CHAPTER 9: AUXILIARY SYSTEMS

9.1 FUEL STORAGE AND HANDLING

9.1.1 New-Fuel Storage

9.1.1.1 Design Bases

The new-fuel storage racks, as shown in Figure 9.1-1, are designed to maintain sufficient spacing between the new-fuel assemblies under fully loaded conditions to ensure that the array will limit the effective multiplication factor (k_{eff}) to $\leq$0.90 for the dry condition and to $\leq$ 0.95 in the event of complete flooding of the storage vault. New-fuel storage racks are supplied for 30 percent of the full core fuel load.

These racks are designed to withstand combined loadings, including impact and seismic disturbance, to ensure against damage to the racks or distortion of the fuel storage arrangement.

The new-fuel storage vault is designed to preclude flooding of the new-fuel assemblies.

9.1.1.2 Facilities Description

9.1.1.2.1 New-Fuel Storage Vault

After receipt, uncrating, and transfer to the operating floor, the new fuel may be placed in dry storage in racks. These racks are contained in a Category I new-fuel storage vault. The new fuel may be placed in the new fuel storage vault, provided criticality concerns are addressed as outlined in letter EF2-61,906. The vault, shown in Figure 9.1-2, accepts 23 new-fuel storage racks, each of which accommodates 10 new-fuel assemblies. The 230-assembly capacity of the vault amounts to 30 percent of the 764 assemblies in the reactor. The vault dimensions are shown in Figure 9.1-2.

The vault is closed at the top by a shield plug 12 in. thick. The shield plug is divided into five sections, each with redundant lifting rings. The openings and shield plugs are steel lined. The plugs extend 4 in. above the refueling floor. The vault floor slopes to an open drain located in the center of the vault floor.

The new-fuel vault is served by the reactor building crane.

9.1.1.2.2 New-Fuel Storage Racks

The new-fuel storage racks provide a place for storing new fuel in the new-fuel storage vault, as shown in Figure 9.1-1. The location of the new-fuel storage vault within the station complex is shown in Figure 9.1-3. Each new-fuel storage rack holds up to 10 channeled or unchanneled fuel assemblies in a row, spaced nominally 6.625 in. apart, center-to-center.

The new-fuel storage racks are designed so that arrangement in rows on a nominal 11.5-in. center-to-center spacing between rows limits the k_{eff} of the array to $\leq$ 0.90 for the dry condition. The k_{eff} is $\leq$ 0.95 in the event of complete flooding of the storage vault. The fuel assemblies are loaded into a rack through a hole in the top of each rack. Each hole for a fuel

assembly has adequate clearance for the insertion or withdrawal of the assembly while enclosed in a protective plastic wrapping. Guides are provided to guide the fuel element spacers the full length of their insertion into the rack so that damage to the fuel assemblies is precluded. The spacers and the upper tie plate of the fuel element rest against the rack to provide lateral support. The design of the racks prevents accidental insertion of the fuel assembly in a position not intended for the fuel. The weight of the fuel assembly is supported by the lower tie plate which is seated in a chamfered hole in the rack base. The new-fuel racks can withstand an upward force of 6000 lb.

9.1.1.3 Safety Evaluation

Calculations of k_{eff} are based upon the geometrical arrangements of the fuel array, and subcriticality does not depend upon the presence of neutron-absorbing materials. The arrangement of the fuel assemblies in the fuel storage racks results in a k_{eff} below 0.90 in a dry condition or in the absence of moderator. In an abnormal condition, if the fuel array were to be flooded with water, k_{eff} would not exceed 0.95. The criticality analysis for initial licensing of the new-fuel storage vault is provided in Reference 1. Use of the new fuel storage vault is currently restricted as discussed in Section 9.1.1.2.1.

The new-fuel storage racks are designed to meet Category I requirements as described in Section 3.2. Stresses in a fully loaded rack will not exceed stresses specified by ASTM Specifications (B108, B179, B209, and B221) on light-weight metal alloys when subjected to a 1.5g horizontal acceleration.

The storage rack structure is designed to withstand the impact resulting from a falling object possessing 2000 ft-lb of kinetic energy. The structural arrangement is such that no lateral displacement of the fuel occurs; therefore, subcritical spacing is maintained.

The new-fuel racks are designed to be restrained by hold-down bolts in case a stuck fuel assembly is inadvertently hoisted and to ensure that rack spacing does not vary under specified loads. The rack structure and hold-down bolts are designed to maintain the minimum required cell spacing due to forces that might occur if a fuel bundle were to jam in the rack during removal.

The new-fuel storage racks are made from aluminum. All welds are in accordance with GE standards which are based on ASME Section IX and ASTM Standards Part 6. The material choice is based on a consideration of the susceptibility of various metal combinations to electrochemical reaction. When considering the susceptibility of metals to galvanic corrosion, aluminum and 300-series stainless steel are relatively close together, insofar as their coupled potential is concerned. The use of stainless steel fasteners in aluminum to avoid detrimental galvanic corrosion is a recommended practice and has been used successfully for many years by the aluminum industry.

9.1.1.4 Testing and Inspection

The new-fuel storage racks do not require any special periodic testing or inspection for nuclear safety purposes.

9.1.2 Spent Fuel Storage

9.1.2.1 Design Bases

Spent fuel storage space is provided to accommodate 3590 fuel assemblies. Stainless steel high-density fuel storage racks are provided for all fuel assemblies. The spent fuel assemblies are placed in racks designed to ensure a k_{eff} equivalent of $\leq$0.95 with the spent fuel pool filled with unborated water at 68°F for both normal and abnormal storage conditions.

The calculated k_{eff} includes margins for uncertainty in the calculations, including mechanical tolerances, which are statistically combined such that the true k_{eff} will be less than 0.95 with a 95 percent probability at a 95 percent confidence level.

To ensure that the analysis followed a conservative approach, the criticality calculations for the high-density racks were performed with the following criteria:

a. Initial uniform enrichment of 4.9 weight percent ^{235}U with credit for gadolinia burnable poison normally present

b. Maximum reactivity evaluated at the point of peak reactivity over burnup

c. Both unchanneled and channeled fuel with maximum expected distortion

d. Abnormal and accident conditions considered

e. Lattice of storage racks is infinite in all directions; that is, no credit for axial or radial neutron leakage

f. Unborated water at 20°C.

To simplify the analysis, no credit is taken for:

a. Neutron absorption in minor structural members; that is, spacers and Inconel springs are replaced by water in the calculation

As indicated in Section 4.0 of Reference 1 and Section 4.0 of Reference 3, the results of the criticality analysis for the spent fuel racks show that for all normal and abnormal storage conditions, the calculated k_∞ is below the criterion of $k_{eff} \leq 0.95$.

The spent fuel storage racks (SFSR) are designed such that no fuel assembly can be placed within the rack array in other than a design storage location.

The spent fuel pool and storage racks, containing their full complement of fuel, are designed to meet Category I requirements.

Reference 1 documents the Abnormal and Accident Conditions analyzed for the Holtec High Density Racks. Reference 3 documents similar analyses performed for the Joseph Oat High Density Racks.

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

Shielding for the spent fuel storage arrangement is sufficient to protect plant personnel so that exposure to radiation is well within the Occupational Limits of 10 CFR 20.101.

The spent fuel storage facilities provide the capability to limit the potential offsite exposures in the unlikely event of significant release of radioactivity from the stored fuel to a fraction of 10 CFR 100 limits.

9.1.2.2 Facilities Description

9.1.2.2.1 Spent Fuel Pool

The two main functions of the Category I spent fuel pool are to provide a storage place for irradiated fuel and other radioactive equipment requiring shielding and to provide a convenient area for performing work on selected radioactive equipment. The spent fuel pool is shown in Figure 9.1-3. The spent fuel pool has an inside length of 40 ft, an inside width of 34 ft, and a height of 38 ft 9 in. The surface of the water is maintained at Elevation 683.5 ft (New York Mean Tide, 1935) by scuppers that act as skimmers and wave suppressors. This results in a minimum water depth for shielding of 7 ft above the top of the fuel while it is being moved over storage racks. The Technical Specifications require that a minimum of 22 ft of water be maintained over the top of irradiated fuel assemblies in the spent fuel storage pool racks during movement of irradiated fuel assemblies in the spent fuel storage pool. Pool water-level indication is painted on the north and east walls of the spent fuel pool starting at 18 ft above the stored fuel assemblies.

The spent fuel pool is of poured reinforced-concrete construction with an all-welded stainless steel plate liner. The water in the spent fuel pool is filtered, demineralized, and cooled as described in Subsection 9.1.3.2.

The stainless steel spent fuel pool liner is designed in accordance with the following codes and standards: ASME Boiler and Pressure Vessel (B&PV) Code Section VIII, Division 1, Subsection B (for welds); and ACI 347, recommended practice for concrete formwork (for tolerances). The liner can withstand thermal loads due to operating temperatures of 125° to 150°F (assume installation temperature of 70°F) and thermal loads due to abnormal temperatures inside the pool of 212° and external temperatures of 150°F. The liner plate is designed based on an acceptance criterion that neither the construction allowable tensile stress (f_t), nor the allowable compressive stress (f_c), exceeds 0.67 yield stress (f_y). The normal allowable compressive membrane strain is 0.003 in./in. and the abnormal allowable compressive membrane strain is 0.005 in./in. All welded seams in the pool liner are backed by channels to collect any possible leakage. All the channels are interconnected to the bottom peripheral drain with four separate outlet drains that are used to monitor leakage. The leaktight integrity of the spent fuel pool liner is verified to be upheld in a postulated fuel assembly drop accident. (Analysis shows that the fuel assembly can be dropped 17 ft 7 in. without penetrating the liner or causing overall slab instability.)

Two self-sealing gates with a monitored drain between them separate the spent fuel pool from the reactor well pool.

Should there be leakage, this arrangement would permit the repair of a gate without disturbing the integrity of the spent fuel pool. Each gate has two solid-rubber seals that seal to the spent fuel pool liner.

Spent fuel assemblies will be stored in thirteen (13) Joseph A. Oat supplied high-density stainless steel racks (designated F1600E002A through H and F1600E002J through N) (See Figures 9.1-4 for the design details of these racks.) These racks have a nominal 6.22-in. center-to-center distance between assemblies (pitch). Each of these racks each contain 169 storage cells. There are nine (9) additional Holtec supplied racks (designated F1600E011A through I) with a nominal 6.23-in. pitch, (See Figure 9.1-15 for the design details of these racks). The spent fuel storage rack description, individual rack size and the number of storage cells are summarized below:

Spent Fuel Pool

Rack and Storage Capacity Summary

Rack Description	Size	Total Number of Cells
F1600E002A through H	13 X 13	169 X 8 = 1352
F1600E002J through N	13 X 13	169 X 5 = 845
F1600E011A and B	9 X 11	99 X 2 = 198
F1600E011C	19 X 19	361 X 1 = 361
F1600E011D	12 X 17	204 X 1 = 204
F1600E011E	9 X 17	153 X 1 = 153
F1600E011F	9 X 12 minus 28 Cell cutout	80 X 1 = 80
F1600E011G	9 X 13 minus 28 Cell cutout	89 X 1 = 89
F1600E011H	9 X 14	126 X 1 = 126
F1600E011I	13 X 14	182 X 1 = 182
Storage Cell Total		3590

Two of the above cells may be used for Boraflex in-service surveillance program.

The arrangement of the fuel storage modules is shown in Figure 9.1-3 (Sheet 3). Four dual purpose cells in F1600E011F and six dual purpose cells in Rack F1600E011G are provided to store defective fuel assembles, control rods, control guide tubes and other equipment and components.

The spent fuel pool is also used to store 104 control rods that are suspended in a vertical position from 52 hooks. These hooks are mounted on frames located adjacent to the spent fuel shipping cask restraining framework on the west wall and on the south wall of the pool. A total of 114 control rods (104 suspended from hooks plus 10 in the dual-purpose cell racks) can be accommodated in the spent fuel pool. If storage for more than 114 rods should ever be required, the rods can be supported by mounting additional hooks on the equipment lugs on the east wall of the pool or control rod curb hanger(s) may be utilized. Temporary storage racks may also be used, as needed, to store control rod blades.

The area in the vicinity of the north wall of the spent fuel pool is laid out as a working area. In Figure 9.1-3, Sheet 3, two fuel-preparation machines are shown, an outline of which is shown in Figure 9.1-8. The function of the fuel-preparation machines is to remove and replace fuel bundle channels. The fuel preparation machines are used for fuel inspections and new fuel receipt/transfer activities. Strict administrative control on the fuel preparation machine's full-up end stop is required for personnel protection. Holtec high density Racks F1600E011C and F1600E011H are designed to support a specially engineered overhead platforms referred to as "Holtec Overhead Platforms (HOPs)" on top of the rack which permits storage of miscellaneous objects up to five tons total dry weight without interfering with the normal functions of the module. The structural and thermal-hydraulic qualification of these racks includes the appropriate consideration of the overhead platform.

A special storage area is provided on the west wall of the spent fuel pool to accommodate the spent fuel shipping cask. Details of the storage area are given in Subsection 9.1.4.2.1.

The spent fuel pool is supplied with several types of underwater lights; some provide general illumination and others provide specific local illumination.

The spent fuel pool shielding design objectives and design criteria are presented in detail in Subsections 12.1.1 and 12.1.2. Special provisions for ventilation of the fuel pool area are discussed in Subsection 12.1.1.1. Estimates of exposure to plant personnel in the fuel pool area are presented in Subsection 12.1.5.2.2.

9.1.2.2.2 Spent Fuel Storage Racks

Spent fuel storage racks provide a place in the spent fuel pool for storing spent fuel assemblies received from the reactor pressure vessel (RPV). There are two types of high density spent fuel storage racks (Holtec and Oat) being used. Both are full-length top entry racks, designed to preclude the possibility of criticality under normal and abnormal conditions.

The original Oat high-density spent fuel storage racks are of welded stainless steel construction with a neutron absorber sandwiched between the stainless steel sheets. The neutron absorber is marketed under the trade name of Boraflex, supplied by BISCO of Park Ridge, Illinois. The original high-density spent fuel racks are designed and fabricated by the Joseph Oat Corporation of Camden, New Jersey.

The basic philosophy of the high-density spent fuel storage rack design is consistent with the NRC Position Paper (Reference 2), General Design Criteria (GDC) 61 and 62, and Standard Review Plan (SRP) Sections 9.1.1 and 9.1.2. Seismic classification and analysis are in accordance with Regulatory Guides 1.29, 1.61, and 1.92.

The modules of the Holtec high density spent fuel racks are square cross-section boxes. Each box is equipped with Boral neutron absorber panels on its sides to form a composite box.

Reference 3 contains a detailed discussion of various aspects of the Oat high-density fuel rack design, analysis, and fabrication.

The high-density spent fuel storage racks provide for a total of 3590 storage locations arranged in 22 racks. All the racks are freestanding; that is, they are not anchored to the pool floor or connected to the pool wall through snubbers or lateral restraints. For the new racks,

the minimum gap between adjacent racks is 1.0 in. at all locations, and the nominal gap between the fuel pool wall and storage rack is 2.39 in. The respective gaps for the Oat racks are 3.625 and 24 to 28 inches.

Of the 3590 storage locations, 3588 locations are intended for the storage of spent fuel assemblies. Two locations are designated for the Boraflex neutron absorber material surveillance program. This program is described in plant procedures. An additional 53 locations are inaccessible due to interferences. This will reduce the available storage locations to 3535. If the two Holtec Overhead Platforms (HOPs) are installed, an additional four cells per HOP will be unavailable.

The Oat high-density fuel storage racks are constructed from SA-240, type 304, austenitic stainless steel sheet material; SA-240, type 304, austenitic stainless steel plate material; and SA-182, type F304, austenitic stainless steel forging material. Boraflex, a patented brand name product of BISCO, serves as the neutron absorber material in the Oat high-density racks. Boraflex material has been tested in a fuel-pool-like environment to a minimum of 1.03×10^{11} rad and found to perform satisfactorily. Boraflex has been observed, however, to shrink and develop gaps. These effects have been evaluated in Reference 7. Alterations in physical properties and offgassing due to irradiation and material chemical or galvanic interactions with the rack structure have been considered in the design of the rack.

The Holtec high density racks are constructed from SA-240, type 304L, austenitic stainless steel sheet and plate material; and male spindles (lower part of support feet) of SA-564-630; age hardened at 1100°F. The storage cells are 6.035 ± 0.04 in. square cross-section. Modules F1600E011E-I are 3 inches shorter than the other Holtec racks with the result that the bale handle is exposed. This is for ease of fuel handling operations. They employ Boral as the neutron absorber material. Boral is a thermal neutron poison material composed of boron carbide and 1100 alloy aluminum. The composite boxes containing the Boral panels and enveloping sheathing are arrayed in a vertical fixture over a solid monolithic baseplate. Boron-10 is the neutron-absorptive agent in both Boraflex and Boral. The boron carbide and aluminum materials in Boral do not degrade as a result of long-term exposure to radiation.

The Oat high-density fuel storage rack contains storage cells that have a 6-in. minimum (+0.125 in., -0 in.) internal cross-sectional opening. These cells are straight to within ± 1/8 in. These dimensions ensure that fuel assemblies with maximum permissible out-of-straightness can be inserted into the storage cells without interference.

To illustrate the elements that make up a typical Oat rack, Figure 9.1-10 shows a horizontal cross section of an array of 3 x 3 cells. (A typical Oat high-density fuel storage rack has an array of 13 x 13 cells.)

The construction of the Oat racks may be best described by the basic building blocks of the design, namely the "cruciform," "ell," and "tee" elements, shown in Figure 9.1-11. The cruciform element is made of four angular subelements, "A" (Figure 9.1-12), with the neutron absorber material tightly sandwiched between the stainless steel sheets. The long edges of the cruciform are welded using a 0.070-in.-thick stainless steel backing strip as shown in Figure 9.1-13. The bottom of the cruciform assembly has a 4-1/4-in.-high stainless steel strip that prevents slippage of the poison material downward due to gravitation loads or operating conditions. The top of the cruciform is also end welded, using a spacer strip as shown in Figure 9.1-13. Skip welding at the top ensures proper venting for off-gassing.

The "ell" and "tee" elements are constructed similarly using angular subelements “A” and “B” and flat subelements "C" (Figures 9.1-12 and 9.1-14). Having fabricated the required quantities of the "cruciform," "tees," and "ells," the assembly is welded in a specially designed fixture that serves the vital function of maintaining dimensional accuracy while fillet welding the adjacent spokes of all elements. In this manner, the cells are produced that are bonded to each other along their long edges, thus, in effect, forming an "egg crate." The following manufacturing deviations were detected subsequent to installation in the spent fuel pool: a cruciform was installed upside down in F1600E002J (Module A9), and a tee was installed upside down in F1600E002L (Modules A11). The effects of these manufacturing deviations have been analyzed in Reference 7 and determined to be acceptable for use.

The bottoms of the cell walls are welded to the base plate, which has 4.75-in.-diameter holes concentric with cell center lines. Carefully machined sleeve elements are positioned in the base plate and are fillet welded to the base plate (Figure 9.1-15). The conical machined surface of the sleeve provides a contoured seating surface for the "nose" of the fuel assembly. Thus, the contact stresses at the fuel assembly nose bearing surface are minimized.

The central hole in the sleeve provides the coolant flow path for heat transport from the fuel assembly cladding. Lateral holes in the cell walls (Figure 9.1-15, Sheet 1) provide the redundant flow path in the unlikely event that the main coolant flow path is clogged.

The composite box assemblies of the Holtec high density racks are arrayed in a vertical fixture over a solid monolithic baseplate which is machined with an array of equispaced cylindrical holes containing tapered crowns (Figure 9.1-15, Sheets 2 and 3). These tapered holes serve as the seating surface for the nose of the fuel assembly.

The high-density fuel storage rack assembly is supported at four corners. The supports elevate the rack base plate 7.5 in. above the pool floor level, thus creating the water plenum for coolant inventory (Figures 9.1-4 and 9.1-5).

The high-density fuel storage racks are designed to meet Category I requirements, in accordance with Regulatory Guide 1.29. They are required to remain functional during and after a safe-shutdown earthquake (SSE). As noted previously, these high-density fuel storage racks are neither anchored to the pool floor nor attached to the side walls. The individual rack modules are not interconnected. Furthermore, a high-density fuel storage rack may be completely loaded with fuel assemblies (which corresponds to greatest rack inertia), or it may be partially loaded so as to produce maximum geometric eccentricity in the structure.

Dynamic simulation analyses involving nine Holtec high density and thirteen Oat high density spent fuel storage racks have been performed to establish the structural margins of safety. Six simulations modeled 22 high density fuel racks (13 Oat and 9 Holtec) in the pool for campaign II with a comprehensive Whole Pool Multi Rack (WPMR) model. References 1 and 3b presents the incorporation of all relevant physical data into the computer code DYNARACK which then handles simultaneous simulation of all racks in the pool as a WPMR 3-D analysis. Some classical single rack runs were also performed. Parameters varied were interface coefficients of friction and extent of storage locations occupied by spent nuclear fuel, ranging from nearly empty to full. A WPMR run was also performed with four Holtec racks installed in the SFP for the same scenario that resulted in the greatest displacement from among the parametric runs. The results show that all stresses are well

below their corresponding ASME Section III, Subsection NF limits and there is no rack-to-rack or rack-to-wall impact. Results also show that rack overturning is not a concern.

Nonlinearities of fuel-to-rack impact and, if appropriate, rack-to-rack impact are included. The racks are designed as freestanding and the effects of rack slide are addressed. Hydrodynamic effects are also included. No additional credit for structural damping is taken unless substantiated by testing or detailed analysis.

Synthetic time-histories in three orthogonal directions (N-S, E-W, and vertical) are generated in accordance with the provisions of SRP 3.7.1. The SRP calls for both the response spectrum and the power spectral density corresponding to a generated acceleration time-history to envelop their target (design basis) counterparts with only finite enveloping infractions. The time-histories for the pool have been generated to satisfy this preferred (and more rigorous) criterion. The seismic files also satisfy the requirements of statistical independence mandated by SRP 3.7.1. Time-history accelerograms were generated for a 20-sec duration of OBE and SSE events, respectively. These artificial time-histories are used in all the non-linear dynamic simulations of the racks.

The time-history data were generated from the floor response spectra given in Section 3.7. These spectra are enveloped with a smooth design spectra. For a complete time-history analysis of the equipment situated on the pool floor, artificial time-history accelerations in three orthogonal directions were generated so that their corresponding response spectra will envelop the smoothed design spectra mentioned above. These artificial time-history series were also verified to be statistically independent.

Figure 9.1-16 displays the dynamic model for the high-density fuel storage racks. Features of the dynamic model are as follows.

a. The fuel rack structure motion is captured by modeling the rack as a 12 degree-of-freedom structure. Movement of the rack cross-section at any height is described by six degrees-of-freedom of the rack base and six degrees-of-freedom at the rack top. In this manner, the response of the module, relative to the baseplate, is captured in the dynamic analyses once suitable springs are introduced to couple the rack degrees-of-freedom and simulate rack stiffness

b. Rattling fuel assemblies within the rack are modeled by five lumped masses located at H, .75H, .5H, .25H, and at the rack base (H is the rack height measured above the baseplate). Each lumped fuel mass has two horizontal displacement degrees-of-freedom. Vertical motion of the fuel assembly mass is assumed equal to rack vertical motion at the baseplate level. The centroid of each fuel assembly mass can be located off-center, relative to the rack structure centroid at that level, to simulate a partially loaded rack

c. Seismic motion of a fuel rack is characterized by random rattling of fuel assemblies in their individual storage locations. All fuel assemblies are assumed to move in-phase within a rack. This exaggerates computed dynamic loading on the rack structure and, therefore, yields conservative results

d. Fluid coupling between rack and fuel assemblies, and between rack and wall, is simulated by appropriate inertial coupling in the system kinetic energy.

Inclusion of these effects uses the methods of References 3c and 3d for rack/assembly coupling and for rack-to-rack coupling

e. Fluid damping and form drag are conservatively neglected

f. Sloshing is found to be negligible at the top of the rack and is, therefore, neglected in the analysis of the rack

g. Potential impacts between the cell walls of the new racks and the contained fuel assemblies are accounted for by appropriate compression-only gap elements between masses involved. The possible incidence of rack-to-wall or rack-to-rack impact is simulated by gap elements at the top and bottom of the rack in two horizontal directions. Bottom gap elements are located at the baseplate levation. The initial gaps reflect the presence of baseplate extensions, and the rack stiffnesses are chosen to simulate local structural detail

h. Pedestals are modeled by gap elements in the vertical direction and as "rigid links" for transferring horizontal stress. Each pedestal support is linked to the pool liner (or bearing pad) by two friction springs. The spring rate for the friction springs includes any lateral elasticity of the stub pedestals. Local pedestal vertical spring stiffness accounts for floor elasticity and for local rack elasticity just above the pedestal. Details of the derivation and computation of the element stiffnesses are given in Reference 4

i. Rattling of fuel assemblies inside the storage locations causes the gap between fuel assemblies and cell wall to change from a maximum of twice the nominal gap to a theoretical zero gap. Fluid coupling coefficients are based on the nominal gap in order to provide a conservative measure of fluid resistance to gap closure

j. The model for the rack is considered supported, at the base level, on four pedestals modeled as non-linear compression only gap spring elements and eight piecewise linear friction spring elements; these elements are properly located with respect to the centerline of the rack beam, and allow for arbitrary rocking and sliding motions.

The high-density spent fuel storage racks are designed to withstand the most severe environmental, loading, and seismic conditions assumed to occur simultaneously. Load combinations are in accordance with SRP Section 3.8.4.

The structural acceptance criteria are in accordance with ASME B&PV Code Section III, Subsection NF.

The breakdown of the load combinations and acceptance limits is as follows:

D + L	Normal limits of NF 3231.1a
D + L + E	Normal limits of NF 3231.1a
D + L + T_o	The lesser of 2 S_y and S_u
D + L + T_o + E	The lesser of 2 S_y and S_u
D + L + T_a + E	The lesser of 2 S_y or S_u

$D + L + T_a + E^1$ Faulted condition limits of NF 3231.1c

where

D	=	dead loads of racks, buoyant rack weight
L	=	live loads, buoyant fuel weight
E	=	operating-basis earthquake (OBE) seismic loads including impact of fuel due to clearance within rack
T_o	=	operating thermal load
T_a	=	accident thermal load based on pool temperature of 212°F
E^1	=	safe-shutdown earthquake seismic loads including impact of fuel due to clearance within rack

In addition to thermal and seismic loads, the high-density spent fuel storage racks are designed to withstand each of the following loadings superimposed on the submerged rack dead weight plus the weight of any stored fuel:

a. A 1200-lb uplift force applied at the top of the rack in the "weakest" storage location. The force is assumed to be applied on one wall of the storage cell boundary as an upward shear force. The damage, if any, is limited to the affected storage locations

b. Fuel assembly dropped on top of the rack with an impact energy of 2000 ft-lb. The impact energy is assumed to correspond to a buoyant mass of 600 lb dropped from a height of 40 in. The impact is assumed to occur on the top ridge of the rack

c. A horizontal force of 1000 lb applied at the most vulnerable location on the top of the rack. The load is assumed to act over the width of one storage cell. The subcriticality of all fuel assemblies is to be maintained with a $k_{eff} \leq 0.95$

d. A fuel assembly (assumed buoyant weight = 600 lb) dropping 16 ft 9 in. through a storage location and impacting the base. Local failure of the base plate is acceptable; however, an impact with the pool liner is not allowed. The subcriticality of all fuel assemblies is to be maintained with a $k_{eff} \leq 0.95$.

The allowable stress criteria are in accordance with ASME B&PV Code Section III, Subsection NF. The high-density spent fuel storage racks were checked for normal operating conditions, severe environmental conditions (OBE), extreme environmental conditions (SSE), and abnormal plant conditions (pool temperature = 212°F), and were found to be satisfactory.

The design of the Fuel Pool Cooling and Cleanup System (FPCCS) is described in Subsection 9.1.3. The decay heat load generated by the stored spent fuel is calculated in accordance with NRC Branch Technical Position ASB 9-2 (BTP ASB 9-2).

The spent fuel pool capacity expansion mentioned in Subsection 9.1.2.2 will occur over a series of campaigns, with the storage capacity increasing after each campaign. The bounding configuration from a thermal-hydraulic standpoint is the final, maximized configuration. This involves the largest number of stored assemblies and therefore the highest SFP decay

heat load and lowest net SFP thermal capacity. All analyses discussed in this section are performed for the final configuration and thereby bound all intermediate configurations.

Thermal-hydraulic qualification analyses for the modified rack array are as follows:

a. Evaluation of the maximum bulk temperature. The bulk temperature is limited to 150°F for all conditions where forced cooling is available

b. Evaluation of loss-of-forced cooling scenarios, to establish minimum times to perform corrective actions and maximum makeup water requirements

c. Determination of the maximum local temperature in the pool to establish that localized boiling in the SFSRs is not possible

d. Evaluation of the maximum temperature difference between the fuel rod cladding and the local SFP water, to establish that nucleate boiling is not possible while SFP forced cooling is operating.

For each of the above analyses, evaluation is performed for an analytically bounding scenario as detailed below.

During normal SFP operation, the maximum normal bulk temperature of 150°F is based on the Fuel Pool Cooling and Cleanup System with the SFP gates installed.

In the scenario addressed, an emergency full core discharge comprised of 764 assemblies is discharged into an SFP that already contains 4016 previously discharged assemblies. This analyzed fuel inventory (4780) conservatively exceeds the maximum possible inventory of 4608 assemblies. The minumum decay time of the previously discharged fuel assemblies for this scenario is 12 months. In addition to those mentioned above, the following conservative framework is applied in the maximum pool bulk temperature calculation:

a. The decay heat load is based on a discharge schedule with bounding parameters (i.e., maximum irradiation time and batch size) for all projected discharges.

b. Design temperatures are used for the coolant water flow inlet to the FPCCS and RHR System heat exchangers.

For evaluating the minimum time-to-boil and corresponding maximum boiloff rate, the following conservatisms are applied:

a. Loss of forced cooling is assumed to occur coincident with the SFP peak decay heat generation.

b. The thermal capacity of the SFP is based on the net SFP water volume only. The energy storage capability of the fuel racks, fuel assemblies and pool structure is neglected.

c. Makeup water supplied to maintain the SFP water level is assumed to be at the coincident SFP bulk temperature. No credit is taken for the difference in enthalpy between the SFP and the cooler makeup water, maximizing the boiloff rate.

For the same pool inventory described above for the maximum pool temperature case, the results of the analysis show that, in the extremely unlikely event of a complete failure of both the FPCCS and RHR System, there would be sufficient time available for corrective actions.

The maximum water boiloff rate is less than the minimum available makeup capacity of 100 gpm available from the condensate storage tanks. Additional sources of makeup are also available.

In order to determine an upper bound on the maximum local water temperature, a series of conservative assumptions are made:

a. The walls and floor of the SFP are modeled as adiabatic surfaces, neglecting conduction heat loss through them.

b. Heat losses by thermal radiation and natural convection from the SFP surface are neglected.

c. The rack-to-wall gaps are modeled as 2 inches wide. The actual rack-to-wall gaps are larger.

d. The bottom plenum gap used in the model is approximately 50 percent of the actual gap.

e. No downflow is assumed to exist between the rack modules.

f. The hydraulic resistance of every SFSR cell is determined based on the most hydraulically limiting fuel assembly type.

g. The hydraulic resistance of every SFSR cell is determined based on the most restrictive water inlet geometry of the cells over the rack support pedestals.

h. The hydraulic resistance of every SFSR cell includes the effects of blockage due to an assumed dropped fuel assembly lying horizontally on top of its rack. This condition bounds the effects of contemplated overhead platforms, which add little extra flow resistance because of the large (~16 in.) spacing between the cell exit and the platform. Blockage due to a dropped assembly also bounds that of a vertically blocked gate, because the width of a channeled assembly is larger than the gate thickness.

i. With a full core discharged into the SFP racks and placed approximately equidistant from the coolant water inlet and outlet, the remaining cells in the spent fuel pool are postulated to be occupied with previously discharged fuel.

j. The in-pool sparger piping is modeled as truncated above the elevation of the racks.

k. In the evaluation of local water temperatures in dual-purpose rack cells containing loaded damaged fuel containers, only two of the cell baseplate holes are credited. This conservatively neglects the two additional baseplate holes and the eight cell side holes, thereby yielding greater than 100 percent redundancy in these cells.

The maximum local water temperature is substantially lower than the local boiling temperature at the top of the SFSRs and nucleate boiling does not occur anywhere within the Fermi 2 SFP.

Dose calculations (Subsection 9.1.3.3) performed based on the boiling condition indicate that the spent fuel pool cooling and cleanup system is adequate to provide reasonable assurance

that the plant can be operated without undue risk to the health and safety of the public, consistent with the requirements of Criterion 2 of 10 CFR 50, Appendix A.

9.1.2.2.3 ISFSI Storage Pad

The function of the Independent Spent Fuel Storage Installation (ISFSI) storage pad is to provide a level resting surface for dry fuel storage casks. The pad is a 141' by 141' square reinforced concrete structure that is two feet thick designed to accommodate sixty four dry storage casks. The pad is compliant with ACI 349, "Code Requirements for Nuclear Safety-Related Concrete Structures," 2001, and designed in accordance with 10CFR Part 72-Licensing requirements for the independent storage of spent nuclear fuel, high-level radioactive waste, and reactor-related greater than Class C waste. The pad includes a surrounding fence with signage indicating that the pad is a radiologically restricted area. There is a subsurface drainage system surrounding the pad to help prevent the soil under the pad from being displaced as a result of freeze and thaw cycles. The subsoil in the area to the north of the pad has also been prepared for possible future expansion of the pad to allow additional placement of up to thirty two dry storage casks.

9.1.2.3 Safety Evaluation

The design of the spent fuel pool and storage racks meets the requirements of Regulatory Guide 1.13, except as noted in Appendix A of the UFSAR. Moreover, the spent fuel storage racks are designed to meet Category I requirements as described in Section 3.2.

The SFP will contain original Oat racks and Holtec racks introduced in Subsection 9.1.2.2.1. A wide range of conditions has been addressed in the design validation of all the racks. Examples relevant to the Oat racks include SFP water temperature increase to 212°F, the effect of manufacturing deviations (inverted cruciform and tees) and Boraflex panel degradation. Every Boraflex panel is assumed to have experienced an 11-12-in. gap (or reactivity equivalent) randomly distributed in the axial direction and losses in width and areal B-10 density due to thinning within the reactivity margin that is provided in the criticality analysis (see Reference 7). Reference 7 provides a discussion of the techniques and assumptions used in the criticality analysis of the Oat racks. The K_{eff} for the existing spent fuel storage racks, including an allowance for uncertainties is shown in the above references to be less than or equal to the 0.95 limit.

The criticality analyses for the high density fuel storage racks were performed with the MCNP code. MCNP (Monte Carlo N-Particle Transport) is a continuous energy Monte Carlo code. See Section 3.13.3.17.

As the SFP capacity expansion progresses, a greater and greater percentage of the pool will be occupied by Holtec high-density racks. Neutronic coupling between the Oat Boraflex racks and the Holtec Boral racks is precluded by the water gap between modules and by the absorber panels on both sides of the gap. Accordingly, to ensure that the design of the spent fuel storage racks provides for a $k_{eff} \leq 0.95$, the following normal and abnormal conditions are addressed in Reference 7b for the new racks:

a. The nominal design case, normal positioning of spent fuel assemblies in the spent fuel storage rack to act as a baseline

b. Temperature and water density effects: analyses are performed for a temperature decrease to 4°C (maximum water density), boiling (giving rise to voids) at various levels in the SFP. The calculations confirm that the reference temperature of 4°C is most conservative

c. Abnormal location of a fuel assembly was shown to have a negligible reactivity effect, while the nominal case was shown to yield a higher reactivity than a case with the fuel assembly moved to the corner of the storage rack cell (a four assembly cluster at closest approach)

d. The drop of a fuel assembly on a spent fuel storage rack. Such event was found to result in an insignificant increase in the calculated k_{eff}

e. The effect on reactivity of tolerances with respect to rack manufacture (stainless steel thickness, lattice spacing), B-10 loading, Boral width, fuel (enrichment and depletion) uncertainties has also been assessed.

The maximum calculated reactivity, including allowance for all uncertainties, was below 0.95.

Consideration has been given to various objects that might be dropped into the spent fuel pool and impact with the spent fuel storage racks. The spent fuel storage racks are designed to withstand the dropping of a fuel assembly as documented in Reference 1 for Holtec Racks and Reference 3 for Joseph Oat racks. A pool gate drop is evaluated to assess damage to the stored fuel assemblies in a fuel rack, the drop of an overhead platform during installation onto the top of a rack is evaluated, and an additional evaluation was also performed to consider the ability of the rack to withstand the uplift force from a stuck fuel assembly. The drop accident events postulated for the Fermi 2 fuel pool were analyzed and found to produce localized damage well within the design limits for the racks. Consequently, the spent fuel storage racks have a very large margin of safety.

The materials of the high-density spent fuel storage racks are described above in Subsection 9.1.2.2.2.

Provision is made to limit potential offsite exposures in case of significant release of radioactivity from the spent fuel. This would be done by using the standby gas treatment system (SGTS), when necessary, to control the release rate of radioactivity from the fuel storage area.

By maintaining the minimum spent fuel pool water level and the use of the normal reactor building ventilation, the exposure to plant personnel is maintained below the limits of 10 CFR 20.

Subsection 9.1.3.3 discusses the radiological consequences of loss of cooling to the spent fuel pool.

9.1.2.4 Testing and Inspection

A detailed and rigorous inspection of the Holtec high-density spent fuel storage modules is carried out at the fabrication facility prior to their release. The racks are also receipt inspected at the site before the racks are installed in the pool.

The design incorporates provisions for periodic testing of the Boraflex poison material throughout the life of the plant to verify the continued presence of a sufficient amount of neutron absorber in the spent fuel storage racks to maintain a $k_{eff} \leq 0.95$.

In situ verification of the poison material was performed at initial installation for the Oat racks to confirm the presence of the neutron absorber (Boraflex) in the spent fuel storage racks.

9.1.2.5 Reactivity of Fuel in Storage

The basic criterion associated with the storage of both irradiated (spent) and new fuel is that the effective multiplication factor of fuel stored under normal conditions will be ≤ 0.90 for the low density racks and ≤ 0.95 for the high density racks. Abnormal storage conditions are limited to a k_{eff} of 0.95 for both high and low density racks.

For the low density racks removed from the Spent Fuel Pool during Cycle 12, these storage criteria were satisfied if the uncontrolled lattice k_∞ calculated in the normal reactor core configuration met the following condition:

$k_\infty \leq 1.31$ for 20°C to 100°C. (Reference 7a)

For the Oat high density racks, these storage criteria will be satisfied if the uncontrolled lattice k_∞ calculated in the normal reactor core configuration (standard cold core geometry) is less than or equal to 1.3191 (Reference 7). For the Holtec racks, the design-basis hypothetical bundle has a k_∞ of 1.3392 in standard cold core geometry. The net result is the k_∞ condition to meet the storage criteria is higher for the Holtec racks than for the Oat and the original low density racks. Thus, the Oat and low density racks are more limiting. The maximum k_∞ in the normal reactor core configuration at cold conditions for fuel assemblies in the spent fuel storage racks is 1.31, based on the Technical Specifications.

The peak uncontrolled lattice k_∞ in normal reactor core configuration is calculated by the fuel fabricator for each bundle type.

9.1.3 Fuel Pool Cooling and Cleanup System

9.1.3.1 Design Bases

The fuel pool cooling and cleanup system (FPCCS) is designed to remove the decay heat produced by stored spent fuel assemblies during all anticipated conditions of plant operation and during plant refueling outages 18 days after reactor shutdown. This includes refueling using either a full-core offload or core shuffle method. The system consists of two identical trains, which include pumps, heat exchangers, and filter-demineralizers.

The heat load in the spent fuel pool is anticipated to increase subsequent to each progressive refuel outage until the pool is filled to the maximum capacity where all of the storage locations are filled. The spent fuel pool temperature is maintained at or below 125°F during normal operation and 150°F during refuel outages. In anticipation of installing additional storage locations in the pool, the spent fuel pool and the FPCCS have been evaluated for a normal operating temperature of 150°F. However, additional engineering evaluation is

required if the spent fuel pool is to be maintained at a temperature greater than 125°F during normal operating conditions.

The design criteria for the FPCCS are as follows:

a. The spent fuel pool storage capacity is, nominally, 3.0 cores (2383 assemblies), plus room for removing a full core. Additional racks are required to be installed in the pool to accommodate additional spent fuel.

b. Both one-quarter core 12-month and one-third core 18-month fuel discharges were originally considered for Fermi 2 refueling cycles; plant operation began at Cycle 1 with one-third core 18-month fuel discharge cycles. The spent fuel pool and the FPCCS have been analyzed assuming a maximum spent fuel population composed of the following:

    1. 6.25 cores composed of 20 groups of spent fuel assemblies, each of the first 19 groups containing from 176 to 228 assemblies discharged approximately every 18 months, ending with a full core discharge 30 years after the first cycle.

c. The spent fuel assemblies have a power history giving the discharge batch an average irradiation less than or equal to 50,000 MWd/MTU

d. The system heat load removal capacity is based on heat exchangers sized for a design heat load of 16.66 x 10^6 Btu/hr with a 55°F hot-to-cold side inlet temperature differential and two trains operating. The system is managed to maintain the spent fuel pool bulk temperature at or below 125°F during normal plant power operation with up to 3.0 cores of spent fuel stored in the spent fuel pool.

e. The decay heat was calculated for 4780 assemblies, exceeding the actual storage locations, based on BTP ASB 9-2. The following assumptions are made to calculate decay heat load to the spent fuel pool.

    1. Each discharged assembly has been irradiated for 5.2 years

    2. During the irradiation period, the reactor is operating at 100 percent power

    3. After shutdown, the RHR cooling system is used as the primary decay heat removal system for up to 18 days while the reactor head is off and refueling/maintenance operations are proceeding, including full-core offload when scheduled

    4. In applying BTP ASB 9-2, the uncertainty factor K for irradiation time $t > 10^7$ sec is taken to be 0.1

f. Refer to Table 3.2-1 for seismic and quality group for FPCCS system.

g. The FPCCS is designed to achieve the following additional functions.

1. Minimize corrosion product buildup and control water clarity, through filtration and demineralization so that the fuel assemblies can be efficiently handled underwater
2. Minimize fission product concentration in the water that could be released from the spent fuel pool to the reactor building environment
3. Monitor spent fuel pool water level and maintain a water level above the fuel sufficient to provide shielding for normal building occupancy
4. Maintain the bulk water temperature at less than 150°F, with the heat loading resulting from the removal of a full core either during plant refueling (greater than 18 days cooling) or in a plant outage following a normal refueling. This may be achieved by being able to interconnect the RHR system and the FPCCS to supplement spent fuel pool cooling.
5. Preclude siphoning the spent fuel pool by providing siphoning breakers on all lines penetrating the spent fuel pool.

9.1.3.2 System Description

The FPCCS cools the spent fuel pool by transferring decay heat through heat exchangers to the reactor building closed cooling water system (RBCCWS), as shown in Figure 9.1-23. Water purity and clarity in the spent fuel pool, reactor well pool, and dryer-separator storage pool are maintained by filtering and demineralizing the pool water as shown in Figure 9.1-24.

The FPCCS is composed of two trains. Each train is designed to remove 8.33×10^6 Btu/hr with a 55°F (FPCCS to RBCCW inlet flows) temperature differential, 550 gpm tube flow and 800 gpm (RBCCW) shell flow (Table 9.1-1). The system consists of two fuel pool cooling pumps; two heat exchangers; two filter-demineralizers; two skimmer surge tanks; and associated piping, valves, and instrumentation. The two fuel pool cooling pumps are connected in parallel, as are the two heat exchangers. The pumps and heat exchangers are located in the reactor building below the level of the bottom of the spent fuel pool.

The filter-demineralizer units are located in the radwaste building in separate shielded cells, with enough clearance to permit removing filter elements from the vessels. Each cell contains only the filter-demineralizers and piping. All air-operated valves (such as inlet, outlet, recycle, vent, and drain) are located on the outside of one shielding wall of the cell, together with necessary piping and headers, instrument elements, and controls. Penetrations through shielding walls are located so as not to compromise radiation shielding requirements (Subsection 12.1.2).

The pumps circulate the spent fuel pool water in a closed loop, taking suction from the skimmer surge tanks through the heat exchangers, circulating the water through the filter-demineralizers, and discharging nominally 7' -6" below the normal water level in the fuel storage pool. The cooled water traverses the pool, picking up heat and debris before starting a new cycle by discharging over the skimmer weirs and scuppers into the skimmer surge tanks. The normal makeup water source for the system is provided from the condensate storage tank to the skimmer surge tanks.

Backup cooling is provided to the spent fuel pool by means of a permanently piped cross tie to the RHR system. In this mode of operation, one RHR pump and the corresponding RHR division heat exchanger will provide the means to cool the spent fuel pool. This cooling circuit is established by opening cross-tie valves V8-3264, G4100F016 and V8-3029, G4100F036 and closing FPCCS valves V8-3006, G4153F004 and V8-3253, G4100F011 (Figure 9.1-23). For the designed piping configuration, the RHR pump flow is throttled with valve G4100F231 to a maximum of 3500 gpm. If the fuel pool gates are removed and the reactor cavity is flooded up, the RHR discharge may be configured to split the flow between the reactor cavity and fuel pool by opening G4100F036. The RHR suction will be from the operating SDC loop, therefore G4100F016 is closed. During this split flow configuration, the FPCC is operated in parallel with RHR SDC as FPCC is discharging to the reactor cavity, G4153F004 and G4100F039 are open and G4100F011 is closed. An RHR heat exchanger will remove the total spent fuel pool decay heat load of approximately 42.22 x 10^6 Btu/hr of a full-core offload, completed at 156 hrs decay cooling since reactor shutdown, with about a 49°F temperature differential (Table 9.1-1). To ensure the availability of backup cooling via the RHR system, the cross-tie piping, the FPCCS piping from the skimmer tanks to the first anchor downstream of valve V8-3006, and the FPCCS piping from the first anchor upstream of valve V8-3253 to the spent fuel pool are Seismic Category I.

Both FPCCS heat exchangers operating in parallel are designed to remove the maximum heat load produced by various combinations of spent fuel discharged from the equilibrium fuel cycle at the time the RHR system is isolated from the spent fuel pool, plus the heat being released by batches discharged at previous refueling (see Subsection 9.1.3.1). The maximum heat load 18 days after a reactor shutdown with about 4276 fuel bundles in the spent fuel pool is 14.29 x 10^6 Btu/hr. The maximum heat load 18 days after reactor shutdown with 9 1/3 core off loads in the spent fuel pool is approximately 11.9 x 10^6 Btu/hr. This load is within the capacity of the FPCCS heat exchangers with a temperature differential of about 39°F.

During refueling outages (up until mode change for plant restart), when spent fuel pool circulating flow is interrupted to drain the reactor well and the dryer/separator storage pool or when the FPCCS becomes incapacitated, either of the RHR system heat exchangers may be used to supplement spent fuel pool cooling in the event the pool bulk temperature cannot be maintained below 150°F. During refueling outages, the RHR system can provide necessary supplemental cooling of the spent fuel pool until the RHR system is isolated from the spent fuel pool to restore low-pressure coolant injection (LPCI) standby mode. After RHR is isolated, the spent fuel pool decay heat is managed to remain within the FPCCS duty capability. Table 9.1-1 also lists the characteristics of an RHR subsystem in the fuel pool cooling assist mode.

The design of the spent fuel pool is such that the top of the stored fuel is at a lower elevation than the bottom of the gate between the reactor well and spent fuel pool. There are no connections to the spent fuel pool that could drain the pool below the elevation of the bottom of the gate when the gate is removed for refueling, or below the normal spent fuel pool level when the gate is in place. To prevent water from being siphoned out of the pool, the piping entering the spent fuel pool is fitted with normally submerged vents which will break a siphon before the minimum required water coverage over the stored fuel is lost. A level indicator, mounted at the valve rack, monitors reactor well water during refueling. A high

rate of leakage through the refueling bellows assembly, drywell to reactor seal, or the spent fuel pool gates is indicated on the operating floor instrument racks.

Spent fuel pool water is continuously recirculated during normal FPCCS operation. The circulation patterns within the reactor well and spent fuel pool are established by the placement of the diffusers and skimmers to sweep particles dislodged during refueling operations away from the work area and out of the pool.

For refueling operations, the reactor well and dryer-separator storage pools are filled by transferring water from clean stored condensate. After the vessel head is removed, the fill water is transferred through the reactor vessel by flooding vessel level up into the reactor well. Clarity and purity of the pool water are maintained by a combination of filtering and ion exchange. The cleanup system has sufficient capacity to ensure pool water clarity and purity. The water purity is maintained by monitoring the demineralizer conductivity and differential pressure with periodic sampling and analysis of spent fuel pool water. The filter-demineralizers maintain water purity within the chemical limits specified below.

	Fuel Pool Chemical Limits	Demineralizer Effluent
Conductivity	≤ 3 μmho/cm at 25°C	≤ 1 μmho/cm at 25°C
Chloride	≤ 500 ppb	≤ 50 ppb
pH	5.3-7.5 at 25°C	6.0-7.5 at 25°C
Total insolubles	≤ 1 ppm	

Demineralizer differential pressure operating limit is 30 psi, and an alarm is provided at 25 psi. No radiochemical limits are needed to monitor the spent fuel pool water and initiate corrective action for the following reasons:

a. Crud buildup that would contribute to gross gamma activity is minimized by the filter-demineralizer

b. Iodine-131, with an assumed concentration of 64 μCi/g, is the most radiologically significant nuclide; doses from the other nuclides, by comparison, are relatively negligible

c. The assumed concentration of 64 μCi/g of ^{131}I is almost 10^{-6} of the specific activity if its solubility limit is attained. Therefore, assuming a partition factor of 10 and a removal efficiency of 99 percent by the SGTS, ^{131}I would not be a problem.

The system flow rate is larger than that required for two complete water changes per day of the spent fuel pool, or one change per day of the fuel storage, reactor well, and dryer-separator pools.

The maximum system flow rate is twice the flow rate needed to maintain the specified water quality. Particulate matter is removed by powdered ion-exchange resin-fiber mixtures. Alternatively, a combination of powdered resin and precoated material such as cellulose may be used as the disposable filter medium. The filter elements are stainless steel mesh elements

mounted vertically in a tube sheet and replaceable as a unit. The filter vessel is constructed of carbon steel and coated with a phenolic resin material.

Spent fuel pool water and demineralizer effluent are sampled and analyzed once per week.

Instrument readings for conductivity and differential pressure are taken once per shift. Alarms sound in the control room if demineralizer conductivity, flow, or differential pressure limits are attained so that corrective action may be initiated. Backwashing and precoating operations are controlled from a local panel in the radwaste building. The spent filter medium is removed from the elements by backwashing with air and condensate and then is flushed to the phase separator tank.

A poststrainer in the effluent stream of the filter-demineralizer limits the migration of filter material. The filter-holding element can withstand a differential pressure greater than the developed pump head for the system.

System instrumentation is provided for both automatic and remote manual operations. A low-low level switch stops the circulating pumps when surge tank reserve capacity is low. Manual control for the circulating pumps is either from local panels or the control room panel. Pump low suction pressure automatically turns off the pumps.

The FPCCS has alarm functions for cooling pump low discharge pressure, refueling bellows seal leakage, spent fuel pool gate reactor well seal leakage, skimmer surge tank high level, spent fuel pool high level, and skimmer surge tank low level. All of these functions give a common alarm signal to the main control room; for example, spent fuel pool cooling system trouble. Each function also has a light, located on local control panels, which determines the cause of the common alarm in the main control room. In addition, there are specific alarms in the control room for spent fuel pool high temperature, spent fuel pool low level, and spent fuel pool demineralizer trouble.

The local control panels receive power from a standby source if normal power is not available. Circulating pump motor loads are considered nonessential loads and will be operated as required under accident conditions.

9.1.3.3 Safety Evaluation

The FPCCS maintains the peak spent fuel pool bulk temperature below 150°F with the maximum design decay heat load during an outage and at or below 125°F during normal plant power operation. Although the spent fuel pool and the FPCCS are evaluated for a normal operating temperature of 150°F, additional engineering evaluation is required if the spent fuel pool is to be maintained at a temperature greater than 125°F during normal operating conditions. The FPCCS and RBCCW pumps are powered from redundant buses; this ensures continued normal cooling operation. The RHR system provides a safety source of emergency makeup water and redundant heat removal capability.

No inlets, outlets, or drains are provided that would permit the spent fuel pool to be drained below a safe shielding level. Lines extending below this level are equipped with siphon breakers, check valves, or other suitable devices to prevent inadvertent pool drainage. The line draining the space between the two gates is sufficiently high to preclude draining excessive water above the spent fuel storage racks.

Except during refueling operations, the spent fuel pool will be isolated from the reactor head cavity and dryer-separator storage pool by two redundant watertight gates that close the opening through which spent and new fuel is transported to and from the spent fuel pool. The bottom of the gate opening is above the top of the fuel storage racks in the bottom of the spent fuel pool to ensure that the stored fuel can never be uncovered.

The only interconnection between the cooling and cleanup subsystems is the spent fuel pool itself. The FPCCS return lines to the spent fuel pool are provided with siphon breakers.

The decay heat load in the spent fuel pool may vary widely because of various possible combinations of the following:

a. The number of groups and the respective irradiation periods of spent fuel assemblies in the racks (see Subsection 9.1.3.1.b. and e.)

b. The duration of time-after-shutdown for each of the spent fuel groups.

Decay heat has been calculated for a bounding emergency full core offload in Reference 7c for 4780 assemblies using the assumptions of Subsection 9.1.3.1 resulting in a greatest heat release to the spent fuel pool of 42.65 x 10^6 Btu/hr (12.5 MWt). The FPCCS does not have the heat removal capacity to maintain the pool temperature to less than 125°F. Engineering evaluation is required to operate with this heat load in the spent fuel pool.

The FPCCS is normally capable of maintaining the spent fuel pool temperature at or below its maximum normal design temperature of 125°F. Under full-core offload with spent fuel pool decay heat load above the system design capacity, the required differential temperature across the FPCCS heat exchangers exceeds the nominal design temperature differential. In the event of other system abnormal conditions of decay heat load higher than available FPCCS removal capacity due to other refueling outage activities and/or FPCCS capacity restriction (e.g., due to maintenance), the heat load may also result in higher temperature differential across the operating FPCCS heat exchanger(s). This would cause the temperature of the spent fuel pool to rise. Should FPCCS not be able to maintain spent fuel pool temperature below 150°F, then an RHR loop would be aligned to take suction from, and discharge to, the spent fuel pool. If the fuel pool gates are removed and the reactor cavity is flooded up, then the suction remains from the shutdown cooling line and the discharge may be split between the recirculation loop and the fuel pool. The use of the RHR system in the fuel pool cooling assist mode makes both low pressure coolant injection (LPCI) subsystems inoperable.

FPCCS and natural circulation have been analyzed to be capable of serving as an alternate method of decay heat removal to enable RHR Shutdown Cooling to be taken out of service for maintenance during refueling. When operating in this alternate shutdown cooling mode, the fuel pool gates are removed and the RPV cavity is flooded. Entry into this mode requires satisfying the refuel technical specification associated with high RPV water level. FPCCS is normally operated with two pumps and two heat exchangers in service. In this capacity, FPCCS and natural circulation maintain FPCCS suction temperature less than 140°F, cooling both the old and freshly off-loaded assemblies in the fuel pool as well as those remaining in the RPV. RWCU may also be placed in operation with the regenerative heat exchanger bypassed to provide additional cooling and in-vessel mixing. This ability to enter this mode of FPCCS operation for RHR maintenance activities is evaluated on a per cycle basis using

the expected vessel and spent fuel pool heat loads. The activity is managed such that normal shutdown cooling can be restored within 8 hrs. This is an arbitrary time frame that conservatively assures cooling can be restored prior to the onset of pool and core boiling. In addition, the operation of this mode restricts the operation of temporary auxilliary pool water filtration units such that the flow discharge does not interfere with the core exit flow and thereby impede natural circulation cooling.

The heat load to the spent fuel pool is caused by the decay heat of the fission products and the activated heavy elements (^{239}U and ^{239}Np) contained in the spent fuel assemblies stored in the racks, including those temporarily stored for a refueling outage. Since different combinations of assemblies are reinserted into the core than were originally removed (possibly including some reinserted assemblies that were removed at prior refuelings), the decay heat load will vary from cycle to cycle. For purposes of the design analysis, a conservatively higher number of assemblies are assumed in the spent fuel pool storage racks than there are actual storage locations as presented in Subsection 9.1.2.2.2 above. Table 9.1-2 presents the fractional decay heat as a function of time after shutdown for this bounding case determined with the method given in BTP ASB 9-2.

The data in the table for the prior discharges are based on a full-power operating period of 5.2 years that is conservatively consistent with an approximately one-third core, 18-month equilibrium fuel cycle. The number of fuel assemblies and the decay heat contribution for each discharge are also given in Table 9.1-2. The decay heats in MW for fuel assemblies discharged in normal refuelings for the entire plant cycle, ending with a normal partial core unload of 260 assemblies, are presented in Table 9.1-3 and are computed as follows:

$$\text{QDKP}\ (t_s) = \text{RTP MWt} \times \frac{P}{P_o}(t_o, t_2) \times \frac{N}{764}$$

where

QDKP (t_s)	=	decay heat of fuel assemblies that have been stored in spent fuel pool racks for t_s sec, MW_t
RTP MWt	=	100 percent of the rated thermal output of core
$\frac{P}{P_o}(t_o, t_2)$	=	fractional decay heat
$\frac{N}{764}$	=	fraction of full core discharged per Refueling/Unload

Table 9.1-3 gives the cumulative spent fuel pool heat load and quantity of spent fuel stored in the racks versus time after the initial discharge to the spent fuel pool. To develop a conservative maximum spent fuel pool heat load, the case of a complete operating cycle followed by an emergency full core offload is considered. The total number of assemblies in the spent fuel pool is taken as 4780. This maximum heat load case includes 19 approximately one-third core discharges from previous refuelings plus a full-core offload of the last operating cycle started at 2-1/2 days decay cooling from reactor shutdown. The maximum spent fuel pool heat load in this case would be 12.50 Mwt (~42.65 x 10^6 Btu/hr; see Table 9.1-1). In this case, one loop of the RHR system would be needed in the fuel pool cooling assist mode to maintain bulk temperature below 150°F.

Upon completion of refueling activities, FPCCS is evaluated to determine that it is capable of maintaining the spent fuel pool temperature below 125°F. Insufficient decay heat removal capability may occur due to FPCCS performance degradation, capacity limitation due to insufficient temperature differential from high RBCCW service water temperature, larger than normal core discharge, and/or less than design cooling time from reactor shutdown. If this situation should exist, then one loop of RHR would continue to be employed to control the spent fuel pool temperature until the FPCCS decay heat removal capacity is sufficient to allow plant restart (RHR LPCI is considered inoperable and Technical Specifications limits with RHR LPCI inoperable apply to plant operations).

Should two active components in the spent fuel pool cooling system be unavailable and the RHR system be unavailable for cooling, the spent fuel pool water temperature would be inherently limited to 212°F, on boiling.

