
FIGURE 10.4-7

10.5 FUEL POOL COOLING AND CLEANUP SYSTEM

10.5.1 Power Generation Objective

The objective of the Fuel Pool Cooling and Cleanup System is to remove the decay heat from the fuel assemblies and maintain the fuel pool water within specified temperature limits.

10.5.2 Safety Design Basis

The Fuel Pool Cooling and Cleanup System shall be designed to remove decay heat from the fuel assemblies and maintain fuel pool water within specified temperature limits. Seismic design requirements are discussed in Section 10.5.5 safety evaluation.

10.5.3 Power Generation Design Basis

1. The Fuel Pool Cooling and Cleanup System shall minimize corrosion product buildup and control water clarity, so that the fuel assemblies can be efficiently handled underwater.

2. The Fuel Pool Cooling and Cleanup System shall minimize fission product concentration in the water which could be released from the pool to the Reactor Building environment.

3. The Fuel Pool Cooling and Cleanup System shall monitor fuel pool water level and maintain a water level above the fuel sufficient to provide shielding for normal building occupancy.

10.5.4 Description

The Fuel Pool Cooling and Cleanup System is shown in Figures 10.5-1a, 10.5-1b sheets 1, 2, 3, and 4, 10.5-1c, and 10.5-1d. The system cools the fuel storage pool by transferring the spent fuel decay heat through heat exchangers to the Reactor Building Closed Cooling Water System (see Table 10.5-1, Fuel Pool Cooling and Cleanup System Specifications. Water purity and clarity in the storage pool, reactor well, and dryer-separator storage pit are maintained by filtering and demineralizing the pool water through a filter demineralizer, which is shown in Figures 10.5-2 sheets 1, 2, 3, and 4.

The system for each fuel pool consists of two circulating pumps connected in parallel, two heat exchangers, one filter demineralizer subsystem, two skimmer surge tanks, and the required piping, valves, and instrumentation. Each pump has a design capacity equal to, or greater than, the system design flow and is capable of simultaneous operation. Four filter demineralizers are provided including one spare

filter demineralizer shared between the three active units, each with a design capacity equal to or greater than the design flow rate for a fuel pool. The pumps circulate the pool water in a closed loop, taking suction from the surge tanks, circulating the water through the heat exchangers and filter demineralizer, and discharging it through diffusers at the bottom of the fuel pool and reactor well (as required during refueling operations). The water flows from the pool surface through skimmer weirs and scuppers (wave suppressers) to the surge tanks. The fuel pool pumps and heat exchangers are located in the Reactor Building below the bottom of the fuel pool. The fuel pool filter demineralizers, which collect radioactive corrosion and fission products, are located in the Radwaste Building. The fuel pool concrete structure and metal liner are designed to withstand earthquake loads per project seismic requirements as a Class I system.

Fuel pool water is normally recirculated. The heat exchangers are designed to remove the decay heat load of the normal discharge batch of spent fuel (see Section 10.5.5). The heat exchangers in the Residual Heat Removal System are used in conjunction with the Fuel Pool Cooling and Cleanup System to supplement pool cooling in the event that a larger than normal amount of fuel is stored in the pool. Normal makeup water for the system is transferred from the condensate storage tank to the skimmer surge tanks to make up evaporative and leakage losses. A seismic Class I qualified source of makeup water is provided through the crosstie between the RHR system and the fuel pool cooling system. (The intertie between the RHRSW system and the RHR can be utilized to admit raw water as makeup.) Also, a standpipe and hose connection is provided on each of the two EECW headers which provides two additional fuel pool water makeup sources. Each hose is capable of supplying makeup water in sufficient quantity to maintain fuel pool water level under conditions of no fuel pool cooling.

The reactor well and dryer-separator pit are filled for refueling by transferring water from condensate storage to the reactor vessel via various pathways such as the Condensate and Feedwater Systems, the Core Spray System, and the Residual Heat Removal System. During the filling operation, water temperature is monitored for reactivity/shutdown margin considerations. A flow path can be established from the reactor well through the skimmer surge tanks to the fuel pool coolant system filters/demineralizers. This flow path allows filtering of water in the reactor well when filled. Following refueling, the water is drained to condensate storage.
The circulation patterns within the reactor well and storage pool are established by the placement of the diffusers and skimmers so as to sweep particles dislodged during refueling operations away from the work area and out of the pools. The normal flow pattern may be altered by taking suction from the bottom of the dryer-separator storage pit to control particles dislodged from parts transferred to the dryer-separator storage pit. Suction may also be taken from the bottom of the reactor well. A portable, submersible-type, underwater vacuum cleaner is provided to remove crud and miscellaneous objects from the pool walls and floor.

Pool water clarity and purity are maintained by a combination of filtering and ion exchange. The filter demineralizer maintains a pH range of 5.6 to 8.6 for compatibility with fuel racks and other equipment.

Material is removed from the circulated water by the pressure precoat filter demineralizer unit in which finely divided powdered ion exchange resin serves as a disposable filter medium. The resin is replaced when the pressure drop is excessive, the ion exchange resin is depleted, or as required by plant conditions. Backwashing and precoating operations are controlled from the Radwaste Building. The spent filter medium is flushed from the elements and transferred to the waste backwash receiver tank by backwashing with air and condensate. New ion exchange resin is mixed in a precoat tank and transferred as a slurry by a precoat pump to the filter where the solids deposit on the filter elements. The holding pump maintains circulation through the filter in the interval between the precoating operation and the return to normal system operation.

The filter demineralizer units are designed to operate with water flowing at approximately 2 gpm/sq ft. Powdered ion exchange resin or resin mixed with cellulose is used as a filter medium. The holding element for the filter material is a stainless steel mesh, mounted vertically in a tube sheet and replaceable as a unit. Venting is possible from below the tube sheet and from the upper head of the filter vessel. The upper head is removable for installation and replacement of the holding elements. The filter vessel is constructed of phenolic resin-coated carbon steel. A poststrainer is provided in the effluent stream of the filter demineralizer to limit the migration of the filter material. The strainer element is capable of withstanding a differential pressure greater than the developed pump head for the system.

The ion exchange resin is a mixture of finely ground cation and anion resins in proportions as determined by service requirements. The cation resin is supplied in the fully regenerated hydrogen form. The anion resin is supplied in a fully regenerated hydroxide form.

The maximum pressure drop across the filter and associated process valves and piping at the time for filter media replacement should not exceed the value shown in Table 10.5-1. A holding pump is connected to each filter demineralizer. This pump starts automatically to maintain sufficient flow through the filter media to retain it on the filter elements during loss of system flow. The holding flow rate is 0.1 gpm/sq ft of filter area. The backwash system is used to completely remove resins and accumulated sludge from the filter demineralizers with a minimum volume of water. Backwash slurry is drained to a local waste backwash receiver tank. The precoat system is designed to rapidly apply a uniform precoat of filter media to the holding elements of a filter demineralizer. The precoat tank is carbon steel coated with phenolic materials and sized to provide adequate volume for one precoating. An agitator is furnished with the tank for mixing. One centrifugal precoat pump and associated piping and valves are provided to precoat any filter demineralizer and

recirculate the water to the precoat tank or suction side of the precoat pump at a rate of 1.5 gpm/sq ft of filter area. The precoat system is also capable of cleaning or decontaminating any filter demineralizer unit with a detergent or citric acid solution. The filter demineralizer units are located separately in shielded cells. Sufficient clearance is provided to permit removal of the filter elements from the vessels. Each cell contains only the filter demineralizer and piping. All inlet, outlet, recycle, vent, drain, and other valves are located on the outside of one shielding wall of the room, together with necessary piping and headers, instrument elements and controls. Penetrations through shielding walls are located so as not to compromise radiation shielding requirements.

The system instrumentation is provided for both automatic and remote-manual operations. Fuel pool skimmer surge tank high, low, and low-low water level switches are provided. The fuel pool skimmer surge tank high and low level switches provide level change indications in the control room and the pump area and annunciates in the Main Control Room. Control of flow to or from the reactor well is accomplished manually during refueling. A level indicator mounted at the valve rack is provided to monitor reactor well water level during refueling. A fuel pool high water level switch operates a local indicator light. A fuel pool high water level condition would be indicated by the fuel pool skimmer surge tank level alarms in the Main Control Room. A fuel pool low water level switch alarms in the Main Control Room as a Fuel Pool System Abnormal Common Alarm. The trip point is adjustable over the range of skimmer weir adjustment. Seismic requirements are discussed in Section 10.5.5 safety evaluation.

The Fuel Pool System Abnormal Main Control Room Common Alarm initiates on pump low discharge pressure, low fuel pool level, Gate seal or drywell to reactor well seal leakage, or refueling bellows leakage and for Unit 1 the individual alarm status is available on ICS. Spent fuel pool level requirements are provided in Technical Specifications Section 3.7.6. Verification that fuel pool level is within allowed limits is performed by direct observation or by use of the low level alarm.

The pumps are controlled from the control room. Pump low-suction pressure automatically turns off the pumps in case of improper valve alignment. A pump low-discharge pressure alarm indicates in the Main Control Room. The controls for the remote manually controlled valves which discharge the fuel pool water to the condenser hotwell and condensate storage tank are located on the pump local panel. The open or closed condition of each of these valves is indicated by lights on the pump local panel.

The flow rate through each of the filter demineralizers is indicated by flow indicators in the pump area and on the radwaste panel. The flow indicators on the pump area panel can be seen by the operators from the vicinity of the fuel pool cooling system.

A high rate of leakage through the refueling bellows assembly, drywell to reactor seal, or the fuel pool gates is indicated by lights on the instrument rack in the pump area and is alarmed in the Main Control Room. The refueling bellows drains have been welded closed. Seismic requirements are discussed in Section 10.5.5 safety evaluation.

The filter demineralizers are controlled from a panel in the Radwaste Building. Differential pressure and conductivity instrumentation is provided for each filter demineralizer unit to indicate when backwash is required. Suitable alarms, differential pressure indicators, and flow indicators are provided to monitor the condition of the filter demineralizers.

Instrumentation is provided to monitor wide range spent fuel pool level between the top of the spent fuel assemblies and the normal water level. The system consists of two redundant level channels utilizing guided wave radar to measure spent fuel pool level. Each channel includes a level element at the fuel pool, a level indicating transmitter, and a remotely located indicator. The instrumentation system is designed to meet NRC Order EA-12-051 requirements.

10.5.5 Safety Evaluation

Unloading the reactor core and the associated increase in fuel pool heat load is a controlled evolution. Administrative controls are used to ensure that fuel pool heat load does not exceed available cooling capacity. The capacity of the Fuel Pool Cooling and Auxiliary Decay Heat Removal (see FSAR Section 10.22) Systems, considering seasonal cooling water temperatures and current heat exchanger conditions, is utilized to maintain the fuel pool temperature at or below 125°F during normal refueling outages (average spent fuel batch discharged from the equilibrium fuel cycle). The RHR System can be operated in parallel with the Fuel Pool Cooling System (supplemental fuel pool cooling) to maintain the fuel pool temperature less than 150°F if a full core off load is performed.

The flow rate is designed to be larger than that required for two complete water changes per day of the fuel pool or one change per day of the fuel pool, reactor well, and dryer-separator pit. The maximum system flow rate is twice the flow rate needed to maintain the specified water quality.

The majority of the Fuel Pool Cooling and Cleanup System and the Reactor Building Closed Cooling Water System are not seismically Class I qualified; however, these are seismically robust systems that are expected to be able to remain functional or be restored to operation following a seismic event. Due to the large thermal capacitance of the spent fuel pool there is sufficient time following a loss of FSP cooling that forced cooling of the FSP can be restored prior to the FSP water boiling (references 1 and 2).

To assure adequate makeup under all normal and off normal conditions, the RHR/RHRSW crosstie provides a permanently installed seismic Class I qualified makeup water source to the FSP. Two additional sources of FSP water makeup are provided via a standpipe and hose connection on each of the two EECW headers. The RHR/RHRSW cross tie and each EECW header is capable of supplying makeup water in sufficient quantity to maintain fuel pool water level under conditions of no fuel pool cooling.

Two diffusers are placed in both the reactor well and the fuel pool to distribute the return water as efficiently and with as little turbulence as possible, and to minimize stratification of either temperature or contamination. Flow control valves at the operating floor enable the operator to achieve optimum recirculation patterns to control and maintain the specified water quality and operational conditions.
The circulating pump motors are powered from their corresponding unit 480-V shutdown board. These boards receive power from the diesel generators on loss of normal auxiliary power. They are considered nonessential loads and will be operated as required under accident conditions.

Automatic isolation and bypassing of the nonseismically designed filter demineralizer portion of the system (that portion of the system in the Radwaste Building) and automatic isolation of the nonseismically designed reactor well recirculation piping are actuated upon a low-level signal in the skimmer surge tank. Reactor well recirculation piping diffusers through the first normally closed isolation valve is seismically qualified.

Redundant level instrumentation is provided in the skimmer surge tank for actuation of the motor operated valves on low level as well as for annunciation in the control room of high and low levels. The fuel pool skimmer surge tank high and low level switches provide level change indications in the control room and the pump area for Units 2 and 3. For Unit 1, the fuel pool skimmer surge tank high and low level switches provide change indications in the pump area and are inputs to the Fuel Pool System Abnormal Main Control Room Common Alarm.

The dryer/separator pit and the reactor well structures have been analyzed and it was determined that a design basis earthquake will not cause a failure during the refueling mode (MODE 5) with the well pit full of water. (This analysis includes the refueling bellows, drywell-to-reactor-well seal, bulkhead plates, and the upper portion of the drywell.)

Each fuel storage pool is designed so that no single failure of structures or equipment will cause inability to (1) maintain irradiated fuel submerged in water, (2) reestablish normal fuel pool water level, or (3) safely remove fuel from the plant. In order to limit the possibility of pool leakage around pool penetrations, each pool is lined with stainless steel. In addition to providing a high degree of integrity, the

lining is designed to withstand abuse that might occur when the spent fuel cask is moved about. Drains in the drywell, reactor well, and fuel transfer canal are seismically qualified through the first normally closed valve or temporary plugs are inserted before initiation of the refueling mode (MODE 5). The 1-1/2-inch refueling bellows drains are permanently sealed. The bellows is drained by means of a portable pump after refueling. Each check valve in the fuel pool cleanup return diffuser lines is provided with a siphon breaking vent pipe in order to prevent siphoning of fuel pool water to no more than 6 inches below the normal water level. Interconnected drainage paths are provided behind the liner welds. These paths are designed to (1) prevent pressure buildup behind the liner plate, (2) prevent the uncontrolled loss of contaminated pool water to relatively cleaner locations within the secondary containment, (3) provide expedient liner leak detection and measurement. These drainage paths are formed by welding channels behind the liner weld joints and are designed to permit free gravity drainage to the floor drain collection tank via the floor drain sump.

References:

1. CNL-15-162, Letter from TVA to the NRC, dated August 27, 2015, “Status of Effort to Resolve Non-Conformance Related to Potential Loss of Spent Fuel Pool Cooling,” (L44 150827 002)

2. Letter from NRC to Mr. David Lockhbaum, dated November 2, 2015 (ADAMS Accession No. ML15132A625)

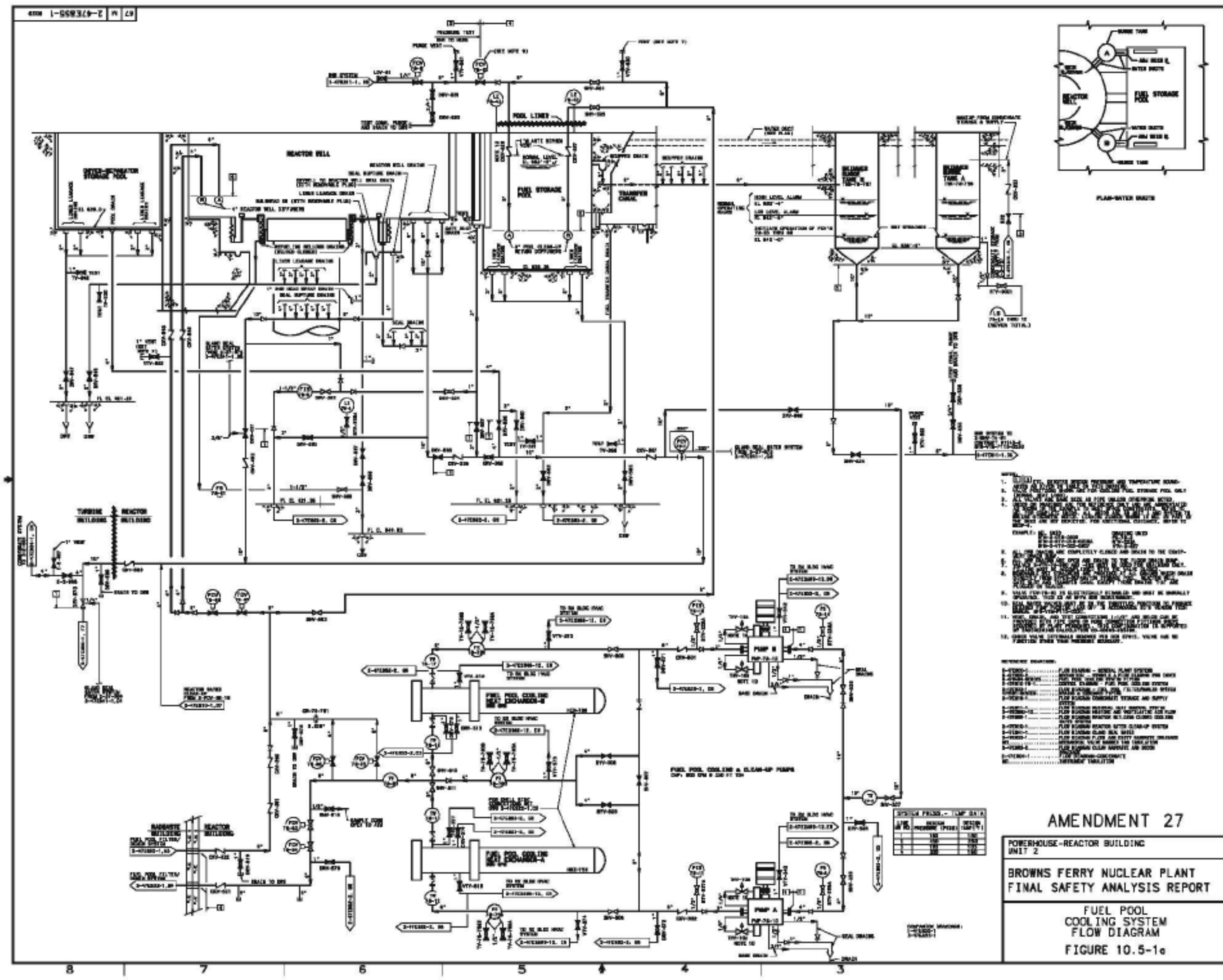
10.5.6 Inspection and Testing

No special tests are required because at least one pump, heat exchanger, and filter demineralizer are normally in operation while fuel is stored in a pool. The spare unit is operated periodically to handle abnormal heat loads or to replace a unit for servicing. Routine visual inspection of the system components, instrumentation, and trouble alarms is adequate to verify system operability.

TABLE 10.5-1

FUEL POOL COOLING AND CLEANUP
SYSTEM SPECIFICATIONS

SYSTEM FUNCTION	SYSTEM SPECIFICATION
RATED CORE THERMAL POWER	3,952 MWt
TOTAL POOL, WELL, AND PIT VOLUME	106,881 cu ft
FUEL STORAGE POOL VOLUME	51,340 cu ft
SYSTEM DESIGN FLOW (1 PUMP)	600 gpm
MAXIMUM FLOW (2 PUMPS)	1,200 gpm
PUMP CHARACTERISTICS (1 PUMP)	600 gpm, 330 ft TDH
FPC HEAT EXCHANGER (1) - DESIGN CAPACITY	4.4 x 10^6 Btu/hr/Hx.
Fuel Pool Flow: 600 gpm; Temp. 125°F / RBCCW Flow: 750 gpm; Temp. 100°F	
RHR HEAT EXCHANGER (1) - DESIGN CAPACITY	19.3 X 10^6 Btu/hr/Hx.
Fuel Pool Flow: 800 gpm; Temp. 150°F / RHRSW Flow: 4500 gpm; Temp. 95°F	34.0 X 10^6 Btu/hr/Hx.
Fuel Pool Flow: 2000 gpm; Temp. 150°F / RHRSW Flow: 4500 gpm; Temp. 95°F	
FILTER DEMINERALIZER	270 sq ft, 550 gpm 20 psi max dp (dirty)
HOLDING PUMP FLOW	27 gpm
PRECOAT FLOW	405 gpm
FUEL POOL SPECIFIC CONDUCTIVITY	≤ 10.0 μmho/cm
FUEL POOL CHLORIDE	≤ 500 ppb



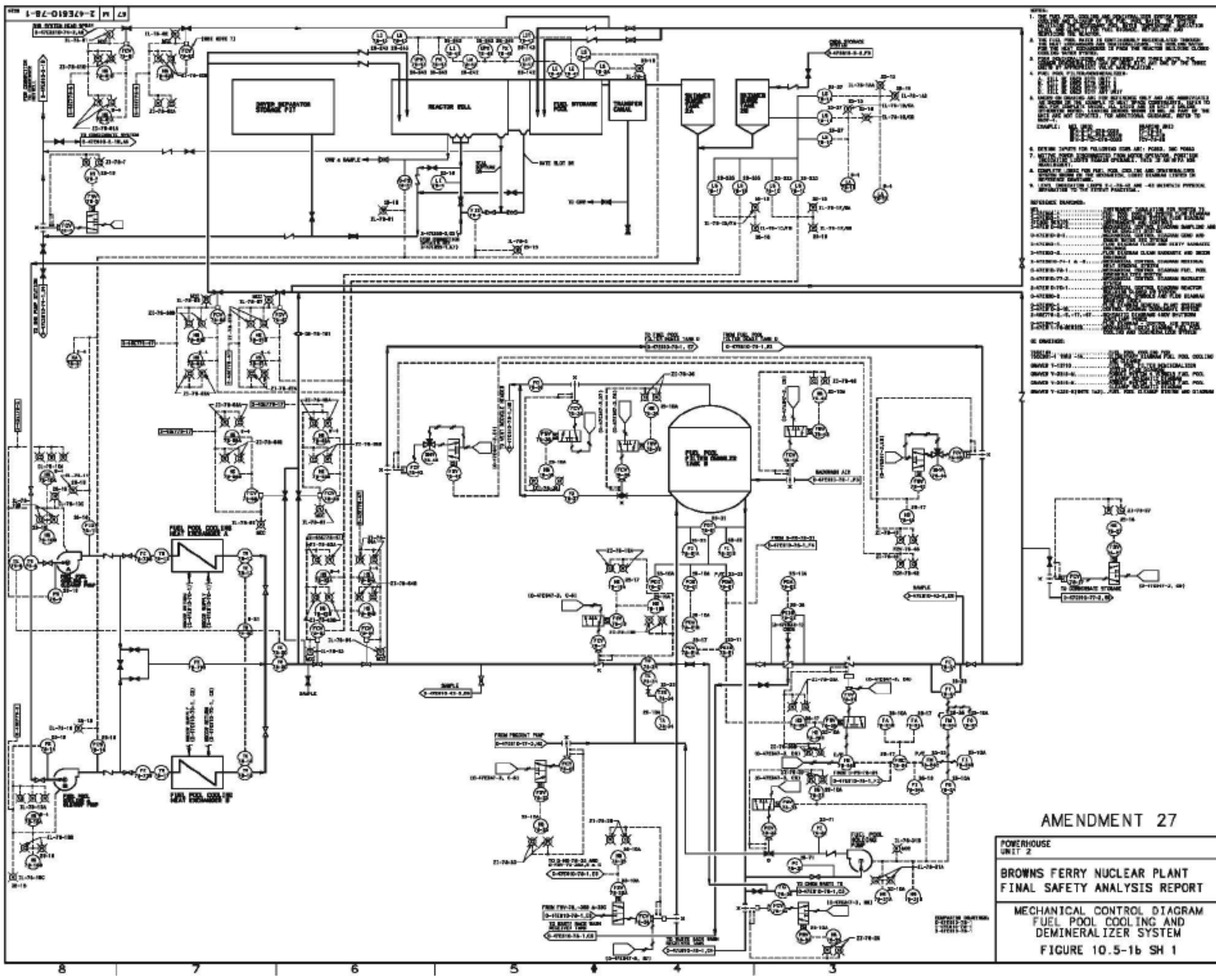
AMENDMENT 27

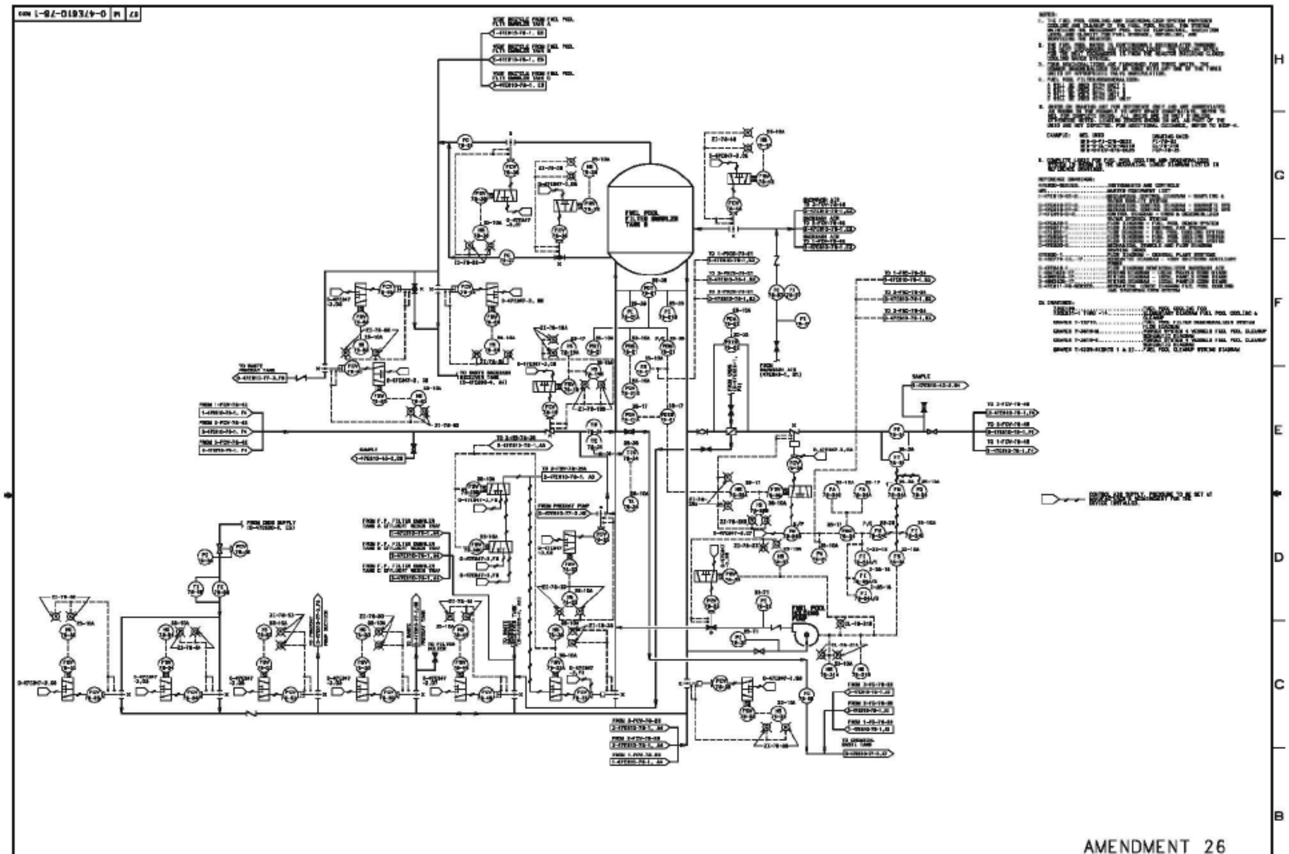
POWERHOUSE-REACTOR BUILDING
UNIT 2

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

FUEL POOL
COOLING SYSTEM
FLOW DIAGRAM

FIGURE 10.5-1c


AMENDMENT 27
POWERHOUSE
UNIT 2
BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT
MECHANICAL CONTROL DIAGRAM
FUEL POOL COOLING AND
DEMINERALIZER SYSTEM
FIGURE 10.5-1b SH 1
8
7
6
5
4
3
H
G
F
E
D
C
B
A



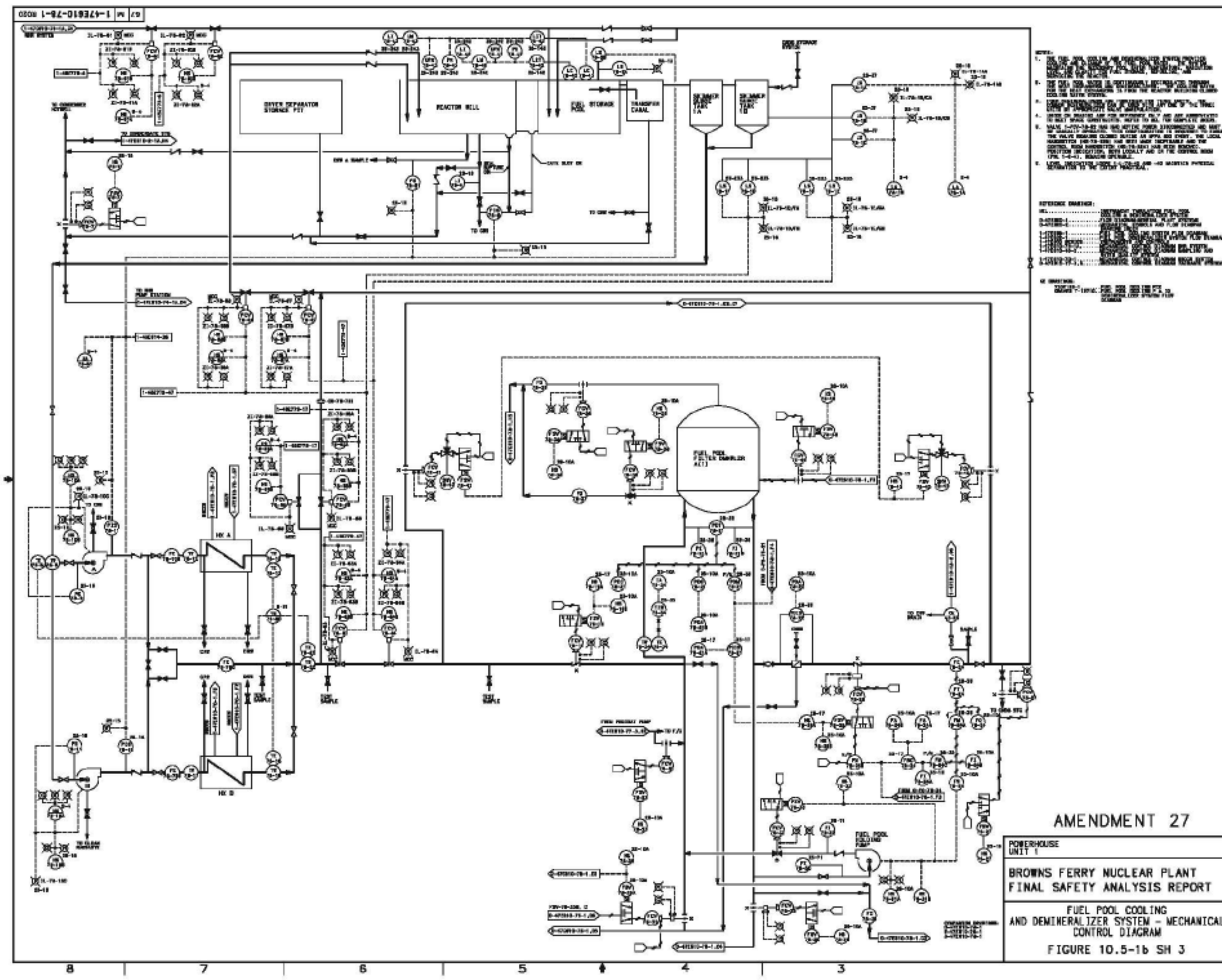
AMENDMENT 26

POWERHOUSE
UNIT 0

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

FUEL POOL COOLING
AND DEMINERALIZER SYSTEM
MECHANICAL CONTROL DIAGRAM

FIGURE 10.5-1b SH 2



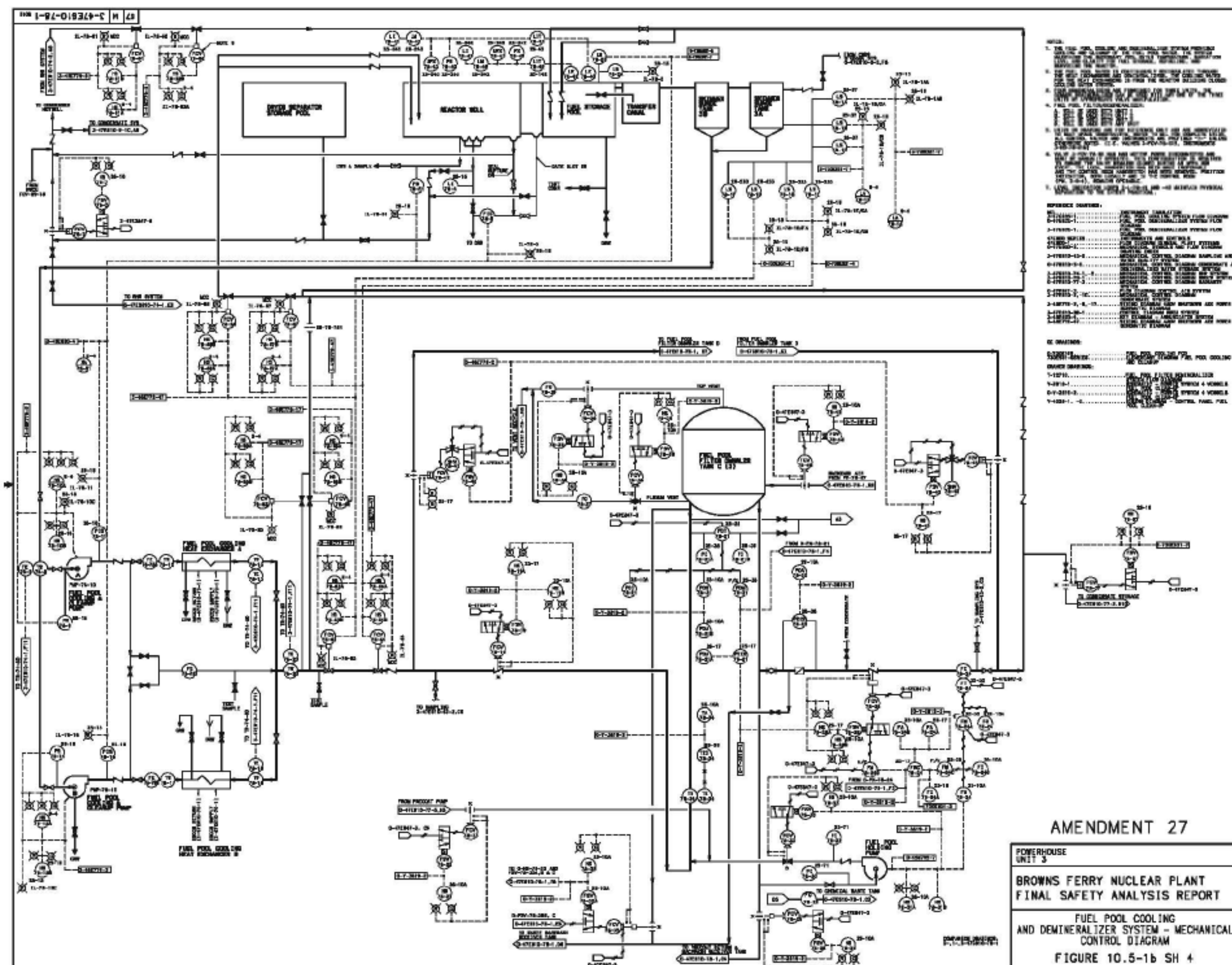
AMENDMENT 27

POWERHOUSE
UNIT 1

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

FUEL POOL COOLING
AND DEMINERALIZER SYSTEM - MECHANICAL
CONTROL DIAGRAM

FIGURE 10.5-1b SH 3



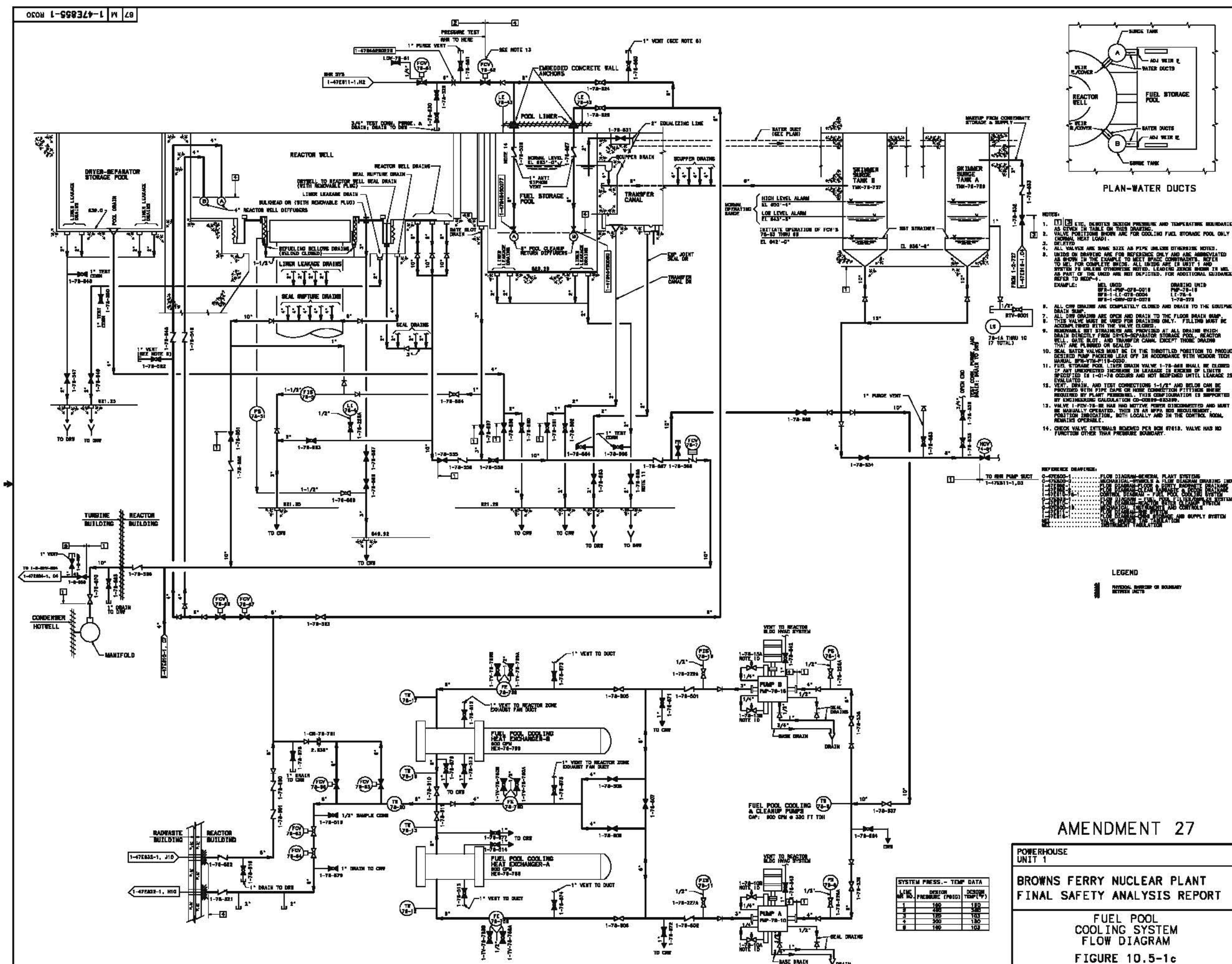
AMENDMENT 27

POWERHOUSE UNIT 3

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

FUEL POOL COOLING
AND DEMINERALIZER SYSTEM - MECHANICAL
CONTROL DIAGRAM

FIGURE 10.5-1b SH 4



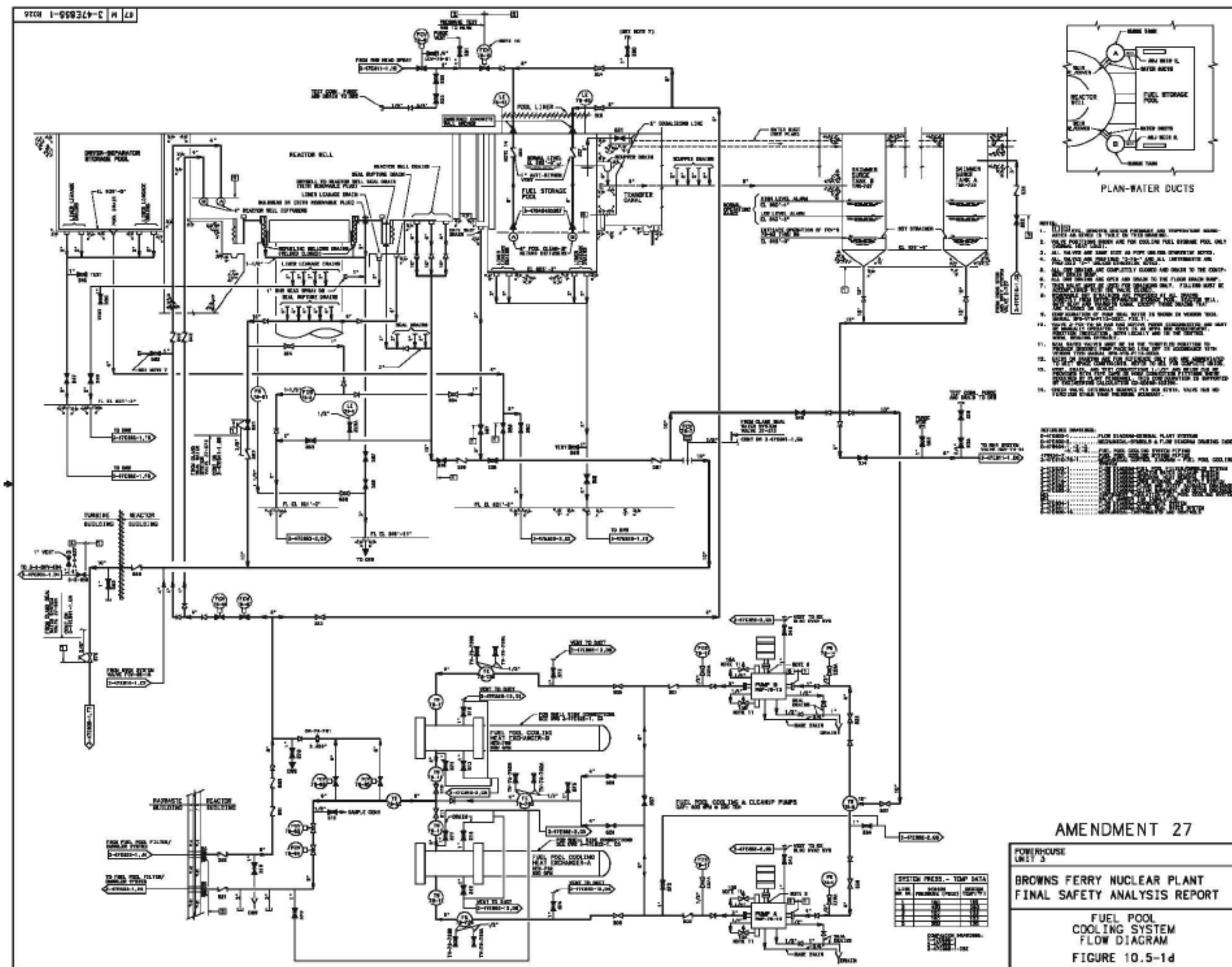
AMENDMENT 27

POWERHOUSE
UNIT 1

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

FUEL POOL
COOLING SYSTEM
FLOW DIAGRAM

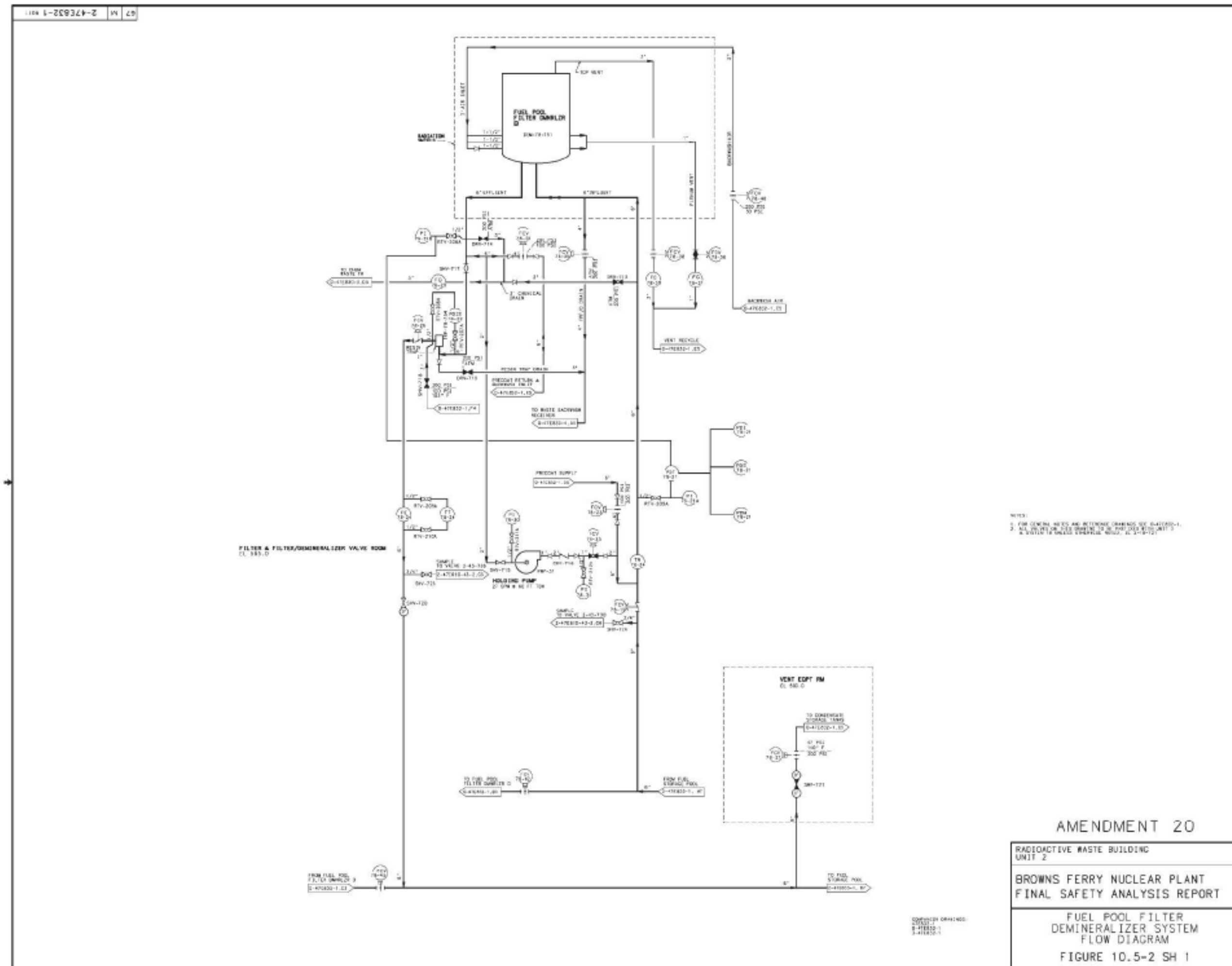
FIGURE 10.5-1c



BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

FUEL POOL COOLING SYSTEM FLOW DIAGRAM

FIGURE 10.5-1d



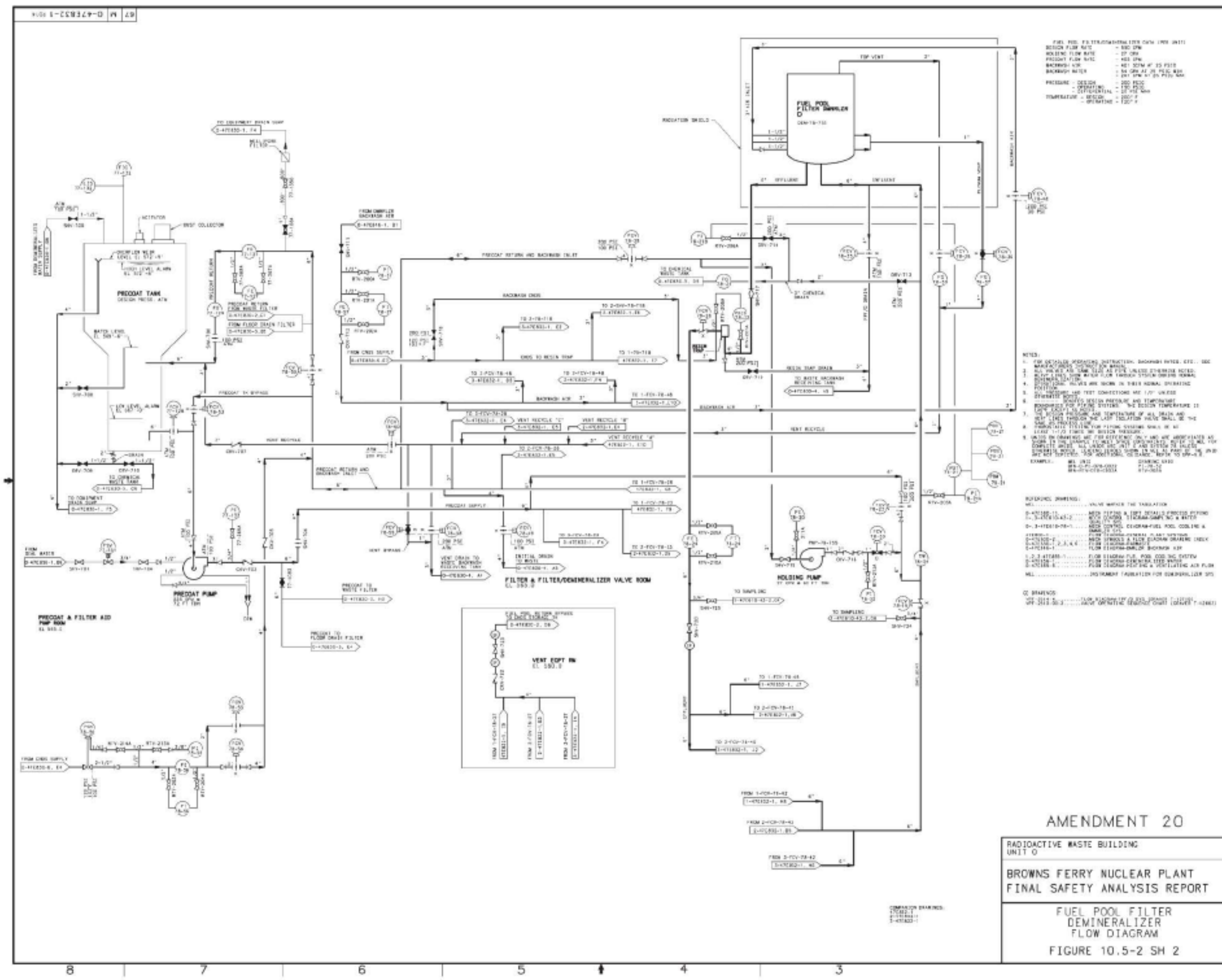
AMENDMENT 20

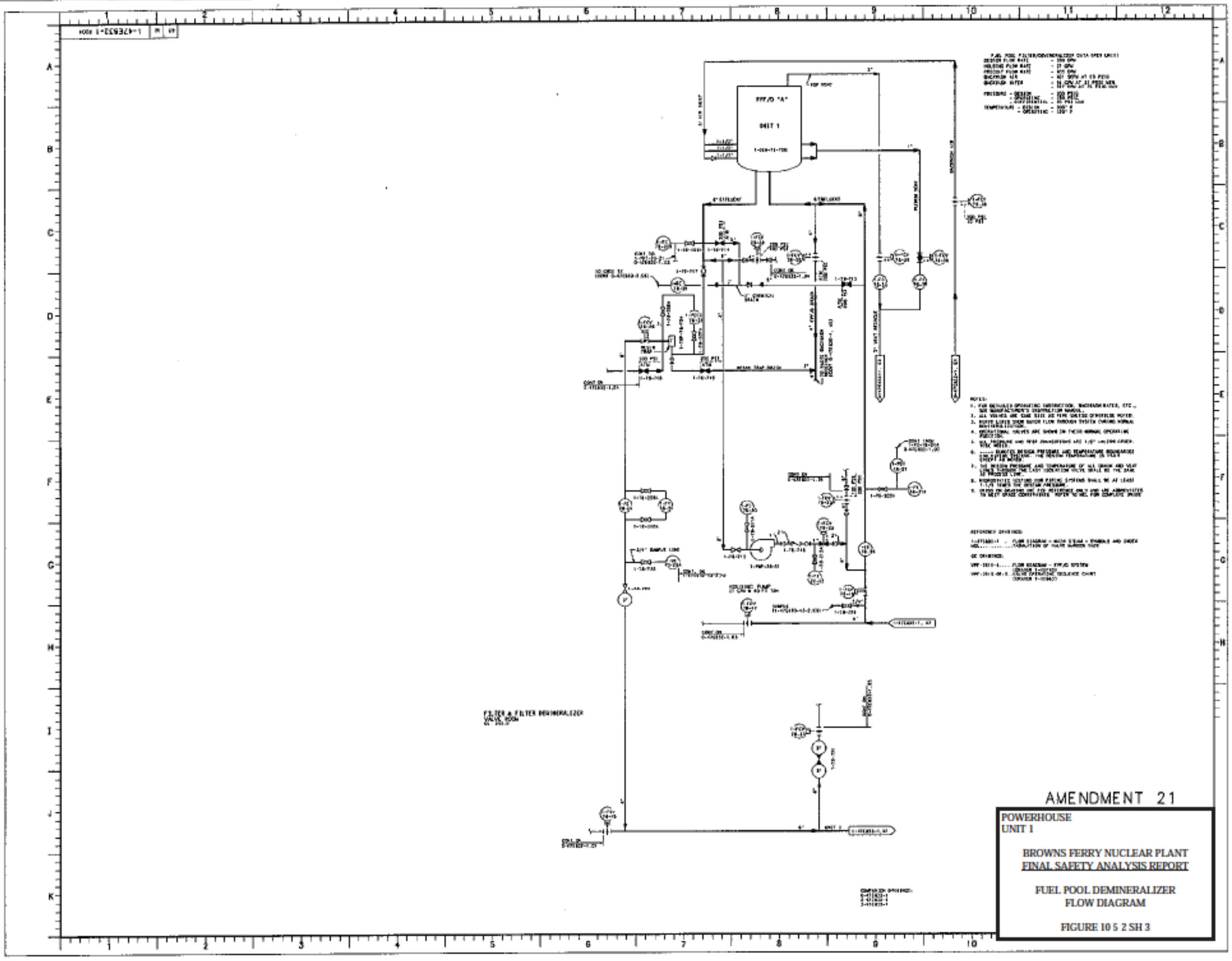
RADIOACTIVE WASTE BUILDING
UNIT 2

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

FUEL POOL FILTER
DEMINERALIZER SYSTEM
FLOW DIAGRAM

FIGURE 10.5-2 SH 1


PRECOAT TANK
PRECOAT PUMP
PRECOAT & FILTER AID PUMP ROOM
FILTER & FILTER/DEMINERALIZER VALVE ROOM
VENT EQPT RM
HOLDING PUMP
AMENDMENT 20
RADIOACTIVE WASTE BUILDING
UNIT 0
BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT
FUEL POOL FILTER
DEMINERALIZER
FLOW DIAGRAM
FIGURE 10.5-2 SH 2



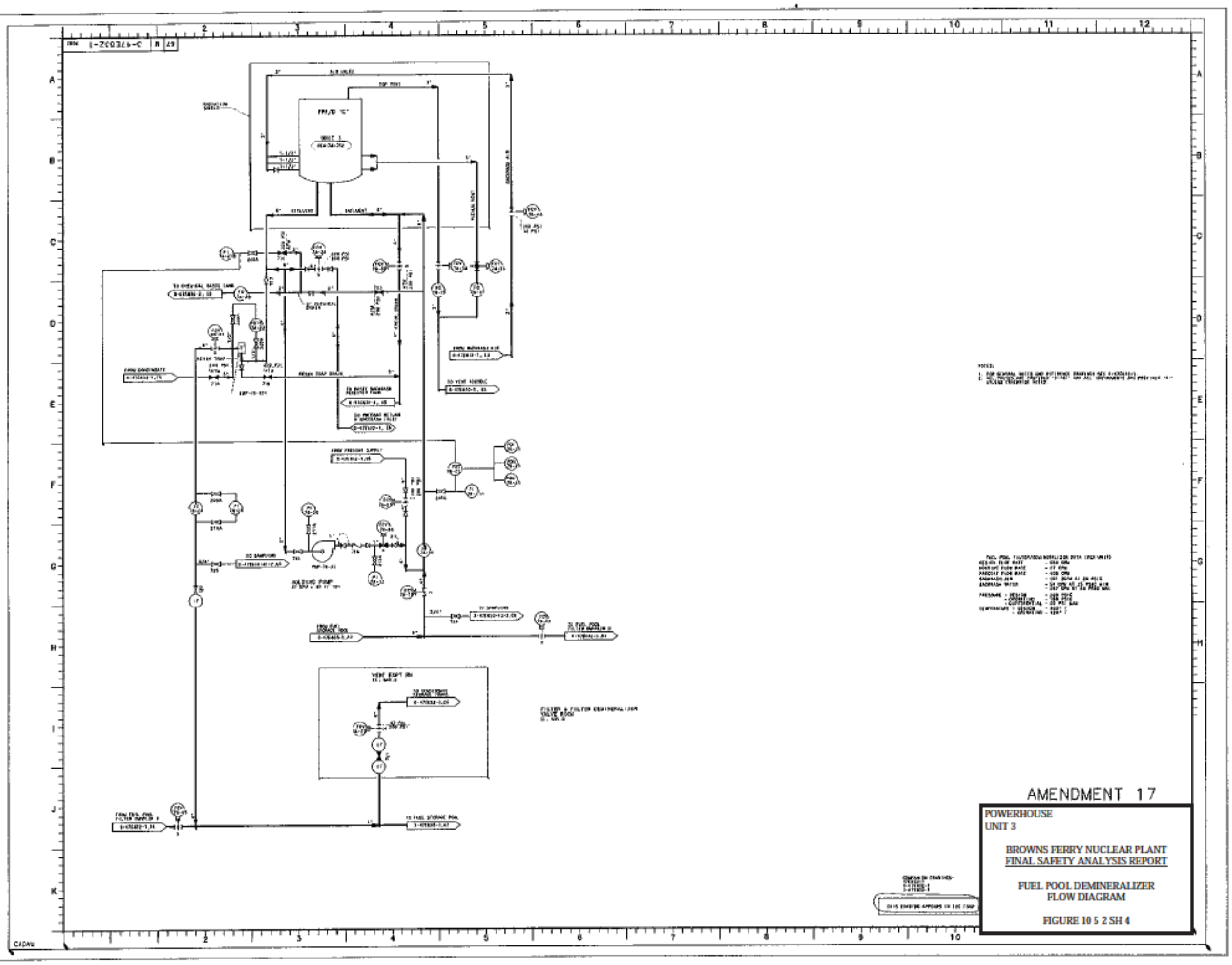
AMENDMENT 21

POWERHOUSE
UNIT 1

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

FUEL POOL DEMINERALIZER
FLOW DIAGRAM

FIGURE 10 5 2 SH 3



POWERHOUSE
UNIT 3

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

FUEL POOL DEMINERALIZER
FLOW DIAGRAM

FIGURE 10 5 2 SH 4

Figure 10.5-3

Deleted by Amendment 7.

Figure 10.5-4

Deleted by Amendment 10

Figures 10.5-4a through 10.5-4d
(Deleted by Amendment 22)

10.6 REACTOR BUILDING CLOSED COOLING WATER SYSTEM

10.6.1 Power Generation Objective

The power generation objective of the Reactor Building Closed Cooling Water System (RBCCWS) is to provide a continuous supply of cooling water to designated plant equipment located in the primary and secondary containments.

10.6.2 Power Generation Design Basis

1. The RBCCWS shall be designed to cool auxiliary plant equipment over the full range of reactor power operation.

2. The RBCCWS shall be designed to limit the possibility of radioactive material release to the raw cooling water.

3. The RBCCWS shall be designed so that failure of the offsite power supply will not impair cooling water supply to equipment within the primary containment.

10.6.3 Safety Design Basis

The portion of the RBCCWS inside the drywell out to and including the containment isolation valves shall be designed so that an earthquake would not impair the System's ability to function as a primary containment pressure boundary.

The portion of the Unit 1 and Unit 3 RBCCW system from the Secondary Containment Boundary penetration to the Secondary Containment Isolation valves shall be designed so that an earthquake would not impair the system's ability to function as a secondary containment pressure boundary.

10.6.4 Description

The Reactor Building Closed Cooling Water System is shown in Figures 10.6-1a, -1b, and -1c. The system consists of pumps, heat exchangers, and necessary control and support equipment. The cooling water pumps, located in the Reactor Building, are centrifugal-type with mechanical seals. The materials used in construction are listed in Table 10.6-1. Each of the three cooling water pumps delivers 1700 gpm; two pumps can provide 100 percent of the flow requirements for a unit during normal plant operation. One pump is provided as a common spare for all three units and is located in the Unit 1 Reactor Building. Selected heat exchangers, which are located in the primary and secondary containments, are connected to the RBCCWS as indicated in Figures 10.6-1a, -1b, and -1c. The RBCCWS heat exchangers are of the straight tube type, with tubes rolled into the tube sheets. The heat exchangers are designed to allow for expansion and contraction of all parts and are constructed of the materials given in Table 10.6-1

The heat exchangers are designed with raw-cooling-water flow on the tube side and corrosion-inhibited demineralized-water flow on the shell side. The two heat exchangers per unit can transfer 100 percent of the heat load requirements for the unit over the full range of reactor power operation. The heat load requirements for the RBCCWS are based on a RWCU flow of 270 gpm. RWCU flow can be increased up to 340 gpm under conditions when the RBCCWS can accommodate the additional heat load. One heat exchanger is provided as a common spare for all three units. Table 10.6-2 lists the operating conditions of the RBCCWS heat exchangers.

All equipment outside the primary containment is provided with manual flow-control valves (accessible during plant operation) or fixed orifices. They are located on the RBCCWS side of the heat exchangers for flow regulation. Fixed orifices are placed so that high velocity water from the orifice does not impinge on valves and fittings. Additionally, the non-regenerative heat exchangers have a TCV downstream of the heat exchangers to maintain RWCU temperatures within limits.

Suitable water level is maintained in the surge tank by manual control. Demineralized water is used for RBCCWS makeup to the surge tank to restore the level. High and low levels are alarmed in the control room to indicate a leak into or out of the System. The surge tank low level alarm occurs in time to permit manual refilling.

Each RBCCWS pump discharge is equipped with a pressure gauge. Low discharge pressure in the common discharge header closes the nonessential loop isolation valve and is alarmed in the Main Control Room. High common suction header temperature is also alarmed in the Main Control Room. Local temperature indicators, accessible during plant operation, are provided as shown in Figures 10.6-1a, -1b, and -1c.

The portion of the RBCCWS inside primary containment out to and including the containment isolation valves is seismic Class I to ensure that the system's ability to function as a primary containment pressure boundary is not compromised as a result of an earthquake. Cooling is maintained on all equipment inside primary containment during failure of offsite a-c power. At such times, cooling to the equipment outside primary containment can be stopped by closing the sectionalizing valve shown in Figures 10.6-1a, -1b, and -1c. Electrical power for operating the RBCCWS during such periods is supplied by the diesel generators. A list of the components cooled and their cooling requirements are given in Table 10.6-3a.

Raw cooling water pumps are the normal supply to the RBCCW system. An alternate cooling water supply for the RBCCWS heat exchangers is provided from the Emergency Equipment Cooling Water System (see Subsections 10.7 and 10.10).

Following a 480-V shutdown board load shed in either Unit 1 or 2, cooling water is automatically restored to equipment in both drywells. Drywell cooling, which is accomplished by the drywell atmosphere cooling coil blowers, is automatically restored to the nonaccident Unit 1 or 2. On the accident-affected unit, four blowers may be manually restarted (all other accident unit blowers are locked out).

A Unit 3 480-V shutdown board load shed signal has no effect on Units 1 or 2. On Unit 3, cooling water is automatically restored and four drywell blowers may be manually restarted.

The RBCCWS is monitored by a process radiation monitor (see Subsection 7.12) to detect possible in leakage from the cooled systems. Early detection of any increase in radioactivity in the RBCCWS limits the possibility of radioactive material release to the raw cooling water System.

To maintain the Unit 1 and/or Unit 3 drywell at a comfortable temperature during plant outages, when the RBCCW will not facilitate personnel activities the drywell can be cooled with the drywell coolers supplied with chilled water via RBCCW piping. The chilled water is supplied from a chiller/pump combination located outside the Unit 3 Reactor Building. The chiller/pump combination contains two air cooled water chillers with two in-line circulating pumps.

The chiller/pump is operable only during plant outages, when cooling is not required for the Reactor Recirculation Pumps and the Drywell Sump Heat Exchangers. During plant operation the supply and return lines from the chiller will be isolated from the operating RBCCW system by closed isolation valves located between the chiller and RBCCW piping.

10.6.5 Safety Evaluation

As described in paragraph 10.6.4, the portion of the RBCCWS inside the primary containment isolation boundary meets the safety design basis by designing to Seismic Class I specifications.

10.6.6 Inspection and Testing

The spare RBCCWS pump and spare RBCCWS heat exchanger may be periodically placed on the line to assure operability. System components located outside the drywell are located where routine visual inspections may be conducted to verify system operability.

Flow measuring devices may be temporarily installed in all major equipment cooling water headers. The flow measuring devices can then be used to balance RBCCW water flows, and with temperature indicators the heat load being removed from the drywell can be determined when necessary.

