SECTION 9

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SECTION 9

9.0 AUXILIARY SYSTEMS

The auxiliary systems required to support the reactor coolant system during normal operation of the Davis-Besse Nuclear Power Station are described in the following sections. Some of these systems are described in detail in Chapter 6 since they serve as engineered safety features. The information in this chapter deals primarily with the functions served during normal operation.

Most of the components within these systems are located within the auxiliary building. Those systems with connecting piping between the containment and the auxiliary building are equipped with containment isolation valves as described in Subsection 6.2.4.

The codes and standards used, as appropriate, in the design, fabrication; and testing of components, valves, and piping are as follows:

a. ASME Boiler and Pressure Vessel Code, Section II, Material Specifications.

b. ASME Boiler and Pressure Vessel Code, Section III, Nuclear Components (1971.)

    1. Nuclear Components Class 1.

    2. Nuclear Components Class 2.

    3. Nuclear Components Class 3.

c. ASME Boiler and Pressure Vessel Code, Section VIII, Pressure Vessels, and ASME Nuclear Code-Case Interpretations.

d. ASME Boiler and Pressure Vessel Code, Section IX, Welding Qualifications.

e. Standards of the American Society for Testing and Materials (ASTM.)

f. ANSI, B31.1.0 - 1967, Power Piping.

g. ANSI, B31.7 - 1969, Nuclear Power Piping.

h. Standards of the American Institute of Electrical and Electronics Engineers (IEEE).

i. Standards of the National Electrical Manufacturers Association (NEMA).

j. Hydraulic Institute Standards.

k. Standards of Tubular Exchanger Manufacturers Association.

l. Air Moving and Conditioning Association Codes (AMCA).

m. ANSI, B96.1, Aluminum Tanks.

n. American Gear Manufacturers Association Standards (AGMA).

o. American Water Works Association Standards (AWWA).

p. Draft ASME Code for Pumps and Valves for Nuclear Power, November 1968.

q. National Fire Protection Association (NFPA) Standards.

r. ANSI Standard C 50.20 - Test Code for Polyphase Induction Motors and Generators.

s. ANSI Standard C 50.2 for Alternating Current Motors, Induction Machines in General and Universal Motors.

t. Heating, Ventilating, and Air Conditioning Guide; American Society of Heating, Refrigerating and Air Conditioning Engineers (ASHRAE).

u. The pressure-containing parts of all engineered safety features systems pumps of stainless steel material were liquid penetrant-examined in accordance with Appendix VIII of Section VIII of the ASME Code. The pressure-containing welds of all engineered safety feature systems pumps were radiographically examined in accordance with Paragraph UW-51 of Section VIII of the ASME Code.

v. Valves and piping were designed and fabricated per requirements of ANSI B16.5 or MSSS SP-66, and ANSI B31.1.0.

w. American Welding Society (AWS).

x. ASME Power Test Codes, PTC 8.2-1965.

y. Pipe Fabrication Institute Standards (PFI).

z. Recommended Standards for Water Works.

aa. State of Ohio Building Code, Chapter BB-51, Plumbing.

bb. The National Plumbing Code.

cc. Sheet Metal and Air Conditioning Contractors National Association (SMACNA).

dd. ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels (1968).

ee. Service Water System valves SW-1424, SW-1429, and SW-1434 Component Cooling Water valves CC1407C, CC1411C, CC1568, Core Flooding valve CF2C, Makeup and Purification valve MU60D, and Pressurizer Quench Tank valve RC229C were purchased utilizing the provisions of NRC Generic Letter 89-09, ASME Section III Component Replacements. Additionally, a spare valve which can be used as either SW3962 or SW3963 has been purchased and upgraded utilizing the provisions of Generic Letter 89-09.

ff. ANSI/FCI 70-2-1976, American National Standard for Control Valve Seat Leakage.

gg. ANSI B16.34-1981, Valves - Flanged, Threaded, and Welded End.

hh. ASME Boiler and Pressure Vessel Code, Section II, Materials Specifications, 1971 through 1989 Editions.

ii. ASME Boiler and Pressure Vessel Code, Section III, Rules for Construction of Nuclear Power Plant Components, 1971 through 1986 Editions.

jj. ASME Boiler and Pressure Vessel Code, Section V, Nondestructive Examination, 1986 Edition.

kk. ASME Boiler and Pressure Vessel Code, Section IX, Qualification Standard for Welding and Brazing Procedures, Welders, Brazers, and Welding and Brazing Operators, latest Edition and Addenda.

ll. ASME Boiler and Pressure Vessel Code, Section III, Rules for Construction of Nuclear Power Plant Components, 1989 Edition, No Addenda.

To aid in utilizing the system drawings, a standard set of symbols and abbreviations has been used.

A cross reference identifying the system components to which the above codes and standards apply is given below in Table 9.0-1.

TABLE 9.0-1

Cross Reference of Components and Design Codes

Component	Code (Class**)
SERVICE WATER SYSTEM	
Pumps	a, b(3), e, j
Piping	b(2), b(3), e, f
Valves	a, b(2), b(3), e, ee, ff, gg, hh, jj, kk
Motors	h, i, r, s
COMPONENT COOLING WATER	
Pumps	a, e, j, p
Heat Exchangers	a, c, d, e, k
Tanks	a, c, d, e
Piping	a, b(2), b(3), d, e, ll***
Valves	a, b(2), b(3), d, e, ee
Motors	h, i, r, s
SPENT FUEL POOL COOLING SYSTEM	
Pumps	a, b(3), e, j
Heat Exchangers	a, c, d, e, k, dd
Filters	a, b(3), d, e, g
Demineralizer Tanks	a, b(3), d, e, g
Piping	a, b(3), d, e
Valves	a, b(3), d, e
Motors	h, i, r, s
SAMPLING SYSTEM	
Piping	b(1), b(2), b(3), e, f
Heat Exchangers	e
Valves	b(1), b(2), b(3), e, f
HEATING, VENTILATION AND AIR CONDITIONING	
Fans	l, t, q, cc
Filters	l, t, q
Ductwork	l, t, q, cc
Heating and Cooling Coils	l, t
Piping	b(2), e, f
Valves	b(2), e, f
Motors	h, i, r, s
STATION AND INSTRUMENT AIR SYSTEM	
Receivers	a, c, d, e
Piping	b(2), e, f
Valves	a, b(2), e, v
Motors	h, i, r, s

** Where applicable to ASME Code Section III

***A portion of the CCW lines (3”-HCB-41 and 3”-HBB-13) in containment that interfaces with the CRDM Stator Cooling Water System was replaced by ECP 10-0470. The construction code for the replacement CCW piping is ASME III 1989 Edition with No Addenda. ASME Code required reconciliation has been performed and is documented in ECP 10-0470.

TABLE 9.0-1 (Continued)

Cross Reference of Components and Design Codes

Component	Code (Class**)
AUXILIARY FEEDWATER SYSTEM	
Pumps	a, b(3), e, j, x
Tanks	e, o
Piping	b(2), e, f
Valves	a, b(2), e
Turbines	e, i
FIRE PROTECTION SYSTEM	
Pumps	e, j, q
Piping	e, q
Valves	e, q
Motors	h, i, r, s, q
Diesel	q
DECAY HEAT REMOVAL SYSTEM	
*Pumps	a, e, j, p, u
Heat Exchangers	a, c, d, e, k, dd
Piping	a, b(1), b(2), d, e,
*Valves	b(2), e, g, p, v
Valves	a, b(1), b(2), d, e
*Motors	h, i, r, s
DIESEL GENERATOR FUEL OIL SYSTEM	
Pumps	a, b(3), e, j
Tanks	a, b(3), e, q
Piping	a, b(3), v
Valves	a, b(3), v
Motors	h, i, r, s
CHEMICAL ADDITION SYSTEM	
*Pumps	a, e, j, p, u
*Tanks	None
Tanks	a, d, e, dd
Piping	a, b(3), d, e, f, v
*Valves	e, f, v
Valves	a, b(3), d, e, f, v
Motors	h, i, r, s

* Equipment by NSS Supplier
** Where applicable to ASME Code Section III

TABLE 9.0-1 (Continued)

Cross Reference of Components and Design Codes

Component	Code (Class**)
MAKEUP AND PURIFICATION SYSTEM	
*Pumps	a, e, j, p, u
*Heat Exchangers	a, c, d, e, k, dd
*Filters	a, d, e, dd
*Demineralizer Tank	a, d, e, dd
Demineralizer Tank	a, b(3), d, e
*Tanks	a, d, e, dd
Piping	a, b(1), b(2), b(3), d, e
*Valves	b(1), e, g, p, v
Valves	a, b(2), b(3), d, e, ee
*Motors	h, i, r, s
DEMINERALIZED WATER MAKEUP SYSTEM	
Pumps	e, i, j
***Clarifiers	e, i, z
***Filters	e, z
***Demineralizer	c, e
Piping	e, v
Valves	e, v
Motors	h, i, r, s
POTABLE AND SANITARY WATER SYSTEMS	
Pumps	e, i, j
Tanks	c, e
Piping	e, aa, bb, o
Motors	h, i, r, s
EQUIPMENT AND FLOOR DRAINAGE SYSTEM	
Pumps	e, j
Piping	e, d, v
Valves	e, d, v
Motors	h, i, r, s
CONDENSATE STORAGE FACILITIES	
Tanks	o, w
Piping	v
Valves	v
EMERGENCY FEEDWATER SYSTEM	
Pump	e, i, j
Tank	e
Piping	a, b(2)(3), e, kk, v
Valves	a, b(2)(3), e, ii, v

Note: The EFWS does not support the RCS during normal plant operation but is added here as a new major plant system.

* Equipment by NSS Supplier
** Where applicable to ASME Code Section III
***Equipment is no longer used

9.1 FUEL STORAGE AND HANDLING

9.1.1 New Fuel Storage

9.1.1.1 Design Bases

The new fuel storage facility is capable of storing a maximum of 80 new fuel assemblies. However, due to optimum moderation ("mist") criticality considerations, rows "C" and "F" must be blocked so that new fuel assemblies cannot be stored in those rows. Therefore, the new fuel storage area can effectively store only 60 new fuel assemblies. The new fuel assemblies are stored vertically in parallel rows with a nominal 21 inch center-to-center fuel assembly spacing in both directions. New fuel assemblies containing up to 5.00 weight percent uranium-235 (wt% U-235) may be stored in the new fuel storage facility. Although the new fuel storage facility is normally dry, this spacing, combined with the blocking of rows "C" and "F" to new fuel assemblies, is sufficient to maintain a k_{eff} of less than or equal to 0.95 when flooded with unborated water, and to maintain a k_{eff} of less than or equal to 0.98 if the storage area is immersed in a hydrogenous "mist" producing optimum moderation (i.e., highest value of k_{eff}) which includes a conservative allowance of 1 percent delta k/k for uncertainties. The design loads to be withstood are seismic, tornado, and thermal loads and are discussed in Subsection 9.1.1.3.

The criticality analyses discussed above assumed that fuel assemblies were uniformly loaded with 5.0 wt% U-235 fuel rods. For zone-loaded fuel assemblies (fuel assemblies containing fuel rods with multiple U-235 enrichments), specific analyses are required to demonstrate that the analysis with a uniform 5.0 wt% U-235 enrichment loading remains bounding.

9.1.1.2 Description

The new fuel storage area is located inside the fuel-handling area in the auxiliary building. The location of the new fuel storage is shown in Figure 1.2-4 and Figure 1.2-5.

The storage racks are of individual cells assembled in four 20-rack assembly units which are braced together to form a rack frame as shown in Figure 9.1-2. The rack assemblies are constructed entirely of type 304 stainless steel.

9.1.1.3 Safety Evaluation

The new fuel storage area and racks are Seismic Class I structures which are designed to withstand seismic loadings of 0.40 g in horizontal and 0.11 g in vertical directions acting simultaneously at the floor. Details of seismic, tornado, and thermal loads are discussed in Section 3.7 and Section 3.8. Figure 9.1-3 shows the mass model for the new fuel racks.

The spacing of the racks is such as to preclude the possibility of criticality even if flooded with unborated water.

The fuel assemblies are removed from the racks without exerting any uplift force on the rack. The fuel assemblies make free contact at the bottom and on the sides. All the projections and corners on the racks are carefully tapered to eliminate any possibility of fuel assembly sticking in the rack. Removal of the new fuel assembly from the new fuel storage racks is done manually. The operator can stop the crane at any time during handling of the new fuel assemblies. The racks can withstand an uplift force of 2700 pounds.

There is no equipment located adjacent to the racks whose failure can damage the racks or the fuel assemblies. The only equipment in close proximity to the new fuel racks is the spent fuel handling bridge which is restrained to Seismic Class I rails by Seismic Class I restraints to prevent its jumping the tracks in the event of earthquake. The winch mechanism is also anchored to the floor by Seismic Class I anchors.

A 4-inch-diameter open floor drain at the bottom of the pit is provided to prevent any possibility of accumulation of water.

The following codes and standards have been followed, as applicable, in the design of the new fuel storage facility:

AISC	American Institute of Steel Construction, Inc. - Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings.
AWS	American Welding Society - Welding in Building Construction.
ASTM	American Society for Testing and Materials specification and standards A 36 Structural Steel A 240 Stainless Steel Plates A 276 Stainless Steel Bars and Shapes.

9.1.2 Spent Fuel Storage

9.1.2.1 Design Bases

The spent fuel pool facility is designed to assure the safe storage of irradiated fuel assemblies under normal and accident conditions. The spent fuel pool structure provides for the containment, confinement, and prevents significant reduction in coolant inventory under normal and accident conditions.

Spent Fuel Pool

The 21 high density Spent Fuel Pool racks store a maximum of 1624 fuel assemblies. The stainless steel rack cells are arranged in a rectangular array of parallel rows with a center-to-center spacing of 9.22 inches, and with walls of 0.075 inches nominal thickness. Boral®, which is a neutron absorber, is attached to all four sides of each cell. Between individual racks, and between peripheral racks and the Pool walls, there is gap. This gap forms a “flux trap” which reduces neutron movement between fuel assemblies in adjacent racks. Holtec Report No. HI-2002359 concluded the combination of Boral and flux traps provides the option to position each fuel assembly in three different patterns within the racks - Mixed Zone Three Region, Checkerboard, and Homogeneous Loading. These loading patterns maintain k_{eff} less than or equal to .95 for fuel assemblies with initial nominal enrichments less than or equal to 5.05 weight percent Uranium-235, assuming the Spent Fuel Pool water is unborated which involves a conservative allowance for manufacturing tolerances and calculation uncertainty. (Fuel stored in the Spent Fuel Pool is administratively limited to a maximum nominal enrichment of 5.0 weight percent Uranium-235. This limit allows compliance with subsection “b” of 10CFR50.68, Criticality Accident Requirements.) The position of a fuel assembly within a pattern is dependent upon its “category”. The category is based the burnup/initial enrichment restrictions contained in the Technical Specifications. The detailed requirements for use of each pattern, or combination of patterns, are contained in administrative procedures.

During accident conditions credit may be taken for the Boron in the Spent Fuel Pool water when showing k_{eff} is maintained less than or equal to .95. The above Holtec report, supplemented by calculation C-NF-062.02-027, determined a Boron concentration of 627 ppm in the Spent Fuel Pool water is required to maintain $k_{eff} \leq .95$ for the worst-case fuel handling accident, (i.e., a 5.05 weight percent enriched assembly misloaded in a Checkerboard pattern). This 627 ppm minimum value is bounded by the Technical Specification 3.7.15 requirement of ≥ 630 ppm.

The maximum heat generation rates for storage of fuel in the Spent Fuel Pool are specified in the USAR Technical Requirements Manual.

Seismic classification of the Spent Fuel Pool racks is discussed in Subsection 9.1.2.2.

Dry Fuel Storage

There is also a Dry Fuel Storage Facility (DFSF) onsite. After a minimum required storage period in the spent fuel pool, the spent fuel assemblies may be transferred to the onsite dry fuel storage facility. The design of the DFSF is described in Section 9.1.6.1.

9.1.2.2 Description

After removal from the reactor, the spent fuel is stored under water within the spent fuel storage pool. The storage pool facilities have reinforced concrete pools lined with 1/4-inch-thick stainless steel. This facility is located inside the fuel handling area in the auxiliary building as shown in Figure 1.2-4 and Figure 1.2-5. The auxiliary building, as well as the storage pool, is a Seismic Class I structure which is designed to withstand seismic, tornado, and thermal loads as discussed in Section 3.7 and Section 3.8. The spent fuel pool storage racks are also Seismic Class I structures which are designed to withstand seismic loadings. The mass model is shown in Figure 9.1-3B. The fuel handling area is also protected against tornado-generated missiles and other potential missiles.

With water level below the bottom of the spent fuel pool/transfer pit gate (elevation 578’ 0”), in the transfer pit, struts must be installed across the pit to prevent overstressing of the 3-foot wall between the spent fuel pool and the fuel transfer tube pit during a seismic event. The struts do not interfere with fuel handling operations or necessary maintenance.

Adequate shielding is provided for station personnel by the 5-1/2-foot-thick concrete walls and water in the pool. The radiation zones around the spent fuel pool are shown in Figures 12.1-2, 12.1-3, and 12.1-4.

The spent fuel pool racks are arranged in a 25 storage cell x 66 storage cell array. To accommodate the pool gates, 26 cells have been deleted from the array. The arrangement is shown in Figure 3.8-1. The location of the storage pools within the station complex is shown in Figures 1.2-4 and 1.2-5.

A separate space is provided for loading the dry fuel storage canister or a spent fuel shipping cask. The spent fuel cask pit is independent of and separated from the spent fuel pool by a 3-foot-thick concrete wall. The only communication between the spent fuel pool and the cask pit is through the 36-inch-wide slot opening provided for the transfer of the spent fuel assemblies from storage to the shipping cask; or to the dry fuel storage canister. This opening is provided with a watertight bulkhead which can isolate the spent fuel pool when needed.

Following sufficient decay, the spent fuel assemblies can be removed from storage and loaded into the spent fuel shipping cask or dry fuel storage canister under water for removal from the site. Casks up to 130 tons in weight can be handled by the spent fuel cask crane.

A cask-wash-and-decontamination area is also provided adjacent to the cask pit. In this area, outside surfaces of the cask can be decontaminated before movement to the onsite Dry Fuel Storage Facility (DFSF) or shipment from the site. The DFSF is described in Section 9.1.6.1.

9.1.2.3 Safety Evaluation

The spent fuel pool storage racks are designed for noncriticality by use of Boral panels. Boral is an aluminum matrix containing boron carbide sandwiched between, and bonded to, aluminum plates. The boron carbide contains a high areal loading of the boron -10 isotope, which provides appropriate neutron attenuation between adjacent storage cells. In addition, the design of the rack section base plates ensures that there is a flux trap between fuel assemblies stored in adjacent rack sections. Placement of fuel in the spent fuel pool prior to installation of all rack modules has been analyzed for the accidental lowering of a fuel assembly outside the rack modules. The analysis concluded that the criticality limit would not be exceeded. After the installation of all rack modules, the spent fuel pool rack design will prevent the accidental insertion of a fuel assembly in other than prescribed locations.

All spent fuel assembly transfer operations are normally conducted under a minimum of 9-1/2 feet of borated water above the top of the active fuel to ensure adequate biological shielding. All piping penetrations into the spent fuel pool, cask pit, and transfer pit penetrate at least 9 feet above the top of the fuel assemblies to avoid any possibility of completely draining the pool in case of a pipe rupture. Isolation valves are provided on the pipes penetrating the pool and pits. These valves are located as close to the concrete wall as practicable to minimize the possibility of pipe failure between the isolation valves and the pool. The spent fuel pool, cask pit, and transfer pit drain valves are locked closed per plant procedure.

The spent fuel pool water is cooled by the spent fuel pool cooling system as discussed in Subsection 9.1.3.

The spent fuel cask crane is electrically interlocked to prevent the crane from inadvertently traveling over the spent fuel pool.

This interlock can be bypassed with a key. The key allows use of the crane’s non-single-failure-proof 20 ton auxiliary hook and single-failure-proof main hook over the spent fuel pool area and cask pit. Even with the interlock bypassed, the use of the 1 ton non-single-failure-proof monorail hoist will not be allowed over the spent fuel pool, cask pit, or fuel transfer pit. Movement of loads in excess of 2430 lbs over the spent fuel pool, but not over the fuel assemblies in the spent fuel pool, is procedurally controlled. Either (1) the main hook and 20 ton auxiliary hook may be operated over the spent fuel pool, cask pit, or transfer tube pit under administrative controls as part of a non-single-failure-proof handling system provided an engineering evaluation of the potential load drop has been completed, or (2) the single-failure-proof main hook may be used as part of a single-failure-proof handling system to carry the load, and a load drop does not have to be postulated or evaluated.

The cask pit is separated and isolable from the pool to preclude the possibility of draining the spent fuel pool in case of damage to the cask pit by an accidental load drop in the pit. The base of the cask pit is solid concrete extending down to bed rock. Thus, a load drop is not postulated to cause any significant damage to the structure.

Additionally, the cask wash pit and train bay floor have been evaluated for a load drop. Since these two areas are located west of the cask pit, there is no possibility that a load drop in these areas could impact the spent fuel pool. There are no safety related equipment/components located in either of these rooms. Nor are there any rooms located beneath either of these two areas. Therefore, any postulated load drop accident in the cask wash pit or the Auxiliary Building train bay will not adversely impact the Spent Fuel Pool or any safety related equipment/components. The evaluation of the Dry Fuel Storage Facility (DFSF) components is described in the Davis-Besse Site Certified Safety Analysis Report (CSAR) for the AREVA / TN NUHOMS-24P Dry Shielded Canisters (DSCs), the AREVA / TN Standardized NUHOMS Updated Final Safety Analysis Report (UFSAR) NUH-003, Appendix U, for the 32PTH1 DSCs, and the AREVA / TN NUHOMS® EOS System Safety Analysis Report, for the EOS-37PTH DSCs.

Spent fuel storage racks are designed to eliminate any possibility of fuel assembly sticking in the racks. All projections and corners are properly tapered and rounded off. The spent fuel assemblies are placed into, and removed from, the racks by the spent fuel handling bridge crane. Since the fuel assemblies make free contact with the storage cells, there would be no uplift force exerted on the racks. The spent fuel handling bridge crane is provided with an overload interlock on the hoist which shuts off the power to the hoist any time the load on the hoist exceeds 2700 pounds. The racks are designed to withstand this uplift force.

The design codes and regulatory guides used in the design and analysis of the spent fuel pool storage rack are identified in References 3.12-84 and 86 through 91.

9.1.3 Spent Fuel Pool Cooling and Cleanup System

9.1.3.1 Design Bases

The spent fuel pool cooling and cleanup system is designed to remove the heat generated by the stored spent fuel assemblies, and to maintain the optical clarity of the water in the spent fuel pool. The system is designed for continuous service whenever spent fuel is stored in the pool.

Two fuel pool heat exchangers and two fuel pool cooling pumps are provided to remove the heat produced by the spent fuel stored in the fuel pool. In the case of a partial core discharge, the pool temperature is kept below 133°F during maximum normal heat load conditions (Section 9.1.3.3.1). During maximum abnormal conditions (Section 9.1.3.3.2) with a full core removed from the reactor, from the reactor, the temperature will be below boiling. Purification systems are provided to maintain optical clarity of the spent fuel pool.

The decay heat removal system described in Section 6.3 serves as a Seismic Class I backup system to the spent fuel pool cooling system. It is also available to remove the decay heat from the spent fuel pool should it be necessary when the entire core is off-loaded into the spent fuel pool.

The two systems are connected with a permanently installed 10-inch line. Two normally closed gate valves are used to provide isolation between the decay heat removal system and the spent fuel pool cooling system.

In addition to its primary function, the spent fuel pool cooling system provides purification by removing fission and corrosion products and by maintaining water clarity of the spent fuel pool facility water, the fuel transfer canal water, and the contents of the borated water storage tank.

The spent fuel pool cooling water has the following quality standard:

Boron Concentration, ppm	≥ 1800
Max Iron (membrane), ppm	1.0
Max Turbidity, NTU	1.0
Max chlorides as Cl-, ppm	0.1
Max fluorides as F-, ppm	0.1
pH at 25°C	~4.5

Turbidity includes corrosion products, fission products, suspended matter, boric acid, and matter deposited on the spent fuel elements which may slough off into the water.

The radiation levels and shielding are described in Chapter 12.

9.1.3.2 System Description

The spent fuel pool cooling system is shown in Figure 9.1-5. It consists of two half capacity recirculating pumps and two half capacity heat exchangers, associated valves, piping and instruments. A bypass system consists of a demineralizer and a filter.

System performance data are shown in Table 9.1-1. Major components of the system are briefly described below.

9.1.3.2.1 Spent Fuel Pool Heat Exchangers

The spent fuel pool heat exchangers are designed to maintain the temperature of the spent fuel pool water as discussed in Subsection 9.1.3.1.

9.1.3.2.2 Spent Fuel Pool Pumps

The spent fuel pool pumps take suction from the spent fuel pool and recirculate the water back to the pool after it passes through the heat exchangers, demineralizer and/or filter in various combinations, as required.

9.1.3.2.3 Spent Fuel Pool Demineralizer

The spent fuel pool demineralizer can remove approximately fifty percent of the fission products contained in the spent fuel pool water in 34 hours, independent of the number of fuel assemblies in the pool. The assumptions used to derive this number are:

a. Uniform mixing of demineralizer effluent with the spent fuel pool water.

b. The concentration of fission products in the effluent of the demineralizer is negligible.

c. Fission product decay is neglected.

d. No additional fission products are added to the spent fuel pool water.

e. The flow rate through the demineralizer is 100 gpm.

f. The volume of water in the spent fuel pool is 300,000 gallons.

9.1.3.2.4 Spent Fuel Pool Filter

The spent fuel pool filter is designed to remove particulate matter from the spent fuel pool water. The filter is sized for the same flow rate as the demineralizer (100 gpm).

9.1.3.2.5 Borated Water Storage Tank Recirculation Pump

The borated water storage tank recirculation pump recirculates water from the borated water storage tank through the spent fuel pool cleanup system for demineralizing and filtering. The pump may also be used for demineralization and filtering the water in the fuel transfer canal during a transfer of fuel.

The pump is used for draining a portion of the refueling canal, fuel transfer pit and cask pit after completion of the fuel transfer operation.

9.1.3.2.6 Spent Fuel Pool Skimmers

Surface skimmers are provided in the spent fuel pool to facilitate the removal of accumulated particulate matter from the surface of the spent fuel pool water.

9.1.3.2.7 Portable Filtration Unit

A submersible, self-contained filtration unit may be used in the Refueling Canal, Fuel Transfer Canal, Spent Fuel Pool or Cask Fill Pit in order to facilitate the removal of accumulated particulate matter from the floor of these tanks and to provide additional water clarity support.

9.1.3.3 Modes of Operation

9.1.3.3.1 Normal Operation

The spent fuel pool cooling system serves two main functions. The first is to remove the decay heat generated by spent fuel stored in the pool as a result of normal refueling conditions. The second function is to provide purification of the spent fuel cooling water.

The first function is accomplished by recirculating spent fuel cooling water from the spent fuel pool through the pumps and heat exchangers and back to the pool. The spent fuel pool pumps take a suction from the pool, circulate the pool water through the tubeside of two heat exchangers, and discharge back to the pool.

With a heat load of 12.4×10^6 Btu/hr and both pumps and heat exchangers operating, the spent fuel pool cooling system is capable of maintaining the spent fuel pool at 125°F or less. With one pump and two heat exchangers operable the pool can be maintained at 140°F or less, and with one pump and one heat exchanger operable the pool can be maintained at 155°F or less under the maximum normal heat load conditions.

In Cycle 13, the spent fuel pool was reracked to provide 1624 spent fuel storage locations. The thermal-hydraulic analysis, Reference 9, has evaluated the spent fuel discharge scenarios described in Table 9.1-2. This table shows the spent fuel pool water temperature would be approximately 133°F (with CCW at 95°F), with both spent fuel cooling system pumps and heat exchangers in operation. The coincident decay heat load is 15.89 x 10^6 Btu/hr. See Reference 9 for the conditions that were assumed in determining the maximum expected normal heat load.

The cleanup function is accomplished by a bypass purification system. The bypass loop branches off from the spent fuel pool pump discharge cross-connect line, bypassing the heat exchangers. After demineralizing and/or filtering, the bypass flow is directed into the normal line downstream of the heat exchanger and returns to the pool.

The cleanup system is provided with two sample connections, one upstream and one downstream of the system. Samples will normally be taken weekly upstream of the cleanup system to ensure that the spent fuel pool water quality is maintained. The sampling frequency may be increased during certain evolutions such as during refueling or spent fuel handling.

The system will also be utilized to purify the water in the borated water storage tank following refueling, and to maintain clarity in the fuel transfer canal during refueling. Water from the borated water storage tank or fuel transfer canal can be purified by using the borated water storage tank recirculation pump. Purification of the borated water storage tank using the spent fuel pool cleanup system can be performed in any mode using a flow path with a high elevation point, such that failure of the Seismic Class II spent fuel pool cleanup system will not result in the draindown of the borated water storage tank below the Technical Specification level. The high elevation point is furnished with an open vent line in order to prevent a siphoning effect in the event of a break in the lower-elevation Seismic Class II piping.

When the operation of the cleanup system is not needed, the cleanup system is isolated by closing the upstream and downstream manual diaphragm valves of the cleanup system.

9.1.3.3.2 Abnormal Conditions

Since the abnormal heat load resulting from core offload can exceed the spent fuel pool cooling system capacity, dependent on the inlet temperature of component cooling water available to the heat exchanger, the decay heat removal system is designed to be available to remove the decay heat from the spent fuel pool.

The modification of the plant to provide 1624 storage locations in the spent fuel pool increased the potential Spent Fuel Pool Cooling System heat load. The thermal-hydraulic analysis for the reracked pool, Reference 9, with the decay heat system cooling the pool, is based on the spent fuel pool containing 1714 fuel assemblies, which included a recent whole core discharge of 177 assemblies. The peak heat load associated with these 1714 assemblies was determined to be 30 x 10^6 Btu/hr (Reference 3.12-84). Assuming one train of the Decay Heat System cooling the spent fuel pool coolant, the analysis yielded a peak pool bulk temperature of approximately 151.5°F with component cooling water at 95°F. The analysis results indicated the spent fuel pool bulk temperature would be above 150°F for less than 28 hours. This short term exceedance of the 150°F long term temperature limit for fuel structure is acceptable per Appendix A of ACI 349, as described in USAR Section 3.8.1.5.1. Per the revised Ultimate Heat Sink evaluation, the bulk temperature would be approximately 154°F, and remain above 150°F for approximately 80 hours (based on component cooling water at 97°F), (Reference 3.12-91).

9.1.3.3.3 Minimum Time-to-Boil and Maximum Boiloff Rate

For discharge/cooling Scenarios 1 and 4A, the calculated time-to-boil and maximum boiloff rates are summarized in Table 9.1-2. These results show that, in the extremely unlikely event of a complete failure of both the Spent Fuel Pool (SFP) Cooling System and Decay Heat Removal System (DHRS), there would be at least 3.78 hours (based on component cooling water at 95°F) available for corrective actions prior to the onset of boiling. Per Reference 91 in Section 3.12 of the USAR, this time would decrease by approximately 20 minutes with component cooling water at 97°F. The maximum water boiloff rate is less than 70 gpm. The DHRS is permanently connected to the Seismic Class I boundary of the SFP. By already-proceduralized valve line-ups, the DHRS can provide borated makeup water to the SFP from the Borated Water Storage Tank, (by pumped or gravity-fill methods). SFP makeup water is also available from the Seismic Class II Demineralized Water Storage Tank or the Clean Waste Receiver Tank. Therefore, it can be concluded that sufficient time for remedial actions is available and that makeup capacity will exceed the makeup demand.

In the unlikely event the establishment of makeup to the SFP was delayed following a loss-of-forced-cooling event, approximately 25 hours of boiling would be required to reduce SFP level from the Technical Specification minimum level of 23 feet above the top of fuel assemblies seated in the storage racks, to the level corresponding to 9-1/2 feet above the top of fuel stored in the racks, given a SFP plan area of approximately 1057 $feet^2$, and assuming a constant boiloff rate of 70 gpm. A minimum of 9-1/2 feet of borated water above the top of active fuel stored in the racks will ensure adequate biological shielding. In summary, 25 hours provides operators with more than sufficient time to intervene with available means to maintain or restore the SFP water level.

9.1.3.4 Reliability Considerations

The spent fuel pool cooling system provides adequate capacity and component redundancy to ensure the reliable cooling of spent fuel stored in the spent fuel pool. Ample time is available to ensure that cooling can be restored even in the unlikely event of multiple component failures or complete cooling loss. The system is so arranged that no uncontrolled, complete loss of water from the pool is possible by piping or component failures. The system performs no emergency functions and is not directly connected to the reactor coolant system.

The decay heat removal system, which has a higher heat removal capacity, serves as a backup system to the spent fuel cooling system.

9.1.3.5 Codes and Standards

Each component of this system is designed to the code or standard, as applicable, as noted in Table 9.0-1.

9.1.3.6 Fuel Leakage Considerations

If a leaking fuel assembly is transferred from the refueling canal to the spent fuel pool, a small quantity of fission product activity may enter the spent fuel pool cooling water, even though the assembly's cladding temperature is lowered, and leakage should be minimized. Also, fuel assembly repair, as described in Section 9.1.4.2.3, may result in an additional release of a small quantity of fission and/or activation product activities into the spent fuel pool cooling water. The purification loop removes these fission products and other contaminants from the pool water. Radiological evaluation is presented in Chapter 11 and Chapter 12.

The ventilation flow for the fuel handling and storage area housing the spent fuel pool is normally exhausted to the environment through the station vent. However, proper interconnections have been provided with the emergency ventilation system so that charcoal filters can be utilized in case the iodine concentration in the air exceeds acceptable limits. Provisions have been made in the design to air-test the flanged end of the fuel transfer tube for leak-tightness after use.

9.1.3.7 System Isolation

The spent fuel pool cooling system has no process lines connecting to the reactor coolant system. Its major penetrations to the containment are through two fuel transfer tubes. Except during fuel transfer operations, each of these fuel transfer tubes is isolated inside the containment by a blind flange in the fuel transfer canal and a normally closed manual gate valve outside the containment in the spent fuel pool.

9.1.3.8 Failure Considerations

Failure of a single component in this system will not cause complete loss of cooling of the stored spent fuel under normal operating conditions since the system is designed with two 50 percent capacity components. A complete loss of cooling is not considered credible, since the decay heat removal system serves as a backup to the spent fuel pool cooling system, should one be required.

9.1.3.9 Operational Limits

There are alarms provided for the spent fuel pool cooling system to indicate high or low pool and containment refueling canal water level. The spent fuel pool water is maintained at a normal level of El. 601 feet 6 inches. The setpoints of the high and low level alarms are El. 601 feet 7 inches and El. 601 feet 2 inches, respectively. Operator action will be required in case of low or high level alarm. Plant procedures provide guidance to mitigate high or low levels.

The level change in the spent fuel pool could occur due to spent fuel pool cooling heat exchanger tube rupture, spent fuel pool cooling system pipe rupture, or leakage to the spent fuel pool system from BWST (past two normally closed valves). The spent fuel pool water temperature is monitored, and a high temperature alarm is provided. Differential pressure alarms are provided across system components, as well as system flow rate alarms.

9.1.3.9.1 Alarm Setpoints

High and low level alarms are provided for both the spent fuel pool and the refueling canal. The low level alarm assures a minimum of 23’ of water is maintained above the fuel assemblies, the high level alarm is provided to prevent overfill.

High temperature alarms will occur if the temperature reaches 125°F in the spent fuel pool or at the inlet of the spent fuel pool demineralizer.

The differential pressure across the spent fuel pool demineralizer, skimmer filter and the containment refueling canal skimmer is monitored. A differential pressure of 10 psi across any of these components will activate an alarm.

The differential pressure across the spent fuel pool demineralizer filter is monitored until 10 psi across the filter activates an alarm. Once the alarm is activated, the differential pressure indicator is removed from service. Purification flow and filter radiation levels are then monitored to determine the optimum point to change the filter. Once the filter is changed, the differential pressure indicator is returned to service.

A low flow alarm set at 700 gpm is provided for the spent fuel pool heat exchanger combined outlet. The spent fuel pool purification system has flow alarms set at 60 gpm (low flow) and 110 gpm (high flow).

9.1.3.10 Testing and Inspection

Each component is inspected and cleaned prior to installation into the system. Demineralized water is used to flush the system.

Instruments were calibrated prior to testing. Alarm functions were checked for operability and setpoints during preoperational testing.

The system was operated and tested initially with regard to flow paths, flow capacity, and mechanical operability. At least one pump of each type was tested to demonstrate head and capacity.

Data are taken periodically during normal station operation to confirm heat transfer capabilities and purification efficiency.

The spent fuel pool, fuel transfer pit and cask pit liners are checked periodically for leaks using the floor and wall monitoring system.

9.1.3.11 Loss of Water from the Fuel Pool

Loss of water from the spent fuel pool facility is unlikely, since these pools and piping within the pools are Seismic Class I. Seismic Class I design of the piping terminates at the isolation valves on the suction line from, as well as on, the return line to the pool (refer to Figure 9.1-5). The spent fuel pool, cask pit, and transfer pit drain valves are locked closed per plant procedure to ensure that these pools are not inadvertently drained. The large bore seismic class II piping (>2.5” diameter) is located such that any failure of this pipe will provide sufficient water cover over the spent fuel elements. Any temporary lines used in the spent fuel pool will be evaluated for siphon breaker protection. Also, all the penetrations into the fuel pool are made at least 9 feet above the top of the fuel assemblies, thereby eliminating the possibility of draining the pool below that level.

The decay heat system, a Seismic Class I system, is permanently connected to the Class I boundary of the spent fuel pool cooling system. This system will serve to make up the spent fuel pool water by supplying the borated water from the BWST to the fuel pool to prevent uncovering of the fuel should the need ever arise. Figures 9.1-5 and 6.3-2A show the interconnecting piping.

The Seismic Class I makeup water source is also available for spent fuel pool water makeup as shown in Figures 9.1-5 and 6.3-2A, and is further illustrated in Figure 9.1-6.

Spent fuel pool makeup water is also available from the Seismic Class II demineralized water storage tanks and the Clean Waste Receiver Tanks.

9.1.4 Fuel Handling System

9.1.4.1 Receiving New Fuel

The new fuel assemblies are received by truck in specially designed shipping containers. The container design is licensed by the fuel supplier to ensure it meets all design requirements.

Upon receipt, the shipping containers are stored in an area where access is limited to authorized personnel only. The shipping containers are opened one at a time and the assemblie(s) are given an inspection for any damage incurred during shipment and are then stored. Storage is in the new fuel storage racks until they are moved to the new fuel elevator for placement in the Spent Fuel Pool or Transfer Mechanism.

9.1.4.2 Refueling

9.1.4.2.1 Preparation

Refueling of the reactor core takes place approximately every 24 months. At this time, as dictated by the fuel management program, spent and partially-spent fuel assemblies are replaced with new fuel assemblies. In addition, some partially-spent fuel assemblies in the spent fuel pool may be re-inserted into the reactor core to enhance fuel utilization or replace partially-spent fuel assemblies which were determined to have defective fuel rods. The actual reactor core configuration shall be identified by the cycle Reload Report incorporated as USAR Appendix 4B.

Prior to the start of the refueling cycle, the following steps are taken to ensure proper and safe refueling operation.

a. All refueling equipment and tools are given routine and scheduled maintenance and inspection to ensure proper functioning and availability.

b. All refueling personnel are thoroughly trained in the use and maintenance of all handling equipment and tools.

c. All refueling personnel are thoroughly briefed on all refueling operations.

9.1.4.2.2 Handling Equipment

The major components of the fuel handling system are shown in Figures 9.1-7 and 9.1-8.

The following Standards and Codes are used in the design, fabrication, installation, and testing of cranes, rails, supporting structures, bridge trolley, hoists, cables, lifting hooks etc.

Crane Manufacturers Association of America (CMAA):

Specification 70

American National Standards Institute Inc. (ANSI):

B30.2.0, 1967

Safety Code for Overhead and Gantry Cranes

American Welding Society (AWS):

AWS - D14.1,70

"Specification for Welding Industrial and Mill Cranes"

American Society for Testing of Material (ASTM):

A36 Steel plates (girder, girt, end truck)

A273 Forged steel (hook)

A235, class E Forged steel (hook)

A307, 325, 409, Bolting

A242 Steel plates (girder, end tie)

A514 Steel plates (end tie)

American Iron & Steel Institute (AISI):

4147, 4150 - Alloy steel

1035, 1045, 1070 Forged steel (hook and sheave)

Federal Specifications:

RR-W-410 for wire rope

Steel Structures Painting Council (SSPC):

SSPC-SP-6-63 "Commercial Blast Cleaning"

SSPC-VIS-1 "The Pictorial Surface Preparation Standards for Painting Steel Structures"

AISC 171 lbs (Bethlehem) Crane rail (bridge)

175 lbs (USS & Bethlehem) crane rails (bridge and trolley)

The following is the list of equipment and tools required during the refueling and servicing operations:

a. Containment vessel polar crane

b. Spent fuel handling bridge and hoist mechanism

c. Auxiliary fuel handling bridge and hoist mechanism

d. Fuel transfer mechanism, including carriage drive system, fuel basket rotation system, and controls

e. Main fuel handling bridge and hoist mechanism

f. Fuel grapple

g. Spent fuel cask crane

h. Portable underwater (submarine) lights

i. Rod assembly handling tool

j. Long-handled hook and guide tool

k. Long-handled wrench tool

l. Phones and intercoms

m. New fuel handling tool

n. New fuel elevator

o. Stud tensioners

p. Stud tensioner slings

q. Stud and nut handling equipment

    1. Stud handling tools and stud handling adapters

    2. Slings

    3. Elongation gauges

    4. Stud impact wrench and adapter

    5. Stud nut wrench

    6. Stud nut handling fixture

r. Portable auxiliary neutron detectors and supports

s. Stud hole chase tool

t. Portable radiation monitors

u. Ladders

v. In-core instrument tube plug tool

w. Spare gaskets - drive housing and instrumentation flanges

x. Binoculars and cameras

y. A sufficient stock of protective clothing and badges for the maximum number of personnel required.

z. Blotting paper, cleaning equipment, rags, plastic bags, tape, log books, etc.

aa. Complete complement of tools

bb. Removable Auxiliary Work Platform

The above list is not meant to be all inclusive but is indicative of the type of tools and equipment required for refueling and servicing operations.

The containment vessel polar crane and fuel handling cask crane are provided with Seismic Class I bridge and trolley structures, crane hook brakes, and rigging. The main fuel handling bridge, auxiliary fuel handling bridge, and spent fuel handling bridge are all provided with Seismic Class I railing and restraints to prevent the bridge and associated structure from tipping over in the event of a seismic occurrence. None of the other tools or equipment are designed to Seismic Class I specifications, since they are not related to nuclear plant safety. All miscellaneous tools and equipment are stored in locations in which no damage could occur to essential equipment during a seismic event.

Two horizontal transfer tubes are provided to convey fuel between the containment vessel and the fuel storage pool. Each tube contains tracks for the fuel transfer carriage, a gate valve on the spent fuel pool side, and a flanged closure on the containment vessel side. Each fuel transfer tube penetrates into the refueling canal, inside the containment vessel, where space is provided for the rotation of the fuel transfer carriage basket, containing a fuel assembly. The other end terminates in the spent fuel storage pool, where space is provided for rotation of the fuel transfer carriage basket.

The refueling canal is a passageway in the containment vessel extending from the reactor vessel to the fuel transfer tube. This reinforced concrete enclosure, lined with stainless steel, forms a canal above the reactor vessel, which is filled with borated water for refueling. The refueling canal is also used for storage of the reactor vessel upper plenum and core barrel assemblies.

Fuel assemblies are handled, in the refueling canal, by a pneumatically actuated grapple attached to a grapple tube. A hoist mechanism raises and lowers the grapple assembly inside of a fixed outer mast. The entire mast assembly can be manually rotated through an angle of 270 degrees. This assembly is mounted on a trolley which can move laterally on a set of rails on the main bridge.

Rod assemblies are transferred, in the refueling canal, by an electric hoist driven rod grapple assembly mounted inside of an inner mast. Control, orifice and burnable poison rod assemblies can be handled by a grapple attached to the rod grapple assembly. The inner mast is raised or lowered by a hoist similar to the fuel assemblies and mounted on the main and auxiliary fuel handling bridge trolleys adjacent to the fuel mast. Rod assemblies may be transferred using

alternate manual methods, in the refueling canal or spent fuel pool. Administrative controls shall exist to ensure adequate shielding is maintained, and to ensure that excessive force is not applied to control components when using alternate manual methods. Alternate manual methods shall not be used for transferring rod assemblies within the reactor vessel.

The main and auxiliary fuel handling bridges span the refueling canal. The main fuel handling bridge is used to shuffle fuel and rod assemblies between the reactor core and fuel transfer mechanisms or refueling canal racks. It can also be used to shuffle fuel and rod assemblies within the reactor core. During refueling operations, when the main fuel handling bridge is away from the reactor core, the auxiliary fuel handling bridge may be used to shuffle fuel and rod assemblies within the reactor core, as specified by the fuel management program.

Fuel assemblies inserted or removed from the reactor core are transported between the fuel transfer tube pit and refueling canal via the fuel transfer tubes by means of a fuel transfer mechanism. There are two fuel transfer mechanisms, each consisting of a fuel transfer carriage, fuel basket, and two tilt mechanisms. The fuel transfer carriage is cable/motor driven on tracks extending from the fuel transfer tube pit to the refueling canal. The fuel basket on the carriage is rotated to a horizontal or vertical position in either the fuel transfer tube pit or refueling canal by a hydraulically operated tilt mechanism. The horizontal position is for carriage transport through the fuel transfer tube, and the vertical position is for fuel assembly insertion or withdrawal.

The spent fuel handling bridge spans the spent fuel pool and fuel transfer tube pit or cask pit, which enables the bridge to transfer fuel assemblies between the spent fuel pool storage rack positions, fuel transfer mechanisms, new fuel elevator, and the cask pit. The spent fuel handling bridge is equipped with two systems for moving a fuel assembly. The first is the mast system which is similar to the refueling canal bridge mast described above. The other system employs auxiliary hoists mounted to the north and south side of the bridge's overhead frame.

The new fuel elevator is in the fuel transfer tube pit and is provided to lower new fuel and rod assemblies to an underwater position for handling by the spent fuel handling bridge (for fuel assemblies) or by the long handled manual control rod handling tool (for rod assemblies).

The spent fuel cask crane auxiliary hook is used to transport new fuel and rod assemblies from the new fuel storage racks to the new fuel elevator, in its elevated position, by use of a new fuel or control component handling tool. The spent fuel cask crane auxiliary hook, with the long handled manual control rod handling tool, is used to shuffle new fuel rod assemblies between the new fuel elevator, spent fuel storage position and the fuel transfer mechanisms. This arrangement is also used to shuffle irradiated rod assemblies between the spent fuel storage positions and the fuel transfer mechanisms.

9.1.4.2.3 Loading and Removing Fuel

Following the reactor shutdown, containment vessel entry, and missile shield removal, the refueling process is begun. The major preparations include removing the reactor closure head and control rod drives and their service structure, installation of the seal plate access port covers, removal of the plenum assembly, and filling the refueling cavity with borated water. Other maintenance activities may be performed in the OTSG’s (e.g., tube inspection) concurrent with refueling activities by placing nozzle dams in the cold leg outlets of the OTSG.

Head removal and replacement time is minimized by the use of stud tensioners. The stud tensioner is a hydraulically operated device that permits preloading and unloading of the reactor

closure studs at cold shutdown conditions. The studs are tensioned to their operational load in two steps in a predetermined sequence. Stud elongation is verified after tensioning.

Following removal of the studs from the reactor vessel tapped holes, the studs and nuts may be supported in the closure head bolt holes with specially designed spacers. The studs and nuts are removed from the reactor closure head for inspection and cleaning using special stud and nut handling fixtures. Stud storage racks are provided.

The reactor closure head assembly is handled by a handling fixture supported from the containment vessel polar crane. It is lifted out of the canal onto a head storage stand located on the operating floor. The stand is designed to protect the gasket surface of the closure head. The lift is guided by at least two closure head alignment studs installed in some of the vessel stud holes. These studs also provide proper alignment of the reactor closure head with the reactor vessel and internals when the closure head is replaced after refueling.

The annular space between the reactor vessel and the bottom of the refueling canal is sealed off before the canal is filled by the installation of the eight access port covers.

The plenum assembly is removed from the reactor by the polar crane, using an internals handling adapter. The plenum assembly may be lifted with the canal flooded or dry. It is stored under water on one of two stands in the refueling canal.

Prior to moving any of the fuel in the reactor vessel, the refueling canal is filled with borated water. The refueling operations of the reactor core offload and reload, or in-core shuffle are conducted by main and auxiliary fuel handling bridges, two fuel transfer mechanisms, and a spent fuel handling bridge. Reactor core offload and reload are performed when plant maintenance activities require all fuel assemblies to be removed from the reactor core. Fuel assembly movement is performed in accordance with a plant procedure that directs fuel handling equipment operations described in Section 9.1.4.2.2.

If it is suspected that there are partially-spent fuel assemblies with defective fuel rods in the reactor core, then the fuel assemblies may be tested. The testing for defective fuel rods is performed in the refueling canal, spent fuel pool, fuel transfer tube pit, or cask pit as directed by the fuel management program. The testing is performed with a leakage/failure detection system and associated vendor procedures which are site approved. When a fuel assembly is identified with defective fuel rods, the repair process of either fuel reconstitution or recaging may be implemented during or after the refueling outage. In reconstitution, the defective fuel rods are replaced. In recaging, all of the sound fuel rods are transferred to a new fuel assembly cage and the defective fuel rods are replaced. Both repair processes are limited to a maximum number of replacement rods per fuel assembly by Technical Specifications. The repair processes are controlled by fuel vendor procedures that are site approved and implemented under work control administrative site procedures. The site approval of vendor procedures is based on a safety evaluation of the repair process prior to procedure approval. Fuel assemblies with defective fuel rods may be reloaded without repair.

New fuel assemblies may be prestaged in the spent fuel pool for temporary storage. Otherwise, new fuel assemblies are placed in the fuel transfer mechanism as described in Section 9.1.4.2.2 concurrently with other fuel handling refueling operations.

The discharged fuel assemblies are stored in the spent fuel pool to decay prior to being placed in on-site dry fuel storage or off-site shipment. Some of the discharged fuel assemblies may be

reused in later reactor core configurations to enhance fuel utilization or to replace partially-spent fuel assemblies determined to have defective fuel rods.

Once refueling is completed, one of the two cables to each fuel transfer carriage is disconnected and the carriage is parked, permitting the gate valve in the fuel transfer tube pit and the blind flange in the refueling canal, on each fuel transfer tube, to be closed and installed, respectively. The gate valve closure enables the refueling canal water to be drained and pumped to the borated water storage tank allowing access to the refueling canal floor to install the blind flange and reactor closure head assembly.

9.1.4.3 Shipping Spent Fuel

The spent fuel assemblies will be stored in the spent fuel pool prior to their being placed in onsite dry fuel storage or shipment offsite.

The spent fuel shipping cask can be received at the site by truck or rail.

Upon arrival, the cask is inspected for any evidence of physical damage. Any loose dirt and grime is washed off, then the cask and transport vehicle are moved inside the auxiliary building. The cask is then unloaded from the transport vehicle using the main hook on the spent fuel cask crane and placed in the cask wash area as shown in Figure 1.2-5. The cask is washed, scrubbed, and steam cleaned to remove all road dirt and grime. After thorough cleaning, the lid on the cask is unbolted, removed and stored. The cask is lifted from the wash area and lowered into the cask pit. Figure 9.1-9 shows the limits of the main hook travel on the spent fuel cask crane. If the cask pit is empty to start with it is filled with the borated water from the borated water storage tank. The bulkhead between the spent fuel pool and the cask pit is removed to establish communication between the two. The spent fuel is now picked up from the storage racks by the spent fuel bridge crane and placed into the cask. When the cask is fully loaded, still in the cask pit, the lid is placed on top of the cask to provide shielding when the cask is lifted out of the water. The cask is now lifted out of the pit and placed in the cask wash area. If needed the cask is connected to a cooling system for the removal of decay heat from the fuel assemblies. After all of the head bolts are installed and properly torqued, the cask is washed and decontaminated, and the surface radiation level is checked. When it is below the Department of Transportation limits specified in 49 CFR 171-178 it is ready for shipment. The cask is then placed on the transport vehicle and connected to its cooling system if needed and shipped offsite.

Transfer of spent fuel to the onsite dry fuel storage facility is described in Section 9.1.6.2.

9.1.4.4 Safety Provisions

Safety provisions are designed into the fuel handling system to prevent the development of hazardous conditions in the event of component malfunctions, accidental damage, or operational and administrative failures during refueling or transfer operations. A mechanical lock prevents disengagement of the fuel assembly grapple latches as long as a fuel assembly weight is carried by the grapple mechanism. The main fuel handling bridge restricts fast speed bridge and trolley movement until the fuel assembly has been completely withdrawn into the protective mast tube, but does allow the bridge or trolley to move at jog speed after the fuel assembly has been raised free of the lower grid of the core support assembly. The bridge and trolley jog speeds are low enough to preclude fuel damage by transverse impact to the exposed fuel assembly.

This auxiliary hoist is equipped with a load brake and manual operation features to ensure the safe handling of fuel during a loss of power, etc., to the hoist motor. The auxiliary hoist uses a manually actuated fuel grapple to handle fuel. The grapple design has a mechanical interlock on the grapple fingers to ensure that a fuel assembly cannot be inadvertently disengaged while suspended. Administrative controls will be used to ensure that spent fuel handling bridge drive motor is not used while fuel is being raised or lowered with the auxiliary hoist.

The new and spent fuel storage facilities are designed for noncriticality as described in Sections 9.1.1 and 9.1.2. Fuel storage racks are designed to prevent insertion of a fuel assembly in other than the prescribed locations. This design ensures a safe geometric array.

The spent fuel pool storage racks are designed for noncriticality by use of Boral neutron absorbing material. These racks are designed to prevent criticality even if non-borated water were present in the pool. The spent fuel pool storage racks are designed to prevent insertion of a fuel assembly in other than prescribed locations. The racks are designed to safely store fuel assemblies meeting the enrichment and burnup requirements specified in the Technical Specifications and the additional requirements specified in the TRM.

All spent fuel assembly transfer operations are conducted under water. The water level in the refueling canal normally provides a minimum 8-1/2 feet of water over the top of the active fuel in the spent fuel assemblies during movement from the core into storage. The depth of the water over the fuel assemblies, as well as the thickness of the concrete walls of the refueling canal, is sufficient to limit the maximum continuous radiation levels in the working area to values consistent with the radiation zoning described in Chapter 12.

The spent fuel pool storage facility water is cooled by the spent fuel cooling system as described in Subsection 9.1.3. A power failure during the refueling cycle will create no immediate hazardous condition due to the large water volume in the refueling canal, spent fuel storage pool, cask pit and transfer pit.

During the refueling period the water level in the refueling canal and the spent fuel storage facility is the same, and the fuel transfer tube valve is continuously open. This eliminates the necessity for an interlock between the fuel transfer carriage and fuel transfer tube valve operations except to verify full open valve position.

The simplified movement of a transfer carriage through the horizontal fuel transfer tube minimizes the danger of jamming or derailing. All operating mechanisms of the system are located in the fuel handling area for ease of maintenance and accessibility for inspection before the start of refueling operations.

During reactor operation, a bolted closure plate and o-ring seal on the containment vessel flange of the fuel transfer tube and the fuel transfer tube valve on the fuel handling area end of the tube provide containment vessel isolation as described in Subsection 6.2.4. The spent fuel storage facility and the refueling canal are completely lined with stainless steel for sealing and ease of decontamination. The fuel transfer tubes are appropriately attached to these liners to maintain leak integrity. The spent fuel storage facility cannot be accidentally drained.

The fuel transfer mechanism is designed to permit initiation of the carriage fuel basket rotation from the building in which the carriage fuel basket is being loaded or unloaded.

All electrical gear is located above water for greater reliability and ease of maintenance. The hydraulic system that actuates the rotating fuel basket uses demineralized water for operation to minimize contamination.

During the refueling process, the refueling canal and spent fuel cooling water will have a boron concentration high enough to ensure $\leq 0.95\ k_{eff}$. In the refueling canal, the requirement of $0.95\ k_{eff}$ may be met using a combination of boron concentration and control rods. However, the boron concentration shall provide a 1% $\Delta k/k$ subcritical condition if all control rods were removed. This ensures that the Startup Accident initial conditions in Section 15.2.1.1 are satisfied. The cycle specific boron concentration and control rod configuration may be obtained from the cycle specific Refueling Specification. Although not required for safe storage of spent fuel assemblies, in the spent fuel racks, the spent fuel storage cooling water is borated so that the refueling canal water will not be diluted during fuel transfer operations.

Each fuel handling bridge mast travel is designed to limit the maximum lift of a fuel assembly to a safe shielding depth with the water at normal refueling level. The raise limit switch setting and a stop block on the lifting chain of the spent fuel handling bridge auxiliary hoist are used to provide a safe shielding depth with the pool water at normal refueling level.

Relief valves are provided on each stud tensioner to prevent over tensioning of the studs due to excessive pressure.

Suspected leaking fuel assemblies can be removed from the core, verified for leakage, and placed in the spent fuel pool for repair, if required. During the fuel assembly repair process, the site approved fuel vendor procedures and site procedures shall ensure the following: the depth of shielding water over the fuel rods during extraction/insertion limit the radiation levels in the work area to values consistent with the radiation zoning levels described in chapter 12 and accident levels described in chapter 15, the thermal characteristics within the fuel assembly at the repair station shall not result in a boiling condition, the fuel assembly repair process shall not compromise the fuel assembly's structural integrity or reliability during operation, and the fuel repair process shall have been evaluated by the 10CFR50.59 process prior to implementation.

Structurally failed fuel is placed in a failed fuel container prior to its transfer to the spent fuel storage rack. Offsite shipment, following a suitable decay period will require that fuel be transferred to a shipping container compatible with the shipping cask design to comply with 10 CFR 71.

9.1.5 Control of Heavy Loads

9.1.5.1 Introduction/Licensing Background

Compliance with the applicable guidelines of NUREG 0612, “Control of Heavy Loads at Nuclear Power Plants”, to ensure safe handling of heavy loads over the reactor core, over spent fuel or over safety-related systems is achieved by the following. Details of DBNPS compliance to NUREG 0612 Phase I are discussed in Serial letter 774 dated 02/1/82. The NRC issued a Safety Evaluation (Log 1634 dated 10/29/84) that concluded the Phase I requirements of NUREG 0612 were satisfied for DBNPS.

NUREG 0612 Phase II was cancelled by the NRC (Log 1777). Phase II had required the evaluation of postulated load drops to evaluate the structure’s adequacy or the affect on potentially impacted systems. Regulatory Issue Summary (RIS) 2005-25 (including Supplement 1) was issued by the NRC to clarify the expectations for load drop evaluations used

to establish heavy load handling procedural guidance. The RIS states that “that licenses incorporate into the FSAR a description of the consequence evaluation and of the underlying analyses necessary to make the description complete and accurate”. The descriptions of this section provide the load drop evaluation information specified in the RIS.

9.1.5.2 Safety Basis

The safety basis that ensures that the risk associated with load-handling failures is acceptably low, based on the following evaluations and controls.

9.1.5.2.1 NUREG 0612 Phase I

NUREG 0612 Phase I required compliance with NUREG 0612 Section 5.1.1. This section specified seven defense-in-depth guidelines to reduce the risk associated with the handling of heavy loads. The NRC issued a Safety Evaluation (Log 1634) which concluded that the guidelines of NUREG 0612 Section 5.1.1 and Section 5.3 were satisfied. Each of the seven defense-in-depth guidelines and the DBNPS method of compliance is discussed Section 9.1.5.4 below.

9.1.5.2.2 NUREG 0612 Phase II

NUREG 0612 Phase II requires documentation of the specific requirements for overhead handling systems operating in the vicinity of fuel storage pools, inside containment, and in plant areas containing equipment required for reactor shutdown, decay heat removal, or spent fuel pool cooling. This documentation demonstrates that adequate measures are in place that either the likelihood of damage from a load drop is extremely small, or that the consequences of a load drop will not exceed the limits set by the evaluation criteria of NUREG 0612, Section 5.1, Criteria I, Criteria II and Criteria III. Details of these requirements and the postulated heavy load drop evaluations are discussed in Serial letter 952 dated June 10, 1983 and plant analyses, the evaluations are summarized below.

A) Containment Load Drop Evaluations

For a number of postulated load drops in containment a “system evaluation” approach was used to address the consequences of the load drop in accordance with Criterion IV of NUREG 0612 Section 5.1. For drop scenarios inside containment with the reactor vessel head installed, the system evaluations considered the reactor depressurized to about 300 psig, on decay heat removal, and cooled to less than 200°F. With the reactor vessel head removed, the system evaluations considered the plant in either “cold shutdown” or “refueling”. The following conservative assumptions were then used in evaluating the loss of equipment.

a. A direct loss of the decay heat removal system was investigated as the first step. In addition to a direct loss of the decay heat removal system, a LOCA could occur resulting in a loss of the decay heat removal system due to loss of inventory and recirculation from the sump would be required.

b. All equipment in a given containment region, including all elevations, was initially assumed lost.

c. For RCS piping or connecting piping in the region of a load drop, a RCS pipe break was assumed to occur and its effects on the core cooling was evaluated assuming

the simultaneous loss of other equipment that could be impacted in the region. If an isolation valve (normally closed valve or check valve) was located between the RCS and the drop region, a break was not assumed. In addition, potential for RCS pipe crimping was evaluated for attached RCS cooling lines.

d. In cases where electrical cabling for motor operated valves were predicted to be impacted and disabled, it was assumed the motor operated valves fail as is. The circuit protection will operate as necessary, and unless a large pipe break could occur concurrently and prevent access to the valve, manual operation of the valve could still be performed.

e. With respect to piping, it was assumed that if a pipe was in a region of concern it would be lost unless it could be shown to be safe from damage based on some other consideration such as physical distance from actual impact point of load drop.

f. With respect to pipe breaks (other than RCS) the affected system or subsystem was assumed lost unless the capability to isolate, repair, or plug the break could be shown.

g. Loss of electrical control and interlock signals, other than for required indication, were not investigated. It was assumed that if power feeds to large motors remained intact, then manual operation at switchgear or motor control centers could be performed if automatic control and interlock signals were lost.

h. Instrumentation circuits were investigated to verify that if any indication of interest were lost, redundant indication would be available. Typical verification involved tracing two redundant circuits of interest to verify that, for the region involved, at least one would not be impacted.

<u>Containment Polar Crane</u>

Due to limit switches, weight indicators, and duel electric brakes a load drop of the load block on the main and auxiliary polar crane hoist are not considered feasible. When the reactor vessel is loaded with fuel, load lifts above the reactor vessel are limited to the reactor vessel head, plenum, missile shields, reactor vessel test equipment, and associated lifting devices. Several load drops above the reactor vessel were structurally evaluated. The reactor vessel head drop was evaluated for a load of 360,000 lbs at a height of 8 feet above the reactor vessel flange with the canal drained. Since the refueling canal is drained during the reactor vessel head lifts, the head falls through air and no energy dissipation from drag was considered, and no other cushioning medium is present or accounted for in the analysis. The plenum has been evaluated for a load drop of 119,000 lbs at a height of 73.5 inches above the reactor vessel flange. The plenum load drop evaluation conservatively neglected the energy dissipation from water drag and the “dashpot” effect caused by the close tolerance between the plenum and core barrel. The missile shields were evaluated for a load drop of 94,500 lbs from a height just clear of the missile shield hold down studs. Based on procedural controls during the removal and installation, the four center missile shields would come to rest on the D-ring support ledge after a postulated drop. The remaining two outer missile barriers will not impact the service structure after a postulated drop due to their location and procedural controls during removal and installation. The results of these three

structural load drop evaluations conclude that Criteria I, Criteria II and Criteria III of NUREG 0612 Section 5.1 are satisfied for load drops above the reactor vessel.

The polar crane hoist shall not be operated over the refueling canal during fuel movements in the refueling canal. Unless necessary and in accordance with approved operating procedures stating the purpose of such use, the polar crane hoist shall not be operated over the refueling canal with fuel in the reactor vessel and the closure head removed. The polar crane hoists shall not be operated over the steam generator compartments when the reactor coolant system is above 300 psig and 200°F. Based on the location and the load lifts performed at each location, the areas of containment accessible to the polar crane have been divided into nine regions. These regions are described in Serial 952 and are procedurally controlled. The basis for acceptance of load drops in these areas is determined by either structural or system evaluations.

Reactor Service Crane

The reactor service crane shall not be used to lift loads directly over the reactor vessel except as required for control rod drive mechanism or reactor service tools. The reactor service crane shall not be operated during fuel movements. All potential load drops from the reactor service crane are enveloped by the load drop evaluations for the polar crane.

Containment Equipment Jib Cranes

The containment jib cranes shall not be operated over the refueling canal during fuel movements in the refueling canal. The containment jib cranes shall not be operated over the refueling canal with fuel in the reactor vessel and the closure head removed, unless necessary and in accordance with approved operating procedures stating the purpose of such use. The containment jib cranes shall not be operated over the steam generator compartments when the reactor coolant system is above 300 psig and 200°F. With the above procedurally controlled operating restrictions, all potential load drops from the containment jib cranes are enveloped by the load drop evaluations for the polar crane.