An analysis has been performed to determine the radiological doses at the site boundary which might result as a consequence of a complete loss of cooling of the spent fuel pool (see Reference 15). Such a complete loss of pool cooling is not considered a Design Basis Accident, and it is not considered a design basis for the fuel pool or for the FPCCS. The subject radiological analysis is currently included in the UFSAR for information only, and is not part of the basis for NRC acceptance of the FPCCS design. Details of the analysis are as follows:

a. Heat released from the spent fuel is conservatively assumed to have a constant value for a period of 30 days after the assumed loss of the FPCCS. For purposes of evaluating the radiological dose consequences only, it is conservatively assumed that no other heat removal method is available except for spent fuel pool boiling and that the time to achieve pool boiling is zero. Makeup water is assumed to be provided at a rate equal to that of boiling and thus maintains the spent fuel pool water volume at a constant value of approximately 48,000 ft^3. Potential makeup sources are the RHR service water, condensate storage, and fire protection system.

b. There is 1.3×10^{-2} μCi/g of ^{131}I in the reactor water during power production (see Table 11.1-3). The temperature, pressure, and flow rate of the spent fuel pool water are much lower than those of the water in the reactor under full-power conditions. Spent fuel in the spent fuel pool storage racks should not cause a water iodine concentration greater than the reactor water concentration, even assuming the spent fuel pool begins to boil. Notwithstanding the above, a ^{131}I concentration of 64 μCi/g was assumed and used. This concentration is more than 5000 times the ^{131}I concentration in the spent fuel pool water during full-power operation and adequately accounts for an "iodine spike."

c. The variation of iodine concentration in the spent fuel pool as a function of time is calculated realistically to account for decay, boiling, and the addition of makeup water. Two cases are considered: one assumes the makeup water to contain no radioactivity, and the other assumes an unlimited supply of makeup water at an initial concentration of 64 μCi/g.

d. The spent fuel pool water volume is about 48,000 ft^3. Based on a concentration of 64 μCi/g, there would be about 87,300 Ci of ^{131}I in the spent fuel pool water.

Doses were calculated from other nuclides (gas and particulate) and it was concluded that ^{131}I is the most radiologically significant radionuclide, the others by comparison being negligible.

e. Iodine in the spent fuel pool water is assumed to be released from the pool at a rate that corresponds to the initial concentration and boiling rate with the application of a partition factor of 10.

f. The use of a partition factor of 10 is justified as follows: the solubility limit for iodine in hot water is 0.078 g/100 ml of water. If this amount of ^{131}I were to be dissolved in water, the specific activity would be 9.9×10^7 μCi/ml. The iodine concentration used in this analysis was 64 μCi/g. However, assuming the spent fuel pool water contained an amount of stable iodine equal to the radioiodine that would exist in the water during this postulated accident, the spent fuel pool water would still be able to accept almost one million times more iodine before the solubility limit was reached. On the basis of solubility, therefore, a partition factor of 10 is justified.

g. Iodine removal efficiency credit for the SGTS is assumed to be 99 percent, consistent with Regulatory Guide 1.52. The SGTS is designed to accommodate an inlet relative humidity of 100 percent; that is, the secondary containment is conservatively assumed to be filled with saturated air.

h. The meteorological condition assumed for the accident is the fifth percentile short-term (accident) χ/Q's for actual site meteorological data provided from Edison's 60-m tower and as reported to, and accepted by, the NRC staff (NRC letter dated April 26, 1976, G. W. Knighton to H. Tauber, Reference 8). These data are presented in Table 2.3-27.

i. The calculations estimate the 2-hr thyroid (inhalation) dose at the site boundary to be 0.17 rem for both radioactive and nonradioactive makeup water. The 30-day thyroid (inhalation) dose at the low-population zone for radioactive makeup is 0.186 rem; whereas for nonradioactive makeup, the 30-day dose is 0.134 rem.

Results indicate that the dose from this postulated accident would not exceed a fraction of 10 CFR 100 limits.

Thermal-hydraulic calculations confirm that the peak clad temperature for the hottest assembly, offloaded after 2-1/2 days of decay heat cooling from reactor shutdown, will remain below the local saturation temperature assuming a bundle inlet temperature at the maximum spent fuel pool temperature of 150°F. In addition, the calculations confirm that at the time of maximum spent fuel pool decay heat loading, with surface temperature approaching boiling (bulk temperature approximately 200°F), the hottest assembly peak clad temperature would still not exceed local saturation temperature. Should the spent fuel pool water temperature increase to the surface boiling point, the peak fuel cladding temperature would be slightly higher than the local saturation temperature ($T_{sat[racks]} \approx 240°F$). This fuel cladding temperature is a fraction of the fuel cladding temperature during normal plant operation. The physical characteristics of the fuel and the integrity of the fuel cladding would not experience changes that could cause an activity concentration in the spent fuel

pool water in excess of the activity in the reactor water during full-power operation. Dose calculations performed by Edison, based on the above, indicate that the design criteria applied to the spent fuel pool cooling system are adequate to provide reasonable assurance that the plant can be operated without undue risk to the health and safety of the public, consistent with the requirements of Criterion 2 of Appendix A to 10 CFR 50.

In summary, the spent fuel pool cooling system's design, siphon-breaking piping arrangement, redundant transfer gates, emergency makeup water supply from the RHR service water system, and RHR backup capability provide a completely reliable system for the storage and cooling of spent fuel.

9.1.3.4 Testing and Inspection

Prior to power operation following a refueling outage, a determination will be made that the heat generation rate in the spent fuel pool is within the current capacity of the FPCCS with both trains in normal operation at a spent fuel pool bulk temperature less than or equal to 125°F.

No special tests are required for instrumentation on the FPCCS. The instrumentation will be subjected to routine testing. The FPCCS Preoperational Test program is discussed in Chapter 14.

9.1.4 Fuel Handling System

9.1.4.1 Design Bases

The fuel handling system provides a safe and effective means for transporting and handling fuel from the time it reaches the plant until the time it leaves the plant after postirradiation cooling.

9.1.4.2 Equipment Description

Table 9.1-5 is a listing of tools and servicing equipment supplied with the nuclear system. The following paragraphs briefly describe the use of some of the major tools, servicing equipment, spent fuel shipping cask, and reactor building crane. Where applicable, safety aspects of the design are discussed. For a historical discussion of the reactor building crane and spent fuel cask-handling details, see Reference 9. The procedure for load testing at 125 percent rated load described in Section 2.3.2 of Reference 9 has been modified in accordance with the guidelines established in NUREG-0554, ANSI B30.2, and NRC BTP ASB 9-1.

9.1.4.2.1 Spent Fuel Shipping Cask

Spent Fuel Shipping Cask Description

Edison does not now contemplate owning its own spent fuel shipping cask, but intends to use a licensed cask from an authorized approved vendor.

Arrangements are being made for the shipment and reprocessing of spent fuel. Since two types of spent fuel shipping casks are presently being used, the equipment and the handling techniques have been developed to utilize either type. To ensure the adequacy of the

equipment and techniques, the reactor building is designed to accept the larger spent fuel cask weighing not more than 125 tons.

The spent fuel shipping cask has a cylindrical configuration with a maximum diameter of 8 ft and a maximum length of 21 ft. Two pairs of lifting lugs are furnished to provide redundancy to the lifting mechanism. All four lugs are used simultaneously for lifting the cask. The cask is designed to conform to 10 CFR 71 with regard to structural design; radiological releases, effects, and protection; allowable spent fuel shipping conditions; shielding; and continuity of decay heat removal capacity for all credible cask accident events.

Spent Fuel Shipping Cask Handling

The fuel cask is delivered through the airlock into the reactor building by truck. The truck is positioned under the equipment access hatch. The cask is upended from its horizontal shipping position by the reactor building crane main hoist.

After upending is completed, the cask is attached to the redundant hook system via either a lifting device or two sets of slings (Figure 9.1-26) that engage the cask on all four lifting lugs and to the redundant hook system. The method of providing a redundant link between the hooks and the cask will be based upon the type of cask used.

The cask is then hoisted from the first floor grade elevation to the fifth floor operating level and traversed to the cask-washdown area, where the cask head is removed and the cask is prepared for fuel storage pool entry. Depending on the cask head removal details, the cask may not have to be disengaged from the crane during this handling step.

The cask enters the pool by traversing the crane from the washdown bay due north until it is in line with the pool storage area for the cask. This line will be marked on the operating floor surface, as shown in Figure 9.1-27. The trolley will then be traversed due east for the cask centerline to follow the marker line until the cask is suspended over the cask storage area and completely clear of the pool edge. The crane operator receives his instructions from a signalman stationed on the operating floor level adjacent to the cask. The signalman remains in visual and voice contact with the crane operator.

Prior to being lowered into the pool, the cask is steadied. Lowering of the cask will be done at minimum speed until the cask has completely cleared the pool edge. Underwater lights will be used to illuminate the cask-setdown area. A 9-ft by 9-ft by 1-in.-thick stainless steel plate is provided in the pool bottom liner to accept the cask.

All of the above-described steps are reversed when the cask is extracted from the pool.

Spent Fuel Shipping Cask Storage Area

There is no spent fuel cask storage pit as committed to in response to PSAR Question 3.2.6. A detailed discussion regarding the elimination of the spent fuel cask storage pit is presented in Reference 3. Except when in transit or when being washed down, the spent fuel shipping cask is kept submerged in its storage area in the northwest corner of the spent fuel storage pool. It will be conveyed there from its truck by the redundant crane system. The spent fuel shipping cask storage area is described above under "Spent Fuel Shipping Cask Handling."

9.1.4.2.2 Reactor Building Crane

An overhead traveling (reactor building) crane is utilized in the Fermi 2 reactor building to handle heavy objects, including the spent fuel shipping and transfer casks. The essential design bases applicable to Fermi 2 spent fuel cask handling are:

a. To minimize, to the maximum extent practical, the probability of dropping heavy objects into the fuel storage pool resulting in damage to fuel or compromising the integrity of the pool

b. To prevent a spent fuel shipping cask drop from exceeding the design limits for the cask as set forth in 10 CFR 71

c. To minimize the probability and the effect of dropping heavy objects, including the spent fuel shipping and transfer casks, during movement through the reactor building, so that damage is prevented to structures, systems, and components important to safety.

In order to obviate the possibility and to minimize the probability, to the greatest extent practical, of occurrence of events a, b, or c above, the special crane design features and improvements that have been incorporated are the following:

a. A completely redundant hoisting system

b. An upgrading of the crane for SSE and design-basis tornado

c. Upgrading of the crane quality assurance criteria

d. Crane control redundancy

e. A crane surveillance and test program

f. Administrative control of crane movements.

Crane operations over the spent fuel storage pool when fuel assemblies are stored therein are not allowed when either of the following conditions occur:

a. less than 22 feet of water over the top of irradiated fuel assemblies seated in the spent fuel storage pool racks.

b. less than the Technical Specification required ac electrical power sources operable, when in modes 4, 5 and when handling irradiated fuel in the secondary containment.

Prior to suspending crane operations, fuel assemblies shall be placed in a safe condition.

The reactor building crane is of the single trolley top running type, carried on two main girders. The girders have a rated lifting capacity of 125 short tons and a span of 113 ft 9 in. Power is applied by twin hoist motors through two gearboxes to the two drum gear rings, located on each end of the drum. In this manner, the hoist mechanism is duplicated. In normal operation, the twin hoist trains share the load, but each is separately able to carry the rated load at allowable code stresses, thus providing adequate safety should one gear train fail. Both hoist trains are provided with electrical and electromechanical type brakes; each of the latter is capable of sustaining the load should a mechanical failure occur in a gear train. Each mechanical brake is sized for 150 percent motor torque or 300 percent for redundant

systems. (This is based on a required brake torque of 277.5 ft-lb, and a rating of each of the two 13-in. brakes of 550 ft-lb. The required brake torque, B_T, is calculated by using

$$B_T = 1.5 \times 33{,}000 \ {}^{P_{hp}}/_{2\pi N}$$

where P_{hp} is the motor horsepower of 20, and N is equal to 570 rpm.)

The electrical brakes (Magnetorque) complement the mechanical brakes. The Magnetorque brakes can limit the hoist-lowering speed to 1.6 fpm at rated load in the event of failure of both the redundant mechanical brakes. If there should also be a loss of normal power to the Magnetorque brakes, an integrated alternator generates enough power to the Magnetorque units to prevent the lowering speed from exceeding the fully rated load speed.

For the main hoist, this speed is 4.7 fpm at rated load. The reactor building floor and the floor under the equipment hatch have sufficient strength to withstand the impact of a fully rated load at this speed.

The redundant wire rope system consists of two balanced reeving systems utilizing two individual wire ropes. These two wire ropes are reeved side by side from double-scoured drum groovings at each end of a single drum through the upper and lower block sheaves and to the double-sheave-type equalizer. Breakage of one cable system would reduce the factor of safety, but since each system is reeved to both sides of the bottom block and upper block system, there is no swinging or pendulum action of the block upon failure of one system. Equalizer sheaves are used in preference to equalizer bars so that ropes may more readily adjust to differences in length without the need for physical maintenance. Each of the equalizers is hung from a main pivot mechanism which is designed to be redundant within itself.

For the Fermi 2 crane, the wire rope safety factor for each single wire rope is a minimum of 10. This is determined by dividing the design rated load (125 tons) by the number of load-carrying parts (16) and the efficiency factors (0.933) and comparing the result with the published breaking strength of 102 (nominal) tons for the 1-1/4 in. (nominal) diameter rope. The design of the dual reeving system is consistent with paragraph 3.f of BTP ASB 9-1.

In both the lower and upper blocks, the sheaves are mounted in a structural cage system having supporting plates on each side of each sheave. Thus, the load being carried by the sheave pin is shared by each of these support plates. Should a pin fail on any one particular sheave group, the adjacent sheave still maintains its integrity. This allows either reeving system to take over the entire load.

The main hook block is provided with a conventional hook, and the redundant feature is provided by two smaller hooks, each capable of sustaining 50 percent of the rated load at code stresses. The two additional hooks are individually mounted on their own pins and supported directly in the main block frame. They are intended for use only when handling the fuel cask.

To ensure against damage due to a tornado, the crane is provided with electrically operated locking bars that effectively connect the unloaded crane to the runway when it is not in use. These locking bars are capable of withstanding a tornado windforce of 410 lb/ft^2 intensity at a maximum of 90 percent of the yield strength of the crane components.

Earthquake protection is provided by restraints on the crane and trolley to prevent either from leaving its respective rails due to horizontal and/or vertical displacement. Seismic responses of the crane, based on its fundamental frequency in the vertical and two horizontal directions (perpendicular and parallel to the girder), have been calculated for the SSE and are 0.65g.

The crane is designed to accommodate SSE forces and deflections with the rated load suspended in the cask-hoisting position. Crane accelerations for the vertical SSE in the unloaded condition were also determined and found in all cases to be less than 1.0g. Seismic uplift forces are therefore not encountered.

The crane responses to the SSE, as determined above, are well below the design limits of the reactor building crane. Thus, the crane will remain within its restraints if subjected to an SSE.

Crane control can be either from the cab or by radio control. In the event that the crane cab becomes uninhabitable, control may be continued by means of the remote radio control provided. The only crane components that are actuated by the crane electrical control system and are an integral part of the mechanical load-retaining hoist system are the two shoe-type hoist holding brakes. The two electrical control components that actuate the hoist holding brakes are either the raise or lower hoist reversing contactors. If either the raise or lower hoist reversing contactor fails to open when called upon, the backup is the stop button in either the cab or in the radio control, which will interrupt the main power to the crane, causing the two independent hoist holding brakes to set and thereby stop the load.

To ensure that crane control can only be executed from one position at a time, a master control transfer switch is situated on the bridge, just outside the cab. This switch must be manually operated by the operator and thus interrupts all of the control circuits so there can be no simultaneous operation of the crane from both the radio control and the cab control.

The crane control system is protected from actuation by signals from an outside source by use of a Security Start circuit. With this feature, the control system cannot be enabled until multiple conditions have been met which are unique to each receiver and its companion transmitter. To activate the equipment under control, the specific companion transmitter must be used. With this security start feature, there is no possibility of an outside source radio transmitter interfering with this system or causing inadvertent actuation since these foreign signals could not match the security circuit's multiple enabling conditions.

The crane test and surveillance programs include both preoperational tests and periodic testing, surveillance, and inspection programs.

Preoperational tests include crane hook certification to 100 percent overload, gear train no-load running tests, and complete functional tests after final crane assembly.

Periodic testing, surveillance, and inspection programs will be performed no more than 1 year prior to any usage of the crane. However, these tests and inspections will be performed just prior to each major refueling outage. Periodic testing will be conducted not more than 1 month prior to lifting of the first cask for a spent fuel transfer. The programs include magnetic particle or liquid penetrant examination of all hook surfaces; inspection of wire ropes for wear or damage, and measurements of wire rope diameters; other periodic testing, maintenance, and surveillance conducted in accordance with Occupational Safety and Health Administration (OSHA) requirements as set forth in 29 CFR 1910.179, Paragraphs (j), (k),

(m), and (n); and periodic inspections as recommended by the crane manufacturer. Testing prior to refueling also includes a full test run of all motions of a typical fuel cask unloading and loading sequence.

The spent fuel cask-handling operation is performed under strict procedural control and under the direct supervision of the Shift Manager or his designated operator. The crane operator receives his instructions from the flagman by verbal communication. All operations that cannot be visually observed by the crane operator from his cab are transferred to radio control.

Personnel carrying out cask- and fuel-handling operations are qualified to meet the guidelines set forth in Regulatory Guide 1.8.

The reactor building crane is designed in accordance with the requirements of:

a. EOCI No. 61, Class A Service, and the structural guidelines of CMAA Specification No. 70

b. Seismic response spectra for Fermi 2

c. Material Specifications: ASTM; AISI; SAE; ASA

d. Electrical Specifications: N.E.C.; NEMA; IEEE; NBFU

e. Welding: AWS

f. Federal, State, and local codes, including OSHA.

Welding specifications used in the crane fabrication are as follows:

a. Manual shielded metal-arc welding (SMAW) in accordance with AWS D1.1 and AWS D14.1 for welding of structural steel of unlimited thickness and base metals of ASTM-A36, ASTM-A242, ASTM-A441, and ASTM-A572. The required preheat and interpass temperatures are as follows:

Plate Thickness (in.)	Minimum Preheat and Interpass Temperature (°F)		
Up to 3/4	50	50	
3/4 to 1-1/2	150	70	For FCAW
1-1/2 to 2-1/2	225	150	
Over 2-1/2	300	225	

(Reference: P&H welding procedure WP-SC of September 1972)

b. Flux cored arc welding (FCAW), same application as above

c. Submerged arc welding (SAW), same application as above, with preheat and interpass temperatures as for FCAW

d. Joint welding procedure classification tests were performed for all welding processes, including groove and fillet type welds

e. No postweld heat treatment was performed.

Girders, trolley frame, and general structures are constructed of ASTM, A-36 steel. The end ties are of ASTM, A-514 material.

9.1.4.2.3 Fuel Servicing Equipment

Two fuel-preparation machines are used to remove the channels from fuel assemblies and to reinstall the channels on fuel bundles. Additionally, the fuel preparation machines are used for fuel inspections and new fuel receipt/transfer activities. Strict administrative control on the fuel preparation machine's full-up end stop is required for personnel protection. These machines are designed to be removed from the pool for servicing.

The new-fuel transfer crane is a 1500 lb, wall-mounted, traveling-hinged boom crane which services the area (B, E, 15, 17) in Figure 9.1-3.

A new fuel uprighting stand is used to hold the steel shipping box in a vertical position while the fuel assembly is removed. A new-fuel inspection stand is used to restrain the fuel bundle in a vertical position for inspection. The inspection stand can hold two bundles. The new fuel uprighting stand and the inspection stand are approximately designated by point C,15 in Figure 9.1-3.

The general-purpose grapple is a small, hand-actuated tool used generally with the fuel. The grapple can be attached to the Reactor building auxiliary hoist and the auxiliary hoists on the refueling platforms. The general-purpose grapple or approved equivalent is used to remove new fuel from the vault, place it in the inspection stand, and transfer it to the fuel storage pool. It also can be used to shuffle fuel in the pool and to handle fuel during channeling.

A channel-handling boom with a spring-loaded takeup reel is used to assist the operator in supporting a portion of the weight after the channel is removed from the fuel assembly. The boom is set between the two fuel-preparation machines. With the channel-handling tool attached to the reel, the channel may be conveniently moved between fuel-preparation machines.

9.1.4.2.4 Servicing Aids

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

Portable underwater vacuum/filter units are provided to assist in removing crud and miscellaneous particulate matter from the pool floor or from the RPV. These units may be completely submerged for extended periods. Fuel pool tool accessories are also provided to meet servicing requirements.

9.1.4.2.5 Reactor Vessel Servicing Equipment

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

The head strongback is used for lifting the vessel head. The strongback is designed to keep the head level during lifting and transport. It is cruciform in shape with four equally spaced lifting points. The strongback is designed so that no single component failure can cause the load to drop or to swing uncontrollably. The head strongback meets the requirements of NUREG-0612. The strongback, including hook pins and turnbuckles, has been load tested to three times its rated capacity of 93 tons in accordance with ANSI N14.6-1978, Paragraph 6.3.

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

The head-holding pedestals are designed to properly support the vessel head and permit seal removal and replacement, seal surface cleaning, and inspection. The mating surface between vessel and pedestal is selected to minimize the possibility of damaging the vessel head.

The RPV ventilation equipment consists of a portable unit that is attached to the RPV head for the purpose of removing trapped radioactive gases under the head during removal. After the head nuts and washers are removed, the RPV ventilation system is attached. As the head is raised, the trapped gases are drawn from the area under the head, passed through chemical filters, and exhausted. This eliminates possible inhalation doses to personnel during RPV head removal.

9.1.4.2.6 In-Vessel Servicing Equipment

The instrument strongback is attached to the reactor building crane auxiliary hoist and is used to lift replacement in-core detectors from their shipping containers.

The auxiliary hoist on the refueling platform is used with appropriate grapples to handle control rods, flux monitors, sources, and other internals of the reactor. Interlocks on both the grapple hoist and auxiliary hoist are provided for safety purposes. The refueling interlocks are described and evaluated in Subsection 7.6.1.1.

The Reactor Cavity Work Platform is used during the In-service Inspection of the vessel and other refueling outage related activities. This platform remains on the Reactor Building Fifth Floor during normal operation, secured safely to the reactor cavity concrete shield blocks. During refueling outages the platform will be installed in the reactor cavity, supported by eight (8) legs resting freely on the refueling deck. The leak-tight work area of the platform remains partially submerged in the flooded reactor cavity. The jib crane associated with this platform can be used to handle objects weighting up to 500 pounds.

The Reactor Cavity Work Platform is considered Seismic Category II/I since it is not required to ensure the three requirements of Category I system as discussed in Section 3.2.1. This Work Platform is designed to accommodate safe-shutdown earthquake (SSE) forces and deflections. Dynamic analysis using the Fermi 2 site characteristics for the refueling floor of the reactor building verifies that the Reactor Cavity Work Platform can withstand the SSE for the Fermi 2 site and will remain supported by the eight legs resting on the refueling deck.

The lifting lugs for the Reactor Cavity Work Platform are designed so that no single component failure can cause the load to drop or to swing uncontrollably. The lifting devices meet the requirements of NUREG-0612. The lift and handling system, including hook, pins and turnbuckles, has been load tested to three times its rated capacity of approximately 28 tons, in accordance with ANSI N14.6-1978, Paragraph 6.3.

9.1.4.2.7 Refueling Equipment

The refueling platform is used as the principal means of transporting fuel assemblies between the reactor well and the fuel storage pool. The platform travels on tracks extending along each side of the reactor well and the fuel storage pool. The platform supports the refueling grapple and auxiliary hoists. The grapple is suspended from a trolley system that can traverse the width of the platform. Platform operations are controlled from an operator station on the trolley. The platform contains a position-indicating system that indicates the position of the fuel grapple over the core.

The refueling platform is designed to accommodate safe-shutdown earthquake (SSE) forces and deflections. It has been designed to withstand a 1.5g horizontal and a 0.14g vertical acceleration based on static analysis. Dynamic analysis using the Fermi 2 site characteristics for the refueling floor of the reactor building verifies that the refueling platform can withstand the SSE for the Fermi 2 site and will remain on the rails. However, the refueling platform is considered Seismic Category II/I since it is not required to ensure the three requirements of Category I system as discussed in Subsection 3.2.1.

To ensure access to the drywell for inspection and maintenance during spent fuel transfer, a refueling shield bridge is utilized. A U-shaped trough lined with a nominal 6 in. of lead is placed across the gap between the RPV flange and the inner edge of the fuel transfer canal. When in place, the refueling shield bridge provides sufficient shielding to ensure continuous access to the drywell during spent fuel transfer.

9.1.4.2.8 Storage Equipment

Specially designed fuel storage racks are provided. For a description of fuel storage racks and fuel arrangement, see Subsections 9.1.1 and 9.1.2.

If sipping indicates a fuel assembly with defects of a large enough magnitude, the defective-fuel assembly is placed in a defective-fuel storage container. The defective-fuel storage containers (containing defective fuel) are stored in the Dual Purpose cells of the fuel storage racks. These are used to isolate leakage of defective fuel while in the fuel storage pool and during shipping. A defective-fuel storage container containing a fuel bundle may be moved.

The channel is removed from the defective-fuel assembly before it is placed in the container.

9.1.4.2.9 Under Reactor Vessel Servicing Equipment

The necessary equipment to remove several control rod drives (CRDs) during a refueling outage is provided. An equipment-handling platform with a rectangular open center is provided. This platform can rotate to provide space under the vessel so that a CRD can be lowered and removed. If a control rod guide tube must be removed, the thermal sleeve within the CRD housing must be rotated to disengage the guide tube. A thermal sleeve tool that

permits installation or complete removal at the thermal sleeve is provided for this purpose. Special tools and instruments to service and test individual control rod hydraulic units are also provided.

Miscellaneous tools are provided to install and remove the neutron detectors. A drain can be opened after in-core insertion to drain any residual water. Correct seating of the in-core string is indicated when drainage ceases.

Additional tools and servicing equipment not covered in these paragraphs are listed in Table 9.1-5.

9.1.4.2.10 Dry Storage Cask Servicing Equipment

A variety of ancillary equipment is used to lift, move, and prepare the dry storage transfer cask and MPC as discussed in the HI-STORM 100 System FSAR. This includes such items as lifting devices, (e.g., lift yoke, lift links, and slings), draining, drying, and backfill equipment, and welding equipment. After a dry storage cask loading campaign is completed the ancillary equipment is either removed from the site or stored in the dry cask equipment storage building near the ISFSI.

9.1.4.3 Description of Fuel Transfer

The fuel handling system provides a safe and effective means for transporting and handling fuel from the time it reaches the plant until it leaves the plant after postirradiation cooling. The previous subsection described the equipment and methods utilized in fuel handling. The following paragraphs describe the integrated fuel transfer system.

9.1.4.3.1 Arrival of Fuel on the Fermi Site

Fuel arrives on the Fermi site by truck. The fuel elements are shipped in steel boxes that support the fuel element along its entire length. The stainless steel box is contained in a stainless steel overpack. Cushioning material and a support frame positions the stainless steel box in the overpack. Two fuel assemblies are contained in each shipping container. Each shipping container is designed to ensure subcritical geometry in handling as required by 10 CFR 71.

A specific criticality safety analysis, as identified in reference 12 herein, was performed for safe storage, handling and transport of GE BWR nuclear fuel shipping containers during new fuel receipt for Fermi 2. A new generic criticality evaluation, identified as reference 16 herein, has been performed for the new stainless steel shipping container. The updated analysis provides assurance that an inadvertent criticality is highly improbable during onsite storage, handling and transportation of new fuel within shipping containers. This meets the criterion of GDC 62, "Prevention of Criticality in Fuel Storage and Handling." The former safety analysis provided is the bases for Fermi 2's exemption from the requirements of 10 CFR 70.24, as granted by the Nuclear Regulatory Commission and identified by reference numbers 13 and 14 herein. The exemption requires criticality monitoring in areas where fuel is handled outside the inner metal shipping containers (the refuel floor). In contrast, the exemption allows administrative controls, such as the use of geometrically safe configurations as bound by the aforesaid former safety analysis for areas in which the new

fuel remains in the inner metal shipping containers (the yard and the reactor building up to the refuel floor). The updated criticality evaluation provides for similar controls. Therefore, monitoring for an inadvertent criticality while handling or transporting new fuel is not required in the yard or during transport to the refuel floor.

The fuel can be handled by wearing gloves and other protective clothing. The containers are lifted to the refueling floor through the equipment hatch using the reactor building crane. Once the fuel is removed from the inner shipping containers on the refuel floor, criticality monitoring is required. Monitoring for an inadvertent criticality event on the refuel floor, is provided by two redundant detectors (D21-N115 and D21-N117). These detectors are high sensitivity gamma ray detectors (GM tubes) and are located on the east wall approximately 9 ft to 12 ft in the air. The alarm trip setting on these detectors is in the proscribed range of 5-20 mR/hr, which is adequate to detect the minimum accident of concern as described in 10 CFR 70.24 and ANSI/ANS 8.3-1986. The alarm circuitry of these detectors is arranged in a fail safe mode such that any malfunction of the detectors or a loss of power results in an alarm condition. Additionally, the detectors have a meter pegging circuit which precludes a downscale low reading (no foldover) during saturation of the GM tube due to high intensity radiation fields. Periodic performance tests are conducted to confirm instrument response to radiation and the operability of the alarm signal generator.

The aforementioned design meets the criterion of GDC 63, "Monitoring Fuel and Waste Storage." Additionally, Fermi 2 personnel are instructed to evacuate areas in which radiation or criticality alarms are activated. Evacuation of plant areas is periodically tested by the conduct of emergency response drills.

Depending on the laydown area, the metal containers can be placed in the new fuel uprighting stand using the auxiliary hoist, the new-fuel transfer crane, or a mobile crane. Any of these cranes can be used to transfer fuel from the new fuel uprighting stand to the inspection stand and to the fuel pool. Transfer of fuel from the new fuel storage vault can be done only with the auxiliary hoist. However, due to the lack of criticality detector redundancy, Fermi 2 does not strictly comply with 10 CFR 70.24 with regard to the new fuel storage vault. Accordingly, the spent fuel pool is used for storage of new fuel rather than the new fuel storage vault.

9.1.4.3.2 Refueling Procedure

Figure 9.1-28 defines, in general, the steps that make up a refueling outage. The heavy lines on the chart define the critical path in a normal outage. Deviations from this path may be encountered under normal circumstances for various reasons, such as scheduling and convenience. The reactor shall be determined to have been subcritical for at least 24 hours by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor pressure vessel.

9.1.4.3.3 Departure of Fuel From the Fermi Site

Spent fuel assemblies may be shipped off site in two different ways: 1) directly from the spent fuel pool into a shipping cask or 2) after a period of storage at the ISFSI and then in a shipping cask. For direct shipping, fuel assemblies from the spent fuel pool are conveyed by the fuel-handling bridge crane into the spent fuel cask located in the fuel storage pool. After

insertion into the spent fuel cask, the cask head is replaced, and the flooded container with fuel is raised out of the pool by the reactor building crane for transfer to the cleaning station. The cleaning station is a depression in the floor adjacent to the pool and is designed for 1000 pounds per square foot load. The cask head is bolted down, and the cask is thoroughly cleaned. Final transfer from the cleaning station down the shaft to the vehicle-loading station is by crane. The cask is laid on its side on a flatbed, one to a flatbed, for return to the fuel processing/storage facility.

Spent fuel to be stored at the ISFSI before shipment offsite is prepared for storage and moved to the ISFSI which is described in Section 9.1.2.2.3. At some time in the future, the fuel at the ISFSI will be shipped offsite. The storage overpack containing the fuel-loaded MPC will be moved to a location where the MPC can be transferred directly from the storage overpack to an NRC-licensed transportation overpack, which is designed and licensed to ship the MPC pursuant to 10 CFR 71. The MPC inside the shipping overpack is loaded on a conveyance (e.g., rail car) for direct shipment offsite. The fuel inside the MPC does not need to be removed and re-packaged in the Fermi spent fuel pool before shipment.

9.1.4.4 Control of Heavy Loads in Close Proximity to Irradiated Fuel or Safety Systems

The NRC in Reference 10 concluded that Fermi 2 meets the guidelines of NUREG-0612 for the handling of heavy loads near spent fuel. Travel paths for the handling of these loads have been graphically described, and the procedures controlling adherence to these travel paths have been identified.

The reactor building crane, Subsection 9.1.4.2.2, main hoist is single-failure proof. There are no heavy-load handling applications at Fermi 2 other than those that can be handled by the main hoist, that require handling within single-failure-proof guidelines. In order to meet NUREG-0612 guidelines, the reactor building crane auxiliary hoist has a load-limit feature that restricts the hoist from handling heavy loads (greater than 2000 lb) over the spent fuel pool and open reactor vessel.

The training and qualification of crane and hoist operators are in accordance with NUREG-0612 guidelines. The testing, inspection, and maintenance of these cranes and hoists also conform to these guidelines. Hoisting of all heavy loads around critical equipment will be covered by written procedures.

Cranes, hoists, and slings used to handle heavy loads around critical equipment are in conformance with the standards specified in NUREG-0612. The matrix analysis performed on all heavy load hoist combinations has identified all potentially affected safety system components and has defined the hazard elimination category under the NUREG-0612 guidelines for each of these components.

The special lifting devices at Fermi 2 include the reactor pressure vessel head strongback, the dryer/separator lifting device, and the spent fuel transfer cask lifting yoke. These special lifting devices, except for the lifting yoke, were found acceptable by the NRC in Reference 10. All lifts of the spent fuel transfer cask are made with a single-failure-proof lifting system to ensure the likelihood of a drop of either load is so low as to be considered not credible. A single-failure-proof lifting system consists of the crane, lifting devices (e.g., lifting yoke, lift links or brackets, slings, etc.), and interfacing lift points (e.g., cask lifting trunnions and MPC lift cleats). The design of the RB crane lifting system for lifts of a spent fuel transfer cask or

canister inside the power plant meets the guidelines for a single-failure-proof lifting system in NUREG-0612, Section 5.1.6.

Periodic testing of these special lifting devices meets the guidelines of NUREG-0612 by following ANSI N14.6-1978 and the NRC's interpretation of the NUREG-0612 guidelines provided with Reference 11. Testing and/or inspection of the RB crane lifting system components used to lift and move dry spent fuel storage cask equipment inside the power plant is performed in accordance with NUREG-0612, Section 5.1.6.

9.1 FUEL STORAGE AND HANDLING

REFERENCES

1. Holtec Report HI-992154, "Licensing Report for Spent Fuel Rack Installation at Fermi Unit 2", (DECo File Number R1-7696)

1a. Letter from Holtec to B. Cummings, Impact of Proposed Rack Height Reduction on Existing Calculations, March 17, 2006 (DECo File Number P1-16952)

2. NRC Position Paper, subject: "Fuel Pool Storage and Handling Application," April 1978 and amended January 1979.

3. Joseph Oat Corporation, Licensing Input on High Density Spent Fuel Racks for Fermi II Project, Report TM-586, Camden, New Jersey.

3a. Deleted

3b. Holtec Proprietary Report HI-961465 – WPMR Analysis User Manual for Pre&Post Processors & Solver, August, 1997.

3c. Singh, K.P. and Soler, A.I., Dynamic Coupling in a Closely Spaced Two-Body System Vibrating in Liquid Medium: The Case of Fuel Racks, 3rd International Conference on Nuclear Power Safety, Keswick, England, May 1982.

3d. Fritz, R.J., The Effects of Liquids on the Dynamic Motions of Immersed Solids, Journal of Engineering for Industry, Trans. of the ASME, February 1972, pp 167-172.

4. S. Levy and J. P. D. Wilkinson, The Component Element Method in Dynamics with Application to Earthquake and Vehicle Engineering, McGraw-Hill, New York, New York, 1976.

5. Joseph Oat Corporation, A Method for Hydro-Thermal Analysis of High Density Fuel Racks, Standard Document No. 20, Camden, New Jersey.

6. Southern Science Applications, Inc., Benchmark Calculations for Spent Fuel Storage Racks, Report SSA-127, Dunedin, Florida.

7. Global Nuclear Fuel, GE-14 Boraflex Spent Fuel Storage Rack Criticality Analysis for Fermi 2 Unit 2. (R1-7976)

7a. General Electric Company, General Electric Standard Application for Reactor Fuel, Latest Revision.

7b. Global Nuclear Fuel, GE-14 Boral Spent Fuel Storage Rack Criticality Analysis for Fermi 2 Unit 2. (R1-7687)

7c. Holtec Report HI-992207, Bulk SFP Thermal-Hydraulic Analyses For Reracking of Fermi Unit 2. (R6-422)

7d. Global Nuclear Fuel, GE-14 Defective Fuel Storage Rack Criticality Analysis for Fermi 2 Unit 2. (R1-7686)

8. Letter from G. W. Knighton, NRC, to H. Tauber, Detroit Edison, April 26, 1976.

9. Detroit Edison Co., "Summary Report, FSAR Stage Open Item, Fuel Cask Storage Pool Reactor Building, Crane Redundancy Fuel Cask Drop Accident," EF2-25622, June 25, 1974.

9.1 FUEL STORAGE AND HANDLING

REFERENCES

10. Letter from B. J. Youngblood, NRC, to Wayne Jens, Detroit Edison, subject: Issuance of Supplement No. 5 to NUREG-0798-Fermi 2, March 21, 1985.

11. Letter from B. J. Youngblood, NRC, to H. Tauber, Detroit Edison, subject: Control of Heavy Loads at Fermi 2 in Accordance with NUREG-0612, November 1, 1983.

12. Letter from General Electric (RDW:98-037) to Detroit Edison, "Detroit Edison Company Fuel Handling Criticality Assessment," dated April 9, 1998. Superseded by Reference 16.

13. Detroit Edison Letter NRC-98-0063, "Request for Exemption from 10 CFR 70.24, Criticality Accident Requirements," dated April 17, 1998.

14. NRC Letter, "Fermi 2 - Issuance of Exemption from the Requirements of 10 CFR 70.24 (TAC NO. MA1645)," dated June 2, 1998.

15. Detroit Edison Design Calculation DC-4231 Vol I, “Loss of Decay Heat Removal – High Density Fuel Storage Pool.”

16. RAJ II Safety Analysis Report, Docket No. 71-9307, Chapter 6, Criticality Evaluation.

17. NUREG-0612, “Control of Heavy Loads at Nuclear Power Plants,” U.S. NRC, July, 1980.

18. HI-STORM 100 System 10 CFR 72 Certificate of Compliance 1014, Amendment 5.

19. HI-STORM 100 System 10 CFR 72 Final Safety Analysis Report, Revision 7.

20. Detroit Edison Company Independent Spent Fuel Storage Installation 10 CFR 72.212 Evaluation Report, as updated.

TABLE 9.1-1 FUEL POOL COOLING AND CLEANUP SYSTEM

Total pool, well, and pit volume	107,000 ft^3	
Fuel storage pool net water volume[c]	42.030 ft^3	
Operating heat load	9.12×10^6 Btu/hr	
Design heat load[c]	16.66×10^6 Btu/hr	
Maximum heat load (core offload)[c]	42.65×10^6 Btu/hr	
	Fuel Pool Cooling Water Pumps	
Quantity	2	
Type	Horizontal, centrifugal	
Design flow/TDH (each)	550 gpm/300 ft	
Motor hp	60 hp	
	Fuel Pool Cooling Heat Exchangers	
Quantity	2	
Design code	ASME B&PV, Section VIII	
	Shell Side	Tube Side
Fluid circulated	RBCCW[a]	Spent fuel pool water
Sizing Temperature	95 °F	125 °F
Sizing Fluid flow	800 gpm	550 gpm
Number of passes	1	2
Material	CS, SA-106B	SS-304, SA-249
Design system pressure	150 psig	200 psig
Design system temperature	150 °F	150 °F
	Spent Fuel Pool Cooling Capacity of FPCCS	
FPCCS to RBCCW Inlet temperature differential	30 °F	55 °F
Cooling Capacity, Btu/hr:		
1 pump/1 H-X, design service rated	4.56×10^6	8.33×10^6
2 pump/2 H-X, design service rated	9.12×10^6	16.66×10^6
	Fuel Pool Filter-Demineralizers	
Type	Pressure precoat	
Quantity	2	
Design filter area	270 ft^2	
Filter capacity	550 gpm	
Maximum pressure drop	30 psi	
Design code	ASME B&PV, Section VIII	

TABLE 9.1-1 FUEL POOL COOLING AND CLEANUP SYSTEM

Holding pump flow	150 gpm
Precoat flow	>400 gpm

Spent Fuel Pool Cooling Capacity of RHR[b]

RHR to RHRSW Inlet ΔT	36 °F	49 °F	61 °F
Cooling capacity, Btu/hr			
RHR/FPC-Assist @ 3,500 gpm	30.72×10^6	42.22×10^6	52.51×10^6
RHR/SDC @ 10,000 gpm	41.6×10^6		

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[a] Maximum design temperature of RBCCW is 95°F at 85°F lake water temperature. When lake water temperature is 60°F or below, the RBCCW is controlled to 70°F.

[b] All RHR design capacity values assume 9,000 gpm RHR Service Water flow and fully fouled (service rated) heat exchanger tubes.

[c] These values assume additional storage locations are added in the spent fuel pool to be consistent with Tables 9.1-2 and 9.1-3.

TABLE 9.1-2 FRACTIONAL DECAY HEAT VERSUS TIME AFTER SHUTDOWN, 5.2 YEARS' IRRADIATION, ONE-THIRD CORE, 18-MONTH CYCLE WITH EMERGENCY CORE OFFLOAD AT 3430 MWt AND 3499 MWt

Time After Shutdown, t_s (days)	T_s (sec)	$\frac{P}{P_o}$	Number of Assemblies Discharged to Pool	Decay Heat per Discharge QDKP 3430 MWt	Decay Heat per Discharge QDKP 3499 MWt
1.08E+04	9.33E+08	6.109E-05	220	0.0603	0.0615
1.02E+04	8.81E+08	6.333E-05	228	0.0648	0.0661
9.67E+08	8.35E+08	6.563E-05	224	0.0660	0.0673
9.21E+03	7.96E+08	6.763E-05	228	0.0692	0.0706
8.27E+03	7.15E+08	7.193E-05	176	0.0568	0.0579
7.48E+03	6.46E+08	7.573E-05	220	0.0748	0.0763
6.93E+03	5.99E+08	7.853E-05	224	0.0790	0.0806
6.38E+03	5.51E+08	8.138E-05	224	0.0818	0.0834
5.40E+03	4.67E+08	8.434E-05	224	0.0849	0.0866
6.29E+03	4.57E+08	8.746E-05	224	0.0879	0.0897
4.74E+03	4.10E+08	9.063E-05	224	0.0911	0.0929
4.20E+03	3.63E+08	9.395E-05	200	0.0844	0.0861
3.65E+03	3.15E+08	9.740E-05	200	0.0875	0.0893
3.10E+03	2.68E+08	1.011E-04	200	0.0908	0.0926
2.55E+03	2.20E+08	1.053E-04	200	0.0945	0.0964
2.01E+03	1.74E+08	1.112E-04	200	0.0998	0.1018
1.46E+03	1.26E+08	1.237E-04	200	0.1111	0.1133
9.12E+02	7.88E+07	1.627E-04	200	0.1460	0.1489
3.65E+02	3.15E+07	3.423E-04	200	0.3074	0.3135
6.50E+00	5.62E+05	3.107E-03	764	10.6600	10.8732

TABLE 9.1-3 CUMULATIVE POOL HEAT LOAD AND QUANTITY OF FUEL STORED IN POOL AT END OF NORMAL REFUELING CYCLE AT 3430 MWt AND 3499 MWt

Time After Initial Discharge (years)	Decay Heat per Discharge QDKP (MWt)	Quantity of Fuel Stored After Discharge (assemblies)	Bulk Pool Heat Load After Dicharge 3430 MWt	Bulk Pool Heat Load After Discharge 3499 MWt
30.0	0.060	220	0.060	0.061
28.5	0.064	448	0.124	0.126
27.0	0.065	672	0.189	0.193
25.7	0.068	900	0.257	0.262
23.2	0.056	1076	0.313	0.319
21.0	0.074	1296	0.387	0.395
19.5	0.078	1520	0.465	0.474
18.0	0.081	1744	0.546	0.557
16.5	0.084	1968	0.630	0.643
15.0	0.087	2192	0.717	0.731
13.5	0.090	2416	0.807	0.823
12.0	0.083	2616	0.890	0.908
10.5	0.086	2816	0.977	0.997
9.0	0.090	3016	1.066	1.087
7.5	0.093	3216	1.160	1.183
6.0	0.098	3416	1.257	1.282
4.5	0.106	3616	1.363	1.390
3.0	0.129	3816	1.493	1.523
1.5	0.218	4016	1.711	1.745
0.008	1.864	4276	3.575	3.647

TABLE 9.1-4 HAS BEEN INTENTIONALLY DELETED

TABLE 9.1-5 TOOLS AND SERVICING EQUIPMENT

Fuel Servicing Equipment

Fuel-preparation machines
New fuel inspection stand
Channel bolt wrenches
Channel-handling tool

Fuel inspection fixture

General-purpose grapples

Servicing Aids

Pool tool accessories
Actuating poles
General area underwater lights
Local area underwater lights
Drop lights
Underwater camera and monitor system
Underwater vacuum/filter units
Viewing aids
Lights support brackets
In-core detector cutting tool
In-core manipulator

Reactor Pressure Vessel Servicing Equipment

RPV servicing tools
Steam line plugs
Shroud head bolt wrenches
RPV nut-handling tool
Head-holding pedestals
Head nut plus washer racks
Head stud rack
Dryer-separator sling
Head strongback
Steam line plug installation tool
RPV head ventilation equipment
Reactor Cavity Work Platform

In-Vessel Servicing Equipment

Instrument strongback
Control rod grapple
Control rod guide tube grapple
Fuel support grapple
Grid guide
Control rod latch tool
Instrument-handling tool
Orifice grapple (peripheral)
Control rod guide tube seal
In-core guide tube seals
Orifice holder (peripheral)
Blade guides

Refueling Equipment

Refueling equipment servicing tools
Refueling platform equipment
Refueling shield bridge

Storage Equipment

Spent fuel storage racks
Storage racks (control rod)
Defective-fuel storage containers

Under Reactor Pressure Vessel Servicing Equipment

CRD servicing tools
CRD hydraulic system tools
Neutron monitoring system servicing tools
CRD handling equipment
Equipment-handling platform
Thermal sleeve installation tool
In-core flange seal test plug

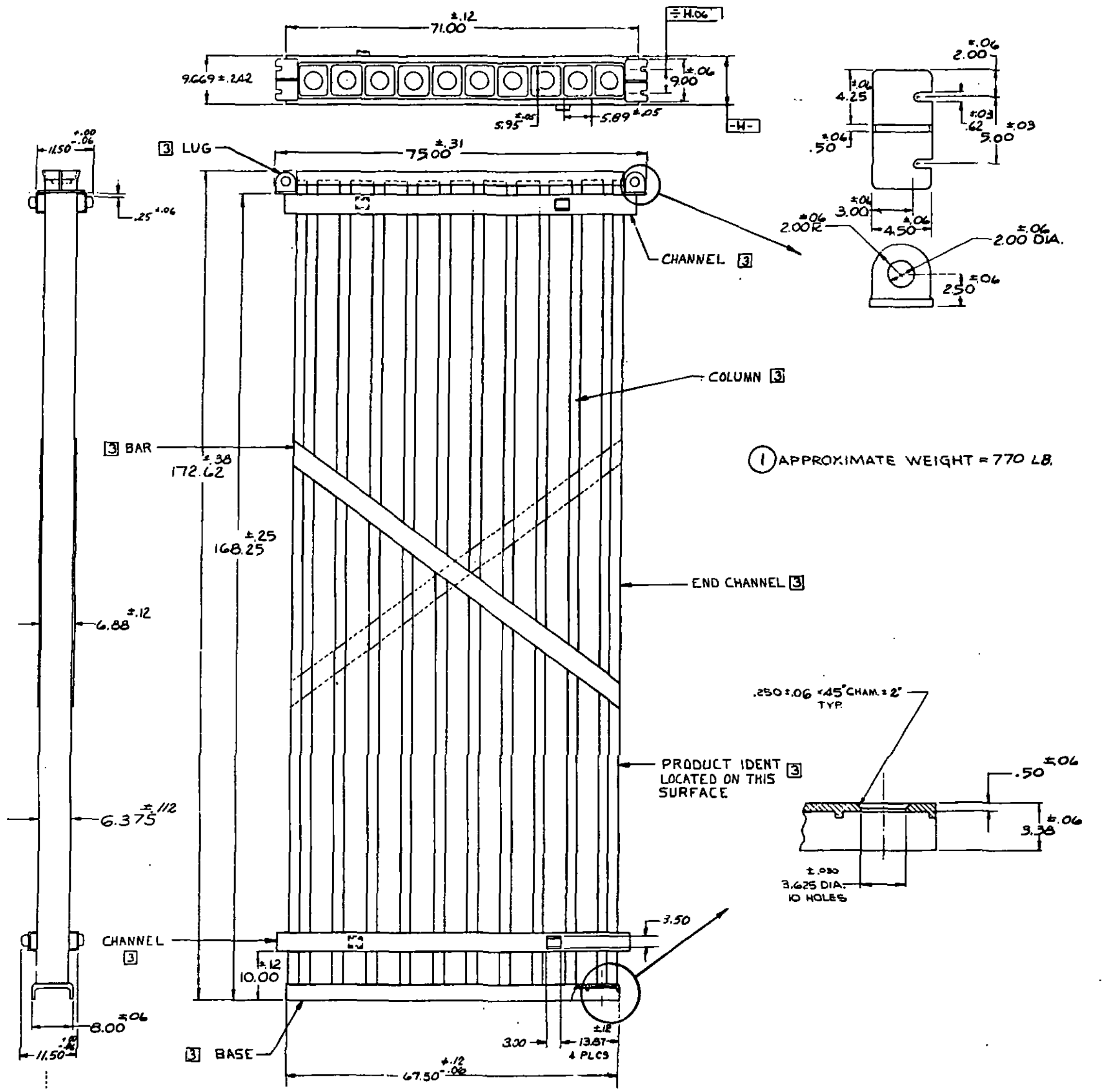

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FIGURE 9.1-1

NEW-FUEL STORAGE RACK

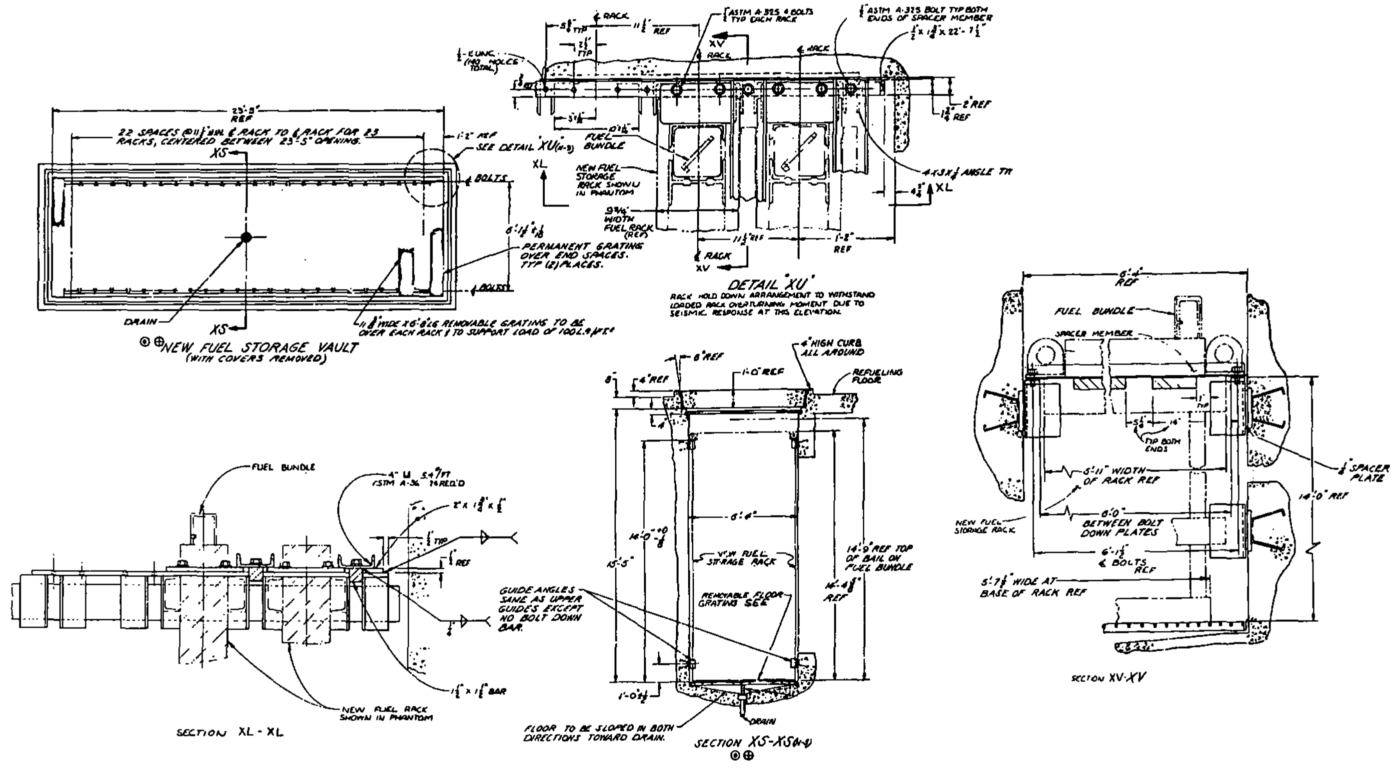

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FIGURE 9.1-2

NEW-FUEL STORAGE VAULT

Redacted in accordance with 10 CFR 2.390

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FIGURE 9.1-3, SHEET 1

REFUELING FACILITIES

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FIGURE 9.1-3, SHEET 2

REFUELING FACILITIES

Redacted in accordance with 10 CFR 2.390

NOTES:

1. FUEL POOL FLOOR LOADING SHOULD INCLUDE WEIGHT OF STORED FUEL ELEMENTS.

2. NEW FUEL STORAGE VAULT FLOOR GRATE LOADING, 1,500#/SQ. FT.

3. RAIL INSTALLATION TO BE DESIGNED FOR 10,000#/WHEEL WITH 10'-0" MIN. WHEEL SPAN.

4. APPROX. SIZE & WEIGHT OF SHIPPING CASK 7'-0" X 18'-6" X 100 TON FLOOR REINFORCEMENT SHOULD BE ADEQUATE TO SUPPORT 100 TON OR REACTOR BUILDING CRANE RATING, WHICH EVER IS GREATER.