Table 10.6-1

REACTOR BUILDING CLOSED COOLING WATER SYSTEM

AND EQUIPMENT DATA

RBCCWS Design Pressure/Temperature	Piping - 150 psig/230°F Equipment - 150 psig/200°F
Raw Cooling Water	
Type	River Water
Normal Design Inlet Temperature	90°F
Maximum Inlet Temperature	95°F
RBCCWS Heat Exchangers	
Number Provided	2/unit - 1 common spare
Materials	
Tubes	Admiralty - SS for RBCCW HX1A and HX1B only
Tube Sheets	Carbon Steel - SS for RBCCW HX1A and HX1B only
Shell	Carbon Steel
RBCCWS Pumps	
Number Provided	2/unit - 1 common spare
Materials	
Casing	Cast Steel
Impeller	Bronze
Shaft	Heat Treated High Carbon or Alloy Steel

Table 10.6-2

REACTOR BUILDING CLOSED COOLING WATER SYSTEM

HEAT EXCHANGER OPERATING CONDITIONS

Heat Transfer (Btu/hr X 10^6)/HX		Number of HX's in operation per unit
Normal	15.75	2
Startup	15.96	2
Cooldown	11.10	2
Shutdown	3.31	2

Normal Flow (GPM)

Shell Side	1685
Tube Side	2550

Fluid

Tube Side (river water) Additives to minimize corrosion

Shell Side (demineralized water)

Additives to minimize corrosion

Seismic Coefficients

1.0g	horizontal
0.07g	vertical

TABLE 10.6-3a
REACTOR BUILDING CLOSED COOLING WATER SYSTEM HEAT LOADS
(3952 MWt)

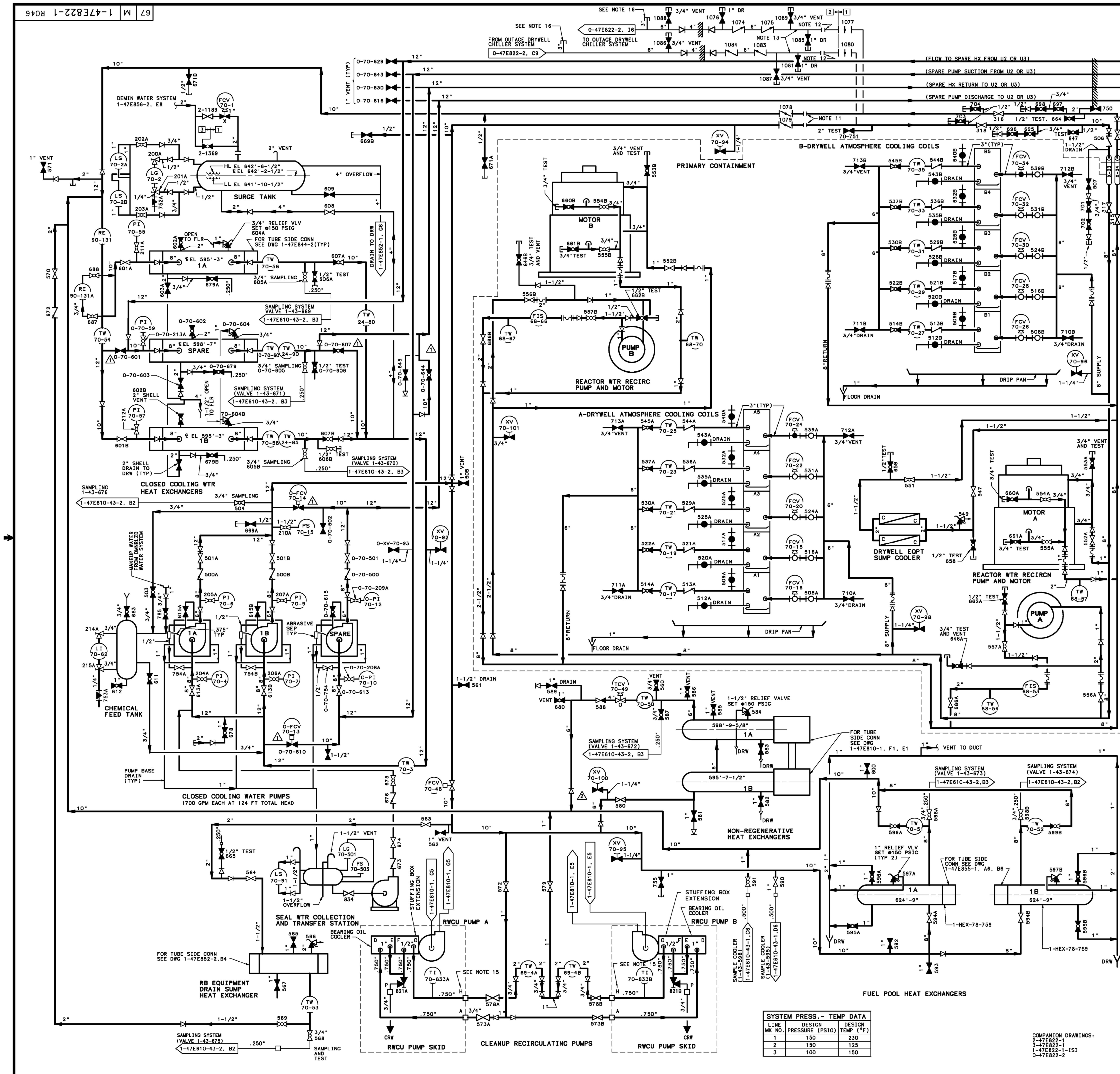
One Unit

MODE OF SERVICE		NORMAL OPERATION			COOLDOWN			SHUTDOWN			STARTUP			AC POWER FAILURE		
System Equipment	No. of Units	ΔP(Pal) / Total 10^6Btu/hr	T_{in}°F	T_{out}°F / Total GPM	ΔP(Pal) / Total 10^6Btu/hr	T_{in}°F	T_{out}°F / Total GPM	ΔP(Pal) / Total 10^6Btu/hr	T_{in}°F	T_{out}°F / Total GPM	ΔP(Pal) / Total 10^6Btu/hr	T_{in}°F	T_{out}°F / Total GPM	ΔP(Pal) / Total 10^6Btu/hr	T_{in}°F	T_{out}°F / Total GPM
Fuel Pool Heatr Exchange	2	9	100	112	0.1	100	112	0.1	100	112	9	100	112	-	-	
		8 8		1500	1 0		167	1.0		167	8 8		1500	-	-	
Reactor Recirculation Pump and Motor	2	10	100	115	10	100	115	-		-	10	100	115	10	100	115
		0 96		120	0 96		120	-		-	0 96		120	0.96		120
Drywell Atmosphere Cooler	8*	16.8	100	110	15.0	100	110	16 8	100	110	16.8	100	107	15 0	100	110
		5.19		1038	6.4		1274	5.19		1038	5.19		1038	6.4		1274
Reactor Bldg. Equipment Drain Tank Cooler	1	1 5	100	125	1 5	100	125	-		-	1 5	100	125	-		-
		0 5		40	0 5		40	-		-	0 5		40	-		-
Drywell Equipment Drain Sump Cooler	1	1 5	100	125	1 5	100	125	-		-	1 5	100	125	1.5	100	125
		0 5		40	0 5		40	-		-	0 5		40	0.5		40
Sample Cooler	4	3	100	111	3	100	111	-		-	3	100	111	-		-
		0 04		6 5	0 04		6.5	-		-	0 04		6.5	-		-
Cleanup Recirculating Pump Cooler	2	10	100	120	10	100	120	10	100	120	10	100	120	-		-
		0.17		15	0.17		15	0.17		15	0.17		15	-		-
Cleanup System Nonregenerative Heat Exchanger	1	8	100	150	7 5	100	143	-		-	5	100	171	-		-
		15.35**		610	12.7**		587	-		-	15.35**		482	-		-
Closed Cooling Water Heat Exchanger	2	4	90	102.4	0.75	90	110.2	0.1	90	104.2	4	90	102.5	0.1	90	107.3
Raw Cooling Water Loop - Tube Side		31.51		5100	22.2		2200	6.62		900	31.91		5100	7.8		900
Closed Cooling Water Heat Exchanger	2	12	118.7	100	5	118.9	100	1.5	110.4	100	10	119.7	100	2.1	110.9	100
CCW Loop - Shell Side		31.51		3369 5	22.2		2249.5	6.62		1220	31.91		3241.5	7.8		1434

*This number of drywell atmosphere coolers is the total number in operation which must provide cooling for heat loads listed with a 25 percent standby capability (two coolers and fans not in operation). Spare fans may be placed in service at operator discretion to provide additional margin.
There is a total of ten drywell atmosphere coolers located in each unit's drywell.
**Heat transfer rate corresponds to an RWCU flow of 270 gpm. Increasing the flow up to the maximum of 340 gpm results in an additional heat load of 7 X 10^6 Btu/hr. Operation of RWU above 270 gpm depends on the ability of RBCCW to accommodate the added heat load.

TABLE 10.6-3b

(Deleted)



SYSTEM PRESS.- TEMP DATA		
LINE MK NO.	DESIGN PRESSURE (PSIG)	DESIGN TEMP (°F)
1	150	230
2	150	125
3	100	150

COMPANION DRAWINGS:
2-47E822-1
3-47E822-1
1-47E822-1-ISI
0-47E822-2

NOTES:
1. DESIGN PRESSURE: 150 PSI, DESIGN TEMPERATURE: 230° F (PIPING) 200°F (EQUIPMENT). TEST PRESSURE 225 PSIG, UNLESS OTHERWISE NOTED.
2. ALL VALVES ARE SAME SIZE AS PIPING UNLESS OTHERWISE NOTED.
3. ALL PRESSURE AND TEST CONNECTIONS ARE 1/2" UNLESS OTHERWISE NOTED.
4. ALL VALVE NUMBERS HAVE UNIT NO. 1, SYSTEM 70 PREFIX "1-70-", EXCEPT AS OTHERWISE NOTED. ALL INSTRUMENT NUMBERS HAVE UNIT NO. 1 PREFIX "1-", EXCEPT AS OTHERWISE NOTED.
5. THE SPARE PUMP AND HEAT EXCHANGER CAN BE USED WITH ANY UNIT.
6. DRYWELL COOLING COILS A5 AND B5 SHOWN AS SPARES-ANY COIL CAN BE DESIGNATED SPARE.
7. 1 DESIGNATES VALVE MAY BE OPEN IF SYSTEM IS IN OPERATION (OPERATING INSTRUCTION OI-70, ATTACHMENT 1).
8. DELETED
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. FOR DESIGN CONDITIONS FOR DRYWELL OUTAGE CHILLER PIPING, SEE FLOW DIAGRAM 0-47E822-2, NOTE 1.
11. FLANGES DENOTE USE OF 10" SPECTACLE BLINDS (NORMALLY OPEN FOR RBCCW OPERATIONS AND MAY BE CLOSED FOR OUTAGE DRYWELL COOLING OPERATIONS.)
12. FLANGES DENOTE USE OF 6" SPECTACLE BLINDS (NORMALLY OPEN FOR OUTAGE DRYWELL COOLING OPERATIONS AND MAY BE CLOSED FOR RBCCW OPERATIONS.)
13. LOCATION FOR RECOMMENDED, FUTURE TIE-IN TO UNIT 2 RBCCW.
14. UNIDS ON DRAWING ARE FOR REFERENCE ONLY AND ARE ABBREVIATED TO MEET SPACE CONSTRAINTS. REFER TO MEL FOR COMPLETE UNIDS.
15. THESE LETTER DESIGNATIONS CORRESPOND TO THE INTERFACE POINTS SHOWN ON INGERSOLL-RAND DRAWINGS 63604227, 63604235 AND 63604284 (CONTRACT NO. 9748).
16. THE TEMPORARY INSTALLATION OF THE BACK-UP CHILLER, AND REQUIRED ANCILLARY EQUIPMENT TO OPERATE IT, SHALL BE DONE UNDER A WO, ADMINISTRATIVELY CONTROLLED WITH THE TEMPORARY ALTERATION PROCESS TO BE IN PLACE FOR LESS THAN 90 DAYS. THIS TEMPORARY CHILLED WATER SUPPLY AND RETURN WILL TIE INTO THE 6 INCH RBCCW PIPING USING THE 3 INCH PIPE TAPS. THE TEMPORARY CHILLER AND PUMP WILL BE CONNECTED TO THE PERMANENT RBCCW PIPING SYSTEM USING HOSES WITH A MINIMUM PRESSURE RATING OF 150 PSI. HOSES MUST BE SECURED TO PREVENT EXCESSIVE MOVEMENT DURING OPERATION. FILLING, VENTING, AND DRAINING OPERATIONS WILL BE PERFORMED UNDER THE SUPERVISION OF RADCON AND CHEMISTRY TO ENSURE THAT POTENTIALLY CONTAMINATED WATER IS NOT RELEASED TO THE ENVIRONMENT, OR USED WITH THE TEMPORARY CHILLER. FILLING, VENTING, AND DRAINING OPERATIONS MAY ALSO REQUIRE A SECONDARY BREACH PERMIT.

REFERENCE DRAWINGS:

MEL....................VALVE MARKER TAG TABULATION
1-47E610-70-1........CONTROL DIAGRAM-REACTOR BUILDING CCW SYSTEM
MEL....................INSTRUMENT TABULATION FOR CCW SYSTEM
0-47E800-1...........FLOW DIAGRAM-GENERAL PLANT SYSTEMS
47W464 SERIES........PHYSICAL PIPING DRAWINGS
1-47E810-1...........FLOW DIAGRAM-REACTOR WATER CLEANUP SYSTEM
1-47E610-43-1 & 2....CONTROL DIAGRAM-SAMPLING AND WATER QUALITY
1-47E844-2...........FLOW DIAGRAM-RAW COOLING WATER
1-47E852-1,-2........FLOW DIAGRAM-DRAINAGE
1-47E855-1...........FLOW DIAGRAM-FUEL POOL COOLING
1-47E856-2...........FLOW DIAGRAM-DEMINERALIZED WATER
1-47E610-32-2........CONTROL DIAGRAM-CONTROL AIR SYSTEM
0-47E800-2...........MECHANICAL SYMBOLS AND FLOW DIAGRAM DRAWING INDEX
2-47E867-3...........FLOW DIAGRAM-SAMPLING & WATER QUALITY SYSTEM
3-47E867-3...........FLOW DIAGRAM-SAMPLING & WATER QUALITY SYSTEM

LEGEND

A PERPENDICULAR LINE REPRESENTS UNIT 1/UNIT 2 AND UNIT 2/UNIT 3 BOUNDARY ISOLATION POINT. VALVES CLOSED FOR ISOLATION PURPOSES.

PHYSICAL BARRIER OR BOUNDARY BETWEEN UNITS

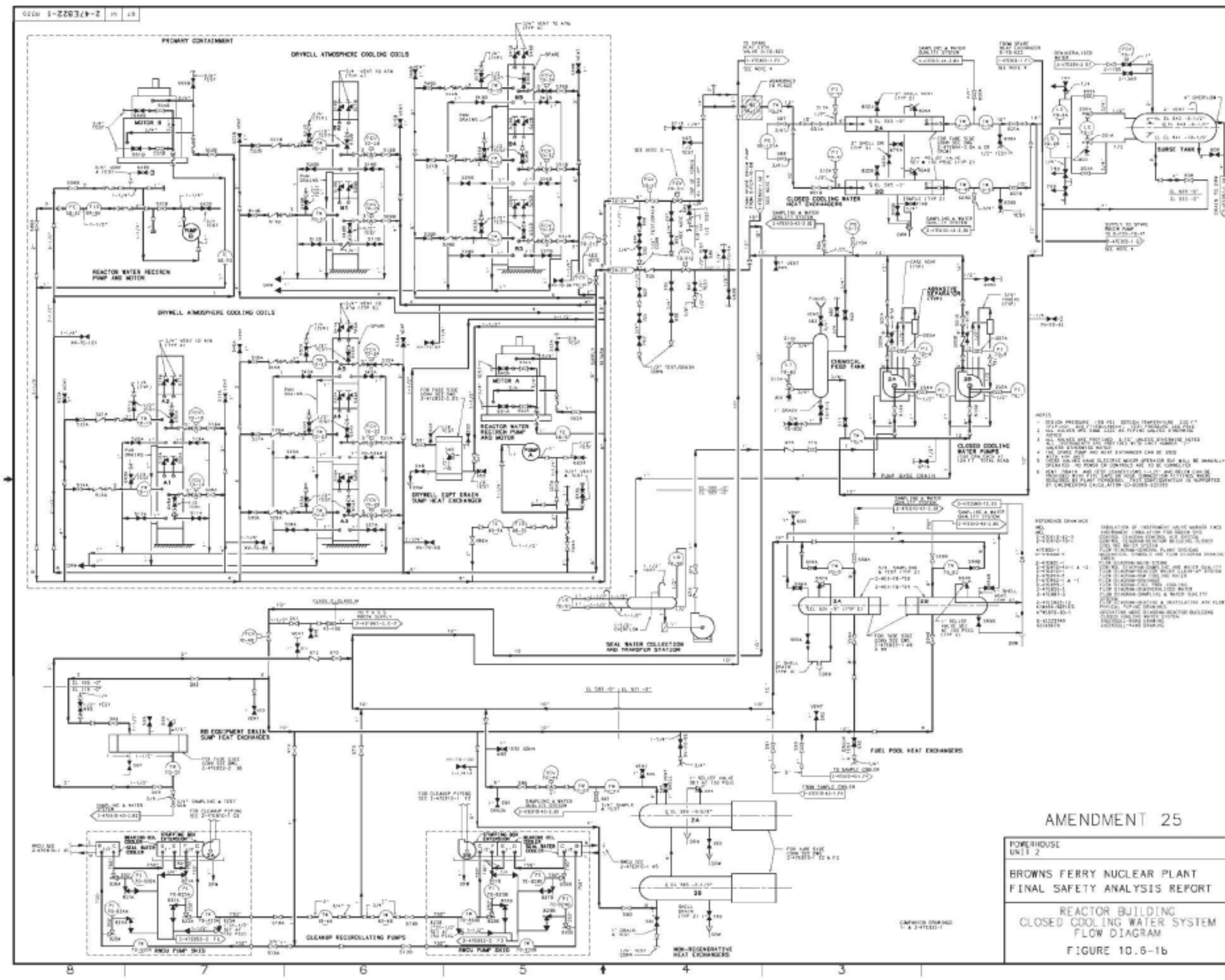
AMENDMENT 28

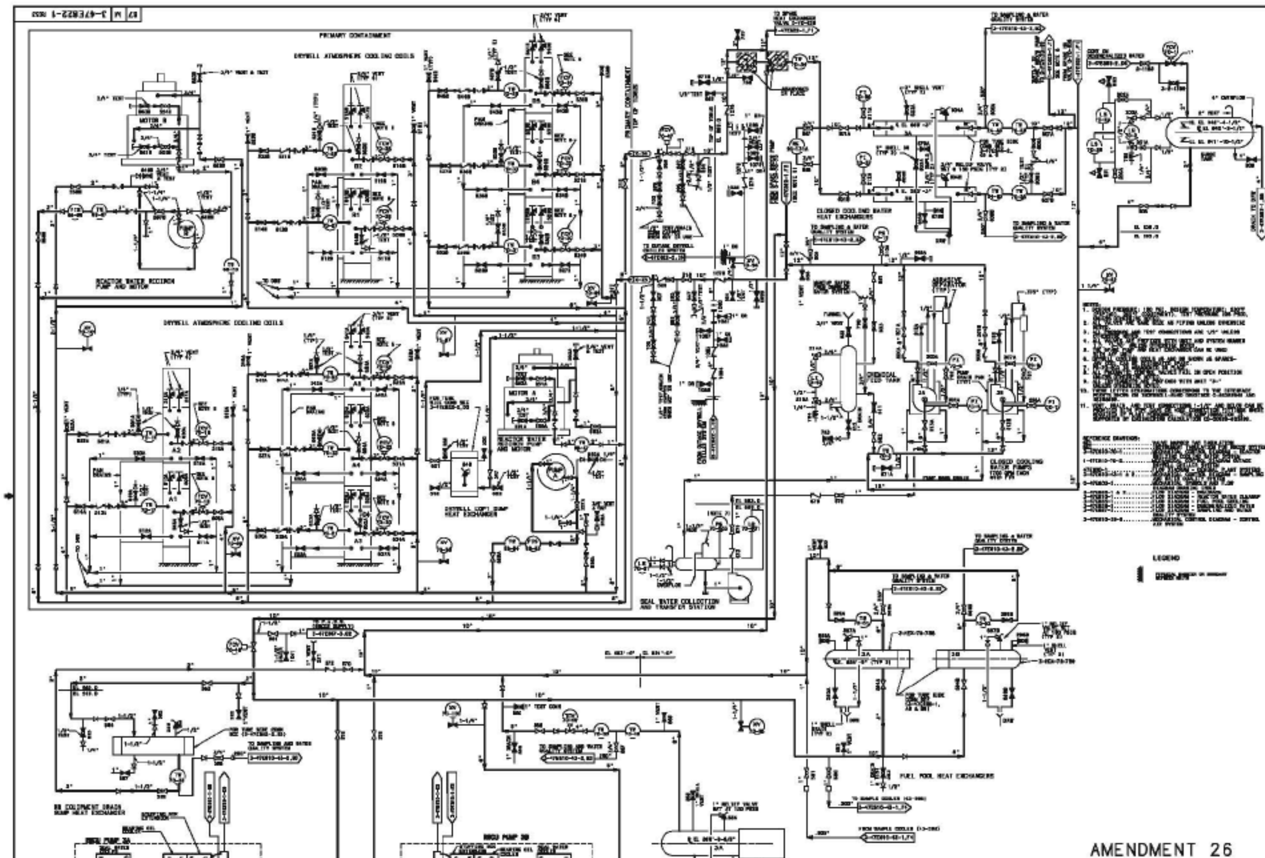
POWERHOUSE
UNIT 1 & 0

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

REACTOR BUILDING
CLOSED COOLING WATER SYSTEM
FLOW DIAGRAM

FIGURE 10.6-1a


PRIMARY CONTAINMENT
DRYWELL ATMOSPHERE COOLING COILS
REACTOR WATER RECIRC PUMP AND MOTOR
MOTOR B
MOTOR A
REACTOR WATER RECIRC PUMP AND MOTOR
DRYWELL EQPT DRAIN SUMP HEAT EXCHANGER
SEAL WATER COLLECTION AND TRANSFER STATION
CLOSED COOLING WATER HEAT EXCHANGERS
SURGE TANK
CHEMICAL FEED TANK
CLOSED COOLING WATER PUMPS
RB EQUIPMENT DRAIN SUMP HEAT EXCHANGER
FUEL POOL HEAT EXCHANGERS
CLEANUP RECIRCULATING PUMPS
RWCU PUMP SKID
NON-REGENERATIVE HEAT EXCHANGERS
AMENDMENT 25
POWERHOUSE
UNIT 2
BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT
REACTOR BUILDING
CLOSED COOLING WATER SYSTEM
FLOW DIAGRAM
FIGURE 10.6-1b



AMENDMENT 26

POWERHOUSE
UNIT 3

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

REACTOR BLDG CLOSED COOLING WATER
SYSTEM - FLOW DIAGRAM

FIGURE 10.6-1c

Figures 10.6-2a through 10.6-2d
(Deleted by Amendment 22)

10.7 RAW COOLING WATER SYSTEM

10.7.1 Power Generation Objective

The objective of the Raw Cooling Water System 024 is to remove heat from turbine-associated equipment and accessories located in and adjacent to the Turbine Building, from the Reactor Building Closed Cooling Water heat exchangers, and from other reactor-associated equipment that utilizes raw cooling water.

10.7.2 Power Generation Design Basis

1. The system shall be capable of supplying the flow requirements of the equipment it serves during the full range of operation.

2. Under loss of power conditions the system shall not be required to perform any essential function, but shall be available for equipment protection and to facilitate restarting.

3. After a design basis accident the system shall not be required for safe shutdown.

10.7.3 Description

The Raw Cooling Water System (shown in Figures 10.7-1a sheets 1, 2, and 3 and 10.7-1b sheets 1, 2, 3, 4, 5, 6, and 7) furnishes cooling water to the following:

a. Turbine lube oil coolers,
b. Generator stator water coolers,
c. Generator hydrogen coolers,
d. Reactor feed pump turbine oil coolers,
e. Service and control air compressors,
f. Steam jet air ejector precoolers,
g. Generator exciter air coolers,
h. Air-conditioning condensers,
i. Recirculation pump VFDs Heat Exchangers,
j. Reactor Building Closed Cooling Water heat exchangers,
k. Condensate booster pump motor heat exchanger and
l. Other miscellaneous coolers
m. Condensate booster pump motor heat exchanger (Unit 2)
n. Condensate booster pump motor heat exchanger (Unit 3)

There are 12 main Raw Cooling Water System pumps, located in the Turbine Building at elevation 557 ft., and they are supplied with river water from the condenser circulating water conduits of each unit, which is passed through fine

mesh strainers. The suction headers for Units 1 and 2 are interconnected. The Unit 3 suction header is served separately due to physical limitation. Three pumps are required per unit and one is provided as a spare for Units 1 and 2. There is a total of 5 RCW pumps in Unit 3. All pumps discharge into a common header. Automatic temperature controls are provided for most of the coolers supplied by the Raw Cooling Water System. The system control diagram is shown in Figures 10.7-2 sheets 1, 2, 3, 4, 5, and 6.

Under loss of power conditions two of the 12 raw cooling water pumps (1D & 3D) may be operated, powered by standby diesel generators, but only if standby diesel-generated power reserve margin is available. They would supply water to selected turbine auxiliary equipment to prevent over heating and subsequent equipment damage and to assist getting back into operation. The booster pumps are not required at this time. The raw cooling water pumps are not started automatically in the loss-of-power mode since they are not required for safe shutdown. The automatic start feature is also defeated when offsite power is available in the design basis accident mode to minimize the loading on the safety-related buses. RCW pump 1D is tripped by the Common Accident Signal Logic to prevent overloading 4kV Shutdown Bus 1. As ECCS pumps are secured and loads are reduced on the 4kV Shutdown Buses, the RCW pump may be manually re-started by operators as desired to support long term post accident recovery and shutdown of the non-accident units.

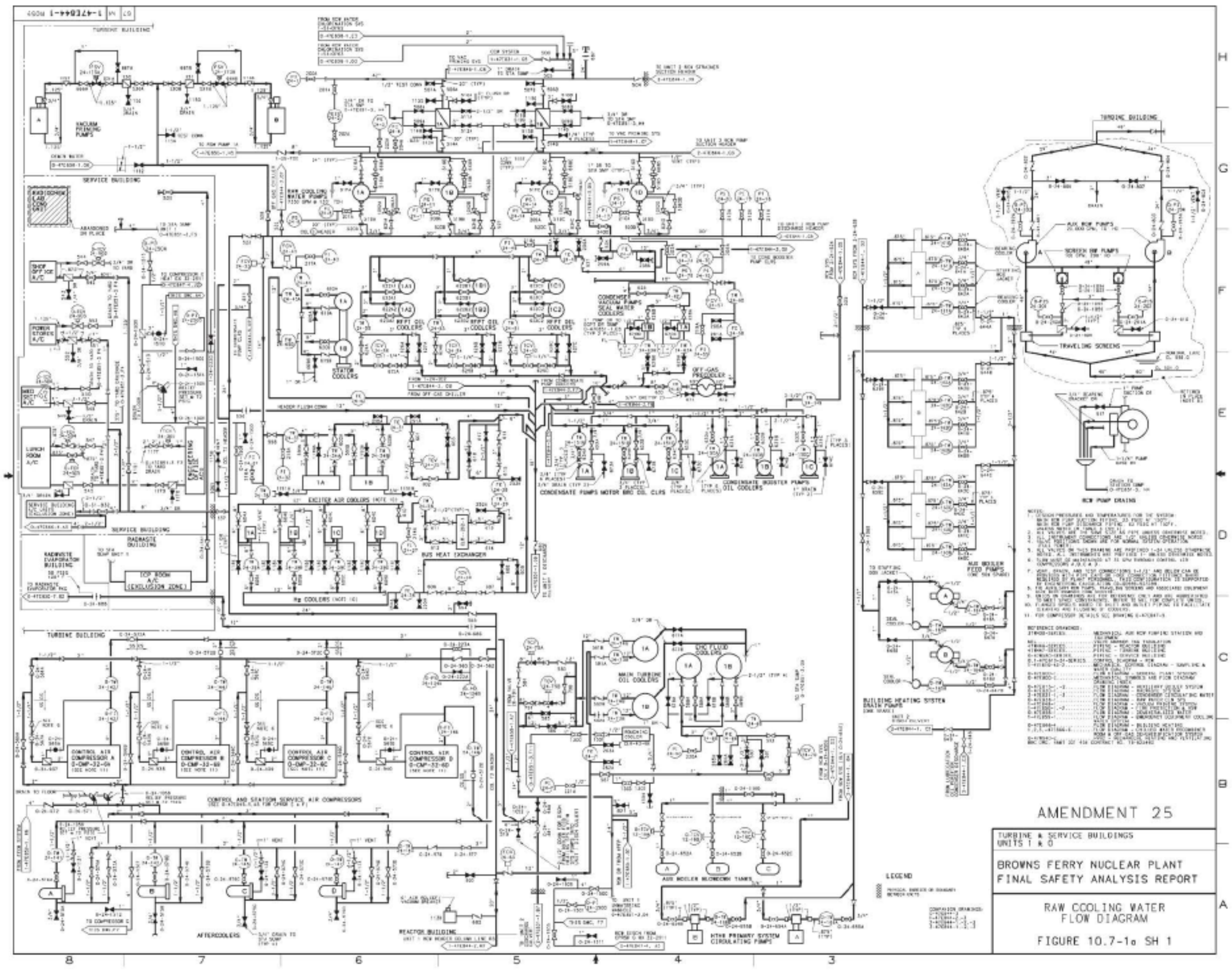
A backup cooling water supply for the Reactor Building Closed Cooling Water heat exchangers and control air compressors is provided from the Emergency Equipment Cooling Water System 067. The EECW system automatically supplies water in place of raw cooling water if the Reactor Building Closed Cooling Water System or Air Compressor System raw cooling water pressure is lost when EECW header pressure is above a specified minimum pressure.

There are four cooling water inlet penetrations of secondary containment, each meeting seismic Class II pressure boundary retention requirements.

Raw cooling water is chemically treated consistent with NPDES permit limitations. In accordance with NRC Bulletin 81-03, raw cooling water is also chemically treated during peak clam spawning periods to ensure clam control.

10.7.4 Inspection and Testing

No special tests are required. Routine visual inspection of the system components, instrumentation, and trouble alarms are adequate to verify system operability.



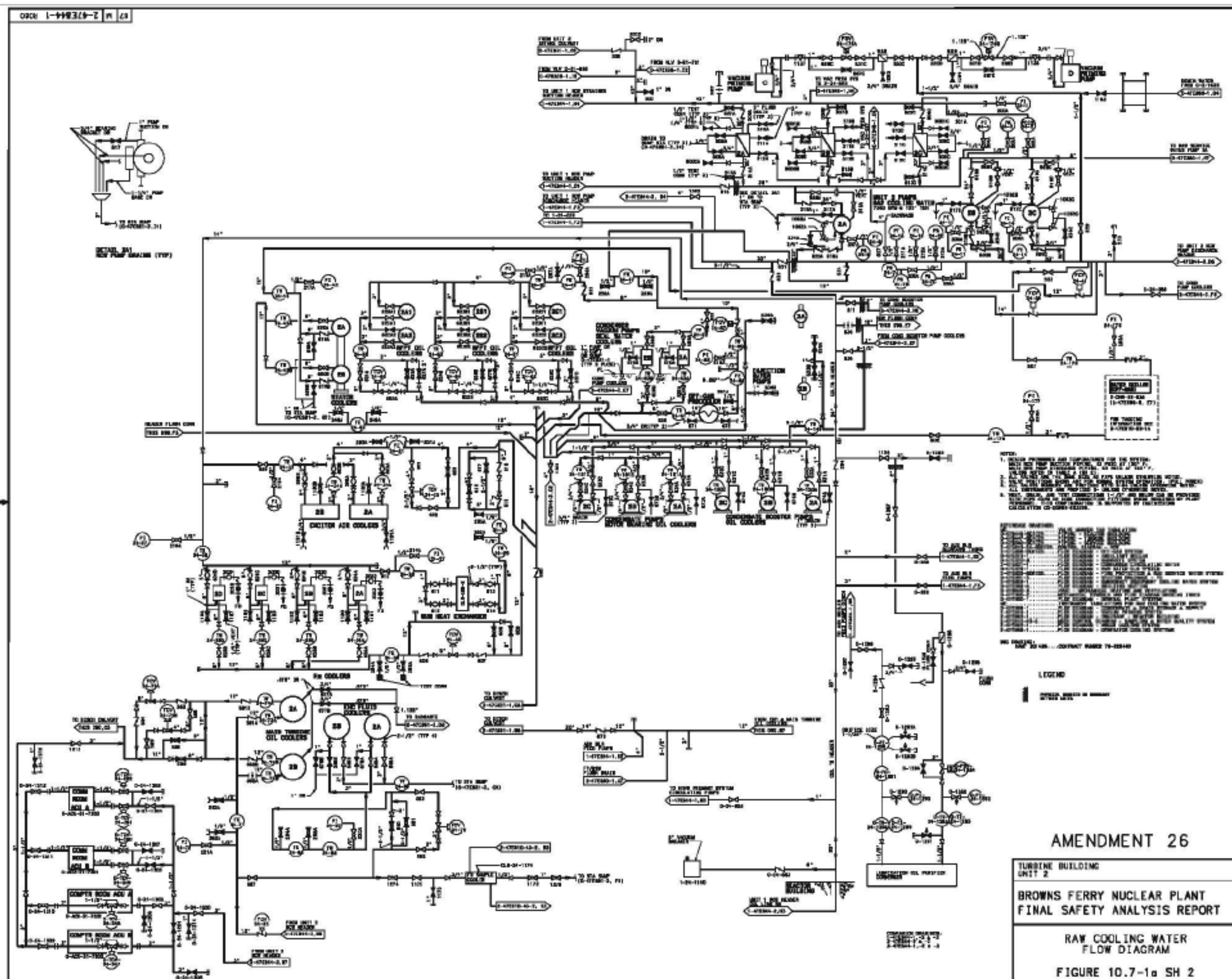
AMENDMENT 25

TURBINE & SERVICE BUILDINGS
UNITS 1 & 0

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

RAW COOLING WATER
FLOW DIAGRAM

FIGURE 10.7-1a SH 1



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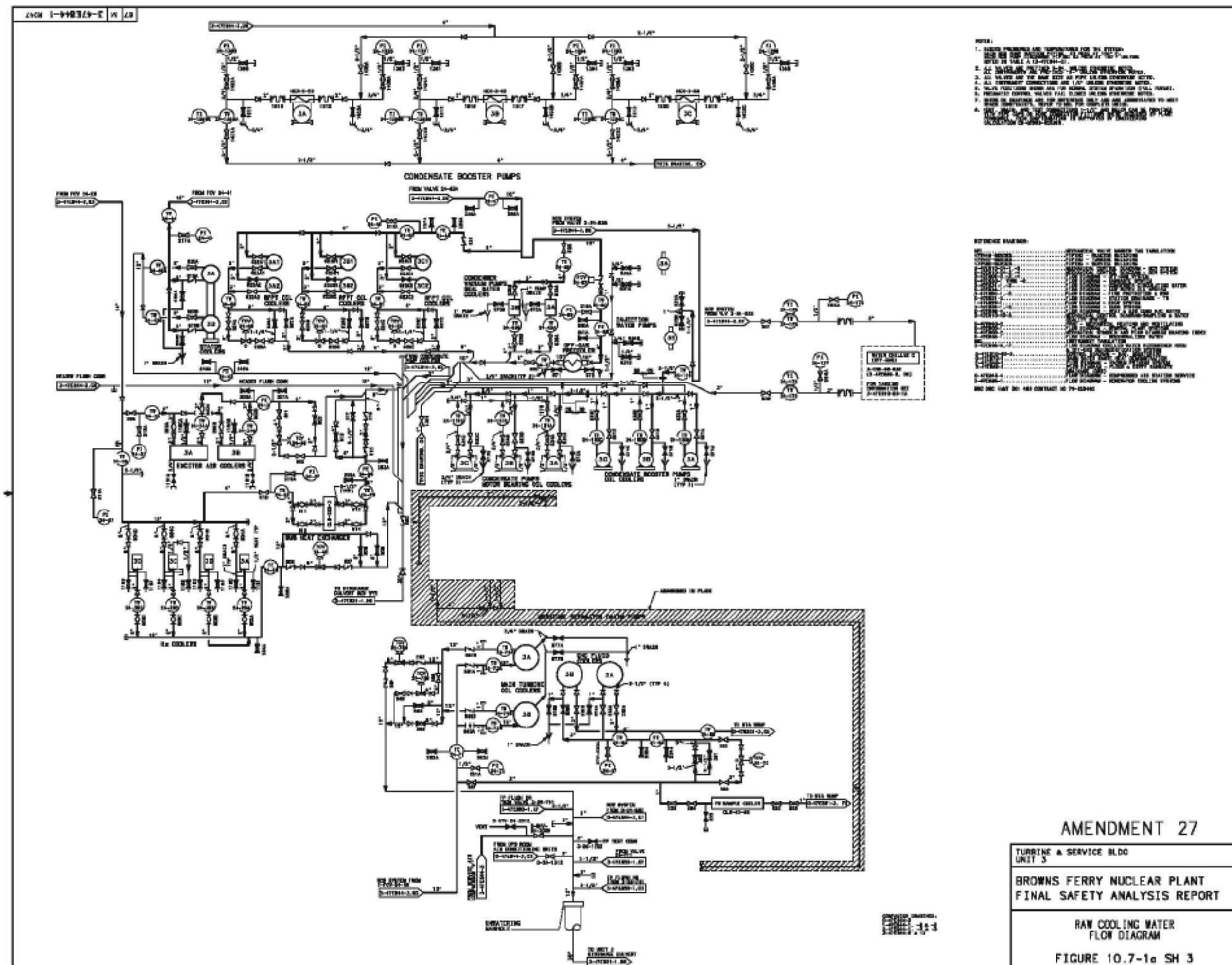
TURBINE BUILDING
UNIT 2

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

RAW COOLING WATER
FLOW DIAGRAM

FIGURE 10.7-1a SH 2



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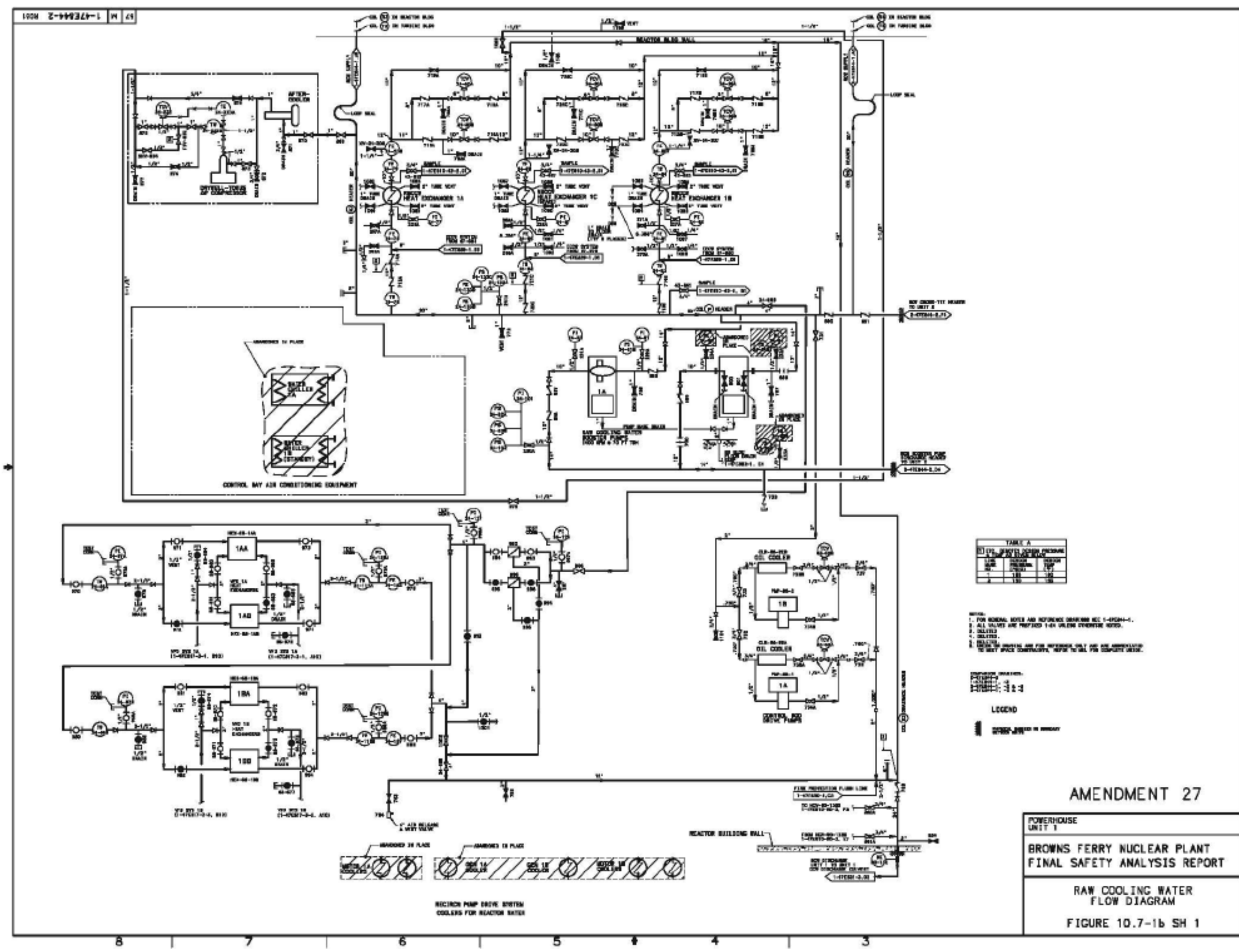
TURBINE & SERVICE BLDG
UNIT 3

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

RAW COOLING WATER
FLOW DIAGRAM

FIGURE 10.7-1a SH 3


AMENDMENT 27
POWERHOUSE
UNIT 1
BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT
RAW COOLING WATER
FLOW DIAGRAM
FIGURE 10.7-1b SH 1

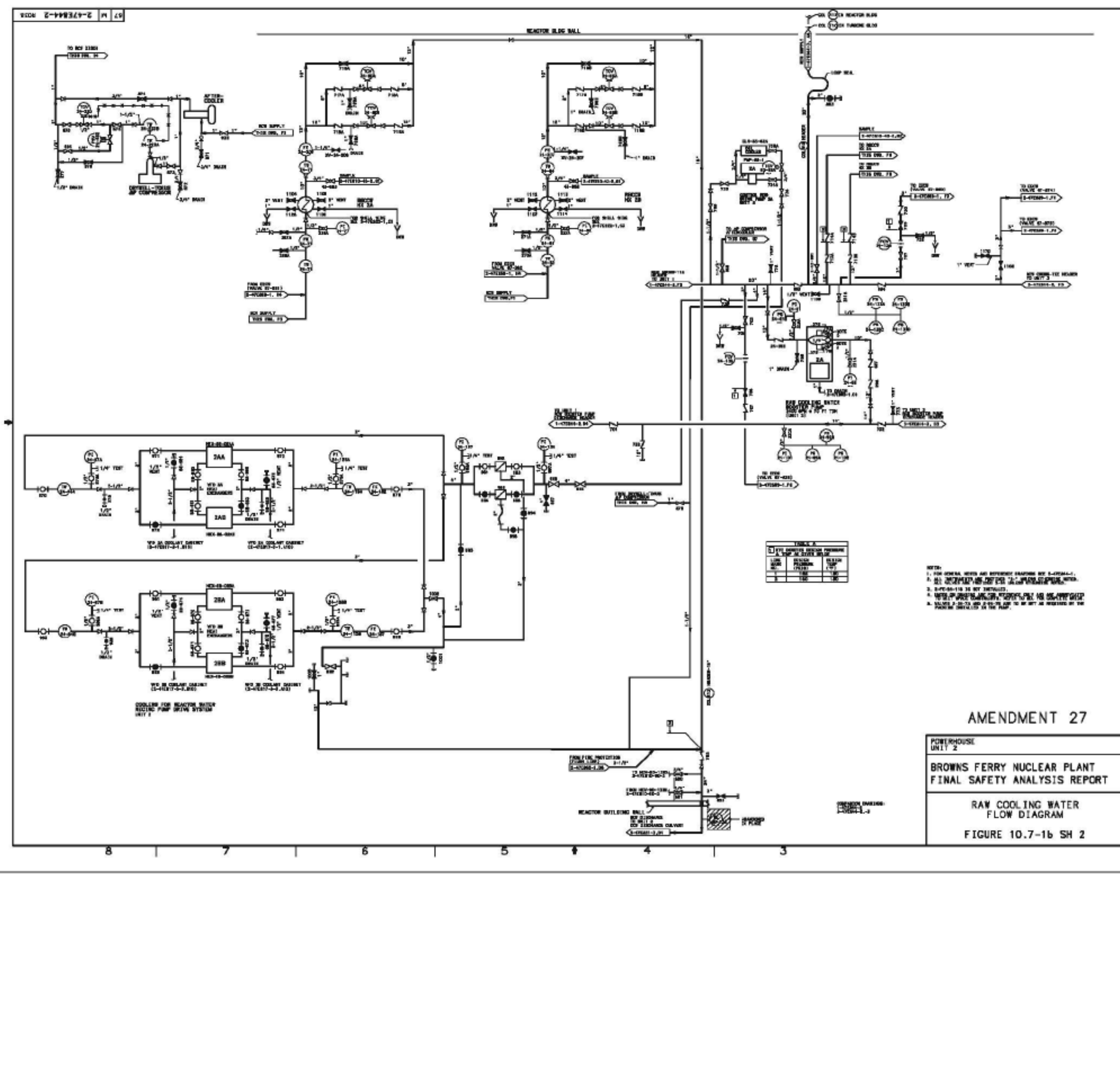
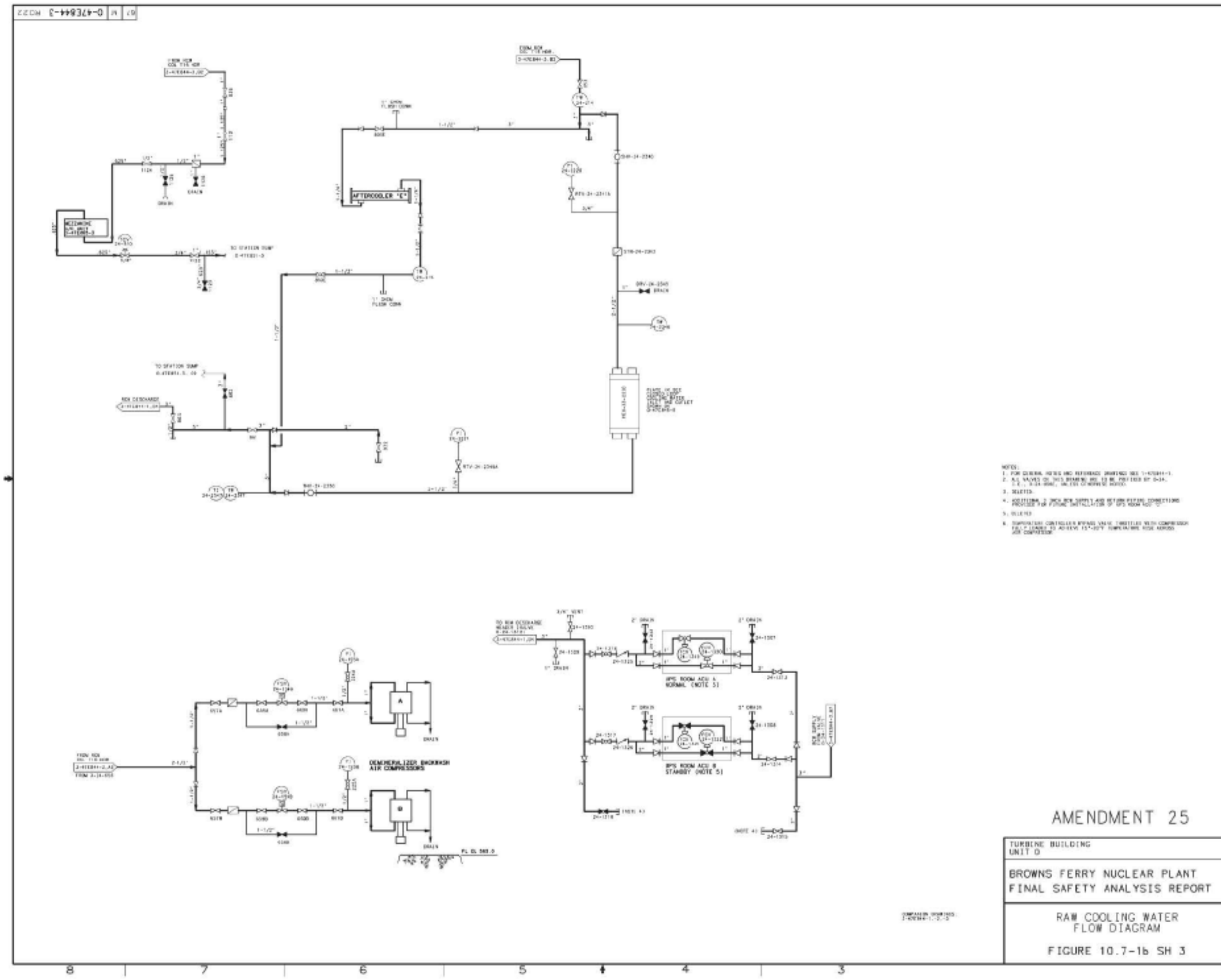