Containment Auxiliary Crane

The containment auxiliary crane shall not be operated when the reactor coolant system is above 300 psig and 200°F. The containment auxiliary crane shall not be operated over the reactor vessel when there is fuel in the core and the reactor head is removed. The containment auxiliary crane shall not be operated over the refueling canal during fuel movements in the refueling canal. The containment auxiliary crane shall not be operated over the refueling canal with fuel in the reactor vessel and the closure head removed, unless necessary and in accordance with approved operating procedures stating the purpose of such use. Movement of loads is limited to the minimum height required to clear the area and will follow a direct and safe load path to the lay down area. This minimizes the time and height required for each lift. The Auxiliary and Polar Cranes may handle heavy loads simultaneously within the restrictions and requirements of the applicable plant procedures, including periods when the reactor is in a defueled condition. With the above procedurally controlled operating restrictions, all potential load drops from the containment auxiliary crane are enveloped by the load drop evaluations for the polar crane.

B) Load Drop Evaluations Outside Containment

Spent Fuel Cask Crane

Loads in excess of 2430 lbs are prohibited from travel over fuel assemblies in the spent fuel pool. The spent fuel cask crane is electrically interlocked to prevent travel over the spent fuel pool, cask pit, and portions of the fuel transfer pit. The by-pass key (KSW6) for this interlock is controlled by the operations department. The key allows use of the crane's non-single-failure-proof auxiliary hook and single-failure-proof main hoist over the spent fuel pool area and cask pit. Even with the interlock bypassed, the use of the 1 ton non-single-failure-proof monorail hoist will not be allowed over the spent fuel pool, cask pit, or fuel transfer pit. Movement of loads in excess of 2430 lbs over the spent fuel pool, but not over the fuel assemblies in the spent fuel pool, is procedurally controlled.

The weight of the gates used to isolate the spent fuel pool from the transfer pit and cask pit is greater than 2430 lbs. The spent fuel pool to cask pit gate cannot be accessed by the crane auxiliary hook. The required actions for movement of the gates, including those to allow use of the appropriate crane hook, are specifically addressed in a procedure. Although the gates are not lifted directly above the fuel assemblies, the impact with the canal liner and the potential impact with the fuel assemblies caused by an unfavorable fall orientation of the divider gate have been evaluated. These drop evaluations used a weight of 8000 lbs at a maximum lift height above the pool floor of 42 feet and above the fuel of 28 feet. The drop evaluation used to calculate the number of fuel assemblies damaged by this postulated load drop considered the drag effect of the water and the energy absorbing capability of each fuel rod. Based on this evaluation, it was determined that the number of fuel assemblies damaged would not exceed the dose guidelines of NUREG 0612. The evaluation of the spent fuel pool base slab conservatively neglected the energy dissipation from drag and the structural resistance of the steel liner plate. The results of this evaluation concluded that perforations and scabbing of the floor slab were not probable and leakage of the pool is not expected.

A heavy load drop accident is not required to be postulated and the consequences do not have to be evaluated when using the single-failure-proof main hoist on the 130 ton spent fuel cask crane in the Auxiliary Building as part of a single-failure-proof handling system for cask handling operations. The drop of a loaded or empty dry spent fuel transfer cask or a dry shielded canister top shield plug in the cask pit while using the spent fuel cask crane main hoist together with a single-failure-proof handling system does not have to be postulated or evaluated.

A single-failure-proof handling system is defined in NUREG-0800 Section 9.1.5 Rev 1, Overhead Heavy Load Handling Systems, which references ASME B30.9-2003 for slings and ANSI N14.6-1993 for special lifting devices. In the Auxiliary Building a single-failure-proof handling system consists of the single-failure-proof main hoist on the spent fuel cask crane and special lifting devices meeting the enhanced safety factors of ANSI N14.6-1993 or ASME B30.9-2003 slings selected to support twice the weight of the lifted load. As discussed in NUREG-0800 Section 9.1.5 Rev 1, these single-failure-proof handling system components meet the requirements of ANSI N14.6, ASME B30.9, and applicable guidelines of NUREG 0612 for the DBNPS heavy loads program.

The postulated drop of a heavy load over the cask pit not containing fuel has been evaluated. The cask pit is located immediately adjacent the spent fuel pool. The cask

pit structure was evaluated for the drop of a heavy load weighing 97 tons at a height of 47 feet above the cask pit floor. The heavy load was considered to fall through air and no energy dissipation from drag was considered, and no cushioning medium was accounted for in the analysis. The cask pit structure was found to be adequate for the imposed load with no postulated damage to the adjacent spent fuel pool.

Other non-frequent heavy load movements over fuel assemblies are controlled procedurally and require an engineering evaluation and approval to ensure that the effects of a load drop are within the acceptance criteria if a single-failure-proof handling system is not being used.

Component Cooling Pump Monorails

To ensure the protection of the service water and component cooling water piping below the floor slab, a structural evaluation was performed for the component cooling water room floor slab in combination with a “system evaluation” approach. A postulated load drop of 5400 lbs for the component cooling water pump was evaluated from the maximum height obtainable by the monorail. The evaluation contains no energy dissipation from air drag or other cushioning medium. The results of this evaluation concluded perforations or scabbing of the floor slab are not probable and if scabbing were to occur the affect on the equipment and piping below would be insignificant. The “system evaluation” approach was also used to address the impact of a load drop on redundant trains within the component cooling water room. The “system evaluation” considered only the component cooling water and service water systems, and considered the reactor in any operating condition, including full power operation. The conclusion of this evaluation was that the redundant trains of the component cooling water and service water systems within the room are protected from load drops by their physical separation.

Intake Structure Gantry Crane

Load drop evaluations of the intake gantry crane are concerned with the operability of the service water system. The “system evaluation” for the load drop from the intake gantry crane considers the reactor in any operating condition including full power operation. Procedural controls restrict heavy loads from passing over an open roof hatch, except for the heavy load passing through the open hatch. The structural evaluations of the service water pump and valve room roofs for the postulated drops considered no energy dissipation from air drag or other cushioning medium. The structural evaluation determined the maximum safe load drop height for each of the heavy loads handled by this crane. Safe load paths have been developed for each of the heavy loads. These safe load paths are reflected and controlled in the crane operating procedure to ensure the operability of the service water system.

9.1.5.3 Scope of Heavy Load Handling Systems

The following is a listing of the handling systems at DBNPS that are subject to NUREG 0612 compliance:

- Containment polar crane/auxiliary hoist
- Reactor service crane

- Containment equipment jib cranes
- Spent fuel cask crane/auxiliary hoist
- Component cooling pump monorails
- Intake structure gantry crane
- Containment auxiliary crane

Other handling systems are not subject to NUREG 0612 compliance since, 1) they have no safe shutdown equipment located in their proximity, 2) they do not carry loads over other safety-related equipment, and are only used when the respective components have been placed out of service in accordance with plant procedures or specifications or 3) associated component handling over safety-related equipment is performed only after the plant has been safely shut down.

9.1.5.4 Control of Heavy Loads Program

9.1.5.4.1 Commitments in Response to NUREG 0612, Phase I Elements

For cranes that are within the scope of NUREG 0612, the seven defense-in-depth guidelines defined in NUREG-0612 Section 5.1.1 must be satisfied. The DBNPS commitments for establishing and controlling these guidelines were reviewed and found to be acceptable as documented in the NRC Safety Evaluation for Phase I, reference Log 1634 dated 10/29/84.

a. The movement of heavy loads is controlled using defined load paths or use of general load paths which are defined in approved plant procedures. Any deviations are administratively controlled.

b. The supervision of heavy load lifts are controlled by a designated individual.

c. Crane operators shall meet the training and qualifications that satisfy the requirements of ANSI B30.2-1976.

d. The reliability of special lifting devices is ensured by application of ANSI N14.6 design safety margins, and periodic inspection and examination using approved procedures.

e. Standard lifting devices (slings) that are selected, inspected, and maintained in accordance with ANSI B30.9-1971.

f. The inspection, testing, and maintenance of cranes is performed in accordance with ANSI B30.2-1976.

g. The design of cranes that handle heavy loads is equivalent to the requirements of CMAA-70 and ANSI B30.2-1976.

9.1.5.4.2 Reactor Pressure Vessel Head Lifting Procedures

The handling of the reactor vessel head is performed in accordance with approved plant procedures. The approved handling procedure contains the lift height restriction of 8 feet maximum over the vessel and the maximum lift weight of 180 tons. As described in Section 9.1.5.2.2.A above, the refueling canal is drained and no cushioning medium is used in support handling of the head. These procedural requirements assure that the handling of the reactor vessel head is performed in a manner that is consistent with the load drop analysis. This provides assurance that the vessel supports are adequate for the postulated head drop loads and that the reactor core will remain covered and cooled.

9.1.5.5 Safety Evaluation

The control of heavy loads is performed at DBNPS in a safe manner, using procedural controls to ensure they are performed consistent with the associated load drop analyses. The bases for this conclusion are:

- The DBNPS controls established to implement the NUREG 0612 Phase I elements (seven defense-in-depth guidelines) that make the risk of a load drop very unlikely.

  AND

- The DBNPS load drop evaluations, described, above, that conclude in the event of a postulated load drop, the consequences are acceptable. The restrictions on load height, load weight, and any required medium under the load are reflected in DBNPS procedures.

  AND

- The risk associated with the movement of heavy loads is evaluated and controlled by plant procedures.

9.1.6 <u>Spent Fuel Dry Cask Storage</u>

9.1.6.1 System Description

The Davis-Besse Nuclear Power Station (DBNPS) has established an onsite Dry Fuel Storage Facility, with the Independent Spent Fuel Storage Installation (ISFSI) pad located inside the plant protected area south of the plant, as shown on USAR Figure 1.2-12. The ISFSI pad is situated south of the Auxiliary Building and Low Level Radioactive Waste Storage Facility, and east of the Personnel Access Facility that controls access to the protected area.

Spent fuel dry cask storage operations at the DBNPS are conducted under a general license in accordance with Subpart K of 10 CFR Part 72. The general license issued by 10 CFR 72.210 authorizes a 10 CFR Part 50 nuclear power plant licensee to store spent fuel at an onsite ISFSI. Subpart K of 10 CFR Part 72 also includes 10 CFR 72.212, “Conditions of general license issued under §72.210,” which requires the use of a dry cask storage system that is pre-approved by the Nuclear Regulatory Commission, as evidenced by its listing in 10 CFR 72.214. DBNPS is required to comply with all of the general license conditions specified in 10 CFR 72.212.

The DBNPS ISFSI uses the TN Americas, LLC (TN) Standardized NUHOMS Horizontal Modular Storage System that has been approved by the Nuclear Regulatory Commission in Certificate of Compliance (CoC) No. 1004, which includes the Technical Specifications for this system. The Standardized NUHOMS Horizontal Modular Storage System is a canisterized spent fuel storage system that consists of a dry shielded canister (DSC), a reinforced concrete horizontal storage module (HSM) that encloses the DSC during long-term storage, and a steel and lead transfer cask (TC) that is used for fuel loading and DSC handling operations within the plant and DSC transfer operations outside the plant. These components are described and evaluated in the TN Updated Final Safety Analysis Report (FSAR) for the Standardized NUHOMS system (designated NUH-003.0103).

Three Standardized NUHOMS dry spent fuel storage systems were deployed at the ISFSI under the initial CoC No. 1004 (not amended) for the storage system, and these are still in service. These consist of TN NUHOMS-24P DSCs that are stored inside of HSM-80 horizontal storage modules located on the ISFSI pad, with the doors of the HSMs facing the center of the ISFSI pad for shielding and access. Each of the NUHOMS-24P DSCs contains 24 DBNPS spent fuel assemblies. Each HSM-80 has a design heat removal capacity of 24 kW. There is a fourth HSM-80 horizontal storage module that is empty and situated next to the three loaded modules for shielding purposes.

After the three HSM-80s were loaded, DBNPS installed high-density spent fuel storage racks in the spent fuel pool to increase its capacity for wet storage of spent fuel. However, with the DBNPS spent fuel pool approaching its capacity, DBNPS resumed movement of spent fuel from wet storage to dry storage using more advanced DSCs and HSMs that the NRC had reviewed and certified for use under a general license. These storage systems use NUHOMS-32PTH1 DSCs loaded into advanced, shielded HSM-H horizontal storage modules by means of the OS200 TC, included in more recent amendments to CoC No. 1004 for the Standardized NUHOMS dry spent fuel storage system. These 32PTH1 DSCs are loaded into HSM-H horizontal storage modules with the HSM-H doors also facing the center of the ISFSI pad for shielding and access. Four NUHOMS-32PTH1 DSCs were loaded in HSMs in the northeast corner of the ISFSI pad in the 2017 campaign.

In the future, DBNPS will move spent fuel from the wet storage to dry storage using the TN Americas (TN) NUHOMS EOS System that has been approved by the NRC in CoC No. 1042, which includes the Technical Specifications for this system. The NUHOMS EOS System is a canisterized spent fuel storage system that consists of a DSC, a reinforced concrete HSM that encloses the DSC during long-term storage, and a steel and lead TC that is used for fuel loading and DSC handling operations within the plant and DSC transfer operations outside the plant. These components are described and evaluated in the TN Updated Final Safety Analysis Report (FSAR) for the NUHOMS EOS System. The NUHOMS EOS System uses NUHOMS EOS-37PTH DSCs loaded into advanced, shielded EOS-HSM horizontal storage modules by means of the EOS-TC125 TC. These EOS-37PTH DSCs are loaded into EOS-HSM horizontal storage modules with the EOS-HSM doors also facing the center of the ISFSI pad for shielding and access.

The DSC provides confinement, an inert environment, structural support, and criticality control for the fuel assemblies. Each canister consists of a DSC stainless steel shell assembly (cylindrical canister shell, canister shell bottom, inner top cover, outer top cover) and a basket assembly. The primary confinement boundary for the DSC consists of the DSC shell, the inner top and inner bottom cover plates, the siphon and vent block, the siphon and vent port cover plates, and the associated welds. The DSC is designed to transfer the decay heat from the fuel to the canister body via the basket and ultimately to the ambient air via either the HSM in

storage mode or the TC during movement. The DSC basket assembly consists of plates or tubes that make up a grid of fuel compartments, designed to accommodate spent fuel assemblies.

The metal TC provides a means to lift and handle the DSC, providing the principal biological shielding, protection from potential hazards and heat rejection mechanism for the DSC during spent fuel loading, unloading, and movement of the DSC from the cask pit to the HSM at the ISFSI.

The HSM provides environmental / structural protection, shielding, and heat rejection of the DSC during storage. The HSM is a low profile, reinforced concrete structure designed to withstand all normal condition loads as well as the abnormal condition loads created by earthquakes, tornado winds and missiles, flooding, and other natural phenomena. The decay heat of the spent fuel assemblies during storage in the HSM is removed from the DSC by natural circulation convection and by conduction through the HSM walls and roof. Air enters near the bottom of the HSM, circulates and rises around the DSC and exits through shielded openings near the top of the HSM. A loaded DSC is stored within the HSM in a horizontal orientation. The DSC support structure, a structural steel frame with rails, is installed within the HSM.

9.1.6.2 Spent Fuel Loading and Unloading Operations

Loading the DSCs with spent fuel assemblies takes place in the Auxiliary Building. A detailed, step-by-step discussion of the spent fuel loading and unloading operations is included in the NUHOMS FSAR, with the site-specific process specified in the DBNPS loading and unloading procedures.

The TC is brought into the Auxiliary Building train bay in the horizontal position by means of the cask transporter. It is lifted to the vertical position and off the cask transporter's cask support skid by means of the Auxiliary Building overhead crane using the crane's single-failure-proof main hoist and the TC lift yoke attached to the TC upper trunnions (the TC lower trunnions pivot on the cask support skid pillow blocks as the cask is uprighted), and set down on a support pedestal on the floor of the cask wash pit.

After the TC is set in the cask wash pit, an empty DSC is lifted with the crane main hoist over the cask wash pit and lowered into the TC, where the TC and DSC are prepared for immersion in the cask pit for fuel loading operations. The DSC is filled with borated water and the annulus between the outer shell of the DSC and inside of the TC is filled with demineralized water and the inflatable annulus seal placed on top of the annulus to prevent contaminated water in the cask pit from entering the annulus.

The TC containing the empty DSC is lifted by the Auxiliary Building crane main hoist from the cask wash pit using the TC lift yoke and placed on a pedestal on the floor of the cask pit. Spent fuel assemblies are then transferred by means of the Spent Fuel Handling Bridge from their storage racks in the spent fuel pool through the cask pit wall opening into the cask pit and loaded into the DSC basket.

After the DSC is loaded with spent fuel assemblies, the Auxiliary Building crane main hoist is then used to place the steel shield plug on the DSC and using the TC Lift Yoke connected to the TC upper trunnions to lift the TC out of the cask pit and transfer it back to the cask wash pit.

The TC and accessible portions of the DSC are decontaminated as necessary. The DSC is drained, dried, backfilled with helium and welded shut with the inner and outer top covers welded in place using the automatic welding machine, and the vent and siphon port covers manually welded in place over the ports in the inner top cover. Helium leak testing of the DSC is performed, the top cover plate is bolted onto the TC, and the TC containing the loaded DSC is then lifted from the cask wash pit and placed on the cask transporter in the Auxiliary Building train bay, where it is lowered into the horizontal position on the cask support skid supported by the skid's upper and lower trunnion pillow blocks.

The TC is transported on the cask transporter through the rollup door out of the Auxiliary Building to the ISFSI by way of the haul path. At the ISFSI, the cask transporter positions the loaded TC in front of the empty HSM whose door is removed in preparation for DSC transfer operations. The TC top cover plate is unbolted and removed and the TC bottom ram penetration cover plate is unbolted and removed. The transfer cask is closely aligned with the HSM opening using hydraulic means mounted on the cask transporter, and TC/HSM restraints are installed. The hydraulic ram that is mounted on the cask transporter is used to push the DSC from the TC into the HSM where it slides onto the HSM rails into its storage position. The HSM shield door is then lifted in place by a mobile crane and bolted onto the front of the HSM so that the NUHOMS system is ready for long-term storage.

Unloading of spent fuel from the DSC back to the spent fuel pool, if needed, would involve essentially the reverse operation of the loading operations summarized above, with several key differences. The hydraulic ram on the cask transporter has grapple fingers that are used to grasp the permanent ring on the bottom of the DSC and pull the DSC into the TC. The DSC interior is accessed to expose the siphon and vent port quick connects. A gas sample of the atmosphere inside the DSC is taken to determine radiological characteristics. Based on the results of the analysis, ALARA requirements for the DSC unloading process would be established with special precautions taken if significant fuel failure is indicated. During DSC re-flood, the DSC is filled with borated water though its siphon port, with the vent port open and effluents routed to the plant's off-gas system. The rate of DSC fill is controlled to ensure the DSC vent pressure does not exceed the allowable pressure. The seal weld of the outer top cover plate to the DSC shell is removed by means of plasma arc-gouging, a mechanical cutting system or other suitable means.

For dry cask loading or unloading activities in the Auxiliary Building, heavy load cask handling operations associated with these activities are conducted using a single-failure-proof handling system as defined in NUREG-0800 Section 9.1.5 Rev 1, Overhead Heavy Load Handling Systems, which references ASME B30.9-2003 for slings and ANSI N14.6-1993 for special lifting devices. In the Auxiliary Building a single-failure-proof handling system consists of the single-failure-proof main hoist on the spent fuel cask crane and special lifting devices meeting the enhanced safety factors of ANSI N14.6-1993 or ASME B30.9-2003 slings selected to support twice the weight of the lifted load. As discussed in NUREG-0800 Section 9.1.5. Rev 1, these single-failure-proof handling system components meet the requirements of ANSI N14.6, ASME B30.9, and applicable guidelines of NUREG-0612 for the DBNPS heavy loads program described in USAR Section 9.1.5, Control of Heavy Loads.

TABLE 9.1-1

Spent Fuel Pool Cooling System Design Parameters
and Major Equipment Data
(Equipment Capacities are for Single Component)

Spent Fuel Pool Cooling System Performance Data	
Normal system cooling capacity, Btu/hr	10.5×10^6
System design pressure, psig	10-150
System design temperature, °F	200
Spent fuel pool volume, gallons	300,000 (gross)
Spent Fuel Pool Heat Exchangers	
Number	2
Type	Shell and Tube
Heat transferred, Btu/hr	5.25×10^6
Shell side (component cooling water)	
Temperature, inlet/outlet, °F	95/111.2
Flow rate, gpm	650
Tube side (spent fuel pool water)	
Temperature, inlet/outlet, °F	120/109.5
Flow rate, gpm	1,000
Material, tube/shell	S.S./C.S.
Spent Fuel Pool Cooling Pumps	
Number	2
Type	Horizontal, Centrifugal
Rated capacity, gpm	1,100
Rated head, ft H_20	40
Motor horsepower	20
Material	S.S.
Spent Fuel Pool Filter	
Number	1
Design flow rate, gpm	100
Design pressure, psig	150
Design temperature, °F	200
Material	S.S.

TABLE 9.1-1 (Continued)

Spent Fuel Pool Cooling System Design Parameters
and Major Equipment Data
(Equipment Capacities are for Single Component)

Spent Fuel Pool Demineralizer

Number	1
Type	Mixed bed
Design flow rate, gpm	100
Resin volume, ft^3	42.5
Material	S.S.

Borated Water Storage Tank Recirculation Pump

Number	1
Type	Horizontal, Centrifugal
Rated capacity, gpm	200
Rated head, ft H_2O	120
Motor KW	15
Material	S.S.

TABLE 9.1-2

Results of Transient Temperature Evaluations++

Scenario	Discharge Type	Coolant System Alignment	Bulk Temperature			Minimum Time to Boil (hrs)	Maximum Boil-off Rate (gpm)
			Maximum Bulk Temperature (°F)	Coincident Decay Heat Load (Btu/hr)	Time After Reactor Shutdown (hrs)		
1	Partial Core – 2 yrs at Full Power #	2 SFPCS Pumps and Heat Exchangers	132.98	15.89×10^6	183	10.42	34.45
2	Partial Core – 2 yrs at Full Power #	1 SFP Pump and Heat Exchanger	169.32	15.55×10^6	197	N/A**	N/A**
3A+	Full Core – 65 days at Full Power #	2 SFPCS Pumps and Heat Exchangers	165.87	29.66×10^6	205*	N/A**	N/A**
3B	Full Core – 2 yrs at Full Power ###	2 SFPCS Pumps and Heat Exchangers	164.90	29.28×10^6	205	N/A**	N/A**
4A	Full Core – 65 days at Full Power ##	1 DHRS Train	151.42	29.75×10^6	203*	3.78	69.57
4B	Full Core – 2 yrs at Full Power ###	1 DHRS Train	150.67	29.38×10^6	203	N/A**	N/A**

* Time for these scenarios, which evaluate an unplanned reactor shutdown 65 days after a planned refueling (which discharged 72 fuel assemblies), is measured from the start of the unplanned shutdown.

** Boiling evaluations are not performed for Scenarios 2, 3A, and 3B as they are non-typical operating lineups. Boiling evaluations are performed of Scenario 1 and the most limiting of Scenarios 4A and 4B (highest bulk temperature and decay heat flux).

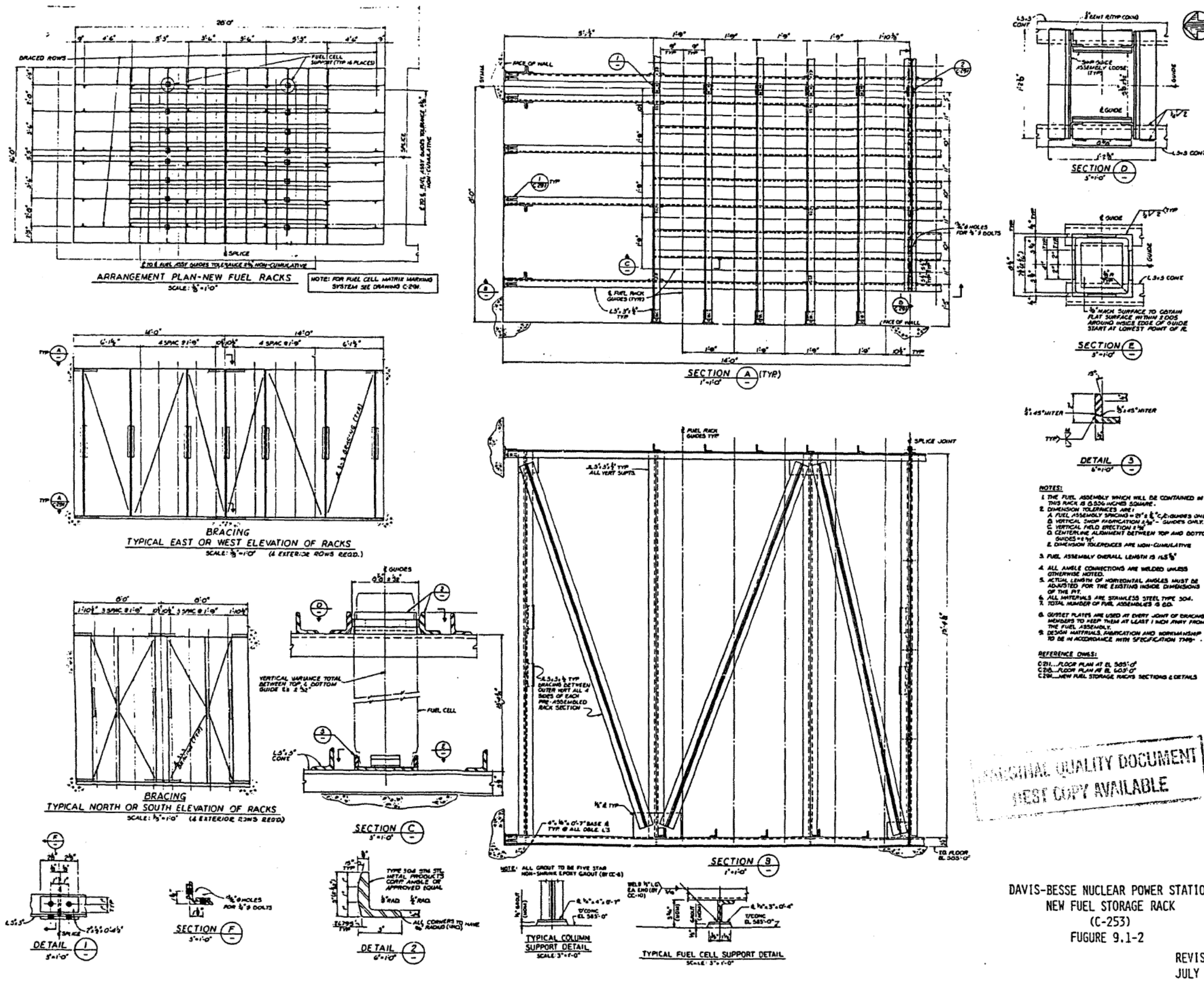
\# Discharge of 72 assemblies which had been a power 2 yrs., following a refueling outage 2 years earlier that discharged 72 assemblies which had been a power 2 yrs.

\## Discharge of 177 assemblies which had been at power 65 days, following a refueling outage 65 days earlier that discharged 72 assemblies which had been at power 2 yrs.

\### Discharge of 177 assemblies which had been at power 65 days, following a refueling outage 2 years earlier that discharged of 72 assemblies which had been at power 2 years.

\+ The maximum heat load – 30.15×10^6 – occurred in Scenario 3A prior to reaching the maximum bulk temperature.

++ These evaluations are based on the Component Cooling Water temperature being 95°F. (See Reference 3.12-91 for results with CCW at 97°F due to re-evaluation of the Ultimate Heat Sink).



NOTES:

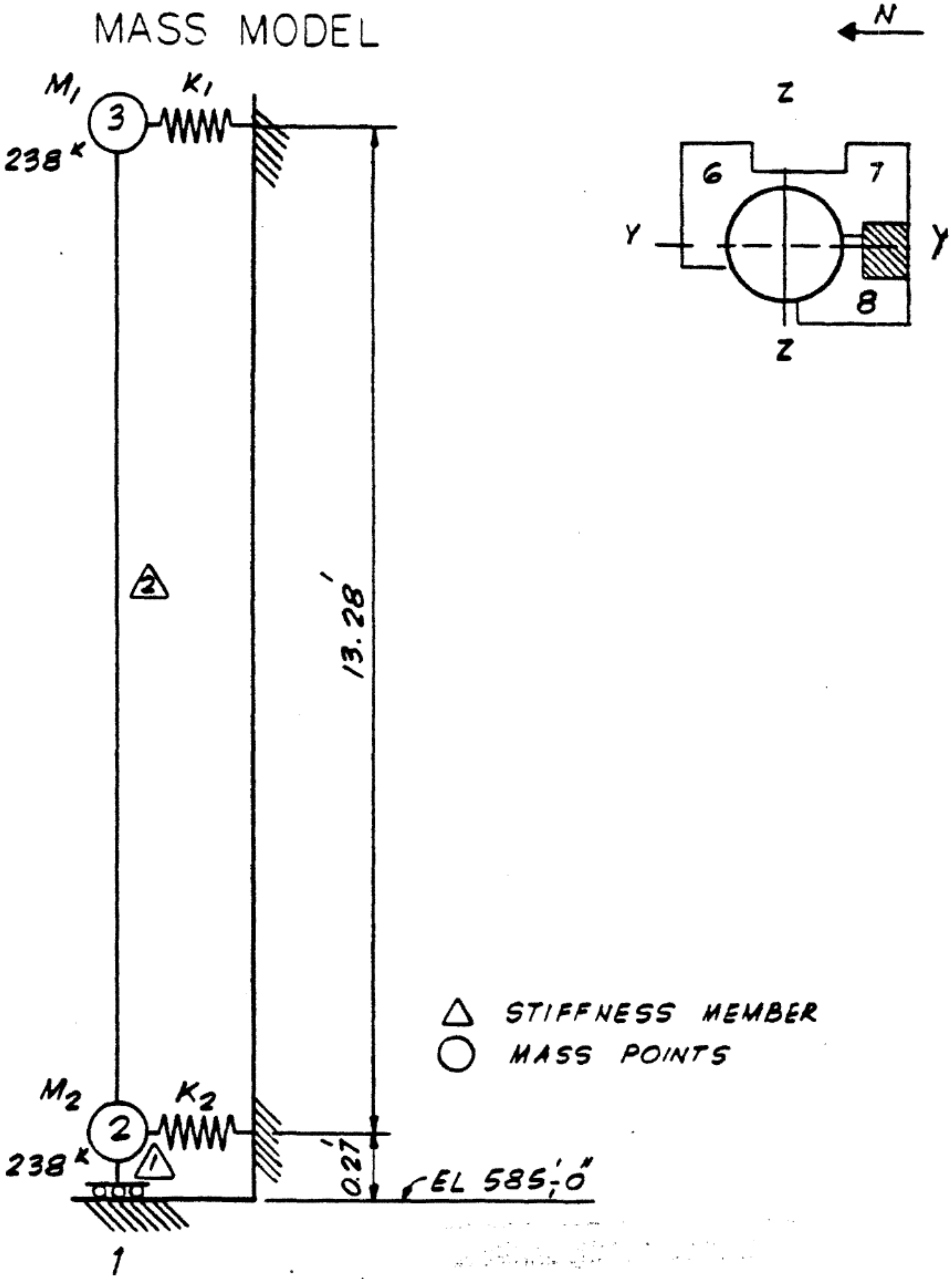
1. THE FUEL ASSEMBLY WHICH WILL BE CONTAINED IN THIS RACK IS 8.536 INCHES SQUARE.
2. DIMENSION TOLERANCES ARE:
   A. FUEL ASSEMBLY SPACING = 21" ± 1/8" C/C - GUIDES ONLY.
   B. VERTICAL SHOP FABRICATION ±1/32" - GUIDES ONLY.
   C. VERTICAL FIELD ERECTION ±1/8"
   D. CENTERLINE ALIGNMENT BETWEEN TOP AND BOTTOM GUIDES = ±1/32".
   E. DIMENSION TOLERENCES ARE NON-CUMULATIVE
3. FUEL ASSEMBLY OVERALL LENGTH IS 165 5/8"
4. ALL ANGLE CONNECTIONS ARE WELDED UNLESS OTHERWISE NOTED.
5. ACTUAL LENGTH OF HORIZONTAL ANGLES MUST BE ADJUSTED FOR THE EXISTING INSIDE DIMENSIONS OF THE PIT.
6. ALL MATERIALS ARE STAINLESS STEEL TYPE 304.
7. TOTAL NUMBER OF FUEL ASSEMBLIES IS 60.
8. GUSSET PLATES ARE USED AT EVERY JOINT OF BRACING MEMBERS TO KEEP THEM AT LEAST 1 INCH AWAY FROM THE FUEL ASSEMBLY.
9. DESIGN MATERIALS, FABRICATION AND WORKMANSHIP TO BE IN ACCORDANCE WITH SPECIFICATION 7749-.

REFERENCE DWGS:

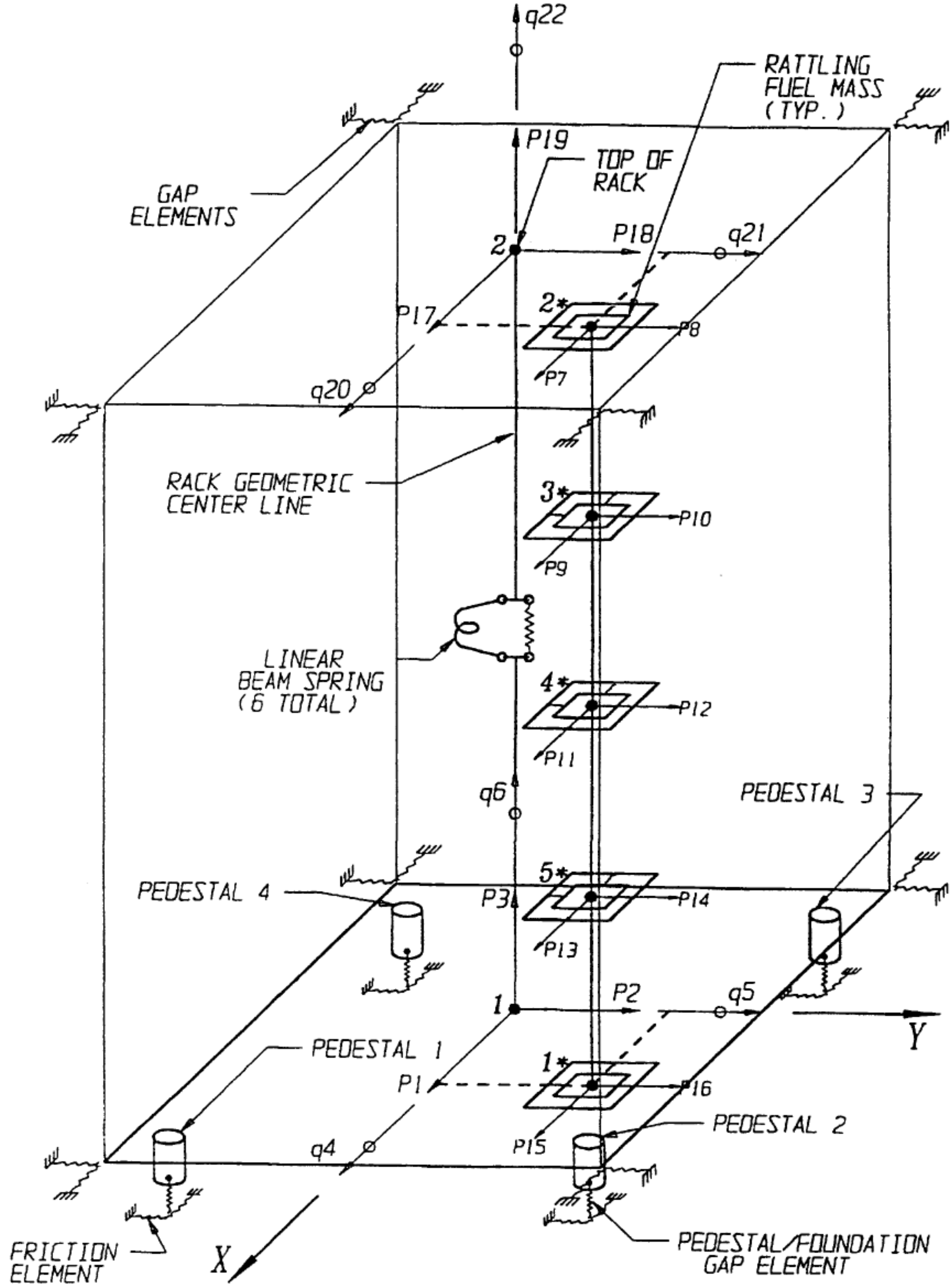
C211...FLOOR PLAN AT EL 585'-0"
C218...FLOOR PLAN AT EL 603'-0"
C291...NEW FUEL STORAGE RACKS SECTIONS & DETAILS

DAVIS-BESSE NUCLEAR POWER STATION
NEW FUEL STORAGE RACK
(C-253)
FUGURE 9.1-2

REVISION 0
JULY 1982



DAVIS-BESSE NUCLEAR POWER STATION
MASS MODEL FOR NEW FUEL STORAGE RACKS
FIGURE 9.1-3

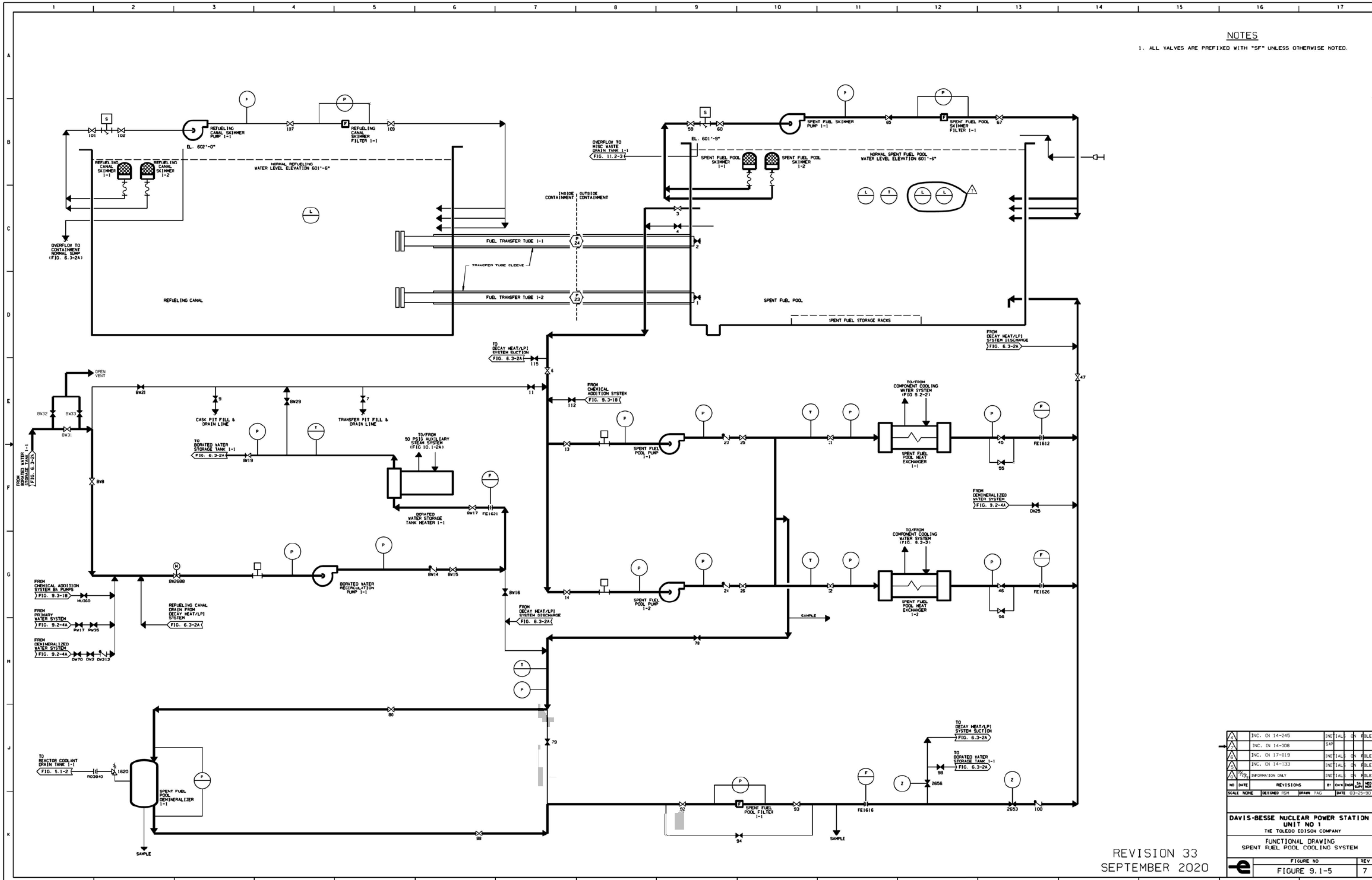


REPORT # HI-981933

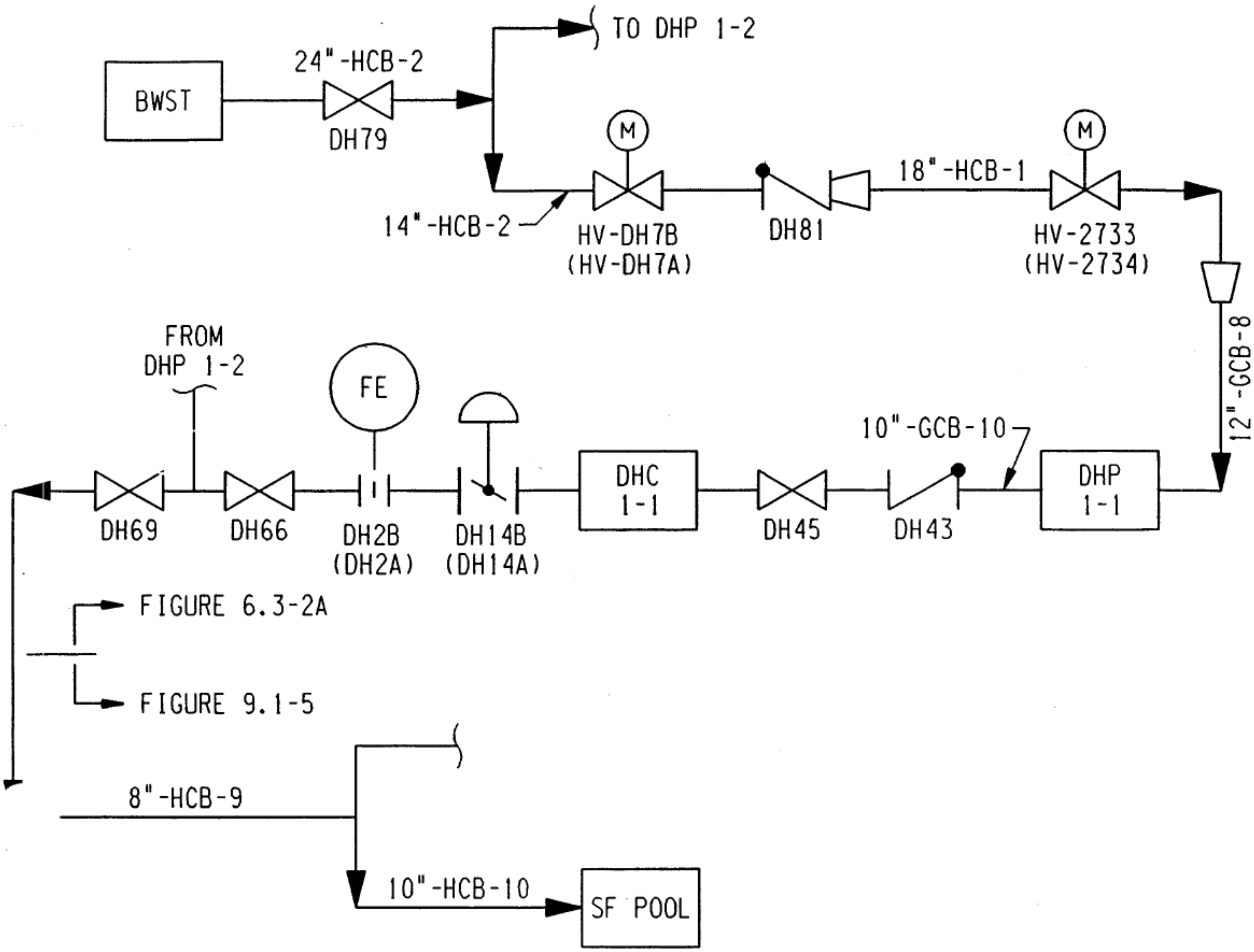
DAVIS-BESSE NUCLEAR POWER STATION
SCHEMATIC OF THE DYNAMIC MODEL OF A
SINGLE RACK MODULE USED IN DYNARACK
FIGURE 9.1-3B

REVISION 22
NOVEMBER 2000

8-02-00 DFN=/G/USAR/UFIG913B.DGN/CIT



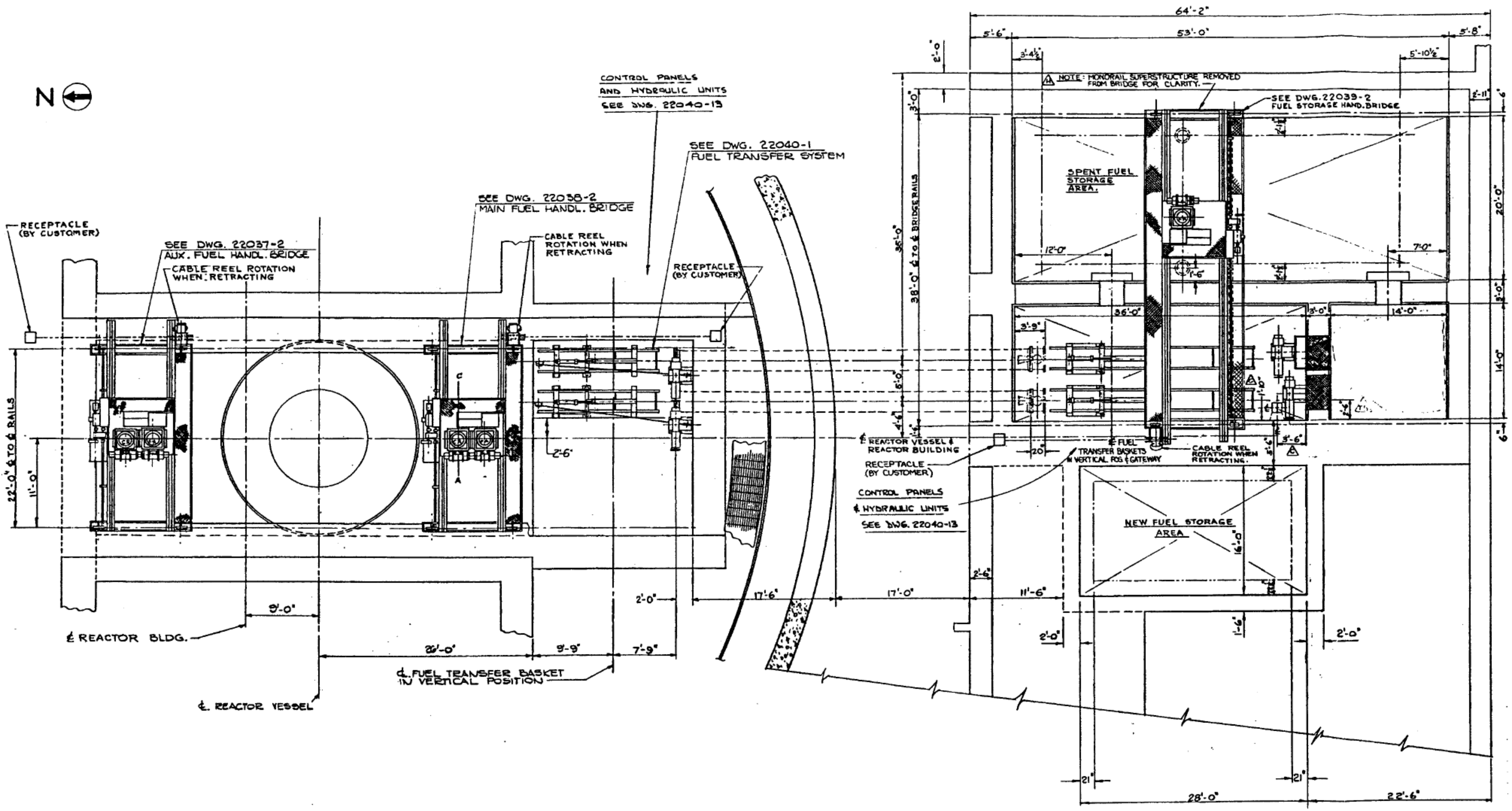
REVISION 33
SEPTEMBER 2020



NOTES:
1. THE VALVE NUMBERS FOR MANUAL VALVES ARE IDENTIFIED FOR DECAY HEAT PUMP 1-1 STRING.
2. THE VALVE NUMBERS IN PARENTHESES ARE FOR THE DECAY HEAT PUMP 1-2 STRING.
3. THE ABOVE VALVE LINE-UP SHOWS THE ASSOCIATED PUMP, PIPING AND VALVES USED TO SUPPLY THE SPENT FUEL POOL MAKE-UP WATER, THE OTHER UNRELATED LINES HAVE BEEN OMITTED.

DAVIS-BESSE NUCLEAR POWER STATION
SPENT FUEL POOL MAKE UP WATER FROM
SEISMIC CLASS 1 SYSTEM
FIGURE 9.1-6

REVISION 21
NOVEMBER 1998



NOTES:
1.) FOR ELEVATION SEE DWG. 22036-1
2.) UPPER STRUCTURE NOT SHOWN IN BRIDGE PLAN VIEWS.
3.) REF. DWGS. B&W. DWG. NO 136186E REV. 3
136187E REV. 3
136188E REV. 3

TOLERANCES UNLESS OTHERWISE SPECIFIED					
	[illegible]	6" TO 12" INC'L	12" TO 36" INC'L	36" TO [illegible] INC'L	[illegible] & OVER
DECIMAL DIM	[illegible]	[illegible]	[illegible]	[illegible]	
FRAC DIM MACH	[illegible]	[illegible]	[illegible]	[illegible]	
FRAC DIM NON MACH	[illegible]	[illegible]	[illegible]	[illegible]	
⊕ OF DRILL & TAP HOLES	[illegible]	[illegible]	[illegible]	[illegible]	
ANGLES	[illegible]	[illegible]	[illegible]	[illegible]	
FINISHES MACHINED	[illegible]				

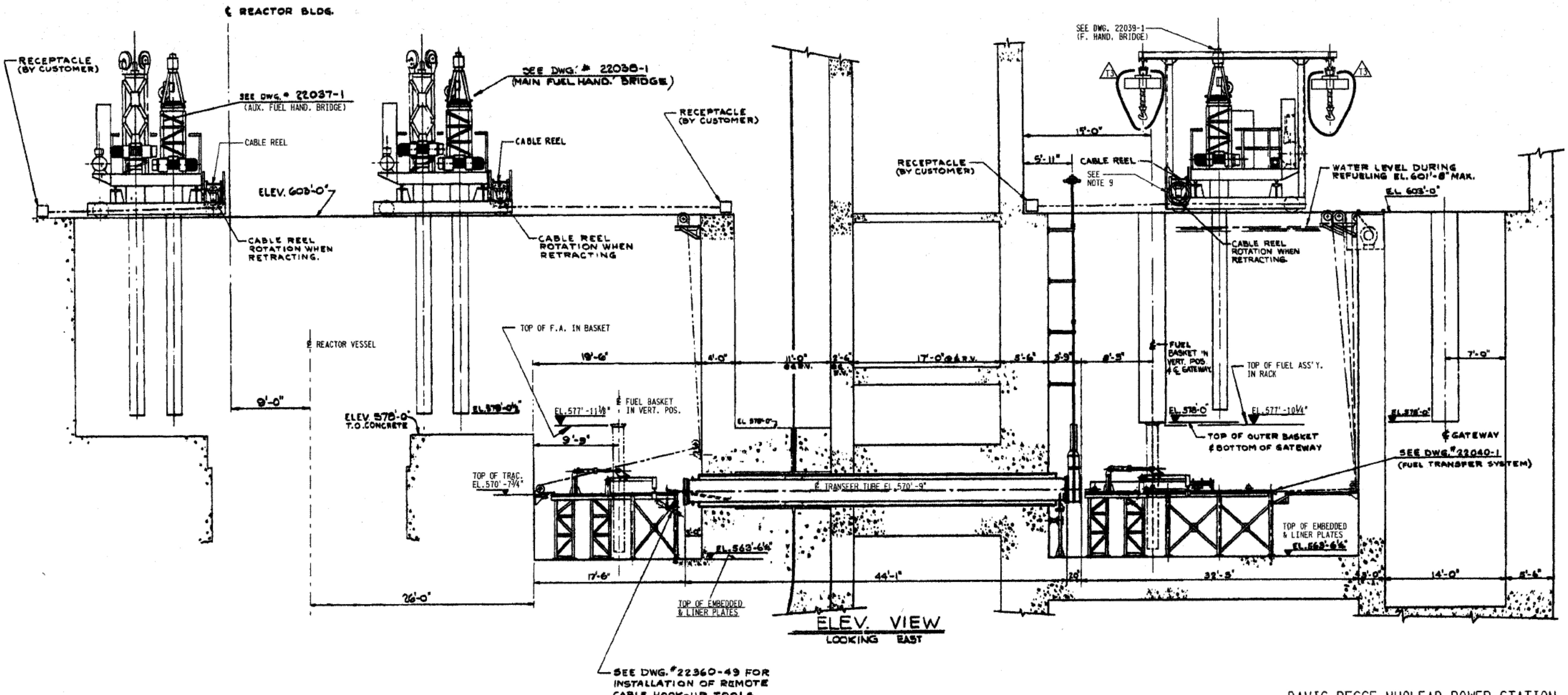
DAVIS-BESSE NUCLEAR POWER STATION

FUEL HANDLING SYSTEM (PLAN)

FIGURE 9.1-7

REVISION 24
JUNE 2004

NOTES:
1. FOR PLAN VIEW SEE DWG. 22036-2
2. FUEL T. MECH. ANCHOR BOLT LAYOUT DWG.22036-3 & 22036-8.
3. FUEL HAND. BRIDGES ANCHOR BOLT LAYOUT (REACTOR) DWG.22036-4.
4. SPENT FUEL F.S.H. BRIDGE ANCHOR BOLT LAYOUT DWG.22036-5.
5. REF. DWG'S. B.& W. NO. 136196E, REV.3, 136187E, REV.3 & 136188E, REV.3.
6. FOR "CABLE DRIVE INSTALLATION ASS'Y. ARRANGEMENT", SEE DWG. 22040-38.
7. FOR "CABLE HOOK-UP INSTALLATION" SEE DWG. 22360-49.
8. FOR "CARRIAGE RAIL MODIFICATION ASS'Y" SEE DWG. 22040-40.
9. CABLE REEL IS SHOWN FOR ROTATION REFERENCE FOR LOCATION AND MOUNTING DETAILS SEE VENDOR DWG. M-519-82.



DAVIS-BESSE NUCLEAR POWER STATION
FUEL HANDLING SYSTEM (SECTION)
M-519-9
FIGURE 9.1-8
REVISION 27
JUNE 2010

DB 03-29-10 DFN=/J/RASDGN/M5199.DGN/CIT

Removed in Accordance with RIS 2015-17

DAVIS-BESSE NUCLEAR POWER STATION
SPENT FUEL CASK CRANE TRAVEL
FIGURE 9.1-9

9.2 WATER SYSTEMS

9.2.1 Service Water System

9.2.1.1 Design Bases

The service water system is designed to serve two functions during station operation. The first function is to supply cooling water to the component cooling heat exchangers, the containment air coolers, and the cooling water heat exchangers in the turbine building during normal operation. The second function is to provide, through automatic valve sequencing, a redundant supply path to the engineered safety features components during an emergency. Only one path, with one service water pump, is necessary to provide adequate cooling during this mode of operation.

The Seismic Class I service water pumps are sized to provide cooling water to the component cooling heat exchangers, containment air coolers, and the emergency core cooling system room cooling coils. Two redundant pumps, of 100 percent capacity each, are provided to back up the operating pump.

The service water system also provides a backup source of water to the auxiliary feedwater system and the Motor Driven Feedwater pump (MDFP). During normal operation service water discharge provides makeup for the circulating water system.

The portion of the system required for emergency operation, including the intake structure, is designed to the ASME Code, Section III, Nuclear Class and Seismic Class I, as applicable. This includes protection from a tornado and tornado missiles. The associated containment penetrations are Nuclear Class 2.

Applicable design codes and standards are listed in Table 9.0-1. Design parameters for the system are listed in Table 9.2-1. The degree of compliance with the single-failure criterion is discussed in Subsection 9.2.1.3. The arrangement of the service water pumps, important dimensions of the pump room, and the minimum and extreme high water levels, including the maximum flood, are provided in Figure 2.4-21.

9.2.1.2 System Description

The service water system provides water for the following (Figure 9.2-1):

Normal Operation:

a. Component cooling heat exchanger cooling

b. Cooling water heat exchanger cooling

c. Turbine condensate polishing demineralizer system (as necessary)

d. Containment air cooler cooling

e. Emergency core cooling system room cooling coils (as necessary)

f. Cooling tower makeup

g. Water jet exhausters

h. Blowdown condenser

i. Motor Driven Feedwater Pump lube oil & seal water coolers

Emergency Operation:

a. Component cooling heat exchangers

b. Containment air coolers

c. Emergency core cooling system room cooling coils

d. Auxiliary feedwater system

e. Control room emergency condenser cooling

f. Component cooling water system makeup

g. Hydrogen dilution blowers

h. Motor Driven Feed water Pump

Three service water pumps are part of the system. They are installed in the intake structure and use Lake Erie as a source of water. The intake structure is chlorinated to prevent slime and algae growth in the system. Two pumps are used in normal operation. Motor-operated strainers at the pump outlets filter any material that may plug heat exchanger tubes and the orifices of the Auxiliary Feedwater pump bearing oil cooler, turbine bearing cooler, and governor oil cooler.

The combined flow leaving the system is normally returned to the circulating water system as makeup. This flow may also be diverted to the intake structure to prevent icing in winter. All Class I piping which passes through the turbine building is enclosed in a Class I tunnel.

The three service water pumps are piped to two separate interconnected but isolated supply paths. Interconnecting switching paths are provided on each of the system heat exchangers. Connections are also provided on the system so that a source of water is available to (1) the component cooling water system, (2) condensate polishing demineralizer system, (3) the suction of the auxiliary feed pumps, (4) the water jet exhausters, (5) Motor Driven Feedwater pump, (6) Motor Driven Feedwater Pump lube oil & seal water coolers and (7) the blowdown condenser.

The service water discharge can be redirected from the cooling tower to the forebay when required to maintain water level in the forebay above elevation 564 feet International Great Lakes Datum (IGLD).

If the supply pipe from Lake Erie to the intake structure is lost during an earthquake, the system will use the intake forebay as a reservoir and cooling pond. This is accomplished by motor-operated valves which block the system return flow to the cooling tower and open another return path to the intake forebay. Operation of the system under this condition is as follows:

Although not part of the Licensing Basis for a combination of events, if a LOCA and seismic event are considered to occur together when the forebay level is above elevation 564 feet IGLD, the actions of the operator to ensure the return of service water to the forebay are dependent on the consequences of the seismic event.

a. If the seismic event causes the blockage of service water as a result of the failure of the nonseismic line to the cooling tower, a pressure switch will cause the intake forebay return line valve to open. Should this valve fail to open, another switch will cause the intake structure de-icing line valve to open after a time delay. With the discharge through the de-icing line, the water temperature will reach the maximum 113°F in approximately 2 days. This time is based on an analysis assuming El. 562 feet IGLD. The data used in the analysis are as follows:

Water temperature (initial), °F	90
Service water flow rate, gpm	10,000
Forebay water level, ft IGLD	562
Heat load, Btu/hr.	Variable (LOCA load)
Surface area, ft^2	225,031
Dry bulb temperature (max.), °F	81.6
Dew point temperature (max.), °F	73.7
Wind speed, mph	7.4
Mean solar radiation, Btu/ft^2	2,362
Time for water temperature to reach 113°F, days	2

The calculated temperature would be lower if the water elevation is higher than 562 feet IGLD in the forebay.

If there is a failure of the intake forebay return valve to open, the operator must leave the control room to open it manually.

b. A seismic event could cause simultaneous breaks in the nonseismic pipes connected to the Service Water system as follows: (1) a break in the nonseismic portion of the discharge line in operation to the cooling tower or collection box, and, (2) a break in the Turbine Plant Cooling Water (TPCW) system piping with failure of the TPCW system to isolate from the Service Water system (i.e., closure of SW1395 or SW1399). If these breaks are not blocked the operator must, (1) open the intake forebay discharge line valve (SW2930) and close the cooling tower or collection basin discharge line valve to stop flow out of the break and, (2) close SW45 or SW46 to isolate the break in the TPCW system piping. Since the nonseismic forebay connection to the lake must be considered nonfunctional, these two breaks will cause the forebay to drop from elevation 564 to 562 feet IGLD in 2.2 hours. This time period is sufficient to open/close valves as needed to switch the nonseismic discharge line to a seismic discharge line and to isolate the break in the TPCW system under any conditions. Failure of the intake forebay return line valve to open would require action outlined in Paragraph (a).

c. The water entrapped in the Seismic Class I area plus 1/3 of the non-Seismic Class I area of the forebay is sufficient to provide plant cooldown without using the Condensate Storage Tanks. The results of the analysis to substantiate this fact are outlined in Subsection 9.2.5. It is noted that although the use of minimum usable volume of the Condensate Storage Tanks is considered in Subsection 9.2.5.1, it is not necessary for the tanks to be available in order to

provide the necessary cooling required. The minimum usable volume of the Condensate Storage Tanks of 270,300 gallons is small compared with the approximately 6,700,000 gallons of water contained in the Seismic Class I area plus 1/3 of the non-Seismic Class I area of the forebay. There is sufficient margin available in the intake forebay volume to provide the extra 270,300 gallons for use by the Auxiliary Feedwater system.

d. Operation of the steam generators for 24 hours while injecting raw water would evaporate about 230,000 gallons of water. Based on raw water chemistry data from October 1995 through October 1997, this would result in a maximum of 1462 pounds of solids accumulating in each steam generator. This has been documented in B&W Canada calculation 205S-A161. This quantity of solids would not adversely affect the heat transfer capability of the steam generators. This capability has been documented in the following reports:

   1. "Secondary Side Chemistry Investigation in the 37 Tube Nuclear OTSG", Research Center Report 1459-2/6/70.

   2. "19 Tube OTSG Fouling Test", E.W. Dotson, ARC Letter Report 69-2218-45-2, 12/31/69.

   3. Babcock & Wilcox Canada Engineering Calculation 205S-A161, "Davis-Besse Unit 1 ROTSG Heat Transfer Evaluation Using Lake Erie Water"

The flow area of the intake forebay is many orders of magnitude larger than that of the service water return header. In turn, the velocity at the intake forebay is much less than the return header. Therefore, any particulate matter that could cause pipe plugging will settle out before reaching the service water pumps. Should an occasional large particle (over 1/16 inch) enter the service water pumps, the service water pump strainers will prevent it from entering the system.

Chlorination of the intake structure serves to prevent slime and algae growth in the system. A Sodium Bromide solution is mixed with the Sodium Hypochlorite solution to enhance the biocidal effectiveness of the intake structure water treatment without increasing the level of chlorine. Should the Sodium Bromide portion of the system not be available, Sodium Hypochlorite solution may be used alone.

The low water velocities at the intake structure, which result in settling of discrete matter upstream of the screens, the service water pump strainers, and chlorination with Sodium Bromide prevent any plugging of the system, including the 30-inch service water return header. Furthermore, as illustrated in Table 9.2-2 the intake water contains only a small amount of suspended solids, which also reduces the possibility of plugging in the service water lines.

The piping takeoff to the auxiliary feed pumps is from two independent Class I lines.

The piping takeoff to the Motor Driven Feedwater Pump is from the Service Water Loop 1 Class I line prior to Component Cooling Water Heat Exchanger 1-1.

The service water system is designed to prevent any component failure from curtailing emergency operation. It is possible to isolate all heat exchangers and pumps on an individual basis. Additionally, the dilution pump, P-180, can supply water to the Service Water System from the intake structure in the event of a fire disabling the Service Water Pumps.

The cooling water supply and return lines to and from the containment air coolers are provided with remotely operated isolation valves. The cooling water supply and return lines are also provided with components to eliminate or minimize the effects of a postulated waterhammer or transient hydraulic event in the event of loss of internal service water pressure.

A check valve is provided in the containment air cooler (CAC) 1-1 service water supply line to prevent water from draining and voiding the CAC 1-1 service water piping. Vacuum breakers are provided in the CAC 1-1 service water return piping to admit air and prevent voiding of the service water piping.

A check valve is provided in the containment air cooler (CAC) 1-2 service water supply line to prevent water from draining and voiding the CAC 1-2 service water piping. Vacuum breakers are provided in the CAC 1-2 service water return piping to admit air and prevent voiding of the service water piping.

Vacuum breakers are provided in the common service water return header to admit air and prevent voiding of the service water piping.

The Service Water System return piping in the service water tunnel has been designed as Seismic Class I.

9.2.1.3 Safety Evaluation

The service water pumps have been designed to operate with Lake Erie low water level. In the event of extremely high water levels, pump operation is ensured since the pump room is sealed to prevent flooding.

Pump head requirements are based on clean-tube friction factors. The containment air coolers have been designed with a cleanliness factor of 75 percent.

The Service Water piping and components in containment are seismic category 1 and therefore eliminates the possibility of radioactivity leaking out of containment through this source in the event of a LOCA. The possibility of radioactive leakage to the service water side of the component cooling heat exchangers is not considered because this would mean a concurrent primary system leak and tube leakage from the component cooling heat exchanger.

No corrosion inhibitors are used in the system. To prevent failure of the piping system due to corrosion, the piping wall thickness used is greater than the minimum required for the design pressures and temperatures.