BASED ON GENERAL ELECTRIC COMPANY DRAWING NO. R6-222

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FIGURE 9.1-3, SHEET 3

REFUELING FACILITIES

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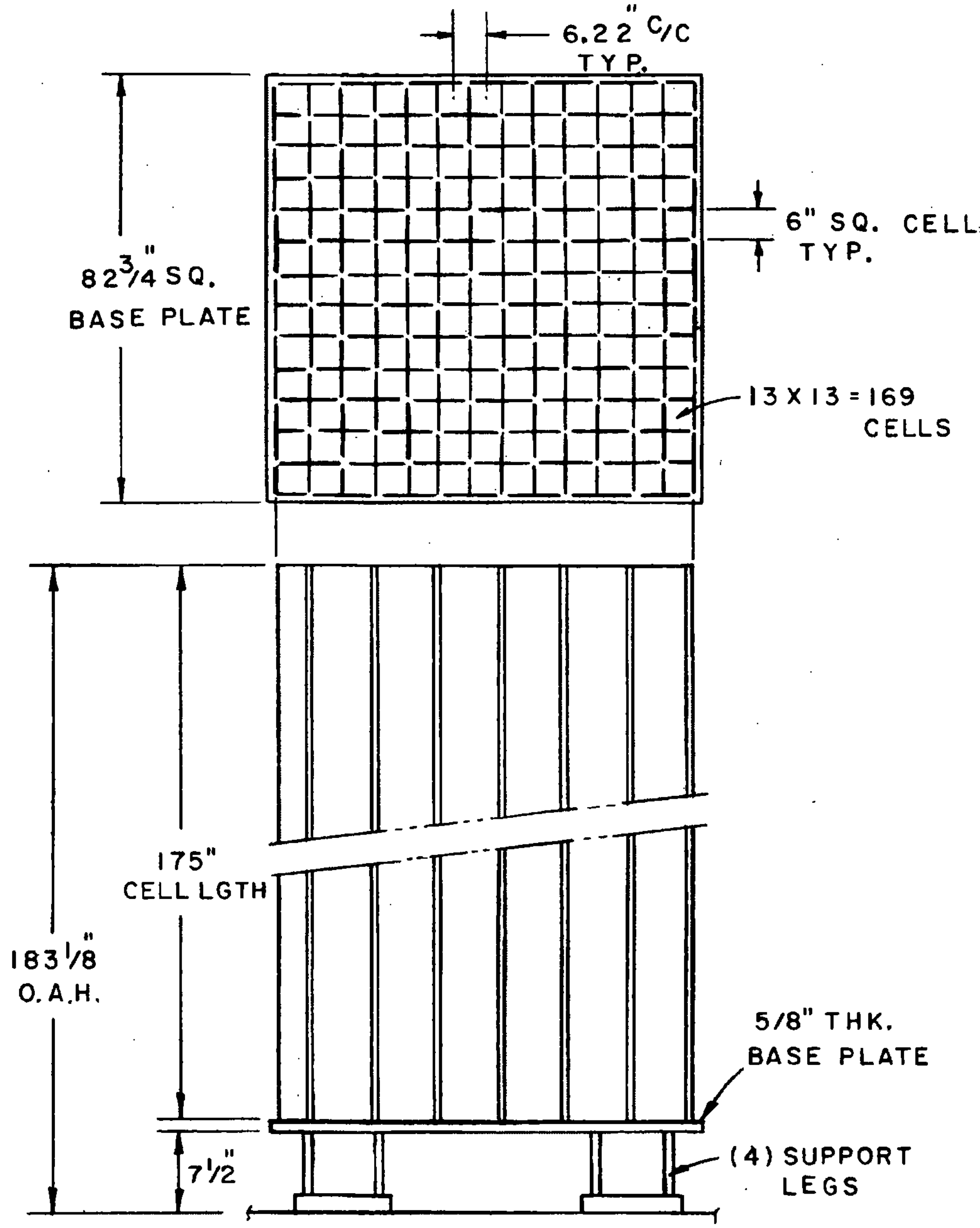

NOTE: DIMENSIONS ARE NOMINAL AND FOR INFORMATION ONLY

FIGURE 9.1-4

MODULE TYPE "A" (169 CELLS)
OAT HIGH-DENSITY SPENT FUEL RACKS

FIGURES 9.1-5 THROUGH 9.1-7
HAVE BEEN INTENTIONALLY DELETED

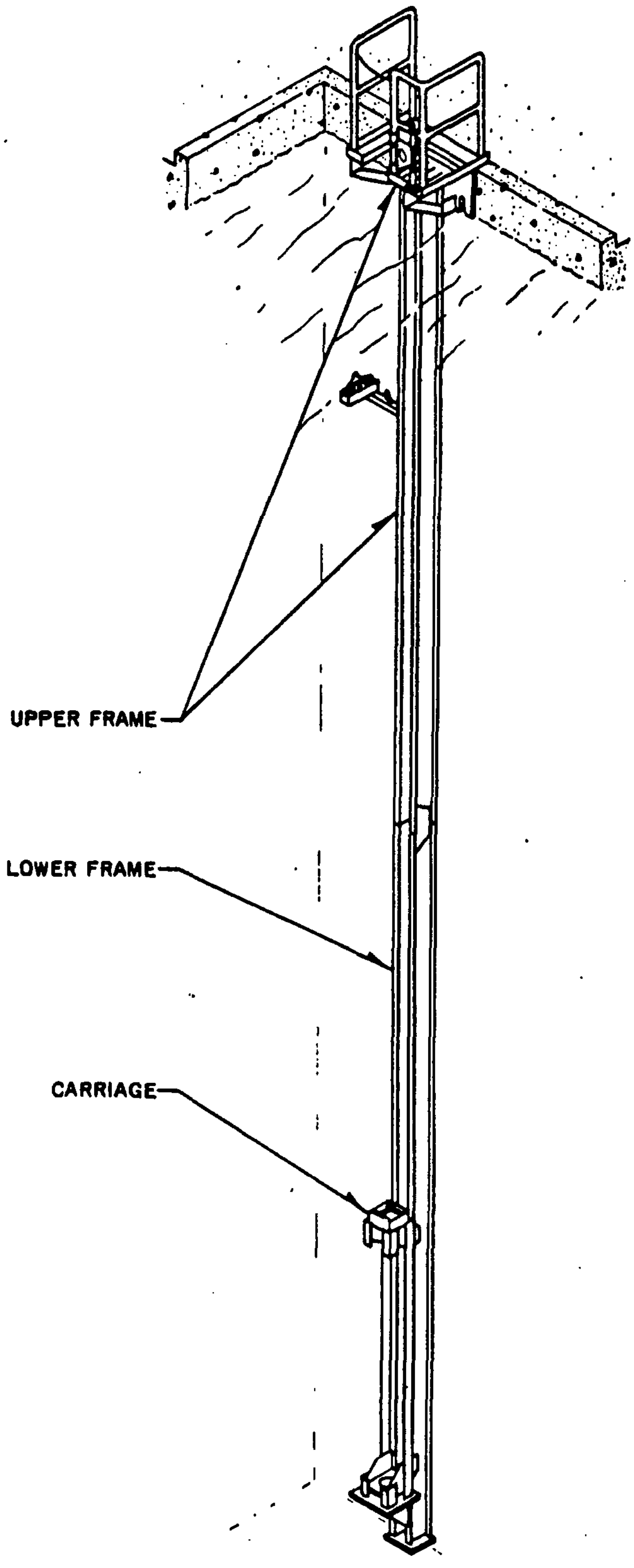

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FIGURE 9.1-8

FUEL PREPARATION MACHINE

FIGURE 9.1-9 HAS BEEN INTENTIONALLY DELETED

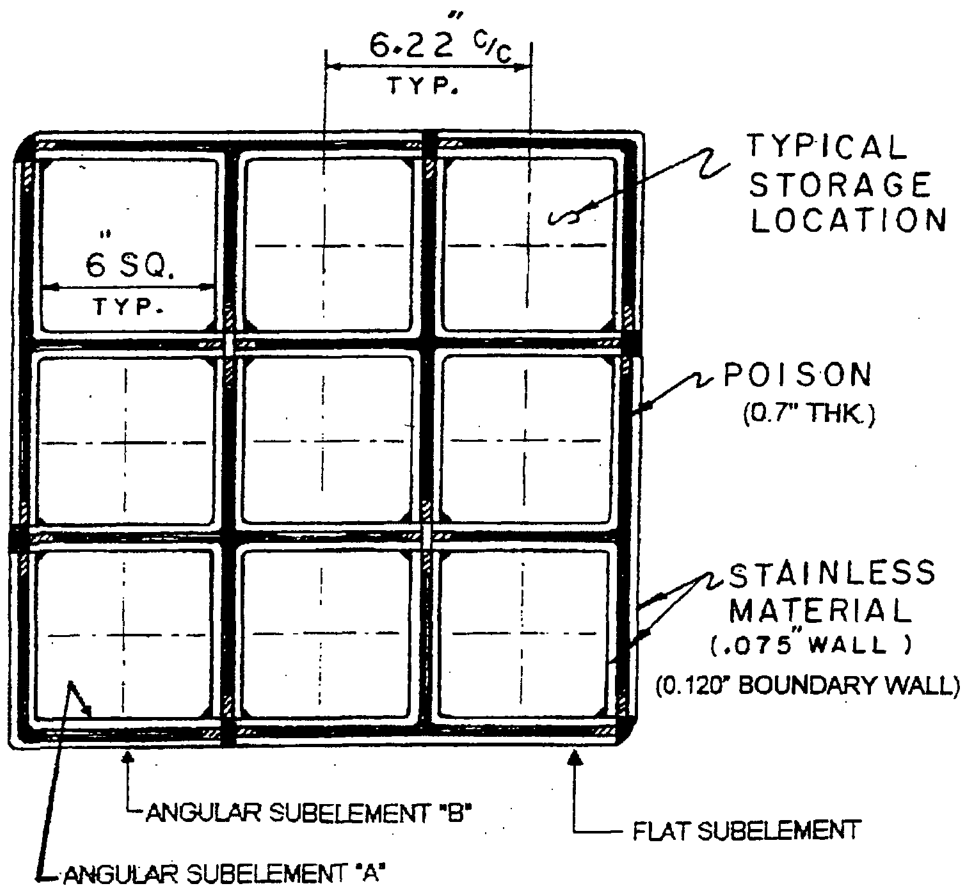

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FIGURE 9.1-10

ARRAY OF CELLS (3x3) OAT HIGH DENSITY SPENT FUEL RACKS

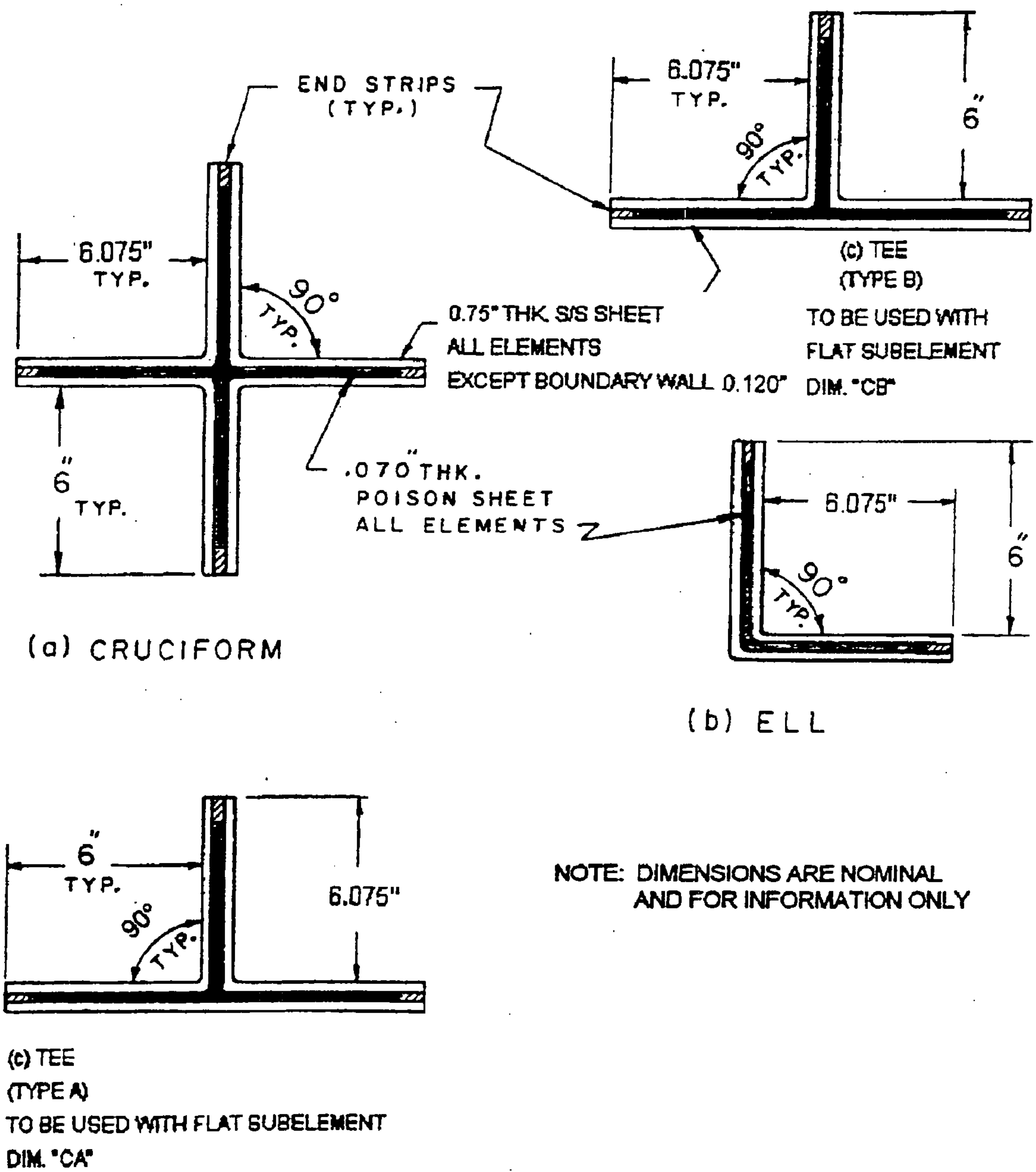

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FIGURE 9.1-11

ELEMENTS CROSS SECTION OAT HIGH DENSITY SPENT FUEL RACKS

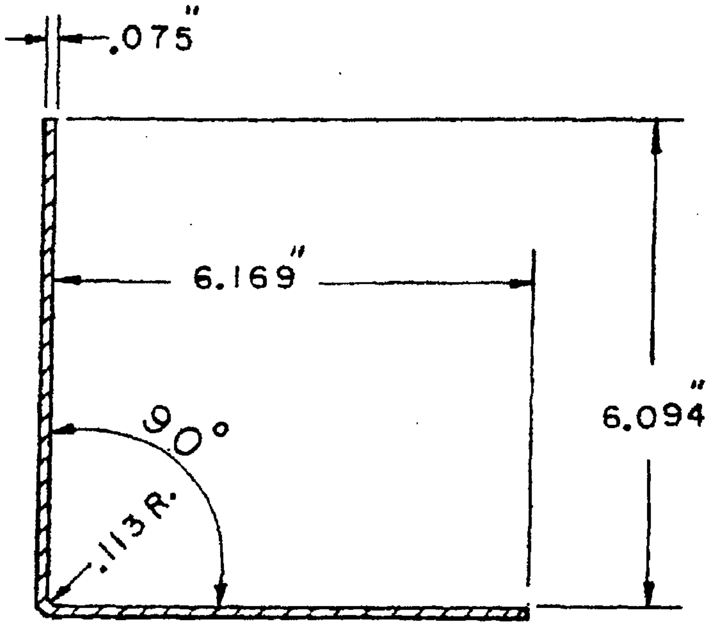

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FIGURE 9.1-12

ANGULAR SUBELEMENT "A" OAT HIGH DENSITY SPENT FUEL RACKS

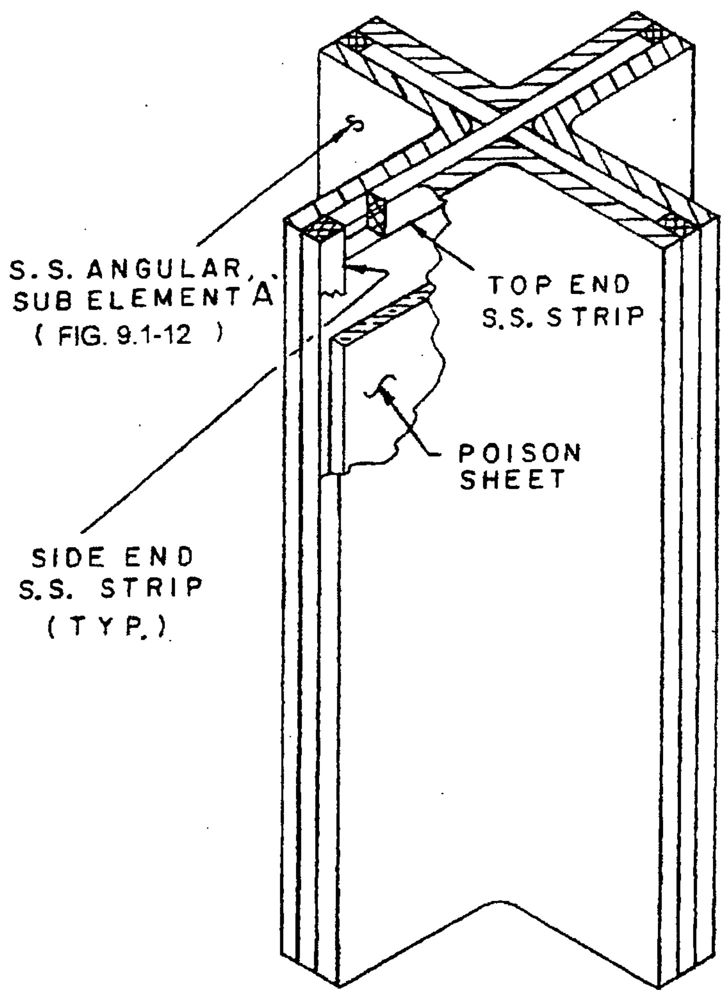

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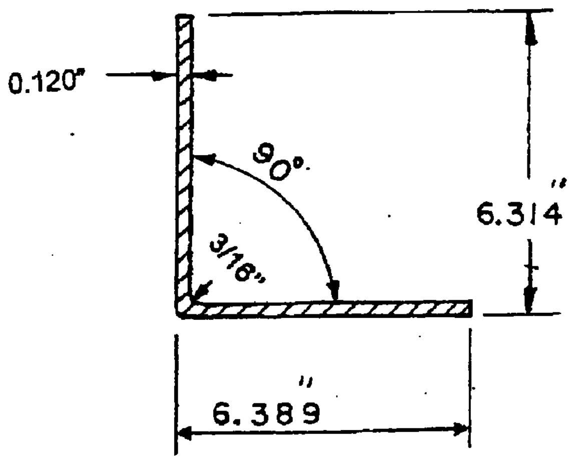
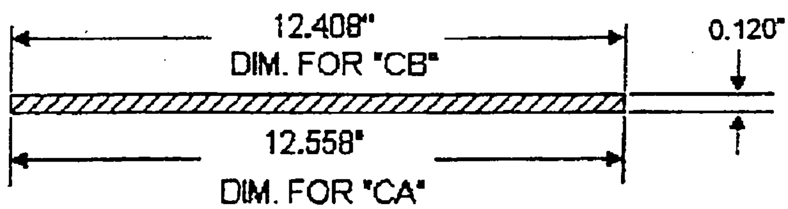
FIGURE 9.1-13

CRUCIFORM ELEMENT (ISOMETRIC VIEW) OAT HIGH-DENSITY SPENT FUEL RACKS



(a) ANGULAR SUB ELEMENT "B"



(b) FLAT SUB ELEMENT "C"

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FIGURE 9.1-14

SUBELEMENTS – OAT HIGH DENSITY SPENT FUEL RACKS

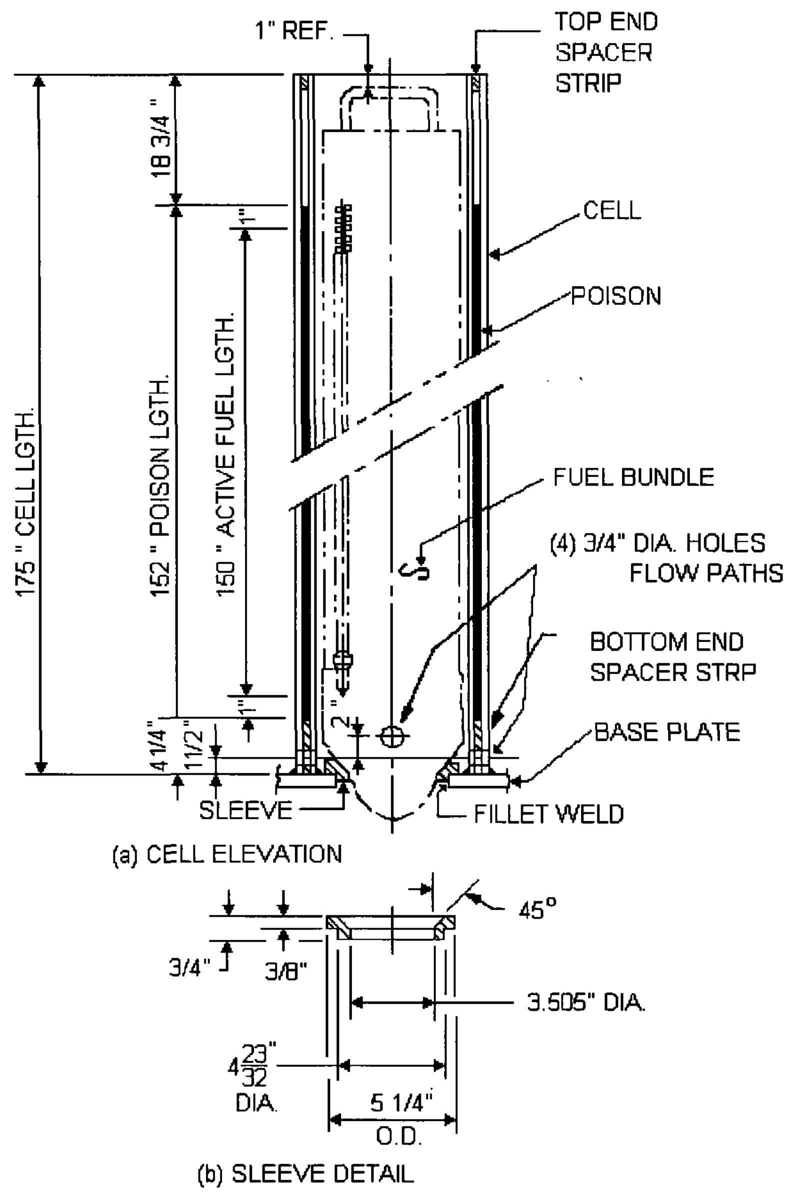

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FIGURE 9.1-15, SHEET 1

TYPICAL CELL ELEVATION – OAT HIGH DENSITY SPENT FUEL RACKS

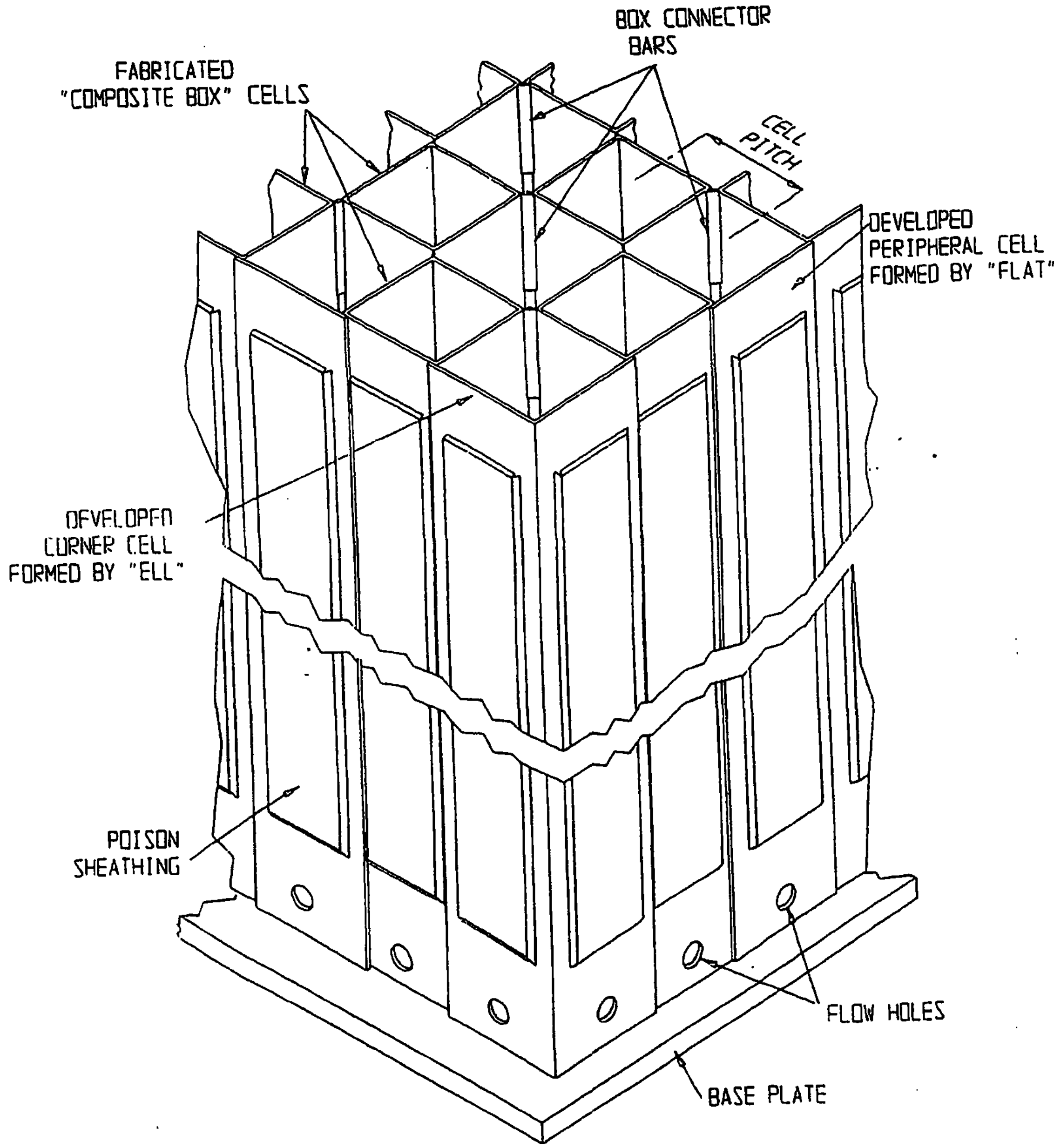

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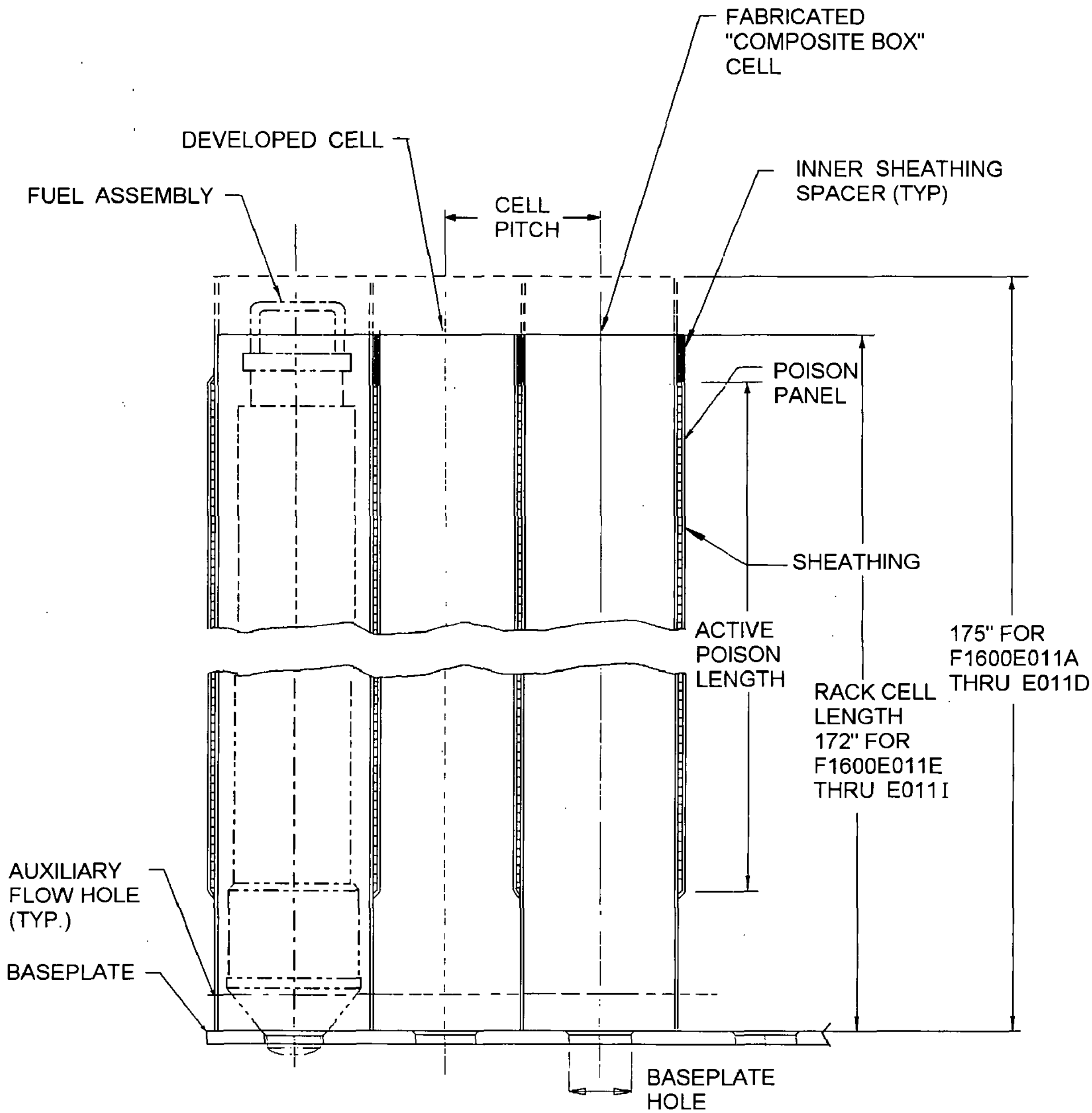
FIGURE 9.1-15, SHEET 2

TYPICAL ARRAY OF HOLTEC HIGH DENSITY STORAGE CELLS (NON-FLUX TRAP CONSTRUCTION)



Fermi 2

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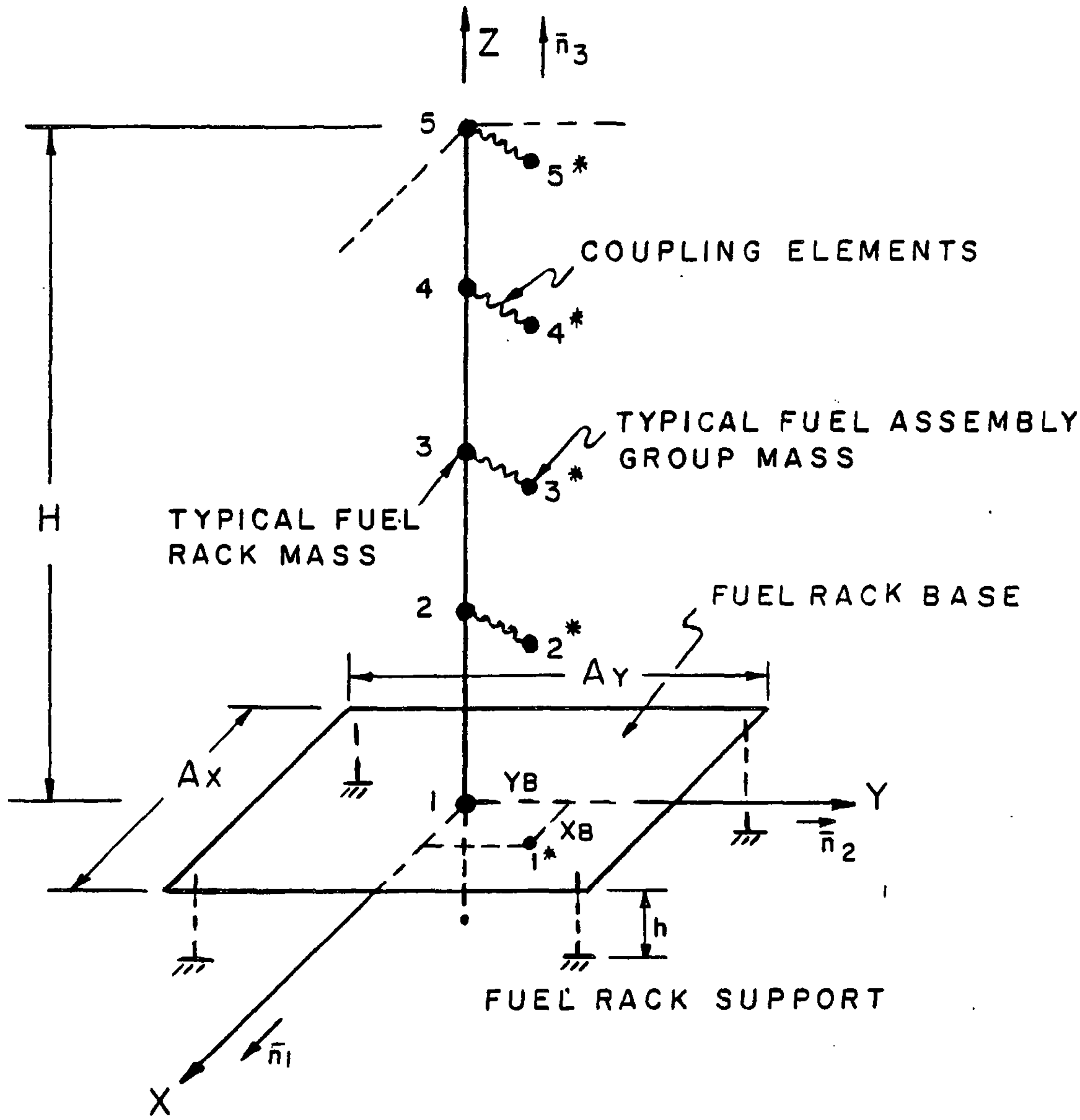
FIGURE 9.1-15, SHEET 3

ELEVATION VIEW OF A TYPICAL HOLTEC HIGH DENSITY STORAGE RACK MODULE



XB, YB - LOCATION OF CENTROID OF FUEL ROD GROUP MASSES - RELATIVE TO CENTER OF FUEL RACK

n_i = UNIT VECTORS

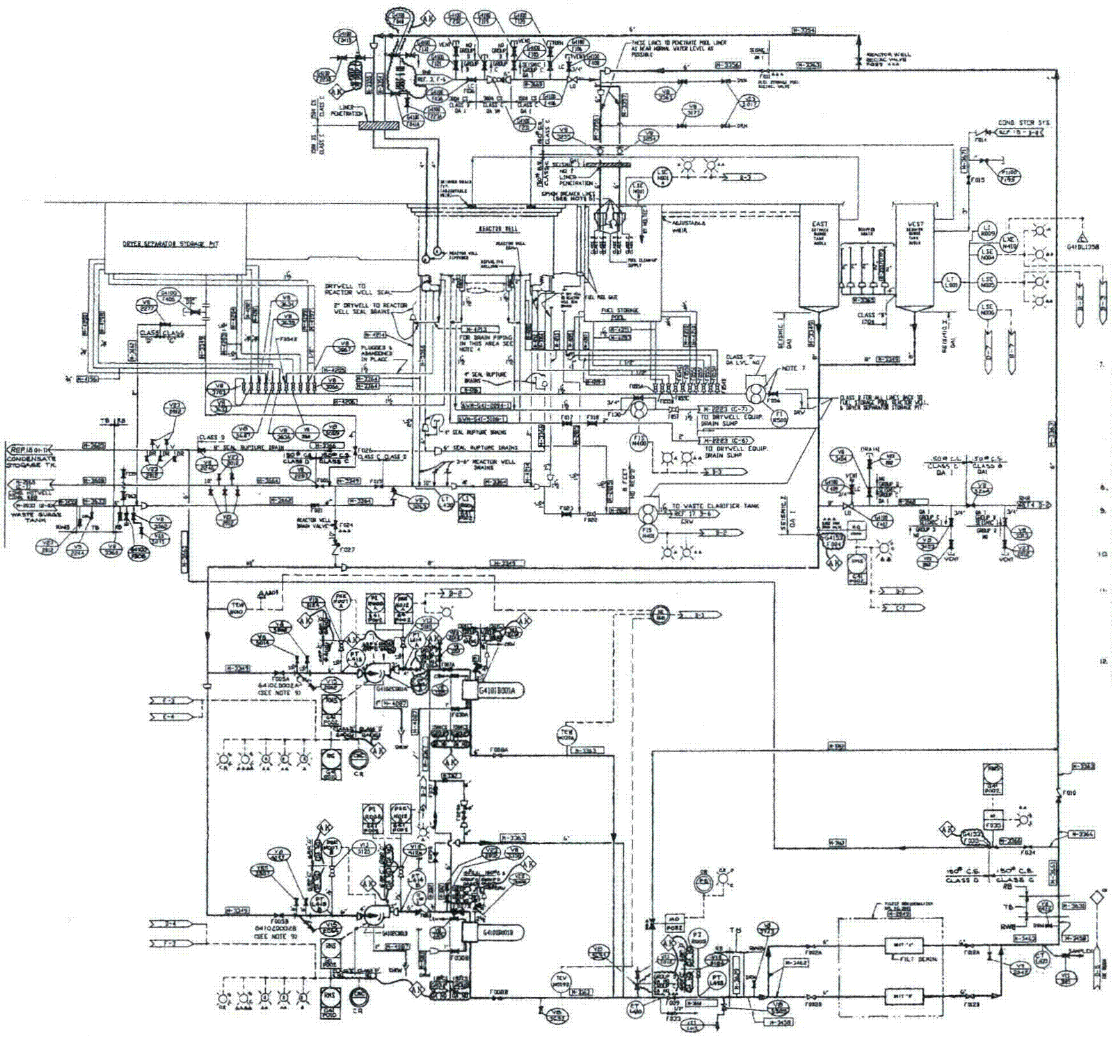
SOURCE: FIGURE 6.1, JOSEPH OATS REPORT
TM 586 (REFERENCE 1)

Fermi 2
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 9.1-16

DYNAMIC MODEL – HIGH-DENSITY SPENT FUEL RACKS

FIGURES 9.1-17 THROUGH 9.1-22
HAVE BEEN INTENTIONALLY DELETED



DIAGRAMS			
REF. NO.	TITLE	D. E. CO. NO.	G. E. NO.
1	DIAGRAM-CONTROL ROD DRIVE HYDRAULIC SYSTEM (CRD)	[illegible]	[illegible]
2	DIAGRAM-STAND-BY LIQUID CONTROL SYSTEM (SLC)	[illegible]	[illegible]
3	DIAGRAM-RESIDUAL HEAT REMOVAL SYSTEM-DIVISION I (RHR)	[illegible]	[illegible]
4	DIAGRAM-RESIDUAL HEAT REMOVAL SYSTEM-DIVISION II (RHR)	[illegible]	[illegible]
5	DIAGRAM-CORE SPRAY SYSTEM (CSS)	[illegible]	
6	DIAGRAM-HIGH PRESSURE COOLANT INJECTION SYSTEM (HPCI)	[illegible]	[illegible]
7	DIAGRAM-HIGH PRESSURE COOLANT INJECTION SYSTEM BAROMETRIC CONDENSER	[illegible]	[illegible]
8	DIAGRAM-REACTOR CORE ISOLATION COOLING SYSTEM (RCIC)	[illegible]	[illegible]
9	DIAGRAM-REACTOR CORE ISOLATION COOLING SYSTEM BAROMETRIC CONDENSER	[illegible]	[illegible]
10	DIAGRAM-REACTOR WATER CLEAN-UP (RWCU)	[illegible]	[illegible]
11	DIAGRAM-REACTOR WATER CLEAN-UP FILTER DEMINERALIZER (RWCU)	[illegible]	[illegible]
12	DIAGRAM-FUEL POOL COOLING SYSTEM (FPC)	[illegible]	[illegible]
13	DIAGRAM-FUEL POOL COOLING SYSTEM FILTER DEMINERALIZER	[illegible]	[illegible]
14	DIAGRAM-REACTOR BUILDING CLOSED COOLING & EMERGENCY COOLING SYSTEM	M-3358	
15	[illegible]	[illegible]	
16	DIAGRAM-NUCLEAR BOILER SYSTEM	[illegible]	[illegible]
17	[illegible]	[illegible]	[illegible]
18	DIAGRAM-CONDENSATE STORAGE SYSTEM	[illegible]	
19	DIAGRAM-STATION AIR SYSTEM	[illegible]	
20	DIAGRAM-FIRE WATER SYSTEM	[illegible]	
21	DIAGRAM-REACTOR RECIRCULATING SYSTEM		[illegible]
22	DIAGRAM-RESIDUAL HEAT REMOVAL SERVICE WATER SYSTEM	[illegible]	
23	DIAGRAM-NUCLEAR BOILER SYSTEM	[illegible]	[illegible]
24	DIAGRAM-WASTE COLLECTOR SYSTEM	[illegible]	[illegible]

NOTES:

1. THIS DIAGRAM REPLACES THE GENERAL ELECTRIC DIAGRAM Nº 729E609AB REV 5 EDISON FILE #R1-10
2. SPECIFIC SYSTEM DESIGN REQUIREMENTS ARE GIVEN IN THE FUEL POOL COOLING & CLEANUP SYSTEM DESIGN SPECIFICATION Nº 3071-509
3. THE PLANT IDENTIFICATION NUMBER FOR THE FPC SYSTEM IS G4100
4. DRAIN PIPING EMBEDDED IN CONCRETE SHALL BE 150# S.S. ALL OTHER DRAIN PIPING TO BE 150# C.S.
5. VACUUM RELIEF LINES ARE OPEN ENDED.
6. MATERIALS, PIPE SCHEDULES, PRESSURES AND TEMPERATURES ARE GIVEN ON THE ISOMETRIC DRAWINGS
7. FUEL POOL DRYER-SEPARATOR PIT AND REACTOR WELL LINERS ARE TO BE DIVIDED INTO SECTIONS, EACH WITH ITS OWN LINER SUMP. THESE DRAINS ARE TO BE PIPED TO AN ACCESSABLE AREA WHERE SHUTOFF VALVES ARE TO BE LOCATED. A LINER LEAK MAY BE LOCALIZED BY SEQUENTIALLY OPENING THE SHUTOFF VALVES TO DRAIN THE LINES INTO THE HEADER. A SIGHT GLASS IS PROVIDED IN THE HEADER TO OBSERVE THE LEAKAGE.
8. SEE DWG. No M-2057 & 2059 FOR LOCATION OF LINER DRAINS.
9. REMOVE STRAINER BASKETS AFTER START-UP. (SEE CECO)
   FUEL POOL PUMP STRAINERS ORDERED ON B/M P1-63M
10. ALL EQUIPMENT AND PIPING COMPONENTS ARE DESIGNATED QUALITY ASSURANCE LEVEL II "UNLESS NOTED OTHERWISE"
11. ALL SEAL WELDS THAT CONNECT DRAIN LINES TO THE LINERS OF THE DRYER SEPARATOR STORAGE PIT, REACTOR WELL, OR FUEL STORAGE POOL, ARE CONSIDERED TO BE PART OF THE LINER, NOT THE PIPE, AND ARE THEREFORE CLASSIFIED AS GROUP "D" WELDS.
12. G4153F004 CAN BE MANUALLY CLOSED TO ALIGN FOR RHR FPCU ASSIST. MOTOR AND POWER ARE NO.

INSTRUMENT & CONTROL SYSTEM NOTES

A. UNLESS OTHERWISE SHOWN ALL INSTRUMENT NO'S ON THIS DIAGRAM ARE FOR SYSTEM G41
B. NOTE REMOVED
C. NOTE REMOVED
D. CR IS H11P601

REFERENCES:

1. PIPING & INSTRUMENT SYMBOLS-GE 1778687 (1-25)
2. LEGEND OF SYMBOLS & INSTR. IDENT M-2001-FOR PLANT SYSTEM DIAGRAM

LEGEND

▲ THIS INSTRUMENTATION IS TO BE LOCATED ON THE OPERATING FLOOR PANEL P001

▲▲ THIS INSTRUMENTATION IS TO BE LOCATED IN THE FUEL POOL PUMP PANEL P002

▲▲▲ THIS EQUIPMENT IS TO BE LOCATED IN VICINITY OF FUEL POOL PUMPS AND HEAT EXCHANGERS.

▲▲▲▲ THIS INSTRUMENTATION IS TO BE LOCATED IN THE RADWASTE BUILDING FILT/DEMIN PANEL G41P010.

DWG Nº — IDENTIFIES PIPING ISOMETRIC FOR FABRICATION & ERECTION.

Fermi 2

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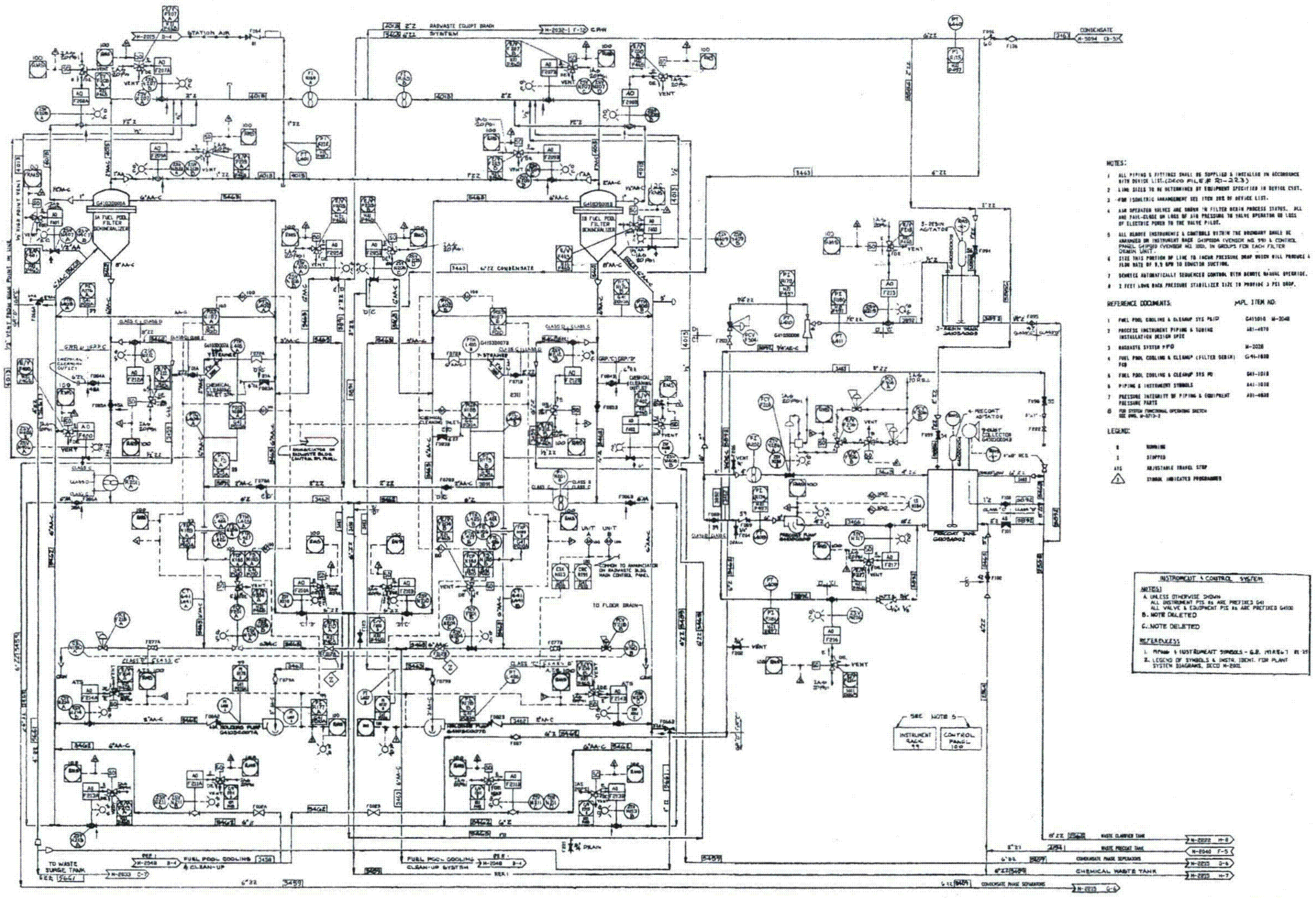
FIGURE 9.1-23

FUEL POOL CLEANING AND CLEANUP SYSTEM P&ID

DETROIT EDISON COMPANY DRAWING NO. 6M721-2046, REV. AK

REV 19 10/14



Fermi 2

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FIGURE 9.1-24

FUEL POOL FILTER/DEMINERALIZER

DETROIT EDISON COMPANY DRAWING NO. 6M721-2049, REV. Z

FIGURE 9.1-25 HAS BEEN DELETED

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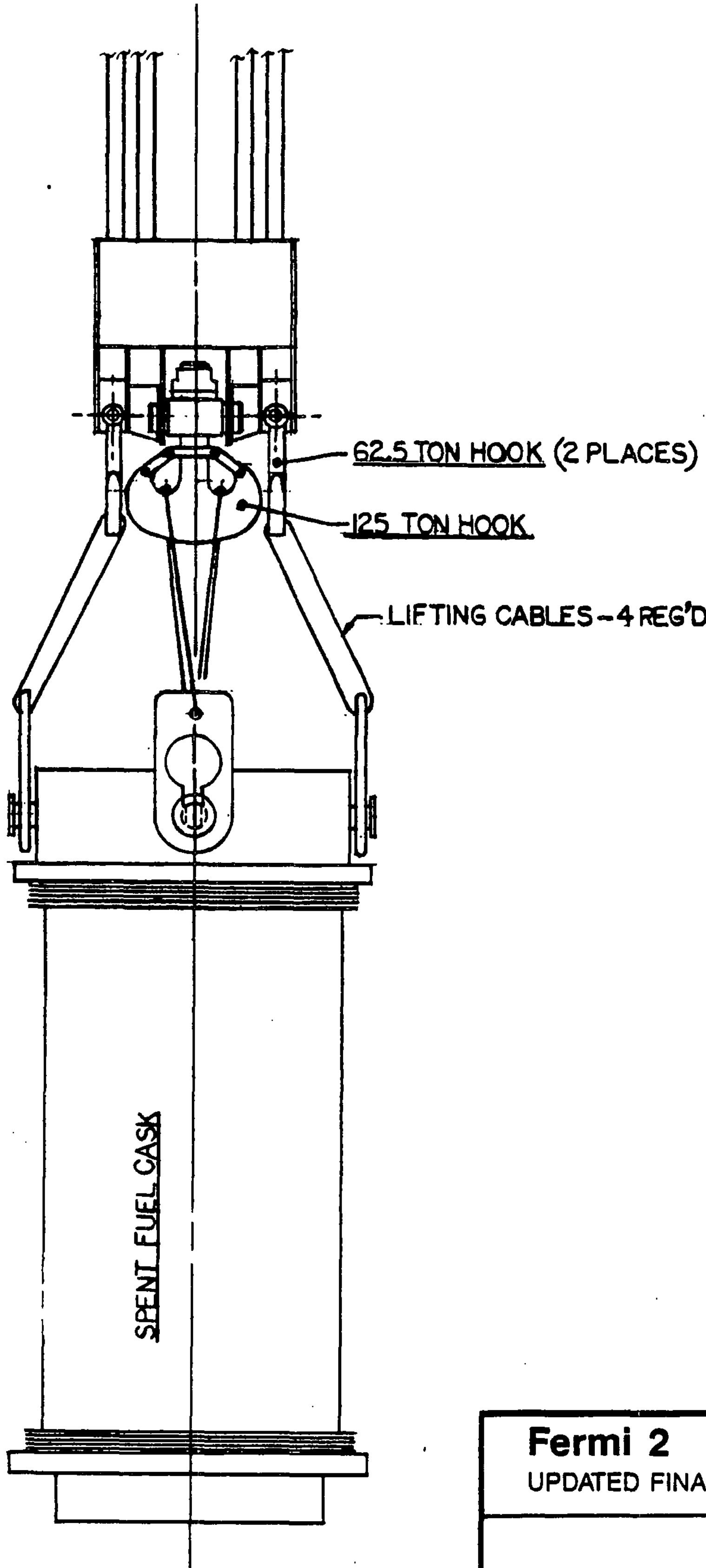

Fermi 2
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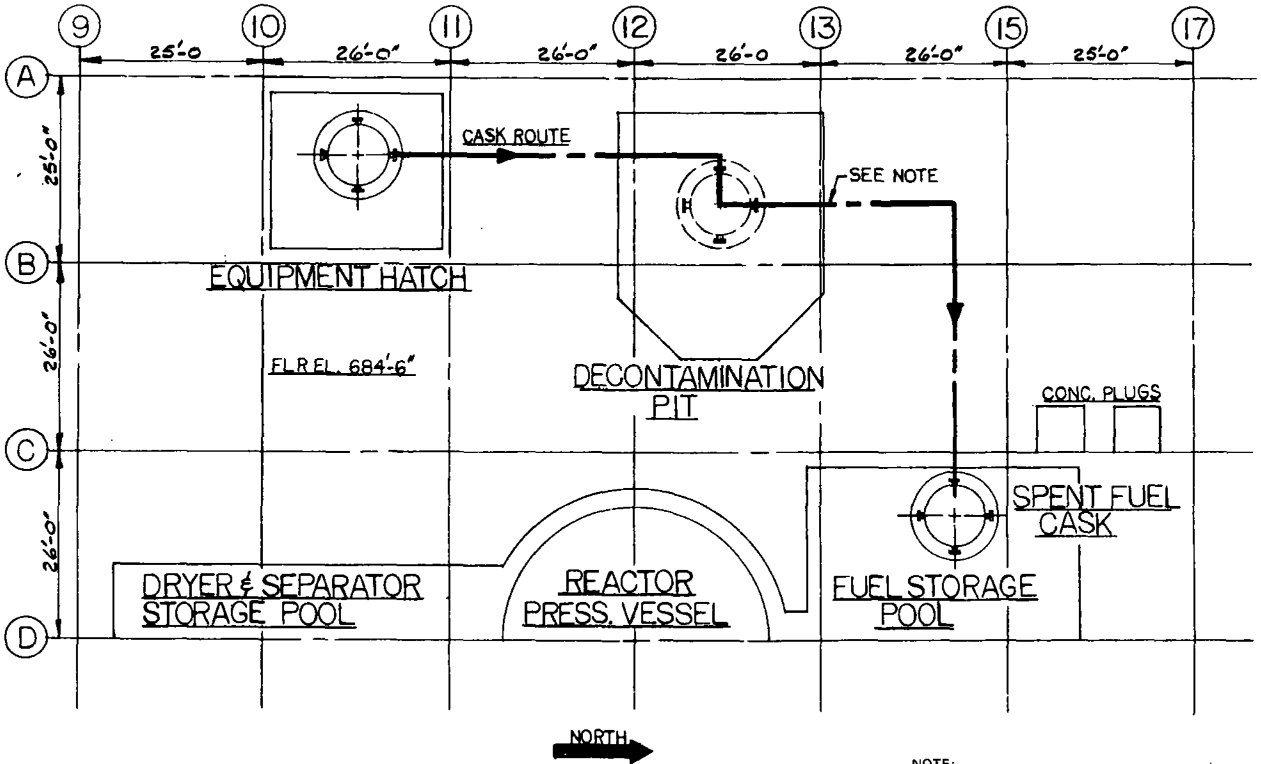
FIGURE 9.1-26

CASK RIGGING SCHEME



NOTE:
WIDE BLACK TAPE OR PAINT ON FLOOR TO GUIDE CRANE OPERATOR.

Fermi 2
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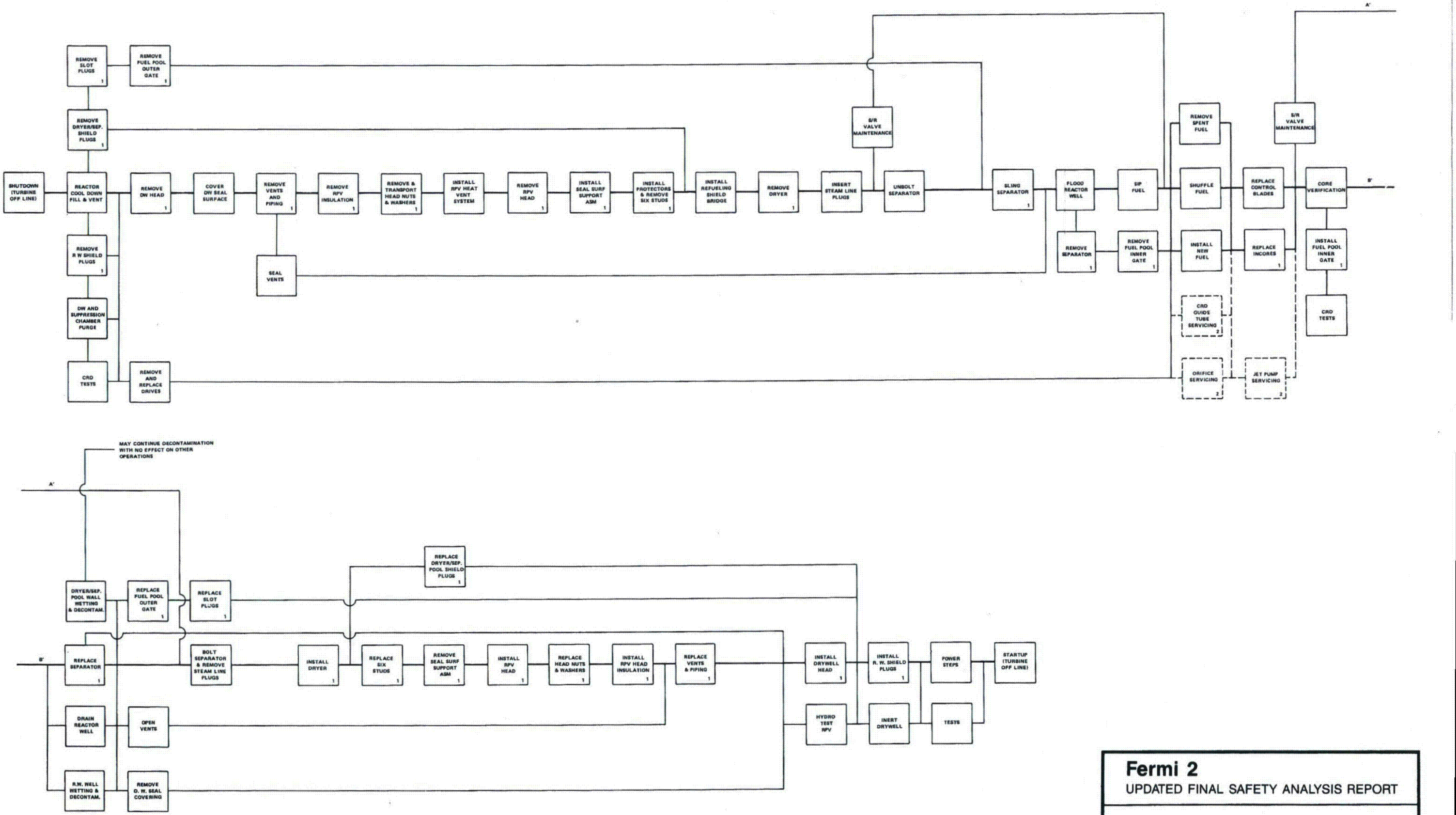
FIGURE 9.1-27

OPERATING FLOOR CASK TRAVELING PATH



Fermi 2

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FIGURE 9.1-28

PLANT REFUELING OUTAGE – FLOW DIAGRAM

GENERAL ELECTRIC COMPANY DRAWING NO. 197R568, REV. D

9.2 WATER SYSTEMS

9.2.1 General Service Water System

9.2.1.1 Design Bases

The general service water (GSW) system is designed to remove various plant heat loads principally from the reactor, turbine, and radwaste buildings during normal station operation. Cooling water is drawn directly from Lake Erie, passed through traveling screens, pumped throughout the plant, and is then returned to the circulating water system where the heat is ultimately rejected to the atmosphere via the plant's natural draft cooling towers.

The GSW system is designed to operate at a higher pressure than the systems it cools, to provide protection against potential outleakage of radioactive contaminants to the environment.

The GSW system is classified as a nonnuclear system and is constructed in compliance with standards for Quality Group D components. This criterion is met by designing the system to ASME Section VIII and ANSI B31.1.0 code requirements. The system is nonseismic, except that portion of the system within the reactor building, auxiliary building, and RHR complex which is designated as Seismic Category II/I.

9.2.1.2 System Description

The GSW system, as shown in Figure 9.2-1, Sheet 1 is designed to remove heat from or provide water to the following equipment on a continuous basis:

a. The reactor building closed cooling water system (RBCCWS) heat exchangers
b. The turbine building closed cooling water system (TBCCWS) heat exchangers
c. The turbine oil coolers
d. The generator hydrogen coolers
e. The radwaste evaporator condenser
f. Reactor building and turbine building room coolers
g. Circulating water pump bearing cooling water and lubricating water.
h. GSW biocide injection system
i. Supplemental cooling chilled water system

The GSW system also provides, on an intermittent basis, water for the following systems or functions:

a. Circulating water biocide injection system
b. Traveling water screen backwashing and deicing
c. Fire protection system (FPS) makeup (via the FPS jockey pump or the GSW to FPS cross-tie line)

d. Residual heat removal (RHR) reservoir makeup

e. Lawn sprinklers

f. Sump flushing.

g. The Sanitary Sewage Treatment Facility (outside protected area)

h. Side Stream Liquid Radwaste Processing System (SSLRPS)

GSW system pump data is shown in Table 9.2-1.

The GSW system takes its water from Lake Erie. The lake water is drawn into an intake canal, passed through a trash rack and a traveling screen, and enters the GSW pump pit. The five GSW pumps take suction from the intake pit and discharge the water into a common header. The GSW pumps operate continuously, maintaining pressure in the GSW header. Minimum pump flow protection is provided by relief valves at each pump's discharge to prevent overheating in the event that the pump is inadvertently run deadheaded. Relief valves are provided in the system piping to prevent overpressurizing the system.

The GSW pump house is located on the existing intake canal serving Fermi 1. It houses the five 25 percent-capacity GSW pumps, two 100 percent-capacity circulating water reservoir makeup pumps, and two 100 percent-capacity fire pumps. The two stage GSW pumps are of the vertical wet-pit type, rated for 7700 gpm flow, and a tested head between 241 and 270 ft. Since the flow demand varies seasonally, the pumps are manually started and stopped by the operator from the main control room. Each pump has a basket strainer located in its discharge line to remove suspended material that has been carried through the traveling screens at the pump house inlet. The strainers are provided with automatic self-backwashing feature.

The GSW system is treated with a biocide to inhibit slime and algae growth and to control organic and inorganic fouling of heat exchanger and piping surfaces. The biocide injection system is shown in Figure 9.2-1, Sheet 2.

Traveling screens and stationary racks are provided to keep floating debris from entering the GSW intake pit. A line from the GSW supply header automatically provides high-pressure water to each screen for backwashing whenever the differential pressure across the screens rises above a predetermined value. A screen deicing line, tapped off the GSW discharge header just prior to its connection into the main condenser circulating water line, provides warm water to keep ice from forming around the screens.