FIGURE 10.7-1b SH 2

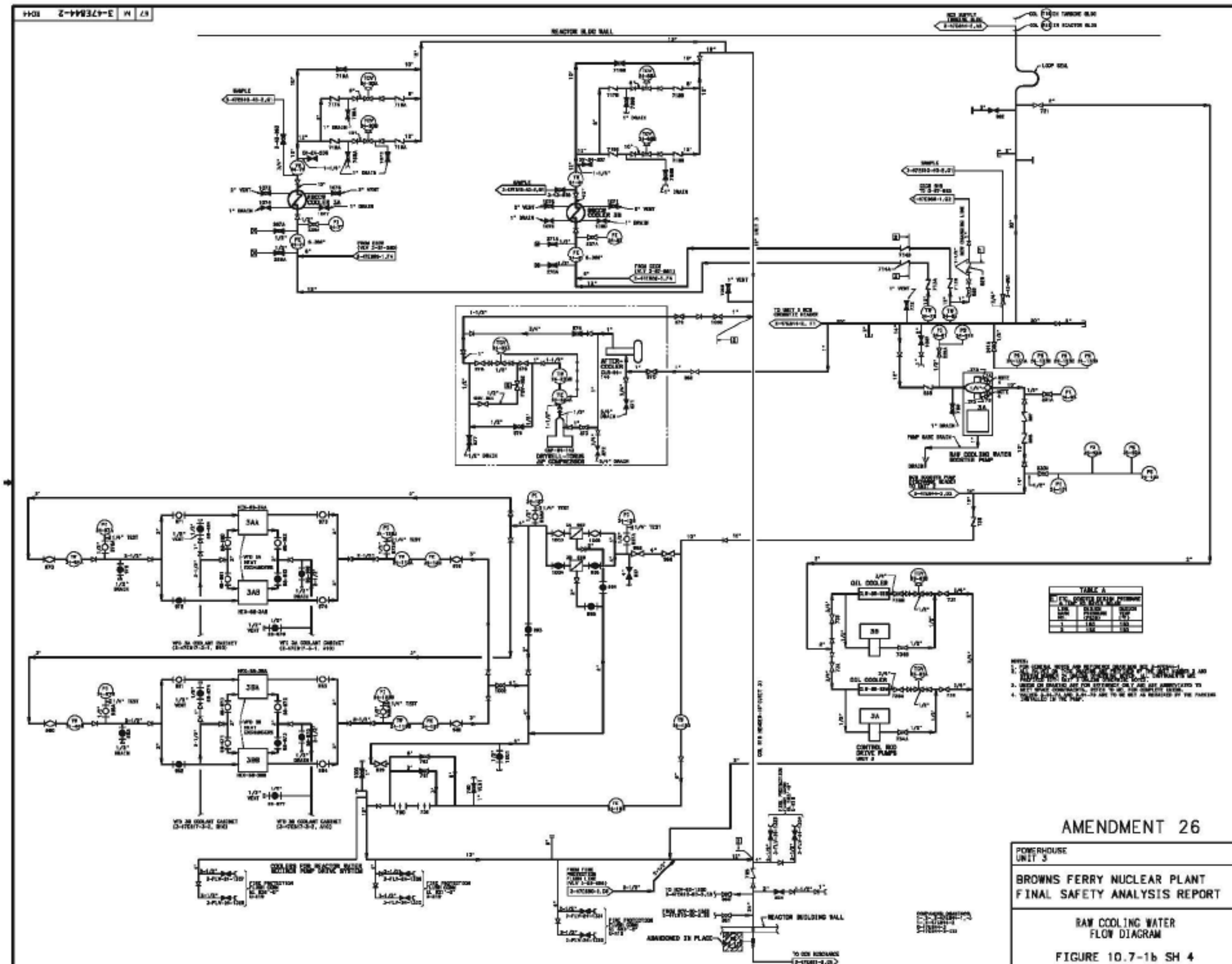


TURBINE BUILDING
UNIT 0

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

RAW COOLING WATER
FLOW DIAGRAM

FIGURE 10.7-1b SH 3



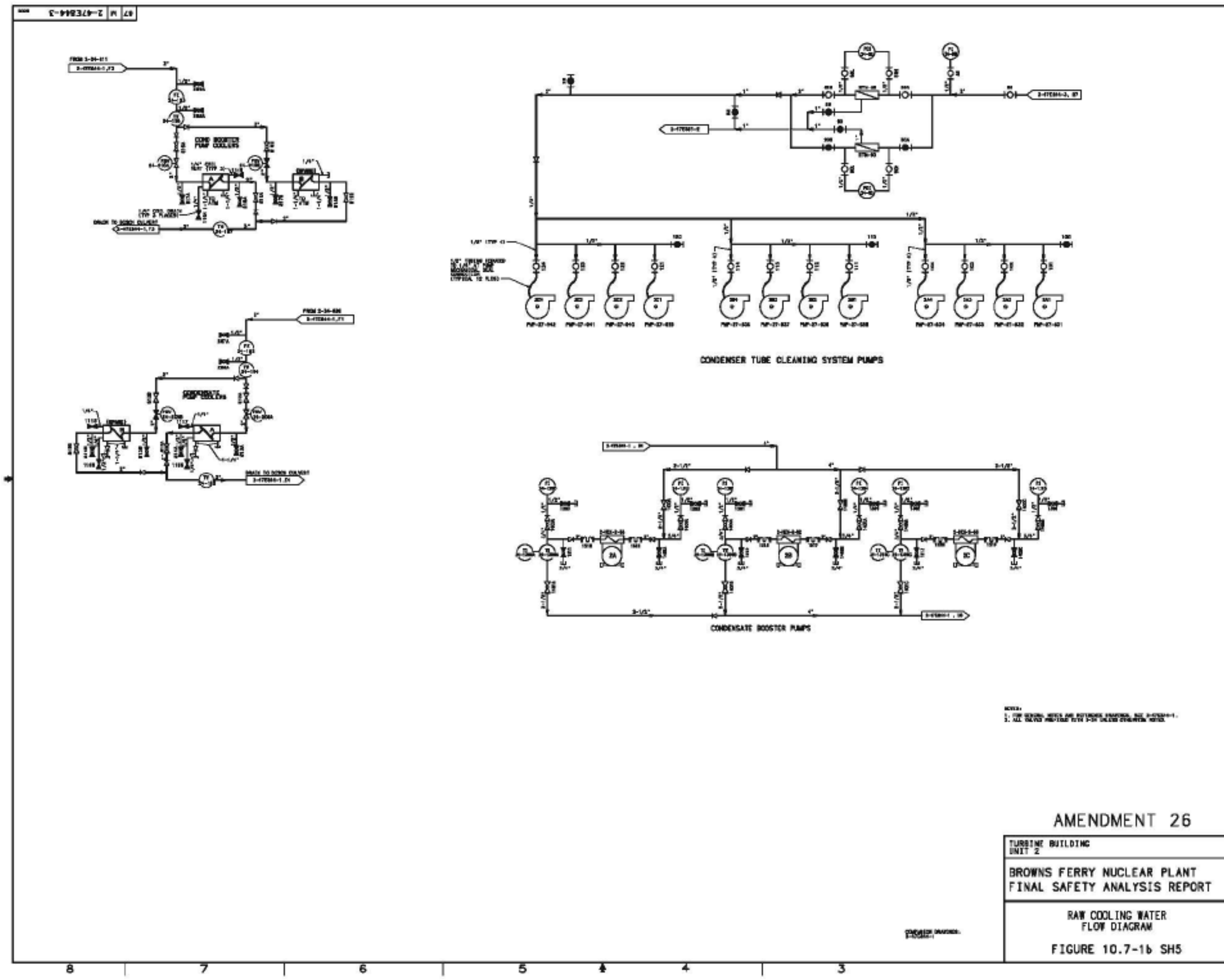
AMENDMENT 26

POWERHOUSE
UNIT 3

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

RAW COOLING WATER
FLOW DIAGRAM

FIGURE 10.7-1b SH 4



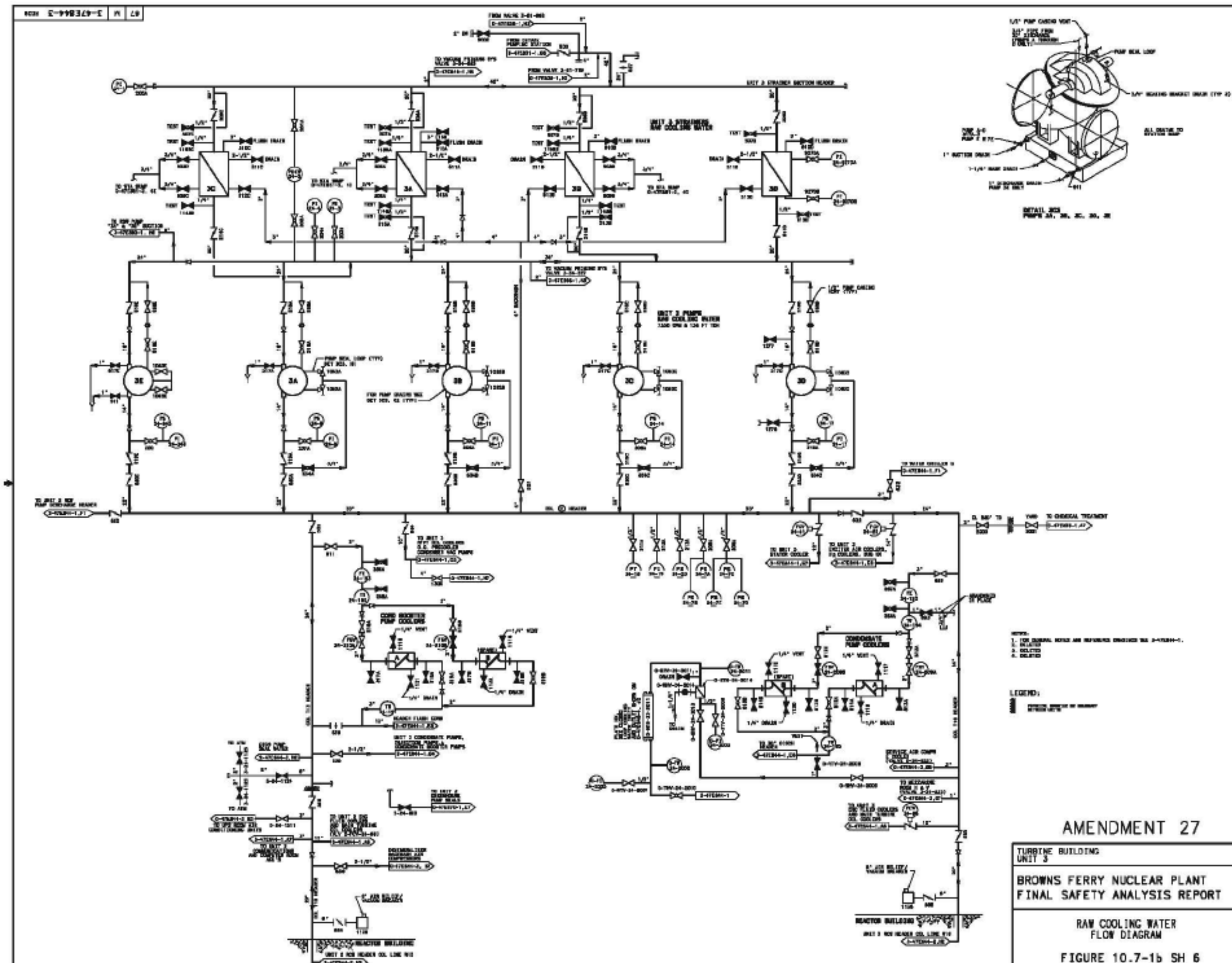
AMENDMENT 26

TURBINE BUILDING
UNIT 2

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

RAW COOLING WATER
FLOW DIAGRAM

FIGURE 10.7-1b SH5



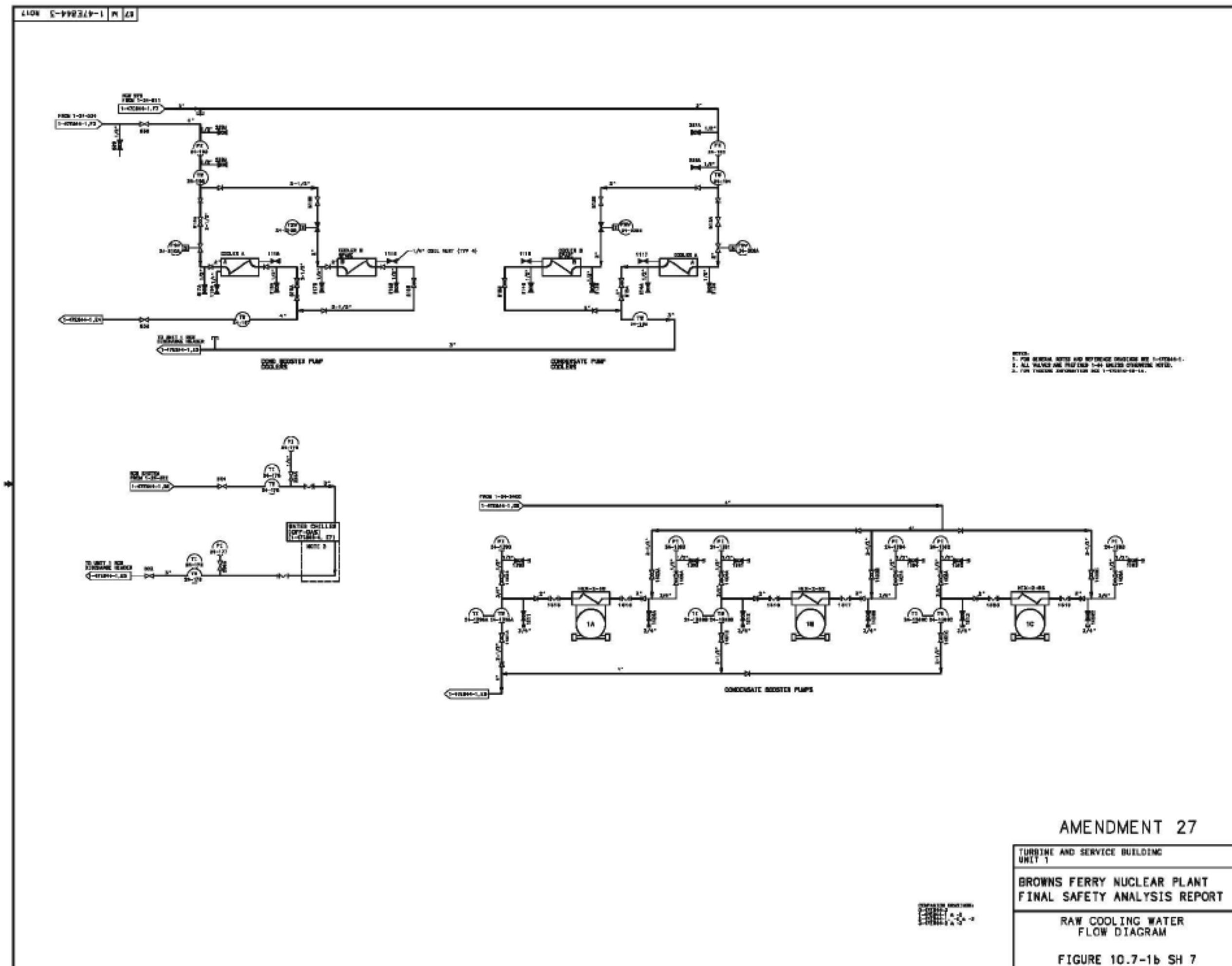
AMENDMENT 27

TURBINE BUILDING
UNIT 3

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

RAW COOLING WATER
FLOW DIAGRAM

FIGURE 10.7-1b SH 6



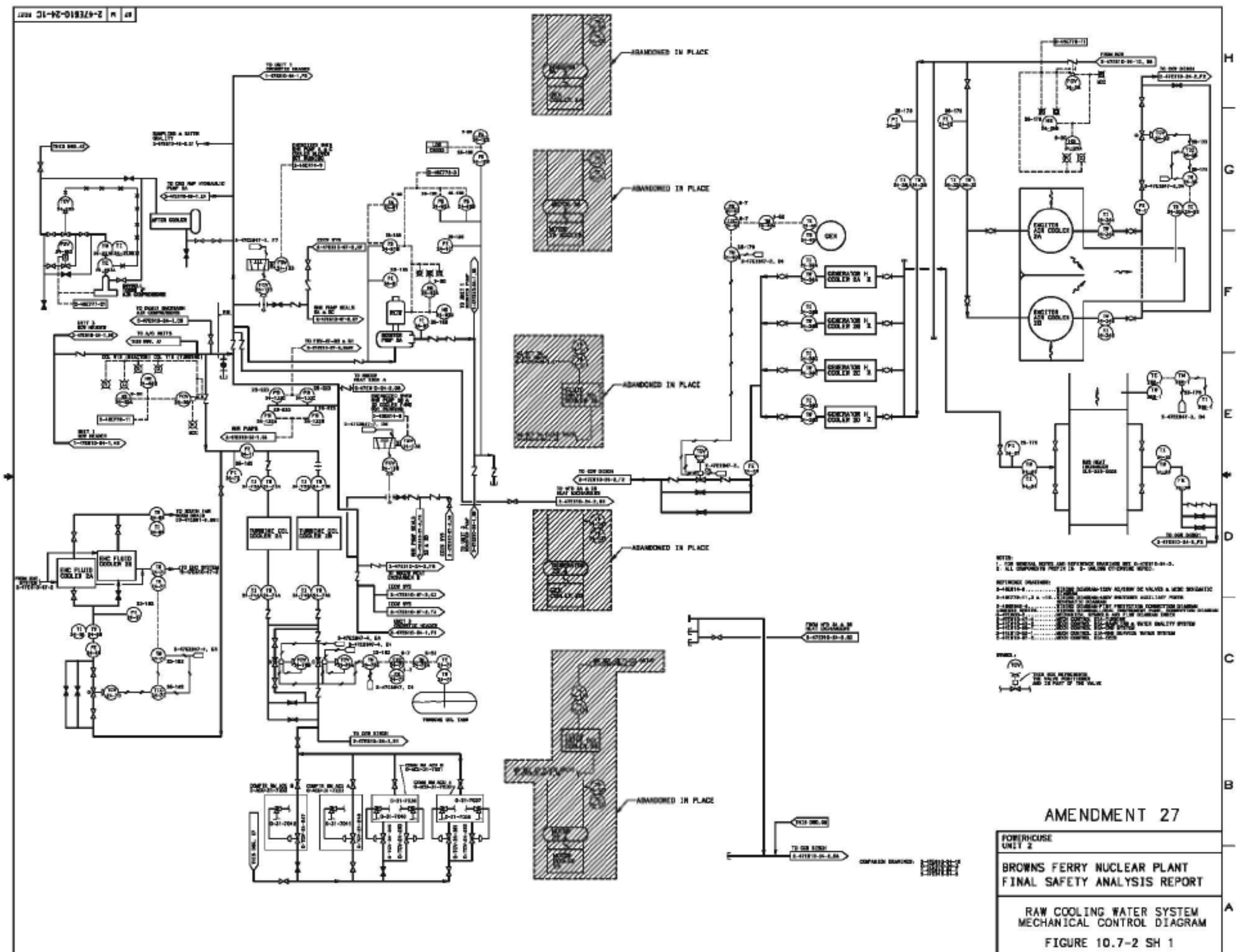
AMENDMENT 27

TURBINE AND SERVICE BUILDING
UNIT 1

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

RAW COOLING WATER
FLOW DIAGRAM

FIGURE 10.7-1b SH 7



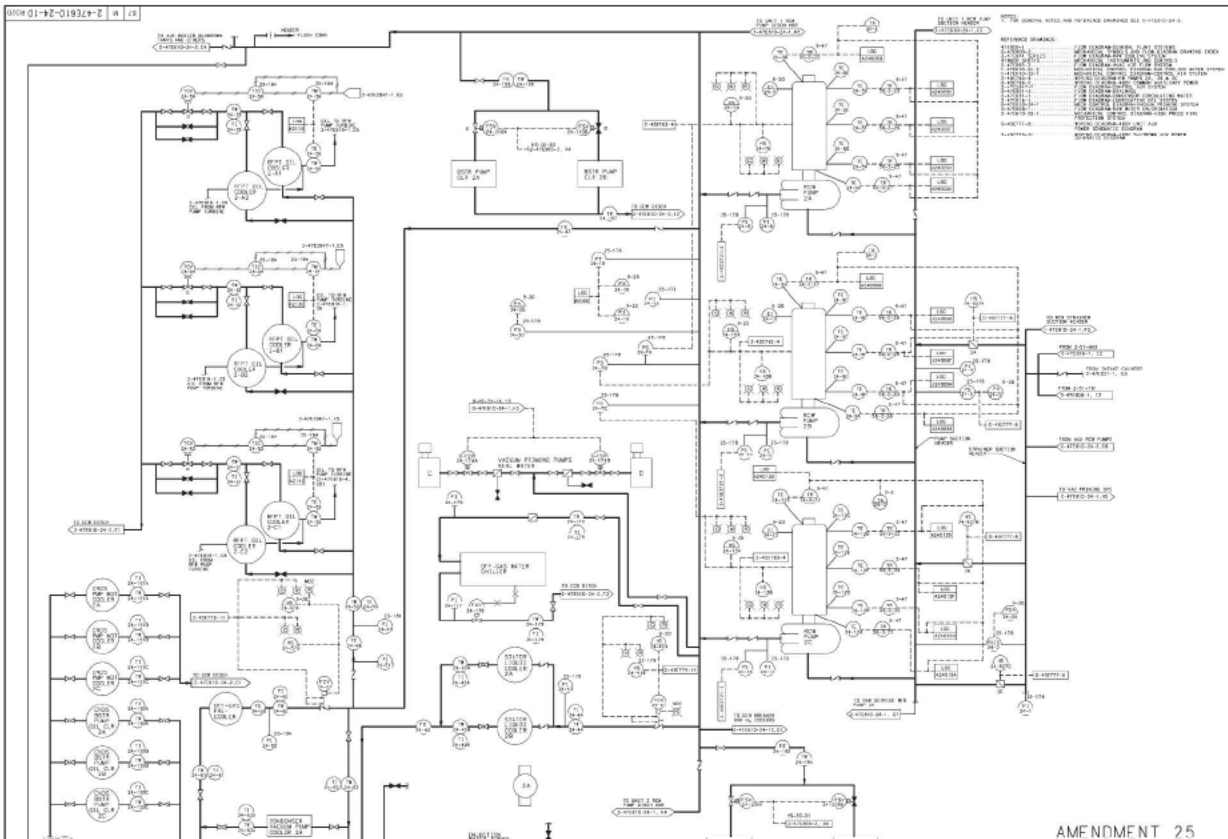
AMENDMENT 27

POWERHOUSE
UNIT 2

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

RAW COOLING WATER SYSTEM
MECHANICAL CONTROL DIAGRAM

FIGURE 10.7-2 SH 1



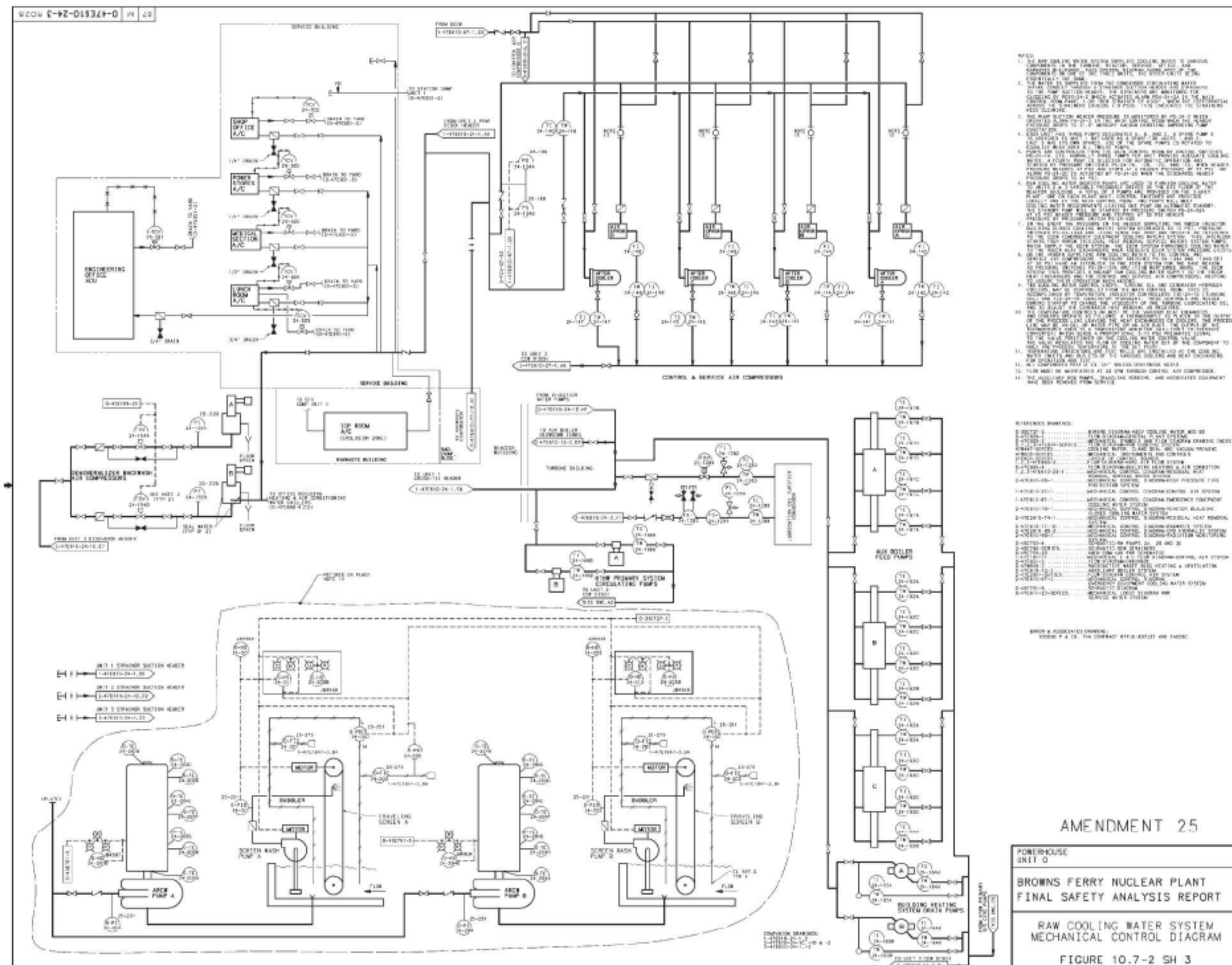
AMENDMENT 25

POWERHOUSE UNIT 2

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

RAW COOLING WATER SYSTEM
MECHANICAL CONTROL DIAGRAM

FIGURE 10.7-2 SH 2



POWERHOUSE
UNIT 0

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

RAW COOLING WATER SYSTEM
MECHANICAL CONTROL DIAGRAM

FIGURE 10.7-2 SH 3

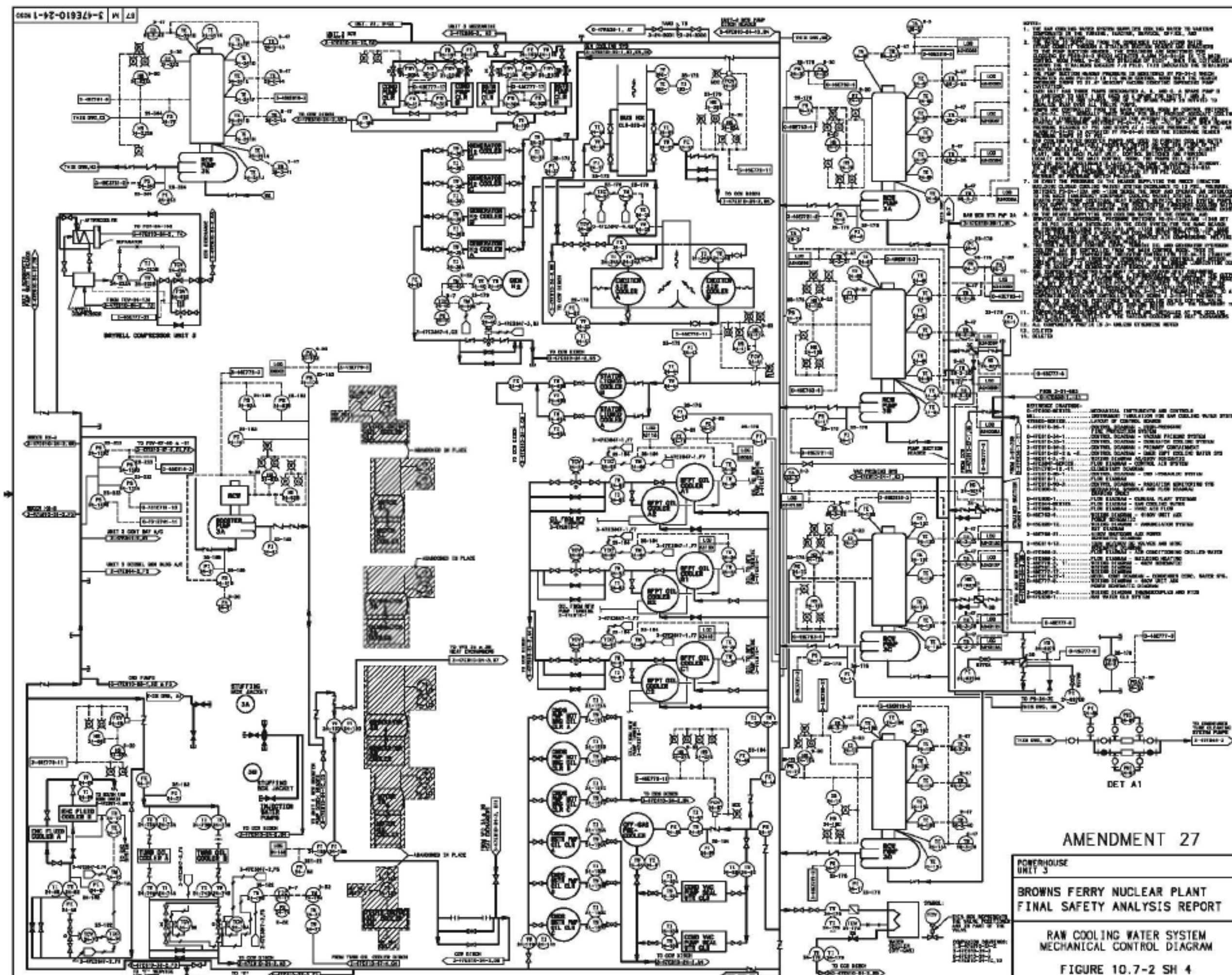
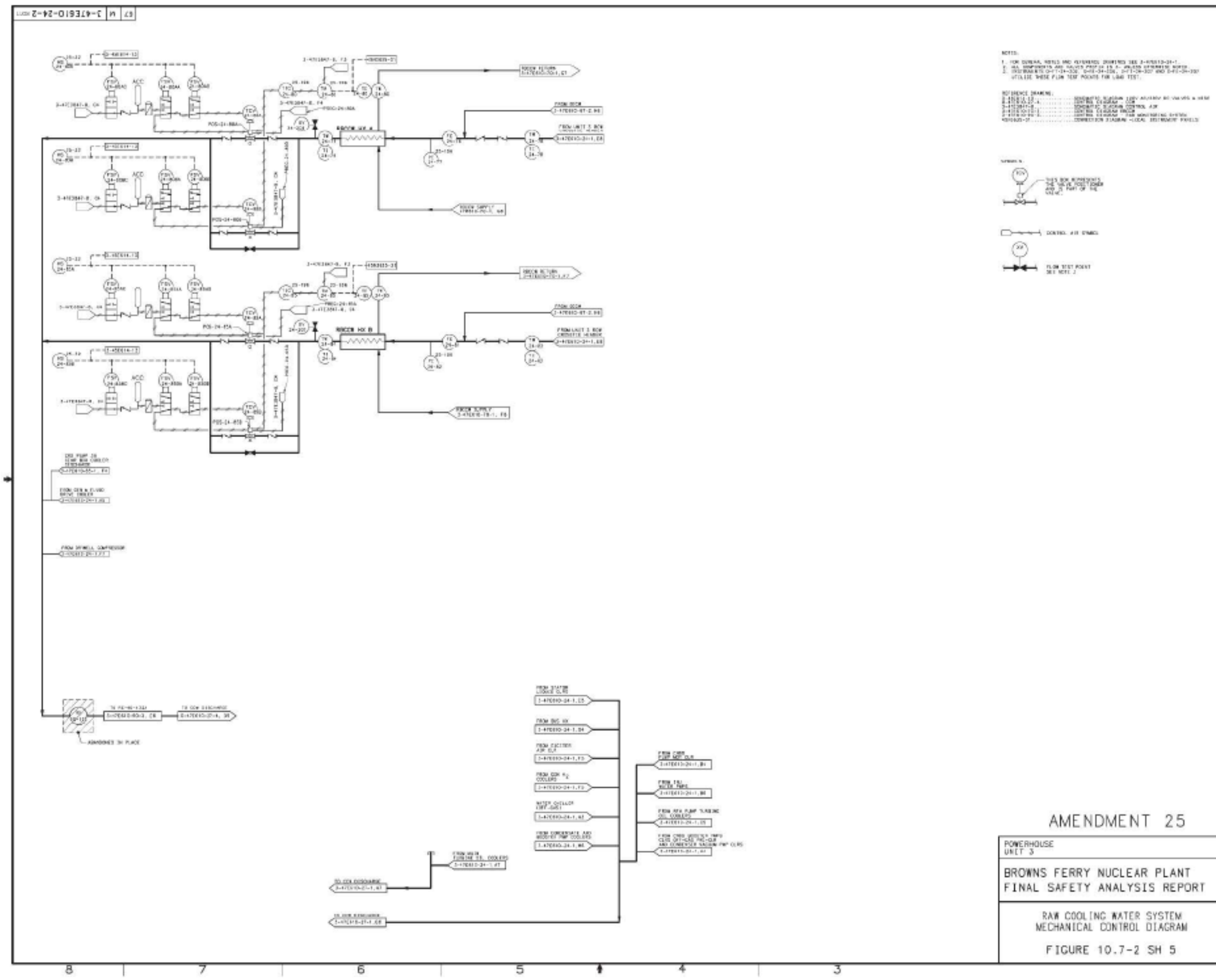


FIGURE 10.7-2 SH 4

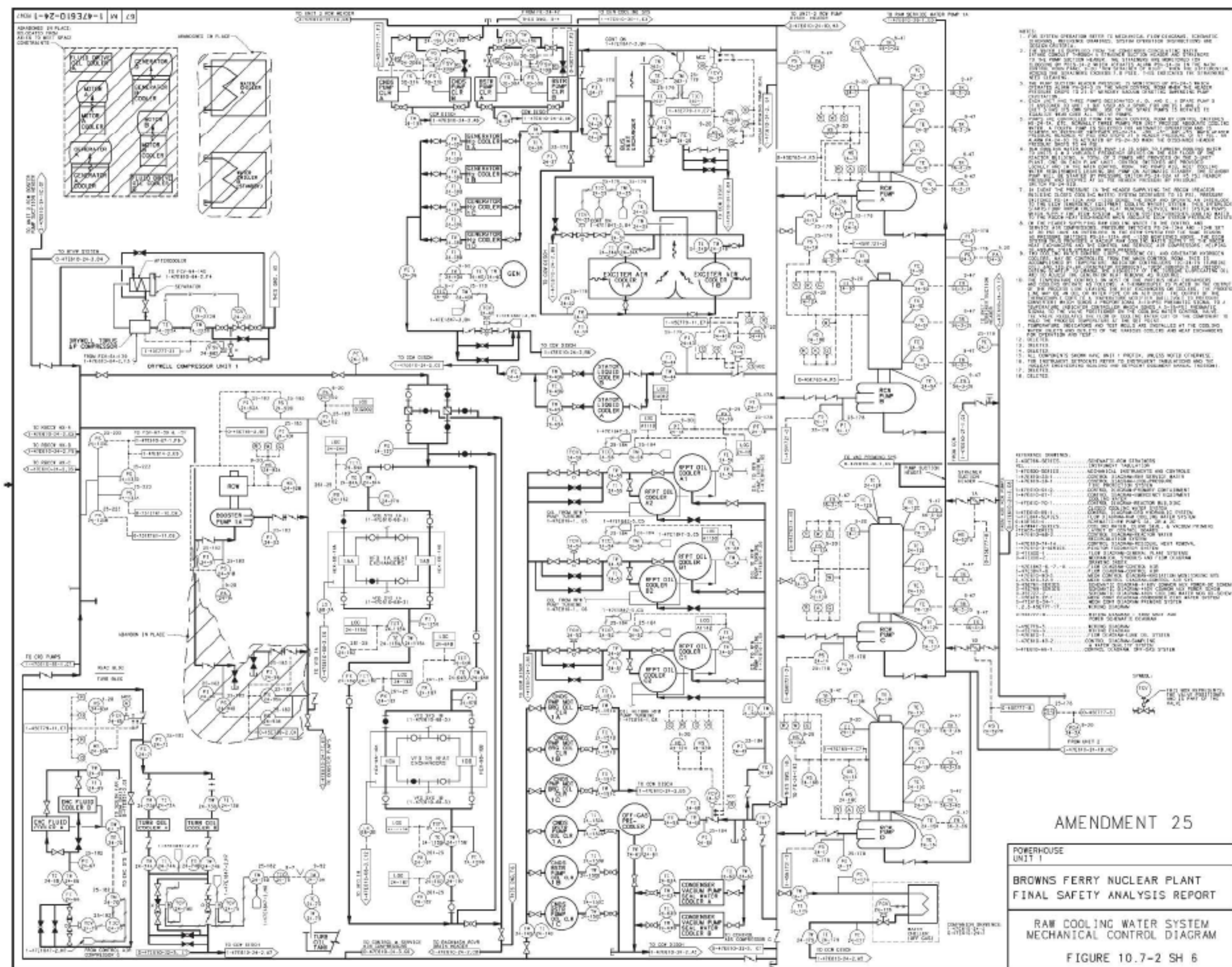


POWERHOUSE
UNIT 3

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

RAW COOLING WATER SYSTEM
MECHANICAL CONTROL DIAGRAM

FIGURE 10.7-2 SH 5


AMENDMENT 25
POWERHOUSE
UNIT 1
BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT
RAW COOLING WATER SYSTEM
MECHANICAL CONTROL DIAGRAM
FIGURE 10.7-2 SH 6

10.8 RAW SERVICE WATER SYSTEM

10.8.1 Power Generation Objective

The objective of the Raw Service Water System is to supply river water for yard-watering, cooling for plant equipment which the Raw Cooling Water System may not conveniently serve, and to function as a keep-fill system for the raw water Fire Protection System.

10.8.2 Power Generation Design Basis

1. The system shall be capable of supplying all normal plant requirements for raw service water.

2. The system shall not be required for safe shutdown.

10.8.3 Description

The Raw Service Water System furnishes water for yard-watering, cooling for miscellaneous plant equipment which require small quantities of cooling water, and functions as a keep-fill system for the raw water fire protection system. The Raw Service Water System is supplied river water from the condenser circulating water inlet conduit through a strainer section to the main raw cooling water pump suction header for each unit. Unit 1 and Unit 2 each have one RSW pump and Unit 3 has two RSW pumps. Therefore, four pumps (375 gpm 200-foot-tdh) supply the common plant system. The pumps discharge into a distribution system common to the raw service water and fire protection systems. Two 10,000-gallon capacity storage tanks are located atop the reactor building. Water level in the tanks controls operation of the raw service water pumps except during operation of the high-pressure fire protection pumps or when the RSW pumps are in manual with the storage tanks isolated from the system. When the high-pressure fire protection pumps are operating, the raw service water pumps will automatically be deenergized (unless in manual) and the raw service water storage tanks will also be isolated from the system.

The raw service water pumps are powered from the 480-V turbine MOV boards 1C, 2C, 3B, and 3C. During the loss-of-power mode they may be backfed through these boards, if power is available from the diesel generators.

For flow diagrams of this system, refer to Subsection 10.11, "Fire Protection Systems."

10.8.4 Inspection and Testing

No special tests are required. Routine visual inspection of the system components, instrumentation, and trouble alarms is adequate to verify system operability. The Raw Service Water System is chemically treated consistent with NPDES permit limitations.

10.9 RHR SERVICE WATER SYSTEM

10.9.1 Safety Objective

The ultimate safety objective of the RHR Service Water System is heat removal from the primary water of the RHR Systems, using cooling water from Wheeler Reservoir delivered by service water pumps located within the intake pumping station. The system also provides standby core and containment cooling and supplies water to the Emergency Equipment Cooling Water System.

10.9.2 Safety Design Basis

1. The system shall be manually operated to supply cooling water to the primary RHR System heat exchangers and provide standby core and containment cooling, and shall be automatically operated to supply water to the EECW System.

2. Piping and equipment, including support structures, shall be designed to withstand the effects of an earthquake without failure.

3. The pumps on the intake pumping station shall be designed to withstand tornado winds and weather.

10.9.3 Description

The RHR Service Water System, although it is a plant-shared system, is remote-manually operated from each of the three unit control rooms to pump raw river water from the intake station to the RHR heat exchangers and provide standby core cooling (see Subsection 4.8). The RHR Service Water System operates automatically to supply water to the EECW System (see Subsection 10.10).

The RHRSW is a twelve-pump, four-header system with four pairs of pumps normally assigned to the RHR System and four additional pumps normally assigned to the EECW System. Each of the pairs (assigned to the RHRSW System) feeds one independent RHR service water header which, in turn, feeds one RHR heat exchanger in each unit. The four pumps assigned to the EECW (A3, B3, C3, D3) are also paired, one pair serving each of the two EECW headers. If necessary, one of each pair of pumps normally assigned to the RHRSW can be manually realigned to the EECW. The entire system is seismic Class I. On a per-unit and -plant basis, the system provides several ways to supply water to shut down equipment. Each RHR service water and EECW header is physically, mechanically, and electrically independent of the alternate headers performing the same function. (See Appendix F for more details.) A flow diagram is shown in Figures 10.9-1a sheets 1, 2, and 3. System instrumentation is shown in Figures 10.9-1b, 10.9-2a, 10.9-2b, 10.9-2c, 10.9-2d, and 10.9-2e.

The RHRSW pumps take suction below the breach of Wheeler Dam (El. 529.0) and will, therefore, remain operable in the unlikely event of failure of the dam. Each pump is rated at 400-hp with a capacity of 4500 gpm at 275-foot total head. Each pump has the capacity to supply 100 percent of the cooling water required by one RHR heat exchanger.

The RHRSW-EECW system has the capability of utilizing FLEX (Flexible and Diverse Coping Mitigation Strategies) pumps to provide make-up water to the RPV and SFP, Drywell Spray header, and cooling water to the RHR heat exchanger and other essential equipment such as the Core Spray Room Coolers. The FLEX connections to RHRSW headers in Intake Pump Station Rooms B and D are designed to be utilized during a Beyond Design Basis External Event (BDBEE) condition for a loss of off-site power event and if the site Emergency Diesel Generators are inoperable.

The RHRSW-EECW system has the capability of utilizing FLEX (Flexible and Diverse Coping Mitigation Strategies) pumps to provide make-up water to the RPV and SFP, Drywell Spray header, and cooling water to the RHR heat exchanger and other essential equipment such as the Core Spray Room Coolers. The Flex connection to the RHRSW header in the Intake Pumping Station Room B is designed to be utilized during a Beyond Design Basis External Event (BDBEE) condition when a loss of off-site power and the site Emergency Diesel Generators are inoperable.

RHRSW pumps A2 and C2 are tripped by the Common Accident Signal Logic to prevent overloading 4kV Shutdown Bus 1. As ECCS pumps are secured and loads are reduced on the 4kV Shutdown Buses, the RHRSW pumps may be manually re-started by operators as desired to support long term post accident recovery and shutdown of the non-accident units. Within 1 hour following a design basis accident, six RHR service water pumps will be required to supply cooling water to the RHR heat exchangers and two to supply EECW requirements.

Connections are provided on the RHR Service Water System for use as a supply of cooling water to the standby coolant supply system. Standby coolant for Units 1 and 2 is provided by the "D" service water header and by the "B" header for Units 2 and 3.

Although individual service water supply headers are provided to each RHR heat exchanger, the discharges from two adjacent exchangers (A and C, or B and D) are joined as a common discharge. Prior to discharge into Wheeler Reservoir, this common discharge is split and subsequently recombined with the discharge from the other pair of heat exchangers. This configuration provides a more uniform temperature of discharge into Wheeler Reservoir. The layout of the discharge lines from the RHR heat exchangers to the reservoir is shown on FSAR Figure 10.9-1b.

Each common discharge is equipped with a radiation monitor. Drain and fill connections are provided on the service-water side of the RHR heat exchangers in order to drain or fill the heat exchangers.

The piping and valves from the heat exchanger inlet check valve to the discharge motor-operated globe valve are designed for 450 psig at 350°F, with the remainder of the supply piping and the Reactor Building discharge piping designed for 185 psig at 150°F. The yard discharge piping is designed for 80 psig at 150°F. The portion of piping from the discharge motor-operated globe valves to the point where the individual discharge lines form a common header is designed for 185 psig at 350°F.

Redundant RHRSW sump pumps are located within a 195 cu. ft. sump pit in rooms A, B, C, and D of the intake pumping station. Each sump pump is designed to remove 300 gpm of water at a total design head of 20 feet. The sump pumps are safety related and protect essential equipment in the pump rooms from damage due to flooding of the intake station pump rooms as a result of local maximum 1-hour rainfall.

10.9.4 Safety Evaluation

Power for the RHR service water pumps is provided by the four 4160-V shutdown boards (two pumps per board) in Units 1 and 2 and four 4160-V shutdown boards in Unit 3 (one pump per board). Each shutdown board is supplied by its individual standby diesel generator in the event of failure of the Normal Auxiliary Power System.

Two RHR service water pumps (A3 and C3) provide direct feed to the EECW north header and are powered from the shutdown boards in Unit 3. Another pair (B3 and D3) provide direct feed to the EECW south header and are powered from the shutdown boards in Units 1 and 2. One RHR service water pump (A2, B2, C2, and D2) on each RHR service water header is powered from the shutdown boards in Units 1 and 2. The other service water pumps on the B and D headers (B1 and D1) are powered from Unit 3 shutdown boards. The other service water pumps on the A and C headers (A1 and C1) are powered from Units 1 and 2 shutdown boards.

The RHR service water pumps are deck-mounted on the intake structure in an accessible location so that maintenance may be performed under emergency conditions. They will remain operable under flood conditions to about El. 578, which is approximately 6 feet above the maximum possible flood level of El. 572.5. The pumps are designed to operate in severe wind and weather such as during tornadoes. The pumps are widely dispersed and physically separated into groups of three on the intake deck to prevent common damage from one or more missiles.

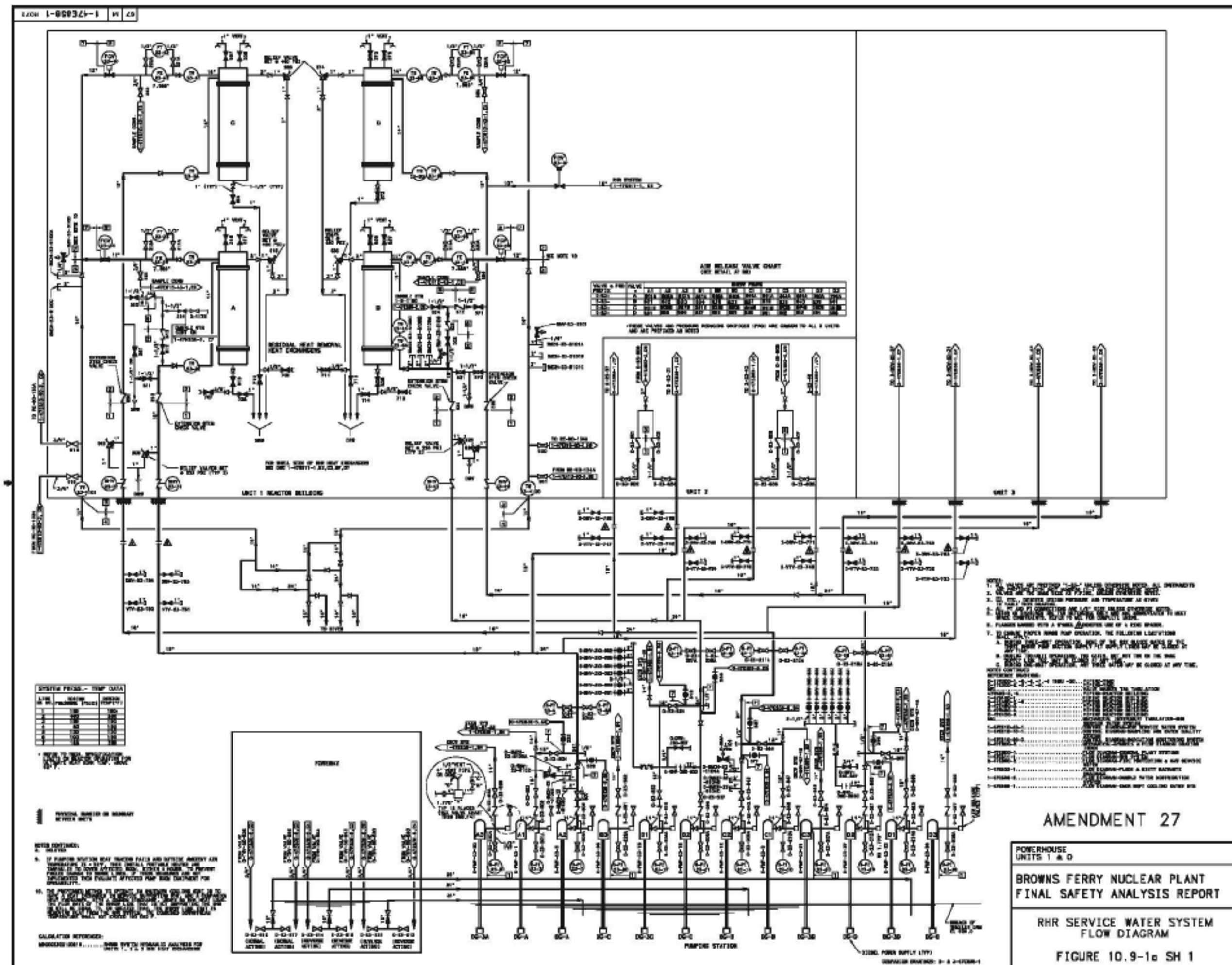
The RHR Service Water System is designed as a Class I system for withstanding the specified earthquake loadings (see Appendix C). The system piping and equipment are designed, installed, and tested in accordance with USAS B31.1.0, Section 1, 1967 edition with the exception of the following: RHRSW Pumps D2 and D3 are designed and constructed in accordance with ASME Section III, Class 3 (non-stamped), 1974 Edition through 1974 Winter Addenda.

For power uprated units, the analyzed RHRSW effluent temperature from the RHR heat exchanger is greater than analyzed for the non-power uprated condition. In order to maintain the same value for post LOCA peak suppression pool temperature as calculated prior to the power uprate, the reactor power level will be limited when the RHRSW input (Wheeler Lake) temperature is above the technical specifications specified value.

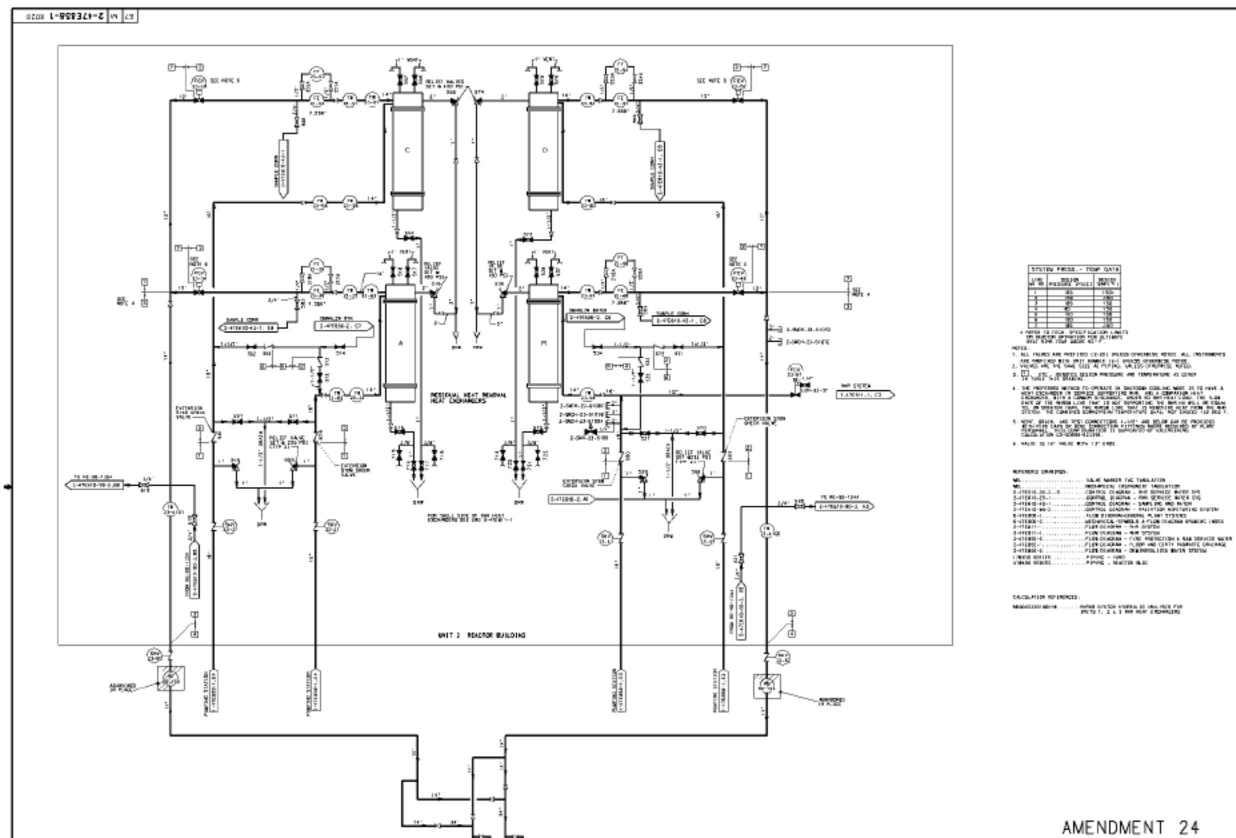
10.9.5 Inspection and Testing

The RHR service water pumps and piping are tested periodically to verify operability. The system is tested during initial startup to simulate breach of Wheeler Dam. The flow through each heat exchanger is measured for compliance with the specifications. After testing or use, the heat exchangers are left filled and the motor-operated valve downstream of the heat exchangers is closed.

The RHR service water system piping and components are monitored by routine inspections, general housekeeping practices, and system operability testing which maintains system leakage to an as-low-as-possible level.



UNIT 1 REACTOR BUILDING
RESIDUAL HEAT REMOVAL HEAT EXCHANGERS
UNIT 2
UNIT 3
PUMPING STATION
AMENDMENT 27
POWERHOUSE
BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT
RHR SERVICE WATER SYSTEM
FLOW DIAGRAM
FIGURE 10.9-1c SH 1



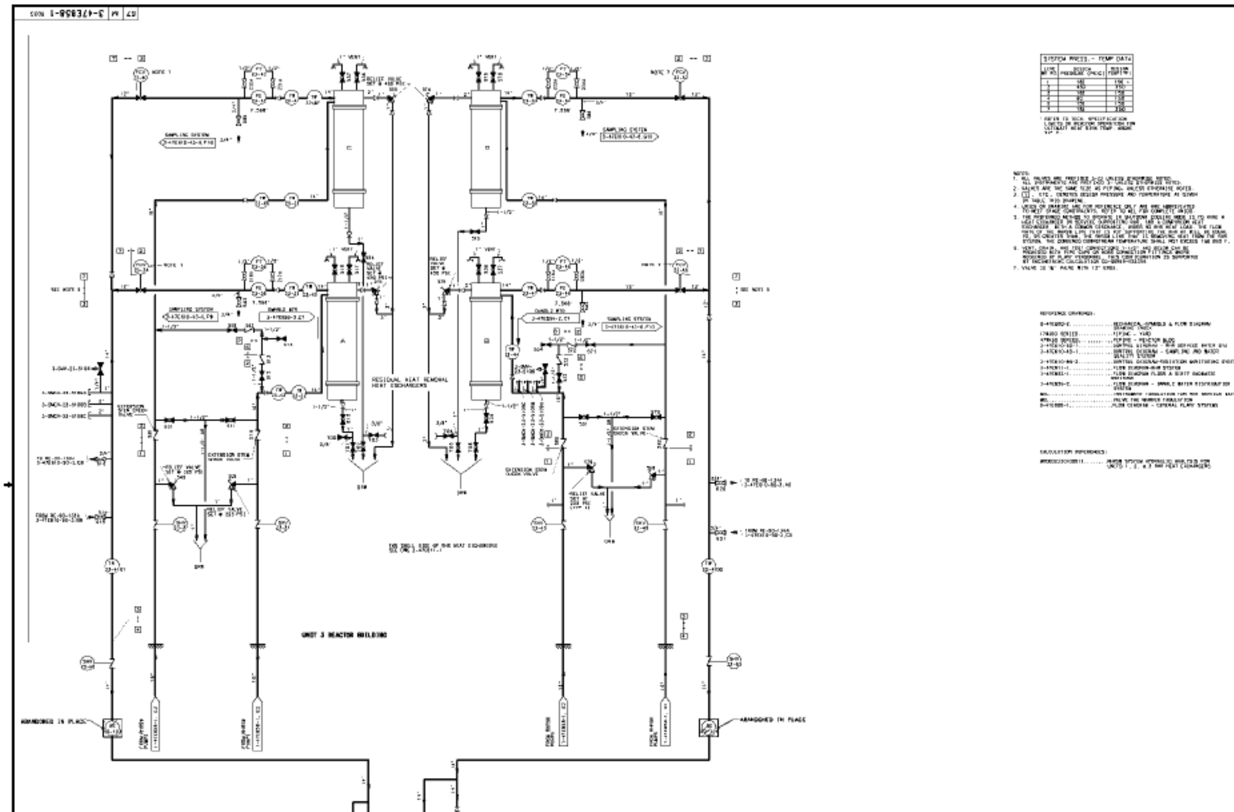
AMENDMENT 24

POWERHOUSE
UNIT 2

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

RHR SERVICE WATER SYSTEM
FLOW DIAGRAM

FIGURE 10.9-1a SH 2



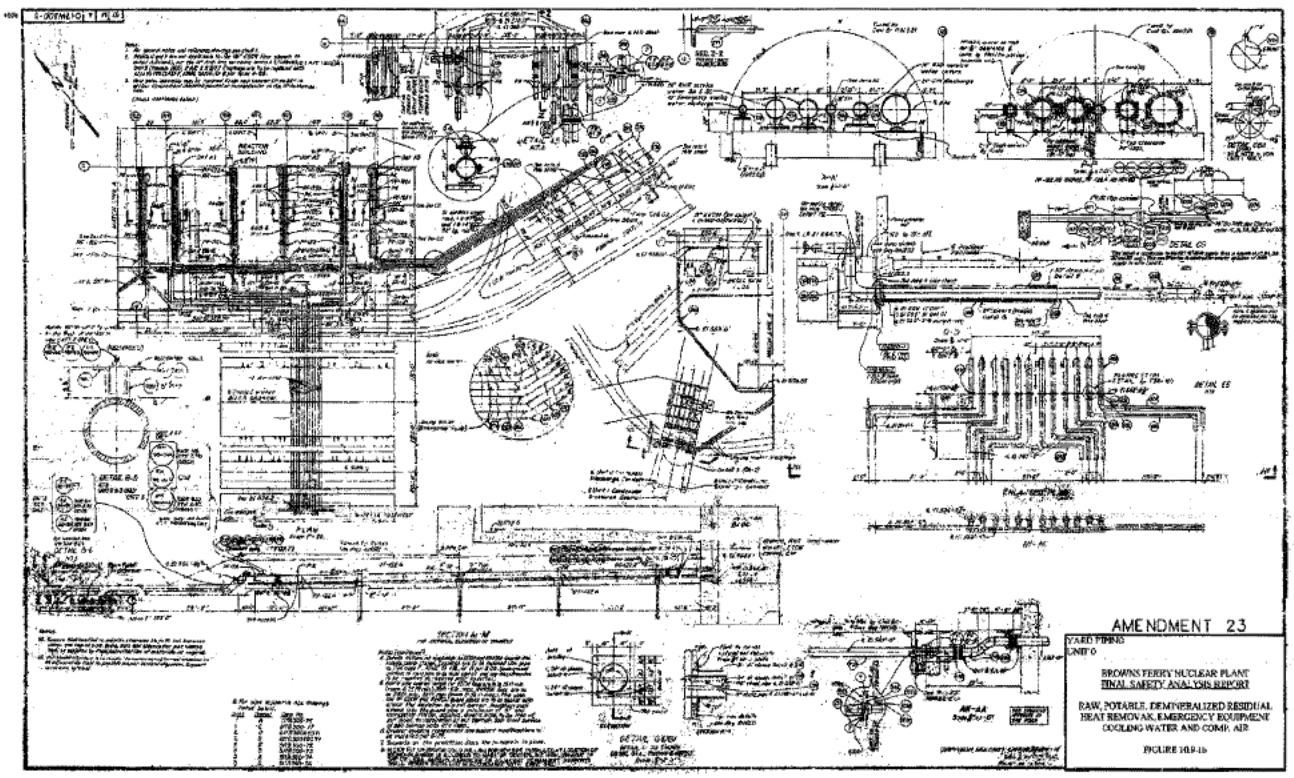
AMENDMENT 24

POWERHOUSE
UNIT 3

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

RHR SERVICE WATER SYSTEM
FLOW DIAGRAM

FIGURE 10.9-1a SH 3



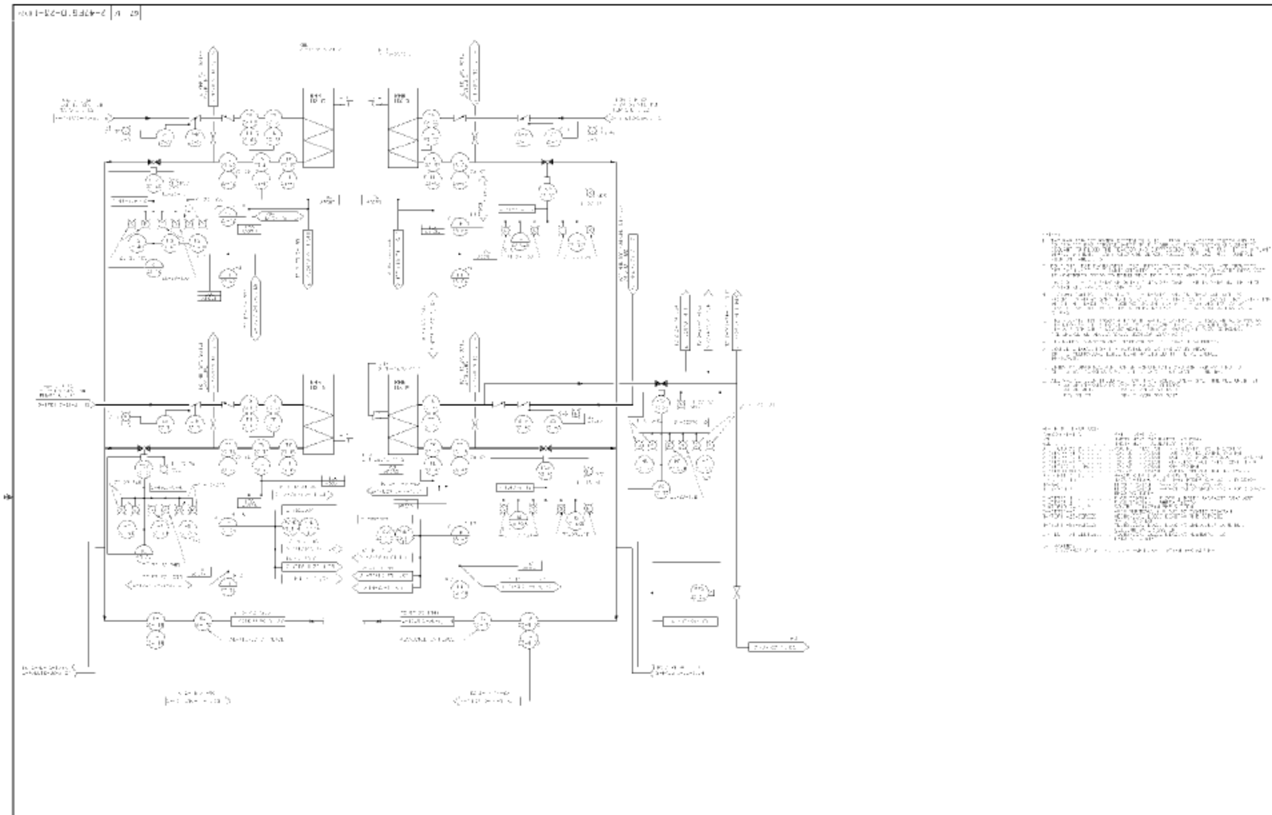
AMENDMENT 23

YARD PIPING
UNIT 0

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

RAW, POTABLE, DEMINERALIZED RESIDUAL HEAT REMOVAK, EMERGENCY EQUIPMENT COOLING WATER AND COMP. AIR

FIGURE 10.9-1b



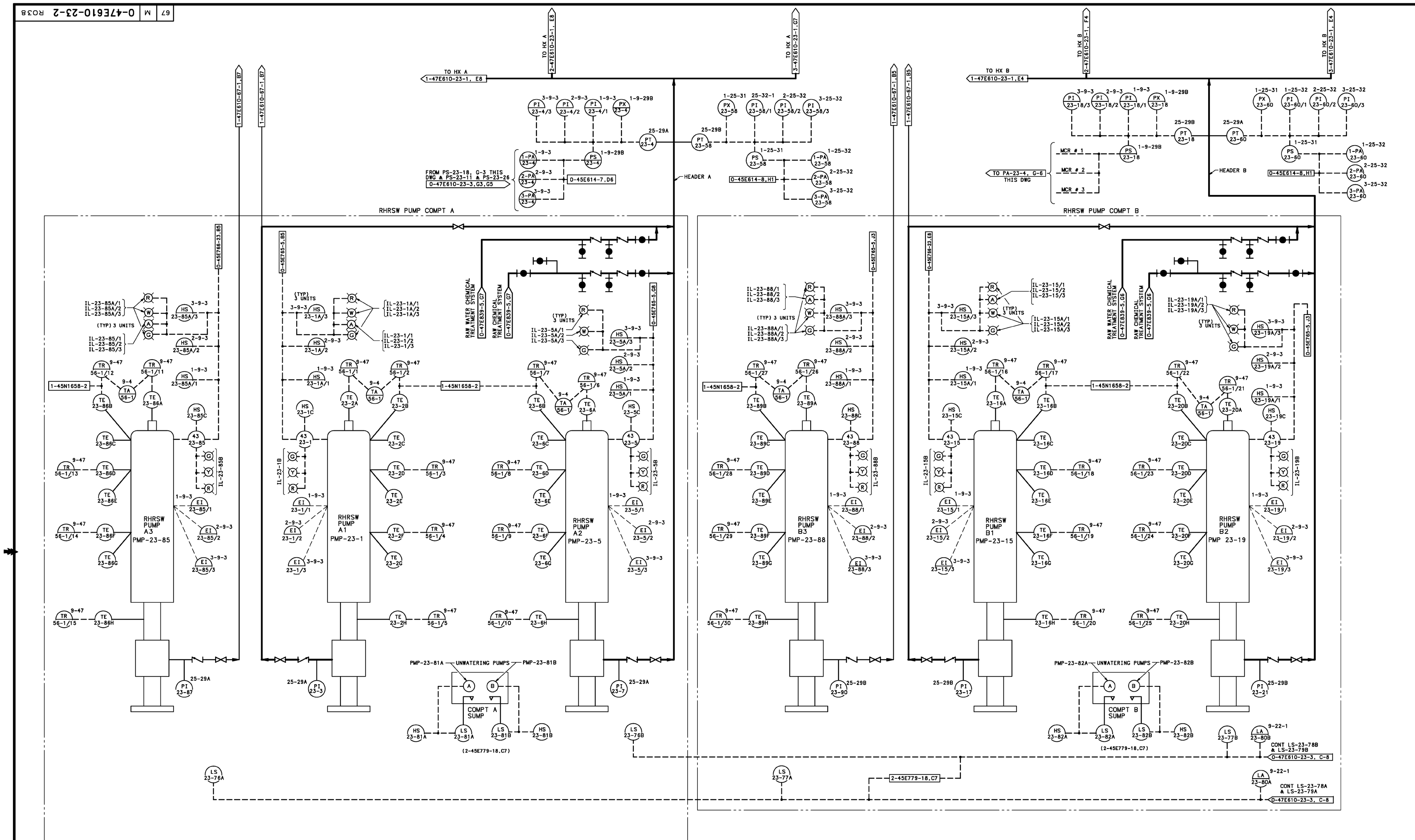
AMENDMENT 20

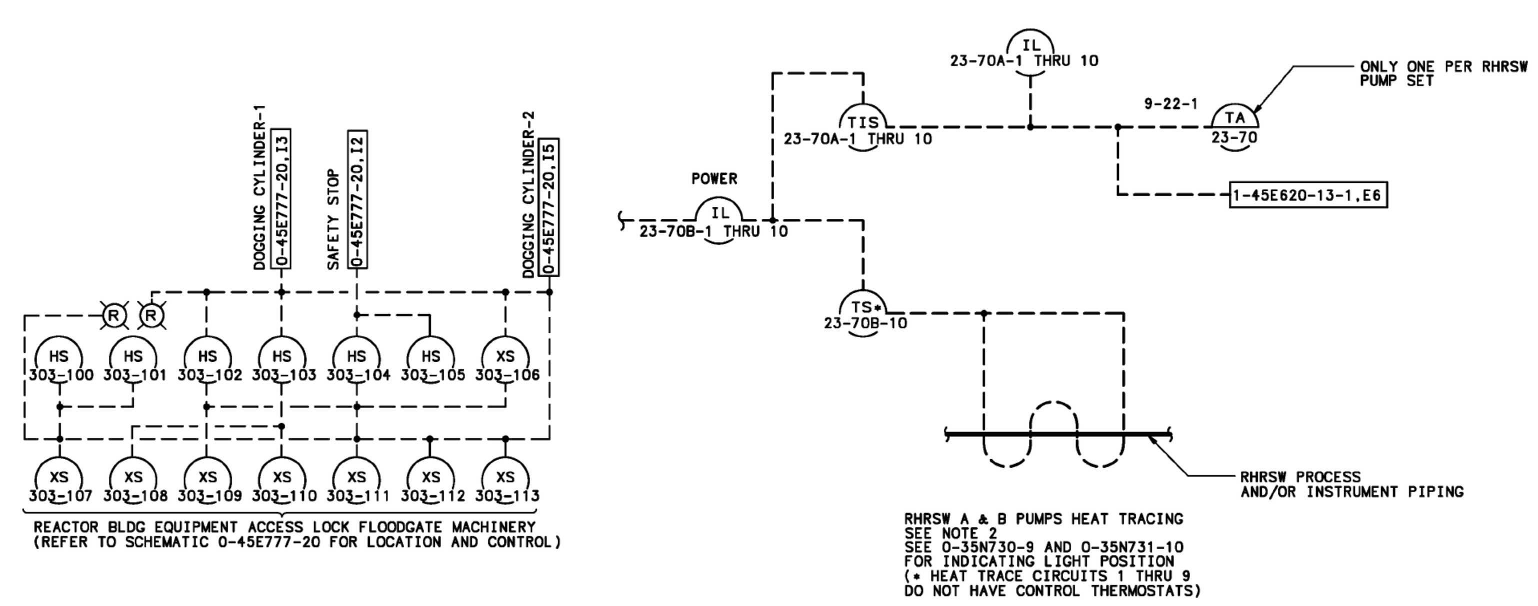
POWERHOUSE
UNIT 2

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

RHR SERVICE WATER SYSTEM
MECHANICAL CONTROL DIAGRAM

FIGURE 10.9-2a





NOTES:

1. FOR GENERAL NOTES AND OTHER REFERENCE DRAWINGS SEE COMPANION DRAWINGS 1-47E610-23-1, 2-47E610-23-1 & 3-47E610-23-1.
2. RHRSW A PUMPS HEAT TRACING AS SHOWN. B PUMPS SIMILAR WITH LOOP 71 INVOLVED.
3. ALL INSTRUMENTS ARE TO BE PREFIXED WITH "O" UNLESS OTHERWISE NOTED.
4. UNITS 2 AND 3 AUTO-INITIATE LOCKOUT WARNING LIGHTS ARE AMBER.
5. UNIDS ON DRAWING ARE FOR REFERENCE ONLY AND ARE ABBREVIATED TO MEET SPACE CONSTRAINTS. REFER TO MEL FOR COMPLETE UNIDS.
6. FOR ALL COMPONENTS DESIGNATED AS 'COMMON' OR 'TYPICAL', UNIDS SHOULD CONFORM WITH THE FOLLOWING FORMAT: O-XX-23-XX/UNIT NUMBER (EXAMPLE O-HS-23-15A/3 FOR HANDSWITCH 23-15A MOUNTED ON A UNIT 3 CONTROL PANEL).

REFERENCE DRAWINGS:

0-45E765-5..........4160V SHTDN AUX. PWR. SCHEMATIC DIAGRAM
0-45E766-23.........4160V SHTDN AUX. PWR. SCHEMATIC DIAGRAM
1-,2-,3-45E779-18...480V SHTDN AUX. PWR. SCHEMATIC DIAGRAM
0-35E720-2..........WIRING DIAGRAM 480V SUPPLY AUX. PWR. SCHEMATIC DIAGRAM
0-35E720-6..........WIRING DIAGRAM 480V SUPPLY AUX. PWR. SCHEMATIC DIAGRAM
0-45E614-7 & -8.....RSW RELAYS & INSTR. LOOPS PRESS. & FLOW INDICATION AND ALARM
1-45N1658-2.........CONNECTION DIAGRAM

COMPANION DRAWINGS:
0-47E610-23-3
1-47E610-23-1
2-47E610-23-1
& 3-47E610-23-1

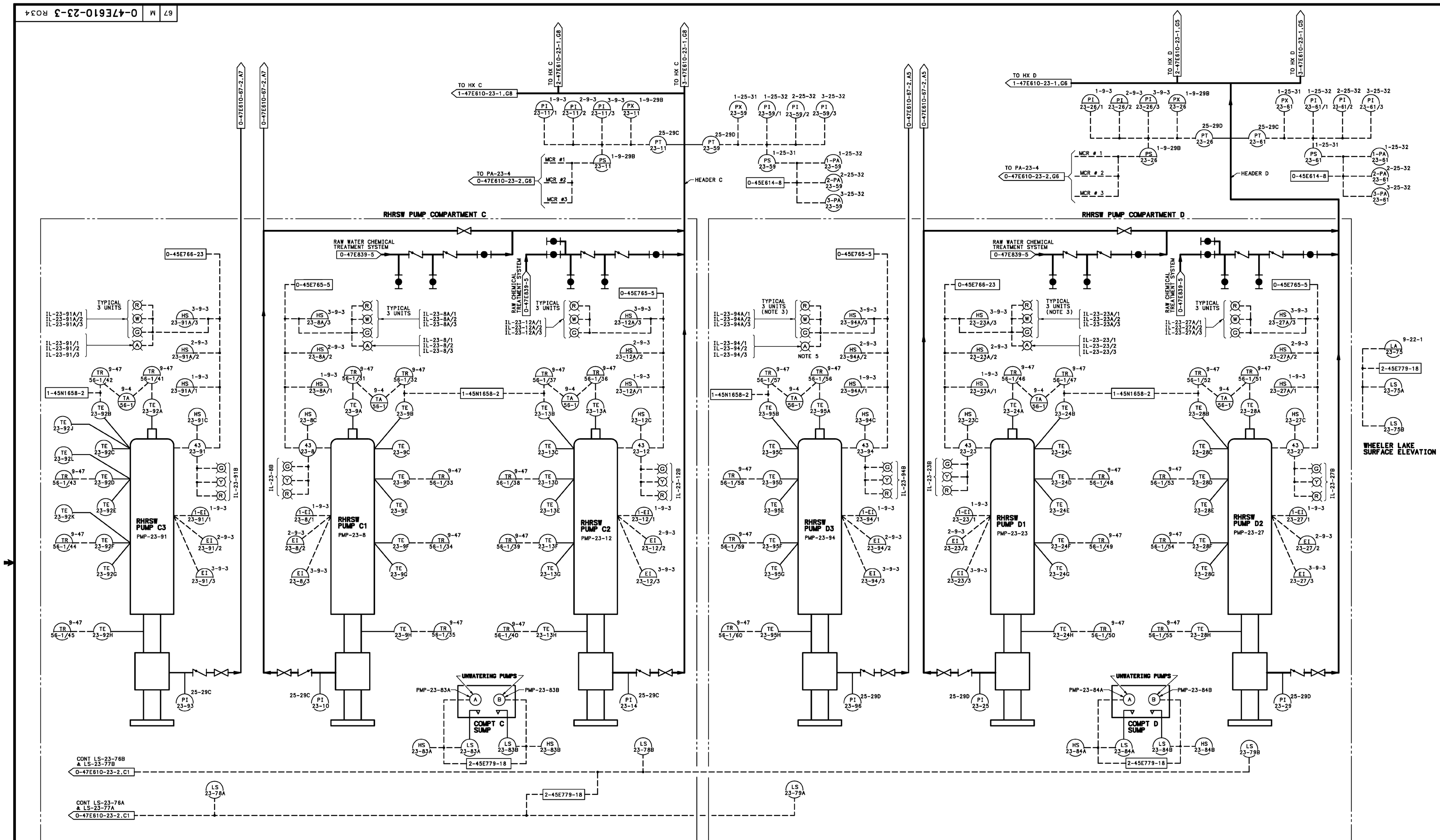
AMENDMENT 28

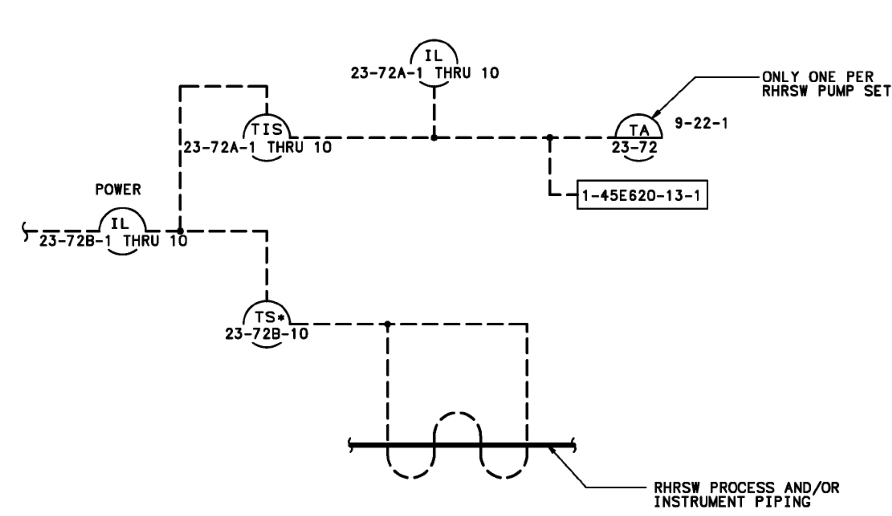
POWERHOUSE
UNIT 0

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

RHR SERVICE WATER SYSTEM
MECHANICAL CONTROL DIAGRAM

FIGURE 10.9-2b





RHRSW C & D PUMPS HEAT TRACING
PUMP C AS SHOWN, PUMP D SIMILAR WITH LOOP 73
INVOLVED SEE 0-35N731-11 AND 0-35N732-10
FOR INDICATING LIGHT POSITION
(*HEAT TRACE CIRCUITS 1 THRU 9
DO NOT HAVE CONTROL THERMOSTATS)

NOTES:

1. FOR GENERAL NOTES AND REFERENCE DRAWINGS SEE COMPANION DRAWINGS 1-, 2- & 3-47E610-23-1.
2. ALL INSTRUMENTS ARE TO BE PREFIXED WITH UNIT "0" UNLESS OTHERWISE NOTED.
3. UNIT 2 & 3 AUTO-INITIATE LOCKOUT WARNING LIGHTS ARE AMBER.
4. UNIDS ON DRAWING ARE FOR REFERENCE ONLY AND ARE ABBREVIATED TO MEET SPACE CONSTRAINTS. REFER TO MEL FOR COMPLETE UNIDS.
5. FOR ALL COMPONENTS DESIGNATED AS "COMMON" OR "TYPICAL", UNIDS SHOULD CONFORM WITH THE FOLLOWING FORMAT: 0-XX-23-XX/UNIT NUMBER (EXAMPLE: 0-HS-23-15A/3 FOR HANDSWITCH 23-15A MOUNTED ON A UNIT 3 CONTROL PANEL).
6. ANY OF THE SPARE THERMOCOUPLES ASSOCIATED WITH MOTOR WINDINGS MAY BE USED AS INPUT TO THE ASSOCIATED TEMPERATURE RECORDERS.

REFERENCE DRAWINGS:

0-35E720-2...........480V SUPPLY AUX POWER SCHEMATIC DIAGRAM
0-35E720-6...........480V WATER SUPPLY SCHEMATIC DIAGRAM
0-45E614-8...........120V AC/250V DC VALVES & MISC. SCHEMATIC DIAGRAM
0-45E614-7...........120V AC/250V DC VALVES & MISC. SCHEMATIC DIAGRAM
0-45E765-6...........4160V SHUTDOWN AUX POWER SCHEMATIC DIAGRAM
0-45E766-23..........4160V SHUTDOWN AUX POWER SCHEMATIC DIAGRAM
1-,2-,3-45E779-18....480V SHUTDOWN AUX POWER SCHEMATIC DIAGRAM
0-45E620-13..........WIRING DIAGRAM-ANNUNCIATOR SYSTEM

COMPANION DRAWINGS:
0-47E610-23-2,
1-, 2- & 3-47E610-23-1

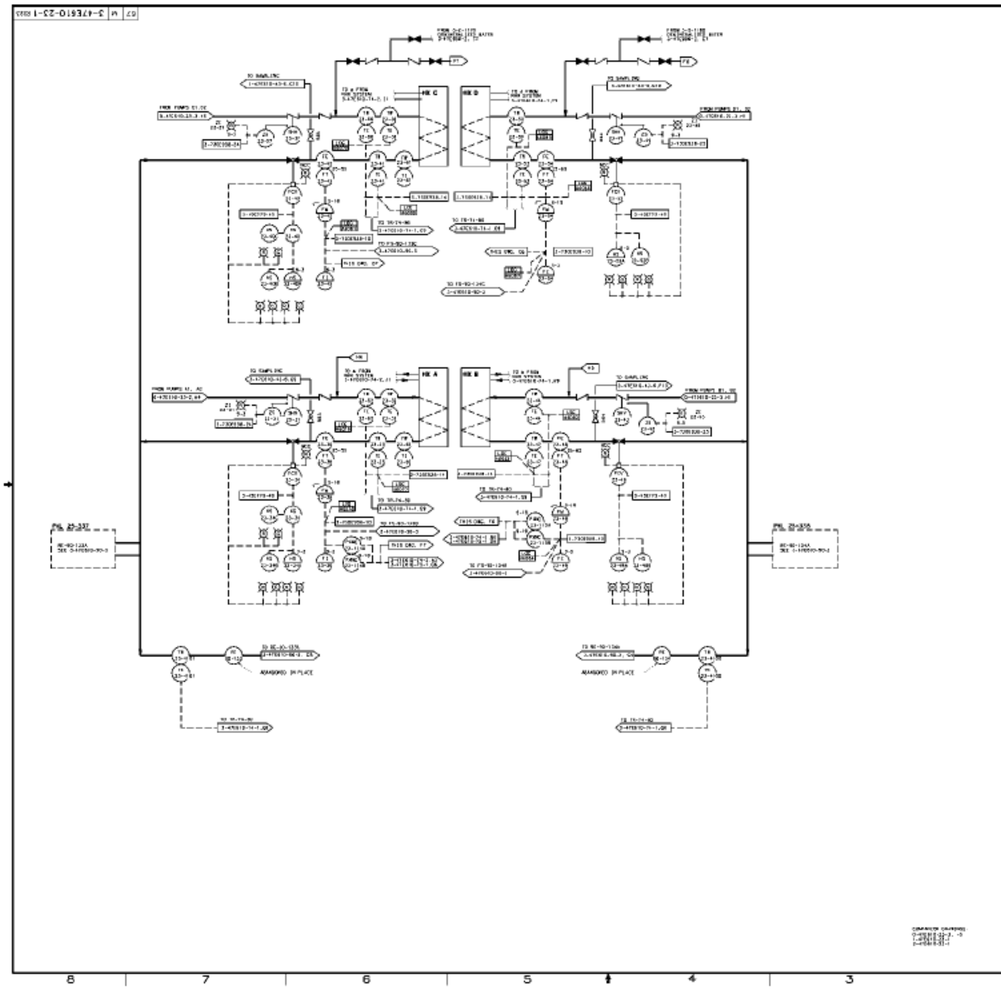
AMENDMENT 28

POWERHOUSE
UNIT 0

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

RHR SERVICE WATER SYSTEM
MECHANICAL CONTROL DIAGRAM

FIGURE 10.9-2c



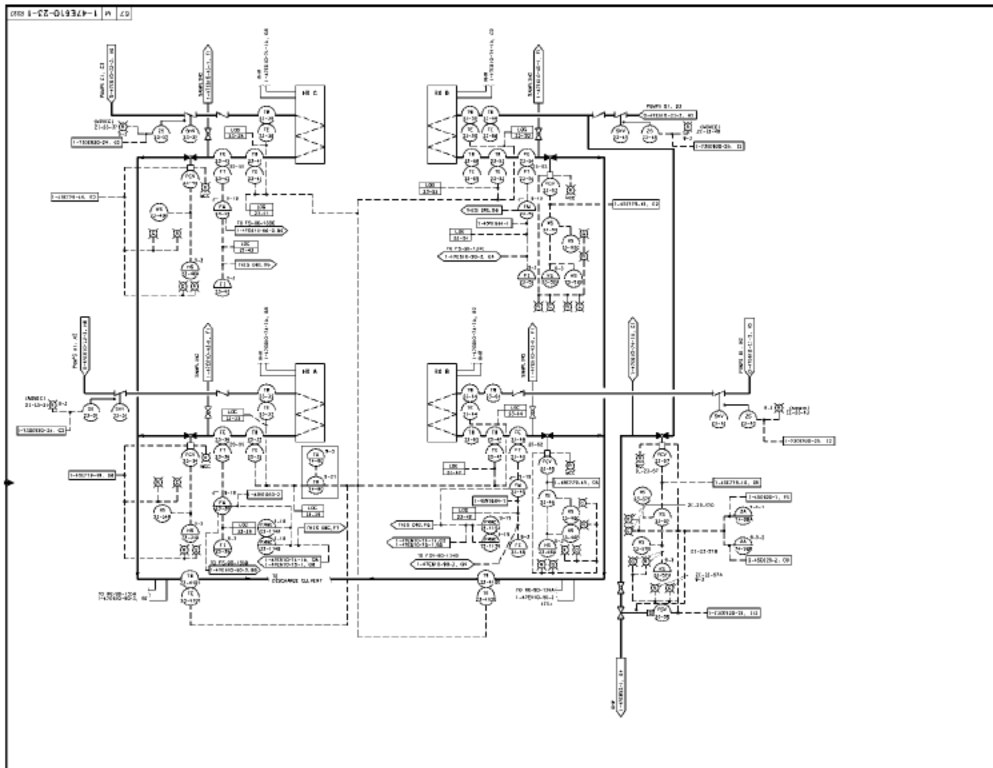
AMENDMENT 24

POWERHOUSE
UNIT 3

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

RHR SERVICE WATER SYSTEM
MECHANICAL CONTROL DIAGRAM

FIGURE 10.9-2d



AMENDMENT 24

POWERHOUSE
UNIT 1

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

RHR SERVICE WATER SYSTEM
MECHANICAL CONTROL DIAGRAM

FIGURE 10.9-2a

Figures 10.9-3 and 10.9-4
(Deleted by Amendment 22)

10.10 EMERGENCY EQUIPMENT COOLING WATER SYSTEM

10.10.1 Safety Objective

The safety objective of the Emergency Equipment Cooling Water (EECW) System is to provide cooling water to the Standby Diesel engine coolers, the Residual Heat Removal (RHR) pump seal coolers and pump room coolers, the Core Spray pump room coolers, the Unit 3 CB chillers, Unit 3 Shutdown Board Room Chillers, Unit 3 Electrical Board Room Air Conditioning Units, and additional makeup for the fuel pool.

In addition the system provides, as a backup to the Raw Cooling Water (RCW) System, cooling water for the Reactor Building Closed Cooling Water (RBCCW) heat exchangers and the Control Air compressors. The system can be aligned to provide cooling water to the Unit 1/2 Emergency Condensing Unit (ECU), if required.

