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9.0 AUXILIARY SYSTEMS

9.1 FUEL STORAGE AND HANDLING

The equipment and evaluation presented in this section are applicable to either unit unless noted otherwise.

The major components of this system are the new fuel storage vault,
spent fuel pool, dryer/separator storage pit, spent fuel pool cooling pumps and heat exchangers, reactor building overhead crane, and the refueling platform. Refer to Section 1.2 for drawings showing the complete layout of the operating floor.

Refer to Section 9.1.2.2.4 for a description of spent fuel storage and handling of Dry Cask Storage (DCS) systems and the Independence Spent /Fuel Storage Installation (ISFSI).

Dresden Station complies with criticality accident requirements, as described in 10 CFR 50.68(b).

9.1.1 New Fuel Storage

9.1.1.1 Design Bases

The design objective of the new fuel storage system is to provide a clean, dry storage vault for new fuel. To achieve this objective, the new fuel storage system is designed using the following bases:

A. New fuel is stored in a manner which precludes inadvertent criticality.

B. Normal reactor refueling involves replacement of 25% to 35% of the core.

C. New or undamaged used fuel channels are installed on new fuel bundles. Present channel management strategy is to only use new fuel channels on new fuel.

D. The new fuel storage vault is designed to withstand earthquake loading of a Class I structure.

E. There will be no release of contamination or exposure of personnel to radiation in excess of 10 CFR 20 limits.

9.1.1.2 Facilities Description

The new fuel dry storage vault is common for Unit 2 and 3 fuel. The vault contains aluminum, unpoisoned, full length, top entry storage racks. The vault can store a maximum of 320 new fuel bundles in an upright position. The minimum center-to-center spacing of fuel bundles within a given row is 6.5 inches and the minimum spacing between fuel bundles in adjacent rows is 10 inches. Racks in the vault are designed to prevent an accidental critical array, even in the event that the vault becomes flooded. Vault drainage is provided by an open drain in the vault bottom to prevent possible water collection. A lip around the top of the vault prevents water on the refueling floor from draining into the vault.

The new fuel storage vault is a reinforced concrete Class I structure, accessible only through top hatches. All entrances to the vault, including hatches and personnel openings, are capable of being locked. An area radiation monitor is located in the vault.

Refer to Section 9.1.2.2.4 for a description of spent fuel storage and handling of Dry Cask Storage (DCS) systems and the Independent Spent /Fuel Storage Installation (ISFSI).

9.1.1.3 Safety Evaluation

The spacing of fuel bundles in the new fuel storage vault maintains k_{eff} less than or equal to 0.90 dry and k_{eff} less than or equal to 0.95 flooded. SVEA-96 Optima2 assemblies can be safely stored in the Dresden 2/3 new fuel storage vault and meet the above criteria of K_{eff} less than 0.90 dry and 0.95 fully flooded. These conditions will be met for any SVEA-96 Optima2 lattice with a nominal enrichment less than 2.80, and for any lattice with an enrichment between 2.80 and 4.95 that has a minimum Gadolinia loading of at least 8 Gadolinia rods at a minimum nominal concentration of 5.5 Gd_2O_3. If a Gadolinia rod is face adjacent to another Gadolinia rod, it shall not be counted toward meeting the minimum number of Gadolinia rods (two face adjacent Gadolinia rods can only be counted as one rod in meeting this requirement).

ATRIUM 10XM assemblies can be safely stored in the Dresden 2/3 new fuel storage vault and meet the criteria of k_{eff} less than 0.90 dry and 0.95 fully flooded. Reference 23 provides the lattice enrichment and Gadolinia loading criteria for ATRIUM 10XM assemblies to be safely stored in the Dresden 2/3 new fuel storage vault.

In addition, controls have been implemented to further reduce the probability of a criticality occurrence, i.e., the storage array will be in a moderation controlled area. A moderation control area limits the amount of hydrogenous material in the area. Administrative controls as generally defined in SIL 152[6] have been incorporated for the area. The vault floor drain prevents flooding. A radiation monitor in the new fuel storage vault provides warning of any radiation level increase.

9.1.2 Spent Fuel Storage

9.1.2.1 Design Bases

The design objectives of the station spent fuel storage system are as follows:

A. To provide a maximum underwater storage capability for 7074 fuel assemblies;

B. To provide for underwater storage of reactor vessel internals; and

C. To provide adequate protection against the loss of water from the fuel pools.

D. To safely store fuel in Dry Cask Storage (DCS) systems per 10CFR72.

To achieve these objectives, the spent fuel storage system is designed using the following bases:

A. There will be no release of contamination or exposure of personnel to radiation in excess of 10 CFR 20 limits;

B. The storage space in each of the Unit 2 and Unit 3 spent fuel pools is designed for a maximum of 3537 irradiated fuel assemblies;

C. It is possible, at any time, to perform limited work on irradiated components; and

D. Space is provided for used control rods, flow channels, and other reactor components.

E. The spent fuel pool is designed to withstand earthquake loadings of a Class I structure.

F. The spent fuel assembly racks are designed to ensure subcriticality in the storage pool. A maximum K_{eff} of 0.95 is maintained with the racks fully loaded with fuel of the highest anticipated reactivity and flooded with unborated water at a temperature corresponding to the highest reactivity.

G. Spent fuel storage in Dry Cask Storage (DCS) systems per 10CFR72.

9.1.2.2 Facilities Description

The major components of the spent fuel storage system are the spent fuel pool, the reactor well, and the dryer/separator pit. These pools are located in the reactor building on elevation 613'-0". Refer to Section 1.2 for detailed drawings of the spent fuel storage system arrangement.

Refer to Section 9.1.2.2.4 for a description of spent fuel storage and handling of Dry Cask Storage (DCS) systems and the Independence Spent /Fuel Storage Installation (ISFSI).

9.1.2.2.1 Dryer/Separator Pit

The dryer/separator pit provides a location for underwater transfer and storage of the reactor moisture separator and steam dryer assemblies during refueling operations.

A raised step is provided at the bottom of the pit to:

A. Prevent particulate material from entering the reactor well from the dryer/separator pit; and

B. Ensure at least a 6-inch water coverage over the reactor shroud head, which becomes very radioactive during plant operations.

The pit is lined with stainless steel. The space between the storage pit liner and the concrete walls and floor has a drain which may be used to detect liner leakage. The annulus drainage flows through "tell-tale" sightglasses to the reactor building floor drain sump.

9.1.2.2.2 Reactor Well

The purpose of the reactor well is to provide a space which can be flooded to permit the removal and underwater transfer of the steam dryer, moisture separators and fuel. Removable shield plugs are installed over the reactor well during normal operation. The plugs are provided to reduce the operating floor radiation during plant operation to insignificant levels. To flood the reactor well for refueling, the refueling bulkhead, in conjunction with a system of bellows seals, provides a watertight barrier to permit flooding above the reactor while preventing water from entering the drywell. As shown in Figure 9.1-1, the bulkhead is a flat, circumferential plate, which is fixed rigidly to the inside of the containment. The bulkhead contains ventilation duct hatches, which allow the drywell cooling system to cool the area above the bulkhead (within the drywell head) during normal plant operations.

The drywell-to-reactor-building bellows seal (see Figures 9.1-1 and 9.1-2) accommodates the differential expansion that occurs between the drywell and the reactor building concrete during plant heatup and cooldown. The seal is a cylindrical, one-piece, stainless steel bellows that seals the annulus between the drywell concrete wall and the drywell liner. To facilitate leak detection, flow through a drain line at the low point on the reactor building side of the seal is monitored.

The reactor-vessel-to-drywell seal accommodates the differential expansion that occurs between the reactor vessel and the drywell during reactor heatup and cooldown (Figure 9.1-2). The seal is a cylindrical, one-piece, stainless steel bellows. One end is welded to a special skirt on the reactor vessel, while the other end is welded to the refueling bulkhead. It seals the opening between the reactor vessel head flange and the drywell to allow flooding the reactor cavity above. To facilitate leak detection, a flow switch is provided to monitor leakage to a drain line located at the low point on the outside of the seal. The drain line is piped to the drywell equipment drain sump.

9.1.2.2.3 Spent Fuel Pool

The spent fuel pool has been designed to withstand the anticipated earthquake loadings as a Class I structure. Each unit has its own spent fuel pool measuring 33 ft x 41 ft. Each pool is a reinforced concrete structure, completely lined with seam-welded stainless steel plates welded to reinforcing members (channels, I beams, etc.) embedded in concrete. The stainless steel liner, minimum 3/16-inch thick, will prevent leakage even in the unlikely event that the concrete develops cracks. To avoid unintentional draining of the spent fuel pool, there are no penetrations that would permit the spent fuel pool to be drained below a safe storage level, and all lines extending below this level are equipped with suitable valving to prevent backflow. As shown in Figure 9.1-3 (Drawing M-31) and Figure 9.1-4 (Drawing M-362), the passage between the spent fuel pool and the refueling cavity above the reactor vessel is provided with two double-sealed gates with a monitored drain between the gates. This arrangement permits detection of leaks from the passage and repair of a gate in the event of such leakage. The normal depth of water in the spent fuel pool is 37 feet, 9 inches and the depth of water in the transfer canal during refueling is 22 feet, 9 inches. A reinforced area is provided in one corner of each spent fuel pool for loading the spent fuel cask. The liner is 1-inch thick at this location to ensure liner integrity. The cask set down pad is sized to accommodate the 100-ton NL 10/24 spent fuel shipping cask and HI-TRAC 100 transfer cask. The pad is reinforced above the pool liner by stainless steel plates welded in place. The water in the spent fuel pool is continuously filtered and cooled by the spent fuel pool cooling and cleanup system described in Section 9.1.3.

In addition to the current capacity for fuel assemblies, the spent fuel pool holds discarded local power range monitors (LPRMs), control blades (also control blades to be reinserted in the core), small reactor vessel components, and miscellaneous tools and equipment as necessary. Additional storage for large components, e.g. steam dryer and separator, is provided in the dryer/separator pit (see Section 9.1.2.2.1) adjacent to the reactor cavity.

9.1.2.2.3.1 Spent Fuel Pool Liner and Sumps

The spent fuel pool system incorporates design features to compensate for damage to the spent fuel pool liner. In addition to the system design capabilities, there are procedural controls to prevent damage from occurring.

The system design provides for minor cracks in the stainless steel liner. Beneath each liner seam weld is a drainage trough which directs leakage to the spent fuel pool liner drain network. These drains lead from beneath the liner to the reactor

building floor drain system. Each of the four pool drain outlets are valved open and a flow glass is provided downstream of each valve. This arrangement aids in locating problem areas and controls flow to the reactor building floor drain sumps. These drain sumps are capable of removing up to 100 gal/min, which is greater than any anticipated seam or liner crack leakage.

9.1.2.2.3.2 Spent Fuel Pool High Density Racks

Each spent fuel pool contains 33 high-density spent fuel storage racks which provide storage for 3537 fuel assemblies. There are 18 racks arranged in a 9x11 array and 15 racks arranged in a 9x13 array. See Figure 9.1-5 for the relative location of the racks in each spent fuel pool. The racks are constructed to form tubes of adequate size for fuel storage. The tubes are welded together along their length with angles or clips to provide the inter-tube connection. The center-to-center distance between assemblies stored in tubes is 6.30 in. x 6.30 in.

The fuel storage tube is constructed of stainless-steel-bearing Boral neutron absorbing material. Boral is a sandwich-type plate (see Figure 9.1-6) that has outer surfaces of Type 1100 aluminum and a core of boron carbide (B_4C) uniformly dispersed in a matrix of Type 1100 aluminum. These plates are enclosed by inner and outer tubes made of Type 304 stainless steel designed to permit spent fuel pool water to enter and exit the Boral area. The inner and outer tubes maintain the Boral plate structural integrity during vibratory events. The plates are not required to carry load. The individual neutron absorbing tubes are connected in a checkerboard pattern forming the rack assembly.

The rack assembly is shown in Figure 9.1-7. Each rack consists of a base assembly with legs and with plates along the edges and across the midpoint. A fuel support plate fabricated from 3/16-inch plate is provided in each storage position to hold one fuel assembly. The support plate is elevated approximately 12.37 inches above the spent fuel pool floor and is welded to the lower end of the tube. Cooling water flows through holes and/or slots in the sides of the support plates into the storage tubes to cool the stored fuel. Along the side of the rack, a filler plate assembly is welded between the absorber tube assemblies to enclose the space between neutron absorbing tubes. The racks are designed to prevent application of excessive vertical forces from the fuel handling system.

Each rack has six legs. The bottom of the legs are 11 1/2-in. x 11 1/2-in. pads. The legs are made of 1/2-inch plates welded to 3/4-inch plates forming the rack base. There are no lateral interties and no rack-to-wall interties or bumper plates. The racks rest on the spent fuel pool liner, except for the racks which are located over the trench. Racks spanning the trench are supported on one side by their pads which rest on the trench bridging grid and on the other side by their pads which rest on the liner.

Typical gap between racks is 2 1/2 inches and the minimum rack-to-wall gap is 3 1/2 inches. The tops of the tubes are straight without any flared guide provisions.

9.1.2.2.3.3 Spent Fuel Pool Blade Guide Racks

Control blade guides for refueling are stored in racks in the Unit 2 and 3 spent fuel pools. They are supported in the racks both vertically and laterally at the bottom by a support plate and laterally at the top by a gridwork of beams. The storage system, which is not safety related, consists of four stainless steel storage racks (two small racks and two large racks). Each small rack can store 36 guide pairs in a 3 x 12 array and each large rack can store 56 guide pairs in a 4 x 14 array for a total storage of 184 guide pairs. The racks can be placed in either of the Unit 2 or 3 spent fuel pools on the cask pad. They can be lifted fully loaded with blade guides by the 9-ton auxiliary hoist using a sling assembly. The racks are free-standing, i.e., they do not require attachment to the walls or floor of the spent fuel pool. They are seismically designed due to their location within the Seismic Category I spent fuel pool structure adjacent to the category I spent fuel storage racks.

9.1.2.2.4 Dry Cask Storage (DCS)

Dry Cask Storage (DCS) and the Independent Spent Fuel Storage Installation (ISFSI) SSCs provide the means for long term onsite storage of Dresden Unit 1, 2 or 3 spent fuel. The DCS systems include the HI-STORM 100 Cask System, HI-STAR 100 Cask System, HI-TRAC 100 Transfer Cask and Multi-Purpose Canisters (MPCs). The ISFSI includes the concrete storage and staging pads and the Cask Transfer Facility (CTF) that provides the means for transferring MPCs containing spent fuel between the transfer cask and the storage casks. Use of the DCS SSCs is granted upon issuance of a Certificate of Conformance (CoC) from the NRC. Use of the ISFSI/CTF SSCs for storage and handling of spent fuel is granted upon compliance with the conditions of the General License issued under 10 CFR 72, Subpart K. Use of the ISFSI/CTF SSCs for storage and handling of spent fuel shall be in accordance with the Dresden Nuclear Power /Station (DNPS) Units 1, 2 & 3 10 CFR 72.212 Evaluation Report.

The ISFSI cask storage pads may also be used for fabrication and staging of Dry Cask Storage Systems not loaded with fuel.

9.1.2.2.5 Low Level Waste (LLW)

Storage racks and equipment have been designed to process and store hardware in the Spent Fuel Pool (SFP) until a LLW dry cask storage system is implemented. This equipment is designed as Category II/I to not impact the SFP liner during a seismic event.

9.1.2.3 Safety Evaluation

This section discusses the safety evaluation associated with the fuel rack design. The analysis of radiological consequences for fuel handling accidents are discussed in Section 15.7.

9.1.2.3.1 High-Density Fuel Racks

The high-density fuel racks (HDFRs) were designed to ensure k_{eff} is maintained less than or equal to 0.95 for all normal and abnormal fuel storage conditions. The 6.30-in. x 6.30-in. spacing of the fuel assemblies in the racks assures this.

Criticality analyses have been performed for each fuel type stored in the spent fuel pools to assure compliance with the k_{eff} criteria. These criticality analyses include allowances for uncertainty, which are described in the criticality analysis reports applicable to the spent fuel pool (Reference 22). Critical parameters have been established to evaluate variations in the neutronic design of actual assemblies. A spent fuel storage criticality validation is performed for each reload to demonstrate that the reload fuel assemblies meet these critical parameters and the rack k_{eff} requirements for storage.

With the introduction of ATRIUM l0XM fuel at Dresden 2/3, an analysis (Reference 22) was performed to ensure that the spent fuel pool design criteria of k_{eff} less than or equal to 0.95 is met. This analysis determines the bounding lattice for all fuel types used at Dresden 2/3 and uses this lattice to demonstrate compliance with the k_{eff} criteria. An ATRIUM l0XM lattice was determined to bound all earlier fuel designs and is the basis of the analysis. The following conditions were analyzed for the spent fuel pool storage racks containing the bounding lattice:

a. Fuel racks fully loaded with bounding lattice at its peak reactivity burnup.

b. Bounding depletion parameters.

c. Most reactive moderator temperature and density.

d. Normal positioning in the spent fuel storage array.

e. Eccentric positioning in the spent fuel storage array.

f. Fuel manufacturing tolerances.

g. Spent fuel storage rack manufacturing tolerances.

h. Fuel geometry changes.

1. Clean and unborated water in the spent fuel pool.

J. Fuel rack lateral movement.

k. Dropped fuel assembly.

1. Missing BORAL panel.

m. Mislocated assembly.

The reactivity of the Dresden 2/3 spent fuel pool storage racks has been calculated using the computer codes CASM0-4 and MCNP5- l .51. CASM0-4 was used to determine the exposure-dependent pin-by-pin isotopic fuel compositions. MCNP5-l .5 l was used with the CASM0-4 pin-specific isotopic fuel compositions and ENDF/B -VII cross-section data to compute peak in- rack reactivity.

In support of the criticality safety calculations, a set of critical experiments were analyzed using MCNP5-l.51 to provide a definitive determination of the methodology bias and bias uncertainty. The critical experiments included a wide range of compositions and geometries that are representative of spent fuel rack designs and BWR fuel geometry.

The Reference 22 analysis demonstrates that all fuel assemblies of the Optima2 and earlier fuel designs used in the Dresden 2 and 3 reactors can be safely stored in the spent fuel pools (with or without a channel) and comply with the Technical Specifications in-rack k-infinite limit (added in Amendment Number 249/242). Additionally, this analysis (Reference 22) provides a method for demonstrating that ATRIUM l0XM fuel can be safely stored in the spent fuel pool and comply with the Technical Specifications in-rack k-infinite limit (added in Amendment Number 249/242) for a range of enrichment and gadolinium loadings.

Before any HDFR was placed in the spent fuel pools, each storage location was checked with a mandrel to confirm that the minimum dimension between the lead-in clips which provide the intertube connection at the top of each storage location was at least 5.758 inches. The storage clips were ground down as necessary to ensure this dimension was achieved.

In situ (i.e., racks installed) neutron attenuation tests were performed to verify that tubes and racks contained a sufficient number of Boral plates such that k_{eff} will not be greater than 0.95 when the spent fuel is in place. Acceptance criteria required the associated tube to be blocked to prohibit insertion of a fuel assembly if one Boral plate was detected missing. If more than one Boral plate was detected missing per spent fuel pool, the storage rack or racks containing any additional missing Boral plates were required to be removed from the spent fuel pool. These storage racks were not to be replaced in the spent fuel pool until a specific criticality analysis covering the proposed corrective action had been submitted to and approved by the NRC.

The following loads were considered in the design and evaluation of the racks in their installed condition in the pool:

A. Deadweight,

B. Buoyance,

C. Operating basis earthquake,

D. Safe shutdown earthquake (SSE),

E. Fuel bundle drop from 12 inches,

F. Thermal gradient, and

G. Pool water temperature.

The racks are classified as Seismic Category I structures. Structural adequacy of the racks was verified for the applicable loading combinations listed in Standard Review Plan Section 3.8.4 and for stress allowables and service limit designations of ASME Section III, Division 1, Subsection NF 3300. Elastic working stress method of analysis was used for all loading combinations except for loads due to fuel bundle drop. An elasto-plastic method of analysis was performed to evaluate the consequence of this accidental loading.

Four different analyses were performed on the racks to evaluate their structural adequacy. Since the racks rest freely on the pool floor, the maximum horizontal movement of each rack when subjected to the vibratory motion of the most severe postulated earthquake (i.e., SSE) was calculated. To ensure that stresses in the racks when subjected to different load combinations are within the allowable stress limits, elastic stress analyses were performed. Rack stability analyses were performed to determine the factor of safety against overturning due to the action of the most severe postulated earthquake. Finally, the consequences of accidentally dropping a fuel bundle on a fuel rack from a height of 12 inches were also evaluated.

Results of the upper bound sliding analysis showed that the maximum sliding distance under SSE loads is 1.012 inches, which is far less than the minimum gap between two adjacent racks (2.5 inches). This is also smaller than the minimum gap between any rack and the adjacent pool wall (3.5 inches). Thus, the proposed racks, when arranged as shown in Figure 9.1-5, would not impact each other or the pool walls.

The maximum combined stress calculations show that the racks meet all stress acceptance criteria.

From the stability analyses of the loaded racks, the safety factor against overturning was found to be 18.9. This is far in excess of the minimum required value of 1.5. Thus, the racks are safe against overturning.

The top corners of the racks were found to be the most critical locations for evaluating the consequence of dropping a fuel bundle. When the fuel bundle drops on the rack, the cross-sectional area of the cell walls absorbing the impact energy increases as the load is transmitted downward. Since the cross-sectional area is smallest when the fuel bundle drops on a corner, the top corner constituted the most critical location.

For evaluating the consequences of fuel bundle drop from 12 inches, the bundle configuration was assumed to be vertical at impact.

In the event that a fuel bundle drops straight through a tube, it will impact on the fuel support plate inside the tube. Each fuel support plate is a thin rectangular plate with a circular hole at the center. This is attached to the inside of the tube with fillet welds. The energy required to cause the plate to yield is small compared to the energy with which the bundle will impact the fuel support plate. Hence, the plate will yield under the impacting bundle. However, the fuel support plates can be dispensed with since these are not essential for the structural integrity of the rack. The stress analysis was performed without considering the stiffening effect of the fuel support plates.

The elasto-plastic analyses of the racks for the accidental fuel bundle drop showed that the maximum length of the rack, measured from the top, which might be stressed beyond elastic limits is 2.1 inches; whereas, the available length of the racks above the active fuel length is about 13 inches.

An Additional evaluation was performed showing that a potential energy of 30240 in-lbs will result in a deformation length of 4 inches. If an assembly weight of 1000 lbs is assumed (which bounds the actual weight of a channeled fuel bundle), a drop height of 30 inches would result in the same potential energy and deformed length. After accounting for a deformed length of 4 inches, the undamaged rack length containing neutron absorbing material is about 9 inches above active fuel. However, a deformation length of 4 inches or more is not expected since the equipment and operating procedures used to handle the fuel bundles limit the maximum height of fuel bundle drop while over the fuel racks to approximately 15 inches.

To ensure the HDFRs are utilized in accordance with the safety analysis assumptions, no loads heavier than the weight of a single spent fuel assembly and handling tool will be carried over fuel stored in the spent fuel pool.

A corrosion sampling program to verify the integrity of the neutron absorber material employed in the HDFRs in the long-term environment has been implemented. The test conditions represent the vented conditions of the spent fuel tubes. Samples are located adjacent to the fuel racks and suspended from the spent fuel pool wall. Eighteen test samples were installed in the Unit 3 pool. One or more samples are tested each testing interval in accordance with a predefined schedule. The samples may be subjected to a B-10 loading analysis.

Additionally, two full length vented fuel storage tubes are suspended in the Unit 2 and 3 spent fuel pools and may be examined should the sample program indicate any reduction of neutron absorber material to below 0.02 g/cm^2, B-10.

This corrosion surveillance program for the HDFRs ensures that any loss of neutron absorber material and/or swelling of the storage tubes will be detected.

The effects of the additional loading on the storage pool structure due to the HDFR and equipment have been analyzed. The spent fuel pool structures including the two stainless steel grid structures spanning the trench were evaluated as Seismic Category I structures in accordance with Regulatory Guide 1.29. Their structural adequacy was verified in accordance with Standard Review Plan Section 3.8.4.

Results of spent fuel pool structural analyses and analyses of grid structures spanning the floor trench show adequacy of the pool structures to withstand the added loads resulting from increased spent fuel pool storage capacity. The stresses in the two grid structures over the trench are within the permissible limits.

The additional thermal loading on the spent fuel pool cooling system was evaluated assuming decay heat energy release rates specified in ANS 5.1, Standard Decay Heat Curve (with two sigma uncertainty) for EPU conditions. The calculated maximum heat loads for a partial core offload and a full core offload from 24 month fuel cycles were based on discharging fuel bundles to the spent fuel pool after a 100-hour cooling period at a transfer rate of ten fuel bundles per hour. Various combinations of operating equipment were evaluated. In addition, heat exchanger performance calculations demonstrate that one shutdown cooling heat exchanger operating at 1500 gal/min in parallel with the spent fuel pool cooling system can maintain the pool water temperature below 150 °F for the full core offload condition. For a planned offload with the cooling systems operating, and assuming the limiting single failure, the temperature of the fuel pool will be kept below 141°F. Because the Dresden shutdown safety management procedure requires the ability to align a spare loop of SDC to the SFP within eight hours of the loss of the operating SDC loop, the failure of a FPC pump is considered the limiting single failure.

In the event of a complete loss of spent fuel pool cooling capability with the fuel pool gate open, the pool temperature would reach the boiling point after 8 hours. [19] This conclusion assumes the following:

A. A 100-hour cooling time following reactor shutdown, prior to discharging fuel to the spent fuel pool;

B. A transfer rate of ten fuel bundles per hour;

C. Initial fuel pool temperature of 110°F; prior to fuel offload,

D. The pool water bulk temperature is at its maximum temperature of 150°F, and

E. Complete mixing of the water.

In lieu of assumptions A and B above, the heat load of spent fuel to be stored in the pool may be calculated prior to each fuel offload based upon the number of fuel assemblies to be discharged, the time between reactor shutdown and the start of fuel offload, and the rate of fuel offload. The calculated heat load includes decay heat from all previously discharged spent fuel assemblies in the fuel pool. Heat removal rates of the fuel pool cooling system and the shutdown cooling fuel pool cooling assist mode are calculated based on the maximum allowable fuel pool outlet temperature of 141 °F for a partial core offload (normal) and 150 °F for a full core offload (abnormal). Conservative values of service water temperature and reactor building closed cooling water (RBCCW) temperature are assumed based on the time of year during which the refueling outage occurs. The calculated heat removal rate is compared to the calculated heat load to ensure that the heat removal rate is higher. For a planned offload with the cooling systems operating, and assuming the limiting single failure, the temperature of the fuel pool will be kept below 141°F. Because the Dresden shutdown safety management procedure requires the ability to align a spare loop of SDC to the SFP within eight hours of the loss of the operating SDC loop, the failure of a FPC pump is considered the limiting single failure. Acceptance limits for the time-to-boil (greater than 8 hours after loss of all spent fuel pool cooling), boil-off rate (less than 70 gpm during bulk boiling conditions), and fuel temperature (less than 350 °F during bulk boiling) must be satisfied. Additionally, calculations must demonstrate that no local boiling occurs on the surface of any fuel assembly, k_{eff} <0.95, and the effects of higher local temperatures on fuel rack structural integrity are acceptable. Administrative controls are procedurally implemented to ensure compliance with the analysis assumptions described above throughout the offload. Additionally, to provide margin against exceeding the acceptance limits, the fuel pool temperature is administratively limited to a temperature less than the acceptance limits of 141° F and 150 °F or else fuel moves are suspended.

The NRC has evaluated the HDFRs and determined that with respect to structural and mechanical design, the HDFRs satisfy the applicable requirements of General Design Criteria 2, 4, 61, and 62 of 10 CFR 50, Appendix A.

The refueling floor is monitored for abnormal radiation levels. Refer to Section 12.3 for a description of the radiation monitoring system.

9.1.2.3.2 Storage of Unit 1 Spent Fuel in the Unit 2 and 3 Spent Fuel Storage Racks

The structural, criticality, and thermal hydraulic effects of storing Unit 1 spent fuel in the Unit 2 and 3 high-density spent fuel racks has been evaluated.[2] Storing Unit 1 spent fuel in Units 2 and 3 spent fuel pools will not increase k_{eff} above 0.95 because Unit 1 fuel is less reactive than Unit 2 and 3 fuel. Since a Unit 1 fuel assembly is dimensionally smaller than a Unit 2 and 3 fuel assembly, it is necessary to insert an adapter in the Unit 2 and 3 spent fuel rack.

It was concluded that Unit 1 fuel assemblies can be safely stored provided an adapter is used and the prescribed storage pattern is followed. The storage pattern requires that Unit 1 fuel assemblies be stored symmetrically starting in the center of the rack and be limited to 63 fuel assemblies per 9x13 rack array and 49 fuel assemblies per 9x11 rack array. The calculations show that the HDFRs are safe and meet the original design requirements when Unit 1 fuel is stored in the limiting pattern.

9.1.2.3.3 Spent Fuel Pool Blade Guide Racks

Design of the control blade guide racks has been carried out in accordance with the applicable portions of Appendix D to the USNRC Standard Review Plan Section 3.8.4, "Technical Position on Spent Fuel Racks." The detailed seismic analyses were performed which demonstrate that rack stresses and displacements meet the acceptance criteria. Although the racks are not classified as safety related, they have been ruggedly designed and carefully analyzed similar to a seismic Category I structure. The rack lifting lugs and lifting sling have been designed with a safety factor of ten with respect to ultimate strength for the maximum combined concurrent static and dynamic load in accordance with the guidelines of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants." The malfunction of these racks under normal and abnormal loading is, therefore, not anticipated. Additionally, the safe load path for the racks will avoid movement of the racks over the spent fuel being stored in the pool.

9.1.3 Spent Fuel Pool Cooling and Cleanup System

9.1.3.1 Design Bases

The design objectives of the spent fuel pool cooling and cleanup system are to handle the spent fuel cooling load and to maintain pool water clarity. To achieve these objectives the system has been designed as follows:

A. To remove the decay heat from a partial or full core discharge of fuel and maintain the spent fuel pool water temperature at or below 150°F with one shutdown cooling heat exchanger operating in parallel with the spent fuel pool cooling system.

B. To provide enough filtering capacity to filter the spent fuel pool water volume every 12 hours.

The results of the HDFR licensing report show the spent fuel pool water temperatures for both refueling cases will remain below the above limits with one shutdown cooling heat exchanger in parallel with the spent fuel pool cooling system. The calculated maximum heat load for both refueling cases is based on discharging fuel bundles to the spent fuel pool after 100-hour cooling period at a transfer rate of ten fuel bundles per hour.

In lieu of assumptions of a 100-hour cooling period and a transfer rate of ten fuel bundles per hour, the heat load of spent fuel to be stored in the pool may be calculated prior to each fuel offload based upon the number of fuel assemblies to be discharged, the time between reactor shutdown and the start of fuel offload, and the rate of fuel offload; the calculated heat load includes decay heat from all previously discharged spent fuel assemblies in the fuel pool. Heat removal rates of the fuel pool cooling system and the shutdown cooling fuel pool cooling assist mode are calculated based on the maximum allowable fuel pool outlet temperature of 141 °F for a partial core offload (normal) and 150°F for a full core offload (abnormal). Conservative values for service water temperature and reactor building closed cooling water (RBCCW) temperature are assumed based on the time of year during which the refueling outage occurs. The calculated heat removal rate is compared to the calculated heat load to ensure that the heat removal rate is higher. For a planned offload with the cooling systems operating, and assuming the limiting single failure, the temperature of the fuel pool will be kept below 141°F. Because the Dresden shutdown safety management procedure requires the ability to align a spare loop of SDC to the SFP within eight hours of the loss of the operating SDC loop, the failure of a FPC pump is considered the limiting single failure. Acceptance limits for the time-to-boil (greater than 8 hours after loss of all spent fuel pool cooling), boil-off rate (less than 70 gpm during bulk boiling conditions), and fuel temperature (less than 350 °F during bulk boiling) must be satisfied. Additionally, calculations must demonstrate that no local boiling occurs on the surface of any fuel assembly, $k_{eff} < 0.95$, and the effects of higher local temperatures on fuel rack structural integrity are acceptable. Administrative controls are procedurally implemented to ensure compliance with the analysis assumptions described above throughout the offload. Additionally, to provide margin against exceeding the acceptance limits, the fuel pool temperature is administratively limited to a temperature less than the acceptance limits of 141 °F and 150 °F or else fuel moves are suspended.

The maximum fuel pool temperature has been chosen to maintain a comfortable working environment in the pool area, to keep the filtering system at an operable temperature, and to maintain clarity of the air above the pool. The filtration flow may be varied to maintain spent fuel pool water clarity and purity. The system has been designed using the following bases:

Design flowrate (maximum)	1400 gal/min
Pump design code	ASME Section VIII
Heat exchanger design code	ASME Section III, Class C
Minimum size of particles filtered	25 microns

9.1.3.2 System Description

The spent fuel pool cooling and cleanup system consists of two circulating pumps; two heat exchangers; a filter; a deep-bed demineralizer; and the required piping, valves, and instrumentation.

The pumps, heat exchangers, and demineralizer are located in the reactor building near the spent fuel pool. The fuel pool filter, which may become radioactive as it collects corrosion products, is located in the radwaste building.

The spent fuel pool cooling and cleanup system is shown in Drawings M-31, M-50, M-362, and M-373). Water from the spent fuel pool overflows via scuppers and an adjustable weir into a pair of crosstied skimmer surge tanks. Foreign material entering the spent fuel pool will either sink to the bottom (where it may be removed by a portable vacuum cleaner) or float about in the pool and eventually enter the skimmers, surge tanks, and filtering loop. The pumps take suction from the skimmer surge tanks (located at the top of the spent fuel pool); continuously skim the water from the surface; and circulate the water to the heat exchanger, filter, and demineralizer before discharging the water through two lines back to the spent fuel pool. During refueling operations, the spent fuel pool cooling system may be aligned to discharge into the reactor refueling cavity via manual isolation valves.

The precoat-type filter in each unit uses stainless steel elements. The precoat material slurry is added to the filter and is circulated through the filter vessel until a uniform coating of precoat material covers the elements. The filter is then placed in service until differential pressure signals the need for backwashing. The backwashing process consists mainly of first valving off and draining the dome of the filter, then filling the filter with high-pressure air. All vents are closed during this filling, and air is trapped in the filter dome above the elements. When the pressure in the filter dome reaches approximately 100 psig, the drain valve is quickly opened, and the filter cake, together with trapped impurities, washes into the filter sludge tank. From the sludge tank, the suspension of impurities and water is processed through the radwaste system.

The deep-bed demineralizer consists of a mixture of cation and anion resins. When the resins are depleted they are either sluiced out and regenerated or disposed of as solid radioactive wastes.

Aside from its normal function of cooling and purifying the spent fuel pool water, the system can be used after reactor refueling to drain the dryer/separator storage pit and reactor cavity. Drain lines allow transport of the water to either the contaminated condensate storage tanks or to the radwaste disposal system for processing, depending upon water condition. Usually water from the dryer/separator storage pit and reactor cavity is drained to the torus via the shutdown cooling system.

The system maximum heat load capacity is approximately 7.25×10^6 Btu/hr at a pool temperature of 125 °F. Each pump and heat exchanger is rated at 700 gal/min and will handle the heat load imposed on the system during normal spent fuel storage, which is approximately 3.65×10^6 Btu/hr. In the event that a pump or heat exchanger should become inoperative, the cooling load can be handled by the remaining pump and/or heat exchanger until such time as the failed equipment can be repaired. Either one or both loops may be used, dependent upon the spent fuel heat load in the pool, for a total design flowrate of 1400 gal/min.

The shutdown cooling system may be connected in parallel with the spent fuel pool cooling system to assist in cooling the pool during periods of extremely high heat loads, such as immediately after refueling or a full core discharge. The shutdown cooling system consists of three pumps and heat exchangers. Each heat exchanger is rated at 27×10^6 Btu/hr. Refer to Section 5.4.7 for a description of the shutdown cooling system.

The two spent fuel pool heat exchangers, with one shutdown cooling heat exchanger, are capable of handling the decay heat load of a full core discharge plus the two most recently discharged batches of fuel. The cross connection is designed for 1500 gal/min flow from the spent fuel pool to the shutdown cooling system.

Cooling water is supplied to the fuel pool heat exchangers (and shutdown cooling heat exchangers) from the reactor building closed cooling water (RBCCW) system. The RBCCW design flowrate to the spent fuel pool cooling heat exchangers is 1600 gal/min per heat exchanger and is designed for 105°F maximum temperature with 95°F service water temperature. A sample point is incorporated to determine any tube leakage.

The spent fuel pool filters and the skimmer surge tanks are shielded with concrete to give a design radiation level of 5 mrem/hr outside the shielded area.

The spent fuel pool cooling system is controlled from a local panel. The operator is provided with indications of system flow, pool water level, water temperature, skimmer surge tank level, and valve positions. Annunciator alarms are provided for spent fuel pool high level, skimmer surge tank low level, spent fuel pool cooling pump discharge low pressure, spent fuel pool filter high differential pressure and low flow, spent fuel pool demineralizer high differential pressure and high effluent conductivity, and spent fuel pool cooling system temperature before and after the heat exchangers.

Initial filling and level maintenance in the spent fuel pool and surge tanks is from the condensate transfer system. When the spent fuel shipping cask (or any other object) is placed in the pool, the water level will proportionately rise in the surge tank. When these objects are withdrawn and the skimmer surge tank level drops to maintain pool level, a low-level alarm on the surge tank will notify operating personnel if this level is not within limits.

A float-type level switch in the spent fuel pool will actuate an alarm in the control room on high spent fuel pool level. There is no pool low-level alarm other than that inherent in the skimmer surge tank low-level alarm.

The spent fuel pool cooling and cleanup system is in continuous operation.

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9.1.3.3 Safety Evaluation

Various precautions are taken to minimize water loss from the system. All penetrations into the pool are located above a fixed height from the bottom such that there must always be a safe level of water above the fuel.

The two spent fuel pool cooling system return lines to the spent fuel pool each have openings in the pipe about 6 inches below the pool surface to act as anti-syphon devices by allowing air into the pipe to break the vacuum if siphoning begins. This precludes uncontrolled draining of the spent fuel pool in the event of a pipe break. Skimmer surge tank level is monitored and a low-level alarm, which could be indicative of a loss of water from the spent fuel pool, is provided in the main control room.

Refer to Section 9.1.2.3 for an evaluation of the spent fuel pool temperature response to a complete loss of spent fuel pool cooling and maximum spent fuel pool temperature for various combinations of operating pumps and heat exchangers.

The chemistry of the spent fuel pool is maintained in accordance with Dresden Chemistry procedures.

Although not a regulatory obligation, the FPC system can be restarted without operator action inside the reactor building in case the FPC system temporarily loses power concurrent with a LOCA. This capability is needed because the high dose rates following a LOCA will make operator entry into the reactor building undesirable. Although the extended loss of spent fuel pool cooling concurrent with a LOCA has been evaluated as beyond the design and licensing bases for the plant, the capability to restart the FPC system without operator action inside the reactor building may be maintained as a prudent action.

9.1.4 Fuel Handling System

9.1.4.1 Design Bases

The design objectives of the fuel handling system are to receive and transfer nuclear fuel in a way that precludes inadvertent criticality and to provide equipment for handling both new and irradiated fuel. To achieve these objectives the fuel handling equipment is designed to handle fuel assemblies and other reactor components described in Chapter 4. There will be no release of contamination or exposure of personnel to radiation in excess of 10 CFR 20 limits.

The reactor building overhead cranes were designed such that all crane parts meet or exceed design criteria as established by Crane Manufacturers Association of America (CMAA) Specification 70 and are compatible with the requirements of the Occupational Safety and Health Act of 1970, as amended in 1971, as well as ANSI B30.2.0.

9.1.4.2 System Description

The major components of this system are the refueling platform and the reactor building overhead crane.

Underwater vacuum cleaning equipment is available for removal of dirt and small particles from sections of the spent fuel pool floor not obstructed by racks or other equipment. A variety of special tools is provided for remote handling of fuel and reactor internals and for exchanging fuel channels. Refer to Section 1.2 for drawings showing a complete layout of the operating floor.

9.1.4.2.1 Refueling Platform

Each unit is provided with a refueling platform equipped with a refueling grapple, two ½-ton auxiliary hoists, and associated equipment. Either of these hoists can be positioned for servicing the reactor cavity or the spent fuel pool.

The refueling platform equipment assembly provides a rigid superstructure which moves on parallel tracks to the location of components either in the spent fuel pool or in the reactor well. Using the fuel grapple or designated accessories, a component can be grappled and transported to a storage or installation position.

The refueling platform consists of a track-mounted bridge which includes an electric motor drive manual and optional semiautomatic controls, instrumentation, and service facilities required to support the operation of the fuel grapple or the handling equipment used at the auxiliary hoists. The refueling platform bridge includes a walkway; railings; a trolley-mounted control cab located on the forward (fuel pool) side; a main grapple hoist; the adjacent frame-mounted auxiliary hoist; a reverse-mounted (reactor side) monorail auxiliary hoist; a hinged jib arm power winch; and the reels, drives, pulleys, and sheaves required for the hoist cables and the service air lines from the self-contained, refueling-platform-mounted air compressor. Service connections include power, service air, and communication receptacles. Bridge motion is in the "X" (east-west) direction. Trolley motion is in the "Y" (north-south) direction. The Hoist motion is in the "Z" (up-down) direction. Bridge and trolley motion controls are provided for fixed speed operation and variable speed operation. The fixed speed control (creep speed) is preset at 10 ft/min. The variable speed control permits operation up to 30 ft/min for the trolley and 50ft/min for the Bridge.

9.1.4.2.1.1 Main Hoist

The main hoist is attached to the trolley and is positioned horizontally by the bridge and trolley motion controls. The fuel grapple consists of a mast and head assembly and accessories. The sectional, telescopic mast is gimbal-mounted to a bearing and hanger assembly installed in the upper floor of the control cab trolley and suspended through the slot, thus providing for lateral (Y-axis) movement. The primary equipment controls, except for the monorail auxiliary hoist, are mounted to the tower section in the operator's cab and optional semiautomatic controls from the touch screen. The position of the main fuel grapple with respect to the core grid may be measured by machine-mounted transducers and displayed to the operator on the digital display. To prevent raising the mast to an overhoist position, prohibitive interlocks are established on the mast. Two limit switches are installed on the upper mast section. One of these switches is the normal-up limit and the other the overhoist limit. The grapple normal-up limit may be overridden to allow raising the hoist to the overhoist limit. A mechanical stop at the top of the mast provides a hard-stop capability. The mast is powered from the main hoist motor assembly. Grapple vertical movement is controlled by a variable speed circuit. The variable speed controller permits operation up to 40 ft/min.

The main hoist drive is mounted on a base located above the operator cab but below the main trolley upper deck. In addition to the motor brake, a second safety brake is provided.

The main grapple head at the end of the telescoping mast includes counteracting primary and secondary hooks integrated to double latch the fuel bundle bail from both sides (Figure 9.1-16). Limit switches are used to provide open or closed indication for the main grapple hooks during operation. Each hook is provided with a welded eye-lug for manual operation if required.

9.1.4.2.1.2 Auxiliary Hoists

There are two auxiliary hoists on the refueling platform. The first, known as the frame-mounted hoist, is mounted on the main trolley along with the main fuel hoist. The second, known as the monorail auxiliary hoist, is mounted on its own trolley which travels the length of the platform and works over the side of the platform opposite the main and frame-mounted hoists.

The frame-mounted hoist cable allows travel of approximately 85 feet. The end points of travel are from an elevation approximately at the upper portion of the personnel walkway deck handrails to approximately 85 feet below that point. The hoist has a 1000-pound load interlock on raising. Control is a dual-speed, momentary-contact control at 10 and 30 ft/min. The raising motion of the hoist is blocked by a drum revolution counter switch at a point about 14 feet below the personnel walkway deck. Raising the hoist above that point can be accomplished only by simultaneously pushing the HOIST RAISE and the RAISE OVERRIDE pushbuttons.

The frame-mounted hoist raising motion is also blocked by an adjustable jamming button, a cylinder-shaped weight with tapered ends which is bolted onto the hoist cable. It is designed to lift a hairpin limit switch mounted alongside the cable on the underside of the trolley upper deckplate. In the event the hairpin limit switch does not block the raising of the hoist electrically, the jamming button will mechanically jam the hoist and stall the motor.

The frame-mounted hoist lowering motion is blocked at a point about 85 feet below the personnel walkway deck by the drum revolution counter switch. The hoist is also supplied with dual air hoses with a spring motor take-up reel to follow the hoist cable up and down under tension.

The monorail hoist is mounted on the monorail trolley which travels on a rail attached to the backside of the upper walkway. It is operated from the personnel walkway by means of a control pendant suspended from the monorail hoist trolley. Its line of travel is parallel to the personnel walkway, opposite the operator's cab.

The monorail hoist moves parallel to the personnel walkway at 30 ft/min controlled from the monorail hoist controller. The other functions and controls are identical to those of the frame-mounted hoist.

9.1.4.2.1.3 Instrumentation and Control

All control of the refueling bridge is PLC bused. Control can be from one of the three control stations but at only one Station at a time for the Bridge and Trolley motion. Which station is controlling is selected at the main hoist operator's console. Platform control can be from either the main console, the frame-mounted auxiliary control pendant, or the monorail auxiliary control pendant. Optional semiautomatic control is also provided for positioning of the bridge and main hoist.

The main hoist, monorail hoist, and frame-mounted hoist use a load cell in conjunction with the PLC. The load cell is an electric system which provides a hoist load readout anytime a load is applied to the grapple. All three hoists can operate independently.

The main hoist utilizes a load cell which provides slack cable indication. The force switch provides a rod block interlock for the hoist-loaded indication and another provides the hoist jam indication.

The load cell switch arrangement for the auxiliary monorail and frame-mounted hoists is the same as the arrangement for the main hoist. The load cell is also used to provide hoist-loaded indication, rod blocks and hoist jam indication.

The main hoist grapple position in the horizontal and vertical planes are displayed on digital readouts and swing arm display as follows:

A. Bridge position - Provides X (east-west) position; and

B. Trolley position - Provides Y (north-south) position; and

C. Hoist position - Provides Z (up-down) position; and

The bridge and trolley digital readouts are designed to provide reactor core coordinates to the platform operator. A backup system of X and Y position indication may be provided along the bridge tracks and the trolley cab by lighted or fixed pointers.

9.1.4.2.2 Reactor Building Overhead Crane

The operating floor is serviced by the reactor building overhead crane, which is equipped with a 125-ton main hoist and a 9-ton auxiliary hoist. These hoists can reach any equipment storage area on the operating floor. The main hoist is operated by a main motor. A second motor operates the auxiliary hoist.

The reactor building overhead crane handling system consists of an overhead, bridge-type crane, spent fuel cask lifting devices, and controls (Figures 9.1-17 and 9.1-18). The overhead crane handling system is used for lifting and transporting the spent fuel cask between the spent fuel pool and the cask decontamination/work area. It is also used to lift and move other equipment and reactor components accessible from the refueling floor. The overhead crane is located in a controlled environment in the reactor building. The crane main hoist system consists of a dual load path through the hoist gear train, the reeving system, and the hoist load block along with restraints at critical points to provide load retention and minimization of uncontrolled motions of the load in the event of

failure of any single hoist component. Redundancy has also been designed into the main hoist and trolley brakes, the spent fuel cask lifting devices, and crane control components. The system will prevent all postulated credible single component failures over the entire supporting load path.

Both the bridge and trolley meet the CMAA fatigue loading requirements. These requirements are stated in Table 3.3.3.1.3-1 of CMAA Specification No. 70. The service classification for the reactor building overhead crane is Class A1 which is designed for 100,000 loading cycles. The weldments fall into categories B and C which permit a stress range of 28,000 - 33,000 psi.

The reactor building overhead crane is classified as Safety Class II equipment and is not seismically qualified.

The main and auxiliary hoists have power control braking as well as two holding brakes. The brakes provided are dc-magnet-operated, electric-shoe-type with a maximum torque rating of 200% of motor torque. The brakes are applied whenever the dc solenoids are deenergized. One dc brake is provided to stop the bridge when deenergized. A manually operated hydraulic brake is also capable of stopping the bridge.