The majority of GSW flow to equipment being cooled is controlled by temperature control valves. Each valve is modulated in response to the exit temperature of the process equipment that the GSW is cooling. The flow to remaining GSW loads is modulated by manual flow valves.

9.2.1.3 Safety Evaluation

The GSW system is not required to be operable in order to effect the safe shutdown of the reactor. Thus, the GSW system is not designed for a single active or passive failure as required of a safety or safety-related system, but sufficient redundancy and automatic protective features are provided to ensure efficient plant operation and availability. Since the

GSW system is not an engineered safety feature (ESF), it is not powered by an essential power bus.

The only portion of the GSW system directly involved in reactor operation and shutdown is the section serving the RBCCWS (via the RBCCW shell and tube heat exchangers and the supplemental cooling chilled water system chiller condensers). If the GSW system becomes inoperative, the emergency equipment service water system (EESWS) takes over to serve the equipment essential to safe reactor shutdown through the emergency equipment cooling water system (EECWS) (described in Subsection 9.2.2). The EECWS and the EESWS are powered off the essential buses. No failure in the GSW system can prevent a safe shutdown of the reactor.

The GSW intake structure is designed for operation during low lake levels by drawing water through a 54-in. line from the circulating water reservoir. On low lake level, an alarm alerts the operator of the condition. If necessary, the operator can supply GSW from the circulating water reservoir by opening the normally closed valve in the 54-in. connecting line between the circulating water reservoir and the GSW pump intake pit, and simultaneously closing the sluice gates to isolate the intake canal from the intake pit. The GSW and circulating water systems can be operated for a limited period of time in this mode to support plant load reduction and shutdown. Subsection 2.2.3.1 further discusses the low water considerations.

Radioactive contamination of GSW is avoided by using closed heat exchangers between the service water and the closed cooling water systems. The GSW remains uncontaminated by operating at higher pressure than the cooled system, and any leakage would be from the GSW system to either the TBCCWS or the RBCCWS. In addition, further protection is provided by activity detection equipment located in the circulating water discharge line downstream from the GSW system discharge connection so that both systems are monitored for radioactive contamination.

9.2.1.4 Tests and Inspections

Initial construction tests such as hydrostatic leak tests were conducted per ASME Section VIII and ANSI B31.1.0 code requirements. Initial system flow distribution, valve operability, instrumentation and control loop checks, and alarm setpoints were done in accordance with the Preoperational Test program. After plant startup, heat exchanger operating performance is observed using actual system heat loads.

Periodic testing of the GSW pumps is done during normal system operation by utilizing each pump's test lines and orifice. Each pump's head and flow point is then compared to its initial flow characteristic curve, and an assessment is made for any deterioration to determine the need for any corrective pump maintenance. Instruments and controls are inspected periodically. No periodic leak tests are planned since the system is continuously operating. Periodic visual inspection of the system will detect minor leakages, such as those from valve stems, flanges, and instrument tubing connections.

9.2.1.5 Instrumentation

Sufficient instrumentation is provided to allow the plant operator to assess the status of the GSW system. The GSW pumps are manually started and shut down from the control room

operating panel via coordinated manual control (CMC) switches. Pump status and motor amperage readouts are provided in the control room.

The discharge from each GSW pump flows through an automatic self-cleaning strainer and a discharge isolation valve. GSW system pressure is regulated by changing the number of pumps in service, adjusting the GSW flow through heat exchangers, and/or bypassing some flow back to the pit as necessary. The discharge header pressure is indicated with high and low pressure alarms in the control room.

Water levels in the GSW pump house are measured on each side of the traveling screens to provide additional pump operating intelligence to plant operators. This level instrumentation controls traveling screens and alarms the control room operator of abnormal inlet water levels.

The GSW flow to the major GSW users, turbine oil-coolers, generator hydrogen coolers, TBCCW coolers, and RBCCW coolers, are modulated by temperature control valves. The process temperatures are also provided with a high and low temperature alarm in the control room. The other GSW loads are modulated by manual controls.

9.2.2 Cooling System for Reactor Auxiliaries

9.2.2.1 Design Bases

The RBCCWS is designed to remove heat from the auxiliary equipment housed in the reactor building and auxiliary building during normal plant operation. The RBCCWS is cooled by the GSW system, and makeup is supplied by the demineralized makeup water system. The supplemental cooling chilled water system provides a source of chilled water for cooling the water supplied to each division of EECW serviced by the RBCCW supplemental cooling loops. The GSW system provides service water for condenser cooling of the SCCW chillers. The RBCCW supplemental cooling loops are intended for operation when the GSW supply temperature is greater than 60°F.

In the event of a mechanical failure of the RBCCWS, high drywell pressure, or upon loss of offsite electrical power, the EECWS will start automatically (or may be manually initiated) to cool equipment needed for reactor shutdown. In addition, the EECWS may be used to augment RBCCW for the purpose of assisting in equipment cooling. The EECWS is cooled by the EESWS which is supplied by the RHR reservoir.

To provide for reactor shutdown under severe natural environmental conditions, as well as upon loss of normal offsite power and failure of the RBCCWS, two full-capacity Emergency Equipment Cooling Water loops are provided.

Motor-operated isolation valves are provided to isolate the nonessential loads on the RBCCWS from each EECW loop.

The RBCCW system is operated at a pressure lower than the GSW system to prevent leakage of potentially radioactive water to the environment. Continuous surveillance of the quality and activity level of the RBCCWS is maintained to detect inleakage of GSW or inleakage from the cooled reactor building auxiliary components.

During emergency situations when EECW is in operation, the EECW pressure is slightly greater than the EESW pressure at the EECW heat exchangers. It would take multiple equipment failures to create a situation where radioactive contamination would enter the EESW. First a component being cooled by EECW would have to leak contaminated water into EECW. This would have to be accompanied by a failure in the EECW heat exchanger in order to release radioactive material into the RHR reservoir. If this were to happen, drift losses from the cooling towers could cause a radioactive release to the environment. Given the fact that it would take multiple equipment failures and that monitoring and sampling provisions exist (as described in Subsections 11.4.3.9.2.3 and 11.4.3.9.2.4), the potential for an unmonitored radioactive release is minimal.

The makeup line between the EECW makeup tank and the EESW system for each division is furnished with a check valve and a normally closed air-operated isolation valve. The test return line is provided with two isolation valves that are closed during normal operation. These valves would minimize the potential of radioactively contaminated water leaking into the EESW system during normal operation. The check valve in the makeup line and the closed test return line isolation valves would also minimize the potential for radioactively contaminated water leaking into the EESW system during a design basis accident. The check valves installed as boundary valves on the nitrogen inerting (T48) and demineralized makeup water (P11) systems would minimize the potential for radioactively contaminated EECW water leaking into these systems.

System construction for cooling the essential equipment necessary for reactor shutdown is in compliance with standards for Quality Group C or Quality Group B. Components and equipment not essential to reactor shutdown are built as a minimum to Quality Group D standards. The EECWS is Category I, and is designed in accordance with ASME Section III, Class 2 and Class 3 requirements. The RBCCWS is Seismic Category II/I and is designed to ASME Section VIII and ANSI B31.1.0 code requirements.

9.2.2.2 System Description

The RBCCWS, as shown in Figure 9.2-2, is designed to remove heat from reactor auxiliaries that fall into two categories: those that are essential to reactor shutdown and those that are non-essential to reactor shutdown. Table 9.2-2 lists RBCCWS component design parameters.

The RBCCWS outside of the RBCCW supplemental cooling loops consists of two RBCCW heat exchangers and three 50 percent-capacity pumps. The two, divisional RBCCW supplemental cooling loops are each designed with one RBCCW supplemental cooling (plate-and-frame) heat exchanger and two 100 percent-capacity RBCCW supplemental cooling pumps (see Figure 9.2-2(2)). During normal operation, two heat exchangers and two pumps operate to provide cooling to all the essential and nonessential heat loads. The third pump is in standby and is designed to be started manually on low RBCCWS pressure. When the RBCCW supplemental cooling loops are in operation, one RBCCW supplemental cooling pump will operate for each EECW division. The second pump in each division is in standby and is designed to automatically start on loss of the operating pump. The RBCCW supply header temperature is maintained nominally at 70°F by a temperature control valve modulating the GSW flow through the RBCCW heat exchanger. During the summer season,

the RBCCW temperature may be higher. When these higher temperature conditions occur, the RBCCW supplemental cooling loops may be used to cool the water that is supplied to EECW. The water temperature exiting each RBCCW supplemental cooling heat exchanger will be maintained by a temperature control valve which may be operated in automatic or manual mode. Both divisional loops of RBCCW supplemental cooling exchange heat with the supplemental cooling chilled water (SCCW) system. SCCW may be placed in operation when the GSW supply temperature exceeds 60°F. The system thermal capacity is based on a nominal 85°F RBCCW temperature. System pressure is controlled by a differential PCV located in the bypass line between the suction and discharge headers of the RBCCW pumps. Makeup to the system as well as system expansion and contraction resulting from load changes are provided by a makeup tank. Makeup water is automatically supplied via a level control valve. Normal makeup is from the demineralized water system and alternatively from the condensate storage system.

Reactor auxiliaries impose a maximum cooling load on the RBCCW heat exchangers of approximately 68 x 10^6 Btu during normal operation. This requires approximately 10,000 gpm of GSW, assuming a maximum service water temperature of 85°F. Circulation on the RBCCW side of the heat exchanger is approximately 8000 gpm; heat exchanger rate is based on temperatures of 112°F in and 95°F out.

The GSW is not used directly because the relatively high impurity level in this system might result in fouling of equipment heat transfer surfaces. Furthermore, the intermediary loop between contaminated reactor auxiliaries and the GSW system provides additional protection against radioactive water leakage into the environment.

The EECW section of the RBCCWS consists of two redundant full-capacity loops, each with two (2) 100 percent capacity heat exchangers, pump, and makeup pump and tank, as shown in Figures 9.2-3 and 9.2-4. One heat exchanger is manually aligned for service. The second heat exchanger is provided as a backup. The twin systems designated as Division I and Division II are cooled by the EESWS. The EESWS, described in Subsection 9.2.5, is powered off the essential buses and is designed to be redundant throughout. Upon loss of offsite power, high drywell pressure, or failure of the RBCCWS, both divisions of the EECWS are automatically activated; that is, pumps start, makeup tanks isolation valves open, and valves isolate the nonessential portion of the RBCCWS. The makeup tanks isolation valves do not start to open until the divisional isolation valves are closed. Upon loss of RBCCWS differential pressure between the supply and return headers, either Division I and/or Division II EECW loops will start automatically, depending on the portion of the RBCCWS affected. The EECWS may also be manually initiated. Component design parameters of the EECWS are shown in Table 9.2-3.

The EECW heat exchangers are designed for a maximum heat load of 13.6 mBtu/hr. This requires approximately 1450 gpm of EESW, assuming a maximum service water inlet temperature of 89°F. Circulation on the EECW side of the heat exchanger is approximately 1700 gpm; heat exchanger rate is based on temperatures of 111.1°F in and 95°F out. The EECWS temperature is maintained nominally at 70°F in a similar manner as the RBCCW heat exchanger by modulating the heat exchanger exit cooling water flow.

The replacement of the original shell-and-tube EECW heat exchangers with a plate-and-frame design increased nominal heat transfer capability, but reduced the minimum flow

channel dimension from 0.78-in. diameter tubes to 0.0732 inch plate spacing; thus, making the new units potentially more susceptible to plugging. The design analyses that define minimum EECW heat exchanger thermal performance consider the potential effects of initial plugging and plugging rate to establish the thermal performance and heat exchanger differential pressure vs. flow test criteria necessary to ensure the accident mission can be accomplished with credit for only one of the two identical units provided in each division. Once the maximum allowed normal operating plugging limit on a unit is reached, the EECW and EESW flows may be aligned to the clean spare heat exchanger in each division; thereby facilitating maintenance without interrupting normal plant operation.

The RBCCW makeup tank and the EECW makeup tanks are supplied with demineralized water and pressurized with nitrogen during normal plant operation. Normal makeup to the tanks is supplied automatically from the demineralized makeup water system by a level control valve. The pressure regulating valve of the normal nitrogen supply system maintains a nitrogen blanket in the tank at a pressure which will keep the EECW loop full to the upper elements. Nitrogen is provided to prevent leakage of oxygen into the system, thereby retarding corrosion of the closed cooling water system.

The EECW (Division I and Division II) system makeup tank is connected with a makeup line to the EESW system to provide an alternate makeup supply for each division when the normal makeup supply to this tank is lost during and after the design basis accident. The isolation valves for the alternate makeup supply consist of a check valve and an air-operated valve which opens automatically on a makeup pump start, a loss of air or a loss of electrical power. This valve is normally closed to prevent EESW, low quality water from entering the EECW, high quality water system during winter operation when TCV F400A/B is nearly closed. Each makeup pump auto starts and provides the makeup water to the tank. A check valve is installed in this makeup supply line to prevent a reverse flow from the EECW makeup tank to the EESW system. The test return line is provided with two isolation valves that are closed during normal operation and accident conditions to minimize the potential for radioactively contaminated water leaking into the EESW system. Inadvertent injection to the EECW system is minimized by system initiation setpoints and permissives. Inadvertent injection of EESW water into the EECW system during testing is minimized by isolation of the tank during testing. The makeup tank for Division I is also provided with the emergency backup nitrogen supply cylinders which automatically provide nitrogen to the makeup tank whenever the normal nitrogen supply is lost and/or the nitrogen supply pressure is reduced below approximately 26 psig. The check valves are added as the boundary valves between the makeup tank and nonsafety-related nitrogen inerting and demineralized water makeup systems to reduce the potential loss of the makeup tank inventory or pressure due to the loss of the nonsafety-related systems. The backup nitrogen supply cylinders are sized to maintain nitrogen pressure in the makeup tank until the makeup tank has refilled the EECW loop during an Appendix R fire. This will maintain a positive suction head on the EECW pump and provide protection against momentum transients when the EECW pump experiences a delayed start during the dedicated shutdown scenario.

The following equipment, considered essential to reactor shutdown, can be cooled either by the RBCCWS or, in an emergency, by at least one division of the EECWS:

a. RHR pumps (two out of four)

b. Core spray pumps (two out of four)

c. Reactor auxiliary space coolers (three in Division I or four in Division II)

d. Standby control air compressor, aftercooler, and space cooler (one out of two sets)

e. Post-LOCA thermal recombiner system coolers (one out of two)

f. Switchgear room space coolers (two out of four)

g. Standby gas treatment room space cooler (one out of two)

h. Control center air conditioning equipment (one out of two)

i. Auxiliary building battery charger area space coolers (one out of two).

The nonessential components of the RBCCW that are connected to the EECWS piping have automatic isolation valves installed in their supply lines. The following equipment, considered nonessential to reactor shutdown, is cooled only by the RBCCWS:

a. Two in-series reactor water cleanup nonregenerative heat exchangers

b. Water sample station cooler

c. Twin reactor water cleanup pump seals and bearings

d. Twin fuel pool heat exchangers

e. Twin recirculating pump motor-generator coupling cooler heat exchangers

f. Recirculating pump motor-generator ventilation cooling coils

g. Steam tunnel cooler

h. Drywell equipment sump heat exchanger

i. Reactor building equipment sump heat exchangers (two)

j. Control rod drive (CRD) pumps

k. Battery room space cooler

l. Drywell penetration cooling (eight)

m. High-pressure cask-washdown pump heat exchanger

n. Instrument rack H21-P284.

NOTE: All twin units are sized for 100 percent redundancy.

Two additional loads, drywell coolers (seven per division) and the reactor recirculation pumps, are normally cooled by RBCCW. Flow to this equipment is maintained upon activation of the EECWS. Should EECWS activate in conjunction with a high drywell pressure signal, the supply valve to the drywell will close, thus ensuring cooling of the essential loads.

9.2.2.3 Safety Evaluation

The EECW Division I and Division II portions of the RBCCWS are designed to provide cooling to equipment required for reactor shutdown in spite of a single active or passive failure. Division I and Division II loops are completely isolable from each other. Each loop of the EECWS is operable from a separate emergency bus. Single-failure analysis for the RBCCW and EECW systems is presented in Table 9.2-4. Upon activation of the EECWS, all nonessential loads of the RBCCWS will be isolated except for the drywell coolers and the reactor recirculation pumps. These loads can be manually isolated from the control room and the supply valve will be automatically closed if a high drywell pressure occurs.

The EESWS cooling the EECWS is also completely redundant and powered off separate emergency buses. This system is discussed in Subsection 9.2.5.

The EECWS components of Division I and Division II are located in different areas of the reactor building to preclude failure of both systems due to pipe whip, jet forces, and generated missiles. Physical separation also provides protection against common failure induced by fire, as described in Appendix 9A.

To detect leakage from the RBCCW and EECW systems, the makeup tanks are provided with low-level alarms and an alarm on the makeup valve stem position. Excessive opening of the makeup valve will be indicative of a substantial system leak. Inleakage of GSW or SCCW will be indicated by a high-level alarm in the RBCCW makeup tank. The RBCCWS is continuously monitored for radioactivity. Leakage to GSW and SCCW is minimized by operating the RBCCW at a relatively low pressure.

The use of demineralized water for makeup and nitrogen capping of the makeup tanks gives reasonable assurance against long-term degradation caused by impurities in the circulating loops. Additionally, corrosion inhibitors are added for pH and oxygen control. Alternatively, the cooling water system may be maintained with pure demineralized water only, with no chemicals added.

The makeup line between the EECW makeup tank and EESW system in each division provides emergency makeup water to the makeup tank by automatically starting the makeup pump and opening the air-operated valve. This makeup system is initiated on either low makeup tank pressure or low makeup tank level when the makeup tank isolation valve is open, and normal pump suction pressure is achieved. In Division I the backup nitrogen supply system to the EECW makeup tank is automatically actuated, based on the makeup tank pressure, upon loss of the nonsafety-related nitrogen inerting system and maintains the nitrogen pressure throughout a dedicated shutdown fire scenario. The check valves installed in the nonsafety-related water and nitrogen supply lines will protect the makeup tank from the potential loss of water inventory or nitrogen pressure.

Both EECW loops are automatically started on high drywell pressure or upon loss of normal offsite power. Upon failure of the RBCCWS, such as pipe rupture, redundant differential pressure switches automatically start the EECW pump(s), depending on the location and severity of the break, and initiate appropriate loop isolation consistent with the operating EECW pump(s).

EECW may be manually initiated with the nonessential loads subsequently restored to facilitate RBCCW heat exchanger cleaning, to enhance drywell cooling during high lake water (GSW) temperature, for testing, or to provide RHR Reservoir freeze protection during extreme cold weather. EECW auto-start on high drywell pressure (i.e., a LOCA) or on a loss of offsite power is unaffected by this mode of operation; therefore, these signals will initiate the automatic protective action of reisolating the nonessential portions of RBCCW piping located inside the EECW system envelope. A loss of RBCCW while EECW is operating in this mode will not reinitiate EECW or reisolate the nonessential loads. This action is not required, however, since this is not a condition requiring protective action as described in Section 7.1.2.1 and EECW remains capable of supporting the safe shutdown of the plant in this configuration.

9.2.2.4 Tests and Inspections

Initial construction tests such as hydrostatic leak tests were conducted per applicable code requirements for the RBCCW and EECW systems. Initial system flow distribution, valve operability, instrumentation and control loop checks, and alarm setpoints were done in accordance with the Preoperational Test program as discussed in Chapter 14. Heat exchanger operating performance will be observed during plant operation. Availability of the RBCCW standby pump and automatic start of the EECW pumps and makeup pumps are tested periodically. Periodic inspections and testing will be performed to monitor heat exchanger performance and cleanliness. Individual pump performance will be assessed in regularly scheduled inspections that can be performed without interruption of plant operation. Isolation of the EECWS will be tested periodically by simulating the initiating events that automatically bring the EECW and EESW systems into operation.

No periodic leak tests will be performed because the system is under continuous pressure during operation. Periodic visual inspection of the system will detect minor leakages such as those from valve stems, flanges, and instrument tubing.

9.2.2.5 Instrumentation

The RBCCW pump motors are equipped with standard controls and protective devices, and are monitored from the main control room. Readouts to observe pressure and inlet and outlet temperatures in the RBCCW and EECW systems are also provided in the main control room. Interlocks are provided on the RBCCW pumps to prevent their starting upon low makeup tank level and/or low suction pressure. High/low pump differential pressure and high/low makeup tank level are alarmed for the RBCCW and EECW systems. Low suction pressure is alarmed only on the EECW pumps. Individual components are equipped with local temperature indicators to periodically monitor performance. The temperature of the RBCCWS (or the EECWS) during operation is controlled by modulating the discharge flow of tube-side cooling water through the respective heat exchangers.

High- and low-level alarms and alarms for low system pressure are provided on the makeup tanks to alert the operator of system leakage and EECW makeup pump failure.

The RBCCW supplemental cooling pumps will be controlled and monitored from a control panel located in the basement of the turbine building. When one or more of these pumps trip, a common trouble alarm will be generated in the main control room. An interlock is

provided to trip the RBCCW supplemental cooling pumps on low flow to the RBCCW header. A trip of the operating pump will automatically start the standby pump supplying the same division. A low level in the RBCCW makeup tank will prevent operation of all RBCCW supplemental cooling pumps. During the normal mode of operation when both loops of supplemental cooling are in operation, an interlock is provided to prevent these pumps from operating without the SCCW chillers in operation. Should no SCCW chillers be in operation, none of the pumps will start and all operating RBCCW supplemental cooling pumps will be tripped.

During the normal mode of operation one of the following two (2) modes are applicable. These two (2) modes of operation are dependent upon a single SCCW chiller's Full Load Amps (FLA) reading. Mode 1 allows the operation of any one single chiller and two (2) chilled water pumps to provide cooling to both RBCCW-SCS loops SCS-1 and SCS-2 when GSW inlet temperature is low. Mode 2 requires the operation of two (2) chillers and two (2) chilled water pumps to provide cooling for both RBCCW SCS loops.

If it is desired to operate only the supplemental cooling pumps or chilled water pumps for the purpose of maintaining water quality, a maintenance switch is provided which permits operation of the pumps without the chillers running.

The water temperature exiting each RBCCW supplemental cooling heat exchanger is maintained at the required value by a temperature control valve that controls the bypass of RBCCW flow around the heat exchanger.

9.2.3 Demineralized Water Makeup System

9.2.3.1 Design Bases

The demineralized water makeup system is designed to deionize water and store it for makeup to the reactor coolant system and plant auxiliary system and services, and is designed as a direct source of water for flushing and cleaning operations.

The raw water is supplied from the potable water system. The influent and effluent water qualities are shown in Tables 9.2-5 and 9.2-6, respectively.

The demineralized water makeup system is nonseismic, and is constructed in compliance with standards for Quality Group D components. This criterion is met by designing the system to ASME Section VIII and ANSI B31.1.0 code requirements.

9.2.3.2 System Description

The demineralized water makeup system consists of a 300-gal potable water holding tank, a packaged skid-mounted reverse osmosis unit, and two raw water booster pumps with associated distribution piping.

Normal operation of the make-up demineralizer is manually initiated from the reverse osmosis (RO) unit control panel. The booster pump takes suction from the raw water holding tank and sends it to the RO water treatment system. The RO system consists of three separate skid mounted units. The first skid is the pretreatment unit which contains the media filters, carbon filters and the softeners. The second skid is the main RO unit which includes

both passes of the reverse osmosis system and the cleaning pump. The last skid, the post treatment unit, contains the transfer pump, ultraviolet light and deionization (DI) bottles.

The normal operation of the RO units is in series. The system operated in the series mode, will provide approximately 25 gpm of purified water. Only one of the two booster pumps needs to operate to supply water to the RO unit. The purified water is stored in the demineralizer make-up water storage tank. The concentrate which is the reject water of this system is simply concentrated potable water and is discharged into the boiler blowdown sump.

Potable water flow into the holding tank is controlled automatically by a level control inlet valve. Operation of the system depends on the water level in the makeup demineralized water storage tank. Makeup demineralized water storage tank level indication with a low-level alarm is provided in the main control room.

Connections are provided to recycle the contents in the demineralized storage tank through the makeup demineralizer to upgrade the quality of the water as it deteriorates due to CO_2 absorption during storage.

9.2.3.3 Safety Evaluation

The demineralized water makeup system is not required for reactor shutdown and as such is not a safety-related system. Redundancy to ensure continuity of design function is not required. Because of the processes involved in this facility, only high-purity water is handled, and no long-term degradation of equipment is anticipated.

9.2.3.4 Tests and Inspections

Initial system flow checks, valve operability, instrumentation and control loop checks, and alarm setpoints were performed in accordance with the Preoperational Test program as discussed in Chapter 14. Flow meters are provided to ascertain pump performance as are on-line conductivity monitors to determine water quality. Grab samples will be taken periodically to confirm water quality and to verify instrument accuracy.

Visual inspection will detect minor leakages such as those from valve stems, flanges, and instrument tubing. Potable water booster pumps will be operated on a rotating basis.

9.2.3.5 Instrumentation

The demineralized water makeup system is operated from the RO unit control panel which is a local panel located in the auxiliary boiler house. Local flow meters provide indication of raw water flow through the system, including the flow through the various skids of the RO unit. On-line conductivity monitors provide information on the various skids of the RO unit performance. The various RO unit trouble alarms and the raw water holding tank low level alarm will shut down the RO unit operation. Any one of these alarms will also initiate the make-up demineralizer trouble alarm in the main control room.

Switchover and operation of the regenerative and backwashing cycle are manual and done from the local RO unit control panel.

All process instrumentation, including that required to maintain temperature, pressure, and flow for the process cycle and the regenerative cycle, is locally indicated.

9.2.4 Potable Water System

9.2.4.1 Design Bases

The potable water system for Fermi 2 is composed largely of existing facilities at Fermi 1, with extended underground distribution lines.

The potable water system is designed to comply with Quality Group D Standards and State of Michigan Health Department code requirements.

9.2.4.2 System Description

The potable water system for Fermi 2 consists of an underground distribution header with branches to the various facilities that require service.

Fermi 2 demand is supplied by the Frenchtown Water System which is normally transferred to the 100,000 gallon elevated storage tank on site. Potable water is used in Fermi 2 to supply the demineralized water makeup system described in Subsection 9.2.3, sanitary plumbing, drinking fountains, washrooms, kitchen facilities, safety showers, and TBHVAC evaporative coolers.

9.2.4.3 Safety Evaluation

The potable water system has no apparent source of contamination. There are no interconnections between the potable water system and any other systems, except that the potable water supplies the demineralized water makeup system and TBHVAC evaporative air coolers. Potential contamination is precluded by an open break in the fill pipe for the demineralized water make-up system and the TBHVAC evaporative air cooler. Because the facility is specifically intended to handle and eliminate impurities, no long-range degradation of equipment is foreseen. Apart from the demineralized water makeup system (which is not critical to the operation or safe shutdown of the reactor), end users of the potable water system are plant personnel. There is, therefore, no requirement for redundancy in order to maintain uninterrupted service. Adequate water supply to the safety showers and eyewash stations is ensured by the large potable water storage tank.

9.2.4.4 Tests and Inspections

No special test or inspections are required for the Potable and Sanitary Water System.

9.2.4.5 Instrumentation

Indicating instruments are read out locally in the Potable Water Building.

9.2.5 Ultimate Heat Sink

The ultimate heat sink is provided by the RHR complex, which contains the RHR service water (RHRSW) system, the EESWS, the diesel generator service water system, the mechanical draft cooling towers, the emergency ac power system (diesel generators), and the reservoir. The systems are shown in Figure 9.2-6. The ultimate heat sink design conforms to the requirements of Regulatory Guide 1.27.

9.2.5.1 Design Bases

The RHRSW system is designed for the following functions:

a. With the RHR system, to remove decay heat and residual heat from the nuclear system so that refueling and nuclear system servicing can be performed

b. With the RHR system, to supplement the fuel pool cooling system with additional cooling capacity

c. With the RHR system, to remove decay heat and residual heat from the nuclear system by cooling the suppression pool water, following a postulated LOCA

d. To provide a method to flood the reactor pressure vessel (RPV), acting as a backup in the extremely unlikely event that all RHR (low pressure coolant injection [LPCI] mode) and core spray pumps fail to operate following a postulated LOCA

e. To provide a method to flood primary containment so that the fuel can be removed from the RPV following a postulated LOCA.

The EESWS is designed to provide a cooling water source for the EECWS. The system functions only during a loss of offsite power, high drywell pressure, or upon failure of the RBCCWS.

The diesel generator service water system is designed to provide a cooling water source for the emergency diesel generators (EDGs) during testing and emergency operation.

The ultimate heat sink system structures are designed to comply with Category I requirements. System construction is designed to comply with requirements for Quality Group C components. Piping, valves, and pumps conform to the ASME Section III Class 3 Code requirements.

The ultimate heat sink system is sized to provide sufficient cooling for 7 days following a reactor shutdown without makeup water addition to the RHR reservoir.

The system structures (pump house, diesel generator building, reservoir, and cooling towers) are designed so that the equipment is physically separated or separated by barriers to ensure against multiple damage from missiles, pipe whip, and fire.

The ultimate heat sink is designed to withstand severe natural phenomena (safe-shutdown earthquake, tornado, storm, flood, and freezing). It is designed to withstand any single failure of manmade structures or components. All necessary electrical equipment is served by the essential buses in the event of loss of offsite power.

The RHRSW supply pressure is less than the pressure in the recirculating RHR system during a normal shutdown or under accident conditions. Therefore, radiation detectors are attached to return lines from the RHR heat exchangers to the cooling towers to monitor for leaks in the exchangers.

9.2.5.2 System Description

The RHR complex consists of a single highly reliable water supply (reservoir); a means for heat rejection (cooling towers); a standby power source comprising four EDGs; a makeup and decanting system; and associated pumps, piping, and instrumentation.

9.2.5.2.1 RHR Complex Reservoir

The RHR complex reservoir consists of two one-half-capacity reinforced-concrete structures of Category I construction, each with a capacity of 3.41×10^6 gal of water at elevation 583 ft. The reservoirs are connected by redundant valved lines to permit access to the combined inventory of the two reservoirs to either RHR division in the event of a mechanical failure in one of the RHR divisions. Each line contains two isolation valves of Category I construction that are remotely operable from the plant main control room.

Normal reservoir water level is at Elevation 583 ft (New York Mean Tide, 1935). Waterproof construction of the walls is provided to Elevation 590 ft (New York Mean Tide, 1935) for protection against flooding from Lake Erie. Subsection 2.4.2.2.3 contains a further discussion of the RHR complex flood protection. Each division of the reservoir is fitted with a floodproof nonsiphon overflow to eliminate excess water. Makeup water delivery ports are designed to prevent siphon losses in the event of a break in makeup water supply piping.

Reservoir water loss due to leaks in the RHRSW, EESW, or diesel generator service water lines is detected by redundant level indicators in each division of the reservoir. Comparison between expected water level due to cooling tower losses and actual indicated level will provide sufficient data to determine any system leakage.

9.2.5.2.2 Cooling Towers

A two-cell induced-draft cooling tower is located over each division reservoir. The towers are of Category I fireproof construction with reinforced-concrete shells, cement board fill, and mist eliminators. Each tower is designed to cool one division of the plant load (one RHR heat exchanger, one EECW heat exchanger, and two EDGs), thus providing complete redundancy. Component design parameters for each tower are given in Table 9.2-7.

Each RHRSW cooling tower cell fan is driven by a 150-hp two-speed motor. The motor is connected to the ESF bus of the EDGs for a redundant power supply, and is manually started and stopped from the main control room.

The towers and fan drives are provided with a reinforced-concrete protective shell for tornado, earthquake, and missile protection.

The fans are provided with a brake system to prevent overspeed from the design-basis tornado. The fan drive shaft is provided with a shield to protect it from tornado missiles.

The cooling tower structure is designed to withstand horizontal and vertical tornado missiles. The cooling fan motor is enclosed in a concrete cubicle designed to repel both types of missiles, and the cooling tower gear hub and shaft are protected by missile shields. Using the guidelines in Standard Review Plan (SRP) Section 3.5.1.4, the only design missile that can be elevated to the top of the towers is the 1-in.-diameter by 3-ft, 8-lb rebar. This missile could damage the cooling tower fan blades if the velocity is sufficiently high. However, analysis has shown that the probability of damaging fans in both cooling tower divisions by rebar tornado missiles is very low (see subsection 3.5.1.3 for more detail). Not withstanding this low probability, two spare sets of two RHR cooling tower fan blades and the necessary tools to install them are stored in the RHR complex building in a location protected from the tornado and tornado missiles. In the event that the cooling tower fan blades are damaged, the blades can be replaced and the fan restored to an operating condition. Plant safe shutdown will not be precluded in the event of tornado missile damage to all four of the RHR cooling tower fans, including assuming a loss of offsite power and a single independent failure. The plant organization estimates that it would take six hours or less to replace a set of cooling tower fan blades. If no fans are available for six hours, reservoir temperature is calculated rise to approximately 100 degrees F. All essential equipment cooled by the UHS is capable of performing its required safety functions at the higher reservoir temperature. One cooling tower fan can maintain hot standby and two cooling tower fans can achieve cold shutdown under these conditions.

In addition to missiles, miscellaneous debris can fall into the tower from the tornado. The debris would not damage the fan blades or other structural components of the towers. The debris would be removed while the blades are being replaced.

9.2.5.2.3 Emergency Diesel Generators

The EDGs are located as a part of the RHR complex. Two divisional pairs of two EDGs are provided; only one divisional pair is required for a safe plant shutdown. The divisional separation is maintained in the EDG system; each EDG division powers only equipment of that same division, and is cooled by that same division. In this manner, no postulated single failure can affect more than one division. A more detailed description of the system is provided in Subsection 8.3.1.1.8 and Subsections 9.5.4 through 9.5.7.

The EDG building is a Category I reinforced-concrete structure. An isolation wall is provided between each EDG for fire and missile protection. Independent fire detection and automatic fire- fighting systems are provided for each EDG.

Diesel generator cooling water is supplied from the RHR reservoirs with each diesel generator supplied by its own pump. Supply lines are also independent for each diesel generator. The diesel generator service water pumps start and stop automatically in conjunction with the diesel generators. The diesel generator service water supplies cooling water to the lube oil heat exchanger, the engine inlet air cooler heat exchanger, and the engine jacket coolant heat exchanger. Demineralized water with corrosion inhibitors is used for the closed loop engine jacket coolant. Makeup is provided by the demineralized water storage tank. The diesel generator service water flows through the tube side of the three-stage heat exchanger. The first stage cools the engine inlet air coolant system, then the second stage cools the lube oil, and finally the third stage cools the jacket coolant.

9.2.5.2.4 Makeup and Blowdown Systems

The makeup system is provided to replace evaporation and blowdown losses during normal shutdown cooling. The makeup system is not designed to withstand accidental and natural phenomena nor to function in the event of a single failure. The system is designed to fill and replenish the reservoir as required, to prevent flooding of the reservoirs, and to prevent siphon losses from the reservoirs in the event of a pipe break. The water makeup and the decanting system are shown in Figure 9.2-7.

Normal makeup water will be supplied by the plant GSW system. Normal water level in each division of the reservoir will be maintained automatically by regulating supply valves.

Five GSW (7700 gpm each) pumps are available to supply makeup water, using installed GSW system piping.

The blowdown system is provided to control the buildup of solids in the reservoir water during normal shutdown cooling. The piping is designed to prevent siphoning from the reservoirs in the case of a line break or other incident. Decanting pumps route blowdown from the reservoir to the main condenser circulating water reservoir. Details of blowdown from the main condenser circulating water reservoir are described in Subsection 10.4.5. Details of effluent monitoring are given in Section 11.4.

9.2.5.2.5 Pumps

Each division of the complex is provided with full-size vertical turbine pumps and a separate reinforced-concrete pump house. All pumps are mounted to ensure adequate net positive suction head (NPSH) under all anticipated operating modes. The pump vendor indicates that a minimum submergence at Elevation 554.6 ft will prevent vortexing of the inlet water to the suction bell of the emergency diesel generator service water (EDGSW) pumps, assuming rated flow at 100°F. The other service water pumps, the EESW and RHRSW pumps, require a minimum submergence at Elevation 554.9 ft and 555.7 ft, respectively. The pump motors and electrical switchgear are located at Elevation 590 ft (New York Mean Tide, 1935), ensuring that the system will continue to operate even if the reservoir is breached during the postulated site high-water event. The pumps and pump houses are Category I construction. Column bracing is provided as required to limit stress to allowable values. All pumps are connected to the essential bus for redundant power supply.

The RHRSW pumps are started and stopped manually from the main control room. Each pair of pumps is capable of delivering 9000 gpm* to the RHR heat exchangers and then to the cooling towers. In the flooding mode, the head of water is sufficient to fill 300,000 ft^3 of air space in the drywell in 1 week, at a rate of approximately 250 gpm. In the event of failure of all four of the 10,000-gpm RHR pumps, the RHRSW pumps in Division II will be capable of backup to the RHR pumps in the LPCI mode at the rate of 3250 gpm.

The 1600-gpm EESW pumps are started automatically on demand of the EECWS. One 800-gpm diesel generator service water pump is provided for each of the four diesel generators. These pumps start and stop automatically, corresponding to the operation of the respective EDG.

Since the reservoir is covered, no automatic backwash strainers are provided on the pump discharge.

NOTE: *RHRSW pump flow reduces below 9000 gpm with time due to the RHR reservoir evaporative and drift losses.

9.2.5.2.6 Piping

The piping system consists of two redundant loops. Each loop serves one division of the system.

Separate piping systems supply the service water from the pumps to the RHR and EECW heat exchangers. The EDG units also have individual cooling water supply lines. The RHRSW and EESW return lines are combined into a single header for each division and are routed to the reservoir via the cooling towers. Diesel generator cooling water return lines for each division also join these two common return headers to the cooling towers. The piping conforms to the following conditions:

a. Piping is Quality Group C and Category I (except for the makeup line and the overflow line). Thermal stress and seismic analysis calculations are made in accordance with the ASME Code Section III, Subsection ND for Class 3 components

b. Pressure indicators are provided on the discharge side of the RHRSW and EDGSW pumps, with PCVs for minimum-flow protection. The PCV air operators are supplied with interruptible air, and the valves fail closed on loss of air

c. Heat removal rate is controlled by a remote manually controlled globe valve on the RHR (for RHRSW), an automatic control valve on the EESW, and a manually controlled globe valve on the diesel generator service water system

d. The cross tie required for primary containment flooding is located in Division II between the discharge side of one pair of RHRSWS pumps and the discharge pipe of the shell side of an RHR system heat exchanger

   Keylock dual isolation valves are provided on the cross tie and are normally closed to prevent the service water from entering the RHR system loop. A testable check valve is provided to prevent the RHR system water from leaking into the service water loop. The cross tie is sized for a flow of 3250 gpm. The two isolation valves are motor operated from the main control room

e. Provision is made for process radiation monitoring on the service water discharge of each RHR heat exchanger division

f. The EESW pumps are provided with spring to close pressure regulating valves for minimum flow protection.

9.2.5.3 Safety Evaluation

The ultimate heat sink consists of a single highly reliable water source with fully redundant cooling towers, pumps, and conduits capable of providing sufficient cooling for 7 days to

permit safe shutdown and cooldown of the nuclear unit in the event of a design-basis accident. Procedures for ensuring continued cooling availability after 7 days are available. The RHR complex is designed for a single active or passive failure of any fluid system component without loss of safety function. The ultimate heat sink is capable of withstanding the effects of the most severe natural phenomenon associated with the site. Other applicable site-related events have been analyzed, including a single failure of man-made structural features. System failure analysis is summarized in Table 9.2-8.

9.2.5.3.1 Protection Against Natural Phenomena

The physical separation of RHRSWS Division I and Division II contributes to the reliability of system performance in the event of damage by natural phenomena (earthquake, tornado, storm, or flood).

9.2.5.3.1.1 Earthquakes

The RHRSWS is designed to meet Category I requirements. The method used in the seismic analysis of the RHR complex is similar to that of the reactor building as outlined in Section 3.7.

9.2.5.3.1.2 Tornadoes

A design-basis tornado and the missiles it might generate are described in Sections 3.3 and 3.5.

9.2.5.3.1.3 Freezing

The RHR building is designed to protect the reservoirs from direct exposure to winter weather. The floors of the RHR building cover a large portion of the reservoir surfaces and the remaining portion of the reservoirs is covered by floor gratings and tower baffles and is protected by high walls. In addition, 80 to 90 percent of the reservoir water is below the frost line. Drain lines provide passive freeze protection for the Mechanical Draft Cooling Tower (MDCT) spray distribution headers by allowing standing water to drain to the RHR Reservoir subsequent to MDCT operation.

The pump columns below the pump room are protected from freezing by an enclosure which is installed at the two open sides of the area below the pump room floor. The enclosed air volume temperature is locally monitored from the pump room.

During unit operation, the RHR complex reservoirs would receive heat from surveillance testing of plant systems such as the EDG, high pressure coolant injection (HPCI), and reactor core isolation cooling (RCIC) systems, and from other plant activities (such as operating the torus water management system). This heat would go directly to the RHR complex or the torus. Heat sent to the torus is removed by sending it to the RHR complex. During unit shutdowns, the RHR complex would receive reactor decay heat that is sufficient to prevent the reservoirs from freezing. The RHR complex also has cold weather bypasses around the cooling towers that can direct service water to the reservoirs instead of the RHR complex cooling towers. This would help retain the heat sent to the reservoirs.

The reservoir temperatures are monitored by readout in the control room and alarm at 43°F. The heat added to the reservoir during normal operation and testing is expected to be sufficient for this temperature to be exceeded during the winter. If needed, additional heat may be used to increase the reservoir temperatures by operating an EECWS or a temporary system installed for that purpose. Surveillance requirements of the reservoir temperature and required action in the event of low temperatures are contained in the Technical Specifications.

9.2.5.3.1.4 Floods

The reactor/auxiliary building and the RHR complex are designed to withstand the maximum postulated flood-water level and associated wave actions as described in Subsection 2.4.2.2 and Section 3.4.

The reservoir overflow is a nonsiphon floodproof port. Sidewalls are waterproofed to Elevation 590 ft (New York Mean Tide, 1935) and are above the Lake Erie stillwater level at the plant site. All active equipment that could be damaged by water (pump motors, switchgear, diesel generators) is located above the maximum flood-water level. The site flood considerations and plant protective structures are discussed in Subsections 2.4.2, 2.4.3, and 2.4.5.

9.2.5.3.1.5 Snow and Ice

The RHR complex roof structure is designed for the probable maximum snow and ice (including cooling tower drift) loads. The roof structure is capable of supporting a maximum loading of 70 lb/ft^2.

9.2.5.3.2 Protection Against Accident Phenomena

The RHR complex is designed to withstand the effects of the most severe natural phenomena associated with the Fermi plant site, as stated in Subsection 9.2.5.3.1 above. Other applicable site- related events such as river blockage, river diversions, reservoir depletion, or transportation accidents are not applicable to the design or postulated to occur. Flooding of the plant by surface runoff is not possible (Subsection 2.4.3.5). Transportation accidents are expected to have no effect on the complex because

a. The nearest main-line railroad and interstate highway are at least 3 miles from the plant site

b. The nearest ship channel is at least 4-1/2 miles from the plant site; the lake is of insufficient depth for commercial traffic in the plant vicinity; and the RHR complex is approximately 1100 ft inland from the lake shore

c. No significant aircraft operations occur in the plant vicinity.

A single failure of man-made structures such as the cooling tower or the pump house would not result in the loss of capability of the heat sink to accomplish its safety functions, because of the redundancy and separation of these components. A breach in the reservoir retaining wall above grade elevation would not compromise the reservoir's 7-day capacity since the reservoir capacity is contained below grade of 583 ft. With the reservoir divided into two

one-half-capacity reservoirs, only one side would be affected by a below-grade structural failure and then only to a limited degree since the damaged reservoir would leak only until ground-water elevation is reached. The 7-day capacity includes allowance for a below grade structural crack in both reservoir basins. Stability of ground-water level is discussed in Subsection 2.4.13.

The RHR complex structures, systems, and components are designed so that the minimum performance requirements of the complex can be met in case the postulated turbine missile strikes the complex.

Typical missiles that could be ejected from the EDGs will be small auxiliary items knocked loose from the engine exterior by blows from within. The maximum velocity of the missiles would be 40 fps, with a maximum mass of 5 lb each. The walls of the EDG rooms are designed to withstand such missiles and contain them within the room. Refer to Subsection 3.5.1 for further discussion of these postulated missiles.

9.2.5.3.3 System Reserve Capacity

The ultimate heat sink system was originally sized to provide sufficient cooling for 30 days following an accident without make-up water addition to the RHR reservoir. Regulatory Guide 1.27 states that a UHS capacity of less than 30 days may be acceptable if it can be demonstrated that replenishment can be effected to ensure the continuous capability of the sink to perform its safety functions, taking into account the availability of replenishment equipment and the limitations that may be imposed on freedom of movement following an accident. In order to provide additional head for the service water pumps, a 7-day reservoir replenishment was reviewed and was found to satisfy the R.G. 1.27 guidelines.

The Fermi 2 UHS design evolved long before the post-TMI improvements in Emergency Preparedness. Those improvements are reflected in the Detroit Edison Radiological Emergency Response Preparedness Plan. One of the objectives of this program is effective and timely implementation of emergency measures. Detroit Edison now has the resources of the Emergency Response Organization to rapidly identify the need for reservoir replenishment and to direct procurement of material and field implementation. This change significantly improves the ability to provide reservoir replenishment within 7 days as it relates to resolving problems associated with freedom of movement following an accident or occurrence of severe natural phenomena.

The 7-day make-up provision for the RHR reservoir is consistent with the 7-day make-up provisions allowed for replenishment of the diesel generator fuel supply. Therefore, this period of time is sufficient to recover from the effects of natural phenomena such as tornado, storm, earthquake or flood and restore site access for replenishment activities.

The reservoir replenishment procedure requires that reservoir make-up be established within 7 days following exceeding the Technical Specification reservoir level limit. Make-up will be provided by the normal make-up system or using RHR Complex fire hoses. If these systems are not available, temporary equipment will be used. The necessary pumps and hoses are commercially available from many sources and 7 days is sufficient time to procure and install the equipment. The procedure requires redundant replenishment equipment so that a single failure will not interrupt make-up. Necessary equipment requirements and vendors are listed to allow rapid procurement. Temporary pumps are located in lower

postulated post-accident dose rate areas so that access in available for monitoring and periodic refueling. The water source will be either Lake Erie, the Fermi 1 discharge canal, the circulating water reservoir, the on-site Quarry Lake or Swan Creek. Projected dose for hose installation is below allowable limits. The temperature and quality of make-up water is maintained to ensure that the service water systems and cooling towers perform as required. Siphon of the reservoir is prevented by ensuring that hoses are not placed into the reservoir water.

The reservoir will continue to store approximately 6 million gallons of water which was the previous 30-day supply. However, the level below that needed for 7 days of operation will be used to provide additional service water pump head margin. Therefore, if the level were to go below that required for 7 days of operation, a slow degradation in service water pump performance (discharge head) below design requirements would be possible.

The 7-day supply calculations utilize the Marley design and test data for cooling tower drift and evaporative water losses. In addition, the seven day supply also assumes a below grade structural crack in both reservoir basins and losses for EECW makeup using EESW. To maximize drift and evaporative losses, the reservoir basins are assumed to be cross-connected and both divisions of EDGs, RHR, EECW/EESW, and RHRSW cooling towers are assumed to be operating. The RHR heat exchanger was assumed to be clean (unfouled) to maximize heat loads on the ultimate heat sink. Constant historical worst-case meteorological data is used to compute evaporative water losses. The 7-day supply also assumes initial reservoir level at the technical specification limit of 580'-0" versus the normal operations level of between 582'-0" and 583'-0" which provides additional conservatism.

The RHR reservoirs are sized to provide for the evaporative and drift losses from the RHR cooling towers for 7 days following a design-basis recirculation line break, assuming a total loss of offsite power for the 7-day period. Evaporative losses are calculated as a function of cooling tower range using computer generated curves based on data supplied by the cooling tower manufacturer.

Drift losses are assumed to be 0.05 percent of flow. The required reservoir volume is based on the following tabulation.

Item Generating Cooling Water Loss	Reservoir Loss in 7 Days (gal)
Evaporative losses from decay and sensible heat, EECW, EDGs and pump energy	2,232,000
Drift loss	123,500
Leakage and EECW Makeup Losses	269,500
Spent fuel pool*	0
Total	2,625,000

* The cooling load for the spent fuel pool is not included since the normal burden is insignificant. When all or part of the core is unloaded, the cooling load in the spent fuel pool will increase, and the cooling load imposed by decay heat in the core will decrease proportionately.

The RHR reservoirs are nominally maintained between 583 ft and 582 ft elevation by an automatic makeup system. The Technical Specifications limit is established at 580 ft, or 5,980,000 gal. The 7-day water loss will result in a water level above Elevation 567'-6". This level is above the minimum submergence level required for the service water pumps as indicated in Subsection 9.2.5.2.5.

The heat load into the suppression pool as a function of time for the first 24 hours post LOCA was taken from the suppression pool peak temperature calculations described in Section 6.2.1.3.3. This suppression pool heat load includes decay heat (based on a pre-trip power level of 3499 MWt, which is 102% of 3430 MWt), sensible/blowdown energy, and RHR and core spray pump heat. The heat input to the suppression pool after 24 hours is from decay heat and pump heat. The decay heat after 24 hours is determined using the Standard Review Plan, Section 9.2.5, Branch Technical position ASB 9-2. The fractions of decay heat (as a fraction of operating power) versus time were converted to decay heat using a pre-trip reactor power level of 3499 MWt, which is 102% of 3430 Mwt.

The heat from the station auxiliary systems includes heat from the EECWS, the EDGs, and pump energy. The heat from the EECWS includes the emergency core cooling system (ECCS) pump cooling, control room air conditioning, air compressor cooling, ECCS room coolers, combustible gas control system (CGCS), and standby gas treatment system (SGTS) room coolers. The heat rejected by the EDGs is based on loads commensurate with the function required during each stage of the 7-day period. The pump energy created by the work input of the RHR, core spray, RHRSW, EECW, EESW, and EDGSW pumps has been considered.

The LOCA coincident with a loss of offsite power is the worst-case condition for reservoir water usage because:

a. The main condenser is unavailable for removal of any core decay energy or primary system energy

b. The EDGs run at the highest loads, resulting in highest heat rejection to the complex

c. The EECWS is operating (in lieu of RBCCWS)

The reservoir is not sized to supply the water required for flooding the core or primary containment to allow access to the core for accident recovery. Flooding of the primary containment is a long-term action, initiated many days or weeks after an accident, following an administrative decision. Such a decision would not be made until offsite power is available and the makeup water system is restored to service.

9.2.5.3.4 Multiple Water Sources

Two half-sized divisions of the reservoir are provided, each with a capacity of 3,410,000 gal of water at elevation 583 ft. Redundant valves remotely operable from the main control room are provided to permit access to the total water supply in the event of a mechanical failure of one division. Each division of the system has a separate piping system with adequate separation such that a failure of one will not induce failure of the other. The reservoirs are designed to withstand all applicable site-related natural and accidental phenomena, and there is no retaining "dam," as such, to fail. All of the water is stored below

site grade level and approximately 90 percent of the total water volume is stored below site ground-water level. The two divisions of the reservoir provide additional reliability over one reservoir. Therefore, it is concluded that there is an extremely low probability of losing the 7-day cooling capability of the reservoirs.

9.2.5.4 Tests and Inspections

Initial construction tests such as hydrostatic leak tests of the RHRSWS were conducted per applicable code requirements. Initial system flow checks, valve operability, instrumentation and control loop checks, and alarm setpoints were done in accordance with the Preoperational Test program as discussed in Chapter 14. After startup, a test will be run to observe the heat exchanger (including cooling tower) performance. Periodic tests are made to assess continuing pump performance and to demonstrate piping system integrity. The RHRSWS is periodically checked during its normal system function of cooling down the plant, such as during refueling or other outages. The EESWS will be tested in conjunction with the EECWS. The diesel generator service water system is tested in conjunction with the diesel generator testing. The reservoir and cooling tower are periodically inspected for macro and micro biological fouling and are treated, as required. Sampling and surveillance for control of dissolved solids and macro and micro biological fouling are performed by the Chemistry Section.

9.2.5.5 Instrumentation

System temperatures, pressures, and flows are monitored either locally or in the main control room. Pressure, temperature, and flow indicators are provided with local readouts. Pump and fan motors and controls incorporate standard protection devices.

Makeup for the RHRSW reservoir is automatic (except under accident conditions), controlled by level monitors in the two RHRSW reservoir divisions. These level monitors signal low-level alarms in the main control room.

9.2.6 Condensate Storage and Transfer System

9.2.6.1 Design Bases

The condensate storage and transfer system is designed to store and distribute condensate and demineralized water for use throughout the plant during normal and shutdown plant conditions. To provide for high plant availability, two full-capacity 600,000-gal storage tanks are provided, one designed as the normal storage tank and the other as the return tank. Demineralized water is stored in a 50,000-gal storage tank.

The condensate storage and return tanks are located near the turbine building and the auxiliary boiler house. They are arranged to permit gravity feed to condensate supply pumps and to the HPCI, RCIC, CRD, standby feedwater (SBFW), and core spray systems. During normal station operation, hotwell level is raised as necessary by vacuum dragging water to the hotwell from the CST or CRT. When the plant is shutdown, or when a greater flow is required, the normal, or if necessary the emergency, hotwell supply pumps will start and stop automatically depending on hotwell level. The condensate storage tank is designed to deliver its last 150,000 gal only to the HPCI or RCIC system (see Section 6.3.2.6).

A containment wall surrounds the tanks. Surveillance of surface conditions and radioactive content will be performed during plant operation.

The makeup demineralizer storage tank is located near the auxiliary boiler house and gravity feeds to the demineralized water transfer pumps and jockey pump. These pumps provide demineralized water to the service risers and the condensate storage tank.

Collection and distribution piping for the condensate and demineralized water is carbon and stainless steel. A cathodic protection system is supplied for the piping. The condensate tanks are fabricated of corrosion-resistant, high-strength aluminum alloy.

Piping to the HPCI and RCIC systems conforms to Quality Group B standards and is built to ASME Section III, Class 2, requirements. The balance of the condensate storage facilities conforms to Quality Group D standards and is built to ASME Section VIII and ANSI B31.1.0 code requirements. The tanks are designed to withstand a 100-mph wind when empty. They conform to USAS B96.1, "Welded Aluminum-Alloy Field-Erected Storage Tanks," code requirements. A minimum water temperature of 40°F is maintained in the insulated condensate storage tank by steam from the auxiliary boiler.

9.2.6.2 System Description

The condensate storage and transfer system, as shown in Figure 9.2-10, consists principally of two large storage tanks and three pumps, with associated receiving and distribution lines, one demineralized water tank, and three pumps with associated receiving and distribution lines. Component design parameters are given in Table 9.2-9. The condensate return and the condensate storage tanks are 600,000-gal aluminum tanks with open vents. The condensate storage tank is insulated and has sufficient heating capacity to maintain a water temperature of at least 40°F, which is the design limit for thermal shock to the RPV nozzles. Both tanks are located inside a containment wall near the turbine building. The condensate storage tank receives demineralized water from the demineralized water makeup system and may also receive low-conductivity water from the condensate return tank. There is also a normally closed balance line connecting these two tanks to allow gravity transfer from one tank to the other above the 150,000-gal limiting level of the storage tank.

Containment for any condensate loss that might be experienced is provided by a containment wall in the immediate area around the condensate storage tanks, as shown in Figure 9.2-11. All valves associated with either condensate storage tank are located in valve pits at the base of the tank. Any leakage into either valve pit is automatically pumped through a 4-in. drain line to the waste collector tank in the radwaste building.

A single wall, 3 ft high above the normal grade level, encloses both condensate storage tanks. This forms a contained area approximately 109 ft wide by 232 ft long. The contained area is also excavated 3 ft below grade level. The enclosed area has been sealed with a Hypalon liner (waterproof barrier) to contain all spillage of contaminated water from the condensate system and prevent it from entering the soil in the condensate storage tank and condensate return tank diked area.

Relief valves are installed in the condensate return tank valve pit to prevent inlet piping overpressurization. A 30 gpm relief valve discharges directly to the CRT valve pit. Relief

valves with 600 and 5000 gpm relief capacities discharge into the lined dike area surrounding the CRT and CST for subsequent cleanup and processing.

Direct access to Lake Erie by lost condensate seeping into the ground is prevented by the clay fill seal beneath the shore barrier. Initial movement of any seepage would be downward to mix and dilute with the ground water from the dolomite aquifer. Thereafter the diluted material would move into and through the aquifer at the same rate of flow and direction of movement as the transient ground water. The direction of movement would be to the east, at a rate of 0.24 ft per day or less. In essence, this would be the same sequence of events as that documented for the loss of all radwaste water to the aquifer in Edison's response to AEC Question 10.2 in the Fermi 2 PSAR, except that the condensate water would be orders of magnitude less radioactive.

The storage tanks are provided with horizontal slots (weirs) in the sides of the tanks with ducting to channel overflow down the side of the tank and eliminate spray. An alarm system is provided that alarms in the control room to indicate that a tank is in the process of overflowing. This alarm system is independent of any other instrumentation or alarms associated with the condensate storage tank or condensate return tank.

Recycle streams are treated and monitored prior to transfer to the condensate return tank. If water in the condensate return tank requires further treatment, it can be transferred to the radwaste system by gravity for processing, or it can be sent through the polishing demineralizer via the hotwell and condensate system.

It is possible for the following water sources to be transferred directly to the condensate tanks:

a. Return from radwaste system
b. Return from CRDs
c. HPCI pump test return
d. RCIC pump test return
e. SBFW pump test return

The inlet valves on the condensate storage tank and condensate return tank are provided with an interlock to prevent the two valves from being simultaneously closed. This interlock provides protection against overpressurization of the tank inlet piping by the CRD, HPCI, RCIC, or SBFW pump.

Treated recycled condensate water is normally routed to the condensate return tank. Typical sources are:

a. Reactor well drain
b. Return from drywell seal rupture
c. Spent fuel storage pool drain
d. Main condenser high-level relief

The demineralized water makeup system supplies only the heated condensate storage tank, the auxiliary boiler, and the demineralized water distribution system.

The condensate storage tank is sized for the following duty:

Fuel pool and reactor well	410,000 gal
Reserve for the HPCI and RCIC pumps (see Section 6.3.2.6)	150,000 gal
Additional reserve	40,000 gal
Total	600,000 gal

The condensate storage tank provides a source of water for the HPCI, RCIC, SBFW, core spray, and CRD pumps. Either condensate tank can supply the low-pressure turbine hood spray pump through a common header.