10.10.2 Safety Design Basis

1. The system shall be capable of operating automatically to maintain sufficient cooling water to all essential components (users) listed below:

   Unit 1/2 Standby Diesel engine coolers
   Unit 3 Standby Diesel engine coolers
   Units 1,2,3 Core Spray pump room coolers
   Units 1,2,3 RHR pump room coolers
   Units 1,2,3 RHR pump seal coolers*
   Unit 3 Electric Board Room ACU Condensers
   Unit 3 Electric Board Room chillers
   Unit 3 Control Bay chillers

   * When EECW is isolated from the shell side of a RHR pump seal cooler, the operability of the affected RHR pump is not compromised provided that the seal water flow is maintained.

2. Piping and equipment, including support structures, are designed to withstand effects of the design basis earthquake without failure.

10.10.3 Description

The Emergency Equipment Cooling Water System has automatic actuation to distribute cooling water supplied by the RHR Service Water System pumps which have been assigned as the principal supply to the EECW System (RHRSW pumps A3, B3, C3 and D3). Flow diagrams for this shared system are shown in Figures 10.10-1a, -1b, -1c, and -1d. System instrumentation is shown in Figures 10.10-2 and 10.10-3 sheets 1, 2, 3, and 4.

The required design EECW flow for the three unit plant is satisfied by RHRSW pumps rated at 400-hp with a capacity of 4500 GPM at a 275-foot head. This includes cooling non-essential components (i.e., the RBCCW heat exchangers and Control Air compressors). Three of the four RHRSW pumps assigned to EECW are necessary to supply the EECW System maximum design flow requirements. Two pumps feed each EECW supply header. There are two completely redundant and independent headers (north and south headers) in a loop arrangement inside and outside the Reactor Building. These headers also provide additional makeup for the fuel pool via normally closed fire hose connections. Also, the north header and south header have a connection to the RCW and Raw Service Water (RSW) Systems, respectively, for charging purposes. To insure absence of air voids in the EECW piping, each header is maintained flooded by the continuous operation of an assigned RHRSW pump. The piping system is shown on Figures I0.I0-1a, -1b, -1c, and -1d. These figures provide the design pressure and temperature conditions of the various portions of the system. The two RHRSW pumps (A3 and C3) assigned to EECW and powered from shutdown boards in Unit 3 will start automatically within specified limits after starting of a diesel generator or a core spray pump in Unit 3.

Similarly, the two RHRSW pumps (B3 and D3) assigned to EECW and powered from shutdown boards in Units 1 and 2 will start automatically within specified limits after starting of a diesel generator or a core spray pump in Units 1 and 2. When a high drywell pressure plus low reactor vessel pressure or low-low-low reactor water level signal is received in any unit, all four RHRSW pumps will start. In addition, the signals that start the A3 and C3 pumps and the B3 and D3 pumps also start the B1 and D1 pumps and the A1 and C1 pumps, respectively, when they are valved into the EECW header and the associated mode switch is selected for EECW operation.

Each EECW supply header from the intake station has a continuous self-cleaning strainer to prevent clogging of the various coolers. The straining media is selected with a nominal 1/8 inch diameter hole mesh design that may include manufac turing anomalies (i.e., elongated holes at bends and conjoined holes at seams). In all cases, the holes will prevent intrusion of particles which are larger in diameter than the smallest tube or orifice serving downstream users.

Each EECW supply header is provided with manual motor-operated sectionalizing valves to permit isolation of leaking equipment in a unit area, while retaining flow to the essential components. Branch runouts and equipment configuration are arranged for complete redundancy, with the exceptions of specific branch lines supplying specific non-essential components normally cooled by the Raw Cooling Water (RCW) system.

These exceptions are (i) a single available branch line to each unit’s set of RBCCW heat exchangers, and (II) a single installed branch line for the control air compressors. The air compressors, located in a Class II structure, are not essential

to shutdown. Each compressor may be isolated in case of an accident. Fire Protection restrictions identified in the NFPA 805 DCD result in the availability of only one supply branch line to each RBCCW heat exchanger. FSAR Section 5.2 and FSAR Appendix F describe design features addressing alternative means of heat removal for specific RBCCW supplied components during failure of a unit's RBCCW system.

Branch runouts, except the ones to the RBCCW heat exchangers and the normally isolated Unit 1/2 CB ECU, are fitted with double-check valves to prevent back or cross flow into alternate headers. Protection against excess flow is provided at the larger branches in the event of a piping or equipment failure.

During normal operation, the RCW System provides cooling water to the RBCCW heat exchangers and the Control Air compressors. If the pressure in the RCW system falls below a predetermined minimum level, the aligned RHRSW pumps receive auto-start signals. In this manner the Class I EECW System provides an automatic backup supply for the closed cooling water heat exchangers and the air compressor coolers. The runouts to the Reactor Building Closed Cooling Water heat exchangers are provided with pneumatically operated flow control valves controlled by EECW and RCW header pressure. The runout to the control air compressors is provided with a pneumatically operated flow control valve controlled by EECW and RCW header pressure.

These valves are designed to shut off flow to the equipment on low header pressure in order to guarantee adequate flow to the essential components.

The design provides multiple paths for the return of water from the coolers to a point outside the Reactor Building.

The EECW System has the capability of utilizing the FLEX (Flexible and Diverse Coping Mitigation Strategies) system to provide make-up water to the SFP and cooling water to essential equipment such as the Unit 3 Chiller and Core Spray Room Coolers. This additional capability is provided in the event of a loss of all the RHRSW pumps due to a Loss of Off-Site Power, and site Emergency Diesel Generators are inoperable.

10.10.4 Safety Evaluation

To assure power to the RHR service water pumps serving the Emergency Equipment Cooling Water System, two pumps (A3 and C3) are connected to 4160-V shutdown boards in Unit 3 and two pumps (B3 and D3) to boards in Units 1 and 2. Each shutdown board is supplied by its individual standby diesel generator in the event of failure of the normal auxiliary power system.

The Emergency Equipment Cooling Water System is designed as a Class I system for withstanding the specified earthquake loadings (see Appendix C). The system piping and equipment are designed, installed, and tested in accordance with USAS B31.1.0, Section I, 1967 edition. The Emergency Equipment Cooling Water System is chemically treated consistent with NPDES permit limitations. In accordance with NRC Bulletin 81-03, the Emergency Equipment Cooling Water System is also chemically treated during peak clam spawning periods to ensure clam control.

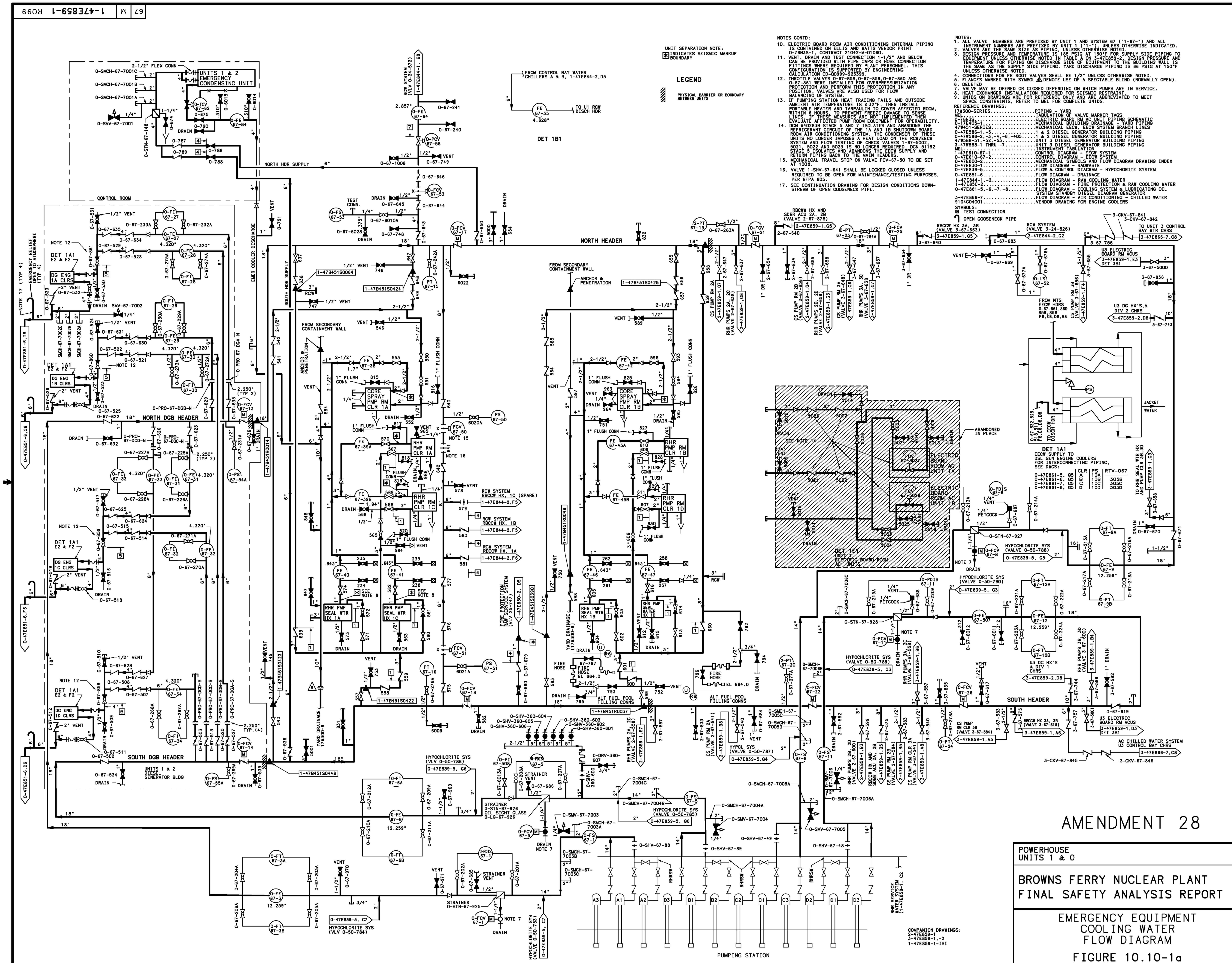
10.10.5 <u>Inspection and Testing</u>

The Emergency Equipment Cooling Water System will be tested periodically to verify operability.

The system was tested during initial startup by simulating the breach of Wheeler Dam. The flow through each cooler is regulated by a manual throttling or self-regulating valve to maintain required minimum flow under all operating conditions. Each manual throttling valve will remain under administrative control as initially adjusted unless future testing indicates inadequate flow is being received by the equipment; at which time, necessary throttle valves will be readjusted.

The Emergency Equipment Cooling Water System piping and components are monitored by routine inspections, general housekeeping practices, and system operability testing which maintains system leakage to an as-low-as-possible level.

The Unit 2 Emergency Equipment Cooling Water System has stainless steel butt welds which have exhibited the presence of Microbiologically Induced Corrosion (MIC). As a result of MIC, the Unit 2 EECW System will be monitored for MIC progress each operating cycle to ensure structural integrity of the system.



NOTES:

1. ALL VALVE NUMBERS ARE PREFIXED BY UNIT 1 AND SYSTEM 67 ("1-67-") AND ALL INSTRUMENT NUMBERS ARE PREFIXED BY UNIT 1 ("1-"), UNLESS OTHERWISE INDICATED.
2. VALVES ARE THE SAME SIZE AS PIPING, UNLESS OTHERWISE NOTED.
3. DESIGN PRESSURE AND TEMPERATURE IS 185 PSIG AT 150°F FOR SUPPLY SIDE PIPING TO EQUIPMENT UNLESS OTHERWISE NOTED IN TABLE A ON 3-47E859-2. DESIGN PRESSURE AND TEMPERATURE FOR PIPING ON DISCHARGE SIDE OF EQUIPMENT TO THE BUILDING WALL IS THE SAME AS THE SUPPLY SIDE PIPING. YARD DISCHARGE PIPING IS 66 PSIG AT 150°F UNLESS OTHERWISE NOTED.
4. CONNECTIONS FOR FE ROOT VALVES SHALL BE 1/2" UNLESS OTHERWISE NOTED.
5. FLANGES MARKED WITH SYMBOL 5 DENOTE USE OF A SPECTABLE BLIND (NORMALLY OPEN).
6. DELETED
7. VALVE MAY BE OPENED OR CLOSED DEPENDING ON WHICH PUMPS ARE IN SERVICE.
8. HEAT EXCHANGER INSTALLATION REQUIRED FOR SEISMIC RESTRAINT.
9. UNIDS ON DRAWINGS ARE FOR REFERENCE ONLY AND ARE ABBREVIATED TO MEET SPACE CONSTRAINTS, REFER TO MEL FOR COMPLETE UNIDS.

REFERENCE DRAWINGS:

17W300-SERIES................PIPING - YARD
MEL..........................TABULATION OF VALVE MARKER TAGS
0-76N3S......................ELECTRIC BOARD RM AC UNIT PIPING SCHEMATIC
0-17E405-1...................MECHANICAL BUILDING DRAINAGE - YARD PIPING
47W451-SERIES................MECHANICAL EECW, EECW SYSTEM BRANCH LINES
0-47E586-1,-5................1 & 2 DIESEL GENERATOR BUILDING PIPING
0-47W586-2,-3,-4,-6,-405.....1 & 2 DIESEL GENERATOR BUILDING PIPING
47W588-51,-52,-53............UNIT 3 DIESEL GENERATOR BUILDING PIPING
3-47W588-1 THRU -7...........UNIT 3 DIESEL GENERATOR BUILDING PIPING
MEL..........................INSTRUMENT TABULATION
1-47E610-67-1................CONTROL DIAGRAM - EECW SYSTEM
0-47E610-67-2................CONTROL DIAGRAM - EECW SYSTEM
0-47E800-2...................MECHANICAL SYMBOLS AND FLOW DIAGRAM DRAWING INDEX
0-47E830-1...................FLOW DIAGRAM - RADWASTE
0-47E839-5...................FLOW & CONTROL DIAGRAM - HYPOCHORITE SYSTEM
0-47E851-6...................FLOW DIAGRAM - DRAINAGE
1-47E844-1,-2................FLOW DIAGRAM - RAW COOLING WATER
1-47E850-2...................FLOW DIAGRAM - FIRE PROTECTION & RAW COOLING WATER
0-47E861-5,-6,-7,-8..........FLOW DIAGRAM - COOLING SYSTEM & LUBRICATING OIL SYSTEM STANDBY DIESEL DIAGRAM GENERATOR
3-47E866-7...................FLOW DIAGRAM - AIR CONDITIONING - CHILLED WATER
9104C04001...................VENDOR DRAWING FOR ENGINE COOLERS

SYMBOLS:

⊠ TEST CONNECTION

OPEN GOOSENECK PIPE

NOTES CONTD:

10. ELECTRIC BOARD ROOM AIR CONDITIONING INTERNAL PIPING IS CONTAINED ON ELLIS AND WATTS VENDOR PRINT 0-76N3S-1, CONTRACT 21042-M-0106Q.
11. VENT, DRAIN AND TEST CONNECTION 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. THROTTLE VALVES 0-67-858,0-67-859,0-67-860 AND 0-67-861 WERE INSTALLED FOR OVERPRESSURIZATION PROTECTION AND PERFORM THIS PROTECTION IN ANY POSITION. VALVES ARE ALSO USED FOR FLOW BALANCING OF SYSTEM.
13. IF PUMPING STATION HEAT TRACING FAILS AND OUTSIDE AMBIENT AIR TEMPERATURE IS ≤ 32°F, THEN INSTALL PORTABLE HEATER AND TARPAULIN TO COVER AFFECTED ROOM, WITHIN 6 HOURS, TO PREVENT FREEZE DAMAGE TO SENSE LINES. IF THESE MEASURES ARE NOT IMPLEMENTED THEN EVALUATE AFFECTED PUMP ROOM EQUIPMENT FOR OPERABILITY.
14. DCN W40283B STAGE 5 AND 7 ISOLATES AND ABANDONS THE REFRIGERANT CIRCUIT OF THE 1A AND 1B SHUTDOWN BOARD ROOM AIR CONDITIONING SYSTEM. THE CONDENSER OF THESE UNITS NO LONGER IMPOSES A HEAT LOAD ON THE RCW/EECW SYSTEM AND FLOW TESTING OF CHECK VALVES 1-67-5002, 5021, 5022 AND 5023 IS NO LONGER REQUIRED. DCN 51192 STAGE 5 ISOLATES AND ABANDONS THE EECW SUPPLY AND RETURN PIPING BACK TO THE MAIN HEADERS.
15. MECHANICAL TRAVEL STOP ON VALVE FCV-67-50 TO BE SET AT 100%.
16. VALVE 1-SHV-67-641 SHALL BE LOCKED CLOSED UNLESS REQUIRED TO BE OPEN FOR MAINTENANCE/TESTING PURPOSES, PER NFPA 805.
17. SEE CONTINUATION DRAWING FOR DESIGN CONDITIONS DOWNSTREAM OF OPEN GOOSENECK PIPE.

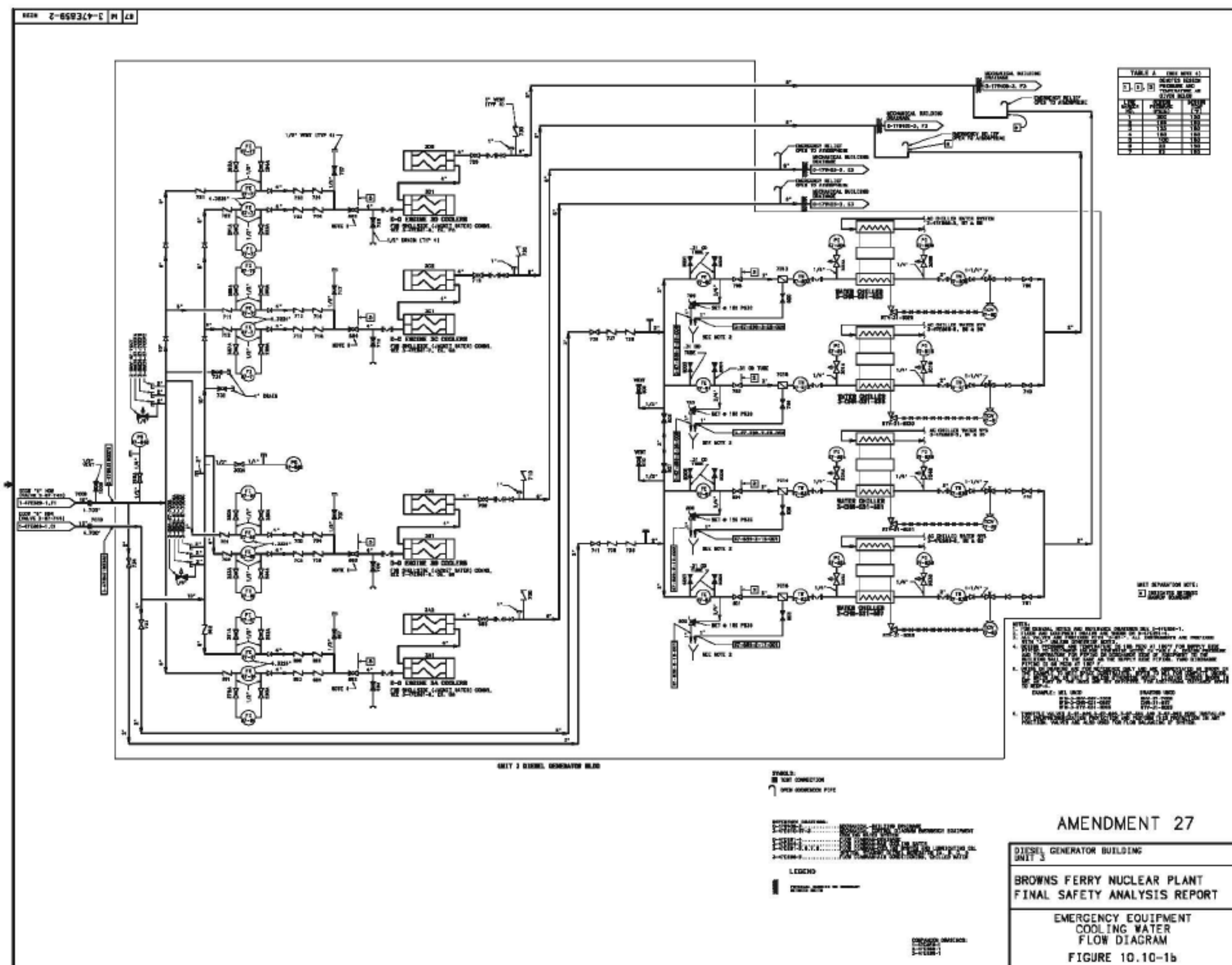
AMENDMENT 28

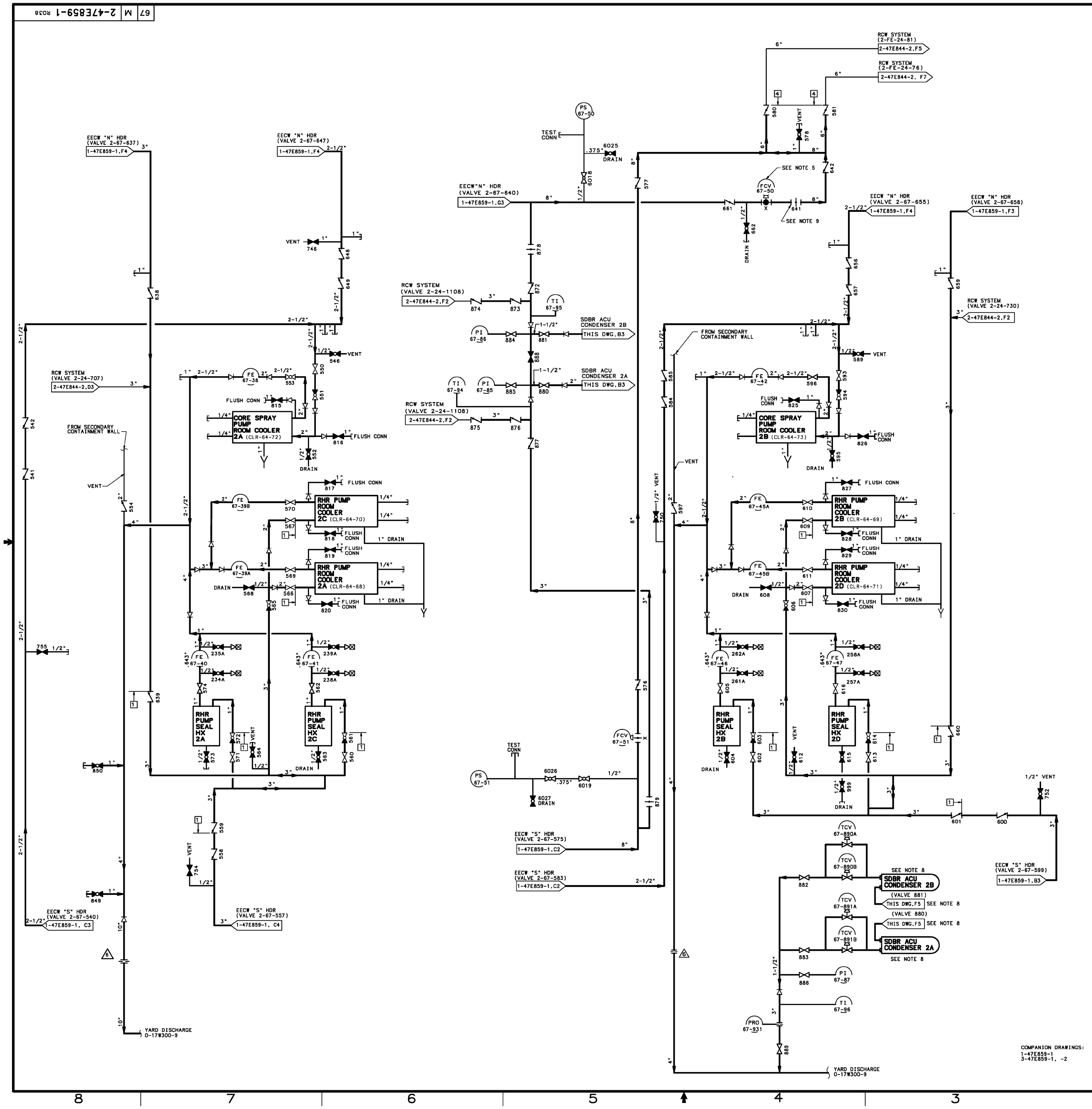
POWERHOUSE
UNITS 1 & 0

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

EMERGENCY EQUIPMENT
COOLING WATER
FLOW DIAGRAM

FIGURE 10.10-1a


UNIT 3 DIESEL GENERATOR BLDG
LEGEND
AMENDMENT 27
DIESEL GENERATOR BUILDING
UNIT 3
BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT
EMERGENCY EQUIPMENT
COOLING WATER
FLOW DIAGRAM
FIGURE 10.10-1b



NOTES:

1. ALL VALVE NUMBERS, EXCEPT AS NOTED, TO HAVE UNIT 2 AND SYSTEM 67 ("2-67-") PREFIX IE; 2-67-500. ALL INSTRUMENTS ARE PREFIXED "2-" EXCEPT AS NOTED.
2. VALVES ARE THE SAME SIZE AS PIPING UNLESS OTHERWISE NOTED.
3. DESIGN PRESSURE AND TEMPERATURE IS 185 PSIG AT 150°F FOR SUPPLY SIDE PIPING TO EQUIPMENT UNLESS OTHERWISE NOTED IN TABLE A ON 3-47E859-2. DESIGN PRESSURE AND TEMPERATURE FOR PIPING ON DISCHARGE SIDE OF EQUIPMENT TO THE BUILDING WALL IS THE SAME AS THE SUPPLY SIDE PIPING. YARD DISCHARGE PIPING IS 66 PSIG AT 150°F.
4. FLANGES MARKED WITH SYMBOL 6 DENOTES USE OF A SPECTACLE BLIND (NORMALLY OPEN).
5. MECHANICAL TRAVEL STOP ON VALVE 2-FCV-67-50 TO BE SET AT ≥ 31%, ±1% OF MAXIMUM OPENING WITH EECW FLOW TO RBCCW HEAT EXCHANGER ≥ 900 GPM WITH THE DOWNSTREAM TCV FULLY OPEN.
6. DELETED
7. VENT, DRAIN AND TEST CONNECTIONS 1-1/2" AND BELOW CAN BE PROVIDED WITH PIPE CAPS OR HOSE CONNECTION FITTINGS WHERE REQUIRED BY PLANT PERSONNEL. THIS CONFIGURATION IS SUPPORTED BY ENGINEERING CALCULATION CD-Q0999-923399.
8. DCN W40283B STAGE 6 AND 8 ISOLATES AND ABANDONS THE REFRIGERANT CIRCUIT ON THE 2A AND 2B SHUTDOWN BOARD ROOM AIR CONDITIONING SYSTEM. THE CONDENSER OF THESE UNITS NO LONGER IMPOSES A HEAT LOAD ON THE RCW/EECW SYSTEM AND FLOW TESTING OF CHECK VALVES 2-67-872, 873, 874, 875, 876 AND 877 IS NO LONGER REQUIRED. ADDITIONAL CHANGES TO THE EECW SYSTEM WILL BE ADDRESSED IN A FUTURE PIC AGAINST DCN W40283B.
9. VALVE 2-SHV-67-641 SHALL BE LOCKED CLOSED UNLESS REQUIRED TO BE OPEN FOR MAINTENANCE/TESTING PURPOSES, PER NFPA 805.

REFERENCE DRAWINGS:

MEL....................TABULATION OF VALVE MARKER TAGS
MEL....................INSTRUMENT TABULATION-SYSTEM 67
2-47E610-67-2..........CONTROL DIAGRAM - EMERGENCY EQUIPMENT COOLING WATER SYSTEM
0-47E800-2.............MECHANICAL SYMBOLS AND FLOW DIAGRAM DRAWING INDEX
2-47E844-2.............FLOW DIAGRAM - RAW COOLING WATER
0-17W300-9.............MECHANICAL ISOMETRIC EMERGENCY EQUIPMENT COOLING WATER SYSTEM
47W451-SERIES..........MECHANICAL EECW, EECW SYSTEM BRANCH LINES HANGERS AND RESTRAINTS
47W586-SERIES..........PIPING - UNITS 1 & 2 DIESEL GENERATOR BUILDING
2-47E2610-76-3.........MECHANICAL CONTROL DIAGRAM CONTAINMENT INERTING SYSTEM

COMPANION DRAWINGS:
1-47E859-1
3-47E859-1, -2

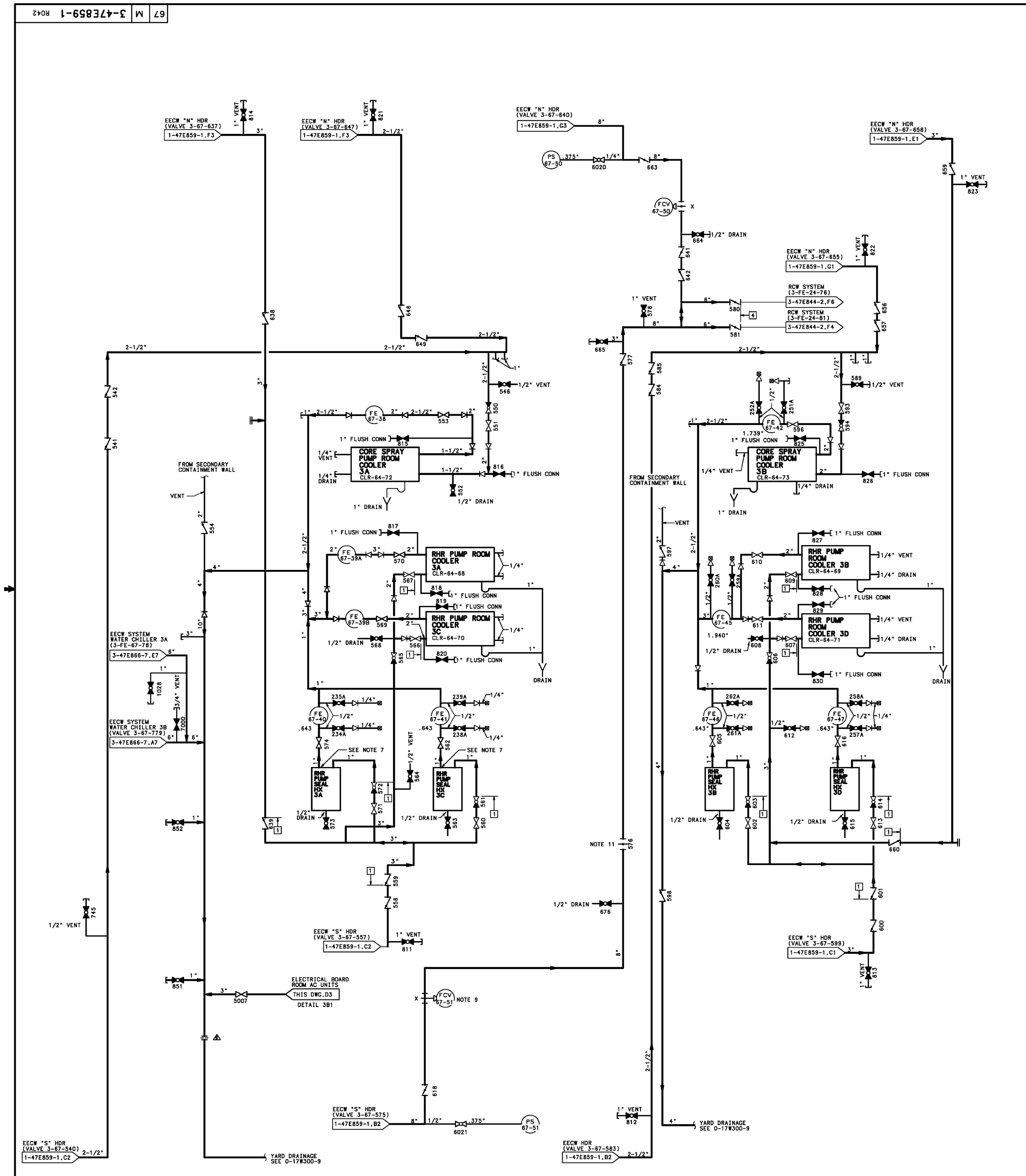
AMENDMENT 28

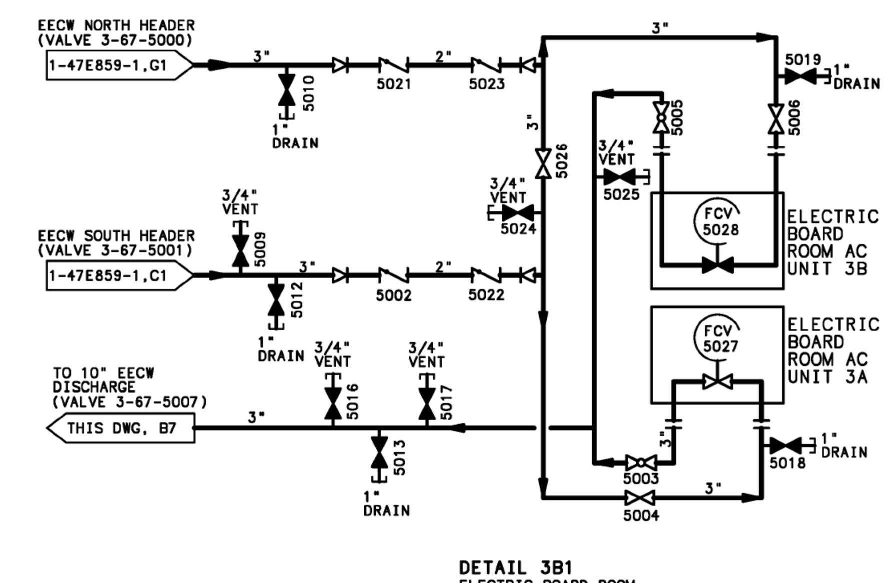
POWERHOUSE
UNIT 2

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

EMERGENCY EQUIPMENT
COOLING WATER
FLOW DIAGRAM

FIGURE 10.10-1c





DETAIL 3B1
ELECTRIC BOARD ROOM AC UNITS

NOTES:

1. ALL INSTRUMENTS HAVE A UNIT 3 PREFIX EXCEPT AS NOTED. ALL VALVE NUMBERS, EXCEPT AS NOTED, TO HAVE UNIT 3, SYSTEM 67 PREFIX.
2. VALVES ARE THE SAME SIZE AS PIPING UNLESS OTHERWISE NOTED.
3. DELETED
4. DESIGN PRESSURE AND TEMPERATURE IS 185 PSIG AT 150°F FOR SUPPLY SIDE PIPING TO EQUIPMENT UNLESS OTHERWISE NOTED IN TABLE A ON 3-47E859-2. DESIGN PRESSURE AND TEMPERATURE FOR PIPING ON DISCHARGE SIDE OF EQUIPMENT TO THE BUILDING WALL IS THE SAME AS THE SUPPLY SIDE PIPING. YARD DISCHARGE PIPING IS 66 PSIG AT 150°F.
5. DELETED
6. FLANGES MARKED WITH SYMBOL ⚠ DENOTES USE OF A SPECTACLE BLIND (NORMALLY OPEN)
7. HEAT EXCHANGER INSTALLATION REQUIRED FOR SEISMIC RESTRAINT.
8. VENT, DRAIN AND TEST CONNECTIONS 1-1/2" AND BELOW CAN BE PROVIDED WITH PIPE CAPS OR HOSE CONNECTION FITTINGS WHERE REQUIRED BY PLANT PERSONNEL. THIS CONFIGURATION IS SUPPORTED BY ENGINEERING CALCULATION CD-Q0999-923399.
9. MECHANICAL TRAVEL STOP ON VALVE FCV-67-51 TO BE SET AT 26%, ±1% OF MAXIMUM OPENING WITH EECW FLOW TO RBCCW HEAT EXCHANGER ≥ 900 GPM WITH THE DOWNSTREAM TCV FULLY OPEN.
10. UNIDS ON DRAWING ARE FOR REFERENCE ONLY AND ARE ABBREVIATED TO MEET SPACE CONSTRAINTS, REFER TO MEL FOR COMPLETE UNIDS.
11. VALVE 3-SHV-67-576 SHALL BE LOCKED CLOSED UNLESS REQUIRED TO BE OPEN FOR MAINTENANCE/TESTING PURPOSES, PER NFPA 805.

REFERENCE DRAWINGS:

0-17W300-9..............PIPING - YARD
0-17W405-3..............MECHANICAL BUILDING DRAINAGE (YARD)
3-17W925-4 & -5.........MECHANICAL - HEATING, VENTILATING & A/C SYSTEM
MEL.....................VALVE MARKER TAG TABULATION
MEL.....................INSTRUMENT TABULATION FOR EECW SYSTEM
47W451-SERIES...........PIPING - REACTOR AND DIESEL GEN BLDG
0-47E586-1, 5. ....PIPING - UNITS 1&2 DIESEL GEN BLDG
0-47W586-2 THRU 4,-6....PIPING - UNITS 1&2 DIESEL GEN BLDG
47W586-5B...............PIPING - UNITS 1&2 DIESEL GEN BLDG
3-47W588-1 THRU 7.......PIPING - UNIT 3 DIESEL GEN BLDG
47W588-51,52,53.........PIPING - UNIT 3 DIESEL GEN BLDG
3-47E588-8, -404........PIPING - UNIT 3 DIESEL GEN BLDG
1-47E610-67-1...........CONTROL DIAGRAM - EECW SYSTEM
3-47E610-67-2...........CONTROL DIAGRAM - EECW SYSTEM
3-47E610-67-3...........CONT DIAGRAM EMERG EQPT COOLING WATER
47E800-1................FLOW DIAGRAM - GENERAL PLANT SYSTEMS
0-47E800-2..............MECHANICAL SYMBOLS & FLOW DIAGRAM DRAWING INDEX
1-47E844-2..............FLOW DIAGRAM - RAW COOLING WATER
3-47E844-2..............FLOW DIAGRAM - RAW COOLING WATER
0-47E851-4..............FLOW DIAGRAM - DRAINAGE
3-47E866-5,-7...........FLOW DIAGRAM - AIR CONDITIONING CHILLED WATER

COMPANION DRAWINGS:
1-47E859-1
2-47E859-1
3-47E859-2
1-,2-,3-47E859-1-ISI

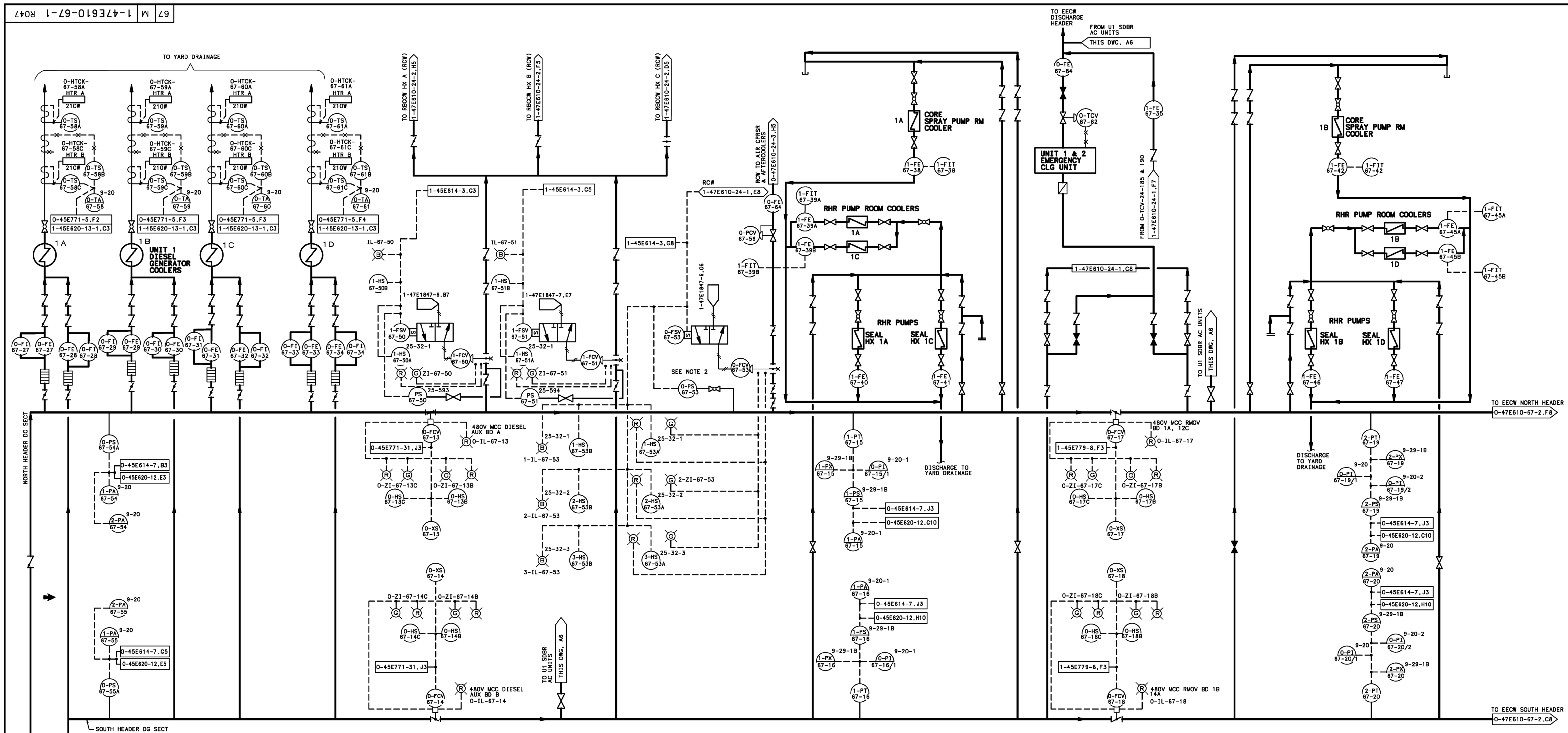
AMENDMENT 28

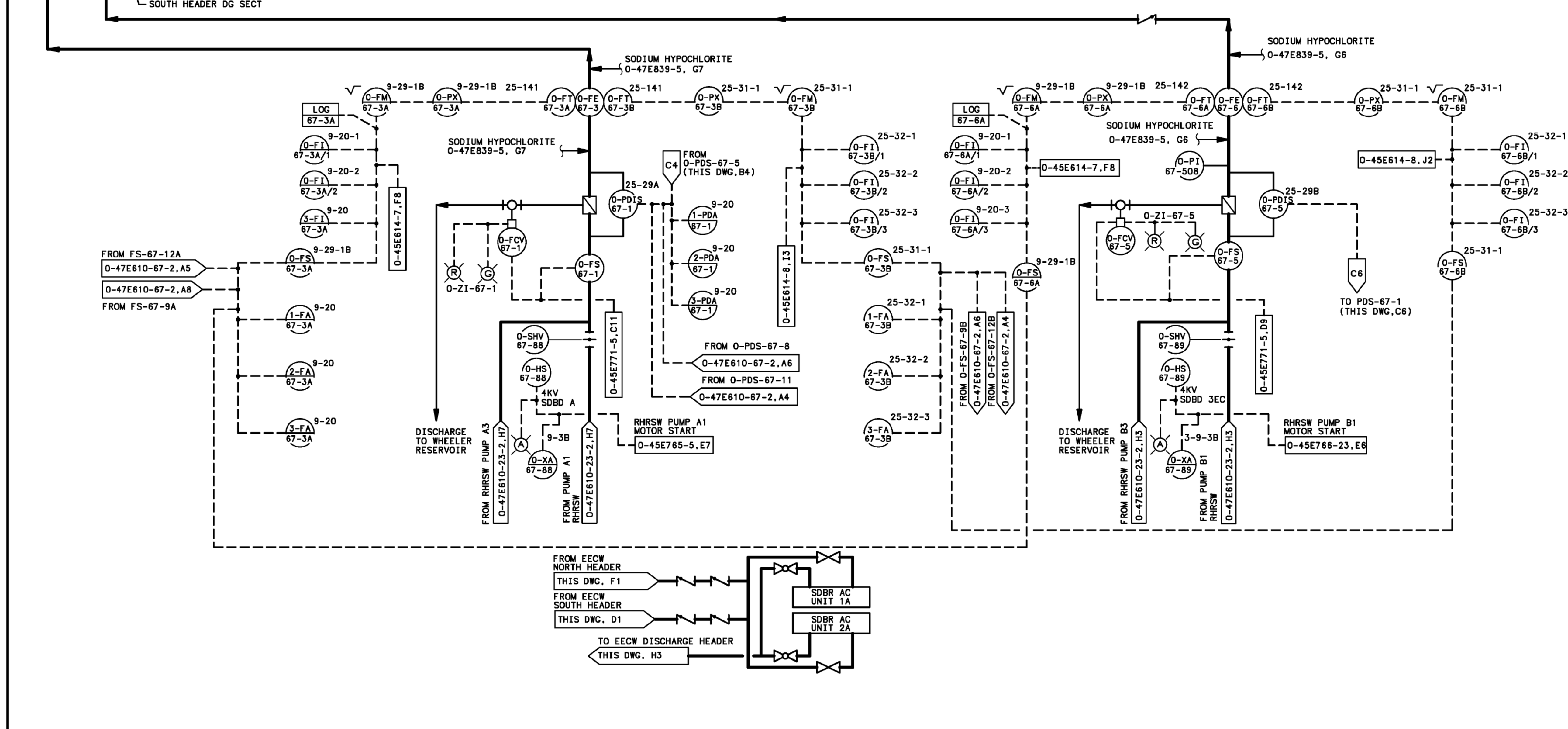
POWERHOUSE
UNIT 3

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

EMERGENCY EQUIPMENT COOLING WATER
FLOW DIAGRAM

FIGURE 10.10-1d





NOTES:

1. THE EMERGENCY EQUIPMENT COOLING WATER SYSTEM (EECW) IS A SYSTEM OF PIPING, VALVES, AND INSTRUMENTATION DESIGNED TO DISTRIBUTE COOLING WATER FROM THE RHR SERVICE WATER SYSTEM TO THE DIESEL GENERATORS, CORE SPRAY SYSTEM EQUIPMENT, AND RHR SYSTEM EQUIPMENT. IT ALSO PROVIDES RAW COOLING WATER BACKUP TO THE CONTROL ROOM CHILLER, CONTROL & SERVICES AIR COMPRESSORS, AND THE REACTOR BUILDING CLOSED COOLING WATER HEAT EXCHANGERS.
2. THE TWO HEADERS (NORTH AND SOUTH) BOTH FEED THE EECW SYSTEM AND THE CONTROL ROOM CHILLER. ONLY THE NORTH HEADER FEEDS THE AIR COMPRESSORS AND AFTER COOLERS WITH EECW.

   EECW BACKUP VALVES AND BACKUP CONTROL POWER TRANSFER SWITCH

   THE EECW BACKUP VALVES ALLOW COOLING WATER TO FLOW THROUGH THE RBCCW HEAT EXCHANGERS WHEN THE NORMAL SOURCE, THE RAW COOLING WATER (RCW) SYSTEM, FAILS. PRESSURE SWITCHES PS-24-133C AND PS-24-133D CLOSE CONTACTS ON LOW HEADER PRESSURE IN THE RCW HEADER TO THE RBCCW HEAT EXCHANGERS AND PS-67-50 CONTACTS CLOSE ON NORMAL EECW HEADER PRESSURE. THIS ENERGIZES FSV-67-50 AND OPENS FCV-67-50 WHEN HS-67-50 IS IN THE AUTO POSITION, ALLOWING THE EECW TO FLOW INTO THE HEADER FOR THE RBCCW HEAT EXCHANGER. LOW EECW HEADER PRESSURE OPENS PRESSURE SWITCH PS-67-50 CONTACTS, DE-ENERGIZES FSV-67-50, CLOSES FCV-67-50 AND ENERGIZES RELAY RLY-67-50. THIS PREVENTS FCV-67-50 FROM CYCLING IF PRESSURE SWITCH PS-67-50 CONTACTS RESET UPON INCREASE OF EECW PRESSURE ABOVE SETPOINT. THE CIRCUIT MUST BE MANUALLY RESET BY OPERATING HS-67-50B AT THE BACKUP CONTROL CENTER PANEL 25-32. THE FOREGOING DESCRIPTION FOR FSV-67-50 AND FCV-67-50 APPLIES ALSO TO FSV-67-51 AND FCV-67-51 EXCEPT FOR THE INSTRUMENT NUMBERS. FCV-67-50 SUPPLIES EECW BACKUP WATER FOR THE RBCCW HEAT EXCHANGERS FROM THE NORTH HEADER AND FCV-67-51 SUPPLIES WATER FROM THE SOUTH HEADER.

   THE BACKUP CONTROL CENTER PANEL 25-32 HAS A NORMAL AND ALTERNATE SUPPLY FROM DIFFERENT BATTERY BOARDS. A TRANSFER SWITCH ALLOWS THE OPERATOR TO MANUALLY TRANSFER FROM NORMAL TO ALTERNATE SUPPLY. AND PROVIDES AN OFF POSITION WHERE THE POWER SUPPLY IS DE-ENERGIZED.
3. BOTH EECW HEADERS HAVE SECTIONALIZING VALVES TO ISOLATE ANY UNIT THAT IS LEAKING WHILE SUPPLYING THAT UNIT FROM THE OTHER HEADER.
4. 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 SPECIFICATION 22A1470).
5. DELETED.
6. UNIDS ON DRAWING ARE FOR REFERENCE ONLY AND ARE ABBREVIATED TO MEET SPACE CONSTRAINTS. REFER TO MEL FOR COMPLETE UNIDS.
7. VALVE OPERATOR (MVOP) UNIDS HAVE THE SAME ADDRESS AS THE VALVE (FCV) UNID, e.g., 0-MVOP-67-1 FOR 0-FCV-67-1.
8. RHRSW PUMPS A3, B3, C3, & D3 ARE DEDICATED TO EECW. RHRSW PUMPS A1, B1, C1, & D1 ARE NORMALLY ALIGNED TO RHRSW BUT CAN BE RE-ALIGNED TO EECW AS LONG AS THERE IS NOT ANOTHER #3 PUMP ALIGNED TO THAT PARTICULAR STRAINER (E.G. CANNOT HAVE A1 USED FOR EECW WITH A3 PUMP IN USE).

REFERENCE DRAWINGS:

MEL..................INSTRUMENT TABULATION SYSTEM 24
MEL..................INSTRUMENT TABULATION SYSTEM 23 & 67
0-47E610-23-2.......CONTROL DIAGRAM-RHR SERVICE WATER
1-47E610-24-1 & -2..CONTROL DIAGRAM-RCW
0-47E610-24-3.......CONTROL DIAGRAM-RCW
0-47E800-1..........FLOW DIAGRAM-GENERAL PLANT SYSTEMS
0-47E800-2..........MECH SYMBOLS & FLOW DIAGRAM DRAWING INDEX
0-47E839-5..........FLOW AND CONTROL DIAGRAM-SODIUM HYPOCHLORITE
1-47E844-1 & -2.....FLOW DIAGRAM-RCW
1-47E859-1..........FLOW DIAGRAM-EECW
0-47E866-3..........FLOW DIAGRAM-HEATING & AIR CONDITIONING HOT & CHILLED WATER
3-47E866-7..........FLOW DIAGRAM-HEATING & AIR CONDITIONING HOT & CHILLED WATER
1-45E751-3 & -5.....WIRING DIAGRAMS-480V REACTOR AUX POWER SCHEMATIC DIAGRAM
0-45E765-5..........WIRING DIAGRAM-4160V SHUTDOWN AUX POWER SCHEMATIC DIAGRAM
0-45E771-4..........WIRING DIAGRAM-480V DIESEL AUX POWER SCHEMATIC DIAGRAM
1-45E779-8..........WIRING DIAGRAM-480V SHUTDOWN AUX POWER SCHEMATIC DIAGRAM
1-45E614-3..........WIRING DIAGRAM 120V AC/250V DC VALVES & MISC SCHEMATIC DIAGRAM
0-45E614-7..........WIRING DIAGRAM 120V AC/250V DC VALVES & MISC SCHEMATIC DIAGRAM
0-45E771-31.........WIRING DIAGRAM 480V DIESEL AUX POWER SCHEMATIC DIAGRAM
0-45E614-8..........WIRING DIAGRAM 120V AC/250V DC VALVES & MISC SCHEMATIC DIAGRAM
1-45E620-13-1.......WIRING DIAGRAM ANNUNCIATOR SYSTEM KEY DIAGRAM
0-45E620-12.........WIRING DIAGRAM ANNUNICATOR SYSTEM KEY DIAGRAM
0-45E771-5..........WIRING DIAGRAM 480V DIESEL AUX POWER SCHEMATIC DIAGRAM
0-45E766-23.........WIRING DIAGRAM 4160V SHUTDOWN AUX POWER SCHEMATIC DIAGRAM
0-76N3S-1...........VENDOR AIR CONDITIONING UNIT P & ID, CONTRACT 21042-M-0106Q

AMENDMENT 28

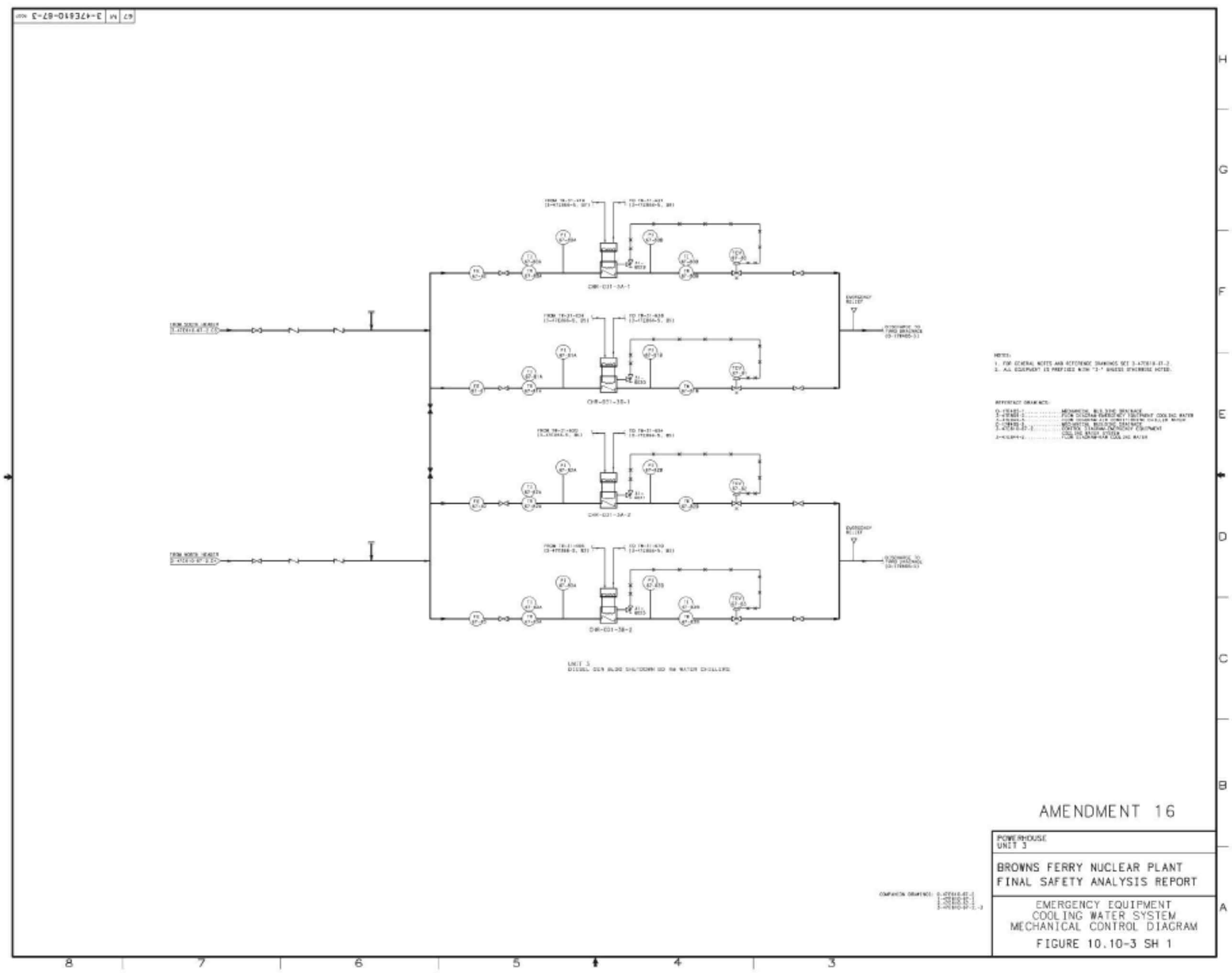
COMPANION DRAWINGS:
0-47E610-67-2
2-47E610-67-2
3-47E610-67-2,-3

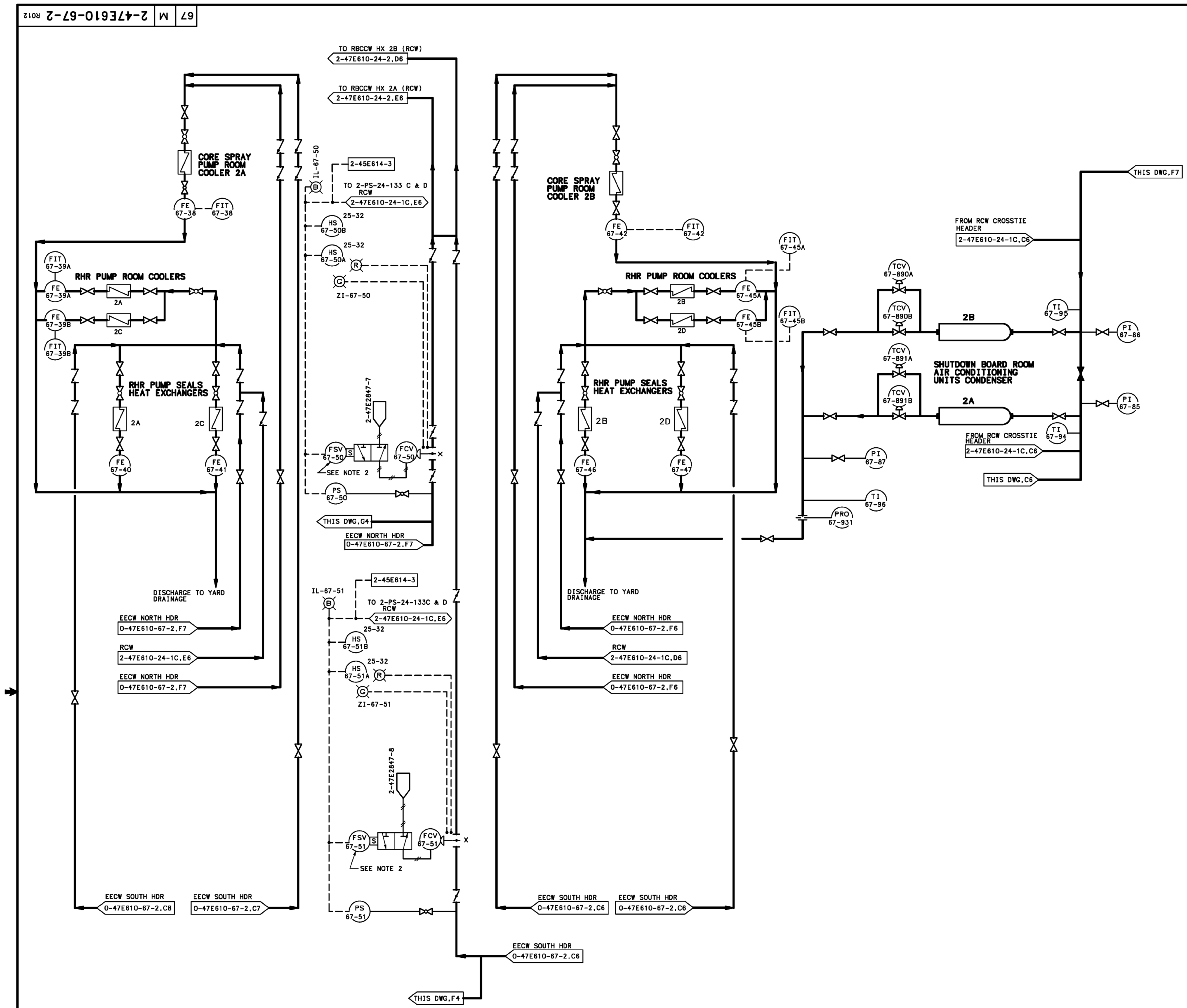
POWERHOUSE
UNITS 1 & 0

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

EMERGENCY EQUIPMENT
COOLING WATER SYSTEM
MECHANICAL CONTROL DIAGRAM

FIGURE 10.10-2


AMENDMENT 16
POWERHOUSE
UNIT 3
BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT
EMERGENCY EQUIPMENT
COOLING WATER SYSTEM
MECHANICAL CONTROL DIAGRAM
FIGURE 10.10-3 SH 1



NOTES:

1. THE EMERGENCY EQUIPMENT COOLING WATER SYSTEM (EECW) IS A SYSTEM OF PIPING, VALVES, AND INSTRUMENTATION DESIGNED TO DISTRIBUTE COOLING WATER FROM THE RHR SERVICE WATER SYSTEM TO THE DIESEL GENERATORS, CORE SPRAY SYSTEM EQUIPMENT, AND RHR SYSTEM EQUIPMENT. IT ALSO PROVIDES RAW COOLING WATER BACKUP TO THE CONTROL ROOM CHILLERS, CONTROL & SERVICE AIR COMPRESSORS, AND THE REACTOR BUILDING CLOSED COOLING WATER HEAT EXCHANGERS. SEE DWG 1-47E610-67-1.

2. THE TWO HEADERS (NORTH AND SOUTH) BOTH FEED THE EECW SYSTEM AND THE CONTROL ROOM CHILLERS. ONLY THE NORTH HEADER FEEDS THE AIR COMPRESSORS AND AFTERCOOLERS.

   EECW BACKUP VALVES AND BACKUP CONTROL POWER TRANSFER SWITCH

   THE EECW BACKUP VALVES ALLOW EECW COOLING WATER TO FLOW THROUGH THE RBCCW HEAT EXCHANGERS AND THE AIR COMPRESSORS AND THE AFTER-COOLERS WHEN THE NORMAL SOURCE, THE RAW COOLING WATER (RCW) SYSTEM, FAILS.

   PRESSURE SWITCHES PS-24-133C AND PS-24-133D CLOSE CONTACTS ON LOW HEADER PRESSURE IN THE RCW HEADER TO THE RBCCW HEAT EXCHANGERS AND PS-67-50 CONTACTS CLOSE ON NORMAL EECW HEADER PRESSURE. THIS ENERGIZES FSV-67-50 AND OPENS FCV-67-50 WHEN HS-67-50 IS IN THE AUTO POSITION, ALLOWING THE EECW TO FLOW INTO THE HEADER FOR THE RBCCW HEAT EXCHANGER. LOW EECW HEADER PRESSURE OPENS PRESSURE SWITCH PS-67-50 CONTACTS, DE-ENERGIZES FSV-67-50, CLOSES FCV-67-50 AND ENERGIZES RELAY RLY-67-50. THIS PREVENTS FCV-67-50 FROM CYCLING IF PRESSURE SWITCH PS-67-50 CONTACTS RESET UPON INCREASE OF EECW PRESSURE ABOVE SETPOINT. THE CIRCUIT MUST BE MANUALLY RESET BY OPERATING HS-67-50B AT THE BACKUP CONTROL CENTER PANEL 25-32.

   THE FOREGOING DESCRIPTION FOR FSV-67-50 AND FCV-67-50 APPLIES ALSO TO FSV-67-51 AND FCV-67-51 EXCEPT FOR THE INSTRUMENT NUMBERS. FCV-67-50 SUPPLIES EECW BACKUP WATER FOR THE RBCCW HEAT EXCHANGERS FROM THE NORTH HEADER AND FCV-67-51 SUPPLIES WATER FROM THE SOUTH HEADER. FCV-67-53 SUPPLIES EECW BACKUP WATER FOR THE AIR COMPRESSORS FROM THE NORTH HEADER. THIS VALVE IS ON UNIT 1 ONLY.

   THE BACKUP CONTROL CENTER PANEL 25-32 HAS A NORMAL AND ALTERNATE SUPPLY FROM DIFFERENT BATTERY BOARDS. A TRANSFER SWITCH ALLOWS THE OPERATOR TO MANUALLY TRANSFER FROM NORMAL TO ALTERNATE SUPPLY AND PROVIDES AN OFF POSITION WHERE THE POWER SUPPLY IS DE-ENERGIZED.

3. BOTH EECW HEADERS HAVE SECTIONALIZING VALVES TO ISOLATE ANY UNIT THAT IS LEAKING WHILE SUPPLYING THAT UNIT FROM THE OTHER HEADER.

4. 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 SPECIFICATION 22A1470).