During emergency operation each normally operating containment air cooler fan receives a Safety Features Actuation Signal (SFAS) to start in slow speed. The slow speed operation of the fans is interlocked with the service water valves of the associated containment air coolers, resulting in the normally open motor operated inlet valves being commanded to open, and the pneumatic operated outlet valves being fully opened. With a safety actuation signal also blocking flow to the cooling water heat exchangers, the system is ensured of having redundant paths available for safe operation.

However, in the event of a loss of offsite power (LOOP), a possibility exists for a water hammer in the Containment Air Coolers Service Water piping due to the stopping and subsequent restarting of the Service Water Pumps. To mitigate the pressure transient experienced due to

the water hammer event and prevent damage to the Service Water piping and Containment Air Cooler coils, this piping and the Containment Air Cooler coils are isolated following a LOOP. To accomplish this, the Containment Air Cooler Service Water inlet valves (SW1366, SW1367 and SW1368) are signaled to close upon restoration of electrical power following the LOOP event. Service Water flow to the Containment Air Coolers is then manually restored if the CAC fans were operating in fast speed prior to the LOOP. If the CAC fans were operating in slow speed prior to the LOOP, Service Water flow will be automatically restored (in the same sequence that occurs when a LOCA signal is present, as described in the following discussion). If a LOCA signal is present in conjunction with the LOOP, the inlet valves automatically open, following a time delay, to a preset throttled position, if the associated CAC fan is running, to refill the Service Water piping. Once the piping is refilled, the Containment Air Cooler Service Water inlet valves fully open to establish design flow rates through the coils.

Although LOCA and a seismic event are not assumed to occur together, non-seismic equipment can not be credited to mitigate consequences of a LOCA. Since the instrument air system is not seismically qualified it can not be credited following a LOCA. Post-LOCA dose rates in the mechanical penetration rooms are high and will prevent entry to manually close the pneumatic operated outlet valves. Therefore, to cope with the postulated loss of instrument air, the pneumatic operated outlet valves are provided with backup nitrogen tanks to support the containment cooling and containment isolation functions. The air tanks are sized to support three valve strokes and maintain the valves closed for thirty days.

The forebay supply line from Lake Erie and the Service Water Discharge lines to the cooling Tower are not seismically qualified. Therefore, although LOCAs and seismic events are not postulated to occur concurrently, these pipes may not be credited for accident mitigation. Analyses have been performed to demonstrate that all post-LOCA safety functions will be performed despite the adverse function of these parts of the system. In the event that one of the non-seismic Service Water discharge lines is in use when a LOCA occurs, the line may be partially or completely blocked. If the pressure in the safety-related common discharge header rises above the setpoint of PSH2930 and PSH2929, one of the seismic flowpaths will be automatically established. Should the common discharge header pressure remain below the pressure switch setpoint, administrative controls have been established for the operators to manually establish a safety-related Service Water discharge flowpath. The automatic transfer and manual actions assure that a safety-related Service Water discharge flow path is always established. To prevent Forebay level from falling below 562 feet IGLD, the administrative controls also require that flow through the non-seismic path must then be terminated within 2.7 hours of reaching a Forebay level of 564 feet during a LOCA.

Service Water Pump 1 is supplied from Essential Bus CI, Pump 2 from Essential Bus D1, and Pump 3 from either Bus C1 or D1 by means of manual transfer switches CD. The Service Water pumps will restart after a Loss of Offsite Power. A time delay is included in the circuit. See Section 9.2.7.3 for additional discussion of the time delay design basis.

The valves and strainers in the independent paths are energized from two independent electrical channels. Valves for Service Water Pump 1 are supplied by Channel 1 via Motor Control Center E12. Valves and strainers for Service Water Pump 2 are supplied by Channel 2 via Motor Control Centers F12. Valves and strainers for Service Water Pump 3 are supplied from either Channel 1 or 2 by means of manual transfer switches.

A single-failure analysis has been made on components of the system to show that a single failure of any component as shown in Table 9.2-3 will not prevent fulfilling the design functions.

When Service Water inlet temperature is less than the maximum design value, it is possible to provide the required heat removal duty of safety and non-safety related heat exchangers with Service Water flow rates below the design value. At sufficiently low temperatures, Service Water may be manually bypassed through the spare CCW heat exchanger to maintain more desirable Service Water pump flow and discharge pressure. When operating in this manner, operating limits ensure that all safety related heat exchangers are capable of immediately providing adequate cooling capacity following a safety actuation signal. Following accidents where it is required, the Service Water system will be manually realigned to terminate bypass flow operation. This will be accomplished before design heat removal capability can be adversely impacted by any subsequent increase in forebay temperature.

9.2.1.4 Tests and Inspection

Except for the underground piping, the equipment, piping, valves, and instrumentation are arranged so that all items can be visually inspected. The containment air cooling units and associated piping are located outside the secondary concrete shield around the reactor coolant system loops. Personnel could enter this area of the containment vessel during station operation for emergency inspection and maintenance of this equipment. Operational tests and inspection were performed prior to initial startup to demonstrate the capability of the Service Water System to respond to an SFAS signal. The method used was to insert an SFAS signal with the system in normal operation to initiate SFAS operation.

9.2.1.5 Instrumentation

The operation of the service water system is monitored by the following instrumentation:

a. Pressure indicators and low-pressure alarms on pump discharge lines

b. High differential pressure alarms on strainers in pump discharge lines

c. Temperature indicators in supply headers to the service water system

d. Pressure indicators for the supply to each emergency core cooling system room cooling coil units

e. Deleted

f. Temperature indicators on the return lines from the containment air coolers, component cooling, and cooling water heat exchangers

g. Remote indication of service water pump motor bearing and stator temperature

h. A test point in the service water tunnel to determine system flow

i. Pressure indicators on the return side of the component cooling water heat exchangers

j. Flow elements and pressure differential indicators in piping to determine flow to individual and combined flow to the ECCS Room Cooler coils

k. Flow elements and pressure differential indicators in piping to determine individual flows to containment air coolers

l. Computer alarms to ensure proper operation of the service water return header valves

m. Redundant alarm channels warn of abnormal forebay level. Each channel has a high-level, low-level, and low-low-level alarm. Each channel is powered and routed separately.

n. Flow elements and associated pressure differential indicators to determine flow through individual CREVS water cooled condensers during tests.

Pressure differential switches on service water pump strainers reset to 1.9 psid (strainer motor starts and blowdown valve opens) and 2.5 psid (high D/P alarm). Previous settings were too low for the actual pressure differential experienced in the system at normal flows and with a clean strainer.

Setpoints were changed on PSH 2917A, 2918A, and 2919A to 116 psig, 105 psig, and 114 psig respectively; and on PSH 2917, 2918, and 2919 to 98 psig, 93 psig and 98 psig respectively. This change allows the strainer blowdown valves to open properly before the relief valves SW 3963 are activated.

9.2.2 Component Cooling Water System

9.2.2.1 Design Bases

The Component Cooling Water (CCW) System is designed to provide cooling water to reactor auxiliaries and ECCS systems during normal station operation and Design Basis Accident (DBA) conditions. The components of the system are sized on the basis of removing the maximum heat load during normal station operation with 90°F service water temperature, and removing maximum heat loads from ECCS components during DBA conditions with service water at the ultimate heat sink conditions.

The system is designed to provide maximum reliability during normal operation and to meet single failure criteria during DBA conditions.

In addition, the CCW System provides cooling water to support makeup pump operation as described in Section 9.3.4.2 during feed-and-bleed operations.

Design parameters for the major components of the system are listed in Table 9.2-5.

9.2.2.2 System Description and Evaluation

9.2.2.2.1 System Description

The functional drawing for the Component Cooling Water System is shown in Figure 9.2-2.

The part of the system required during DBA conditions is separated into two redundant loops, with each loop capable of supplying 100 percent of the cooling water required under those conditions.

During normal operation, one of the loops will supply cooling water to reactor auxiliaries with the other loop in a standby capacity. During DBA conditions the nonessential portion of the system is automatically isolated from both loops.

The CCW system provides cooling water to the reactor auxiliaries and ECCS systems listed in Table 9.2-4.

Three CCW pumps and heat exchangers are provided so that any one of the pump heat exchanger units can be removed from service for maintenance or repair without reducing the capability or redundancy of the system. Thus, the third pump can take the place of either No. 1 or No. 2 pump in all respect. The electrical scheme for the third pump is described in Subsection 8.3.1.1.3.

During normal station operation one pump is operating and one pump is in standby (in the redundant loop). The third pump is electrically disconnected from the system. Failure of the operating pump initiates an automatic transfer to the standby pump in the redundant loop. Manual valve and electrical alignment is initiated to place the third pump in a standby status in place of the affected pump.

Under DBA conditions, one CCW pump runs in each loop and nonessential components are isolated from the system. No single failure in a loop affects the other loop.

To support makeup pump operation during feed-and-bleed (non-DBA condition) as described in Section 9.3.4.2, a cross tie to the essential CCW header is provided. During normal operation, cooling to the makeup pumps is supplied via the nonessential header which may be isolated during conditions requiring feed-and-bleed operations.

Both CCW loops may be in operation while cooling the primary system below 280°F and maintaining primary temperature below 140°F for refueling operations. The length of time two pumps are required is dependent on decay heat load, the auxiliary load, and service water temperature. Refer to Section 9.3.5.2.2 for additional information.

The makeup water for the system is supplied from the demineralized water storage tank. Seismic Category I makeup is provided from the service water system.

9.2.2.2.2 Codes and Standards

The equipment in this system is designed to the applicable codes and standards tabulated in Table 9.0-1.

9.2.2.2.3 System Isolation

The component cooling water lines which penetrate the containment have remotely operated isolation valves inside and outside the containment. Note that check valves on each of the three component cooling water lines penetrating containment were installed for the purpose of preventing a post LOCA overpressure condition of the containment penetration piping segments in response to NRC Generic Letter 96-06.

Connections with other systems are provided for makeup, N_2 blanketing, venting, and draining.

All equipment vent and drain lines are equipped with normally closed, manually operated valves.

In the event that a loss of coolant accident results in DBA, the lines supplying the non-engineered safety feature systems will be isolated automatically at SFAS Levels 3 and 4.

The decay heat cooler, the letdown cooler, and the RC pump bearing internal heat exchanger are the only components having a single barrier between the component cooling water system and the reactor coolant system.

The design pressure and temperature of the barriers confining the reactor coolant exceed operating ranges of the reactor coolant in these components and are as follows:

Decay heat cooler		
	Pressure, psig	450
	Temperature, °F	350
Letdown cooler		
	Pressure, psig	2500
	Temperature, °F	600
Reactor coolant pump bearing internal heat exchanger		
	Pressure, psig	2500
	Temperature, °F	650

The letdown coolers have higher design pressure and temperature than the operating pressure and temperature of the reactor coolant system. Further protection has been afforded the letdown coolers by the installation of a redundant valve on the common inlet to the coolers. This valve, in addition to the valve on the line to each cooler, provides the necessary redundancy for isolation of the letdown coolers in the event of a tube rupture. Redundant high pressure switches on the CCW side of the coolers actuates the closing of the inlet valves to the coolers.

The decay heat system is isolated during normal station operation. The decay heat system design pressure and temperature is based on the operating condition in the decay heat removal mode and, therefore, is considerably lower than the normal reactor coolant system operating condition. The operation of the decay heat system and relationship to the reactor coolant are described in USAR Section 9.3.5.

Depending on the extent of reactor coolant system cooldown, a rupture (failure) of the reactor coolant barrier decay heat cooler tube will have the following consequences:

a. Component cooling water system pipe rupture due to overpressurization:

Component cooling surge tank low level
Component cooling water high radiation
Component cooling pump discharge line low flow

b. Component cooling water system pipe does not rupture:

Component cooling surge tank high level
Component cooling water high radiation
Component cooling water system high temperature

The above conditions will actuate an alarm in the control room.

The redundant component cooling water and decay heat removal loops would then be utilized for decay heat removal. The CCW system piping has the primary rating of 150 psig at 500°F. Therefore, the possibility of pipe rupture is extremely remote.

The reactor coolant pump internal heat exchanger has higher design pressure and temperature than the operating pressure and temperature of the reactor coolant system. However, if the heat exchanger tube ruptured, the RC pump seal water will leak into the lower pressure component cooling water on the shell side of the heat exchanger. Upon high pressure, a pressure switch in the outlet line will close the motor operated valve on the outlet line of the heat exchanger. The check valve on the inlet line, at the same time, will prevent the RC pump seal water from leaking into the component cooling water system, making it impossible for the pump seal water pressure to extend beyond these two isolation valves.

9.2.2.2.4 Leakage Consideration

A small amount of normal leakage is expected from the CCW system. The operator will be alerted to a small increase in leakage rate by an increased frequency of makeup to the surge tank. Leakage detection and isolation for leakage rates in excess of makeup capacity is provided by a series of alarms and automatic valve closures initiated by level switches on the CCW surge tank.

These are, in order of descending level: high level alarm, low level alarm, automatic isolation of nonessential components not required for reactor operation, and automatic isolation of all nonessential components. The automatic isolation circuitry meets seismic and single failure criteria. In addition, no single failure in the isolation circuitry will curtail cooling water to those components required for reactor operation.

The high level alarm would alert the operator to leakage into the CCW system from another system. If the leakage came from a radioactive component, the radiation monitors (refer to Subsection 11.4.2.2.3), located on the suction header of each loop, will alert the operator to this condition. Those components exposed to normal primary system pressure on one side (the reactor coolant pump seal coolers and the letdown coolers) will be automatically isolated from the CCW system in the event of a tube rupture.

Seismic Category I makeup is available from the service water system in the event that the non-Seismic I makeup supplies are not available after a Design Basis Accident.

9.2.2.2.5 Failure Analysis

The single failure analysis presented in Table 9.2-6 was based on the assumption that a loss of coolant accident had occurred. It was then assumed that an additional malfunction or failure occurred either in the process of actuating the emergency system or as a secondary accident. All credible failures were analyzed. In general, the types of failures analyzed should be unlikely because vital components of the component cooling water system are serving normal functions, and a program of periodic testing is incorporated into the station operating procedures to ensure the operational consistency of each vital component.

9.2.2.2.6 Environmental Consideration

All components serving the emergency function operation in this system are missile protected and are designed for Seismic Class I requirements.

9.2.2.2.7 Prevention of Long-Term Corrosion

The component cooling water is a demineralized water containing corrosion inhibitor. A chemical pot feeder provides chemicals to the system to maintain the concentration of corrosion inhibitor and to control the pH value. Corrosion is kept to a minimum on all components in this system.

9.2.2.3 Tests and Inspection

Preoperational tests and inspection were performed on all components prior to station initial startup.

All components in this system are hydrostatically tested in accordance with the applicable code.

The operated components are all accessible for visual inspection for leaks from pump seals, valve packing, and flange joints.

The leakage tests were conducted on all containment penetration isolation valves prior to station initial startup.

Electrical components, such as switchgear and starting controls, are tested periodically.

9.2.2.4 Instrumentation

The operation of the component cooling water system is monitored by the following instrumentation:

a. Temperature indicators on the inlet and outlet lines of the component cooling heat exchanger and high temperature alarms on the outlet lines.

b. Pressure indicators on the inlet and outlet line of the component cooling pumps and heat exchangers.

c. Level indicators on each compartment of the component cooling surge tank and high/low level alarm.

d. Flow indicator and low flow alarm on the outlet line of the component cooling pumps.

e. Radiation monitor and alarm on the suction header of each loop.

f. Temperature and pressure indicators on the inlet line of cooling water loops, and temperature indicator on return line.

g. Temperature and flow indicators on the outlet lines of selected components being cooled.

9.2.3 Makeup Water Treatment System

9.2.3.1 Design Bases

The makeup water treatment system is designed to supply high-quality water in sufficient quantity for primary and secondary plant makeup.

Applicable design codes and standards are listed in Table 9.0-1.

9.2.3.2 System Description

The functional drawing for the Demineralized Water System is shown in Figure 9.2-4A.

Under normal operation, Lake Erie water which may be chlorinated at the Intake Structure is delivered to one of two Chlorine Detention Tanks. Sodium Hypochlorite may also be injected into the tanks. From the Chlorine Detention Tank the water is sent to a vendor supplied processing system. The vendor's system provides all necessary equipment and components to produce demineralized water for makeup to the demineralized water storage tank.

The Clearwell and Clearwell Transfer pumps, originally part of the Makeup Water System, are no longer used to produce primary and secondary plant makeup water. The Clearwell is fed from the Carroll Township Water System. The Clearwell is used as a source of makeup water for the Fire Water Storage Tank.

One of three pumps is then used, as required, to transfer the demineralized water in the storage tank to various points throughout the station, such as the condenser hotwell, condensate storage tanks, and for miscellaneous flushing operations.

Regeneration of the demineralizers is accomplished through the use of sulfuric acid and sodium hydroxide. Regenerant wastes from the demineralizers are transferred to the backwash sump where they are diluted before being discharged to the collection box via the settling basin.

Demineralizer system effluent does not exceed the following limits:

Conductivity	0.3 μmho/cm max.
Soluble silica	20 ppb as SiO_2

Note: other system effluent limits are controlled by procedure.

9.2.3.3 Safety Evaluation

The makeup water treatment system is not interconnected with any safety features system and is not essential for safe shutdown of the station.

9.2.3.4 Tests and Inspection

All pressure vessels and piping were hydrostatically tested in accordance with the ASME Code, Section VIII and ANSI B31.1.0, respectively. After the equipment was put into service, a performance test was run to ascertain that the system and equipment were performing satisfactorily.

9.2.3.5 Instrumentation

Operating instrumentation provided to monitor performance of this system includes the following:

a. Level indication and alarms on all tanks

b. Conductivity indication on the effluent lines of all demineralizers

c. pH indication on the effluent of the neutralizing tank

d. Flow indication for proportionate feed of chemicals

e. Discharge pressure indication of all pumps

9.2.4 Potable and Sanitary Water Systems

9.2.4.1 Design Bases

The domestic water system is designed to furnish potable, sanitary, and area wash water throughout the station.

Applicable design codes and standards are listed in Table 9.0-1.

9.2.4.2 System Description

The functional drawing for the Domestic Water System is shown in Figure 9.2-5.

The source of water for the Domestic Water System is the off-site Carroll Township Water System. This water is taken from Lake Erie west of the Davis-Besse site, filtered and treated to meet the requirements of the Ohio EPA. The Carroll Township system pressure is maintained by the use of an elevated 500,000 gallon storage tank with a maximum water level of 742.5 feet ILG which provides sufficient pressure to supply all station needs.

Radioactive contamination or chemical/biological contamination of Domestic Water is prevented by the actions listed below.

1. There are no direct connections to any contaminated systems.

2. Backflow prevention devices are used when connecting to industrial uses of the system.

3. Use of hose connections to potentially radioactive contaminated sources is controlled by station procedures.

9.2.4.3 Safety Evaluation

The domestic water system is not interconnected with any safety features system and is not essential for safe shutdown of the station.

9.2.4.4 Tests and Inspection

Inspection and testing conformed to the requirements of the prevailing revision of the Ohio Building Code to ensure compliance with Chapter BB-51, plumbing. All pressure vessels were hydrostatically tested in accordance with the ASME Code, Section VIII.

9.2.4.5 Instrumentation

Operating instrumentation provided to monitor performance of this system includes the following:

a. Domestic Water Header Pressure

b. Temperature indication of hot water system

9.2.5 Ultimate Heat Sink

The ultimate heat sink for this station is Lake Erie, which is the source of cooling water for the service water system. This single water source is utilized for both normal and emergency shutdown conditions. Lake water is conducted through the intake water system to the intake structure, where the service water pumps are located.

An open forebay area ahead of the intake structure serves as a reservoir for an ensured source of water in case of an extreme lowering of the lake due to meteorological conditions or collapse of the intake canal or submerged pipes. The effects of high and low lake levels and maximum probable wave action are discussed in Subsection 2.4.5 and Section 3.4.

The system described herein complies with Safety Guide 27. A further discussion of compliance is given in Section 2.27 of Appendix 3D.

9.2.5.1 Loss of Intake Canal

The most severe natural phenomenon, which will cause partial loss of the ultimate heat sink, is a loss of the intake canal due to an earthquake. Since the intake canal is categorized as Seismic Class II beyond approximately 700 feet from the intake structure, it has been postulated that the intake from the lake collapses as well as an incredible collapse of the sides of the Seismic Class II portion of the intake canal. All water flow from the lake to the intake canal was assumed stopped. The seismic class II intake canal collapse was assumed to leave one-third of the water surface area and one-third of the water volume from the seismic class II portion of the intake canal.

In the unlikely event of loss of the intake canal, the reactor will be tripped, and station will be maintained at hot standby with the auxiliary feedwater pumps for as long as condensate storage and other demineralized water storage is available. Assuming the minimum condensate storage tank volume allowed by Technical Specifications being available for the auxiliary feedwater pumps, the station can be kept at hot standby for 13 hours (additional condensate storage tank water would be available, but the transfer to the ultimate heat sink after 13 hours is conservative for this analysis). In this situation. Auxiliary Feedwater pump suction will be transferred to the Service Water system a minimum of 13 hours after loss of the intake canal.

The water stored in the intake canal forebay below elevation 562 feet will provide sufficient cooling surface to continue cooling the station by evaporation for at least 30 days. However, it

is estimated that 14 days should provide sufficient time to reestablish direct water flow communication between the lake and the station intake structure via the intake water system.

The design incorporates a Seismic Class I return line from the service water system to the intake canal Seismic Class I area forebay.

In order to support License Amendment 242 to increase the service water temperature from 85°F to 90°F, additional evaluations of the ultimate heat sink have been performed (Reference 6), which considered a LOCA with an intake water system loss of connection of Lake Erie. These analyses form the design basis for the ultimate heat sink temperature of 90°F at the time of accident initiation. The transient ultimate heat sink temperature following the accident, Figure 9.2-6, is used to determine the containment pressure, temperature response and equipment qualifications in containment. Consequently, old analyses have been removed from the USAR.

The ultimate heat sink transient temperature analysis has been performed using the VPLUG computer program, which is a Bechtel Corporation standard computer program. VPLUG program is a one dimensional multilayer computer program that simulates the transient temperature response of the cooling ponds based on plant operating conditions and assumed meteorological conditions. The major assumptions used in the computer program are:

1. All mixing in the pond, including interfacial mixing between layers, can be simulated by a single constant entrainment ratio.

2. All heat exchange with the environment is through the free water surface of the pond.

3. Longitudinal mixing is negligible.

The ultimate heat sink computer model was developed and calibrated using onsite meteorological data and observed ultimate heat sink temperature data from the summer of 1995. Then the model was tested using the onsite meteorological data from 1988. These two years were selected because the ultimate heat sink temperature approached 85°F during these years. The results of the analysis showed a good correlation between the predicted and observed ultimate heat sink temperatures. The same ultimate heat sink model is used to calculate the post-LOCA transient temperature of the pond using the same conservative meteorological data that was used at the time of original plant licensing.

The time dependent heat input to the ultimate heat sink is conservatively estimated using the energy removal rate from a design basis hot leg break, assuming a constant service water temperature of 85°F. The use a constant 85°F service water temperature maximizes the heat removal from the containment and thus the heat input to the ultimate heat sink. A service water temperature of 85°F is only used to determine the maximum heat load to the UHS. The heat removal rate and the integrated heat removed are given in Tables 9.2-7 and 9.2-8 respectively. The other assumptions used in the analysis are as follows.

Maximum Temperature:

a. Meteorological Data

Maximum Clear Sky Radiation H_{clear}	750 LA/day
Percentage of possible sunshine, S	0.6
Maximum Net Sky Radiation $H_{net} = H_{clear} * (1\text{-}65\ (1.S)\text{^}2$	672 LA/day 2478 Btu/Ft^2/Day
Ambient air temperature	81.6 °F
Dew point	73.7 °F
Wet bulb temperature	75.7 °F
Wind speed	7.4 mph

b. Plant Design Data

Elevation	562 ft. IGLD
Surface area	5.166 acres
Water volume, gallons	12,358,686 gallons
Total decay heat (for 30-day period), Btu	31 x 10^9 BTU
Minimum time after reactor shutdown the service water pumps are required (hr)	0 Hours
Service water temperature at the start of transient	90 °F
Maximum UHS temperature	107.6 °F
Time of maximum temperature	2.3 days
Final Intake canal elevation	559.7 ft

Additional analyses have been performed to determine the maximum evaporation rate from the ultimate heat sink assuming a dewpoint temperature of 64°F and a wind speed of 8.8 mph (Reference 15). These evaluations show that the ultimate heat sink will contain an adequate amount of water to cool the plant for more than 30 days.

The determination of the minimum forebay water elevation required by the Service Water pumps is documented in Reference 10. The analysis was performed with a bounding pump flowrate and fluid temperature. Also, the minimum forebay elevation at the end of the 30-day post-accident analysis period (i.e., maximum evaporation case) was determined by Reference 15. The results show that the forebay elevation at the end of the 30-day post-accident analysis

period is greater than the minimum water elevation required by the Service Water pumps. Therefore, pump NPSH and submergence are acceptable at the end of the 30-day post-accident analysis period.

Basis for Meteorological Conditions:

No weather record was examined; the design criteria were obtained from References 1, 3, and 5. Temperature data in Reference 1 was based on a 30-year period of record from 1931 to 1960. Seventeen stations around Lake Erie were used. Humidity and wind data were as listed in Reference 3 and were based on a 15-year period of record.

The design ambient air, dew point, and wet bulb temperatures and wind speed were determined as follows:

a. Ambient Air Temperature

The mean dry bulb temperature (Td) at Davis-Besse in July (the hottest month) is about 75°F (Ref. 1). The standard deviation(s) of Td is about 2.2°F (Ref. 1). Since temperature is a normally distributed parameter, 99.73 percent of the cases are included between

Td - 3s and Td + 3s (Ref. 2)

so a very conservative estimate for the highest daily average

Td is Td + 3s or 75 + 6.6 = 81.6°F

This is the extreme mean July temperature. It is exceeded only 0.27 percent of the time, or about once in 370 years.

b. Dew Point

1. Maximum Evaporation

When the highest possible evaporation rate for determining water loss, is of interest, the relative humidity (R.H.) for calculating evaporative losses should be a very conservative estimate for the 24 hour average R.H. The average relative humidity at Toledo in July is 71 percent. At 1300 local standard time (LST), it is 55 percent R.H. (Reference 3). To maximize evaporation, 55 percent R. H. was assumed to persist for the full 30 days without diurnal variation.

At 81.6°F, Td, and 55 percent R.H. the corresponding TW and TDP at 30" atmospheric pressure are (Reference 4):

Wet bulb temp (TW) = 69.7°F

Dew point temp (TDP) = 64.0°F

2. Maximum Temperature

When the highest forebay temperature is of interest, the relative humidity (R.H.) for calculating evaporative losses should be a very conservative estimate for the 24-hour average. The average relative humidity at Toledo in July is 71 percent. The extreme condition of R.H. is 76 percent, in August rather than July (Reference 5). At 0700 LST in August, it is 89 percent. To minimize heat transfer, 89 percent was assumed to persist for the full 30 days without diurnal variation. The following temperatures were determined at 89 percent humidity and a dry bulb temperature equal to the July mean plus one standard deviation, i.e., 75°F + 2.2°F = 77.2°F

Wet bulb temp (TW) = 75.7°F

Dew point temp (TDP) = 73.7°F

c. Wind Speed

1. Maximum Evaporation

For the highest evaporation rate, the use of the average wind speed in July is conservative. Near the Great Lakes, extreme high temperatures are rarely coincident with high winds. With the hot period hypothesized, it would not be possible to have winds as high as the average. They would have to be less than average. Use of the average wind speed is probably unrealistically conservative, however, the average was used. The average wind speed for July is 8.8 mph (Reference 5).

2. Maximum Temperature

The wind speed used is that for the extreme conditions in August, 7.4 mph (Reference 5). August is used instead of July because it is more conservative. This is the lowest average wind speed for the year, although it would be reasonable to assume even a lower speed. However, there is sufficient conservatism already built into this analysis so that hypothesizing a lower speed is not necessary.

It is not possible to precisely determine the recurrence interval or percent of time the design criteria can be expected to be equaled or exceeded but the lower bound has to be less than 0.27 percent of the time with a recurrence interval of about once in 370 years. It is estimated to be less than 0.10 percent with a recurrence interval of less than once in 1000 years. This far exceeds the intended requirements in Regulatory Guide 1.27, Rev. 1 in which the requirements are normally based on worst conditions observed in only 40 to 100 years of data.

9.2.6 Condensate Storage Facilities

9.2.6.1 Design Bases

Condensate Storage Tanks provide the primary water source for the Auxiliary Feedwater System. The capacity is based on an assumed available inventory sufficient to remove decay

heat for thirteen hours plus a subsequent cooldown to less than 280°F, under normal conditions. Condensate Storage also has the capability to provide makeup to the turbine cycle.

The storage facility is Seismic Class II. No radioactivity concentration is anticipated in this facility; therefore, only normal surveillance of the storage tank and valves for leakage is required.

The storage system is subjected to ambient conditions of 50 to 120°F and 100 percent humidity.

The storage tank design conforms to AWWA D100 and AWS D5-2.

9.2.6.2 Description

The condensate storage system is shown in Figure 10.4-11. Two, 250,000-gallon tanks are provided. The tanks are located within a building adjacent to the turbine building. Normally, both tanks are in use, being interconnected by piping and normally locked opened isolation valves. The tanks provide the suction of the auxiliary feed pumps, motor driven feed pump and startup feedwater pump. The condensate storage system also has the capability to provide makeup to the turbine cycle.

Level is normally maintained by makeup directly from the 140,000 gallon demineralized water storage tank. The capability also exists to provide makeup through the condenser hotwell. Three 200-gpm demineralized water transfer pumps are available for makeup supply to the tanks with interlocks permitting any two of the pumps to be operating. The pumps are available provided a sufficient level is maintained in the demineralized water storage tank.

The two 64,000 gallon (each) Deaerator Storage Tanks typically hold an additional 106,000 gallons of condensate.

9.2.6.3 Safety Evaluation

Failure of the condensate storage facilities will not preclude a safe shutdown of the reactor. Backup water supplies are discussed in Section 9.2.7.3.

9.2.6.4 Inspection

The entire facility is readily available for in-service inspection.

9.2.6.5 Instrument Application

The following instrumentation is provided to monitor this facility:

a. High/low tank level alarms

b. Remote and local tank level indication

9.2.7 Auxiliary Feedwater System

9.2.7.1 Design Bases

The auxiliary feedwater system is designed to provide feedwater to the steam generators when the turbine-driven main feedwater pumps are not available or following a loss of normal and

reserve electric power. All components and piping in the system are designed to Class I requirements, except the condensate storage tank supply sources, and are tornado protected.

Applicable design codes and standards are shown in Table 9.0-1.

9.2.7.2 System Description

The functional drawing for the auxiliary feedwater system is shown in Figure 10.4-12A. On station shutdown, the auxiliary feedwater pumps can be used to remove decay heat until the decay heat removal system can be placed in service. The auxiliary feedwater system consists of two steam turbine-driven feedwater pumps, condensate storage tanks, suction and discharge water piping, steam piping, valves, and associated instrumentation and controls. The pumps take suction from the condensate storage tanks, or from the Seismic Class I service water system. A connection is provided to allow the fire protection system to supply water to the pump suctions. The turbine driver receives steam from the steam generators and exhausts to the atmosphere. The condensate storage capacity is sized so that a total condensate inventory may be available to the pumps sufficient to remove decay heat for approximately thirteen hours plus a subsequent cooldown to less than 280°F under normal conditions (i.e., no loss of offsite power). The steam supply for the turbine drivers is from the main steam headers as shown in Figure 10.3-1 and Figure 10.4-12A. Following a complete loss of normal and reserve power, the auxiliary feedwater system supplies water directly to the steam generators through the auxiliary feedwater nozzles to remove reactor decay heat. Reactor decay heat removal after coastdown of the reactor coolant pumps is provided by the natural circulation characteristics of the reactor coolant system. Use of the auxiliary feedwater system for cooldown is discontinued when the reactor coolant system temperature decreases to about 280°F; further cooldown is accomplished by the decay heat removal system. The Emergency Feedwater System (EFWS) ties into the AFWS to provide a water supply should the AFWS fail during a Beyond-Design-Basis External Event (BDBEE) (see Subsection 9.2.9).

The required pumping capacity to the steam generators is determined by the decay heat removal requirements for a total loss of main feedwater flow transient assuming infinite irradiation at 2,772 MWt which equates to 100.37% of 2817 MWt. Each pump is a horizontal, centrifugal pump with 1050-gpm capacity at 1050-psi head. One pump meets the capacity requirement.

The maximum particulate size that will be present in the Auxiliary Feedwater Pump (AFP) cooling water lines is determined by the maximum size particulate present in the cooling water source upstream of valves AF13, AF14, SW9 and SW10. The water upstream of these valves is the cooling water source for the AFP cooling water lines. There are two sources of cooling water available for the AFP cooling water. The first water source is the common section of the AFP suction piping from the Fire Protection System (FPS) and the Condensate Storage Tanks (CSS). The second water source is the Service Water System connection isolated by normally closed valves SW9 and SW10 respectively. Therefore the particulate size present in the AFP cooling water lines is dependent upon the size of the particulate matter in the AFP suction line or the SW connection to the AFP cooling water lines.

Strainer S257 is located in the common section of the AFP suction piping from the CSS and the FPS. This strainer has a wire mesh size of 0.120 inches and has been installed to protect the AFP from the large entrained particulate matter from the CSS or FPS that could damage the pumps. This strainer limits the size of the particulate in the water source to 0.120 inches.

Filters F15-1, F15-2 and F15-3 located on the Service Water Pump (SWP) discharge lines limit the particulate size contained in the SW cooling water source to 0.0625 inches. However, during Chlorine System outages zebra mussel infestations can occur. Microbiologically induced corrosion and small infestations of zebra mussels have been discovered downstream of the filters during the Chlorine System outages. The zebra mussel shells range in size from very small up to 0.25 inches long.

The corrosion products that result from M1C are rust fragments. These rust fragments can be as large as 0.25 inches in width. Based on the M1C phenomena, it is possible for the SW source of cooling water to contain fragments as large as 0.25 inches.

To prevent plugging of the pressure reduction orifices RO4979 and RO4980 and plugging of a single stage strainer, two strainers of varying size screen mesh were installed in the AFP cooling water lines. Strainers S503 and S504 have been installed upstream of orifices RO4979 and RO4980 and limit the size of particulate in the cooling water to 0.125 inches. These strainers are the first stage of filtering and prevent the passage of particles large enough to plug the 0.131 inch orifice opening. Strainers S203 and S204 are installed downstream o0020f orifices RO4979 and RO4980. These strainers contain baskets with a mesh size of 0.0625 inches. The purpose of these strainers is to limit the size of particulate matter entering the bearing oil coolers.

9.2.7.3 Safety Evaluation

The Auxiliary Feedwater (AFW) system at DB-1 consists of two safety-grade AFW pumps capable of being actuated and controlled by safety-grade signals that ensure the availability of feedwater to at least one steam generator, under the assumed conditions of a single failure. System reliability is achieved by the following features:

a. Two turbine-driven pumps are provided.

b. Steam is supplied by separate steam lines from separate steam generators. The lines are physically separated to meet single-failure criterion.

c. In the event of loss of water supply from the condensate storage tanks, an automatic backup is provided from the service water system.

   The service water system provides the Seismic Class I backup supply. In addition, a backup supply of water from the fire protection system is available to the auxiliary feedwater system via a manual valve.

d. Feedwater to the steam generators is supplied through lines separate from the main feedwater lines and through separate steam generator nozzles. These lines are also physically separated to meet single-failure criterion. Refer to Section 15.2.8 for the required system flowrate (600 gpm).

e. The suction and discharge feedwater lines and main steam lines are designed to Class I. The interface with non-Class I piping is shown on the flow diagram.

f. The turbines are designed to absorb water slugs carried over with wet steam without injurious effects.

g. The turbines are provided with mechanical hydraulic governors.

h. The turbine pump units are physically separated by a pressure-retaining wall and door (see Subsection 3.6.2.7).

i. DC power is supplied to the auxiliary feed pump steam inlet valve MS-106 and pump outlet valve AF 3870 to ensure divers electric power sources to the valves of the auxiliary feedwater system required to actuate for system functioning.

j. The Steam Generator Level Control Valves (AF 6452/6451) fail open on a loss of power to the valves.

k. Two auxiliary feedwater flow indication systems are provided for each steam generator. Both are Class 1E systems. The Class 1E systems consist of a common orifice plate and two flow transmitters in each AFW line to the steam generator. One flow indicator per AFW train is located on the PAMS panel. Another flow indicator for each AFW train is located on the Center Console in the Control Room.

l. The Steam Admission Valves to the AFPTs fail safe by opening on loss of air or loss of DC power to the solenoid.

m. The Steam Admission Valves are located close to the turbine to minimize the amount of cold piping and the potential for water slugs entering the turbines.

n. Steam traps have been installed downstream of the Steam Admission Valves. This ensures that any leakage across the Steam Admission Valves does not accumulate in the turbine casing.

o. If the Auto Steam Generator Level Control System should fail, then the Level Control Valves (AF 6452/6451) can be de-energized (fail open) and Steam Generator Level can be controlled manually from the Control Room (HIS 520A/521A) by varying AFPT speed.

p. The number of valves that require repositioning upon an SFRCS actuation is minimized by having MS 106A, and MS 107A, AF 3870 and AF 3872 in the open position during modes 1-3.

q. The AFW Isolation Valves (AF 608 and AF 599) are no longer actuated by SFRCS. The valves are normally open motor operated valves with control power removed.

r. To increase the reliability of check valves MS-734 and MS-735 a continuous minimum steam flow may be diverted to feedwater heater E6-2. The continuous flow through the valves lifts the discs off the seats, reducing seat wear. The minimum flow line is shown in Figure 10.4-12A.

Steam Generator level is controlled by modulating solenoid control valves. These valves assume the automatic level control function and the AFPT controller maintains a constant speed at its high speed stop (HSS) setting. Automatic level control is accomplished by comparing a S/G level signal with a level setpoint providing an output signal to the valve controller to position the valve. The level control valves and the AFPT speed can be controlled manually from the control room.

A cavitating venturi is also provided in each AFW line to the steam generator downstream of the motor driven feedwater pump (MDFP) discharge tie-in. The venturi is designed to limit the maximum flow rate to a depressurized steam generator to 800 gpm.

A main steam line break accident inside containment concurrent with a single active failure will prevent the isolation of AFW flow to the failed steam generator. Operator action would be required to isolate AFW flow to the affected steam generator. This event is analyzed in Section 6.2.1.3 based on assuming a flow rate of 800 gpm to the affected steam generator.

The AFW System is inter-related with several other plant systems due to its design requirements. It interfaces with the Service Water System for a seismic backup suction supply. This also causes indirect interaction with the Essential Electrical Distribution System. AFW also interfaces with the Main Steam System which provides the motive force for the turbine. Due to these interfaces, several sequencing details are included in the design basis of the AFW system to ensure that the systems are properly coordinated. The sequencing is discussed in detail in the following paragraphs.

In the event of a loss of communication with CST, the volume of protected water within nuclear safety related portion of condensate supply to the AFW Pumps will sustain pump operation until the transfer to an alternate source of water, the Service Water System, is complete. Failures in the pump suction normal supply can occur either due to fouling of the common AFW Suction Strainer, S257, or because the non-nuclear safety related CSTs and the non-nuclear safety related piping from the CSTs to the AFWS are damaged by postulated events prior to or during AFP operation.

Failures in the pump suction supply are initially mitigated by transfer to an alternate source of water, the Service Water System. Low-pressure switches provided on the auxiliary feedwater pump suction line, upon sensing low pressure (5.3 psig for 10 seconds), will automatically open the service water inlet valve. If suction pressure remains low (3.8 psig for 60 seconds) the steam supply valves will close to protect the AFP from significant damage. The low-low suction pressure switches that close the steam supply valves also open contacts in the steam valves' auto-open circuit to prevent an SFRCS "open" signal from causing the valves to cycle, which could cause valve motor operator damage.

The 60 second delay allows for the return of a Service Water pump following a Loss of Offsite Power. The time delay in closing the steam isolation valves is to be longer than the time required to start and load the Emergency Diesel Generator and to restart the Service Water pump. The low-low pump suction pressure trip time delay therefore has to be coordinated with the time delay in the restart of the Service Water pump. The Service Water pump restart time delay is required to control the sequencing of loads being placed on the Emergency Diesel Generator. The steam supply valves to AFPT 1-2 will automatically re-open if the low-low suction pressure condition clears and if an SFRCS "open" signal is present. This automatic reopening is not a design function. This feature is not incorporated in the steam valve control circuits for AFPT 1-1, and manual action will be needed to re-open its steam supply valves if they are closed by a low-low suction pressure switch actuation.

A time delay is provided in the Main Steam to AFW Pump Turbine Line low steam pressure trip for Train 2. This time delay provides protection for the specific scenario of a Steam Line Break on Steam Generator 1 with a single failure of Auxiliary Feedwater Pump 1 and a loss of offsite power (see Section 3.6.2.7.1.5 for additional details regarding the mitigation of Steam Line Breaks). If MS-5889B, No. 2 Turbine Steam Admission valve, opens prior to power being available to open MS-107, Main Steam to AFWP Turbine 2 steam isolation valve, the steam line

will begin to depressurize and the low steam pressure switches' setpoint may be reached. However, when off-site power is lost, the time delay relay loses power and drops out. This prevents the low steam pressure switches, PSL107A-D, from generating steam valve close/open-inhibit signals even if steam pressure fails below their setpoint. When power is restored, MS-107 receives power and begins opening if either an SFRCS or a manual open signal is present, and the steam line will begin to repressurize. Power will also be restored to the time delay relay. If pressure fell below the steam pressure switches’ setpoint, the steam line will repressurize adequately before the time delay expires, resulting in MS107 remaining open. The Train 1 side of AFW does not have a similar time delay because valve MS-106 is DC powered and can respond immediately to an open signal. This prevents the steam line depressurization scenario described above from occurring on Train 1.

9.2.7.4 Tests and Inspections

All active components of the system are accessible for inspection during normal station operation. The auxiliary feedwater pumps will be tested periodically during station operation in accordance with Davis-Besse Technical Specifications.

9.2.7.5 Instrumentation

Operating instrumentation provided to monitor performance of this system includes the following:

a. Turbine speed

b. Pump suction and discharge pressure

c. Pump and turbine high-vibration computer point

d. Pump bearing oil and turbine bearing metal temperatures.

e. Turbine control valve steam pressure

f. Low condensate storage tank level alarm

g. SG startup LVL

h. SG press outlet

i. SG AFW inlet flow

j. Minimum recirculation and test line flow indication

k. Condensate level in AFPT casing

The auxiliary feedwater system instrumentation is discussed in Subsections 7.3 and 7.4.1.3.

9.2.8 Motor Driven Feedwater Pump

9.2.8.1 Design Basis

The Motor Driven Feedwater Pump (MDFP) provides feedwater to the steam generators during normal plant startup and shutdown (see Section 10.4.7.2). The MDFP is also designed to provide a backup supply of feedwater to the steam generator in the event of a total loss of both auxiliary and main feedwater. The MDFP is non-safety related. However, the pump provides a diverse means of supplying auxiliary feedwater to the steam generators and thus functions as a backup to the nuclear safety related auxiliary feedwater system.

9.2.8.2 System Description

The functional drawing for the MDFP is shown in Figure 10.4-12. The pump can be aligned to take suction from the condensate storage tanks, deaerator storage tanks, or the service water system. The pump discharge can be aligned to either the auxiliary feedwater system or the main feedwater system. During plant operation when reactor power is greater than 40%, the MDFP is aligned as a backup auxiliary feedwater pump capable of delivering water to both steam generators.

The MDFP is a horizontal, eight stage centrifugal pump with approximately an 800 gpm capacity at 1050-psi head.

9.2.8.3 Safety Evaluation

The Motor Driven Feedwater Pump is non-safety related. It provides a diverse means of supplying auxiliary feedwater to the steam generators thus improving the reliability and availability of the auxiliary feedwater system.

Increased auxiliary feedwater system reliability and availability is achieved by the addition of the MDFP through the following features:

a. The pump is motor driven to provide a power source diverse from the steam turbine driven main and auxiliary feedwater pumps.

b. The pump can take suction from the condensate storage tanks and discharge water to either steam generator.

c. The pump is sized to provide approximately the same capacity as one auxiliary feedwater pump.

d. The pump and auxiliary components are capable of being supplied by either emergency diesel generator in the event of a loss of offsite power.

e. The capability exists to start the pump and control auxiliary feedwater flow to the steam generators from the Control Room.

In the event of a line break in the steam supply piping of one auxiliary feedwater pump turbine and a single failure in the redundant auxiliary feedwater train, the motor driven feedwater pump is capable of providing auxiliary feedwater to the steam generators.

9.2.8.4 Tests and Inspections

All active components of the system are accessible for inspection during normal station operation. The MDFP will be tested periodically during station operation in accordance with Davis-Besse Technical Specifications.

9.2.8.5 Instrumentation

Operating instrumentation provided to monitor performance of the MDFP includes the following:

a. Pump suction and discharge pressure

b. Pump flow

c. Pump motor current draw

9.2.9 Emergency Feedwater System

9.2.9.1 Design Basis

The Emergency Feedwater (EFW) system is designed to support the site's Diverse and Flexible Coping Strategies (FLEX) for a Beyond Design Basis External Event (BDBEE). The EFW System does not perform any design basis functions. The EFW System provides feedwater to the steam generators should the Auxiliary Feedwater (AFW) system become unavailable. All components and piping in the system are designed to Seismic Class I requirements and are tornado protected. Applicable design codes and standards are shown in Table 9.0-1.

The Nuclear Energy Institute (NEI) established the requirements to maintain the capability for core cooling for beyond design basis events in the NRC-endorsed NEI 12-06, Diverse and Flexible Coping Strategies (FLEX) Implementation Guide (Ref. 16). The company evaluated, per the guidance of INPO Event Report (IER) 11-4, the ability to cope with a beyond design basis event consistent with the NEI 12-06 requirements. The assessment considered the consequences of seismic, wind, or flooding events that result in the loss of non-safety related equipment concurrent with a total site loss of all AC power. Utilizing only installed and portable equipment protected from seismic, wind, and flooding events consistent with the guidance of NEI 12-06, the company concluded a robustly protected auxiliary feedwater source is required for DBNPS. The EFW Facility (EFWF) and EFW System provides the auxiliary feedwater source and the motive force to provide water to the steam generators and to support the FLEX strategies.

9.2.9.2 System Description

The functional drawing for the emergency feedwater system is shown in Figure 9.2-7. The primary function of the emergency feedwater system is to serve as a backup source of feedwater to the steam generators following a BDBEE. In addition, the system is designed to supply non-borated makeup water to the spent fuel pool or to the reactor coolant system. The emergency feedwater system consists of one diesel engine-driven pump, emergency water storage tank, suction and discharge water piping, fuel oil piping, valves, and associated instrumentation and controls. The pump takes suction from the emergency water storage tank. A connection is provided to the storage tank to allow refill from the demineralized water system, the fire protection system, or other available water supply. The diesel engine receives fuel oil

from the 6,000-gallon storage tank located on 603’-0” elevation of the Emergency Feedwater Facility.

The 290,000-gallon emergency water storage tank is sized so that a sufficient inventory of water is available to remove decay heat, cooldown the primary system to 280°F, and makeup any spent fuel pool boiloff for more than twenty-three hours (Ref. 17 – C-ME-050.05-001).

The system is equipped to operate with or without plant supplied power. Following manual initiation, the emergency feedwater system supplies water directly to steam generator 1-1 through the auxiliary feedwater nozzles to remove reactor decay heat. Feedwater supply is available to steam generator 1-2 following Operator action to reposition manual isolation valves. As with the auxiliary feedwater system, it is anticipated that the use of the emergency feedwater system for cooldown will be discontinued when the reactor coolant system temperature decreases to about 280°F; further cooldown is accomplished by the decay heat removal system.

The flow to the steam generators is manually controlled to maintain the desired steam generator level and cooldown rate. The emergency feedwater pump is a horizontal, eleven (11) stage centrifugal pump with a minimum of 600-gpm capacity to the system with the steam generator(s) at 1050-psi head.

Demineralized water is the preferred source, while the system is designed to be refilled from other sources as they are available (e.g., Fire Protection System (FPS), Service Water System (SWS), etc.).

9.2.9.3 Safety Evaluation

The Emergency Feedwater (EFW) system consists of one non-safety, seismically qualified pump capable of being actuated and controlled either locally or from the Control Room. System reliability is achieved by the following features:

a. One diesel engine-driven pump, seismically qualified, with its own dedicated battery start system.

b. Fuel oil to the diesel is from a 6,000-gallon storage tank located inside of the robustly designed EFWF.

c. The source of feedwater is from a seismically qualified, missile-protected, 290,000 gallon tank which is integral to the EFWF.

d. Feedwater to the steam generators is supplied through the existing AFW lines. The minimum required flow rate of 600 gpm is the same as for the AFWS as noted in Section 15.2.8.

e. The suction and discharge feedwater lines are designed to seismic Class I, and contained in either the protected EFWF, Auxiliary Building, or routed underground between the two buildings.

f. DC power is supplied to a solenoid operated flow control valve from the normal EFWF power source, backup emergency generator, or batteries. The valve is normally open, fails open, with the throttle position manually controlled.

g. The emergency feedwater flow indication to each steam generator is available locally. Indication of total flow to both steam generators is available locally and in the Control Room.

h. There are no valves that require repositioning upon manual initiation for automatic delivery of feedwater to steam generator 1-1.

i. A cavitating venturi is not required for the EFW system since it is manually initiated and controlled.

Refill of the emergency water storage tank is a manual operation, making use of any available source.

9.2.9.4 Tests and Inspections

All active components of the system are accessible for inspection during normal station operation. The emergency feedwater pumps will be tested periodically during station operation.

9.2.9.5 Instrumentation

Operating instrumentation provided to monitor performance of this system in the EFWF includes the following:

a. Local and remote diesel engine operating indication

b. Local pump suction and discharge pressure

c. Local emergency feedwater storage tank level

d. Local emergency feedwater storage tank temperature

e. Local and remote system flow indication

f. Local flow indication to steam generators 1-1 and 1-2

g. Local fuel oil storage tank level

h. Local flow indicator for EFW minimum flow line

i. High emergency feedwater facility sump level

TABLE 9.2-1

Service Water System Design Parameters for Major Equipment

Inlet water temperature, °F	90[4]

Service Water Pumps

Number	3 (100% capacity)
Number normally operating	2
Type	Vertical turbine
Rated capacity, gpm	10,250
Rated head, ft	160
Motor horsepower	600
Number of stages	2

Emergency Core Cooling System Room Cooling Coils

Number of units	4
Number normally operating	as required
Number for emergency operation	2 in the same room
Flow per unit, gpm (emergency)	140
Max. head loss, ft	11

Containment Air Coolers

Number	3
Number normally operating	2
Head loss/gpm each normal operation	2.0 ft/540 (total 1080 gpm)
Number required for emergency operation	1
Head loss/gpm for emergency operation	11.6 ft/1600 (total 1600 gpm)[2]
Max Service Water outlet temperature, clean unit emergency operation	221.3°F[5]

Component Cooling Heat Exchangers

Number	3
Number normally operating	1
Head loss	10.4 ft @ 8000 gpm
Number required for emergency operation	1
Flow per unit, gpm	7500

(1) Deleted

(2) Refer to Subsection 6.2.1.3.2 for analysis using reduced service water flow for emergency operation.

(3) Deleted

(4) In support of License Amendment 242 a variable inlet water temperature up to 90°F was evaluated and determined to be acceptable.

(5) The maximum outlet temperature was determined assuming a CAC fan flowrate of 58,000 cfm. This is conservative. The design basis slow speed CAC fan flowrate has been revised to 45,000 cfm.

TABLE 9.2-2

Average Composition of Existing Lake Erie Water
at the Davis-Besse Station*

Calcium (Ca)	45	ppm
Magnesium (Mg)	11	ppm
Sodium (Na)	12	ppm
Chloride (Cl)	22	ppm
Nitrate (NO_3)	12	ppm
Sulfate (SO_4)	37	ppm
Phosphate (PO_4)	1.5	ppm
Silica (SiO_2)	2	ppm
P Alkalinity as $CaCO_3$	6	ppm
M.O. Alkalinity as $CaCO_3$	101	ppm
Total Hardness as $CaCO_3$	154	ppm
Free Mineral Acidity as $CaCO_3$	76	ppm
pH	8.1	
Suspended Solids	131	ppm
Dissolved Solids	225	ppm
Dissolved Oxygen	10	ppm

* Based on samples from November, 1968 to October, 1970 and analyzed by the Toledo Edison Company.

TABLE 9.2-3

Single Failure Analysis - Service Water System

	Component	Failure	Results
1	Offsite Power	Not available	Emergency diesels start and supply electrical load to system.
2	Emergency diesels	One not available	The operative diesel supplies necessary power to one of the redundant system flow paths.
3	Service water pumps	One not available	The redundant Service Water train supplies adequate cooling to its components.
4	Service water piping	Rupture	Passive failures of Service Water System piping are not postulated during or following design basis accidents. The following discussion pertains to the postulation of line break events during normal operation. As discussed in Table 3.6-1, pipe breaks are not postulated in seismic category I systems with fluid pressures less than 275 psig and fluid temperatures less than 200°F. Since the essential portions of the Service Water system are seismic category I, and the system pressure and temperature are less than 275 psig and 200°F, rupture of the essential Service Water piping is not postulated. Rupture of the non-essential, seismic class II SW piping is postulated. The analysis is presented in USAR Section 3.6.2.7.2.16.
5	Essential component SW isolation valve (e.g., CAC, CCW, or Aux Feedwater Pump suction)	Fails to open or spuriously closes	The redundant component supplied by the other Service Water train provides the required function.
6	Turbine Plant Cooling Water Heat Exchanger isolation valve	Valve SW1395 or SW1399 fails to close	Service Water flow to components in the affected loop would be reduced. The redundant Service Water train provides adequate cooling to its components.

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TABLE 9.2-3 (Continued)

Single Failure Analysis - Service Water System

	Component	Failure	Results
7	CREVS cooler outlet Valve	Fails to open or spuriously closes	The redundant CREVS cooler will provide adequate cooling to the control room. In addition, a safety-related, air-cooled system is available as a backup to the service water system.
8	SW Strainer Backwash valve	Fails to open	This failure could result in the loss of one service water pump. The redundant Service Water train supplies adequate flow to its components.
9	Valve SW2945, Service Water strainer blow down valve to intake structure	Fails closed or fails to remain open	This failure is not credible since the air supply to the valve operator has been isolated and the valve is permanently in its failed open position. Failure of this valve to remain open is not postulated. (See Note 1)
10	Service water return path isolation valves (i.e., SW 2929, SW 2931, and SW 2932)	Fails to close via MOV (due to either automatic or manual actuation of MOV)	Operator repositions the valve manually. Postulated failure of a valve to change position when manually operated was not required as part of the original licensing basis. (See Note 1)
11	SW Discharge to Intake Forebay isolation valve, SW 2930	Fails to open	SW 2929, SW Discharge to Intake Structure (de-icing line) isolation valve opens automatically due to high pressure in the return header. Operator subsequently manually opens valve SW2930 and closes SW2929. Postulated failure of a valve to change position when manually operated was not required as part of the original licensing basis. (See Note 1)

TABLE 9.2-3 (Continued)

Single Failure Analysis - Service Water System

	Component	Failure	Results
12	Manual valve with function common to multiple components (e.g., SW82, ECCS room cooler common header outlet valve, or SW2930A, Instrumentation root isolation valve.)	Fails closed	Passive Failure of manual valves (i.e., failure to remain open) is not postulated under Davis-Besse's licensing basis. (See Note 1)
13	Non-Seismic Discharge Pipe Downstream or SW2931 and SW2932	Partially or Completely blocked	SW2930 or SW2929 (winter only) opened by manual action or automatic action. Operator subsequently closes SW2931 or SW2932.

Note 1: Davis-Besse's licensing basis does not include passive valve failures that postulate the disc separating from the stem and dropping into the valve seat, blocking flow (i.e., stem-to-disc separation) or failures that postulate a valve failing to change position when locally, manually operated. This licensing basis is consistent with the guidance provided in regulatory documents that were used during the time of Davis-Besse's license application processing (References 9.6.11 and 9.6.12). These documents conclude that passive valve failures need not be postulated as part of post-accident response.

TABLE 9.2-4

Reactor Auxiliaries and ECCS Systems Supplied by Component Cooling Water

	DBA Conditions	Normal Operation
Decay heat removal coolers	Yes	Yes[(1)]
Emergency diesel generator jacket heat exchangers	Yes	Yes
Decay heat removal pump bearing housing cooling	Yes	Yes
High pressure injection pump bearing oil cooling	Yes	Yes
Containment gas analyzer system heat exchangers	Yes	Yes
Letdown coolers	No	Yes
Seal return coolers	No	Yes
Radwaste and reactor sample coolers	No	Yes
Spent fuel pool heat exchangers	No	Yes
Control rod drive coolers	No	Yes
Pressurizer quench tank cooler	No	Yes
Boric acid evaporator packages	No	Yes
Degasifier package	No	Yes
Waste evaporator package[(4)]	No	Yes
Waste gas compressors	No	Yes
Reactor coolant pumps (motor air, upper bearing, lower bearing, and pump seal)	No	Yes
Makeup pump lube and gear oil coolers	No[(3)]	Yes

Notes: (1) During cool-down only.
(2) Deleted
(3) Required during feed-and-bleed. (Non-DBA condition)
(4) Waste evaporator has been abandoned in place.

TABLE 9.2-5

Component Cooling Water System Design Parameters and Major Equipment Data
(Equipment capacities are for single components)

Component Cooling Pumps	
Number	3
Number normally operating	1
Type	Horizontal, Centrifugal
Rated capacity gpm	7,860
Rated head, ft H2O	150
Motor horsepower	400
Material	CS
Component Cooling Heat Exchangers[(2)]	
Number	3
Number normally operating	1[(1)]
Type	Shell and Tube
Heat transferred, Btu/hr	27.4×10^6
Shell side (component cooling water)	
Normal inlet/outlet temperature, °F	102/95
Flow rate, gpm	7,860
Tube side (service water)	
Normal inlet/outlet temperature, °F	90/97.3
Flow rate, gpm	7500
Material, tube/shell	SS/CS

(1) During winter months, SW may be manually bypassed through the spare CCW heat exchangers as described in Section 9.2.1.3.

(2) Values indicated area performance values determined by approved calculations. Original equipment data may be obtained from specification data sheets.

TABLE 9.2-6

Single Failure Analysis - Component Cooling Water System

	Components	Malfunction	Comments and Consequences
1.	Component Cooling Pump	Fails (Stops)	This type of failure is considered unlikely, since the same pump performs normal function. Standby pump will serve same function.
2.	Component Cooling Heat Exchanger	Fails to perform, tube rupture	Standby unit will perform the same function.
3.	Surge Tank	Leaks	The Surge Tank has adequate volume to accommodate minor, operational leakage from the system during normal operation and post-LOCA. The surge tank is divided by a partition plate into two equal compartments. Each compartment serves a separate loop. This design assures that leaks on one loop do not affect the other loop. Level switches are provided to isolate the non-essential portion of the system upon detecting low levels in the surge tank. Additionally, system make-up capability exists to compensate for leakage and to reduce the dependence on the redundant loop. Additional details on system leakage considerations are discussed in Section 9.2.2.2.4. Passive failure of the Surge Tank is not postulated.
4.	Decay Heat Removal Cooler	Fails to perform	Standby unit will perform the same function.
5.	Emergency Diesel Generator Heat Exchanger	Fails to perform	The second emergency diesel generator is available, the component cooling water system will be switched to the standby loop.
6.	Component Cooling Pump Suction or Discharge Valve	Sticks closed	This type of failure is considered unlikely, since the same pump performs normal function. Standby pump will serve same function.
7.	Remotely-operated Isolation Valve on Decay Heat Removal Cooler Outlet Line	Fails to open	Alternate line (loop) will serve the same function.
8.	Deleted		

TABLE 9.2-6 (Continued)

Single Failure Analysis - Component Cooling Water System

	Components	Malfunction	Comments and Consequences
9.	Pipe Line	Rupture	Passive failures of CCW System piping are not postulated during or following design basis accidents. Note: Section 9.2.2.2.5 specifies that this table is based on the assumption that a LOCA has occurred. The following discussion presents information pertaining to the postulation of line break events during normal operation. This information is not related to post-LOCA single failures, but is included for completeness and clarity relative to passive failures that are postulated in the CCW system. As discussed in Table 3.6-1, pipe breaks are not postulated in seismic category I systems with fluid pressures less than 275 psig and fluid temperatures less than 200°F. Since the CCW piping is seismic category I (except in Containment) as stated in Section 3.6.2.7.2.8, and the system pressure and temperature are less than 275 psig and 200°F, respectively, pipe breaks outside Containment are not postulated. Ruptures (i.e., critical cracks) are postulated in non-seismic category I portions of the system. Should a critical crack in the non-seismic category I portion of the system inside Containment, a level alarm on the surge tank and flow indicators on the line will both indicate the abnormal condition. Level switches on the CCW surge tank will isolate the leak by closing isolation valves between the non-seismic and the seismic category I portions of the system. This isolation protection meets single failure criteria.
10.	Remotely-Operated or Self-Actuated Containment Isolation Valve	Fails to close	There are nine isolation valves on component cooling water lines penetrating containment - three on each of the two supply lines to containment and three on the combined return from containment. One valve failing to close does not affect the proper function of the system.

TABLE 9.2-6 (Continued)

Single Failure Analysis - Component Cooling Water System

	Components	Malfunction	Comments and Consequences
11.	Remotely-Operated or (HV-1495) on Auxiliary Equipment Cooling Header	Fails to close	Procedural actions in response to SFAS level 3 or low surge tank level alarms require HV-1495 to be verified closed. Operator closes the manual gate valve upstream of this control valve. Alternate line (loop) will serve the same function.
12.	Other essential components served by CCW (such as Hydrogen Analyzers, Decay Heat Pump lube oil coolers, High Pressure Injection Pump lube oil coolers)	CCW fails to provide adequate cooling	Redundant component will perform the required function.