The condensate return tank is the same size as the condensate storage tank in order to provide operational flexibility so that one tank may be substituted for another for certain functions as described herein. The condensate return tank has sufficient capacity for:

Fuel pool and reactor well	410,000 gal
Additional reserve	190,000 gal
Total	600,000 gal

The primary function, however, of the return tank is to receive condensate from the plant and to store any temporary excess condensate. Excess water from the condenser hotwell is normally relieved to this tank. During normal station operation, hotwell level is raised as necessary by vacuum dragging water to the hotwell from the CST or CRT. When the plant is shut down, or when a greater flow is required, the normal, or if necessary the emergency, hotwell supply pumps will start and stop automatically depending on hotwell level. Makeup to the hotwell is normally supplied by the hotwell supply pump drawing condensate from the condensate return tank. During normal station operation, the hotwell supply pump starts and stops automatically, depending on the hotwell level of the condenser.

Condensate for distribution to other plant areas is supplied from the condensate pump discharge header, downstream of the condensate polishing demineralizers, via a pressure-reducing valve. Condensate at 100 psig feeds the condensate storage system distribution header.

The condensate storage and transfer system distribution pumps are housed in the turbine building and are supplied from the same header, feeding from either condensate tank. The three pumps required for distribution are described as follows:

a. The condensate storage jockey pump (one 100-gpm pump) is used to maintain condensate pressure during startup/ shutdown and supply water to the condensate distribution header whenever the supply from the condensate system is not available. The condensate storage jockey pump has minimum flow protection to provide sufficient pump cooling in the event of low or zero flow in the condensate distribution system. The condensate distribution system supplies the following:

    1. Radwaste building

    2. The turbine building backwash tank

3. A reactor building supply header. This portion of the reactor building supply header is intended for preoperational flushing and fill, and not for makeup.

   It provides RHR system flushing and maintains the RHR keep fill system. As such, the various distribution branches to the following items are normally valved off:

   (a) RBCCW makeup tank

   (b) EECW makeup tank (two).

   The reactor building supply header supplies makeup to the following:

   (a) Cleanup phase separators

   (b) Reactor water cleanup filter-demineralizer

   (c) Fuel pool skimmer surge tank

b. The normal hotwell supply pump is a 600-gpm pump discharging to the main condenser hotwell level control station. The normal hotwell supply pump can be used to supply water to the condensate distribution system when either of the other two sources (jockey pump or condensate system supply) are insufficient or out of service. The normal hotwell supply pump has a minimum flow valve which provides flow protection for a limited time. Should plant outage activities require prolonged operation of the pump with minimum flow, a supplemental, temporary flow path is established to provide adequate minimum flow protection.

c. The emergency hotwell supply pump is a 2000-gpm pump with an independent distribution line to the main condenser hotwell and branch to the TBCCWS for flushing. This emergency pump can also discharge into the condensate return line to the storage tanks. In this way, it can be used to transfer condensate water from one tank to the other.

The makeup demineralized water tank supplies demineralized water for the following:

a. Condensate storage tank

b. Auxiliary boiler deaerator makeup

c. TBCCWS makeup

d. RBCCWS makeup

e. EECWS makeup

f. Standby liquid control tank in reactor building

g. Service risers in turbine building, reactor building, auxiliary building, and radwaste building

h. Plant instrument backflush

i. Cask washdown in reactor building

j. Health Physics and chemistry laboratories

k. Deleted

l. Emergency showers in the radwaste building

m. Core spray system charging, and for maintaining the keep fill system.

The demineralized water is distributed to the condensate storage tank and throughout the plant by two 100-gpm transfer pumps and by a 20-gpm jockey pump. Normally the jockey transfer pump operates continuously with bypass flow unloading on low water demand. The jockey pump thereby provides a minimum hydrostatic pressure of 30 psi throughout the system. On increasing flow demand (system pressure decreases), one 100-gpm transfer pump starts automatically. If flow demand is still not satisfied and header pressure continues to drop, the second transfer pump starts.

9.2.6.3 Safety Evaluation

Adequate protection from environmental conditions is provided by designing the tanks to withstand a 100-mph wind when empty. The tanks are designed for -10°F to 95°F ambient temperatures.

Condensate water within the plant is separated from the demineralized water system and confined to areas of limited access. Health Physics surveillance is maintained on this equipment. An exclusion fence surrounds the condensate storage and condensate return tanks. Because of these conditions, personnel injury from radiation is extremely unlikely. Construction materials are corrosion resistant and should serve without failure during the 40-year life of the plant. The tank is constructed of aluminum alloy, and the HPCI and RCIC pump suction lines are stainless steel. Carbon steel piping is provided with cathodic protection.

The design of the demineralized water makeup system precludes radioactive contamination of the storage tank. The transfer line between the uncontaminated demineralized water storage tank and the potentially contaminated condensate storage tank terminates in the condensate storage tank above the normal operating water level. This feature, along with check valves at the transfer pump discharge lines and the normally closed motor-operated isolation valve, provides reasonable assurance against inadvertently contaminating the demineralized water makeup system. Contamination of the potable water system is prevented by an open break to the potable water holding tank.

Active functioning of the condensate storage and transfer system is not required during a reactor shutdown. Suction for RCIC (or HPCI) can be manually transferred from the condensate tank to the Category I suppression pool. In addition, on low level in the tank, suction is automatically transferred. Therefore, it is not necessary to install redundant power sources and redundant water sources throughout the condensate system. Nevertheless, the system is designed with certain redundancies (e.g., cross- connections, standby pumps) to reduce, as far as practical, the probability of causing a plant shutdown because of some failure in the condensate storage and transfer system.

Normally, water from the plant is returned to the condensate return tank after treatment and analysis. The water in the return tank will be analyzed to ensure that it is of sufficiently high quality. In the event that water has excessive radioactivity or conductivity readings, it will be transferred to either the polishing demineralizer via the hotwell or to radwaste for further treatment. These operating procedures will ensure a source of high-quality water for station operation.

Leakage is controlled by utilizing welded construction for the storage tanks, and as much as is practicable for the piping. Each penetration on the tank is supplied with an isolation valve. Levels in the two storage tanks are recorded and alarmed in the main control room.

Accidental release of liquids in the condensate storage tank would not result in concentrations in Lake Erie exceeding the limits of 10 CFR 20. Any accidental release that is not retained by the lined containment around the storage tanks will infiltrate the site fill; however, horizontal permeability of the soil will provide sufficient holdup to attain the required decontamination factor by radioactive decay before entering Lake Erie. No credit is taken for filtration or ion exchange through the soil.

9.2.6.4 Tests and Inspections

Initial construction tests such as hydrostatic leak tests were conducted per applicable code requirements for the condensate storage system. Initial system flow tests, valve operability, instrumentation and control loop checks, and alarm setpoint checks were performed in accordance with the Preoperational Test program as discussed in Chapter 14.

Periodic tests are conducted to confirm pump performance and operation of automatic controls. Inspection for system leakage is coincident with pump testing and routine monitoring activities.

9.2.6.5 Instrumentation

Pump motors are equipped with standard controls and protective devices and are controlled and monitored from the main control room. Each pump's flow is indicated locally. The level and temperature of the condensate storage tanks and the demineralized water storage tank are continuously recorded in the main control room. High-and low-water-level and overflow alarms are provided. Level and temperature indicators are provided locally at the tanks.

The storage tank temperature can be manually controlled from the main control room by continuously circulating stored condensate through the condenser hotwell.

9.2.7 Cooling System for Turbine Auxiliaries

9.2.7.1 Design Bases

The TBCCWS is designed to remove heat from auxiliary equipment housed in the turbine building. The TBCCWS is cooled by the GSW system and makeup is supplied by the demineralized water system. The TBCCWS is designed to operate at a lower pressure than the GSW system to prevent leakage to the environs.

The TBCCWS is nonseismic and is constructed in compliance with standards for Quality Group D components. This criterion is met by designing the system to ASME Section VIII and ANSI B31.1.0 code requirements.

9.2.7.2 System Description

The TBCCWS, as shown in Figure 9.2-12, consists of two 100 percent-capacity TBCCWS heat exchangers and three 50 percent-capacity pumps. The TBCCWS component design parameters are listed in Table 9.2-10. During normal operation, one heat exchanger and two pumps operate to remove all the equipment heat loads. The third pump is in standby and is designed to be started manually on low TBCCWS pressure. The TBCCW supply header temperature is maintained by a temperature control valve modulating the GSW flow through the TBCCW heat exchanger. System pressure is controlled by a differential PCV located in the bypass line between the suction and discharge headers of the TBCCW pumps. Makeup to the system as well as system expansion and contraction due to load changes are provided by a makeup tank. Makeup water is automatically supplied via a tank level control valve from the demineralized makeup water system. A pressure-regulating valve maintains nitrogen overpressure in the tank. Nitrogen is provided to prevent the introduction of oxygen into the system, thereby retarding corrosion of the closed cooling water system, and to maintain a positive suction head on the pumps.

Turbine auxiliaries impose a maximum cooling load on the TBCCW heat exchanger of approximately 45 x 10^6 Btu/hr during normal operation. This requires approximately 9000 gpm of GSW, assuming a maximum GSW temperature of 85°F. Circulation on the TBCCW side of the heat exchanger is approximately 6000 gpm; design temperatures are 110°F in and 95°F out. Normal operating temperature is 80°F out. The operating temperature range is 75°F to 88°F as measured at the TBCCW pump suction. The TBCCW header temperature control high-alarm setpoint is 88°F and the low-alarm setpoint is 75°F.

The GSW is not used directly because the relatively high impurity level in this system might result in fouling of equipment heat transfer surfaces. Furthermore, the intermediary loop between potentially contaminated turbine auxiliaries and the GSW system provides additional protection against radioactive water leakage into the environment. The following equipment is cooled by the TBCCWS:

a. First floor:

    1. Station air compressors with associated coolers
    2. Heater feed pump motors
    3. Heater drain pump motors
    4. Condenser mechanical vacuum pumps
    5. Oil coolers for each reactor feed pump and turbine drive
    6. Coolers for air conditioning unit serving the Health Physics laboratory and radwaste control room
    7. Cooler for the chemical sampler

8. Office building air conditioning unit.

b. Second floor:

1. Generator bus duct cooler heat exchanger
2. Hydrogen seal oil coolers
3. Stator winding coolers
4. Excitation equipment area air coolers
5. Offgas system aftercoolers.

c. Third floor:

1. Ring water coolers for the offgas vacuum pumps
2. Adsorber room air conditioner cooler for the offgas system
3. Unitized actuators cooling.
4. Offgas Chiller Refrigeration Unit N6200D010 Condenser.

9.2.7.3 Safety Evaluation

Turbine auxiliaries housed in the turbine building are not considered essential to safe reactor shutdown. An alternative power source for TBCCW pumps and controls is not required. Because a reactor shutdown can be safely executed without depending upon turbine auxiliaries, no alternative water supply to the TBCCWS is required. Redundancy is therefore properly limited to two normally operating pumps and one standby pump to facilitate periodic tests and maintenance and to ensure plant availability.

The design of the TBCCWS avoids direct application of GSW to cooling components and heat exchangers. This excludes impurities from the turbine auxiliaries and service piping and reduces the chances of long-term degradation from fouling or corrosion. Additionally, corrosion inhibitors are added for pH and oxygen control. Alternatively, the TBCCW system may be maintained with pure demineralized water only with no chemicals added.

The TBCCWS operates at a lower pressure than does the GSW system to protect against leakage into the GSW system. Leakage from the TBCCWS into the building is detected by a low-level alarm in the makeup tank and by an alarm on the makeup level control valve stem position. Excessive opening of the makeup valve is indicative of a substantial system leak. Inleakage of GSW will be indicated by a high-level alarm in the makeup tank.

9.2.7.4 Tests and Inspections

Initial construction tests such as hydrostatic leak tests were conducted per applicable code requirements for the TBCCWS. Initial system flow distribution, valve operability, instrumentation and control loop checks, and alarm setpoints were done in accordance with the Preoperational Test program as discussed in Chapter 14. Heat exchanger performance is observed during plant operation using actual system heat loads.

Instruments and controls are inspected periodically. No periodic leak tests are planned since the system is continuously operating and pressurized. Periodic visual inspection of the system will detect minor leakages, such as those from valve stems, flanges, and instrument tubing.

9.2.7.5 Instrumentation

Pump motors are equipped with standard controls and protective devices, and are monitored from the main control room. Readouts to observe pressure and inlet and outlet temperatures in the TBCCW are provided in the main control room. Interlocks are provided on the TBCCW pumps to prevent their starting on low makeup tank level and/or low suction pressure. The temperature of the TBCCWS during operation is controlled by modulating the discharge flow of GSW through the TBCCW heat exchangers. Closed loop pressures are regulated by pump unloading (bypass) valves.

High- and low-level alarms are provided on the makeup tank as are alarms for low system pressure, to alert the operator of system leakage or loss of nitrogen capping.

9.2.8 Torus Water Management System

9.2.8.1 Design Bases

The torus water management system (TWMS) is designed to provide thermal mixing of the torus water, torus water volume inventory control, torus water quality maintenance, and to drain and fill the torus to facilitate inside torus recoating, inspections, and repair work. The TWMS design flow, makeup, and discharge rates are based on draining or filling the 1,000,000 gal of torus water in 48 hr and circulating the torus water at a rate of 500 gpm (maximum). Figure 9.2-13 is the system schematic.

The TWMS pumps (refer to Table 9.2-11) are located in the subbasement of the reactor building in the HPCI room. The interconnected piping required to perform the circulating, cleaning, draining, and filling functions of the TWMS is located in the reactor, turbine, and auxiliary buildings.

The TWMS primary containment penetrations and the associated isolation valves are classified as ASME Section III, Class 2, and designated as Category I. The balance of the TWMS is classified as ASME Section VIII for pressure vessels and ANSI B31.1.0 for piping, and is designated as nonseismic.

9.2.8.2 System Description

The TWMS pumps take suction from two torus connections placed at a 180° angle around the torus from each other. Water is similarly returned to the torus using two different torus connections also at a 180° angle from each other. These torus connections were selected to maximize thermal mixing efficiency to the extent practical.

The TWMS normally maintains torus water quality. The TWMS pumps will transfer torus water to the condensate system at the main condenser continuously or intermittently, as selected by the operator. Clean condensate from the condensate reject line to the condensate storage tank provides return water to the torus. The rate at which torus water is transferred to

the condensate system for cleaning and subsequently returned to the torus is regulated by the control room operator to maintain the proper torus water level and desired quality. One facet of the primary containment monitoring system provides the operator with a wide-range torus water level indication and one aspect of the TWMS provides a narrow-range torus level indication.

As required for recoating, inspections, and repair work, the TWMS is used to drain and fill the torus. Torus draining is accomplished by using the TWMS pumps to transfer torus water directly to the main turbine condenser for storage. Torus filling is accomplished using a condensate system pump to transfer water back to the torus through the TWMS return piping. During the filling operation, one of the eight polisher/demineralizers may be placed into service to clean the water returned to the torus.

9.2.8.3 Safety Evaluation

The TWMS is not required for reactor shutdown or accident mitigation and as such is not a safety-related system. However, the availability of the TWMS increases the reliability and availability of the plant. The reliable operation of the TWMS is ensured through the redundancy of the TWMS pumps (two at 250 gpm each) and two 50 percent-capacity suction and discharge lines with required isolation valves.

To ensure that the TWMS will not impair the safety function of the torus, the TWMS primary containment isolation valves automatically close. These valves trip in response to selected primary containment isolation system isolation signals (refer to Table 6.2-2) and to the high-high level alarm of the drywell floor drains or the torus room floor drain sump. The power supplies for containment isolation valves are arranged so that loss of one supply cannot prevent automatic isolation of a TWMS suction or return line when required. Torus water level and temperature limits and alarms are monitored and provided by the primary containment monitoring system, which is designated as a safety-related system.

Administrative controls and other constraints are provided to ensure that the suppression pool is not drained by the TWMS when the need for the ECCSs could be required. The limiting conditions for the draining of the suppression pool are specified in the Technical Specifications. Operational procedures for the TWMS include detailed information on the draining of the suppression pool. The TWMS pumps will automatically trip, preventing a torus water-level decrease, when the torus water level low-low alarm setpoint is reached at an elevation of 556.83 ft (2 in. below normal level) except when in the torus drain mode of TWMS.

Because the TWMS is considered a moderate energy system, flooding and spraying effects from postulated cracks in the system piping have been evaluated. Flooding and spraying effects have been determined to be enveloped by the limiting RHR pump discharge line crack (refer to Subsection 3.6.2.3.4.1.2). System overpressure protection is maintained by pump discharge relief valves.

9.2.8.4 Tests and Inspections

Initial system flow checks, valve operability, and instrumentation and control loop checks were performed in accordance with the test program as discussed in Chapter 14. A flow

meter in the TWMS transfer line to the condensate system is provided to indicate the torus water removal rate and establish TWMS pump operability. Conductivity monitors on the discharge side of the polisher/ demineralizers provide continuous monitoring of the water returned to the torus. Minor leakages such as those from valve stems, flanges, and instrument tubing are detected through visual inspection.

9.2.8.5 Instrumentation

The TWMS is operated from the control room. Flow meter indication of the torus water flow to the condensate system is provided in the main control room. Position indication for the TWMS primary containment isolation valves and the control valves on the transfer and return lines are also provided in the main control room. Torus water management system controls and instrumentation are augmented by the containment monitoring functions of the primary containment monitoring system and the water quality monitoring devices in the turbine building.

9.2.9 Supplemental Cooling Chilled Water System

9.2.9.1 Design Basis

The SCCW system is a closed cooling water system which during normal plant operation will provide chilled water that will be used to lower the temperature of the cooling water supply to EECW that is normally cooled by RBCCW. The SCCW system is designed to remove 100 percent of the normal heat produced by the EECW system during normal operation. The SCCW system transfers the heat it has removed to the GSW system via mechanical chillers.

The SCCW system is designed to operate at a higher pressure than the RBCCW system to provide protection against potential outleakage of radioactive contaminants from the RBCCW system.

The SCCW system is classified as a non-nuclear system and is constructed in compliance with standards for Quality Group D components. This criterion is met by designing the system to ASME Section VIII and ANSI B31.1.0 code requirements. The system is nonseismic.

9.2.9.2 System Description

The SCCW system, as shown on Figure 9.2-14, is designed to remove heat from the RBCCW system headers that supply cooling water to the EECW headers and from the fan coil that cools the SCCW chiller area (turbine building basement).

The SCCW system consists of three 50 percent-capacity chillers (two normally operating), only one may be operating during low load conditions, three 50 percent-capacity chilled water pumps (two normally operating), a chilled water expansion tank and associated valves and controls. The system is designed to be started when RBCCW temperature first exceeds its nominal control temperature and left in operation until the RBCCW temperature can be maintained at or below its nominal control temperature. Table 9.2-12 provides SCCW system design parameters.

After a manual startup from the local control panel, system operation does not require operator intervention unless it is desired to rotate equipment that is operating or a trouble signal is annunciated in the control room. In the event a chiller were to trip, the standby chiller would automatically start. A trip of one of the operating chilled water pumps would automatically start the standby pump. The chiller trip logic is tied to the chilled water pump operation to ensure an adequate number of pumps are operating.

An expansion tank is provided to accommodate changes in water volume as the temperature of the chilled water varies. This tank will also maintain a constant chilled water pump suction pressure.

The demineralized water system will provide system make-up water.

9.2.9.3 Safety Evaluation

The SCCW system is not required for the safe shutdown of the reactor or for accident mitigation. Therefore, the SCCW system is not designed for a single active or passive failure as required of a safety-related system, but sufficient redundancy and automatic protective features are provided to ensure efficient system operation and availability. Since the SCCW system is not an engineered safety feature (ESF), it is not powered by an essential power bus.

The SCCW system is involved in normal plant operation. During periods when RBCCW temperature is above its nominal control point, it may be used to cool the cooling water supply to EECW that is normally cooled by RBCCW. If the SCCW system trips off or is removed from service, RBCCW and/or EECW, as required, will perform the cooling function. The RBCCW system is nonsafety and its pumps are not powered by an essential bus. If the RBCCW cooling water temperature is not available, the EECW and EESW systems can be used to provide the required cooling of the drywell and other equipment. The EECW and EESW systems are powered off essential buses. A failure of the SCCW system will not prevent a safe shutdown of the reactor.

The possibility of radioactive contamination of the SCCW system is reduced by using plate heat exchangers between the RBCCW supplemental cooling and SCCW systems. The SCCW system operates at a higher pressure than the RBCCW supplemental cooling system and therefore any leakage would be from the SCCW system into the RBCCW supplemental cooling system. The SCCW system does not contain a radiation monitor. Due to the design of a plate type heat exchanger, any leakage is likely to be to the ambient (auxiliary building) rather than to the SCCW system. The SCCW system is manually sampled to detect any developing problems. Should the barrier between RBCCW and SCCW fail, there is no potential release path to the environment. The SCCW system interfaces with the GSW system via mechanical chillers. The closed refrigerant system which is between the SCCW and GSW systems has a relief valve which exits the turbine building. However, that relief valve setpoint is much higher than any pressure that can be developed in the SCCW system and therefore does not constitute a release path.

9.2 WATER SYSTEMS

REFERENCES

1. AEC letter Docket 50-341, from Voss A. Moore, Assistant Director for Boiling Water Reactors/Directorate of Licensing, to The Detroit Edison Company, dated November 30, 1973.

2. Applicant's Responses to the July 9, 1973 AEC letter with 17 questions regarding the RHRSW Pond (Complex) Design, Docket 50-341, dated August 10, 1973.

TABLE 9.2-1 GENERAL SERVICE WATER SYSTEM PUMP DATA

General Service Water Pump

Number supplied	Five
Type	Vertical, wet-pit, turbine
Fluid	Chlorinated lake water
Capacity, gpm	7700
Total head, ft	
(Tested two stage pump)	241 to 270
Motor	
Type	Vertical, dripproof, induction
Horsepower	900
Speed	1779
Voltage/frequency/phase	4000/60/3

Circulating Water Makeup Pump

Number supplied	Two
Type	Vertical, wet-pit, turbine
Fluid	Chlorinated lake water
Capacity, gpm	15,000
Total head, ft	32
Motor	
Type	Vertical, dripproof, induction
Horsepower	200
Speed	880
Voltage/frequency/phase	460/60/3

TABLE 9.2-2 REACTOR BUILDING CLOSED COOLING WATER SYSTEM COMPONENT PARAMETERS

RBCCW Pumps

Number supplied	Three
Type	Horizontal, single-stage centrifugal
Fluid	Demineralized water
Capacity, gpm	4000
Total head, ft	167
Motor	
Type	Horizontal 445 TS Frame
Horsepower	200
Speed, rpm	1770
Voltage/frequency/phase	460/60/3

RBCCW Supplemental Cooling Pumps

Number supplied	Four (Two per EECW Division)
Type	Horizontal, single-stage centrifugal
Fluid	Demineralized water
Capacity, gpm	1557 for EECW Division I 1715 for EECW Division II
Total head, ft	260
Motor	
Type	Horizontal 445 TS Frame
Horsepower, HP	150
Speed, rpm	1785
Voltage/frequency/phase	460 V/60 Hz/3

RBCCW Heat Exchangers

Number supplied	Two
Type	Shell and tube, single pass
Heat transfer duty, Btu/hr	67.8×10^6
Heat transfer area, ft^2	12,780
Design code	ASME Section VIII, TEMA Class C
Shell	
Fluid	Demineralized water
Design pressure, psig	150
Design temperature, °F	120
Flow, gpm	8000
Inlet temperature, °F	112
Outlet temperature, °F	95
Material	Carbon steel A-285-C
Tube	
Fluid	Lake water

TABLE 9.2-2 REACTOR BUILDING CLOSED COOLING WATER SYSTEM COMPONENT PARAMETERS

Design pressure, psig	175
Design temperature, °F	120
Flow, gpm	10,000
Inlet temperature, °F	85
Outlet temperature, °F	99
Material	304 stainless steel

RBCCW Supplemental Heat Exchangers

Number supplied	Two 1 @ 100% capacity for EECW Division I 1 @ 100% capacity for EECW Division II
Type	Plate heat exchanger
Heat transfer duty, Btu/hr	10×10^6 for EECW Division I 11.5×10^6 for EECW Division II
Heat Transfer Area, ft^2	1229 for EECW Division I 1272 for EECW Division II
Design code	ASME Section VIII
Material	Plates: 304 SS Nozzles: 316 SS
Cold Side	
Fluid	SCCW, demineralized water
Design pressure, psig	150
Design temperature,°F	150
Flow, gpm	1100 for EECW Division I 1300 for EECW Division II
Inlet temperature,°F	60.2
Outlet temperature,°F	78.4 for EECW Division I 77.9 for EECW Division II
Hot Side	
Fluid	RBCCW Supplemental, demineralized water
Design pressure, psig	150
Design temperature,°F	150
Flow, gpm	1557 for EECW Division I 1715 for EECW Division II
Inlet temperature,°F	82.9 for EECW Division I 83.4 for EECW Division II
Outlet temperature,°F	70

Note: The heat duties are the maximum expected values and will not occur simultaneously.

RBCCW Makeup Tank

TABLE 9.2-2 REACTOR BUILDING CLOSED COOLING WATER SYSTEM COMPONENT PARAMETERS

Number provided	One
Type	Horizontal, elliptical dished heads
Design pressure, psig	100
Design temperature, °F	120
Operating pressure, psig	45
Internal volume, gal	600
Liquid volume, gal	300
Pressurizing gas	Nitrogen
Material	Carbon steel ASTM-A515 GR70

TABLE 9.2-3 EMERGENCY EQUIPMENT COOLING WATER SYSTEM COMPONENT DESIGN PARAMETERS

EECW Pumps	
Number supplied	Two
Type	Horizontal, centrifugal
Fluid	Demineralized water
Capacity, gpm	1775
Total head, ft	167
Motor	
Type	Induction, dripproof
Horsepower	100
Speed, rpm	1785
Voltage/frequency/phase	460/60/3
EECW Heat Exchanger	
Number supplied	Four (Two per EECW Division)
Type	Single Pass, Plate and Frame
Heat transfer duty, Btu/hr	13.6×10^6
Heat transfer area, ft^2	4214.1 ft^2
Design code	ASME Section III, Class 3,
Hot Side	
Fluid	Demineralized water
Design pressure, psig	150
Design temperature, °F	150
Flow, gpm	1700
Inlet temperature, °F	111.1
Outlet temperature, °F	95
Material	T-316 Stainless Steel
Cold Side	
Fluid	RHR service water
Design pressure, psig	175
Design temperature, °F	150
Flow, gpm	1450
Inlet temperature, °F	89

TABLE 9.2-3 EMERGENCY EQUIPMENT COOLING WATER SYSTEM COMPONENT DESIGN PARAMETERS

Outlet temperature, °F	107.9
Material	T-316 Stainless Steel

EECW Makeup Tank

Number provided	Two
Type	Horizontal, elliptical dished heads
Design pressure, psig	100
Design temperature, °F	140
Operating pressure, psig	36
Internal volume, gal	600
Liquid volume, gal	300
Pressurizing gas	Nitrogen
Material	Carbon steel SA-515 Grade 70

EECW Makeup Pump

Number provided	Two
Type	Motor driven horizontal centrifugal, vert. disch
Power	480 Vac/3Ph, 3.0 HP
Design pressure	60 ft TDH
Design flow	20 gpm
Material	Stainless steel

EECW Makeup Pressure Regulator Valve

Number provided	Two
Type	Discharge regulator, self-actuated
Size	1-1/2 in.
Setpoint (discharge)	36 psig
Maximum design flow	25 gpm (Div I): 15 gpm (Div II)
Minimum design flow*	10.7 gpm (Div I): 9.8 gpm (Div II)`
Design makeup flow to EECW Head Tank*	3.2 gpm (Div I, or Div II)

* Coincident flows

TABLE 9.2-4 REACTOR BUILDING CLOSED COOLING WATER AND EMERGENCY EQUIPMENT COOLING WATER SYSTEMS FAILURE ANALYSIS

Component	Failure Mode	Consequences on Safety
RBCCW pumps	Loss of pumping due to loss of offsite power	Loss of offsite power automatically starts both EECW pumps and initiates isolation of the nonessential loads.
RBCCW pumps	Trip on low suction pressure or low makeup tank water level	Each EECW pump will start on low differential pressure between its respective supply and return headers. Isolation of nonessential loads is also initiated. (a)
RBCCW piping	Pipe rupture in the RBCCWS	Each EECW pump will start on low differential pressure between its respective supply and return headers. Isolation of nonessential loads is also initiated. (b)
EECW pump	Fails to start due to failure of one set of diesel generators	Redundant full-capacity EECW loop is provided, powered off the second emergency bus.
EECW piping	Piping leak or rupture in the EECWS	Automatic makeup valve will open to maintain tank level. If leak exceeds makeup capacity, the low tank level or low suction pressure will be alarmed. Redundant full-capacity EECW loops are provided that are isolable from each other.
EECW Makeup Tank	Loss of nonsafety-related demineralized water and nitrogen inerting systems	Safety related EESW makeup water to the makeup tank restored by automatic start of EECW makeup pump if makeup tank has low pressure or level and makeup tank isolation valve is open, and normal pump suction pressure is achieved. Nitrogen pressure in the makeup tank will be automatically maintained by the backup nitrogen supply for Division I until the makeup tank Nitrogen leaks off and the Makeup tank is filled and Pressurized with EESW water via the makeup pump

TABLE 9.2-4 REACTOR BUILDING CLOSED COOLING WATER AND EMERGENCY EQUIPMENT COOLING WATER SYSTEMS FAILURE ANALYSIS

Component	Failure Mode	Consequences on Safety
Isolation valve between EECW and RBCCW [P4400F601A(B), P4400F603A(B)]	Fails to close for isolating nonessential loads	Water may be lost from the EECW loop if a break exists in the RBCCWS. Redundant EECW loop available.
Isolation valve on makeup tank outlet line	Fails to open on automatic system initiation	Low suction pressure to the EECW pump may alarm, requiring manual shutdown, but the other redundant EECW division is available. Makeup pump is disabled.
Isolation valve on nonessential load in essential loop	Fails to close on demand	No adverse effects since the heat loads and flow requirements for the nonessential loads affected are small. (c)
Control valves	Loss of control air or instrument power supply	The EESW temperature control valve will fail open. The EECW will continue to operate and provide the necessary cooling water. The Div I controller will continue to operate on a loss of offsite power to fail the TCV open. NIAS is available for Temperature Control Valve (TCV) P44F400A/B.
Isolation valve for Drywell Loads	Fails to close on high drywell pressure/signals	Redundant full-capacity EECW loop is provided plus the valve can be closed manually.

TABLE 9.2-4 REACTOR BUILDING CLOSED COOLING WATER AND EMERGENCY EQUIPMENT COOLING WATER SYSTEMS FAILURE ANALYSIS

Component	Failure Mode	Consequences on Safety

(a) The differential header pressure sensors are located inside the EECW system boundary; thus, if EECW has been manually initiated with the nonessential loads subsequently restored (either for RBCCW heat exchanger cleaning, enhanced drywell cooling during periods of high lake water temperature, testing, or to provide RHR Reservoir freeze protection during extreme cold weather), a loss of RBCCW pumps while EECW is operating in this mode would not reinitiate EECW or re-isolate the nonessential loads. This protective action is not required, however, since this is not a condition requiring protective action as described in Section 7.1.2.1 and EECW remains capable of supporting the safe shutdown of the plant in this configuration. EECW auto-start on high drywell pressure (i.e., a LOCA) or on a loss of offsite power is unaffected by this mode of operation; therefore, these signals will initiate the automatic protective action of reisolating the nonessential portions of RBCCW piping located inside the EECW system envelope.

(b)The differential header pressure sensors are located inside the EECW system boundary; thus, if EECW has been manually initiated with the nonessential loads subsequently restored (either for RBCCW heat exchanger cleaning, enhanced drywell cooling during periods of high lake water temperature, testing, or to provide RHR Reservoir freeze protection during extreme cold weather), a rupture of the RBCCW piping outside of the EECW system envelope while EECW is operating in this mode would not reinitiate EECW or reisolate the nonessential loads. This protective action is not required, however, since this is not a condition requiring protective action as described in Section 7.1.2.1 and EECW remains capable of supporting the safe shutdown of the plant in this configuration.

(c) If a rupture of the RBCCW piping located inside the EECW system envelope were to occur (with EECW either in standby or in operation for RBCCW heat exchanger cleaning, enhanced drywell cooling during periods of high lake water temperature, testing, or to provide RHR Reservoir freeze protection during extreme cold weather), it is unlikely that the loss in differential header pressure would be sufficient to cause an EECW auto-start due to the small bore of these nonessential lines. It is also unlikely that the RBCCW head tank would deplete to the low level RBCCW pump trip setpoint since the normal makeup capacity exceeds the predicted leak rates. These events rely on the normal EECW makeup supply to feed the break until operators locate and isolate the leak. Again, the protective actions of initiating EECW and isolating the nonessential loads are not required since this is not a condition requiring protective action as described in Section 7.1.2.1 and EECW remains capable of supporting the safe shutdown of the plant during the period required to locate and isolate the break.

Consequences default to those of a rupture of EECW piping as described in the table above.

TABLE 9.2-5 MAKEUP DEMINERALIZED WATER SYSTEM TYPICAL INFLUENT WATER QUALITY ANALYSIS

Major Cation Constituents	Calcium Carbonate $CaCO_3$ (ppm)
Calcium (Ca^{++})	42
Magnesium (Mg^{++})	8
Sodium (Na^{+})	11
Total cations	61

Major Anion Constituents	Calcium Carbonate $CaCO_3$ (ppm)
Bicarbonate (HCO_3^-)	74
Carbonate ($CO_3^=$)	Not Detected
Chloride (Cl^-)	20
Fluoride (F^-)	0.1
Hydroxide (OH^-)	0
Sulfate ($SO_4^=$)	45
Total anions	139

Additional Analysis

pH at 25°C	7.6
Specific conductivity, mmho/cm at 25°C	275[a]
Total solids, ppm	160
Total hardness, ppm as $CaCO_3$	124
Total alkalinity, ppm as $CaCO_3$	87
Iron, ppm as Fe	Trace
Soluble silica, ppm as SiO_2	0.4
Insoluble silica, ppm as SiO_2	0.07
Turbidity, Jackson Turbidity Units	<0.1
Free carbon dioxide, ppm as CO_2	0
Free available chlorine, ppm as Cl_2	1.1
Total Phosphate, ppm as PO_4	0.2[b]
Chemical oxygen demand, ppm as O_2	12

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[a] This value will vary with the season, with a maximum of 350 mmho/cm during periods of heavy runoff.

[b] The total phosphate figure may vary based on the actual treatment at the Frenchtown Plant

TABLE 9.2-6 DEMINERALIZED WATER MAKEUP SYSTEM TYPICAL EFFLUENT WATER QUALITY

Specific conductivity, mmho/cm at 25°C	0.1
pH at 25°C	6.5 to 7.5
Chloride (ppb as Cl^-)	2
Silica (ppb as SiO_2)	<5
Total metallic impurity (ppb of which 2 ppb maximum is Cu)	<10

TABLE 9.2-7 ULTIMATE HEAT SINK COMPONENT DESIGN PARAMETERS

RHR Service Water Pumps

Number supplied	Four
Type	Vertical, turbine type
Fluid	Service water
Capacity, gpm	4500
Total head, ft	185
Motor	
Type	Vertical, induction, dripproof
Horsepower	300
Speed, rpm	1800
Voltage/frequency/phase	4000/60/3

Emergency Equipment Service Water Pumps

Number supplied	Two
Type	Vertical, turbine
Fluid	Service water
Capacity, gpm	1600
Total head, ft	145
Motor	
Type	Vertical, induction, dripproof
Horsepower	100
Speed, rpm	1760
Voltage/frequency/phase	460/60/3

Diesel Generator Service Water Pumps

Number supplied	Four
Type	Vertical, turbine
Fluid	Service water
Capacity, gpm	800
Total head, ft	115
Motor	
Type	Vertical, induction, dripproof
Horsepower	50
Speed, rpm	1760
Voltage/frequency/phase	460/60/3

Cooling Towers

Number supplied	Two
Type	Induced Draft
No. of cells/tower	Two

TABLE 9.2-7 ULTIMATE HEAT SINK COMPONENT DESIGN PARAMETERS

Design flow, gpm	13,000
Design heat load, Btu/hr	160×10^6
Water inlet temperature, °F	116
Water outlet temperature, °F	89
Ambient air dry bulb, °F	92
Ambient air wet bulb, °F	76
Fan motor horsepower	150
Fan type	Eight blades, two-speed
Motor electrical requirements	460/60/3

TABLE 9.2-8 ULTIMATE HEAT SINK FAILURE ANALYSIS

Component	Failure Mode	Consequences on Safety
RHRSW, EESW, and DGSW pumps	Loss of pumping due to loss of offsite power	Power is automatically supplied by the emergency buses fed by the diesel generators.
RHRSW, EESW, or DGSW pump	Loss of pumping due to mechanical failure	RHRSW has one-half capacity still available in one division, completely redundant division still intact. A check valve in pump discharge prevents loss of flow through malfunctioning pump. EESW has full capacity pump in redundant division. DGSW pump failure will cause loss of the particular EDG it services. One half of the electrical division plus full redundant electrical division still remain.
RHRSW, EESW, and DGSW pumps	Do not start due to failure of one divisional pair of diesel generators to start on loss of offsite power	Redundant RHRSW, EESW, and DGSW pumps are provided which are powered off the redundant divisional pair of diesel generators.
RHRSW, EESW, and DGSW pumps	Do not start due to failure of <u>one</u> EDG to start on loss of offsite power	The RHRSW pump associated with the particular EDG will not start; 150 percent cooling capacity still provided. The associated DGSW pump will not start but is not needed. The EESW pump is normally run off a particular EDG. Associated EDG failure causes loss of associated EESW pump. Manual throw-over to other EDG within a division is provided to increase reliability during EDG maintenance. Full-capacity redundant division pump intact.
Valve or piping in Division I or Division II	Loss of flow path due to pipe break or failure of valve to open	Fully redundant flow path with separate supply and return lines is provided to redundant RHR heat exchanger, redundant EECW heat exchanger, and redundant diesel generator.

TABLE 9.2-8 ULTIMATE HEAT SINK FAILURE ANALYSIS

Component	Failure Mode	Consequences on Safety
RHR complex structures	Local structural failure due to tornado-borne missiles, turbine missiles, or EDG missiles	Division I and Division II components are physically separated and divided by a divisional barrier wall. Structure will withstand external missiles. Each EDG is protected by interior walls designed to withstand EDG generated missiles.
Cooling tower structure	Collapse or damage from tornado-borne or turbine missiles	Full-capacity redundant cooling tower provided. Physical separation prevents loss of both divisions.
Cooling tower fan motor	Mechanical failure of fan blades or motor	Each cooling tower has two one-half capacity cells. With redundant cooling tower, capacity of 150 percent still available.
	Failure to start due to diesel generator failure to respond upon loss of offsite power	The particular fan motor not needed as service water pump capacity also reduced; 150 percent cooling capacity still available.

TABLE 9.2-9 CONDENSATE STORAGE SYSTEM COMPONENT DESIGN PARAMETERS

Normal Hotwell Supply Pump	
Number supplied	One
Type	Centrifugal
Fluid	Condensate
Capacity, gpm	600
Total head, ft	246
Motor	
Type	Dripproof, induction
Horsepower	60
Speed, rpm	3550
Voltage/frequency/phase	460/60/3
Emergency Hotwell Supply Pump	
Number supplied	One
Type	Centrifugal
Fluid	Condensate
Capacity, gpm	2000
Total head, ft	108
Motor	
Type	Dripproof, induction
Horsepower	75
Speed, rpm	1750
Voltage/frequency/phase	460/60/3
Condensate Storage Jockey Pump	
Number supplied	One
Type	Centrifugal
Fluid	Condensate
Capacity, gpm	100
Total head, ft	246.2
Motor	
Type	Dripproof, induction
Horsepower	15
Speed, rpm	3500
Voltage/frequency/phase	460/60/3
Condensate Tanks	
Number provided	Two
Type	Vertical, cylindrical
Design code	USAS B96.1
Design pressure, psig	Hydrostatic head
Design ambient temperature, °F	-10 to 95
Operating pressure, psig	Atmospheric

TABLE 9.2-9 CONDENSATE STORAGE SYSTEM COMPONENT DESIGN PARAMETERS

Internal volume, gal	600,000
Dimensions	
Diameter, in.	644 I.D.
Height, in.	432
Material	Aluminum alloy, B-209-5454

Demineralized Water Storage Tank

Number provided	One
Type	Vertical, cylindrical
Design code	USAS B96.1
Design pressure	Hydrostatic head
Design ambient temperature, °F	-10 to 95
Operating pressure	Atmospheric
Internal volume, gal	50,000
Dimensions	
Diameter, in.	228 I.D.
Height, in.	288
Material	Aluminum alloy, SB-209-5454

TABLE 9.2-10 TURBINE BUILDING CLOSED COOLING WATER SYSTEM COMPONENT DESIGN PARAMETERS

TBCCW Pumps

Number supplied	Three
Type	Horizontal, single-stage centrifugal
Fluid	Demineralized water
Capacity, gpm	3000
Total head, ft	57.8
Motor	
Type	Open dripproof
Horsepower	60
Speed, rpm	1770
Voltage/frequency/phase	460/60/3

TBCCW Heat Exchangers

Number supplied	Two
Type	Shell and tube, single pass
Heat transfer duty, Btu/hr	45×10^6
Design code	
Shell	TEMA Class C
Fluid	Demineralized water
Design pressure, psig	150
Design temperature, °F	120
Flow, gpm	6000
Inlet temperature, °F	110
Outlet temperature, °F	95
Material	Carbon steel
Tube	
Fluid	Lake water
Design pressure, psig	175
Design temperature, °F	120
Flow, gpm	9000
Inlet temperature, °F	85
Outlet temperature, °F	95
Material	Admiralty SB-111

TBCCW Makeup Tank

Number provided	One
Type	Horizontal, elliptical dished heads
Design pressure, psig	20
Design temperature, °F	120

TABLE 9.2-10 TURBINE BUILDING CLOSED COOLING WATER SYSTEM COMPONENT DESIGN PARAMETERS

Operating pressure, psig	15
Internal volume, gal	600
Liquid volume, gal	300
Pressurizing gas	Nitrogen
Material	Carbon steel ASTM A-515 Grade 70

TABLE 9.2-11 TORUS WATER MANAGEMENT SYSTEM COMPONENT DESIGN PARAMETERS

TWMS Pumps

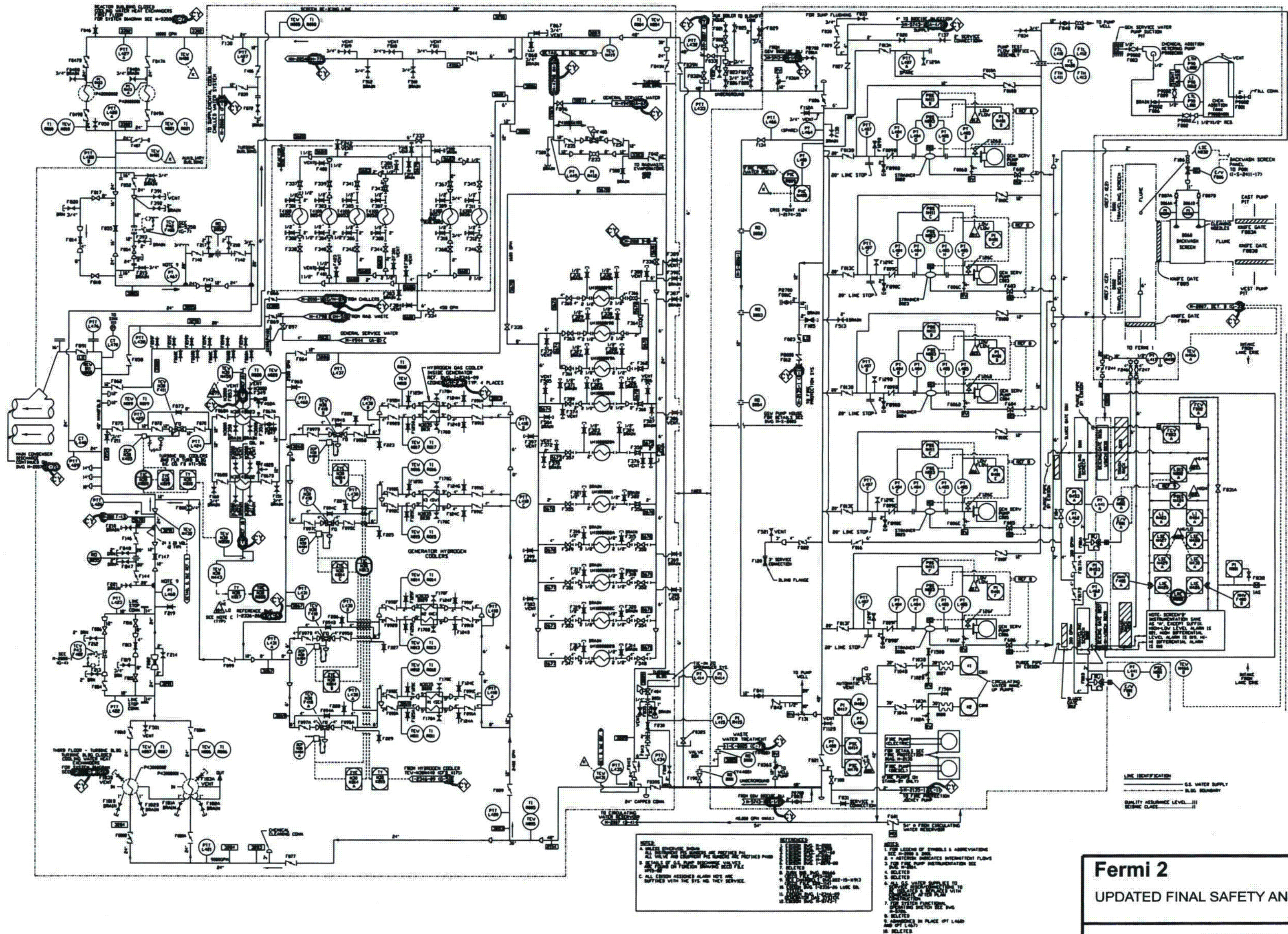
Number supplied	Two
Type	Horizontal, single-stage centrifugal
Fluid	Torus water
Capacity, gpm	250
Total head, ft	480 (rated) 500 (by test)

Motor

Type	Open dripproof
Horsepower	75
Speed, rpm	3550
Voltage/frequency/phase	460/60/3

9.2-12 SUPPLEMENTAL COOLING CHILLED WATER SYSTEM DESIGN PARAMETERS

A. Chillers	
Type	Centrifugal, water cooled
Quantity	Three, 50% capacity each
Refrigerant	R-134a (HFC 134a)
Capacity, tons refrigeration	800 each
Input Power, kw	505
Evaporator	
Chilled water source	SCCW, demineralized
Chilled water flow, gpm	1230
Chilled water temperature, °F	75.8 in/60.2 out
Chilled water pressure drop, ft	16.1
Condenser	
Cooling water source	GSW
Cooling water flow, gpm	2000
Cooling water temperature, °F	85 in/96.2 out
B. Chilled Water Pumps	
Number supplied	Three, 50% capacity each
Type	Centrifugal single stage, horizontal split case
Fluid	Demineralized water
Capacity, gpm	1230
Total head, ft	110
Motor	
Type	Horizontal 324T Frame
Horsepower, HP	40 hp
Speed, rpm	1775
Voltage/Frequency/Phase	460 V/60 Hz/3
C. Expansion Tank	
Number provided	One
Type	Vertical with diaphragm
Design pressure, psig	125
Design temperature, °F	125
Operating pressure, psig	30
Total volume, gal	134
Acceptance volume, gal	46
Pressurizing gas	Air
Material	Carbon steel



DETROIT EDISON COMPANY DRAWING NO. 6M721-2010, REV. CY

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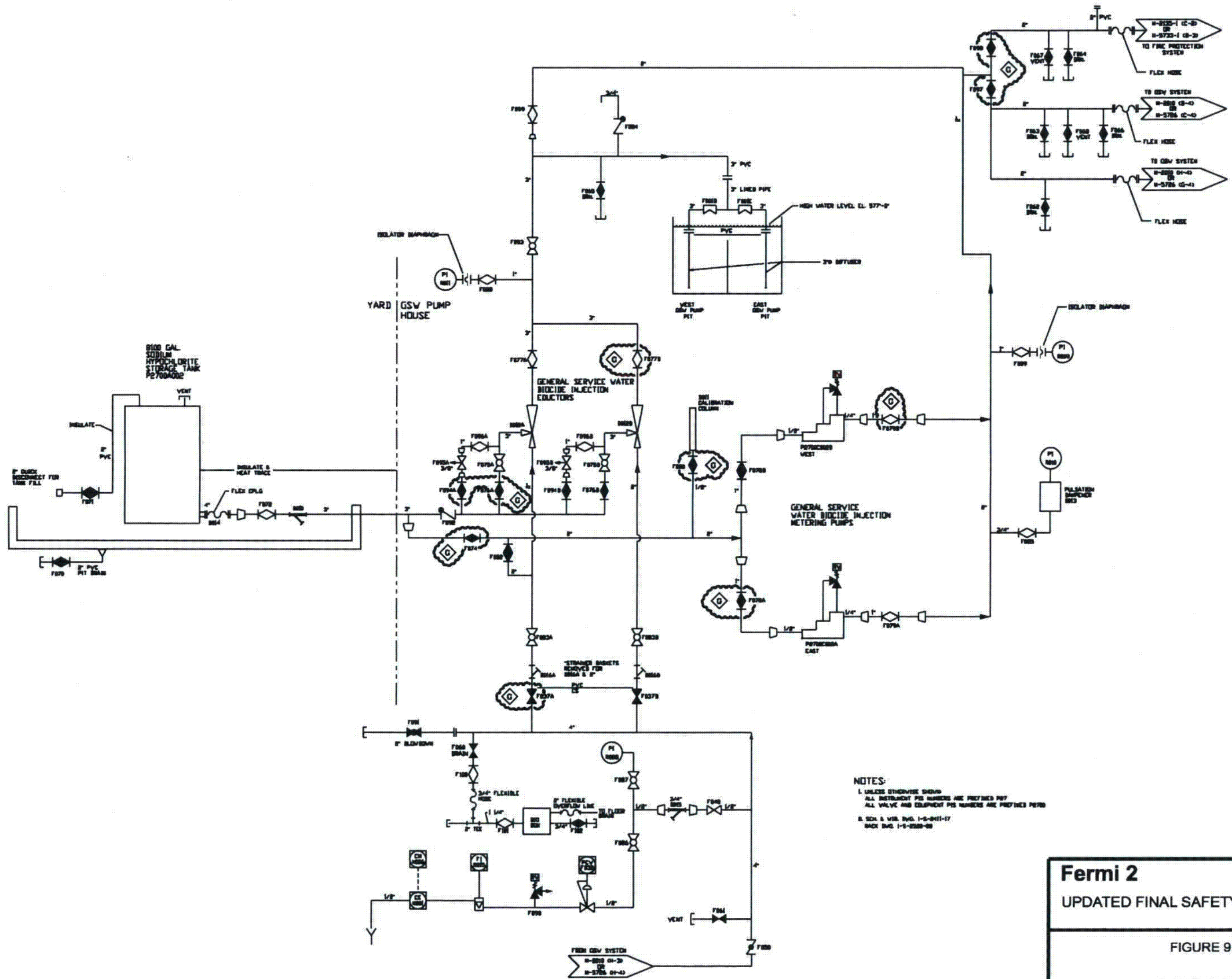
FIGURE 9.2-1, SHEET 1

GENERAL SERVICE WATER



P2700

GSW BIOCIDE INJECTION SYSTEM

DETROIT EDISON COMPANY DRAWING NO. 6M721-5743-1, REV. G

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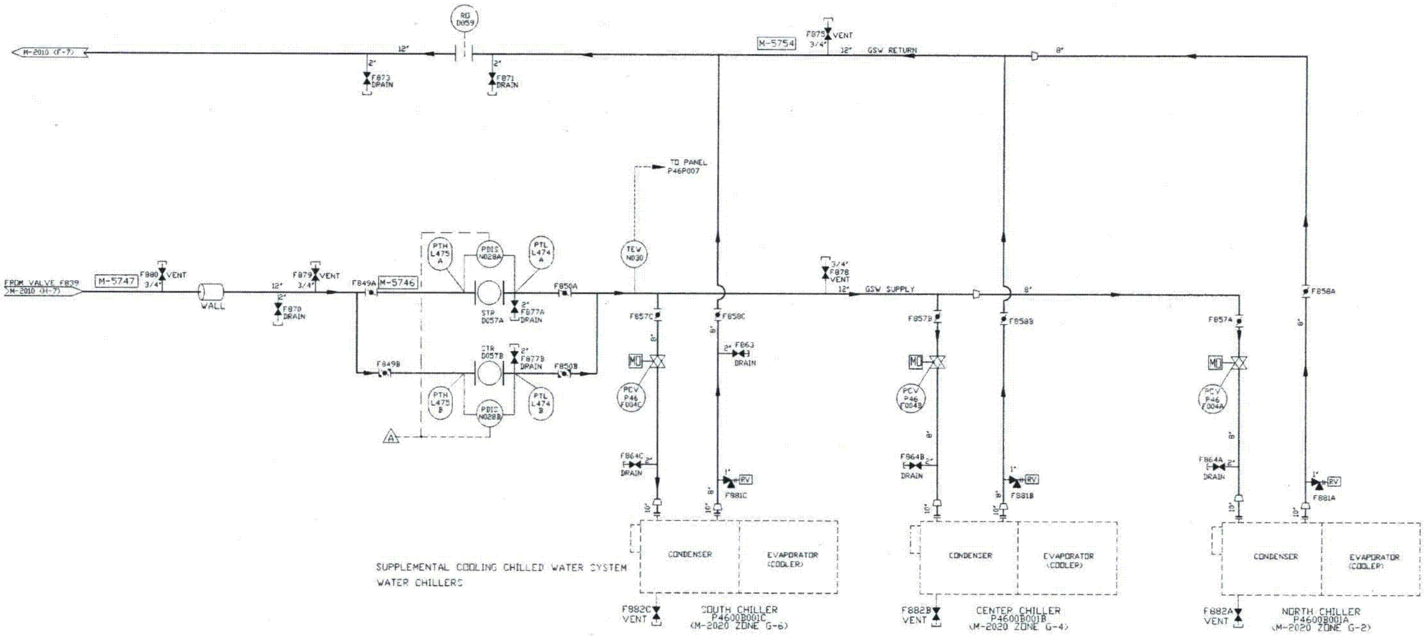
FIGURE 9.2-1, SHEET 2

GENERAL SERVICE WATER
BIOCIDE INJECTION SYSTEM

REV 18 10/12



NOTES:

1. UNLESS OTHERWISE SHOWN:
   ALL INSTRUMENT PIS NOs. ARE PREFIXED P41
   ALL VALVE AND EQUIPMENT PIS NOs. ARE PREFIXED P4100
2. FOR LEGEND OF SYMBOLS AND ABBREVIATIONS SEE M-2000 & M-2001

REFERENCE DRAWINGS:

M-2010	DIAG. GENERAL SERVICE WATER UNIT 2
M-5726	GENERAL SERVICE WATER SYSTEM
M-5726-1	GENERAL SERVICE WATER SYSTEM
M-5745	SUPPLEMENTAL COOLING CHILLED WATER SYSTEM FDS
M-2020	SUPPLEMENTAL COOLING CHILLED WATER SYSTEM

Fermi 2

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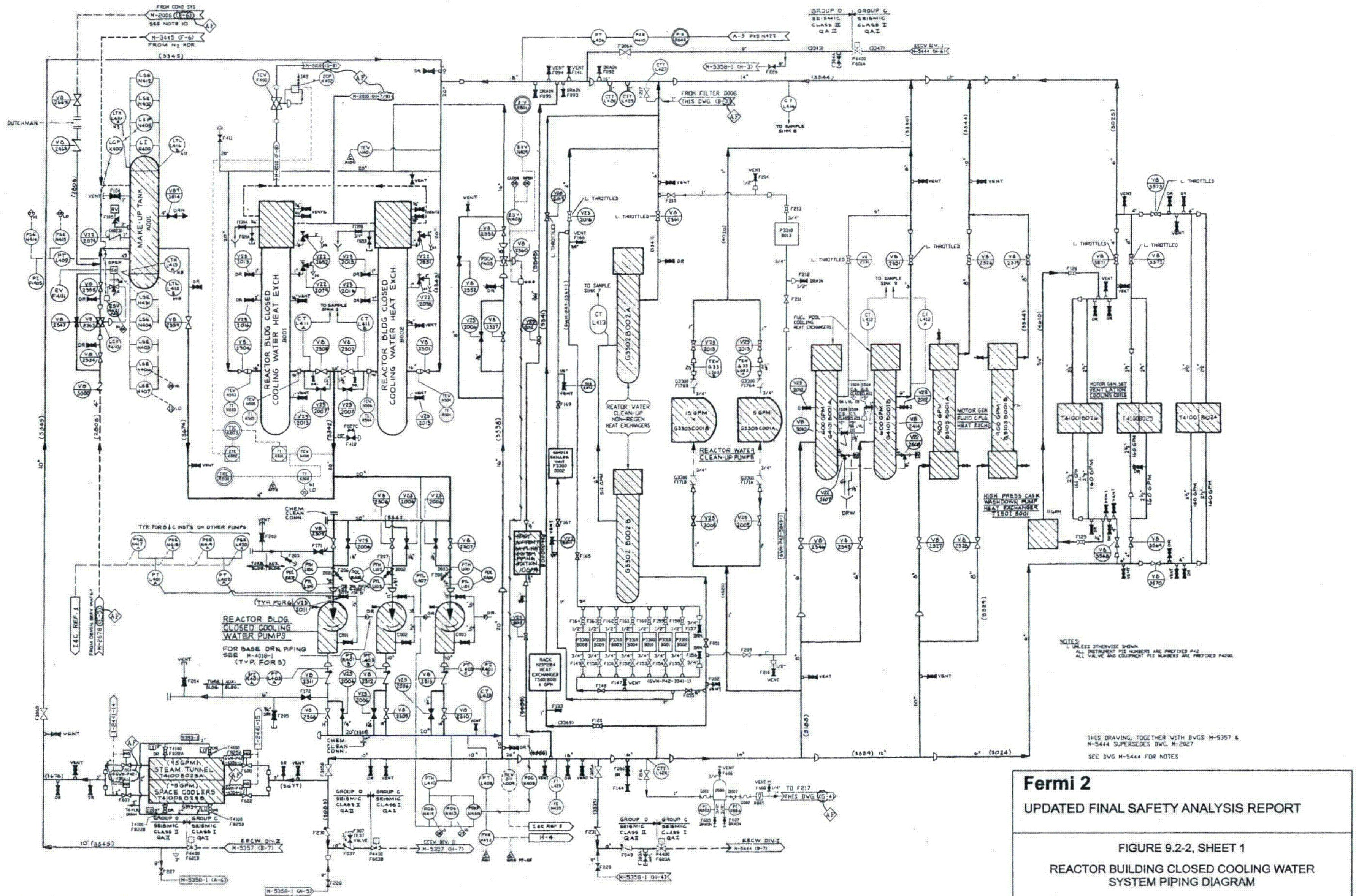
FIGURE 9.2-1, SHEET 3