Spring bumpers effective for both directions of travel are provided on the outboard ends of the bridge truck. Crane runway stops with four spring-type trolley bumpers are mounted on runway girders at the ends of the runway rails. The reactor building overhead crane has safety lugs on both the bridge and the trolley that prevent derailment. The stops also would prevent the reactor building overhead crane trolley from falling into the spent fuel pool.

The reactor building overhead crane is provided with three emergency shutdown switches located on the reactor building walls (elevation 613'-0").

A cable train equalizer feature guards against lifting loads that are not positioned directly beneath the main hook or when the load is off balance. If, during crane operation, the respective lengths of the redundant cable trains become excessively different, the equalizer circuit will disable all crane functions and activate a rotating red light. Following such a trip, all crane functions except raising and lowering may be restored by operating the EQUALIZER BYPASS keyswitch located in the cab.

The main hoist and auxiliary hoist each have two upper limit switches for the lifting circuit and one limit switch for the lowering circuit which inhibit the operation of the hoist in either direction when the upper or lower limit is reached. One set of the main hoist limit switch contacts is used for the main hoist motor control. This switch is used to restrict lifting to a predetermined limit in the crane's restricted mode.

A digital-type weight indicator for the main hoist is provided. High- and low-load limits can be set manually on the unit with contact closures available for the set weight limits. The contacts operate the main hoist motor. When the weight to be lifted is above the setpoint on the weight indicator, the control circuit for the main hoist motor will prevent its operation and the main hoist brakes will set. As an alternative to the digital load limiter, station procedures require supervising personnel to ensure load hangups do not occur during reactor building crane operation.

A radiation monitor is mounted on the crane bottom and is interlocked with the crane controls. A radiation monitor reading of less than 0.1 mrem/hr will give a low-radiation alarm indicating equipment malfunction. This will actuate an interlock which prevents further upward movement of either hoist. A radiation monitor reading of 30 mrem/hr or greater will give a high-radiation alarm and actuates an interlock which also prevents further upward movement of either hoist. This interlock prevents upward hoisting movement when a highly radioactive component is lifted too close to the surface of the pool water, although lowering or lateral movements are still permitted.

The high-radiation interlock may be bypassed in the crane cab. In the bypass position, movement of a high-radiation load is permitted after the high-radiation interlock has stopped the crane.

The crane can be operated in either a NORMAL or RESTRICTED mode. In the RESTRICTED mode, the crane bridge and trolley movement is restricted to ensure the crane remains within a predefined pathway. Limit switches are provided for both bridge and trolley forward and reverse directions of travel. When the limits are reached, the limit switches deenergize the control circuits. In the NORMAL mode the pathway limit switches are bypassed.

Fuel cask handling above the 545-foot level of the reactor building is considered a "restricted load" and must be performed, to the extent practical, in the restricted mode. The restricted mode may be exited to permit maneuvering the fuel cask around permanent components or structures. Fuel cask handling in other than restricted mode due to emergency or equipment failure situations is permitted to the extent necessary to move the cask to the closest acceptable and stable location.

A RESTRICTED MODE/NORMAL MODE key selector switch activates the restricted pathway limit switches and the main hoist restricted mode upper limit switches. The main hoist must be centered over the pathway when the RESTRICTED MODE is selected. Once on the pathway, the main hoist must also be raised to engage the restricted mode upper limit switches before the bridge or trolley will move. Either one of the two limit switches failing to function properly will prevent movement of the bridge or trolley. The restricted mode upper limit switches must be adjusted so that the restricted load (spent fuel cask) will be lifted a minimum of 6 inches above the refuel floor to avoid bumping into concrete curbs and other obstructions along the haul path (Reference 20 and 21). This adjustment will depend on the height of the spent fuel cask to be lifted.

For carrying a restricted load, the RESTRICTED MODE/NORMAL MODE selector switch RESTRICTED MODE is selected. The hoist FAST/SLOW switch SLOW position is selected to disable the main hoist fast speed circuitry.

Limit switches on the hoist equalizer bar will shut off power to the bridge, trolley, and the hoist when they open. Either one of the two limit switches will open the power circuit for the crane.

Administrative control is exercised by the key-operated restricted mode switch and the key-operated bypass switch for the equalizer bar limit switches.

The brakes for the hoist will set for any one of the following conditions and keep the load in a safe position:

A. AC power supply failure;

B. Either one of the two limit switches on the equalizer bar open;

C. Main hoist restricted area upper and final upper limit switches open after reaching the set limit;

D. Main hoist weight upper limit switch;

E. Main hoist area upper limit switch;

F. High load contact of the digital indicator control panel;

The circuitry is designed such that only the bridge or trolley can be operated at one time. The bridge and trolley brakes will set for any one of the following conditions and keep the restricted load in a safe position:

A. AC power supply failure;

B. Either one of the two limit switches on the equalizer bar open;

C. Main hoist restricted area upper and final upper limit switches open after reaching the set limit;

D. Field loss relay trip;

E. Instantaneous overcurrent relay trip;

F. Bridge or trolley forward/reverse limit switch open; and

9.1.4.2.3 Fuel Handling Operations

A variety of specialized tools and servicing equipment are utilized for fuel handling. Table 9.1-1 is a list of equipment typically (not all inclusive) used. Use of this equipment for fuel handling is outlined below.

New fuel is brought into the reactor building through the equipment entrance. A rail crane is provided in the equipment access area for removal of new fuel from trucks and, if necessary, for movement of the multiassembly transfer basket. The new fuel is hoisted to the upper floor for storage utilizing the reactor building overhead crane.

Prior to refueling, the new fuel is transferred to the spent fuel pool using the reactor building overhead crane or new fuel transfer crane for Unit 2 and the reactor building overhead crane for Unit 3. If new fuel channels are to be used, the fuel is channeled on the new fuel inspection stand before being placed in the spent fuel pool. If previously irradiated channels are to be used, the fuel is placed into the pool, then channeled under water with special tools. Present channel management strategy is to only use new flow channels on new fuel.

Prior to initial reactor fueling, the spent fuel pool, reactor head cavity, and reactor internals storage pit were filled with water and checked for leakage. Dummy fuel assemblies were run through a complete cycle from the new fuel storage vault to the spent fuel pool. Prior to fuel handling, all hoists, cranes, and tools are inspected and tested to assure safe operation.

In preparation for refueling, the concrete shield plugs in the reactor cavity and the transfer canal are removed using the reactor building overhead crane. The drywell head, head insulation, and reactor vessel head are removed, using the same crane. The steam dryer assembly may be moved to the dryer/separator storage pit either before or after flooding of the storage pit. The steam separator assembly is unbolted from the core structure using a special hand tool. Contaminated demineralized water from any of several sources, as convenient, is pumped into the reactor until the reactor cavity and the dryer/separator storage pit are flooded to the normal level of the spent fuel pool. The steam separator assembly is then transferred to the dryer/separator storage pit.

The reactor water cleanup system (see Section 5.4.8) may be operated to assure sufficient water clarity for fuel movement, then the spent fuel pool gates are removed. Spent fuel is removed from the reactor using the main fuel grapple and placed in racks in the spent fuel pool. The same equipment is used to transfer the new fuel and exposed fuel to be reloaded from the spent fuel pool to the reactor. The racks in which fuel assemblies are placed are designed and arranged to ensure subcriticality in the pool.

When reactor refueling is complete, the spent fuel pool gates are set back in place. The steam separator assembly is returned to the reactor. The steam separator assembly is bolted down and the steam dryer assembly is replaced. The water in the dryer/separator storage pit is pumped down to the raised step at the bottom of the pit, and reactor cavity is pumped down to the reactor vessel flange. The reactor vessel head, insulation, drywell head, and concrete shield blocks are replaced.

During core alterations, direct communication is maintained between the control room and refueling platform personnel.

During and after refueling operations, some fuel channels may be removed from the fuel using the underwater channeling equipment. Fuel channels on spent fuel may be taken off and put onto new fuel or may be swapped with those on partially spent fuel, if they are relatively undeformed (i.e., not bowed, bulging, or twisted past certain limits). However, present channel management strategy is to only use new channels on new fuel. Channels which can no longer be used are temporarily stored in the spent fuel pool and eventually disposed of offsite as solid radwaste.

When the spent fuel is to be loaded for onsite storage or offsite shipping, a spent fuel cask is used. After confirmation of the operational acceptability of the reactor building overhead crane as described in Section 9.1.4.4.2, the fuel cask will be hoisted to the refueling floor and moved over a controlled path to the decontamination pad and then to its position in the spent fuel pool. The spent fuel is loaded into the cask under water. The cask is closed and removed from the pool and moved over to the cask wash-down area.

After decontamination, the spent fuel cask is lowered by the reactor building overhead crane to a truck, railway car or similar transport device in the equipment access area.

9.1.4.3 Safety Evaluation

This section discusses the safety aspects of the refueling platform, reactor building overhead crane, and associated equipment design. The analyses of the radiological consequences for fuel handling accidents are discussed in Section 15.7.

9.1.4.3.1 Refueling Platform

Protective interlocks prevent handling of fuel over the reactor when a control rod is withdrawn, and another set of interlocks prevents control rod withdrawal when fuel is being handled over the reactor. Optional boundary zones generated by the PLC prevent moving the bridge outside of the fuel pool area or reactor cavity without additional operator action. The telescoping fuel grapple in the normal up position cannot lift any load, including fuel assemblies, which would result in less than seven feet of water coverage at normal fuel pool water level. For a fuel assembly, the top of active fuel (pellets) is about 18 inches below the bale handle. This results in about 8.5 feet of water coverage over fuel in the normal up position.

Fuel stored in the spent fuel pool is covered with sufficient water for radiation shielding, and fuel being moved is at all times covered by a minimum depth of 8 feet of water. Spent fuel will not be handled with an inadequate depth of water shielding. However, the geometry of the TN9.1 fuel transfer cask requires that less than 8 feet of water be above a spent fuel bundle during transfer to the cask.

An evaluation of the dose rates at the spent fuel pool surface as a function of fuel bundle age for a 5-foot water shield above a spent fuel assembly has been completed. For calculational purposes, zero age was taken to be reactor shutdown. The active fuel length is 12 feet. The plenum above the fuel was assumed to be filled with normal density water. The assembly was assumed to be 5 inches on a side and to have an average power of 3.54 MWt. This was based on an equilibrium exposure of a 2561 MWt core containing 724 assemblies. A peaking factor of 1.5 was used to consider dose rates associated with a highly activated assembly. The water shield was placed 5 feet above the assembly, not above the active fuel. The neutron activation contribution to the dose rate was based on 1000 grams of Zircaloy-2 (wt.%: 97.89 Zr, 0.06 Ni, 0.15 Cr, 0.20 Fe, and 1.7 Sn) located 5 feet below the surface of the fuel pool.

The self-shielding properties of UO_2 were not considered in calculating the dose rates given in Table 9.1-2 and shown in Figure 9.1-19. With self-shielding considered, the fuel geometry is such that the dose rate calculations would yield a lower dose rate along the axis of the assembly. However, some points on the fuel pool surface are essentially unaffected by the UO_2 self-shielding effect. The calculation assumed the fuel assembly was filled with fission products associated with a 3.54 MWt fuel bundle but with normal density water as the self-shielding material.

If operations dictate that the assemblies be moved in 90 days, the dose rates at the surface of the spent fuel pool for the average and the highly activated assembly would be approximately 14 and 21 mrem/hr respectively.

During actual fuel transfer to/from the cask, the actual water shield thickness is 5 feet, 10⅜ inches. The expected radiation dose will then be decreased by approximately a factor of 10.

The above dose rates and the dose rates in Table 9.1-2 and Figure 9.1-19 for a spent fuel assembly will increase by approximately 20% following power uprate to 2957 MWt and a 24 month fuel cycle. It is noted that the fission products in a long-cooled extended burn fuel bundle will increase more than 20% because of the increase in long-lived isotopes. However, it is estimated that the conservative modeling of ignoring the self-shielding of UO_2 will compensate for dose rate increases beyond the estimated 20% discussed above.

9.1.4.3.2 Reactor Building Overhead Crane

The 125-ton capacity reactor building overhead crane main hoist is designated as a single failure proof crane for 110-ton loads. The NRC has approved use of the reactor building overhead crane during power operations to lift a total load up to 116 tons for removal and installation activities for the reactor shield blocks prior to and during Unit 3 refueling outage D3R17. Within the dual load path, the design criteria are such that all dual elements comply with the CMAA Specification No. 70 for allowable stresses, except for the hoisting rope which is governed by more stringent job specification criteria. With several approved exceptions, single element components within the load path (i.e. the crane hoisting system) have been designed to a minimum safety factor of 7.5, based on the ultimate strength of the material. Components critical to crane operation, other than the hoisting system, have been designed to a minimum safety factor of 4.5, based on the ultimate strength of the material. Table 9.1-3 lists the results of the crane component failure analysis.

The reactor building overhead crane and spent fuel cask yoke assemblies meet the intent of NUREG-0554 for loads less than or equal to 110 tons.

All analyses for handling spent fuel casks, performed relative to the overhead crane handling system loads have been based on the National Lead (NL) 10/24 spent fuel shipping cask which weighs 100 tons (Figure 9.1-18) and the HI-TRAC 100 transfer cask which weighs less than 100 tons (Section 9.1.2.2.4). If larger casks are used, additional analyses will be required to assure safety margins are maintained.

Administrative controls and installed limit switches restrict the path of travel of the crane to a specific controlled area when moving the spent fuel cask. The controls are intended to assure that a controlled path is followed in moving a cask between the decontamination and hatchway area and the spent fuel pool. Administrative controls also ensure movement of other heavy loads such as the drywell head, reactor vessel head, and dryer separator assembly is over preapproved pathways. If the Technical Specifications required AC sources during shutdown conditions become inoperable while moving irradiated fuel in the Secondary Containment, then crane operations over the Spent Fuel Storage Pool must be suspended until the required AC sources are returned to operable status. Additionally, crane operations over the Spent Fuel Storage Pool are required to be suspended when Spent Fuel Storage Pool level is not within the Technical Specification limit and irradiated fuel assemblies are in the Spent Fuel Storage Pool.

Technical Specifications state refueling requirements. Station procedures prohibit movement of heavy loads over the spent fuel pools or open reactor cavity except under Special Procedures

The crane reeving system does not meet the recommended criteria of Branch Technical Position APCSB 9-1 (now incorporated into NUREG-0554) for wire rope safety factors and fleet angles. The purpose of these criteria is to assure a design which minimizes wire rope stress wear and thereby provides maximum assurance of crane safety under all operating and maintenance conditions. Because the crane reeving system does not meet these recommended criteria, there is a possibility of an accelerated rate of wire rope wear occurring. Accordingly, to compensate in these design areas, a specific program of wire rope inspection and replacement is in place.

The inspection and replacement program assures that the entire length of the wire rope will be maintained as close as practical to original design safety factors at all times. This inspection and replacement program provides an equivalent level of protection to the methods suggested in wire rope safety and crane fleet angle criteria and will assure that accelerated wire rope wear will be detected before crane use.

"Two blocking" is an inadvertently continued hoist which brings the load and head block assemblies into physical contact, thereby preventing further movement of the load block and creating shock loads to the rope and reeving system. A mechanically operated power limit switch in the main hoist motor power circuit on the load side of all hoist motor power circuit controls provides adequate protection.

against "two blocking" in the event of a fused contactor in the main hoist control circuitry. This power limit switch will interrupt power to the main hoist motor and cause the holding brakes to set prior to "two blocking."

The reactor building refueling floor has been designed for a live load of 1000 lb/ft^2. The entire reactor building refueling floor (with the exception of the fuel pool and open reactor cavity) is considered a safe load path zone.

A 9-ton load drop has been analyzed (Reference 17 through 18). The results show that the refueling floor can survive a drop from 7 feet without scabbing damage. Procedures limit the 9-ton lift height to a maximum of 7 feet. Existing procedural controls limit both the height of a lift to clear obstacles and require the use of the most direct path to laydown areas.

A load drop analysis has been performed in Reference 11 for handling the top layer of the Units 2 and 3 reactor cavity shield blocks weighing up to 116 tons for the designated safe load path that demonstrates that a postulated load drop will not affect any safety-related equipment, since there will be no scabbing or perforation of the refueling floor, and the overall response of the floor system will be acceptable. This load drop analysis was performed in accordance with the guidelines of NUREG-0612, Appendix A. The load drop analysis for the shield blocks was approved by the NRC for Dresden in Reference 12, and the methodology can be used to adjust the Reference 11 calculation for variation in the weight of the shield blocks. Extension of the methodology is not approved for application to ether heavy loads or load paths.

The load drop analysis used the following key assumptions and methods.

- The weight of the dropped top half-layer reactor cavity shield block is considered to be l16 tons, including the slings and other rigging used to lift the cavity shield block, and excluding the weight of the crane load block. The maximum drop height for the 116-ton shield block is assumed to be 1' - 0" above the floor.

- The overall adequacy of the impacted structural elements is determined by calculating the total strain energy in the impacted elements corresponding to an allowable ductility limit, and comparing this energy to the impact energy imparted to the impacted elements.

- The kinetic energy of the reactor cavity shield block at impact is conservatively assumed to be transferred entirely to the impacted structural elements.

- The energy absorption of the impacted elements is calculated using constructed elasto-plastic load-deflection diagrams for the elements. The ductility limit is determined using Reference 13, Appendix C, Section C.3, and the area under the load-deflection diagram up to the applicable ductility limit is used as the measure of the energy absorption capacity of the elements.

- The shear failure load is estimated using Reference 14. The shear failure load is at least 1.20 times the flexural resistance load in order to use the flexural mode of failure to calculate the strain energy. Otherwise, the ductility ratios given in Reference 13, Section C.3.7 or C.3.9 are used.

- The calculation uses the actual concrete compressive strength, as noted in station documents.

- The potential for scabbing of the underside of the refueling floor is investigated based on drop of the cavity shield block when one of the three lift points of the lifted cavity shield block fail. The calculation is based on the local damage equations given in Reference 15.

The designated safe load path, hoisting height restrictions, and the maximum weight of the reactor cavity shield block and rigging are described in applicable procedures. When handling the top layer of shield blocks weighing more than 110 tons, crane controls incorporate travel limits and hoisting height restrictions.

The reactor building overhead crane meets the single-failure criteria stated in NUREG-0612 for heavy loads of 110-tons. The NRC has approved use of the reactor building overhead crane during power operations to lift a total load up to 116 tons for removal and installation activities for the reactor shield blocks prior to and during Unit 3 refueling outage D3R17. As required by CMAA-70, the maximum crane load weight plus the weight of the bottom block, divided by the number of parts of rope does not exceed 20% of the manufacturer's published breaking strength.

The reactor building overhead crane main hook has:

A rated load capacity	=	250,000 lb
Block and rope weight	=	20,500 lb
Total weight lifted	=	270,500 lb

This weight is supported by 12 parts of wire rope with a published breaking strength of 175,800 pounds.

$$\frac{\text{Total weight lifted/Number of parts of rope}}{\text{Breaking strength of rope}} = \frac{270{,}500}{12 \times 175{,}800} = 12.8\% \qquad (1)$$

As can be seen by Equation 1, this is less than the 20% CMAA-70 requirement.

A detailed analysis of the possibility of horizontal displacement of the cask in the event one of the redundant rope trains fails has been conducted. It has been confirmed that the horizontal load displacement will not exceed 2.5 inches throughout the critical elevations of lift. At the high point of the lift, with the cask above the operating floor, the static displacement of the load is approximately 0.5 inch with a total static plus dynamic displacement of approximately 1 inch. The total horizontal displacement of the load when the cask is submerged in the spent fuel pool is approximately 2.5 inches. A larger total horizontal displacement, approximately 9 inches, can occur with the load at its lowest elevation, that is with the load at the grade elevations. However, it should be noted that the NL 10/24 100-ton cask and the HI-TRAC 100 cask, which are the heaviest loads to be lifted through the equipment hatchway, are 7.33 feet in diameter and 7.83 feet across the cask yoke and approximately 8.25 feet in diameter and 8.5 feet across the cask yoke respectively. The equipment hatchway has a minimum 20.08 foot square opening (See Figure 9.1-20). Local protrusions of ductwork along the vertical path of the cask through the hatchway reduce the cross section to approximately 19.5 feet. Since the path of the cask is controlled by limit switches which restrict the position of the cask during lifting to ±6 inches from the center line of the hatchway, lateral clearances in excess of 4 feet are available.

9.1.4.4 Inspection and Testing Requirements

Surveillance requirements for the Refueling Platform are stated in Technical Specifications. These are summarized below.

9.1.4.4.1 Refueling Platform

The refueling platform interlocks are functionally tested prior to any fuel handling with the head off the reactor vessel. They are tested at intervals as defined in the Surveillance Frequency Control Program until these interlocks are no longer required. The refueling platform interlocks are tested following any repair work or maintenance associated with the interlocks.

9.1.4.4.2 Reactor Building Overhead Crane

To ensure safe operation during cask transfer, restricted mode testing is performed prior to the activity. This testing ensures the crane will not operate except within approved load paths. The travel will never be over the reactor vessel or the fuel racks in the spent fuel pool. Travel over the spent fuel pool is limited to that small area allocated for cask storage.

Visual inpsections of crane components are performed prior to cask lifting activities. Also, load testing is performed prior to a fuel cask transfer. The load test is conducted by raising a load no higher than 6 inches off its transport for a fixed period of time. This is to verify proper crane operation.

9.1.5 References

1. Deleted.

2. Quadrex Corporation, Qualification Report, QUAD-3-83-002, Revision 1.

3. Deleted.

4. Deleted.

5. Letter, J. A. Zwolinski (NRC) to D. L. Farrar (ComEd), "Technical Specifications Relating to Storage of New and Spent Fuel in the High Density Fuel Storage Racks," dated December 12, 1985. Transmitted Amendment No. 91 to DPR-19 and Amendment No. 85. to DPR-25.

6. GE SIL No. 152, "Criticality Margins For Storage Of New Fuel," dated March 31, 1976.

7. Letter RS-01-161, K. A. Ainger (Exelon) to NRC Document Control Desk, “Additional Plant Systems Information Supporting the License Amendment Request to Permit Uprated Power Operation,” dated August 13, 2001.

8. Safety Evaluation Tracking No. 1998-04-247.

9. Deleted.

10. Deleted.

11. Calculation DRE02-0064 (Rev 0, 0A, 0B), “D2/3 Load Drop Evaluation of the Reactor Shield Plugs.”

12. Dresden Nuclear Power Station, Units 2 and 3 – “Issuance of Amendments – Heavy Loads Handling (TAC Nos. MB7840 and MB7841),” dated October 10, 2003.

13. American Concrete Institute (ACI) 349-97, “Code Requirements for Nuclear Safety Related Concrete Structures,” 1997 Edition.

14. ACI 318-99, “Building Code Requirements for Structural Concrete,” 1999 Edition.

15. American Society of Civil Engineers, “Second ASCE Conference on ‘Civil Engineering and Nuclear Power, Volume V: Report of the ASCE Committee on Impactive and Impulsive Loads,” September 1980.

16. OPTIMA2-TR025DR-SFP, Revision 0A, “New Fuel Vault and Spent Fuel Pool Criticality Analyses for Dresden Units 2 & 3 with SVEA-96 Optima 2 Fuel.” May 2016.

17. Calculation 8.31.0-2 (Rev 0, 1, 2, 3), “Load Drop Analysis for Reactor and Turbine Buildings.”

18. Calculation 8.31.0-4 (Rev 0, 1, 2), “Load Drop Evaluation in the Hatchway of the Reactor Building.”

19. GE-NE-A22-00103-49-01, Rev 01, “Task T0603, Dresden-Fuel Pool Cooling and Cleanup system (DRE and QDC Extended Power Uprate).”

20. Letter from JS Abel (ComEd) to DL Ziemann (US Atomic Energy Commission), application for Dresden License Amendment 22 with attached Dresden Special Report No. 41, dated November 8, 1974.

21. License Amendment No. 204 to DPR-19 for Unit 2 and License Amendment No. 196 to DPR-25 for Unit 3 via Safety Evaluation Report (SER) dated March 2, 2004.

22. DRE14-0049, Revision 0, "Spent Fuel Pool Criticality Analysis for ATRIUM l0XM Fuel – HI-2146153".

23. DRE14-0044, Revision 1, “New Fuel Vault Criticality Analysis for ATRIUM 10XM Fuel”.

Table 9.1-1

TOOLS AND SERVICING EQUIPMENT

Fuel Servicing Equipment

Fuel preparation machines
New fuel inspection stand
Channel bolt wrenches
Channel handling tool
Fuel pool sipper
Channel gauging fixture
General purpose grapples
Fuel inspection fixture
Channel transfer grapple
Channel storage adapter
New fuel transfer stand

Servicing Aids

Pool tool accessories
Actuating poles
General area underwater lights
Local area underwater lights
Drop lights
Underwater TV monitor
Underwater vacuum cleaner
Viewing aids
Light support brackets
Incore detector cutter
Incore manipulator
Underwater viewing tube
Pole Handling System
Mast Mounted Camera

Refueling Equipment

Refueling equipment servicing tools
Refueling platform equipment
Scorpion II 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
Control rod guide tube seal
Incore guide tube seals
Blade guides
Fuel bundle sampler
Peripheral orifice grapple
Jet pump service tools
Peripheral fuel support plug
Peripheral orifice holder

Storage Equipment

Spent fuel pool high density storage racks
Spent fuel pool blade guide storage racks
Channel storage racks
Storage racks (control rod/defective fuel)
In-vessel racks
New fuel storage rack
Defective fuel storage containers
Spent fuel rack adapter lifting tool
Unit 1 canister lid tool
Unit 1 canister handling tool
Unit 1 fuel assembly tool

Table 9.1-2 (1)

DOSE RATE AT FUEL POOL SURFACE (mrem/hr)

Time Post-Shutdown (days)	Average Fuel Bundle	Zircaloy-2	TOTAL	Hot Fuel Bundle	Zircaloy-2	TOTAL
30	34.1	12.5	46.6	51.2	18.8	70.0
60	10.6	10.0	20.6	15.9	15.0	30.9
120	3.90	5.90	9.80	5.88	8.90	14.8
180	2.83			4.25		
210	2.51	2.50	5.01	3.77	3.80	7.57
240	2.24	1.90	4.14	3.36	2.85	6.21
300	1.82	1.03	2.85	2.73	1.55	4.28
360	1.52	0.58	2.10	2.28	0.87	3.15

(1) For spent fuels from the uprated core (2957 MWt), the dose rate would increase by approximately 20%.

Table 9.1-3

COMPONENT FAILURE ANALYSIS
125/9-TON CAPACITY REACTOR BUILDING OVERHEAD CRANE

Item No.	Description	Factor of Safety (Ultimate)	Factor of Safety (Yield)	Redundant(1) Yes/No	Failure Protection If No Redundancy Is Provided	Result of Failure If No Redundancy Is Provided
1.	Sister hook	8.5	4.3	Yes		
2.	Lifting eye	7.4	3.7	Yes		
3.	Hook swivel	8.5	3.8	No	Block housing	Sheaves will be displaced vertically by about 1/4 inch
4.	Hoist rope	7.8	5.3	Yes		
5.	First reduction pinion	62.8	43.6	Yes		
6.	First reduction gear	31.0	13.0	Yes		
7.	First reduction shaft	35.0	23.3	Yes		
8.	Second reduction pinion	26.5	19.0	Yes		
9.	Second reduction gear	12.7	5.3	Yes		
10.	Second reduction shaft	47.3	36.4	Yes		
11.	Drum pinion shaft	9.7	6.4	Yes		
12.	Pinion key	11.7	6.3	Yes		
13.	Gear key	13.5	7.3	Yes		
14.	Extension shaft	39.3	26.2	Yes		

Table 9.1-3 (Continued)

COMPONENT FAILURE ANALYSIS
125/9-TON CAPACITY REACTOR BUILDING OVERHEAD CRANE

Item No.	Description	Factor of Safety (Ultimate)	Factor of Safety (Yield)	Redundant(1) Yes/No	Failure Protection If No Redundancy Is Provided	Result of Failure If No Redundancy Is Provided
15.	Coupling key	15.9	8.9	Yes		
16.	Extra reduction pinion	14.8	10.7	Yes		
17.	Extra reduction gear	12.0	6.2	Yes		
18.	Main drum	7.6	3.8	No	Increased safety factor	Uncontrolled descent of load
19.	Drum shaft	12.3	9.7	No	Safety lugs to constrain drum hubs	Drum will be displaced vertically by about 1/8 inch
20.	Drum bearing support	11.48	8.8	No	Increased safety factor	Uncontrolled descent of load
21.	Drum pinion bearing support	31.7	14.5	No	Increased safety factor	Uncontrolled descent of load
22.	Rope anchor	11.3	8.4	Yes		
23.	Equalizer bar	24.2	12.6	Yes		
24.	Equalizer bushing	22.6	17.0	No	Equalizer bar will rest on pin	Equalizer bar will be displaced vertically by 3/8 inches
25.	Equalizer pin	12.9	8.6	No	Equalizer bar will rest on support frame	Equalizer bar will be displaced by 1¼ inches

Table 9.1-3 (Continued)

COMPONENT FAILURE ANALYSIS
125/9-TON CAPACITY REACTOR BUILDING OVERHEAD CRANE

Item No.	Description	Factor of Safety (Ultimate)	Factor of Safety (Yield)	Redundant(1) Yes/No	Failure Protection If No Redundancy Is Provided	Result of Failure If No Redundancy Is Provided
26.	Equalizer bar support frame	9.3	4.2	No	Load girt will catch equalizer bar	Equalizer bar will be displaced by 1¼inches
27.	Load sensing sheave frame	24.4	14.7	No	Increased safety factor	Uncontrolled descent of load
28.	Sheave pins	40.5	26.9	No	Sheave frame will catch sheaves	Sheaves will be displaced vertically by ½ inch
29.	Load cell pins	7.5	5.9	No	Load girt will catch sheave frame	Sheave frame will be displaced vertically by ½ inch
30.	Rod eye	6.5	3.3	No	Load girt will catch sheave frame	Sheave frame will be displaced vertically by ½ inch
31.	Load cell bracket	6.7	3.1	No	Load girt will catch sheave frame	Sheave frame will be displaced vertically by ½ inch
32.	Main load girt I	11.9	6.2	No	Increased safety factor	Uncontrolled descent of load
33.	Main load girt II	6.5	3.4	No	Increased safety factor	Uncontrolled descent of load

Table 9.1-3 (Continued)

COMPONENT FAILURE ANALYSIS
125/9-TON CAPACITY REACTOR BUILDING OVERHEAD CRANE

Item No.	Description	Factor of Safety (Ultimate)	Factor of Safety (Yield)	Redundant[1] Yes/No	Failure Protection If No Redundancy Is Provided	Result of Failure If No Redundancy Is Provided
34.	Trolley truck	7.5	3.9	No	Increased safety factor	Load girts will be displaced vertically about 1 inch
35.	Trolley wheel axle	11.9	7.9	No	Increased safety factor	Truck will be displaced vertically by 1 inch
36.	Bridge truck	8.6	4.5	No	Increased safety factor	Truck will be displaced vertically by 1 inch
37.	Bridge wheel axle	10.9	6.9	No	Increased safety factor	Truck will be displaced vertically by 1 inch
38.	Bridge girders	4.7	2.5	No	Increased safety factor	Truck will be displaced vertically by 1 inch

Notes:

1. Redundant implies backup system provided.

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9.2 WATER SYSTEMS

The auxiliary water systems for Dresden Units 2 and 3 include the following:

A. Containment cooling service water (CCSW) system;

B. Service water system;

C. Reactor building closed cooling water (RBCCW) system;

D. Demineralized water makeup system;

E. Ultimate heat sink;

F. Condensate storage facilities;

G. Turbine building closed cooling water (TBCCW) system; and

H. Standby coolant supply system.

9.2.1 Containment Cooling Service Water System

9.2.1.1 Design Bases

The design bases of the containment cooling service water system are as follows:

A. To provide the containment cooling function to meet containment capability requirements;

B. To provide redundancy in critical components to meet reliability requirements;

C. To operate without reliance upon external sources of power;

D. To provide for periodic testing and inspection of each component in the system to demonstrate system availability.

The CCSW system also supports the safety grade cold shutdown capability described in Section 6.3.1.2. Containment temperature and pressure capability requirements are described in Section 6.2.1. Containment cooling information is provided in Section 6.2.2.

9.2.1.2 System Description

The CCSW system is an open loop cooling water system consisting of four CCSW pumps and associated valves, piping, controls, and instrumentation. Schematic

diagrams of the CCSW systems for Units 2 and 3 are shown in Drawings M-29, Sheet 2 and M-360, Sheet 2, respectively.

The CCSW system provides cooling water for the containment cooling heat exchangers during both accident and nonaccident conditions, as described in Section 6.2.2. System piping is arranged to form two separate, two pump, flow networks (loops) until the piping downstream of the differential pressure control valve on the discharge of the heat exchanger. At this point, the piping from both loops merge into a common discharge line to the service water 48" header. Each pair of CCSW pumps takes a suction from the crib house via separate supply piping. Two CCSW pumps discharge into a common header which routes the cooling water to that loop's associated heat exchanger. At the heat exchanger, heat is transferred from the low pressure coolant injection (LPCI) subsystem to the CCSW system, and subsequently to the river.

During normal plant operation, the CCSW system is not operating. Following an accident or other plant evolution which requires containment heat removal, the CCSW system is manually started. Each CCSW pump is rated at 500 hp with a service factor of 1.15. The CCSW pumps are powered by normal ac or diesel generator ac power. Additional CCSW pump information is provided in Table 9.2-1.

The CCSW pumps develop sufficient head to maintain the cooling water heat exchanger tube side outlet pressure 7 psi higher than the LPCI subsystem pressure on the shell side when the flow from one LPCI pump (20 psi higher when the flow from two LPCI pumps) is passing through the heat exchanger. The ΔP is maintained by the manual operation of a differential pressure control valve in the CCSW outlet piping from the LPCI heat exchanger. Maintaining this pressure differential prevents reactor water leakage into the service water and thereby into the river. A minimum of 5000 gpm is necessary to maintain containment cooling.

The four CCSW pumps are located in the turbine building. Two of the four CCSW pumps (pumps B and C) are located in a single, common watertight vault for flood protection. To prevent the CCSW pump motors from overheating, the vault has two vault coolers. The cooling water for each cooler is provided from its respective CCSW pump discharge line through a four-way valve. This valve also permits flow reversal of the cooling water through these coolers to help clean the tubes. Refer to Section 3.4 for a discussion of the flood protection features at Dresden.

A continuous fill of the CCSW system is provided by the service water system or, in the case of a loss of power to the service water pumps, the diesel generator cooling water system may be aligned to provide the continuous fill. This eliminates the potential for water hammer upon CCSW system startup. The diesel generator cooling water system is discussed in Section 9.5.5.

The Unit 2 CCSW loops also provide a safety-related source of service water to the Control Room Emergency Ventilation System (CREVS) Refrigeration Condensing Unit (RCU). Only one CCSW pump is required to support the CREVS RCU. The minimum flow to support operation of a single CCSW pump is 350 gpm, or 10 percent of the design flow through a CCSW pump. If a flowpath through the LPCI heat exchanger is not available, with one CCSW pump supplying cooling water to the CREVS RCU, then other CCSW pump flowpaths will be necessary since the CREVS RCU only uses 121 gpm. Other flowpaths include the CCSW Pump Vault cooler flowpath, CCSW to ECCS Room Cooler flowpath and flowpaths from CCSW loop vents/drains to floor drains. Since the CCSW Pump Vault coolers are only available to the 2B and 2C CCSW Pumps, then the 2A and 2D CCSW Pumps can only supply CREVS if a flowpath is available through the LPCI heat exchanger.

The Unit 2 CCSW B Loop is normally aligned to provide the backup cooling to the CREVS RCU since the 2C CCSW Pump, the 2D CCSW Pump and the CREVS are powered from the Division II electrical system. The 2A and 2B CCSW Pumps are powered from the Division I electrical system.

If the 2A or 2B CCSW Pumps are the backup source of cooling for the CREVS, the RCU function could be lost if either Division I or Division II power is lost whereas, if the 2C or 2D CCSW Pump are the cooling water source, then only a loss of Division II power would result in the RCU being lost. Each Unit has a diesel driven SBO EDG, which is connectable to each Unit's safety-related 4kv Bus. To compensate for any increased chance that the CREVS RCU could be lost due to a loss of power to the 2A or 2B CCSW Pumps, SBO EDG availability, including connectability, is verified whenever the 2A or 2B CCSW Pumps are the safety-related cooling water supply to the CREVS RCU. Refer to Sections 6.4 and 9.4.1 for a description of the control room ventilation system.

The CCSW system also supplies a safety related source of river water to the LPCI and HPCI room coolers as a backup to the service water system.

CCSW heat exchanger supply lines 2(3)-1510-16"-D have 10-inch hose connections to enable an alternative cooling water supply from portable submersible pumps in the Unit 2/3 intake canal. These hose connections, 10-inch hoses, and submersible pumps (hydraulically driven by a diesel powerpack) would be used during a postulated seismic event that fails the Dresden Lock and Dam and creates a 1-inch break LOCA in each unit. This event is further described in Section 9.2.5.3.3.

9.2.1.3 Safety Evaluation

Containment cooling is not immediately required following a design basis loss-of-coolant accident (LOCA). The required timing of the initiation of containment cooling functions by CCSW is described in Section 6.2.1.3.2. One of the two heat exchangers, two CCSW pumps, and one LPCI pump all in the same loop are the minimum requirements for containment cooling.

The CCSW system contains two redundant trains, and any single active failure would not prevent the system from performing its function.

If normal ac power is not available, the CCSW system can be operated using power provided by diesel generators. Section 6.3.2.2.2 describes conditions required to start two CCSW pumps using diesel generator power.

An evaluation was made of the number of CCSW pump vault cooling fans required for adequate vault temperature control. It was determined that two of the four cooling fans are required for each operating CCSW pump with an entering cooling water temperature of 95°F.

9.2.1.4 Testing and Inspection Requirements

Periodic testing of CCSW system pumps and selected valves is performed to demonstrate system operability. Pump testing includes demonstrating delivery of rated flow at a specific pressure.

Piping and heat exchanger infestation by Corbicula (Asiatic clams) has been identified as a hazard to the Dresden safety-related service water systems. This problem has been studied since the 1970's. Dresden Station has implemented a program to trend infestation characteristics. This information is used to ensure flow blockage will not occur in safety-related systems using river water. The program includes:

A. Periodic inspection and cleaning of the intake bays;

B. Periodic biocide injection into the intake bay or service water distribution header;

C. Periodic flushing of infrequently used or stagnant lines in safety-related service water systems;

D. Annual water and substrate sampling;

E. Periodic testing, inspection, and cleaning of safety-related heat exchangers; and

F. Periodic inspection of high- and low-flow service water piping for corrosion, erosion, silting, and biofouling.

This program has been applied to the CCSW system.

9.2.1.5 System Instrumentation

Control room instrumentation includes CCSW loop flow indicators and differential pressure indicators between the tube side and shell side of the heat exchangers. Pressure gauges are installed on the tube side of the containment cooling heat

exchanger to monitor for blockage. Pressure gauges are provided on the CCSW pump vault coolers to monitor for flow blockage and to determine the need for heat exchanger cleaning.

The CCSW pumps and the CCSW heat exchanger discharge motor-operated valves are controlled from the main control room.

Radiation monitoring of the CCSW system is provided to detect leakage of radioactive water from the containment cooling heat exchanger. This radiation monitor is located in the service water system at the discharge of the CCSW system. Radiation monitoring information is provided in Section 11.5.

9.2.2 Service Water System

9.2.2.1 Design Bases

The design objective of the service water system is to provide strained river water of suitable quantity and quality for plant equipment cooling requirements. To achieve this objective, the service water system was designed to:

A. Cool the RBCCW system under all operating conditions;

B. Meet cooling water requirements during the reactor shutdown mode, which represents the most severe condition and is used as the design basis;

C. Operate at a higher pressure than any of the loads which it serves; and

D. Provide an inexhaustible supply of water to the condenser hotwell, via the standby coolant supply valves, so that feedwater flow can be maintained to the reactor in the event of a LOCA. A description of the standby coolant supply system is provided in Section 9.2.8.

9.2.2.2 System Description

The service water system is shared by Units 2 and 3. The purpose of the system is to provide strained river water for various plant equipment. The components and systems which are cooled by or receive water from the service water system are listed in Table 9.2-2. The service water systems for Units 2 and 3 are shown in Drawings M-22 and M-355, respectively. As shown in these drawings, the system consists of five pumps, three strainers, and a common distribution header. Two service water pumps are provided per unit and the fifth shared pump is used as a backup. A chemical treatment system and the necessary control and support equipment are also provided.

Normally two pumps and one strainer for each unit are in operation while the fifth pump and one strainer are in standby. The system is cross-tied between Units 2 and 3. The pumps take their suction from a flooded pit in the common intake structure (crib house) for Units 2 and 3. The water supply from the river and/or

lake is described in detail in Section 2.4. The service water pumps from both units discharge into a common header serving both units. From this common header, service water is routed through the strainers to a common distribution header which routes the service water to the plant equipment listed in Table 9.2-2.

The system is a once-through flow network. Each heat exchanger discharges into one of two standpipes which connect to the plant discharge flume. The larger of the two standpipes is monitored for radioactive contamination.

In the event of a failure of the service water pumps to maintain header pressure, motor-operated isolation valves may be closed to isolate nonessential loads (TBCCW heat exchangers, main generator stator cooling water heat exchangers, and hydrogen coolers) from the service water system.

The service water pumps are mounted vertically and are driven by 1000-hp electric motors. Each has a capacity of approximately 15,000 gal/min at 91 psig. A Unit 2 LOCA signal trips the SW 2A pump. A Unit 3 LOCA signal trips the SW 3A pump.

The pumps are powered by 4-kV buses 23(33) and 24(34). The 2/3 service water pump can be powered by either Unit 2 or Unit 3. In the event of loss of power to any of these buses, the associated service water pumps will trip on undervoltage. Once power is restored to the bus, the pump can be manually restarted. There are no automatic start features for the service water pumps.

The service water strainers consist of wire mesh elements which undergo automatic self-cleaning when strainer differential pressure reaches a preset limit. The strainers can also be backwashed manually.

The chemical injection system is used to inject a biocide to control clam growth and remove slime-causing bacteria. The chemical injection system inhibits growth of algae which would reduce heat transfer capability of the heat exchanger tubes.

The service water system keeps the station fire protection system header pressurized when the fire water system is not in use. One or more of the following pumps will be used to maintain fire water header pressure if the service water pumps fail to do so. Manually operated Unit 1 screen wash pump, automatically operated Unit 1 diesel-driven fire pump or automatically operated Unit 2/3 diesel-driven fire pump.

If the diesel-driven fire pump fails, the service water system may be used to provide required fire water flow through a normally closed isolation valve. Refer to Section 9.5.1 for details of the fire protection system.

The service water system supplies water to the standby coolant supply system, which in turn provides an inexhaustible supply of water to the hotwell (under accident conditions). Standby coolant supply system information is provided in Section 9.2.8.

Service water or Fire Protection water provides a backup supply of cooling water to the CRD pumps in the event TBCCW is unavailable for normal cooling.

Under normal operating conditions, the service water system provides cooling water to the ECCS room coolers and pressurizes the CCSW pump keep fill line.

The CCSW system provides a safety related backup source of cooling water to the ECCS room coolers if service water is lost.

The HPCI and X-area coolers have been provided with four-way valves to allow reversing the service water flow through the coolers to keep the coolers and associated piping clean. This modification improved the efficiency of these room coolers.

The service water system is the primary source of cooling water for the RBCCW and TBCCW heat exchangers. Alternate cooling may be provided to the RBCCW and TBCCW heat exchangers and consequently the loads they cool. The fire header system or CCSW system can be used to provide cooling via temporary connections to the RBCCW and TBCCW heat exchangers to service equipment required to be operational.

Permanent tap connections on the TBCCW and RBCCW heat exchanger service water inlet lines facilitate connection of temporary cooling water sources during service water system outages. The taps are installed on the inlet lines for the 2B, 2/3 and 3B RBCCW heat exchangers and the 2A, 2B, 3A and 3B TBCCW heat exchangers. Each tap consists of a normally closed isolation valve and a flange. The RBCCW heat exchanger service water lines have 8-inch taps and the TBCCW heat exchanger inlets have 2½-inch taps. During normal heat exchanger operation, each flange is blind flanged and the corresponding isolation valve is closed.

9.2.2.3 Safety Evaluation

In the event of loss of auxiliary power on a unit, the standby service water pump will be manually started using power from the other unit. The standby pump and the two pumps on the remaining unit are sufficient to provide service water to both the tripped unit and the operating unit.

In the event of loss of auxiliary ac power to the 4kV safety-related electrical buses, the emergency diesel generators will automatically repower the safety-related electrical buses, and the operator will manually start a service water pump if the SW pumps were using underground piping on emergency power after securing a safe shutdown of the units. (A LOCA or failure of either of the diesels to start concurrent with the loss of auxiliary ac power will affect the speed with which the operator can put discretionary loads onto the emergency power system.) Service water is not necessary for a safe shutdown of the units, but it is needed to attain a normal cold shutdown condition.

In the event of a LOCA and a loss of all power where the service water system is unavailable, the HPCI and LPCI room coolers will receive backup cooling water from the CCSW system. Once the CCSW pumps are loaded on the emergency diesel generators and started, cooling water is diverted from the main system header to the service water header that feeds all three room coolers. This ensures that heat can be removed from the HPCI and LPCI rooms almost continuously.

The thermal response of the HPCI room has been evaluated for a postulated LOCA and LOOP assuming continuous system operation for an extended period of time (four hours). The HPCI room temperature setpoint at which the system would isolate was never reached, and the system would be always available.

9.2.2.4 Testing and Inspection Requirements

The station service water system is normally in operation, and no functional tests are required.

The service water effluent gross activity monitor is required to be operable; otherwise, a grab sample must be taken and analyzed every 12 hours. This radiation monitor is verified operable once per 24 hours.

9.2.2.5 System Instrumentation

The service water pumps can be controlled either from the main control room or locally at their associated breakers. Local control has been provided for safe shutdown concerns in the event the control room becomes inaccessible.

Pressure and temperature instrumentation is provided on selected heat exchangers cooled by the service water system.

Principal measurements such as service water header pressure, supply pressure, and pump motor current are indicated in the control room. Local pressure and temperature gauges are provided for flow balancing and equipment cooling control by manual valve adjustment. For that equipment which requires a controlled temperature, local automatic temperature controllers are provided to control service water flow through the equipment. As shown in Drawings M-22 and M-355, temperature control valves are included for the following:

A. The TBCCW heat exchanger common service water discharge;

B. Turbine oil coolers common service water discharge;

C. Main generator hydrogen coolers common service water discharge;

D. The RBCCW individual heat exchanger service water discharge; and

E. Each control room air conditioner condenser service water discharge.

Abnormal conditions, such as low service water pressure, a service water pump trip, or an X-area cooler trip are annunciated in the main control room. This provides the operator with information to assess the abnormal condition and initiate corrective actions.

Instrumentation is provided for the operation of the automatic backflush of the service water strainers on high differential pressure.

Radiation monitoring instrumentation is provided at the outlet of the service water system to monitor the radioactive discharges to the environment. Process radiation monitoring information is provided in Section 11.5.

9.2.3 Reactor Building Closed Cooling Water System

9.2.3.1 Design Bases

The performance objectives of the RBCCW system are to provide cooling for equipment and systems in the reactor buildings and to minimize the release of radioactive material from the reactor equipment to the service water. To achieve these objectives, the system has been designed with the following specifications:

A. Design flowrate (maximum) of 8800 gal/min per pump and

B. Design code for the heat exchangers - ASME Section VIII.

9.2.3.2 System Description

The RBCCW system is a closed loop system which consists of piping, pumps, heat exchangers, an expansion tank, a chemical feeder, and the necessary control and support equipment. Diagrams of the Unit 2 and 3 RBCCW systems are shown in Drawings M-20 and M-353, respectively.

The system's purpose is to provide cooling under various modes of operation and shutdown for the reactor auxiliaries listed in Table 9.2-3. The RBCCW system is available to cool all plant cooling loads in the reactor building under all operating conditions.