5. FOR GENERAL NOTES AND REFERENCE DRAWINGS SEE 1-47E610-67-1.

REFERENCE DRAWINGS:

47W611-24-1	LOGIC DIAGRAM - RCW
2-47E844-1 & 2	FLOW DIAGRAM - RCW
MEL	INSTRUMENT TABULATION - EECW
2-47E859-1	FLOW DIAGRAM - EECW
2-47E610-24-1C	MECHANICAL CONTROL DIAGRAM - RCW
47E800-1	FLOW DIAGRAM - GENERAL PLANT SYSTEMS
0-47E800-2	MECHANICAL SYMBOLS AND FLOW DIAGRAM DRAWING INDEX
2-45E614-3	WIRING DIAGRAMS - 120V AC/250V DC VALVES AND MISC SCHEMATIC DIAGRAM

COMPANION DRAWINGS: 0-47E610-67-2, 1-47E610-67-1, 3-47E610-67-2 & -3

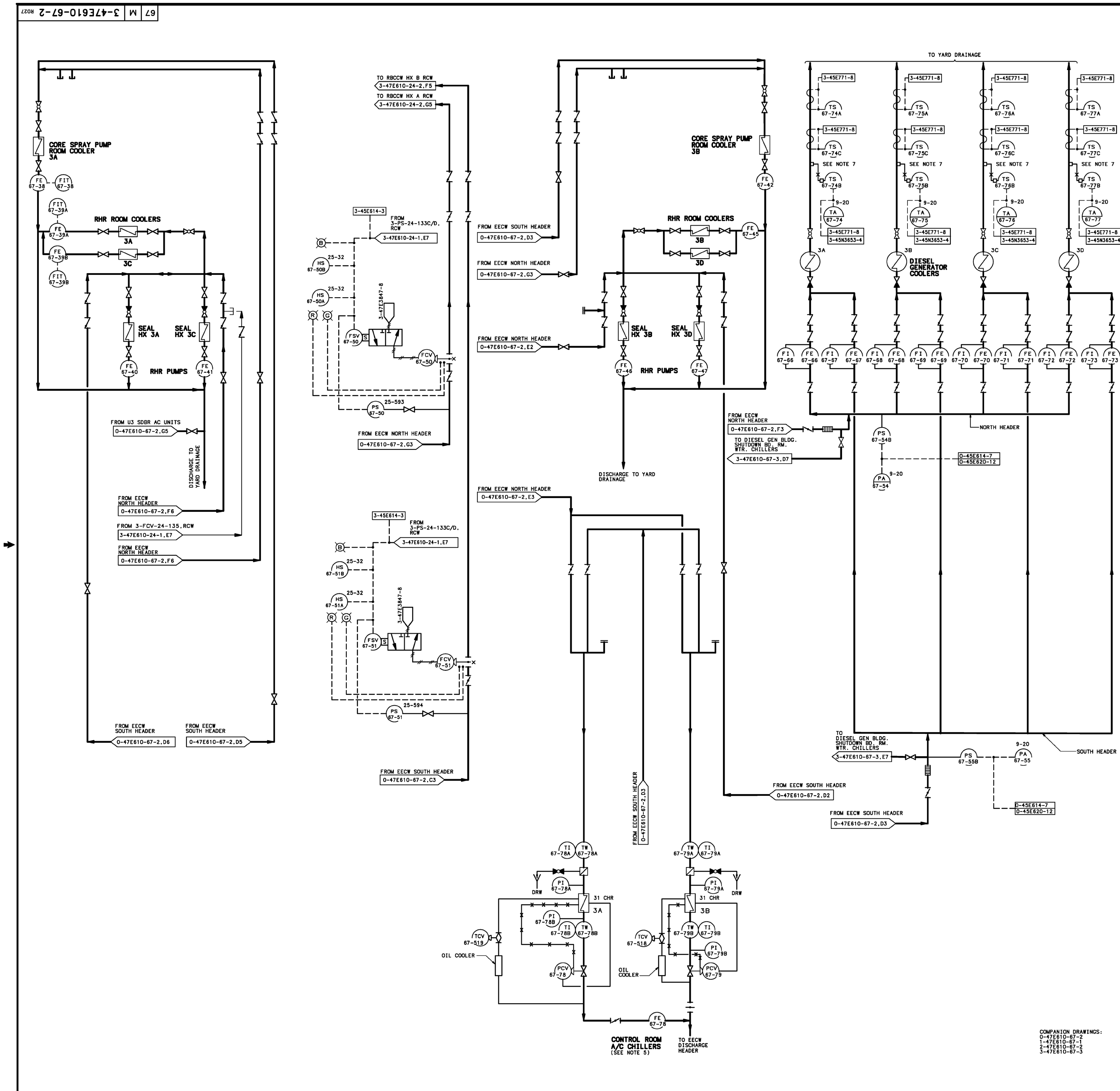
AMENDMENT 28

POWERHOUSE UNIT 2

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

EMERGENCY EQUIPMENT COOLING WATER SYSTEM MECHANICAL CONTROL DIAGRAM

FIGURE 10.10-3 SH 2



NOTES:

1. THE EMERGENCY EQUIPMENT COOLING WATER SYSTEM (EECW) IS A SYSTEM OF PIPING, VALVES, AND INSTRUMENTATION DESIGNED TO DISTRIBUTE COOLING WATER FROM THE RHR SERVICE WATER SYSTEM TO THE DIESEL GENERATORS, CORE SPRAY SYSTEM EQUIPMENT, AND RHR SYSTEM EQUIPMENT. IT ALSO PROVIDES RAW COOLING WATER BACKUP TO THE CONTROL ROOM CHILLER, CONTROL AND SERVICE AIR COMPRESSORS AND THE REACTOR BUILDING CLOSED COOLING WATER HEAT EXCHANGERS.
2. THE TWO HEADERS (NORTH AND SOUTH) BOTH FEED THE EECW SYSTEM AND THE CONTROL ROOM CHILLER. ONLY THE NORTH HEADER FEEDS THE AIR COMPRESSORS AND AFTER COOLERS WITH EECW. THIS RCW BACKUP VALVE IS OPENED BY THE PRESSURE IN THE EECW NORTH HEADER WHEN THEIR SOLENOID HAS BEEN ENERGIZED BY EITHER OF THE TWO PRESSURE SWITCHES SENSING THE RAW COOLING WATER PRESSURE SUPPLY AT THE EQUIPMENT.
3. BOTH EECW HEADERS HAVE SECTIONALIZING VALVES TO ISOLATE ANY UNIT THAT IS LEAKING WHILE SUPPLYING THAT UNIT FROM THE OTHER HEADER.
4. 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 SPECIFICATION 22A1470).
5. FOR INSTRUMENTATION ASSOCIATED WITH THE CONTROL ROOM AIR-CONDITIONING CHILLERS, SEE FLOW DIAGRAM 3-47E866-7 AND INSTRUMENT TABULATION 0-47B601-24-0.
6. FOR INSTRUMENTATION ASSOCIATED WITH THE DIESEL GENERATOR BUILDING SHUTDOWN BOARD ROOM AIR-CONDITIONING CHILLERS, SEE FLOW DIAGRAM 3-47E866-5.
7. SURFACE MOUNTED CAPILLARY TUBE TEMPERATURE ELEMENT.
8. ALL COMPONENTS HAVE A PREFIX "3-" UNLESS OTHERWISE NOTED.
9. EECW BACKUP VALVES AND BACKUP CONTROL POWER TRANSFER SWITCH.
THE EECW BACKUP VALVES ALLOW COOLING WATER TO FLOW THROUGH THE RBCCW HEAT EXCHANGERS WHEN THE NORMAL SOURCE, THE RAW COOLING WATER (RCW) SYSTEM, FAILS. PRESSURE SWITCHES PS-24-133C AND PS-24-133D CLOSE CONTACTS ON LOW HEADER PRESSURE IN THE RCW HEADER TO THE RBCCW HEAT EXCHANGERS AND PS-67-50 CONTACTS CLOSE ON NORMAL EECW HEADER PRESSURE. THIS ENERGIZES FSV-67-50 AND OPENS FCV-67-50 WHEN HS-67-50 IS IN THE AUTO POSITION, ALLOWING THE EECW TO FLOW INTO THE HEADER FOR THE RBCCW HEAT EXCHANGER. LOW EECW HEADER PRESSURE OPENS PRESSURE SWITCH PS-67-50 CONTACTS, DE-ENERGIZES FSV-67-50, CLOSES FCV-67-50 AND ENERGIZES RELAY RLY-67-50. THIS PREVENTS FCV-67-50 FROM CYCLING IF PRESSURE SWITCH PS-67-50 CONTACTS RESET UPON INCREASE OF EECW PRESSURE ABOVE SETPOINT. THE CIRCUIT MUST BE MANUALLY RESET BY OPERATING HS-67-50B AT THE BACKUP CONTROL CENTER PANEL 25-32.
THE FORGOING DESCRIPTION FOR FSV-67-50 AND FCV-67-50 APPLIES ALSO TO FSV-67-51 AND FCV-67-51 EXCEPT FOR THE INSTRUMENT NUMBERS, FCV-67-50 SUPPLIES EECW BACKUP WATER FOR THE RBCCW HEAT EXCHANGERS FROM THE NORTH HEADER AND FCV-67-51 SUPPLIES WATER FROM THE SOUTH HEADER.
THE BACKUP CONTROL CENTER PANEL 25-32 HAS A NORMAL AND ALTERNATE SUPPLY FROM DIFFERENT BATTERY BOARDS. A TRANSFER SWITCH ALLOWS THE OPERATO MANUALLY TRANSFER FROM NORMAL TO ALTERNATE SUPPLY AND PROVIDES AN OFF POSITION WHERE THE POWER SUPPLY IS DE-ENERGIZED.

REFERENCE DRAWINGS:

0-17E405-1............MECHANICAL BUILDING DRAINAGE
MEL...................INSTRUMENT TABULATION - RCW (SYSTEM 24)
MEL...................INSTRUMENT TABULATION - EECW (SYSTEM 67)
3-47E610-24-1,-2......CONTROL DIAGRAM - RCW
0-47E800-2............MECHANICAL SYMBOLS AND FLOW DIAGRAM DRAWING INDEX
47E800-1..............FLOW DIAGRAM - GENERAL PLANT SYSTEMS
3-47E844-2............FLOW DIAGRAM - RCW
3-47E859-1 & -2.......FLOW DIAGRAM - EECW
3-47E866-7............FLOW DIAGRAM - HEATING AND AIR CONDITIONING HOT AND CHILLED WATER
3-45E614-3............WIRING DIAGRAM - 120V AC/250V DC VALVES & MISC SCHEMATIC DIAGRAM
0-45E614-7............WIRING DIAGRAM - 120V AC/250V DC VALVES & MISC SCHEMATIC DIAGRAM
0-45E620-12...........ANNUNCIATOR KEY DIAGRAM
0-45E771-5............WIRING DIAGRAM - 480V DIESEL AUX POWER SCHEMATIC DIAGRAM
3-45E771-8............WIRING DIAGRAM - 480V DIESEL AUX POWER SCHEMATIC DIAGRAM
3-45N3653-4...........WIRING DIAGRAM - 480V DIESEL AUX PWR SCHEMATIC DIAGRAM
3-47E866-5............FLOW DIAGRAM - AIR CONDITIONING CHILLED WATER
3-47E866-7............FLOW DIAGRAM - AIR CONDITIONING CHILLED WATER
3-47E844-2............FLOW DIAGRAM - RAW COOLING WATER
3-47E610-67-3.........CONTROL DIAGRAM-EECW
0-47E610-67-2.........CONTROL DIAGRAM-EECW
1-47E859-1............FLOW DIAGRAM-EECW

COMPANION DRAWINGS:
0-47E610-67-2
1-47E610-67-1
2-47E610-67-2
3-47E610-67-3

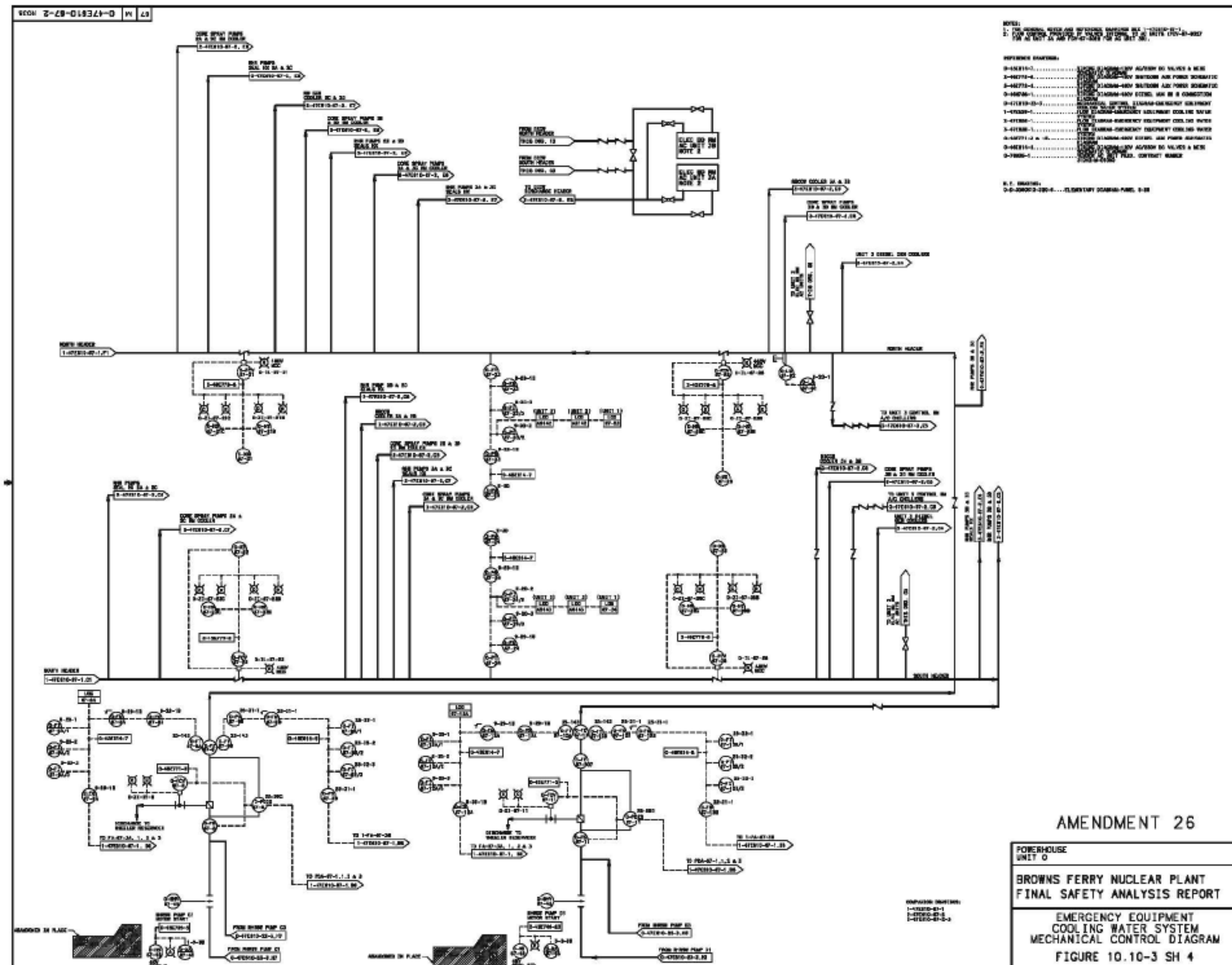
AMENDMENT 28

POWERHOUSE
UNIT 3

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

EMERGENCY EQUIPMENT
COOLING WATER SYSTEM
MECHANICAL CONTROL DIAGRAM

FIGURE 10.10-3 SH 3



AMENDMENT 26

POWERHOUSE
UNIT 0

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

EMERGENCY EQUIPMENT
COOLING WATER SYSTEM
MECHANICAL CONTROL DIAGRAM

FIGURE 10.10-3 SH 4

Figure 10.10-4

Deleted by Amendment 10

Figures 10.10-4a and 10.10-4b
(Deleted by Amendment 22)

10.11 FIRE PROTECTION

The fire protection program is based on the NRC requirements and guidelines, Nuclear Electric Insurance Limited (NEIL) Property Loss Prevention Standards and related industry standards. With regard to NRC criteria, the fire protection program meets the requirements of 10 CFR 50.48(c), which endorses, with exceptions, the National Fire Protection Association's (NFPA) 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants – 2001 Edition." BFN has further used the guidance of NEI 04-02, "Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program under 10 CFR 50.48(c)" as endorsed by Regulatory Guide 1.205, "Risk-Informed, Performance Fire Protection for Existing Light-Water Nuclear Power Plants."

Adoption of NFPA 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants", 2001 Edition in accordance with 10 CFR 50.48(c) serves as the method of satisfying 10 CFR 50.48(a) and General Design Criterion 3. Prior to adoption of NFPA 805, General Design Criterion 3, "Fire Protection" of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Licensing of Production and Utilization Facilities," was followed in the design of safety and non-safety related structures, systems, and components, as required by 10 CFR 50.48(a).

NFPA 805 does not supersede the requirements of GDC 3, 10 CFR 50.48(a), or 10 CFR 50.48(f). Those regulatory requirements continue to apply. However, under NFPA 805, the means by which GDC 3 or 10 CFR 50.48(a) requirements are met may be different than under 10 CFR 50.48(b). Specifically, whereas GDC 3 refers to SSCs important to safety, NFPA 805 identifies fire protection systems and features required to meet the Chapter 1 performance criteria through the methodology in Chapter 4 of NFPA 805. Also, under NFPA 805, the 10 CFR 50.48(a)(2)(iii) requirement to limit fire damage to SSCs important to safety so that the capability to safely shut down the plant is satisfied by meeting the performance criteria in Section 1.5.1 of NFPA 805.

A Safety Evaluation was issued on October 28, 2015, by the NRC, that transitioned the existing fire protection program to a risk-informed, performance-based program based on NFPA 805, in accordance with 10 CFR 50.48(c).

10.11.1 Design Basis Summary

10.11.1.1 Defense-in-Depth

The fire protection program is focused on protecting the safety of the public, the environment, and plant personnel from a plant fire and its potential effect on safe reactor operations. The fire protection program is based on the concept of defense-

in-depth. Defense-in-depth shall be achieved when an adequate balance of each of the following elements is provided:

(1) Preventing fires from starting;
(2) Rapidly detecting fires and controlling and extinguishing promptly those fires that do occur, thereby limiting fire damage; and
(3) Providing an adequate level of fire protection for structures, systems, and components important to safety, so that a fire that is not promptly extinguished will not prevent essential plant safety functions from being performed.

10.11.1.2 NFPA 805 Performance Criteria

The design basis for the fire protection program is based on the following nuclear safety and radiological release performance criteria contained in Section 1.5 of NFPA 805:

- Nuclear Safety Performance Criteria. Fire protection features shall be capable of providing reasonable assurance that, in the event of a fire, the plant is not placed in an unrecoverable condition. To demonstrate this, the following performance criteria shall be met.

  (a) Reactivity Control. Reactivity control shall be capable of inserting negative reactivity to achieve and maintain subcritical conditions. Negative reactivity inserting shall occur rapidly enough such that fuel design limits are not exceeded.

  (b) Inventory and Pressure Control. With fuel in the reactor vessel, head on and tensioned, inventory and pressure control shall be capable of maintaining or rapidly restoring reactor water level above top of active fuel such that fuel clad damage as a result of a fire is prevented.

  (c) Decay Heat Removal. Decay heat removal shall be capable of removing sufficient heat from the reactor core or spent fuel such that fuel is maintained in a safe and stable condition.

  (d) Vital Auxiliaries. Vital auxiliaries shall be capable of providing the necessary auxiliary support equipment and systems to assure that the systems required under (a), (b), (c), and (e) are capable of performing their required nuclear safety function.

  (e) Process Monitoring. Process monitoring shall be capable of providing the necessary indication to assure the criteria addressed in (a) through (d) have

been achieved and are being maintained.

- Radioactive Release Performance Criteria. Radiation release to any unrestricted area due to the direct effects of fire suppression activities (but not involving fuel damage) shall be as low as reasonably achievable and shall not exceed applicable 10 CFR, Part 20, Limits.

Chapter 2 of NFPA 805 establishes the process for demonstrating compliance with NFPA 805.

Chapter 3 of NFPA 805 contains the fundamental elements of the fire protection program and specifies the minimum design requirements for fire protection systems and features.

Chapter 4 of NFPA 805 establishes the methodology to determine the fire protection systems and features required to achieve the nuclear safety performance criteria outlined above. The methodology shall be permitted to be either deterministic or performance-based. Deterministic requirements shall be “deemed to satisfy” the performance criteria, defense-in-depth, and safety margin and require no further engineering analysis. Once a determination has been made that a fire protection system or feature is required to achieve the nuclear safety performance criteria of Section 1.5, its design and qualification shall meet the applicable requirement of Chapter 3.

10.11.1.3 Codes of Record

The codes, standards and guidelines used for the design and installation of plant fire protection systems are as follows: (for specific applications and evaluations of codes refer to the Fire Protection Report)

a) American National Standards Institute (ANSI)
b) American Society for Testing Materials (ASTM)
c) Factory Mutual (FM) Research Fire Protection Equipment Approval Guide
d) Institute of Electrical and Electronic Engineers (IEEE)
    - IEEE 383 – 1974, Standard for Type Test of Class 1E Electric Cables, Field Splices, and Connections for Nuclear Power Generating Stations
    - IEEE 384 – 1974, Criteria for Separation of Class 1E Equipment
    - IEEE 634 – 1978, Standard Cable Penetration Fire Stop Qualification Test
e) National Fire Protection Association (NFPA)
    - NFPA 10 – 1967 Ed., Installation of Portable Fire Extinguishers
    - NFPA 12 – 1966 Ed., Carbon Dioxide Extinguishing Systems
    - NFPA 13 – 1985 Ed., Standard for the Installation of Sprinkler Systems (Control Bay and Intake Pumping Station)

- NFPA 13 – 1987 Ed., Standard for the Installation of Sprinkler Systems (Unit 2 Reactor Building and Units 1, 2, 3 Battery and Battery Board Rooms)
- NFPA 13 – 1991 Ed., Standard for the Installation of Sprinkler Systems (Unit 3 Reactor Building, Units 2 & 3 HPCI Rooms, and Cable Spreading Rooms A & B)
- NFPA 13 – 2002 Ed., Standard for the Installation of Sprinkler Systems (Unit 1 Reactor Building and Unit 1 HPCI Room)
- NFPA 14 – 1986 Ed., Standpipe and Hose Systems
- NFPA 15 – 1985 Ed., Standard for Water Spray Fixed Systems for Fire Protection
- NFPA 15 – 2001 Ed., Standard for Water Spray Fixed Systems for Fire Protection
- NFPA 20 – 1987 Ed., Standard for the Installation of Stationary Pumps for Fire Protection
- NFPA 20 – 1999 Ed., Standard for the Installation of Stationary Pumps for Fire Protection
- NFPA 22 – 1971 Ed., Standard for Water Tanks for Private Fire Protection
- NFPA 24 – 1984 Ed., Standard for the Installation of Private Fire Service Mains
- NFPA 30 – 1969 Ed., Flammable and Combustible Liquids Code
- NFPA 50A – 1984 Ed., Standard for Gaseous Hydrogen at Consumer Sites
- NFPA 72 – 1990 Ed., Standard for the Installation, Maintenance, and Use of Protective Signaling Systems
- NFPA 72 – 2002 Ed., Standard for the Installation, Maintenance, and Use of Protective Signaling Systems
- NFPA 80 – 1986 Ed., Standard for Fire Doors and Fire Windows
- NFPA 80 – 1992 Ed., Standard for Fire Doors and Fire Windows
- NFPA 90A – 1989 Ed., Standard for Installation of Air Conditioning and Ventilating Systems
- NFPA 600 – 2000 Ed., Standard on Industrial Fire Brigades
- NFPA 805 – 2001 Ed., Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants

10.11.2 <u>System Description</u>

10.11.2.1 <u>Required Systems</u>

Nuclear Safety Capability Systems, Equipment, and Cables

Section 2.4.2 of NFPA 805 defines the methodology for performing the nuclear safety capability assessment. The systems, equipment and cables required for the nuclear safety capability assessment are contained in TVA calculation EDQ099920110010, NFPA 805 – Nuclear Safety Capability Analysis.

Fire Protection Systems and Features

Chapter 3 of NFPA 805 contains the fundamental elements of the fire protection program and specifies the minimum design requirements for fire protection systems and features. Compliance with Chapter 3 is documented in the Fire Protection Report.

Chapter 4 of NFPA 805 establishes the methodology and criteria to determine the fire protection systems and features required to achieve the nuclear safety performance criteria of Section 1.5 of NFPA 805. These fire protection systems and features shall meet the applicable requirements of NFPA 805 Chapter 3. These fire protection systems and features are documented in the Fire Protection Report.

Radioactive Release

Structures, systems, and components relied upon to meet the radioactive release criteria are documented in the Fire Protection Report.

10.11.2.2 Definition of “Power Block” Structures

Where used in NFPA 805 Chapter 3 the terms “Power Block” and “Plant” refer to structures that have equipment required for nuclear plant operations. For the purposes of establishing the structures included in the fire protection program in accordance with 10 CFR 50.48(c) and NFPA 805, the plant structures listed in Table 10.11-1 are considered to be part of the ‘Power Block’.

10.11.3 Safety Evaluation

The Fire Protection Report documents the achievement of the nuclear safety and radioactive release performance criteria of NFPA 805 as required by 10 CFR 50.48(c). This document fulfills the requirements of Section 2.7.1.2 “Fire Protection Program Design Basis Document” of NFPA 805. The document contains the following:

- Identification of significant fire hazards in the fire area. This is based on NFPA 805 approach to analyze the plant from an ignition source and fuel package perspective.
- Summary of the Nuclear Safety Capability Assessment (at power and non-power) compliance strategies.
    - Deterministic compliance strategies

  - Performance-based compliance strategies (including defense-in-depth and safety margin)
- Summary of the Non-Power Operations Modes compliance strategies.
- Summary of the Radioactive Release compliance strategies.
- Summary of the Fire Probabilistic Risk Assessments.
- Key analysis assumptions to be included in the NFPA 805 monitoring program.

10.11.4 Fire Protection Program Documentation, Configuration Control and Quality Assurance

In accordance with Chapter 3 of NFPA 805 a fire protection plan documented in FPDP-6 defines the management policy and program direction and defines the responsibilities of those individuals responsible for the plan's implementation. TVA procedure FPDP-6:

- Designates the senior management position with immediate authority and responsibility for the fire protection program.

- Designates a position responsible for the daily administration and coordination of the fire protection program and its implementation.

- Defines the fire protection interfaces with other organizations and assigns responsibilities for the coordination of activities.

- Identifies the appropriate authority having jurisdiction for the various areas of the fire protection program.

- Identifies the procedures established for the implementation of the fire protection program, including the post-transition change process and the fire protection monitoring program.

- Identifies the qualifications required for various fire protection program personnel.

- Identifies the quality requirements of Chapter 2 of NFPA 805.

Detailed compliance with the programmatic requirements of Chapter 3 of NFPA 805 are contained in the Fire Protection Report.

Table 10.11-1
Power Block Structures

Unit 1 Reactor Building
Unit 2 Reactor Building
Unit 3 Reactor Building
Control Building
Turbine Building
Radwaste Building
Unit 1/2 Diesel Generator Building
Unit 3 Diesel Generator Building
Intake Pumping Station
Off-Gas Building & Off-Gas Stack
Standby Gas Treatment Building
Cooling Towers & Channel Diesel Fire Pump Building
Yard
Reactor Building Air Intake Plenum – Units 1, 2, 3
161kV and 500kV Switchyards
Drywell Containment

Figure 10.11-1

Deleted by Amendment 9.

Figures 10.11-1a through 10.11-1b

Deleted by Amendment 9.

Figures 10.11-2 through 10.11-12

Deleted by Amendment 9.

10.12 HEATING, VENTILATING AND AIR-CONDITIONING SYSTEMS

10.12.1 Safety Objective

The safety objective of the Heating, Ventilating, and Air Conditioning Systems is to maintain the control bay environment required for protection of equipment and for occupancy throughout the life of the plant.

10.12.2 Power Generation Objective

The power generation objective of the Heating, Ventilating, and Air Conditioning Systems is to maintain all areas at the environmental conditions required for operation of equipment, to maintain all occupied areas at conditions required for comfort and safety of personnel, and to limit the spread of contamination during power and shutdown operation of the plant.

10.12.3 Safety Design Basis

The control bay ventilation and air conditioning systems shall maintain the temperature of the control and electrical board rooms within the acceptable limits for operation of instruments and for uninterrupted safe occupancy under all plant conditions.

The diesel generator building's HVAC systems shall be capable of maintaining the required conditions for safety-related equipment.

10.12.4 Power Generation Design Basis

1. The systems shall maintain all plant areas at the temperature conditions necessary for operation of equipment.

2. The systems shall maintain all occupied areas within the temperature and humidity range required for human occupancy.

3. The systems shall limit the spread of radioactive contamination and provide for the filtration of air before exhaust from areas where significant airborne activity is expected.

4. The systems shall provide for the safe disposal of combustible or otherwise undesirable vapors and gases.

5. The systems shall limit the spread of radioactive airborne contamination to the control bay by providing the capability for automatic control bay isolation and emergency pressurization.

10.12.5 Description

10.12.5.1 General

Reactor Building, Turbine Building, Radwaste Building, and Diesel Generator Buildings have separate systems for year-round ventilation. The Control Bay is equipped with an air conditioning system. A common plant heating system serves the Reactor, Turbine, and Radwaste Buildings. The Diesel Generator Buildings are equipped with electric resistance heaters. A separate, electric hot-water heating system serves the Control Bay. The plant heating system is a forced hot water system designed to maintain indoor temperatures necessary for the protection of equipment and for personnel comfort. It is also designed to preheat the fresh air supply to the buildings, as required, during winter conditions.

The following paragraphs of this FSAR section provide descriptions of the Ventilation systems for the Turbine Building, Radwaste Building, and Diesel Generator Buildings. In addition, a description is provided of the Control Bay HVAC system.

The Reactor Building ventilation system is described in paragraph 5.3.3.6.

The Drywell Ventilation and Cooling System is described in 5.2.3.7.

10.12.5.2 Turbine Building

The Turbine Building is heated, cooled, and ventilated during normal and shutdown operations by a circulating air system. Fresh air heating coils and unit space heaters are served by the building heating hot water system.

The building heating, ventilating, and cooling systems were designed to provide a summer maximum inside temperature of 105°F with outside conditions at 97°F, and a winter inside minimum temperature of 55°F when the outside temperature is 5°F. These systems are shared in the sense that the turbines are in a common ventilation zone.

All outside air enters the building through fan room roof hoods, is filtered, passes across hot water coils for winter heating, through evaporative coolers for summer cooling, and hence to the supply fans. Air flow is routed to areas of progressively greater radioactive contamination potential. In rooms where offgases might pose a potential health problem to personnel, a slight vacuum is created by the ventilation system so that any offgas, such as from the hydrogen analyzers in the offgas monitor panel, is carried out through the exhaust fans and checked for high radiation levels. The air passes through door grilles (with the exception of the offgas monitor cubicle which has a labyrinth entry arrangement and the offgas recombiner room which has a gap between door and room floor), with backflow or air-check dampers and adjustable dampers for manually balancing air flow quantities. Each plant unit has two 100-percent capacity Turbine Building exhaust fans located on the Reactor Building roof. These fans discharge through a fan stack with the top at El. 735. The air is monitored before release. The turbine room roof ventilator fans provide

additional exhaust capacity in the summer season (see Figures 10.12-1, 10.12-7, and 10.12-8).

Per plant unit, the Turbine Building exhaust fan and the nine turbine room roof-ventilator exhaust fans are collectively capable of exhausting a maximum of approximately 269,000 CFM of building air to the outdoors. The Turbine Building discharge dampers and Reactor Building exhaust fans are pneumatically activated. These dampers are located on the Reactor Building roof. The turbine-spaces supply fan, the mechanical-spaces supply fan, the two electrical-spaces supply fan, and the five turbine room supply fans are collectively capable of supplying a maximum of approximately 255,000 CFM of outdoor air to the building. The building can thus be maintained at a slight negative pressure relative to the outdoors to minimize possible exfiltration of contaminated air.

During cold weather, the two-speed, building exhaust fan may be operated at half-speed and the roof-ventilator exhaust fans turned off, by groups, to maintain any of various building exhaust air flow rates to a minimum of 62,500 CFM. The two-speed, mechanical-spaces supply fan may be operated at half-speed; the two-speed supply fans may be reduced in speed or turned off; and the single-speed supply fans may be either operated or turned off, in various operating combinations, to maintain slightly less air-supply flow rate to the building than is exhausted.

10.12.5.3 <u>Control Building</u>

<u>Toxic Gas Protection</u>

The evaluation of control room habitability included consideration of possible hazards created by the accidental release of potentially toxic chemicals. The evaluation considered chemicals stored both onsite and offsite within a 5-mile radius of BFN. Possible shipments of toxic chemicals by pipeline, barge, rail, or road routes within a 5-mile radius were also considered. Methods of analysis used were those outlined in NRC Regulatory Guide 1.78, "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release."

All chemicals stored onsite which are considered to be potentially hazardous to the control room personnel were analyzed utilizing the approach outlined in NRC Regulatory Guide 1.78. The analysis (1) confirmed that these chemicals have no effect on the main control room habitability. Additional chemicals are stored onsite, however, it was assumed that these chemicals do not constitute a hazard to control room personnel due to the fact that they are stored in small quantities, are solids, are liquids with a very low vapor pressure at ambient temperatures, or the operators would have sufficient time to don protective equipment in the event of a release.

There are no industrial or military facilities located within a 5-mile radius of BFNP where stored chemicals could cause a potential hazard to the plant[1].

The only rail line or road passing within a 5-mile radius of the plant is Alabama State Route 20, which at its closest point, is 4-1/2 miles from the plant.

These analyses[1] indicated that the worst-case accident would be from nearby barge traffic transporting benzene, ethyl benzene, toluene, vinyl acetate, acrylonitrile, and chlorine. Analyses were performed utilizing the approach outlined in NRC Regulatory Guide 1.78. Major assumptions included Pasquill stability Class G and adverse wind direction. Wind speed was chosen to maximize the 2 minute concentration at the control room air intakes. The analysis[1] indicated that, upon a release of these chemicals from a barge accident, the concentration in the control room would pose a potential threat to the control room operators. With the exception of chlorine, these chemicals can be detected by smell by the control room operators in sufficient time to don protective equipment without experiencing any physical impairment. Upon a chlorine release from a barge accident, the concentration in the control room would impact control room habitability. However, the probability of a barge accident which results in a chlorine concentration in the control room which exceeds the concentration limits in NRC Regulatory Guide 1.78 is less than 1.0E-6 events per year. At this probability value, the chlorine release can be excluded from consideration in the control room habitability analysis.

SECURITY RELATED INFORMATION
TEXT WITHHELD UNDER 10 CFR 2.390

Control Building HVAC

The Control Building Heating, Ventilating, and Air Conditioning Systems serve the three floors in the control bay and the six shutdown electrical board rooms in the Reactor Building immediately adjacent to, and normally entered from, the control bay. There are several separate subsystems serving these areas.

The Control Building air conditioning is divided into eight general areas. Four of these areas have separate air supply systems, each serving a room or group of rooms with cooling and heating thermostatically controlled. These areas are the Units 1 and 2 Control Room, Units 1 and 2 elevation 593, relay room, Unit 3 Control Room. The Unit 3 elevation 593 area is heated and cooled with a separate air supply system, but it is not thermostatically controlled. The air supply systems for the remaining three areas serve a group of rooms with only cooling. These areas are the Unit 1 Electric Board Rooms, Unit 2 Electric Board Rooms, and the Unit 3 Electric Board Rooms. Three of the eight control building air conditioning systems are shared, in that one system serves the Units 1 and 2 Main Control Room, the second serves the Units 1 and 2 elevation 593, and the third serves the Switchyard

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1 Calculation ND-Q0031-890038

Relay Room common to all three units. (For additional information refer Appendix F.) Each air supply system is equipped with two 100-percent capacity air handling units or air conditioning units (see Figures 10.12-2a and -2c).

The Control Room Emergency Ventilation System (CREVS) processes outside air needed to provide ventilation and pressurization for the Control Room Habitability Zone (CRHZ) during isolated conditions. When the CRHZ is isolated, a fixed amount of outside air is processed through a HEPA filter bank, air heater, charcoal adsorbers, and post filters. A seismically-qualified safety-related Control Room Emergency Ventilation System (CREVS), composed of two redundant trains, is provided (as shown in Figure 10.12-2b) in the Unit 2 control bay area. This system of filtered outside air aids in positive pressurization of the CRHZ with respect to the outdoors. Test facilities to conduct standard DOP and Freon leak tests are provided for this system. Carbon sample canisters are provided for Laboratory Carbon Sample Analysis.

The CREVS is started automatically by a primary containment isolation signal or high radiation signal, or it can be started manually at any time. The CREVS, once activated, continues to operate until shut down manually.

The control bay HVAC flow diagrams are shown in Figures 10.12-2a, 10.12-2b, and 10.12-2c. Cooling of the atmosphere in the Main Control Room is provided by a recirculation air system with refrigeration units. During normal operation, a small stream of makeup air drawn through NBS dust filters is used to maintain a slight positive pressure in the control room. Upon receipt of a primary containment isolation signal or high radiation signal, the normal control room pressurization and makeup network is automatically isolated from the CRHZ. This same signal automatically starts the operation of the CREVS. The trip setting for the Control Building intake duct radiation monitors is based on the Technical Specifications Section 3.3.7.1 allowable value of 270 cpm above background, which is a radiation level corresponding to about 10^{-5} mci/cc of Xenon-133 (about 1 mRem/hr). The nominal trip setpoints are determined to account for appropriate instrument errors (e.g., drift) and are specified in the setpoint calculations. The initial setpoint was based on manufactures empirical formulas.

Outside air for the CREVS is drawn from both of the main outside air intake ducts supplying ventilation tower 1 and ventilation tower 3. Outside air pulled from these two intakes passes through a HEPA filter bank located in ventilation tower 2.

The CREVS is activated by a primary containment isolation signal or high radiation signal from the Control Building intake duct radiation monitors, the same signals also initiates the isolation of the CRHZ. The two 100 percent redundant filter trains are safety-related and are powered from separate divisions of normal and emergency diesel power. Only one train operates following auto actuation with the other train on standby.

In each train, a Class 1E electric duct air heater is mounted upstream of charcoal adsorber filters to maintain the incoming air's relative humidity to below 70 percent for high charcoal adsorption efficiency. The CREVS is designed to process outside air post DBA for 30 days without danger of saturation.

The control room air handling units provide ventilation to the main control room area. Two 100-percent capacity air handling units are provided, each containing: heating and cooling coils, a humidifier, controls, and motor-operated dampers. The dampers isolate the air handling unit when on standby. The air handling cooling coils are equipped with vent and drain valves. Room return air is proportionally mixed with fresh air by manual dampers and filtered by renewable media filter cells rated at 85-percent NBS.

Fresh air is mechanically supplied for makeup to air-conditioning systems, for ventilating system requirements, and for pressurizing the Control Building. Fresh air supply systems separately serve the Units 1 and 2 air-conditioned spaces except the Electric Board Rooms, the Unit 3 air-conditioned spaces except the electric board rooms and spreading rooms. Each of the air-conditioned spaces has two 100-percent capacity supply fans.

Each spreading room is ventilated by one 100-percent capacity fresh-air supply fan. Two 100-percent capacity exhaust fans serve both spreading rooms. The fresh air is filtered. The air flow is balanced with the exhaust flow exceeding the total supply flow in order to prevent a positive pressure in the spreading rooms in relation to the Control Bay Habitability Zone (CBHZ) thus precluding the possibility of unfiltered air inleakage into the CBHZ. Dampers exist in the ventilation exhaust lines from the spreading rooms. Manual provisions exist to restart one spreading room ventilation system independent of the other.

Two 100-percent capacity, air-cooled, water-chilling units, located on the Unit 1 and 2 Diesel Generator Building roof, provide essential cooling for the Units 1 and 2 control bay air-conditioning systems.

Two 100-percent capacity chilled water pumps for Unit 3 are each designed to circulate water through a water cooled water chilling unit. Chilled water is then circulated through a chilled water piping loop to each air-handling unit's cooling coil. The system is equipped with test wells for temperature monitoring.

Two 100-percent capacity hot water pumps are each designed to circulate water that is heated by a hot water generator located in the Unit 1 mechanical equipment room. The hot water is circulated through a hot water piping loop to various air-handling unit heating coils and reheat coils mounted in branch air supply ducts.

To prevent overheating of essential electrical equipment as a result of loss of cooling due to failure of both water chillers serving Units 1 and 2, or by the rupture of the chilled water loops, supplementary cooling is provided. Water-cooled condensing units are connected to direct-expansion cooling coils mounted in each air-conditioning system return air separate duct. Condenser cooling water is taken from the Emergency Equipment Cooling Water System.

The Unit 3 control bay ventilation and heating instruments are strategically located to sense a mixture of return and supply air for optimum system performance.

The rooms containing the Unit 1 shutdown boards and the rooms containing the Unit 2 shutdown boards, and the rooms containing the Unit 3 480V shutdown boards are each cooled by two 100-percent capacity air-conditioning units or air handling units located in each of the Reactor Buildings. The air distribution system and damper configuration allows each room to be independently isolated and cooled. Fire dampers allow each board room to be independently isolated when an excessively high temperature is detected. Smoke detectors in the return ducts of the Unit 2 board rooms alarm in the main control room.

The 4-kV electric boards for Unit 3 are located in rooms within the Unit 3 Diesel Generator Building. The Unit 3 electric board rooms are cooled by redundant air-conditioning units. Induced fresh air and the recirculated air are filtered by NBS-rated filters. Fire dampers allow each board room to be independently isolated when an excessively high temperature is detected.

The Unit 3 electric board room air conditioning units described above are seismically qualified and are powered from emergency electric power sources. Condenser cooling water is supplied by the EECW System.

The battery rooms, motor-generator set rooms for Unit 3 only, and Units 1 and 2 mechanical equipment room are ventilated for the removal of heat and dangerous fumes by two exhaust systems, one serving Units 1 and 2 and the other serving Unit 3. Each system contains two 100-percent capacity exhaust fans located on the control bay roof. A third exhaust fan for each system is provided within the control bay structure. Fire dampers are present in all ventilation ducts which penetrate the perimeter of the auxiliary battery rooms for all 3 units.

Cooling for the Switchyard Relay Room is provided by two safety related air handling units. Normally, Relay Room Air Handling Unit A provides cooling for the Relay Room. This unit contains a prefilter, heating and cooling coils, a humidifier, controls and a motor operated damper. Ventilation is provided for the Relay Room from the Unit 1 board room normal or emergency supply fan through air handling Unit A. Relay Room Air Handling Unit B automatically starts on high relay room temperature, which is annunciated as a common A/C alarm in the Unit 1 Main Control Room, and will continue to operate until it is manually shut off. Relay Room Air Handling Unit B, located in the Relay Room, is a local room cooler which contains a prefilter, cooling coil, controls and a backdraft damper at its discharge. During emergencies, if offsite power is available, the operating air handling unit will continue to run. Upon loss of offsite power, either Relay Room Air Handling Unit can be started locally within three hours to assure sufficient cooling.

The non-safety-related Process Computer Room, Units 1/2 Computer Room, Communications Battery Board Room, and Communications Room are cooled by redundant 100-percent capacity non-safety-related air-conditioning units located in the Turbine Building. Damper configuration allows each room to be independently isolated and cooled.

The Unit 3 Computer Room is cooled by the safety-related air supply system which serves the Unit 3 Auxiliary Instrument Room.

The Control Bay ventilation towers are located outside on the Control Bay roof and these towers are not fully protected from the effects of a tornado. Units 1 and 3 vent towers house safety-related equipment to ensure safe shutdown, prevent radioactive release to the environment, and support long term cooling. A probabilistic analysis documents that the occurrence and consequences of a tornado on the Units 1 and 3 vent towers are less than the NRC acceptance criteria of 10^{-7} per year. Therefore, protection of the Units 1 and 3 vent towers from the effects of a tornado is not required.

10.12.5.4 Radwaste Building

The Radioactive Waste Building ventilating system consists of two 50-percent capacity supply fans which supply a total of 28,000 CFM of filtered air to central areas on the various floor levels. Two 50-percent capacity exhaust fans with a total rated capacity of 30,000 CFM exhaust air from individual spaces in the Radioactive Waste Building through roughing and HEPA filters. Exhaust air is routed through the Reactor Building by a separate exhaust duct before being discharged on the Reactor Building roof. The filters are arranged in two separate plenums, each of which can handle 50-percent of the capacity and each of which can be isolated for servicing. When one plenum is isolated, one supply fan and one exhaust fan are stopped and the system operates on 50-percent capacity. Vents from tanks, sumps, and hoods are routed to the exhaust ducts. All exhaust grilles are equipped with opposed blade dampers for balancing, and doors used for air intakes to spaces are equipped with backdraft dampers (see Figure 10.12-4).

Generally, the Radioactive Waste Building is heated by tempering the building air supply with hot water coils located in the fan room. Sufficient heat is furnished to preheat the supply air for equipment protection and reasonable personnel comfort. The power stores, Elevation 580.0, which have a separate ventilation system are heated by suspended electric heaters. Additional electric heat is furnished for personnel comfort in the following areas: clothes change, ventilating equipment, and fan rooms.

Comfort air-conditioning is provided for the radio-chemical laboratory and the radioactive waste control room. The system consists of an air-handling unit, water-cooled condenser, and electric duct heaters located in the Service Building. A 100-percent fresh air supply with no return is used to prevent possible buildup of contaminants. The air supplied is exhausted through the laboratory hoods each of which is connected to an exhaust fan and HEPA filters. The fans and filters are located in the fan room at Elevation 580.0 and discharge to the roof of the Radioactive Waste Building.

10.12.5.5 Diesel Generator Building

The Diesel Generator Building (DGB) Heating, Ventilation and Air Conditioning (HVAC) system is designed to maintain the required environmental conditions for safety related equipment located in the Units 1, 2, and 3 DGB.

The protective safety functions are accomplished through the various ventilation methods described below. Ventilation cooling and fume removal from each of the

eight (8) (DG) rooms is provided by one of two redundant exhaust fans (A & B) with associated room inlet and outlet, and fan discharge motor operated dampers. These fans discharge into a common exhaust plenum, which is open to the atmosphere.

Although not classified as protective safety related equipment, each DG room contains a separate battery vent hood exhaust fan. This fan is normally operating to prevent hydrogen gas buildup during battery charging operation.

Each Unit 1 and 2 Diesel Auxiliary Board room (480-V A and B) have roof mounted exhaust fans and air intakes. Each exhaust fan and each air intake have motor operated dampers which close when fan is off.

The Unit 3 480-V Diesel Auxiliary Board rooms (3EA and 3EB) are ventilated similar to Unit 1 and 2 with additional exhaust duct connections from room 3EA to an adjacent toilet room, and additional ventilation exhaust ducting from the pipe and electrical tunnel to the 3EB fan.

Unit 1 and 2 DGB Emergency Transformer, located in the pipe and electrical tunnel area, is cooled by a separate roof mounted exhaust fan with motor operated damper.

The Unit 3 250-V Battery Room 3EB, is ventilated and maintained at a negative pressure with two redundant roof mounted exhaust fans, each with associated back-draft dampers.

The Central Diesel Information Center, which is located in Units 1 and 2 DGB, is cooled by a roof mounted exhaust fan with an associated motor operated damper.

The Electrical Access and Miscellaneous Equipment Room, which is located in Unit 3 DGB, is cooled by a roof mounted exhaust fan with an associated motor operated damper.

For the four 4160-V Shutdown Board Rooms of Unit 3 (3EA, 3EB, 3EC, and 3ED) and the Bus Tie Board Room air conditioning is provided. In addition, outdoor air is available for pressurization and exhaust.

10.12.6 Safety Evaluation

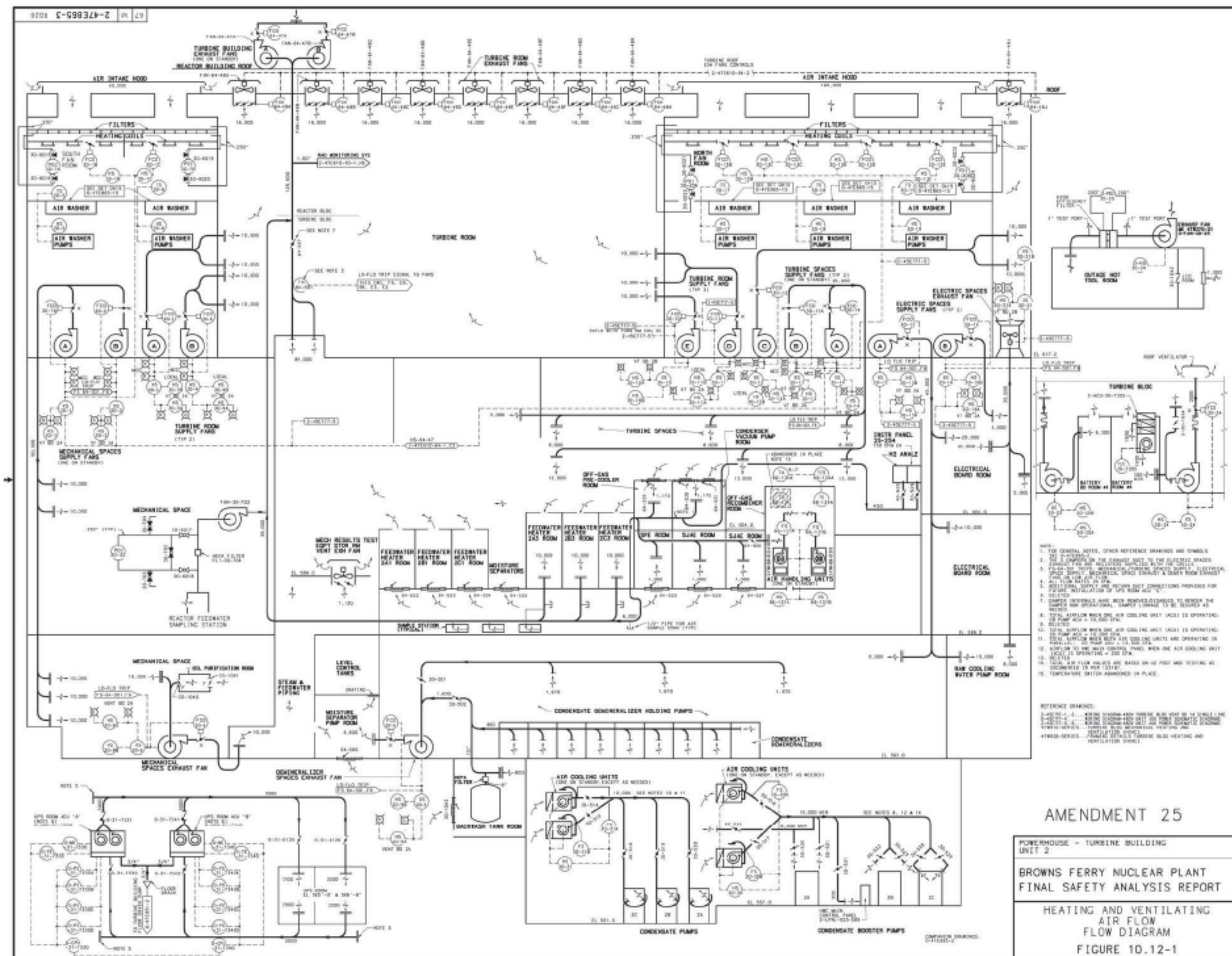
The air-conditioning and ventilation systems of the Control Building, Diesel Generator Building, and shutdown board rooms shown on Figures 10.12-2a, -2b, -2c, 10.12-5, and 10.12-6 of the FSAR, are provided to condition the environment so that safety-related controls and electrical equipment will remain operable at all times. Sufficient equipment redundancy and alternate power sources ensure that tolerable limits are maintained and, therefore, extreme environmental conditions are not part of the design bases. These systems are discussed in Subsection 10.12.5.3 of the FSAR. (The Unit 3 4-kV shutdown board room air conditioning systems do not have single failure proof power sources. If required, manual actions ensure area temperatures remain within limits.) In general, two water chiller units or air conditioning units are provided, whereas only one is required; and at least two

blowers are provided with one being a standby. Alternate electrical boards feed power from the offsite or emergency supplies to the redundant units. The dampers are motor- or air-operated; however, they can be easily and quickly positioned manually in the event of a operator malfunction. Thus, no criteria or design analyses are needed to cope with extreme environmental conditions associated with a loss of all air-conditioning and ventilation systems servicing the control rooms and equipment rooms.

Environmental control of the control bay is maintained for personnel occupancy and for continuous operation of plant equipment during any type of accident and throughout the life of the plant. All cooling facilities have redundant or backup systems. All essential functions are provided by two or more separate and redundant systems or subsystems. Power supplies for essential equipment are taken from separate shutdown boards and routed through separate boards and trays. Cooling water for essential equipment is taken from the Emergency Equipment Cooling Water System.

10.12.7 Inspection and Testing

The control bay and shutdown board room air-conditioning and ventilating systems are in continuous operation and are accessible for periodic inspection. Essential electrical components, switchovers, and starting controls are tested initially and periodically.



AMENDMENT 25

POWERHOUSE - TURBINE BUILDING
UNIT 2

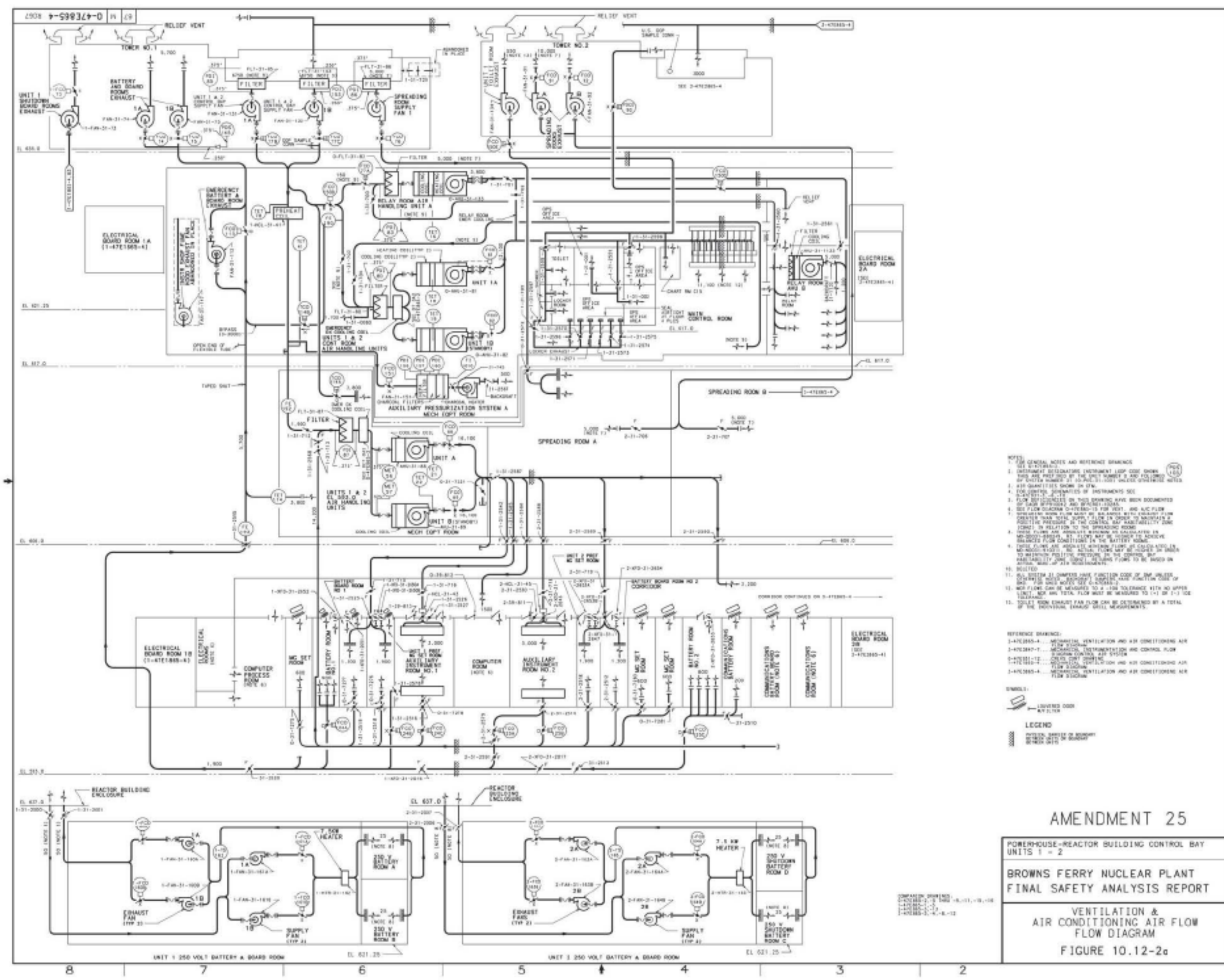
BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

HEATING AND VENTILATING
AIR FLOW
FLOW DIAGRAM

FIGURE 10.12-1

Figure 10.12-2

Deleted by Amendment 9.

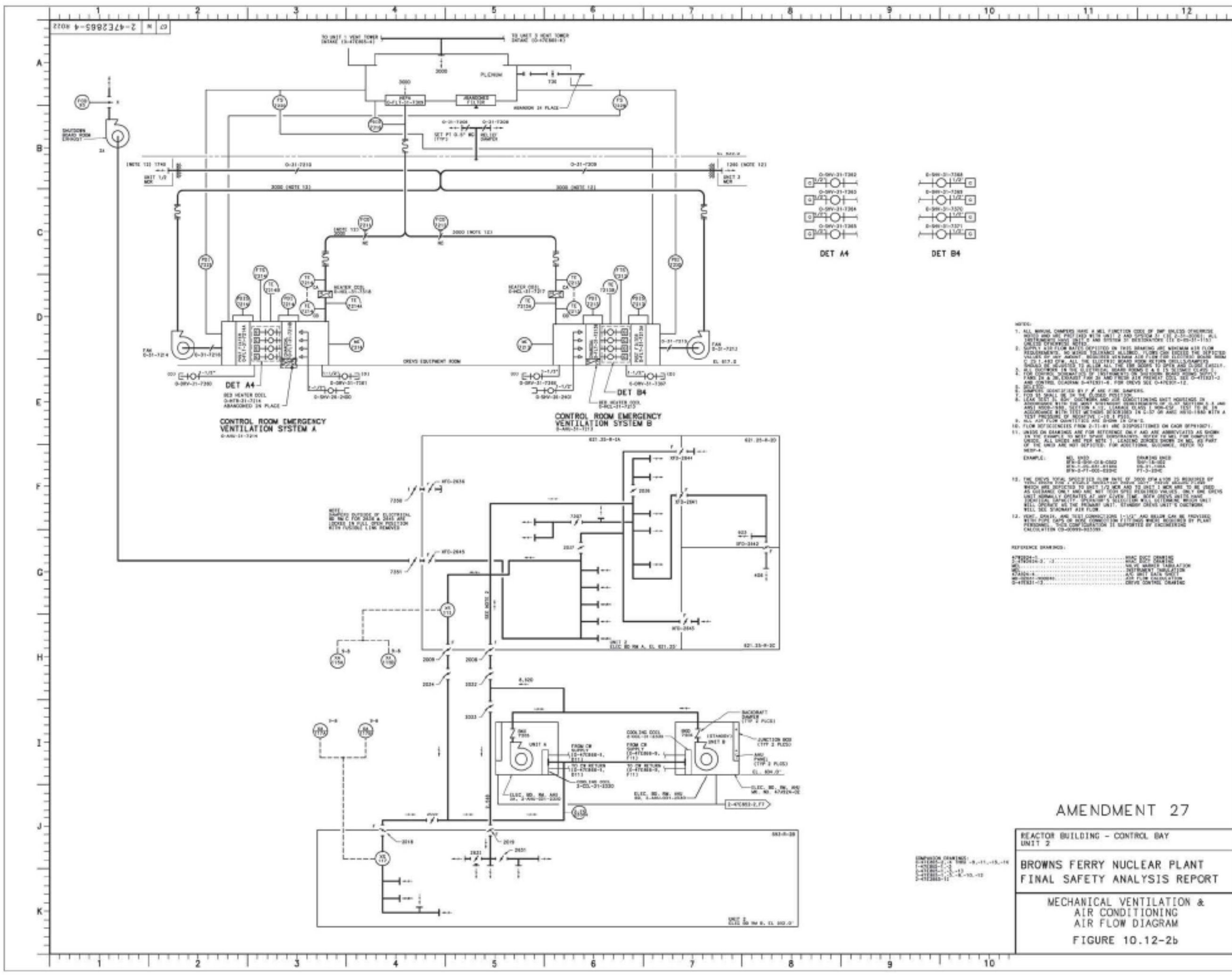


AMENDMENT 25

POWERHOUSE-REACTOR BUILDING CONTROL BAY
UNITS 1 - 2

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

VENTILATION &
AIR CONDITIONING AIR FLOW
FLOW DIAGRAM
FIGURE 10.12-2a

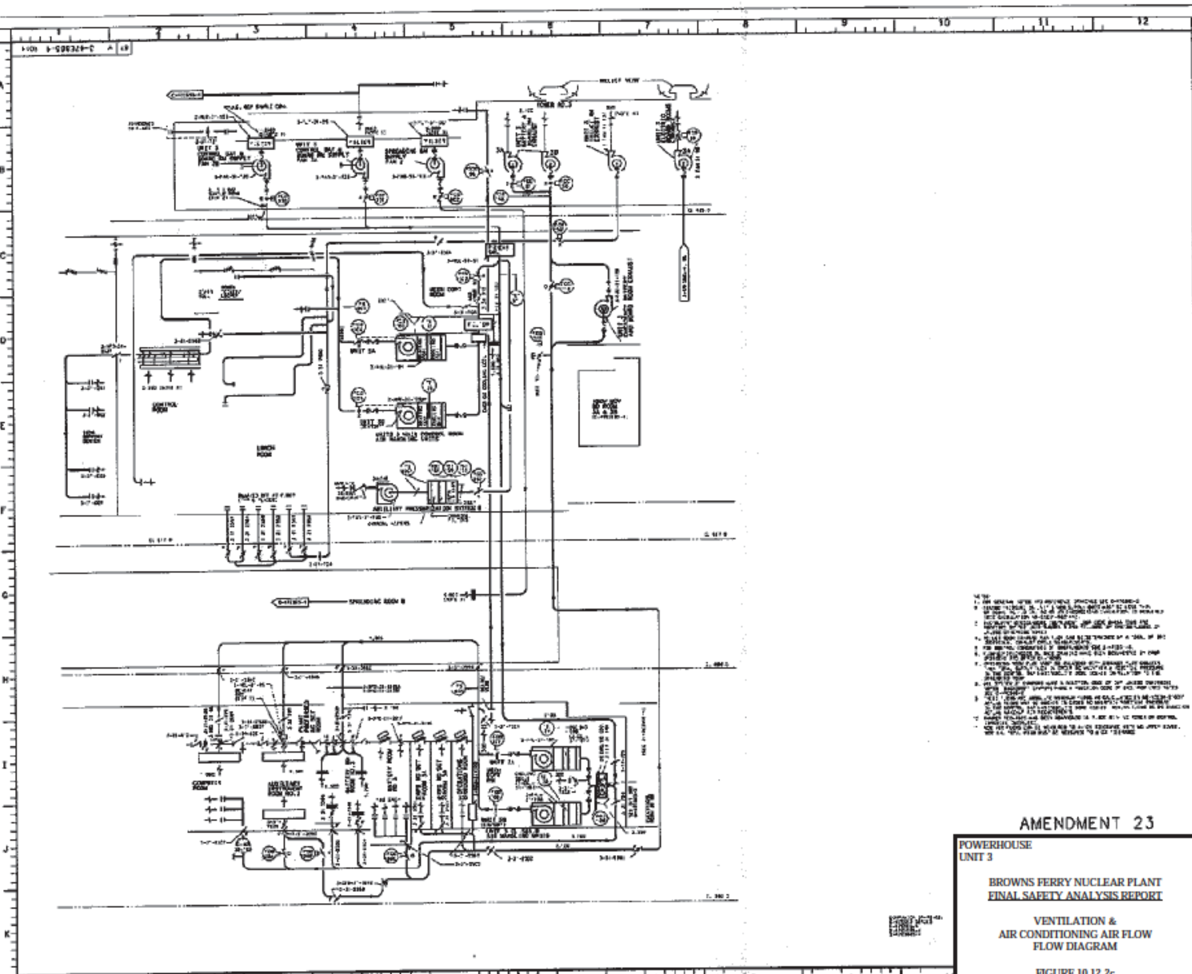


REACTOR BUILDING - CONTROL BAY
UNIT 2

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

MECHANICAL VENTILATION &
AIR CONDITIONING
AIR FLOW DIAGRAM

FIGURE 10.12-2b



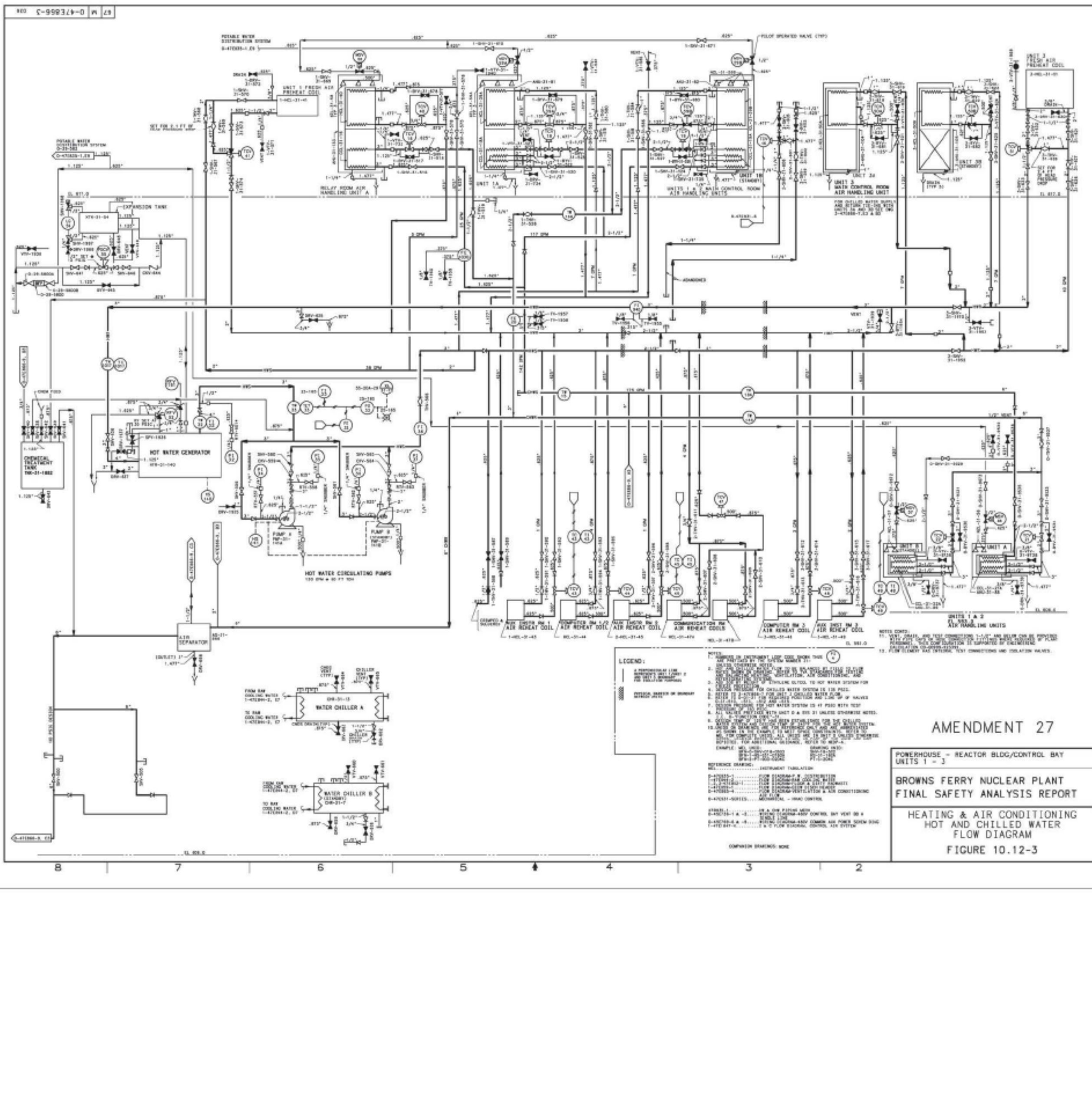
AMENDMENT 23

POWERHOUSE
UNIT 3

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

VENTILATION &
AIR CONDITIONING AIR FLOW
FLOW DIAGRAM

FIGURE 10 12 2c



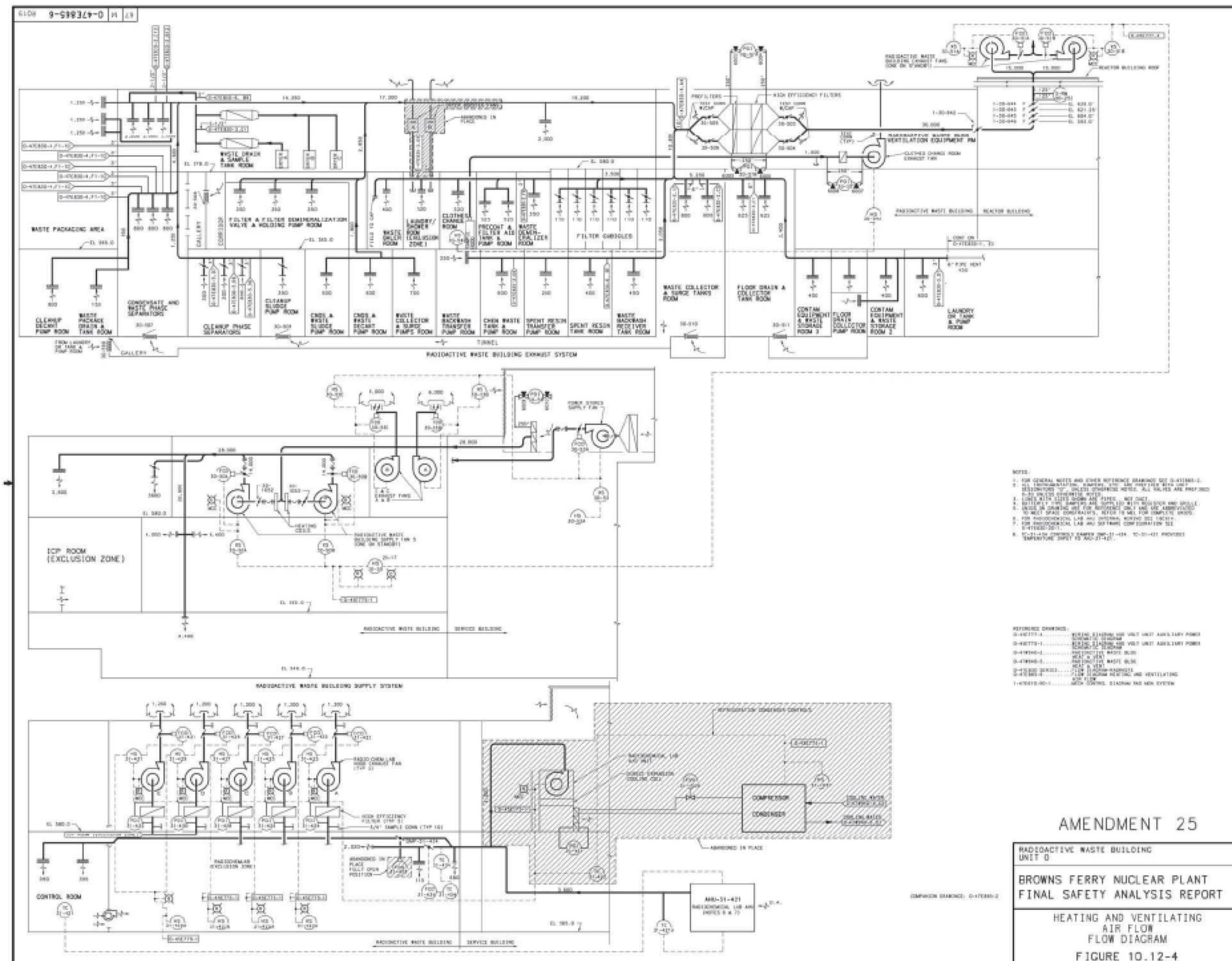
AMENDMENT 27

POWERHOUSE - REACTOR BLDG/CONTROL BAY
UNITS 1 - 3

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

HEATING & AIR CONDITIONING
HOT AND CHILLED WATER
FLOW DIAGRAM

FIGURE 10.12-3



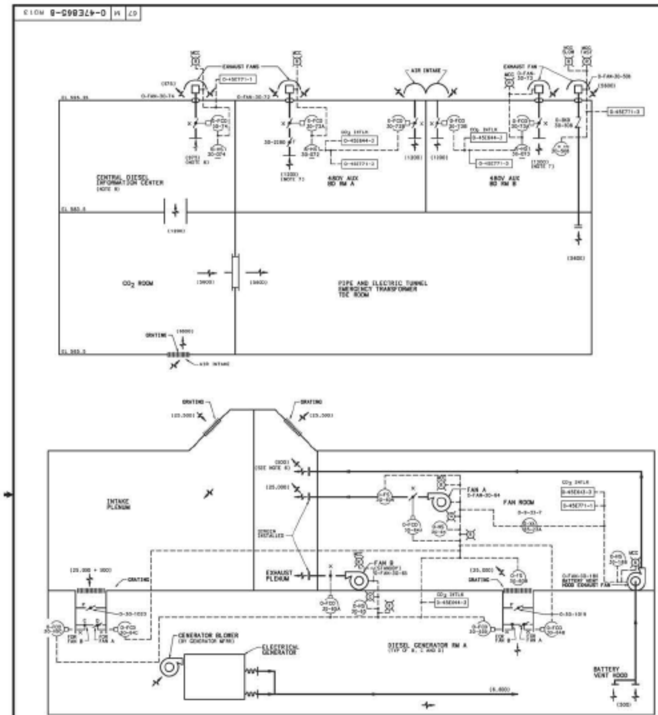
AMENDMENT 25

RADIOACTIVE WASTE BUILDING
UNIT 0

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

HEATING AND VENTILATING
AIR FLOW
FLOW DIAGRAM

FIGURE 10.12-4



DG ROOM	DG COOLING EXHAUST FAN A				DG COOLING EXHAUST FAN B				BATTERY VENT EXHAUST FAN (HS)	BATTERY VENT EXHAUST FAN	FAN ROOM INLET DAMPERS			DG ROOM INLET DAMPERS		
	(FAN)	(FS)	(FCO)	(HS)	(FAN)	(FS)	(FCO)	(HS)			DAMPER 0-30-	OPERATOR (FCO)	OPERATOR (FCO)	DAMPER 0-30-	(FCO)	(FCO)
A	FAN-84	[illegible]	[illegible]	[illegible]	FAN-85	[illegible]	[illegible]	[illegible]	[illegible]	FAN-186	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
B	FAN-86	[illegible]	[illegible]	[illegible]	FAN-87	[illegible]	[illegible]	[illegible]	[illegible]	FAN-187	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
C	FAN-88	[illegible]	[illegible]	[illegible]	FAN-89	[illegible]	[illegible]	[illegible]	[illegible]	FAN-188	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
D	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	FAN-189	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]

NOTE: ALL TAG NUMBERS PROCEEDED BY 0-XX-30

NOTES:

1. FOR GENERAL NOTES AND REFERENCE DRAWINGS SEE 0-47E865-2.
2. NUMBERS IN PARENTHESIS INDICATE AIR FLOW RATES IN CUBIC FEET PER MINUTE.
3. INSTRUMENT NUMBERS ARE LISTED FOR DIESEL GENERATORS A THROUGH D RESPECTIVELY.
4. UNLESS OTHERWISE SHOWN FAN DAMPERS HAVE THE SAME ADDRESS AS THE FLOW CONTROL OPERATORS (FCO).
5. UNIDS ON DRAWING ARE FOR REFERENCE ONLY AND ARE ABBREVIATED TO MEET SPACE CONSTRAINTS. REFER TO MEL FOR COMPLETE UNIDS.
6. THE AIR FLOW RATE FOR THE DIESEL GENERATOR BATTERY EXHAUST FAN IS A
7. MINIMUM REQUIRED FLOW.
8. REQUIRED FLOW FOR THE DG INFORMATION CENTER FAN IS 975 CFM
9. OPERABILITY NOTE: IN ORDER TO BE MAINTAINED AT OR BELOW 110°F DURING NORMAL OPERATION, ADDITIONAL COOLING WILL BE REQUIRED WHEN THE OUTSIDE TEMPERATURE REACHES 85°F. LIKEWISE, TO MAINTAIN THE ROOM AT OR BELOW 130°F ADDITIONAL COOLING WILL BE REQUIRED WHEN THE OUTSIDE TEMPERATURE REACHES 95°F. TEMPORARY FAN(S), RATED AT A MAXIMUM OF 2,500 CFM, WOULD BE PLACED IN THE ELECTRICAL ACCESS ROOM TO DELIVER THE FLOW TO THE DG 1 INTAKE AIR PLENUM.