GENERAL SERVICE WATER SYSTEM PIPING DIAGRAM

REV 10 11/00

DETROIT EDISON COMPANY DRAWING NO. 6M721-2010-1, REV. 0



DETROIT EDISON COMPANY DRAWING NO. 6M721-5358, REV.AF

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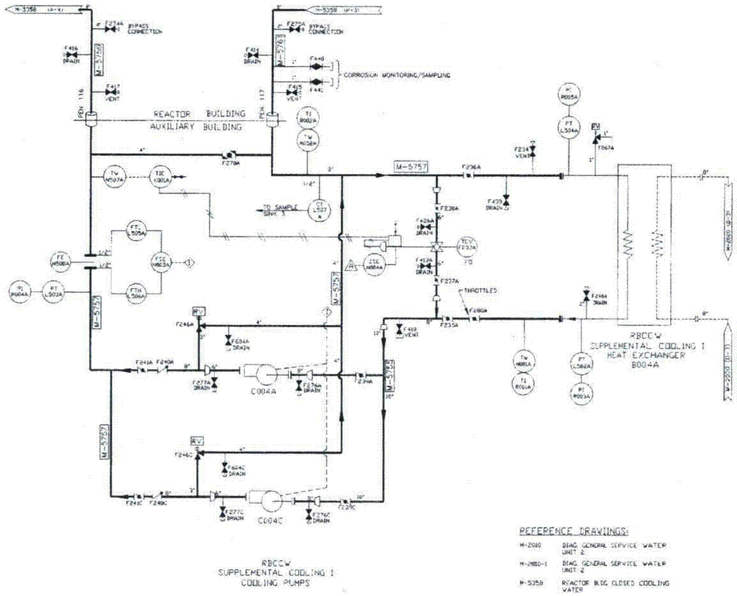
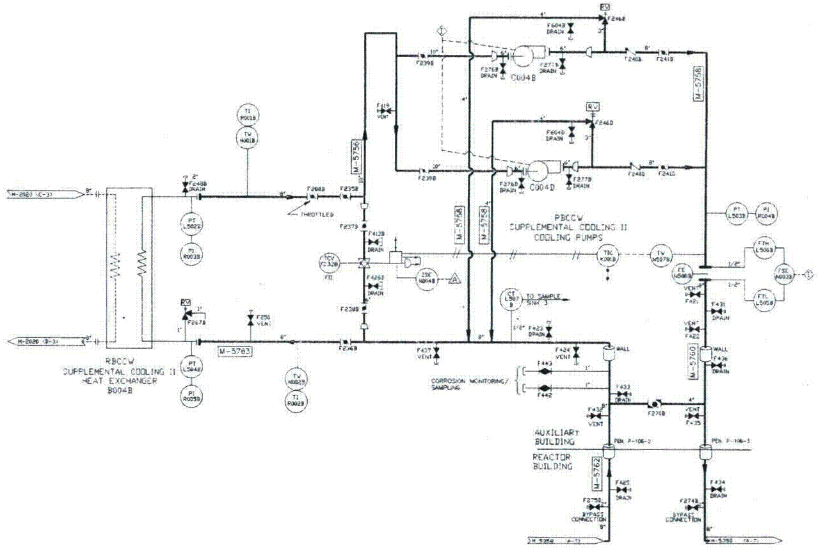
FIGURE 9.2-2, SHEET 1

REACTOR BUILDING CLOSED COOLING WATER SYSTEM PIPING DIAGRAM





REFERENCE DRAWINGS:

M-2010	DIAG. GENERAL SERVICE WATER UNIT 2
M-2010-1	DIAG. GENERAL SERVICE WATER UNIT 2
M-5358	REACTOR BLDG CLOSED COOLING WATER
M-5727	RBCCW SYS FDS
M-5727-1	RBCCW SYS FDS
M-5726	GENERAL SERVICE WATER SYSTEM
M-5726-1	GENERAL SERVICE WATER SYSTEM
M-5745	SUPPLEMENTAL COOLING CHILLED WATER SYSTEM FDS
M-2020	SUPPLEMENTAL COOLING CHILLED WATER SYSTEM, FLOW SWITCHES AND RBCCW MAKEUP TANK LOW LEVEL

NOTE:

1- UNLESS OTHERWISE SHOWN
ALL INSTRUMENT PIS NOs ARE PREFIXED P42
ALL VALVE AND EQUIPMENT PIS NOs ARE PREFIXED P4200

2- UNLESS OTHERWISE SHOWN
ALL VENTS AND DRAINS ARE 3/4"

3- FOR LEGEND OF SYMBOLS AND ABBREVIATIONS SEE M-2000 AND M-2001

LEGEND:

1 INTERLOCK BETWEEN SUPPLEMENTAL COOLING PUMPS, FLOW SWITCHES AND RBCCW MAKEUP TANK LOW LEVEL

DETROIT EDISON COMPANY DRAWING NO. 6M721-5358-1, REV. 0

Fermi 2

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FIGURE 9.2-2, SHEET 2

REACTOR BUILDING CLOSED COOLING WATER PIPING DIAGRAM

REV 10 11/00

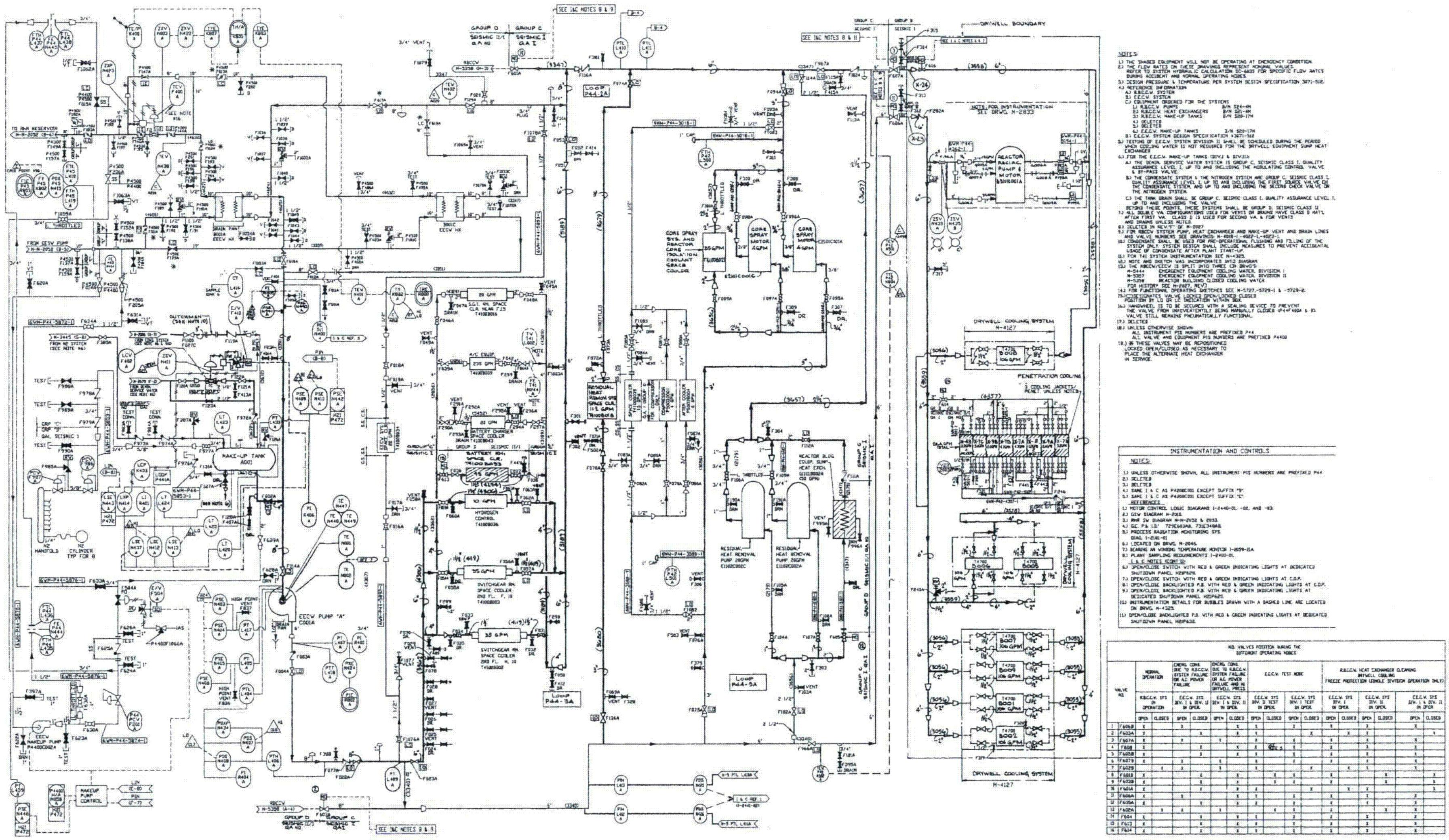

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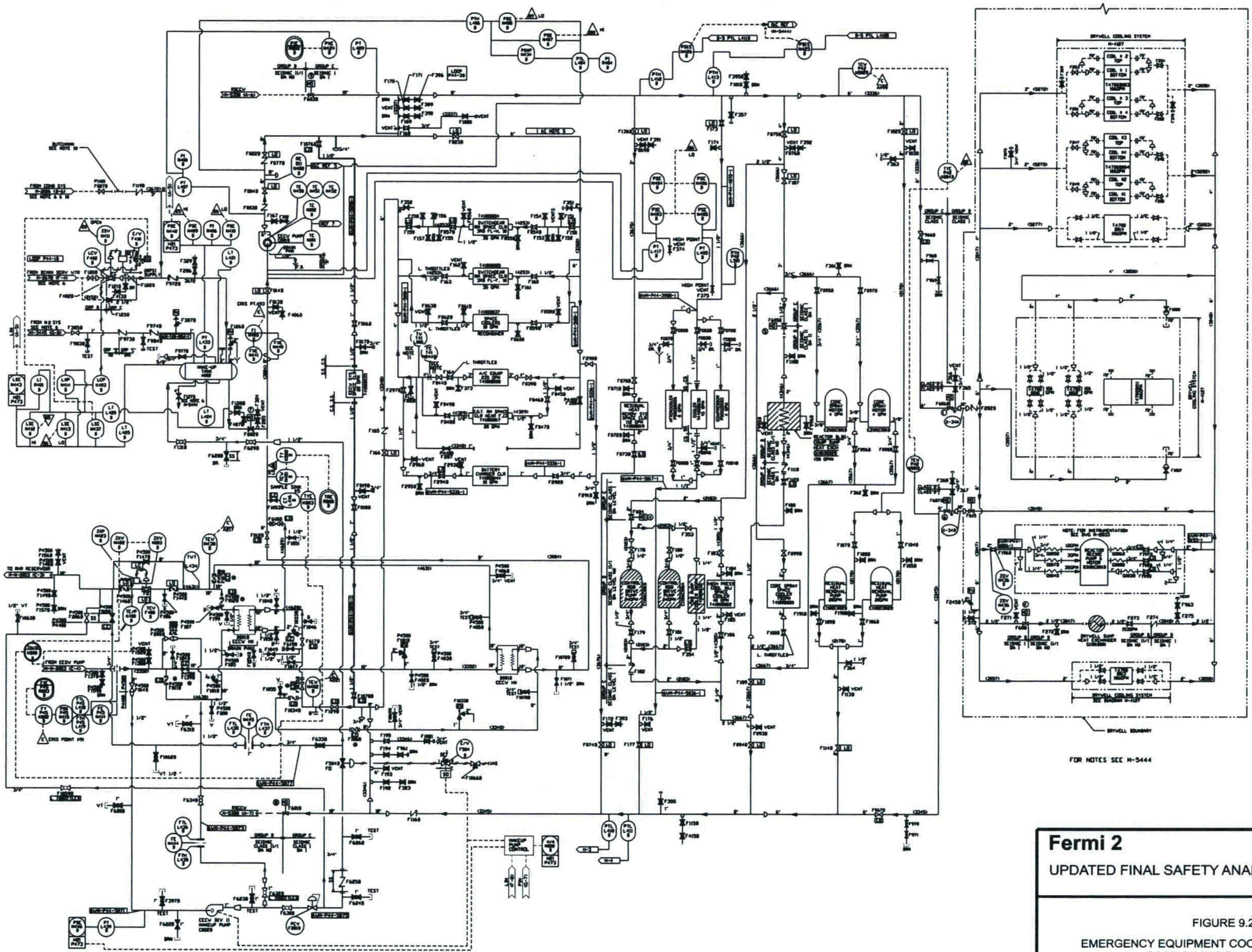
FIGURE 9.2-3

EMERGENCY EQUIPMENT COOLING WATER SYSTEM - DIVISION 1 PIPING DIAGRAM

DETROIT EDISON COMPANY DRAWING NO. 6M721-5444, REV. S

REV 19 10/14



DETROIT EDISON COMPANY DRAWING NO. 6M721-5357, REV. BP

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FIGURE 9.2-4

EMERGENCY EQUIPMENT COOLING WATER SYSTEM
DIVISION II- PIPING DIAGRAM

REV 19 10/14

FIGURE 9.2-5 HAS BEEN DELETED

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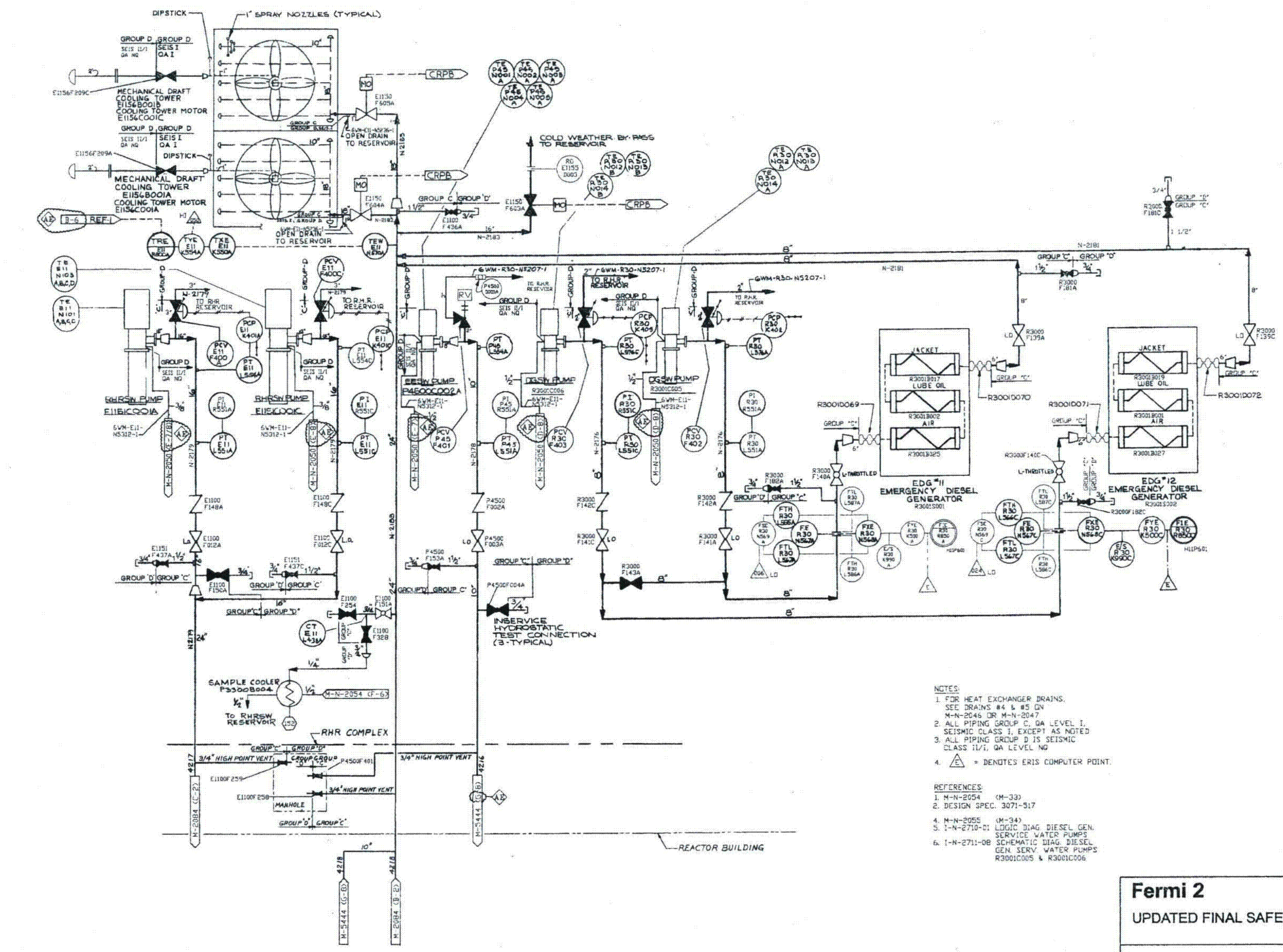

NOTES:

1. FOR HEAT EXCHANGER DRAINS, SEE DRAINS #4 & #5 ON M-N-2046 OR M-N-2047
2. ALL PIPING GROUP C, QA LEVEL I, SEISMIC CLASS I, EXCEPT AS NOTED
3. ALL PIPING GROUP D IS SEISMIC CLASS II/I, QA LEVEL NQ
4. (E) = DENOTES ERIS COMPUTER POINT.

REFERENCES:

1. M-N-2054 (M-33)
2. DESIGN SPEC. 3071-517
4. M-N-2055 (M-34)
5. I-N-2710-01 LOGIC DIAG. DIESEL GEN. SERVICE WATER PUMPS
6. I-N-2711-08 SCHEMATIC DIAG. DIESEL GEN. SERV. WATER PUMPS R3001C005 & R3001C006

DETROIT EDISON COMPANY DRAWING NO. 6M721N-2052, REV.AE

Fermi 2

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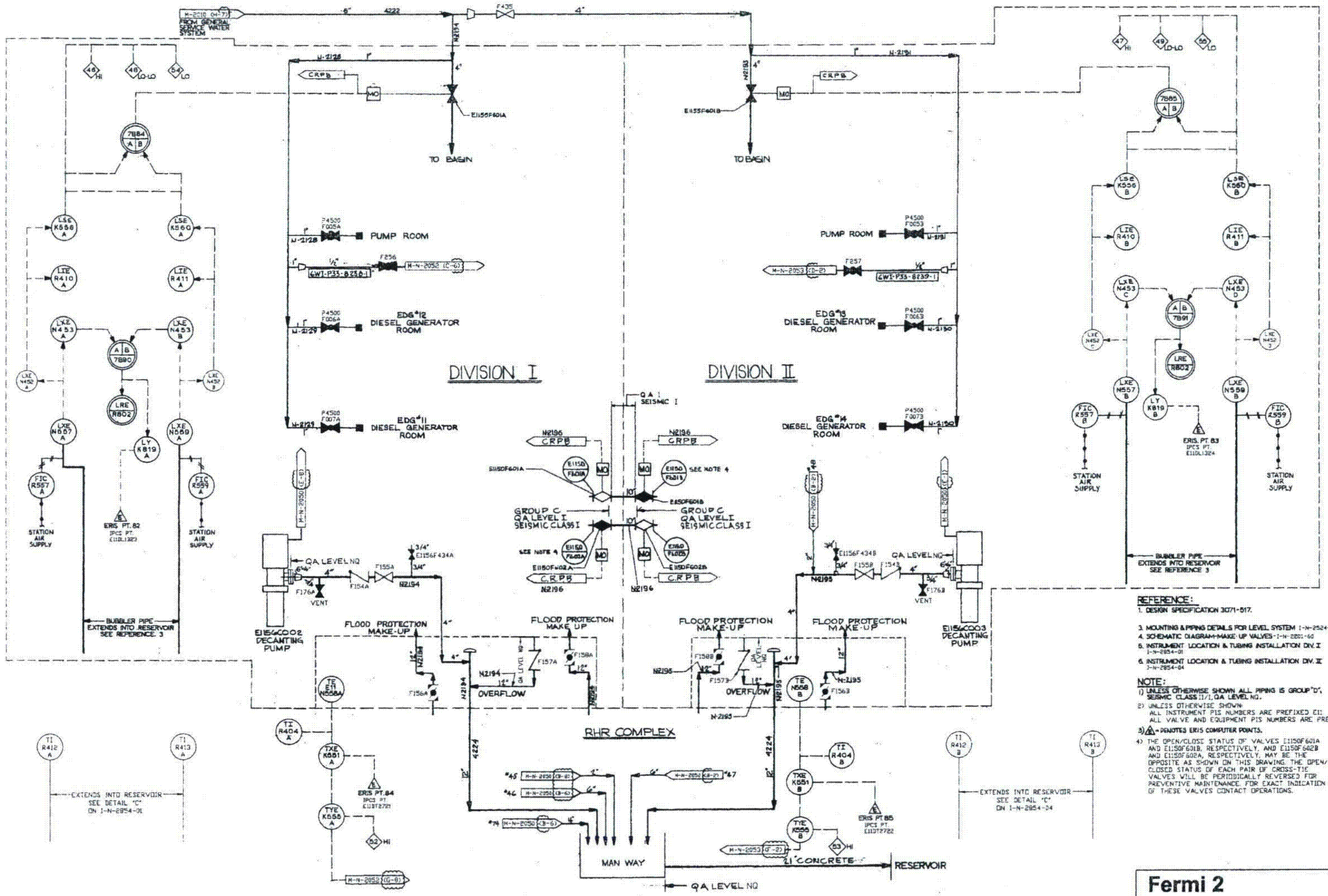
FIGURE 9.2-6

RESIDUAL HEAT REMOVAL SERVICE WATER SYSTEM (DIVISION 1) RESIDUAL HEAT REMOVAL COMPLEX



REFERENCE:

1. DESIGN SPECIFICATION 3071-517.

3. MOUNTING & PIPING DETAILS FOR LEVEL SYSTEM I-N-2524-03
4. SCHEMATIC DIAGRAM-MAKE UP VALVES-I-N-2201-60
5. INSTRUMENT LOCATION & TUBING INSTALLATION DIV. I I-N-2854-01
6. INSTRUMENT LOCATION & TUBING INSTALLATION DIV. II I-N-2854-04

NOTE:

1) UNLESS OTHERWISE SHOWN ALL PIPING IS GROUP "D", SEISMIC CLASS II/I, QA LEVEL NQ.
2) UNLESS OTHERWISE SHOWN: ALL INSTRUMENT PIS NUMBERS ARE PREFIXED E11 ALL VALVE AND EQUIPMENT PIS NUMBERS ARE PREFIXED E1100
3) E - DENOTES ERIS COMPUTER POINTS.
4) THE OPEN/CLOSE STATUS OF VALVES E1150F601A AND E1150F601B, RESPECTIVELY, AND E1150F602B AND E1150F602A, RESPECTIVELY, MAY BE THE OPPOSITE AS SHOWN ON THIS DRAWING. THE OPEN/CLOSED STATUS OF EACH PAIR OF CROSS-TIE VALVES WILL BE PERIODICALLY REVERSED FOR PREVENTIVE MAINTENANCE. FOR EXACT INDICATION OF THESE VALVES CONTACT OPERATIONS.

Fermi 2

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FIGURE 9.2-7

SERVICE WATER, MAKEUP, DECANT, AND OVERFLOW SYSTEMS - DIVISIONS I AND II RESIDUAL HEAT REMOVAL COMPLEX

DETROIT EDISON COMPANY DRAWING NO. 6M721-N-2054, REV. W

FIGURE 9.2-8 HAS BEEN DELETED

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FIGURE 9.2-9 HAS BEEN DELETED

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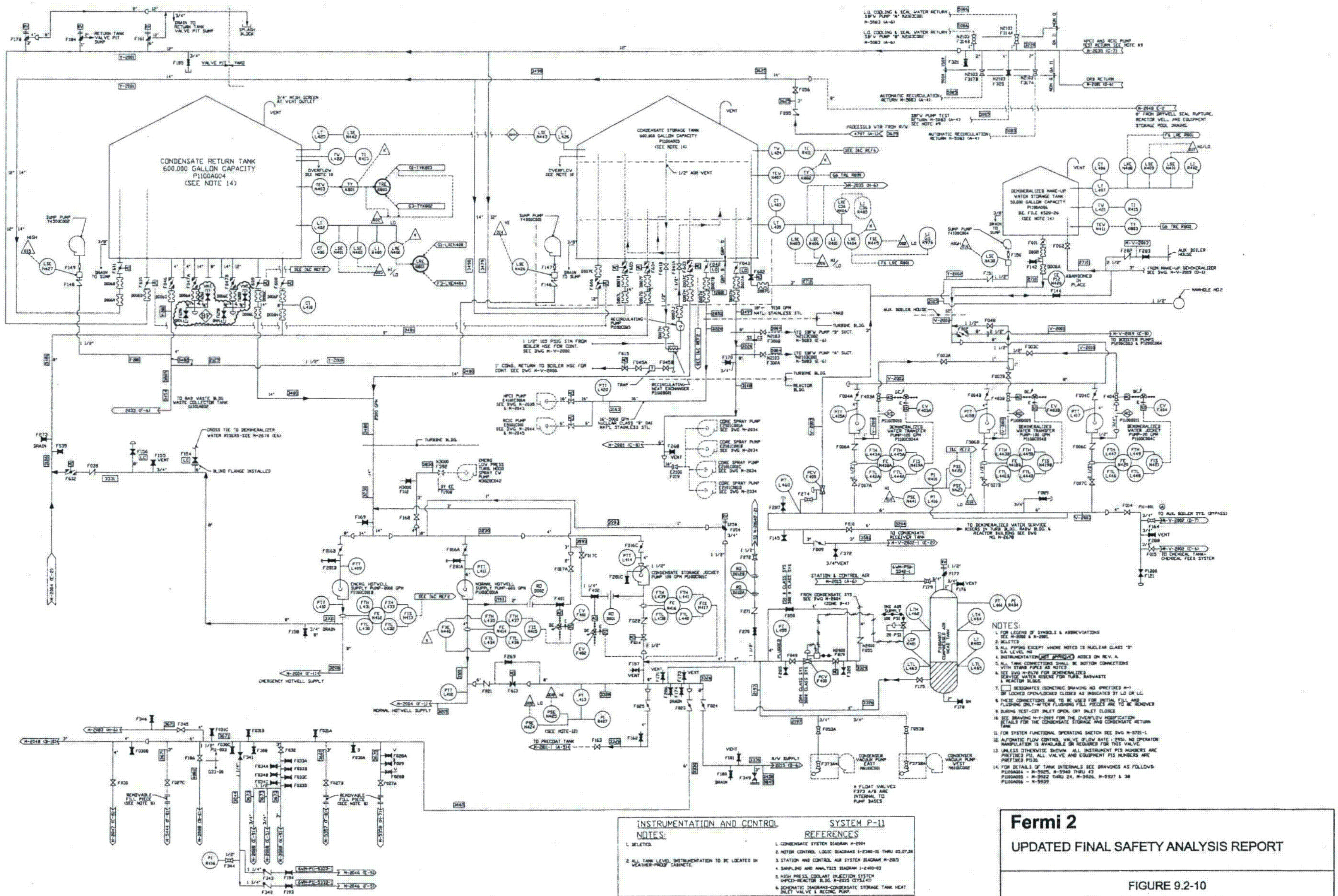

Fermi 2

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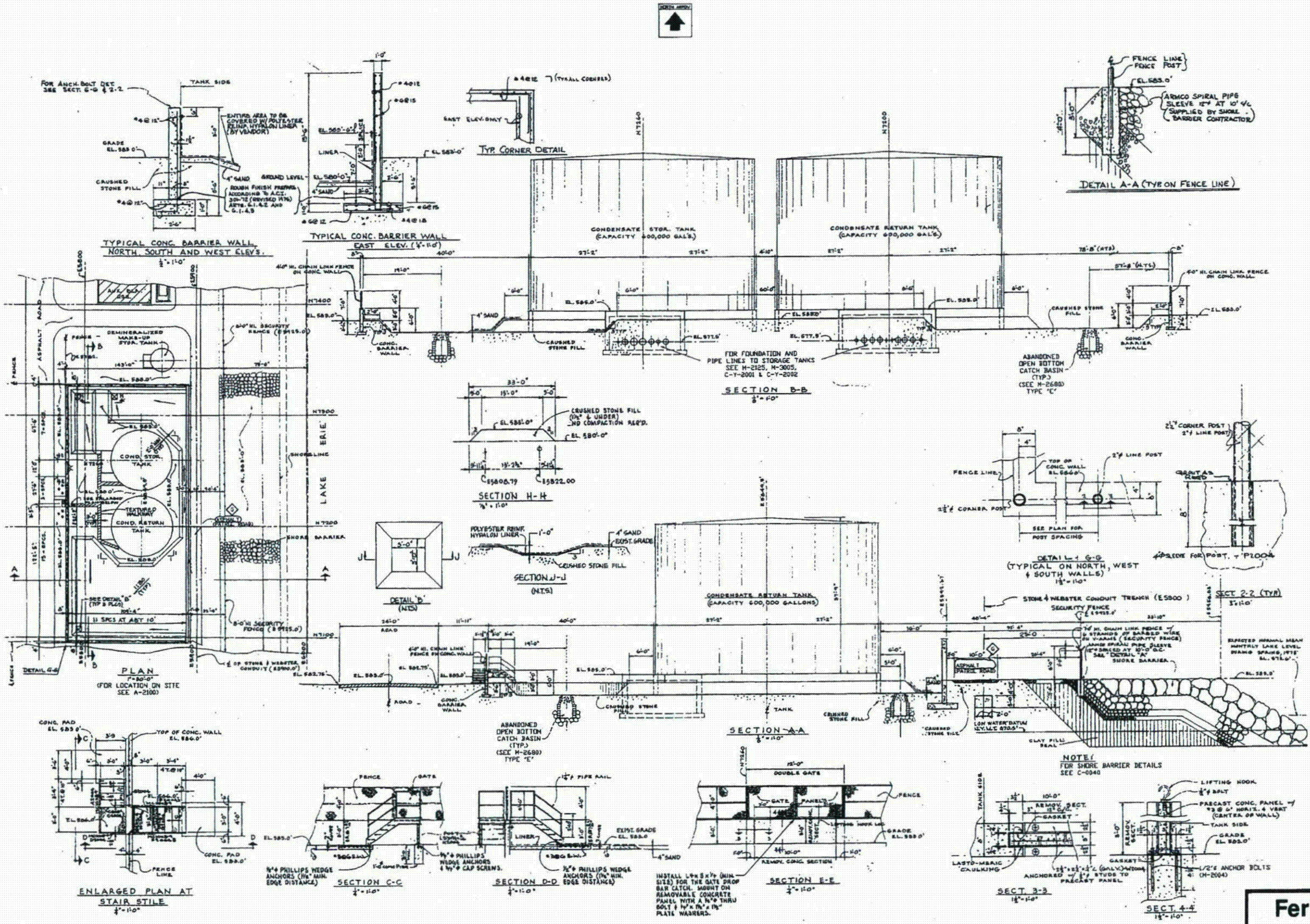
FIGURE 9.2-10

CONDENSATE STORAGE AND TRANSFER

DETROIT EDISON COMPANY DRAWING NO. 6M721-2006, REV. BF



DETROIT EDISON COMPANY DRAWING NO. 6C721Y-2003, REV. G

Fermi 2

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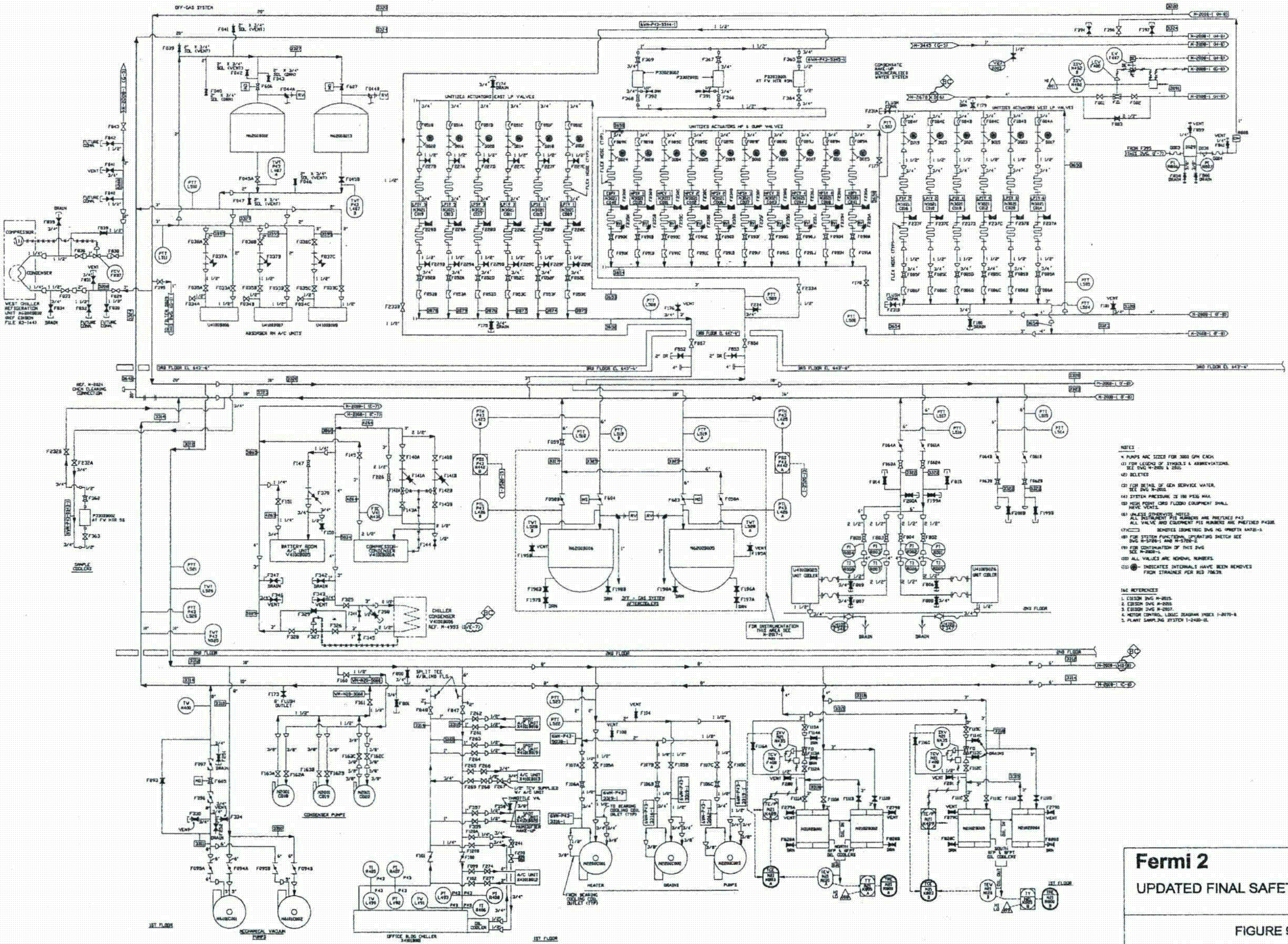
FIGURE 9.2-11

CONTAINMENT WALL AROUND CONDENSATE STORAGE TANKS



Fermi 2

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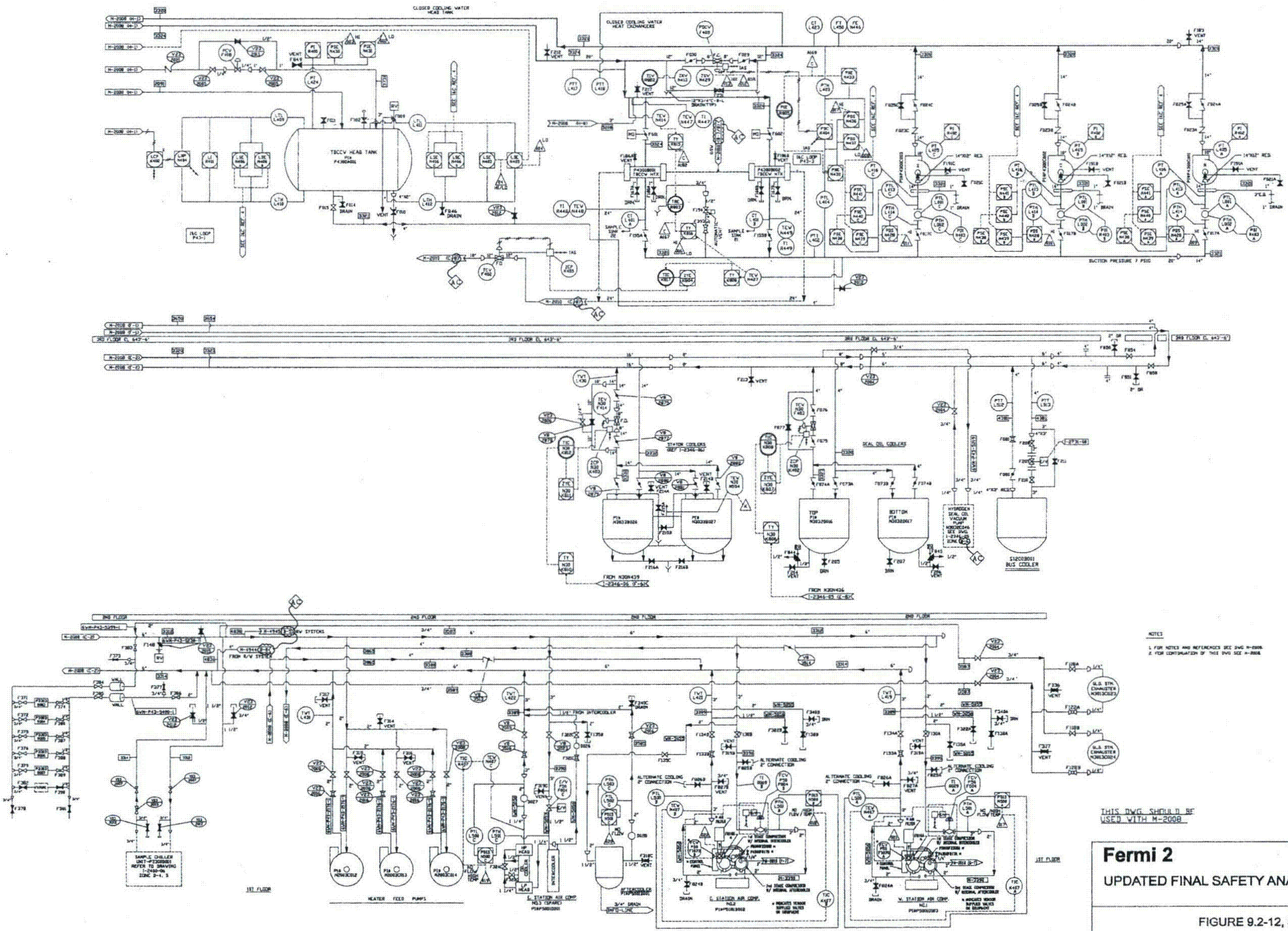
FIGURE 9.2-12, SHEET 1

TURBINE BUILDING CLOSED COOLING WATER SYSTEM PIPING DIAGRAM

DETROIT EDISON COMPANY DRAWING NO. 6M721-2008, REV. BC



Fermi 2

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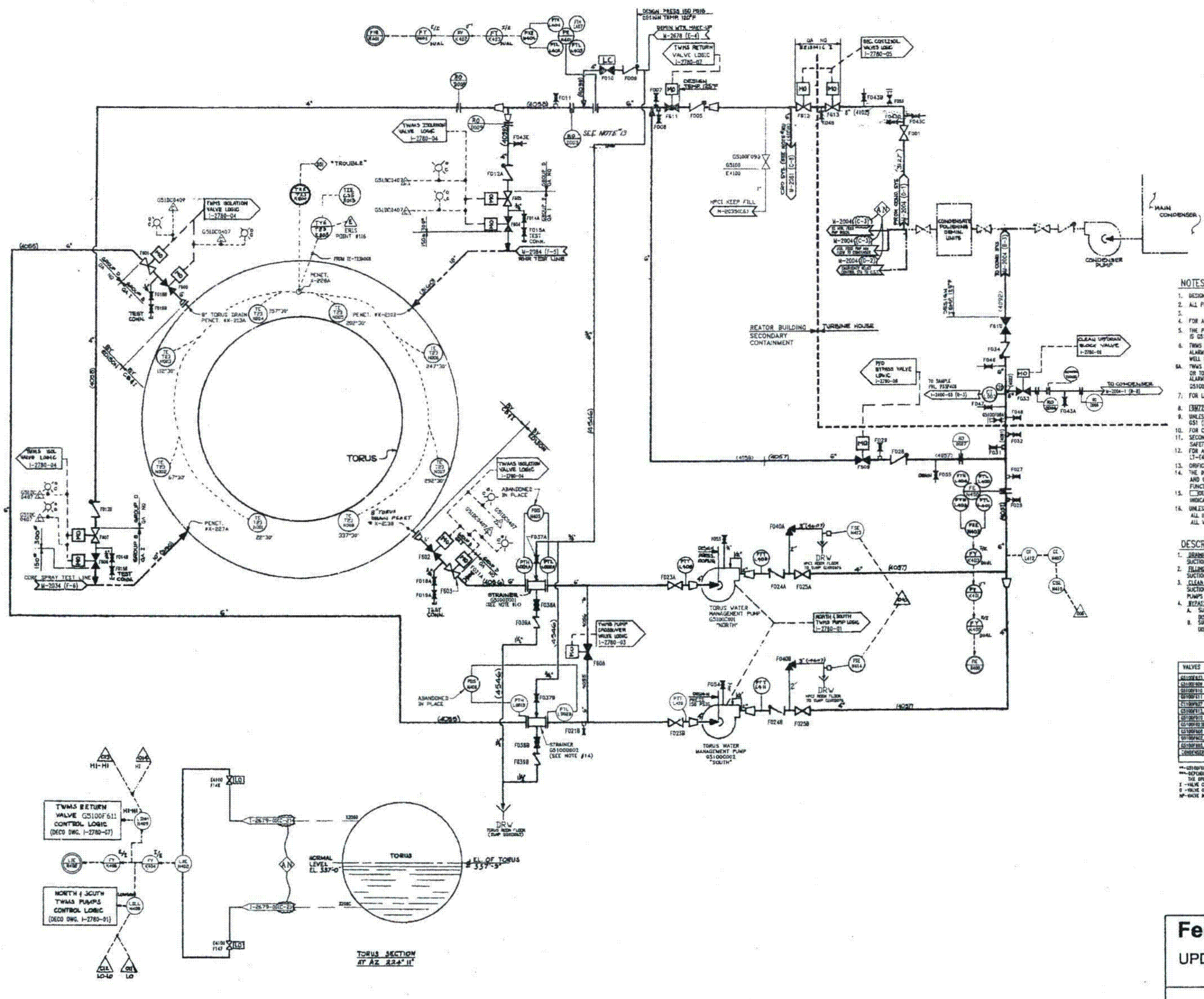
FIGURE 9.2-12, SHEET 2

TURBINE BUILDING CLOSED COOLING WATER SYSTEM PIPING DIAGRAM

DETROIT EDISON COMPANY DRAWING NO. 6M721-2008-1, REV. AC



NOTES:

1. DESIGN PRESS. 350 PSIG, DESIGN TEMP: 195°F EXCEPT WHERE NOTED.
2. ALL PIPING EXCEPT WHERE NOTED IS GROUP 'D', QA LEVEL NO.
3.
4. FOR ASSEMBLY DWG NO'S SEE KEY PLAN M-2320.
5. THE PLANT IDENTIFICATION NUMBER FOR THE TORUS WATER MANAGEMENT SYSTEM IS G5100.
6. TWMS CLASS "B" ISOLATION VALVES TO CLOSE ON "HIGH HIGH LEVEL" ALARM IN DRYWELL OR TORUS ROOM FLOOR DRAIN SUMPS, HIGH DRYWELL PRESSURE OR LOW RPV LEVEL 2 ALARM.

6A. TWMS PUMP OPERATION TO STOP ON "HIGH HIGH LEVEL" ALARM IN DRYWELL OR TORUS ROOM FLOOR DRAIN SUMPS, TORUS WATER LEVEL LOW-LOW ALARM, EITHER PUMP SUCTION ISOLATION VALVE IS ≤95% OPEN OR VALVES G5100F033 & G5100F609 BOTH ACCIDENTALLY CLOSE (≤5%).

7. FOR LEGEND OF SYMBOLS & ABBRIVATIONS SEE M-2000 & 2001.
8. [6M721-   ] (DWG. NO.) IDENTIFIES PIPING ISO FOR FABRICATION & ERECTION.
9. UNLESS OTHERWISE INDICATED ALL ALARM NO'S SHOWN ON THIS DWG. ARE SUFFIXED G51 (EXAMPLE 008G51)
10. FOR CONTROL LOGIC DEVELOPMENT SEE DWG. NOS. I-2780-01 THRU 07.
11. SECONDARY CONTAINMENT ISOLATION VALVES G5100F612 AND G5100F613 ARE NOT NUCLEAR SAFETY RELATED, BUT ARE SEISMIC I, QA LEVEL II.
12. FOR ADDITIONAL INSTRUMENTS CONNECTED TO TORUS LEVEL TAPS/ LT-E41L450 AND LT-E41L452, SEE SYSTEM DIAGRAM M-2035 AND I-2879-01.
13. ORIFICE RO-0003 IS RESIZED AND REPLACED TO MEET TWMS CLEANUP MODE DESIGN CONDITION.
14. THE INTERNAL STRAINER SCREENS TO THE TWMS PUMP SUCTION STRAINERS (G5100D001 AND G5100D002) HAVE BEEN REMOVED. THE STRAINERS HAVE NO STRAINING TYPE OF FUNCTION.
15. [ ] DESIGNATES VALVE POSITION LOCKED OPEN OR LOCKED CLOSED BY LO OR LC INDICATION WITHIN BOX.
16. UNLESS OTHERWISE SHOWN:
    ALL INSTRUMENT PIS NUMBERS ARE PREFIXED G51.
    ALL VALVE AND EQUIPMENT PIS NUMBERS ARE PREFIXED G5100.

DESCRIPTION OF MODES

1. DRAINING
   SUCTION FROM THE TORUS, DISCHARGE INTO THE CONDENSER
2. FILLING
   SUCTION FROM THE CONDENSER, USING CONDENSER PUMP, FILLING THE TORUS
3. CLEAN UP
   SUCTION FROM THE TORUS THROUGH THE MAIN CONDENSER, WITH CONDENSER PUMPS RUNNING, AND DISCHARGING INTO THE TORUS.
4. BYPASS
   A. SUCTION FROM THE TORUS, THROUGH BYPASS, CONDENSER PUMPS RUNNING, DISCHARGING INTO THE TORUS.
   B. SUCTION FROM THE TORUS, THROUGH BYPASS, CONDENSER PUMPS NOT RUNNING, DISCHARGING INTO THE TORUS.

VALVES	1	2	3	4 A	4 B
G5100F033	0	X	0	X**	X**
G5100F609	X	X	X	0**	0**
G5100F610	X	X	X	X	X
G5100F611	0	0	0	X	X
C1100F027 (SEE M-2051)	X	X	0	0	X
G5100F612, F613, F001	0	0	0	0	0
G5100F010	X	X	X	X	X
G5100F023B, F023A, F025B, F025A	0	0	0	0	0
G5100F608	X	X	X	X	X
G5100F602, F600, F603, F601	0	X	0	0	0
G5100F605, F604, F606, F607	X	0	0	0	0
CONDENSER PUMP DISCHARGE	NP	***	NP	NP	X

**- G5100F608 TO OPEN PRIOR TO THE CLOSURE OF G5100F033
***- DEPENDING ON WHICH CONDENSER PUMPS ARE RUNNING THOSE VALVES TO BE IN THE OPEN POSITION, THE REST IN NP.
X - VALVE CLOSED
0 - VALVE OPEN
NP - VALVE IN NORMAL POSITION

Fermi 2

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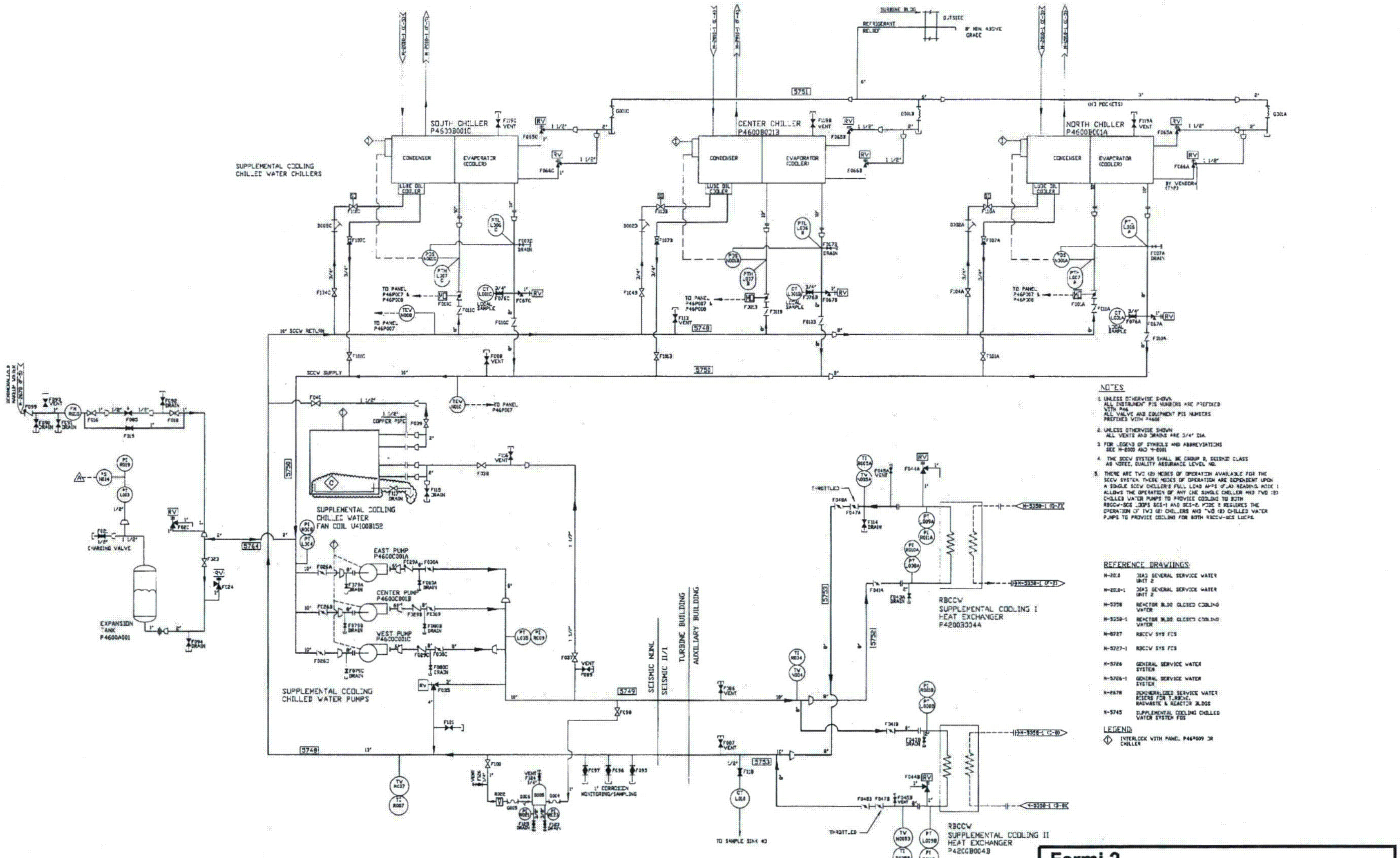
FIGURE 9.2-13

TORUS WATER MANAGEMENT SYSTEM

DETROIT EDISON COMPANY DRAWING NO. 6M721-4100, REV. AN



DETROIT EDISON COMPANY DRAWING NO. 6M721-2020, REV. C

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FIGURE 9.2-14

SUPPLEMENTAL COOLING CHILLED WATER SYSTEM - PIPING DIAGRAM

9.3 PROCESS AUXILIARIES

9.3.1 Compressed Air System

9.3.1.1 Design Bases

The Fermi 2 station and control air system provides the plant with a reliable source of clean, dry, oil-free compressed air for plant operation. Control air system is designed to provide oil and dirt-free air with a dewpoint of -40°F (at pressure). The control air compressors, aftercoolers, dryers, and receiver tanks are provided to supply air to some of the engineered safety feature (ESF) equipment in the plant when the normal supply of control air is not available. Because the noninterruptible portion of the control air system provides control air to ESF equipment, it is classified as a safety-related system.

The station air and interruptible control air systems are constructed in compliance with standards for Quality Group D components. The criteria are met by designing the systems to ASME Section VIII and ANSI B31.1.0 code requirements. These systems are nonseismic.

The noninterruptible control air system is constructed in compliance with upgraded standards for Quality Group D components. These criteria are met by designing this system to ASME Section III, Class 3 requirements. The system is Category I.

9.3.1.2 System Description

The air system is composed of two subsystems. The first is the supply and distribution of station air and the second is the supply and distribution of interruptible and noninterruptible control air. The station air and interruptible control air supply equipment is located in the turbine building. The non-interruptible control air system is located in the auxiliary building. The station and control air systems are the source of compressed air for use in routine maintenance operations, in equipment process cycles such as demineralizer backwashing, and as an instrument and control media. The compressed air system is shown in Figure 9.3-1.

The station air system consists of three, two stage, nonlubricated compressors equipped with inlet filter-silencers, and intercoolers and aftercoolers. Two 150-ft^3-capacity air receivers and the station air distribution piping, valves, and fittings complete the station air equipment.

In operation, ambient air from the turbine building is drawn into the station air compressors via the inlet filter-silencers. This air is compressed, cooled, and discharged into the station air receivers. Normal practice is to have one compressor running and one lined up in automatic. The running compressor maintains near constant pressure (100 psig) in the air receivers while the compressor in automatic is available to start if more capacity is required. A connection is provided in the 8" inlet line to the west Air Receiver tank (P5001A002) for installing and operating an alternate air source at any time when an additional source of compressed air is desired to supply or supplement the needs of the compressed air system. The use of this tap is administratively controlled and, when not in use, a blank flange is installed.

From the station air receiver, the station air is distributed throughout the plant via the station air header/riser system. The station air system is sized to minimize the pressure loss of air at the point of use.

The noninterruptible control air portion of the system consists of two 100 percent-capacity 100 scfm, single-stage nonlubricated reciprocating air compressors; two 100 percent-capacity parallel strings of oil filters, air dryers, and afterfilters; two control air receivers; and associated piping, fittings, and valves. During normal plant operation, the source of noninterruptible and interruptible control air is through interconnections between the station and control air systems. Compressed air from the station air system is supplied through one of these interconnections to the Division I and II noninterruptible control air compressor discharge headers. The air then flows from each header through its divisional 100 percent-capacity filter and dryer. It is cleaned of all particles of dirt $\geq 0.5 \mu m$ (nominal), $\geq 0.9 \mu m$ absolute, and then dried by a regenerative desiccant-type dryer which is designed to establish a -40°F dewpoint (at line pressure). After leaving the filter/dryer, the noninterruptible control air flows to its divisional control air receiver from which it eventually flows to its point of use through its divisional noninterruptible control air distribution system.

Another station air connection supplies the interruptible control air system. The interruptible control air system contains two 100 percent redundant dryers. Each dryer has its own prefilter, afterfilter, and instrumentation. Each dryer unit is capable of supplying the same quality of instrument air as the noninterruptible control air system. Redundancy allows for maintenance to be performed on one unit without jeopardizing the system's air quality or quantity. Dryer redundancy improves the reliability of the interruptible control air system. The interruptible control air flows to the interruptible control air receiver, which supplies the interruptible control air distribution system. The station and control air compressors, air receivers, filters, and dryers are designed to operate in an ambient temperature range of 60°F to 100°F, a range of 20 percent to 100 percent relative humidity, and a radiation field of 1mR/hr.

The control air distribution system is divided into two distinct parts: interruptible and noninterruptible. Noninterruptible control air (NIAS) supplies, through two separate distribution systems (Divisions I and II), equipment in the following systems:

a. Standby gas treatment system (SGTS)

b. Control center air conditioning system (CCACS)

c. Primary containment atmosphere monitoring system (PCAMS)

d. Emergency equipment cooling water system (EECWS)

e. Primary containment pneumatic supply system

f. Torus to reactor building vacuum relief system.

g. Railroad bay airlock door seals.

In addition, Division I NIAS provides control air for the following:

a. Primary containment isolation of drywell equipment and floor drain sump pump discharge lines

b. Back-up supply for Division I (N_2) pneumatic supply to the primary containment.

Division II NIAS supplies, in addition, air operated valves in the following systems:

a. High pressure coolant injection system (HPCI)

b. Reactor core isolation cooling system (RCIC).

c. Standby Gas Treatment Primary Containment Isolation Valves which support Torus Venting.

d. Torus Vent Secondary Containment Isolation Valves.

All other control air users are connected to the interruptible control air distribution system. Interruptible control air (IAS) is supplied through its own set of filters, dryer, and receiver tank, which is fed from the station air system.

The station air compressors and their associated coolers are cooled by the turbine building closed cooling water system (TBCCWS). The control air compressors and aftercoolers are cooled by the reactor building closed cooling water system (RBCCWS) or the EECWS.

During normal operation, any one of three installed station air compressors will be in operation. One of the other two will be in "auto" and the third compressor will be in the "off" position. Normal operating pressure from the station air compressors is nominally 100 psig. If the station air header pressure drops below 95 psig, the compressor in "auto" will automatically start.

If the pressure drops to 90 psig, an alarm in the control room will be initiated and the third compressor may be manually started from the control room panel. If the station air header pressure continues to decrease, at 85 psig the station air supply header isolates and only supplies the IAS and NIAS. An alarm is initiated in the control room.

Should station air supply pressure to either division of NIAS decrease to 85 psig, its division's control air compressor automatically starts. If the pressure continues to decrease, at 75 psig the station air supply isolates from the NIAS and alarms. Each division of NIAS is supplied at this point by its own control air compressor.

There is a normally locked closed intertie between Divisions I and II of the noninterruptible control air system. During a maintenance outage on a control air compressor, after cooler, filters, or dryer of one division the intertie may be opened so that the division having the outage may be fed by the other division. Similarly, the normally closed interruptible control air intertie to Division II noninterruptible control air system may be opened during a Division II supply maintenance outage (i.e., Division II compressor, after cooler, filters/dryer outage). In this latter case, loss of offsite power or any other station air failure would render Division II of the noninterruptible control air system inoperable. The intertie auto isolation valve will close on loss of power or low header pressure, thus maintaining Division II noninterruptible air receiver tank integrity. In addition, the isolation valve for the station air supply to the noninterruptible control air system interconnection has a normally locked closed bypass valve and a normally locked open outlet valve. These valves may be unlocked and repositioned (i.e., the bypass valve opened and the outlet valve closed) to provide an alternate lineup for station air supply to the noninterruptible control air system to support normal plant operation in the event the isolation valve is unavailable.

9.3.1.3 Safety Evaluation

The noninterruptible portion of the control air system is required to effect a safe reactor shutdown; it is also required for control during long-term recovery. The station air system and interruptible control air system are not required to effect a safe reactor shutdown. The pneumatic supply to the primary containment is normally fed from the nitrogen inerting system (Subsection 9.3.6). An intertie is provided to permit Division I noninterruptible control air to be used as an emergency backup to the Division I containment pneumatic supply system.

Bottled nitrogen can also be connected to both containment pneumatic supply divisions as an additional backup supply source.