Five RBCCW pumps are provided: two for each unit and one extra, designated 2/3, which is shared by Units 2 and 3. The shared pump is located on the Unit 2 side. Each pump is a centrifugal pump with a capacity of 8800 gal/min at 80 psig (50% capacity) driven by a 300-hp electric motor. All three pumps on Unit 2 have a common suction header and a common discharge header; both pumps on Unit 3 also have a common suction header and common discharge header. From each unit's common discharge header, the cooling water is routed to the various plant equipment. The RBCCW systems for Units 2 and 3 are normally isolated from each other but may be crosstied at the suction and discharge of the 2/3 RBCCW pump.

The RBCCW system water returning from the loads is routed through the heat exchangers prior to returning to the pump suction. Similar to the RBCCW pump arrangement, five 50% capacity heat exchangers are provided: two for each unit and one extra, designated 2/3, which is shared between Units 2 and 3. The 2/3 heat exchanger is located on the Unit 2 side.

The RBCCW heat exchangers provide means for heat rejection from RBCCW to the service water (see Section 9.2.2). Each heat exchanger is a two pass, counterflow heat exchanger rated at 78×10^6 Btu/hr and is capable of removing the maximum heat load required.

An expansion tank is provided for each unit, connected to the suction line of the associated unit pumps. The expansion tank allows for water expansion from temperature and pressure changes within the RBCCW system. The tank is located above the highest point in the system and provides adequate net positive suction head (NPSH) to the RBCCW pumps. The tank also provides a storage volume of water that is replenished from the clean demineralized water tank and provides a means for RBCCW to overflow to the reactor building floor drain sump. Leakage into or out of the RBCCW system can be detected by low and high tank level alarms. The tank is provided with a vent pipe to prevent pressure buildup.

The RBCCW piping and loads are divided into three loops by isolation valves which can control the flow through the associated loop. Table 9.2-3 lists the loads associated with the individual loops.

A. Loop I - Primary containment critical loads. Loss of RBCCW to the primary containment may require immediate reactor shutdown.

B. Loop II - Shutdown cooling system heat exchangers.

C. Loop III - Reactor building auxiliary equipment.

Motor-operated valves are provided for loop isolation. The containment isolation valves do not receive containment isolation signals and, therefore, remain open unless closure is manually initiated. The motor-operated valve located on the outlet of the shutdown cooling heat exchangers may be throttled to control the temperature of the water injected to the reactor vessel by the shutdown cooling system.

The RBCCW pumps are not provided with a pump minimum flow valve. Prior to pump operation, a minimum flow path of 900 gal/min is required.

The RBCCW system is used to cool several reactor auxiliary systems and related equipment. With the exception of the Drywell Equipment Drain Sump Heat Exchanger, the Reactor Building Equipment Drain Tank Heat Exchanger, the Reactor Recirculation Pump Motor Oil Coolers, and the Reactor Recirculation Pump Seal Coolers, the RBCCW system operating pressure is lower than the processes served. The RBCCW system operating pressure is lower than the service water system operating pressure in the RBCCW heat exchangers. Except for the components listed above, any leakage will be into the RBCCW loop from both the equipment being cooled, and from the service water system. In all cases, however, the design prevents the accidental discharge of potentially radioactive water into the service water system, and thereby into the river.

A radiation monitor that records and alarms in the control room is located on the inlet piping to the heat exchangers to detect the leakage of any radioactive process water into the RBCCW system. Another method to evaluate leakage from equipment to the closed loop is via the grab sampling station located near the outlet of each major component of the cooling water system. A high-level alarm in the expansion tank would also detect leakage into the system.

RBCCW provides cooling under various modes of plant operation and shutdown. Pump and heat exchanger requirements for each mode of service will depend on plant conditions of operation or shutdown.

A chemical feeder provides a means for inhibiting rust development and controlling pH. Sodium nitrite ($NaNO_2$) is used to prevent rust development.

9.2.3.3 Safety Evaluation

The pumps and heat exchangers should be used in equal numbers. If two pumps and one heat exchanger are used, the design capacity of the heat exchanger is exceeded and causes excessive vibration. With two heat exchangers and one pump, the load limits for the pump can be exceeded as the pump approaches runout conditions.

Loss of RBCCW cooling flow to the primary containment loads (Loop I) requires immediate plant shutdown if flow cannot be reestablished within 1 or 2 minutes. Damage to electrical equipment and reactor recirculation pump seals and bearings may result.

Loss of RBCCW to the drywell coolers results in drywell atmosphere heatup and subsequent drywell pressure increase. The RBCCW pump logic provides for operation in all circumstances except when an accident signal is received concurrent with a loss of auxiliary power (see Section 8.3). The continuous supply of water to the drywell coolers will support the reduction of drywell pressure following a scram. The cooling water will also continue to be supplied to the

reactor recirculation pumps and motors during a pump coastdown (loss of RBCCW could possibly cause seizure of a recirculation pump).

Loss of RBCCW to the drywell equipment drain sump heat exchangers may result in an increase in airborne activity in the primary containment.

In the event of an ac power failure, one RBCCW pump (obtaining power from the emergency diesel) and one heat exchanger will provide cooling for the reactor recirculation pump, drywell sump, and the drywell coolers (these are considered to be critical loads). The RBCCW system (one pump) is considered in the diesel generator support of a nonaccident safe shutdown (refer to Section 8.3). The RBCCW is not required to perform any post-accident heat removal functions. In the case of a coincident loss of offsite power and a LOCA, power is not automatically provided to the RBCCW pump motors.

The heatup of the drywell during a postulated loss of coolant accident could, in turn, heatup the volume of liquid trapped between the RBCCW drywell return line containment isolation valves. Heatup of this trapped volume could overpressurize and fail the associated piping, creating a bypass path for the primary containment. To prevent the potential overpressurization of this piping, a relief valve has been installed between the containment isolation valves to protect against the consequences of thermal expansion of the trapped fluid. Another relief valve has been installed on the non-safety related loop to prevent the pressure buildup that could affect the inboard isolation valves.

9.2.3.4 Testing and Inspection Requirements

Since the system is operating at all times, no testing is required except for periodic operation of the spare pump.

9.2.3.5 System Instrumentation

Major components serviced by the cooling water system are provided with high-temperature alarms and/or temperature transmitters to aid in regulating cooling water flow.

As shown in Drawings M-20, M-22, M-353, and M-355, RBCCW temperature is automatically controlled by local temperature controllers which regulate service water flow through the RBCCW heat exchangers. The RBCCW pump discharge header temperature and pressure are indicated in the main control room. Low RBCCW system pressure is detected by a pressure switch which monitors the RBCCW pump discharge header pressure and annunciates in the main control room. High temperature at the RBCCW heat exchanger outlet is annunciated in the main control room. Local pressure and temperature gauges are provided for balancing flow through equipment by manual valve adjustment.

Instrumentation located in the main control room provides information allowing the operator to assess annunciated abnormal conditions and initiate corrective measures. Temperature of the cooling water to equipment located in the drywell is recorded in the control room along with cooling water outlet temperature from each drywell cooler. Low RBCCW flow from the reactor recirculation pumps is annunciated in the control room. Cooling water outlet temperatures from recirculation pump motor cooling coils and pump seals are recorded, and abnormally high temperature is annunciated.

Water level in each RBCCW expansion tank is indicated locally. Water level is sensed by tank-mounted level switches. One level switch controls the automatic level control valve for filling the expansion tank. A second level switch monitors abnormally high and low water level conditions and annunciates in the control

room. Leakage into and out of the RBCCW system, beyond the control capability of the level control valve, is detected by these high- and low-level switches.

Leakage from equipment into the RBCCW system can be detected by the radiation monitor located at the inlet to the RBCCW heat exchangers. This monitor records and alarms in the control room. The outlet of each major component of the RBCCW system is also provided with a grab sample station which can be used to locate the source of a leak.

9.2.4 Demineralized Water Makeup System

The demineralized water makeup system consists of all equipment required to transfer water from the well water storage tank, through the dual media filters and the portable demineralizer, and into the various water storage tanks onsite.

9.2.4.1 Design Bases

The design objective of the demineralized water makeup system is to provide the desired quantity of reactor quality water for pre-operation and normal operation of the power plant. The system provides makeup water to the clean demineralized water storage tanks and the contaminated condensate storage tanks.

9.2.4.2 System Description

The demineralized water makeup system is common to both Units 2 and 3 and consists of pumps, storage tanks, demineralizers, and the necessary control and support equipment. The system pumps are of adequate size to provide the maximum expected flowrates. Drawings M-35, Sheet 1 and the M-423 series depict the demineralized water makeup system.

The system takes well water from the existing 200,000-gallon Unit 1 well water storage tank. This tank is filled from two deep wells. Well water is pumped from the well water storage tank, by any of the three well water transfer pumps, to the makeup demineralizer system. As shown in the M-423 series, the makeup demineralizer system consists of three 33 1/3% capacity dual media filters whose combined effluent is routed to a portable demineralizer. The portable demineralizer effluent is routed through a direct acting pressure control valve to the 200,000-gallon clean demineralized water storage tank. Water may also be routed to any of the contaminated condensate storage tanks.

As shown in Drawing M-423, Sheets 6 through 8, the demineralizer system is provided with necessary storage tanks and pumps to regenerate the demineralizer resin beds in place.

The demineralized water makeup system operates on demand at infrequent intervals to replenish demineralized water in the storage tanks.

9.2.4.3 Safety Evaluation

The demineralized water makeup system has been evaluated by the NRC under Systematic Evaluation Program (SEP) Topic VI-10.B, Shared Systems. This evaluation determined that this system is not required to function for any safety-related purpose.

The heatup of the drywell during a postulated loss of coolant accident could, in turn, heatup the volume of water trapped between the demineralized water drywell supply line containment isolation valves. Heatup of this trapped volume could overpressurize and fail the associated piping, creating a bypass path for the primary containment. To prevent the potential overpressurization of this piping, a relief valve has been installed between the containment isolation valve to protect against the consequences of thermal expansion of the trapped fluid.

9.2.4.4 Testing and Inspection Requirements

The makeup water system operates intermittently during operation of the plant and no testing of the system is required.

9.2.4.5 System Instrumentation

As shown in Drawings M-35, Sheet 1, and the M-423 series, the clean demineralized water storage tank levels are indicated locally. Pumps and valves for the demineralized water makeup system are controlled from the makeup demineralizer building and the portable demineralizer. The demineralized water from the portable demineralizer is tested for conductivity.

9.2.5 Ultimate Heat Sink

9.2.5.1 Design Bases

The design objective of the ultimate heat sink is to provide sufficient cooling water to the station to permit operation of the CCSW system when the normal heat sink (the river) is unavailable. The CCSW system is described in Section 9.2.1.

In addition to supplying the CCSW system, the ultimate heat sink may also be used for cooling other plant equipment if desired.

9.2.5.2 System Description

The ultimate heat sink consists of water sources for the cooling water systems, necessary retaining structures, and the canals connecting the water sources with the intake structures.

The cooling water from the lake and/or the Des Plaines and Kankakee rivers, provides the normal heat sink for the disposal of unusable energy inherent in the thermodynamic cycle of the two operating Dresden reactor and turbine-generator plants. The cooling water also provides the principal means for removal of fission product decay heat following a unit shutdown. Aside from the main condenser circulating water system, four systems depend on the use of the cooling water. These systems are the LPCI system in the containment cooling mode, the service water system, the fire protection system, and the diesel generator cooling water system.

The Kankakee River is the normal source of emergency cooling water for Dresden Station. In the event of a loss of this water source, there is a limited supply of water trapped, by design, in the intake and discharge canals.

Due to the topography of the circulating water canals and piping, approximately 9 million gallons of river water are trapped within the canals, not including water in the cooling lake (see Figure 2.4-1). This is due to the topographical high points in both the intake and discharge flumes. As the Dresden pool level falls, backflow from the discharge flumes stops at elevation 498'-0", and backflow from the intake flumes stops at elevation 494.2 feet. However, the water level in the discharge canal will equalize in level with the intake canal at an elevation of 494.2 feet. This will occur from any of the following reasons: 1) leakage through the flow diverter gates in the discharge canal, 2) backflow through circulating water system piping, and 3) opening of deicing line.

Lake water and/or this impounded river water would be used as a heat sink for the long-term removal of decay heat from the reactors. The flow control station, cooling lake spillway gates, lift pumps, etc. would be adjusted to prevent loss of lake water to the river.

Additional cooling canal and lake information is provided in Section 2.4.

9.2.5.3 <u>Safety Evaluation</u>

The original FSAR before amendments did not address how Dresden Station would achieve safe shutdown or mitigate the consequences of an accident following a dam failure. The following Sections 9.2.5.3.1 and 9.2.5.3.2 are based on coping scenarios which were sent to the NRC on February 28, 1969 in response to Question I.F in Amendment 9/10 of Dresden's original FSAR. Question I.F reads:

> "With regard to A.10., above, we understand that failure of the Dresden Island Lock and Dam could reduce the Kanakakee River to a level below the intake of the main cooling water canal. If this failure were to occur as a result of an earthquake which disabled all Class II systems, it appears that the availability of an ultimate heat sink for all units would be uncertain. On this basis a complete safety evaluation of the consequences of such a failure should be provided for our review. The evaluation should include elevation drawings showing the dam, the intake structures, and the levels at which service water pumps would lose suction.
>
> 1. Provide an evaluation of the seismic design of the lock and dam indicating whether the structure is considered Class I.
> 2. Provide an evaluation of the ability of Units 2 and 3 to cope with the effects of an earthquake during normal operation which causes coincident failures of the dam, all Class II systems, and offsite site electrical power. How would the consequences be affected by the availability of offsite power?

3. Repeat the evaluations assuming in addition that the earthquake results in the design basis loss-of-coolant accident in one of the two units."

The conditions in Question 3 for a dam failure coincident with a LOCA are beyond the design basis of Dresden Units 2 & 3. Therefore, calculations providing design basis requirements are satisfied are not required to support assumptions and conclusions in Section 9.2.5.3.2. However, changes to plant procedures and/or structures, systems, and components must be evaluated against information in both of the following sections through the provisions of 10CFR50.59 to ensure that no unreviewed safety questions result.

9.2.5.3.1 Dam Failure During Normal Plant Operation

If a catastrophic failure of the Dresden Lock and Dam occurred, with no seismic activity both Units 2 and 3 could be safely shut down. The lake provides a large reservoir of cooling water. In addition, even if the lake were unavailable, a substantial amount of cooling water would be trapped in the intake and discharge head works (as described later in this section) and could be recirculated using the deicing line. The Class I sections of Class II systems were considered only in bringing the plant to a safe condition. It is immaterial whether the shutdown is carried out on normal auxiliary power or on the diesel generators.

As an example, assume the case with both Dresden Units 2 and 3 in operation with normal outflow of electrical power and normal alignment of the auxiliary power network to each unit. At this point complete failure of the dam takes place with a rapid decrease in the pool level. The first indication of trouble would be a drop in power requirements of the circulating water pumps and service water pumps. Vacuum on each unit condenser would decrease and the reactors would scram on condenser low vacuum. With the loss of the main heat sink, reactor pressure would increase and the isolation condenser on each unit would actuate.

During this sequence of events, the operator is required to trip the circulating and service water pumps to prevent pump damage. Other equipment, which either adds heat to the primary system or which is cooled directly or indirectly by river water, would be removed from service.

Following the reactor scram on Units 2 and 3, the relief valves from the primary system to the suppression chamber would open to prevent overpressurizing the reactor vessel. Level in the reactor would be maintained by reactor feed pumps; control rod drive pumps; or, in the case of loss of auxiliary power, the HPCI system. With the initiation of the isolation condenser, depressurization of the primary system would start.

Each of the reactors could be depressurized at a controlled rate using the isolation condensers. The primary system temperature could be reduced to 212°F in 8 - 12 hours and be maintained at this condition with the isolation condensers. Generally, the temperature could not be reduced below this point since the system depends on steam flow to remove the core decay heat.

Preferred makeup water to the isolation condenser is from the clean demineralized water storage tank via two diesel driven makeup pumps or the clean demineralized water pumps, the latter of which are discussed in Section 9.2.6. With this water source unavailable, river water would be pumped to the isolation condensers by diesel-driven fire pumps, through a crosstie with the 2/3 DGCW line, or by portable engine-driven pumps pumping into the fire system. Contaminated condensate water is also available to provide makeup water to the isolation condensers; however, the use of this water source is less desirable than the other sources. It is also possible to obtain portable pumps which would draw suction from the intake canal and discharge to the fire protection system.

The fire protection system is considered a Class II system; however, parts of this system can meet the requirements of a Class I system. Using existing valves, it is possible to sectionalize the system to isolate the failed parts.

Operation of the diesel generators is assured since the diesel generator cooling water pumps' suction lines are at elevation 487'-8". The diesel fire pump of Units 2 and 3 has its suction above elevation 494.2 feet. Therefore, the compartment that contains the suction of the fire pump and the CCSW intakes must be reflooded using the travelling screen refuse pumps.

Reference should be made to Drawing M-10. The diesel fire pump and suction lines for the CCSW pumps take suction from a compartment between column row B and C with its center line on column line 4. River water enters this compartment through two screened openings to the left and right of column line 4. The floor of this compartment is at elevation 493'-8" and the ceiling is at elevation 509'-6". The two openings extend between these two elevations. The wire mesh screens from each of these openings would be lifted out of place and replaced with stop logs. The stop logs are stored in the crib house at a location which would minimize the time required for their installation by station operators should a dam failure occur.

Dewatering valves, located at elevation 480'-0", would be opened to permit river water to flow from the compartments under the circulating water pumps and intake piping to the trash rake refuse pit located between column row C and D and column line 7 and 8. The floor of this pit is at elevation 477'-0". Thus, the water in this pit would rise to elevation 494.2 feet, the level in the intake canal.

Two refuse pumps take suction from this pit. The pumps are located in a compartment adjacent to the pit with their suction at elevation 479'-0". Each pump has a discharge capacity of 2400 gal/min. A permanent pipe line is installed between the discharge line of these pumps and the compartment with the CCSW pumps. By proper electrical switching, the refuse pumps can be operated off the diesel generator.

The capability of the ultimate heat sink (UHS) was reassessed in 2001 prior to extended power uprate (EPU). Only the part of the intake canal trapped inventory above the suction of the diesel generator cooling water pumps (2 million gallons) is considered available from the UHS, as the makeup path potentially relies on onsite power. This inventory lasts approximately four days following EPU. If the offsite power is restored within the 4-day period, the time would increase because the entire volume of the intake canal would be available with no diesel generator cooling required.

An additional small amount of cooling water for diesel generator cooling is also required but could be recirculated to the intake structure after dissipating its heat to the environment. Loss of impounded river water, due to evaporation, could be made up by a portable, low-head, high- volume, engine-driven pump. Dresden Station can obtain several suitable pumps from many sources in the Northern Illinois area, such as other company-owned facilities and pump rental companies.

9.2.5.3.2 Dam Failure Coincident with a Large Break LOCA

A dam failure coincident with a loss of coolant accident was not considered in the original plant design. Although not required for licensing, a coping study was completed which evaluated a dam failure coincident with a loss of coolant accident. The coping study was not used in the original licensing of the plant. Additionally the systematic evaluation program and the safety evaluation for the full term operating license did not recommend that this event be considered. Based on a probabilistic safety assessment, a loss of coolant accident and seismic event are not postulated to occur concurrently. The probability of occurrence of an earthquake coincident with a Large Break LOCA was below the criteria specified in the standard review plan. The frequency of occurrence for Units 2 and 3 is 1.6E-07.

9.2.5.3.3 Seismically Induced Dresden Dam Failure Coupled With a Small Break LOCA

As a part of the Individual Plant External Event Evaluation, a method was developed to safely shut down following a small break loss of coolant accident (LOCA) concurrent with a loss of dam. Consistent with the guidance in EPRI NP-6041-SL "A Methodology for Assessment of Nuclear Power Plant Seismic Margin (Revision 1)", Dresden Station has the capability to safely shutdown following a seismic event that fails the Dresden Dam and causes a 1-inch break LOCA in each unit. The Isolation Condenser system and available Emergency Core Cooling Systems (ECCS), such as the HPCI and LPCI systems, are sufficient to mitigate the small break LOCA for a period of 24 hours. Cooling water to the CCSW heat exchangers is required 24 hours after the event to remove heat from the suppression pool.

If the isolation condenser shell-side makeup sources of water identified in Section 9.2.5.3.1 are inoperable due to the seismic event, the Unit 2 DGCW system can supply makeup water through hose connections on the Unit 2 DGCW discharge piping. Hose connections on both the A and B isolation condenser makeup pump suction piping allow the Unit 2 DGCW system to supply water through two, 4-inch hoses to either pump. Both the DGCW and isolation condenser makeup systems are seismically qualified and will be available following a seismic event. The hoses needed to connect the two systems are stored in an area that can withstand the seismic event. A procedure directs plant workers and operators on how to set up the hoses and operate the systems following the seismic event.

Because complete failure of the Dresden Lock & Dam will cause the intake canal level to lower below the CCSW intake pipes, cooling water to the CCSW heat exchangers will be supplied from the intake canal by portable submersible pumps. These submersible pumps, one for Unit 2 and one for Unit 3, are hydraulically driven by a diesel powerpack. The pumps, diesel powerpacks, and sufficient lengths of 10-inch diameter hose to connect the submersible pumps to the hose connections on the CCSW piping, which are located on the second floor of the turbine buildings, are also stored in an area that can withstand the seismic event. A procedure directs plant workers and operators on how to set up and operate the equipment following the seismic event. These actions and equipment provide the capability to mitigate the effects of a seismic event that fails the Dresden Lock & Dam and results in a 1-inch break LOCA in each unit for the 72-hour time frame given in EPRI NP-6041-SL.

9.2.5.4 Testing and Inspection Requirements

A surveillance is performed every third refueling outage to verify the ability of the system to accomplish its intended purpose of providing a source of water for the shells of the isolation

condensers of both units and also verify that the required manual operations described in Section 9.2.5.3.1 can be performed before onsite sources of shell-side makeup for the isolation condensers are depleted.

9.2.6 Condensate Storage Facilities

The condensate storage facilities include all equipment necessary to store potentially contaminated demineralized water and condensate and to transfer water from the clean and contaminated water storage tanks throughout the plant for various uses.

9.2.6.1 Design Bases

The design objective of the condensate storage facilities is to provide water of a quality and quantity required for preoperation and operation of the power plant.

The system is designed to ensure a minimum of 90,000 gallons of water is available from each contaminated condensate storage tank (CCST) for use by HPCI.

9.2.6.2 System Description

As shown in Drawings M-35, Sheet 1 and M-366, the condensate storage facilities consist of two 250,000-gallon capacity CCSTs (CCST 2/3A and 2/3B), one 200,000-gallon capacity contaminated demineralized water storage tank (T-105A), and two clean demineralized water pumps shared between Units 2 and 3. Two of the tanks, CCST 2/3A and 2/3B, are each normally maintained at levels which make 90,000 gallons available to each HPCI system that is required to be operable. 90,000 gallons satisfies the HPCI system makeup water assumptions in an NRC Systematic Evaluation Program analysis regarding safe shutdown using only Class I systems. A minimum combined water volume in CCST 1, CCST 2/3A, and CCST 2/3 B of 130,000 gallons (single plant operation) and 260,000 gallons (dual plant operation) is required to meet Appendix R safe shutdown requirements. Refer to Section 6.3 for a discussion of the HPCI system. Each unit has two condensate makeup pumps, two condensate transfer pumps, and one condensate transfer jockey pump with associated piping and valving to transfer condensate throughout the plant.

Two clean demineralized water pumps, each rated at 250 gal/min, take suction from the clean demineralized water storage tank and discharge into a common header. The clean demineralized water transfer system provides water for multiple uses including the following:

A. Decontamination;

B. Floor washdown in areas containing radioactive drain systems;

C. Laboratories;

D. Filling of cooling water systems;

E. Purposes requiring demineralized water where radioactive contamination is not desired;

F. Makeup water to the isolation condenser.

All three of the contaminated water tanks are capable of being crosstied and the two CCSTs are normally connected. Both CCST 2/3A and CCST 2/3B may supply water to HPCI subsystems for emergency core cooling systems (ECCS) for Unit 2 and Unit 3. The safety related water supply for HPCI is the suppression pool.

The two CCSTs provide a source of condensate for use by the following systems of both units:

A. Main condenser hotwell,

B. Control rod drive hydraulic system,

C. HPCI,

D. Core spray, and

E. LPCI.

The condensate makeup pumps take suction from the CCSTs and provide the driving force to transport makeup water to the main condenser hotwell whenever there is no condenser vacuum.

The two redundant condensate transfer pumps, each rated at 500 gal/min, and the 70-gal/min condensate jockey pump take suction from the CCSTs and provide the motive force to supply the following systems with condensate:

A. Isolation condenser alternate makeup,

B. Fuel pool cooling and cleanup,

C. Reactor water cleanup,

D. Radwaste, and

E. ECCS fill header alternate supply.

9.2.6.3 Safety Evaluation

The entire clean demineralized water system piping arrangement to the isolation condenser utilizes normally open, manually operated valves, with the exception of the final power-operated isolation valve at the isolation condenser shell side inlet. Power to this valve is supplied from 250-Vdc motor control center (MCC) 2A (Unit 2), or 250-Vdc MCC 3A (Unit 3). The isolation valve is also accessible for manual operation in case its power supply or motor malfunctions.

The clean demineralized water system includes two pumps, neither of which are on buses supplied by emergency power (they are both on MCC 25-2). However, emergency power can be aligned to MCC 25-2 by the plant operators, diesel load permitting.

Both condensate transfer pumps are provided power from buses which are fed from the emergency diesel generators upon loss of ac power. The pumps can be locally controlled in the event the control room and the auxiliary electric equipment room are inaccessible. All valves in the supply system to the isolation condenser are manually operated, with the exception of the final isolation valve, which is supplied from 250-Vdc power. This valve is accessible should its motor or power supply fail, thus requiring manual operation.

Sufficient pure water is stored onsite to perform a plant cooldown in a reasonable amount of time in accordance with Branch Technical Position RSB 5-1.

The potential for the ECCS condensate supply lines to freeze has been evaluated. The majority of the CCST piping to ECCS is buried 5 to 6 feet below surface grade. To prevent freezing, the ECCS lines above ground are well insulated and heat traced. The valves on these lines are contained in insulated permanent enclosures. All other safety-related process instrument and sampling lines are indoors and not exposed to subfreezing temperatures.

9.2.6.4 Testing and Inspection Requirements

Water quality in the clean and contaminated condensate storage tanks is periodically analyzed in accordance with station chemistry procedures.

9.2.6.5 System Instrumentation

As shown in Drawing M-35, Sheet 1, each CCST level is indicated in the control room and low-level alarms alert the operator to excessive use of condensate or when normal makeup is required. High-level alarms are provided for each tank to indicate the filled condition. Each storage tank is electrically heated and thermostatically controlled locally.

The condensate makeup pumps, condensate transfer pumps, and the condensate transfer jockey pumps are remotely operated from the control room. Each is provided with circuitry to annunciate a tripped condition in the control room. The condensate makeup pumps will automatically start due to a low hotwell level.

Condensate transfer pump discharge header pressure and demineralized water pump discharge header pressure are indicated in the control room. Low condensate transfer header pressure and low demineralized water header pressure signals actuate alarms in the control room.

9.2.7 Turbine Building Closed Cooling Water System

9.2.7.1 Design Bases

The purpose of the TBCCW system is to provide a means of heat rejection from systems located in the turbine building and crib house.

9.2.7.2 System Description

The TBCCW is a closed loop system which consists of pumps, heat exchangers, an expansion tank, a chemical feeder, and associated control and support equipment. Separate, independent systems are provided for Units 2 and 3. The TBCCW system is shown in Drawings M-21 and M-354, Sheets 1 and 2.

The TBCCW system consists of two pumps which circulate the cooling water throughout the unit. An expansion tank piped to the TBCCW pump suction line is located on the turbine building ventilation fan floor (elevation 549'-0"). Its elevation above the TBCCW pumps ensures adequate NPSH for the pumps. It also provides a surge volume for the system as the cooling water density varies. Expansion tank level is maintained by an automatic level control valve which supplies demineralized water to the tank as level decreases. An internal overflow for the tank is routed to the 48-inch service water discharge header.

The TBCCW pumps discharge to a common header which supplies cooling water to the various equipment listed in Table 9.2-4.

The electrohydraulic control (EHC) fluid coolers have a temperature control valve on the common cooling water outlet of the heat exchangers which automatically controls the cooling water flow in response to EHC fluid effluent temperature.

The sparging air compressor aftercoolers, the Unit 2 service air compressor, and reactor feedwater pump oil coolers have temperature control valves on the cooling water inlet lines which control the cooling water flow through these components, maintaining proper temperatures. The instrument air compressors and the Unit 3 service air compressor have a solenoid valve which opens automatically when the compressor is started. The cooling water flow is adjusted with a manual valve on the cooling water outlet line, for component temperature control.

Other loads have no automatic temperature control, but flow through them may be manually throttled.

Return flow from the various loads enters a common header which is routed to one of two TBCCW heat exchangers. The cooling water flows through the shell side of the selected heat exchanger and the effluent is routed to the TBCCW pump suction. Service water provides the cooling medium on the tube side of the heat exchanger. An air-operated temperature control valve, common to both heat exchangers, throttles the service water in response to the temperature of the cooling water effluent. Refer to Section 9.2.2 for a description of the service water system. A chemical feeder at the suction to the pumps provides a mechanism to add a corrosion inhibitor to the system.

9.2.7.3 Safety Evaluation

The equipment cooled by the TBCCW system is not considered essential. It is important to note that loss of TBCCW cooling to the control rod drive pumps does not affect the function of the control rod scram function. Service water or Fire Protection water backs up the TBCCW supply to the CRD pumps depending on the use of Service water, underground or above ground header respectively.

9.2.7.4 Testing and Inspection Requirements

The TBCCW system operates continually and requires no operability checks. The cooling water is sampled periodically in accordance with station chemistry procedures. A corrosion-inhibitor is added via the chemical feeder when required, as determined by the sample analysis.

9.2.7.5 System Instrumentation

As shown in Drawings M-21 and M-354, Sheets 1 and 2, a level switch is provided to automatically open the demineralized water makeup valve to fill the TBCCW expansion tank. A second level switch actuates a common high or low expansion tank level annunciator in the control room.

The TBCCW pumps are remotely controlled from the main control room. Discharge header pressure is indicated and low header pressure is annunciated in the control room. Discharge header temperature is also indicated and high temperature is annunciated in the control room.

The TBCCW pumps are powered by normal ac power supplies (480-V MCCs) and are protected by thermal overload trips. A trip of a TBCCW pump annunciates in the control room. There are no automatic start features for the pumps; thus, if only one TBCCW pump is operating, the idle TBCCW pump will not automatically start if the originally operating pump trips.

Local temperature and pressure indicators are provided throughout the system to allow for flow balancing and determination of individual heat exchanger performance.

9.2.8 Standby Coolant Supply System

9.2.8.1 Design Bases

The purpose of the standby coolant supply system is to provide an inexhaustible supply of water to the condenser hotwell so that feedwater flow to the reactor can be maintained in the event it is needed for core flooding and/or containment flooding following a postulated LOCA.

9.2.8.2 System Description

The system consists of piping between the service water system and the condenser hotwell (see Drawings M-22 and M-355), as well as associated valves, and instrumentation. The service water system is described in detail in Section 9.2.2. This equipment supplies approximately 15,000 gal/min of screened and strained river water to the hotwell. Two motor-operated isolation valves are used in the interconnected piping to provide the capability for testing the valves and to prevent leakage of river water to the condenser.

A hydrostop flange has been added upstream of standby coolant supply isolation valve, MO-2-3901, and MO-3-3901, to permit valve maintenance. The hydrostop plug that remains in the system is in the tee out of the main flow stream. The line water pressure acts on the plug to force it into the tee. This ensures the plug will not enter the main flow stream.

The standby coolant supply system is actuated manually from the control room. The operator remotely opens the two isolation valves in the line between the service water system and the condenser hotwell. The control room switches are covered with plastic sleeves to prevent inadvertent actuation. Water can then be pumped from the hotwell with the condensate booster pumps (and feedwater pumps if necessary) to the reactor vessel for core flooding at reactor vessel pressures up to 200 psig. The containment vessel can be flooded up to the top of the core by continued operation of the pumps, if desired. Gases in the drywell would be vented through the standby gas treatment system to the chimney by remote manual operation from the control room.

The functioning of this system is dependent upon the availability of auxiliary power and is not considered in the sizing of the diesel generator.

9.2.8.3 Safety Evaluation

To maintain core cooling capability during a LOCA, it is necessary to supply large amounts of water to the reactor vessel. This can be accomplished by the emergency core cooling systems or the feedwater system, depending upon the type and conditions of the accident. Additional information on the details and the consequences of a LOCA can be found in Sections 6.2.1, 6.3, and 15.6.

Flooding of the entire containment vessel to a water level above the reactor core is an ultimate assurance that core cooling can be maintained. In addition, containment flooding provides the means to achieve post-accident recovery under any conditions that might prevent filling of the reactor vessel.

The main condenser hotwell normally contains a volume of water equivalent to the volume that can be pumped by the condensate/feedwater system in approximately 3 minutes of operation at 100% rated power (2957MWt) feedwater flowrates. The contaminated condensate storage tanks provide an additional large volume of water (see Section 9.2.6). The hotwell and condensate storage tank volumes, in conjunction with the condensate/feedwater system, can be used for containment flooding.

The standby coolant supply connection to the hotwell provides the capability for continuous flow through the condensate/feedwater system for rapid containment flooding and serves as a backup to other core cooling systems. With the hotwell being supplied with makeup water from the service water system, the condensate/feedwater system provides an additional coolant supply in excess of those required to satisfy the reactor vessel reflooding and containment flooding design objectives.

9.2.8.4 Testing and Inspection Requirements

Most of the components supporting the standby coolant supply system are normally and continually in operation, thereby eliminating the need for periodic testing. These include the service water system, the reactor feedwater system, and the hotwell condensate system.

The valves in the system are motor-operated and can be exercised periodically and their capability demonstrated.

9.2.8.5 System Instrumentation

The two standby coolant supply motor-operated valves are individually controlled from the main control room. Each valve is monitored for valve position and an alarm is actuated in the control room if either valve is not fully closed.

Table 9.2-1

CONTAINMENT COOLING SERVICE WATER EQUIPMENT SPECIFICATIONS

Containment Cooling Service Water Pumps

Number	4 (2 needed to provide required cooling capacity)
Type	Horizontal, centrifugal
Power source	Auxiliary transformer or emergency diesel
Capacity	3,500 gal/min each
Head (approximately)	470 feet

Table 9.2-2

SERVICE WATER SYSTEM LOADS

A. RBCCW heat exchangers (3)

B. TBCCW heat exchangers (2)

C. Traveling screen wash spray

D. Fire protection system

E. Turbine oil coolers (2)

F. Deleted

G. Generator hydrogen coolers (4)

H. Generator stator water coolers (2)

I. Standby coolant supply

J. Control room air conditioning condensers (2)

K. Auxiliary electric equipment room air conditioning condenser (1)

L. Maximum recycle concentrator condensers (2)

M. Off-gas glycol chillers (2)

N. X-area coolers (steam tunnel coolers) (8)

O. CCSW keep fill

Table 9.2-2

SERVICE WATER SYSTEM LOADS

P. HPCI room coolers

Q. LPCI room coolers

R. Off-gas filter building manifold sample system heat exchanger

S. Radwaste max recycle condensate holding tanks (2)

T. Chlorinator pumps (2)

U. Control rod drive pump coolers (alternate cooling is provided by SW or FP system. TBCCW is normal source)

V. Integrated leak rate test air compressor (Unit 3 only)

Table 9.2-3

MAJOR RBCCW SYSTEM LOADS

LOAD
LOOP I
A. Drywell coolers
B. Recirculation pump seals and pump motor oil coolers
C. Drywell equipment drain sump heat exchanger
LOOP II
A. Shutdown cooling heat exchangers
LOOP III
A. Reactor building auxiliary equipment
1. Shutdown cooling pump seal coolers
2. Fuel pool heat exchangers
3. Clean-up system non-regenerative heat exchangers
4. RWCU main pumps lube oil coolers
5. RWCU auxiliary pump seal cooler
6. RWCU precoat loop cooler
7. Pump back air compressor and aftercooler
8. Reactor building equipment drain tank heat exchanger

Table 9.2-4

TBCCW SYSTEM LOADS

A. Control rod drive pumps
B. Reactor feed pump oil coolers and pump seals
C. Plant instrument and service air compressors
D. Circulating water pumps^(1)^
E. Bus duct coolers
F. Main generator exiter air cooler
G. Condensate pumps
H. Condensate booster pumps
I. EHC fluid coolers
J. Radwaste air sparger aftercooler (Unit 2 only)
K. Sample coolers
L. Resin transfer air compressor
M. Decontamination facility air conditioner (Unit 2 only)

Note:

1. Unit 2 TBCCW system normally supplies both Unit 2 and 3 circulating water pumps. Through available valving, the Unit 3 TBCCW system can supply both Unit 2 and 3 circulating water pumps instead.

9.3 PROCESS AUXILIARIES

This section includes descriptions of the following systems:

A. Compressed air systems (9.3.1);

B. Process sampling systems (9.3.2);

C. Equipment and floor drainage systems (9.3.3); and

D. Standby liquid control system (9.3.5).

9.3.1 Compressed Air Systems

The primary compressed air systems include the instrument air system (oil-free) and the service air system. Each of the units (2 and 3) has its own independent instrument and service air systems which, if needed, can be cross-connected with each other for reliability. Unit 1 has an independent system that is not reliable enough for full time service. The Unit 2 service air system will normally supply the Unit 1 service air system when the Unit 1 service air compressor is not operating.

Other compressed air systems include the pump-back and drywell pneumatic supply systems. The pump-back system and drywell pneumatic supply system supply primary containment gases (instead of air) to the drywell. The radwaste area has its own air sparging system.

9.3.1.1 Design Bases

The design objective of the compressed air systems is to ensure the availability of air (of suitable quality and pressure) for power plant operation. To achieve this objective, the compressed air systems are designed in accordance with the following specifications:

A. Design temperature of 105°F,

B. Design operating pressure of 110 psig, and

C. Design capacity of 500 ft^3/min per system.

9.3.1.2 Instrument Air System

9.3.1.2.1 System Description

The purpose of the instrument air system is to supply clean, dry, compressed air for air-operated control devices and instruments.

All major instrument air system components except those associated with the 3C compressor train are located at elevation 517'-6" in the turbine building. The 3C compressor train is located on the Unit 3 538'-0" elevation. Unit 2 has two compressor trains, 2A and 2B; Unit 3 has three, 3A, 3B, and 3C. Each train consists of a compressor, an aftercooler, air receiver tanks, dryers, prefilters, afterfilters and the necessary control and support equipment.

The instrument air system supplies air to:

A. Turbine building loads,

B. Reactor building loads,

C. Radwaste building loads,

D. Crib house loads, and

E. Off-gas filter building loads.

The instrument air compressors are two-stage, water cooled, rotary screw compressors, which deliver oil-free, pulsation-free air. The 2B, 3B, and 3C are 100 Hp electric motor driven, the 3A is 125 Hp electric motor driven, and the 2A is 201 Hp electric motor driven. The 2B, 3A, 3B, and 3C compressors are rated at 460 CFM at 100 psig discharge pressure. The 2A compressor is rated at 843 CFM at 100 psig discharge pressure. These compressors will operate on the compressor's pressure switch to load and unload. The compressors have no interlock with the Unit 2(3) instrument air main receiver pressure switch.

Cooling for all compressors is provided by the turbine building closed cooling water (TBCCW) system.

The instrument air supply system is operable at all times during plant operations. With the compressor in NORMAL mode and the control switch set to close, the following sequence occurs:

A. The instrument air compressor starts and runs unloaded (magnetic unloader deenergized);

B. If after a predetermined time the pressure is less than a predetermined setpoint, the magnetic unloader energizes and loads the compressor;

C. The instrument air compressor runs loaded until the proper pressure is established. At that time the magnetic unloader deenergizes and the compressor unloads; and

D. The instrument air compressor continues to load and unload as the pressure commands until it auto trips, the control switch is placed to TRIP, or the breaker opens.

Instrument air receivers are used as buffers between compressor discharge and the rest of the system. In this configuration, they dampen pressure pulses from the compressor and provide a smooth flow of air to the system. They also provide storage capacity to accommodate intervals when demand exceeds the capacity of the compressors.

Relief valves are set as follows:

A. 2A, 2B, 3B, 3C Compressor discharge - 135 psig,

B. 3A Compressor discharge – 120 psig

C. Local air receivers - 135 psig, and

D. Main air receivers - 125 psig.

The service air crosstie provides instrument air backup from the service air header. It operates automatically when instrument air pressure falls below 85 psig. Once operated, it must be manually reset when the air pressure is again greater than 85 psig. A control room alarm is provided to indicate low instrument air pressure.

9.3.1.2.2 Safety Evaluation

The postulated worst-case scenario for instrument air failure was identified as an event that would cause a total loss of instrument air as a result of a nonsafety-related system failure. Total loss of instrument air would cause a forced power reduction or plant shutdown, but all safety-related devices requiring instrument air would perform as designed.

Safety-related equipment design considers properly sized air accumulators as an effective means to supply residual air, pressurized nitrogen as an alternate motive force, and selection of appropriate component fail-safe positions to demonstrate acceptable consequences of the loss of air. The instrument air system provides high-quality air (i.e., free of moisture, particulates, and oil), thereby minimizing the potential for safety-related valve failure resulting from a lack of air due to blocked air supply lines. The station's air quality monitoring and preventative maintenance programs ensure that consistently high-quality instrument air is supplied.

In addition to these design features and administrative controls, the Unit 2 and 3 instrument air systems can be cross-connected during those rare occasions when one air compressor becomes unavailable and there is an unusually high and sudden demand on the instrument air system. Under normal circumstances one unit's air system is isolated from the other unit's with the exception of the 3C compressor, which may be aligned with either unit's system for additional support. Typically, the 3C compressor is aligned to Unit 2 to support the additional house loads on that system. This dedicated configuration isolates failure events and prevents them from affecting another dedicated system.

9.3.1.2.3 Testing and Inspection Requirements

A periodic survey is performed to test the service-air-to-instrument-air backup valve.

Periodic air quality monitoring is performed to ensure that high-quality instrument air is supplied to the plant.

9.3.1.3 Service Air System

9.3.1.3.1 System Description

The purpose of the service air system is to:

A. Provide air to certain process equipment and breathing air manifolds;

B. Provide air for mixing and agitating functions at reduced pressure;

C. Provide an emergency backup supply of air to the plant instrument air system; and

D. Provide pneumatic pressure to control the operation of the service air compressors.

E. Provide operating air to Unit 1 air systems when the Unit 1 service air compressor is not operating.

The major service air system components are located at elevation 517'-6" in the turbine building. Units 2 and 3 each have one compressor train. Each train consists of a compressor, an aftercooler, a moisture separator, an air receiver tank, and necessary control and support equipment (see Figure 9.3-5).

The service air system supplies air to:

A. Turbine building loads,

B. Reactor building loads,

C. Radwaste building loads,

D. Crib house loads, and

E. Off-gas filter building loads.

F. Unit 1 air systems.

The service air compressors are an oil-free, rotary screw, water cooled, two stage compressor powered by a 125 H.P. electric motor. The Unit 2 service air compressor is rated for 574 ft^3/min at 110 psig. The Unit 3 service air compressor is rated for 460 ft^3/min at 100 psig. Cooling is provided by the TBCCW system. An intercooler, after cooler, moisture separator and drain traps are contained within the compressor compartment.

The service air compressors utilize an unloader solenoid valve assembly for loading and unloading.

Service air receivers are used as buffers between compressor discharge and the rest of the system. In this configuration, they dampen pressure pulses from the compressor and provide a smooth flow of air to the system. They also provide storage capacity to accommodate intervals when demand exceeds capacity of the compressor.

Relief valves in this system (i.e., at air receivers) are set at 125 psig.

A pressure regulating valve supplies air to Unit 1 air systems.

The service air system also has a carbon monoxide monitor to protect workers using the breathing air feature of the system.

9.3.1.3.2 Testing and Inspection Requirements

A periodic survey is performed to test the service-air-to-instrument-air backup valve.

A periodic survey is performed to test the service-air to Unit 1 pressure regulating valve closing pressure.

9.3.1.4 Pump-Back System

The pump-back system maintains a 1-psi drywell-to-torus differential pressure in order to decrease the amount of hydro-shocking of the torus support structure during drywell pressurization.

The pump-back system includes two 100% capacity compressors and a receiver tank. The suction for the system is from the torus, via the nitrogen supply line. The discharge is to the drywell. All controls are in the control room.

The pump-back system is cross-connected with the drywell pneumatic supply system and supplies drywell gas (typically nitrogen) to the associated instruments and actuators. The reverse is not true; that is, the drywell pneumatic supply system cannot be used as a backup to the pump-back system.

9.3.1.5 Drywell Pneumatic Supply System

The drywell pneumatic supply system has a function similar to the instrument air system, but it supplies drywell gas (instead of air) to control devices and instruments in the drywell.

Suction to the drywell pneumatic supply system is from the drywell rather than from outside air, thereby reducing the need for venting the drywell due to continuous bleeding of air from pneumatic components (air which would otherwise lead to an increase in drywell pressure and excessive oxygen concentration in the inerted atmosphere). The drywell pneumatic compressors have been permanently removed from service. The crosstie to the pump back system supplies the pressurized drywell gas to this system. The nitrogen purge system can also supply the drywell pneumatic components.

9.3.1.6 Radwaste Air Sparging System

The radwaste air sparging system is an independent air system. The system contains its own air compressors, coolers, and filters. The system is designed for a pressure of 13 psig and a capacity of 1300 ft^3/min.

The radwaste disposal system uses large amounts of low-pressure compressed air for agitating materials in waste tanks. Materials are agitated to prevent packing or solidification in the tanks.

The system is also connected with the off-gas system to provide air for system purging.

9.3.2 Process Sampling Systems

9.3.2.1 High Radiation Sampling System

Dresden license amendments #197 and #190 approves the elimination of the requirement to have and maintain HRSS in accordance with NEDO-32991. The following items were committed to as part of this amendment.

1. Dresden has developed contingency plans for obtaining and analyzing highly radioactive samples of reactor coolant, suppression pool, and containment atmosphere. The contingency plans will be contained in the Dresden chemistry procedures and implemented with the implementation of the license amendment. Establishment of the contingency plans is considered a regulatory commitment.

2. The capability for classifying fuel damage events at the Alert level threshold will be established at a level of core damage associated with radioactivity levels of 300 micro-curies/gm dose equivalent iodine. This capability will be described in the emergency plans and emergency plan implementing procedures and implemented with the implementation of the license amendment. The capability for classifying fuel damage events is considered a regulatory commitment.

3. Dresden has established the capability to monitor radioactive iodine that has been released offsite to the environs. This capability is described in the emergency plans and emergency implementing procedures. The capability to monitor radioactive iodine is considered a regulatory commitment.

All information contained in the UFSAR regarding the past Regulatory Requirements for post accident sampling utilizing the HRSS Buildings is being retained for historical purposes. Applying to UFSAR Sections 9.3.2.1.1 through 9.3.2.1.2.4.3.

9.3.2.1.1 Design Bases

The High Radiation Sampling System (HRSS) or Post-Accident Sampling System (PASS) is provided to:

- provide contingency capability to determine the degree of core damage under degraded core accident conditions through the collection and analysis of reactor coolant and containment atmosphere samples,

- provide contingency capability for the analysis of reactor coolant to verify the injection of standby liquid control into the reactor system, and

- provide contingency capability to assess the corrosion potential of post-accident reactor coolant on components and materials in contact with the coolant.

These capabilities are met through the installation of the HRSS or other procedurally approved processes, and the establishment of a program for the collection and analysis of reactor coolant and containment atmosphere samples and development of a core damage assessment procedure.

The Post-Accident Sampling (PAS) program provides the capability to sample, transport, and/or analyze reactor coolant and/or containment atmosphere samples from either unit under degraded core accident conditions.