TABLE 9.2-7

Energy Removal Rate for 14.14 Ft^2 Hot-Leg Break (without ECCS Quenching)

Time (sec)	Containment Air Cooler Energy Removal Rate (Btu/hr)	Decay Heat Cooler Energy Removal Rate (Btu/hr)	Total Energy* Rejected Rate Btu/hr)
100	0.64×10^8	-	0.933×10^7
500	0.64×10^8	-	0.731×10^8
1,000	0.64×10^8	-	0.731×10^8
2,000	0.64×10^8	-	0.731×10^8
3,000	0.64×10^8	-	0.731×10^8
4,500	0.50×10^8	1.01×10^8	1.604×10^8
6,000	0.57×10^8	1.03×10^8	1.692×10^8
8,000	0.61×10^8	1.07×10^8	1.773×10^8
10,000	0.63×10^8	1.10×10^8	1.821×10^8
20,000	0.43×10^8	1.04×10^8	1.57×10^8
50,000	0.22×10^8	0.72×10^8	1.037×10^8
100,000	0.15×10^8	0.54×10^8	0.781×10^8
200,000	0.11×10^8	0.43×10^8	0.631×10^8
400,000	0.79×10^7	0.33×10^8	0.503×10^8
800,000	0.57×10^7	0.25×10^8	0.401×10^8
1,000,000	0.54×10^7	0.24×10^8	0.384×10^8

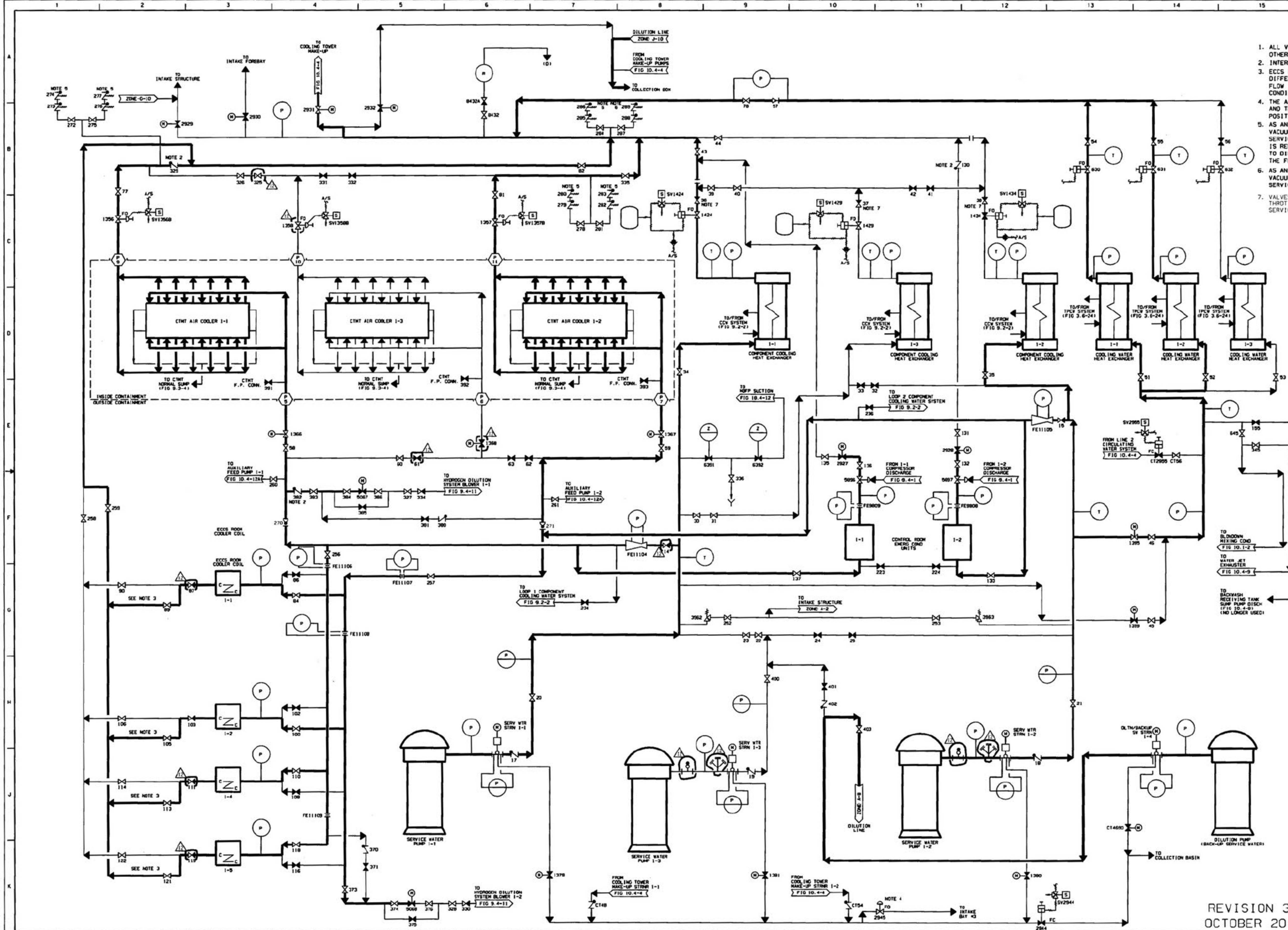
*Includes containment air coolers, decay heat coolers, and auxiliary loads

TABLE 9.2-8

Energy Removed After a 14.14 Ft^2 Hot-Leg Break (without ECCS Quenching)

Time (sec)	Energy Removed by the Containment Air Coolers (Btu)	Energy Removed by the Decay Heat Coolers (Btu)	Total Energy* Removed (Btu)
1,000	0.16×10^8	-	0.19×10^8
2,000	0.34×10^8	-	0.39×10^8
3,000	0.51×10^8	-	0.59×10^8
4,000	0.69×10^8	-	0.79×10^8
4,500	0.78×10^8	-	0.90×10^8
6,000	0.10×10^9	0.43×10^8	1.59×10^8
10,000	0.17×10^9	0.16×10^9	3.55×10^8
20,000	0.31×10^9	0.46×10^9	8.26×10^8
50,000	0.56×10^9	0.12×10^{10}	18.70×10^8
100,000	0.81×10^9	0.20×10^{10}	30.49×10^8
200,000	0.12×10^{10}	0.34×10^{10}	50.61×10^8
500,000	0.19×10^{10}	0.12×10^{10}	95.20×10^8
1,000,000	0.27×10^{10}	0.10×10^{11}	153.34×10^8

*Includes containment air coolers, decay heat coolers, and auxiliary loads



NOTES

1. ALL VALVES ARE PREFIXED WITH "SW" UNLESS OTHERWISE NOTED.
2. INTERNALS REMOVED.
3. ECCS ROOM COOLER DISCHARGE VALVES MAY BE POSITIONED DIFFERENTLY THAN DEPICTED. PROVIDING THAT ADEQUATE FLOW IS ASSURED UNDER DESIGN BASIS ACCIDENT CONDITIONS WITHOUT OPERATOR INTERVENTION.
4. THE AIR SUPPLY HAS BEEN DISCONNECTED FROM SW2945 AND THE VALVE SHALL REMAIN IN ITS FAILED OPEN POSITION FOR SINGLE ACTIVE FAILURE CONCERNS.
5. AS AN ACCEPTABLE TEMPORARY CONDITION. OUTBOARD VACUUM BREAKER VALVES ARE NOT NEEDED TO SUPPORT SERVICE WATER SYSTEM OPERATION. WHEN OUTBOARD VALVE IS REMOVED. A FLANGED PIPE ELBOW SHALL BE INSTALLED TO DIRECT ANY POTENTIAL INBOARD VALVE LEAKAGE TOWARD THE FLOOR.
6. AS AN ACCEPTABLE TEMPORARY CONDITION. OUTBOARD VACUUM BREAKER VALVES ARE NOT NEEDED TO SUPPORT SERVICE WATER SYSTEM OPERATION.
7. VALVES SW36. SW37. AND SW38 ARE REQUIRED TO BE THROTTLED TO PROVIDE RUNOUT PROTECTION FOR THE SERVICE WATER PUMPS.

NO	DATE	REVISIONS	BY	CH'K	ENGR	SA SUPV	MED MGR
9		INC. UCN'S 09-106. AND 10-004	"INITIALS ON FILE"				
8		INC. UCN'S 06-0250. 06-0260. 06-0270 AND 06-0310	"INITIALS ON FILE"				
12		INC. CN'S 14-0170. 14-224. 14-332. & 14-333	SBW				
11		INC. CN's 12-072. 12-174. 12-175. 12-216 & 12-217	"INITIALS ON FILE"				
10		INC. CN 11-156. CN 12-004 & CN 12-165	"INITIALS ON FILE"				
0		INFORMATION ONLY	"INITIALS ON FILE"				

SCALE NONE | DESIGNED FXG | DRAWN PAG | DATE 06-14-90

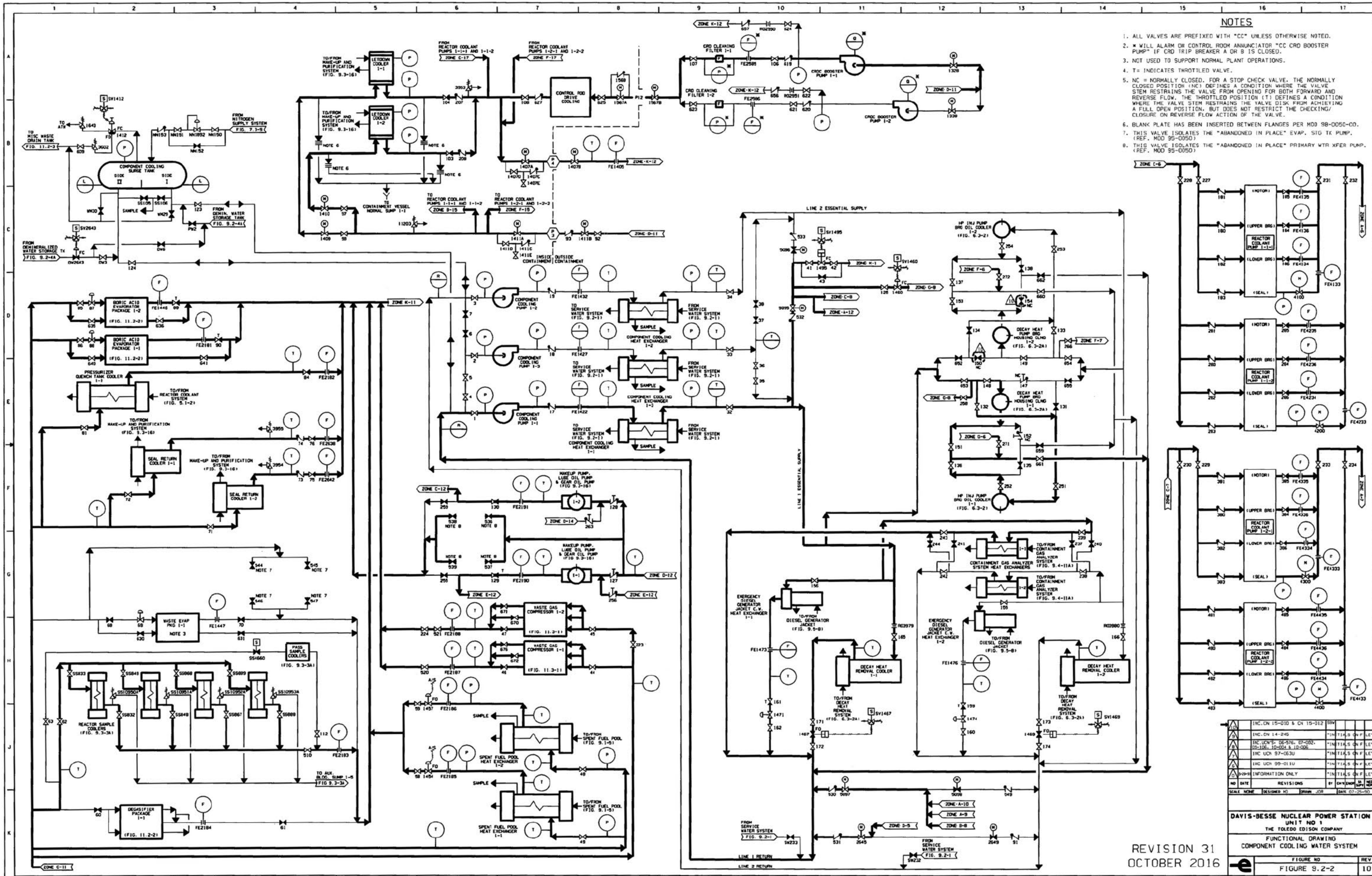
DAVIS-BESSE NUCLEAR POWER STATION
UNIT NO 1
THE TOLEDO EDISON COMPANY

FUNCTIONAL DRAWING
SERVICE WATER SYSTEM

REVISION 31
OCTOBER 2016

FIGURE NO: FIGURE 9.2-1 | REV 12

DFN=G/USAR/UFIG921.DGN
DB 03-17-16



NOTES

1. ALL VALVES ARE PREFIXED WITH "CC" UNLESS OTHERWISE NOTED.
2. * WILL ALARM ON CONTROL ROOM ANNUNCIATOR "CC CRD BOOSTER PUMP" IF CRD TRIP BREAKER A OR B IS CLOSED.
3. NOT USED TO SUPPORT NORMAL PLANT OPERATIONS.
4. T= INDICATES THROTTLED VALVE.
5. NC = NORMALLY CLOSED. FOR A STOP CHECK VALVE, THE NORMALLY CLOSED POSITION (NC) DEFINES A CONDITION WHERE THE VALVE STEM RESTRAINS THE VALVE FROM OPENING FOR BOTH FORWARD AND REVERSE FLOW. THE THROTTLED POSITION (T) DEFINES A CONDITION WHERE THE VALVE STEM RESTRAINS THE VALVE DISK FROM ACHIEVING A FULL OPEN POSITION, BUT DOES NOT RESTRICT THE CHECKING/ CLOSURE ON REVERSE FLOW ACTION OF THE VALVE.
6. BLANK PLATE HAS BEEN INSERTED BETWEEN FLANGES PER MOD 98-0050-00.
7. THIS VALVE ISOLATES THE "ABANDONED IN PLACE" EVAP. STG TK PUMP. (REF. MOD 95-0050)
8. THIS VALVE ISOLATES THE "ABANDONED IN PLACE" PRIMARY WTR XFER PUMP. (REF. MOD 95-0050)

NO	DATE	REVISIONS	BY	CH'K	ENGR	SA SUPV	NED MGR
10		INC. CN 15-010 & CN 15-012	SBW				
9		INC. CN 14-245	"INITIALS ON FILE"				
8		INC. UCN'S: 06-576, 07-002, 09-106, 10-004 & 10-006	"INITIALS ON FILE"				
7		INC UCN 97-063U	"INITIALS ON FILE"				
6		INC UCN 99-011U	"INITIALS ON FILE"				
0	5-29-90	INFORMATION ONLY	"INITIALS ON FILE"				

SCALE NONE | DESIGNED HO | DRAWN JOR | DATE 07-25-90

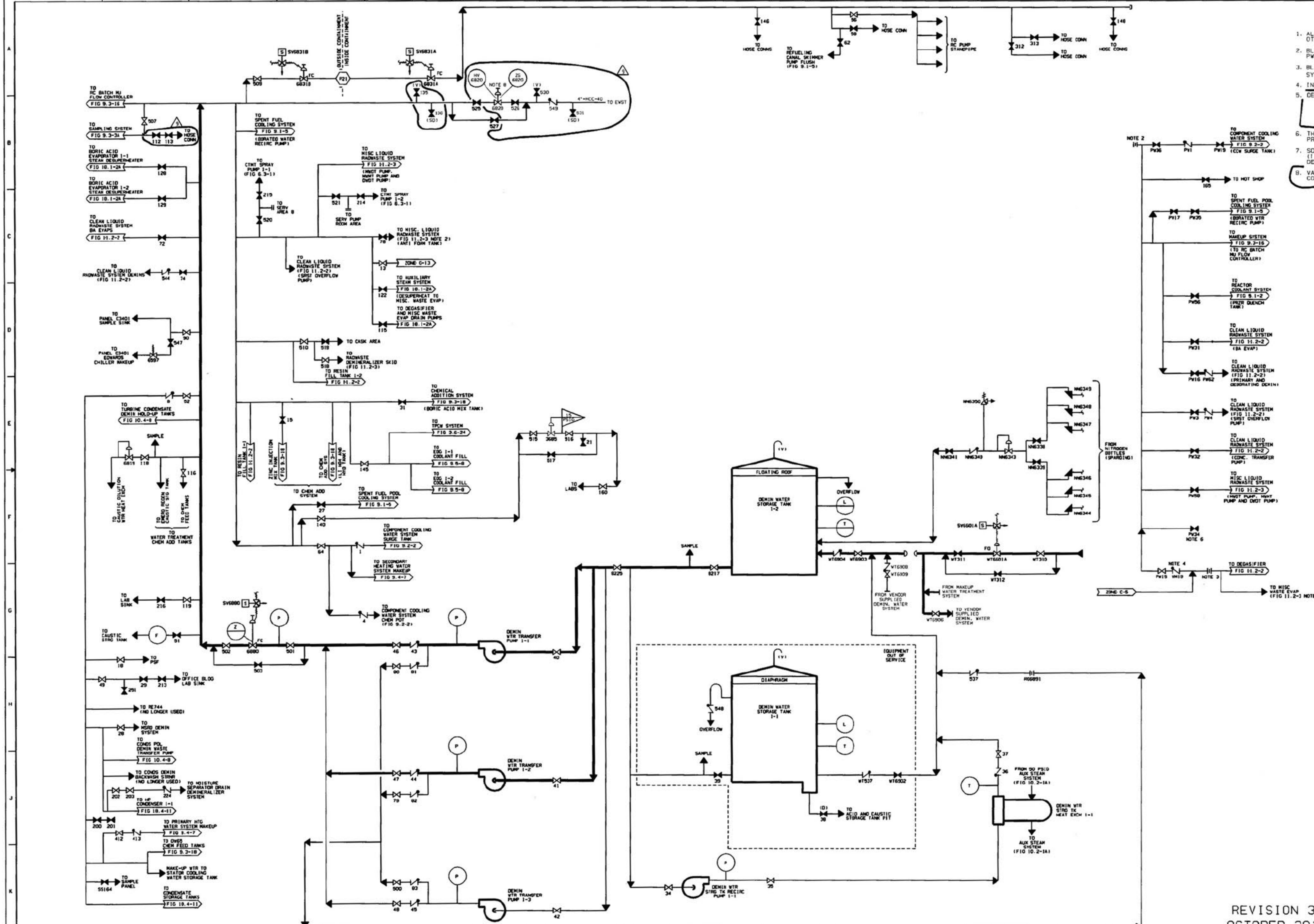
REVISION 31
OCTOBER 2016

DAVIS-BESSE NUCLEAR POWER STATION
UNIT NO 1
THE TOLEDO EDISON COMPANY

FUNCTIONAL DRAWING
COMPONENT COOLING WATER SYSTEM

FIGURE NO	REV
FIGURE 9.2-2	10

DB 04-04-16
DFN=G/USAR/UF10922.DGN



NOTES

1. ALL VALVES ARE PREFIXED WITH "DW" UNLESS OTHERWISE NOTED.
2. BLANK PLATE BETWEEN FLANGES TO ISOLATE PW (ABANDONED) AND DW SYSTEMS.
3. BLANK PLATE BETWEEN FLANGES TO ISOLATE DW SYSTEM FROM MISC. WST. EVAP. AND DEGASSIFIER.
4. INTERNALS REMOVED.
5. DELETED
6. THIS VALVE ISOLATES THE "ABANDONED IN PLACE" PRIMARY WATER SYSTEM. (REF. MOD 95-0050)
7. SOME VALVES, NUMBERED WITH A "PW" DESIGNATION (I.e. PW32), ARE NOW USED TO SUPPLY DEMINERALIZED WATER. (REF. MOD 95-0050)
8. VALVE DW6820 RESTORED AND ASSOCIATED ADV CONTROLS ABANDONED.

REVISION 31
OCTOBER 2016

NO	DATE	REVISIONS	BY	CHK	ENGR	SA SUPV	MED MGR
6		INC. UCN 10-064	"INITIALS ON FILE"				
5		INC. UCN 97-063U, 97-121, 01-040U & 01-044U	"INITIALS ON FILE"				
4		INC. UCN'S 95-113U & 95-131U	"INITIALS ON FILE"				
9		INC. CN'S 13-232, 14-180 & 15-032	SBW				
8		INC. CN'S 11-140, 11-146 & 12-163	"INITIALS ON FILE"				
7		INC. CN 10-145 & CN 11-144	"INITIALS ON FILE"				
0		INFORMATION ONLY	"INITIALS ON FILE"				

SCALE NONE | DESIGNED DHG | DRAWN PAG | DATE 06-18-90

DAVIS-BESSE NUCLEAR POWER STATION
UNIT NO 1
THE TOLEDO EDISON COMPANY

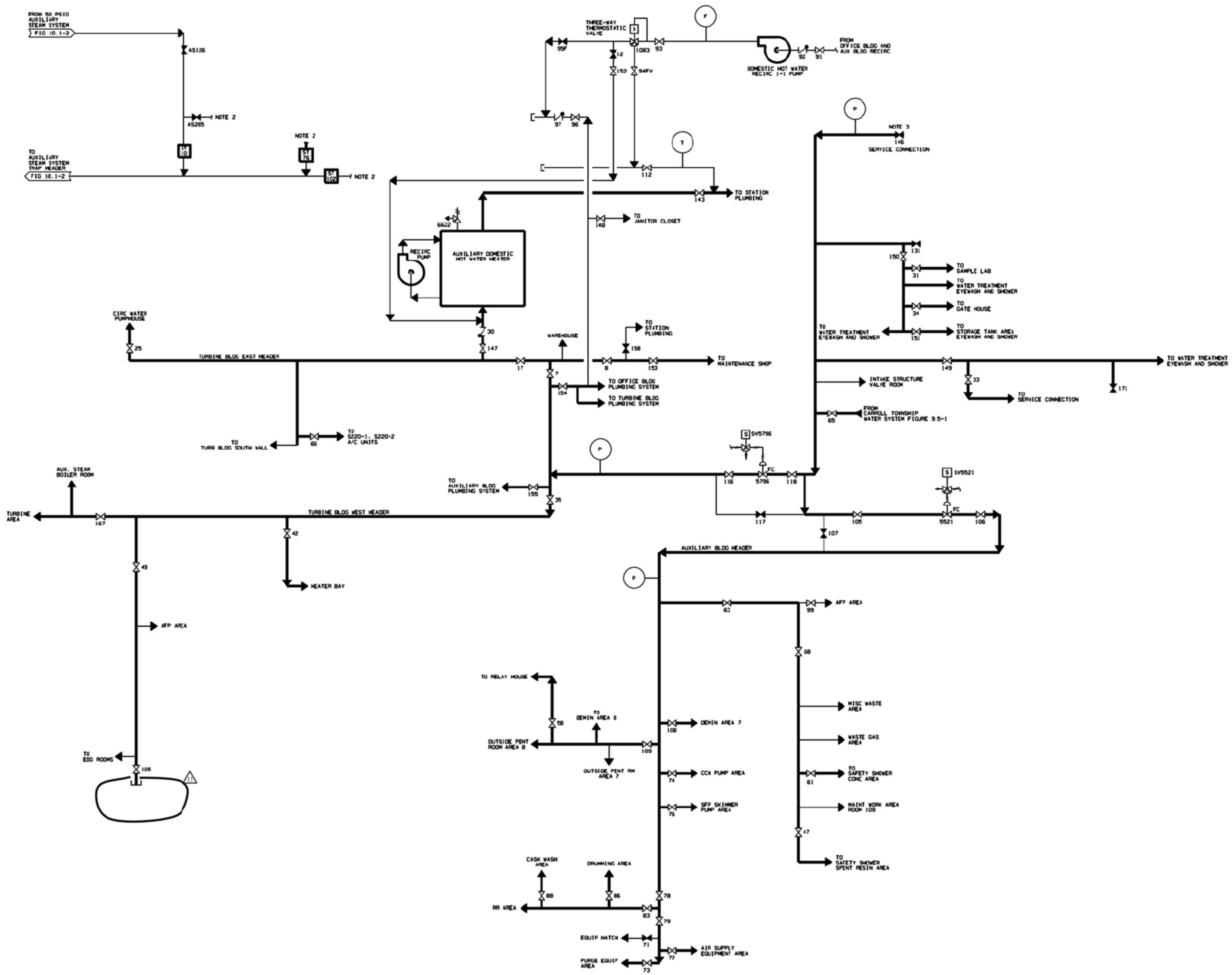
FUNCTIONAL DRAWING
DEMINERALIZED
WATER SYSTEM

FIGURE NO: FIGURE 9.2-4A | REV: 9

DFN-/USAR/UFIG924A.DGN
DB 07-18-16

NOTES

1. ALL VALVES ARE PREFIXED WITH "DW" UNLESS OTHERWISE NOTED.
2. FROM ABANDONED EQUIPMENT.
3. FORMER SUPPLY FROM ABANDONED DOMESTIC WATER PUMPS.



REVISION 33
SEPTEMBER 2020

NO	DATE	REVISIONS	BY	CH'K	ENGR	SA SUPV	MED MGR
8		INC. CN 14-245	"INITIALS ON FILE"				
11		INC. CN 19-064	SAP				
10		INC. CN 17-137	"INITIALS ON FILE"				
9		INC. CN 14-315	"INITIALS ON FILE"				
0		INFORMATION ONLY	"INITIALS ON FILE"				

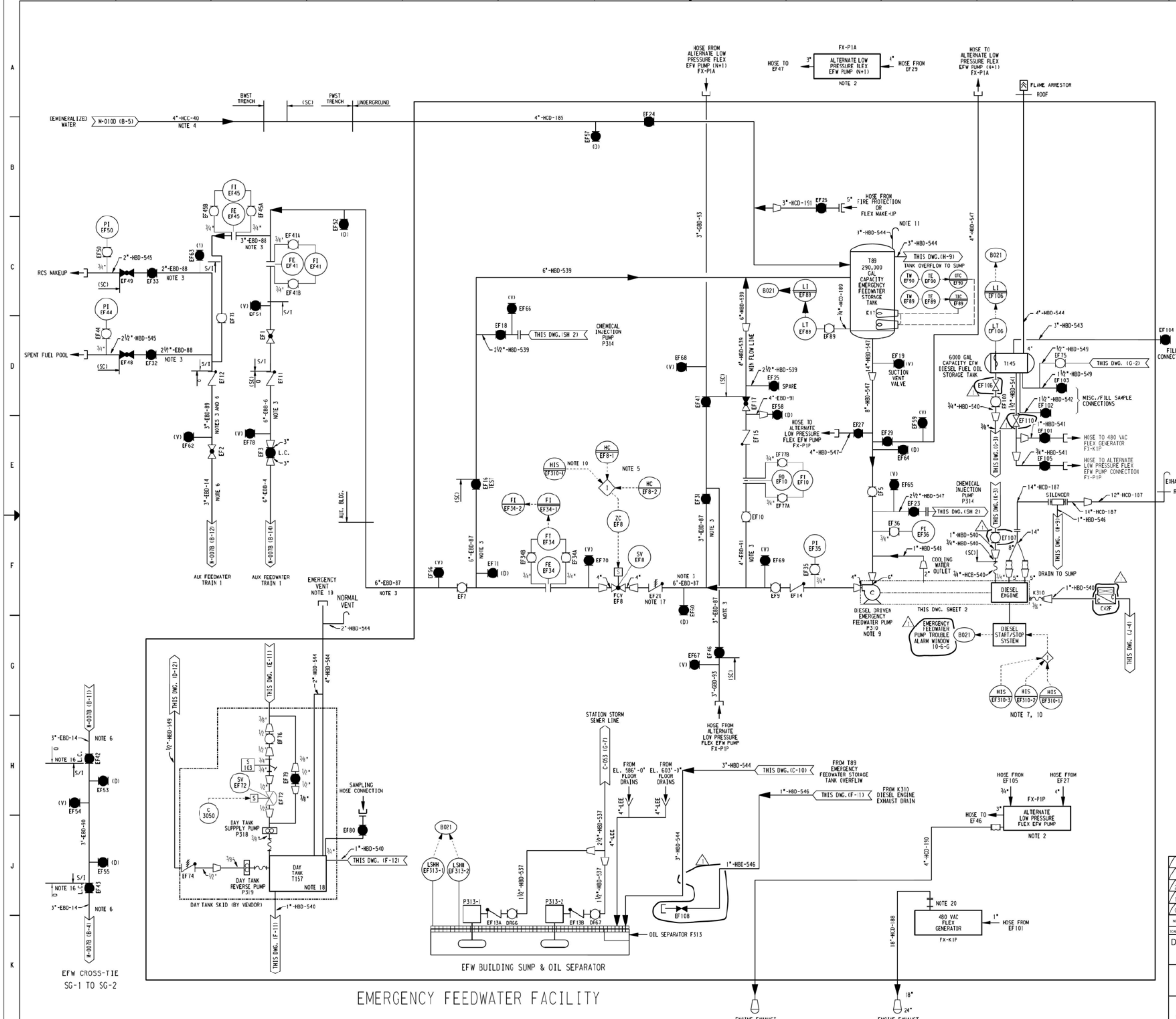
SCALE NONE | DESIGNED EMG | DRAWN PAG | DATE 05-07-90

DAVIS-BESSE NUCLEAR POWER STATION
UNIT NO 1
THE TOLEDO EDISON COMPANY

FUNCTIONAL DRAWING
DOMESTIC WATER SYSTEM

FIGURE NO	REV
FIGURE 9.2-5	11

DFN=G/USAR/UFIG925.DGN
DB 08-04-20



NOTES:

1. FOR GENERAL NOTES, PIPING SYMBOLS, AND P&ID INDEX, SEE DWG. M-001.
2. ALTERNATE LOW PRESSURE FLEX EFW PUMP FX-P1P IS STORED IN THE EFW FACILITY. FILL CONNECTIONS WILL BE PROVIDED AS SHOWN. ALTERNATE LOW PRESSURE FLEX EFW PUMPS FX-P1P AND FX-P1A (N+1) WILL BE OPERATED AS PART OF THE DBNPS BDBEE RESPONSE.
3. ALL EBB AND EBD PIPE ASSOCIATED WITH THIS PIPE SEGMENT HAS MINIMUM 900 LB VALVES AND PIPING IS SCHEDULE 80 FOR ALL PIPE SIZES.
4. LINE CLASS HCC-40 DOWNGRADED TO ANSI B31.1 (M-200Q).
5. HCEF8-1 IS LOCATED AT EFW LOCAL PANEL C3046. HCEF8-2 IS LOCATED AT EFW MCR PANEL C5732
6. CRITICAL, SAFETY RELATED Q ANSI B31.1, EQUIVALENT TO ASME, SECTION III, CLASS 3 (DCM SECTIONS III.E.2.1.1.1 & III.E.2.1.1.1.3)
7. FOR DESCRIPTION OF EFW SYSTEM CONTROLS, REFER TO OS-0062 AND OS-0053.
8. ALL SAFETY RELATED AND ANSI B31.1 CRITICAL PIPING SHALL BE CLASSIFIED AS SEISMIC CLASS I. REMAINDER OF EFW PIPING SHALL BE DESIGNED TO SEISMIC CLASS I.
9. EMERGENCY FEEDWATER PUMP P310 DRAINS TO SUMP VIA TEMPORARY TUBING.
10. HISEF310-1 IS USED FOR START/STOP OF P310 FROM EFW LOCAL PANEL C3046.

    HISEF310-2 IS USED FOR START/STOP OF P310 FROM EFW MCR PANEL C5732.

    HISEF310-3 IS USED FOR EMERGENCY STOP OF P310 FROM EFW MCR PANEL C5732.

    HISEF310-4 IS USED FOR REMOTE/LOCAL OPERATION OF FLOW CONTROLLERS HCEF8-1 AND HCEF8-2 AND IS LOCATED AT EWF LOCAL PANEL C3046. WHEN HISEF310-4 IS IN "LOCAL", FLOW CONTROL WILL BE PERFORMED FROM HCEF8-1. WHEN HISEF310-4 IS IN "REMOTE", FLOW CONTROL WILL BE PERFORMED FROM HCEF8-2.
11. EWST VENTS TO THE FUEL OIL STORAGE ROOM.
12. UNLESS OTHERWISE NOTED, ALL DRAINS WILL BE 1" & ALL VENTS 3/4". LINE IDENTIFICATION SAME AS HEADER.
13. FOR INSTRUMENTATION SYMBOLS, SEE DWG. M-002.
14. FOR INSTRUMENT LIST, SEE DWG. M-720I.
15. FOR PIPING AND MATERIAL SPEC. SHEET SEE DWG. NO'S. M-601 AND M-602 AND SPECIFICATION M-200.
16. LOCKED CLOSED POSITION OF VALVES EF42 AND EF43 AND SYSTEM FUNCTIONS SHOWN ARE FOR NORMAL OPERATION USING AUXILIARY FEEDWATER TRAIN #1 COMPONENTS.
17. VALVE EF20 IS A LIFT CHECK VALVE.
18. SUPPLY AND RETURN I&C IS NOT SHOWN.
19. EMERGENCY VENT SPRING-OPERATED CAP WITH OPENING PRESSURE OF 0.5 PSIG AND FULL OPEN PRESSURE OF 2.5 PSIG.
20. 18"-HCD-188 IS HOODED OVER THE 480VAC FLEX GENERATOR EXHAUST NOZZLE.

REVISION 33
SEPTEMBER 2020

NO.	DATE	REVISIONS	BY	CH'K	ENGR.	ENGR. SUPV.
2		INC. CN 18-160	SAP			
1		INC. CN 16-133	INITIALS ON FILE			
0		INFORMATION ONLY (UCN 16-065)	INITIALS ON FILE			

SCALE NONE | DESIGNED N/A | DRAWN N/A | DATE N/A

DAVIS-BESSE NUCLEAR POWER STATION
UNIT NO. 1
Q

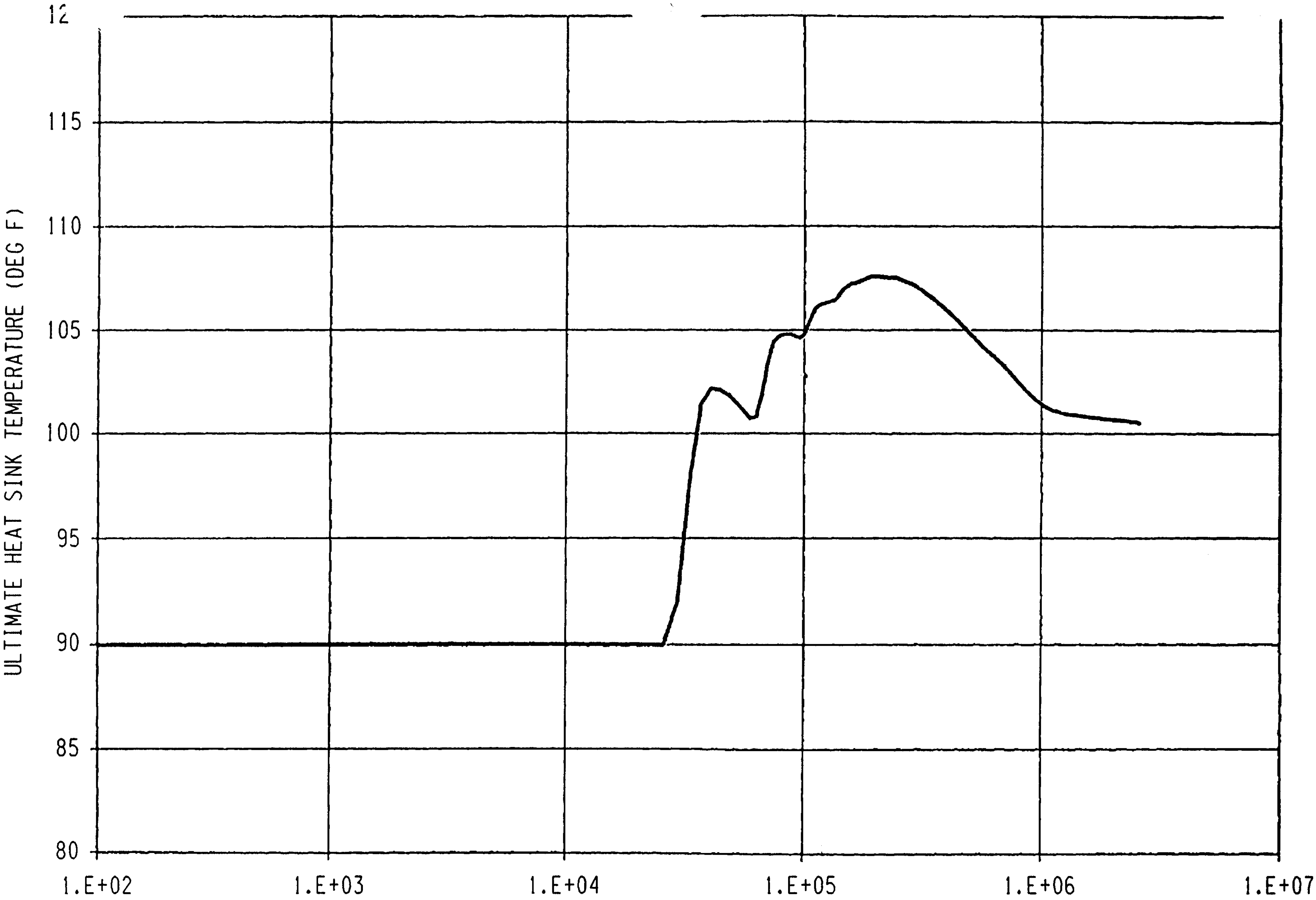
PIPING & INSTRUMENTATION DIAGRAM
EMERGENCY FEEDWATER SYSTEM

FENOC

FIGURE NO.	SHEET NO.	REV.
FIGURE 9.2-7		2

DB 04-21-20

DFN=G/USAR/UFIG927.DGN



NOTE: DISCONNECTION FROM LAKE ERIE IS ASSUMED.

REVISION 23
NOVEMBER 2002

DAVIS-BESSE NUCLEAR POWER STATION
ULTIMATE HEAT SINK TEMPERATURE
AFTER A 14.14 FT^2 HOT LEG BREAK
(WITHOUT ECCS QUENCHING)
FIGURE 9.2-6

9.3 PROCESS AUXILIARIES

9.3.1 Station and Instrument Air System

9.3.1.1 Design Bases

The station and instrument air system is designed to provide a reliable continuous supply of dry, oil-free compressed air for pneumatic instrument operation and for control of pneumatic valves. The system also supplies air to service outlets through the station for operation of pneumatic tools or other requirements.

The Emergency Instrument Air Compressor and the Station Air Compressors supply air at a pressure of a maximum of 118 psig to the station and instrument air systems.

The system is designed to handle air at 120°F ambient temperature and 100% relative humidity. The instrument air dryers are capable of delivering dry air with a moisture content corresponding to a dew point of not more than -40°F at 100 psig. The filters are capable of removing 98 percent of suspended particles larger than 0.07 microns.

9.3.1.2 System Description

Figure 9.3-1 is a functional drawing of the system. The system has two station air compressors and one emergency instrument air compressor. The capacities for Station Air Compressors 1-1 and 1-2 are 1306 scfm and 1306 scfm respectively . During normal operation, one station air compressor will operate to supply station and instrument air requirements. The remaining air compressors are placed in "standby" and will automatically load upon decrease of supply air header pressure. To ensure instrument air supply, the station air header is automatically isolated when the station air header pressure drops to a preset value.

A 787 scfm capacity, oil-free Emergency Instrument Air Compressor is provided to supply instrument air during a malfunction of the Station Air Compressors. In the event of a Loss of Offsite Power, both the Station and the Emergency Instrument Air Compressors will trip. However, the Emergency Instrument Air Compressor has provisions to be powered from the Diesel Generators and manually restarted. In addition, the Emergency Instrument Air Compressor can be utilized to supply station air in the event of a malfunction of the Station Air Compressors.

Temporary Air Compressor(s) can be connected to the system upstream of valve SA57 to provide another source of air.

All three air compressors have their own individual air receivers.

Protection against loss of instrument air is provided by redundancy and/or bypass provisions in compressors, receivers, filters, dryers, and supply headers. In addition, in the event of loss of all instrument air supply, all pneumatically operated valves are either arranged to assume their respective "fail-safe" positions or are provided with a safety related air accumulator to actuate the component in an emergency.

The instrument air supply is fed into a loop header and branch lines are taken off the loop header to supply all areas of the station. This loop header will enable maintenance on any one instrument air branch line at a time without interrupting the air supply to other branch lines. An instrument air receiver is connected to the loop header.

The design codes and standards applicable to the station and instrument air system components are listed in Table 9.0-1.

9.3.1.3 Air Cleanliness

Compressed air is filtered and dried before being used as instrument air. This is achieved by the use of air dryers and filtration equipment. There are two dual chamber air dryers installed in parallel. Air dryer T80-1 and T80-2 uses one prefilter, which can be manually bypassed on high differential pressure and two afterfilters which are connected in parallel.

Dryer T186-1 and T186-2 has one prefilter and one afterfilter, the filters can be manually bypassed in case of high differential pressure. The dual chamber air dryers are heatless type using activated alumina desiccant. The air downstream of the afterfilter exceeds the guidelines presented in ANSI/ISA S-7.3, in that the prefilters are 100 percent efficient for particulates larger than 0.6 microns and the afterfilters are 98 percent efficient for particulates larger than 0.07 microns. Also, the maximum moisture content corresponds to a dew point of not more than -35 degrees°F at the normal system pressure.

Normally only one air dryer is in operation at a time with the other dryer in a standby mode.

9.3.1.4 Safety Evaluation

None of the engineered safety features depends on the supply of instrument air for its operation.

Table 9.3-1 lists the safety-related air-operated valves that are required for safe shutdown. All these valves are designed to assume a safe position by spring actuation and/or by a Q-listed accumulator system following a loss of instrument air. All safety-related air-operated valves are designed to fail in the safe position listed in Table 9.3-1.

9.3.1.5 Tests and Inspections

Regular maintenance of the equipment is performed to ensure cleanliness. This includes the regular blowing down of receivers, changing filter cartridges, and checking on the performance of the air dryer unit.

Each compressor is inspected periodically to ensure equipment operability. The system is periodically tested to demonstrate proper operability of the system.

During normal operation, one Station Air Compressor is operated. The remaining air compressors are placed in “standby” and will automatically load upon decrease of supply air header pressure. The emergency instrument air compressor and other station air compressor are checked and operated periodically without interruption of the station air supply.

9.3.1.6 Instrumentation and Controls

Instrumentation (a portion of which is in the Control Room) to monitor the station and instrument air system includes:

a. Trouble alarm for each compressor

b. Trouble alarm for the instrument air dryer package

c. High air temperature alarm on discharge of each aftercooler

d. High station air compressor jacket water temperature alarm

e. Instrument and station air header low-pressure alarms

Pressure control valves, provided on non-vital headers, automatically shut if the pressure reaches a very low level. This ensures the availability of an instrument air supply to important instruments.

9.3.2 Process and Post-Accident Sampling Systems

9.3.2.1 Process Sampling Systems

9.3.2.1.1 Design Bases

The station sample system provides samples from the reactor coolant system, the makeup and purification system, the chemical addition systems, the secondary steam system, and containment areas. The samples of primary importance are drawn to central stations in the auxiliary and turbine buildings, eliminating the need for personnel entry into the containment vessel or the radwaste areas during normal operation. Samples of secondary importance are grab samples, taken locally at the sample connections provided at appropriate locations throughout the station.

The samples are collected and prepared for laboratory and/or continuous analysis. Typical analyses performed are:

a. Reactor coolant boron concentration

b. Fission product activity level

c. Dissolved gas content

d. Corrosion product concentration

e. Condensate activity

The analytic results are used for regulating boron concentration adjustments, evaluating the integrity of fuel rods and the performance of the demineralizers, and regulating chemical addition to the reactor coolant.

All materials, fabrication, installation testing, and examination were in accordance with applicable portions of the latest issue of the following codes.

ASME Section III	Nuclear Power Plant Components
ASME Section II	Material Specification
ASME Section IX	Welding Qualifications

ANSI Code for Pressure Piping

B16.11	Forged Steel Fittings, Socket Weld and Threaded
B31.1.0,1967	Power Piping

9.3.2.1.2 Description and Operation

Figures 9.3-3A and 9.4-11A are the functional drawings of the sample system. Each system permits the drawing of its respective samples manually during normal station power and shutdown operations.

Four alternate paths have been provided for the containment air samples. These air samples are drawn to the containment gas hydrogen analyzers and to the containment atmospheric gas and particulate radioactivity monitors.

During normal operation the following samples can be obtained from the designated systems. The samples marked with an asterisk (*) may be continuously monitored.

a. Reactor Sample System

    1. Reactor Sample Hood

        a. Core flooding tank

        b. Seal return cooler

        c. Makeup filter

        d. Purification demineralizer inlet

        e. Purification demineralizer outlet

        f. Makeup tank, liquid

        g. Decay heat removal cooler

        h. Pressurizer, liquid and vapor space and letdown coolers outlet

        i. Makeup tank, gas

        j. Pressurizer quench tank

b. Feedwater and Steam System

    *1. Condensate pump common discharge

    *2. Feedwater heater E-2, common outlet

    *3. Condensate polishing demineralizer, inlet

    *4. Condensate polishing demineralizer, outlet

    *5. Deaerator storage tanks 1 + 2, inlets

    *6. Deaerator storage tanks 1 + 2, outlets

*7. Feedwater heaters E-6, common outlets

*8. Feedwater heater 1-1-6, outlet

*9. Feedwater heater 1-2-6, outlet

*10. Steam generator 1-1, water (available only below 15 percent power)

11. Deleted

*12. Steam generator 1-2, water (available only below 15 percent power)

13. Deleted

14. Moisture separator reheater drain tank 1-1

15. Moisture separator reheater drain tank 1-2

16. Feedwater heater 1-1-4 drain

17. Feed water heater 1-2-4 drain

*c. Containment vessel gas analyzer system. Containment vessel volume atmosphere (any of four sample points)

Secondary system samples are purged to the seal water drain tank or the condenser pit. Normal primary samples are purged to the RCDT or Miscellaneous Waste Drain Tank via the Auxiliary Building Sump. Samples taken on an intermittent time basis are circulated for a predetermined time period (a function of the sample stream flow rate and transfer line volume) to provide a valid sample at the central panel. A portion of this stream is directed into a closed sample container for collection. Samples are collected in containers designed for full operating temperature and pressure and at flow velocities which ensure transport of suspended particles where appropriate. The non recoverable sample drains are drained to the Auxiliary Building sump area.

Samples requiring constant monitoring have their analyzers installed in parallel with a continuously flowing sample stream. A portion of this stream is conditioned per the analyzer requirements and passed through the analyzer. This parallel path arrangement minimizes response time by permitting sample flow rates greater than the individual requirements of each analyzer.

The sample take-off point design is per the ASME Performance Test Code 19.00-1970 Section III.

The sample system construction is per Section III of the ASME code. The tubing is of stainless steel construction with sizes determined per operating conditions.

The systems do not employ any automatic sequencing operations for sample selection. All multiple sampling operations are manual with check valves on the inlet and return manifolds to prevent incorrect reverse flow.

9.3.2.1.3 Safety Evaluation

9.3.2.1.3.1 Inherent Design Features

All feedwater and steam samples are reduced in pressure and temperature to protect the process analyzing instruments. Safety valves are included where necessary to protect these devices.

Gaseous leakage is collected by placing the reactor sampling station under a hood provided with a fan and HEPA/charcoal filter exhausting to the radwaste area ventilation system. Liquid leakage from the valves in the hood and sampling effluents are drained to the Auxiliary Building Sump area.

Each sample line of the Reactor Sample System is properly shielded to reduce the total emitted radiation to below 2.5 mr/hr.

The Containment Vessel Gas Analysis System is classified as Seismic Class I and is designed and fabricated accordingly. Portions of the Reactor Sample System are classified as Seismic Class 1. The Steam and Feedwater Sample System is not classified as Seismic Class 1, but is designed and fabricated to reduce its vulnerability to earthquakes.

9.3.2.1.3.2 Operational Failure Considerations

To evaluate system safety, several failures and malfunctions were assumed concurrent with a loss-of-coolant accident, and the consequences were analyzed. The analysis is summarized in Table 9.3-3.

9.3.2.1.4 Instrumentation Applications

The normal sampling systems perform no emergency functions other than monitoring the reactor containment vessel atmosphere for proximity to the lower explosive limit for hydrogen. The reactor coolant quality is maintained within the limits given in Table 9.3-4, and the steam generator feedwater quality is maintained within the limits given in Table 9.3-5.

Each continuous analyzer in the subject systems has an alarm initiation function for off-normal or hazardous level conditions.

9.3.2.2 Post Accident Sampling System

9.3.2.2.1 Design Bases

The design basis requirements for post accident sampling were eliminated by License Amendment 264. Post accident sampling is no longer a required design function. The following discusses the capabilities and operation of the installed PASS equipment.

9.3.2.2.2 Description and Operation

The design basis requirements for the PASS were eliminated by License Amendment 264.

The PASS-liquid has capability to obtain grab samples from the following sources.
-Pressurizer liquid and vapor spaces
-RCS cold leg loop-2

-Containment Emergency sump via the decay heat removal DHR system
-Letdown System

The PASS-containment air has capability to obtain grab samples from containment air, via the containment radiation monitoring system.

The PASS-liquid and the PASS-containment air are non-safety related systems. The technicians wear self-contained breathing apparatus during sampling operation.

The PASS-liquid and the PASS-containment air functional drawings are shown in Figure 9.3-3A and Figure 9.4-11A respectively.

The PASS-liquid is located in Rooms 106 and 106A (545' el) of the Auxiliary Building. The PASS-liquid provides capability to sample RCS, DHR and letdown systems. The fluid from appropriate sample locations in these systems are routed to a sample cave through sample coolers. The sample coolers cool the sample fluid to approximately 120°F. The system is purged either to the Reactor Coolant Drain Tank or Pressurizer Quench tank. The sample cave consists of a 30 cc sample flask and a 150 cc gas expansion tank from which liquid and off-gas samplers are collected in 12 ml vials. The system is flushed with demineralized water each time after taking a sample. In addition, the PASS-liquid has the capability to obtain high pressure liquid samples in a shielded shipping cask. This shipping cask is used to transport the samples to off-site analytical laboratory.

The PASS-liquid control panel is shielded from the sample coolers by 8" concrete and 4" lead shield walls. A 4" lead shield wall separates the control panel and the sample cave. The sample cave provides a 3" lead shielding to minimize radiation exposure during sampling. In addition, flow gauges are repeated in an adjacent low radiation area so that the technician need not stay at the control panel during system purging with post accident fluids. The sample coolers are flushed with demineralized water to minimize radiation fields when samples are being drawn from the sample cave.

The PASS design does not required an isolated auxiliary system to be placed into operation in order to draw samples. The valves used in the PASS are qualified for the environmental conditions in which they are expected to operate. A flow restricting orifice is provided in the RCS cold leg sample line to limit the loss of reactor coolant in case of a sample line rupture. In addition, the pressure reducing valve in the system will also act as a flow restrictor.

The PASS-containment air is located in the fuel handling area, Room 300 (585' el) in the Auxiliary Building. The containment air is recirculated back to the containment through this system using a pump. The samples are obtained through a septum using a syringe. The system has provisions to obtain cartridge samples to quantify iodine and particulate activities when these activities can not be quantified from the air sample. The operation of this system does not require an isolated auxiliary system be placed into operation.

9.3.2.2.3 Safety Evaluation

The design basis requirements for the PASS were eliminated by License Amendment 264. Post accident sampling is no longer a required design function.

9.3.2.2.4 Instrumentation Application

The PASS does not contain any online monitors that are used for post-accident monitoring. The PASS-liquid sample path contains an in-line monitor which can be used to determine the hydrogen concentration in the RCS when the sample is being drawn. The containment atmosphere is monitored for hydrogen by redundant in-line hydrogen monitors (see Section 6.2.5). In the event these monitors are not available, hydrogen concentration can be determined from the off-gas and containment atmosphere samples. Onsite analytical capability is provided to determine radioactivity, boron, chloride and hydrogen. If needed, dissolved oxygen and pH can also be performed onsite. The samples are diluted in the laboratory as necessary to meet the instrumentation sensitivity limitations.

The PASS-liquid has been provided with instrumentation to monitor flow, pressure and temperatures while samples are being drawn. Flow gauges are repeated in a low radiation area. The PASS-containment air is equipped with instrumentation to monitor the flow of containment air through the system.

9.3.3 Equipment and Floor Drainage System

The auxiliary building drainage system includes equipment drains, floor drains, sumps, oil interceptor with oil storage tank, acid neutralizing tanks, and sump pumps with associated sump level controls. Radioactive equipment drains and floor drains either discharge directly into the Miscellaneous Waste Drain Tank (MWDT) or discharge into sumps which are pumped into the MWDT for radioactive waste processing. Low-radioactivity wastes from designated laboratory sinks, decontamination facilities, and floor drains are collected in the detergent waste drain tank for radioactive waste processing. The sump pumps in the Emergency Core Cooling System (ECCS) pump rooms, which normally discharge to the MWDT, are provided with a bypass to discharge directly to the Clean Waste Receiving Tanks (CWRT). These ECCS sump pumps and piping are designed to Seismic Class I. Disposal of collected radioactive drainage is discussed in Chapter 11. Figure 11.2-3 is a functional drawing of the Miscellaneous Liquid Radwaste System. Nonradioactive areas, such as the electrical penetration rooms, electrical switchgear rooms, cable spreading room, and control room, discharge to the station storm sewer. An oil interceptor is provided for the emergency diesel generator rooms drains and diesel fuel day tank rooms drains. Acid-neutralizing tanks are provided for both battery room drains.

The turbine building drainage system includes equipment drains, floor drains, sumps, oil interceptors with oil storage tanks, and sump pumps with associated sump level controls. Turbine building equipment drains and floor drains are collected in low-point sumps and pumped through oil interceptors before discharging into the station storm sewer. Some drains, where oil is not expected, discharge directly to the station storm sewer. In the event of radioactivity in the secondary system, sump contents can be pumped to the settling basins or to the condensate demineralizer holdup tanks. In addition to the normal condenser pit sump pumps, there are two flood pumps that operate on high-high sump level and discharge to the settling basins or alternatively to the transformer collection tank. The condenser pit flood pumps cannot pump against the head of the sump pumps when aligned to their normal flow path to the settling basin.

The containment building drainage system includes floor drains, equipment drains, the normal sump, and submersible type sump pumps with associated sump level controls. The normal sump in the containment vessel is pumped directly into the MWDT or alternatively may be aligned to be pumped to the CWRT.

Figure 9.3-4 and Figure 9.3-5 are schematics of the Station Drainage System. Disposal of collected radioactive drainage is discussed in Chapter 11.

As shown on Figure 9.3-4, most sumps are provided with a duplex pump system with an alternator to automatically start, stop, and alternate each of the pumps and to start the second pump any time the one in operation is unable to carry the load. The exceptions are the following: the Unit Lube Oil Storage Room Sump, which is pumped as necessary using a portable sump pump; the Water Treatment Backwash Sump, which utilizes three sump pumps (two of which alternate and the third of which is manually controlled); the Water Treatment Building Sump, which has had the sump pumps removed and is now pumped manually by a portable pump on an as-needed basis; and, the marsh transformer vault and the two 345KV switchyard manholes, which utilize only one pump each. All station drainage system sumps are provided with high-level alarms with the exception of the marsh transformer vault and the Water Treatment Building Sump (alarm removed).

Applicable design codes and standards are listed in Table 9.0-1.

Table 9.3-6 provides a list of certain rooms containing essential equipment and the associated floor drain and pipe routing drawings. This table was generated in response to FSAR question 9.3.4 to identify rooms outside of containment that contained essential equipment and were considered susceptible to flooding. See Figure 9.3-6 for a definition of symbols. Also, see Subsection 3.6.2.7.2 for a discussion of all Seismic Class II piping which could cause flooding in the auxiliary building and intake structure.

All potential sources of flooding in compartments housing safety-related equipment are discussed in Section 3.6. Flooding due to a single failure will not affect safe shutdown of the plant. Precautions taken by the installation of walls, curbs, and pressure doors are discussed in the above referenced section and responses. In addition to control room high level alarms on all sumps, the operator would be further alerted to the malfunction of a process system by applicable indications or alarms from tank levels, flow, pressure, and/or temperature sensors.

The sumps and sump pumps serving compartments containing safety-related equipment are not sized to handle a major component rupture. However, the ECCS room sump pumps are Q-listed pumps. These pumps serve a large portion of the compartments containing safety-related equipment. Other sump pumps serving compartments with safety-related equipment are located in large volume areas where flooding would not affect safety-related equipment. Figure 9.3-7 shows schematically the drainage system from certain safety-related compartments. This figure was generated in response to FSAR question 9.3.4 to describe the drainage system from rooms outside of containment that contained essential equipment and were considered susceptible to flooding.

As noted in Figure 9.3-5, large portions of the roof and floor drainage system in the auxiliary building have been seismically supported to prevent flooding from external sources.

9.3.3.1 Station Drainage and Discharge System

The operation and functioning of the Station Drainage and Discharge System and its components for Davis-Besse Unit No. 1 are described in the sections that follow. There are various components and routes which make up the System. These components and their route of discharge are designed to meet the Federal and State regulations in discharging to the

environment. Due to the complexity of this system, it is broken down into sections. Each section and a brief description is given in the following:

1. Station drains going to the Toussaint River via the storm sewer system and their respective oil interceptor.

   Any area where oil can be reasonably postulated to be introduced into the drainage system is protected by an oil interceptor located between the drains and storm sewer system. The oil interceptor collects any oil by means of a skimmer and transfers it to an oil tank located near the oil interceptor. The oil tanks are equipped with a pump suction line to remove oil for proper disposal. Plant personnel check the oil tank levels and have the tanks pumped out as necessary.

2. Station and Storm drains going to the Toussaint River via the storm sewer system.

   Roof drains and drains from areas where oil and radioactivity is not typically introduced into the drainage system discharge directly to the storm sewer system, which drains by gravity into the Toussaint River.

3. Sewage plant effluent going to the Settling Basin #3.

   The plant sewage collects in wet-wells and the lift stations pump the wet-well contents to the Sewage Treatment Plant for processing. The processed effluent collects in the Sewage Treatment Plant Sump and is pumped to Settling Basin #3, which overflows into the Settling Basin Sump and is then pumped to Lake Erie via the Collection Box.

4. Settling Basin effluent going to the Collection Box.

   The effluent from the Water Treatment Bldg Backwash Sump discharges into the Settling Basins. Two pumps in the sump start and stop automatically and the third pump must be started locally when needed. The effluent from the turbine building sumps (east and west condenser pit sumps) can also be aligned to discharge into the Settling Basins. The Settling Basin water overflows into the Settling Basin Sump and is pumped to Lake Erie via the collection box.

5. Station parking lot drains going to the north marsh via their respective oil interceptor.

   The drains from the station's north parking lot are routed through an oil interceptor before draining into the north marsh to prevent any oil from getting into the marsh.

6. Transformer Collection Tank operation.

   The retention areas of the Main, Auxiliary, Bus-tie (AC & BD) and Startup 01 transformers drain into the transformer collection tank. Plant personnel check the tank level and have the tank pumped out as necessary. Any oil that has accumulated is collected for proper disposal. The Condenser Pit Flood Pumps may be aligned to discharge into the transformer collection tank.

7. Collection box discharge to Lake Erie via the 72-inch concrete discharge pipe.

   The effluent discharging from the following systems discharges into Lake Erie via the collection box: Settling Basins, Processed Radwaste, Circulating Water Blowdown. The Dilution Pump is normally used to dilute the effluent going to the lake. If desired, the Cooling Tower Makeup Pump discharge may be directed to the lake via the collection box.

8. Screen Wash Catch Basin discharge to a permitted outfall.

   Screen wash water discharges into the Screen Wash Catch Basin. Debris washed from the screens will settle in the basin while the water will overflow to a Federal and State environmental regulation permitted outfall. Periodically the basin will be de-watered and the sediment removed for disposal.

9. Switchyard drains discharging to the north marsh and Toussaint River.

   The switchyard is graded high in the middle and slopes down to the east and to the west. The west slope drains to the north marsh via an open ditch while the east slope drains to the Toussaint River via the storm sewer system.

10. Auxiliary Building and Containment Drains.

    The floor and equipment drains in the Auxiliary Building and Containment drain to their respective sump which discharges to the Miscellaneous Waste Drain Tank (MWDT), except for selected floor and equipment drains in the Auxiliary Building which drain directly to the MWDT or to the Detergent Waste Drain Tank (DWDT). Contents of the MWDT and DWDT are processed and the effluent is pumped to the collection box.

11. Deleted

12. Battery Rooms and Relay House Drains.

    All Battery Rooms and Relay House drains are routed through acid neutralizing tanks prior to discharging to the Storm Sewer System, to prevent any acids from being introduced into the Storm Sewer System.

13. Marsh Level Control.

    The levels in the marshes are determined and directed by the U.S. Fish and Wildlife Dept. These levels are maintained by station personnel by the use of the marsh pumps located in the marshes. These marsh pumps discharge directly into Lake Erie.

14. Radiation monitoring of the Station Drainage

    A Radiation monitoring device is installed in the storm sewer discharge line. The monitor is provided with an annunciator alarm as well as a computer point. Each individual radwaste line contains a radiation monitor which, during releases of radioactive waste is set to alarm and initiate automatic closure of its isolation valve prior to exceeding applicable limits. Operation of the monitors is covered in

Process Radiation Monitors (Section 11.4). A sample line is provided in the collection box discharge line to the lake to permit a grab sample to be taken.

15. Effluent disposal from manholes, retention areas, etc.

    Station personnel periodically check the precipitation accumulation in the various manholes, retention pits, flow test box, etc. When a measurable amount of precipitation collects, these areas are pumped out with a portable submersible pump. Any oil that has accumulated is collected for proper disposal.

16. Temperature control of the discharge water from the collection basin to the lake.

    Technical Specification requirement deleted by License Amendment 55.

17. Acid Neutralizing Tank Operation.

    The function of the acid neutralizing tank located in the relay house and the two (2) acid neutralizing tanks located north of the auxiliary building is to neutralize any acid that would leak from the batteries into the drainage system. The neutralizing tank and sumps are filled with approximately 600 pounds of limestone pieces one (1) to three (3) inches in size. The limestone, which is the neutralizing agent, will raise the pH of the acid waste to 4.50 to 7.00. The neutralizing tanks are checked periodically by station personnel and new limestone added when necessary. Complete removal and replacement of new limestone will be every ten (10) years.

9.3.3.2 Post-LOCA Sump pH-Control

The Davis-Besse Unit 1 design utilizes Trisodium Phosphate for pH control. It is stored in baskets located on the 565' elevation of containment adjacent to the normal sump and in the normal sump area. The design calculations to determine the required amount of Trisodium Phosphate were conducted assuming the maximum Technical Specification boron concentrations and water volumes for water that will be injected into containment in the event of a LOCA. It was found that 250 cubic feet of Trisodium Phosphate was required to assure a pH of between 7 and 8 within four (4) hours. On the other extreme, a calculation based on 325 cubic feet (original plus new basket capacity) was conducted to assure that the fluid pH will not exceed the upper limit of 11 if the injected water was at its minimum Technical Specification boron concentration and volume.

9.3.4 Makeup and Purification System

9.3.4.1 Design Bases

The Makeup and Purification System is designed to accommodate the following functions during normal reactor operation:

a. To control the Reactor Coolant System (RCS) inventory during all phases of normal reactor operation

b. To receive, purify and recirculate reactor coolant water

c. To maintain the required boron concentration in the RCS (to control reactivity)

d. To provide seal injection water for the 4 reactor coolant pumps

e. In conjunction with the pressurizer, accommodate changes in reactor coolant volume due to small temperature changes

f. To maintain the proper concentration of hydrogen and hydrazine for oxygen control, zinc for source term reduction, and lithium for pH control

g. Provide makeup to the RCS for protection against small breaks in the RCS pressure boundary

h. To degas the RCS

i. To add borated water to the Core Flooding Tanks.

In addition, the Makeup and Purification System provides feed and bleed capability to maintain core cooling in the event of a loss of all secondary side cooling.

9.3.4.2 System Description

9.3.4.2.1 General

The Makeup and Purification System is shown schematically in Figure 9.3-16. Tables 9.3-7 and 9.3-8 list the system performance and component design characteristics.

Letdown Cooler:

The letdown cooler reduces the temperature of the letdown flow from the reactor coolant system to a temperature suitable for demineralization and injection to the reactor coolant pump seals. The letdown coolers are two shell and tube heat exchangers arranged in parallel. Each is designed to handle half the maximum letdown flow. Heat in the letdown cooler is rejected to the component cooling water passed through the shell side of the coolers.

Letdown Flow Control:

The normal letdown flow rate at reactor operating pressures is controlled by a fixed block orifice sized for the normal purification rate. A parallel, normally closed, remotely operated valve can be opened to obtain flow rates up to the maximum letdown capability, e.g., reactor coolant boron concentration adjustment. This valve is also used to maintain the desired letdown rate at reduced reactor coolant pressure, i.e., during startup and shutdown. In addition, there is a second parallel, normally closed valve which may be manually positioned for flow control.

Design maximum letdown flow is 140 gpm, and each letdown cooler is sized for 70 gpm. The remotely operated valve can pass 188 gpm in the fully open position, and total flow, including the block orifice path, is 258 gpm. To exceed 140 gpm, letdown flow would require operator error or controller failure in the remotely operated valve controller (this valve fails closed on loss of air). A relief valve on the letdown line downstream of the flow control devices provides overpressure protection for the 150 psig downstream system. The relief valve with a setpoint of 150 psig has a capacity of 570 gpm with 10 percent accumulation. The following letdown indications are provided:

a. Letdown flow indication.

b. Letdown line pressure high alarm (downstream of flow control devices).

c. High ΔP alarm for makeup filters.

d. High temperature alarm downstream of letdown coolers along with high temperature interlock to close cooler inlet valve at 135°F.

Downstream relief valve has ample capacity at 570 gpm to prevent overpressure even if block orifice and remotely operated valve both become fully open.

At full RCS pressure, it is not considered credible that the remotely operated and the manually operated valves will be open simultaneously.

During plant conditions with reduced RCS pressures, it may be desirable to maintain 140 gpm letdown flow by opening all three parallel letdown flow control valves.

Letdown Flow Radiation Monitor:

The letdown flow radiation monitor is discussed in Subsection 11.4.2.

Purification Demineralizer Filter:

The purification demineralizer filter is designed to remove particulate matter from the letdown stream prior to entering the purification demineralizer. This filter minimizes any accumulation of radioactive crud in the purification resin and on the downstream piping of the purification system and waste disposal system. The filter may be bypassed during normal operation. The purification demineralizer filter and bypass are sized to handle the maximum letdown flow rate.

Purification Demineralizers (Mixed-bed or Cation):

There are three purification demineralizers, each capable of holding mixed bed or cation resin. These become boric acid saturated after a short period of use and remove other reactor coolant anions and cations. Since the reactor coolant may be contaminated with fission and corrosion products, the resins will remove certain radioactive impurities. Chapter 11 describes coolant activities, coolant handling and storage, and expected limits on activity discharge. Demineralizers 1-1 and 1-2 are each sized to handle one-half the maximum letdown flow rate. Demineralizer 1-3 is sized to handle the maximum letdown flow and can be used instead of, or in series with, the other two purification demineralizers. Demineralizer 1-3 is typically used for delithiating the reactor coolant system and either demineralizer 1-1 or 1-2 being used for normal purification while the other is in standby to be used for normal purification or delithiation as needed.

Makeup Filters:

Two makeup filters are installed in parallel to remove particulates from the effluent of the purification demineralizers. This prevents solids from entering the makeup tank and thus entering the reactor coolant pump seals and the reactor coolant system. Each filter is sized for maximum letdown flow.

Makeup Tank:

The makeup tank serves as a receiver for letdown, seal return, chemical addition, and makeup. The tank also accommodates temporary changes in system coolant volume. The volume of the tank is sized such that the useful tank volume, in conjunction with the pressurizer, accommodates the expected expansion and contraction of the reactor coolant system during normal power transients (between 15% and 100% power).

Makeup Pumps:

The makeup pumps are designed to return the purified flow to the reactor coolant system and supply the seal water flow to the reactor coolant pumps. One pump provides normal makeup to the reactor coolant system plus the required seal water flow. Both pumps are driven by motors receiving power from redundant essential bus.

Seal Return Coolers:

The seal return coolers are sized to remove the heat added by the makeup pump recirculation and the heat picked up in passage through the reactor coolant pump seals. The two shell and tube coolers are arranged in parallel and each is sized for full flow. Heat from these coolers is rejected to the component cooling water passing through the shell side of the coolers.

Seal Return Cooler Makeup Pump Operation:

During normal plant operation only one seal return cooler needs to be in operation. This method of operation prevents lowering the makeup tank water temperature. However, if both makeup pumps are operating longer than 1/2 hour, the second seal return cooler is started.

Seal Injection Filters:

Two filters of the disposable cartridge type are installed in parallel in the seal injection line. These filters remove particulates which could enter the reactor coolant pump seals and result in increased seal wear. Each filter is sized to handle normal seal injection flow.

9.3.4.2.2 Mode of Operation

The makeup and purification system is operated during all phases of the Nuclear Steam Supply Systems (NSSS) operating life, including startup, power operation, and shutdown. The system is also operated during refueling by employing the purification equipment through interconnections to the decay heat removal system.