On loss of offsite power, the control air compressors are automatically started with power supplied from the emergency diesel generators (EDGs). Enough receiver capacity is provided to supply 10 minutes of noninterruptible control air before control air compressor load pickup by the diesel generators is required. With normal offsite power available, the control air compressors start immediately on low noninterruptible control air header pressure.

Maximum plant availability and control air system reliability are ensured by providing three station air compressors and two standby control air compressors. Additionally the control air compressors are powered by independent ESF power sources, and each division includes a 10-minute receiver tank reserve capacity.

Control air accumulators are located so as to maximize protection for the associated valve and nearby safety-related equipment. Physical separation criteria for the associated system of the valve were also maintained in determining the accumulator location. Inside the drywell, the accumulators have been integrally supported and welded to drywell support steel; outside the drywell, anchor bolts have been used to secure the welded accumulator support structures in position. The accumulator supports and anchor system were analyzed for stressed conditions resulting from seismic excitation, thrust loading from a tank rupture or supply line rupture, and external jet impingement during a LOCA environment. In each of the above loading conditions, the support and anchor designs were found to be adequate to preclude the accumulators from becoming missiles.

9.3.1.4 Tests and Inspections

Initial construction tests such as air leak tests were conducted per applicable code requirements for the station and control air systems. Initial system flow checks, valve operability, instrumentation and control loop checks, and alarm setpoints for the control air subsystem were done in accordance with the Preoperational Test program as discussed in Chapter 14. The station air subsystem was subjected to similar acceptance testing. The quality of the air delivered by the filter/dryer units was also determined.

Periodic examinations of filters and dryers and periodic replacement of filter cartridges are scheduled to ensure control air quality. Periodic inspections are made of compressors to ensure performance of these active units.

Periodic inspections of receiver tanks are performed. Inspection of instruments is made to confirm actuation of relief valves, isolation valves, automatic switchovers, and alarms. Automatic compressor starts are also demonstrated.

9.3.1.5 Instrumentation

Local (turbine building) instrumentation in the station and interruptible control air systems is provided to monitor line and receiver air pressure, pressure drop across filters, compressor airflow rates, and temperature of the compressed air and cooling water. Similar local instruments in the reactor/auxiliary building are provided for the noninterruptible control air system.

Main control room instrumentation consists of pressure indication of station air and control air headers (with low-pressure alarms), selector switches to isolate either division of noninterruptible air, and control switches for the control air compressors.

The station air compressors are started in the main control room and controlled locally.

9.3.2 Process Sampling

Figures 9.3-2, 9.3-3, and 9.3-4 illustrate the sampling systems in the reactor building, turbine building, and radwaste building, respectively. Details of the process radiation monitoring system (RMS) are given in Section 11.4.

9.3.2.1 Design Bases

The Fermi 2 process sampling system is designed to permit samples to be taken for the following purposes:

a. To maintain radiological surveillance

b. To provide analog measurement signals to controls for process equipment

c. To evaluate the performance of system equipment

d. To measure the quality of the process fluid.

Rad Protection supervision is provided where required (some samples will be radioactive). Wherever samples are delivered through shielding walls, backflushing facilities are provided to confine the radioactive material to the shielded area. Where necessary to avoid health hazards to operators, the system is designed with special safeguards, such as one or more remote air-operated block valves with remote position indicators. The system is designed to permit continuous sampling and minimize plate-out or decay that could bias analyses.

Where feasible, the system piping and sample taps are designed to permit mixing and sampling before process inventories are transferred in the process train.

To ensure that the samples taken are representative, the following considerations are provided for in the design:

a. Line lengths are minimized and the smallest practical line diameter is used to reduce lag time and to minimize plating-out of sample

b. Sampling lines avoid traps, deadlegs, and dips

c. The sample flow rates and line sizes are chosen to ensure flow in the turbulent-flow regime.

Sample lines are type 304 stainless steel tubing. After the source or isolation valves, lines connected to Quality Groups A, B, and C systems are constructed to meet Quality Group D requirements. Lines connected to Quality Group D systems are constructed to meet Quality Group D requirements.

All sample tubing or piping, from the point where it connects to a process system to and including the source valve (or if inside primary containment, from the source to the isolation shutoff valve outside primary containment) is the same piping class as the system piping to which it connects. Further, sample lines are either pitched to drain or are equipped with vents and drains, and are designed to prevent damaging water hammer in operation. External lines are heat traced to prevent freezing and all hot lines are stress analyzed to accommodate thermal movement.

9.3.2.2 System Description

Tables 9.3-1 through 9.3-4 describe the process sampling system by listing, for each system sampled, the sample locations, the analyses to be performed, and anticipated pressures and temperatures. Grab samples are taken for laboratory analysis. Grab samples may be taken locally near the process point or remotely at a central sampling station. For remote grab samples, a sample line is routed from the process pipe to the central sampling station. For local grab samples, a sample line is routed from the process pipe to the nearest accessible area for plant personnel. To determine whether a grab sample should be remote or local, the samples are put into the following classification or criteria:

a. Classification or criteria for remote grab sample:
   1. Sample is taken frequently
   2. Sample point is inaccessible during operation
   3. Sample has to be conditioned
   4. Sample may be radioactive
   5. Entrained gases must be vented through a hood.

b. Classification or criteria for local grab samples:
   1. Sample is taken infrequently
   2. Sample point is accessible
   3. Sample taken only during shutdown
   4. Sample tends to form deposits which would cause plugging of longer lines.

Remote grab samples are routed to a central sampling station, where they are temperature conditioned to 120°F or less and are provided with manual flow control. Continuous samples are routed to the central sampling station where the samples are regulated for proper flow and

are temperature controlled as required by the instrument manufacturer. Continuous samples are provided with a means for taking grab samples at the central sampling station and are designed so that grab samples do not reduce flow below the design requirements of the continuous analysis instrumentation.

After sample conditioning, except for those samples recovered directly into the process flow, the samples flow through the analysis instrumentation to the radwaste floor drain system.

A special sample drain collection/recovery system has been designed to reduce radwaste burden by collecting and recovering certain sample drains which are of sufficient water quality to allow recovery into the condensate process flow without radwaste processing. The system consists primarily of a single tank with 240 gallon working volume and 20 gpm pump. Normally this system discharges to the condensate pump suction header but defaults to turbine building floor drains.

Central sampling stations are located in the radwaste, turbine, and reactor buildings. This is done to minimize the length of the sample lines and therefore shorten the transport time for the samples. Each central sampling station contains the remote grab and continuous samples as discussed in previous paragraphs, and the temperature conditioning equipment and analysis instrumentation. The central sampling stations are provided with exhaust hoods to draw air from the sample sinks. Airflow is 100 to 150 linear ft per minute.

All remote sample lines, where possible, are pitched 1/4 in./ft in direction of flow. The sample line lengths are as short as possible and the routing avoids traps, deadlegs, and dips upstream from the sample discharge. Sample flow is in the turbulent-flow region to minimize deposition and to ensure representative samples. Local samples are located in well-ventilated accessible areas. Drain funnels are provided to carry sample streams, which are not recoverable, to the floor drain system.

9.3.2.3 Safety Evaluation

Samples that require special handling, and all sample lines that flow continuously, lead to central sampling stations in the reactor building, the turbine building, or the radwaste building. The central sampling stations are equipped with ventilation hoods, backflushing facilities, and pressure and temperature controls. Remote air-operated sample valves are controlled from these central sampling stations.

High-pressure sample lines are required to pass hydrostatic tests with the process units they serve and must conform to the same construction standards.

All sample lines have a shutoff valve located as close as possible to the sample source connection. This valve is manually operated if accessible, and solenoid operated where inaccessible.

Solenoid valves are designed to fail closed. Soft-seated solenoid valves are provided to ensure minimum leakage because leakage could go undetected for long periods of time. Since all samples have a potential for becoming radioactive, the following special precautions are taken to minimize radiation hazards to plant personnel:

a. Sample lines are routed wherever possible in shielded areas where plant personnel have little or no access

b. Equipment that tends to trap activated "crud" is kept behind shield walls

c. Provisions are made for sample line backflushing with demineralized water at the sample stations

d. Ventilated hoods are provided at the sample station

e. Local grab samples are located in well-ventilated areas that are accessible to plant personnel

f. Remote samples are extended through shield walls if located near radioactive equipment or if the sample line creates significant radiation field.

9.3.2.4 Tests and Inspections

During plant operation or shutdown, no special tests or inspections are required for sample lines and sample stations beyond inclusion in the test and inspection programs conducted on the systems they serve. Continuous analysis instrumentation will be periodically checked and recalibrated.

9.3.2.5 Instrumentation

Pressure controls and remotely operated valves are procured to the same specification as the lines they are sampling. The continuous monitors installed in various sample stations are identified by function in Tables 9.3-1 through 9.3-4.

9.3.3 Plant Equipment and Floor Drains

9.3.3.1 Design Bases

The plant equipment and floor drainage systems are designed to collect and remove all waste liquids from their points of origin to a suitable disposal area in a controlled and safe manner. Water from radioactive drains is collected for sampling and analysis prior to disposal to the environment in accordance with 10 CFR 20. Drain line penetrations through containment barriers are designed to maintain containment during normal operation and design-basis accidents (DBAs).

In the reactor, auxiliary, turbine, and radwaste buildings, most drain water is considered potentially radioactive and is accumulated for periodic discharge to the radwaste system for treatment. In general, drainage from production equipment is of high purity and high activity relative to floor drain discharge, and is collected separately from the floor drain discharges. In the radwaste process, cleanup of the floor drain accumulations may be more complex and could require more unit separation than do the equipment drain accumulations routed to the radwaste waste collector tank.

Equipment drain water of relatively high purity and high activity is separately collected and discharged to the radwaste waste collector tank and subsequent cleanup train. If the effluent from this cleanup train is of satisfactory quality, the purified stream is normally recycled to the 600,000-gal condensate return tank. Floor drain water of relatively low purity is collected in separate sumps and periodically discharged to the radwaste floor drain collector tank and cleanup train. If this water is of satisfactory quality, the purified stream may also be

recycled to the CST or exhausted to the plant circulating water reservoir decanting line that flows into Lake Erie.

The normal equipment and floor drain water in each quadrant of the reactor building sub-basement is collected in the local sump of the respective quadrant. The drain water from the NW and SE quadrant sumps is discharged to the radwaste waste collector tank. The drain water from the SW and NE quadrant sumps is discharged to the radwaste floor drain collector tank.

Equipment drain connections are generally through open funnels (sight drains) at those locations where it is considered desirable to verify performance at a glance, where periodic temperature observations may be required, or where the coolant water system is a high pressure system and might overpressurize drain lines and equipment.

Drain system piping effecting drywell isolation is constructed to meet standards for Quality Group B components. They are designed to ASME Section III, Class 2 code requirements. The balance of the drain system is either Quality Group C designed to ASME Section III, Class 3 code requirements or Quality Group D, designed to ANSI B31.1.0, except for the recirculating sump heat exchangers. Their piping is designed to ASME Section VIII and to ANSI B31.1.0 code requirements.

All the equipment drain piping above the floor in the reactor building sub-basement is designed to ANSI B31.1.0 code requirements.

9.3.3.2 System Description

NOTE: Pump rates are nominal flow rates.

The Fermi 2 drainage system is designed for accumulation of discharges from equipment and floor drains inside the reactor building, auxiliary building, turbine building, and radwaste building, and for periodic transfer of these accumulations to the liquid radwaste system.

Within the reactor building, seven separate drain collection systems operate, each with an independent sump. The reactor and auxiliary buildings drain systems are shown in Figures 9.3-5 and 9.3-6.

An equipment drain collection system from primary coolant components terminates in a 1100-gal nominal capacity sump located in the drywell area under the reactor pressure vessel (RPV) with twin parallel 50-gpm transfer pumps that discharge to the radwaste waste collector tank. The sump is closed and vented, with a recirculating bypass capability from the transfer pump discharge header line returning to the sump. This bypass flows through a heat exchanger cooled by RBCCW. The sump fluid is automatically recirculated on a signal from a temperature sensor in the sump, in order to protect radwaste-system resins from deleterious overheating. The sump liquid setpoint is 135°F. Periodic sump discharge is initiated automatically on a signal from the sump level controller. The discharge header to the radwaste waste collector tank penetrates the primary containment wall. In order to preserve the integrity of primary containment, this line is sealed by the submerged pump suction lines inside primary containment.

These lines are also protected by one air-operated isolation valve and one motor-operated isolation valve installed in tandem in the discharge header; one valve is located inside

containment, the other outside. Each valve is fed from a different division. These isolation valves are automatically closed by a rise in pressure inside primary containment and by other primary containment isolation signals (See Table 6.2-2).

Equipment drains from secondary containment spaces in the reactor building and auxiliary building are also collected and discharged to the radwaste waste collector tank. Two drain sumps, each holding 1500 gal (nominal capacity), are provided, each with twin submersible pumps and bypass heat exchangers.

The fourth drain system in the reactor building draws from a trench drain and an undervessel drain and exhausts to the radwaste floor drain collector tank. This system is similar to the equipment drain systems located in the drywell discussed previously. Dual isolation valves ensure the integrity of primary containment, but the bypass cooling heat exchanger is omitted. Sump capacity is 1000 gal (nominal capacity).

The fifth and sixth drain systems in the reactor building collect from the floor drains in secondary containment areas and exhaust through twin parallel pumps to the radwaste floor drain collector tank. These systems, like the other floor drain system, have no sump cooling provision. Each sump has a 1500-gal nominal capacity.

The seventh reactor building drain system consists of a sump in the torus area (with no collection piping). This system discharges through twin parallel transfer pumps and an external water seal to the radwaste floor drain collector tank. This sump has a 900-gal nominal capacity.

Equipment and floor drains in the emergency core cooling system (ECCS) pump rooms, in the subbasement of the reactor building, have been physically separated to prevent possible flooding between ECCS Division I and Division II equipment through the drain lines in the event of an accident that causes one of the rooms to flood.

Equipment and floor drains in the emergency core cooling system (ECCS) pump rooms,in the sub-basement of the reactor building, are collected in each room's sump to prevent possible inter-divisional flooding between ECCS Division I and Division II, with the exception of HPCI room. The floor and equipment drains from the HPCI room are collected in the RHR Division II pump room sump. A motor operated auto-close flood control valve is provided in the floor and equipment drain line to prevent possible flooding between the two rooms in the event of an accident that causes one of the rooms to flood. The flood control valve will normally be open, but will close on high-high sump level to prevent water from backing up into the subbasement floor and equipment drains. The valves will reopen on low sump level. Selected RHR pump and rack H21-P596B drains in the southwest corner room are hard piped to the southeast corner sump, but are isolated by normally closed manual valves.

Flooding of any individual corner room or the HPCI room due to a line break in either room can be confined to that corner room. The configuration of the motor operated flood control valve and its associated sump is shown in Figure 9.3-6. The level switch data for the motor operated flood control valve is given in Table 9.3-5.

The motor-operated flood-control valve and its limit switches are tested periodically to ensure their satisfactory performance. This testing is done as required by the Performance Evaluation Procedures of the overall plant surveillance program. Maintenance procedures

cover the testing of the valves. Switches and other pertinent instrumentation are covered by a section of the overall balance-of-plant (BOP) preventive maintenance program.

The turbine building has eight separate radioactive drain collection systems, each with an independent sump. The drain system is shown in Figures 9.3-8 and 9.3-9.

Two equipment drain sumps, with nominal capacities of 400 and 4400 gal, collect oil-free radioactive liquids from equipment and piping systems. Each sump has twin 50-gpm sump pumps that periodically discharge to the waste collector tank in the radwaste building.

A third 2300-gal nominal capacity service water drain sump is provided to collect nonradioactive liquids from such systems as the general service water (GSW) system, and TBCCWS piping. This sump can be emptied into the liquid waste holding pond in the yard.

Two floor drain sumps, with nominal 1600-gal and 4400-gal capacities, are provided to collect oil-free liquids, and each has twin 50-gpm sump pumps discharging to the floor drain collector tank in the radwaste building.

Finally, three sumps, with approximate capacities of 1900, 2200, and 3000 gal, are provided to collect oil-contaminated liquids. These sumps are each provided with two 50-gpm to 64-gpm pumps as well as a 200-gpm or a 250-gpm emergency pump. The discharge is normally routed to an oil-water separator prior to treatment in the radwaste building. The emergency pumps can be used to empty the sumps to the liquid waste holding pond, if desired.

The radwaste building contains an equipment and a floor drain sump, each with a 900-gal nominal capacity. First-floor leakages drain directly into the waste collector tank or into the floor drain collector tank located in the basement. Basement leakages are collected in the appropriate sump and pumped out by twin 50-gpm pumps. The system drains are shown in Figures 9.3-10 and 9.3-11.

The RHR complex drain system is segregated into two types of wastes, oil-free water and oil-contaminated water. Equipment and floor drains that are potentially contaminated with oil drain to a manway which is connected by an overflow line to the liquid waste holding pond. Equipment and floor drains that are oil free drain to another manway which is connected by an overflow line to the circulating water reservoir. The piping pits are provided with sump pumps which discharge to the clear-water manway. The RHR system drains are shown in Figure 9.3-12.

9.3.3.3 Safety Evaluation

All potentially contaminated internal drain water is processed through the radwaste purification trains before release or recycle. The integrity of primary containment is ensured by tandem isolation valves. The drainage system is protected from overpressure by open sight funnel drains at most collection points.

To further ensure performance, the high-temperature drains in the reactor building are cooled by the RBCCWS. This ensures an acceptable net positive suction head (NPSH) at the transfer pumps.

9.3.3.4 Tests and Inspections

The drain lines are all welded and all required tests for joint soundness were carried out in accordance with applicable codes. For this reason, the closing field welds are in accessible positions.

Because spare pumps are installed, no periodic qualifying tests were undertaken. Completed piping has been hydrostatically tested in the field.

9.3.3.5 Instrumentation

Each sump is equipped with a high-high-level alarm to signal automatic initiation of the second pump. The starting of the second pump would be indicative of a system leakage. In addition, temperature controls are provided to cool critical sumps by actuating the flow of sump water through heat exchangers.

All reactor building sumps have leak-detection instrumentation. Timers monitor the operation of the sump pumps both for frequency and for length of operation. Leakage is detected by a pump operating before the timers time out or by a pump operating too long. Leakage is alarmed in the main control room.

9.3.4 Chemical, Volume Control, and Liquid Poison Systems

The only BWR systems that are related to this general class of systems are the standby liquid control system (SLCS) and the reactor water cleanup (RWCU) system.

The SLCS is described in Subsection 4.5.2.4 and the RWCU system is described in Subsection 5.5.8.

9.3.5 Failed Fuel Detection System

In the event of gross rod failure, the increased activity in the coolant would be transferred to the steam and detected by the main steam line RMS. Downstream of the steam line monitors are the offgas RMS and the reactor building exhaust vent RMS. The design bases, system description, safety evaluation, tests and inspections, and instrumentation applications for each of these subsystems are found in Section 11.4.

9.3.6 Nitrogen Inerting System

9.3.6.1 Design Bases

The Fermi 2 nitrogen inerting system provides and maintains a nitrogen atmosphere inside the primary containment and also provides pressurized nitrogen for pneumatic service inside the primary containment and distribution throughout the plant. The system schematic is shown in Figures 9.3-13 and 9.3-14.

The nitrogen inerting system supply is located outside the reactor building on the west side. The components are shown in Figure 9.3-15. The remainder of the system is located in the reactor building. The nitrogen inerting system supplies nitrogen gas at the proper pressure

and temperature for inerting the primary containment and for distribution throughout the plant.

The nitrogen inerting system design requirements are the following:

a. To provide nitrogen gas at the proper temperature and pressure to inert the primary containment to a minimum of 97 percent by volume of nitrogen. The nitrogen gas will be injected into the primary containment and the existing atmosphere will be displaced out through the reactor/auxiliary building ventilation system or through the SGTS. Mixing of the injected nitrogen will be accomplished by the use of the drywell cooling system (see Subsection 9.4.5).

b. To provide nitrogen makeup for atmospheric leakage out of the primary containment during normal operation and to ensure that a positive pressure is maintained inside the primary containment with respect to the secondary containment. Makeup requirements to some degree will be taken care of by the bleed-off of nitrogen gas from the pneumatic instrumentation inside the primary containment. Provisions for nitrogen addition to the primary containment atmosphere have been made at the drywell and suppression chamber supply lines through a separate on-line purge system. This system controls the pressure of the drywell and torus through vent/makeup of nitrogen

c. To provide nitrogen gas for the pressurized distribution system for the following services:

    1. To provide pressurized nitrogen for the pneumatic instrumentation inside the primary containment. During normal operation, nitrogen will be supplied to this instrumentation from the nitrogen inerting system. In the event of a loss of nitrogen supply, bottled nitrogen will be available for emergency use for the pneumatic requirements inside the primary containment

    2. To provide pressurized nitrogen to any other remaining services requiring nitrogen throughout the plant.

Air purging of the primary containment to the breathable limit will be accomplished by the use of the reactor/auxiliary building ventilation system or the SGTS.

9.3.6.2 System Design

The nitrogen inerting system primary containment penetrations and the associated isolation valves are classified as ASME Section III, Class 2. The pneumatic supply system inside primary containment is classified as ASME Section III, Class 3. The balance of the nitrogen inerting system pressure vessels are classified as ASME Section VIII, and the piping is classified as ANSI B31.1.0.

The nitrogen inerting system primary containment penetrations and associated isolation valves and the pneumatic supply system inside primary containment are designated as Category I. The remainder of the system is classified as nonseismic.

The nitrogen inerting system has been designed in accordance with the following criteria.

a. Liquid nitrogen requirements are based on the following usage:

   1. To inert the primary containment to less than 3 percent by volume of oxygen

   2. To provide additional nitrogen to the primary containment to compensate for leakage.

   3. To provide nitrogen for the pressurized distribution system.

b. The inerting and air purging procedures for the primary containment will be completed in approximately 6 hr.

c. The minimum distribution temperature of the nitrogen gas for all phases of operation of the nitrogen inerting system is controlled. The vaporizing medium during the primary containment inerting procedure is saturated steam at 15 psig from the plant auxiliary boilers. Heat for the pressurized distribution system will be provided electrically

d. The design capacity of the liquid storage tank is based on the service requirement of the pressurized distribution system for Fermi 2 and the vapor loss from the storage tank during the interval of storage

e. The receiver usable capacity will be designed to allow a system flow rate of 50 cfm for a period of 5 minutes if the liquid nitrogen source should be out of service. A full-capacity standby receiver is available

f. The pressurized distribution system is designed to allow connection of bottles as a backup source of nitrogen

g. The design flow of the nitrogen gas to the primary containment for the inerting procedure is 3000 scfm.

9.3.6.3 <u>Design Evaluation</u>

The system fluid will be commercial 99-percent pure nitrogen. The fluid will not be radioactive. System components for the handling of liquid nitrogen have been constructed of materials suitable for temperatures of -320°F.

The liquid nitrogen storage tank provides the source of supply for pressurized nitrogen distribution. The tank is equipped with a pressure build coil and an auxiliary pressure build vaporizer. The pressure build coil will transfer heat to the liquid nitrogen to generate saturated nitrogen vapor.

The nitrogen inerting system has a steam vaporizer and electric heat exchanger. The steam vaporizer will be used only when nitrogen is required for the inerting of the primary containment. The electric heat exchanger is used to supply gaseous nitrogen for pressurized distribution.

The nitrogen receivers provide temporary storage to meet sudden demands for pressurized nitrogen throughout the plant. One receiver will be in full standby to allow maintenance without disturbing normal plant operation.

The source of system pressure is the liquid nitrogen storage tank. The vapor pressure in the tank will be regulated to provide the required system pressure. All pressure-retaining components of the system are equipped with properly sized pressure relief valves. Piping that is handling liquid nitrogen has pressure relief valves installed in any segment where liquid nitrogen could become entrapped between closed valves. All liquid nitrogen transfer lines are sloped upward in the direction of flow to prevent vapor pocket buildup at the nitrogen source.

The nitrogen inerting system is not required for the safe shutdown of the reactor, and hence is not required to protect the health and safety of the public. However, the continuous operation of the plant is contingent upon the nitrogen inerting system maintaining the required nitrogen atmosphere inside the primary containment. Therefore, to ensure that nitrogen gas is always available to meet the primary containment nitrogen requirements, small amounts of bottled, high-pressure nitrogen will be stored at the site as a secondary source of nitrogen supply.

9.3.6.4 Tests and Inspections

The liquid storage and vaporizing facilities for the nitrogen inerting system are located outside the reactor building and are accessible for inspection. The nitrogen receiver tanks and bottled nitrogen tanks are located in the reactor building and are accessible for inspection during normal plant operation. Initial system checks, valve operability, instrumentation and control loop checks, and alarm setpoints for the nitrogen inerting system were done in accordance with the Preoperational Test program as discussed in Chapter 14. The temperature and pressure of nitrogen delivered by the steam and electric vaporizers have also been determined.

Periodic inspections of receiver tanks and the passive Division II backup nitrogen supply system will be performed. The inspection of instruments will be made to confirm the actuation of relief valves and alarms. The system and its components will be periodically tested and maintained as appropriate for the system safety classification.

9.3.6.5 Instrumentation Requirements

When the primary containment is being inerted, pressure and temperature control will be maintained in the following manner:

a. A pressure control valve located downstream of the liquid storage tank discharge and the steam vaporizer automatically maintains a discharge pressure of approximately 30 psig

b. A temperature indicator is located in the condensate discharge line of the steam vaporizer as is a low-temperature switch that shuts down the nitrogen discharge from the vaporizer at preset temperature.

Pressure and temperature control of the pressurized nitrogen distribution system will be maintained as follows:

a. A pressure control station located between the liquid storage tank and the electric heat exchanger automatically maintains a downstream pressure of approximately 110 psig

b. A variable setpoint temperature controller on the discharge side of the electric heat exchanger maintains a nitrogen discharge temperature

c. A pressure control station located downstream of the receivers maintains a downstream pressure

d. The drywell makeup station will sense the pressure of the primary containment and the secondary containment and with manual action, a positive pressure will be maintained in the primary containment

e. The provision for a bottle backup station will include a manually operated pressure regulator to maintain the receiver pressure when required. However, Division II is backed up by a passive nitrogen supply using bottles in the secondary containment. This capability supports manual operation of Division II SRVs from the control room for certain post-fire shutdowns requiring low pressure makeup systems

f. A pressure indicator is provided to monitor pressure downstream of the receivers. When the pressure of the receiver in operation reaches a setpoint, an alarm is provided to indicate low receiver pressure.

The primary containment isolation valves will automatically isolate on high drywell pressure, low reactor level, or high reactor building radiation.

TABLE 9.3-1 PROCESS SAMPLING SYSTEM

Item No.	Description	Sample tap Location	Purpose of Sample	Type of Analysis	Analysis Mode	Source Operating Temperature (°F)	Source Operating Pressure (psig)
1	TBCCW (2)	E and W heat exchanger outlet	Tube leaks	Conductivity Laboratory	Local grab	95/130	35
2	Condenser	Condenser leak troughs	Spare tap not used				
3	Condensate (3)	Condensate pumps discharge north-center-south	Condensate quality and tube leaks	Laboratory	Grab station	94	213
4	Condensate	Condensate pumps discharge header	Spare tap not used			94	213
5	Condenser (6)	Condenser leak troughs inlet and outlet each quadrant turbine	Spare taps -not used			100	-14
6	Condenser	Condenser leak trough	Spare tap not used			125	147
7	Condenser circulating water system (2) A or B	NE and SE water box influent	Water analysis	pH, Biocide Residual laboratory	Local Grab	95	50
8	Condenser circulating water system (2) A or B	E and W water boxes effluent	Water analysis	Biocide Residual Conductivity, pH, Total solids Laboratory	Local Grab	100	50
9	Condensate polishing demineralizer	Polishing demineralizer inlet header	Condensate quality and tube leaks	Conductivity Cation, Dissolved 0_2 Corrosion Products Laboratory	Continuous Grab station	94	213
10	Condensate polishing demineralizer	Polishing demineralizer outlet header	Treated condensate quality	Conductivity Dissolved 0_2 Corrosion Products Laboratory	Continuous Grab station	94	213
11	Feedwater heaters (4) 11, 11A, 11B, 11C	No. 2 FWH effluent header 2N, 2C, 2S	Water analysis	Laboratory	Grab station	388 170	498 634
12	Reactor feedwater system (2) A and B	After last heater 6A and 6B (2)	Water analysis	Laboratory	Continuous Grab station	425	1106
13	Heater feed	Heater feedpump discharge header	Water analysis	Laboratory	Grab station	94	700
14	Main steam (2) (A or B)	Main steam line	Steam conditions	Conductivity Laboratory	Continuous Grab station	549	1020
15	Drains cooler	Drain discharge to condenser	Water analysis	Laboratory	Local grab	134 104	-12 psia
16	Deaerating No. 5 heater (2) drain	No. 5N + 5S drain outlet	Drain water quality for pumping drains forward	Corrosion Products when required Laboratory	Local grab or tie continuous with item 25	392	224 210
17	Feedwater heaters (12)	Condensate inlet and outlet to heaters	Water analysis	Laboratory	(a)	105-400	580 634
18	Feedwater heaters (12)	Drains inlet and outlet to heaters	Water analysis	Laboratory	(b)	105-400	-13.5-345

TABLE 9.3-1 PROCESS SAMPLING SYSTEM

Item No.	Description	Sample tap Location	Purpose of Sample	Type of Analysis	Analysis Mode	Source Operating Temperature (°F)	Source Operating Pressure (psig)
19	Condensate polishing demineralizer	Polishing demineralizer inlet header	Condensate quality	Conductivity Laboratory	Continuous Grab station	94	213
20	Condensate polishing demineralizer	Polishing demineralizer effluent header	Demineralizer efficiency	Conductivity Laboratory	Continuous Grab station	94	213
21	General service water header	Effluent header to circulating water	Water analysis	Biocide Residual	Local grab	85	80-125*
22	Reactor Feedwater System	36 in. header after heater 6A and 6B	Water analysis	Conductivity Dissolved 02 Turbidity Corrosion Products Laboratory Dissolved H2	Continuous Grab station	425	1116
23	Condensate (4)	Hotwell discharge pipe each quadrant, condenser	Tube leaks	Conductivity Sodium Laboratory	Continuous Grab station	91.7	-9
24	Circulating water decant	Circulating water decant line	Sample of decant to Lake Erie	Laboratory	Local grab	85	50
25	Feedwater heater drains	Heater drain pumps discharge header	Evaluating heater drain contribution to feedwater	Dissolved 0_2 Turbidity Corrosion Products	Continuous Grab station	392	791
26	Circulating water decant before radwaste	Discharge of decant pumps	Water analysis	Corrosion Products Laboratory	Local grab	95	50
27	Makeup demineralizer storage tank	Tap on tank	Tank water purity	Laboratory	Local grab	Ambient	Tank head
28	Condensate storage tank	Tap on tank	Tank water purity	Laboratory	Local grab	Ambient (>40°F)	Tank head
29	Condensate return tank	Tap on tank	Tank water purity	Laboratory	Local grab	Ambient (95°F)	Tank head
30	Inlet line to condensate return tank	CRT return line	CRT supply purity	Laboratory	Local grab	95	58
31	Torus water management	Discharge of torus water management pumps	Water analysis	Laboratory	Grab station	160	210
32	SCCW Chilled Water (3)	Outlet of chilled water evaporator	Water analysis	Laboratory	Local grab	60	100

(a) Local grab for Sample No. 17a, c, d and e

(b) Local grab for Sample No. 18c and d

* Bounds plant operating procedures

TABLE 9.3-2 PROCESS SAMPLING SYSTEM

Item No.	Description	Sample tap Location	Purpose of Sample	Type of Analysis	Analysis Mode	Source Operating Temperature (°F)	Source Operating Pressure (psig)
33	SCCW (1)	Chilled water common return	Water analysis	Laboratory	Local grab	76	100
34	RBCCW (3)	RBCCW Return Headers from EECW	Water analysis	Laboratory	Local grab	85	80
35	RBCCW (2)	Heat exchanger outlet (2) N and S	Tube leaks	Conductivity	Local grab	85	80
36	RBCCW	Pump discharge header	Tube leaks	Conductivity	Grab station	85	80
37	Reactor primary coolant water	Main recirculating system pipe	Monitor reactor water when cleanup is isolated	Conductivity Dissolved 0_2 pH, Corrosion Products Laboratory Dissolved H_2	Continuous Grab station	540	1230
38	Reactor water cleanup filter-demineralizer	Filter inlet pipe	Reactor water quality	Conductivity Laboratory	Continuous Grab station	120	1214
39	Reactor water cleanup filter-demineralizer (2) (A or B)	Filter outlet pipe	Demineralizer efficiency	Corrosion Products Conductivity Laboratory	Continuous Grab station	120	1214
40	Suppression pool (4)	RHR pump suction A, B, C, D	Water analysis	Laboratory	Local grab	90	Atm
41	Standby liquid control	Dip from tank	Test for boron concentration	Laboratory	Dip sample	90	Atm
42	Reactor shutdown cooling system (2) (A and B)	RHR heat exchanger outlet A & B	Water analysis	Conductivity Dissolved 0_2 pH, Laboratory Dissolved H_2	Continuous Grab station	335	480
43	Cleanup phase separator decant	Decant line to waste collector tank	Process data	Laboratory	Local grab	125	130
44	Cleanup phase separator sludge	Cleanup sludge discharge mix pump	Process data	Laboratory	Local grab	70-130	70
45	Fuel pool water	Dip from fuel Storage pool	Water analysis	Laboratory	Dip sample	70	Atm

TABLE 9.3-2 PROCESS SAMPLING SYSTEM

Item No.	Description	Sample tap Location	Purpose of Sample	Type of Analysis	Analysis Mode	Source Operating Temperature (°F)	Source Operating Pressure (psig)
46	EECW Division I Outlet (2)	Heat exchanger	Plate Leaks	Laboratory	Local grab	85-95	80
47	EECW Division II Outlet (2)	Heat exchanger	Plate Leaks	Laboratory	Local grab	85-95	80
48	Reactor water cleanup heat exchangers (2) cooling water	Cooling water (RBCCW) outlet, A and B RWCU heat exchanger	Tube leaks	Laboratory	Local grab	110	80
49	Fuel pool heat exchangers (2) and cooling water	Cooling water (RBCCW) outlet, A and B fuel pool cooling and clean- up heat exchanger	Tube leaks	Laboratory	Local grab	110	80
50	Reactor water cleanup	Cleanup pump discharge (RWCU Inlet)	Reactor water quality	Conductivity Dissolved 02 pH Corrosion Products Laboratory Dissolved H2	Continuous Grab station	537	1220
51	Reactor water cleanup	RWCU Outlet header (before addition to feedwater	Cleanup system operation	Corrosion Products Conductivity Laboratory	Continuous Grab station	537	1220
52	Spent fuel pool circulating system	Fuel pool pump discharge (2) A and B	Water quality	Laboratory	Local grab	130	130
53	Service water discharge from RBCCW heat exchangers	Service water discharge header from heat exchangers	Tube leaks	Laboratory	Local grab	100	80-125*
54	Control rod drive	CRD filter outlet	CRD water quality	Conductivity Dissolved oxygen	Grab Station		
55-56	Not used						

TABLE 9.3-2 PROCESS SAMPLING SYSTEM

Item No.	Description	Sample tap Location	Purpose of Sample	Type of Analysis	Analysis Mode	Source Operating Temperature (°F)	Source Operating Pressure (psig)
57	Reactor Water Cleanup	Cleanup Pump Suction	Reactor Water Quality	Laboratory Conductivity Dissolved O2 ph Dissolved H2	Grab Station, Continuous	537	1050

* Bounds Plant Operating Procedures

TABLE 9.3-3 PROCESS SAMPLING SYSTEM

Item No.	Description	Sample tap Location	Purpose of Sample	Type of Analysis	Analysis Mode	Source Operating Temperature (°F)	Source Operating Pressure (psig)
58	Floor drain demineralizer	Outlet pipe	Demineralizer efficiency	Laboratory	Local grab	140	40-140
59	Turbine building floor drain oil separator	Discharge to floor drain collector tank	Process data	Laboratory	Local grab		
60	Turbine building floor drain sumps (3)	Discharge to oil separator	Process data	Laboratory	Local grab		
61	Turbine building floor drain sumps (3)	Discharge to trash pond	Process data	Laboratory	Local grab		
62	Not used						
63	Floor drain sumps (7)	Sump pump discharge to floor drain collector tank	Process data	Laboratory	Local grab		
64	Equipment drain sumps (6)	Sump pump discharge to waste collector tank	Process data	Laboratory	Local grab		
65	Radwaste building emergency drains sump (5)	Sump pump discharge	Process data	Laboratory	Local gab	140	40-140
66	Radwaste evaporator (2) (A and B)	Concentrate pump discharge A and B	Process data	Laboratory	Local grab	165	40
67	Waste surge tank pump discharge	Waste surge tank	Process data	Laboratory	Grab station	140	40-140
68	Waste collector tank	Waste collector tank pump discharge	Process data	Laboratory	Grab station	140	40-140
69	Floor drain collector tank pump discharge	Floor drain collector tank pump discharge	Process data	Laboratory	Grab station	80	100
70	Not used						
71	Not used						
72	Waste sample tanks (2) (A and B)	Waste pump dis-charge	Discharge suitability	Laboratory	Grab station	40-140	40-140
73	Waste sample tank	Recirculating line to waste sample tank	Discharge suitability	Laboratory	Grab station	40-140	40-140

TABLE 9.3-3 PROCESS SAMPLING SYSTEM

						Source Operating	
						Temperature	Pressure
Item No.	Description	Sample tap Location	Purpose of Sample	Type of Analysis	Analysis Mode	(°F)	(psig)
74	Waste collector filter-demineralizer	Filter-demineralizer outlet	Filter-demineralizer efficiency	Laboratory	Local grab	140	140
75	Not used						
76	Waste demineralizer	Demineralizer outlet	Demineralizer efficiency	Laboratory	Local grab	140	40-140
77	Floor drain filter-demineralizer	Filter-demineralizer outlet	Filter-demineralizer efficiency	Laboratory	Local grab	140	40-140
78	Not used						
79	Condensate phase separator	Condensate decant pump discharge	Process data	Laboratory	Local grab	80	Atm
80	Chemical waste tank	Chemical waste pump discharge	Process data	Laboratory	Grab station	80	40
81	Fuel pool cooling and cleanup filter-demineralizer	Inlet pipe	Fuel pool water quality	Laboratory	Local grab	130	130
82	Fuel pool cooling and cleanup filter-demineralizer	Outlet pipe efficiency	Filter	Laboratory	Local grab	125	130
83	Not used						
84	Not used						
85	Distillate (2)(A and B) surge tank	Distillate surge tank (A and B)	Distillate data	Laboratory	Grab station	40	40
86	Radwaste effluent	Discharge line to decant line	Discharge data	Laboratory	Grab station	150	50
87 to100	See Table 9.3-4						
101	Waste collector etched-disk filter	Discharge to etched-disk filter	Filter efficiency	Laboratory	Local grab	40-140	55-167
102	Waste collector oil coalescer filter	Discharge of oil coalescer	Oil coalescer efficiency	Laboratory	Local grab	0-140	55-167
103	Floor drains etched-disk filter	Discharge of etched-disk filter	Filter efficiency	Laboratory	Local grab	40-140	22-100
104	Floor drains oil coalescer	Discharge of oil coalescer	Oil coalescer efficiency	Laboratory	Local grab	40-140	22-100

TABLE 9.3-3 PROCESS SAMPLING SYSTEM

Item No.	Description	Sample tap Location	Purpose of Sample	Type of Analysis	Analysis Mode	Source Operating Temperature (°F)	Source Operating Pressure (psig)
105	Distillate surge tank A	Discharge of distillate A pump	Distillate data	Laboratory	Local grab	40	40
106	Distillate surge tank B	Discharge of distillate B pump	Distillate data	Laboratory	Local grab	40	40
107	Evaporator feed surge tank	Discharge of evaporator feed-pumps	Process data	Laboratory	Grab station	80	100
108	Centrifuge	Decant line to waste clarifier	Process data	Laboratory	Local grab	140	Atm
109	Extruder/evaporator distillate	Discharge line to waste clarifier	Process data	Laboratory	Local grab	212	0
110	Floor drain demineralizer	Demineralizer outlet before strainer	Distillate quality	Conductivity	Continuous	140	40-140
111	Waste demineralizer	Waste demineralizer discharge	Distillate quality	Conductivity	Continuous	140	40-140
112	Waste collector oil coalescer filter	Discharge of oil coalescer	Water effluent quality	Conductivity	Continuous	40-140	55-167
113	Floor drain oil coalescer	Discharge of oil coalescer	Process data	Conductivity	Continuous	40-140	22-100
114 to 119	Not used						
120	Fuel pool cooling and cleanup demineralizer A	Demineralizer A effluent	Demineralizer efficiency	Conductivity	Continuous	140	40-140
121	Fuel pool cooling and cleanup demineralizer B	Demineralizer B effluent	Demineralizer efficiency	Conductivity	Continuous	140	40-140
122	Circulating water	Circulating water pumps discharge header	pH control	pH	Continuous with recirculating pump operation	60	50
123	Not used						
124	Floor drain demineralizer	Floor drain demineralizer outlet before recycle valve	Demineralizer efficiency	Conductivity	Continuous	140	40-140
125	Waste demineralizer	Waste demineralizer discharge	Demineralizer efficiency	Conductivity	Continuous	140	40-140

TABLE 9.3-3 PROCESS SAMPLING SYSTEM

Item No.	Description	Sample tap Location	Purpose of Sample	Type of Analysis	Analysis Mode	Source Operating Temperature (°F)	Source Operating Pressure (psig)
126	TBCCW supply	Discharge header from pumps	Process data	Laboratory	Local grab	95	50
127	Radwaste evaporator (2) (A and B)	Discharge lines from A and B distillate pumps to distillate coolers	Process data	Laboratory	Local grab	135	Atm
128	Evaporator drains holdup tank	Discharge line from evaporator drains pump	Process data	Laboratory	Local grab	165	Atm
129	Not used						
130	Radwaste system fuel pool filter-demineralizer A outlet	Line to fuel pool cooling cleanup system	To check water purity	Laboratory	Local grab	140	40-140
131	Radwaste system fuel pool filter-demineralizer B outlet	Line to fuel pool cooling cleanup system	To check water purity	Laboratory	Local grab	140	40-140
132	RHR heat exchanger B	Discharge to RPV	Water analysis	Conductivity	Continuous	335	480
133	RHR heat exchanger A	Discharge to RPV	Water analysis	Conductivity	Continuous	335	480
134	RHR heat exchanger B(service water)	Discharge to RHR	Radiation water tube leaks	Isotopic chloride	Continuous grab	155	80
135	RHR heat exchanger A(service water)	Discharge to RHR	Tube leaks	Isotopic chloride	continuous grab	155	80
136 to 151	Not used						
152	RHR Division I	RHR service water return, Division I	Tube leaks	Laboratory	Grab station	155	80
153	RHR Division II	RHR service water return, Division II	Tube leaks	Laboratory	Grab station	155	80
154 to157	Not used						
158	Main and reheat system	52-in. manifold	Spare tap			542	997
159	Not used						

TABLE 9.3-3 PROCESS SAMPLING SYSTEM

Item No.	Description	Sample tap Location	Purpose of Sample	Type of Analysis	Analysis Mode	Source Operating Temperature (°F)	Source Operating Pressure (psig)
160	Offgas vacuum and recombiner chain	20-in. manifold	Spare tap			94	-14.2
161	Offgas vacuum and recombiner chain	2.2-minute delay pipe from precooler	Monitor hydrogen and oxygen	Hydrogen oxygen	Continuous	70	-14.2
162	Offgas vacuum and recombiner chain	2.2-minute delay pipe from precooler	Spare tap			70	
163	Stator Winding Cooling de-oxygenating unit	Inlet/outlet of contactors	Monitor dissolved oxygen	Oxygen	Grab Sample	Ambient	80
164	Stator Winding Cooling demineralizer unit	Vent/drain stator winding cooling unit	Oxygen & metallic imputities	Conductivity	Grab Station	150	180
165	Station and control air	2-in. air header	Monitor control air moisture	Dewpoint hygrometer	Continuous	75	110
166 to 169	Not used						
170	Primary containment monitoring system	In reactor drywell	To check quality of reactor atmosphere	Hydrogen oxygen content	Continuous	135	2
171	Primary containment monitoring system	In reactor drywell	To check quality of reactor atmosphere	Hydrogen-oxygen content	Continuous	135	2
172	Primary containment monitoring system	In suppression pool	To check quality of atmosphere in suppression pool	Hydrogen-oygen content	Continuous	150	2
173	Primary containment monitoring system	In suppression pool	To check quality of atmosphere in Suppression pool	Hydrogen-oxgen content	Continuous	150	2
174	Not used						
175	Not used						

TABLE 9.3-4 PROCESS SAMPLING SYSTEM

Item No.	Description	Sample tap Location	Purpose of Sample	Type of Analysis	Analysis Mode	Source Operating Temperature (°F)	Source Operating Pressure (psig)
87	Auxiliary boiler steam (2)	N and S steam drum	Steam quality	Laboratory	Local grab	341	105
88	Auxiliary boiler feedwater	Feedwater inlet header	Feedwater quality	Laboratory	Local grab	220	125
89	Makeup demineralizer anion exchanger	Discharge from anion exchanger	Demineralizer efficiency	Conductivity	Continuous	80	50
90	Makeup demineralizer mixed bed	Discharge from mixed- bed exchanger	Demineralizer efficiency	Conductivity	Continuous	80	40
91	Makeup demineralizer	Makeup demineralizer outlet	Demineralizer efficiency	Conductivity	Continuous	80	40
92	Makeup demineralizer potable water	Raw water booster pump discharge	Raw water data	Laboratory	Grab station	80	50
93	Makeup demineralizer carbon filter	Carbon filter discharge	Filter efficiency	Laboratory	Local grab	80	65
94	Makeup demineralizer cation exchanger (2)	Discharge from cation exchanger	Demineralizer efficiency	Laboratory	Grab station	80	60
95	Makeup demineralizer anion exchanger (2)	Discharge from anion exchanger	Demineralizer efficiency	Laboratory	Grab station local grab	80	50
96	Makeup demineralizer mixed bed (2)	Discharge from mixed bed exchanger	Demineralizer efficiency	Laboratory	Grab station local grab	80	40
97	Makeup demineralizer system	Makeup deminer-	Demineralizer alizer outlet	Laboratory efficiency	Grab station	80	40
98	Makeup demineralizer acid solution	Discharge to mixed bed and cation exchangers	Acid concentration	Laboratory	Grab station	80	50
99	Makeup demineralizer caustic solution	Discharge to mixed bed and anion exchangers	Process data caustic concentration	Laboratory	Grab station	80	50
100	Not used						

TABLE 9.3-5 REACTOR BUILDING: FLOOD CONTROL VALVE

Division	Sump	Isolation Valve	Level Switch[a]
II	DO76 (Floor and equip. drains)	T4500F601	LSE-N076-B

[a] Switch limit points:

High-high	45 in. (valve closes)
High	39 in.
Low	24 in. (valve opens)
Low-low	22 in.

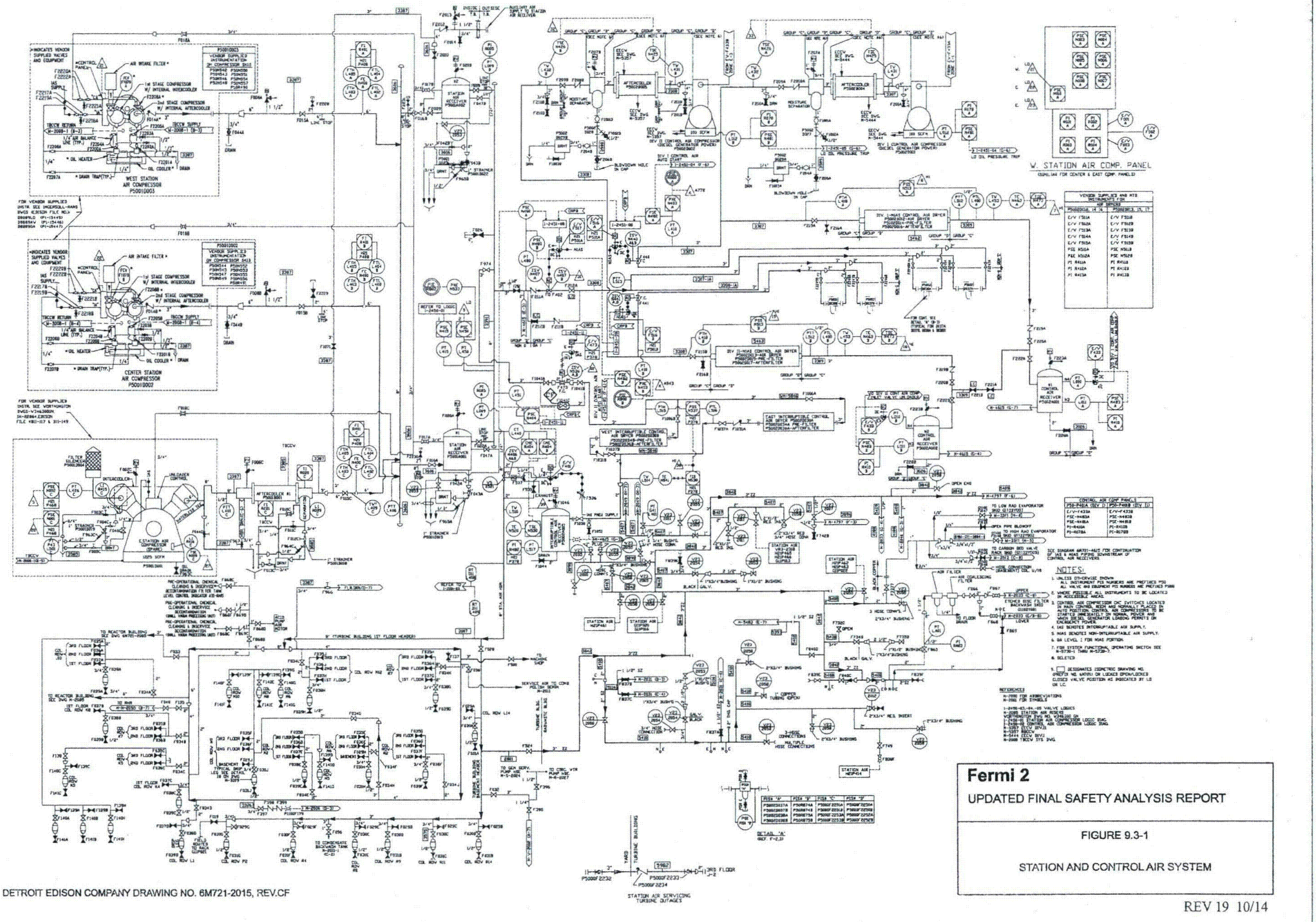

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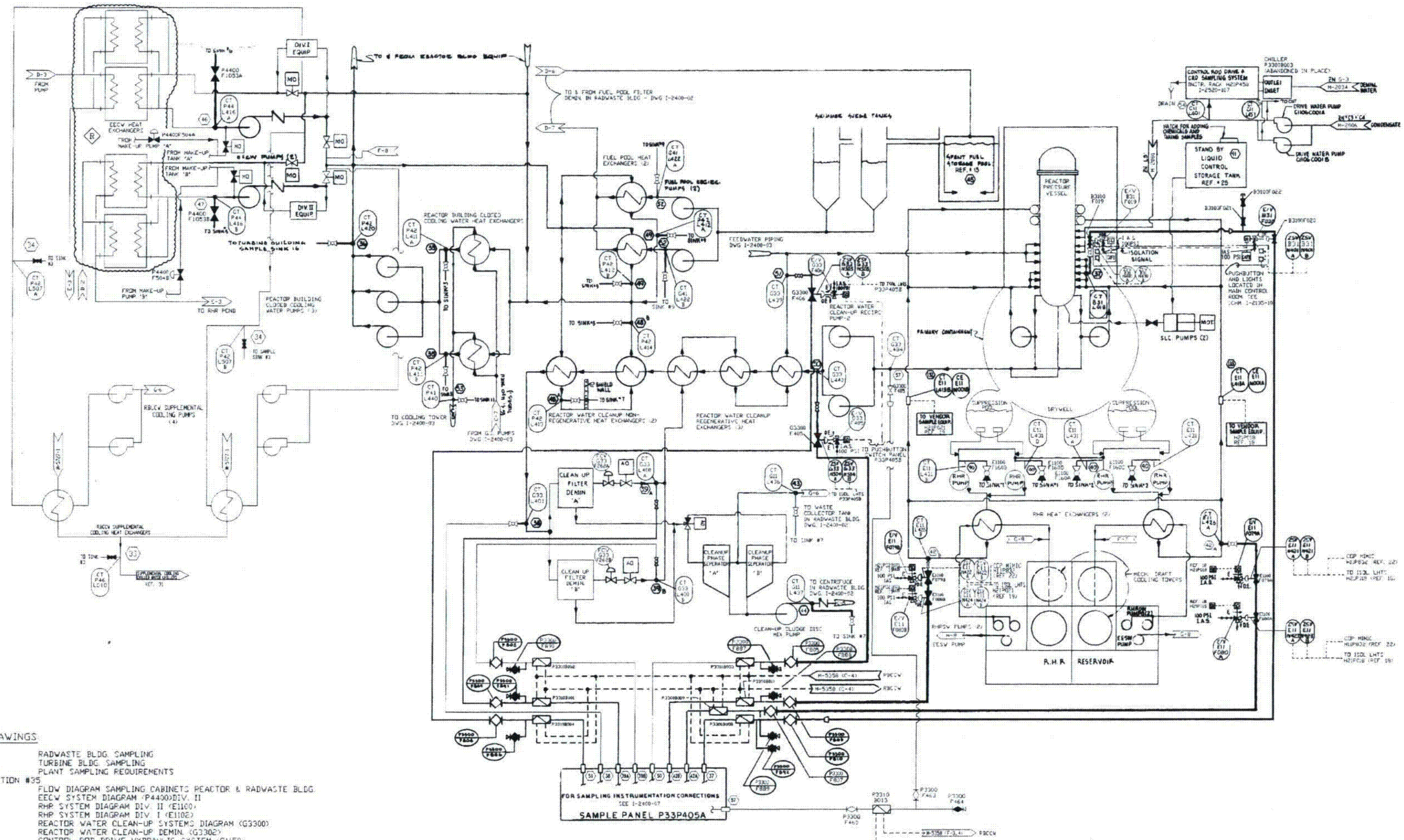
FIGURE 9.3-1

STATION AND CONTROL AIR SYSTEM



REFERENCE DRAWINGS

1. I-2400-02 RADWASTE BLDG. SAMPLING
2. I-2400-03 TURBINE BLDG. SAMPLING
3. I-2400-01 & 05 PLANT SAMPLING REQUIREMENTS
4. DESIGN INSTRUCTION #35
5. I-2400-07 FLOW DIAGRAM SAMPLING CABINETS REACTOR & RADWASTE BLDG.
6. M-5357 EECW SYSTEM DIAGRAM (P4400)DIV. II
7. M-2083 RHR SYSTEM DIAGRAM DIV. II (E1100)
8. M-2084 RHR SYSTEM DIAGRAM DIV. I (E1102)
9. M-2046 REACTOR WATER CLEAN-UP SYSTEMS DIAGRAM (G3300)
10. M-2047 REACTOR WATER CLEAN-UP DEMIN. (G3302)
11. M-2081 CONTROL ROD DRIVE HYDRAULIC SYSTEM (C1150)
12. M-5358 RBCCW SYSTEM DIAGRAM (P4200)
13. M-5444 EECW SYSTEM DIAGRAM DIV. I (P4400)
14. M-2010 GENERAL SERVICE WATER SYSTEM DIAGRAM (P4100)
15. M-2048 FUEL POOL CLEANING & CLEAN-UP SYSTEM DIAGRAM (G4100)
16. M-2088 REACTOR WATER CLEAN-UP PHASE SEPARATORS SYSTEM DIAGRAM (G1101)
17. M-2833 REACTOR RECIRC. NUCLEAR BOILER SYSTEM DIAGRAM (B3100)
18. I-2282-23 INSTRUMENT RACK H21P018 (WIRING DIAGRAM)
19. I-2282-24 INSTRUMENT RACK H21P021 (WIRING DIAGRAM)
20.
21.
22. I-2052-21 C.O.P. PANEL H11P832
23. I-2832-02 TUBE RTG. 2ND FLOOR REACTOR BUILDING
24. I-2520-107 INSTRUMENT RACK H21P450 CRD SAMPLING SYSTEM
25. M-2082 STANDBY LIQUID CONTROL SYSTEM DIAGRAM (C4100)
26. I-2281-17 INSTRUMENT RACK H21P018 (TUBING CONN.)
27. I-2281-19 INSTRUMENT RACK H21P021 (TUBING CONN.)
28. I-2401-03 ANALYTICAL INSTRUMENT DIAGRAM SAMPLE STATION P33P403A & B
29. M-2006 CONDENSATE STORAGE & TRANSFER SYSTEM (P1100)
30. M-2678 DEMINERALIZED SERVICE WATER RISERS FOR TURB. RADWST. & REAC. BLDG'S (P1100)
31. M-2020 SUPPLEMENTAL COOLING CHILLED WATER SYSTEM

NOTES:

1. ⬡ DESIGNATES SAMPLE TAP IDENTIFICATION LISTED ON I-2400-01 & 05.
2. FOR EXACT LOCATIONS OF SAMPLE TAPS SEE RESPECTIVE SYSTEM DIAGRAM FOR ASSOC. SYSTEM LISTED UNDER REFERENCE DRAWINGS.