The worst case source term as defined by Regulatory Guide 1.3 was used for shielding design and sampling/analysis considerations. The source terms used were as follows;

- The reactor coolant source term is based on the release to the coolant of 100% of the noble gas radionuclides, 50% of the halogen radionuclides, and 1% of the particulate radionuclides in an equilibrium reactor core operating at 2561 MWt; and

- The containment atmosphere source term is based on the release to the containment of 100% of the noble gas radionuclides and 25% of the halogen radionuclides in an equilibrium core operating at 2561 MWt.

- The capability to obtain and analyze a sample of reactor coolant or containment atmosphere without radiation exposures to any individual exceeding the criteria of General Design Criteria (GDC) 19[5] (Appendix A, 10 CFR Part 50), (i.e. 5 rem. whole body, 75 rem extremities).

- Re-analysis of the LOCA was performed using the guidance in Regulatory Guide 1.183. Acceptance criteria from 10 CFR 50.67 are used for Alternative Source Term.

The design of the PASS classified the system and components as non-safety-related except where tie-ins are made to a safety-related system. In the latter case, the sample piping up to the first remotely operated isolation valve was classified as safety-related. The sampling system piping and supports were designed to ANSI B31.1. The system components were not designed to seismic Category I requirements but did consider seismic loads due to the potential of routing over safety-related systems. All PASS piping in the reactor building is seismically supported.

9.3.2.1.2 System Description

9.3.2.1.2.1 Program Description

In the event post accident monitoring instrumentation is incapable of performing its function contingencies have been established for use of the HRSS system. HRSS system provides:

- The capability to obtain samples as defined within the site's program.

- The capability to obtain a reactor coolant sample.

- obtain information to be used for the determination of the degree of core damage including the following;
  - reactor coolant radionuclide data including iodines and particulate activity,
  - containment atmosphere radionuclide data including noble gas activity, and
  - containment atmosphere hydrogen concentrations.
- ensure that reactor coolant and containment atmosphere samples can be obtained and analyzed.

A core damage procedure has been developed that utilizes industry accepted practices, including the use of radionuclide data to assess the condition of the core during accident conditions.

9.3.2.1.2.1.1 Analytical Program

9.3.2.1.2.1.1.1 Radionuclides

Gamma spectroscopy instrumentation is utilized for the identification of radionuclides in reactor coolant samples, for iodines and particulate, and containment atmosphere samples (LOCA conditions only), for noble gases. Backup systems as well as locations exist in the event that radiological conditions prohibit the use of the instrumentation.

9.3.2.1.2.1.1.2 Hydrogen

The station has two (2) in-line hydrogen monitors for the drywell atmosphere. These monitors meet category I requirements as defined in Regulatory Guide 1.97. One of the monitors is primary, the second is the redundant backup. Hydrogen concentrations in the containment atmosphere are quantified (in percent by volume) via in-line monitoring with these monitors. A description of this system can be found in section 6.2.5.3.2. In addition to functions described in section 6.2.5.3.2, the hydrogen concentration in containment atmosphere is used in the core damage assessment procedure for estimating core damage.

9.3.2.1.2.1.1.3 Chloride

The PAS program provides for backup sampling and laboratory analysis capabilities through grab samples to ensure that sample analyses can be performed.

9.3.2.1.2.1.1.4 Boron

The PAS program provides for backup sampling and laboratory analysis capabilities through grab samples to ensure that sample analyses can be performed.

9.3.2.1.2.1.1.5 Dissolved Hydrogen

Reactor coolant dissolved hydrogen analysis is not performed for the following reasons. The core damage assessment procedure does not require this data for assessing the degree of damage to the core. Additionally, the reactor system is designed to remove non-condensible gases during the process of cooling down. Therefore, any dissolved hydrogen concentrations obtained would not be indicative of the hydrogen inventory generated as a result of degrading core conditions. Use of this data would lead to a non-conservative estimate of the degree of core damage.

9.3.2.1.2.1.1.6 Dissolved Oxygen

Reactor coolant dissolved oxygen analysis is not performed for the following reasons. The core damage assessment procedure does not require this data for assessing the degree of damage to the core. The usefulness of this parameter is in the assessment of the corrosion potential of reactor water to materials and components in contact with the coolant. Verification of reactor coolant dissolved oxygen concentrations below 100 ppb is required by NUREG 0737. In a shutdown condition (normal or post-accident), dissolved oxygen concentrations greater than 100 ppb will exist since there is no means available to reduce the concentration to below 100 ppb.

9.3.2.1.2.1.1.7 pH

NUREG-0737 does not require pH analysis of reactor coolant.

During the accident management phase, reactor coolant pH analysis is not performed for several reasons. The core damage assessment procedure does not require this data for assessing the degree of damage to the core and emergency procedures in place to estimate the potential radionuclide inventory available for release do not consider water chemistry conditions such as pH. Emergency procedures follow guidelines established by NUREG-1228[6] for estimating source terms during the accident response period. Factors to be used for quantifying the iodine source term in the gaseous phase available for release are pre-defined and do not require reactor coolant pH as an input. When needed, grab samples of undiluted reactor coolant can be obtained for laboratory analysis during the recovery phase when radiation levels have decreased.

9.3.2.1.2.1.2 Quality Control Program

Station chemistry procedures are in place to ensure the accuracy of the data and the functionality of the system. These procedures ensure that all instrumentation can produce accurate results. Elements of the analytical and radioanalytical instrumentation quality control program include: maintenance, calibration, performance checks and periodic use of the instrumentation. Whenever practical, the instrumentation and/or portions of the PASS system are integrated into normal day to day operational activities to ensure functionality and availability.

The PASS employs a minimum number of valves which will become inaccessible for repairs after an accident. These valves are not within the scope of 10CFR50.49 and are therefore exempt from the requirement for formal documentation. These valves however have been procured to design specifications appropriate for the expected post-accident environmental conditions in which they will operate.

The PAS program is tested periodically to verify that the program objectives can be met. In addition, various components of the PASS such as gauges, valves, indicators, switches, and regulators are periodically verified for functionality.

9.3.2.1.2.1.3 Sample Storage and Control

The PAS program original design ensured that equipment provided for backup sampling was capable of providing at least one sample/day for 7 days following onset of the accident and at least one sample/week until the accident condition no longer exists. A place for storage of these samples has been established onsite and incorporates the use of shielding to minimize the buildup of radiation fields within the immediate area. In some case, the samples may be transported to an off-site facility.

9.3.2.1.2.1.4 Alternative Power Source

A motor control center (MCC) is located in the operating area of the HRSS building and provides a 480-V power supply to the HRSS and HVAC equipment and a 208/120-V power supply for controlling lighting, and heat tracing the sample tubing. This MCC is powered from 480-V bus 26 (unit 2) or bus 36 (unit 3).

Should a loss of off-site power event occur, the PAS program provides for an alternative power source that can be linked up. Standby diesel power is available for the HRSS building and the MCC can be energized to meet the time limits for sampling and analysis under post-accident conditions.

9.3.2.1.2.1.5 Radiation Exposure Minimization

The program considers the need to meet GDC 19 requirements. In the development of the program, information regarding the worst case scenario for plant radiation fields during the accident has been used with time motion studies to verify that activities including preparation, sample collection, sample transport, sample analysis, and sample disposal will not result in personnel exposures in excess of GDC 19 requirements.

9.3.2.1.2.2 Post-Accident Sampling System Description

The post-accident sampling system (PASS) provides for the following operational capabilities:

- transfer a sample fluid from the source to the sampling area,
- control the temperature and pressure of the sample,
- obtain a reactor coolant grab sample in a shielded container suitable for transport to an onsite or offsite laboratory for analysis,
- obtain a grab sample of the containment atmosphere in a shielded container suitable for transport to an onsite or offsite laboratory for analysis, and
- store, handle, and return to the plant waste generated by the sampling operations.

9.3.2.1.2.2.1 PASS Components

The PASS consists of systems and equipment needed to safely obtain reactor coolant samples and containment atmosphere samples. The major components of the PASS include the following;

- Liquid Sample Panel (LSP),
- Containment Air Sample Panel (CASP),
- Chemical Analysis Panel (CAP),
- Chemical Monitoring Panel (CMP),

- a cooling rack for thermally hot samples,
- chilled water system,
- a sample waste collection system,
- valves and piping for the HRSS,
- an independent heating, ventilation, air conditioning (HVAC) system,
- controls for the entire system, and
- a communication system to the control room.

P&ID drawings M-1234, sheets 2, 3, and 4, M-1235, M-1236, M-1237, M-1239 sheets 2, 3, and 4, M-1240, M-1241, and M-1242 provide schematic details of the layout of the HRSS.

9.3.2.1.2.2.2 General Arrangement

PASS components for each unit are housed in a dedicated (HRSS) building which is located adjacent to the respective reactor building. A connecting trench extends the piping and electrical lines from the reactor building to the HRSS building.

The HRSS building is designed in accordance with the Uniform Building Code requirements for Zone I. The HRSS buildings for both units are located south of the corresponding reactor buildings. Each building is free standing and arranged into four separate areas as described in the following.

The HRSS building equipment layout is based on dividing the building area into the following four distinct radiation zones;

1. The vestibule area - where preparations are made for entry to the sampling areas. The building is entered through the vestibule area which contains a clothing change area and a portal radiation monitor. The vestibule is separated from the operating area by a wall with a door.

2. The operating area - where all control and sampling manipulations at the panels are performed. The operating area contains the control panels for liquid and containment air sampling, the motor control center, the LSP, the CAP, and the CASP. The HVAC system control panel is located adjacent to the vestibule. An aisle in front of these panels is provided for manual operations such as valve alignment at the panels, calibration, and shielded cask cart movement.

3. The maintenance aisle - which serves as access to the rear of the sampling panels for maintenance purposes. The maintenance aisle behind the sampling panels is separated from the operating area by a combination of concrete shield walls and a shield door.

4. The pit area - which contains the waste tank and pumps, and serves as the pipe and valve gallery. The pit area houses the sample waste tank, the waste pumps, the sample coolers, the chilled water system, and the HRSS building sump. This area is adequately shielded in view of the very high radiation levels associated with post-accident sample wastes that are collected in the waste tank. A 5-foot wide, 3-foot deep concrete trench with removable 2-foot thick concrete covers connects the reactor building and the pit area. Piping carrying process samples, demineralized water, instrument air, electrical power, control cables, and other services are located in the trench.

The interior finishes of the HRSS building are sealed and painted to provide for decontamination of wall and floor surfaces. This will provide surfaces which minimize the penetration of any spilled radioactive liquids into the concrete and allow ease of decontamination of areas.

9.3.2.1.2.2.3 High Radiation Sampling System Building Environmental Control

The HRSS building HVAC system provides heating and cooling, filtered and unfiltered exhaust systems, and positive control of airflows. Conditioned air is supplied to the HRSS building to offset the environmental and internal loads seen by the building. A single filter bypass fan is provided for routine operation to prevent the filters from loading. Control of airflows is provided to assure that the HRSS building is maintained at a negative pressure with respect to the environment. The exhaust air flow rate is maintained at approximately 1000 ft^3/min while the intake air flow rate is maintained at an adjustable differential to ensure infiltration into the HRSS building. To control airborne contamination, the building ventilation is designed such that the air flows from the lesser to the higher contaminated zone, i.e., from the vestibule to the operating area to the maintenance aisle and finally to the pit where it is exhausted outside the building to the plants ventilation system. Under normal conditions the exhaust is not filtered. During a post-accident condition, the exhaust would be routed to the exhaust filter unit. For enhanced reliability, redundant exhaust fans are provided on the filtered train. In both cases, the exhaust is routed to the station's 310 foot chimney and is tied into the ventilation duct in the base.

By design, air is exhausted from the three sample panels to control inleakage at approximately 100 ft^3/min for the CAP, 300 ft^3/min for the CASP and 360 ft^3/min for the LSP to control internal leakage. The exhaust air may be passed through a combined high efficiency particulate air (HEPA) and activated carbon filter train. The HRSS building ventilation system is shown in P&ID drawing M-1236 for unit 2 and M-1241 for unit 3.

All components of the HRSS, with the exception of tubing and valves in the reactor building, are located in the HRSS building. The HRSS building temperature is maintained at approximately 75 degrees F. No severe environmental conditions are imposed on the design of the system components. The heating, ventilation, and air conditioning (HVAC) equipment is located outdoors and is designed for -20 to 105 degrees F, and snow and wind loads.

9.3.2.1.2.2.4 Radiation Shielding

The PASS is designed to provide the capability to extract, monitor, analyze, and dispose of samples of reactor coolant and containment atmosphere during post-accident conditions with radiation exposures well below the criteria of General Design Criteria (GDC) 19 (10 CFR 50, Appendix A). To meet GDC 19 requirements, the following criteria were used in the design and construction of the shielding for the HRSS building, the sample panels, and sampling processes:

- limit the dose rate to 15 mrem/hr in general occupancy areas and 100 mrem/hr in areas infrequently occupied except directly in front of the sample panels, and

- limit the whole body exposure to 100 mrem per technician per sampling exercise in the HRSS building.

The HRSS building is provided with 3 foot thick external walls and a 2 foot thick roof to limit the radiation dose inside the building due to the post-accident radiation sources within the reactor building. Within the HRSS building, concrete shield walls protect the technician in the operating area from radiation sources due to sample flow in tubing, panels, and waste collection tanks.

The LSP is provided with a front panel shield consisting of 7 inches of lead shot (0.09 inches in diameter) sandwiched between two ½ inch steel plates. Shield glass viewing ports are provided for observing the sample bottle needle area and the gauges. The integral steel base consists of 5 inches of lead shot (0.09 inches in diameter) sandwiched between two ½ inch steel plates.

Additional radiological protection features include the following;

- The CAP is provided with front panel and base shield similar in size and configuration to the LSP.

- Provisions exist to purge the sample lines in both the LSP and CAP with demineralized water once the sampling and in-line analysis has been completed.

- The CASP has a front panel of 3 inch thick steel plate which provides adequate shielding from radiation fields present within the CASP hardware.

- The sample tubing raceway in the maintenance aisle is provided with a 4 inch thick steel cover to reduce the dose contribution from this source.

To prevent radiation streaming from the gaps around the LSP, CAP, or CASP, these gaps are packed with lead wool. Laboratory procedures and localized shielding are utilized to maintain doses to laboratory workers well below the allowable levels in GDC 19.

9.3.2.1.2.3 PASS Function Descriptions

9.3.2.1.2.3.1 Reactor Coolant Sample Lines

The LSP is designed to obtain samples during degraded core accident conditions from the following sample points:

- the reactor recirculation discharge line of the "B" loop of unit 2 and the "A" loop of unit 3 at a point downstream of the pump discharge isolation valve,
- the low pressure coolant injection (LPCI) discharge header downstream of the containment cooling heat exchangers, and
- downstream of the shutdown cooling (SDC) system heat exchangers.

These sample points ensure that reactor coolant samples can be obtained under the following plant post-accident conditions:

- post-accident with no coolant loss,
- post-accident, ECCS during a small loss-of-coolant accident (LOCA), and
- post-accident, ECCS during a large LOCA.

Reactor coolant sampling during post-accident conditions does not require an isolated auxiliary system (e.g., reactor water cleanup system (RWCU) to be placed in operation in order to use the sampling system. The sample lines include containment isolation valves that will close upon initiation of a containment isolation signal or a safety injection signal. These valves can be remotely controlled from the control room to facilitate sampling during and after an accident. P&ID drawings M-26, M-29, M-32, M-357, M-360, and M-363 provide detailed information on the process sample lines.

The sample is transferred from the source to the sampling panel through stainless steel tubing. The sample lines are ½ inch OD Type 304 stainless steel tubing of all welded construction up to the sample panels. Optimum sample velocities have been specified to minimize settling and plateout, and to keep sample lines from clogging.

The heatup of the drywell during a postulated loss of coolant accident could, in turn, heatup the volume of liquid trapped between the containment isolation valves of the sample line from the reactor recirculation discharge line. Heatup of this trapped volume could overpressurize and fail the associated piping, creating a bypass path for the primary containment. The effect of this thermal pressurization has been analyzed using Appendix F of Section III of the ASME B&PV Code, 1977 Edition through S'77 Addenda. The results demonstrate that the stresses remain within Appendix F allowables. Therefore, the pressure boundary integrity of primary containment is maintained.

9.3.2.1.2.3.2 Temperature Control

Cooling of the sample fluid to 120 degrees F has been provided for liquid sample lines having a post-accident temperature greater than 120 degrees F. The cooling is accomplished by shell and tube type heat exchangers.
The sample cooling water is provided by a chilled water system that includes two redundant air-cooled condensing units and direct expansion coils which are immersed in the two chilled water storage tanks (see P&ID drawings M-1237 and M-1242). The chilled water is constantly recirculated and passed through the expansion coils by a set of recirculation pumps. A second set of pumps provide a chilled water supply to the sample cooling rack. The temperature of the chilled

water is maintained at 60 degrees F. Thermal storage capacity is provided in the tanks which will allow obtaining at least two high temperature samples even in the event of complete failure of the refrigeration equipment.

9.3.2.1.2.3.3 Containment Atmosphere Sample Lines

The CASP is designed to obtain samples during degraded core accident conditions from the following sample points:

- the drywell atmosphere at three different sample locations: east coolers area, west coolers area, and reactor head vent area,
- the torus atmosphere, and
- the standby gas treatment system.

The sample lines include containment isolation valves that will close upon initiation of a containment isolation signal or a safety injection signal. These valves can be remotely controlled from the control room to facilitate sampling during and after an accident. Flow through the sample line is established through the use of an eductor that uses nitrogen gas. The flow through the sample line is returned to the containment as a means to manage the radioactive gas. P&ID drawing M-1235 and M-1240 provide detailed information on the process sample lines.

The main sample line is a ½ inch OD Type 304 stainless steel tubing tied into existing sample points downstream of the containment isolation valves. To minimize radiation field buildup in the sample line resulting from plateout, the sample tubing is run with large radius bends, flow velocities are maintained at 10 ft/s, and the tubing is heat traced to maintain it at 275 degrees F. To ensure that representative sampling is achieved, welded tubing is used up to the panel. The heat tracing is also needed to prevent condensation from occurring within the sample line.

9.3.2.1.2.3.4 Liquid Sample Panel

The sample extraction takes place in the LSP. The LSP is a free-standing, self-supporting structure containing the necessary sample tubing, valves, and gauges within a totally enclosed panel.

The LSP contains a reactor coolant sampling module that receives the different sources of primary reactor coolant entering, one at a time, at a maximum of 120 degrees F and 1600 operating (2300 design) psig. Design flow rates through the panel are: 1900 cc/min during purging, and 200 cc/min during sampling. The module has power operated valves to automatically stop either purge or sample flow in the event of excessive sample temperatures resulting from failure of the chilled water system.

The LSP has the following capabilities:

- Collection of an undiluted depressurized reactor coolant in a sealed bottle. The bottle is remotely lowered into a shielded cask. The cask is removed from the panel. Depending on the radiation levels, this sample may be analyzed onsite.

- Collection of a diluted (1 to 1000) depressurized sample in a sealed bottle. The bottle is remotely lowered into a shielded cask. The cask is removed from the panel and transported to the laboratory for chemical and isotopic analysis.

The reactor coolant sample is drawn from the appropriate sample point after considering plant conditions.

The LSP sample lines are ¼ inch OD Type 304 stainless steel and can be flushed with demineralized water. The purge and flush volumes can be stored in the HRSS waste tank before pumping the wastes to the drywell floor drain sump or to the reactor building equipment drain tank. The design minimizes the potential for leakage of samples. Should a rupture of the reactor coolant line occur anywhere along the sample line outside containment, the containment isolation valves can be remotely closed. The volume of reactor coolant released over time is limited by the sample line size. Leakage within the PASS panels are contained and routed to the HRSS waste tank.

9.3.2.1.2.3.5 Chemical Analysis Panel

The CAP is located next to the LSP and is interconnected with the LSP. The sample input to the CAP is from the LSP where it has been conditioned, i.e., cooled and depressurized to the design requirements of the CAP. The CAP is a free-standing, self-supporting structure. A graphic display showing the sample and support services flow paths, flow and pressure indicators, calibration reagent tanks, and other components are mounted on the front face of the panel. The effluent from the CAP is routed to the waste tank.

To reduce radiation levels, the tubing within the panel is ¼ and _ inch OD Type 304 stainless steel. Provisions have been made for flushing the sample lines with demineralized water.

9.3.2.1.2.3.5.1 Chloride

Dresden license amendments #197 and #190 have removed the post-accident requirement for this sample.

9.3.2.1.2.3.5.2 Boron

Dresden license amendments #197 and #190 have removed the post-accident requirement for this sample.

9.3.2.1.2.3.6 Containment Atmosphere Sample Panel

The containment atmosphere sampling system consists of a control panel with plant valve indications, the containment atmosphere sample (CAS) control panel, the CASP, and the gas partitioner controls. The CASP is an enclosed cabinet with provisions for the connection of a gas partitioner for collecting a sample. The panel encloses a network of valves, tubing (¼ inch OD Type 304 stainless steel), fittings, instruments, and quick-connect couplings. The CASP routes part of the gaseous sample from the main sample line to the gas partitioner. The CAS control panel operates the valves at the CASP and is located in the operating space area. All of the CASP operations, with the exception of the operations mentioned above, are performed in this space. The gas partitioner is designed to separate the iodine and particulate from the noble gas. A gas collection vial is used to capture the noble gas for radionuclide analysis. Gas cylinders required for operation of the CASP are located outside the HRSS building.

9.3.2.1.2.3.7 Control and Monitoring Panels

Three individual control panels for the operation of the LSP, CAP, and CASP are located in the operating area of the HRSS building and shielded from the sample panels by a 3 foot thick concrete wall. Most of the operations for sampling and monitoring are performed from the following panels to limit the radiation dose to the technician from the radioactive fluids in the sample panels.

- PASS control panel - the PASS control panel consists of three sections. Annunciator windows indicating various alarm conditions are located in the top section. The midsection contains a graphic layout displaying the liquid and gaseous sample system flow paths, valves, pumps, and other equipment. All hand switches with indicating lights for operating valves,

pumps, and HVAC equipment are located in the lower section of the control panel.

- Chemical monitor panel (CMP) - the CMP is an auxiliary panel which contains the indicating and recording equipment for the cells and analyzers which are mounted in the CAP. The panel permits the technician to work with, and observe the indicating and recording equipment from a remote location to reduce exposure under post-accident conditions.

- CAS control panel - the CAS control panel contains selector switches, pilot lights, an annunciator system, and a pressure controller and gauge. A mimic diagram of the CASP flow paths, valves and equipment is also provided on the panel. The technician uses this control panel to select, initiate, and control sample filling exercises.

9.3.2.1.2.3.8 Waste Handling System

The HRSS waste handling system is provided to handle both liquid and gaseous wastes resulting from the sampling operations. In addition to the spent sample itself, waste is generated in the purging of the sample lines. At the conclusion of the sampling sequence, the lines are flushed to reduce the background activity. Each sample extraction produces approximately ten gallons of waste fluid.

The waste handling system consists of a 250-gallon stainless steel collection tank supplied with two horizontal centrifugal discharge pumps which will handle approximately one week of sampling operation. Liquids enter the tank via a 2 inch drain header. The discharge of the tank is directed to the reactor building equipment drain tank during normal operation and may be directed into the drywell floor drain sump during the post-accident mode.

During post-accident conditions, the incoming samples may contain large quantities of dissolved hydrogen which will accumulate in the waste tank. Noble gases dissolved in the sample will also be stripped and will accumulate in the tank. Inerting and evacuation features are provided to control the concentration of these gases. Since the hydrogen concentration can be approximately 30% by volume, the tank is inerted with nitrogen prior to filling to preclude an explosive gas mixture. Since the tank's atmosphere is not monitored, a rupture disc is provided as backup in the event of detonation of a combustible mixture in the tank. For control of gaseous radionuclides, an evacuating compressor will vent the tank's contents back to the drywell. During normal operation, the tank is vented to the HRSS HVAC and operates on a nominal ¼ inch H_2O negative pressure.

9.3.2.1.2.4 Analytical and Radioanalytical Capabilities Description

Analytical and radioanalytical methods used for post-accident sample analyses are reviewed for the following criteria:

- chemical effect of the post-accident coolant matrix,

- time and radiological dose limitations of analyses,
- radiation effect on method and/or equipment,
- compliance with sensitivity and range requirements,
- sample size requirements, and
- accuracy of the analysis methods.

When practical, instrumentation used on a routine basis will be utilized also for fulfilling post-accident analysis requirements. This practice should help to increase the availability and reliability of the method.

9.3.2.1.2.4.1 Facilities

The hot laboratory is located in the chemistry building on the ground floor elevation. A fork lift will be required to transfer the sample cask from the HRSS building (unit 2 or 3) to the chemistry building. The hot laboratory will be used unless radiation fields do not permit the use of the facility. Then backup laboratory instrumentation in the unit 3 HRSS building will be utilized. The area where the samples will be taken for radionuclide analysis is dependent on the radiation fields present in the main counting facilities. Samples will be counted in the counting room adjacent to the hot laboratory unless the radiation fields are greater than 2.5 mR/hr. Then the samples will be shipped off-site for radionuclide analysis.

9.3.2.1.2.4.2 Laboratory Instrumentation

Instrumentation and procedures exist onsite to measure for boron concentrations of 0.5 to 3.0 ppm with an accuracy of ± 20%. The analysis is performed on a 1 to 1,000 diluted reactor coolant sample in the hot laboratory or taken to an off-site facility as backup.

Instrumentation and procedures exist onsite to measure for chloride concentrations within a range of 0.1 to 20 ppb with an accuracy of ± 20% for concentrations above 0.5 ppb and ± 0.05 ppb for concentrations under 0.5 ppb. The analysis is performed on a 1 to 1000 diluted reactor coolant sample in the hot laboratory.

9.3.2.1.2.4.3 Radioanalytical Instrumentation

The counting room instrumentation is capable of acquiring nuclear spectra and identifying constituents of the spectrum as well as quantifying each constituent. Both systems consist of a computer system tied in with a germanium detector and are capable of determining radionuclide concentrations in liquids and gases of varying sample sizes and storing this information for future use. The radionuclide analysis capability include provisions to identify and quantify the

isotopes of the nuclide categories discussed above to levels corresponding to the source terms given in Regulatory Guide 1.3. Sensitivity of onsite liquid sample analysis capability permits measurement of nuclide concentrations in the range from approximately 1 μCi/g to 10 Ci/g. The PAS program ensures that background levels of radiation in the counting facility are less than 2.5 mR/hr so that the radionuclide analysis will provide results with an acceptably small error (approximately a factor of 2).

9.3.2.2 Turbine Building Sampling System

The turbine building sampling system provides for in-line monitoring of the following parameters for each unit. Hotwell liquid is monitored for conductivity from two sample locations. The effluent from each of the seven condensate demineralizers, the condensate makeup, and the main condensate demineralizer header is monitored for conductivity. Effluent from the 100% condensate filtration system is periodically monitored for insoluble iron. Effluent from the reactor feedwater line, condensate pump discharge, and control rod drive pump discharge is monitored for conductivity and dissolved hydrogen and oxygen. A high-conductivity condition in either or both the main condensate demineralizer header or condensate pump discharge effluent is annunciated by an alarm in the control room.

All of the turbine building sampling system sampling lines have provisions for manually extracting samples of effluent. Three of the sample lines: the line from the main condensate demineralizer header, the reactor feedwater line, and the condensate pump discharge header, also have provisions for manually extracting filtered samples of effluent.

The turbine building sampling system is equipped with the necessary auxiliary components to support the sampling process, such as coolers, pressure and flow indicators, pressure and flow controllers, and supplies of demineralized water and instrument air. Additional sampling taps are provided on each sampling line for the installation of any additional measuring devices in the future.

9.3.2.3 Reactor Building Sampling System

The reactor building sampling system provides for inline monitoring of the following parameters for each unit. The effluent from each of the reactor water cleanup demineralizers, fuel pool demineralizer inlet and outlet, and the fuel pool filter inlet, is monitored for conductivity. The effluent from reactor water recirculation is monitored for conductivity and dissolved hydrogen and oxygen. The effluent from the inlet of the reactor water cleanup filter is monitored for conductivity, dissolved oxygen, and pH. A high-conductivity condition in the effluent samples from the reactor water cleanup filter inlet, reactor water recirculation or reactor water cleanup demineralizer, is annunciated in the control room.

All of the reactor building sampling system sampling lines have provisions for manually extracting samples of effluent. Taps are available which are specifically designated for connecting to an ion chromatograph; effluent samples from reactor

water recirculation, reactor water cleanup filter inlet, and reactor water cleanup demineralizer can be routed to the chromatograph for additional analyses.

The reactor building sampling system is equipped with the necessary auxiliary components to support the sampling process, such as coolers, pressure and flow indicators, pressure and flow controllers, and supplies of demineralized water and instrument air. Additional sampling taps are provided on each sampling line (except for the reactor water cleanup filter inlet) for the installation of any additional measuring devices in the future.

9.3.2.4 Off-Gas Sampling System

Each unit's off-gas sampling system is constructed in two parts: a vial sampler and the sampling rack. The vial sampler provides the ability to extract a sample of effluent from the following:

A. Absorber train bypass line;

B. Inlet to A charcoal absorber;

C. Inlet to B charcoal absorber;

D. Inlet to C charcoal absorber;

E. Inlet to F charcoal absorber;

F. Inlet to J charcoal absorber;

G. Inlet to buried shielded filters; and

H. Absorbers recirculation line.

The vial sampler is equipped with a remotely operated pump, a grab sample vial, and an in-line filter.

The sampling rack provides the ability to extract a sample of effluent from only the inlet to the off-gas system's buried and shielded filters (which is also the discharge of the third and final absorber train; all three trains are connected in series). The rack is equipped with two remotely operated pumps, in-line suction-side and discharge-side particulate filters, flow and pressure indicators and pressure switches.

Samples from the vial sampler and the sampling rack are discharged into a common sample return header, which directs samples back to the outlet of the off-gas system's prefilter.

9.3.2.5 Off-Gas Filter Building Air Particulate Sampling System

The off-gas filter building air particulate sampling system is common to both Units 2 and 3. It is designed to extract effluent samples from various off-gas filter building cubicles and valve aisles. This system utilizes one vacuum pump, which draws samples through a particulate air sample and charcoal filter, and which functions on the same water seal principle employed in the primary containment particulate sampling system.

9.3.2.6 Primary Containment Oxygen Sampling System

Each unit's primary containment oxygen sampling system provides the ability to extract gas samples from the vicinity of two drywell cooler intake louvers, from the drywell vent head return opening (near the reactor bell seal), and from the torus. These supply sample lines are each equipped with two automatic containment isolation valves which close automatically on a Group II isolation signal.

Sample extraction and analysis are done in a recirculation mode, whereby samples are drawn through a filter using a sample pump; transferred to and through a cooler (with cooling water from the reactor building closed cooling water [RBCCW] system), moisture collector, and oxygen analyzer; and then are pumped back through a common sample return header to the drywell. This common return header is isolated by two pneumatic drywell isolation valves which close automatically on a Group II isolation signal. The oxygen analyzer can be calibrated using gas standards, introduced into the system from portable gas-standard bottles. Each sampling line is equipped with a bypass line which diverts samples to and through a flow-measuring device (with local flow readout) and a bypass pump. Bypass flow is diverted back to the drywell via the common return header. All sampling lines are heat-traced up to the sampling panel.

9.3.2.7 Primary Containment Particulate Sampling System

9.3.2.7.1 Design Bases

The primary containment particulate sampling system (PCPSS) constitutes one of three subsystems of the primary containment leak detection (PCLD) system (the other two are the primary containment drain flow monitoring system and air temperature monitoring system). The PCPSS is designed to extract air samples from various locations in primary containment and the steam tunnel. Four of the sampling supply lines and one common sampling return line are each isolated using two pneumatically controlled containment isolation valves. Each second isolation valve along the sampling line and away from the drywell marks the safety-related boundary of the PCPSS; that is, all five lines are designated safety-related from the point at which samples are extracted (or returned) in the drywell to the second containment isolation valve. All of the pneumatically controlled containment isolation valves close automatically upon the initiation of a Group II containment isolation signal. All other sampling supply lines rely on double, manual valve isolation for containment isolation. Automatic containment isolation valves are not provided for these sample lines since they were originally designated for occasional use. Separation and redundancy criteria were applied only in the design of the PCPSS and its impact on containment isolation.

All of the sampling lines, except for one, were installed using stainless steel tubing. The one exception, the steam tunnel sampling line, was installed using copper tubing. The safety-related portions of PCPSS were designed to function during and after postulated seismic events (operating basis earthquake [OBE] and safe shutdown earthquake [SSE]). Safety-related components were supported in accordance with seismic Category I requirements. Nonsafety-related components that can affect the safety function of safety-related components were also supported in accordance with seismic Category I requirements.

9.3.2.7.2 System Description

Each unit's primary containment particulate sampling system provides the ability for extracting gas samples from various primary containment areas and from the steam tunnel. These samples are drawn through filters which collect air particulates and can be removed for analysis.

Gas samples from the vicinity of two of the drywell cooler intake louvers, the vent return opening near the reactor bell seal, and the torus are extracted for particulate sampling via the sampling lines of the primary containment oxygen sampling system. Gas samples from these four areas are diverted to the particulate sampling system from a point just upstream of the oxygen sampling system's panel. Each of these four sampling lines has its own PCPSS entry point, which can be isolated with two in-series, manually operated valves (in addition to the upstream automatic isolation valves near primary containment). Gas samples from all other primary containment areas are drawn through area-designated, occasional-use sampling lines and sampling filters. These lines can be selectively opened and closed in an effort to identify potential sources of primary containment leakage.

This system is equipped with the necessary flow, level, and pressure indicators; a clean demineralized water supply; and provisions for connecting a portable sampling pump.

On Unit 3, samples are drawn through the filters using a vacuum sampling pump, which employs the water seal principle to function. Having passed through the filters, sample air is pushed through the pump and into a separator by the water's centrifugal action. Once in the separator, water is drawn off and recirculated back to the sampling pump in a closed loop fashion via a cooler (with cooling water from the reactor building closed cooling water [RBCCW] system). Separated air is then forced back to and through the oxygen process sampling system common return header to the drywell.

The Unit 2 Drywell Particulate Sample Pump and associated skid components have been abandoned in place. A portable sampling pump must be connected to existing sample tubing when the need for sampling arises.

9.3.2.8 Turbine Building Air Particulate Sampling System

Each unit's turbine building air particulate sampling system provides the ability for extracting gas samples from various turbine building areas. Samples are drawn through particulate and charcoal filters and can be removed for analysis. Sample extraction is performed in the same way as for primary containment particulate air extraction.

9.3.2.9 Feedwater Particulate Sampling System

Each unit's feedwater particulate sampling system provides the ability for extracting a sample of feedwater from a point downstream of the feedwater heaters (upstream of the reactor). Samples are drawn through a particulate filter which can be removed for analysis. Since samples are discharged directly to the condenser, no sampling pump is required; the difference in pressure between the feedwater header and the condenser provides the motive force. This system is equipped with the necessary flow indicators and cooler (with water from the TBCCW system on the shell side).

9.3.2.10 Radwaste Building Sampling System

The radwaste building sampling system is common to both units and provides grab sample points for the following:

A. At the radwaste sample sink:

   1. Floor drain collector pump outlet,
   2. Floor drain filter outlet,
   3. Waste collector pump outlet,
   4. Two waste collector filter outlets,
   5. Two waste neutralizer tank outlets,
   6. Waste surge tank outlet,
   7. Waste demineralizer outlet,
   8. Three waste sample tank outlets, and
   9. Two floor drain sample tank outlets.

B. At the maximum recycle demineralizer sample sink:

   1. Two floor drain demineralizer inlets, and
   2. Two floor drain demineralizer outlets.

C. At the maximum recycle sample sink:

   1. Two floor drain neutralizer tank outlets, and
   2. Two concentrator condenser outlets.

All three sampling sinks are covered by protecting hoods which are connected to the HVAC exhaust system. Clean demineralized water is supplied to each of the sampling taps for flushing. All sampling operations are performed manually without the assistance of any continuous monitoring or process indicating devices.

9.3.3 Equipment and Floor Drainage Systems

The objective of the equipment and floor drainage system is to collect and control all waste liquids and ensure they are processed or disposed of properly. The system drain sumps also provide detection of leakage from other systems. The sumps are sampled periodically to assist in the determination of leakage sources.

The equipment and floor drainage systems consist of equipment and floor drains in the drywell, reactor building, turbine building, and radwaste building. Radioactive waste sumps associated with the drainage systems are used to accumulate and transfer the liquids collected from the reactor process systems. These liquids are processed according to their purity (conductivity levels); therefore, it is necessary to segregate the collective process as much as possible in order to provide a safe, efficient and economical method of treatment.

In general the collected liquids are divided as follows:

A. The equipment drains and sumps normally contain low-conductivity waste. Process fluids collected in the equipment drains and sumps are generally at elevated temperatures which, in some cases, require a cooler. Liquid entering the equipment drain system typically comes from anticipated equipment leakage, such as pump seal leakoff, vents and drains from major pieces of equipment, and relief valve outlets, which are hard-piped directly to the drain system.

B. The floor drains and sumps normally contain liquid with moderate- to high-conductivity levels. Liquid flowing into the floor drain system typically comes from the floor drain hubs and reflects leakage from unanticipated sources which are not hard-piped into the drain system.

The various drain systems and sumps are addressed in the following subsections.

9.3.3.1 Drywell Equipment Drains and Sumps

Each unit has one drywell equipment drain sump (DWEDS) with two pumps located beneath the reactor vessel. A heat exchanger cooled by RBCCW provides temperature control of the DWEDS by automatically opening a heat exchanger recirculation valve on the discharge of the pump and closing the discharge to radwaste valve when the high-temperature setpoint is reached. Once the temperature of the water in the sump is reduced to an acceptable level, flow is automatically established to radwaste. The high-temperature setpoint for the automatic recirculation valve is adjustable in the control room. A high-temperature alarm is also provided.

The DWEDS can be used in combination with other leak detection equipment to determine the source of leakage in the drywell. Leakage into the DWEDS is considered identified leakage and is composed of the normal pump seal and valve packing leakage and does not represent a safety consideration as long as the leakage is small compared to the reactor coolant makeup capacity available.

Drywell equipment drain sump pump and controls are located in the main control room. A total flow integrator is provided in the main control room to measure flow from the DWEDS. This total flow is used to determine total primary system leakage for comparison with Technical Specification limits. A run time meter and pump indicating lights are provided for each sump pump at a radwaste control room panel.

The drywell equipment drain liquid is pumped to the waste collector tank through a common line with the reactor building equipment drains. One sump pump starts on a sump high-level signal with the second pump starting on a sump high-high level signal. A high-high level alarm is provided in the main control room. The pumps trip on sump low-level or when the primary containment valves are not fully open.

Since the primary containment isolation valves are normally in the closed position, the pumps do not normally automatically start as a result of a sump high or high-high level signal. An operator normally empties the drywell sump periodically by opening the primary containment isolation valves from the control room. A pump will then automatically start if a sump high-level signal is present (two pumps start if a high-high level signal is present). If a sump high-level signal is not present, the operator manually starts a pump.

If the equipment drain sump leak detection is unavailable, it is possible to monitor total leakage with the floor drain sump leakage detection alone. In this case the equipment drain sump will overflow and leakage will be collected in the floor drain sump. Under this scenario, all leakage will be conservatively treated as unidentified leakage as discussed in Section 5.2.5.6.4.

However, a control interlock built into the DWEDS heat exchanger recirculation valve control allows for recirculating sump liquid whenever liquid temperature exceeds a high setpoint and the primary containment valves are closed. Drywell equipment drain liquid can be diverted to the radwaste floor drain system via the floor drain collector tank.

Drywell equipment drain sump containment isolation valves close when a Group II primary containment isolation signal is received and fail closed on a loss of instrument air. Valve closure not only isolates the drywell but also prevents the pumping of potentially contaminated liquid from the drywell. For this reason a seal-in circuit is included on these valves to keep the valves closed after the isolation signal is reset.

The heatup of the drywell during a postulated loss of coolant accident could, in turn, heatup the volume of liquid trapped between the drywell equipment sump discharge valve and the drywell equipment drain sump containment isolation valves. Heatup of this trapped volume could overpressurize and fail the associated piping, created a bypass path for the primary containment. A relief valve has been installed in this section of piping to prevent the potential overpressurization from the thermal expansion of the trapped fluid.

9.3.3.2 Drywell Floor Drains and Sumps

Each unit has one drywell floor drain sump (DWFDS) with two pumps located beneath the reactor vessel. Flow into the DWFDS is considered unidentified leakage. Drywell floor drain sump pump controls are located in the main control room. A total flow integrator is provided in the main control room to measure flow from the DWFDS. This total flow is used to determine if unidentified leakage exceeds Technical Specification limits. A run time meter and pump indicating lights are provided for each sump pump at a radwaste control room panel.

The DWFDS liquid is pumped to the floor drain collector tank through a common line with the reactor building floor drains. This line can be cross-connected to flow to the waste collector tank; however, connecting the lines is not desirable since the floor drain water quality is generally poorer than the equipment drain water quality. One sump pump starts on a sump high-level signal with the second pump starting on a sump high-high level signal. A high-high sump level alarm is provided in the main control room. The pumps trip on sump low-level or primary containment valves not fully open.

If the floor drain sump leak detection is unavailable, it is possible to monitor total leakage with the equipment drain sump leak detection alone. In this case the floor drain sump will overflow and leakage will be collected in the equipment drain sump. Under this scenario, all leakage will be conservatively treated as unidentified leakage as discussed in Section 5.2.5.6.4.

Since the primary containment isolation valves are normally in the closed position, the pumps do not normally automatically start as a result of a sump high or high-high level signal. An operator normally empties the drywell sump periodically by opening the primary containment isolation valves from the control room. A pump will then automatically start if a sump high-level signal is present (two pumps start if a high-high level signal is present). If a sump high-level signal is not present, the operator manually starts a pump.

Drywell floor drain sump containment isolation valves close when a Group II primary containment isolation signal is received and fail closed on a loss of instrument air. As with the drywell equipment drains, the signal not only isolates the drywell but also prevents the pumping of potentially contaminated liquid from the drywell. The same seal-in feature is used as with the equipment drain sumps.

9.3.3.3 Reactor Building Equipment Drains and Tanks

Each unit has one reactor building equipment drain tank (RBEDT) located in the reactor building basement:

A. Unit 2 RBEDT is located in the northwest corner room and

B. Unit 3 RBEDT is located in the northeast corner room

The single pump on each tank starts automatically on a high tank level signal or a high tank temperature signal and trips on a low tank level signal or a low tank temperature signal. Tank level is indicated on an RBEDT level recorder.

A heat exchanger cooled by RBCCW water provides temperature control of the RBEDT fluid by automatically opening a heat exchanger recirculation valve on the discharge of the pump and closing the discharge valve to radwaste when the high temperature setpoint is reached. The temperature switch also starts the pump if it is not running. Once the temperature of the water in the tank is reduced to acceptable level, flow is rerouted to radwaste, unless the pump had been started by the temperature switch, in which case the pump trips. The high-temperature setpoint for the automatic recirculation valve is adjustable in the main control room. The discharge valve fails open on loss of air while the recirculation valve fails closed. The RBEDT pump is interlocked off when a Group II primary containment isolation signal is received.

The reactor building equipment drain liquid is pumped to the waste collector tank through a common line with the drywell equipment drains. A run time meter is provided to monitor the flow to the collector tank. A collector tank high-temperature alarm is provided in the main control room.

9.3.3.4 Reactor Building Floor Drains and Sumps

Each unit has two reactor building floor drain (RBFD) sumps with two pumps in each sump. Both sumps are located in the reactor building basement with RBFD sump A being on the east side and RBFD sump B on the west.

The reactor building floor drain liquid is pumped to the floor drain collector tank through a common line with the drywell floor drains. A run time meter is provided to monitor the flow to the floor drain collector tank. In each sump, the pump starts on a high sump level signal and trips on a low sump level signal. The pumps also trip when a Group II primary containment isolation signal is received. A high sump level alarm is provided in the main control room.

The sump pumps from the southeast and southwest (low pressure core injection [LPCI] and core spray) corner room sumps discharge sump fluid to the RBFD sumps. Individual sump high-level alarms annunciate on a control room panel.

9.3.3.5 Turbine Building Equipment Drains and Sumps

Each unit has one equipment drain sump with two pumps located in the condensate pump pits (elevation 469'-6"). Sump pumps are started automatically by a mechanical alternator level switch. One sump pump starts on a sump high-

level signal and the second pump starts on a sump high-high level signal. A high-high sump level alarm is provided in the main control room. Both pumps trip on sump low level. Run time indication is provided in the radwaste control room.

The Unit 2 and Unit 3 turbine building equipment drain sumps have a 2 inch normal pipe size transfer line which permits their contents to be pumped into the adjacent floor drain sump pit.

9.3.3.6 Turbine Building Floor Drains and Sumps

Each unit has one floor drain sump with two pumps located in the condensate pump pit. Sump pumps are started automatically by a mechanical alternator level switch. One pump starts on a high sump level signal and the second pump starts on a high-high sump level signal. A high-high sump level alarm is provided in the main control room. Both pumps trip on sump low level. Run time indication is provided in the radwaste control room.

9.3.3.7 Radwaste Floor Drains and Sumps

Two sumps, each with one pump, are located in the radwaste basement near the west wall. Each pump is started automatically by a level float switch on a high sump level signal and trips on a low sump level signal. Run time indication is provided in the radwaste control room. A high sump level alarm is also provided in the radwaste control room.

9.3.3.8 High-Pressure Coolant Injection Room Floor Drains and Sumps

Each unit has one sump and sump pump located in their respective high-pressure coolant injection (HPCI) rooms. Each sump pump is started automatically by a float switch on a high-level signal and the pump trips on a low-level signal. Flow is directed to the reactor building floor drain sump. A high sump level alarm is also provided in the main control room.

9.3.4 Chemical and Volume Control System

This section is not applicable to Dresden Station.

9.3.5 Standby Liquid Control System

9.3.5.1 Design Bases

The standby liquid control (SBLC) system fulfills two performance objectives.

The first performance objective of the standby liquid control (SBLC) system is to bring the reactor to a shutdown condition at any time during core life independent of control rod system capabilities. To accomplish this objective the SBLC system is designed to inject sufficient sodium pentaborate solution into the reactor to shutdown from full power to a subcritical condition, with a reactor core boron concentration of 600 ppm.

The quantity of liquid control is determined by the negative reactivity required to render and maintain the reactor subcritical with the control rods withdrawn to their full power position. Allowance for nonuniform mixing of the liquid poison injected into the reactor coolant has been provided.

The design of the SBLC system assumes certain conditions as the bases. It is assumed that the reactor is operating at maximum power, at xenon equilibrium with the control rods in a normal operation pattern. Further, it is assumed that the operator is unable to insert control rods either by scram or by the normal mode of insertion. The SBLC design results in the reactor being subcritical by at least 1% Δk (k_{eff} less than or equal to 0.99) at the most reactive condition during the cycle. The analysis is based on a 160°C xenon-free core with the control rods in the configuration defined below and utilizes a boron concentration equivalent to 600 ppm at 68°F. The analysis assures existence of a shutdown margin of 1% Δk, which is adequate to account for uncertainties.

Control rod drive malfunctions sufficiently severe to prevent insertion of a single rod are highly unlikely, and the coincidental occurrence of such malfunctions in all fully or partially withdrawn drives is more unlikely. Therefore, the assumption that no control rods can be inserted is extremely conservative, as is the design shutdown margin imposed. In view of this inherent conservatism, an additional assumption of peak xenon conditions is not warranted.

The maximum licensed reactor power for Units 2 and 3 is 2957 MWt. Raising power to the average power range monitor (APRM) rod block line at 100% flow would be in violation of the operating license; therefore, operation at the rod block line would not occur. The present analysis is valid whether or not the local peak power approaches rod block monitor (RBM) limits. Only large changes in total initial power would materially affect the boron concentration determinations.

The basis assumed for the reactor water level is the normal operating level. This water level is assumed because there is no loss of feedwater or vessel level control during insertion of the liquid control solution.