During normal NSSS operation, one makeup pump continuously supplies high pressure water from the makeup tank to the seals of the reactor coolant pumps, and to a makeup line which is connected to the reactor inlet by a high pressure injection line. This line is the only interconnection between the makeup system and the high pressure injection system normally used during power operations. Makeup flow to the reactor coolant system is regulated by the makeup control valve, which operates on signals from the liquid level controller of the reactor coolant system pressurizer. A minimum makeup flow is maintained through a manually set bypass around the makeup control valve. This minimizes thermal fatigue of the makeup nozzle thermal sleeve by maintaining a minimum flow of the makeup water.

A control valve in the Reactor Coolant (RC) pump seal injection line automatically maintains the desired flow rate to the seals. Needle valves in the individual seal injection lines are used to manually throttle flow to the seals of each pump. A part of the water supplied to the seals leaks into the reactor coolant system. The remainder returns to the makeup tank after passing through one of the two seal return coolers.

Seal water inleakage to the reactor coolant system requires a continuous letdown of reactor coolant to maintain the desired coolant inventory. In addition, letdown of reactor coolant is required for removal of impurities and boric acid from the reactor coolant and to accommodate volume changes in the reactor coolant system during change in power level. During normal power operations, reactor coolant is removed from one of the reactor inlet lines, cooled during passage through letdown cooler(s), passed from the containment vessel through a containment isolation valve, reduced in pressure during flow through the letdown flow control station, and then passed through a purification demineralizer. Normally the letdown flow is routed through the “operational” purification demineralizer, either 1-1 or 1-2, which is used as a mixed bed, to maintain good chemistry (low chlorides and fluorides) and to remove radiodines to keep the dose equivalent iodine low. That purification demineralizer will also remove radiocesium and other gross activity. The “operational” purification demineralizer will be Li-7 saturated and the “spare” purification demineralizer will be normally unsaturated with Li-7. Purification demineralizer 1-3 is normally used for delithiating the reactor coolant system. The “spare” purification demineralizer can be used for additional cation and/or anion contaminant and Li-7 removal when needed. A three-way valve (MU 11) directs the coolant either through the makeup filter to the makeup tank or to the clean radioactive waste disposal system. The normal letdown flow rate is approximately 70 gpm. This permits recirculation of greater than one reactor coolant system volume through the purification train during a 24 hour period. The maximum letdown flow rate allowed at full reactor pressure is 140 gpm. This flow rate permits changing boron concentration by bleeding coolant from the reactor coolant system during xenon peaking following a 50 percent power change. During this period, unborated reactor grade water is added to the reactor coolant system to dilute the boric acid concentration in the reactor coolant system. This is done to compensate for the negative reactivity addition resulting from the xenon peaking.

Normally, the three-way valve is positioned to direct the letdown flow to the makeup tank. If the boric acid concentration in the reactor coolant is to be reduced, the three-way valve is positioned to direct the letdown flow to the clean radioactive waste disposal system. Boric acid removal is accomplished in the clean radioactive waste disposal system either by directing the letdown flow through a deborating demineralizer with the effluent returned directly to the makeup tank, or by directing the letdown flow to a clean waste receiver tank. The level in the makeup tank is maintained with deborated water from storage or with demineralized water from the station demineralized water storage tank. The flow of demineralized water is measured and totaled by an inline flow integrator and associated instrumentation. The flow of demineralized water to the makeup tank is controlled remotely by the makeup tank feed flow valve. During normal operation the flow integrator (batch controller), the integrated control system interlock, or the operator will terminate dilution. The above procedure for reducing the reactor coolant system boric acid concentration is the feed and bleed method.

When in service the purification demineralizer filter (i.e., the prefilter) can process all reactor coolant letdown from the primary system. The purification demineralizer filter is normally bypassed during plant operation, and is placed in service during heatup, cooldown and outage periods.

During cooldown operations (when there is a relatively greater possibility of crud severely contaminating purification demineralizer resins), the spare makeup filter can be positioned to serve as backup for the purification demineralizer filter. Should the purification demineralizer filter require maintenance, the spare makeup filter can be used in its place. With this arrangement, prefiltration is omitted only in the unlikely event that neither of the two filters is available for service.

The makeup tank also receives chemicals for addition to the reactor coolant. Chemicals in solution are injected into the letdown flow upstream of the makeup filters and then pass into the makeup tank which serves as a final mixing location. A hydrogen overpressure is maintained in the tank to ensure that a predetermined amount of dissolved hydrogen remains in the reactor coolant.

System control is accomplished remotely from the control room with the exception of the seal return coolers, and purification demineralizer 1-3 for operating in series with the other purification demineralizers. The letdown flow rate is established by the block orifice during normal operation, but may be increased by opening the letdown control valve. The fixed block orifice is sized to pass the normal flow rate of approximately 70 gpm at the operating pressure of 2200 psia. The parallel remotely operated control valve is throttled to obtain total flow rates above the block orifice flowrate. The parallel manually operated valve is normally closed and is used only if the remotely operated valve is isolated and is out of service and letdown flow greater than 70 gpm is needed or both the block orifice and remotely operated valve are inoperable. This latter circumstance is considered extremely unlikely.

During unit operations when RC system pressure is less than normal operating pressure, the remotely operated control valve is used in addition to or instead of the block orifice to obtain necessary letdown flow because the block orifice is a fixed restriction designed to pass approximately 70 gpm at normal RC system pressure. The manually operated valve can also be used along with the remotely operated valves if needed at reduced RCS pressure to maintain 140 gpm. During normal operation, the letdown path is through the block orifice only. Boron concentration changes will require the use of the remotely operated control valve in parallel to obtain flow rates above 70 gpm. The spare purification demineralizer can be placed in service by remote positioning of the demineralizer isolation valves. Purification demineralizer 1-3 can be placed in service with one of the other purification demineralizers by remote positioning of the isolation valve. To place purification demineralizer 1-3 in service in series with one of the other purification demineralizers requires using manual valves. Diverting the letdown flow to the clean radioactive waste treatment system is accomplished by remote positioning of the three-way valve and the valves in the clean radioactive waste disposal system. The control valve in the injection line to the reactor coolant pump seals is automatically set by a flow controller to maintain the desired total flow rate to the seals.

The reactor coolant makeup control valve is automatically controlled by the pressurizer level controller. The makeup pumps are remotely controlled from the control room. Switch-over from one makeup filter to the other is remotely controlled from the control room. The purification demineralizer filter can be bypassed by remote operation from the control room. Operation of the seal return coolers requires manual positioning of the valves.

Coolant at the refueling boron concentration is supplied to the reactor coolant system for preoperational fill by using the boric acid pumps and the clean waste receiver transfer pumps or the demineralized water supply pumps. The fill line bypasses the makeup tank and makeup pumps and connects into the RC System through the normal makeup control valve. When the

fill operation is completed, the auxiliary fill line is secured; makeup and inventory control is then continued by operation of a makeup pump.

The makeup and purification system provides makeup to the reactor coolant system to replenish inventory lost due to a small rupture in the reactor coolant system pressure boundary. The makeup control valve senses a decrease in pressurizer level and positions itself to maintain level. A high flow alarm is associated with the increase in additional makeup. Following this alarm will be a low makeup tank level alarm. Additional makeup is provided by sources of reactor coolant grade water upstream of the makeup tank.

The Makeup and Purification System is monitored by the radiation monitoring system. This is used to monitor the reactor coolant for fission products due to failed fuel during operation.

The Makeup and Purification System is not assumed to operate to mitigate design basis accidents as presented in Chapter 15 of the USAR. However, in the unlikely event that all steam generator cooling capability is lost, i.e., a simultaneous loss of both Main Feed Pumps, both Auxiliary Feed Pumps and the Motor-Driven Feed Pump, core cooling will be accomplished by feed and bleed. This event is not a design basis accident. The Makeup Pumps are the only existing pumps which can pump against the normal Reactor Coolant System (RCS) pressure.

9.3.4.3 Safety Evaluation

9.3.4.3.1 Reliability Consideration

This system provides important functions for the normal operation of the unit. Redundant components and alternate flow paths have been provided to improve system reliability.

In addition to the letdown orifice, the system has two full-capacity control valves in parallel with the orifice. One of these control valves is manually operated and one is remotely operated.

Three mixed bed or cation purification demineralizers are provided. One demineralizer is normally in service as the “operational” demineralizer. One demineralizer is used to removal lithium (Li-7). One remaining demineralizer is in standby as a spare.

There are two makeup pumps, each capable of supplying the required reactor coolant pump seal and makeup flow. One is normally in operation while the other, kept on standby status, is used as needed.

There are two letdown coolers and two seal return coolers. Based on system requirements, one or both of each may be in operation. Similarly, there are two makeup filters and two seal injection filters provided.

The purification demineralizer filter is provided for use as required. Redundancy in these components is not necessary.

9.3.4.3.2 Malfunction Analysis

A malfunction analysis of the makeup and purification system is given in Table 9.3-9, it demonstrates that, in the event of any credible single active failure, the system can still meet its normal operating design requirements.

9.3.4.3.3 System Isolation

The letdown line and the reactor coolant pump controlled bleed-off line are outflow lines which penetrate the containment vessel. These lines contain electric motor-operated isolation valves inside the containment vessel and solenoid actuated air operated isolation valves outside the containment vessel which are automatically closed by a safety features actuation signal.

The injection lines to the reactor coolant pump seals are inflow lines penetrating the containment vessel. Each of the four seal injection lines contains a stop check valve inside the containment vessel and a solenoid actuated air operated valve outside the containment vessel. The solenoid actuated air operated valve outside the containment vessel is actuated by a safety features actuation signal and is designed to fail closed on loss of its air supply. With the presence of a safety feature actuation signal and a simultaneous failure of the air supply, the isolation valve will be closed by the accumulator air pressure. An air accumulator is provided to keep this valve open in the event of a failure in the instrument air supply system. This prevents unnecessary loss of seal injection flow and the possible reactor coolant pump damage which may result. If during normal operation, a failure of the air supply occurs, the isolation valve will remain open until commanded to close. The air accumulator provides additional air pressure to that trapped under the piston operator (air under piston to open isolation valve). Even if this air pressure should eventually be lost, the seal injection fluid pressure in the isolation valve body will hold the isolation valve open. No action is required by the control room operator to avoid reactor coolant pump damage because the isolation valve will remain open.

The reactor coolant pump can operate indefinitely with the loss of either seal injection water or component cooling water (pump cooling). However, the motor operation is limited by loss of component cooling water to approximately five minutes. The line providing makeup flow to the reactor coolant system does so via one of the high pressure injection lines. Isolation for the makeup line is accomplished by a stop check valve in the high pressure injection line inside the containment vessel and a motor-operated isolation valve outside the containment vessel.

Check valves in the discharge of each makeup pump provide further backup for containment vessel isolation. Also, the solenoid-operated isolation valves outside the containment vessel in the letdown line, and reactor coolant pump controlled bleedoff line, are designed to fail closed on loss of air supply.

9.3.4.3.4 Leakage Considerations

Design and installation of the components and piping in the makeup and purification system consider radioactive service. Except where flanged connections have been installed for ease of maintenance, the system is of all-welded construction. Principal valves have provisions for leak-off connections.

The consequences of a pipe failure in the makeup and purification system depend upon the location of the rupture. If the rupture were to occur between the reactor coolant loop and the first isolation valve or check valve, it would lead to uncontrolled loss of coolant from the reactor coolant system. This accident is evaluated in Chapter 6. If the rupture were to occur beyond the first isolation valve or outside the containment vessel the release of radioactivity would be limited by the small line sizes and by closing of the isolation or check valve. Leakage detection in the makeup and purification system is achieved by monitoring of the makeup tank level as a function of time.

Possible leakage of hydrogen from the makeup tank has been taken into consideration. The tank and connected piping were hydrostatically tested to demonstrate leaktightness during preoperational testing. Thereafter, the only likely source of leakage will be the safety valve, and it will be piped to the radioactive waste system.

Control of leakage for major loss of coolant accident is described in Chapter 6. The capability of the makeup and purification system to handle small breaks as required by NRC Criterion 33 is also discussed in Chapter 6.

9.3.4.4 Tests and Inspections

Active and passive components of the makeup and purification system will be examined periodically to determine their operating condition. Periodic visual inspections and preventive maintenance will be conducted according to sound maintenance practice.

9.3.4.5 Instrumentation Applications

9.3.4.5.1 Operational Limits

Alarms or interlocks are provided to limit variables or conditions of operation that might affect system safety. The variables of conditions of operation which are limited are as follows:

a. Makeup Tank Level and Pressure

   High and low water level in the makeup tank are alarmed at 86 and 55 inches, respectively. A loss of level in this tank could cause a loss of NPSH to the makeup pumps, thus causing loss of makeup flow which could have a serious effect on the reactor coolant system if it occurred during cooldown or when makeup for contraction of the reactor coolant system is required. A low level interlock with the three-way valve is set at 25 inches and signals the valve to switch from bleed operation to makeup operation to prevent draining of the tank. In addition, the suction of the makeup pumps switches automatically from the makeup tank to the borated water storage tank at a makeup tank level of 17 inches. If the makeup pump suction shift is not accomplished within 45 seconds of reaching the 17-inch level, the makeup pumps trip to prevent cavitation.

   In the event that the Control Room must be evacuated, the local makeup tank level is provided to allow appropriate operator actions to prevent makeup pump trip on low level.

   The makeup tank level is measured by redundant transmitters. The signals are transmitted to the control room for indication and alarm.

   Plant operating procedures call for the makeup tank pressure to be maintained at or above 15 psig for all modes of operation. A low pressure switch (PS MU 21-1) actuates an alarm in the main control room when makeup tank pressure falls to approximately 17 psig.

b. Letdown Line Temperature

   A high temperature alarm is set at 135°F and signals a high temperature in the letdown line downstream of the letdown coolers to protect the purification

demineralizers' resins. Hot reactor coolant, if allowed to pass through the demineralizers, could damage the demineralizer resins.

The letdown temperature is determined largely by component cooling water temperature and flow, while the letdown pressure is a function of the performance of the pressure breakdown equipment. Therefore, these two parameters are independent, and their relationship can vary widely.

Excessive temperature and pressure will be observed through alarms in the control room, an increasing amount of liquid waste generation and through increased leakage from the system.

c. Dilution Control

Initiation of the dilution cycle must be by the operator. Several safeguards are incorporated into the design to prevent inadvertent excessive dilution of the reactor coolant boric acid concentration. These safeguards are as follows:

1. The process of continuous deboration of the reactor coolant cannot start unless specific control rod groups are withdrawn to a certain point which allows for deboration. This control rod group position interlock through the integrated control system either permits or prohibits continuous dilution depending upon the control rod group position. Because of this interlock, the demineralized water makeup valve and the three-way valve can be operated simultaneously only when the control rod group is withdrawn to a preset position. The demineralized water makeup valve is automatically closed, and the three-way valve position is automatically changed when the rods have inserted to a preset position.

2. The dilution valves are interlocked so that the operator must preset the desired dilution batch size before initiating the dilution cycle. The dilution cycle will automatically terminate when the dilution flow has integrated to the preset batch size.

3. The operator is able to manually terminate the dilution cycle at any time.

9.3.4.5.2 Instrumentation Application

The instrumentation in the makeup and purification system provides measurements which are used to indicate, record, alarm, interlock and control process variables such as level and flow as follows:

a. The following process variables are measured and a signal is transmitted that provides indication in the control room.

1. Letdown flow.

2. Makeup pump discharge header pressure.

3. Makeup flow (low range and high range).

4. Makeup tank level.

5. Seal injection flow.

b. The following process variables are measured and a signal is transmitted that will actuate alarms and provide indication in the control room.

1. Seal injection filter differential pressure.

2. Makeup tank pressure.

3. Makeup flow (wide range).

4. Makeup filter differential pressure.

c. The letdown temperature is measured and signals are transmitted that will actuate alarms and provide indication in the control room. Temperature switches are provided at the discharge of the letdown coolers, and one is provided further downstream on the letdown line. The switches on the discharge of the coolers will close the inlet valves to the coolers upon detection of high temperatures.

d. The following process variables are measured and locally indicated.

1. Makeup pump discharge pressure.

2. Purification demineralizer filter differential pressure.

3. Makeup tank level

e. The RC pump seal bleed-off flow is measured and a signal is transmitted that will actuate alarms and provide indication in the control room.

f. The RC pump seal inlet flow is measured and a signal is transmitted that will actuate alarms and provide indication in the control room.

g. The following process control valves are manual/electric positioned from the control room.

1. Letdown flow control valve.

2. Makeup system three-way feed and bleed control valve.

h. Signals from the following process variables are transmitted to the plant computer for indication and/or alarm.

1. Letdown temperature.

2. Letdown flow.

3. Letdown pressure.

4. Makeup tank temperature.

5. Makeup tank level.

6. Letdown radiation.

7. DELETED

8. Makeup tank pressure.

9. Makeup pump discharge pressure.

10. Makeup pump discharge temperature.

11. Seal injection flow.

12. Makeup flow.

13. Seal return flow.

i. The makeup system feed and bleed controls (batch controller) is a device that measures the amount of boric acid or demineralized water added to the reactor coolant system. The feed and bleed controls (batch controller) will automatically terminate the addition when the quantity preselected by the operator is reached.

9.3.5 Decay Heat Removal System

9.3.5.1 Design Bases

The decay heat removal system removes decay heat from the core and sensible heat from the reactor coolant system during the later stages of cooldown. The system also provides auxiliary spray to the pressurizer for complete depressurization, maintains the reactor coolant temperature during refueling, and provides a means for filling and partial draining of the refueling canal. In the event of a loss-of-coolant accident, the system injects borated water into the reactor vessel for long-term emergency cooling. The emergency functions of this system are described in Chapter 6.

9.3.5.2 System Description

9.3.5.2.1 General

The decay heat removal system is shown schematically in Figure 6.3-2A. Tables 9.3-10 and 9.3-11 list the system performance and component design characteristics.

Decay Heat Pumps:

Two decay heat pumps are arranged in parallel and are designed for continuous operation during the period required for removal of decay heat for refueling. The design flow rate is that required to cool the reactor coolant system from 280°F to 140°F in approximately 26 hours. Prior to bringing the decay heat removal system on line, the steam generators are designed to cool the reactor coolant system from operating temperature to 280°F and a pressure below 260 psig in a 6-hour period.

Additional performance data for the decay heat pumps are given in Chapter 6.

Decay Heat Removal Coolers:

The decay heat removal coolers remove the decay heat from the circulated reactor coolant during a routine shutdown. Operating both coolers provides the design capability to reduce the reactor coolant temperature to 140°F in approximately 26 hours after the reactor coolant temperature reaches 280°F. Both coolers are available for emergency operation.

Additional performance data for the decay heat removal coolers are given in Chapter 6.

Borated Water Storage Tank:

The Borated Water Storage Tank (BWST) is located outside the containment vessel and the auxiliary building. It contains a minimum of 2600 ppm boron in solution and is used both for emergency core injection and filling the refueling canal during refueling. The BWST also supplies borated water for emergency cooling to the containment spray system, decay heat removal (LPI) system, and high pressure injection system. It also supplies makeup water to the spent fuel pool cooling system and can serve as source for the makeup pumps.

The BWST is provided with a heater located in the auxiliary building which normally maintains the borated water at a temperature >50°F. The minimum temperature of the BWST is 35°F. The outdoor portion of the BWST piping is installed inside the pipe trench 10 feet below ground level and is covered with a steel plate; the trench depth is well below the frostline, and, therefore, no tracing is required.

9.3.5.2.2 Mode of Operation

Two pumps and two coolers perform the decay heat cooling function. After the steam generators have reduced the reactor coolant temperature to less than 280°F, decay heat cooling is initiated. Normally one pump takes suction from the reactor coolant outlet line and discharges through a cooler into the reactor vessel. The other pump is normally aligned to the BWST. To control the decay heat return temperature a portion of the reactor coolant must be bypassed around the decay heat coolers. The equipment utilized for decay heat cooling is also used for low pressure injection during loss-of-coolant accident conditions as described in Chapter 6.

The time required to cool the reactor coolant from 280°F to 140°F starting six hours after shutdown from an initial power level of 2817 MW thermal, using one decay heat pump and cooler is approximately 175 hours. The pump and cooler requirements for maintaining a refueling temperature of 140°F are dependent on the time after shutdown. One pump and cooler can maintain a refueling temperature of 140°F or less after 197 hours from shutdown. For the time period between 20 and 197 hours after shutdown, two pumps and two coolers are required to maintain a temperature of 140°F or less.

During refueling, the decay heat from the reactor is rejected to the decay heat removal coolers in the same manner as it is during cooldown to 140°F. At the beginning of the refueling period, both coolers and both pumps are required to maintain 140°F in the core and refueling canal. Later, as core decay heat decreases, one cooler and pump can maintain the required 140°F.

The fuel canal is normally filled using one of the decay heat removal pumps while the other decay heat removal pump is in service for decay heat removal from the core. In this mode of

operation, the suction of one of the decay heat pumps is aligned to the borated water storage tank, with the pump discharge aligned to the refueling canal.

Alternatively, the refueling canal can be filled by switching the suction of the decay heat pumps from the reactor outlet to the borated water storage tank. As the borated water passes through the reactor vessel into the refueling canal it absorbs the decay heat from the core. When the refueling canal is filled, suction to the pumps is switched back to the reactor outlet pipe. The refueling canal is drained after refueling by switching the discharge of one of the pumps from the reactor injection nozzle to the borated water storage tank. The other pump will continue the recirculation mode of decay heat removal.

After the decay heat removal system has been brought on line, the reactor coolant pumps are secured. This results in the loss of depressurization due to the loss of normal spray to the pressurizer. An auxiliary spray connection from the decay heat removal system to the pressurizer is provided in order to continue reactor coolant system depressurization. The auxiliary spray connection originates downstream of the decay heat removal cooler, penetrates the containment vessel, and connects to the normal spray line between the pressurizer and the normal spray line throttle valve. A remote operated throttle valve is provided in the auxiliary spray line for flow control.

9.3.5.3 Safety Evaluation

9.3.5.3.1 Reliability Considerations

Since the equipment is designed to perform both normal and emergency functions, separate and redundant flow paths and equipment are provided to prevent a single component failure from reducing the system performance below a safe level.

For normal operation of the decay heat removal system, sufficient redundancy is obtained by having a common suction line from the reactor coolant loop through the containment vessel. The common suction line then splits into two separate loops each containing one decay heat pump and one decay heat removal cooler and separate reactor vessel injection lines.

For emergency operation of the decay heat removal system, reliability considerations are discussed in Chapter 6.

9.3.5.3.2 Malfunction Analysis

Failures and malfunctions in the decay heat removal system in conjunction with a loss-of-coolant accident are discussed in Chapter 6. This analysis is also valid for the normal decay heat removal mode since essentially all the components are common for both the normal and the emergency operations.

During cooldown of the reactor coolant from 280°F to 140°F, the pressurizer is normally cooled by spray from the decay heat removal system. The absence of spray to the pressurizer does not render the decay heat removal unable to cool the primary system, and the operator may periodically duplicate the depressurizing effect of the spray by opening the pressurizer vent line and discharging steam to the quench tank. Although cooldown may be slower than normal, the continued use of the decay heat removal system and the pressurizer vent line on the pressurizer will provide a safe and orderly cooldown of the primary system in the complete absence of pressurizer spray.

During the winter, if the BWST heater fails to operate, the borated water temperature will be decreased at a rate of 22°F per day, assuming the worst-case environmental conditions of 0°F ambient temperature and 30 mph wind velocity. Temperature indicators are provided so that corrective action can be initiated in the event of a tank heater failure.

Rupture of the BWST and/or total release of borated water from the BWST is analyzed in Chapter 3. An analysis of the radionuclide content of the BWST is also presented in Chapter 3.

9.3.5.3.3 System Isolation

The decay heat removal system is connected to the reactor outlet line on the suction side and to the reactor vessel, through the core flooding lines, on the discharge side. The system is isolated from the containment vessel on the suction side by two normally closed, electric motor-operated stop valves in series located inside the containment vessel and two normally closed, electric motor-operated stop valves, in parallel, one in each decay heat pump suction line, located outside the containment vessel. On the discharge side, each line can be isolated from the containment vessel by closing an electric motor-operated stop valve outside the containment vessel. There are also two check valves inside the containment vessel. In the event of a loss-of-coolant accident the valves between the reactor vessel and the suction of the pumps remain closed until required for long-term boration control.

The decay heat removal system is also connected to the reactor coolant system via the auxiliary spray line to the pressurizer. This line contains an electric motor-operated throttle valve outside the containment vessel and an electric motor-operated stop valve and one check valve inside the containment vessel. These two valves are normally closed and are open only at lower reactor coolant pressures to provide auxiliary spray.

9.3.5.3.4 Leakage Considerations

During reactor power operation, the decay heat removal system is idle. Under loss-of-coolant accident conditions, fission products may be recirculated in the coolant through the exterior piping system. Potential leaks have been evaluated to obtain the total radiation dose due to leakage from this system. The evaluation is discussed in Chapter 15.

9.3.5.4 Tests and Inspections

The system has provisions for periodic flow testing. This provision is explained in Chapter 6.

Each component of the system is inspected according to the applicable codes or standards noted in Table 9.3-11.

9.3.5.5 Instrumentation Applications

9.3.5.5.1 Operational Limits

Alarms or interlocks are provided to limit variables or conditions of operation that might affect system or station safety. These variables or conditions of operation are discussed below.

Decay Heat Removal Flow Rate

Low flow from the decay heat pumps during the decay heat removal mode of operation is alarmed to signify a reduction or stoppage of flow and cooling of the core. The alarm is

interlocked with the corresponding decay heat removal pump switchgear to prevent initiation of the low flow alarm when the pump is not in use. The alarm may be defeated when the plant is shutdown to provide indication of pump breaker opening. The annunciator low flow alarm is set to warn the operators of approaching low flow conditions during normal decay heat removal operations and may be reduced below the normal setpoint for low RCS level operations.

High temperature at the suction of the decay heat pump is alarmed to warn the operator that the temperature is above normal.

Decay Heat Removal Cooler Exit Temperature

High temperature from the decay heat removal cooler is alarmed to signal a loss of cooling capability in the respective cooler. The annunciator high temperature alarm setpoint is varied based on RCS temperature.

Decay Heat Removal System Overpressurization Protection

Both valves from the reactor coolant system in the suction line to the decay heat removal pumps are interlocked with the reactor coolant system (hot leg) pressure as described in USAR Section 7.6.1.1.2. This interlock prevents opening of the valves and automatically closes on RC pressurization and prevents the resulting overpressurization of the decay heat system piping while the reactor coolant system pressure is above the decay heat removal system design pressure. During shutdown when decay heat removal system operation is initiated, the suction valves are opened prior to starting the decay heat removal pumps. An interlock is provided to trip the pressurizer heaters if primary system pressure reaches the setpoints of Subsection 7.6.1.1.2.

A relief valve in the suction line to the decay heat pumps, PSV 4849 (see Figure 6.3-2A), has been sized to pass 1800 gpm at a set pressure of 320 psig. The flow rate is based on the maximum developed runout flow (900 gmp per pump) with both high pressure injection (HPI) pumps running simultaneously. This flow rate is considered to cause the worst credible pressure transient. The opening of a Core Flooding Tank isolation valve was not considered because power is removed from the valve once it is closed upon plant cooldown and depressurization as described in Subsection 6.3.2.15. Other postulated occurrences ,makeup control valve failing open, loss of DHR system cooling, all pressurizer heaters energizing, do not produce pressure excursion as a severe as that produced by the two HPI pumps. Although the pressurizer, by procedure, cannot be solid, for the purpose of analysis it was considered to go solid during the transient. The relief valve installed is a Seismic Class I Nuclear Class 2 bellows type of safety-relief valve. It should be noted that the postulation of both HPI pumps starting during DHR system operation, is made only for the purpose of sizing PSV 4849. The possibility of this event occurring due to either a single operator error or a single spurious signal is precluded by the design of the safety features actuation system. See Section 7.3 for details.

Containment Vessel Emergency Sump Suction Line Overpressurization Protection

The motor-operated valve in the decay heat removal pump suction line from the BWST/containment vessel emergency sump (DH2733, DH2734) is interlocked with the corresponding motor-operated valve for the decay heat removal line to the pump (DH1517, DH1518). This interlock prevents opening of the RCS/Decay Heat Removal Suction valve when the sump suction line valve is not fully closed. This in turn prevents inadvertent overpressurization of the sump suction line during cooldown.

Decay Heat Removal Isolation Valve Setpoints

The interlock setpoints for decay heat removal isolation valves DH11 and DH12 are described in Section 7.6.1.1.2.

9.3.6 Chemical Addition System

9.3.6.1 Design Bases

Chemical addition operations are required to alter the concentration of various chemicals in the reactor coolant and auxiliary systems. The chemical addition system is designed to add boric acid to the reactor coolant system for reactivity control, lithium hydroxide for pH control, zinc for source term reduction, and hydrazine for oxygen control. The system may also be used to add hydrogen peroxide to the Reactor Coolant to achieve forced oxidation after shutdown for source term reduction. The system also provides boric acid for other station components, and is sized to be able to add sufficient boric acid to maintain the core 1% Dk/k subcritical at any time during life.

9.3.6.2 System Description

9.3.6.2.1 General

The chemical addition system is shown schematically in Figure 9.3-18. Table 9.3-12 lists historical system performance data. The Technical Requirements Manual shows the current cycle's system performance requirements with respect to minimum boron concentration and volume for Modes 1-4. Table 9.3-13 lists component design characteristics of the chemical addition system. The equipment in the chemical addition system is designed to the applicable codes or standards as noted in Table 9.3-13.

Boric Acid Mix Tank

The boric acid mix tank is used to mix and supply concentrated boric acid to the boric acid addition tanks. The tank is supplied with an agitator for mixing and an electrical heater to maintain the temperature for boric acid solubility. The boric acid mix tank supplies the boric acid addition tank by gravity flow.

Boric Acid Addition Tanks

Two boric acid addition tanks are provided for storage of concentrated boric acid solution. The volume of the tanks provides sufficient concentrated boric acid solution to increase the reactor coolant system boron concentration to that required for cold shutdown with no xenon. Electric heaters in the tanks, heating the rooms in which the tanks are located, and electric heat tracing of the transfer lines maintain the fluid temperature above that required to ensure solubility of the boric acid. The tanks store, for eventual use, boric acid from the boric acid mix tank and the concentrate storage tank.

Boric Acid Pumps

Two centrifugal boric acid pumps are provided to facilitate transfer of the concentrated boric acid solution from the boric acid addition tanks to the borated water storage tank, makeup tank, clean waste receiver tanks or the spent fuel storage pool. The pumps are sized so that when

both are operating, one complete charge of concentrated boric acid solution from the boric acid addition tanks may be injected into the reactor coolant system in 12 hours.

Lithium Hydroxide Mix Tank

This tank contains an agitator and is used for mixing and storing lithium hydroxide for addition to the makeup and purification system for reactor coolant pH control. This tank is sized such that the reactor coolant Li-7 concentration can be maintained within proper limits. This tank can also be used to add hydrogen peroxide to the Reactor Coolant System after shutdown to induce forced oxidation.

Lithium Hydroxide Pump

The lithium hydroxide pump is a diaphragm, variable stroke pump which transfers lithium hydroxide from the lithium hydroxide mix tank to the letdown line upstream of the makeup filters in the makeup and purification system. This pump can also be used to transfer hydrogen peroxide from the lithium hydroxide mix tank to the makeup and purification system. The pump's power is supplied from a lighting distribution panel.

Hydrazine Pump

The hydrazine pump is a diaphragm, variable stroke pump which transfers hydrazine to the letdown line upstream of the makeup filters in the makeup and purification system. This pump can also be used to transfer hydrogen peroxide from the lithium hydroxide mix tank to the makeup and purification system. The pump’s power is supplied from the lighting distribution panel.

Zinc Injection Skid

The zinc injection skid contains equipment to inject a zinc acetate solution to the Reactor Coolant System via the Makeup and Purification System. Major components on the skid are:

- Zinc Injection Mix Tank (T99)

  The purpose of the Zinc Injection Mix Tank is to mix and store zinc acetate for addition to the Makeup and Purification System. The tank is sized to support zinc injection at the maximum required flow rate for one week without needing to refill the tank.

- Zinc Injection Pumps (P295-1 and P295-2)

  The purpose of the Zinc Injection Pumps is to transfer zinc acetate from the Zinc Injection Mix Tank to the letdown line upstream of the makeup filters in the Makeup and Purification System. The Zinc Injection Pumps are positive displacement pumps designed to meter known quantities of chemical into systems of varying pressures.

- Zinc Injection Mix Tank Mixer (S86)

  The purpose of the Zinc Injection Mix Tank Mixer is to thoroughly agitate the contents to produce a uniform solution.

- Zinc Injection Control Panel (C3741)

  The purpose of the Zinc Injection Control Panel is to provide a connection for the power source of the skid, to distribute power the Zinc Injection Mix Tank Mixer and the Zinc Injection Pumps, and to provide local instrumentation and controls. The control panel has a pump selector switch and pump on/off switches with indicator lights, pump leak detection system local indication, Zinc Injection Mix Tank low and low-low liquid level indication, mixer on/off switch indicator lights, and an emergency skid shutoff button.

Heating Systems

To maintain the solubility of concentrated boric acid contained in the components of this system, several different types of heating systems are utilized. These are:

a. Internal Tank Heaters: The boric acid mix tank, and boric acid addition tanks all have full capacity, internal heating systems.

b. Space Heaters: The boric acid addition tanks, boric acid pumps and a portion of the associated interconnecting piping are located in a room which is equipped with two independent, full capacity infrared space heating systems. This space heating system has a primary and secondary power supply.

c. Heater Tracing: Pumps, piping, and valves which normally handle concentrated boric acid and which are not located in the room with the space heaters, are equipped with two independent, full capacity heat tracing systems. Concentrated boric acid is infrequently handled by non-heat traced piping when additions are made to one of the clean waste receiver tanks. The piping and valves which do not have a heat tracing system are flushed with demineralized water after the transfer of concentrated boric acid.

Each of the full capacity heating systems is capable of maintaining the temperature of the liquid in the components it services at 15°F above the crystallization temperature of 7% by wt. boric acid. Therefore, the boric acid mix and addition tanks are typically maintained at an average temperature of 110°F. Additionally, sufficient margin exists at the Technical Requirements Manual upper limit of 7.5 weight percent (13,125 ppm) and the minimum temperature of 105°F to prevent crystallization. Where two, or more, full capacity systems are provided for the same component, one is normally used while the other(s) serves as a standby.

9.3.6.2.2 Mode of Operation

The chemical addition system delivers the necessary chemicals to other systems as required. Concentrated boric acid is supplied to the spent fuel pool, borated water storage tank, clean waste receiver tanks and makeup tank for leakage or to change the concentration of boric acid in the associated systems.

Lithium hydroxide is added to the makeup tank for reactor coolant pH control. Proper reactor coolant pH is obtained by maintaining the lithium concentration as described in Table 9.3-4.

Hydrazine may be added to the makeup tank for reactor coolant oxygen control during subcritical, cooldown, or shutdown conditions. Due to the temperature dependence of the hydrazine-oxygen reaction, hydrazine is most effective between 200-400°F, with decomposition occurring above 400°F. Proper reactor coolant oxygen control is obtained by maintaining the

hydrazine concentration at 0.1-1.0 ppm N_2H_4 during those stages of operation in which hydrazine is to be used.

Hydrogen peroxide may be added to the Reactor Coolant System (RCS) to induce forced oxidation when RCS cold leg temperature is less than 180°F with at least one reactor coolant pump in operation; or when the RCS is on decay heat cooling with no reactor coolant pumps in operation and the decay heat cooler outlet temperature is ≤140°F, thus allowing the use of purification demineralizers to clean up ionic species liberated in the Reactor Coolant System.

Zinc Acetate is added to the makeup tank to reduce the radiation source term and for mitigation of Primary Water Stress Corrosion Cracking (PWSCC) of Alloy 600 components.

9.3.6.3 Safety Evaluation

9.3.6.3.1 Reliability Considerations

The chemical addition system is not credited for mitigation of any USAR Chapter 15 accidents. The boric acid addition system is credited for providing safe shutdown capability in the event the Borated Water Storage Tank is lost by a tornado missile. Independent essentially powered boric acid pumps and independent boric acid addition lines are provided to guard against a single component failure. Boric acid is available for boration from the two boric acid addition tanks as well as directly from the concentrate storage tank. To facilitate the handling of concentrated boric acid, flush connections are provided that allow the lines carrying it to be cleaned when necessary.

The lithium hydroxide and hydrazine pumps are interconnected such that, if one of them fails, the other can be used in it place.

The zinc injection skid contains two injection pumps which are capable of independent operation and can be mechanically and electrically removed from service without interrupting zinc injection.

The concentrated boric acid handled by this system can be maintained above its crystallization temperature by any one of at least two independent, full capacity heating systems. The boric acid mix tank has only one full capacity heating system available to maintain it above the crystallization temperature. For piping systems not heat traced, refer to Section 9.3.6.2.1.c. Even if boric acid solidification did make the system inoperable, the reactor can still be shut down safely. In such a situation, sufficient boron could be added to the primary system by compensating for the temperature contraction of the reactor coolant with water, via the high pressure injection pumps, from the borated water storage tank.

9.3.6.3.2 Malfunction Analysis

A malfunction analysis of the Chemical Addition (CA) system is given in Table 9.3-14 to demonstrate that the system has sufficient redundancy to maintain the normal operating design requirements with any credible single active failure.

Following is a discussion of the effects of system malfunction on safety-related equipment.

Makeup of borated water from the Borated Water Storage Tank (BWST) may be required during cooldown after a malfunction of the CA system. The reactor coolant system boron concentration requirements for shutdown are as follows for hypothetical worst case conditions

(minimum full power boron concentration, maximum shutdown boron concentration), maximum worth control rod stuck out.

Hot Full Power (582°F) Concentration		Hot Standby (532°F) Requirement	Hot Shutdown (280°F) Requirement	Cold Shutdown (70°F) Requirement
BOL	1800 ppm	2100 ppm	2250 ppm	2300 ppm
EOL	5 ppm	300 ppm	600 ppm	700 ppm

Injection of the contraction volume from the BWST through the high pressure injection system raises the reactor coolant boron concentration as follows:

	Hot Standby (582°F - 532°F)	Hot Shutdown (582°F - 280°F)	Cold Shutdown (582°F - 70°F)
BOL	1824 ppm	1961 ppm	2005 ppm
EOL	83 ppm	528 ppm	671 ppm

Injection of borated water through the high pressure injection system may begin after the reactor coolant system pressure has been reduced to 1600 psia. Cooldown from hot zero power may commence when this pressure is reached. After the contraction volume has been injected, the reactor coolant may be borated to the final shutdown boron concentration by feed and bleed. Maximum total boric acid volumes for the hot and cold shutdown procedures are as follows (2600 ppm boric acid solution):

	Hot Standby (582°F - 532°F)	Hot Shutdown (582°F - 280°F)	Cold Shutdown (582°F - 70°F)
BOL	31,027 gallons	63,326 gallons	78,025 gallons
EOL	8,323 gallons	19,114 gallons	22,268 gallons

The two 103,000 gallon clean waste receiver tanks will accept the required reactor coolant bleed.

The above concentration and volume requirements take no credit for xenon and assume the maximum worth control rod is stuck out. These values are less than the required minimum BWST volume of 500,100 gallons in Modes 1-4; therefore, the BWST provides adequate CA system backup capability. For post-LOCA conditions, with 50% minimum tripped rod worth, the required minimum volume to be injected from the BWST for reactivity control is assumed to be 360,000 gallons, which is sufficient to borate to a shutdown margin of -1% $\Delta k/k$. This value is also less than the required minimum BWST volume.

Other chemicals added by the CA system are hydrazine, zinc, and lithium hydroxide, which are used to inhibit corrosion, and source term reduction in the reactor coolant system and are not required for safe shutdown. Hydrogen peroxide can also be added by the CA system to induce forced oxidation of the Reactor Coolant System after shutdown for source term reduction. Hydrogen peroxide is not required for safe shutdown.

It is concluded that there are no safety-related functions which would be adversely affected by failures in the CA system.

9.3.6.3.3 System Isolation

No containment vessel isolation is required of this system since its boundaries do not penetrate the containment vessel.

9.3.6.3.4 Leakage Considerations

The system delivers additives to the spent fuel storage pool, clean waste receiver tanks, makeup tank and borated water storage tanks. Backflow from the tanks to the boric acid pumps is prevented by check valves and normally-closed shutoff valves between them. Additive lines to the spent fuel storage pool contain check valves, thereby preventing backflow.

Backflow to the zinc injection system is prevented by check valves and the positive displacement pumps.

9.3.6.4 Tests and Inspections

Active and passive components of the chemical addition system will be examined periodically to determine the operating condition. Periodic visual inspections and preventive maintenance will be conducted according to sound maintenance practice.

9.3.6.5 Instrumentation

9.3.6.5.1 Operational Limits

There are certain conditions of operation which must be limited to ensure system operation. These conditions are as follows:

a. The boric acid mix tank, addition tanks, and associated transfer piping must be maintained at a temperature above the crystallization temperature of the concentrated boric acid solution that is consistent with the Technical Requirements Manual. The required minimum temperature is maintained by use of infrared space heaters and/or level and temperature interlocked electric heaters in the tanks and/or electric heat tracing of the associated piping.

b. During any stage of core life, the amount of concentrated boric acid solution maintained in the boric acid addition tanks must be sufficient to borate the reactor coolant system to a 1% Ak/k Shutdown Margin at hot shutdown condition with maximum worth control rod stuck out and no xenon. The boric acid addition tanks are supplied with level indication and low level alarms in the control room which permit the operator to monitor the volume of concentrated boric acid in storage.

9.3.6.5.2 Applications

The instrumentation of the chemical addition system provides measurements which are used to indicate, record, alarm, interlock and control process variables such as level and flow as follows:

a. The boric acid mix tank temperature is measured, indicated locally and a signal is transmitted that will actuate alarms and provide indication on a panel not located in

the control room. A signal is also provided for heater control to maintain fluid temperature within a predetermined range.

b. The boric acid mix tank level is measured and indicated locally and a signal is transmitted that will actuate high or low alarms in the Control Room. An interlock signal is also provided to de-energize the heaters on low level.

c. The boric acid addition tank temperature is measured and a signal is transmitted that will actuate high or low alarms and provide indication both in the control room and on a panel located outside the control room. A signal is also provided for heater control to maintain fluid temperature within a predetermined range.

d. The boric acid addition tank level is measured, and a signal is transmitted that will actuate alarms and provide indication both in the control room and on a panel located outside the control room. Two interlocks are provided, one to de-energize the heaters on low level and the other to shut off the boric acid pump on low-low level.

e. The flow control valve on the primary boric acid addition line is able to be positioned from the control room.

f. The following process variables are measured and locally indicated:

1. Hydrazine pump discharge pressure
2. Hydrazine pump quantity
3. Lithium hydroxide pump discharge pressure
4. Lithium hydroxide pump quantity
5. Lithium hydroxide mix tank level
6. Boric acid pump discharge pressure
7. Boric acid emergency flow
8. Differential pressure across boric acid pump suction strainer
9. Zinc injection pump operation
10. Injection pump diaphragm failure
11. Zinc injection mix tank tank liquid level
12. Zinc injection pump discharge pressure

g. The flow in the primary boric acid addition line is measured and a signal transmitted that will provide indication in the control room.

h. Signals from the following process variables are transmitted to the plant computer for indication and/or alarm:

1. Temperature of room containing boric acid addition tanks and pumps
2. Boric acid addition tank level
3. Boric acid addition tank temperature
4. Temperature in boric acid addition lines
5. Differential pressure across boric acid pump suction strainer
6. Primary boric acid addition line flow

TABLE 9.3-1

Safety Related Air-Operated
Valves Required for Safe Shutdown

System	Valve Number	Safe Position	Function
Steam generator secondary water sample line	HV-607	Closed	Containment isolation
Containment air cooler service water outlet**	TV-1356 TV-1357 TV-1358	Open Open Open	Maintains service water through the coolers.
Letdown line to purification demineralizer	HV-MU3	Closed	Containment isolation
Containment vessel equipment vent header	HV-1719A HV-1719B	Closed Closed	Containment isolation Containment isolation
Steam generator secondary water line	HV-598	Closed	Containment isolation
Demineralizer water supply line	HV-6831A HV-6831B	Closed	Containment isolation
Reactor coolant system drain line to RC drain tank	HV-1773A HV-1773B	Closed	Containment isolation
Containment vessel purge inlet line	HV-5005 HV-5006	Closed Closed	Containment isolation Containment isolation
Containment vessel purge outlet line	HV-5007 HV-5008	Closed Closed	Containment isolation Containment isolation
Main steam line*	FV-100 FV-101	Closed Closed	Containment isolation Containment isolation
Main Steam to Condenser Warmup Valves	HV-375 HV-394	Fail Closed Fail Closed	Containment isolation Containment isolation
MSIV Bypass Line	HV-100-1 HV-101-1	Fail Closed Fail Closed	Containment isolation Containment isolation
Main Steam to Auxiliary Feed Pumps	HV-5889A HV-5889B	Fail Open Fail Open	Ensures Main Steam to Auxiliary Feed Pumps
Atmospheric Vent Valves	PVICS-11A PVICS-11B	Fail Closed Fail Closed	Containment isolation Containment isolation
Pressurizer quench tank circulation inlet line	HV-232	Closed	Containment isolation
Service air supply line	HV-2010	Closed	Containment isolation

TABLE 9.3-1 (Continued)

Safety Related Air-Operated
Valves Required for Safe Shutdown

System	Valve Number	Safe Position	Function
Instrument air supply line	HV-2011	Closed	Containment isolation
Core flooding tank fill and nitrogen supply lines	HV-1541	Closed	Containment isolation
	HV-1544	Closed	Containment isolation
Core flooding tank sample lines	HV-1545	Closed	Containment isolation
Core flooding tank vent line	HV-1542	Closed	Containment isolation
Pressurizer quench tank circulating outlet line	HV-229A	Closed	Containment isolation
	HV-229B		
Reactor coolant pump seal water supply	HV-MU66A	Closed	Containment isolation
	HV-MU66B	Closed	Containment isolation
	HV-MU66C	Closed	Containment isolation
	HV-MU66D	Closed	Containment isolation
Reactor coolant pump seal water return	HV-MU38	Closed	Containment isolation
Pressurizer quench tank sample line	HV-235A	Closed	Containment isolation
	HV-235B		
Nitrogen supply to pressurizer quench tank	HV-236	Closed	Containment isolation
Ventilation of the control room	HV-5301E	Closed	Isolates the control room from outside
	HV-5311E	Closed	
	HV-5361B		
	HV-5362B		
	HV-5301A		
	HV-5311A		
	HV-5301B		
	HV-5311B		
	HV-5301C		
	HV-5311C		
	HV-5361A		

TABLE 9.3-1 (Continued)

Safety Related Air-Operated
Valves Required for Safe Shutdown

System	Valve Number	Safe Position	Function
Ventilation of the Control room	HV-5362A HV-5301D HV-5311D HV-5301F HV-5311F HV-5301G HV-5311G HV-5301H HV-5311H	Closed	Isolates the control room from outside.
Decay heat removal system	HV-DH14A HV-DH14B HV-DH13A HV-DH13B	Fail Open Fail Open Fail Closed Fail Closed	Maintains the operation of decay heat removal system. Closes the bypass on decay heat exchanger.
Component cooling system	HV-1469 HV-1471 HV-1467 HV-1474 HV-1495 HV-1460	Fail Open Fail Open Fail Open Fail Open Fail Closed Fail Closed	Maintains flow through the decay heat removal and diesel generator coolers. Isolates non-essential load on the system.
Service water system	TV-1424 TV-1429 TV-1434	Fail Open Fail Open Fail Open	Maintains service water through the component cooling heat exchanger.

* On loss of air, the open position of these valves can be maintained by using nitrogen from the individual nitrogen bottles. Each bottle holds enough gas to maintain the open position, if required, for five days.

** Each of these valves is provided with a backup safety related nitrogen tank, sized to support three valve strokes and to maintain the valve closed for 30 days.

TABLE 9.3-3

Failure Analysis – Process Sampling System

Component	Failure	Comments and Consequences
Pressurizer Samples Liquid or Vapor Space	Electrically operated isolation valve inside reactor containment vessel fails to close.	The isolation valves operated by the Safety Actuation System satisfy the Single Failure criteria, the electrically operated isolation valve external to the containment vessel will close.
Steam Generator, Secondary Side, water and steam	Isolation valve external to the containment vessel fails to close.	Sample line is not connected directly to the reactor coolant system. The steam generator provides the first barrier.
Core Flooding Tank	Electrically operated isolation valve inside the containment vessel fails to close	Sample line is not connected directly to the reactor coolant system; core flooding line check valves provide first boundary. Pneumatically operated isolation valve external to the containment vessel will close.
Pressurizer Quench Tank Vapor Space	Electrically operated isolation valve inside the containment fails to close.	The electrically operated isolation valve external to the containment will close.
Sample line from any of the above components	Line breakage, inside reactor containment vessel, downstream of any electrically operated isolation valve.	During a loss-of-coolant accident all external isolation valves included in the Safety Actuation System will be closed.

TABLE 9.3-4

Reactor Coolant Quality*

Parameter	Value
Boron, ppm	Appendix 4B
Lithium as ^{7}Li, ppm	5.0 ppm(Note 1)
pH at 582°F	6.9 – 7.4
Dissolved 02 (max.), ppm	0.10 (Limit not applicable with $T_{ave} \leq 250°F$)
C1 (max.), ppm	0.05
H_2, std cc/kg water	25 – 50
F (max.), ppm	0.050
Hydrazine as N_2H_4, ppm	Critical – not applicable; Subcritical (less than 200°F) – as required to control O_2
SO_4 (max.), ppm	0.050
Fe (membrane)	(used to monitor crud in RCS, diagnostic only)
Conductivity	(diagnostic only)
Total Dissolved Gas, STD cc/Kg water	100
Zinc (max), ppb**	10

* Values were derived following EPRI PWR Primary Water Chemistry Guidelines and provide an environment that is compatible with reactor coolant materials. Chemistry limiting conditions for operation are contained in the Technical Requirements Manual.

** Zinc limit is a steady state limit. The EPRI PWR Primary Water Chemistry Guideline does not address the zinc limit.

Note 1: The operating cycle is initiated with a maximum lithium concentration of 6.4 ppm at zero (0) Effective Full Power Days (EFPD) and 5.0 ppm at 4 EFPD. After 4 EFPD 300°C "at temperature" pH transitions to 7.2 without exceeding 5.0 ppm lithium. When 300°C "at temperature" pH of 7.2 is achieved, the Li/B ratio is controlled to maintain pH at 7.2. The minimum 300°C "at temperature" shall be greater than 7.0 whenever nuclear heat is produced and the reactor is critical.

TABLE 9.3-5

Steam Generator Feedwater Quality*

pH at 25°C	≥9.3[(1)]
Dissolved O_2 (max.), ppm	0.003
SiO_2 (max.), ppm	0.01
Fe (membrane) (max.), ppm	0.01
Cu (max.), ppm	0.001
Sodium (max.), ppm	0.003
Chloride (max.), ppm	0.005
Fluoride (max.), ppm	0.005
Lead, ppm	<0.001
Sulfate, (max.), ppm	0.003
Iron, (total) (max.), ppm	0.005
Corrected Cation conductivity (max.), μmho/cm (due to inorganic anions)	0.2
Hydrazine (N_2H_4), ppb, and/or an amine to ensure a reasonable residual in the event of an oxygen in leakage transient	≥20[(2)]

* Values were derived for Steady State operation following EPRI PWR Secondary Water Chemistry Guidelines.

(1) No upper limit is given because pH is regulated to maintain an optimum balance between corrosion control and containment minimization.

(2) Applicable when N_2H_4 is used for the oxygen scavenger. No upper limit is given as elevated concentrations may be used if additional corrosion protection is required.

TABLE 9.3-6 (Note 1)

Floor Drain System

Room Nos.	Equipment Arrangement Fig. No.	Piping or Floor Drainage Fig. No.	Source of Flooding, Effect, Precautions, and Detection
601, 602 (Note 2)	3.6-2	9.3-8	See Subsection 3.6.2
603	3.6-2	9.3-8	See Subsection 3.6.2
500, 501, 515	3.6-6	9.3-8	See Subsection 3.6.2
427	3.6-7	9.3-9	See Subsection 3.6.2
404	3.6-7	9.3-9	See Subsection 3.6.2
314	3.6-3	9.3-10	See Subsection 3.6.2
304	3.6-3	9.3-10	See Subsection 3.6.2
303	3.6-3	9.3-10	See Subsection 3.6.2
318, 319	3.6-3	9.3-10	See Subsection 3.6.2
320A, 321A			
313	3.6-3	9.3-10	See Subsection 3.6.2
236	3.6-4	9.3-11	See Subsection 3.6.2
237, 238	3.6-4	9.3-11	See Subsection 3.6.2
208	3.6-4	9.3-11	See Subsection 3.6.2
209	3.6-4	9.3-11	See Subsection 3.6.2
225	3.6-4	9.3-11	See Subsection 3.6.2
105, 113, 115	3.6-5	9.3-12	See Subsection 3.6.2
112, 116, 117 119, 122, 123	3.6-5	9.3-12 9.3-13	See Subsection 3.6.2
Service Pump Room and Tunnel	3.6-18	9.3-14	See Subsection 3.6.2
Tunnel	--	9.3-15	See Subsection 9.2.1.2

Note 1: This table was generated in response to FSAR question 9.3.4 to identify rooms outside of containment that contained essential equipment and were considered susceptible to flooding.

Note 2: Rooms 705 and 706 are located above, and separated by a metal grating from, Rooms 602 and 601, respectively; thus, for this analysis, Rooms 705 and 706 are considered to be part of Rooms 602 and 601, respectively. Likewise, Room 600 is not separated from Room 601, so Room 600 is considered to be part of Room 601.

TABLE 9.3-7

Makeup and Purification System Performance Data

Normal letdown flow, gpm	70
Maximum letdown flow, gpm	140
Total flow to each reactor coolant pump seal, gpm	8
Seal inleakage to reactor coolant system per reactor coolant pump, gpm	7
Temperature to reactor coolant pump seals, °F	120
Purification letdown fluid temperature, °F	120
Makeup tank normal operating pressure range, psig	25 – 45
Makeup tank nominal water volume, ft^3	400

TABLE 9.3-8

Makeup and Purification System Component Data

Makeup Pump

Quantity	2
Type	Centrifugal, Mechanical Seal
Rated capacity, gpm	150
Rated head, ft at sp gr =1	5800
Motor horsepower	450
Pump material	SS (Wetted Parts)
Design pressure, psig	3050
Design temperature, °F	200
Seismic Class	I
Code	Draft ASME B&PV, III, Class 2

Letdown Cooler

Quantity	2 Half-Capacity (Maximum Letdown)
Type	Shell and Tube
Heat transferred, Btu/hr	16.2×10^6
Letdown flow lb/hr	3.5×10^4
Letdown temperature change, °F	557 to 120
Material, shell/tube	CS/SS
Design pressure, shell/tube, psig	200/2500
Design temperature, shell/ tube, °F	350/600
Cooling water temperature change, °F	95-176
Seismic Class	I (shell and anchors)
Code	ASME Section III-Class 3 and VIII

Seal Return Cooler

Quantity	2 Full Capacity
Type	Shell and Tube
Heat transferred, Btu/hr	40×10^4
Seal return cooler tubeside flow , lb/hr	2.122×10^4
Material, shell/tube	CS/SS
Design pressure, shell/tube, psig	150/150
Cooling temperature, shell/tube, °F	250/200
Cooling water flow, lb/h	1.9628×10^4
Cooling water temperature change, °F	95-115
Seismic Class	I
Code	ASME Section III-C and VIII

TABLE 9.3-8 (Continued)

Makeup and Purification System Component Data

Makeup Tank

Quantity	1
Volume, ft^3	600
Design pressure, psig	100
Design temperature, °F	200
Material	SS
Seismic Class	I
Code	ASME Section III-C

Purification Demineralizer Filter

Quantity	1
Capacity, gpm	140
Vessel material	SS
Vessel design pressure, psig	150
Vessel temperature, °F	200
Absolute rating, microns	0.1 - 10
Vessel seismic Class	II
Vessel code	ASME III-Class 3

Purification Demineralizer

Quantity	2
Type	Mixed or Cation Bed, boric acid saturated
Vessel material	SS
Usable resin volume, ft^3	50
Design flow, gpm	75
Vessel design pressure, psig	150
Vessel design temperature, °F	200
Vessel seismic Class	I
Vessel design code	ASME Section III-C

Makeup Filter

Quantity	2
Capacity, gpm	140
Vessel material	SS
Vessel design pressure, psig	150
Vessel design temperature, °F	200
Absolute rating, microns	0.1 - 10
Vessel seismic Class	II
Vessel design Code	ASME III-Class 3

TABLE 9.3-8 (Continued)

Makeup and Purification System Component Data

Seal Injection Filter

Quantity	2
Capacity, gpm	50
Vessel material	SS
Vessel design pressure, psig	3050
Vessel design temperature, °F	200
Absolute rating, microns	0.45-23
Vessel seismic Class	I
Vessel design Code	ASME III-Class 2

Purification Demineralizer

Quantity	1
Type	Mixed or Cation Bed
Design flow, gpm	140
Usable resin volume, ft^3	51
Vessel design pressure, psig	150
Vessel design temperature, °F	200
Vessel material	304 SS
Vessel seismic Class	II
Vessel design Code	ASME III, Class 3

TABLE 9.3-9

Malfunction Analysis of Makeup and Purification System

	Component	Malfunction	Comment
1.	Letdown Cooler	Tube rupture in one cooler	Pressure switches are installed on the shell side of the letdown coolers. A pressure switch is installed on each letdown cooler shell (CCW) side and initiates closure of the associated cooler's tube side (letdown) inlet isolation valve (MU1A/MU1B). A diverse pressure switch is installed on each letdown cooler shell side outlet piping (CCW) which will initiate the closure of the letdown coolers common inlet (letdown) isolation valve (MU2B). These switches will detect pressurization in the component cooling water system due to a tube rupture. In the event that both coolers became isolated letdown is restored to the unaffected cooler. The unaffected letdown cooler is sufficient to meet normal makeup operations.
2.	Letdown Coolers	Loss of cooling water flow due to failure of component cooling water system downstream of containment vessel isolation valve.	This malfunction results in loss of the capability for feed and bleed. However, cold shutdown can still be achieved. Boric acid may be added in combination with the required demineralized water so that the total added quantity injected will produce the required soluble poison concentration level as well as the required makeup for contraction (582°F to 140°F) and is approximately equal 3250 ft^3q. Required volume of 7875 ppm boric acid solution (assuming the CRA of highest worth stuck out of the core) is approximately 1711 ft^3 (12,800 gallons).
3.	Block orifice	Fails	Either of the two full flow control valves in parallel with the block orifice have capability of maintaining normal letdown flow.
4.	Makeup Pump	Fails while operating	Adequate makeup and seal injection flow is provided by the redundant pump.

TABLE 9.3-9 (Continued)

Malfunction Analysis of Makeup and Purification System

	Component	Malfunction	Comment
5.	Makeup Pump	Fails to start	Adequate makeup and seal injection flow is provided by the redundant pump.
6.	Seal Return Cooler	Tube rupture in one Cooler	The failed cooler can be manually isolated, and the spare cooler can be manually brought on line to provide sufficient seal return and makeup pump re-circulation cooling.
7.	Solenoid actuated, air operated isolation valves outside Containment Vessel	Loss of air supply	Air accumulators are provided on the air inlet to the valves so that the valves will remain in the same position as they were prior to the loss of air. Enough air is available so that the valves will close upon receipt of SFAS signal. The isolation valves in the seal injection lines are equipped with air accumulators to ensure seal injection flow is maintained to the Reactor Coolant Pumps.

TABLE 9.3-10

Decay Heat Removal System Performance Data

Reactor Coolant Temperature at	
Startup of Decay Heat Removal, °F	280
Time to Cool Reactor Coolant System	
From 280°F and a pressure below 260 psig	
to Refueling Temperature, hr	26*
Maximum Refueling Temperature, °F	140
Boron Concentration in the Borated	
Water Storage Tank, minimum ppm boron	2600

* This number was changed to 26 hours based on the DB MUR Summary Report. (References 13 & 14)

TABLE 9.3-11

Decay Heat Removal System Component Design Data

Decay Heat Pumps

Number	2
Type	Single Stage, Centrifugal
Capacity, gpm	3,000
Head at rated capacity, ft	350
Motor horsepower, hp	400
Material	SS (Wetted parts)
Design pressure, psig	450
Design temperature, F	350
Seismic Class	I
Code	Draft ASME B&PV, III Class 2

Decay Heat Removal Coolers

Number	2
Type	Shell and Tube
Heat transferred, Btu/hr	30×10^6 (26.9×10^6)*
Reactor coolant flow (tube), gpm	3,000
Cooling water flow (shell), gpm	6,000
Temperature change, tube/shell, °F	140-120/95-105 (140-122/95-104)*
Material, shell/tube	CS/SS
Design pressure, shell/tube, psig	150/450
Design temperature, shell/tube, °F	250/350
Seismic Class	I
Code	ASME III-C & VIII

* The values in parentheses are from B&W Document 51-1172856-00, dated August 3, 1988, based on input from Atlas Industrial Manufacturing Company for the design normal case heat load assuming degraded Decay Heat Removal Cooler performance as discussed in Section 6.3.1.2.

Borated Water Storage Tank

Number	1
Capacity, gal.	550,000
Material	SS
Design pressure	Atmospheric
Design temperature, °F	125
Seismic Class	I
Code	AWWA D100

TABLE 9.3-12

Chemical Addition System Performance Data

Boric Acid Storage Concentration, wt%	7
Required System Temperature to Prevent Boric Acid Crystallization, °F	95
Volume of Concentrated Boric Acid for Cold Shutdown at Beginning of Life (Feed and bleed method), ft^3	813
Volume of Concentrated Boric Acid for Cold Shutdown Near End of Life (Feed and bleed method), ft^3	773
Volume of Concentrated Boric Acid for Hot Shutdown at Beginning of Life (Feed and bleed method), ft^3	574
Volume of Concentrated Boric Acid for Hot Shutdown Near End of Life (Feed and Bleed method), ft^3	524

This table lists historical system performance data. The Technical Requirements Manual shows the current cycle's system performance requirements with respect to minimum boron concentration and volume for Modes 1 through 4.