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FIGURE 9.3-2

SAMPLES IN REACTOR BUILDING

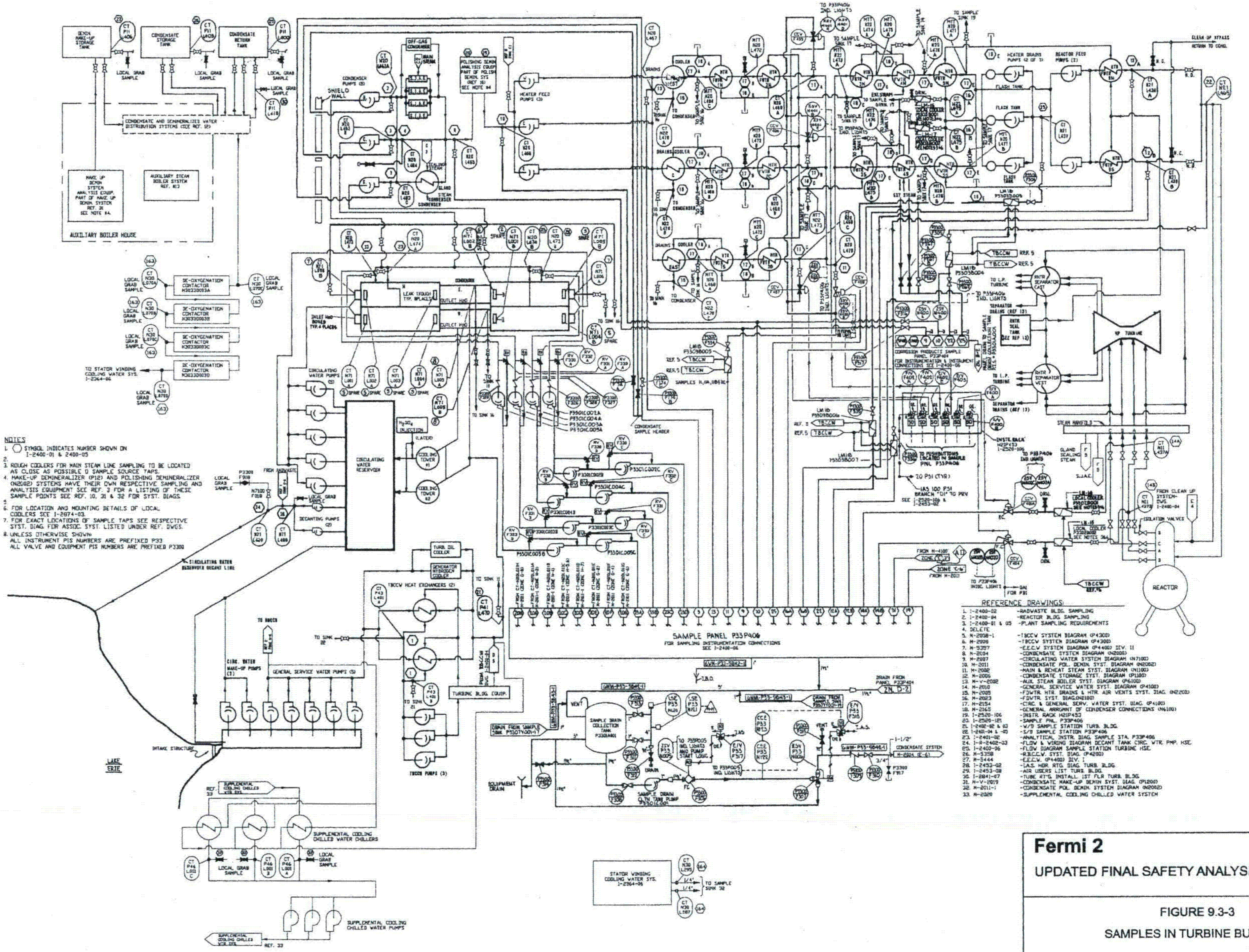

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FIGURE 9.3-3

SAMPLES IN TURBINE BUILDING

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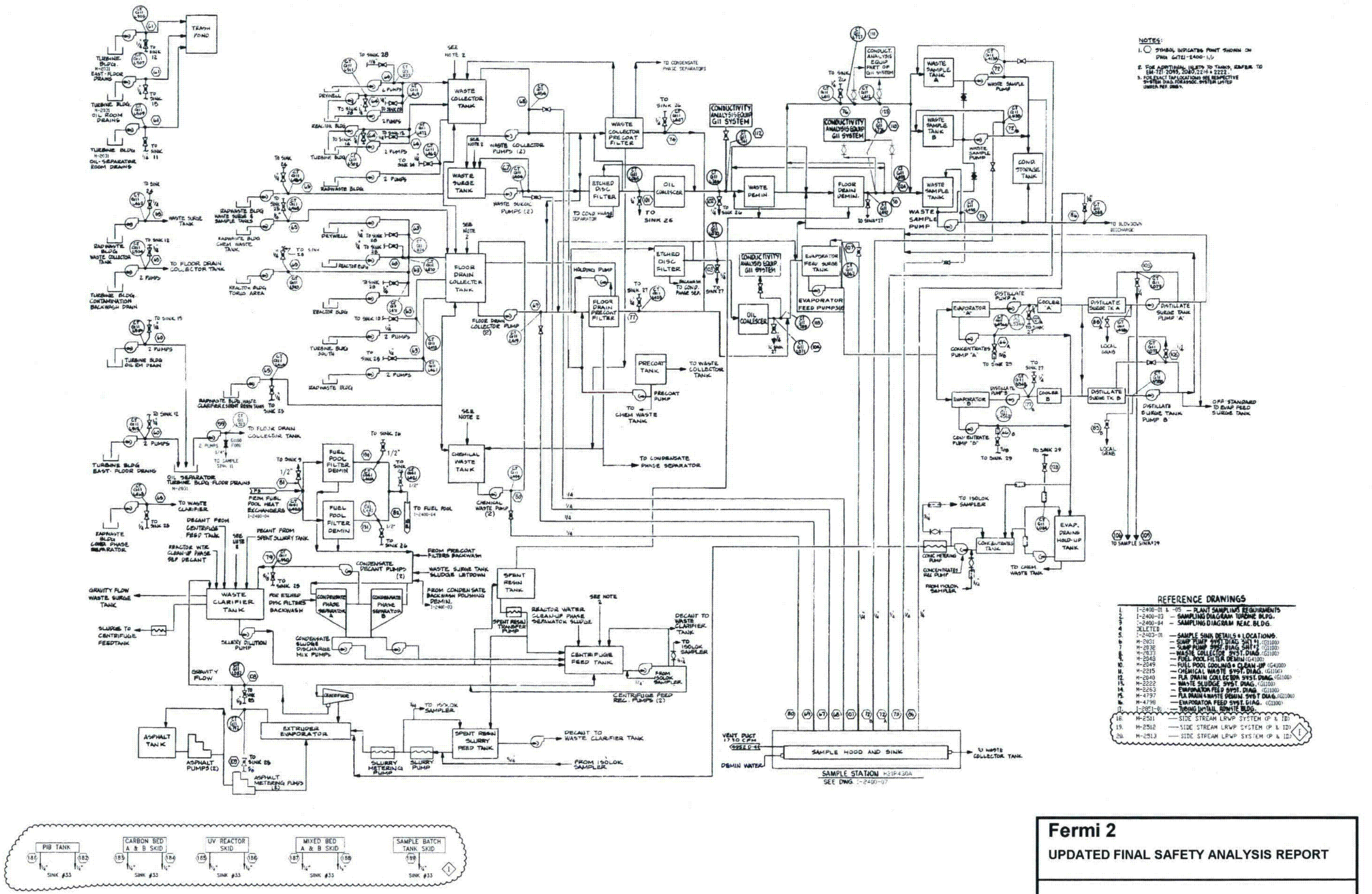

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FIGURE 9.3-4

SAMPLES IN RADWASTE BUILDING

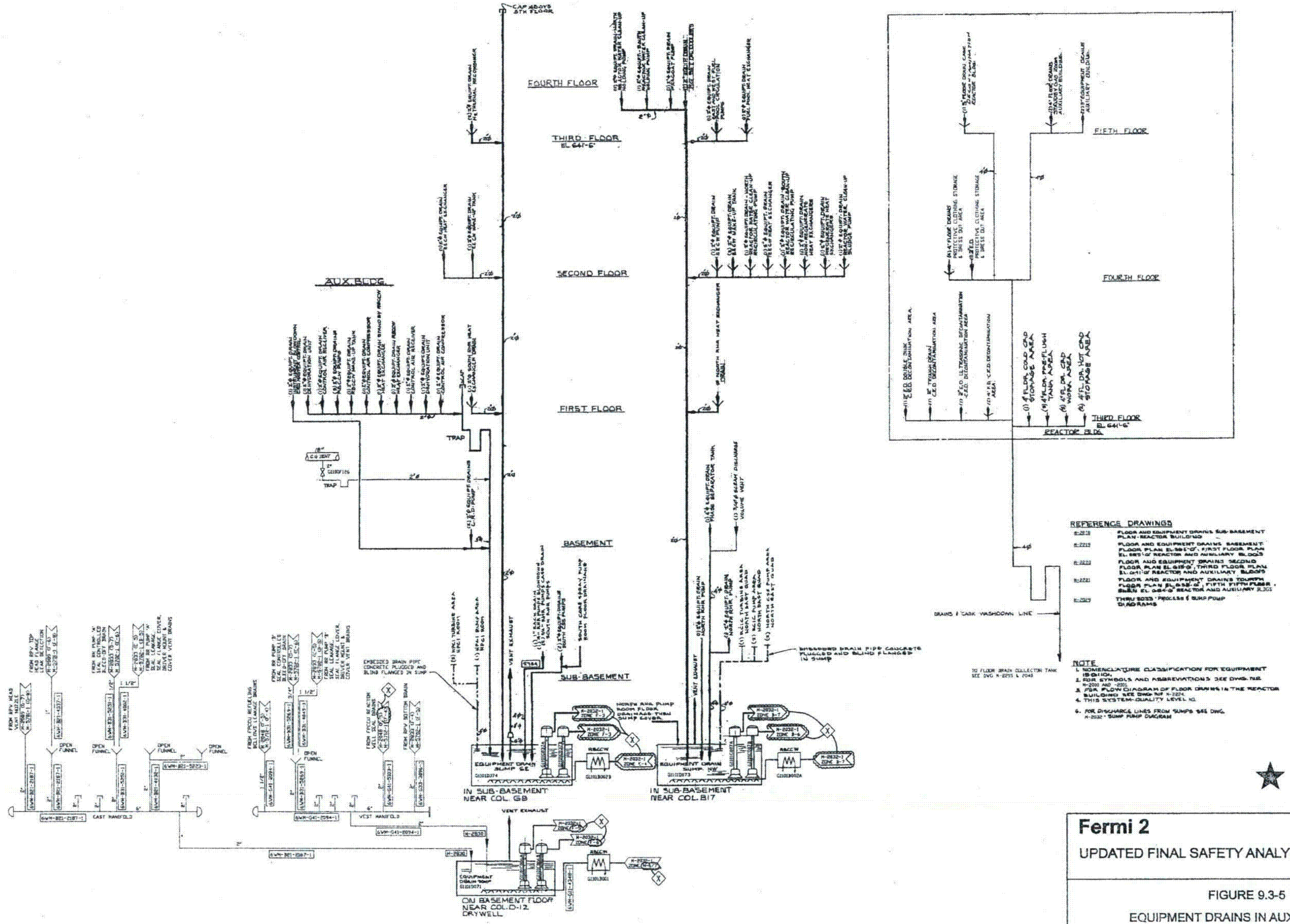

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FIGURE 9.3-5

EQUIPMENT DRAINS IN AUXILLARY AND REACTOR BUILDING

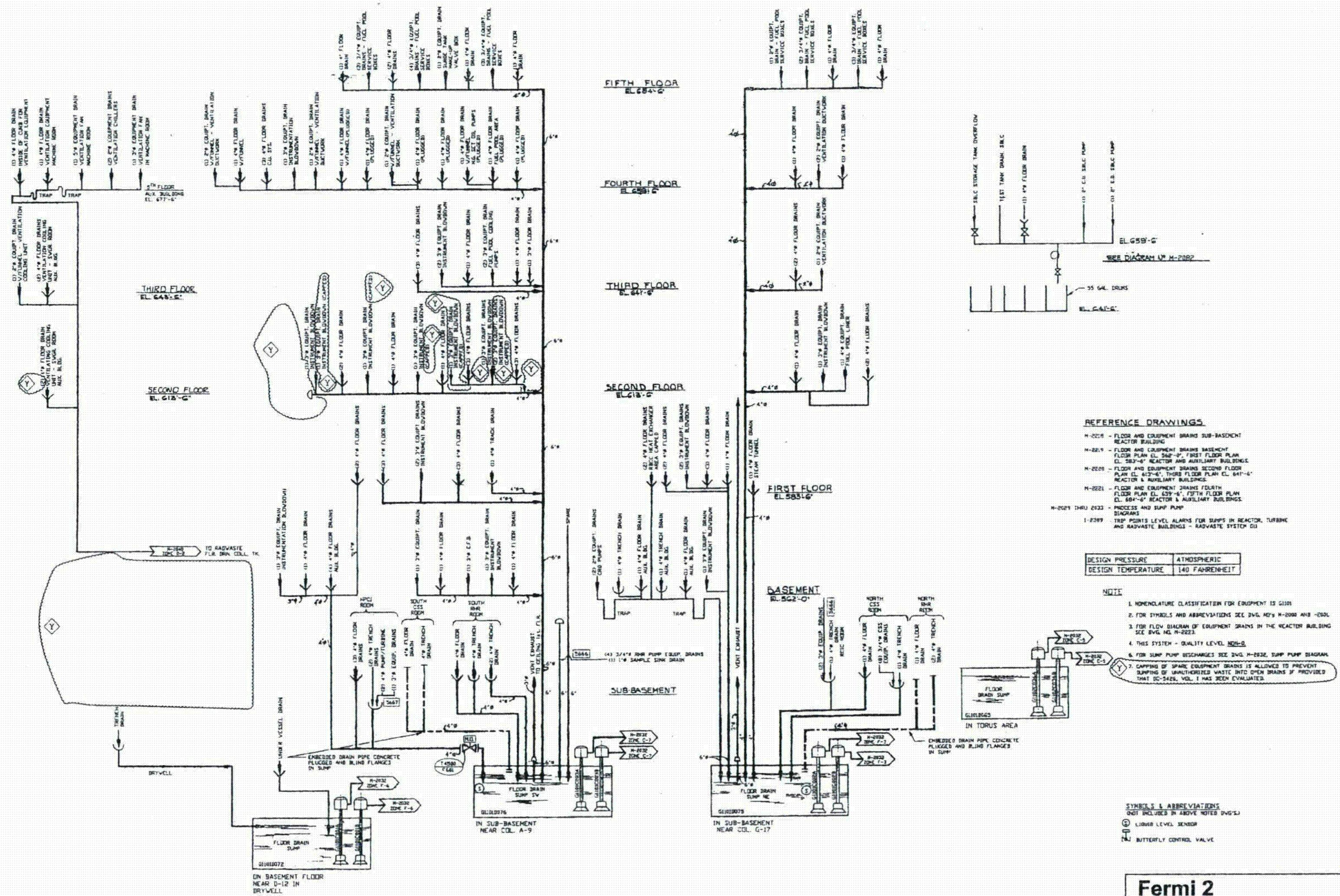

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FIGURE 9.3-6

FLOOR DRAINS IN AUXILIARY AND REACTOR BUILDINGS

REV 19 10/14

Redacted in accordance with 10 CFR 2.390

NOTES:

1. ALL PIPE ELEVATION ARE FOR THE CENTER OF PIPE
2. USE 1½ RADIUS, 90° ELBOW UNLESS OTHERWISE NOTED
3. ALL THE PIPES ENTERING THE SUMP SHALL BE SEAL WELDED
4. PITCH ALL LINES TO ELEVATIONS SHOWN
5. FOR TRENCH DRAIN SEE TYPICAL TRENCH DRAIN DETAIL 11 ON THIS DRAWING
6. FOR CONTINUATION OF ALL PIPING FROM ABOVE SEE DRAWING M-2219
7. [illegible]
8. DELETED
9. WELDING PROCEDURE SHALL BE IN ACCORDANCE WITH D.E. CO. STANDARD SPECIFICATION 3.11 REV 4
10. ALL CLEANOUTS SHALL BE FLUSH WITH FINISHED FLOOR
11. DELETED
12. TERMINATE EQUIPMENT DRAINS 12" ABOVE FINISHED FLOOR
13. ALL CLEANOUTS ARE OF 4" SIZE
14. FOR GENERAL CONDITIONS OF THIS CONTRACT SEE D.E. SPEC. 3071
15. DELETED
16. DELETED
17. ALL SYSTEMS ARE QUALITY LEVEL 3
18. OUT OF ROUND OPENINGS IN SUMP LINERS FOR PIPE INSERTION TO BE FIELD WELD FILLED AND DYE PENETRANT TESTED FOR APPROVAL.
19. 3/4" 3'-0" SECTION OF STYROFOAM TO BE CLAMPED EACH END & CENTER. 90° ELLS 3 PLACES, 45° ELLS 2 PLACES
20. IF NECESSARY CUT STYROFOAM TO CLEAR RE-STEEL
21. GENERAL PLANT IDENTIFICATION NUMBER FOR SUMPS IS G11
22. ALL FLOOR & GUTTER DRAINS S.S. CASTINGS ASTM SPEC A296 OR A351 TYPE 304
23. ALL PIPING 3" & LARGER S.S. SCH. 40 ASTM SPEC A312 TYPE 304
24. ALL PIPING 2½ & SMALLER S.S. SCH. 40 ASTM SPEC A312 TYPE 304
25. ALL FITTINGS 3" & LARGER S.S. SCH. 10-S ASTM SPEC A403 TYPE 304 OR 304 L
26. ALL B.W. FITTINGS 2" & SMALLER S.S. SCH. 40 ASTM SPEC A403 TYPE 304 "W.P."
27. DELETED
28. DELETED
29. FOR DETAILS AND SECTIONS NOT SHOWN ON THIS DRAWING SEE M-2225 & M-2226
30. WASTE SUMP DRAIN HUBS AND FUNNELS ARE CUT OFF FLUSH WITH FLOOR AND CONCRETE PLUGGED PER EDP-5299 (SIMILAR TO DETAIL H4-H4) & TYP PLUGGING DETAIL (6-2)
31. CAPPING OF SPARE EQUIPMENT DRAINS IS ALLOWED TO PREVENT DUMPING OF UNAUTHORIZED WASTE INTO OPEN DRAINS PROVIDED THAT DC-5426, VOL. I HAS BEEN EVALUATED

DETROIT EDISON COMPANY DRAWING NO. 7M721-2218, REV. X

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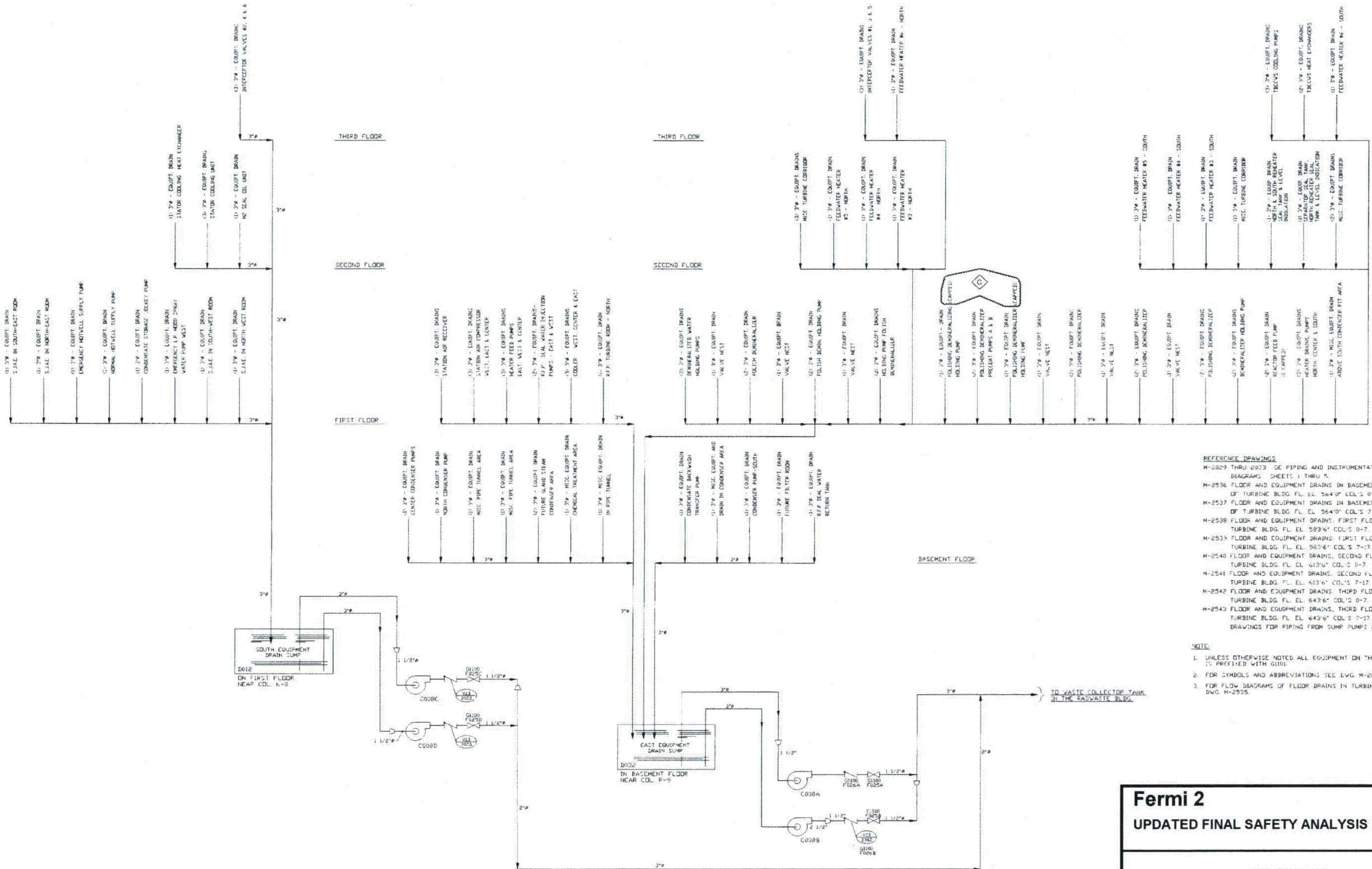
FIGURE 9.3-7

FLOOR AND EQUIPMENT DRAINS IN REACTOR BUILDING SUBBASEMENT



REFERENCE DRAWINGS

M-2029 THRU 2033 GE PIPING AND INSTRUMENTATION DIAGRAMS SHEETS 1 THRU 5.
M-2536 FLOOR AND EQUIPMENT DRAINS IN BASEMENT OF TURBINE BLDG. FL. EL. 564'0" COL'S 0-7.
M-2537 FLOOR AND EQUIPMENT DRAINS IN BASEMENT OF TURBINE BLDG. FL. EL. 564'0" COL'S 7-17
M-2538 FLOOR AND EQUIPMENT DRAINS, FIRST FLOOR TURBINE BLDG. FL. EL. 583'6" COL'S 0-7.
M-2539 FLOOR AND EQUIPMENT DRAINS, FIRST FLOOR TURBINE BLDG. FL. EL. 583'6" COL'S 7-17.
M-2540 FLOOR AND EQUIPMENT DRAINS, SECOND FLOOR TURBINE BLDG. FL. EL. 613'6" COL'S 0-7.
M-2541 FLOOR AND EQUIPMENT DRAINS, SECOND FLOOR TURBINE BLDG. FL. EL. 613'6" COL'S 7-17.
M-2542 FLOOR AND EQUIPMENT DRAINS, THIRD FLOOR TURBINE BLDG. FL. EL. 643'6" COL'S 0-7.
M-2543 FLOOR AND EQUIPMENT DRAINS, THIRD FLOOR TURBINE BLDG. FL. EL. 643'6" COL'S 7-17.
DRAWINGS FOR PIPING FROM SUMP PUMPS LATER.

NOTE:

1. UNLESS OTHERWISE NOTED ALL EQUIPMENT ON THIS DIAGRAM IS PREFIXED WITH G1101
2. FOR SYMBOLS AND ABBREVIATIONS SEE DWG. M-2000 & 2001
3. FOR FLOW DIAGRAMS OF FLOOR DRAINS IN TURBINE BLDG. SEE DWG. M-2535.

Fermi 2

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FIGURE 9.3-8

EQUIPMENT DRAINS IN TURBINE BUILDING

DETROIT EDISON COMPANY DRAWING NO. 6M721-2534, REV. G

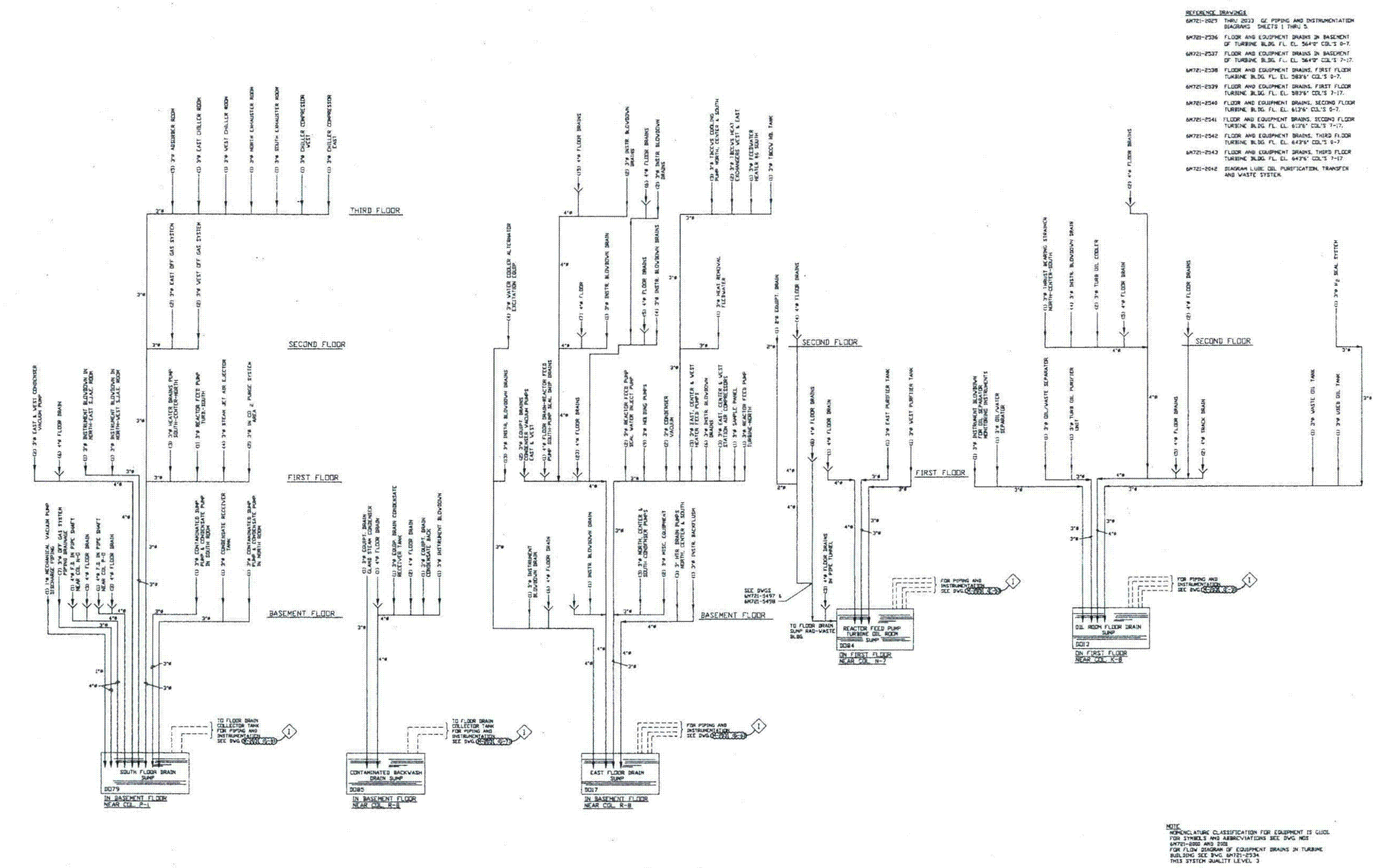

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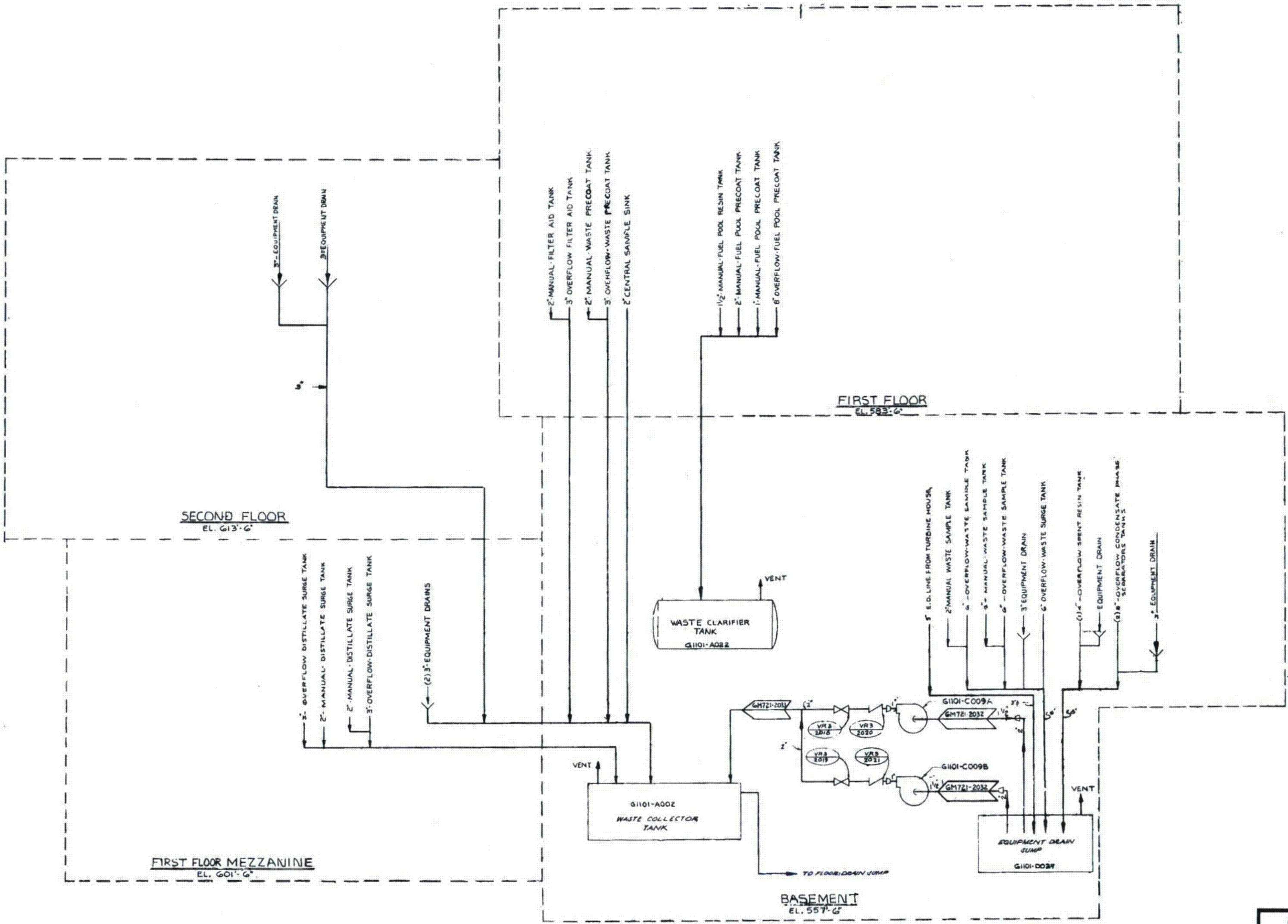
FIGURE 9.3-9

FLOOR DRAINS IN TURBINE BUILDING

DETROIT EDISON COMPANY DRAWING NO. 6M721-2535, REV. I



NOTE:

FOR SYMBOLS AND ABBREVIATIONS SEE DWG. NOS. 6M721-2000 AND -2001.
NOMENCLATURE CLASSIFICATION FOR EQUIPMENT IS G1101.
THIS SYSTEM QUALITY LEVEL 3.

Fermi 2

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FIGURE 9.3-10

EQUIPMENT DRAINS IN RADWASTE BUILDING

DETROIT EDISON COMPANY DRAWING NO. 6M721-2550, REV. F

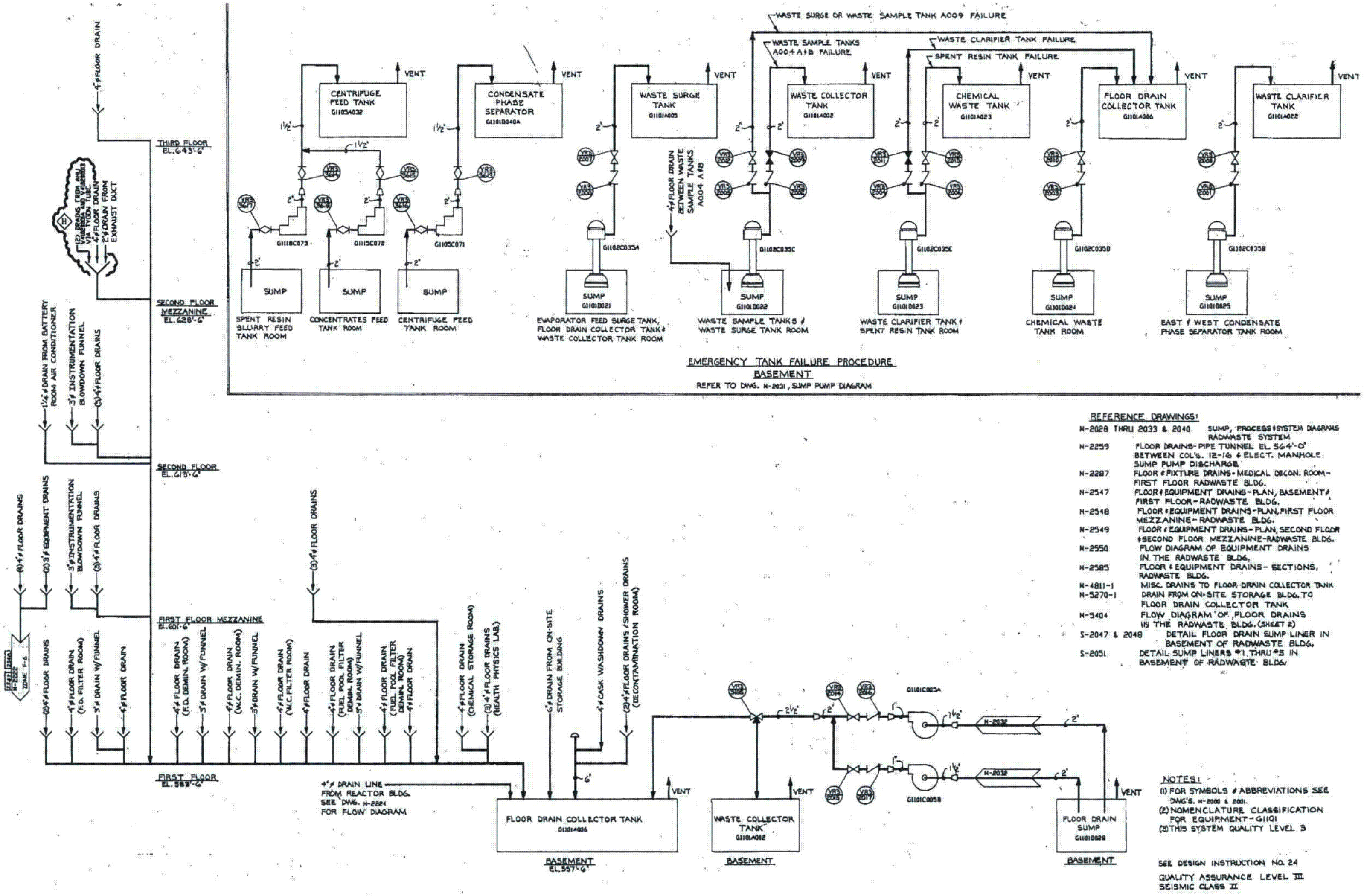

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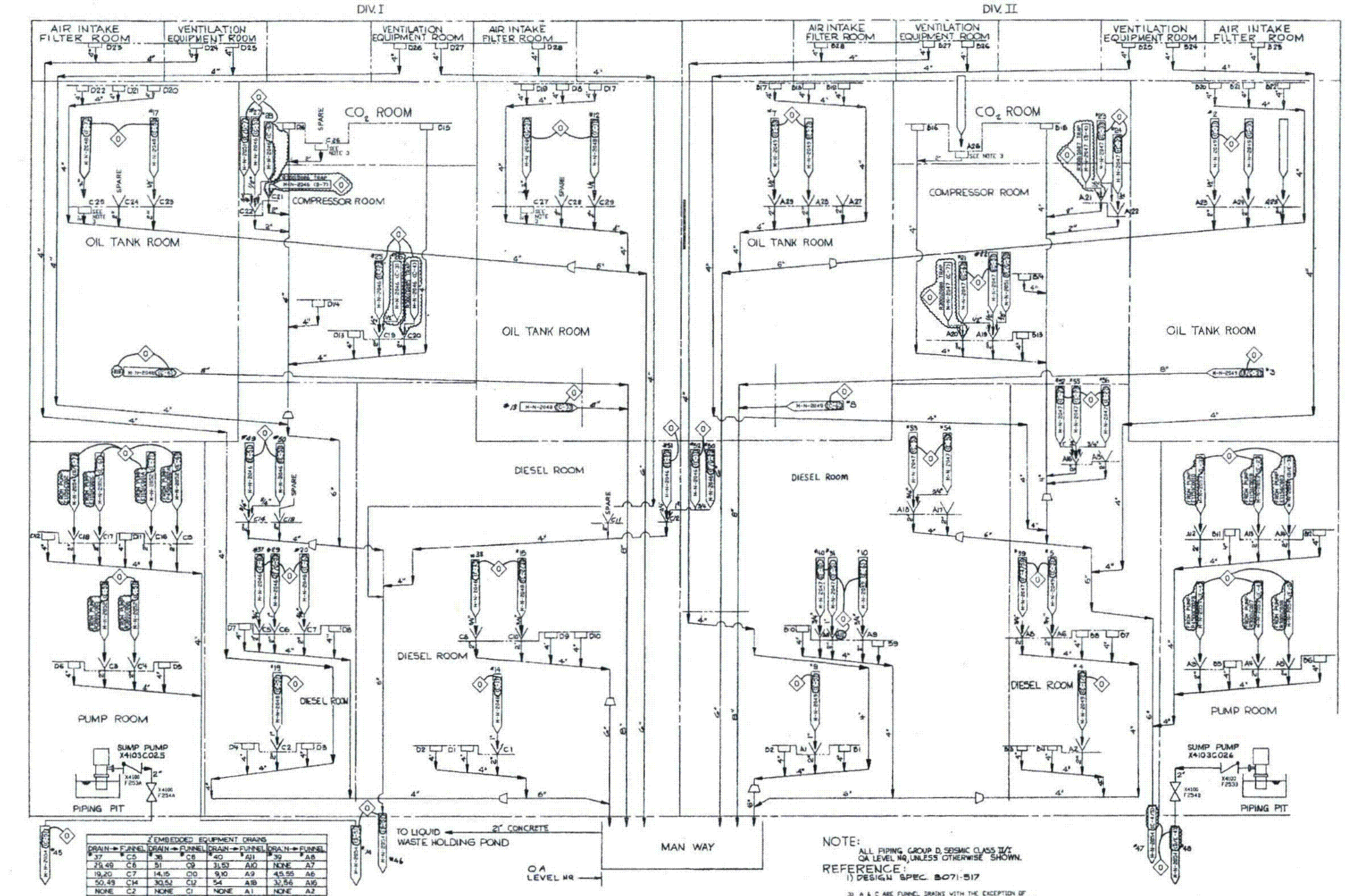
FIGURE 9.3-11

FLOOR DRAINS IN RADWASTE BUILDING

DETROIT EDISON COMPANY DRAWING NO. 6M721-2551, REV. H



2" EMBEDDED EQUIPMENT DRAINS							
DRAIN →	FUNNEL	DRAIN →	FUNNEL	DRAIN →	FUNNEL	DRAIN →	FUNNEL
#37	#C5	#38	#C8	#40	#A11	#39	#A8
29,49	C6	51	C9	31,53	A10	NONE	A7
19,20	C7	14,15	C10	9,10	A9	45,55	A6
50,49	C14	30,52	C12	54	A18	32,56	A16
NONE	C2	NONE	C1	NONE	A1	NONE	A2
NONE	C13	NONE	C11	NONE	A17	NONE	A15

NOTE:
ALL PIPING GROUP D, SEISMIC CLASS II/I QA LEVEL NQ, UNLESS OTHERWISE SHOWN.

REFERENCE:
1) DESIGN SPEC. 3071-517

3) A & C ARE FUNNEL DRAINS WITH THE EXCEPTION OF DRAINS A26, C25, C26 AND C27 WHICH ARE TRUNCATED AT OR NEAR THE FLOOR.

4) B & D ARE FLOOR DRAINS

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Fermi 2

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FIGURE 9.3-12

EQUIPMENT DRAINS AND FLOOR DRAINS (DIVISIONS I AND II) RESIDUAL HEAT REMOVAL COMPLEX - P&ID

REV 19 10/14

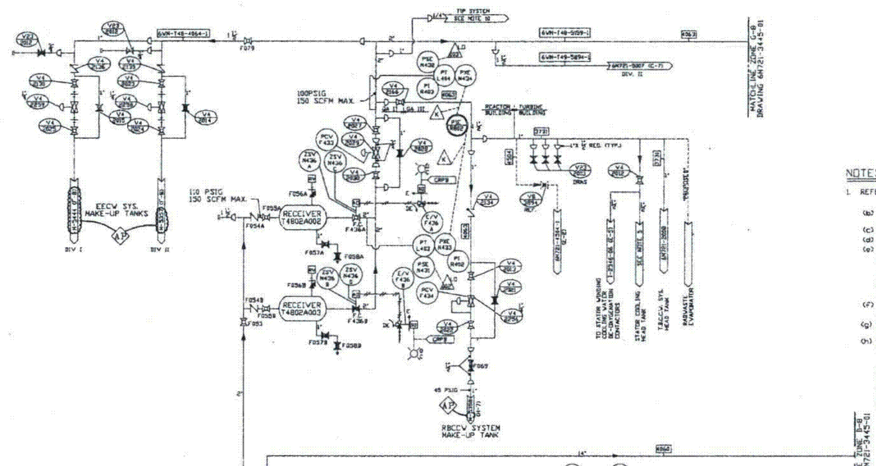


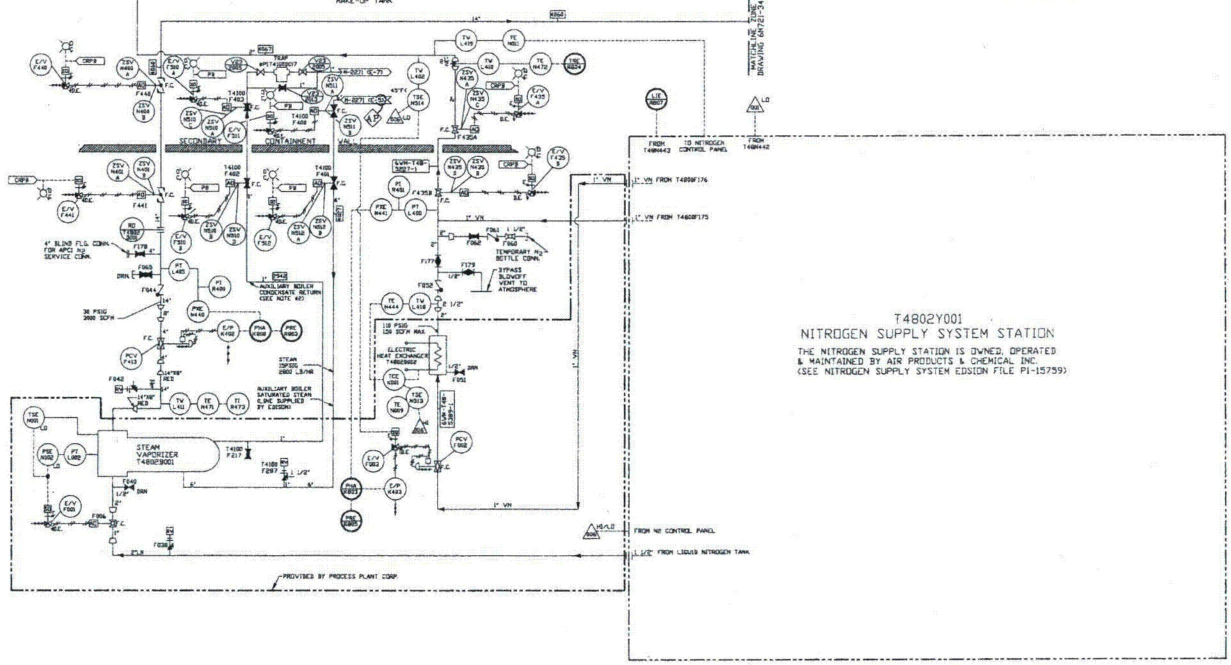
TABLE										
VACUUM BREAKER VALVES					N_2 SUPPLY ISOLATION VALVES					
VACUUM BREAKER VALVE NO.	SOLENOID VALVE INSTRUMENT NO.	VACUUM BREAKER LIMIT SWITCHES INSTRUMENT NUMBERS			PENETRATION NUMBER	ISOLATION VALVE NO.	LIMIT SWITCH INSTRUMENT NO.	SOLENOID VALVE INSTRUMENT NO.	CHECK VALVE NUMBER	TEST CONN. VALVE NUMBER
T2300F400A	E/V-T23F400A	ZSV-T23N400A	ZSV-T23N401A	ZSV-T23N402A	X-204a	T4800F416	ZSV-T48N416A & B	E/V-T48F416	T4800F113A	T4800F114A
T2300F400B	E/V-T23F400B	ZSV-T23N400B	ZSV-T23N401B	ZSV-T23N402B	X-204b	T4800F417	ZSV-T48N417A & B	E/V-T48F417	T4800F113B	T4800F114B
T2300F400C	E/V-T23F400C	ZSV-T23N400C	ZSV-T23N401C	ZSV-T23N402C	X-204c	T4800F418	ZSV-T48N418A & B	E/V-T48F418	T4800F113C	T4800F114C
T2300F400D	E/V-T23F400D	ZSV-T23N400D	ZSV-T23N401D	ZSV-T23N402D	X-204d	T4800F419	ZSV-T48N419A & B	E/V-T48F419	T4800F113D	T4800F114D
T2300F400E	E/V-T23F400E	ZSV-T23N400E	ZSV-T23N401E	ZSV-T23N402E	X-204e	T4800F420	ZSV-T48N420A & B	E/V-T48F420	T4800F113E	T4800F114E
T2300F400F	E/V-T23F400F	ZSV-T23N400F	ZSV-T23N401F	ZSV-T23N402F	X-204f	T4800F421	ZSV-T48N421A & B	E/V-T48F421	T4800F113F	T4800F114F
T2300F400G	E/V-T23F400G	ZSV-T23N400G	ZSV-T23N401G	ZSV-T23N402G	X-204g	T4800F422	ZSV-T48N422A & B	E/V-T48F422	T4800F113G	T4800F114G
T2300F400H	E/V-T23F400H	ZSV-T23N400H	ZSV-T23N401H	ZSV-T23N402H	X-204h	T4800F423	ZSV-T48N423A & B	E/V-T48F423	T4800F113H	T4800F114H
T2300F400J	E/V-T23F400J	ZSV-T23N400J	ZSV-T23N401J	ZSV-T23N402J	X-204j	T4800F424	ZSV-T48N424A & B	E/V-T48F424	T4800F113J	T4800F114J
T2300F400K	E/V-T23F400K	ZSV-T23N400K	ZSV-T23N401K	ZSV-T23N402K	X-204k	T4800F425	ZSV-T48N425A & B	E/V-T48F425	T4800F113K	T4800F114K
T2300F400L	E/V-T23F400L	ZSV-T23N400L	ZSV-T23N401L	ZSV-T23N402L	X-204l	T4800F426	ZSV-T48N426A & B	E/V-T48F426	T4800F113L	T4800F114L
T2300F400M	E/V-T23F400M	ZSV-T23N400M	ZSV-T23N401M	ZSV-T23N402M	X-204m	T4800F427	ZSV-T48N427A & B	E/V-T48F427	T4800F113M	T4800F114M

NOTES:

1. REFERENCE INFORMATION
   (b) NITROGEN INERTING SYSTEM SPECIFICATION NO. 3071-10L
   (c) PURCHASE B/M P4-97M
   (d) ALL EDISON PIPING IS 150# ANSI RATED.
   (e) ISOLATION VALVES IN THE PNEUMATIC SUPPLY LINE TO THE SUPPRESSION POOL VACUUM BREAKER VALVES WILL ONLY BE OPENED WHEN TESTING OF THE VALVE IS REQUIRED & DURING INERTING.
   (f) VACUUM BREAKER SPECIFICATION NO. 3071-86
   (g) PURCHASE B/M P1-41M-VALVE DECO FILE NO'S. P1-784&785.
   (h) PROCESS PLANTS CORPORATION DRAWINGS:

TITLE	DECO FILE NO.
GENERAL ARRG'MT	P4-51
INCONNECTING PIPING	P4-52
STEAM VAPORIZER	P4-53
NITROGEN VAPORIZER	P4-54
ELEC-HTX	P4-55
STORAGE TANK	P4-125

2. AUXILIARY BOILER CONDENSATE RETURN LINE DRAIN VALVES SHALL BE MAINTAINED IN THE OPEN POSITION WHEN STEAM VAPORIZER IS NOT IN USE. THE RETURN LINE PIPING SHALL BE SLOPED TO THESE DRAIN POSITIONS.
3. VALVE T4800F130 SHALL OPEN ONLY DURING CONTAINMENT INERTING MODE.
4. TO INERT THE DRYWELL, VALVE T4800F459 IS REMOTELY CLOSED. AFTER INERTING IS COMPLETED, VALVE T4800F440 IS CLOSED AND VALVE T4800F459 IS OPENED.
5. FOR CONT. SEE TERMINAL POINT "DE" ON STATOR HEAD TANK DWG. EDISON FILE #T1-777. ALSO SEE ENGLISH ELECT CO DIAGRAM, EDISON FILE #R1-610.
6. FOR LEGEND OF SYMBOLS & ABBREVIATIONS SEE DWG. 6M721-2000 & 6M721-2001.
7. ▭ DENOTES ISOMETRIC DRAWING NUMBER (PREFIX NO. 6M721-▭) OR VALVE LOCKED POSITION EITHER LOCKED OPEN/ LOCKED CLOSED AS INDICATED BY 'LO' OR 'LC'
8. INSTRUMENTATION ADDED ON REV. D.
9. UNLESS OTHERWISE SHOWN: ALL INSTRUMENT PIS NUMBERS ARE PREFIXED T48. ALL VALVE AND EQUIPMENT PIS NUMBERS ARE PREFIXED T4800.
10. FORMER N_2 SUPPLY TO TIP SYSTEM, CAPPED AND ABANDONED IN PLACE PER EDP 4940. SEE DRAWING 6M721-4063-1 FOR DETAILS. FOR NEW TIP SYSTEM N_2 SUPPLY, SEE 6M721-5740.
11. PRIMARY CONTAINMENT MONITORING SYSTEM 6I721-2679-01.
12. THIS REMOVABLE SPOOL PIECE TO BE REMOVED WHEN PERSONNEL ARE ON BREATHING AIR INSIDE DRYWELL.
13. SEE BLEED-OFF TUBING DWG. 6I721-2516-03 (DET. 13).
14. FOR SYSTEM FUNCTIONAL OPERATING SKETCH SEE DWG. 6M721-5739-1.
15. NITROGEN INERT SYSTEM CONTROL BUILDING 6I721-2659-1.
16. THE VENT HEADER AND DOWNCOMER SYSTEM INTERNAL TO THE TORUS WAS DESIGNED, FABRICATED AND ERECTED IN ACCORDANCE WITH THE ASME CLASS B PRESSURE VESSEL CODE 1968 THROUGH SUMMER 1969 ADDENDA BUT IS NOT LEAK TESTED.
17. NOTE DELETED
18. POSITION-LOCKED CLOSED (LC), METHOD-SEALED CLOSED BARRIER VIA ELECTRICALLY DE-ENERGIZED.



FOR CONTINUATION SEE DRAWING 6M721-3445-01

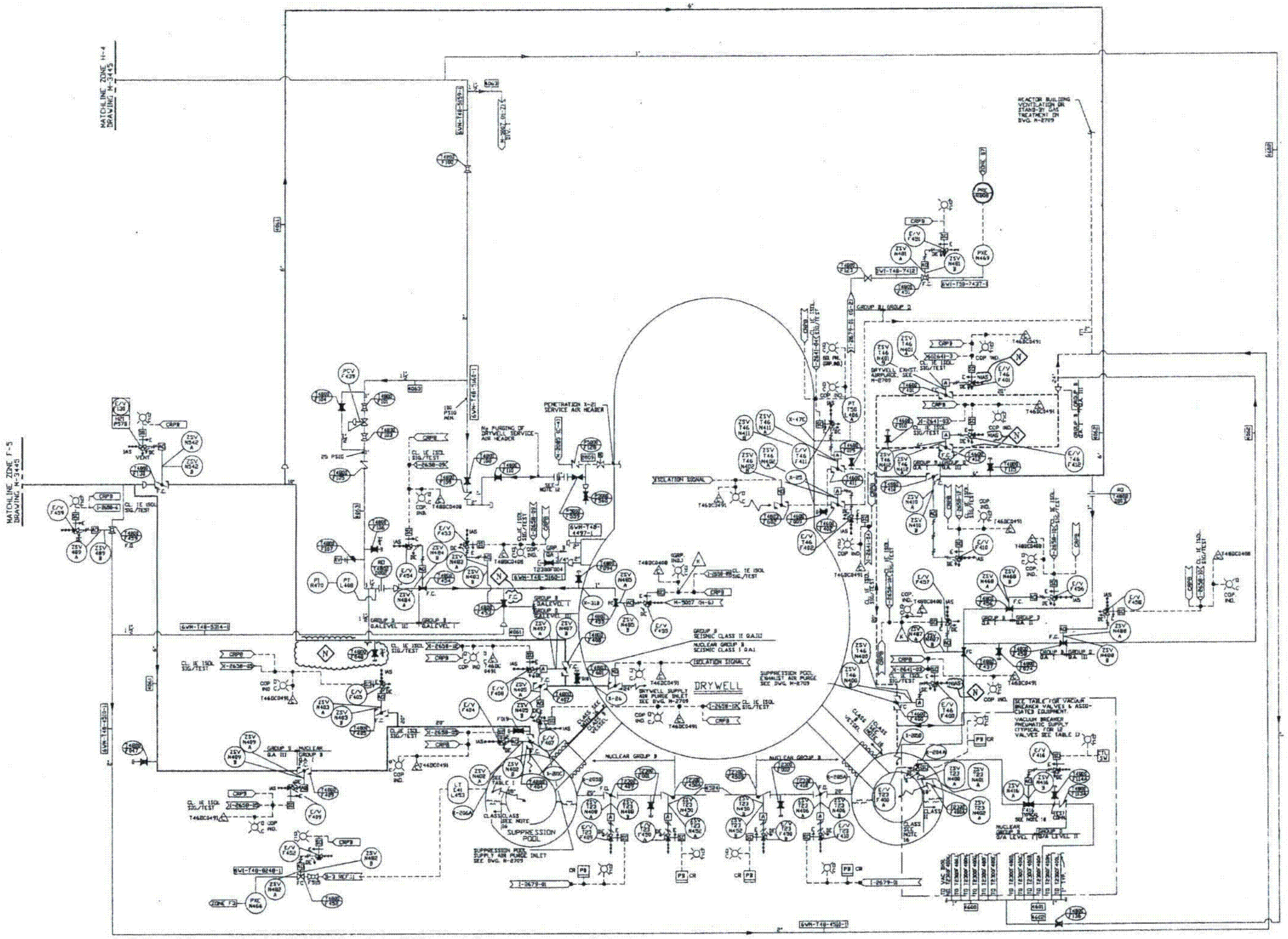
Fermi 2

UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 9.3-13

NITROGEN INERTING SYSTEM

DETROIT EDISON COMPANY DRAWING NO. 6M721-3445, REV. AP



THIS DRAWING ISSUED AS A RESULT OF REDRAWING OF ORIGINAL M-3445. FOR CONTINUATION, SEE DRAWING M-3445.

Fermi 2

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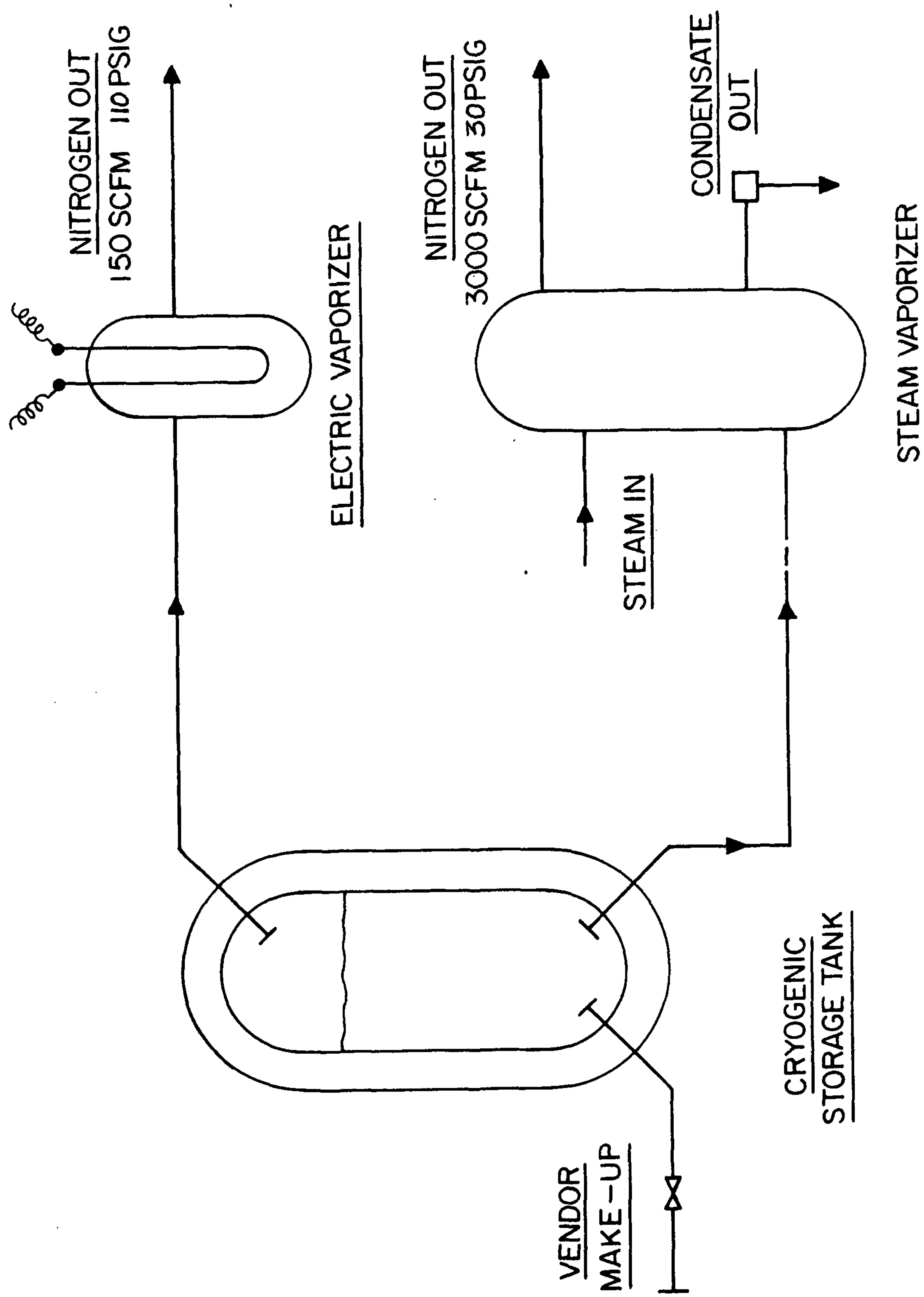
FIGURE 9.3-14

NITROGEN INERTING SYSTEM

DETROIT EDISON COMPANY DRAWING NO. 6M721-3445-1, REV. N



Fermi 2

UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 9.3-15

NITROGEN INERTING SYSTEM SUPPLY