An additional operational criterion imposed by 10 CFR50.62 (Reference 9.3.6.7) requires the system to deliver 86 gal/min of 13% (minimum) sodium pentaborate solution containing natural B-10 or equivalent. In order to meet this requirement, Dresden Units 2 and 3 have modified the content in the SBLC tanks to utilize sodium pentaborate solution enriched to ≥45 atomic percent B-10. At 14wt % (minimum) solution, a system flow rate (using a single pump) of 40 gal/min provides the reactivity control that is equivalent to 86 gal/min of 13% solution.

With the use of enriched sodium pentaborate solution, the amount of enriched B-10 solution that can be injected into the reactor vessel by the SBLC system has an equivalent natural boron concentration of 1398 ppm. The system design provides for an additional margin of 25% boron to compensate for possible losses and imperfect mixing of the chemical in the reactor water. This results in an average concentration of 1747 ppm of boron in the reactor core. The cycle specific fuel reload credits a boron concentration of 918 ppm in the shutdown margin analysis.

In order to provide the shutdown requirements indicated above, a gross volume of 3391 gallons of 14% sodium pentaborate solution at a temperature of 100°F is required. This volume includes an additional volume of solution contained below the pump suction that is not available for injection. Other equivalent combinations of increased concentration and reduced volume have been evaluated for temperature and net positive suction head requirements. See Table 9.3-3.

The second objective is to maintain the pH of the suppression pool at a value greater than 7 in the event of a design basis LOCA. If a DBA LOCA occurred, the contents of the SBLC system tanks are injected into the suppression pool for pH control which then ensures that the particulate iodine deposited into the pool during a DBA LOCA does not re-evolve and become airborne as elemental iodine. This role of the SBLC system is described in UFSAR section 15.6.5.5.

9.3.5.2 System Description

The equipment for the SBLC system is located in the reactor building and consists of an unpressurized tank for low-temperature sodium pentaborate solution storage; a storage tank heater; two positive displacement pumps; two explosion-actuated

shear plug valves; the poison sparger ring; and the necessary piping, including pump suction piping heat tracing, valves, and instrumentation. This system does not require external cooling or power for such cooling. A diagram of the standby liquid control system is shown on Figure 9.3-8. Table 9.3-3 summarizes the principal design parameters.

The SBLC tank is equipped with a top cover, vent, and drain. Redundant SBLC system pump suction lines are arranged and constructed to minimize entry of particulate material which might settle on the tank bottom. Heaters are provided to heat the water during initial mixing and to maintain temperature as required during normal operation. The tank has a capacity of 5250 gallons. The neutron absorber solution used is a 14% percent solution, with a saturation temperature of 62°F. As required by Technical Specifications, the solution temperature is maintained at least 20 degrees F above this saturation temperature as added margin against boron precipitation. The solution storage tank is heated by an immersion heater and the pump suction piping is heat traced. The ambient temperature of the solution is maintained below 110°F to ensure adequate NPSH. Temperature and liquid level alarms for the system are annunciated in the control room.

The sodium pentaborate solution is delivered to the reactor by either one or both of two 40-50 gal/min, 1500-psi for Unit 2 and 1765 psi for Unit 3, positive displacement stainless steel pumps. The pumps and piping are protected from overpressure by two relief valves set at approximately 1500 psig for Unit 2 and 1586 psig for Unit 3, which discharge back to the SBLC tank.

The explosion-actuated valves are double squib-actuated shear plug valves. A low-electrical monitoring system gives visible (pilot light) indication of circuit continuity through both firing squibs in each valve.

The two explosion-actuated injection valves provide high assurance of opening when needed and ensure that boron does not leak into the reactor even when the pumps are being tested. Each explosion-actuated valve is closed by a plug in the inlet chamber. The plug is circumscribed with a deep groove so the end is readily sheared off by the valve plunger, opening the inlet hole through the plug. The sheared end is pushed out of the way by the plunger; it is shaped so it does not block the ports after release. The shearing plunger is actuated by an explosive charge, with dual ignition primers, inserted in the side chamber of the valve.

The operator operates a five position key switch if it is determined that neutron absorber solution should be injected into the reactor. A turn from the neutral to the first position (single pump position) in either direction starts one pump and opens one of the two parallel valves for use in terminating an anticipated transient without scram (ATWS) event. (See Sections 7.8 and 15.8 for additional description of ATWS). Turning the switch to the second position on either side of neutral (two pump position) starts both pumps and opens both valves for use in alternate water injection for vessel level control. Use of the key switch minimizes the probability of an accidental injection of the neutron absorber solution.

A red light beside the keylock switch illuminates when liquid is flowing through an orifice flow switch downstream of the explosion-actuated valves. Cross-piping and check valves assure a flow path through either pump and either explosion-actuated valve. Either pump will start even though its local switch at the pump is in the "stop" position.

A test tank and demineralized water supply are an integral part of the system to facilitate system testing and flushing. All tanks and piping in the system are designed in accordance with ASME codes.

9.3.5.3 Safety Evaluation

Reactor Shut Down. The reactivity requirements for the SBLC system are sufficient to shut down the reactor from full power in the absence of any control rod motion. The resulting reactivity in the shutdown condition is K_{eff} less than or equal to 0.99. The shutdown requirement can be achieved with 600 ppm boron in the moderator. With the use of enriched B–10 sodium pentaborate solution, the available boron concentration that can be injected into the reactor vessel is equivalent to 1398 ppm (after discounting the 25% margin to compensate for loss and imperfect mixing). The cycle specific fuel reload credits a boron concentration of 918 ppm in the shutdown margin analysis. To achieve a core average concentration of 1398 ppm with normal moderator level in the reactor vessel, 3391 gallons of a 14.0 weight % sodium pentaborate solution (enriched to ≥45 atomic percent B-10), or equivalent, is required in the SBLC tank.

Using the single pump injection rate, the SBLC system is capable of reducing power at a rate of 1% per minute; thus the time required to reach the zero power or hot shutdown condition from full rated power is 80 to 100 minutes. The maximum xenon decay and burnup causes a reactivity change equivalent to a power increase of only 0.5% per minute. Therefore, the rate of boron addition causes a steady, constant rate power decrease, even if actuated during the maximum xenon decay and burnup removal transient.

The rate of power change with time is expected to be constant until the power is too low to produce boiling in the core. When the reactor becomes subcritical, the injection rate is sufficient to maintain the reactor subcritical indefinitely.

Stability considerations impose an upper limit on the rate of solution injection. Power oscillations, resulting from boron concentration oscillations, would be possible if the time for solution injection were equivalent to the recirculation loop transit time. Because the loop transit time is on the order 10 to 15 seconds, injection of sufficient solution to reduce power to zero over an interval of 100 minutes is slow enough to assure that the poison concentration in the core does not oscillate but increases monotonically.

In the event that the SBLC system interlock fails to isolate the reactor water cleanup system, the cleanup system would start to remove the boron from the core. This removal rate would be extremely small because of the flow path of the boron; the boron is inserted into the bottom of the vessel, moves up through the core, then to the outside of the core shroud. If the recirculation pumps are still in operation, one-third of the flow outside the core shroud is taken out through the recirculation loop; the remaining two-thirds are inserted back into the bottom of the vessel. Of the one-third taken out through the recirculation loop, only 2% is diverted to the cleanup system. Therefore, of the amount of boron which gets outside the core shroud, only about 0.7% is removed by the cleanup system.

A failure of the reactor water cleanup system to isolate would be observed in the control room and remote manual isolation of the cleanup system would be completed. Conservative SBLC system design assumes failure of the operator to isolate the cleanup system. This type of failure could result in a cleanup system

removal of 4.5% of reactor coolant boron before the demineralizers became saturated to the extent that no more boron could be removed. This small amount is easily accounted for by the 25% nonuniform mixing factor present in the original amount of boron solution.

The SBLC system is not designed to respond at a rate which would shut down the reactor during a transient situation. The simultaneous failure of 177 individual control rod assemblies and the loss of the main condenser vacuum would cause a power transient which is beyond the design capabilities of the nuclear system.

In the event that the reactor was critical with the reactor open and flooded to the normal water level during refueling, as is possible during low-power tests (power level less than 0.5 MWt) and insertion of the control rods was found to be impossible, the SBLC system would be started. The additional volume of water would diminish the boron concentration to 250 ppm which would produce a k_{eff} less than 0.95 in the cold, xenon-free core.

Since the SBLC system is to be operable in the event of an auxiliary power failure, it can be powered from the diesel generator. When fuel is in the reactor, the SBLC system is operable in Mode 1, 2 and 3. At least one pump and one explosion-actuated shear valve must be operable for the system to be considered operable.

The availability of the SBLC system is assured by the continual monitoring of the system key parameters, the system's testability, and the two independent sets of active components (pumps and explosion-actuated injection valves and their actuation circuits) provided to inject boron into the system.

Post accident Suppression pool pH. The ability of the SBLC system to maintain the suppression pool pH at a value above 7 in the event of a design basis LOCA is discussed in UFSAR Section 15.6.5.5.

9.3.5.4 Tests and Inspections

The system must be tested periodically to establish the operability of all components. To avoid introducing boron to the reactor, the test is done in two parts. With the injection valves closed, each pump may be started locally and the solution may be pumped from the storage tank and returned to the tank. This demonstrates the ability of the pumps to remove solution from the tank at the required flowrate. By valving out the liquid poison supply and valving in demineralized water, the system can be flushed to prevent any boron precipitation in the pumps and lines.

The system can be tested for complete continuity during a shutdown when demineralized water can be pumped from the test tank into the reactor vessel. Testing necessitates replacement of the explosive charges in the shear plug valves.

The SBLC tank solution temperature is monitored periodically to ensure that boron (in the form of sodium pentaborate) does not precipitate out of solution and occlude pump supply lines. Ensuring adequate solution temperature also verifies the operability of tank heaters.

The containment isolation valves, two check valves located in series near the drywell penetration, are tested in accordance with the IST Program.

Should the SBLC system ever be used to shut down the reactor, the sodium pentaborate would be removed from the primary system by flushing for gross dilution and by operation of the reactor water cleanup system for final polishing.

Boron concentration of the solution is periodically determined by chemical analysis.

A quarterly inservice test (IST) is conducted to verify the operational readiness of the SBLC pumps. It also verifies operability of the SBLC pump discharge check valves.

Quarterly testing is performed on the SBLC pumps to identify any potential operational degradation.

9.3.5.5 Instrumentation Requirements

The SBLC system consists of boron injection pumps, explosion-actuated injection valves, and related piping and instrumentation. The instrumentation and controls for the SBLC system are designed to:

A. Actuate the injection pumps,

B. Open the boron injection line into the vessel,

C. Maintain the boron solution above its saturation temperature,

D. Function so that the system is testable during operation,

E. Provide indication to the operator of system operation and operations status, and

F. Isolate reactor water cleanup system.

The SBLC system is actuated by a five-position keylock switch on the control room console. This assures that switching from the "off" position is a deliberate act. Switching to either side starts that injection pump, opens the corresponding explosion-actuated valve, and closes the reactor cleanup system isolation valves to prevent boron dilution. Switching to the far right and left positions on either side of the switch starts both pumps and opens both valves, as well as isolating the reactor cleanup system.

Automatically controlled heaters are provided in the SBLC tank to maintain adequate margin above the boron solution saturation temperature during initial mixing and during normal plant operation.

The SBLC system functional control logic is shown on Figure 9.3-9.

Indicators are provided in the control room to verify the operability of the system and to verify operation if the system should ever be used.

Each explosion-actuated injection valve's ignition circuit continuity is monitored by a trickle current, and an alarm occurs in the control room if either circuit opens. Indicator lights show which primer circuit opened. A green light at the local start switch in the plant indicates that power is available to the pump motor contactor. A red light indicates the contactor is closed (pump running). When an SBLC pump is started, a red light in the control room will indicate the pump motor contactor is energized. There is a red light for each pump. Also in the control room, a red light will energize indicating flow through the pipes. A level switch is provided in the SBLC tank which actuates an alarm in the control room on high or low liquid level. Also, a temperature sensor in the SBLC tank actuates an alarm in the control room on high or low liquid temperature. A pressure indicator downstream from the check valves from each pump indicates the pump pressure during testing or actual system operation.

9.3.6 References

1. M12-2(3)-84-119 and Onsite Reviews Nos. 86-26 and 87-24.

2. Letter to L. O. DelGeorge (CECo) from J. D. Neighbors (NRC), July 22, 1982, "Post Accident Sampling System NUREG-0737, II.B.3 Evaluation Criteria Guidelines"; Regulatory Guide 1.97, Revision 2, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," U.S. Nuclear Regulatory Commission (NRC), Office of Standards Development, December 1980.

3. "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," Regulatory Guide 1.97, Revision 2, U. S. Nuclear Regulatory Commission (NRC), Office of Standards Development, December 1980.

4. "Design Specification for High Radiation Liquid and Gas Sampling System for Normal and Post-Accident Operations - Model A," June 1981, Sentry Equipment Corp. (SEC); "System Design Descriptions for Commonwealth Edison Company Dresden Nuclear Station Units 2 and 3," NUS Corp., March 10, 1981, Rev. 0, Document No. 5308-SDD.

5. 10CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix A, "General Design Criteria 19 for Nuclear Power Plants," USNRC, May, 1977, Wash. D.C.

6. NUREG-1228, "Source Term Estimation During Incident Response to Severe Nuclear Power Plant Accidents," United States Nuclear Regulatory Commission, October, 1988.

7. M12-2(3)-84-119 and Onsite Reviews Nos. 86-26 and 87-24.

8. NED-M-MSD-2, “Revised Sodium Pentaborate Requirements for the SLCS, Dresden 2 & 3, Quad Cities 1 & 2.”

9. DRE05-0009 Rev 01, “Shutdown Boron Capability for the Standby Liquid Control Systems.”

Table 9.3-1

ESTIMATED INTEGRATED DOSE PER SAMPLE

Reactor Coolant Sample

Activity	Time (min)	Dose Rate (mrem/hr)	Dose (mrem)
Assemble in vestibule	10	2.5	0.42
Perform valve lineup at control panel	10	2.5	0.42
Perform manipulations on the LSP	35	100-400	58.33 - 233.33
Withdraw shielded cart to vestibule for transport to hot lab	5	100	8.33
Total	60		67.50 - 242.50

It is estimated that the operator will receive a maximum dose of 500 mrem during the 30-minute time period spent outside the HRSS building in getting to and from the plant or obtaining a grab sample in the shielded cart.

Containment Air Sample

	Time (min)	Dose Rate (mrem/hr)	Dose (mrem)
Assemble in vestibule	10	2.5	0.42
Perform valve lineup, initiate auto sequencer	10	2.5	0.42
Capture sample	8	2.5	0.33
Wait for sequencer to complete panel purge	7	2.5	0.29
Withdraw sample cartridges(1)	1	1000	16.67
Prepare partitioner for purge	0.5	1000	8.33
Purge partitioner	7	2.5	0.29
Transport sample in shielded cart to vestibule	1	15	0.25
Total	44.5		27.00

Note:
1. During this activity, the dose to operator's hands is estimated at 1.22 rem.
2. Dose rates presented are based on conservative assumptions and remain valid for uprate to 2957MWt.

Table 9.3-2

HRSS EQUIPMENT PARAMETERS

Liquid Sample Panel	
Quantity	1 per reactor unit
Sample Inputs	
Reactor coolant module	5
Demineralizer module	3
Radwaste module	5
Design Pressure/Temperature	
Reactor coolant module	2300 psig/120°F
Demineralizer module	1250 psig/120°F
Radwaste module	150 psig/120°F
Materials	
Tubing	1/4-inch, Type 304 stainless steel
Shielding	
Panel	7 inches of 0.09-inch diameter lead shot
Base	5 inches of 0.09-inch diameter lead shot
Panel walls	1/2-inch steel plates (2)
Dimensions	
Height	7 feet
Depth	4 feet
Width	8 feet
Ventilation	360 ft^3/min through panel

Table 9.3-2 (Continued)

HRSS EQUIPMENT PARAMETERS

<u>Chemical Analysis Panel (*)</u>: equipment is nonfunctional and not maintained

Quantity		1 per reactor unit
Instrument		Range
*	Hydrogen gas chromatograph	10 to 2000 cc/kg
*	Chloride ion chromatograph	0 to 20 ppm
*	Boron ion chromatograph	0 to 1000 ppm
*	Dissolved oxygen	0 to 20 ppm
	pH probe	pH 1 to 13
	Conductivity probe	0.1 to 500 μmho/cm
Material		
	Tubing	1/4-inch or 1/8-inch Type 304 stainless steel
Shielding		
	Panel	7 inches of 0.09-inch diameter lead shot
	Base	5 inches of 0.09-inch diameter lead shot
	Panel walls	1/2-inches steel plates (2)
Dimensions		
	Height	7 feet
	Depth	3 feet
	Width	4 feet
Ventilation		100 ft^3/min through panel

Table 9.3-2 (Continued)

HRSS EQUIPMENT PARAMETERS

Containment Air Sample Panel	
Quantity	1 per reactor unit
Materials	
Tubing	1/4-inch Type 304 SS
Shielding	
Front panel	3-inch thick steel plate
Dimensions	
Height	7 feet
Depth	2 feet
Width	3 feet
Ventilation	300 ft^3/min through panel

Table 9.3-3

STANDBY LIQUID CONTROL SYSTEM PRINCIPAL DESIGN PARAMETERS

System	
Design negative reactivity (% Δk/k)	3.0
Available reactor boron concentration	1747 ppm
Poison injection rate per pump	40 gal/min
Poison compound	$Na_2B_{10}O_{16} \cdot 10H_2O$
Standby liquid control tank capacity	5393 gal
Pumps	
Type of pump (postitive displacemnt)	Triplex plunger
Number	2 (two required)
Normal Operating Conditions (each pump)	
Capacity	40 gal/min
Total developed head	1275 - 1552 psi
Suction pressure	Atmospheric
Pumping temperature	70°F
Available net positive suction head	25 feet
Type of drive	Electric motor
Rating	50 hp
Power Sources	
Pumping and control	Unit Auxiliary Transformer 21 (31), Reserve Auxiliary Transformer 22 (32) or diesel-generator [MCC 28-1(38-1) for 2A(3A) standby liquid control pump; MCC 29-1(39-1) for 2B(3B) standby liquid control pump]

Table 9.3-3 (continued)

STANDBY LIQUID CONTROL SYSTEM PRINCIPAL DESIGN PARAMETERS

Standby Liquid Control Tank Volume Requirements (gallons):

Temperature	14.0% Solution	15.4% Solution	16.5% Solution
110	3391	3097	2901
120	3830	3532	3333
130	4388	4086	3883

9.4 AIR CONDITIONING, HEATING, COOLING, AND VENTILATION SYSTEMS

The design objectives of the station heating, ventilation, and air conditioning (HVAC) systems are to provide protection to plant personnel and equipment from extreme thermal environmental conditions and to provide personnel protection from airborne radioactive contaminants. To achieve these objectives, the following design bases were used:

A. The design of equipment and components complies with the requirements of the applicable codes and standards of the major regulatory organizations, such as:

   1. Underwriters Laboratory;

   2. National Bureau of Standards; and

   3. American Society of Heating, Refrigerating and Air Conditioning Engineers.

B. A minimum of 1/4-in.H_2O pressure differential is maintained at all times during plant operation between those areas which are clean and normally accessible and those areas designated as having the greatest contamination potential. This requirement does not necessitate "cascading" of several successive differential pressures of this magnitude. In many cases flow control can be maintained satisfactorily by individual differentials of considerably less than 1/4-in.H_2O. The HVAC systems are designed to maintain differentials smaller than 1/4-in.H_2O between adjacent areas where several steps in series are present before the area of greatest contamination potential is reached. Some systems (as identified in that particular section) simply maintain airflow from low to higher areas of potential contamination.

Normal plant HVAC equipment will be in continuous operation and will not need periodic testing.

It is normal practice to flow air from points of less contamination to points of greater contamination. This practice has been used throughout the design of the ventilation system. In some cases proper air flow direction cannot be maintained. Regardless of air flow direction, station radiological programs (e.g., radiological surveys, plant decontamination) are in place to monitor and limit the spread of contamination. Radioactive airborne activity is monitored continuously. Redundancy of equipment and recorders are used for reliable sampling and to provide a record of the reactor building ventilation stack release. The fuel pool area is also monitored to allow for trend indication and a permanent record of condition. The area radiation monitoring system is described in Section 12.3.

9.4.1 Control Room Area Ventilation System

The control room HVAC system provides conditioned air for personnel comfort, safety, and equipment reliability. In addition, the control room ventilation system provides habitability during toxic gas or radioactive gas releases, as described in Section 6.4. This section discusses operation of the control room HVAC system under non-emergency conditions. The control room HVAC system is shown in

Drawings M-273, Sheets 1 and 2, and M3121).

9.4.1.1 Design Bases

The control room ventilation system is designed to:

A. Maintain the control room between 70°F and 80°F with outside temperatures varying between -6°F and +93°F;

B. Provide adequate radiation protection to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of the General Design Criteria (GDC) 19 or Standard Review Plan (SRP) 6.4 limits;

C. Provide protection from toxic gas release;

D. Provide protection from fire and smoke; and

E. Provide HVAC to the control room emergency zone as described in Section 6.4.2.1.

9.4.1.2 System Description

The control room HVAC system consists of two independent HVAC subsystems sharing some common ductwork: one multizone subsystem (Train A) and one single zone subsystem (Train B). Train A is the primary temperature control and air distribution subsystem for the control room emergency zone. Train B is a backup system which serves the control room emergency zone when Train A is not available. Train B is described in Section 6.4.

The Train A HVAC system provides individual zone temperature control by blending air from the hot and cold decks of the air handling unit (AHU) to satisfy the individual room thermostats. During cooling operation, the air is cooled and dehumidified by means of a direct expansion coil within the AHU. During heating operation, air is passed over a hot water coil within the AHU. A steam humidifier is provided in the discharge duct to the control room and auxiliary computer room to maintain the rooms at a minimum of 40% relative humidity. The system supplies the air to three separate zones as follows:

A. Zone 1 - This zone was permanently isolated from the control room emergency zones,

B. Zone 2 - control room, and

C. Zone 3 - Train B HVAC equipment room.

The Train A air distribution system is divided into three separate ducting arrangements, each serving one of the areas indicated above. Zone 1 has been permanently isolated and does not serve any area. The system is sized to dissipate both the internal and external heat loads within each space, and the design of 75°F is based on maintaining personnel comfort. Return air from each zone is collected and routed back through a common duct system via a vane-axial

return air fan. The Train A unit can be operated in the event of a loss of offsite power (LOOP) if the 4-kV nonessential switchgear is backfed.

During normal operation of the Train A HVAC system, the control room emergency zone is maintained at a positive pressure by a maximum inlet flow of 2000 ft^3/min through the inlet damper. The exhaust damper remains closed during normal operation.

During normal operation of the Train B HVAC system, the control room emergency zone is maintained at a positive pressure. Outdoor air is introduced at the rate of 2000 ft^3/min into the control room emergency zone to maintain the positive pressure. Emergency operation of Train B requires operator action as described in Section 6.4.4.1.

9.4.1.3 Safety Evaluation

Section 6.4 contains an evaluation of the control room HVAC system.

9.4.1.4 Inspection and Testing Requirements

Section 6.4 describes inspection and testing of the control room HVAC system.

9.4.2 Spent Fuel Pool Area Ventilation System

The spent fuel pool area ventilation system is an integral part of the reactor building ventilation system described in Section 9.4.5. Filtered, tempered air is supplied to the refueling floor, which includes the spent fuel storage pool and the dryer and separator storage pool areas. Exhaust air flow from the operating floor is regulated by a series of manually operated dampers. The exhaust ducts carry the effluent directly to the reactor building vent stack or, if secondary containment is isolated, to the standby gas treatment system (SBGTS).

9.4.3 Radwaste Facility Ventilation System

The radwaste area ventilation system is comprised of subsystems which service the following areas:

A. Radwaste building;

B. Radwaste control room;

C. Radwaste solidification building; and

D. Maximum recycle radwaste building.

These are described in the following subsections.

9.4.3.1 Radwaste Building Ventilation

The radwaste building ventilation system has been designed to maintain the following conditions with outside temperatures varying between -6°F and +93°F:

A. Occupied areas	Minimum 50°F Maximum 103°F
B. Cells and collector tank room	Minimum 50°F Maximum 120°F
C. Concentrator and concentrator waste tank cells	Minimum 50°F Maximum 150°F

The radwaste building HVAC system is shown in Drawing M-272.

The outside air for the radwaste building system is drawn into the building through stationary louvers, filtered, heated as necessary, and distributed by supply air ducts and registers throughout the building. Normally the airflow sequence starts from the clean areas and moves progressively to the areas of greater potential contamination. The system includes three 50% capacity supply fans and three 50% capacity exhaust fans. Normally, the system operates on two exhaust fans and either one or two supply fans as needed to maintain adequate differential pressures between the general area relative to outside and contaminated areas relative to clean areas. The standby supply and exhaust fans will start automatically on low airflow or auto trip of a running fan. The system is balanced to maintain a slight negative pressure in the general areas of the building, with certain areas of highest contamination potential maintained at 1/4-in.H_2O negative pressure. All supply fans trip on building overpressure, and all exhaust fans trip on excessive building vacuum.

The system exhausts air out of the building to the 310-foot chimney. The exhaust fans draw building air into the exhaust vents located throughout the building and through two parallel filter trains, each composed of a prefilter followed by a high efficiency particulate air (HEPA) filter before discharging the air to the 310-foot chimney. Each supply and discharge fan is equipped with backdraft dampers that open when the fan is in operation.

9.4.3.2 Radwaste Control Room HVAC

The radwaste control room is cooled by a dedicated refrigeration unit which rejects heat to the outside atmosphere. The unit is equipped with an electric heater coil for use in cold weather. Control room air is recirculated through a filter and supply fan, and air lost by exfiltration is made up by outside air. The radwaste control room HVAC system is shown in Drawing M-760.

9.4.3.3 Radwaste Solidification Building HVAC

A fan in each of the two trains of the radwaste solidification building ventilation system draws outside air into the building through a rough prefilter, a heat

recovery heat exchanger, two trains of supply filters and electric heaters, and discharges to the truck bay, and the high and low level drum storage areas. Air infiltration to the truck bay is expected. The ventilation system circulates the air through the drum storage and processing areas of the building, and discharges the exhaust air through two trains of exhaust fans, prefilters, and HEPA filters before being directed to the heat recovery heat exchanger and the 310-foot chimney. A slight negative pressure is maintained in the solidification building by modulating the exhaust fan dampers.

A separate air handling unit in the solidification building control room recirculates air through a filter, twin cooling coils, and an electric heating coil. Air lost by exfiltration is made up by outside air. The HVAC condensing unit is located on the roof of the control room.

In addition, the HVAC equipment room has its own dedicated air conditioning unit.

Electric power is supplied by nonessential motor control centers. Each flow path of the air supply and exhaust systems is equipped with two fail-closed dampers, one on each end of each flow path. In addition, isolation dampers on the inlet and discharge of each flow path are employed to isolate the supply and exhaust systems when not in operation.

The radwaste solidification building ventilation system is shown in Drawings M-850, M-851, and M-852.

9.4.3.4 Radwaste Maximum Recycle Building Ventilation

The maximum recycle building ventilation system draws outside air by a single supply fan through a filter, heat recovery heat exchanger, and heating and cooling coils. The air is ducted through the building from area to area in the direction of increasing radioactivity potential. Air is discharged from the building through two redundant flow paths, each including a flow damper, prefilter, HEPA filter, exhaust fan, and butterfly isolation damper. This exhaust fan system discharges the air to the 310-foot chimney through the heat recovery heat exchanger. A slight negative pressure is maintained in the building.

Isolation in this system is primarily isolation of the ventilation from the 310-foot chimney by dampers on each of the redundant exhaust fan outlets.

The maximum recycle building ventilation system is shown in Drawing M-760.

9.4.4 Turbine Building Area Ventilation System

The turbine building ventilation system has been designed to maintain area temperature between 65°F and 120°F with outside temperatures varying between -6°F and +93°F.

The turbine building ventilation systems are made up of the main turbine room ventilation system, the reactor feedwater pump ventilation system, the

east turbine room ventilation system, the off-gas recombiner room ventilation system, the auxiliary electrical equipment room/auxiliary computer room cooling system, and the battery room ventilation system. These are separate systems with separate intake and exhaust points. Only the main turbine room system and off-gas recombiner room ventilation systems are exhausted to the 310-foot chimney.

9.4.4.1 Main Turbine Room Ventilation System

The supply air to the main turbine room in each unit is provided by the south turbine room system and the north turbine room system. The two 100%-capacity south turbine room supply fans and three 50%-capacity north turbine room supply fans draw outside air through separate filters, steam heating coils, and evaporative air wash units. This air is then ducted throughout the main turbine area, including the operating floor, the ground floor, and mezzanine areas. The Unit 3 south turbine room system also provides ventilation to the Unit 3 battery charger room. Pneumatically operated backdraft dampers isolate fans not in use. A control systems, using differential pressure sensors, operate pneumatically actuated flow control dampers on the exhaust system to maintain the turbine operating floor at a slight negative pressure relative to the atmosphere. The exhaust fans draw air directly from the low pressure heater bays, moisture separator area, and steam jet air ejector rooms to keep these LHRAs with potential contamination at a negative pressure in relation to the building. There are fans (Condensate Pump Floor Exhaust and HP Heater Bay Supply Fans) that discharge air directly into the LHRAs but these are not operated unless the Turbine Building Ventilation Exhaust Fan(s) are in operation. There are area dp control dampers which throttle to ensure there is adequate dp between these areas and the radiologically clean areas in the building.

In the exhaust system, there are three 50% capacity fans, two operating and one on standby, which draw air from the feedwater heater, moisture separator, and steam jet air ejector areas and discharge to the 310-foot chimney. The system has the capability to operate with a reduced number of supply and exhaust fans in response to cold weather, support of maintenance, and differential pressure requirements. Operating at power with no exhaust fans for a specific unit is highly undesirable and these allowances in lineups ensures minimal time with no ventilation in service.

Exhaust air from the clean and dirty oil storage tank room is discharged to the atmosphere. This ductwork is routed through the abandoned U2 RRMG set ductwork and out the abandoned U2 RRMG set exhaust gooseneck.

The evaporative air wash system has proven ineffective in this climate for helping maintain acceptable building temperatures without creating undesirable/unacceptable humidity levels in the building and are no longer used.

The main turbine room ventilation systems are shown in Drawings M-270 and M-530, Sheets 1 and 2.

9.4.4.2 Reactor Feedwater Pump Ventilation System

The reactor feedwater pump ventilation system removes heat generated by the reactor feedwater pump motors. This is a separate ventilation system housed in the turbine building. Two 100% capacity ventilation fans draw outside air through an air filter and duct it to the three reactor feedwater pump motors. A recirculation air duct is provided. Temperature-controlled dampers are provided to allow the air to be recirculated back to the inlet of the supply fans or replaced with up to 100% outside air. One of the ventilation fans must be operating before a reactor feedwater pump motor can be started. Exhaust air is discharged directly to the atmosphere separately from the main turbine room system.

This system is not essential for safe shutdown, and upon loss of offsite power, the ventilation system will shut down. However, should it be necessary, the system can be connected to the emergency diesel generator power by operator action.

The reactor feedwater pump room ventilation system is shown in Drawing M-270 and M-530, Sheets 1 and 2.

9.4.4.3 Deleted

9.4.4.4 East Turbine Building Ventilation System

The east turbine building ventilation system draws outside air into the building through fixed louvers. This air is filtered, heated as necessary, and distributed through plenums and ducts by three 50% capacity supply and exhaust fans. The system can meet all heating needs with a single supply and exhaust fan. The system can meet cooling needs with a single supply and exhaust fan in all but the most extreme hot weather. As a result, it is allowable to operate the system with a reduced fan line-up.

Air is directed to the north and south HVAC equipment rooms, switchgear rooms, Unit 2 battery rooms, Unit 2 battery panel room, mask cleaning room, Unit 2 turbine building access hallway. The supply and exhaust systems are balanced to provide a slight positive differential pressure relative to the atmosphere with exhaust fan flow control to maintain pressure if required. Depending on the temperature of the return air, it is either recirculated or fully exhausted to the atmosphere. The system can be used to ventilate or purge the auxiliary electric equipment room. .

The east turbine building ventilation system is shown in Drawing M-936.

9.4.4.5 Battery Room Ventilation System

The Unit 2 battery room, located above the main control room, is served by the east turbine room ventilation system. Air is ducted into the battery room and exhausted to the north HVAC equipment room through an opening in the wall.

The Unit 3 battery and battery charger rooms are located in the southwest corner at the mezzanine level. Air is ducted into the charger room by the south turbine building ventilation system and discharged into the turbine room by an exhaust fan. The battery room has its own dedicated air conditioning system to maintain the room temperature between 70°F and 85°F for reliable battery operation. Air is discharged into the turbine room by a fan installed in an opening in the wall.

Following a loss of offsite power, ventilation to the battery rooms will be lost. Hydrogen gas emitted from the batteries as they are recharged by the diesel generators will accumulate in the rooms, but the hydrogen concentration will not reach the lower flammability limit.

The battery room ventilation systems for both units are shown in Drawing M-973.

9.4.4.6 Off-Gas Recombiner Rooms

The off-gas recombiner area ventilation system for each unit draws outside air into the system through stationary louvers. This air is filtered, heated as necessary, and discharged to the upper level operating aisle ("penthouse"). The system consists of two parallel supply and exhaust trains, one supply and exhaust train in operation and one in standby. The system has controls to trip the operating supply fan if the coil air outlet approaches freezing. The system can draw air from the turbine building which is distributed at each elevation at the stairwell opening. The use of a supply fan is only required during hot weather. Operation without a supply fan greatly improves dp to outside for the Off-Gas Recombiner and Turbine Buildings.

The parallel exhaust fans draw from the recombiner rooms, condenser rooms, and the off-gas instrument and control panel. This air is discharged without further filtering to the 310-foot chimney. The system is isolated by dampers in each parallel inlet and exhaust duct, with one set of exhaust dampers normally closed. The off-gas instrument and control panel is ventilated with a separate exhaust blower. Additional heating is supplied by steam area heaters.

The off-gas recombiner room ventilation system is shown in Drawings M-625 and M-633.

9.4.4.7 Auxiliary Electrical Equipment Room and Auxiliary Computer Room Cooling Systems

The AEER and computer room are cooled by the AEER/ACR HVAC. The AEER/ACR HVAC consists of an air handling unit located in the mask cleaning room of the east turbine building and an air cooled condensing unit located outside. In addition to this cooling system, backup ventilation to the AEER is provided by the east turbine building ventilation (See Section 9.4.4.4).

9.4.5 Reactor Building Ventilation System

The reactor building ventilation system is designed to maintain the reactor building area temperatures between 65 °F and 103 °F based on outside temperatures varying between -6 °F and +93 °F.

The reactor building is also divided into distinct zones based on environmental conditions. Refer to Table 3.11-1, Environmental Zone Parameters for Normal Service Conditions.

Special ventilating and ducting systems are used on the refueling floor to continually exhaust air from the spent fuel storage pool area, dryer/separator storage pit, and drywell head cavity. See Section 9.4.2.

The normal ventilation system provides at least one free-volume change of air per hour in the reactor building. Normally the air flows from the filtered supply through ducts to the uncontaminated areas, through potentially contaminated areas, and then is returned and exhausted through the exhaust fans to the reactor building ventilation stack. Cooling and heating units in various rooms of the reactor building provide for personnel comfort and equipment protection.

The reactor building ventilation system is designed to supply filtered, tempered outside air and distribute it through all working areas and equipment rooms in the reactor building. The system maintains a negative pressure of at least 1/4-in.H_2O within the building. The system also maintains a differential pressure of at least 1/4-in.H_2O between clean and potentially contaminated areas. The system is designed to trip automatically on a secondary containment isolation signal, as described in Section 6.2.3.

The Unit 2 and 3 air supply systems tempers the filtered outside air by means of a non-freezing type steam heating coil or by chilled water cooling coils. Cooling water for the chilled water coils is provided by a piping system from air-cooled chiller units located outside the reactor building (M-269, Sheet 3 and M-529, Sheet 3). Tempered air is directed by three 50% capacity fans through a redundant pair of pneumatically operated isolation butterfly valves to the building's duct system. Each of these three 50% capacity supply fans is equipped with a pneumatically actuated backdraft damper to isolate each fan when the fan is not in operation. Power to the fans is supplied by electrical buses which may be connected to the emergency diesel power in the event of an offsite power failure.

The conditioned supply air is then directed to all working areas and equipment rooms of the reactor building as shown in Drawings M-269 Sheet 2 and M-529, Sheet 2. Flow to and within these areas is indicated and controlled by means of differential pressure indicators and pneumatically operated dampers.

Exhaust air is collected by a network of ducts throughout the reactor building and is returned to the exhaust fan plenum through a pair of pneumatically operated isolation dampers. The air exhaust system consists of a set of three 50% capacity fans which exhaust to the reactor building vent stack. The drywell purge system is also ducted to these exhaust fans so that drywell effluent is exhausted by this system when the drywell is inerted, purged, or opened for access. The three exhaust fans are powered by emergency diesel-powered electric buses. Reactor building infiltration is discussed in Section 6.2.3.

9.4.6 Emergency Core Cooling System Ventilation System

The emergency core cooling system (ECCS) rooms consist of two low pressure coolant injection (LPCI)/core spray pump rooms and one high pressure coolant injection (HPCI) pump room per unit, all of which are served by the reactor building ventilation system. Each of the rooms has a water-cooled heat exchanger/fan unit which functions as a room cooler. The fans for these units are powered by the emergency bus and are capable of operating following a LOOP. The LPCI/core spray rooms are normally maintained at less than 104°F. The HPCI rooms are normally maintained less than 140°F. These temperatures are below the qualification temperatures of the mechanical and electrical components in the rooms that are required for safe shutdown of the plant. Section 3.11 contains a further discussion of equipment qualification.

The LPCI/core spray pump room cooler fan motors are supported from rod hangers in a seismically qualified pendulum fashion from the room ceiling.[1]

During normal plant operating conditions, the cooling water for the ECCS room coolers is provided by the service water system. The service water system is discussed in Section 9.2.1. The containment cooling service water system provides backup cooling water to the ECCS room coolers. The CCSW system is described in section 9.2.1.

If the reactor building ventilation system is shut down due to secondary containment isolation or LOOP, the equipment in the ECCS rooms are cooled solely by the room coolers. Loss of the room coolers is discussed in Section 3.11.4.

9.4.7 Diesel Generator RoomVentilation System

Each diesel generator room (Unit 2, Unit 3, and the 2/3 diesels) has an independent ventilation system. These ventilation systems are shown on P&ID M-974. The outside air temperature range for this system is -6 to 93°F.

A. Diesel Generator Ventilation Units 2 and 3

The Unit 2 and 3 diesel generator rooms have identical ventilation systems. The system components include a ventilation fan, inlet ductwork, pneumatically controlled isolation (for CO^2 isolation from fire detectors) and modulation dampers (outside air and general area air from turbine building are blended), heat activated fire control dampers (fire dampers), failed open system isolation dampers (which also serve a CO^2 isolation function), and a pneumatic damper control system for temperature control of the modulating dampers. The system(s) share an intake penthouse with the U2 and U3 Reactor Feed Pump Ventilation systems. The high connection point to the Reactor Feed Pump Ventilation ensures that warm air being recirculated by the Reactor Feed Pump Ventilation system is not recaptured into the intake for the Diesel Generator Ventilation system. The recirculated component of the air comes from the general vicinity of the Reactor Feed pump Ventilation system rollamatic filter.

During normal plant operation, when the diesel generator is in standby, ventilation for the diesel generator room is provided via a duct (with isolation damper and fire damper) from the South Turbine Building supply system for that unit. This air is exhausted by normal leakage around and under the room doors along with isolation damper leakage. The ventilation system operates when the Diesel Engine is in service but can be operated with the engine secured if required. Once the fan is energized, the ventilation from the south turbine ventilation system is secured, the isolation dampers open, and the temperature control modulating dampers control the temperature in the room.

Air supply to the ventilation control panel for damper actuators and instrumentation is provided via the instrument air system. Nitrogen is available upon a loss of instrument air pressure to supply the ventilation control panel.

In the event of a fire in the diesel generator room, the ventilation fan is tripped, all inlet and exhaust dampers are closed, and carbon dioxide is injected in to the room. In addition, each isolation damper is complimented with a heat activated fire damper.

B. Diesel Generator Ventilation Unit 2/3

The 2/3 diesel generator room ventilation system is similar to the Unit 2 and 3 diesel generator room ventilation systems except for the following:

During normal plant operation, the 2/3 diesel generator room is not provided forced air from another ventilation system since it is located outside of the reactor and turbine buildings.

Intake for the ventilation fan is taken from the outside atmosphere only (no connection to another system) and the exhaust from the room goes directly outside. A modulating damper upstream of the fan is used to maintain temperature control.

There is an override switch provided to allow for overriding CO^2 isolation to allow for ventilation system operation if a fire is not present and fire protection system closed the dampers.

The 2/3 EDG diesel generator ventilation system does not have fire dampers because the ventilation ducts do no penetrate a fire barrier. See FHA VOL I Section 3.7.7 and 4.7.1 for more information.

9.4.8 Drywell Ventilation System

The drywell ventilation system has been designed to maintain the drywell at an average temperature of 135°F during normal operations and an average temperature of 105°F eight hours after shutdown, with outside temperatures varying from -6°F to +93°F.

The drywell ventilation system is shown in Drawing M-273, Sheet 3 and 4.

The drywell ventilation system contains seven air coolers (water-cooled heat exchanger/fan units) which cool the drywell atmosphere to approximately 135°F. The drywell atmosphere is circulated through the coolers and throughout the drywell by fans and ductwork. The reactor building closed cooling water (RBCCW) system provides a source of cooling water to remove heat from the coolers. By maintaining drywell ambient air temperature less than 150°F during normal plant operation, the insulation on motors, isolation valves, operators and sensors, instrument cable, electrical cable and sealants used at the penetrations will have a sustained life without premature degradation.

The seven drywell coolers are checked for leakage under normal cooling water pressure at each major outage. Operation of the fan is also observed at this time. During normal reactor operation, the temperature sensors in the drywell monitor the effectiveness of the coolers.

Provisions have been made to ensure that the drywell ventilation system remains operational in the event of a loss of station ac power. Drywell cooler fans are fed by the emergency power system and are therefore available following a loss of power. The fans will trip on a loss-of-coolant accident (LOCA). Spare cooling fan capability is provided with this system. Drywell temperature monitoring is described in Section 6.2.1.

9.4.9 Technical Support Center HVAC System

The Technical Support Center (TSC) heating, ventilating, and air conditioning system is a constant volume system and provides a radiological habitable environment similar to the control room during normal and emergency conditions to protect personnel and equipment in the TSC.

The TSC HVAC system maintains the TSC at least 1/8 inch water gauge positive pressure with respect to the outdoors when the make up air filter unit is operating (during emergency) and positive pressure with respect to the outdoors during normal plant operation.

The TSC ventilation system is designed to limit the TSC temperature to 80°F during normal operation with outdoor temperatures varying between -10°F and +95°F and provide filtration and adsorption to maintain radiation levels below maximum permissible concentration per GDC Criteria 19 (5 Rem to the whole body or its equivalent) or 5 Rem TEDE for accidents analyzed using Alternative Source Term.

The HVAC system consists of the following components: makeup air filter unit, air handling unit, and condensing unit. The makeup filter unit capacity is 1000±10% scfm, with a maximum of 250 scfm taken from the outdoors to maintain the TSC pressure during emergency conditions.

9.4.10 References

1. Bechtel Calculation 113-C-001.

9.5 OTHER AUXILIARY SYSTEMS

The descriptions and evaluations of other auxiliary systems are included in this section. Other auxiliary systems include the fire protection system; communications system; lighting system; and the diesel generator fuel oil storage and transfer, cooling water, starting air, lubrication, and combustion air intake and exhaust systems.

9.5.1 Fire Protection System

The design bases, system descriptions, safety evaluations, inspection and testing requirements, NFPA conformance reviews, personnel qualifications, and training are described in Reference 1.

9.5.2 Communication Systems

9.5.2.1 Design Bases

The objective of the station communication system is to provide reliable and convenient communications among onsite personnel and with offsite locations. The following bases have been used to design the communication system:

A. Convenient communications to offsite locations;

B. Convenient intraplant communication and public address system;

C. Evacuation and station fire annunciation system with alarms at all strategic points throughout the plant; and

D. Radio communication.

9.5.2.2 Description

Telephone communication to offsite is Voice Over Internet Protocol (VOIP) system through an underground cable system. The Internet demarcation point is in the administration building. The internet lines are provided from the towns of Joliet and Morris.

A cellular phone system including internal and external antenna system with enhancers was added to the Control Room, Technical Support Center, and Operational Support Center. The Cellular Phone System is designed to provide reliable back-up communication in the event of a loss of the normal phone system.

The telephone system within the station is a VOIP system which is owned by AVAYA. The normal power feed for the system is from the security electrical bus. Phones are provided to the control room, all offices and routinely occupied areas, and central areas in the plant.

Dedicated communications systems at Dresden Station allow effective coordination of any emergency response. These systems include:

A. A nuclear accident reporting system (NARS) which links the control room; the technical support center (TSC); the emergency operations facility (EOF); the Illinois Emergency Management Agency (in Springfield) (IEMA); the Illinois Department of Nuclear Safety (in Springfield) (IDNS); the Grundy County emergency operating center (EOC); the Grundy County Sheriff's Department; the Will County EOC; the Will County Sheriff's Office; the Kendall County EOC; and the Kendall County Sheriff's Office.

B. The operations status line links control room, the TSC, and the EOF.

C. A Technical Conference line that enables communication between the TSC, and the EOF.

D. A Damage Control line that enables communication between the control room, the TSC, and the operational support center (OSC).

E. A radio voice channel between the control room, the TSC, the EOF, mobile vehicles, and portable, two-way radios in the field.

F. An emergency notification system (ENS), and a health physics network (HPN) that allow communications between the station and the NRC. This system is owned by the NRC and is part of the Federal Telecommunications System (FTS).

An intercom phone and loud speaker system is also provided within the Unit 2, and 3 areas. This system is independent of the VOIP system, and can be accessed from the paging phones. This system provides voice communication between the control room and selected areas in the plant. The power source for this system is from the emergency ac system.

An independent sound power phone system is also provided in the plant. Phone jacks are located throughout the two-unit area to permit instrument calibration between control room and local sensors. Under emergency conditions sound power phones can be used to maintain communications between the control room and the affected area.

The 900 MHz intra-plant radio system is digitally processed and controlled. This is a trunked, ten channel system with four repeater transmitters. If two repeater transmitters are lost, the system will switch to the remaining two and remain fully functional. The hand held radio units have a "talk around" mode that permits direct communication between the hand held units even if the radio base station equipment is not functional.

Base stations are located in the Control Room and Security Alarm Stations. Radio desk sets are placed in various locations – e.g. U2 NSO desk, U3 NSO desk, Work Execution Center, Technical Support Center, and Operations Support Center. All work groups in the station are provided with portable two-way hand held radios. Repeater antennas are located at various locations in the plant.

Key personnel are provided with pagers and can receive messages via a transmitter accessed from the VOIP system.

Located in the turbine building is a 250-W, 37.6-Mhz FM radio base station with a control station in the main control room. This radio permits direct contact with the system load dispatcher. Contact is also maintained with the alternate system load dispatcher.

Microwave and telephone communication equipment is provided to interface with the EOF. Signals from the computer systems (Process and Station Mini Computer) and designated phones located in the TSC and station are the sources of information. The interface between station equipment and microwave/land line equipment is through modems.

For emergency conditions the station has cellular telephones available for use by Operations and Emergency Response personnel as needed.

9.5.2.3 Performance Analysis

The evacuation and station fire annunciation system includes alarms located at strategic points throughout the plant to warn of a nuclear incident or other emergency conditions.

During normal operation and in particular during emergency conditions, communications offsite and onsite are of paramount importance. The communication systems for the Dresden Station are of such a diversity of design that the operating group is assured of maintaining voice contact both onsite and offsite. For a discussion of maintaining communications within the plant during a fire see, Section 7.6 of the Safe Shutdown Report (FPR Volume 2).

9.5.2.4 Testing and Inspection Requirements

The evacuation and station fire annunciation system is tested periodically.