TABLE 9.3-13

Chemical Addition System Equipment Data

Boric Acid Mix Tank

Quantity	1
Type	Vertical Cylindrical
Volume, gal	972
Design Pressure, psig	Atmospheric
Design Temperature, °F	200
Material	SS
Seismic Class	II
Code	N/A

Boric Acid Addition Tanks

Quantity	2
Type	Horizontal Cylindrical
Volume, gal	7600
Design Pressure, psig	15
Design Temperature, °F	200
Material	SS
Seismic Class	II
Code	ASME III-C

Lithium Hydroxide Mix Tank

Quantity	1
Type	Vertical Cylindrical
Volume, gal.	50
Design Pressure, psig	Atmospheric
Design Temperature, °F	150
Material	SS
Seismic Class	II
Code	NA

TABLE 9.3-13 (Continued)

Chemical Addition System Equipment Data

Boric Acid Pumps

Quantity	2
Type	Centrifugal
Capacity, gpm	25
Head, ft	180
Design Pressure, psig	100
Design Temperature, °F	200
Material	SS
Seismic Class	II
Code	Draft ASME B&PV, III, Class 3

Lithium Hydroxide Pump

Quantity	1
Type	Diaphragm, Variable Stroke
Capacity, gph	0-10
Head, ft	231
Design Pressure, psig	150
Design Temperature, °F	200
Material	SS
Seismic Class	II
Code	NA

Hydrazine Pump

Quantity	1
Type	Diaphragm, Variable Stroke
Capacity, gph	0-10
Head, ft	231
Design Pressure, psig	150
Design Temperature, °F	200
Material	SS
Seismic Class	II
Code	NA

Zinc Injection Skid

Zinc Injection Pump

Quantity	2
Type	Positive displacement dual diaphragm
Capacity, gph	0.01-0.10
Design Pressure, psig	150
Design Temperature, °F	150
Material	SS
Seismic Class	II
Code	N/A

TABLE 9.3-13 (Continued)

Chemical Addition System Equipment Data

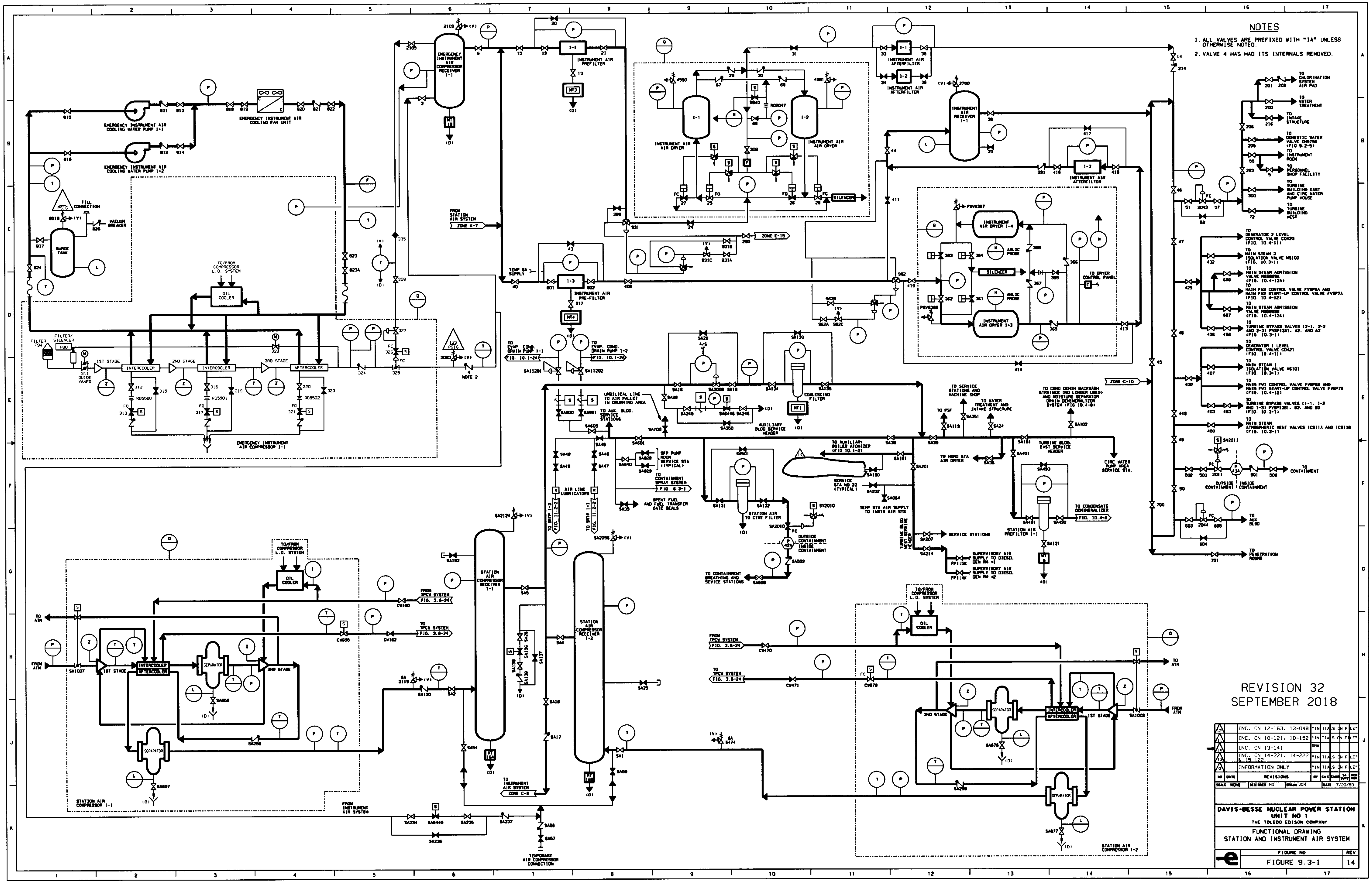
Zinc Acetate Storage Tank

Quantity	1
Type	Vertical Cylinder
Volume, gal	60
Design Pressure, psig	Atmospheric
Design Temperature, °F	150
Material	SS
Seismic Class	II
Code	N/A

TABLE 9.3-14

Malfunction Analysis of the Chemical Addition System

	Component	Malfunction	Comment
1.	Hydrazine Pump	Fails to start or stops	The lithium hydroxide pump can be manually lined up to supply sufficient hydrazine to the reactor coolant.
2.	Lithium Hydroxide Pump	Fails to start or stops	The hydrazine pump can be manually lined up to supply sufficient lithium hydroxide to the reactor coolant.
3.	Boric Acid (BA) Pump	Fails to start or stops	Adequate boric acid for feed and bleed operation is provided by the redundant pump.
4.	Boric Acid Injection Line Flow Control Valve	Fails to open	The throttle valve in the bypass line can be manually opened to permit flow.
5.	Boric Acid Mix Tank Mixer	Fails to operate	Utilize concentrated boric acid from boric acid addition tanks or boric acid concentrate storage tank.
6.	Piping from BA Mix Tank to BA Addition Tank	Break	Same as above.
7.	Zinc Injection Pump	Fails to start or stops	The zinc injection pumps can be manually lined up to supply sufficient zinc acetate to the reactor coolant.
7.	BA Addition Tank and Lines to BA Pumps	Break	The parallel tank and pump will supply the required flow.
8.	2” SS Line Connecting the Discharge of the Two Pumps	Break	Same as 7.
9.	Tracing	Failure to heat tanks or lines	Same as 7.



REVISION 32
SEPTEMBER 2018

NO	DATE	REVISIONS	BY	CH'K	ENGR		
12		INC. CN 12-163, 13-048	"INITIALS ON FILE"				
11		INC. CN 10-121, 10-152	"INITIALS ON FILE"				
14		INC. CN 13-141	SBW				
13		INC. CN 14-221, 14-222 & 15-122	"INITIALS ON FILE"				
0		INFORMATION ONLY	"INITIALS ON FILE"				

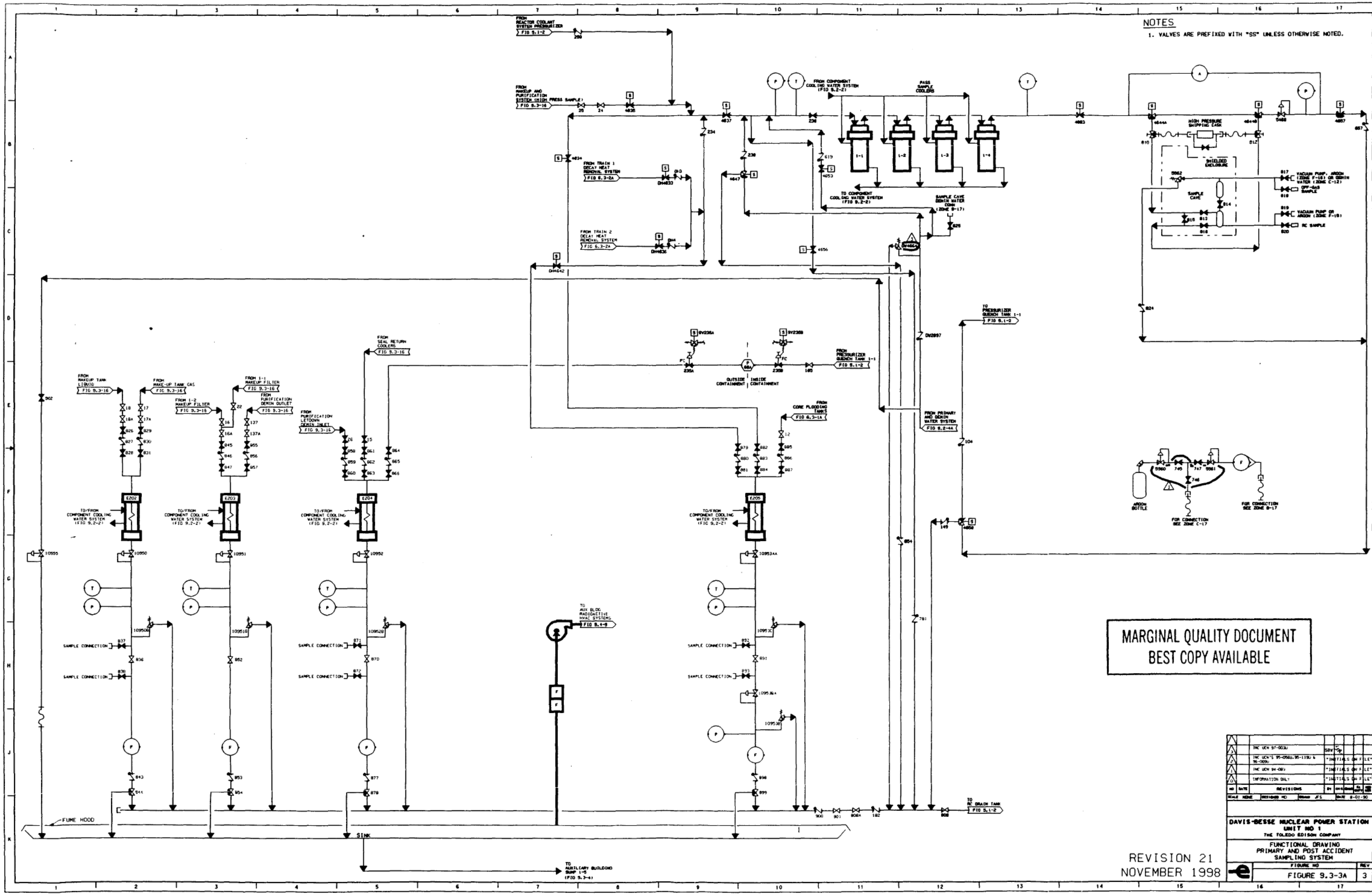
SCALE NONE | DESIGNED MD | DRAWN JOR | DATE 7/20/90

DAVIS-BESSE NUCLEAR POWER STATION
UNIT NO 1
THE TOLEDO EDISON COMPANY

FUNCTIONAL DRAWING
STATION AND INSTRUMENT AIR SYSTEM

FIGURE NO: FIGURE 9.3-1 | REV: 14

G:/USAR/UFIG931.DGN DB 11-301-7



NO	DATE	REVISIONS	BY	CHK			
3		INC UCN 97-003U	SBV				
2		INC UCN'S 95-056U, 95-119U & 96-009U	"INITIALS ON FILE"				
1		INC UCN 94-08?	"INITIALS ON FILE"				
0		INFORMATION ONLY	"INITIALS ON FILE"				

SCALE NONE | DESIGNED HD | DRAWN JFS | DATE 8-01-90

DAVIS-BESSE NUCLEAR POWER STATION
UNIT NO 1
THE TOLEDO EDISON COMPANY

FUNCTIONAL DRAWING
PRIMARY AND POST ACCIDENT
SAMPLING SYSTEM

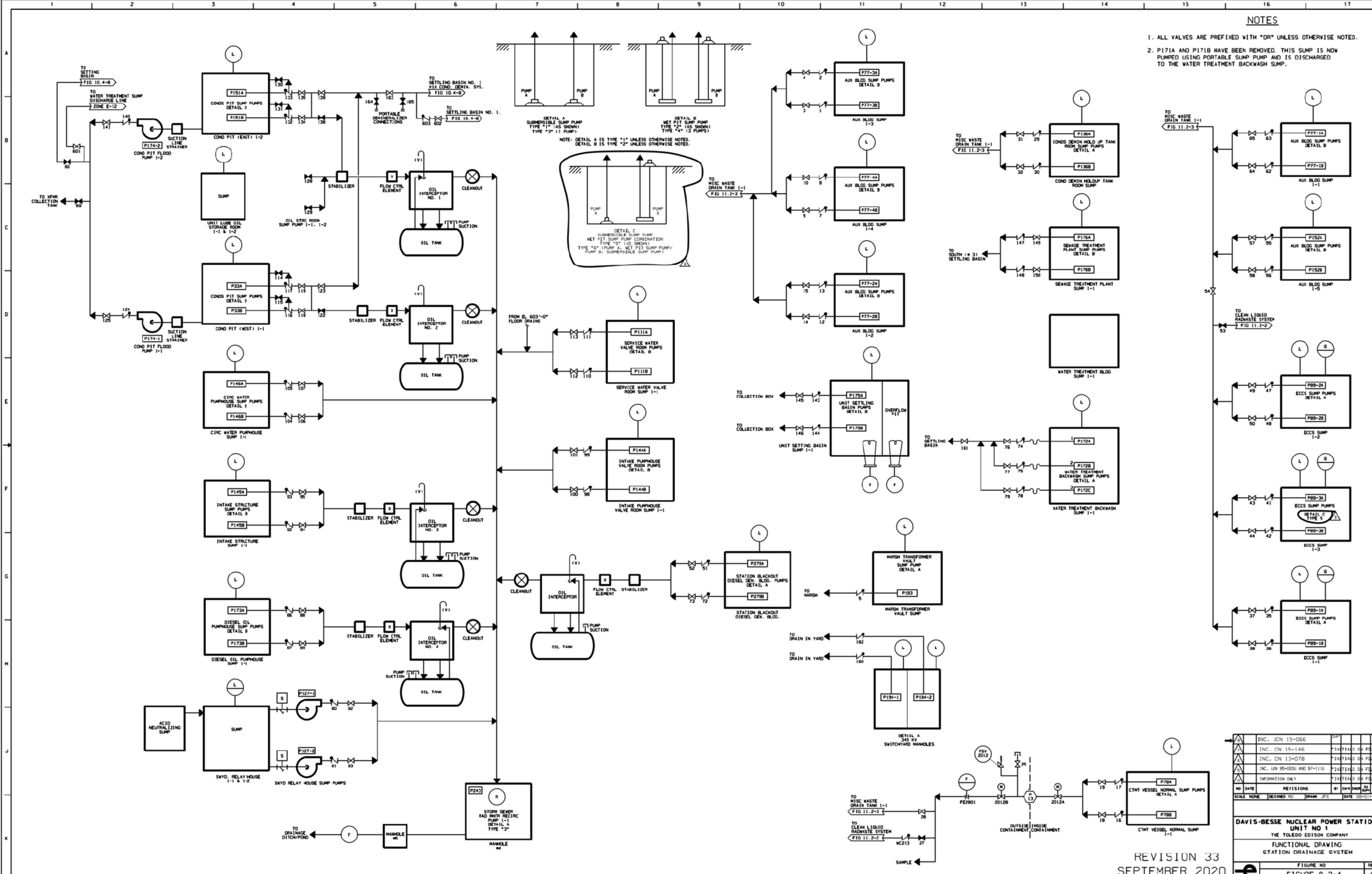
FIGURE NO: FIGURE 9.3-3A | REV 3

REVISION 21
NOVEMBER 1998

DB: 08-04-98
DFN=G/USAR/UFIG933A.DGN

NOTES

1. ALL VALVES ARE PREFIXED WITH "DR" UNLESS OTHERWISE NOTED.
2. P171A AND P171B HAVE BEEN REMOVED. THIS SUMP IS NOW PUMPED USING PORTABLE SUMP PUMP AND IS DISCHARGED TO THE WATER TREATMENT BACKWASH SUMP.



NO	DATE	REVISIONS	BY	CHK	ENGR	SA SUPV	MED MGR
8		INC. UCN 15-066	SAP				
7		INC. CN 15-146	"INITIALS ON FILE"				
6		INC. CN 13-078	"INITIALS ON FILE"				
5		INC. UCN 95-003U AND 97-111U	"INITIALS ON FILE"				
0		INFORMATION ONLY	"INITIALS ON FILE"				

SCALE NONE | DESIGNED MD | DRAWN JFS | DATE 08-01-90

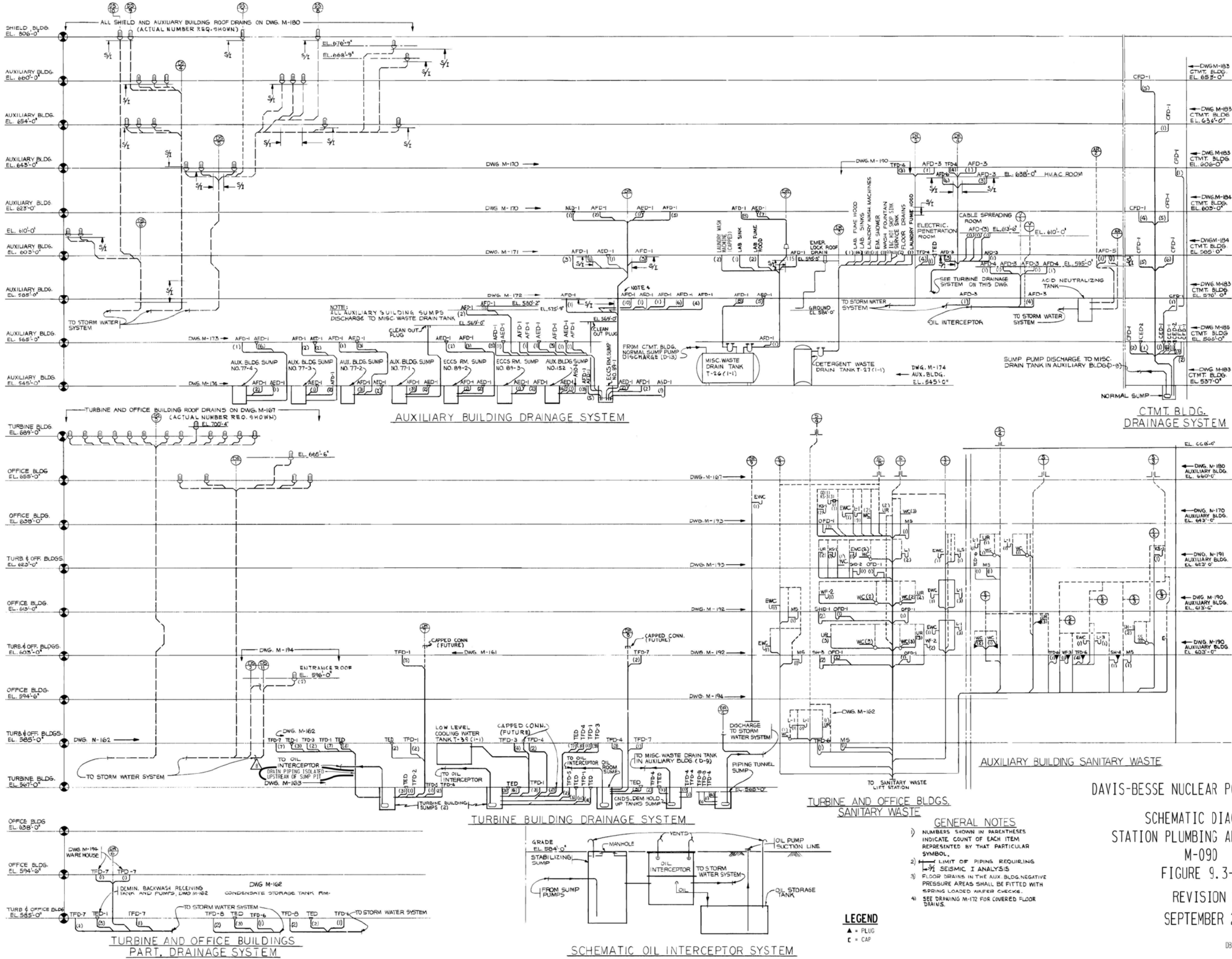
DAVIS-BESSE NUCLEAR POWER STATION UNIT NO 1
THE TOLEDO EDISON COMPANY

FUNCTIONAL DRAWING
STATION DRAINAGE SYSTEM

FIGURE NO: FIGURE 9.3-4 | REV: 8

REVISION 33
SEPTEMBER 2020

DB 04-27-20
DFN=G/USAR/UFIG934.DGN



GENERAL NOTES

1) NUMBERS SHOWN IN PARENTHESES INDICATE COUNT OF EACH ITEM REPRESENTED BY THAT PARTICULAR SYMBOL.
2) LIMIT OF PIPING REQUIRING S/I SEISMIC I ANALYSIS
3) FLOOR DRAINS IN THE AUX. BLDG. NEGATIVE PRESSURE AREAS SHALL BE FITTED WITH SPRING LOADED WAFER CHECKS.
4) SEE DRAWING M-172 FOR COVERED FLOOR DRAINS.

LEGEND

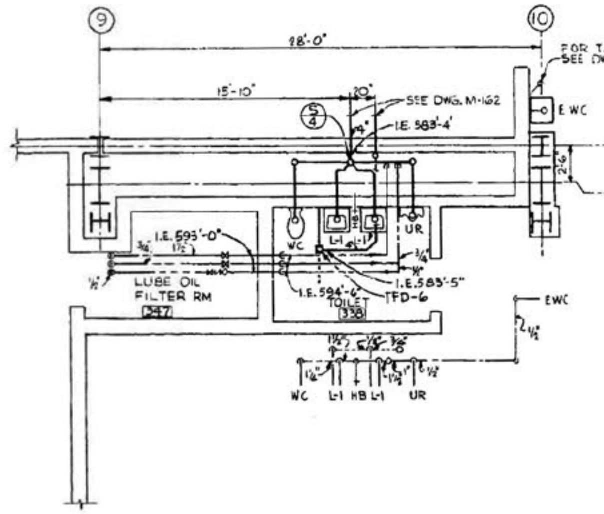
▲ = PLUG
C = CAP

DAVIS-BESSE NUCLEAR POWER STATION

SCHEMATIC DIAGRAM
STATION PLUMBING AND DRAINS
M-090
FIGURE 9.3-5

REVISION 33
SEPTEMBER 2020

DB 08-17-20 DFN=/J/RASDGN/M90.DGN/CIT



TURBINE BUILDING
PARTIAL FLOOR PLAN DETAIL AT EL 585'-0"
SCALE: 1/4"=1'-0"

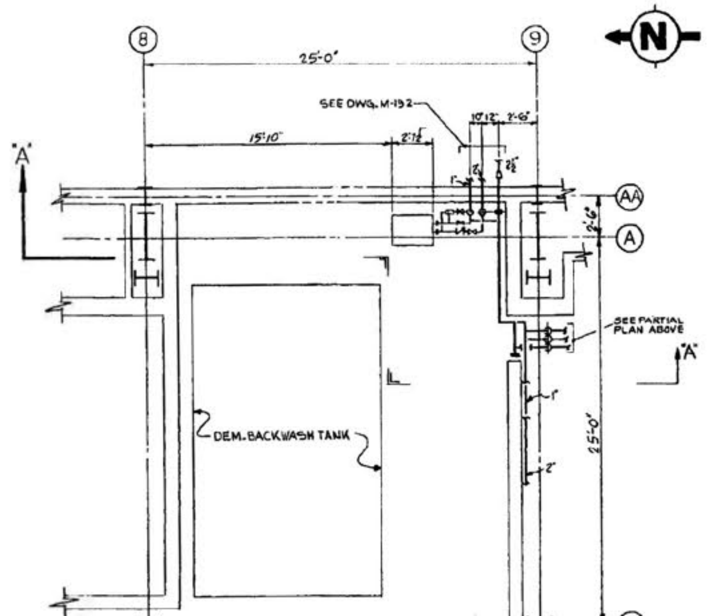
PLUMBING FIXTURE CONNECTIONS						
ITEM	FIXTURE	MOUNTING HEIGHT (X)	WATER COLD	WATER HOT	TRAP SIZE	REMARKS
L-1	LAVATORY	31"	1/2"	1/2"	1 1/2"	WALL HUNG
WC	WATER CLOSET	—	1 1/4"	—	—	WALL HUNG
WC-1	WATER CLOSET	—	1/2"	—	—	TANK OPERATED FLOOR MOUNTED
L-2	LAVATORY	—	1/2"	1/2"	1 1/2"	COUNTERTOP
			1/2"	1/2"	3"	
EWC	EL. WATER COOLER	40"	1/2"	—	1 1/2"	WALL HUNG
EWC-1	EL. WATER COOLER	—	1/2"	—	1 1/2"	FLOOR MOUNTED
SH-1	SHOWER	—	1/2"	1/2"	2"	SINGLE HEAD
SH-2	SHOWER	—	3/4"	3/4"	—	2 HEADS (WALL HUNG)
SH-3	SHOWER	—	1"	1"	3"	6 HEADS COLUMN
UR	URINAL	24"	1"	—	—	WALL HUNG
			1"	1"	2"	
WF-2&3	WASH FOUNTAIN	—	1"	1"	2"	CIRCULAR
ILS	INSTR. LAB SINK	—	1/2"	1/2"	1 1/2"	COUNTERTOP
SH-4	SHOWER	—	1"	1"	3"	5 HEADS COLUMN
ES	EMERGENCY SHOWER	—	1 1/4"	—	—	FLOOR DRAIN
KS-1	KITCHEN SINK	—	1/2"	1/2"	1 1/2"	WITH GARBAGE DISPOSER
KS-2	KITCHEN SINK	—	1/2"	1/2"	1 1/2"	DOUBLE BOWL
SS	SERVICE SINK	28"	1/2"	1/2"	3"	WALL HUNG
EWH	ELECTRIC WATER HTR.	—	3/4"	3/4"	—	FLOOR MOUNTED
LW	LAUNDRY WASH. MACH.	—	1/2"	1/2"	2"	
DW	DISHWASHER	—	1/2"	1/2"	—	CONNECT WASTE TO GARBAGE DISPOSER OUTLET
WH	WALL HYDRANT		3/4"	—	—	"JOSAM" NO. 1410-U.
GD	GARBAGE DISPOAL	—	—	—	2"	
GT	GREASE TRAP	—	—	—	—	
KS-3	KITCHEN SINK	—	1/2"	1/2"	2"	SINGLE BOWL

(X) TO FLOOD RIM OF FIXTURE

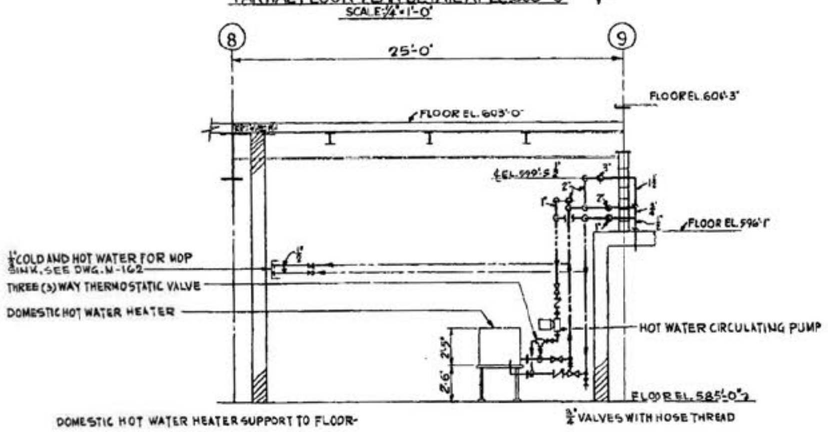
PLUMBING SYMBOLS		
SYMBOL	ABBREV.	ITEM
	CW	POTABLE COLD WATER PIPING
	HW	POTABLE HOT WATER PIPING
	HWC	POTABLE HOT WATER CIRCULATING PIPING
	HB	HOSE BIBB
	WH	WALL HYDRANT
		GATE VALVE IN PLAN
—N—		CHECK VALVE
		BALANCING FITTING
		GATE VALVE IN VERTICAL
W		WATER RISER
		SOIL WASTE PIPING
		VENT PIPING
	CO	CLEAN OUT
	TT	TEST TEE
S		SOIL STACK, VENT THRU ROOF
—AR—		ACID RESISTING WASTE PIPING
----AR----		ACID RESISTING VENT PIPING
AR		ACID RESISTING VENT PIPING THRU ROOF
DR		DRAINAGE PIPING RISER
		STORM WATER OR DRAINAGE PIPING
DS		DOWNSPOUT
		DRAINAGE PIPING EMBEDDED IN CONCRETE
	RD	ROOF DRAIN
	TFD-1	TURBINE BLDG. FLOOR DRAIN 'JOSAM' 524 OR EQUAL
	TFD-2	TURBINE BLDG. FLOOR DRAIN 'JOSAM' 545 OR EQUAL
	TFD-3	TURBINE BLDG. FLOOR DRAIN 'JOSAM' 774 OR EQUAL
	TFD-4	TURBINE BLDG. FLOOR DRAIN 'JOSAM' 654 OR EQUAL
	TFD-5	TURBINE BLDG. FLOOR DRAIN 'JOSAM' 666YY OR EQUAL
	TFD-6	TURBINE BLDG. TOILET RM. 'JOSAM' 300-35
	TFD-7	TURBINE BLDG. FLOOR DRAIN 'JOSAM' 524 WITH 4" CAST IRON 'P' TRAP
	TFD-8	TURBINE BLDG. FLOOR DRAIN 'JOSAM' 526
	RFD-2	RELAY HOUSE FLOOR DRAIN WITH BRASS STRAINER
	RED-1	RELAY HOUSE EQUIPMENT DRAIN (ACID RESISTING)
	OFD-1	OFFICE BLDG. FLOOR DRAIN
	S.H.D.-1	SHOWER DRAIN Ø 3"
	S.H.D.-2	SHOWER DRAIN Ø 2"
	RFD-1	RELAY HOUSE FLOOR DRAIN (ACID RESISTING)
	CD	CANOPY DRAIN 'JOSAM 4943-B' OR EQUAL
	S.H.D.-1	SHOWER DRAIN
	T.E.D.	TURBINE BLDG. EQUIPMENT DRAIN HUB OF CAST IRON PIPE 2" ABOVE FLOOR
	T.O.D.E.	TOP OF DRAIN ELEVATION
	I.E.	INVERT ELEVATION
—GS—		GALVANIZED STEEL PIPING
	AP	ACCESS PANEL
—A—		COMPRESSED AIR PIPING
—V—		VACUUM
—G—		GAS PIPING (LP)
—DW—		DEMINERALIZED WATER PIPING
	TED-1	TURB. BLDG. EQUIPT. DRAIN WITH 4" CAST IRON 'P' TRAP
SP		STANDPIPE RISER
		STANDPIPE HOSE VALVE IN PLAN. HOSE VALVES SHALL BE "KENNEDY" Ø 1 1/2" FIG. 930 SCREWED WITH CAP AND CHAIN OR EQUAL.
		FLANGES OR ISOLATION FLANGES
		PLUG
		CAP
	FC	FLOW CONTROL FITTING

GENERAL NOTES

1. PLUMBING SYSTEM SHALL BE INSTALLED IN ACCORDANCE WITH OHIO STATE PLUMBING CODE.
2. FOR OFFICE BUILDING AND TURBINE BUILDING SANITARY, WATER AND DOWNSPOUTS RISER DIAGRAMS SEE DWG. M-195.
3. FOR EXACT LOCATION OF PLUMBING FIXTURES SEE ARCHITECTURAL DRAWINGS.
4. FOR SANITARY WASTE PIPING KEEP SLOPE 1/4" PER FOOT.
5. FOR FLOOR AND SHOWER DRAINS INSTALLED ON ELEVATIONS ABOVE 585'-0" FURNISH AND INSTALL FLASHING AND CLAMPING DEVICES. FOR FLASHING USE SHEET LEAD NOT LESS THAN 4 POUNDS PER SQUARE FOOT. FLASHING SHALL EXTEND 10" MIN. IN ALL DIRECTIONS.
6. HANGERS FOR COPPER PIPING SHALL BE INSIDE COPPER GALVANIZED OR USE SHEET LEAD BETWEEN PIPE AND HANGER.
7. SOLDER: FOR COLD WATER COPPER PIPING, SOLDERING SHALL BE MADE OF CLEAN METAL AND MAY BE 50 PERCENT LEAD AND 50 PERCENT IN BY WEIGHT. FOR HOT WATER AND HOT WATER RECIRCULATING COPPER PIPING, USE SOLDER OF 95 PERCENT TIN AND 5 PERCENT ANTIMONY BY WEIGHT.
8. INSULATION: ALL COLD WATER, HOT WATER AND HOT WATER RECIRCULATING PIPING SHALL BE COVERED WITH 1/2" THICKNESS FIBERGLASS WITH UNIVERSAL JACKET.
9. PERFORATED METAL SCREENS INSTALLED IN FLOOR DRAINS TO PROVIDE FOR FOREIGN MATERIAL EXCLUSION (FME) ARE ACCEPTABLE. (ECP 07-0164-00)



TURBINE BUILDING
PARTIAL FLOOR PLAN DETAIL AT EL. 585'-0"
SCALE: 3/8"=1'-0"



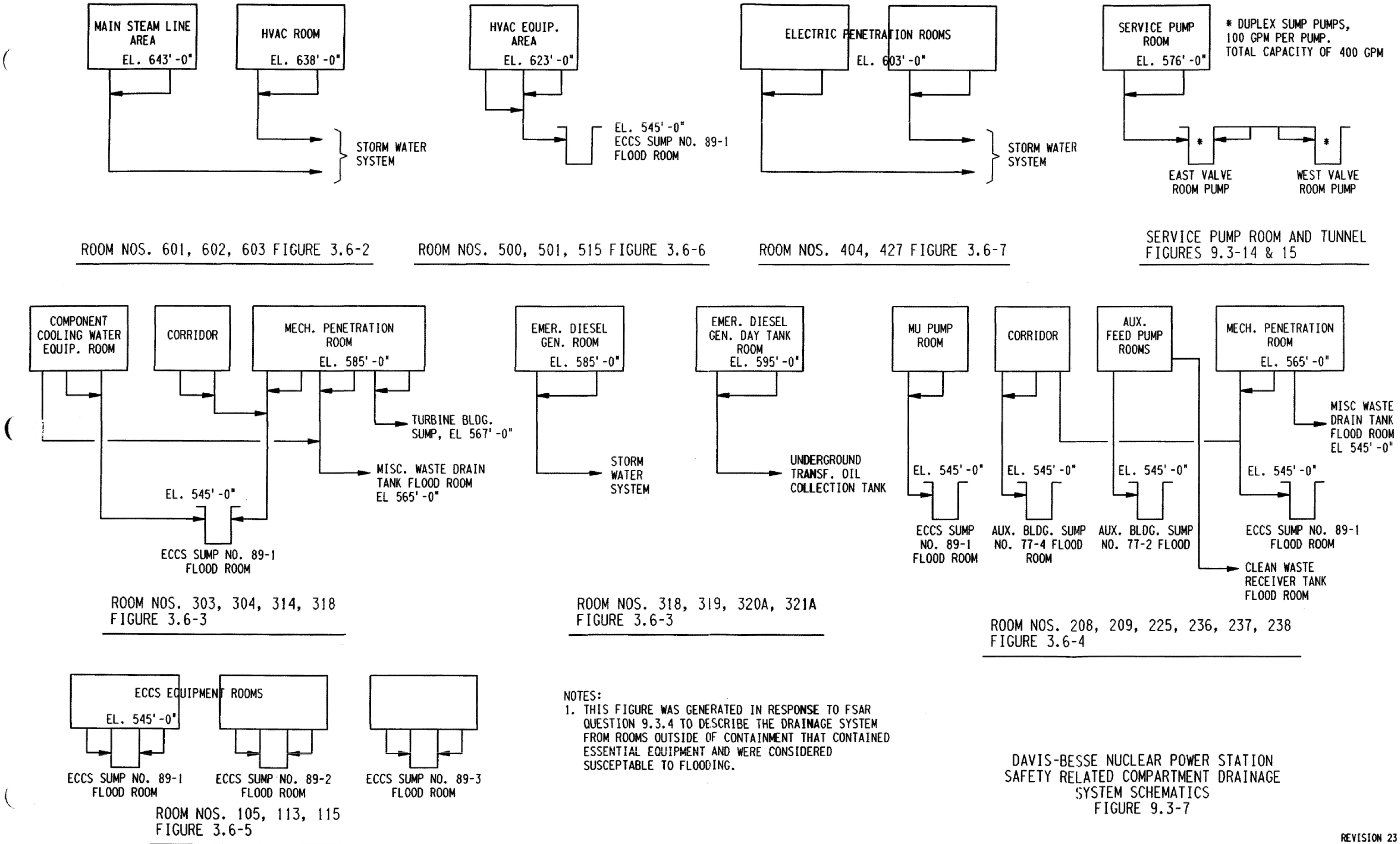
SECTION "A-A" LOOKING EAST
DOMESTIC HOT WATER HEATER WATER PIPING ARRANGEMENT
SCALE: 3/8"=1'-0"

DAVIS-BESSE NUCLEAR POWER STATION

SYMBOLS SCHEDULES AND DETAILS
M-160
FIGURE 9.3-6

REVISION 31
OCTOBER 2016

DB 03-21-16 DFN=/J/RASDGN/M160.DGN/CIT



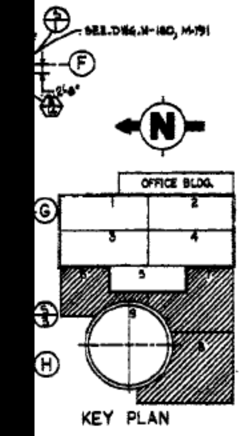
NOTES:
1. THIS FIGURE WAS GENERATED IN RESPONSE TO FSAR QUESTION 9.3.4 TO DESCRIBE THE DRAINAGE SYSTEM FROM ROOMS OUTSIDE OF CONTAINMENT THAT CONTAINED ESSENTIAL EQUIPMENT AND WERE CONSIDERED SUSCEPTABLE TO FLOODING.

DAVIS-BESSE NUCLEAR POWER STATION
SAFETY RELATED COMPARTMENT DRAINAGE SYSTEM SCHEMATICS
FIGURE 9.3-7

REVISION 23
NOVEMBER 2002

DB 10-30-01 DFN: G/USAR/UFIG937.DGN

[redacted]

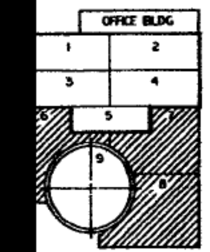


-BESSE NUCLEAR POWER STATION
AUXILIARY BUILDING
PLANS EL.623'-0", 638'-0"
AND 643'-0"
M-170
FIGURE 9.3-8
REVISION 30

OCTOBER 2014

DB 02-10-14 DFN=/H/PLDSGN/M170.DGN/TIF

[redacted]



KEY PLAN
EL. 603'-0"

PENETRATION LIST	
℄ EL. OF SLEEVE	REMARKS
ROOF EL. 610'-0"	SEE DWG. M-180
EL. 597'-1 3/4"	SEE DWG. M-180
ROOF EL. 610'-0"	SEE DWG. M-180
FLOOR EL. 595'-0"	SEE DWG. M-172
FLOOR EL. 595'-0"	SEE DWG. M-172
FLOOR EL. 595'-0"	SEE DWG. M-172
FLOOR EL. 595'-0"	SEE DWG. M-172
EL. 601'-1 1/4"	
GRATING EL. 603'-0"	
EL. 601'-1 3/4"	
EL. 600'-7 1/4"	THRU WF BEAM
EL. 600'-6 1/4"	THRU WF BEAM
EL. 598'-2 1/2"	
FLOOR EL. 603'-0"	
EL. 599'-3 1/4"	
FLOOR EL. 603'-0"	
FLOOR EL. 603'-0"	
EL. 597'-2 1/2"	
EL. 601'-1 1/2"	
FLOOR EL. 603'-0"	
ROOF EL. 610'-0"	SEE DWG. M-180
FLOOR EL. 6[illegible]-0"	
EL. 601'-1"	THRU WF BEAM

, SYMBOLS AND DETAILS SEE
-195
S SHALL BE "CLASS MSD" EXCEPT

SHALL BE AFD-1, EXCEPT AS NOTED.
ATION = 602'-10 1/2" EXCEPT AS NOTED.
LABS. AND ACCESS CONTROL RM.
COLS. 8-11 AND F-H SEE DWG-M-190.
CTED TO DR-1 USE STANDARD TEES.
ROVIDED UNDER CONSTRUCTION
D.
OF SEISMIC I PIPING.
TORS ON THIS DRAWING ARE FOR INFORMATION
AND AREA DESIGNATORS WILL NOT BE
S DRAWING. FOR ROOM AND AREA DESIGNATORS
103.

IS-BESSE NUCLEAR POWER STATION

NAGE SYSTEM AUXILIARY BUILDINGS
PLAN ELEV. 603'-0"
M-171
FIGURE 9.3-9

REVISION 30
OCTOBER 2014

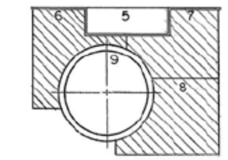
DB 07-17-14 DFN=/J/RASDGN/M171.DGN/CIT

1 2 3 4 5 6 7 8 9 10 11

271'-0"

[redacted]

N



KEY PLAN
EL. 585'0"

PENETRATION LIST			
PEN NO	SIZE O.D.	℄ EL. OF SLEEVE	REMARKS
B 1	8"	581'-3 1/4"	
B 3	8"	581'-3 1/4"	
B 4	8"	580'-11 3/8	
B 5	14"	IN FLOOR EL. 585'-0"	
B 8	6"	IN FLOOR EL. 585'-0"	
B 11			
B 13	8"	582'-2 1/4"	
B 16			
B 18	14"	579'-11 3/8"	
B 19	8"	IN FLOOR EL. 585'-0"	
B 20	8"	579'-8 1/2"	

BLDG.
00

SYMBOLS AND DETAILS SEE DWG. M-175.
SHALL BE CLASS MSD EXCEPT AS NOTED ON DWG.
E TYPE AFD-1 UNLESS OTHERWISE NOTED ON DWG.
CTLY RELATED TO DRAINAGE SYSTEMS ARE FOR
NOT NECESSARILY CURRENT.
S ABOVE EL. 585'-0" FURNISH AND INSTALL
NG DEVISES EXCEPT STAINLESS STEEL DRAINAGE
ATIVE PRESSURE AREAS NOTED ARE TO BE
G LOADED CLOSED WAFER CHECK UNDER THE DRAIN
S THIS DWG.) THESE WAFER CHECKS ARE SAFETY
OM 328) FLOOR DRAINS ARE TO BE FITTED WITH
D WAFER CHECKS UNDER THE DRAIN GRATES AND THE
TO BE CAPPED PER MOD 88-0055
OF THIS DRAWING ARE NOT WITHIN THE SCOPE OF
IONS TO ANY ITEM ON THIS DRAWING WHICH IS OUT
DRAWING WILL NOT BE REFLECTED.
OR DETAILS OF COVER PLATE.
UNDER THE FLOOR AND IS NOT IN SERVICE.

AVIS-BESSE NUCLEAR POWER STATION
DRAINAGE SYSTEM
AUXILIARY BUILDING
M-172
FIGURE 9.3-10

REVISION 33
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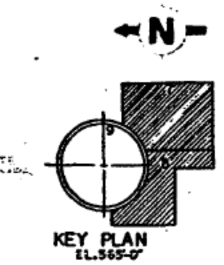
DB 04-23-20 JFN=/H/PLDSGN/M172.DGN/TIF

[redacted]



MARGINAL QUALITY DOCUMENT
BEST COPY AVAILABLE

...ENNERALIZERS

...TES
...GENERAL NOTES, SYMBOLS, PENETRATIONS
...DETAILS SEE DWG. M-175...
...FLOOR DRAINS TYPE AFD-1 UNLESS OTHERWISE
...TED ON DWG.
...OR DRAINS IN NEGATIVE PRESSURE AREAS NOTED ARE TO BE
...TED WITH A SPRING LOADED CLOSED WAFER CHECK UNDER THE
...GRATE TYP. 3 PLCS. THIS DWG. THESE WAFER CHECKS ARE SAFETY RELATED.
...IPMENT DRAINS IN NEGATIVE PRESSURE AREAS NOTED
...TO BE SEALED CLOSED. (TYP. 3 PLACES THIS DWG.)
...OR DRAINS IN THE CLEAN LIQUID WASTE MONITOR TANK ROOMS
...OMS 200 & 201) ARE TO BE FITTED WITH A SPRING LOADED CLOSED
...ER CHECK UNDER THE DRAIN GRATE PER MOD [illegible]. THESE WAFER
...KS ARE NOT SAFETY RELATED. ALSO, THE FLOOR DRAIN GRATE IN ROOM
...IS TO HAVE A [illegible] INCH HIGH, SIX INCH DIAMETER RISER
...ACHED (REFERENCE SKETCH [illegible].)
...D OUT PORTIONS OF THIS DRAWING ARE NOT
...IN THE SCOPE OF THIS DRAWING. REVISIONS TO
...ITEM ON THIS DRAWING WHICH IS OUT OF THE
...E OF THE DRAWING WILL NOT BE REFLECTED.
...R DRAINS IN ROOMS 237 & 238 ARE TO BE FITTED WITH A SPRING LOADED
...D WAFER CHECK VALVE UNDER THE GRATE.
...IPMENT DRAINS IN ROOMS 237 & 238 ARE TO BE PLUGGED.
...ES 7 & 8 ARE FOR MOD 96-0057.

13

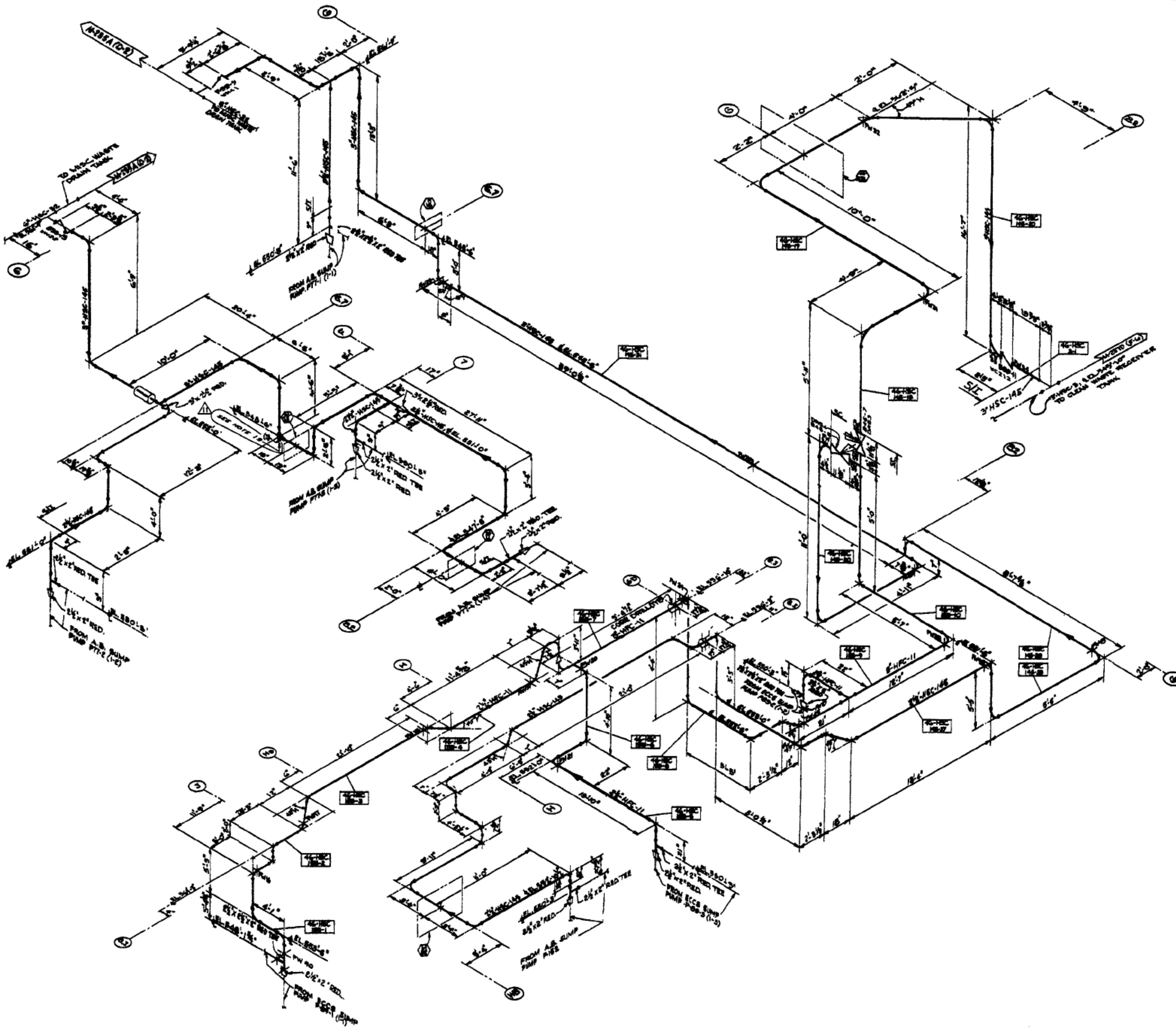
DAVIS - BESSE NUCLEAR POWER STATION
DRAINAGE SYSTEMS - AUXILIARY BUILDING
M – 173
FIGURE 9.3-11

REVISION 23
NOVEMBER 2002

Removed in Accordance with RIS 2015-17

DAVIS-BESSE NUCLEAR POWER STATION
DRAINAGE SYSTEMS - AUXILIARY BUILDING
M-174
FIGURE 9.3-12

DB 10-13-16 DFN H/PLDSGN/M174.DGN/TIF



NOTES:

2. FOR MATERIAL DESCRIPTION OF PIPE, VALVES AND FITTINGS, SEE PIPING CLASS SHEETS M-601 OF SPEC. 7749-M-198.
3. FOR PIPING AND INSTRUMENT DIAGRAM SEE DWG. M-039 & M-046.
4. FOR EQUIPMENT LIST SEE M-600.
6. NO ALLOWANCE HAS BEEN MADE FOR WELDED JOINTS OR GASKETS AND ALL PIPING IS DIMENSIONED IN THE COLD ERECTED POSITION UNLESS OTHERWISE NOTED ON THE DRAWING.
7. FOR PIPING SYSTEMS COMPOSITE DRAWING(S) THIS AREA SEE M-260A & M-260B.
8. FOR MATERIAL DESCRIPTION OF INSULATION SEE SPEC. 7749-M-197 AND 7749-M-198.
9. PIPE BENDS, UNLESS OTHERWISE NOTED, HAVE A MINIMUM CENTER LINE RADIUS OF FIVE (5) TIMES THE NOMINAL PIPE DIAMETER.
10. FOR PIPING ISOMETRIC AND COMPOSITE DWG. INDEX SEE DWG. M-200.
11. FOR PIPING AND INSTRUMENT SYMBOLS SEE DWG. M-001 AND M-002.
13. UNLESS OTHERWISE NOTED, ALL VENT CONNS. ARE 3/4" AND DRAIN CONNS. ARE 1".
14. THIS ISOMETRIC TO BE WORKED WITH DWGS. M-235A, M-237D AND M-246A.
15. DELETED.
16. ⊢—S/I DENOTES LIMITS OF SEISMIC CLASS 1 PIPING.
17. THIS DWG. CONTAINS INFORMATION PREVIOUSLY SHOWN ON ITT GRINNELL DWG. NO. PW-246B.
18. FOR HANGER LOCATIONS SEE DRAWING HL-246B.
19. PERMANENT SHIELDING INSTALLED ON VALVE DR142. SEE CALC 96A.

11

DAVIS-BESSE NUCLEAR POWER STATION

SUMP PUMP DISCHARGE SYSTEM
M-246B
FIGURE 9.3-13

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DAVIS-BESSE NUCLEAR POWER STATION

DRAINAGE SYSTEMS - PIPING TUNNEL
AND INTAKE STRUCTURE
(M-165)
FIGURE 9.3-14

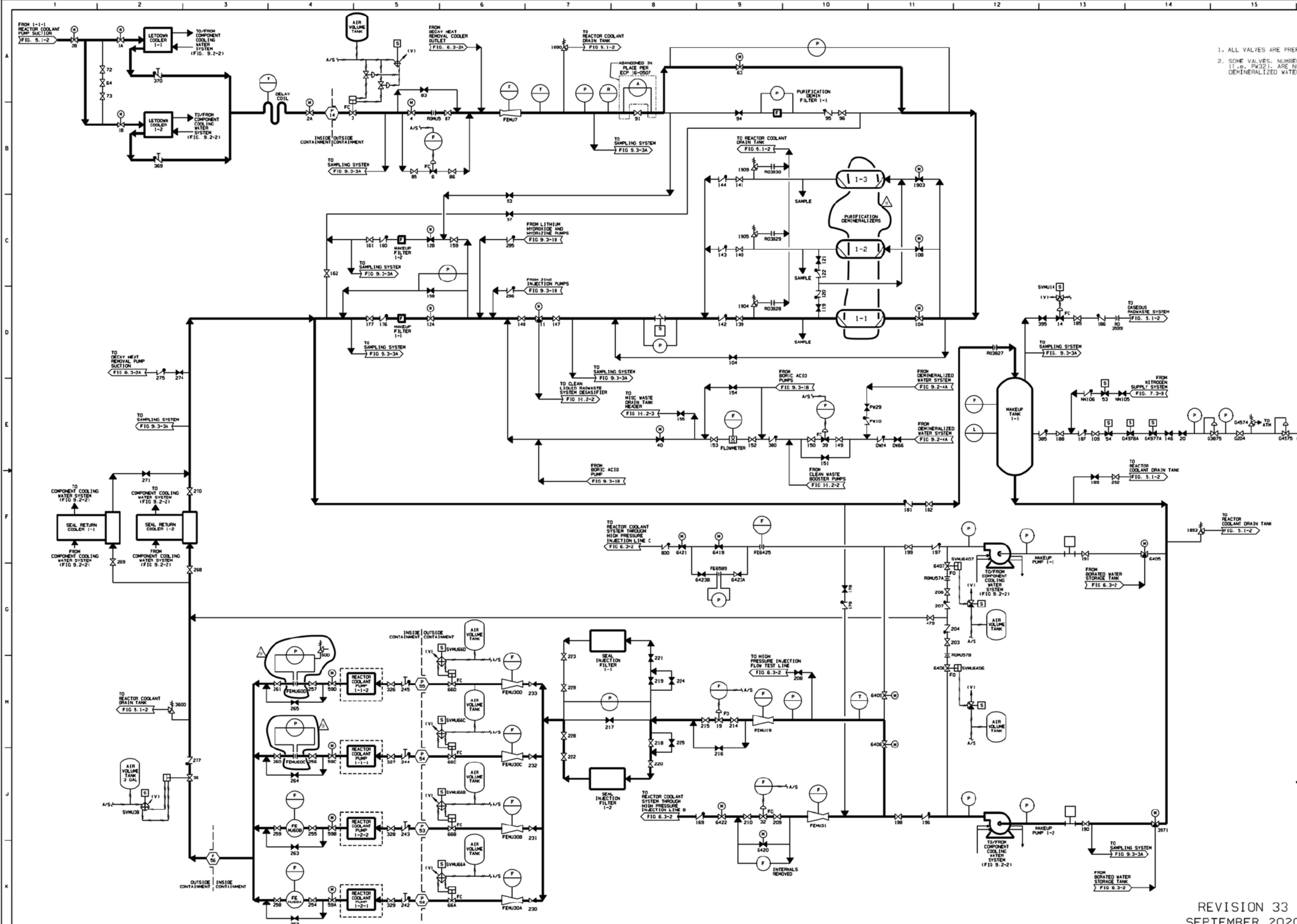
DB 08-13-12 DFN= N/PLDSGN/M165.DGN/TIF

Removed in Accordance with RIS 2015-17

DAVIS-BESSE NUCLEAR POWER STATION

DRAINAGE SYSTEM - TURBINE BUILDING
PLAN AT ELEV. 567'-0"
(M-163)
FIGURE 9.3-15

DB 04-02-12 DFN= H/PLDSGN/M163.DGN/TIF



NOTES

1. ALL VALVES ARE PREFIXED WITH "MU" UNLESS OTHERWISE NOTED.
2. SOME VALVES, NUMBERED WITH A "PW" DESIGNATION (i.e. PW32), ARE NOW USED TO SUPPLY DEMINERALIZED WATER. (REF. MOD 95-0050)

NO	DATE	REVISIONS	BY	CHK	ENGR	SA SUPV	MED MGR
5		INC. UCN'S 97-063U, 03-025U AND UCN 03-047U	"INITIALS ON FILE"				
9		INC. CN'S 19-043, 19-044 & 20-019	SAP				
8		INC. CN 16-153	"INITIALS ON FILE"				
7		INC. CN 14-245	"INITIALS ON FILE"				
6		INC. CN 11-144	"INITIALS ON FILE"				
0		INFORMATION ONLY	"INITIALS ON FILE"				

SCALE NONE | DESIGNED MJD | DRAWN BEW | DATE 04-09-90

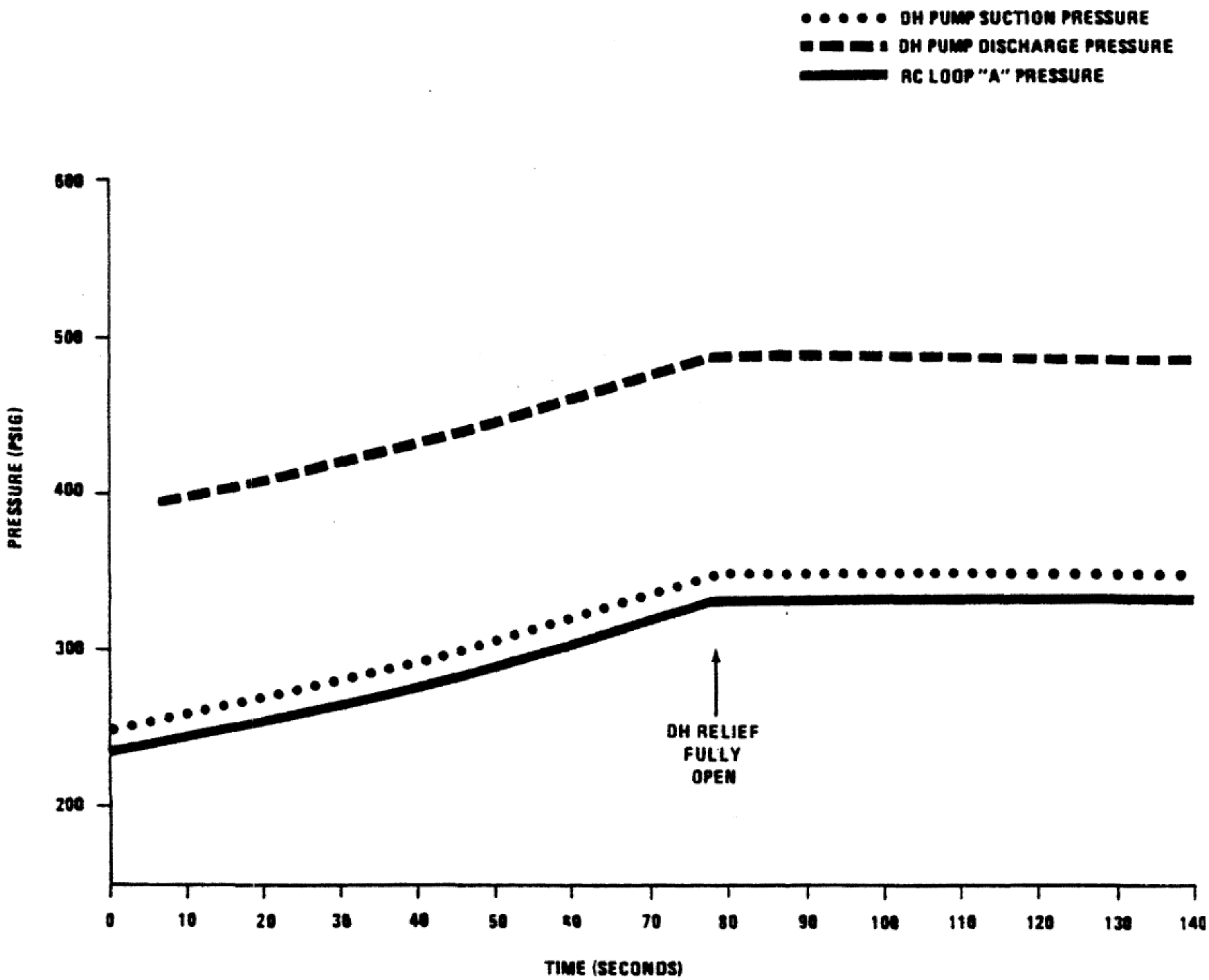
DAVIS-BESSE NUCLEAR POWER STATION UNIT NO 1
THE TOLEDO EDISON COMPANY

FUNCTIONAL DRAWING
MAKEUP AND PURIFICATION SYSTEM

FIGURE NO: FIGURE 9.3-16 | REV: 9

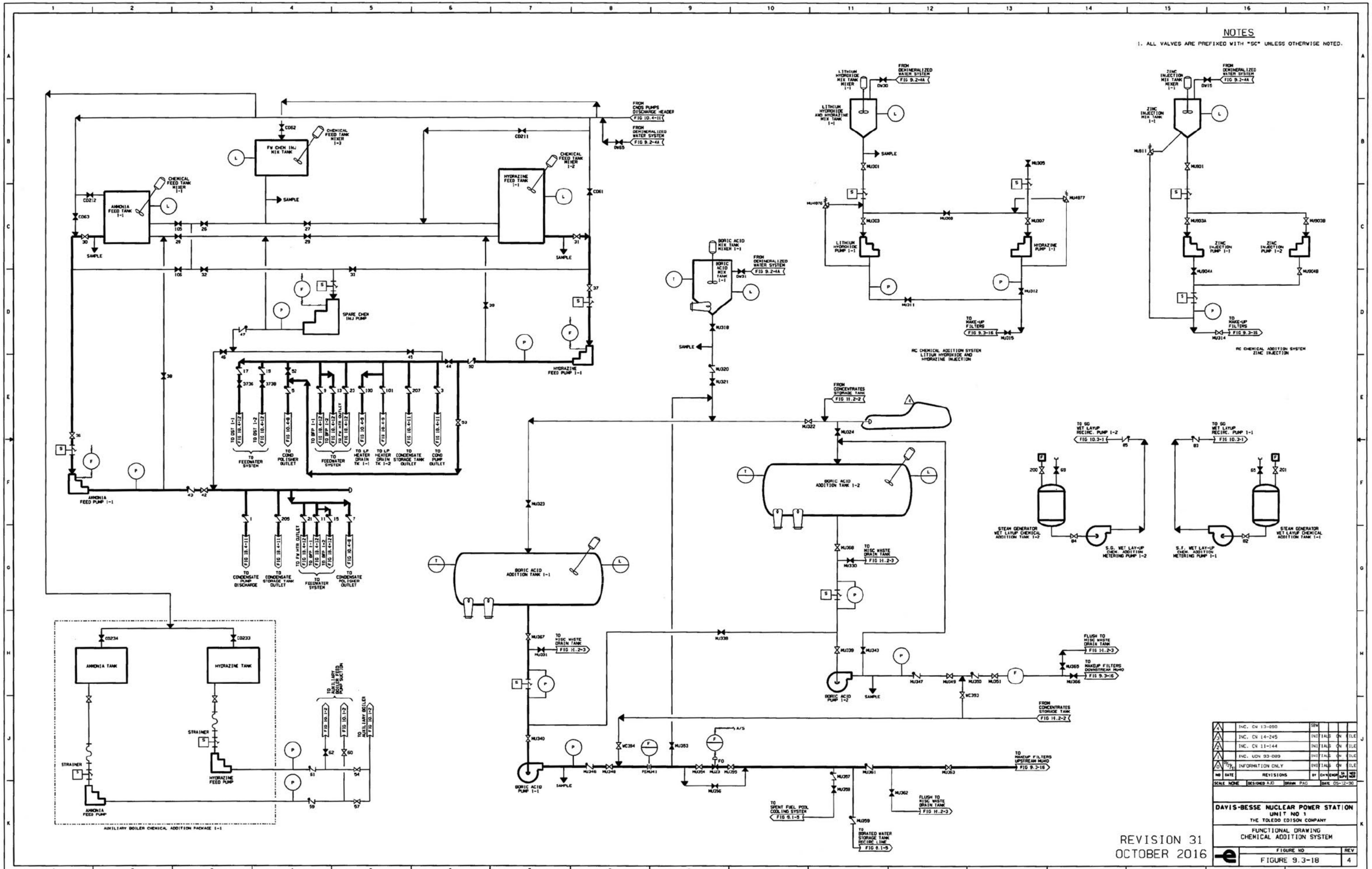
REVISION 33
SEPTEMBER 2020

DFN=/USAR/UFIG9316.DGN
DB 07-14-20



DAVIS-BESSE NUCLEAR POWER STATION
PRESSURES AFTER STARTUP OF DH SYSTEM AT 280 F
WITH HPI SYSTEM INADVERTENTLY ACTUATED
FIGURE 9.3-17

REVISION 0
JULY 1982



REVISION 31
OCTOBER 2016

FIGURE 9.3-18 CHEMICAL ADDITION SYSTEM

9.4 AIR CONDITIONING, HEATING, COOLING, AND VENTILATING SYSTEMS

9.4.1 Control Room

9.4.1.1 Design Bases

The heating, ventilating, and air conditioning systems for the control room are designed to provide a suitable environment for equipment and station operator comfort and safety.

The ventilating systems and equipment are designed in accordance with the recommended practices of the American Society of Heating, Refrigerating, Ventilating, and Air Conditioning Engineers Guide, the Air Moving and Conditioning Association, and the National Fire Protection Association. The ductwork is designed and fabricated in accordance with the Sheet Metal and Air Conditioning Contractors' National Association (SMACNA) standards.

During normal operation, the control room normal air conditioning system utilizes recirculation of return air with a suitable fresh-air makeup. The system has the capability of heating and ventilating with outside air or cooling using mechanical refrigeration. The system is capable of introducing one hundred percent outside air. The control room is maintained at approximately 75°F and 50 percent relative humidity in the summer and 75°F and 30 percent relative humidity in the winter. Radioactivity limits and radiation protection are discussed in Subsection 9.4.1.3. The operation of the control room emergency ventilation system is required to mitigate the consequences of design basis accidents.

The control room emergency ventilation system provides two main functions. These functions are separated into two systems in the Technical Specifications and identified as the Control Room Emergency Ventilation System (CREVS) and the Control Room Emergency Air Temperature Control System (CREATCS). The CREVS provides a protected environment from which occupants can control the unit following an uncontrolled release of radioactivity, hazardous chemicals, or smoke. The CREATCS provides temperature control for the control room following isolation of the control room. The control room emergency ventilation system or CREVS as described in UFSAR refers to the complete system, which supports both functions.

9.4.1.2 System Description

The system, shown in Figure 9.4-1 and Figure 9.4-7, consists of two parts:

a. The control room normal air conditioning system consists of redundant air-handling units with heating and cooling coils. Each air-handling unit has a prefilter, final filter, hot water preheat coil, and a cooling coil. One unit will be operating with the other unit available for manual actuation in the event of failure of the operating unit. The total design flow rate for each unit is 21,920 cfm.

b. The control room emergency ventilation system consists of two 100 percent-capacity (3300 cfm) redundant fan-filter assemblies. Each filter system includes a roughing filter, high-efficiency filter, and charcoal adsorber. A cooling coil and water-cooled condensing unit are provided for each system to provide suitable temperature conditions in the control room for operating personnel and safety-related control equipment. Two 100 percent-capacity redundant air-cooled condensing units are provided as a backup to the water-cooled condensing units. On high refrigerant head pressure, the Service water valve

closes and the refrigerant solenoid valves align the air-cooled condensing unit automatically.

The control room emergency ventilation system is capable of removing the heat loads listed in Table 9.4-2 while maintaining a temperature of 95°F or below in the control room and shift manager's office. The system is capable of filtering 3300 cfm of control room air in a recirculation mode or introducing up to 300 cfm of outside air and 3000 cfm of recirculated air into the control room after filtering. The system is capable of maintaining 0.125 inches w.g. positive pressure in the control room with an intake of 300 cfm outside air. The systems are designated as Seismic Class I and seismic failure is not considered credible.