REFERENCE DRAWINGS:

COMPANION DRAWINGS:
0-47E865-2

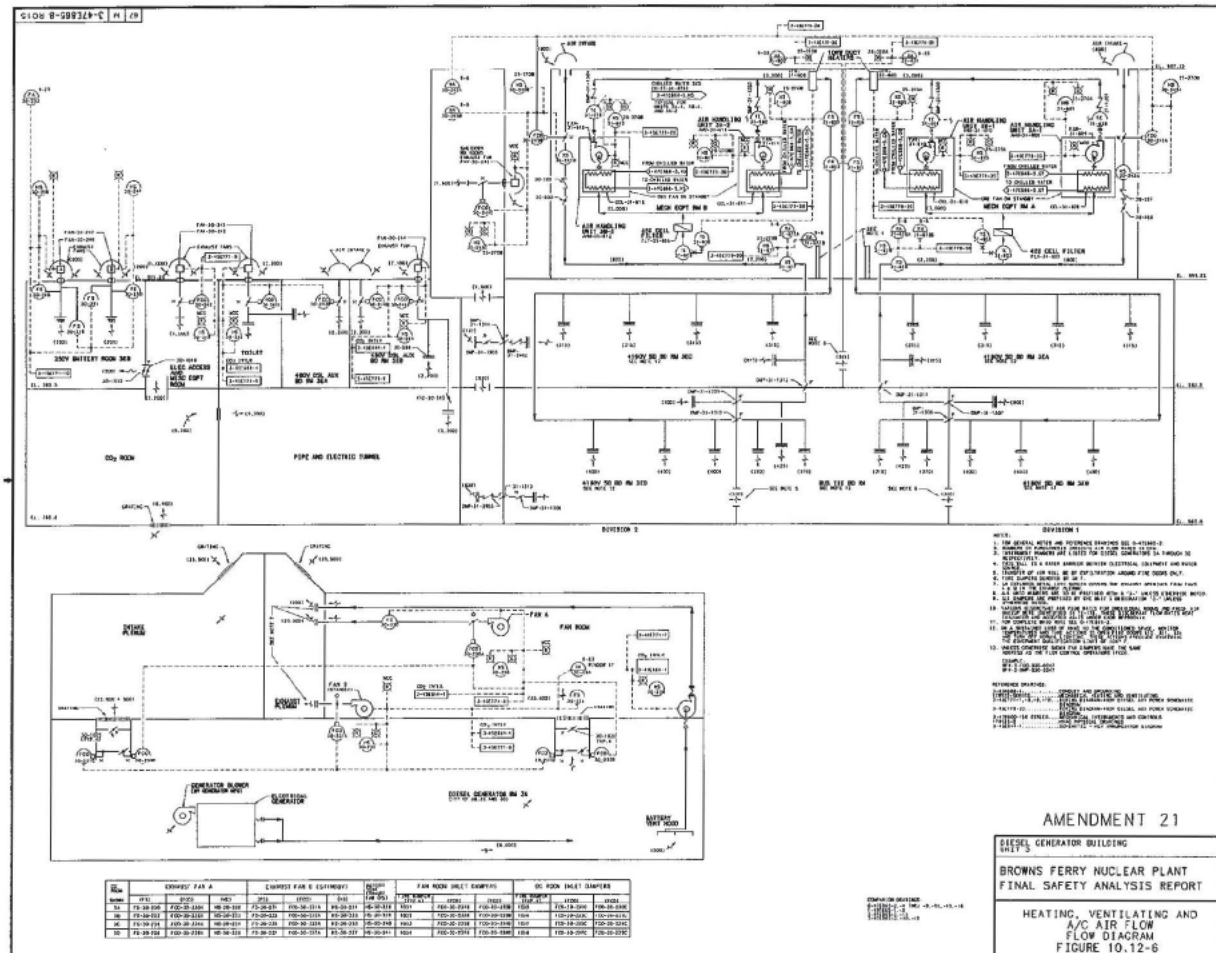
AMENDMENT 27

DIESEL GENERATOR BUILDING
UNIT 1 & 2

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

HEATING, VENTILATING AND
A/C AIR FLOW
FLOW DIAGRAM

FIGURE 10.12-5

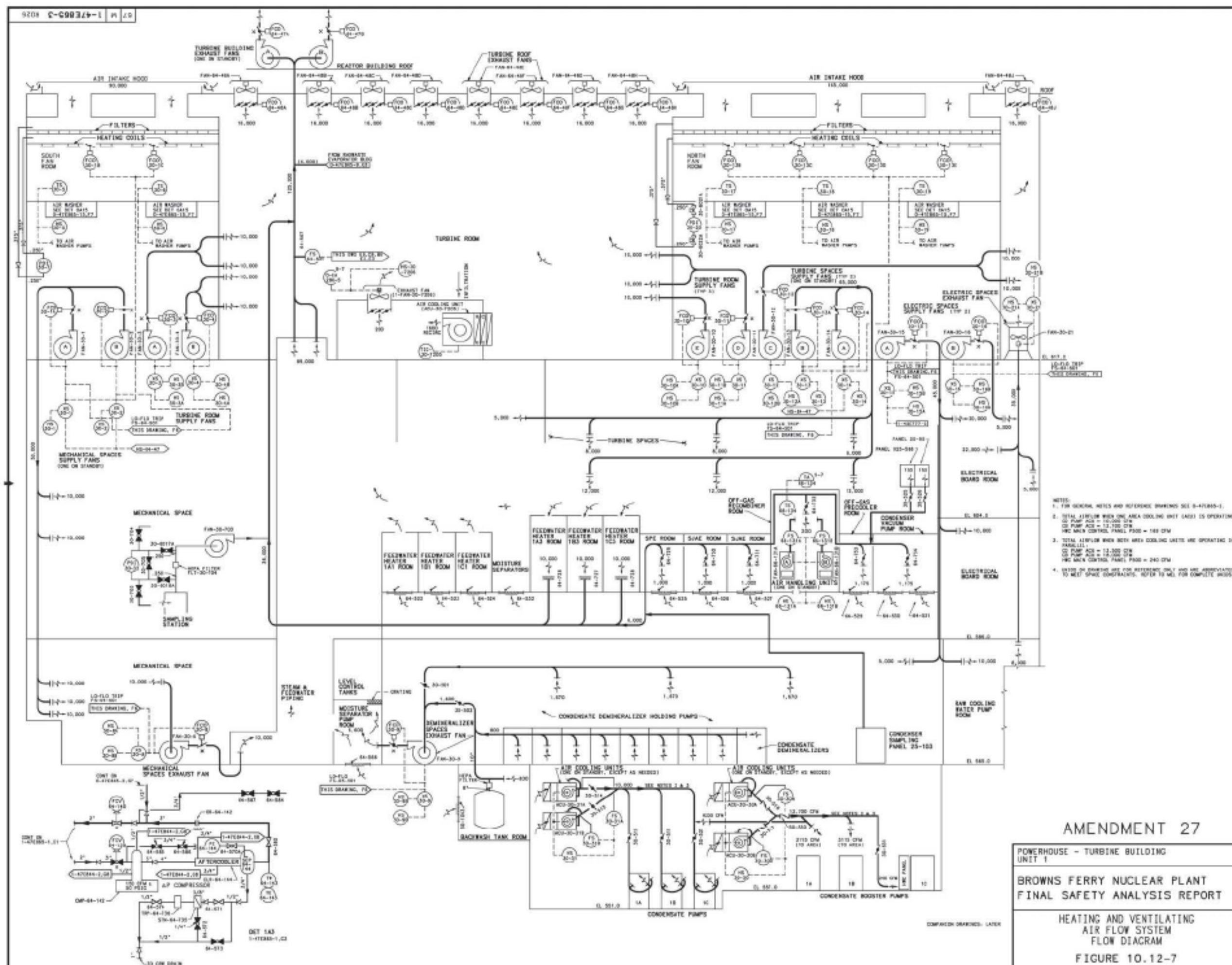


AMENDMENT 21

DIESEL GENERATOR BUILDING
UNIT 3

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

HEATING, VENTILATING AND
A/C AIR FLOW
FLOW DIAGRAM
FIGURE 10.12-6



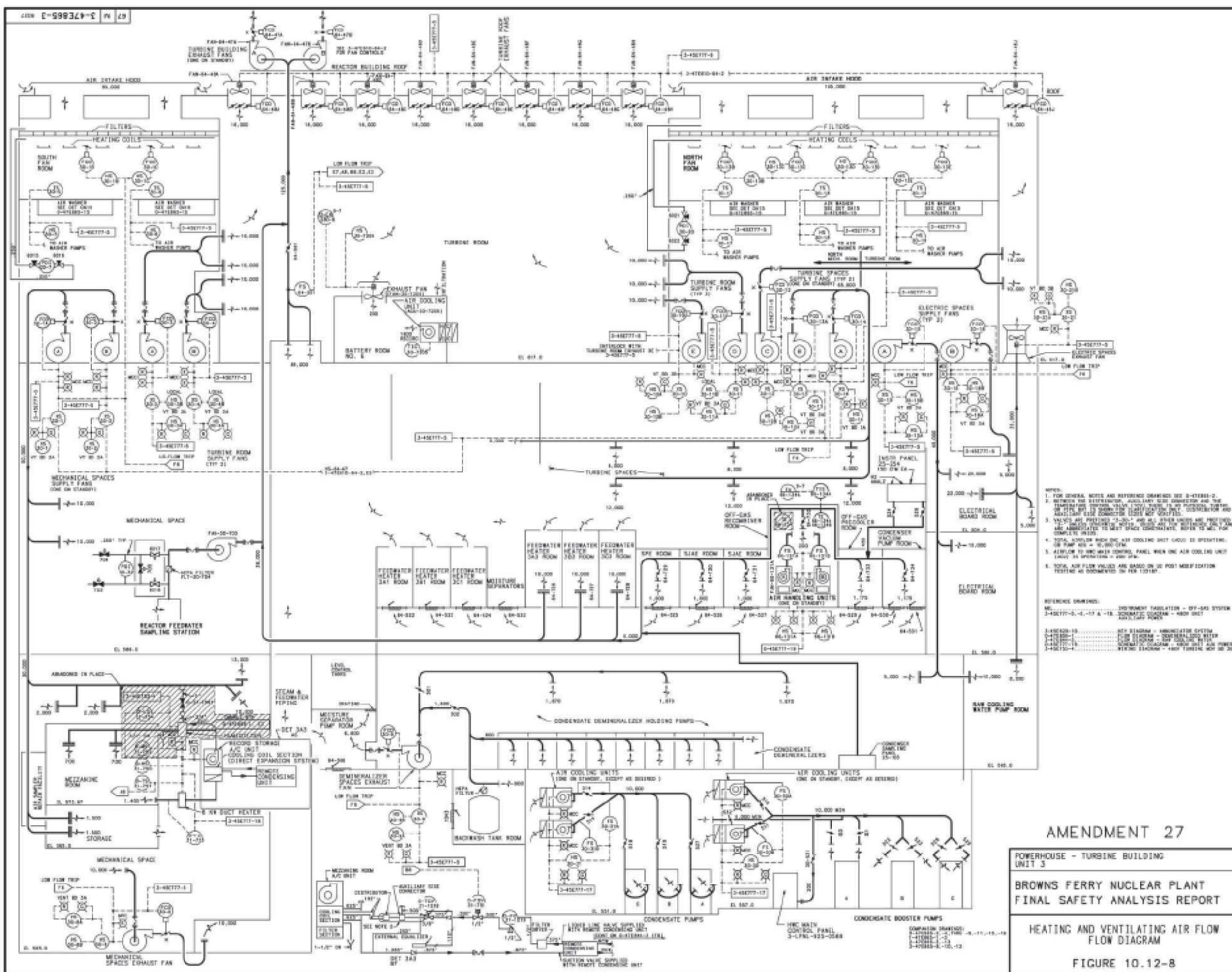
AMENDMENT 27

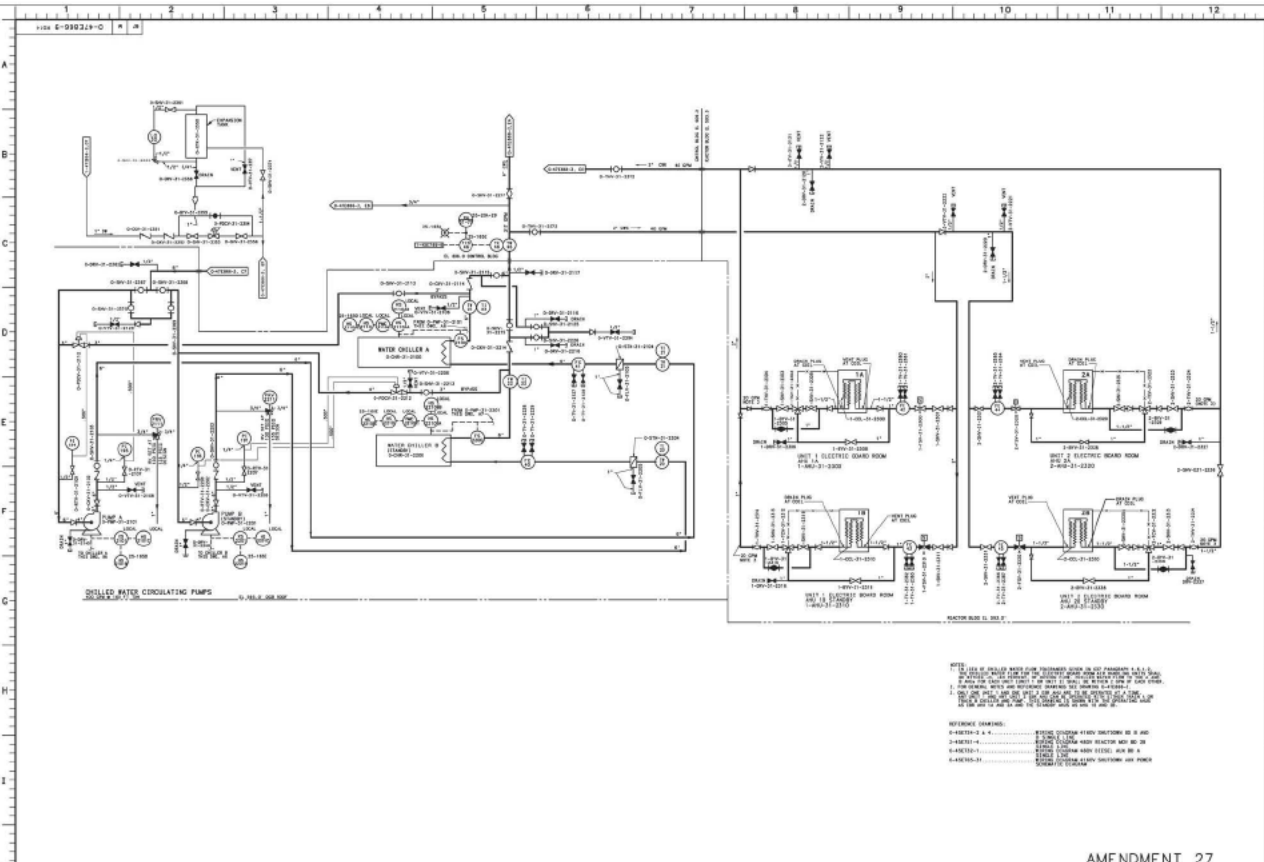
POWERHOUSE - TURBINE BUILDING
UNIT 1

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

HEATING AND VENTILATING
AIR FLOW SYSTEM
FLOW DIAGRAM

FIGURE 10.12-7


AMENDMENT 27
POWERHOUSE - TURBINE BUILDING
UNIT 3
BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT
HEATING AND VENTILATING AIR FLOW
FLOW DIAGRAM
FIGURE 10.12-8



AMENDMENT 27

POWERHOUSE
UNIT 3

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

FLOW DIAGRAM, CHILLED WATER
CIRCULATING PUMPS

FIGURE 10.12-9

10.13 DEMINERALIZED WATER SYSTEM

10.13.1 Power Generation Objective

The objective of the Demineralized Water System is to provide a supply of high purity water which is free of radioactive material. This shared system provides high purity water to all three units and to the common plant facilities such as the radiochemical laboratory. All secondary containment penetrations are designed to prevent seismically induced failure from causing a breach of containment.

10.13.2 Power Generation Design Basis

The system shall be capable of supplying all normal plant requirements for demineralized water.

10.13.3 Description

Demineralized water is used for makeup to various systems including the Primary Coolant Systems, the Reactor Building Closed Cooling Water Systems, and the Standby Liquid Control Systems. The water is piped to the radiochemical and health physics laboratories, sampling sinks, and various points in the plant where radioactive decontamination work is performed. Demineralized water is used extensively during preoperational cleaning of the reactor and piping systems.

Makeup demineralized water required to support plant operations is supplied by a makeup demineralized water unit.

Demineralized water is pumped from the makeup demineralized water unit to a 375,000-gallon outdoor storage tank. The makeup demineralized water unit is operated as needed to keep this tank full. Water needed for makeup in the Primary Coolant System is pumped from the demineralized water storage tank to one or more of the three condensate storage tanks. Other requirements for demineralized water are supplied by a distribution system drawing from three 10,000-gallon head tanks located on the roof of the Reactor Building. Level controls on the head tanks actuate pumps which supply water from the demineralized water storage tank to the head tanks.

The Demineralized Water System is designed to prevent the intrusion of water from other sources. Tanks containing demineralized water are of aluminum, pumps are of stainless steel, and valves and piping are either aluminum, stainless steel, or other non-corrosive material. These measures assure that purity of the water is maintained.

Under loss-of-power conditions, the system serves no essential functions. Primary system makeup requirements are met from the condensate storage tanks.

10.13.4 Inspection and Testing

The Demineralized Water System requires no inspection other than normal maintenance. Operation of the system to meet construction and preoperational cleaning needs satisfies testing requirements.

DEMINERALIZED WATER STORAGE TANKS WATER SPECIFICATIONS

Specific Conductivity .... ≤ 1.0 μhos/cm

Chloride .... ≤ 50 ppb

Silica .... ≤ 50 ppb

Filterable Iron.. ≤ 50 ppb

Figure 10.13-1 Sheets 1 and 2

(Deleted by Amendment 17)

10.14 CONTROL AND SERVICE AIR SYSTEMS

10.14.1 Power Generation Objective

1. To provide oil-free, control air, dried to a low dewpoint and free of foreign materials, to all pneumatically-operated instruments and controls and final operators, such as control valves, throughout the entire plant and yard.

2. To provide service air to hose connections throughout the plant and yard, and to miscellaneous equipment in the Standby Liquid Control System, Amertap Condenser Tube Cleaning System, Condensate Demineralizer Air Surge System, and the Radwaste System.

10.14.2 Power Generation Design Basis

1. The system shall be capable of supplying all normal plant requirements for control and service air.

2. Loss of control air pressure in any single unit shall not result in tripping of any other unit.

3. Control air shall be filtered and dried. Service air does not require special moisture removal (except moisture separators, traps, and drains) beyond the aftercooler.

10.14.3 Safety Design Basis

Accumulators shall be provided in the containment drywell to assure the Automatic Depressurization System main steam relief valves will be held open, and the inboard main steam isolation valves may be closed following control air failure (see Subsections 4.4 and 4.6). Redundancy for these ADS main steam relief valves is achieved by operation cables that are routed along different paths. Accumulators shall also be provided in the steam and feedwater valve room to assure the outboard main steam isolation valves may be closed.

10.14.4 Description

10.14.4.1 Control and Service Air Systems

Plant control air consists of four 524-SCFM (nominal), 125-psig, and one 1445-SCFM, 120-psig, air compressors, each designed for continuous operation, are connected to a common discharge header which supplies three 266-ft^3 control air receivers. Plant service air has one 910-SCFM (nominal) and one 642-SCFM (nominal), 100-psig air compressors, each designed for continuous operation, are connected to a common discharge header. The 910-SCFM and 642-SCFM

compressors supply one 266-ft^3 and one 48-ft^3 service air receivers. The air receivers are provided with moisture traps.

Service Air for the Radwaste Building is supplied by a separate system. This system consists of three skid mounted compressors with integral air receivers, located in the Radwaste Building. Two of the compressors are 45 CFM displacement type units, with 10.7 ft^3 receivers. The third compressor is a 113 CFM displacement type unit, with a 16 ft^3 receiver. The operation of the compressors is staged to keep the Radwaste Building Service Air System pressure between 95 and 125 psig. The design pressure and temperature for the system is 150 psig and 150^oF, respectively. Each receiver is provided with a moisture trap.

Control air compressors A-D, and service air compressors E and F are equipped with intercoolers between stages. The second stage discharge of the A-D compressors is connected to the horizontal, two-stage, water-cooled aftercooler of shell and tube design with moisture separator, which, in turn, is connected to the associated discharge header. The final stage discharge of the E compressor is connected to a horizontal pipeline-type shell and tube design water-cooled aftercooler with a moisture separator, which, in turn, is connected to the associated discharge header. The final stage of the F compressor is connected to an internal shell and tube water cooled after cooler with moisture separator which is connected to the discharge header. Control air compressor G and service air compressor F are electric motor driven, two stage, centrifugal package units; and service air compressor E is a three stage centrifugal package unit. Integral to each compressor are the oil sump, inter/after coolers, oil coolers, and control systems.

The Service Air System provides backup control air through a check valve and a backpressure control valve which opens if control air pressure drops below 85 psig. Thus, the Control and Service Air Systems are normally separate, with the service air acting as a backup to control air. This backpressure control valve can be manually operated from the Main Control Room. Service air is piped from the receiver to conveniently located service outlets throughout the plant (see Figures 10.14-2a and 10.14-2b).

The discharge header from the control air receiver supplies four air dryers, one in each unit plus one standby in Unit 1. The dryers are fully automatic with a timed regeneration cycle based on air dryer size; regeneration purge rate maintains the required dewpoint. There is one dryer for each unit, with one additional standby air dryer which can provide supplemental air to any unit. Each discharge from the Unit 1, Unit 2, Unit 3, and the standby dryers is routed through a cartridge-type filter. Each unit dryer filter station discharges into a 4-inch header which runs the length of the unit. The standby dryer filter station discharges through separate lines to each unit header, with check valves arranged to prevent loss of air pressure in one unit from affecting another unit.

From these unit headers, the control air is routed through 1-1/2-inch branch headers to the various locations in the unit. All of these branch headers are provided with shutoff valves to facilitate additional future control air requirements without requiring complete system shutdown. In addition, the 4-inch control air header for each unit is connected to the adjacent unit through manual valves and an air operated stop valve (between Units 2 and 3 only) which are normally open. The air operated stop valve closes upon loss of air from either Unit 2 or 3 and opens only when air pressure is restored.

During emergencies, the accumulators installed in the steam and feedwater valve room provide sufficient air for closure of the outboard main steam isolation valves. Since air-handling equipment is located in the Turbine Building (a Class II structure), the compressors and air receivers could conceivably be lost during an earthquake. Therefore, for Units 2 and 3, the headers routed to the Reactor Building are provided with manual valves and valves operated from the Main Control Room to facilitate system isolation. For Unit 1, the control air headers routed to the Unit 1 Reactor Building are provided with locally operated manual isolation valves in the Turbine Building. Check valves located inside the Reactor Building are credited for secondary containment isolation upon failure of the control air system following a seismic event, or the lines have been analyzed to maintain the secondary containment inleakage rate less than the SGTS capacity when the building is subjected to an internal negative pressure of 0.25 inch of water (see subsection 5.3.3.5).

An emergency control air compressor is located in the Unit 1 Control Bay to supply control air for the Unit 1 - Unit 2 Control Room Air-Conditioning System (see Figure 10.14-3). The receiver for this compressor rides on the air supply line leading into the control room. Unit 3 has a similar piping arrangement to provide this emergency air (using the same compressor).

Service air is provided throughout the entire plant.

Two control air compressor motors are powered from the 480-V common station service boards, two are powered from the 480-V shutdown boards, and one is powered from the 4-kV shutdown boards. The two service air compressor motors are powered from the common station service board. Compressors must be manually restarted when shutdown boards are supplied from the diesel generators.

10.14.4.2 Drywell Control Air System

Each reactor unit has a Drywell Control Air (DCA) System that provides control air for the equipment inside the drywell. A nitrogen supplied system is used that has DCA receiver tanks supplied dry, oil free nitrogen at a regulated pressure from the Containment Inerting system (see subsection 5.2.3.8) nitrogen makeup header in the Reactor Building. Pressure regulators are sized to limit flow to prevent over-

pressurizing primary containment should the DCA header pressure boundary fail inside the drywell during an accident. This nitrogen supply will not cause dilution of an inerted drywell. For the volumes of the drywells, pressure increases in the drywell due to operation of pneumatic equipment inside the drywell are very small. Drywell pressure may actually decrease depending on the amount of containment leakage, nitrogen makeup usage, and environmental conditions. Primary containment ventilation, purging and nitrogen makeup are monitored and controlled from the control room. The DCA receivers are sized to provide sufficient control air to assure operation of the drywell equipment in the event of an interruption of the nitrogen source. For each unit on the DCA supply lines upstream and downstream of the drywell penetration, two check valves exist which function as isolation valves. Also, a test connection and two normally open manual valves are present on the supply line. The manual valves are useful in providing backup to the check valves for positive containment isolation. All of this DCA equipment is located in the Reactor Building to protect it from natural phenomena.

The flow path for the DCA System is shown in Figures 10.14-4 sheets 1, 2, 3, 4, 5, and 6. Containment Inerting system nitrogen is continuously supplied at 100 psig via the DCA receiver tanks to the drywell control air headers.

In the event of a Beyond Design Basis External Event (BDBEE) to meet diverse and flexible coping strategies (FLEX) requirements, the Unit 1 DCA system may be supplied pressurized nitrogen through the existing permanent test connection isolation valve.

In the event of a Beyond Design Basis External Event (BDBEE) to meet diverse and flexible coping strategies (FLEX) requirements, the Unit 2 DCA system may not be supplied pressurized nitrogen through the existing permanent test connection isolation valve.

In the event of a Beyond Design Basis External Event (BDBEE) to meet diverse and flexible coping strategies (FLEX) requirements, the Unit 3 DCA system may be supplied pressurized nitrogen through the existing permanent test connection isolation valve.

The line through drywell penetration X-48 is used only for connecting a temporary air compressor to pressurize containment during performance of the Appendix J containment Integrated Leak Rate Test (ILRT). This line is normally closed with a 3-inch blind, double o-ring flange. The double o-ring flange is equipped with a test connection for Appendix J leak testing.

The DCA air receivers discharge into a common header which in turn supplies compressed air through two penetrations into the drywell and provides compressed air to each of the DCA headers. Each DCA header section, inside the drywell, supplies pneumatic pressure to one-half of the DCA users including three Automatic

Depressurization System (ADS) main steam relief valves, two inboard main steam isolation valves, and either three or four of the remaining main steam relief valves (MSRVs). Because of the control air requirements of the ADS main steam relief valves and inboard main steam isolation valves, accumulators are installed in the drywell to assure sufficient air in emergencies. Each DCA supply line has a check valve inboard and outboard of the drywell. These valves serve as primary containment isolation valves. A test connection and block valve are installed on both sides of the penetration for Appendix J leak testing. An emergency source of nitrogen is furnished at each penetration through a normally closed valve from the Containment Atmospheric Dilution (CAD) system with each CAD train supplying one of the DCA header sections.

Normally, the DCA System furnishes control air for the drywell equipment and the plant Control Air System provides control air for the outboard main steam isolation valves. However, provisions, with closed isolation valves and check valves, are made to use the plant Control Air System to supply the drywell control air and to use the DCA System to supply the outboard main steam isolation valves if the need arises.

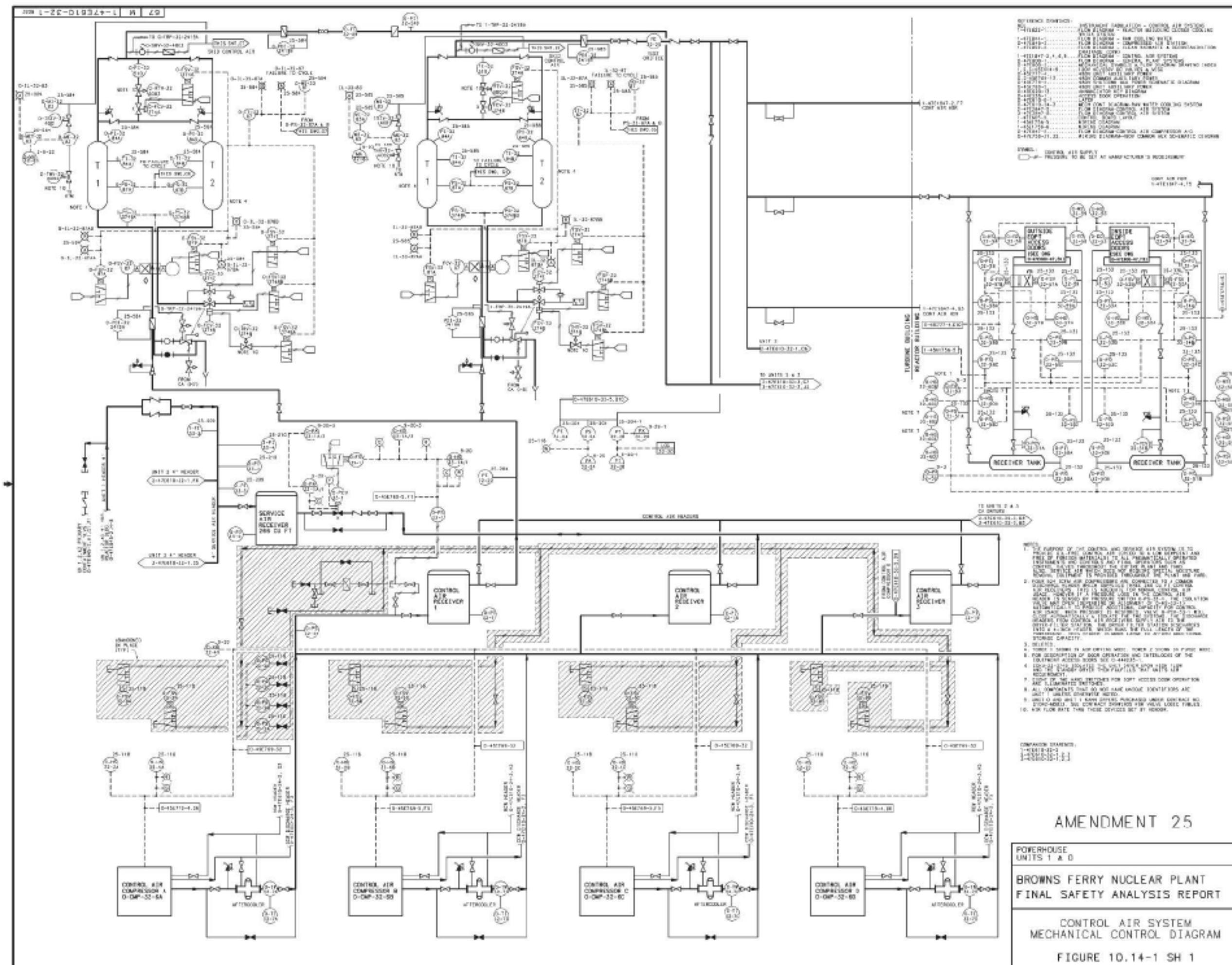
10.14.5 Safety Evaluation

The Control and Service Air Systems are not essential to safe shutdown except for the Control Air System accumulators serving the ADS main steam relief and main steam isolation valves. These accumulators are designed as Class I for withstanding the specified earthquake loadings (see Appendix C). In the event of control air failure, accumulator air is trapped by check valves. An emergency compressor is provided to maintain Control Building instrument air so that the Control Building environment can be held within safe limits for the equipment and personnel.

Although the DCA System is not essential for safe shutdown of the plant, it could be effectively utilized during this operation. Therefore, redundant components and a backup source of control air from the plant Control Air System are provided to increase the reliability of these drywell air systems. An emergency backup source of nitrogen from the CAD system is provided to each units' DCA system. Also, portions of the DCA System outside of the drywells are designed to meet Seismic Class I requirements. These portions are for each DCA supply header, from the primary containment penetration out to the outboard check valve.

10.14.6 Inspection and Testing

No special tests are required. Routine visual inspection of the system components, instrumentation, and trouble alarms is adequate to verify system operability.

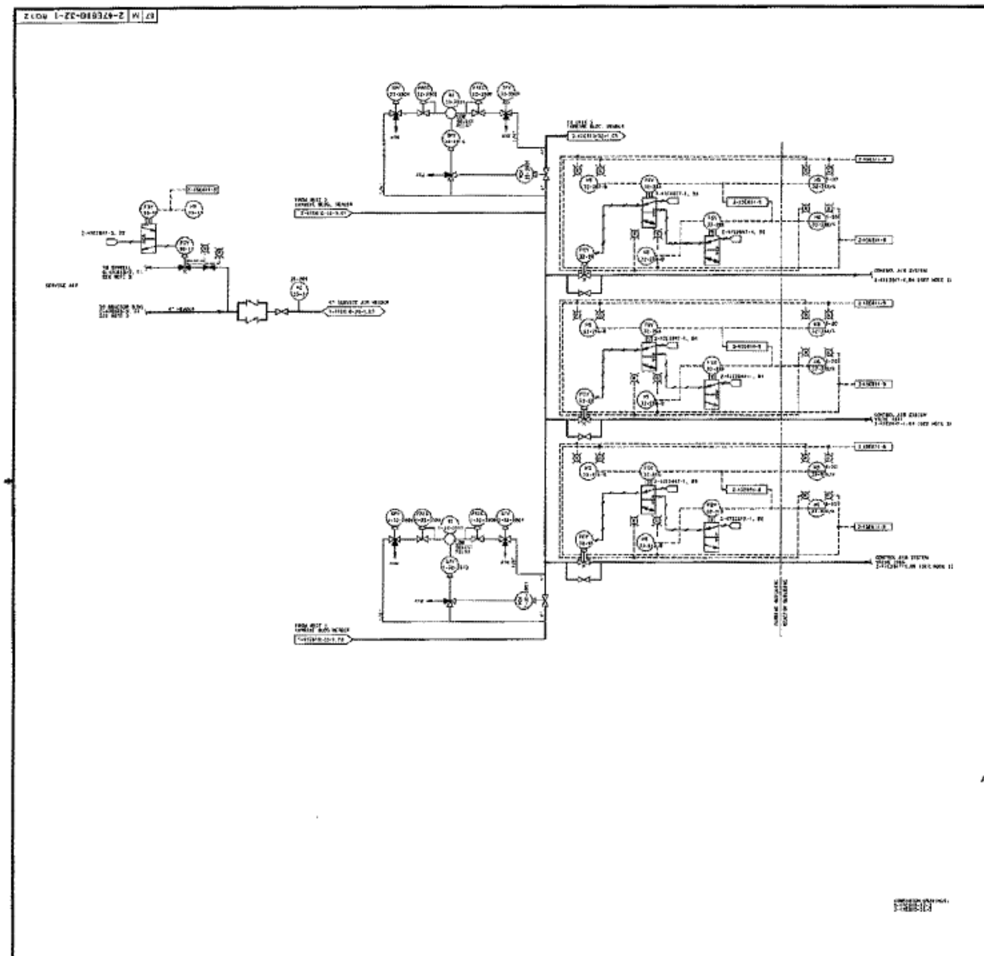


POWERHOUSE
UNITS 1 & 0

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

CONTROL AIR SYSTEM
MECHANICAL CONTROL DIAGRAM

FIGURE 10.14-1 SH 1



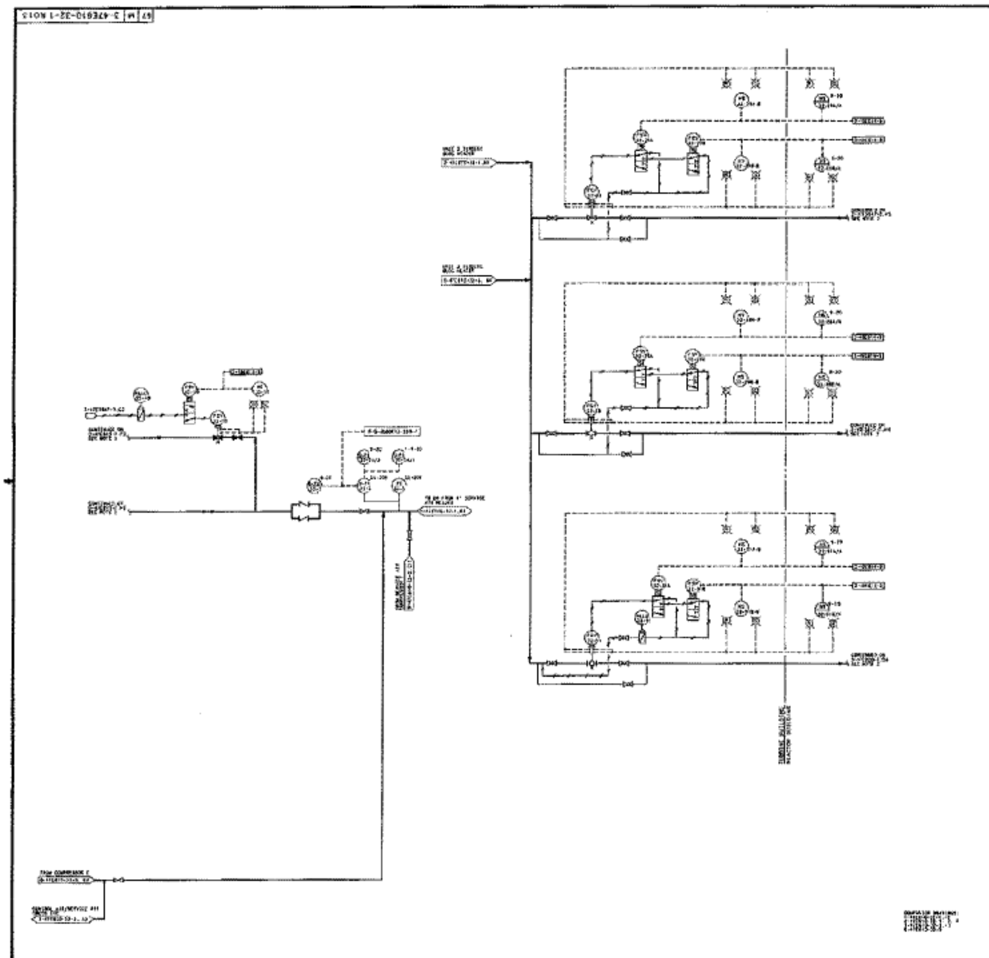
AMENDMENT 22

POWERHOUSE
UNIT 2

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

CONTROL AIR SYSTEM
MECHANICAL CONTROL DIAGRAM

FIGURE 10.14-1 SH 2



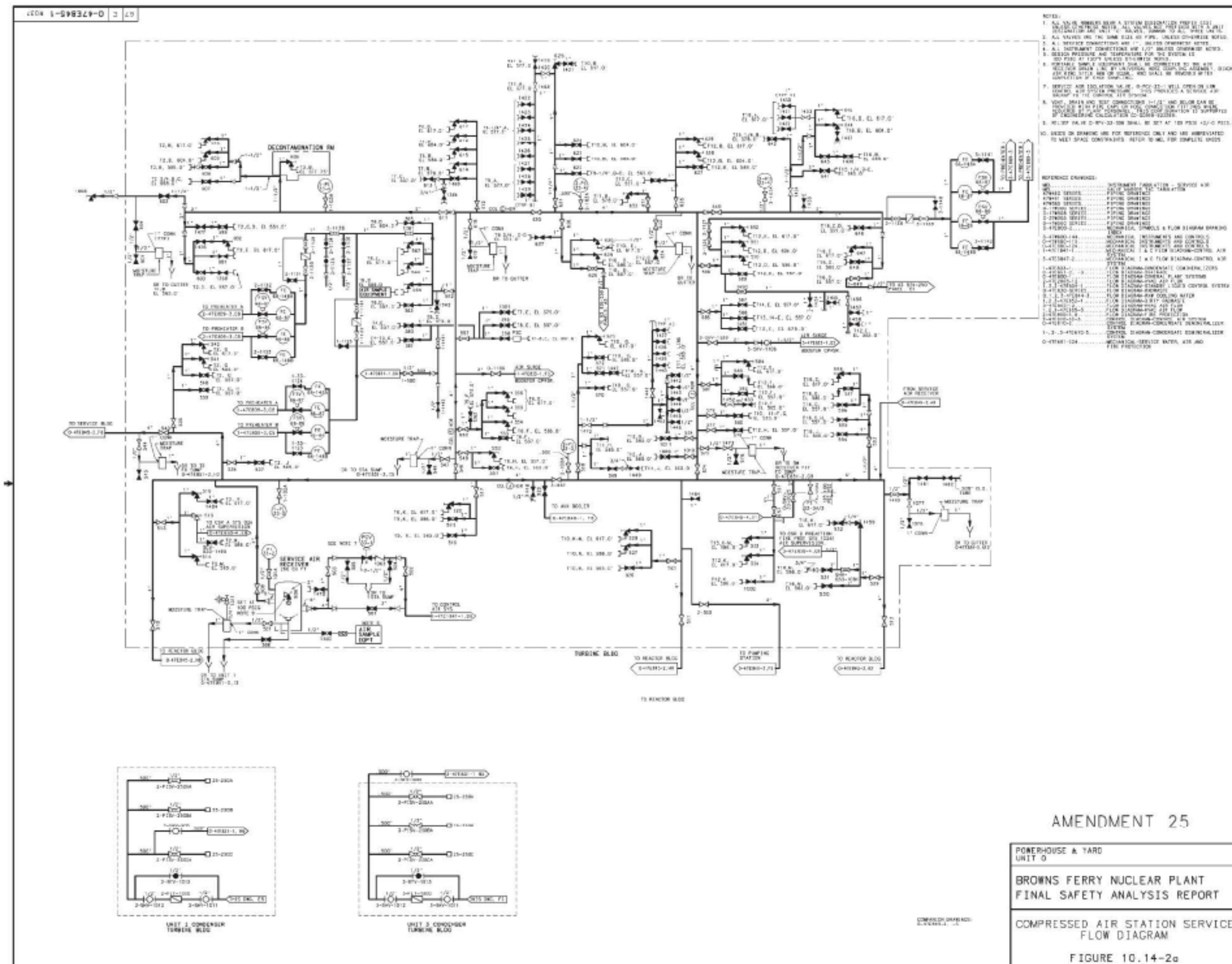
AMENDMENT 22

POWERHOUSE
UNIT 3

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

CONTROL AIR SYSTEM
MECHANICAL CONTROL DIAGRAM

FIGURE 10.14-1 SH 3



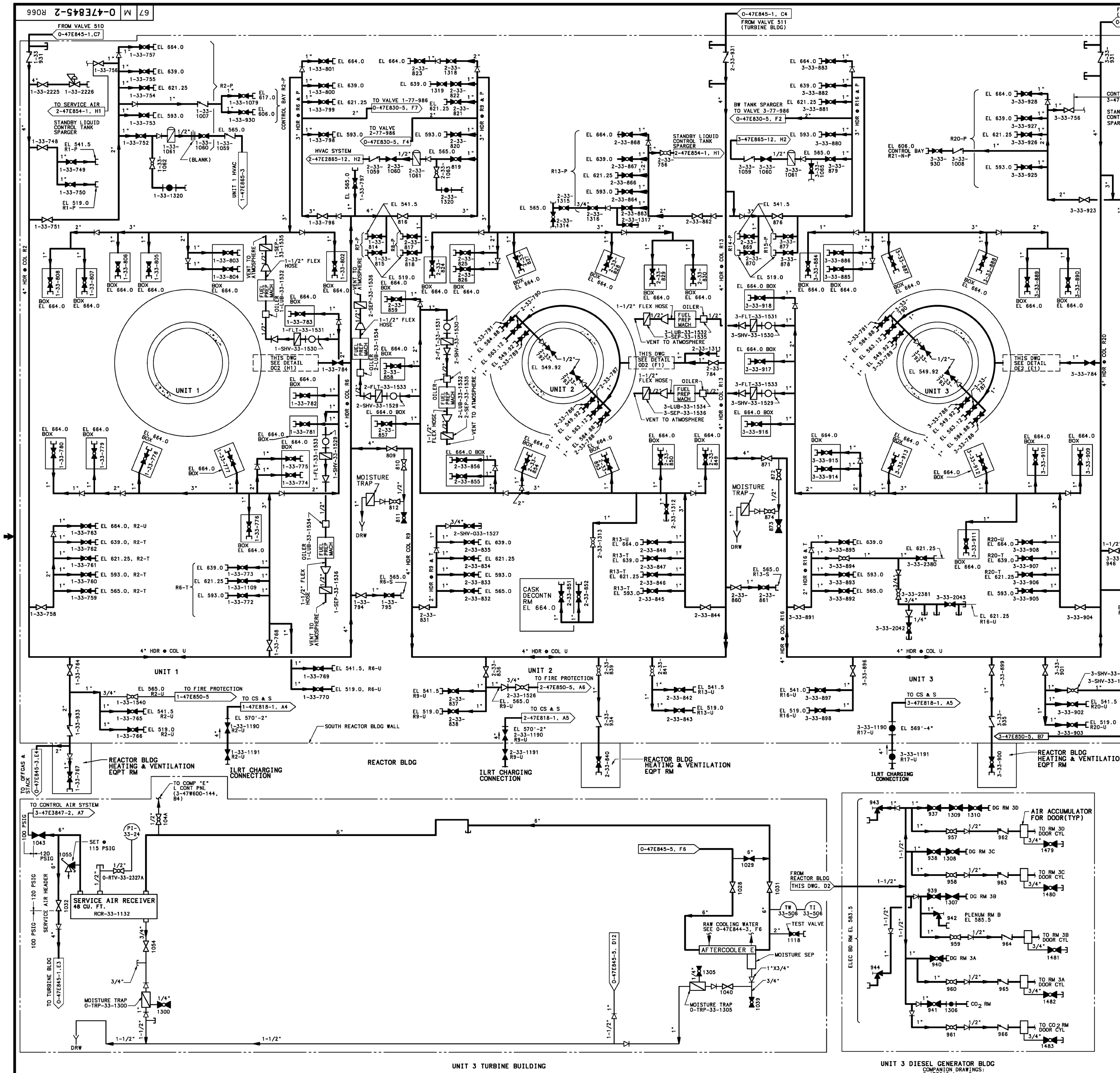
AMENDMENT 25

POWERHOUSE & YARD
UNIT 0

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

COMPRESSED AIR STATION SERVICE
FLOW DIAGRAM

FIGURE 10.14-2a



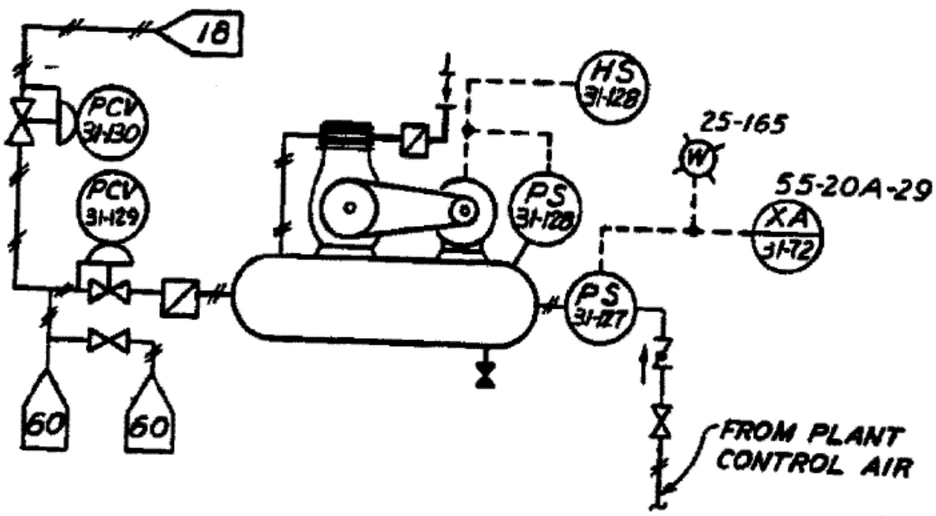
AMENDMENT 28

POWERHOUSE
UNIT 0

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

COMPRESSED AIR STATION SERVICE
FLOW DIAGRAM

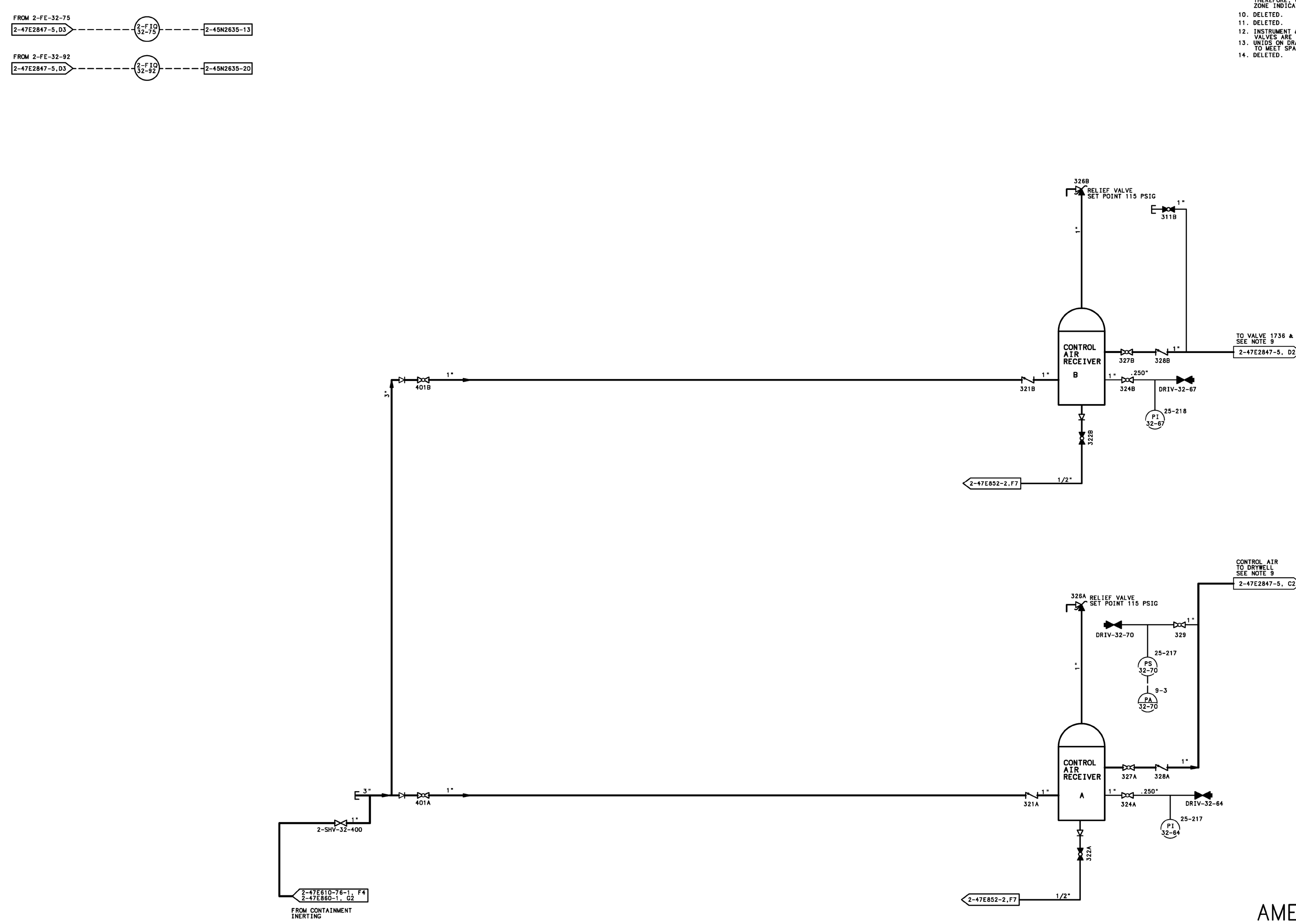
FIGURE 10.14-2b



AMENDMENT 16

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

Emergency Control Air Compressor
for Control Room Ventilation System
FIGURE 10.14-3



NOTES:

1. FOR GENERAL NOTES AND REFERENCE DRAWINGS SEE 2-47E610-32-1 AND 2-47E2847-1.
2. THIS SYSTEM PROVIDES CONTROL AIR FOR THE PNEUMATIC INSTRUMENTATION INSIDE THE DRYWELL. CROSS-CONNECTIONS ARE PROVIDED TO THE MAIN PLANT CONTROL AIR SYSTEM AND TO THE OUTBOARD MAIN STEAM ISOLATION VALVES.
3. DELETED.
4. DELETED.
5. DELETED.
6. DELETED.
7. DELETED.
8. ALL VALVES SHALL BE TAGGED. VALVE NUMBERS SHALL BE PREFIXED WITH THE SYSTEM NUMBER, I.E. 32-321A. VALVES ON THIS DRAWING ARE FOR UNIT 2, UNLESS OTHERWISE NOTED.
9. NO CONTROL DRAWING IS REQUIRED DOWNSTREAM OF THIS POINT THEREFORE, CONTINUE ON REFERENCED FLOW DIAGRAM AT THE ZONE INDICATED.
10. DELETED.
11. DELETED.
12. INSTRUMENT & CONTROL COMPONENTS ARE PREFIXED "2-" AND MANUAL VALVES ARE PREFIXED "2-32-" UNLESS OTHERWISE NOTED.
13. UNIDS ON DRAWING ARE FOR REFERENCE ONLY AND ARE ABBREVIATED TO MEET SPACE CONSTRAINTS, REFER TO MEL FOR COMPLETE UNIDS.
14. DELETED.

AMENDMENT 28

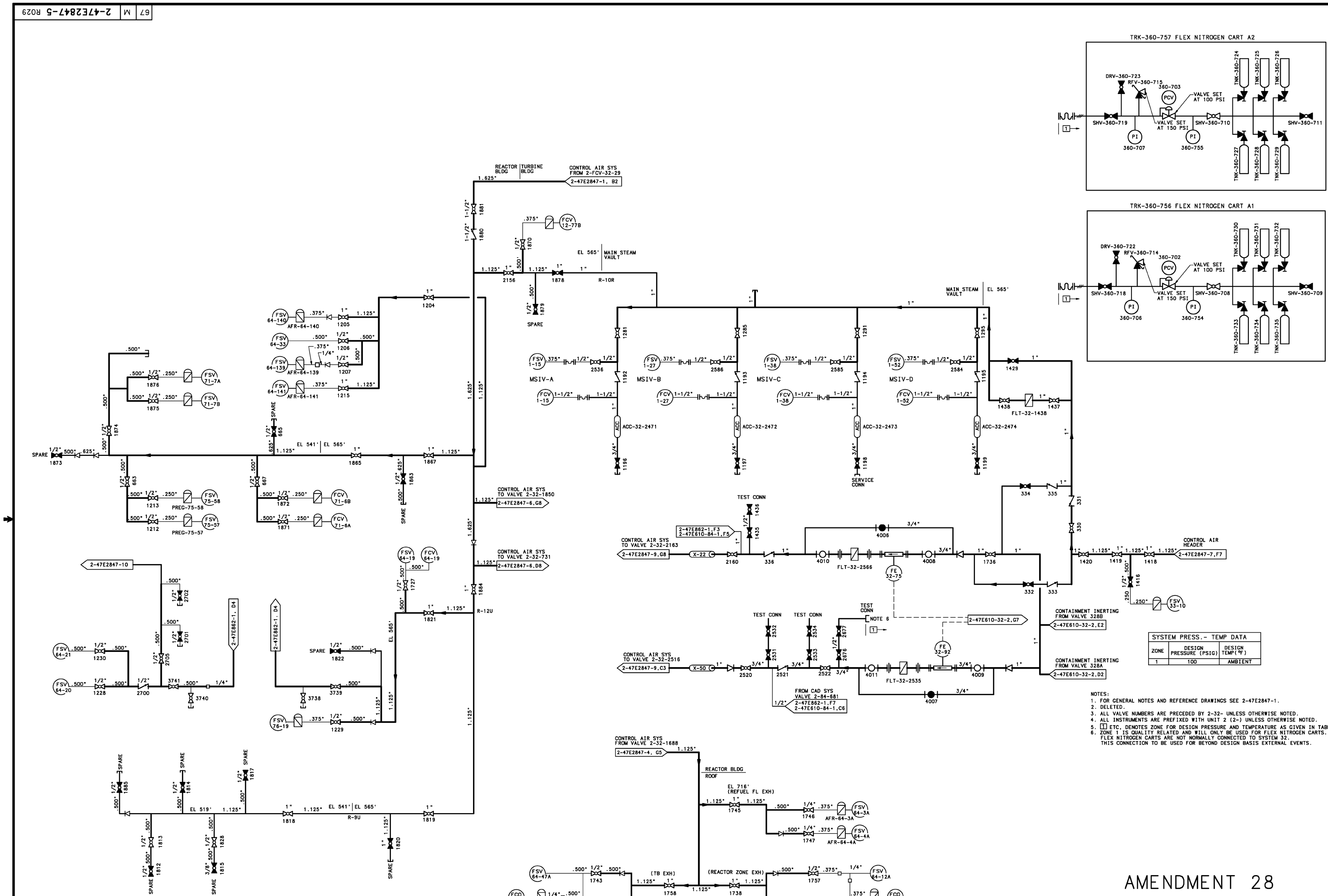
POWERHOUSE
UNIT 2

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

CONTROL AIR SYSTEM
MECHANICAL CONTROL DIAGRAM

FIGURE 10.14-4 SH 1

2-47E2847-5 R029 | M | 67



SYSTEM PRESS.- TEMP DATA		
ZONE	DESIGN PRESSURE (PSIG)	DESIGN TEMP(°F)
1	100	AMBIENT

NOTES:
1. FOR GENERAL NOTES AND REFERENCE DRAWINGS SEE 2-47E2847-1.
2. DELETED.
3. ALL VALVE NUMBERS ARE PRECEDED BY 2-32- UNLESS OTHERWISE NOTED.
4. ALL INSTRUMENTS ARE PREFIXED WITH UNIT 2 (2-) UNLESS OTHERWISE NOTED.
5. 1 ETC. DENOTES ZONE FOR DESIGN PRESSURE AND TEMPERATURE AS GIVEN IN TABLE.
6. ZONE 1 IS QUALITY RELATED AND WILL ONLY BE USED FOR FLEX NITROGEN CARTS. FLEX NITROGEN CARTS ARE NOT NORMALLY CONNECTED TO SYSTEM 32. THIS CONNECTION TO BE USED FOR BEYOND DESIGN BASIS EXTERNAL EVENTS.

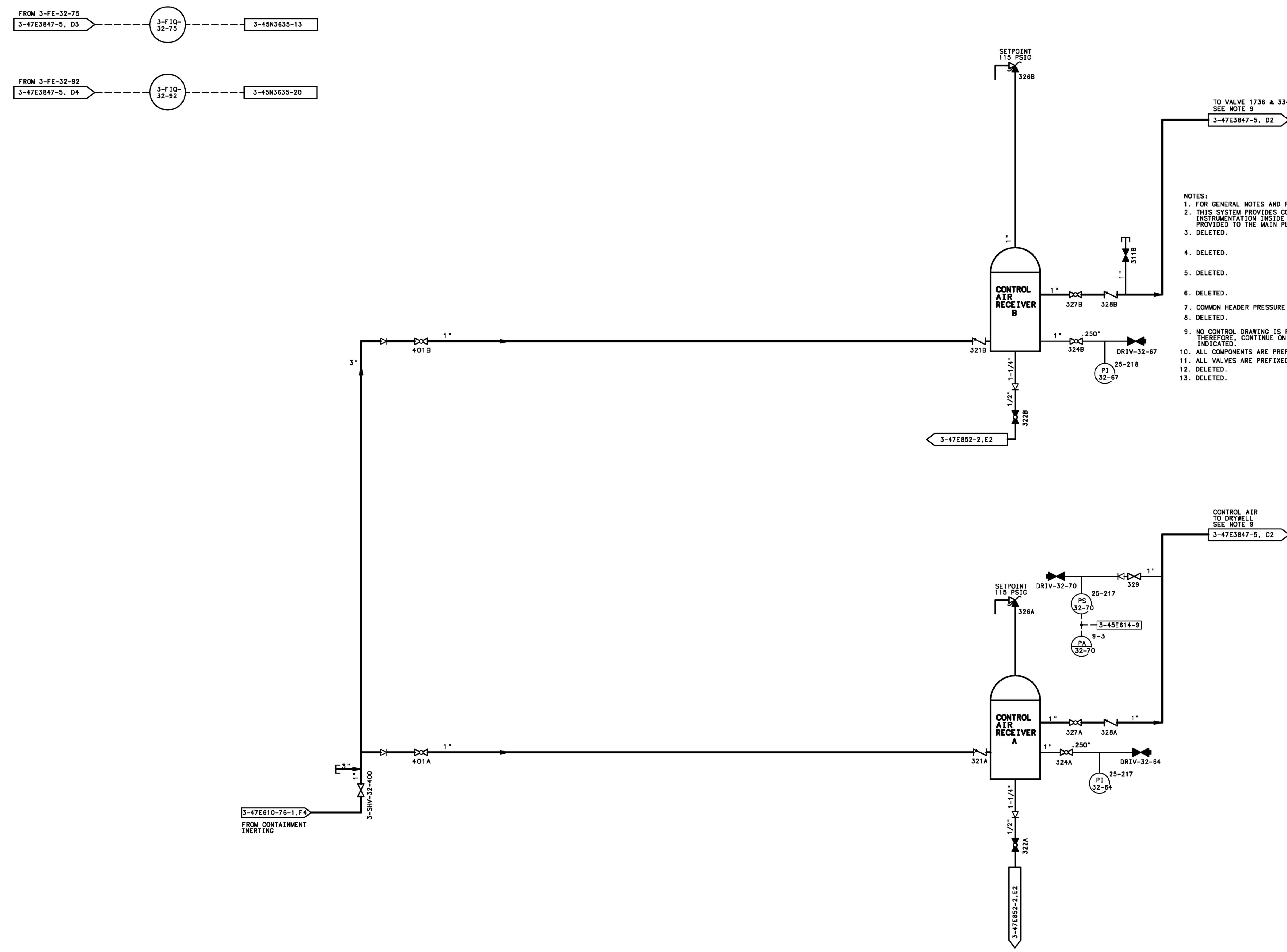
AMENDMENT 28

REACTOR BUILDING
UNIT 2

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

CONTROL AIR SYSTEM
FLOW DIAGRAM

FIGURE 10.14-4 SH 2



NOTES:
1. FOR GENERAL NOTES AND REFERENCE DRAWINGS SEE 3-47E610-32-1
2. THIS SYSTEM PROVIDES CONTROL AIR FOR THE PNEUMATIC INSTRUMENTATION INSIDE THE DRYWELL. A CROSS-CONNECTION IS PROVIDED TO THE MAIN PLANT CONTROL AIR SYSTEM.
3. DELETED.
4. DELETED.
5. DELETED.
6. DELETED.
7. COMMON HEADER PRESSURE LOSS IS ALARMED IN THE MCR BY PA-32-70.
8. DELETED.
9. NO CONTROL DRAWING IS REQUIRED DOWNSTREAM OF THIS PRINT THEREFORE, CONTINUE ON REFERENCED FLOW DIAGRAM AT THE ZONE INDICATED.
10. ALL COMPONENTS ARE PREFIXED BY "3-" UNLESS OTHERWISE NOTED.
11. ALL VALVES ARE PREFIXED BY "3-32-" UNLESS OTHERWISE NOTED.
12. DELETED.
13. DELETED.