9.5.3 Lighting Systems

9.5.3.1 Design Basis

The objective of the station lighting system is to provide adequate lighting for the operation and maintenance of equipment.

The Main Control Room lighting is designed to provide illumination levels sufficient for task performance, consistent with human factors guidelines found in the Human Factors Engineering Design Criteria and Standards Manual. See Section 7.5.4 for further discussion of the Detailed Control Room Design Review.

The minimum standby lighting requirements (in footcandles) are shown in Table 9.5-2.

9.5.3.2 System Description

The station lighting system is designed to provide adequate lighting for the normal operation of the plant. The standby lighting system uses both ac power from the diesel generators and dc power from self-contained lighting units and station batteries.

Eight-hour emergency lighting is installed in all areas containing safe shutdown equipment that needs to be operated following a fire. Eight-hour emergency lighting is also located on access and egress routes for this equipment.

9.5.3.3 Performance Analysis

Standby lighting is necessary to provide adequate lighting in all areas essential for the safe operation of the plant during loss of normal ac power. In order to conform with 10 CFR 50 Appendix R requirements, a walkdown was performed on all the primary and alternate access routes that the station identified for each manual actuation of safe shutdown equipment at the station. This walkdown was performed (in accordance with the guidelines in Dresden station special procedure SP-84-7-62) with all lighting deenergized except the battery pack lights. The 8-hour battery-powered emergency lighting unit locations resulting from the walkdown are shown on Drawings F-201-6 through F-214-6.

9.5.3.4 Tests and Inspections

Station procedures provide for periodic testing of the standby lighting to demonstrate its availability. Appendix R Emergency Lighting is tested under the fire protection program (see FPPDP Volume 7, Section 5.6).

Normal station maintenance provides adequate inspection of normal lighting equipment.

9.5.4 Diesel Generator Fuel Oil Storage and Transfer System

A separate, safety-related and seismically qualified fuel oil storage and transfer system is provided for each diesel generator (DG). Each storage and transfer system includes a 15,000-gallon underground diesel fuel oil storage tank and a 750-gallon diesel oil day tank. The fuel oil storage tanks are protected against tornado missiles by virtue of their location underground. The day tank and the piping and equipment downstream of the day tank are classified as safety-related.

Fuel is transferred from the 15,000-gallon diesel fuel oil storage tank to the 750-gallon diesel oil day tank with the diesel oil transfer pump. Transfer is accomplished automatically by level switches on the day tank. Diagrams of the Unit 2, Unit 3, and the Unit 2/3 DG fuel oil storage and transfer system are shown on Drawing M-41, Sheet 2 and Figure 9.5-2.

Each day tank contains sufficient fuel to sustain emergency diesel generator operation for 1 hour of operation at 110% of rated load and 61.2 Hz. The configuration of the system is such that the minimum normal operating level in the day tank is above the low level alarm setpoint. The low level alarm setpoint is maintained above the minimum required 1 hour fuel storage level. This ensures that the day tank provides a minimum 1 hour of fuel for operation of the diesel generator at 10% above rated load and 2% above the normal frequency of 60 Hz (245 gallons). The fuel oil transfer system, which is safety-related and seismically qualified, ensures the delivery of fuel for diesel generator operation beyond the 1-hour supply of the day tank. Each diesel fuel oil storage tank has a capacity adequate to sustain system operation pending normal commercial deliveries of fuel. The diesel generator uses a fuel which is readily available.

The Technical Specifications require a minimum of 10,000 gallons of diesel fuel to be kept on site for each DG. Figure 9.5-3 shows diesel fuel consumption versus load. At the 10% overload condition (2860 kW) and 2% above normal frequency (61.2 Hz) each DG will consume 211 gallons of fuel per hour; at a rated load of 2600 kW, each DG will consume 192 gal/hr, and at 50% rated load, 105 gal/hr. An original FSAR analysis, using original diesel loads, postulated that two DGs are connected to the emergency buses in a unit which has experienced a design basis accident concurrent with a loss of offsite power and that a third DG is connected to the non-accident unit which is being shut down after having also experienced a loss of offsite power. This analysis conservatively estimated a total fuel consumption of 340 gallons for the first hour. After the first hour the reactor will probably have been sufficiently cooled such that some emergency core cooling system (ECCS) pumps can be shut off and the loads diminished on the diesels, resulting in a fuel consumption of about 250 gal/hr. The analysis estimated that the maximum onsite diesel fuel supply (47,250 gallons total) would last 7.9 days and that the minimum onsite diesel fuel supply, as required by Technical Specifications, would last 5.2 days.

To provide an adequate margin of safety beyond the anticipated 8-hour delivery time for fuel oil, the actual basis for the minimum onsite fuel supply specification is a two-day supply to each diesel with the diesel operating at full load. This basis resulted in the specified 10,000 gallon minimum supply for each diesel.

The diesel oil day tank level is sensed by a level switch, which initiates a signal to automatically start or stop the fuel oil transfer pump. The fuel oil transfer pump starts automatically when its respective day tank level drops below a predetermined level.

Upon initiation of a DG start signal, fuel from the fuel oil day tank is supplied through a strainer via an electrically-driven fuel oil priming pump and a duplex filter to the injectors. Any excess fuel is returned to the fuel oil day tank. At an engine speed of approximately 200 rpm, the priming pump deenergizes and the engine-driven fuel oil pump continues to supply fuel oil.

The Unit 3, diesel oil transfer system also supplies fuel oil to the Unit 2/3 diesel fire pump day tank. There is sufficient fuel oil stored onsite to support the simultaneous operation of the fire pump and the Unit 3 DG. The transfer of fuel from the diesel fuel oil storage tank to the fire pump day tank requires opening a manual isolation valve which is administratively controlled.

The connection of the diesel fuel oil transfer pumps to the emergency buses is automatic. Each diesel generator has one fuel oil transfer pump which starts automatically when its respective day tank level drops to 32 inches nominal (two-thirds full). Figure 9.5-4 shows the electrical schematic and control from each DG to its respective fuel oil transfer pump.

A three-position switch (Unit 2, Normal, and Unit 3) on the 2/3 DG auxiliary control panel isolates, controls, and protects the 2/3 DG fuel oil transfer pump in the event of a fire. With the switch in the normal position, the 2/3 DG fuel oil transfer pump receives power from MCC 28-1. If the power is unavailable from MCC 28-1, the power automatically transfers to MCC 38-1. If the switch is in either the Unit 2 or Unit 3 position, the opposite unit's power source is electrically isolated. This would be performed if a fault has occurred or is anticipated on MCC 28-1 or MCC 38-1 such as if a major fire is located at the MCC or near the cabling of the fuel oil transfer pump.

A curb around the 2/3 DG fuel oil day tank provides a means of containing oil and water leaks or fires, and facilitates leak disposal. For additional information regarding fire protection, see Section 9.5.1. Also, as shown in Figure 9.5-2, an emergency fuel cutoff valve is provided for each fuel oil day tank in the event of a fire or other emergency.

Operability of the DG fuel oil transfer pumps is verified during DG testing. Diesel fuel oil additives are used to maintain fuel oil quality. Diesel fuel oil is sampled and tested once a month to verify quality.

9.5.5 Diesel Generator Cooling Water System

The diesel generator cooling water (DGCW) system provides cooling water to each of the three DGs. In addition, the system provides an alternate water supply for the containment cooling service water (CCSW) keep fill system. The Unit 2 DGCW system, via hose connections on its discharge piping, provides a seismically-qualified emergency supply of water to either the A or B isolation condenser makeup pump if all other sources of makeup to the isolation condenser shells are unavailable. Refer to Figure 9.5-9.

The closed loop portion of the DGCW system circulates cooling water through cored passages in the diesel's cylinder liners, cylinder heads, turbocharger aftercoolers, and lube oil cooler. Diesel engine cooling water temperature is maintained relatively constant by the temperature regulating valve which controls the flow of engine water through the DGCW heat exchangers. A bypass line provides fast engine warmup and a constant flow of diesel engine cooling water. The heat exchanger cooling water is provided by the open loop portion of the DGCW system. A water expansion tank provides a surge volume and makeup capability. The motive force for this cooling water is provided by two gear-driven centrifugal pumps. During diesel standby readiness, a 15-kW immersion heater is provided to warm the diesel lube oil. Natural circulation forces the cooling water through the immersion heater. The oil cooler then becomes an oil "heater" and the flow of oil

from the circulating pump is warmed to keep the oil system in standby readiness. A portion of closed loop cooling water is constantly routed through a temperature switch manifold; temperature switches on the manifold monitor cooling water temperature and transmit high temperature signals to a local alarm panel and the DG trip circuitry. Diagrams of the DGCW system are shown on Figures 9.5-5, 9.5-6, 9.5-7, and 9.5-8 (Figures 9.5-6, 9.5-7 and 9.5-8 are the same as Drawing M-517, Sheets 1, 2, and 3).

The open loop portion of the DGCW system provides cooling water to the DG heat exchangers. A separate DGCW pump is provided for each cooling water system. The open loop portion of the DGCW system consists of three pumps (one for the Unit 2 DG, one for the Unit 3 DG, and one for the shared 2/3 DG) in the crib house, the components cooled by the system and the associated piping and valves. The open loop portion of the DGCW system is discharged into the service water system (SWS) discharge header.

The DGCW pumps can be cross-connected so that each pump can supply any of the other pumps' cooling loads, except for the DGs, but they are normally isolated from each other and operated as separate subsystems. Thus, a failure in one subsystem would not affect the safety function of the other cooling water subsystems.

The pumps are driven by submersible, canned rotor motors which ensure pump availability in the event of flooding. The motors are cooled by the discharge of the pump which is returned to the pump suction. The pump suction is taken from the circulation water bays. A crosstie header allows the pumps to take suction from alternate bays while maintenance is being performed on a circulation water bay.

The DGCW pumps are located in Class II structures, but have been afforded Class I protection. They are located at elevation 490'-8" in the crib house where the circulating water pumps are located. This floor is more than 8 feet below the ground level in an area where a 2-foot thick reinforced concrete slab is directly above the DGCW pumps. The remaining part of the system's piping and valves traverse to and from the missile-protected diesel and reactor buildings via a reinforced concrete tunnel that runs below ground. Therefore, the DGCW system is adequately protected against tornado missiles. Further discussion of tornado missiles and components within the crib house can be found within UFSAR Section 3.5. The concrete structure of the crib house would not be affected by earthquake.

Under normal operating conditions, the SWS supplies flow to the DGCW system loads through check valves. Normally closed manual valves in the crib house can be opened to cross-connect the discharge piping of any DGCW pumps to any or all DGCW heat loads, except for the DGs. There are no power-operated valves in the DGCW system. The DGCW pump jackets and bearings are cooled by the pumped fluid, and the pumps do not depend on other systems for cooling or lubrication.

Also, a connection is provided to the CCSW keep fill system by the emergency core cooing system (ECCS) room cooler crosstie header. The CCSW keep fill system is normally supplied by the SWS, but the DGCW pumps may be used as an alternate water supply.

A crosstie between the Fire Protection System also allows the 2/3 DGCW discharge flow to serve as an alternate makeup water source for the isolation condensers, spent fuel pools and reactor pressure vessels.

The capacity of each DGCW pump is 1100 gal/min at a total discharge head of 115 feet. It has been determined from various evaluations that safety related, DGCW pump flows required to service the respective heat loads are dependent on inlet cooling water temperatures. The minimum flow versus inlet temperature based on the S & L calculation number ATD-0400 are as follows:

TEMPERATURE	FLOW*
95°F	800 GPM
90°F	725 GPM
85°F	660 GPM
80°F	615 GPM
75°F	575 GPM

* For Unit 2 and Unit 3 DGCW Pumps, the minimum flow rates are 30 gpm higher to reheat the additional CCSW keep fill function flow requirements.

This calculation is based on not exceeding a high cooling water temperature alarm (190°F) and to prevent a high cooling water temperature trip (200°F) during normal operation. The high cooling water temperature trip is bypassed in the "Auto Start" mode. The flow identified above are the minimum DGCW flow versus inlet temperature necessary to maintain engine coolant temperature less than 190°F alarm set point.

The DGCW pump trip alarms in the control room, and remotely, on the generator relay and metering panel. A restriction orifice type flow indicator is also provided in the DGCW pump discharge line in accordance with Regulatory Guide 1.97.

The Unit 2 pump receives electrical power from 480-V motor control center (MCC) 29-2. The Unit 3 pump receives electrical power from MCC 39-2. The 2/3 DGCW pump normally receives power from MCC 28-3, but an automatic device connects the pump to MCC 38-3 (Unit 3) if MCC 28-3 is deenergized. The pumps can be operated in manual mode or in automatic start mode. In the event of a loss of offsite power, the pumps are connected to the DG bus.

Each DG can operate without cooling water for 3 minutes at full load with a speed of 900 rpm assuming an initial cooling water temperature of 100°F prior to engine start. The DG operating time increases to 10 minutes with no load on the generator (at 900 rpm). At its idling speed each diesel generator can run for 42 minutes without cooling water, again assuming an initial water temperature of 100°F.

9.5.6 Diesel Generator Starting Air System

The purpose of the diesel generator (DG) starting air system is to store and deliver sufficient air to start the diesel under all conditions. The safety function of the air start piping is to provide a means to start the diesel engine in case of a loss of offsite power.

The DGs are started by air-driven starting motors. A separate starting air system is provided for each DG. Each DG starting air system has two starting air compressors and two air-driven starting motors. If the starting solenoid valve is energized, two air-driven starting motors engage a flywheel ring gear. After the two air-driven starting motors engage, the air start relay valve opens and four air receiver units supply the air which cranks the starting motors. Two air compressors maintain the air receiver pressure at greater than or equal to 220 psig. At an engine speed greater than 200 rpm, the starting solenoid valve deenergizes, interrupting the air to the starting motors and venting off the pressure which causes the air motors to stop and disengage. If the air receiver pressure is reduced to 175 psig, sufficient pressure would remain to start the DG once with no air compressor action. A diagram of diesel starting air piping is shown on Figure 9.5-10 (Drawing M-173). The safety-related portion of the system is shown on Drawing M-173.

Some minor modifications to the DG starting air system resulted from design concerns raised by the Dresden Safety System Functional Inspection (SSFI). These concerns have been addressed in the "Operability Assessment of SSFI Report Concerns".[2] The modifications have enhanced the DG starting air system, and have been implemented on the Unit 2 and Unit 3 DGs and the Unit 2/3 swing diesel generator. These modifications included the following:

A. Addition of a single 1½-inch check valve to the combined discharge header of both air receiver units in each starting air train;

B. Removal and replacement of the air receiver inboard isolation drain valves with higher pressure rated valves in each diesel starting air train; and

C. Resupport of drain piping from each DG starting air receiver.

The diesel start-up air piping (P&ID M-173) is designed to provide starting air from both A and B receivers for diesel starts. Should either of the receivers become depressurized, check valves in the piping will isolate the depressurized receiver from the pressurized receiver; thus allowing air from the pressurized receiver to start the diesel.

The second modification replaced each air receiver inboard isolation drain valve (200-psi gate valve) with higher pressure rated valves. Since the system pressure is 250 psi, replacement with higher pressure valves would allow using these valves as isolation points without undue potential for leakage across the valve seat.

The third modification resupported the air receiver drain piping as a design enhancement. Pipe supports for the air receiver drain line were added as a part of the modification.

Power for the Unit 2 DG starting air compressors is supplied by 480-V motor control centers (MCCs) 28-2 and 29-2. Power for the Unit 3 DG starting air compressors is supplied by 480-V MCC 38-2 and 39-2. Power for DG starting air compressor 2/3A is supplied by 480-V MCC 28-1, and power for DG starting air compressor 2/3B is supplied by 480-V MCC 38-4.

During a diesel engine starting sequence, the start failure relay energizes if the engine does not reach 200 rpm within 15 seconds after the start signal is initiated. An alarm annunciates on the engine panel and a DG fail-to-start alarm annunciates in the control room. Also, when the manual shutoff valve on the diesel starting air system is closed, the closure annunciates an alarm in the control room.

During DG testing, the diesel starting air compressor is checked for operation and for its ability to recharge the air receivers.

9.5.7 Diesel Generator Lubrication System

A separate lubrication system is provided for each diesel generator (DG). During operation, the diesel engine drives three oil pumps. The engine-driven oil pumps are the scavenging pump, the main lube oil pump, and the piston oil pump. During standby conditions, the DG lubrication system is kept warm by heat transferred via natural circulation from a 15-kW immersion heater (located in the cooling water system) to the DG oil coolers which, in a standby mode, act as heaters to heat circulating oil. Two electrically driven oil pumps circulate the warmed oil through the lubrication system. Diagrams of the Unit 2, Unit 2/3, and Unit 3 DG lubrication system are shown on Drawing M-478, Sheets 1, 2, and 3).

The scavenging oil pump takes a suction from the diesel engine oil pan, pumps the oil through a filter and cooler, and provides a suction to the piston oil pump, and the main oil pump.

The piston oil pump supplies oil for the cooling of the piston and lubrication of the piston pin bearing surfaces.

The main oil pump supplies oil for the other moving engine parts such as the main bearings, gear train, cam shaft, and rocker arms.

Standby DG oil circulation provides continuous lubrication of the engine crankshaft and turbocharger bearings and maintains lube oil system accessories filled with oil at all times. Standby lubrication minimizes premature bearing wear during engine starts by providing the proper amount of prelube. Prelube also alleviates the possibility for engine wear caused by restarting the engine within 15 minutes to 3 hours following shutdown. The 15-minute to 3-hour vulnerability period is when changes in oil viscosity result in a flow imbalance of standby lube oil between the turbocharger bearings and engine accessory rack components.

To counteract the flow imbalance effect, a motor-driven turbocharger lube oil circulating pump provides a dedicated flow of lube oil to the turbocharger for bearing lubrication during standby periods and for residual heat removal following engine shutdown. A motor-driven lube oil circulating pump circulates oil through the accessory rack components (i.e., lube oil filter, cooler, etc.) to maintain their oil level, and it provides pressurized oil to the main crankshaft bearings and engine gallery.

The turbocharger lube oil circulating pump was installed to upgrade the standby lube oil system. The portions of the upgrade modification which were not safety-related included the electrical feeds downstream of the Class 1E circuit breakers at motor control centers (MCC) 28-3, 29-4 and 38-3 up to and including the new pump motors, and the annunciation circuit for the two discharge pressure switches. However, the switches themselves were classified as safety-related since they maintain the pressure boundary of the lube oil system.

The standby lube oil system modification has no effect on the start and load rate capability, overall capacity, or operability status of the DG units as outlined in the Technical Specifications. The lube oil system modification enhances reliability by minimizing the degrading effect of frequent start and operability tests as described in the Dresden Technical Specifications.

The standby lube oil system was designed to minimize the potential for cumulative bearing wear due to inadequate lubrication during engine starts. On a functional level, the circulating lube oil system is not essential to the operability of the engines. The system does, however, makeup part of the engine's lube oil system pressure boundary which must be maintained to ensure successful operability of the engine.

Power for the Unit 2 lube oil circulating and turbocharger lube oil circulating pumps is supplied by 480-V MCC 28-3. Power for the Unit 3 lube oil circulating and turbocharger lube oil circulating pumps is supplied by 480-V MCC 38-3. Power for the 2/3 lube oil circulating and turbocharger lube oil circulating pumps is supplied by 480-V MCC 29-4.

Local and remote annunciation is provided to alert operators in the event of a disruption in circulating lube oil system flow. Low lubricating oil temperature and/or low main bearing oil pressure initiate a DG trouble alarm in the main control room. One of two main bearing oil pressure switches can also initiate a DG shutdown when oil pressure decreases below the switch's predetermined setpoint.

Operability of the DG lubrication system is verified during DG testing.

9.5.8 Diesel Generator Combustion Air Intake and Exhaust System

A diagram of diesel generator (DG) combustion air intake and exhaust piping is shown in Drawing M-40. As shown in the drawing, outside air is drawn through an oil-bath filter to the DG for combustion. Diesel engine exhaust gases are directed through a turbocharger to the exhaust silencer. The safety-related portion of the system is shown in Drawing M-40.

Outside air is drawn to the DGs through the turbochargers to support combustion of diesel fuel oil. The turbochargers are driven by two motive sources. The primary turbo driving force is derived from high-energy exhaust gas flowing through the turbine section of the turbocharger. At the time of engine startup and during the period preceding load application, there is a relatively low level of energy generated in the exhaust gas. During this interim period, the engine uses the turbocharger gear train to drive the turbocharger at 18 times engine speed to provide compressed air for combustion.

Normally, high driving torque is transmitted through the turbocharger gear train for only a short time because, as engine load is applied, the exhaust gas energy proportionally increases its share of the drive burden. When full rated load is applied, the exhaust energy is sufficient to drive the turbocharger without gear train assist. The overrunning clutch then allows the turbocharger to speed up and run independently of the gear train; the gears continue to freewheel without transmitting torque.

Protection of the combustion air intake and exhaust system from tornado missiles was reviewed under Topic III-4.A of the Systematic Evaluation Program (SEP) performed for Unit 2. A probabilistic assessment of tornado missiles impacting the system was performed and it was concluded that the probability of exceeding the requirements of 10 CFR 100 is very low and, as such, acceptable.[4] This topic is further discussed in selection 3.5.4.

9.5.9 Station Blackout System

9.5.9.1 Station Blackout System Description

The station blackout system is a non-safety-related, independent source of additional on-site emergency ac power. The System consists of 2 diesel-driven generator sets, each set having a continuous rated output of 4350 kw at 4160 volts ac and a 0.8 power factor. Each generator is connectable, but not normally connected, to the safe shutdown equipment on one nuclear unit, but can also be connected to the opposite unit via the safety-related 4kv cross-ties. The station blackout diesel generators (SBO DGs) must be manually started and manually connected to the appropriate safe shutdown loads. The start and load function of the SBO DG can be performed from the main control room.

The SBO diesel generators and auxiliaries are located in the station blackout building, which also houses the Unit 2 safety-related 125 Vdc alternate battery. For this reason, the SBO Building is a Category 1, safety-related structure. The SBO building protects against weather-related events which could initiate an SBO event and provides physical isolation from safety related components. The SBO Building is physically separated from emergency systems, thus avoiding the consequences of multiple failures of on-site diesel generator systems due to severe weather-related events. The 4 kV power cables between the SBO and safety-related switchgear are located in underground cable ducts.

The SBO DG auxiliaries include: a fuel oil system, starting air system, engine lubrication system, engines cooling system, engine combustion air and exhaust, a distributed control system (DCS), local engine control panels, dc battery and switchboard, uninterruptible power supply and panel, 4kv switchgear, synchronizing equipment, 480 volt motor control center, HVAC, building drain and sump system, and fire detection/protection system.

9.5.9.1.1 Engine/Generator Set

The configuration of engines is a tandem DG set with a 12-cylinder and 16-cylinder engine on a common shaft with the generator in the middle.

The synchronous generator is rated at 4350 kw, 4160 volts, 60 hertz, 0.8 power factor. Its maximum output current is 755 amps. The rotor has an 8-pole field rotating at the synchronous speed of 900 rpm. The exciter is a brushless design. A permanent magnet generator provides a source of power to the voltage regulator and exciter and obviates the need for dc field flashing circuitry.

9.5.9.1.2 Fuel Oil System

The purpose of the fuel oil system is to provide for storage and transfer of fuel oil from the storage tank to day tank and from the day tank to the engines. The fuel oil system for the SBO DGs is entirely independent of any other fuel oil systems at Dresden.

The two SBO DGs share a common 15,000 gallon underground storage tank and short sections of common suction and return piping. Each SBO DG has its own fuel oil transfer pump which takes suction on the storage tank and discharges to a 1095-gallon day tank. The transfer pump is an ac-driven, positive-displacement pump. The discharge piping of each transfer pump can be cross-tied to allow for filling the opposite unit's day tank. The storage tank and each of the day tanks is equipped with a vent line to atmosphere through a flame arrestor to prevent overpressurization of the tanks. Overfilling of the storage tank will initiate in alarm, locally and in the control room. An overflow line from each of the day tanks back to the storage tank mitigates the consequences of overfilling of the day tanks.

Fuel oil is transferred from the day tanks to the engines using an engine-driven pump. In parallel with the engine-driven fuel pump is an ac-powered fuel oil backup pump and a manual priming pump.

The following fuel oil consumption data for the Unit 2 SBO DG, using No. 2 diesel fuel oil, was obtained during factory testing. The data for Unit 3 SBO DG is similar to these values.

50% load: Observed consumption 154 gallons/hr
75% load: Observed consumption 207 gallons/hr
100% load: Observed consumption 271 gallons/hr
110% load: Observed consumption 307 gallons/hr

Thus, the day tank contains enough fuel for about 4 hours of single-unit operation at 100% load without refilling. The filled storage tank, accounting for the unfilled head space, has enough fuel to operate both SBO DGs at 100% load for about 24 hours.

Manual and automatic operation of the fuel oil transfer pump is available. In the automatic mode, the pump will operate upon a signal from the distributed control system (DCS), indicating a low-normal level in the day tank. The DCS also starts the pump upon an engine shutdown. The pump stop signals are received from the DCS upon sensed high-normal level. The ac-powered fuel oil backup pump will operate upon sensed low fuel oil pressure while at rated speed. This pump will sustain engine-operation in case the engine-driven pump fails.

The fuel oil transfer pump and the ac backup pump are powered from MCC 65-1 (75-1).

9.5.9.1.3 Starting air system

The SBO starting air system, consisting of two independent trains, provides full redundancy in both components and connections to supply air to the engine starting motors. Each redundant train can provide, at a minimum, five (5) full cranking cycles without recharging its air receivers. Therefore, each DG set can receive 10 cranking cycles, independent of ac power.

Operation of the starting air system to start the SBO DG is accomplished either locally or from the main control room. A local control switch is used to select the lead starting air train. Starting air logic is as follows. The engines will crank using the selected lead train by energizing the two (2) dc-operated solenoid pilot valves for that train. If after 4 seconds, the controller does not sense at least 80 rpm engine speed, it will automatically swap over to the opposite train. This swapping will continue for a total cranking time of 10 seconds, after which the start controls will lock out and provide an overcrank alarm both locally and in the control room. Swapping is inhibited as long as the sensed engine speed is above 80 rpm. The starting motors will continue to crank until 200 rpm engine speed or the 10 second overcrank alarm and lock out is enabled. Above 200 rpm, the starting motors disengage by de-energizing the solenoid pilot valve, venting off the pressure which causes the air motors to stop and the pinion gear to retract.

All ac- power components of the starting air system are fed from MCC 65-1 (75-1).

9.5.9.1.4 Engine Lubrication System

A separate lubrication oil system is provided for each diesel engine, with all components on either the main skid or the engine accessory rack. During operation, the engine drives three oil pumps: the scavenging oil pump, the piston cooling oil pump, and the main bearing oil pump.

The scavenging oil pump takes a suction from the diesel engine oil sump, pumps the oil through a filter and cooler, and provides a suction to the piston cooling oil pump and main bearing oil pump.

The piston cooling oil pump supplies oil for the cooling of the piston and lubrication of the piston pin bearing surfaces.

The main bearing oil pump supplies oil for the other moving engine parts such as the main bearings, gear train, cam shaft, and rocker arms. This supply stream exits the engine and is filtered through an engine-mounted turbo lube oil filter and supplies the turbocharger bearings.

During standby conditions, the engine lubricating oil is kept warm by heat transferred from the jacket water system via the engine oil coolers which, in standby, heat circulating oil. The circulating oil is driven by a 1 hp, ac lube oil circulating pump.

Unlike the emergency diesel engine, the SBO engine does not provide lubrication of the crankshaft bearings during standby conditions, only the turbocharger bearings. However, each engine is equipped with a pre-lube function which can be used prior to non-emergency engine starts to lubricate crankshaft bearings, camshaft bearings and rocker arms.

The ac lube oil circulating pumps are fed from MCC 65-1 (75-1) and the dc lube oil circulating pumps are fed from Panel 6A-1 (7A-1). Low lube oil pressure during operation will annunciate both locally and in the control room and will trip the engines if in non-emergency conditions.

9.5.9.1.5 Engine cooling system

Heat rejection from the engine jacket water cooling system is provided by dedicated radiators. Each engine has its own independent flow path and radiator. Included in this flow path is a hotwell tank which serves as a source of cooling water so that the engine can operate without ac-driven radiator fans for a short time.

The engine-driven jacket water pumps provide the only forced circulation in the jacket water system. There are two pumps per engine with each pump supplying loads on its half of the engine.

An engine trip will automatically be initiated on jacket water low pressure at rated speed or on jacket water high-high temperature while at rated or idled speed. These trips are accompanied by alarms both locally and in the control room. During emergencies, these trips are automatically bypassed. A high temperature alarm and low expansion tank level will alarm at any time, regardless of engine operating status, locally and in the control room.

During standby, a 15-kw immersion heater heats the jacket water directly and the lube oil indirectly for more reliable starts.

The immersion heaters and radiator fans are powered form MCC 65-1 (75-1).

9.5.9.1.6 Local engine control panels

The main engine control panels (2202 (3)-104, 105, 106, 107) are housed in a common structure. These panels contain engine and generator control, monitoring, and status functions. Local operation of the generator set can be performed from the -105 panel. This panel also contains the local annunciator board and a programmable logic controller (PLC), in which many of the control and monitoring functions are embedded. The -107 panel contains the generator protective relays and the neutral grounding transformer.

9.5.9.1.7 DC battery and switchboard

An independent 125 volt dc battery system for each generator unit will provide the necessary power for 4 kV switchgear control and indications, diesel generator control and indications, lube oil standby circulation, and the uninterruptible power supply (UPS) inverter.

The battery is a 125 Vdc lead-acid battery, and is contained in its own bermed, ventilated room. The ampere-hour capacity of battery is adequate to supply expected SBO system loads for one hour without taking credit for the battery chargers. The battery, its 300 A stationary battery charger, and a shared maintenance charger connect to dc Switchboard 6A(7A). The switchboards provide feeds to the various dc loads.

The ac source for the stationary battery chargers is MCC 65-1 (75-1).

9.5.9.1.8 Uninterruptible Power Supply (UPS) and panel

The UPS is an independent system which will provide 15 kva single phase power to the local DG control panel and the Distributed Control System (DCS) in the event of loss of normal ac power. The normal source for the UPS panelboard is a 15 kva inverter and dc switchboard. A bypass source is available through a 15 kva single-phase isolimiter transformer whose primary side is fed from MCC 65-1 (75-1).

9.5.9.1.9 4 kv switchgear

Figure 9.5-14 is a single-line diagram showing the 4kv connections between the SBO DGs and the plant safe shutdown buses for both Units 2 and 3.

Dedicated 4 kv switchgear centers 61 (71) are provided for 4 kv connections from the SBO diesel generator to the safety-related 4 kv buses. Bus 61 can be connected to either Bus 23 or 24, but not both simultaneously. Bus 71 can be connected to either Bus 33 or 34, but not both simultaneously. These connections to the safety-related buses are isolated via 2 breakers, one safety-related (located at the safety-related bus) and one non-safety-related (located at Bus 61 (71)), controllable from the main control room.

The safety-related feed breakers from SBO at Bus 23 (33) and 24 (34) all operate similarly. These breakers must be closed manually from control room panel 923-74 before the associated breaker at Bus 61 (71) can be closed. The breakers are tripped manually from control room panel 923-74. The breakers will also trip upon Bus 23 (33) or 24 (34) undervoltage or upon unit LOCA signal being generated, if the SBO mode switch is in the normal mode, as it would be during surveillance test conditions.

The tie breakers at Bus 61 (71) to Buses 23 (33) and 24 (34) all operate similarly. When Bus 23 (33) is deenergized and the SBO mode switch at panel 923 - 74 is in the SBO mode, the feed breaker can be "dead bus " closed. The tie to Bus 24 (34) works similarly. These breakers can be manually tripped under all conditions and will trip automatically should the diesel generator feed breaker trip.

The SBO DG 2(3) generator is connectable to Bus 61 (71) from the main control room. When Bus 61 (71) is deenergized, the SBO mode switch at panel 923-74 can be placed in the SBO mode and the feed breaker can be "dead bus" closed. Once the breaker is closed in this mode, and the diesel generator running in emergency mode, only a manual trip, a bus fault or a select few diesel generator faults will trip this breaker. If the diesel generator is secured from the emergency mode at panel 923-74, a manual trip or any diesel generator trip or normal stop of the diesel generator will trip the breaker.

The normal SBO building power source is supplied from Unit 1 Bus 11 to Bus 61 and 71, when the SBO DG is in standby. This feed is tripped on undervoltage of the respective bus or when the respective mode switch at panel 923-74 is placed in SBO mode.

Buses 61 and 71 can be cross-tied to prevent blacking out half of the SBO Building when normal feed breaker for that half is out-of-service. Interlocks prevent completing the cross-tie if any of the Bus 61 or 71 feed breaker to the safety buses are closed or the SBO DG output breaker is closed. These breakers trip if the respective SBO mode switch is placed in SBO mode or if an undervoltage condition exists on either Bus 61 or 71.

9.5.9.1.10 480 Volt Motor Control Center (MCC)

Dedicated 480 Vac motor control centers (MCC 65-1(75-1)) are provided for distribution of station blackout auxiliary power. The MCC is fed from Bus 61 (71) through 4 kv/480 v transformer 65 (75). Load distribution is provided for the diesel auxiliaries, heating, ventilation and air conditioning, battery chargers, lighting, and welding receptacles.

9.5.9.1.11 Distributed Control System (DCS)

The DCS provides remote operation of some SBO systems, primarily the diesel generator and switchgear control, from the main control room. The DCS receives analog signals and contact open/close signals for monitoring of the SBO systems status. The DCS sends digital outputs for system control via contact open/close changes.

The DCS central processing unit, touch screen control console, and other controls are located in main control room panel 923-74.

9.5.9.1.12 Fire protection system

The SBO Building and equipment are protected from fire by fire hoses in the main rooms of the building and a wet pipe sprinkler system in each of the diesel rooms and day tank rooms. The source of water is the main fire ring header. Alarms are generated when these water systems are actuated.

A smoke and heat detection system is installed in the rest of the SBO building and supervised by the plant XL-3 Pyrotronics system. The power for this function comes from MCC 75-1.

9.5.9.2 Capacity of SBO Diesel Generators

Two SBO diesel generators are installed at Dresden. The capacity of each of these generators is 4350 kw continuous, with a 2000-hour rating of 4785 kw. This capacity is sufficient to power one division of safe shutdown loads needed in dealing with a station blackout event. The SBO event loading scenario was developed on a per station per division basis with the aid of station operators and considered a reactor initially a full power and an indefinite SBO event duration (except for refilling the fuel oil storage tank). The worst-case divisional load requirements at either Dresden or Quad Cities was then used as the design basis for the minimum generator capacity. This capacity easily envelopes the load requirements for a LOOP or LOCA event as defined in the UFSAR Tables 8.3-2 and 8.3-3. Additional capacity to reduce the need for operator manual load-shedding and provide a contingency margin of 10% was added to this minimum capacity.

Initial system tests have demonstrated that the capacity of 4785 kw is achievable. In addition, a more rigorous initial test of the Unit 2 generator capacity was performed by picking up plant loads on isolated plant buses in a stepwise fashion, more accurately simulating emergency conditions. During this test, the following loads were successfully accelerated to rated speed without violating the criteria for allowable voltage and speed drop during the transient.

2 CCSW pumps,
2 LPCI pumps,
1 Core spray pump,
1 Service water pump,
3 Drywell coolers,
2 RB exhaust fans,
1 RB vent fan, and
Various other 480 volt loads - for a total loading of 3650 kw (2200 kvar).

The Unit 3 generator response and capacity is similar and was similarly tested. Actual loading of the SBO generators, during a station blackout event or a less severe event, will be done manually by the operators under the guidance of procedure DGA-12 and Dresden Operating Procedures.

9.5.9.3 Connectability to Safety-Related 4 KV Buses

Although not normally connected to any bus, the configuration of the SBO 4 kv distribution system allows connection of each SBO DG to any of the four safety-related emergency buses. Figure 9.5-14 shows that the Unit 2(3) SBO DG feed, Bus 61(71) in the SBO Building which then connects to either Bus 23(33) or 24 (34). These connections will be isolatable via a two breaker system, that is, one safety-related and one non-safety related breaker in each line-up. During emergency conditions, Bus 23-1(33-1) or 24-1(34-1) will then be connected to the SBO DG. If conditions require, the safety-related cross-tie between Bus 23-1 and 33-1 or between 24-1 and 34-1 can be utilized so that the Unit 2 SBO DG can supply power to Unit 3 loads and vice versa.

9.5.9.4 Modes of Operation

The control logic for starting and loading the emergency diesel generators is not affected by any of the station blackout controls.

Under conditions of total or partial loss of offsite power, the SBO DGs can either be remotely or locally started in the emergency mode in which the engine will immediately accelerate to rated speed and almost all trips are bypassed. Dead bus connections can be made to allow the SBO DGs to provide power to safe shutdown plant loads. Emergency operation is designed to utilize the programmable logic controller (PLC) in panel 2202(3)-105. If a failure is detected in the PLC, the generator set can still be started and controlled locally in the PLC Bypass mode; however, the emergency start pushbutton must be depressed which provides for a fast start and bypassed trips. The trips which are not bypassed during emergency condition, are:

Engine overspeed
Generator ground fault (59G)
Generator differential (87)

The generator trips require resetting a lockout relay at the SBO switchgear and the overspeed trip requires resetting an engine latch.

Prior to starting the engines for surveillance tests (i.e. non-emergency conditions), the lube oil valves can be manually configured to provide prelubrication of the engine bearings and rocker arm assemblies while the engine is barred over. The engines can then be remotely started in the normal mode in which the engine heatup for 5 minutes at idle speed prior to going to rated speed with all trips enabled. Synchronized breaker closings connect the generator set to the plant safety buses and off-site power for loading. A local normal (non-emergency) start is also possible; however, synchronized breaker closings must still be performed remotely. In addition to the trips enabled in the emergency mode, the following trips are enabled in the normal mode:

Engine A/B Lube Oil Pressure LO-LO
Engine A/B Jacket Water Pressure LO
Engine A/B Jacket Water Temperature HI-HI
Engine A/B Crankcase Pressure HI
Generator Overcurrent
Generator Negative Sequence
Generator Loss of Field
Generator Neutral Overvoltage
Generator Reverse Power

A local or remote emergency stop is provided which will immediately stop the engine and trip the output breaker. A local or remote normal stop can also be performed, if the diesel generator is in the normal mode, which will trip the generator feed breaker and run the engine at idle speed for a 10 minute cooldown before stopping.

A four-position control switch ("LOCKOUT/REMOTE/LOCAL/PLC BY-PASS") at panel 2202(3)-105 controls some of the generator set functions. In the LOCKOUT position, the engine starting air solenoids are prevented from being energized. This switch position is used when taking the generator set out-of-service. In the REMOTE position, normal starting and stopping of the generator set can only be done from the main control room. In the LOCAL position, normal starting and stopping can only be done from the SBO Building. Emergency starting or stopping can be done either remotely or locally regardless of the switch position in LOCAL or REMOTE. In the PLC BYPASS position, critical engine controls (starting, stopping, non-bypassed trips) which otherwise are provided by the PLC, are bypassed in favor of hard-wired control circuitry. Only a local emergency start is possible and the start pushbutton must be depressed continually until the starters disengage automatically above 200 rpm.

Control of generator loading via voltage and speed controls can be done remotely or locally regardless of the REMOTE/LOCAL switch position.

9.5.9.5 Remote Control of SBO Diesel Generators

The SBO system is designed to provide remote operations from the main control room. These operations include start and control of the diesel generators and auxiliaries, manipulation of the 4 kv connections, and loading of the diesel generators. This remote operation is accomplished via a Distributed Control System (DCS). The diesel generators can be manually started and manually loaded from the DCS and main control room panels.

9.5.10 References

1. EGC, "Dresden Units 2 and 3 Fire Protection Reports, Volumes 1 through 5, and Fire Protection Program Documentation Package, Volumes 1 through 13," Current Amendment.

2. QAL 12-87-197 Quality Assurance, Dresden Station, August 28, 1987.

3. Deleted

4. Integrated Plant Safety Assessment Final Report, Systematic Evaluation Program, Dresden Nuclear Power Station, Unit 2, NUREG-0823, Supplement No. 1, October 1989, Section 2.3.2.

Table 9.5-1

TABLE INTENTIONALLY DELETED

Table 9.5-2

MINIMUM STANDBY LIGHTING REQUIREMENTS

Location	Illumination (fc)	
	AC Power Source	DC Power Source
Control room	10	10
Safety-related equipment and control areas	3	None
Standby ac-equipment areas	3	3 (local)
Access routes	2	Silhouette

9.6 CONTROL OF HEAVY LOADS

9.6.1 Introduction/Licensing Background

In July of 1980 the NRC issued NUREG 0612 entitled "Control of Heavy Loads at Nuclear Power Plants" (Reference 1). All Nuclear Power Stations (NPS) were required to respond to NRC concerns for Phase I (NUREG-0612, Section 5.1.1) and Phase II (all other Sections). ComEd (CE) engaged in a series of submittals/correspondence with the NRC for NUREG 0612, Phase I (References 2 through 6) and Phase II (Reference 7).

The NRC accepted Dresden's Unit 2 and Unit 3 (Dresden) Phase I Commitments to NUREG 0612 on July 11, 1983 per SER / TER-C5506-350/351, June 2, 1983 and Revised June 8, 1983 as developed by the Franklin Research Center (Reference 8).

June 28, 1985 the NRC suspended all reviews of NUREG 0612, Phase II submittals under General Letter 85-11 (Reference 9).

On April 11, 1996 the NRC issued BULLETIN 96-02: "Movement of Heavy Loads Over Spent Fuel, Over Fuel in the Reactor Core, or Over Safety-Related Equipment" (Reference 10). All NPS were required to respond. ComEd responded May 14, 1996 by stating that activities were within the current licensing basis (Reference 11). The NRC accepted ComEd's response for this Licensing Action on May 20, 1998 (Reference 12).

On March 2, 2004 the NRC granted certain heavy load licensing amendments for inclusion in Dresden's UFSAR (Reference 13), which were incorporated into UFSAR Section 9.1.

In July of 2008 the Nuclear Energy Institute (NEI) issued, "Industry Initiative on Control of Heavy Loads" under the NEI 08-05 (Reference 26). The purpose of this document was "to address NRC staff concerns regarding the interpretation and implementation of regulatory guidance associated with heavy load lifts." The intent of NEI 08-05 was to increase NPS industry consistency in the interpretation of NUREG-0612. NEI 08-05 was endorsed by the NRC on December 1, 2008 under RIS 2008-28 (Reference 27).

9.6.2 Safety Basis

Refer to UFSAR Section 9.6.1 for a description of license commitments to Phase I of NUREG 0612, "Control of Heavy Loads at Nuclear Power Plants."

The main hoist of the reactor building overhead crane has a 125-ton capacity and is designated as a single failure proof crane for 110-ton loads. In addition, the NRC has approved use of the reactor building overhead crane during power operations to lift a total load up to 116-tons for removal and installation activities for the reactor shield blocks (Reference 16). Refer to UFSAR Sections 9.1.4.2.2, 9.1.4.3.2 and 9.1.4.4.2 for a description of the reactor building overhead crane.

Specific heavy loads, weights, handling devices and associated procedures are identified in station procedures (Reference 14).

Refer to UFSAR Section 9.1.4.3.2 for a description of the load drop analyses in the reactor building (References 15 through 18), which includes a drop of reactor shield blocks weighing up to 116-tons at a maximum height of 1'-0" above the floor.

Standby Gas Treatment (SBT), Containment Cooling Service Water (CCSW), and Emergency Diesel Generator (EDG) are safety related systems in the Turbine Building. The Turbine Building Cranes are not designated single failure proof. Analyses are performed when appropriate, including postulated load drops in the Turbine Building (References 18 through 24).

The movements of heavy loads are controlled by station procedures within the licensing basis.

9.6.3 Scope of Heavy Load Handling Systems

The following heavy load handling systems were evaluated and found acceptable to the NRC during the NUREG-0612, Phase I submittal process (Reference 8).

- Units 2/3 Reactor Building Overhead Crane
- Unit 2 Refuel Floor Hatchway Jib Crane
- Unit 3 Refuel Floor Hatchway Jib Crane
- Unit 2 Refuel Platform Hoist
- Unit 3 Refuel Platform Hoist
- Units 2/3 New Fuel Storage Vault Jib Crane
- Unit 2 Reactor Building Hatchway Jib Crane
- Units 2/3 Reactor Service Platform Jib Crane
- Unit 2 Turbine Building Overhead Crane
- Unit 3 Turbine Building Overhead Crane

In 2003 the NRC reaffirmed the Units 2/3 Reactor Building Overhead Crane (RBOC) is qualified single-failure-proof for lifting heavy loads per NUREG-0554 and NUREG-0612 during seismic events without backfitting either the RBOC or the Reactor Building to meet modern day codes. Furthermore, a 7.798 factor of safety for the wire rope of the RBOC is acceptable (Reference 25).

Refer to UFSAR Section 9.1.2.2.4 for a description of spent fuel storage and handling of Dry Cask Storage (DCS) systems and Independent Spent /Fuel Storage Installation (ISFSI).

Refer to UFSAR Section 9.1.4 for a description of the Fuel Handling System.

9.6.4 Control of Heavy Loads Program

The Control of Heavy Loads Program consists of the following:

A. Refer to UFSAR Section 9.6.1 for a description of license commitments to Phase I elements of NUREG 0612, "Control of Heavy Loads at Nuclear Power Plants."
B. Reactor Pressure Vessel Head (RPVH) and Spent Fuel Cask (SFC) lifts are made with the reactor building overhead crane, which is designated as single failure proof for 110-ton loads. The RPVH and SFC weigh less than 110-tons (Reference 14).
C. The NRC has approved use of the reactor building overhead crane during power operations to lift a total load up to 116-tons for removal and installation activities for the reactor shield blocks (Reference 13).
D. The effects of load drops are analyzed to verify that lift height limitations allow floors to act as protective ceilings OR protective barriers are in place OR damage to safe shutdown equipment would not preclude operation of sufficient equipment to achieve safe shutdown (References 13 through 24).

E. Configuration management activity considerations and administrative controls established under maintenance Rule 10 CFR 50.65(a)(4) "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants" is an acceptable means of addressing the movement of heavy loads (References 26 through 28).

9.6.4.1 Dresden Commitments in Response to NUREG 0612, Phase I Elements

1. Refer to UFSAR Section 9.1.4.3.2 for a description of safe load paths for Reactor Shield Plugs, Reactor Pressure Vessel Heads and Spent Fuel Casks. The entire reactor building refueling floor (with the exception of the fuel pool and open reactor cavity) is considered a safe load path zone.
2. Technical Specifications and Station procedures prohibit movement of heavy loads over the spent fuel pools or open reactor cavity except under Special Procedures according to UFSAR Section 9.1.4.3.2.
3. Crane operators are qualified, trained and conduct themselves in accordance with ANSI B30.2-1976 per station procedures.
4. Special lift handling devices are designed to comply with ANSI, N14.6-1978 per station procedures.

   Exceptions:
   The NRC has granted certain exceptions to ANSI, N14.6-1978 as documented in their endorsement letter dated July 11, 1983 of TER-C5506-350/351 (Reference 8). Stress design safety factors of 3 for minimum yield strength and 5 for ultimate strength were used in the design of the reactor head strongback, moisture separator hook box, and dryer/separator lifting rig. In addition, Section 2.1.5.a.of TER C5506 350/351 indicate load tests have been performed on these lifting devices to the weights specified:

   1. reactor head strongback - 129 tons (129% of load)
   2. moisture separator hook box - 129 tons (180% of load)
   3. dryer/separator lifting rig - 130 tons (180% of load)

   For the dryer/separator lifting rig, the calculated value of the ratio of ultimate strength to rated load is greater than 5 for all components except for the weld of the lug to the W6x25 beam (value 2.17) and the weld of the lug to the hook box (value 2.6).

5. Slings “not specifically designed” comply with ANSI/ASME B30.9-2010 per station procedures.
6. Periodic crane inspections, testing and maintenance are conducted per UFSAR Section 9.1.4.4 and station procedures.
7. The reactor building crane design is designated single failure proof and meets the requirements of ANSI B30.2 and CMAA-70 per UFSAR Section 9.1.4.1. In addition, the reactor building overhead crane and spent fuel cask yoke assemblies meet the intent of NUREG-0554 for loads less than or equal to 110 tons per UFSAR Section 9.1.4.3.2.