To protect pieces of electronic equipment, thermostats and humidistats controlling space heating are set at maximums of 80°F and 60% relative humidity, respectively.

Layout of system equipment and air flow guidance ducts is shown on Figures 9.4-3, 9.4-4, and 9.4-5. The control room normal air conditioning system utilizes pneumatic and electro-pneumatic controls. Equipment that is designed for a safety-related function utilizes controls that are supplied from essential power sources.

Safety-related systems that are comprised of redundant components are provided with independent control systems to assure that the failure of any single component will not prevent the fulfilling of the design functions.

The description of major system components is given below:

a. Control Room Normal Air-Conditioning System

<u>Supply Fans</u>

Type	Centrifugal
Capacity, cfrn	21,920
Static pressure, in. w.g.	3.6
Brake horsepower, bhp	18.6
RPM	1029

<u>Motor</u>

Type	Induction
Horsepower rating, hp	25
Voltage, V	480
Enclosure	TEFC
Insulation class	B

<u>Heating Coil</u>

Type	Hot water
Capacity, Btu/hr	334,000

<u>Cooling Coil</u>

Type	Chilled water
Capacity, Btu/hr.	754,000

<u>Package Water Chiller</u>

Capacity, tons	79.7
Refrigerant	HFC-134a
Condenser	Air-cooled

<u>Computer Room HVAC Unit</u>

Capacity	75,900 Btu/hr
Chilled Water	15 gpm
AirFlow	3000 cfm

b. Control Room Emergency Ventilation System

<u>Fans</u>

Type	Centrifugal
Capacity, cfm	3300
Static pressure, in. w.g.	5.5
Brake horsepower, bhp	4.4
RPM	2752

<u>Motors</u>

Type	Induction
Horsepower rating, hp	5
Voltage, V	480
Enclosure	Drip-proof
Insulation class	B

<u>Prefilters</u>

Quantity per unit	4
Rated flow per filter unit, cfm	1000
Type	Replaceable
Media	Reinforced nonwoven fire-retardant cotton fabric
Average efficiency	70%
Rating basis	NIST (formerly NBS) dust-spot method
Rated pressure drop unloaded (in. w.g.)	0.20

HEPA Filters

Quantity per unit	4
Rated flow per filter unit, cfm	1000
Type	High efficiency, dry
Media	Glass fiber (water-proof, fire retardant)
Cell side material	Stainless steel
Face guards	4-mesh galvanized hardware cloth
Seal	High Viscosity fluid seal
Deficiency	99.95% with 0.3 micron diameter DOP
Rating basis	MIL-STD-282
Rated pressure drop unloaded, in. w.g.	1.0
Codes:	ASME AG-1-1997 (with exception of casing size) MIL-STD 282 UL-586

Charcoal

Quantity per unit	12
Rated flow per charcoal element, cfm	333.3
Type	Activated coconut shell, impregnated
Particle size	# 6 - # 18 sieve
Ignition temperature, °C	330 minimum
Charcoal per element, lb	43
Maximum moisture content, %	3
Gasketing material	ASTM D 1056, Gr. SCE-43
Casing	Type 304 stainless steel
Penetration %: (unused activated carbon)	0.1% maximum molecular iodine, at 40 fpm, 30°C and 95% relative humidity. 3% maximum methyl iodide, 30°C, 95% relative humidity
Retentivity, seconds	0.25 minimum
Rated pressure drop, in. w.g.	1.00 Maximum per bank
Air face velocity, fpm	42 approximate
Codes	ANSI N509-1980

Water-Cooled Condensing Unit

Capacity, ton	10
Refrigerant	R-12

Cooling Coil

Type	DX Coil
Capacity, Btu/hr	120,000

Air-Cooled Condensing Unit

Capacity, tons	10
Refrigerant	R-12

9.4.1.3 Safety Evaluation

Should a failure of both normal refrigerating units occur during warm weather and while the station is operating normally, the control room normal air conditioning systems can be utilized in an outside air mode. The air conditioning systems have 100-percent outside air capability.

The worst-case environment is based on the following assumptions:

a. Outside summer design conditions of 95°F dry bulb and 76°F wet bulb.

b. A temperature of 110°F in the surrounding auxiliary building.

c. Control room heat loads as listed in Table 9.4-2.

d. Control room and shift manager's office temperature of 95°F.

During normal operation, the control room emergency ventilation system is held on standby. Under normal operating conditions, the control room will be free of airborne radioactivity. A description of the methods utilized for airborne radioactivity monitoring is provided in Subsection 12.2.4. In the event of a LOCA, the control room normal air conditioning system is automatically shutdown by an SFAS signal. The control room normal air conditioning system is also automatically shutdown by a high radiation signal from the station vent radiation monitors. The control room emergency ventilation system fans are manually activated from the control room.

During emergency isolation of the control room, the normal supply and return fans are shut down automatically and all control room isolation dampers are closed to preclude the admission of airborne contaminants to the control room. The control room operator has manual controls for initiating the control room emergency ventilation system to ensure satisfactory control room conditions following an accident. Closure of the normal ventilation system dampers to accomplish isolation of the control room is described in Chapter 15. The control room emergency ventilation system filters have a total efficiency not less than 95 percent so that the limits of Criterion 19 of 1CFR50 Appendix A are met. The CREVS can either be operated in the recirculation mode or outside air intake mode. However, to minimize the unfiltered inleakage into the control room, the CREVS is operated in the outside air intake mode following a LOCA. See Section 15.4.6.4.

This enables the control room to be maintained at a one-eighth inch w.g. higher pressure than the other areas of the auxiliary building. Technical Specification Administrative Controls establish the Applicability Requirements to both allow temporary opening of the Control Room boundary and the associated controls of the temporary opening.

When the Control Room Emergency Ventilation System is manually started, the system will start in the water cooled condensing mode. The associated compressor will start and the service water valve feeding the condenser will automatically open. The system will be maintained in the water-cooled condensing mode. The manual isolation valves for the air cooled condenser will

be opened to align the system for standby operation prior to or after the system has been started. In the event of an earthquake, during which the Seismic Category II supply line from the lake to the intake canal is lost, Reactor shutdown and cooldown will be initiated. The Service Water will be recirculated to the forebay to preserve the water in the forebay. As the heat from the plant is rejected to the forebay the recirculated Service Water temperature in the forebay (the heat sink) would start rising. As a result of the rising Service Water temperature, the Control Room Emergency Ventilation System refrigerant head pressure would rise. When head pressure reaches the condenser switchover pressure setpoint, the system will automatically shift to the air-cooled mode. The air-cooled condenser is qualified for all postulated accidents except a tornado. A tornado and an earthquake are not postulated to occur simultaneously.

The control room operator has remote manual control for selecting the outside air damper positions to ensure satisfactory control room conditions following an accident.

The control room normal air conditioning system can be started manually in a recirculation mode under the following conditions:

a. The temperature in the control room is 100°F and is continuing to rise.

b. The radiation level in the atmosphere is too high to allow any outside air to enter into the control room.

c. The control room normal air conditioning system is operable.

The outside air dampers and exhaust air damper will be maintained closed and the control room isolation dampers will be opened before starting the control room normal air conditioning system in the recirculation mode.

Manual action will be required to turn off all nonessential control room and control cabinet room equipment, nonessential lighting, and part of the essential lighting to maintain the control room temperature at or below 110°F under the following conditions:

a. The control room temperature is approaching 110°F. This will take on the order of hours; therefore, the operator will have ample time.

b. The control room emergency ventilation system is functioning in a recirculation mode.

c. The control room normal air conditioning system is not operable.

d. The radiation or toxic contaminant concentration in the atmosphere is too high to bring in any outside air.

e. The control room has been isolated by a signal from the radiation detectors.

Therefore, if for any reason the control room temperature should reach 110°F and continue to rise during normal station operation, administrative procedure will require a station shutdown.

The control room integrity for leak-tightness is maintained by a leak-tight door with alarm in the vestibule on the turbine building side and a water/airtight door with alarm in the north wall on the auxiliary building side. An alarmed door also exists between the Turbine Building and the Shift

Supervisors Office. The details of the water/airtight door to the auxiliary building are shown in Figure 9.4-6.

The reactor protection system is designed for continuous operation in a room environment of 40°F to 110°F and up to 75 percent relative humidity. Since the reactor protection system will perform its design functions within seconds, and since the control room will not reach 110°F before substantial time has elapsed, it follows that the reactor protection system is not affected by loss of air conditioning and/or ventilation as postulated.

All portions of the safety features systems (not including the reactor protection system) that are located in the control room are designed to operate in ambient conditions of 110°F and 80 percent relative humidity.

The control room area ventilating and air conditioning equipment is located remote from the control area. Closure of the control room normal air conditioning system dampers to accomplish isolation of the control room is described in Chapter 15.

The discharge of smoke from the control room may be accomplished by manually overriding the smoke detectors and operating the control room normal air conditioning system in the 100-percent outside air mode. System flow rates appear in Subsection 9.4.1.2. Fire dampers are provided at ductwork penetrating through fire walls. Redundant portions of the system are operated on an alternating schedule to ensure constant operability.

A single-failure analysis has been made on all active and passive components of the control room emergency ventilation systems to show that failure of any single component will not prevent fulfilling of the design functions. This analysis is shown in Table 9.4-1.

An analysis of dose levels in the control room under accident conditions is included in Chapter 15.

Control Room entranceways and other openings to vital areas will be kept closed at all times, with the following exceptions: (1) during the passage of authorized personnel; (2) if required for maintenance to be opened, the entranceways and other openings will be controlled to prohibit the entrance of unauthorized personnel.

When CREVS is operating in the recirculation mode assuming 1/8 inch w.g. pressure differential across all leak paths and the maximum operation pressure differential across dampers upstream of active fans, the infiltration rate is as follows:

Source	In-leakage (CFM)
Dampers, Ductwork	1 (10 pairs @ 0.1 cfm ea.)
Control Room Doors	6 (3 doors@ 2 cfm ea.)
Electrical Penetrations	Negligible
Piping Penetrations	Negligible
Floor-Wall-Roof Joints and Cracks	Negligible
Ingress and Egress	Negligible

There are 20 isolation dampers in the control room normal air conditioning system with a maximum leakage rate of 0.10 cfm per damper. Dampers are installed in series, in pairs of 2.

All electrical, piping, and ductwork penetrations are sealed to preclude in-leakage to the control room.

9.4.1.4 Tests and Inspection

All components of control room normal air conditioning and control room emergency ventilation systems are accessible for periodic inspection and maintenance.

The control room emergency ventilation system prefilters are of the type which exhibit an average efficiency of 70 percent dust-holding capacity when tested with the National Institute of Standards and Technology (formerly National Bureau of Standards) test using a mixture of Cottrell precipitate and lint.

The HEPA filters are of the type which meet the requirements of Appendix FC-I of AG-1-1997.

The HEPA filters are shop and acceptance tested for the efficiency-penetration test with homogeneous particles of dioctyl phthalate (DOP) in accordance with MIL-STD-282. The filters are shop tested to measure the pressure drops across filter banks. The filters have a removal capability of 99.95 percent of the DOP smoke when shop tested with 0.3 micron diameter DOP. Also, the filters are in-place tested with DOP. Less than 1% penetration and bypass leakage of DOP by each entire HEPA filter unit shall constitute acceptable performance for in-place testing.

The charcoal adsorbers are initially shop performance tested with methyl iodide tracers for efficiency and freon for bypass leakage. The adsorbers have been inspected for potential leakage and were found to be in satisfactory condition. The screens and all housings are welded, rather than riveted, to produce separation; no cell or housing deformation that could cause loss of charcoal or channeling was observed.

The fans are statically and dynamically balanced. Fan ratings are in accordance with AMCA Standard Test Code 211-A.

The control room emergency ventilation system was given a preoperational test prior to startup as follows:

a. Fans and air-cooled and water-cooled condensing units were operated continuously for at least one hour, and all related louvers and other mechanical components were proven operable.

b. The HEPA filter banks were in-place tested at design flow for efficiency penetration with homogenous particles of dioctyl phthalate (DOP).

c. The charcoal adsorber banks were tested in place at design flow with Freon-112 or equivalent) to ensure that there is less than one percent penetration and bypass leakage across the filter banks and that the charcoal adsorber elements are not damaged. Carbon samples are analyzed as required by an independent laboratory to determine remaining charcoal adsorber life and replacement requirements in accordance with station Technical Specification requirements.

d. The ductwork was tested for leakage.

e. The system were balanced, adjusted, and tested for performance.

The system was tested to demonstrate its capability to maintain a minimum of one-eighth inch w.g. positive pressure across the control room pressure boundary, using a pressurization flow rate of 300 cfm. This testing supports the control room envelope habitability analysis. The free volume of the control room and the control room envelope air exchange rate are utilized to evaluate the conditions following a radiological event, hazardous chemical release, or a smoke challenge to assess the acceptability of continuous occupation of the control room.

Periodic testing is covered by Technical Specification Surveillances.

The control room emergency ventilation system will be periodically tested to ensure that it will perform its design function. These tests will include visual inspection, pressure drop measurements across each filter component, flow rate verification, filter efficiency, and filter bypass checks. In addition, a test signal will be applied to demonstrate proper shutdown of the control room normal air conditioning system and manual actuation of the control room emergency ventilation system. Fans and water-cooled condensing units will be operated and all related louvers and other mechanical components will be proven operable as described in the station Technical Specifications.

9.4.1.5 System Interrelationships

The design ventilation capacities required for the control room, equipment and cable spreading room ventilation systems, including flow rates and cooling requirements, are given below:

Room	Flow Rates (cfm)	Design Cooling Loads (Btu/hr)
Cable Spreading Room	2,640	57,044
Computer Room	1,260	27,580
Control Cabinet Room	11,550	249,910
Control Room	4,100	88,710

The heating coil in the control room normal air-conditioning unit is sized to provide 334,000 Btu/hr heating to the control room area. With this heating capacity, the required ambient conditions in the control room are maintained.

9.4.2 Auxiliary Building

9.4.2.1 Nonradwaste Areas

9.4.2.1.1 Design Bases

The heating and ventilating systems for the nonradwaste areas are designed to provide a suitable environment for equipment and personnel. The following sections give environmental conditions maintained in each room.

The systems are generally designed in accordance with the recommended practices of the American Society of Heating, Refrigerating, Ventilating, and Air Conditioning Engineers (ASHRAE) Guide, the Air Moving and Conditioning Association, and the National Fire Protection Association (NFPA). NFPA deviations are identified in DBNPS Correspondence to NRC dated July 31, 1989 (Serial Number 1685). The ductwork is designed and fabricated in accordance with the Sheet Metal and Air Conditioning Contractors National Association (SMACNA) standards.

The ASHRAE outdoor air temperature philosophy stipulates that short term excursions higher than design can occur for 1% of the time in a year (approximately 30 hours). Rooms containing the required equipment that could potentially be susceptible to short term (approximately 24 hours as often as once a year) elevated outdoor temperature were evaluated. It was determined that equipment, with the exception of the Emergency Diesel Generators (EDGs), would continue to perform required functions at the outdoor temperature of 104°F. The EDGs were evaluated for short term temperature excursions resulting from exceeding the outdoor design temperature of 95°F. The evaluation concluded that the EDGs will continue to function at room temperatures up to and including 131°F which corresponds to an outdoor temperature of 101.9°F (Train 1) and 103.9°F (Train 2) at 110% EDG loading.

Seismic classification is discussed below.

9.4.2.1.2 System Descriptions

9.4.2.1.2.1 Battery Rooms

Each battery room receives ventilation air from its respective low voltage switchgear room through a transfer grill and is continuously exhausted through duct work by roof mounted non-safety-related battery room exhaust fans. The exhaust fan in each room and its associated duct work is non-seismic. Fans are energized from hand indicating switches and are designed to run continuously to maintain the room temperatures and to purge the hydrogen and oxygen gas in the room generated by the battery charging. Additionally, two non-safety related electric heaters are provided in Battery Room B to add heating capability to the room during the winter months. A local temperature switch automatically starts each battery room electric heater at predetermined low temperature setpoint. Due to the relatively small size of the battery rooms, low normal mode room heat loads, and the comparatively large outside wall area, a continuous supply of ventilation air from the low voltage switchgear rooms may be required to maintain the battery rooms above 60°F during severe winter weather. The HVAC systems are sized to maintain the battery rooms between 60°F and 104°F year-round during all modes of operation.

One safety-related battery room ventilation fan is provided in each battery room to exhaust the room following a loss of off-site power, a postulated accident or failure of the normal, non-safety-related exhaust fans. The safety-related fan motor and exhaust damper are supplied from an essential motor control center to ensure an uninterrupted power supply. This portion of the system is Seismic Class I. Low flow switches, located in the fan exhaust duct work, alarm through the control room annunciator and computer if neither operating fan (one fan per battery room) is energized, or if the energized fan fails to provide the required air flow. A temperature switch automatically starts each safety-related battery room ventilation fan at predetermined high temperature setpoint. Each safety-related ventilation fan can be started manually from its local control station to purge hydrogen and oxygen gas following a failure of the normal fan or loss of offsite power when normal ambient temperatures and room heat loads may not be sufficient to activate the high temperature switch.

9.4.2.1.2.2 Low Voltage Switchgear Rooms

The non-radwaste area ventilation system serving the low voltage switchgear rooms utilizes recirculation of return air along with outside air for all normal modes of operation.

The low voltage switchgear ventilation system consists of non-radwaste area supply and return fan, two safety-related low voltage switchgear room ventilation fans, three safety-related motor

operated outside air dampers, two safety-related exhaust dampers and associated controls and duct work. The safety-related portion of the system is designated as Seismic Class I. The non-radwaste area ventilation system is non-seismic.

The normal ventilation system consisting of non-radwaste fans operates continuously through temperature controllers which modulates supply, return and exhaust dampers to maintain the average temperature in the non-radwaste areas between 60°F and 104°F for all normal modes of operation.

Safety-related 100% capacity low voltage switchgear room ventilation fans are provided to ensure adequate cooling of the low voltage switchgear room following a loss of off-site power, postulated accident or failure of normal ventilation system. Each ventilation fan is started automatically by a temperature switch at a predetermined temperature setpoint which simultaneously opens outside air supply louvers and exhaust air dampers. Each safety-related ventilation train is designed to maintain its low voltage switchgear average room temperature between 60°F and 104°F year-round during all modes of operation, including post-accident.

An electrical isolation room is located within each of the low voltage switchgear rooms. Normal ventilation is maintained by the same non-radwaste area ventilation system as described above. Following a loss of off-site power, postulated accident or failure of normal ventilation system, these rooms are indirectly served by the safety-related low voltage switchgear room ventilation through open doorways. The average temperature of the rooms is normally maintained between 60°F and 104°F. During loss of ventilation, the room temperature may exceed 104°F. The equipment located in the low voltage switchgear and electrical isolation rooms can remain operable at higher temperatures for limited periods of time. Administrative controls shall be taken under these circumstances and if safety-related or non safety-related ventilation systems are inoperable.

A smoke detector is provided for the main supply duct. Fire dampers are provided at ducts through fire walls.

9.4.2.1.2.3 Emergency Diesel Generator Rooms

The Emergency Diesel Generator (EDG) Room Ventilation System consists of two (2) safety-grade, 50% capacity supply air fans in each EDG room. The system fans, duct work and temperature controls are designated as Seismic Class I. The fans are started automatically when the respective diesel engine is started. The ventilation systems are sized to provide adequate outside air cooling to maintain each "operating" diesel generator room at 125°F assuming 95°F outside air and EDG operating at 110% load. Combustion air for the diesel engine is piped from a roof-top mounted, missile-protected air cleaner and is not dependent upon operation of the ventilation system. Each ventilation system includes safety-grade modulating supply, return and exhaust air dampers which are interlocked through room temperature controllers. The dampers automatically modulate to maintain the room temperature between 60°F and 125°F for all operating conditions. The supply and exhaust air dampers fail closed, and the return air damper fails open, to prevent freezing temperatures in the EDG Room.

The EDGs were evaluated for short term temperature excursions resulting from exceeding the outdoor air design temperature of 95°F. The evaluation concluded that EDGs will continue to function at room temperatures up to and including 131°F.

Winter heating in the EDG room is accomplished by means of an electric heating system. This heating system is non-seismic.

9.4.2.1.2.4 Auxiliary Feedwater Pump Rooms

The auxiliary feedwater (AFW) pump room ventilation system consists of one 100% capacity, safety-related exhaust fan, and temperature switch in each room. This ventilation system is designated as Seismic Class I. Missile protected supply air transfer grills are provided in the turbine building floor slab and exhaust air is discharged from each room through similar openings so as to preclude short circuiting of supply and exhaust air. Each exhaust fan is started automatically by its pump room temperature switch at a predetermined temperature setpoint and is sized to maintain its pump room between 60°F and 120°F during all modes of operation including post accident, utilizing supply air from the turbine building at ≤ 110°F.

9.4.2.1.2.5 Component Cooling Water Pump Rooms

The component cooling water (CCW) pump room ventilation system consists of safety-related and non-safety-related systems.

The safety-related (Seismic Class I) system provides two 100% capacity CCW pump room ventilation fans, and electrohydraulic actuator operated exhaust and recirculation dampers. Electrohydraulic actuator operated supply air louvers are not safety-related, but they are Seismic Class I. Safety-related cooling and ventilation is ensured by one of these two 100% capacity safety-related CCW pump room ventilation fans. The temperature switch starts one of the ventilation fans at a predetermined temperature setpoint and opens the motor operated outside air louvers. The temperature element modulates exhaust and recirculation air dampers through temperature indicating controllers. The fan shuts off when the temperature in the CCW pump room drops below a predetermined setpoint. This system is designed to maintain the CCW pump room temperature between 60°F and 104°F year-round. The CCW pump room was evaluated with one fan operating, short term temperature excursions of the outdoor design temperature exceeding 95°F and with outdoor air flow reduced by the Elevator Equipment room fan running. Discussion of using an outside air temperature exceeding 95°F is included in Section 9.4.2.1.1. Results show that the room temperatures are less than 120°F, which is the temperature that the equipment in the CCW pump room can operate for short excursions, with no reduction in equipment reliability or plant safety.

The non-safety-related (non-seismic) system consists of the Elevator Machinery Room Exhaust Fan, which is kept normally shutdown with its damper closed. This restriction is administratively applied to prevent drawing steam laden air into the CCW pump room in the event of a high energy line break in the turbine building. The supply air for this fan is drawn from the turbine building through a transfer grill located in the north elevator machinery room wall and exhausted into the CCW pump room. The Turbine Building Roof Exhaust Fan (non-safety related and non-seismic) suction flow path from the CCW pump room has been isolated by installation of a steel plate. The steel plate is installed to isolate the CCW pump room from the turbine building in the event of a high energy line break in the turbine building.

9.4.2.1.2.6 High Voltage Switchgear Rooms and Auxiliary Shutdown Panel Room

The non-radwaste area ventilation system is the only system providing ventilation to the High Voltage Switchgear Rooms and Auxiliary Shutdown Panel Room (rooms 323, 324, and 325). With the ventilation system in operation, the rooms will be maintained between 60°F and 104°F with design outside conditions. These rooms contain safety related equipment required to

operate under accident conditions. Following a loss of off-site power, postulated accident or failure of the non-safety grade ventilation system, calculations have shown that the ambient temperature in these rooms will not exceed 120°F under extreme (higher than design) outside air temperature conditions. It is determined that the equipment in these rooms will safely and reliably perform their safety function for room temperatures up to 120°F.

9.4.2.1.3 Safety Evaluation

A single failure analysis has been performed on all active and passive components of the non-radwaste area ventilation systems. Safety-related systems that are comprised of redundant components are provided with independent control systems to assure that the failure of any single component will not prevent the fulfilling of the design functions. Safety evaluations for each system is shown below.

9.4.2.1.3.1 Battery Rooms

The following considerations were made regarding the potential for buildup of explosive mixtures of hydrogen and oxygen in the battery rooms. The maximum rate of H_2 generated from each 60 cell-125 Volt NCX 1500 (or equivalent) battery at equalizing voltage (up to 2.39 volts per cell) is 0.013 CFM. This would amount to 0.026 CFM in each battery room. On normal float charge (approximately 2.2 volts per cell), the normal generation rate is approximately 0.0022 CFM per room. Exhaust fans located close to the ceiling draw supply air into each room through a transfer grill. The introduction of a continuous supply of air into the room will suppress the volume fraction of gases given off by the batteries in the room and will avoid the possibility of buildup of explosive mixtures in the room. The coefficient of diffusion of H_2 in air is 0.634 cm^2 per second, compared to the coefficient of diffusion of water vapor, which is 0.239 cm^2 per second. This indicates that even without agitation, considerable mixing will take place.

Loss of all forced ventilation to either battery room (429B or 428A) may result in an increased hydrogen concentration in the room. Room 428A is fitted with a solid door which is normally open and a wire mesh door. Due to free air exchange through this door and through a transfer grill into the much larger room 428, significant hydrogen accumulation in room 428A (battery room “B”) should not occur. Loss of forced ventilation to room 429B (battery room “A”) for an extended period may result in hydrogen concentrations approaching the administrative limit of 2 volume percent for a hydrogen/air mixture. When on float charge, this would require over two weeks with no credited air exchange. Additional time would be required to reach the combustible limit of 4 percent. Air exchange associated with normal room access or the transfer grill would extend these periods. Due to the very low combustible gas buildup rate, operability of the affected battery train is not immediately impacted, even with no forced ventilation available. Precautions are taken to ensure that each battery room has either a forced ventilation system operating in manual or that the battery room door is opened to provide a flow path. (If door 429 to room 429B is temporarily left opened, additional compensatory action may be required by plant procedures due to impairment of barrier functions.) In addition, equalize charging is normally terminated if no continuous forced ventilation system is operating.

Since the battery room ventilation systems are also designed to maintain a suitable room temperature year-round, loss of the safety grade ventilation system in either room may, in extremely cold or hot weather, result in inability to tolerate an additional single failure. Therefore, the affected component(s) should be replaced or repaired as soon as practical. The affected battery train could still provide power during a station blackout condition. However, the battery capacity and voltage characteristics are not assured when the initial battery electrolyte

temperature is below 60°F or above 104°F. In addition, without a forced ventilation fan operating in manual, the batteries should not be recharged at “equalize” voltage following a station blackout condition. See single failure analysis in Table 9.4-7.

9.4.2.1.3.2 Low Voltage Switchgear Rooms

Loss of either low voltage switchgear room ventilation train will result in the system’s inability to tolerate an additional single failure and the affected component(s) should be repaired or replaced as soon as possible. The affected switchgear can remain operable once the associated room temperature exceeds 104°F for limited periods of time, when administrative controls are undertaken. The analysis is shown in Table 9.4-3.

9.4.2.1.3.3 Emergency Diesel Generator Rooms

In case of a failure of one safety-related emergency diesel generator EDG room ventilation system (two 50% capacity fans), a full capacity safety-related ventilation system is available for redundant EDG to maintain its room temperature between 60°F and 125°F, assuming outside air at 95°F (at Meteorological Tower) or less.

Based on the design basis heat load for the EDG rooms, a supply of outside air temperature of 72.7°F (Train 1) and 76.8°F (Train 2) or less will enable one EDG room ventilation fan to maintain the room design temperature of 131°F for all operating conditions, provided that there is no flow of hot room air through the fan which is out of service. Reverse flow through the fan which is out of service could cause excessive recirculation of hot room air through the operable fan which would result in reduced cooling. Reverse flow through an inoperable fan is prevented by placing a temporary blanking plate on the discharge of the inoperable fan. A single failure analysis is presented in Table 9.4-4.

The EDGs were evaluated for short term temperature excursions resulting from exceeding the outdoor air design temperature of 95°F. The evaluation concluded that EDGs will continue to function at room temperatures up to and including 131°F.

9.4.2.1.3.4 Auxiliary Feedwater Pump Rooms

Loss of either auxiliary feedwater (AFW) pump room ventilation system will result in the system’s inability to tolerate an additional single failure and the affected components should be repaired or replaced as soon as possible. Due to the small size of the AFW pump rooms and the magnitude of the room heat loads, these rooms cannot be maintained indefinitely at or below 120°F following a failure of their respective ventilation system, regardless of the turbine building air temperature. See single failure analysis in Table 9.4-9.

9.4.2.1.3.5 Component Cooling Water Pump Rooms

Loss of either safety-related train of component cooling water (CCW) pump room ventilation will result in the system's inability to tolerate an additional single failure, and the affected system component(s) should be repaired or replaced as soon as possible. Due to the magnitude of the CCW pump room heat load and the limited exterior wall area available for heat transfer, the CCW pump room cannot be maintained indefinitely at or below 104°F following a loss of both redundant trains of safety-related ventilation, regardless of outside air temperature. A single failure analysis is presented in Table 9.4-8.

9.4.2.1.4 Tests and Inspections

The systems were given a preoperational test before the station produced power. The components of the ventilating systems are accessible for periodic inspection and maintenance.

9.4.2.2 Fuel-Handling Area

9.4.2.2.1 Design Bases

The ventilation system for the fuel-handling area is independent of that used in any other areas and is designed on a once-through basis to control and direct all potentially contaminated air to the station vent stack via roughing and high-efficiency particulate air filters. Exhaust air from the fuel-handling area is monitored before it is discharged from the station through the vent stack. Redundant exhaust fans are provided in this area.

The fuel-handling ventilation system is designed to provide an average of 20 air changes per hour across the surface of the spent-fuel pool. Summer outside design conditions are 95°F DB and 76°F WB and in winter are -10°F DB. The fuel-handling area is maintained at between 60 and 110°F. The system was designed on the basis that the fuel-handling area is potentially radioactive.

Based on the re-racking of the spent fuel pool, an evaluation of the ventilation system (Holtec Report No. HI-992221) was performed assuming the maximum pool bulk temperature (conservatively assumes 1714 assemblies). The evaluation concluded the building air temperature in the vicinity of the pool will be maintained less than or equal to 110°F.

The systems and equipment are designed in accordance with the recommended practices of the American Society of Heating, Refrigerating, Ventilating, and Air Conditioning Engineers Guide, the Air Conditioning Association, and the National Fire Protection Association. The ductwork is designed and fabricated in accordance with the Sheet Metal and Air Conditioning Contractors National Association (SMACNA) standards.

9.4.2.2.2 System Description

The fuel-handling area ventilation system consists of a supply-air unit and redundant exhaust fans. The supply-air unit provides 100 percent outside air without a recirculation mode. The supply air to the area is heated by a hot water heating coil to maintain a temperature of 60°F to 70°F in the space with a design outside temperature of -10°F dry bulb. The system is shown in Figure 9.4-9.

The fuel-handling area filter consists of prefilters and HEPA filters. The fuel-handling area is connected to the emergency ventilation system filters by means of ductwork bypasses and dampers. During normal operation, the exhaust from the fuel-handling area is passed through the fuel-handling area exhaust filter and discharged through the station vent stack.

Two redundant radiation detectors powered from essential sources, which are simple ionization chambers, are installed close to the suction end of fuel handling area ducting.

During normal operation, the dampers on the upstream side of the emergency ventilation system (EVS) filters (CV5024 and CV5025) are open. In the event of a fuel-handling accident, the radiation detectors would send signals to the essential solenoid valves and to the EVS fans. The essential solenoid valves would cause dampers on the suction and discharge side of

fuel-handling area supply and exhaust fans to close (this would cause supply and exhaust fans to stop) and the dampers in the bypass ducting (HV5430A and HV5430B) to open. This action will ensure that the fuel-handling area supply and exhaust ducting are isolated and the EVS fans are started automatically to pull a negative pressure in the fuel handling area. This will ensure the accident doses at the site boundary will be well below the 10CFR100 guidelines.

When the containment equipment hatch is open, the inside of the containment vessel becomes part of the fuel handling area negative pressure boundary. With the equipment hatch open and both doors of the personnel air lock open, fuel handling area negative pressure boundary integrity is considered to exist, provided that at least one air lock door is capable of being closed and a designated individual is available immediately outside the personnel air lock to close the door (Reference 7, Letter Serial 2690). Hoses or cables running through the containment personnel air lock must employ a means to allow safe, quick disconnect or severance, and be tagged at the air lock with specific instructions to expedite removal during crane operation with loads over the spent fuel pool, cask pit, or transfer pit or operations involving movement of fuel in the spent fuel pool, cask pit, or transfer pit, with the equipment hatch and personnel air lock open (Reference 8, Letter Serial 2756). Following a fuel handling accident with the containment equipment hatch open, the containment purge system is shutdown and containment isolation dampers CV 5005, CV 5006, CV 5007, and CV 5008 are closed to establish fuel handling area negative pressure boundary integrity.

The FDS/RMS console and printer will provide alarm indication in the main control room on high radiation in the fuel handling area.

The connections between the emergency ventilation system and this system are automatically closed by the safety features actuation system for post-LOCA operation.

The supply and exhaust fans are returned manually as soon as the radioactivity levels in the area are within acceptable limits.

9.4.2.2.3 Safety Evaluation

Air is exhausted from the fuel-handling area at a greater rate than it is being supplied to maintain the area under a negative pressure and thus ensure that all leakage is into the fuel-handling area rather than out of it.

The system is manually started from the local control panel. The system logic requires one exhaust fan to be operating prior to initiation of the supply-air unit to ensure negative pressure within the area. Redundant exhaust fans are provided for this area. One exhaust fan will be operating with the other fan available for manual actuation in the event of failure of the operating fan.

Loss of both exhaust fans stops the supply fan.

Smoke detectors are provided in the return duct and main supply duct in compliance with the requirements of NFPA 90A. Fire dampers are provided at penetrations through firewalls.

The system is designed so that in the event of a fuel handling accident resulting in release of radioactivity, the fuel handling area supply and exhaust units will be stopped and the exhaust of the fuel handling area will be automatically transferred to the emergency ventilation system. A failure modes and effects analysis of components in the fuel handling area ventilation system is provided in Table 9.4-5. As shown in Table 9.4-5, the system contains redundant components

such that a single failure of an active component (i.e., one that is required to change its position to accomplish the safety function) will not prevent the fulfillment of system functional capability.

9.4.2.2.4 Tests and Inspections

The prefilters are of the type which exhibit an average efficiency of 70 percent dust holding capacity when tested with the National Institute of Standards and Technology (formerly National Bureau of Standards) test using a mixture of Cottrell precipitate and lint.

The HEPA filters are of the type which meet the requirements of Appendix FC-1 of AG-1-1997.

The HEPA filters are shop and acceptance tested for the efficiency-penetration test with the homogeneous particles of dioctyl-phthalate (DOP) in accordance with MIL-STD-282. The filters are shop-tested to measure the pressure drops across them at rated flow. The filters are also in-place tested with DOP. The filters have a removal capability of 99.97 percent of the DOP smoke.

The fans are statically and dynamically balanced. Fan ratings are in accordance with AMCA Standard Test-Code 211-A. The ductwork is leak-tested, and the entire systems are balanced, adjusted, and tested for performance.

The systems were given a preoperational test. The components of the systems are accessible for periodic inspection.

A discussion of application of instrumentation is given in Subsection 12.2.4.

9.4.3 Radwaste Area

9.4.3.1 Design Bases

The ventilation system for the radwaste area is independent of that used in any other areas and is designed on a once-through basis to control and direct all potentially contaminated air to the station vent stack via roughing and high-efficiency particulate filters. Exhaust air from the radwaste area is monitored before it is discharged from the station through the vent stack. Redundant exhaust fans are provided for this area.

The radwaste ventilation system is designed to maintain an average of five air changes per hour within the potentially radioactive areas of the auxiliary building. The maximum design temperature in these areas is 110°F. During winter, the air is tempered to maintain a minimum auxiliary building temperature of 60°F. Chemistry laboratories and health physics monitor areas in the auxiliary building are air conditioned to 75°F dry bulb. The maximum design temperature in the emergency core cooling system rooms is 104°F. The electrical penetration room number one is maintained at a minimum temperature of 60°F in winter and 75°F maximum in summer.

Summer outside design conditions are 95°F DB and 76°F WB and in winter are -10°F DB.

The calculated rates of heat gain, both sensible and latent heat loads, include the anticipated loads due to heat sources within the conditioned space such as people, lighting, power equipment, etc.

The ventilating systems and equipment are generally designed in accordance with the recommended practices of the American Society of Heating, Refrigerating, Ventilating, and Air

Conditioning Engineers Guide, the Air Moving and Conditioning Association, and the National Fire Protection Association (NFPA). NFPA deviations are identified in DBNPS correspondence to NRC dated July 31, 1989 (Serial Number 1685). The ductwork is designed and fabricated in accordance with the Sheet Metal and Air Conditioning Contractors National Association (SMACNA) standards. The maximum settings on all equipment which has space heater control is: 80°F thermostat.

9.4.3.2 System Description

The radwaste area ventilation system is required for building ventilation during station operation and during shutdown operation. The system consists of a supply-air unit and redundant exhaust fans. The supply-air unit distributes fresh outside air to the potentially contaminated areas at all levels of the auxiliary building. The unit provides 100 percent outside air without a recirculation mode. A hot water heating coil is provided for tempering outside air when required.

The radwaste area exhaust filter consists of prefilters and HEPA filters. The radwaste area is connected to the emergency ventilation system filters by means of ductwork bypasses and dampers. During normal operation, the exhaust from the radwaste area is passed through the radwaste area exhaust filter and discharged through the station vent stack. In the event radioactivity levels exceed acceptable limits, the supply and exhaust fans are stopped and the ducting from the radwaste area to the emergency ventilation system is opened automatically. The radiation-monitoring system will provide alarm indication in the main control room on an alarm printer. The dampers on the upstream side of the emergency ventilation system filters are normally open and are operable from the control room. The emergency ventilation system is initiated manually by the operator in the control room to channel the exhaust air through the emergency ventilation system HEPA and charcoal filters to reduce the doses at the site boundary well below the 10CFR100 guidelines. The connections between the emergency ventilation system and the radwaste area ventilation system are automatically closed by the safety features actuation system for post-LOCA operation.

Ventilation air required for the RRA entrance area beyond the PSF passage way is supplied by the radwaste ventilation system supply fan. A booster supply fan and heating and cooling coils are provided for comfort heating and cooling of the area. Exhaust air from the hot and cold laboratories and health physics monitor areas is passed through the hood exhaust filter system which consists of roughing filter, HEPA filter, and charcoal adsorber banks. Air from the RRA entrance area beyond the PSF passage way is exhausted without recirculation.

The refrigeration chiller units provided for the control room provide the chilled water required for cooling the electric penetration room number one on El. 603' level and the laboratory and health physics monitor areas in the auxiliary building. Hot water heating coils and electric heating coils are provided to maintain a temperature of 60 to 75°F in these spaces. The makeup pump room has an independent cooling unit which is designed to prevent the room temperature from exceeding 75°F.

The Emergency Core Cooling System (ECCS) rooms cooling units maintain a suitable environment for the electric-motor drivers of high-pressure injection pumps, decay heat pumps, and containment spray pumps. Each cooling unit consists of a fan and cooling coil. The cooling units are designated as Seismic Class I. Room air is circulated over the water-cooling coils by the fans and discharged back into the room. The cooling units are automatically energized by an increase in the room temperature.

The system is shown in Figure 9.4-9.

9.4.3.3 Safety Evaluation

Air is exhausted from the radwaste area at a greater rate than it is being supplied to ensure that the radwaste area is maintained under a negative pressure with respect to the clean areas and that leakage will be from the clean areas to the potentially contaminated areas.

The radwaste area ventilation system is manually started from the local control panel. The system logic requires one exhaust fan to be operating prior to initiation of the supply air unit to ensure a negative pressure within the auxiliary building. Redundant exhaust fans are provided for this area. One exhaust fan operates normally with the other fan available for manual actuation in the event of failure of the operating fan. Loss of both exhaust fans stops the supply fan. A single failure of any component of the system will not result in damage to any safety-related systems.

Redundant equipment is operated on an alternating basis to provide continuing assurance of operability.

Fire dampers are provided at ductwork penetrations through fire walls.

Smoke detectors used in the ventilation systems are self-contained units arranged for mounting directly on the ventilation ducts. The units are located in the return air ducts prior to exhausting from the building or being diluted by outside air and in the main supply ducts on the downstream side of the filters. The units are designed to operate into the ventilation system controls to provide safe shutdown of the ventilation system, actuation of alarms, etc. The unit includes alarm contacts, trouble contacts, and an alarm test switch, housed in a sheet metal cabinet with a rubber gasket to seal the detector housing to the duct.

For post-accident conditions, the leakage assumptions used for areas housing ECCS equipment are discussed in Subsection 15.4.6.5. The leakage considered is essentially that release which would result from a pipe rupture. As discussed in Subsection 15.4.6.5, the emergency ventilation system is capable of handling any potential radioactive iodine release to the penetration rooms.

Only a few of the tanks in the radwaste system are considered potential sources of significant quantities of radioactive materials. A description of the tanks and a discussion of the quantities of gas that could be released in the event of a tank rupture are provided below:

Waste Gas Decay Tanks

There are three waste gas decay tanks, each of which can contain up to 1013 cubic feet of gas at 150 psig pressure. Normally, all three tanks would not be full at the same time. The tanks are Q-listed.

Waste Gas Surge Tank

The waste gas surge tank can contain up to 1026 cubic feet of gas at 15 psig pressure. Normally, the gas pressure within the tank is about 5 psig. The tank is Q-listed.

Clean Waste Receiver Tanks

There are two clean waste receiver tanks, each of which can contain up to about 16,400 cubic feet of gas at 15 psig pressure. The tanks are interconnected by a normally open equalization line, and rupture of one tank would probably vent both tanks. It should be noted that the clean waste receiver tanks normally operate at a pressure less than 15 psig. When the pressure approaches the 15 psig upper limit, the vapor space in the tanks is reduced due to the presence of liquid.

Clean Waste Monitor Tanks:

There are two clean waste monitor tanks, each of which can contain up to 3,230 cubic feet of gas at 15 psig pressure. The tanks are interconnected by a normally open equalization line. All comments made on the operation of the clean waste receiver tanks are applicable in this case also.

The exhaust system ductwork in the vicinity of the tank is subject to possible rupture or collapse in the event of a pressure pulse caused by a violent rupture of one or more of the tanks in the radwaste system. Extensive damage to the radwaste building exhaust system ductwork, however, is limited primarily to the immediate vicinity of the ruptured tank with the remaining portions of the system still in operation. The contaminants from the ruptured tank may enter into the adjoining areas, which are within radwaste exhaust system boundary, through the spaces left by the damaged ductwork and other openings.

Physical separations are provided to prevent contaminants from entering the control room, cable spreading room, emergency diesel generator room, switchgear rooms, and other adjoining clean areas of the auxiliary building.

Exhaust air from the radwaste area is monitored before it is discharged from the station through the vent stack. In the event radioactivity levels exceed acceptable limits, the supply and exhaust fans are stopped and the ducting from the radwaste area to the emergency ventilation system is opened automatically. The emergency ventilation system is initiated manually by the operator in the control room to channel the exhaust air through the emergency ventilation system HEPA and charcoal adsorbers.

Air is exhausted from the radwaste area at a greater rate than it is being supplied to ensure that the entire radwaste area is maintained under a negative pressure, thus preventing contaminants from being delivered to other areas of the auxiliary building.

A single failure analysis has been made on components of the Emergency Core Cooling System (ECCS) room cooling units to show that failure of any single component as shown in Table 9.2-3 and Table 9.4-6 will not prevent fulfilling of the design functions.

9.4.3.4 Tests and Inspections

The prefilters are of the type which exhibit an average efficiency of 70 percent dust holding capacity when tested with the National Institute of Standards and Technology (formerly National Bureau of Standards) test using a mixture of Cottrell precipitate and lint.

The HEPA filters are of the type which meet the requirements of Appendix FC-I of AG-1-1997.

The HEPA filters are shop and acceptance tested for the efficiency-penetration test with the homogeneous particles of dioctyl-phthalate (DOP) in accordance with MIL-STD-282. The filters are shop-tested to measure the pressure drops across them at rated flow. The filters are also in-place tested with DOP. The filters have a removal capability of 99.97 percent of the DOP smoke.

The fans are statically and dynamically balanced. Fan ratings are in accordance with AMCA Standard Test-Code 211-A. The ductwork is leak-tested, and the systems are balanced, adjusted, and tested for performance.

The systems were given a preoperational test. The components of the systems are accessible for periodic inspection.

9.4.3.5 Instrumentation Application

A discussion on the application of instrumentation is given in Subsection 12.2.4.

9.4.4 Turbine Building

9.4.4.1 Design Bases

The heating and ventilating systems for the turbine building are designed to provide a suitable environment for equipment and personnel. Airborne radioactivity levels inside the turbine building are not significant. The exhaust air from this area is released into the atmosphere without any special treatment.

The turbine building ventilation system is designed to maintain the turbine building environment between 60 and 110°F.

Summer outside design conditions are 95°F DB and 76°F WB. The design condition for winter is -10°F DB.

The ventilating systems and equipment are designed in accordance with the recommended practices of the American Society of Heating, Refrigerating, Ventilating, and Air Conditioning Engineers Guide, the Air Moving and Conditioning Association, and the National Fire Protection Association. The ductwork is designed and fabricated in accordance with the Sheet Metal and Air Conditioning Contractor National Association (SMACNA) standards.

9.4.4.2 System Description

Turbine building ventilation is provided by means of several heating and ventilating units and ventilation-only units which conduct air to motors and equipment. The air is recirculated in cold weather to conserve heat.

The supply air to the area it heated by hot water heating coils to maintain a temperature of 60 to 70°F in the space with an outside design temperature of -10°F dry bulb. Space heating is provided during station shutdown by the heating and ventilating units operating in a recirculation mode.

Summer conditions require 100 percent outside air to limit the temperature in the area to 110°F. Air is exhausted through the turbine building roof exhaust fans.

The turbine building ventilation system consists of 5 heating and ventilating units, 6 ventilating-only units, and 6 roof exhaust fans. The system is manually started by operator action at locally mounted control panels.

Unit heaters, outside-air louvers, and exhaust fans are provided in the circulating-water pump house and in the auxiliary boiler room area to maintain the environment between 60°F and 110°F. Exhaust air from the auxiliary boiler room is discharged into the turbine building. Air is exhausted from the circulating water pump house through roof exhaust fans. The systems are shown in Figure 9.4-10. The components of the ventilation systems are accessible for periodic inspection and maintenance.

9.4.4.3 Safety Evaluation

Smoke detectors are provided in the return duct and in the main supply ducts in compliance with the requirements of NFPA 90A. Fire dampers are provided at ductwork penetrations through fire walls. A single failure of any component in the system will not result in damage to any safety-related systems.

Smoke detectors and fire detectors are provided to detect unsafe conditions in the turbine building in accordance with the requirements of NFPA 90A. The units are located in the return air ducts prior to exhausting from the building or being diluted by outside air and in the main supply ducts on the downstream side of the filters.

Fire detectors are manual reset limit devices which break electrical contacts as the air temperature reaches the predetermined cut-out level.

Freeze protection thermostats are provided to detect malfunction of the turbine building hot water heating system.

Fire dampers or fire doors, as required, are provided at ductwork penetrations through fire walls. The units are spring loaded devices, held open by a fusible link, and designed to close when the surrounding air temperatures reach levels indicative of a fire.

Actuation of the smoke detectors, fire detectors, and/or freeze protection thermostats will de-energize the respective ventilation system fan and cause the system dampers to close.

9.4.5 Service Water Pump Room Ventilation System

9.4.5.1 Design Bases

The system is designed to maintain the service water pump room between 40°F and 104°F and the diesel-driven fire pump room between 40°F and 120°F year-round for all modes of operation including post-accident at design outside conditions.

The diesel-driven fire pump room will only exceed 104°F at design outside conditions when the diesel-driven fire pump is operating. The safety related equipment in the diesel driven fire pump room was evaluated for temperature excursions associated with diesel-driven fire pump operation and found acceptable.

The American Society of Heating, Refrigeration, Ventilating, and Air Conditioning Engineers (ASHRAE) outdoor air temperature philosophy stipulates that short term excursions higher than design can occur for 1% or the time in a year (approximately 30 hours). The safety

related equipment in the service water pump room and diesel driven fire pump room was evaluated for the short term (approximately 24 hours as often as once a year) outdoor air temperature excursions and found acceptable.

9.4.5.2 System Description

The service water pump room ventilation system (Seismic Class I) consists of four safety-related service water pump room ventilation fans, with associated temperature switches and controls. Each fan is sized at 50% of the capacity required to maintain the service water pump room at or below 104°F and the diesel-driven fire pump room at or below 120°F based on a 95°F outside air supply. Each channel of fans is started automatically by temperature switches at a predetermined temperature setpoint. The missile protected supply air penthouse is sized to ensure adequate supply air with all four supply fans operating simultaneously.

Winter heating in the service water pump room is accomplished by means of a steam heating system. The unit heater uses low pressure auxiliary steam, except during station shutdown when steam is taken from the auxiliary boiler. This heating system is non-seismic.

9.4.5.3 Safety Evaluation

Loss of either safety-related channel of service water room ventilation may result in the system's inability to tolerate an additional single failure, and the affected component(s) should be repaired or replaced as soon as possible.

A heat load/heat transfer analysis of the service water pump and diesel-driven fire pump rooms indicates that one service water pump room ventilation fan is of adequate capacity to maintain the service water pump room at or below 104°F and the diesel-driven fire pump room at or below 120°F provided the outside air temperature is 86°F or less.

Due to the magnitude of the room heat load, the service water pump room cannot be maintained indefinitely at or below 104°F following a loss of all ventilation regardless of the ambient outside air temperature, nor can the operability of the service water pumps be ensured once the room temperature exceeds 104°F. See single failure analysis presented in Table 9.4-10.

9.4.5.4 Tests and Inspections

The system was given a preoperational test after major modifications were completed. The components of the ventilating system are accessible for periodic inspection and maintenance.

9.4.6 <u>Containment Recirculation System</u>

The Containment Recirculation System consists of two redundant fans with independent duct distribution systems. It is controlled from the control room. All components of the system are designed as Seismic Class I. The capacity of each containment recirculation fan is 8,000 cfm.

The containment recirculation fans may be used during normal operation to prevent temperature stratification.

TABLE 9.4-1

Single Failure Analysis –
Control Room Emergency Ventilation System

	Component	Failure	Comments and Consequences
1.	Offsite power	Not available	Emergency diesels start and supply electrical load to systems.
2.	Emergency diesels	One not available	The operative diesel supplies necessary power to one of the redundant system flow paths.
3.	Control room emergency ventilation systems	One not available	The standby, 100 percent capacity (for emergency load), control room emergency ventilation system is available to provide suitable temperature conditions in the control room for operating personnel and safety-related control equipment.
4.	Control room emergency ventilation systems	Rupture of equipment casings and/or ducts	Consideration has been given in the detailed design to withstand the design basis temperature, pressure, and seismic forces during a post-accident situation. The equipment and components are also inspectable and protected against credible missiles.
5.	Control room emergency ventilation systems	Rupture of system piping	Rupture is not considered credible since all piping is designed to withstand the design basis temperature, pressure, and seismic forces during a post-accident situation and is inspectable and protected from missiles.

TABLE 9.4-2

Control Room Emergency Ventilation System Heat Loads

	Source		Heat Load (Btu/hr)
1.	Essential Cabinets		
2.	Essential Lighting (partial)		
3.	People (8)		
4.	Heat Transmission Load		
5.	Heat introduced into control room (at 95°F) by the intake of 300 cfm of outside air (at 95°F) (Latent Heat)		
6.	Fan load		
		Total	104,862
	Total heat removal capability for one emergency unit (10 tons nominal)		120,000

TABLE 9.4-3

Single Failure Analysis –
Low Voltage Switchgear Room Ventilation Systems

	Component	Failure	Comments and Consequences
1.	Offsite power	Not available	Emergency diesels start and supply electrical load to systems.
2.	Emergency diesels	One not available	The operative diesel supplies necessary power to the redundant low voltage switchgear room ventilation system.
3.	Low-voltage switchgear room ventilation systems	One not available	The standby, full capacity, safety-related ventilation system is available to maintain an average temperature between 60°F and 104°F in the second low-voltage switchgear room.
4.	Low-voltage switchgear room ventilation fans	Rupture of casing	Consideration has been given in the detailed design to withstand the design basis temperature, pressure and seismic forces during a post-accident situation. The equipment and components are also inspectable and protected against credible missiles.

TABLE 9.4-4

Single Failure Analysis –
Emergency Diesel Generator Room Ventilation Systems

	Component	Failure	Comments and Consequences
1.	Offsite power	Not available	Emergency diesels start and supply electrical load to systems.
2.	Emergency diesels	One not available	The operative emergency diesel generator supplies necessary power to its own ventilation system.
3.	Emergency diesel generator room ventilation systems	One not available	The standby, full capacity safety-related ventilation system is available to maintain room temperature between 60°F and 125°F using outside air at 95°F or less for essential equipment in the second emergency diesel generator room.
4.	Emergency diesel generator room ventilation fans	One not available	One fan in the diesel generator room will maintain 131°F with 70°F or less outside temperature, and no reverse flow through the inoperable fan.
5.	Emergency diesel generator room ventilation system	Rupture of equipment casings and/or ducts	Consideration has been given in the detailed design to withstand the design basis temperature, pressure and seismic forces during a post-accident situation. The equipment and components are also inspectable and protected against credible missiles.

TABLE 9.4-5

Failure Mode and Effect Analysis –
Fuel Handling Area Ventilation System

Component	Failure Mode	Effect and Comments
1. Fuel Handling Area Exhaust Fans	One not available	The standby, full-capacity, fuel handling area exhaust fan is available to maintain a suitable environment in the fuel handling area.
2. Fuel Handling Area Supply Fan	Not available	The lack of normal makeup air results in a greater negative pressure being produced in the fuel handling area and infiltration of air from the adjoining radwaste areas.
3. Fuel Handling Area Exhaust Filter	Not available	The fuel handling area is connected to the emergency ventilation system filters by means of ductwork bypasses and dampers to exhaust the fuel handling area via that system.
4. Dampers in ductwork from the fuel handling area to EVS (HV 5430A and HV 5430B)	One not available	The second bypass ductwork damper establishes flow path to EVS.
5. Essential solenoid valves in main pneumatic air line	One not available	Second solenoid valve (in series on a separate channel) is capable of de-energizing on high radiation and bleeding air from respective valves.
6. Dampers on the upstream side of EVS filters (HV 5024 and HV 5025)	Power failure during fuel handling	The dampers are manually opened just prior to fuel handling operation. Since the dampers failed position is “as is”, they will remain open when power failure occurs during fuel handling operation, thus assuring flow path to EVS.

TABLE 9.4-6

Single Failure Analysis –
ECCS Room Cooling Units

	Component	Failure	Comments and Consequences
1.	Offsite power	Not available	Emergency diesels start and supply electrical load to systems.
2.	Emergency diesels	One not available	The operative diesel supplies necessary power to one of the redundant system flow path.
3.	ECCS pump room cooling units	One not available	Cooling capacity in one room is reduced to 50 percent of design. The standby, full capacity, ECCS Room Cooling Units are available to maintain a suitable environment for essential equipment in the second ECCS pump room.
4.	ECCS room cooling units	Rupture of equipment casings	Consideration has been given in the detailed design to withstand the design basis temperature, pressure and seismic forces during a post-accident situation. The equipment and components are also inspectable and protected against credible missiles.
5.	ECCS room cooler combined service water outlet header valve	Fails closed	This is considered incredible due to the fact that the valve is a manual valve.

TABLE 9.4-7

Single Failure Analysis –
Battery Room Ventilation System

	Component	Failure	Comments and Consequences
1.	Offsite power	Not available	Emergency diesels start and supply electrical load to systems.
2.	Emergency Diesel Generator	One not available	The operative diesel supplies necessary power to the redundant battery room ventilation. Affected battery discharges per design. Adequate ventilation is later made available to support recharging.
3.	Battery Room Ventilation System Train/Fan	One fails	Long term redundancy is provided by the alternate train batteries and associated safety-related ventilation system. Annunciator should prompt compensatory action prior to loss of affected battery operability.
4.	Battery Room Ventilation System Train	Rupture of duct	Consideration has been given in the detailed design to withstand the design basis temperature, pressure and seismic forces during a post-accident situation. The equipment and components are also inspectable and protected against missiles.

TABLE 9.4-8

Single Failure Analysis –
CCW Pump Room Ventilation System

	Component	Failure	Comments and Consequences
1.	Offsite power	Not available	Emergency diesels start and supply electrical load to systems.
2.	Emergency Diesels	One not available	The operative diesel supplies necessary power to the redundant CCW room ventilation system.
3.	CCW Pump Room Ventilation Trains/Fan	One fails	Redundancy is provided by the alternate train 100% capacity safety-related ventilation fan and outside air louvers, exhaust and recirculation air dampers and controls to maintain room temperature between 60°F and 104°F. Short term (less than 24 hours, as often as once a year) temperature excursions to 120°F are acceptable.
4.	CCW Pump Room ventilation system	Rupture of ducts	Consideration has been given in the detailed design to withstand the design basis temperature, pressure and seismic forces during a post-accident situation. The equipment and components are also inspectable and protected against credible missiles.

TABLE 9.4-9

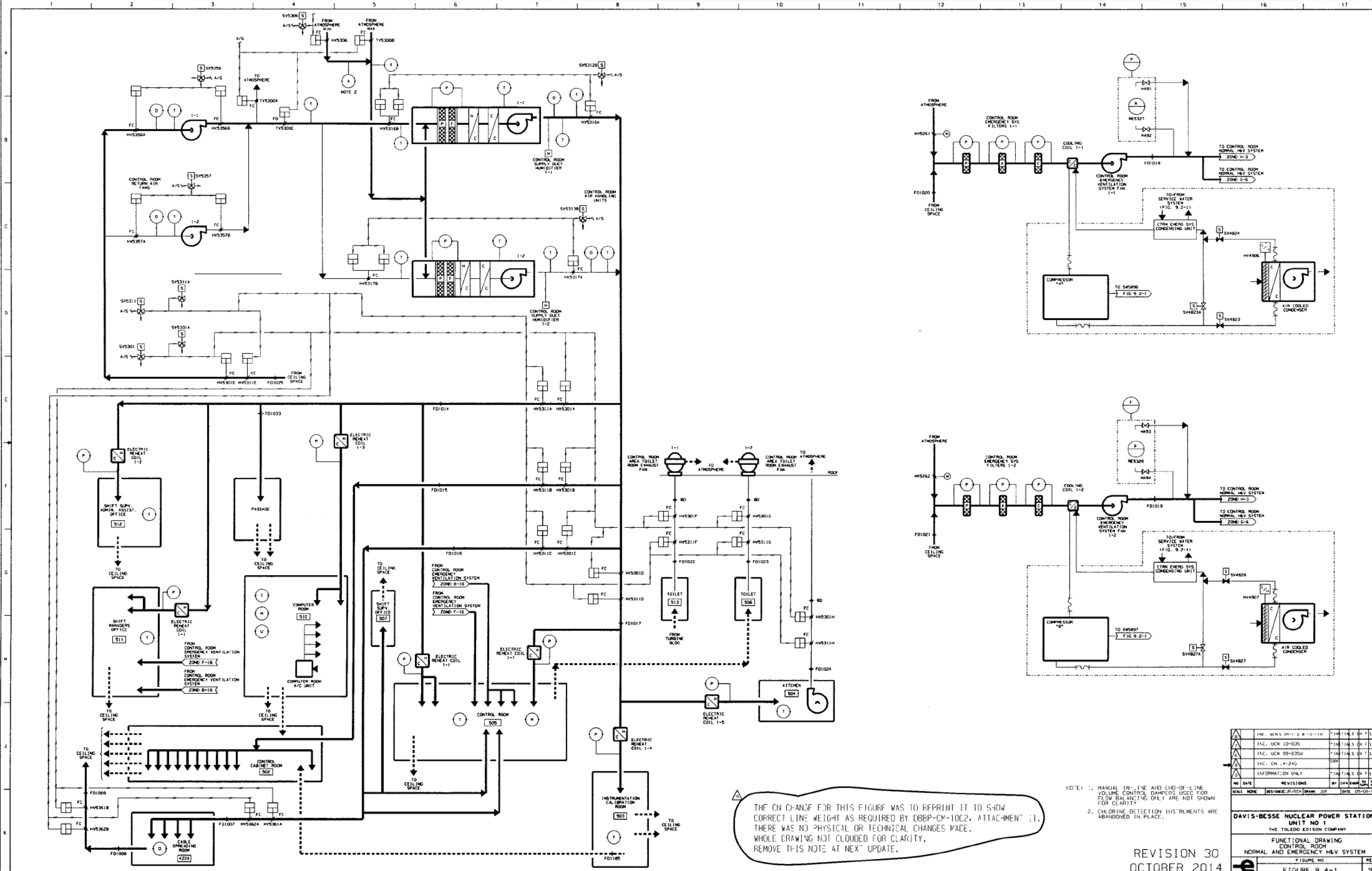
Single Failure Analysis –
Auxiliary Feedwater Pump Room Ventilation System

	Component	Failure	Comments and Consequences
1.	Offsite power	Not available	Emergency diesel start and supply electrical load to systems.
2.	Emergency Diesel	One fails	The operative diesel supplies necessary power to the redundant auxiliary feedwater room ventilation system.
3.	Aux. Feedwater Pump Room Ventilation Exhaust Fan or Train	One fails	Redundancy is provided by the alternate train 100% capacity safety-related exhaust fan and its associated ventilation system to maintain between 60°F and 120°F utilizing supply air from the turbine building ≤110°F.
4.	Aux. Feedwater Pump Room Ventilation System	Rupture of ducts	Consideration has been given in the detailed design to withstand the design basis temperature, pressure and seismic forces during a post-accident situation. The equipment and components are also inspectable and protected against credible missiles.

TABLE 9.4-10

Single Failure Analysis –
Service Water Pump Room Ventilation System

	Component	Failure	Comments and Consequences
1.	Offsite power	Not available	Emergency diesel start and supply electrical load to systems.
2.	Emergency Diesels	One not available	The operative diesel supplies necessary power to the service water pump room ventilation system.
3.	Service Water Pump Room or Diesel-Driven Fire Pump Ventilation Channel (2 fans)	One fails	Redundancy is provided by the alternate safety-related ventilation channel. Two 50% capacity fans per channel have adequate capacity to maintain the service water pump room at or below 104°F and the diesel driven fire pump room at or below 120°F based on a 95°F outside air supply. One fan per channel has adequate capacity to maintain the service water pump room at or below 104°F and the diesel-driven fire pump room at or below 120°F provided the outside air temperature is 86°F or less.
4.	Service Water Pump Room or Diesel-Driven Fire Pump Room Ventilation System	Rupture of duct	Consideration has been given in the detailed design to withstand the design basis temperature, pressure, flow and seismic forces during a post-accident situation. The equipment and components are also inspectable and protected against credible missiles.



THE CN CHANGE FOR THIS FIGURE WAS TO REPRINT IT TO SHOW CORRECT LINE WEIGHT AS REQUIRED BY DBBP-CM-1002, ATTACHMENT 11. THERE WAS NO PHYSICAL OR TECHNICAL CHANGES MADE. WHOLE DRAWING NOT CLOUDED FOR CLARITY. REMOVE THIS NOTE AT NEXT UPDATE.