SEE 3-47E610-32-1 FOR COMPANION DRAWINGS

AMENDMENT 28

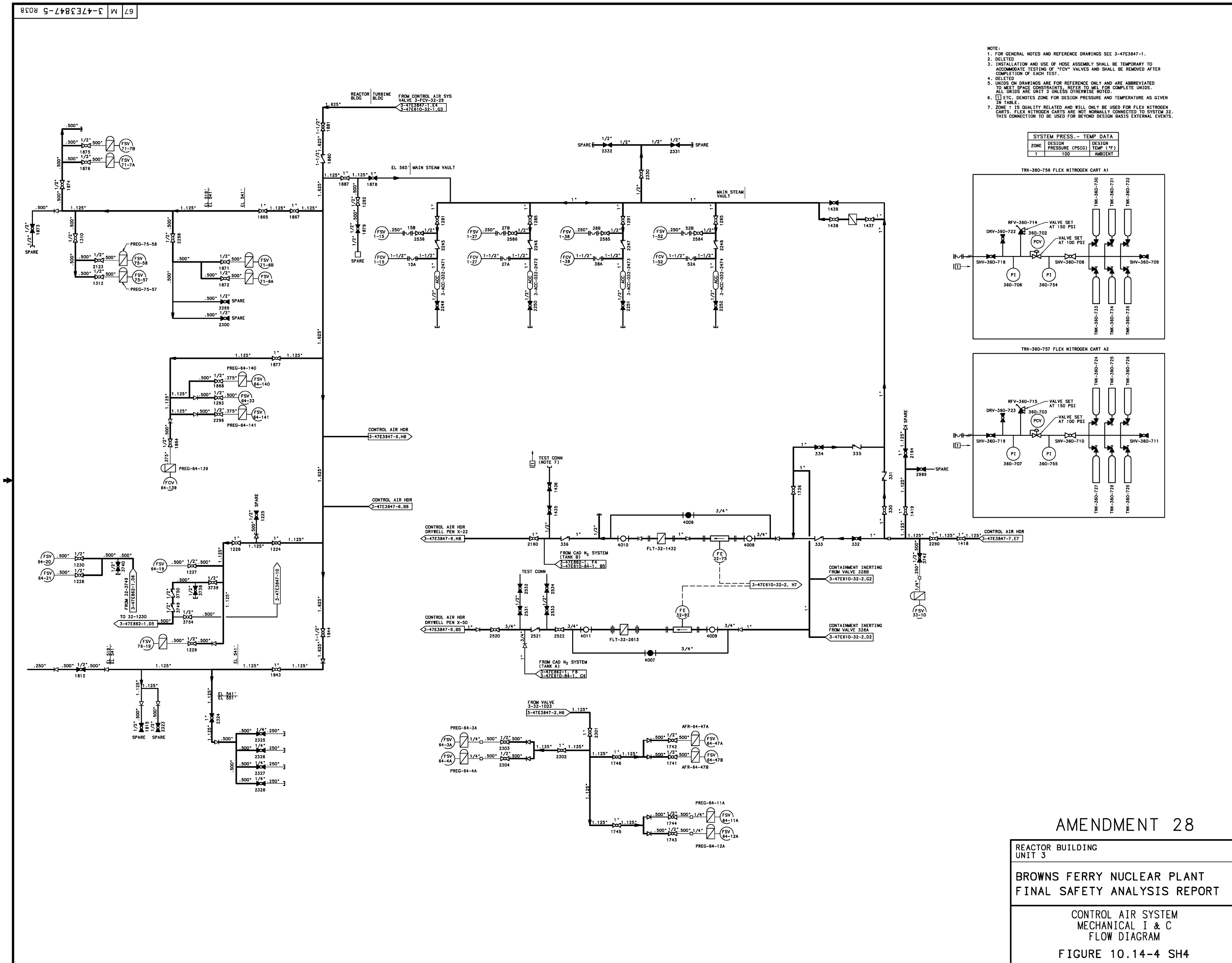
POWERHOUSE
UNIT 3

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

CONTROL AIR SYSTEM
MECHANICAL CONTROL DIAGRAM

FIGURE 10.14-4 SH 3

67 | M | 3-47E610-32-2 R022



NOTE:
1. FOR GENERAL NOTES AND REFERENCE DRAWINGS SEE 3-47E3847-1.
2. DELETED
3. INSTALLATION AND USE OF HOSE ASSEMBLY SHALL BE TEMPORARY TO ACCOMMODATE TESTING OF "FCV" VALVES AND SHALL BE REMOVED AFTER COMPLETION OF EACH TEST.
4. DELETED
5. UNIDS ON DRAWINGS ARE FOR REFERENCE ONLY AND ARE ABBREVIATED TO MEET SPACE CONSTRAINTS. REFER TO MEL FOR COMPLETE UNIDS. ALL UNIDS ARE UNIT 3 UNLESS OTHERWISE NOTED.
6. $\boxed{1}$ ETC. DENOTES ZONE FOR DESIGN PRESSURE AND TEMPERATURE AS GIVEN IN TABLE.
7. ZONE 1 IS QUALITY RELATED AND WILL ONLY BE USED FOR FLEX NITROGEN CARTS. FLEX NITROGEN CARTS ARE NOT NORMALLY CONNECTED TO SYSTEM 32. THIS CONNECTION TO BE USED FOR BEYOND DESIGN BASIS EXTERNAL EVENTS.

SYSTEM PRESS.- TEMP DATA

ZONE	DESIGN PRESSURE (PSIG)	DESIGN TEMP (°F)
1	100	AMBIENT

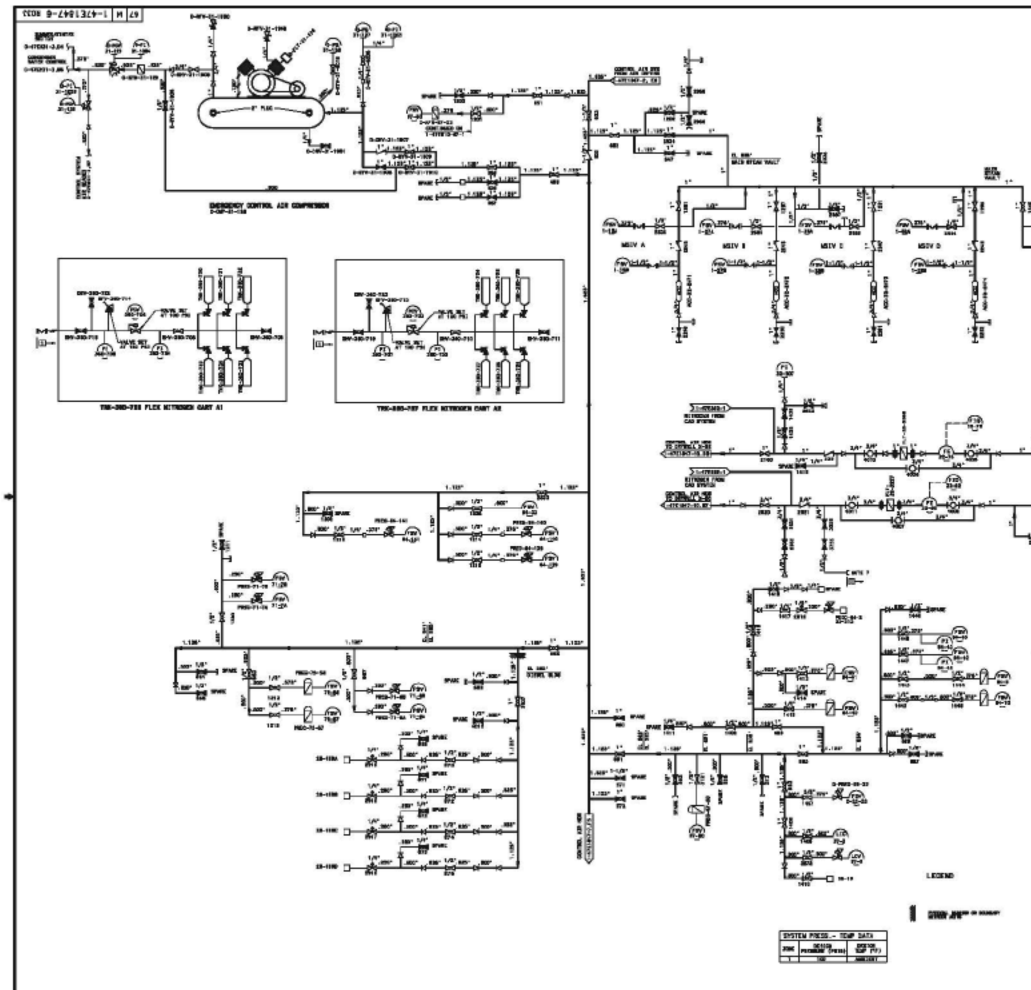
AMENDMENT 28

REACTOR BUILDING
UNIT 3

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

CONTROL AIR SYSTEM
MECHANICAL I & C
FLOW DIAGRAM

FIGURE 10.14-4 SH4



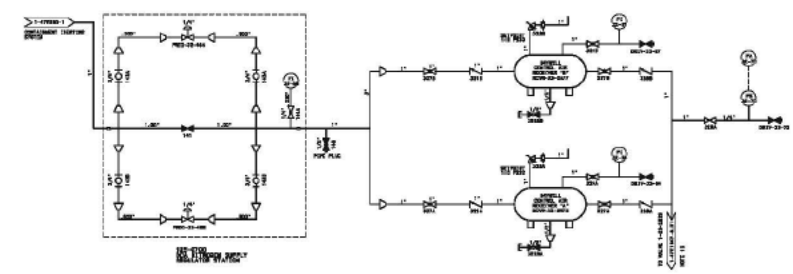
AMENDMENT 27

REACTOR BUILDING
UNIT 1

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

CONTROL AIR SYSTEM
MECHANICAL I & C
FLOW DIAGRAM

FIGURE 10.14-4 SH 5



AMENDMENT 27

POWERHOUSE
UNIT 1

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

CONTROL AIR SYSTEM
MECHANICAL CONTROL DIAGRAM

FIGURE 10.14-4 SH 6

10.15 POTABLE WATER AND SANITARY SYSTEMS

The potable water for use in the plumbing systems is supplied by the city of Athens, Alabama. Obtaining water from this state-approved water supply was more economical than constructing and operating either a temporary and permanent purification plant.

Backflow preventers have been installed at each cross connection to other systems to protect the potable water supply from possible contamination due to backflow. Sewage from the project is collected and treated.

These shared systems do not influence the operational safety of the plant. The portions of the systems which penetrate secondary containment were evaluated to ensure they could prevent seismically induced failure from causing a breach of containment. The lines are provided with valves to insure the containment integrity if the piping systems are damaged by seismically induced loads.

10.16 EQUIPMENT AND FLOOR DRAINAGE SYSTEMS

10.16.1 Power Generation Objective

The objective of the drainage systems is to collect and remove from the plant all liquid wastes from their points of origin to the river directly, or, if necessary, to the Radioactive Waste Building (see Subsection 9.2), where they are treated and returned for reuse or discharged to the river.

10.16.2 Power Generation Design Basis

1. Liquid wastes shall be collected and discharged in a manner such that the operation or availability of the plant is not limited thereby.

2. The Reactor Building floor and equipment drain, and drywell floor drain sump pumps shall be powered from diesel-backed power sources to allow operation in a loss of normal auxiliary power situation.

10.16.3 Safety Design Basis

The drainage systems shall be designed to prevent the inadvertent release of significant quantities of liquid radioactive material from the restricted area of the plant so that resulting radiation exposures are within the guideline values of 10 CFR 20.

10.16.4 Description

10.16.4.1 General

The plant drainage is handled through two completely separate drainage systems of the following categories:

a. Radioactive drainage, and
b. Nonradioactive drainage.

The Reactor Building floor drainage system (Radioactive Drainage) is shared in respect to the common drain header into which the unitized Reactor Building floor drainage pumps discharge. Two Reactor Building floor drainage pumps are provided for each unit. These are powered from diesel-backed power sources to allow operation in a loss of normal auxiliary power situation. They are designed to accommodate several times the maximum expected drainage flow. Each pump has a capacity of 160 gpm at 90 ft of head, and both pumps may operate concurrently.

10.16.4.2 Radioactive Drainage

Radioactive drainage consists of both equipment and floor drainage. Equipment drainage consists of waste leakage from equipment such as rotating shaft glands, miscellaneous line drains, and equipment drains for maintenance. With the exception of Turbine Building equipment drains, these drains are collected in closed piping systems which terminate in closed and shielded sumps located where necessary to accommodate a gravity drainage system. In the Turbine Building, equipment drains from equipment that might contribute to airborne contamination are connected into closed headers (no funnels) and routed to equipment drain sumps. Equipment drains that are not considered to have this potential for airborne contamination are collected into open headers and routed separately to equipment drainage sumps. From these sumps the waste is pumped to the Radwaste Building, where it enters the waste collector tank to be held for treatment.

The radioactive floor drainage system drains areas which may contain radioactive materials. These are collected and piped to shielded sumps in a manner similar to equipment drains. Floor drains are separated into individual systems according to the level of contamination from a higher to a lower contaminated level area. Each separate drain header is terminated below minimum water level in the sump to effectively seal it from other drains. From these sumps, waste liquid is pumped to radwaste, where it enters the floor drain collector tank to be held for treatment.

10.16.4.3 Nonradioactive Drainage

The nonradioactive floor drains or drains from unlimited access areas are further divided into two collection systems:

a. Nonradioactive, noncontaminated drains, and
b. Nonradioactive drains of possible contamination.

The nonradioactive, noncontaminated drains are collected in drain sumps located conveniently throughout the plant where level controlled sump pumps pump this drainage into the condenser circulating water discharge tunnels.

Drains of possible nonradioactive contamination, such as floor drains installed below oil-filled transformers or lubricating oil tanks where accidental oil spills could take place, are collected in a separate drainage system and sump. A very small amount of potentially radioactive drainage may be directed to this sump. However, radioactive drainage to sump is removed from sump on a

noncontinuous basis and may be disposed of by the liquid radwaste system which ensures that all plant releases are within the limits specified in 10 CFR 20.

10.16.4.4 Power Supply

All Reactor Building drainage system pumps are fed from the reactor 480-V MOV boards and are immediately available in a loss of normal auxiliary power situation. The Turbine Building drainage system pumps are fed from the turbine 480-V MOV boards and are available for operation by manual backfeed when power is available.

10.16.4.5 Diagrams

For flow diagrams of the radwaste drainage, refer to Subsection 9.2, "Liquid Radwaste System."

10.16.4.6 Evaluation for Flooding due to Failure of Low Energy Piping Systems Outside Primary Containment

The following is TVA’s response to the letter from Roger S. Boyd of the AEC to James E. Watson of TVA, dated August 11, 1972, requesting an evaluation of Browns Ferry nuclear facility and systems to determine (1) whether the failure of any non-class I equipment, particularly in the circulating water system and fire protection system, could result in flooding which would adversely affect Class I equipment, and (2) whether failure of any equipment could cause flooding such that common mode failure of redundant equipment would result.

It has been revised to reflect guidance for pipe failure evaluation provided later by the AEC in a letter from A. Giambusso of the AEC to James E. Watson of TVA dated December 18, 1972, and errata provided by letter from the AEC dated January 19, 1973 and a letter from John F. Stolz of the AEC to James E. Watson of TVA, dated May 22, 1973, that summarized discussions between the AEC and TVA at a meeting held May 3, 1973. These letters are identified in Supplement 3 to the Safety Evaluation Report for BFN Units 1, 2, and 3.

Failures in the pressure boundaries of the water systems in either Diesel Generator Building will not prevent a safe shutdown of a unit because penetrations, including the large equipment access doors into the Diesel Generator Buildings, are sealed except the personnel access hatch into the Control Building at EL 593. An emergency drainage system exists that has sufficient capacity to prevent the accumulation of water to a level that would adversely affect safety related systems, structures, and components in the Diesel Generator Buildings.

Failures in the pressure boundaries of any water system (including fire protection systems) in a Reactor Building or inadvertent leakage into a building (most likely from the Turbine Building) will not prevent a safe shutdown of the unit. The reasons are given below:

1. All penetrations between Reactor Buildings below plant grade (EL 565) are sealed; therefore, flooding is confined to one building.

2. All penetration into the periphery of the Reactor Buildings are sealed and made watertight to EL 578, including the personnel locks and equipment accesses.

3. Flooding in the lower portion of each building will be detected by the following three mechanisms: 1) automatic initiation of the Reactor Building floor and drain system sump pumps; 2) the water level switches which actuate when the water level is approximately two inches on the EL 519 floor in the torus area, the HPCI room, and the four corner rooms, and/or; 3) inspection. Evidence of a break will alert the control room operator to take corrective measures.

4. There are a number of ways of interrupting the power to the affected system(s).

5. Also, two RHR pump-heat exchanger combinations and two RHRSW pumps from an adjacent unit can be used to provide reactor cooling in an emergency.

Loss of the engineered safety systems pumps, RHRSW, on the intake structure cannot prevent the shutdown of a unit because of the following:

1. The high walls forming the four compartments around the RHRSW pumps provide protection against natural phenomena such as tornadoes and wind waves in conjunction with floods (probable maximum flood plus waves from 45 mph winds).

2. Failure of the pressure boundaries of the water systems inside one compartment will not cause the water to overflow into an adjacent compartment because the wall design is such that the water preferentially overflows the rear wall (rear walls are one foot lower than other walls).

Failures in the pressure boundaries of any water system in the Turbine, Service, Radwaste, and Offgas Buildings and the Stack will not prevent a safe shutdown of a unit because of the following reasons:

1. There are no engineered safety systems located inside these areas.

2. Sumps are included in each area with high level alarms that annunciate in the Radwaste Building, which will be manned continuously, so that any necessary corrective actions can be initiated.

3. All penetrations between these areas and the Reactor Building below EL 572.5 are sealed, including the personnel accesses which are watertight, bulkhead-type doors. (This allows at least 20 minutes to detect the worst failure, a condenser intake culvert open-ended break in the Turbine Building, and to stop the main condenser circulating water pumps.) This is 7.5 feet above plant grade, and the water would be drained to the plant yard through the outside doors.

10.16.5 Safety Evaluation

For the safety evaluation of safety design basis , refer to Subsection 9.2.

10.16.6 Inspection and Testing

No special tests are required. Routine visual inspection of the system components, instrumentation, and trouble alarms is adequate to verify system reliability.

10.17 PROCESS SAMPLING SYSTEMS

10.17.1 Power Generation Objective

The objective in sampling process liquids and gases is to provide representative samples for testing to obtain data from which the performance of the plant, items of equipment, and systems may be determined.

10.17.2 Power Generation Design Basis

1. Sampling points shall be located so as to provide information needed for plant operation as well as for special tests.

2. The sampling systems shall be designed to provide samples which are representative of the fluids being sampled.

10.17.3 Description

10.17.3.1 Liquid Sampling Systems

Table 10.17-1 gives a list of liquid samples taken (additional samples may be obtained as warranted by plant conditions), the location of the sample connection, and the primary purpose for which the sample is taken. Figures 10.17-1a, 10.17-1b, 10.17-1c sheets 1 and 2, 10.17-1d, 10.17-1e, and 10.17-1f provide diagrams of the systems.

Normally, liquid grab samples which could contain radioactivity in excess of background are taken within sampling hoods. Most hoods have doors which are normally kept closed except when sample bottles are being inserted or removed. Handles of most sample valves are located outside the hood. The hoods have windows for observation of the sampling. Most sample points have a method of flushing to ensure representative samples.

Hoods are located close to sample points to minimize the lengths of sample lines. Sampling hoods are located in various locations throughout the plant including the Reactor Building, Turbine Building, and Radwaste Building.

10.17.3.1.1 Reactor Water Cleanup System

When the Reactor Water Cleanup System is in operation, reactor water quality is monitored by means of a sample connection located upstream of the two cleanup filter/demineralizers.

Performance of the individual filter/demineralizers can be assessed on a comparison of samples taken downstream of each filter/demineralizer to those samples taken

upstream. At each of these sample points, a continuously flowing stream is passed through a cooling system and then through a conductivity cell. The streams from the three sample points are combined and routed back into the cleanup system or to radwaste. Grab samples can be taken at each of the three sample points for determination of radiochemical constituents of interest.

When the Reactor Water Cleanup System is not in operation, grab samples of recirculation system reactor water can be taken at a sample connection located on Recirculation Loop A. The reactor water recirculation system sample is cooled by a cooler which is supplied with cooling water from the Reactor Building Closed Cooling Water System.

10.17.3.1.2 Main Steam System

Reactor steam may be sampled for determination of radioactivity and content of noncondensables. Samples may be withdrawn from each of the four steam lines at points upstream of the respective stop valves.

10.17.3.1.3 Condensate System

Inleakage of condenser cooling water is indicated by an increase in conductivity of the condensate or reactor water. With the waterbox out-of-service, condenser leaks may be located by use of a Helium leak detection system.

Continuously flowing sample streams from the filter/demineralizer inlet and outlet sample points and from the outlets of filter/demineralizer units are monitored for conductivity with values being recorded. Exhaustion of the ion exchange capacity of a demineralizer results in an increase in conductivity in the outlet of that demineralizer.

Grab samples can be taken from each of the continuously flowing stream samples.

10.17.3.1.4 Feedwater

Samples may be taken from the combined flow from the No. 1 heaters before the feedwater line penetrates the primary containment. A portion of the combined-flow sample can be passed continuously through particulate filters and ion exchange papers with another portion being passed through a conductivity cell. A sample cooler is used for cooling samples at the feedwater sampling point.

Grab samples can be taken from the feedwater sample point.

10.17.3.1.5 Radwaste and Fuel Pool Systems

Samples used for monitoring the Radwaste and Fuel Pool Cooling and Cleanup Systems are piped to a hooded sink located in the Radwaste Building. All liquids to be released to the river, from the Radwaste System, are sampled and analyzed before release. Water to be returned to process is checked for quality before it is sent to condensate storage. Samples can also be taken of liquids before and during processing.

When any of the fuel pool filter/demineralizers, the waste demineralizer, or the floor drain filter is in operation, continuously flowing samples of the effluents can be passed through inline conductivity cells. Conductivity readings are recorded. High-conductivity alarms are provided for the demineralizer.

10.17.3.1.6 Other Liquid Systems

The Reactor Building Closed Cooling Water Systems are provided with grab sample connections which are used to determine whether inleakage of radioactive liquid is taking place.

Raw water used in cooling the Residual Heat Removal System heat exchangers is monitored continuously for radioactivity while the system is in operation. The monitor is supplemented by grab sampling and analysis. Grab sample points are located downstream of each RHR heat exchanger.

Auxiliary systems such as the water treatment plant, the various raw water cooling systems, and the auxiliary boilers, are provided with sampling connections needed for their proper operation.

10.17.3.1.7 Recorders and Alarms

Recorders/Displays giving the readouts of conductivity and other analytical instrumentation, and associated alarms, are located on local panels, in the Main Control Room, or for Units 2 and 3 in the radio-chemical laboratory.

10.17.3.2 Gas Sampling Systems

Table 10.17-2 gives a list of gas samples taken, the location of the sample connection, and the primary purpose for which the sample is taken.

10.17.3.2.1 Air Ejector Offgas

A sample is withdrawn from the offgas line at a point downstream of the air ejector after-condenser. A diagram of the air ejector offgas sampler is shown in Figure 10.17-2. This sampler supplies the gas sample for the Air Ejector Offgas Radiation Monitoring System (see Subsection 7.12) and also provides a grab sample of the

offgas for laboratory analysis. Determination of radioactive constituents can be made as desired.

10.17.3.2.2 Modified Offgas Filters

Sample connections are provided upstream and downstream of the prefilters and after filters. These connections may be used for taking grab samples and for conducting DOP (dioctyl phthalate) smoke tests to determine filter efficiency (see Subsection 9.5) as appropriate. The differential pressure across each set of filters is monitored and annunciated in the Main Control Room.

10.17.3.2.3 Stack

A sample is withdrawn from the stack at a point where offgas and dilution air are well mixed. The sample is delivered to the Stack Radiation Monitoring System (see Subsection 7.12).

10.17.4 Inspections and Tests

No special inspections or tests are required for the sampling systems.

10.17.5 Power Generation Evaluation

The sampling systems, as designed, meet the objective and design bases stated in paragraphs 10.17.1 and 10.17.2.

TABLE 10.17-1

LIQUID SAMPLES

(Sheet 1)

Item	Description	Location	Purpose	Remarks
1	Reactor Water	Recirculation Pipe	Monitor reactor water when cleanup system is isolated	Grab Sample Sample Cooling
2	Reactor Water Cleanup			
	a. Filter-Demineralizer Influent	Filter-demineralizer inlet pipe	Monitor reactor water conductivity	Cont. (a), C.E.(b) Grab Sample
	b. Filter-Demineralizer Effluent	Outlet piping from each filter demineralizer	Filter-demineralizer efficiency	Cont. (a), C.E.(b) Grab Sample
3	Steam Samples	Main steam lines (before stop valve)	Various	Grab Sample
4	Standby Liquid Control System			
	a. Tank	At tank	Borate Concentration	Grab Sample
	b. Recirculation Line	Recirculation pipe	Borate Concentration	Grab Sample
5	Pressure Suppression Pool	Suppression Recirculation and equipment test line	Check radiochemical concentrations	Grab Sample
6	Feedwater	After last heater	Corrosion studies	Grab Sample Sample Cooling
7	Closed Loop Cooling Water			
	a. Closed Loop Cooling Water	Outlet of each major heat exchanger	Determine location of heat exchanger leaks	Grab Sample
	b. Closed Loop Cooling Water	Pump Discharge	Check corrosion inh bitor concentration	Grab Sample

* For notes, see final sheet of Table 10.17-1

TABLE 10.17-1

LIQUID SAMPLES

(Sheet 2)

Item	Description	Location	Purpose	Remarks
8	Waste Disposal			
	a. Waste Surge Tank	Outlet Pipe	Process data	Grab Sample
	b. Waste Collector Tank	Pump Discharge	Process data	Grab Sample
	c. Floor Drain Collector Tank	Pump Discharge	Process data	Grab Sample
	d. Chemical Waste Tank	Pump Discharge	Process data	Grab Sample
	e. Waste Sample Tank	Pump Discharge	Recycle/Discharge suitability	Grab Sample
	f. Floor Drain Sample Tank	Pump Discharge	Discharge suitability	Grab Sample
	g. Fuel Pool Filter-Demineralizer Influent	Inlet Pipe	Fuel pool quality	Grab Sample
	h. Fuel Pool Filter-Demineralizer Effluent	Outlet Pipe	Filter-Demineralizer Efficiency	Cont.(a),C.E.(b) Grab Sample
	i. Floor Drain Filter Effluent	Outlet Pipe	Filter Efficiency	Cont.(a),C.E.(b) Grab Sample
	j. Waste Demineralizer Effluent	Outlet Pipe	Demineralizer Efficiency	Cont.(a),C.E.(b) Grab Sample
9	Condensate			
	a. Condensate	Condensate Pump Discharge	Condensate quality and tube leaks	Cont.(a),C.E.(b) Grab Sample
	b. Condensate Demineralizer Effluent	Condensate Booster Pump Discharge	Treated condensate quality	Cont.(a),C.E.(b) Grab Sample
10	Raw Cooling Water	Discharge from Closed Cooling Water Heat Exchanger	Determine no radioactivity release	Grab Sample
11	Makeup/Reuse			
	a. Demineralized Water Storage Tank	At Tank	Water quality	Grab Sample
	b. Condensate Storage Tank	At Tank	Water quality	Grab Sample

* For notes, see final sheet of Table 10.17-1

TABLE 10.17-1

LIQUID SAMPLES
(Sheet 3)

Item	Description	Location	Purpose	Remarks
12	Special Samples			
	a. Laundry Drain Tanks	Pump Discharge	Discharge suitability	Grab Sample
13	RHR Heat Exchanger	Service Water Discharge	No Radioactive release	Grab Sample

NOTES:

a. Continuously flowing sample.
b. Sample line conductivity element.

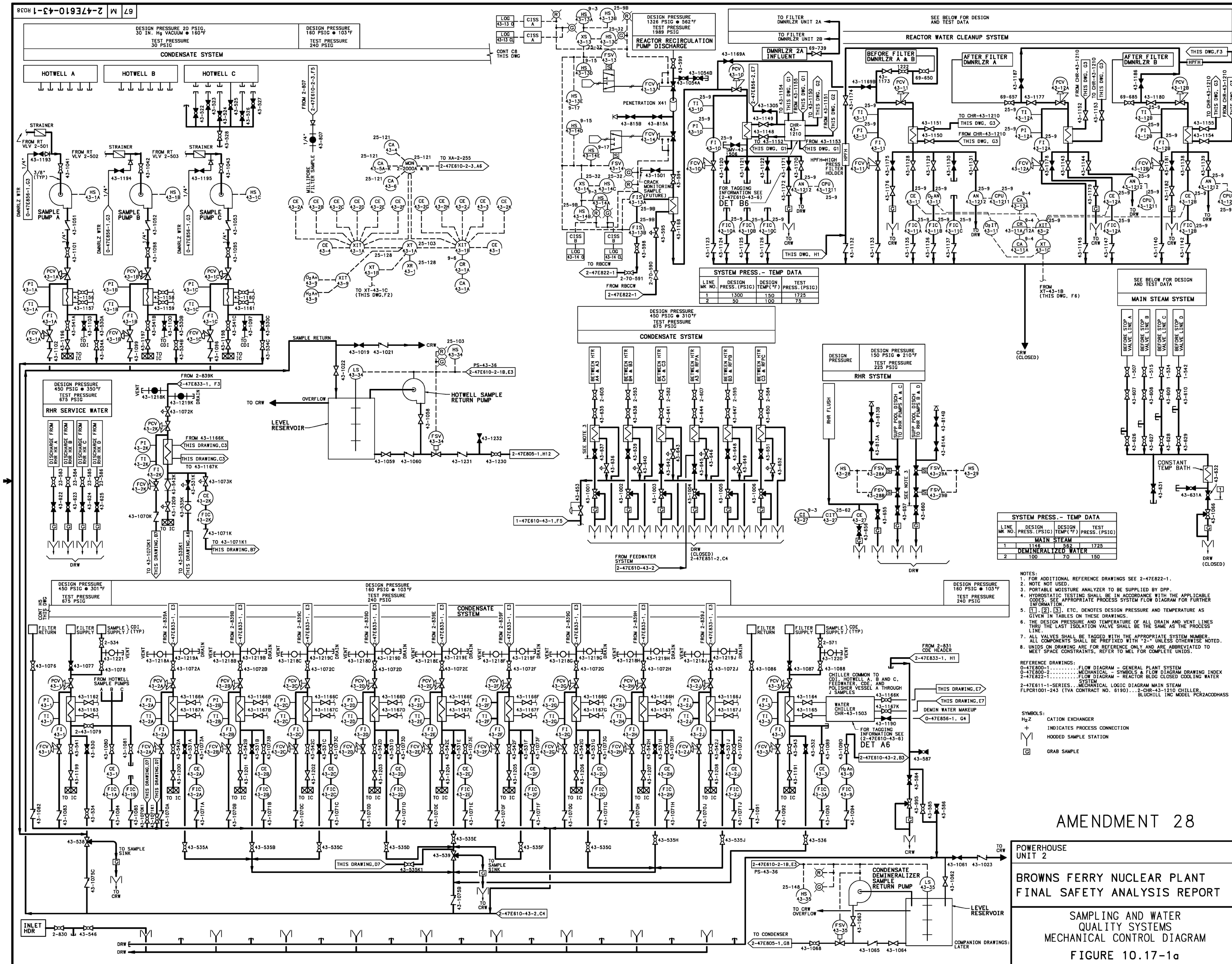
TABLE 10.17-2

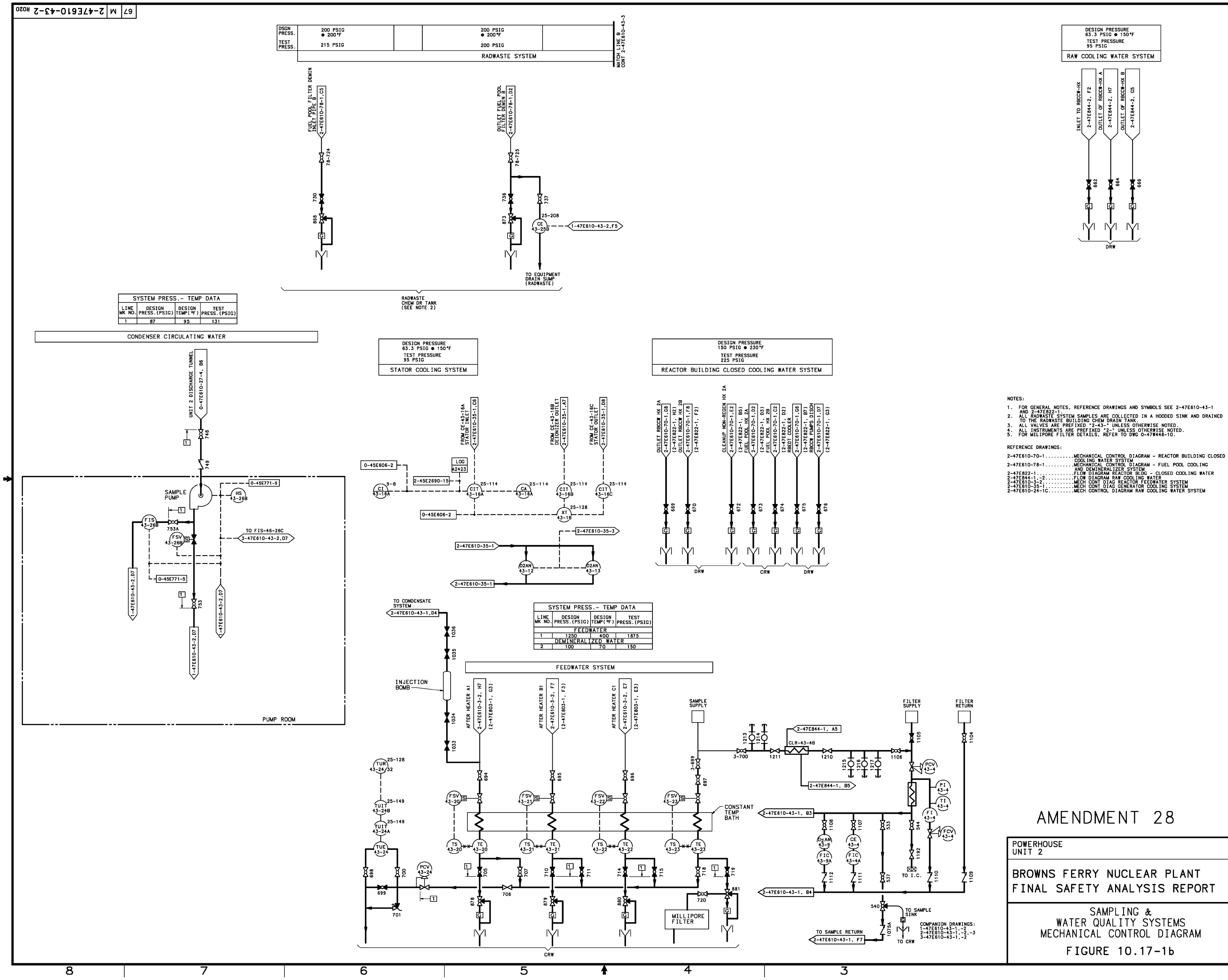
GASEOUS SAMPLES

Activity Item	Description	Location	Purpose	Remarks
1	Air Ejector Off-gas Sample (c)	After Air Ejectors (Before 30 min. holdup)	Activity Release	Cont.(a),R.E.(b) Grab Sample
2	Offgas Filter Samples	Inlet and Outlet (After 30 min. holdup)	Determine filter efficiency	Grab Sample
3	Stack Sample	Stack		
	a. Noble Gas		Activity release	Cont. (a), R.E.(b)
	b. Particulate		Particle release	Grab Sample (c)
	c. Iodine		Iodine release	Grab Sample (c)
4	Ventilation			
	a. Reactor Bldg.	Fan Discharge	Activity release	Cont.(a),R.E.(b) Grab Sample
	b. Radwaste Bldg.	Fan Discharge	Activity release	Grab Sample
	c. Turbine Bldg.	Fan Discharge	Activity release	Grab Sample
5	Standby Gas Treat-ment System	Before and after each filter	Filter efficiency	Grab Sample

NOTES:

a. Continuous flowing sample, continuously monitored.
b. Radiation element on sample line, continuously monitored.
c. Post-accident high range, continuously monitored.


2-47E610-43-1
AMENDMENT 28
SYSTEM PRESS.- TEMP DATA
LINE MK NO. | DESIGN PRESS.(PSIG) | DESIGN TEMP(°F) | TEST PRESS.(PSIG)
MAIN STEAM
1 | 1146 | 562 | 1725
DEMINERALIZED WATER
2 | 100 | 70 | 150
NOTES:
1. FOR ADDITIONAL REFERENCE DRAWINGS SEE 2-47E822-1.
2. NOTE NOT USED.
3. PORTABLE MOISTURE ANALYZER TO BE SUPPLIED BY DPP.
4. HYDROSTATIC TESTING SHALL BE IN ACCORDANCE WITH THE APPLICABLE CODES. SEE APPROPRIATE PROCESS SYSTEM FLOW DIAGRAM FOR FURTHER INFORMATION.
5. 1, 2, 3, ETC. DENOTES DESIGN PRESSURE AND TEMPERATURE AS GIVEN IN TABLES ON THESE DRAWINGS.
6. THE DESIGN PRESSURE AND TEMPERATURE OF ALL DRAIN AND VENT LINES THRU THE LAST ISOLATION VALVE SHALL BE THE SAME AS THE PROCESS LINE.
7. ALL VALVES SHALL BE TAGGED WITH THE APPROPRIATE SYSTEM NUMBER. ALL COMPONENTS SHALL BE PREFIXED WITH "2-" UNLESS OTHERWISE NOTED.
8. UNIDS ON DRAWING ARE FOR REFERENCE ONLY AND ARE ABBREVIATED TO MEET SPACE CONSTRAINTS, REFER TO MEL FOR COMPLETE UNIDS.
REFERENCE DRAWINGS:
0-47E800-1..........FLOW DIAGRAM - GENERAL PLANT SYSTEM
0-47E800-2..........MECHANICAL - SYMBOLS & FLOW DIAGRAM DRAWING INDEX
2-47E822-1..........FLOW DIAGRAM - REACTOR BLDG CLOSED COOLING WATER SYSTEM
2-47E611-1-SERIES...MECHANICAL LOGIC DIAGRAM MAIN STEAM
FLPCR1001-243 (TVA CONTRACT NO. 6190)...2-CHR-43-1210 CHILLER, BLUCHILL INC MODEL PCR2ACODHASS
SYMBOLS:
H2Z CATION EXCHANGER
INDICATES PROCESS CONNECTION
HOODED SAMPLE STATION
GRAB SAMPLE
POWERHOUSE
UNIT 2
BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT
SAMPLING AND WATER
QUALITY SYSTEMS
MECHANICAL CONTROL DIAGRAM
FIGURE 10.17-1a



SYSTEM PRESS.- TEMP DATA

LINE MK NO.	DESIGN PRESS.(PSIG)	DESIGN TEMP(°F)	TEST PRESS.(PSIG)
1	87	95	131

SYSTEM PRESS.- TEMP DATA

LINE MK NO.	DESIGN PRESS.(PSIG)	DESIGN TEMP(°F)	TEST PRESS.(PSIG)
FEEDWATER			
1	1250	400	1875
DEMINERALIZED WATER			
2	100	70	150

NOTES:

1. FOR GENERAL NOTES, REFERENCE DRAWINGS AND SYMBOLS SEE 2-47E610-43-1 AND 2-47E822-1.
2. ALL RADWASTE SYSTEM SAMPLES ARE COLLECTED IN A HOODED SINK AND DRAINED TO THE RADWASTE BUILDING CHEM DRAIN TANK.
3. ALL VALVES ARE PREFIXED "2-43-" UNLESS OTHERWISE NOTED.
4. ALL INSTRUMENTS ARE PREFIXED "2-" UNLESS OTHERWISE NOTED.
5. FOR MILIPORE FILTER DETAILS, REFER TO DWG 0-47W448-10.

REFERENCE DRAWINGS:

2-47E610-70-1..........MECHANICAL CONTROL DIAGRAM - REACTOR BUILDING CLOSED COOLING WATER SYSTEM
2-47E610-78-1..........MECHANICAL CONTROL DIAGRAM - FUEL POOL COOLING AND DEMINERALIZER SYSTEM
2-47E822-1.............FLOW DIAGRAM REACTOR BLDG - CLOSED COOLING WATER
2-47E844-1,-2..........FLOW DIAGRAM RAW COOLING WATER
2-47E610-3-2...........MECH CONT DIAG REACTOR FEEDWATER SYSTEM
2-47E610-35-1..........MECH CONT DIAG GENERATOR COOLING SYSTEM
2-47E610-24-1C.........MECH CONTROL DIAGRAM RAW COOLING WATER SYSTEM

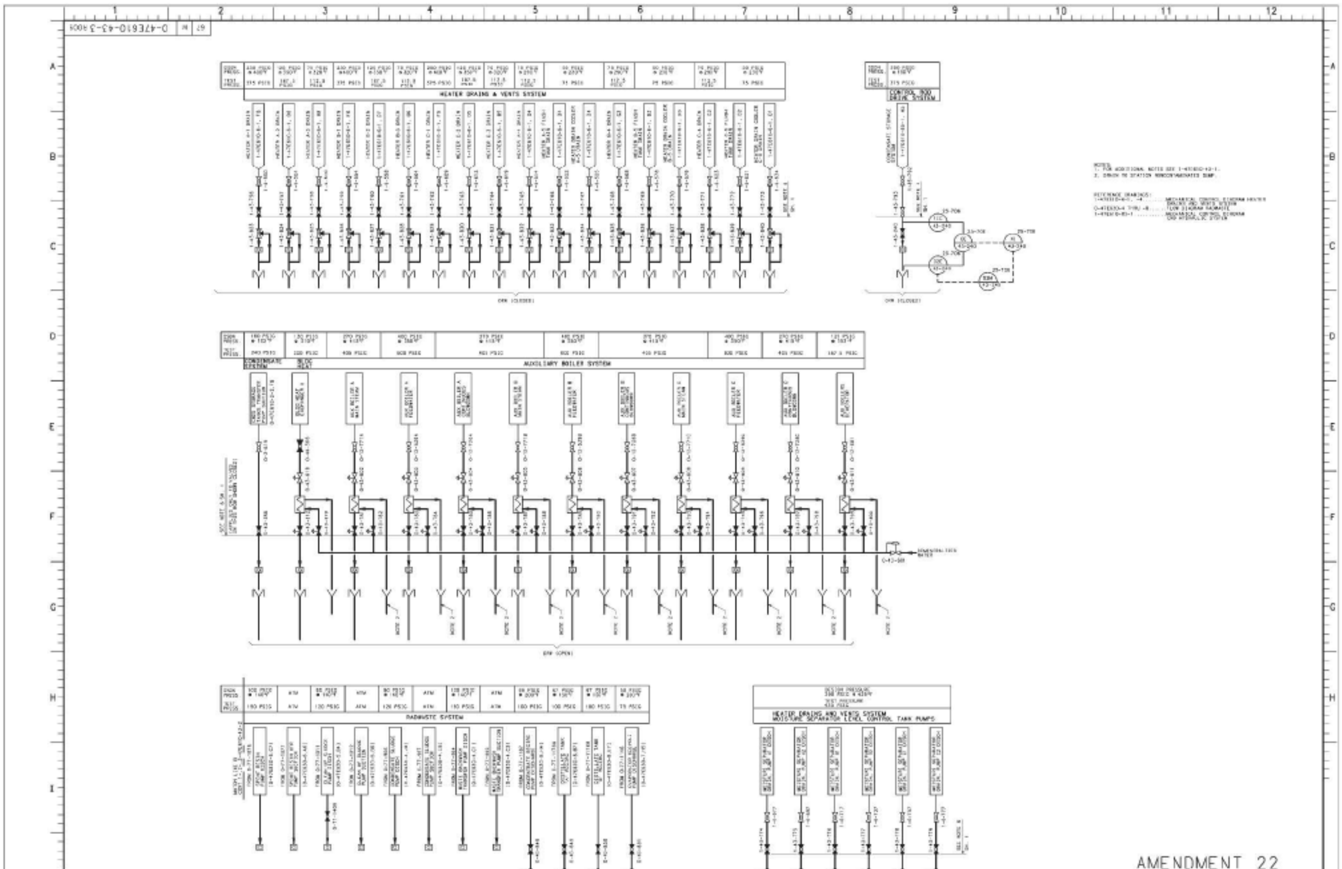
AMENDMENT 28

POWERHOUSE
UNIT 2

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

SAMPLING &
WATER QUALITY SYSTEMS
MECHANICAL CONTROL DIAGRAM

FIGURE 10.17-1b



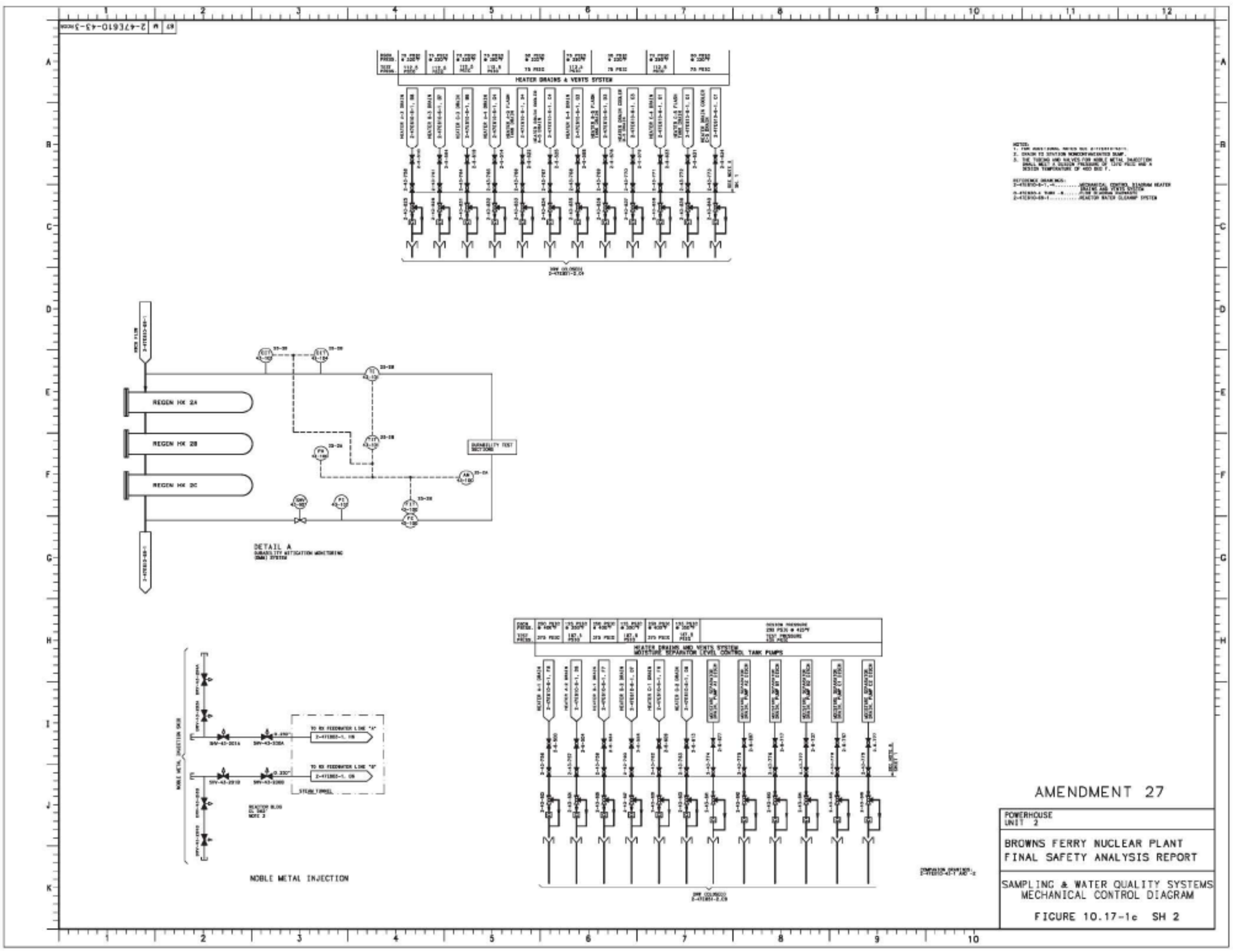
AMENDMENT 22

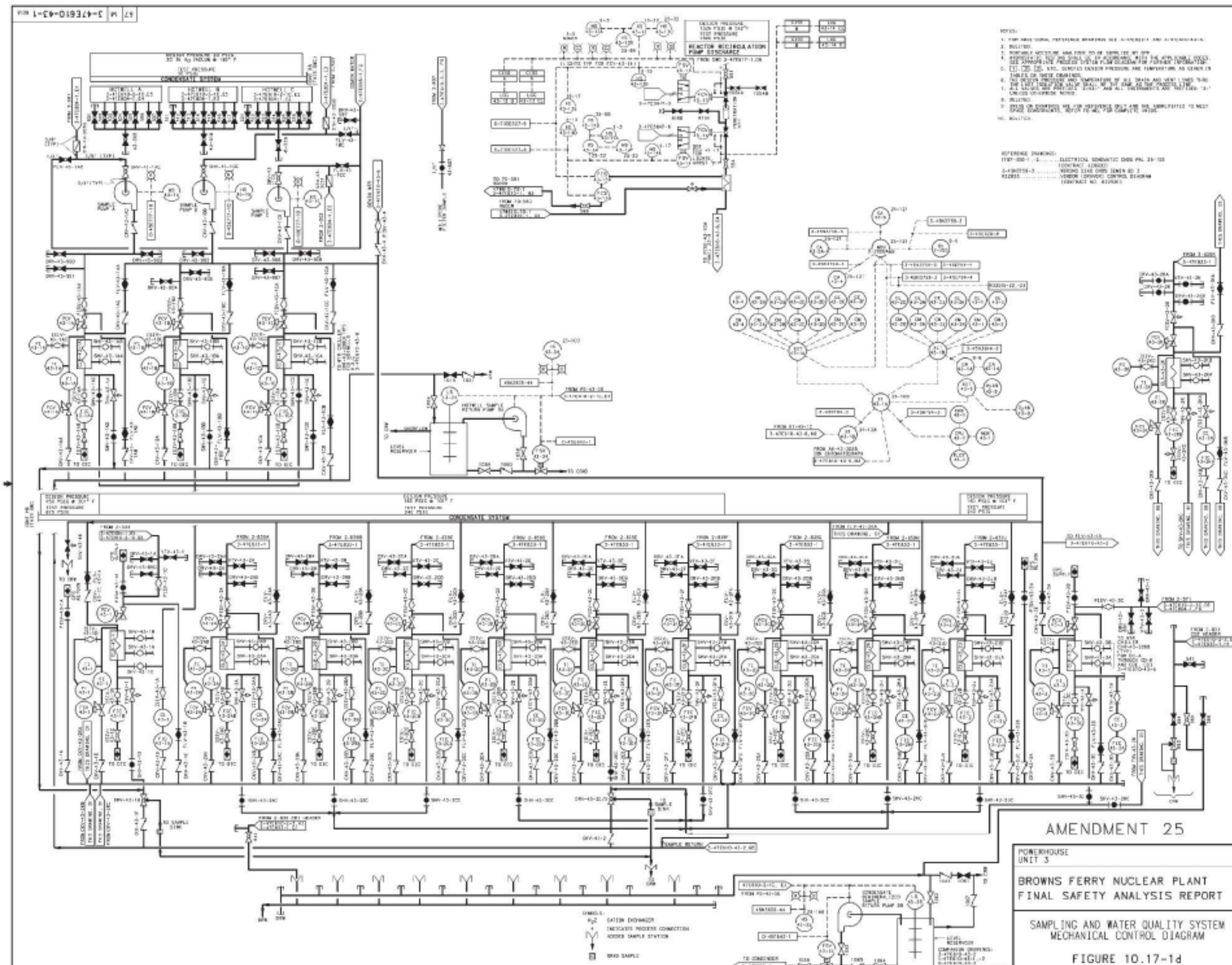
POWERHOUSE
UNITS 1 & 0

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

SAMPLING AND WATER
QUALITY SYSTEM
MECHANICAL CONTROL DIAGRAM

FIGURE 10.17-1c SH 1


HEATER DRAINS & VENTS SYSTEM
REGEN HX 2A
REGEN HX 2B
REGEN HX 2C
DETAIL A
NOBLE METAL INJECTION
AMENDMENT 27
POWERHOUSE
UNIT 2
BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT
SAMPLING & WATER QUALITY SYSTEMS
MECHANICAL CONTROL DIAGRAM
FIGURE 10.17-1c SH 2

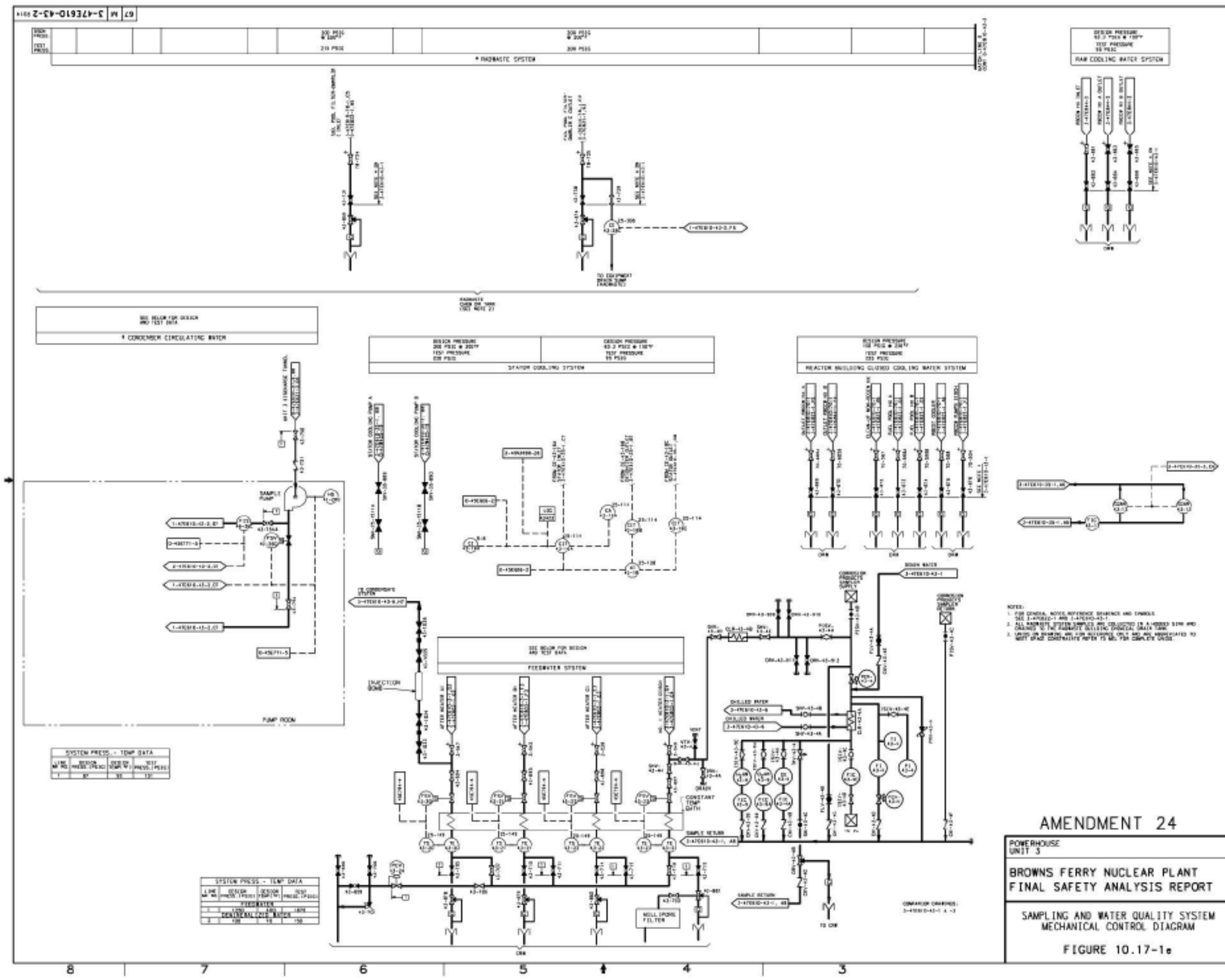


POWERHOUSE
UNIT 3

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

SAMPLING AND WATER QUALITY SYSTEM
MECHANICAL CONTROL DIAGRAM

FIGURE 10.17-1d



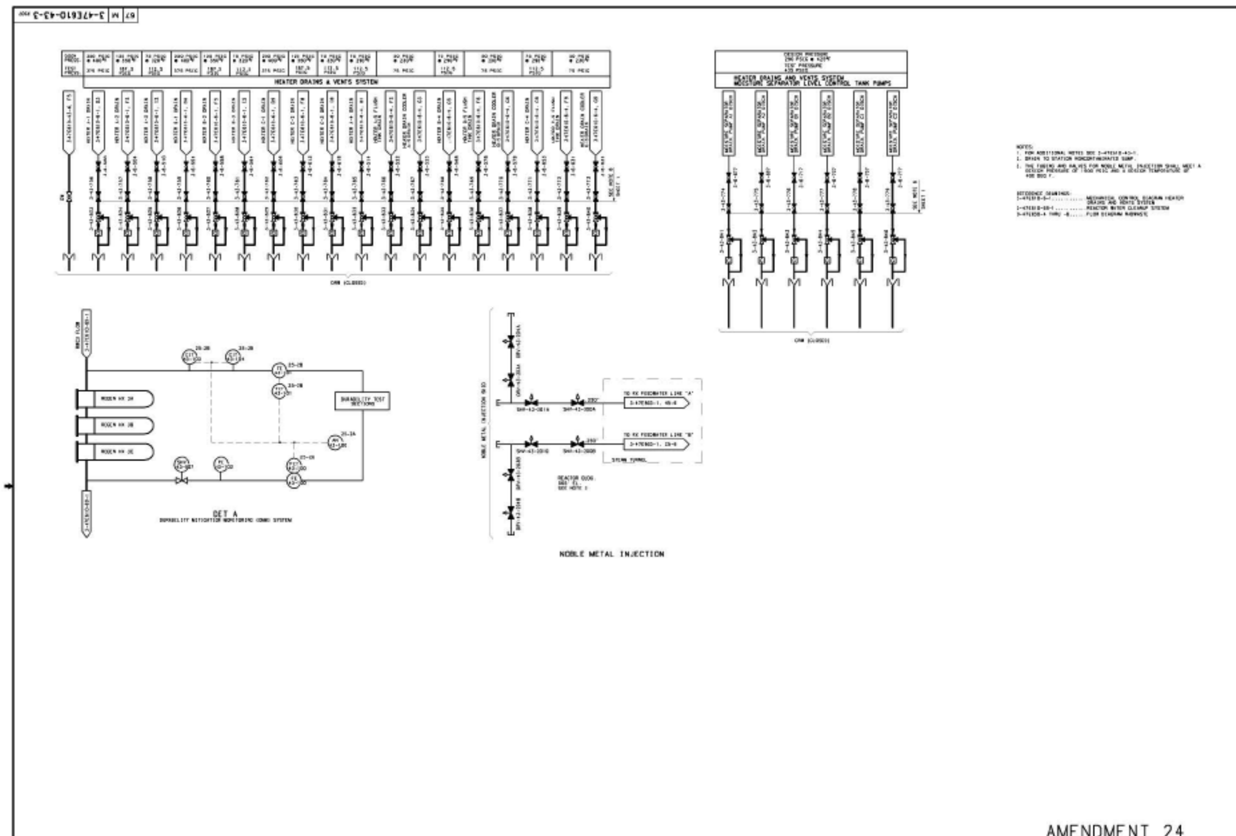
AMENDMENT 24

POWERHOUSE
UNIT 3

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

SAMPLING AND WATER QUALITY SYSTEM
MECHANICAL CONTROL DIAGRAM

FIGURE 10.17-1e



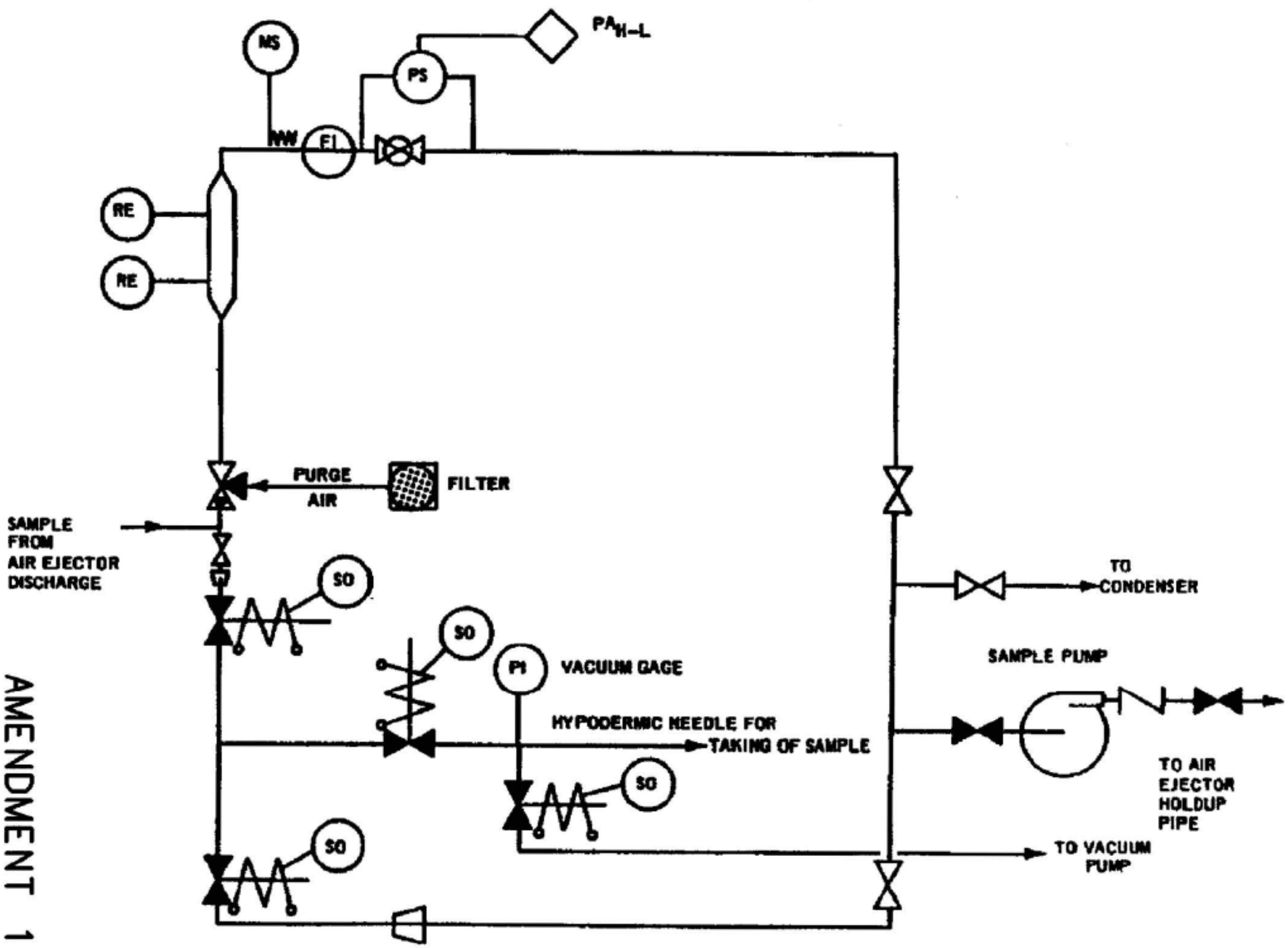
AMENDMENT 24

POWERHOUSE
UNIT 3

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

SAMPLING AND WATER QUALITY SYSTEM
MECHANICAL CONTROL DIAGRAM

FIGURE 10.17-1f



AMENDMENT 16

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

Air Ejector Offgas Sampler
FIGURE 10.17-2

10.18 PLANT COMMUNICATIONS SYSTEM

10.18.1 Design Basis

The design of the plant communications system is such that the requirements for safe operation and safe shutdown are met and that compliance with the Radiological Emergency Plan is adequate. Plant communications are divided into two areas: onsite and offsite. The evacuation alarm is discussed in Section 10.18.6.

10.18.2 Onsite Communications - General

The primary onsite communications needs are furnished by the telephone switching system, the in-plant radio system which includes paging, the loudspeaker paging system, and an independent sound-powered telephone system which can be used to shut down the reactor units when the main control room has been abandoned and other communications systems are inoperative.

10.18.2.1 Onsite Communications - Specific

10.18.2.1.1 Telephone Switching System

The telephone switching system provides the primary means for two-way voice communications throughout the site. Multiple nodes or switches are interconnected by a network of copper and fiber optic cable and are powered by battery-backed supplies. This concept lends itself to a relatively easy expansion or contraction of capacity as dictated by changing plant needs. Node 1 primarily serves the powerhouse area and the other switching nodes primarily serve office/administrative type areas. The system is capable of providing service for several thousand voice and data lines and is equipped with provisions for the following:

- Regular two-way voice conversations
- Data circuits
- Access to loudspeaker paging
- Access to radio paging
- Access to off-site telephone lines
- Various convenience options
- Code Call (Fire and Medical Emergency Alarm)

The power for the Node 1 telephone switch is supplied from two independent 120V AC sources. One of the sources is an Uninterruptible Power Supply (UPS) and diesel backed. The battery is capable of providing power for a minimum of three hours should both power sources fail. Each of the other switching nodes is powered by a UPS. The switchover from primary source to battery is automatic for all nodes

10.18.2.1.2 Sound-Powered Telephone Systems

Backup Control Center Sound-Powered System

The backup control center system consists of two completely redundant systems. Each subsystem is wired directly and independently of any other communication system. Wiring routes avoid the spreading room, unit control rooms, and auxiliary instrument room. Sound-powered equipment and circuits are provided in the diesel generator building, electrical board rooms "A" through "F" in the reactor building, Units, 1, 2, and 3 backup control centers, and reactor core isolation cooling control panel 25-31 in each unit area.

There are telephones equipped with magnetos and magneto howlers provided in each system. The selector switch on the calling telephone is placed on the assigned number of the called telephone and then the magneto on the calling telephone is operated to sound the howler on the called telephone. There is no signaling on the jacks in this system, but jacks are near enough to telephones so that the howler on the telephone can be heard.

RADCON Sound-Powered System

A circuit is provided between the RADCON Laboratory, the Units 1 and 2 control rooms, and the Unit 3 control room. Two pairs are provided between these locations--one for magneto signaling and one for talking over sound-powered handsets.

10.18.2.1.3 Inplant Radio Systems

Nuclear Security VHF Radio System

The Nuclear Security Service (NSS) Radio System is part of the UHF/VHF trunked radio system. A system is provided for communication between nuclear security officers, the central alarm station, the secondary alarm station, and Nuclear Security personnel located within the BFN reservation. Coverage includes the reservation and plant areas. The level of coverage is obtained through an inplant repeater system and inplant/external antenna systems that direct transmissions to portable radios.

Inplant Radio System

The Inplant Radio System consists of a UHF/VHF trunked system and an independent VHF channel (F4). The UHF/VHF trunked system and F4 channel are accessed by handheld radios. F4 channel is also accessed by hard-wired remote control units in various areas of the plant. The trunked system allows individual users to integrate large groups into specific talk groups for coordination (i.g., Fire

Brigade, Operations, Maintenance, and Nuclear Security). The UHF/VHF trunked system consists of 6 UHF channels and 2 VHF channels. The power supply to UHF/VHF trunked system consists of two, redundant systems backed by emergency diesel generators. The four UHF channels of trunked systems are backed by an uninterruptible power supply (UPS). One channel is assigned as a control channel. Three channels are available for voice communication. The F4 channel is supplied by a separate uninterruptible power supply (UPS).