9.6.4.2 Reactor Pressure Vessel head (RPVH) Lifting Procedures

Refer to UFSAR Sections 9.1.4.2.2, 9.1.4.3.2 and 9.1.4.4.2 for a description of the reactor building overhead crane and RPVH handling procedures.

9.6.4.3 Single Failure Proof Cranes for Spent Fuel Casks

Refer to UFSAR Sections 9.1.4.2.2, 9.1.4.3.2 and 9.1.4.4.2 for a description of the reactor building overhead crane and Dry Cask Storage (DCS) system handling procedures.

9.6.4.4 Turbine Building Overhead Cranes

The turbine building deck at elevation 561'-6" is serviced by the Unit 2 and Unit 3 turbine building overhead cranes. The Unit 2 crane is equipped with a 125-ton main hoist and a 10-ton auxiliary hoist. The Unit 3 crane is equipped with a 175-ton main hoist and a 25-ton auxiliary hoist. Each main and auxiliary hoist is operated independently with its respective set of controls.

Each turbine building overhead crane handling system consists of an overhead, bridge-type crane, trolley, and controls. They are used to lift and move tools, equipment and components accessible from the turbine deck floor and equipment hatches. The overhead cranes are located in a controlled environment in the turbine building.

The Unit 2 and Unit 3 turbine building overhead cranes are classified as Non-Safety equipment and are not seismically qualified. Neither crane nor their respective hoists are rated single failure proof or equivalent.

The effects of load drops in the turbine building use the following key assumptions and methods.

- Load drop analyses conform to the guidelines of NUREG 0612, Appendix A.
- The orientation of a dropped heavy load may be controlled when there is single failure proof rigging below the hook.
- During postulated scabbing/spalling events the damaged concrete on the underside of concrete slabs/floors falls vertically.
- Credit may be taken for shields designed to provide load drop protection.
- Credit may be taken for electrical cable tray covers and for overhead cable trays during postulated concrete spalling events. It is assumed that covers and top trays do not protect lower cable trays during postulated floor penetration events.
- Pipes that bend or dent due to scabbed/spalled concrete are considered functional. It is assumed that pipes are ruptured during postulated floor penetration events.
- Heavy loads may be moved as a configuration management activity with administrative controls in accordance with Maintenance Rule 10 CFR 50.65(a)(4).

9.6.5 Safety Evaluation

Controls implemented by NUREG 0612 Phase I elements make the risk of a load drop very unlikely.

In the event of a postulated load drop, the consequences are acceptable, as demonstrated by load drop analyses. Restrictions on load height, load weight and medium under the load are reflected in plant procedures and engineering change evaluations (EC-Eval).

The use of a single failure proof crane in the reactor building makes the risk of a load drop extremely unlikely and acceptably low.

The risk associated with the movement of heavy loads per NEI 08-05 guidelines and Maintenance Rule 10 CFR 50.65(a)(4) considerations are evaluated and controlled by station procedures.

9.6.6 References

1. D.G. Eisenhut (NRC), Letter to All Licensees. Subject: NUREG 0612 initiated, December 22, 1980.

2. E.D. Swartz (CE), Letter to D.G. Eisenhut (NRC). Subject: Initial Phase I response to NUREG 0612, June 22, 1981.

3. E.D. Swartz (CE), Letter to D.G. Eisenhut (NRC). Subject: Submittal of "Change-Out" pages only in response to NUREG-0612, December 11, 1981.

4. E.D. Swartz (CE), Letter to D.G. Eisenhut (NRC). Subject: Response to concerns raised in TER Draft, May 4, 1982.

5. E.D. Swartz (CE), Letter to D.G. Eisenhut (NRC). Subject: Supplemental response to concerns raised in TER Draft, November 18, 1982.

6. B. Rybak (CE), Letter to D.G. Eisenhut (NRC). Subject: Final Phase I completion in response to NUREG-0612, May 19, 1983.

7. E.D. Swartz (CE), Letter to D.G. Eisenhut (NRC), Subject: Supplemental response to NUREG-0612, September 22, 1981.

8. D.M. Crutchfield (NRC), Letter to D.L Farrar (CE). Subject: Final Phase I acceptance NUREG-0612, July 11, 1983. Attachments SER, July 11, 1983 and TER-C5506-350/351, June 2, 1983 and Revised June 8, 1983 developed by the Franklin Research Center were enclosed.

9. H.L. Thompson, Jr. (NRC), Letter to All Licensees. Subject: Completion of Phase II, NUREG 0612 (GL 85-11), June 28, 1985.

10. NRC Bulletin 96-02 to all Licensees. Subject: Movement of Heavy Loads Over Spent Fuel, Over Fuel in the Reactor Core, or Over Safety-Related Equipment, April 11, 1996.

11. J.B. Hosmer (CE), Letter to the NRC. Subject: Response to NRC Bulletin 96-02, May 13, 1996.

12. R.M. Pulsifer (NRC), Letter to O.D. Kingsley (CE). Subject: Completion of Licensing Action for NRC Bulletin 96-02, May 20, 1998.

13. M. Benerjee (NRC), Letter to C. R. Crane (Exelon). Subject: Dresden Unit 2 and Unit 3 Licensing Amendments Nos. 204 and 196 for Facility Operating Licenses Nos. DPR-19 and DPR-25 respectively, March 2, 1998.

14. Procedure DMP 5800-18 (Rev 28): Load Handling of Heavy Loads and Lifting Devices.

15. Calculation DRE02-0064 (Rev 0, 0A, 0B): D2/3 Load Drop Evaluation of the Reactor Shield Plugs.

16. M. Banerjee (U.S. NRC) letter to J. L. Skolds (Exelon Generation Company, LLC), "Dresden Nuclear Power Station, Units 2 and 3 – Issuance of Amendments – Heavy Loads Handling (TAC Nos. MB7840 and MB7841)," dated October 10, 2003.

17. Calculation 8.31.0-4 (Rev 0, 1): Load Drop Evaluation in the Hatchway of the Reactor Building.

18. Calculation 8.31.0-2 (Rev 0, 1, 2, 3): Load Drop Analysis for Reactor and Turbine Buildings.

19. Calculation DRE99-0077 (Rev 0): Evaluation of a Shipping Container Drop from the Turbine Deck onto the Cable Tunnel Ceiling Slab.

20. Calculation DRE02-0046 (Rev 0): Evaluation of Turbine Services Sea Vans for Drop from the Turbine Deck onto the Cable Tunnel Ceiling.

21. Calculation DRE04-0026 (Rev 2): Structural Evaluations Associated with Moving the Unit 2 Generator Rotor.

22. Calculation DRE08-0022 (Rev 0, 1, 2, 3, 4, 5): Load Drop Evaluation on the Turbine Floor Elevation 561'6".

23. Calculation DRE08-0029 (Rev 0, 1): Load Drop Evaluation Associated with Heavy Load Movement in Support of Refueling Outage D3R20.

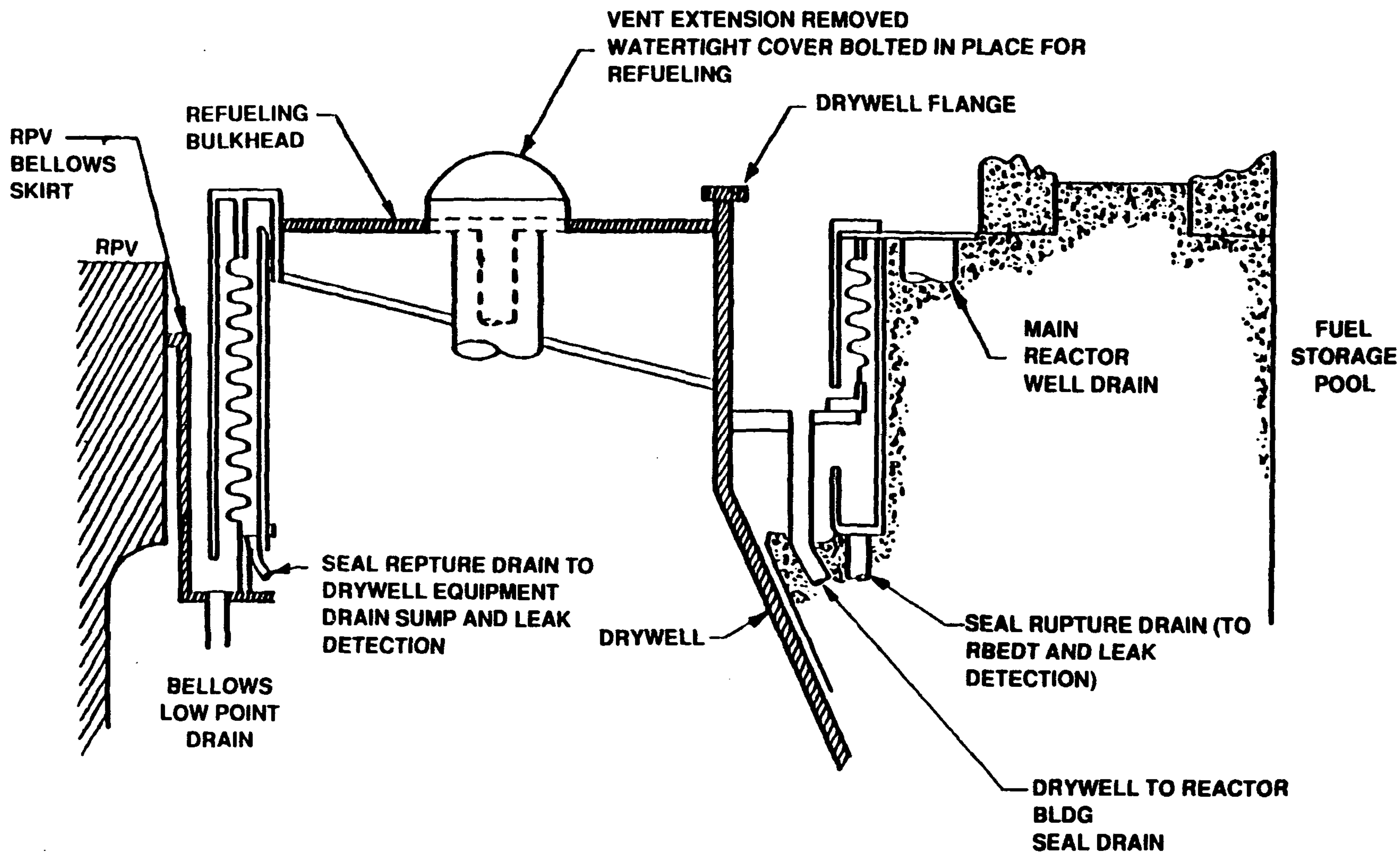
24. Calculation DRE09-0002 (Rev 1): Design of Protection for the Turbine Floor Center Court.

25. L. B. Marsh (U.S. NRC) letter to M. L. Dapas (Region III, NRC) "NRR Response to TIA 2001-13, Backfitting Requirements for Dresden Units 2 and 3 Reactor Building Crane (TAC Nos. MB3923 and MB3024)," dated February 21, 2003.

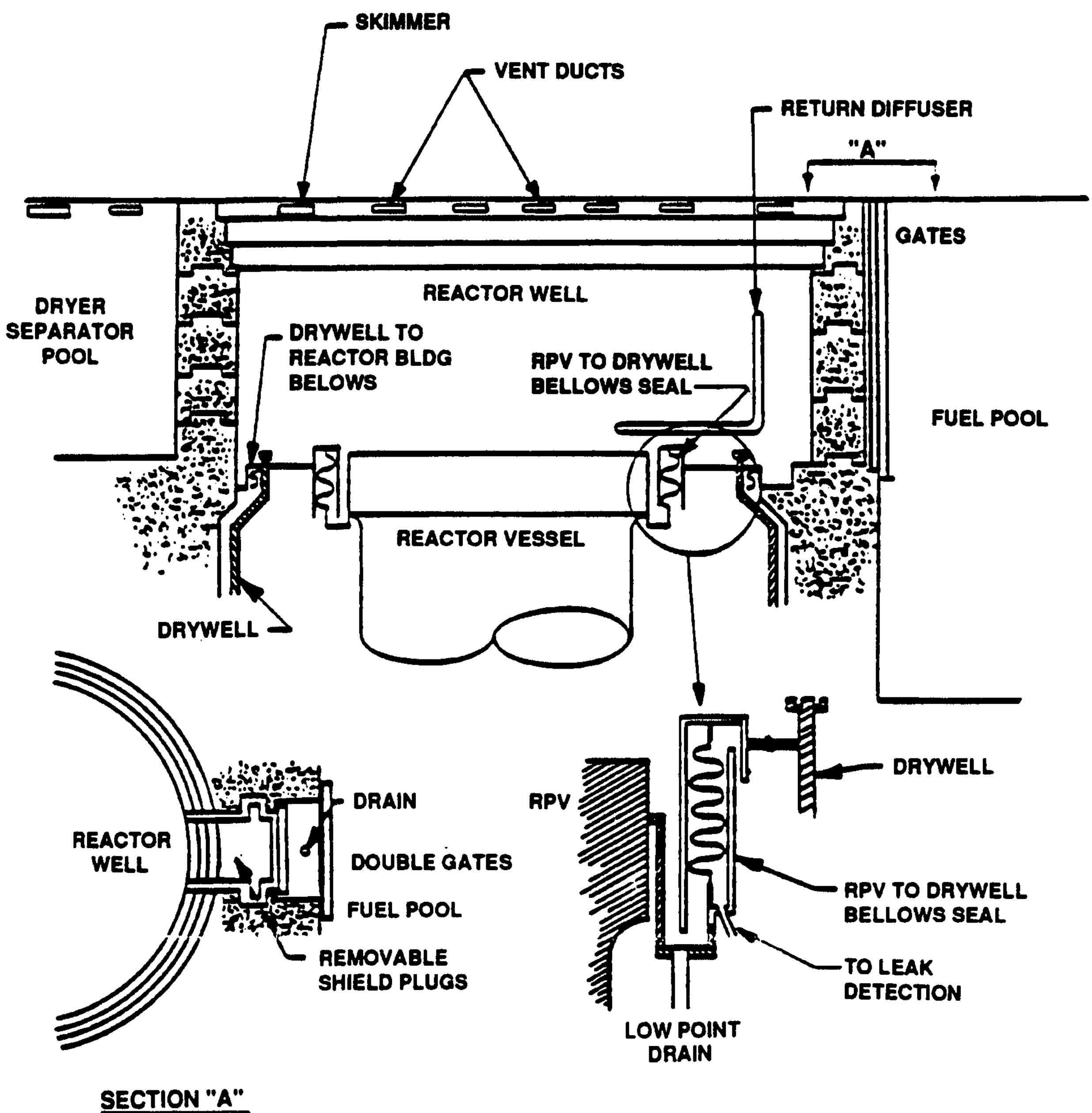
26. NEI 08-05 (Rev 0): "Industry Initiative on Control of Heavy Loads" dated July 2008.

27. "NRC Regulatory Issue Summary 2008-28 Endorsement of Nuclear Energy Institute Guidance for Reactor Vessel Head Heavy Load Lifts" (RIS 08-28) dated December 1, 2008.

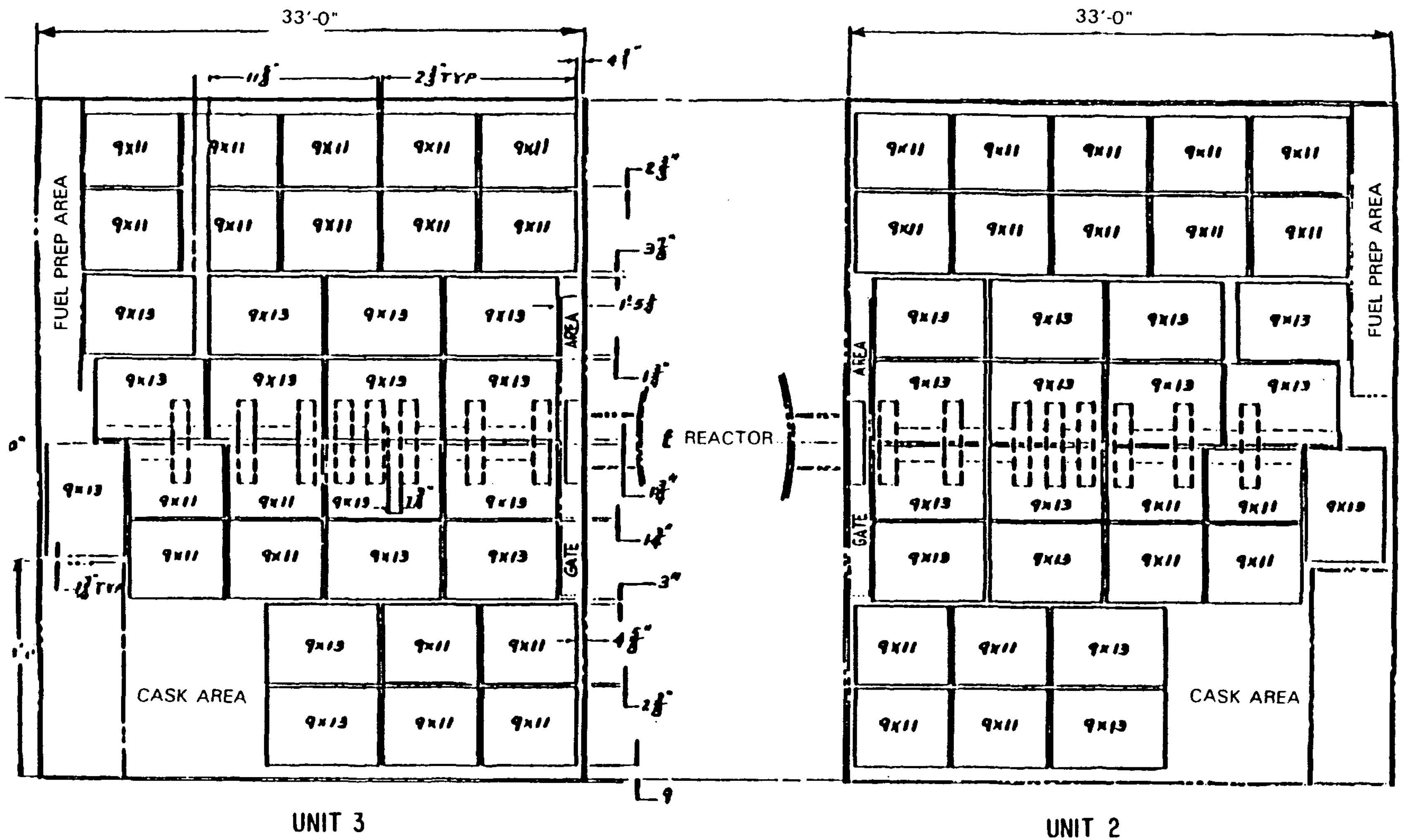
28. Maintenance Rule 10 CFR 50.65(a)(4): "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants."



DRESDEN STATION UNITS 2 & 3
REFUELING BULKHEAD AND ASSOCIATED BELLOWS (DETAILS)
FIGURE 9.1-1



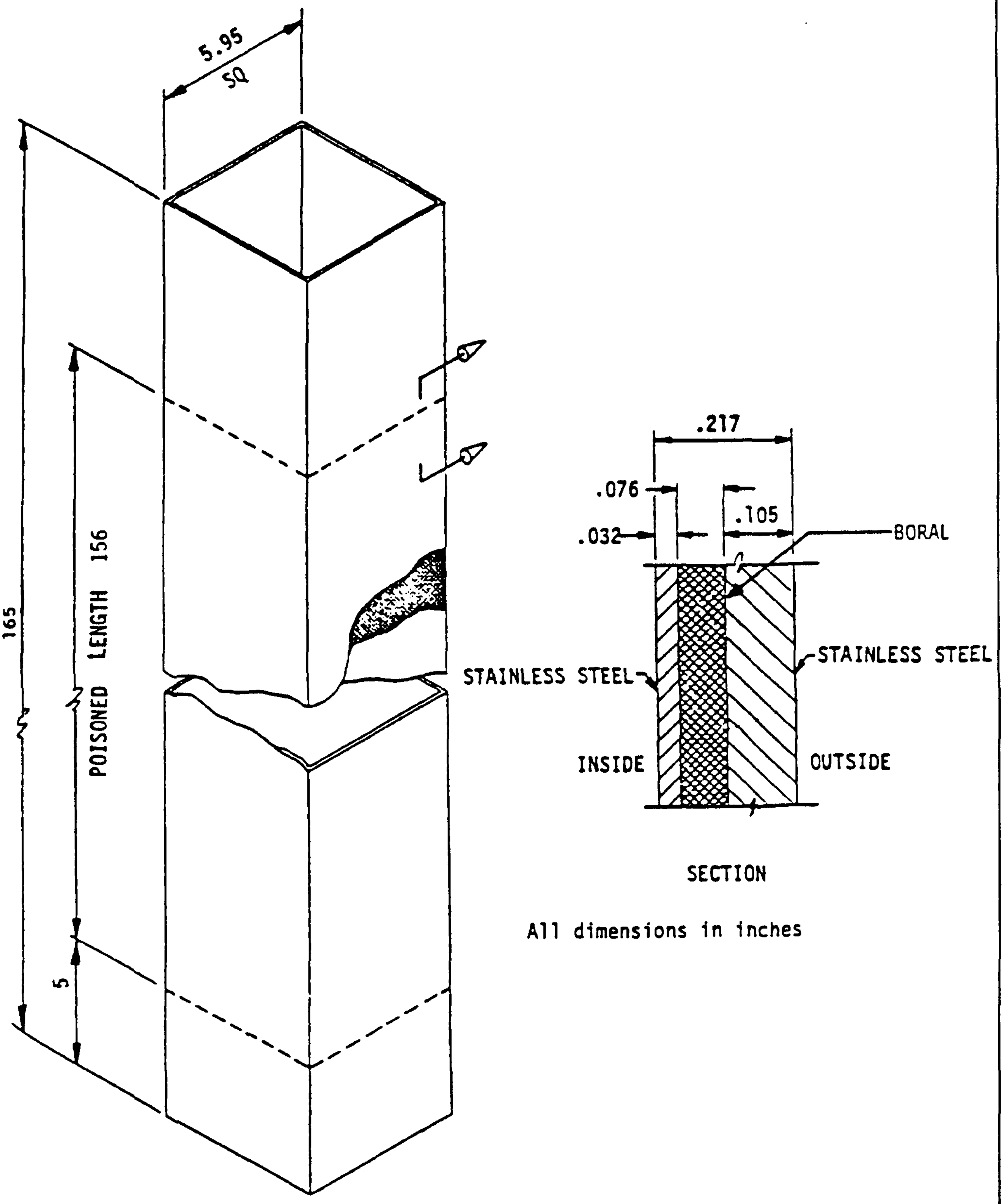
DRESDEN STATION UNITS 2 & 3
REFUELING BULKHEAD AND BELLOWS
FIGURE 9.1-2



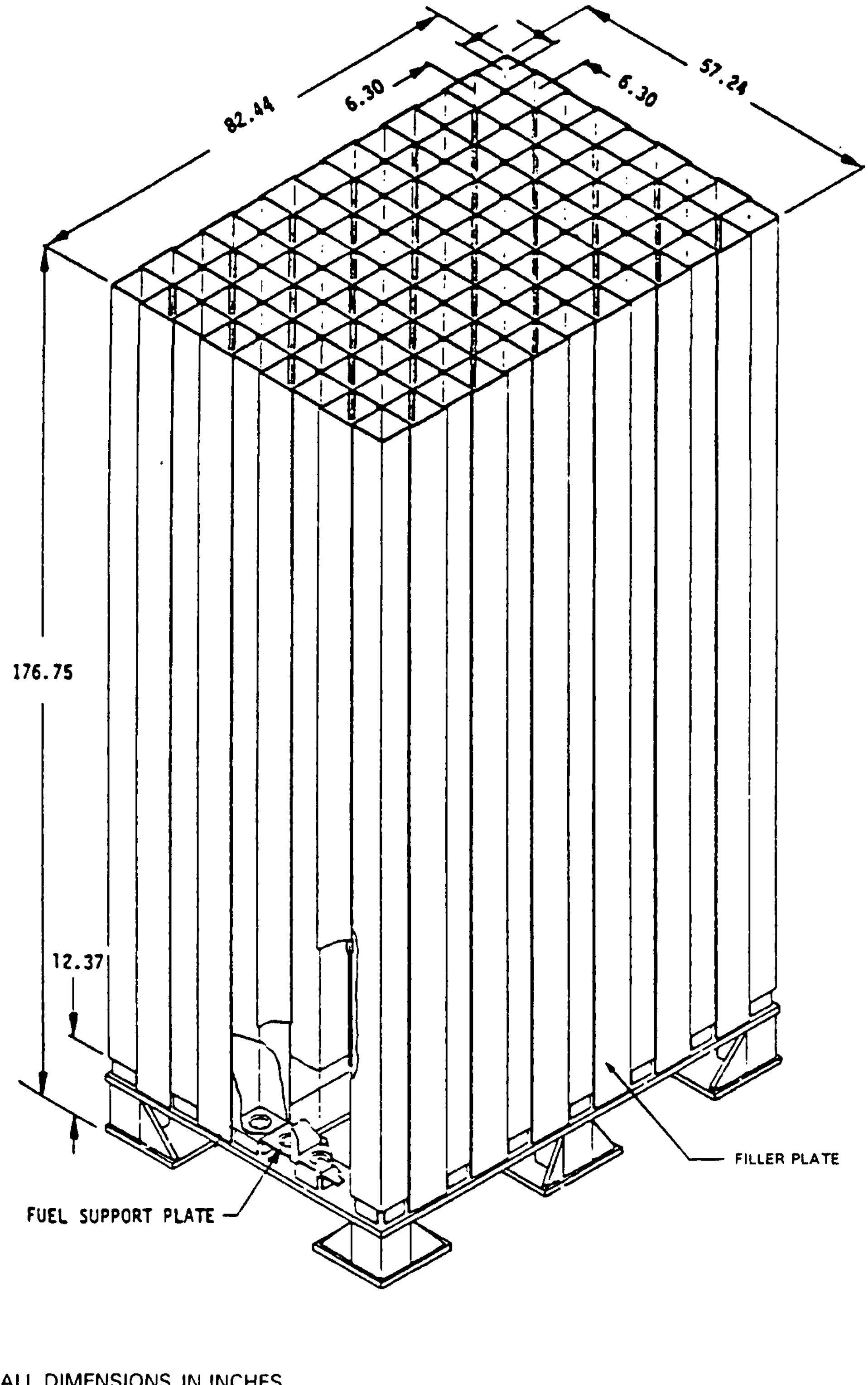
DRESDEN STATION
UNITS 2 & 3

SPENT FUEL STORAGE POOL
ARRANGEMENT

FIGURE 9.1-5

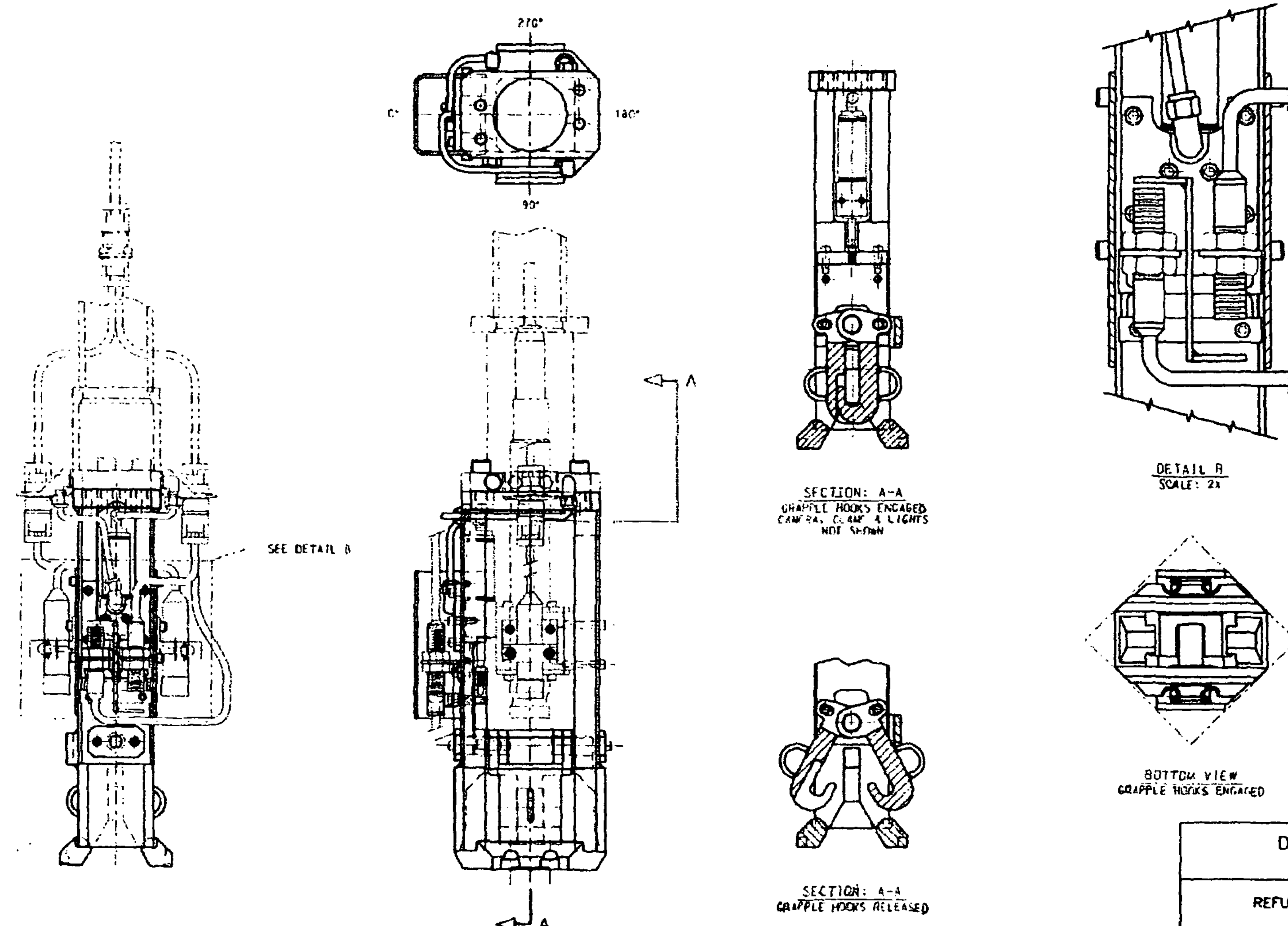


DRESDEN STATION UNITS 2 & 3
STAINLESS STEEL TUBE WITH BORAL CORE FOR HIGH DENSITY SPENT FUEL RACKS
FIGURE 9.1-6



ALL DIMENSIONS IN INCHES

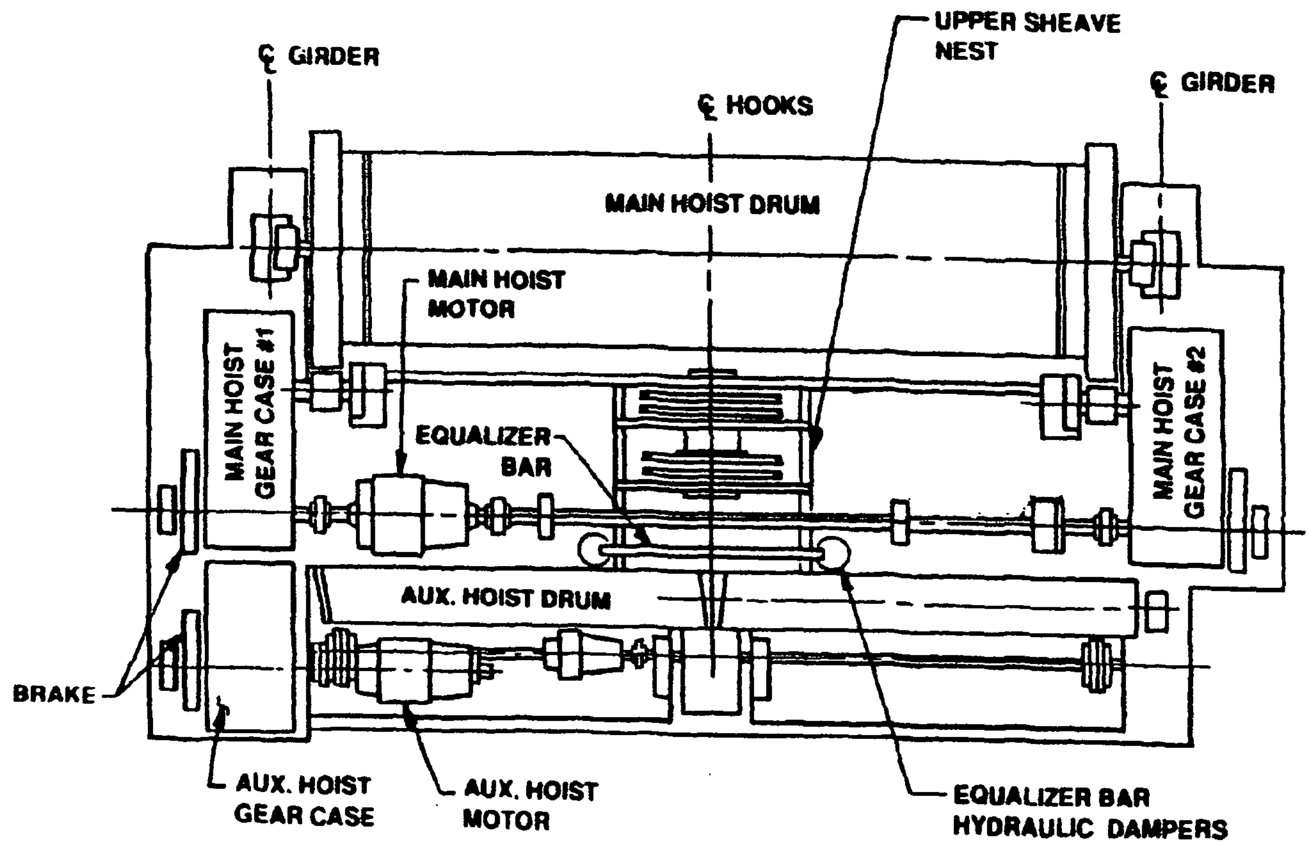
DRESDEN STATION UNITS 2 & 3
HIGH DENSITY SPENT FUEL RACK - 9 x 13
FIGURE 9.1-7



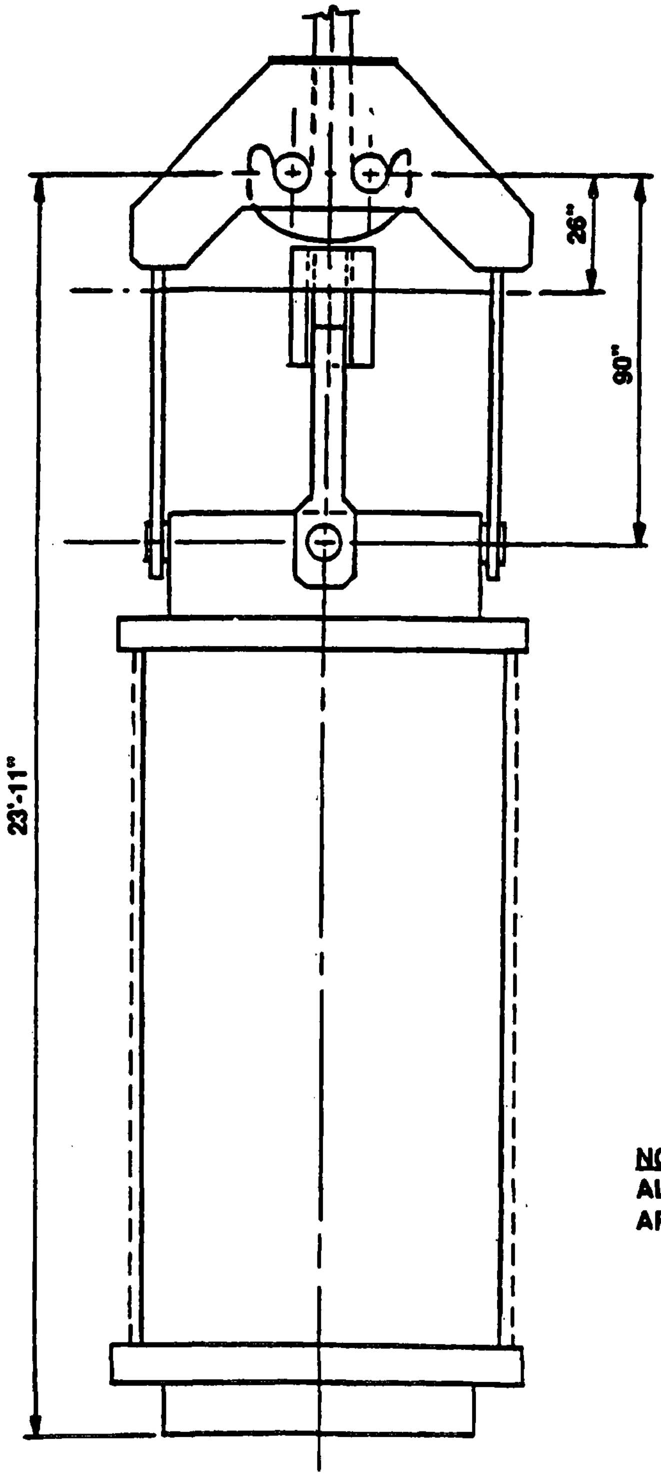
DRESDEN STATION
UNITS 2 & 3

REFUEL BRIDGE MAIN HOIST
GRAPPLE HEAD

FIGURE 9.1-16

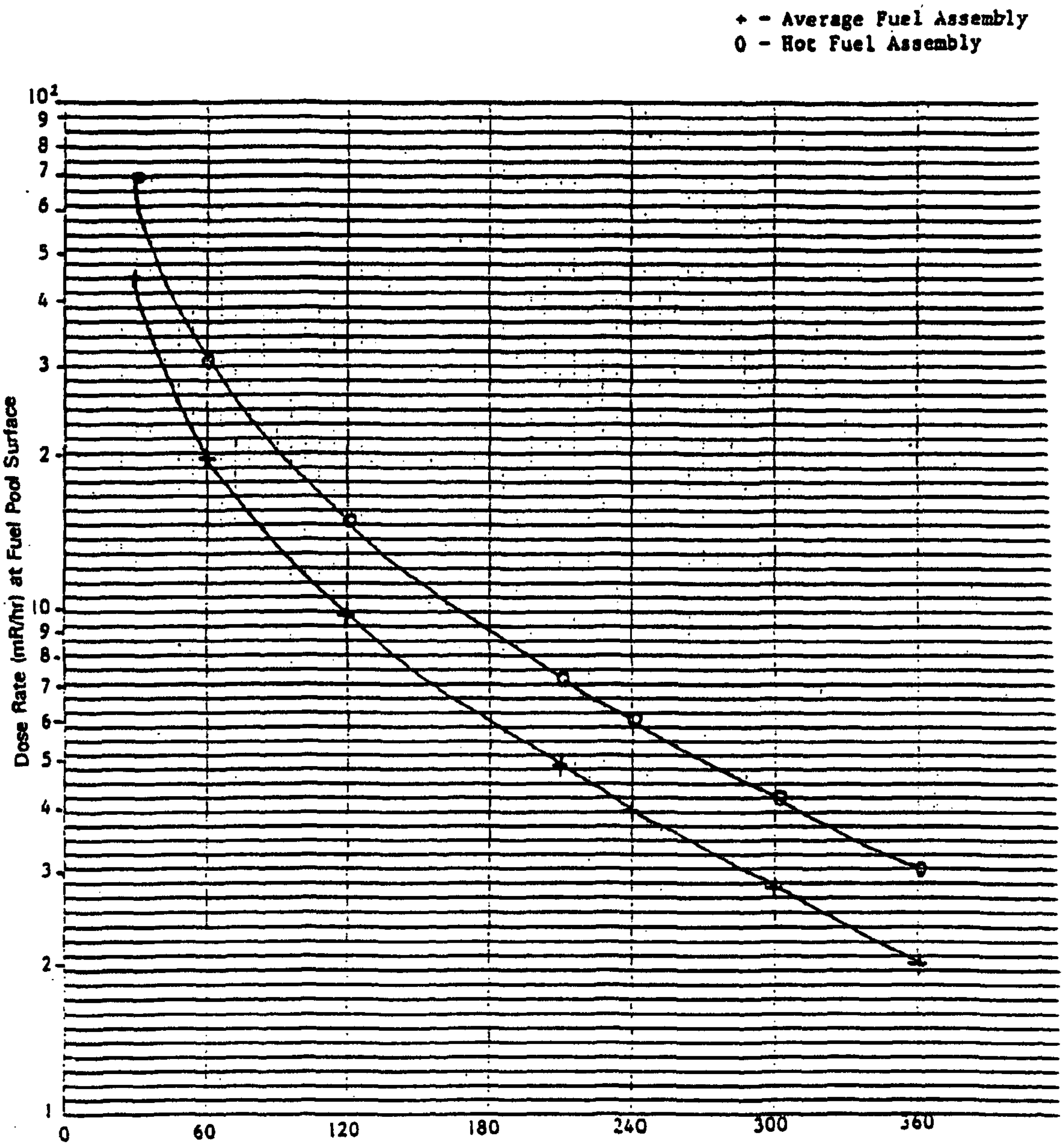


DRESDEN STATION UNITS 2 & 3
PLAN VIEW OF REACTOR BUILDING OVERHEAD CRANE TROLLEY WITH DUAL LOAD PATH MAIN HOIST
FIGURE 9.1-17
REV 7 JUNE 2007



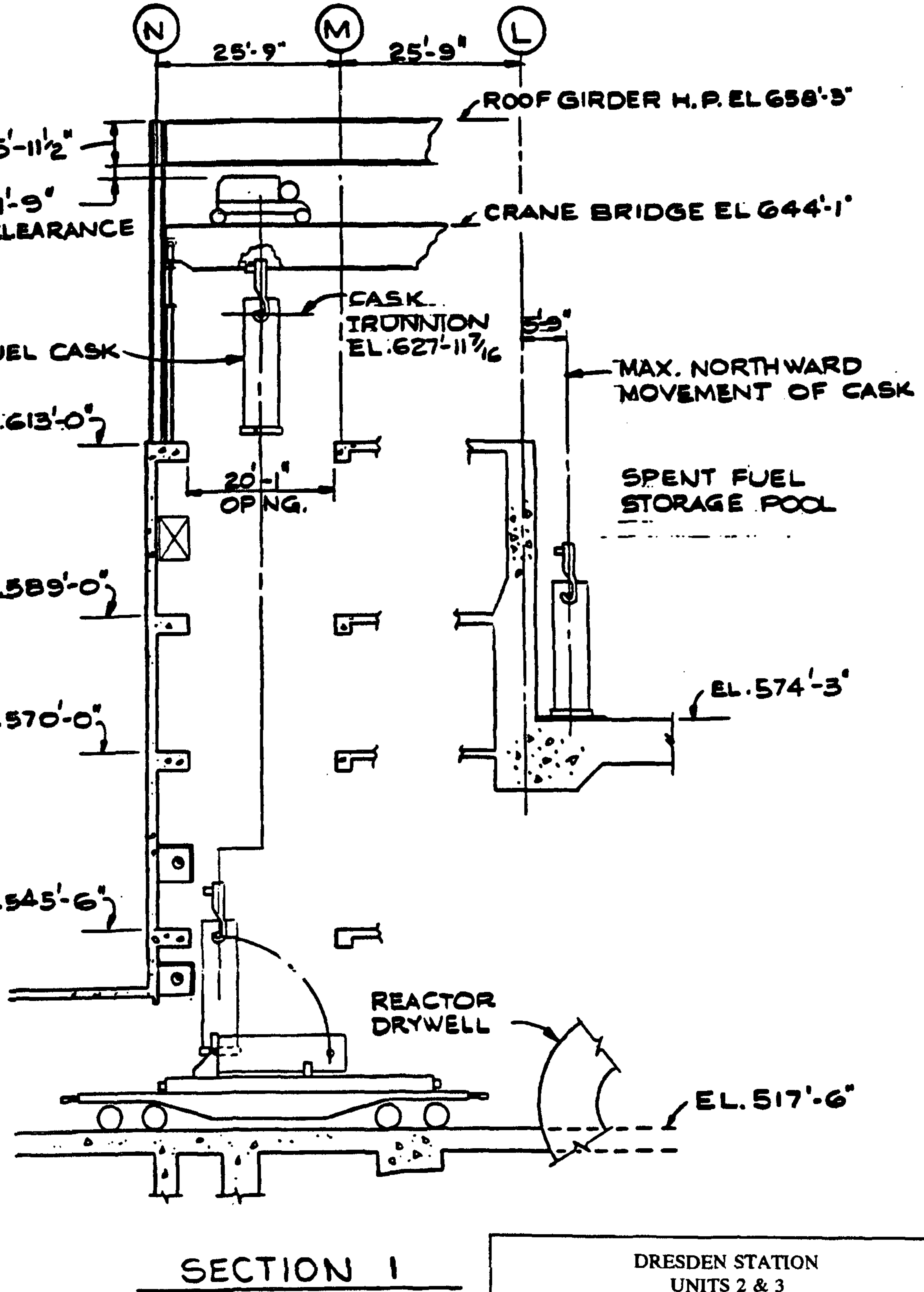
NOTE:
ALL DIMENSIONS
ARE APPROXIMATE

DRESDEN STATION UNITS 2 & 3
NL 10/24 FUEL CASK REDUNDANT LIFT RIG WITH REDUNDANT HOOK
FIGURE 9.1-18

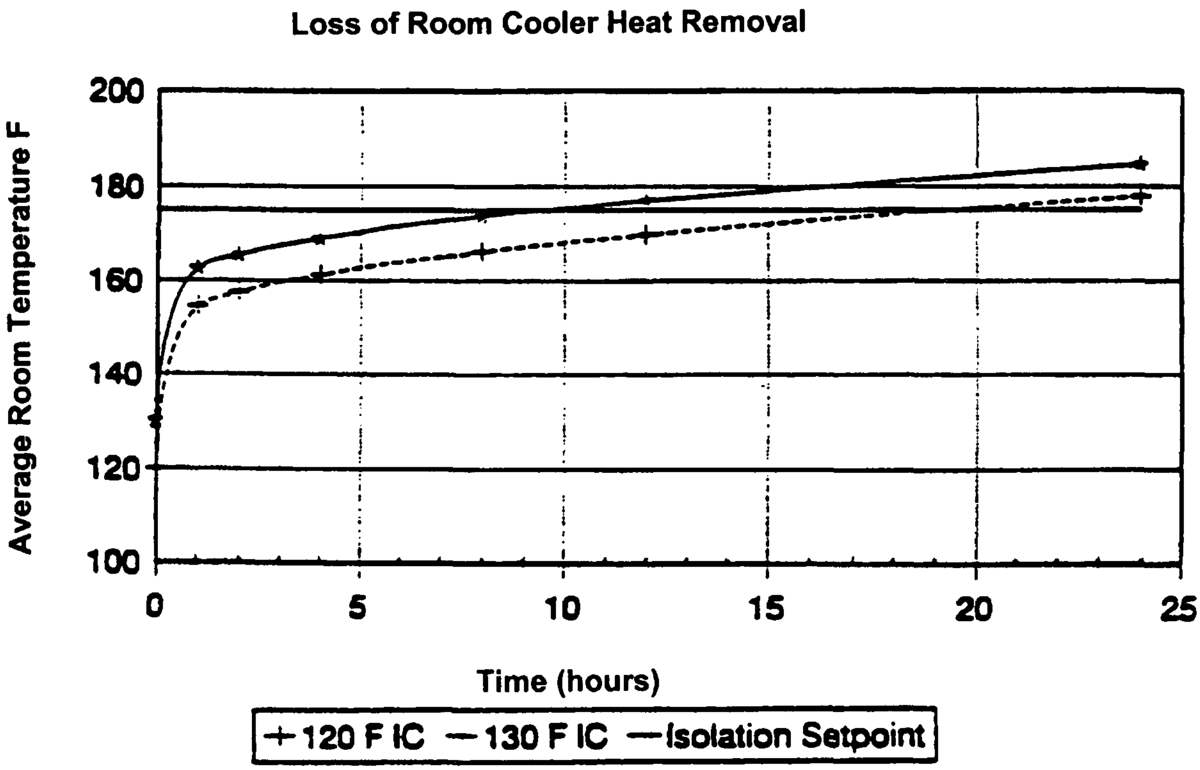


Note: For spent fuels from the uprated core, the dose rates would increase by approximately 20%.