NOTE: 1. MANUAL IN-LINE AND END-OF-LINE VOLUME CONTROL DAMPERS USED FOR FLOW BALANCING ONLY ARE NOT SHOWN FOR CLARITY

2. CHLORINE DETECTION INSTRUMENTS ARE ABANDONED IN PLACE.

REVISION 30
OCTOBER 2014

NO	DATE	REVISIONS	BY	CHK	ENGR	SA SUPV	MED MGR
8		INC. UCN'S 09-172 & 10-110	"INITIALS ON FILE"				
7		INC. UCN 10-005	"INITIALS ON FILE"				
6		INC. UCN 99-035U	"INITIALS ON FILE"				
9		INC. CN 14-245	SBV				
0		INFORMATION ONLY	"INITIALS ON FILE"				

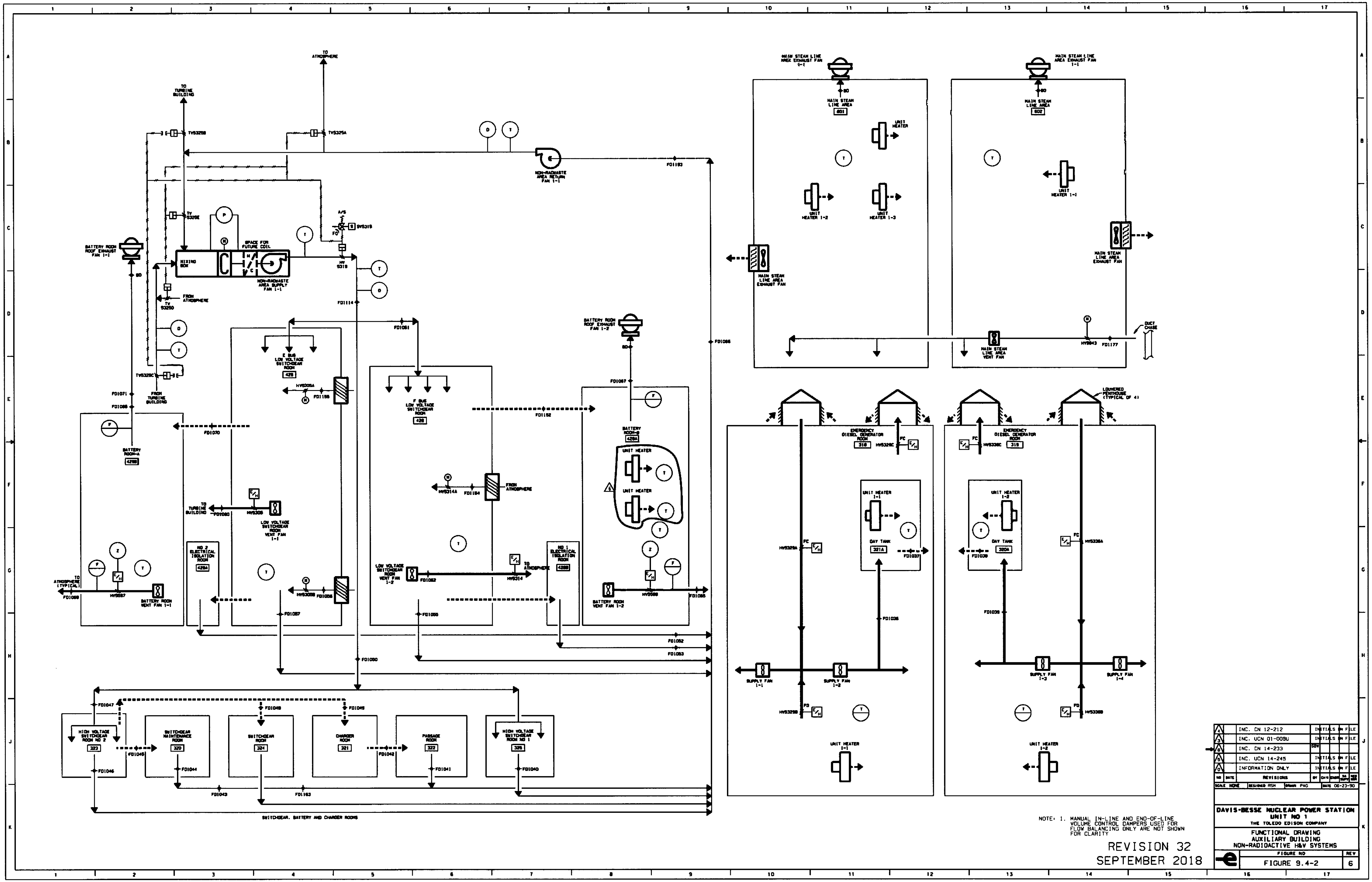
SCALE NONE | DESIGNED CJR/RSM | DRAWN JOR | DATE 05-09-90

DAVIS-BESSE NUCLEAR POWER STATION
UNIT NO 1
THE TOLEDO EDISON COMPANY

FUNCTIONAL DRAWING
CONTROL ROOM
NORMAL AND EMERGENCY H&V SYSTEM

FIGURE NO: FIGURE 9.4-1 | REV: 9

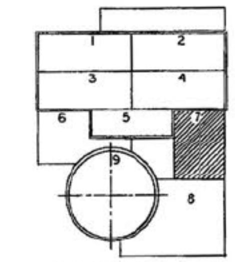
DFN-/USAR/UFIG941.DGN
DB 10-01-14



REVISION 32
SEPTEMBER 2018

DFN=/G/USAR/UFIG942.DGN
DB 10-10-17

[redacted]



KEY PLAN
EL. 638'-0"

NOTES:

1. FOR AIR FLOW DIAGRAM THIS SYSTEM SEE DRAWING NO. M-027A.
2. FOR SYMBOLS, ABBREVIATIONS AND GENERAL NOTES SEE DWG. NO. M-400.
3. FOR PENTHOUSE DETAIL SEE DRAWING NO. M-453
4. FOR BLOCKOUT LOCATIONS SEE DWG. NO. C-235.
5. ALL DUCTS PASSING THRU SLAB AT EL. 638'-0" SHALL HAVE FIRE DAMPERS.
6. CONTROL ROOM EMERGENCY VENTILATION SYSTEM FILTERS F22-1 & F22-2 SHALL BE FURNISHED BY OWNERS, INSTALLED BY CONTRACT 7749-11.
7. CHILLERS SHALL HAVE 5" STAINLESS STEEL FLEXIBLE COUPLINGS ON WATER SIDE PIPING CONNECTIONS.
8. FOR SEISMIC CLASS I DUCTWORK SUPPORT LIMITS SEE P&ID NO. M-027A.
9. FOR FIRE DAMPER SCHEDULE SEE DRAWING M-467.
10. DELETED
11. NEW AND REPLACEMENT INSULATION SHALL BE PER SPECIFICATION M-198N.
12. SEE SPECIFICATION G-0090 FOR DUCT JOINT SEALING MATERIALS.

DAVIS-BESSE NUCLEAR POWER STATION

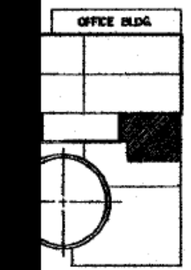
HEATING, VENTILATING AND AIR CONDITIONING
CONTROL ROOM - EQUIPMENT ROOM
PLAN ELEV. 638'-0"
M-450
FIGURE 9.4-3

REVISION 31

OCTOBER 2016

DB 06-06-16 DFN=/J/MECH/M450.DGN/TIF

[redacted]



PLAN
623'-0"

ON DIAGRAM THIS SYSTEM SEE DRAWING

S, ABBREVIATIONS AND GENERAL NOTES
D. M-400.

RILLES SHALL BE 12" X 12" UNLESS
RWISE.

PASSING THRU SLAB AT EL. 638'-0"
F.D.'S (3HR.).

UT LOCATIONS SEE DWG. NO. C-228.

ER AND GRILLE LOCATIONS SEE
CLG. PLAN DWG. A-39.

AMPER SCHEDULE SEE DRAWING M-467.

NDICATES LIMIT OF DUCTWORK REQUIRING
ANALYSIS (OTHER THAN "Q-LISTED".

DOOR MODIFICATIONS FOR FIRE DAMPERS
D FD-1158 SEE DWGS. M-468 & M-470.

000 CFM DIFFUSER IN THE COMPUTER ROOM
NECK SIZE IS ACCEPTABLE.

S DOORS AND PANELS FOR HEATING COILS
6-2, E76-3, E76-4, E76-5, E110 AND E111
NG 7749-M-410-45.

FICATION G-009Q FOR DUCT JOINT
ATERIALS.

DAVIS-BESSE NUCLEAR POWER STATION

HVAC CONTROL ROOM -
PLAN AT ELEVATION 623'-0"
M-451
FIGURE 9.4-4

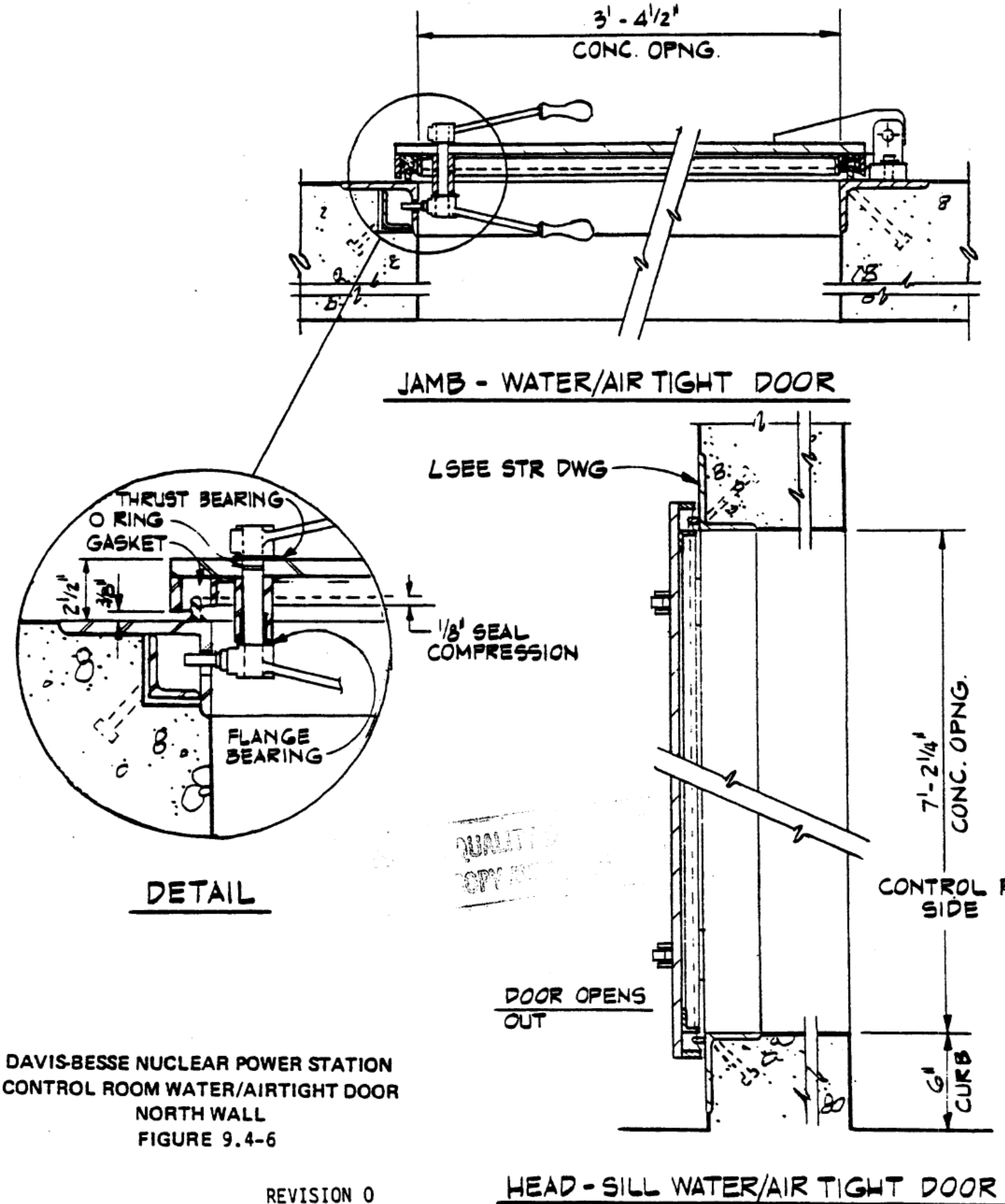
REVISION 27
JUNE 2010

DB 03-23-10 DFN=/J/MECH/M451.DGN/TIF

Removed in Accordance with RIS 2015-17

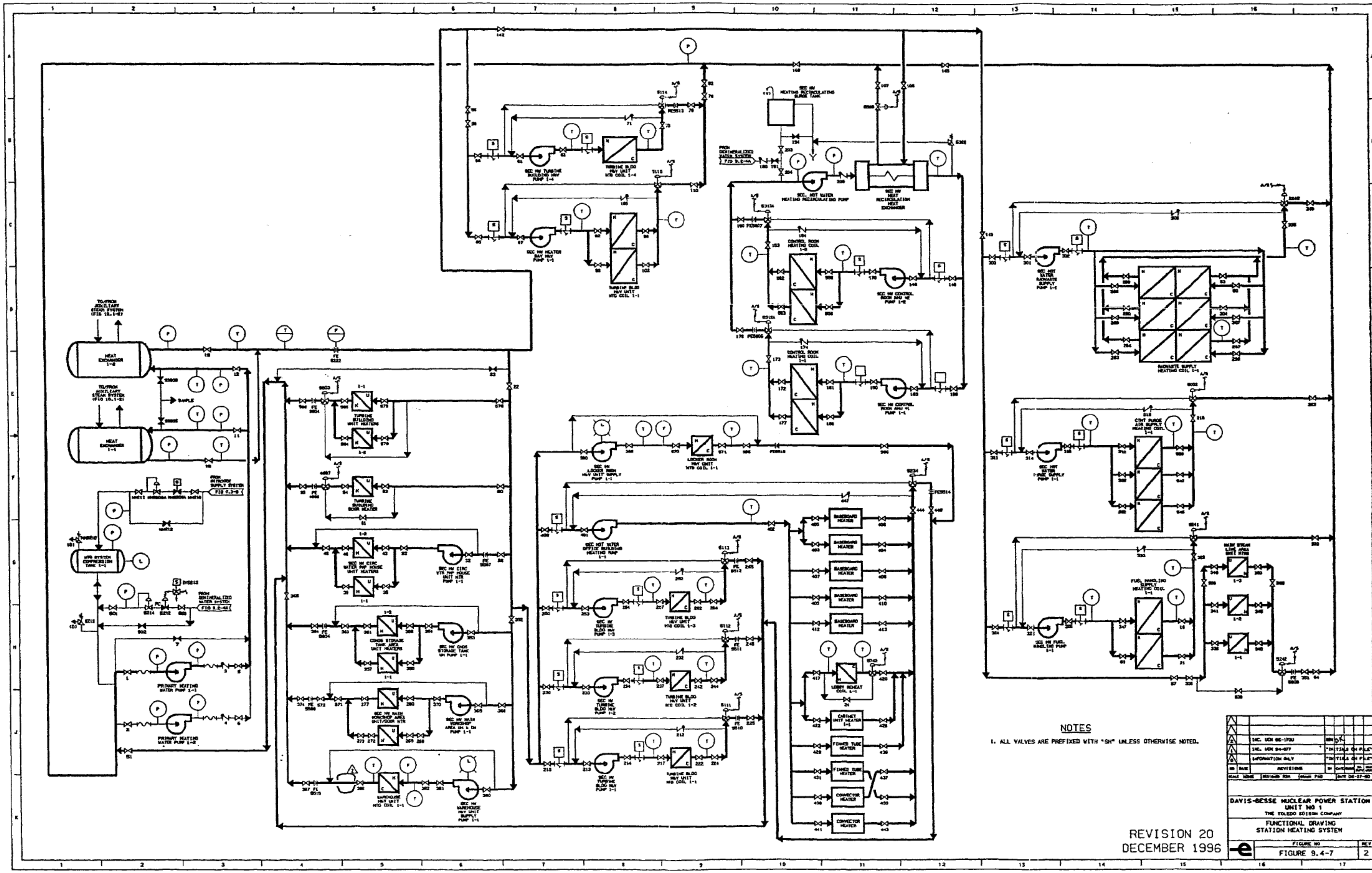
DAVIS-BESSE NUCLEAR POWER STATION

HEATING, VENTILATING AND AIR CONDITIONING
CONTROL ROOM SECTIONS AND DETAILS
M-453
FIGURE 9.4-5



DAVIS-BESSE NUCLEAR POWER STATION
CONTROL ROOM WATER/AIRTIGHT DOOR
NORTH WALL
FIGURE 9.4-6

REVISION 0
JULY 1982



NOTES

1. ALL VALVES ARE PREFIXED WITH "SH" UNLESS OTHERWISE NOTED.

REVISION 20
DECEMBER 1996

NO	DATE	REVISIONS	BY	CHK'D			
2		INC. UCN 96-1700					
1		INC. UCN 94-077	"INITIALS ON FILE"				
0		INFORMATION ONLY	"INITIALS ON FILE"				

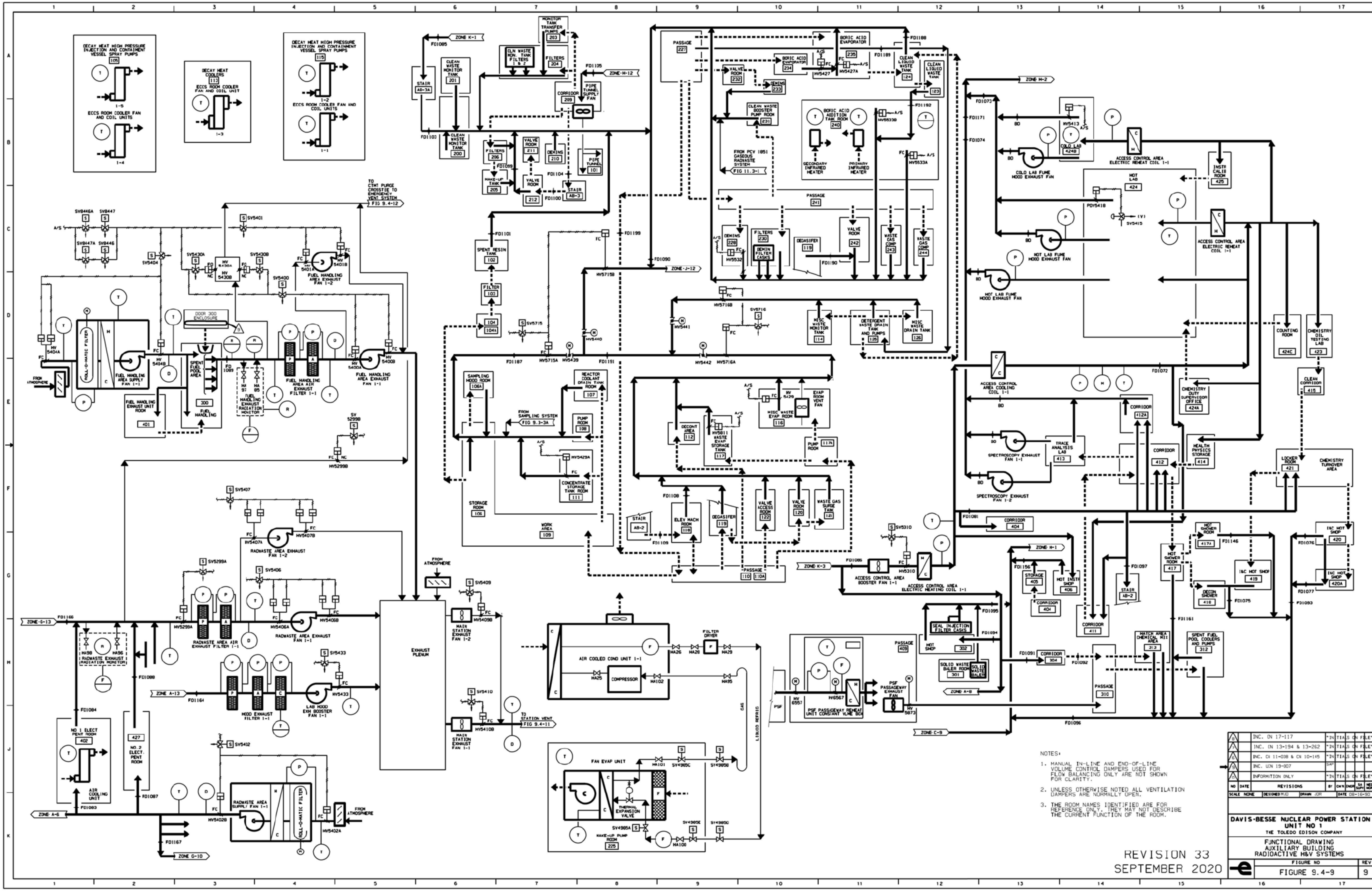
SCALE NONE | DESIGNED RSM | DRAWN PHG | DATE 06-27-90

DAVIS-BESSE NUCLEAR POWER STATION UNIT NO 1
THE TOLEDO EDISON COMPANY

FUNCTIONAL DRAWING
STATION HEATING SYSTEM

FIGURE NO: FIGURE 9.4-7 | REV: 2

DFN-/DBNFS2/USAR/UF10547.DGN
08 11-04-96



NOTES:

1. MANUAL IN-LINE AND END-OF-LINE VOLUME CONTROL DAMPERS USED FOR FLOW BALANCING ONLY ARE NOT SHOWN FOR CLARITY.
2. UNLESS OTHERWISE NOTED ALL VENTILATION DAMPERS ARE NORMALLY OPEN.
3. THE ROOM NAMES IDENTIFIED ARE FOR REFERENCE ONLY. THEY MAY NOT DESCRIBE THE CURRENT FUNCTION OF THE ROOM.

NO	DATE	REVISIONS	BY	CHK	ENGR	SA SUPV	MED MGR
8		INC. CN 17-117	"INITIALS ON FILE"				
7		INC. CN 13-194 & 13-262	"INITIALS ON FILE"				
6		INC. CN 11-038 & CN 10-145	"INITIALS ON FILE"				
9		INC. UCN 19-007	SAP				
0		INFORMATION ONLY	"INITIALS ON FILE"				

SCALE NONE | DESIGNED MJD | DRAWN JDR | DATE 08-16-90

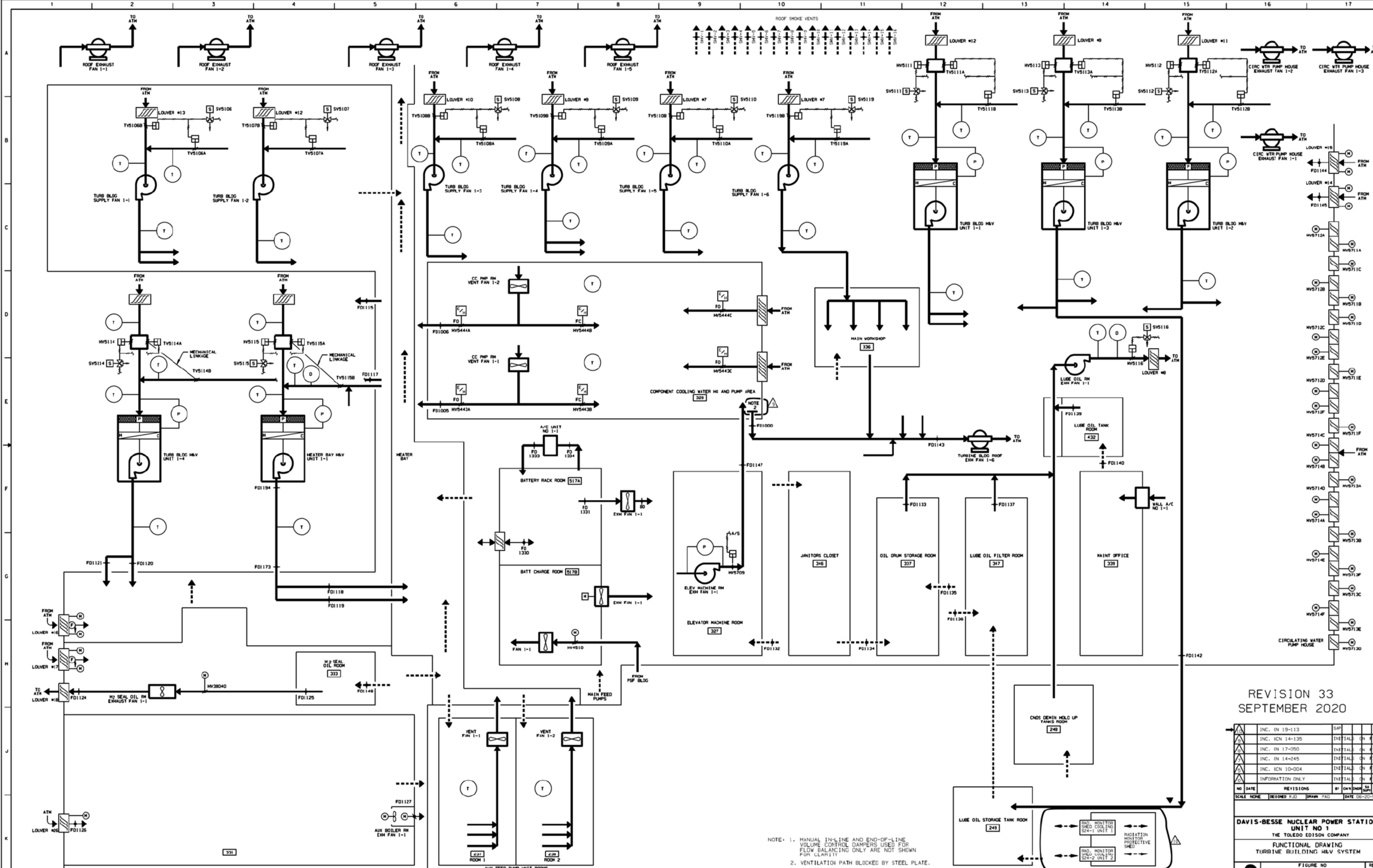
DAVIS-BESSE NUCLEAR POWER STATION UNIT NO 1
THE TOLEDO EDISON COMPANY

FUNCTIONAL DRAWING
AUXILIARY BUILDING
RADIOACTIVE H&V SYSTEMS

FIGURE NO: FIGURE 9.4-9 | REV: 9

REVISION 33
SEPTEMBER 2020

DB 04-27-20
DFN: G/USAR/UFIG949.DGN



NOTE: 1. MANUAL IN-LINE AND END-OF-LINE VOLUME CONTROL DAMPERS USED FOR FLOW BALANCING ONLY ARE NOT SHOWN FOR CLARITY

2. VENTILATION PATH BLOCKED BY STEEL PLATE.

REVISION 33
SEPTEMBER 2020

NO	DATE	REVISIONS	BY	CHK	ENGR	SA SUPV	MED MGR
10		INC. CN 19-113	SAP				
9		INC. ECN 14-135	INITIALS ON FILE				
8		INC. CN 17-050	INITIALS ON FILE				
7		INC. CN 14-245	INITIALS ON FILE				
5		INC. ECN 10-004	INITIALS ON FILE				
0		INFORMATION ONLY	INITIALS ON FILE				

SCALE NONE | DESIGNED MJD | DRAWN PAG | DATE 06-20-90

DAVIS-BESSE NUCLEAR POWER STATION UNIT NO 1
THE TOLEDO EDISON COMPANY

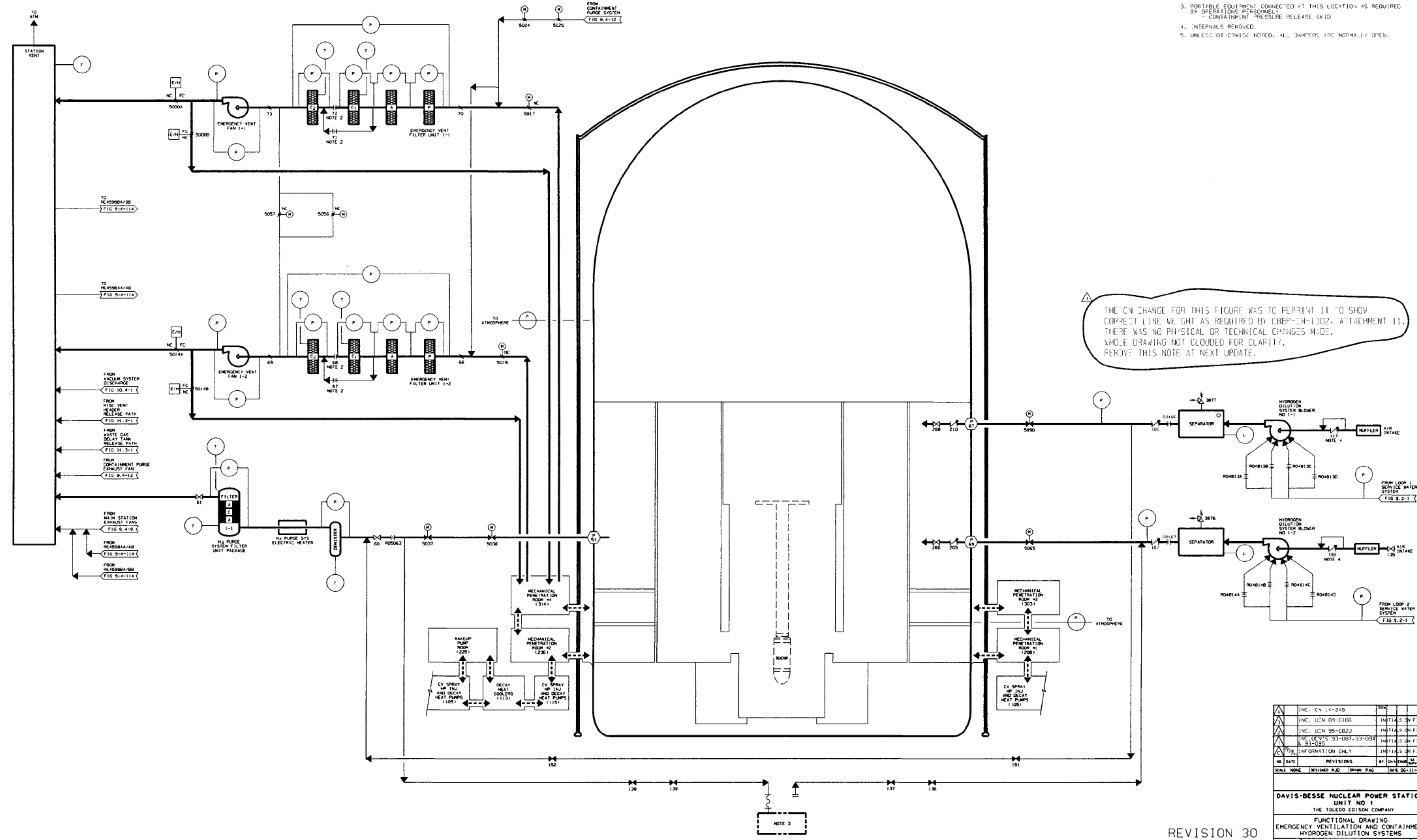
FUNCTIONAL DRAWING
TURBINE BUILDING H&V SYSTEM

FIGURE NO: FIGURE 9.4-10 | REV: 10

DFN=/G/USAR/UFIG9410.DGN
DB 08-14-20

NOTES

1. ALL VALVES ARE PREFIXED WITH "CV" UNLESS OTHERWISE NOTED.
2. FLANGES 67 AND 71 ARE BLANKED OFF.
3. PORTABLE EQUIPMENT CONNECTED AT THIS LOCATION AS REQUIRED BY OPERATIONS PERSONNEL:
   - CONTAINMENT PRESSURE RELEASE SKID
4. INTERNALS REMOVED.
5. UNLESS OTHERWISE NOTED, ALL DAMPERS ARE NORMALLY OPEN.



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NO	DATE	REVISIONS	BY	CHK	ENGR	SA SUPV	MED MGR
4		INC. CN 14-245	SBW				
3		INC. UCN 09-0106	INITIALS ON FILE				
2		INC. UCN 95-0820	INITIALS ON FILE				
1		INC. UCN'S 93-087, 93-094 & 93-095	INITIALS ON FILE				
C	12/27/91	INFORMATION ONLY	INITIALS ON FILE				

SCALE NONE | DESIGNED NJD | DRAWN PAG | DATE 06-11-90

DAVIS-BESSE NUCLEAR POWER STATION UNIT NO 1
THE TOLEDO EDISON COMPANY

FUNCTIONAL DRAWING
EMERGENCY VENTILATION AND CONTAINMENT HYDROGEN DILUTION SYSTEMS

FIGURE NO: FIGURE 9.4-11 | REV 4

REVISION 30
OCTOBER 2014

DFN=/G/USAR/UFIG9411.DGN
DB 10-01-14

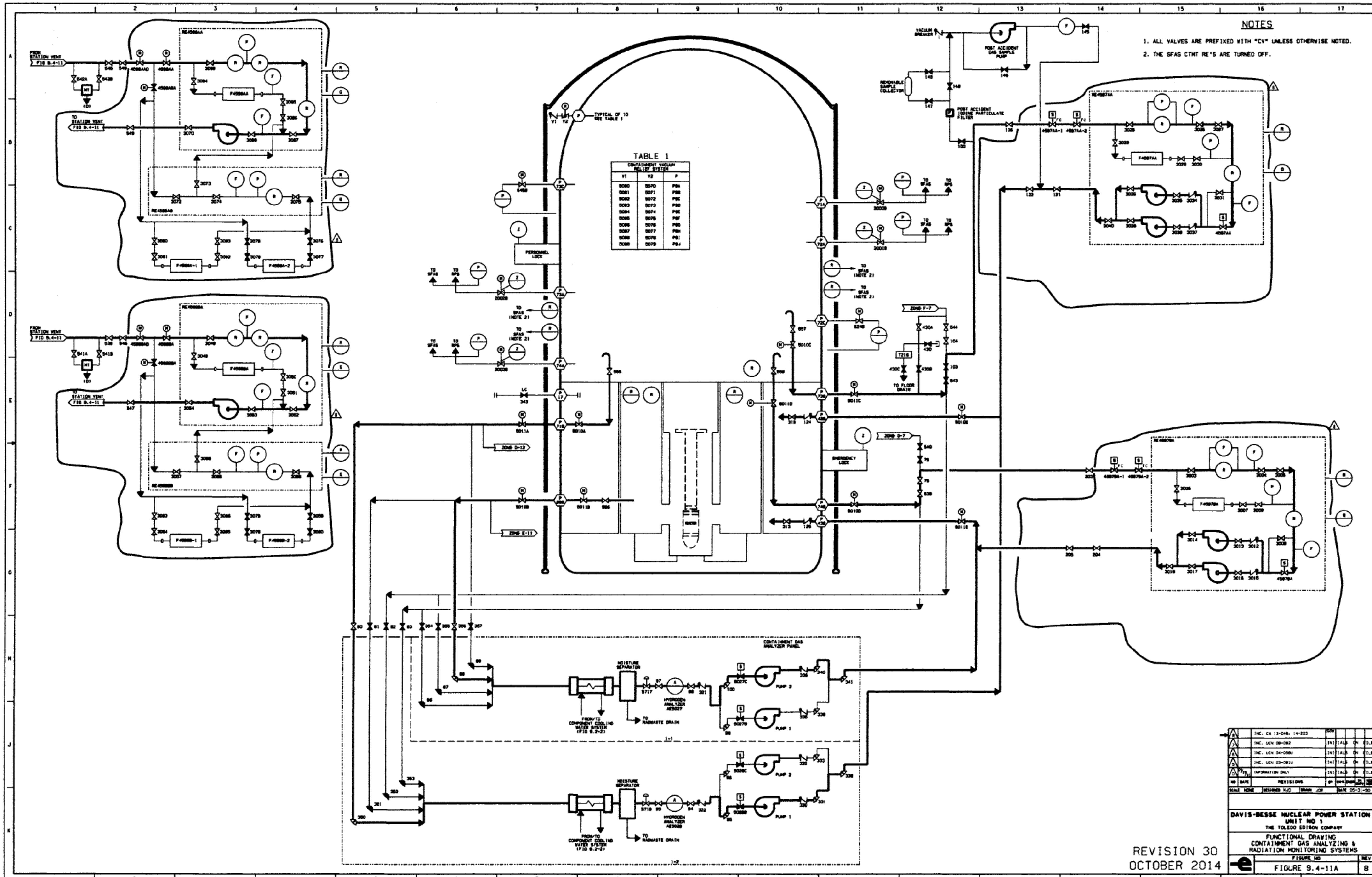


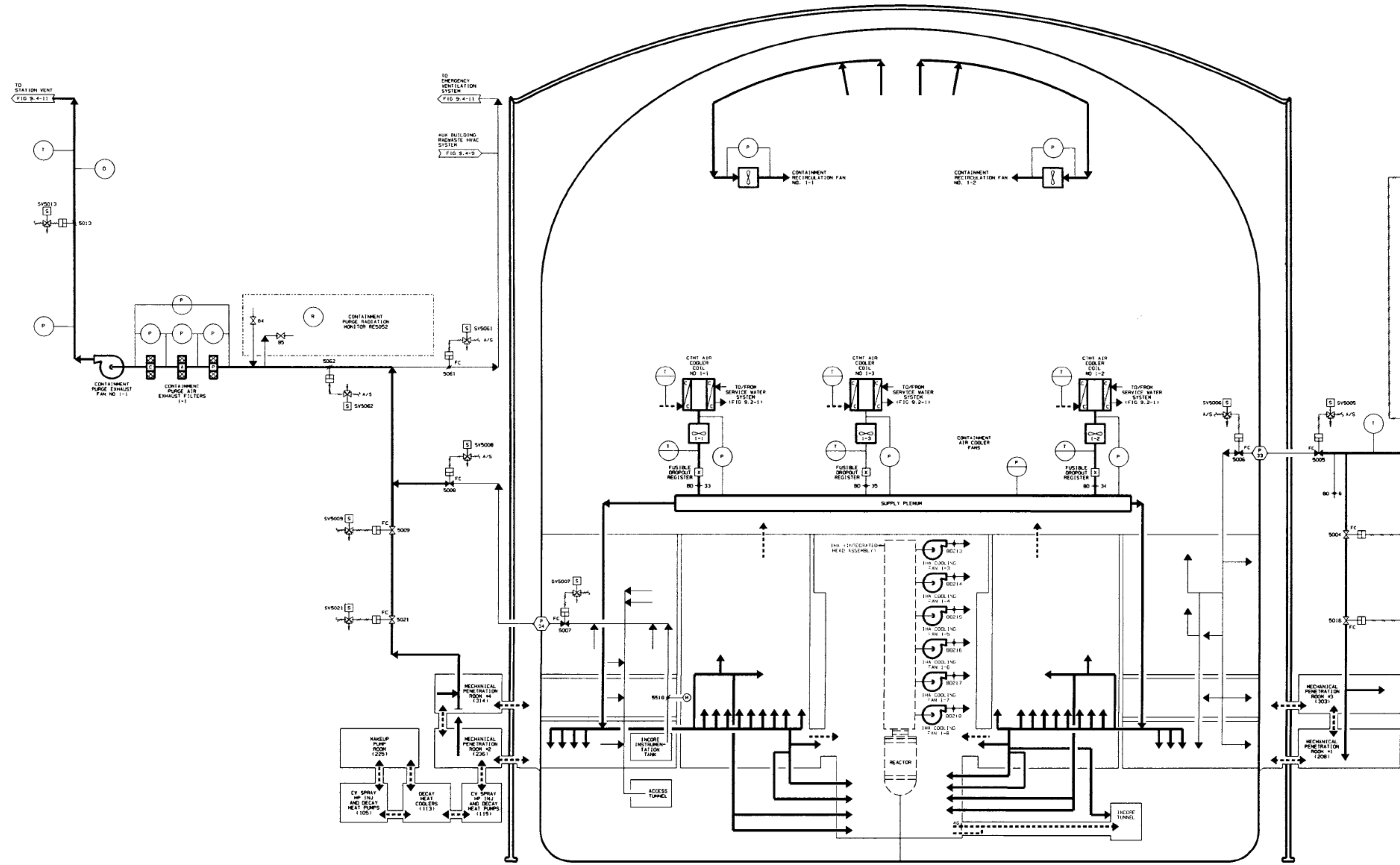
TABLE 1

CONTAINMENT VACUUM RELIEF SYSTEM		
V1	V2	P
5060	5070	P5A
5061	5071	P5B
5062	5072	P5C
5063	5073	P5D
5064	5074	P5E
5065	5075	P5F
5066	5076	P5G
5067	5077	P5H
5068	5078	P5I
5069	5079	P5J

DFN=/O/USAR/UF19411A.DGN
DB 08-22-14

NOTES

1. ALL VALVES ARE PREFIXED WITH "CV" UNLESS OTHERWISE NOTED.
2. MANUAL IN-LINE AND END OF LINE VOLUME CONTROL DAMPERS USED FOR FLOW BALANCING ONLY ARE NOT SHOWN.



THE CN CHANGE FOR THIS FIGURE WAS TO REPRINT IT TO SHOW CORRECT LINE WEIGHT AS REQUIRED BY DBBP-CM-1002, ATTACHMENT 11. THERE WAS NO PHYSICAL OR TECHNICAL CHANGES MADE. WHOLE DRAWING NOT CLOUDED FOR CLARITY.
REMOVE THIS NOTE AT NEXT UPDATE.

NO	DATE	REVISIONS	BY	CHK	ENGR	SA SUPV	NED MGR
4		INC. CN 11-122	INITIALS ON FILE				
3		INC. UCN 10-007	INITIALS ON FILE				
2		INC. UCN 95-820	INITIALS ON FILE				
5		INC. CN 14-245	SBW				
0		INFORMATION ONLY	INITIALS ON FILE				

SCALE NONE | DESIGNED KJD | DRAWN PAG | DATE 06-12-90

DAVIS-BESSE NUCLEAR POWER STATION UNIT NO 1
THE TOLEDO EDISON COMPANY

FUNCTIONAL DRAWING
CONTAINMENT AIR COOLING, PURGE AND RECIRCULATION SYSTEMS

REVISION 30
OCTOBER 2014

FIGURE NO	REV
FIGURE 9.4-12	5

DFN:/G/USAR/UF169412.DGN
DB 10-01-14

9.5 OTHER AUXILIARY SYSTEMS

9.5.1 Fire Protection Program

The DBNPS fire protection program is described in NG-DB-00302, DBNPS Fire Protection Program. The program describes the overall fire protection effort and establishes the fire protection policy for the protection of Appendix R safe shutdown equipment required in the event of a fire and delineates the requirements for controlling changes to the program. The program identifies the various positions, responsibilities and authorities within the organization and addresses Fire Brigade training, controls over combustibles and ignition sources and includes procedures for fire fighting, equipment testing and quality assurance. The program also describes the specific implementation features and the means to limit fire damage to Appendix R safe shutdown equipment required in the event of a fire so that the capability to safely shut down is ensured.

The Fire Hazard Analysis Report (FHAR), which is part of the overall program, documents the analysis that ensures compliance with 10CFR50 Appendix R. Sections III.G, III.J, III.L, and III.O. It also documents the review of BTP APCSB 9.5-1, Appendix A requirements. Additionally, the FHAR contains the fire protection requirements for operability and surveillances and is incorporated by reference into the Davis-Besse Updated Safety Analysis Report (USAR).

The FHAR Section 9.0, Fire Protection Safety Analysis, contains the description of the fire protection systems previously found in this section.

9.5.2 Communications Systems

9.5.2.1 Offsite

Offsite communication are primarily accomplished by utilizing telephone systems and a private Asynchronous Transfer Mode (ATM) network system with built in redundancy.

A third type of off-site communication is provided by using the Toledo Edison mobile radio frequency communication system. This UHF Radio System also provides onsite communications (see Subsection 9.5.2.2.5).

9.5.2.2 Onsite

9.5.2.2.1 Normal Station Communications System

Internally, the station utilizes a system primarily composed of individual, solid-state amplifier units. The system functions for paging, alarm signaling, and party-line-type voice communications. In plant areas, five channels are provided for regular communications, and an additional five channels are provided for establishing maintenance circuits. Office area amplifier units only have the regular plant area channels. Each station has individual amplifiers for the handset and local speaker(s), and the system has been designed such that failure of any one station will not affect the remainder. Should a station fail, its modular design allows for rapid replacement of components to return it to service.

The Normal Station Communications System includes a station in the Acid Injection Building, so that a person who contacted acid or discovered a leak could communicate the information quickly and conveniently.

A Gai-Tronics handset unit is provided in the Emergency Lock (to containment) for the possibility of a person becoming trapped there.

9.5.2.2.2 Telephone System

Telephones have been provided for in the normally manned areas, including the Operational Support Center. These phones serve as a backup to the main internal station communications system and function as the primary offsite administrative circuit. The system is powered separately from the main page system.

9.5.2.2.3 Fuel Handling Circuit

An entirely separate loop circuit is provided by the Fuel Handling System for the exclusive use of personnel directly involved with fuel handling operations. A remote station is supplied in the control room for monitoring. The components used in this circuit are similar to those of the normal station system. The fuel handling circuit may be supplemented by radio communications.

9.5.2.2.4 DELETED

9.5.2.2.5 Portable Communications (UHF Radio) System

An additional communications system in the form of Ultra High Frequency (UHF) radio is provided.

Portable transceivers arc carried by personnel, and remote fixed transceivers are located at various key locations around the site.

Communications are possible in six different “modes” as follows:

Mode 1	Portable to Portable
Mode 2	Portable to Repeater to Remote/Portable
Mode 3	Remote to Lindsey Repeater
Mode 4	Remote to Acme Repeater
Mode 5	Remote to Ottawa County Sheriff
Mode 6	Remote to Remote

Operating instructions for this system can be located in plant procedures.

9.5.2.2.6 Sound-Powered Phones

Sound-powered phones are installed to supplement the Normal Station Communications System (Gai-Tronics) which may be lost due to a fire in the Control Room. These phones are hard-wired independent of the Control Room and are provided with headsets that would be plugged in at key locations, including the Auxiliary Shutdown Panel. The headsets are maintained at the key locations. The phones utilize the human voice to generate electric current to ensure communication, and no external power source is required.

9.5.2.2.7 Alarms

Three different alarms, designated Fire, Access Control Area Evacuation, and Initiate Emergency Action Procedures, may be sounded throughout appropriate areas of the station

over the paging system. These signals will be initiated by the control room operator and have separate distinctive characteristics. All alarms will run for a preset period of time and may be overridden in case they fail to de-energize.

Visual signals have been installed to indicate alarm and page signals in the Diesel Generator Rooms, Diesel Generator Day Tank Rooms, and the Fire Pump Diesel Room.

9.5.2.2.8 Testing

Periodic maintenance and testing of the amplifier and loudspeakers of the public address system is performed to ensure that each unit is in working condition. The three alarm systems are tested weekly to confirm that each is operable. (See Chapter 14 for initial test procedure.)

The Off-site Communications are tested according to the requirements as stated in the Davis-Besse Emergency Plan and Physical Security Plan.

9.5.2.2.9 Single Failure Criteria

The communications system does not have a safety related function as defined by 10CFR50 General Design Criteria and, therefore, has not been designed to meet single failure criteria. However, there is a sound-powered phone system located in designated areas of the plant that can be used for communications necessary to control shutdown. This system would remain operable in the event of a serious control room fire that disables the inplant communication system. In addition, the telephone system (Subsection 9.5.2.2.2) and the portable radio system (Subsection 9.5.2.2.5) are considered backups to the main internal system. Also, some sections of the main internal system are supplied by two redundant power feeders from the uninterruptible instrumentation distribution panel. As a minimum, these sections can be found in the Turbine Building, Auxiliary Building, and Containment.

9.5.3 Lighting Systems

The station lighting system utilizes five basic subsystems: 1) normal station and security lighting; 2) AC emergency lighting; 3) AC/DC emergency lighting; 4) Battery-powered lighting; and 5) hand-held lighting.

9.5.3.1 Normal Station and Security Lighting

Normal station lighting is provided by buses C2 and D2 through a double-end fed distribution substation (EF5) with a split-bus arrangement. The bus tie breaker connecting the two buses of this substation is normally open so that each bus is fed separately through a full capacity 4160-480/277 volt transformer. Upon failure of either source, the bus tie breaker will be manually closed and both buses will be fed from the remaining source.

Cooling Tower and switchyard lighting, which is actuated by photoelectric sensors, is provided by a yard distribution center and a switchyard distribution center.

Parking lot lighting is provided by a distribution center located in the Personnel Processing Facility.

Selection of lighting fixtures has been based on the particular area of application. Sodium or mercury vapor fixtures are used only for high bay lighting or out of doors. Fluorescent and mercury lighting is utilized only in areas where no possible contamination of the reactor coolant

system or its components could occur, such as the control room and office areas. In the containment and portions of the auxiliary building where such contamination could occur, only incandescent or LED fixtures are used.

The security lighting system has been brought into compliance with NRC requirements.

Some industrial type fluorescent lighting fixtures in the Auxiliary Building have exposed cord connecting the fixture to the lighting circuit outlet box or conduit. The separation distances between these exposed cords (cables in free air) and Class 1E raceway comply with the minimum separation distances given in IEEE Std 384-1992. Additionally, a review by fire protection has determined that the amount of combustible loading caused by these exposed cords does not pose an unanalyzed fire hazard.

While the normal and station lighting may be available, this lighting has not been evaluated for availability in the event of a fire.

9.5.3.2 AC Emergency Lighting

AC Emergency Lighting consists of two divisions of lighting circuits fed from essential motor control centers E11C and F11A and serving the Containment, the Auxiliary Building and the Control Room. The lighting circuits for the Auxiliary Building and the Control Room are automatically fed from the Diesel Generators. Lighting for containment must be manually activated.

This lighting is in addition to that provided by the normal station lighting from substation EF5. Redundant feeders and penetrations into the containment are used to preclude the possibility of total loss of lighting.

During modes 5 or 6, the capability exists for powering Containment lighting from non-essential sources when the normal essential sources are deenergized for maintenance. Both non-essential sources will not be used simultaneously so as to preclude a total loss of Containment lighting.

9.5.3.3 AC/DC Emergency Lighting

The AC/DC emergency lighting consists of two divisions of lighting circuits serving the Turbine and Auxiliary Buildings and the Control and Cable Spreading Rooms. Emergency lighting is used to provide sufficient illumination for safe evacuation during emergencies. The fixtures are 120V incandescent and will normally be supplied by substation EF5 through an automatic transfer switch. On failure of the EF5 source of power, the switch will cause automatic transfer of the emergency lighting to the 125V DC source for uninterrupted service. The essential DC sources are charged by the Diesel Generators and would be available in excess of 72 hours.

9.5.3.4 Battery-Powered Lighting

The battery-powered lighting unit system consists of numerous self-contained sealed beam units located throughout the Auxiliary and Turbine Building. The units are subject to a periodic 8-hour discharge test and are maintained operable in accordance with plant procedures and manufacturer's recommendations.

The lighting units are rated at 8 hours for discharge to 87.5% of nominal system voltage (6 volts) or 5.25V. The battery charger is fully automatic and capable of restoring the battery to full

capacity from 87.5% of the nominal battery voltage. Pilot lights indicate the state of the charge of the battery.

9.5.3.5 Hand-Held Lights

A minimum of ten hand-held lighting units are available for distribution to operators during a fire emergency. The operability of these units is assured by plant procedures. Hand-held lights would be used by the operators for indoor plant areas and access and egress for outside plant areas. Hand-held lights are provided in accordance with current procedure practices to each operator implementing manual safe shutdown actions due to a fire. The use of the hand-held lights would include those areas exposed to a fire in indoor areas that would be entered to perform manual safe shutdown actions after the fire is extinguished. These hand-held lights are provided for manual operator actions in indoor areas as a precautionary/backup measure in case the permanent emergency lighting would not be available as anticipated. The portable lighting in outside areas can provide an equivalent level of Lighting as a permanent lighting system. In addition, in outdoor areas, portable Lighting provides greater flexibility than a permanent lighting system.

9.5.4 Diesel Generator Fuel Oil Storage and Transfer System

9.5.4.1 Design Bases

The fuel supply for the emergency diesel generators is designed to meet the requirements of IEEE-308. The system is designed to withstand damage or loss of function caused by earthquake or tornado. The system includes seven days' storage of fuel oil for each emergency diesel generator. Applicable design codes and standards are listed in Table 9.0-1 and additional detail is given in Subsection 8.3.1.1.4.

9.5.4.2 Description

The emergency diesel generator fuel oil storage and transfer system is shown in Figure 9.5-8. The physical arrangement of the emergency diesel generator day tanks is shown in Figure 9.5-10.

The diesel fuel oil storage and transfer system is comprised of two separate trains. Each train consists of one supply tank, one fuel oil transfer pump, one day tank, and piping between the supply tank and day tank.

Each supply tank has a gross capacity of approximately 40,000 gallons. The tanks are installed above grade elevation; with tornado missile protection provided by a truncated pyramid of structural backfill built around the tanks. Corrosion of the tanks will be prevented by protective coatings, and by cathodic protection if necessary.

The emergency diesel generator day tanks are filled automatically via separate transfer systems which receive fuel oil from the two Emergency Diesel Fuel Oil Storage Tanks.

Each transfer pump is a submersible centrifugal pump suspended from the supply tank manhole. The pumps have a capacity which is greater than the fuel consumption of its associated emergency diesel generator at its maximum rated load. The fuel oil transfer pump discharge lines run directly to the associated diesel day tank. The underground portion of these lines will be protected from corrosion by protective coatings and cathodic protection. Each pump will be controlled automatically by level switches on the associated day tank. Manual

control for each pump is available at either the associated storage tank (below the outer manway) or the associated day tank room.

Each of the two diesel generator day tanks has a capacity of approximately 5,000 gallons, measured from the “start” level for the transfer pump.

Each day tank is located in a separate enclosure, with wet pipe, fusible head type automatic fire protection sprinklers. Flow indicating switches in each sprinkler system are provided for alarm. The day tanks are elevated to provide gravity flow to the suction of the fuel oil pumps for the engine.

The diesel oil system meets or exceeds all requirements of IEEE-308. The system meets ANSI proposed standard N195, “Fuel Oil Systems for Standby Diesel Generators,” (1974 Draft) with the following exceptions:

a. No overflow line is provided from the day tank back to the emergency diesel generator fuel oil storage tank.

b. The emergency diesel generator fuel oil storage tanks do not have high level alarms. Operator monitoring during filling of local level indication precludes overfilling.

c. No pressure indicator is provided at the discharge of the transfer pumps. Proper operation of these pumps can be verified by observing the fill rate in the day tanks.

d. Fuel Samples are analyzed in accordance with diesel manufacturer recommendations. Out of specification conditions are corrected by filtration, water removal, or fuel replacement as appropriate.

e. Fuel Oil Storage Tanks are drained and cleaned at a maximum interval of ten years.

In addition, the emergency diesel generator fuel oil storage tanks do not have strainers installed on the fill lines.

9.5.4.3 Safety Evaluation

The fuel oil storage required to maintain one emergency diesel generator in operation at is continuous rated load for seven days is approximately 40,000 gallons. To conservatively establish the storage capacity, this volume is based on consumption 10 percent higher than the fuel consumption measured during testing at the supplier’s factory; and a conservatively assumed specific gravity of the fuel oil. The required fuel oil capacity is based on both fully loaded emergency diesel generator consumption and fully loaded emergency diesel generator consumption plus 10 percent. A basis for the Emergency Diesel Generator Day Tank capacity, ANSI-N195, requires sufficient fuel oil capacity to sustain fully loaded emergency diesel generator operation for 60 minutes at 110 percent full load consumption. The basis for the combined capacity of the Emergency Diesel Generator Day Tank and the Fuel Oil Storage Tank, IEEE-308, requires sufficient fuel oil capacity to sustain continuous, full loaded emergency diesel generator operation for 7 days.

There is sufficient fuel oil in each day tank to operate its associated diesel generator for more then 19 hours at the continuous rated load.

The two Emergency Diesel Generator supply trains are completely independent except for a cross connect downstream of the day tanks which permits either diesel engine to be supplied with fuel oil from either tank in an emergency situation. In the event of an extreme emergency, the day tanks can be filled directly from a supply facility through the emergency fill connection.

There is no interconnection between the diesel generator engine fuel oil system and any nonsafety related system.

Each emergency diesel generator engine is operated monthly for test purposes, which keeps the internal surfaces of the system flooded or wetted with oil to reduce any potential corrosive condition. The tanks in the system are included in the station sampling program, and provisions are provided for the removal of any moisture in the system. A single failure of any component in the system will not result in damage to any safety-related systems. This analysis is shown in Table 9.5-3.

Level indication and low level alarms for the emergency diesel generator day tanks are provided in the main control room, as well as a low level alarm for the storage tanks. Level indication and high/low level indicating lights are provided locally at each storage tank.

Offsite diesel oil supply is available from several sources for onsite delivery within a safe margin of time to maintain the continuous operation of the emergency diesel generators.

With the exception of underground piping and tanks, all equipment and components are readily available for inspection.

9.5.4.4 Tests and Inspections

The storage tanks, transfer pumps, day tanks, and transfer piping receive tests and inspections in accordance with the applicable construction code.

Fuel quality and component operability will be verified at regular intervals in accordance with Technical Specifications.

9.5.5 Emergency Diesel Generator Cooling Water System

Subsection 8.3.1.1.4.1 describes the operation of the diesel generator jacket water cooling system.

9.5.6 Emergency Diesel Generator Starting System

Subsection 8.3.1.1.4.1 describes the diesel generator starting system.

9.5.7 Emergency Diesel Generator Lubrication System

Subsection 8.3.1.1.4.1 describes the diesel generator lubrication system.

TABLE 9.5-3

Single Failure Analysis –
Diesel Generator Fuel Oil Storage and Transfer System

	Component	Failure	Comments and Consequences
1.	Fuel oil transfer	Fails (stops)	Redundant subsystem is operable. Emergency diesel generator operable by affected subsystem has sufficient fuel for more than 19 hours of operation.
2.	Pipe line	Rupture	Should pipe rupture occur, level alarm on the day tank will indicate the abnormal condition. Redundant sub-system is not affected.
3.	Storage tank	Leaks or rupture	Should tank leaks or rupture occur, level alarm will indicate the abnormal condition. Alternate tank will serve the same function.

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DAVIS-BESSE NUCLEAR POWER STATION
UNIT NO 1
THE TOLEDO EDISON COMPANY

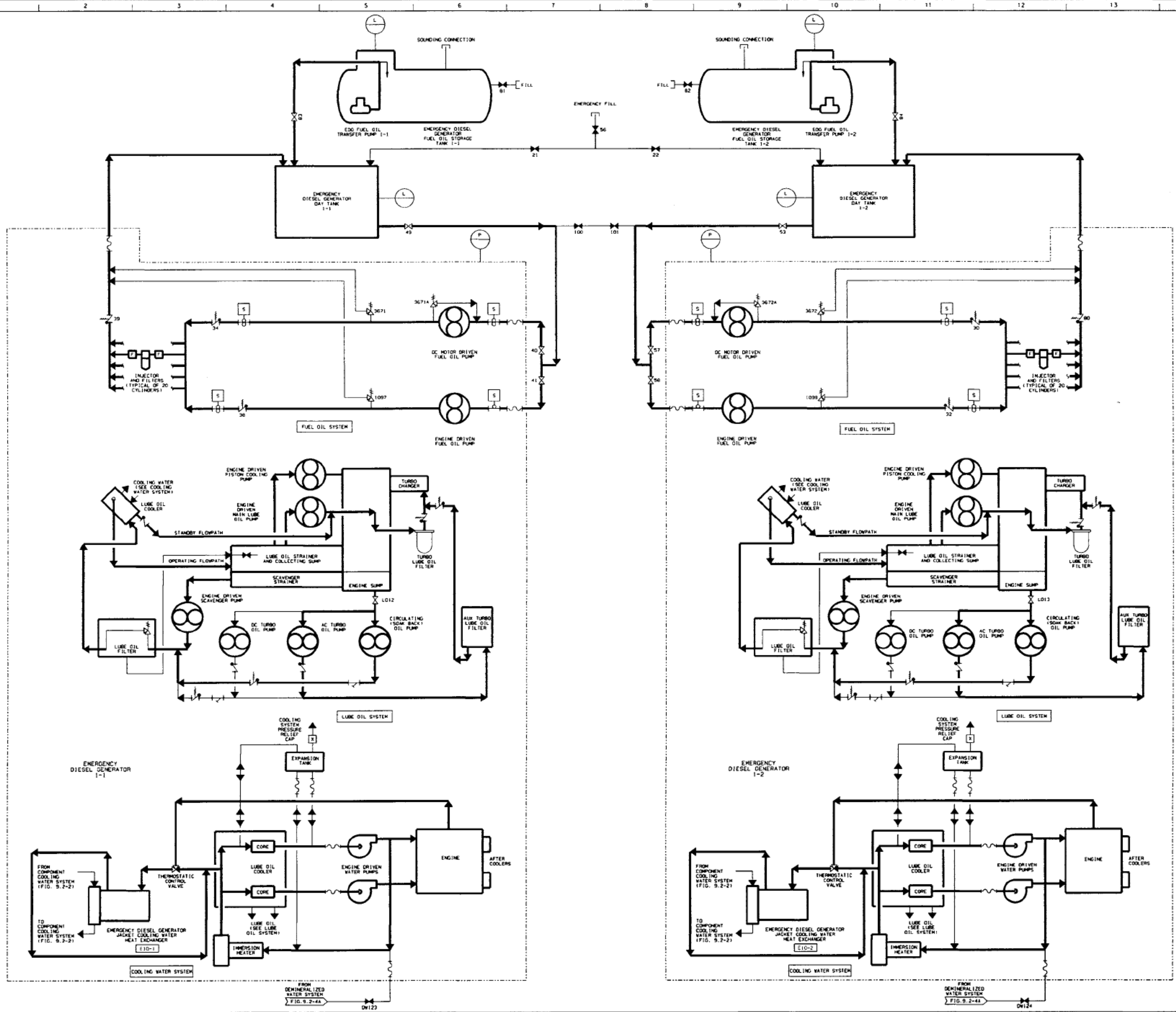
FUNCTIONAL DRAWING
STATION FIRE PROTECTION SYSTEM

FIGURE NO
FIGURE 9.5-1

DB:05-27-16
DFN:G/USAR/UFIG951.DGN

NOTES

1. ALL VALVES ARE PREFIXED WITH "DO" UNLESS OTHERWISE NOTED.



THE CN CHANGE FOR THIS FIGURE WAS TO REPRINT IT TO SHOW CORRECT LINE WEIGHT AS REQUIRED BY DBBP-DM-1002, ATTACHMENT 11. THERE WAS NO PHYSICAL OR TECHNICAL CHANGES MADE. WHOLE DRAWING NOT CLOUDED FOR CLARITY. REMOVE THIS NOTE AT NEXT UPDATE.

NO	DATE	REVISIONS	BY	CHK	ENGR	SA SUPV	MED MGR
3		INC. CN 14-245	SBW				
2		INC. UCN 10-006	INITIALS ON FILE				
1		INC. UCN 93-086	INITIALS ON FILE				
0	05/29/98	INFORMATION ONLY	INITIALS ON FILE				

SCALE NONE | DESIGNED MJD | DRAWN BEW | DATE 05-19-90

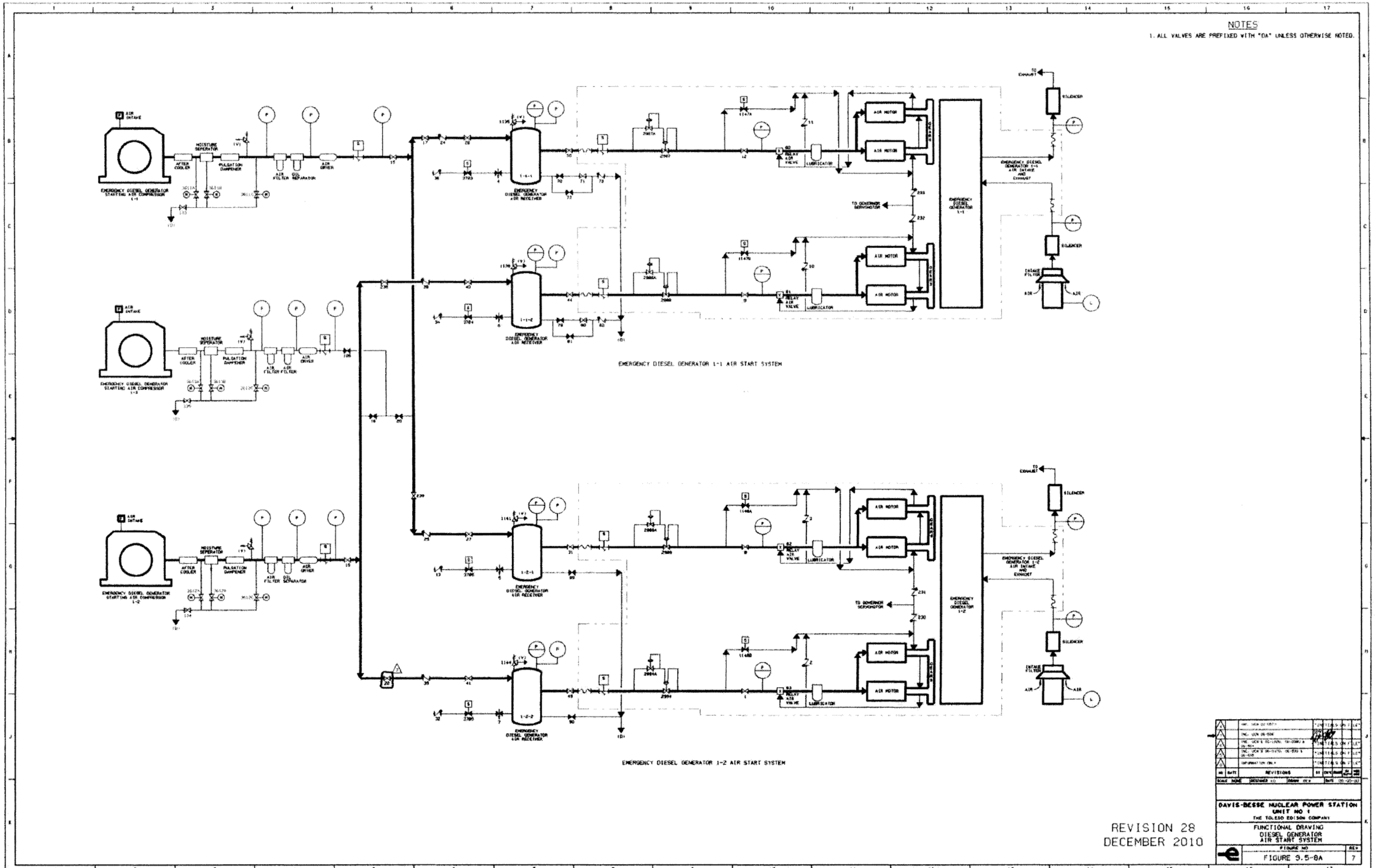
DAVIS-BESSE NUCLEAR POWER STATION UNIT NO 1
THE TOLEDO EDISON COMPANY

FUNCTIONAL DRAWING
EMERGENCY DIESEL GENERATOR AUXILIARY SYSTEMS

FIGURE NO: FIGURE 9.5-8 | REV: 3

REVISION 30
OCTOBER 2014

DFN=G/USAR/UF1G958.DGN
DB 10-07-14


NOTES
1. ALL VALVES ARE PREFIXED WITH "DA" UNLESS OTHERWISE NOTED.
EMERGENCY DIESEL GENERATOR 1-1 AIR START SYSTEM
EMERGENCY DIESEL GENERATOR 1-2 AIR START SYSTEM
AIR MOTOR
EMERGENCY DIESEL GENERATOR 1-1
EMERGENCY DIESEL GENERATOR 1-2
SILENCER
TO EXHAUST
TO GOVERNOR SERVOMOTOR
LUBRICATOR
RELAY AIR VALVE
AFTER COOLER
MOISTURE SEPERATOR
PULSATION DAMPENER
AIR FILTER
OIL SEPARATOR
AIR DRYER
AIR INTAKE
EMERGENCY DIESEL GENERATOR AIR RECEIVER
REVISION 28
DECEMBER 2010
DAVIS-BESSE NUCLEAR POWER STATION
UNIT NO 1
THE TOLEDO EDISON COMPANY
FUNCTIONAL DRAWING
DIESEL GENERATOR
AIR START SYSTEM
FIGURE 9.5-8A

Removed in Accordance with RIS 2015-17

DAVIS-BESSE NUCLEAR POWER STATION

DIESEL GENERATOR DAY TANK LOCATION

FIGURE 9.5-10

9.6 REFERENCES

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