10.18.2.1.4 Paging System (Loudspeaker)

Paging speakers are installed throughout the Reactor, Turbine, and Control Buildings.

This equipment may be accessed by telephone. The speaker-amplifiers in each unit area are fed in parallel from a local AC lighting source.

10.18.2.1.5 Radio Page System

A radio page system is provided to signal key plant personnel throughout the BFN reservation. The telephone system is equipped with signaling trunks to permit the dialing of assigned radio page numbers from any plant telephone.

10.18.3 Offsite Communications - General

The offsite communications are accomplished by several diverse means, each having a redundant pathway. Dedicated telephone lines link the plant to the major administrative offices in Chattanooga for daily plant operations, and to the Central Emergency Control Center (CECC) to comply with the Radiological Emergency Plan (REP). An optical fiber based communications system serves as a redundant path for these lines. The plant also interfaces with the public switched network which provides local and long distance access. The emergency radio network and the Transmission Power Supply (TPS) radio serve as a voice pathway to the area surrounding the plant site. The offsite radio paging serves the same area for contacting key personnel carrying pagers.

10.18.3.1 Offsite Communications - Specific

10.18.3.1.1 Public Telephone Network

The public telephone system circuits enter the plant through a buried fiber-optics transmission cable. Power for the fiber-optics interface equipment is 120-V AC. A battery provided by an offsite vendor powers the system upon loss of AC power. Connections to the plant telephone system are made in the communications room.

Several types of connections are provided into the plant. These include:

Direct inward dial trunks;
Direct outward dial trunks;
Dedicated tie lines to Chattanooga TVA telephone network;
Dedicated tie lines to Chattanooga CECC.

10.18.3.1.2 Optical Fiber Based Communications

An optical fiber based communications system from the Athens Substation to the Node 3 Building is provided as a redundant means of providing communications circuits to the Chattanooga Administrative Telephone Network and the TVA Direct Distance Dial Network.

10.18.3.1.3 Emergency Radio System

This system is provided to communicate with field radiological monitoring teams, provide backup communication to the Chattanooga CECC, and provide mobile communications for key plant managers. Access to the system is provided from the main control room, the RADCON laboratory, the technical support center, the Operations support center, and the plant simulator.

Two independent repeaters are provided, accessing redundant, spatially diverse VHF transmitters. Field radiological monitoring vans are provided with VHF transceivers. The approximate area of coverage extends fifty miles from the plant.

10.18.3.1.4 Transmission Power Supply (TPS) UHF Radio System

This system is provided to assist the coordination of power transmission system control and maintenance activities. Control stations are provided in the technical support center (TSC) and the main control room. This system is independent of all other interplant communications systems and may be used in an emergency to communicate with the CECC in Chattanooga.

10.18.3.1.5 Sheriff's Radio System

A UHF radio is provided for communication between BFN Nuclear Security and the Limestone County Sheriff's Department. Access is provided from the Central Alarm Station and the Secondary Alarm Station.

10.18.3.1.6 Radio Paging System

This system is used to signal an employee carrying a pager in the surrounding areas of the plant. The range is approximately the same as the emergency radio system and the operation is the same as the onsite paging.

10.18.3.1.7 (Deleted)

10.18.3.1.8 NRC EMERGENCY NOTIFICATION SYSTEM (ENS)

The NRC Emergency Notification System (ENS) provides dedicated communication lines for emergency communication functions.

10.18.4 Evaluation

Onsite Systems

The telephone switching system, which is one of the primary onsite systems, is designed so that failures in individual processors or lines do not interrupt service. Such failures are annunciated and repairs are made promptly. The Node 1 switching equipment, which primarily serves the powerhouse area, is located in the communications room which is in a Seismic Class I building.

The sound-powered telephone systems are completely independent of power, each other, and all other systems provided. As long as a complete metallic path exists between instruments, communications can be maintained since the instruments supplied with these systems are very rugged and will successfully withstand high shocks, negligence, and abuse. If permanently installed wires are rendered unusable for any reason, a temporary pair of wires can be used with the sound-powered instruments.

The Nuclear Security radio must provide continuous coverage from all areas of the plant to the Central Alarm Station (CAS) and the Secondary Alarm Station (SAS). The Sheriff's radio is a separate system with radios in the CAS and in the SAS and a transmitter in the turbine building to send to the Limestone County repeater.

The Transmission Power Supply (TPS) radio is located in the Telecommunication & Computer Center (Node 3) Building with a remote terminal unit in the TSC and in the Main Control Room. This radio operates on 120-V AC lighting power feed and is backed by a 48-V DC source in the T&CC Building. This radio system keys the repeater at Monte Sano Mountain which provides communication to the Transmission Dispatch Control Center in Chattanooga. The transmitter at the Monte Sano repeater is primarily powered by 120-V AC and emergency backed by motor generator.

The loudspeaker paging equipment is dispersed in the control building and powerhouse areas. Single or multiple open circuits or amplifier failure in individual units will not prevent the remaining equipment from functioning.

Both onsite and offsite radio paging are accomplished through a fully redundant paging terminal located in Chattanooga and powered from an uninterruptable power supply. Access is provided by dedicated circuits with diversely routed backups.

Offsite Systems

Commercial telecommunications circuits enter the plant through a buried cable. The demarcation point is in a hut provided by an offsite vendor just outside the security fence. If this cable was cut, an alternative offsite route would be the TVA fiber optic system. Both systems are redundantly powered.

The Emergency Radio System operates with either of two separate and spatially diverse repeater sites. These sites are redundant with respect to operating area and access to the CECC.

10.18.5 Inspection and Tests

The most effective test of the telecommunications at the plant is the constant daily usage of this system. User troubles are reported and repaired promptly. In addition to daily usage, the Emergency Preparedness Staff performs periodic checks of the TSC telephones.

10.18.6 Accountability Alarm System

Plant-wide siren coverage is provided for signaling accountability to the plant personnel. These sirens are operated in the undulating mode for accountability.

There are two completely separate control stations provided for the system and also redundant automatic timers. The timers may be manually bypassed in case both fail.

The motor-driven sirens operate in groups by separate contactors, each of which has two diesel-backed, 480-V AC power supplies and two redundant actuating relays.

The electronic sirens and strobe lights operate in groups. Each group has two redundant actuating relays and is powered from an uninterruptible power supply. Uninterruptible power supply 0-BUPS-244-0004 is disabled by TACF 0-08-003-244. Uninterruptible power supply 1-BUPS-244-0001 is disabled by TACF 1-08-007-244. Uninterruptible power supply 1-BUPS-244-0002 is disabled by TACF 1-08-008-244. Uninterruptible power supply 2-BUPS-244-0002 is disabled by TACF 2-08-007-244.

The design of the siren system is such that it will not likely be inoperative for the following reasons.

1. Two independent, widely separated operating centers are provided.

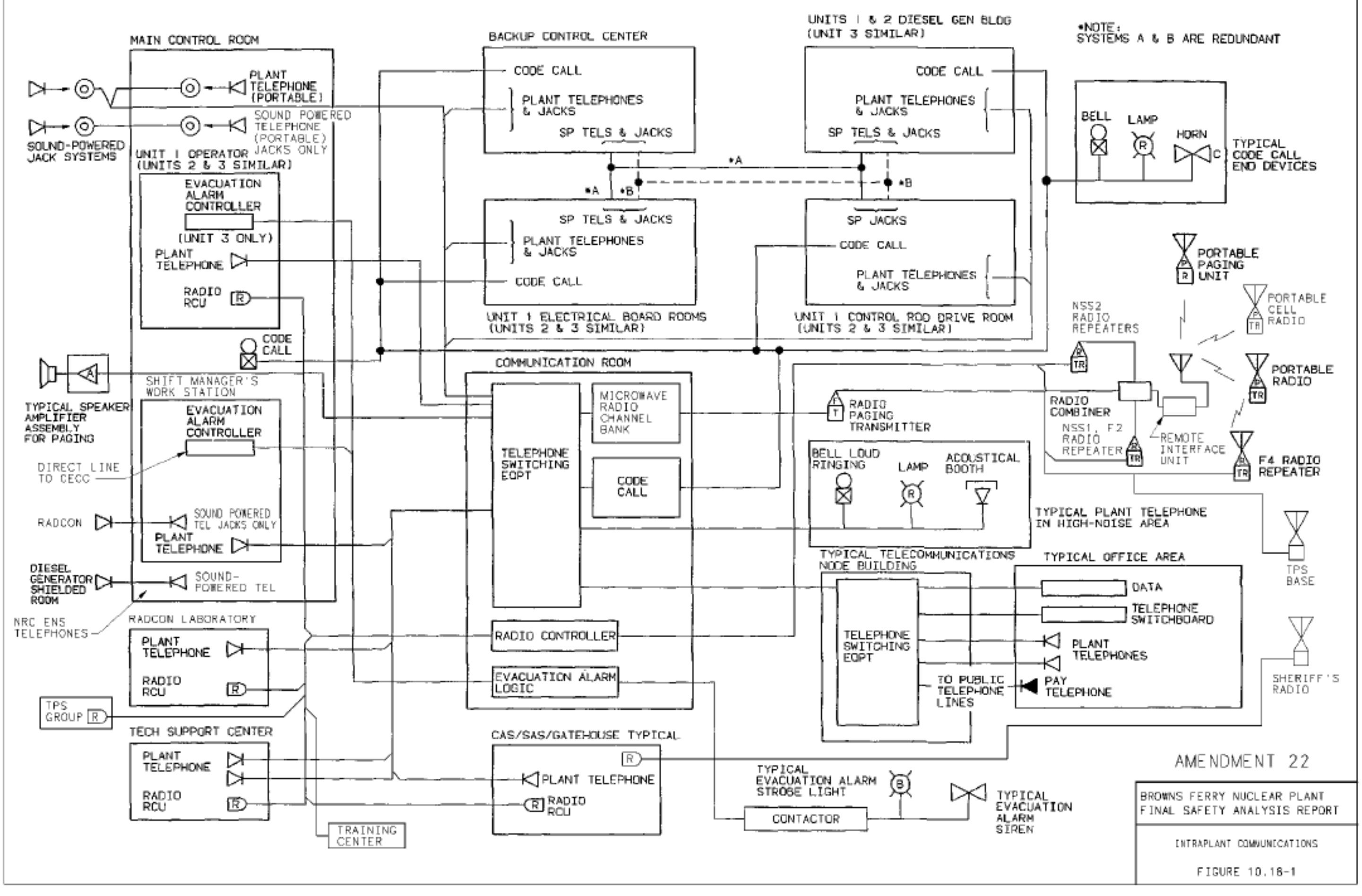
2. Duplicate timers located in widely separated bays are furnished with provision for manual override in case both fail. Each timer is powered by a separate ac source.

3. Duplicate actuating relays are provided in each remote control unit.

4. Independent, remote-control units are provided, each controlling a group of sirens or sirens and strobe lights. Failure of one control unit will not affect the others.

5. Two sources of diesel-backed AC power are provided for each control unit, with provisions for annunciation upon failure of each source. The control units that control the sirens and strobe lights are supplied from an uninterruptible power supply.

6. Power failure to the timers and to the remote control unit actuating relays is annunciated.

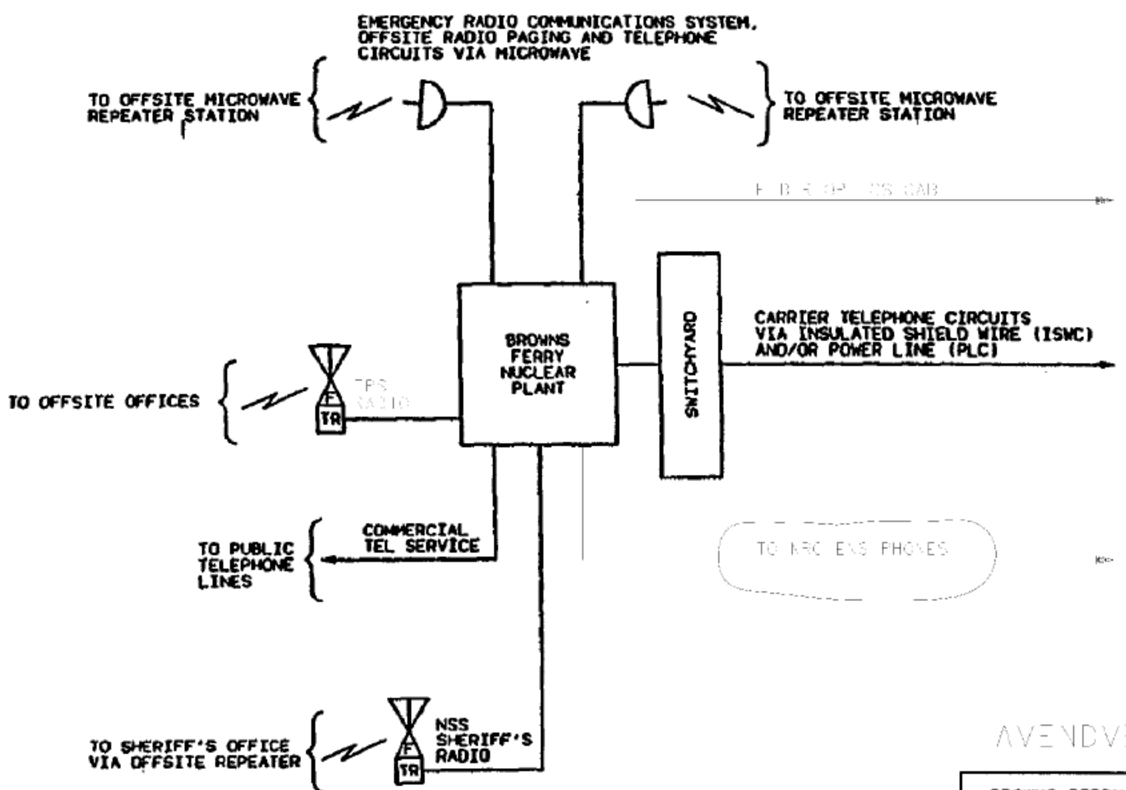


AMENDMENT 22

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

INTRAPLANT COMMUNICATIONS

FIGURE 10.16-1



AMENDMENT 19

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

PLANT TO OFFSITE COMMUNICATIONS

FIGURE 10.18-2

Figure 10.18-3

Deleted by Amendment 9.

POSTULATED CONDITIONS	OFFSITE COMMUNICATIONS									ONSITE COMMUNICATIONS									
	NRC ENS PHONES	MICROWAVE RADIO (MW)		COMMERCIAL TELEPHONE LINES	TPS RADIO		EMERGENCY RADIO SYSTEM	NSS SHERIFF'S RADIO	OFFSITE RADIO PAGING	NSS1, NSS2 RADIO (NSS2 BACKUP)	ONSITE RADIO PAGING	CELL RADIO (NOTE 5)	INPLANT RADIO SYSTEM (F2)	INPLANT RADIO SYSTEM (F4)	TELEPHONE SYSTEM (NODES 1 2 & 3)	LOUDSPEAKER PAGING SYSTEM	SOUND-POWERED SYSTEM (BCCSPS)	EVACUATION ALARM SYSTEM	CODE CALL (FIRE & MEDICAL EMERGENCY)
FIRE IN COMMUNICATION ROOM (TOTAL DESTRUCTION)								x		x		x	x	x	NODES 2 & 3 SURVIVE		x		
FIRE IN CABLE TUNNEL TO SWITCHYARD		x					x	x	x	x	x		x	x	PARTIAL	x	x	PARTIAL	PARTIAL
FIRE IN CONTROL ROOM	PARTIAL	PARTIAL		PARTIAL			x	x	x	x	x	x	x	x	PARTIAL	PARTIAL	PARTIAL	PARTIAL	PARTIAL
LOSS OF ALL AC POWER	x	x		x	x					NSS2 SURVIVE		x	x	x	x		x		
DESIGN BASIS ACCIDENT	x	x		x	x		x	x	x	x	x	x	x	x	x	x	x	x	x
SAFE SHUTDOWN EARTHQUAKE										MOBILES & PORTABLES SURVIVE		SEE NOTE 4	SEE NOTE 3	SEE NOTE 3					
LOSS OF OFFSITE POWER	x	x		x			x	x	x	x	x	x	x	x	x		x	x	
PROBABLE MAXIMUM FLOOD		x					x	x	x	x	PARTIAL	x	PARTIAL	x	PARTIAL		x	PARTIAL	PARTIAL
TORNADO (MICROWAVE ANTENNAS & REFLECTORS DESTROYED)	x			x	x			x	x	x	x	SEE NOTE 4	x	x	X LOSS OF MW TRUNKS	x	x	x	x
FIRE IN TURBINE BUILDING (TOTAL DESTRUCTION)	x			x								SEE NOTE 4		x	PARTIAL	PARTIAL	x	PARTIAL	PARTIAL
FIRE IN REACTOR BUILDING FIRE ZONE 2-4	x	x		x	x		x	x	x	x	x	x	x		PARTIAL	x	PARTIAL	PARTIAL	PARTIAL

NOTES.

1. X IN BLOCK INDICATES SURVIVAL OF THE SERVICE DURING THE POSTULATED CONDITION
2. PARTIAL IN BLOCK INDICATES PARTIAL AVAILABILITY DUE TO THE LOSS OF THAT PORTION OF THE SYSTEM WHERE THE CONDITION EXISTS. THE SURVIVING EQUIPMENT WILL REMAIN FUNCTIONAL.
3. MOST PORTABLE UNITS WILL SURVIVE AND WILL OPERATE IN SIMPLEX MODE
4. CELL RADIO WILL OPERATE RADIO TO RADIO ON 900MHZ BAND ONLY.
5. AVAILABLE BUT NOT CREDITED.

AMENDMENT 22

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

COMMUNICATION EQUIPMENT AVAILABILITY TABLE

FIGURE 10.18-4

Figures 10.18-5a through 10.18-5b

Deleted by Amendment 9.

Figure 10.18-6

Deleted by Amendment 9.

10.19 LIGHTING SYSTEM

The plant lighting system is comprised of the normal, standby, and emergency lighting subsystems. The normal lighting subsystem is fed from 480-V three phase lighting board panels through 480-240/120-V single phase lighting transformers. The normal lighting subsystem provides adequate illumination during normal plant operation. This common system is ordinarily powered from the Normal Auxiliary Power System through the 480-V Station Service System (see UFSAR Figure 8.4-1a).

The standby lighting subsystem is supplied power from the Standby Auxiliary Power System and provides a dual purposes light source. Specifically, its primary function is to provide adequate, long term duration, diesel-backed AC lighting for personnel safety and continuity of essential functions in the absence of the normal lighting subsystem. It has a secondary function of supplementing the normal lighting subsystem. The standby lighting subsystem is supplied power through 480-240/120-V single phase transformers and is diesel-backed.

The emergency lighting subsystem is comprised of a 250-V DC emergency lighting system and individual battery pack lights used to supplement the 250-V DC emergency lighting system. A portion of the individual battery pack lights are required to meet fire protection requirements in accordance with the appropriate design criteria. The 250-V DC emergency lighting system is supplied power from 240-V AC lighting boards. In the event of an AC power supply failure, the emergency lights automatically energize or transfer to their DC power source.

Standby and/or emergency lighting is provided in critical areas of the Control Bay, Reactor, Turbine, Diesel Generator, Radwaste, and Standby Gas Treatment Buildings.

Local 480-240/120-V single phase transformers supply normal lighting power for remote areas including the circulating water intake structure. Emergency lighting for these remote areas is supplied from 120-V AC preferred service power.

10.20 AUXILIARY BOILER SYSTEM

10.20.1 Power Generation Design Basis

The Auxiliary Boiler System shall be designed to supply steam for the following:

a. Building heating
b. HPCI and RCIC testing
c. Steam seal regulator at startup
d. Condenser hotwell deaeration and heating at startup (Unit 1 and Unit 3)
e. Steam jet air ejector (SJAE) operation at startup
f. Radwaste evaporator (in place but not generally used)
g. Offgas preheater at startup
h. N_2 evaporator
i. Sellers jet
j. Pegging steam for auxiliary deaerator

10.20.2 Description

There are three shop-assembled, water-tube, natural-circulation boilers designed for pressure firing with oil. Each boiler has a rated capacity of 50,000 lb/hr at 250 psig saturated steam (406°F). Flue gases from the boilers are discharged into a common steel stack extending above the Reactor Building roof. The three boilers may be operated individually or simultaneously depending on steam requirements.

The auxiliary boilers provide the only steam source to the powerhouse building heating system. They also provide steam to the turbine seal system and the SJAE when the nuclear steam supply is incapable of furnishing adequate steam to these systems, such as during cold startup, low power reactor operations and main steam isolations. The auxiliary steam system may be used in the startup phase for deaeration of the condensate in the condenser hotwell, should the condensate oxygen level exceed startup limits, (Unit 1 and Unit 3) and for deaeration of the radwaste evaporation system.

Steam supplied to the building heating system is returned as condensate to the Auxiliary Boiler System. Condensate from several other uses of the Auxiliary Boiler System directly contributes to the condensate inventory of the reactor system.

Makeup water for the Auxiliary Boiler System is provided by the Demineralized Water System.

Removable pipe sections and blanking flanges are provided in the steam supply lines to the HPCI and RCIC turbines, and a check valve and a gate valve are provided in the steam lines to the SJAE and the steam seal regulator to assure that

no radioactive steam or condensate can backfeed to the auxiliary boiler when the lines are not in use. This system provides a common means to test the HPCI and RCIC steam turbines at low pressure. Blanking flanges are provided in the deaeration lines to the Unit 1 and Unit 3 condenser hotwells to assure that the chemicals used to treat the boiler feedwater do not leak into the condenser hotwell. The function to provide deaeration to the condenser hotwell is no longer utilized for Unit 2. The addition of a temporary spool piece is required for the use of the Unit 1 and Unit 3 deaeration line. These operations are not critical in a time sense. The outage of this system does not influence the capability of the plant to shut down in any mode. Instrument and test connections throughout the system facilitate operation and performance testing.

10.21 POSTACCIDENT SAMPLING SYSTEM

10.21.1 Power Generation Objective

The power generation objective of the postaccident sampling system for BFN Unit 2 and Unit 3, is to provide representative samples of reactor coolant, torus liquid, drywell atmosphere, torus atmosphere and secondary containment atmosphere after a LOCA and during testing and training.

The power generation objective of the postaccident sampling system for BFN Unit 1 is to provide representative samples of reactor coolant, torus liquid, drywell atmosphere, and torus atmosphere after a LOCA and during testing and training.

10.21.2 Power Generation Design Basis

PASS is designed to fulfill the requirements of NUREG-0737, Item II.B.3 (Units 2 and 3 only), and to be able to obtain and analyze the required samples at any time without exceeding the personnel exposure limits of GDC 19 in Appendix A of 10CFR50 (Units 1, 2, and 3). In addition, the PASS shall be available to perform the following tasks:

1. To safely cope with and obtain fluid samples contaminated by a Regulatory Guide 1.3 release and using source terms determined by TID-14844 methodology,

2. To allow the safe transport of grab fluid samples to the on-site laboratory,

3. To provide for the required analysis of fluid samples in accordance with the requirements of Regulatory Guide 1.97, and to provide pertinent data to the operator with adequate accuracy, range and sensitivity to describe the radiological and chemical status of the reactor coolant and containment systems,

4. To safely dispose of the sampling and flushing fluids by returning those to the torus, with the exception of the secondary containment atmosphere which is returned back to secondary containment (Units 2 and 3); to safely dispose of sampling fluids by returning the gas to the torus and the liquid to a floor drain in the residual heat removal (RHR) heat exchanger room (Unit 1).

5. To reduce activity following sampling, by using nitrogen gas to purge gas sample return lines and demineralized water to flush liquid sample return lines (Units 2 and 3),

6. To provide for the packaging of the samples for shipment to an offsite facility for additional or backup analyses,

7. To ensure obtaining representative fluid samples, by maintaining turbulent flow through the lines,

8. To minimize the volume of fluid to be taken from containment, by keeping sample line lengths as short as reasonably achievable, and

9. To provide adequate cooling and ventilation of the sample station, by connecting to a two-inch valved vent pipe included in the penetration and exhausting to the secondary containment for processing through the standby gas treatment system (Units 2 and 3); to provide adequate ventilation of the sample station, by connecting a one and one half-inch valved vent pipe included in the penetration exhausting to the secondary containment (Unit 1).

10.21.3 Description

10.21.3.1 Units 2 and 3

As shown on Figures 10.21-1, 10.21-2, 10.21-3, and 10.21-4, grab samples can be obtained from the reactor coolant, RHR liquid, drywell gas, torus gas and secondary containment atmosphere. From the various sources in the reactor building, the samples are routed to the piping station, through a seismically qualified secondary containment penetration and to the sampling station installed in the turbine building.

Located nearby the sampling station is the PASS control panel which provides readouts for conductivity and radiation level; indications for flow, pressure and temperature; and switches for the various control valves. A graphic display is also provided on the control panel to show the status of the pumps and valves (except for the isolation valves). The control panel is designed for sequential and manual operation. A selector switch is provided which allows selection of either liquid or gas sampling mode. Table 10.21-1 provides a listing of PASS in-line instrumentation.

The sample station consists of a wall mounted frame and enclosures which includes an upper gas sampler and a lower liquid sampler. Lead shielding is provided for the gas and liquid sampling compartments.

The piping station, which is installed in the reactor building, consists of sample coolers and control valves serving the liquid samples.

Primary containment isolation valves with isolation signals from the PCIS are provided for the RHR liquid sample line and the liquid/gas return line to the torus. System isolation valves are also furnished for the torus gas, drywell gas and the RBCCW cooling supply lines. These isolation valves are remote manually controlled from the main control room using 1E control power and 1E qualified components.

Although the reactor coolant sample is taken from the jet pump instrument line downstream from an orifice and an excess flow check valve, a system isolation valve is also provided as a precautionary measure and to allow ease in component maintenance. This isolation valve is remote manually controlled from the main control room using non-1E power.

10.21.3.2 Unit 1

As shown on Figures 10.21-5 and 10.21-6, manual grab samples can be obtained from the RHR liquid, drywell gas, and torus gas. From the various sources in the reactor building, the samples are routed through a seismically qualified secondary containment penetration and to the sampling station installed in the turbine building.

The RHR liquid sample is reactor coolant when the RHR system is in shutdown cooling mode and torus liquid when the RHR system is in suppression pool cooling mode.

The gas sample is obtained via the H_2O_2 analyzer sample return in the containment inerting system. A selector switch on the H_2O_2 analyzer control panel is provided which allows selection of either a drywell or torus gas sample.

The sample station consists of a wall mounted frame and enclosures which includes an upper gas sampler and a lower liquid sampler. Lead shielding is provided for the gas and liquid sampling compartments.

Primary containment isolation valves with isolation signals from the PCIS are provided for the RHR liquid sample line. System isolation valves are also furnished for the gas supply and return lines. These isolation valves are remote manually controlled from the main control room using 1E control power and 1E qualified components.

10.21.4 Safety Evaluation

10.21.4.1 Units 2 and 3

The PASS is not required to support safe shutdown of the plant. Thus, the PASS does not perform a safety function, with the exception of providing primary containment isolation and meeting the requirements for the interface points with the drywell/torus gas sample lines tied-in to the H2/02 analyzer system, RBCCW and the secondary containment penetration. A system isolation valve is provided for the reactor coolant sample line, as an additional safety feature, even though the tie-in point is downstream from an orifice and an excess flow check valve. This design meets the requirements of NUREG 0737, Item II.B.3 and Regulatory Guide 1.97.

10.21.4.2 Unit 1

The PASS is not required to support safe shutdown of the plant. Thus, the PASS does not perform a safety function, with the exception of providing primary containment isolation and meeting the requirements for the interface points with the H_2O_2 analyzer gas sample return line and the secondary containment penetration.

10.21.5 Inspections and Tests

PASS equipment is to be operated at least semiannually (+25%) to ensure the capability to obtain and analyze reactor coolant and primary containment atmosphere samples under accident conditions. This semiannual operation provides adequate familiarity and training for plant personnel to assure operational skills when required.

Table 10.21-1

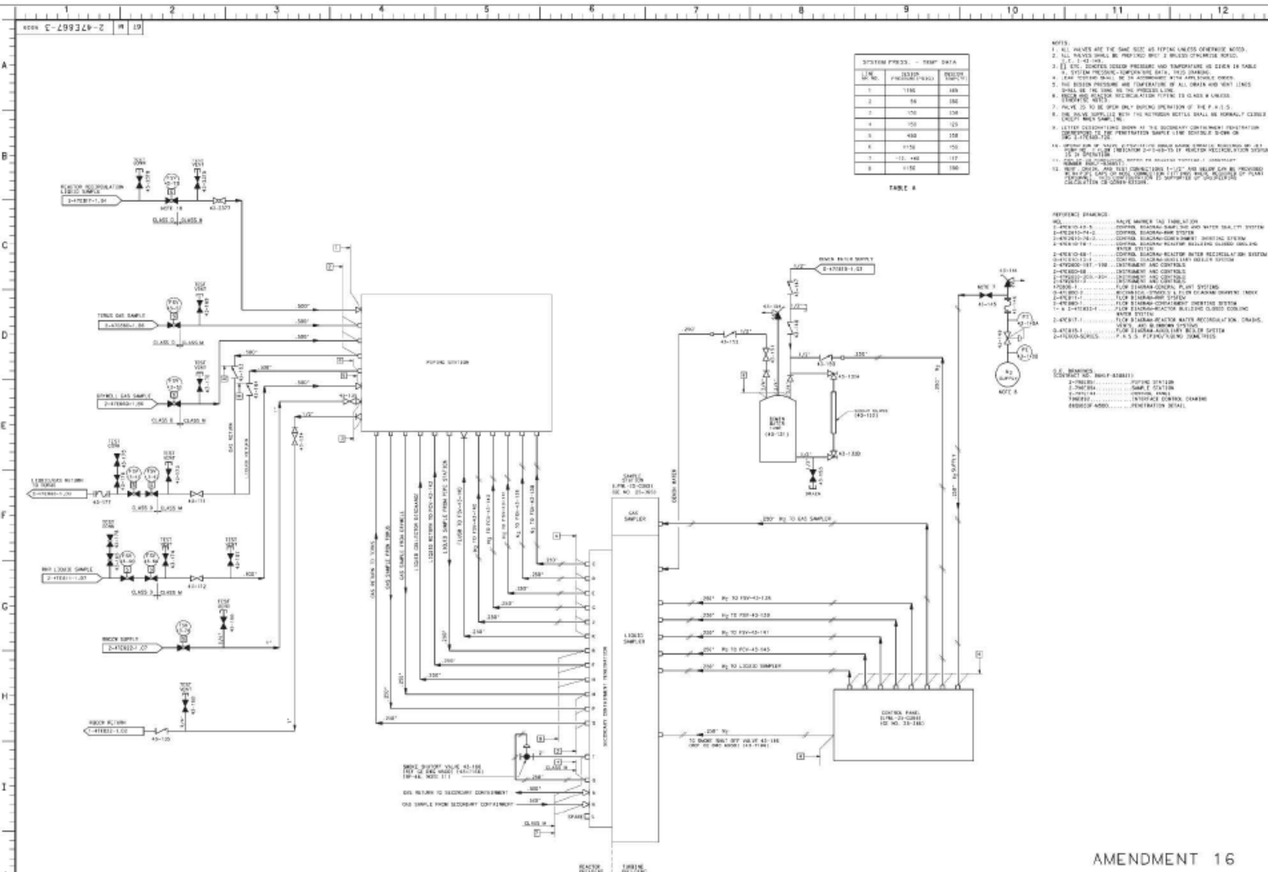
PASS In-Line Instrumentation

Liquid Sample Mode

Instrument	Description	Range	Purpose
FI-664	Flow	0-2 gpm	Line flush flow rate
CI-663	Conductivity	0-10 µs/gpm	Sample conductivity
PI-661	Pressure	0-1200 psig	Sample pressure
PI-662	Pressure	0-150 psia	Dissolved gas collection chamber pressure
RI-665	Radiation	10-100,000 mr/hr	Approximate radiation level of sample line in liquid sampler, used also to verify system flush
TI-660	Temperature	50-150°F	Sample line temperature

Gas Sample Mode

Instrument	Description	Range	Purpose
FI-725	Flow	2-25 slpm	Calibrate pump discharge pressure versus sample flow rate (not used during normal operation
PI-708	Pressure	0-50 psia	Sample pressure
TI-724	Temperature	50-150°F	Sample line temperature



AMENDMENT 16

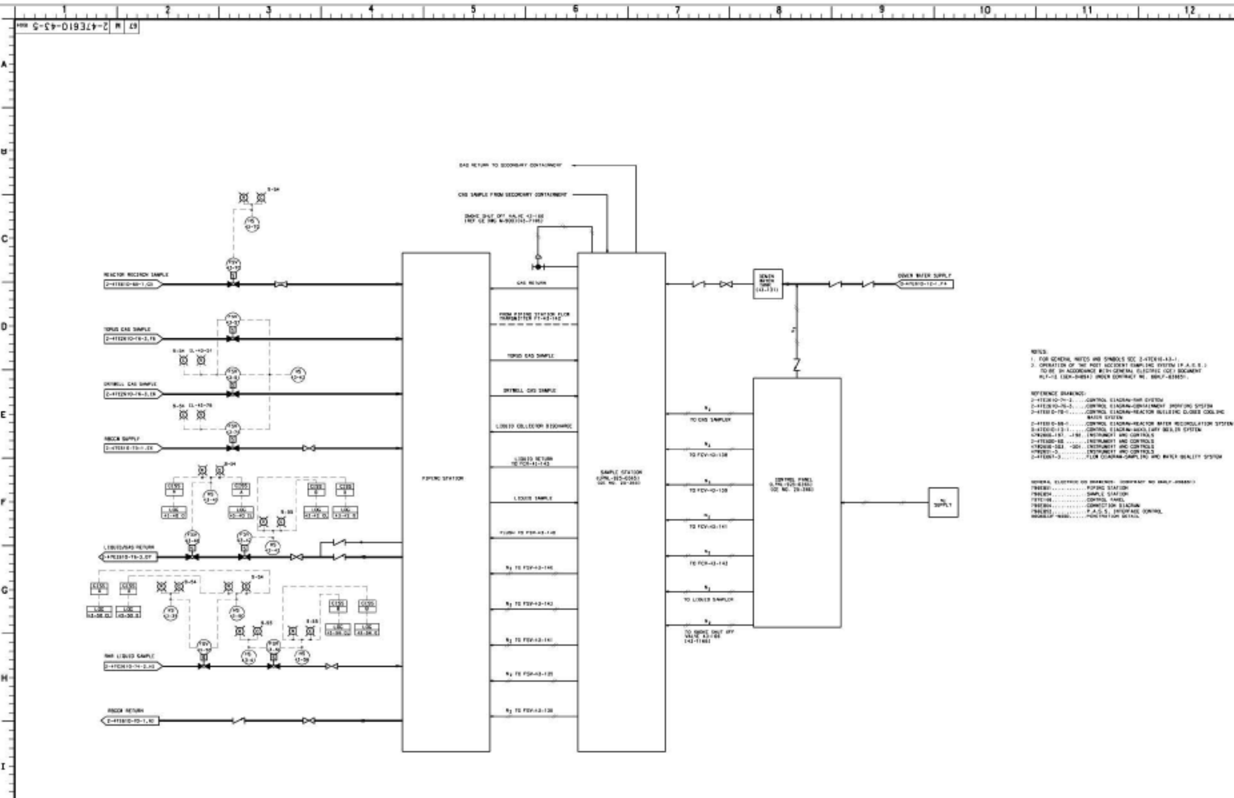
POST ACCIDENT SAMPLING SYSTEM (P.A.S.S.)

POWERHOUSE
UNIT 2

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

FLOW DIAGRAM - PASS

FIGURE 10.21-1



POST ACCIDENT SAMPLING SYSTEM (P.A.S.S.)

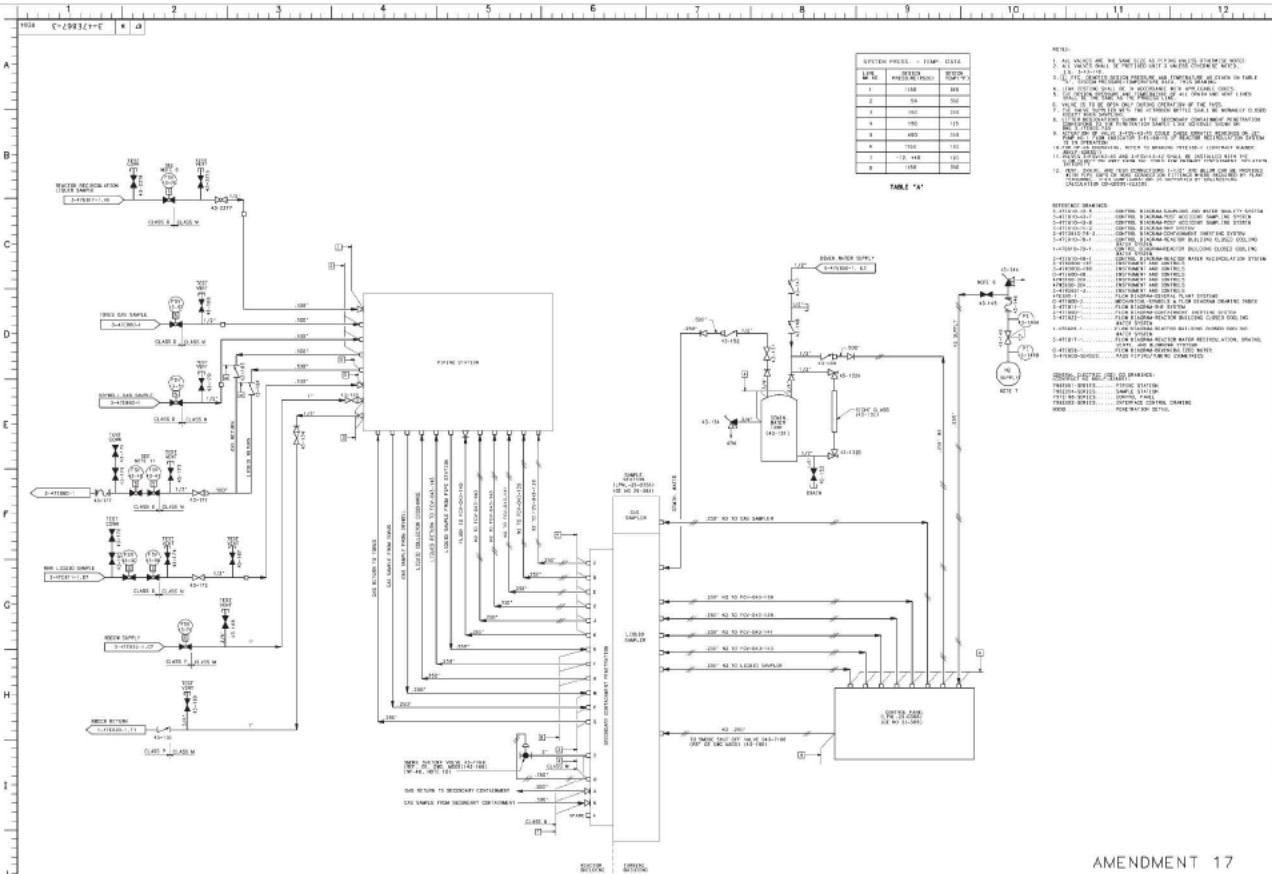
AMENDMENT 24

POWERHOUSE
UNIT 2

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

MECHANICAL CONTROL
DIAGRAM - PASS

FIGURE 10.21-2



POST ACCIDENT SAMPLING SYSTEM (PASS)

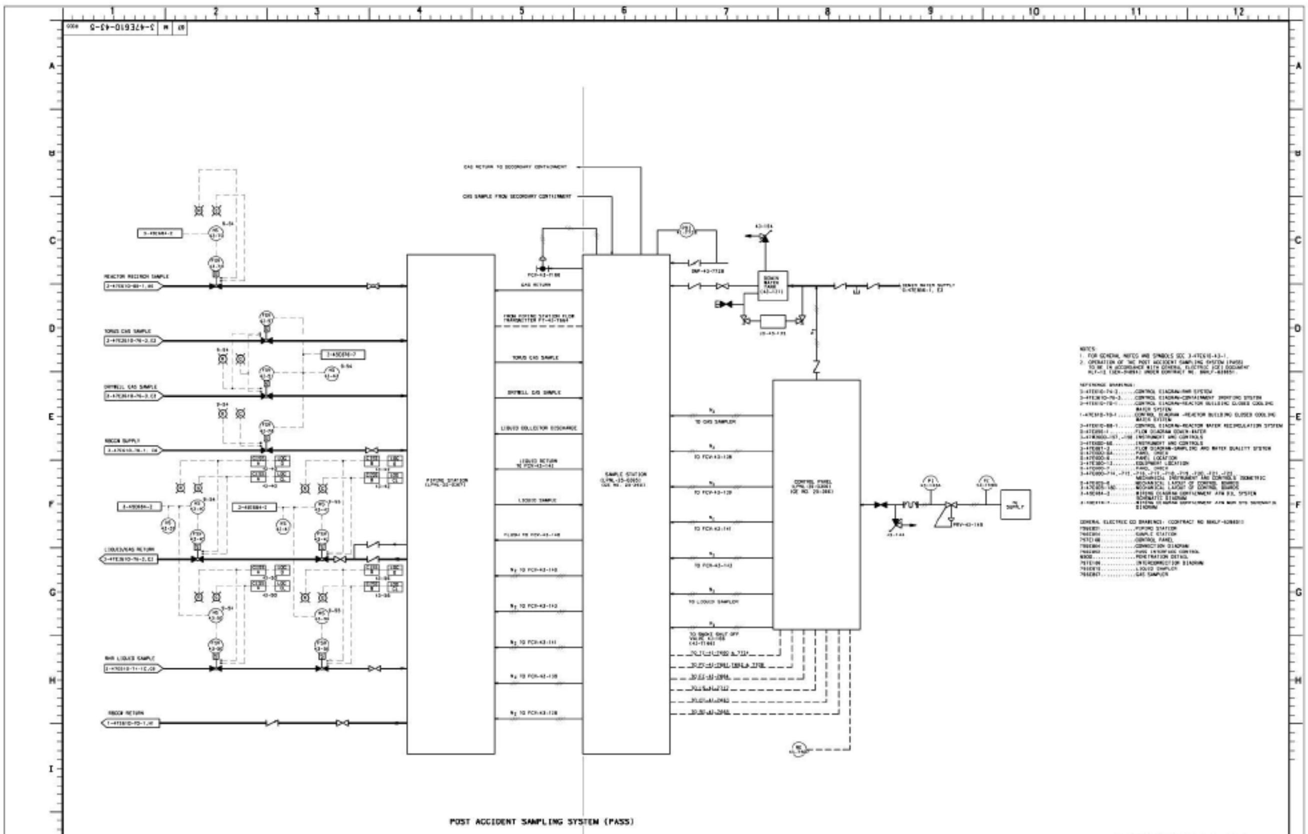
AMENDMENT 17

POWERHOUSE
UNIT 3

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

SAMPLING AND WATER QUALITY SYSTEM
FLOW DIAGRAM

FIGURE 10.21-3



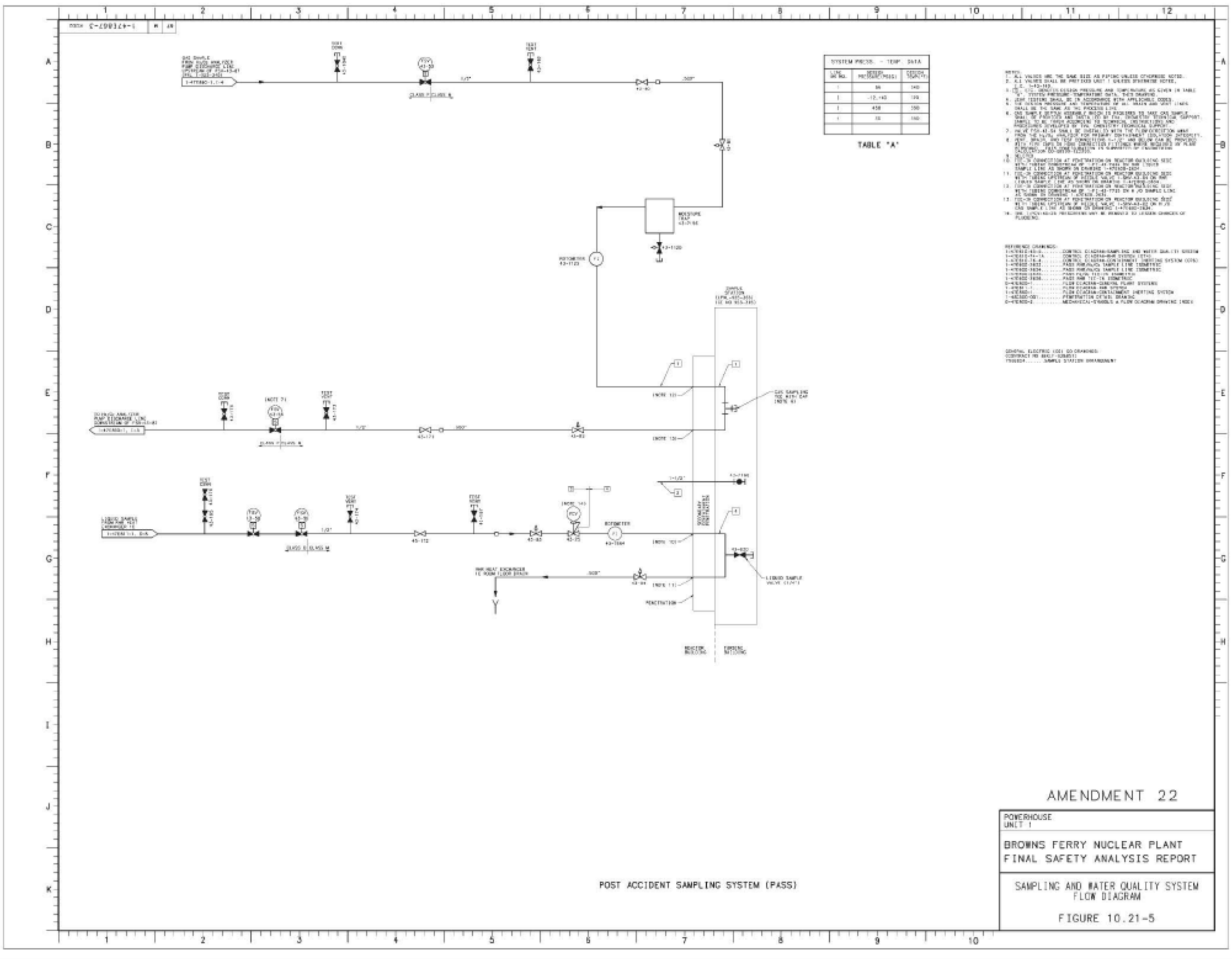
AMENDMENT 24

POWERHOUSE
UNIT 3

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

SAMPLING AND WATER QUALITY SYSTEM
MECHANICAL CONTROL DIAGRAM

FIGURE 10.21-4



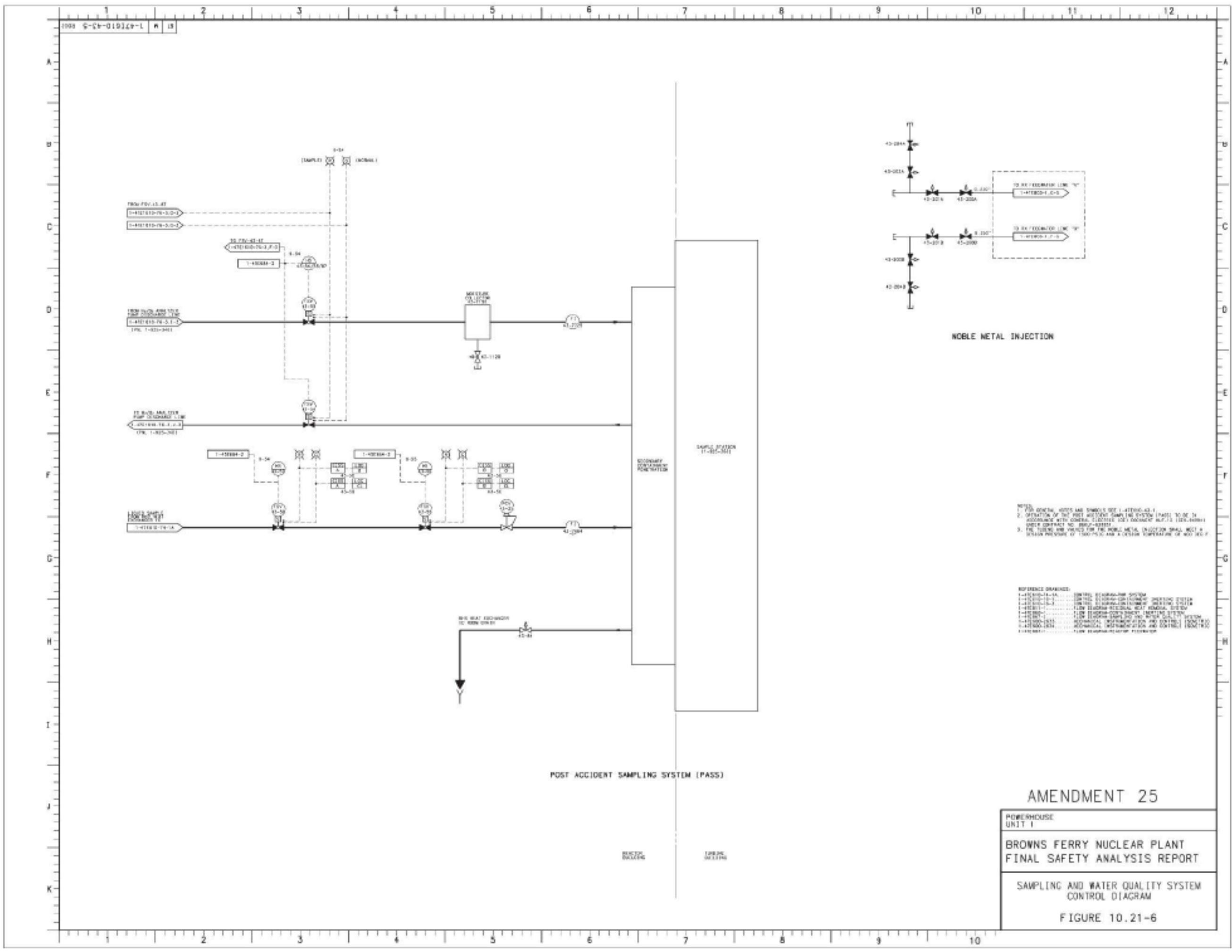
AMENDMENT 22

POWERHOUSE
UNIT 1

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

SAMPLING AND WATER QUALITY SYSTEM
FLOW DIAGRAM

FIGURE 10.21-5


NOBLE METAL INJECTION
POST ACCIDENT SAMPLING SYSTEM (PASS)
AMENDMENT 25
POWERHOUSE
UNIT 1
BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT
SAMPLING AND WATER QUALITY SYSTEM
CONTROL DIAGRAM
FIGURE 10.21-6

10.22 AUXILIARY DECAY HEAT REMOVAL SYSTEM (ADHR)

10.22.1 Power Generation Objective

The Auxiliary Decay Heat Removal System provides a non-safety related means to remove decay heat and residual heat from the spent fuel pool and reactor cavity of either BFN Units 1, 2, or 3. The ADHR allows servicing of the Residual Heat Removal System (RHRS) components earlier in an outage, thus, potentially reducing the outage duration.

10.22.2 Power Generation Design Basis

The ADHR is designed with sufficient capacity to limit temperature in the spent fuel pool (SFP). The ADHR is supplemented by operation of the Fuel Pool Cooling and Cleanup System (FPCS), and assistance from RHRS for a period immediately after the gates between the SFP and the reactor cavity are opened. The ADHR can be utilized to pre-cool the SFP prior to opening gates between SFP and the reactor cavity.

10.22.3 Description

One ADHR System is provided to serve either Unit 1, Unit 2, or Unit 3. The ADHR System consists of two (2) cooling water loops. The primary cooling loop circulates SFP water entirely inside the Reactor Building and rejects heat from the SFP to a secondary loop by means of a heat exchanger. The secondary loop transfers heat to the atmosphere outside the Reactor Building by means of evaporative cooling towers.

The primary loop suction piping from a point near the SFP to a point near pumps, and the discharge piping from a point near the heat exchangers to SFP, is removable. This piping can be installed temporarily for operation of the ADHR and can be removed and stored.

The secondary cooling loop is operated at a higher pressure than the primary loop to prevent cross contamination of the secondary loop by a leak from the primary loop. Differential pressure switches (primary to secondary loop) detect conditions which could have the potential for cross contamination.

10.22.4 Safety Evaluation

The ADHR is not required to support safe shutdown of the plant. Thus, the ADHR does not perform a safety function, with the exception of meeting the requirements for secondary containment penetrations.

10.22.5 Inspections and Testing

The ADHR requires no special inspection or testing.

10.23 HYDROGEN WATER CHEMISTRY SYSTEM (HWC)

10.23.1 Power Generation Objective

The Hydrogen Water Chemistry (HWC) System injects hydrogen into the feedwater system at the suction of the condensate booster pumps to mitigate Intergranular Stress Corrosion Cracking (IGSCC) in the recirculation piping and the reactor internals. The injected hydrogen causes a reduction in the dissolved oxygen within the reactor internals and recirculation piping and lowers the radiolytic production of hydrogen and oxygen in the vessel core region.

10.23.2 Power Generation Design Basis

The purpose of the Hydrogen Water Chemistry (HWC) System at Browns Ferry Nuclear Plant (BFN) is to reduce rates of IGSCC in recirculation piping and lower reactor vessel internals. The corrosion potential in an operating BWR can be reduced by injecting hydrogen into the reactor feedwater system. In high gamma radiation regions, such as in the downcomer, excess hydrogen reacts with H_2O_2, oxygen and other oxidizing species to form water. Therefore, hydrogen injection will result in a less oxidizing environment and, thus, lower corrosion potentials. The target Electrochemical Corrosion Potential (ECP) is - 230 mV Standard Hydrogen Electrode (SHE). At this target potential, and normal BWR water quality, new cracks should not initiate for piping and vessel internals, and existing cracks will have extremely low and tolerable crack growth rates.

Noble Metal Chemical Application (NMCA) involves an occasional batch injection of a small amount of noble metal into the reactor coolant. The noble metal acts as a catalyst for hydrogen and oxygen recombination reactions and allows for a lower HWC feedwater hydrogen injection rate. The lower HWC feedwater hydrogen injection rate with NMCA lowers personnel operating doses and provides improved corrosion protection of reactor vessel internals (lower ECPs).

The storage system is designed to perform the following functions at all times. Failure modes associated with gas supply system will not impact ability of the station to safely shutdown. The cryogenic hydrogen storage tanks are qualified to withstand seismic loads and tornado winds but are assumed to fail in place from tornado missiles. The Cryogenic Oxygen tank may fail under either seismic or tornado wind loads. Gaseous hydrogen storage tubes are qualified to withstand either seismic loads, tornado wind loads, and tornado missiles.

The balance of the HWC System is designed to perform the following functions:

a. Inject sufficient hydrogen into the feedwater stream to be capable of maintaining up to a 2.7 ppm dissolved hydrogen concentration in order to minimize the potential for stress corrosion cracking of lower vessel internal components.

b. Inject sufficient oxygen into the offgas system to ensure that the excess hydrogen in the offgas stream is recombined.

10.23.3 Description

The HWC System injects hydrogen into the reactor feedwater stream at the suction of each condensate booster pump. Hydrogen addition to the feedwater results in an excess ratio of hydrogen to oxygen at the entrance to the Offgas System. Therefore, the HWC System also injects oxygen between the last stage steam jet air ejector and offgas preheater to maintain a stoichiometric mixture of hydrogen and oxygen in the recombiner. The net result is that the Offgas System operates at the same stoichiometric conditions as without HWC. The offgas monitor panel contains dual hydrogen and oxygen analyzers to monitor the recombiner exit conditions to alert plant personnel on abnormal oxygen or hydrogen concentrations and shutdown the HWC system.

Because oxygen is reduced in the steam, condensate oxygen also is reduced. At condensate oxygen concentration values less than approximately 15 ppb, carbon steel corrosion rates are accelerated due to stripping of the oxide layer. To counter this effect, a small amount of oxygen is injected into the condensate pump suction header.

The HWC System control logic, which is contained within the HWC Main Control Panel, provides processing of signal inputs, determination of alarm and shut down conditions, and output of data and shutdown signals. The panel supplies alarm status and display, process parameter displays, and operator interface capabilities. Output terminals are supplied that provide signals to the main control room for remote (plant control room) shutdown and alarm status. Process data is stored locally in the HWC Main Control Panel computer and can be output on a portable disk drive. Capability is provided to supply output of process parameter data to a remote data acquisition system or station computer. The controller is capable of automatically adjusting hydrogen and oxygen flows as a function of reactor power, which is based on feedwater flow, to maintain a constant hydrogen concentration in the feedwater.

Hydrogen and oxygen gases are supplied by an onsite Cryogenic Tank Farm Facility, which is owned, operated, and maintained by the gas supply contractor. The liquid is vaporized as needed and in the case of hydrogen, compressed to the required supply pressure. Oxygen is not compressed but supplied to the process at the liquid tank pressure. Hydrogen is supplied from a nominal 15,000 gallon cryogenic bulk liquid tank which is administratively restricted to less than 10,000 lb capacity and the oxygen is supplied from a nominal 11,000 gallon cryogenic bulk liquid tank. These tanks are designed and constructed in accordance with the ASSME Boiler & Pressure Vessel Code, Section VIII, Division 1. Hydrogen storage

capacity is limited to less than 10,000 lbs which is below the threshold limit of an OSHA process safety plan and an EPA Risk Management Plan per OSHA 1910.119 and EPA 40 CFR 68.

Excess flow check valve protection is provided for all hydrogen and oxygen piping outside of the storage facilities to ensure that adequate separation is maintained between a large leakage source and any safety related structures or air pathways into safety related structures. The hydrogen water chemistry system trips on offgas system isolation and on low feedwater flow, which will occur soon after all scram trips. Isolation valves fail close on loss of electric or pneumatic power to prevent inadvertent injection of gases during any station transient.

The hydrogen water chemistry system includes six hydrogen gas concentration detectors, with two attached to the hydrogen flow control module, one mounted above each of the feedwater pumps (three total) and one above the hydrogen injection piping entrance to the turbine building. These detector units continuously monitor ambient air and upon detecting a high hydrogen gas concentration will isolate the hydrogen supply line which results in an HWC system trouble signal in the main control room.

Connections are provided to allow hydrogen piping to be completely purged of air before hydrogen is introduced into the line. Suitable valves are provided to cross connect the purge outlet to a hydrogen vent line, which contains a flame arrestor.

10.23.4 Safety Evaluation

The hydrogen water chemistry system, including the gas vendor equipment, is designed and installed in accordance with EPRI Report NP-5283-SR-A, “Guidelines for Permanent BWR Hydrogen Water Chemistry Installations - 1987 Revision.” This system serves no safety function. It is not required to effect or support the safe shutdown of the reactor or perform in the operation of reactor safety features. Systems analysis has shown that failure of this system will not compromise any safety-related systems or prevent safe shutdown. There is no equipment redundancy associated with the injection control functions for the hydrogen water chemistry system. Equipment redundancy is only provided at the tank farm facility to ensure a reliable hydrogen supply tot he hydrogen water chemistry and generator hydrogen cooling systems.

The hydrogen storage vessels are designed to maintain integrity during a design basis earthquake. The liquid hydrogen tank, including all integral tank attached piping, is designed and qualified to Uniform Building Code (UBC) seismic zone 4 requirements. The liquid hydrogen filled piping between the tank, hydrogen pumps, and vaporizers is qualified to UBC seismic zone 1 requirements. Both of these are conservative with respect to the BFN design basis earthquake acceleration values.

The design of foundations for the hydrogen storage tank includes tank seismic loading.

Foundations for permanent liquid hydrogen and oxygen storage tanks and gaseous hydrogen storage vessels in support of the station HWC System are designed to keep the associated vessel in place during a design basis tornado. Vessel failure with the corresponding loss of all contents is permitted during the tornado. Siting considerations for the storage facilities included evaluation of impact to nearby safety related structures due to fireball or explosion of hydrogen, and oxygen vapor cloud ingestion into safety related air pathways. Leakage from or failure of either facility was evaluated as acceptable with no adverse impact to safe station shutdown. Redundant pressure relief (hydrogen and oxygen) and vent stack design (hydrogen only) vessels provides protection of storage vessels and liquid filled piping from thermal overpressure due to external fire. Due to the explosion potential of hydrogen leaks, there is no fire abatement system. Hydrogen fires are to be allowed to burn until hydrogen source is isolated.

The location of the liquid hydrogen tank was selected to prevent loss of power lines to/from the switchyard or damage station transformers due to hydrogen facility catastrophic failure. Based on the minimum separation distance of over 2,300 feet the overpressure wave associated with tank farm facility failure is bounded by the design tornado wind loads for these structures/components.

10.23.5 Inspection and Testing

The functional operability of the HWC gas storage system and HWC injection equipment was initially tested at the time of system installation to confirm system operation and functional trips.

The gas tank farm facility is owned, maintained, and operated by the selected gas vendor. Therefore, this equipment is not considered permanent plant equipment.

10.24 SUPPLEMENTAL DIESEL GENERATOR SYSTEM

The Supplemental Diesel Generator (SDG) system provides a non-safety related highly reliable backup source of power for the Emergency High Pressure Makeup (EHPM) system and auxiliaries (see section 10.YY). The Supplemental DG system is manually started and aligned to provide an alternate source of power for the EHPM system and auxiliaries during fire events where the normal power supply to the EHPM system has failed. Each DG is capable of supplying the EHPM system loads for all three units.

10.24.1 Safety Objective

The objective of the Supplemental DG system is to increase the availability of the EHPM system in fire events involving loss of Offsite Power, or fire damage to the normal EHPM power supply. The Supplemental Diesel Generator system performs no nuclear safety related or important-to-safety functions. The Supplemental Diesel Generator system is credited in the Fire Protection Report (FPR) discussed in FSAR section 10.11, Fire Protection.

10.24.2 System Description

The Supplemental DG system consists of two (2) 4160V 3,250KW air-cooled diesel generators and associated equipment in an enclosure located in the yard. All support equipment required to maintain standby readiness (battery charger, jacket water heaters, etc) and normal utility loads (lighting, heating, etc) are normally powered by an external utility power feed and are automatically transferred to the Supplemental DGs in emergency conditions. Power required to start the DGs and energize the SDG distribution system is supplied by DG mounted batteries. This allows the SDG system to be placed in emergency service without dependence on external support systems. Additionally, an external 480V connection is provided for connection of a temporary external power source to the Supplemental DG distribution system.

The Supplemental DG system is manually initiated and controlled. Initiation and control is performed from either the EHPM system panels located in the Main Control Room (MCR) in each unit, or via the EHPM system Local Control Panels (LCPs) located in the EHPM equipment room in each unit. Indication and annunciation is provided in each remote operation location for use in the manual control scheme. The control circuits are designed such that fire damage in the control building will not cause failure of controls located in the EHPM equipment rooms. SDG equipment will remain functional following a fire in the control building.

The SDG enclosure is constructed atop a steel base equipped with an approximately 5,800 gal fuel tank sized to operate one SDG at full load for 24 hours. The fuel system is equipped with a fuel maintenance system for removal of particulates and water from the diesel fuel oil. The enclosure is equipped with an external fuel fill box to allow refueling from a fuel truck.

The Supplemental DG enclosure skid is provided with fire detection and suppression.

10.24.3 Safety Evaluation

The SDG system has sufficient capacity to power the EHPM system loads on all 3 units at the same time. The SDG system does not rely on external support systems for emergency operation. It is therefore independent of offsite power which is the source of the normal EHPM feed, and is not vulnerable to fire damage to typical support systems such as pneumatic supply, cooling water, motive power and control power. This ensures that the SDG system is highly reliable for the scenarios in which it is credited in the Fire Probabilistic Risk Analysis (FPRA) described in the FPR.

Fuel capacity is sufficient to support the SDG system for 24 hours, which satisfies the mission time required for the FPRA and also allows time for refueling for extended mission time.

10.24.4 Inspection and Testing

Periodic inspection and maintenance of the Supplemental DGs is based upon manufacturer's recommendations and sound maintenance practices. Engine related parameters are provided on the controller units mounted on the Supplemental Diesel Generators, and alarms are logged for monitoring. The SDGs can be tested unloaded, with EHPM loads during EHPM testing (EHPM suction and discharge via the Condensate Storage Tanks (CSTs)), or a via load bank testing (connection point is provided for a separate load bank connection).

10.25 EMERGENCY HIGH PRESSURE MAKEUP (EHPM) SYSTEM

10.25.1 Power Generation Objective

The Emergency High Pressure Makeup (EHPM) System provides makeup water to the reactor vessel during shutdown and isolation from the main heat sink to replace the normal makeup sources or Emergency Core Cooling Systems during a fire event for the sole purpose of reducing the Core Damage Frequency and Large Early Release Frequency.

10.25.2 Power Generation Design Basis

The EHPM system operates manually to maintain sufficient inventory in the reactor vessel in response to a fire event such that normal and emergency inventory systems are not available. Provision is made for remote manual operation of the system by an operator. The power supply for the system is provided by energy sources of high reliability in order to provide a high degree of assurance that the system will operate when necessary. Provision is made such that periodic testing can be performed during plant operation, in order to provide a high degree of assurance that the system will operate when necessary.

10.25.3 Safety Objective

The objective of the EHPM System is to reduce the Core Damage Frequency and Large Early Release Frequency for an event that involves the loss of the Emergency Core Cooling Systems (ECCS) in conjunction with a fire event. The EHPM system has no nuclear safety functions. The EHPM system has no important to safety functions.

10.25.4 Description

The EHPM System consists of a motor driven pump unit and associated valves and piping capable of delivering makeup water to the reactor vessel. The EHPM System includes dedicated medium and low voltage electrical components to ensure EHPM system power independence from existing site power distribution to the extent practicable. The pump takes suction from the condensate system via the bottom of the condensate storage tank (CST) and discharges into either the reactor feedwater line for delivery to the reactor vessel or to a full flow return test line to the condensate storage tank. A minimum flow bypass line to the full flow return test line is also provided for pump protection. The EHPM system provides emergency makeup water from the CST to the reactor vessel during fire events where the fire results in the normal (e.g. reactor feedwater) and emergency (e.g., emergency core cooling systems and RCICs) methods of reactor vessel inventory control are non-functional or ineffective.

Following any reactor shutdown, steam generation continues due to heat produced by the radioactive decay of fission products. The EHPM system provides inventory to a shutdown and isolated reactor vessel to compensate for inventory loss due to boil-off and reactor coolant leakage. During fire events where the normal (e.g., reactor feedwater) and emergency (e.g., emergency core cooling systems and RCICs) methods of reactor vessel inventory control are non-functional or ineffective, the EHPM system has a makeup capacity sufficient to maintain the core in a safe and stable state.

The EHPM system is manually initiated and controlled. This initiation and control is capable of being performed from either the EHPM system panel located in the main control room or locally in the vicinity of the pump via the EHPM system local control panel. The EHPM system is provided with its own designated unit specific reactor vessel level and pressure indication at both of these locations for use in its manual control scheme. The pump controls provide automatic trip of the EHPM pump motor upon receiving a low-low suction pressure signal, to prevent damage to the pump. The EHPM system piping is designed in accordance with USAS B31.1.0, 1967 edition.

10.25.5 Safety Evaluation

Due to the non-safety related classification of the EHPM system, no portion of this system requires protection from the adverse effects of design basis external events such as earthquakes, tornadoes, floods, rain, or transportation accidents to support the safe shutdown of the plant.

10.25.6 Inspection and Testing

A design flow functional test of the EHPM System may be performed during plant operation by taking suction from the condensate storage tank 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. Control system design provides manual return from test to operating mode if system initiation is required during testing. Periodic inspection and maintenance of the motor-drive pump unit are based on manufacturer’s recommendations and sound maintenance practices. System process indication, as well as system alarms, are displayed in both the control room and the local control panel for the pump unit.