DRESDEN STATION UNITS 2 & 3
DOSE RATE AT FUEL POOL SURFACE VS. AGE OF FUEL ASSEMBLY
FIGURE 9.1-19



DRESDEN STATION UNITS 2 & 3
NLI 10/24 FUEL CASK LOADING SECTION
FIGURE 9.1-20

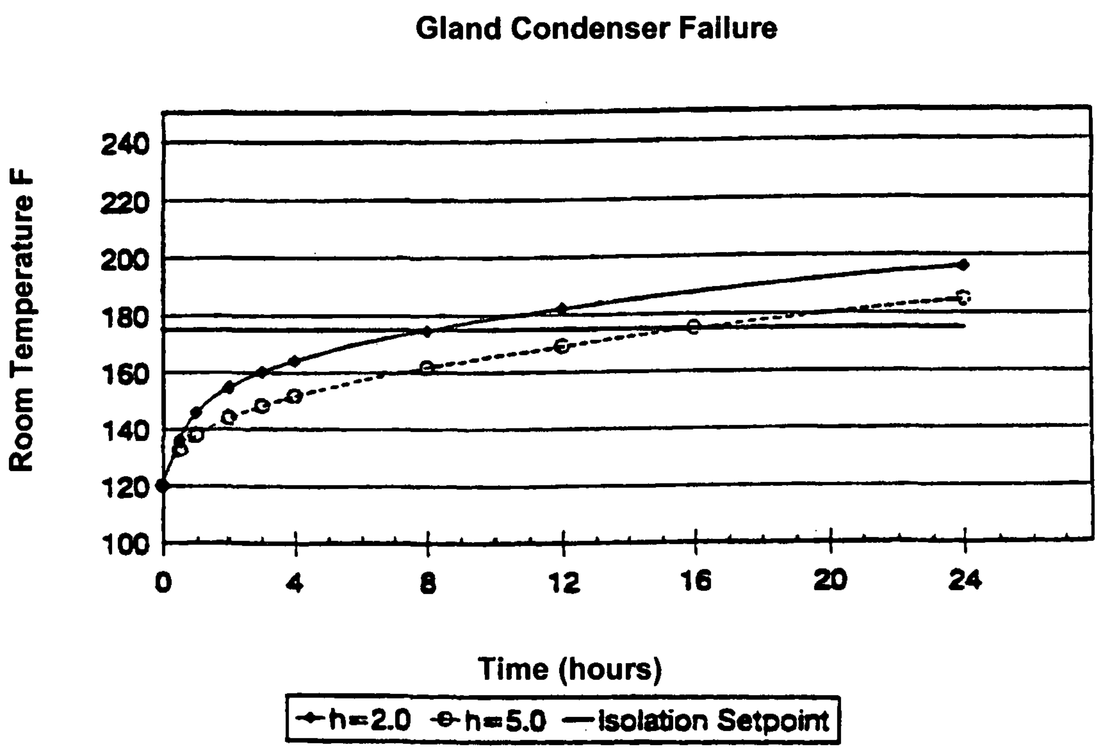


Minimum Temperature Limit (with uncertainty)

Evaluation performed at 2527 MWt.

DRESDEN STATION UNIT 3
HPCI ROOM THERMAL RESPONSE
FIGURE 9.2-5 REVISION 5, JANUARY 2003

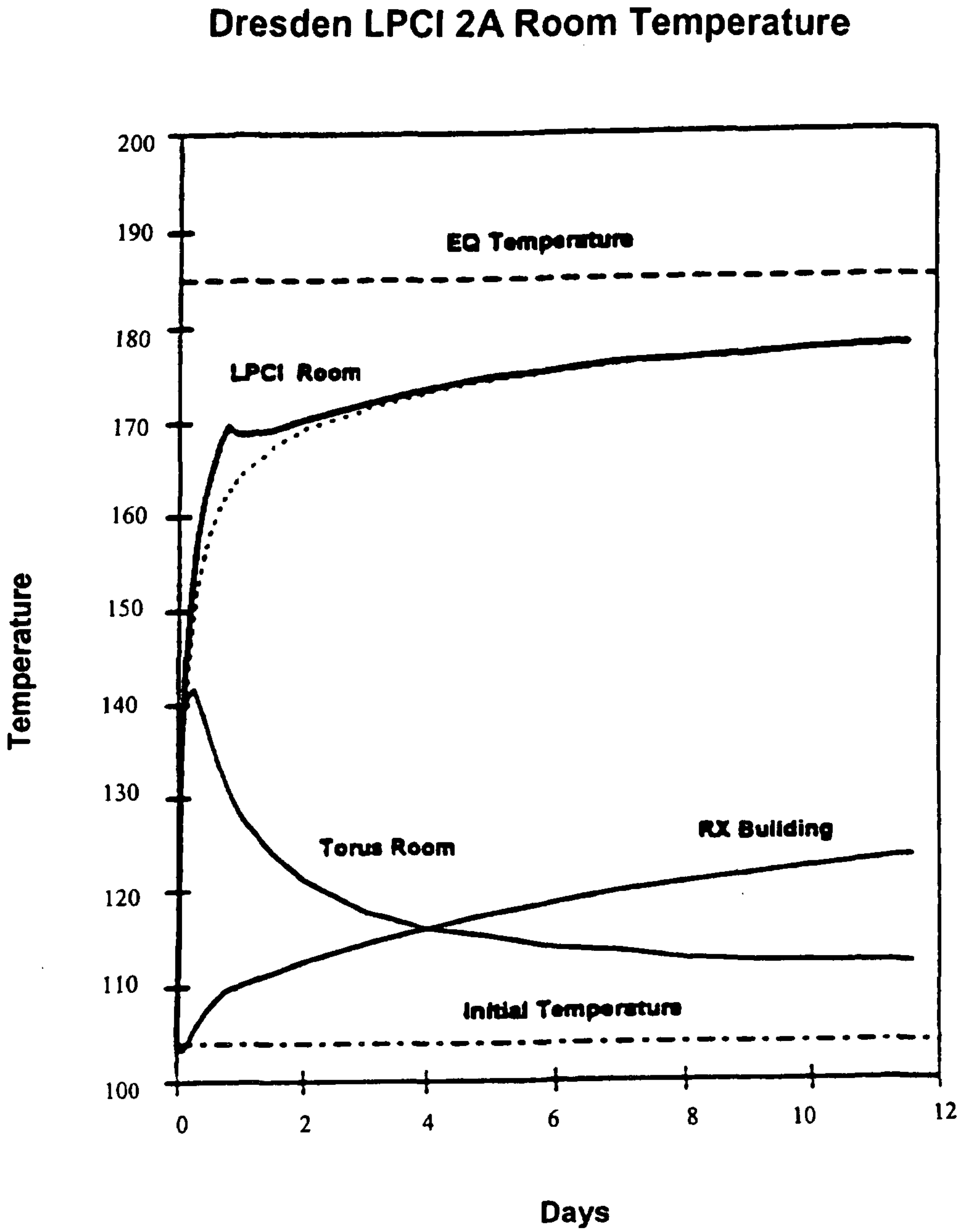
HPCI Room Thermal Response



216 lb/hr seal leakage

Evaluation performed at 2527 MWt.

DRESDEN STATION UNIT 3
HPCI ROOM RESPONSE TO GLAND SEAL CONDENSER LEAKAGE
FIGURE 9.2-6 REVISION 5, JANUARY 2003

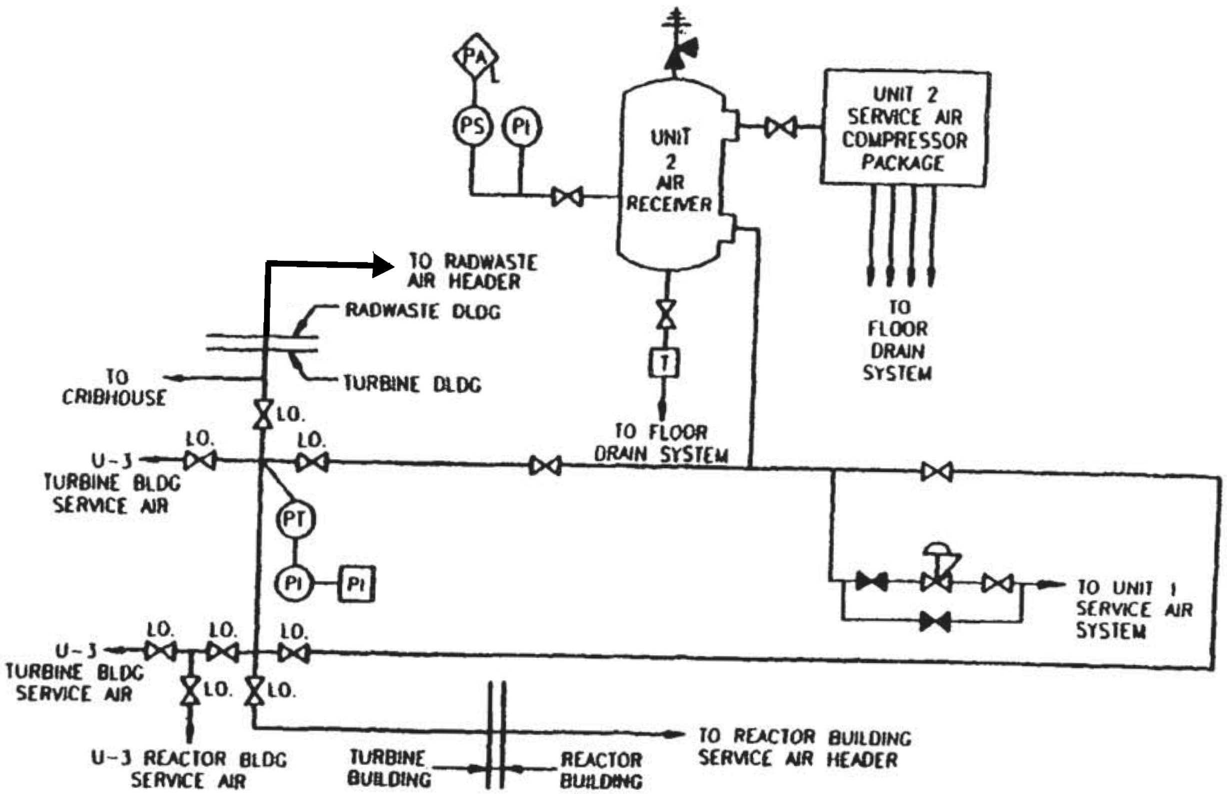


Evaluation performed at 2527 MWt.

DRESDEN STATION
UNIT 3

LPCI ROOM TEMPERATURE RESPONSE DUE TO LOSS OF ROOM COOLER

FIGURE 9.2-7
REVISION 5, JANUARY 2003

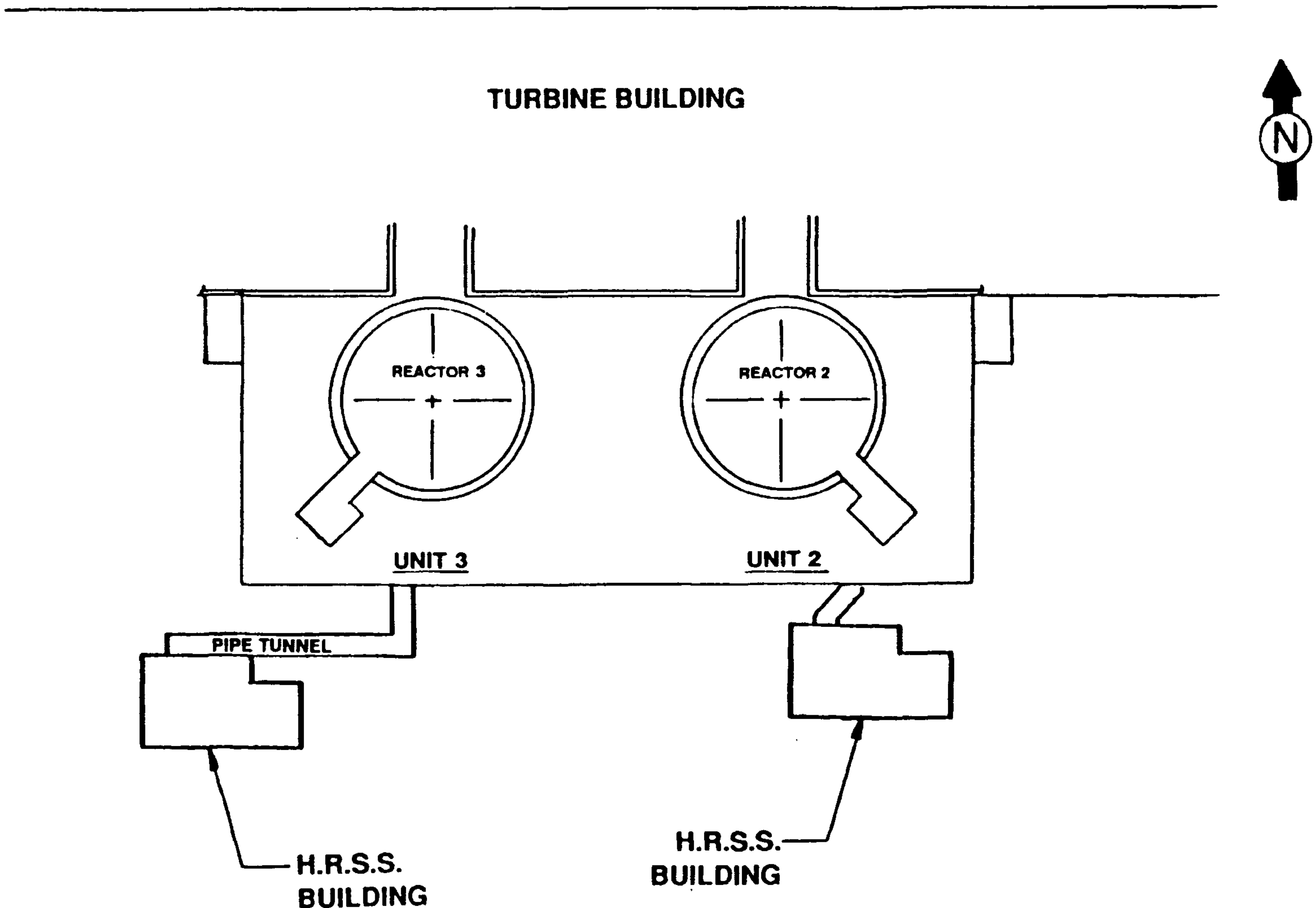


REVISION 12, JUNE 2017

DRESDEN STATION
UNITS 2& 3

SERVICE AIR SYSTEM
(TYPICAL OF UNIT 3)

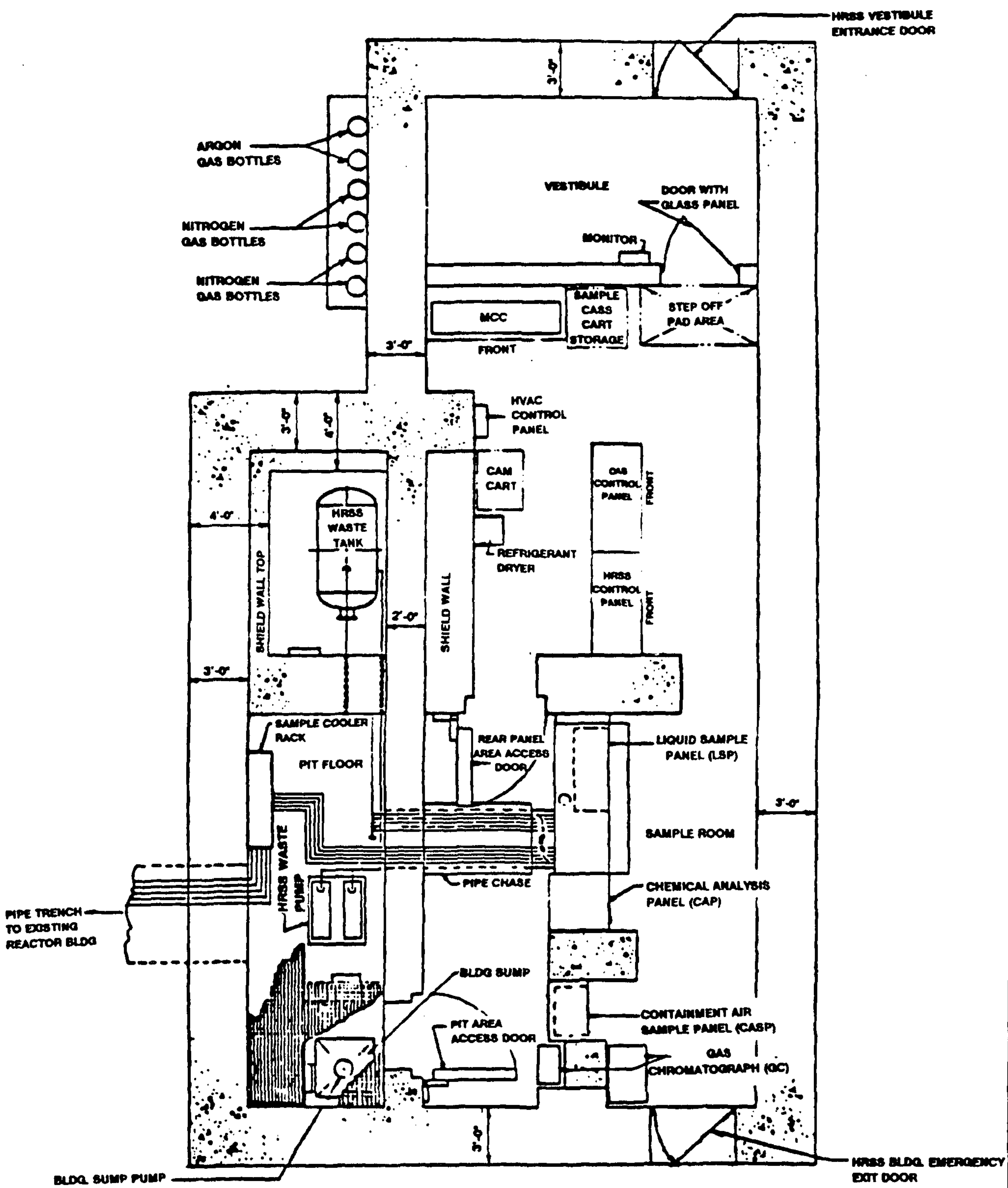
FIGURE 9.3-5



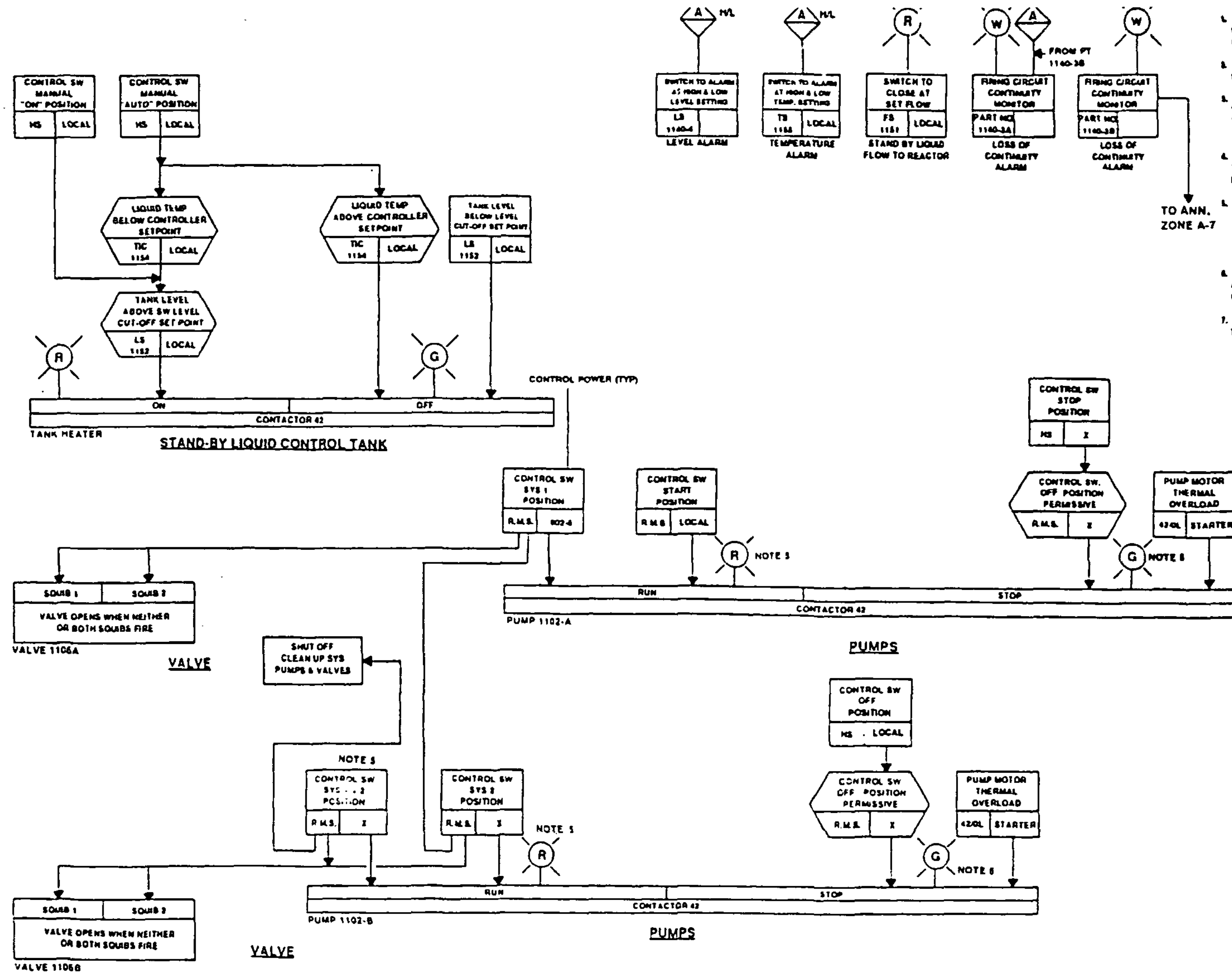
DRESDEN STATION
UNITS 2 & 3

LOCATION OF HRSS BUILDING
DRESDEN 2,3

FIGURE 9.3-6

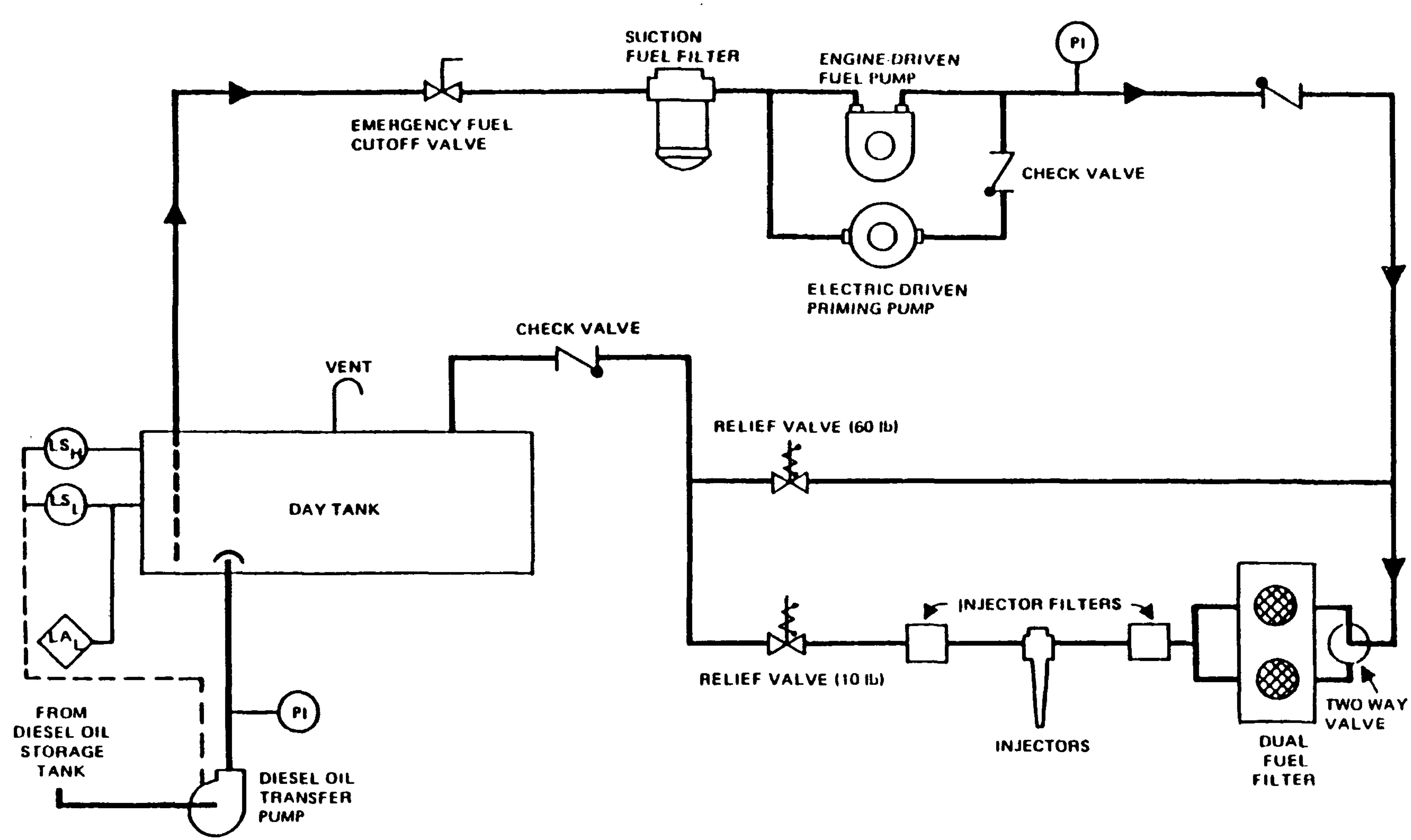


DRESDEN STATION UNITS 2 & 3
SHIELDING DESIGN FOR THE HRSS BUILDING
FIGURE 9.3-7



1. SYSTEM ELECTRICAL POWER SHALL BE EMERGENCY AC BUSSES WITH PUMP 1102A AND VALVE 1106A ON A DIFFERENT BUS THAN PUMP 1102B AND VALVE 1106B.
2. CONTROL POWER FOR THE STANDBY LIQUID PUMP AND EXPLOSIVE VALVE SHALL BE TAKEN FROM THE BUSSES FEEDING THE PUMP MOTORS.
3. RMS CONTROL SWITCH IS KEYLOCK IN OFF POSITION. MAINTAIN CONTACT IN "SYS" POSITIONS. SWITCH BYPASSES CONTINUITY MONITOR IN "SYS" POSITIONS TO FIRE VALVE SQUIBS. MOTOR STARTERS ARE INTERLOCKED TO PREVENT SIMULTANEOUS OPERATION OF BOTH PUMPS.
4. CONTINUITY INDICATING LIGHTS ON WHEN FIRING CIRCUIT HAS CONTINUITY THROUGH BOTH SQUIBS. ANNUNCIATOR TRIPS & LIGHT GOES OFF UPON LOSS OF CONTINUITY IN EITHER SQUIBS.
5. PUMP PILOT LIGHTS SHALL FUNCTION AS FOLLOWS:
   RED - "ON" WHEN PUMP RUNNING
   GREEN - "ON" WHEN PUMP STOPPED
   BOTH "OFF" WHEN PUMP MANUALLY SHUTS OFF FOR BREAKER FOR MAINTENANCE
6. CONTINUITY MONITOR TO BE CURRENT LIMITED AND FUSED TO PREVENT ACCIDENTAL FIRING OF VALVES BY ANY MODE OF FAILURE OF THE MONITOR CIRCUIT.
7. LIGHT TO BE "ON" WHEN STANDBY LIQUID SOLUTION IS FLOWING TO REACTOR THUS INDICATING THAT SQUIBS VALVES HAVE OPENED.

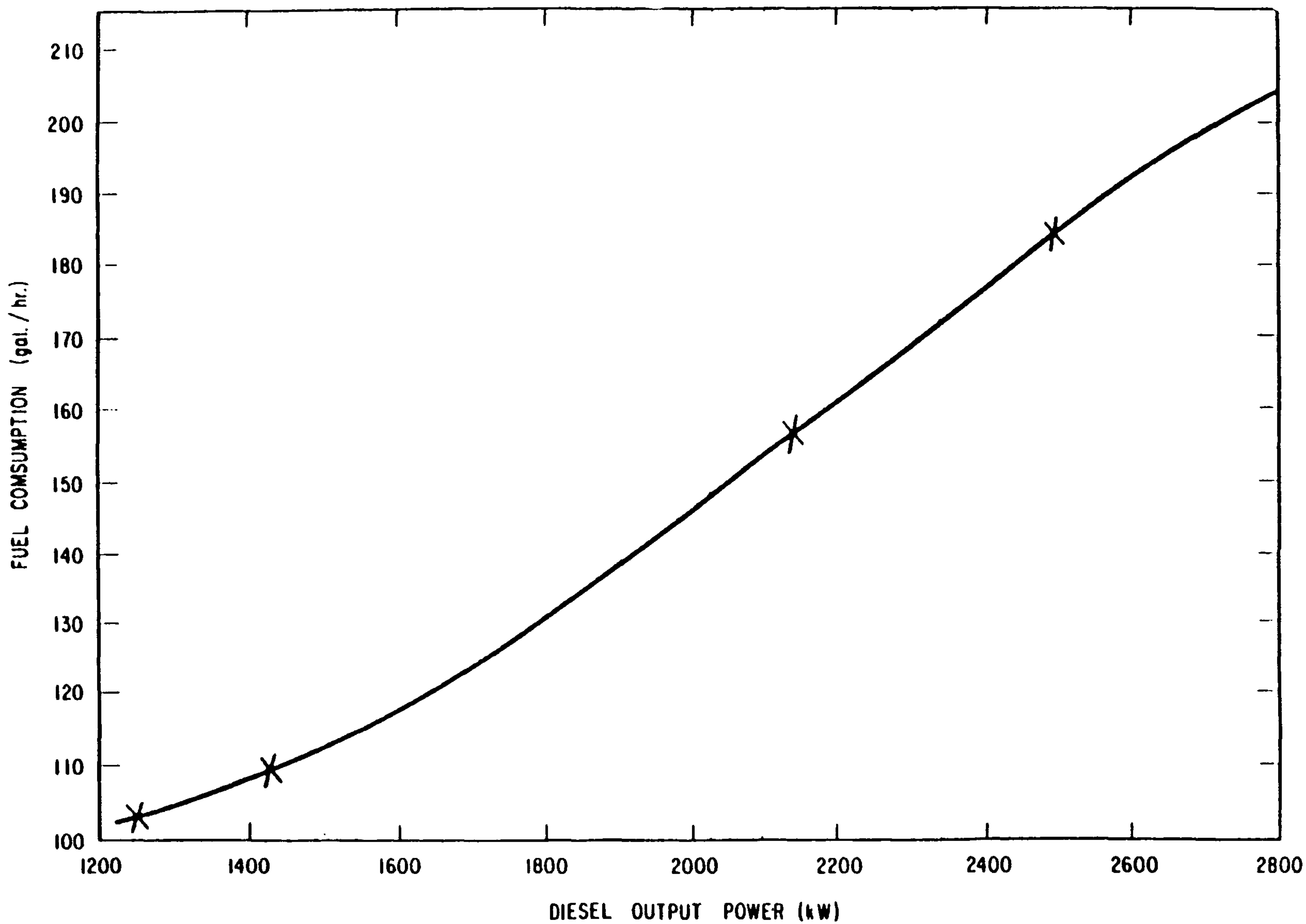
DRESDEN STATION
UNITS 2 & 3
STANDBY LIQUID CONTROL SYSTEM
FUNCTIONAL CONTROL DIAGRAM
FIGURE 9.3-9



DRESDEN STATION
UNITS 2 & 3

TYPICAL DIESEL GENERATOR
FUEL OIL SYSTEM

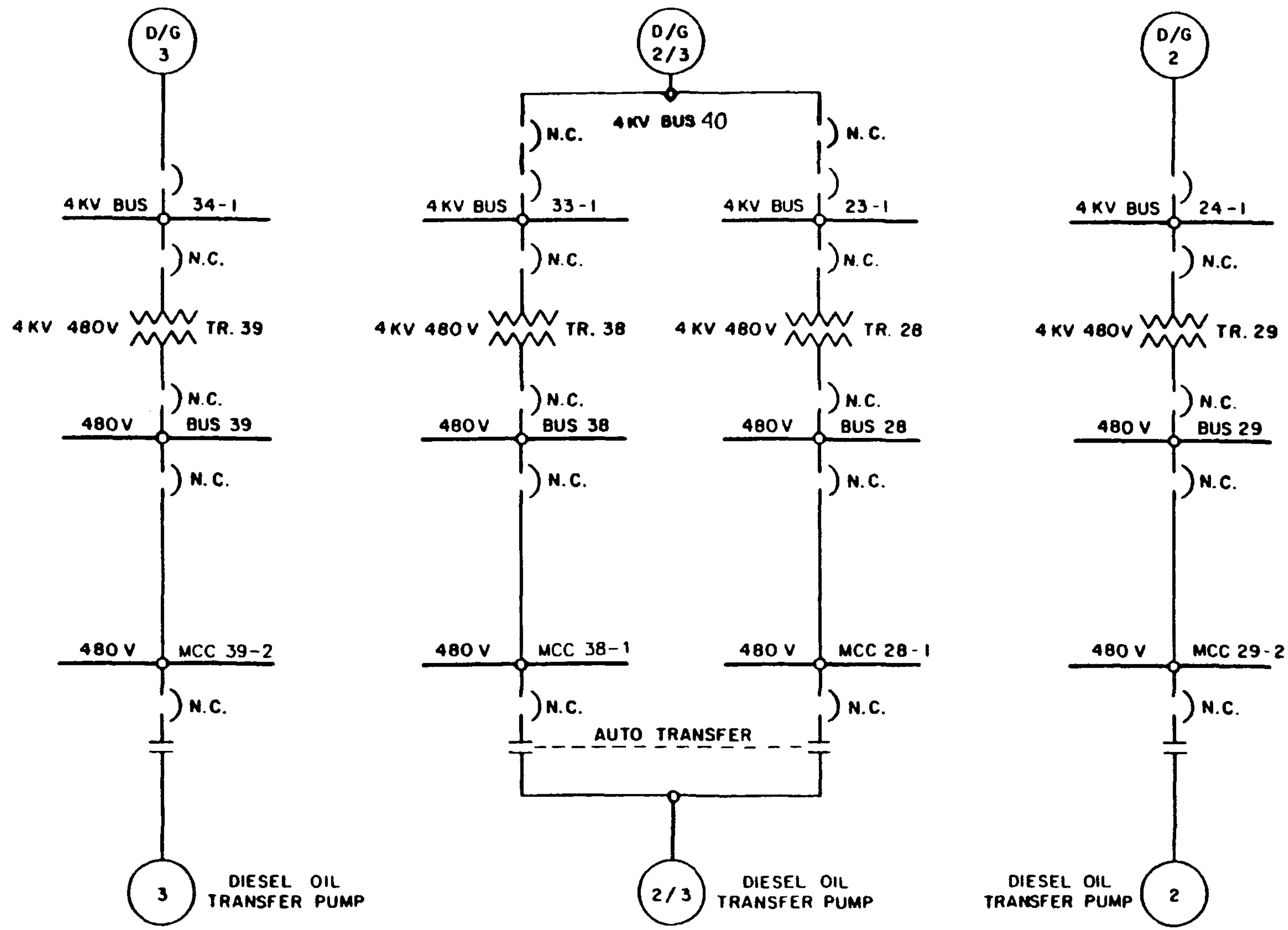
FIGURE 9.5-2



DRESDEN STATION
UNITS 2 & 3

FUEL CONSUMPTION -
S20EGW DIESEL GENERATOR

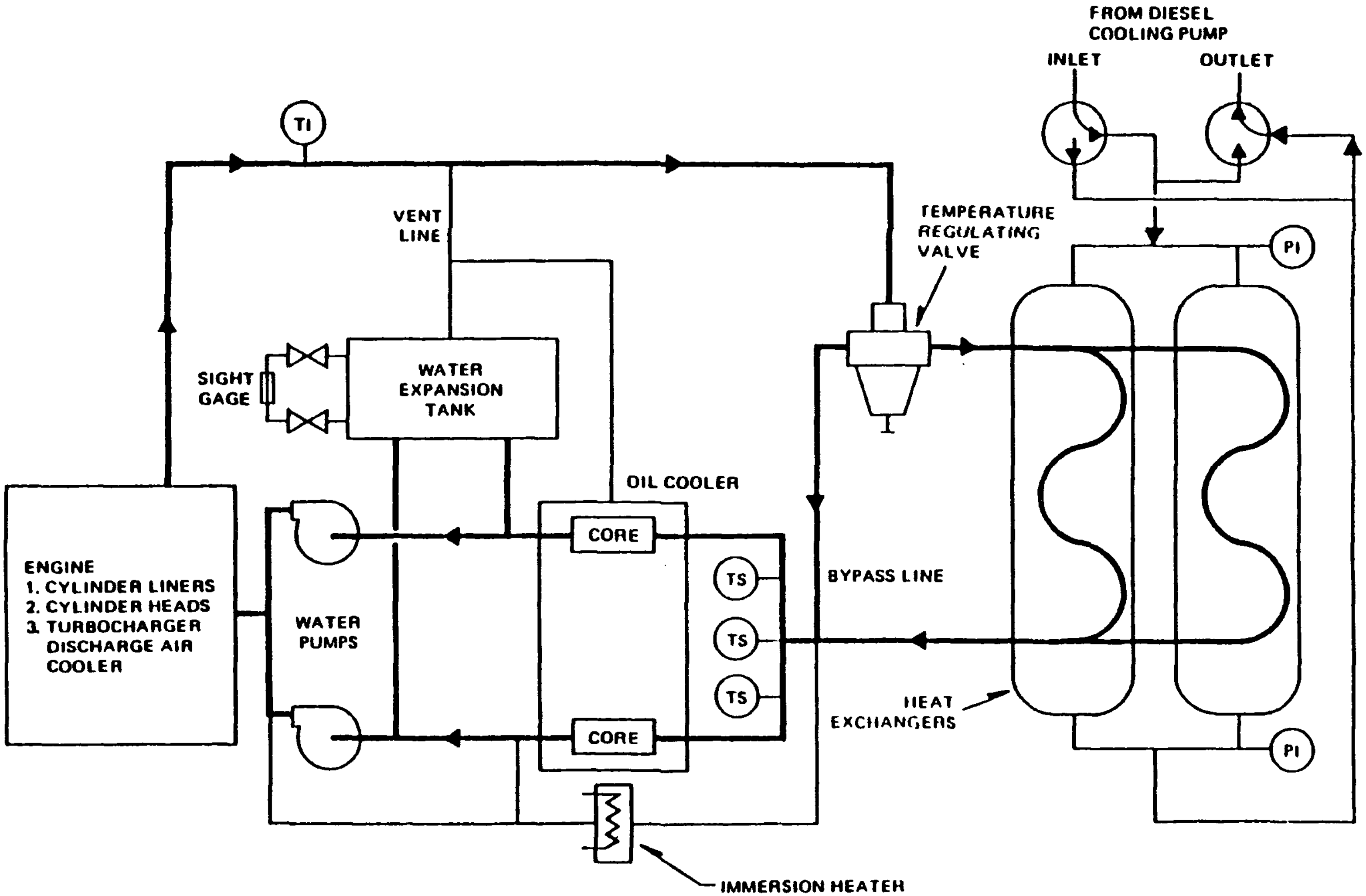
FIGURE 9.5-3



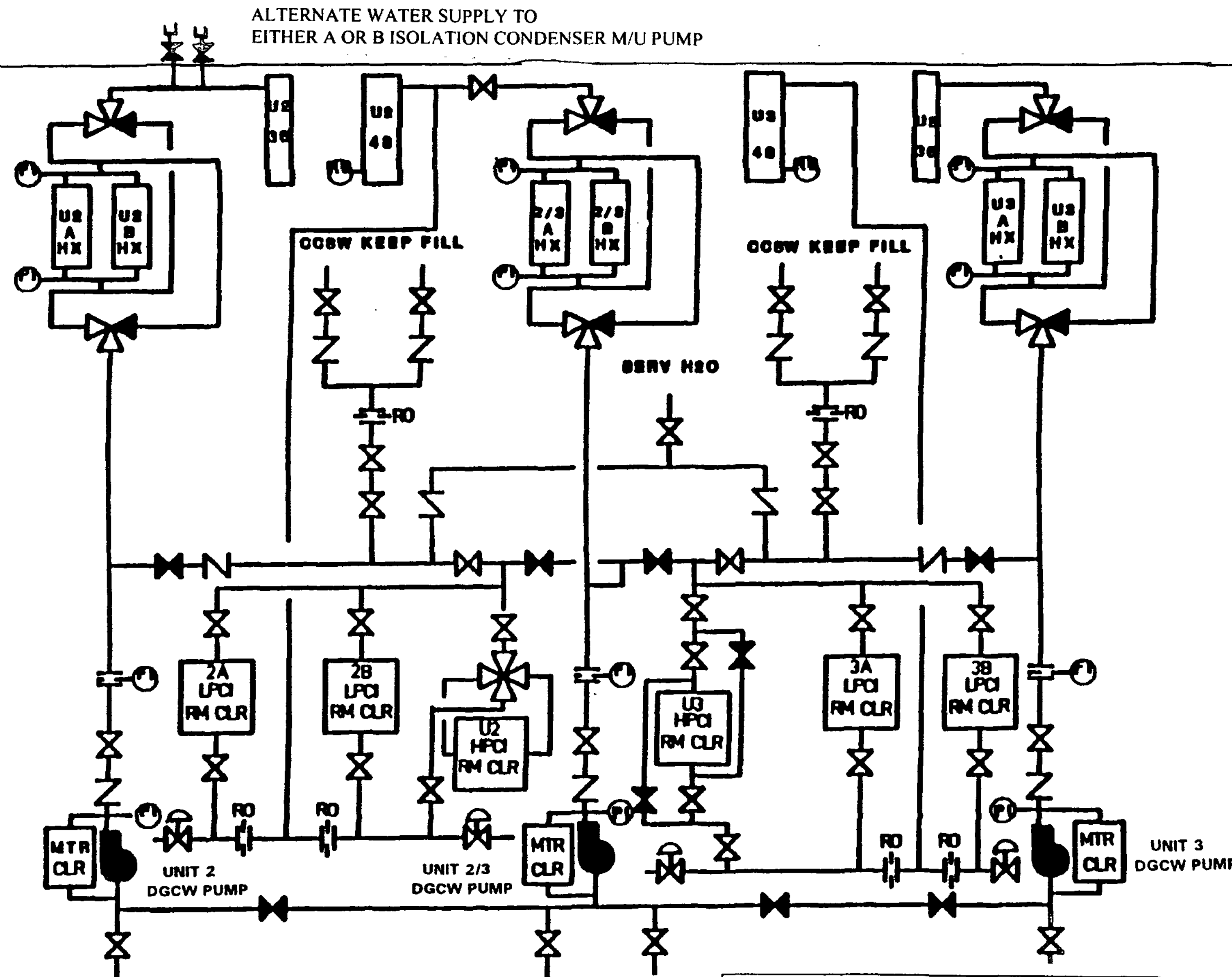
DRESDEN STATION
UNITS 2 & 3

POWER SUPPLIES TO
DIESEL OIL TRANSFER PUMPS

FIGURE 9.5-4



DRESDEN STATION UNITS 2 & 3
TYPICAL DIESEL GENERATOR COOLING WATER SYSTEM
FIGURE 9.5-5

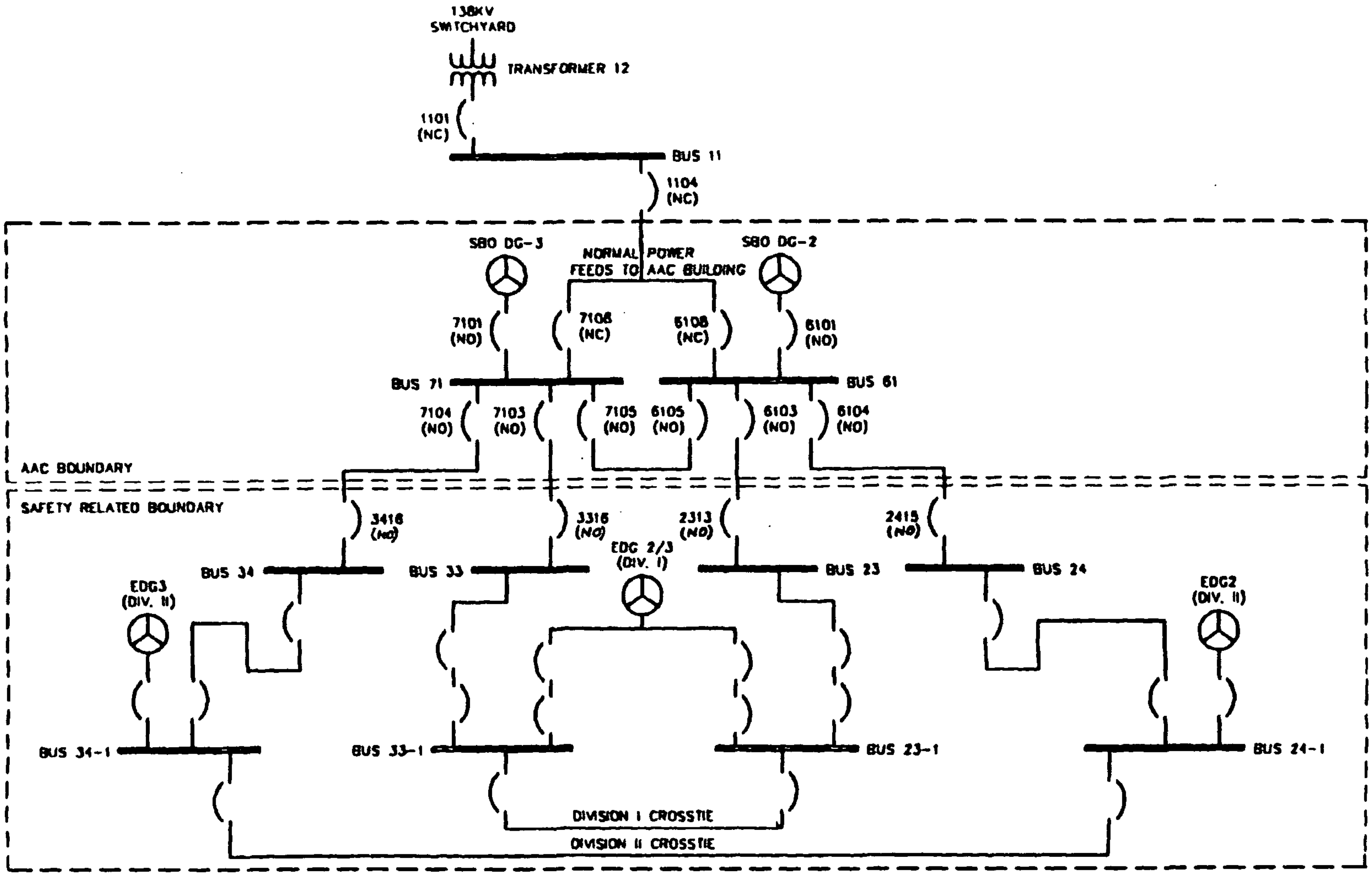


UFSAR REVISION 6, JUNE 2005

DRESDEN STATION UNITS 2 & 3

DIESEL GENERATOR / ECCS ROOM COOLING SYSTEMS

FIGURE 9.5-9



DRESDEN STATION
UNITS 2 & 3

SINGLE-LINE ELECTRICAL DIAGRAM OF STATION BLACKOUT GENERATOR TIES TO PLANT AUXILIARY ELECTRIC SYSTEM

FIGURE 9.5-14